ACRS - 1622

PDR 9/13/79

TABLE OF CONTENTS
MINUTES OF THE
227TH ACRS MEETING
MARCH 8-10, 1979

7909260372

1028 020

Table of Contents
Minutes of the 227th ACRS Meeting

Table of Contents
Minutes of the 227th ACRS Meeting

Appendixes to 227th ACRS Meeting

Table of Contents
Minutes of the 227th ACRS Meeting

1028 023

Table of Contents
Minutes of the 227th ACRS Meeting

1028 024

Commission by a toll-free telephone call to Western Union at (800) 325-6000 (in Missouri (800) 342-6700). The Western Union operator should be given Datagram Identification Number 3737 and the following message addressed to Robert Reid: (petitioner's name and telephone number); (date petition was mailed); (plant name); and (publication date and page number of this FEDERAL REGISTER notice). A copy of the petition should also be sent to the Executive Legal Director, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, and to William H. Cuddy, Esquire, Day, Berry and Howard, Counselors at Law, One Constitution Plaza, Hartford, Connecticut 06103, attorney for the licensees.

Nontimely filings or petitions for leave to intervene, amended petitions, supplemental petitions and/or requests for hearing will not be entertained absent a determination by the Commission, the presiding officer or the Atomic Safety and Licensing Board designated to rule on the petition and/or request that the petitioner has made a substantial showing of good cause for the granting of a late petition and/or request. That determination will be based upon a balancing of the factors specified in 10 CFR 2.714(a)(1)-(v) and 2.714(d).

For further details with respect to this action, see the application for amendment dated December 15, 1978, which is available for public inspection at the Commission's Public Document Room, 1717 H Street, N.W., Washington, D.C., and at the Waterford Public Library, Rope Ferry Road, Route 156, Waterford, Connecticut.

Dated At Bethesda, Maryland this 9th day of February, 1979.

For the Nuclear Regulatory Commission.

ROBERT W. REID,
Chief, Operating Reactors Branch No. 4, Division of Operating Reactors.

[FR Doc. 79-5224 Filed 2-20-79; 8:45 am]

[7590-01-M]

[Docket No. 50-382-OL]

LOUISIANA POWER & LIGHT CO.

Establishment of Atomic Safety and Licensing Board To Preside in Proceeding

Pursuant to delegation by the Commission dated December 29, 1972, published in the FEDERAL REGISTER (37 FR 28710) and §§ 2.105, 2.700, 2.702, 2.714, 2.714a, 2.717 and 2.721 of the Commission's Regulations, all as amended, an Atomic Safety and Licensing Board is being established in the following proceeding to rule on petitions for leave to intervene and/or requests for hearing and to preside over the proceeding in the event that a hearing is ordered.

Louisiana Power & Light Company

(Waterford Steam Electric Station, Unit 3), Construction Permit No. CPPR-103.

This action is in reference to a notice published by The Commission on January 2, 1979, in the FEDERAL REGISTER (44 FR 125-126) entitled "Receipt of Application for Facility Operating License; Availability of Applicant's Environmental Report; Consideration of Issuance of Facility Operating License; and Opportunity for Hearing".

The Chairman of this Board and his address is as follows:

Sheldon J. Wolfe, Esq., Atomic Safety and Licensing Board Panel, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555.

The other members of the Board and their addresses are as follows:

Dr. Walter H. Jordan, 881 W. Outer Drive, Oak Ridge, Tennessee 37830.

Dr. Harry Foreman, Director, Center for Population Studies, Box 395, Mayo, University of Minnesota, Minneapolis, Minnesota 55455.

Dated at Bethesda, Md., this 12th day of February 1979.

ROBERT M. LAZO,
Acting Chairman, Atomic Safety and Licensing Board Panel.

[FR Doc. 79-5358 Filed 2-20-79; 8:45 am]

[7590-01-M]

REGULATORY GUIDE

Issuance and Availability

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

Regulatory Guide 1.28, Revision 2, "Quality Assurance Program Requirements (Design and Construction)," describes a method acceptable to the NCR staff for complying with the Commission's regulations with regard to overall quality assurance program requirements during design and construction of nuclear power plants. This guide endorses ANSI N45.2-1977, "Quality Assurance Program Requirements for Nuclear Facilities."

Comments and suggestions in connection with (1) items for inclusion in guides currently being developed or (2) improvements in all published guides are encouraged at any time. Comments should be sent to the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555, Attention: Docketing and Service Branch.

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

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

Dated at Rockville, Md., this 13th day of February 1979.

For the Nuclear Regulatory Commission.

ROBERT B. MINOGUE,
Director, Office of Standards Development.

[FR Doc. 79-5357 Filed 2-20-79; 8:45 am]

[7590-01-M]

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Meeting

In accordance with the purposes of sections 29 and 182b. of the Atomic Energy Act (42 U.S.C. 2039, 2232 b.), the Advisory Committee on Reactor Safeguards will hold a meeting on March 8-10, 1979, in Room 1046, 1717 H Street, NW, Washington, DC. Notice of this meeting was published on January 19, 1979 (44 FR 4056.)

The agenda for the subject meeting will be as follows:

THURSDAY, MARCH 8, 1979

8:30 A.M.-12:15 P.M.: Executive Session (Open)—The Committee will hear and discuss the report of the ACRS Chairman regarding miscellaneous matters relating to ACRS activities.

The Committee will discuss proposed reports to the NRC regarding the status of unresolved generic matters applicable to light water reactors and the combination of dynamic loads as a design basis for nuclear facilities.

The Committee will discuss the qualifications of candidates proposed for appoint-

1028 025

ment to the Committee. Portions of th' se- sion will be closed as necessary to prot in- formation the release of which would represent an unwarranted invasion of privacy.

1:15 P.M.-1:45 P.M.: Executive Session (Open)—The Committee will hear and discuss the report of its Subcommittee and consultants who may be present regarding proposed resolution of Anticipated Transients Without Scram.

Portions of this session will be closed as required to permit discussion of Proprietary Information related to this matter.

1:45 P.M.-6:15 P.M.: Anticipated Transients Without Scram (Open)—The Committee will hear reports from and hold discussions with representatives of the NRC Staff and the nuclear industry regarding proposed corrective action to resolve Anticipated Transients Without Scram.

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

FRIDAY, MARCH 9, 1979

8:30 A.M.-10:00 A.M.: Executive Session (Open)—The Committee will discuss the role and responsibilities of the ACRS in the NRC regulatory process.

10:00 A.M.-1:30 P.M.: Meeting with NRC Staff (Open)—The Committee will meet with members of the NRC Staff to hear reports on and to discuss recent operating experience and licensing actions. The Committee will also hear and discuss reports regarding generic matters related to the regulatory process including the criteria and procedures for imposition of Civil Penalties (10 CFR Part 2.205) and criteria for consideration of primary coolant pipe failures inside and outside containment.

The future schedule for ACRS activities will also be discussed.

2:30 P.M.-3:00 P.M.: Executive Session (Open)—The Committee will hear and discuss the report of its Subcommittee and Consultants who may be present regarding the request for an Operating License for the William H. Zimmer Nuclear Generating Station Unit 2.

Portions of this session will be closed as required to discuss Proprietary Information related to this facility and arrangements for the physical security of this station.

3:00 P.M.-4:30 P.M.: William H. Zimmer Nuclear Generating Station Unit 2 (Open)—The Committee will hear reports from and hold discussions with representatives of the NRC Staff and the Applicant regarding proposed operation of this unit.

Portions of this session will be closed as required to discuss Proprietary Information related to this facility and arrangements for the physical protection of this station.

SATURDAY, MARCH 10, 1979

8:30 A.M.-12:30 P.M.: Executive Session (Open)—The Committee will discuss its proposed reports to the NRC on the Zimmer Nuclear Station and the proposed resolution of Anticipated Transients Without Scram. Portions of this session will be closed as necessary to discuss Proprietary Information and matters involved in an adjudicatory proceeding.

The Committee will also discuss proposed comments and positions regarding other items discussed during this meeting including a revision of its report on the status of unresolved generic matters applicable to light-water reactors, and use of dynamic load combinations as a design basis for nuclear facilities.

The Committee will hear and discuss reports of its Subcommittees on recent activities related to:

- Improved Safety Systems
- Evaluation of Licensee Event Reports
- Evaluation of Systems Interactions
- Regulatory Activities

The Committee will propose changes to the timing and scope of its annual report to Congress on the NRC Safety Research Program.

The Committee will discuss proposed comments regarding NRC policies related to requirements for shutdown and decay heat removal in nuclear reactors and the use of probabilistic assessment in the licensing process.

Procedures for the conduct of and participation in ACRS meetings were outlined in the FEDERAL REGISTER on October 4, 1978, page 45926. 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.

I have determined in accordance with Subsection 10(d) Pub. L. 92-463 that it is necessary to close protions of this meeting as noted above to protect Proprietary Information (5 U.S.C. 552b(c)(4)), to preserve the confidentiality of classified and Proprietary Information related to safeguarding of special nuclear material and the arrangements for Physical protection of the Zimmer Station (5 U.S.C. 552b(c) (1) and (4)), and to protect information the release of which would constitute a clearly unwarranted invasion of personal privacy (5 U.S.C. 552b(c)(6)) and to permit discussion of matters involved in an adjudicatory proceeding (5 U.S.C. 552b(b)(10)).

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

Dated: February 16, 1979.

JOHN C. HOYLE,
Advisory Committee Management Officer.

[FR Doc. 79-5527 Filed 2-20-79; 8:45 am]

[8010-01-M]

SECURITIES AND EXCHANGE COMMISSION

[Release No. 15566; SR-CSE-79-1]

Cincinnati Stock Exchange

Filing of and Order Approving Proposed Rule Change

FEBRUARY 13, 1979.

Pursuant to Section 19(b)(1) of the Securities Exchange Act of 1934, 15 U.S.C. 78(s)(b)(1) (the "Act"), notice is hereby given that on February 13, 1979, the Cincinnati Stock Exchange ("CSE"), 205 Dixie Terminal Building, Cincinnati, Ohio 45202 filed with the Commission copies of a proposed rule change which would limit the liability of the CSE and of any person providing electronic trading services for the CSE's Multiple Dealer Trading System ("MDTS") on the CSE's behalf in the event that any persons having electronic means of direct access to the MDTS should incur losses as a result of their use of MDTS services.[1] The CSE states that this rule, which is similar to provisions in effect at other exchanges[2] is necessary at this time because a third party is about to undertake to provide certain essential electronic services for the MDTS, and it would not be possible for the undertaking to occur unless the potential liability of the CSE and those providing services on its behalf can be limited in the manner proposed.

Interested persons are invited to submit written data, views and arguments concerning the submission on or before March 26, 1979. Persons desiring to make written comments should file six copies thereof with the Secretary of the Commission, Securities and Exchange Commission, 500 North Capitol Street, Washington, D.C. 20549. Reference should be made to File No. SR-CSE-79-1.

Copies of the submission, all subsequent amendments, all written statements with respect to the proposed rule change which are filed with the Commission, and of all written communications relating to the proposed rule change between the Commission and any person, other than those which may be withheld from the public in accordance with the provisions of 5 U.S.C. section 552, will be

[1] Rule 9D3(l) limits the CSE's liability only with respect to those persons having electronic means of direct access to the MDTS (generally, those persons who are "Users" of the MDTS system as defined in the CSE rules and persons associated therewith) and should not be construed to expand this limitation to any other parties.

[2] See, e.g., Chicago Board Options Exchange, Inc., Rule 6.7; New York Stock Exchange, Inc., Consitution, Article IX, Section 9; and Philadelphia Stock Exchange, Inc., By-Laws, Section 12-11.

POOR ORIGINAL

1028 026

regulatory efforts in accomplishing the general purposes set forth in the Clean Air Act;

—Appropriate automobile emission standards and best available technologies needed to meet them;

—Most appropriate and practical means of preserving air quality in areas in which the air is now cleaner than the national ambient air quality standards;

—Most appropriate and practical means of enhancing air quality in those areas in which established air quality standards are not met;

—Special problems of small business and governmental agencies in obtaining reductions of emissions from existing sources to offset increased emissions from new sources;

—Alternatives to regulation as a means of reducing pollution;

—Inherent problems in efforts to diminish pollution in high altitude areas; and

—Relationship of established environmental regulations to national energy policies.

Those wishing to testify should notify Paul Freeman at (202) 634-7138 by March 7 in order to schedule a time for submission of prepared oral testimony and should send at least 50 copies of such testimony no later than March 14 to the attention of Paul Freeman at the office of the National Commission on Air Quality, 1730 K Street, N.W., Suite 207, Washington, D.C. 20006.

NATIONAL COMMISSION ON AIR QUALITY,
WILLIAM H. LEWIS, Jr.,
Director.

FEBRUARY 28, 1979.

[FR Doc. 79-6590 Filed 3-1-79; 11:31 am]

[7590-01-M]

NUCLEAR REGULATORY COMMISSION

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS, SUBCOMMITTEE ON EMERGENCY CORE COOLING SYSTEMS (ECCS)

Meeting

The ACRS Subcommittee on Emergency Core Cooling Systems will hold a meeting on March 19-20, 1979 at the Travelodge International Hotel, 9750 Airport Blvd., Los Angeles, CA 90045. Notice of this meeting was published February 23, 1979.

In accordance with the procedures outlined in the FEDERAL REGISTER on October 4, 1978 (43 FR 45926), 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 agenda for subject meeting shall be as follows:

MONDAY, MARCH 19, AND TUESDAY, MARCH 20, 1979

8:30 A.M. UNTIL THE CONCLUSION OF BUSINESS EACH DAY

The Subcommittee may meet in Executive Session, with any of its consultants who may be present, to explore and exchange their preliminary opinions regarding matters which should be considered during the meeting and to formulate a report and recommendations to the full Committee.

At the conclusion of the Executive Session, the Subcommittee will hear presentations by and hold discussions with representatives of the NRC Staff, and their consultants, regarding the following topics:

(1) Code Work on Transient Two-Phase Flow
(2) Status of Physical Inputs to Codes
(3) Analysis of LOFT L2-2 Test
(4) Status of ECCS Related Research Programs
(5) Standard Problem Program
(6) ODYN Code Review
(7) Status of Analysis of Assymetric Blowdown Forces
(8) Status of Current Licensing Actions

In addition, it may be necessary for the Subcommittee to hold one or more closed sessions for the purpose of exploring matters involving proprietary information. I have determined, in accordance with Subsection 10(d) of Pub. L. 92-463, that, should such sessions be required, it is necessary to close these sessions to protect proprietary information (5 U.S.C. 552b (c)(4)).

Further information regarding topics to be discussed, whether the meeting has been cancelled or rescheduled, the Chairman's ruling on requests for the opportunity to present oral statements and the time allotted therefor can be obtained by a prepaid telephone call to the Designated Federal Employee for this meeting, Dr. Andrew L. Bates (telephone 202/634-3267) between 8:15 a.m. and 5:00 p.m., EST.

Dated: February 26, 1979.

JOHN C. HOYLE,
Advisory Committee Management Officer.

[FR Doc. 79-6374 Filed 3-1-79; 8:45 am]

[7590-01-M]

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS NUCLEAR REGULATORY COMMISSION

Revised Notice of Meeting

Regarding the previous FEDERAL REGISTER Notice (published on February 21, 1979, Volume 44, P. 10557) for the meeting of the Advisory Committee on Reactor Safeguards to be held on March 8-10, 1979, in Washington, D.C., a change for the items being discussed on Friday, March 9, 1979 has been made as follows.

FRIDAY, MARCH 9, 1979

8:30 a.m.-10:00 a.m.: Executive Session (Open)—The Committee will discuss the role and responsibilities of the ACRS in the NRC regulatory process.

10:00 a.m.-1:15 p.m. and 2:15 p.m.-3:00 p.m.: Meeting with NRC Staff (Open)—The Committee will meet with members of the NRC Staff to hear reports on and to discuss recent operating experience and licensing actions including proposed changes in the Technical Specifications for the Dresden Nuclear Power Station Unit 2. The Committee will also hear and discuss reports regarding generic matters related to the regulatory process including the criteria and procedures for imposition of Civil Penalties (10 CFR Part 2.205), the Nonproliferation Alternative Systems Assessment Program and the shipment of spent reactor fuel elements through densely populated areas.

The future schedule for ACRS activities will also be discussed.

3:00 p.m.-3:30 p.m.: Executive Session (Open)—The Committee will hear and discuss the report of its Subcommittee and Consultants who may be present regarding the request for an Operating License for the William H. Zimmer Nuclear Generating Station Unit 1.

Portions of this session will be closed as required to discuss Proprietary Information related to this facility and arrangements for the physical security of this station.

3:30 p.m.-7:00 p.m.: William H. Zimmer Nuclear Generating Station Unit 1 (Open)—The Committee will hear reports from and hold discussions with representatives of the NRC Staff and the Applicant regarding proposed operation of this unit.

Portions of this session will be closed as required to discuss Proprietary Information related to this facility and arrangements for the physical protection of this station.

1028 027

Dated: February 26, 1979.

JOHN C. HOYLE,
Advisory Committee Management Officer.

[FR Doc. 79-6375 Filed 3-1-79; 8:45 am]

[7590-01-M]

(Byproduct Material License No. 45-02808-04)

ATLANTIC RESEARCH CORP.

Oral Argument

Notice is hereby given that, in accordance with the Appeal Board's order of February 22, 1979, oral argument on the appeal of Atlantic Research Corporation, Alexandria, Virginia, from the decisions of the Administrative Law Judge in this civil penalty proceeding will be heard at 10 a.m. on Thursday, March 22, 1979, in the Commission's Hearing Room, 5th floor, 4350 East-West Highway, Bethesda, Maryland.

Dated: February 26, 1979.

For the Appeal Board.

MARGARET E. DU FLO,
Secretary to the Appeal Board.

[FR Doc. 79-6378 Filed 3-1-79; 8:45 am]

[7590-01-M]

DRAFT REGULATORY GUIDES AND NUREG REPORTS

Issuance and Availability

The Nuclear Regulatory Commission has prepared draft Regulatory Guides and NUREG Reports to aid licensees in implementing proposed amendments to 10 CFR Part 73 (§ 73.20, 73.25, 73.26, 73.45, 73.46), which were published in the FEDERAL REGISTER August 9, 1978. These documents have been assembled into 3 volumes:

"Fixed Site Physical Protection Upgrade Rule—Guidance Compendium, Volume I"

"Fixed Site Physical Protection Upgrade Rule—Guidance Compendium, Volume II"

"Standard Format and Content Guide for Physical Protection of Strategic Special Nuclear Material In Transit".

These draft volumes are being made available to concerned parties so that they may review the materials and provide comments and suggestions early in the development of this guidance. The NRC anticipates that these documents will be revised in response to the comments, and will be made final concurrently with the effective date of the aforementioned amendments to 10 CFR Part 73, in mid-1979.

A seminar is scheduled for March 27-28, 1979 in Richmond, Virginia, to orient potential users in the application and content of these documents. Present licensees will be contacted regarding seminar arrangements. Other interested parties should contact Mr. L. J. Evans, Jr., Chief, Requirements Analysis Branch, Division of Safeguards, Office of Nuclear Material Safety and Safeguards, U. S. Nuclear Regulatory Commission, Washington, D.C. 20555, telephone number (301) 427-4043 by March 9, 1979.

Copies of these documents will be available for public inspection at the NRC's Public Document Room at 1717 H Street, N.W., Washington, D.C. 20555.

Dated at Silver Spring, Md. this 14th day of February 1979.

For the Nuclear Regulatory Commission.

WILLIAM J. DIRCKS,
Director, Office of Nuclear Material Safety and Safeguards.

[FR Doc. 79-6373 Filed 3-1-79; 8:45 am]

[7590-01-M]

[Docket Nos. 50-498A, 50-499A, 50-445A, 50-446A]

HOUSTON LIGHTING & POWER CO., ET AL., (SOUTH TEXAS PROJECT, UNITS 1 AND 2), TEXAS UTILITIES GENERATING CO., ET AL., (COMANCHE PEAK STEAM ELECTRIC STATION, UNITS 1 AND 2)

Prehearing Conference and Arguments on Motion To Quash Subpoenas

FEBRUARY 23, 1979.

Pursuant to Board's Order dated December 5, 1978, a prehearing conference will be held at 10:00 a.m., local time, on March 20, 1979 in the Nuclear Regulatory Commission's Hearing Room, 4350 East-West Highway, 5th Floor, Bethesda, Maryland to consider and review progress made by the parties in completing discovery and preparing for an early commencement of the evidentiary hearing.

Commencing at 9:00 a.m. on March 29, 1979 at the same location mentioned above, the Board will hear arguments on the Joint Motion to Quash Subpoenas, filed by counsel for Air Products and Chemicals, Inc.; E. I. DuPont de Nemours & Co.; Monsanto Company; PPG Industries, Inc.; and Union Carbide Corporation dated February 16, 1979.

Dated at Bethesda, Md., this 23rd day of February 1979.

It is so ordered.

For the Atomic Safety and Licensing Board.

MARSHALL E. MILLER,
Chairman.

[FR Doc. 79-6376 Filed 3-1-79; 8:45 am]

[7590-01-M]

PUBLIC SERVICE ELECTRIC & GAS CO. (SALEM NUCLEAR GENERATING STATION, UNIT 1)

[Docket No. 50-272]

Order Rescheduling Prehearing Conference; Proposed Issuance of Amendment to Facility Operating License No. DPR-70

Notice is hereby given that, pursuant to 10 CFR 2.752, the prehearing conference in the above-referenced matter which was originally scheduled for February 22, 1979, shall be held at 1:30 p.m. on Thursday, March 15, 1979, in the Main Courtroom (1st Floor), Old Salem Courthouse, Broadway and Market Streets, Salem, New Jersey.

The parties are directed to be prepared to discuss the items listed in 10 CFR 2.752. The Licensee shall also be asked to arrange a visit to the facility by the Board.

Dated at Madison, Wisconsin, this 26th day of February 1979.

It is so ordered.

For the Atomic Safety and Licensing Board.

GARY L. MILHOLLIN,
Chairman.

[FR Doc. 79-6377 Filed 3-1-79; 8:45 am]

[7590-01-M]

[Docket No. 50-272]

PUBLIC SERVICE ELECTRIC & GAS CO. (SALEM NUCLEAR GENERATING STATION, UNIT 1)

Order Rescheduling Special Prehearing Conference for Limited Appearances; Proposed Issuance of Amendment to Facility Operating License No. DPR-70

By its Order of December 15, 1978, this Board granted a motion to hold a special prehearing conference for the purpose of receiving statements from persons who wish to make limited appearances under 10 CFR 2.715. The conference, scheduled for February 22, 1979, was cancelled because of bad weather.

Notice is hereby given that, pursuant to 10 CFR 2.751a and 10 CFR 2.715, the special prehearing conference will be held at 9:30 a.m. on Friday, March 16, 1979, in the Main Courtroom (1st Floor), Old Salem Courthouse, Broadway and Market Streets, Salem, New Jersey. The Board will also meet at this same location at 7:00 p.m. on Thursday, March 15, 1979, to accept appearances by persons who are unable to appear during normal working hours.

All persons desiring to make limited appearances in this proceeding shall attend this special prehearing conference. If the Chairman so determines, persons desiring to make their state-

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

February 28, 1979

SCHEDULE AND OUTLINE
FOR DISCUSSION
227TH ACRS MEETING
MARCH 8-10, 1979
WASHINGTON, DC

Thursday, March 8, 1979, Room 1046, 1717 H Street, NW, Washington, DC

1) 8:30 A.M. - 12:30 P.M.

Executive Session (Open)

1.1) 8:30 A.M. - 9:30 A.M.: Report of ACRS Chairman

1.2) 9:30 A.M. - 11:00 A.M.: Report of ACRS Subcommittee on proposed revision of ACRS Report on the Status of Generic Items Related to Light-Water Reactors

1.3) 11:00 A.M. - 12:00 Noon: Discuss proposed report to NRC regarding combination of Dynamic Loads as a Design Basis for Nuclear Facilities

1.4) 12:00 Noon - 12:30 P.M.: Discuss proposed candidates for appointment to the ACRS (Portions of this session will be closed as necessary to protect information the release of which would represent an unwarranted invasion of personal privacy.)

2) 12:30 P.M. - 1:30 P.M. LUNCH

1028 029

3) 1:30 P.M. - 2:00 P.M.

Executive Session (Open)

3.1) Report of ACRS Subcommittee on resolution of Anticipated Transients Without Scram (Portions of this session will be closed as necessary to permit discussion of Proprietary Information related to this matter.)

4 2:00 P.M. - 6:30 P.M.

Anticipated Transients Without Scram (Open)

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

Friday, March 9, 1979, Room 1046, 1717 H Street, NW, Washington, DC

5) 8:30 A.M. - 10:00 A.M.

Executive Session (Open)

5.1) Discuss the role and responsibilities of the ACRS in the NRC regulatory process

6) 10:00 A.M. - 1:15 P.M.

Meeting with NRC Staff (Open)

6.1) 10:00 A.M.- 10:30 A.M.: Report on recent Operating Experience and Licensing Actions

6.1-1) Diablo Canyon Nuclear Station - Instrument line integrity

6.1-2) Dresden Nuclear Power Station Unit 2 - Proposed change in Technical Specifications regarding secondary containment leak rate

6.2) 10:30 A.M. - 11:30 A.M.: Report on criteria and Procedures for Imposition of Civil Penalties (10 CFR Part 2.205)

	6.3) <u>11:30 A.M. - 1:15 P.M.</u>: Report on Nonproliferation Alternative Systems Assessment Program
7) 1:15 P.M. - 2:15 P.M.	LUNCH
8) 2:15 P.M. - 3:00 P.M.	<u>Meeting with NRC Staff (Open)</u> 8.1) <u>2:15 P.M. - 2:45 P.M.</u> Report on proposed DOT criteria regarding shipment of radioactive materials 8.2) <u>2:45 P.M. - 3:00 P.M.</u>: Future Schedule
9) 3:00 P.M. - 3:30 P.M.	<u>Executive Session (Open)</u> 9.1) Report of ACRS Subcommittee on William H. Zimmer Nuclear Generating Station Unit 1 (Portions of this session will be closed as required to discuss Proprietary Information related to this facility and provisions for the physical protection of this Station.)
10) 3:30 P.M. - 7:00 P.M.	<u>William H. Zimmer Nuclear Generating Station Unit 1 (Open)</u> (Portions of this session will be closed as required to discuss Proprietary Information related to this facility and provisions for the physical protection of this Station.)

Saturday, March 10, 1979, Room 1046, 1717 H Street, NW, Washington, DC

11) 8:30 A.M. - 4:30 P.M.

Executive Session (Open)

11.1) Discuss proposed ACRS reports to NRC regarding:
- Wm. H. Zimmer Station
- ATWS
- Revision of ACRS report on Unresolved Generic Matters Applicable to LWR's
- Combination of Dynamic Loads as a Design Basis for Nuclear Plants

11.2) Reports of ACRS Subcommittees on:
- Improved Safety Systems
- Evaluation of Licensee Event Reports
- Evaluation of Systems Interactions
- Regulatory Activities

11.3) Discuss proposed changes in the timing and scope of the ACRS Annual Report to Congress on the RSR Safety Research Program

11.4) Discuss ACRS comments-recommendations regarding candidates proposed for appointment to the Committee

11.5) Di cuss proposed comments/re mmendations regarding:
- RC Staff policies related to requirements for shutdown and decay heat removal using safety grade equipment

- Use of probabilistic assessment in the licensing process
- Criteria and procedures for imposition of civil penalties

(Portions of these sessions will be closed as necessary to discuss Proprietary Information and provisions for physical security at the facilities noted; to permit discussion of material involved in an adjudicatory proceeding; and to protect information the release of which would represent an unwarranted invasion of personal privacy.)

oOo

Issue Date:

June 4, 1979

MINUTES OF THE 227TH ACRS MEETING MARCH 8-10, 1979 WASHINGTON, DC

The 227th meeting of the Advisory Committee on Reactor Safeguards, held at 1717 H St. N.W., Washington, DC, was convened at 8:30 a.m., Thursday, March 8, 1979.

[Note: For a list of attendees, see Appendix I. Mr. Moeller was not present on Saturday.]

The Chairman noted the existence of the published agenda for this meeting, and the items to be discussed. He noted that the meeting was being held in conformance with the Federal Advisory Committee Act (FACA) and the Government in the Sunshine Act (GISA), Public Laws 92-463 and 94-409, respectively. He noted that no requests had been received from members of the public to present oral statements. He also noted that copies of the transcript of some of the public portions of the meeting would be available in the NRC's Public Document Room at 1717 H St. N.W., Washington, DC, within approximately 24 hours.

[Note: Copies of the transcript taken at this meeting are also available for purchase from Ace Federal Reporters, Inc., 444 North Capitol St. N.W., Washington, DC, 20001.]

I. Chairman's Report (Open to Public)

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

A. Reviewers

The Chairman named Messrs. Moeller and Okrent as reviewers and Mr. Siess as alternate reviewer for the 227th ACRS Meeting.

B. Honors to ACRS Members and ACRS Staff

1. Mr. Plesset

The Chairman announced that Mr. Plesset has been elected to the National Academy of Engineering.

1028 034

2. Mr. Shewmon

The Chairman announced that Mr. Shewmon has been elected to the National Academy of Engineering.

3. R. Muller

The Chairman presented a 30 year service pin and certificate to R. Muller, ACRS Staff.

4. R. L. Wright

The Chairman noted that R. L. Wright was planning to leave his position on the ACRS Staff in the near future. Members expressed their gratitude to him for the service and cooperation he extended to the Members while he was employed here.

C. Floating Nuclear Plant

The Chairman informed the Committee that Offshore Power Systems, Inc., applicant for a Preliminary Design Approval for the Floating Nuclear Plant, has agreed to provide a core retention system for the proposed plant, and that the AS&LB hearings have been rescheduled.

D. ECCS History

The Chairman suggested, and the Committee agreed, that a discussion regarding the history of ECCS and ECCS research be scheduled for the 229th ACRS Meeting (May). Persons closely involved with this matter in its early days, such as S. H. Hanauer, H. S. Isbin, A. J. Pressesky, S. A. Szawlewicz, and personnel working in the AEC's Division of Reactor Safety Research in its early days are to be invited to participate.

E. Appearance of ACRS Consultants at the Diablo Canyon Hearings

The Chairman informed the Committee that two ACRS consultants, M. Trifunac and E. Luco, were subpoenaed, and have appeared before the AS&LB hearings on Diablo Canyon. The hearings went well, and there appeared to be no problems regarding their appearance. Legal counsel was provided for these consultants

Mr. Okrent requested that the ACRS Staff ascertain from D. Allison, Diablo Canyon Project Manager, whether any new issues were raised during these hearings. If new issues were raised, D. Allison should be requested to inform the Committee of these issues at the 228th ACRS Meeting (April).

F. Overturning a California Nuclear Power Plant Statute

Mr. Plesset informed the Committee that the Federal District Court of California has declared the California law that prohibited the future siting of nuclear power plants, unless the waste disposal problem was solved (Chapter 196 of the 1976 Laws of California, Assembly Bill 2822, Section 25524.2), invalid.

G. Testimony Before House and Subcommittee on Energy and the Environment

The Chairman informed the Committee that he and Messrs. Moeller, Plesset and Siess testified on February 22, 1979 before the House Committee on the Interior and Insular Affairs, Subcommittee on Energy and the Environment, regarding the NRC Research Program.

H. NRC Staff Documents

The Chairman informed the Committee that H. R. Denton is willing to provide the Committee with the type documents that were requested at the 226th ACRS Meeting, provided the ACRS Staff would identify the type documents the Committee desires.

II. Meeting on Anticipated Transients Without Scram (ATWS) (Open to Public)

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

A. Subcommittee Report

Mr. Kerr, ATWS Subcommittee Chairman, discussed the proposed NRC resolutions for the ATWS problem, vendor recommendations, and ACRS consultants' recommendations (see Appendix IV). He noted that comments have been received from several utilities, vendors, and the Atomic Industrial Forum (see Appendix V).

Several Members noted that the NRC Staff had sent a 58-page questionnaire to nuclear power plant operators in an effort to obtain data to be applied to the proposed NRC Staff ATWS resolutions. The time allowed for response to this questionnaire appeared to be rather short.

E. G. Case, NRC Staff, admitted that the questionnaire was long, but indicated that the time frame in which answers would be required was negotiable between the NRC Staff and the specific utility.

B. NRC Staff Presentation

E. G. Case discussed the NRC Staff's current position regarding ATWS, as described in NUREG 0460 (Vol. 3) (see Appendix VI).

C. Industry's Presentations

1. Atomic Industrial Forum

J. Sorensen, Washington Public Power Supply System, representing the Atomic Industrial Forum, discussed the ATWS problem from the industry point of view. He noted concern that the industry input has been discounted by the NRC Staff, that ATWS consequences are overstated by the NRC Staff, that an inadequate value/impact assessment has been placed on the Staff's proposed fixes, and that NRC procedures for orderly regulation have not been followed. (For his verbatim remarks, see Appendix VII.)

W. Owen, Duke Power Co., representing the Atomic Industrial Forum, noted the opinion that ATWS is a significant problem to utility executives, in that it is a good example of the regulatory uncertainties that have plagued the industry for years. He discussed the problems of modifying existing plants and plants under construction, along with the additional problem that duplicate plants may not, in fact, be duplicates if they were licensed after a certain date. (For W. Owen's verbatim presentation, see Appendix VIII.) In answer to a question, W. Owen said that the most bothersome part of the NRC Staff proposed requirements are their retrofit portions.

2. General Ele ric Company (GE)

E. Stroupe (GE) discussed the cost to utilities for the proposed NRC Staff ATWS resolutions, boiling water reactor mitigation capabilities, boiling water reactor scram system reliability, shutdown considerations, factors in achieving high reliability, inherent capability of the BWR, BWR ATWS mitigation capability, the current capabilities of the plants, the capabilities of the BWR plant equipped with automatic rod insertion systems, and the responses of BWR plants to the proposed resolutions (see Appendix IX).

3. Westinghouse's Presentation

R. Steitler said that Westinghouse (W) believes that ATWS is not a safety problem. In W plants, several diverse sensors detect rod insertion for anticipated events. He noted his opinion that the original NRC Staff ATWS document, WASH-1270, used unrealistic data, and had an unrealistic goal. At that time W again concluded that there was no safety problem. The NRC Staff has never shown that an ATWS event in a W plant will lead to core-melt or serious consequences. No consequences are found beyond a short period of pressure higher than operating but below the danger point.

R. Steitler said that to meet the NRC's Staff alternates 2, 3, and 4, some hardware changes would be required. Different analysis shows that the risk for W plants is essentially the same for alternates 2, 3, and 4. W believes that no requirements beyond alternate 2 should be required for W plants.

S. H. Hanauer, NRC Staff, said that W and the NRC Staff are in agreement; both believe that W plants already meet alternates 2, 3, and 4. The purpose for the Staff's proposed requirement of analysis for new plants is to try to obtain more assurance that an ATWS will not produce severe consequences.

W. Lipinski, ACRS consultant, noted a recent failure of an undervoltage breaker in a W plant. This is the type of breaker that would trip the plant in the event of an ATWS. If a new breaker were selected, data would have to be provided for its reliability, as theoretical analysis, alone, cannot provide this type of reliability.

1028 033

A. C. Thadani, NRC Staff, noted that the same type of breaker is used trip the turbine in a W plant.

4. Babcock and Wilcox (B&W)

A. McBride, B&W, discussed the frequency of an ATWS event, the consequences and the risk, methods for reducing risks, and recommendations to the Committee (see Appendix X).

5. Combustion Engineering (CE) Presentation

W. E. Burchill, CE, discussed the CE ATWS position, flaws in NUREG-0460 (Vol. 3), a supplementary protection system functional description, and CE recommendations to the Committee (see Appendix XI).

D. ACRS Consultants' Comments

1. S. Ditto's Comments

S. Ditto said that he believes that the addition of the Automatic Rod Insertion (ARI) system to the GE plants would improve the reliability of scram. He noted that although he was not certain of the details of the logic system in ARI, the idea of dumping the scram header volume is a sufficiently diverse mechanism from the normal air pressure release of the scram valves to be considered a redundant system. He compared the ARI system as equivalent to the tripping of the motor generator sets in a PWR. He said that a manually operated air dump valve might be better than an automatic dump valve, or a manual system used to introduce poison. He suggested that there is a negative incentive for an operator to dump poison into the core if an inadvertent action is going to cost the utility $25 to $30 million dollars. He suggested that if spurious scrams were to cost $25 to $30 million dollars, operators would find ways to assure that the scram systems did not operate. He said that he has very little confidence that a good automatic poison injection system that can be tolerated will also be reliable. He also noted his skepticism that one can solve a problem that cannot be clearly identified

In answer to a question, S. Ditto said that he believes that if the rod drive systems received a signal to insert, there is only a very small probability that the rods will fail

1028 039

to insert. He said he believes that failure would result from a malfunction of the sensor or electrical system failing to receive the signal to insert.

2. W. Lipinski

W. Lipinski noted that the effectiveness of the ARI System depends upon a reactor pump trip having occurred. The principle is that after the reactor pump has been tripped, an alternate method for insertion of the rods is available.

3. E. P. Epler

Mr. Epler noted his lack of confidence in the reliability of reactor operators performing in the specific prescribed manner. Rather, he said he preferred automated systems.

E. NRC Staff Response

E. G. Case noted that the NRC Staff has heard nothing at this meeting that would cause it to alter its current position on ATWS.

S. H. Hanauer noted the following points of agreement between the NRC Staff and the reactor vendors:

- the need for considering value impact, the use of best estimate analysis, provided conservative estimates are made where inadequate data exist;
- the necessity for quenchers for suppression pool integrity in ATWS events in BWRs;
- that if W plants can assure turbine trip and feedwater initiation, these plants can meet the ATWS requirements specified by the NRC Staff; and
- high pressure does not necessarily yield core melt - the NRC Staff is anxious to see real analyses of this matter.

S. H. Hanauer noted the following points of disagreement between the NRC Staff and the reactor vendors:

- that ATWS is a safety concern, and that there is a difference between alternatives 3 and 4;

1028 040

- NRC is not rushing to judgment in this matter; there have been many opportunities for commenting in various forms, and the proposal to go to rulemaking negates the complaint;

- the validity of reliance on prompt operator actions in less than 10 minutes;

- the introduction of an ATWS requirement in the spirit of NUREG-0460 (Vol. 3) will not give rise to a whole spectrum of new requirements;

- the degree of improvement that ARI can provide in a BWR scram system, although the NRC Staff would not object to installation of an ARI system;

- the BWR has not been demonstrated to have less susceptibility to ATWS, nor a higher reliability scram system;

- the industry will find some way to mitigate against a $25 million cost of a spurious actuation of a safety system;

- the integrity of a BWR suppression pool at a temperature above 200 °F;

- the acceptability of containment rupture; and

- that an hour is available to replenish condensate under all ATWS circumstances.

S. H. Hanauer identified the following matters which fall into neither of the above categories:

- The purpose of the NRC Staff's request for analyses in its February 15 letter was to provide the analyses needed in a different mode to avoid the necessity of calculating for each reactor or each reactor reload. The Staff does not intend to make ATWS a design basis accident.

- The NRC Staff is not entirely decided on the date for which alternate 4 will be effective. There is some flexibility in this matter.

. The NRC Staff believes that alternative 4 provides substantially more reliable mitigating equipment that has higher performance than that of alternative 3. The NRC Staff intends to follow the Commission's admonition to use probabilistic calculations with extraordinary care.

. Moderator temperature coefficients of reactivity on some reloads seem to be higher than had been considered earlier.

F. Caucus

The Committee determined that it would need additional information before it could reach a consensus on ATWS. It agreed to continue consideration of this problem at the 228th ACRS Meeting (April). Members identified the following matters to be discussed:

For BWRs

. The maximum temperature and pressure transients the torus or pressure suppression pools of GE containments can accept without rupture. What are the consequences if failure of the torus or suppression pool occurs?

. Evaluation and comparison of the effects of 43, 86, and 400 GPM liquid boron injection rates on the predicted transients in the pressure suppression pool or torus.

. The effects of various time delays of boron injection (to 10 minutes) on the predicted transients in the pressure suppression pool or torus.

For PWRs

. Water hammer potentials in PWR primary cooling systems during ATWS events, and their predicted effect on safety relief valves, for example.

. Effects of exceeding Service Level C stresses on the reactor pressure vessel during the ATWS pressure pulse. In the event that leakage around head gaskets could occur, what are the consequences of such leakage?

III. Meeting on William H. Zimmer Nuclear Power Station Unit 1 (OL) (Open to Public)

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

A. Subcommittee Report

Mr. Bender, Zimmer Subcommittee Chairman, discussed the status of the Subcommittee's review of the application for an operating license for the Zimmer Nuclear Power Station, and included a site description, a summary of the Mark II Containment lead plant load evaluation and acceptance criteria, a Mark II pool dynamic load summary, technical evaluations by ACRS consultants, a summary of the Mark II Containment Reassessment Program, and inspection reports and correspondence relating to allegations of noncompliance with NRC criteria and regulations (see Appendix XII). He noted that some questions had been raised regarding quality assurance at Zimmer, but that an active program has been established to correct any of the difficulties, that the NRC Staff is monitoring this program, and at this time is satisfied.

[Note: J. Flynn, Cincinnati Gas and Electric Company (CG&E) coordinated presentations for the Applicant; I. Peltier for the NRC Staff.]

B. Status of NRC Staff Review

I. Peltier discussed the status of the NRC Staff's review, and noted that the review of the Zimmer application has been completed, and subject to satisfactory resolution of very few remaining issues, the plant may be operated without undue risk to the health and safety of the public. (For a list of outstanding and confirmatory items, see Appendix XIII.)

J. Schultz, NRC Staff, explained the reason for the NRC Staff's revision of the safety evaluation report. He said that for the past several months, the Office of the Executive Legal Director (OELD) has been concerned that an adequate basis for the conclusions presented in the SER has not been provided. This matter was brought to the attention of the OELD by the AS&LB.

The NRC Staff incorporated an improved description of the bases for the conclusions drawn in the SER, and as a result a revision was issued that concurred with the requirements set forth by OELD. Approximately half of the text in the report was affected. Most of the changes occur in two categories, the citing of approved references or the detailed descriptions of the bases used in the report. No substantive changes in the conclusions occurred as a result of this effort.

J. Schultz noted that this revision problem will affect most of the plant reviews in the near future.

C. Applicant's Presentations

1. Operating Organization

E. Borgmann, CG&E, provided an organization chart for the operation of the Zimmer Plant (see Appendix XIV).

In answer to a question, he noted that among supervisory personnel there is accumulated 75 man years of experience, and among operators there is also approximately 75 man years of experience in nuclear plants.

2. Plant and Site

H. Brinkmann, CG&E, discussed the location and the layout of the Zimmer plant and the site upon which it is located. He noted that the site is approximately one-half mile from the village of Moscow, Ohio, and is located near the Ohio River on its flood plain. He noted that U. S. Highway 52 runs adjacent to the site, as does the Ohio River, on which there is freight barge traffic. Using aerial photographs, he identified the major structures on the site. He noted that discharged steam from the safety valves is piped through the reinforced concrete floor that separates the dry well from the wet well, and is discharged into the suppression pool through quenchers. For loss of coolant accidents, downcomers are provided to exhaust steam from the dry well to the suppression pool. Thirteen safety relief valves are provided.

Dr. Krishnaswamy, Sargent and Lundy, discussed the engineering details of the Mark II containment system.

3. BWR/V Changes from BWR/IV

R. Johnson, GE, discussed the differences between the BWR/V Nuclear Steam Supply System (NSSS) and the BWR/IV NSSS (see Appendix XV).

He noted that the solid state manual control system is being applied for the first time at Zimmer.

4. Personnel Training

J. Schott, CG&E, discussed Zimmer's initial plant staff training program, its requalification program, and its replacement training program (see Appendix XVI).

Mr. Okrent noted that since Zimmer is the first nuclear plant that the Applicant will operate, and since there is a relatively small number of personnel experienced in nuclear plant operations on board, he recommended that the Applicant appoint at least one experienced person onto the Offsite Review Committee.

I. Peltier noted that the Applicant meets ANSI Standard 18.1 for both operator and supervisor personnel training.

J. Schott also provided an organization chart for the operating force at Zimmer.

D. NRC Staff Report on the GE Mark II Containment System

C. Anderson, NRC Staff, discussed the state of the Staff's review of the Mark II Containment System, and the findings to date (see Appendix XVII). He noted that the review is essentially complete, and that although not all of the documentation is complete, the NRC Staff sees no problem with licensing this containment in this plant. He noted that the review is documented in NUREG-0487.

J. A. Kudrick, NRC Staff, noted that since no data were available on Mark II, the NRC Staff used conservative values. He said that the criteria will be revised when adequate new data are available.

I. Peltier said that the NRC Staff is satisfied that the Mark II containment is designed to accept pool dynamic loads.

E. General Questions

J. Kovacs, NRC Staff, said that the containment and its equipment will be evaluated on the basis of the absolute sum method of calculation. Some components, that do not meet the criteria using absolute sum calculations, but do meet these criteria using square root of the sum of the squares calculations, will be considered case by case.

Mr. Okrent requested that information be provided him regarding the capability of instrumentation to follow the course of an accident to measure radiation levels inside containment for three postualted events: 1) Class-9 Accidents, 2) 10 CFR 100 Accidents, and 3) a best estimate for a LOCA. He also asked for estimated radiation levels for each of these events. J. Flynn agreed to provide this information.

Mr. Ray requested information on grid load flow and stability in CG&E's electric distribution system, and asked for a copy of the load swing curve for the most marginal case for grid operation (the slowest rate of return to stable conditions following a loss of Zimmer from the grid). Mr. Ray also requested a diagram of CG&E's 345 kv network. J. Flynn agreed to provide this information.

In answer to a question regarding settlement of structure, I. Peltier said that differential settlement of the base mat is not a major problem. Differential settlement between adjacent buildings should and will be monitored. He noted that measurements made on March 1 were of the same value as those made in December.

In answer to a question, I. Peltier said that there are two separate remote shutdown panels. These panels can be used only for shutdown.

J. Borgmann said that CG&E believes that it has a solid power plant operating background, and that it has adequate staffing for the Zimmer plant. CG&E has a commitment to, and will provide efficient, safe operation.

F. Caucus

Members were polled, and agreed that they could try to write a report on the Zimmer Nuclear Power Station Unit 1. Members identified items that they believed should be included in the report.

IV. Meeting With Members of the NRC Staff on Recent Operating Experience, Licensing Activities, and Future Agenda (Open to Public)

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

A. Dresden 2: Technical Specification Change to Allow Temporary Operation With Detached Blow-Out Panels

P. O'Connor, NRC Staff, discussed the circumstances surrounding a temporary Technical Specification amendment issued for operation of Dresden 2 with blow-out panels detached. These panels had become detached from the reactor building framing in the area above the refueling floor on Friday, February 2, 1979.

This detachment resulted in an opening in the reactor building of about 22 feet by 40 feet. Repairs could not be completed by Sunday, February 4, and the Licensee requested an emergency 24 hour Technical Specification amendment to allow this unit to commence startup operations before the panels were replaced. The volume beneath the refueling floor was temporarily sealed and internal pressure in this volume was maintained at a negative pressure (-0.2 in. water) while the repairs to the reactor building were completed before expiration of the 24 hour period. The cause of the blow-out panel detachment is still under investigation.

P. O'Connor noted that at the time of the event, Unit 2 was operating at 700 MWe, and Unit 3 was in a cold shutdown condition. Unit 2 was immediately brought to a cold shutdown condition in accordance with Technical Specification requirements. (For additional details, see Appendix XVIII.)

B. Procedures for Imposing Civil Penalties on Licensees for Violations of Requirements

[Note: At the 226th meeting, the Committee heard a report from the Office of Inspection and Enforcement (I&E) concerning a fine imposed against the Wisconsin Public Service Corp. for an incident at the Kewaunee Nuclear Power Plant during which a shift refueling supervisor received an exposure of 2900 mrem when he entered the reactor cavity without a radiation work permit, and without a health physics escort. For background on this matter, including pertinent sections of the Code of Federal Regulations and an information report to the Commission, see Appendix XIX.]

E. Jordan, NRC Staff, discussed the procedures used by I&E in imposing civil penalties on licensees for violations of NRC requirements, and also discussed the avenues of appeal available to the licensees (see Appendix XX).

C. Non-Proliferation Alternative Assessment Program (NASAP)

1. Overview

R. Hartfield, NRC Staff, provided an overview of both the NASAP and International Fuel Cycle Evaluation (INFCE) programs (see Appendix XXI).

2. Office of Nuclear Reactor Regulation Participation

J. Meyer, NRC Staff, discussed the areas of participation of the Office of Nuclear Reactor Regulation in the NASAP and INFCE programs (see Appendix XXII).

In answer to a question, J. Meyer said that no effort is currently being used to develop regulatory requirements for a non-proliferation fuel cycle.

3. Office of Reactor Safety Research Participation

C. Kelber, NRC Staff, noted that $800,000 for Advanced Reactor Safety Research has been requested to be reprogrammed for the fiscal year 1979 budget. This request is currently before Congress, and the money is not yet available. Current research work is limited to the scope of the problems that have been identified by the NRC Staff as key factors in their evaluation. The Office of Reactor Safety Research will draw upon resources already available in the current program for these purposes.

C. Kelber said that the main area for research in the case of LWRs is improved fuel management by multiple batch reloading cycles. He suggested that if reloading time could be reduced, less burnup would be required to provide equivalent plant availability.

C. Kelber said that if the $800,000 becomes available, RES will focus on three new concepts: the heavy water reactor, the light-water breeder reactor, and the gas-cooled fast reactor. He noted that some of the liquid metal fast breeder reactor work that is being done is also applicable to the gas-cooled fast reactor. He said also that the Staff has identified two key issues for early scoping with respect to the heavy-water reactor (HWR), and that RES will attempt to provide the technical assistance to the NRR Staff. RES will attempt to identify the potential safety issues concerning HWR pressure-tube leaks before breaks, and its relationship to current U. S. licensing criteria.

He noted also that since six months of the current fiscal year have already passed, it is unlikely that all of the requested research funds could be spent this year even if they do become available. He said that there has been much cooperation between the heavy-water reactor vendor in this country, Combustion Engineering, and Atomic Energy of Canada, Limited.

With regard to the gas-cooled fast reactor, RES is reviewing past work and issues that have been identified by the vendor, General Atomic, and in addition, RES plans to provide technical assistance using LMFBR Programmatic support to some of the more conventional accident analyses.

C. Kelber said that RES also plans to support the efforts of the Office of Nuclear Material Safety and Safeguards (NMSS).

4. Office of Nuclear Materials Safety and Safeguards Participation

K. Black, NRC Staff, discussed the participation of NMSS in the NASAP and INFCE programs (see Appendix XXIII).

Mr. Okrent requested that safety-related reports issued by this program be provided to the Committee. J. Meyer, NRC Staff, agreed to provide the reports. Mr. Okrent also requested that the Committee be provided with six copies of the letter setting up NASAP, the draft and final version of the NASAP report, and the public and agency comments on the report.

D. Proposed Department of Transportation (DOT) Criteria for Shipment of Nuclear Materials

R. Bernero, NRC Staff, discussed the proposed DOT criteria and rules relating to the shipment of radioactive materials (see Appendix XXIV). He noted that these criteria and rules may impact on some of the operations of the NRC.

It was the Committee's consensus that it should not plan to review additional matters related to the transportation of radioactive materials and to relations between the NRC and the DOT, unless the Commission sees compelling reasons for further ACRS involvement.

E. Future Agenda

The Committee agreed upon a future agenda for the review of cases (see Appendix II).

1028 049

V. Executive Sessions (Open to Public)

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

A. Meetings with Foreign Safety Groups

The Chairman noted that the ACRS's foreign travel budget will support only one foreign trip to meet with a safety committee during the current fiscal year. It was the consensus of the Committee that the trip to Japan, scheduled for April, would be the most useful to the Committee at this time.

B. Status of Generic Items Relating to Light-Water Reactors: Report No. 7

The Committee reviewed its report, Status of Generic Items Relating to Light-Water Reactors: Report No. 6 and declared the following items in this report resolved:

- II-5A, Monitoring for Loose Parts inside the Pressure Vessel;
- IIB-2, Qualification of New Fuel Geometries;
- IIC-6, Maintenance and Inspection of Plants; and
- IID-1, Safety-Related Interfaces Between the Reactor Island and the Balance-of-Plant.

The Committee referred to the appropriate subcommittee, for the development of a position for Committee consideration, the following items from the above report:

- II-2, Effective Operation of Containment Sprays in LOCA (to Radiological Effects and Site Evaluation Subcommittee) and
- II-A-4, Periodic (10 yr.) Review of all Power Reactors (to Reactor Operations Subcommittee).

The Committee agreed to combine items II-8, BWR Pump Overspeed During a LOCA and IIA-2, PWR Pump Overspeed During a LOCA.

The Committee prepared its report, Status of Generic Items Relating to Light-Water Reactors: Report No. 7 (see Appendix XXV). During the preparation of this report, the Committee approved a revised numbering system by which the generic items are identified. The Committee also agreed to consider, at a later meeting, abolition of its generic items report, and a merger of the generic items report with the NRC Staff's Task Action Plans. The Procedures Subcommittee will consider the suggestion that the Committee no longer reference generic items as such in its reports on specific projects.

C. Report to Commissioners Regarding Combination of Loads as a Design Basis For Nuclear Facilites

Mr. Bender recalled that he had been requested by Commissioner Kennedy to prepare a report regarding the basis for combining loads under accident conditions in nuclear power stations. He identified the problems and complexities in determining for structural analyses the input loads and loading conditions that should be considered to meet accident conditions (see Appendix XXVI). He also described the NRC Staff's load combination evaluation program. He suggested that a proper evaluation of this complex problem was a greater than one man task.

The Committee established an ad hoc subcommittee to consider this generic item, consisting of Mr. Bender, Chairman, and Messrs. Okrent, Plesset, Shewmon and Siess. A letter to the Commissioners was prepared informing them that this ad hoc subcommittee has been formed to pursue the matter of combination of dynamic loads as a design basis for nuclear facilities.

J. Knight, NRC Staff, indicated that the Staff would like to discuss the matter, the current NRC Staff positions, and the history of how the Staff arrived at these positions, sometime in the future. He noted that there is an NRC Staff task action plan regarding load combinations, and that it is among the top twenty items receiving high priority. The NRC Staff at this time is focusing attention on this plan which is being written.

Mr. Okrent suggested that it might be useful to make a probabilistic analysis of the events to determine what needs to be done. He noted that the current methods used by the NRC Staff are really judgmental (square root sum of the squares method) because there is no knowledge of the actual events as a function of time.

J. Knight requested that the NRC Staff be scheduled to meet with the Committee on this matter at the 228th ACRS Meeting, but the Committee declined the request and elected to keep its scheduled information meeting on pipe breaks.

D. The Role of the ACRS in the Regulatory Process

Members discussed the past and current role of the ACRS in the regulatory process, its relations to the Commissions it has advised (AEC and NRC), the perceptions it has had over the years regarding its approach to safety matters, its changing relations with the NRC Staff (and the AEC Regulatory Staff), and how it should operate and approach safety matters in the forseeable future. Members suggested the following:

- The Committee should examine the depth and breadth of its review process to determine if major problems (e.g., stress corrosion cracking) are given appropriate attention. Are problems that are identified by reactor operation given adequate attention by the Committee?

- The Committee should take a broad view of its charter with respect to the area that it surveys to identify significant safety issues, but should be selective to allow for in-depth examination of important safety matters. Neither should the Committee attempt to assume the role of policy maker regarding a total safety philosophy.

- The mechanism by which matters worthy of consideration are selected needs to be reexamined as well as the procedures for members and/or subcommittee chairmen to pursue items of concern. The Committee should identify and concern itself only with major safety issues and policy, and not become tangled in details that can be addressed better by the NRC Staff.

- The Committee should not duplicate the work of the NRC Staff, but should maintain cognizance of the capability of the NRC Staff and the quality of its work. The Committee should review NRC Staff safety and regulatory positions to assure that they "make sense in the real world".

. Specific areas of Committee interest included in the past, and also of current interest are listed below:

- ALARA,
- safety features of reactors,
- reactor pressure vessel integrity,
- continued development of codes and standards,
- inspection requirements,
- fracture mechanics,
- probabilistic methods,
- seismic dynamic analyses,
- hydrodynamic analyses,
- regulatory guides, Commission rules, etc.,
- project reviews,
- legislation related to, or impacting upon, safety, and
- waste management

. Additional areas for the Committee to consider are

- "acceptable risks" need to be considered (e.g., designate a subcommittee to pursue this).
- The safety design approach for LMFBRs should be considered (e.g., are any major design changes needed?).
- Are existing regulatory requirements appropriately based on current knowledge of reactor safety phenomena, methodology, experience, etc.?
- Should the ACRS exercise a more active role regarding the identification and resolution of safety issues?

1028 053

- An improved procedure is needed to permit Members to debate the issues among themselves, particularly where the NRC Staff and the vendors/applicants are in disagreement. Is too much time spent during ACRS meetings listening to presentations on items of an informational nature?

- Communications with the Commissioners need to be improved so that the ACRS is aware of current Commission policy and interests.

- What is the appropriate scope for ACRS activities? In some areas the Committee may have "stretched itself too thin". At the same time, questions related to items such as the reliability of off-site power supplies, other forms of energy generation, etc., are areas not being examined.

This subject was referred to the Procedures Subcommittee for further consideration.

E. Subcommittee Reports

1. Licensee Event Reports (LER) Subcommittee

Mr. Moeller, Subcommittee Chairman, noted that the Subcommittee met on March 1-2, 1979, to organize the effort in the review of LERs. The Subcommittee has recommended procedures for the review, a proposed meeting was scheduled, and a scope of the Committee's report was proposed (see Appendix XXVII).

2. Regulatory Activities

Mr. Siess, Subcommittee Chairman, noted the Subcommittee's recommendation, and the Committee concurred in the NRC Staff's regulatory position on the following regulatory guides:

- Regulatory Guides 1.137 (Rev.1), Fuel Oil Systems for Standby Diesel-Generators, and

- Regulatory Guide 1.143 (Rev.1), Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants (see Appendix XXIX).

1028 054

Mr. Siess noted difficulties with proposed Regulatory Guide 1.40 (Rev. 1), and the Committee recommended that the Regulatory Activities Subcommittee consider this Guide again after requested changes are made by the NRC Staff.

3. Improved Safety Systems

Mr. Siess reported to the Committee on discussions held with the NRC Staff by the Improved Safety Systems Subcommittee. He noted that RES has requested for fiscal year 1979 a $800,000 budget, $400,000 to come from reprogramming funds, and $400,000 from unspent funds from other projects. At this time, no funds have been made available.

RES has proposed the following research program and budget:

1. Vented Containments - $300,000

2. Alternate Heat Removal Systems - $200,000, and

3. Value Impact Methodology - $300,000.

Mr. Siess noted that DOE has a $4 million budget for improved reactor safety research, and that RES will try to coordinate its programs with DOE.

4. Reactor Safety Research Subcommittee

Mr. Siess, Subcommittee Chairman, said that the Subcommittee would try to provide information regarding the ACRS recommendations on the RES budget to the Commission by July as requested by Commissioner Gilinsky. He suggested that each working group in the Subcommittee review their appropriate subject areas and recommend priorities. This information needs to be obtained by April or early May, and will have to be considered at the June Subcommittee meeting. He suggested that the Committee's report to the Commission might take the form of an interim report to be completed in July, and to be presented orally to the Commissioners in either July or August. He said that it will be necessary to obtain budget information from the NRC Staff as soon as possible.

F. Meeting with Office of Inspection and Enforcement

The Committee agreed to schedule for the 228th ACRS Meeting (April) a meeting with I&E management to further discuss I&E policies with respect to the imposition of fines and other civil penalties.

G. ACRS Reports and Letters

1. Status of Generic Items Relating to Light-Water Reactors: Report No. 7

The Committee prepared its report to the Commissioners, Status of Generic Items Relating to Light-Water Reactors: Report No. 7 (see Appendix XXV).

2. Combination of Dynamic Loads as a Regulatory Design Basis

The Committee prepared an interim letter to the Commissioners informing them that the Committee is considering the generic item, Combination of Dynamic Loads as a Design Basis for Nuclear Facilities (see Appendix XXVIII).

3. Regulatory Guides

The Committee approved a memorandum to the NRC Executive Director for Operations informing him that the Committee concurs in the NRC Staff's regulatory position of the following Regulatory Guides:

- Regulatory Guide 1.137 (Rev. 1), Fuel Oil Systems for Standby Diesel Generators, and
- Regulatory Guide 1.143 (Rev. 1), Design Guidance for Radioactive Waste Management Systems, Structures, and Components Installed in Light-Water-Cooled Nuclear Power Plants (see Appendix XXIX).

4. Requirements for Shutdown and Decay Heat Removal Using Safety-Grade Equipment

The Committee approved a memorandum to the NRC Executive Director for Operations recommending that a limited probabilistic study be made to develop information for the evaluation of the NRC Staff's requirements for achieving cold shutdown and decay heat removal through the use of safety-grade equipment (see Appendix XXX).

1028 056

5. Transportation of Radioactive Materials

The Committee prepared a memorandum to the NRC Executive Director for Operations recommending that the NRC Staff assure that information concerning risks from both radioactive and non-radioactive shipments be made available to the NRC and the Dept. of Transportation (DOT) for use in their respective studies of highway routing regulations (see Appendix XXXI).

6. Transportation of Radioactive Materials, ACRS Participation

The Committee prepared a memorandum to R. Bernero, NRC Staff, informing him that the Committee does not plan to review additional matters related to transportation of radioactive materials and to relations between the NRC and the DOT, unless the Commission sees compelling reasons for further ACRS involvement (see Appendix XXXII).

VI. Executive Sessions (Closed to Public)

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

A. New Members

The Committee agreed to propose the names of [redacted] and [redacted] to the Commissioners for nomination to fill the current vacancy on the Committee.

B. William H. Zimmer Nuclear Power Station Unit 1

The Committee prepared a report informing the Commissioners that it believes that, subject to certain specified conditions, the William H. Zimmer Nuclear Power Station Unit 1 can be operated without undue risk to the health and safety of the public (see Appendix XXXIII).

The 227th ACRS Meeting was adjourned at 3:45 p.m., Saturday, March 10, 1979.

oOo

ACRS
1622

APPENDIXES
TO
MINUTES OF THE 227TH ACRS MEETING
MARCH 8-10, 1979

227th ACRS Meeting

Meeting Dates:
March 8-10, 1979

APPENDIX I

ATTENDEES

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Max W. Carbon, Chairman
Milton S. Plesset, Vice-Chairman
Myer Bender
Harold Etherington
William Kerr
Stephen Lawroski
J. Carson Mark
William M. Mathis
Dade W. Moeller
David Okrent
Jeremiah J. Ray
Paul G. Shewmon
Chester P. Siess

ACRS STAFF

Raymond F. Fraley, Executive Director
Marvin C. Gaske, Assistant Executive Director
James M. Jacobs, Technical Secretary
Herman Alderman
John H. Austin
Andrew L. Bates
Paul A. Boehnert
Sam Duraiswamy
Elpidio G. Igne
Morton W. Libarkin
Richard K. Major
Thomas G. McCreless
John C. McKinley
Robert E. McKinney
Ragnwald Muller
Gary R. Quittschreiber
Jean A. Robinette
Richard P. Savio
Hugh E. Voress
Robert L. Wright

CONSULTANTS

E. P. Epler
W. Lipinski
S. Ditto
Z. Zudans

NRC ATTENDEES

227TH ACRS MTG.

March 8, 1979

Div. of Project Management

L. P. Crocker
S. Varga

Div. of System Safety

W. Minners
J. Knight
T. M. Novak
A. C. Thadani
T. M. Su
R. L. Tedesco
D. F. Thatcher
F. C. Cherry
J. Kovacs

Div. of Operating Reactors

C. Wichman
V. L. Rooney
P. O'Connor

Nuclear Reactor Regulation

F. Schroeder
S. Hanauer
C. H. Berlinger

Office of Stds. Development

G. A. Norberg

NRC ATTENDEES

227TH ACRS MTG.

March 9, 1979

Div. of Project Management

J. F. Stolz
J. N. Wilson
I. A. Peltier
R. Trevino
D. B. Vassallo
L. P. Crocker

Div. of Systems Safety

N. H. Wagner
J. Kudrick
L. Ruth
C. Anderson
R. L. Tedesco

Div. of Site Safety & Env. Analysis

N. A. Eisenberg
D. O. Nellis
F. J. Hebdon

Inspection & Enforcement

E. L. Jordan

Safeguards
C. Sawyer

Nuclear Regulatory Research

C. N. Kelber
W. Lahs

Of. of Stds. Development

R. M. Bernero

Executive Legal Director

J. Lieberman
S. Burns

I&E, Region III

J. Menning

ICSB

R. F. Scholl

Div. of Operating Reactors

R. Clark
F. Pagano
J. Millex

Nuclear Material Safety and Safeguards

J. Giarratana
K. Black

MPA

R. A. Hartfield

ARB

J. Long

Nuclear Reactor Regulation
D. F. Rou
J. F. Meyer
P. F. Riehm

APPLICANT ATTENDEES

227th ACRS MEETING

March 8, 1979

Vermont Yankee

C. Sayha
R. E. Sayla

Toledo Edison

T. Myers

Stone & Webster

T. Myers
D. Jaqnetts

Northwest Utilities

W. Romberg

General Electric

Elwood P. Stroupe
J. V. Woodford
L. J. Sobon
E. C. Eckert
W. P. Sullivan
A. L. Armih
J. V. Woodford
H. C. Pfefferlen

Sargent and Lundy

R. M. Crawford
G. T. Kitz

Westinghouse
B. D. Sloane
R. W. Steitler

Bechtel
N. Willoughby

SCS
R. Soyle

Gulf States Utilities

J. Leavines

Duke Power

R. Wardell
W. H. Owen

Combustion Engineering

W. E. Burchill

WPPSS

G. C. Sorensen

Babcock & Wilcox

J. H. Taylor
A. McBride

AIF

F. T. Stetson

PSE&G

C. W. Vepreb

TVA

J. A. Domer

JCP&L
K. R. Goddard

Boston Edison
C. S.Ondash

A-4

APPLICANT ATTENDEES

227TH ACRS MEETING

March 9, 1979

Stone and Webster

G. F. Dawe
F. Ogden
L. L. Dietrich
R. E. Cotta
G. T. Kitz
R. M. Crawford
C. N. Krishnaswamy
R. J. Pruski
R. L. Givan
R. F. Scheibel
M. E. Jackson
S. Rurka
A. E. Meligi

General Electric

R. Villa
S. Mark
T. Mark
B. E. Woodward
W. E. Smith
R. B. Johnson
E. Carroll
L. J. Sobon

Cincinnati Gas & Electric

E. A. Borgmann
J. D. Flynn
J. J. Seibert
J. R. Schott
W. W. Schwiers
J. C. Herman
H. Brinkmann

PUBLIC ATTENDEES

227TH ACRS MEETING

March 8, 1979

P. M. Abraham, Duke Power Co., Charlotte, NC
R. Borsum, B&W, Derwood, MD
W. W. Bowers, Philadelphia Electric Company, Media, PA
Charles Brinkman, Combustion Engr., Gaithersburg, MD
C. P. Chen, PASNY, New York, NY
Craig Grochmal, Stone & Webster, 7315 Wisconsin Av., Bethesda, MD
Hiroyoshi, Hamada, The Tokyo Electric Power, 1901 L St., NW, Wash., DC
Richard A. Hill, General Electric, Ben Lomand, Ca
Roger W. Huston, Consumers Power Co., Jackson, MI
Richard B. Johnson, GE, 175 Carmer, San Jose, CA
Vincent P. Manno, Am. Electric. Power Service, Corp., NY, NY
Robert L. McGuinness, Northeast Utilities, 170 Rolling Hill Rd. Southington, CT
Frank McPhatter, B&W, Madison Hights, VA
T. D. Martin, NUTECH, Vienna, VA
R. C. L. Olson, Baltimore Gas & Electric Co., Lutherville, MD
S. L. Rosen, Boston Edison, Waban, MASS 02168

PUBLIC ATTENDEES

227TH ACRS MEETING

March 9, 1979

Walter Batchelor, Am. Psychological Assoc., Wash., DC
R. Borsum, B&W, Derwood, MD
Troy B. Conner, Jr., CG&E, Rockville, MD
J. P. Morin, LILCO, Hamppange, NY
W. J. Museler, LILCO, 105 Scraggy Hill, Pt. Jefferson, NY 11777
Joseph P. Novarro, LILCO, Wading River, NY
James Rivello, Long Island Light Co., Shoreham, NY
A. R. Smith, General Electric Co., San Jose, CA
Michael Stern, Wisconsin Public Service Corp., Green Bay, Wis.
J. E. McEwen, Jr

PUBLIC ATTENDEES

227TH ACRS MEETING

<u>Saturday, March 10, 1979</u>

Nancy B. Willoughby, Bechtel
Richard Aaron, Self
Mark Wetterhahn, CM&C
L. S. Gifford, GE

A-8

APPENDIX II

ACRS FUTURE AGENDA 3/2/79

ACRS MEETING PROJECT	TYPE OF REVIEW	REACTOR VENDOR	SER ISSUE DATE
APRIL			
SEQUOYAH 1 & 2	OL	W	3/2/79
PALO VERDE 4 & 5	CP	CE	3/1/79
MAY			
MILLSTONE 2	STRETCH POWER	CE	4/2/79
JUNE			
NONE			
JULY			
SHOREHAM	OL	GE	6/1/79
LASALLE 1 & 2	OL	GE	6/1/79
FNP 1-8	ML	W	6/1/79
AUGUST			
MIDLAND 1 & 2	OL	B&W	7/2/79
SAN ONOFRE 2 & 3	OL	CE	7/2/79
WATTS BAR 1 & 2	OL	W	7/2/79

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

APPENDIX III

March 10, 197

ACRS Members

SCHEDULE OF ACRS SUBCOMMITTEE MEETINGS AND TOURS

The following is a list of tours and Subcommittee meetings currently scheduled, subject to the approval of the Advisory Committee Management Officer. If you are listed and cannot attend a meeting, or if you are not listed but would like to attend, please advise the ACRS Office as soon as possible.

Most hotels currently being used by ACRS Members in the downtown Washington and Bethesda areas require a guaranteed reservation if arrival is scheduled after 6:00 p.m. Failure to use a room under these conditions involves forfeiture of the cost. Please advise the ACRS Office as soon as possible if you cannot attend a meeting for which you are scheduled so that reservations can be cancelled in time to avoid this.

MW Libarkin

M. W. Libarkin
Assistant Executive Director
for Project Review

cc: ACRS Technical Staff
M. E. Vanderholt
B. Dundr
R. F. Fraley
M. C. Gaske

A-10

1028 060

MARCH

12	Sequoyah Nuclear Power Station (RS) - CM, WM, CS
15	Ft. St. Vrain, Longmont, CO (JCM) - CS, MC, WM, PS
19-20	ECCS (Los Angeles, CA) (AB) MP, HE, DO
23-24	LER's (AB) - DM, MB, HE, SL, WK, CM, WM, JR
26	Trp. Aboard Nuclear-Powered Surface Ship (RFF/GRQ) MC, HE, SL, JM, DO, CS, WM
29	Palo Verde, Units 4 and 5 (Phoenix, AZ) (GRQ/PB) - PS, MC, WM, DO, JR
30	Power & Electrical Systems (Phoenix, AZ) (GRQ) - WK, CM, WM, DO, JR

APRIL

4	Regulatory Activities (A.M.) (GRQ/SD) - CS, MB, HE, WK, DM
4	Consideration of Class 9 Accidents (P.M.) - (GRQ) - WK, MB, SL, DM, DO, CS
5-7	228th ACRS Meeting
14-22	Trip to Japan - (RFF) - MWC, SL, JCM, MP, PS, CPS
18-20	Waste Mgt. (Hanford, WA) - (RM) - DM, WK, WM, JR
26-27	LER's (AB) - DM, SL, WK, JR, WM, HE, JCM

(Continued)

MAY

4	ECCS (Los Angeles, CA) (AB) - MP, DO
9	Reg. Activities (A.M.) (GRQ/SD) - CPS, MB, WK
9	Dynamic Load Combinations (P.M.) (EI) - MB, CS, PS
9	Reactor Operations (Millstone 2 Str. Pwr.) (RKM) - HE, WM, DO, JR
10-12	229th ACRS meeting
17	Fluid Dynamics (AB) - MP, HE, CS
24-25	LER's (AB) - DWM, SL, WK, JR, WM, HE, JCM, MP

Note: A meeting of the Subcommittee on Safeguards & Security is being planned for late May.

ADDITIONAL SCHEDULED LER SUBCOMMITTEE MEETINGS

June 28-29 - LER's (AB) - DM, SL, WK, JR, WM, HE, JM, MP

July 19 - LER's (AB) - DM, SL, WK, JR, WM, HE, JCM, MP

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

March 7, 1979

APPENDIX IV
ATWS: STATUS REPORT

ACRS Members

ATWS PLANT MODIFICATIONS

I have attached for your information three tables that may be helpful during the Committee's review of NUREG-0460. The first table has the Staff ATWS requirements, the second has the vendors' recommendations and the third table has the ACRS consultants' comments. I must assume the responsibility for the information on the second and third tables. These are entirely my opinion of what the vendors and consultants were saying.

Thomas G McCreless

Thomas G. McCreless, Chief
Project Review Branch No. 2

Attachments As Stated

NRC Staff Requirements

TABLE 1

Alternate Plant Modifications

	1	2	3	4
B&W	Nothing	. BUSS[2] . AMSAC[3]	. BUSS[2] . AMSAC[3] . Analysis[8]	. AMSAC[3] . Add safety valves . Analysis[8]
CE	Nothing	. SPS[2] . AMSAC[3]	. SPS[2] . AMSAC[3] . Analysis[8]	. AMSAC[3] . Add safety valves . Analysis[8]
W	Nothing	. AMSAC[3]	. AMSAC[3]	. AMSAC[3] . Analysis[8]
GE	Nothing	. ARI[2] . SD[7] . RPT[1] . Logic[4]	. ARI[2] . RPT[1] . Logic[4] . Automatic 86 gpm SLCS[5] . SD[7] . Analysis[8]	. RPT[6] . Automatic, high capacity liquid poison injection . Analysis[8]

[1] The approved Monticello design is an acceptable RPT design for all BWR 4 plants. The approved Zimmer design is an acceptable RPT design for all BWR 5 and 6 plants. There may be other acceptable designs which must be treated on a plant specific basis.

[2] A system which is diverse and independent from RPS, meeting IEEE-279 and acting as backup to the electrical portion of the current scram system.

[3] ATWS mitigating system actuation circuitry satisfying criteria in Appendix C.

[4] Changes in logic to reduce vessel isolation events and permit feedwater runback.

[5] Modified SLCS piping to assure delivery of 86 gpm of poison and automatic actuation circuitry satisfying parts A through H of Appendix C with reliability equivalent to the mechanical portion of the SLCS.

[6] Recirculation pump trip satisfying criteria in Appendix C.

[7] Modification of scram discharge volume.

[8] Analysis remains to be performed and reviewed to confirm expected mitigation capability as described in Sections 2.2 and 2.3.

Vendor Recommendations

	1	2	3	4
B&W	ATWS is not a safety problem; diverse instrumentation could be added to future plants.			
CE	ATWS is not a safety problem.			
W	-	AMSAC	AMSAC	AMSAC
		(Same for Alternate 2, 3 and 4)		
GE	-	Same as NRC; Should suffice for all current reactors	Similar to NRC: No need to automate SLCS; Should suffice for all future reactors	Unnessary; too costly and defeats Standardization

1028 074

Consultants' Recommendations

PWRs:

E. Epler and S. Ditto:	Both agree with the NRC Staff alternates; Ditto recommends re-examination of criteria of when a plant should use Alternate 3 or 4.
J. Lee	Supports Alternates 2 and 3; Alternate 4 is unnecessary.

BWRs:

E. Epler

- In all of the alternatives the RPT must work. Presently the RPTs are activated by high pressure or low water; a low water indication is not expected to be very effective for some ATWS and and pressure sensors may be damaged by the high pressures of an ATWS.
- A good manual scram is also necessary. ARI is an important feature. Scram discharge volume is a potential weak point.
- Manually actuated liquid poison system would be worthless but an automated system is currently unattainable.
- Based on the above:
  - Alternate 2 is acceptable for existing plants.
  - New fix is needed for future plants.

S. Ditto

- ATWS should not be treated as a DBE as ATWS is a class of events and each event should be examined separately.
- ARI appears to be a good fix. A manually operated scram wired directly to the air-dump valves would be better than ~~ARI.~~ automatic poison injection.
- Scram discharge volume is a weak poin in scram system.
- Considering the above:
  - Alternate 2 is acceptable for existing plants.

W. Lipinski

- Encourages improvements to plant protection systems but doubts that statistical adequacy can be shown.
- SLCS should not be automated.
- From a cost/benefit standpoint, it is unnecessary to back fit the 11 oldest plants which are subject to Alternate 2.

J. Lee

- Supports Alternates 2 and 3; however, would like to see additional analyses comparing 43 vs. 86 gpm SLCS. Alternate 4 is unnecessary.

Westinghouse Electric Corporation

Power Systems Company

February 28, 1979
NS-TMA-2046

Mr. Harold Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

SUBJECT: Anticipated Transients Without Scram

Dear Mr. Denton:

We are writing in response to the Commission's notice in 44 Fed. Reg. 6816 (February 2, 1979), which states that the Regulatory Requirements Review Committee (RRRC) has issued recommendations concerning Anticipated Transients Without Scram (ATWS), and which affords the public an opportunity to appeal those recommendations prior to a decision on implementation. This letter initiates our appeal from the RRRC recommendations.

The existing ATWS mitigation equipment on Westinghouse plants provides reasonable assurance that the health and safety of the public is protected. The addition of new equipment to these plants will not provide any significant increase in protection which would be sufficient to justify the additional cost. Moreover, there is no "substantial, additional protection afforded," as required by Commission regulations, to support a decision to backfit.

In view of the public notice, we were distressed to receive a letter from Dr. Mattson, dated February 15, 1979, two weeks before the return date for appeal, stating:

> "The Regulatory Requirements Review Committee has concurred with the generic analysis approach and the Director of the Office of Nuclear Reactor Regulation has authorized the Staff to proceed".

If the Staff has been given authorization to proceed, the Commission's notice of opportunity to appeal has been, in effect, nullified.

There are several items regarding the decision of the RRRC which we consider to be detrimental. Our comments and recommendations with respect to these are as follows:

1. With respect to Westinghouse plants, the same hardware is proposed for alternatives 2, 3 and 4. The implementation of alternative 4 requires more stringent analyses and criteria; but no corresponding decrease in ATWS risk is credited by the NRC. Westinghouse has already submitted analyses which are sufficient to demonstrate the adequacy of present ATWS mitigation systems to meet the requirements of alternative 4. Clearly, alternative 4, with increased analytical costs and with no lessening of ATWS risk, cannot be shown to be cost effective. No additional analyses should be required to demonstrate compliance with alternative 4 unless Westinghouse implements substantive NSSS design changes.

2. Appendix F of Volume 3 (NUREG-0460) shows a significantly lower ATWS risk for Westinghouse plants as compared to the risk for other Vendors' plants. For example, plants after Calvert Cliffs #2 (alternative 3), the risk calculated for Westinghouse is a factor of ten (10) lower than for BWR's and a factor of forty (40) lower than for other PWR vendors. Because of the large number of Westinghouse units (38% of total) and the fact that Westinghouse plants pose a much lower risk than others, a low total risk to the public is a direct result of the conservative Westinghouse design. The Staff can only realize its total risk of 2×10^{-3} in 1990 based on requiring a "safer" Westinghouse plant than all the other vendors. We realize that the risk analysis done by the Staff is preliminary; however, advantages of the Westinghouse design over other vendor's is expected to persist assuming the Staff does a more detailed calculation. The Staff should develop consistent requirements for all vendors. This should be done for Westinghouse plants by not requiring alternative 4 for any plant and by relaxing the reliability goal for the proposed hardware modifications.

3. Despite the NRC's policy to the contrary, Volume 3 conclusions and the implementation guidelines endorsed by the RRRC are made without the benefit of an appropriate value-impact analysis of the alternatives considered. The Staff's attempt at such an analysis in Volume 2 of NUREG-0460 was replete with shortcomings. In Volume 3 (Page 44), the Staff rejects the use of value-impact analysis and replaces it with their "engineering judgment". This is unacceptable in view of the long history of the ATWS issue and the Commission's stated policy to require completion of a value-impact analysis prior to implementation of any significant new requirements. Moreover, there has been no showing that an ATWS event in a Westinghouse plant can lead to any severe condition such as core melt. Because ATWS events in Westinghouse plants do not lead to severe conditions, the value of preventing such an event is moot; hence, no cost impact can be justified. Consquently, the value of reducing the consequences of ATWS is zero for Westinghouse plants; therefore, the impact should also be zero. Westinghouse believes that each of the proposed four alternatives must be justified by appropriate value-impact analysis.

4. Volume 3 of NUREG-0460 proposes an implementation policy that is in direct disagreement with the goals of standardization as stated in NUREG-0427. For example, plants with CP's after January 1, 1978 that replicate plants with CP's before January 1, 1978 will be required to have additional hardware and a more stringent analysis basis. Westinghouse believes that replicate plants should meet identical requirements.

In summary, we do not believe that any of the alternatives are required for protection of the health and safety of the public. If the Commission persists in action to impose ATWS requirements, they should not be backfit to plants under construction or to standard plants with PDA's. With regard to requirements for future plants, we have the following recommendations.

1. Alternative 2 should be adopted on a generic basis for future Westinghouse plants and additional generic analysis should only be required as necessary to confirm continued adequacy if there are substantive changes in the Westinghouse design,

2. Each vendor should be required to provide the same level of protection,

3. An appropriate value-impact analysis should be completed prior to a decision to justify the level of protection ultimately required, and

4. Approved standard designs should be governed by the Commission's standardization policy absent a finding of a requirement for substantial additional protection required to justify backfit.

Westinghouse would be pleased to discuss these concerns and our recommendations with you and your staff.

Very truly yours,

T.M. Anderson

T. M. Anderson, Manager
Nuclear Safety Department

/rd

cc: Prof. William Kerr
Mr. Carsen Mark

Consumers Power Company

COPY

General Offices: 212 West Michigan Avenue, Jackson, Michigan 49201 • Area Code 517 788-0550

March 2, 1979

RECEIVED
ADVISORY COMMITTEE ON
REACTOR SAFEGUARDS U.S. N.R.C

MAR 8 1979

AM PM
7,8,9,10,11,12,1,2,3,4,5,6

Mr H R Denton, Director
Office of Nuclear Reactor Regulation
US Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr Denton

This letter provides comments for your consideration concerning the Regulatory Requirements Review Committee (RRRC) recommendations dealing with Anticipated Transients Without Scram (ATWS). This letter is submitted in response to the notice appearing in the Federal Register, February 2, 1979.

The RRRC endorsed the recommendations contained in Volume 3 of NUREG 0460 concerning ATWS. Consumers Power Company is deeply concerned that these recommendations will fail to close this long-standing issue satisfactorily. In addition, Consumers Power Company considers that the cost of plant modifications which would be required as a result of these recommendations is not commensurate with the benefits to be gained.

Consumers Power Company's concerns that the actions recommended by the RRRC will not resolve this issue are based on the requirements for additional analyses and have been heightened by NRC correspondence originating after the RRRC deliberation. It is not clear that more analyses, as required by NUREG 0460, Volume 3, will finally answer all of the Staff's questions on plant capabilities with respect to mitigation. A copy of the requirements concerning the generic analyses to be performed for plants for which Alternative 3 of NUREG 0460 is applicable has been received only in the last few days and has been only briefly reviewed. These requirements were formulated following RRRC review of NUREG 0460, Volume 3. Despite the brevity of review, the following points can be made:

1. The extent and detail of work required to comply with the request is overwhelming. The requested submittal dates indicate no comprehension of the magnitude of work involved.

2. In light of the extent of analysis required, the NUREG 0460, Volume 3, treatment of ATWS as compared to that which would be required if it was classified as a design basis accident (DBA) appears subtle. It would

A-22

1028 [illegible]

appear, contrary to Volume 3, that the NRC staff considers ATWS a DBA in everything except name.

3. There is no assurance that this analysis will finally resolve this issue, or that it will not simply generate requests for still further information. This concern is caused by a lack of specific mitigation criteria or definition of an ATWS event.

4. Requirements for the plant-specific analyses for those plants for which Alternative 2 is applicable are still undefined. The example posed by the Alternative 3 requirements does not encourage us to expect realistic requirements for these Alternative 2 analyses.

More important than these comments, however, is the basic fact that the requirement for these analyses presupposes that an ATWS event contributes significantly to overall public risk. Consumers Power Company considers that this assumption is incorrect, and that elimination of this predisposition makes mitigative analytical work unnecessary.

Consumers Power Company considers that analyses and studies performed by reactor vendors and EPRI have demonstrated that the probability of an ATWS event is much lower than that considered in NUREG 0460. These analyses and studies have been previously submitted to the NRC staff. Even without consideration of this area of disagreement, however, the expensive modifications recommended in NUREG 0460 appear unnecessary in consideration of the consequences of an ATWS event:

1. In a PWR, peak pressures following an ATWS event in which all rods fail to insert (a small subset of the already low probability event) would in most instances be of a magnitude only slightly above hydrostatic test pressure and would exist for only brief periods (on the order of half a minute). The assumption inherent in NUREG 0460 that these pressures completely incapacitate emergency cooling system isolation valves is unjustified. In fact, Consumers Power Company concludes the emergency core cooling system could be relied on following such an ATWS event and would be available, along with the steam generators, to cool the plant. Therefore, postulating a core melt as a likely consequence of this scenario is without basis.

2. Our Big Rock Point BWR is of a somewhat unique design. It incorporates relief valve capacity sufficient to prevent pressures following an ATWS event from exceeding the highest relief valve setting. Thus, a rupture of the primary coolant system or incapacitation of emergency cooling isolation valves would not occur. An ATWS event would not automatically lead to core melt as inferred in NUREG 0460 for all BWRs.

3. The staff has been extremely reluctant to grant credit for operator action to mitigate ATWS. In fact, Consumers Power Company considers that prompt operator action would provide significant mitigation. Operator response to any scram, regardless of ATWS considerations, is to observe control rod position indication and, if appropriate, initiate

a separate scram. This action would terminate most ATWS events. In the event this effort also fails, the indications that an ATWS event has occurred would be numerous. Some obvious indications would be control rod position indication, reactor power and pressure indications and the noise associated with relief valve operation. Our experience has indicated that operators would clearly recognize that an abnormal situation existed and their immediate actions would be directed toward effecting plant shutdown. Consumers Power Company concludes that operator action (within a few minutes) can be relied upon to safely mitigate this event, obviating the need for expensive plant modifications.

In conclusion, Consumers Power Company hereby "appeals" the recommendations of RRRC as invited in the Federal Register notice. Consumers Power Company encourages you to reject requirements for further analyses and high cost modifications since information already available to the Staff is sufficient to conclude that the probability of an ATWS event is already sufficiently small and its consequences would be acceptable.

David A Bixel (Signed)

David A Bixel
Nuclear Licensing Administrator

CC MCarbon, Chairman, ACRS

Atomic Industrial Forum, Inc.
7101 Wisconsin Avenue
Washington, D.C. 20014
Telephone: (301) 654-9260
Cable: Atomforum Washingtondc

AIF

March 2, 1979

Mr. Harold Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Harold:

The purpose of this letter is to appeal the decisions made by the Regulatory Requirements Review Committee (RRRC) concerning Anticipated Transients Without Scram (ATWS). This appeal is in response to Federal Register notice 7590-01-M which appeared in FR Vol. 44, No. 24, on February 2, 1979, concerning the RRRC meeting of January 2, 1979, reported in the meeting summary dated January 18, 1979.

The AIF Committee on Reactor Licensing and Safety urges you to reconsider the recommendations of the RRRC on ATWS in their entirety. As you know, it is our strongly held judgment, supported by our detailed review, that NUREG-0460 is seriously flawed by its lack of objectivity and its predisposition to support the earlier held conclusions of a segment of the NRC staff. The complexity of the ATWS issue tends to mask both the value and impacts of any technical resolution, and this predisposition calls into question the usefulness of NUREG-0460 as a policy-making vehicle.

The pivotal issue is the real contribution of ATWS risk to overall public risk. The presumption (or predisposition) that ATWS is indeed a significant contributor to risk biases both the methods used and the results obtained throughout the NUREG-0460 report. We believe that much of the report would have been neither relevant nor necessary if no such predisposition had been present.

Now that the NRC Staff has endorsed engineering judgment as the basis for ATWS resolution, it should be noted that the engineering judgments of thousands of industry engineers, working for the NSSS vendors, architect-engineers, utilities, the Electric Power Research Institute, and private consultants, have been applied to the ATWS issue over the past ten years. These engineers have been directly involved in the generation and detailed review of the massive documentation accumulated on ATWS, and it is from this direct involvement that their judgments have been formed. It is the consensus of these judgments, in turn, that forms the basis for the position of the AIF Committee on Reactor Licensing and Safety.

Part of the ATWS disagreement between the NRC and the nuclear industry concerns technical issues. These issues have been much debated, and we do not intend to debate them further here, but rather merely to call your attention to some of the major technical issues, as follows:

- the lack of a convincing technical demonstration that ATWS is a serious safety problem
- the lack of a technically defensible linkage between ATWS and core melt
- the excessive conservatism in the NRC Staff's consequence calculations
- the frequencies of anticipated transients of interest
- scram failure probabilities
- the failure of the NRC Staff to demonstrate that overall public risk would be reduced if their recommendations were implemented.

The remaining part of the NRC-industry disagreement concerns quasi-technical and policy matters. Most prominent in this category is the subject of value-impact assessment. It is our judgment that the value-impact assessment in NUREG-0460 is inadequate and incomplete. Further, we feel that a proper V-I assessment should be a primary tool for resolving the NRC-industry dispute. In addition. we are mystified by recent statements by the staff before the ACRS that they have "abandoned" any real attempt to generate a thorough value-impact analysis.

The NRC Staff actions in this instance contradict the NRC value-impact policy described in Chairman Hendrie's July 21, 1978 letter to the President, and ignore other Staff instructions in this regard. Value-Impact must not be ignored by NRC Management because it is an essential component of effective regulation. We have noted your personal efforts in support of implementing real value-impact assessments in the past, and we trust that your review of the ATWS issue will continue the momentum in this direction.

Standardization is another policy matter that is thwarted by the Staff's recommendations. We note that the RRRC deferred to you the decision on whether to apply different requirements to standardized plants at different stages of licensing. We urge you to maintain the integrity of the NRC standardization policy by not requiring different fixes for standardized plants.

Our recommendations concerning the resolution of the ATWS issue are contained in our January 31, 1979 presentation to the ACRS Subcommittee on ATWS. A copy of this presentation, along with our ACRS presentations of July 13, 1978 and September 8, 1978, are attached.

In summary, it is our judgment that the NRC Staff report, NUREG-0460, now endorsed in substance by the RRRC, seriously overstated the potential reduction in risk that further ATWS prevention and mitigation could provide, and seriously understates the extensive impacts that implementation of the Staff recommendations would produce on nuclear power plant designs and owner/applicant resources. Further, the Staff report does not fully consider the overall impact of recommended ATWS related design changes on plant safety. We vigorously oppose the Staff's overall approach and look to you to underscore the technical and policy inconsistencies contained in NUREG-0460.

We are prepared to discuss this matter further with you and your staff.

Very truly yours,

John E. Ward, Chairman
Committee on Reactor Licensing and Safety

JEW:skh
Enclosures

DUKE POWER COMPANY

ELECTRIC CENTER, BOX 33189, CHARLOTTE, N. C. 28242

W. C. DAIL
VICE PRESIDENT
DESIGN ENGINEERING

March 2, 1979

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Anticipated Transients Without Scram
Regulatory Requirements Review Committee Recommendations

Dear Mr. Denton:

The purpose of this letter is to bring to your attention Duke Power Company's concerns on the course of action recommended by the Regulatory Requirements Review Committee (RRRC) during its January 2, 1979 meeting on Anticipated Transients Without Scram (ATWS).

In its January 2, 1979 meeting the RRRC agreed with the use of engineering judgment as the primary basis for reaching decisions on the ATWS issue and recommended that certain general requirements for ATWS protection be established by a notice and comment rulemaking. The proposed requirements include, among other things, modifications of operating plants and design changes in plants under construction to incorporate protection and mitigation features for ATWS. Although we are encouraged by the NRC's intent to use engineering evaluations and engineering judgment to reach decisions on the ATWS issue, we are concerned that engineering evaluations and engineering judgments have not been utilized to determine whether ATWS should be regarded as a safety problem.

We believe that the shutdown systems of current designs of light water reactors are sufficiently reliable such that ATWS events are extremely unlikely and that ATWS is not a safety problem requiring consideration in the licensing process. The reasons for our conclusion that ATWS is not a safety issue include the following:

1. Engineering judgment and experience relative to the design and manufacturing, inspection, installation, operation and testing practices of the shutdown systems indicate that ATWS events are not credible.
2. Extensive operating experience demonstrates that the shutdown systems are sufficiently reliable to preclude consideration of ATWS events. There are no instances on record where a shutdown system failed when required to perform.

A-28

1028 086

3. No specific deficiencies in the shutdown systems have been identified to warrant consideration of failure to properly perform, and there are no indications that the systems cannot perform their intended function.

4. Results of detailed analyses by EPRI and reactor vendors of the reliability of the shutdown systems, transient event data, and ATWS risk indicate that the shutdown systems are highly reliable and that the risk of ATWS is acceptably small.

5. To date no meaningful value-impact assessment has been developed to demonstrate the need for ATWS modifications.

On the basis of these considerations, we believe that any requirement to implement plant modifications for ATWS protection is unnecessary and unjustified.

We recommend that if the Commission concludes that some measures be taken to enhance ATWS protection, despite the general industry conviction that ATWS is not a problem, that these measures be considered only for new plants. Currently operating plants and those under construction should be excluded from any ATWS requirements. This approach is supported by the NRC Staff conclusion that "the present likelihood of severe consequences arising from an ATWS event is acceptably small and presently there is no undue risk to the public from ATWS" (Cf: NUREG-0460, Vol. 3, pp. 42-43). We believe also that this approach would preclude unnecessary costs to the present plants, provide stability of the licensing process, and be consistent with efficient use of NRC Staff and industry resources. With regard to the measures which could be applicable to future plants, although not considered necessary as discussed above, Duke Power Company agrees with the purpose stated by the staff in NUREG-0460, Volume 3, to provide resolution of the ATWS issue in a cost effective manner.

Specifically concerning our Cherokee and Perkins Nuclear Stations (Docket Nos. 50-488 through 493) which are under design and construction and which are being licensed as duplicate plants, we consider that the most cost effective resolution for ATWS is one of prevention versus mitigation. We also have specific comments on the second paragraph of item (3a) of the minutes of the January 2, 1979 RRRC meeting. The Preliminary Design Approval for CESSAR was issued December, 1975 and the Construction Permit for Cherokee was issued December, 1977; therefore, the proposed ATWS requirement for Cherokee would be the implementation of Alternative 3. Because Perkins has not yet received a Construction Permit, due to prolonged hearing delays, Alternative 4 would appear to apply. The difference of principle (i.e., prevention versus mitigation) of resolution has a significant effect in this case. Not only is the NSSS standard design proceeding toward completion but also the Perkins design is doing so by virtue of the Cherokee design moving forward. This is the spirit of the standardization concept which we strongly support and have heard the NRC support many times. To require different modifications for the two facilities would be contrary to standardization; to require Alternative 4 for any currently licensed, standard design would not be cost effective and would be subject to future regulatory instability. We have discussed resolution of ATWS with our NSSS vendor, Combustion Engineering, and share their feeling that a system like the Supplementary Protection System as specified in Alternatives 2 and 3 of NUREG-0460, Volume 3 provides a viable, cost effective ATWS resolution for Cherokee and Perkins.

In summary, Duke Power Company continues to consider that ATWS is not a valid safety concern and that no further regulatory action is necessary. Should the Commission determine that plant modifications are required relative to the ATWS issue we believe that they should be applicable only to future plants.

Very truly yours,

L. C. Dail

L. C. Dail
Vice President
Design Engineering

RFW/jmi

cc: Dr. Max Carbon, Chairman
Advisory Committee on Reactor Safeguards
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. William Kerr, Chairman
Subcommittee on ATWS
Advisory Committee on Reactor Safeguards
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

bcc: P. M. Abraham
K. S. Canady
D. C. Holt
W. O. Parker
L. C. Dail
R. F. Wardell
W. H. Rasin
J. E. Beall
J. A. Honey (CE)
Fred Stetson (AIF)
Herb Feinroth (DOE)
Files: P81-1412.01 and A-41

POWER AUTHORITY OF THE STATE OF NEW YORK

10 COLUMBUS CIRCLE   NEW YORK, N. Y. 10019

(212) 397-6200

March 5, 1979

Mr. H. R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Mr. Denton:

In response to Federal Register Notice 7590-01-M, we offer our comments to you and the Regulatory Requirements Review Committee (RRRC). We are deeply concerned about the conclusions and recommendations concerning ATWS reached in NUREG-0460 which has been endorsed by the RRRC.

We believe the NRC Staff has so intently focused on required mitigation hardware that they have not properly responded to the fundamental question as to whether or not ATWS is a real safety concern. Their conclusions and recommendations appear to be based on the a priori assumption that ATWS is a serious safety problem. NUREG-0460 reaches conclusions which disregard actual operating data and statistical analyses supplied by the industry.

We are especially concerned by the Staff response to questions on the reliability of control rods and control rod drives. In Volume 3 of NUREG-0460, pages D-8 and D-9, the staff states "Scram systems experience statistics are insufficient to decide the question of rod and drive failure probability or vulnerability to common mode failures. The modes of failure, if they occur, are likely to be surprises." In reaching this conclusion, the Staff has disregarded the engineering judgment of the industry and statistical analyses supplied by the industry which show that the probability of the coincident failure of a sufficiently large number of control rods, such that plant shutdown is impaired, is extremely small. This is especially true in view of the fact that each control rod mechanism is periodically subjected to surveillance testing and there is virtually no likelihood that a common mode failure of the control rods would occur between the time of the last surveillance test and their use to effectuate a safe plant shutdown.

A-31

1028 [illegible]

To date, none of the control rod drive failure events has come anywhere close to constituting a scram failure. Nevertheless the NRC Staff recommends mitigation modifications to eliminate the non-existing problem. If such modifications are required, they introduce severe reliability problems and may introduce additional safety problems. We urge you to disapprove a technical decision which is based upon such conjecture.

Since the publication of Volumes 1 and 2 of NUREG-0460 there has been a large public record of the ACRS and its subcommittee meetings on ATWS. As pointed out by G.S. Lellouche of EPRI during the ACRS subcommittee meeting on January 31, 1979 the appendices on Volume 3 of NUREG-0460 purport to answer questions raised by various organizations and to provide a risk analysis. Unfortunately, many of the questions are not answered and the risk analysis is of questionable value. In accordance with the Commission practice, such conclusions should be subjected to proper peer review in order that the decision may be sound and correct. We urge that you take whatever additional time and measures may be necessary to obtain adequate peer review. We urge the Commission not to decide the ATWS issue until the ACRS has made available its conclusions and recommendations after such peer review.

We are also concerned about the rigid target dates for rule making and the schedule requirements recommended by the Staff for performing generic analysis prior to the Commission's consideration of a proposed rule. We understand that the Staff has prepared a list of questions running to some 58 pages which required approximately two months for preparation. In view of the fact that we have not seen these questions, we submit that it is impractical, if not impossible, to adequately respond to these undoubtedly complex detailed questions within the limited allotted time.

We understand that the present Staff position will either require ATWS to be included as a design basis accident or that the maximum mitigation alternative will be required for everyone unless the response to this long list of questions is timely and pursuasive. In view of the very substantial costs which would be incurred in either considering ATWS as a design basis accident or in providing for the maximum mitigation, we believe that a reasonable amount of time should be provided in preparing our response so that the best considered judgment of all parties may be included in reaching the final conclusion.

Very truly yours,

Joseph R. Schmieder

J. R. Schmieder
Chief Engineer

A-32

cc: Dr. Max Carbon, Chairman
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
1717 H Street Northwest
Washington, D.C. 20555

Dr. William Kerr
Advisory Committee on Reactor Safeguards
U.S. Nuclear Regulatory Commission
1717 H Street Northwest
Washington, D.C. 20555

Dr. Joseph Hendrie, Chairman
U.S. Nuclear Regulatory Commission
1717 H Street, Northwest
Washington, D.C. 20555

Dr. Roger Mattson, Director
Division of Systems Safety
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. Lee V. Gossick, Executive Director
Operations
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Mr. Edson G. Case, Chairman
Regulatory Requirements Review Committee
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

ILLINOIS POWER COMPANY

500 SOUTH 27TH STREET, DECATUR, ILLINOIS 62525

March 6, 1979

RECEIVED
ADVISORY COMMITTEE ON
REACTOR SAFEGUARDS U.S. NRC

MAR 8 1979

AM 7,8,9,10,11,12,1,2,3,4,5,6 PM

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

Dear Dr. Carbon:

The Atomic Industrial Forum has requested Utility Companies to communicate their views to you on the subject of ATWS and specifically to respond to questions attributed to Dr. Kerr regarding the possible use of scientific and/or probability analysis to calculate risks and to compare those calculations with an acceptable risk. Enclosed is a copy of a letter of July 7, 1978, which I wrote to Dr. Lawroski in which I described my views on the ATWS question. I did not discuss the questions raised by Dr. Kerr and will attempt to do so here.

The basic decision as to whether ATWS is an acceptable risk to society is a societal decision, but to the extent that a corporate organization must make such a decision for the organization, it is a management decision. A variety of judgments (engineering, scientific, financial) become inputs in the decision process, but they are inputs, not decisions. I am concerned by the implication that can be drawn from Dr. Kerr's question, that some scientific process can produce a decision. At best, it can provide additional inputs. At worst, it can produce confusion and lead to a poor decision. Although the Rasmussen Report is considered a scholarly and professional study (with the possible exception of the Executive Summary), I do not believe that it can or will substitute for the management (or societal) decisions as to whether nuclear power is an acceptable risk to society.

I believe that similar questions could be raised related to the acceptability of the risk of a 747 airplane crashing into the Sears Tower in Chicago. I am not aware of any scientific or probabilistic studies of this possible occurrence by the Boeing Company, the airline companies, or Sears, Roebuck & Company, even though it has a finite probability of occurrence and many people could be killed.

A-34

1028 092

The decisions to build the airplane, to fly it, and to build the Sears Tower involved the consideration of this risk (implicitly or explicitly). A scientific evaluation would not change that risk, and as a practical matter, I question whether such a study would have helped or hindered the decision process.

Dr. Kerr asked how the Utility Industry, as a body, concluded that ATWS does not pose unacceptable risks to society. I would assume that the process is similar to that used by the airlines where to my knowledge they all concluded that the risk of a 747 airplane crash into Sears Tower was not an unacceptable risk to society. It is interesting to note that Sears also concluded that this event was not unacceptable for a particular segment of society which is at greater risk; the people who work in the building and who affirm this conclusion by doing so.

I apologize for this lengthly discussion, but I believe that the growing obsession to achieve "zero risk" is destroying our National perspective. I would again urge the ACRS to examine the merits of the industry's conclusions on ATWS rather than the basis for forming these conclusions. As I tried to emphasize in my earlier letter, there are hundreds of reactors of all types in operation around the world and thousands of knowledgeable people who have judged that their operation without an ATWS requirement is an acceptable risk. Although he does not specifically address ATWS, Herbert Inhaber in his paper, "Risk of Energy Production" (AECB-119/Rev.2) concludes that nuclear power (presumably without ATWS requirements) and natural gas had the lowest risk of the eleven energy technologies considered and up to 100 times less risk than some.

In my earlier letter, I expressed the concern that ATWS requirements may not contribute to reactor safety when considered in total. However, even if they would, and thus make nuclear power even safer than natural gas (per Inhaber's evaluation) and thus make their risk more than 100 times less than other energy technologies, would this be a significant factor in determining if nuclear power (or ATWS) represents an unacceptable risk to society?

I believe that the questions I raised at the end of my earlier letter are still central to consideration of ATWS. I sincerely hope that the ACRS will address them.

Sincerely,

L. J. Koch

L. J. Koch
Vice President

Enclosure

cc: Dr. Stephen Lawroski
Dr. William Kerr

A-35

1028 093

ILLINOIS POWER COMPANY

500 SOUTH 27TH STREET, DECATUR, ILLINOIS 62525

July 7, 1978

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

Dear Dr. Lawroski:

I am writing in response to the invitation from Mr. T. G. McCreless to send comments on the report, "Anticipated Transients Without Scram for Light Water Reactors," NUREG-0460, to the Advisory Committee on Reactor Safeguards. This is a difficult letter to write because I believe the primary consideration of ATWS should be to place the subject in perspective, but because of past actions, this may be more difficult than to review the subject report.

It is my general impression that the subject of "Anticipated Transients Without Scram" was initiated as an academic treatment of a hypothetical event. There was considerable discussion at the time this concept was first introduced as to its credibility and if it merited consideration. It is my impression that a study was initiated without resolving the basic question of whether or not ATWS should be treated as a real event, because it was anticipated that the results might be "interesting." Now, some eight or nine years and thousands of pages later (a record of minutes, letters, and reports is available for purchase in 13 volumes and some 5000 pages for $1390.), it is difficult to address the basic questions: Is this a credible incident? Is this accident expected to happen? Should reactors be desfgned to accommodate this hypothetical condition? I believe that these are primary questions that must be considered in an evaluation of ATWS.

There are hundreds of reactors operating around the world; production, civilian, military, research, power, etc., which are believed to be operating safely and which have not been conditioned by an ATWS requirement. The accepted operation of these reactors represents the considered and collective judgment of the most knowledgeable people in the world. I am not aware of any significant change in position by these people, and I believe that the vast majority of them would consider the experience to date as support of their judgment.

The "cost-benefit" of ATWS should also be considered. I believe there are two major costs which must be considered:

1. The actual cost of any additional system which would be added to accommodate this hypothetical event, and

2. The implicit cost (including adverse impact on reliability and safety) of an ATWS mitigating system for which inadvertent operation would have a deleterious impact.

I am concerned about the second category because the design and operation of safety-type systems should not include a conflicting motivation to prevent inadvertent operation. For example, a poison-injection system which must respond automatically to an input signal (or signals) will be complicated in design and will be a burden in operation if there is a strong incentive to avoid "spurious" actuation. In addition, if inadvertent operation also creates secondary problems such as difficult clean-up or abnormal manipulations or operations, reliability of subsequent plant operation could be impaired. Since the probability of inadvertent operation of an ATWS mitigating system is many orders of magnitude higher than the probability of legitimate operation, this factor warrants careful consideration. (An approximation of the difference in probabilities can be derived from the thousands of total reactor years of operating experience during which an ATWS condition has not occurred as compared to the number of "spurious scrams" which have occurred.)

I believe that the ACRS has an obligation to evaluate the overall ATWS concept to determine if, in fact, reliability and safety of nuclear reactors will be enhanced by the addition of more systems and requirements. I do not believe it is enough for ACRS to review the Report, or to "validate" the estimates of probabilities and other conclusions produced by the study. I recognize that this will be a difficult question for the Committee and that you will be subjected to much pressure and abuse if you do. I believe the ACRS not only has this obligation, but is in a unique position to bring the necessary stature and statesmanship to this question. I recommend, therefore, that each member of the Committee ask himself the following questions:

1. Is this a real potential accident that I believe has a realistic probability of occurrence?

2. Do I believe that the addition of this requirement to nuclear reactors will enhance reliability and safety?

I make this recommendation with full recognition of the difficult circumstances surrounding this task, but with firm conviction that it is absolutely essential.

Sincerely,

L. J. Koch
Vice President

cc: Dr. Joseph Hendrie

A-38

STAFF ATWS CONCLUSIONS

1. ATWS IS A SAFETY ISSUE

2. BASES FOR DETERMINING REQUIRED ATWS PROTECTION

   PRIMARY BASES

      ENGINEERING ANALYSES AND JUDGMENT

   SUPPORTING BASIS

      RISK ASSESSMENT

3. VALUE-IMPACT SHOULD BE CONSIDERED

4. ATWS SHOULD BE RESOLVED BY RULE MAKING

STAFF ATWS RECOMMENDATIONS

1. FOR EARLY OPERATING PLANTS 1/

    ALTERNATIVE #2 MODIFICATIONS

    ADDITIONAL REQUIREMENTS --

        COST EFFECTIVE FEATURES BASED ON PLANT-UNIQUE ANALYSES

2. FOR OTHER OPERATING PLANTS AND PLANTS WITH CP'S ISSUED PRIOR TO JANUARY 1, 1978

    ALTERNATIVE #3 MODIFICATIONS

    TO BE CONFIRMED BY GENERIC ANALYSES FOR CLASSES OF PLANTS

3. FOR PLANTS WITH CP'S ISSUED ON OR AFTER JANUARY 1, 1978 AND NEW PLANTS

    ALTERNATIVE #4 MODIFICATIONS

1/ DRESDEN 1, YANKEE ROWE, INDIAN POINT 1, HUMBOLDT BAY, BIG ROCK POINT, CONNECTICUT YANKEE, SAN ONOFRE 1, LACROSSE, NINE MILE POINT, OYSTER CREEK.

ALTERNATIVE ATWS MODIFICATIONS

VENDOR	ALTERNATIVES 2	3	4
B&W	DIVERSITY IN ELECTRICAL PORTION OF RPS (BUSS) PLANT-UNIQUE ANALYSES	BUSS 1/ GENERIC CONFIRMATORY ANALYSES	AMSAC 2/ SAFETY VALVES GENERIC CONFIRMATORY ANALYSES
C-E	DIVERSITY IN ELECTRICAL PORTION OF RPS (SPS) 1/ PLANT-UNIQUE ANALYSES	SPS 1/ GENERIC CONFIRMATORY ANALYSES	AMSAC 2/ SAFETY VALVES GENERIC CONFIRMATORY ANALYSES
W	AMSAC 2/ PLANT-UNIQUE ANALYSES	AMSAC 2/ GENERIC CONFIRMATORY ANALYSES	AMSAC 2/ GENERIC CONFIRMATORY ANALYSES
GE	DIVERSITY IN RPS (ARI) RECIRC PUMP TRIP (RPT) REDUCE TRANSIENT FREQ. (LOGIC) IMPROVE SENSOR DIVERSITY IN SCRAM DISCH. VOLUME (SD) PLANT-UNIQUE ANALYSES	ARI RPT LOGIC SD SLCS CHANGES 3/ GENERIC CONFIRMATORY ANALYSES	RPT FAST-ACTING, HIGH CAPACITY POISON INJECTION GENERIC CONFIRMATORY ANALYSES

A-41

1028 399

ALTERNATIVE ATWS MODIFICATIONS -- FOOTNOTES

1/ WITH THIS MODIFICATION, ADDITIONAL DIVERSE MITIGATING SYSTEM ACTUATION CIRCUITRY MAY NOT BE REQUIRED.

2/ DIVERSE ATWS MITIGATING SYSTEM ACTUATION CIRCUITRY.

3/ CHANGES TO ASSURE AUTOMATIC DELIVERY OF ~86 GPM POISON.

A-42

1028 100

NEAR-TERM ATWS SCHEDULE

MARCH 1979	ACRS ATWS REPORT
MAY 1, 1979	PRELIMINARY VENDOR ANALYSES ON NEED FOR ADDITIONAL ATWS MITIGATION FEATURES, IF ANY, BEYOND THOSE PROVIDED BY ALTERNATIVE #3.
MAY 31, 1979	NRR RECOMMENDATION TO COMMISSION -- START OF RULE MAKING PROCESS.

AIF COMMITTEE ON REACTOR LICENSING AND SAFETY

ACRS PRESENTATION ON ATWS

March 8, 1979

Part I

Jerry Sorensen

Supervisor of Licensing Engineering

Washington Public Power Supply System

GOOD AFTERNOON, MY NAME IS JERRY SORENSEN. I AM SUPERVISOR OF LICENSING ENGINEERING FOR THE WASHINGTON PUBLIC POWER SUPPLY SYSTEM (WPPSS). I HAVE RECENTLY BEEN ASKED TO CHAIR THE AIF SUBCOMMITTEE ON ATWS. IT IS IN THAT CAPACITY THAT I AM HERE TODAY.

THE AIF PRESENTATION TODAY WILL BE HANDLED IN TWO PARTS. FIRST, I WILL BRIEFLY SUMMARIZE SOME OF THE CONCERNS THAT HAVE BEEN EXPRESSED BY THE AIF IN THE RECENT PAST. FOLLOWING MY PRESENTATION, MR. WARREN OWEN OF DUKE POWER COMPANY WILL PROVIDE THE UTILITY MANAGEMENT PERSPECTIVE ON ATWS.

AS YOU GENTLEMEN ARE WELL AWARE, THE INDUSTRY INVOLVEMENT WITH THE ATWS ISSUE HAS BEEN GOING ON FOR A CONSIDERABLE PERIOD OF TIME. OUR TECHNICAL POSITION IS WELL KNOWN AND HAS BEEN PRESENTED IN SOME DETAIL AT PREVIOUS MEETINGS. IT IS NOT MY PURPOSE TO REPEAT THAT TECHNICAL MATERIAL, BUT MERELY TO SUMMARIZE SOME OF OUR MAJOR AREAS OF CONCERN. WE WOULD LIKE TO THINK THAT THE ISSUE IS ABOUT TO BE CLOSED SUCH THAT ALL OF US CAN DIRECT OUR EFFORTS TO OTHER ACTIVITIES. UNFORTUNATELY, WHILE WE MAY BE APPROACHING THE END OF A PHASE, WE DO NOT SEE THIS AS THE END OF THE ISSUE.

YOUR CONTINUING DELIBERATIONS ON THIS ISSUE HAVE BEEN MOST HELPFUL IN BRINGING TO LIGHT ITS VARIOUS ASPECTS AND, IN PARTICULAR, IN CAUSING WIDESPREAD EVALUATIONS BY ALL SEGMENTS OF THE INDUSTRY OF THE FUNDAMENTAL QUESTION: DOES ATWS PRESENT A SIGNIFICANT THREAT TO THE HEALTH AND SAFETY OF THE PUBLIC? THE ANSWER TO THIS QUESTION IS,

OF COURSE, THE CORNERSTONE OF YOUR CONCLUSIONS AND ON IT WILL BE BUILT YOUR RECOMMENDATIONS TO THE COMMISSIONERS REGARDING REGULATIONS PROPOSED BY THE NRC STAFF.

WE, AS AN INDUSTRY, HAVE RE-EXAMINED THIS QUESTION IN DEPTH SINCE THE PUBLICATION OF NUREG-0460 ALMOST A YEAR AGO. WE HAVE STATED TO YOU OUR CONCLUSIONS AND HAVE RECOMMENDED SPECIFIC CHANGES TO THE DESIGN AND OPERATION OF CERTAIN PLANTS NOW OPERATING AND UNDER CONSTRUCTION WE BELIEVE THAT THOSE CHANGES ARE SUFFICIENT TO ACCOMMODATE THE DIFFERENCE IN OPINION BETWEEN THE STAFF AND THE INDUSTRY OR THE ANSWER TO THE FUNDAMENTAL QUESTION. WE ARE CONCERNED THAT CHANGES THAT GO BEYOND THOSE WHICH WE HAVE RECOMMENDED ARE UNFOUNDED AND IN FACT WILL PREVENT US FROM GETTING THE MAXIMUM SAFETY FROM AVAILABLE INDUSTRY AND STAFF RESOURCES. IT IS IN THAT CONTEXT THAT WE SUMMARIZE FOR YOU TODAY OUR CRITICISMS OF THE REGULATIONS WHICH HAVE BEEN PROPOSED BY THE NRC STAFF.

SINCE THE ISSUANCE OF NUREG-0460, VOLUMES 1 AND 2 IN APRIL OF 1978, THE INDUSTRY HAS VOICED A NUMBER OF CONERNS WITH THE APPROACH BEING TAKEN BY THE STAFF TO ACHIEVE ATWS RESOLUTION. SOME OF THESE CONCERNS, WHICH I WILL ADDRESS TODAY ARE AS FOLLOWS:

1) INDUSTRY INPUT (EPRI REPORTS AND OTHERS) HAS BEEN DISCOUNTED BY STAFF
2) ATWS CONSEQUENCES OVERSTATED BY STAFF
3) INADEQUATE VALUE/IMPACT ASSESSMENT BY STAFF
4) NRC PROCEDURES FOR ORDERLY REGULATION HAVE NOT BEEN FOLLOWED

1) INDUSTRY INPUT (EPRI AND OTHER REPORTS) HAS BEEN DISCOUNTED BY STAFF

IN ORDER TO DEMONSTRATE TO THE NRC THAT THE OPERATING EXPERIENCE AT NUCLEAR POWER PLANTS SUPPORTED THE INDUSTRY POSITION THAT ATWS DOES NOT REPRESENT A MAJOR SAFETY CONCERN, REPORTS DEVELOPED BY THE ELECTRIC POWER RESEARCH INSTITUTE WERE SUBMITTED FOR STAFF REVIEW ALTHOUGH THESE REPORTS WERE AVAILABLE TO THE STAFF DURING THE TIME THAT NUREG-0460 WAS BEING PREPARED THERE APPEARS TO HAVE BEEN NO SERIOUS CONSIDERATION OF THE DATA. REPORTS AND EVALUATIONS PREPARED INDEPENDENTLY BY THE NSSS VENDORS APPEAR TO HAVE BEEN DISMISSED IN A SIMILAR MANNER.

2) ATWS CONSEQUENCES OVERSTATED BY STAFF

THE STAFF HAS STATED (NUREG-0460, VOL 3, PG 21) THAT "FOR PLANTS SUPPLIED BY GE, THE MOST LIKELY RESULT OF AN ATWS EVENT WOULD BE A CORE MELT DOWN. FOR SOME WESTINGHOUSE PLANTS AND ALL B & W AND CE PLANTS, THE RESULT WOULD MOST LIKELY BE EITHER EXCESSIVE PRIMARY SYSTEM PRESSURE OR CORE MELT." EVALUATIONS OF ATWS EVENTS BY THE VENDORS DO NOT SUPPORT THESE CONCLUSIONS. THESE EVALUATIONS OF ATWS CONSEQUENCE ARE NOT TAKEN LIGHTLY BY THE UTILITIES WHO MUST ULTIMATELY OPERATE THE PLANTS. OBVIOUSLY, IF AN ATWS SHOULD OCCUR AND THE CONSEQUENCES WERE AS STATED BY THE STAFF, IT IS THE UTILITY-NOT THE VENDOR OR REGULATOR-WHO WOULD SUFFER THE IMMEDIATE ECONOMIC LOSS BUT IT IS THE ENTIRE INDUSTRY THAT WOULD SUFFER THE RESULTANT PUBLIC WRATH.

3) <u>INADEQUATE VALUE/IMPACT ASSESSMENT BY STAFF</u>

THE NRC HAS STATED ITS INTENT TO SUPPORT NEW REGULATIONS WITH VALUE/IMPACT STATEMENTS. IN DEVELOPING THE VALUE/IMPACT STATEMENT FOR ATWS, THE STAFF DID MEET WITH INDUSTRY REPRESENTATIVES TO GET INDUSTRY FIGURES ON THE COST FOR VARIOUS PROPOSED ATWS FIXES. THE IMPACTS NOTED IN NUREG-0460 (VOL 3) REFLECT DIRECT COSTS WITH ONLY PASSING MENTION OF INDIRECT COSTS - WHICH GENERALLY FAR OUTWEIGH THE DIRECT COSTS. THE COSTS TO IMPLEMENT THESE ATWS FIXES MUST ULTIMATELY BE BORNE BY THE RATEPAYERS, WHO ALREADY CONSIDER RATES TO BE UNREASONABLE.

4) <u>NRC PROCEDURES FOR ORDERLY REGULATION HAVE NOT BEEN FOLLOWED</u>

THE NRC STAFF HAS PREPARED AND PUBLISHED PROCEDURES WHICH PROVIDE FOR THE PUBLIC (INCLUDING THE INDUSTRY) TO REVIEW AND COMMENT ON ACTIONS PROPOSED TO BE TAKEN BY THE REGULATORY REQUIREMENTS REVIEW COMMITTEE (RRRC). ON FEBRUARY 2, 1979, A FEDERAL REGISTER NOTICE WAS ISSUED, REGARDING AN OPPORTUNITY TO APPEAL RRRC ACTIONS ON ATWS. THE COMMENT PERIOD FOR THIS ITEM WOULD EXPIRE ON MARCH 2, 1979. THUS, WE WERE DISMAYED THAT THE OPPORTUNITY TO PRESENT AN APPEAL REGARDING THE PROPOSED ATWS "SOLUTIONS" SEEM TO HAVE BEEN PROVIDED ONLY AS A PROCEDURAL FORMALITY ON THE ROAD TO A RULE-MAKING HEARING. A LETTER FROM DR. MATTSON, DIRECTOR OF DSS, HAS BEEN ISSUED TO THE FOUR REACTOR VENDORS ACCOMPANIED WITH SOME FIFTY-EIGHT PAGES OF REQUESTS FOR ATWS ANALYSES. THE LETTER, DATED FEBRUARY 15, 1979,

INDICATES THAT THIS NRC REQUEST IS AUTHORIZED BY THE DIRECTOR, NRR. WE DO NOT UNDERSTAND HOW ON THE ONE HAND, THE STAFF CAN REQUEST APPEALS TO BE FILED BEFORE PROCEEDING WITH IMPLEMENTATION OF THE RRRC RECOMMENDATIONS, YET ON THE OTHER HAND THE STAFF APPEARS TO BE PROCEEDING POSTHASTE TO IMPLEMENT THOSE RECOMMENDATIONS. THE FEBRUARY 15 LETTER REQUESTS THAT THE NSSS VENDORS REDO OR EXPAND THEIR ATWS ANALYSES, REJUSTIFY THE ASSUMPTIONS, PERFORM NEW ANALYSES ON RCPB VALVES, INCLUDE NEW DOSE AND CONTAINMENT CALCULATIONS, BECAUSE OF REVISED ASSUMPTIONS, AND SUBMIT RESULTS WITHIN AN UNREALISTIC TIME FRAME. HOWEVER, IT IS OUR UNDERSTANDING THAT THERE MAY BE ADDITIONAL TIME ALLOWED TO RESPOND TO THIS REQUEST. MOREOVER, A CAREFUL READING OF THAT LETTER INDICATES THAT THE REQUESTED ANALYSES INHERENTLY CONTAIN THE POTENTIAL FOR FUTURE ATWS RATCHETS, WITH NO END IN SIGHT.

THE ABOVE REPRESENT SOME OF OUR CONCERNS THAT HAVE BEEN EXPRESSED OVER THE PAST YEAR WITH REGARD TO ATWS.

THE NRC STAFF HAS LISTED THE ATWS ISSUE AS IT'S NUMBER 1 GENERIC ITEM AND HAS CLEARLY STATED THAT "ATWS WILL BE RESOLVED." WE TOTALLY CONCUR THAT ATWS MUST BE RESOLVED, OUR DISAGREEMENT IS WITH THE STAFF'S POSITION TOWARD RESOLUTION!

WE HAVE BROUGHT OUR DEBATE OF THE ATWS ISSUE BEFORE THE ACRS BECAUSE THIS IS A PUBLIC FORUM AVAILABLE TO DEVELOP A RECORD OF OUR CONCERN. WE ALSO RECOGNIZE THE ACRS AS AN INDEPENDENT TECHNICAL REVIEW GROUP. THE NRC STAFF HAS

CHARTED A COURSE WHICH THEY APPEAR DEDICATED TO FOLLOW.

IN REACHING YOUR CONCLUSIONS ON ATWS, WE REQUEST THAT THE ACRS CONSIDER THE FOLLOWING:

- THAT AN ATWS EVENT WITH SIGNIFICANT CONSEQUENCES IS NOT A CREDIBLE EVENT, NOR DESERVING OF ANY CONSIDERATION AS A DBA, BASED ON ACCEPTABLY LOW ATWS PROBABILITY AND CURRENT PLANT CAPABILITY (WITH BWR RPT) TO MITIGATE ATWS CONSEQUENCES.
- THAT THE ACRS ENDORSE THE REGULATORY STAFF CONCLUSION THAT ATWS PROBABILITIES ARE ACCEPTABLY LOW FOR THE CURRENT POPULATION OF NUCLEAR PLANTS, FOR THEIR REMAINING LIFE-TIME.
- THAT THE PRESENTATIONS GIVEN BY THE NSSS VENDORS TO THE ACRS HAVE SHOWN CURRENT PLANT CAPABILITY (WITH BWR RPT) TO ACCEPTABLY MITIGATE ATWS EVENTS, SHOULD AN ATWS EVER OCCUR.
- CONCUR THAT NUREG-0460 IS PREDISPOSED TO A SUPPOSITION OF ATWS SIGNIFICANCE.
- THAT PROPER AND THOROUGH VALUE/IMPACT ASSESSMENTS OF THE BACKFITS RECOMMENDED IN NUREG-0460 VOLUME 3 HAVE NOT BEEN PERFORMED.
- THAT PROPER AND THOROUGH VALUE/IMPACT ASSESSMENTS SHOULD BE PERFORMED AND REVIEWED AS PART OF THE BASES FOR DECISIONS ON ATWS BACKFITS.
- THAT PLANT STANDARDIZATION EFFORTS SHOULD NOT BE COMPROMISED AS A RESULT OF SUCCESSIVE ATWS BACKFIT REQUIREMENTS.
- THAT CREDIBLE OPERATOR ACTION CAN BE RECOGNIZED AS PROVIDING A PROBABLE SIGNIFICANT RESPONSE WITHIN SEVERAL MINUTES, GIVEN THE PROPER ATWS TRAINING AND PROCEDURES,

IN VIEW OF THE BACKFIT ALTERNATIVES

- THAT THE STAFF SHOULD NOT REQUIRE DBA TYPE OF ATWS ANALYSIS FROM THE NSSS VENDORS OR UTILITIES; THAT "BEST-ESTIMATE", PRODUCT LINE GENERIC ANALYSIS SHOULD BE SUFFICIENT AS A BASIS FOR CONCLUDING CURRENT PLANT ACCEPTABILITY (WITH BWR RPT), IN VIEW OF ALREADY ACCEPTABLY LOW ATWS PROBABILITY.

WE ARE REALISTIC ENOUGH NOT TO EXPECT THE ACRS TO DECLARE THE STAFF POSITION TO BE WRONG - HOWEVER, WE DO HOPE THAT YOU WOULD POINT OUT THE EXCESSES IN THEIR POSITION. AS POINTED OUT IN OUR LAST PRESENTATION TO THE ATWS SUBCOMMITTEE, WE RECOMMEND THAT EXISTING PLANTS AND PLANTS UNDER CONSTRUCTION ARE SAFE WITH THE CURRENT DESIGN; THUS ALTERNATIVE I WOULD APPLY. IF THERE IS A FEELING THAT IMPROVEMENTS ARE REQUIRED FOR FUTURE PLANTS, THEN ALTERNATIVE NUMBER 2 WOULD ADEQUATELY MEET THOSE CONCERNS.

THIS CONCLUDES MY COMMENTS. I WILL BE HAPPY TO ENTERTAIN ANY QUESTIONS OR COMMENTS. IF THERE ARE NONE, I WILL RELINQUISH MY POSITION TO MR OWEN.

AIF COMMITTEE ON REACTOR LICENSING AND SAFETY

ACRS PRESENTATION ON ATWS

March 8, 1979

Part II

Warren Owen

Senior Vice President

Duke Power Company

GOOD AFTERNOON. MY NAME IS WARREN OWEN AND I AM SENIOR VICE PRESIDENT, ENGINEERING AND CONSTRUCTION FOR DUKE POWER COMPANY. I HAVE APPEARED BEFORE THIS COMMITTEE NUMEROUS TIMES IN THE PAST - ALWAYS IN CONNECTION WITH ONE OF DUKE'S NUCLEAR PROJECTS. I AM STILL REPRESENTING DUKE POWER COMPANY HERE TODAY, BUT MY OBSERVATIONS ON THE ATWS ISSUE ARE REPRESENTATIVE OF THE OPINIONS OF MY SENIOR MANAGEMENT COLLEAGUES IN THE INDUSTRY.

I'M NOT GOING TO TELL YOU ATWS IS <u>NOT</u> A PROBLEM--IT IS, BUT IT'S OBVIOUS FROM JERRY'S PRESENTATION THAT WE BELIEVE CURRENT PLANTS DO NOT NEED TO BE MODIFIED BEYOND WHAT WE'VE ALREADY PRESENTED. I VIEW THE ISSUE AS A SIGNIFICANT PROBLEM TO UTILITY EXECUTIVES WHOSE COMPANIES ARE OPERATING, CONSTRUCTING OR CONTEMPLATING NUCLEAR FACILITIES. UNCERTAINTY IN COSTS, SCHEDULES AND REGULATORY REQUIREMENTS ARE THE MAJOR OBSTACLES FACING THOSE UTILITIES TRYING TO UTILIZE THE NUCLEAR OPTION FOR THE GENERATION OF ELECTRICITY. ATWS IS A GOOD EXAMPLE OF THE REGULATORY UNCERTAINTIES THAT HAVE PLAGUED THE INDUSTRY FOR YEARS. I BELIEVE THE ACRS HAS THE OPPORTUNITY TO HELP REMOVE THIS ONE UNCERTAINTY.

THE UTILITY BUSINESS HAS MUCH AT STAKE IN SEARCH FOR A REASONABLE SOLUTION TO THE ATWS ISSUE. WE MUST HAVE SAFE AND RELIABLE PLANTS IN ORDER TO PERFORM OUR FUNCTION OF PROVIDING ADEQUATE ELECTRICAL ENERGY TO OUR CUSTOMERS. NOW AND IN THE FUTURE. WE ARE ALWAYS WILLING TO CONSIDER ANY REASONABLE ARGUMENTS IN SUPPORT OF INCREMENTAL IMPROVEMENTS IN SAFETY. OUR GOAL IS TO HAVE THESE NUCLEAR PLANTS CONTINUE THEIR EXCELLENT SAFETY RECORD, -- FOR ECONOMIC

AS WELL AS PUBLIC ACCEPTANCE REASONS. THE CURRENTLY OPERATING PLANTS ARE MAKING A SUBSTANTIAL CONTRIBUTION TO OUR ELECTRIC SYSTEMS AND I WOULD HATE TO SEE SOMETHING OF MARGINAL VALUE CAUSE CHANGES WHICH WOULD NEGATIVELY IMPACT THIS CONTRIBUTION.

I BELIEVE THERE OUGHT TO BE OVER-RIDING REASONS TO SUPPORT ANY DESIGN OR HARDWARE CHANGES ON CURRENTLY OPERATING REACTORS. WE ARE ALSO CONCERNED ABOUT THE IMPOSITION OF A WHOLE NEW SPECTRUM OF DESIGN REQUIREMENTS IN MID-STREAM FOR THOSE REACTORS UNDER CONSTRUCTION. FOR REACTORS NOT YET LICENSED, REQUIREMENTS WHICH RESULT IN INCREMENTAL IMPROVEMENTS IN SAFETY CERTAINLY DESERVE SERIOUS CONSIDERATION, BUT SHOULD ONLY BECOME REGULATION IS F THEY CAN BE FULLY COST JUSTIFIED.

WE HAVE WATCHED THE EVOLUTION OF THIS ISSUE OVER THE PAST TEN YEARS AND ARE CONCERNED THAT <u>ANY</u> IMPOSED SOLUTIONS BE APPLIED CONSISTENTLY. INDEED, NRC MUST BE PARTICULARILY SENSITIVE TO STANDARDIZED PLANTS TO AVOID LOSING THE ECONOMIC AND SAFETY BENEFITS WE ALL EXPECT TO ACHIEVE BY THAT CONCEPT. AN EXAMPLE OF THIS POTENTIAL THREAT TO STANDARDIZATION IS THE CASE OF MY OWN COMPANY'S "SIX PACK"-CALLED THE CHEROKEE AND PERKINS PLANTS. THESE PLANTS WERE LICENSED AS DUPLICATES AND ALSO UTILIZED A STANDARD NSSS; HOWEVER, BECAUSE OF DIFFERENCES IN CONSTRUCTION PERMIT DATES STEMMING FROM NRC REOPENED ENVIRONMENTAL HEARINGS ON ONE PLANT, DUPLICATION AND STANDARDIZATION WILL BE LOST WITH THE PROPOSED NRC ATWS RESOLUTION. NOT ONLY ARE THE SPECIFIC FIXES DIFFERENT BUT THE PRINCIPALS OF RESOLUTION AS WELL. THIS CASE IS NOT UNLIKE OTHERS IN VIOLATION OF THE BASIC TENET OF STANDARDIZATION A SITUATION WHICH CAN BE REPAIRED BY USING THE LEAD UNIT AS

THE LICENSING BASE.

IN VOLUME 3 of NUREG-0460 THE REGULATORY STAFF SUGGESTS THAT ENGINEERING JUDGMENT, RATHER THAN NUMERICAL GOALS, BE THE BASIS FOR DETERMINING AN ATWS RESOLUTION. I BELIEVE THIS TO BE REASONABLE, BUT I SUGGEST TO YOU THAT A VALUABLE SOURCE OF ENGINEERING JUDGMENT RESIDES WITH OUR INDUSTRY AND I ASK YOU TO GIVE THAT JUDGMENT FULL CONSIDERATION IN YOUR DELIBERATIONS.

RESOLUTION OF THE ATWS ISSUE SEEMS TO HAVE STRAYED FROM THE PRUDENT USE OF ENGINEERING JUDGMENT. I SENSE A DESIRE BY SOME TO USE AN NRC-IMPOSED "SOLUTION" REGARDLESS OF THE NUCLEAR INDUSTRY'S OBJECTIONS AND LOGIC. MY LAWYERS WOULD WANT ME TO CALL THIS "ARBITRARY AND CAPRICIOUS." STAFF EFFORTS SINCE PUBLICATION OF NUREG-0460 IN APRIL 1978, AND THE LATEST VERSION IN DECEMBER 1978, SEEM TO HAVE BEEN AIMED AT FORCING A RESOLUTION AT A PRECIPITOUS PACE. WE COMMEND THE ACRS, AND ITS WORKING GROUP ON ATWS CHAIRED BY PROFESSOR KERR, FOR ALLOWING AN ORDERLY PRESENTATION OF THE ISSUES WITH A CHANCE FOR FULL INDUSTRY INPUT. HAD THE NRC SUCCEEDED IN THEIR ORIGINAL SCHEDULE ANNOUNCED IN APRIL 1978, WE MIGHT WELL FIND OURSELVES TODAY INVOLVED IN A RULE-MAKING HEARING CONCERNING ATWS AS A DBA, FROM WHICH POSITION THE NRC PROPERLY RETREATED ONLY LAST DECEMBER.

IT IS IMPORTANT THAT NEW REGULATORY REQUIREMENTS BE BASED ON NEED, AND IF NEEDED, BE LOGICALLY DEVELOPED. IN MY OPINION, THIS HAS NOT BEEN DONE WITH THE PROPOSED ATWS FIX. I SEE NO REASON WHY RISK ACCEPTABLE TODAY MUST BE SIGNIFICANTLY LOWER IN FUTURE YEARS WITHOUT REGARD FOR COSTS IMPOSED ON

OUR CUSTOMERS. PUBLIC HEALTH AND SAFETY OBJECTIVES MUST FULLY CONSIDER ECONOMIC IMPLICATIONS ON THAT SAME PUBLIC, IF THE PUBLIC IS TO BE WELL SERVED. TODAY, INFLATION IS THIS COUNTRY'S WORST ENEMY AND, IN MY VIEW, INFLATION IS FUELED BY OVER-REGULATION.

AS A UTILITY EXECUTIVE, THE DEVELOPMENT OF DESIGN REQUIREMENTS, WITH OBVIOUS EXPENSE BUT WITHOUT A BENEFIT TO THE PUBLIC OUTWEIGHING THAT EXPENSE, LEAVES ME WITH DOUBTS AND QUESTIONS: HOW CAN I JUSTIFY ATWS FIXES TO MY STOCKHOLDERS, CUSTOMERS AND UTILITY COMMISSIONS IF I CAN'T JUSTIFY THEM IN MY OWN MIND? WHAT WILL THE TOTAL, EVENTUAL COST OF A "FIXED" PLANT BE? HOW OFTEN WILL WE BE REQUIRED TO BACKFIT HARDWARE WITHOUT A REAL VALUE/IMPACT ASSESSMENT BY THE NRC? IT IS OUR JUDGMENT THAT THE VALUE-IMPACT ASSESSMENT IN NUREG-0460 IS INADEQUATE AND INCOMPLETE. FURTHER, I FEEL THAT A PROPER COST BENEFIT ASSESSMENT SHOULD BE A PRIMARY TOOL FOR RESOLVING THE NRC-INDUSTRY DISPUTE. IN ADDITION, WE ARE MYSTIFIED BY RECENT STATEMENTS BY THE STAFF BEFORE THE ACRS THAT THEY HAVE "ABANDONED" ANY REAL ATTEMPT TO GENERATE A THOROUGH VALUE-IMPACT ANALYSIS. VALUE-IMPACT MUST NOT BE IGNORED BY NRC MANAGEMENT BECAUSE IT IS AN ESSENTIAL COMPONENT OF EFFECTIVE REGULATION.

THE ATWS ISSUE HAS THE POTENTIAL TO DE-STABILIZE THE LICENSING PROCESS. RECENTLY, WITH MY STAFF AND JERRY'S COMMITTEE, I HAVE REVIEWED THE EVOLUTION OF ATWS OVER THE PAST FEW YEARS. THE ATWS ISSUE HAS FLUCTUATED OVER THE

YEARS FROM A GENERIC CONSIDERATION RESPONDED TO VIA NSSS VENDORS ANALYSES OF RELIABILITY; TO A WASH-1270 DEMAND FOR SCRAM SYSTEM IMPROVEMENTS AND/OR MITIGATION CAPABILITY; TO THE 1975 STATUS REPORTS REQUIRING ATWS ANALYSES DONE ONLY PER NRC DICTATES ON PARAMETERS AND MODELS; TO A NUREG-0460 (VOLUMES 1-2) DESIGN BASIS ACCIDENT SCENARIO REQUIRING MITIGATION ONLY; TO TODAY'S VOLUME 3 NUREG-0460 REQUIREMENTS FOR MITIGATION PLUS PREVENTION, OR MITIGATION ONLY. FURTHERMORE, OUR UNDERLYING CONCERN IS THAT THERE IS NO "FINAL" RESOLUTION OF ATWS EVIDENT IN THE NRC STAFF DOCUMENTS. BASED ON THE ENTIRE HISTORY OF AEC/NRC LICENSING BEHAVIOR. I AM GRAVELY CONCERNED THAT THE NRC'S PROPOSED FIXES TODAY ARE JUST THE BEGINNING OF FUTURE COSTLY RE-REVIEWS, RE-ANALYSES AND RETROFITS.

THE ATWS ISSUE WILL BE AN INDICATOR TO THE INDUSTRY OF OUR REGULATORY FUTURE, AND AS SUCH, ITS RESOLUTION IS MORE IMPORTANT THAN JUST THE IMMEDIATE FINANCIAL BURDEN UPON UTILITY RATEPAYERS. IF THE REGULATORY STAFF'S PROPOSED RESOLUTION OF ATWS IS SUPPORTED BY THE ACRS AND THE COMMISSION, THE FUTURE OF NUCLEAR DEVELOPMENT IS FURTHER ENDANGERED. WHAT WE NEED IS A SIGNAL TO THE UTILITY INDUSTRY; A SIGNAL TO EXPECT REASONABLE REGULATION THAT CAREFULLY CONSIDERS PUBLIC HEALTH AND SAFETY, INCLUDING A RIGOROUS VALUE-IMPACT EVALUATION FOR EACH NEW REQUIREMENT, AND A WELL-SUPPORTED FINDING THAT EACH REQUIREMENT WILL PROVIDE SUBSTANTIAL AND NECESSARY ADDITIONAL PROTECTION.

CONTINUED GROWTH OF THE NUCLEAR INDUSTRY IS VITAL TO MEETING FUTURE DEMANDS FOR ELECTRICITY AND I BELIEVE THIS GROWTH CAN BE ACCOMPLISHED SAFELY--BUT I'M CONVINCED THAT

THE IMPOSITION OF EXPENSIVE NEW REGULATORY REQUIREMENTS WITHOUT CLEAR JUSTIFICATION WILL ENDANGER ONE OF THE FEW OPTIONS REMAINING FOR THE UTILITY INDUSTRY.

GENERAL ELECTRIC COMPANY

PRESENTATION ON ATWS TO THE ACRS

MARCH 8, 1979

- INTRODUCTION
- BWR SCRAM SYSTEM RELIABILITY
- INHERENT CAPABILITY OF BWR
- ASSESSMENT OF NUREG 0460, VOL 3 REQUIREMENTS
- SUMMARY

TOTAL ATWS COSTS

COSTS IN MILLIONS OF DOLLARS (1)

		PLANT STATUS	
ALTERNATE	VENDOR	OPERATING	NEW
#2	B&W, CE	1.4-2.1	.9-1.8
	WESTINGHOUSE	0-2.1 (4)	0-1.8
	GE	2.0-3.3	2.0-3.3
#3	B&W, CE	1.5-2.4	1.1-2.0
	WESTINGHOUSE	0-2.4	0-2.0
	GE (2)	3.8-5.8	3.8-5.8
#4	B&W, CE	N/A	3.0-3.8
	WESTINGHOUSE	N/A	0-2.0
	GE (3)	N/A	9.5-14.0

(1) TOTAL COSTS ARE APPROXIMATELY 2.5 TIMES DIRECT COSTS FROM VOL. III NUREG 0460.

(2) ADDITIONAL DOWNTIME OF UP TO 25 DAYS FOR REPIPING OPERATING PLANTS $\$7 \times 10^6$ - $\$25 \times 10^6$ AND POTENTIAL DOWNTIME FROM SPURIOUS ACTUATION NOT INCLUDED.

(3) POTENTIAL DOWNTIME FROM SPURIOUS ACTUATION NOT INCLUDED.

(4) FOR PLANTS WITH RPT INSTALLED ALREADY — COST $< \$1 \times 10^6$

1028 113

OVERVIEW

THE BWR COST PENALTY FOR ATWS MODIFICATIONS IS NOT CONSISTENT WITH:

- o THE MORE RELIABLE BWR SCRAM SYSTEM
- o THE OVERALL LOW SUSCEPTABILITY OF THE BWR TO EVENTS WHICH COULD LEAD TO CORE UNCOVERY.

OVERALL BWR CAPABILITY IS IGNORED IN THE ATWS MITIGATION APPROACH

BWR SCRAM SYSTEM RELIABILITY

A-62

CONCLUSIONS OF RELIABILITY STUDY

- o CURRENT SCRAM SYSTEM (WITHOUT ARI) HIGHLY RELIABLE (< 10^{-6}/DEMAND UNRELIABILITY)
- o MECHANICAL PORTION MOST RELIABLE (<< 10^{-7}/DEMAND UNRELIABILITY)
- o MANY WAYS TO SCRAM RODS
- o FAILURE OF SOME RODS TO INSERT IS ONLY OF MINOR CONSEQUENCE
- o ADDITION OF ALTERNATE ROD INSERTION (ARI) IMPROVES OVERALL RELIABILITY BY APPROXIMATELY TWO ORDERS OF MAGNITUDE
- o TOTAL SCRAM FAILURE IS INCREDIBLE

$\boxed{\text{ATWS IS NOT A SAFETY CONCERN}}$

INHERENT CAPABILITY OF BWR

A-64

BWR ATWS MITIGATION CAPABILITY

- BWR MITIGATION CAPABILITY DEMONSTRATED BY RESPONSE TO

  TURBINE TRIP WITH BYPASS (TTw/BP)

  MAIN STEAMLINE ISOLATION VALVE CLOSURE (MSIV)

- RECIRCULATION PUMP TRIP ACTUATED BY

  HIGH VESSEL PRESSURE

  LOW REACTOR WATER LEVEL

- REACTOR PRESSURE MAINTAINED BELOW SERVICE LEVEL C LIMITS

- FUEL TEMPERATURE MAINTAINED BELOW ~1400°F (LESS THAN 20% OF FUEL EXPERIENCES BOILING TRANSITION)

- SUPPRESSION POOL OR CONDENSER PERFORMS AS HEAT SINK

- SUPPRESSION POOL HEATING NOT EXPECTED TO AFFECT CORE COOLING CAPABILITY

DOES NOT LEAD TO CORE UNCOVERY MUCH LESS CORE MELT

1028

BWR ATWS MITIGATION CAPABILITY

TURBINE TRIP WITH BYPASS SCENARIO:

- RPT PREVENTS OVERPRESSURE AND LEADS TO SIGNIFICANT POWER REDUCTION
- FEEDWATER RUNBACK
- STANDBY LIQUID CONTROL INITIATED BY OPERATOR AT 2 MINUTES
- RHR INITIATED AT ~ 10 MINUTES
- REPLENISH CONDENSATE STORAGE TANK AT ~ 1 HOUR
- CONDENSER IS PRIMARY HEAT SINK AND WATER SOURCE

TURBINE TRIP CONSEQUENCES:

- CORE COOLING MAINTAINED AT ALL TIMES
- SUPRRESSION POOL TEMPERATURE PEAKS AT <200°F AFTER 20 MIN
- REACTOR SHUTDOWN AT 1 HR
- NO CORE UNCOVERY

BWR ATWS MITIGATION CAPABILITY

MSIV SCENARIO:

- RPT PREVENTS OVERPRESSURE AND LEADS TO SIGNIFICANT POWER REDUCTION
- HPCI/S INITIATED
- STANDBY LIQUID CONTROL INITIATED BY OPERATOR AT 2 MINUTES
- RE-ESTABLISH BYPASS TO CONDENSER AT ~ 10 MINUTES
- RHR INITIATED AT ~ 10 MINUTES
- REPLENISH CONDENSATE STORAGE TANK AT ~ 1 HOUR

MSIV CONSEQUENCES:

- CORE COOLING MAINTAINED AT ALL TIMES
- SUPPRESSION POOL TEMPERATURE PEAKS AT ~190°F AFTER 10 MINUTES
- REACTOR SHUTDOWN AT 1 HR
- NO CORE UNCOVERY

ASSESSMENT OF NUREG 0460, VOL 3 REQUIREMENTS

A-68

NUREG 0460 ALTERNATE #2 REQUIREMENTS

I. RECIRC PUMP TRIP

II. ALTERNATE ROD INSERTION

III. LOGIC CHANGES TO REDUCE NUMBER OF MSIV ISOLATIONS

IV. INCREASE DIVERSITY OF SCRAM DISCHARGE INSTRUMENTATION

BWR RESPONSE WITH ARI

- ARI RESULTS IN "15 SECOND" ROD INSERTION DELAY

- SOME TRANSITION BOILING INITIALLY, NO FUEL FAILURES

- INSIGNIFICANT POOL TEMPERATURE INCREASE RELATIVE TO NORMAL SCRAM (<10°F)

- ADDITION OF ARI IMPROVES ROD INSERTION RELIABILITY BY TWO ORDERS OF MAGNITUDE

CONSEQUENCES TOTALLY ACCEPTABLE

NUREG 0460 REQUIREMENTS

ALTERNATE #3

SAME AS ALTERNATE #2 PLUS TIMED "TWO PUMP" BORON MITIGATION SYSTEM

ALTERNATE #4

ELIMINATES ALTERNATE ROD INSERTION IN FAVOR OF HIGH CAPACITY BORON MITIGATION SYSTEM

MITIGATION WITH ALTERNATE #3

- REACTOR PRESSURE MAINTAINED BELOW SERVICE LEVEL C LIMITS
- FUEL TEMPERATURE MAINTAINED BELOW ~ 1400°F (LESS THAN 20% OF FUEL EXPERIENCES BOILING TRANSITION, HOWEVER, NO FUEL FAILURE)
- SUPPRESSION POOL PERFORMS AS HEAT SINK
- POOL TEMPERATURE CONSIDERATIONS FOR WORST CASE SCRAM FAILURE

CONTAINMENT TYPE	BULK POOL TEMPERATURE*
MK I	~ 200°F
MK II	~ 180°F
MK III	~ 165°F

*MITIGATION INITIATED AT TWO MINUTES

DOES NOT LEAD TO CORE UNCOVERY MUCH LESS CORE MELT

ALTERNATE 3 vs ALTERNATE 4

- MITIGATION OF MSIV PLUS ADDITIONAL FAILURES USING ALTERNATE 3
  - SORV PEAK POOL TEMPERATURE 185°F
  - RHR PEAK POOL TEMPERATURE 175°F

- RELIABILITY OF MAKE UP WATER SYSTEMS ARE SUFFICIENTLY HIGH TO PRECLUDE ASSUMING FAILURE
  - HPCS UNRELIABILITY $\sim 6 \times 10^{-3}$/DEMAND
  - OTHER WATER SOURCES AVAILABLE FOR MOST EVENTS

- ARI MAKES ATWS EVENT EXTREMELY REMOTE

<u>ALTERNATE 4</u>

- DEFEATS STANDARDIZATION
- <u>NOT</u> COST-EFFECTIVE
- ALTERNATE 4 NOT LOWER RISK FOR NEWER PLANTS

SUMMARY

- o BWR SCRAM SYSTEM RELIABILITY IS SUFFICIENT TO PRECLUDE ANY ATWS EVENT

- o IF ATWS MUST BE CONSIDERED, ALTERNATE 2 IS ALL THAT IS REQUIRED FOR ANY CURRENT BWR

- o IF REQUIRED FOR LICENSING PURPOSES FOR FUTURE BWR's ALTERNATE 3 IS MORE THAN SUFFICIENT

- o ALTERNATE 4 IS OVERLY CONSERVATIVE AND NOT COST EFFECTIVE AND SHOULD NOT BE CONSIDERED FOR ANY PRESENT OR FUTURE BWR

2-27-79

RECOMMENDED ACRS ACTION

THE ACRS LETTER SHOULD STRONGLY RECOMMEND:

- ALTERNATE 2 IS TOTALLY ADEQUATE FOR ALL CATEGORY BWR PLANTS
- IF ADDITIONAL MITIGATION CAPABILITY IS REQUIRED TO RESOLVE THE ATWS ISSUE:
    - MANUAL START AND DELIVERY OF BOTH SLC PUMPS IS ADEQUATE AND SHOULD ONLY BE REQUIRED ON PLANTS UNDER CONSTRUCTION WHERE IT IS COST EFFECTIVE TO DO SO.
    - NOTHING BEYOND ALTERNATE 3 SHOULD BE APPLIED TO PLANTS WITHOUT CP's AND FUTURE PLANTS.
- STANDARDIZATION SHOULD BE PRESERVED BY TREATING ALL GESSAR PLANTS ALIKE (IF MODIFICATIONS ARE REQUIRED ALTERNATE 3 IS SUFFICIENT).
- ALTERNATE 4 SHOULD NOT BE REQUIRED FOR ANY PLANT SINCE COMPARABLE PROTECTION IS PROVIDED WITHIN THE REQUIREMENTS OF ALTERNATE 3 AND AT MUCH LOWER COST.

B&W PRESENTATION TO THE ACRS
MARCH 8, 1979

- RISK DUE TO ANTICIPATED TRANSIENTS WITHOUT SCRAM (ATWS)

- METHODS OF REDUCING ATWS RISK

RISK DUE TO ATWS

FREQUENCY · CONSEQUENCE < SAFETY GOAL

OR

$P(\text{TRANSIENT}) \cdot P(\text{SCRAM FAILURE}) \cdot P(P>P_{MAX}/\text{ATWS}) < \text{SAFETY GOAL}$

SAFETY GOAL PREVIOUSLY SET AT APPROXIMATELY 10^{-6} EVENTS/YEAR.

FREQUENCY OF ATWS

FREQUENCY = P(TRANSIENT) · (P(CMF OF RPS) + P(CMF OF CRDM))

P(TRANSIENT) <0.5 EVENTS/YEAR FOR LOFW AND LOOP

0.5 EVENTS/YEAR DERIVED FROM EPRI AND B&W DATA FOR LOFW AND LOOP TRANSIENTS

P(CMF OF RPS) $<1.1\times10^{-5}$ FAILURES/DEMAND (BAW-10019,1970)
1.5×10^{-5} FAILURES/DEMAND (NUREG 0460,1978)

P(CMF OF CRDM) $\ll 1.5\times10^{-5}$ FAILURES/DEMAND

THEREFORE:

FREQUENCY (0.5) · (1.5×10^{-5}) $<1\times10^{-5}$

CONSEQUENCE

CONSEQUENCE $\equiv$ $P(P > P_{MAX} / ATWS)$

EXTRAPOLATING EXISTING ANALYSES B&W CONCLUDES:

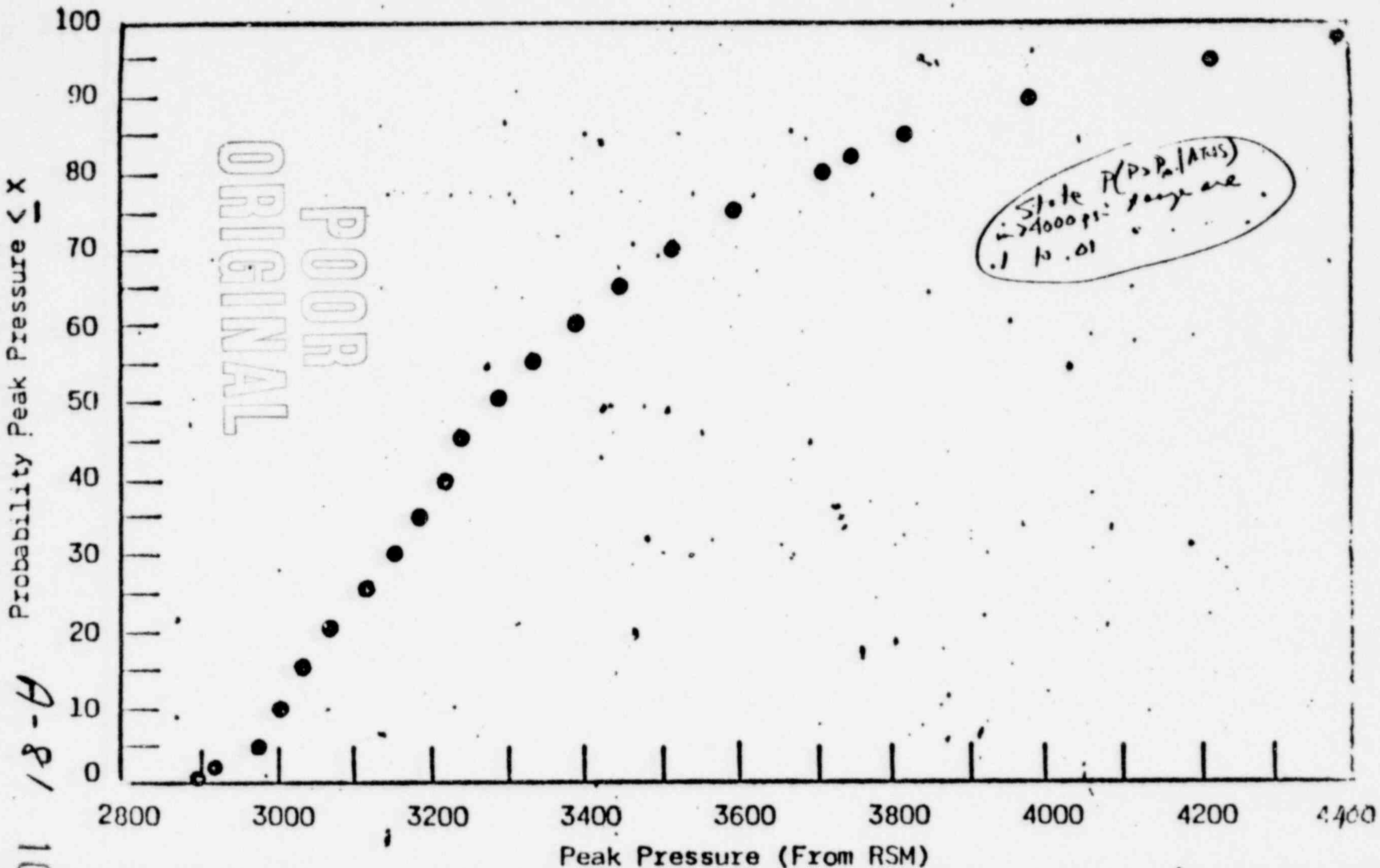
$$P_{MAX} > 4000 \text{ PSI}$$

BASED ON:

- ASME SERVICE LEVEL C STRESS LIMITS FOR FERRITIC RCPB COMPONENTS.
- BURST TEST DATA ON STEAM GENERATOR TUBES.
- PRESSURE RETAINING INTEGRITY OF RC PUMPS.
- OPERABILITY OF CRITICAL BOUNDARY VALVES.

FOR PEAK PRESSURE CONDITIONAL ON ATWS

(3800 MWt, 205 FA - [illegible]FW)



A-81

1028 130

SUMMARY

RISK DUE TO ATWS = FREQUENCY · CONSEQUENCE

$<(1\times10^{-5})\ (0.1) \leq 10^{-6}$ EVENTS/YEAR SAFETY GOAL

B&W THEREFORE CONCLUDES THAT THE RISK DUE TO ATWS IS ACCEPTABLY SMALL.

<u>METHODS OF REDUCING RISK</u>

- RISK MAY BE DECREASED BY REDUCING THE <u>FREQUENCY</u> OF ATWS:

  THE NRC IS PROPOSING DIVERSE SCRAM INSTRUMENTATION AS A MEANS OF REDUCING ATWS FREQUENCY ON 121 PWR's.

- RISK MAY BE DECREASED BY REDUCING THE <u>CONSEQUENCE</u> OF ATWS:

  THE NRC IS PROPOSING ADDITIONAL SAFETY VALVES AS A MEANS OF REDUCING ATWS PEAK RCS PRESSURES ON 41 PWR's.

RECOMMENDATIONS TO THE ACRS

B&W RECOMMENDS THE FOLLOWING:

- A 10^{-6} SIGNIFICANT EVENT/YEAR SAFETY GOAL BE ESTABLISHED FOR ALL EXISTING PLANTS -- AND CONSEQUENTLY THAT NO PLANT DESIGN MODIFICATIONS BE REQUIRED FOR B&W PWR's.
- THAT THE SAFETY GOAL BE RE-EVALUATED FOR FUTURE STANDARD PLANT DESIGNS.
- THAT DESIGN MODIFICATIONS WHICH MAY BE REQUIRED TO MEET INCREASED SAFETY GOALS BE LEFT TO THE ENGINEERING JUDGEMENT OF THE PLANT DESIGNER.

COMBUSTION ENGINEERING PRESENTATION TO THE ACRS

MARCH 8, 1979

WILLIAM E. BURCHILL
MANAGER
C-E ATWS TASK FORCE

1. C-E ATWS POSITION
2. FLAWS IN NUREG-0460, VOLUME 3
3. SPS FUNCTIONAL DESCRIPTION
4. C-E RECOMMENDATIONS TO ACRS

C-E HAS EVALUATED THE SYSTEM 80 DESIGN AND HAS FOUND ATWS NOT TO BE A SIGNIFICANT SAFETY CONCERN BECAUSE OUR CALCULATIONS SHOW THAT:

THE RISK FROM ATWS IS NEGLIGIBLE.

ATWS WOULD NOT PRODUCE VIOLATION OF 10CFR100 CRITERIA.

ATWS WOULD NOT PRODUCE A CORE MELT.

A-86

1028 144

CONSEQUENCES OF ATWS EVENTS

NO OVER-PRESSURIZATION (NO TRIP REQUIRED)

- UNCONTROLLED BORON DILUTION
- EXCESS LOAD
- PRIMARY SAMPLE LINE BREAK

SLIGHT OVER-PRESSURIZATION (LESS THAN SERVICE LEVEL C)

- FULL POWER CEA WITHDRAWAL
- IDLE LOOP STARTUP
- PARTIAL LOSS OF FEEDWATER
- LOSS OF STATION POWER

OVER-PRESSURIZATION WHICH MAY EXCEED SERVICE LEVEL C

- ZERO POWER CEA WITHDRAWAL
- LOSS OF COOLANT FLOW
- LOSS OF EXTERNAL LOAD
- COMPLETE LOSS OF FEEDWATER

ATWS ANALYSIS RESULTS FROM CENPD-158, REVISION 1

TABLE 8-2

SUMMARY OF ATWS CONSEQUENCES FOR 3800 MWt CLASS NSSSs

ATWS Event	Radiological Release, rem: Whole Body	Radiological Release, rem: Thyroid	Pressurizer Peak Pressure, psia	Reactor Fuel: Clad Collapse	Reactor Fuel: Enthalpy cal/gram	Fuel Cladding Minimum DNBR	Containment Peak Pressure, psig
ATWS Criteria	<25	<300	<3200	no	<280	>1.0*	<50
Full Power CEA Withdrawal	$<3.6 \times 10^{-4}$	<0.31	2613	no	<280	≥2.0	<18.1
Zero Power CEA Withdrawal	$<3.6 \times 10^{-4}$	<0.31	3761	no	<280	≥5.4	<18.1
Uncontrolled Boron Dilution	$<3.6 \times 10^{-4}$	<0.31	2688	no	<280	≥1.7	<18.1
Partial Loss of Coolant Flow	$<3.6 \times 10^{-4}$	<0.31	3609	no	<280	≥0.97**	<18.1
Idle Loop Startup	$<3.6 \times 10^{-4}$	<0.31	2509	no	<280	≥6.3	<18.1
Loss of External Load	$<3.6 \times 10^{-4}$	<0.31	3883	no	<280	≥2.9	<18.1
Partial Loss of Feedwater	$<3.6 \times 10^{-4}$	<0.31	3138	no	<280	≥2.9	<18.1
Complete Loss of Feedwater	$<3.6 \times 10^{-4}$	<0.31	4087	no	<280	≥2.9	<18.1
Loss of Station Power	$<3.6 \times 10^{-4}$	<0.31	2575	no	<280	≥1.5	<18.1
Excess Load	$<3.6 \times 10^{-4}$	<0.31	2525	no	<280	≥2.6	<18.1
Primary Sample Line Break	$<3.6 \times 10^{-4}$	<0.31	2577	no	<280	>2.8	<18.1

* Based on W-3 CHF Correlation (see Section 1.2)
**Peak cladding temperature = 700F

A-8.8

1028 146

C-E/ACRS

ATWS

ATWS WOULD NOT PRODUCE VIOLATION OF 10CFR100 CRITERIA OR CORE MELT

THE REACTOR COOLANT PRESSURE BOUNDARY WILL REMAIN FUNCTIONAL

NO CMF CAN DISABLE BOTH THE SCRAM FUNCTION AND THE OPERABILITY OF THE EMERGENCY SAFETY FEATURES

ALL EMERGENCY SAFETY FEATURES WILL REMAIN FUNCTIONAL

REACTOR SUBCRITICALITY CAN BE MAINTAINED BY BORON INJECTION BY THE NORMAL CHARGING SYSTEM

PLANT COOLDOWN CAN BE ACHIEVED FOLLOWING USUAL EMERGENCY PROCEDURES

CALCULATED CONSEQUENCES ARE LESS THAN 1.0% OF 10CFR100 CRITERIA

1028 147

C-E/ACRS
ATWS
MARCH 8, 1979

A-89

NUREG-0460, VOLUME 3 MISUSES RISK ASSESSMENT

RISK = FREQUENCY X CONSEQUENCE

$$\text{RISK} = \left\{ \begin{bmatrix}\text{FREQUENCY OF ANTICIPATED TRANSIENTS WITH}\\ \text{POTENTIALLY SERIOUS CONSEQUENCES}\end{bmatrix} \times \begin{bmatrix}\text{PROBABILITY OF ADVERSE INITIAL}\\ \text{CONDITIONS, EG, MTC}\end{bmatrix} \times \begin{bmatrix}\text{PROBABILITY OF FAILURE TO}\\ \text{AUTOMATICALLY INSERT ONE}\\ \text{PERCENT NEGATIVE REACTIVITY}\end{bmatrix} \right\}$$

$$\times \left\{ \begin{matrix}\text{PROBABILITY OF VIOLATION OF 10CFR100}\\ \text{CRITERIA OR CORE MELT}\end{matrix} \right\}$$

CONSEQUENCE $\ll$ 1.0, NOT 1.0, CORE MELT PER EVENT

THEREFORE, RISK FROM ATWS IS $\ll$ OTHER RISKS

A-9u

1028 148

C-F/ACRS

ATWS

GENERIC ATWS ANALYSES IN 2/15/79 NRC REQUEST ARE TANTAMOUNT TO MAKING ATWS A DESIGN BASIS EVENT

ANALYSES OF ALL PLANT TYPES

CONSERVATIVE PARAMETER ENVELOPE

CONTROL OF PLANT OPERATING PARAMETERS

MITIGATING EQUIPMENT QUALIFICATION

REAFFIRMATION OF RESULTS FOR RELOADS

WHY IS C-E SO ADAMANTLY OPPOSED TO THE ATWS DBE OPTION?

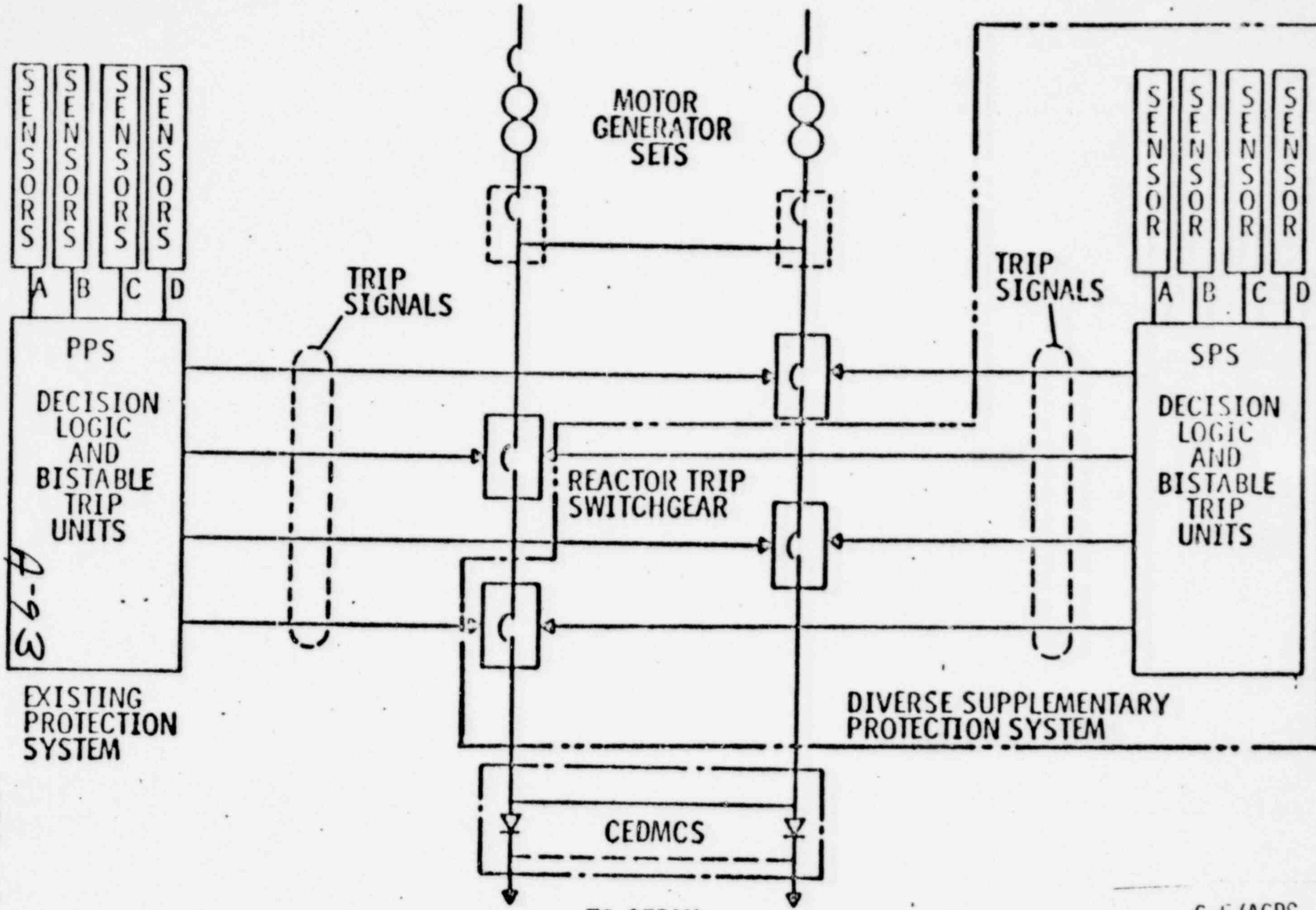
BECAUSE C-E BELIEVES THAT IT WILL COST LARGE, CONTINUING, AND UNNECESSARY AMOUNTS OF TIME, MONEY, AND MANPOWER WITH NO DEFINITIVE INCREASE IN SAFETY.

A-92

1028 150

C-E/ACRS

ATWS



A-93

1020-151

C-E/ACRS
ATWS

C-E HAS INCORPORATED INTO THE SYSTEM 80 DESIGN
A SUPPLEMENTARY PROTECTION SYSTEM AND PROVISION FOR CEDM/CEA TESTING

SPS PROVIDES DIVERSE AND REDUNDANT SENSORS (HIGH PRIMARY PRESSURE) AND LOGIC AND DIVERSE TRIP SWITCHGEAR AND POWER INTERRUPT

WITH SPS, NO CMF CAN DISABLE BOTH REACTOR SCRAM AND AUTOMATIC ACTUATION OF EMERGENCY SAFETY FEATURES

A-94

PERIODIC CEDM/CEA MOTION TESTS AND ROUTINE CEA MOTION ASSURES CEDM AND CEA/CORE INTERFACE SCRAM ABILITY

PERIODIC INSPECTION AND MAINTENANCE ASSURES CEDM AND CEA/CORE INTERFACE SCRAM ABILITY

SEARCH FOR POTENTIAL MECHANICAL CMFs PRODUCED NO CREDIBLE SCRAM FAILURE

C-E/ACRS
ATWS

1028 152

C-E RECOMMENDS THAT THE ACRS:

1. CONCUR THAT ATWS IS NOT A SAFETY PROBLEM FOR THE SYSTEM 80 REACTOR DESIGN.

2. CONCUR THAT THE DESIGN BASIS REQUIREMENT SOLUTION IS PREFERRED OVER THE DESIGN BASIS EVENT SOLUTION FOR ATWS.

3. CONCUR WITH THE C-E CONCLUSION THAT THE SUPPLEMENTARY PROTECTION SYSTEM PLUS TESTING IS A PROPER RESPONSE TO AN ATWS DESIGN BASIS REQUIREMENT FOR SYSTEM 80.

A 95.

1028 153

C-E Power Systems
Combustion Engineering, Inc.
1000 Prospect Hill Road
Windsor, Connecticut 06095

Tel. 203/688-1911
Telex 9-9297

POWER
SYSTEMS

February 28, 1979
LD-79-014

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Subject: Regulatory Requirements Review Committee ATWS Recommendations

Dear Mr. Denton:

Thank you for your letter of February 12, 1979, concerning the comments and recommendations which we provided to you regarding the Staff's report on anticipated transients without scram (ATWS), NUREG-0460, Volume 3. In response to your letter and in accordance with Notice 7590-01-M in the Federal Register, Vol. 44, No. 24, Page 6816 on February 2, 1979, we offer the following further comments and recommendations.

The Regulatory Requirements Review Committee (RRRC) has agreed with the Staff that engineering judgment should be used as the primary basis for reaching decisions on the ATWS issue with quantitative risk assessment used in a supportive role. However, we believe that the NRC staff has misused risk assessment to provide support for their recommendations for ATWS resolution. In Appendix F of NUREG-0460, Volume 3, the Staff evaluates the risk from an ATWS event by assuming that all overpressurization events lead to core melt. This implies that in the equation - risk equals frequency times consequence - the consequence term has been set equal to one core melt per event.

We believe that there would be a negligible threat to the primary coolant pressure boundary integrity or the ability to establish long-term shutdown cooling following an ATWS event. Our calculations documented in topical report CENPD-158, Revision 1, which have been reviewed by the NRC staff, indicate that even using conservative assumptions, the radiological releases which would be experienced following an ATWS event are no more than approximately one percent of the guideline values contained in 10 CFR 100. Thus, we believe that the consequence term, and thus the risk from ATWS, is several orders of magnitude lower than that which has been stated by the Staff in NUREG-0460, Volume 3. We, therefore, believe that no nuclear power plant modifications are required.

C-E/ACRS
ATWS
March 8, 1979

A-96 1028 154

We believe that the RRRC recommendation to provide analyses which are required under alternatives 3 and 4 of NUREG-0460, Volume 3, will produce an unending regulatory review. Such a review will undermine the stability of the regulatory process for as long as those analyses are considered in licensing actions. This belief, which we expressed to you in our letter of January 12, 1979, has been strongly reinforced by the letter we have received from the Division of Systems Safety dated February 15, 1979, which enclosed generic questions and guidelines for those analyses. That a request for such an extensive amount of analyses, which we believe are tantamount to making ATWS a design basis event, can be made under the description of early verification stated in NUREG-0460, Volume 3, indicates that the future course of ATWS regulation under alternatives 3 and 4 will be long and unstable.

We believe that the RRRC recommendations concerning standard plant designs are contradictory to established NRC policy on standard plant licensing. The RRRC recommended amendment of all currently effective Preliminary Design Approvals so as to provide the modifications of alternative 4 (and provision for the modifications of alternative 4 in all Final Design Approvals that are associated with those amended Preliminary Design Approvals).

In order to follow these recommendations, we would need to amend the Combustion Engineering Standard Safety Analysis Report (CESSAR) which was issued preliminary design approval No. PDA-2, dated December 31, 1975, on Docket No. STN-50-470 and defend that amendment. We would also need to submit two versions of the final CESSAR and obtain a final design approval for each version. The necessity for production of such separate designs and the added effort to obtain licensing approval for them would undermine the realization of benefits which attend the standardization process. Furthermore, the two designs thus produced would be based on entirely different principles: one on ATWS prevention, the other on ATWS mitigation. This would violate the standardization of design philosophy which is of added significance due to the possibility that it could establish a precedent.

After reviewing NUREG-0460, Volume 3, we have concluded that, if ATWS must be considered, prudent engineering judgment dictates a solution by prevention rather than one by mitigation. It was based on this same belief that we modified the System 80 standard design to include the supplementary protection system (SPS) and proposed this design modification for NRC review. We believe that such an approach provides regulatory stability because it does not depend on detailed engineering analyses, the results of which are strongly dependent upon plant operating and design parameters.

We recommend that the proposed ATWS regulation exempt our pre-CESSAR plants from any design modification because the record supports the conclusion that these plants pose an insignificant societal risk due to an ATWS event. We also recommend that, if the proposed regulation is to require design modifications of the standard design described in CESSAR, that such modifications be applied uniformly and include no more than provision of the supplementary protection system as specified in NUREG-0460, Volume 3. We hope that the regulation would be specific and clearly defined in order not to be subject to future differences in interpretation of ATWS requirements.

C-E/ACRS

A-97 1028 155 ATWS

March 9, 1979

Our comments and recommendations are made in the context of the mutual effort by NRC and the nuclear industry to bring the long discourse on ATWS to a conclusion, and because of the necessity to eliminate ATWS as a destabilizing influence on the regulatory process. We continue to believe that ATWS in our nuclear steam supply systems does not present a valid concern for the health and safety of the public today, nor will it in the future regardless of the number of such systems which are in operation. This belief and our recommendations are supported by our extensive engineering evaluations of both the reliability of current and planned reactor protection systems and our predicted consequences of an ATWS should it actually occur.

We would be pleased to discuss the bases for our comments and recommendations more fully with you. It is our sincere hope that such discussion would assist in the expeditious resolution of the ATWS issue.

Very truly yours,

COMBUSTION ENGINEERING, INC.

A. E. Scherer
Licensing Manager

AES:lgw

cc: Professor Max Carbon (ACRS)
Professor William Kerr (ACRS)
Mr. Edson G. Case (RRRC)
Dr. Roger J. Mattson (NRC)

C-E/ACRS
ATWS
March 8, 1979

A-98 1028 156

C-E Power Systems
Combustion Engineering, Inc.
1000 Prospect Hill Road
Windsor, Connecticut 06095

Tel. 203/688-1911
Telex: 9-0297

POWER
SYSTEMS

February 28, 1979
LD-79-015

Dr. Roger J. Mattson, Director
Division of Systems Safety
Office of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

SUBJECT: NRC REQUEST FOR GENERIC ATWS ANALYSES

Dear Dr. Mattson:

This letter responds to your letter dated February 15, 1979, and the enclosed generic questions and guidelines concerning anticipated transients without scram (ATWS). Your letter requests that we identify generic classes of Combustion Engineering nuclear steam supply systems (NSSSs), place each NSSS which we have designed into one of these classes, specify design modifications for each NSSS class according to the requirements of NUREG-0460, Volume 3, and provide analyses of the performance of the modified plants during ATWS events. Most analyses are required to be submitted by April 15, 1979 with the balance due by June 1, 1979.

Our initial evaluation has shown that the time allowed is insufficient to respond to the requests in your letter. We are also concerned that response to certain of the requests will necessitate engineering techniques which have not been previously reviewed by NRC and hence will lead to an escalation of review, thus jeapordizing the timely resolution of the ATWS issue which we both seek. For example, we are particularly concerned with the request for assurance of performance of pressurizer safety and relief valves starting on page 32 of the enclosure to your letter. Finally, we have thus far received no authorization from our customers to do the activities outlined above. Such authorization is even more necessary due to the request for extensive analyses of equipment which is outside of our normal scope-of-supply.

In general, it is our belief that commitment of the large amount of engineering manpower, time, and resources which would be necessary to respond to your request is counterproductive because of the associated diversion of these resources from other important tasks

C-E/ACRS

1028 157

ATWS

A-99

March 2, 1979

We, therefore, request that you reconsider the scope of effort, as well as the schedule, set forth in your letter. We believe that a rule can be formulated without the need for such extensive and detailed additional analyses.

In accordance with your request, we will be happy to meet with your staff on March 1, 1979, to discuss our technical concerns.

Very truly yours,

COMBUSTION ENGINEERING, INC.

A. E. Scherer
Licensing Manager

AES:lgw

cc: Mr. Harold R. Denton, NRC
Professor Max Carbon, ACRS
Professor William Kerr, ACRS

C-E/ACRS
1028 158 ATWS
A-100
MARCH 9, 1979

SCHEDULE
WM. H. ZIMMER NUCELAR POWER STATION, UNIT 1
WASHINGTON, DC

I. Subcommittee Report 3:00 pm - 3:30 pm

1. Status of Zimmer Review - M. Bender

2. Status of Mark II Generic Review - M. Plesset

II. Discussions with the NRC Staff and the Applicants 3:30 pm - 7:00 pm

1. NRC Staff Report on Zimmer Review 3:30 pm - 4:15 pm (45 minutes)

a. Revisions to Zimmer SER

b. Open Issues

c. Status of Mark II Review

d. Interim ATWS Position on Zimmer

2. Technical Presentations by Applicant 4:15 pm - 5:00 pm (45 minutes)

a. Organization

b. Site Description

c. Plant Description with Emphasis on Mark II Containment and Significant Changes to NSSS (Discussions of the Recirculation Pump Trip and the Reactor Flow Control System Should Be Included).

d. Training Programs, Emergency Plannings, and QA and QC Programs

e. Plant Staffing

BREAK 5:00 pm - 5:15 pm

3. NRC Staff Report - Mark II Containment — 5:15 pm - 6:00 pm (45 minutes)

   a. Lead Plant Acceptance Criteria

   b. Zimmer Design Assessment

   c. Zimmer SRV Tests

   e. Generic Acceptance Criteria

4. Applicant Response to Items 1 and 3 — 6:00 pm - 6:30 pm (30 minutes)

5. General Discussions and Conclusions — 6:30 pm - 7:00 pm (30 minutes)

March 2, 1979

PROJECT STATUS REPORT
WILLIAM H. ZIMMER NUCLEAR POWER STATION, UNIT 1

The William H. Zimmer Nuclear Power Station, Unit 1 is located in Ohio on the Ohio River approximately 24 miles southeast of Cincinnati and 1/2 mile north of the small town of Moscow, Ohio. The site is approximately 632 acres and is located between the Ohio River and a secondary road which parallels the river. (See Attachment A). The application was filed by the Cincinnati Gas and Electric Company, the Columbus and Southern Ohio Electric Company, and the Dayton Power Light Company. The Cincinnati Gas and Electric Company is authorized to act as the agent for the other two power companies and is primarily responsible for design, construction, and opetation of this station. The original application was for two units. Construction of second unit, however, has been cancelled. The architect-engineer is Sargent & Lundy, the construction contractor is Kaiser Engineers, and the nuclear steam supply system supplier is the General Electric Company. The nuclear steam supply system is a BWR/5 type utilizing the 8 X 8 General Electric fuel that is similar to the Hatch, Unit 2. (See Attachment B). The containment system will be a Mark II type and is the first of this type to be considered by the Committee for a license to operate. The Mark II containment design and the NRC Staff's acceptance criteria for the design basis pool dynamic loads have been reviewed by the Zimmer Subcommittee and the Fluid Dynamics Subcommittee. A summary of Mark II containment lead plant load evaluation and acceptance criteria and a report issued by Dr. Thomas Eaton, a former ACRS Fellow, are included under this tab in this notebook.

The application for a construction permit was docketed on April 7, 1970 and the construction permit was issued on October 27, 1972. A copy of the Committee's letter is included as Attachment C. The application for the operating license was docketed on September 10, 1975. There will be a hearing on this application. The construction is estimated to be about 95% complete and the Applicants have scheduled fuel loading for June of 1979. It is not clear at this point that the date for fuel loading will be met.

The plant is designed to withstand a SSE of 0.2g and an OBE of 0.1g. The bedrock surface at the site is relatively flat and at an approximate elevation of 410 ft. above mean sea level. The Staff has required that existing foundation materials to the 450 ft. elevation be removed and replaced with compacted fill. The site lies on a 0.5 mile wide alluvial plain on the Ohio River with the approximate elevation of the flood plain at the 500 ft. elevation. Plants structures are to be located at an average final plant grade elevation of 520 ft. The compacted fill will be dewatered. The Applicants have not yet accepted the Staff's requirement for a dewatering of the fill to an elevation of 457 ft. It would, however, appear that this disagreement will be resolved without involving any significant technical problems.

The NRC Staff has issued acceptance criteria for the Mark II containment design to accommodate pool dynamic loads. The Applicants have taken issue with two of these criteria (The treatment of the bubble release on the quencher air clearing loads and the treatment of the LOCA jet submerged drag). The Staff has the matter under review and it appears that it will be resolved in the very near future. The generic acceptance criteria for the Mark II containment design are considered by the Staff as applying all lead plants. It is possible, however, that the continued research will make it possible to modify these criteria as to be less conservative. The Staff has required that the Applicants evaluate all loads using the linear sum load combination method. Nearly all of the structures in the Zimmer plant are acceptable under this criteria.

The Zimmer design will utilize valve flow control rather than pump speed flow control (such as was used on the BWR/4 design) to regulate the flow of the primary coolant. The procedure used is to start the recirculation pumps on the 100% speed power source to unseat the pump bearings. The suction and block valves are fully opened and the flow control valves are in the minimum position. When the pumps are near full speed, the main power source is tripped and the pumps are allowed to coast down to near 25% speed where the low frequency motor generator set will power the pumps and motors at 25% speed. The flow control valves are then opened to the maximum position at which point the reactor heatup and pressurization can commence. When the reactor power is greater than 30%, the low feedwater flow interlock is cleared and the recirculation pumps get switched to the 100% speed power source. The flow control valves are closed to the minimum position before the speed change to prevent large increases in reactor core power. An interlock has been installed on each pump to prevent system startup or transfer from 25% to 100% pump speed unless the flow control valve is in the minimum position. This is to prevent a reactivity insertion due to the sweeping of the voids from the core should the transfer to the 100% speed occur with the flow control valve in a maximum position.

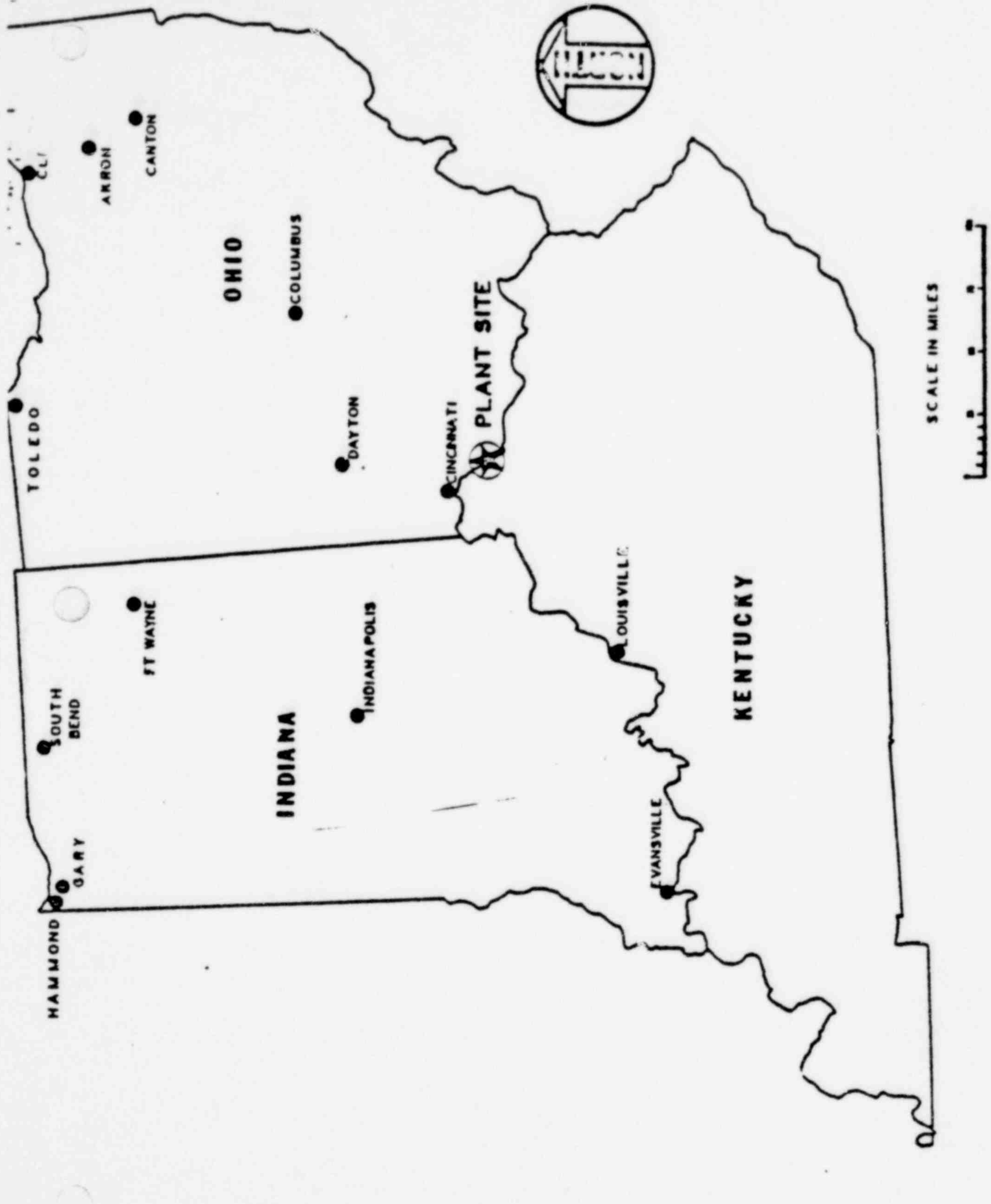
An article appeared in the Cincinnati Post on February 15, 1979 which reported an interview with a Mr. I. T. Yin, an NRC inspector from the Region III Office. The article dealt with design deficiencies in the pipe hanger supports and QA and QC deficiencies at the Zimmer plant. Mr. Vandel and Mr. Yin were at the February 27, 1979 Subcommittee meeting representing the Region III Office and these matters were discussed with them. A copy of the newspaper article and material summarizing the Region III concerns is included as Attachment C.

The Hearing Board has allowed intervention petitions on the Zimmer application from the City of Cincinnati, Dr. David B. Frankhauser, Mrs. M. B. Snell, and the Miami Valley Power Project. The Subcommittee has received no written statements or requests for time for oral statements. No significant differences of opinion among the NRC Staff have been identified.

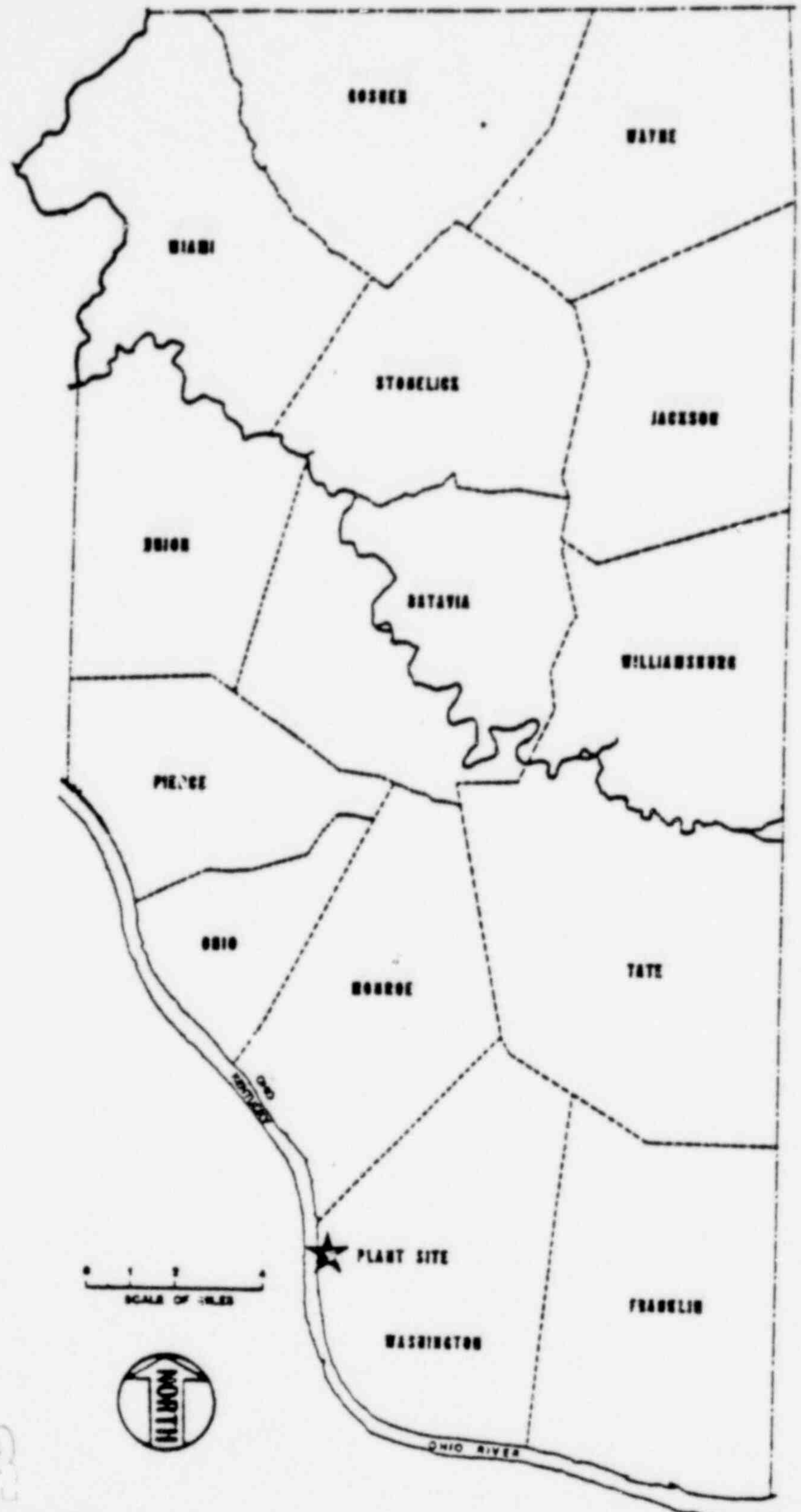
$\boxed{1}$

ATTACHMENT A

A-XII-6


OHIO
TOLEDO
AKRON
CANTON
COLUMBUS
DAYTON
CINCINNATI
PLANT SITE
INDIANA
FT WAYNE
SOUTH BEND
INDIANAPOLIS
GARY
HAMMOND
LOUISVILLE
EVANSVILLE
KENTUCKY
SCALE IN MILES

A-XII-7



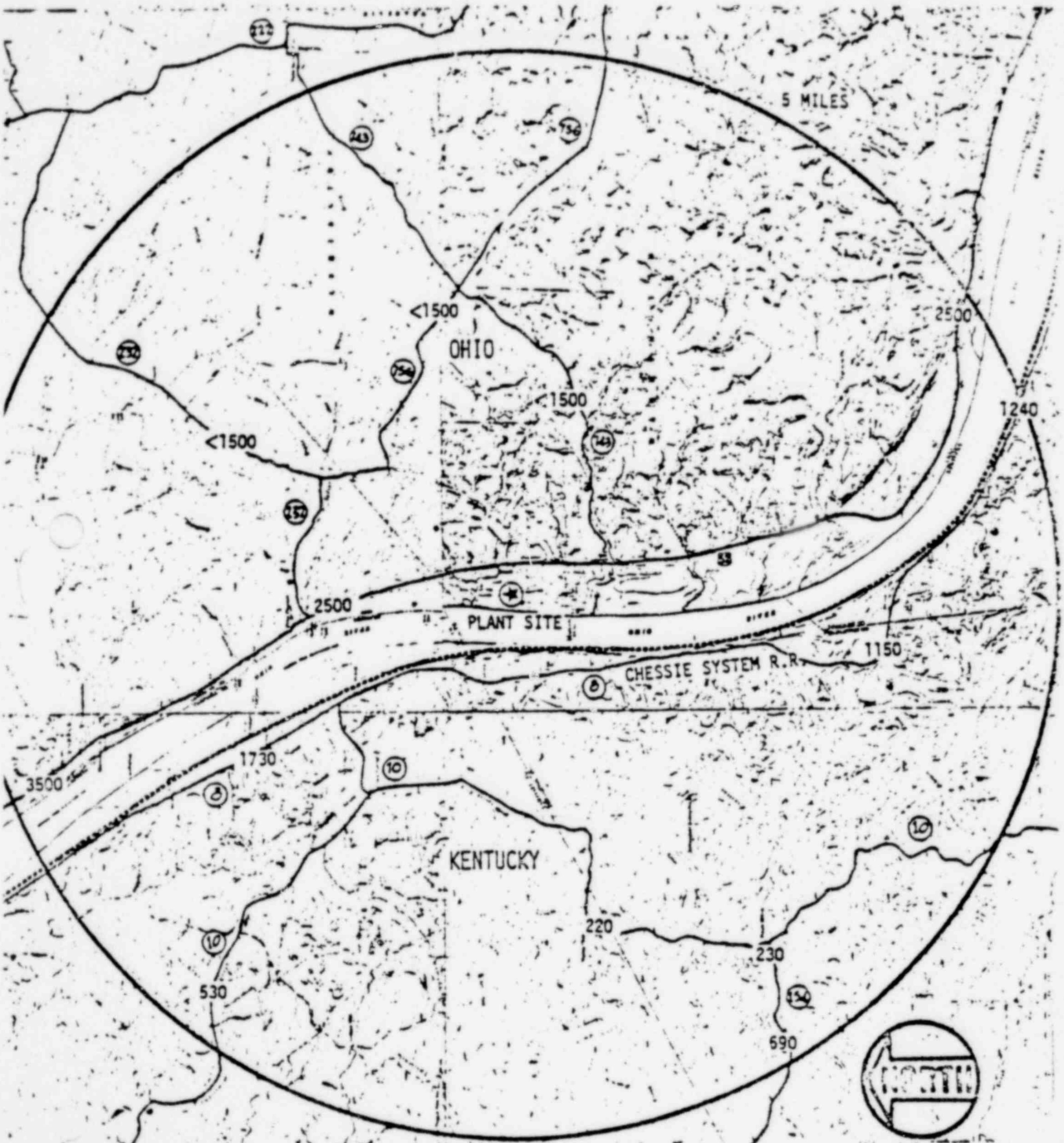
POOR ORIGINAL

WM. H. ZIMMER NUCLEAR POWER STATION, UNIT 1
FINAL SAFETY ANALYSIS REPORT

FIGURE 2.1-2

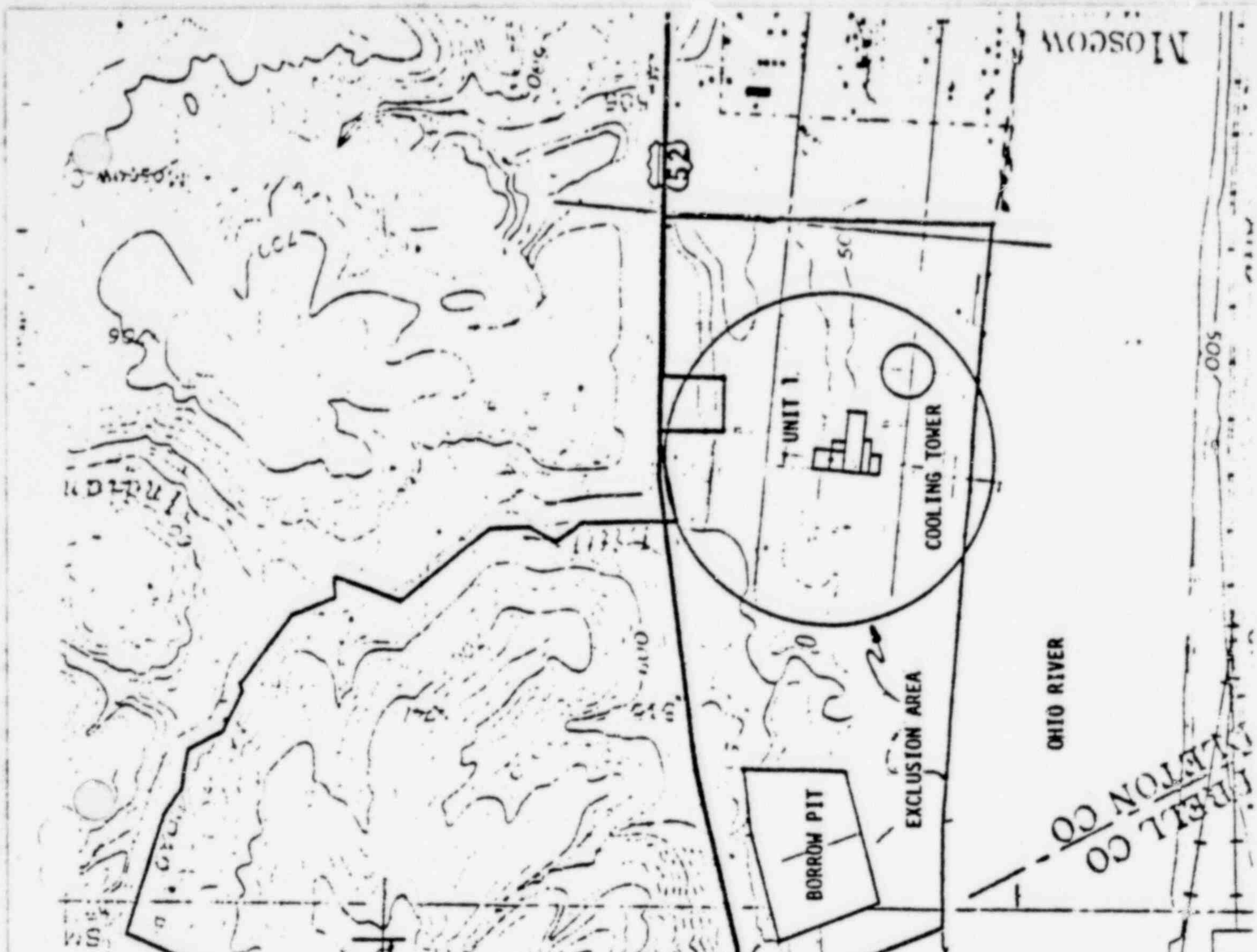
LOCATION OF THE SITE IN

A-XII-8



POOR ORIGINAL

A-XII-9



UNIT 1
COOLING TOWER
EXCLUSION AREA
BORROW PIT
OHIO RIVER
METEOROLOGICAL TOWER
PROPERTY LINE
METEOROLOGICAL TOWER
BARGE LANDING SLIP (KENTUCKY SIDE)
Moscow
52
IVER

Attachment B

A XII - 11

1028 169

TABLE 1-1 (Continued)

Design Feature	Hatch Unit 2	Hatch Unit 1	Zimmer Unit 1
Number of Recirculation Loops	2	2	2
Recirculation Loop Inside Diameter, inches	28	28	20
Recirculation Pump Capacity, gallons per minute	45,200	45,200	33,880
Number of Jet Pumps	20	20	20
Number of High Pressure Coolant Injection Pumps	1	1	1[#]
Number of Core Spray Loops	2	2	2[##]
Number of Low Pressure Coolant Injection Pumps	4	4	3
Number of Containment Spray Loops	2	2	2
Maximum Heat Flux, British thermal units per square foot per hour	361,594	428,300	354,000
Average Heat Flux, British thermal units per square foot per hour	145,528	164,410	143,900
Maximum Power per Fuel Rod Length, kilowatts per foot	13.4	18.5	13.4
Average Power per Fuel Rod Length, kilowatts per foot	5.39	7.11	5.45
Maximum Fuel Temperature, degrees Fahrenheit	3435	4380	3325
Minimum Critical Power Ratio	1.30	1.32	1.24
Total Peaking Factor	2.49	2.60	[illegible].43

[#] High pressure core spray used on Zimmer.

[##] One low pressure core spray used on Zimmer.

TABLE 1-1

COMPARISON OF PRINCIPAL DESIGN FEATURES
OF ZIMMER AND SIMILAR FACILITIES

Design Feature	Hatch Unit 2	Hatch Unit 1	Zimmer Unit 1
Rated Thermal Power, megawatts	2436	2436	2436
Gross Electrical Output, megawatts	822	813	839
Net Electrical Output, megawatts	795	786	797
Main Steam Flow Rate, pounds per hour	10,470,000	10,030,000	10,470,000
Total Reactor Core Flow Rate, pounds per hour	77,000,000	78,500,000	78,500,000
Feedwater Temperature, degrees Fahrenheit	420	387.4	420
Reactor Operating Pressure, pounds per square inch gauge	1005	1005	1020
Fuel Lattice	8x8	7x7	8x8
Number of Fuel Assemblies	560	560	560
Number of Control Rods	137	137	137
Reactor Vessel Inside Diameter, inches	218	218	218
Reactor Vessel Inside Height, feet	69.3	69.3	69.3
Reactor Vessel Design Pressure, pounds per square inch gauge	1250	1250	1250
Reactor Vessel Wall Thickness, inches	5-17/32	5-17/32	5-3/8

SUMMARY OF MARK II CONTAINMENT
LEAD PLANT LOAD EVALUATION AND ACCEPTANCE CRITERIA

The Zimmer Plant has the first Mark II containment to be reviewed by the ACRS for an Operating License. The Mark II design (Figure 1) consists of an over/under drywell/wetwell pressure suppression containment system. A LOCA within the primary system causes a pressurization of the drywell with an air-steam-water mixture which flows through the downcomer pipes into the wetwell. The water in the wetwell condenses the steam flowing in from the drywell and pressures remain below those found in dry containments. Discharge lines from the reactor vessel safety relief valves (SRVs) are routed to the wetwell for condensation of steam.

During testing of the Mark III containment design certain loads associated with the injection of air and steam into the water pool were identified. The Mark II containment was reevaluated to treat these newly identified loads.

The loads identified are discussed below and Figure 2 shows the time sequence of the loads.

DESCRIPTION OF LOCA-RELATED HYDRODYNAMIC PHENOMENA

Assuming the instantaneous rupture of a steam or recirculation line, a sonic wave exits the broken primary system pipe and expands into the drywell atmosphere. This wave rapidly attenuates as a front expanding spherically outward into the drywell. The wave then enters the vent system, progressing into the pool.

Since there would be a very rapid drywell pressure increase associated with the postulated LOCA, a compressive wave could be formed in the water that initially occupies the downcomers. Prior to clearing of this water from the downcomers, this compressive wave could propagate through the suppression pool and result in a dynamic loading on the suppression chamber and structures within the suppression pool.

As the drywell pressure increases, the water initially in each main vent downcomer accelerates into the pool clearing the vents of water. During this water clearing process, a jet forms in the suppression pool which creates water jet impingement and drag loads on structures near the vent outlet and on the suppression pool basemat. In addition, jet formation can occur asymmetrically leading to lateral reaction loads on the vents. During the vent-clearing transient, the diaphragm will be subject to a downward pressure differential. Immediately following vent water clearing, a bubble of air from the drywell starts to form at the vent exit. The steam in the air-steam mixture flowing through the vents condenses in the pool. As the air bubble forms, its pressure is nearly equal to the drywell pressure at the time of vent clearing. This results in a pressure disturbance in the pool. The dynamic bubble pressure is geometrically attenuated through the suppression pool water and results in loads on submerged structures and on the suppression pool structure.

When the air flows from the drywell through the vent system, the bubbles initially formed expand. Continued injection of drywell air and expansion of the air bubble results in a rise of the suppression pool surface. Structures close to the pool surface experience impact loads as the rising pool surface strikes the lower surface of the structures, followed by drag loads as the pool surface continues to rise past the structures. In addition, the rising pool surface compresses the air in the upper half of the suppression chamber causing a net upward load on the diaphragm.

As the pool surface rises, the air bubble collapses, terminating the potential for the upward loading, and the water slug breaks up. Breakup of the slug occurs at a height of about 1.5 times the initial submergence of the vents. Subsequent pool swell evolves into a two-phase air water froth. There is no substantial froth pool swell due to the compression of the air space above the pool surface. Gravity induced fall back of the froth returns the pool to the original pre-LOCA elevation.

Following air carryover, there will be a relatively long period of decreasing steam flow through the vent system. During this time, vent flow occurs in three distinct phases:

1. High mass flux, characterized by nearly steady-state condensation;
2. Medium mass flux, characterized by periodic variations in condensation rate; and
3. Low mass flux chugging, characterized by intermittent condensation.

During steam condensation, the vents experience a lateral loading caused by random movement of the steam-water interface. The magnitude of this load varies with steam mass flux and suppression pool temperature. Maximum lateral loads in a postulated LOCA occur toward the end of blow down. The same condensation phenomenon also results in pressure loadings on the suppression pool boundary.

Shortly after a postulated LOCA, the ECCS will automatically pump condensate water and/or suppression pool water into the reactor vessel. This water floods the reactor core and subsequently cascades into the drywell through the postulated break in the pipe. The time at which this will occur depends upon break size and location. Because the drywell will be full of steam when the vessel is reflooded, the sudden introduction of water into the drywell causes steam condensation and depressurization. As the drywell pressure falls below the suppression chamber pressure, the vacuum relief system will allow air from the suppression chamber to reenter the drywell. Eventually, sufficient air will return to equalize the drywell and suppression chamber pressures.

The magnitude and timing of LOCA pool swell and steam condensation pool dynamic loads depends on the break size. A spectrum of LOCA break sizes was considered in order to establish the limiting design conditions for Mark II containments. The LOCA conditions which were considered include the following accident conditions:

A XII-16

1. Design Basis Accident (DBA), a double-ended break of a recirculation line or main steam line.
2. Intermediate Break Accident (IBA), a break such that the high pressure subsystem of the ECCS cannot maintain reactor water level; however, vessel depressurization does not occur. An IBA corresponds to a liquid or steam line break of about 0.1 $ft.^2$
3. Small-Break Accident (SBA), a break that will not result in reactor depressurization due either to loss of reactor fluid or automatic operation of the ECCS.

The DBA is the design limiting case for the pool swell related pool dynamic loads including jet, drag, impact and fallback loads. The IBA and SBA cases have a much lower rate of drywell pressurization. Therefore, for these cases the IBA and SBA pool swell loads are correspondingly lower. However, LOCA related steam loads can occur over a wider spectrum of breaks since the maximum condensation loads occur at low vent mass flux. Condensation oscillations and chugging may occur over an extended period of time for small breaks as a result of the reduced reactor vessel depressurization rate compared to a DBA.

DESCRIPTION OF SRV-RELATED HYDRODYNAMIC PHENOMENA

BWR plants are equipped with safety/relief valves (SRV) to control primary system pressure. Small pressure variations can be controlled by changing power level and/or load. However, more rapid transients such as a turbine trip cannot be handled by such means. For these transients, SRVs mounted on the main steam line are actuated to divert either a portion or all of the generated steam into the suppression pool. These valves are actuated at individual pre-set pressure levels or by an external signal (ADS). The series of SRVs are individually set at pressures over a range, such that only the required number of valves to control the pressure transient will actuate. Upon SRV actuation, the air column within the partially submerged SRV discharge line is compressed by the high pressure steam and, in turn, accelerates the water column into the suppression pool. The water jet or jets thus formed create pressure and velocity transients which are manifested as drag or jet impingement loads on submerged structures.

Following water clearing, the compressed air is also accelerated into the suppression pool forming a high pressure air bubble. This bubble executes a number of oscillatory expansions and contractions before rising to the suppression pool surface. The associated transients again create drag loads on submerged structures as well as pressure loads on the submerged boundaries. These loads are referred to as SRV air clearing loads.

Following the air clearing phase essentially pure steam is injected into the pool. Experiments indicate that the steam jet-water interface which exists at the discharge line exit during this phase is relatively stationary so long as the local pool temperature is low. Thus, the condensation proceeds in a stable manner and no significant loads are experienced. Continued steam blowdown into the pool will increase the local pool temperature. The condensation rates at the turbulent steam-water interface are eventually reduced to levels below that needed to readily condense the discharged steam. At this "threshold" level, the condensation process becomes unstable; i.e.,: steam bubbles are formed and shed from the pipe exit, the bubbles oscillate and collapse giving rise to severe pressure oscillations which are imposed on the pool boundaries. Current practice to deal with this phenomenon in BWR plants is to restrict the allowable operating temperature envelope via the Technical Specifications such that the threshold temperature is not reached. This restriction is referred to as the pool temperature limit.

The Mark II plants have committed to the use of a T-quencher device which contains many small holes to break up the steam flow. This allows stable condensation nearly up to the boiling point of water.

Following identification of these loads, the Mark II Owners and GE initiated a program of testing to evaluate the magnitude of the loads. In November 1975, they submitted the Dynamic Forcing Function Report (DFFR) to the NRC describing a generic methodology for determining Mark II pool dynamic loads. In May 1977, the program was modified to include a Lead Plant Program (LPP) for Zimmer, Shoreham, and Lasalle and a Long Term Program (LTP). The LPP has concentrated on establishing conservative design bases for the lead plants. The LTP will provide a more realistic load evaluation for design and construction of the plants to be licensed following the lead plants. Documentation of the load evaluation for the LPP was completed in the second quarter of 1978 and the NRC Staff issued its report, NUREG-0487, "Mark II Containment Lead Plant Program Load Evaluation and Acceptance Criteria," in October 1978.

ACRS Subcommittee meetings on July 7-8, 1977; November 30, 1977; May 23, 1978; and November 28-30, 1978 reviewed the Mark II and SRV load definition and acceptance criteria.

There are approximately 39 load specifications covered by the NRC acceptance criteria; 14 were derived from the original Mark II Owners proposed criteria, 5 involve plant unique analysis, and 20 were developed by the NRC. Of the 20 developed by the NRC, 8 have been adopted by the Owners Group, 6 were recently resolved, and 6 issues remain open with resolution pending.

The NRC Staff indicated at the February 27, 1979 Zimmer Subcommittee meeting that they expect resolution of the last 6 items by the end of March. Agreement has been reached in general on the items, however, some documentation is still outstanding from the Applicants. The Staff expects to issue a generic report closing out the open items in March following their review of the needed documentation.

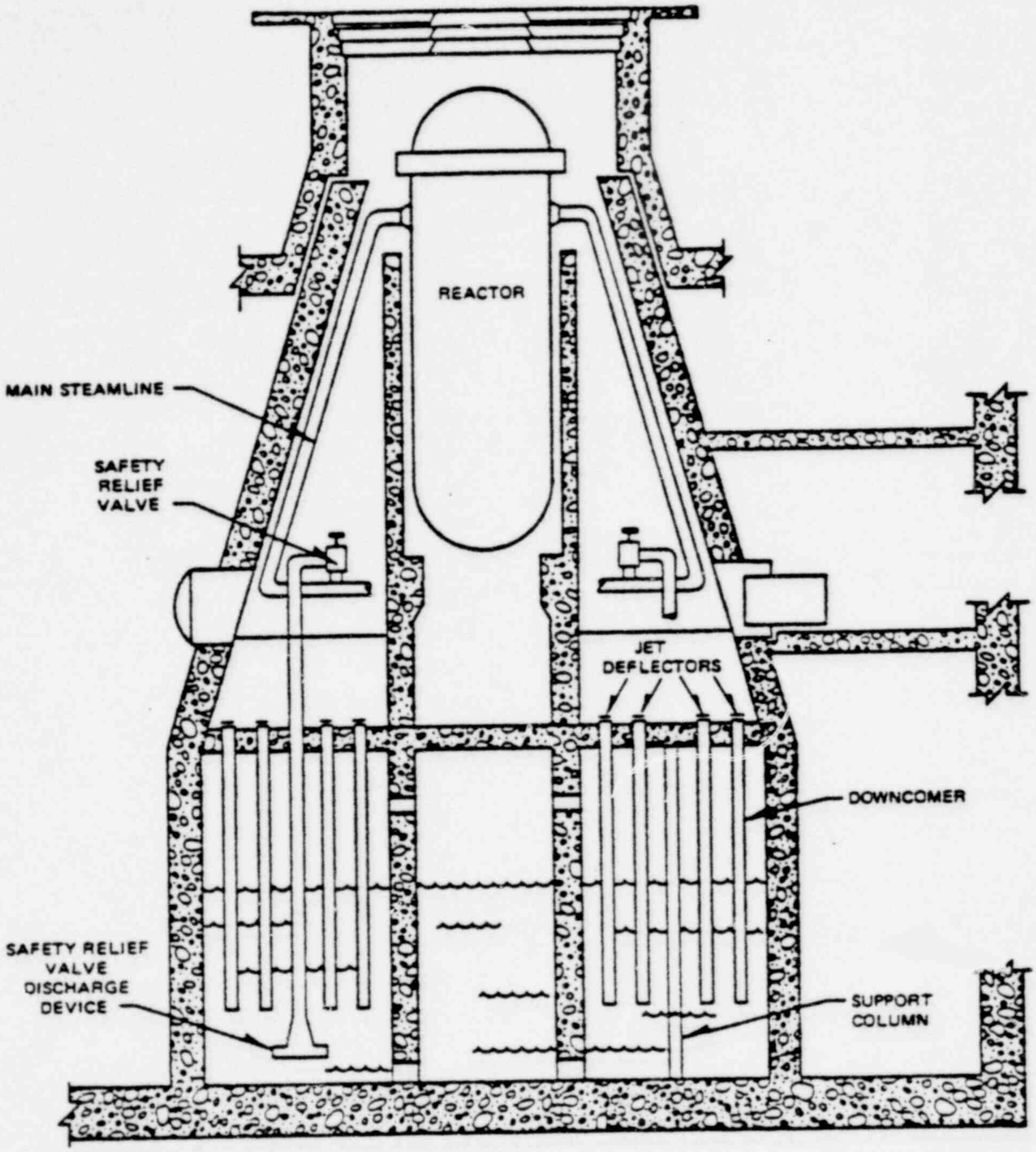
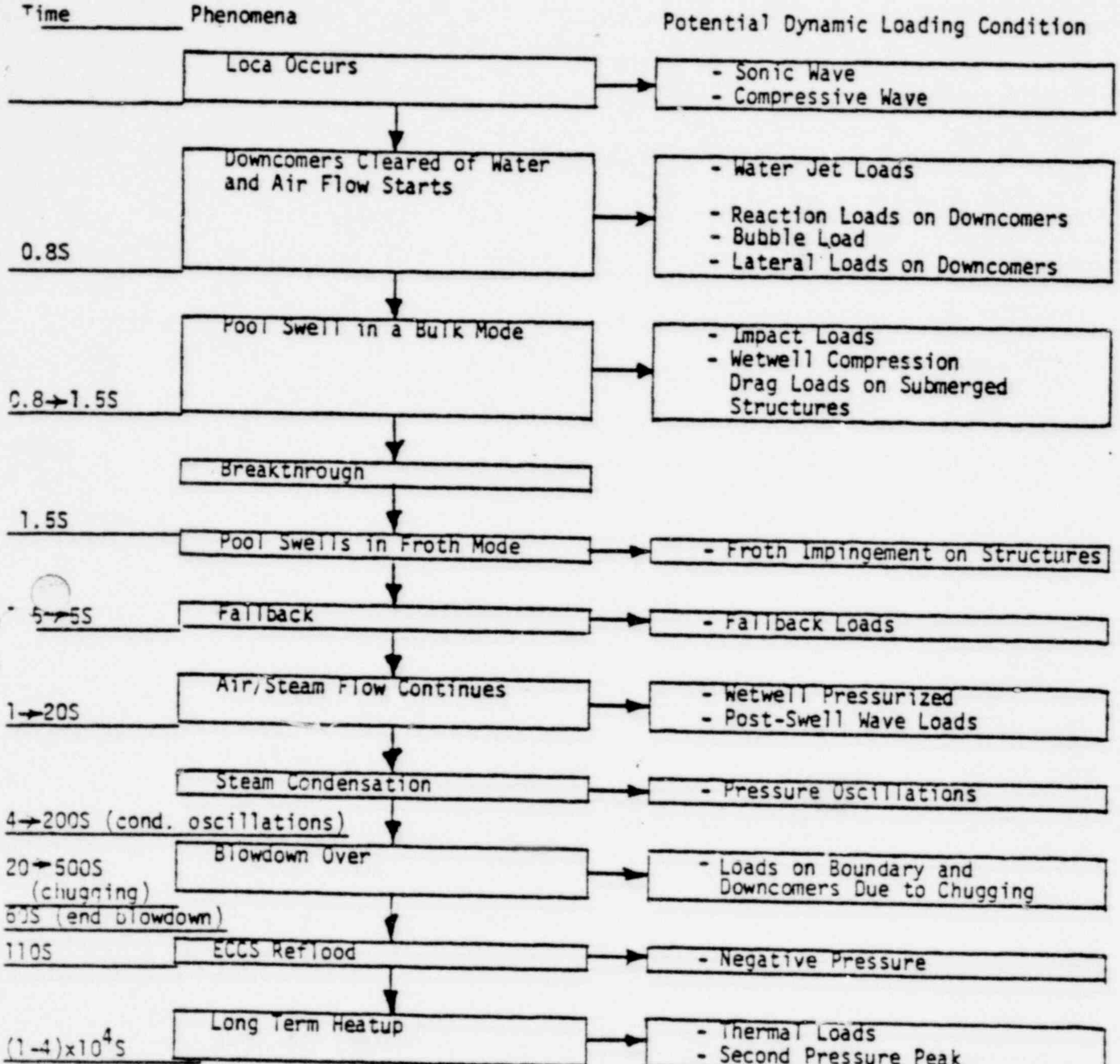


Figure II-1 Typical Mark II Pressure Suppression Containment

Figure II-2 LOCA Sequence of Events



Peak drywell and wetwell pressure @ 50S

Maximum diaphragm Δ P down @ 0.7S

Maximum diaphragm Δ P up @ 2.0S

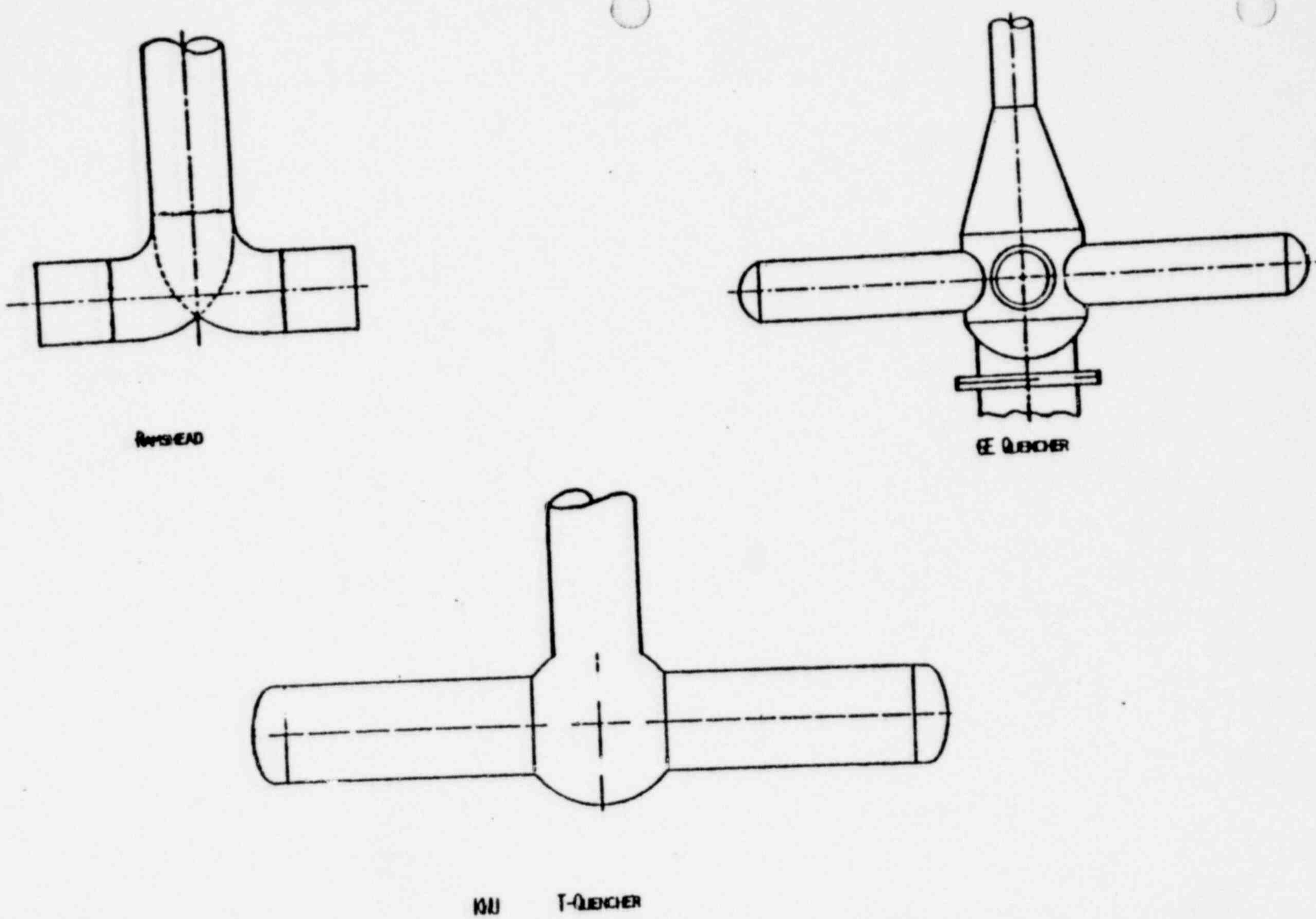


Figure III-3, Safety Relief Valve Discharge Devices

A XII-22

Table IV-1

Mark II Pool Dynamic Load Summary Table

Load or Phenomenon	Mark II Owners Group Load Specification	Reference	NRC Review Status	LER Section
I. LOCA-Related Hydrodynamic Loads				
A. Submerged Boundary Loads During Vent Clearing	33 psi over-pressure added to local hydrostatic below vent exit (walls and basemat) - linear attenuation to pool surface	DFFR - Rev. 2	Acceptable	III.B.2
B. Pool Swell Loads				
1. Pool Swell Analytical Model				
a) Air Bubble Pressure	Calculated by the Pool Swell Analytical Model(PSAM) Used in calculation of submerged boundary loads.	DFFR - Rev. 2 NEDE-21544-P	Acceptable	III.B.3.a.1
b) Pool Swell Elevation	1.5 x submergence	DFFR - Rev. 2	NRC Criteria I.A.1	III.B.3.a.2
c) Pool Swell Velocity	Velocity history vs. pool elevation predicted by the PSAM used to compute impact loading on small structures and drag on gratings between initial pool surface and maximum pool elevation and steady state drag between vent exit and maximum pool elevation. Analytical velocity variation used up to maximum velocity. Maximum velocity applies thereafter up to maximum pool swell.	DFFR - Rev. 2 NEDE-21544-P	NRC Criteria I.A.2	III.B.3.a.3
d) Pool Swell Acceleration	Acceleration predicted by the PSAM. Pool acceleration is utilized in the calculation of acceleration drag loads on submerged components during pool swell.	DFFR - Rev. 2 NEDE-21544-P	Acceptable	III.B.3.a.4
e) Wetwell Air Compression	Wetwell air compression is calculated by the PSAM. Defines the pressure loading on the wetwell boundary above the pool surface during pool swell.	DFFR - Rev. 2 NEDE-21544-P	Acceptable	III.B.3.a.5
f) Drywell Pressure History	Plant unique. Utilized in PSAM to calculate pool swell loads.	Plant Unique FSAR NEDM-10320	Acceptable if based on NEDM-10320. Otherwise plant unique reviews required.	III.B.3.a.6

Mark II Pool Dynamic Load Summary Table

Load or Phenomenon	Mark II Owners Group Load Specification	Reference	NRC Review Status	LER Section
2. Loads on Submerged Boundaries	Maximum bubble pressure predicted by the PSAM added uniformly to local hydrostatic below vent exit (wells and basemat) linear attenuation to pool surface. Applied to walls up to maximum pool swell elevation.	DFFR - Rev. 2 NEDE-21544-P	Acceptable	III.B.3.b
3. Impact Loads				
a) Small Structures	1.5 x Pressure-Velocity correlation for pipes and I beams. Constant duration pulse	DFFR - Rev. 2	NRC criteria I.A.6	III.B.3.c.1
b) Large Structures	None - Plant unique load where applicable	FSAR	Plant unique review where applicable	III.B.3.c.2
c) Grating	No impact load specified. P_{drag} vs. open area correlation and velocity vs. elevation history from the PSAM.	DFFR - Rev. 2	NRC Criteria I.A.3	III.B.3.c.3
4. Wetwell Air Compression				
a) Wall Loads	Direct application of the PSAM calculated pressure due to wetwell compression.	DFFR - Rev. 2 NEDE-21544-P	Acceptable	III.B.3.d.1
b) Diaphragm Upward Loads	2.5 psid	DFFR - Rev. 2	NRC Criteria I.A.4	III.B.3.d.2
5. Asymmetric Load	None	DFFR - Rev. 2	NRC Criteria I.A.5	III.B.3.e
C. Steam Condensation and Chugging Loads				
1. Downcomer Lateral Loads				
a) Single Vent Loads	8.8 KIP static	DFFR - Rev. 2	NRC Criteria I.B.1	III.B.4.a.1
b) Multiple Vent Loads	Prescribes variation of load per downcomer vs. number of downcomers	DFFR - Rev. 2	NRC Criteria I.B.2	III.B.4.a.2

A-XII-24

1028 182

Mark II Pool Dynamic Load Summary Table

Load or Phenomenon	Mark II Owners Group Load Specification	Reference	NRC Review Status	LER Section
2. Submerged Boundary Loads				
a) High Steam Flux Loads	Sinusoidal pressure fluctuation added to local hydrostatic. Amplitude uniform below vent exit-linear attenuation to pool surface. 4.4 psi peak-to-peak amplitude. 2-7 Hz frequencies.	January, 1977 Application memorandum	Acceptable	III.B.4.b.2
b) Medium Steam Flux Loads	Sinusoidal pressure fluctuation added to local hydrostatic. Amplitude uniform below vent exit-linear attenuation to pool surface. 7.5 psi peak-to-peak amplitude. 2-7 Hz frequencies.	January, 1977 Application memorandum	Acceptable	III.B.4.b.3
c) Chugging Loads	Representative pressure fluctuation taken from 4T test added to local hydrostatic.	January, 1977 Application memorandum	Acceptable pending resolution of FSI concerns.	III.B.4.b.4
- uniform loading condition	Maximum amplitude uniform below vent exit-linear attenuation to pool surface. +4.8 psi maximum overpressure, -4.0 psi maximum under pressure, 20-30 Hz frequency.	" "	"	
- asymmetric loading condition	Maximum amplitude uniform below vent exit-linear attenuation to pool surface. 20 psi maximum overpressure, -14 psi maximum underpressure, 20-30 Hz frequency, peripheral variation of amplitude follows observed statistical distribution with maximum and minimum diametrically opposed.	" "	" "	

A XII - 25

1028 183

Mark II Pool Dynamic Load Summary Table

Load or Phenomenon	Mark II Owners Group Load Specification	Reference	NRC Review Status	LER Section
II. SRV-Related Hydrodynamic Loads				
A. Pool Temperature Limits for KWU and GE four arm quencher	No temperature limit	DFFR Revision 2	NRC Criteria II.1 and II.3	III.C.1
B. Quencher Air Clearing Loads	Mark II plants utilizing the KWU quencher use an interim load specification consisting of the ramshead calculational procedure. Mark II plants utilizing the four arm quencher use quencher load methodology described in DFFR.	DFFR Revision 2	NRC Criteria II.2	III.C.2.b III.C.2.c
C. Quencher Tie-Down Loads				
1. Quencher Arm Loads				
(a) Four Arm Quencher	Vertical and lateral arm loads developed on the basis of bounding assumptions for air/water discharge from the quencher and conservative combinations of maximum/minimum bubble pressure acting on the quencher.	DFFR Revision 2	Acceptable	III.C.2.e.1
(b) KWU T Quencher	KWU "T" quencher not included in Mark II O.G. Program. T quencher arm loads not specified at this time.	N/A	Review Continuing	

A-XII-26

Mark II Pool Dynamic Load Summary Table

Load or Phenomenon	Mark II Owners Group Load Specification	Reference	NRC Review Status	LER Section
2. Quencher Tie-Down Loads				
(a) Four-Arm Quencher	Includes vertical and lateral arm load transmitted to the basemat via the tie downs. See II.C.1.a above plus vertical transient wave and thrust loads. Thrust load calculated using a standard momentum balance. Vertical and lateral moments for air or water clearing are calculated based on conservative clearing assumptions.	DFFR Revision 2	Acceptable	III.C.2.e.2
(b) KWU "T" Quencher	KWU "T" quencher not included in Mark II O.G. program. T quencher tie-down loads not specified at this time	N/A	Review Continuing	

A-XII-27

1028 185

Mark II Pool Dynamic Load Summary Table

Load or Phenomenon	Mark II Owners Group Load Specification	Reference	NRC Review Status	LER Section
III. LOCA/SRV Submerged Structure Loads				
A. LOCA/SRV Jet Loads				
1. LOCA/Ramshead SRV Jet Loads	Methodology based on a quasi-one-dimensional model.	NEDE-21730 NEDE-21471	NRC Criteria III.A.1	III.D.1.a III.D.1.b
2. SRV-Quencher Jet Loads	No loads specified for lead plants. Model under development in Long Term Program.	N/A	NRC Criteria III.A.2	III.D.1.c
B. LOCA/SRV Air Bubble Drag Loads				
1. LOCA Air Bubble Loads	The methodology follows the LOCA air carryover phase from bubble charging, bubble contact, pool rise and pool fallback. The drag calculations include standard and acceleration drag components.	DFFR - Revision 2 NEDE-21471 NEDE-21730	NRC Criteria III.B.1.	III.D.2.a
2. SRV-Ramshead Air Bubble Loads	The methodology is based on an analytical model of the bubble charging process including bubble rise and oscillation. Acceleration drag alone is considered.	NEDE-21471	NRC Criteria III.B.2	III.D.2.b
3. SRV-Quencher Air Bubble Loads	No quencher drag model provided for lead plants. Lead plants propose interim use of ramshead model (See III.B.2 above). Model will be developed in long term program.	N/A	NRC Criteria III.B.3.	III.D.2.c
C. Steam Condensation Drag Loads	No generic load methodology provided. Generic model under development in long term program.	N/A	Lead plant load specification and NRC review will be conducted on a plant unique basis with confirmation in long term program using generic model.	III.D.3

POOR ORIGINAL

A-XII-28 7028 186

Mark II Pool Dynamic Load Summary Table

Load or Phenomenon	Mark II Owners Group Load Specification	Reference	NRC Review Status	LER Section
IV. Secondary Loads				
A. Sonic Wave Load	Negligible Load - none specified	DFFR - Revision 2	Acceptable	III.E.1
B. Compressive Wave Load	Negligible Load - none specified	DFFR - Revision 2	Acceptable	III.E.2
C. Post Swell Wave Load	No generic load provided.	N/A	Plant unique load specification and NRC review	III.E.3
D. Seismic Slosh Load	No generic load provided.	N/A	Plant unique load specification and NRC review.	III.E.4
E. Fallback load on Submerged Boundary	Negligible load - none specified	DFFR - Revision 2	Acceptable	III.E.5
F. Thrust Loads	Momentum balance	DFFR - Revision 2	Acceptable	III.E.6
G. Friction Drag Loads on Vents	Standard friction drag calculations	DFFR - Revision 2	Acceptable	III.E.7
H. Vent Clearing Loads	Negligible Load - none specified	DFFR - Revision 2	Acceptable	III.E.8

A·XII - 29

A PRELIMINARY TECHNICAL EVALUATION OF THE GENERAL ELECTRIC MARK II PRESSURE SUPPRESSION CONTAINMENTS

FINAL REPORT

for

Work Performed Under

U.S. Nuclear Regulatory Commission Contract No. 11-78-632

Performed by

Thomas E. Eaton, PE, ScD
Consulting Engineer
680 Berry Lane
Lexington, Kentucky 40502

31 December 1978

Submitted to:

Mr. Marvin C. Gaske
Assistant Executive Director
ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
U.S. Nuclear Regulatory Commission
1717 H Street, N.W.
Washington, D.C. 20555

A-XII-30

1028 133

A PRELIMINARY TECHNICAL EVALUATION OF THE GENERAL ELECTRIC MARK II PRESSURE SUPPRESSION CONTAINMENT SYSTEM

FINAL REPORT

U.S. NRC Contract 11-78-632 .

by

Thomas E. Eaton

Abstract

The Mark II BWR pressure suppression containment designs are being reassessed after important structural loadings were identified which resulted from suppression pool fluid-dynamic phenomenon.

The suppression pool fluid-dynamic problems arise from the complex nature of the multi-component (air/steam/water), two-phase (gas/liquid) flow associated with postulated LOCA's or safety rel' .f valve actuation.

The seven Mark II (domestic) owner utilities and the General Electric Company formulated a three part program to reassess Ma.k II containments, i.e., the definition of dynamic forcing functions, the preparation of plant-unique containment design assessment reports, and the establishment of a Mark II Containment Supporting Program. To date, the Mark II reassessment program has lead to modifications in the lead plants and to a refinement in pool dynamic technology.

Although belated, the evaluation of suppression pool fluid dynamic effects in Mark II plants appears to have been rigorous and thorough. It is my opinion that these effects will be adequately accommodated in the Mark II plants upon obtaining an operating license from the U.S. Nuclear Regulatory Commission.

Introduction

This report is the final report for U.S. NRC Contract 11-78-632 and presents the preliminary findings of a technical evaluation of the General Electric Mark II Pressure Suppression Containment concept and of the NRC's licensing evaluation of the Mark II design.

Contents

This report attempts to identify the important issues pertaining to the licensing evaluation of Mark II containments; to discuss the differences between Mark II and Mark III containments, and between the various Mark II plant designs; to present questions regarding the suppression pool hydrodynamic phenomenon and the Mark II containment design; and to compare the various design parameters for the seven Mark II plants.

The Mark II Pressure Suppression Containment

The Mark II pressure suppression containment was designed and developed by the General Electric Company as part of the 1969 product line which utilized the BWR/5 Nuclear Steam Supply System, see Figures 1 and 2, as well as Table 1.

The Mark II containment utilizes an over/under design arrangement, i.e., a conical (inverted) shaped drywell over a cylindrical wetwell, see Figure 1. Importantly, the two primary containment building chambers, i.e., the drywell and the wetwell, are separated by a diaphram (the drywell floor) which supports long vent pipes (24 inch diameter, nominal) which extend from the drywell floor down into the suppression pool. The wetwell is partially filled with water which is the energy absorbing media for the pressure suppression containment system.

The suppression pool is used to absorb the energy from and thereby to condense any steam released from the NSSS. Steam may be released accidentally during a hypothetical Loss of Coolant Accident (LOCA) or as the result of actuation of the Safety Relief Valves on the reactor primary system.

Mark II Containment Advantages and Disadvantages

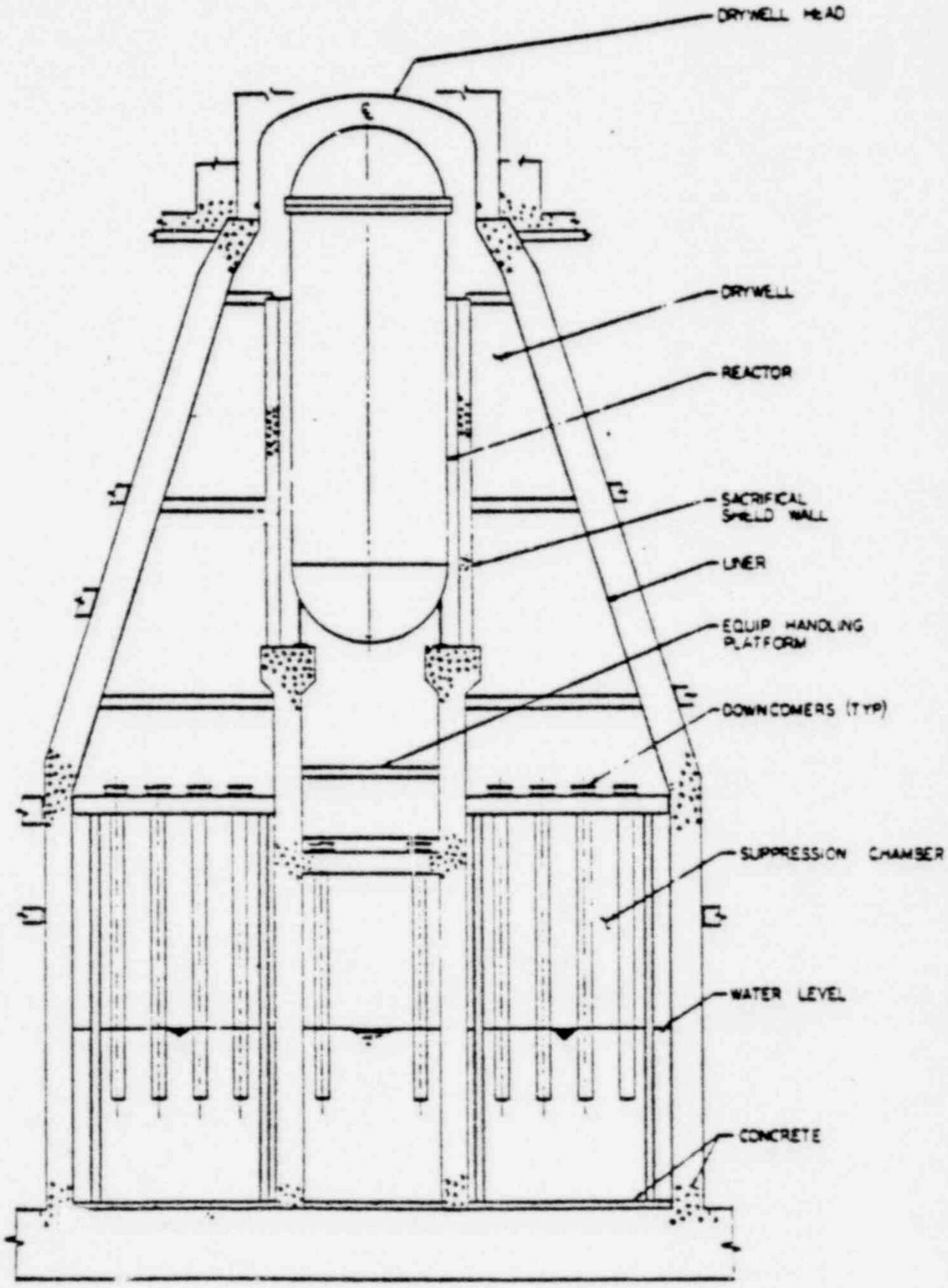
The principal advantage of the Mark II containment compared to a typical dry containment is reduction of containment pressures which occur during and after a hypothetical LOCA. The pressure suppression pool also is a convenient receiver for condensing any primary steam released by the reactor vessel's safety relief valves.

FIGURE 1

A TYPICAL MARK II PRESSURE SUPPRESSION CONTAINMENT

PRIMARY CONTAINMENT

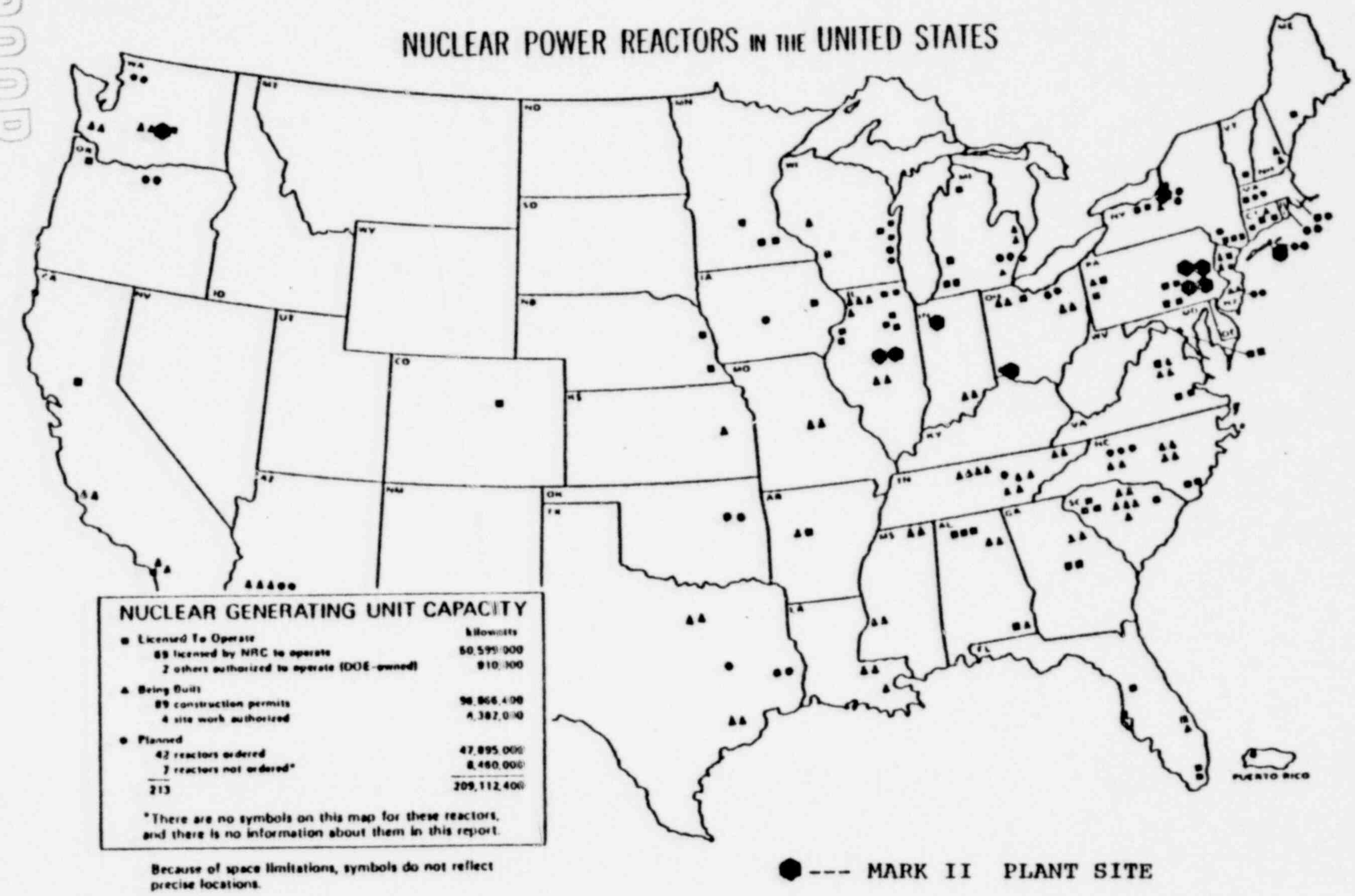
NINE MILE POINT NUCLEAR STATION-UNIT 2
NIAGARA MOHAWK POWER CORPORATION
PRELIMINARY SAFETY ANALYSIS REPORT



PRIMARY CONTAINMENT

A-XII-34

FIGURE 2 - LOCATION OF DOMESTIC MARK II PLANTS



1028 192

TABLE 1

THE MARK II PLANTS

Unit Name	Owner	Capacity		Start-up
BAILLY, Unit No. 1	Northern Indiana Public Service Co.	645 MWe	1931 MWth	Indef.
LASALLE, Unit Nos. 1*& 2	Commonwealth Edison Co.	1078 MWe	3293 MWth	1979/80
LIMERICK, Unit Nos. 1 & 2	Philadelphia Electric Co.	1065 MWe	3293 MWth	1983/85
NINE MILE POINT, Unit No. 2	Niagara Mohawk Power Corp.	1100 MWe	3323 MWth	1983
SHOREHAM*	Long Island Lighting Co.	819 MWe	2436 MWth	1980
SUSQUEHANNA, Unit Nos. 1 & 2	Pennsylvania Power & Light Co.	1050 MWe	3293 MWth	1980/81
WPPSS, Unit No. 2	Washington Public Power Supply System	1100 MWe	3323 MWth	1980
ZIMMER*	Cincinnati Gas & Electric Co.	810 MWe	2436 MWch	1979

---

* Lead Plants (LPP)

A XII-35

1028 193

The condensation of steam during its release allows the containment size to be smaller than that of a dry containment. Typically, Mark II containments are designed for 45 psig internal pressure and less than 5 psi negative pressure.

The principal disadvantages of the Mark II containment arise from the smaller size as compared to dry containments and from the complex two component, two phase flow (air/steam/water) which arises during the steam pressure suppression process.

The small primary containment volume interferes with the assembly of the plant, in-service inspection and surveylance, and plant maintenance. It also complicates hydrogen gas concentration control, etc.

In order to function properly, there must be no steam by-passing the suppression pool, i.e., no direct flow path for steam to enter the wetwell air space without condensing. Also, a mechanism must be provided to assure that noncondensible gases can return to the drywell after an accident without returning through the downcomer vent pipes.

Mark II Reassessment Program

In early 1975, as the result of suppression pool fluid-dynamic data generated from the Mark III containment evaluation program, it was determined by the U.S. NRC Staff that the Mark II plants should be reassessed. The details of the reassessment program are given in Table 2.

Variations in Mark II Containment Designs

The seven different Mark II plants (with eleven different reactors) are each essentially unique. Although each plant represents the same class of containment design, i.e, Mark II, the differences between the plants are significant with regard to the licensing evaluation of the plants. Since the details of the containment design influence the assessment of various pool fluid dynamic-related forces, each plant must have a unique design evaluation.

Table 3 lists some of the design variations between the various Mark II plants. Among the important differences are the design of the following: Diaphram seal, Reactor pedistal, Wetwell wall, Vacuum breakers, Diaphram support, Downcomer vent support, Suppression pool hardware, SRV quencher device, and SRV quencher support.

Table 2

THE MARK II CONTAINMENT REASSESSMENT PROGRAM [1,2]

In April 1975, the U.S. NRC Staff determines a need for a complete reassessment of the Mark II facilities

In early 1975, the Mark II Owners Group is established to respond to NRC inquires

A Lead Plant Program (LPP) was initiated which provided for conservative load definitions for those plants (Zimmer, Shoreham, LaSalle) which would be completed before the Mark II containment supporting program could be completed

A Long Term Program (LTP) was initiated to justify the reduction of certain pool dynamic loads used in the LPP

The Mark II Owners Group and the General Electric COmpany establish a three element Mark II containment reassessment program to determine the additional information required to reassess the Mark II containmnets [2]. The three elements are

1. The Dynamic Forcing Funcitons Information Report (DFFR), NEDE-21061-P and its revisions and addenda: The DFFR "provides a suitable methodology" for conservatively estimating the suppression pool hydrodynamic loadings on Mark II containments

2. A Plant-Unique Design Assessment Report (DAR) was prepared by each plant and issued by early 1976. Up-dates and revisions were or will be issued as required. The DAR's used the design basis methodology presented in the DFFR

3. A "Mark II Containment Supporting Program" was implimented to confirm the adequacy of the DFFR as a design basis methodology [2]

Certain Mark II plant modifications have been identified and are being made.

NUREG-0414: "Mark II Containment Pool Dynamic Loads" forms the basis of the NRC Staff's evaluation of the LPP and LTP

A Mark II Generic Safety Evaluation Report is planned for mid-1980

TABLE 3

MARK II PLANT DESIGN VARIATIONS

A. DIAPHRAM-TO-WALL SEAL DESIGN

Monolithic
Inflated Seal
Welded with Expansion Joint

B. REACTOR PEDISTAL

Solid Concrete
Water Filled

C. REACTOR POWER / PHYSICAL SIZE

D. WETWELL WALL DESIGN

Rigid
Flexible

E. VACUUM BREAKER DESIGN

In Downcomer Pipe
In Drywell Floor (Diaphram)

F. DIAPHRAM (DRYWELL FLOOR) SUPPORT

Reactor Pedistal and Columns
Reactor Pedistal, Columns, and Primary Vessel Walls

G. DOWNCOMER VENT SUPPORT

Cantilevered
Bottom Bracing Structure Network

H. SECONDARY CONTAINMENT BUILDING DESIGN

I. SUPPRESSION POOL HARDWARE ARRANGEMENT

Piping, Walkways, Intake Screens, etc.

J. SUPPRESSION POOL HARDWARE DESIGN & SUPPORT

K. SRV QUENCHER DEVICE DESIGN AND SUPPORT

L. DOWNCOMER SUBMERGENCE DEPTH (8-1/2 to 13 FT)

M. NUMBER OF SRV DISCHARGE LINES

Table 4 provides a numerical comparison of the various Mark II plant design parameters.

Mark II and Mark III Containment Designs

A comparison of the design features of Mark II and Mark III containments is given in Table 5. This table attempts to identify the differences and similarities between these different pressure suppression containment design concepts.

Briefly, the Mark III containment, see Figure 3, uses a containment building design similar to that of a conventional dry containment with the drywell volume enclosed within the wetwell. The suppression pool is an annulus around the base of the secondary containment. The containment building has drywell and wetwell volumes which are separated by reinforced concrete walls.

Air and/or steam are vented from the drywell into the wetwell through short horizontal penetrations in the drywell walls and below (at varying submergences) the suppression pool level. The design of the Mark III suppression pool leads to considerably modified dynamic behavior as compared to the Mark II design. The Mark III design has a significantly larger wetwell air space than the Mark II design so that many of the Mark II design's disadvantages are not as severe in the Mark III plants.

Questions Regarding the Mark II Plants

During the course of this contract work, various questions have arisen regarding the Mark II plant and its suppression pool fluid dynamic phenomenon.

A listing of some of the general questions on the Mark II containment which should be considered in a review of the Mark II licensing evaluation are listed in Table 6.

A listing of questions concerning various specific aspects of the Mark II containment design are given in Table 7.

A question regarding dissolved gas release during suppression pool heatup is addressed in the next section.

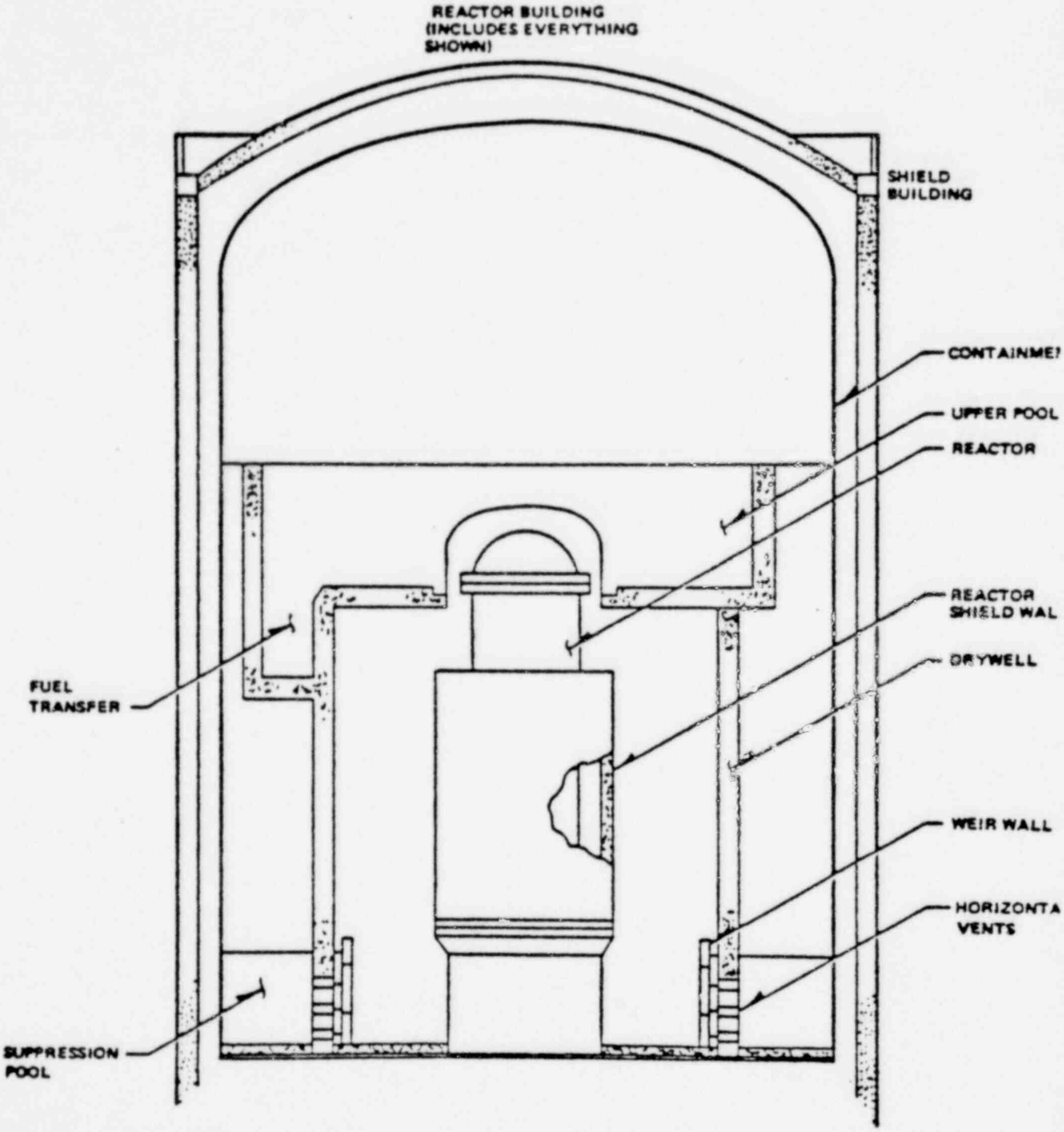


FIGURE 3

MARK III REACTOR BUILDING DESIGN

A XII - 40

TABLE 5

A COMPARISON OF MARK II AND MARK III

PRESSURE SUPPRESSION CONTAINMENTS

MARK II	MARK III
Uniform downcomer submergence	Vents at three different submergences
Vertical downcomer vent pipes-cantilevered from diaphram or braced	Horizontal vent pipes in drywell wall
Long vertical vent pipes	Short horizontal vent pipes
Suppression pool is a large pool below drywell	Suppression pool is an annular pool around drywell
Small wetwell air space	Large wetwell air space
Unique Over/Under design	Design similar to dry containment
Suppression pool bounded by reactor building walls	Suppression pool requires weir wall
Tall reactor pedistal	Short reactor pedistal
Drywell is part of primary containment	Drywell is enclosed inside primary containment
Jet deflectors required over downcomer vent entrances	No deflectors required
Vent submergence 10-12 ft (constant)	Variable vent submergence-5, 10, 15 ft, nominal
Diaphram (Drywell/wetwell separator) supported from reactor pedistal, columns and (in some cases) the containment walls	Drywell walls supported in part from containment walls
Pool swell limited by wetwell air compression	Wetwell air compression is insignificant
Wetwell air compression reverse loads diaphram	"

TABLE 5
(Concluded)

MARK II	MARK III
Design pressure - 45 psig	Design pressure - 15 psig
Wetwell free volume - 160,000 cu. ft., nominal	Wetwell free volume - 1,100,000 cu. ft., nominal
Pool depth - 25 ft., nominal	Pool depth - 20 ft., nominal
Drywell free volume - 225,000 cu. ft., nominal	Drywell free volume - 274,000 cu. ft., nominal
Suppression pool water vol. - 120,000 cu. ft., nominal	Suppression pool water vol. - 116,000 cu. ft., nominal
More sensitive to hydrogen concentration limits	Larger volume for hydrogen dilution
Drywell liner mandatory	Drywell liner not required
Sensitive to steam by-pass around diaphram	Not as sensitive to steam by-pass
Poor accessibility for construction and inspection	Better accessibility for construction and inspection
High reactor building	Lower reactor building (50 ft.)
Gratings, columns, downcomer vents, SRV lines and quenchers, reactor pedistal all subjected to suppression pool dynamic loads	Less submerged structures

TABLE 6

GENERAL QUESTIONS PERTAINING TO THE

REVIEW OF THE LICENSING EVALUATION OF

MARK II CONTAINMENTS

1. Have all of the suppression pool fluid-dynamic phenomenon and their related forces been identified?

2. Are the magnitudes of the pool dynamic forcing functions adequate to conservatively evaluate their effects on Mark II containment structures?

3. How should the various structural loads in Mark II plants be combined in order to establish the design of the containment building?

4. Have the dynamic interactions between the containment structures and the suppression pool been properly evaluated over the entire history of the accident events of interest?

5. Have the problems associated with the vibrations and acoustics during a LOCA or SRV actuation been adequately determined?

6. Have the consequences of the hostile primary containment environment in Mark II plants following accidental steam releases been adequately assessed?

7. What pool dynamic phenomenon can be predicted from first principles?

8. Have scaling effects in the experimental programs been properly identified and incorporated into the development of analytical models for full size plants?

9. Have all of the safety problems which arise because of the small size of the pressure suppression containments been identified?

10. Should upper and/or lower suppression pool temperature limits be established?

11. Might new suppression pool dynamic effects be identified if the full spectrum of potential break sizes were considered?

TABLE 7

QUESTIONS PERTAINING TO

MARK-II PRESSURE SUPPRESSION CONTAINMENTS

A. Suppression Pool Dynamic Loads

Is the chugging load on the downcomer vent conservatively defined?

Is the effect of non-condensible bubble dynamics included in the hydrostatic pressure resistance to downcomer flow? Could it provide a mechanism for downcomer flow variations?

Does two-phase choked flow ever occur in the downcomer vents?

Does liquid water released into the drywell affect the downcomer vent flow?

Do large bubbles breaking through the liquid surface produce noteworthy pool dynamic loads?

How do non-condensible bubbles rising in the suppression pool influence the various dynamic loads? Do such bubbles have a dampening effect?

Would the submergence of permanent passive energy-absorbing materials in the suppression pool serve to reduce the significance of pool dynamic loads?

B. Pool Swell Phenomenon

Is noncondensible gas forced back into the downcomer vent during pool fallback? If so, could multiple pool swell events occur during a given accident?

Is it possible that vapor composition variations at the vent inlets in the drywell could lead to a nonuniform pool swell height? Could this change the nature of the breakthrough and pool fallback phenomenon?

At the end of the pool swell event, gas breakthrough produces intimate vapor/liquid contact. If the gas contained condensible vapor would this influence the pool fallback behavior?

TABLE 7
(Continued)

C. Fluid-Structure Interactions

Fluid-structures interactions (FSI) effects have been assessed independent of other loads occurring simultaneously; would FSI results be changed if a more detailed structural analysis were performed?

Since Fluid-structure interaction effects are plant-unique, have they been adequately assessed in each plant?

Have FSI effects incorporated in data from the various experimental facilities been properly considered in model development work?

D. Safety Relief Valves

Do the Safety Relief Valve (SRV) discharge lines have multiple vacuum breaker valves? If not, what are the consequences of a failure of a SRV vacuum breaker? Of particular interest are the water hammer and potential boiling phenomenon (due to cold water contact with hot pipes) during an accidental SRV line reflood?

Should SRV quencher devices be supported from the wetwell floor?

Do any Mark II plants intend to use SRV ramshead devices? Should Mark II plants be licensed if they use SRV ramsheads instead of quencher devices?

E. Downcomer Vents

Can physical damage to the downcomer vent pipes occur as the result of pool dynamic loads?

What transverse support (bracing), if any, should be required on the downcomer vents?

TABLE 7
(Concluded)

F. Small Pipe Break Accidents

What is the drywell temperature history and the wetwell air space temperature history during a Small Break Accident (SBA)? Include the effects of thermal stratification, drywell condensation and downcomer heat transfer in the analysis.

Is it possible to identify a pipe break size below which the pressure suppression function is ineffective?

Does non-condensible transport to the gas-liquid interface significantly interfere with the pressure suppression process if the differential pressure is insufficient to force vapor into the wetwell air space?

Has the effect of the hostile environment-resulting from the various accidents postulated - on equipment, instruments, and components inside the containment been thoroughly evaluated?

ADDENDUM

Can the suppression pool dynamic loads be enhanced by either sudden condensation from ECCS Cooling Water spillage or from vapor generated by liquid water impingement on hot primary system components?

What would be the consequences of a SRV line break inside the we'well air space? If sufficient steam were admitted to displace the noncondensible gases through the pressure relief vents, would activation of the wetwell spray system produce as yet unidentified pool dynamic loads, i.e., pool swell impact on the drywell floor?

What should be the specific volume exponent in the wetwell air compression pressure / volume relationship?

Suppression Pool Dissolved Gas Release

The water in the Mark II suppression pool contains a certain amount of dissolved gases at temperatures below saturation. As water is warmed, its ability to dissolve gases decreases. At the boiling point, it no longer can dissolve gases.

The gases dissolved in the suppression pool of Mark II containments will contribute to the noncondensible gas inventory as it is released during pool heat-up.

Using data on the soluability of gases in water, it was estimated that the maximum potential air release would be about two per cent (2%) of the water volume if the gas were released after being dissolved at atmospheric pressure. For realistic conditions in a Mark II containment, it was found that about 0.5 - 1.0% of the volume of the water would be released as gas at STP.

Should, for any reason, the partial vapor pressure of carbon dioxide in the wetwell air space increase substantially above that of normal air, the suppression pool water's gas release would increase markedly because of the high soluability of CO_2 in water.

Future Work

For this report, time does not permit the completion of various tables associated with the design evaluation of the Mark II plants; these tables could be completed in a short time, however.

With this, the preliminary evaluation of the Mark II containment should turn to a review of the Mark III containment evaluation program. This Mark III program review will be required in order to respond to questions recieved from ACRS Member Myer Bender in a letter dated 28 September 1978, see Attachment A.

Upon completion of the Mark II / Mark III containment licensing evaluations, additional guidance and instruction concerning the project's goals will be required in order to pursue the technical evaluation of the Mark II containment systems.

A partial listing of the things which might be done to continue the evaluation are listed below

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

September 28, 1978

Dr. Thomas Eaton
Assistant Professor
Department of Mechanical Engineering
College of Engineering
University of Kentucky
Lexington, Kentucky 40506

Dear Tom:

I am enclosing a copy of the minutes of a meeting between the Mark II Containment Owner's Group and NRC personnel which provides a fragmented picture of the status of the licensing evaluation for this containment concept. The Zimmer Nuclear Power Plant will be the first installation requesting an operating license to use the Mark II containment arrangement. Consequently, it may be the area where your initial efforts could be of most use to the ACRS. We need to know a number of things.

1. How does the Mark II containment evaluation approach compare to the Mark III approach which has been subjected to more detailed scrutiny than the Mark II installation? Most of the safety issues were initially identified during the Mark III review.

2. How well do the experimental programs to verify Mark II containment capability correlate with the Mark III programs, especially configuration simulation of the sort being conducted by G.E. at San Jose?

3. Is there a direct basis for comparison between the fluid dynamic and structural response considerations for Mark I, Mark II, and Mark III containments?

4. Since concern has sometimes been expressed about the effects of suppression pool temperature on the suppression pool response to syst m blowdown, are these factors properly identified and analyz for the Mark II containment?

5. Are ther any matters affecting suppression pool bypass for small LOCAs that deserve special attention in the Mark II design?

The above are probably the matters deserving most immediate review attention for the Zimmer Plant and yo assistance in bringing the matter into focus would be of considerable value.

A XII - 48

Dr. Thomas Eaton 2 September 28, 1978

In the longer term, there may be other questions more subtle in nature but possibly of more significance to public safety that are worth investigation. For example, the concerns about suppression pool bypassing generally relate to containment overpressurization as a result of a small LOCA, but the public consequences of containment overpressurization in a small LOCA may not be meaningful if the radioactivity release from the reactor system is small. It may be worthwhile to determine whether the consequence boundary used as a basis for determining performance is appropriate. A second point of some interest may be the size of the blowdown and the effect of rate of fluid release on pool swell and containment structural response. Since many of the discussions are premised on limiting cases, there may be some value in determining whether probabalistic considerations have been taken into account properly when concern is expressed for these phenomenological effects.

Hopefully, during the period of time while arrangements are being made for : . to accept the fellowship assignment to the ACRS in Washington, you will be able to make some headway in assisting us through this type of study effort. I have not yet had an opportunity to discuss the above matters with other members of the ACRS and there may be other interests or other approaches that might be considered. Please call me if you wish to expand on your understanding of the subject matter.

Sincerely,

M. Bender

M. Bender

MB/mh

Enclosures

cc: R. F. Fraley - ACRS Distribution

A. Provide a detailed report on the differences between the various Mark II containments and the significance of these differences regarding the safe operation of the plants.

B. Undertake a detailed comparison of the Mark II and Mark III containment design assessment programs and NRC licensing evaluations.

C. Establish a technical basis for comparison of the various Mark II plants and for comparison of the Mark II and Mark III plants.

D. Evaluate the structural analysis of the diaphram and the wetwell chamber as performed for combined loadings which include pool dynamic forces.

References

[1] NUREG-0474: "A Technical Update on Pressure Suppression Type Containments in Use in U.S. LIght Water Reactor Nuclear Power Plants," Division of Systems Safety, Division of Operating Reactors, Office of Nuclear Reactor Regualtion, U.S. Nuclear Regulatory Commission, July 1978.

[2] NEDO-21297: "Mark II Containment Supporting Program Report, "Nuclear Energy Projects Division, General Electric Co., Revision 1, March 1978.

Los Pinos, the Mexican White House. In the afternoon a special outing was arranged with Carter going by helicopter to the village of Ixtlilco ... where he will mingle with farmers, lunch on some local specialties and stroll

a surprising mixture of interest, disdain and fear, much like the ... vague fears you yourselves inspire in certain areas of our national subconscious," he said.

ramp. Nor was there the traditional "abrazo" — the Latin American big bear ... Americans have grown accustomed to seeing on such occasions.

blown ... the em... sacre" ... al decis...

Cincinnati Post - 2/15/79 - Front Page

Zimmer opening may be delayed

By Douglas Starr
Post staff reporter

Safety problems at the Zimmer nuclear station may delay the plant's start-up date, according to federal inspectors.

Correcting the problems will involve reviewing specifications and perhaps strengthening some of the thousands of supports that hold miles of pipe throughout the Moscow, Ohio plant.

Some of these supports may not be strong enough, federal officials say. Inspectors note that in addition to finding some support components too weak, they found that the calculations and inspections required to check the strength of the supports were not completed.

SOME OF THE PIPES involved carry water essential to the safe shutdown of the reactor. But federal inspectors said if the plant opened with these safety problems, and an accident occurred, the reactor probably could be safely shut down.

Broken pipes, however, whipping round after an accident, could damage important equipment at the plant, federal sources said.

In rare cases, pipes carrying mildly radioactive water could conceivably leak, inspectors said, but probably not beyond the confines of the plant.

"This job is very screwed up," said Isa T. Yin, an inspector for the Nuclear Regulatory Commission's Chicago office. "This is a common problem, but Zimmer seems to be the worst case."

OFFICIALS of the Cincinnati Gas & Electric Co., the majority owner of the plant, said the problems are being corrected and said they do not expect major delays.

Inspector Yin visited the plant four times since August 1978, at which time he issued six citations connected with construction and inspection practices.

In his citations, Yin noted that the utility does not know how much strain the pipe supports will take.

- In many cases, he said, workers did not install supports according to the designers' drawings. That is permitted—but any changes should first have been cleared with the design firm.

Yin later found some of these supports to be improperly anchored to the concrete walls.

- In other cases, the design firm —Sargent & Lundy of Chicago— never calculated how much strain the supports would take.

- Yin further found that one of the inspection programs at Zimmer —through which the problems should have been spotted—was inadequate.

- Finally, Yin noted that about 50 of the plant's more than 1000 "snubbers"—anti-shock devices that cushion the pipes from earthquakes—were of a make he considered unsound. Utility officials dispute this opinion.

THESE PROBLEMS earned CG&E six federal "infractions." Under law, an infraction is the second most serious of three possible violations involving nuclear construction. The most serious, a "violation," involves the actual spillage of radioactiove material.

Earl Borgman, vice president of engineering and electric generation at CG&E, said the problems are well on the way to being solved. He said many supports have been strengthened, and others have been added.

The utility checked about 2000 anchor expanders—part of the pipe-to-wall supports—and replaced 75, Borgman said. The plant's designers have reviewed about half the pipe-to-ceiling hangers, he said, and promised to complete the job by April.

HE ADDED that CG&E has ordered 50 new anti-shock snubbers, but would wait to test the old ones before installing the new.

Borgman said he expects only a slight delay, if any, in the $661 million station's June 1979 fuel loading date. (The station is expected to produce electricity sometime in 1979.) He said the corrections will cost "a lot—that's all I can say."

But commission sources said they expect a delay closer to six months.

Cooli... clear...

POOR ORIGINAL

A XII-5'

1028 209

Inside

focus

Family plight: living in the sha...

By Ginny Hunter
Post staff reporter

Her son lies on a hospital bed, immobilized.

She stays with him in his private room, watching as the attendants turn him on his stomach, then his back, every four hours, day and night.

Months of therapy, and pain, have been living in the shadow of cancer.

"People who give in don't survive," she said. "We take it as it comes."

It first came in 1967. Her youngest son, Jeff, died of leukemia, which is cancer of the bone marrow. The day before Jeff died, doctors found a lump on the covering

Q PDR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

OCT 3 1978

Docket No. 50-358

Cincinnati Gas and Electric
Company
ATTN: Mr. Earl A. Borgmann
Vice President Engineering
139 East 4th Street
Cincinnati, OH 45201

Gentlemen:

This refers to the investigation conducted by Messrs. J. E. Foster, I. T. Yin, and E. J. Gallagher of this office on August 9-11, 15 and 16, 1978, of activities at the Zimmer Unit 1 construction site, authorized by NRC Construction Permit No. CPPR-88, and to the discussion of our findings with Messrs. B. K. Culver and W. W. Schwiers and others of your staff at the conclusion of the investigation.

This investigation concerned allegations of improper design and installation of pipe hangers, restraints, and snubbers at the Zimmer Unit 1 site. The enclosed copy of our investigation report identifies those areas examined during the investigation. Within these areas, the investigation consisted of a selective examination of procedures and representative records, observations, and interviews with personnel.

During this investigation, certain of your activities appeared to be in noncompliance with NRC requirements, as described in the enclosed Appendix A.

This notice is sent to you pursuant to the provisions of Section 2.201 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations. Section 2.201 requires you to submit to this office within thirty days of your receipt of this notice a written statement or explanation in reply, including for each item of noncompliance: (1) corrective action taken and the results achieved; (2) corrective action to be taken to avoid further noncompliance; and (3) the date when full compliance will be achieved. Item 4 in Appendix A is a recurrent item. Therefore, in your response please give this matter your particular attention.

As a result of our investigation, you issued Stop Work Orders relative to installation of mechanical snubbers, hydraulic snubbers and concrete expansion anchors. We understand that you will not release these Stop Work Orders until you have determined that a

A-XII-52

7811170260

Q

1028 210

quality program has been implemented to control these activities and we are informed of these actions. We are aware that your Stop Work Order related to the installation of concrete expansion anchors was released on September 6, 1978. We will review these matters during subsequent inspections.

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and the enclosed inspection report will be placed in the NRC's Public Document Room, except as follows. If this report contains information that you or your contractors believe to be proprietary, you must apply in writing to this office, within twenty days of your receipt of this letter, to withhold such information from public disclosure. The application must include a full statement of the reasons for which the information is considered proprietary, and should be prepared so that proprietary information identified in the application is contained in an enclosure to the application.

We will gladly discuss any questions you have concerning this investigation.

Sincerely,

for Glen W. Roy
James G. Keppler
Director

Enclosures:
1. Appendix A, Notice of Violation
2. IE Investigation Rpt No. 50-358/78-18

cc w/encls:
Mr. J. R. Schott, Plant Superintendent
Central Files
Reproduction Unit NRC 20b
PDR
Local PDR
NSIC
TIC
U. Young Park, Power Siting Commission

A-XII-53

Appendix A

NOTICE OF VIOLATION

Cincinnati Gas and Electric Company

Docket No. 50-358

Based on the results of an NRC investigation conducted on August 9-11 and 15-16, 1978, it appears that certain of your activities were not conducted in full compliance with NRC requirements as noted below. These items are infractions.

1. 10 CFR 50, Appendix B, Criterion V requires, in part, that activities affecting quality shall be accomplished in accordance with documented instructions, procedures and drawings. Paragraph 17.1.5 of the PSAR states, in part, "Activities affecting the quality of the facility are accomplished in accordance with written instructions, procedures, or drawings which prescribe acceptable methods for carrying out the activities, make reference to appropriate inspections and tests, and include acceptance criteria . . ."

   Contrary to the above, numerous (at least 3) anchor bolts for safety-related supports and restraints were installed in a manner contrary to the instructions detailed on construction drawings.

2. 10 CFR 50, Appendix B, Criterion IV, requires, in part, that measures shall be established to assure that requirements which are necessary to assure adequate quality are suitably referenced or included in the documents for procurement of material. Paragraph 17.1.4 of the PSAR states, "QAS determines that measures have been established to ensure applicable . . . design basis and other requirements to ensure adequate quality are suitably included or referenced in the documents for procurement of essential material, equipment and services . . ."

   Contrary to the above, the procurement documents for safety-related concrete expansion anchor bolts used for anchorage of supports and restraints do not include or reference requirements necessary to ensure adequate quality nor do the procurement documents require the suppliers to have a quality assurance program.

3. 10 CFR 50, Appendix B, Criterion X requires, in part, that a program for inspection activities affecting quality shall be

7811170262

A-XII-54

1028 212

established and executed to verify conformance with the drawings. Paragraph 17.1.10 of the PSAR states, in part, "Inspections and tests are performed in accordance with written procedures which include requirements for check lists and other appropriate documentation of the inspections and tests performed."

Contrary to the above,

a. An inspection program has not been documented or executed to verify that the concrete expansion anchor bolts have been installed adequately, such as verification of embedment depth, torque installation requirements, bolt spacing or minimum edge distance requirements.

b. The hangers, snubbers, and restraints inspection program had not been executed to verify that the installation of these items were in conformance with the design drawings.

4. 10 CFR 50, Appendix B, Criterion III requires, in part, that measures shall be established to assure that the design basis for structures, systems and components are correctly translated into specifications, drawings, procedures and instructions. Paragraph 17.1.3 of the PSAR states, in part, "Materials will be selected which correctly meet the requirements for the design and intended application of safety-related components," and that "Parts for equipment and components will be selected which correctly meet the requirements of the design, and their intended applications are reviewed as set forth . . . to ensure their correct application and workability for their intended function is a reliable and safe manner."

Contrary to the above,

a. Design drawings Nos. M-126-7H-15, M-126-10H-58 and M-126-10H-57 do not include the required detail information to assure adequate installation of safety-related supports, in that, the design drawings do not indicate the bolt length nor embedment depth of the concrete expansion anchor bolts transmitting the load to the concrete structure.

b. The design drawing Nos. M-448-6H-41 and M-488-8H-20 indicate that safety-related supports and restraints are to be anchored to nonsafety related masonry block walls using concrete expansion anchors.

c. Installation of hangers and snubbers were not in accordance with design drawings. The installation was completed prior to design change review and approval.

The use of drawings with incomplete design basis was cited once previously as an item of noncompliance in IE Inspection No. 50-358/78-09).

5. 10 CFR 50, Appendix B, Criterion VI requires, in part, that measures shall be established to control the issuance of documents, such as instructions. Paragraph 17.1.6 of the PSAR states, in part, " . . . changes to . . documents are reviewed for adequacy and are distributed in a manner similar to the original document."

Contrary to the above, the use of the Inter-Office Memorandum (IOM) to issue QA manual procedure change was not considered appropriate in that the content of the IOM did not receive engineering and QA review, and that the IOM is not distributed and updated in accordance with the QA manual procedure.

6. 10 CFR 50, Appendix B, Criterion XIII requires, in part, that measures shall be established to control the preservation of equipment in accordance with instructions to prevent damage. Paragraph 17.1.13 of PSAR states, in part, " . . . equipment manufacturers' instructions prescribe controls for the onsite handling, . . and preservation of material and equipment in accordance with work and inspection instructions as necessary to prevent damage or deterioration."

Contrary to the above, several installed Bergen-Paterson hydraulic snubbers were observed without accumulator indicator protective covers as required by the manufacturer.

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

REGION III

Report No. 50-358/78-18

Docket No. 50-358 License No. CPPR-88

Licensee: Cincinnati Gas and Electric Company
139 East 4th Street
Cincinnati, OH 45201

Facility Name: Zimmer, Unit 1

Investigation At: Zimmer 1 Site, Moscow, OH

Investigation Conducted: August 9-11, 15 and 16, 1978

Investigator: C. E. Norelius for E. Foster 9/28/78

Inspectors: D. H. Danielson for I. T. Yin 9/28/78

R. L. Spessard for E. J. Gallagher 9/28/78

Reviewed By: C. E. Norelius
Charles E. Norelius
Assistant to the Director 9/28/78

D. H. Danielson for
Duane H. Danielson, Chief
Engineering Support Section 2 9/28/78

R. L. Spessard
Richard L. Spessard, Chief
Engineering Support Section 1 9/28/78

Investigation Summary

Investigation on August 9-11, 15 and 16, 1978 (Report No. 50-358/78-18)
Areas Inspected: Special, unannounced inspection of procurement, design control, inspection program, and installation procedures for pipe hangers, restraints, and snubbers; review of pertinent records, inspection of

7811170264
A-XII-57

1028 215

installed components and interviews with personnel. The investigation involved 84 inspector-hours onsite by three NRC inspectors.

Results: Six items of noncompliance (all infractions) were identified in the following areas; (Installation of concrete expansion bolts for hangers and snubbers not in accordance with drawings - Section III, Paragraph 2; Inadequate procurement documents for the concrete expansion bolts - Section III, Paragraph 3; Inadequate QC inspections for installation of hangers, snubbers and concrete expansion bolts - Section II, Paragraph 1 and Section III, Paragraph 2; Inadequate design control and review for hangers, snubbers, and concrete expansion anchor bolts - Section II, Paragraph 2 and Section III, Paragraph 1; Inadequate control of issuance of procedure changes - Section II, Paragraph 3; Inadequate measures to protect hydraulic snubber components from damage - Section II, Paragraph 4).

A-VII-58

INTRODUCTION

The Zimmer Unit 1 nuclear power plant, licensed to the Cincinnati Gas and Electric Company, is under construction near Moscow, Ohio. Sargent and Lundy is the Architect-Engineering firm for the plant, which is being constructed by Kaiser Engineering, Inc.. The facility will utilize a Boiling Water Reactor (BWR) designed by General Electric Company.

REASON FOR INVESTIGATION

On June 27, 1978, representatives of NRC contacted Individual "A" by telephone, and discussed concerns which he had relative to the design and installation of pipe hangers, restraints, and snubbers at the Zimmer 1 plant. During the conversation, Individual "A" alleged various problems related to design and installation of this equipment. An investigation was initiated into these allegations.

SUMMARY OF FACTS

On June 27, 1978, an individual contacted the NRC Division of Nuclear Reactor Regulation in Bethesda, MD, and indicated that Individual "A" had concerns related to design and installation of piping equipment at the Zimmer site. NRC personnel contacted Individual "A" on that date, discussed his concerns, and received several allegations of improper construction. Information from this discussion was transmitted to NRC Region III (RIII) for action.

On June 29, 1978, RIII personnel contacted Individual "A", an employee at the Zimmer 1 site, and discussed concerns relative to pipe hanger, restraint, and snubber design and installation. General information as to the various concerns was obtained, and Individual "A" was requested to note specific locations of nonconforming equipment.

RIII representatives interviewed Individual "A" on July 11, 1978. Discussion indicated that his concerns related to what he felt was inadequate design, and improper installation of pipe hangers, restraints, and snubbers. Design concerns expressed included: insufficient thickness of base plates for pipe supports insufficient bolt diameter, overall hanger geometry, and lack of consideration given to design loading conditions. Installation problems discussed included improper embedment length of expansive concrete anchors, installation of pipe supports attached to block walls, and lack of lock nuts where required.

Individual "A" provided the RIII representatives with a number of locations where allegedly deficient pipe hangers and snubbers could be located. He indicated that this list was a selection of observed problem locations, not a full list of all locations. Individual "A" stated that limiting of NRC inspection to these areas would not be necessary, as a general inspection of pipe hangers and snubbers would identify similar defects.

Discussion with personnel of the NRC Office of Standards Development indicated that possible deficiencies related to design to pipe support base plates had been identified as a generic problem affecting all reactor sites, and was under review. As such, the concern related to base plate design will not be treated in this report.

A Zimmer site inspection performed during August 9-11, 15 and 16, 1978, reviewed the allegations. During the inspection, a selection of the locations provided by Individual "A", and several additional locations selected at random were inspected.

The inspection revealed that significant deficiencies did exist relative to design control and review, installation, and inspection of pipe hangers, restraints, and snubbers, as alleged. Four items of noncompliance with NRC requirements were identified in areas where deficiencies were alleged, and two items were identified during inspection of related areas.

The licensee was advised of the findings of the investigation on August 11, 1978, and further discussions were held during August 15 and 16, 1978. On August 15, 1978, the licensee advised RIII representatives that a stop work order had been issued for installation of concrete expansion bolts. On August 28, 1978, the licensee informed RIII that a stop work order had been issued for hydraulic snubbers. A stop work order for installation of mechanical snubbers was issued previously. It was indicated that the stop work orders would remain in effect until evaluations had been completed and corrective measures had been established. As part of the corrective action program, Sargent and Lundy design engineers have been assigned to the Zimmer site to review and coordinate design of pipe hangers and snubbers. A design review committee will be formed to review future designs, and an inspection team will be formed to review completed installations. This program will be reviewed by RIII inspectors during subsequent inspections.

CONCLUSIONS

1. Adequacy of pipe support base plate design was not reviewed as a part of this investigation, but will be treated as a part of a generic issue.

2. Six items of noncompliance with NRC requirements were identified, four of which were directly related to allegations made by Individual "A". These items of noncompliance were related to: concrete anchor bolt installation, pipe hanger and concrete anchor bolt inspection, quality documentation for concrete anchor bolts, pipe hanger design control, document control and review, and protection of equipment during construction.

DETAILS

Section I

Prepared by J. E. Foster

Reviewed by C. E. Norelius, Assistant to the Director

1. Personnel Contacted

   Cincinnati Gas and Electric Company

   B. K. Culver, Project Manager
   D. C. Kramer, Quality Assurance Engineer
   J. R. Schott, Station Superintendent
   W. W. Schwiers, Principal Quality Assurance and Standards Engineer
   J. F. Weissenberg, Quality Assurance Engineer

   Kaiser Engineering, Incorporated

   E. Arnett, Pipefitter
   W. Garner, Pipefitter
   R. Marshal, Construction Manager
   W. Puckett, Lead Mechanical Quality Assurance Inspector
   K. T. Shinkle, Hanger and Mechanical Inspector
   R. Turner, Quality Assurance Manager

   Individuals

   Individual "A"

2. Scope

   This investigation focused on the expressed concerns of Individual "A", relative to pipe hangers, restraints, and snubbers at the Zimmer 1 site. Design and installation of pipe hangers and snubbers were principal areas of interest. An expressed concern relative to base plate design (thickness) was not considered as within the scope of this investigation, as it is being treated as a generic issue.

   On June 27, 1978, NRC Headquarters personnel contacted Individual "A".

A-XII - 62

On June 29, 1978, RIII personnel contacted Indiviaual "A" by telephone.

On July 11, 1978, RIII personnel interviewed Individual "A".

During August 9-11, 15 and 16, 1978, an on-site inspection was performed.

3. Initial Contact with Individual "A"

On June 29, 1978, RIII personnel contacted Individual "A" by telephone. Individual "A" indicated that he was employed at the Zimmer site, and was involved with work on pipe hangers, restraints, and snubbers. He stated that he felt that there were major problems related to piping equipment at the site. These problems included both design and installation of hangers, snubbers, and restraints.

Individual "A" indicated that he questioned much of the criteria apparently utilized in the design of plant piping suspension systems. He discussed concerns related to the size of bolts used on pipe hangers, thickness of pipe support base plates, snubbers possibly misaligned with anticipated load axes, pipe supports acting as anchors, and the adequacy of pipe support anchorages mounted to block walls.

He also indicated that, in many cases, installation of pipe hangers, snubbers, and restraints was improper. He indicated that craftsmen frequently installed components not in accordance with the design drawings. The deviations were recorded in as-built drawings and were sent to Sargent and Lundy for approval. Individual "A" also stated that concrete expansion bolt type anchors were improperly installed, locknuts were missing in some hanger locations, and seismic snubbers were improperly installed.

4. Interview of Individual "A"

On July 11, 1978, Individual "A" was interviewed by RIII personnel. He repeated the comments made previously, and provided details as to locations and drawing numbers for a number of pipe hangers and snubbers alleged to be deficient. Individual "A" indicated that this was not a full listing of locations of deficient components, and a general inspection of hangers and snubbers would easily identify the problems he mentioned.

Individual "A" provided additional information as to improper installation of concrete expansion anchors. He indicated that there were many examples of incorrect embedment depth for the anchors, and locations where the anchor bolts had been cut off to disguise improper embedment depth (the top of the bolt would be cut off to make it appear that the proper length of the bolt was embedded). Individual "A" stated that he believed that "Phillips red head" concrete anchor bolts had been substituted for "Hilti-quick" anchor bolts in some locations.

Discussion with Individual "A" provided additional information as to possible design problems. He indicated that in many places, pipe restraints designed as rigid supports were actually "anchors" as they would not allow pipe movement in any direction. Concerns related to the thickness of pipe support base plates were discussed, and Individual "A" was advised that base plate thickness (rigid plate analysis) had been identified as a generic issue applicable to many architect-engineering firms, and was being pursued by the NRC.

Individual "A" indicated that an additional design problem was indicated by the fact that site specifications prohibited welding of attachments across the flange of a steel beam, and yet this had been done on the snubbers for the main steam lines.

Individual "A" stated that a possible design deficiency existed where attachments were welded to Bergen-Paterson pipe clamps rather than utilizing additional clamps. He stated that this may overstress the pipe clamps.

5. Inspection

Information developed through contacts with Individual "A" was provided to RIII inspectors for review during a site inspection of pipe hangers, restraints, snubbers, and related piping components. The details of inspection findings are covered in Sections II and III of this report. An RIII Investigation Specialist accompanied the inspectors during the initial portion of the inspection. During the inspection, a number of the locations provided by Individual "A" were inspected.

6. Concerns not covered by this report

Several concerns indicated by Individual "A" are not treated in this report. These apply to items not yet fabricated, or installation problems such as the lack of locknuts or other easily correctable problems. It was found that the licensee had not performed final inspections of any safety-related hangers, snubbers, and restraint installations, and thus had not had the opportunity to identify and correct minor nonconformances.

Section II

Prepared by I. T. Yin

Reviewed by D. H. Danielson, Chief
Engineering Support Section 2

1. <u>Hanger and Snubber Inspection Program</u>

More than 50% of all the safety related hangers and snubbers have been installed, however, none of these items have received final quality control inspection and signoff. Many of these installed items were inspected for proper welding procedures, materials, weld size, and surface conditions. The inspector observed approximately 30 safety related hangers, snubbers, and restraints and identified the following:

a. <u>Installations that Deviated From Design Drawings</u>

1WS - 027SR - The snubber weld attachment was bolted to the wall instead of welded to a embeded plate as shown on the drawing.

1WS - 033SR - Same deviation as 1WS-027SR.

1WS - 138HA - The vertical anchor structure plate was fastened to the concrete by 2 bolts on one side and welded to a embeded plate on the other side instead of being completely welded to the embed plate as shown on the drawing.

1WS - 025HV - Two of the four pipe riser shear lugs were not resting on the pipe clamp.

1RT - 014SR - An additional structural member was welded between the web of the I-beam and the snubber weld attachment.

1RT - 016SR - The snubber was attached to a horizontal beam instead of vertical post as shown on the drawing. The attachment differed from that shown on the drawing.

1WR - 215HR - The horizontal pipe line was supported on a column instead of resting on a structural system consisting of a horizontal beam, vertical hanger, and a U-bolt.

1WR - 200 HR - The pipe line was supported on a column instead of by a structural system consisting of a swivel linkage.

1WR - 214HR - Same deviation as 1WR - 200HR.

1RH - 011HR - The rigid strut was welded to the web of W 12 x 27 below instead of being welded to an embedment plate above. The filler weld size was measured to be 1/4", same as the web thickness of W 12 x 27. The weld measured to be 1/4" was not in accordance with the requirements of KEI QACMI M-12.

A Design Document Change (DDC) was written for 1RH-011HR and was approved by the Architect Engineer and the licensee. The actual installation differed from that approved by the DDC. A DDC has been written for 1WS-033, but was not yet approved. Two DDC's for 1RT-014SR and 1RT-016SR were in preparation.

b. Conflict Between Design and Construction Requirements

Welds crossing a beam flange were observed at many locations including the following snubber and hanger beam attachments:

1RT - 008SR	1RH - 485HR
1RT - 009SR	1RH - 486HR
1RH - 006SR	1RH - 487HR
1RH - 004SF	1RH - 002SR

The welding was as specified on the design drawings, however welding across a beam flange is in conflict with common industrial practice, and the requirements of KEI QAMCI M-12.

In view of the above identified problems, the licensee's hanger and snubber inspection program appears to be inadequate in that, repeated nonconformances were permitted to continue because none of the installation has received final inspection as of the date of NRC inspection.

This is considered an item of noncompliance with 10 CFR 50, Appendix B, Criterion X, and Wm. H. Zimmer, Unit 1, FSAR Paragraph 17.1.10 requirements. (358/78-18-01)

2. Review of Design Changes

An Inter-office Memorandum (IOM), entitled "Essential Piping Hanger Installation Criteria and Inspection Requirements," issued by the KEI QA Manager, dated September 22, 1977, was reviewed by the inspector. Among the instructions, it states that "Modifications to hanger designs will be accomplished by issue of a DDC. Inspection verification of compliance with DDC's will be recorded on the inspection record copy of the drawing. A copy of the DDC must be available to Quality Assurance prior to performing the construction change. In the event construction modification is to be accomplished prior to formal approval of the DDC, Quality Assurance will perform inspections in accordance with the unapproved DDC's requirements. In these cases final acceptance will be withheld until an approved copy of the DDC is received."

This instruction deviated from the Cincinnati Gas and Electric Company QA Manual, dated May 6, 1977, Paragraph 3.12.2(c), which states that "DDC's are used when it is desired to obtain expedited approval of drawing or specification changes without waiting for a formal revision of the affected document."

As noted in Paragraph 1.a above, hangers and snubbers have been installed contrary to design requirements. In discussion with with KEI QA inspection group, the inspector was informed of the site common practice of "red lining" as-built deviations after component installation and submitting these "red lined" drawings to the design engineer's office in hopes that the changes would be acceptable.

This is considered an item of noncompliance with 10 CFR 50, Appendix B, Criterion III, and Wm H. Zimmer, Unit 1, FSAR Paragraph 17.1.3 requirements. (358/78-18-02)

3. Document Control

The use of IOM to document work instruction as described in Paragraph 2 above without issuance, review, and approval as well as other document controls such as updating and distribution is considered an item of noncompliance with 10 CFR 50, Appendix B, Criterion VI, and Wm H. Zimmer, Unit 1, FSAR Paragraph 17.1.6 requirements. (358/78-18-03)

The inspector questioned if work instructions for other safety related construction had been issued through IOM's? It was further noted that all instructions documented in IOM's should be placed in controlled manuals or procedures. The licensee agreed to review this matter.

4. Protection for Hydraulic Snubbers

Bergen-Paterson (B-P) snubbers numbered 1RT-017SR, 1RT-015SR, 1RH-006HR, 1RH-001SR and 1WS-044SR, which represent approximately 20% of all the snubbers observed, were observed without the required accumulator indicator protective covers. The requirement for installing these covers was discussed during a previous RIII inspection on May 31 and June 1-2, 1978, (Report No. 50-358/78-10, Paragraph 5.b).

Failure to protect safety related component during installation is considered an item of noncompliance with 10 CFR 50, Appendix B, Criterion XIII, and Wm. H. Zimmer, Unit 1, FSAR Paragraph 17.1.13 requirements. (358/78-18-04)

5. Design Review

The following hanger, snubber, pipe whip restraint, and attached structures were observed by the inspector.

1RT - 016SR - The installation deviated from design drawings (Para. 1.a). The torsional moment at the horizontal I-beam may be excessive.

1RT - 014SR - The installation deviated from design drawings (Para. 1.a). This snubber and snubber 1RT-016SR were welded at the same location. The attachment of snubber 1RT-016SR to the horizontal beam could impose an additional torsional moment. The 1/4" fillet weld on the W 8 x 17 beam having a web thickness of 1/4" is not in accordance with KEI QACMI M12 which requires the weld at the web to be 1/16" less than the web thickness.

1 WS - 138HA - The installation deviated from design drawings (Para. 1.a). The structural arrangement may not be sufficient for the loadings shown on the drawing.

1 RH - 011HR - The installation deviated from design drawings (Para. 1.a). The loading shown on the drawing may be excessive for the as-built condition.

A XII 68

1028

1 WR - 322HA - The anchor attachment to the pipe may not be sufficient to withstand the loading shown on the drawing.

1 RH - 271HV - The two tack welds on the pipe saddle may not be sufficient to withstand system vibration loadings.

1 WS - 025HV - The installation deviated from design drawings (Para. 1.a). Excessive loading could be imposed on the pipe riser lugs.

RHFR - 1003 - A portion of the pipe whip restraint ring section (2'-6" dia. x 8" wide x 2" thick) was removed to provide clearance for the pipe. The DDC S-1276 allowed a 1 3/4" x 1'-3" cut with radius corners. The starting edge of the cut was shown to start 3" from the edge of positioning structure. The actual cut started 9 3/4" away from the edge of positioning structure and extended outward at approximately 45° without radius corners. The inside dimension of the cut was approximately 2 7/8" x 1' -10" (near center) and 2' -1" (at the edge of the ring). The outside dimension of the cut was approximately 2" x 1' -10" (near center) and 2' -1" (at the edge of the ring). The cut was tapered at all locations. A nonconformance report had not been written.

Beam Cuts - One side of the I-beam flanges on a W24 x 68 radial structural beam at El. 534' was cut out to the edge of the web. The cut length was 12" at the top and 24" at the bottom. The flange of a W14 x 314 beam was welded to the web of the W24 x 68. No reinforcement was at the cut. The cut was made to clear the recirculation loop discharge valve actuator. The structural arrangement is shown on S&L Drawing S-398.

1RH-017SR - A DPE load of 4686 lbs. is shown acting on the 1/4" thick web of a W 8 x 17 beam. The structural adequacy is questioned. In addition, a 1/4" fillet weld is shown on the drawing, which is in conflict with KEI requirements, (see 1RT-014SR above for explanation).

1RH-443SR and 1RH-455SR - Two horizontal seismic restraints are installed on each side of valve 1E12F073B and valve 1E12F074B. The pipe line is 1RH 56BB 1 1/2". The inspector questioned the following:

a. The adeqauacy of the W 8 x 17 beam to resist the torsional moment imposed by this installation.

b. The weight of each valve including the motor operator is 220 lbs. $\pm$ 10% and is about 13" off the centerline of the 1 1/2" pipe. The adequacy of the pipe section to withstand the total offset loading of 440 lbs., taking into consideration local pipe stress and seismic restraint requirements.

Beam Stiffeners - A number of beam gusset plates were welded to I-beam sections in the area of the RHR Heat Exchanger Room West. The plates were welded on three sides without opening at the corners. This is not considered an acceptable industry practice because the triaxial stresses imposed due to welding could result in cracking at the weldment.

The licensee agreed to perform a complete review of hanger, restraint and attachment structure designs used on all safety related piping systems. This is considered an unresolved item. (358/78-18-05)

Section III

Prepared by E. J. Gallagher

Reviewed by R. L. Spessard, Chief
Engineering Support Section 1

1. Review of Design/Construction Drawings For Installation of Anchorage of Supports Using Concrete Expansion Anchor Bolts

The inspector reviewed selected drawings being used to install safety-related supports and restraints anchorage to concrete using concrete expansion anchor bolts. The following are the results:

a. Drawing Nos. M-126-10H-58 (1WS 172SR), M-126-10H-57 (1WS 171SR), and M-126-7H-15 (1WR 137HR) are Sargent and Lundy detail construction drawings which do not include instructions critical to the anchorage detail to assure anchorage capacity, i.e., expansion bolt length nor embedment depth. These drawings have received design review and QA review.

b. The following drawings reviewed do not include the embedment depth to which the anchor bolts are to be installed to assure adequate anchorage capacity: M-448-3H-29 (1WR 205HR), M-126-7H-19 (1WS 220HR), M-448-3H-64 (1WR 175SR) and M-448-3H-61 (1WR 236HR). It was brought to the inspector's attention that none of the detail drawings issued to date include these instructions to assure proper installation.

c. Specifications, procedures or instructions have not been written to date, regarding the use, installation or testing required when using concrete expansion anchor bolts for anchorage of safety-related supports or restraints. Instructions that would be required include minimum embedment depth for each diameter bolt, minimum spacing requirements, minimum edge distance, instructions when in contact with reinforcing steel, relocation instructions and torque requirements to assure the bolts are capable to develop the tensile capacity required.

d. The inspector reviewed design/construction drawings which indicate anchorage of safety-related supports and restraints to non-safety related block walls both hollow core block (concrete filled) and high density shielding block walls bonded together with a mortar mix. Specification H-2174, Section 4-1 (masonry work) includes hollow core block and high density block walls. This specification was identified as (nonessential) non-safety related work.

Examples of this condition are shown on drawings M-448-6H-41 (1WR08HR) and M-488-8H-20 (1WR232SR). The safety-related supports are designed to transmit loads to structural members.

This failure to assure that applicable design bases, as indicated above in a through d, for safety-related supports and restraints are correctly translated into specifications, procedures, drawings or instructions is considered an item of noncompliance with 10 CFR 50, Appendix B, Criterion III. (358/78-18-06)

This same condition was previously identified in RIII Inspection and Enforcement Report No. 358/78-09 (Item 358/78-09-04).

2. Observation of Anchorage of Supports and Restraints Installed

The following table is a list of support and restraints installed that the inspector observed to ascertain the quality of the installation versus the details on the design drawings:

Design Drawing No. (Restraint No.)	Observations
1WR087SR (M-448-6H-24)	(1) Three of four expansion bolts are saw cut, therefore the length of the bolt is questionable.
	(2) The plate is not fully bearing on the concrete due to improper embedment depth. The support at this time is unrestrained and free to move under load conditions.
1MS0104SR (M-401-9H-63)	Main steam line snubber is welded across the flange which conflicts with requirements of inspection procedure QACMI-M12, Section 5.2.12(a).
1WR175SR (M-448-3H-64)	At least three holes are drilled next to installed anchor bolt plate. Holes reduce the effective concrete stress cone resisting anchor plate loads.

1WS220HR (M-126-7H-19)	(1)	Plate is 3/4 inch off floor slab. Detail drawing indicates bearing directly on slab.
	(2)	Due to condition in (1) improper embedment depth is evident.
1WR013SR (M-448-5H-42)	(1)	Drawing indicates use of embed plate where restraint installed uses expansion anchor bolts.
	(2)	Washers under nut turns yet nut has been tightened.
	(3)	Two nuts have not been tightened or torqued.
	(4)	One bolt has a stack of four washers, therefore reducing embedment depth required.
	(5)	Bolts have improper embedment depth as per manufactures instructions, i.e., 4.5 times diameter of bolt
	(6)	Bolts installed on angle reducing capacity of bolt.
	(7)	Bolts installed violate manufacturers minimum spacing requirements, i.e., 10 times anchor diameter
1WR205HR (M-448-3H-29)		Drawing indicates base plate under pipe stanchion; no base plate installed and pipe is completely restrained to walls. Base plate unable to be installed in this situation.
1WS033SR (M-126-13H-32)	(1)	Nuts on bolts not fully engaging threads; potential cause for thread failure under load.
	(2)	Drawing indicates use of embedded plate and restraint installed uses four expansion bolts of undetermined length. No design change issued for this restraint.

1WS025HV Bolts installed violate minimum edge distance requirement of manufacturer, i.e., minimum 5 times anchor diameter.

Based on the above observations, work has not been accomplished in accordance with the construction details provided by the applicable drawings.

This failure to accomplish activities affecting quality in accordance with instructions or drawings is considered an item of noncompliance with 10 CFR 50, Appendix B, Criterion V. (358/78-18-07)

In addition to the observations made on the above specific supports and restraints the following two general observations were made:

a. The inspector determined that the bolts installed to date have had no inspection performed to assure that the correct length bolt has been installed according to design drawings. The bolts do not include a length identification marker that is available from the supplier of the bolts to be inspected subsequent to installation nor has any inspection been performed during installation to ascertain the use of the correct length bolts.

b. Interviews with two pipe fitters responsible for installing the anchors indicated that the expansion bolts have been installed without applying a torque to the specified ranges of values as required by Design Document Control (DDC) No. SLS-266 dated April 5, 1977. No inspection had been provided to assure the use of a calibrated torque wrench for setting the bolts to the prescribed torque range. The torque appli[illegible] to the bolt directly affects the tensile capacity of the [illegible]t.

Based on the above observations of installed supports and restraints and interviews with the pipe fitters performing the installation of the concrete expansion anchors, Kaiser Engineer, Inc., (KEI) has not provided an inspection program to assure that anchorage of safety-related components has been installed to the design drawings and manufacturers minimum requirements.

This failure to provide an inspection program to assure adequate installation of safety-related items is considered an item of noncompliance with 10 CFR 50, Appendix B, Criterion X. (358/78-18-08)

3. <u>Review of Procurement Documents for Concrete Expansion Anchor Bolts</u>

The inspector reviewed the procurement documents used to purchase concrete expansion anchor or bolts used for anchorage of safety-related supports. The procurement documents do not indicate quality assurance requirements in the space provided for such requirements.

The procurement agent informed the inspector that Sargent & Lundy Specification H-2174, Section 5-4.6.4 requires the use of Hilti "Kwik Bolts" and does not specify any quality requirements. The supplier has therefore not submitted a QA Manual nor has KEI or CG&E performed any QA audits of the supplier. The procurement documents do not reference quality standards for the material supplier nor a certificate of conformance.

This failure to assure that requirements which are necessary to assure adequate quality are suitably included or referenced in the documents for procurement of material and services is considered an item of noncompliance with 10 CFR 50, Appendix B, Criterion IV. (358/78-18-09)

PDR

RECEIVED

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
1978 DEC - 299 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137
OFFICE OF THE SECRETARY D.C.

OCT 25 1978

Docket No. 50-358 Q

Cincinnati Gas and Electric
Company
ATTN: Mr. Earl A. Borgmann
Vice President Engineering
139 East 4th Street
Cincinnati, OH 45201

Gentlemen:

This refers to the inspection conducted by Messrs. I. T. Yin and E. J. Gallagher of this office on September 28-29, 1978, of activities at the Wm. H. Zimmer Power Station authorized by NRC Construction Permit No. CPPR-88 and to the discussion of our findings with Mr. W. W. Schwiers and others of your staff at the conclusion of the inspection.

The enclosed copy of our inspection report identifies areas examined during the inspection. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations, and interviews with personnel.

No items of noncompliance with NRC requirements were identified during the course of this inspection.

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and the enclosed inspection report will be placed in the NRC's Public Document Room, except as follows. If this report contains information that you or your contractors believe to be proprietary, you must apply in writing to this office, within twenty days of your receipt of this letter, to withhold such information from public disclosure. The application must include a full statement of the reasons for which the information is considered proprietary, and should be prepared so that proprietary information identified in the application is contained in an enclosure to the application.

A-XII-76

7811300257

1028 234

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

R. F. Heishman

R. F. Heishman, Chief
Reactor Construction and
Engineering Support Branch

Enclosure: IE Inspection
Report No. 50-358/78-22

cc w/encl:
Mr. J. R. Schott, Plant
Superintendent
Central Files
Reproduction Unit NRC 20b
PDR
Local PDR
NSIC
TIC
U. Young Park, Power
Siting Commission

A-XII-77

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

REGION III

Report No. 50-358/78-22

Docket No. 50-358 License No. CPPR-88

Licensee: Cincinnati Gas and Electric Company
139 East 4th Street
Cincinnati, Ohio 45201

Facility name: Zimmer, Unit 1

Inspection at: Zimmer 1 Site, Moscow, Ohio

Inspection conducted: September 28-29, 1978

Inspectors: I. T. Yin 10/23/78

E. J. Gallagher 10/23/78

Reviewed by: D. H. Danielson, Chief 10/23/78
Engineering Support Section 2

<u>Inspection Summary</u>

<u>Inspection on September 28-29, 1978 (Report No. 50-358/78-22)</u>

<u>Areas Inspected</u>: Followup inspection of problem areas relative to the safety related hangers, restraints, and concrete expansion anchor bolts installation which were identified during previous RIII inspections. The inspection involved 28 inspector-hours onsite by two NRC inspectors.

<u>Results</u>: No items of noncompliance or deviations were identified.

A-XII-78

811300...

1028 236

DETAILS

Persons Contacted

Principal Licensee Employees

*W. W. Schwiers, Principal QA and Standards Engineer
*J. F. Weissenberg, QA and Standards Engineer
*B. K. Culver, Project Manager
*D. C. Kramer, Quality Assurance Engineer

Kaiser Engineer, Inc. (KEI) Employee

*R. E. Turner, QA Manager

The inspector also contacted other employees and craftsmen during the inspection, including representatives of General Electric Company, and Kaiser Engineers, Incorporated.

*Denotes those present at the Exit Interview.

Licensee Action on Previously Identified Items

(Closed) Unresolved Item (358/78-01): Environment qualification hydraulic snubber seals. The inspector reviewed the subject concern and determined that no seal materials in the market to date can withstand the radiation and operation environment conditions inside the power reactor containment for 40 years of service life. It is a common practice for the hydraulic snubber vendors to select the best seal materials available through laboratory testing and plant operating experience, and to provide service and replacement procedures in case of material deterioration identified. The present control and selection of seal materials by the major snubber vendors are considered acceptable.

(Open) Noncompliance Item (358/78-10-01): Inadequate specification for procuring mechanical snubbers. The inspector reviewed Supplement 5 to Specification H-2259, dated June 6, 1978, and considered it inadequate. This was based on the fact that unit activation parameters were addressed but no mention of (1) the unit bleed rate of the hydraulic snubbers, and (2) equivalent load relieﬁng characteristics of the mechanical snubbers. Further, the inclusion of cold position settings for the snubbers in the S & L installation drawings will not be completed until November 1, 1978. In addition, the inspector stated that although snubber hot position setting (HPS) is not required during installation, the HPS should be verified during system hot functional testing.

A - XII 79

1028 237

(Closed) Unresolved Item (358/78-10-02): Purchase specification for E-System hydraulic snubbers. The inspector reviewed S & L Spec. H-2897, "Hydraulic Snubbers for Reactor Recirculation and Main Steam Piping" dated April 5, 1978, and consider it acceptable.

(Open) Noncompliance Item (358/78-10-04): Installation of INC mechanical snubbers without adequate installation and inspection procedures. The inspector reviewed: (1) Kaiser Engineers, Inc. (KEI) Procedure 2-126, "Installation of Mechanical Shock and Vibration Arrestors", Rev. 0, dated July 26, 1978, and (2) KEI Procedure 2-127, "Installation of Hydraulic Shock an Sway Arrestors", Rev. 0, dated August 1, 1978, and considered them acceptable. The item remains open because (1) the update of KEI, QACMI, M-12 has not been reviewed and approved for use, and (2) the re-inspection of the installed mechanical snubbers based on the latest procedure has not been initiated.

(Closed) Unresolved Item (358/78-10-05): Qualification test reports for the ITT-Grinnell hydraulic snubbers. During a licensee audit of General Electric Company, (GE), San Jose, CA on June 5-7, 1978 (Audit Report 78-04), the Acton Environmental Test Corporation reports 12215-1 (dated January 31, 1976, relative to the environmental testings) and 12215-5 (dated February 17, 1976, relative to the seismic testings) on a 2 1/2" bore by 5" stroke snubber was reviewed by the licensee and considered acceptable. The Test Report 12215-4, dated April 21, 1976, relative to the largest snubber provided under GE Spec. 21A9422 was also reviewed and accepted by the licensee.

(Closed) Unresolved Item (358/78-10-06): E-System hydraulic snubber qualification reports. The inspector reviewed the technical reports issued by E-System and considered them acceptable. For details see Section I, Paragraph 2.

(Open) Unresolved Item (358/78-10-07): International Nuclear Safeguards Corporation (INC) mechanical snubber environmental transient and performance tests. The INC Report No. 116, "Summary of Design Data, Operational Characteristics and Test Results of the Mechanical Shock and Vibration Arrestor", Rev. 1, dated June 16, 1976, was reviewed by the inspector. The dynamic functional characteristics of the A, AS, D, and DS type snubbers, the preventive measures for jamming up, and the applicability of the general type report to the specific purchase specification were not apparent. A meeting with INC in their engineering office arranged through licensee to discuss these issues was requested by the inspector.

(Closed) Noncompliance Item (358/78-10-10): Inadequate indoctrination and training records. The inspector reviewed the training records dated September 15, 1978, for installation of mechanical and hydraulic snubbers, and considered it acceptable.

X-XII-80

1028 238

3. NRC Staff Report - Mark II Containment	5:15 pm - 6:00 pm (45 minutes)
a. Lead Plant Acceptance Criteria	
b. Zimmer Design Assessment	
c. Zimmer SRV Tests	
e. Generic Acceptance Criteria	
4. Applicant Response to Items 1 and 3	6:00 pm - 6:30 pm (30 minutes)
5. General Discussions and Conclusions	6:30 pm - 7:00 pm (30 minutes)

(Open) Unresolved Item (358/78-18-05): Design review for safety related pipe suspension. The licensee performed an audit in S & L office and identified several problems. For details, see Section I, Paragraph 1.

Functional or Program Areas Inspected

Functional and program areas inspected are documented in Section I and Section II of this report.

A XII - 81

1028 240

SECTION I

Prepared by I. T. Yin

Reviewed by D. H. Danielson, Chief
Engineering Support
Section 2

1. Design Review for Safety Related Pipe Suspension

The adequacy of the subject matter was questioned by the inspector during an investigation on August 9-11, and 15-16, 1978 (RIII Report 78-18). Subsequently, the licensee performed an audit relative to the concerns at the S & L office, Chicago, on September 6-7, 1978. The inspector reviewed the Audit Report No. 78/07, and considered some of the findings to be significant. These included:

a. Insufficient implementation of document review procedures.

b. Re-evaluation of the hangers inside the auxiliary building was scheduled for completion by September 28, 1978.

c. Re-analysis of hangers inside the containment was scheduled for completion by November 30, 1978.

d. S & L has not maintained a record of support design calculations. Many of the support designs resulted in torsional stresses which were higher than the allowables.

e. Inadequate review for Design Document Changes (DDC's).

A followup licensee audit in the same areas will be conducted at S & L office on October 16-17, 1978. The inspector noted that he would like to observe the audit.

No items of noncompliance or deviations were identified.

2. Review of E-System Hydraulic Snubber Qualification Test Reports

During the visit, the inspector reviewed the following technical test reports submitted by the vendor to the licensee. No problem areas were identified during the review.

a. No. 152000-600, "Test Report on Non-Metallic Seal Material for use in Snubbers", Rev. A, dated October 12, 1977.

b. No. 152000-620, Volume 1 of 9, "Summary Report, Product Qualification Test Report, GE Pipe Suspension Snubber", Rev. B, dated January 20, 1978.

c. No. 152000-620, Volume 2 of 9, "Administrative Data, Production Qualification Test Report, GE Pipe Suspension Snubber", Rev. A, dated December 8, 1977.

d. No. 152000-620, Volume 4 of 9, "Test Data, 20 Kip Snubbers, Qualification Test Report, GE Pipe Suspension snubber", Rev. A, dated December 8, 1977.

e. No. 152000-620, Volume 6 of 9, "Test data, 50 Kip Snubber, Qualification Test Report, GE Pipe Suspension Snubber", Rev. A, dated December 8, 1977.

f. No. 152000-620, Volume 7 of 9, "Test Data, 70 Kip Snubber, Qualification Test Report, GE Pipe Suspension Snubber", Rev. A, dated December 8, 1977.

SECTION II

Prepared by E. J. Gallagher

Reviewed by R. L. Spessard, Chief
Engineering Support
Section 1

1. Status of Work on Installation of Anchorage of Pipe Supports and Restraints

Subsequent to the IE investigation conducted at the Zimmer plant on August 9-11, 15-16, 1978, CG&E issued a stop-work order No. 78-02 after a number of deficiencies were identified related to the use, installation and inspection of concrete expansion anchors used to anchor safety-related pipe supports and restraints.

This stop-work order was lifted effective September 7, 1978, based on the corrective action taken, in particular, the initiation of procedures for installation and inspection of expansion bolts, training of craftsmen installing the bolts, identification of quality assurance requirements for the procurement of the product and the application of a length identification stamp on the head of each bolt using a permanent die stamp.

2. Review of Specification and Procedures for Installation of Concrete Expansion Anchor Bolts

The inspector reviewed the following specification and procedures being used for the installation and inspection of concrete expansion anchor bolts used for anchorage of safety-related (essential) supports and pipe restraints:

a. DDC No. SLS-315 (August 31, 1978) and attached Sargent and Lundy specification entitled, "Concrete Expansion Anchors: Installation and Inspection Procedure," Rev. 2 dated August 31, 1978.

b. QACMI M-12 Rev. 1 entitled, "Inspection Instructions for Pipe Hangers and Support Installation."

c. Field Construction Procedure FCP 2-128 Rev. 4 dated August 31, 1978.

d. QACMI M-15 Rev. 1 entitled, "Concrete Expansion Anchor Post-Installation Procedure".

The inspector was informed that QACMI M-15 will be used to inspect expansion anchor bolts installed prior to the issuance of DDC SLS-315 (August 31, 1978) and FCP 2-128 (August 31, 1978) and that DDC SLS-315, QACMI M-12 and FCP 2-128 will be used for the installation and inspection of expansion anchor bolts installed after August 31, 1978.

QACMI M-15 requires an inspection to be performed on bolts installed prior to August 31, 1978, and includes ultrasonic testing to determine the length of the installed anchors as well as inspecting bolt spacing, edge distance and embedment depth for all bolts and inspecting the applied torque on a frequency of one bolt per hanger. If this one bolt is unacceptable, the procedure requires testing of all bolts for that particular hanger. In addition, all bolts that have been saw cut or show excessive projection shall be checked for torque and embedment depth.

The following items relative to the specification and procedures were discussed and were not able to be resolved during this inspection:

a. S & L specification, Section 2.2.3, Table E lists the minimum testing torque requirements which are much less than the installation requirements in Table D, e.g., a 3/4 inch bolt is required to be installed to 125 to 175 foot-pounds and tested to 81 foot-pounds. The inspector requested the engineering justification for the established values. The licensee agreed to make this information available during the follow-up inspection.

b. QACMI M-15, Rev. 1, Section 3.6 states that, "bolts installed out-of-plumb by greater than 5° shall be unacceptable." S & L specification for installation does not include a tolerance or requirement for installation plumbness. Craftsmen are being trained in accordance with S & L spec. This requirement is under evaluation by the licensee.

The above items are considered unresolved until the information is made available at a subsequent inspection, (358/78-22-01).

3. Calibration of Torque Wrenches Used for Installation of Expansion Anchors

S & L specification for concrete expansion anchors, Rev. 2, Section 2.2.1 requires torque wrenches to be used for inspection

A-XII-85

1028 244

and to be calibrated on a weekly basis, if using snap-type torque wrenches. The inspector reviewed the records of five of the eight torque wrenches to be used by the craftsmen and found the calibration records to be satisfactory.

4. Procurement Documents for Concrete Expansion Anchors

The inspector reviewed purchase requisition No. 25333 dated September 13, 1978, for concrete expansion anchor bolts manufactured by Hilti Fastener Systems. The purchase order identified the quality assurance requirements, in particular, the requirement for the supplier to issue a certificate of conformance for material properties and a requirement for a length identification marker to be stamped on the head of each bolt. This stamp is in the form of a letter, e.g., "L" which corresponds to a length of 4 3/8 " or "R" (6 1/4"). The inspector observed in the warehouse a supply of anchor bolts with the length identification marker applied.

5. Training of Craftsmen on the Installation of Expansion Anchors

Field Construction procedure FCP 2-128, Rev. 4, Section 3.1.1 requires the craft superintendent to instruct the craftsmen in accordance with installation procedures and maintain a record log of the qualified craftsman. The inspector reviewed this log with the craft superintendent, and he as interviewed two craftsmen installing the anchors in the field. Discussion with these craftsmen indicated that a training session had been performed and that they were familiar with the installation requirements of the procedure. Torque wrenches were not being used as they were in for calibration.

No items of noncompliance were identified in the above areas inspected.

Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, items of noncompliance or deviations. One unresolved item disclosed during the inspection is discussed in Section II, Paragraph 2.

Exit Interview

The inspectors met with site staff representatives (denoted in the Persons Contacted paragraph) at the conclusion of the inspection on September 29, 1978. The inspectors summarized the purpose and findings of the inspection. The licensee acknowledged the findings reported herein.

PDR

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III

DEC 21 1978

Docket No. 50-358

Cincinnati Gas and Electric
Company
ATTN: Mr. Earl A. Borgmann
Vice President Engineering
139 East 4th Street
Cincinnati, OH 45201

Gentlemen:

This refers to the investigation conducted by Messrs. J. E. Foster, T. E. Vandel and H. M. Wescott of this office on September 18-22 and 28-29, 1978, of activities at the Zimmer Unit 1 construction site, authorized by NRC Construction Permit No. CPPR-88, and to the discussion of our findings with you, Messrs. B. K. Culver and W. W. Schwiers and others of your staff at the conclusion of the investigation.

This investigation concerned allegations of inadequate materials and welding of cable trays, pans and fittings supplied to the Zimmer Unit 1 site. The enclosed copy of our investigation report identifies those areas examined during the investigation. Within these areas, the investigation consisted of a selective examination of procedures and representative records, observations, witnessing of tests, and interviews with personnel.

During this investigation, one of your activities appeared to be in noncompliance with NRC requirements, as described in the enclosed Appendix A.

This notice is sent to you pursuant to the provisions of Section 2.201 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations. Section 2.201 requires you to submit to this office within thirty days of your receipt of this notice a written statement or explanation in reply, including for each item of noncompliance: (1) corrective action taken and the results achieved; (2) corrective action to be taken to avoid further noncompliance; and (3) the date when full compliance will be achieved.

7902010327

A XII 87

1028 246

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter, the enclosures, and your response to this letter will be placed in the NRC's Public Document Room, except as follows. If the enclosures contain information that you or your contractors believe to be proprietary, you must apply in writing to this office, within twenty days of your receipt of this letter, to withhold such information from public disclosure. The application must include a full statement of the reasons for which the information is considered proprietary, and should be prepared so that proprietary information identified in the application is contained in an enclosure to the application.

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

R. F. Heishman

R. F. Heishman, Chief
Reactor Construction and
Engineering Support Branch

Enclosures:
1. Appendix A, Notice of Violation
2. IE Investigation Rpt No. 50-358/78-21

cc w/encls:
J. R. Schott, Plant Superintendent
Central Files
Reproduction Unit NRC 20b
PDR
Local PDR
NSIC
TIC
U. Young Park, Power Siting Commission

A XII 88

1023 247

Appendix A

NOTICE OF VIOLATION

Cincinnati Gas and
Electric Company

Docket No. 50-358

Based on the results of a NRC investigation conducted on September 18-22, and 28-29, 1978, it appears that certain of your activities were not conducted in full compliance with NRC requirements as noted below. This item is a deficiency.

10 CFR 50, Appendix B, Criterion IX requires, in part, that "Measures shall be established to assure that special processes, including welding,..., are controlled and accomplished by qualified personnel using qualified procedures in accordance with applicable codes, standards, specifications, criteria, and other special requirements."

Paragraph 17.1.9.2 of the Quality Assurance Program documented in the ZPS-1 FSAR states, in part, "Special processes are accomplished and controlled by qualified personnel using qualified procedures in accordance with applicable codes, ...

Section IX of the ASME Code states that changes in essential variables to the welding procedure specification require requalification of the procedure and welder. Section IX further lists shielding gas and filler material size as essential variables.

1. Husky Products, Incorporated, Welding Procedure No. 2, QAP 107, dated October 18, 1974, Revision No. 01 specifies that welding grade carbon dioxide shielding gas and 0.035" diameter filler metal be used.

   Contrary to the above, the inspector determined by review of records that two (2) of the essential variables had been changed. For a period of approximately four (4) weeks in November and December 1974, the shielding gas mixture and the size of the filler material was changed without benefit of requalification of the procedure.

2. In addition, two welders had made several steel TIG weldments with neither a qualified welding procedure specification nor qualification of the welders.

7902010332

A XII 89

1028 243

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

REGION III

Report No. 50-358/78-21

Docket No. 50-358 License No. CPPR-88

Licensee: Cincinnati Gas and Electric Company
139 East 4th Street
Cincinnati, OH 45201

Facility Name: Wm H. Zimmer Nuclear Power Plant

Investigation At: Zimmer site, Moscow, Ohio and Husky Products, Inc., Florence, Kentucky

Investigation Conducted: September 18-22, and 28-29, 1978

Investigator: J. E. Foster

Inspectors: T. E. Vandel 12-18-78

R. M. Wescott 12-18-78

Reviewed by: D. W. Hayes, Chief
Projects Section

C. E. Norelius
Assistant to the Director

Investigation Summary

Investigations on September 18-22 and September 28-29, 1978 (Report No. 50-358/78-21)
Areas Inspected: Review of cable trays, pans and fittings located at the Zimmer site and at the Husky Products, Inc. plant; review of activities at the Husky Products, Inc. plant; and observation of testing activity at independent test labs. The investigations involved 143 inspector-hours by three NRC inspectors.
Results: One item of noncompliance (a deficiency) was identified in the control of special processes (welding). Details, Section III.

7902010335 A-XII-90

1028 249

INTRODUCTION

The Zimmer Unit 1 nuclear power plant, licensed to the Cincinnati Gas and Electric Company, is under construction near Moscow, Ohio. Sargent and Lundy is the Architect-Engineering firm for the plant, which is being constructed by Kaiser Engineering. The facility will utilize a Boiling Water Reactor (BWR) designed by General Electric Company.

The Husky Products Division (Husky) of the Burndy Corporation has supplied electrical cable pans for the Zimmer plant. These cable pans are utilized to route both safety-related and nonsafety-related electrical cables.

REASON FOR INVESTIGATION

On August 31, 1978, a copy of a letter written by Individual "A", a former Husky employee, was received at the NRC Region III (RIII) office (Exhibit I). This letter expressed concerns relative to the quality of electrical cable pans produced by Husky for use in the Zimmer and Clinton nuclear power plants, and alleged the use of weak materials and improper welding in cable pan construction. An NRC investigation was initiated into these allegations.

SUMMARY OF FACTS

Individaul "A" was contacted by RIII personnel on September 8, 1978, and his concerns were discussed in general. These concerns related to the use of low strength materials and improper welding as contained in the letter attached as Exhibit I.

During September 18-20, 1978, RIII inspector visually inspected electrical cable pans at the Zimmer site, and found the welding of the pans to be acceptable. Site personnel agreed to have samples of the cable pan materials tested for material strength, and to have sections of cable pan destructively tested to determine the strength of the welds. Cable pans to be tested were then selected at random (by NRC and Utility representatives).

Cable tray samples selected were tensile tested, with the tests witnessed by an RIII inspector. All of the samples tested were found to exceed the specified yield point (test results attached as Exhibit V).

Destructive testing of welds was performed on a sample of the cable pans at the Zimmer site, also witnessed by RIII personnel. These tests indicated that the welds were of acceptable strength and size according to American Welding Society criteria.

A XII - 91

Individual "A" was interviewed by RIII personnel. He indicated that the use of low strength material was a one-time occurrence which took place during the manufacture of cable pans for the Zimmer plant. Individaul "A" stated that a shipment of steel was found to be of low strength, and the decision was made to use the shipment for "fittings" (curved sections of cable pan) only, but the shipment was not properly segregated. The shipment was inadvertantly used in the production of straight sections of cable pan, he indicated.

Individual "A" was critical of the manual welding performed by Husky welders, and the welding certification program conducted by Husky. He indicated that the Husky welders had difficulty in passing the certification tests, and welded differently during the test than in production welding.

In addition, comments were received which related to work at the Clinton plant, and are covered in a separate report (IE report No. 50-461/78-06).

RIII personnel made two visits to the Husky facility in Florence, Kentucky. During plant visits, the manufacturing areas were toured, work in progress was observed, pertinent records were reviewed, and interviews were held with Husky personnel.

Records reviewed, and interviews held with Husky personnel indicated that Husky welders had been qualified as required by the American Society of Mechanical Engineers Code for Boilers and Pressure Vessels, Section IX (ASME Section IX). No information relative to the use of low strength materials could be developed.

On September 22, 1978, RIII personnel visited the Union Testing and Research Laboratory, where material samples had been tested for Husky during production of cable pans for the Zimmer plant. Records relating to all tests of material for Husky for the years 1974-1976 inclusive were reviewed. None of the test reports reflected that materials to be used in the Zimmer plant cable pans did not meet the specified yield strength requirements.

During a second visit to the Husky facility, signed statements were obtained from three Husky employees. The personnel interviewed indicated that they had no knowledge of any low strength materials being used in construction of cable pans for the Zimmer site. (See Exhibits II, III and IV).

During document review, it was found that the shielding gas and diameter of the filler material utilized for the welding process differed from the qualified welding procedure for a period of approximately four weeks. This is in noncomformance with ASME Section IX in that a variable of the welding process was changed without subsequent requalification of the welding procedure and welders.

Husky personnel stated that they would have their welding procedure qualified with the alternate shielding gas and filler material, to demonstrate that the quality of the welds was not affected by the changes in weld procedure. Later contacts with Husky personnel indicated that some manual welding had been performed prior to procedure qualification.

CONCLUSIONS

1. No evidence was developed that low strength material had been utilized in fabrication of electrical cable pans for the Zimmer plant.

2. Materials and welding for cable pans supplied by Husky to the Zimmer plant were tested and found to be acceptable.

3. Welder certification had been performed as required by Section IX of the ASME Boiler and Pressure Vessel Code.

4. Welding wire and shield gas were not as specified in the qualified welding procedure for a period in 1974. In addition, two welders performed welding without benefit of prior qualification. This is in nonconformance with 10 CFR 50, Appendix B, Criterion IX and Section IX of the ASME code. (See Details Section III).

POOR ORIGINAL

A-XII-93

1028 252

DETAILS

Section I

Prepared by J. E. Foster
Reviewed by C. E. Norelius
Assistant to the Director

1. Personnel Contacted

Cincinnati Gas and Electric Company

E. A. Borgman, Vice President
B. K. Culver, Project Manager
R. P. Ehas, Quality Assurance and Standards Engineer
D. C. Kramer, Quality Assurance and Standards Engineer
J. R. Schott, Station Superintendent
W. W. Schweirs, Principal Quality Assurance and Standards Engineer
W. D. Waymire, General Engineering Department

Kaiser Engineers, Inc.

R. Turner, Quality Assurance Manager

Husky Products

Fred L. Banta, Engineering R&D Manager
Don Dietrich, Tool Engineer
Clare F. Duncan, Quality Control Manager
Ronald C. Johnson, Production Foremen
Randy Pratt, Industrial Engineer
Ken Rigley, Welding Operator
Duane Ring, President
Barry Schuster, Utilities Market Manager

The William Powell Co. (Union Testing and Resea[illegible]boratory)

Steven L. Fogle, Assistant Manager of Laborato[illegible]
Edwin E. Winterfeldt, Corp. Manager of Quality [illegible]rance

Individuals

Individuals "A" through "J"

A-XII-94

Sargent and Lundy

M. E. Schuster

Cincinnati Post-Enquirer

Douglas Starr, Staff Reporter

Hetcutt Research Associates

L. J. Fritz, Material Testing Supervisor
R. E. Duvall, Testing Technician

F&S Machining Services, Inc.

J. Foster, President

2. SCOPE and CHRONOLOGY

This investigation centered on the allegations provided by Individual "A", relative to the use of low strength materials and improper welding by Husky. This report covers those allegations and inspections which pertain to the Zimmer Unit 1 plant. Allegations made which pertain to the Clinton 1 plant will be reported in a separate report.

On August 31, 1978, a copy of a letter by Individual "A" was received at RIII.

On September 8, 1978, Individual "A" was contacted by RIII personnel.

During September 19-20, 1978, inspections were made at Clinton and Zimmer.

On September 20, 1978, Individual "A" was interviewed by RIII personnel.

During September 20-22, 27-29, 1978, RIII personnel visited the Husky facility.

On September 21, 1978, Individual "A" was contacted by telephone.

On September 22, 1978, RIII personnel visited the Union Test Lab.

On September 25, 1978, a second letter from Individual "A" was received at RIII (Exhibit VII).

On September 27, 1978, Individual "A" was re-interviewed by RIII personnel.

During September 27-29, RIII personnel visited the Husky facility.

On September 28, 1978, tests were performed on cable pans from the Zimmer site.

On September 29, 1978, RIII personnel visited Modern Welding and Sheet Metal.

3. Initial Contact with Individual "A"

On September 8, 1978, RIII personnel contacted Individual "A" by telephone. Individual "A" indicated that he had been the Manager of Industrial Engineering for the Husky Products Company. He stated that he had worked for the company approximately five years, but was laid off on August 4, 1978.

Individual "A"'s concerns, as delineated in his letter of August 18, 1978, were discussed in general terms.

4. Interview of Individual "A"

On September 20, 1978, Individual "A" was interviewed by RIII personnel. Individual "A" indicated that the order for cable pans for the Zimmer plant was the first contract for which Husky had to meet nuclear requirements. He stated that these requirements included a special design requiring wrap-around splice plates, and pan side rails made from material with a minimum tensile strength of 35,000 pounds per square inch.

Individual "A" stated that for the Zimmer project, Husky procured steel from the Central Steel Company or J&L steel, purchasing commercial quality steel, and then testing the steel to see that it met the minimum strength requirements. The steel supplier would take a "master" coil, and slit it into six (on the average) production coils for Husky usage. Samples would be taken from the steel when it arrived at Husky, and the shipment would be placed on hold until the results of the tests were received. Individual "A" indicated that these material tests had been performed by the Powell Valve Company test lab in Cincinnati (The Union Testing and Research Laboratory).

Individual "A" stated that it was found that commercial quality steel varied in strength, and that one shipment was found to be of low tensile strength steel. He stated that Individual "B" made the decision to use this low strength steel in "fittings" or curved sections of cable pan, where strength is not crucial, and that a memo to this effect had been written. Individual "A" stated that on approximately February 10, 1976, he found that the low tensile strength material mentioned had not been properly segregated, and had inadvertently been made into straight sections of electrical cable pan.

Individual "A" indicated that he had informed Individual "D" that the low strength material had been used to manufacture cable pan. and produced a handwritten note (see Exhibit VI) which he indicated had been given to Individual "D". He also indicated that he had informed Individuals "B", "C", "G", and "I" that this had happened. He stated that this one-time occurrence had been the subject of discussion among Husky personnel for several years.

Individual "A" stated that the manual welds used to manufacture fittings were poorly done, and that the welder certification program was a "farce". He stated that welders who were to work on cable pans for the Zimmer contract were required to pass a qualification test as required by Section IX of the ASME Code. When initially tested by Gladstone Laboratories, he said, the welders could not pass the qualification test, and generally succeeded in passing the test after multiple attempts. Individual "A" stated that the welders did not perform their production welding any differently after passing the welder certification test.

Individual "A" indicated that several knowledgable people had been critical of the welding performed by Husky welders, including Individual "J" (whose report is attached as part of Exhibit I). Individual "A" indicated that Individual "J" would have no part of training Husky welders unless they attended the full training course that his welding school provided.

RIII personnel advised Individual "A" that the technical specification for the cable pans to be used in the Zimmer plant (specification H-2199, Division 2, Section 202.1) required that the materials be of a minimum yield strength of 30,000 pounds per square inch (yield strength is usually less than tensile strength). The comment regarding 35,000 lb/square inch tensile strength is incorrect. Individual "A" was also advised that the specification would not allow the use of low strength material for cable pan fittings.

5. Investigation at Husky Products

During September 20-22, 1978, RIII personnel visited the Husky Products facility in Florence, Kentucky.

Discussion with Husky personnel indicated that, due to the special design of cable pans for the Zimmer contract, steel rolls utilized in their construction were of unique size (7.7 and 5.7 inch wide rolls) not used for any other contract. As such, it was indicated, the 14 and 22 gauge material for the Zimmer contract could be easily traced through the receipt, testing, and manufacturing process, and such documentation could be identified by Husky Order No. 3995.

RIII personnel toured the Husky facility, observed the fabrication of sections of electrical cable pan, and inspected equipment utilized in the forming and welding processes. Storage and receipt inspection procedures were also reviewed.

Husky personnel indicated that they had no knowledge of any low strength steel being received or utilized by Husky for any contract. It was indicated that during 1974-1976, Husky purchased commercial quality steel, and then took samples from the material, which would be placed on hold until testing indicated that it met the contract requirements. Husky personnel stated that they had experienced some problems with low strength aluminum, and some steel had been returned to the vendor for roll flaws, but no 14 or 22 gauge steel had been found to be of low yield strength.

Husky personnel stated that no decision had been made to use low strength material on cable pan fittings on the Zimmer contract or any other contracts.

Husky personnel did indicate that half of one shipment of coiled steel had been returned to the vendor for coil defects known as "coil breaks". They stated that the coil breaks do not affect the strength of the material, but cause problems during manufacture, and detract from the visual appearance of finished products. Two Husky officials noted that it was possible that it was decided to use rolls with coil breaks for fittings, as the coil breaks could be cut out during the manufacturing process. However, none of the individuals interviewed recalled such a decision.

A review of the Zimmer contract file indicated that part of a shipment of 14 gauge steel for the Zimmer contract had been returned to the vendor for having "'ad waves" (improper winding of the steel which would cause manufacturing problems) Additionally, a steel shipment received on February 10, 1976, was found to be .002 inches too thick, and was accepted.

RIII personnel reviewed documents relative to receipt of materials, shipment of materials to the Zimmer site, production records covering Zimmer cable pans manufactured during 1976, returned shipments of roll steel, correspondence with steel vendors concerning coil breaks, discrepancy reports, and internal memoranda. None of the documents reviewed indicated that unacceptable materials had been utilized by Husky.

RIII personnel also reviewed welding procedure and welder qualification documentation.

It was found that manual welding for the Zimmer plant was performed using a Metal Inert Gas (MIG) procedure, and steel filler wire, using semi-automatic equipment. On this type of equipment, welding parameters are set on the welding machine, and the welder positions the welding gun and pulls a trigger. The equipment then operates automatically, controlling shielding gas flow, electric current, filler wire feed rate, and time of the weld. Manual welding was performed on "f ttings" (curved sections of cable pan) only, with the bulk of cable pan being straight sections welded by automatic resistance welding equipment.

Welding records reviewed met the requirements of the American Society of Mechanical Engineers Boiler and Pressure Vessel Code. Section IX (ASME Section IX), which was imposed on Husky by its inclusion in their Quality Assurance Manual.

ASME Section IX prescribes methods and procedures to be followed in welding procedure and welder qualification. Individaul "A"'s comment that the Husky welders did not qualify in the same manner as they produced welds is correct, but is in conformance with ASME requirements. Qualification was performed to a butt weld procedure, per the requirements of ASME Section IX, and production welds were spot welds.

6. Visit to Union Testing and Research Laboratory

On September 22, 1978, RIII representatives visited the Union Testing and Research Laboratory, a division of the William Powell Company

Powell personnel indicated that they had performed material tests for Husky during the years 1974-1976, and followed the procedure of calling the company and informing them of the test results from handwritten forms, then typing the test forms and sending a copy to Husky for their records.

RIII personnel reviewed Powell files for Husky covering 1974-1976. All test reports reviewed indicated 14 and 22 gauge steel was tested and found to be in excess of 30,000 pounds per square inch yield strength. Typical values for such material ranged from 35,000 to 40,000 pounds per square inch. Records for the years 1975 and 1976 indicated one test of 16 gauge steel was tested and to have 29,400 lbs/square inch yield, and one sample of aluminum was tested and found to have 15,650 lbs/square inch yield strength.

Powell personnnel stated that they did not recall any 14 or 22 gauge steel which they had tested which did not exceed 30,000 lbs/square inch yield strength. They indicated that this was typical of 14 and 22 gauge steel, and that steel vendors have no difficulty producing such material.

7. Contact with Individual "A"

Individual "A" was contacted by telephone by the RIII investigator on September 21, 1978, and asked to provide additional detail regarding his alleged discovery of the use of low strength material. Individual "A" stated that he had been aware of the existence of low strength material through receipt of inspection reports which had been routed through his office. He stated that some of the material was marked "return to vender", and some of it was marked "use for fittings only - segregate". He indicated that he was in the Husky material storage area on February 10, 1976, and asked a worker where the Zimmer low strength material was stored. The worker did not know what he was talking about, Individual "A" said, and he asked the worker's supervisor the same question, with similar results. Individual "A" stated that he then advised Individual "B" of the occurrence, and wrote the note attached as Exhibit VI to Individual "D". He indicated that Individual "D" went to look into the matter, and later returned the note with a verbal comment to "forget it".

Individual "A" commented that he had not actually read the written specification for the Zimmer cable pans, but he understood that the specification required material with a minimum tensile strength of 35,000 lbs. per square inch. He was again advised of the actual specification requirements.

8. Contact with Individual "J"

Individual "J", of the Technicron School of Welding, was contacted by the RIII investigator on September 7, 1978.

Individual "J" indicated that his school utilized Gladstone Laboratories (Gladstone) to certify his welders, and that when Husky welders had difficulty passing weld certification tests, Gladstone had recommended him to Husky.

Individual "J" stated that he did not remember all of the details of his review of Husky, but he recalled that most but not all of their problems involved the welding of aluminum. He indicated that he had fewer concerns relative to steel welding. He stated that he had looked at Husky from the viewpoint of a consultant, with a view towards training their welders at his school.

Individual "J" indicated that he had not refused to train welders from Husky, but he had wanted the welders to take the entire training course which his school offerred. He stated that Husky management only wanted their welders to be schooled in the two weld procedures (MIG and TIG) which they utilized. Individual "J" indicated that he did have some reservations that the older Husky welders would not benefit from training at his school.

During the discussion Individual "J" indicated that he was not aware that his report had been attached to Individual "A"'s letter. He indicated that Individual "A" had not contacted him, and that he had not been in contact with the Husky company since the date of his report.

9. Interview with Individual "A" on September 27, 1978

Individual "A" was interviewed on September 27, 1978, and discussions were held on the progress of the NRC investigation.

Individual "A" was advised that no evidence of low strength material had been developed, and was requested to provide any additional information which would aid in the investigation. Individual "A" indicated that in early 1975 prior to the shipment of low strength steel which was inadvertently used for cable pans, another shipment had been tested, found to be of low strength material, and was properly returned to the vendor. He stated that he believed that the

A XII 101

1028 260

shipment which was improperly utilized was a small shipment, possibly of six coils of steel, which was delivered during the months of December 1975 or January 1976.

Individual "A" indicated that he had also recalled an occurrence in November 1975, when Husky sent Zimmer material to Modern Welding and Sheet Metal (Modern), a specialty welding firm which did not have welders qualified to ASME Section IX at the time. Individual "A" stated that this was done because the Husky plant was on strike, and the company felt that they had to meet their contract to supply the cable pans. He stated that the order comprised over 100 pieces of equipment, of three-piece construction. He indicated his understanding that the welders for Modern were not qualified to ASME Section IX until sometime in 1976.

Individual "A" provided the RIII investigators with the name and telephone number of a former Husky employee who, it was indicated, might have some recollection of the alleged use of low strength material during manufacture of equipment for the Zimmer plant.

10. Contact with Individual "D"

Individual "D" was contacted by the RIII investigator on September 29, 1978.

Individual "D" was questioned as to his knowledge of the use of low strength materials in the fabrication of cable pans for the Zimmer plant. He stated that he did not recall the use of any low strength material on any of the Husky nuclear contracts. He indicated that he did not believe that anyone at Husky would knowingly allow such an occurrence, especially those in the Quality Control department.

The scenario of the discovery of the use of the low strength material as described by Individual "A" was discussed with Individual "D", and the note allegedly sent to him was read. Individual "D" stated that he had no recollection of any such note, and indicated that it would be unusual for him to return such a note without some kind of written comment, as he disliked verbal communications.

Individual "D" recalled occurrences where shipments of steel were found to have various problems such as excessive oil, roll problems such as ripples or twists, or were rejected because of steel thickness variations. He indicated that he also recalled the incidence of some low strength aluminum, and steel pre-galvanized with an aluminum-zinc coating which was banned from inclusion in the Zimmer equipment.

He stated that the aluminum-zinc coated material (Galvalume) was to be made into cable pan covers, but Husky personnel recognized that the 1.6% aluminum content of the coating was undesirable due to its large surface area, and a program was set up to insure that no Glavalume pan covers were shipped to the Zimmer site. Individual "D" indicated that on at least one occasion, covers were inadvertently fabricated of this material, were identified, and had to be re-fabricated.

11. Visit to Husky Products during September 27-29, 1978

RIII personnel visited the Husky facility during September 27-29, 1978. During this visit, documentation related to welder qualification testing, production records, material tests, deficiency reports, internal memoranda of the Industrial Engineering section, and weld procedure qualifications were reviewed. Interviews were held with Husky personnel, and three signed statements were obtained. (See exhibits II, III and IV).

None of the documents reviewed, and none of the statements received during interviews indicated that low strength materials had been utilized during manufacture of the Zimmer plant cable pans.

Welding certification was reviewed as pertaining to welding procedure and welder qualification to Section IX of the ASME Boiler and Pressure Vessel Code. Welder qualification records and welder qualification test pieces (stored at Husky) were considered acceptable. Records indicated that welders had made several qualification attempts in many cases. This is acceptable under ASME Section IX.

During document reviews at Husky, it was found that the welding procedure for manual welding on Zimmer equipment had been qualified using carbon dioxide shielding gas and .035 inch diameter filler material, but a mixture of shielding gas and .045 inch diameter filler material had been utilized for the period of November 14 - December 3, 1974. The is in nonconformance with ASME Section IX, which required requalification of the welding procedure when these variables were changed.

12. Interview with Individual "E"

Individual "E", Husky Purchasing Agent, was interviewed by RIII personnel on September 28, 1978, at Husky.

Individual "E" stated that to his knowledge, Husky had not received nor returned any steel which did not meet the appropriate yield strength requirements. He stated that since the steel that was purchased during the manufacture of the Zimmer equipment was purchased to commercial steel specifications, and then tested, it would not have been returned if it did not meet the minimum strength requirements. No minimum strength requirements are imposed on the steel vendor when commercial grade steel is purchased.

Individual "E" stated that flat stock steel was purchased and controlled in the same fashion as roll stock i.e., to commmerical grade requirements, and then tested to insure that it met the minimum strength requirements.

Individual "E" stated that the Central Steel Company had supplied all of the 14 gauge steel utilized for the Zimmer cable pans.

13. Visit to Modern Welding and Sheet Metal

On Septemeber 29, 1978, RIII representatives visited the Modern Welding and Sheet Metal Company.

Discussions were held with Individual "F", one of the managers for the firm. Individual "F" indicated that the majority of the work that his firm does for Husky is specialty welding of seperators, junction boxes, cable bus, and aluminum welding. He indicated that to the best of his knowledge, his firm had not performed any welding on cable pans for Husky at any time.

Individual "F" was requested to review his files for work performed for Husky for the years 1975 and 1976, with attention to any work on electrical cable pans. Individual "F" stated that he could not find any orders concerning electrical cable pans, and the Husky identification number (3995) for the Zimmer project was not found in his review of his files.

On October 12, 1978, the RIII investigator contacted ~~Individual~~ "F" and requested that he again review his files, and provide the NRC with information as to any products manufactured for Husky during November, 1975. Individual "F" provided this information, which indicated that tap boxes and cable separators had been fabricated by his firm for Husky, but no work had been done on cable pans, and none of the Husky tags applied to the work had referenced the Zimmer identification number.

14. Contact with Individual "G"

Individual "A" had advised RIII personnel that Individual "G" might have information concerning the use of low strength material in the Zimmer equipment. This individual was contacted by the RIII investigator on October 5, 1978.

Individual "G" stated that he had been in the hospital during time period of the alleged use of low strength materials. He indicated that he had no knowledge of such an occurrence, and that he had not heard anyone at the Husky plant discuss such an occurrence while he was employed there (his employment terminated in February, 1978).

15. Contact with Individual "H"

Individual "H", an employee of Hobart Welding who had acted as a consultant to Husky on welding and welding qualification, was contacted on September 29, 1978.

Individual "H", indicated that his first contact with Husky was approximately five years ago, and that Individual "I" had been trained in the Hobart school. He stated that Husky had long been involved in welder qualification and in upgrading their welding. Individual "H" advised that five or six years ago, the Husky welders did have some welding problems, and that they did acceptable welding on the production line, but made poor qualification test pieces.

Individual "H" stated that he believed that Husky had a good program for welding qualification testing, and had used the program to "weed out" the poorer welders.

16. Discussions with Individual "A"

Several telephone discussions were held with Individual "A" concerning the findings of the investigation. Individual "A" expressed dissatisfaction with the findings of the investigation, and provided additional allegations concerning Husky.

Individual "A" stated that the Husky welders had not qualified on both the vertical and horizontal welding positions, and had performed vertical welding during cable pan manufacture.

Individual "A" indicated that he felt that the Husky welds had been required to be of pressure vessel quality. He was advised that the specification had not required welds of pressure vessel quality. Welds of pressure vessel quality require non-destructive examination such as magnetic particle, radiographic, liquid penetrant, or ultrasonic testing, as a verification of their quality, and no such inspections were required.

Individual "A" also indicated that he felt that the company had not met all of the requirements of Code of Federal Regulations, Title 10, Part 50, Appendix B, Quality Assurance Criteria for Nuclear Power Plants (a copy of this regulation had been provided to him by RIII personnel). RIII personnel explained that all of the requirements of this regulation were imposed on utilities, but the provisions of the Husky Quality Assurance Manual were the requirement imposed on Husky after approval of the manual by utility representatives.

17. Contact with Husky Personnel

Telephone contacts with Husky personnel indicated that some cable channels had been fabricated by Modern, with the order being processed during November, 1975, and completed in later months. Husky personnel indicated that this material was for another nuclear power plant, and was fabricated prior to the particluar utility's imposition of a requirement for work done by welders qualified to Section IX of the ASME Code.

Husky personnel also indicated that virtually all of their welding was done in the horizontal welding position, and they did not recall any pieces for the Zimmer contract which necessitated vertical welding.

A review of Husky welder certifications for the horizontal and vertical positions indicated that one Husky welder was not qualified in the MIG procedure vertical (3G) welding position. Welders previously indicated by Husky personnel as having produced the majority of the Manual MIG welding for the Zimmer project (at work center 35) were recorded as having been qualified in both horizontal (2G) and vertical (3G) positions. Qualification to the "3G" vertical position also qualified a welder to perform flat (1G) welding per ASME Section IX.

18. Contacts with Husky Personnel

Telephone discussions with Husky personnel on October 24, and 29, 1978, provided additional information on low strength aluminum materials.

Husky personnel indicated that aluminum materials were ordered to 6063T6 requirements, which include a minimum 30,000 lbs. per square inch yield strength (as shown by mill certificates). They stated that a shipment of the material was thought to be of low strength, and sample test pieces sent to their test lab confirmed that the material was below requirements. Husky personnel indicated that as a result of this, the entire lot of material was returned to the vendor, and the balance of their orders with the vendor were cancelled.

Husky personnel stated that the rejections of this material occurred in October and November 1977, with the original discrepancy report being generated in September of 1977. They stated that in January 1978, representatives of the vendor visited the Husky facility and discussed the problem.

Section II

Prepared by T. E. Vandel
Reviewed by D. W. Hayes, Chief
Projects Section

1. Site Review Activities

The following Zimmer site activities were performed by the inspector relative to the allegations regarding inadequate material and welding of Husky Products, Inc. (Husky) cable trays, pans and fittings:

a. A review was conducted of the licensee source evaluation, surveillance and auditing activities performed regarding Husky. It was established that the licensee program for vendor evaluation and auditing had been accomplished in that the Husky Quality Assurance program and Welding procedures had been reviewed and approved by licensee representatives. Additionally, an audit by the licensee was performed of the implementation of the program at the Husky plant prior to start of fabrication.

   In response to questioning, the inspector was informed that no source inspection of material was done prior to shipment since the material was readily amenable to inspection upon receipt at the site. It was added that the material was considered so standard and unsophisticated as to not warrant shop inspection.

b. In review of the cable trays, pans and fittings on site, it was established that essentially all of the material has been installed and indeed have been filled with cables. During visual inspection of the installed trays no faulty or inadequate trays were identified. In discussions with the licensee representatives regarding the difficulty of visual inspection of welds now covered by galvanizing, it was concluded that testing of selected random samples of material would be a more meaningful test. Therefore, the following list of samples, randomly selected by the licensee representative and the NRC inspectors, was picked for testing by either tension pull tests (yield strength) or by weld tear testing or both.

Type Components	P.O. Number	Stock Number	Tests
Straight tray 18"	7070-27655	55M1-18-144	Two yields, one tear
Straight tray 24"	7070-27303	55M1-24-144	One yield, two tear

Fitting	7070-27223	55N1-12-H30°	One yield, one tear
Straight tray 24" (from control room)	Route #1276K (P.O. unknown)	55M1-24-144	One yield, one tear
Fitting	7070-27655	55N1-24-VI90°-12	One yield one tear
Fitting	7070-28009	55N1-24-VI30°-12	One yield*

*No tear test was considered necessary since the fitting had inadvertently been torn during handling and the results of those weld tears showed adequate welding.

It was further agreed that the yield strength testing would be done by an independent testing laboratory in accordance to ASTM standard E-8 Tension Testing of Metallic Materials and that the minimum strength acceptance criteria will be the S&L specification H-2199 requirement of paragraph 202.1; i.e., yield strength to be a minimum of 30,000 psi. In addition, the weld tear tests would also be done by an independent facility and that the acceptability of the welds would be judged as outlined in AWS standard C-1.1.

2. Witness of Testing

The inspector witnessed the following testing at independent laboratories of the samples previously selected at the site.

a. Yield strength testing was conducted on September 28, 1978, at Metcut Research Associates facility. The inspector reviewed the qualifications of the operator, the calibration and adequacy of the testing machine and the QA program standards of the facility and considered them to be acceptable for the test. It was further learned that the tensile specimens had been prepared in accordance with the ASTM E-8. The results of the tests are as follows.

Metcut Number	Site Sample Number	Yield Strength Pounds per Square inch	Ultimate Strength	Percent Elongation
T-2 1162	1276K	40,700	48,100	34.9
T-2 1163	55N112-H30°	42,600	47,800	30.7
T-2 1164	55N24VI90-12	43,100	48,900	28.3
T-2 1165	55N1-24VI30-12	42,400	47,600	32.6
T-2 1166	55M1-24-144	42,100	44,700	33.0
T-2 1167	55M1-18-144(No. 1)	42,200	44,900	30.4
T-2 1168	55M1-18-144(No. 2)	41,400	44,800	33.7

As can be noted from the table above, the yield strength values were well above the minimum yield value of 30,000 psi and therefore all test samples were deemed acceptable.

b. Also on September 28, 1978, the weld tear tests of the resistance spot welds, were witnessed by the NRC inspector at the F&S Machine Company, located in Moscow, Ohio.

A test rig had been assembled whereby the test assembly was anchored to the floor and by use of a fork lift truck the assembly was pulled apart at the welds (side panels to tray bottom welds). The test method performed adequately with the following results established.

Site Sample Number	Number of Welds in Tear Test	Results of Testing
55M1-24-144 (No. 1)	five	Acceptable welds
55M1-24-144 (No. 2)	three	Acceptable welds
55M1-24-144 (Note 1)	three	Acceptable welds
MK 1276K 55M1-24-144	three	Acceptable welds
55M1-12-H30° fitting	seven	One weld had a reduced spot section, see Note 2
55M1-24VI90°-12 fitting	eight	Two welds had a reduced spot, see Note 2
55M1-18-144	three	Acceptable welds

Note 1: An additional test assembly, available for test in addition to the two planned to be tested, was also tested for a total of seven test assemblies tested.

Note 2: The reduced spot section welds were subsequently measured and found to be adequate per the minimum size specified in AWS C-1.1. A total of seven test assemblies were tested with a total of 32 welds being tested. All welds were determined to be adequate with three spots being evaluated as being acceptable to AWS C-1.1.

Section III

Prepared by H. M. Wescott
Reviewed by D. W. Hayes, Chief
Projects Branch

1. Review of Welding Requirements and Observation of Installed Cable Tray

The inspector reviewed selected documents and made observations of safety related cable tray and fittings, as follows:

a. Review of Sargent and Lundy specification H-2199, dated March 16, 1973, Revised July 17, 1973, titled, "Specification for Cable Pans".

b. Review of NEMA Standard VE1-1971 used in conjunction with the specification.

c. Review of the Husky Products, Inc. Quality Control Manual, Section IX "Control of Special Processes", issue date December 18, 1974, revised January 15, 1975.

d. Review of Wm H. Zimmer Unit 1 "Documentation Check Lists" (Form QAS-106).

e. Review of certificates of compliance.

f. Review of Galvanizing Inspection reports.

g. Review of Wm H. Zimmer receiving inspection plans (KEI Form No. QA-8).

h. Observations made of cable tray installed and in storage area.

i. Participated in selection of randomly selected cable tray and fittings to be tested for minimum yield strength and weldment strength tests.

2. Review of Welding Procedures, Qualifications and Observations at Burndy/Husky

The inspector reviewed welding procedure specifications, procedure qualifications records, welder performance qualifications, and selected documents pertaining to safety related cable tray and fittings, as follows:

a. Review of all welder qualifications.

b. Review of Welding Procedure specification QAP-107, Welding Procedure No. 2 "Manual Gas Metal Arc Welding Process," effective date October 18, 1974, Revision No. 01.

c. Review of QAP 104 "Procedure for Inspection of Resistance Spot Welding", effective date August 18, 1974, Revision No. 01.

d. Review of inter-office correspondence concerning welding, that indicated QAP-107 should be requalified to reflect changes in essential variables.

e. Discussion with management and shop personnel.

f. Observations made in the shop area of fabrication in progress.

g. Review of in process inspection records.

Review of a Burndy/Husky memorandum from the Husky welding engineer dated November 14, 1974, Subject "Welder Performance Qualification" indicated that a 75% argon and 25% carbon dioxide shielding gas mixture and .045 filler material was substituted for the welding grade carbon dioxide shielding gas and 0.35 filler material that was specified in QAP-107 "Manual Gas Metal Arc Welding Process", dated October 18, 1974, Revision No. 01. The memo further stated that, "The ASME Section says that if this occurs, the procedure must be requalified along with the performance tests. (Section QW 281.2, QW 281.3 and QW 281.4)".

An Inter-Office letter dated December 3, 1974, stated that the argon/carbon dioxide gas mixture would be used until the supply was exhausted at which time the welding grade carbon dioxide would be used.

The argon/carbon dioxide shielding gas mixture was used for approximately four weeks with no requalification of the welding procedure specification and welders.

Husky management personnel indicated that QAP-107 would be requalified using the 75% argon and 25% gas mixture using the .045 filler material.

This is considered to be an item of noncompliance to 10 CFR, Part 50, Appendix B, Criterion IX. (50-258/78-21-02)

Subsequent to the investigation telephone contacts with Husky personnel by the investigation specialist established that steel TIG welding had been performed on cable tray prior to qualification of the welding procedure specification by two welders that had not qualified for the process. Husky personnel were requested to review the qualification records of the personnel who had performed the welding and inform RIII of the results of their review.

Husky personnel informed RIII of the review by telephone, and followed with written notification dated November 10, 1978. The Husky review indicated that the two welders had performed TIG welding on equipment for the Zimmer plant prior to the welding procedure qualification for the TIG process.

The steel TIG welding procedure was qualified on August 26, 1975, by one of the two welders. The second welder was qualified to the procedure on March 10, 1976. Both welders had made several steel TIG welds prior to being qualified.

These conditions were contrary to 10 CFR 50, Appendix B, Criterion IX of the ASME Code. (358/78-21-01)

Exit Interview

The inspectors and the Chief, Reactor Construction and Engineering Support Branch, met with licensee representatives noted in Details, Section 1, under Personnel Contacted, at CG&E Co. on September 22, 1978. The inspectors summarized the scope and findings of the investigation and the licensee acknowledges the findings.

Attachments: Exhibits
I through VII

A XII - 114

1028 273

August 18, 1978

Public Interest Research Group
2000 P Street N. W.
Washington, D. C. 20036

Attention: Mr. John Abbotts

Dear Mr. Abbotts:

I am writing this as a former employee of Husky Products Inc. of Florence, Kentucky to report serious and deliberate non-conformance to 10 CFR 50 Nuclear Requirements and Engineering Specifications based on the above requirements. To make it even worse they send out notarized Certificates of Compliance with the full knowledge they are false.

In May of this year I had occasion to visit the Zimmer Nuclear Containment area and to see the various control areas and in particular to see Husky cable trays in position and many filled with the cables.

Since this visit I have been disturbed by two aspects of Husky's non-conformance, particularly as they relate to the safe operation of this plant after completion of construction.

These two important aspects are as follows:

1. Use of inferior and weak material completely out of specifications.

2. Trays welded by incompetent welders with every type weld defect present in every tray assembly.

The following illustrates these two aspects in more detail. They are related to the Zimmer job specifically which was the original job with the 10 CFR 50 requirements. On this job flagrant and serious non-conformance occurred and with this as a pattern it has occurred on all subsequent jobs.

MATERIAL:

All tray is designed with a load capacity which includes a safety factor. The tensile strength of the side rails largely determines this capacity. On the Zimmer job the tensile strength of the side rail material was to be in excess of 35,000 pounds. Husky received and tested material as low as 18,000 pounds and a considerable amount in the range of 20 to 23,000 pounds. Some was rejected, some accepted on the basis it would be used for fittings where strength is not as critical.

POOR ORIGINAL

A-XII-115

1028 274

Instead the material was not kept separate and thus many very weak side-rails were made up into long straight assemblies. After finding out that common mill steel varied so widely in tensile strength no more testing was done so that they could remain "unaware" of this condition. Incidentally some testing of T-6 aluminum was also performed and a wide range of tensile strength was also found. This was also ignored as above. What this adds up to is that Husky has built tray that will not carry the rated load even with safety factor included.

WELDING:

The Zimmer job was the first job requiring the use of Certified welders in order to insure good welds. Husky contracted with Gladstone Laboratories of Cincinnati to set up a welder certification program. They did this and then tested all the welders. Without exception they failed the tests miserably. Husky then called in various welding Engineers and Mr. Ind "J" of Tech-nicron School of Welding in Cincinnati who submitted a written report of findings. A copy of his report is attached. In general all the weld Engineers concurred with Mr. Ind. "J" report. Mr. Ind. "J" was asked if he could or would train the welders. He refused, stating that it is very difficult, if not nearly impossible to untrain people first, then try to retrain, than it is to start fresh with a person having no prior welding knowledge or experience.

Husky then proceeded to work on their own in crash programs in which the weldors finally welded one piece which would pass a bend test. This welder then became "Certified" by Husky. However., what is critically important is that nothing occurred to the quality of the production welds! In fact it remains to date in the same sad state as Mr. Ind. "J"s findings dated October 30, 1974. Just a few weeks ago one welder was "tested" over 60 times before he finally made a test piece which was only marginally acceptable. Now he is a Husky "Certified" welder!

Starting in July and continuing this month a new type of non-conformance is presently in process on the Clinton job. Fittings are being Mig spot welded contrary to specific Engineering requirements. In addition Aluminum Bronze filler rod is being used with full knowledge that aluminum is not permitted in the containment area. Even worse the position of the spot is in such a manner the weld is less than 35% effective!

Substantiation of all these charges can be accomplished thru examination of Husky documents in relation to Material and to the Welders by the records, visual examination of the welds and by retesting the so called "Certified" welders by a competent Welding Engineer. Visual inspection of the Clinton fittings will substantiate the charges outlined.

A-XII-116

1028-275

What disturbs me even more than the actual incidents described is the fact so many top management people see nothing wrong in all these actions. So little real concern is shown to producing a truly quality product within the specifications. This should become even more particularly so when nuclear safety is directly involved.

Yours truly,

Individual "A"

Distribution as follows:

Engineering Companies that may or may not be concerned.

Ebasco
United Engineers and Constructors
Bechtol Corp.
Brown & Root
~~[illegible]~~
Sargent & Lundy
Stone & Webster
Black & Veatch
This may not be complete, however to the best of my knowledge it is.

Government Agencies:

Nuclear Regulatory Commission
Congressional Joint Atomic Energy Committee

Private Group:

Public Interest Research Group

Exhibit I
Page 3 of 7

A XII - 117

1028 276-

Repo of the Findings at
Husk roducts Incorporated
or tober 30, 1974

Submitted by: Technichron Inc., School of Welding

It was generally found that the reason your company has had difficulty in certifying your weldors is due to the fact that while some of your men are qualified weldors, they suffer from the ills of an employee that is offering an incentive program.

In order for an employee of your company to meet his required production level, plus benefit by the incentive program it was found that their welding machines were set at maximum output allowable, which is just below the point of blowing holes in the parent metal. This condition creates improper welding methods, and instead of establishing good welding, you have a situation of blasting the metal together. These extreme amperage settings also make it necessary to use higher gas flow in order to control the arc. This has to be extremely costly to your company.

ause of the conditions that exist (welding machine settings and as flows) it was observed that improper welding is a common occurance t Husky Products. The welds are not structurally sound.

Aluminum Welding:

All the welds have craters and it was observed that most of these craters show the common condition known as "crater cracking". It was further observed that there were many welds that had both cracking conditions in the weld as well as the crater. These conditions are primarily caused by the extremely high amperages and gas coverage. Your weldors are running extremely hot welds due to speed and thus you have rapid cooling conditions and cracking. The high gas flows (while costly) also causes rapid cooling and thus cracking.

Generally it was observed that the weldors in your aluminum welding ar s had good welding techniques however lack knowledge in setting up the proper welding conditions before welding.

POOR ORIGINAL

Exhibit I
Page 4 of 7

A XII - 118

These men lacked the following knowledge:

1. Setting the welding machine
2. Setting the proper gas flow
3. Balling the tungsten rather then pointing it
4. Controlling the weld to prevent craters
5. Cleaning the parent metal before welding

Steel Welding:

Four men were observed in the steel welding areas. One man had the knowledge of proper machine and gas flow settings however he lacked the welding techniques. This man was one of your oldest weldors. The other three men had very little knowledge about proper settings and one of the three lacked the proper welding techniques. This man was your oldest employee in your welding department. Again it was apparent that all conditions existed to turn out maximum production.

As long as you have these conditions you will find that certifying welders is going to be extremely difficult. When observing several of the test coupons run by your weldors it was found that the following conditions existed:

1. Crystallizations of the weld
2. Porosity
3. Penetration that exceeded 100%
4. Undercut
5. Weaking of parent metal in the heat effected zone

All the conditions are created directly by running too high of amperages, too high of gas flows, and dirty metal.

POOR ORIGINAL

Exhibit I
Page 5 of 7

A-XII-119

Other Observations:

1. The using of fans in the welding areas is common practice. This condition causes the gas shield to be blown away, thus causing porosity in the welds. This is another reason for the high gas flow pressures which is costly since larger volumes of gas are used then necessary.

2. It was noted that Argon/CO^2 mix was being used in your M.I.G. welding operations on steel. This again is costly because CO^2 would be adequate for your operation. Straight CO^2 costs about 1/6 of what 75/25 Argon/CO^2 mix costs.

3. Many of your employees do not use eye protection or face protection. I'm certain you must have frequently absenteeism due to eye flash injuries.

4. No use of safety glasses in the entire plant. Welders must wear safety glasses under their welding hood. (An OSHA Standard).

5. The plant is not in compliance with OSHA Standards. This could cause extreme hardship in the future especially if you have a severe injury of one of your employees.

Suggetion:

Husky Products Inc., should consider a training program for thoses individuals employed in their welding department. This program should emphasize welding methods as well as welding techniques.

Any success arising from this training program is highly questionable, since proper welding methods and techniques would cut production. The present attitude in your welding department is quantity not quality. Sound certified quality welds will definitely reduce quantity, however the savings in cost of materials will most likely improve or equalize profits.

A-XII-120

I am submitting this report with the intension of creating many constructive suggestions and have no intension to sound like I am being critical. You realized you had some concerns or you would have never contacted Technichron in the first place. Therefore, I sincerely hope that I have been of service to your company and that we may serve you again in the future.

Thank you.

Respectively Submitted

Individual "J"

Technichron School of Welding

Exhibit I

Page 7 of 7

A - XII - 121

I, Individual "B" , make the following written voluntary statement to James Foster who has identified himself to me as an investigation specialist of the Nuclear Regulatory Commission. I understand that I do not have to make a statement and that any statement I do make may be used in legal proceedings.

I have no knowledge of low yield strength steel, below 30,000 lbs. per square inch, having been present at the Husky Products Plant nor of such material having been utilized in the production of cable pans for the Zimmer Nuclear Power Plant.

I have read the preceding statement consisting of one page and made corrections where necessary. It is a true representation.

Signed Individual "B"

Date 9-28-78

Witness [signature] 9-28-

Witness James E. Foster 9/28/78

Exhibit II

A XII - 122

I, Individual "C" , make the following written voluntary statement to James Foster who has identified himself to me as an investigation specialist of the Nuclear Regulatory Commission. I understand that I do not have to make a statement and that any statement I do make may be used in legal proceedings.

I have no knowledge of low yield strength steel, below 30,000 lbs. per square inch, having been present at the Husky Products Plant nor of such material having been utilized in the production of cable pans for the Zimmer Nuclear Power Plant.

I have read the preceding statement consisting of one page and made corrections where necessary. It is a true representation.

Signed Individual "C"

Date 9-28-78

Witness James E. Foster 9/28/78

Witness [illegible]

Exhibit III

A-XII-123

1028 282

I, Individual "I" make the following voluntary written statement to James E. Foster, who has identified himself to me as an Investigation Specialist of the Nuclear Regulatory Commission. I understand that I do not have to make a statement, and any statement that I do make may be used in legal proceedings. I am presently employed by Husky Products (as) an Industrial Engineer.

To the best of my knowledge, no low yield point material has ever been utilized in the manufacture of equipment for the Zimmer Nuclear Power Plant, Unit 1. I have been directly involved with the in-house welder certification program since its inception. This program has been properly conducted, and follows the provisions of ASME Section IX for welder certification. I did not object to my participation in this program, but had to become knowledgeable in welding before becoming centrally involved in the program. I feel that welder certification has been honestly conducted.

Welding procedures and welders have not been re-qualified when weld shield gas or gas mixtures have been changed. I pointed out to Individual "A" that this had not been done. After 3-4 weeks, Husky started using CO_2 gas strictly as the procedure calls for.

Exhibit IV

Page 1 of 2

A-XII-124

1028 293

I was aware that the Aluminum-Bronze MIG spot weld process had not been qualified as to process or welders. I felt that these qualifications were not necessary, as the process is similar to resistance welding in that it is semi-automatic. The welding parameters are set, and the welder only aims the welding gun.

I have read this voluntary statement, consisting of two (2) pages, and made corrections where necessary. It is a true representation.

Witness: James E. Foster 9/28/78 Signed Individual "I"

Harvey M. Wescott 9/28/78 Date 9/28/78

Exhibit IV

Page 2 of 2

A-XII-125

Report No. [illegible]

ile Testing of (7) ______ ~~Bet~~ Sheet Specimens ~~Manufactured~~ to

~~Me...t Drawing~~ No. Supplied by CG&E

im. Gage Section: 0.08" x 0.50" x 2.0" lg

pe...ure: R.T.

in Rate through 0.2% Yield: 0.005 in. /in. /min.

l Rate thence to Failure: 0.05 in. /min.

RAI No.	Spec. No.	U.T.S. (ksi)	.02% Y.S. (ksi)	Yield Point (ksi)	Elong. (%)	~~R.A. (%)~~
21167	SSM 1-18-144	44.9		42.2	30.4	
21168	SSM 1-18-144	44.8		41.4	33.7	
21166	SSM 1-24-144	44.7		42.1	33.0	
21162	SSM 1-24-144-1276K	48.1		40.7	34.9	
21163	SSN 1-12-H30	47.8		42.6	30.7	
21164	SSN 1-24VI-90-12	48.9		43.1	28.3	
21165	SSN 1-24VI-30-12	47.6		42.4	32.6	

otes:

POOR ORIGINAL

Exhibit V

Page 1 of 1

LJF
Project Engineer

RED
Laboratory Technician

A-XII-126

1028 [illegible]

2-10-76

Low tensile Zimmer
stringers mixed
in stores and
now being used
for straights!

Returned with
verbal reply
to "forget it"
2-10-76

Text: Individual "D": 2-10-76

Low tensile Zimmer
stringers mixed
in stores and
now being used
for straights!
Individual "A"
Returned with
verbal reply
to "forget it"
2-10-76.

Exhibit VI

A-XII-127

September 22, 1978

Husky manufactures Cable Trays to NEMA Standards as per a catalog as a commercial item. It also manufactures modifications of Standard items and specials to a customers specifications.

Zimmers were special in 4 important ways as follows.:

1. They required special wrap around splice plates with different bolt holes to strengthen the joints where 2 trays come together.

2. They specified side rail material to have a minimum tensile strength of 35,000 pounds.

3. Welding was to be Mig Welded in accordance with ASME Section 9 and to be performed by certified welders.

4. All pertinent records relative to Quality are to be retained on long term retention basis.

In respect to the welding this meant that the welds were to have a quality level equal to that required for boilers and pressure vessels. These were to be top quality welds with good fusion, structurally sound and with minimum of defects. These were to be welded by qualified welders certified as such thru testing as called out in Section 9 of ASME.

Husky welders are competent to produce commercial type welds for an ordinary commercial product where defects and lack of fusion is acceptable. This is the type of weld done daily on our commercial work. We have Incentive Standards on this work and our weldors earn from 160 to 200% day in and day out. This is the type welding described in Mr. Ind. "J"s report.

Testing of our welders established their incompetence to produce quality welds at pressure vessel standards. Husky worked with the weldors until they made one good piece which would pass a bend test. The welder is then certified and then goes right back to production making commercial type welds for Incentive which is the only type weld ever made. Outside of making this one test piece they have no production experience in this type weld. Based on their difficulty in passing the test they need considerably more training, followed with actual production experience. before they can be competent to produce a high quality type of weld.

Quality welding would greatly increase the manufacturing cost, particularly if we changed all welding to become quality type. A second alternative would be to produce quality welds when required on nuclear work and commercial quality on all other work. Husky's decision was to certify the weldors but produce only the normal commercial type welds on all work. We would tell people we weld to Section 9 of ASME with certified weldors. This has never changed. We have never made any effort to produce pressure vessel quality welds.

Exhibit VII
Page 1 of 2

SEP 25 1978

A-XII-128

This was done on the Zimmer job and was incorporated into the Quality Control Manual that Husky Welding is in conformance with ASME section 9 and the welds are made by certified welders. This is misleading in that people think that they will get quality welds. Instead everybody gets commercial quality welds made by a welder who once made one quality weld piece. On this basis Husky has secured additional nuclear work.

The top Managers of Husky are on a bonus setup. Anything that adds cost subtracts from profit which in turn reduces their bonus. To produce quality would be very expensive and would reduce their bonus. It is entirely possible the decision not to produce the specified quality welds was based entirely on the cost required to do so. The reason given to me and my people was, "that it is completely unnecessary.

Individual "A"

September 22, 1978

Exhibit VII
Page 2 of 2

A - XII - 129

UNITED STATES
NUCLEAR REGULATORY COMMISSION
REGION III
799 ROOSEVELT ROAD
GLEN ELLYN, ILLINOIS 60137

JAN 0 3 1979

Docket No. 50-358

Cincinnati Gas and Electric
Company
ATTN: Mr. Earl A. Borgmann
Vice President
Engineering Services
and Electric Production
139 East 4th Street
Cincinnati, OH 45201

Gentlemen:

This refers to the inspection conducted by Mr. I. T. Yin of this office on November 16-17, 1978, of activities at the Wm. H. Zimmer Power Station authorized by NRC Construction Permit No. CPPR-88 and to the discussion of our findings with Mr. B. K. Culver and others of your staff at the conclusion of the inspection. This also refers to the investigation and inspection conducted by Mr. I. T. Yin on November 21, 1978, at Sargent and Lundy Engineers office in Chicago, relative to the document control provisions for pipe stress reports.

The enclosed copy of our inspection report identifies areas examined during the inspection. Within these areas, the inspection consisted of a selective examination of procedures and representative records, observations, and interviews with personnel.

During this inspection, certain of your activities appeared to be in noncompliance with NRC requirements, as described in the enclosed Append A.

This notice is sent to you pursuant to the provisions of Section 2.201 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations. Section 2.201 requires you to submit to this office within thirty days of your receipt of this notice a written statement or explanation in reply, including for each item of noncompliance: (1) corrective action taken and the results achieved; (2) corrective action to be taken to avoid further noncompliance; and (3) the date when full compliance will be achieved.

A-XII-130

1028 289

Cincinnati Gas and Electric Company JAN 03 1979

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter, the enclosures, and your response to this letter will be placed in the NRC's Public Document Room, except as follows. If the enclosures contain information that you or your contractors believe to be proprietary, you must apply in writing to this office, within twenty days of your receipt of this letter, to withhold such information from public disclosure. The application must include a full statement of the reasons for which the information is considered proprietary, and should be prepared so that proprietary information identified in the application is contained in an enclosure to the application.

We will gladly discuss any questions you have concerning this inspection.

Sincerely,

R. F. Heishman, Chief
Reactor Construction and
Engineering Support Branch

Enclosures:
1. Appendix A, Notice of Violation
2. IE Inspection Report No. 50-358/78-27

cc w/encls:
Mr. J. R. Schott, Plant Superintendent
Central Files
Reproduction Unit NRC 20
PDR
Local PDR
NSIC
TIC
U. Young Park, Power Siting Commission

POOR ORIGINAL

OFFICE	RIII	RIII	RIII	RIII	RIII
SURNAME	Yin/bk	Danielson	Heishman	Vandel	Hayes
DATE	12/27/78				

NRC Form 318B (R III) (1-78) NRCM 0240 U.S. GOVERNMENT PRINTING OFFICE: 1978—

H-XII.131

1028 290

Appendix A

NOTICE OF VIOLATION

Cincinnati Gas and
Electric Company

Docket No. 50-358

Based on the results of an NRC inspection conducted on November 16-17, and 21, 1978, it appears that certain of your activities were not conducted in full compliance with NRC requirements as noted below. The item is an infraction.

10 CFR 50, Appendix B, Criterion VIII requires, in part, that measures shall be established for identification and control of materials, parts and components. These measures shall assure that identification of the item is maintained by heat number, part number, serial number, or other appropriate means, either on the item or on records traceable to the item, as required throughout fabrication, erection, installation, and use of the item. Paragraph 17.1.8.2 of the FSAR states, "Essential materials, parts, and components. . . bear identification as to heat number, part number, . . . and complete traceability exists between the item and quality control records."

Contrary to the above, among the six hangers and restraints observed and the records reviewed by the inspector, two rigid supports welded to essential piping did not have heat numbers for material and weld filler metal nor did they have welders' identifications. One other rigid seismic restraint having the similar noncompliance was identified recently by the KEI QC inspector.

U.S. NUCLEAR REGULATORY COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT

REGION III

Report No. 50-358/78-27

Docket No. 50-358

License No. CPPR-88

Licensee: Cincinnati Gas and Electric Company
139 East 4th Street
Cincinnati, OH 45201

Facility Name: Wm. H. Zimmer Power Station, Unit 1

Inspection At: Wm. H. Zimmer 1 Site, Moscow, Ohio; and Sargent and Lundy Office, Chicago

Inspection Conducted: November 16-17 and 21, 1978

Inspector: /s/ T. T. Yin — 12/28/78

Reviewed By: /s/ D. H. Danielson, Chief
Engineering Support Section 2 — 12/28/78

Inspection Summary

Inspection on November 16-17 and 21, 1978 (Report No. 50-358/78-27)
Areas Inspected: Inspection of safety-related hangers and restraints and document control provision for pipe stress calculations and reports including: (1) Review of welding and NDE procedures, (2) Observation of weld control and performance, (3) Review of welding and material records, (4) Observation of hanger component outdoor storage, and (5) Review of document control measures for pipe stress analyses. The inspection involved a total of 13 inspector-hours onsite, and 5 inspector-hours at the Sargent and Lundy offices by one NRC inspector.
Results: Of the five areas inspected, no apparent items of noncompliance were identified in four areas; one item of noncompliance was identified in one area (Infraction - rigid supports welded to essential piping were without material and welder's identification - Paragraph 3.d).

A-XII-133

DETAILS

Persons Contacted

Inspection on Site During November 16-17, 1978

Principal Licensee Employees (CG&E)

*B. K. Culver, Project Manager
W. W. Schwiers, Principal QA and Standards Engineer
*J. F. Weissenberg, QA and Standards Engineer
*R. L. Wood, QA and Standards Engineer
*J. B. Vorderbrueggen, Hanger Engineer

Kaiser Engineer, Inc. (KEI)

*R. Marshall, Project Manager
R. E. Turner, QA Manager
*M. G. Franchuk, Mechanical QA Engineer
K. T. Shinkle, Hanger and Mechanical Inspector

Investigation and Inspection at Sargent and Lundy Office, Chicago on November 21, 1978

Licensee Representative (CG&E)

**W. W. Schwiers, Principal QA and Standards Engineer

Sargent and Lundy Engineers (S&L)

**E. B. Branch, Head, Engineering Mechanics Division (EMD)
**J. B. Adee, Jr., QA Coordinator
S. Rurka, Senior Structural Project Engineer
J. M. McLaughlin, Assistant Manager, Structural Department
**G. T. Kitz, Section Superviosr, EMD
**R. J. Pruski, Project Manager
**R. F. Scheibel, Project Director
S. G. Carlson, Mechanical Project Engineer
R. P. Pauliukonis, Administrator, EMD
A. P. Gillis, Senior QA Coordinator
**H. S. Taylor, Assistant Head, QA Division
**W. G. Hegener, Manager, Mechanical Department

USNRC Region I

A. N. Fasano, Reactor Inspector

*denotes those present at site Exit Interview on November 17, 1978.
**denotes those present at site Exit Interview on November 21, 1978.

A-XII-134

Licensee Action on Previously Identified Items

(Open) Noncompliance Item (358/78-18-01): Inadequate hanger, snubber, and restraint inspection program. The details of the problem areas observed are recorded in RIII Report No. 78-18, Section II, Paragraph 1. During this inspection samples of the KEI "Construction Inspection Plan Daily Work Sheet, Inprocess Inspection", and the CG&E "Mechanical Construction Test Procedure" MC-5 entitled "Pipe Support Final Inspection," Revision 0, dated September 5, 1978, were reviewed by the inspector and are considered unacceptable based on the following findings:

1. The inprocess inspection checklist relative to the specific inspection areas was not prepared and approved by supervision or engineering.

2. No requirements for documenting deviations if they are not within specific acceptance tolerance.

3. Inspections that should be carried out during inprocess installations to ensure timely corrections and prevent recurrence were placed in final inspection period when adjustment of components are required prior to testing and operations.

Functional or Program Areas Inspected

1. Review of Welding and NDE Procedures for Hangers and Restraints

   The following procedures from the Kaiser Engineers, Inc. (KEI) Special Process Procedure Manual (SPPM) were reviewed by the inspector relative to the welding and NDE work for hangers and restraints.

   a. SPPM No. 3.1.51, "Shielded Metal Arc, AWS Carbon Steel to Carbon Steel Structural Shapes Charpy Tested A-588 Gr. B (As rolled)", dated February 17, 1975.. This procedure was qualified on January 24, 1975.

   b. SPPM No. 4.6, R.3," Visual Examination", dated May 23, 1978.

   c. SPPM No. 3.3, R.4, "Welding Filler Materials Control Procedure", dated April 9, 1975.

   No items of noncompliance or deviations were identified.

2. Observation of Weld Controls and Performance

   a. The inspector observed the weld filler metal issuance station east of the reactor building. Items reviewed included calibration of electrode hol... oven, control of portable heated containers, segregat... of electrodes by type and size, and certifications of electrodes.

b. The inspector observed the finished shop and field welds on hangers and snubbers number 1WS044SR, 1WR091SR, 1WR214HR, 1WR200HR, and 1WS311HA. No rejectable weld surface conditions had been identified.

No items of noncompliance or deviations were identified.

3. Review of Welding and Material Records

Selective areas of weld filler metal and attachment hanger materials for the following items were reviewed by the inspector for certification of compliance.

a. Snubber 1WS044SR:

The heat number on the pipe attachment, a section of 8" sch. 40 pipe, is 31565. It was shop welded to line 1WS15A18.

b. Rigid Restraint 1WR091SR:

The heat number on the pipe attachment is 252661, shop welded to line 1WR03A14.

c. Anchor 1WS311HA:

The heat numbers on the shear lugs are 69D782 and 71A088. The material certification, including tensile testing and chemical analysis for Heat No. 69D782, was reviewed by the inspector.

d. Rigid Supports 1WR200HR, and 1WR214HR:

The heat numbers for the rigid support pipe members welded to the Closed Cooling System lines 1WR06A14 and 1WR07A14 designated as pipe class "C" and seismic class "B" essential components were not available. Further, the heat numbers for the weld filler metal and welders' identification were not marked on the supports or recorded in the installation documents.

e. Rigid Seismic Restraints 1FC037SR:

There were no heat numbers on the eight shear lugs. Noncompliance Report No. E 1445, dated November 8, 1978, was written to document this.

A-XII - 136

1028 295

The lack of control and identification of safety related materials as described in item d above is an apparent item of noncompliance identified in Appendix A. (358/78-27-01).

Except as noted, no items of noncompliance or deviations were identified.

4. Observation of Hanger Component Outdoor Storage

The condition of components for hangers and restraints stored outdoors at the laydown area, east of the reactor buildings, was considered questionable. Rust was observed at the ball bushing coated structural members. Components were observed off the dunnage on the sandy soil. A Nonconformance Report, No. E-851 identified piping storage problems at the same area on October 10, 1977. This report was still open. The inspector stated that he had concerns relative to: (1) the acceptability of the rusty and frozen bushings which may require close gap clearance, and (2) the timely resolution of noncomformances.

This is an unresolved item pending more in-depth inspection during a future visit. (358/78-27-02)

No items of noncompliance or deviations were identified.

5. S&L Pipe Stress Report Document Control

The inspector received a telephone call on October 17, 1978, from an individual who alleged that the control for documentation of the stress reports was not adequate. The inspector performed investigations relative to the subject matter on November 21, 1978, at the S&L office, Chicago. With the exception of one specific problem relative to the engineer not keeping an up-to-date procedure, no apparent problems were identified during the investigation and inspection.

Areas Inspected:

a. Procedures Review

The following manual procedures were reviewed by the inspector:

(1) S&L Organization Manual - Section 2H, dated October 15, 1978, a description of functional authorities and reporting responsibilities.

A-XII-137

1028 296

(2) S&L Organization Position Description Manual - Section 2H, dated October 15, 1976. This section included responsibilities of Mechanical Department, Engineering Mechanics Division (EMD), the division that is responsible for piping analysis and issuance of stress reports. Specific job descriptions within EMD reviewed included System Analysis Supervisor, Project Engineer, Piping Engineer, and Engineering Analyst.

(3) EMD Technical Procedure No. 11, "Standard EMD Checklist for Piping System Stress Report", Revision 3, dated July 3, 1978. This procedure included Group QA procedure GQ - 3.08, Design Calculation Requirements.

(4) EMD Administrative Procedure No. 5, "Procedure for Filing Reports, Memoranda, and Analyses", Revision 2, dated December 7, 1976.

(5) EMD Administrative Procedure No. 6, "EMD Report Number Assignment", Revision 1, approved on December 18, 1976.

b. Review of S&L Audits

The following S&L internal audit reports, including corrective actions, were reviewed by the inspector:

(1) Report on Internal Audit No. 23 performed on April 1, 2, and 5, 1976, relative to personnel compliance with procedures for design calculation, system and structure design review, and QA documentation.

(2) Report on Internal Audit No. G-17 performed on September 30, 1976, relative to personnel compliance with QA documentation requirements.

(3) Report on Internal Audit No. G-39 performed on October 24, 1977, relative to personnel compliance with QA procedures and department standards.

(4) Report on Internal Audit No. 31 performed on April 14, 14, 18 and 19, 1977, relative to personnel compliance with procedures for design calculations, and system design reviews.

(5) Report on Internal Audit No. G-59 performed on June 28-30, 1978, relative to personnel compliance with various department standards.

(6) Report on International Audit No. 41 performed after problems identified at the site in August, 1978, relative to the adequacies of hanger component design. Areas audited included work implementation of the following QA procedures:

GQ - 3.04	Design Criteria
GQ - 3.07	S&L Drawings
GQ - 3.08	Design Calculation
GQ - 4.01	Procurement Specification

(7) QA Division Corrective Action Reports (CAR)

CAR 66, dated June 10, 1977
CAR 67, dated June 10, 1977
CAR 93, dated October 22, 1976

(8) Reports on Internal Reaudits

. No. 31.1, dated July 15 and 21, 1977 for CAR No. 66, 67, 68 and 69.

. No. 17.1, dated November 26, 1976 for CAR No. 93.

c. Review of Stress Reports

The status of stress reports is controlled by card index files with computer printout updates as work tool and reference. Three approved stress reports selected at random were reviewed by the inspector in the area of document control.

(1) EMD-4130-WR-09, "Reactor Building Closed Cooling Water", dated November 22, 1977.

(2) EMD-4130-HD-6, "From Second Stage RHTR Drain Tank to Condensor HD-6", Revision 1, dated June 12, 1978.

The above two reports were signed off by the qualified preparer and reviewer and approved by the supervisor. QA calculation checklist was included and signed-off. Computer printout configuration, code commitments, and reference drawing number were included in the report. The computer program used is PIPSYS 09.5.065-3.4, dated September 8, 1977. The load combination data appeared to be complete.

(3) Report on "10-inch Containment Spray Header (Upper), Run 5".

A-XII, 39

1028 298

RH-05-TH-3, Revision 2, dated September 27, 1973.
RH-05-SE-2, dated September 27, 1973.
RH-05-WT-2, dated March 30, 1973.

Although no calculation checklist was included in the above reports, it was considered acceptable because the calculations were performed prior to the establishment of such a requirement. The reports include system configuration. The load combination for primary and secondary stresses appeared to be in order. The code requirements were stated in the reports.

d. Staff Interview

The inspector interviewed the Engineering Analyst (EA) who reviewed the reports EMD-4130-WR-09, and EMD-4130-HD-6. The latest revision of the procedure for reviewing stress calculations was not in procession of the EA, even though the Interoffice Memorandum of July 24, 1978, from the Head, to EMD to the staff specifically stated that EMD Technical procedure No. 11, Revision 3, approved on July 3, 1978, should be maintained and used by the Staff. The inspectors concern included: (1) the outdated EMD Procedure No. 11, dated September 6, 1977, was not removed from the EA's location, (2) the EA's apparent unawareness of the procedure updating, and (3) the EA's difficulty in retrieving the required procedures. This is an unresolved item. (358/78-27-03)

No items of noncompliance or deviations were identified.

Unresolved Items

Unresolved items are matters about which more information is required in order to ascertain whether they are acceptable items, items of noncompliance or deviations. Two unresolved items disclosed during the inspection are discussed in Paragraph 4 and 5.d.

Exit Interview

The inspector met with licensee representatives (denoted in the Persons Contacted paragraph) at the conclusion of the inspection on November 17 and 21, 1978. The inspector summarized the purpose and findings of the inspection. The licensee acknowledged the findings reported herein.

A-XII 140

1028 299

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS
UNITED STATES ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

September 17, 1971

Honorable James R. Schlesinger
Chairman
U. S. Atomic Energy Commission
Washington, D. C. 20545

Subject: REPORT ON WILLIAM H. ZIMMER NUCLEAR POWER STATION, UNIT 1

Dear Dr. Schlesinger:

At its 137th meeting, September 9-11, 1971, the Advisory Committee on Reactor Safeguards completed its review of the application from the Cincinnati Gas and Electric Company, the Columbus and Southern Ohio Electric Company, and the Dayton Power and Light Company for a permit to construct the William H. Zimmer Nuclear Power Station, Unit 1. The Cincinnati Gas and Electric Company is responsible for the design, construction, and operation of the plant and is authorized to act as sole agent during construction and for licensing negotiations. The project was considered at Subcommittee meetings on August 27, 1971, at the plant site, and on September 1 and September 8, 1971, in Washington, D. C. During its review the Committee had the benefit of discussions with representatives and consultants of the applicants, Sargent and Lundy, the General Electric Company, and the AEC Regulatory Staff. The Committee also had the benefit of the documents listed below.

The Zimmer Station will be located in Ohio on a 635-acre site on the Ohio River approximately 24 miles southeast of Cincinnati and one-half mile north of Moscow, Ohio. The population of Moscow is estimated by the applicants to be 348. The nearest population center is Covington, Kentucky which is located 20 miles northwest of the site and has a population of 60,000. The low population zone radius is 3.0 miles within which the 1960 population was less than 1,900 and the projected 1985 population less than 2,800. The projected 1985 population within 10 miles of the site is 30,100. The exclusion zone has a minimum radius of 1,250 feet, is bounded on the north by U. S. Route 52, and includes a small manufacturing plant located on the periphery. Provisions have been made to evacuate the employees of this plant in the unlikely event of an accident.

A - XII -141

1028 300

The Zimmer Station will utilize a General Electric boiling water reactor to be operated at a power level of 2436 MWt. It is the first reactor of the GE 1969 product line reviewed by the Committee. Waste heat is rejected to the atmosphere by a natural-draft cooling tower.

The primary containment is of the over-under pressure suppression type similar to those of the previously reviewed Limerick and Shoreham units. The drywell is a steel-lined prestressed concrete truncated cone; the pressure suppression chamber is a cylinder of similar construction. The drywell and pressure suppression chamber are separated by a reinforced concrete deck penetrated by 88 vent pipes. The reactor building is constructed of reinforced concrete up to the refueling floor and of structural steel and paneling at higher levels. The design is intended to limit inleakage to 100% of the building volume per day at a pressure of 1/4 inch of water during operation of the standby gas treatment system. This system, which includes provisions for circulating air throughout the reactor building, exhausts through redundant sets of double high efficiency particulate air filters and deep-bed activated carbon sorbers.

The emergency core cooling system of the GE 1969 product line incorporates several changes. The high pressure injection system has been modified to inject water through a sparger directly over the top of the core, rather than into the downcomer region via the feedwater line. Also, an electric motor drive instead of a steam turbine drive is used for the pump. This system now also serves as one of the two core spray systems. The low pressure coolant injection system has been modified to inject water from the suppression pool directly into the core region through three separate lines, each of which is supplied water by a separate pump. The maximum diameter of the reactor recirculation piping has been reduced from 28 to 20 inches.

The applicants have proposed to design the main steam lines and turbine stop and bypass valves to requirements which are substantially similar to AEC quality assurance Classification Group B. The Committee believes that the main steam lines should be designed to retain their integrity during a design basis earthquake. The applicants propose to install a sealing system, designed as an engineered safety feature, in connection with the main steam line isolation valves to minimize leakage. These matters should be resolved in a manner satisfactory to the Regulatory Staff prior to completion of construction of the station.

The applicants have studied design features to make tolerable the consequences of failure to scram during anticipated transients, and have concluded that automatic tripping of the recirculation pumps and injection of boron could provide a suitable backup to the control rod system for this type of event. The Committee believes that this recirculation pump trip represents a substantial improvement and should be provided for the Zimmer reactor. However, further evaluation of the sufficiency of this approach and the specific means of implementing the proposed pump trip should be made. This matter should be resolved in a manner satisfactory to the Regulatory Staff and the ACRS during construction of the reactor.

The radioactive waste disposal systems process high and low conductivity liquid wastes by demineralizers or evaporators and the decontaminated effluent is recycled to the condensate storage tank for reuse. Chemical and detergent wastes normally are to be processed through evaporators and, if necessary, further processed by demineralizers before discharge. The gaseous waste treatment system includes a high temperature catalytic recombiner followed by a 30-minute holdup system. The applicants will provide an additional holdup system which results in the substantial reduction of all isotopes except long-lived krypton. The applicants have stated that both the liquid and gaseous waste handling systems will be used to the fullest extent and will limit releases of radioactivity or exposures to man to values less than those specified in the proposed 10 CFR 50, Appendix I. An environmental monitoring program has been established, and the applicants have stated that it will permit the calculation of radiation exposures to man from records of radioactivity released from the plant.

The applicants have stated a system will be provided to control the concentration of hydrogen in the primary containment that might follow in the unlikely event of a loss-of-coolant accident. The Committee believes that the containment should be inerted and that the hydrogen control system should be designed to maintain the hydrogen concentration within acceptable limits using the assumptions listed in the AEC Safety Guide 7, Control of Combustible Gas Concentrations in Containment Following a Loss of Coolant Accident.

The applicants' pipe whip criteria consider both longitudinal and circumferential pipe breaks and provide for the installation of piping restraints as required to prevent damage to essential reactor coolant systems and equipment or to the containment.

Other problems related to large water reactors have been identified by the Regulatory Staff and the ACRS and cited in previous ACRS reports. The Committee believes that resolution of these items should apply equally to the Zimmer Station.

The Committee believes that the items mentioned above can be resolved during construction and that, if due consideration is given to these items, the William H. Zimmer Nuclear Power Station, Unit 1 can be constructed with reasonable assurance that it can be operated without undue risk to the health and safety of the public.

Sincerely yours,

Spencer H Bush

Spencer H. Bush
Chairman

References

1. Cincinnati Gas and Electric Company, Columbus and Southern Ohio Electric Company, and The Dayton Power and Light Company, License Application and Preliminary Safety Analysis Report (Volumes 1 through 5) for the William H. Zimmer Nuclear Power Station

2. Amendments 1 through 7 and 9 through 19 to the License Application for the William H. Zimmer Nuclear Power Station

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

February 16, 1978

Honorable Joseph M. Hendrie
Chairman
U. S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: REPORT ON EDWIN I. HATCH NUCLEAR PLANT, UNIT NO. 2

Dear Dr. Hendrie:

During its 214th meeting, February 9-11, 1978, the Advisory Committee on Reactor Safeguards completed its review of the application of the Georgia Power Company, Oglethorpe Electric Membership Corporation, Municipal Electric Authority of Georgia and the city of Dalton, Georgia (the Applicants) for a license to operate the Edwin I. Hatch Nuclear Plant, Unit No. 2. The plant will be operated by Georgia Power Company. The application was reviewed at Subcommittee meetings on January 27 and 28, 1978 in Washington, D.C. During its review, the Committee had the benefit of discussions with representatives and consultants of the Nuclear Regulatory Commission (NRC) Staff; General Electric Company; Southern Company Services, Incorporated; Bechtel Power Corporation; and the Applicants. The Committee also had the benefit of the documents listed.

The Edwin I. Hatch Nuclear Plant is a two-unit station located on the south bank of the Altamaha River approximately 11 miles north of Baxley, Georgia. The two units are virtually identical except that Hatch Unit No. 1 utilizes 7X7 fuel assemblies while Hatch Unit No. 2 will utilize 8X8R (Retrofit) fuel assemblies. The rated thermal power for each unit is 2436 MW(t). Each unit includes a General Electric Company BWR/4 boiling water reactor. The Committee reported on the application for a construction permit for Unit No. 2 on November 3, 1971.

Hatch Unit No. 2 is the first reactor scheduled to use the new General Electric 8X8R fuel on a core-wide basis. This fuel design is a slightly modified version of the General Electric 8X8 fuel assembly design currently in use in a number of boiling water reactors. These modifications include, among others, an increase in fuel length, use of natural uranium at the top and bottom of the fuel rod and the addition of a second water rod to each fuel assembly. These changes improve the shutdown and thermal margins, provide flatter local power distribution, and improve fuel cycle efficiency. Four of the 8X8R fuel assemblies have been operating in Peach Bottom Unit No. 2 since May 1976 and two assemblies have been

A XII - 145

operating in Vermont Yankee since August 1976. The NRC Staff has concluded that the 8x8R fuel assembly design is acceptable for use in Hatch Unit No. 2. The Committee concurs.

The NRC Staff has identified a number of safety-related items which will require resolution prior to a decision on the issuance of an operating license. These matters should be resolved in a manner satisfactory to the NRC Staff.

With regard to the generic problems listed in the Committee's report, "Status of Generic Items Relating to Light-Water Reactors - Report No. 6," dated November 15, 1977, items considered relevant to Edwin I. Hatch Nuclear Plant, Unit No. 2 are: II-1, 4, 5A, 5B, 6, 7, 8, 10; IIA-4; IIB-2, 4; IIC-1, 3A, 3B, 5, 6, 7; IID-2. These problems should be dealt with by the NRC Staff and the Applicants as solutions are found.

The Advisory Committee on Reactor Safeguards believes that if due consideration is given to the items mentioned above, and subject to satisfactory completion of construction and preoperational testing, there is reasonable assurance that the Edwin I. Hatch Nuclear Plant, Unit No. 2 can be operated at power levels up to 2436 MW(t) without undue risk to the health and safety of the public.

Sincerely yours,

Stephen Lawroski

Stephen Lawroski
Chairman

References

1. Edwin I. Hatch Nuclear Plant, Unit No. 2, Final Safety Analysis Report, with Amendments 18 through 41.
2. Report to the Advisory Committee on Reactor Safeguards by the Office of Nuclear Reactor Regulation, U. S. Nuclear Regulatory Commission in the matter of Georgia Power Company, et al, Edwin I. Hatch Nuclear Plant, Unit No. 2, dated January 4, 1978.
3. General Electric Company, "Lattice Physics Methods," NEDE-20913A, January, 1977.
4. General Electric Company, "Lattice Physics Methods Verification," NEDO-20939A, January, 1977.
5. General Electric Company. "BWR Simulator Methods Verification," NEDO-20946A, January, 1977.
6. General Electric Company, "Three-Dimensional BWR Core Simulator," NEDO-20953A, January, 1977.

A XII -14L

7. General Electric Company, "BWR/6 Fuel Design," NEDE-20948-P, June, 1976, and Amendment No. 1, November, 1976.
8. General Electric Company, "BWR/4 and BWR/5 Fuel Design," NEDE-20944-P, September, 1976.
9. General Electric Company, "BWR Fuel Channel Mechanical Design and Deflection," NEDE-21354-P, September, 1976.
10. General Electric Company, "BWR/6 Fuel Assembly: Evaluation of Combined Safe Shutdown Earthquake (SSE) and Loss-of-Coolant Accident (LOCA) Loadings," NEDE-21175-P, November, 1976 and Amendment 1, April, 1977.

A XII -147

OUTSTANDING ISSUE

. MARK II ACCEPTANCE CRITERIA

- LOAD CRITERIA

- DESIGN ASSESSMENT

(VIEWGRAPH 1)

CONFIRMATORY ITEMS

- TOXIC CHEMICALS (ROUTE 52)
- QUALIFICATION OF EQUIPMENT
- TRANSIENT ANALYSIS (ODYN CODE)
- REACTOR FLOW CONTROL SYSTEM
- CONTROL ROD DRIVE TUBES
- INSERVICE INSPECTION
- RECIRCULATION PUMP TRIP EFFECTS
- EXEMPTIONS TO 10 CFR 50 APP. G, H, J
- LPCI FLOW DIVERSION EFFECTS
- PHYSICAL SEPARATION AND ELECTRICAL ISOLATION
- PROTECTION OF MOTOR/GENERATOR SETS
- AUTOMATIC ACTUATION OF WETWELL SPRAYS
- SAFETY RELATED DISPLAY INSTRUMENTATION
- USE OF NON-SAFETY GRADE EQUIPMENT
- FIRE PROTECTION
- INDUSTRIAL SECURITY
- INITIAL TEST PROGRAM

(VIEWGRAPH 2)

ITEM OF DISAGREEMENT

- DEWATERING OF COMPACTED BACKFILL
  (457' VS 480' WATER LEVEL)

Resolved at Compromise
460' msl

(VIEWGRAPH 3)

A-103

ITEMS RESOLVED SINCE SER ISSUANCE

OUTSTANDING ISSUE

- EMERGENCY CORE COOLING
  - TWO LOOP TEST APPARATUS

CONFIRMATORY ITEM

- FINANCIAL

(VIEWGRAPH 4)

A-104

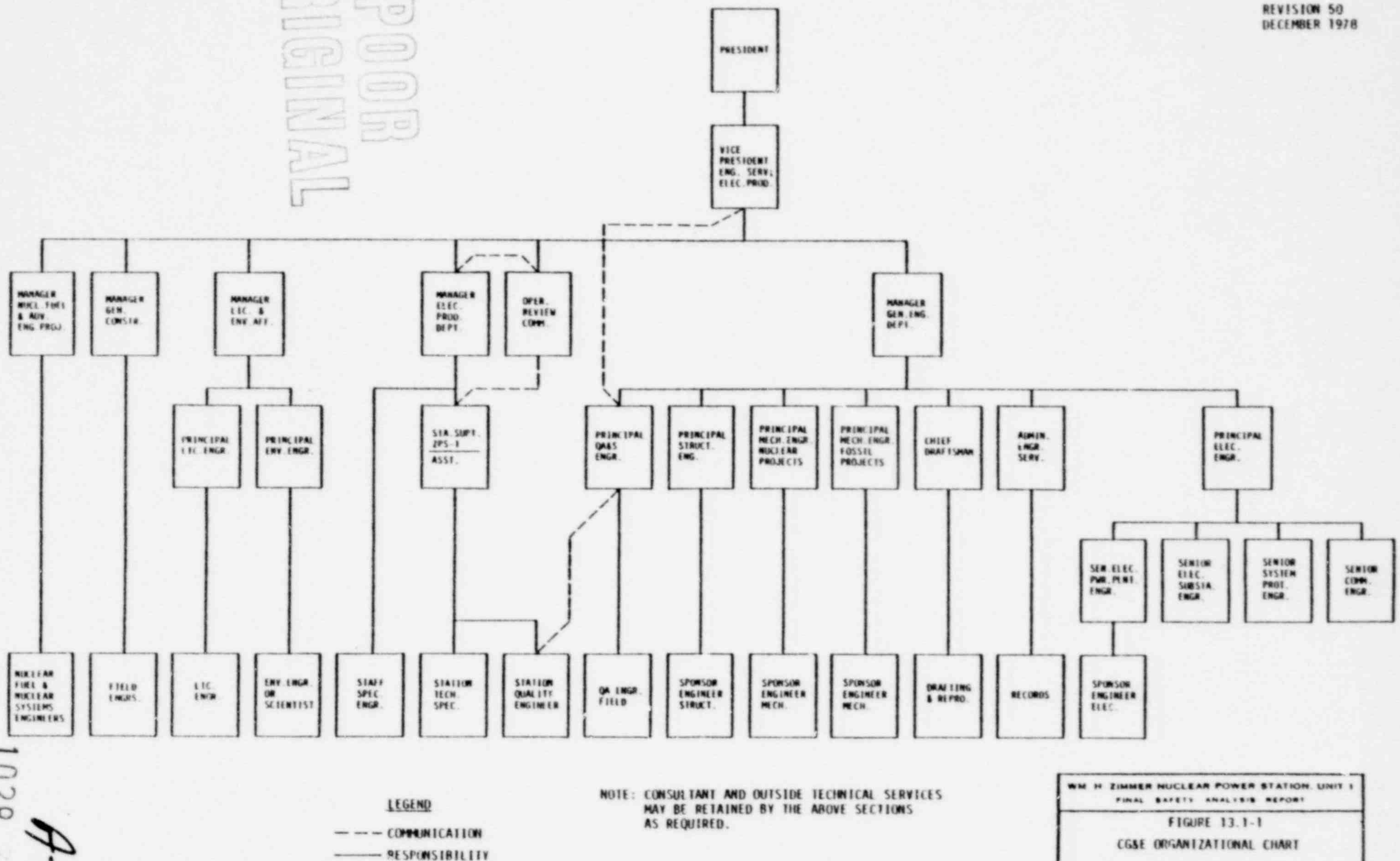
ACRS GENERIC CONCERNS

CONSIDERED IN SER

ACRS DESIGNATION	SUBJECT	SER SECTION
II-1	TURBINE MISSILES	3.5
II-5A	LOOSE PARTS MONITOR	4.4.1
II B-2	QUALIFICATION OF NEW FUEL GEOMETRIES	4.2
II B-4	STRESS CORROSION CRACKING IN BWR PIPING	5.2
II C-4	VESSEL SUPPORT STRUCTURES	5.2.1
II C-5	WATER HAMMER	6.3.2
II C-6	MAINTENANCE AND INSPECTION OF PLANTS	12.3

(VIEWGRAPH 5)

REVISION 50
DECEMBER 1978



NOTE: CONSULTANT AND OUTSIDE TECHNICAL SERVICES MAY BE RETAINED BY THE ABOVE SECTIONS AS REQUIRED.

WM. H. ZIMMER NUCLEAR POWER STATION, UNIT 1
FINAL SAFETY ANALYSIS REPORT

FIGURE 13.1-1
CG&E ORGANIZATIONAL CHART

APPENDIX XIV

ZIMMER: OPERATING ORGANIZATION

A-106

1028 312

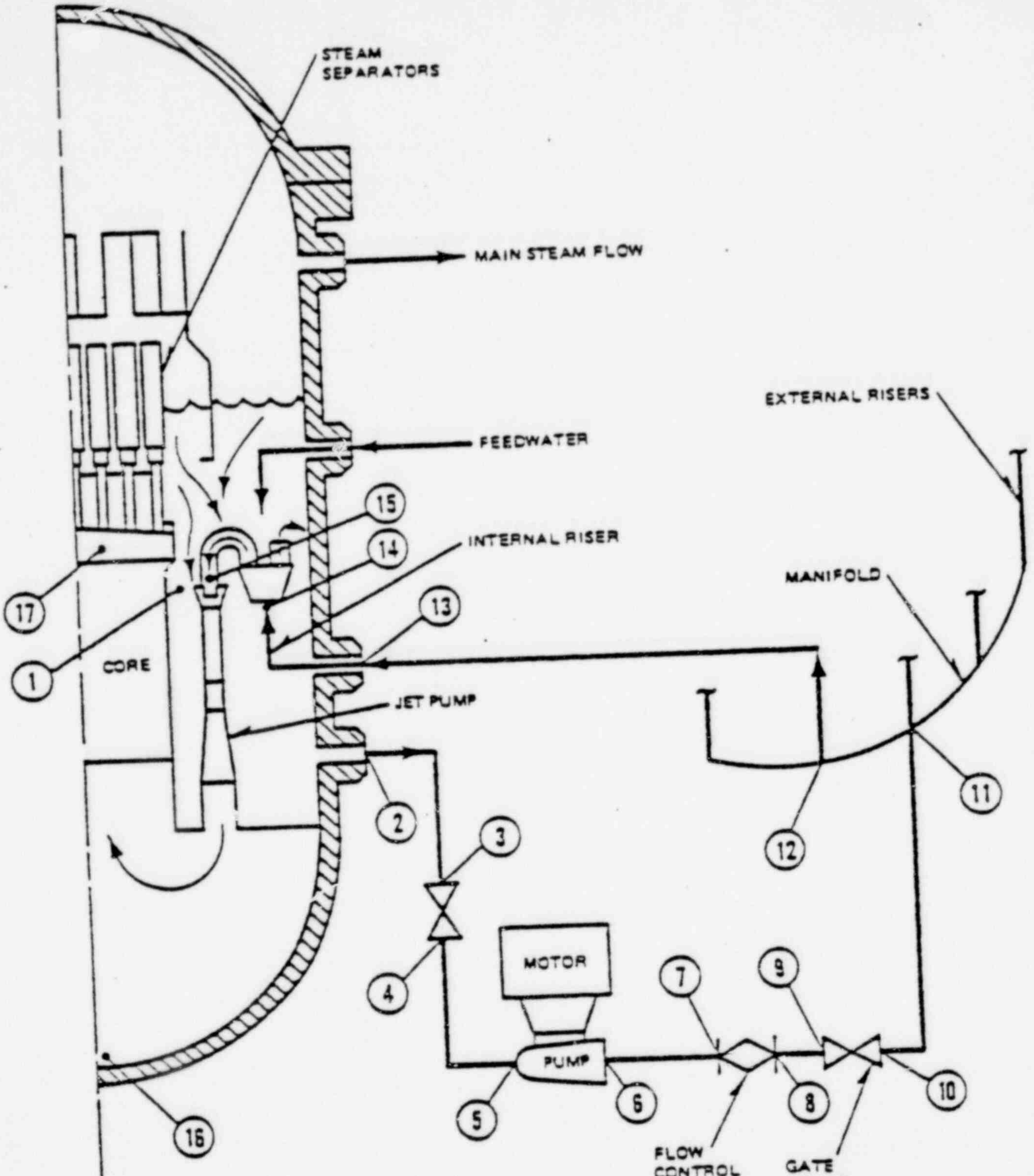
BWR/5 NSSS DESIGN FEATURES
INTRODUCED IN 1969*

APPENDIX XV
ZIMMER: BWR/5 NSSS DESIGN FEATURES
INTRODUCED IN 1969

- HIGH PRESSURE CORE SPRAY
- SOLID STATE REACTOR MANUAL CONTROL SYSTEM
- RECIRCULATION FLOW CONTROL SYSTEM
- APRM CHANNEL EFFECTIVENESS IN STARTUP RANGE

*BWR/5 = GE-1969 PRODUCT LINE

RBJ:MM/1382
3/6/79



WM. H. ZIMMER NUCLEAR POWER STATION, UNIT
FINAL SAFETY ANALYSIS REPORT

FIGURE H.2.1-1

SCHEMATIC OF RECIRCULATION SYSTEM

A-108

1028 514

BWR/5

<u>REACTOR RECIRCULATION FLOW CONTROL SYSTEM</u>

<u>GENERAL</u>

- SYSTEM PROVIDES VARIABLE COOLANT FLOW FOR ADJUSTING REACTOR POWER LEVEL TO MEET LOAD FOLLOWING REQUIREMENTS.

- SYSTEM IS FOR POWER GENERATION OPERATIONAL CONTROL AND IS NONESSENTIAL

- IMPROVED SYSTEM DESIGN FEATURES

  <u>JET PUMPS</u> - FIVE NOZZLE JET CONFIGURATION

  <u>FLOW CONTROL</u> - CONSTANT SPEED PUMP/MOTOR WITH VARIABLE FLOW CONTROL VALVE

DAJ/877
3/6/79

BWR/5

REACTOR RECIRCULATION FLOW CONTROL SYSTEM

SUMMARY OF DESIGN CHANGES

BWR/3 & BWR/4	VS	BWR/5
• SINGLE NOZZLE JET PUMPS		• FIVE NOZZLE JET PUMPS
• FLOW CONTROL BY VARIABLE SPEED PUMP/MOTOR (M-G SET VARIES FREQUENCY)		• FLOW CONTROL BY VARIABLE CONTROL VALVE ON DISCHARGE OF CONSTANT SPEED PUMP/ MOTOR. (FLOW VARIED BY INCREASE OR DECREASE IN PRESSURE DROP)
• PUMP COASTDOWN GOVERNED BY INERTIA OF PUMP/MOTOR AND INERTIA OF M-G SET		• PUMP MOTOR ROTOR INERTIA INCREASED BY ~20% BECAUSE NO M-G SET (TO OBTAIN SIMILAR COASTDOWN CHARACTERISTICS)
• RECIRCULATION BYPASS LINE USED TO THROTTLE PUMP DURING PLANT HEATUP AND LOW POWER OPERATIONS		• L-F-M-G SET TO MINIMIZE PUMP HEAT INPUT DURING PLANT HEATUP AND LOW POWER OPERATION (PUMP OPERATES AT 25% OF NORMAL SPEED)

DAJ/878
3/6/79

BWR/3 & BWR/4	VS	BWR/5
• RECIRC PIPING 28" O.D.		• RECIRC PIPING 20" O.D. (REDUCED TO ALLOW MINIMUM DRYWELL SPACE REQUIREMENTS, ALSO CONFIGURATION MODIFIED TO ACCOMMODATE FCV)
• PUMP FLOW MEASUREMENT BY FLOW ELEMENT NOZZLE		• PUMP FLOW MEASURED BY ELBOW PRESSURE TAP FLOW ELEMENT
• RPT ELIMINATES POWER SOURCE TO PUMP MOTOR		• RPT ACTIVATES 25% SPEED SOURCE (LOW FREQ. M-G SET) & TRIP 100% SPEED SOURCE

DAJ/879
3/6/79

BWR/5

REACTOR RECIRCULATION FLOW CONTROL SYSTEM

LICENSING REVIEW

- COMPREHENSIVE DESCRIPTION PROVIDED IN APP. H (JULY 77) TO ZIMMER FSAR

- PRINCIPAL NRC REVIEW AREAS

  - CAVITATION PROTECTION

  - RECIRCULATION FLOW INCREASE TRANSIENT

  - FLOW CONTROL VALVE POSITION DURING LOCA

- NRC REVIEW COMPLETE AND SYSTEM IS ACCEPTABLE (FEB. '79 ACRS SUBCOMMITTEE MEETING-ZIMMER)

DAJ/880
3/6/79

A-112

A. INITIAL PLANT STAFF TRAINING PROGRAM

B. REQUALIFICATION PROGRAM

C. REPLACEMENT TRAINING PROGRAM

A. <u>INITIAL PLANT STAFF TRAINING PROGRAM</u>

1. <u>OPERATIONS GROUP</u>
   - A. INITIAL COLD LICENSE TRAINING PHASES I THRU VI
   - B. NONLICENSED OPERATOR TRAINING

2. <u>SUPERVISORY STAFF</u>
   - A. INTRODUCTION TO NUCLEAR POWER
   - B. ACCELERATED NUCLEAR POWER PREPARATORY TRAINING
   - C. STATION NUCLEAR ENGINEERING
   - D. BWR CHEMISTRY
   - E. BWR MAINTENANCE
   - F. NUCLEAR INSTRUMENTATION
   - G. PROCESS INSTRUMENTATION AND CONTROL
   - H. BWR OPERATING FUNDAMENTALS
   - I. OBSERVATION AND TRAINING AT OPERATING FACILITIES

3. <u>PLANT TECHNICIANS</u>
   - A. ZIMMER ORIENTATION
   - B. NUCLEAR FUNDAMENTALS
   - C. RADIATION PROTECTION
   - D. SPECIFIC COURSES
     - I. - ELECTRONIC FUNDAMENTALS
     - II. - NUCLEAR INSTRUMENTATION
     - III. - DIGITAL LAB
     - IV. - SYSTEMS TRAINING
     - V. - GENERAL MAINTENANCE (CENTRIFUGAL PUMPS, VALVE LAPPING & PACKING, RIGGING & LIFTING, ETC.)
   - E. PARTICIPATION IN PREOP & STARTUP TESTING: LAB & SHOP SET-UP; ON-THE-JOB IN THEIR SPECIALTY.

B. <u>REQUALIFICATION PROGRAM</u>

1. LICENSED (RO OR SRO) PERSONNEL

A. PRE-PLANNED LECTURES

I.- THEORY; PRINCIPALS OF OPERATION

II.- GENERAL AND SPECIFIC OPERATING CHARACTERISTICS

III.- INSTRUMENTS & CONTROLS

IV.- PROTECTION SYSTEMS

V.- ESF

VI.- PROCEDURES

VII.- RADIATION CONTROL AND SAFETY

VIII.- TECH. SPECS

IX.- QUALITY

B. REACTIVITY MANIPULATIONS

I. -REACTOR STARTUP & SHUTDOWN

II. -CR SEQUENCE CHANGES

III. -SD MARGIN CHECKS

IV. -CR SCRAM TIMING

V. -REFUELING

C. APPARATUS OPERATION

D. PLANT CHANGES (DESIGN, PROCEDURES, T.S., ETC.)

E. PROCEDURE REVIEW
(ABNORMAL & EMERGENCY)

EMERGENCY CONDITIONS

A. PERSONNEL EMERGENCY

1. APPLICABILITY - INDIVIDUALS WITHIN THE SITE BOUNDARY REQUIRE EMERGENCY TREATMENT

B. STATION EMERGENCY

1. APPLICABILITY - PHYSICAL OCCURRENCE WITHIN THE PLANT. VERY UNLIKELY THAT OFFSITE HAZARDS WILL RESULT.

EXAMPLES

A. FIRE

B. EXPLOSION

C. RELEASE OF TOXIC GAS

D. NATURAL PHENOMENA

C. GENERAL EMERGENCY

THREE CATEGORIES DIVIDED ACCORDING TO SEVERITY. GENERAL EMERGENCY INVOLVES A RADIOACTIVE RELEASE INTO THE AIR, WATER, OR GROUND SUCH THAT INITIAL ASSESSMENT INDICATES OFFSITE AGENCY NOTIFICATION IS NECESSARY.

1. NOTIFICATION - EXCEEDS REPORTABLE TECH. SPEC. VALUES

2. MOBILIZATION ALERT -

   A. SHORT TERM (UP TO 24 HRS)
      WHOLE BODY DOSE >50 MREM
      <1 REM
      THYROID >300 MREM
      < 5 REM

   B. SHORT TERM TO RIVER UNIDENTIFIED GROSS BETA - GAMMA ACTIVITY CONCENTRATION >1 MPC AFTER DILUTION AT THE DOWNSTREAM WATER INTAKES

   C. ACTUAL OR PREDICTED OFFSITE CONTAMINATION DUE TO STATION OPERATION FOR WHICH FRC PAG's MAY BE EXCEEDED FOR AGRICULTURAL PATHWAYS.

3. OFFSITE GENERAL EMERGENCY ACTUAL OR IMMINENT ATMOSPHERIC RELEASES LIKELY TO RESULT IN DOSES EXCEEDING EPA PAG's.

STATION ADMINISTRATION

CHEMICAL/RADIOCHEMICAL

DESIGN AND MODIFICATIONS

DOCUMENT CONTROL

EQUIPMENT CONTROL

EMERGENCY PLAN

SPECIAL TESTS AND EXPERIMENTS

FIRE PROTECTION

HOUSEKEEPING & CLEANLINESS CONTROL

INSTRUMENT MAINTENANCE

MEASURING AND TEST EQUIPMENT CALIBRATION

MECHANICAL AND ELECTRICAL MAINTENANCE

NUCLEAR ENGINEERING

OPERATIONS

PROCUREMENT CONTROL

QUALITY ASSURANCE

RADWASTE OPERATIONS

RADIATION PROTECTION

RECORDS MANAGEMENT

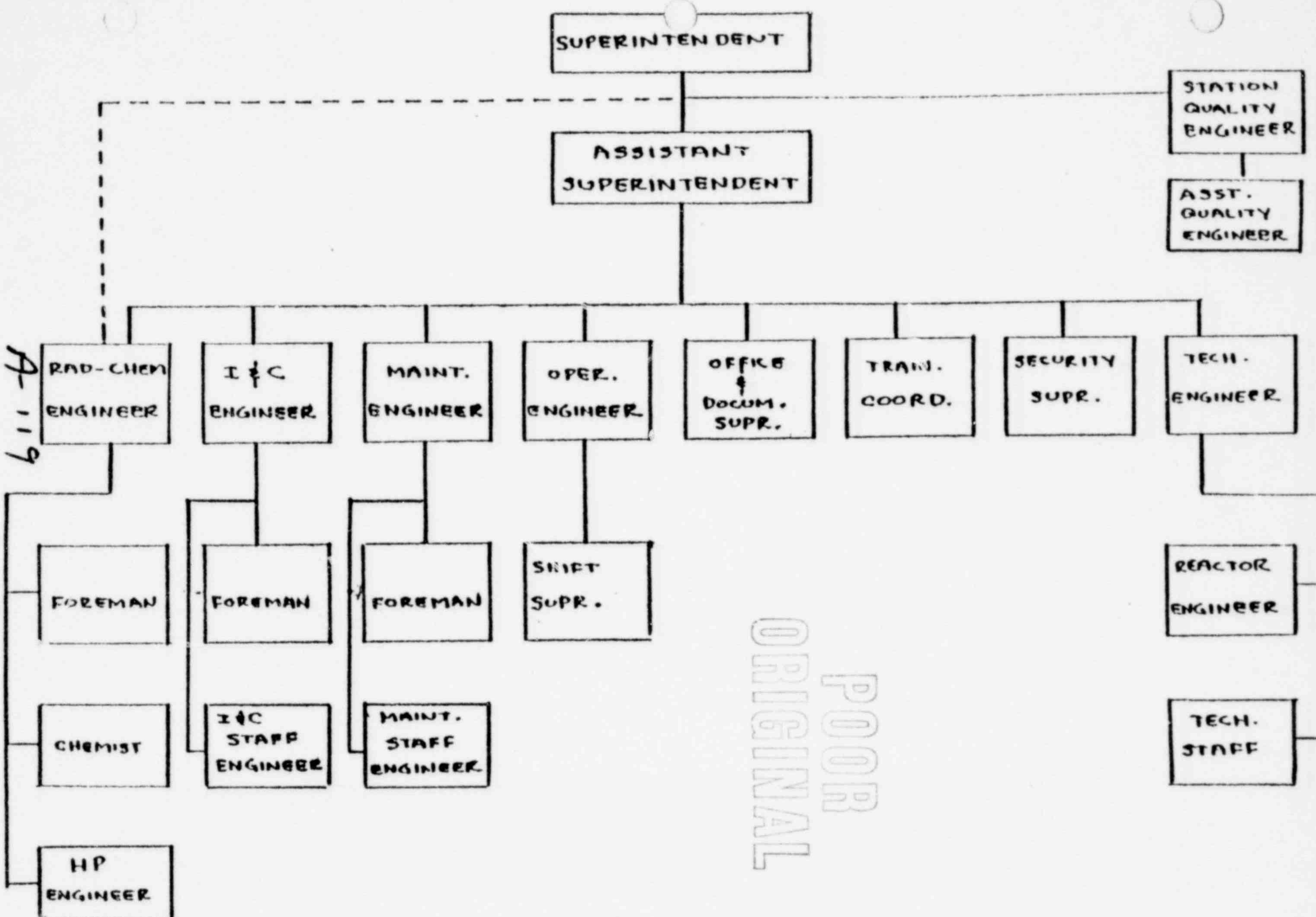
RELIABILITY

REPORTS MANAGEMENT

SECURITY

SPECIAL PROCESSES

TRAINING



A-119

POOR ORIGINAL

1028 325

ZIMMER

MARK II

LEAD PLANT LOAD CRITERION

OVERVIEW

39 LOAD OR PHENOMENA SPECIFICATIONS

39
- 14 ORIGINAL MK II CRITERIA ACCEPTABLE
- 5 PLANT UNIQUE REVIEW
- 20 NRC DEVELOPED CRITERIA

20
- 8 ADOPTED BY MK II O.G.
- 6 RECENTLY RESOLVED
- 6 UNDER REVIEW (2 SIGNIFICANT AREAS)

ZIMMER

MK II

LOADS RECENTLY

RESOLVED

LOAD/PHENOMENON	RESOLUTION
1. SUBMERGED BOUNDARY DURING VENT CLEARING	- EVALUATION OF ZIMMER CONTAINMENT - LETTER REPORT - MARCH
2. SMALL STRUCTURE IMPACT	ADOPTED NRC LOAD CRITERIA
3. ASYMMETRIC POOL SWELL	- EVALUATION OF ZIMMER CONTAINMENT - LETTER REPORT - MARCH
4. "T" QUENCHER ARM LOADS	USE DFFR METHOD FOR FOUR ARM QUENCHER
5. "T" QUENCHER TIE-DOWN LOADS	USE DFFR METHOD FOR FOUR ARM QUENCHER
6. "T" QUENCHER ZONE OF INFLUENCE	CYLINDRICAL ZONE OF INFLUENCE

A-121

ZIMMER

MK II LOADS

UNDER REVIEW

LOAD/PHENOMENA	RESOLUTION
1. QUENCHER AIR CLEARING LOADS	- 6 LOAD CASES MEET INTENT OF NRC CRITERIA OF SRV LOAD MAGNITUDE, FREQUENCY AND PHASING. CONFIRM WITH KWU TEST DATA. MARCH 1979. - IN-PLANT CONFIRMATORY TEST PLANNED.
2. LOCA JET SUBMERGED DRAG	- NEW RING VORTEX MODEL - PRELIMINARY STAFF REVIEW MARCH 1979
3-5. LOCA/SRV AIR BUBBLE DRAG	- ACCELERATION DRAG COEFFICIENTS - EQUIV. VELOCITY IN A UNIFORM FLOW FIELD. - INTERFERENCE EFFECTS CLOSELY SPACED STRUCTURES - GENERIC REPORT MARCH 1979
6. CHUGGING FSI	- GENERIC RESPONSE TO NRC QUESTIONS MARCH 1979 - IP CONFIRMATION

A-122

1028 323

Pool Dynamic Load Confirmatory Program

- Zimmer In-Plant Tests
  - Test Plan March 1979.
- Extended 4T Tests
  - Condensation Oscillations
  - Prototypical Vent Length
  - Report 4Q 1980
- New GKM II - Test Program
  - Condensation Oscillations
  - Vent Lateral Loads
  - Prototypical of a Specific Plant
  - Data, May 1980

A-123

CONCLUSIONS

ZIMMER

POOL DYNAMIC LOADS

- ZIMMER ADOPTED LARGE MAJORITY OF NRC CRITERIA

- ANTICIPATE NO PROBLEMS IN RESOLVING FEW OPEN ITEMS

- ZIMMER SER SUPPLEMENT MARCH 1979

- GENERIC SUPPLEMENT TO NUREG 0487 APRIL 1979

- MK II CONFIRMATORY PROGRAM AND ZIMMER IN-PLANT TESTS CONFIRM LEAD PLANT LOADS

Commonwealth Edison
One First National Plaza, Chicago, Illinois
Address Reply to: Post Office Box 767
Chicago, Illinois 60690

February 4, 1979

APPENDIX XVIII

DRESDEN 2: TECHNICAL SPECIFICATIONS CHANGE TO ALLOW FOR TEMPORARY OPERATION WITH DETACHED BLOW-OUT PANELS

Director of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Subject: Dresden Station Unit 2
Emergency Change to License DPR-19
Appendix A, Technical Specifications
NRC Docket No. 50-237

Dear Sir:

On February 2, 1979 at 7:15 P.M., several of the Unit 2/3 Reactor Building blow-off panels became detached as a result of a ventilation system malfunction which pressurized the Reactor Building. This created a 22'x40' opening in the Reactor Building superstructure at the refueling floor breaching the secondary containment. At this time, Unit 2 was operating at 700 MWe and Unit 3 was in the cold shutdown condition.

Unit 2 was immediately brought to a cold shutdown condition in accordance with Technical Specification requirements. Repairs have been initiated but are not expected to be complete until late on February 4 or early February 5, 1979.

Our System Power Supply office forecasts a 900 megawatt shortfall in meeting our Monday, February 5, 1979 load without Dresden Unit 2. When the required reserve margin is taken into account the projected deficiency would be 1300 megawatts. A survey of surrounding utilities showed that the prospect of buying emergency power is not good because of weather related problems on other systems.

In order to have Unit 2 carrying sufficient load to prevent a system load emergency on February 5, startup of the unit must commence by 2:00 P.M. on February 4, 1979 -- at least several hours prior to the completion of the repairs to the blowout panels.

In an effort to establish a secondary containment boundary, the refueling floor (613' elev.) has been isolated from the remainder of the Reactor Building by sealing hatchways, doorways, and ventilation systems which penetrate to the floors below or connect Units 2 and 3. As a result of these efforts, the Standby Gas Treatment system has been demonstrated to be capable of maintaining a negative pressure of .2 inches of water in the Unit 2 secondary containment structure (exclusive of the refueling floor). In order to allow startup, a temporary Technical Specification change to allow operation under the above stated conditions is hereby requested. The changes appear as Attachment I to this letter. These changes have received on-site and off-site review and approval.

We believe that operation under these conditions does not create any undue hazard to the health and safety of the public for the following reasons:

1. The probability of the occurrence of a design basis event requiring the secondary containment boundary during the time the temporary change is in effect is very small.

2. With the exception of the refueling floor areas, a secondary containment boundary for Unit 2 will be in effect until total secondary containment is in effect; no fuel movements or operations involving the fuel pools will be performed.

3. Unit 3 is in cold shutdown and Unit 2 is limited to 84% power due to operation of the unit in the coastdown mode, making previous accident analyses performed at 100% power conservative.

Based on the above, we believe that operation under the temporary Technical Specification change causes no undue hazard to the health and safety of the public.

A-126

Pursuant to 10 CFR 170, Commonwealth Edison has determined this to be a Class III Amendment and a fee remittance of $4,000.00 is enclosed.

Please direct any questions concerning this matter to this office.

Three (3) signed originals and thirty-seven (37) copies of this letter are provided for your use.

Very truly yours,

Cordell Reed
Assistant Vice-President

Attachment

SUBSCRIBED and SWORN to
before me this ___ day
of ________________, 1979.

____
Notary Public

A-127

operation except when all of the following conditions are met.

a. The reactor is subcritical and Specification 3.3.A is met.

b. The reactor water temperature is below 212°F and the reactor coolant system is vented.

c. No activity is being performed which can reduce the shutdown margin below that specified in Specification 3.3.A.

d. The fuel cask or irradiated fuel is not being moved in the reactor building.

2. The doors of the core spray and LPCI pump compartments shall be closed at all times

a. A preoperational secondary containment capability test shall be conducted after isolating the reactor building and placing either standby gas treatment system filter train in operation. Such tests shall demonstrate the capability to maintain a 1/4 inch of water vacuum under calm wind (<5 mph) conditions with a filter train flow rate of not more than 4000 cfm.

b. Additional tests shall be performed during the first operating cycle under an adequate number of different environmental wind conditions to enable valid extrapolation of the test results.

c. Secondary containment capability to maintain a 1/4 inch of water vacuum under calm wind (<5 mph) conditions with a filter train flow rate of not more than 4000 cfm, shall be demonstrated at each refueling outage prior to refueling.

d. For the 24 hour period commencing on February 4, 1979 at 1:00 P.M., secondary containment integrity shall be demonstrated by the ability to maintain 0.2 inches of water negative pressure in the Unit 2 Reactor building areas below the refueling floor.

2. Whenever the LPCI and core spray subsystems are required to be operable, the

A-128

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

February 27, 1979

.ket No. 50-237

Mr. Cordell Reed
Assistant Vice President
Commonwealth Edison Company
Post Office Box 767
Chicago, Illinois 60690

Dear Mr. Reed:

The Commission has issued the enclosed Amendment No. 40 to Provisional Operating License No. DPR-19 for Dresden Nuclear Power Station, Unit No. 2. The amendment is in response to your request of February 4, 1979. You were previously notified of this license amendment by telephone and letter on February 4, 1979.

The amendment revises the Technical Specifications to permit operation of the reactor for a period of 24 hours from 1:00 p.m. on February 4, 1979, with a negative pressure of 0.2 inches of water maintained in areas of the Reactor Building below the refueling floor.

Copies of our related Safety Evaluation and the Notice of Issuance are also enclosed.

Sincerely,

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 40 to DPR-2
2. Safety Evaluation
3. Notice of Is[illegible]nce

cc w/enclosures:
See next page

A-129

1028 [illegible]

cc w/enclosures:
Mr. John W. Rowe
Isham, Lincoln & Beale
Counselors at Law
One First National Plaza, 42nd Floor
Chicago, Illinois 60603

Mr. B. B. Stephenson
Plant Superintendent
Dresden Nuclear Power Station
Rural Route #1
Morris, Illinois 60450

Anthony Z. Roisman
Natural Resources Defense Council
917 15th Street, N. W.
Washington, D. C. 20005

U. S. Nuclear Regulatory Commission
ATTN: Jimmy L. Barker
P. O. Box 706
Morris, Illinois 60450

Susan N. Sekuler
Assistant Attorney General
Environmental Control Division
188 W. Randolph Street
Suite 2315
Chicago, Illinois 60601

Morris Public Library
604 Liberty Street
Morris, Illinois 60451

Chairman
Board of Supervisors of
Grundy County
Grundy County Courthouse
Morris, Illinois 60450

*Department of Public Health
ATTN: Chief, Division of
Nuclear Safety
535 West Jefferson
Springfield, Illinois 62761

Director, Technical Assessment
Division
Office of Radiation Programs
(AW-459)
U. S. Environmental Protection
Agency
Crystal Mall #2
Arlington, Virginia 20460

U. S. Environmental Protection
Agency
Federal Activities Branch
Region V Office
ATTN: EIS COORDINATOR
230 South Dearborn Street
Chicago, Illinois 60604

*(w/cy. of incoming dtd. 2/4/79)

POOR
ORIGINAL

A-130

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

COMMONWEALTH EDISON COMPANY

DOCKET NO. 50-237

DRESDEN NUCLEAR POWER STATION UNIT NO. 2

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 40
License No. DPR-19

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by the Commonwealth Edison Company (the licensee) dated February 4, 1979, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Provisional Operating License No. DPR-19 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendix A, as revised through Amendment No. 40, are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance: February 27, 1979

ATTACHMENT TO LICENSE AMENDMENT NO. 40

PROVISIONAL OPERATING LICENSE NO. DPR-19

DOCKET NO. 50-237

Change the Technical Specifications contained in Appendix A by removing Page 120 and inserting the enclosed Page 120. The revised page contains the captioned amendment number and a vertical line indicating the area of change.

A-133

1028 339

3 LIMITING CONDITION FOR OPERATION	4.7 SURVEILLANCE REQUIREMENT
operation except when all of the following conditions are met.	
a. The reactor is subcritical and Specification 3.3.A is met.	a. A preoperational secondary containment capability test shall be conducted after isolating the reactor building and placing either standby gas treatment system filter train in operation. Such tests shall demonstrate the capability to maintain a 1/4 inch of water vacuum under calm wind (<5 mph) conditions with a filter train flow rate of not more than 4000 cfm.
b. The reactor water temperature is below 212°F and the reactor coolant system is vented.	b. Additional tests shall be performed during the first operating cycle under an adequate number of different environmental wind conditions to enable valid extrapolation of the test results.
c. No activity is being performed which can reduce the shutdown margin below that specified in Specification 3.3.A.	c. Secondary containment capability to maintain a 1/4 inch of water vacuum under calm wind (<5 mph) conditions with a filter train flow rate of not more than 4000 cfm, shall be demonstrated at each refueling outage prior to refueling.
d. The fuel cask or irradiated fuel is not being moved in the reactor building.	d. **For the 24 hour period commencing on February 4, 1979, at 1:00 p.m., reactor operation is permitted provided that a negative pressure of 0.2 inches of water is maintained in the Unit 2 reactor building areas below the refueling floor.**
2. The doors of the core spray and LPCI pump compartments shall be closed at all times	2. Whenever the LPCI and core spray subsystems are required to be operable, the

POOR ORIGINAL

A-134

1028

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 40 TO PROVISIONAL OPERATING LICENSE NO. DPR-19

COMMONWEALTH EDISON COMPANY

DRESDEN UNIT NO. 2

DOCKET NO. 50-237

Introduction

By letter received on February 4, 1979, Commonwealth Edison Company (CECo) proposed an amendment to the Dresden Unit No. 2 Technical Specifications. The proposed change requested authorization to operate Dresden Unit No. 2 after a negative pressure of 0.2 inches of water was established in the Reactor Building areas below the refueling floor to demonstrate secondary containment integrity. This negative pressure was maintained in the Reactor Building area for 24 hours following initiation at 1:00 p.m. on February 4, 1979.

Evaluation

The existing Dresden Unit No. 2 Specification 4.7.C.1.c requires that the secondary containment be capable of maintaining 0.25 inches of water vacuum at each refueling outage prior to refueling.

On February 2, 1979, several blowout panels on the Reactor Building for Dresden Unit Nos. 2 and 3 became detached as a result of a ventilation system malfunction that pressurized the Reactor Building. The over-pressurization caused the blowout panels to fail, which resulted in a 22' x 40' opening in the Reactor Building superstructure at the refueling floor level.

We reviewed CECo's request and obtained additional related information from licensee representatives by a telephone conversation on February 4, 1979. The following factors related to the proposed change were considered in our review:

1. The Reactor Building air space has been isolated from the refueling floor air space by temporary barriers. With these barriers in place, CECo is able to maintain a negative pressure of 0.2 inches of water in the Reactor Building exclusive of the refueling floor by operating the standby gas treatment system.

2. The secondary containment performance requirement (0.25 inches of water negative pressure) was selected to prevent leakage from the Reactor Building caused by localized areas of wind-induced negative pressure. These areas are located in the upper corners of the lee side of the Reactor Building refueling floor during high winds. These areas are now isolated from the Reactor Building volume by temporary barriers. We have determined that a negative pressure of 0.2 inches of water in the balance of the Reactor Building (excluding the refueling floor) is adequate to prevent exfiltration from the building.

3. The accident analysis for Dresden assumes that accidents occur under stagnant meteorological conditions with a low wind velocity. The current wind conditions present at Dresden Station involve 10 to 15 mile per hour winds with turbulent flow.

4. Most of the primary containment penetrations are located in the Reactor Building below the refueling floor and do not communicate directly with the refueling floor air space. Therefore, in the event of a LOCA, the leakage through the containment penetrations (a) will be contained within the portion of the Reactor Building secondary containment that maintains its integrity and (b) will be processed through the standby gas treatment system.

5. CECo has committed to stop all fuel handling and cask handling activities on the refueling floor until secondary containment integrity is restored by repairing the breach in the Reactor Building wall above the refueling floor level.

6. CECo has indicated that they are unable to purchase sufficient replacement power to prevent system voltage reduction and possible subsequent load shedding in the area.

We have considered the preceding factors and have concluded that CECo's procedure to maintain Reactor Building integrity (as previously described) provides an equivalent level of protection against the exfiltration of airborne radioactive material should a LOCA occur within the 24 hours that this Technical Specification is in effect.

We have further concluded that the proposed operation is in the public interest because of possible load shedding in the area that might be required if Dresden Unit No. 2 is not available during this 24-hour period.

The NRC staff has determined that an additional measure of protection will be afforded the public by the continuous operation of the standby gas treatment system during the period that this specification is in effect. We have discussed this action with CECo and they have agreed to operate the system continuously. We have modified the proposed specification to require that the negative pressure be maintained continuously.

In evaluating the above considerations, we have concluded that the proposed change, as modified by the NRC staff, is acceptable.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact and pursuant to 10 CFR §51.5(d)(4) that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) because the amendment does not involve a significant increase in the probability or consequences of accidents previously considered and does not involve a significant decrease in a safety margin, the amendment does not involve a significant hazards consideration, (2) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (3) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: February 27, 1979

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-237

COMMONWEALTH EDISON COMPANY

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 40 to Provisional Operating License No. DPR-19, issued to the Commonwealth Edison Company (the licensee), which revised the Technical Specifications for operation of Unit No. 2 of Dresden Nuclear Power Station (the facility) located in Grundy County, Illinois. The license amendment is effective as of its date of issuance.

The amendment revises the Technical Specifications to permit operation of the reactor for a period of 24 hours from 1:00 p.m. on February 4, 1979, with a negative pressure of 0.2 inches of water in areas of the Reactor Building below the refueling floor.

The application for the amendment complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment. Prior public notice of this amendment was not required since the amendment does not involve a significant hazards consideration.

The Commission has determined that the issuance of this amendment will not result in any significant environmental impact and that pursuant to 10 CFR §51.5(d)(4) an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated February 4, 1979, (2) Amendment No. 40 to License No. DPR-19, and (3) the Commission's related Safety Evaluation. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D. C. 20555 and at the Morris Public Library, 604 Liberty Street, Morris, Illinois 60451. A single copy of items (2) and (3) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, this 27th day of February, 1979.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

A-139

ACCESSION NBR:7902130147 DOC.DATE: 79/02/04 NOTARIZED: NO DOCKET #
FACIL:50-237 Dresden Nuclear Power Station, Unit 2, Commonwealth E 05000237
AUTH.NAME AUTHOR AFFILIATION
REED,C. Commonwealth Edison Co.
RECIP.NAME RECIPIENT AFFILIATION
Office of Nuclear Reactor Regulation

6.1-2

SUBJECT: Describes 790202 incident in which several reactor bldg blow off panels became detached. Requests approval of temporary Tech Spec change to allow operation w/in secondary containment boundary. Believes no hazard would be created.

DISTRIBUTION CODE: A001S COPIES RECEIVED:LTR 3 ENCL 3 SIZE: 4
TITLE: GENERAL DISTRIBUTION FOR AFTER ISSUANCE OF OPERATING LIC

NOTES: [illegible] CHECK FOR $ [illegible]

	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL	RECIPIENT ID CODE/NAME	COPIES LTTR	COPIES ENCL
ACTION:	05 BC ORB #2	7	7			
INTERNAL:	01 REG FILE	1	1	02 NRC PDR	1	1
	12 I&E	2	2	14 TA/EDO	1	1
	15 CORE PERF BR	1	1	16 AD SYS/PROJ	1	1
	17 ENGR BR	1	1	18 REAC SFTY BR	1	1
	19 PLANT SYS BR	1	1	20 EEB	1	1
	21 EFLT TRT SYS	1	1	22 BRINKMAN	1	1
EXTERNAL:	03 LPDR	1	1	04 NSIC	1	1
	23 ACRS	16	16			

VERY POOR ORIGINAL

TOTAL NUMBER OF COPIES REQUIRED: LTTR 38 ENCL 38

A-140

1028 346

6.1-2

IMPOSITION OF CIVIL PENALTIES

REGULATIONS AND CRITERIA GOVERNING

POOR ORIGINAL

Subpart B cedure for Imposing Requirements by Order, or for Modification, Suspension, or Revocation of a License, or for Imposing Civil Penalties

§ 2.200 Scope of subpart.

(a) This subpart prescribes the procedures in cases initiated by the staff, or upon a request by any person, to impose requirements by order on a licensee or to modify, suspend, or revoke a license, or for such other action as may be proper.

(b) This subpart also prescribes the procedures in cases initiated by the staff to impose civil penalties pursuant to section 234 of the Act and section 206 of the Energy Reorganization Act of 1974.

[36 FR 16896, Aug. 26, 1971, as amended at 39 FR 12353, Apr. 5, 1974; 42 FR 28893, June 6, 1977]

§ 2.201 Notice of violation.

A-142

(a) Before instituting any proceeding to modify, suspend, or revoke a license or to take other action for alleged violation of any provision of the Act or this chapter or the conditions of the license the Director, Office of Inspection and Enforcement, will serve on the licensee a written notice of violation, except as provided in paragraph (c) of this section. The notice of violation will concisely state the alleged violation and will require that the licensee submit, within twenty (20) days of the date of the notice or other specified time, a written explanation or statement in reply including:

(1) Corrective steps which have been taken by the licensee, and the results achieved;

(2) Corrective steps which will be taken; and

(3) The date when full compliance will be achieved.

(b) The notice may require the licensee to admit or deny the violation and to state the reasons for the violation, if admitted. It may provide that, if an adequate reply is not received within the time specified in the notice, the Director, Office of Inspection and Enforcement, may issue an order to show cause why the license should not be modified, suspended or revoked or such other action be taken as may proper.

(c) When the Director, Office of Inspection and Enforcement, finds that the public health, safety, or interest so requires, or that the violation is willful, the notice of violation may be omitted and an order to show cause issued.

[27 FR 10836, Nov. 7, 1962]

§ 2.202 Order to show cause.

(a) The Director of Nuclear Reactor Regulation, Director of Nuclear Material Safety and Safeguards, Director, Office of Inspection and Enforcement, as appropriate, may institute a proceeding to modify, suspend, or revoke a license or for such other action as may be proper by serving on the licensee an order to show cause which will:

(1) Allege the violations with which the licensee is charged, or the potentially hazardous conditions or other facts deemed to be sufficient ground for the proposed action;

(2) Provide that the licensee may file a written answer to the order under oath or affirmation within twenty (20) days of its date, or such other time as may be specified in the order;

(3) Inform the licensee of his right, within twenty (20) days of that date of the order, or such other time as may be specified in the order, to demand a hearing;

(4) Specify the issues; and

(5) State the effective date of the order.

(b) A licensee may respond to an order to show cause by filing a written answer under oath or affirmation. The answer shall specifically admit or deny each allegation or charge made in the order to show cause, and may set forth the matters of fact and law on which the licensee relies. The answer may demand a hearing.

(c) If the answer demands a hearing, the Commission will issue an order designating the time and place of hearing.

(d) An answer or stipulation may consent to the entry of an order in substantially the form proposed in the order to show cause.

(e) The consent of the licensee to the entry of an order shall constitute a waiver by the licensee of a hearing, findings of fact and conclusions of law, and of all right to seek Commission and judicial review or to contest the validity of the order in any forum. The order shall have the same force and effect as an order made after hearing by a presiding officer or the Commission.

(f) When the Director of Nuclear Reactor Regulation, Director of Nuclear Material Safety and Safeguards, Director, Office of Inspection and Enforcement, as appropriate, finds that the public health, safety, or interest so requires or that the violation is willful, the order to show cause may provide, for stated reasons, that the proposed action be temporarily effective pending further order.

[27 FR 377, Jan. 13, 1962, as amended at 28 FR 10153, Sept. 17, 1963]

§ 2.203 Settlement and compromise.

At any time after the issuance of an order designating the time and place of hearing in a proceeding to modify, suspend, or revoke a license or for other action, the staff and a licensee or other person may enter into a stipulation for the settlement of the proceeding or the compromise of a civil penalty. The stipulation or compromise shall be subject to approval by the designated presiding officer or, if none has been designated, by the Chief Administrative Law Judge, according due weight to the position of the staff. The presiding officer, or if none has been designated, the Chief Administrative Law Judge, may order such adjudication of the issues as he may deem to be required in the public interest to dispose of the proceeding. If approved, the terms of the settlement or compromise shall be embodied in a decision or order settling and discontinuing the proceeding.

[36 FR 16896, Aug. 26, 1971]

§ 2.204 Order for modification of license.

The Commission may modify a license by issuing an amendment on notice to the licensee that he may demand a hearing with respect to all or any part of the amendment within twenty (20) days from the date of the notice or such longer period as the notice may pr The amendment will become effe he expiration of the period dur ch the licensee may demand a hearing, or, in the event that he demands a hearing, on the date specified in an order made following the hearing. When the Commission finds that the public health, safety, or interest so requires, the order may be made effective immediately.

[28 FR 10153, Sept. 17, 1963]

§ 2.205 Civil penalties.

(a) Before instituting any proceeding to impose a civil penalty under section 234 of the Act, the Director of Nuclear Reactor Regulation, Director of Nuclear Material Safety and Safeguards, Director, Office of Inspection and Enforcement, as appropriate shall serve a written notice of violation upon the person charged. This notice may be included in a notice issued pursuant to § 2.201. The notice of violation shal specify the date or dates, facts, an the nature of the alleged act or omission with which the person is charged and shall identify specifically the particular provision or provisions of th law, rule, regulation, license, permit or cease and desist order involved i the alleged violation and shall stat the amount of each penalty which th Director of Nuclear Reactor Regulation, Director of Nuclear Materi Safety and Safeguards, Directo Office of Inspection and Enforcemen as appropriate proposes to impos The notice of violation shall als advise the person charged that th civil penalty may be paid in th amount specified therein, or the proposed imposition of the civil penalt may be protested in its entirety or i part, by a written answer, either deny ing the violation, or showing extenua ing circumstances. The notice of viol tion shall advise the person charge that upon failure to pay a civil penalt subsequently determined by the Com mission, if any, the penalty ma unless compromised, remitted or mit gated, be collected by civil action, pu suant to section 234c of the Act.

(b) Within twenty (20) days of th date of a notice of violation or oth time specified in the notice, the pers

POOR ORIGINAL

charged [illegible] ther pay the penalty in the amou[illegible] roposed or answer the notice of violation. The answer to the notice of violation shall state any facts, explanations, and arguments, denying the charges of violation, or demonstrating any extenuating circumst[illegible]ces, error in the notice of violation, or other reason why the penalty should not be imposed and may request remission or mitigation of the penalty.

(c) If the person charged with violation fails to answer within the time specified in paragraph (b) of this section, the Director of Nuclear Reactor Regulation, Director of Nuclear Material Safety and Safeguards, Director, Office of Inspection and Enforcement, as appropriate, will issue an order imposing the civil penalty in the amount set forth in the notice of violation described in paragraph (a) of this section.

(d) If the person charged with violation files an answer to the notice of violation, the Director of Nuclear Reactor Regulation, Director of Nuclear Material Safety and Safeguards, Director, Office of Inspection and Enforcement, as appropriate, upon consideration of the answer, will issue an order dismissing the proceeding or imposing, mitigating, or remitting the civil penalty. The person charged may, within twenty (20) days of the date of the order or other time specified in the order, request a hearing.

(e) If the person charged with violation requests a hearing, the Commission will issue an order designating the time and place of hearing.

(f) If a hearing is held, an order will be issued after the hearing by the presiding officer or the Commission dismissing the proceeding or imposing, mitigating, or remitting the civil penalty.

(g) The Director of Nuclear Reactor Regulation, Director of Nuclear Material Safety and Safeguards, Director, Office of Inspection and Enforcement, as appropriate, may compromise any civil penalty, subject to the provisions of § 2.203.

(h) If the civil penalty is not compromised, or is not remitted by th[illegible] rector of Nuclear Reactor Regula[illegible] Director of Nuclear Material Safety and Safeguards, Director, Office of Inspection and Enforcement, as appropriate, the presiding officer or the Commission, and if payment is not made within ten (10) days following either the service of the order described in paragraph (c) or (f) of this section, or the expiration of the time for requesting a hearing described in paragraph (d) of this section, no such request having been made, the Director of Nuclear Reactor Regulation, Director of Nuclear Material Safety and Safeguards, Director, Office of Inspection and Enforcement, as appropriate, may refer the matter to the Attorney General for collection.

(i) Except when payment is made after compromise or mitigation by the Department of Justice or as ordered by a court of the United States, following reference of the matter to the Attorney General for collection, payment of civil penalties imposed under section 234 of the Act shall be made by check, draft, or money order payable to the Treasurer of the United States, and mailed to the Director of Nuclear Reactor Regulation, Director of Nuclear Material Safety and Safeguards, Director, Office of Inspection and Enforcement, as appropriate.

[36 FR 16896, Aug. 26, 1971; 36 FR 18173, Sept. 10, 1971]

§ 2.206 Requests for action under this subpart.

(a) Any person may file a request for the Director of Nuclear Reactor Regulation, Director of Nuclear Material Safety and Safeguards, Director, Office of Inspection and Enforcement, as appropriate, to institute a proceeding pursuant to § 2.202 to modify, suspend or revoke a license, or for such other action as may be proper. Such a request shall be addressed to the Director of Nuclear Reactor Regulation, Director of Nuclear Material Safety and Safeguards, Director, Office of Inspection and Enforcement, as appropriate, and shall be filed either (1) by delivery to the Public Document Room at 1717 H Street NW., Washington, D.C., or (2) by mail or telegram addressed to the Director of Nuclear Reactor Regulation, Director of Nuclear Material Safety and Safeguards, Director, Office of Inspection and Enforcement, as appropriat[illegible] U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. The requests shall specify the action requested and set forth the facts that constitute the basis for the request.

(b) Within a reasonable time after a request pursuant to paragraph (a) of this section has been received, the Director of Nuclear Reactor Regulation, Director of Nuclear Material Safety and Safeguards, Director, Office of Inspection and Enforcement, as appropriate shall either institute the requested proceeding in accordance with this subpart or shall advise the person who made the request in writing that no proceeding will be instituted in whole or in part, with respect to his request, and the reasons therefor.

(c) (1) Director's decisions under this section will be filed with the Office of the Secretary. Within twenty (20) days after the date of a Director's decision under this section that no proceeding will be instituted or other action taken in whole or in part, the Commission may on its own motion review that decision, in whole or in part, to determine if the Director has abused his discretion. This review power does not limit in any way either the Commission's supervisory power over delegated Staff actions or the Commission's power to consult with the Staff on a formal or informal basis regarding institution of proceedings under this section.

(2) No petition or other request for Commission review of a Director's decision under this section will be entertained by the Commission.

(Sec. 201 as amended Pub. L. 93-438, 88 Stat. 1242; Pub. L. 94-79, 89 Stat. 413 (42 U.S.C. 5841))

[39 FR 12353, Apr. 5, 1974, as amended at 42 FR 36240, July 14, 1977]

Subpart [illegible] served]

Subpart D—Additional Procedu[illegible] Applicable to Proceedings for [illegible] Issuance of Licenses To Constru[illegible] or Operate Nuclear Power Plan[illegible] of Duplicate Design at Multip[illegible] Sites

Source: 40 FR 2976, Jan. 17, 1975, unl[illegible] otherwise noted.

§ 2.400 Scope of subpart.

This subpart describes procedur[illegible] applicable to licensing proceedin[illegible] which involve the consideration [illegible] hearings of a number of application[illegible] filed by one or more applicants pursu[illegible] ant to Appendix N of Part 50 of th[illegible] chapter, for licenses to construct an[illegible] operate nuclear power reactors of e[illegible] sentially the same design to be locate[illegible] at different sites.

§ 2.401 Notice of hearing on application[illegible] pursuant to Appendix N of Part 50 f[illegible] construction permits.

(a) In the case of applications pursu[illegible] ant to Appendix N of Part 50 of thi[illegible] chapter for construction permits fo[illegible] nuclear power reactors of the type de[illegible] scribed in § 50.22 of this chapter, th[illegible] Secretary will issue notices of hearin[illegible] pursuant to § 2.104.

(b) The notice of hearing will als[illegible] state the time and place of the hear[illegible] ings on any separate phase of the pro[illegible] ceeding.

§ 2.402 Separate hearings on separat[illegible] issues; consolidation of proceedings.

(a) In the case of applications pursu[illegible] ant to Appendix N of Part 50 of thi[illegible] chapter for construction permits fo[illegible] nuclear power reactors of a type de[illegible] scribed in § 50.22 of this chapter, the Commission or the presiding officer may order separate hearings on particular phases of the proceeding, such as matters related to the acceptability of the design of the reactor, in the context of the site parameters postulated for the design; environmental matters; or antitrust aspects of the application.

(b) If a separate hearing is held on a particular phase of the proceeding, the Commission may, pursuant to § 2.716, consolidate for hearing on that

A-14-3

1028 349

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

SECY-77-491

INFORMATION REPORT

For: The Commissioners

From: Ernst Volgenau, Director
Office of Inspection and Enforcement

Thru: for Executive Director for Operations

Subject: CRITERIA FOR ISSUANCE OF CIVIL PENALTIES AND APPEAL PROCESS AVAILABLE TO THE LICENSEE

Purpose: This responds to questions raised by the Commission on August 11, 1977,* concerning the procedures involved in determining the need for and the process of issuing a civil penalty; the process by which a licensee may appeal an adverse enforcement action; the mechanism for convening a Hearing Board; and the extent of the Commission's involvement at each stage of the appeal process.

Discussion: The Director of the Office of Inspection and Enforcement, the Director of the Office of Nuclear Reactor Regulation, and the Director of the Office of Nuclear Material Safety and Safeguards have the authority to impose civil penalties, where appropriate. However, the Director of the Office of Nuclear Material Safety and Safeguards and the Director of the Office of Nuclear Reactor Regulation have not previously exercised this authority.

1. Need for Civil Penalty Action

There are three formal sanctions available to the Commission in the exercise of its enforcement responsibility. These three sanctions are (1) notices of violation, (2) civil penalties, and (3) orders of various kinds such as orders to modify, suspend or revoke licenses and orders directing cessation of specified activities. Each of these sanctions is described in the attached "Criteria for Determining Enforcement Action and Categories of Items of Noncompliance with AEC Regulatory Requirements - Modifications" (Criteria for Enforcement Action), issued December 31, 1974. This document is sent to all licensees and was noticed in the Federal Register. 40 FR 820 (January 3, 1975). (Copy attached as

POOR ORIGINAL

Contact:
T. W. Brockett, IE
?-27246

*Budget Session, 8/11/77 (ref: SECY memo to EDO, OGC, 8/17/77)

A-141 Sheet 1 of 33
Attachment 3

Appendix A.) The Office of Inspection and Enforcement's Manual Chapter 0800, available in the Public Document Room, provides guidance to the staff in its implementation of the Criteria for Enforcement Action.

The sanction for a given case is selected in accordance with the Criteria for Enforcement Action and the guidance of MC 0800. The determination of the appropriate sanction is in the last analysis, however, a matter of judgment exercised in accordance with the established criteria and the guidance in MC 0800. The specific action decided upon is dependent on the facts and circumstances of each particular case. Factors bearing upon selection of the appropriate enforcement action include the total items of noncompliance, the significance of each individual item of noncompliance and the licensee's previous enforcement history. In selecting the appropriate sanction, emphasis is on corrective action and management controls to assure continued compliance as distinguished from purely punitive action.

The Criteria for Enforcement Action outline eleven examples for which a civil penalty may be the appropriate sanction. Thus a civil penalty is considered where repetitive items of noncompliance with the same general requirement have been noted, where chronic noncompliance is found, where noncompliance has been deliberate, and also where a single instance of noncompliance of the significance level of a "violation"* occurs. Orders are issued in instances of unauthorized uses or activities; where an immediate hazard exists regardless of whether there may be any associated noncompliance with regulatory requirements; in other instances where serious potential safety, security or environmental hazards must be removed; in instances where other enforcement actions have not been effective; in instances where deliberate violations have occurred; or in other similar instances.

*Items of noncompliance have been categorized into three levels of significance: "violation" (most significant), "infraction", and "deficiency" (least significant). These categories are elaborated on in Attachment B to Appendix A.

The legislative history of Section 234 of the Atomic Energy Act of 1954, as amended, is the Commission authority for imposing civil penalties. This authority was intended to be exercised for items of noncompliance which are too significant for a mere notice of violation and yet not significant enough to warrant the suspension or revocation of a license. A brief summary of the legislative history is set forth in Appendix B.

2. Process for Issuing Civil Penalties

Procedures for the civil penalty action are found in 10 CFR 2.205 of the Commission's "Rules of Practice." After consideration of the various factors discussed above and a decision to issue a proposed civil penalty, a notice of violation is prepared citing the specific items of noncompliance and the sections of the regulations in Title 10, Code of Federal Regulations, or license conditions with which the licensee was found to be in apparent noncompliance. Each item of noncompliance is classified as a violation, infraction or deficiency and a dollar amount is assigned to each classification in accordance with guidance in MC 0800. A Notice of Proposed Imposition of Civil Penalties is also prepared by IE with concurrence of the Office of the Executive Legal Director. The appropriate Licensing Office is informed of the proposed action early in the consideration process. A graphic portrayal of the process is contained in Appendix C

3. Timing of Commission and Public Notification

After the proposed civil penalty is signed by the Director of the Office of Inspection and Enforcement, a Notice of Significant Enforcement Action (EN) is dispatched to the Commission five days in advance of the date for mailing the civil penalty notice to the licensee. The Office of Public Affairs is notified also so that a press release can be made, usually two days following the dispatch of the civil penalty notice to the licensee.

4. Process by Which Licensee May Appeal

POOR ORIGINAL

A-146

a. Proposed Civil Penalty Action

The licensee is given twenty days from the date of receipt of the proposed civil penalty notice to respond. If no response is received in the twenty day period the penalties will be imposed in the proposed amount. The licensee may protest the imposition of the penalties in whole or in part. If he chooses to protest the penalties he may (a) deny the items of noncompliance listed in the Notice of Violation, (b) demonstrate extenuating circumstances, (c) show error in the Notice of Violation, or (d) show other reasons why the penalties should not be imposed. He may also request remission or mitigation of the penalties.

When the licensee's response to the Notice of Violation and the Notice of Proposed Imposition of Civil Penalties is received in IE Headquarters, copies are sent to the responsible Regional Office, the appropriate licensing office, the Office of the Executive Legal Director and the Office of Public Affairs. After a review of the licensee's response, IE will either issue an order dismissing the proposed penalty or impose, mitigate or remit the civil penalties.

b. Order Imposing the Civil Penalties

POOR ORIGINAL

Upon receipt of the Order Imposing Civil Penalties, the licensee may, within twenty days, pay the civil penalties or request a hearing on the order. A number of licensees have requested hearings on civil penalty matters, however, these matters have usually been resolved in the prehearing stage. Only three cases have gone beyond the prehearing stage and these are currently in the hearing process.* It should be noted that if a hearing is requested, a hearing must be granted. The mechanism for convening a hearing is the issuance by the Commission of a Notice of Hearing.

*In addition there was the Virginia Electric and Power Company matter involving a civil penalty. There as a result of a stipulation by all parties the order imposing the civil penalty was initiated by the Atomic Safety and Licensing Board rather than the staff. This proceeding was unique and is not pertinent to the present discussion.

A-147

Sheet 4 of 22

As dis ussed below the hearing is held by either the Administrative Law Judge or the Atomic Safety and Licensing Board (ASLB). The decision of the Administrative Law Judge or the ASLB, as appropriate, is appealable to the Appeal Board at the request of either the licensee or the staff. The Appeal Board's decision may in turn be reviewed by the Commission at its discretion. Once the decision becomes final, a licensee may seek redress in the courts.

If payment is not made within the specified time following either the service of an order or the expiration of the time for requesting a hearing, the matter may be referred to the Attorney General for collection through a civil action in District Court. Under Section 234 of the Act, a licensee may refuse payment and the matter may be processed directly in District Court without going through the Commission's administrative process. The Attorney General has the exclusive power to compromise, mitigate or remit civil penalties which have been referred to him for action.

5. <u>Mechanism for Convening a Hearing Board</u>

Normally the Administrative Law Judge (Judge Samuel Jensch) is designated to hear civil penalty cases. The Administrative Law Judge is designated rather than a three man Atomic Safety and Licensing Board because the issues in controversy in these cases tend to be narrow and factual rather than broad and technically complex, as is typical of a reactor licensing proceeding.

The Commission itself appoints the Administrative Law Judge or Board to hear a civil penalty case. It does so in the Notice of Hearing, having acted on a draft of that Notice presented to it by the staff. The Commission also typically uses the Notice of Hearing as the means for authorizing the Appeal Board to perform a review function in the proceeding.

6. <u>Extent of the Commission's Involvement at each Stage of the Appeal Process</u>

As indicated above, the decision of the Appeal Board would be reviewable by the Commission at its discretion, therefore the Commission's <u>ex parte</u> rule limits Commission involvement in a civil penalty matter once a hearing has been requested.

A-118

Coordination: The Office of the Executive Legal Director concurs in this paper. The Office of the General Counsel has no legal objection.

Ernst Volgenau
Director
Office of Inspection
and Enforcement

Enclosures:
Appendices A, B, C

DIS UTION
Com ioners
Commission Staff Offices
Exec Dir for Operations
Regional Offices
ACRS
AS&LBP
AS&LAP

d-2). 9-11-77

UNITED STATES
ATOMIC ENERGY COMMISSION
WASHINGTON, D.C. 20545

December 31, 1974

To: All AEC Licensees

CRITERIA FOR DETERMINING ENFORCEMENT ACTION AND CATEGORIES OF NONCOMPLIANCE WITH AEC REGULATORY REQUIREMENTS - MODIFICATIONS

On November 1, 1972, the Commission issued criteria for enforcement actions to be taken for noncompliance with its rules and with license conditions in accordance with Sections 161, 186, and 234 of the Atomic Energy Act and Subpart B of Part 2, 10 CFR. On June 5, 1973, the Commission notified licensees that categories of violation with AEC regulatory requirements had been established because the Commission and the nuclear industry recognized that the significance of violations varies in the potential for affecting the health and safety of the public, the common defense and security, and the environment.

Based on a review of the experience with the criteria for determining enforcement action and the categories of noncompliance, modifications of the use of these criteria and these categories are being made. Comments explaining the modifications are enclosed as Attachments A and B.

The changes in the criteria and categories are primarily administrative in nature and should result in a higher level of understanding of the enforcement program - and the results of the program - on the part of the public and the industry. The basic purpose of the enforcement program - enhancement of the health and safety of the public, the common defense and security, and the environment - remains the same. The long standing practice of requiring corrective action for each identified item of noncompliance (Violations) is not changed. The enforcement program continues to emphasize corrective action where necessary to assure that regulated activities meet applicable requirements and are conducted with due regard for public health and safety, common defense and security and protection of the environment.

The modifications clarify the enforcement criteria and categories of noncompliance in the areas of safeguards and environmental matters and provide more explicit definitions to aid in a better understanding of the enforcement program. These definitions make clear the applicability of the program in matters of quality assurance, management control, and systems performance. Also, because the Commission relies to a degree on reports from licensees to assure that timely corrective action is taken and to assure that the industry is notified of important matters

APPENDIX A

POOR ORIGINAL

A-150

Sheet 7 of 22

of generic interest, a reporting requirement is viewed from the enforcement standpoint to be of the same level of importance as the matter for which the report is required. As a part of the correspondence between a licensee and the AEC subsequent to an inspection, notifications will be made to a licensee of apparent failures on the part of the licensee to meet his commitments contained in his application or in correspondence to the AEC and of deviations from appropriate codes, standards or guides.

The levels of enforcement actions available to the Commission in the exercise of its regulatory responsibilities are the same as those set forth in the letter of November 1, 1972. These include written notices of violation, civil monetary penalties, and orders to "cease and desist" or for modification, suspension, or revocation of a license.

The criteria for issuance of a "Notice of Violation" are essentially unchanged.

The criteria for civil penalties have been modified to elaborate upon those situations for which civil penalties may be imposed. The amount of civil penalty in any given case, within the confines of the amounts established by the Atomic Energy Act, is determined by consideration of several factors including:

1. Potential or actual consequences associated with the item of noncompliance. This includes consideration of the categories of noncompliance.

2. Type of licensee. This includes the purpose for which licensed and the quantity, form and kind of radioactive material authorized.

3. The licensee's recent enforcement history, if applicable. This includes the nature and number of items of noncompliance, the frequency of noncompliance, whether items of noncompliance were repetitive of the same or similar requirements, promptness of corrective action, and the licensee's management of its program for assuring compliance with regulatory requirements.

The criteria clarify that repetitiveness of noncompliance or history of noncompliance is not an essential ingredient for consideration for civil penalty. In some cases of a single instance of noncompliance, a civil penalty may be the appropriate enforcement sanction.

The criteria for orders emphasize the importance of quality assurance and are broadened to include all aspects of the regulatory program. Under these criteria, an order to suspend a license or a portion thereof may be issued for authorized activities of licensees or permit holders which are performed in such a manner as to constitute an immediate or potential threat to employees or the public; or for construction deficiencies which, if not suspended immediately, could eventually result in significant or essentially irreversible construction defects which impact on safety or which increase the potential for or the potential severity of an accident. If, for example, a quality assurance requirement for a specific construction activity is not implemented, this activity may be suspended until full compliance with the requirement is achieved.

Regulatory Operations Bulletins and Immediate Action Letters have been used not only to disseminate information but also as a means of accomplishing voluntary action on the part of licensees to inspect, report and make commitments to correct problems on a timely schedule. These two communications are recognized in these revisions. If these methods are ineffective in achieving the desired action, an order may be promptly issued requiring the action.

The enforcement record of a licensee may be a consideration in selecting the appropriate enforcement sanction in any given case. A licensee's enforcement history is evaluated in terms of distribution of items of noncompliance by importance and by the degree of repetitiveness of noncompliance with the same basic requirement. However, regardless of the history, consideration will be given to the more significant enforcement sanctions as a result of any inspection that reveals items of particular importance to safety and management.

The former system of severity categorization, which was the subject of a letter to licensees dated June 5, 1973, has been revised to place items of noncompliance with regulatory requirements (Violations) more clearly in perspective with regard to their relative significance to the public health, safety and interest and the common defense and security. As shown in Attachment B to this letter, the revised system for categorizing violations (items of noncompliance) has three levels of relative importance which are designated in descending order as (1) "violation," (2) "infraction," and (3) "deficiency," each of which is a legal violation in the statutory sense.

It should be recognized that the enforcement criteria and the categories of noncompliance apply only to situations where there is an apparent failure on the part of a licensee to meet regulatory requirements. The licensee may also be notified of deviations from commitments and appropriate codes, standards, or guides. The significance of these failures generally is judged against the actual or potential consequences resulting from the failures and from the standpoint of licensee awareness and management of his program. From the viewpoint of enforcement, a licensee failure that results in the potential for consequences is

POOR ORIGINAL

A-152

equally important with the failure that results in the consequencies - both represent instances of failure of the licensee to properly perform. However, from the impact of health and safety, common defense and security, the protection of the environment, actual consequences — when the event did occur — and potential consequences — when the opportunity for occurrences exists but the event did not happen — of a item of noncompliance are quite different. In reporting the more important items of noncompliance, those items that caused or resulted in actual consequences will be differentiated from those that merely provided the potential for the consequences.

The enforcement criteria and the categories of noncompliance apply to situations where there is an apparent failure on the part of a licensee to meet regulatory requirements, commitments, and appropriate codes, standards or guides. There do occur events - such as some equipment malfunctions - at licensee facilities which are not founded in the failure of the licensee to meet requirements, commitments, and appropriate codes, standards, and guides. Such events are not included within the enforcement program.

The enforcement criteria and the categories of noncompliance have been placed in the Public Document Room, 1717 H Street, N.W., Washington, D.C., and a notice has been placed in the Federal Register concerning their availability to all persons upon request.

Sincerely,

Donald F. Knuth

Donald F. Knuth, Director
of Regulatory Operations

Enclosures:
A. Criteria for Determining Enforcement Action
B. Categories of Items of Noncompliance

POOR
ORIGINAL

Sheet 10 of 22

A-153

CRITERIA FOR DETERMINING ENFORCEMENT ACTION

In Connection with Licensing and Regulatory Provisions
of the Atomic Energy Act of 1954, as Amended,
and Regulations and Licenses Issued Thereunder

INTRODUCTION

The purpose of the AEC enforcement program is the enhancement of the health and safety of the public, the common defense and security, and the environment. The enforcement program emphasizes corrective action, where necessary, to assure that regulated activities meet applicable requirements and are conducted with due regard for public health and safety, common defense and security and protection of the environment. Corrective action is required for each identified item of noncompliance.

Results of AEC inspections and investigations of licensed activities have shown that licensees have not in all cases complied with the regulatory requirements, and it has been necessary to take specific enforcement actions commensurate with the items of noncompliance. This document sets out the criteria for enforcement actions to be taken with respect to future noncompliance with the Atomic Energy Commission's requirements in accordance with Sections 161, 186 and 234 of the Atomic Energy Act and Subpart B of Part 2, 10 CFR.

LEVELS OF ENFORCEMENT ACTIONS AVAILABLE TO THE COMMISSION

The formal actions available to the Commission in the exercise of its enforcement responsibilities are of three basic types (notices of violation, civil penalties, and orders) which may be applicable to a specific enforcement situation.

1. Written Notices of Violation (10 CFR 2.201)

   Notices of Violations are written notices to licensees, citing the apparent instances of failure to comply with regulatory requirements (Violations) which for purposes of categorization have been classified violations, infractions and deficiencies. Such items of noncompliance are generally observed or identified during investigations, inspections, or inquiries.

   The same letter enclosing a Notice of Violation may also enclose a notification of apparent deviations from licensee commitments and the provisions of appropriate codes, standards or guides.

2. Civil Monetary Penalties (10 CFR 2.205)

   The Commission may levy civil monetary penalties against licensees for violations, infractions or deficiencies with respect to requirements in licensing provisions of the Act or any rule, regulation,

A-154

Sheet 11 of 22

1028 [illegible]

order, or license issued thereunder. The Commission is required to issue a "notice of violation" to the person charged before instituting proceedings to impose a civil penalty.

3. <u>Orders to Cease and Desist; and Orders for Suspension, Modification, or Revocation of a License (10 CFR 2.202 and 2.204)</u>

The AEC has authority to issue orders to "cease and desist," and orders to suspend, modify, or revoke licenses. Such orders are ordinarily preceded by certain procedural requirements, including a written "notice of violation" to the licensee providing him with an opportunity to respond as to the corrective measures being taken. In the event the licensee fails to respond to the notice or to demonstrate that satisfactory corrective action is being taken, an order to show cause may be issued requiring the licensee to show why the particular order (either of revocation, or modification, or suspension) should not be made effective. In some instances where the health, safety, or interest of employees or the public so requires or deliberate noncompliance with the Commission's regulations is involved, the notice provision may be dispensed with and, in addition, the particular order may be made immediately effective pending further order.

In addition to proceeding by way of order, the Commission may also, pursuant to Section 232 of the Act, request the Attorney General to obtain an injunction or other court order to enjoin licensees from violating the Act or any regulation or order issued thereunder.

<u>NOTICE OF VIOLATION - CRITERIA</u>

Section 2.201 of 10 CFR requires that before any formal enforcement action is taken for alleged noncompliance, the AEC will serve on the licensee a written "notice of violation" except when the Director of Regulation finds that the public health, safety, or interest so requires, or that noncompliance is deliberate, the "notice of violation" may be omitted and an order to show cause issued.

Generally, a "notice of violation" may be considered sufficient enforcement action in those cases where:

a. Items of noncompliance are readily correctable, or

b. Items of noncompliance are not repetitive or numerous, and do not constitute an immediate or serious threat to the health and safety of the licensee's employees or the public, to the environment, or to the common defense and security, and

c. These is no indication that appropriate corrective action will not be taken.

A-155

CIVIL MONETARY PENALTIES - CRITERIA

The Commission may levy civil monetary penalties on licensees who do not comply with the licensing provisions of the Act or any rule, regulation, order, or license issued. Generally, the type of cases that are appropriate for imposing civil penalties are those involving significant items of noncompliance and which represent a threat (but not necessarily immediate) to the health, safety, or interest of the public, or to the common defense or security, or the environment. As a matter of judgment, civil penalties may be used in lieu of license suspension when there is no immediate threat to the health and safety or the common defense and security and license suspension would deprive the licensee or his employees of their means of livelihood, or the public of essential service.

Civil penalties may be the appropriate enforcement action in cases or situations which meet one or more of the following criteria:

a. Those cases of noncompliance with the same basic requirements that were brought to the attention of the licensee in a "notice of violation" following a previous inspection; or

b. Those cases of noncompliance in which the licensee fails to carry out in a timely manner the corrective action the licensee stated would be taken in response to a previous written notice; or

c. Those cases involving the deliberate failure of a person to comply with regulatory requirements;* or

d. Those cases involving items of noncompliance in which (1) the licensee's history is one of chronic noncompliance, or (2) due to the nature and number of items of noncompliance, it is apparent that management, having been afforded an opportunity to correct previous items of noncompliance, is not conducting its licensed activities in conformance with regulatory requirements, or

---

* NOTE: Section 221(b) of the Atomic Energy Act requires the FBI to investigate all suspected or alleged criminal violations of the Act.

POOR ORIGINAL

Sheet 13 of 23

A-156

~~1030 002~~

1028 362

e. Those cases where (1) an order for immediate, but temporary, suspension or to "cease and desist" is issued to remove an immediate threat to the health or safety of the licensee's employees or the public, to the environment or to the common defense and security, and (2) punitive action is deemed necessary to assure future compliance; or

f. Those cases involving activities under construction permits where there are repeated items of noncompliance with regulatory requirements; or

g. Those cases where an item of noncompliance resulted in or contributed to the cause or the seriousness of an accident or an incident; or

h. Those cases involving items of noncompliance in the Violation category; or

i. Those cases where the nature and number of items of noncompliance with the regulatory requirements identified during an inspection or an investigation demonstrate that management is not conducting its licensed activities with adequate concern for the health, safety or interest of its employees or the public or the common defense and security; or

j. Those cases where licensees knowingly use materials which are not authorized by the license or utilize authorized materials for uses which are not authorized; or

k. Those cases where significant matters** were not reported to the Commission in a timely manner as required by the regulatory requirements.

Civil penalties may be assessed for other cases having comparable types of items of noncompliance and situations for which the Commission deems civil penalties to be appropriate and necessary.

** Such significant matters may include, but are not limited to, exposure of personnel to doses in excess of limits, release of radioactive concentrations in effluents in excess of limits, incidents involving an attempt to commit a theft or unlawful diversion of SNM, or to commit an act of sabotage of certain facilities, failure of safety systems, emergency core cooling or other related safety systems to perform their design function, or the MUF of SNM in excess of applicable limits, or similar matters.

IMAGE EVALUATION
TEST TARGET (MT-3)

6"

ORDERS - CRITERIA

The AEC has authority to issue orders to "cease and desist" or to suspend, modify, or revoke licenses. The Commission is empowered to enforce these orders and obtain any other appropriate relief by injunction from Federal district courts, if necessary. Cases involving an immediate threat to the public health and safety, or the common defense and security, require immediate steps to remove the threat and are handled by this type of action. Persons who deliberately violate, attempt to violate, or conspire to violate the Commission's regulations and orders, are, upon conviction of the violations, subject to fine up to $5,000 and imprisonment for not more than two years (Section 223 of the Act).

In the event the licensee fails to respond to a "notice of violation" or to demonstrate that satisfactory corrective action is being taken, an order to show cause may be issued requiring the licensee to show why the particular order (either of revocation, or modification, or suspension) should not be made effective. In those instances where the health, safety, or interest of employees or the public, or the common defense and security so requires, or deliberate noncompliance with the Commission's regulations is involved, the notice provision may be dispensed with and, in addition, the particular order may be made immediately effective pending further order.

a. Orders to Cease and Desist

An order to cease and desist is ordinarily issued when a person is conducting unauthorized activities and has been notified of the need for authorization but fails to terminate the activity and other similar circumstances as appropriate.

b. Orders to Suspend a License

An order is ordinarily issued for immediate suspension of a license, or a portion thereof, as necessary to remove an immediate threat to the health, safety or interest of licensee's employees or the public, or to the common defense and security; or for noncompliance with AEC requirements relating to construction of a facility which, if not corrected immediately, could subsequently result in a significant threat to the health, safety or interest of employees or the public, or the common defense and security.

c. Order to Modify a License

An order for the modification of a license, in whole or in part, is ordinarily issued as an enforcement sanction when it is determined that a licensee's operations or activities must be limited or modified to protect the health, safety, or interest of the licensee's employees or the public, or the common defense and security.

A-158

Sheet 15 of 22

~~1030 004~~

1029 001

d. Orders to Revoke a License

An order is ordinarily issued to revoke a license when:

1. The licensee's performance shows that he is not qualified to perform the activities covered by the license; or

2. Civil penalty proves to be ineffective as an enforcement action; or

3. The licensee refuses to correct items of noncompliance; or

4. A licensee does not respond to a "notice of violation"; or

5. A licensee's response to a "notice of violation" indicates inability or unwillingness to maintain compliance with regulatory requirements; or

6. Any material false statement is made in the application or in any statement of fact required under Section 182 of the Act.

e. Denial of Application for License Renewal

Denial of an application for a license renewal is ordinarily used in lieu of an order for revocation where license renewal is pending or the expiration of the license term is imminent.

f. Orders for Other Items of Noncompliance

Orders to cease and desist, or for suspension, modification or revocation of a license are ordinarily issued for other comparable types of violations, infractions or deficiencies when the Commission deems such sanctions to be appropriate and necessary.

In all cases where orders are issued to impose civil penalties, to require a licensee to "cease and desist," or to suspend, modify, or revoke a license, the person so ordered may demand a hearing under 10 CFR Part 2. The hearing will be granted prior to implementation of the order except in cases where the Commission finds that the violation is deliberate or the public health, safety, or interest requires that the proposed action be temporarily effective pending the outcome of the hearing and/or further order.

A-159

Sheet 16 of 23

1029 002

~~1030 005~~

REGULATORY OPERATIONS BULLETINS - CRITERIA

A Regulatory Operations Bulletin may be issued to a class of licensees requesting specific actions as a result of safety related equipment design inadequacies, defects, operating inadequacies, malfunctions, or failures of a generic nature that have occurred at a similar facility or operation. The Bulletin will specify that licensees inspect for and/or correct the inadequacies described in the Bulletin, notify Regulatory Operations of the corrective action taken or planned, and the date when action was or will be completed. An order may be issued if the response to a Bulletin is not prompt and effective.

IMMEDIATE ACTION LETTERS - CRITERIA

A Regulatory Operations Immediate Action Letter is ordinarily issued to solicit or confirm a licensee's commitment to certain actions for investigating, reporting, controlling, and correcting situations involving defects, deviations, failures, or administrative controls, at the licensee's facility. An order may be issued if the response to an Immediate Action Letter is not prompt and effective.

CATEGORIES OF ITEMS OF NONCOMPLIANCE

The Commission and representatives of the nuclear industry have recognized that the significance of items of noncompliance with AEC requirements varies in the potential for affecting the health and safety of the public, the common defense and security, and the environment. The Commission considers that it is desirable to include in Notices of Violation an indication of the significance of each item of noncompliance cited. As a means of categorizing the items of noncompliance into an order of importance which will express their relative significance, the Commission has established three categories of items of noncompliance as follows:

Violation

A violation is an item of noncompliance of the type listed below, or an item of noncompliance (1) which has caused, contributed to or aggravated an incident of the type listed below, or (2) which has a substantial potential for causing, contributing to or aggravating such an incident or occurrence; e.g., a situation where the preventive capability or controls were removed or otherwise not employed and created a substantial potential for an incident or occurrence with actual or potential consequences of the type listed below:

(a) Exposure of an individual in excess of the radiation dose specified in 10 CFR 20.403(b) or exposure of a group of individuals resulting in each individual receiving a radiation dose which exceeds the limits of 10 CFR 20.101 and a total dose for the group exceeding 25 man-rems.

(b) Radiation levels in unrestricted areas which exceed 50 times the regulatory limits.

(c) Release of radioactive materials in amounts which exceed specified limits, or concentrations of radioactive materials in effluents which exceed 50 times the regulatory limits.

(d) Fabrication, or construction, testing, or operation of a Seismic Category I system or structure in such a manner that the safety function or integrity is lost.

(e) Failure to function when required to perform the safety function or loss of integrity of a Seismic Category I system, or structure; or other component, system, or structure with a safety or consequences limiting function.

(f) Exceeding a safety limit as defined in technical specifications associated with facility licenses.

POOR ORIGINAL

A-161

Sheet 18 of 22

~~1030 007~~

1029 004

(g) Industrial sabotage of utilization or fuel facilities.

(h) Radiation or contamination levels in excess of limits on packages or loss of confinement of radioactive materials in packages offered for shipment on a common carrier.

(i) Diversion or theft of plutonium, uranium 233, or uranium enriched in the isotope U-235.

(j) A breakdown in management or procedural controls as evidenced by items of noncompliance in several areas of the QA criteria and license requirements.

(k) Other similar items of noncompliance having actual or potential consequences of the same magnitude.

Failure to report the above items as required constitutes a violation of the same importance level.

Infractions

An infraction is an item of noncompliance of the type listed below, or an item of noncompliance (1) which resulted in a reduction preventive capability below requirements but redundant contro precluded an item of noncompliance of the violation category. or (2) which caused, contributed to or aggravated an incident of the type listed below, or (3) which has a substantial potential for causing, contributing to or aggravating such an incident or occurrence; e.g., the preventive capability or controls were removed or otherwise not employed and there was substantial potential for an accident or occurrence with actual or potential consequences of the type listed below:

(a) Exposure of an individual or groups of individuals to radiation in excess of permissible limits but less than the values in 10 CFR 20.403.

(b) Release of radioactive materials in concentrations or rates which exceed permissible limits but in amounts less than permissible limits.

(c) Failure to function or loss of integrity of a Seismic Category I system or structure, or other component, system, or structure with safety or consequences limiting function during test; or failure to meet surveillance frequencies.

(d) Fabrication, or construction, testing, or operation of a Seismic Category I system or structure in such a manner that the safety function or integrity is impaired.

POOR ORIGINAL

A-162

Sheet 19 of 22

~~1030 009~~

1029 005

(e) Exceeding limiting conditions for operation (LCO).

(f) Inadequate management or procedural controls.

(g) Safety system settings less conservative than limiting safety system settings.

(h) A quantity of SNM unaccounted for which exceeds permissible limits.

(i) Exceeding limits or limiting conditions for operation in licenses, technical specifications, guides, codes, or standards which are imposed for the purpose of minimizing adverse environmental impact.

(j) Other similar items of noncompliance having actual or potential consequences of the same magnitude.

Failure to report the above items as required constitutes an item of noncompliance of the same category.

Deficiency

A deficiency is an item of noncompliance in which the threat to the health, safety, or interest of the public or the common defense and security is remote; and no undue expenditure of time or resources to implement corrective action is required; and deficiencies include such items as noncompliance with records, posting, or labeling requirements which are not serious enough to amount to infractions.

Failure to report deficiencies as required constitutes an item of noncompliance of the same category.

LEGISLATIVE HISTORY OF SECTION 234 OF THE ATOMIC ENERGY ACT OF 1954, AS AMENDED

Section 234 of the Atomic Energy Act of 1954, as amended, (Act), is the Commission authority for imposing civil penalties. Prior to the enactment in 1969 of Section 234 of the Act, the Commission's enforcement authority was limited to notices of violation and orders to cease and desist and to modify, suspend or revoke licenses.

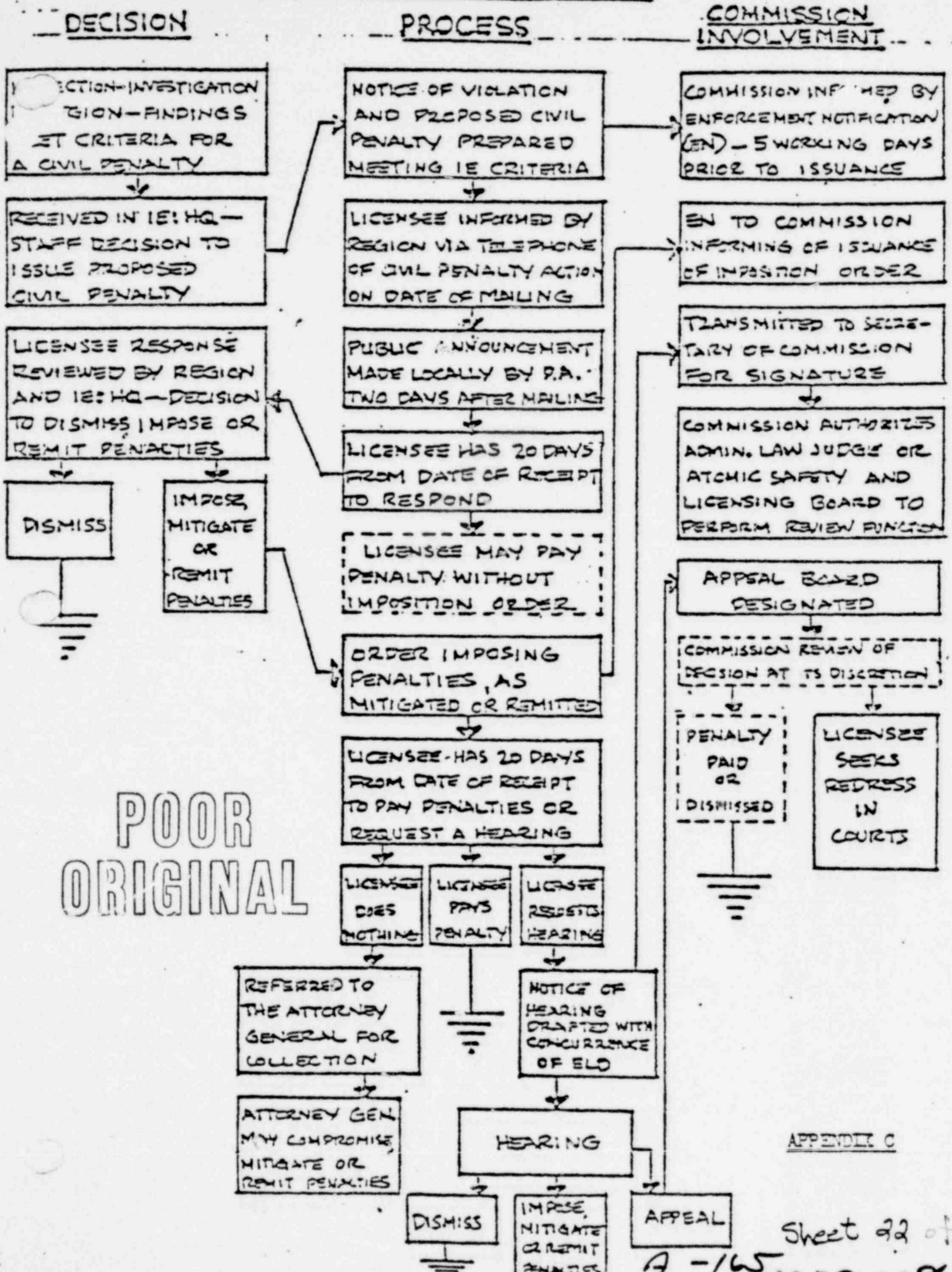
The legislative history of Section 234 of the Act indicates that the Joint Committee on Atomic Energy was concerned that revocation or suspension of a license in some instances "may be too harsh a penalty" and "may penalize the licensee's employees through loss of income without having any significant impact on the licensee itself." S. Report 91-553, H. Report 91-691, at 9, 10. Civil penalties could be imposed "without depriving a licensee of his means of livelihood or without requiring the cessation of an authorized activity which might be of material benefit to the public." id at 10.

The Joint Committee emphasized that civil penalties would not be appropriate for all violations. For example, "where the violation is one that seriously threatens the health or safety of an employee or a member of the public" a civil penalty should not be used. id at 10. However, penalties could be imposed in cases where license suspension or revocation is not in the public interest, but in which the importance of full adherence to regulatory requirements should be emphasized by more than a notice of violation or a cease and desist order. Hearings before JCAE, AEC Omnibus Legislation - 1969, 91st Congress, 1st session, 28 (September 12, 1969).

The purpose of the grant of authority to impose civil penalties is to provide the Commission with enforcement flexibility to deal with items of noncompliance of varying severity thereby "materially assist[ing] the Commission in carrying out its program to protect public health and safety and assure the common defense and security." S. Rept. 91-553, at 10. It should be noted that the Joint Committee stated that "the penalties authorized are civil only and are remedial in nature as opposed to punitive." id at 16. This statement is somewhat enigmatic since civil penalties inevitably have punitive aspects.

Appendix B

CIVIL PENALTY PROCESS



POOR ORIGINAL

APPENDIX C

Sheet 22 of ?

A-165

1029 008

W/ P E. Clin

POOR ORIGINAL

JUL
~~JUN~~ 19 1978

Wisconsin Public Service Corporation
ATTN: Mr. P. Ziemer
President
Post Office Box 1200
Green Bay, Wisconsin 54305

Docket No. 50-305

Gentlemen:

The findings of a recent inspection of the radiation protection program at the Kewaunee Nuclear Power Plant, particularly with regard to the personnel exposure which occurred in the reactor cavity on May 2, 1978, indicate a significant management weakness related to radiation protection. The inspection findings have been discussed with members of your staff by telephone on several occasions since the inspection. More importantly, the Director of our Region III Office met with you on May 1? 1978 to discuss the circumstances surrounding the May 2 exposure. At that meeting we also discussed the three apparent items of noncompliance found during the recent inspection. These noncompliances are set forth in the Notice of Violation attached as Appendix A to this letter.

In our view, the items of noncompliance in Appendix A demonstrate a lack of effective radiation exposure control. The potential for a significant personnel exposure in the reactor cavity was described in IE Circular No. 76-03, "Radiation Exposures in Reactor Cavities," dated September 10, 1976. In your November 12, 1976 response to this circular, you described the controls in effect at the Kewaunee facility to prevent such an exposure. The incident apparently resulted from a breakdown of these controls.

While the actual exposure of 2.9 rems did not exceed the regulatory limit, we consider the May 2 exposure to be very serious because of the potential for an extremely large radiation exposure. Our concern is even greater because our inspection showed that the decision to enter

Certified Mail
Return Receipt Requested

Attachment 4

OFFICE ▶				Sheet 1 of 7	
SURNAME ▶			A-166		

~~1030~~ 012

1029009

the reactor cavity was made by the senior member of management present on site with disregard for the survey required by the regulations, without the Radiation Work Permit required by your procedures, and without the radiation monitoring device required by your Technical Specifications. Consequently, we propose to impose civil penalties in the cumulative amount of Ten Thousand Dollars ($10,000) for these noncompliances. Appendix B of this letter is the Notice of Proposed Imposition of Civil Penalties. You are required to respond to this letter, and in preparing your response you should follow the instruction in Appendix A.

As noted previously, the employee who decided to enter the reactor cavity and who was subsequently exposed, was the senior Wisconsin Public Service Corporation employee on site at the time. Recognizing the natural tendency of other employees to refrain from stopping activities initiated by such an individual, the importance of supervisors' adherence to established requirements cannot be overstated. Inadequate communication between those involved also appears to have been a major contributor to the incident. In responding to the noncompliance items in Appendix A, you should specifically address your plans for strengthening these areas.

I would also like to address another concern. At about 8:30 a.m. on May 3, 1978, upon arrival at the Kewaunee Nuclear Power Station to inspect certain refueling outage activities, our inspectors were informed that a potential radiation overexposure had occurred at about 2:30 a.m. on May 2, 1978. Although aware soon after the incident that a substantial overexposure might have occurred, plant personnel had not informed our assigned project inspector who was present at the plant on May 2. While notification was not required since the exposure did not exceed regulatory limits, we are concerned that we were not promptly informed of this matter in view of our evident interest and the presence on site of our project inspector on the day of the occurrence. We hope that you will freely inform us of any potential problem where the NRC has a legitimate interest.

Your written reply to this letter and Notice of Violation and the findings of our continuing inspections of your activities will be considered in determining whether further enforcement action, such as additional civil penalties or orders to suspend, modify or revoke the license, may be required to assure future compliance.

Sheet 2 of 7

OFFICE ▶						
SURNAME ▶						
DATE ▶		A-147				

1029 010
~~1030 013~~

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and the enclosures will be placed in the NRC's Public Document Room.

Sincerely,

Ernst Volgenau
Director
Office of Inspection
and Enforcement

Enclosures:
1. Appendix A, Notice of Violation
2. Appendix B, Notice of Proposed Imposition of Civil Penalties

Sheet 3 of 7

A-168

1029 011
~~1030 011~~

POOR ORIGINAL

Appendix A

JUL 1 9 1978

NOTICE OF VIOLATION

This refers to the inspection conducted by representatives of the Region III (Chicago) Office at the Kewaunee Nuclear Power Plant, Kewaunee, Wisconsin, of activities authorized by NRC License No. DPR-43.

During this inspection conducted on May 3-5, 18 and June 5, 1978, the following apparent items of noncompliance were identified.

1. 10 CFR 20.201, "Surveys," requires in section (b) that each licensee make or cause to be made such surveys as may be necessary for him to comply with the regulations of 10 CFR 20. As defined in 10 CFR 20.201, section (a) "Survey" means an evaluation of the radiation hazards incident to the production, use, release, disposal, or presence of radioactive materials or other sources of radiation under a specific set of conditions.

   Contrary to the above, you failed to make such surveys as were necessary to assure compliance with 10 CFR 20.101, "Exposure to Individuals to Radiation in Restricted Areas." Specifically, you failed to make such a survey to assure that dose limits would not be exceeded on May 2, 1978 when an employee entered the reactor cavity and moved about in general radiation fields later measured to be as high as 2000 R/hr.

   This violation had the potential for causing a substantial radiation overexposure.

   (Civil Penalty - $4,000)

2. Technical Specification 6.11, "Radiation Protection Program" requires that procedures for personnel radiation protection shall be prepared consistent with the requirements of 10 CFR Part 20, and shall be approved, maintained and adhered to for all operations involving personnel radiation exposure.

   Procedure RC-HP-35, Revision 8, dated April 15, 1976, "Radiation Work Permit," states in section 1.1 that the purpose of a Radiation Work Permit (RWP) is to protect plant personnel by controlling access into areas such as high radiation areas, requires in section 2.1.1 that a RWP be issued for entry into any high radiation area, and specifies in section 6.0 the tasks which must be performed by various personnel prior to approval and issuance of the RWP.

Sheet 4 of 7

OFFICE ▶						
SURNAME ▶			A-169			

~~1030 015~~
1029 012

Contrary to the above, on May 2, 1978 a Radiation Work Permit was not approved and issued to control access into a high radiation area prior to an employee entering the reactor cavity where he moved about in general radiation fields later measured to be as high as 2000 R/hr.

This is an infraction. (Civil Penalty - $3,000)

3. Technical Specification 6.13.1 requires that any individual or group of individuals permitted to enter a high radiation area shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

Contrary to the above, on May 2, 1978, a radiation monitoring device which continuously indicates the radiation dose rate was not provided to an employee who entered the reactor cavity, a high radiation area containing general radiation fields later measured to be as high as 2000 R/hr.

This is an infraction. (Civil Penalty - $3,000)

This notice of violation is sent to you pursuant to the provisions of Section 2.201 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations. You are hereby required to submit to this office, within twenty (20) days of your receipt of this notice, a written statement or explanation in reply, including for each item of noncompliance: (1) admission or denial of the alleged items of noncompliance; (2) the reasons for the items of noncompliance, if admitted; (3) the corrective steps which have been taken by you and the results achieved; (4) corrective steps which will be taken to avoid further noncompliance; and (5) the date when full compliance will be achieved.

Sheet 5 of 7

OFFICE ➤						
SURNAME ➤			A-170			
DATE ➤						

~~1030 015~~

1029013

POOR ORIGINAL

JUN 1 9 1977

Appendix B

NOTICE OF PROPOSED IMPOSITION OF CIVIL PENALTIES

Wisconsin Public Service Corporation, Docket No. 50-305

This Office has considered the enforcement options available to the NRC, including administrative actions in the form of written notices of violation, civil monetary penalties, and orders pertaining to the modification, suspension, or revocation of a license. Based on these considerations we propose to impose civil penalties pursuant to Section 234 of the Atomic Energy Act of 1954, as amended (42 USC 2282), and to 10 CFR 2.205, in the cumulative amount of Ten Thousand Dollars ($10,000) for the specific items of noncompliance set forth in Appendix A to the cover letter. In proposing to impose civil penalties pursuant to this section of the Act and in fixing the proposed amount of the penalties, the factors identifed in the statements of consideration published in the Federal Register with the rule making action which adopted 10 CFR 2.205 (36 FR 16894) August 26, 1971 and the "Criteria for Determining Enforcement Action," which was sent to NRC licensees on December 31, 1974, have been taken into account.

Wisconsin Public Service Corporation may, within twenty (20) days of the date of receipt of this notice, pay the total civil penalties in the cumulative amount of Ten Thousand Dollars ($10,000) or may protest the imposition of the civil penalties in whole or in part by a written answer. Should Wisconsin Public Service Corporation fail to answer within the time specified, this office will issue an order imposing the civil penalties in the amount proposed above. Should Wisconsin Public Service Corporation elect to file an answer protesting the civil penalties, such answer may (a) deny the items of noncompliance listed in the Notice of Violation in whole or in part, (b) demonstrate extenuating circumstances, (c) show error in the Notice of Violation, or (d) show other reasons why the penalties should not be imposed. In addition to protesting the civil penalties in whole or in part, such answer may request remission or mitigation of the penalties. Any written answer in accordance with 10 CFR 2.205 should be set forth separately from your statement or explanation in reply pursuant to 10 CFR 2.201, but you may incorporate by specific reference (e.g., giving page and paragraph numbers) to avoid repetition.

Wisconsin Public Service Corporation's attention is directed to the other provisions of 10 CFR 2.205 regarding, in particular: failure to answer and ensuing orders; answer, consideration by this office, and orders; requests for hearings, hearings, and ensuing orders; compromise; and collection.

OFFICE ▶						
SURNAME ▶			A-171			
DATE ▶						

Upon failure to pay any civil penalty due which has been subsequently determined in accordance with the applicable provisions of 10 CFR 2.205, the matter may be referred to the Attorney General, and the penalty, unless compromised, remitted, or mitigated, may be collected by civil action pursuant to Section 234c of the Atomic Energy Act of 1954, as amended, (42 USC 2282).

Sheet 7 of 7

OFFICE ▶			A-172			
SURNAME ▶						
DATE ▶						

~~1030 010~~

1029 015

WISCONSIN PUBLIC SERVICE CORPORATION

P.O. Box 1200, Green Bay, Wisconsin 54305

August 10, 1978

Mr. Ernest Volgenau, Director
Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Sir:

Wisconsin Public Service Corporation
(Kewaunee Nuclear Plant) Docket No. 50-305
July 19, 1978 Notice of Violation

This written explanation is provided pursuant to the requirements of 10 CFR § 2.201 in response to your letter of July 19, 1978 (apparently erroneously dated June 19, 1978) which transmitted a Notice of Violation and Imposition of Civil Penalties related to an event at the Kewaunee Nuclear Power Plant on May 2, 1978.

As to Item 1, Wisconsin Public Service Corporation (hereinafter "WPSC") denies the allegation of the violation. As to Item 2, WPSC also denies the allegation of an infraction. As to Item 3, WPSC admits an infraction subject to the explanation set forth below (See also the attached Answer to Notice.).

The following is WPSC's description and evaluation of the May 2, 1978, event. On the morning of May 2, 1978, the filling operation of the refueling pool was interrupted with a water level of approximately 8" above the reactor vessel flange to perform an inspection. An operator was dispatched to inspect for leaks. That inspection indicated significant leakage about either the reactor vessel-refueling pool seal or the sand plug covers over the reactor vessel nozzles.

When this information was supplied to the Shift Supervisor, he decided to enter the containment area so as to be able to evaluate the nature and extent of the problem and to determine what corrective measures were indicated. The Shift Supervisor, in concurrence with the Night Refueling Coordinator, determined the most direct way to evaluate the leakage source and the extent of leakage, which appeared large, was to enter the reactor vessel cavity.

CERTIFIED MAIL
RETURN RECEIPT REQUESTED

A-173

Sheet 1 of 8
Attachment 5

1029 016
~~1030 019~~

In accordance with established and approved procedures, the senior Health Physics "H P") man on site was contacted to determine what measures were necessary for the proposed entry. A contracted HP technician was dispatched by the Health Physics Group to the area to perform a survey with a high range radiation monitor and a respirator to use during the entry. By dispatching an HP technician to the area with a respirator and a high range monitor, the senior HP man performed actions which indicated to the contract HP man working for him, to the Shift Supervisor and to the Night Refueling Coordinator that entry was appropriate provided the radiation levels determined in the survey by the HP technician were not beyond reasonable limits.

The HP technician performed a survey which indicated radiation levels in the 50-70 R/hr range. Those readings corresponded to the Health Physics Department posted radiation field strength for the area of 70 R/hr.

Subsequent evaluation disclosed that the results of the survey were inaccurate. Thus, the Shift Supervisor was given erroneous information upon which to base his entry decision. The survey inaccuracy apparently resulted from incomplete performance of the survey by the HP technician in light of the large radiation field variations. Although NRC has surmised that the survey may have been affected by intimidation of the technician by the Shift Supervisor, WPSC review of the incident indicates that the contracted HP technician did not know, until after the completion of the entry, that the person who proposed and made the entry was the Shift Supervisor.

Based upon the field strength disclosed by the survey, entry time limits were discussed. At that time a final decision to perform the entry was made. The survey information showing radiation levels insufficiently high to preclude entry was employed in that evaluation.

At that point it was the responsibility of the HP group to assure that a radiation monitoring device appropriate to the expected radiation field and level of exposure was provided to and worn by the person making the entry. As a result of oversight by all personnel involved, the only devices worn were the 0 to 200 mR range dosimeter (which was offscale following exit) and the TLD (which subsequent analysis found to indicate an exposure of 2.8 rem). Subsequent evaluation of the field strength and the circumstances of the entry provided the conclusion that the Shift Supervisor had a peak exposure to the head of 2.9 rem. See Report No. 50-305/78-07, pages 7-9.

It should be noted that under the procedures established by RC-HP-35 no Radiation Work Permit ("RWP") was required. The entry at issue involved an emergency situation and was of very short duration. In accordance with the alternative procedure available under RC-HP-35 an experienced HP person, kept in constant attendance, was substituted for the RWP requirement. This decision facilitated prompt and expeditious response to a potentially dangerous leak situation while providing the measure of safety mandated by radiation protection procedures.

A-174 Sheet 2 of 8

The precautions decided upon included the decision to make the entry very brief. This resulted in minimization of exposure risk and an actual exposure below regulatory limits.

Following the Shift Supervisor's exit from the cavity, the personal dosimeter offscale reading was identified, an investigation commenced, and NRC was subsequently notified of the event.

The following corrective steps have been and will be taken with regard to the above event:

> During the plant safety meeting held on June 21, 1978, the reactor vessel cavity entry incident was discussed with the members of the plant staff. Included in that review and discussion was the identification of the requirement to carry a properly ranged dosimeter into high radiation areas and other monitoring devices as appropriate. All personnel who are granted unescorted access to radiation areas receive an annual refresher course in health physics. During that refresher course, the responsibilities of each individual to be aware of proper dosimetry and monitoring will be reviewed. The review of the incident with the members of the plant staff which has been completed and the yearly refresher training will provide meaningful assurance that personnel have been adequately trained to avoid such mistakes in the future.
>
> Additionally, as a directive from Corporate Management, the Health Physics Group has been directed to split the day and night responsibility between the two most senior personnel available within that group. The Health Physics Department has also been ordered to review the entire plant for areas similar to the reactor cavity in terms of radiation hazards and assure that the posting of those areas clearly indicates the hazard potential of each area. The specific responsibilities of the Health Physics Group have been delineated such that there will be no misinterpretation of which organization provides assurance with the requirements of the Health Physics Program. Direction has been provided to assure that each proposed entry is fully evaluated such that there can be no misunderstanding as to the extent of the evaluation necessary by the various organizations. A formal inspection board has been established to assure that future investigations of significant incidents are carried out in an organized, complete and independent manner and communication with the NRC inspectors performing a parallel investigation is formally established.

In addition to the foregoing description and evaluation of the May 2, 1978, event and the corrective program undertaken, WPSC wishes to comment on certain assertions and implications evident in NRC reports and correspondence concerning this event. WPSC is particularly concerned with NRC identification of the problem as displaying management weakness. NRC has also indicated the belief that more controls were necessary.

A-175 Sheet 3 of 8

1029 018

~~1030 021~~

In view of the fact that our review and evaluation indicate that a personnel error by a contracted HP technician responsible for the incomplete survey was the cause of the event, we are at a loss to recognize how additional controls, which still depend upon avoidance of similar personnel errors as the only means to assure that reoccurrence will be avoided, provide any additional measure of safety. Associated with increased control is the danger of hampering emergency operations and creating unsafe conditions.

An isolated personnel failure to perform a task accurately, due at least in part to radiation field variation, cannot fairly be characterized as management weakness. Supervisory personnel must be entitled to rely on the validity of survey results reported to them. Evaluation of decisions must be made in light of the facts known to the decision maker at the time of the decision.

Finally, with regard to certain statements, in the letter accompanying the notices, it should be again noted that no overexposure occurred and no violations have been shown.

In conclusion, it is the position of WPSC as to Items 1 and 2 no violation or infraction has been shown. As to Item 3, significant corrective action has been undertaken and WPSC does not feel that any civil penalty is appropriate for Item 3 under applicable NRC guidelines.

Sincerely,

P. D. Ziemer

P. D. Ziemer
President

snf

Enc.

A-176

Sheet 4 of 8

~~1030 022~~
1029 019

UNITED STATES

NUCLEAR REGULATORY COMMISSION

Wisconsin Public Service Corporation (Kewaun : Nuclear Power Plant)	ANSWER TO NOTICE OF OF VIOLATION AND PROPOSED IMPOSITION OF CIVIL PENALTIES Docket No. 50-305

Pursuant to 10 C.F.R. § 2.205 and in answer to the Notice of Violation, Wisconsin Public Service Corporation (herewith "WPSC"), by its undersigned attorneys admits, denies and states as follows:

1. It is alleged that WPSC failed to make a survey required to assure compliance with 10 C.F.R. § 20.101, Section 20.101(b)(1) provides: "During any calendar quarter the dose to the whole body from radioactive material and other sources of radiation in the licensee's possession shall not exceed 3 rems. . ." At no time during the event in question was this limit exceeded. As acknowledged by WPSC and NRC exposure to the individual was about 2.90 rem. (See I E Inspection Report No. 50-305/78-07, page 9.)

The statement that there was a failure to survey is simply factually inaccurate. Prior to making his entry to the reactor vessel cavity, the shift supervisor requested from Health Physics personnel clarification of the safety requirements for such an entry. As a result of that request, a survey of the area (as required by the applicable regulations) was in fact performed. This survey failed accurately to disclose the actual radiation field present, apparently

A-177

Sheet 5 48

1029 020

~~1030~~

because of incomplete performance of the survey by the health physics technician. Nonetheless, in reaction to the survey, an evaluation of radiation exposure was made by the persons responsible prior to entry. As a result of this evaluation, a decision to make the entry very brief in order to minimize exposure was made. This decision allowed and resulted in full compliance with the regulations of Part 20

The inaccuracy of the survey resulted from an isolated failure by health physics personnel. All appropriate procedures were followed in requesting the survey and evaluating its results. No improper management decisions were involved. No violation of Part 20 regulations resulted and thus no civil penalty is warranted.

2. The second alleged item of non-compliance relates to a failure to secure a Radiation Work Permit ("RWP") as allegedly required by Procedure RC-HP-35 Revision B, dated April 15, 1976 in conformance with Technical Specification 6.11. It is agreed that no RWP was obtained prior to the event in question. However, complete examination of the radiation protection program and the established requirements of RC-HP-35 discloses that alternative applicable procedures are available and were followed. Thus, no infraction occurred.

Procedure RC-HP-35 includes the following provisions:

> "NOTE: During jobs of very short duration, emergencies, or where quick action is necessary, a continuous escort by experienced Health Physics personnel may be substituted for the RWP."

> "NOTE: During jobs of very short duration, emergencies or where quick action is necessary or at the discretion of

Sheet 6 of 8
1029 021
~~1030 024~~

Health Physics Supervisor or the
designated alternate a continuous escort
by experienced Health Physics personnel
may be substituted for the RWP."

The purpose of permitting alternative procedures under the circumstances noted is to allow expeditious handling of emergency situations or short term activities where the requirement of documented approvals would be counter productive. When senior members of plant staff determine that immediate action is necessary to assure plant safety, reduce total radiation exposure to plant personnel, or expedite repairs, the procedures thus permit quicker reaction while the presence of the Health Physics personnel provides the measure of safety ordinarily provided by the RWP.

The event in question undeniably involved an emergency situation and a job of very short duration. During the event a contract Health Physics technician was in attendance at the point of entry. That technician was in attendance during the whole period of entry and attempted to monitor the entry path during the event as allowed by the procedure. Therefore, the conditions of the alternative procedure were satisfied and no violation or disregard for procedures existed.

The infraction alleged thus did not occur and no civil penalty is warranted.

3. The third alleged item of non-compliance involves an employee who entered a high radiation area without wearing the prescribed radiation monitoring device. WPSC admits certain personnel failures in this regard. However, significant corrective steps have

A-179

Sheet 7 of 8
1029 022
~~1030 025~~

been taken which assure that further instances of non-compliance will not occur. The non-compliance was the result of oversight by all personnel involved. Steps have been taken to assure compliance with the appropriate procedures. In addition, no safety threat or actual damages was involved in the absence of a proper dosimeter. It should also be noted that the exposure would not have been mitigated by the presence of proper dosimetry.

Because of the isolated nature of this event, because no safety threat or actual danger was created by the event, and because corrective steps have already been taken with regard to the event, WPSC believes that, under NRC criteria for imposing civil penalties, no civil penalty should be imposed by reason of Item 3.

STEVEN E. KEANE
DAVID A. BAKER

By David A. Baker
Attorneys for Wisconsin Public Service Corporation

OF COUNSEL:

FOLEY & LARDNER
777 East Wisconsin Avenue
Milwaukee, Wisconsin 53202

(414) 271-2400

A-180

~~Sheet 8 of 8~~
1029 023
~~1030 025~~

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

ACRS(3)
H-1016

DEC 07 1978

Wisconsin Public Service Corporation
ATTN: Mr. P. Ziemer
President
Post Office Box 1200
Green Bay, Wisconsin 54305

Docket No. 50-305

Gentlemen:

This refers to your letter of August 10, 1978, which responded to the Notice of Violation and Notice of Proposed Imposition of Civil Penalties sent to you with our letter of July 19, 1978 (incorrectly dated June 19, 1978). Our July 19, 1978 letter identified apparent items of noncompliance found during our inspection conducted on May 3-5, 18, and June 5, 1978, at your Kewaunee Nuclear Power Plant.

After careful consideration of your August 10, 1978 letter to Dr. Volgenau and the letters of July 20, 1978 and August 15, 1978, from Mr. E. W. James to Mr. James G. Keppler, we are amending the Notice of Violation and Notice of Proposed Imposition of Civil Penalties sent to you on July 19, 1978, for the reasons given in Appendix C to this letter. The effect of this change is the reduction of the cumulative amount of civil penalties from Ten Thousand Dollars ($10,000) to Seven Thousand Dollars ($7,000).

We are concerned with positions taken in your responses to the May 2 incident. Specifically you: 1) minimize the significance of the incident, 2) appear to condone token efforts to follow procedures, and 3) fail to acknowledge management's responsibility for licensed activities at the Kewaunee plant.

You appear to minimize the significance of the May 2 incident in the final paragraph of your Answer to Notice, which states that "...no safety threat or actual danger was created by the event...". We would emphasize the fact that entry into radiation fields of 2000 R/hr allows an individual to receive a dose at the rate of over 0.5 rems per second. Less than six (6) seconds exposure at this rate would have resulted in a dose that exceeds the regulatory limit. We regard the lack of a significant overexposure in the May 2 incident to be simply fortuitous.

You appear to condone token efforts on the part of employees to follow procedures. In the third paragraph of your Answer to Notice you state, "...a survey of the area (as required by the applicable regulations) was in fact performed." However, you also admit that the survey was

CERTIFIED MAIL
RETURN RECEIPT REQUESTED

~~Sheet 1 of 11~~

A-181 Attachment 6

1029 024
~~1030 027~~

"inaccurate" and "incomplete." Further, you imply that a proper evaluation of the situation was performed prior to entry into the reactor cavity. It is evident that the evaluation performed on May 2 did not give adequate consideration to the radiological conditions which could exist in the reactor cavity when in-core instrument thimbles were in the withdrawn position. This was the case even though our investigation disclosed that involved plant employees were aware of exposure problems encountered during cavity entries at other facilities (IE Circular No. 76-03 dated September 10, 1976). The performance of radiological measurements and evaluation of radiological conditions provide the foundation for an effective radiological protection program. Consequently, measurements must be accurate and complete and evaluations must be thorough. In addition to the above, you also state that the attendance of an HP technician at the point of entry satisfied the requirement for continuous escort by HP personnel. It is apparent to us that neither the literal requirement nor the basic purpose for continuous escort is met by this action.

In your letter of August 10, 1978, you express particular concern with our characterization that this incident was indicative of a significant management weakness and you attempt to transfer the blame to a contract HP technician by describing the cause as an isolated personnel failure. It is our view that the incident resulted from a weakness in the radiation protection program, which we regard as a management responsibility. We are concerned that not one but several individuals, who represented several different plant groups, were involved in the failure to assure that procedures and requirements were being followed. This concern is amplified by the fact that one of the individuals was the senior member of management on the site at the time. We expect members of management, in particular, to stress the importance of and set the example for following procedures and requirements.

Specific comments regarding Mr. James' letters of July 20, 1978 and August 15, 1978, to Mr. James G. Keppler are addressed in Appendix D.

We propose to impose civil penalties in the cumulative amount of Seven Thousand Dollars ($7,000) for the items of noncompliance listed in Appendix A. Appendix B of this letter is the Amended Notice of Proposed Imposition of Civil Penalties. You are required to respond to this letter, and in preparing your response you should follow the instructions in Appendix A.

Your written reply to this letter and Notice of Violation and the findings of our continuing inspections of your activities will be considered in determining whether further enforcement action, such as additional civil penalties or orders to suspend, modify or revoke the license, may be required to assure future compliance.

A-182 Sheet 2 of 11

1029 025

~~1030 023~~

In accordance with Section 2.790 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations, a copy of this letter and the enclosures will be placed in the NRC's Public Document Room.

Sincerely,

John G. Davis

John G. Davis
Acting Director
Office of Inspection
and Enforcement

Enclosures:
1. Appendix A, Amended Notice of Violation
2. Appendix B, Amended Notice of Proposed Imposition of Civil Penalties
3. Appendix C, Comments re. Contested Items of Noncompliance
4. Appendix D, Comments re Ltrs dtd July 20, 1978 and August 15, 1978

cc w/enclosures:
David A. Baker
Foley & Lardner
777 East Wisconsin Ave.
Milwaukee, Wisconsin 53202

A-183

~~Sheet 3 of 11~~
1029 026
~~1030 029~~

Appendix A

Amended Notice of Violation

This refers to the inspection conducted by representatives of the Region III (Chicago) Office at the Kewaunee Nuclear Power Plant, Kewaunee, Wisconsin, of activities authorized by NRC License No. DPR-43.

During this inspection conducted on May 3-5, 18, and June 5, 1978, the following apparent items of noncompliance were identified.

1. Technical Specification 6.11, "Radiation Protection Program," requires that procedures for personnel radiation protection be prepared consistent with the requirements of 10 CFR Part 20, and be approved, maintained, and adhered to for all operations involving personnel radiation exposure.

   Procedure RC-HP-35, Revision B, dated April 15, 1976, "Radiation Work Permit," was issued in implementation of Technical Specification 6.11. A stated purpose of this procedure is to inform workers of the radiation conditions and the protective requirements necessary to safely perform their jobs. Specifically, this procedure requires a radiation work permit for entry into a high radiation area except for jobs of very short duration or emergencies where continuous escort by experienced health physics personnel may be used in lieu of a radiation work permit.

   Contrary to the above, on May 2, 1978, an employee entered the reactor cavity, a high radiation area, without complying with procedure RC-HP-35 in that neither a radiation work permit was issued nor was the employee continuously escorted by an experienced health physics person.

   This violation had the potential for causing a substantial radiation overexposure.

   (Civil Penalty - $4,000)

2. Technical Specification 6.13.1 requires that any individual or group of individuals permitted to enter a high radiation area shall be provided with a radiation monitoring device which continuously indicates the radiation dose rate in the area.

   Contrary to the above, on May 2, 1978, a radiation monitoring device which continuously indicates the radiation dose rate was not provided to an employee who entered the reactor cavity, a high radiation area containing general radiation fields later measured to be as high as 2000 R/hr.

   This is an infraction. (Civil Penalty - $3,000)

A-184 Sheet 4 of 11

1029 027

~~1030 050~~

This amended Notice of Violation is sent to you pursuant to the provisions of Section 2.201 of the NRC's "Rules of Practice," Part 2, Title 10, Code of Federal Regulations. You are hereby required to submit to this office, within twenty (20) days of your receipt of this notice, a written statement or explanation in reply, concerning amended item of noncompliance #1 and include: (1) admission or denial of the alleged item of noncompliance; (2) the reasons for the item of noncompliance, if admitted; (3) the corrective steps which have been taken by you and the results achieved; (4) corrective steps which will be taken to avoid further noncompliance; and (5) the date when full compliance will be achieved. In responding to the Amended Notice of Violation, the responses to the July 19, 1978 Notice of Violation may be incorporated by reference.

A-185

Sheet 5 of 11

1029 028

~~1030 031~~

Appendix B

AMENDED NOTICE OF PROPOSED IMPOSITION OF CIVIL PENALTIES

Wisconsin Public Service Corporation Docket No. 50-305

This Office has considered the enforcement options available to the NRC, including administrative actions in the form of written notices of violation, civil monetary penalties, and orders pertaining to the modification, suspension, or revocation of a license. Based on these considerations we propose to impose civil penalties pursuant to Section 234 of the Atomic Energy Act of 1954, as amended, (42 USC 2282), and to 10 CFR 2.205 in the cumulative amount of Seven Thousand Dollars ($7,000) for the specific items of noncompliance set forth in Appendix A to the cover letter. In proposing to impose civil penalties pursuant to this section of the Act and in fixing the proposed amount of the penalties, the factors identified in the statements of consideration published in the Federal Register with the rule making action which adopted 10 CFR 2.205 (36 FR 16894) August 26, 1971, and the "Criteria for Determining Enforcement Action," which was sent to NRC licensees on December 31, 1974, have been taken into account.

Wisconsin Public Service Corporation may, within twenty (20) days of the date of receipt of this notice, pay total civil penalties in the cumulative amount of Seven Thousand Dollars ($7,000) or may protest the imposition of civil penalties in whole or in part by a written answer. Should Wisconsin Public Service Corporation fail to answer within the time specified, this office will issue an order imposing the civil penalties in the amount proposed above. Should Wisconsin Public Service Corporation elect to file an answer protesting the civil penalties, such answer may (a) deny the items of noncompliance listed in the Notice of Violation in whole or in part, (b) demonstrate extenuating circumstances, (c) show error in the Notice of Violation, or (d) show other reasons why the penalties should not be imposed. In addition to protesting the civil penalties in whole or in part, such answer may request remission or mitigation of the penalties. Any written answer in accordance with 10 CFR 2.205 should be set forth separately from the statement or explanation in reply pursuant to 10 CFR 2.201, but may incorporate by specific reference (e.g., giving page and paragraph numbers) to avoid repetition.

Wisconsin Public Service Corporation's attention is directed to the other provisions of 10 CFR 2.205 regarding, in particular, failure to answer and ensuing orders; answer, consideration by this office, and ensuing orders; requests for hearings, hearings and ensuing orders; compromise; and collection.

A-186 Sheet 6 of 11

1029 029

~~1030~~

Upon failure to pay any civil penalty due which has been subsequently determined in accordance with the applicable provisions of 10 CFR 2.205, the matter may be referred to the Attorney General, and the penalty, unless compromised, remitted, or mitigated, may be collected by civil action pursuant to Section 234c of the Atomic Energy Act of 1954, as amended, (42 USC 2282).

A-187 Sheet 4 of 11

Appendix C

Comments Re Contested Items of Noncompliance

After careful consideration of the information provided in your response of August 10, 1978, to the Notice of Violation and Notice of Proposed Imposition of Civil Penalties dated July 19, 1978, and of the information provided in letters to Mr. James G. Keppler from Mr. E. W. James dated July 20, 1978 and August 15, 1978, we have the following comments:

1. We have deleted the first item of noncompliance, because of its similarity in intended purpose to the modified second item of noncompliance. This modification reflects your reliance on the alternative (continuous escort by experienced health physics personnel) permitted by procedure RC-HP-35 in lieu of the RWP as described in your letter of August 10, 1978.

2. Regarding the second item of noncompliance, you contend that the shift supervisor's entry involved a job of very short duration and emergency and that under such conditions your procedure RC-HP-35 provides for an alternate procedure, which you allege was followed. The alternate procedure allows substitution of a continuous escort by experienced health physics personnel for the completion of the RWP procedure. You contend that attendance of a contract health physics technician at the point of entry satisfied the requirement of the alternate procedure.

   According to your procedure, one of the main purposes of procedure RC-HP-35 is "...to protect plant personnel...by informing the worker of the radiation and contamination conditions..." It is apparent that when the alternate procedure is utilized, the continuous escort by health physics personnel is intended to assure that an adequate survey is performed and that the worker will be informed of the radiological conditions in the work area. In this case, continuous escort was not provided. By remaining at the point of entry, the health physics technician could not measure nor inform the worker of the radiation fields he was entering.

   The amended citation reflects the failure to provide a continuous escort by experienced health physics personnel.

3. Regarding the third item of noncompliance, you acknowledge that the noncompliance occurred. However, you state that, because of the isolated nature of the event, because no safety threat or actual danger was created by the event, and because corrective steps have already been taken with regard to the event, no civil penalty should be imposed.

A-188

Sheet 8 of 11
1029 031
~~1030 03~~

We have addressed your statement concerning the absence of a safety threat in our letter of transmittal. Based on the very real potential for a significant overexposure that was presented by this event and our concern for the demonstrated weakness in the radiation protection program which we also addressed in our letter of transmittal, we conclude that a civil penalty is appropriate and consistent with NRC criteria for imposing civil penalties.

A-189

Sheet 9 8/11

1029 032

~~1030 055~~

Appendix D

Comments Re Letters dated July 20, 1978 and August 15, 1978

These comments address your letters of July 20, 1978 and August 15, 1978 responding to our letters of June 30, 1978 and July 19, 1978, respectively.

In responding to our letter of June 30, 1978 and its enclosed report (50-305/78-09) of the management meeting held on May 18, 1978, you expressed concern that your positions were not completely reflected in the report. As stated in the report, the purpose of the meeting was to review the findings of our inspection following the reactor cavity exposure incident of May 2, 1978, and to discuss your corrective actions. The report did not present our findings, which were detailed in Inspection Report 50-305/78-07 sent to you on July 19, 1978. Nor was the report intended to present your positions, other than your initial corrective actions, which have been documented in your letter of July 20, 1978 and August 15, 1978, to the Regional Office and your letter of August 10, 1978, to Inspection and Enforcement headquarters. For these reasons we do not intend to change the report to reflect your positions. Of course your July 20, 1978 letter becomes a part of the public record on this matter.

Your July 20 and August 15 letters attempt to relieve the shift supervisor of any responsibility for his entry into an unknown, high radiation field. They imply that the shift supervisor should not be encumbered by radiation hazard evaluations during potential emergency situations. We consider this position contrary to prudent radiation protection practices. We believe that sound radiation protection requires proper performance by the Health Physics Staff and cognizance and cooperation by responsible Operations Staff whose actions can result in changing plant conditions which affect radiation levels. In this regard, we point out that according to the shift supervisor's statement to our inspectors, he had read IE Circular No. 76-03, which states, "With the thimbles or detectors withdrawn into the cavity, however, exposure rates of hundreds or possibly thousands of roentgens per hour can exist. Overexposures can occur in seconds." Furthermore, we believe that the shift supervisor and other senior employees should set a good example for the remainder of your staff by ensuring that their actions are consistent with established procedures.

Your August 15 letter suggests that our inspection (50-305/78-07) of May 3-5, 18, and June 5, 1978, failed to include interviews with the refueling coordinator and the auxiliary operator. Our first knowledge of the involvement of the refueling coordinator resulted from your August 10, 1978 letter. Although the refueling coordinator's involvement before the entry appears only to be peripherally related to the radiation protection aspects of the incident, his involvement should have been made known to our inspectors during the inspection.

A-190

Sheet 10 of 11

1029 033

~~1030 036~~

Regarding the auxiliary operator, our report (78-07) clearly shows (Paragraph 4.c) that the auxiliary operator was interviewed during the inspection.

Regarding the three "main points of fact" revealed by your investigation:

1. For whatever reason, the lead health physics technician appears not to have specified that the "necessary equipment" include high range dosimetry and a radiation monitoring device. In our view, such an omission is not consistent with proper control, procedures, and HP practices.

2. Our interview with the health physics technician indicated that he was aware that the person making the reactor cavity entry was a person of authority. However, we are not certain that he knew the person's name and title before the entry.

3. The first paragraph under 4.d of our report 78-07 states our understanding of the health physics supervisor's notifications. As stated earlier, our inspectors were not informed of the refueling coordinator's involvement.

Your August 15 letter also states that you find the conclusions presented in the Inspection Report and the subsequent proposed enforcement action to be in error. In our letter of transmittal and by the Amended Notice of Violation, we acknowledge a change in the circumstances of the event; i.e., alleged implementation of the alternate procedure which allows substitution of continuous escort by experienced health physics personnel for the RWP requirements. However, the basic conclusions of our report remain valid; i.e., there were failures on the part of personnel at the Kewaunee plant to follow procedures and the technical specifications. Our concerns regarding the actions of the shift supervisor have been addressed previously.

A-191

WISCONSIN PUBLIC SERVICE CORPORATION

P.O. Box 1200, Green Bay, Wisconsin 5430[illegible]

RECEIVED

1979 FEB 22 PM 1 [illegible]

U.S. NUCLEAR REG. COMM.
ADVISORY COMMITTEE ON
REACTOR SAFEGUARDS

POOR ORIGINAL

January 2, 1979

Mr. John C. Davis
Acting Director
Office of Inspection and Enforcement
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Sir:

This letter and the two attachments (Response to Amended Notice of Violation, Appendix A, and Answer to Amended Notice of Proposed Imposition of Civil Penalties, Appendix B) respond to your letter of December 7, 1978, and its attachments.

As you will note from the attachments, we continue to take exception to your position which, in error, continues to refer to the action of our Shift Supervisor as a non-compliance with our procedure RC-HP-35. As the attachments indicate, we cannot see that that incident was other than that the H-P technician did not survey the area completely and our Shift Supervisor assumed that when he requested the survey, he was receiving adequate information. Certainly when he was told he would be entering a 75 rem field he knew that he was entering a high radiation area and planned to meet that condition.

It bothers us greatly that this condition is then characterized by you as a significant management weakness. We feel we have acted very responsibly in putting together a strong management team at the Kewaunee Plant. We have supplied the Health Physics Department with the very best of instrumentation. Other nuclear plants have contacted us at the urging of your own inspectors to inquire about programs your inspectors told them were very good at Kewaunee.

Even before this incident occurred, we had originated the Design Change No. 746 project which will put radiation monitors in six areas with potential for significant radiation level changes. These meters will have remote read-out from outside the monitored areas. The Reactor Cavity Area is one of those areas for which instruments are on order.

In the Operating Budget for January 1 through December 31, 1978, we included money for additions to our Health Physics Group to allow H-P Group coverage on a shift basis around the clock. These additional people had been hired and trained in an extensive training program.

CERTIFIED MAIL
RETURN RECEIPT REQUESTED

Sheet 1 of 7
Attachment 7

A-192

1029 035
~~1030 [illegible]~~

Mr. John G. Davis
January 2, 1979
Page 2

We, therefore, do not agree that management lacks concern for the safety of employees.

As it is evident that your organization uses evaluation of events to comparatively rate plants, we do feel that these investigations should be as complete and accurate as possible. It is important that your investigation as well as our own be correctly documented and that fines are assessed only as appropriate.

Sincerely,

Paul D. Ziemer

Paul D. Ziemer
President

snf

POOR
ORIGINAL

A-193

1079 036

~~1030 039~~

Mr. John G. Davis
January 2, 1979
Page 2

We, therefore, do not agree that management lacks concern for the safety of employees.

As it is evident that your organization uses evaluation of events to comparatively rate plants, we do feel that these investigations should be as complete and accurate as possible. It is important that your investigation as well as our own be correctly documented and that fines are assessed only as appropriate.

Sincerely,

Paul D. Ziemer

Paul D. Ziemer
President

snf

POOR
ORIGINAL

A-194

Sheet 2 of 7

1029 037

~~1030 040~~

RESPONSE TO AMENDED NOTICE OF VIOLATION (APPENDIX A)

Pursuant to 10 CFR § 2.201 the following response to the Amended Notice of Violation dated December 7, 1978, is provided.

The response to the July 19, 1978, Notice of Violation submitted to the NRC on August 10, 1978, addresses the events of May 2, 1978, and is incorporated herein by reference.

The procedure RC-HP-35 Revision B dated April 15, 1976, entitled "Radiation Work Permit" identifies its intended function in the control of activities in the statement of Purpose of the procedure which states:

> "The purpose of a Radiation Work Permit (RWP) is to protect plant personnel by controlling access to areas such as high radiation areas, airborne activity areas, contamination areas, etc., by informing the worker of the radiation and contamination conditions and the protective clothing or other requirements necessary to safely perform his job."

The RWP form identifies individuals intending to perform an activity, work to be performed, the results of radiation surveys along with levels of airborne activity and radioactive contamination, protective equipment and other special instructions deemed necessary by Health Physics personnel. The RWP document thereby does provide a method to control individuals and work activities and provides a means to inform the worker of the specific radiological conditions associated with the intended activity and the protective equipment and/or other requirement necessary to provide for safety.

The procedure RC-HP-35 also includes an option to the RWP document such that in an emergency or activity of short duration, where rapid action is necessary or desired, that action would not be precluded due to a

A-195

Sheet 3 of 7

1029 ~~-~~ 038

1030 ~~041~~

requirement to have completed the administrative paperwork of the RWP. That option of the procedure requires a continuous escort by experienced Health Physics personnel. The issue in regard to this matter is what constitutes a continuous escort in the context of the procedure KC-HP-35. Since the purpose of the procedure is clearly to control access and inform the workers of the hazards and protective requirements, it is obvious that the escort is simply to accomplish the same objectives. The manner of accomplishing those objectives could be in essence a hand-in-hand accompaniment, a pre-monitoring of the area by the HP staff member to identify the associated hazards, or if the individual requiring entry is qualified in the use of monitoring equipment the individual could enter alone following a review of limits and precautions with a member of the HP staff at the point of entry. The manner by which the escort is accomplished is dependent upon the specific conditions associated with the desired activity to be performed in light of the requirement of 10 CFR 20.1 to maintain personnel exposure as low as reasonably achievable including consideration for dose to Health Physics personnel. In regard to the May 2, 1978, event a pre-monitoring of the area to be entered was performed by the HP technician acting as the escort. As stated in the August 10 letter, the HP technician who was serving as the escort performed a survey of the area although inaccurately. The provisions of the procedure were complied with. There is no doubt that an error was made in the performance of the associated survey in that the variation in radiation field strength was not detected. That error, however, is not due to a procedural inadequacy which can be asserted to be a management failure, but is a failure in the mechanics of performing the survey and evaluating the radiation hazards of the area of entry. While effort was made to also accompany and monitor entry through use of an extendible probe, the combined effort remained inadequate as the monitored area was exceeded.

Because the above demonstrates conformance to the requirements of our reference procedure in the allegation, the alleged violation is denied.

12/29/78

A-197

1029 040

1030 043

ANSWER TO AMENDED NOTICE OF PROPOSED IMPOSITION OF CIVIL PENALTIES

(APPENDIX B)

Pursuant to 10 CFR § 2.205 in answer to the Amended Notice of Proposed Imposition of Civil Penalties the following is provided in regard to each item:

Item 1. It is alleged that Wisconsin Public Service Corporation (herewith "WPSC") failed to comply with Technical Specification 6.11 in that the provisions of Procedure RC-HP-35 Revision B dated April 15, 1976, which allowed for continuous escort of short term jobs or emergenices were not complied with. The response pursuant to 10 CFR § 2.201, identified as Appendix A, here attached, identifies errors in and denies the allegation as stated in the Amended Notice of Violation. Thus, no civil penalty is warranted.

Item 2. The Answer to Notice of Violation and Proposed Imposition of Civil Penalties stated that in fact a non-compliance did occur, however, under NRC criteria for imposing civil penalties no civil penalty should be imposed for Item 2 of the Amended Notice of Violation.

On December 31, 1974, the U. S. Atomic Energy Commission, predecessor to the NRC issued to all licensees the "Criteria for Determining Enforcement Action." That criteria addressed civil monetary penalties and the specific criteria upon which such penalties could be imposed. The criteria upon which civil penalties may be imposed include:

A. Repeated non-compliance

B. Failure to implement corrective action previously committed to.

C. Deliberate failure to comply with regulations.

D. Chronic non-compliance.

E. Cases where an order was issued to assure health and safety of the public and personnel.

A-198

Sect 6 & 7

1029 041

~~1030~~

F. Actual seriousness of an event was contributed to by the non-compliance.

G. Violation category enforcement action events.

H. Case where the nature <u>and</u> number of events indicates a lack of management concern for safety.

I. Knowing unauthorized use of materials.

J. Failure to report significant matters to the Commission.

The above criteria clearly indicate that enforcement action in the form of civil penalty is intended to be imposed for deliberate or chronic failures of a licensee to comply with the requirements of the Regulations or exhibit adequate concern for safety. While it is recognized that civil penalties may be imposed for cases not specifically listed in the criteria, the non-compliance must be of a similar nature and comparable to the conditions of the criteria. The non-compliance associated with item 2 was not a deliberate or chronic failure, but as indicated in the August 10, 1978, response an isolated oversight by the personnel involved.

12/29/78

A-199

WASHINGTON, D. C. 20555

February 22, 1979

Wisconsin Public Service Corporation
ATTN: Mr. P. Ziemer
President
Post Office Box 1200
Green Bay, WI 54305

Docket No. 50-305

Gentlemen:

This is in response to your letter dated January 2, 1979, which was in response to the Amended Notice of Violation and Amended Notice of Proposed Imposition of Civil Penalties sent to you with our letter dated December 7, 1978.

The December 7, 1978 letter concerned two items of noncompliance found during a Nuclear Regulatory Commission inspection on May 3-5 and 18, and June 5, 1978, of the radiation protection program at your Kewaunee Nuclear Power Plant.

After careful consideration of your January 2, 1979 response, we conclude that the items of noncompliance did occur as described in the Amended Notice of Violation. With regard to item 1, we find no evidence that continuous escort by experienced health physic personnel was provided nor was the purpose of the procedure fulfilled. With regard to item 2, you have admitted noncompliance. Accordingly, we hereby serve the enclosed Order on Wisconsin Public Service Corporation, imposing Civil Penalties in the amount of Seven Thousand Dollars ($7,000).

The two items of noncompliance were related to an incident involving entry into the reactor cavity, an area with the potential for causing a substantial radiation overexposure. The potential was brought to your attention through IE Circular 76-03, "Radiation Exposures in Reactor Cavities," which was acknowledged by you on November 12, 1976. Both items of noncompliance contributed to the seriousness of the incident, which had the potential for causing a substantial radiation overexposure. The imposition of Civil Penalties in this case is consistent with enforcement policy and published criteria.

CERTIFIED MAIL
RETURN RECEIPT REQUESTED

POOR ORIGINAL

Sheet 1 of 6
Attachment 8

A-200

1029 043
~~1030 046~~

We will review your corrective actions regarding the items of noncompliance during future inspections.

Sincerely,

John G. Davis
John G. Davis
Acting Director
Office of Inspection
and Enforcement

Enclosures:
Order Imposing Civil
Penalties

Sect 2 & 6

A-201

1029 044

~~1030~~

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter of Wisconsin Public Service Corporation Post Office Box 1200 Green Bay, WI 54305	))))))	Docket No. 50-305 License No. DPR-43

ORDER IMPOSING CIVIL PENALTIES

I

Wisconsin Public Service Corporation, Green Bay, Wisconsin, (the "licensee"), is the holder of License No. DPR-43 (the "license"), issued by the Nuclear Regulatory Commission (the "Commission"), which authorizes the licensee to operate the Kewaunee Nuclear Power Plant located in Kewaunee County, Wisconsin, in accordance with the conditions specified therein. The license was issued on December 21, 1973, and has a termination date of August 6, 2008.

II

A special inspection of certain of the licensee's activities under the license was conducted on May 3-5, 18 and June 5, 1978. As a result of this inspection, it appears that the licensee has not conducted its activities in full compliance with the requirements of the Technical Specifications. A written Notice of Violation and Notice of Proposed Imposition of Civil Penalties in the amount of $10,000 was served upon the licensee by letter dated July 19, 1978 (incorrectly dated June 19, 1978). The licensee responded August 10, 1978. In view of the licensee's

POOR ORIGINAL A-202 Sheet 3 of 6

1029 045

~~1030 048~~

response, an Amended Notice of Violation was served upon the licensee by letter dated December 7, 1978, appended hereto as Appendix I, specifying the items of noncompliance, in accordance with 10 CFR 2.201. An Amended Notice of Proposed Imposition of Civil Penalties dated December 7, 1978, was served concurrently upon the licensee in accordance with Section 234 of the Atomic Energy Act of 1954, as amended, (42 USC 2282), and 10 CFR 2.205, incorporating by reference the Amended Notice of Violation, which stated the nature of the items of noncompliance and the provision of the NRC regulations with which the licensee was in noncompliance.

An answer from the licensee to the Amended Notice of Violation and to the Amended Notice of Proposed Imposition of Civil Penalties dated January 2, 1979, is appended hereto as Appendix II.

III

Upon consideration of the answer received and the statements of fact, explanation, and argument of mitigation contained therein, the Acting Director of the Office of Inspection and Enforcement has determined that the penalties proposed for the items of noncompliance designated in the Amended Notice of Violation should be imposed.

IV

In view of the foregoing and pursuant to Section 234 of the Atomic Energy Act of 1954, as amended, (42 USC 2282), and 10 CFR 2.205, IT IS HEREBY ORDERED THAT:

POOR ORIGINAL

A-203 Sheet 486

1029 046

~~1030~~

The licensee pay civil penalties in the total amount of Seven Thousand Dollars ($7,000), within twenty (20) days of the date of receipt of this Order, by check, draft, or money order payable to the Treasurer of the United States and mailed to the Acting Director of the Office of Inspection and Enforcement.

V

The licensee may, within twenty (20) days of the receipt of this Order, request a hearing. If a hearing is requested, the Commission will issue an Order designating the time and place of hearing. Upon failure of the licensee to request a hearing within twenty (20) days of the date of receipt of this Order, the provisions of this Order shall be effective without further proceedings and, if payment has not been made by that time, the matter may be referred to the Attorney General for collection.

VI

In the event the licensee requests a hearing as provided above, the issues to be considered at such a hearing shall be:

(a) whether the licensee was in noncompliance with the Commission's requirements as set forth in the Amended Notice of Violation attached hereto as Appendix I; and

A-204 Sheet 5 of 6

~~1030~~ 1029 047

PDR 71-0338

"Nothing To Sell But Integrity"

METILS INC

HOME OFFICE
7645 GULF FREEWAY
P. O. BOX 12585
HOUSTON, TEXAS 77017
(713) 944-8761
TELEX: 77-4226

DOCKETED
USNRC
JUL 1 8 1979
NMSS
MAIL SECTION
DOCKET CLERK

RECEIVED
JUL 17 1979
U.S. Nuclear Regulatory Commission
NMSS
Mail Section

July 16, 1979

Mr. Charles E. MacDonald, Chief
Transportation Branch
Division of Fuel Cycle and Material Safety
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. MacDonald,

Attached is a copy of the QUALITY ASSURANCE PROGRAM of METILS, Inc. implemented for License Number 42-16534-01 in accordance with requirements of 10 CFR Part 71, Appendix E.
Please let me know if other data needs to be submitted for compliance with regulations for shipping radioisotopes designed within the license number designated above.

Respectfully submitted,

METILS, Inc.

Lee Wall

Lee Wall
Radiation Safety Officer and
Operations Manager

LW/ae
encl

778 107

dupe

7908240051

1029 048

~~1030 051~~

13532

QUALITY ASSURANCE PROGRAM

OF

METILS, Inc.

License Number 42-16534-01

FOR COMPLIANCE WITH 10 CFR PART 71, APPENDIX E

1. ORGANIZATION

The final responsibility for the Quality Assurance (QA) Program for Part 71 Requirements rests with METILS, Inc.

Design and Fabrication shall not be conducted under this QA Program. The QA Program is implemented as shown on the attached organization chart.

The Radiation Safety Officer is responsible for overall administration of the program, training and certification, document control and auditing.

The Radiographers are responsible for handling, storing, shipping, inspection, test and operating status and record keeping.

2. QUALITY ASSURANCE PROGRAM

The management of METILS, Inc. establishes and implements this QA Program. Training, prior to engagement, for all QA functions is required according to written procedures. QA Program revisions will be made according to written procedures with management approval. The QA Program will ensure that all defined QA procedures, engineering procedures, and specific provisions of the package design approval are satisfied. The QA Program will emphasize control of the characteristics of the package which are critical to safety.

The Radiation Safety Officer shall assure that all radioactive material shipping packages are designed and manufactured under QA Program approval by Nuclear Regulatory Commission for all packages designed or fabricated after July 1, 1978. This requirement can be satisfied by receiving a certification to this effect from the manufacturer.

3. DOCUMENT CONTROL

All documents related to a specific shipping package will be controlled through the use of written procedures. All document changes will be performed according to written procedures approved by management.

The Radiation Safety Officer shall insure that all QA functions are conducted in accordance with the latest applicable changes to the documents.

dupe

7908240055

778 108

1029 049

~~1030~~

4. HANDLING STORAGE AND SHIPPING

Written safety procedures concerning the handling, storage and shipping of packages for certain special form radioactive material will be followed. Shipments will not be made unless all tests, certifications, acceptances, and final inspections have been completed. Work instructions will be provided for handling, storage, and shipping operations.

Radiography personnel shall perform the critical handling, storage and shipping operations.

5. INSPECTION, TEST AND OPERATING STATUS

Inspection, test and operating status of packages for certain special form radioactive material will be indicated and controlled by written procedures. Status will be indicated by tag, label, marking or log entry. Status of nonconforming parts or packages will be positively maintained by written procedures.

Radiography personnel shall perform the regulatory required inspections and tests in accordance with written procedures. The Radiation Safety Officer shall ensure that these functions are performed.

6. QUALITY ASSURANCE RECORDS

Records of package approvals (including references and drawings), procurement, inspections, tests, operating logs, audit results, personnel training and qualifications and records of shipments will be maintained. Descriptions of equipment and written procedures will also be maintained.

These records will be maintained in accordance with written procedures. The records will be identifiable and retrievable. A list of these records with their storage locations, will be maintained by the Radiation Safety Officer.

7. AUDITS

Established schedules of audits of the QA Program will be performed using written check lists. Results of audits will be maintained and reported to management. Audit reports will be evaluated and deficient areas corrected. The audits will be dependent on the safety significance of the activity being audited, but each activity will be audited at least once per year. Audit reports will be maintained as part of the quality assurance records.

7/16/79
DATE

Lee Wall
Lee Wall
Radiation Safety Officer and
Operations Manager

778 109

1029 050
~~1030 055~~

QUALITY ASSURANCE PROGRAM ORGANIZATION CHART FOR

METILS, Inc.

LICENSE NUMBER 42-16534-01

FOR

COMPLIANCE WITH 10 CFR PART 71, APPENDIX E

PRESIDENT, TOM C. REED

RADIATION SAFETY OFFICER AND OPERATIONS MANAGER
LEE WALL

RADIOGRAPHER'S

DESCRIPTION OF RESPONSIBILITIES

1. PRESIDENT - General management of the company.

2. OPERATIONS MANAGER - Responsible for operations control of the company supervision of employees, cost control, job assignments, personnel relations, training.

3. RADIATION SAFETY OFFICER - Responsible for overall administration of the radiation safety program (including Quality Assurance), personnel radiation safety training and certification, document control, and auditing of the radiation safety program.

4. RADIOGRAPHERS - Responsible for using, storing, shipping, inspection, testing operating status, and record keeping of radioisotope sources and devices in accordance with written procedures of the company as approved by the Operations Manager and Radiation Safety Officer.

778 110

1029 051
~~1030~~