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MINUTES OF THE
216TH ACRS MEETING
APRIL 6-7, 1978
WASHINGTON, DC

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Mr. Glenn O. Bright, Atomic Safety and Licensing Board Panel, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.
Dr. James C. Lamb III, 313 Woodhaven Road, Chapel Hill, N.C. 27514.

Dated at Bethesda, Md., this 16th day of March 1978.

For the Atomic Safety and Licensing Board Panel.

JAMES R. YORE,
Chairman.

[FR Doc. 78-7520 Filed 3-21-78; 8:45 am]

[7590-01]

[Docket No. 50-244]

ROCHESTER GAS & ELECTRIC CORP.

Issuance of Amendment to Provisional Operating License

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 15 to Provisional Operating License No. DPR-18 issued to Rochester Gas & Electric Corp. which revised Technical Specifications for operation of the R. E. Ginna Plant located in Wayne County, N.Y. The amendment is effective as of the date of issuance.

The amendment incorporates fire protection Technical Specifications on the existing fire protection equipment and adds administrative controls related to fire protection at the facility. This action is being taken pending completion of the Commission's overall fire protection review of the facility.

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 issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated July 19, 1977, as supplemented December 13, 1977, (2) Amendment No. 15 to License No. DPR-61, and (3) the Commission's related Safety Evaluation dated November 25, 1977. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street NW., Washington, D.C., and at the Rochester Public Library, 115 South Avenue, Rochester, N.Y. 14627. A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Md., this 1st day or March 1978.

For the Nuclear Regulatory Commission.

DENNIS L. ZIEMANN,
Chief, Operating Reactors Branch No. 2, Division of Operating Reactors.

[FR Doc. 78-7525 Filed 3-21-78; 8:45 am]

[7590-01]

[Docket No. 50-244]

ROCHESTER GAS & ELECTRIC CORP.

Issuance of Amendment to Provisional Operating License

The U.S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 16 to Provisional Operating License No. DPR-18, issued to Rochester Gas & Electric Corp. (the licensee), which revised Technical Specifications for operation of the R. E. Ginna Plant (facility) located in Wayne County, N.Y. The amendment is affective as of its date of issuance.

The amendment changed the Technical Specifications to:

1. Delete the requirement for an Annual Operating Report, while retaining the specific requirement for an Annual Report of Occupational Exposure,
2. Modify the submittal date for the Monthly Operating Report to the 15th instead of the 10th of the month following the calendar month covered by the report,
3. Delete the Respiratory Protection Program based on your compliance with 10 CFR 20.103 since this item is now included in 10 CFR Part 20 of the Commission's regulations, and
4. Add a shock suppressor (snubber) to the safety-related listing of suppressors in Specification 3.13-1.

The applcation 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 issuance of this amendment.

For further details with respect to this action, see (1) the application for amendment dated November 28, 1977, (2) Amendment No. 16 to License No. DPR-18, 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, NW., Washington, D.C., and at the Rochester Public Library, 115 South Avenue, Rochester, N.Y. 14627.

A copy of items (2) and (3) may be obtained upon request addressed to the U.S. Nuclear Regulatory Commission, Washington, D.C. 20555. Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Md., this 8th day of March 1978.

For the Nuclear Regulatory Commission.

DENNIS L. ZIEMANN,
Chief, Operating Reactors Branch No. 2, Division of Operating Reactors.

[FR Doc. 78-7527 Filed 3-21-78; 8:45 am]

[7590-01]

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Meeting

In accordance with the purposes of sections 29 and 182b of the Atomic Energy Act (42 U.S.C. 2039, 2232b, the Advisory Committee on Reactor Safeguards will hold a meeting on April 6-7, 1978, in Room 1046, 1717 H Street NW., Washington, D.C.

The agenda for the subject meeting will be as follows:

THURSDAY, APRIL 6, 1978

8:30 A.M.-9:15 A.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 hear and discuss the report of the ACRS Subcommittee and consultants who may be present regarding the request for an operating license for the Arkansas Nuclear One, unit 2 powerplant. Portions of this session will be closed if necessary to discuss proprietary information applicable to this project.

9:15 A.M.-12:15 A.M. AND 1:15 P.M.-3:15 P.M. ARKANSAS NUCLEAR ONE, UNIT 2 (OPEN)

The Committee will hear and discuss presentations by representatives of the NRC staff and the applicant related to the request for operation of Arkansas Nuclear

One, unit 2. Portions of this session will be closed if necessary to discuss proprietary information applicable to this matter.

3:15 P.M.–3:45 P.M. EXECUTIVE SESSION (OPEN)

The Committee will discuss the report of its Subcommittee and consultants who may be present regarding the liquid pathway generic study for floating and land-based nuclear powerplants (NUREG-0440). Portions of this session will be closed if required to protect proprietary information regarding this matter.

3:45 P.M.–5 P.M. LIQUID PATHWAY GENERIC STUDY (OPEN)

The Committee will hear and discuss reports of representatives of the NRC staff and offshore power systems regarding the liquid pathway generic study (NUREG-0440) concerning impacts of accidental radioactive releases to the hydrosphere from floating and land-based nuclear powerplants. Portions of this session will be closed if required to discuss proprietary information regarding this matter.

5 P.M.–6:30 P.M. EXECUTIVE SESSION (OPEN)

The Committee will hear and discuss reports of Subcommittees, working groups, and members on a number of generic matters related to reactor safety including review of NRC siting policies and practices, proposed changes in NRC regulatory guides, and NRC procedures for review of proposed operations at increased power levels. This portion of the meeting will be open to the public. The Committee will also discuss its proposed report to the NRC regarding operation of Arkansas Nuclear One, unit 2. This session will be closed to discuss matters involved in an adjudicatory proceeding.

FRIDAY, APRIL 7, 1978

8:30 A.M.–11 A.M. MEETING WITH NRC STAFF (OPEN)

The Committee will hear presentations from and hold discussions with members of the Nuclear Regulatory Commission staff regarding recent licensing actions and operating experience including reports of increased microseismic activity in the vicinity of the Oconee nuclear plant, failure of a burnable poison-rod assembly in the Crystal River nuclear plant and the implementation of ACRS recommendations regarding the Davis-Besse nuclear power station, unit 1. The NRC staff will also report to the ACRS on generic matters related to nuclear powerplant safety including development of criteria for instrumentation to follow the course of a serious accident, and criteria for combination of dynamic loads in the design of nuclear powerplants. The future schedule for ACRS activities and topics proposed for consideration by the Committee will also be discussed.

11 A.M.–11:30 A.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 operation of the McGuire nuclear station, units 1 and 2. Portions of this session will be closed if necessary to discuss proprietary information applicable to this project.

11:30 A.M.–12:30 P.M. AND 1:30 P.M.–4 P.M.: MC GUIRE STATION, UNITS 1 AND 2 (OPEN)

The Committee will hear and discuss reports of representatives of the NRC staff and the applicant regarding the request for an operating license for the McGuire nuclear station, units 1 and 2. Portions of this session will be closed if necessary to discuss proprietary information applicable to this matter.

4 P.M.–6:30 P.M.: EXECUTIVE SESSION (OPEN/CLOSED)

The Committee will discuss its proposed reports to NRC regarding Arkansas Nuclear One, Unit 2 and the McGuire nuclear station, units 1 and 2. This session will be closed to discuss matters involved in adjudicatory proceedings.

The Committee will discuss reports of its members regarding miscellaneous ACRS activities such as proposed reorganization of ACRS Subcommittees and working groups and the qualifications of candidates proposed for appointment to the Committee. Portions of this session will be closed to discuss material which if released would represent an unwarranted invasion of personal privacy.

Procedures for the conduct of and participation in ACRS meetings were outlined in the FEDERAL REGISTER on October 31, 1977, page 56972. 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 section 10(d) of Pub. L. 92-463 that it is necessary to close portions of the meeting as noted above to protect proprietary information (5 U.S.C. 552b(c)(4)), to permit discussion of matters involved in adjudicatory proceedings (5 U.S.C. 552b(c)(10)), and to protect information the release of which would represent an unwarranted invasion of personal privacy (5 U.S.C. 552b(c)(6)). Separation of factual information from information considered exempt from disclosure during closed portions of the meeting is not considered practical.

Background information concerning items to be considered during this meeting can be found in documents on file and available for public inspection in the Nuclear Regulatory Commission's Public Document Room, 1717 H Street NW., Washington, D.C. 20555, and in the following public document rooms:

MCGUIRE NUCLEAR STATION UNITS 1 AND 2

Public Library of Charlotte and Mecklenburg County, 310 North Tryon Street, Charlotte, N.C. 28202.

ARKANSAS NUCLEAR ONE, UNIT 2

Arkansas polytechnic college, russellville, Ark. 72801.

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 202-634-1371, between 8:15 a.m. and 5 p.m. e.s.t.

Dated: March 20, 1978.

JOHN C. HOYLE,
Advisory Committee Management Officer.

[FR Doc. 78-7712 Filed 3-21-78; 8:45 am]

[7715-01]

POSTAL RATE COMMISSION

[Docket No. R77-1]

POSTAL RATE AND FEE INCREASES, 1977

Oral Argument

MARCH 15, 1978.

Oral argument in this case previously scheduled for March 24, 1978, has been re-scheduled for Tuesday, March 28, 1978, at 9:30 a.m.

A further notice will be issued advising counsel of the time allotted to each for oral argument.

DAVID F. HARRIS,
Secretary.

[FR Doc. 78-7510 Filed 3-21-78; 8:45 am]

[8025-01]

SMALL BUSINESS ADMINISTRATION

[License No. 06/06-0175]

SMALL BUSINESS INVESTMENT CAPITAL, INC.

Filing of Application for Approval of a Conflict of Interest Transaction Between Associates

Notice is hearby given, pursuant to §107.1004 of the regulations governing small business investment companies (13 CFR 107.1004 (1977)), by the Small Business Administration (SBA) of a conflict of interest transaction between the Small Business Investment Capital, Inc. (Licensee), 10003 New Benton Highway, Little Rock, Ark. 72203, a Federal Licensee under the Small Business Investment Act of 1958, as amended (the Act) (15 U.S.C. 661 et seq.), and an Associate.

The Licensee was licensed by SBA on March 6, 1975. It is wholly owned by Snur-Valu Stamps, Inc., 10003 New Benton Highway, Little Rock, Ark. 72203, which in turn is owned approximately 45 percent by Affiliated Food Stores, Inc., a cooperative of retail grocers, and 55 percent by present and former members of the cooperative.

It is proposed that the Licensee loan $150,000 to Mr. Jerry Kelly to build a

Issue Date:

SEP 14 1978

CERTIFIED

MINUTES OF THE 216TH ACRS MEETING APRIL 6-7, 1978 WASHINGTON, DC

The 216th 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, April 6, 1978.

The Chairman noted the existence of the published agenda for this meeting, and listed 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: M . Isbin was not present Thursday.]

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. Siess and Shewmon as reviewers for the 216th ACRS meeting.

B. New Nuclear Licensing Bills

The Chairman noted that several proposals are before the Congress for changes to the nuclear licensing procedures. He noted two specific items included in the Administration's version of the proposed bill that are of direct interest to the Committee:

- The proposal for non-mandatory review of all license applications is more limiting than that proposed by the Committee. In the current form, the only license applications exempt from mandatory review are those for follow-on standard plants.

- The name of ACRS is proposed to be changed to the Advisory Committee on Reactor Safety.

C. ACRS Fellows Program

The Chairman noted that appropriations for the ACRS Fellows Program are not included in the current appropriations bill for fiscal year 1978. Members suggested that a letter should be prepared outlining the Committee's need for such a program and sent to the appropriate Senate and House Oversight Committees, accompanied by a request that these letters be forwarded to the pertinent committees or subcommittees within each House (see Appendix III).

D. Testimony of Committee Before Senate Subcommittee on Nuclear Regulation

The Chairman noted that the Committee has been invited to testify before the Senate Subcommittee on Nuclear Regulation, the Committee on Environment and Public Works, Senator Hart, Chairman. The Chairman noted that a draft of proposed testimony has been distributed to Members, and requested that they provide comments and suggestions regarding this testimony as soon as possible. He also noted that he would be accompanied to the hearing by the Executive Secretary, the Vice Chairman, and Messrs. Bender and Siess. He welcomed the presence of any other Members who could participate.

E. Vermont Yankee Vs. National Resources Defense Council

The Chairman noted that the U.S. Supreme Court has handed down a decision on the litigation between the Vermont Yankee Nuclear Plant and the National Resources Defense Council. Included in this decision also was the litigation between the NRC regarding the Midland Plant and Aschliman. The decision, copies of the summary of which have been provided, was generally favorable to the position taken by the NRC.

II. Meeting on Arkansas Nuclear One, Unit 2, Nuclear Power Plant (OL) (Open to Public)

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

A. Subcommittee Report

Mr. Carbon, Subcommittee Chairman, discussed the review of the Arkansas Nuclear One, Unit 2 design. He noted that this plant is a CESSAR-80 plant, but is using the newly designed core protection calculator system (CPCS). This system has been designed since the construction permit review of this plant, and therefore was not discussed at that time. He discussed the general design of the plant; the fuel rod configuration, noting that this plant will be the first to use the Combustion Engineering 16x16 fuel rod assembly; the functions of the CPCS, the core operating limits supervisory system (COLSS), which will be used for the first time in this plant; and the outstanding issues. He said that the subcommittee believes that the two major areas of interest are the CPCS, and the large number of outstanding items remaining to be resolved at this stage of the review. (For details, see Appendix IV; for a description of the core protection calculator system, see Appendix V; for consultants' reports, see Appendix VI.)

Mr. Carbon noted that an NRC Staff reviewer, J. Calvo, identified 39 technical issues which needed to be resolved as of last Spring, but the NRC Staff claims that all of these items have since been resolved.

Mr. Kerr offered his opinion that he believes that the CPCS has been very thoroughly reviewed by the NRC Staff, its consultants, and the ACRS' consultants. He noted that the Applicant desires to keep the CPCS connected to the plant computer, which is not part of the plant protection system, and therefore is not safety grade. In this way the Applicant hopes both to monitor the operation of the CPCS, and to use the information gained for other calculations, making more efficient use of information generated by the CPCS. The NRC Staff has concluded that such a connection can be allowed during the startup phase of the reactor, but should not be permitted during normal long-term operation. Mr. Kerr suggested that the Committee may wish to explore this matter.

W. Lipinski, ACRS consultant, suggested that the connection of the CPCS to the plant computer may improve the reliability of the CPCS, and also provide a means for recording the data generated by the CPCS. If the connection is not permitted, there will be no recording of this data.

E. P. Epler, ACRS consultant suggested that the Committee should proceed with caution in permitting a non-safety item to be tied to a safety system. He noted however, that there are advantages to this connection as well as possible pitfalls.

[Note: D. Rueter, Arkansas Power and Light Company (APLC), coordinated presentations for the Applicant; D. Martin, for the NRC Staff.]

B. Applicant's Presentations

1. Introduction

N. Moore, APLC, discussed the status of construction and testing of Arkansas Nuclear One, Unit 2 (ANO-2). He noted that the plant is 93.5% complete, based upon labor requirements, and 99.25% complete, based upon equipment installation. 261 of 287 systems have been accepted from construction for pre-operational testing and startup activity. He noted that pre-core hot functional testing was completed on February 18, 1978. The status of the pre-operational testing program is as follows:

- 135 of 185 tests are complete,
- with regard to tests required for fuel loading, 102 of 136 tests are complete.

The present schedule for fuel loading is May 15, 1978. It is anticipated that criticality will be achieved during the week of July 3, 1978, and it is hoped that commercial operation will begin in late October 1978.

2. Core Protection and Calculator System

A. Spinell, Combustion Engineering (CE), discussed the design and review of the CPCS and discussed the COLSS and how this system aids the plant operator in maintaining some of the limiting conditions for operation (see Appendix VII).

3. Functional Design of the CPCS

W. Gill, CE, discussed the functional design of the CPCS, including the relationships of the calculator to the remainder of the reactor protective systems, the design bases events, system inputs and outputs, system functions, algorithms, power distribution methods, methods to control departure from nucleate boiling, and treatment of uncertainties (see Appendix VIII).

In answer to a question, he noted that the plant will trip for an accident involving a sheared shaft on a main coolant pump, although this is not a design basis accident for the CPCS.

4. CPCS Algorithms and Uncertainties

R. Humphries, CE, discused the algorithms used in the CPCS functions, and discussed the manner in which uncertainties are treated (see Appendix IX).

5. CPCS Hardware and Software Design

E. Brown, CE, discussed the CPCS hardware and software design (see Appendix X). He described how the functional requirements of the CPCS were implemented in the hardware and the software systems of the core protection calculator, the on-line testing features, and the verification program for the CPCS. He said that he believes that the test procedures provide for a wider range of response than hopefully would be obtained from an on-line plant.

In answer to a question, E. Brown said that the CPCS system automatically fails safe on loss of power.

W. Lipinski, ACRS consultant, questioned whether the test procedures actually provide for dynamic response of the system under truly transient conditions.

6. General Questions

In answer to a question, N. Moore, APLC, said that the Applicant believes that any modifications made in the control rod element system as a result of resolutions of the current CE control rod problems, will be capable of being backfitted into ANO-2. For example, additional guide tubes can be inserted in the top of the assemblies to protect the current guide tubes from the full vibrations. If resolution is obtained in the very near future, it is possible that this modification can be made before the reactor goes into operation.

Mr. Shewmon requested that the Applicant provide him with the specification for hold-down bolts for Class-1 equipment.

C. Status of NRC Staff Review

1. Outstanding Items Relating to the CPCS

R. Martin, NRC Staff, said that with review of the CPCS, out of the 27 issues which have been raised by the NRC Staff, only 3 continue to be outstanding and must be resolved prior to the issuance of the operating license:

- Position No. 26: Optical Isolators. Resolution will be required prior to the issuance of the OL. Resolution is anticipated by June 1.

- Position No. 14: Seismic Qualification Review. Resolution is anticipated to have progressed to a point by June 1 that remaining portions of the issue can be addressed with conditions to the OL, without undue impact on the issuance date of the license.

- Position No. 19: Software Change. Resolution should have progressed to a point by June 1 that remaining portions of the issues can be addressed with conditions to the operating license, without undue impact on the issuance date of the license.

2. Review of CPCS

L. Beltracchi, NRC Staff, discussed the Staff's review of the CPCS, including the current status, the safety overview, the criteria for the review, the methodology used, and the test audits (see Appendix XI).

3. Safety Significance of CPCS

W. Hodges, NRC Staff, said that the design basis for the CPCS is the loss-of-flow event. The CPCS is also capable of protecting against a number of the anticipated operational occurrences, which include uncontrolled control element assembly withdrawal, control element assembly misoperation, boron dilution, excess heat removal, and steam generator tube rupture. In addition, the CPCS also provides some protection for a steam-line rupture event with loss of off-site power, and for a main coolant pump shaft seizure when loss of flow can be detected. The NRC Staff has investigated the backup protection that is available for each of these postulated accidents in the event of the failure of the CPCS concurrently with one of these accidents. For several of the postulated accidents, the CPCS does not provide the first trip even when it is operating, therefore there is no change in the consequences to the plant at such times, whether the CPCS is operating or not. The only accident for which there is not automatic backup for the CPCS is a control element assembly misoperation, which requires manual backup by the operator. In such an event, the operator is assisted by alarms from the COLSS, and also by indications on the control panels. If the CPCS does fail to operate in the control element assembly misoperation event, the plant is in the same mode of operation in which it would have been had there not been a CPCS installed. For this particular event, Combustion Engineering plants always required a manual trip.

W. Lipinski asked whether a program has been developed to examine the performance of the CPCS system after it is placed into operation to determine that it does indeed function in the manner in which it is intended to function.

W. Hodges said that during startup testing, a loss of flow test will be run at no power. Power inputs will be simulated to the CPCS at several power levels during these trips and the function of the CPCS will be monitored. There will also be a zero power trip based upon the input from the real pumps. He said that the purpose of the vendor's Phase 2 testing of the system, after the software had gone through a rather extensive testing program to determine that there were no coding errors, was to check the system functionally to see whether the integrated system would perform as designed. During this Phase 2 testing, the simulator that was used was noisy, and may have been nonrepresentative of the type of noisy signals which might be received from an operating plant. Evaluation was obtained of the behavior of the dynamic components of the algorithms themselves used for the DNBR calculations. During these tests, although the scatter of the signals fell outside that predicted, the time to trip was an order of magnitude less than the required time to trip as demonstrated by design analysis codes. Further, the simulator provided a capability to dynamically vary all input simultaneously.

W. Hodges, noted for an example, that the most limiting transient is the core pump loss of flow. This case was analyzed on the simulator, all of the inputs were ramped in a predicted manner based on design calculations, all of the inputs were varied simultaneously, the trip output was obtained, and the time to trip was compared with the required time of trip as predicted by the design codes, and this was found to be conservative. Further, to ensure that there were no coding errors, the test cases were reperformed on the single channel test system at Windsor, where the dynamic inputs could be held steady. The results of these tests were compared with the FORTRAN coding, and the difference was very small between the DNBR outputs and the local power density outputs. Based on these tests, the NRC Staff was satisfied that the scatter arising from the Phase 2 testing was not due to coding errors, but rather to a noisy simulator. However, the NRC Staff is requiring that process noise be evaluated in the plant during startup testing to insure that excessive trips owing to the dynamic component of the thermal power algorithms themselves cannot result.

W. Gill, CE, discussed noise levels in operating CE plants. He noted that measurements have been made in a number o. plants, including St. Lucie, and that a spectrum of frequencies and amplitudes have been identified which would be expected to be found in ANO-2, assuming that ANO-2 will behave as the other plants have. The single channel CPCS system at Windsor, is being exposed to this spectrum of amplitudes and frequencies, and this program is approximately 80% complete.

In a discussion regarding the linking of the CPCS to the reactor computer, F. R. Naventi stated that the NRC Staff made a decision (position 20) not to conduct a safety review of these combined systems in the case of ANO-2. Since this area has not been reviewed as a safety system, and the possible interactions and consequences not considered, the combined use of the two systems will not be authorized during routine operation. As a continuation of this discussion, E. P. Epler suggested that, before use of the CPCS and the reactor computer as a combined system should be allowed, the interactions between the two systems should be considered, so that failure of the safety system from such events as common mode failure or interference by the nonsafety system can be precluded.

In answer to a question regarding the issues raised by J. Calvo, NRC Staff, in his memo of June 24, 1977, L. Beltracchi said that many of these concerns were also NRC Staff concerns. Changes have been made in the ANO design to resolve these issues. R. Tedesco, NRC Staff, said that J. Calvo was not involved in the last part of the ANO-2 review, and that his concurrence on the resolutions of these issues has not been obtained.

4. Data Links to Plant Computer

F. R. Naventi discussed the data links to the ANO-2 computer (see Appendix XII).

5. Status of Project Review

D. Martin, NRC Staff, discussed the overall status of the project review, noting recent resolutions to formerly open items, the status of open items and the expected resolution, and new items that have been identified recently (see Appendix XIII).

6. Applicant's Response

D. Rueter said that the Applicant agrees with the interpretation of the NRC Staff regarding the status of the project in the review.

In answer to a question regarding the number of experienced personnel to be transferred from Unit 1 to Unit 2, R. Terwilliger, APLC, said that five shift supervisors, five plant operators, five assistant plant operators, and four waste control operators who will operate Unit 2 have also had experience on Unit 1. He noted that, in agreement with the NRC Operator Licensing Branch, because of the dissimilarity between Units 1 and 2, there will be very few operators that are licensed for both units.

D. Caucus

The Members provided their opinions concerning matters discussed above and identified those matters that they believed should be addressed in a report concerning this review.

The Committee agreed unanimously that it would try to write a favorable report on the Arkansas Nuclear One, Unit 2, Nuclear Power Plant.

Mr. Kerr requested that the NRC Staff compare the ANO-2 docket with other dockets recently reviewed to determine trends in the number of open items in the reviews being considered by the Committee. L. Crocker, NRC Staff, agreed to provide such a summary.

III. Meeting on Liquid Pathway Generic Study (Open to Public)

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

A. Subcommittee Report

Mr. Moeller, Floating Nuclear Plant Subcommittee Chairman, discussed the history of the development of the Liquid Pathway Generic Study (NUREG-0440), noting that this study compares the radioactive material releases from Class-9 accidents from floating nuclear plants with similar accidents at land-based plants (see Appendix XIV). He noted that reports were received from a number of ACRS consultants (see Appendix XV).

Mr. Moeller stated that the Committee's mission in this meeting is to ascertain whether NUREG-0440 adequately addresses the questions raised by the Committee at previous meetings. Of particular interest is the methodology used in the evaluation. He suggested that subsequent meetings may be needed to review the acceptability of the risks involved or the need for design changes to make the risks acceptable.

R. Foster, ACRS consultant, pointed out that while the results of both atmospheric and liquid pathways have been examined, there is a very important difference in the two pathways in that the liquid pathway may provide a prompt release of sump water, as contrasted with a second more delayed release from the molten core. It is possible that the prompt release of the sump water may in fact lead to consequences which are substantially greater than those from the core melt.

I. Catton, ACRS consultant, offered the following opinions:

- The NRC Staff calculations of leach rate of debris as presented in NUREG-0440 are best estimate calculations, rather than upper bounds.

- Fragmentation of the molten core from a steam explosion has not been adequately considered.

[Note: R. Vollmer coordinated presentation for the NRC Staff.]

B. NRC Staff Presentations

1. Introduction

R. Vollmer summarized the major conclusions drawn from the current revision of NUREG-0440, Liquid Pathway Generic Study (see Appendix XVI). In his conclusion, he stated that the NRC Staff believes that, in the very unlikely event of a core melt, the floating plant presents a greater hazard to the health and safety of the public, in that the sump water, and a large amount of the contained radiation in the core, would be removed through the liquid pathway before effective interdiction could be instituted. Further, the NRC Staff believes that effective liquid pathway interdiction would result in a high social or economic impact. Actual cost figures are currently being developed.

R. Vollmer said that the Liquid Pathway Generic Study represents a last major effort in the completion of a safety evaluation report (SER) and an environmental statement for the floating nuclear plant. It is important to the NRC Staff that the Committee provide a report containing the Committee's view on its assessment on the adequacy of the study, as input to both the environmental impact statement and the SER.

C. Offshore Power Systems Company Conclusions

D. Walter, Offshore Power Systems Company (OPS), discussed the Liquid Pathway Generic Study, specifically both the NRC Staff and the OPS conclusions. Included in this discussion were the dose pathways considered for the study, the FNP liquid pathways dose consequence calculations made by OPS, NRC Staff conclusions as included in NUREG-0440, the scope of the NRC Staff Liquid Pathway Generic Study, the results of recent Sandia Corporation leach tests, calculated dose consequences from Class-9 accidents, a comparison between liquid and air pathways, air pathways dose consequences, the calculated accident risk for air pathways, and a comparison of the risks from air pathways and liquid pathways (see Appendix XVII).

D. Technical Presentations

1. Dose Comparisons

G. Chipman, NRC Staff, compared calculated doses released from Class-9 accidents postulated in both floating and land-based plants (see Appendix XVIII).

In answer to a question, G. Chipman noted that, if the airborne material is released through the bottom of the containment, such release caused by a melt-through, the airborne release for a floating plant would be approximately the same as for a land-based plant.

Mr. Bender suggested that it would be useful to examine a variety of current land-based sites to consider what the economic and environmental consequences of a Class-9 accident might realistically be.

2. Core Melt-Through Penetration Mode and Steam Explosions

T. Spies, NRC Staff, discussed several postulated modes for penetration of containment by a core melt and also the consequences of postulated steam explosions (see Appendix XIX). He addressed the concerns regarding steam explosions that were raised during the March 22 subcommittee meeting.

In a discussion among Members, ACRS consultants, and T. Spies, it was established that very limiting conditions are required in order for a steam explosion to take place. Critical to this interaction are the temperature of both the cooling mass and the heating mass, and the further requirement that the heating mass must be a liquid. One of the prime requirements for a steam explosion is a very high rate of heat transfer.

3. Applicant's Response to NRC Staff's Presentation of Steam Explosions

D. Walker, OPS, said that it is important to note that the calculated releases from the leaching of the molten fuel and those from the sump produce comparable dose consequences. The radionuclides most effective in producing these dose consequences are cesium and strontium. Important to the overall consequences are the leach rate, and whether or not the sump material can be interdicted. If significant leaching takes a week, and the sump material finds itself in the sea water almost immediately, the contribution to dose from the leaching becomes insignificant. He noted the Applicant's disagreement with the NRC Staff conclusion that the realistic concern for dose to the public should not be, as the NRC Staff contends, Category-5 releases. He said that the Applicant believes that both the Category-5 and 7 releases are insignificant because of the low probability of their occurrence.

4. Magnitude of Pressure Transient Following a Steam Explosion

D. Walker, OPS, summarized the Applicant's analysis to determine the magnitude of pressure transients following a steam explosion, noting the calculated initial peak pressure, the physical configuration of the reactor pressure vessel and the barge, time factors involved, and heat transfer calculations (see Appendix XX). He noted that both the Applicant and the NRC Staff have concluded that steam explosions are unlikely in the event of a core melt accident, and that, even if the accident should occur, it is unlikely that the interactions would involve any large fraction of the core debris in a single event.

I. Catton, noting that the model used by the Applicant involved a pressure pulse from a noncondensible gas, questioned whether these calculations were applicable to a steam bubble.

In a discussion regarding possible damage to the second unit at an FNP site from a steam explosion occurring beneath the first unit, B. Haga said that the neighboring plant could be shutdown safely even in the event that its barge was ruptured, sank, and was in a tilted position.

D. Walker said that the study indicated that the time the material would be held in the sumps was in the range of 10 to 35 days. He said that, following the core melt-through, the plant would still float, even with a hole, and that tidal flushing of the sumps would not occur.

5. Coupling of River, Estuarine, and Ocean Doses

J. A. Nutant, OPS, discussed the estimated liquid pathway results (man-rem per core-melt), plant site rankings based on liquid pathway man-rem consequences without interdiction, airborne release estimates for a steam explosion inside the hull, diameters of released particles, and dose consequences from particle transport (see Appendix XXI).

In answer to a question, D. Walker said that the airborne releases are approximately the same for all plants of equivalent size, and that the dose consequences are determined by the population distribution in the vicinity of the plant, and vary from site to site.

D. F. Bunch, NRC Staff, noted that, in these presentations, the Staff made the same assumptions that were used in WASH-1400, so that comparisons could be made in familiar terms.

6. NRC Staff's Response to Applicant's Presentations

G. Chipman noted that the NRC Staff made its comparison on the basis of expected results, what could occur, and what would be expected to occur if the core-melt did occur. The NRC Staff analysis showed that risk from the air pathway for a floating plant is comparable tc that from a land based plant. He said that the NRC Staff disagrees with the Applicant's analysis with regard to sump water, and that more than 10% of the sump water would be released within one week.

P. Haga said that he believes that the Applicant has shown that, when realistic interdiction is considered, the consequences of accidental releases of radioactivity to the liquid pathway are not very important when compared to the consequences of accompanying releases to the air pathway. The conclusion is that, for accidents beyond the design basis, floating nuclear plants have been shown to be comparable to land-based plants.

Therefore, the applicant sees no need for any extra design features, and does not plan any. He offered the hope that this meeting would be sufficient for the Committee to prepare a final report on the OPS application.

R. F. Foster, ACRS consultant, summarized his conclusions as follows:

- The air pathway is going to be at least as important as the liquid pathway, and perhaps as much as an order of magnitude more important.

- The sump release is at least as important as the leach rate from the core, and perhaps five times more important.

- Interdiction is practical in terms of the liquid pathway, although probably not at the source.

- The problems are economic as well as risk to health and safety.

I. Catton summarized his conclusions;

- He questioned the size of the calculated steam explosion.

- He questioned whether the steam explosion was properly addressed with regard to the particular materials involved.

- The overall consequences from a Class-9 accident may be only a factor of two different between floating and land-based plants.

E. Caucus

The Committee agreed unanimously that it would attempt to write a report on NUREG-0440, Liquid Pathway Generic Study, at this meeting.

[Note: Time did not permit the completion of this report at the 216th ACRS meeting. It is anticipated that this report will be completed at the 217th ACRS meeting, in May, 1978.]

IV. Meeting on McGuire Nuclear Station, Units 1 and 2 (OL) (Open to Public)

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

A. Subcommittee Report

Mr. Plesset, discussed the history of the Subcommittee's review of the application for an operating license for the McGuire Nuclear Station, Units 1 and 2, discussed the overall plant parameters and site location, the outstanding issues, and briefly compared this station with other stations of similar design (see Appendix XXII). He noted certain changes made in the ice condenser portion of the plant as a result of the operating experience obtained at the D. C. Cook Nuclear Plant. He noted that McGuire is the first plant to be reviewed that utilizes an Upper-Head Injection (UHI) system as part of the ECCS.

[Note: K. S. Canady coordinated presentations for the Applicant; R. Birkel, for the NRC Staff.]

B. Status of NRC Staff Review

R. Birkel discussed the chronology of the review of the application for an operating license for the McGuire Nuclear Station, and he noted that there were no differing technical views expressed by members of the NRC Staff relating to the review of the McGuire safety analysis report, as summarized in the Safety Evaluation Report (SER), NUREG-0422. He noted that the environmental portion of the public hearing was completed on April 22, 1977, and that it is anticipated that the radiological safety portion of the AS&LB hearing will be resumed in late June, 1978. This hearing is being contested. He noted that, with the exception of a few outstanding issues, the NRC Staff has completed the review of the McGuire Station. He noted the status of the 21 issues listed as outstanding in the SER as of March 1, 1978: 10 issues have been satisfactorily resolved, 3 are partially resolved and generic in nature, and the remaining items are expected to be resolved in the near future. Referring to section 1.6, beginning on page 108 of the SER, he listed the status of outstanding items as follows:

Items Requiring Information from the Applicant:

Item 1: Duke Power Company has orally committed to providing the required information and justification no later than July 1, 1978. NRC Staff finds this acceptable.

Item 2: The Applicant is committed to submit his stress analysis report by October 1, 1978. NRC Staff finds this acceptable.

Item 3: Resolved.

Item 4: Resolved.

Item 5: Resolved.

Item 6: Resolved.

Item 7: Generic, partially resolved.

Item 8: Resolved.

Item 9: The Applicant has stated that a response will be provided by May 8, 1978. The NRC Staff finds this satisfactory.

Item 10: Generic, partially resolved. The Applicant has agreed to submit additional information to identify the model number and the requalification document reference for information for class-1E equipment. The NRC Staff will review this information and compare it with the Westinghouse Topical Report. The Applicant has also agreed to submit additional documentation on balance-of-plant electrical connections and terminals inside containment. The NRC Staff believes that the review of the requalification program and the correction of any equipment deficiencies can be completed prior to the issuance of an OL, currently scheduled for December, 1978.

Item 11: Resolved.

Item 12: Resolved.

Item 13: Resolved.

Item 14: The NRC Staff has not yet completed its evaluation of the steam line break accident which was recently filed as Revision 28 to the FSAR.

Issues Requiring NRC Staff Evaluation

Item 1: This issue is open. The NRC Staff review is in progress, but major problems are not expected.

Item 2: Partially resolved. The NRC Staff has approved the UHI ECCS Generic Evaluation Model. The Applicant will submit the Appendix K analysis using this model by May 8, 1978. Major problems are not expected.

Item 3: This issue is resolved.

Item 4: This issue is resolved.

Item 5: The NRC Staff review of the McGuire fire protection design is in progress. A revised fire protection analysis was submitted by the Applicant on March 22, 1978. A site fire protection review is scheduled for the week of April 17-21, 1978, with staff positions to be issued by May 15, 1978. The Applicant's response is due on June 29, 1978. The review is projected to be completed by mid-August.

Item 6: A request for additional industrial security information was issued April 7, 1978, with response from the Applicant due in mid-May. The review is scheduled to be completed by mid-July.

Item 7: Updated financial information from the Applicant was filed April 7, 1978.

With regard to ACRS generic matters, a discussion of NRC Staff efforts leading to satisfactory resolution is contained in a letter to Chairman Bender dated October 25, 1977. Appendix B of the McGuire SER provides the current status of these matters as they relate to this plant.

R. Birkel concluded that the NRC Staff does not consider any of the remaining eight open items or the three generic items to pose serious or major problems. Each item will be resolved to the NRC Staff's satisfaction well in advance of a decision to issue an operating license.

Mr. Plesset identified two additional items for which the NRC Staff was requested by the subcommittee to be prepared to answer questions:

- The effect of inadvertent injection by the upperhead injection system when the plant is being shutdown and pressure in the primary system falls.

- At the subcommittee meeting, D. Riley, California Environmental Study Group, had noted concern that the headbolts that secure the reactor vessel head to the vessel could fail in an "unzippering" mode of all the bolts, and that the head might be blown through the containment.

In answer to a question, R. Birkel identified the two contentions raised in this case by intervenors:

- financial qualification of the Applicant, and
- the effects on the McGuire Plant of seismic events occuring outside the tectonic province in which the plant is located.

Both issues have been addressed in the SER.

C. Introduction

K. S. Canady, Duke Power Company (Duke) discussed the location of the site, the Duke service area and transmission grid, the Duke generating capacity, the corporate organization of the company, and the general plant layout (see Appendix XXIII).

In answer to a question, K. S. Canady noted that security protection for McGuire will be contracted to an outside organization. He also noted that quality assurance and control organizations report to a corporate quality assurance director, whose organization is outside that of the steam production department. He also noted that there is a decommissioning task force developed within Duke.

In answer to a question regarding the use of emergency diesel generators at McGuire, instead of an emergency line from a nearby hydroelectric station as at Oconee, K. S. Canady said that Duke was unable to reach agreement with the NRC Staff on the use of hydropower for emergency power at this station.

In answer to a question regarding the NRC Staff's reluctance to permit McGuire to rely upon the hydro station at the Collins - Ford Dam for emergency power, K. Kniel, said that the NRC Staff requirements are that emergency power must meet Seismic Category-1 qualifications and tornado missile protection: the Collins - Ford hydro station does not meet these requirements.

In answer to a question regarding hydroelectric generators, C. Wylie, Duke, said that at Keeowee Dam, the experience with the hydro station since the two units went in service in April 1971, was that out of approximately 500 manually initiated starts, there have been 7 failures to start. These failures were not in the hydroelectric unit itself, but in the manual initiation circuits. Once the initiation signal was received, the turbines started. With respect to diesel reliability at McGuire, with 4 diesel generators being tested, and with a criterion of reaching speed in 11 seconds, in 545 starts, 544 were successful, and the other reached speed in 12 seconds.

D. Emergency Planning (Questions Only)

In answer to a question regarding relationships with the North Carolina Department of Human Resources, Radiation Protection Branch, L. Lewis, Duke, said that Duke is working with this cognizant agency in North Carolina to prepare the draft of the State of North Carolina's emergency plan. This plan will comply with the NRC requirements, the NRC Regulatory Guide and Checklist, and supplements to the Checklist. Appendixes will be prepared for each of the nuclear stations within the State of North Carolina. He noted also that Duke has well-established communications with the South Carolina Department of Health and Environmental Control, Radiation Protection Branch, which were developed with the licensing and operation of the Duke's Oconee Nuclear Station in South Carolina. In addition, the Southern Emergency Response Council, made up of representation of all of the Southern States, effectively interacts with the states and provides help to each state in the event of an emergency.

E. ECCS Design

1. UHI Analysis vs. Measurement

N. Lauben, NRC Staff, discussed the Staff's review of the UHI evaluation model from January, 1975 through December, 1977; identified the issues considered since December, 1977, discussed the confirmatory comparisons for the split downcomer model and downcomer model sensitivity; and compared accumulator flow rates, upperhead temperature, guide tube flow, and support column flow, as calculated by the Westinghouse UHI models and the Sandia UHI model (see Appendix XXIV).

2. Specific Plant Analysis

S. Israel, NRC Staff, said that when the Applicant submitted a LOCA analysis for the large break, using the August, 1977 model, the analysis indicated that the double-ended cold-leg guillotine break with a Moody discharge factor of 0.6 was the most limiting break, and calculated a peak clad temperature of 2164 °F. More recently, the Applicant has submitted small break analyses which indicate that a six-inch break is the most limiting break with peak clad temperature of approximately 1500 °F. The large break calculations took credit for a finite containment backpressure based on the LOCA calculations. These calculations are still under review by the NRC Staff. In reviewing the LOCA calculations, the Staff is interested in sensitivity studies regarding the application of UHI. Of interest also is the break spectrum, to assure that the most limiting breaks have been identified. Following this discussion, the Applicant will present analyses of these matters, which the NRC Staff has not seen yet. However, based on the experience obtained during the long review of the UHI evaluation model, the NRC expects that the results will show peak clad temperatures below 2100°F for peaking factors of around 2.3.

3. UHI Analysis

W. Johnson, Westinghouse, discussed the UHI analysis, including predicted ECCS performance, the analysis conditions, compliance with Appendix K, modifications to the ECCS model, loss-of-coolant/temperature analysis code, zircaloy-water problem description, and the calculated peak clad temperatures using the approved UHI evaluation model for a double-ended cold-leg guillotine break (see Appendix XXV). He discussed the recently discovered metal-water problem in the Westinghouse evaluation model. This error was originally discovered by Fromatome, Westinghouse's French licensee, when they found that the total heat generation rate calculated from the metal-water reaction was low by a factor of 2. Further investigation showed that the evaluation model had a logic error. He mentioned the various reasons why Westinghouse had not found this error previously. Westinghouse believes that this error, in terms of licensing requirements, is significant, but that in terms of margins to safety, it is not.

D. Ross, NRC Staff, disagrees with Westinghouse that the margin of safety was unchanged.

In answer to a question, J. Cermak, Westinghouse, said that for non-UHI plants, the effect of this error is in the neighborhood of from 0.15 to 0.25 in peaking factor. With respect to UHI plants, the effect is approximately 100 ° in the Appendix K calculated temperature.

In answer to a question regarding the advantage of higher peaking factors, K. S. Canady said that, with a low peaking factor, the plant must be operated to closer limits than with a higher peaking factor. If there is a relative high peaking factor, the plant can be operated in either a base loaded or load-following mode; the higher peaking factor gives more operational flexibility.

F. Stud Bolts

R. E. Tome, Westinghouse, said that his presentation was prepared as a result of the question of stud bolt failure in the pressure vessel head as raised at the subcommittee meeting. Each reactor vessel has 54 7-in. diameter closure studs threaded at each end. There are 8 threads per in., and the length of the thread engagement between the stud and the vessel flange is 9 in. The length of the engagement between the stud and the nut is 7 in. Studs, nuts, and washers are made from SA-450 Class-3 material with an ASME code specified minimum room temperature yield strength of 130,000 psi, and an ultimate strength of 145,000 psi. The closure stud assemblies for McGuire Units 1 and 2 were designed, analyzed, fabricated, and inspected to the requirements of the 1971 edition of the ASME code, Section III, and December, 1971 addenda. The design conditions for the stud assemblies are to resist an internal pressure of 2500 psi in the reactor vessel at 650°F, while the normal operating conditions are to resist 550 psi pressure at 550 °F. There are several stress limits that must be met by the studs during plant operation. The studs are sized so that the average membrane stress in the studs is less than 1/3 of the code specified minimum yield strength for the design conditions. The shear stress in the threads is less than 0.6 of this allowable membrane stress limit. The maximum membrane stress during normal plant operation occurs at the end of heatup, and is 47.6 ksi, vs. a code allowable of 73.6 ksi, which is 2/3 of the specified minimum yield strength at operating temperature. The maximum membrane plus bending stress during normal plant operations also occurs at the end of plant heatup, and is 98.6 ksi vs. a code allowable limit of 110 ksi, which is the specified minimum yield strength at 550 °F. During steady state operation the stresses in the studs are at a membrane stress of 24.4 ksi, and a combined membrane plus bending stress of 57.5 ksi. The hypothetical case for one stud failing instantaneously at the end of the heatup was considered. The dynamic effect of suddenly applying all the loads from the stored energy in the flange below the failed stud and the pressure load increase from the failed stud on just 2 adjacent studs next to the failed stud was calculated to be an average stress across the two studs of 65.7 ksi, which is still below the code allowable limit of 73.6 ksi for normal operation. The maximum calculated membrane bending stress in those studs was found to increase to 116.4 ksi, which will result in some localized yielding. Therefore, there would not be a progressive stud failure mode, or zipper effect from this assumed stud failure. The most critical stress locations on the studs, the threads, were evaluated using the ASME-specified fatigue reduction factor of 4. The significant stress cycles on the studs were 57 specified startup cycles including a tensioning and untensioning operation, and 143 additional startup cycles without the studs being tensioned or untensioned. The total usage factor for the studs was calculated to be 0.6 vs. code allowable value of 1.0, using the ASME design peak curves. The ASME Code provides design fatigue curves for temperatures between room temperature and 700 °F, which covers the operating temperature of McGuire. The temperature effect is compensated for by code requirements. Bolts will be tensioned to a maximum membrane stress of 55 ksi, the nut will be tightened, and the tensioner relaxed, which would reduce the membrane stress in the stud. The elongation applied to each stud by the tensioner is controlled by a hydraulic pressure gauge on the tensioner pumping unit. Elongations of all studs are held to 0.0051 inches, with a tolerance of plus or minus 0.0002 inches for normal operation. In addition to other requirements for the material, it also must meet a Brinell hardness between 302 and 388. In addition, Section 3 of the ASME Code requires Charpy impact testing be performed on the material. Charpy and tensile tests are performed on samples removed from one bar of each heat of material represented or for each charge of 10,000 lb. of material, whichever is less. A review of the record for the McGuire stud material processed to these requirements indicates that the axial strength and ultimate strength in all cases exceed the code-specified minimum allowables. Periodic nondestructive testing, both volumetric and surface, are required for these studs during fabrication and plant life. The stud material receives a 100% volumetric examination by ultrasonic testing (UT) in both the radial and axial direction after heat treatment and prior to threading and must meet the ASME Section III acceptance standards. After final machining, the studs receive a 100% magnetic particle examination to detect nonaxial surface indications. The studs then receive a 100% UT axial scan after hydrostatic testing in the shop, prior to their shipment to the plant site. The studs receive a 100% visual examination and a 100% UT examination from one end. During the plant life, the studs must be 100% reinspected on both surfaces, either a UT or MT and UT exam, in 10 years with 5 to 33% of the studs being inspected every 40 months.

R. Tome concluded that the Applicant believes that the reactor vessel studs, designed and fabricated to the requirements of Section III of the ASME Code, and inspected to both Section III and XI requirements of the code as required by the federal regulations, will not fail and will operate in a reliable manner for the life of the plant. This conclusion has been borne out by experience on operating Westinghouse plants.

Mr. Siess noted that this presentation did not answer the questions raised by Mr. Riley, regarding possible failure at a pressure of approximately 7000 psi under ATWS conditions, and his claim that 18% of the stud bolts may be below standard strength.

R. Tome claimed that none of the studs are below strength, that any defective studs are replaced before operation, and that there is no concern that the system could reach 7000 psi pressure and rupture the studs, since other components would fail first.

Mr. Shewmon requested informat[illegible] regarding the specifications to which the holddown bolts [illegible]e auxiliary feedwater turbine are manufactured. K.S. Canad[illegible]d to provide the information.

G. Effect on Fuel of Core Radial Diffe[illegible]ressure from Asymmetric LOCA Loads

J. Cermak, Westinghouse, said that Westinghouse has performed a scoping calculation of the forces from a pressure wave which moves the core barrel and then the rarefaction wave moving back across the inside of the core causing a differential pressure across the fuel assembly. This load would be the equivalent of 10% of the hydraulic load on the fuel. Westinghouse believes that the conservatism in the design can easily absorb this 10% pressure, which was not considered in the initial design. Further, calculations of the forces on the fuel assembly of McGuire, considering both the seismic and hydraulic loading on the fuel assembly, totaled less than 2000 lbs. Adding the 10% calculated in the bounding calculation, the total loading would be less than 2200 lbs. Both of these loads are less than the lower limits of the experimental data, which is 3200 lbs. Westinghouse concludes that there would be no significant deformation of the fuel assembly, and therefore believes there is no problem.

H. General Questions

In answer to a question regarding whether an operator action at a remote shutdown panel could cause an unsafe condition, T.C. McMeekin, Duke, said that there is protection against such a situation. A remote shutdown panel alarm will register in the control room if the panel is opened. Further, a set of transfer switches must physically be opened to transfer control from the control room to the remote control panel.

I. Caucus

The Committee agreed unanimously that it believed it could write a letter favorable to the application for an operating license for the McGuire Nuclear Station, Units 1 and 2.

V. Meeting with the NRC Staff on Recent Operating Experience, Licensing Actions, Generic Matters Relating to LWRs and Future Agenda (Open to Public)

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

A. Davis-Besse-1: Implementation of ACRS Recommendations from the ACRS Report of January 14, 1977

L. B. Engle, NRC Staff, discussed the implementation of the ACRS recommendations included in the Committee's report of January 14, 1977. (For background, see Appendix XXVI.) He said that the purpose of this report is to update the Committee to the current status. He noted that Supplement No. 1 to the SER was issued on April 22, 1977, the same date that an OL was issued for Davis-Besse, Unit 1. Although the OL was issued authorizing full power, 960 MWe, the operation of the facility was restricted to a sequence of operational modes until preoperational test, startup test, and other items were completed to the satisfaction of the NRC Staff. This OL stipulated 19 conditions that imposed limitations on plant operations, and required special reports and/or modifications to be completed at specific times following the date of the issuance of the license. Since the issuance of the license, 6 conditions to the license have been removed by amendments, supported by SERs, and 2 conditions have been revised.

The reactor attained criticality on August 12, 1977, initial electricity was produced on August 23, and 75% of full power was attained on January 23, 1978. A reactor power of 90% was obtained for 2 days on February 15 and 16, 1977. However, condenser problems required the Licensee to reduce operating power to 75%, until April 3, when power was increased. 100% of rated power was reached on April 4. Based on recommendations by Babcock and Wilcox (B&W), the NSSS vendor, on April 5 the Licensee reduced plant operation to 3 pumps to reduce flow and is currently operating at about 75% of full rated power. Davis-Besse has burnable poison rod assemblies, and is therefore affected by the failure of one of these assemblies in the Crystal River Plant. The cumulative service factor has been about 75%, and the unit forced-outage rate has been about 25%. The shutdowns were required for repairing and servicing equipment primarily in the secondary system. The first scheduled refueling outage is planned for late 1979.

In answer to a question, L. B. Engle said that the condenser problems to which he referred involved some tube leaks. It has not yet been determined whether these leaks are from flow-induced vibration as a function of power. Tubes will be inspected during a planned outage.

L. B. Engle reviewed the items raised in the Committee's January 14, 1977 report:

- Increase of seismic design basis from 0.15g to 0.2g. The NRC Staff stipulated in the operating license that the Licensee shall submit a seismic reanalysis and evaluation to the NRC for its review in sufficient time to obtain Commission approval of the adequacy of the plant systems needed to accomplish safe shutdown of the plant and continued shutdown heat removal prior to startup following the first regularly scheduled refueling outage. In performing the reanalysis, a safe shutdown earthquake of 0.2g shall be applied at the foundation level of the plant, and the response spectra shall be as specified in Reg. Guide 1.60. The NRC Staff is in the process of developing guidelines for this seismic reanalysis.

- ECCS. The NRC Staff has reviewed revised nucleate boiling logic proposed by B&W, which does not allow return to nucleate boiling after critical heat flux conditions are reached. The NRC Staff has determined that the revised logic was an appropriate change to be incorporated in the B&W evaluation model, that the overall effect of the change on peak clad temperature was insignificant, and that it met the Acceptance Criteria. The Applicant has submitted additional analyses correcting for fuel pin pressure errors and erroneous flow resistance values for the reactor vessel inlet nozzle. The NRC Staff has determined that the ECCS analysis for Davis-Besse, Unit 1, is in accordance with Appendix K.

- Large Break Analysis. The operating license stipulated that, within six months from its issuance, the Licensee shall provide additional supporting analysis for the large break spectrum to document the exact margins, and should provide to the NRC Staff reactor coolant system flow data. The Licensee submitted the large-break spectrum on October 21, but, because of delays in plant operation, the coolant system flow data were not available. The license condition was revised to require that within 30 days following 2 weeks of sustained reactor power operation at a power level of 90% or greater of rated thermal power, the Licensee provide operating reactor system coolant flow data. However, during the 2 days that the plant was at 90% rated thermal power, the Licensee was able to obtain the system flow data, and is getting ready to submit the information to the NRC Staff by mid-April.

- Improved Radiation Surveillance in Ohio. The State of Ohio has indicated to the NRC Staff that it is initiating a program which will eventually qualify for an NRC state contract for technical aid.

- Long Term Capability of Hermetic Seals. Environmental qualification of equipment is being pursued by the NRC Staff as part of a Category A generic activity, Task Action Plan A-24, Qualification of Class IE Safety-Related Equipment, to assure proper performance of seal materials during plant operation.

- Instrumentation to Follow the Course of an Accident. The NRC Staff has issued Regulatory Guide 1.97 (Rev. 1), and a technical activities steering committee was established on August 22, 1977. The NRC Staff has concluded that the instrumentation to monitor post-accident conditions met the NRC Staff criteria and was acceptable. However, Davis-Besse 1 does not meet Position C.3 of Regulatory Guide 1.97, Revision 1, and guidance is being developed in this area.

- ATWS. This is a generic matter. The NRC Staff has prepared a draft generic technical report on ATWS which incorporates the comments and concerns of the industry, including the Babcock and Wilcox Company. This draft report is currently being reviewed by NRC management and the NRC Staff. An open meeting will be held on April 19, 1977, to discuss the studies concerning ATWS.

- Fire Protection. The Licensee has submitted its fire hazard analysis report, and the NRC Staff has determined that the report was not adequate for determining the fire protection program in accordance with Appendix A to the Branch Technical Position. Therefore, a condition was placed on the license, stipulating that within three years from the date of issuance of the license, the Licensee shall increase the level of fire protection in the facility to the levels recommended in Appendix A, or with alternatives acceptable to the NRC Staff. Prior to startup following a first regularly scheduled refueling outage, the Licensee shall implement Section B of Appendix A, Administrative Procedures, Controls, and Fire Brigade, and Section C of Appendix A, Quality Assurance Program. Since the issuance of the OL, the reevaluation of the fire protection program has continued. The Licensee has presented additional information on the installation of fire retardant seals for electric penetrations. The

NRC Staff has concluded that these seals through fire barriers as installed at Davis-Besse are conditionally acceptable. However, the NRC Staff has required that the Licensee perform certain full-scale testing to verify the adequacy of the configuration and installation of the seals that represented the worst-case departure from sections originally tested under the ANSI E119 standard. The NRC Staff is scheduled to complete the Davis Besse Unit 1 fire protection review before the end of of calendar year 1978.

- Industrial Security. The NRC Staff has reviewed the Licensee's amended security plan as required by NCFR 5073.55, and completed their Phase 1 review in September of 1977. The Licensee has submitted a modified security plan. This plan is being evaluated.

B. Oconee: Microseismicity

J. Kelleher, NRC Staff, discussed a microearthquake swarm occurring at the Oconee site in January, 1978. He noted that the cause of this swarm is not clear. The maximum magnitude of these microearthquakes was in the range of 2 to 2.5 on the Richter scale. He noted that there was a network of microseismometers located at this site.

In answer to a question, J. Kelleher said that there are very few areas in Eastern United States that are currently being monitored for microseismicity, and that it is possible that similar swarms of microearthquakes could be taking place elsewhere without their being detected.

Regarding the reported swarm, only one earthquake was reported as being "felt". (For the data presented, see Appendix XXVII.)

C. Combustion Engineering Plants: Control Element Guide Tube Wear

H. Levin, NRC Staff, stated that a problem of cracking of control element guide tubes was first identified in the Millstone 2 Plant on December 14, 1977. He reviewed the chronology of the actions taken since this problem was first identified, and discussed the design of the components, the safety considerations involved, the observations made on the worn components, the interim fixes accepted by the NRC Staff, the bases for continued operation, and the susceptibility of other NSSS designs to guide tube wear (see Appendix XXVIII).

Mr. Bender suggested that applicants establish whether there are suitable out-of-pile vibration test arrangements for all reactor internals configurations to evaluate the effect of representative working conditions.

D. Crystal River 3: Failure of Burnable Poison Rod Assembly

K. Seyfrit, NRC Staff, first discussed events leading up to the identification of a burnable poison rod assembly failure at Crystal River, Unit 3. On September 12, 1977, following recovery from a scram, a quadrant tilt of 7% was noted, which disappeared after several hours of operation. On January 1 and again on January 3, 1978, there was an alarm on the loose parts monitor associated with the B steam generator; this disappeared after a short period of time. On February 17, there was another alarm on the loose parts monitor associated with the B steam generator, which persisted. The Licensee performed a number of investigations, including examining the chemistry of the primary coolant, and looking at other loose parts monitors. No other abnormalities were noted at that time. Because of the persistence of the noise in B steam generator, one of the reactor coolant pumps was shut down, and power was reduced to about 78%. This reduction in power and flow eliminated the noise at that point. A B&W investigation team was called in, evaluated the data that was available, and confirmed that there were some loose parts in the top of the B steam generator. Operation continued from February 18 to March 3, at which time the Licensee was able to determine that there had been a small amount of steam generator tube leakage, on the order of a gallon a day. The unit was taken off-line on March 3, and a cool-down begun. Observation through a manhole on B steam generator identified some loose parts, including the coupling and spider for the burnable poison rod assembly B-47. In addition, there were other parts identified, including pieces of cladding, and some evidence of damage of the tube ends. On March 13, the reactor vessel head was removed, and a second burnable poison rod, B-52, was observed sticking up. Several poison pins were broken off. Examination of the latching mechanism indicated that one of the balls was missing, and grooves were found in the hold-down latch assembly that corresponded to the location of the missing ball. The remainder of the fuel assembly appeared to be in good condition. The Licensee has postulated three possible causes for the damage, and is investigating to determine the actual cause. The first two assumptions involve a manufacturing deficiency and the possibility that the assembly was not initially latched. The third assumption is that the wear was caused by vibration.

K. Seyfrit stated that at this time there are only two other plants in operation using these burnable poison assemblies, Davis-Besse-1 and Three Mile Island-2. It is likely that the burnable poison assemblies will be removed from Davis-Besse within the next ten days to two weeks, when the plant is shut down for other planned maintenance. He noted minor damage to the steam generator, and inferred that part of the problem is the extension of the tubes approximately 3/4 in. above the tube sheets. The probable fix for this problem may be the removal of the extension of the tubes above the tube sheet. (For details, see Appendix XXIX.)

E. Implementation of Regulatory Guide 1.97, "Instrumentation to Follow the Course of an Accident"

F. Hebdon, NRC Staff, stated that when the work had been completed on Regulatory Guide 1.97 (Rev. 1), the NRC Staff began work on Task Action Plan A-34 to develop detailed acceptance criteria and guidance to be used by Applicants, Licensees, and NRC Staff reviewers to support implementation of this guide. During the development of this Task Action Plan, it was recognized that certain instruments were described in the Guide with such clarity that implementation on that part of the Guide could proceed more quickly than could implementation of the entire Guide. Therefore, the NRC Staff decided to divide implementation of Reg. Guide 1.97 into two phases:

- Phase 1 incorporates the recommendations of position C.3 of the Guide. Position C.3 describes the specific instrumentation to be used if accident conditions degrade beyond those assumed in the FSAR. It was believed that position C.3, by itself, constitutes an interim solution that could be implemented on all operating plants in a timely manner, while the more time consuming case-by-case review described in the remainder of the Regulatoy Guide is completed.

- Phase 2 incorporates the remainder of the regulatory positions in Regulatory Guide 1.97. The principle position is position C.1 which states that for postulated accidents listed in Chapter 15 of Regulatory Guide 1.70, the Applicant shall perform a detailed safety analysis to determine the parameters that should be measured to provide the operator with essential information concerning the nature of an accident and the response of available safety systems.

LaSalle County and Watts Bar were selected for Phase 1, position C.3, and Allens Creek and Sundesert were selected for Phase 2, or for full implementation of Regulatory Guide 1.97. Subsequently, in response to the ACRS report on Diablo Canyon, Diablo Canyon was added as a lead plant for implementation of position C.3. In August 1977, Regulatory Guide 1.97 (Rev. 1) was issued. The NRC Staff has characterized this revision as Category 3, backfit required for all applications in review, and further NRC Staff consideration of individual cases required in order to determine the need for backfitting for all operating plants.

Two problems have developed in dealing with Applicants:

1. Technical questions, such as the definition of identifiable release points described in position C.3 have been raised.

2. Philosophical problems concerning the apparent commitment in position C.3 to include instrumentation to monitor accidents that go beyond Class 8 have been raised.

The NRC Staff has made some progress in clarifying its position and in resolving the Applicants' concerns in these areas. However, work remains to be done in both areas. In March, the NRC Staff sent additional guidance to the Applicants and requested the proposal for implementation of Reg. Guide 1.97 be submitted no later than May 1, 1978.

F. Seismic Monitoring on the Eastern Seaboard

C. Stepp, NRC Staff, described the seismic monitoring networks in Eastern United States, noting that the NRC Staff is funding entirely or partially the operation of these networks. He said that there are two principle networks along the eastern seaboard, the Northeastern Network, and a network around Charleston, SC. The Northeastern Network includes about 30 microseismic monitoring stations that will detect local earthquakes of smaller than magnitude Richter 3. The Charleston network includes about 12 microseismic stations. Another network is currently being installed in the central Virginia region, and will consist of 5 microseismic stations. In addition, another 5-station microseismic network is being established in southwestern Virginia, in the Giles County earthquake zone. It is proposed that these microseismic stations in Virginia be operated by Virginia Polytechnic Institute.

C. Stepp said that four of the microseismic instruments currently located in the temporary network at North Anna will be incorporated into the central Virginia network. He noted that there are currently 17 stations in the Lake Anna area. The effect of the reduction of the number of microseismic stations will be loss of accuracy in pinpointing epicenters of seismic disturbances. The accuracy of the proposed network would be only approximately plus or minus 5 km. Because of the concentration of the four stations at North Anna, the accuracy in this particular area would be on the order of from 1 to 1.5 km. (For details see Appendix XXX.)

The Committee raised no objections regarding the proposal to reduce the number of instruments at the North Anna seismic network.

G. Monitoring Neutron Exposure at Nuclear Facilities

S. Block, NRC Staff, said that, as a result of questions arising regarding the exposure of nuclear plant personnel to unknown amounts of neutrons, and the possible inadequacy of neutron detection and recording instruments, this problem is being considered by the NRC Staff. He noted that the Office of Nuclear Regulatory Research has been requested to initiate a program for the purpose of collecting data on the effectiveness of personnel neutron dosimetry programs at the operating nuclear power plants (see Appendix XXXI). He pointed out that, for certain energy ranges, adequate dosimeters have not been developed, and he recommended that such instruments be developed. He reviewed the neutron dosimetry records that are available from operating plants, and was unable to identify a significant problem. He suggested that the likelihood of overexposure, in the range where instrumentation has not been available, is highly unlikely. However he recommended that further investigations be made. (For details on personnel neutron dosimetry methodology, see Appendix XXXII.)

H. Future Agenda

The Committee approved a tentative future schedule (see Appendix II).

L. Crocker, NRC Staff, in discussing the scheduling of a review of Indian Point Nuclear Power Plant, Unit 3, for an increase to full rated power, said that the SER has been published and is being delivered to the Committee. However, at this time, the impact of the programing error discovered in the Westinghouse ECCS evaluation model, noted earlier in the meeting, is not known.

Mr. Bender requested that the Staff provide the Committee with a written statement regarding the details of the error, and how it was found.

The Committee again requested that the NRC Staff inform the Committee of any significant design changes regarding the currently-proposed Allens Creek Nuclear Plant and the design previously reviewed by the Committee.

VI. Executive Sessions (Open to Public)

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

A. Regulatory Activities Subcommittee Report

1. Revision of 10 CFR 50.44

Mr. Siess, Subcommittee Chairman, recalled that at the 215th ACRS meeting, the Committee declined to approve the proposed changes in wording of 10 CFR 50.44, relating to combustible gas control following a LOCA. He said that the NRC Staff has decided that their position to not permit repressurization, along with purging, is the conservative position, and it is their intention to propose the changes to the rule to the Commission without the ACRS' blessing. They do recognize, however, that there are questions about the desirability of repressurization, and therefore have requested the Office of Nuclear Regulatory Research to include a study of repressurization in connection with its proposed research project on advanced containment concepts, involving vented containments. It is hoped that risk information can be developed eventually to resolve the question. The subcommittee does not believe that this matter requires additional action at this time by the Committee.

2. Regulatory Guides

The Committee approved the following Regulatory Guides:

- Regulatory Guide 1.29 (Rev. 3), Seismic Design Classification, and
- Regulatory Guide 1.68 (Rev. 2), Initial Test Program for Water-Cooled Nuclear Power Plants.

During the discussion of Regulatory Guide 1.68 (Rev. 2) Mr. Bender recommended that the NRC Staff allot some technical assistance funds to establish how this Regulatory Guide is being applied throughout the nuclear industry with the anticipation that the Guide will ultimately be modified to define effective preoperational test practices for nuclear power plants. The current Guide does not, in his opinion, provide adequate information for the purposes of regulation.

3. Regulatory Activities Subcommittee Agenda for its May Meeting

Mr. Siess noted that the following items are scheduled to be considered at the May 3, 1978 meeting of the Regulatory Activities Subcommittee:

- Regulatory Guide on Lightning Protection,
- Regulatory Guide 1.9 (Rev. 1), Selection, Design, and Qualification of Diesel Generation Units Used as On-Site Electric Power Systems at Nuclear Power Plants,
- Regulatory Guide 1.63, Electrical Penetration Assemblies,
- Regulatory Guide 1.130, Service Limits and Loading Combinations for Class I Plate and Shell Type Component Supports and
- Proposed Amendment to 10 CFR 55 Appendix A, Codes and Standards.

B. ACRS Quarterly Report to Commissioners

The Chairman noted that Members have received a draft copy of proposed quarterly report to the Commissioners for the period, December, 1977 through March, 1978. He requested that Members provide the ACRS Office with their comments on this report within the next week.

C. Testimony to Senate Subcommittee on Nuclear Regulation

The Chairman and the Executive Director stated that they would prepare a final statement to be used as testimony before the Senate Committee on Environment and Public Works, Subcommittee on Nuclear Regulation, Senator Hart, Chairman. He noted that a draft copy of the testimony had been provided the Members, and requested that their comments be submitted to the ACRS Office as soon as possible. The Chairman noted that he would be accompanied to the hearing by the Executive Secretary, the Vice Chairman, and Messrs. Bender and Siess. He welcomed the presence of any other Members who could participate.

D. Activities of the Members

1. Mr. Shewmon

It was the consensus of the Committee that Mr. Shewmon should not act as a consultant-without-pay to the Westinghouse Material Research Laboratory.

2. Mr. Moeller

The Committee offered no objection to Mr. Moeller's preparing a paper for the British Nuclear Energy Society's November 1978 meeting. The paper will be on radiation protection in the fuel cycle, and will consist primarily of an abridgment of chapter 7 of the Committee's Annual Report to Congress (1977) (NUREG-0392) and new material developed for the 1978 report.

The Committee offered no objection to Mr. Moeller's lecturing at an MIT safety course.

E. Proposed Independent Nuclear Accident Review Board

The Chairman designated a working group to develop a position on a proposed independent nuclear accident review board. Messrs. Bender and Shewmon volunteered to serve on this working group.

F. Proposed Meeting with Groupe Permanent

The Committee agreed to defer until the 217th ACRS meeting the setting of dates for a meeting with the French Groupe Permanent, to give more Members a chance to check their personal appointment calendars.

G. Reorganization of ACRS Subcommittees

The Committee agreed to defer until the 217th ACRS meeting a discussion of the reorganization of ACRS Subcommittees.

H. ACRS Reports and Letters

1. Letter to Dr. E. J. Sternglass

The Committee prepared a letter to Dr. E. J. Sternglass, in response to his request for ACRS review of changes in cancer mortality in the vicinity of several nuclear plants.

(For background material on the request, see Appendix XXXIII; for Committee reply, see Appendix XXXIV.)

VII. 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 nominate [redacted] for appointment to the Committee.

B. ACRS Reports and Letters

1. Arkansas Nuclear 1, Unit 2 Nuclear Power Plant

A report was prepared advising the Commissioners that the Committee believes that, subject to certain conditions, there is reasonable assurance that the Arkansas Nuclear One, Unit 2 Nuclear Power Plant can be operated at core power levels up to 2815 MWt without undue risk to the health and safety of the public (see Appendix XXXVI).

2. McGuire Nuclear Station, Units 1 and 2

A report was prepared advising the Commissioners that the Committee believes that, subject to certain conditions, there is reasonable assurance that the McGuire Nuclear Station, Units 1 and 2, can be operated at power levels up to 3411 MWt without undue risk to the health and safety of the public (see Appendix XXXVI).

3. Liquid Pathway Generic Study

Although the Committee completed its review of the NRC Staff's draft report, NUREG-0440, Liquid Pathway Generic Study, time did not permit completion of a report. The Committee agreed to table completion of this report until the 217th ACRS meeting.

The 216th ACRS meeting was adjourned at 8:00 p.m., Friday, April 7, 1978.

ACRS Meeting

Meeting Dates:
April 6-7, 1978

APPENDIX I

ATTENDEES

ADVISORY COMMITTEE ON REACTOR SAFEGUARDS

Stephen Lawroski, Chairman
Max W. Carbon, Vice-Chairman
Myer Bender
Harold Etherington
Herbert S. Isbin
William Kerr
J. Carson Mark
Dade W. Moeller
Milton S. Plesset
Paul G. Shewmon
Chester P. Siess

ACRS STAFF

Raymond F. Fraley, Executive Director
Marvin C. Gaske, Assistant Executive Director
Herman Alderman
Andrew L. Bates
Paul A. Boehnert
Sam Duraiswamy
Elpidio G. Igne
James M. Jacobs
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

Ivan Catton
Elbert P. Epler
Richard F. Foster
Walter C. Lipinski

NRC ATTENDEES

216TH ACRS MTG.

April 6, 1978

Div. of Systems Safety

J. P. Joyce
M. W. Hodges
R. L. Tedesco
G. Lainas
D. Pickett
B. Turovlin
A. J. Szukiewicz
P. W. Baranowsky
H. E. Polk
H. Conrad
L. Belfracchi
R.Schemel
L. Phillips
T. A. Ippolito
S. Israel

Div. of Project Management

R. E. Martin
D. B. Vassallo
F. R. Naventi
L. P. Crocker
J. F. Stolz
T. P. Speis
C. W. Moon
A. R. Markese
K. Kniel

Div. of Systems Evaluation

R. Codell
G. Chipman
H. Berkson
D. R. Muller
R. A. Vollmer
D. F. Bunch
L. G. Hulman

RESEARCH

R. DiSalvo
F. Manning
M. A. Taylor
J. A. Murphy

NRC Consultants

J. B. Bullock

ACRS Consultants Attending

W. C. Lipinski
E. P. Epler
R. Foster
I. Catton

ATTENDEES - APPLICANT

216TH ACRS MEETING

April 6, 1978

COMBUSTION ENGINEERING
R. R. Mills
W. J. Gill
T. A. Jones
A. B. Spinell, Jr.
H. E. Neuschaefer
E. H. Kennedy
'. R. Hamphries
E. M. Brown
T. M. Starr
J. F. Church
F. C. Sernatinger
T. G. Shultz

ARKANSAS POWER & LIGHT
J. R. Perdue
G. H. Miller
M. Cajanaugh
D. R. Sikes
G. G. Young
D. G. Wardis
Neal A. Moore
J. R. Marshall
D. Rueter
D. H. Williams
B. A. Terwilliger

BECHTEL
E. H. Smith
J. C. Bradford
M. S. Iyer

OFFSHORE POWER SYSTEMS
J.A. Nutant
H. J. Stumpf
K. C. Perry
V. W. Campbell
P. B. Haga
B. Z. Cowan
C. A. Pelletier
D. C. Aabye
M. A. Capo
A. S.Caerdrin
D. Hewaeesee
G. R. Collin
D. Walker
J. E. Tabugen, Consultant to OPS
T. Pudlin, consultant to OPS
E. M. Buchak, consultant to OPS

SAI
R. L. Ritzman

PUBLIC ATTENDEES

216TH ACRS MEETING

Thursday, April 6, 1978

P. E. Grossman, Jr., Ebasco Services, In., NY
R. W. Prados, Louisiana Power & Light Co. 2425 Ramsey Drive,
New Orleans, LA
R. Borsum, B&W, Derwood, MD

P. S.Damerell - MPR Associates
W. W. Little, Westinghouse Hanford Co.
S. J. Weems, MPR Associates

NRC ATTENDEES

216TH ACRS MTG.

April 7, 1978

Div. of Project Management
C. Garland
L. Engle
L. P. Crocker
C. W. Moon
D. B.Vassallo
A. W. Drumerick
J. F. Stolz
K. Kniel
C. Van Niel

Inspection & Enforcement
K. Seyfrit
H. A. Wilber
D. Suy

RSLB
S. Berggren

MIPC
R. Denning

Div. of Systems Eval.
F. Hebdon
G. Chysman
G. B. Staley
W. P. Gammill
J. Grieves

RSR
J. Kelbher

DA
R. K. Grahm

ORB 4
R. M. Reid

Div. of Operating Reactors

R. E. Johnson
G. F. Lauih
E. L. Conner
G. S. Vissing
H. A. Levin
F. D. Coffman
E. Moler

Div. of Systems Safety
J. Stepp
C. F. Miller
H. C. LI
M. Hartzman
G. N. Lauben
W. Milsted
S. Israel
R. G. Fitzpatrick
N. H. Wagner
R. J. Bosnak

C. H. Hofmayer
D. F. Ross
J. P. Knight
M. D. Houston
M. R. Hum
F. C. Cherny
G. Mazetes

Stds. Development
A. Huite

Nuclear Reactor Regulation
B. Grimer
R. B. McMullen

EEB
S. Block

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Applicant Attendees

216TH ACRS Meeting

April 7, 1978

Duke Power

D. B. Blackmon
G. A. Copp
R. F. Wardell
L. Lewis
T. P. Harrall
R. A. Pace
T. C. McMeekin
K. S. Seidle
K. S. Canady
W. J. McCabe

W. F. Beaver
R. B. Priory
D. L. Camup
W. Parker, Jr.
D. G. Owen
C. J. Wylie
R. A. Pearce
L. Dail
P. M. Abraham
G. Cage
D. C. Holt
B. Rice
J. Foley,J r.

Westinghouse

N. J. Kiparalo
E. V. Somers
C. R. Srerrett
W. J. Johnson
J. O. Cermak
R. S. Howard
R. S. Borgraber
F. Landerman
A. Ball, Jr.
R. E. Tome

Toledo Edison
R. E. Sund
W. Lower

Law Engr.
C. E. Sams

PUBLIC ATTENDEES

216TH ACRS MEETING

Friday, April 7, 1978 - A.M.

June Allen - NAEC - Charlottesville, VA
R. Borsum - B&W - Derwood, MD
R. S. Bhatnagar - Duke Power Co. - Charlotte, NC
Paul Grossman - Ebasco Services, Inc. - New York, NY
Richard M. Kacich - Northeast Utilities - Hartford, CT
James B. McIlvaine - Bechtel Power Corp. - Frederick, MD
R. R. Mills - Combustion Engineering - Windsor, CT
R. M. Neil - VEPCO - Richmond, VA
R. C. L. Olson - Baltimore Gas & Electric Co. - Lutherville, MD
Thomas R. Robbins - Pickard Lowe & Garrick (Toledo Edison Co.) - Crofton, MD
Scott Sunde - Greenville News - Greenville, SC
R. E. Schaffstall - GE - Reston, VA
David Sokolsky - Self - San Francisco, CA

P. B. Haga, OPS, Jacksonville, FL
B.A. McIntyre, Westinghouse Electric Corp., Irwin, PA
S. E. Jacobs, Westinghouse, Pittsburgh, PA
M. Young, Westinghouse, Pittsburgh, PA

APPENDIX II

ACRS FUTURE AGENDA 4/3/78

ACRS MEETING PROJECT	TYPE OF REVIEW	REACTOR VENDOR	SER ISSUE DATE
MAY			
MAINE YANKEE	POWER INCREASE	CE	4/3/78
INDIAN POINT 3	FULL POWER	W	4/3/78
JUNE			
NEW ENGLAND 1&2	CP	W	5/1/78
DIABLO CANYON 1&2	OL	W	5/1/78
DAVIS BESSE 2&3	CP	B&W	5/1/78
JULY			
RESAR-414	STD NSSS	W	6/1/78
ALLENS CREEK 1	CP	GE	6/1/78
S8G	SP	-	6/1/78
AUGUST			
ERIE 1&2	CP	B&W	7/3/78
FFTF	SP	-	7/3/78
SUNDESERT 1&2	CP	W	7/3/78
NORTH COAST	ESR	-	7/3/78

ACRS FUTURE AGENDA 4/3/78

ACRS MEETING PROJECT	TYPE OF REVIEW	REACTOR VENDOR	SER ISSUE DATE
SEPTEMBER			
SEQUOYAH 1&2	OL	W	8/1/78

17-9

APPENDIX III
Letter, Rep. T. Bevill to Chmn. Hendrie – on ACRS Fellowship Program

Congress of the United States
House of Representatives
Committee on Appropriations
Washington, D.C. 20515

March 16, 1978

Honorable Joseph M. Hendrie
Chairman
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Chairman:

The Committee has received and considered your February 28, 1978, request to reprogram $300,000 in FY 1978 appropriations for a fellowship program which would assist the Advisory Committee on Reactor Safeguards.

During hearings on the FY 1979 budget request, the Commission identified several important areas where the regulatory program needs improvement, in order to reduce nuclear licensing times and to increase confidence in reactor safety. These include the improvement of internal processing schedules, additional on-site inspections, additional administrative support and others.

In view of these higher priority program needs, the Committee does not approve the reprogramming of funds to initiate a fellowship program.

Sincerely,

Tom Bevill

TOM BEVILL
Chairman
Public Works Subcommittee

17-10

PROJECT STATUS REPORT

ARKANSAS NUCLEAR ONE, UNIT 2

BACKGROUND:

The NRC Staff issued the original Safety Evaluation Report for Arkansas Nuclear One, Unit 2 (ANO-2) Operating License review on November 11, 1977 and a Supplement No. 1 in early March 1978. The ANO-2 project was reviewed at ACRS Subcommittee meetings in Russellville, Arkansas on June 24, 1977 and in Washington, D.C. on February 2, 1978 and March 20, 1978. The Combustion Engineering Core Protection Calculator System (CPCS), which is being reviewed as part of the ANO-2 docket, since ANO-2 will be the first operating plant to use this new system, was reviewed by the CESSAR System 80 Subcommittee on February 28, 1975 in Windsor, Connecticut and by the Electrical Systems, Control and Instrumentation Subcommittee in Windsor Connecticut on May 20, 1977 and in Washington, D.C. on June 30, 1977 and March 20, 1978 (Highlights attached). The CPCS was not reviewed by the NRC or ACRS at the ANO-2 construction permit stage since this is a newly developed system subsequent to the construction permit review.

The ACRS, during its 214th meeting, February 9-11, 1978 partially reviewed the application of Arkansas Power and Light Company for a license to operate ANO-2. All areas pertinent to the oprating license review were covered except for the CPCS/COLSS. At the conclusion of that meeting, it was pointed out that it did not appear that any significant new items were opened up by the Committee beyond those identified by the NRC Staff. The ACRS decided not to write an interim letter on ANO-2 at that meeting due to the large number of outstanding issues and the incomplete review of the CPCS/COLSS.

PLANT DESCRIPTION:

Many features of the design of ANO-2 are similar to Calvert Cliffs 1 and 2 except that the ANO-2 plant will use fuel assemblies with a 16 x 16 fuel rod array, while the Calvert Cliffs 1 and 2 use 14 x 14 fuel rod assemblies. The initial power of ANO-2 core is 2815 MWt (approximately 912 net MWe), compared to the initial power level of 2560 MWt (approximately 810 net MWe) for Calvert Cliffs 1 and 2. A reactor design comparison between ANO-2 and Calvert Cliffs 1 and 2 is included as Attachment 2.

The ANO-2 reactor will be the first to use Combustion Engineering 16 x 16 fuel rod assembly design. This fuel will be longer than the previous Combustion Engineering 14 x 14 design. Because of this and the fact that there will be more fuel rods per fuel assembly, the fuel rods will operate at lower linear heat generation rates. The cladding also has a larger thickness-to-diameter ratio than the 14 x 14 design.

The ANO-2 will be the first in the United States to use a digital computerized Core Protection Calculator System (CPCS) as part of the reactor protection system. The remainder of the reactor protection system is conventional analog hard-wired equipment. The CPCS, in conjunction with the overall reactor protection system, is designed to provide at least the same level of protection to the core as a conventional, hard-wired system. The CPCS is designed to initiate a reactor trip for the following events:

(1) Uncontrolled control element assembly (CEA) withdrawal from a critical condition.

(2) CEA misoperation.

(3) Uncontrolled boron dilution.

(4) Total and partial loss of reactor coolant forced flow.

(5) Excess heat removal due to secondary system malfunction.

(6) Steam generator tube rupture with and without a concurrent loss of offsite alternating current (AC) power.

Backup trips are available to limit the consequences of each of the above events, even with failure of the CPCS, except for the CEA misoperation. It is not clear how limited the consequences will be in the event of CPCS system failure. The NRC Staff has not done an independent evaluation of these consequences.

The ANO-2 will also use a new reactor monitoring system, designated as the Core Operating Limit Supervisory System (COLSS), to continuously monitor important reactor characteristics and establish margins to operating limits. This system will use the output of the incore detector system to synthesize the core average axial power distribution. This is not considered to be a safety system and as a result has not been reviewed in detail by the NRC Staff.

OUTSTANDING ISSUES:

Attachment 3 provides a list of outstanding issues discussed at the last Subcommittee meeting held on March 20, 1978.

HIGHLIGHTS
COMBINED ELECTRICAL SYSTEMS, CONTROL
AND INSTRUMENTATION AND THE
ARKANSAS NUCLEAR ONE UNIT 2 NUCLEAR PLANT
SUBCOMMITTEE MEETING
WASHINGTON, D.C.
MARCH 20, 1978

1. Ten outstanding NRC Staff Positions on the Core Protection Calculator System were outstanding. These include the following:

Position 1	*Uncertainty values in CPC data base must be experimentally qualified - requires measurements at startup
Position 4	Separation Criteria between the optical isolator cards in the CEAC - Contingent on Position 26
Position 5	*Cable Separation - Susceptibility to noise - requires check during startup
Position 12	*Noise susceptability - Resolved pending confirmation measurements during startup
Position 14	Adequacy of seismic loads - report of tests is under NRC Staff review
Position 15	Range limits - Resolved with the exception of limit values on shape annealing matrix constants
Position 18	Software burn-in test - awaiting final NRC Staff review of test data
Position 19	Qualification of Software Change Procedure - Requires fully qualified software change procedure
Position 20	Data links to plant computer - Applicant has just recently agreed to disconnect data links - Resolved
Position 26	Qualification of optical isolator as an isolation device

*Positions 1, 5 and 12 require startup of the plant to obtain data; therefore, they cannot be resolved until after issuance of the operating license.

17-13 Attachment 1

2. The NRC Staffs Supplemental Safety Evaluation Report provided for this meeting reported 24 non CPCS related outstanding issues. Two of these items were resolved at the time of this meeting and four additional items were identified leaving a total of 26 outstanding issues. Both, the NRC Staff and Applicant agreed that all of these items could be resolved prior to the end of May 1978. Scheduled fuel load date is May 15, 1978.

3. Topics discussed during this meeting which appeared to trouble the Subcommittee included:

   a. Determination of periodic CPCS test interval determination without a reliability analysis of the CPCS.

   b. NRC Staff requirement to disconnect the CPCS plant computer data links

   c. Large number of Phase II test cases were outside the initial acceptance criteria

   d. Lack of qualified software change procedure

   e. Noise tests, similar to actual plant noises, should have been used to determine the affect on CPCS operation.

   f. Total number of outstanding issues may be excessive to get a favorable ACRS report in April 1978.

4. The Subcommittees recommended that the CPCS and ANO-2 be brought to the ACRS for consideration at the April 6-8 ACRS meeting. The Applicant was informed that due to the large number of outstanding issues (10 CPCS and 26 non CPCS) the ACRS might not be able to write a favorable report at the April ACRS meeting.

Documents Provided to the Subcommittee for this Meeting

1. Presentation Schedule (Attachment B)

2. Copies of viewgraphs (Attachments 1-33). A complete set of handouts is available in the ACRS official copy of these minutes.

3. Supplement No. 1 to the Safety Evaluation Report for the Arkansas Nuclear One Unit 2 plant, dated March 1973.

REACTOR DESIGN COMPARISON

THERMAL AND HYDRAULIC DESIGN PARAMETERS (NOMINAL)	ANO-2	Calvert Cliffs 1 & 2
Performance Characteristics		
Reactor Core Heat Output, thermal megawatts	2815	2560
Reactor Core Heat Output, millions of British thermal units per hour	9608	8737
System Pressure, pounds per square inch absolute	2250	2250
Minimum DNBR at Nominal Conditions (full power)	2.26	2.18
Coolant Flow		
Total Flow Rate, millions of pounds per hour	120.4	128.8
Average Velocity Along Fuel Rods, feet per second	16.4	14.2
Average Mass Velocity, millions of pounds per hour per square foot	2.60	2.33
Coolant Temperature, degrees Fahrenheit		
Nominal Inlet	553.5	543.4
Vessel Outlet	612.0	595.0
Average in Vessel	582.75	569.2
Nominal Outlet of Hot Channel (full power)	652	642.9
Heat Transfet at 100 percent Power		
Active Heat Transfer, Surface Area, square feet	51,000	48,400
Average Heat Flux, British thermal units per hour per square foot	182,200	178,000
Maximum Heat Flux, British thermal units per hour per square foot	425,800	527,900
Average linear heat rate of fuel rod only, kilowatts per foot	5.34	5.94
Maximum Clad Temperature, degrees Fahrenheit		
Clad Surface at Nominal Pressure	657	657
Fuel Temperature, degrees Fahrenheit		
Maximum at 100 percent Power	3420	4170
CORE MECHANICAL DESIGN PARAMETERS		
Fuel Rod Array	16x16	14x14
Number of Fuel Assemblies	177	217
Fuel Rods per Assembly	224-236	164-176
Fuel Assemblies Overall Dimensions, inches	7.980 x 7.980	7.980/7.980
Number of Spacer Grids per Assembly	12	8
Fuel Rods		
Number	40,716	36,896
Outside Diameter, inches	0.382	0.440
Clad Thickness, inches	0.025	0.026
Clad Material	Zircaloy 4	Zircaloy 4
Fuel Pellets		
Material	Sintered Pellets	Sintered Pellets
Length, inches	0.390	0.650

ATTACHMENT 2

17-16

I STATUS OF PROJECT REVIEW

SAFETY EVALUATION REPORT
ISSUED - NOVEMBER 11, 1977

FIRST ACRS MEETING - FEBRUARY 9, 1978

SUPPLEMENT NO. 1 TO SAFETY EVALUATION
REPORT ISSUED - MARCH 6, 1978

SCHEDULED PROSPECTIVE DECISION DATE
FOR ISSUANCE OF THE OPERATING LICENSE - JUNE 1, 1978

II ITEMS RESOLVED SINCE PREPARATION OF SSER NO. 1

FUEL ASSEMBLY BURNABLE POISON DESIGN VERIFICATION (4.0)

CEA SURVEILLANCE PLAN FOR $Al_2O_3 - B_4C$ (4.0)

III EACH NEW ITEM SINCE PREPARATION OF SSER NO. 1

CONTAINMENT PURGE VALVE CLOSURE (SECTION 6.0)

REGULATORY GUIDE 1.44 (SECTION 5.0)

ENVIRONMENTAL QUALIFICATIONS
FOR POLYETHEYLENE CABLES (SECTION 3.11)

ECCS PUMP ROOM LEAKAGE (SECTION 15.4.6)

Attachment 3

IV EACH OUTSTANDING ISSUE IDENTIFIED IN SSER NO. 1 (21 ITEMS)

SEISMIC QUALIFICATION (SECTION 3.10)

ENVIRONMENTAL QUALIFICATIONS (SECTION 3.11)

CEA GUIDE TUBE WEAR (SECTION 4.0)

CONTAINMENT PRESSURE DUE TO MAIN STEAM LINE BREAK MASS AND ENERGY RELEASES (SECTION 6.2)

CONTAINMENT LEAKAGE TESTING PROGRAM (SECTION 6.2.6)

ENVIRONMENTAL QUALIFICATIONS OF SAFETY RELATED INSTRUMENTATION FOR MAIN STEAM LINE BREAK INSIDE CONTAINMENT (SECTION 6.2.1)

EVALUATION OF EMERGENCY CORE COOLING SYSTEM (SECTION 6.3

CONTAINMENT SUMP TESTS (SECTION 6.3.4)

VERIFICATION OF IMPLEMENTATION OF INSTRUMENTATION & CONTROL SYSTEMS DESIGN (SECTION 7.1)

INPUT FAULT AND SURGE TESTING OF POWER SUPPLIES (SECTION 7.2.2)

EVALUATION OF ADEQUACY OF PARAMETERS ESSENTIAL FOR ACCIDENT AND POST-ACCIDENT MONITORING (SECTION 7.5.1)

REDUNDANT VALVE POSITION INDICATION (SECTION 7.6.3)

SEPARATION CRITERIA FOR CONDUITS (SECTION 7.9.4)
FIRE PROTECTION (SECTION 9.7)
FEEDWATER HAMMER IN STEAM GENERATORS (SECTION 10.6)
PREOPERATIONAL TESTS (SECTION 14.0)
EMERGENCY PLAN (SECTION 13.3)

RCP SEIZURE ANALYSIS USING CESEC CODE (SECTION 15.4.2)

REVIEW OF MAIN STEAM LINE BREAK ANALYSIS (SECTION 15.4.-

FINANCIAL QUALIFICATIONS (SECTION 20.0)

GENERIC ISSUES - SPECIFIC ANO-2 ACTIONS

(1) REACTOR VESSEL SUPPORTS (SECTION 3.9.3)

(2) OVERPRESSURE PROTECTION, LONG TERM (SECTION 5.7)

(4) OFFSITE GRID STABILITY (SECTION 8.2)

V CONCLUSIONS

FOR OPERATING LICENSE ISSUANCE

(A) TOTAL NUMBER OF ITEMS - NON-CPCS (28)

(B) TWO ITEMS RESOLVED AND ARE CURRENTLY AWAITING PUBLICATION IN AN SSER OR APPLICANT DOCUMENTATION

(C) MAJOR ISSUES (8) PROJECTED TO BE UNDER REVIEW THROUGH LATE MAY, 1978

CEA GUIDE TUBE WEAR

CONTAINMENT LEAKAGE TESTING PROGRAM

ENVIRONMENTAL QUALIFICATIONS OF SAFETY RELATED INSTRUMENTATION FOR MSLB INSIDE CONTAINMENT

CONTAINMENT SUMP TESTS

INPUT FAULT AND SURGE TESTING OF POWER SUPPLIES

FIRE PROTECTION

RCP SEIZURE ANALYSIS USING CESEC CODE

OFFSITE GRID STABILITY

(D) CORE PROTECTION CALCULATOR SYSTEM REVIEW STATUS IN THE MARCH 6, 1978 SSER

22 POSITIONS

12 POSITIONS RESOLVED

3 POSITIONS RESOLVED FOR FUEL LOAD (REQUIRE STARTUP DATA)

7 REMAIN OUTSTANDING (THESE INCLUDE SEISMIC QUALIFICATIONS AND POSITION 20)

THE CORE PROTECTION CALCULATOI

1. What is it? A system for on-line calculation of core power distribution and DNBR, and for providing a reactor trip signal when either linear power density or DNBR reach selected levels. The system also calculates CEA position, and primary coolant flow rate, and provides a trip signal when flow declines to a selected value.

2. How does the protection system in a "CPC reactor" differ from that in the reactor without CPC? The CPC system produces 2 trips out of a total of 14 trip functions which can be distinguished in the protection system. The other 12 trip functions are unchanged. The non CPC reactor uses a measurement of ∇P in the steam generator to indicate primary coolant flow rate. The DNBR trip replaces (in some sense) the thermal margin trip in the non CPC systems. It should be noted that each of the trips provided by the CPC has an identified backup so that if the expected trip does not provide the necessary shutdown, a backup trip is available.

3. How does it work? The CPC system makes use of six minicomputers, one in each of four separate channels, to calculate DNBR; and two to calculate CEA position. Using as input CEA position, the readings from 12 (4 sets of three) ex-core neutron detectors, primary flow rate as calculated from primary pump speed, pressurizer pressure core ∇T as determined by the difference between measured values of hot leg and cold leg temperatures. The CPC makes virtually real time calculations of core power distribution and of DNBR.

4. What are the problems? Since this is the first U.S. reactor in which on-line digital computers are to be used as part of the

reactor protection system the NRC staff has been concerned about (a) hardware reliability (b) system independence (both independence of individual channels one-from-the-other, and independence of the protection systems and components, from non-protection systems and components, (c) and about software validity and reliability. Acceptable methods for testing and maintaining both hardware and software have had to be developed by the vendor, and checked and accepted by the NRC staff.

5. <u>Is the new system safer than the old?</u> It may be a standoff. With the new system one should know more about what the core power distribution is than one knew with the old. However, the new system is more complicated than the old one. In my view, the additional information available makes the change worthwhile.

APPENDIX VI
ANC-2: ACRS Consultants' Report

US Department of Energy
ARGONNE NATIONAL LABORATORY
9700 South Cass Avenue, Argonne, Illinois 60439
Telephone 312/972-4639 (4636)
FTS 972-4639 (4636)

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

MAR 31 1978
AM PM
7,8,9,10,11,12,1,2,3,4,5,6

March 29, 1978

Advisory Committee on Reactor Safeguards
U. S. Nuclear Regulatory Commission
Washington, DC 20555

Attention: Mr. G. R. Quittschreiber

Subject: Core Protection Calculator System (CPCS) for Arkansas Nuclear One - Unit 2

Reference: Letter, W. C. Lipinski to G. R. Quittschreiber, subject: *Review of CESSAR System 80*, dated May 5, 1975.

The above referenced letter discussed CPCS issues which were of concern at the time of writing and was based on the review of non-proprietary information. Subsequently proprietary information was reviewed and several areas of concern were resolved. In addition, the NRC staff and its consultants have conducted an in-depth review of the CPCS. The documentation resulting from the NRC review has been transmitted to me and reviewed by me.

It is to be noted that the reactor trip system is based on thirteen (13) trip functions. Each trip function is comprised of four redundant and independent protection channels. Only two (2) of the thirteen (13) trip functions are derived from the CPCS. These are: (1) High local power density and (2) Low departure from nucleate boiling ratio (DNBR). The remaining eleven (11) trip functions are derived from hard wired analog systems.

If the CPCS were to fail, backup trips will function. In order to assess the degree of protection provided by the backup trips, the NRC staff conducted a review beyond that normally performed for reactor protection systems. The results of this review are documented on pages D-1 through D-3 in Supplement No. 1 to NUREG-0308. The NRC staff concluded that the backup to the CPCS is acceptable. I concur with this finding.

During the course of the CPCS review, the NRC staff developed twenty-seven (27) positions. Of these, seventeen (17) positions are resolved and closed, and ten (10) positions are still outstanding. This is not unexpected because for several issues operating data is required.

The outstanding positions are:

1. Uncertainty Associated with the Algorithms. Resolution: Experimentally qualify adequacy of uncertainties by

performing confirmatory measurements during startup to demonstrate the adequacy of the axial power systhesis.

4. CEAC Separation Criteria at the Output of the Optical Isolator Cards. Resolution: Contingent on position No. 26.

5. Cable Separation. Resolution: Applicant will reevaluate design where safety-related control rod drive position sensor cables are run together with nonsafety cable and will advise NRC staff as to its resolution. Concern is that nonsafety cables will induce noise in safety cables.

12. Electrical Noise and Isolation Qualification. Resolution: Noise and EMI readings to be made in plant to verify that the noise spectrum is within the susceptibility envelope used during system test.

14. Seismic Qualification. Resolution: NRC provided applicant with current criteria for multi-frequency input and sine beat tests. Submittal date for a satisfactory seismic qualification plan and a review completion date to be determined.

15. Addressable Constants. Resolution: Software has been redesigned to reject entry of unreasonable constants by operator and was tested by NRC staff during Phase II test audit. Resolved for all addressable constants with the exception of limit values on the nine (9) shape annealing matrix constants.

18. Burn-In Test. Resolution: Software burn-in test on fully configured system completed at ANO-2 during February 1978. A preliminary review indicates no major problems. Final review required for resolution

19. Qualification of Software Change Procedures. Resolution: Qualify software change by either:

    A. Final test on plant system

        (1) Define a test configuration acceptable to NRC staff.

        (2) Define an acceptable test program for each change or for a pre-defined category of software changes.

    or

    B. Final test on a single channel system

        (1) Qualify the single channel system

        (2) Define an acceptable test program for each change or for a pre-defined category of software changes.

Changes to ANO-2 software will be prohibited until a change procedure has been fully qualified in accordance with position No. 19.

20. Data Link to Plant Computer. Resolution: NRC staff will only allow links to plant computer to be connected during initial startup and refueling startups. The applicant must submit procedures and test criteria and methods for NRC review. If this is not acceptable, the NRC will require the applicant to remove the data links.

26. Optical Isolator. Resolution: Optical isolator must be qualified as an isolation device by applying 125 volts alternating current or 125 volts direct current at the input and output of the device. These optical isolators are installed between the two (2) CEAC computers and the four (4) CPC computers.

Of the above outstanding positions, three (Nos. 1, 5, and 12) require rtup data/analysis and seven (Nos. 4, 14, 15, 18, 19, 20, and 26) require solution prior to plant startup. In addition, the NRC staff requires that No. 26 be resolved prior to the issuance of an operating license.

I disagree with the NRC staff resolution of position on No. 20 Data Link to Plant Computer. The NRC staff bases its position on GDC 24 and IEEE 279-1971, Section 4.7.

GDC 24 -- "Separation of protection and control systems. The protection system shall be separated from control systems to the extent that failure of any single control system component or channel, or failure or removal from service of any single protection system component or channel which is common to the control and protection systems leaves intact a system satisfying all reliability, redundancy, and independence requirements of the protection system. Interconnection of the protection and control systems shall be limited so as to assure that safety is not significantly impaired."

IEEE 279-1971 -- "Section 4.7 Control and Protection System Interaction.

4.7.1 Classification of Equipment. Any equipment that is used for both protective and control functions shall be classified as part of the protection system and shall meet all the requirements of this document.

4.7.2 Isolation Devices. The transmission of signals from protection system equipment for control system use shall be through isolation devices which shall be classified as part of the protection system and shall meet all the requirements of this document. No credible failure at the output of an isolation device shall prevent the associated protection system channel from meeting the minimum performance requirements specified in the design bases.

Examples of credible failures include short circuits, open circuits, grounds, and the application of the maximum credible ac or dc potential. A failure in an isolation device is evaluated in the same manner as a failure of other equipment in the protective system.

4.7.3 Single Random Failure. Where a single random failure can cause a control system action that results in a generating station condition requiring protective action and can also prevent proper action of a protection system channel designed to protect against the condition, the remaining redundant protection channels shall be capable of providing the protective action even when degraded by a second random failure.

Provisions shall be included so that this requirement can still be met if a channel is bypassed or removed from service for test or maintenance purposes. Acceptable provisions include reducing the required coincidence, defeating the control signals taken from the redundant channels, or initiating a protective action from the bypassed channel.

4.7.4 Multiple Failures Resulting from a Credible Single Event. Where a credible single event can cause a control system action that results in a condition requiring protective action and can concurrently prevent the protective action from those protection system channels designated to provide principal protection against the condition, one of the following must be met.

4.7.4.1 Alternate channels, not subject to failure resulting from the same single event, shall be provided to limit the consequences of this event to a value specified by the design bases. In the selection of alternate channels, consideration should be given to (1) channels that sense a set of variables different from the principal channels, (2) channels that use equipment different from that of the principal channels to sense the same variable, and (3) channels that sense a set of variables different from those of the principal protection channels using equipment different from that of the principal protection channels. Both the principal and alternate protection channels shall meet all the requirements of this document.

4.7.4.2 Equipment, not subject to failure caused by the same credible single event, shall be provided to detect the event and limit the consequences to a value specified by the design bases. Such equipment shall meet all the requirements of this document."

GDC 24 and IEEE 279 do not forbid the connection of protection and control equipment. (In this case, the term control is used in the broad sense where the operator is used to close the loop between the information he receives from the plant computer and the actions he may take in operating the plant.)

A description of the Core Operating Limit Supervisory System (COLSS), e Core Monitoring Computer, and the Plant Computer appears on pages 9, 10, and 11 in the above referenced letter.

During the NRC staff presentation on March 2?, 1978, the following were offered as reasons to support the NRC staff isolation position:

1. The plant computer is not safety grade, therefore it should not perform safety functions.

2. The CPCS has to have additional programs to generate the output data to be transmitted to the plant computer.

3. The plant computer will send interrupt signals to the CPCS to start data transmission from the CPCS to the plant computer.

I would like to comment on each of the above points:

1. If the CPCS were not digitally implemented, the plant computer would still be used to perform the same calculations. Plant computers are used in all other nuclear power plants to supply the operator with information on plant status. This information, coupled with operator judgement, is used to operate the plant. The plant computers in all cases supplement and do not replace plant protection equipment. If the NRC staff has developed a new position on the use of plant computers in general, this position should be better stated and added to the General Design Criteria if the position has merit.

2. It is true that the CPCS digital program has been expanded to provide for the data transmission to the plant computer. As to whether this added feature has compromised the CPCS can only be determined by examining overall CPCS reliability. More discussion of reliability is presented later under discussion of Position No. 8.

3. The feature by which the plant computer sends an interrupt signal to the CPCS to start data transmission does place the CPCS as a slave to the plant computer rather than the reverse. There is a solution to this problem in which the CPCS would send an interrupt signal to the plant computer to tell it that data is to be transmitted. The applicant should be given the opportunity to discuss whether slaving the plant computer to the CPCS is acceptable.

The NRC staff, in taking its final position in allowing the CPCS to be connected to the plant computer during initial startup and refueling startups, is inconsistent for the following reasons:

1. The CPCS software will not be changed. The same software that is used during startups will be used during power operation. Any concern the staff may have with respect to a reduction in system reliability because of software complexity remains unchanged.

2. The plant computer will continue to execute algorithms and display information to the operator. The operator will use this information to run the plant. The plant computer will not receive data from the CPCS and the algorithms based on this data will not be operational.

It is my recommendation that the applicant be allowed to connect the CPCS to the plant computer with properly qualified isolation devices and that the plant computer be slaved to the CPCS for data transmission.

The NRC staff in Position No. 8, Time Interval of Periodic Testing, has required that the test interval should be significantly more frequent than the proposed 30 days during the first six months of operation and that the applicant develop an acceptable analysis of the CPCS reliability in accordance with applicable IEEE standards. Based on the test data acquired during the first six months and the reliability analysis, the test interval can then be modified.

The NRC staff has not provided the applicant with guidance on a reliability goal. If the test interval is to be modified, it must be done on the basis of meeting a specified requirement. It is recommended that the NRC staff develop a reliability requirement for the CPCS and provide this information to e applicant.

The Phase II Test and Test Report are covered under staff position No. 24. Of the 36 static test cases, 16 were outside of expected DNBR range and 6 were outside of expected LPD range, but based on additional analysis and testing were found to be acceptable by the NRC staff. Ten of the 26 dynamic test cases did not meet acceptance criteria for "time to trip" but explanations are acceptable to NRC staff.

Satisfactory final performance of the CPCS is determined by testing. It is imperative that the system pass all static and dynamic tests without explanations as to why a particular test was not passed. A proper simulation of the reactor should be used such that the test results are not dependent on a poor simulation, and explanations therefore have to be used to qualify a test as acceptable. Furthermore, the dynamic tests have been based on variation of a single input parameter with time and with all other inputs held constant. From previous static and dynamic test cases, coding errors of a fixed point multiplication overflow and a floating point multiplication underflow were detected. It must be clearly demonstrated that similar coding errors still do not exist for the case of variation of multiple input parameters with time. Alternately, it must be shown that if the system is still not properly scaled, that trips will occur sooner because of improper coding or scaling.

Sincerely,

Walter C. Lipinski

Walter C. Lipinski
Senior Electrical Engineer

WCL/at

cc: Dr. William Kerr

17-28

ELBERT P. EPLER
NUCLEAR SYSTEMS CONSULTANT
712 FLORIDA AVENUE
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483-0994

3-30-1978

Wm Kerr,

Chairman, Electrical Systems and I&C Subcommittee

At the March 20 subcommittee meeting, held to review the CPCS for ANO Unit 2, a question arose concerning criteria for the frequency of periodic testing of the Core Protection Calculator System. Inasmuch as this equipment is first of a kind, it has been made a requirement that, during the first six months of operation, the periodic test be significantly more frequent than the customary 30 days. We do, however, have information relative to test frequency which does not appear in the literature. In view of the concerns expressed, it would be worthwhile to present the relevant data.

A similar situation existed at ORNL when a new system for protection of the HFIR was developed, making use of solid state components and calculational techniques, for which no reliability data existed. A test frequency of once per eight hour shift was initially established, based on a very pessimistic estimate of component failure rates. This test frequency posed no problem inasmuch as an on-line system enabled each channel to be tested, from sensor to actuator, by pushing a single button at the operator's console. The test, by causing a local perturbation of the process variable, verifies operability of the channel by interrupting the current in one of the three magnet coils for each rod. The entire system, comprising 21 channels, can be tested in about ten minutes. The once per shift frequency was later changed to once per day. During the 12 years of operation, this has resulted in 100,000 channel tests being performed.

Three failures have occurred in 12 years as follows.

a. As a result of a design error the neutron flux amplifiers were found to be incapable of protecting against a transient having a very short period. When similar equipment was applied to a fast burst reactor, it was discovered that the power burst overloaded and destroyed the field effect transistors at the input to the amplifiers. For approximately two years the reactor had been unprotected against a design basis event. This condition was not discovered by periodic tests.

b. The three ionization chambers of the Faulty Fuel Element System were wetted during the testing of a relief valve. The equipment was dried and returned to service. When the reactor was brought to power and the periodic test performed, it was discovered that one of the three chambers remained inoperable.

c An engineer, in trouble shooting, inadvertently left a clip lead in place. The inoperable channel was discovered by periodic test.

This absense of component failure came as a surprise, as it would be expected that during these 12 years, several dozen component failures would have been discovered. It is a tribute to the excellent quality of surveillance and preventive maintenance that, although components have been replaced as a result of degraded performance, there have been but these two instances of failure to respond.

The consequence of failure of a protective feature depends upon the frequency at which the system is challenged, and on the number of diverse features within the system, which are available to meet each challenge.

During the 12 years of operation there have been but three occasions on which a scram become necessary. These were each brought about by an increase in temperature of the coolant at the reactor inlet. In each case a momentary dip in voltage caused one of the three main circulating pumps to trip. with two pumps remaining the flow was reduced to 80%. The control system dutifully reduced the reactor power to 80% to maintain the proper flux-to-flow ratio, however when one pump tripped its heat exchanger became unavailable. As a result, with 80% flow and 67% heat exchanger capacity, the coolant temperat. rose to the scram point.

Although there have been three challenges to the Inlet Temperature protective feature, which provides the sole protection, i.e., there is no diverse back-up feature, there have been no component failures. The frequency of periodic test has therefore not been a critical matter.

Had the challenges been, not an increase in temperature, but a slow increase in reactor power, the event would have been even less serious as it would have been seen by three diverse features, the Neutron Flux, the Thermal Power (Flow x ΔT) and the Inlet Temperature. There has, however, been no increase in reactor power to challenge the system, nor have there been any component failures.

A failure of the Reactor Low Pressure feature would be much more serious. Failure to scram on loss of pressure due to a leak, would result in fission products being discharged into the reactor building. There is no diverse protection for this event, however there has been neither a challenge nor a component failure.

In establishing a test frequency for this system it can be seen that neither component failure nor frequency of challenge has been a significant factor. The initial test frequency of once per shift was later changed to once per day. The system continues to be tested frequently because the ability to see the system respond to a change in the process variable is a valuable aid to surveillance. The required test frequency, however, is once per core life, i.e, 23 days. If for any reason it should become impossible or undesirable to test, the reactor would not have to be shut down unnecessarily as a result of an unrealistic requirement for periodic testing.

The HFIR has demonstrated that component failure has not been a problem, and that surveillance, not periodic test, has been the important contributor to reliability. The frequency of test can therefore be based entirely on convenience. This can also be demonstrated to be true for commercial reactors.

The BWR expe iences, seven times per year, a severe transient on occurrance of a turbine trip. The reactor power can increase as much as a factor of ten in less than a second, and failure to scram would surely result in melting. These seven challenges can be met by either the Neutron Flux or the diverse Pressure protective features. This is in sharp contrast to the HFIR which has experienced three mild temperature excursions in 12 years. Even so, it can be shown that the convenient test interval of 30 days is adequate.

The two-of-four logic employed in PWRs, and of current interest to this discussion, provides a depth of defense against random component failure. Three failures would be required to prevent protective action. This can be illustrated as follows.

Let it be assumed that each channel fails once per year. In a two-of-four system and a 30 day test interval, this would yield an unavailability of 1.16×10^{-3} for a given protective feature. The turbine trip transient is, however, sensed by both Neutron Flux and Pressure giving a combined unreliability of 1.3×10^{-6}. The 7 transients per year would yield a failure rate of 10^{-5} which would be acceptable. The assumed failure rate of once per year for each channel would, however, be entirely unacceptable, even for household kitchen appliances. Even so, with a two-of-four system and a convenient test interval of 30 days, this highly unreliable hardware would be adequate to meet even seven challenges per year.

The above represents experience with conventional hardware, which does not necessarily apply to the CPCS.

we can see several differences which might affect the ailure rate of the new system.

It has been demonstrated that with high quality surveillance, analog hardware failures can be detected, or even anticipated. It is to be expected that this will hold true for the CPCS digital equipment. It is also characteristic of digital equipment that self checking features will detect the vast majority of failures. These two factors taken together should leave relatively few component failures to be detected by the periodic test.

What is even more important, is the degree of diversity which places a limit on the consequence of failure of a given protective feature. The following is quoted from the SER, D-2 " Limited fuel damage is not considered in itself a significant safety concern. ...... The staff is considering failure of the digital trip system to perform its design function. while our review is incomplete and awaiting submittal of some requested analyses from the applicar t appears that at this stage of our review that <u>backup analog trips</u> and/or inherent shutdown mechanisms would limit the consequences of this type of a failure to prevent undue risk to the public alth and safety".

It would appear that with the retention of hard ~~ed~~ analog equipment that even more than the usual diversity exists with the result that component failure in this system, as in the past, would be of minor concern.

The staff position requiring an increase in test frequency for the initial six months, appears to be sound. Even an increase of a factor of two, in a two-of-four system would reduce failure probability by a factor of eight.

These factors, especially the matter of diverse ~~de~~tection, are peculiar to the ANO system. We can expect in the future to see digital systems standing alone, i.e, without being backed up by hard wired systems. In that case the consequence of component failure would be very much greater. We can, however, expect to learn much from this initial ANO installation so as to be better prepared, in the future, to evaluate the role of periodic test.

We should be mindful that in the past, component failure has never been a significant contributor to protection system failure, and that periodic tests performed for the purpose of detecting component for use may therefore be performed at convenient intervals. We would be both surprised and disappointed if the digital equipment were to prove to be so prone to component failure that periodic testing would need to be performed more frequently

E. P. Epler

CORE PROTECTION CALCULATOR SYSTEM

1. DESIGN AND OVERVIEW

2. FUNCTIONAL DESIGN

3. POWER DISTRIBUTION/DNB METHODOLOGY/UNCERTAINTIES

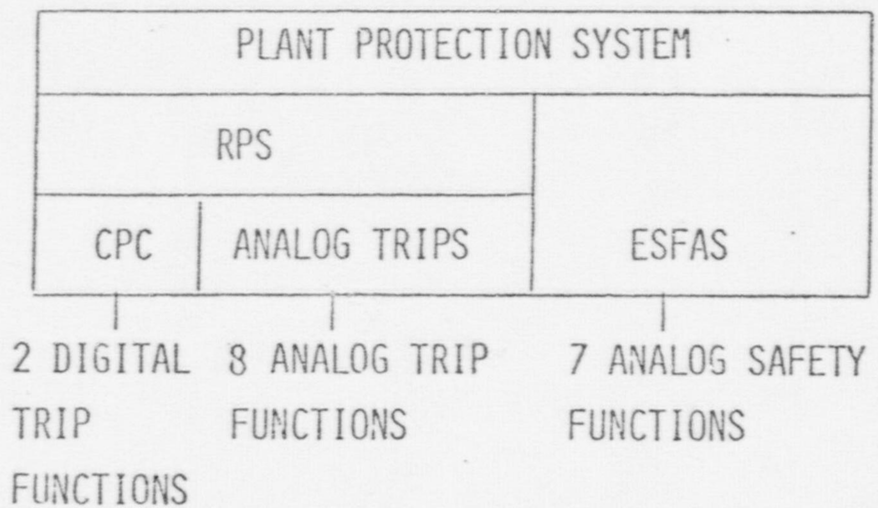
4. HARDWARE/SOFTWARE DESIGN

THE ANO-2 PLANT PROTECTION SYSTEM (PPS) IS COMPOSED OF TWO SUB-SYSTEMS:

1. AN ENGINEERED SAFETY FEATURES ACTUATION SYSTEM (ESFAS), AND

2. A REACTOR PROTECTION SYSTEM (RPS)

THE CORE PROTECTION CALCULATOR INITIATES TWO OF THE TEN TRIPS IN THE REACTOR PROTECTION SYSTEM, THE LOW DNBR TRIP AND THE HIGH LOCAL POWER DENSITY TRIP.



<u>DESIGN CRITERIA AND OVERVIEW</u>

1. DESIGN CRITERIA

2. DESIGN APPROACH

3. CONVERSION TO DIGITAL SYSTEM

4. DESIGN FEATURES

CRITERION NUMBER 10

(10CFR:50, APPENDIX A)

"THE REACTOR CORE AND ASSOCIATED COOLANT, CONTROL, AND PROTECTION SYSTEMS SHALL BE DESIGNED WITH APPROPRIATE MARGIN TO ASSURE THAT SPECIFIED ACCEPTABLE FUEL DESIGN LIMITS ARE NOT EXCEEDED DURING ANY CONDITION OF NORMAL OPERATION, INCLUDING THE EFFECTS OF ANTICIPATED OPERATIONAL OCCURRENCES."

DEFINITION OF ANTICIPATED OPERATIONAL OCCURRENCES

"ANTICIPATED OPERATIONAL OCCURRENCES MEAN THOSE CONDITIONS OF NORMAL OPERATION WHICH ARE EXPECTED TO OCCUR ONE OR MORE TIMES DURING THE LIFE OF THE NUCLEAR POWER UNIT..."

QUOTE FROM 10CFR:50, APPENDIX A

12-44

SPECIFIED ACCEPTABLE FUEL DESIGN LIMITS

1. LHR

   CORRESPONDING TO CENTERLINE MELT

2. DNBR

   EQUAL TO 1.3 (W-3 CORRELATION)

17-45

CRITERION NUMBER 20
(10CFR50, APPENDIX A)

"THE PROTECTION SYSTEM SHALL BE DESIGNED 1) TO INITIATE AUTOMATICALLY THE OPERATION OF APPROPRIATE SYSTEMS INCLUDING THE REACTIVITY CONTROL SYSTEMS, TO ASSURE THAT SPECIFIED ACCEPTABLE FUEL DESIGN LIMITS ARE NOT EXCEEDED AS A RESULT OF ANTICIPATED OPERATIONAL OCCURRENCES AND 2) TO SENSE ACCIDENT CONDITIONS AND TO INITIATE THE OPERATION OF SYSTEMS AND COMPONENTS IMPORTANT TO SAFETY."

CRITERION NUMBER 25
(10CFR50, APPENDIX A)

"THE PROTECTION SYSTEM SHALL BE DESIGNED TO ASSURE THAT SPECIFIED ACCEPTABLE FUEL DESIGN LIMITS ARE NOT EXCEEDED FOR SINGLE MALFUNCTION OF THE REACTIVITY CONTROL SYSTEMS,. SUCH AS ACCIDENTIAL WITHDRAWAL (NOT EJECTION OR DROP-OUT) OF CONTROL RODS."

F7-46

OVERALL REQUIREMENT

THE NSSS DESIGN AND TECHNICAL SPECIFICATIONS WHICH GOVERN ITS OPERATION ARE SUCH THAT:

1. THE SPECIFIED ACCEPTABLE FUEL DESIGN LIMITS (E.G., DNBR = 1.3) AND OTHER SAFETY LIMITS ARE NOT VIOLATED AS A CONSEQUENCE OF ANY ANTICIPATED OPERATIONAL OCCURRENCE (E.G., A ROD DROP), AND
2. THE CONSEQUENCES OF ANY OTHER POSTULATED ACCIDENT (E.G., STEAM GENERATOR TUBE RUPTURE) WILL BE ACCEPTABLE,

<u>PROVIDED THAT</u>

1. ACTUAL PLANT CONDITIONS ARE WITHIN THE LIMITING CONDITIONS FOR OPERATION, AND
2. ACTUAL SAFETY SYSTEM SETPOINTS ARE EQUAL TO OR CONSERVATIVE RELATIVE TO LIMITING SAFETY SYSTEM SETTINGS, AND
3. EQUIPMENT OTHER THAN THAT CAUSING OR DEGRADED BY THE OCCURRENCE OR ACCIDENT OPERATES AS DESIGNED, INCLUDING ALLOWANCE FOR DESIGN MALFUNCTIONS SUCH AS A STUCK ROD OR OTHER SINGLE FAILURE.

THE MARGIN REQUIRED BY CRITERION 10 IS DESIGNED INTO THE NSSS; HOWEVER THE REACTOR OPERATOR MUST OPERATE THE PLANT SUCH THAT THIS MARGIN IS MAINTAINED.

ALLOWED OPERATION IS DEFINED BY TECHNICAL SPECIFICATION LIMITING CONDITIONS FOR OPERATIONS (LCO).

COLSS, A DIGITAL MONITORING SYSTEM, AIDES THE OPERATOR IN MAINTAINING SOME OF THESE LCO's.

LOFA (ANTICIPATED OPERATIONAL OCCURRENCE)

LSSS: LOW DNBR TRIP (CPC)

LCO : CORE OPERATING LIMIT ON THERMAL MARGIN *

WITHDRAWN ROD WORTH

SCRAM DELAY TIMES

ROD DROP TIME

* MAINTAINED WITH HELP OF COLSS

CORE PROTECTION CALCULATORS

(CPC)

THE CORE PROTECTION CALCULATORS ARE DESIGNED TO PROVIDE THE FOLLOWING PROTECTIVE FUNCTIONS:

A. INITIATE AUTOMATIC PROTECTIVE ACTION SUCH THAT THE SPECIFIED FUEL DESIGN LIMITS ON DNBR AND LOCAL POWER DENSITY ARE NOT EXCEEDED DURING SELECTED ANTICIPATED OPERATIONAL OCCURRENCES, AND

B. INITIATE AUTOMATIC PROTECTIVE ACTION DURING CERTAIN ACCIDENT CONDITIONS TO AID THE ENGINEERED SAFETY FEATURES SYSTEM IN LIMITING THE CONSEQUENCES OF SELECTED ACCIDENTS.

EVOLUTION OF LICENSING CRITERIA

NRC INTERPRETATION OF CRITERIA AND INDUSTRY KNOWLEDGE AND VIEWS IN REACTOR PROTECTION DEVELOPED AND CHANGED IN THE EARLY 1970's.

1. IN GENERAL SINGLE FAILURES OF AN ACTIVE COMPONENT SHOULD BE CONSIDERED AS A POSSIBLE INITIATING MECHANISM FOR AN AOO.

   ROD MISOPERATION EVENTS

   - SINGLE ROD WITHDRAWAL
   - OUT OF SEQUENCE INSERTION AND WITHDRAWAL

2. IN MOST CASES, OPERATOR ACTION SHOULD NOT BE RELIED UPON TO PREVENT THE SPECIFIED ACCEPTABLE FUEL DESIGN LIMITS FROM BEING EXCEEDED.
   - AXIAL FLUX PERTURBATIONS

BASED ON THESE CONSIDERATIONS AND THE POTENTIAL IMPACT OF THE RESTRICTIONS IN TERMS OF OPERATION, IT WAS CONCLUDED THAT THE PROTECTIVE SYSTEM MUST:

1. SENSE THE POWER DISTRIBUTION WITH INCREASED ACCURACY.

2. INCLUDE MEASURED CONTROL ROD POSITION AS INPUT.

3. PROVIDE INCREASED ACCURACY IN DNBR THERMAL MARGIN BY ON-LINE INTERPRETATION OF RELEVANT COOLANT SYSTEM PARAMETERS.

TO EFFECTIVELY IMPLEMENT THE ABOVE REQUIREMENTS, DIGITAL PROCESSING TECHNOLOGY WAS INCORPORATED INTO THE PPS.

CPC ADVANTAGES RELATIVE TO ANALOG COUNTERPART

1. IMPROVES PLANT SAFETY

   MORE DIRECT MEASURE OF FUEL DESIGN LIMITS

   REDUCES RELIANCE ON OPERATOR ACTION

17-5-3

2. IMPROVES PLANT PERFORMANCE

   PROVIDES MORE ACCURATE MEASURE OF FUEL DESIGN LIMITS

   PERMITS PLANT PARAMETERS TO BE TRADED OFF AGAINST ONE ANOTHER SUCH THAT MARGIN TO FUEL DESIGN LIMITS IS UNCHANGED

3. IMPROVES PLANT FLEXIBILITY

   SIMPLIFIES TASK OF ACCOMMODATING CHANGING CONDITIONS DURING PLANT LIFE

CPC DESIGN FEATURES

- AN ON-LINE PROTECTION SYSTEM USING 3 LEVELS OF EX-CORE DETECTOR INFORMATION
- A COMPLETE SYSTEM OF DEDICATED DIGITAL CALCULATORS TO PROVIDE FOUR CHANNEL REDUNDANCY
- USES AN AXIAL/RADIAL SYNTHESIS TO CONSTRUCT POWER DISTRIBUTIONS
- USES MEASURED CEA POSITION INPUT
- FLOW DETERMINATION BASED ON RCP SPEED MEASUREMENTS
- LINEAR HEAT RATE AND DNBR CALCULATED ON-LINE
- OPERATOR'S CONSOLE PROVIDES COMPREHENSIVE DATA DISPLAY

W GILL CE

OBJECTIVE

TO PROVIDE A FUNCTIONAL DESCRIPTION OF THE CPC/CEAC SYSTEM

OUTLINE

PART 2 FUNCTIONAL DESCRIPTION

1. RELATIONSHIP OF CPC TO REMAINDER OF RPS
2. CPC DESIGN BASES EVENTS
3. SYSTEM INPUTS AND OUTPUTS
4. FUNCTIONAL BLOCK DIAGRAM
5. ALGORITHMS

PART 3 METHODS AND UNCERTIANTIES

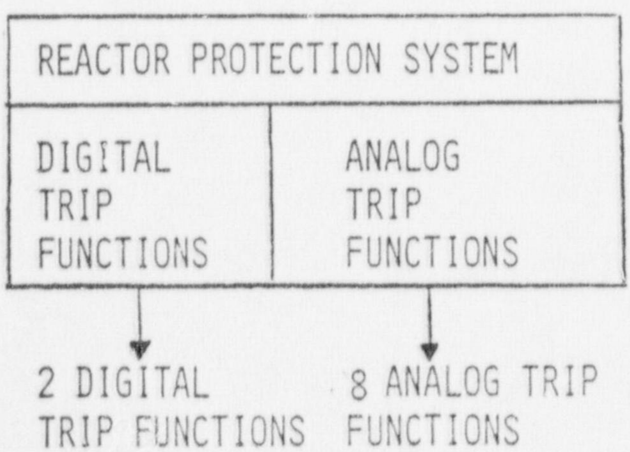
1. POWER DISTRIBUTION METHODS
2. DNB METHODS
3. TREATMENT OF UNCERTAINTIES

REACTOR PROTECTION SYSTEM

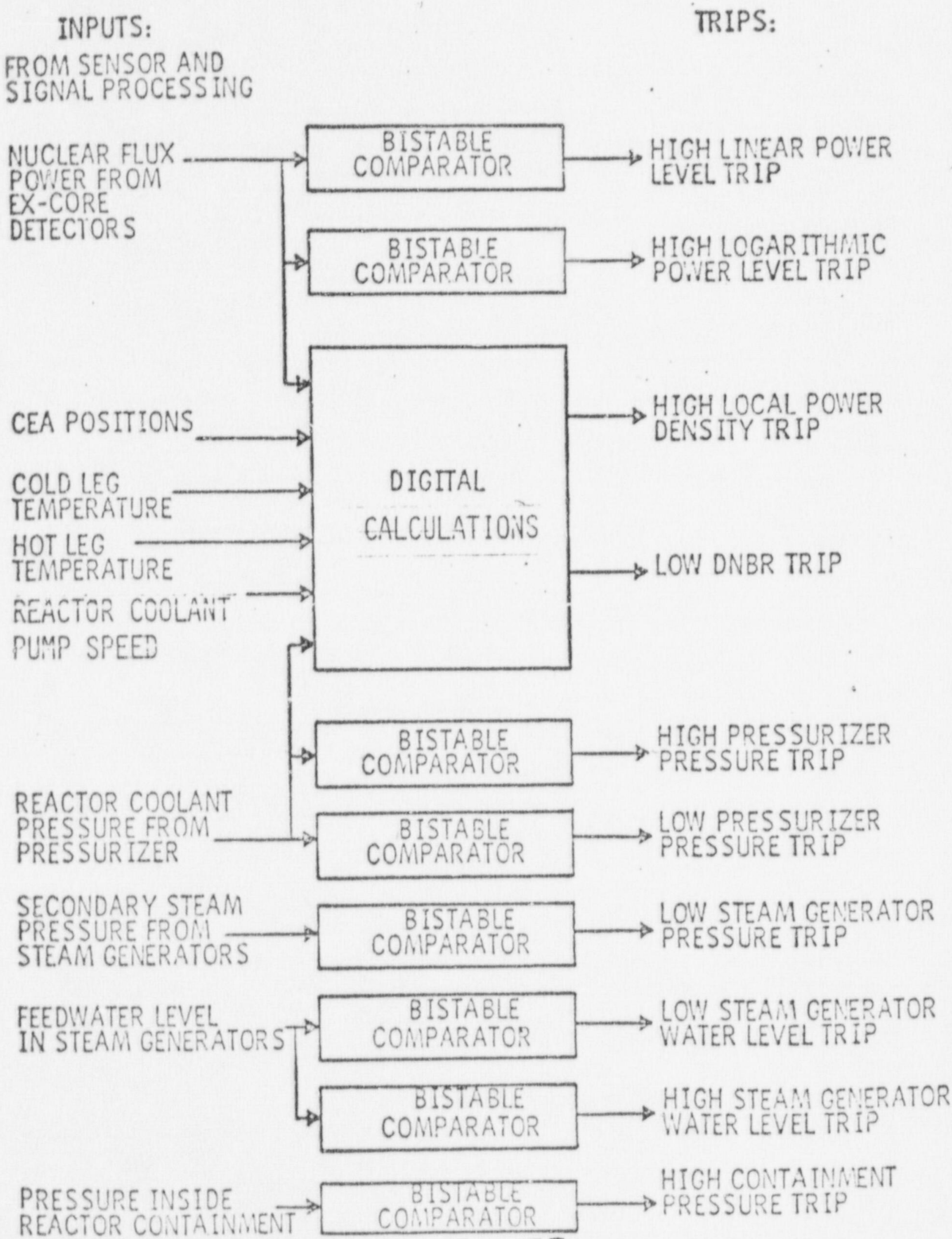
THE ANO-2 REACTOR PROTECTION SYSTEM IS COMPOSED OF TEN TRIP FUNCTIONS. THESE INCLUDE:

A. EIGHT ANALOG TRIP FUNCTIONS CONSITING OF SINGLE VARIABLES (E.G. PRESSURE) WHICH ARE COMPARED TO TRIP SETPOINTS, AND

B. TWO DIGITAL TRIP FUNCTIONS CONSISTING OF MULTI VARIABLES WHICH ARE PROCESSED BY DIGITAL COMPUTERS AND COMPARED TO TRIP SETPOINTS.



ANO-2 REACTOR PROTECTION SYSTEM TRIPS



<u>CPC</u>

THE CORE PROTECTION CALCULATORS ARE DESIGNED TO PROVIDE THE FOLLOWING PROTECTIVE FUNCTIONS:

A. INITIATE AUTOMATIC PROTECTIVE ACTION SUCH THAT THE SPECIFIED FUEL DESIGN LIMITS ON DNBR AND LOCAL POWER DENSITY ARE NOT EXCEEDED DURING ANTICIPATED OPERATIONAL OCCURRENCES, AND

B. INITIATE AUTOMATIC PROTECTIVE ACTION DURING CERTAIN ACCIDENT CONDITIONS TO AID THE ENGIEERED SAFETY FEATURES SYSTEM IN LIMITING THE CONSEQUENCES OF THE ACCIDENTS.

CPC DESIGN BASES EVENTS

MAJOR ANTICIPATED OPERATIONAL OCCURRENCES

1. UNCONTROLLED AXIAL XENON OSCILLATIONS
2. CEA RELATED EVENTS INCLUDING SINGLE ROD WITHDRAWAL, SINGLE DROPPED ROD, SUB-GROUP DEVIATION AND OUT-OF-SEQUENCE WITHDRAWAL AND INSERTION
3. EXCESS LOAD
4. LOSS OF LOAD
5. LOSS OF FORCED REACTOR COOLANT FLOW
6. UNCONTROLLED BORON DILUTION

POSTULATED ACCIDENTS

1. STEAM GENERATOR TUBE RUPTURE
2. REACTOR COOLANT PUMP SHAFT SEIZURE

BACK-UP TRIP FUNCTIONS FOR CPC ANTICIPATED OPERATIONAL OCCURRENCES

FAILURE OF CPC WITH CONCURRENT AOO IS NOT A DESIGN BASES EVENT FOR THE ANO-2 PLANT.

AN EVALUATION BASED ON THE CENPD-158 ATWS REPORT, WAS PERFORMED TO DETERMINE BACK-UP TRIP FUNCTIONS.

RESULT:

EVENT	BACK-UP TRIP
UNCONTROLLED CEA WITHDRAWAL FROM A CRITICAL CONDITION	HIGH PRESSURIZER PRESSURE
UNCONTROLLED BORON DILUTION	HIGH PRESSURIZER PRESSURE
TOTAL AND PARTIAL LOSS OF REACTOR COOLANT FORCED FLOW	HIGH PRESSURIZER PRESSURE
EXCESS HEAT REMOVAL DUE TO SECONDARY SYSTEM MALFUNCTION	LOW STEAM GENERATOR WATER LEVEL
STEAM GENERATOR TUBE RUPTURE	LOW PRESSURIZER PRESSURE
CEA MISOPERATION	MANUAL TRIP

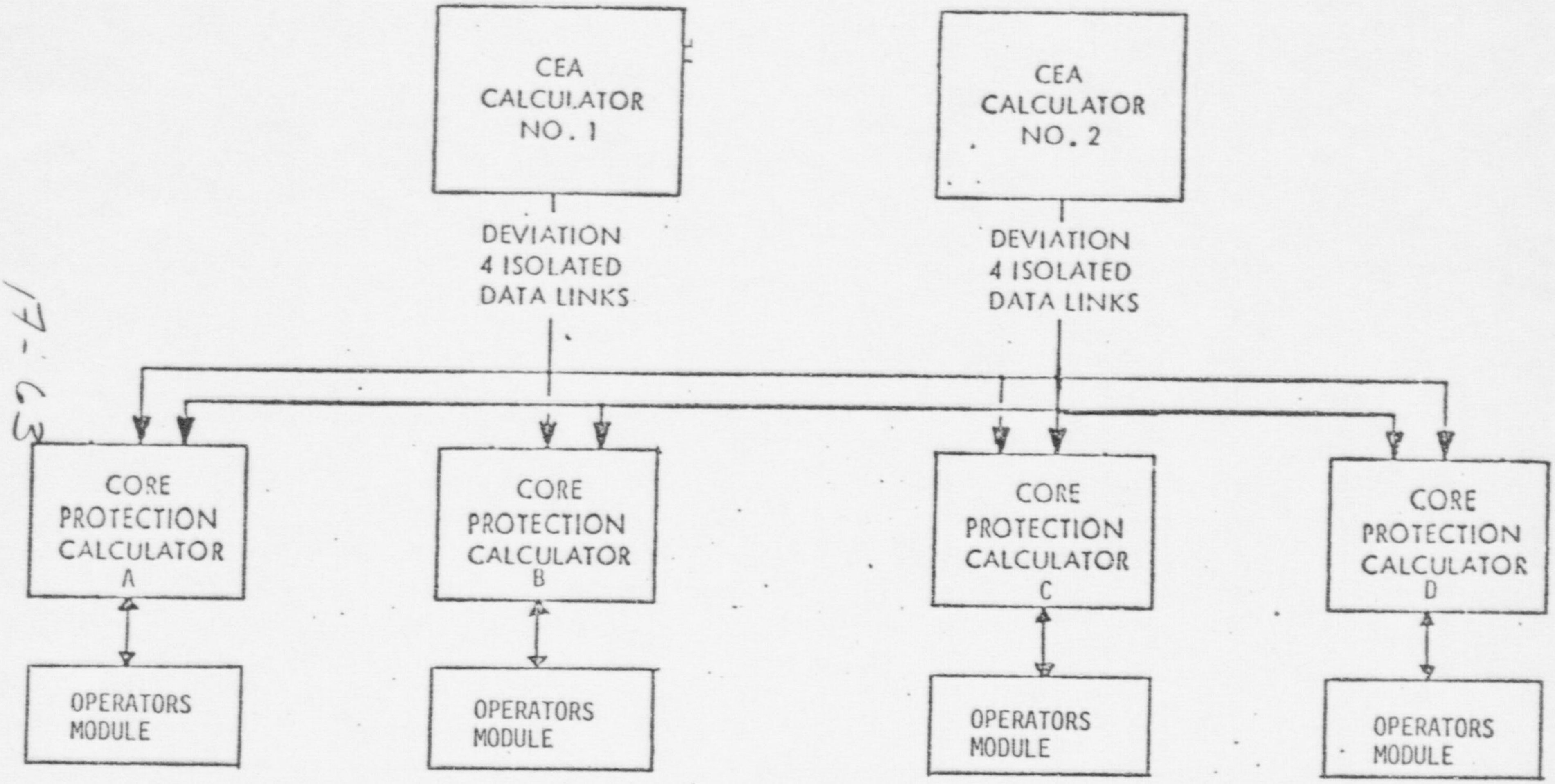
CPC MONITORED PLANT VARIABLES

MONITORED VARIABLE	NUMBER OF SENSORS PER CHANNEL
RCP ROTATIONAL SPEED	4 (1 PER PUMP)
COLD LEG TEMPERATURE	2
HOT LEG TEMPERATURE	2
PRIMARY PRESSURE	1
EX-CORE DETECTOR FLUX	3 DETECTORS, IN AXIAL STACK
CEA POSITION	20 (1 PER CEA SUBGROUP)

CEAC MONITORED PLANT VARIABLES

MONITORED VARIABLE	NUMBER OF SENSORS PER CHANNEL
CEA POSITION	81 (1 PER CEA)

CORE PROTECTION CALCULATOR SYSTEM



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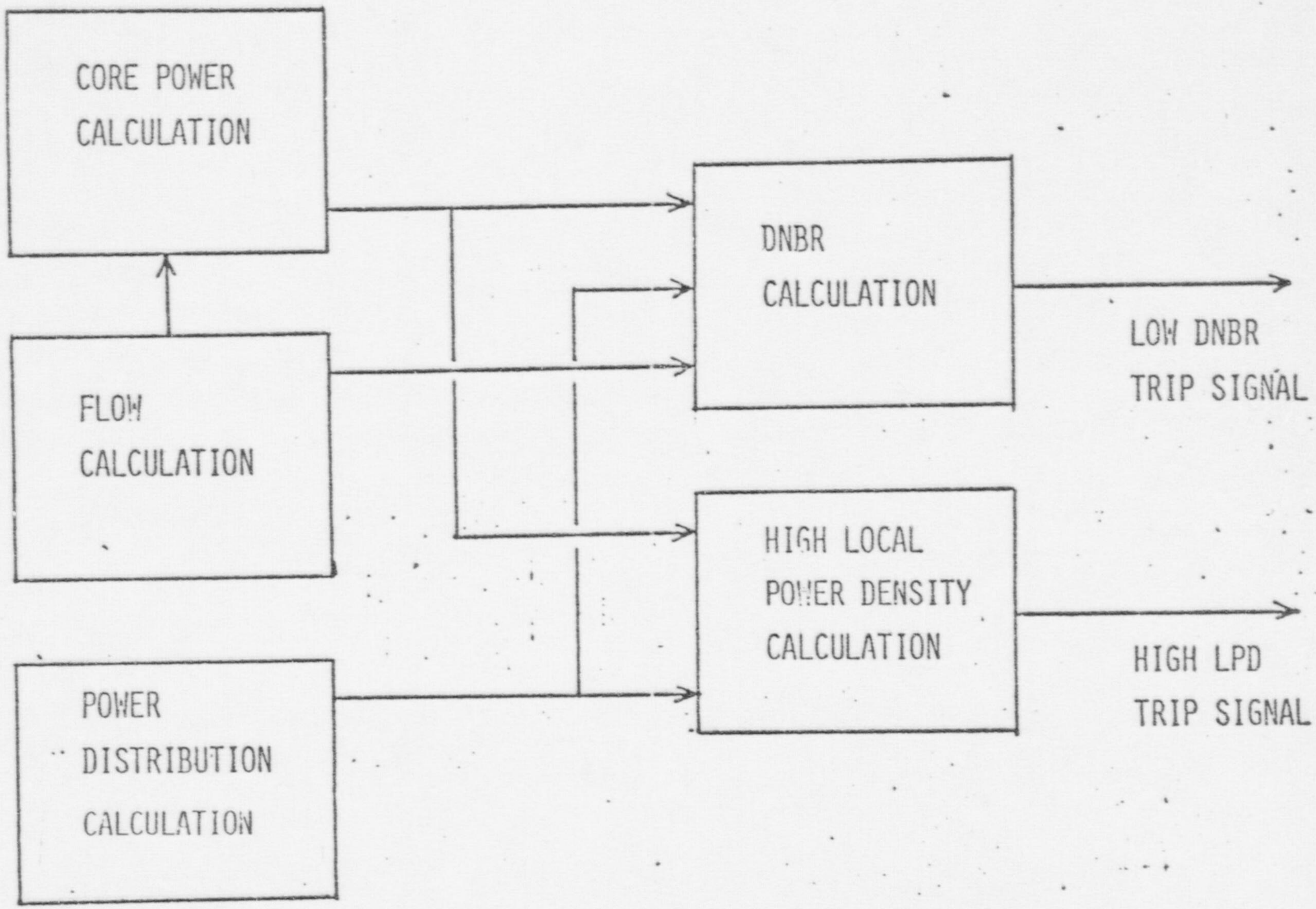
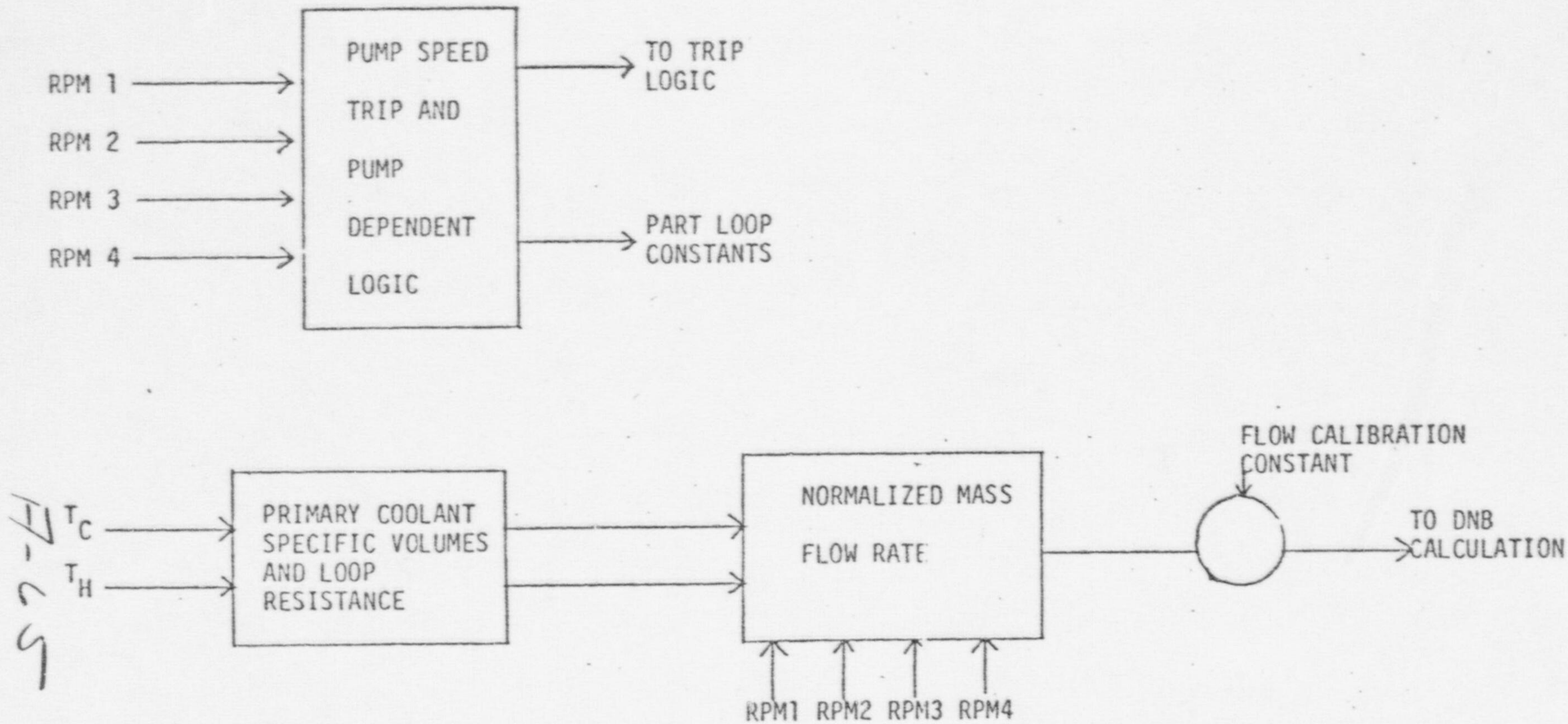
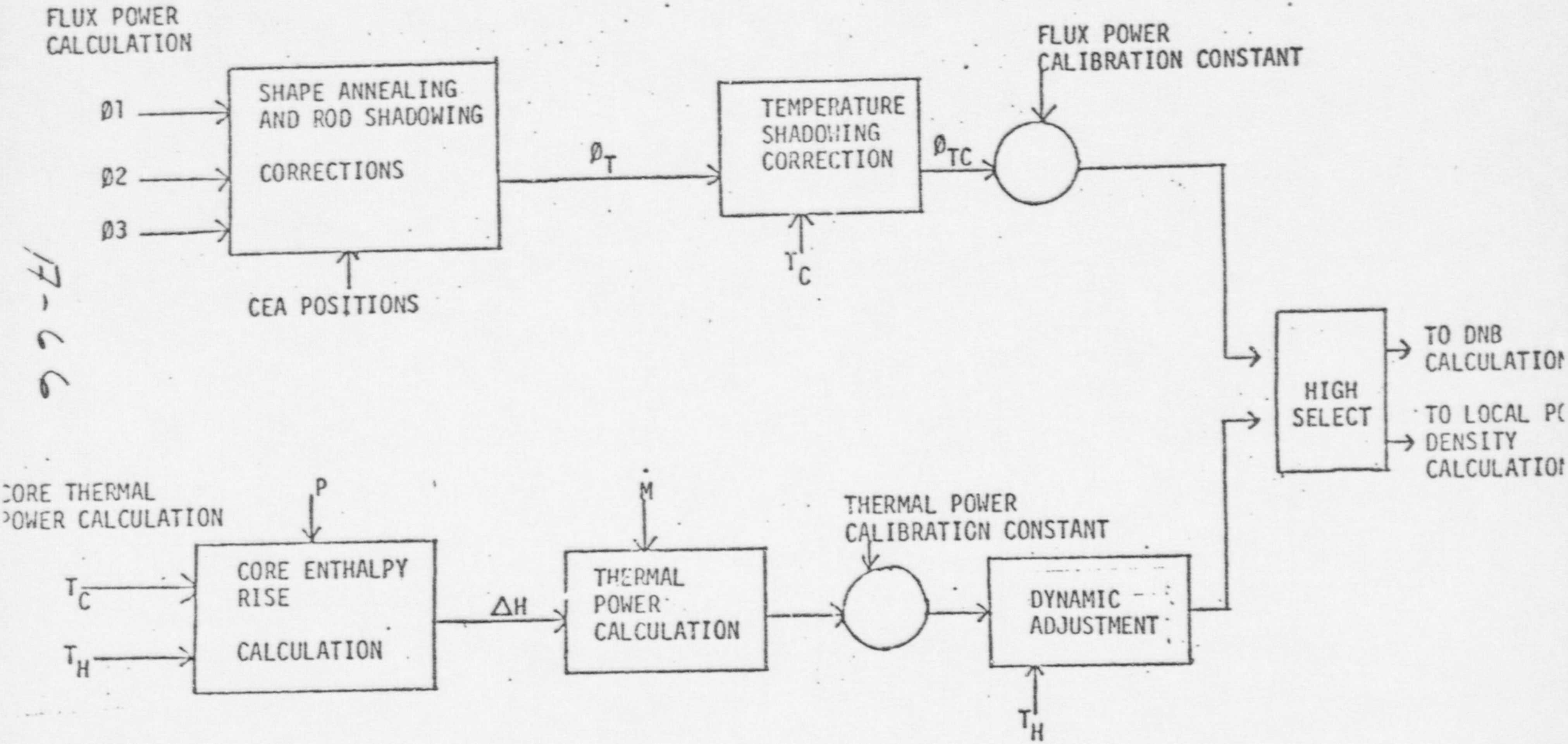
CPC FUNCTIONAL BLOCK DIAGRAM



CPC FLOW CALCULATION



CPC POWER CALCULATION



17-22

CORE PROTECTION CALCULATOR SYSTEMS

ANO-2 DESIGN

5 C-E 3410 MWT CLASS PLANTS

18 C-E SYSTEM 80 CLASS PLANTS

THE CPC DESIGNS FOR THESE PLANTS ARE IDENTICAL TO THE ANO-2 DESIGN EXCEPT FOR DIFFERENCES DUE TO

1. NUMBER OF CONTROL RODS
2. PLANT SPECIFIC DATA BASE CONSTANTS
3. PLANT SPECIFIC HARDWARE QUALIFICATION CRITERIA
4. ADVANCEMENTS IN METHODOLOGY TO IMPROVE PLANT PERFORMANCE

CPC ALGORITHMS AND UNCERTAINTIES

- TOPICS
  - POWER DISTRIBUTION
  - LOCAL POWER DENSITY
  - DNBR
  - UNCERTAINTY ASSESSMENT
- DETAILS
  - CPC UNCERTAINTY TOPICAL REPORT (CENPD - 170 AND SUPPLEMENT 1-P)

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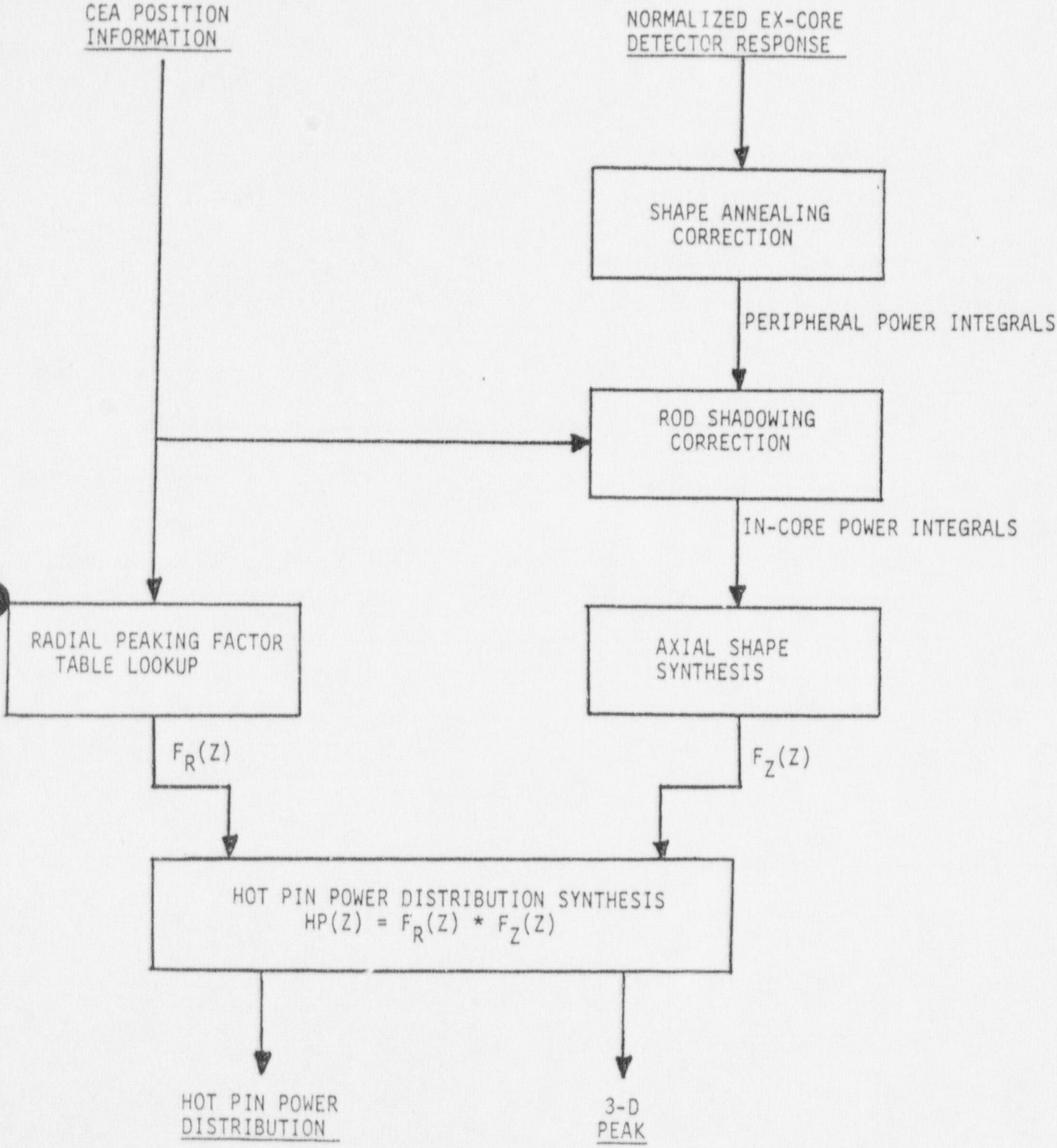
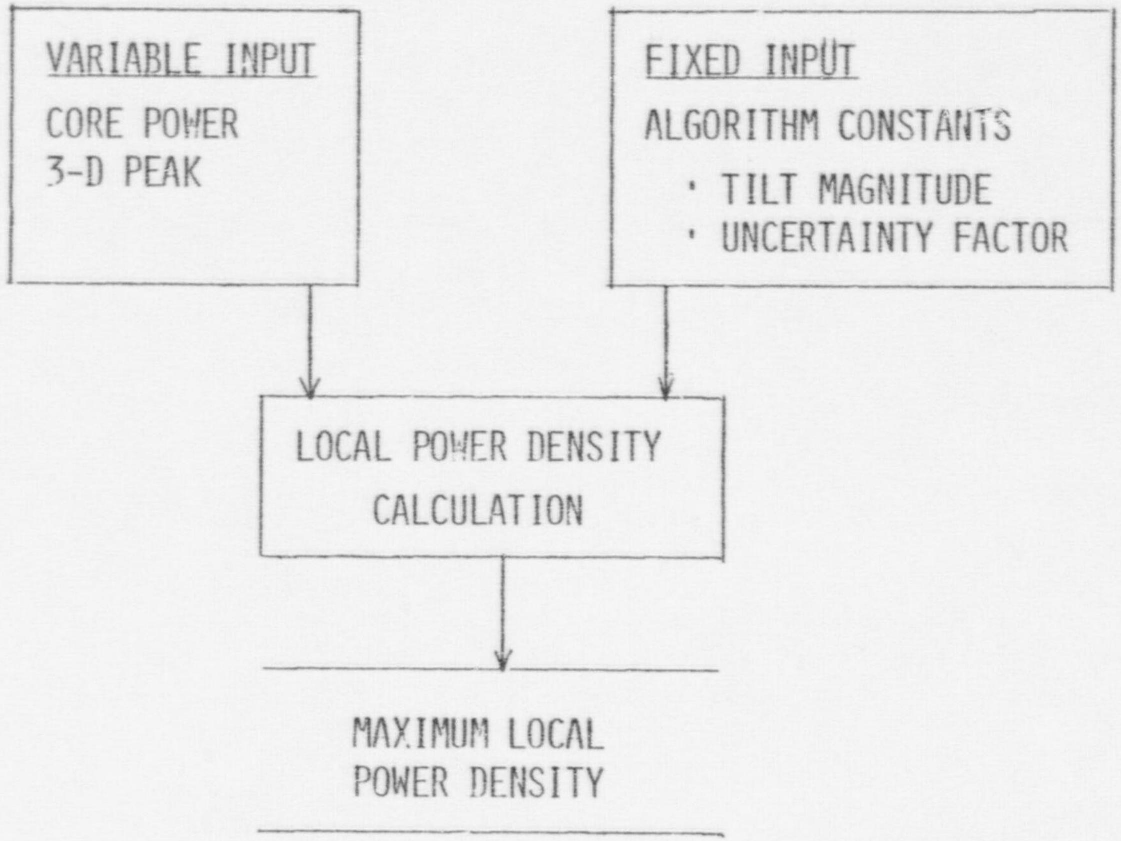
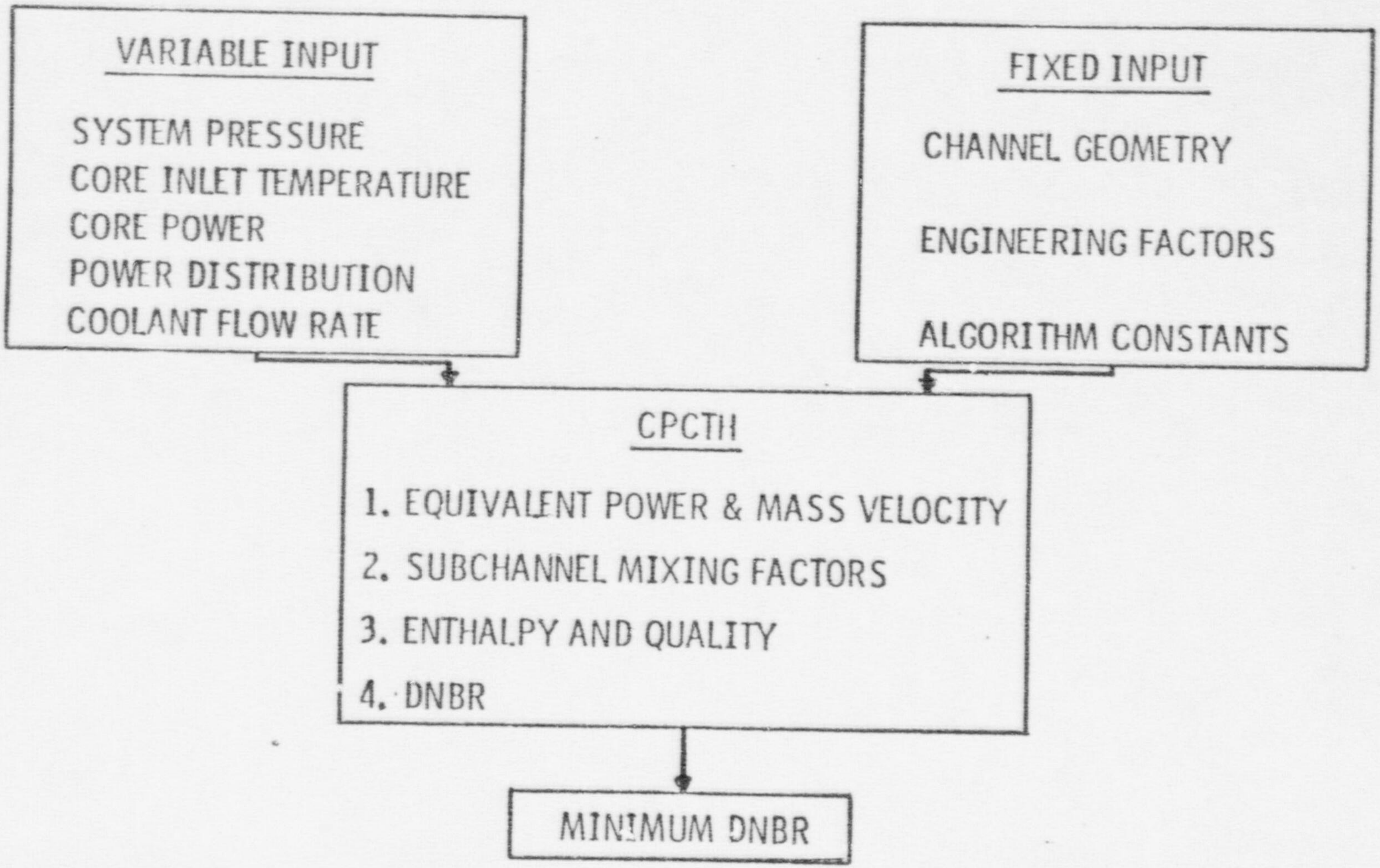
CPC CALCULATION OF CORE POWER DISTRIBUTION



LOCAL POWER DENSITY CALCULATION



DNBR CALCULATION



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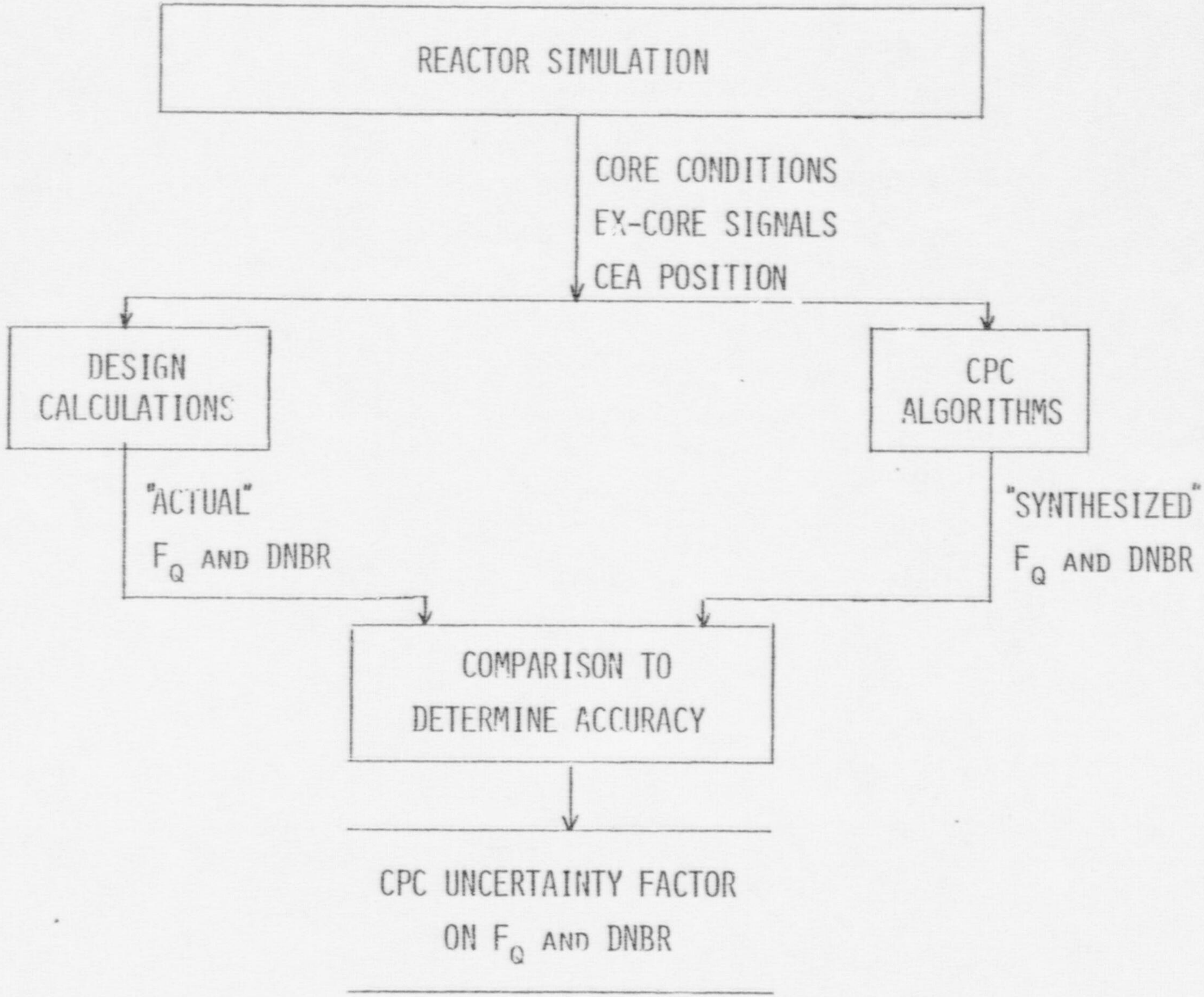
CPC UNCERTAINTY ASSESSMENT



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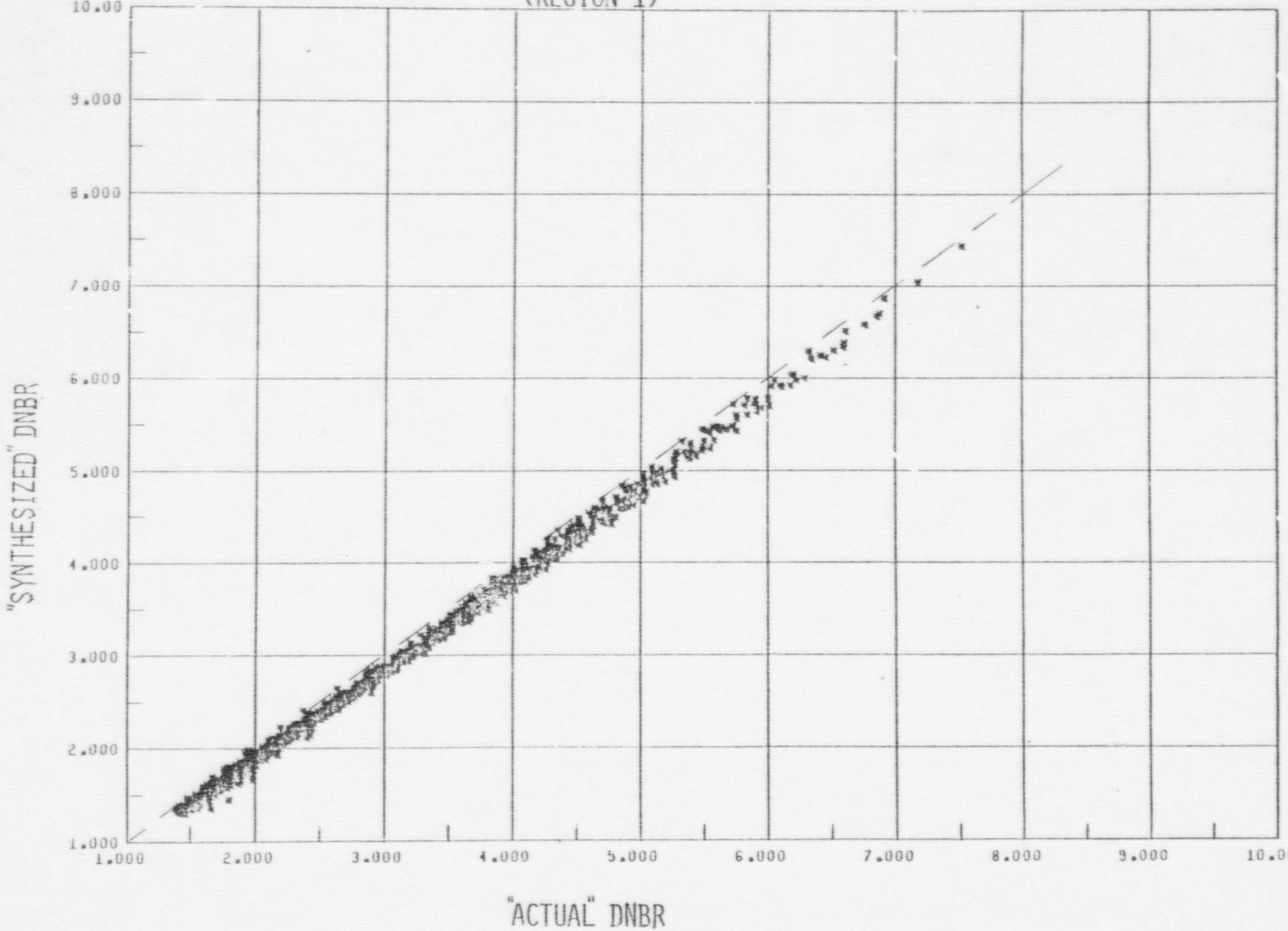
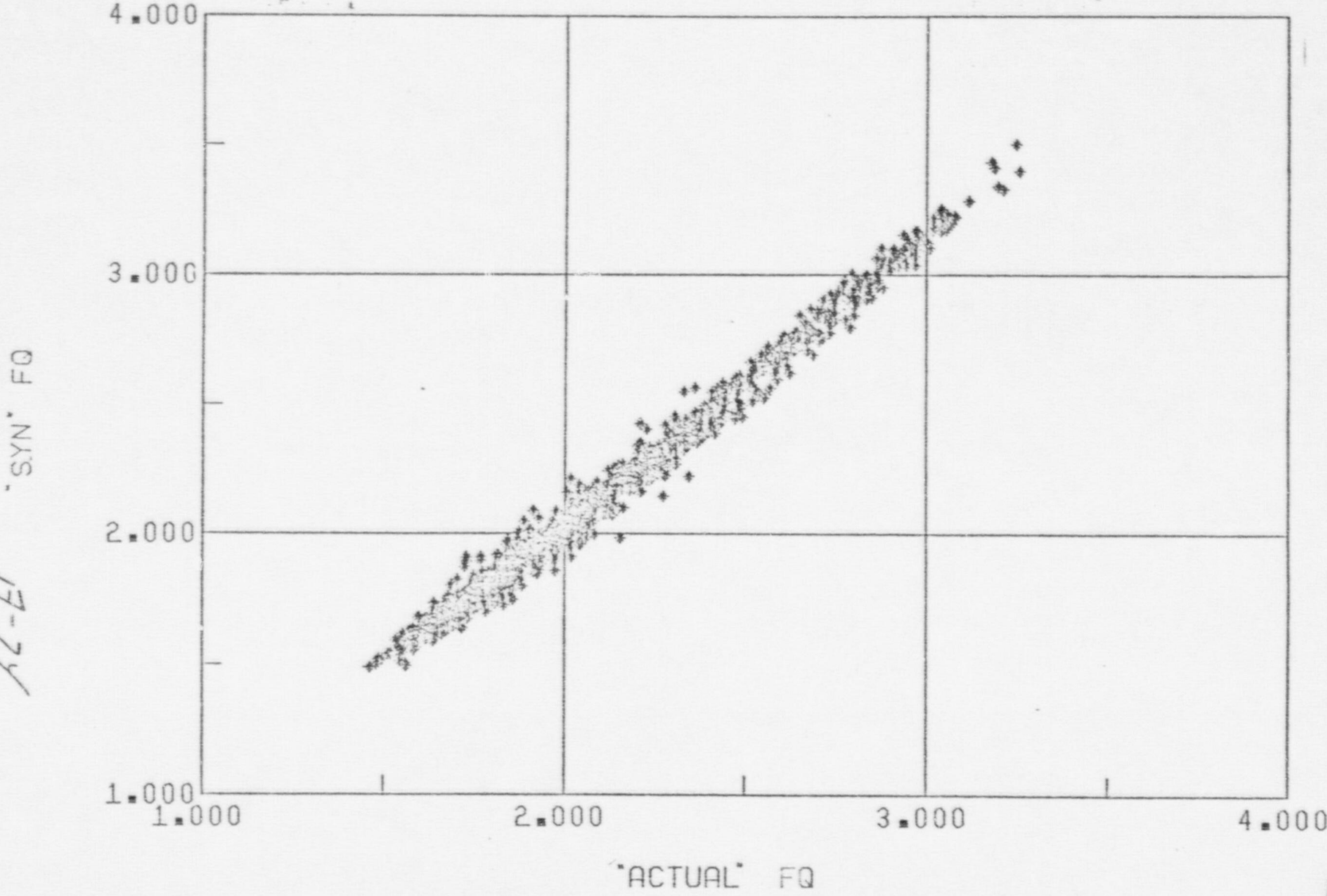
DNBR COMPARISON
(REGION 1)



CPC FQ COMPARISON (MOC)



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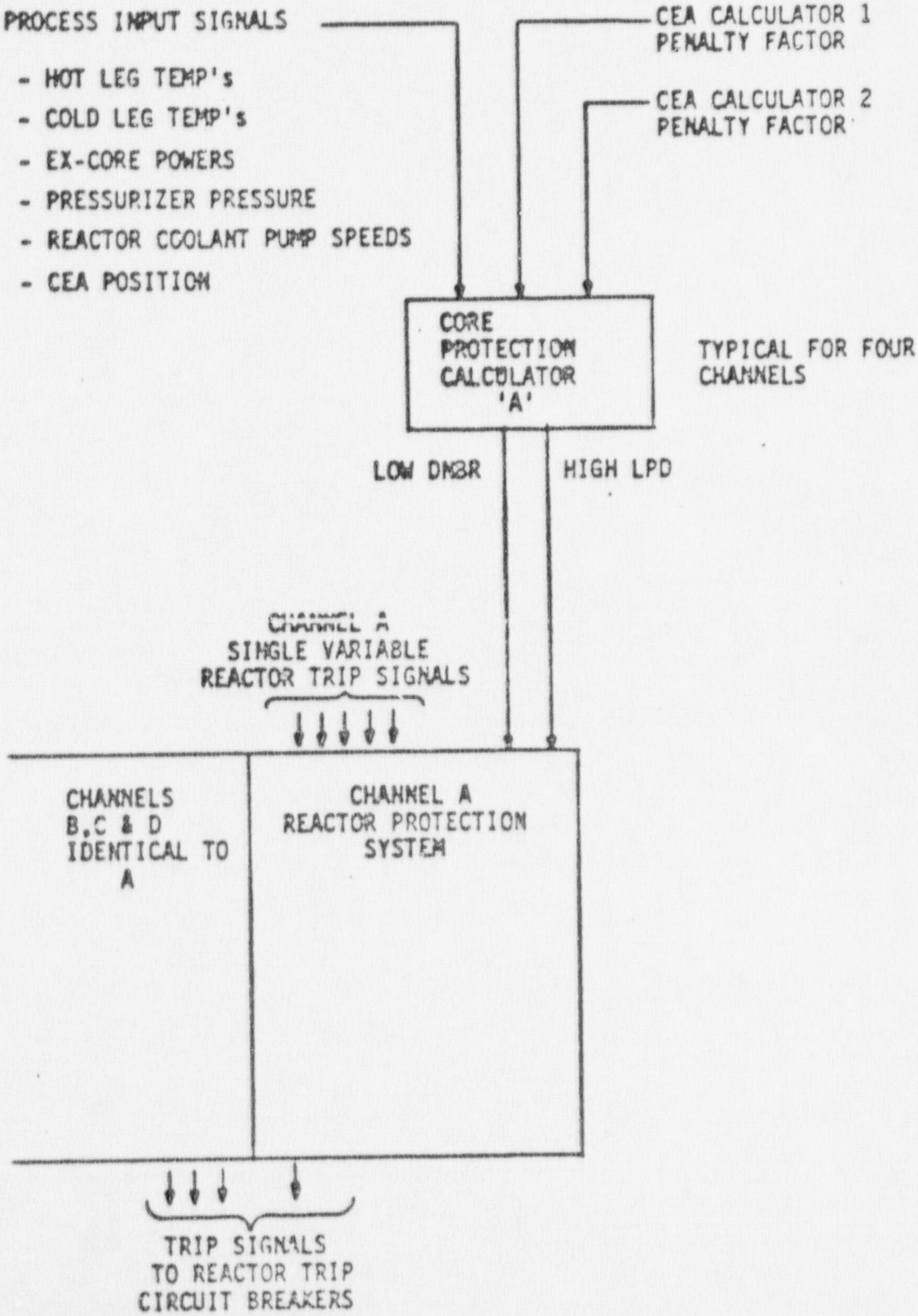
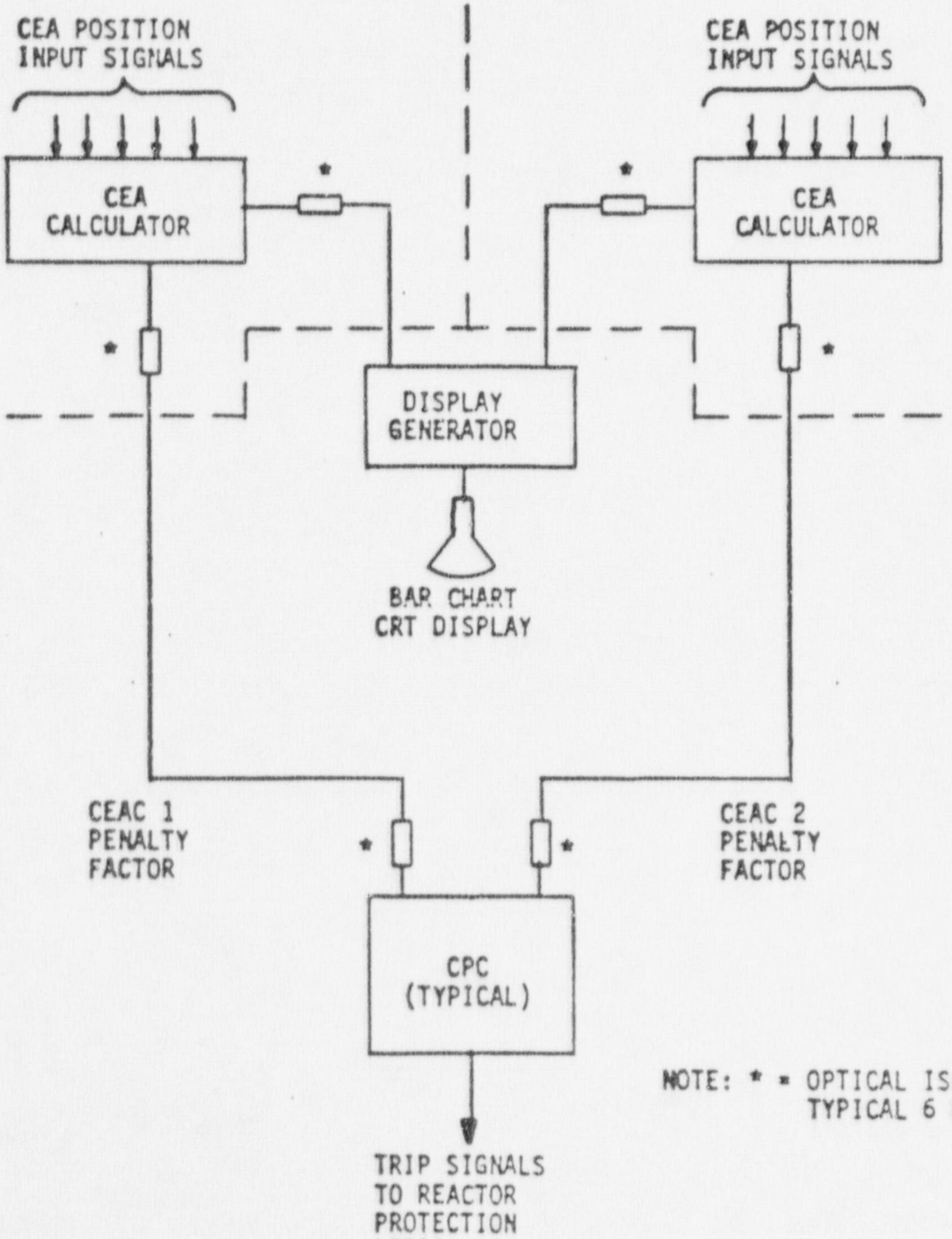
SIMPLIFIED REACTOR PROTECTION CHANNEL
BLOCK DIAGRAM CHANNEL A



SIMPLIFIED CEA CALCULATOR
BLOCK DIAGRAM



NOTE: * = OPTICAL ISOLATORS
TYPICAL 6 PLACES

CORE PROTECTION CALCULATOR SYSTEM
DISPLAY AND INDICATION

OPERATORS MODULE

SYSTEM STATUS INDICATION

OPERATOR DISPLAY OF SETPOINT AND CALCULATED VARIABLES

OPERATING BYPASS CONTROL AND INDICATION

KEYLOCK ADMINSTRATIVE CONTROL FOR CALCULATOR SECURITY

ADDRESSABLE CONSTANT ENTRY FOR CALIBRATION

CPC ANALOG INDICATORS

DEDICATED ANALOG METERS FOR

- DNBR MARGIN TO TRIP SETPOINT
- LPD MARGIN TO TRIP SETPOINT
- CALIBRATED NEUTRON FLUX POWER

ALARM ANNUNCIATORS

STATION ANNUNCIATORS ARE PROVIDED TO INDICATE TRIP STATUS AND OPERABILITY OF THE CPC SYSTEM

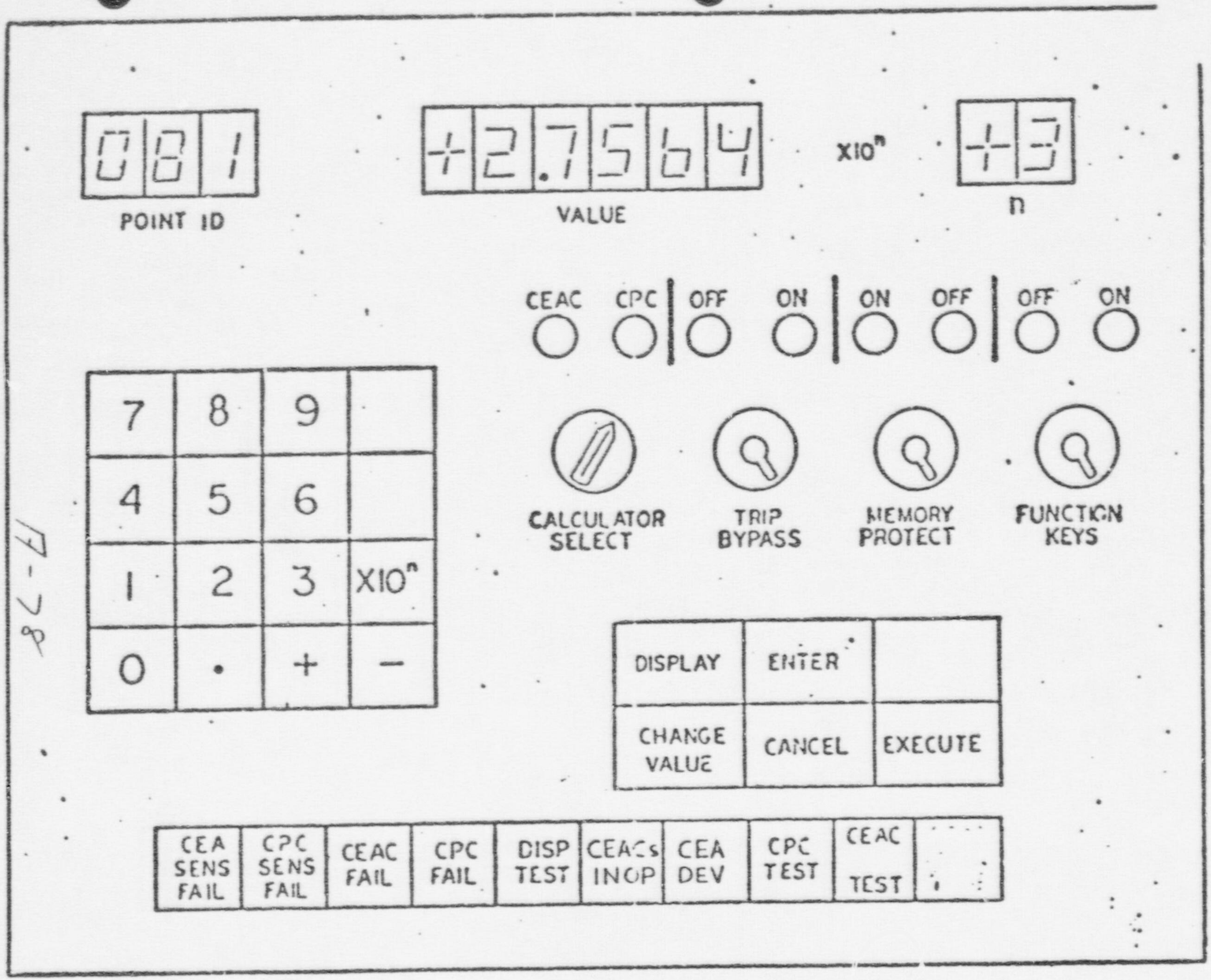
ANALOG PROCESS INDICATION

DEDICATED ANALOG METERS DISPLAY EACH CPC SENSOR INPUT VALUES EXCEPT CEA POSITION AND REACTOR COOLANT PUMP SPEED

CEA POSITION DISPLAY

A CRT DISPLAYS THE POSITION OF ALL 81 CONTROL ELEMENT ASSEMBLIES

THE DISPLAY IS SWITCH SELECTABLE TO EITHER OF TWO REDUNDANT SIGNAL CHANNELS.



Operator's Module

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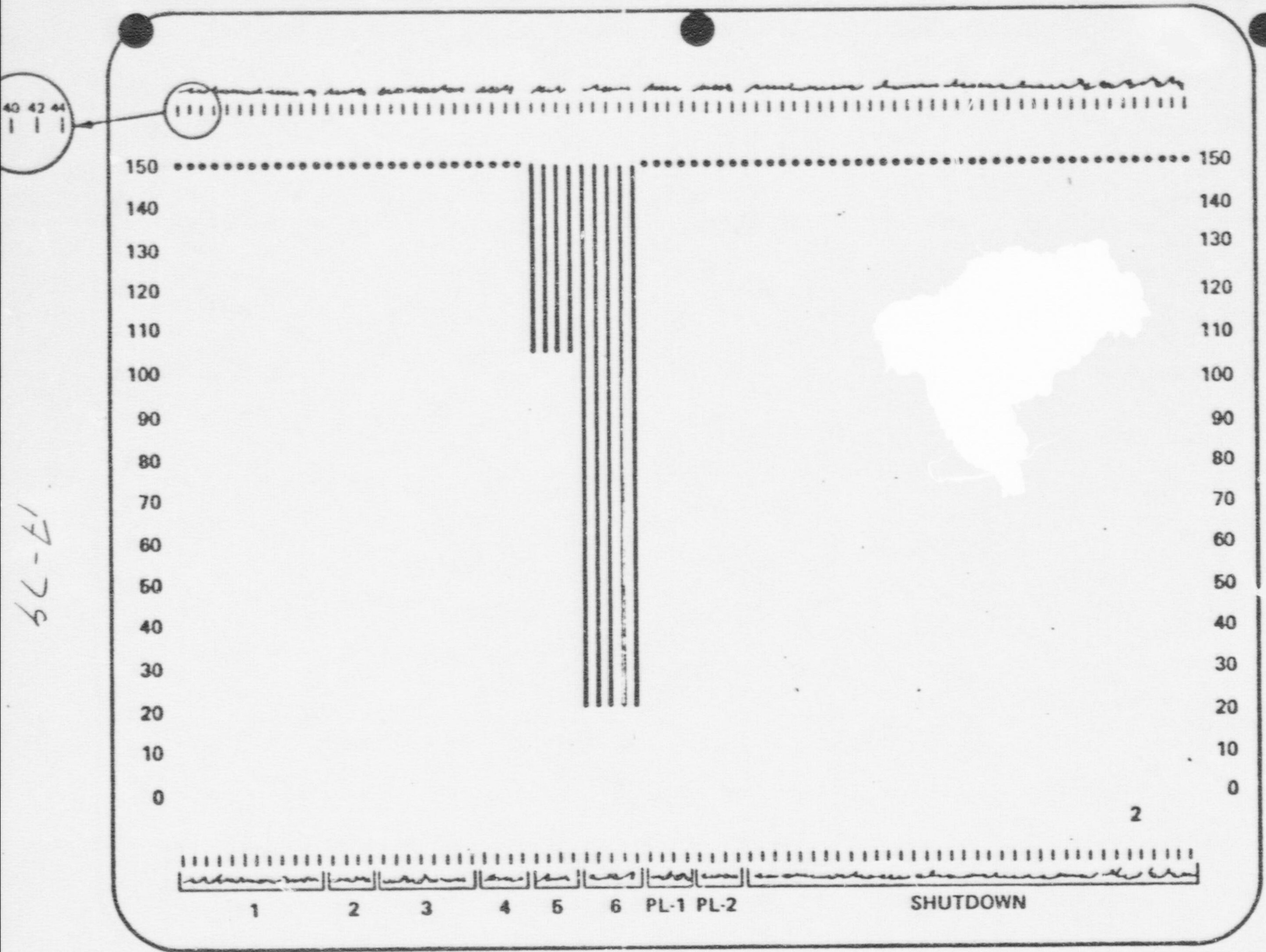

CEA POSITION DISPLAY
Figure 7

CORE PROTECTION CALCULATOR SYSTEM TESTING

AUTOMATIC ON LINE TESTING

EACH CALCULATOR IN THE CPC SYSTEM PROVIDES A RAPID SELF DIAGNOSTIC CAPABILITY TO ASSURE A FAIL SAFE RESPONSE TO DETECTED HARDWARE. FAILURES.

POWER FAIL	AUTOMATIC CHANNEL TRIP ON LOSS OF INPUT POWER
MACHINE MALFUNCTION	AUTOMATIC CHANNEL TRIP ON INTERNAL CALCULATOR MALFUNCTION
INPUT/OUTPUT SUBSYSTEM	AUTOMATIC CHANNEL TRIP ON DEFECTED MALFUNCTION LOSS OF POWER NO RESPONSE CALIBRATION VOLTAGE CHECK
MEMORY	AUTOMATIC CHANNEL TRIP ON PARITY INDICATES HARDWARE FAILURE CHECKSUM INDICATES FAILURE OR CHANGE OF MEMORY CONTENTS
SENSOR RANGE	AUTOMATIC ANNUNCIATION ON SENSOR FAILING HIGH OR LOW
WATCHDOG TIMER	AUTOMATIC CHANNEL TRIP AND LATCH IF THE COMPUTER FAILS REQUIRES MANUAL RESET
CALCULATION REASONABILITY	AUTOMATIC CHANNEL TRIP IF THE RANGE OF A CALCULATION IS EXCEEDED

PERIODIC TESTING

SURVEILLANCE

ALARMS, INDICATORS AND OPERATORS MODULES PROVIDE TIMELY INDICATION OF THE STATUS AND OPERABILITY OF THE CPC SYSTEM

OFF LINE TEST

A COMPREHENSIVE TEST CAPABILITY IS PROVIDED TO ALLOW THE OPERATOR TO CHECK THE HARDWARE AND SOFTWARE OPERABILITY OF THE CPC SYSTEM. THE TEST IS MANUALLY INITIATED WITH AUTOMATIC TEST ROUTINES AND A HARDCOPY PRINTOUT OF TEST RESULTS

SIGNAL INJECTION TEST

THE CAPABILITY FOR INJECTION OF "LIVE" PROCESS SIGNALS IS PROVIDED TO ALLOW PERIODIC VERIFICATION OF THE COMPLETE SIGNAL PATH WITHIN THE CPC SYSTEM

ISOLATION TEST

PERIODIC VERIFICATION OF THE OPTICAL ISOLATORS AND CEA POSITION ANALOG ISOLATORS CAPABILITY FOR ISOLATION IS PROVIDED

QUALIFICATION PROGRAM

THE QUALIFICATION PROGRAM IS DESIGNED TO DEMONSTRATE THAT THE CPC SYSTEM WILL PERFORM ITS REQUIRED FUNCTION CONSISTENT WITH THE DESIGN BASES OF THE NUCLEAR POWER GENERATING STATION.

HARDWARE

A COMPREHENSIVE PROGRAM OF TEST AND ANALYSES HAS BEEN PERFORMED TO DEMONSTRATE THAT THE HARDWARE IS CAPABLE OF PERFORMING ITS REQUIRED FUNCTIONS.

ENVIRONMENT

- SEISMIC TESTING AND ANALYSIS
- TEMPERATURE/HUMIDITY TEST
- ELECTRO-MAGNETIC NOISE TESTS

DESIGN FEATURES

- ISOLATION VERIFICATION TESTING
- ACCURACY/DRIFT TESTING
- DESIGN SPECIFICATION TESTS

RELIABILITY

- 5 MONTH FACTORY BURN IN TEST/ANALYSIS
- SITE BURN IN TEST

SOFTWARE

SIMILAR TO THE HARDWARE, THE SOFTWARE FOR THE CPC SYSTEM HAS UNDERGONE EXTENSIVE TEST AND ANALYSIS TO DEMONSTRATE ITS ADEQUACY.

PHASE I

EACH MODULAR ELEMENT OF THE SOFTWARE WAS TESTED TO ASSURE THAT IT CORRECTLY REFLECTED THE DESIGN REQUIREMENTS

INPUT SWEEP

THE INTEGRATED SYSTEM WAS THOUROUGHLY TESTED OVER THE ENTIRE PARAMETER RANGE OF REQUIRED SYSTEM OPERATION TO ASSURE CORRECTNESS OF IMPLEMENTATION AND TO DETERMINE THE UNCERTAINTY COMPONENT DUE TO THE DIGITAL COMPUTER CALCULATIONS.

ANO-2 INTEGRATED SYSTEM QUALIFICATION/FIELD TEST

PHASE II TESTS

THE ANO-2 CPC SYSTEM WAS THOROUGHLY EXERCISED UTILIZING "LIVE" SIGNALS DRIVEN FROM A SPECIAL PURPOSE SIMULATOR. RESULTS WERE COMPARED TO OFF LINE PREDICTIONS OF CPC PERFORMANCE.

PRE-OPERATIONAL TESTING

THE CPC SYSTEM WAS INSTALLED IN THE FIELD AND THE OPERABILITY OF THE SYSTEM WAS DEMONSTRATED.

SITE BURN-IN TEST

THE ANO-2 CPC SYSTEM WAS TESTED IN THE FIELD WITH A STATIC SIMULATOR TO VERIFY THE SYSTEMS PERFORMANCE AND OPERABILITY UNDER FIELD CONDITIONS.

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ACRS MEETING

AGENDA

CORE PROTECTION CALCULATOR SYSTEM
ARKANSAS NUCLEAR ONE — UNIT 2
DOCKET 50-368

APRIL 6, 1978

SAFETY OVERVIEW

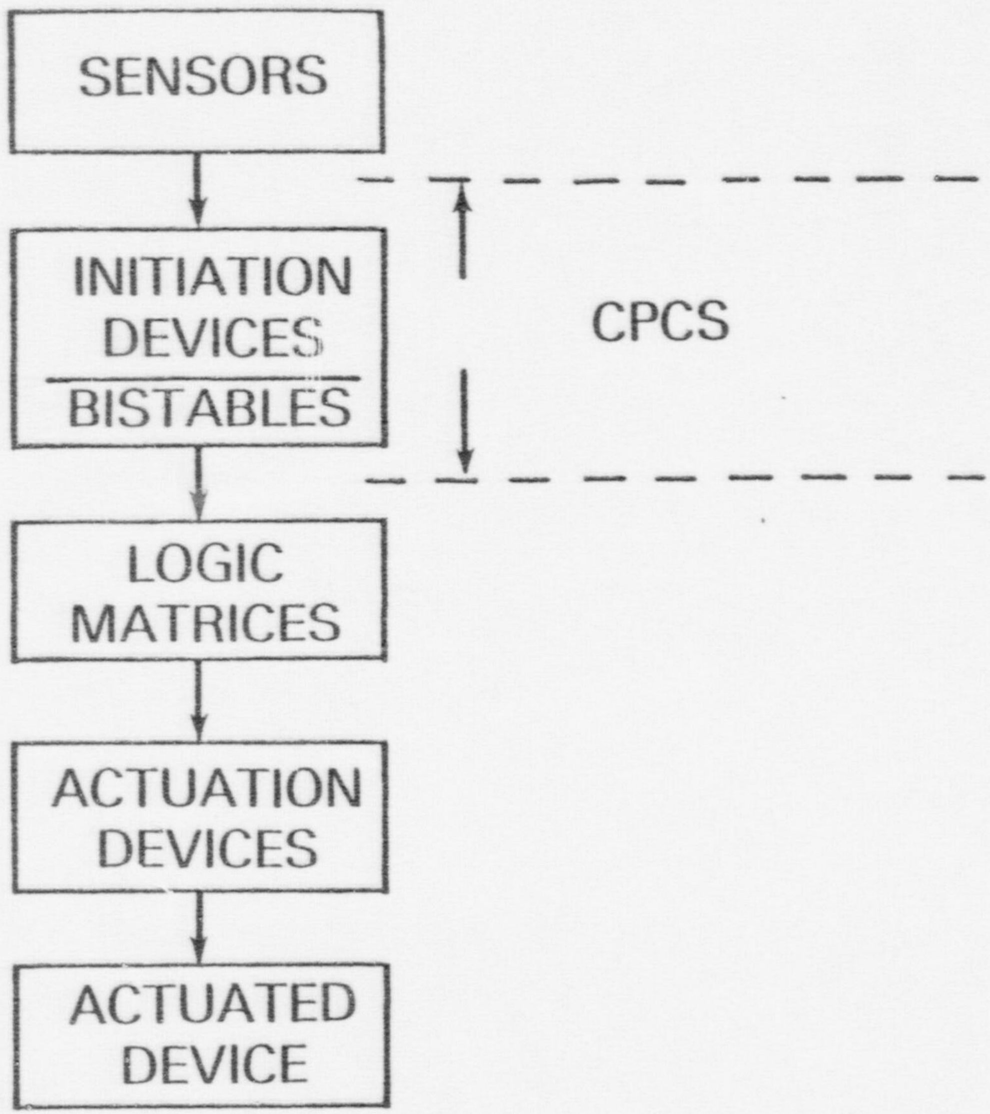
REVIEW CRITERIA

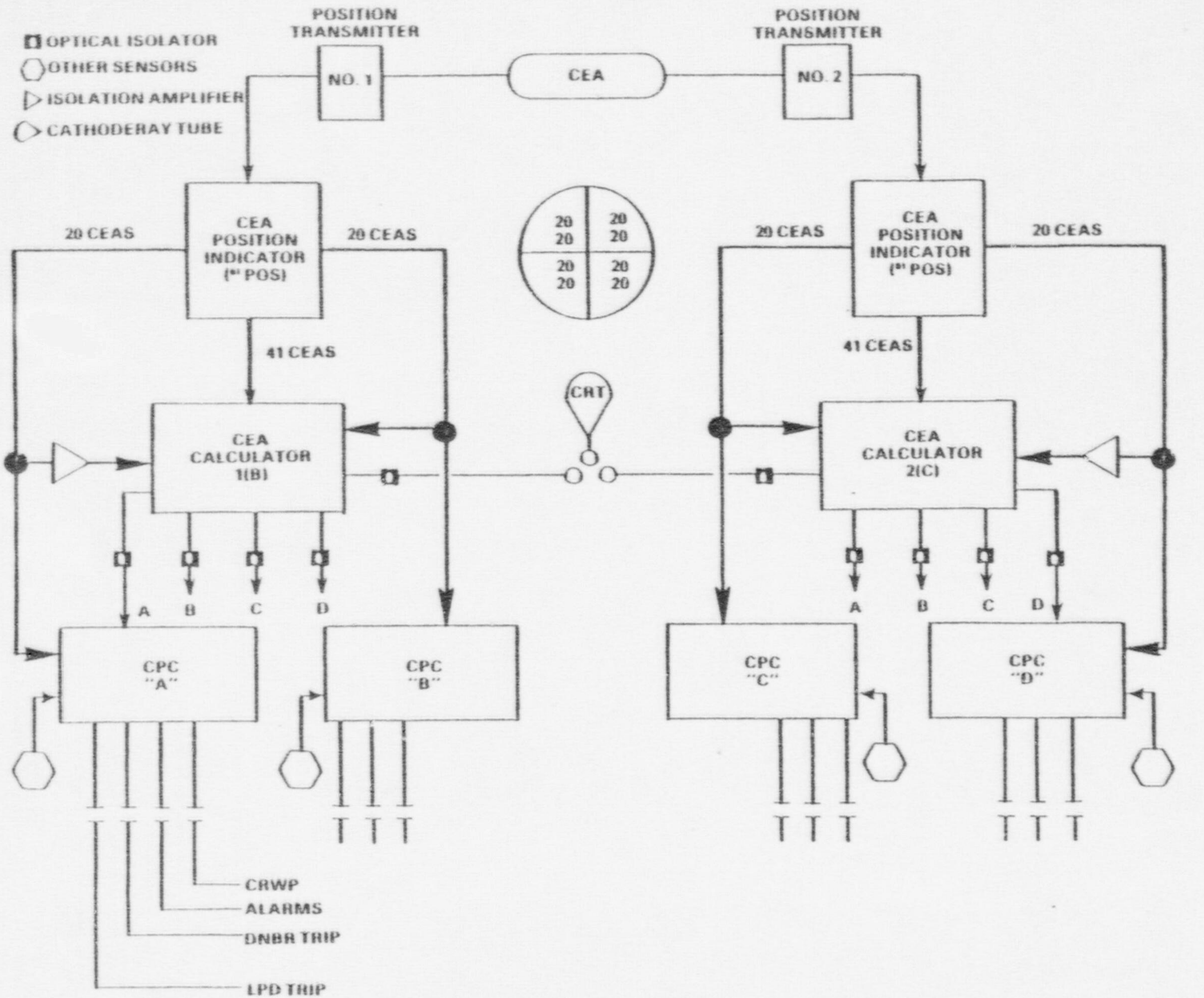
REVIEW METHODOLOGY

REVIEW STATUS

SAFETY OVERVIEW

PROTECTION CHANNEL FUNCTIONAL COMPONENTS





CPS HARDWARE CONFIGURATION BLOCK DIAGRAM

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CHARACTERISTICS

TRIPS

2	Digital	Lo DNBR
		Hi LPD
12	Analog	Hi Flux
		Hi Press
		Etc.

DIGITAL

- Sensor Signals
  - A/D Conversion of Continuous Signals
- Discrete Logic Execution
  - Protection Algorithms
  - Real Time
- Computer Output – Logic Matrix
- Communication – Information Readout
- Periodic Test & Surveillance

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IS IT FUNCTIONALLY ADEQUATE?

WILL IT OPERATE WHEN NEEDED?

CAN IT BE RELIABLY MAINTAINED?

REVIEW CRITERIA

REVIEW CRITERIA

- HARDWARE
  - GDC
  - Reg. Guides
  - Industry Standards
- SOFTWARE
  - GDC
  - Reg. Guides
  - Industry Standards
  - Surveys
    - Nuclear
      - Halden Project
      - KWU
      - Phoenix
      - SRP
    - S-Ware
      - Structure
      - Quality Control
  - Experience

REVIEW METHODOLOGY

REVIEW METHODOLOGY

STAFF

CPB	Physics
AB	T-H
	Design Basis
ICSB	EE

CONSULTANTS

S-Ware	Eng

EXECUTION

Review Plan
Task Force Meetings
Audits — Working Meetings

TRIPS WHICH PROVIDE PROTECTION IN EVENT OF CPC FAILURE

- High Pressurizer Pressure Trip
- Low Pressurizer Pressure Trip
- Low Steam Generator Water Level Trip
- Low Steam Generator Pressure Trip

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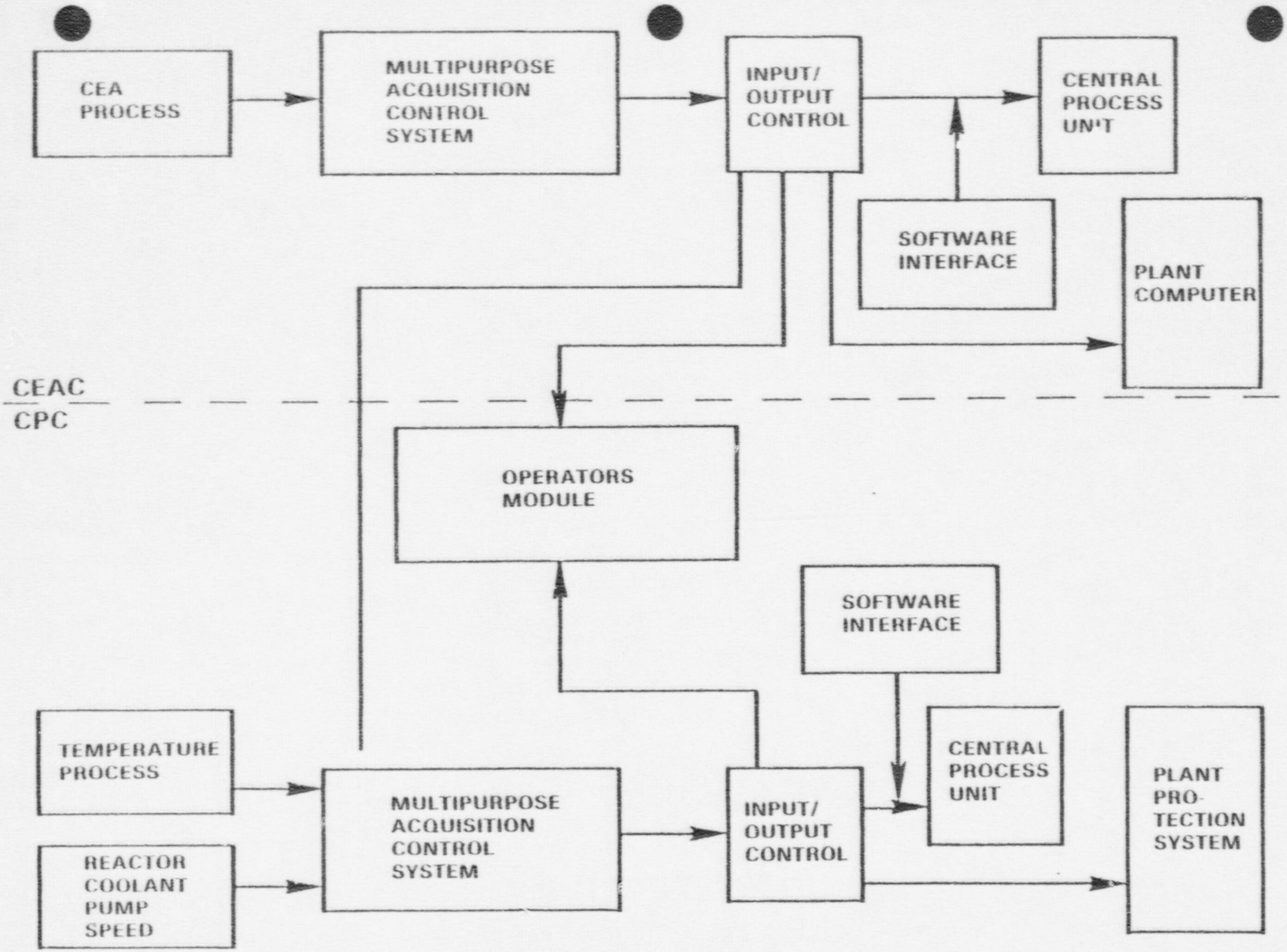
FIRST BACK-UP TRIP FOR EACH EVENT

- CEA Withdrawal — High Pressurizer Pressure (CPC Not 1st Trip)
- Boron Dilution — High Pressurizer Pressure
- LOF — High Pressurizer Pressure
- Excess Heat Removal — Low Steam Generator Water Level (CPC Not 1st Trip)
- Steam Generator Tube Rupture — Low Pressurizer Pressure (1600 PSIA)
- CEA Misoperation — Manual (COLSS Alarms and Control Board Indication)



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CHANNEL B CEAC/CPC

FUNCTIONS AND DOCUMENTS ASSOCIATED WITH REDESIGN AND REQUALIFICATION OF STORED COMPUTER PROGRAMS

Functional Requirements → Design → Development → Test

Functional Requirements	Design	Development	Test
CEN-44(A)-P CPC Functional Description and Supplement 1(P) Supplement 2(P) Supplement 3(P)	CEN-53(A)-P CPC/CEAC Data Base and Supplement 1(P) Supplement 2(P)	CEN-67(A)-P CPC/CEAC Program Assembly Listing	CEN-65(A)-P Phase I Test Audit
CEN-45(A)-P CEAC Functional Description	CEN-57(A)-P CPC Software Specification and Supplement 1(P)		CEN68(A)-P Phase II Test Audit
	CEN-58(A)-P CEAC Software Specification		CEN72(A)-P Phase I Test Report
	CEN-55A Phase II Test Procedure and Supplement 1(P)		CEN73(A)-P Phase II Test Report
	CEN-69(A)-P CPC/CEAC Executive System Software Specification		CEN-60(A) Core Protection Calculator Integrated System Burn-In Test Procedure

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TYPICAL HIGHLIGHTS

QUALITY ASSURANCE PLAN (16)

QUALIFICATION OF SOFTWARE CHANGE PROCEDURE (19)

PHASE II TEST AND TEST REPORT (24)

17-99

BURN IN TEST OF SYSTEM (18)

DATA LINK TO PLANT COMPUTER (20)

OPTICAL ISOLATOR QUALIFICATION (26)

TEST AUDITS

HARDWARE BURN-IN TEST

PHASE I TEST

PHASE II TEST

PROCESS PROTECTIVE CABINET THERMAL TEST

OPTICAL ISOLATOR QUALIFICATION TEST

EMI NOISE IMMUNITY TEST

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REVIEW STATUS

REVIEW STATUS

Positions Defined	27
Positions Reviewed and Closed	21
Positions Outstanding	6

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SUMMARY

Positions Outstanding	6
Start-Up Data/Analysis	3 (1, 5, 12)
Resolution Required Prior to License	1 (26)
License Condition	2 (14, 19)

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Plus

- Detailed Start-Up Procedures
- Start-Up Test Audit
- Start-Up Test Report
- Technical Specifications

DATA LINKS TO PLANT COMPUTER

BASES

GDC - 24

- ADDED DESIGN COMPLEXITY
- ADVERSE FUNCTIONAL FEEDBACK
- DATA COLLECTION FOR DESIGN BASES ANALYSES EVALUATION

RESOLUTION

- FOUR CHANNELS CONNECTED DURING INITIAL STARTUP AND REFUELING STARTUPS
- DISCONNECTED DURING OPERATION

APPENDIX XIII
ANO-2: Data Links to the Plant Computer

GDC 24

FACTORS IMPACTING SAFETY

- ADDED DESIGN COMPLEXITY
  - INCREASE PROBABILITY OF DESIGN ERROR
  - COMMON MODE FAILURE
- ELECTRICAL FAILURE PROPAGATION FROM NON-IE INTO IE
- ADVERSE FUNCTIONAL FEEDBACK
  - ANALOG SYSTEMS
    - SIMPLE PARAMETERS SUBJECT TO OPERATOR JUDGEMENT
  - CPCS
    - COMPLEX MULTIVARIABLE PARAMETERS NOT EASILY EVALUATED BY OPERATOR

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GDC 24

REQUIREMENTS

- ELECTRICAL ISOLATION
  - INDEPENDENCE
  - REDUNDANCY
  - SINGLE FAILURE
- SAFETY IS NOT SIGNIFICANTLY IMPAIRED
  - RELIABILITY AND COMMON MODE FAILURE
  - ADVERSE FUNCTIONAL FEEDBACK
- ADVANTAGES

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PROTECTION AND CONTROL INTERACTION

I. Hardwired Between Protection and Automatic Control System

II. Set Point/Calibration of Protection System Using Operating System

III. Incorporating Additional Non-Safety Design Features into the Protection System

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I. STATUS OF PROJECT REVIEW

- FSAR DOCKETED IN APRIL 1974
- SAFETY EVALUATION REPORT (SER) ISSUED ON NOVEMBER 11, 1977
- SUPPLEMENT ONE TO SER ISSUED ON MARCH 6, 1978
- ACRS ELECTRICAL SYSTEMS, CONTROL AND INSTRUMENTATION SUB-COMMITTEE MEETINGS ON THE CPCS WERE HELD ON MAY 20, 1977, JUNE 30, 1977 AND MARCH 20, 1978
- ACRS ANO-2 SUBCOMMITTEE MEETINGS WERE HELD ON JUNE 24, 1977 AND FEBRUARY 2, 1978
- ACRS MEETING INCLUDING THE CORE PROTECTION CALCULATOR SYSTEM WAS HELD ON FEBRUARY 9, 1978
- ESTIMATED DATE OF COMPLETION OF ALL MATTERS IN PARTS II THROUGH VI IN SUPPORT OF ISSUANCE OF AN OPERATING LICENSE - JUNE 1978

II. ITEMS RESOLVED SINCE FEBRUARY 9, 1978 ACRS MEETING

A SUPPLEMENT TO THE SER HAS NOT YET BEEN PREPARED FOR THE FOLLOWING ITEMS. HOWEVER THE COMMUNICATIONS BETWEEN THE STAFF AND THE APPLICANT IN RECENT WEEKS INDICATE THAT THESE ISSUES HAVE BEEN RESOLVED AND A SUPPLEMENT TO THE SER REPORTING OUR EVALUATION WILL BE PREPARED IN THE NEAR FUTURE.

FUEL ASSEMBLY BURNABLE POISON DESIGN VERIFICATION (4.0)

CEA SURVEILLANCE PLAN FOR $Al_2O_3 - B_4C$ (4.0)

CONTAINMENT PRESSURE DUE TO MAIN STEAMLINE BREAK MASS AND ENERGY RELEASES (6.2)

EVALUATION OF EMERGENCY CORE COOLING SYSTEM PERFORMANCE (6.3)

EVALUATION OF ADEQUACY OF PARAMETERS ESSENTIAL FOR ACCIDENT AND POST ACCIDENT MONITORING (7.5.1)

FINANCIAL QUALIFICATIONS (20.0)

CONTAINMENT SUMP TESTS (6.3.4)

III. NEW ITEMS SINCE FEBRUARY 3, 1978 ACRS MEETING

CONTAINMENT PURGE VALVE CLOSURE (6.0)

REGULATORY GUIDE 1.44 (5.0)

ECCS PUMP ROOM LEAKAGE (15.4.6)

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IV. COMPLETE LISTING OF ANO-2 REVIEW ISSUES (28 ISSUES)

* SEVEN OF THESE ARE RESOLVED

** THREE OF THESE HAVE BEEN IDENTIFIED SINCE FEBRUARY 9, 1978

SEISMIC QUALIFICATION (3.10)

ENVIRONMENTAL QUALIFICATIONS (3.11)

* FUEL ASSEMBLY BURNABLE POISON DESIGN VERICATION (4.0)

* CEA SURVEILLANCE PLAN FOR $Al_2O_3 - B_4C$ (4.0)

CEA GUIDE TUBE WEAR (4.0)

** REGULATORY GUIDE 1.44 (5.0)

** CONTAINMENT PURGE VALVE CLOSURE (6.0)

* CONTAINMENT PURESSURE DUE TO MAIN STEAM LINE BREAK MASS AND ENERGY RELEASES (6.2)

CONTAINMENT LEAKAGE TESTING PROGRAM (6.2.6)

ENVIRONMENTAL QUALIFICATIONS OF SAFETY RELATED INSTRUMENTATION FOR MAIN STEAM LINE BREAK INSIDE CONTAINMENT (6.2.1)

* EVALUATION OF EMERGENCY CORE COOLING SYSTEM PERFORMANCE (6.3)

* CONTAINMENT SUMP TESTS (6.3.4)

VERIFICATION OF IMPLEMENTATION OF INSTRUMENTATION & CONTROL SYSTEMS DESIGN (7.1)

INPUT FAULT AND SURGE TESTING OF POWER SUPPLIES (7.2.2)

* EVALUATION OF ADEQUACY OF PARAMETERS ESSENTIAL FOR ACCIDENT AND POST-ACCIDENT MONITORING (7.5.1)

REDUNDANT VALVE POSITION INDICATION (7.6.3)

SEPARATION CRITERIA FOR CONDUITS (7.9.4)

FIRE PROTECTION (9.7)

FEEDWATER HAMMER IN STEAM GENERATORS (10.0)

PREOPERATIONAL TESTS (14.0)

EMERGENCY PLAN (13.3)

RCP SEIZURE ANALYSIS USING CESEC CODE (15.4.2)

REVIEW OF MAIN STEAM LINE BREAK ANALYSIS (15.4.2)

** ECCS PUMP ROOM LEAKAGE (15.4.6)

* FINANCIAL QUALIFICATIONS (20.0)

OFFSITE GRID STABILITY (8.2)

GENERIC ISSUES - SPECIFIC ANO-2 ACTIONS

REACTOR VESSEL SUPPORTS (3.9.3)

OVERPRESSURE PROTECTION, LONG TERM (5.7)

V. SCHEDULE FOR RESOLUTION OF ITEMS

A. THE FOLLOWING TEN ITEMS ARE EXPECTED TO BE RESOLVED FOR THE ISSUANCE OF THE OL AND REPORTED IN A SUPPLEMENT TO THE SER BY JUNE 1, 1978

ENVIRONMENTAL QUALIFICATIONS (3.11)

REGULATORY GUIDE 1.44 (5.0)

CONTAINMENT SUMP TESTS (6.3.4) Resolved

VERIFICATION OF IMPLEMENTATION OF INSTRUMENTATION AND CONTROL SYSTEMS DESIGN (7.1)

SEPARATION CRITERIA FOR CONDUITS (7.9.4)

FEEDWATER HAMMER IN STEAM GENERATORS (10.6)

EMERGENCY PLAN (13.3)

REVIEW OF MAIN STEAMLINE BREAK ANALYSIS (15.4.2)

REACTOR VESSEL SUPPORTS (3.9.3)

OVERPRESSURE PROTECTION, LONG TERM

B. THE SCHEDULE FOR RESOLUTION OF THE FOLLOWING TWELVE ITEMS IS DEPENDENT PRIMARILY ON THE STAFF'S FINDINGS RESULTING FROM THE REVIEW OF PRESENTLY SUBMITTED INFORMATION OR ON THE DATE OF SUBMITTAL OF CURRENTLY OUTSTANDING INFORMATION.

SEISMIC QUALIFICATION (3.10)

CEA GUIDE TUBE WEAR (4.0)

CONTAINMENT PURGE VALVE CLOSURE (6.0)

CONTAINMENT LEAKAGE TESTING PROGRAM (6.2.6)

ENVIRONMENTAL QUALIFICATIONS OF SAFETY RELATED INSTRUMENTATION FOR THE MSLB INSIDE CONTAINMENT (6.2.1)

INPUT FAULT AND SURGE TESTING OF POWER SUPPLIES (7.2.2)

REDUNDANT VALVE POSITION INDICATION (7.6.3)

FIRE PROTECTION (9.7)

PREOPERATIONAL TESTS (14.0)
(LOSS OF OFFSITE POWER TESTS)

RCP SEIZURE ANALYSIS USING CESEC CODE (15.4.2)
(CESEC VERIFICATION TESTING PROGRAM)

ECCS PUMP ROOM LEAKAGE (15.4.6)

OFFSITE GRID STABILITY (8.2)
(OFFSITE POWER SYSTEM DEGRADATION)

VI. ITEMS WHOSE RESOLUTION FOR THE OL MAY INCLUDE CONDITIONS TO THE OL

ENVIRONMENTAL QUALIFICATIONS (3.11)

CONTAINMENT PURGE VALVE CLOSURE (6.0)

CONTAINMENT PRESSURE DUE TO MSLB MASS AND ENERGY RELEASES (6.2)

REDUNDANT VALVE POSITION INDICATION (7.6.3)

FIRE PROTECTION (9.7)

REOPERATIONAL TESTS (14.0)

RCP SEIZURE ANALYSIS USING CESEC CODE (15.4.2)

REACTOR VESSEL SUPPORTS (3.9.3)

OVERPRESSURE PROTECTION, LONG TERM (5.7)

CPCS POSITIONS:

1. UNCERTAINTY ASSOCIATED WITH ALGORITHMS

5. CABLE SEPARATION

12. ELECTRICAL NOISE AND ISOLATION QUALIFICATION

VII. CONCLUSIONS

FOR OPERATING LICENSE ISSUANCE

1. NON-CPCS TOTAL NUMBER OF ITEMS (28)

   A. SIX OF THESE 28 ITEMS ARE NOW RESOLVED

   B. TEN MORE OF THESE 28 ITEMS ARE EXPECTED TO BE RESOLVED AND REPORTED IN AN SSER BY JUNE 1, 1978

   C. THE DATE OF RESOLUTION OF THE REMAINING TWELVE IS DEPENDENT PRIMARILY ON THE RESULTS OF ONGOING STAFF REVIEWS (4) OR THE DATE OF SUBMITTAL OF CURRENTLY OUTSTANDING INFORMATION (8)

2. CORE PROTECTION CALCULATOR SYSTEM STATUS ON SEVEN OUTSTANDING POSITIONS IN SSER NO. 1

   4. CEAC SEPARATION CRITERIA - SEE POSITION 26

   14. SEISMIC QUALIFICATIONS - OUTSTANDING

   15. ADDRESSABLE CONSTANTS - RESOLVED

   18. BURN IN TEST - RESOLVED

   19. QUALIFICATION OF SOFTWARE CHANGE PROCEDURE - OUTSTANDING

   20. DATA LINKS TO PLANT COMPUTER - RESOLVED

   26. OPTICAL ISOLATORS - OUTSTANDING

HIGHLIGHTS
FLOATING NUCLEAR PLANT SUBCOMMITTEE MEETING
LIQUID PATHWAY GENERIC STUDY
Washington D.C.
March 22, 1978

1. The FNP radioactivity release to the hydrosphere consists of the prompt release and a long term leach release. The NRC Staff has assumed that the prompt release would come from the sump water discharging into the ocean upon hull melt-through and would consist of from 10% to 80% of the iodine and cesium inventory which is about 100 million curies. The leach release for the FNP results from the core debris sitting on the ocean floor and leaching of the cesium and strontium. The NRC Staff has assumed that about 50% of the total cesium and strontium in the debris would leach the first week which is about 10 million curies.

2. Sandia laboratory tests using 7 gram molten corium samples being dropped into seawater had cesium leach rates of 0.075% minimium to 0.80% maximum and strontium leach rates from non-detectable to 2.5% for the first 3 days and about the same amount for an additional 29 days. This indicates that the NRC Staff assumption of 50% leaching within the first week may be conservative.

3. The NRC Staff warns against just looking at the numbers to make a decision regarding the LBP and FNP comparision. They suggest looking at the qualitative conclusions of NUREG-0440.

4. The assumption used in the LPGS is that the probability of a core melt accident is the same for a LBP and FNP and that in case of a core melt the airborne release is the same. No additional consideration was given to the FNP airborne release.

5. The ACRS Consultants present at the end of the Subcommittee meeting all indicated that they had no major disagreements with the existing study. Several suggestions were made for additional consideration but it was felt these would have a minor effect on the overall conclusions.

6. A suggestion by Dr. Foster which appeared to deserve further consideration was that since the prompt sump water release is assumed to release 100 million curies while the leach release is considerably smaller (10 million curies the first week) that consideration should be given to ensure that the sump water which contains the in-rushing seawater after melt-through, remains in the hull.

7. It was noted that no NRC Staff member has expressed disagreement with the results and conclusions of the LPGS report; however, it was also noted that some NRC Staff views indicate that changes in the FNP design or siting configuration may be needed to conclude that the FNP design does not pose an undue risk as a result of a Class 9 event.

8. The Subcommittee recommended that the NRC Staff and OPS come to the ACRS at the April 6-8, 1978 Meeting and discuss LPGS. It was noted that a letter from the ACRS commenting on the LPGS Report would be appropriate.

NRC STAFF CONCLUSIONS IN NUREG-0440

Based on the results and analyses of this study the following conclusions were reached:

- The risks associated with uninterdicted releases to the liquid pathway at an FNP are generally less than for an LBP for the spectrum of design basis events.

- The liquid pathway risks do not involve acute loss of life, although as discussed above, some long-term effects could be manifested and economic impacts could be large. The significance of the differences in the liquid pathway-related risks between FNPs and LBPs depends, in part, on the risks of the liquid pathway as opposed to the air pathway. Based on the information reviewed and the staff's independent analyses, for most sites the risks to the public of any of the various categories of accidents (Class 1-9) are likely to be dominated by the air pathway. However, in the case of the FNP, the release of large quantities of fission products to the water resulting from a coremelt accident is expected to result in economic and other impacts greater than for an LBP (although the impacts may be different in kind) and approaching those associated with the air pathway.

- The expected liquid pathway impacts resulting from a core-melt accident at an FNP are different from and greater than the expected impacts from an LBP. This results primarily from the fact that measures to isolate releases to the immediate vicinity of the site are not feasible for an FNP for the first few days following a core-melt accident. During this time, significant quantities of radioactivity would be released to the open water body with resulting impacts that are greater than those associated with an LBP where isolation (interdiction) at the source would essentially eliminate off-site impacts.

This study has as its objective an examination of the comparability of the risks associated with accidental releases via liquid pathway at an FNP to those at a similarly designed LBP. Based on the present design of the FNP and its site structure design envelopes, the overall conclusion is that, while the liquid pathway risks are small for both types of plants, the core-melt impacts are not comparable with the FNP impacts being greater. The staff results indicate that the consequences associated with core-melt releases to the liquid pathway at an FNP are higher than those associated with an LBP and that prompt interdiction measures to keep the initial releases (within about 1 week) from entering the open waterbody (liquid pathway) are not feasible for an FNP. The staff considers this combination of differences in release magnitude and interdiction potential to be significant. The impacts from releases to the liquid pathway from FNPs could be reduced to the level of impacts from LBPs if the ability to prevent the rapid release of large quantities of activity to the open water body is provided.

An evaluation of the environmental, economic, and social significance of the above findings will be performed as part of the overall assessment of the FNP concept.

SCHOOL OF NUCLEAR ENGINEERING

a, Georgia 30332 (404) 894-3720

March 27, 1978

APPENDIX XV
Consultant's Reports

Mr. G. R. Quittschreiber
Senior Staff Engineer
Advisory Committee on Reactor Safeguards
United States Regulatory Commission
Washington, D.C. 20555

Dear Mr. Quittschreiber:

Since I found it necessary to leave the meeting of the ACRS Subcommittee on the Floating Nuclear Plant at 5:00 p.m. (the scheduled time for conclusion of the meeting), I am not sure the consultants were requested to submit a written statement. However, since this is our usual practice, I am enclosing a few brief comments.

Sincerely,

Karl Z. Morgan
Neely Professor

KZM:rs

Enclosure

Report of Karl Z. Morgan on the Meeting of the Subcommittee on the Floating Nuclear Plant (FNP) Held in Washington, D.C. March 22, 1978

I think it is important to emphasize that for conditions up to Class 9 core melt accidents the FNP has a better radiation safety score than the LBP. I interpret NUREG-0440 to indicate that the risk for airborn radioactive contamination is greater than that for the contamination released to the water and that the risks from airborn contamination are about equal for the FNP and the LBP. I consider that NUREG - 0440 is over conservative for the FNP and that on the average the ocean cited FNP would be more than twice as safe as the LBP from risks associated with airborn contamination and much safer from the standpoint of radioactive water pollution.

I do not agree with NUREG-0440 that in case of a Class 9 core melt the LBP would be safer than the FNP. The difference in the two cases in general is that the radioactive contamination of the water would take place almost immediately in the case of the FNP while with the LBP it might take years, decades, or centuries for the peak of the water radioactive pollution to reach the human environment. NUREG - 0440 considers this time factor a plus for the LBP while I am convinced it would be a negative safety factor. From my own experience with accidents, I have observed that the population dose (man.rem) is less when the risk is well defined and comes early and disappears soon rather than in the case where it comes at some indefinite time in the future and lingers over a long period of time. With the FNP Class 9 accident immediate, effective, and heroic measures (such as I summarized in my report following our September 29, 1977 meeting) would be taken to minimize the man.rem dose. After this, full advantage would be taken of dilution and dispersion in the large body of water and a large fraction of the Cs and Sr would settle and be buried in the mud at the bottom of the body of water. With the LBP, however, the risk probably would show up as radioactive contamination in the water supply of a future unsuspecting public. If there is a serious radiation risk, it is better to face it and dispose of it as soon as possible rather than wait to face it in the indefinite future. Ground water contamination from a Class 9 LBP for 100 years at an average dose of only 10 mrem per year to 5×10^6 persons is 5×10^6 man.rem or about $5\times10^6\times3\times10^{-4}$ = 1500 radiation induced malignancies while 5 rem average to 1000 persons during the year following a Class 9 FNP accident is only 5,000 man.rem or only 1 to 2 malignancies. I think it would be difficult to find an offshore FNP site that would present a cancer risk from radiological pollution of the water that would be as great as that from some of our presently sited land based plants.

Since radioisotopes of iodine present one of the major risks in a Class 9 Core melt accident of a LBP or a FNP, it is easy to show that dilution with stable iodine at the source could almost eliminate the risk of radiation induced thyroid carcinoma. One gram of stable iodine in the proper chemical form would reduce the radiation risk of 100,000 Ci of radioiodine during a major reactor accident, by more than a factor of 2. Isotopic dilution would not be as simple in the case of $^{89,90}Sr$ and $^{134,137}Cs$, but it still could be effective. The stable isotopes could be introduced into the sump water at the time of the accident and the basement floor of the barge could be covered with several feet of silicon sand impregnated with KI. The Si would tend to reduce the solubility of the reactor core mix while the KI would reduce the radiation hazard from radioiodine radionuclides in proportion to the reduction in specific activity (Ci/g).

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

APR 3 1978

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

Battelle
Pacific Northwest Laboratories
P. O. Box 999
Richland, Washington 99352
Telephone (509) 942-5011
Telex 32-6345

March 28, 1978

Mr. G. R. Quittschreiber
Senior Staff Engineer
Advisory Committee on Reactor Safeguards
Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Quittschreiber:

This letter documents my impressions of the potential consequences of the liquid pathway following a major accident at a floating nuclear plant on the basis of the ACRS Subcommittee meeting in Washington, D.C. on March 22, 1978, the material presented in NUREG-0440, and other previous reports and presentations by the staff and the applicant.

It is my understanding that a major purpose for undertaking the Liquid Pathway Generic Study (LPGS) was to determine whether the consequences of a major accident at a floating nuclear plant (FNP) would be substantially greater than for a land based plant (LBP) because of the liquid pathway. Implicit in the purpose would seem to be the objective of reaching a decision as to whether the consequences of the liquid pathway for a FNP are sufficiently adverse that some design change is necessary in order to make such plants acceptable from a health and safety aspect.

Since the LPGs was begun several years ago, a great deal of attention by the ACRS Subcommittee, the staff and the applicant has been given to the parameters, choice of assumptions, modeling methodology and comparability of treatment of the FNPs vs. LBPs. In my view, the major flaws that were identified in earlier reports have been eliminated and, although there are still many uncertainties involved, the radiation doses as calculated and presented in NUREG-0440 represent a reasonable basis for comparing the consequences of accidents at FNPs and LBPs via the liquid pathway. I would have preferred to see the summary tables and figures of NUREG-0440 focus on the dose to individuals rather than "man-rem." It is the dose to individuals that would determine the nature, extent and duration of interdiction and thus the socioeconomic costs. Although man-rem provides a simplistic common unit for comparison, it is so muddled with a range of dose rates (promptly lethal to fractions of natural background), population groups (users of the contaminated beach to consumers of the contaminated fish) and of modes of exposure (external -

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ignoring beta, to long retained internal emitters) that it tends to obscure the underlying factors needed for rational decisions. However, if man-rem is to be the basis for judgment then, as a minimum, the decision makers should be provided with a clear picture of the portions of the total dose that are associated with:

- the sump water vs. the molten core
- Cs, Sr and possibly a few other nuclides
- fish consumption, beach exposure, and swimming.

Most (if not all) of this information is contained in NUREG-0440 or in the Applicant's Report T.R.22A60. However, it is not presented in NUREG-0440 in a way that make the relationships stand out. Such relationships are fundamental to considerations of what needs to be contained at the source and which pathways may require the most effective interdiction.

Another feature of NUREG-0440 that clouds the basis for decisions about the liquid pathway is the absence of perspective in relation to the atmospheric pathway. Apparently the rationale is that the population dose consequences of the atmospheric pathways for LBPs and FNPs are about the same; therefore, they can be eliminated from further comparison and attention can be focused just on the liquid pathway. Such a rationale would be alright if the atmospheric pathway consequences were about the same or substantially less than the liquid pathway consequences. On the other hand, if the atmospheric pathway consequences far outweigh those of the liquid pathway, then the worthiness of directing attention just to the liquid pathway is questionable.

The applicant has provided population dose estimates for the air pathway that are on the order of ten fold higher than liquid pathway. The NRC staff also seemed to believe that consequences from the air pathway would be more severe than for the liquid pathway, but apparently have used the values in WASH-1400, which they point out were not derived in the same manner as the dose for the liquid pathway.

At this point it is not at all clear to me whether the worst case air pathway doses and worst case liquid pathway doses each assume virtually all of the available volatile fission products to be released via that one pathway. Obviously this can not be the case. My perception is that there can be releases to the atmosphere without a core melt-through which would initiate the liquid pathway; but, that there will not be a core melt-through to the basin without prior rupture of the containment (an initiation of the air pathway). We need a better perspective of the relative contributions of the liquid vs. the air pathway for the same "worst case" scenario(s). In order to accomplish the most good (in reduced dose to people in the neighborhood) is it better to:

A. minimize the release to the atmosphere?

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B. minimize the loss of sump water?

or C. minimize the loss of radionuclides from the molten core into the sea water?

Another consideration that warrants some attention under liquid pathways is the potential for small "hot particles" being transported away from the accident site by ocean currents and deposited on the beaches. The nature of this potential source makes it difficult to incorporate into generic dose models, but it would be of interest to have some order of magnitude estimates of the dose rate from a particle that is small enough to be transported by the water and deposited on the shore.

Sincerely yours,

R. F. Foster

R. F. Foster
Senior Staff Advisor

Battelle
Pacific Northwest Laboratories
Battelle Boulevard
Richland, Washington 99352
Telephone (509)
Telex 32-6345

March 31, 1978

Mr. G. R. Quittschreiber
U.S. Nuclear Regulatory Commission
Advisory Committee on Reactor Safeguards
Washington, DC 20555

Dear Mr. Quittschreiber:

I want to thank you for the opportunity of participating in the ACRS subcommittee meeting on the Floating Nuclear Plant Liquid Pathway Generic Study in Washington, DC on March 22, 1978. The presentations made by the NRC staff were excellent and the report, NUREG-0440, clearly documents the findings of the study in a comprehensive manner. I have little doubt that the dose to the public via the liquid pathway would be greater for a floating nuclear plant in the event of a core meltdown than for the same event in a land based nuclear plant. However, I believe the NRC staff has taken an overly conservative stance on the issue of leaching of radioactivity from the core melt debris. Additional tests, such as the leaching tests conducted by Sandia, would be helpful in resolving the issue but I doubt that complete resolution would be possible without costly large scale tests and further confirmation of the core meltdown scenarios. In view of the very low probability of a core melt event, the expense of a large scale testing program does not appear warranted at the present time.

I find it difficult to accept the staffs' view that the leaching characteristics of the core melt debris would more closely resemble calcines or poorly formed concretes than crystalline or glassy material. Calcines and concretes are typically very porous which accounts in a large measure for the relatively high leachability of these materials. Calcines are formed with little or no melting of the final product and concretes involve no melting at all. Melting is important with respect to leach rate since it tends to produce a dense material with a low porosity. Although concrete may be involved in the meltdown, it is highly unlikely that the core melt debris would resemble concrete upon contact with water. The formation of concrete requires very finely divided particles (cement) to form the crystalline hydrates which "glue" the particles, sand, and aggregate together. Meltdown debris would not be expected to exhibit the degree of fineness needed to form concrete. A crystalline or glassy material, such as formed in the small scale preliminary Sandia tests, would, in my opinion, be the more likely result of core melt entering or contacting water. The core melt debris may be

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porous and fractured but will have a porosity much less than either calcines or concrete. The 5% leach fraction in one week used by the applicant in computing doses appear more realistic than the staffs' 50% leach fraction in one week.

I believe the Sandia leach test results, although preliminary, are the best data available for estimating leach rates from core melt debris. Tests where the melt is poured into water may produce different results than quenching in a crucible as in the initial Sandia tests. However, I believe a substantial portion, perhaps most, of the core melt will not be dropping into water but will be covered by water rushing into the breach made by the initial melt-through. Quenching the melt in a crucible would more closely simulate this case than pouring the melt in water.

Very truly yours,

Basil W. Mercer

Basil W. Mercer, Manager
Water and Waste Management

BWM:mae

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

April 6, 1978

TO: G. R. Quittschreiber

FROM: Ivan Catton

SUBJECT: FNP SUBCOMMITTEE MEETING ON THE LIQUID PATHWAY GENERIC STUDY, MARCH 22, 1978

Questions raised at the September 22, 1977 FNP Subcommittee meeting on steam explosions have not yet been fully answered. The Staff position on debris leach rates and that of the Applicant are as far apart as ever. In the following paragraphs, I will reiterate the remaining concerns about steam explosions and my view of the leaching rate.

The steam explosion was assumed to take place under the barge. A very conservative estimate of the energy release was made and calculations of the shock impact on the adjacent barge was made. Two questions were raised. First, the pressure wave will reflect off the water surface as a rarefaction wave and, as pointed out by Dr. Plesset, may cause more damage than the pressure wave. Second, the large steam bubble will collapse and cause a water hammer that may result in damage. The collapsing steam bubble will drive a great deal of water into the barge. This may increase the prompt release.

A mechanistic view of the meltdown process leads one to conclude that the steam explosion could take place in the lower bulkhead. In my opinion, the process is as follows:

(1) The molten core melts through the four foot thick concrete pad below the vessel.

(2) After the concrete pad is penetrated, the concrete-fuel-steel mixture falls twelve feet to the lower bulkhead.

(3) The impact of the fuel debris on the lower bulkhead will cause the heat transfer to be high and the penetration rapid.

(4) The water pressure outside the hull will be equivalent to 30 feet of water (depth below the surface) above the pressure inside. Geysering will follow penetration.

(5) Intimate mixing of the fuel debris and water will result because of the water driving pressure.

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(6) The mixing process - water up and fuel debris down - will be highly susceptible to a steam explosion in the lower bulkhead.

(7) A steam explosion in the lower bulkhead could result in enhanced radioactivity being driven out the hull annulus or up through the hole in the pad.

It is my understanding, from conversations with Dr. Speis and Mr. Marchese, that all of the above aspects of the steam explosion are being resolved. I have not seen the resolutions.

The question of what leach rate is proper is still open. The Staff and the Applicant differ by about a factor of twenty. The Applicants arguments are based on their view of how the Canadian leach data should be used and their interpretation of the recent SANDIA leach data. Details of the SANDIA test are not available as yet. The Applicant visited SANDIA to obtain information for his position.

In many respects the SANDIA leach data are non-prototypic. A small (7gm) sample of concrete-fuel in a platinum crucible was rapidly quenched by immersion in water. The slug of material was then put in water and leach rates were determined. The sample was glass-like with only a moderate amount of cracking. My experience has been that significant cracking and fragmentation occurs when molten glass is poured into water. Further, if the pour results in a steam explosion, very fine fragmentation occurs. The SANDIA experiments found a sample surface area of 100 cm^2/gm and 2.5% loss of Strontium in three days. A moderate interaction of the fuel-concrete mixture will probably increase the surface area by a factor of ten. Temperature effects and buoyancy driven circulation could easily lead to another factor of two so that one obtains the Staff result from suitably interpreting the SANDIA data. My conclusion is that the Staff estimate of the leach rate is a best estimate.

Measurements of leach rates are very difficult and the results are quite variable. At the "meeting of experts on leaching", one of the experts indicated that they usually throw away the first few weeks worth of data because it is too unreliable. He also indicated that the same test repeated in the same laboratory can yield results that differ by a factor of 10. A factor of 10 difference exists between the two sets of data from SANDIA and the time period is that usually considered to be too unreliable to be of interest.

As a final note, the second barge must maintain its cooling and as a result circulation in the basin should be a consideration in any methods of mitigation under study.

APPENDIX XVI

NUREG-0440: Major Conclusions

NRC STAFF ASSESSMENT

ON

LIQUID PATHWAY GENERIC STUDY

PRESENTED AT ACRS MEETING

APRIL 6, 1978

SUMMARY OF MAJOR RESULTS

- IMPACTS VIA LIQUID PATHWAY FROM DBA'S SMALL AND SIMILAR FOR FNP AND LBP

- CONSEQUENCES OF CORE-MELT RELEASES, FROM CORE DEBRIS AND SUMP WATER, HIGHER FOR FLOATING PLANTS

- PROMPT SOURCE INTERDICTION AT FNP NOT LIKELY

- PATHWAY INTERDICTION EFFECTIVENESS PROPORTIONAL TO EFFORT APPLIED

17-131

D Walker

GENERIC LIQUID PATHWAYS STUDY

- STUDY INITIATED TO ADDRESS ACRS CONCERNS
- STUDY INTENDED TO PROVIDE BASIS FOR COMPARING FNP AND LBP DOSE CONSEQUENCES VIA LIQUID PATHWAYS
- OPS AND NRC STUDIES CONSIDER RANGE OF RATIONAL SOURCE TERMS BASED ON ENGINEERING JUDGMENT AND USED TO CALCULATE DOSE CONSEQUENCES
- DOSE CONSEQUENCE CALCULATIONS GENERIC RATHER THAN FOR SPECIFIC SITES
- OPS PERFORMED CALCULATIONS FOR FNP'S, NRC FOR LBP'S
- LIQUID PATHWAYS COMPARISON APPEARS IN NRC REPORT, NUREG 0440
- COMPARISON OF LIQUID AND AIR PATHWAYS RISK IN OPS REPORT BUT NOT IN NRC REPORT

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DOSE PATHWAYS CONSIDERED FOR LIQUID PATHWAYS STUDY

- SEAFOOD CONSUMPTION DOSE PATHWAY
- BEACH EXPOSURE PATHWAY
- SWIMMING EXPOSURE PATHWAY
- DRINKING WATER INGESTION DOSE PATHWAY

17-133

OPS CONCLUSIONS - FNP LIQUID PATHWAY DOSE CONSEQUENCE CALCULATIONS

- FISH INGESTION IS DOMINANT DOSE PATHWAY TO MAN FOR OCEAN, BEACH, AND ESTUARINE SITES
- FOR ESTUARINE SITES, DOSE CONSEQUENCES ARE ABOUT FACTOR OF 3 GREATER THAN OCEAN SITES
- FOR RIVERINE SITES, DRINKING WATER IS PRINCIPAL DOSE PATHWAY; DOSE CONSEQUENCES LESS THAN OCEAN SITES
- POPULATION DOSES AND MAXIMUM INDIVIDUAL DOSES DO NOT INCREASE SIGNIFICANTLY AFTER 2 YEARS
- INTERDICTION TO REDUCE LIQUID PATHWAYS DOSE CONSEQUENCE IS FEASIBLE, COULD BE APPLIED IN SUFFICIENT TIME AND WOULD SUBSTANTIALLY REDUCE DOSE CONSEQUENCES
- RISK VIA LIQUID PATHWAY SIGNIFICANTLY LESS THAN VIA AIR PATHWAY CONSIDERING REALISTIC INTERDICTION

PRINCIPAL CHANGE IN OPS PERSPECTIVE SINCE OPS REPORT

- INCREASED POSSIBILITY FOR CONTINUED PUMPING OF SUMP LIQUID AFTER POSTULATED VESSEL MELT THROUGH

17-134

NRC CONCLUSIONS - NUREG 0440

- RISKS FROM LIQUID PATHWAYS RELEASE 25 to 300 TIMES GREATER FOR LBP FOR THE SPECTRUM OF DESIGN BASIS EVENTS.
- RISKS FROM LIQUID PATHWAYS RELEASE 6 TO 30 TIMES GREATER FOR FNP FOR EVENTS BEYOND DESIGN BASIS.
- UNLIKE RELEASE TO AIR PATHWAYS FOR SEVERE ACCIDENTS, LIQUID PATHWAYS RELEASES DO NOT POSE AN IMMEDIATE RISK (ACUTE FATALITIES).
- UNDER REALISTIC (CONTINUED) USE CONDITIONS, SIGNIFICANT DOSES TO INDIVIDUALS ARE NOT EXPECTED VIA LIQUID PATHWAYS.
- LARGE AND RELATIVELY PROMPT RELEASE OF RADIOACTIVITY IS LIKELY FOR FNP BUT NOT FOR LBP.
- FOR FNP, TIMELY AND EFFECTIVE INTERDICTION OF LIQUID PATHWAYS DOSE ROUTES IS FEASIBLE AND LIKELY (WITH ATTENDANT COSTS).
- FOR FNP, ECOSYSTEM IMPACTS VIA LIQUID PATHWAYS ARE IMMEDIATE, TRANSIENT & REVERSIBLE.
- FOR LBP, ECOSYSTEM IMPACTS VIA LIQUID PATHWAYS ARE DELAYED AND MAY BE PREVENTED. IF THEY OCCUR THEY ARE LIKELY TO BE LONG LASTING.

NRC LPGS STUDY

AREAS OF OPS DISAGREEMENT WITH NUREG 0440 PERSPECTIVE

1. HIGH LEACH RATES
2. DOSE PATHWAYS INTERDICTION CUTOFF AT 5 REM
3. LAKE AND SMALL RIVER LAND BASED SITES NOT INCLUDED IN SUMMARY TABLES
4. NO COMPARISON TO AIR PATHWAY

AREAS OF OPS AGREEMENT, NUREG 0440

1. DESIGN BASIS ACCIDENT EVALUATION
2. MODELS FOR LIQUID PATHWAYS DISPERSION & DOSE
3. STEAM EXPLOSION EVALUATION

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RECENT SANDIA LEACH TESTS

TEST SET #1

CORIUM:CONCRETE (WT RATIO)	CATION	AMOUNT LEACHED IN 3 DAYS (%) 25°C	90°C
9:1	Cs	0.075	0.17
	Sr	1.2	2.5
7:3	Cs	0.099	0.25
	Sr	0.68	1.0
5:5	Cs	0.020	0.80
	Sr	--	1.1

TEST SET #2

1) THREE DAY LEACH VALUES SIMILAR TO THOSE FOR TEST SET #1

2) INCREMENTAL LEACH FOR ADDITIONAL 29 DAYS OF LEACH SIMILAR TO THOSE FOR INITIAL 3 DAY LEACH PERIOD

3) MEASURED SURFACE TO MASS RATIO (BET), 100 cm^2/gm

CALCULATED DOSE CONSEQUENCES, CLASS 9 ACCIDENTS

SITE	PLANT TYPE	ESTIMATED LIQUID PATHWAYS CONSEQUENCES, NO INTERDICTION(2) (MAN-REM)	ESTIMATED LIQUID PATHWAYS CONSEQUENCES, INTERDICTION (MAN-REM)
OCEAN	FNP	10^6	10^3
	LBP	10^5	10^2
ESTUARY	FNP	10^7	10^4
	LBP	10^6-10^7	10^3-10^4
RIVER	FNP	10^6	10^3
	LBP	10^5	10^2
LAKE	LBP	10^6-10^7	(1)
SMALL RIVER	LBP	10^6-10^7	(1)

(DOSE CONSEQUENCE BASED ON CASES DESIGNATED AS "EXPECTED" ON EARLIER TABLES)

(1) NO CALCULATED VALUE IN NUREG 0440
(2) BASED ON TABLES IN NUREG 0440

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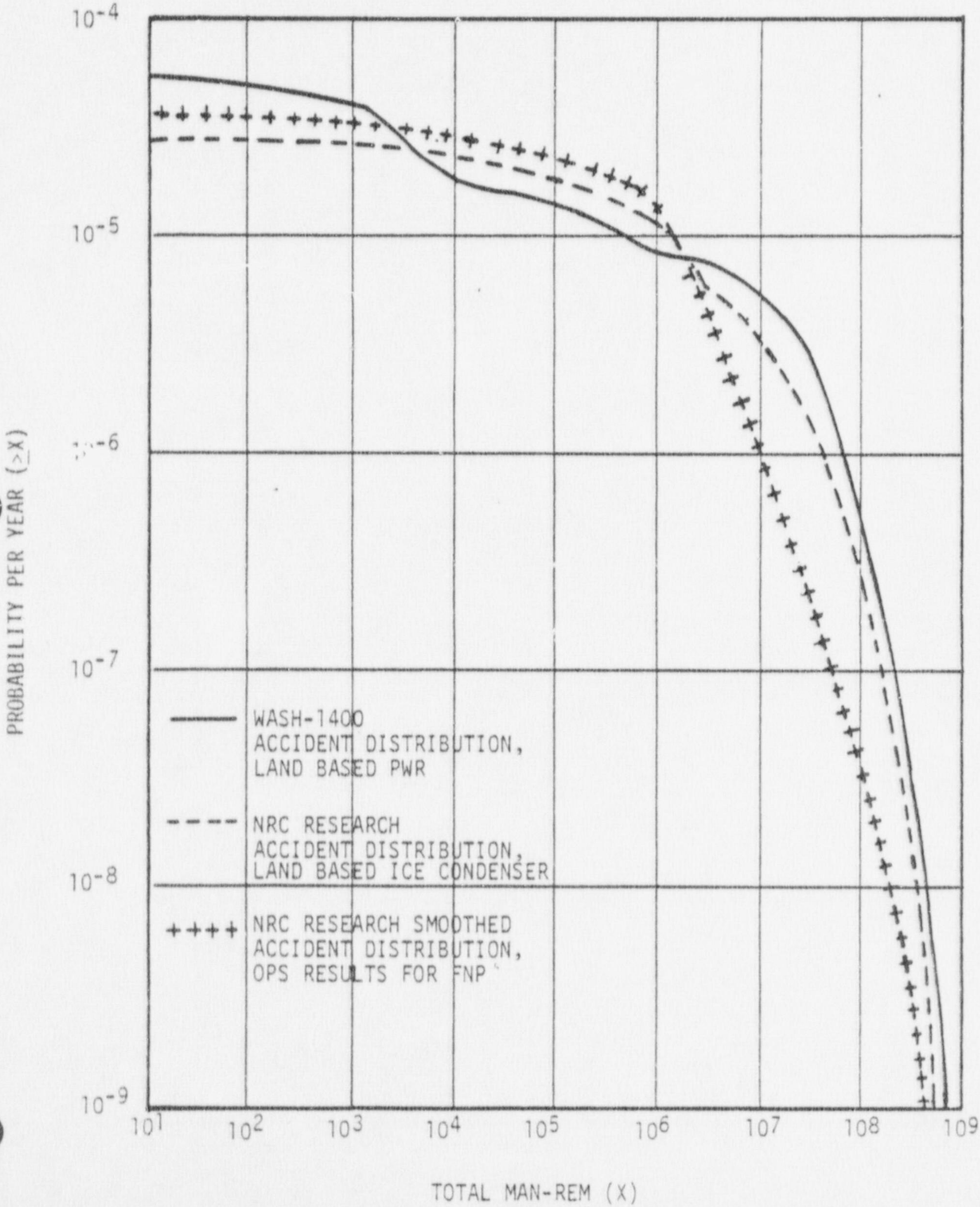
COMPARISON SUMMARY INCLUDING AIR PATHWAY

	MAN REM PER CORE MELT	EFFECTIVE SOURCE INTERDICTION ABILITY	EXPECTED SOCIAL COSTS	BIOTA KILL ZONES	ENVIRON-MENTAL CONTAM-INATION	TIME BEFORE ACTIVITY REACHES PATHWAY	PATHWAY INTER-DICTION ABILITY	CALCULATED ACUTE FATALITIES(1)
STAFF CASE(2) LIQUID PATHWAY								
FNP		NO	HIGH	YES	MAJOR	SHORT	YES	NO
LBP		YES	LOW	NO	MINOR	LONG	YES	NO
APPLICANT CASE(3) LIQUID PATHWAY								
FNP		PARTIAL	MODERATE	NO	MODERATE	SHORT	YES	NO
LBP		YES	LOW	NO	MINOR	LONG	YES	NO
AIR PATHWAY(4)								
FNP		NO	HIGH	PROBABLE	MAJOR	SHORT	NO(4)	YES
LBP		NO	HIGH	PROBABLE	MAJOR	SHORT	NO(4)	YES

(1) ASSUMES NO SOURCE INTERDICTION
(2) ASSUMES HIGH LEACH RATE FOR FNP
(3) ASSUMES 15% TOTAL LEACH, 15% PROMPT RELEASE
(4) NO INTERDICTION FOR DIRECT EXPOSURE OR DIRECT INGESTION; FOOD PATHWAYS INTERDICTION FEASIBLE

AIR PATHWAY DOSE CONSEQUENCES, SEVERE ACCIDENTS (ATLANTIC COAST SITE-ONSHORE)



CALCULATED ACCIDENT RISK FOR AIR PATHWAY

SITE	CASE	MAN REM/YEAR
ATLANTIC COAST SITE, ONSHORE	OPS-FNP	70
	NRC RESEARCH, LBICP	180
	WASH-1400 PWR	270
EASTERN RIVER SITE	WASH-1400 PWR	840
	NRC RESEARCH, LBICP	610
ATLANTIC COAST SITE, OFFSHORE	OPS-FNP	70
	NRC RESEARCH, LBICP	180

LIQUID PATHWAYS - AIR PATHWAYS RISK COMPARISON

MAN-REM/YEAR

		INTERDICTION	NO INTERDICTION
AIR PATHWAY	FNP	75 - 150	100 - 200
	LBP	75 - 600	100 - 800
LIQUID PATHWAY	FNP	10^{-1} to 10^{-2}	50 - 100
	LBP	10^{-2} to 10^{-3}	5 - 70

DOSE COMPARISON FOR MOST LIKELY SENARIO

FNP	LBP	INTERDICTION
10^{6-7} M-R	10^{3-4} M-R	SOURCE PLUS LOW PATHWAY EFFORT
10^{5} M-R	10^{3} M-R	SOURCE PLUS HIGH PATHWAY EFFORT

APPENDIX XVIII
NUREG-0440: Dose Comparisons for Land-Based and Floating Plants.

F7-143

PERSPECTIVE ON MAJOR ISSUES

INTERDICTION

KEY TO THE DIFFERENCES IN IMPACTS LBP VS. FNP

LBP: CONFIDENCE THAT RELEASES CAN BE ISOLATED IN PLANT VICINITY.

FNP: NO ACTION CAN BE TAKEN TO PRECLUDE INITIAL RELEASES (WITHIN ~ 1 WK) FROM ENTERING OPEN WATER BODY. ISOLATION OF LATER RELEASES POSSIBLE.

BOTH: INTERRUPTION OF PATHWAYS TO MAN FOR UNISOLATED RELEASES FEASIBLE.

RELEASES

LARGE QUANTITIES OF ACTIVITY CAN BE RELEASED TO OPEN WATER BODIES BEFORE SOURCE INTERDICTION AT FNP. LBP RELEASES ISOLATED.

PROMPT RELEASE: MORE THAN 10% RELEASED IN ONE WEEK (PROBABLY ~ 80%) ABOUT 100 MILLION CURIES. CONDITIONAL PROBABILITY 0.5.

LEACH RELEASE: LEACH OF SIGNIFICANT FRACTION OF CERTAIN ISOTOPES (Cs & Sr) WITHIN ABOUT ONE WEEK. ABOUT 10 MILLION CURIES

STEAM EXPLOSIONS

LARGE EXPLOSIONS UNLIKELY

EFFECTS OF LARGE EXPLOSIONS

- AIR RELEASES - SMALL EFFECT AT MOST
- DAMAGE TO PLANTS - 2ND UNIT NOT SERIOUSLY DAMAGED
- LIQUID RELEASES - COULD INCREASE BY NO MORE THAN 10X UNDER OPTIMUM CONDITIONS.

COMPARISON SUMMARY: EXPECTED CASE

	LIQUID		AIR	
	FNP	LBP	FNP & LBICP***	PWR
TIMELY SOURCE INTERDICTION ABILITY	NO	YES	NO	NO
PATHWAY INTERDICTION ABILITY	YES	YES	YES	YES
RELEASES TO OPEN ENVIRONMENT	LARGE	SMALL	LARGE	SMALL
TIME BEFORE ACTIVITY REACHES PATHWAY	SHORT	LONG	SHORT	SHORT
MAN REM PER CORE MELT	10^{6-7}* 10^{5}**	10^{3-4}* 10^{3}**	2×10^{6}	4×10^{3}
EXPECTED SOCIAL COSTS	LARGE	SMALL	LARGE	SMALL
BIOTA KILL ZONES	YES	NO	—	—
PATHWAY FATALITIES	NO	NO	NO	NO

---

* SOURCE PLUS MIN. PATHWAY INTERDICTION

** SOURCE PLUS MAX. PATHWAY INTERDICTION

*** SIMILAR RESULTS WILL BE EXPECTED FOR BWRs

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PATHWAY INTERDICTION

- POPULATION DOSE = f(INTERDICTION LEVEL)

- ESTUARY SITE

LEVEL (REM)	POPULATION DOSE REDUCTION FACTOR
5	20
1	100
0.5	200

- OCEAN SITE

LEVEL (REM)	POPULATION DOSE REDUCTION FACTOR
5	10
1	20
0.5	30

AIR PATHWAY COMPARISON

COMPARISONS OF MOST LIKELY SCENARIO	WASH-1400 PWR	ICE CONDENSER*
RELEASE CATEGORY MOST LIKELY TO RESULT	PWR 7	PWR 5
CORE INVENTORY RELEASED	0.6% Xe-kr 2×10^{-3}% I	90% Xe-kr 0.2% I
MAN-REM	4×10^{3}	2×10^{6}
ACUTE FATALITIES	< 1	< 1
COMPARISON MEAN CONSEQUENCES (ALL CATEGORIES)		
MAN-REM	2×10^{6}	8×10^{6}
ACUTE FATALITIES	< 1	< 1

* SIMILAR RESULTS WILL BE EXPECTED FOR BWRs

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IMPORTANCE OF DEBRIS LEACH RATES

SCENARIO	RESULTANT RELEASES	SCENARIO PROBABILITY	AIR PATHWAY[1] EXPECTED CASE	LIQUID PATHWAY[2] MAN-REM SLOW LEACHING	RAPID LEACHING
1. $S_2D-\gamma$ $TML-\gamma$	AIR + DEBRIS + SUMP	36%	2×10^6	2×10^6	3×10^6
2. $S_2H-\gamma$ $S_1H-\gamma$	AIR + DEBRIS	22%	1×10^7	10^5	10^6
3. $S_2HF-\gamma,\delta$	AIR + DEBRIS	17%	3×10^7	10^5	10^6
4. STEAM EXPLOSION IN OR BELOW BULKHEAD	AIR + DEBRIS + SUMP	10%[4]	$10^7 - 10^8$ [3]	10^6	2×10^6

CONCLUSION: DIFFERENCE BETWEEN APPLICANT AND STAFF LEACH ASSUMPTIONS HAVE LITTLE EFFECT ON RISK.

1. Average of several east coast sites, variation less than 50%.
2. Assumes effective source interdiction after one week for FNP and pathway interdiction to 5 rem integrated max individual dose.
3. Estimate.
4. Conservative estimate of likelihood of SE given core melt.

PRINCIPAL FNP CORE MELT SCENARIOS AND RELEASES TO THE ENVIRONMENT

SCENARIO	RESULTANT RELEASES	SCENARIO PROBABILITY	AIR PATHWAY COMPONENT	LIQUID PATHWAY COMPONENT DEBRIS	SUMP
1. $S_2D-\gamma$ $TML-\gamma$	AIR + DEBRIS + SUMP	36%	∿ 100% Xe-kr 1% I 1% Cs ∿ 0% Sr ∿ Ø% Ru	∿ 0% Xe-kr 3% I 5% Cs 89% Sr 92% Ru	∿ 0% Xe-kr 96% I 94% Cs 11% Sr 8% Ru
2. $S_2H-\gamma$ $S_1H-\gamma$	AIR + DEBRIS	22%	∿ 100% Xe-kr 1% I 4% Cs 0.3% Sr 0.9% Ru	∿ 0% Xe-kr 3% I 5% Cs 89% Sr 92% Ru	- 0 - - 0 - - 0 - - 0 - - 0 -
3. $S_2HF-\gamma_1\delta$	AIR + DEBRIS	17%	∿ 100% Xe-kr 6% I 14% Cs 2% Sr 1% Ru	∿ 0% Xe-kr 3% I 5% Cs 89% Sr 92% Ru	- 0 - - 0 - - 0 - - 0 - - 0 -
4. S.E. IN OR BELOW BULKHEAD*	AIR + DEBRIS + SUMP	10%	∿ 100% Xe-kr 6% I 8% Cs 1% Sr 11% Ru	∿ 0% Xe-kr 1% I I 20% Cs 89% Sr 9% Ru	∿ 0% Xe-kr 93% I 72% Cs 10% Sr 80% Ru

* ASSUMES S.E. P = 10%, WATER DF = 10, RELEASES AVERAGE OF SCENARIOS 1-3 INCLUDES 3% VAPORIZATION OF SUMP

COMPARISON OF STEAM EXPLOSION RELEASES
TO MORE PROBABLE RELEASES

	Xe-Kr	I	Cs	Te	Sr	Ru
Average Release from 90% of Core Melts	1.0	.02	.05	.07	.005	.006
Release from SE below RC	1.0	.06	.08	.15	.01	.11
Release from SE in RV	1.0	.27	.67	.40	.08	.43

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DOSE CONSEQUENCE FOR PARTICLE RELEASE

. PARTICLE SIZE - 10 MICRONS

. CORE MASS INVOLVED - 1%

. BIOTA UPTAKE MODEL
  - GI-TRACT TRANSFER (ICRP-2)
  -RETENTION PER NUREG-0440

. CONSEQUENCE TO MAN

ESTUARY

Cs & Sr - $1. \times 10^7$

Particles $5. \times 10^6$

TOTAL $\sim \cancel{1}2. \times 10^7$

OCEAN

Cs & Sr - $3. \times 10^6$

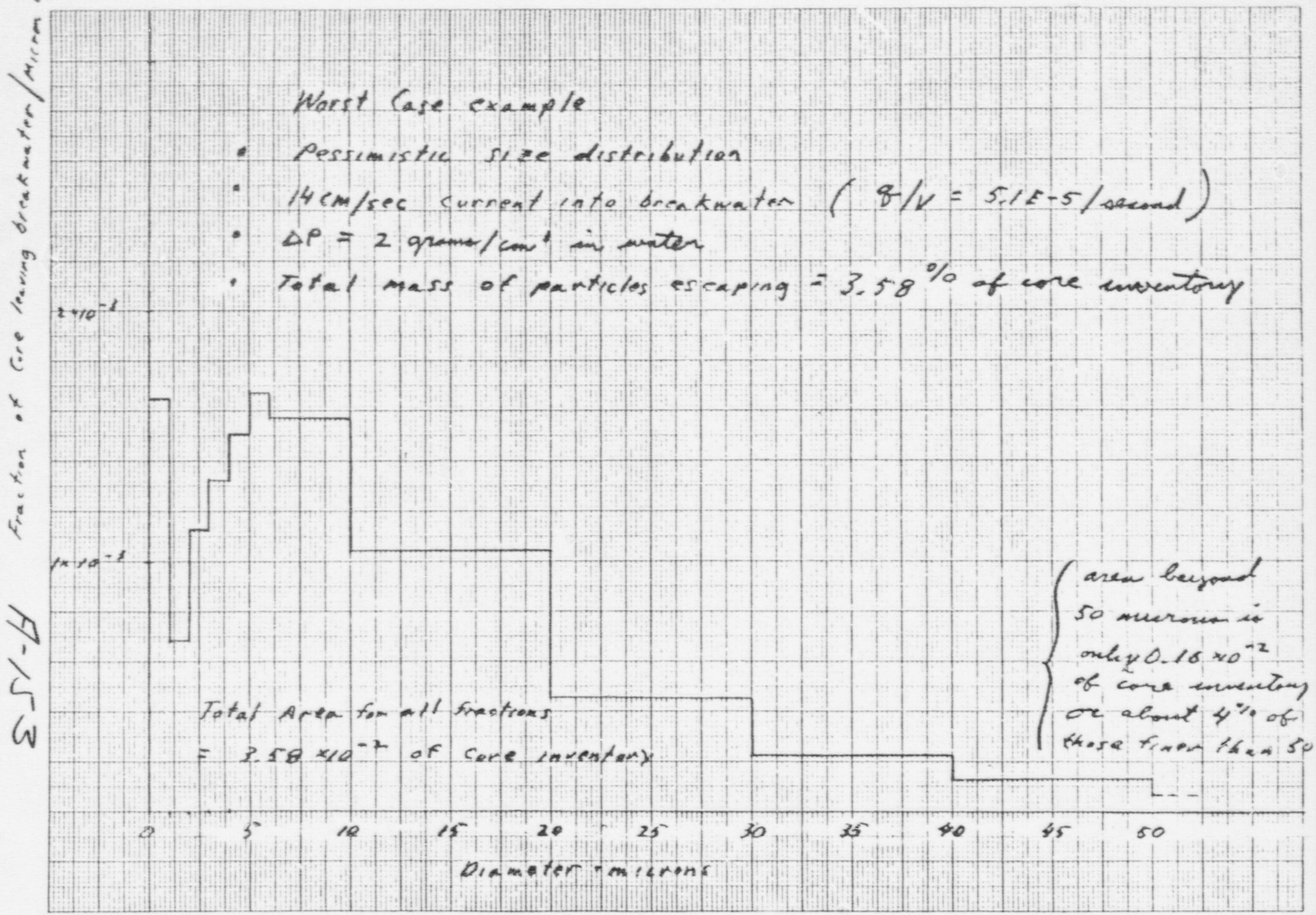
Particles $1. \times 10^6$

TOTAL $4. \times 10^6$

. CONSEQUENCE TO BIOTA
  - GI-TRACT EXPOSURE LIMITING
  - INCREASED ADULT MORTALITY

RIVER-ESTUARY-OCEAN COUPLING

- ESTUARY-OCEAN
  EFFECTS NOT ADDITIVE
- LARGE RIVER-OCEAN
  EFFECTS ADDITIVE
- LARGE RIVER-ESTUARY
  EFFECTS ADDITIVE
- COMPARABILITY OF FNP-LBP NOT CHANGED



Worst Case example
• Pessimistic size distribution
• 14 cm/sec current into breakwater (Q/V = 5.1E-5/second)
• ΔP = 2 grams/cm³ in water
• Total mass of particles escaping = 3.58 % of core inventory
Fraction of core leaving breakwater / Micron diam
2×10⁻¹
1×10⁻¹
Total Area for all fractions
= 3.58 ×10⁻² of core inventory
area beyond
50 microns is
only 0.16 ×10⁻²
of core inventory
or about 4% of
those finer than 50
0
5
10
15
20
25
30
35
40
45
50
Diameter - microns

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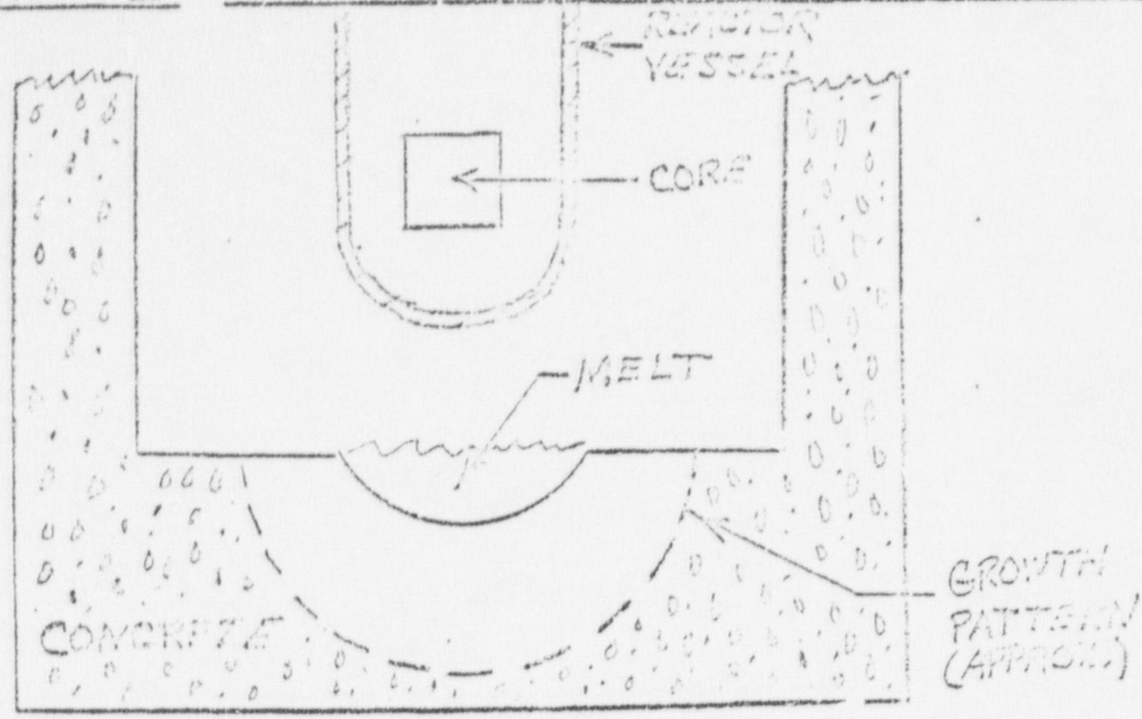
STAFF ASSESSMENT
ON
CORE MELT-THROUGH PENETRATION MODE AND STEAM EXPLOSIONS

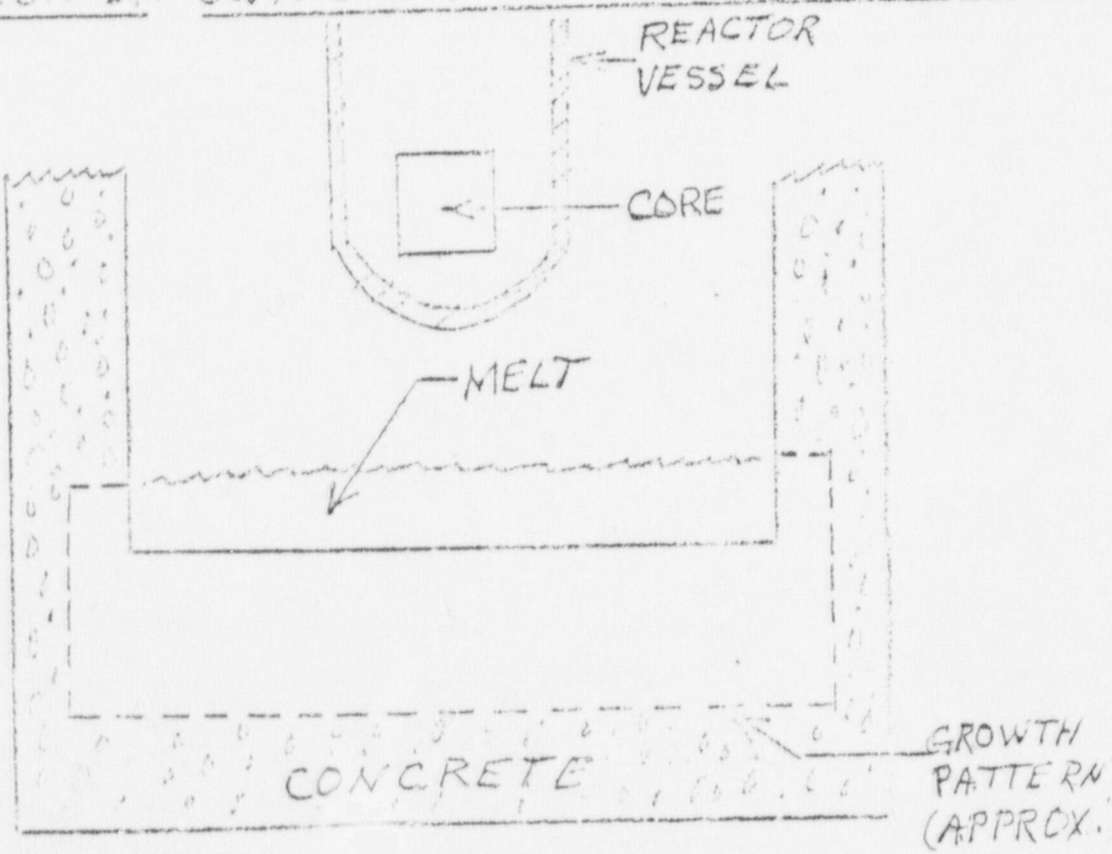
PRESENTED AT ACRS MEETING

APRIL 6, 1978

CASE - 1 : POUR-THROUGH PENETRATION MODE



CASE - 2 : UNIFORM PENETRATION MODE



CORE MELT-THROUGH PENETRATION MODE THROUGH REACTOR VESSEL

- WASH-1400 CORE MELTDOWN SCENARIO ESTIMATES:

  - CASE 1: MOLTEN CORE MATERIAL IS TRANSFERRED (POURED) FROM THE CORE REGION TO THE LOWER HEAD IN A SLOW CONTINUOUS FASHION - THIS TRANSFER (OR POUR) WILL MORE LIKELY LEAD TO THE DEVELOPMENT OF A LOCAL HOLE IN THE REACTOR VESSEL.

  - CASE 2: ON THE OTHER HAND (LESS LIKELY SCENARIO) IF THE MATERIAL IS RAPIDLY COLLECTED IN A LOWER REACTOR VESSEL REGION IN A POOL AND A SLOW MELTING OF THE VESSEL OCCURS, WIDESPREAD FAILURE OF THE LOWER REACTOR VESSEL HEAD IS POSSIBLE.

- STAFF UTILIZED THESE TWO SCENARIOS TO ASSESS THE SUBSEQUENT EVOLUTION OF THE MELTDOWN SCENARIO AND ITS OUTCOME.

CORE MELT-THROUGH PENETRATION MODE THROUGH CONCRETE CONTAINMENT BASEMAT

- DEPENDS ON INITIAL CONDITIONS:

  - CASE 1: INITIAL HEMISPHERICAL MELT GEOMETRY (CAUSED BY MELT BURROWING A HOLE IN CONCRETE FLOOR) WILL CREATE A NEAR HEMISPHERICAL GROWTH PATTERN IN CONCRETE RESULTING IN A GRADUAL POUR-THROUGH PENETRATION MODE.

  - CASE 2: INITIAL CYLINDRICAL MELT GEOMETRY (CAUSED BY MELT SPREADING OVER CONCRETE FLOOR) WILL CREATE A NEAR CYLINDRICAL GROWTH PATTERN IN CONCRETE RESULTING IN A UNIFORM MELT-THROUGH PENETRATION MODE.

CORE MELT-THROUGH PENETRATION MODE THROUGH THE FNP STEEL HULL PLATE

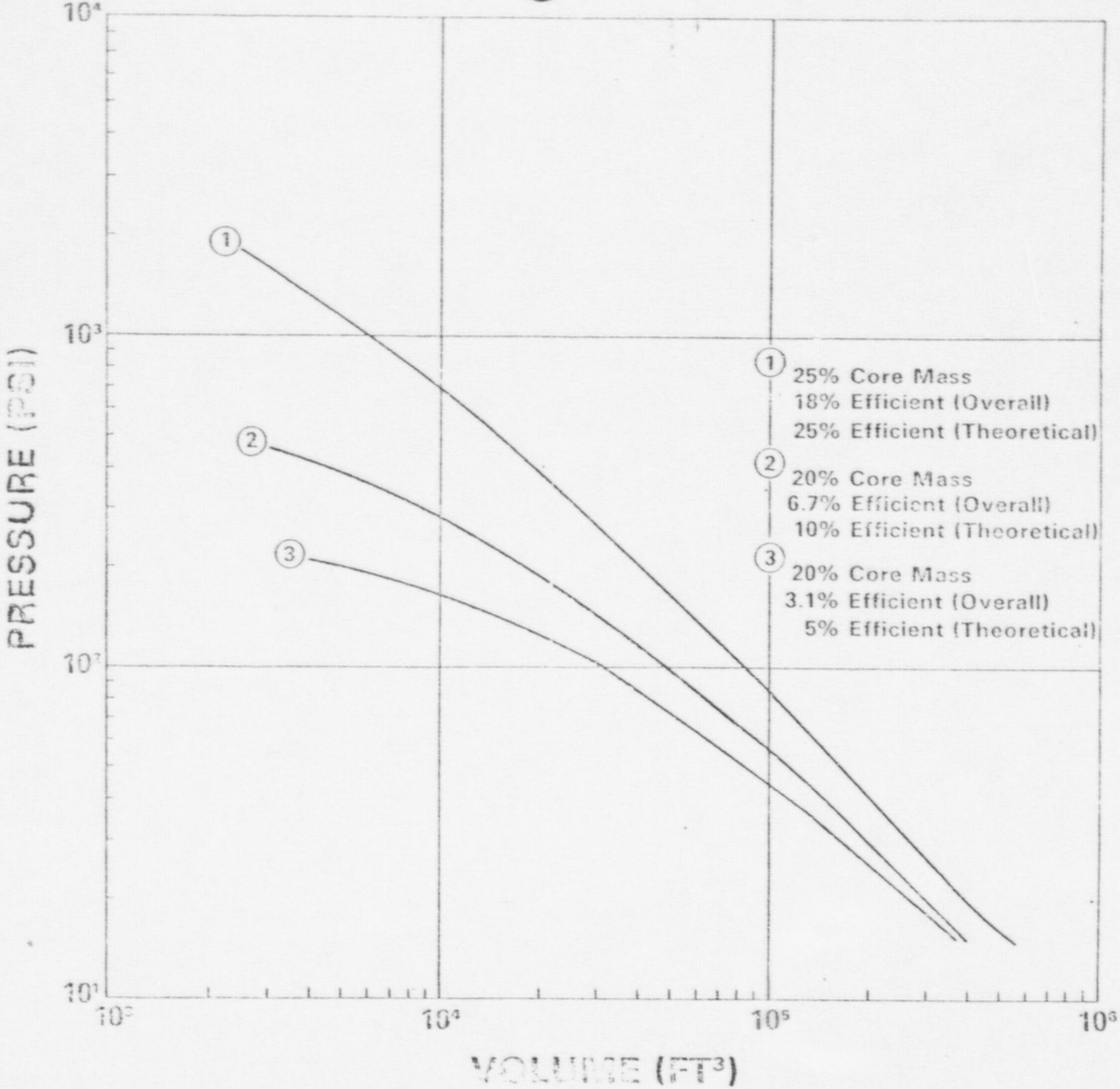
- CASE 1 (POUR-THROUGH PENETRATION MODE OF CONCRETE BASEMAT):
  WILL MOST LIKELY RESULT IN A POUR-THROUGH PENETRATION MODE OF STEEL HULL PLATE

- CASE 2 (UNIFORM PENETRATION MODE OF CONCRETE BASEMAT):
  WILL MOST LIKELY RESULT IN A UNIFORM PENETRATION MODE OF STEEL HULL PLATE

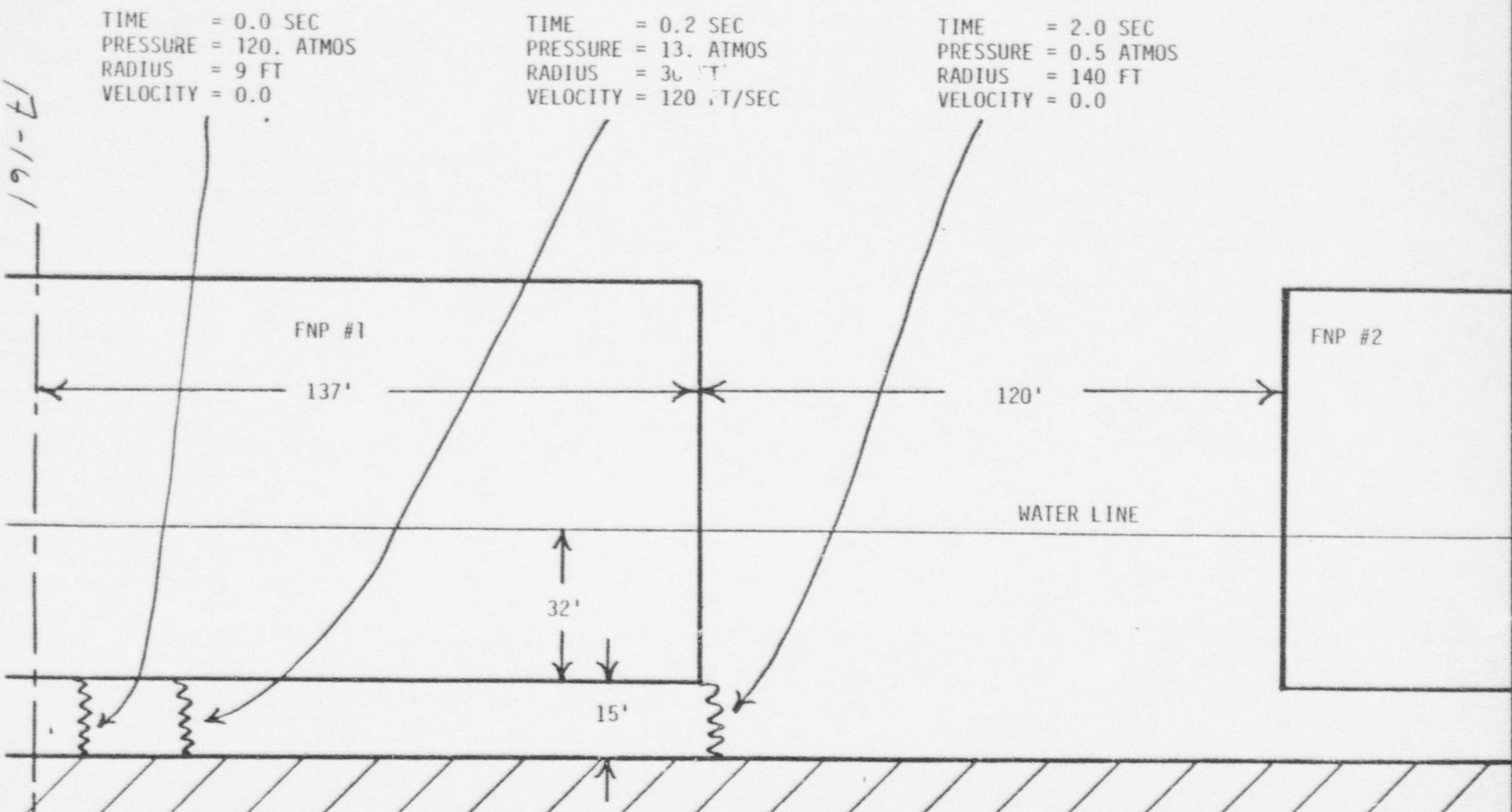
STEAM EXPLOSION (SE) BUBBLE DYNAMICS

- ASSUMPTIONS USED IN THE ANALYSIS
  - NO HEAT LOSSES DURING BUBBLE EXPANSION PHASE
  - NO BUBBLE LEAKAGE INTO BARGE ABOVE SE BUBBLE
  - BARGE BEHAVES AS A RIGID BODY; NO PRESSURE RELIEF
  - BUBBLE EXPANSION IS NON-SPHERICAL (BY VIRTUE OF IMPOSED GEOMETRY)
- OBSERVATIONS
  - RATIO OF $\frac{S}{V}$ SMALL AS BUBBLE RADIUS INCREASES
  - LARGE ENERGY CONTENT IN BUBBLE (BY VIRTUE OF THE CHOSEN INITIAL CONDITIONS)
  - BUBBLE/WATER BOUNDARY VERY HOT

A-160


PRESSURE (PSI)
VOLUME (FT³)
10^4
10^3
10^2
10^1
10^3
10^4
10^5
10^6
1
2
3
1 25% Core Mass
18% Efficient (Overall)
25% Efficient (Theoretical)
2 20% Core Mass
6.7% Efficient (Overall)
10% Efficient (Theoretical)
3 20% Core Mass
3.1% Efficient (Overall)
5% Efficient (Theoretical)

STEAM EXPLOSION BUBBLE EVOLUTION



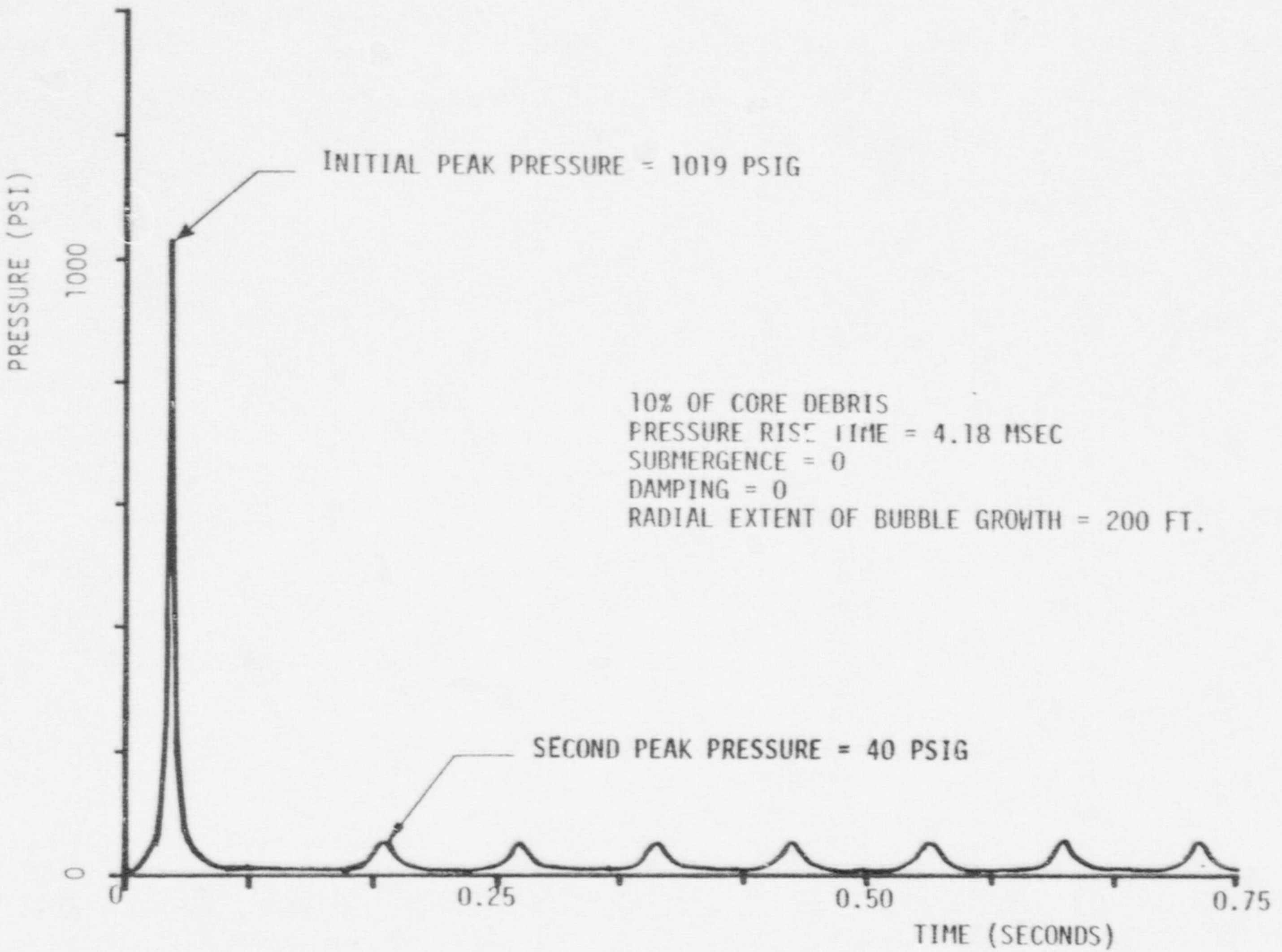
F7-161

SUMMARY OF ANALYSIS TO DETERMINE MAGNITUDE OF FOLLOWING PRESSURE PULSES

AFTER INITIAL STEAM EXPLOSION PRESSURE PULSE

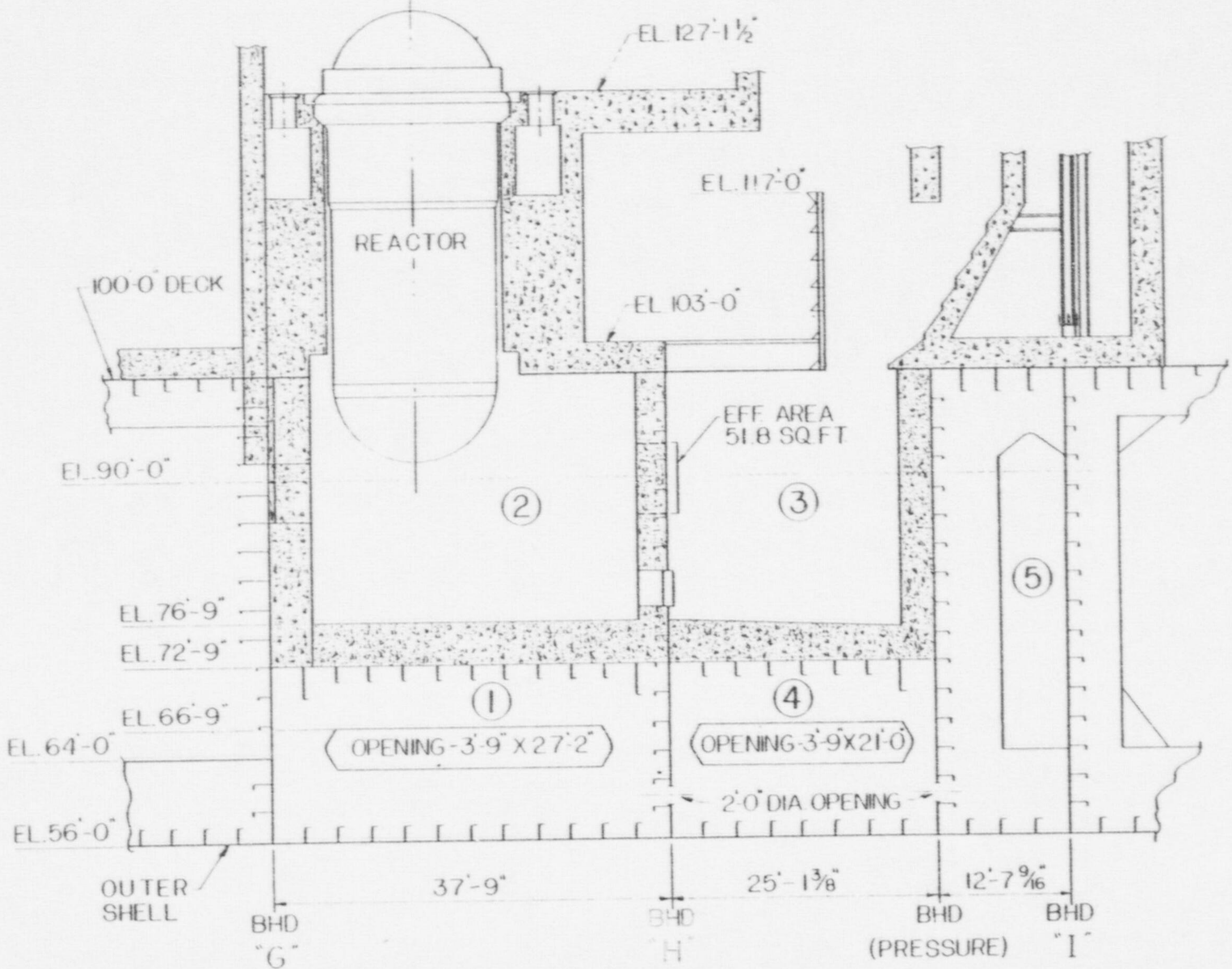
(CALCULATIONS ASSUME 10% OF CORE DEBRIS PARTICIPATES IN INTERACTION)

PRESSURE RISE TIME (MSEC)	RADIAL EXTENT OF BUBBLE GROWTH (FT)	LOCATION OF INTERACTION BELOW PLATFORM (FT)	INITIAL PRESSURE PULSE (PSIG)	SECOND PRESSURE PULSE (PSIG)
4.18	25	0	1019	135
4.18	200	0	1019	40
3.20	200	44	3370	95


INITIAL PEAK PRESSURE = 1019 PSIG
PRESSURE (PSI)
1000
0
10% OF CORE DEBRIS
PRESSURE RISE TIME = 4.18 MSEC
SUBMERGENCE = 0
DAMPING = 0
RADIAL EXTENT OF BUBBLE GROWTH = 200 FT.
SECOND PEAK PRESSURE = 40 PSIG
0
0.25
0.50
0.75
TIME (SECONDS)

F. 6 3

FLOODED COMPARTMENTS AFTER
CORE MELT-THROUGH



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TIME TO DILUTE CONTAMINANTS IN SUMP WATER
IN FLOODED TANKS AFTER HYPOTHETICAL CORE MELT-THROUGH

FRACTION OF SUMP WATER CONTAMINANT RELEASED TO BASIN	DILUTION TIME (HOURS)	
	$C^{(1)} = 0.1$	$C = 0.05$
0.2	900	260
0.5	2900	820
0.8	7500	2200

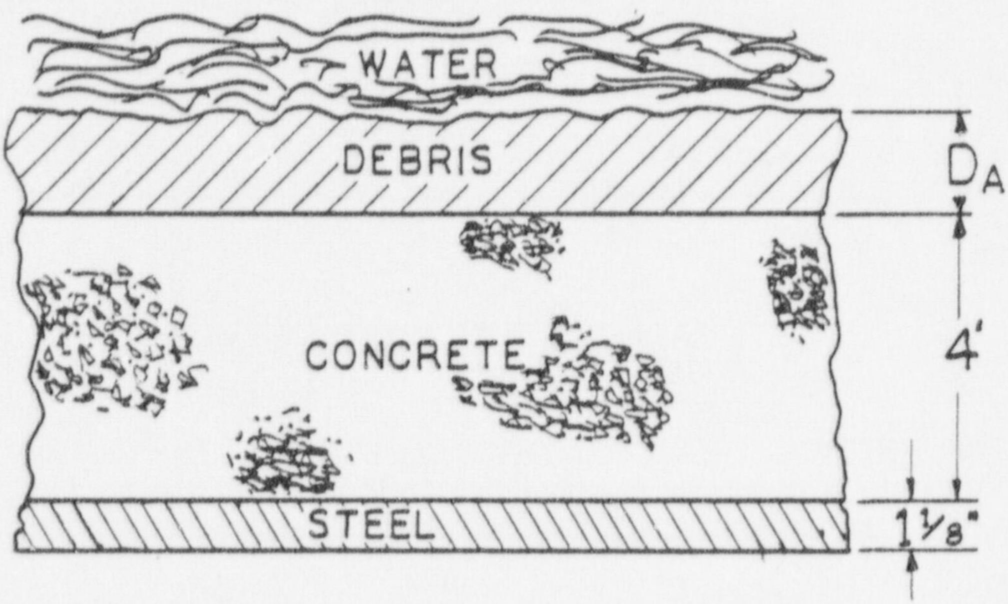
WAVE PERIOD = 8 SECONDS AREA OF HOLE = 20 FT^2

WAVE HEIGHT = 0.7 FEET INFLOW/CYCLE = 7 FT^3

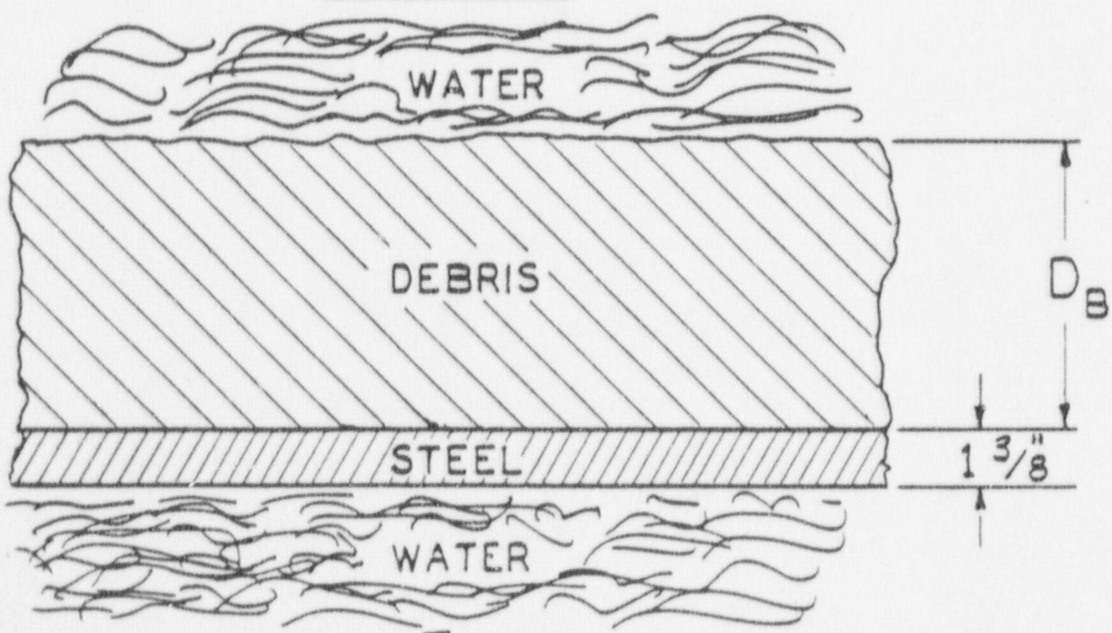
(1) C = CONCENT[illegible]ION OF CONTAMINANTS IN SEA WATER OUTSIDE OF HOLE AS A FRACTION OF CONCENTRATION IN HOLED TANK.

SCHEMATIC DIAGRAMS OF HEAT TRANSFER CALCULATIONS FOR HYPOTHETICAL CORE MELT ACCIDENT

CASE A



CASE B



RESULTS OF HEAT TRANSFER CALCULATIONS
FOLLOWING HYPOTHETICAL CORE MELTDOWN ACCIDENT

	CASE A	CASE B	
D (IN)	$T_M = T_I$ (°C)	T_I (°C)	T_M (°C)
1	240	153	167
2	510	157	212
3	960	160	284
4	1590	164	384
6		172	665
8		179	1056
10		186	1557
13		197	2514

CASE A (REACTOR CAVITY FLOOR):

$K_D = K_{UO2}$ = .005 CAL/SEC-CM-°C

ρ_D = 8.24 GM/CM^3

Q_D''' = .17 CAL/SEC-CM^3

CASE B (HULL PLATE):

K_D = .005 CAL/SEC-CM-C°

ρ_D = 6.57 GM/CM^3

Q_D''' = .085 CAL/SEC-CM^3

(SUBSCRIPT D DENOTES DEBRIS)

TOTAL ESTIMATED LIQUID PATHWAY RESULTS (MAN-REM PER CORE-MELT)

ESTUARY ⟶ OCEAN	FNP	2.4×10^7
	LBP	5.2×10^6
LARGE RIVER ⟶ OCEAN	FNP	6×10^6
	LBP	3×10^5
SMALL RIVER ⟶ ESTUARY ⟶ OCEAN	LBP	7×10^6

PLANT-SITE RANKINGS BASED ON

LIQUID PATHWAY MAN-REM CONSEQUENCES

WITHOUT INTERDICTION

INCREASING MAN-REM CONSEQUENCES ↓

1. LBP - OCEAN
2. LBP - LARGE RIVER
3. FNP - OCEAN
4. LBP - LAKE
5. LBP - ESTUARY
6. FNP - LARGE RIVER
7. LBP - SMALL RIVER
8. FNP - ESTUARY

AIRBORNE RELEASE ESTIMATES FOR STEAM EXPLOSION INSIDE HULL

ASSUMPTIONS

- 3% AND 10% OF DEBRIS MASS PARTICIPATES IN INTERACTION AND IS FINELY FRAGMENTED
- ALL OF STEAM GENERATED IS ASSUMED TO REPRESSURIZE CONTAINMENT
- ACTIVITY IN DEBRIS UNDERGOING INTERACTION ASSUMED TO BE ENTRAINED
- ENTRAINED ACTIVITY REDUCED BY FACTOR OF 10 TO ACCOUNT FOR PLATEOUT (AVAILABLE ACTIVITY)
- FRACTION OF AVAILABLE ACTIVITY RELEASED TO ENVIRONMENT ASSUMED PROPORTIONAL TO FRACTIONAL OVERPRESSURE

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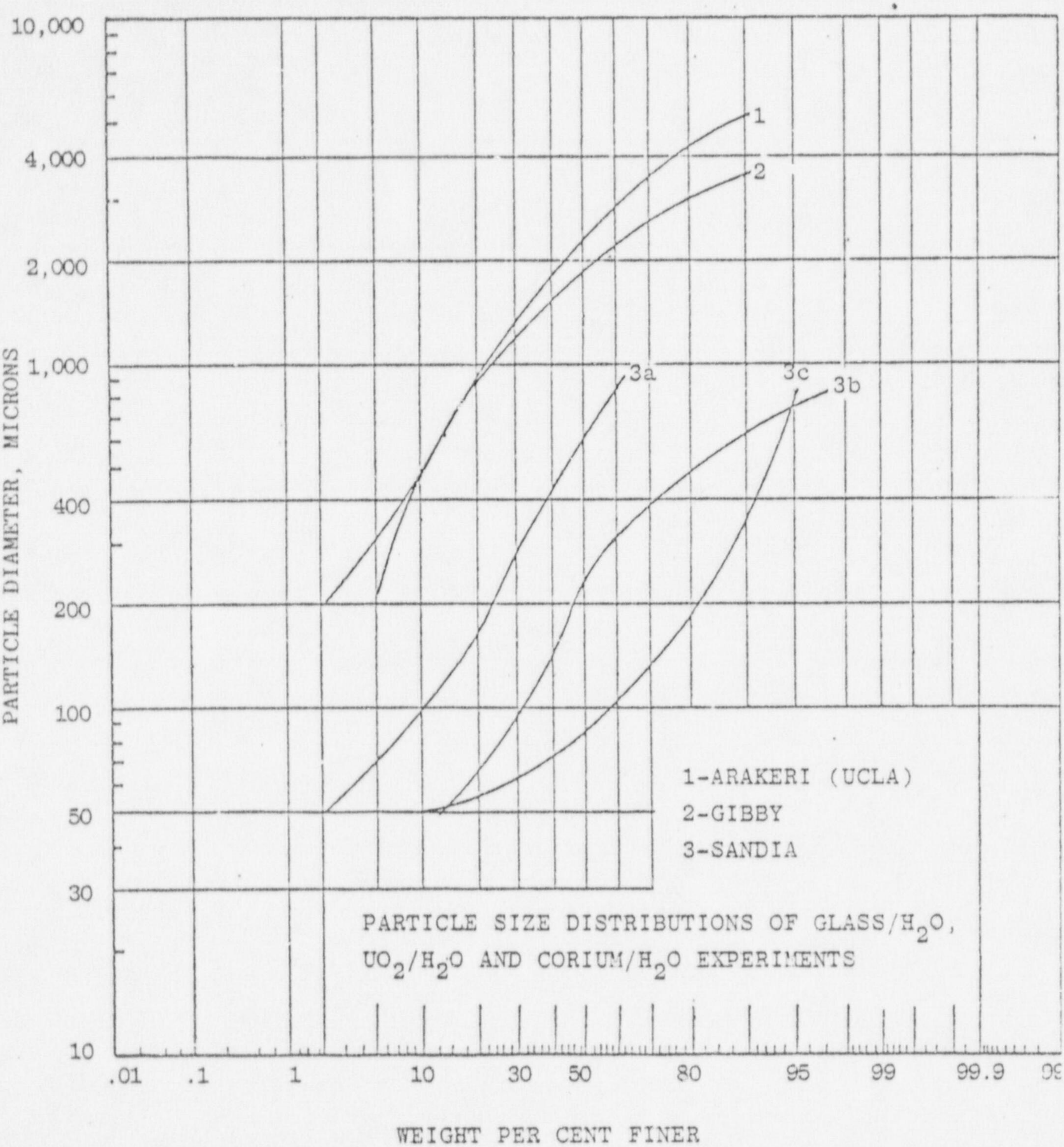
DOSE CONSEQUENCE

DEBRIS MASS PARTICIPATING	REPRESSURIZATION Δ P(PSI)	AVAILABLE ACTIVITY VENTED	ESTIMATED DOSE CONSEQUENCES (MAN-REM)
3%	0.8	6%	10^4
10%	2.6	18%	10^5



PARTICLE SIZE DISTRIBUTIONS OF GLASS/H_2O, UO_2/H_2O AND CORIUM/H_2O EXPERIMENTS

DOSE CONSEQUENCES FROM PARTICLE TRANSPORT

METHODOLOGY

1) ASSUMED 10% OF CORE DEBRIS IS FRAGMENTED WITH A TOTAL OF 10^8 CURIES.

2) ANALYZED PARTICLE SIZES OF 10, 40 AND 100 μ WITH A DENSITY OF 4 G/CM^3 AND UNIFORM ACTIVITY.

RESULTS

1) FNP - OFFSHORE SITE: BEACH POPULATION AND MAXIMUM INDIVIDUAL DOSES ARE EQUAL TO OR LESS THAN WATER SOLUBLE CASE. SEAFOOD INGESTION POPULATION DOSE MAY BE THE SAME ORDER OF MAGNITUDE AND MAXIMUM INDIVIDUAL DOSE MAY BE AN ORDER OF MAGNITUDE HIGHER THAN WATER SOLUBLE CASE.

2) FNP - ONSHORE SITE: MAXIMUM INDIVIDUAL BEACH DOES MAY BE AN ORDER OF MAGNITUDE HIGHER THAN WATER SOLUBLE CASE. OTHER DOSES AS FOR THE OFFSHORE SITE.

STATUS REPORT FOR THE MCGUIRE NUCLEAR STATION

Introduction

The McGuire Nuclear Station is located on the shores of Lake Norman, in Mecklenburg County, North Carolina. The site is approximately 11 miles northwest of the city limits of Charlotte, North Carolina, and is located in a rural area, having no unusual characteristics. The plant is designed for an SSE of .15g and a OBE of .08g.The NSSS and the initial fuel loading will be supplied by the Westinghouse Electric Corporation. The containments will be of ice condensor type and the fuel will be the 17x17 R grid design. McGuire will be the lead plant with UHI. The ice condensor contain ment is similar to that used on Cook Units 1&2 (Cook Unit 1 utilizes the 15x15 grid fuel design. Cook Unit 2 utilizes the 17x17 R grid fuel design). The core design is very similar to that used in the Cook Unit 2 and Trojan. Westinghouse will provide the steam turbines. Tables comparing the design features of McGuire to similar plants and some figures illustrating some features of the design are attached as Attachment 1. The Applicant will act as the architect engineer and construction contractor as per the Applicant's usual practice.

Construction was initiated on June 23, 1971. Unit 1 is now approximately 93% complete, and Unit 2 is approximately 58% complete. Fuel loading for Unit 1 is scheduled for December 1978, with the fuel loading for Unit 2 scheduled for October 1980.

The Staff issued their Safety Evaluation Report ont he McGuire Nuclear Station on March 3, 1978. The outstanding issues which the staff has identified will be discussed in this report. A number of the outstanding issues identified in the Staff's Safety Evaluation Report have since been resolved. The Staff has in addition issued a Safety Evaluation Report on the UHI analytical models and appears to have reached an agreement with the Applicant on the ECCS analyses for McGuire.

Outstanding Issues in the Staff's Safety Evaluation Report

The March Safety Evaluation Report contained a total of 21 outstanding issues, 7 of which are outstanding because the Staff evaluation of these areas has not yet been completed. Six of these outstanding issues have since been resolved and it is likely that others may be resolved in the near future. The current status of the outstanding issues are as follows:

(1) A Staff requirement that the Applicant to submit justifiction for the use of augmented inservice inspection in lieu of complete pipe break protection - The Applicant is seeking exceptions to the Staff criteria (Reg Guide 1.49) for protection against pipe whip and wishes to substitute augmented inservice inspection for the installation of certain pipe restrains. The Applicant has not

formally either identified the pipe locations for which exceptions will be sought or submitted a justification. This information will be submitted by July 1, 1978.

(2) A Staff requirement for the submittal of a summary of the dynamic analysis applicable to Seismic Category I piping which would include the location of all postulated breaks and stress comparison under design load combinations - This is a documentation requirement which the Staff is requiring prior to the completion of their review. The Applicant will submit this information by October 1, 1978.

(3) A Staff requirement for additional information regarding low pressure over-pressurization protection - This issue is now resolved.

(4) A Staff requirement for the installation of leak detection capability for leakage from the reactor coolant system into the research heat removal and safety injection pump systems - The Staff is requiring that the Applicant commit to the installation of suitable leak detection equipment. The Applicant will submit a response by April 7, 1978.

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(5) The Staff is requiring that the Applicant commit to a submit procedures for the leak testing of eight dual bellows penetrations, located outside of the secondary containment, which could constitute potential bypass leakage path - This item is now resolved.

(6) A Staff requirement for a residual heat removal system interlock to prevent pump runout during switch over to the residual spray - In the event that only one containment spray train is operable, ECCS water will be pumped to the auxiliary spray headers. This will be accomplished by isolating the direct injection of the residual heat removal pumps into the cold legs, and diverting this discharge to the safety injection train and the auxiliary spray headers. To preclude potential pump runout at least one of the valves in the cross between the discharges of the residual heat removal pumps must be closed before the residual heat removal sprayline valve can be open. The Staff is requiring that these valves to interlocked. The Applicant will submit a response by April 7, 1978.

(7) A Staff requirement for the evaluation of water hammer potential - The Staff is concerned over potential water hammer in the ECCS (due to void collaspe in the lines) and the steam generator (due to void collaspe in the feedwater preheaters). The Staff is requiring that the Applicant provide additional information which will demonstrate that damaging water hammer would not occur.

(8) A Staff requirement for a means of detecting of post-loss of coolant accident leakage - The Staff will require that the Applicant provide information identifying all possible sources of significant leakage and the means of monitoring this leakage. The Applicant will provide a response by April 7, 1978.

(9) A Staff requirement for the evaluation of ECCS during off design conditions - The Applicant's procedures call for the blocking of the safety injection signal during plant cooldown and closing and locking out power to the cold leg accumulator valves during shutdown operations. The Staff is requiring an analysis of the the ECCS performance for these conditions, and will review the effect of implementing these procedures. The Applicant will provide a response by April 7, 1978

(10) A Staff requirement for additional information on the qualification of Nuclear Steam Supply System Equipment and the balance of planned Class IE equipment - The Applicant has stated that the seismic and environmental qualification of equipment will be in accordance with the WCAP-7744 - 1971 and WCAP-7817 - 1971. The Staff has reviewed these topical reports and has found that some procedures which are specified are not acceptable. The Staff will require that all elements of the Applicant's qualification program be acceptable. Resolution of this issue is expected in the near future.

(11) A Staff requirment for additional information on the offsite power system design - This issue is now resolved.

(12) The Staff is requiring that an alarm be installed on the non-isolation of the unborated water supply during startup or shutdown - The Applicant now uses information from the neutron detectors to indicate the unintentional injection of unborated water into the primary coolant during startup or shutdown. The Staff is requiring a more direct indication of the satisfactory isolation of the unborated water supply. The Applicant will submit a resonse by April 7, 1978.

(13) A Staff requirment for the evaluation of accident trips for new trip items - The Applicant has changed some of the technical specifications on the trip delay times. This issue is now resolved.

(14) The Staff is requiring additional information on the steamline break accident - The Staff is requiring additional documentation which is intended to clarify the Applicant's previous responses on questions on this item. Revision 28 to the FSAR has recently been submitted and should resolve this issue.

The following issues are currently outstanding pending Staff evaluation:

(1) The evaluation of Unit 2 reactor-vessel fracture toughness data - The Staff's review of this item is not yet completed. There appear to be no unique problems in this review.

(2) The evaluation of conformance with Appendix K to 10 CFR Part 50 - These are ECCS/UHI items. The Staff has issued an Safety Evaluation

Report on the ECCS analysis for plants with UHI. It appears that this ECCS analysis for McGuire will show compliance with the Appendix K.

(3) The evaluation of the documentation of the test program results of the electrical penetration qualification - This issue is now resolved.

(4) The evaluation of the Applicants justification for shared electrical power supply system - This item is now resolved.

(5) The evaluation of the Fire Protection Analysis - The Staff Fire protection review has not yet been completed.

(6) The evaluation of the Industrial Security Plan - The Staff's review of the Industrial Security Plan is not yet completed. It is expected that this issue will be rsolved by mid-July.

(7) The evaluation of financial requirements of the Applicant - It is the Staff's practice to not complete this evaluation until time near the end of the review to assure that the most current information has been used in the evaluation. It appears that the Staff's conclusions will be favorable. Completion of this review is expected by mid-May, 1978.

Action on ACRS Generic Matters

The status of the Applicant and the NRC Staff action on the ACRS generic items is as follows: (The items which are marked with a asterisk are recommended for inclusions in the generic paragraph.

Group II:

(1) Turbine Missiles - This items is resolved in that the facility has a peninsular turbine arrangement.

(2) Effective Operation of Containment Sprays in a Loss of Coolant Accident - The McGuire Plant utilizes a ice condensor containment. Sodium Tetraborate has been added to the ice makeup solution to enhance the iodine absolution characteristic of the ice. The technical specifications will require a minimum ice ph of 9.0 whenever the reactor is critical. The containment sprays will use borated water.

*(3) Possible Fracture of the Pressure Vessel Post Loss of Coolant Accident by Thermal S... - This item is not resolved for McGuire and is under generic review by the Staff.

*(4) Instruments to Detect Severe Fuel Failures - The McGuire Station utilizes gamma monitors on a hot leg sampling line. The adequacy of this type of instrumentation to detect failures associated with very rapid events has not yet been established.

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*(5a) Monitoring for Loose Parts Inside the Reactor Vessel - The Applicant has committed to a installation of a loose parts monitoring system.

*(5b) Monitoring for Excessive Vibration Inside the Reactor Pressure Vessel - The Applicant has not committed to either the installation of equipment for neutron noise analysis or the installation of vibration detection equipment on other reactor pressure vessel components, in the event that the usefulness of such devices is established.

*(6) Non Random Multiple Failures - This item is unresolved for this facility.

*(7) Behavior of Reactor Fuel Under Abnormal Conditions - This item is unresolved for this facility.

(8) Boiling Water Reactor Recirculation Pump Over Speed During a Loss of Coolant Accident - This item is not applicable to the McGuire Nuclear Station.

*(9) The Advisability of Seismic Scram - The Applicant has not proposed the use of seismic scram for the McGuire Nuclear Station. The Staff has indicated that they do not intend to require such a scram on McGuire.

(10) Emergency Core Cooling System Capability for Feature Plants - The Staff has indicated that this item is under generic review and is considered to be unresolved for the McGuire Plant. The McGuire design does however utilize the 17x17 R grid fuel and upper head injection.

Group IIA:

*(1) Ice Condensor Containments - The McGuire Plant is the second ice condenser containment station to come before the Committee for a operating license review. (D.C. Cook, Units 1&2 was the first). The Staff has developed some capability for performing a independent analysis of the ice condensor performance but have not used these analysis tools on McGuire. Results to date indicate a favorable comparison with the Westinghouse calculations for D.C. Cook.

*(2) Pressurized Water Pump Overspeed During a Loss of Coolant Accident - The Staff has indicated that this matter is unresolved and is under generic review.

*(3) Steam Generator Tube Leakage - The Staff has indicated that this items is considered to be resolved in part by the requirements for inservice inspection. The steam

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generators used in McGuire were manufactured prior to the implementation of the latest Westinghouse steam generator design fixes.

*(4) ACRS NRC Periodic Ten Year Review of All Power Reactors - This item is unresolved and is under generic review.

Group IIB:

(1) Computer Reactor Protection System - This item is not applicable to the McGuire Station. Devices of this type are not used at the McGurie Station.

*(2) Qualification of New Fuel Geometries - The Westinghouse 17x17 fuel assembly is to be used in the McGuire reactors. A number of tests and surveillance programs are ongoing at this time. The Trojan Reactor a full 17x17 core loading fuel. Commerical operation of Trojan was begun on 5/20/76.

(3) Behavior of Boiling Water Reactor Mark III Containments - This item is not applicable to the McGuire Nuclear Station.

(4) Stress Corrosion Cracking in Boiling Reactor Piping - This item is not applicable to the McGuire Nuclear Power Station.

Group IIC:

*(1) Locking Out of Emergency Core Cooling System Power Operated Valves - The NRC Staff has accepted valve lockout and the

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administrative controls established by the Applicant for the McGuire Station and considers this item to be resolved on this basis. This does not seem to be consistant with the Committee's position on a acceptable resolution for this item.

*(2) Design Features to Control Sabotage - The NRC Staff has not yet completed their evaluation of the applicant's Security Plan. The methods planned for the McGuire Station appear to be consistant with the current state of the art.

*(3a) Decontamination of Reactors - This item is unresolved and is under generic review by the Staff.

*(3b) Decommissioning of Reactors - This items is unresolved and under generic review by the Staff.

(4) Vessel Supports Structures - The load analysis has been performed for this facility using the approved Westinghouse models and the structures have been found to be adequate. The Staff has concurred in this analysis. This items is considered to be resolved for the McGuire Nuclear Station.

*(5) Waterhammer - This item is unresolved and is addressed in item 7 of the Staff's outstanding issues.

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*(6) Maintenance and Inspection of Plants - The NRC Staff considers this item to be resolved for the McGuire Nuclear Station and that the Applicant is in compliance with current NRC requirements. This does not appear to consistant with the Committee position on acceptable resolution for this item.

(7) Behavior of Boiling Water Reactor Mark I Containment - This item is not applicable to the McGuire Nuclear Station.

Group IID:

(1) Safety Related Interfaces Between the Reactor Island and the Balance of Plant - The Staff has indicated that this item is not applicable to the McGuire Nuclear Station since the McGuire Station is a custom design. Duke Power Company acts its own architect engineer and construction contractor and has this advantage in handling interfaces between the NSSS, vendor supplied systems and the balance of plant.

*(2) Assuance of Continuous Long Term Capability of Hermetic Seals on Instrumentation and Electric Equipment - The NRC Staff has indicated that they have not address this item in the Safety Evaluation Report, except as general requirement for environmental qualification of equipment.

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The Subcommittee may wish to discuss this further with the NRC Staff.

Group IIE:

*(1) Soil Structural Interactions - The NRC Staff has indicated that this item is currently being evaluated as part of the Task Action Plan A40 "Seismic Design Criteria". This evaluation is scheduled to be completed by September 1978 and could lead to the modification of current criteria for seismic input and soil structural interaction. The foundations for most of the major structures at McGuire are on sound rock.

Intervenors/Significant Differences of Opinion Among the NRC Staff

The Carolina Environmental Study Group has raised a concern regarding the probability of a major earthquake in the eastern part of North Carolina, and has cited anomalous changes in land elevation and groundwater behavior as possible predictors of such a earthquake. David Stewart, David Dunn, and S. Duncan Heron has raised this issue on the Bunswick Docket and has reported that such anomalous conditions exist in the vicinity of Southport, North Carolina The Carolina Power and Light Company has been operating a micro seismic network in the Southport area for approximately 1-1/2 years and have identified no local earthquakes. The historical record for this area indicates a very low level of seismicity. The McGuire site is approximately 200 miles from the area of the postulated anomalities and is in a different tectonic province.

The Carolina and Environmental Study Group had submitted a written statement to the ACRS on March 6, 1977, (J. Riley to ACRS, see Attachment 2) during the Committee's review of the Perkins/Cherokee application. This material as distribued to the Committee during its review of the Cherokee/Perkins application. The Committee appeared to be satisfied that these issues had been properly addressed by the Applicant and the NRC Staff. Mr. Riley raised these same issues at the McGuire Subcommittee meeting.

The NRC Project Manager has informed me that there are no significant dissent ing technical views remaining within the NRC Staff on McGuire. The questions as to the testing of the large circuit breakers are believed to have been re solved. Robert Pollard was formally the NRC Project Engineer on the McGuire Plant. The NRC Project Manager does not know of any specific dissenting technical views which were expressed by Mr. Pollard on the McGuire Plant.

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DESIGN PARAMETER COMPARISON

DESIGN PARAMETERS	W. B. MCGUIRE 1 AND 2	D. C. COOK 2	TROJAN
NSSS POWER LEVEL, MWT	3425	3403	3423
REACTOR COOLANT PRESSURE, PSIA	2250	2250	2250
REACTOR COOLANT FLOW RATE, 10^6 LB/HR	140.3	134.6	132.6
REACTOR COOLANT TEMPERATURE, $^{\circ}F$			
VESSEL INLET	558.1	541.3	552.[illegible]
VESSEL AVERAGE	588.2	573.8	584.[illegible]
VESSEL OUTLET	618.2	606.4	616.[illegible]
STEAM GENERATOR			
STEAM TEMPERATURE, $^{\circ}F$	544.6	521.1	533.[illegible]
STEAM PRESSURE, PSIA	1000	820	910
STEAM FLOW, 10^6 LB/HR	15.14	14.74	15.07

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DESIGN PARAMETER COMPARISON

DESIGN PARAMETERS	W. B. MCGUIRE 1 AND 2	D. C. COOK 2	TROJAN
MINIMUM DNBR AT NOMINAL CONDITIONS			
TYPICAL FLOW CHANNEL	2.08	3.03	2.04
THIMBLE (COLD WALL) FLOW CHANNEL	1.74	2.70	1.71
MINIMUM DNBR FOR DESIGN TRANSIENTS	≥ 1.30	≥ 1.77	≥ 1.30
DNB CORRELATION	"R" (W-3 WITH MODIFIED SPACE FACTOR)	WRB-1	"R" (W- WITH MO- FIED SP FACTOR)
AVERAGE THERMAL OUTPUT, KW/FT	5.44	5.41	5.44
MAXIMUM LICENSED THERMAL OUTPUT FOR NORMAL OPERATION, KW/FT	12.6	11.8	12.6
LIMITING F_Q VALUE	2.32	2.18	2.32

DESIGN COMPARISON

BASIC COMPONENT	W. B. MCGUIRE 1 AND 2	D. C. COOK 2	TROJAN 1
REACTOR VESSEL, ID, IN	173	173	173
CORE			
NUMBER OF ASSEMBLIES	193	193	193
ROD ARRAY	17 x 17	17 x 17	17 x 17
ROD OD, IN	0.374	0.374	0.374
NUMBER OF GRIDS	8 R TYPE	8 R TYPE	8 R TYPE
ACTIVE FUEL LENGTH, IN	144	144	144
NUMBER OF CONTROL RODS			
FULL LENGTH	53	53	53
PART LENGTH	8	8	8
STEAM GENERATOR			
TYPE	PREHEAT	FEEDRING	FEEDRING
SHELL DESIGN PRESSURE, PSIA	1200	1100	1100
REACTOR COOLANT PUMP			
TYPE	93A	93A	93A
MOTOR HORSE POWER	7000	6000	6000

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SYSTEM AND COMPONENT DESIGN COMPARISON

System/Component	Similar Designs	Principle Differences
Fuel	Trojan, Cook 2 and Sequoyah	None
Reactor Vessel Internals	Sequoyah and Trojan	McGuire and Sequoyah will have Upper Head Injection Systems; McGuire and Trojan will have neutron pads; McGuire and Sequoyah will be modified for increased inlet flow bypass to the upper head.
Reactivity Control	Cook 2, Trojan and Sequoyah	None
Nuclear Design	Cook 2, Trojan and Sequoyah	None
Thermal-Hydraulic Design	Cook 2, Trojan and Sequoyah	Cook 2 has used the Westinghouse improved thermal design procedure
Reactor Coolant System	Cook 2, Trojan and Sequoyah	None
Reactor Vessel	Cook 2, Trojan and Sequoyah	McGuire and Sequoyah vessel heads incorporate upper head penetrations.

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SYSTEM/COMPONENT	SIMILAR DESIGNS	PRINCIPLE DIFFERENCES
REACTOR COOLANT PUMP	COOK, TROJAN AND SEQUOYAH	MCGUIRE HAS HIGHER COOLANT FLOW DUE TO CHANGES TO IMPELLER, DIFFUSER AND INCREASED MOTOR HORSEPOWER
STEAM GENERATORS	COOK, TROJAN AND SEQUOYAH	MCGUIRE WILL HAVE PREHEAT STEAM GENERATORS
RESIDUAL HEAT REMOVAL SYSTEM	COOK, TROJAN AND SEQUOYAH	NONE
PRESSURIZER	COOK, TROJAN AND SEQUOYAH	NONE
CONTAINMENT SYSTEMS	COOK AND SEQUOYAH	MCGUIRE, COOK AND SEQUOYAH HAVE ICE CONDENSER SYSTEMS
EMERGENCY CORE COOLING SYSTEM	COOK, TROJAN AND SEQUOYAH	MCGUIRE AND SEQUOYAH HAVE UPPER HEAD INJECTION SYSTEMS
REACTOR PROTECTION SYSTEM	FUNCTIONS ARE SIMILAR TO COOK 2, TROJAN AND SEQUOYAH	NONE

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SYSTEM/COMPONENT	SIMILAR DESIGNS	PRINCIPLE DIFFERENCES
ENGINEERED SAFETY FEATURE ACTUATION SYSTEMS	FUNCTIONS ARE SIMILAR TO COOK 2, TROJAN AND SEQUOYAH	NONE
ELECTRICAL SYSTEMS REQUIRED FOR SAFE SHUTDOWN	FUNCTIONS ARE SIMILAR TO COOK 2, TROJAN AND SEQUOYAH	NONE
SAFETY RELATED DISPLAY INSTRUMENTATION	FUNCTIONS SIMILAR TO COOK 2, TROJAN AND SEQUOYAH	ACTUAL CONFIGURATION MAY DIFFER
CHEMICAL AND VOLUME CONTROL SYSTEM	COOK 2, TROJAN AND SEQUOYAH	MCGUIRE WILL HAVE BORON THERMAL REGENERATION

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MAJOR DESIGN CHANGES SINCE PSAR

CHANGE IN DESIGN	REASON
FUEL ASSEMBLY DESIGN CHANGED FROM 15 x 15 TO 17 x 17 ARRAY	IMPROVED CORE THERMAL MARGIN
UPPER INTERNALS MODIFICATION	ACCOMMODATE 17 x 17 AND UPPER HEAD INJECTION
THERMAL SHIELD HAS BEEN REPLACED BY NEUTRON PADS	SIMPLIFIED CORE SUPPORT DESIGN AND REDUCED PRESSURE DROP AND VELOCITY
ICE BASKETS, SUPPORTS AND LATTICE FRAMES FOR THE ICE CONDENSER HAVE BEEN REDESIGNED	ACCOMMODATE DESIGN LOADING CONDITIONS BEYOND THOSE COMMITTED IN THE PSAR
UPPER HEAD INJECTION SYSTEM HAS BEEN ADDED	IMPROVED PERFORMANCE OF THE EMERGENCY CORE COOLING SYSTEM
STEAM GENERATOR CHEMISTRY CHANGED FROM PHOSPHATE TO VOLATILE AMINE TREATMENT	PROBLEMS ASSOCIATED WITH PHOSPHATE CHEMISTRY
SEMI-AUTOMATIC SWITCHOVER TO ECCS RECIRCULATION HAS BEEN PROVIDED AS BACKUP TO OPERATOR ACTION	MEET NRC REQUIREMENT TO REDUCE RELIANCE ON OPERATOR ACTION

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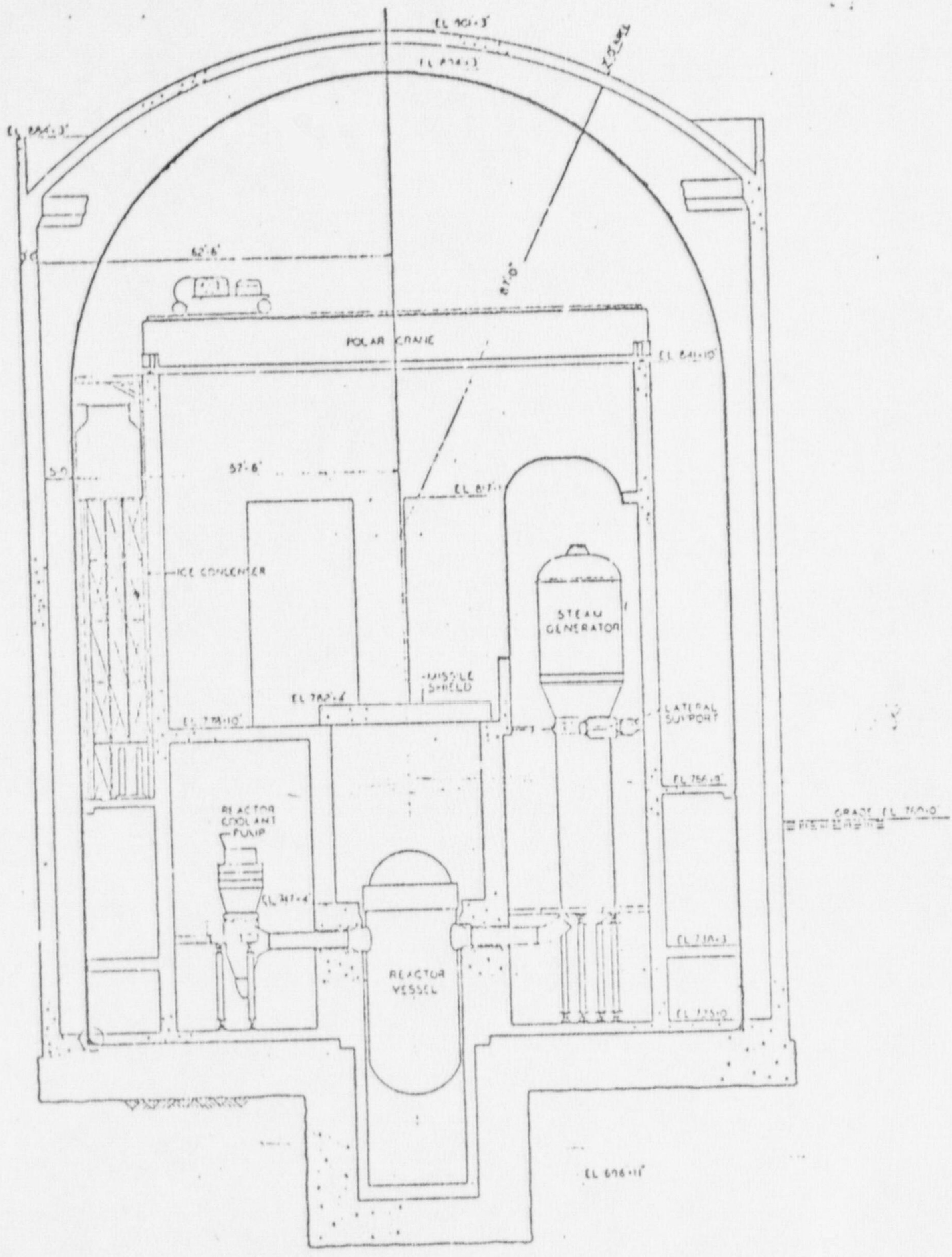
17 x 17 FUEL SURVEILLANCE PROGRAM

SURRY DEMONSTRATION PROGRAM

- (2) 7-GRID 17 x 17 ASSEMBLIES IRRADIATED IN EACH OF THE TWO SURRY UNITS
- FIRST AND SECOND CYCLE EXAMINATIONS COMPLETED FOR BOTH UNITS INCLUDES TELEVISION, ROD BOW MEASUREMENTS, PROFILOMETRY, GAMMA SCAN, EDDY CURRENT, ETC.
- SURRY UNIT 1 DEMO ASSEMBLIES DISCHARGED
- SURRY UNIT 2 DEMO ASSEMBLIES NOW UNDERGOING THIRD CYCLE OF IRRADIATION (EXPECTED DISCHARGE DATE: MARCH, 1979)

17 x 17 FULL CORE EXAMINATIONS

- VISUAL EXAMINATION OF FIRST TWO 17 x 17 FULL CORE REACTORS TO RELOAD FUEL; I.E., TROJAN AND BEAVER VALLEY UNIT 1
- EXAMINATIONS TO INCLUDE OBSERVATIONS FOR CLADDING DEFECTS, ROD BOWING, CORROSION, CRUD DEPOSITION, AND GEOMETRIC DISTORTION



REACTOR BUILDING AND CONTAINMENT STRUCTURAL OUTLINE

MCGUIRE NUCLEAR STATION

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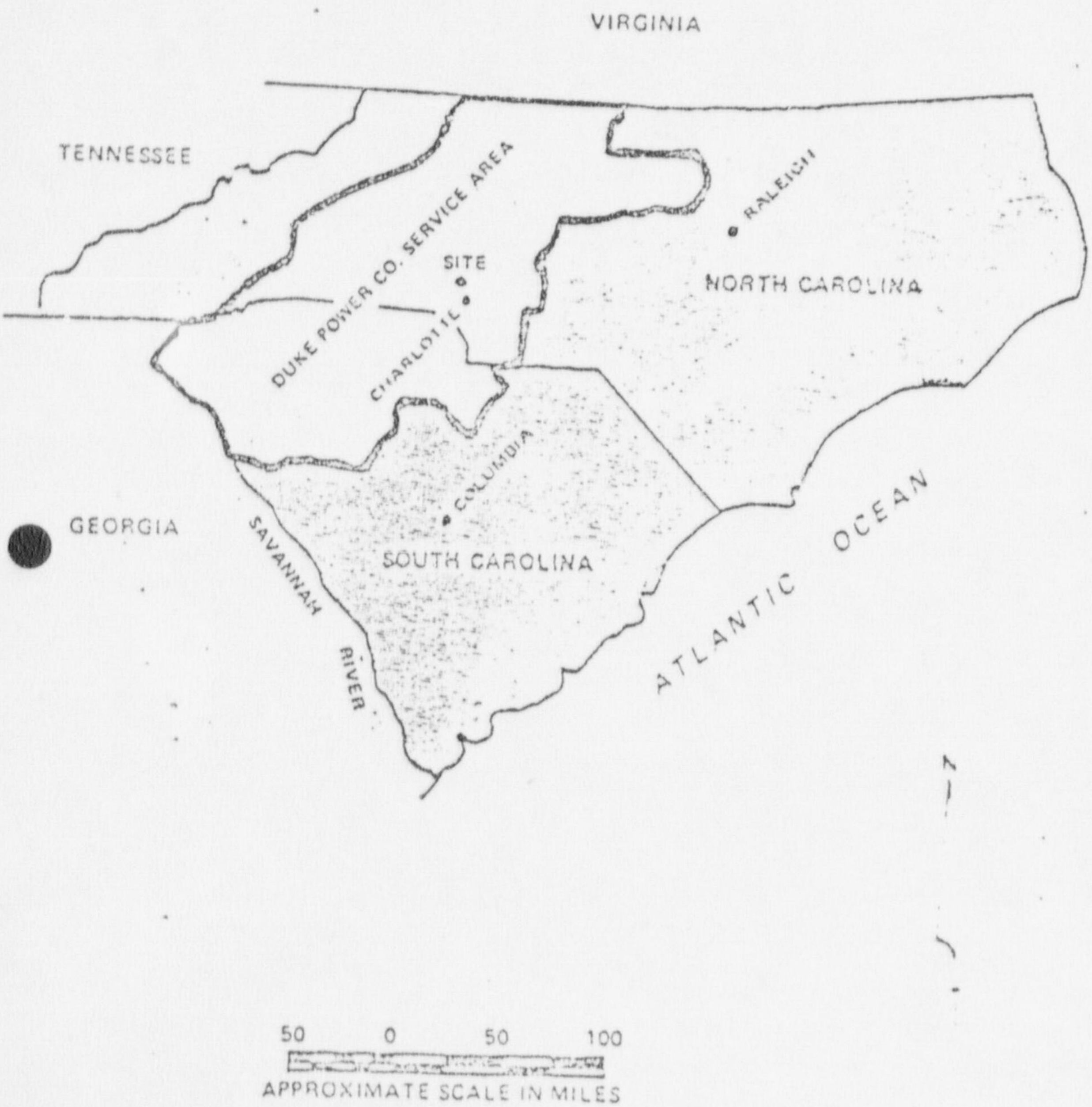
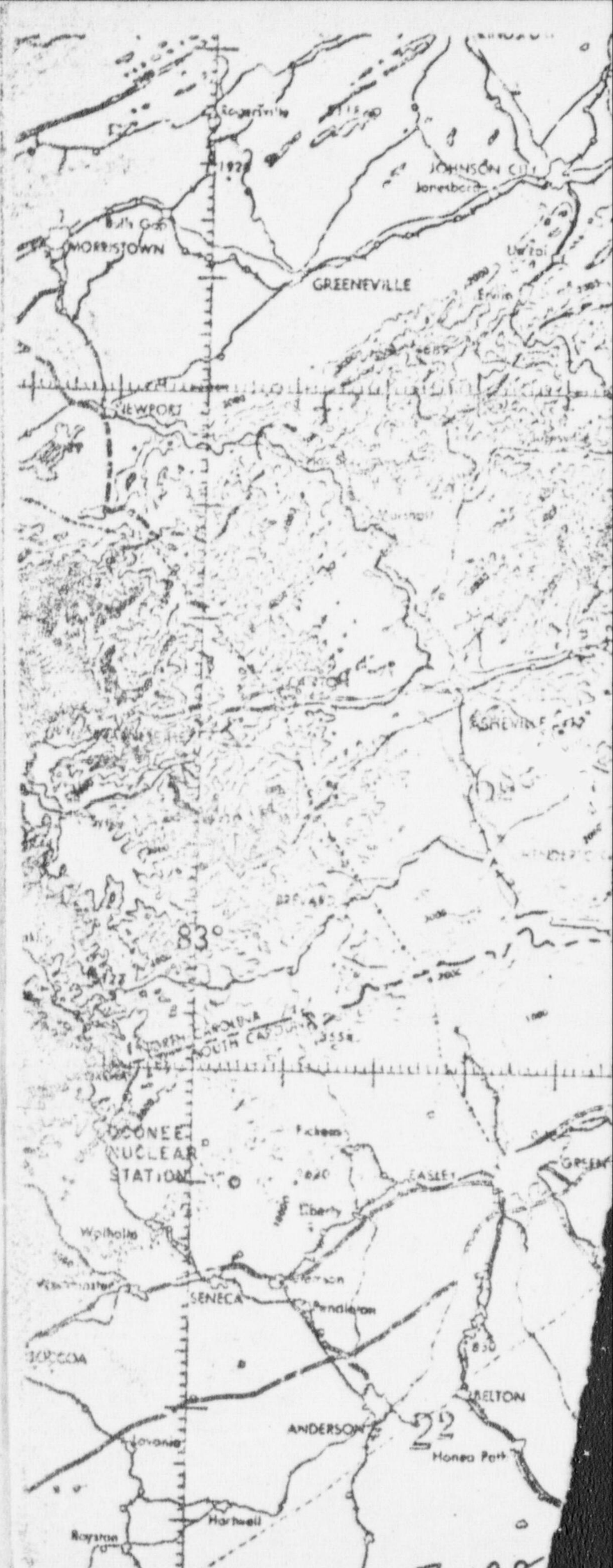


DUKE POWER COMPANY SERVICE AREA

McGUIRE NUCLEAR STATION


Rogersville
Jonesboro
Bull's Gap
MORRISTOWN
GREENEVILLE
Erwin
NEWPORT
ASHEVILLE
83°
NORTH CAROLINA
SOUTH CAROLINA
OCONEE
NUCLEAR
STATION
Pickens
EASLEY
Liberty
Walhalla
Westminster
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Honea Path
Lavonia
Hartwell
Royston

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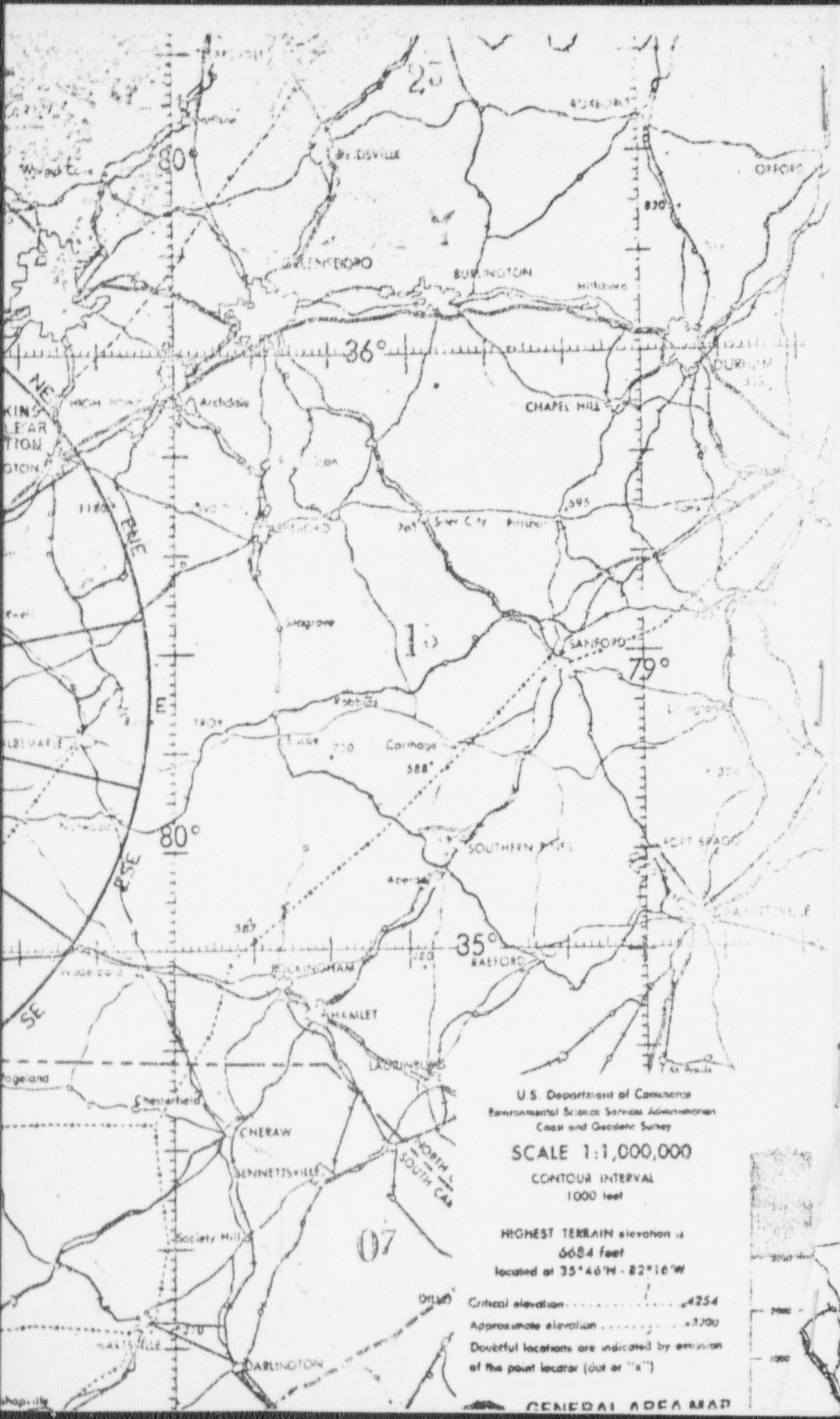
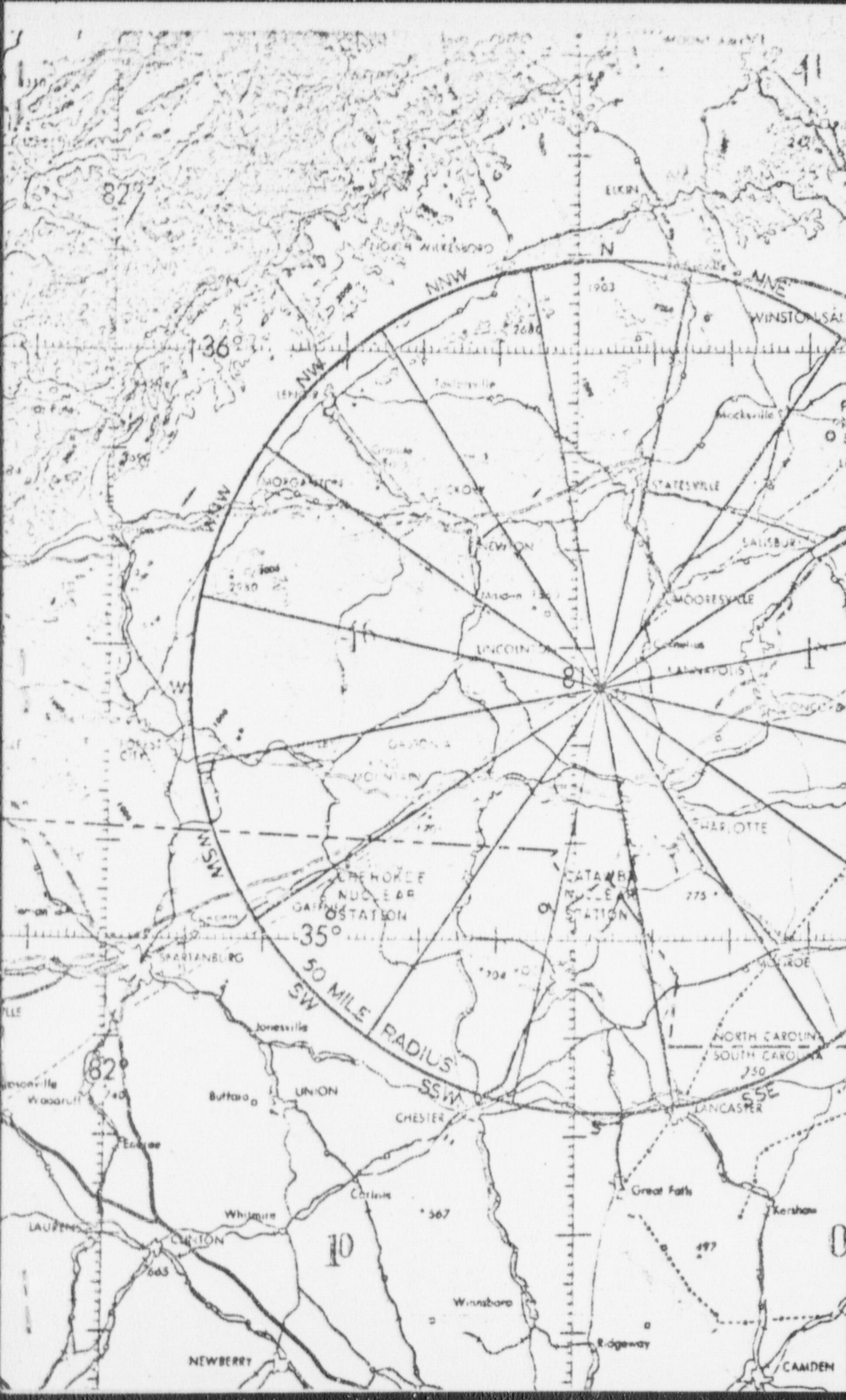
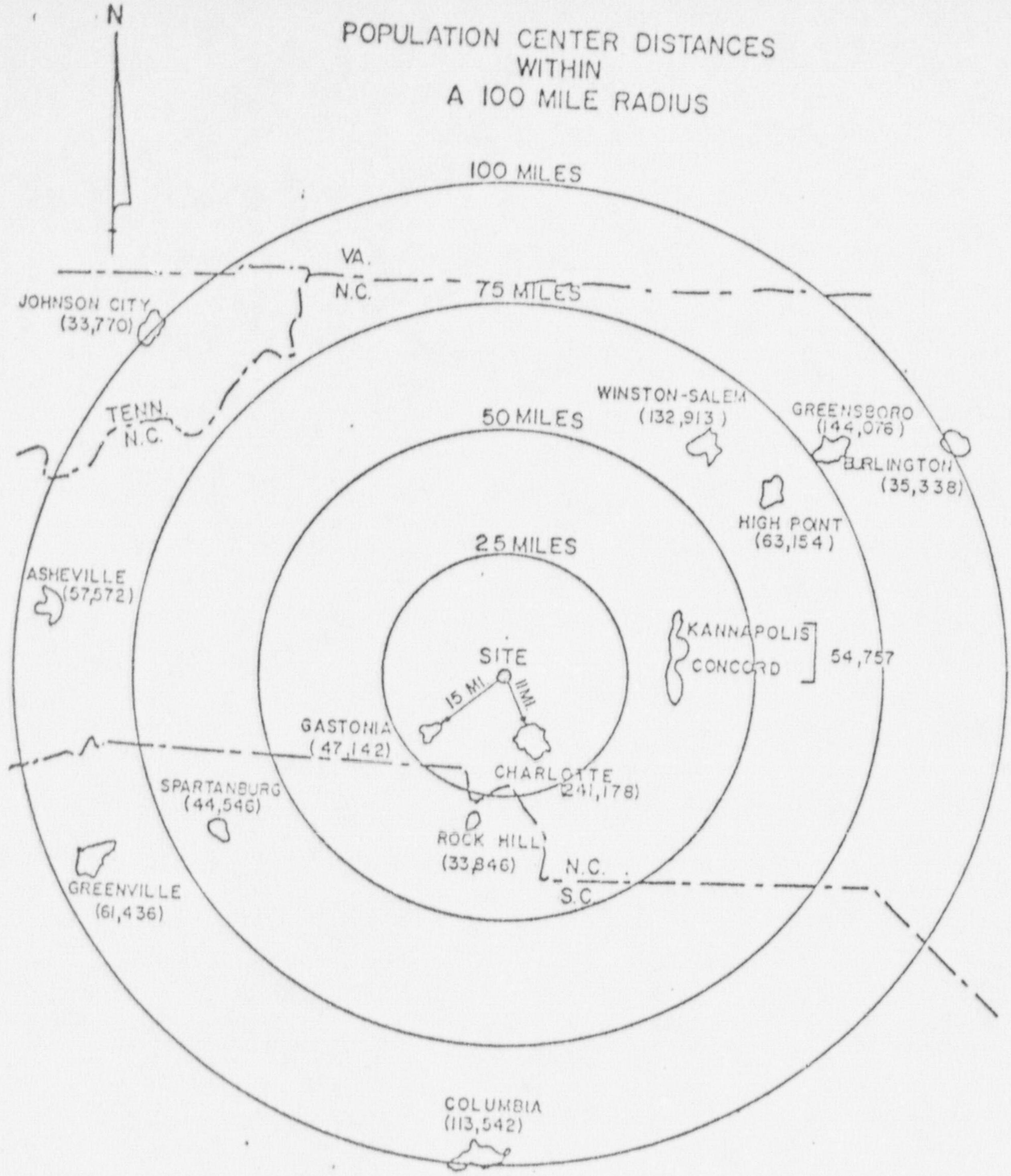


REIDSVILLE
GREENSBORO
BURLINGTON
Hillsboro
DURHAM
CHAPEL HILL
HIGH POINT
Archdale
Siler City
SANFORD
Seagrove
Robbins
Carthage
588
SOUTHERN PINES
FORT BRAGG
Aberdeen
ROCKINGHAM
HAMLET
RAEFORD
LAURINBURG
Pageland
Chesterfield
CHERAW
BENNETTSVILLE
Society Hill
DARLINGTON
NORTH
SOUTH CA
NE
ENE
E
ESE
SE
36°
35°
80°
79°
25
15
07
U.S. Department of Commerce
Environmental Science Services Administration
Coast and Geodetic Survey
SCALE 1:1,000,000
CONTOUR INTERVAL
1000 feet
HIGHEST TERRAIN elevation is
6684 feet
located at 35°46'N - 82°16'W
Critical elevation .4254
Approximate elevation .3300
Doubtful locations are indicated by omission
of the point locator (dot or "x")
GENERAL AREA MAP



ELKIN
NORTH WILKESBORO
WINSTON-SALEM
Mocksville
STATESVILLE
MORGANTON
SALISBURY
MOORESVILLE
LINCOLNTON
Cornelius
GASTONIA
CHARLOTTE
CHEROKEE NUCLEAR STATION
GAFFNEY
CATAWBA NUCLEAR STATION
SPARTANBURG
MONROE
50 MILE RADIUS
NORTH CAROLINA
SOUTH CAROLINA
Jonesville
UNION
CHESTER
LANCASTER
Great Falls
Kershaw
Whitmire
LAURENS
CLINTON
Winnsboro
Ridgeway
NEWBERRY
CAMDEN
N
NNW
NW
WNW
W
WSW
SW
SSW
S
SSE
NNE
36°
35°
82°
81

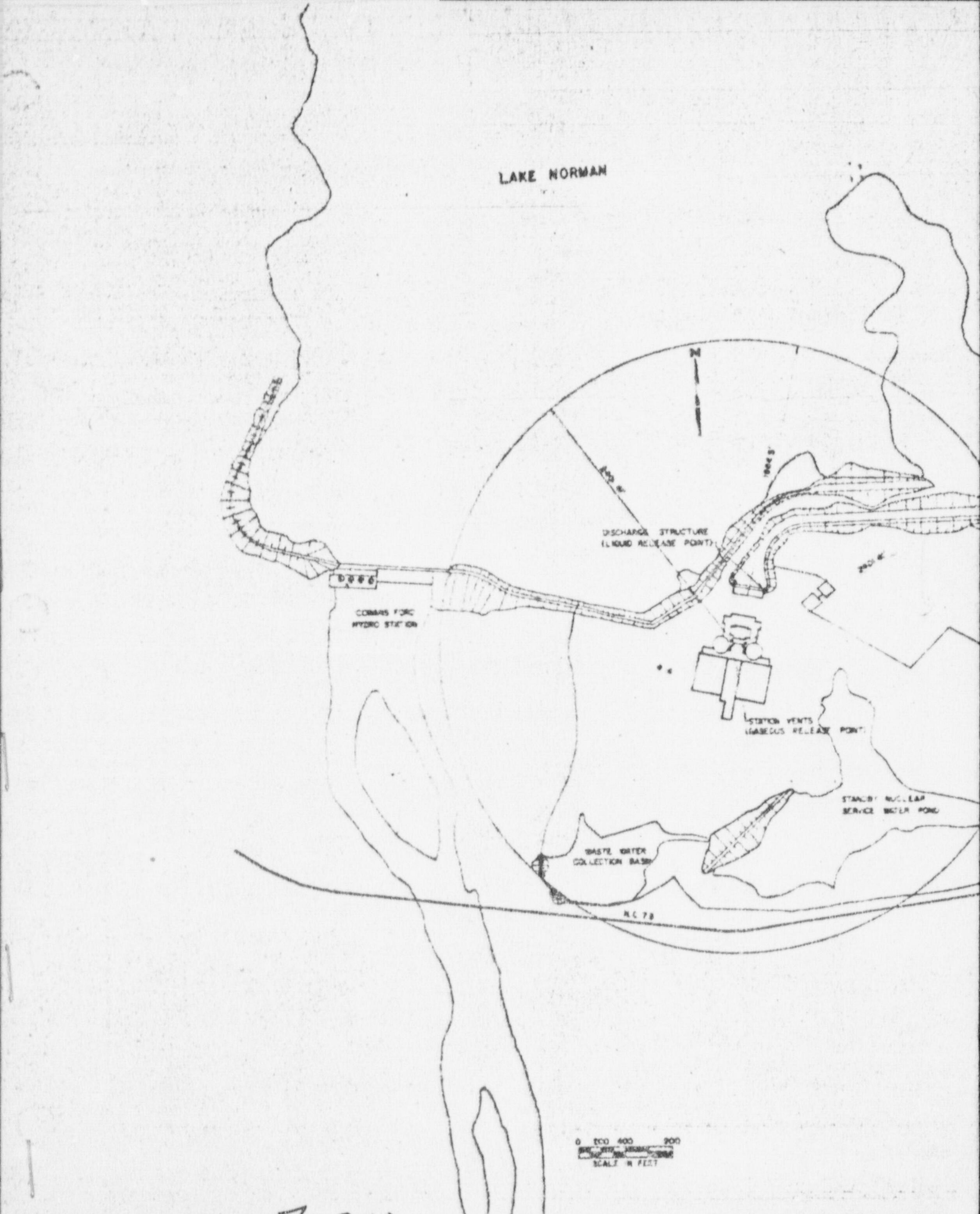


SOURCE: 1970 CENSUS OF POPULATION

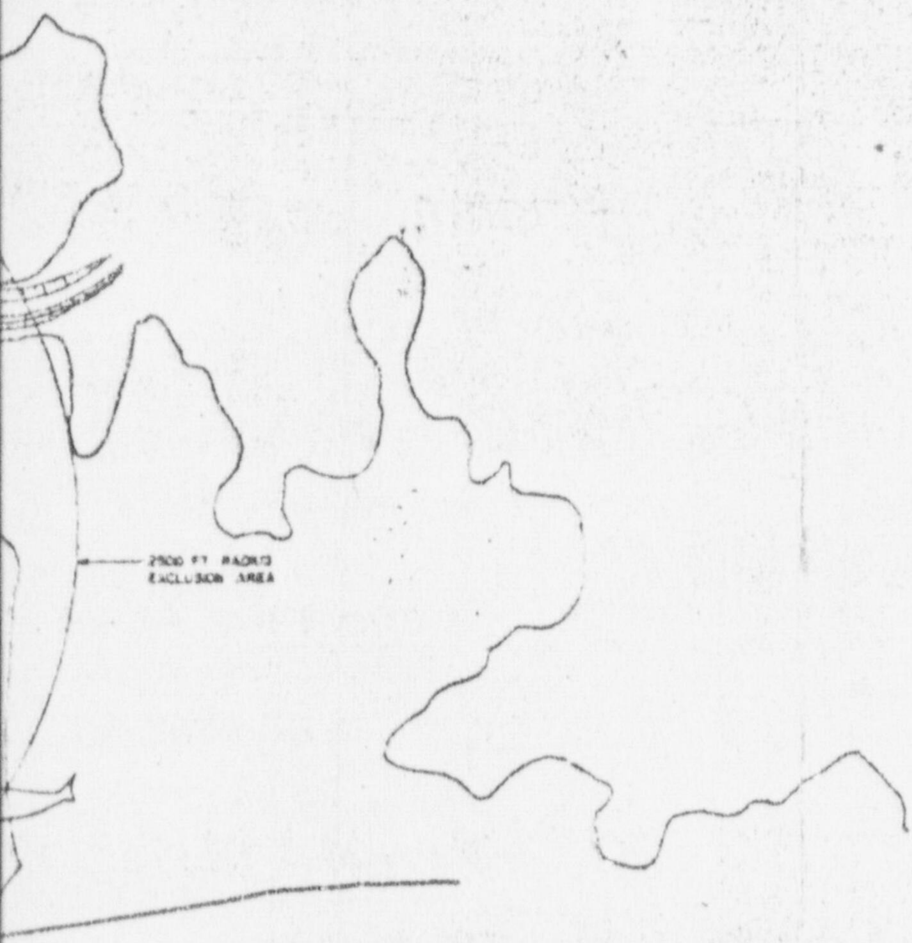
POPULATION CENTER DISTANCES WITHIN A 100 MILE RADIUS

McGUIRE NUCLEAR STATION

Figure 2.1.3-1


LAKE NORMAN
N
DISCHARGE STRUCTURE
(LIQUID RELEASE POINT)
COWANS FORD
HYDRO STATION
STATION VENTS
(GASEOUS RELEASE POINT)
STANDBY NUCLEAR
SERVICE WATER POND
WASTE WATER
COLLECTION BASIN
N.C. 73
SCALE IN FEET

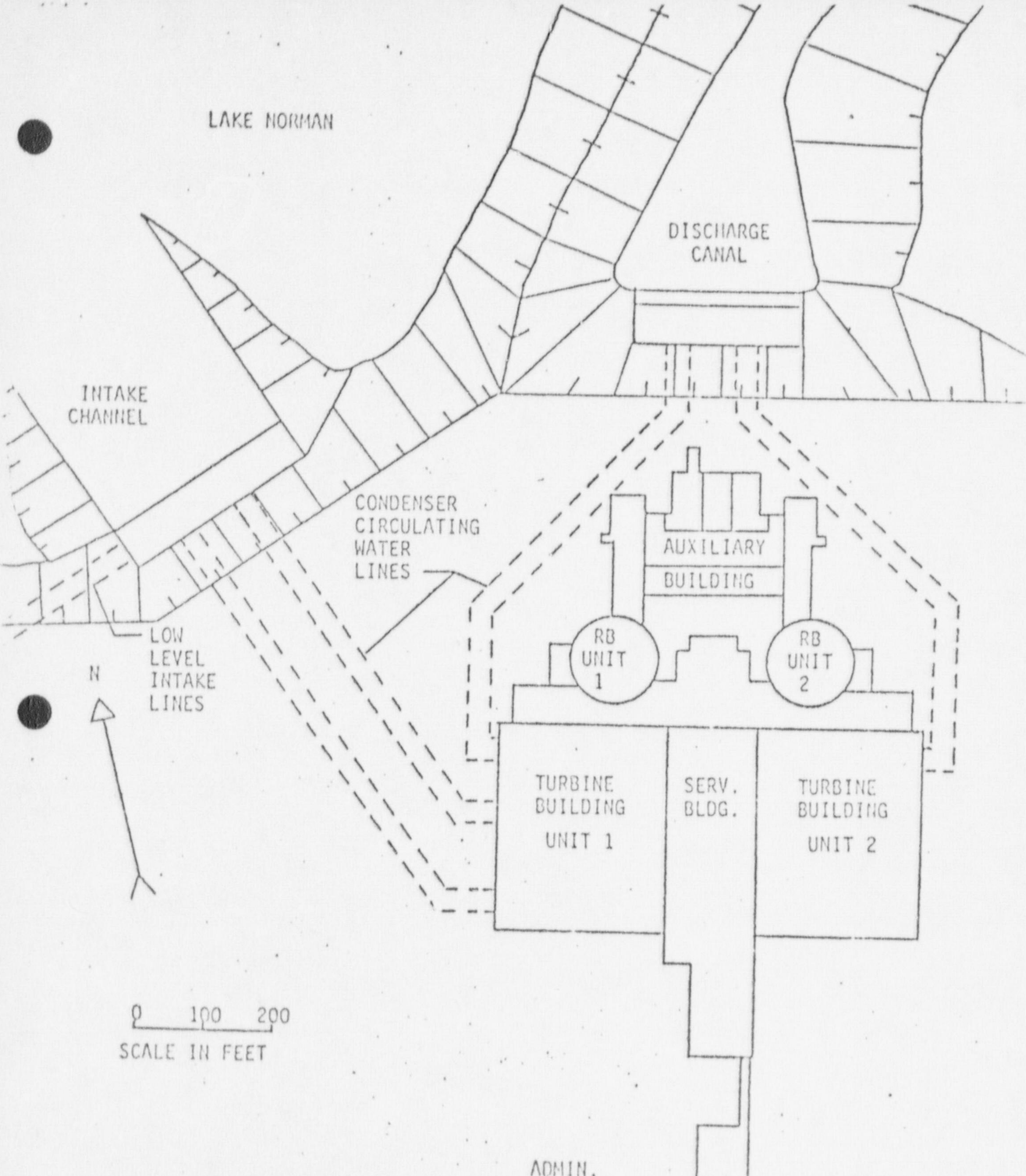
17-200



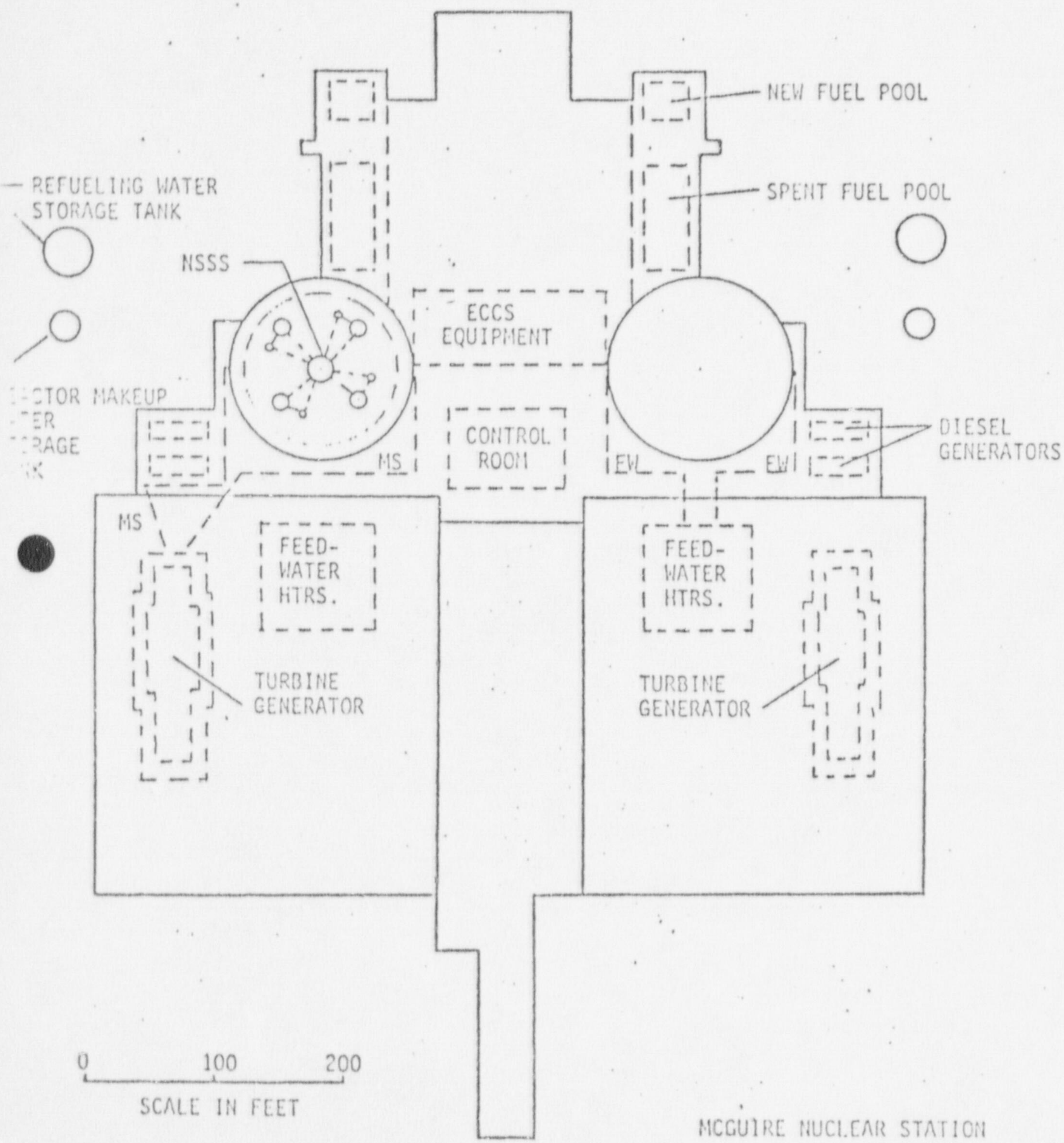
DISTANCES FROM RELEASE POINTS
TO THE EXCLUSION AREA

McGUIRE NUCLEAR STATION

Figure 2.1.2-2

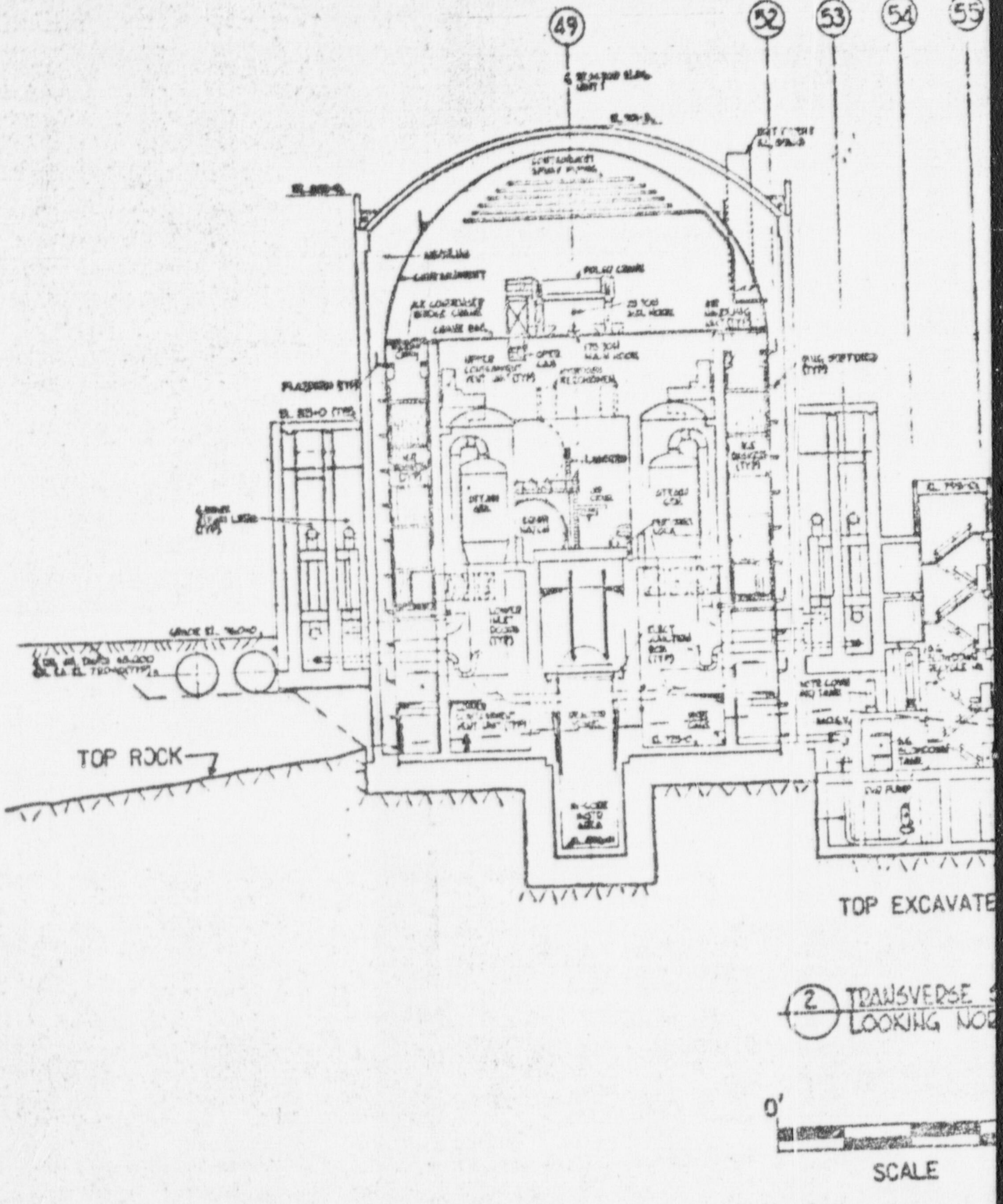


MCGUIRE NUCLEAR STATION



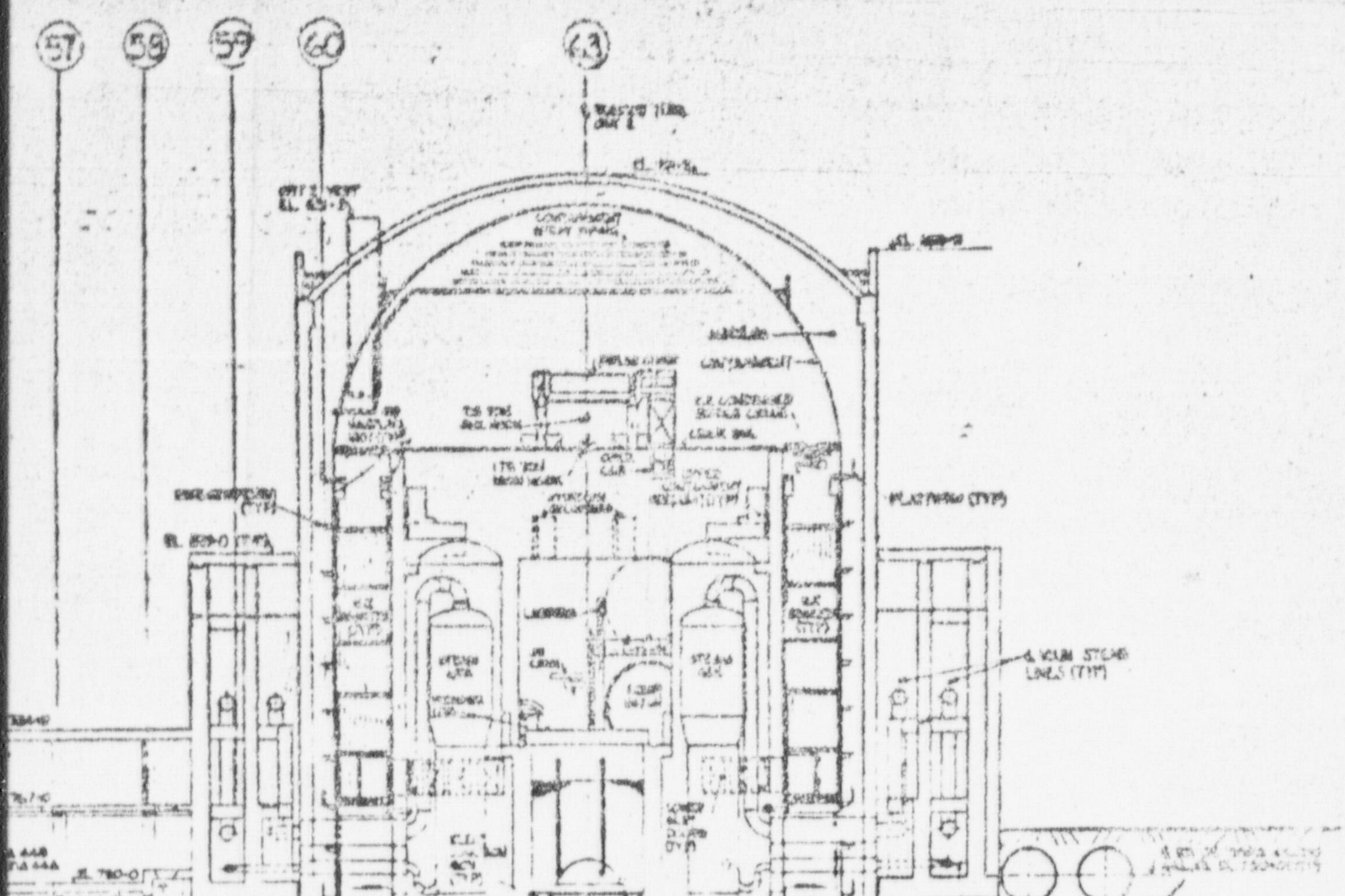
MCGUIRE NUCLEAR STATION

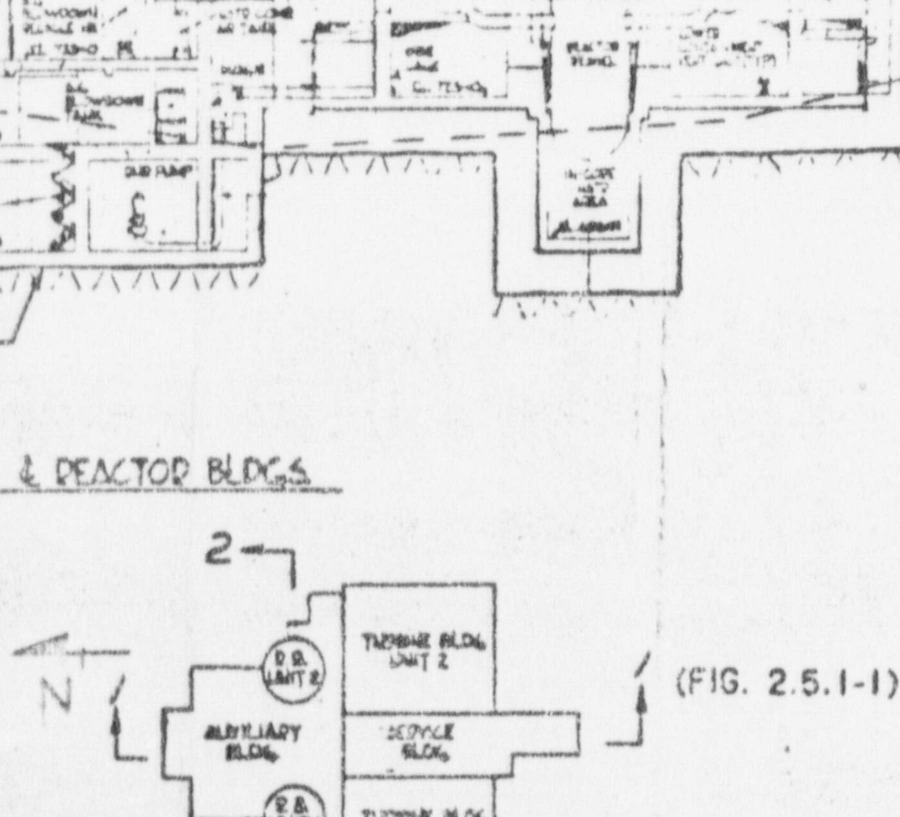
A-202



49
52
53
54
55
TOP ROCK
TOP EXCAVATE
2
TRANSVERSE S
LOOKING NOR
0'
SCALE

A-203

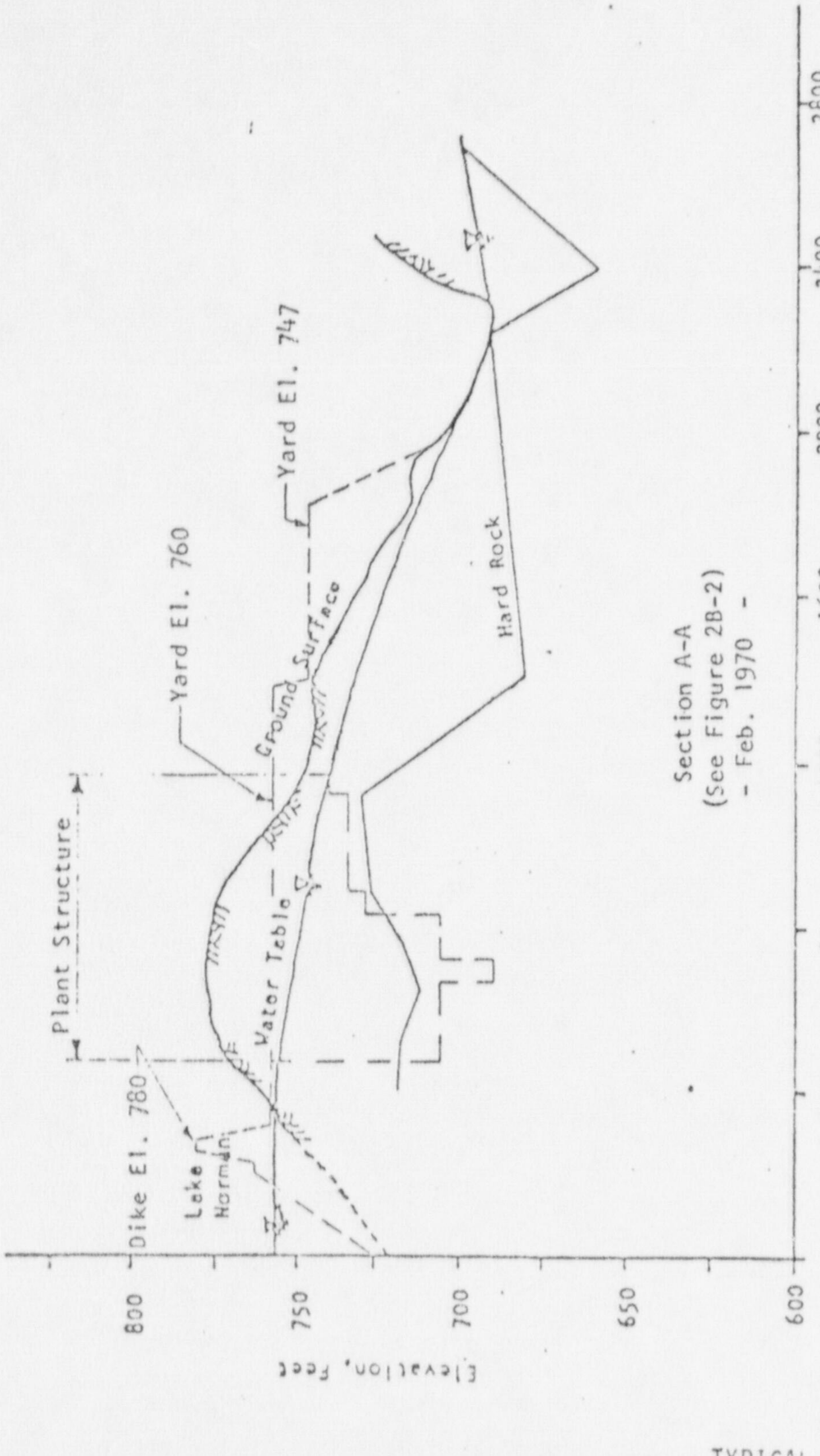




STRUCTURE FOUNDATION PROFILE

McGUIRE NUCLEAR STATION

Figure 2.5.1-2



TYPICAL GROUNDWATER SECTION A-A
McGUIRE NUCLEAR STATION
Figure 2B-3
Revision 5

17-204

THE CAROLINA ENVIRONMENTAL STUDY GROUP

March 6, 1977

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

Gentlemen:

The enclosed twenty copies of questions with respect to the safety of the McGuire, Catawba, Perkins, Cherokee and Oconee nuclear stations were prepared for submission at the scheduled meeting of the ACRS in Charlotte on Jan. 19. The meeting was canceled. As I have not learned of a subsequent scheduling of the meeting I am taking the opportunity to forward these questions to you. I note with interest that none of them were raised in NUREG-0138 or NUREG-0153. I shall feel somewhat less concern if the procedures for projecting life performance of the most heavily fatigued components in the primary coolant system are made at a higher level of sophistication than is provided by the ASME codes now in effect.

Yours very truly,

Jesse L. Riley

Jesse L. Riley
President, CESG

ATTACHMENT 2

A-205

854 HENLEY PLACE · CHARLOTTE · NORTH CAROLINA 28207 · (704) 375-4342

CAROLINA ENVIRONMENTAL STUDY GROUP

January 19, 1977

854 HENLEY PLACE
CHARLOTTE, NORTH CAROLINA 28207
704 378 4342

P. O. BOX [illegible]
WINSTON-SALEM, NORTH CAROLINA 27[illegible]
919 [illegible]

Advisory Committee on Reactor Safeguards
Meeting in Charlotte, N. C.

In the matter of Perkins and Cherokee nuclear stations and with reference to Catawba, McGuire and Oconee nuclear stations

WASH-1270 was concerned with reducing the risk of a common mode failure involving control rod insertion. At both the Hanford and Kahl reactors there had been incidents in which the primary control rod insertion system had failed to perform. WASH-1270 reported estimates of as high 7000 psi in the primary cooling system during ananticipated transient without scram. The Atomic Energy Commission in that report required that, after a specified date, all reactors in order to meet licensing requirements, would have to be equipped with two separate control rod insertion systems which would be entirely different with respect to modes of failure so that the risk of complete insertion failure during a transient would be significantly reduced.

WASH-1270 did not call for retrofit of units licensed before the cutoff date. Was the primary reason for not requiring retrofit economic or concern for generating capacity? The present reserve is such as to permit shutdown and retrofit.

The WASH-1270 requirement appears not to be in effect for Catawba, Perkins or Cherokee. What are the grounds for removing from force the requirements initially promulgated in WASH-1270?

The ASTM code for reactor vessels prescribes the requirements for bolting materials used in closure studs. Single tensile specimens cut from each end of a bolting stock heat must exceed 130 kips at ambient break. For the McGuire and Catawba reactors the variance in accepted individual tensile specimens was such that, using normal statistical inference, the population from which the specimens were drawn contained 18% of individuals less than 130 kips. Several questions should be asked and answered.

The prediction of failure, a serious concern in the NASA space programs, resulted in the development of probabilistic analysis. Why is this more sophisticated, state-of-the-art technique not used in the evaluation of physical test data and the setting of specifications for reactor studs?

The ASTM code assumes that small tensile specimens, area less than 0.5 in^2, will provide critically important information as to the elevated temperature tensile properties of bolts with a minimum cross sectional area of about 40 in^2. Has the relationship between tensile specimen properties obtained according to the code and the elevated temperature tensile characteristics of bolts from the same stock been determined? If so, what were the findings?

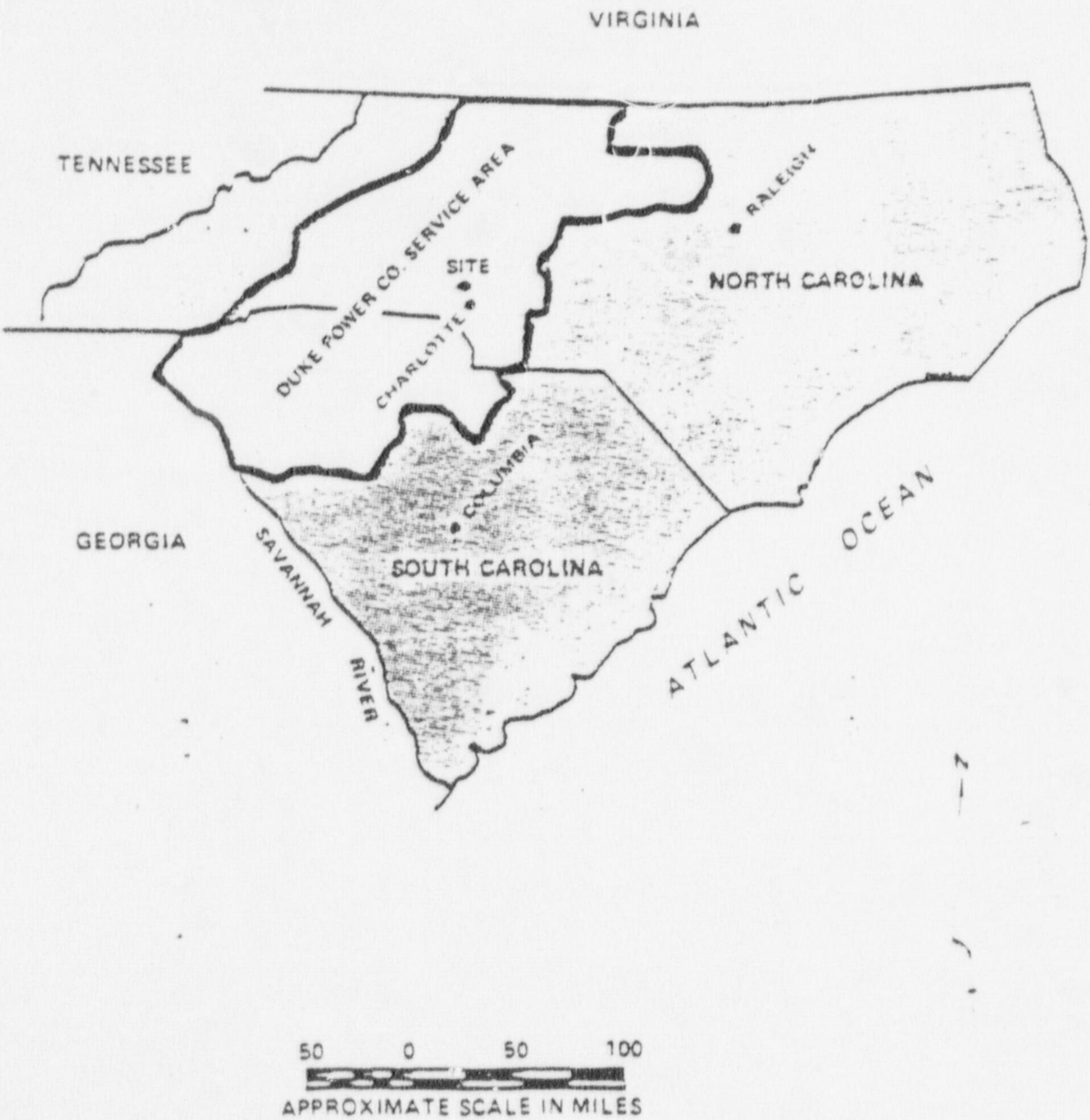
17206

The assumption was made in the case of the McGuire and Catawba reactors that, in operation, no portion of the studbolt would exceed a temperature of 500°F. What assumptions are made for the Perkins and Cherokee reactors? Has an actual determination been made of the maximum operating temperature of stud bolts with particular reference to those portions penetrating the upper flange and threaded into the lower flange?

Does not the departure from the initial requirements for dual and different rod insertions proposed, actually promulgated, in WASH-1270 increase the risk associated with an undetected, weak stud bolt?

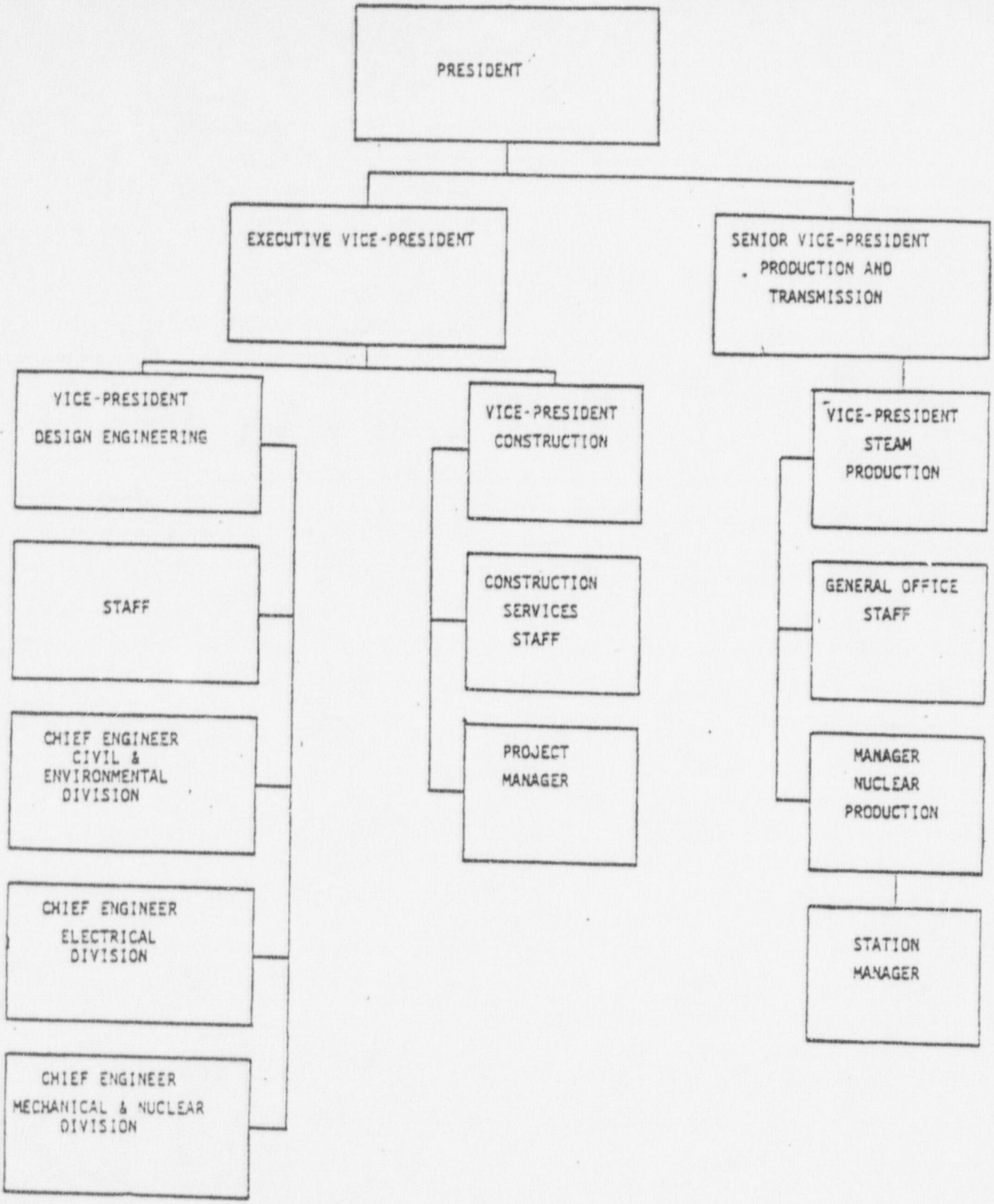
Has an independent, well qualified expert or group of experts calculated the load contribution to studs vicinal to a failed stud of the recovery from compression of the flange region compressed by the nut of the failed stud for a variety of situations including normal operating pressure and the range of pressures anticipated for the full range of transients which can develop in the absence of scram? If so, what were the findings?

Jesse L. Riley
Jesse L. Riley
for CESG



DUKE POWER COMPANY SERVICE AREA

McGUIRE NUCLEAR STATION

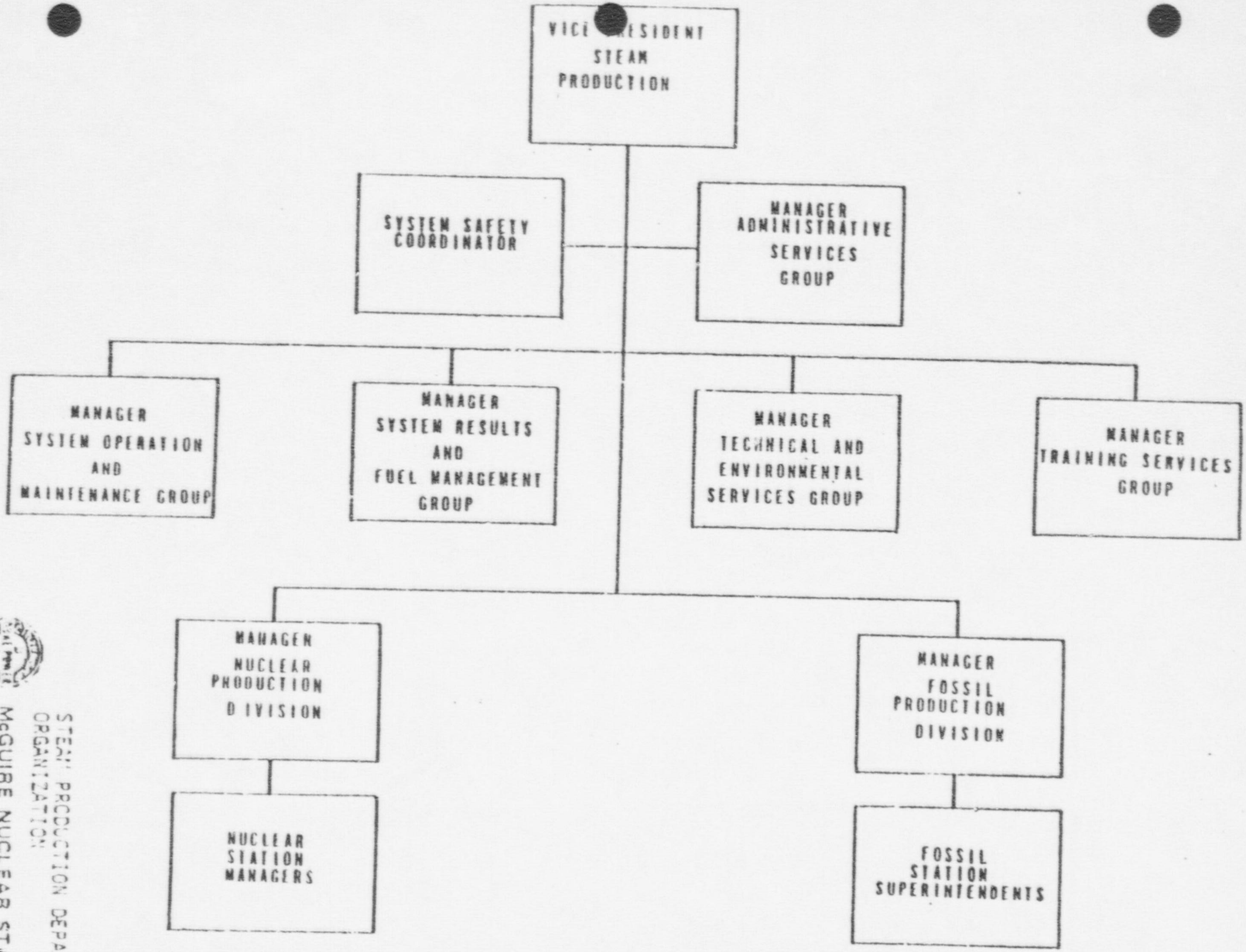


DUKE POWER CORPORATE ORGANIZATION

McGUIRE NUCLEAR STATION

Figure 13.1.1-1

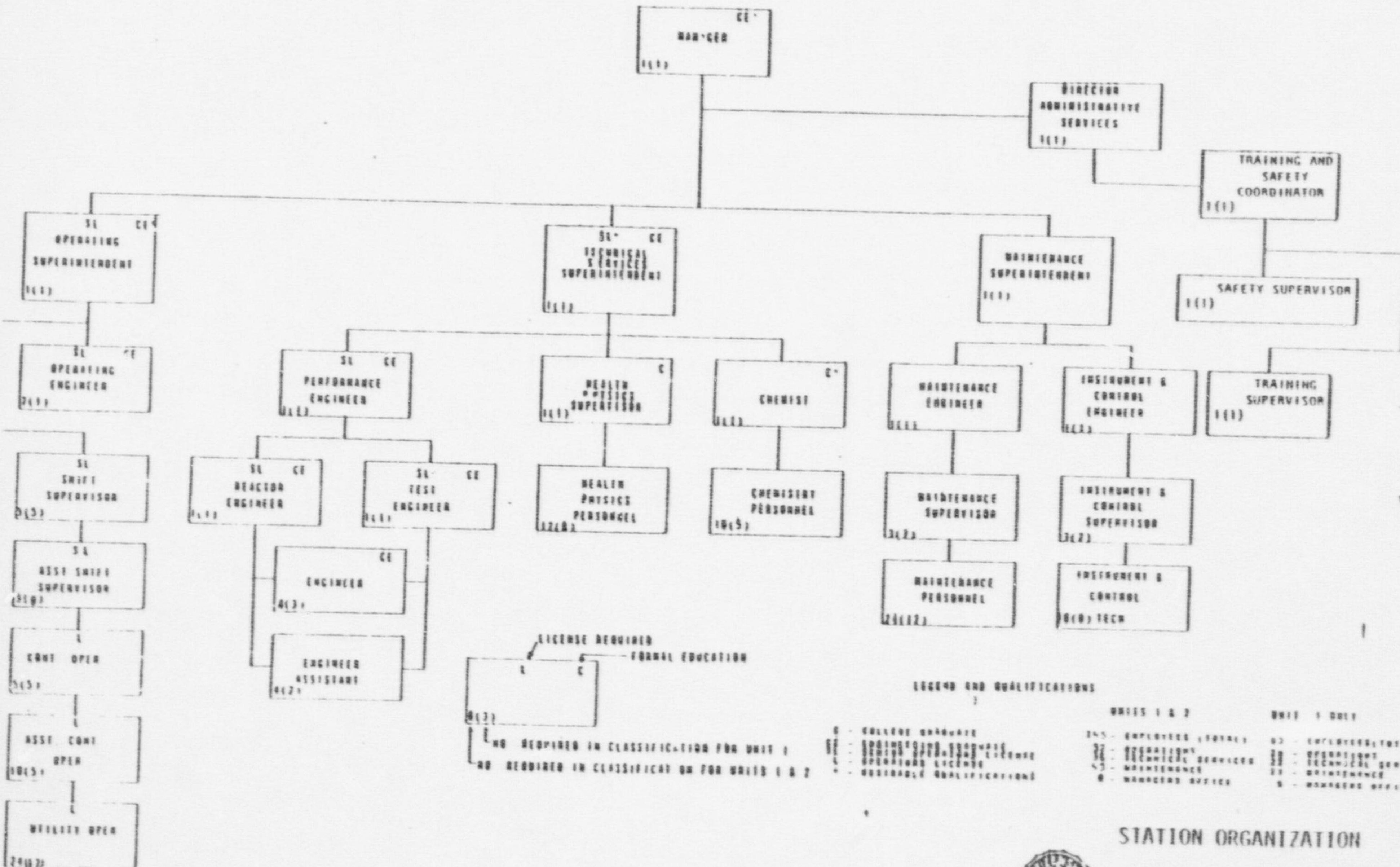
17-209



STEAM PRODUCTION DEPARTME
ORGANIZATION

McGUIRE NUCLEAR STATIO

17.210



STATION ORGANIZATION

McGUIRE NUCLEAR STATION

A-211


BOAT DOCK
EXCLUSION AREA BOUNDARY
BOAT GARAGE
CONSTRUCTION ACCESS ROAD
ENVIRONMENTAL LABORATORY
CONSTRUCTION PARKING
AUXILIARY BUILDING
REACTOR BUILDINGS
TURBINE BUILDINGS
SERVICE BUILDING
STANDBY NUCLEAR SERVICE WATER POND
ACCESS ROAD
ADMINISTRATION BUILDING
DISCHARGE STRUCTURE
SEWAGE TREATMENT PLANT
LOW LEVEL PUMP STRUCTURE
WASTE WATER TREATMENT SYSTEM
ELEVATED WATER TANK
RELAY HOUSE
SWITCHING STATION
MICROWAVE AND METEOROLOGICAL TOWER

SUMMARY OF UHI EVENTS

INITIATED REVIEW OF UHI EVALUATION MODEL	JAN. '75
NRC STATUS REPORTS	SEPT. '75
	AUG. '76
DRAFT SERS	JULY '77
	DEC. '77
ACRS MEETINGS	JUNE '75
	SEPT. '75
	DEC. '75
	MARCH '76
	JUNE '76
	SEPT. '76
	JULY '77
	DEC. '77

Issues Considered Since 12-77

- Verification of Split Downcomer
- Accumulator Delivery
- ROSA-II Analysis
- Semiscale MOD3 UHI Tests
- Error Corrections
  - UHI Low Flow Quench Limit
  - Generic Zirc-Water Reaction
- Head Cooling Jet Modification To Reduce Upper Head Temperature

17-21

Confirmatory Comparisons For Split Downcomer Model

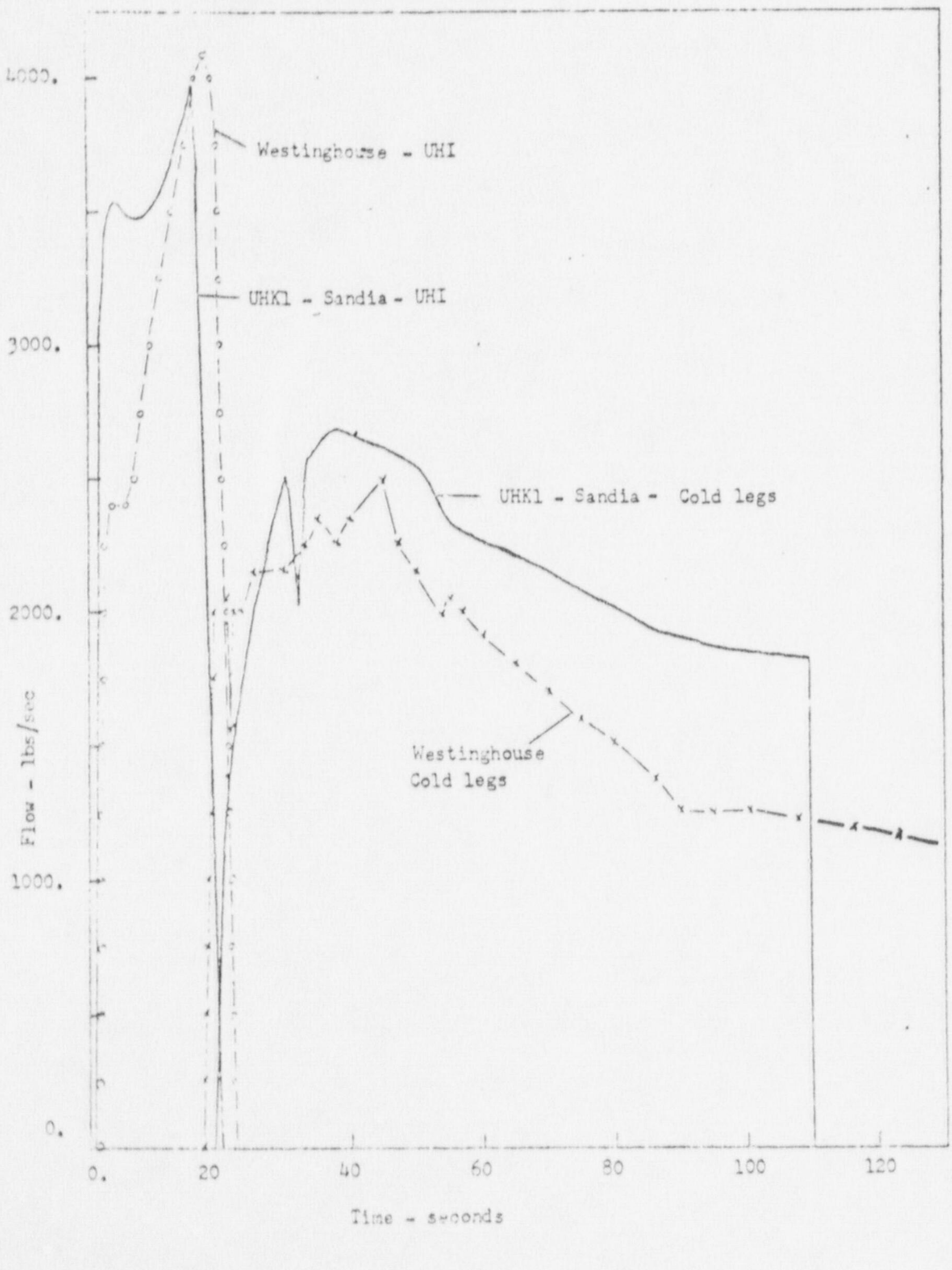
- L1-4 - Lower Plenum Delivery, Lower Plenum And Downcomer Storage, End-Of-Bypass.
- Creare Sweepout - Lower Plenum Storage
- Creare Transient - End-Of-Bypass And Lower Plenum Delivery
- Plant Sensitivities - Pct And System Effects

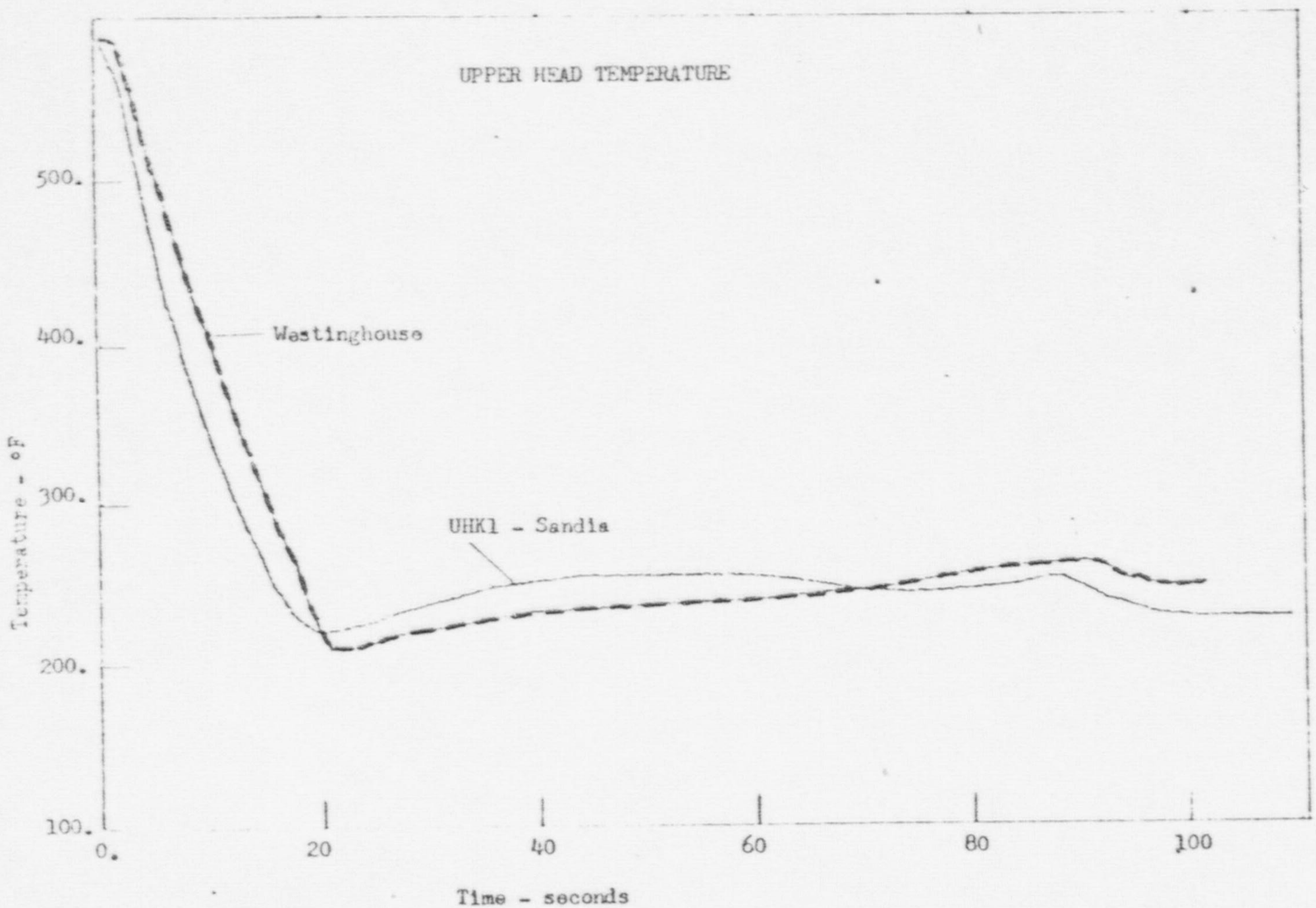
Downcomer Model PCT Sensitivity

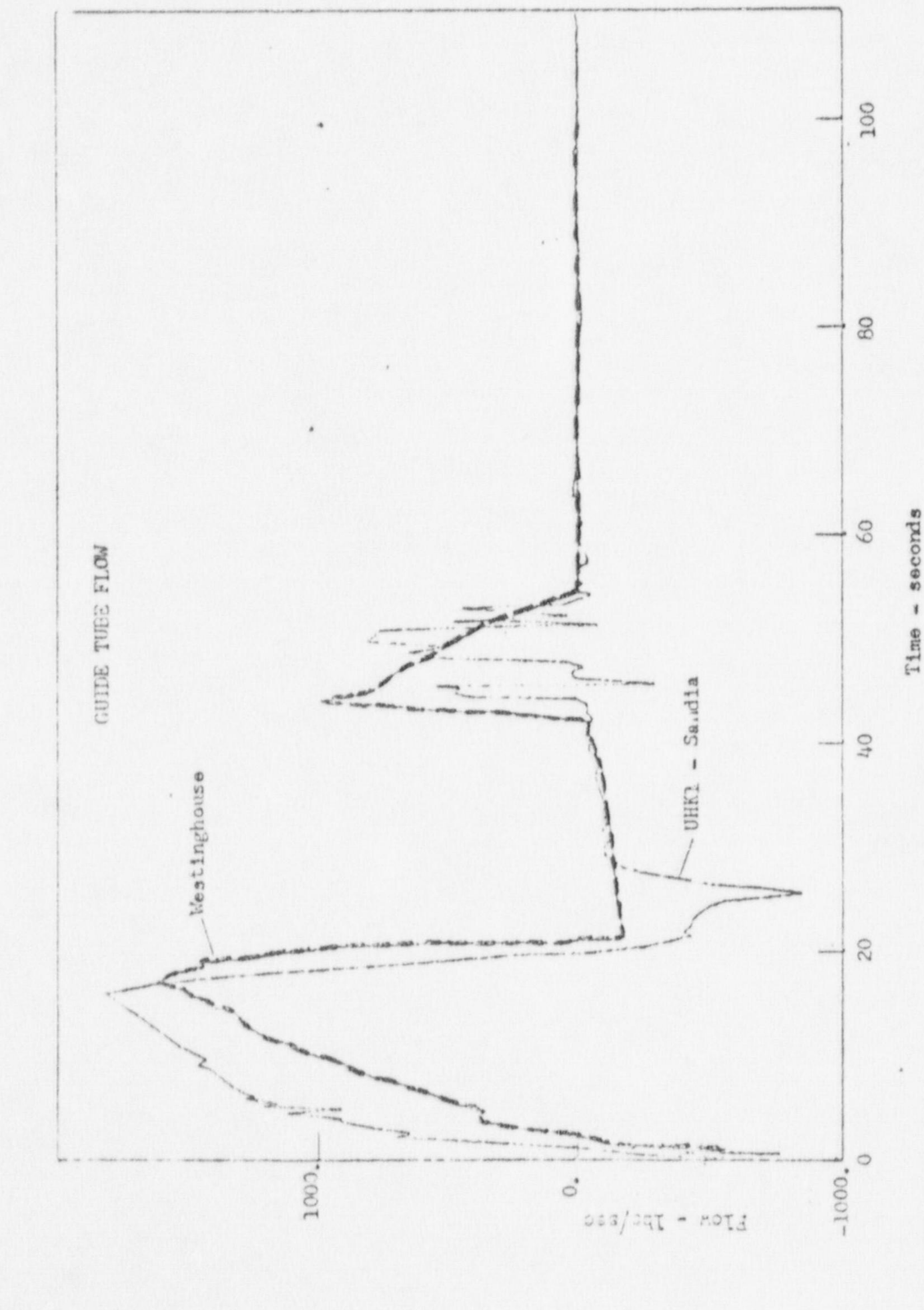
Case	Perfect Mixing		Imperfect Mixing	
	PCT (°F)	ΔPCT (°F)	PCT (°F)	ΔPCT (°F)
Base	2020	-	1800	-
Head Fix	2020	0	1997	197
Lower Plenum Separation	2130	110	2150	153

. Based On This Study And Conservative Comparisons To Data, Revised Model Is Acceptable.

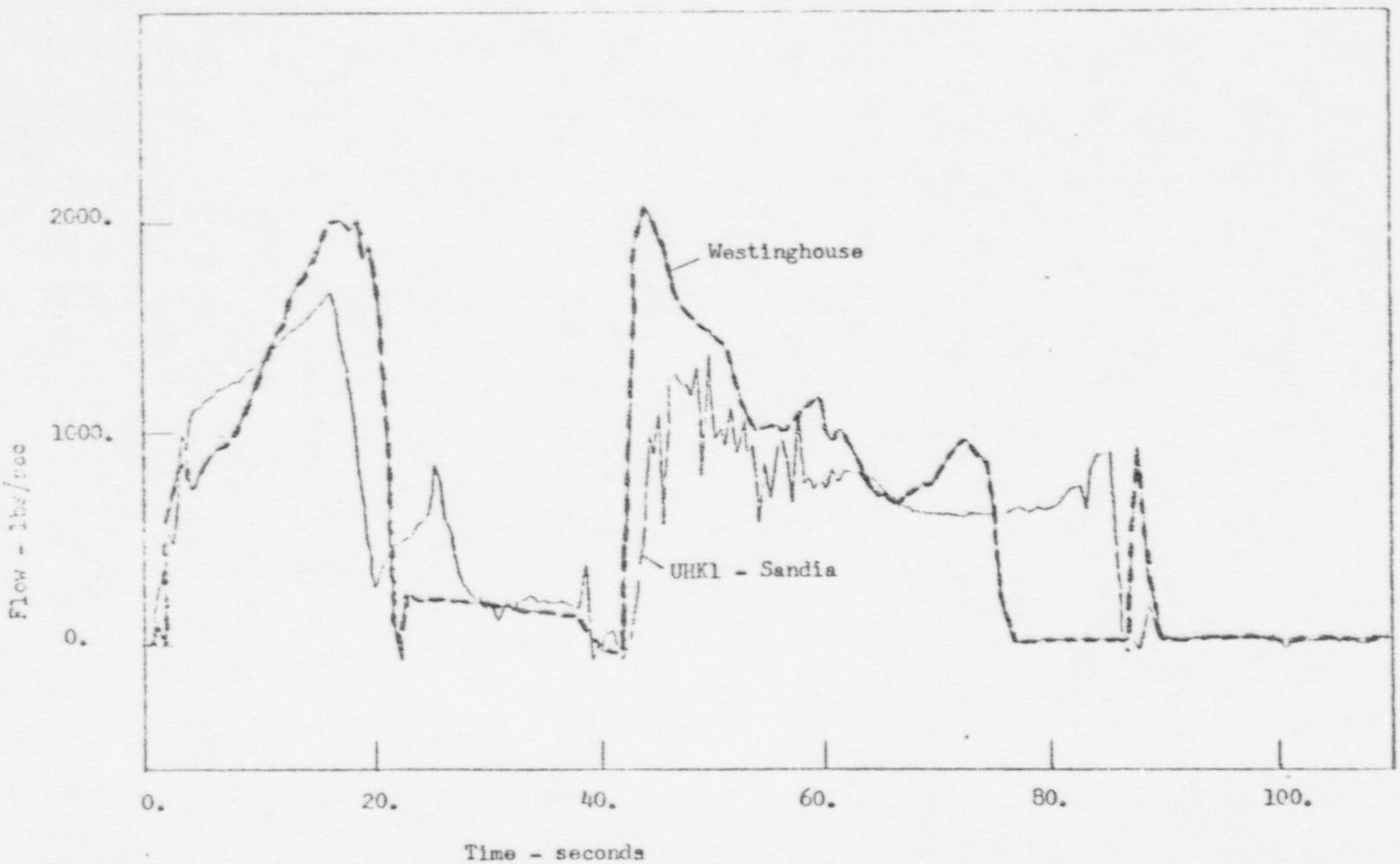
ACCUMULATOR FLOW RATES




UPPER HEAD TEMPERATURE
Westinghouse
UHK1 - Sandia
Temperature - °F
500.
400.
300.
200.
100.
0.
20
40
60
80
100
Time - seconds


GUIDE TUBE FLOW
Westinghouse
UHKI - Sandia
Time - seconds
Flow - lbm/sec
1000.
0.
-1000.
0
20
40
60
80
100

SUPPORT COLUMN FLOW



PREDICTED ECCS PERFORMANCE

- OVERVIEW

- CURRENTLY SUBMITTED RESULTS

- MODIFICATION TO METAL-WATER REACTION RATE CALCULATION

- APPROVED UHI EVALUATION MODEL

- CALCULATED ECCS PERFORMANCE WITH APPROVED MODEL

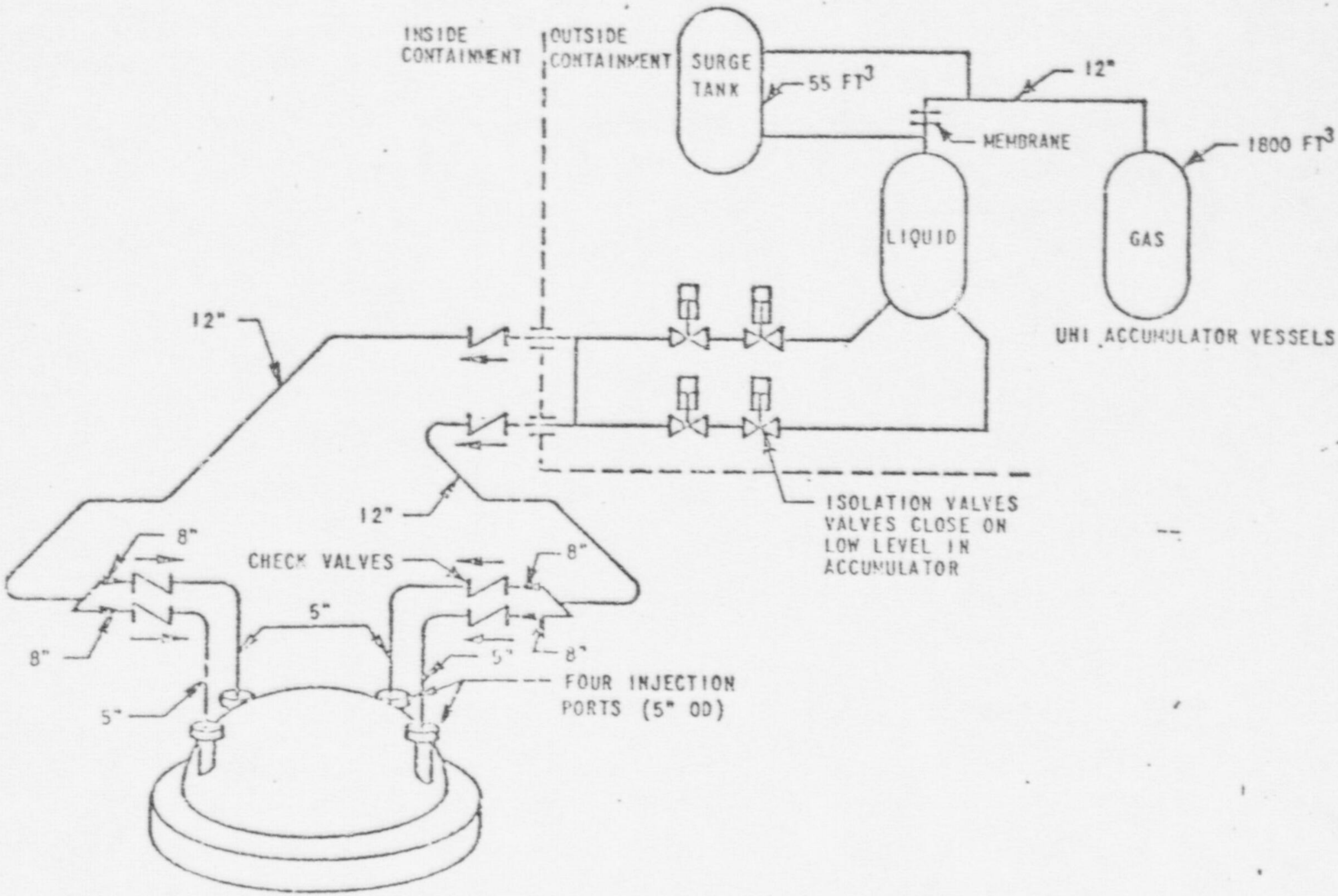


Figure 2.1 Upper Head Injection System Schematic

H-22

7165-1

ANALYSIS CONDITIONS

NSSS POWER (HYDRAULIC ANALYSIS, MWT, 102% OF)	3564
CORE POWER (ROD HEATUP ANALYSIS, MWT, 102% OF)	3411
PEAK LINEAR POWER (KW/FT)	12.63
PEAKING FACTOR (AT LICENSE RATING)	2.32
ACCUMULATOR WATER VOLUME (COLD LEG, NOMINAL, FT^3)	950
ACCUMULATOR WATER VOLUME (UHI, NOMINAL DELIVERED, FT^3)	1000 FT^3 PERFECT MIXING 850 FT^3 IMPERFECT MIXING

COMPLIANCE WITH APPENDIX K 10CFR50.46

RESULT	C_D = 0.6 DECLG PERFECT MIXING	C_D = 0.6 DECLG IMPERFECT MIXING
PEAK CLAD TEMP. (°F)	2164.	2163.
PEAK CLAD TEMP. LOCATION (FT)	9.0	9.0
LOCAL ZR/H2O REACTION (MAX. %)	7.02	6.05
LOCATION OF MAX. LOCAL ZR/H2O (FT)	9.0	9.0
TOTAL ZR/H2O REACTION (%)	<0.3	<0.3
HOT ROD BURST TIME (SEC)	64.0	60.2
HOT ROD BURST LOCATION (FT)	6.0	6.75

MODIFICATIONS TO ECCS MODELS

- MODIFIED METAL-WATER REACTION RATE
- COLD LEG ACCUMULATOR GAS EXPANSION
- DOWNCOMER CROSSFLOW MODIFICATION
- LOWER PLENUM SEPARATION MODEL

ACCOUNT IS MADE OF INCREASED SPRAY NOZZLE FLOW AREA.

LOCTA ZIRC-WATER PROBLEM DESCRIPTION

- ZIRC-WATER REACTION, HEAT SOURCE IS ASSOCIATED WITH SURFACE NODE

- DEFINITION OF VOLUMETRIC HEAT GENERATION BASED ON RADIAL MESH SIZE RESULTS IN TOTAL HEAT GENERATION DUE TO ZIRC-WATER REACTION BEING REDUCED BY FACTOR OF TWO

- PROBLEM HAS BEEN VERIFIED BY EXAMINATION OF HEAT BALANCE AT TIME OF PCT WHEN HEAT IN EQUALS HEAT OUT

- CORRECTION HAS BEEN IMPLEMENTED FOR ALL FUTURE ANALYSES

PEAK CLAD TEMPERATURE RESULTS USING

APPROVED UHI EVALUATION MODEL

DOUBLE ENDED COLD LEG GUILLOTINE BREAK, Cd=0.6

UHI ACCUMULATOR SETPOINTS	PERFECT MIXING IN UPPER HEAD	IMPERFECT MIXING IN UPPER HEAD
PRESSURE	1200 PSI	1300 PSI
VOLUME	~ 1080 FT3	~ 960 FT3
PEAK CLAD TEMPERATURE	2190 DEGREES F	1990 DEGREES F

February 24, 1978

APPENDIX XXVI
Davis Besse 1: Status Report

STATUS REPORT DAVIS-BESSE 1

The Committee reviewed the OL application for DB-1 on January 6, 1977 and issued its report on January 14, 1977.

The plant is now in power ascension, operating up to 75% full power.

STATUS OF ITEMS IN ACRS OL LETTER, January 14, 1977
(item numbers are shown in margin of attached copy of letter) * items are suggested for discussion.

* 1. Not resolved. OL license conditioned to require analysis prior to startup after 1st refueling. NRC Staff has not issued guidelines to licensee for the analysis.

2. Resolved. Additional data on reactor coolant flow and pressure drops is to be provided after the reactor reaches 90% full power.

* 3. Not resolved. Little has been done on this item. The NRC Staff wrote to the State of Ohio (in 1973) suggesting they participate in the program and explaining what assistance is available. No response has been received and n. follow-up action has been taken.

* 4. Not resolved. This is generic item IID-2. The NRC Staff will elaborate.

* 5. Not resolved. Little has been done on this item for DB-1 or other operating plants. The NRC Staff has not required backfitting of RG 1.97.

6. Not resolved. ATWS is generic to all plants.

7. Not resolved. OL license conditioned to require modification prior to startup following 1st refueling. NRC Staff has suggested a modification to resolve this item.

* 8. Not resolved. Much progress has been made on fire protection. OL conditioned to require licensee to meet Appendix A to BTP 9.5.1 within three years of issuance of OL. NRC Staff will elaborate.

9. Resolved. NRC Staff wil elaborate.

STATUS OF ACRS GENERIC ITEMS. (Report No. 4, April 16, 1976)

II-1 Turbine missiles
No change.

II-2 Containment sprays following a LOCA
No change; LOCA doses re-evaluated.

II-3 RPV Failure Post-LOCA by Thermal Shock
No change.

II-4 Instruments to detect severe fuel failures
No change; resolved for limited failures.

II-6 Common mode failures
No change.

II-7 Behavior of fuel under abnormal conditions
No change.

II-9 Seismic scram
No change.

II-11 Instrumentation to follow course of an accident
Resolved generically but DB-1 does not meet RG 1.97
See Above Discussion.

IIA-1 Pressure in containment following a LOCA
Resolved. Reactor cavity pressure for DB-1 resolved.

IIA-4 Rupture of high pressure lines outside containment
Resolved.

IIA-5 PWR pump overspeed during a LOCA
No change.

IIA-7 Steam generator tube leakage
No change; partially resolved by RG 1.83.

IIA-8 10-year review
No change for DB-1

IIC-1 <u>Locking out of power operated ECCS Valves</u>
Isolation of RHR from primary system resolved, single failure criterion for ECCS satisfied. No other changes.

IIC-2 <u>Fire Protection</u>
Progress being made by DB 1.

IIC-3 <u>Features to control sabotage</u>
NRC Staff maintains this is resolved fo DB-1.

IIC-4 <u>Decontamination and Decommissioning</u>
No change.

IIC-5 <u>Reactor vessel supports</u>
No change for DB-1.

IIC-6 <u>Water Hammer</u>
Steam generator water hammer not a problem in B&W plants.

<u>CONDITIONS REMAINING ON OL LICENSE</u>

- Seismic re-analysis of certain plant systems for a 0.2g safe shutdown earthquake and use of Regulatory Guide 1.60 design response spectra.
- Evaluation of fuel rod bowing effects.
- Inadvertent closure of decay heat removal system isolation valves during decay heat removal operation.
- Plant operating restrictions with less than three reactor coolant pumps in operation.
- Evaluation of facility fire protection capability in accordance with Appendix A to Branch Technical Position APCSB 9.5.1.

<u>CONDITIONS RESOLVED TO SATISFACTION OF NRC STAFF, BUT AMENDMENTS REMOVING THE LICENSE CONDITIONS NOT ISSUED.</u>

- Install flow measuring devices to monitor adequacy of boron dilution modes of plant operation.
- Analysis of the reactor coolant system response to pressure transient that can potentially occur during startup and shutdown.

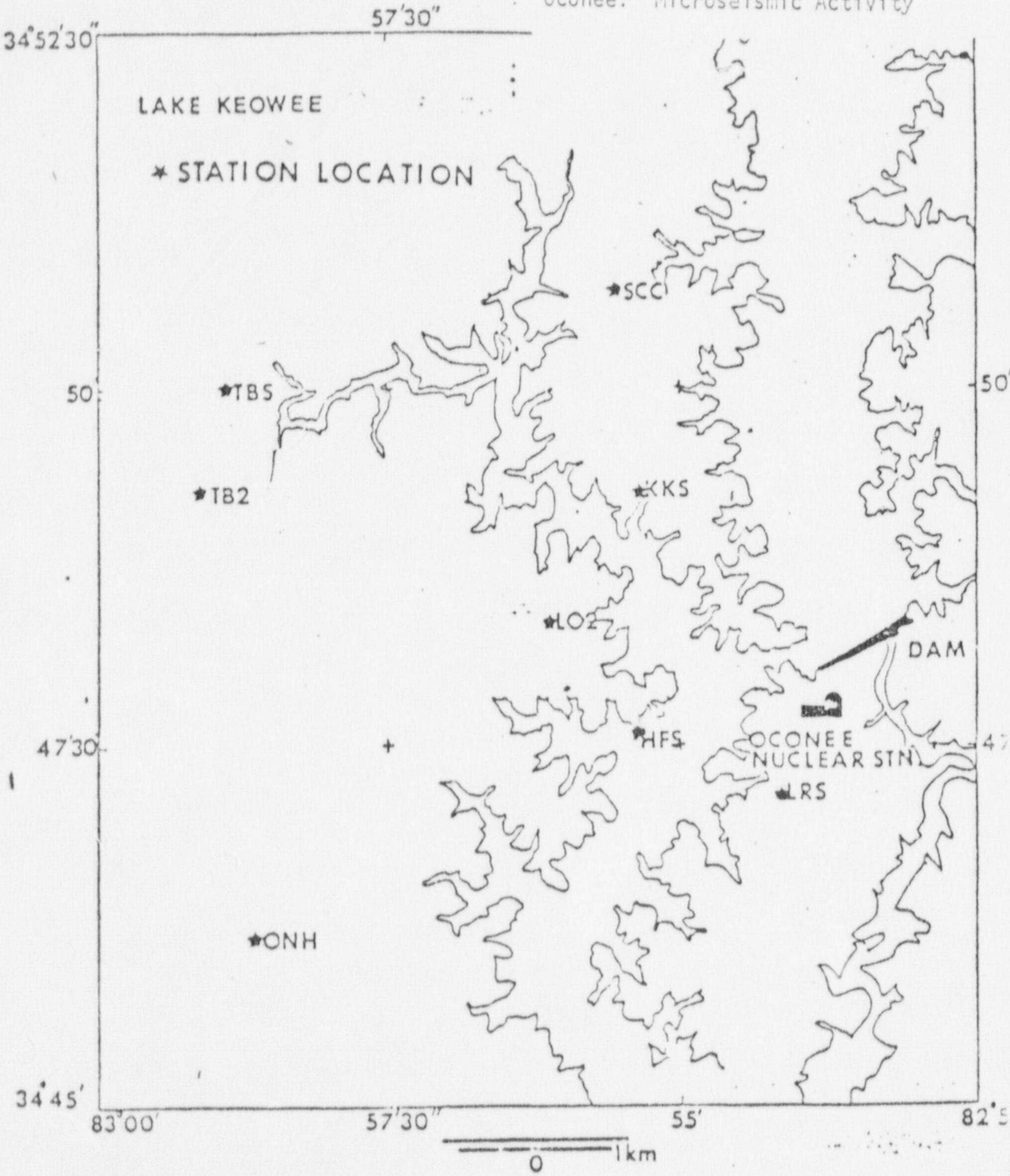
ALL OUTSTANDING ITEMS IN SER HAVE EITHER BEEN RESOLVED OR A CONDITION IMPOSED ON THE OL.

17-230

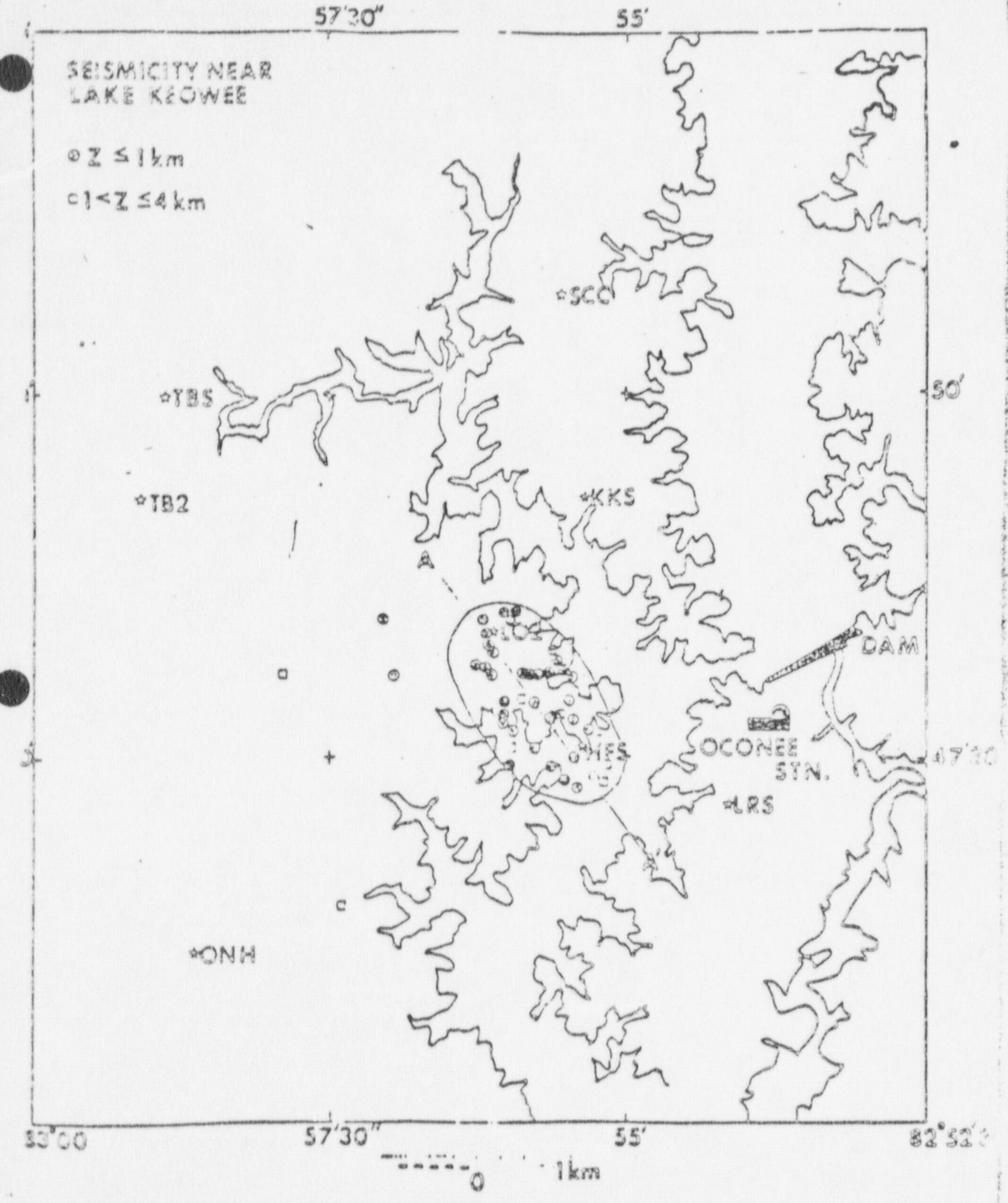
Fig. 1

STATION LOCATIONS OF KEOWEE NET

APPENDIX XXVII
Oconee: Microseismic Activity



A-231



Location of Seismic activity near Lake Keowee (December 30, 1977 - January 16, 1978)

TABLE 1

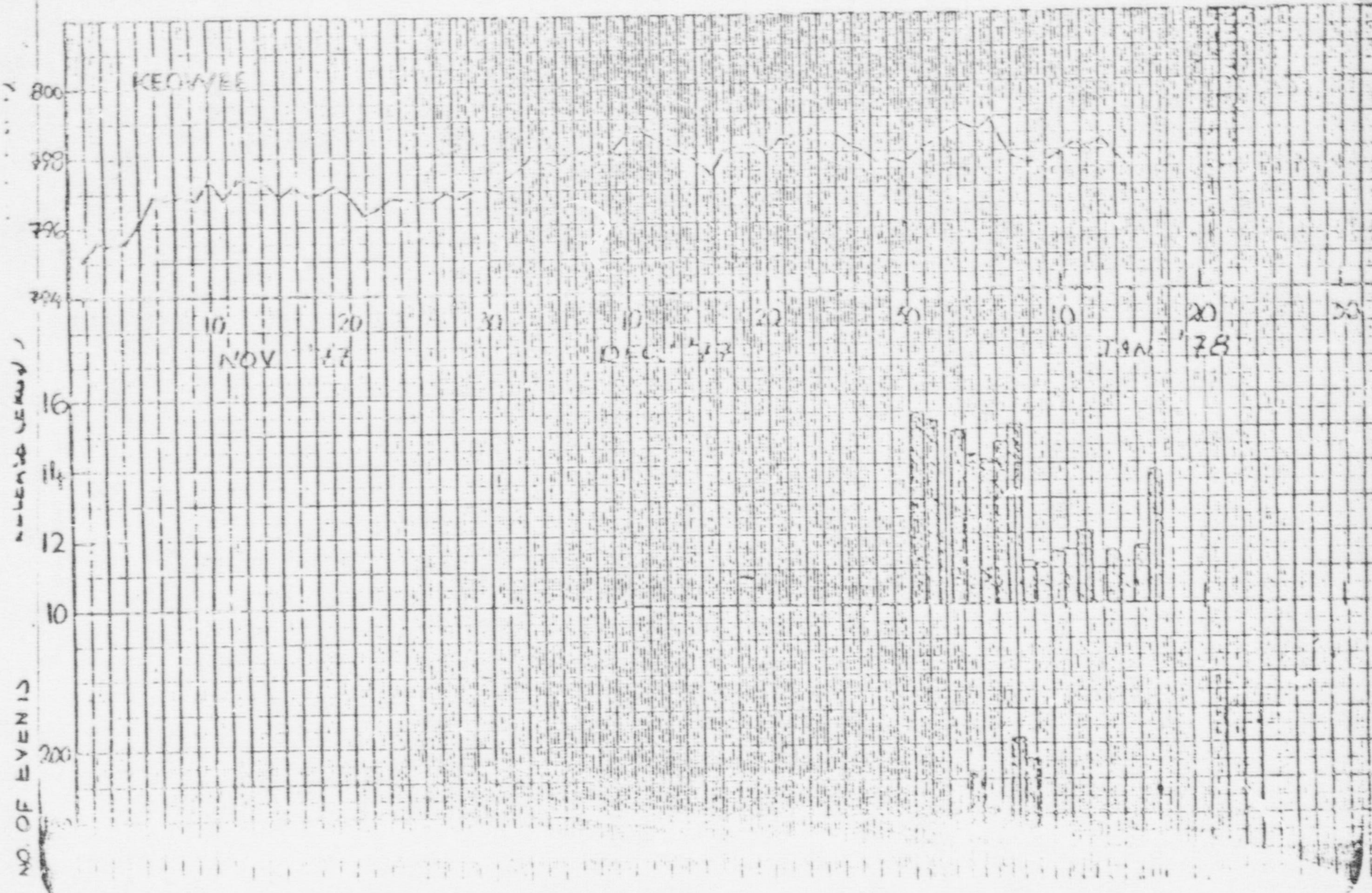
NUMBER OF EVENTS IN KEOWEE AREA

DATE	NUMBER *
December 28, 1977	0
29	2
30	1
31	28
January 1, 1978	16
2	38
3	162
4	109
5	94
6	209
7	179
8	40
9	9
10	55
11	37
12	12
13	13
14	4
15	10
16	45

*Note: The number of events for the period (December 28 - January 2) are those recorded at the Jocassee net, while from January 3, 1978, events recorded in the Keowee net were used.

Fig. 2

Lake Level at Keowee dam, compared with daily seismic energy release and number of earthquakes.



A-234


REVISED APRIL 2 1970
STAGE IV
12/13/69
LEGEND
PROJECTED
ACTUAL FILLING
FT. (MSL)
820
800
780
740
720
700
680
660
640
1968
1969
1970
1971
1972

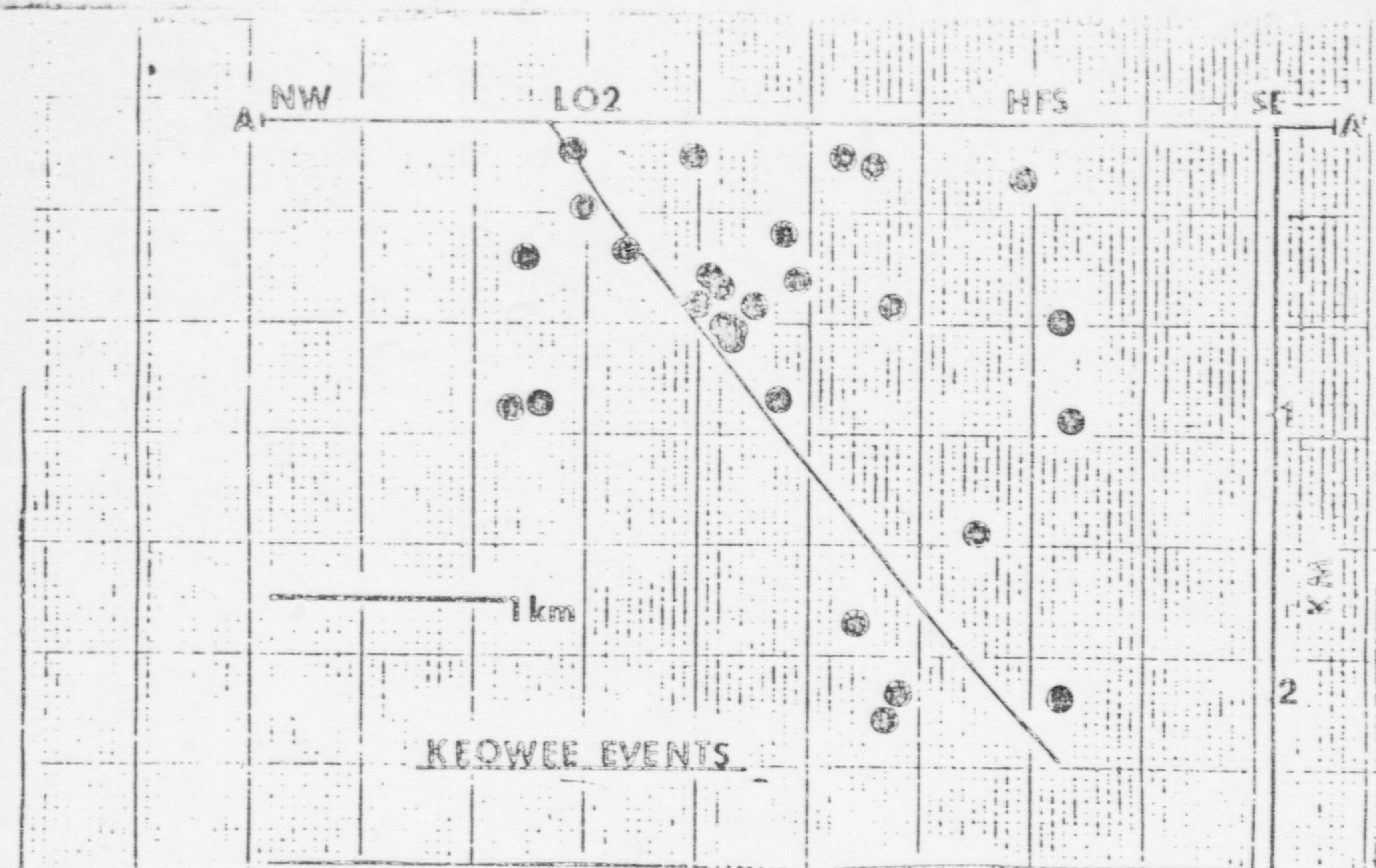
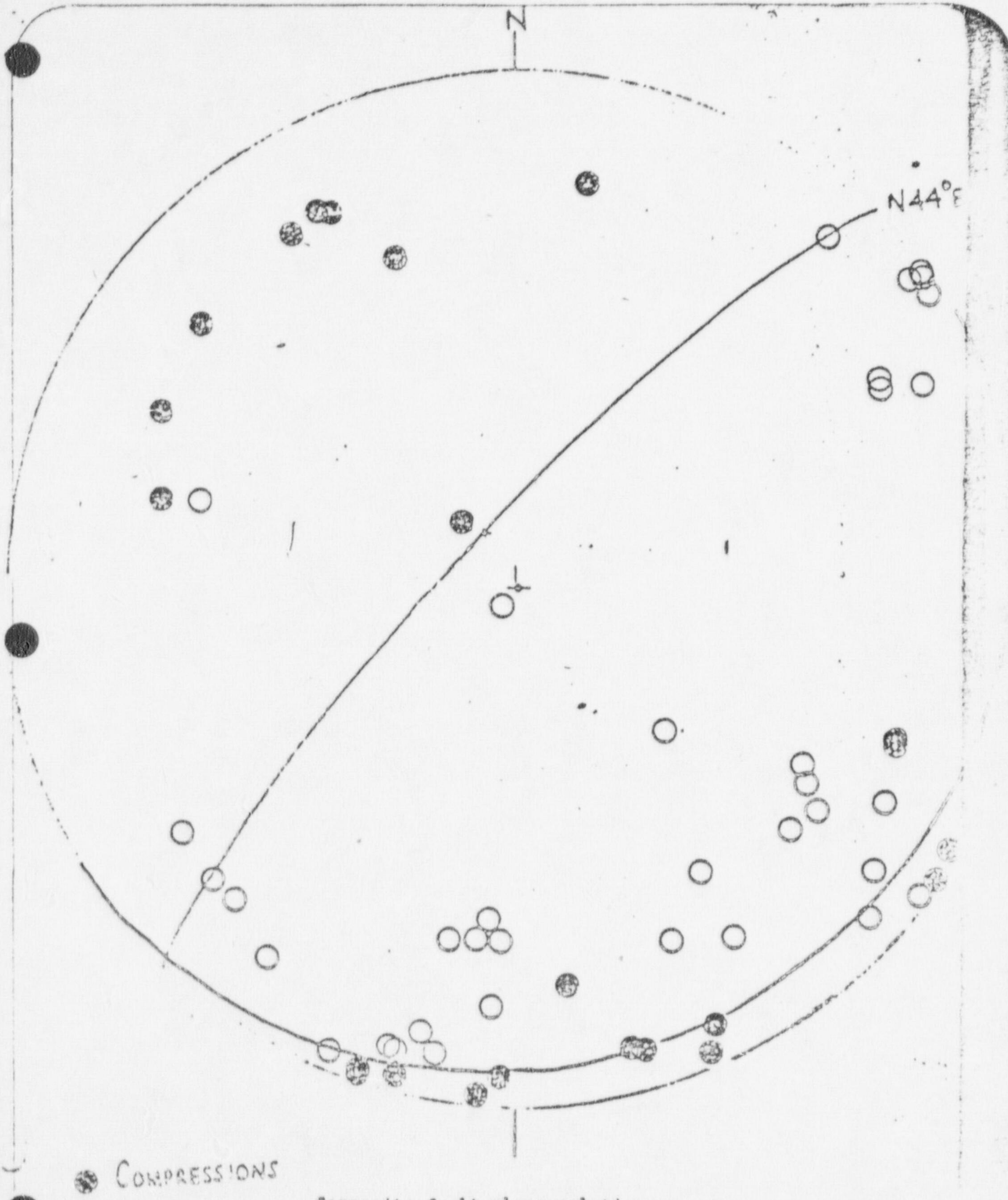


Fig. 8 VERTICAL CROSS SECTION ALONG AA'



Composite fault plane solution -equal area lower hemispher
Only events lying along a plane indicated in Fig. 4 were

A-237

CEA GUIDE TUB

CHRONOLOGY OF ACTIONS

12/14/77	Cracks observed in Control Element Assembling (CEA) guide tube during fuel inspection program at Millstone 2.
12/15/77	Northeast Nuclear Energy Company (NNECO) and Combustion Engineering (CE) met with NRC engineers presenting preliminary findings. We request general meeting to evaluate the guide tube problem.
12/19/77	Representatives from Baltimore Gas and Electric (BG&E), Florida Power and Light (FP&L), Maine Yankee Atomic Power Company (MYAPCO), NNECO, Omaha Public Power District (OPPD), and CE met with NRC to present Millstone 2 data on CEA guide tube wear.
12/20/77	All effected CE licensees with operating facsilities were notified to: (1) insert 1/7 of all CEA's, not fully inserted, at least 10 steps each day, and (2) provide request for amendment to allow CEA insertion beyond the present "full out" position.
12/21/77	Engineering Branch (EB) contracts with Idaho National Engineering Laboratory to provide confirmatory analyses of the guide tube problem.
12/23/77	CE provides proprietary version of slides used at 12/19/77 meeting with the staff.
12/27/77	Received request for Technical Specification (TS) changes from all operating CE licensees.
01/04/78	BG&E provides information on necessary inhibits to allow CEA insertion of three inches. Also excepts proposed TS telecopied by staff.
01/06/78	Amendments for Calvert Cliffs 1 and 2, Ft. Calhoun, Maine Yankee, and St. Lucie were issued authorizing CEA insertion three inches.
01/12/78	Second meeting with facilities and CE was held. CE presented more ECT inspection data and described the possible temporary fix; sleeving of selected CEA guide tubes.
01/16/78	DOR request I&E to ensure the the intent of 1/6/78 amendment letter is implemented at each facility.
01/17/78	CE provides non-proprietary version of slides used at 12/19/77 meeting with the staff.

A238

01/18/78	Twenty-day-letters sent to BG&E, FP&L, MYAPCO and OPPD requiring, pursuant to 10CFR50.54(f), that "justification that excessive guide tube wear does not exist in your facility, or, if unable to assure that such wear does not exist, justification that continued operation of the facility would not create undue risk..." NNECO was sent a similar letter requiring "additional justification for return to operation...."
01/27/78	Engineering Branch (EB) contracts with Battelle Northwest Engineering Laboratory to provide confirmatory analyses of the guide tube problem.
02/03/78	DOR provides operating experience Memorandum No. 11 on guide tube wear to DSS.
02/06/78	The staff prenoticed the "resolution of the operational problems related to CEA guide tube wear prior to return to power operation: for Calvert Cliffs 1 and Millstone 2.
02/14/78	All responses to 20-day letters received. (Note: Some responses late due to adverse weather condition.) BG&E and FP&L submited CEN-79-P for the operating reactors, Calvert Cliffs 2 and St. Lucie. NNECO references CEN-82-P for return to operation of Millstone 2. (CEN-82-P references Calfert Cliffs 1, but it was not submitted by BG&E.) OPPD documents "that there is no significant guide tube wear in the Fort Calhoun Station Unit 1 fuel assemblies." MYAPCO submittal states that "the amount of guide tube wear on Maine Yankee fuel that is similar to the fuel currently residing in the reactor is significantly less than the wear experienced by fuel in earlier cycles.
02/15/78	BG&E met with the staff to inform us that they plan to sleeve the CEA guide tubes in 110 fuel assemblies.
02/17/78	BG&E submitts CEN-83(B)-P on "Calvert Cliffs Unit No. 1 Reactor Operation with Modified CEA Guide Tubes."
02/17/78	CE recommends that BG&E place a hold on the handling of 14 selected fuel assemblies in the Calvert Cliffs 1 reactor.
02/21/78	BG&E and CE met with staff to answer our concerns on sleeving. At conclusion, the staff informed BG&E that operation with sleeved CEA guide tube involves an unreviewed safety question.
02/24/78	NNECO and CE met with the staff to present their plan to sleeve approximately 35 guide tubes. The staff notified them that operation with a sleeved CEA guide tubes involves an unreviewed safety question.

02/27/78 02/28/78	FP&L and BG&E, respectively, notified the staff that "preliminary results of the CE analysis of the more severely worn assemblies identified to date indicate that the stress criteria established in some guide tube during the limiting seismic excitation (SSE)." They also state that test program "results continue to support the conclusion that guide tube wear will not prevent CEA's from inserting following an SEE.
03/08/78	NNECO provides additional information on CEA guide tube wear including Amendment 1 to CEN-79-P, CEN-80(N)-P, CEN-82-P, and CEN-83(B)-P.
03/08/78	NNECO requires an amendment to authorize operation of Misstone 2 with sleeves installed in the CEA guide tubes. The letter also transmits CEN-80(N)-P, "Millstone 2 Reactor Operation with Modified CEA Guide Tubes."
03/15/78	NNECO submits additional information on CEA guide tube wear including Amendment 2-P to CEN-79-P, CEN-80(N)-P, and CEN-83(B)-P and sleeving procedures used at Calvert Cliffs 1 and Millstone 2.
03/16/78	NNECO responds to staff questions and submits Amendment 1 to CEN-79-P, CEN-80(N)-P, CEN-82-P, and CEN-83(B)-P.
03/16/78	BG&E response to staff questions, including Amendment 1 to CEN-79-P, CEN-80(N)-P, CEN-82-P, and CEN-83(B)-P.
03/17/78	BG&E request for amendment to operate with sleeved CEA guide tubes and to remove all part length CEA plus response to staff questions, includes Amendment 2-P to CEN-79-P, CEN-80(N)-P, CEN-82-P, and CEN-83(B)-P, "Additional Information on Guide Tube Wear."
03/20/78	BG&E submits revised reload analyses for Cyle 3 operation.
03/20/78	BG&E responds to staff questions including results of visual examinations of sleeved guide tubes and worse case wear observed at Calvert Cliffs 1.
03/29/78	NRC provides MYAPCO with questions on location of worse possible wear of the guide tubes, fracture mechanics, type analysis, and revised seismic analysis.
03/31/78	NRC issues Amendment 32 to BG&E authorizing Cycle 3 operation for Calvert Cliffs 1 with sleeved CEA guide tubes.

A-240

CE REACTORS

PALISADES	-	CRUCIFORM CONTROL ELEMENTS
FT. CALHOUN	-	LIMITED WEAR OBSERVED
CALVERT CLIFFS 1 AND MILLSTONE 2	-	EXTENSIVE SLEEVING OF NEW AND WORN GUIDE TUBES
ST. LUCIE	-	SHUTDOWN FOR RELOAD INSPECTION IN PROGRESS
CALVERT CLIFFS 2 AND MAINE YANKEE	-	CONTINUING REVIEW

CONTINUING STAFF REVIEW

- ST. LUCIE ECT DATA WITH CEA'S INSERTED 3 INCHES FOR 3 MONTHS.

- ADDITIONAL DATA FROM MAINE YANKEE ON RODDED FUEL ASSEMBLIES VS. RESIDENCE TIME AND LOCATION.

- DECISION ON ADDITIONAL LICENSING ACTIONS ON OPERATING REACTORS.

FUEL ASSEMBLY GUIDE TUBE INTEGRITY

I. PROBLEM

II. DESCRIPTION OF GUIDE TUBES

III. SAFETY AND DESIGN CONSIDERATIONS

A. LOADINGS
B. COOLABILITY
C. SCRAMABILITY

IV. WEAR OBSERVATIONS

A. NON-DESTRUCTIVE EXAMINATIONS
B. DESTRUCTIVE EXAMINATIONS

V. INTERIM FIXES

A. SLEEVING

1. DESCRIPTION OF SLEEVE
2. STRUCTURAL ANALYSES
3. SLEEVING PROCEDURE
4. SAFETY CONSIDERATIONS
5. TESTING

B. 3" INSERTION

VI. BASES FOR CONTINUED OPERATION

A. PLANTS WITH SLEEVED AND UNSLEEVED GUIDE TUBES

B. PLANTS WITH UNSLEEVED GUIDE TUBES

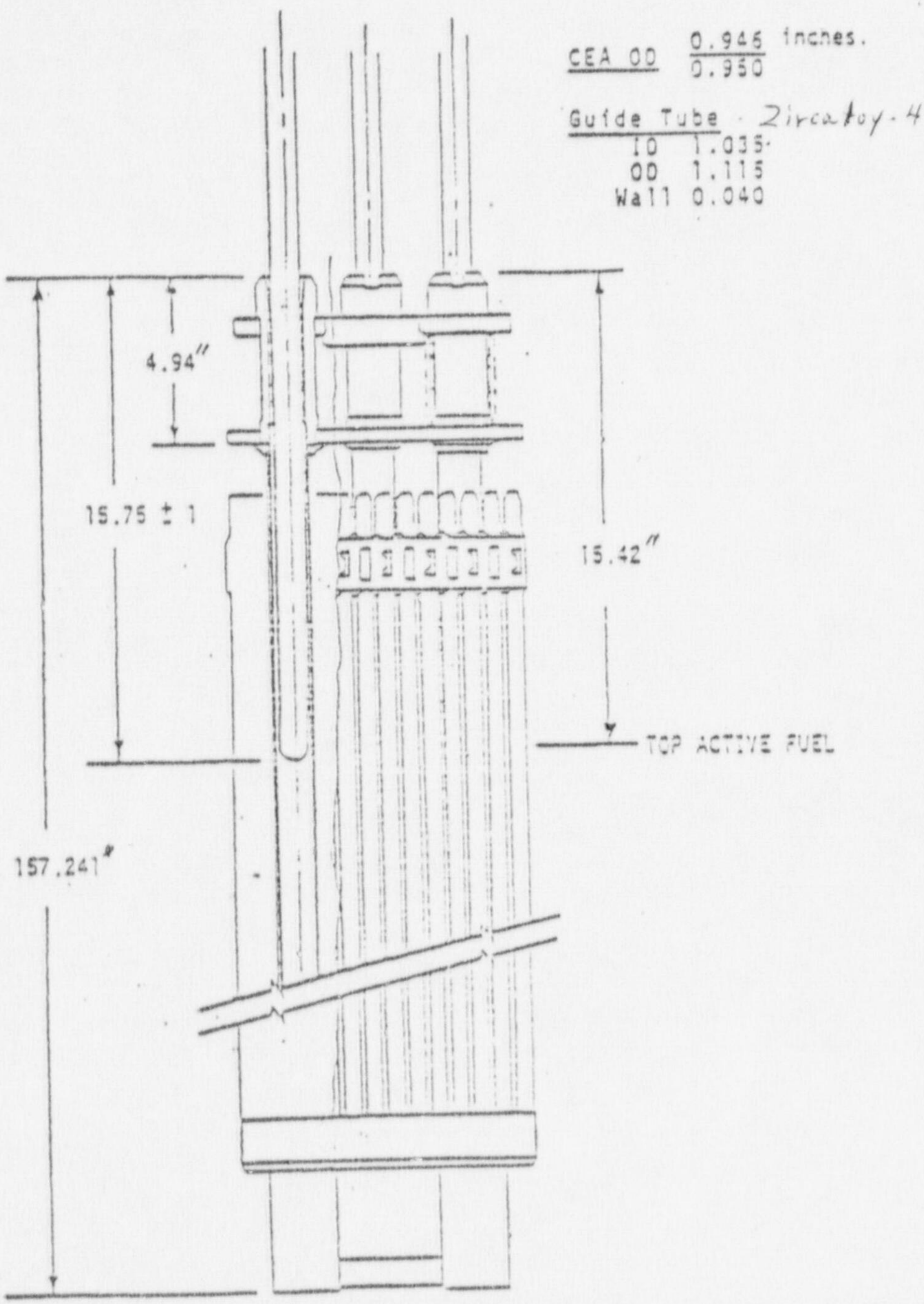
VII. SUSCEPTIBILITY OF OTHER NSSS DESIGNS TO GUIDE TUBE WEAR

FUEL ASSEMBLY GUIDE TUBE INTEGRITY

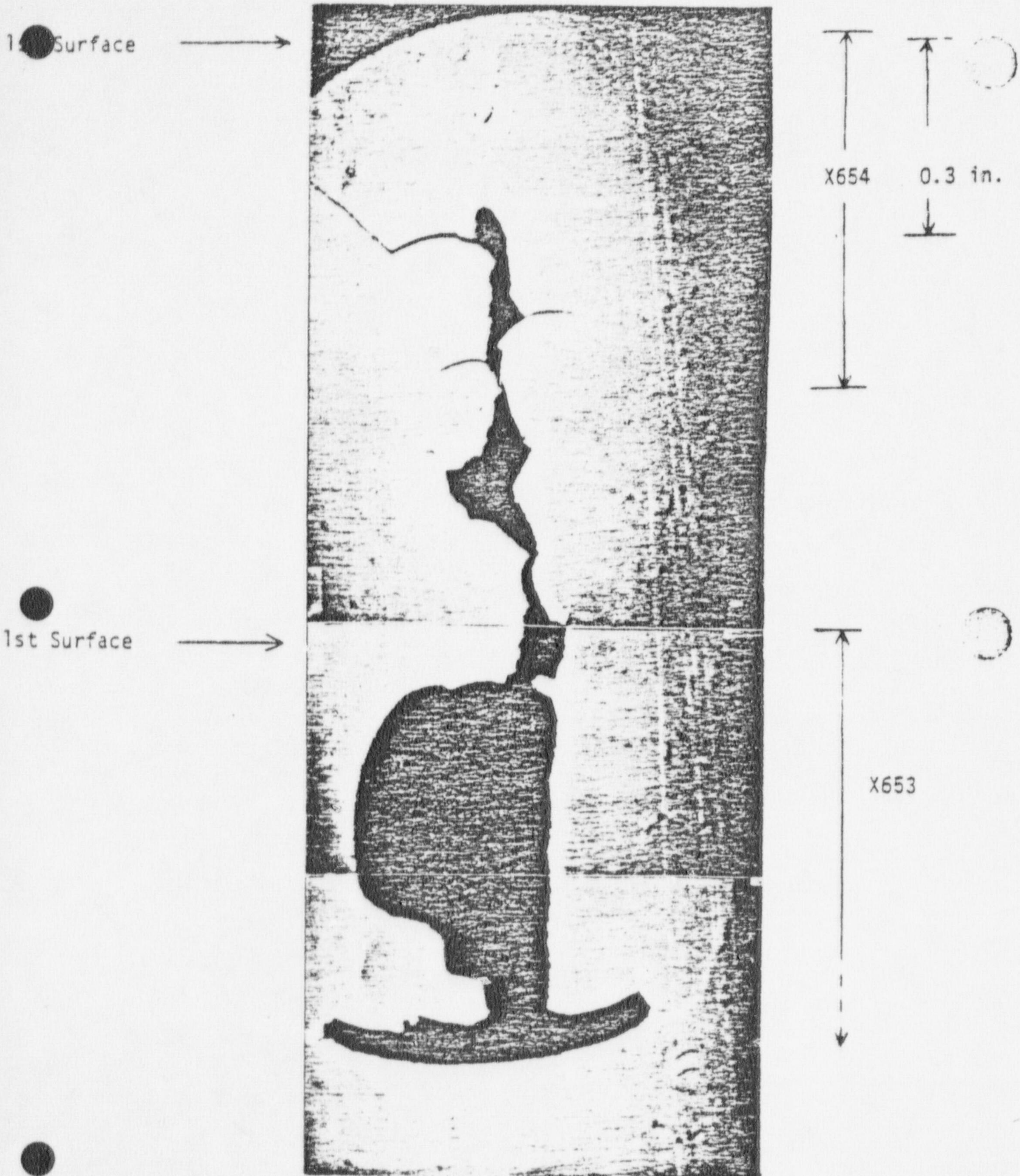
I. PROBLEM

A. On December 13, 1977, cracks were found in the control element assembly guide tubes of three fuel assemblies at Millstone Point Unit No. 2.

II. DESCRIPTION OF GUIDE TUBES



CEA AXIAL POSITIONING

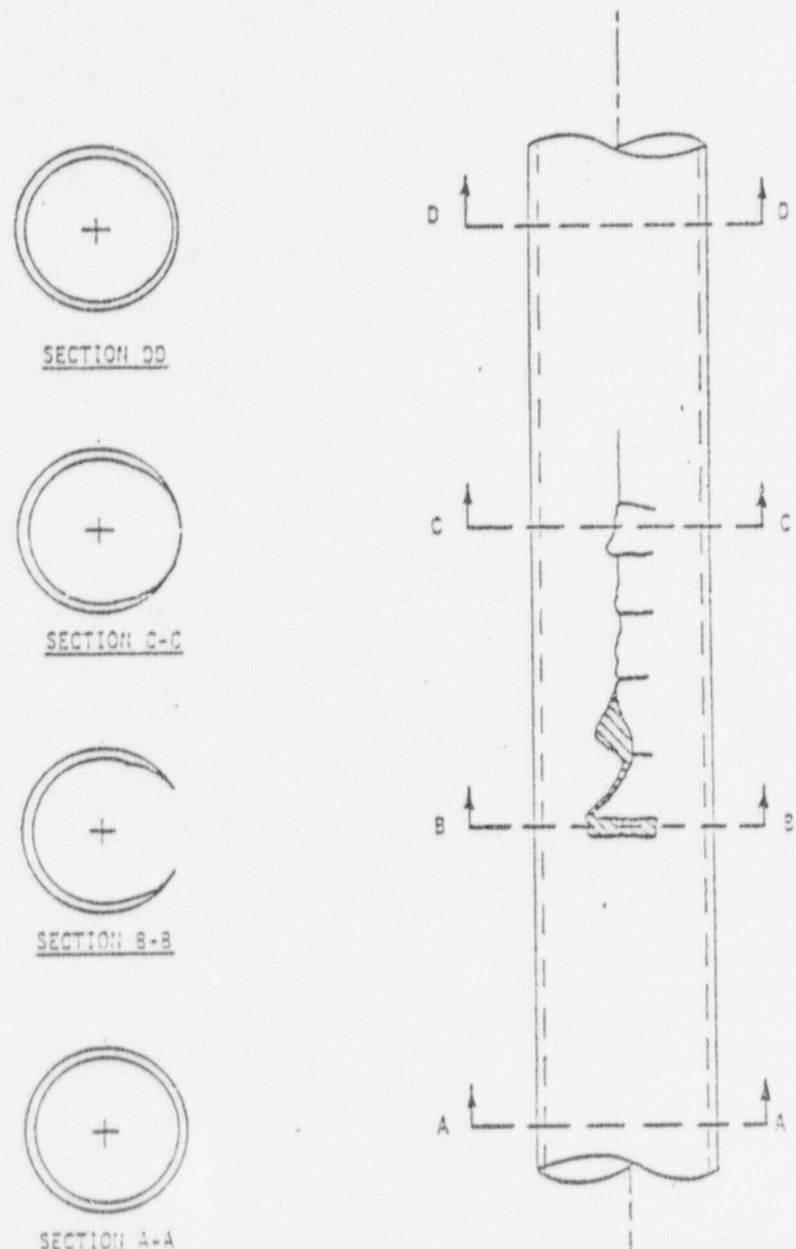


C-E 3-5

OD Appearance of SE Guide Tube Wear

C-E 21-22

Montage of SW Guide Tube ID Showing Wear Opening After Clamshelling



REPRESENTATION OF WARN CEA GUIDE TUBE

III. SAFETY AND DESIGN CONSIDERATIONS

A. SCRAMABILITY

B. COOLABLE GEOMETRY

IV. WEAR OBSERVATIONS

A. Non-destructive Examinations

1. Eddy Current Testing

2. Visual Inspection with Borescope and/or Periscope

B. Destructive Examinations

1. Metalographic Examination in Hot Cell Facility

F.250

EDDY CURRENT TESTING AT OPERATING COMBUSTION ENGINEERING REACTORS

	Number of Guide Tubes Tested	
Plant	Average Wear Probe	Azimuthal Wear Probe
Millstone Point #2	474	120
Maine Yankee	659	93
Calvert Cliffs #1	623	291
Fort Calhoun	140	0

1725

V. INTERIM SOLUTIONS

A. Repair of Worn Guide Tubes by Sleeving

1. Description of Sleeves

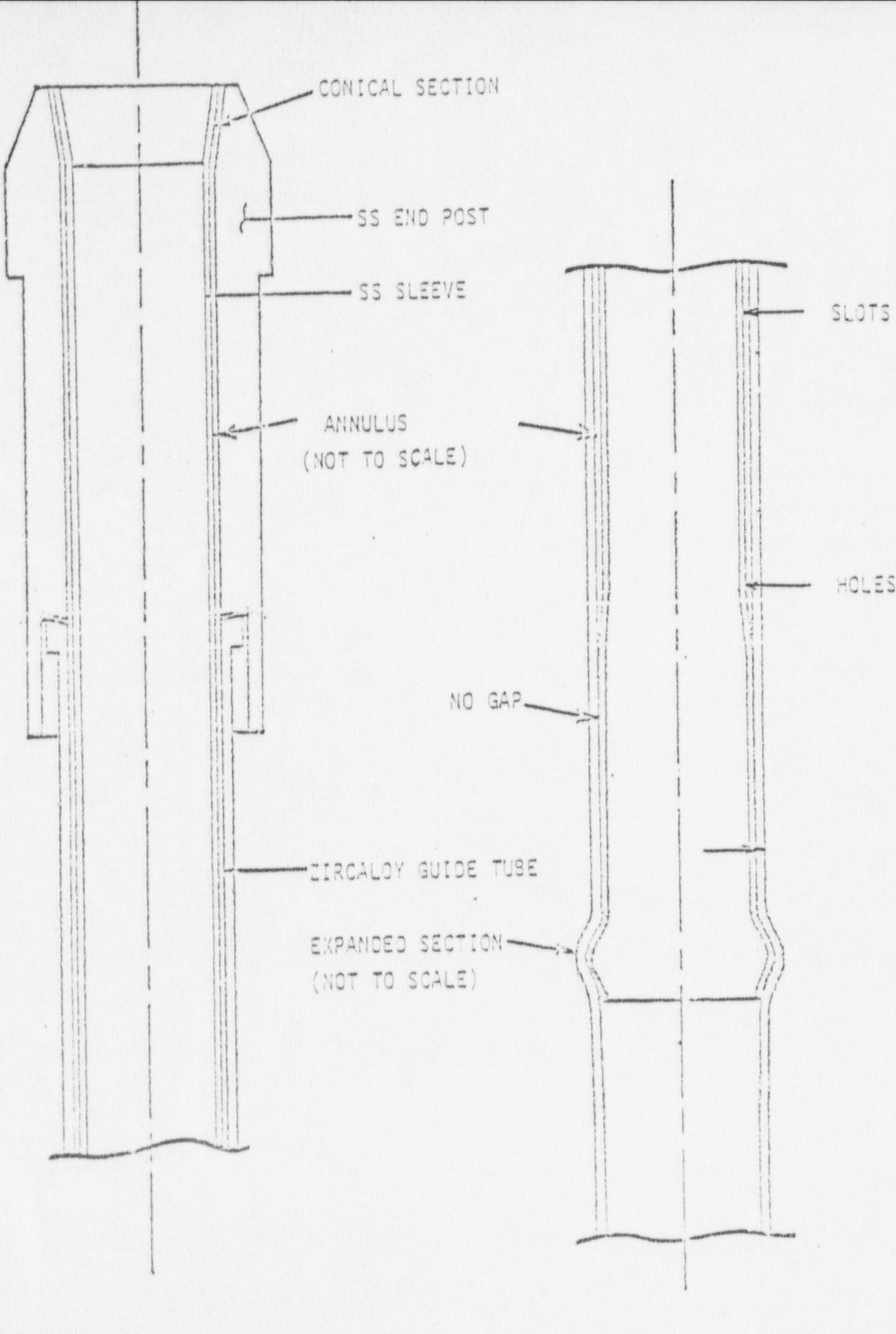
2. Structural Analyses

3. Sleeving Procedure

4. Safety Considerations

5. Testing of Sleeved Assemblies

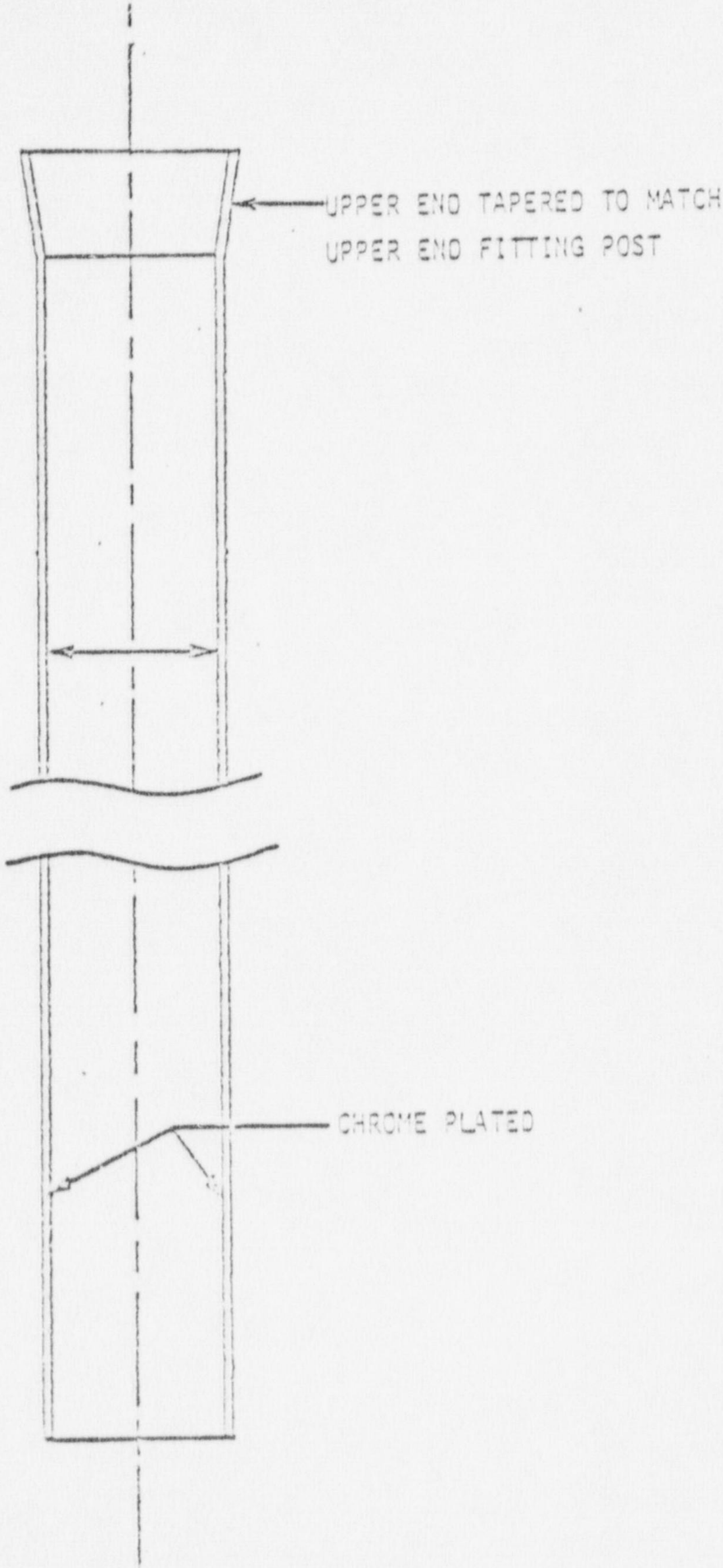
B. 3 Inch Insertion of Control Rods



FUEL ASSEMBLY WEAR SLEEVES

A-253

MATERIAL: CHROME PLATED 304 STAINLESS STEEL



A-254

GUIDE TUBE SLEEVE

QC CHECKS TO ENSURE PROPER SLEEVE INSTALLATION

1. Pull Out Test

2. Visual Inspection

3. I.D. Gauging Operations

4. Periscope Inspection for Cracks

VI. BASES FOR CONTINUED OPERATION

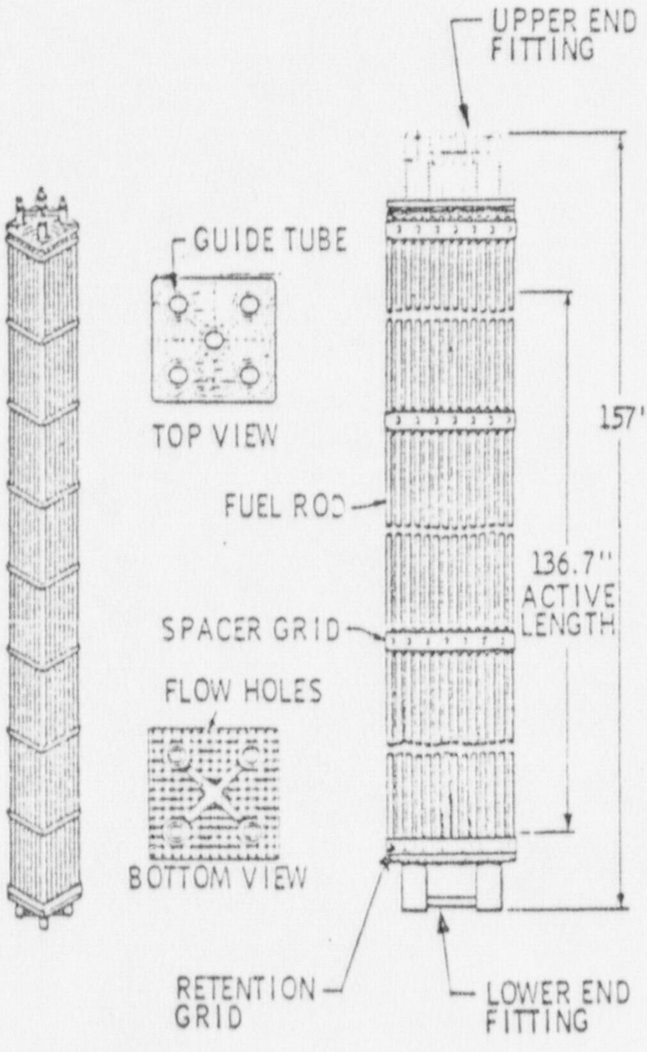
SLEEVING ACCEPTANCE

- STRENGTHENS WORN TUBES
- ELIMINATES FURTHER WEAR
  - NEW TUBE PROTECTION
  - STOPE WORN TUBE DAMAGE
- NO HINDRANCE TO CEA OPERATION
- NO SIGNIFICANT TEMPERATURE OR FLOW CHANGES
- ACCEPTABLE INSTALLATION PROCESS
- OPERATIONAL EVALUATION JUDGED ACCEPTABLE FOR AT LEAST ONE CORE LIFE
- IMPROVES RELOAD ASSEMBLY POSITIONING FLEXIBILITY

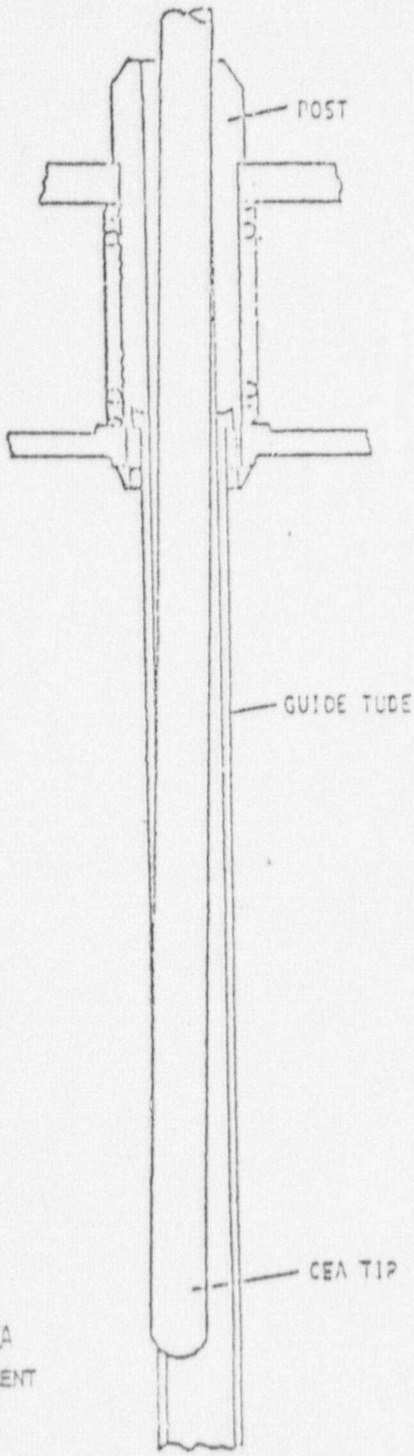
VI. BASES FOR CONTINUED OPERATION

CONTINUED UNSLEEVED OPERATION

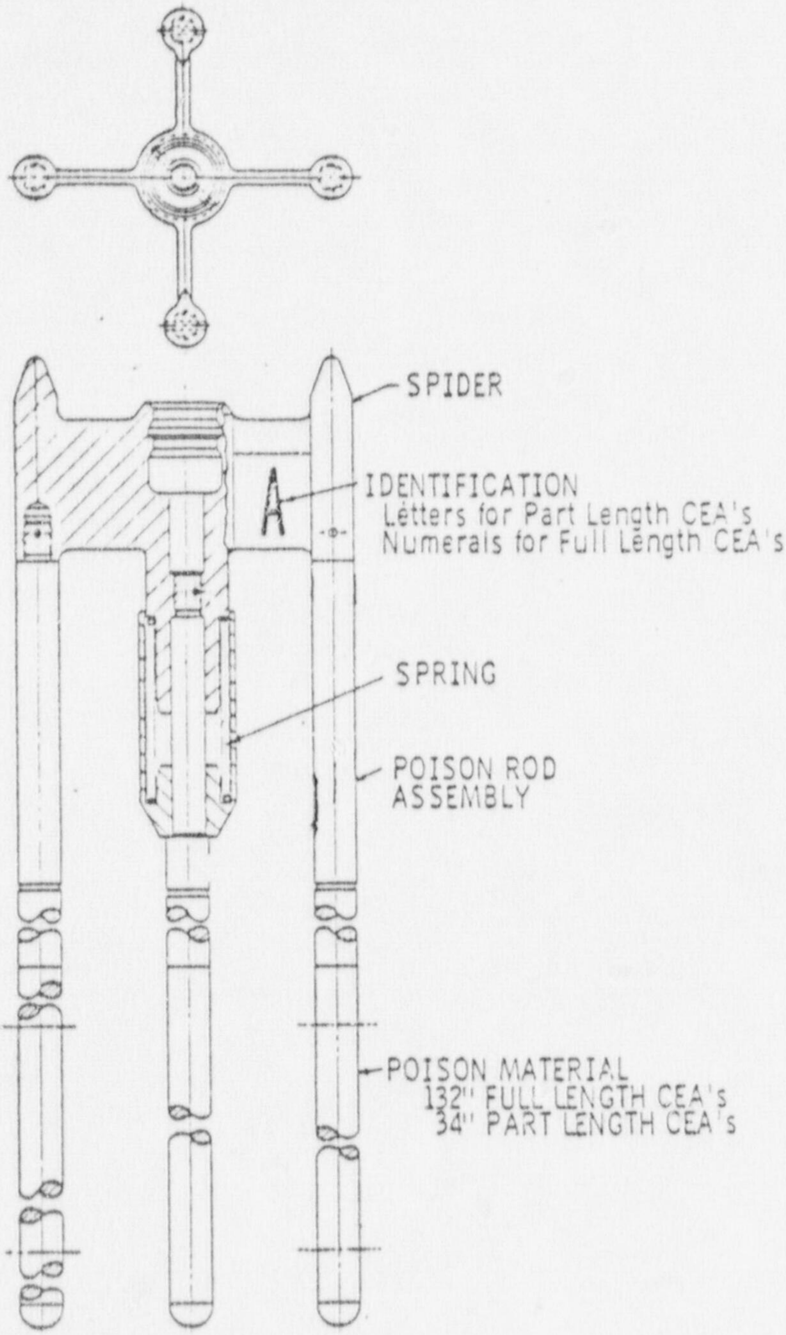
- CEA FULL-OUT POSITION 3 INCHES LOWER
  - ASSURE SCRAM
  - STOP WEAR AT PREVIOUS FULL-OUT LOCATION
  - CONTROL ROD CONFERS STIFFNESS TO TUBE
- POSITIVE RESULTS OF WORN TUBE SCRAM TESTS
- WORN TUBES CONTRIBUTE SUPPORT
  - WEAR IS LOCALIZED
  - NORMAL OPERATION LOADS ARE LOW
- SAFETY REQUIREMENTS CAN BE MET
- SHORT CYCLE OR SMALL CORE CONSIDERATIONS
  - SPECIFIC CORES



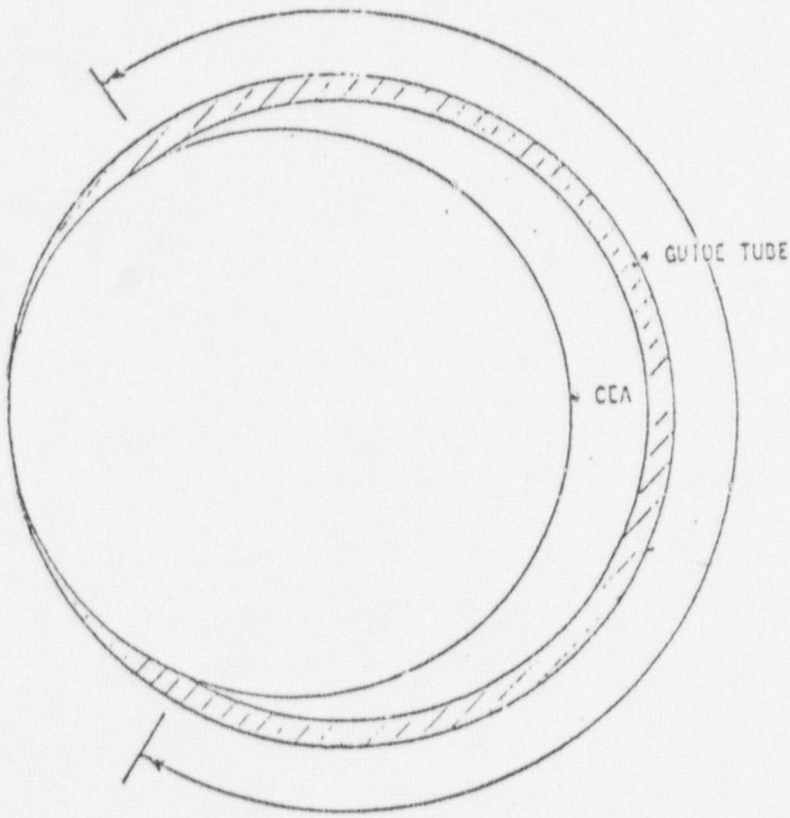
CE FUEL ASSEMBLY



REPRESENTATION OF CEA
MAXIMUM MISALIGHMENT



CONTROL ELEMENT ASSEMBLY (CEA)



REPRESENTATION OF CEA CONFIGURATION AT POINT OF WEAR

March 7, 1978

PRELIMINARY NOTIFICATION OF EVENT OR UNUSUAL OCCURRENCE -- PNO-78-41

This preliminary notification constitutes EARLY notice of an event of POSSIBLE safety or public interest significance. The information presented is as initially received without verification or evaluation and is basically all that is known by IE staff as of this date.

FACILITY: Florida Power Corporation
Crystal River Unit 3
Docket No. 50-302
Citrus County, Florida

APPENDIX XXIX
Crystal River 3: Burnable Poison Rod Assembly Failure

SUBJECT: FIXED BURNABLE POISON ROD COUPLING ASSEMBLY FAILURE

On March 6, while inspecting for possible loose parts in the steam generator upper plenum, Florida Power Corporation (FPC) found a Burnable Poison Rod Coupling Assembly. This coupling apparently broke loose from one of the Burnable Poison Rod Assemblies and was carried in the reactor coolant stream to the steam generator. Sixty-eight Burnable Poison Rod Assemblies are normally locked in fuel assemblies to provide flux shaping during early stages of the fuel cycle. The reactor is shut down and preparations are being made for reactor vessel head removal and inspection. The licensee has not identified any fuel damage. The extent of the shutdown is not known; however, Region II estimates a shutdown in excess of 4 weeks.

Media interest is anticipated due to a possible extended plant shutdown during the coal strike. A news release will be issued by the licensee. The State of Florida has been informed. An NRC inspector is onsite.

Region II (Atlanta) received notification of this occurrence by telephone from station management at 8:10 a.m. on March 7, 1978. This information is current as of 9:00 a.m. on March 7.

Contact: GKlingler, IE x28019 FNolan, IE x28019 JSniezek, IE x28019

Distribution: Transmitted H St 1:28
Chairman Hendrie
Commissioner Kennedy
Commissioner Gilinsky
Commissioner Bradford
S. J. Chilk, SECY
C. C. Kammerer, CA
(For Distribution)

Transmitted: MNBB 1:29
L. V. Gossick, EDO
S. H. Hanauer, EDO
J. J. Fouchard, PA
R. Hartfield, MIPC
R. G. Ryan, OSP
H. K. Shapar, ELD

P Bldg 1:30
E. G. Case, NRR

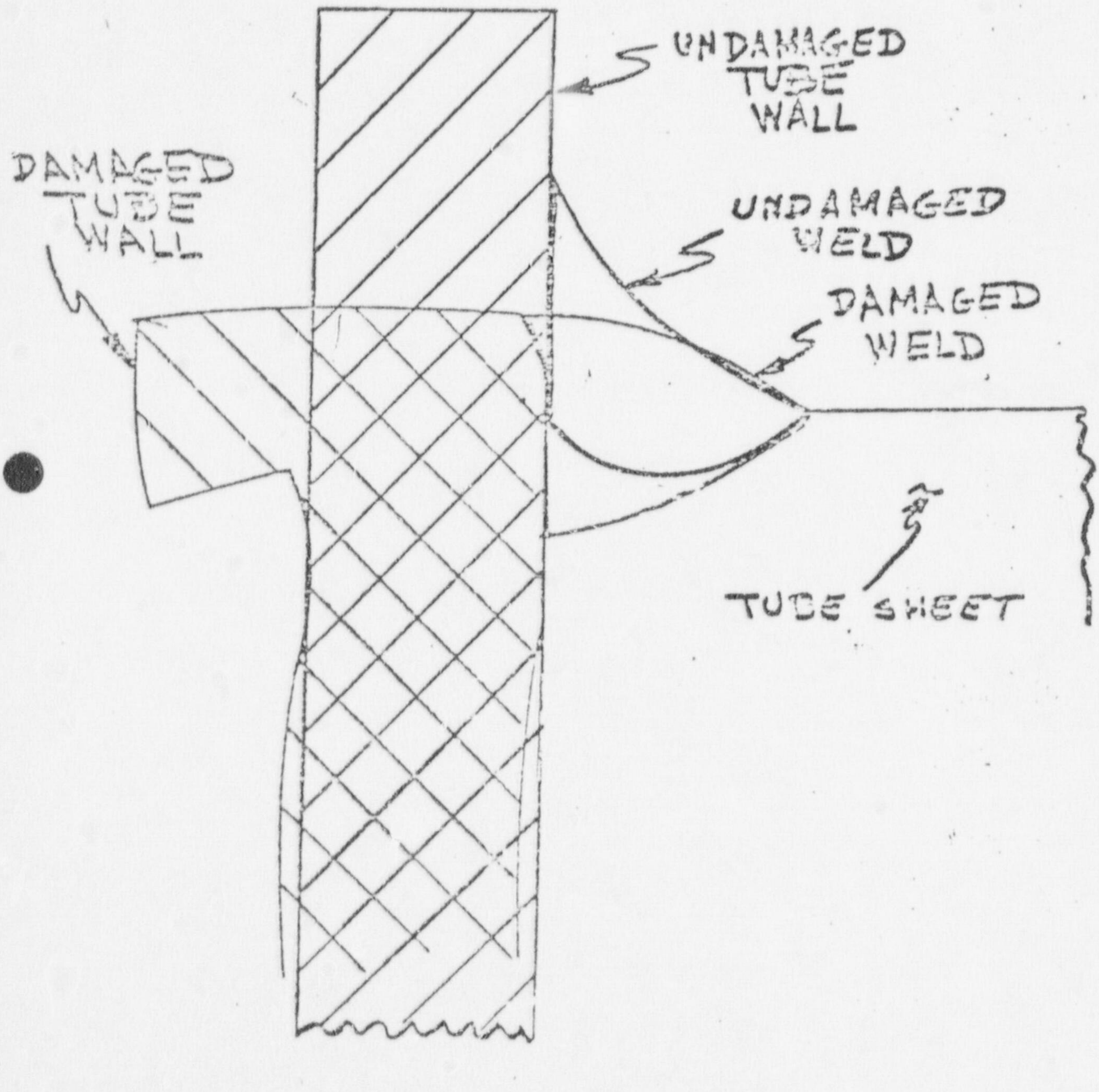
SS Bldg 1:31
C. V. Smith, NMSS

E. Volgenau, IE
Region II 1:33

(MAIL)
T. J. McTiernan, OIA
R. Minogue, SD

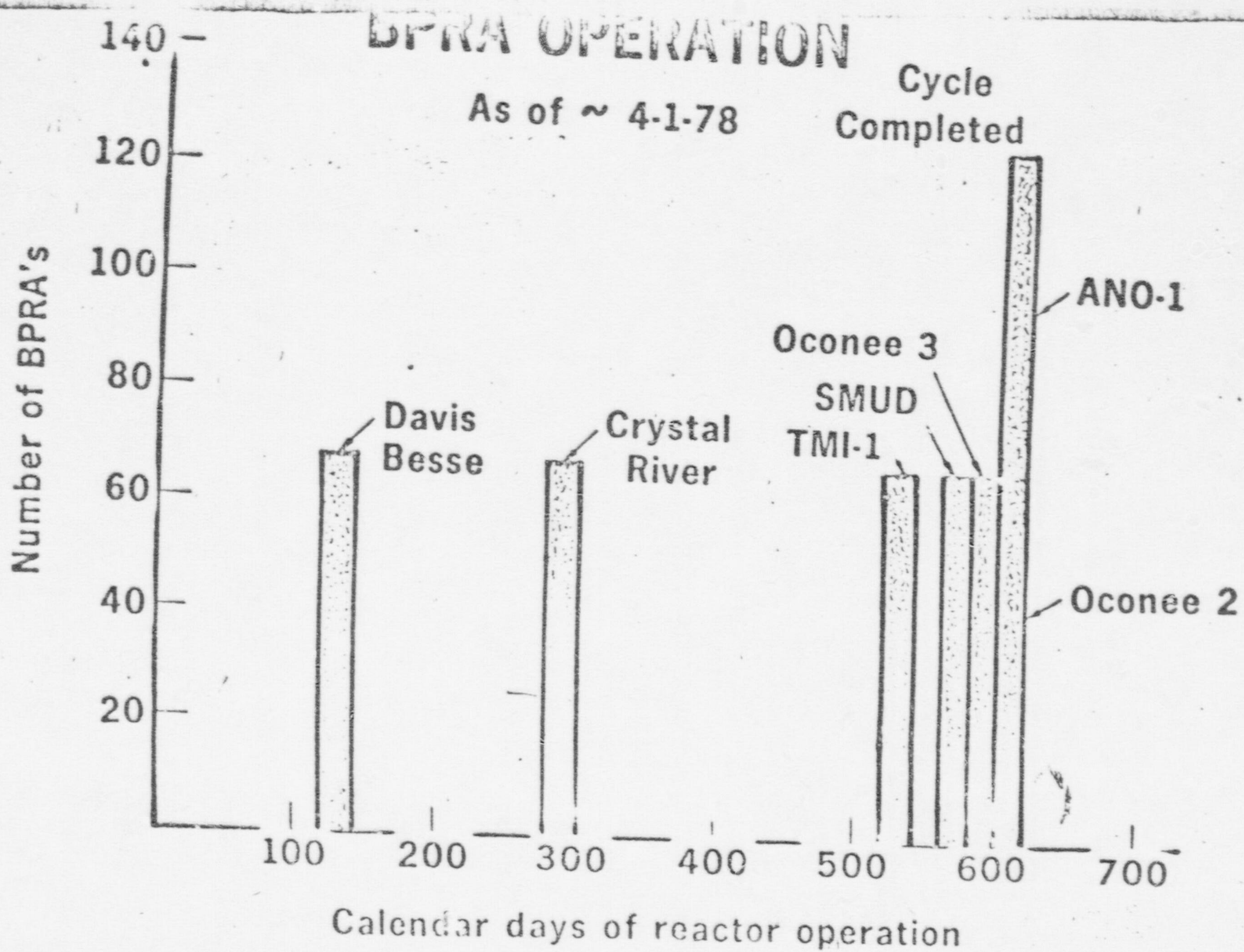
PRELIMINARY NOTIFICATION

1722



FLORIDA TUBE END DAMAGE

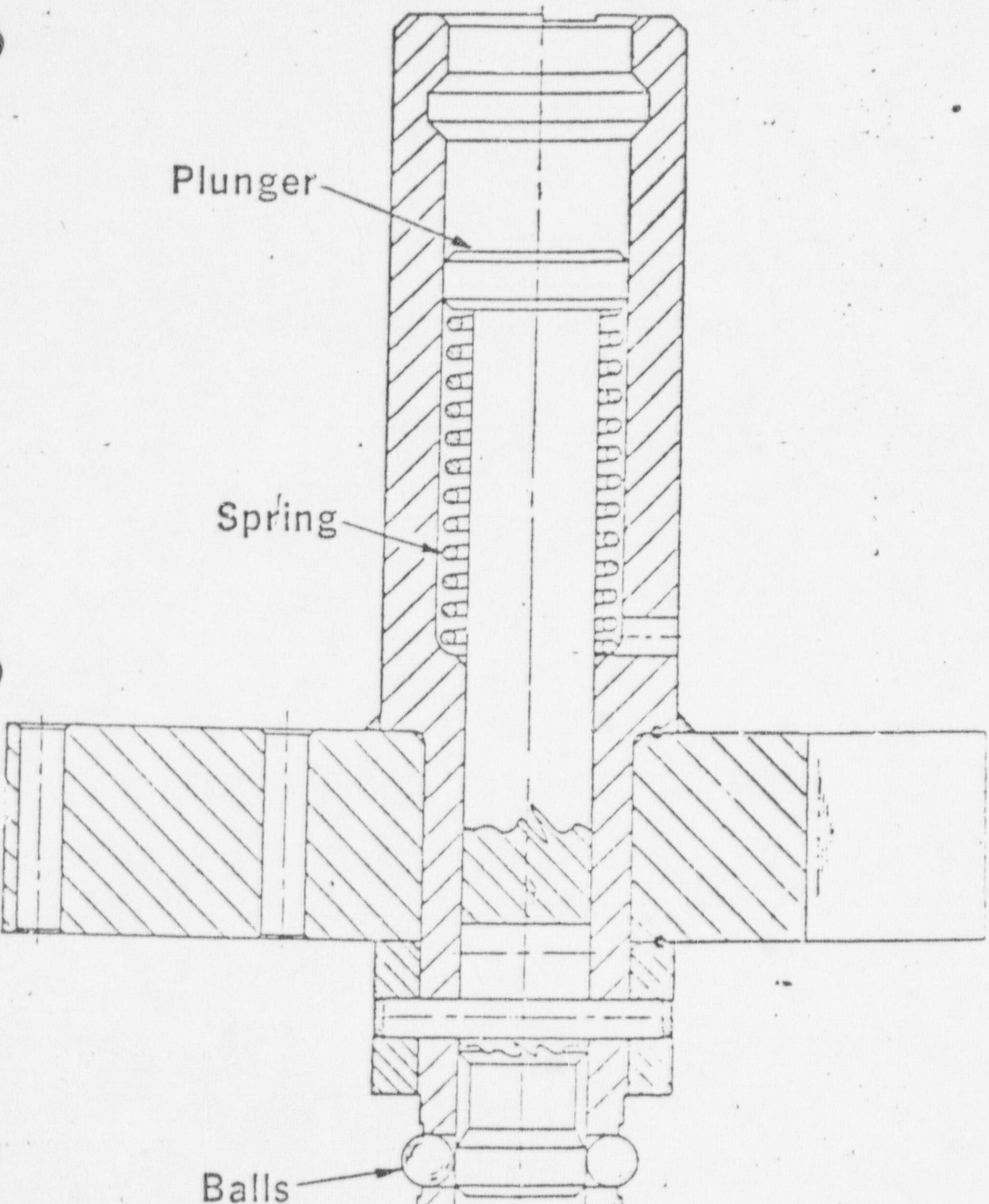
A263


BPRA OPERATION
As of ~ 4-1-78
Cycle Completed
Number of BPRA's
140
120
100
80
60
40
20
Davis Besse
Crystal River
TMI-1
SMUD
Oconee 3
ANO-1
Oconee 2
100
200
300
400
500
600
700
Calendar days of reactor operation

17-265

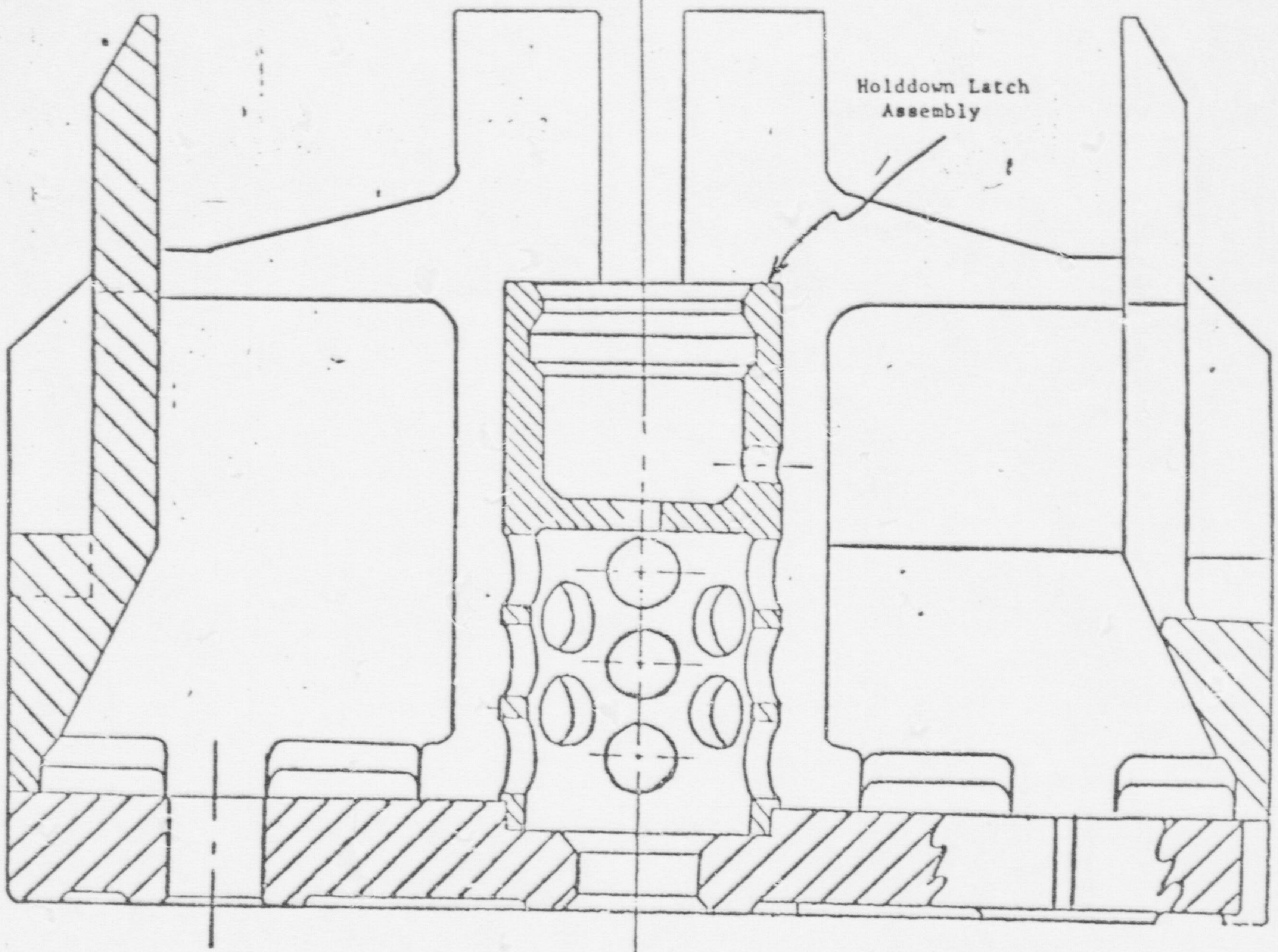
POSSIBLE CAUSES OF LOOSE BPRA AT CR-III

POSSIBLE CAUSE	RELATED INVESTIGATIONS
1. MANUFACTURING DEFICIENCY	• MANUFACTURING RECORDS BEING CHECKED
2. ASSEMBLY NOT LATCHED; FORCED INSERTION BY FUEL ASSEMBLY GRAPPLE; WEAR ENLARGEMENT OF GROOVE UNTIL SEPARATION	• FORCED BPRA INSERTION TEST • EXAMINATION OF HOLDDOWN LATCH ASSEMBLIES AT CR-III, AND OTHER SITES. • INSTRUMENTED BPRA IN HOT FLOW LOOP AT ALLIANCE RESEARCH CENTER
3. WEAR OF RETAINING LAND UNTIL SEPARATION	• SAME AS FOR ITEM 2 ABOVE



Coupling spider ass'y

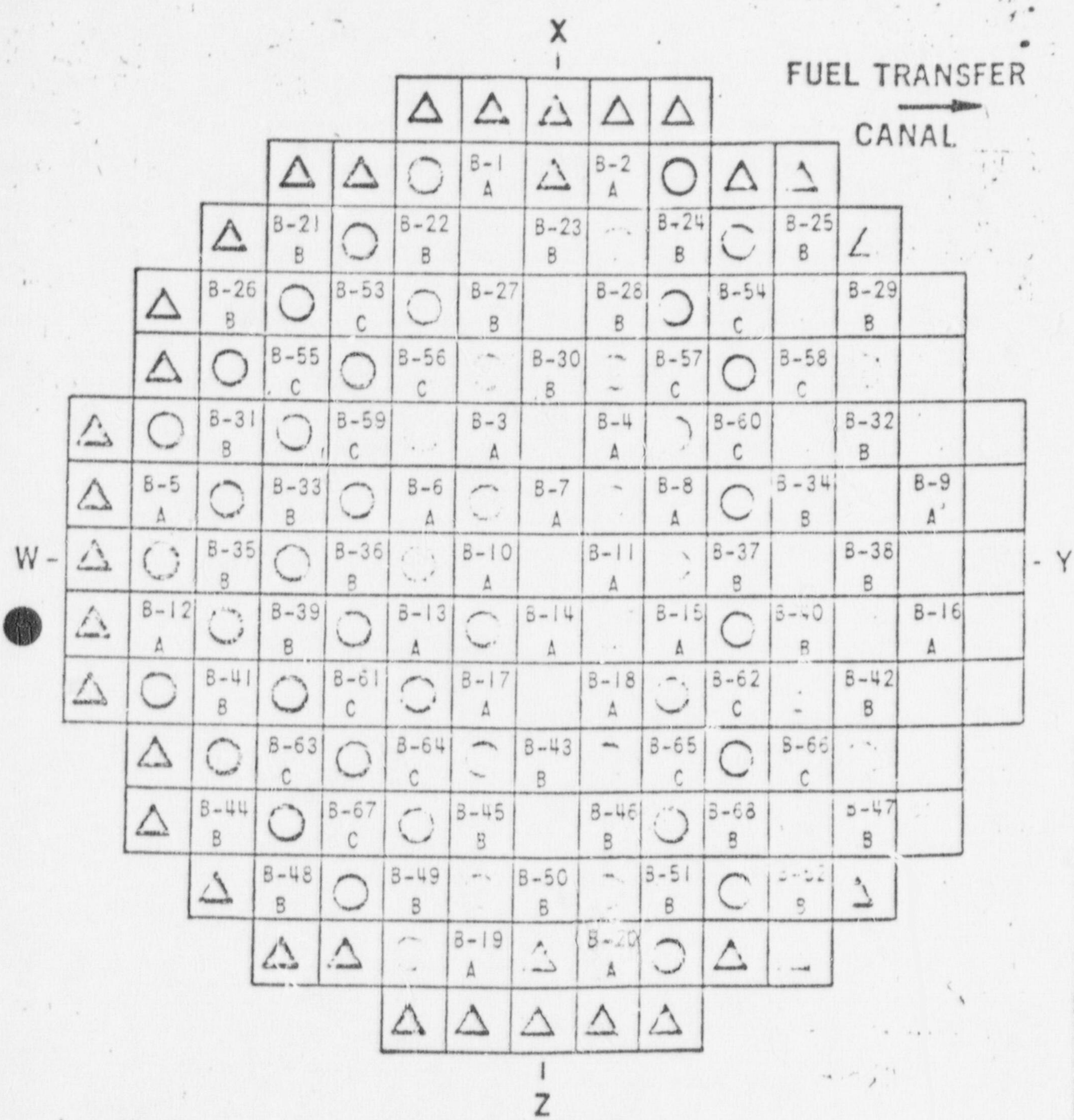
17-266



UPPER END FITTING AND HOLDDOWN LATCH ASSEMBLY

F7-267

FLORIDA BPRA LOADING

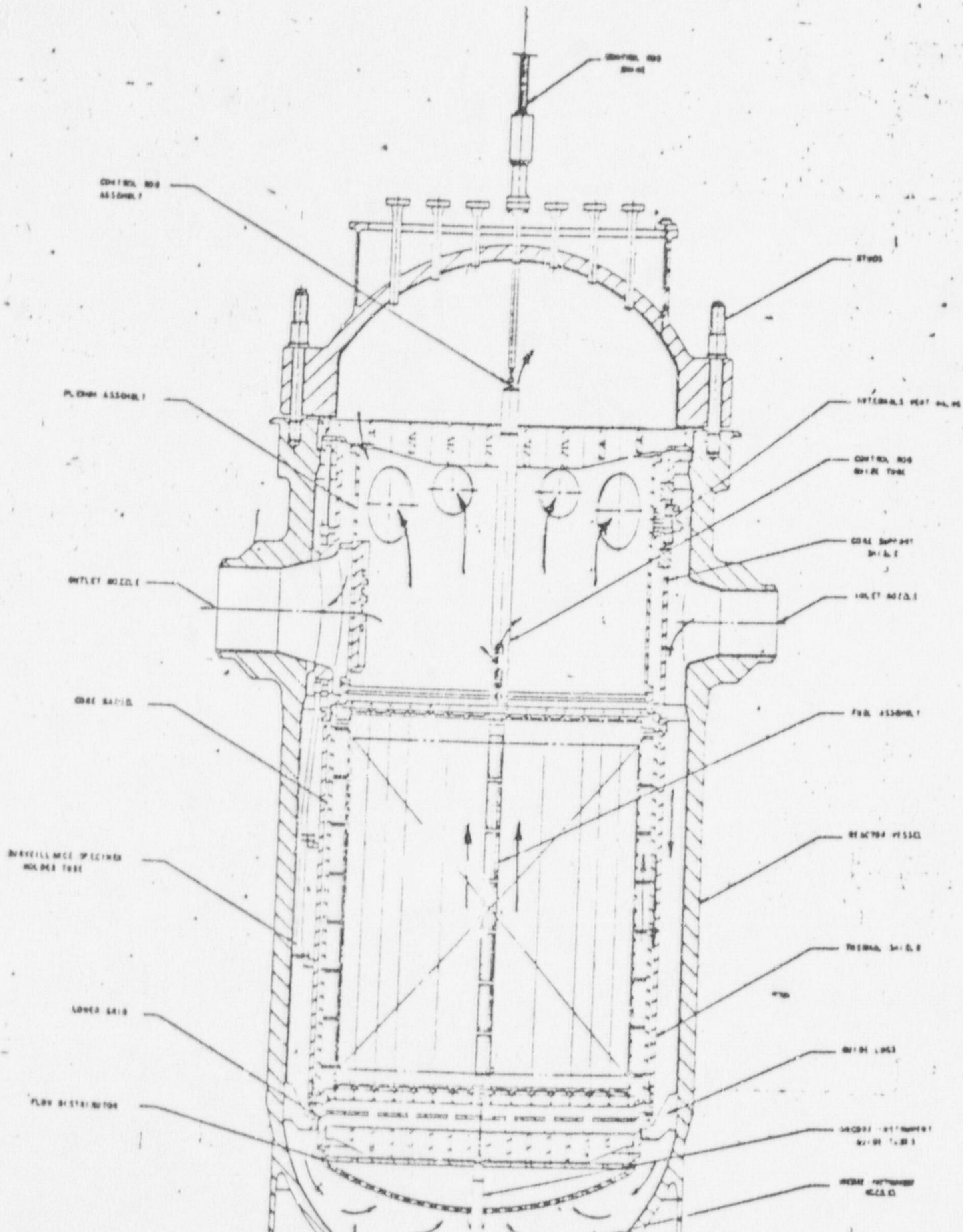


△ ORIFICE ROD ASSEMBLIES

COLUMN WELDMENT/CONTROL ROD POSITIONS

B-X BPRA POSITIONS

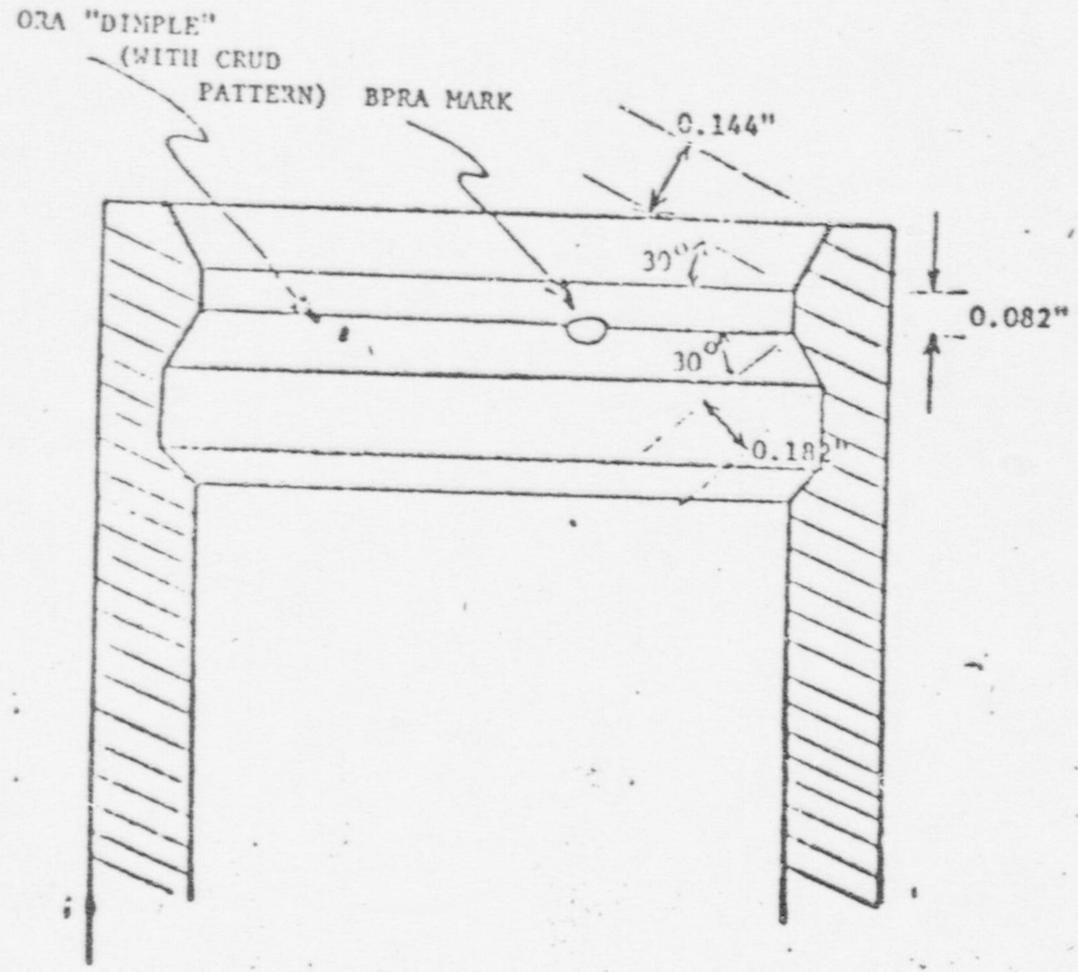
A-268


CONTROL ROD ASSEMBLY
PLENUM ASSEMBLY
OUTLET NOZZLE
SURVEILLANCE SPECIMEN HOLDER TUBE
LOWER GRID
FLOW DISTRIBUTOR
STUDS
INLET NOZZLE
FUEL ASSEMBLY
REACTOR VESSEL
THERMAL SHIELD

A-269

EVALUATION OF FUEL DAMAGE
CRYSTAL RIVER-III SITE WORK

TASK	STATUS	RESULTS
Plenum Area Inspection	Complete	• Observed B-52
Top of Core Inspection	Complete	• All control components appear seated except for B-47 and B-52
Accessible Rod/Debris Removal	Complete	
B-52 Site Inspection	Complete	• One Ball Missing
F.A. C35 and C37 Visual Inspection	Complete	• Grooves in holddown latch assembly • Remainder of assembly appears in good condition
Guide Tube Probe (C35 and C37)	Complete	• 23 out of 32 Guide Tubes Clear
Core Unloading	Complete	• Small debris in 84 assembly lower end fittings
Lower End Fitting Inspection		
Coupling Assembly Pull Test		
Holddown Latch Inspection		
Guide Tube Debris Removal (C35, C37)		
Guide Tube Inspection (C35, C37)		

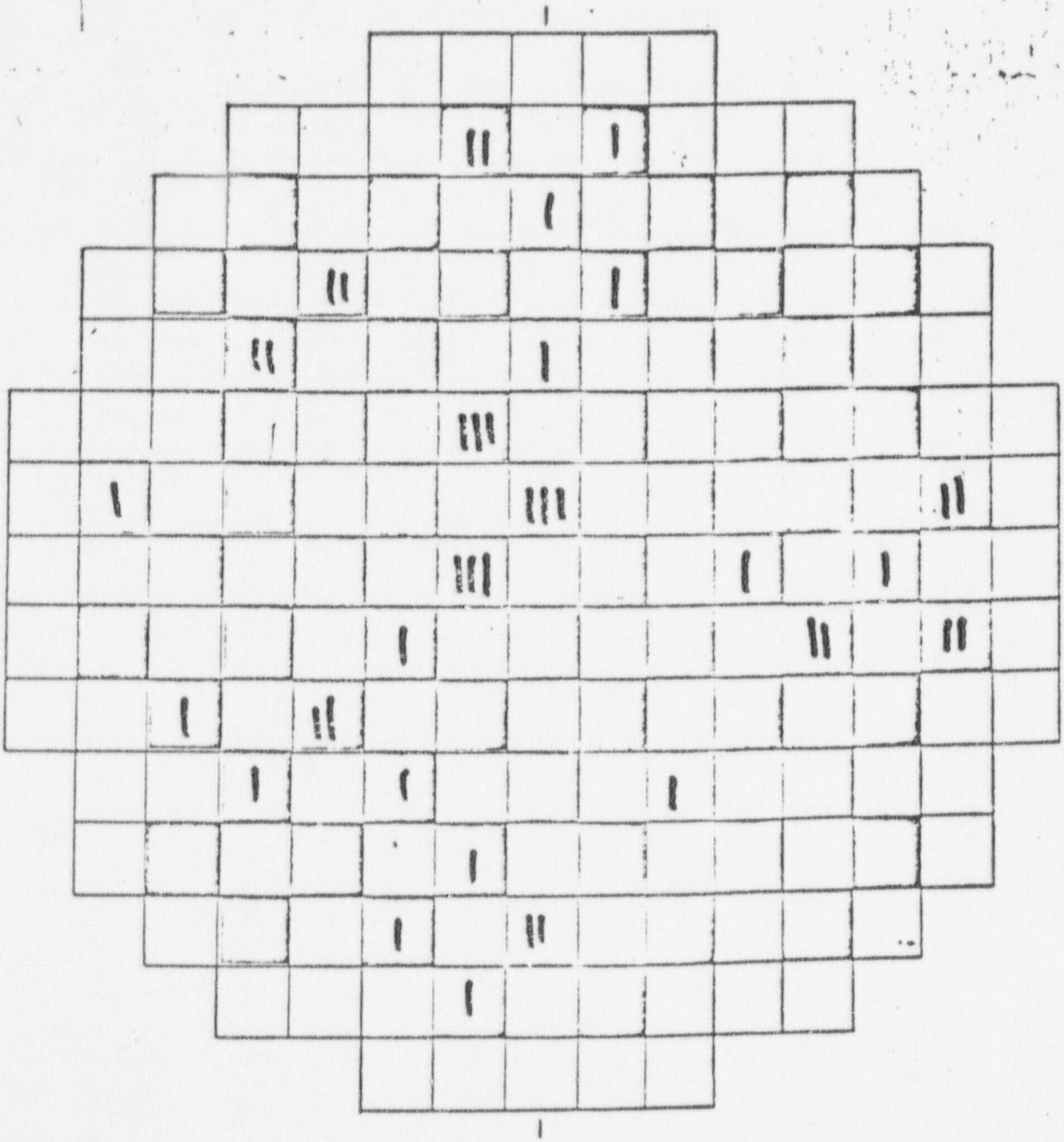


TYPICAL MARKINGS

SUBJECT NO.

1727

WEAR INSPECTION



BPRA LOCATION

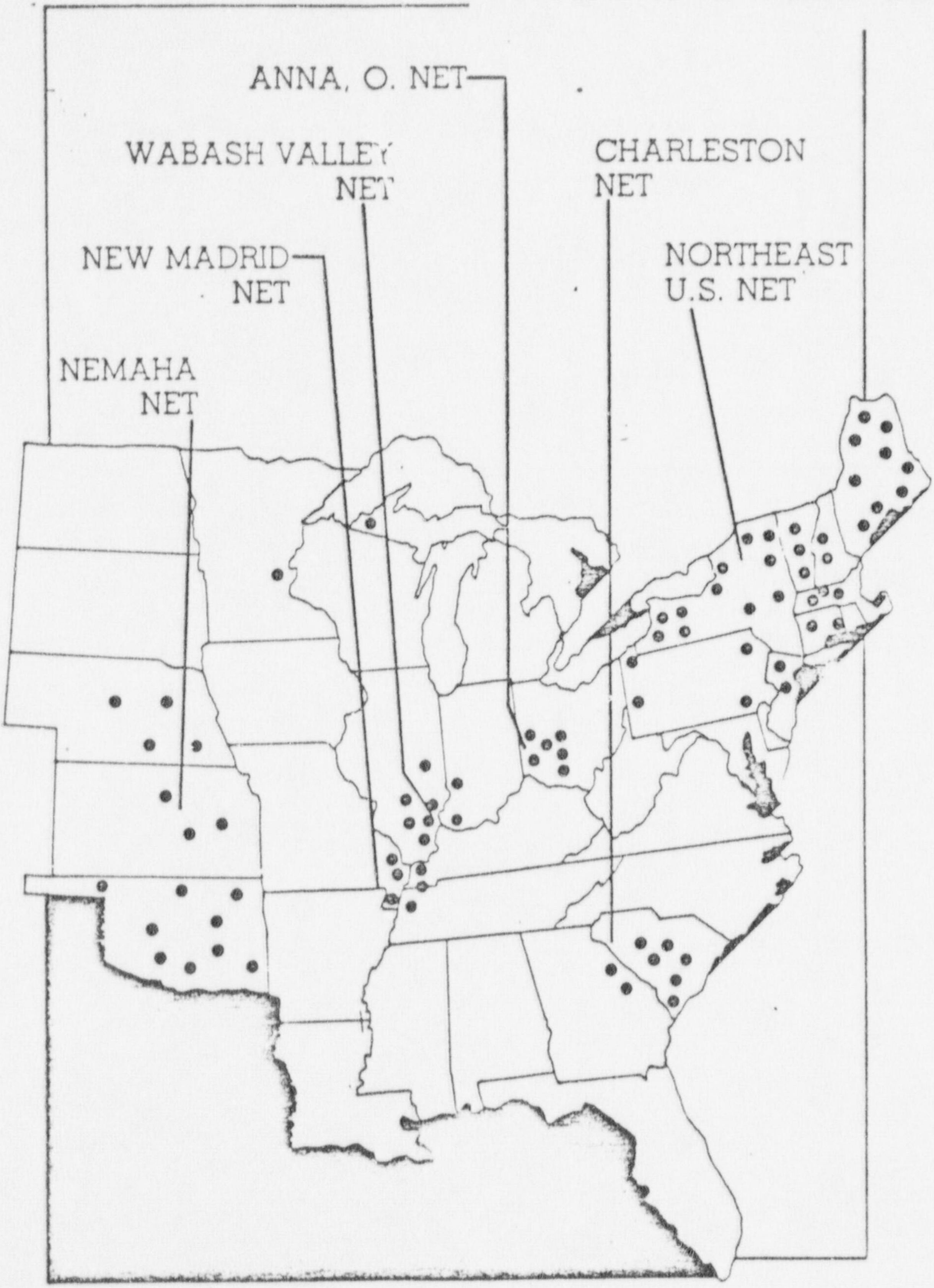
WEAR CATEGORY

I	< 2%
II	2 - 10%
III	> 10%

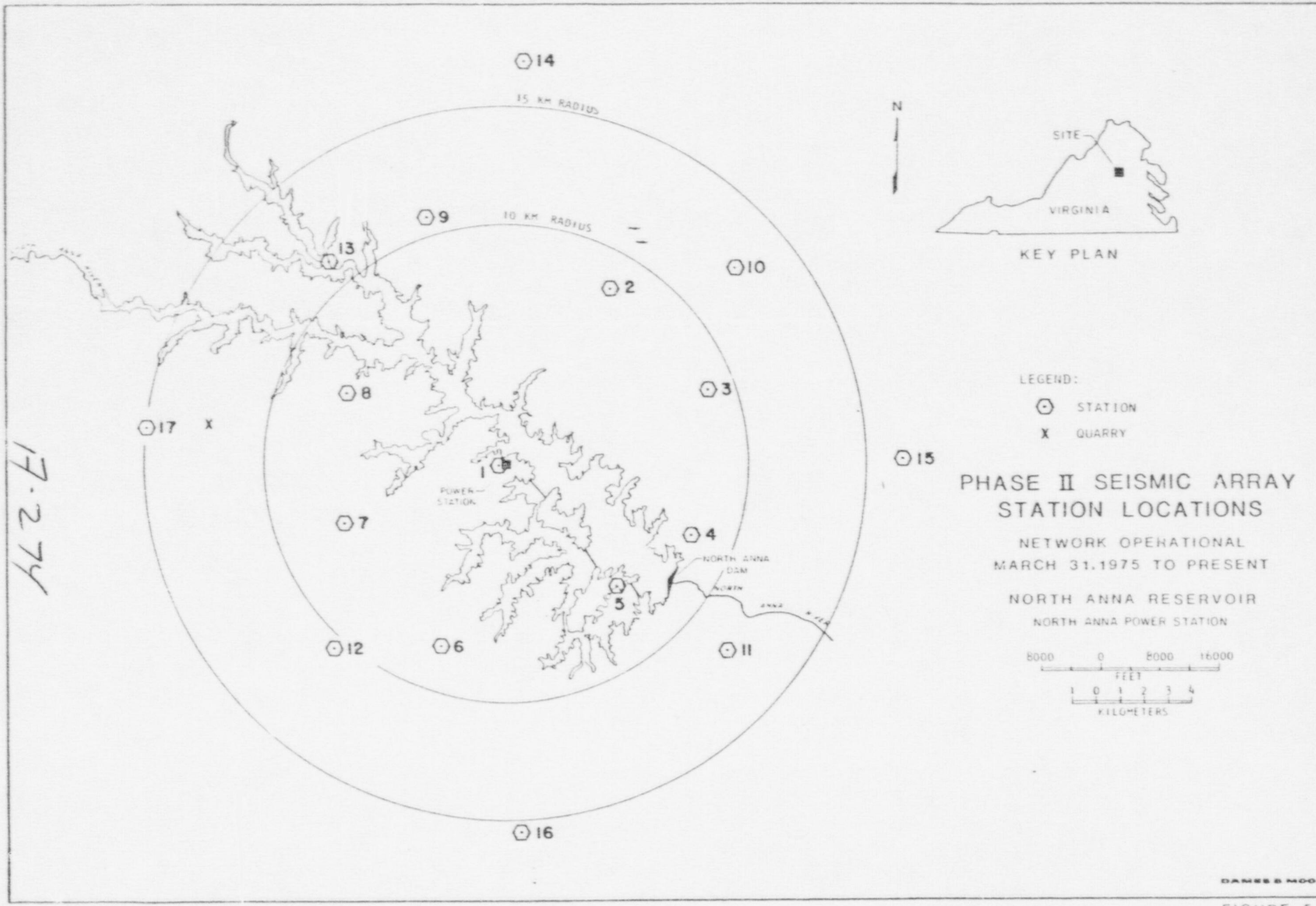
A-272

35-39

APPENDIX XXX
Microseismic Networks in Eastern U.S.



17-273



PHASE II SEISMIC ARRAY STATION LOCATIONS

NETWORK OPERATIONAL MARCH 31, 1975 TO PRESENT

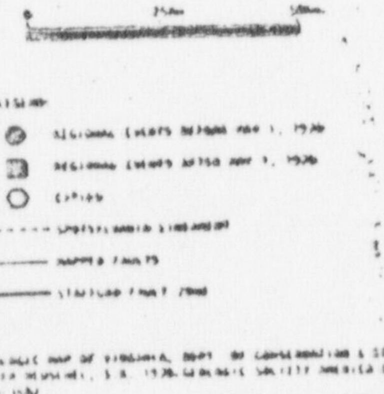
NORTH ANNA RESERVOIR

NORTH ANNA POWER STATION

FIGURE 5

REGIONAL EPICENTER MAP - Δ≤ 100 KM. $M_b \geq 2.0$

PERIOD: JANUARY 21, 1974, THROUGH AUGUST 1, 1977



O = C Va Net

17-275

1. Murphy

APPENDIX XXXI
Neutron Exposure: Request for Office of Nuclear Regulatory Research Program

UNITED STA
NUCLEAR REGULATO
WASHINGTON, D. C. 20555

April 3, 1978

MEMORANDUM FOR: S. Levine, Director, Office of Nuclear Regulatory Research

FROM: E. G. Case, Acting Director, Office of Nuclear Reactor Regulation

SUBJECT: STUDIES TO DETERMINE CAPABILITY OF EXISTING PERSONNEL NEUTRON DOSIMETRY SYSTEMS AT OPERATING NUCLEAR POWER REACTORS TO MONITOR REACTOR NEUTRON ENVIRONMENTS (RR-NRR-78-8)

NRR requests RES to fund a program for the purpose of collecting data on the effectiveness of personnel neutron dosimetry programs at operating nuclear power plants. To achieve this objective there is a need to identify plant areas in which significant neutron levels occur, and to characterize the nutron spectral distribution in order to determine the dose equivalent rates at these locations (e.g., containment areas of PWR's), so that occupational dose estimates, provided by the personnel neutron dosimeter, can be compared with the "true" theoretical dose as determined by the neutron spectrum and respective dose rate per unit flux for each energy interval at these locations. Neutron exposures have seldom been observed (reported) using current measurement techniques at operating reactors. We need to evaluate the adequacy of present neutron monitoring techniques at reactor sites. Obtaining the data in this manner would appear more efficient than requesting all licensees to perform these surveys independently.

Status of Problem

Regulatory Guide 8.14 "Personnel Neutron Dosimeters" requires that licensees supply personnel monitoring equipment to those employees whose exposure to neutrons is likely to exceed 300 mrem in a quarter. The Guide provides criteria for acceptable devices and techniques for neutron personnel monitoring. NTA film, a neutron dosimeter used throughout the nuclear industry, is not sensitive to neutrons below about 0.7 MEV. Therefore, depending upon the spectrum, the dose equivalent can be grossly underestimated. On the other hand, albedo dosimeters, which are not quite as widely used as NTA among power reactor licensees, are quite sensitive to low energy neutrons and can overestimate the dose equivalent by factors of 20 to 50 (again depending on the neutron spectrum and calibration technique). Since most licensees do not routinely measure the neutron spectral distribution at their facilities, the devices worn by the workers, although acceptable by R.G. 8.14, may be providing inaccurate dose estimates.

Contact: S. Block, EEB/DOR
28066

Accurate measurement of the neutron spectrum requires specialized nuclear instrumentation and methods generally not available to the licensee, except through consultants. Therefore, few attempts have been made by licensees to determine spectral distribution. Several PWR reactors (e.g., Calvert Cliffs, St. Lucie, Millstone 2 and Trojan) have neutron streaming problems inside containment and are installing additional neutron shielding. This problem is generic, and considerable staff time has been devoted to its resolution. This ignorance of specific neutron spectral distribution in occupied areas of containment is therefore of concern to the staff, because incorrect dose assessments may result.

Information Needs

A study is therefore needed which can provide the following data:

1) The neutron spectral distribution at selected locations inside and outside containment of operating nuclear power plants. The measurement technique should be of sufficient sophistication to show any structure that may exist in the spectral distribution curve, particularly in the intermediate energy region (i.e., from 10 ev to 100 kev) which may contribute an appreciable fraction of the dose equivalent. The neutron spectrum should also be characterized with respect to geometry and any shielding perturbation that could effect the measurement.

2) The theoretical ("true") dose equivalent rate, at each location, determined from the spectral distribution data of (1) and the Neutron Flux Dose Equivalent parameters of 10 CFR 20.4(4).

3) The neutron dose equivalent rates made at the locations selected in (1) above, using rem counter devices such as the Andersson-Braun neutron survey meter. Other devices that can measure neutron dose or dose equivalent rates with at least the same accuracy as the rem counter, over the neutron energy region of interest, may also be used in parallel.

4) The survey meter measurements, compared with the theoretical values, to show the effectiveness of portable survey meters to read out "true" dose equivalent rates of reactor neutron spectrum.

5) Measurements made using personnel monitoring methods described in Regulatory Guide 8.14 at the selected locations in (1), intercompared with the "true" dose equivalent to determine the accuracy of each method. (Personnel monitoring exposure techniques should be at the discretion of the contractor). Commercial personnel neutron dosimeter systems should be used, as available, for each personnel monitoring performance check (e.g., albedo personnel dosimeters and NTA film).

17-277

6) Conclusions with respect to the accuracy of the various techniques, grouped according to physical geometry and neutron shielding.

Cost and Possible Contractor

Battelle Northwest has submitted a draft 189 working paper to perform a study of this type. Although their scope does not directly address several issues of interest to NRR, it does contain the essence of these interests. Other laboratories that could perform this study include Lawrence Livermore Laboratory, which has developed a portable neutron spectrometer used to study the neutron energy spectral distribution at one nuclear power reactor; Savannah River Laboratory which has done considerable work in albedo personnel dosimetry; Brookhaven National Laboratory with experience in LET dosimetry; and the University of Wisconsin which has TLD expertise to perform these studies. We anticipate that the required information can be obtained at a cost of about $100,000 for a one year study at 6 to 12 reactors. Selection of reactors would be made in conjunction with NRC.

Value Impact

We feel that this study is important in confirming that adequate personnel neutron dosimetry is being performed by nuclear power reactor licensees, consistent with Regulatory Guide 8.14. If it is determined that the spectral distribution is heavily weighted with neutrons of energies less than 0.7 mev, those licensees using NTA film may be grossly underestimating personnel exposures. Appropriate actions could then be taken to change deficient personnel monitoring practices. Conversely, those licensees using albedo dosimetry might have to re-evaluate their calibration procedures if they are grossly overestimating their personnel neutron exposures. The requested study will provide NRR the technical basis for developing any needed additional guidelines or revising existing guidelines.

Sources of Information on Neutron Radiation at Power Plants

Several nuclear power plants have made neutron measurements in containment in conjunction with shield reviews because of their neutron streaming problems. These include Millstone II, Rancho Seco, Calvert Cliffs, Farley, Trojan and St. Lucie. These data can be made available by licensees. Other data have been reported at ANS meetings or have been developed by A&E firms (e.g., Bechtel, Ebasco, and Sargent and Lundy) for utilities in conjunction with shield reviews.

Edson G. Case

Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation

cc: See page 4

APR 0 4 1978

S. Levine - 4 -

cc: V. Stello
R. Minogue
C. Smith, Jr.
E. Volgenau
D. Eisenhut
R. Vollmer
B. Grimes
W. Kreger
T. Murphy
R. Alexander
L. Barrett
E. Adensam
G. Zimmer
J. Kastner
F. Swanberg
L. Cunningham
J. Foulke
Section B/EEB

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R Vollmer 3/24/78

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4/3/78

OFFICE ▶	EEB/DOR	EEB/DOR	EEB/DOR	RAB/DSE	AD/RT-DOR	D/DOR
SURNAME ▶	SBlock:jpf	LBarrett	BGrimes	WKreger	DEisenhut	VStello
DATE ▶	3/15/78	3/16/78	3/16/78	3/[illegible]/78	3/[illegible]/78	3/[illegible]/78

NORTH ANNA ENVIRONMENTAL COALITION

P.O. BOX 3951
CHARLOTTESVILLE, VIRGINIA 229
(717)533-7694 or (804)293-603
March 29, 1978

Mr. Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation
U. S. NUCLEAR REGULATORY COMMISSION
Washington, D. C. 20555

Dear Mr. Boyd:

I am writing in regard to the January 25, 1978 memorandum to you from Mr. Glenn W. Zimmer in the Occupational Health Standards Branch entitled

NEUTRON EXPOSURE AT COMMERCIAL POWER REACTORS.

This memorandum was given to the Advisory Committee on Reactor Safeguards (ACRS) on March 10, 1978 during an NAEC presentation regarding risks that are not currently understood.

The Coalition believes that Mr. Zimmer's new information raises serious questions regarding worker protection from neutron exposure, and thus asks you to answer the following questions at your earliest convenience:

1. What are the names of those reactors where personnel are "receiving some neutron exposure which heretofore has been unknown"? or is this a newly-known problem at all reactors (PWR's)?

2. Since your receipt of this January 25 memo, what new measurement techniques have been instituted at the reactors in question to remedy the "inadequacy" described in the memo?

3. If new measurement techniques have not yet been instituted, what measures are being taken to protect workers from previously unknown neutron exposure?

4. What assessment is being made of potentially inadequate reactor shielding?

5. Have the workers at risk from neutron exposure been so notified and allowed a voice in their assignments? If workers have not been informed of potential neutron exposure, the Coalition hereby requests that such notification be made.

Given the almost daily news stories about the rapidly growing knowledge of long-term effects from low-level radiation exposure, we are sure you have directed your staff to move swiftly on the problem of neutron exposure and will answer the foregoing questions promptly.

Thank you for your professional interest.

Sincerely,

June Allen (Mrs. P. M.)
President, NAEC

Enclosure

A-280

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

NRR
P-712

FEB 0 9 1978

MEMORANDUM FOR: Robert E. Alexander, Chief
Occupational Health Standards Branch, SD

FROM: Glenn W. Zimmer, Senior Health Physicist
Occupational Health Standards Branch, SD

SUBJECT: NEUTRON EXPOSURE AT COMMERCIAL POWER REACTORS

Following my submittal on the subject topic to Roger Boyd, Director, Division of Project Management, Office of Nuclear Reactor Regulation, the following things have occurred:

January 30, 1978

I called Leo Higginbotham, IE, to discuss what effort IE should take to investigate personnel neutron exposure at operating reactors. Mr. Higginbotham was not in so I left a message for him to return my call.

January 31, 1978

I called Mr. Higginbotham again and discussed briefly with him the information which I had forwarded to Roger Boyd on January 25. He told me that Tom Murphy, NRR, had discussed this with L. Cunningham and referred me to Mr. Cunningham. I then called Mr. Cunningham and he indicated that he was currently thinking of requesting information from the various IE regional offices about personnel neutron exposures at operating reactors to find out what information they presently have.

I received a call from Tom Murphy, NRR, who discussed with me the fact that they had been in contact with IE and inquired as to whether or not I knew the name of the reactor where the data were obtained and whether or not the data were personnel monitoring data. I told him that I did not know the name of the reactor and that I was told that the data are personnel monitoring data. He also inquired as to where the data had come from and I told him I had obtained it from Bill Endres and Leo Faust at Battelle, and that it was a copy of a briefing sheet which they had used when they gave a briefing to Sy Block, NRR, and me the morning of December 15, 1977, and again in the afternoon when they briefed Judy Foulke, RES, Charlie Hinson, NRR, me, and for a few minutes Frank Swanberg, RES, and that RES had requested them to submit a 189.

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I received a call from Sy Block, NRR, requesting me to meet with him and Tom Murphy, NRR, to discuss neutron exposure at commercial power reactors. I checked with Mr. Alexander who requested me to postpone it for a day because of the urgency of preparing information for Mr. Minogue's testimony. I notified Sy Block of this and will try to arrange a meeting for tomorrow after completing preparation of the testimony.

February 1, 1978

I called Sy Block, NRR, to inform him that because of urgent work here in preparing background information for Mr. Minogue's testimony that I would not be able to meet with him today. He stated that it was not urgent and that if we could make it tomorrow it would be fine.

February 2, 1978

I called Sy Block, NRR, this morning to set up a meeting. The meeting was established and I met with Sy Block and Tom Murphy, NRR, in Mr. Murphy's office. They were interested in the data that had gone forward with my memo to Mr. Boyd and in how I came to write the memo. I explained the circumstances which led up to the preparation of the memo for Mr. Boyd in conformance with Mr. Minogue's December 14 memo. We then discussed the fact that the data are not conclusive and in fact only demonstrate that personel neutron exposures at reactors should be investigated. The data are nonconclusive because of the calibration and the fact that there were no control badges nor were there any controls exercised by Endres and Faust over the utilization of the badges, hence they did not know exactly how they were used except that they were for personnel exposures. Mr. Murphy stated that he and Mr. Block had talked with Endres and Faust, BNWL, and that BNWL had merely given the badges to the health physicist at the reactor who distributed them and then returned them to Battelle for processing, and that the reactor personnel were then informed by Battelle of the readout of the badges. They felt that since they (Murphy and Block) were aware of neutrons at reactors inside containment, and it was their understanding that these exposures were inside containment, that the problem was not serious because of the fact that personnel were not usually inside containment, although some entries might be made, and that there is a Regulatory Guide (Reg. Guide 8.14) which reactor licensees have to comply with so that personnel neutron exposures are being taken into account. I pointed out that Reg. Guide 8.14 was only applied, to my knowledge, to those reactors applying for license after November 1, 1977, and so I questioned them as to whether or not they knew for sure that DOR had in fact backfitted the Reg. Guide 8.14 into tech specs, or had in some other way made it mandatory at reactors licensed prior to November 1, 1977. Mr. Murphy indicated that he was not sure, but that this was an area that would be explored with DOR. Sy Block stated that perhaps a closer look at personnel neutron dosimetry at reactors was necessary and perhaps TLD should be used instead of NTA film; however the energy dependence of TLD would make it necessary to better

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define the energy spectrum. Mr. Murphy also informed me, as had been indicated to me by Mr. Cunningham, IE, that he had been in touch with IE and that NRR planned to forward my memo to Boyd, NRR, to IE and request them to investigate. I told them that I had been given similar direction by my management, namely Mr. Minogue, for Mr. Alexander and I to look into notifying IE and having IE do an investigation. I also informed them of my earlier discussion with Mr. Cunningham, IE, on January 31 regarding his thinking of requesting information from the various IE regional offices to ascertain what they already knew about neutron exposures at reactors. Mr. Murphy felt that it would be unnecessary for me to forward my memo to IE requesting any action since this was a job that NRR would be taking care of. I told him that was fine and perhaps my request would be unnecessary in view of their action; however, I would comply with the direction I received from my management.

We also discussed the possibility that a controlled study should be made of the neutron energy spectrum, etc., at reactors following the initial investigation by IE. It was suggested that perhaps this should be a research program, and I agreed with this possibility dependent upon what information IE would come up with. The discussion then turned to who might be capable of doing this work, and I stated that the best people in my opinion would be the people at BNWL or Dale Hankins at LLL. Mr. Block was dubious of whether or not BNWL had the necessary equipment and that perhaps there was someone else at LLL better qualified, although he admitted that Dale Hankins certainly has the knowledge of TLD to do the job. He also suggested that Hoy at Savannah River would be qualified, with which I agreed. I then told them that Herb Attix (who is presently at the University of Wisconsin) would also be qualified, with which they agreed. Mr. Block stated that if BNWL did submit a 189 as had been suggested by RES, that NRR would be interested in reviewing it.

February 3, 1978

I drafted a memo from myself to Mr. Leo Higginbotham, IE, forwarding the memo with attachments which I previously sent to Mr. Boyd, NRR, for his use and information. I asked him to please advise us following the initial gathering of information from the regional offices whether or not IE is planning a follow-up study or technical investigation or if IE recommends a research program to define the magnitude and significance of personnel neutron exposures at commercial power reactors.

LHigginbotham, IE	LBarrett, NRR	
JCunningham, IE	CHinson, NRR	
RBoyd, NRR	BGrimes, NRR	Glenn W. Zimmer, Senior Health Physicist
TMurphy, NRR	JFoulke, RES	Occupational Health Standards Branch
WKreger, NRR	FSwanberg, RES	Office of Standards Development
SBlock, NRR		

A-283

OFFICE	SD:OHSB	SD:OHSB	Task #: N/A		
	GWZimmer/pdw	REAlexander			
DATE	2/3/78	2-3-78			

FORM 318 (9-76) NRCM 0240

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

MAR 0 6 1979

MEMORANDUM FOR: L. Higginbotham, Acting Director
Div. of Fuel Facility & Materials Safety Inspection, IE

FROM: D. G. Eisenhut, Assistant Director for Operational Technology, DOR

R. H. Vollmer, Assistant Director for Site Analysis, DSE

SUBJECT: NEUTRON EXPOSURE AT COMMERCIAL POWER REACTORS

We have received the attached memo from Glenn W. Zimmer in our Office of Standards Development and have discussed the issue with him. It is our understanding that the data on which he bases his concerns came from a Battelle Pacific Northwest Laboratory evaluation of neutron albedo dosimeters given to a health physicist of a PWR who used them, in some unknown manner, on reactor personnel. The people from PNL who were responsible for evaluating and reporting the data were Leo Faust and G. W. Enders.

We discussed the data with Faust and Enders, and it was stressed by these individuals that the data did not imply a lack of control by any licensee with respect to neutron exposure. The data was gathered as an aside to a research program on gamma skyshine dose measurements and for most of the dosimeters listed in the attached memo, the exposure was outside the control of Faust and Enders. In addition, it was stressed by the PNL investigators that the energy spectral distribution of the neutrons to which the dosimeters were exposed was not known. This means that the interpretation of the results from the dosimeters is questionable. The albedo dosimeter may overestimate exposures by as much as a factor of 20 to 50 depending on the calibration sources. The purpose of PNL submitting the data was to provide a justification for a research program proposed by PNL to NRC Research to measure spectral distribution of neutrons and the related calculated neutron dose equivalent compared to TLD neutron dose equivalent measurements.

We have no reason to believe that a problem exists at LWR's with respect to personnel neutron dosimetry as long as their Radiation Protection Programs are appropriately implemented. Regulatory Guide 8.14 "Personnel Neutron Dosimeters" gives acceptable methods of measuring neutron doses and dose equivalent exposures. This regulatory guide specifically recommends against use of NTA film for energies less than about 0.7 MEV. The guide provides alternate acceptable methods for determining neutron dose to personnel.

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We request that I&E determine, during normally scheduled inspections, whether or not reactor licensees are performing appropriate neutron measurements. By so doing, we can satisfy ourselves that recommended practices are being carried out.

This review was performed by T. Murphy, RAB/DSE, and S. Block, EEB/DOR.

/s/

Darrell G. Eisenhut, Assistant Director for Operational Technology
Division of Operating Reactors

/s/

Richard H. Vollmer, Assistant Director for Site Analysis
Division of Site Safety and Environmental Analysis

Enclosure:
As stated

cc: E. Case
R. Boyd
R. Mattson
H. Denton
V. Stello
D. Eisenhut
R. Vollmer
R. Alexander
R. Minogue
G. Zimmer
L. Barrett
T. Murphy

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EEB/DOR
LBarrett
2/24/78

AD/SA/DSE
RHVollmer
2/25/78

SEE PREVIOUS YELLOW FOR CONCURRENCES

OFFICE	RAB:DSE	EEB:DOR	RAB:DSE	EEB:DOR	AD:OT:DOR	AD:OT:DOR
SURNAME	TMurphy/mas	SBlock	WEKreger	BGrimes	DEisenhut	MCallaw
DATE	2/16/78	2/23/78	2/ /78	2/ /78	3/ 2 /78	2/

JAN 25 1978

MEMORANDUM FOR: Roger S. Boyd, Director
Division of Project Management
Office of Nuclear Reactor Regulation

FROM: Glenn W. Zimmer
Occupational Health Standards Branch
Office of Standards Development

THRU: Robert B. Minogue, Director
Office of Standards Development

SUBJECT: NEUTRON EXPOSURE AT COMMERCIAL POWER REACTORS

Recently it has come to my personal attention that personnel at some commercial power reactors are receiving some neutron exposure which heretofore has been unknown. Apparently these exposures have gone unnoticed because of the inadequacy of the neutron measurement techniques employed, and insufficient knowledge of this field. I understand that neutron exposures of up to a few hundred millirems in a relatively short period of time (a few hours or days) are possible (see attached Table 1 and Table 2). Additionally, I understand from another source that neutron fields of 25 kev neutrons superimposed on the 1/e neutron spectrum exist at a PWR which is known about. Neutrons of this energy cannot be measured by the NTA film which I understand is in use at that reactor. I do not know if the time-controlled personnel neutron exposures at that facility are being reported to NRC or not.

The significance of this, in my view, may be concern about the adequacy of reactor shielding and the control of exposures to reactor operating personnel. The previously unevaluated neutron contribution to the total dose equivalent may be significant, particularly if the current consideration of the neutron quality factor results in the assignment of higher quality factors thereby causing higher rem values.

Glenn W. Zimmer

Glenn W. Zimmer
Occupational Health Standards Branch
Office of Standards Development

Enclosures: Tables 1 and 2

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SPECIAL STUDY

SPECIAL PURPOSE BADGES AT A PWR SITE

Table 1

Dosimeter ID No.	Penetrating	Skin	Thermal Neutrons	Fast Neutrons	Dose - mrems
505	1.65E+2	1.65E+2	4.76E-1	0	165.4
507	1.42E+2	1.42E+2	4.45E+1	1.43E+3	1616.5
516	1.39E+2	1.39E+2	1.50E+0	5.37E+1	194.2
517	1.33E+2	1.33E+2	5.60E+1	4.67E+2	656
518	1.33E+2	1.33E+2	5.53E+1	4.93E+2	681.3
519	2.42E+2	2.42E+2	1.76E+2	7.71E+2	1189.0
520	4.51E+2	4.51E+2	4.72E+1	1.06E+2	604.2
609	1.36E+2	1.36E+2	4.12E-1	0	136.4
643	1.31E+2	1.31E+2	4.92E-1	0	131.5
645	1.43E+2	1.43E+2	6.27E-2	0	143.06

(Note: E+2, etc., type designation is the exponent for the factor of 10.)

The above data is information that was passed on to me in a personal communication. It is understood that the data as shown above for fast neutrons has not been corrected with a calibration factor. Because the thermoluminescent dosimeters were calibrated against Cf-252 instead of for the spectrum that was thought to pertain at the site, it is expected that the fast neutron data may be high by a factor of 10. Therefore, I have personally applied a factor of 10 reduction to the fast neutron column of figures added the penetrating exposure, thermal neutron exposure, and fast neutron exposure after it had been corrected downward by a factor of 10 to obtain a total mrem dose. This is shown in Table 2.

17-2-88

Table 2

Dosimeter ID No.	Corrected Fast Neutron	Total Neutron	Dose - mrems
505	0	.4	165.4
507	143	187.5	329.5
516	5.3	6.8	145.8
517	46.7	102.7	235.7
518	49.3	104.6	237.6
519	77.1	253.1	495.1
520	10.6	57.8	508.8
609	0	.4	136.4
643	0	.5	131.5
645	0	.06	143.06

The corrected fast neutron was obtained by decreasing the fast neutron listed in Table 1 by a factor of 10.

The total neutron was obtained by adding the thermal neutrons listed in Table 1 to the corrected fast neutrons.

The dose in mrems was obtained by adding the penetrating and thermal neutrons from Table 1 to the corrected fast neutrons from Table 2.

PERSONNEL NEUTRON DOSIMETRY METHODOLOGY

(FROM REGULATORY GUIDE 8.14)

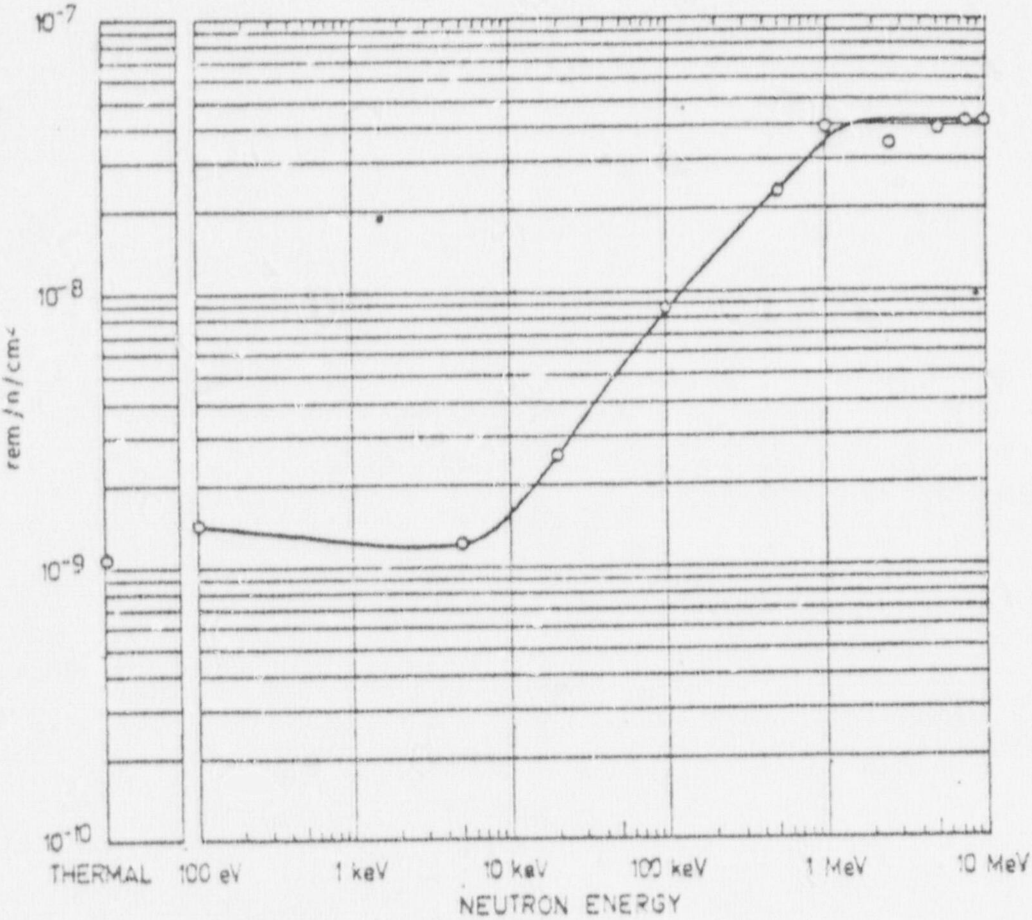
1. NEUTRON FILM BADGE
2. ALBEDO NEUTRON DOSIMETERS
3. CALCULATED NEUTRON DOSE EQUIVALENT BASED ON MEASUREMENT OF NEUTRON DOSE EQUIVALENT RATE AND STAY-TIME
4. CALCULATED NEUTRON DOSE EQUIVALENT BASED ON MEASURED GAMMA/NEUTRON RATIOS

Table A-1. Mean quality factors, $\overline{QF}$*, and values of neutron flux density which, in a period of 40 hours, results in a maximum dose equivalent of 100 mrem. [A5][1]

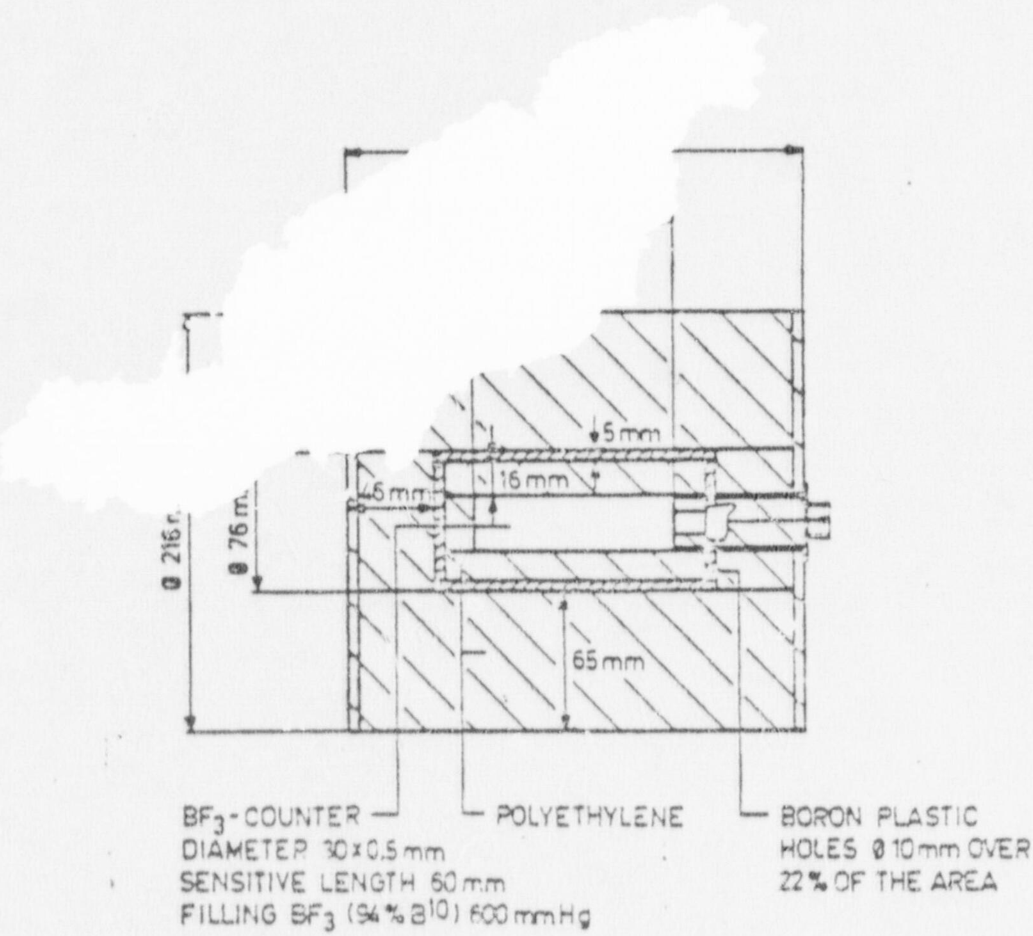
Neutron Energy	$\overline{QF}$	Neutron Flux Density
MeV		$cm^{-2}\ s^{-1}$
2.5×10^{-8} (thermal)	2	680
1×10^{-7}	2	680
1×10^{-6}	2	560
1×10^{-5}	2	560
1×10^{-4}	2	580
1×10^{-3}	2	680
1×10^{-2}	2.5	700
1×10^{-1}	7.5	115
5×10^{-1}	11	27
1	11	19
2.5	9	20
5	8	16
7	7	17
10	6.5	17
14	7.5	12
20	8	11

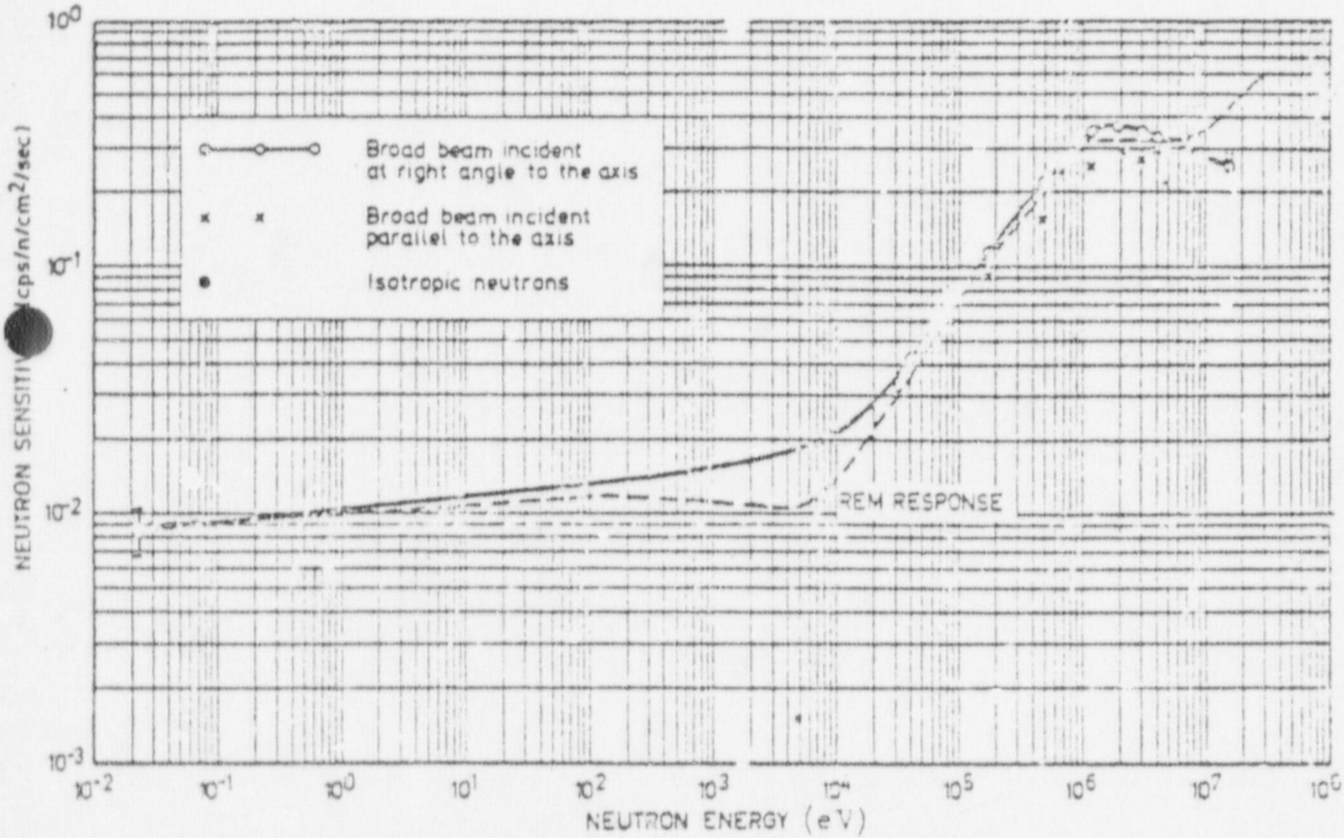
* Maximum value of $\overline{QF}$ in a 30-cm phantom.

[1]umbers in brackets refer to corresponding numbers in Section A-4, References the Text of the Appendix.

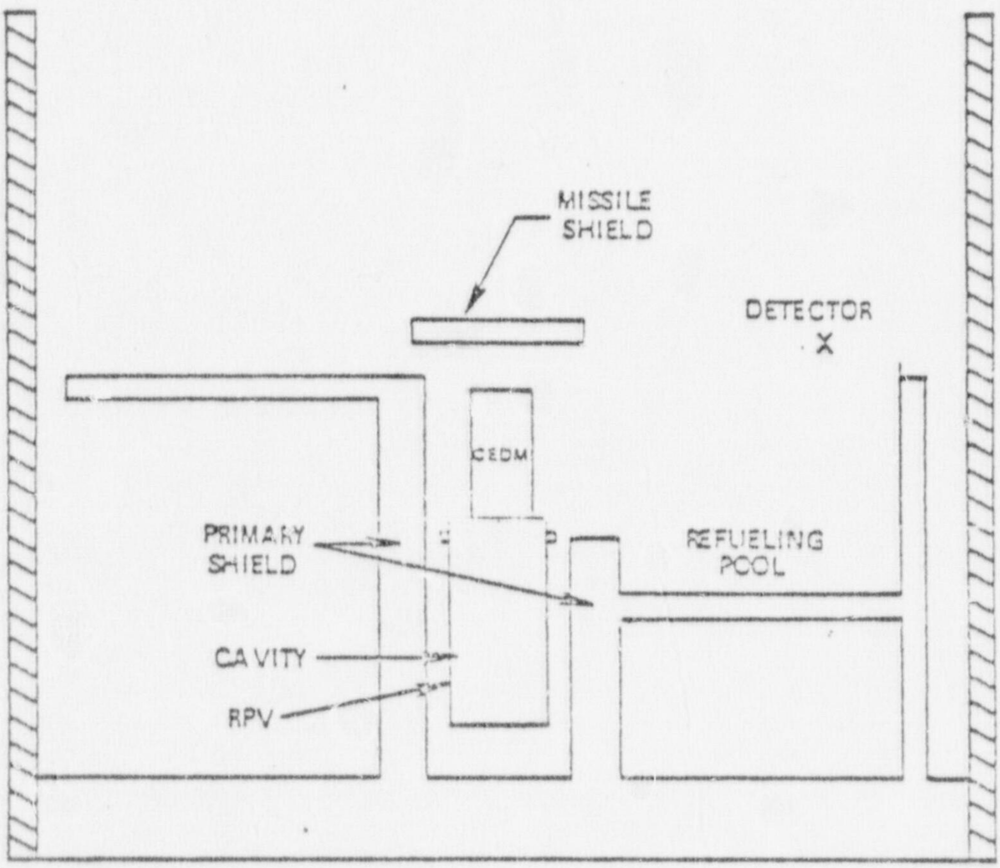


Neutron dose in rem/n/cm^2 as a function of neutron energy.
From NBS Handbook 63, (1957).


5 mm
16 mm
65 mm
BF3-COUNTER
DIAMETER 30 x 0.5 mm
SENSITIVE LENGTH 60 mm
FILLING BF3 (94 % B10) 600 mm Hg
POLYETHYLENE
BORON PLASTIC
HOLES Ø 10 mm OVER
22 % OF THE AREA



Neutron sensitivity as a function of energy



SIMPLIFIED GEOMETRICAL MODEL OF
NEUTRON STREAMING IN PWR's

STUDIES TO DETERMINE CAPABILITY OF NEUTRON DOSIMETRY TECHNIQUES TO MONITOR NUCLEAR POWER REACTOR ENVIRONMENTS

1. MEASURE NEUTRON SPECTRAL DISTRIBUTION AT SELECTED LOCATIONS INSIDE AND OUTSIDE REACTOR CONTAINMENT.

2. CALCULATE THE "THEORETICAL" DOSE EQUIVALENT RATE BASED ON THE SPECTRAL DISTRIBUTION AND THE FLUX TO DOSE EQUIVALENT CONVERSION FACTORS.

3. MEASURE THE NEUTRON DOSE EQUIVALENT RATES AT THE SAME LOCATIONS AS (1) WITH PORTABLE SURVEY METERS SUCH AS THE ANDERSON-BRAUN REMETER, AND A TISSUE PROPORTIONAL COUNTER.

4. COMPARE THE THEORETICAL AND MEASURED DOSE EQUIVALENT RATES TO SHOW EFFECTIVENESS OF SURVEY METERS.

5. USING NEUTRON PERSONNEL DOSIMETRIC TECHNIQUES PREVIOUSLY DESCRIBED, MAKE NEUTRON MEASUREMENTS AT THE SAME LOCATIONS AS (1) ABOVE AND COMPARE THESE MEASUREMENTS WITH THE THEORETICAL AND SURVEY METER VALUES TO DETERMINE THE DEGREE OF ACCURACY OF EACH PERSONNEL DOSIMETRY METHOD.

6. PROVIDE CONCLUSIONS WITH RESPECT TO EACH MEASUREMENT TECHNIQUE.

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SUMMARY

NEUTRON DOSES - WHOLE BODY

POWER REACTORS

YEAR	PERIOD OF EXPOSURE	NUMBER OF INDIVIDUALS	NUMBER OF MAN-REMS
1971	< 6 MOS.	5	0.72
	1/2 - 1 YR.	1	0.02
1972	< 6 MOS.	3	0.11
1973	< 6 MOS.	5	0.065
	1/2 - 1 YR.	5	0.380
1974	< 6 MOS.	68 (4 DOSES > .5)	6.789
	1/2 - 1 YR.	26	1.345
	1 - 2 YRS.	8	0.232
	2 - 3 YRS.	4	0.130
	3 - 4 YRS.	3	0.33
	4 - 5 YRS.	1 (1 DOSE > .5)	0.96
	6 - 7 YRS.	2 (1 DOSE > .5)	1.745
	9 - 10 YRS.	1 (1 DOSE > .5)	1.115

SUMMARY - (CONTINUED)

YEAR	PERIOD OF EXPOSURE	NUMBER OF INDIVIDUALS	NUMBER OF MAN-REMS
1975	< 6 MOS.	34	1.288
	1/2 - 1 YR.	8	0.664
	1 - 2 YRS.	8 (1 DOSE >.5)	2.169
	2 - 3 YRS.	11 (2 DOSES >.5)	2.242
	3 - 4 YRS.	3	0.239
	4 - 5 YRS.	1	0.04
	5 - 6 YRS.	1 (1 DOSE > .5)	0.645
	8 - 9 YRS.	1	0.01
	9 - 10 YRS.	1 (DOSE > .5)	0.55

17-297

APPENDIX XXXIII

UNITED STAT Request by E. J. Sternglass for ACRS
NUCLEAR REGULATOR Review of Changes in Cancer Mortality
ADVISORY COMMITTEE ON RE in the Vicinity of Several Nuclear Plants
WASHINGTON, D.

March 23, 1978

MEMORANDUM FOR: Chairman Hendrie
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Bradford

FROM: Raymond F. Fraley
Executive Director, ACRS

SUBJECT: REQUEST BY DR. ERNEST J. STERNGLASS FOR ACRS REVIEW OF CHANGES IN CANCER MORTALITY IN THE VICINITY OF THE HADDAM NECK (CONNECTICUT YANKEE) AND MILLSTONE NUCLEAR PLANTS

The attached may be of interest in view of the current activity to review the effects of low-level ionizing radiation.

The Committee plans further discussion of this matter during its 216th meeting (April 6-7, 1978) to consider an appropriate course of action.

Attachments:
1. Ltr., Sternglass to Moeller dtd. 3/3/78
   w/encls:
   a. Rpt. by Sternglass, "Cancer Mortality Changes Around Nuclear Facilities in Connecticut"
   b. Rpt. by Sternglass, "Strontium-90 Levels in the Milk & Diet Near Conn. Nuclear Power Plants"
2. Ltr. Fraley to Sternglass dtd. 3/22/78
3. Memo, Fraley to Members dtd. 3/17/78, w/o atts.
4. Ltr., Fraley to Hilberg dtd. 3/22/78, w/o atts.
5. Ltr. Morgan to Sternglass dtd. 2/14/78

cc w/atts.:
L. V. Gossick, EDO
E. G. Case, NRR
C. V. Smith, NMSS
R. B. Minogue, SD
J. Hard, OCM
S. Chilk, SECY

CONTACT:
R. F. Fraley, ACRS
634-1371

A-298

University of Pittsburgh

SCHOOL OF MEDICINE
Department of Radiology
Radiological Imaging Division

March 3, 1978

Dr. Dade W. Moeller
27 Wildwood Drive
Bedford, MD 01730

Dear Dr. Moeller:

At the request of Dr. Karl Z. Morgan, I am sending you copies of two reports dealing with strontium-90 levels and cancer mortality changes around the Haddam Neck and Millstone Reactors for review by the ACRS.

Sincerely yours,

Ernest J. Sternglass

Ernest J. Sternglass, Ph.D.
Professor of Radiological Physics

Enclosure

ES:ss

17-299 ATT. 1

CANCER MORTALITY CHANGES AROUND NUCLEAR FACILITIES IN CONNECTICUT

Dr. Ernest J. Sternglass
Professor of Radiological Physics
Department of Radiology
University of Pittsburgh
School of Medicine

A-300

Testimony presented at a Congressional Seminar on Low-Level Radiation, Feb. 10, 1978, Washington ,D. C.

A detailed study of cancer statistics in Connecticut and nearby New England indicates that cancer mortality increased sharply around two large nuclear reactors in south-eastern Connecticut in direct relation to the measured pattern of accumulated levels of strontium-90 in the local milk. Cancer rates increased most strongly closest to the Millstone Nuclear Power Station located in Waterford where the measured strontium-90 levels reached their highest values, with lesser rises being observed for areas with lower values of strontium-90 in the milk located at increasingly greater distances in every direction away from the Millstone Plant, known to have released the largest amount of radioactive gases ever officially reported for any nuclear plant in the United States.[1]

The Haddam Neck plant started to operate in 1968 and the Millstone plant followed in 1970. Between this time and 1975, the most recent year for which detailed data are available, the cancer mortality rate rose 58% in Waterford where the most heavily emitting Millstone plant is located, 44% in New London five miles to the north-east, 27% in New Haven, 30 miles to the west, and 12% for the State of Connecticut as a whole. Rhode Island, whose border is only 20 to 30 miles east of these two plants rose 8%, Massachusetts some 70 miles to the north-east rose 7%, New Hampshire some 120 miles north-east rose only 1%, while for the State of Maine more than 200 miles in the same direction, the cancer death rate actually declined by 6% during the same period.[2]

Likewise, again following the pattern of decreasing strontium-90 in the milk, cancer mortality declined 1% for Vermont more than 200 miles to the north. Even for heavily polluted New York City located some 120 miles to the south-west, cancer mortality did not rise as one might have expected, but actually decreased by 2% between 1970 and 1975, despite the high levels of sulphur, carbon monoxide, automobile exhaust, heavy metals, cigarettes, organic chemicals, food-additives, pesticides and other types of pollutants.

An examination of the radiation doses received by the population drinking the milk in Waterford and nearby New London using the accepted methods recommended by the International Committee on Radiation Protection indicates that the accumulated doses to the bones of children over the period 1970 to 1975 reached values of about 640 millirads from the milk and other food produced in the area, and about 320 millirads to the bone-marrow.[3] This must be compared with a dose of some 2 millirads to the bone-marrow from a typical chest x-ray, so that the very radiation sensitive bone-marrow of children in the New London area received the equivalent of some 160 chest x-rays in the course of 6 years of their most sensitive period of growth and development.

The dose to the bone-marrow produced by strontium-90 from the milk and food of 320 millirems must also be compared with the dose of 1,200 millirems[4] found by Dr. Alice Stewart of Oxford University to have doubled the normal risk of leukemia and cancer for children who had received diagnostic x-ray exposures during their development in their mother's womb, and some 80 millirems for those irradiated in their first three months of intra-uterine development. Comparable doses have also recently been found to double the normal risk of cancer and leukemia among older atomic workers in a study by Drs. Thomas Mancuso, Alice Stewart and George Kneale as reported at recent Congressional Hearings and published in the November 1977 issue of the journal, "Health Physics". [5]

Since bone-marrow type of leukemia is well known from studies of the Hiroshima and Nagasaki A-Bomb survivors to be induced by radiation, and since measurements of the bones of both children and adults have shown a high correlation with levels of strontium-90 in the milk, one is led to conclude the probable existence of a direct causal relation between the abnormally high levels of strontium-90 in the milk near the two Connecticut Nuclear plants and the pattern of cancer changes in Connecticut and nearby New England.

This conclusion is further supported by the fact that the types of cancers that rose most strongly in the Connecticut area are exactly those types that have been found to be most sensitive to radiation in earlier studies as classified by the International Committee on Radiation Protection. Thus, the types of cancers that increased the most in the time available so far were cancers of the respiratory system, which rose 25%, breast cancers, which increased 12%, and cancers of the pancreas, which rose 32%. Since the peak of cancer mortality for respiratory cancers did not occur among the uranium miners until some 7 to 12 years after the onset of irradiation, it is to be expected that further rises in lung cancer will take place in the next five years.(6)(7)

Additional evidence that the pattern of sharply rising cancers in south-eastern Connecticut and nearby New England is likely to be due to the strontium-90 and other fission products that escaped from the Millstone and Haddam Neck Nuclear reactors comes from the fact that cancer deaths show a much greater rise for women than for men, consistent with the findings of Mancuso, Stewart and Kneale for atomic workers exposed to similarly low levels of radiation over a period of years. Thus, whereas cancer mortality rates in Connecticut increased by only 11% for white males between 1970 and 1975, the increase for white females was 17%.

Still another observation supports the conclusion that the sharp local rises in cancer in Connecticut are connected with the localized releases of airborne radioactivity from defects in the nuclear fuel, comes from the

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evidence that the increases were largest for those who were simultaneously exposed to the highest concentrations of other known cancer promoting pollutants, such as industrial chemicals, dust ,pesticides, sulfates, nitrous oxide and other air-pollutants, both in the area where they live and in the working place.

Such synergistic effects are well-known for the case of uranium miners, where the mortality due to lung cancer is some 5 to 10 times greater for miners who only inhaled the radioactive gases but did not smoke while the rate was 50 to 100 times greater for those who did.(6)

Thus, the combined action of airborne radioactivity and ordinary pollution would be greatest for those who live and work in the most polluted enviroments, who have the lowest socio-economic status and therefore also the poorest medical care, so that they do not receive the benefit of early diagnosis and treatment. It follows that such synergistic effects involving radioactive and other forms of pollution would be expected to affect most heavily the poorest portion of the population, and this is indeed found to be the case in Connecticut.

Thus, while the total number of cancer deaths increased 15% for the white population of the state as a whole, between 1970 and 1975, this number rose 51% for the non-white or predominantly black population.

Furthermore, in accordance with the greater airborne dust and pollution in chemical factories and other heavy manufacturing, mining and construction activities employing men, the greatest increase in the number of cancer cases during the time the radioactive gases were added to the existing pollutants took place for non-white males, namely by the very large amount of 77%. Thus, the observed pattern of cancer mortality changes in Connecticut and nearby New York and New England since the onset of airborne releases by the two large nuclear plants all fit the expected behavior for radiation - related cancers observed in numerous earlier animal experiments and large-scale epidemiological studies carried out over the last thirty years by many scientists all over the world.

Especially significant is the fact that the cancer increases in Connecticut following the rise of local strontium-90 levels occurred most strongly in those age groups which the recent studies of Milham[10] as well as Mancuso, Stewart and Kneale[5] had shown to have the greatest increases among the carefully monitored workers at [illegible]shington. These were particularly the oldest workers, for whom the immune system is known not to be as effective in protecting against the proliferation of cancer cells as in the middle aged adult, just as in the case of the developing fetus and the young infant.

As shown in the accompanying table, whereas there was an overall increase in the cancer mortality rate per 100,000 population of 12%, after correcting for the change in age distribution, the cancer mortality rate for the 25 to 49 year old individuals actually declined 15%, presumably due to their much greater resistance to the effects of chronic irradiation on their immune defenses and the general improvement in environmental factors and medical care.[11]

On the other hand, there was an increase in cancer mortality rates for all older age groups, namely +4% for those 50 to 54 years old, +9% for those 55 to 64 years old, and +14% for those over 65 years at death, a pattern that fits the trend of the data for the atomic workers at Hanford found by the Milham and Mancuso studies.[5]

These findings help to explain why earlier observations on workers, x-ray technicians, and radiologists exposed to radiation indicated a much smaller hazard than is now emerging from studies of entire populations under normal peace-time conditions that include the unborn, the young and the very

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old. NOt only were most workers and radiologists in the least sensitive age group when they were exposed, but they were also receiving relatively brief exposures at considerably higher instantaneous dose-rates than individuals in the general population whose principal dose comes from very low dose, continuous exposure from inhaled or ingested radio-isotopes in their bones and other organs such as strontium-90.

Thus, the range of sensitivity can easily vary by a factor of 100 to 1000, depending on the age and the intensity of the radiation,[12] the effect per unit absorbed dose being most serious for very low-level, protracted environmental exposures to the developing fetus and the individual with reduced immune resistance over 65 years of age,[13] in agreement with the observations of Bross.[14]

This means that the most serious of all radiation exposures are not brief medical x-rays, diagnostic isotope tests for the adult, but prolonged environmental exposures to fallout accumulating in the body from nuclear bomb-testing and releases from nuclear facilities acting slowly on the infant in utero, the young child and the oldest individuals in our society.

As a final test of this conclusion, it follows that the greatest effects on cancer rates in the general population during recent years should not be associated with medical x-rays or other environmental factors but with the releases from large nuclear facilities, especially since world-wide strontium-90 concentrations in the diet from bomb-tests have now declined to levels below those measured around these installations. This is supported not only by the findings around the Connecticut Reactors, but also by the pattern of strontium-90 levels and cancer changes around the nuclear fuel reprocessing facility at West-Valley, N. Y. [15]

Thus, one would expect that in general, the most recent unexplained upswing in U.S. cancer mortality rates should have taken place most strongly in the states that have large nuclear reactors and fuel reprocessing facilities within or close to their borders. At the same time, states that do not have such large nuclear facilities operating for more than five to six years should now show either much smaller increases in cancer rates, or manifest declines if strontium-90 and cesium-137 from fission processes play a key synergistic role with other carcinogens in the environment.

As can be seen from an inspection of Table 5 , this is indeed found to be the case. If one examines the rate of change of cancer mortality in the United States for every state during the most recent 3 year period for which, detailed data is available (1972-75), one finds that the greatest upward changes have taken place for the states that have the largest nuclear facilities such as Hanford (Washington), Oak Ridge (Tennessee), Savannah River (South Carolina), or that have nuclear reactors with known large releases in very densely populated areas such as near the Millstone boiling water reactor (Connecticut and Rhode Island) and the Oyster Creek Reactor in New Jersey, which is also of the boiling water (BWR) type.

In fact, according to the figures published annually in the U.S. monthly Vital Statistics Reports, above average rates of cancer increase in the U.S. occurred exactly in these states: Washington, +5.0%; Connecticut, +8.6%; Tennessee, +8.1%; Rhode Island, +8.0%; New Jersey, +5.7%; South Carolina, +5.4% compared with a U.S. average of +3.4% for this period.

On the other hand, cancer mortality rates actually declined during this same period most strongly in the four states having no nuclear facilities at all: Alaska, -10.6%; Montana, -4.4%; New Hampshire, -2.0%; and Hawaii, -1.5%. For Maine, which has only a single pressurized water reactor (PWR) operating since 1972, cancer rates declined 1.3%. Following the same pattern, Virginia with two recently completed PWR's declined somewhat less or by 1.1%.

But perhaps most significant is the fact that New York City, with two pressurized water reactors that emit fewer radioactive gases than the boiling water type located 30 miles north of the city showed a decline of 1.1% in cancer rates despite its enormous air-pollution and socio-economic problems, clearly supporting the conclusion that when large amounts of radioactive gas releases are missing from the mix of pollutants, the resulting effect on cancer and other chronic diseases is much less than when radioactive gases act synergistically with dust, chemicals, cigarettes and other air-pollutants in the environment.

Clearly, it is difficult to understand this striking pattern of cancer changes in any other way. When states that are as environmentally clean of ordinary air-pollutants as the State of Washington and MOntana are changing in opposite directions, one increasing by 5.0% while the other is decreasing by 4.4%, ordinary air and water pollution by itself can hardly be the crucial factor.

And when a heavily urbanized, densely populated and polluted area such a New York City declines in cancer rates compared to such rural, clean areas as the State of WAshington, the State of Tennessee, or the State of South Carolina, one cannot continue to put the principal blame on sulphur emissions from fossil fuel power-plants, automobile exhaust, drugs, food additives, hair-dyes, cosmetics, particulates, and medical x-rays for the present rise in the U.S. cancer rate without considering the role of radioactive releases.

The facts clearly show that the ordinary types of widely distributed, cancer causing agents cannot be the sole factors involved: they could not explain the highly localized cancer rises around Millstone, West-Valley, Hanford, Oak Ridge, and Savannah River, or the sharp declines in areas far from such sources of man-made radioactive wastes.

As difficult as it is for us to face the new facts that have now come to light as to the unexpectedly high sensitivity to prolonged low-level radiation exposure of some segments of our population, we cannot continue to risk the very survival of our nation on the hope that all these new findings will somehow be explained another way. Each year that we persist in closing our eyes to the new data, we will increase the total amount of Sr-90 accumulated in the soil and thus the biological damage to our newborn and the cancer risk for our older population. But if we should be able to accept these disturbing findings, then the evidence for the declining cancer rates in the least polluted areas of our country clearly points the way to the possibility of greatly reducing the risk of cancer and chronic disease in the years to come as we learn how to prevent the subtle damage from what we once believed were harmless levels of man-made and natural background radiation.

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2. See Table I and Figure I.

3. See Table 10 of Reference I, reproduced in Appendix I.

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9. See Reference 8.

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11. Table 4

12. A. Petkau, "Health Physics", 22, 239, 1972.

13. E. J. Sternglass, "Radioactivity", in "Environmental Chemistry", Chapter 15, p. 477-515; Ed. by J. O'M. Bockris (Plenum Press, N. Y. 1977).

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15. See Appendix II, Figures 1 and 2 for strontium-90 levels around West-Valley and cancer rate changes in the city of Olean, N. Y. located in Cattaraugus county 25 miles southeast of the N.F.S. facility, showing an increase by 95% in cancer rates peaking 9 years after start of large-scale releases of fission products. Similar patterns of high concentration of strontium-90 in the milk decreasing with distance away from the nuclear facility have also been found for the Trojan Nuclear Plant near Portland, Oregon, a pressurized water reactor completed in 1975 (see figures for June-July 1976 and 1977).

Table of Cancer Mortality Rates in Connecticut and New England Before and After Start-Up of the Millstone Nuclear Plant in Waterford, Connecticut

Cancer Death Rate per 100,000 Population

	Approx. Dist. From Millstone	1970	1975	Percent Change
Vermont	200m. NW	176.1	173.9	-1%
Connecticut	35m. NW*	168.1	188.4	+12%
New Haven, Conn.	30m. W	200.9	255.5	+27%
Waterford, Conn.	0	152.6	241.8	+58%
New London, Conn.	5m. E	177.4	255.0	+44%
Rhode Island	50m. NE	200.1	216.0	+8%
Massachusetts	70m. NE	185.0	198.4	+7%
New Hampshire	120m. NE	180.4	182.6	+1%
Maine	200m. NE	197.7	185.0	-6%
U.S.	—	162.0	171.7	+6%
NEW YORK CITY	120m. SW	220.9	216.4	-2%

*Population center of State of Connecticut (Hartford-Waterbury area)

Sources:

Connecticut Health Department, Registration Reports;

U.S. Monthly Vital Statistic Reports.

TABLE 1

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TABLE 19—DEATH RATES PER 100,000 POPULATION FROM SELECTED CAUSES, CORRECTED FOR RESIDENCE FOR TOWNS OF OVER 10,000 POPULATION: CONNECTICUT, 1970

TOWN	Tuberculosis All forms (010–019)	Syphilis (090–097)	Malignant neoplasms (140–209)	Diabetes mellitus (250)	Cardiovascular diseases (390–458)	Diseases of the heart (390–398, 402, 404, 410–429)	Rheumatic fever and rheumatic heart disease (390–398)	Hypertensive heart disease (402, 404)	Ischemic heart disease (410–414)	Chronic endocarditis and myocardial insufficiency (421, 428)	Cerebrovascular Disease (430–438)	Arteriosclerosis (440)	Pneumonia (480–486)	Emphysema (492)	Cirrhosis of liver (571)	Nephritis and nephrosis (580–584)	All accidents¹ (800–949)	Motor vehicle accidents¹
Column Number	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18
CONNECTICUT	1.7	0.2	168.1	16.4	453.2	321.7	7.6	3.1	313.3	1.6	90.4	13.7	26.4	9.2	18.8	2.9	[illegible]	[illegible]
Fairfield County	2.3	0.1	172.6	17.0	438.7	333.0	5.9	2.4	299.7	2.9	81.3	13.5	23.4	8.2	19.3	2.9	27.5	[illegible]
Bethel	....	....	81.7	9.0	525.0	408.0	....	....	408.6	....	72.6	9.1	20.3	....	5.1	9.1	[illegible]	[illegible]
Bridgeport	5.1	....	230.0	25.8	606.9	450.4	10.9	3.2	436.7	0.6	113.7	17.3	26.2	10.9	23.5	3.6	23.7	[illegible]
Danbury	....	....	135.8	23.5	438.6	326.8	....	2.0	315.3	3.9	74.4	13.7	17.0	3.9	15.7	....	[illegible]	[illegible]
Darien	....	....	141.7	4.9	371.5	223.5	4.9	....	213.6	....	63.4	14.7	21.4	....	14.7	....	[illegible]	[illegible]
Fairfield	1.8	....	174.5	7.0	440.0	297.8	7.0	7.0	274.9	1.8	102.2	26.4	25.4	7.0	21.1	....	[illegible]	[illegible]
Greenwich	1.7	....	213.7	15.0	454.1	340.5	....	1.7	267.1	15.0	88.3	16.7	28.7	18.4	21.7	3.3	[illegible]	[illegible]
Monroe	8.2	....	106.7	8.2	123.1	82.0	....	8.2	73.8	....	16.4	16.4	16.4	8.2	....	....	[illegible]	[illegible]
New Canaan	....	....	165.2	17.1	455.7	279.1	5.7	....	267.7	....	119.6	28.5	5.7	5.7	....	....	[illegible]	[illegible]
Newtown	6.7	....	188.6	....	370.4	249.2	26.9	....	215.5	....	80.8	....	67.4	....	6.7	6.7	[illegible]	[illegible]
Norwalk	2.5	....	153.7	20.0	431.5	331.3	3.8	1.3	325.8	2.5	73.6	7.6	31.5	7.6	25.4	2.5	[illegible]	[illegible]
Ridgefield	....	....	59.7	5.4	260.3	222.4	....	5.4	200.1	....	32.5	....	5.4	5.4	21.7	5.4	10.8	[illegible]
Shelton	....	....	191.7	3.7	367.4	293.9	7.3	....	279.3	....	66.1	7.3	22.0	....	14.7	....	29.4	[illegible]
Stamford	3.7	....	149.3	18.3	449.6	341.6	9.2	2.7	316.8	1.8	81.5	11.9	21.7	6.4	22.0	7.3	23.8	[illegible]
Stratford	....	....	172.4	20.0	507.1	372.8	4.0	4.0	354.8	6.0	106.2	12.0	14.0	12.0	18.0	6.0	30.0	[illegible]
Trumbull	....	3.2	145.3	18.9	311.0	252.6	9.5	....	240.0	....	56.8	18.9	12.6	....	3.2	....	15.8	[illegible]
Westport	....	....	148.7	18.1	326.4	253.9	....	....	243.0	3.6	61.7	3.6	7.3	14.5	18.1	3.6	21.8	[illegible]
Wilton	....	....	91.8	7.3	335.5	269.9	....	....	255.3	7.3	65.6	....	7.3	7.3	21.9	....	14.6	[illegible]
Rest of County	....	....	109.7	5.5	233.1	178.2	....	....	167.3	2.7	41.1	....	24.7	11.0	11.0	....	11.0	[illegible]
Hartford County	1.6	0.4	166.6	16.9	421.0	307.8	8.1	4.2	287.5	0.9	83.9	11.5	26.3	10.4	19.7	2.9	32.5	[illegible]
Berlin	....	....	140.6	....	471.1	372.7	7.0	....	323.6	....	84.4	....	14.1	7.0	28.1	....	21.1	[illegible]
Bloomfield	....	....	190.0	32.6	334.3	228.0	....	16.3	206.3	5.4	97.7	10.9	16.3	5.4	5.4	....	54.3	[illegible]
Bristol	....	....	132.5	16.1	430.0	333.7	10.8	3.6	297.8	1.8	77.1	7.2	23.3	7.2	25.7	3.0	[illegible]	[illegible]
East Hartford	....	....	172.6	10.4	328.0	255.5	1.7	1.7	216.9	....	48.3	12.1	27.6	10.0	13.8	1.7	27.6	[illegible]
Enfield	....	....	99.3	4.2	233.4	184.1	2.2	....	174.3	....	24.9	4.4	30.9	2.2	13.2	2.2	37.5	[illegible]
Farmington	6.9	....	158.9	6.9	315.3	255.3	6.9	6.9	231.8	....	62.2	6.9	41.4	20.7	27.6	....	55.3	[illegible]
Glastonbury	4.8	....	144.2	9.6	302.8	201.9	....	....	201.9	....	81.7	9.6	9.6	4.8	....	4.8	14.4	[illegible]
Hartford	2.5	....	202.0	19.0	530.4	403.4	10.8	9.5	378.2	0.6	98.2	15.8	40.5	8.9	33.3	3.2	41.8	[illegible]
Manchester	....	....	160.0	41.5	415.4	233.4	10.4	4.2	224.7	....	121.0	21.9	16.6	16.6	4.2	2.1	[illegible]	[illegible]
New Britain	4.8	1.2	191.7	18.0	523.5	363.0	12.0	....	340.2	1.2	131.8	12.0	23.8	9.6	31.1	7.2	25.4	[illegible]
Newington	....	....	200.5	11.6	337.8	254.5	3.9	....	243.0	....	42.4	....	19.3	11.6	7.7	....	15.4	[illegible]
Plainville	5.9	....	184.3	5.9	439.9	311.8	5.9	....	305.9	....	65.4	11.9	23.3	11.9	17.8	....	29.7	[illegible]
Rocky Hill	....	....	176.4	58.8	401.8	323.4	....	....	313.6	9.8	68.6	....	29.4	9.8	9.8	9.8	35.2	[illegible]
Simsbury	....	....	79.2	17.0	232.2	164.2	....	11.3	141.6	....	51.0	....	22.7	5.7	....	....	45.3	[illegible]
Southington	6.4	....	112.4	12.8	260.7	192.0	6.4	6.4	176.0	....	48.2	12.8	12.8	12.8	19.2	3.2	[illegible]	[illegible]
South Windsor	....	....	82.8	....	254.7	171.9	6.4	....	165.6	....	63.7	6.4	19.1	19.1	12.7	....	12.7	[illegible]
West Hartford	....	1.5	181.3	11.7	573.5	418.1	13.2	2.9	358.7	....	121.3	15.1	21.7	10.3	10.3	1.5	[illegible]	[illegible]
Wethersfield	....	....	205.1	29.8	451.2	316.8	7.5	3.7	[illegible]	3.7	74.6	11.9	18.6	3.7	11.9	3.7	[illegible]	[illegible]
Windsor	....	....	180.0	25.0	339.0	234.7	8.9	....	225.9	....	66.4	13.3	31.0	31.0	17.7	4.4	[illegible]	[illegible]
Windsor Locks	....	....	98.9	6.6	383.9	263.7	6.6	6.6	250.5	....	105.5	13.2	....	....	26.4	6.6	[illegible]	[illegible]
Rest of County	....	2.0	130.1	15.8	321.3	231.6	9.9	3.9	212.9	2.0	69.0	3.9	11.8	7.9	9.9	2.0	31.5	[illegible]
Litchfield County	1.4	....	191.4	21.4	505.6	373.9	6.2	6.2	353.9	2.8	96.1	13.8	33.0	13.8	15.2	2.1	[illegible]	[illegible]
New Milford	....	....	176.1	6.8	467.5	365.9	6.8	6.8	352.4	....	81.3	....	47.4	6.8	13.6	....	27.1	[illegible]
Plymouth	....	....	222.1	48.3	441.2	357.3	....	....	328.3	9.7	67.6	9.7	19.3	9.7	19.3	....	9.7	[illegible]
Torrington	....	....	193.8	18.8	581.0	447.0	3.1	15.6	418.9	3.1	100.0	25.0	50.0	12.5	18.8	3.1	9.4	[illegible]
Watertown	....	....	123.0	26.7	390.3	278.0	....	....	278.0	....	69.5	21.4	10.7	5.3	....	....	10.7	[illegible]
Winchester	....	....	179.8	27.0	512.5	350.7	....	9.0	332.7	9.0	116.9	9.0	27.0	9.0	45.0	....	[illegible]	[illegible]
Rest of County	3.5	....	212.9	19.0	502.1	370.2	12.1	3.5	331.5	1.7	107.2	10.4	32.9	20.8	12.1	3.5	[illegible]	[illegible]
Middlesex County	0.9	....	158.9	9.7	469.7	365.2	9.7	3.5	340.7	4.4	86.0	6.1	25.5	8.8	21.1	0.9	[illegible]	[illegible]
Clinton	....	....	86.4	....	278.3	230.7	....	9.6	201.8	....	38.4	9.6	28.8	9.6	9.6	....	[illegible]	[illegible]
Middletown	....	....	160.3	14.1	503.2	373.0	11.3	2.8	[illegible]	8.5	112.8	2.8	31.0	5.6	[illegible]	....	[illegible]	[illegible]
Rest of County	1.5	....	161.6	8.8	482.1	380.7	10.3	4.4	357.2	2.9	79.4	7.3	22.0	10.3	14.7	1.5	[illegible]	[illegible]
New Haven County	1.3	....	179.5	16.7	516.1	387.0	10.5	2.1	361.7	0.9	97.0	13.3	28.6	7.8	13.3	3.1	31.4	[illegible]
Ansonia	....	....	221.8	23.6	542.6	358.6	14.2	4.7	339.7	....	146.3	18.9	42.5	14.2	23.5	9.4	4.7	[illegible]
Branford	4.9	....	199.6	....	457.6	345.7	4.9	9.7	331.1	....	73.0	24.3	19.5	4.9	4.9	4.9	[illegible]	[illegible]
Cheshire	....	....	198.2	15.9	355.3	270.5	5.3	....	259.9	....	53.3	....	21.2	....	....	....	37.1	[illegible]
Derby	....	....	206.2	15.9	570.9	420.3	23.8	....	380.6	7.9	118.9	15.9	55.5	....	15.9	....	23.8	[illegible]
East Haven	....	....	162.6	19.8	315.1	271.7	....	4.0	219.9	....	51.6	4.0	19.8	....	7.9	....	[illegible]	[illegible]
Guilford	....	....	138.6	33.0	428.5	280.4	16.5	....	271.9	....	90.6	21.7	33.0	8.2	....	....	[illegible]	[illegible]
Hamden	....	....	175.5	8.1	492.2	369.2	12.1	....	351.1	....	101.9	10.1	32.3	6.1	12.1	....	[illegible]	[illegible]
Meriden	....	....	164.3	17.9	582.0	419.6	17.9	....	401.7	....	133.9	12.5	21.4	5.4	32.1	5.4	[illegible]	[illegible]
Milford	3.9	....	162.5	11.7	499.8	338.4	7.8	....	[illegible]	....	78.3	19.6	23.5	3.9	13.7	3.9	[illegible]	[illegible]
Naugatuck	....	....	131.1	17.2	432.6	316.7	13.0	....	[illegible]	....	90.8	8.6	25.9	13.0	[illegible]	....	[illegible]	[illegible]
New Haven	2.9	....	200.9	23.4	597.7	469.6	6.6	2.9	445.6	....	97.6	12.4	[illegible]	6.6	[illegible]	[illegible]	[illegible]	[illegible]
North Branford	....	....	81.7	9.2	220.0	156.3	9.2	....	147.1	....	55.2	....	9.2	18.4	9.2	9.2	[illegible]	[illegible]
North Haven	4.5	....	187.0	4.5	311.5	[illegible]	....	4.5	216.1	4.5	62.8	....	17.9	4.5	8.0	....	[illegible]	[illegible]
Orange	....	....	153.0	14.7	416.9	307.7	7.3	7.3	278.1	....	102.6	14.7	29.3	7.3	....	7.3	[illegible]	[illegible]
Seymour	....	....	163.5	7.8	420.3	[illegible]	....	7.8	272.5	....	109.0	15.6	....	7.8	15.6	....	[illegible]	[illegible]
Wallingford	....	....	139.4	8.4	[illegible]	287.2	16.7	....	[illegible]	....	60.2	8.4	41.8	8.4	13.9	5.6	[illegible]	[illegible]
Waterbury	0.9	....	202.7	11.8	676.5	495.1	[illegible]	4.0	[illegible]	3.7	135.1	21.1	[illegible]	13.0	33.3	2.8	[illegible]	[illegible]
West Haven	....	....	189.4	17.0	591.8	462.2	5.7	....	[illegible]	....	107.9	11.4	15.2	5.7	11.4	1.9	[illegible]	[illegible]
Wolcott	....	....	113.0	23.8	211.5	158.9	7.9	....	151.0	....	31.8	15.9	31.8	15.9	23.8	7.9	[illegible]	[illegible]
Rest of County	2.0	....	160.6	12.5	415.1	325.4	8.3	....	312.9	2.1	50.1	18.8	22.9	12.5	4.2	6.3	[illegible]	[illegible]
New London County	0.9	....	139.9	15.3	420.5	288.8	2.6	3.1	270.6	1.3	101.1	15.7	27.5	9.2	17.0	3.9	[illegible]	[illegible]
East Lyme	....	....	[illegible]	....	351.3	251.7	8.8	....	237.1	....	70.3	17.6	26.3	....	8.8	....	[illegible]	[illegible]
Groton	....	....	85.2	5.2	[illegible]	191.0	....	....	185.9	....	51.6	5.2	[illegible]	7.7	15.5	2.6	[illegible]	[illegible]
Ledyard	....	....	67.6	....	135.3	[illegible]	....	....	[illegible]	....	[illegible]	6.8	6.8	....	6.8	....	[illegible]	[illegible]
Montville	....	....	82.0	....	[illegible]	[illegible]	....	....	195.5	6.3	37.8	6.3	6.3	18.9	18.9	6.3	[illegible]	[illegible]
New London	3.2	....	177.4	12.7	[illegible]	421.3	3.2	....	[illegible]	....	[illegible]	[illegible]	57.0	3.2	[illegible]	....	[illegible]	[illegible]
Norwich	2.4	....	210.5	31.5	[illegible]	[illegible]	2.4	7.3	[illegible]	2.4	115.2	[illegible]	29.0	14.5	16.9	12.1	[illegible]	[illegible]
Stonington	....	....	[illegible]	37.5	[illegible]	[illegible]	....	6.3	[illegible]	....	[illegible]	18.8	[illegible]	25.0	6.3	....	[illegible]	[illegible]
Waterford	....	....	[illegible]	17.6	[illegible]	[illegible]	....	11.7	240.7	....	[illegible]	[illegible]	5.9	5.9	5.9	11.7	[illegible]	[illegible]
Rest of County	....	....	130.8	16.4	429.9	301.4	7.0	2.3	278.0	2.3	105.1	[illegible]	[illegible]	7.0	14.0	....	77.1	[illegible]
Tolland County	1.0	....	129.6	6.8	301.8	211.2	3.9	1.0	203.5	....	63.3	11.7	[illegible]	5.8	10.7	1.0	[illegible]	[illegible]
Mansfield	....	....	91.8	5.4	[illegible]	[illegible]	....	....	[illegible]	....	32.4	....	16.2	5.4	....	5.4	[illegible]	[illegible]
Vernon	....	....	91.6	[illegible]	[illegible]	[illegible]	3.6	....	[illegible]	....	105.5	14.5	[illegible]	7.3	21.8	....	[illegible]	[illegible]
Rest of County	1.8	....	150.5	1.8	[illegible]	232.9	5.3	1.8	218.7	....	52.9	14.1	31.8	10.6	8.8	....	[illegible]	[illegible]
Windham County	4.7	....	162.5	13.0	511.7	367.4	13.0	3.5	[illegible]	....	108.3	41.8	18.8	13.0	21.2	3.5	[illegible]	[illegible]
Killingly	....	....	[illegible]	14.7	[illegible]	[illegible]	[illegible]	....	[illegible]	....	88.0	58.7	22.0	7.3	[illegible]	....	[illegible]	[illegible]
Plainfield	8.3	....	199.4	16.6	[illegible]	[illegible]	8.3	....	[illegible]	....	182.8	49.9	33.2	8.3	....	[illegible]	[illegible]	[illegible]
Windham	[illegible]	....	137.1	....	[illegible]	[illegible]	5.1	10.2	330.1	....	[illegible]	15.2	[illegible]	15.2	[illegible]	5.1	[illegible]	[illegible]
Rest of County	[illegible]	....	151.7	17.7	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	....	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

¹ Death rates from accidents for residents [illegible] are based on allocation to place of occurrence of accidents. Rates for Connecticut on this occurrence basis would be: All accidents, 33.9; Motor Vehicle accidents, 15.0.

TABLE I(a)

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TABLE 19—DEATH RATES PER 100,000 POPULATION FROM SELECTED CAUSES, CORRECTED FOR RESIDENCE FOR TOWNS OF OVER 10,000 POPULATION: CONNECTICUT, 1975

TOWN	Tuberculosis All forms (010–019)	Syphilis (090–097)	Malignant neoplasms (140–209)	Diabetes mellitus (250)	Cardiovascular diseases (390–458)	Diseases of the heart (390–398, 402, 404, 410–429)	Rheumatic fever and rheumatic heart disease (390–398)	Hypertensive heart disease (402, 404)	Ischemic heart disease (410–414)	Chronic endocarditis and myocardial insufficiency (424, 428)	Cerebrovascular Disease (430–438)	Arteriosclerosis (440)	Pneumonia (480–486)	Emphysema (492)	Cirrhosis of liver (571)	Nephritis and nephrosis (580–584)	All accidents[1] (E800–E949)	Motor vehicle accidents[1] (E810–E823)
Column Number	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17	18
CONNECTICUT	0.6	0.2	188.4	14.6	420.2	317.4	6.5	3.2	282.8	0.7	79.5	13.3	22.4	6.8	14.8	2.6	29.8	[illegible]
Fairfield County	0.7	0.2	180.6	15.1	414.7	301.2	6.1	3.5	276.1	1.1	83.9	11.0	21.2	6.3	11.3	2.3	[illegible]	[illegible]
Bethel	....	....	137.5	....	293.7	207.7	[illegible]	....	214.1	7.9	110.2	7.9	[illegible]	15.7	....	....	[illegible]	[illegible]
Bridgeport	1.3	....	207.2	21.1	531.0	412.3	7.9	2.0	332.9	1.3	101.6	23.7	20.3	7.2	17.8	3.9	28.2	[illegible]
Brookfield	....	....	173.8	....	303.0	211.3	17.9	....	187.5	....	71.4	....	....	....	26.8	....	17.[illegible]	[illegible]
Danbury	....	....	205.8	5.3	360.3	79.1	1.8	3.5	229.0	3.5	77.3	10.5	14.1	3.5	19.3	3.9	19.[illegible]	[illegible]
Darien	4.5	....	174.9	13.5	372.9	295.3	9.0	....	17.9	4.5	71.7	22.4	17.9	[illegible]	[illegible]	4.5	[illegible]	[illegible]
Fairfield	....	....	213.9	10.2	441.8	221.1	8.0	10.2	205.4	1.7	95.8	6.8	18.7	1.7	[illegible]	....	[illegible]	[illegible]
Greenwich	....	....	221.3	20.0	374.4	293.4	9.5	3.2	243.3	....	53.7	7.9	22.1	1.6	12.8	1.6	23.[illegible]	[illegible]
Monroe	....	....	138.5	7.[illegible]	246.2	169.2	....	....	169.2	....	61.5	....	15.4	15.4	15.[illegible]	....	23.1	[illegible]
New Canaan	....	....	170.1	10.3	311.4	221.6	6.2	....	201.0	....	77.3	5.2	15.5	....	[illegible]	....	30.1	[illegible]
Newtown	....	....	147.9	5.9	207.7	218.9	....	....	207.1	....	47.3	35.5	17.8	....	[illegible]	....	17.[illegible]	[illegible]
Norwalk	1.3	2.5	151.5	5.0	418.1	330.4	10.0	3.9	307.9	1.3	91.4	14.8	22.5	13.8	16.3	1.3	25.[illegible]	[illegible]
Ridgefield	....	....	132.0	10.2	218.3	177.7	10.2	10.2	147.2	....	30.5	10.2	25.4	....	10.2	....	10.[illegible]	[illegible]
Shelton	....	....	165.5	20.3	439.2	334.5	[illegible]	6.8	301.1	....	81.1	20.3	16.9	....	16.9	....	16.9	[illegible]
Stamford	1.8	....	191.1	16.6	431.0	317.6	15.9	2.8	262.5	....	93.3	12.0	20.3	7.4	10.0	1.[illegible]	33.[illegible]	[illegible]
Stratford	....	....	223.1	15.9	5[illegible]5.9	300.4	13.9	....	348.0	....	95.0	12.0	27.9	6.0	6.0	2.0	31.9	[illegible]
Trumbull	....	....	159.0	20.2	249.0	135.0	8.7	2.9	171.0	....	65.9	5.8	14.5	2.9	11.6	5.8	23.1	[illegible]
Westport	....	....	199.3	27.5	316.2	237.1	....	10.3	213.1	3.4	65.3	3.4	21.1	17.2	13.7	3.4	17.[illegible]	[illegible]
Wilton	....	....	149.7	12.0	231.4	173.7	....	....	161.7	....	101.[illegible]	....	....	....	12.0	6.0	29.9	2[illegible]
Rest of County	....	....	137.1	9.3	220.9	171.3	....	3.1	166.1	....	46.7	18.7	9.3	9.3	3.1	3.1	15.6	[illegible]
Hartford County	1.0	0.1	186.4	12.4	413.8	299.9	5.1	3.3	277.9	0.5	81.8	13.9	23.0	5.2	15.0	1.8	30.9	15
Berlin	....	....	178.8	6.6	461.8	344.4	....	6.6	291.4	....	92.7	....	19.9	....	6.6	....	26.[illegible]	[illegible]
Bloomfield	....	....	172.3	34.7	310.8	232.5	5.0	....	212.0	....	34.7	5.0	....	....	9.9	....	[illegible]	[illegible]
Bristol	....	....	158.8	18.2	[illegible]	[illegible]	9.1	1.8	[illegible]	....	91.2	3.6	12.8	10.9	10.9	....	31.[illegible]	[illegible]
East Hartford	....	....	179.6	16.5	372.2	274.1	7.[illegible]	7.3	211.4	....	37.[illegible]	25.[illegible]	22.2	7.4	1.[illegible]	3.7	13.5	[illegible]
Enfield	....	....	121.[illegible]	11.1	328.9	265.7	6.7	2.2	186.7	....	65.9	4.4	[illegible]	....	11.1	4.4	[illegible]	[illegible]
Farmington	....	....	178.3	6.4	433.1	331.2	10.1	6.4	[illegible]	....	82.8	6.4	25.5	6.4	12.7	....	[illegible]	[illegible]
Glastonbury	....	....	103.4	4.3	[illegible]	185.3	4.3	8.5	[illegible]	....	51.7	12.9	8.6	....	8.6	4.3	17.[illegible]	[illegible]
Hartford	1.2	0.6	211.0	14.[illegible]	[illegible]	321.9	3.2	2.0	300.3	1.3	83.7	12.3	29.1	5.5	30.4	1.9	31.1	[illegible]
Manchester	4.2	....	[illegible]	21.2	487.3	317.5	6.4	12.7	[illegible]	....	91.1	31.8	30.9	10.0	16.9	....	[illegible]	[illegible]
New Britain	....	....	216.1	8.7	215.5	261.5	3.7	1.2	310.4	....	119.3	10.9	26.1	6.7	16.1	3.7	31.[illegible]	[illegible]
Newington	....	....	170.9	3.6	265.5	270.0	....	....	256.2	....	65.5	18.2	18.2	3.6	7.3	7.2	21.8	[illegible]
Plainville	....	....	178.[illegible]	6.0	369.0	273.8	....	....	261.9	....	65.5	....	6.0	....	23.8	....	17.9	[illegible]
Rocky Hill	....	....	282.3	9.0	414.4	360.4	9.0	....	351.4	....	27.0	18.0	18.0	9.0	27.0	9.0	45.0	[illegible]
Simsbury	....	....	79.6	....	223.9	144.3	....	5.0	139.3	....	74.[illegible]	....	19.9	....	11.9	....	24.9	[illegible]
Southington	3.0	....	153.8	11.0	[illegible]	275.1	....	5.9	257.[illegible]	....	59.2	8.9	23.7	....	3.0	....	[illegible]	[illegible]
South Windsor	6.3	....	201.[illegible]	25.2	213.8	144.7	6.3	....	119.5	....	37.7	12.[illegible]	....	....	12.6	....	37.[illegible]	[illegible]
West Hartford	....	....	217.1	11.8	[illegible]	402.9	8.8	1.5	[illegible]	1.5	117.[illegible]	14.7	[illegible]	5.9	10.2	....	30.9	[illegible]
Wethersfield	....	....	179.5	11.0	487.2	350.0	7.3	3.7	343.0	....	[illegible]	[illegible]	11.7	3.7	7.3	....	29.3	[illegible]
Windsor	....	....	159.0	4.2	380.1	[illegible]	4.[illegible]	....	255.2	....	83.7	16.7	[illegible]	8.4	12.0	....	25.1	[illegible]
Windsor Locks	....	....	191.1	6.9	379.3	265.2	6.9	....	241.4	....	75.9	[illegible]	6.0	6.9	20.7	....	[illegible]	[illegible]
Rest of County	1.[illegible]	....	182.8	7.[illegible]	347.8	[illegible]	3.7	1.9	[illegible]	1.9	[illegible]	11.[illegible]	26.1	1.9	[illegible]	1.9	[illegible]	[illegible]
Litchfield County	0.7	....	196.1	10.0	480.2	363.6	7.9	8.6	335.5	....	73.8	14.0	15.0	8.6	14.0	5.3	35.8	[illegible]
New Milford	....	....	117.3	....	305.1	327.2	6.2	....	314.8	....	49.4	....	6.[illegible]	....	....	6.2	21.7	[illegible]
Plymouth	....	....	111.1	38.6	[illegible]	237.0	....	....	259.3	....	74.1	13.5	27.8	13.5	18.5	9.[illegible]	9.3	[illegible]
Torrington	....	....	277.0	23.1	[illegible]	410.1	6.3	18.9	375.4	....	82.0	[illegible]	15.8	15.8	28.4	12.6	31.[illegible]	[illegible]
Watertown	....	....	145.1	10.4	[illegible]	310.9	20.7	10.4	274.6	....	57.0	[illegible]	[illegible]	5.2	....	6.2	46.0	[illegible]
Winchester	....	....	293.2	77.7	531.0	407.1	....	8.8	350.5	....	70.8	8.[illegible]	25.5	....	8.8	....	20.5	[illegible]
Rest of County	1.6	....	189.7	4.9	486.5	371.3	8.1	6.5	343.7	....	81.1	13.0	17.2	8.1	10.2	1.6	48.5	17.[illegible]
Middlesex County	1.7	....	163.2	15.0	461.3	348.9	5.8	3.3	327.2	1.7	84.1	17.5	21.6	10.0	10.2	5.0	32.5	[illegible]
Clinton	....	....	203.7	9.3	324.1	240.7	....	....	222.2	....	46.3	9.3	18.5	....	18.5	....	37.0	[illegible]
Middletown	....	....	139.3	19.5	431.8	323.7	5.8	2.8	303.8	....	91.9	2.8	8.4	10.5	36.2	5.0	23.1	[illegible]
Rest of County	2.7	....	163.9	13.6	405.9	374.6	6.8	4.1	354.2	2.7	83.3	23.9	23.5	6.8	10.9	6.4	35.4	[illegible]
New Haven County	0.8	0.1	225.8	16.6	466.3	360.1	7.7	2.2	223.7	0.3	79.3	11.8	24.7	5.6	15.2	2.3	27.0	11
Ansonia	....	....	222.8	5.0	490.1	371.3	10.3	5.0	316.5	....	69.3	30.6	21.8	9.0	5.0	....	31.7	[illegible]
Branford	4.7	....	205.6	....	562.1	471.[illegible]	....	....	366.2	....	84.5	....	[illegible]	....	9.4	....	[illegible]	[illegible]
Cheshire	4.7	....	137.4	9.5	312.8	232.2	9.5	....	223.8	....	61.6	9.5	23.4	9.5	9.5	....	47.[illegible]	[illegible]
Derby	....	....	260.5	33.5	415.4	269.7	....	8.4	325.1	....	50.4	....	23.2	8.4	....	....	15.6	[illegible]
East Haven	....	....	173.4	[illegible]	391.1	314.5	8.1	....	221.8	....	63.5	....	16.1	8.1	12.1	....	[illegible]	[illegible]
Guilford	7.0	....	118.9	7.0	312.7	279.7	7.0	....	216.8	....	42.0	....	21.0	....	....	....	49.0	[illegible]
Hamden	....	2.0	224.0	20.0	472.0	368.0	10.0	....	221.0	2.0	32.0	14.0	23.0	6.0	10.0	10.0	20.0	[illegible]
Madison	....	....	121.0	....	338.5	272.7	8.3	3.3	217.9	....	57.9	....	21.8	....	8.3	....	8.[illegible]	[illegible]
Meriden	1.8	....	211.4	19.5	[illegible]	390.1	3.0	3.0	[illegible]	....	120.8	4.9	21.3	1.8	19.5	1.8	[illegible]	[illegible]
Milford	....	....	179.7	21.5	[illegible]	298.9	2.0	....	287.1	2.0	63.4	9.8	23.4	2.0	17.6	[illegible]	[illegible]	[illegible]
Naugatuck	....	....	139.4	31.9	466.1	381.[illegible]	4.0	....	312.0	....	87.0	19.9	19.9	8.0	19.9	8.0	31.9	[illegible]
New Haven	1.5	....	253.5	[illegible]	531.4	411.[illegible]	7.6	3.0	[illegible]	....	[illegible]	15.1	37.0	11.3	16.5	2.3	28.0	[illegible]
North Branford	....	....	150.5	8.[illegible]	314.8	200.0	25.1	....	113.0	....	[illegible]	....	....	....	....	....	....	[illegible]
North Haven	....	....	213.0	4.3	328.1	[illegible]	8.7	....	191.3	....	56.5	8.7	8.7	4.3	13.0	....	4.3	[illegible]
Orange	....	....	145.7	13.2	337.4	[illegible]	6.[illegible]	....	178.6	....	59.3	13.[illegible]	[illegible]	....	13.[illegible]	6.6	19.9	[illegible]
Seymour	....	....	223.0	22.4	477.6	350.7	[illegible]	....	328.4	....	111.9	14.9	14.9	....	7.[illegible]	....	17.3	[illegible]
Southbury	....	....	271.8	....	592.2	440.0	48.5	....	[illegible]	....	77.7	....	19.4	9.7	....	....	77.[illegible]	[illegible]
Wallingford	....	....	165.7	17.1	305.7	301.1	11.4	2.0	320.0	....	102.0	....	42.9	8.0	17.1	2.0	31.4	[illegible]
Waterbury	....	....	210.0	11.4	533.8	307.7	7.0	6.1	300.0	....	[illegible]	18.4	18.4	5.3	22.8	1.8	[illegible]	[illegible]
West Haven	....	....	205.6	13.[illegible]	201.7	416.7	3.7	....	311.1	....	53.7	18.5	[illegible]	1.9	[illegible]	3.7	[illegible]	[illegible]
Wolcott	....	....	160.3	[illegible]	282.4	251.9	....	....	236.6	....	[illegible]	....	[illegible]	....	[illegible]	....	[illegible]	[illegible]
Rest of County	....	....	150.5	8.7	379.1	235.1	2.9	....	220.3	....	58.0	8.7	20.3	5.8	[illegible]	2.0	37.7	20.[illegible]
New London County	0.6	....	166.9	11.6	403.6	292.2	7.1	3.8	265.0	1.7	80.6	12.5	21.2	5.0	15.0	2.5	33.4	15.4
East Lyme	....	....	139.5	31.0	[illegible]	[illegible]	31.0	15.5	[illegible]	7.8	[illegible]	[illegible]	21.3	7.8	23.3	....	[illegible]	[illegible]
Groton	....	....	108.4	21.7	[illegible]	181.[illegible]	13.0	2.7	157.2	....	59.6	8.1	[illegible]	2.7	3.1	2.7	19.0	8
Ledyard	....	....	63.7	6.[illegible]	114.[illegible]	82.8	....	....	90.1	....	[illegible]	6.1	6.4	....	....	....	41.0	[illegible]
Montville	....	....	86.4	12.3	210.0	155.8	....	....	111.1	....	[illegible]	6.2	6.2	6.2	6.2	6.2	[illegible]	[illegible]
New London	....	....	[illegible]	[illegible]	[illegible]	[illegible]	3.3	....	[illegible]	....	102.[illegible]	16.0	[illegible]	[illegible]	19.9	3.3	[illegible]	[illegible]
Norwich	....	....	188.3	[illegible]	501.1	382.0	11.2	6.7	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	4.5	[illegible]	[illegible]
Stonington	....	....	[illegible]	11.9	416.4	[illegible]	....	6.0	221.8	6.0	119.0	[illegible]	17.[illegible]	6.0	11.9	....	[illegible]	11
Waterford	....	....	211.8	[illegible]	[illegible]	307.7	....	....	[illegible]	....	[illegible]	23.0	[illegible]	11.0	11.0	....	[illegible]	[illegible]
Rest of County	3.3	....	121.1	11.[illegible]	[illegible]	224.2	3.3	1.6	[illegible]	1.6	58.9	1.6	19.6	4.0	6.[illegible]	1.6	30.0	21.[illegible]
Tolland County	0.9	....	142.5	13.9	251.4	180.7	1.9	0.9	177.6	0.9	40.7	13.0	17.6	4.6	7.4	1.9	[illegible]	11.[illegible]
Mansfield	....	....	91.4	11.1	[illegible]	116.7	5.6	....	111.1	....	11.1	....	11.1	[illegible]	....	....	[illegible]	[illegible]
Stafford	....	....	200.0	20.0	[illegible]	[illegible]	....	....	220.0	....	70.0	30.0	40.0	20.0	10.0	10.0	20.0	[illegible]
Vernon	....	....	[illegible]	10.4	[illegible]	201.4	....	....	191.0	....	[illegible]	13.9	20.8	3.5	10.4	....	[illegible]	[illegible]
Rest of County	1.9	....	128.4	15.6	[illegible]	200.8	1.9	1.9	185.2	1.9	29.2	[illegible]	11.7	1.9	7.8	1.9	[illegible]	21.[illegible]
Windham County	....	1.1	197.0	21.4	489.5	381.8	7.8	1.1	261.8	1.1	70.0	11.1	23.6	4.5	21.0	2.2	40.0	[illegible]
Killingly	....	6.7	[illegible]	13.3	[illegible]	[illegible]	20.0	....	[illegible]	....	80.0	26.7	20.0	6.7	13.3	....	[illegible]	[illegible]
Plainfield	....	....	211.1	[illegible]	491.8	[illegible]	[illegible]	....	[illegible]	....	[illegible]	10.4	16.4	....	8.2	8.2	[illegible]	[illegible]
Windham	....	....	220.6	9.8	511.7	411.8	4.9	4.9	[illegible]	4.9	73.5	9.8	19.6	4.9	31.3	....	28.4	[illegible]
Rest of County	....	....	[illegible]	21.[illegible]	417.[illegible]	315.6	2.4	....	[illegible]	....	[illegible]	4.7	21.1	4.7	11.[illegible]	2.[illegible]	[illegible]	[illegible]

[1] Death rates for residence accident total and motor vehicle;

TABLE I(b)

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CANCER INCREASE IN CONNECTICUT 1970-75

BY SITE AND KNOWN RADIATION SENSITIVITY

	Increase in %	Increase in No.	Fraction of Incr. No.	ICRP Radiation Sensitivity
All Sites Combined	+12%	793	100%	---
Respiratory System	+25%	292	37%	HIGH
Breast Cancer	+12%	127	16%	Not Class.
Digestive System	+6%	155	20%	HIGH
Pancreas	+32%	84	11%	HIGH
L.-Intest.	+11%	71	9%	HIGH

Source: Connecticut Annual Registration Reports

TABLE 2

TABLE XVII—DEATHS AND DEATH RATES PER 100,000 POPULATION FROM MALIGNANT NEOPLASMS, ACCORDING TO SITE: CONNECTICUT, 1966-1975

Site of Disease (International List, Eighth Revision 1967)	1975	1974	1973	1972	1971	1970	1969	1968	1967	1966
Total (140-209)	5909	5691	5474	5383	5185	5116	5102	5017	4818	4673
Digestive organs and peritoneum (150-159)	1777	1736	1600	1687	1562	1622	1516	1578	1556	1535
Respiratory system (160-163)	1284	1185	1173	1102	1024	992	953	920	823	814
Breast (174)	595	552	532	528	534	468	530	481	449	455
Lymphatic and Hematopoietic system (200-209)	541	535	529	504	508	513	530	452	435	395
Female genital organs (180-184)	320	324	319	296	328	353	320	332	353	291
Male genital organs (185-187)	286	254	296	240	266	246	245	223	217	231
Urinary system (188-189)	274	269	263	267	230	231	223	254	240	264
Buccal cavity and pharynx (140-149)	149	150	126	139	150	131	147	158	122	129
Brain, spinal cord, meninges, and eye (190-192)	149	144	120	124	136	127	139	95	98	80
Skin (172-173)	73	75	64	60	60	53	67	71	61	61
Soft tissue (171)	38	36	29	37	23	27	31	46	38	28
Bone, including jawbone (170)	19	32	29	22	26	27	29	23	33	20
Endocrine glands (193-194)	23	26	18	23	24	26	21	15	14	22
Other and unspecified sites (195-199)	381	379	376	359	314	300	351	369	379	348
	Rate per 100,000 population									
Total (140-209)	188.4	182.3	176.1	174.4	169.3	168.1	169.1	169.2	164.5	162.7
Digestive organs and peritoneum (150-159)	56.6	55.4	51.5	54.6	51.0	53.3	50.2	53.2	53.1	53.4
Respiratory system (160-163)	40.9	38.0	37.7	35.7	33.4	32.6	31.6	31.0	23.1	28.3
Breast (174)	19.0	17.7	17.1	17.1	17.4	15.4	17.6	16.2	15.3	15.8
Lymphatic and hematopoietic system (200-209)	17.2	17.1	17.0	16.3	16.6	16.9	17.6	15.2	14.9	13.7
Female genital organs (180-184)	10.2	10.4	10.3	9.6	10.7	11.6	10.6	11.2	12.0	10.1
Male genital organs (185-187)	9.1	8.1	9.5	7.8	8.7	8.1	8.1	7.5	7.4	8.0
Urinary system (188-189)	8.7	8.6	8.5	8.6	7.5	7.6	7.4	8.6	8.2	9.2
Buccal cavity and pharynx (140-149)	4.7	4.8	4.1	4.5	4.9	4.3	4.9	5.3	4.2	4.5
Brain, spinal cord, meninges, and eye (190-192)	4.7	4.6	3.9	4.0	4.4	4.2	4.6	3.2	3.3	2.8
Skin (172-173)	2.3	2.4	2.1	1.9	2.0	1.7	2.2	2.4	2.1	2.1
Soft tissue (171)	1.2	1.2	0.9	1.2	0.8	0.9	1.0	1.6	1.3	1.0
Bone, including jawbone (170)	0.6	1.0	0.9	0.7	0.8	0.9	1.0	0.8	1.1	0.7
Endocrine glands (193-194)	0.7	0.8	0.6	0.7	0.8	0.9	.7	0.5	0.5	0.8
Other and unspecified sites (195-199)	12.1	12.1	12.1	11.6	10.3	9.8	11.6	12.5	12.9	12.1

TABLE 2(a)

A-315

INCREASE IN NUMBER OF

CANCER DEATHS IN CONNECTICUT - 1970-75

AFTER START OF MILLSTONE POINT NUCLEAR PLANT IN 1970

	Total No.	No. White	No. Non-Wh.	Population(Million)
1970	5197	5005	192	3.044
1975	6001	5711	290	3.137
PERCENT INCREASE	+16%	+14%	+51%	+3%

Source: Table 10, State of Connecticut, Department of Health Annual Register Reports

TABLE 3

A-316

RELATIVE EFFECT OF RADIOACTIVE

RELEASES ON CANCER DEATHS IN

DIFFERENT AGE GROUPS IN CONNECTICUT

1970 - 1975

AGE	20-24 yrs	25-49 yrs	50-54 yrs	55-64 yrs	65 + yrs	All AGES
Number in 1970	20	540	412	1207	2929	5197
Number in 1975	23	486	422	1439	3559	6001
% Change in cancer deaths	+15%	-10%	+2%	+19%	+22%	+15%
Change in Pop.	+15%	+5%	-2%	+10%	+8%	+3%
Net Change in cancer rate	0%	-15%	+4%	+9%	+14%	+12%
% of all Cancer Deaths in 1975	0.4%	8%	7%	24%	59%	----

(Source: Connecticut Annual Registration Reports)

TABLE 4

A-317

RECENT CHANGES IN U. S.

CANCER MORTALITY RATES 1972-75

IN STATES WITH AND WITHOUT

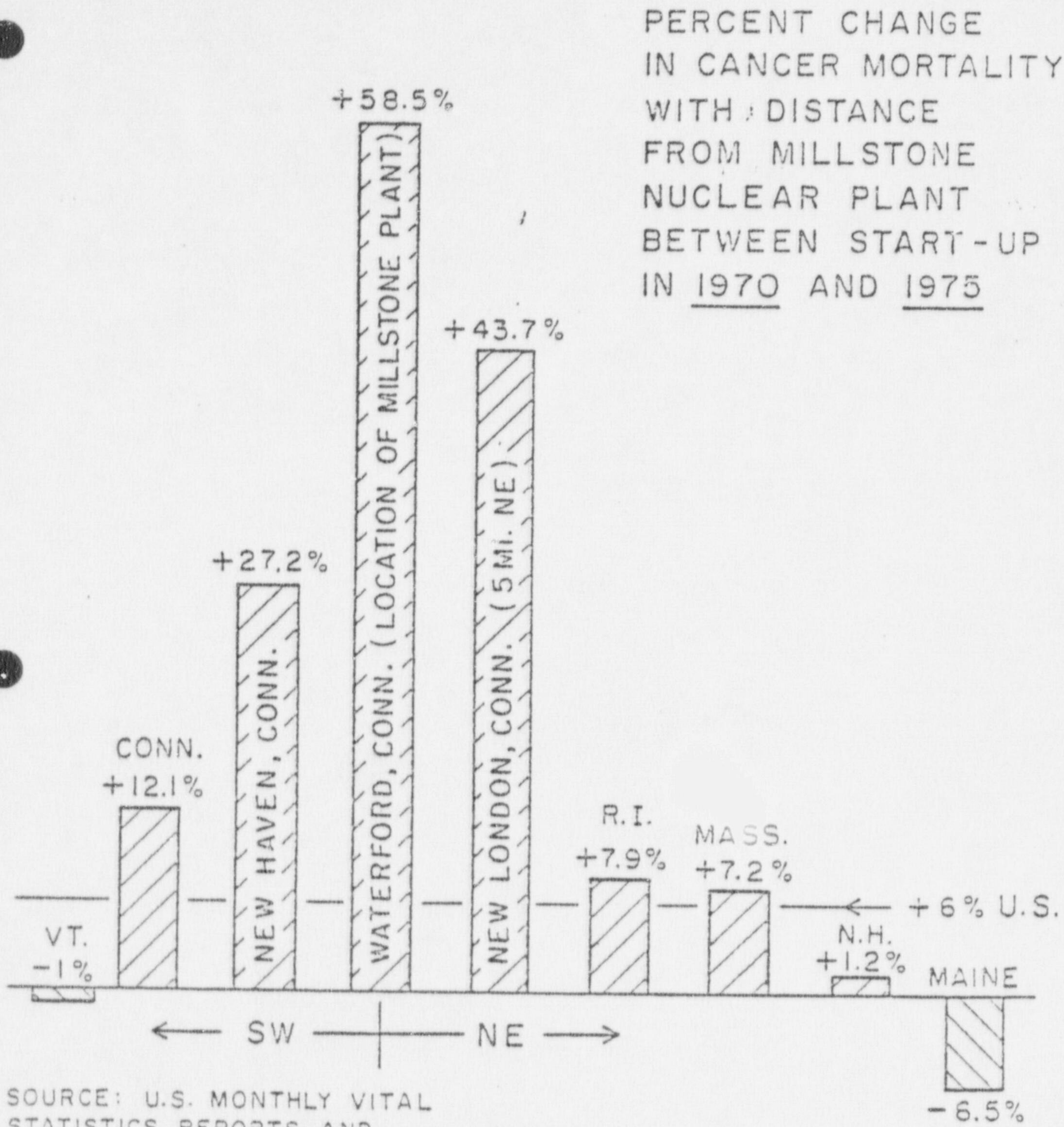
LARGE NUCLEAR RELEASES

AREA	% CHANGE	NUCLEAR FACILITY
Connecticut	+8.6%	Millstone and Haddam Neck
Tennessee	+8.1%	Oak Ridge
Rhode Island	+8.0%	Millstone and Haddam Neck
New Jersey	+5.7%	Oyster Creek (BWR)
S. Carolina	+5.7%	Savannah River
Wash. State	+5.0%	Hanford
U. S. Average	+3.4%	----
N. Y. City	-1.1%	2 PWR (1962;73)
Virginia	-1.1%	2 PWR (1972;73)
Maine	-1.3%	1 PWR (1972)
Hawaii	-1.5%	No Nuclear Reactor
New Hampshire	-2.0%	No Nuclear Reactor
Montana	-4.4%	No Nuclear Reactor
Alaska	-10.6%	No Nuclear Reactor

Source: U.S. Monthly Vital Statistics

TABLE 5

17-318



SOURCE: U.S. MONTHLY VITAL STATISTICS REPORTS AND STATE OF CONN. VITAL STATISTICS

FIGURE 1

A-319

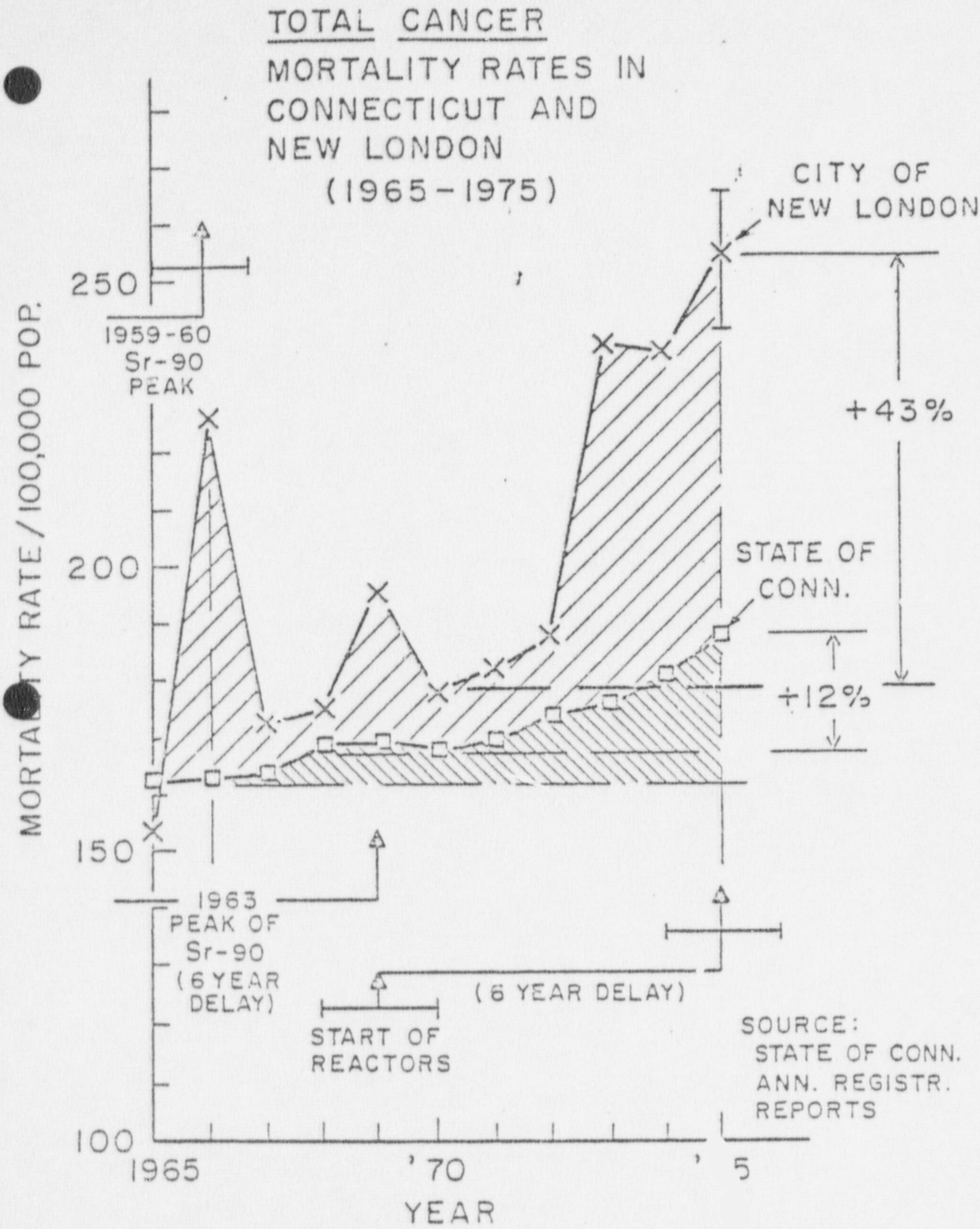


FIGURE 2

A-320

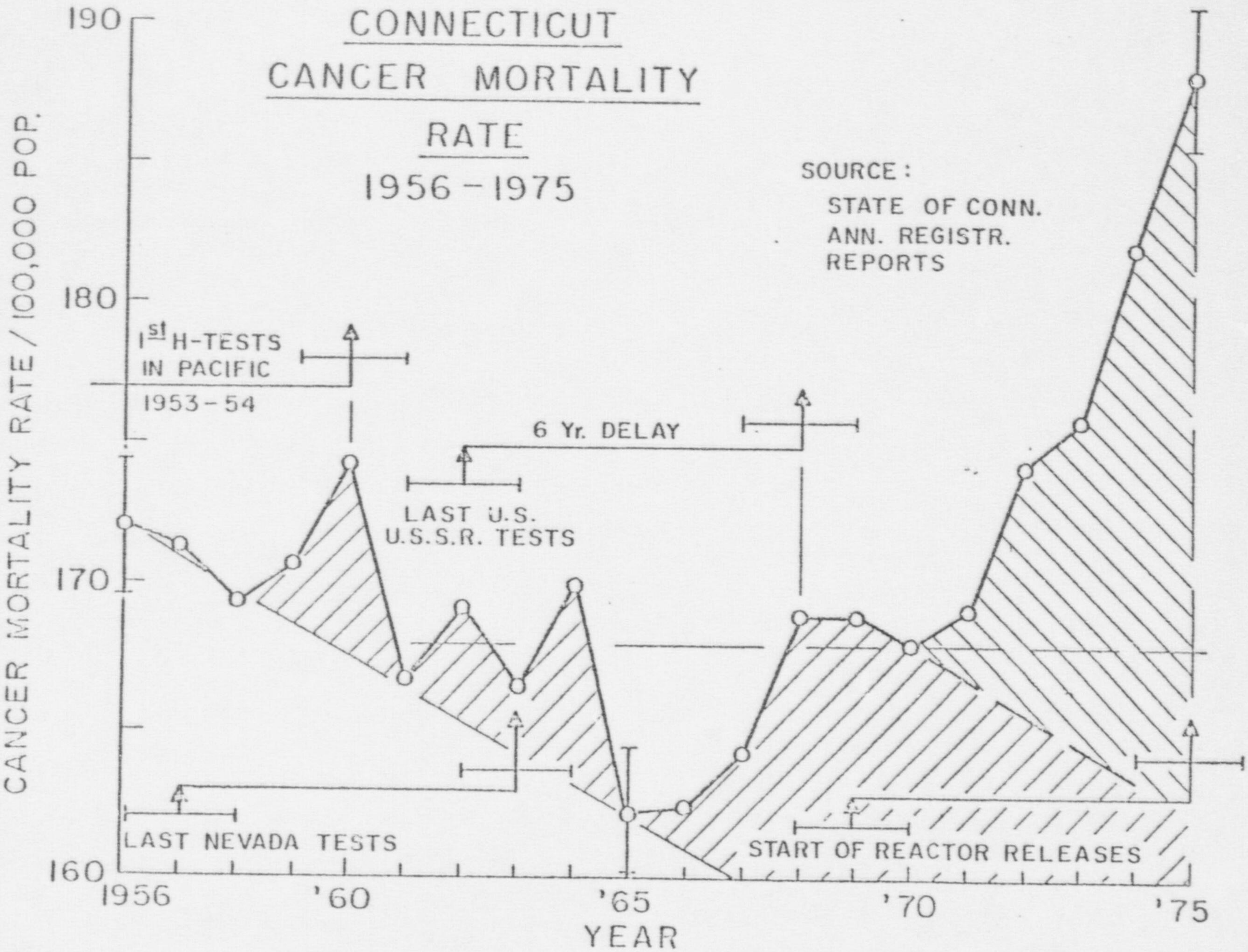
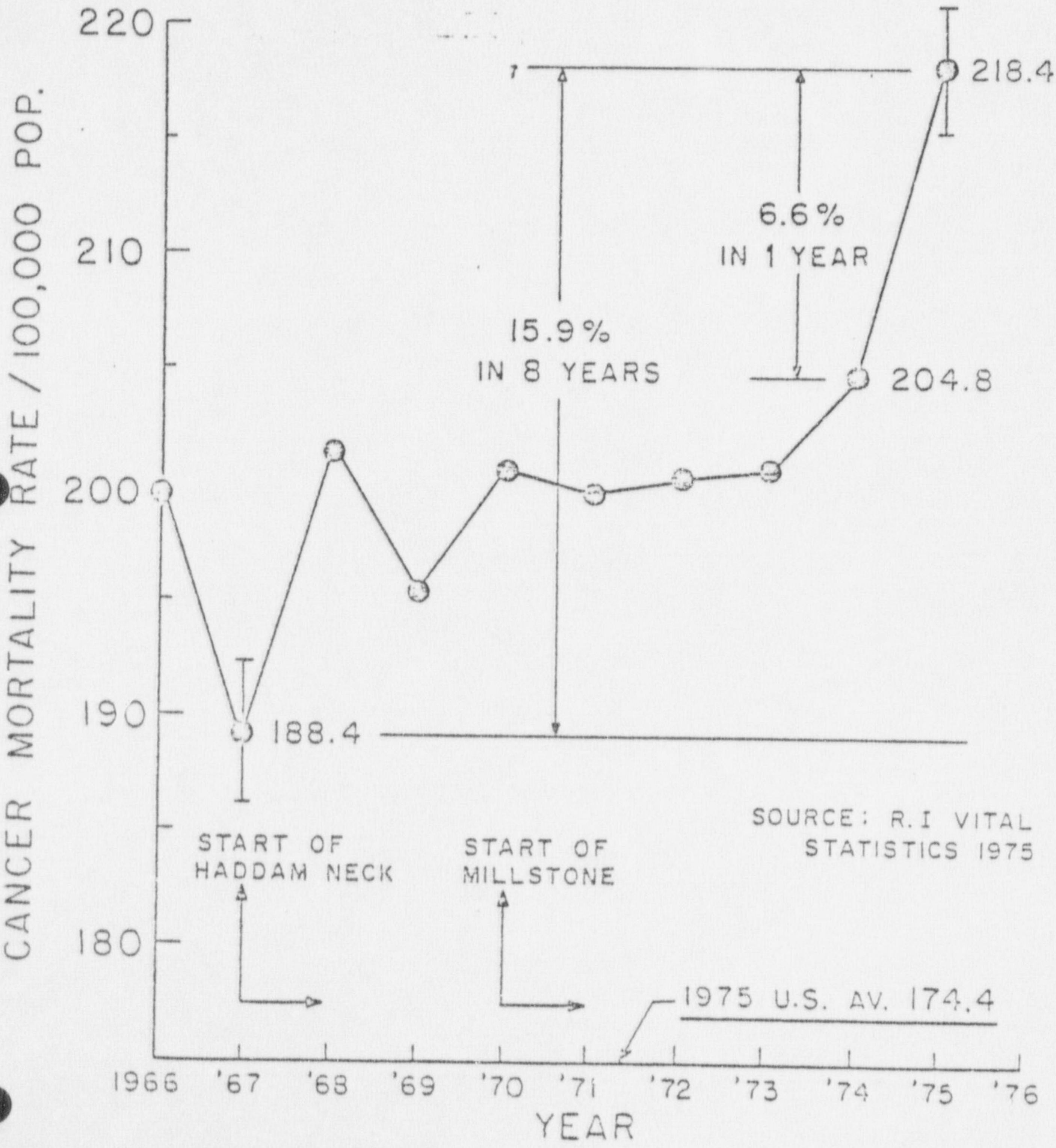

CONNECTICUT
CANCER MORTALITY
RATE
1956 - 1975
SOURCE :
STATE OF CONN.
ANN. REGISTR.
REPORTS
CANCER MORTALITY RATE / 100,000 POP.
190
180
170
160
1st H-TESTS
IN PACIFIC
1953 - 54
6 Yr. DELAY
LAST U.S.
U.S.S.R. TESTS
LAST NEVADA TESTS
START OF REACTOR RELEASES
1956
'60
'65
'70
'75
YEAR

FIGURE 3

F-32

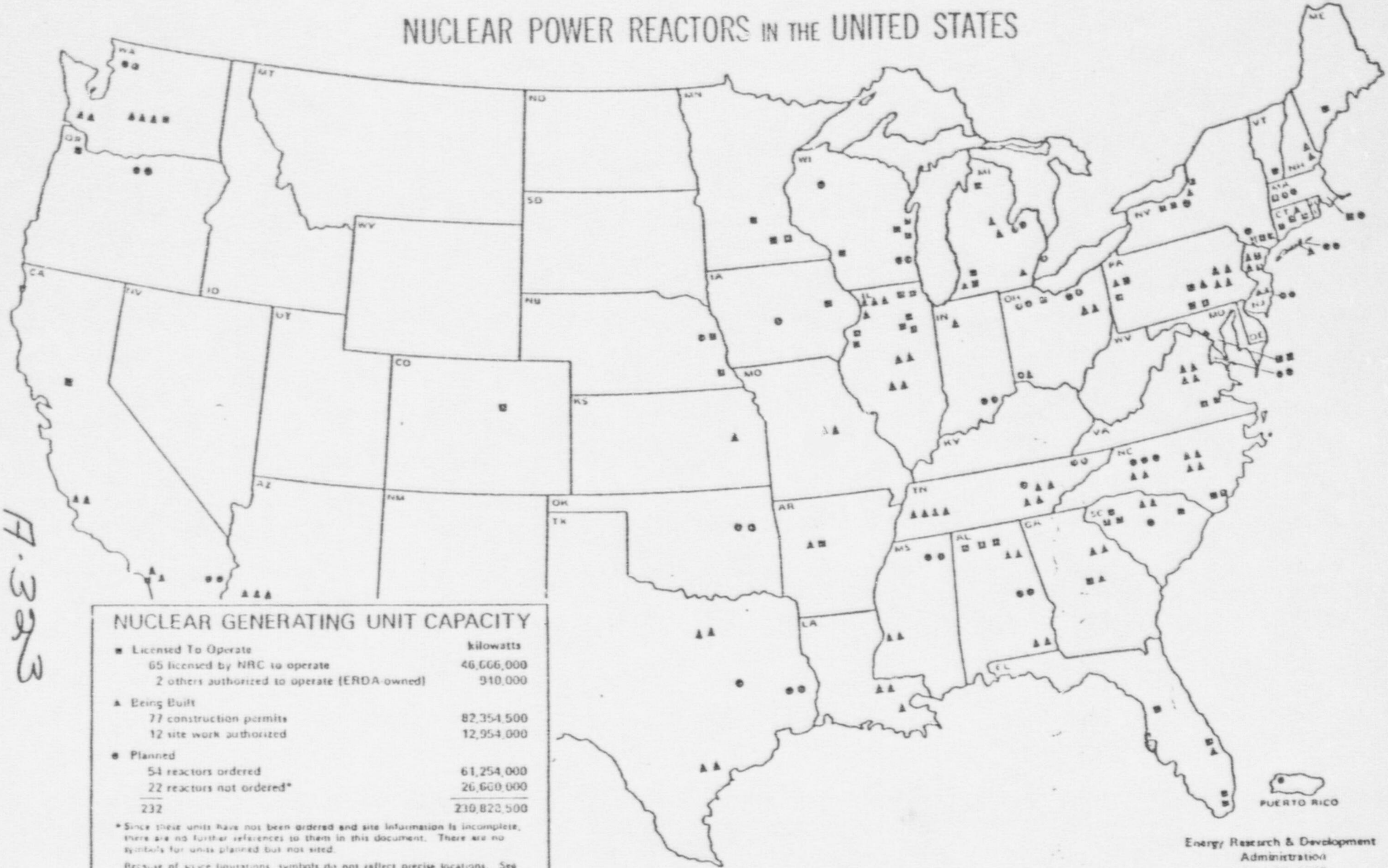
CANCER MORTALITY RATE IN RHODE ISLAND SINCE START OF HADDAM NECK NUCLEAR REACTOR IN 1967



A-322

FIGURE 4

NUCLEAR POWER REACTORS IN THE UNITED STATES



NUCLEAR GENERATING UNIT CAPACITY

	kilowatts
■ Licensed To Operate	
65 licensed by NRC to operate	46,666,000
2 others authorized to operate (ERDA-owned)	910,000
▲ Being Built	
77 construction permits	82,354,500
12 site work authorized	12,954,000
● Planned	
54 reactors ordered	61,254,000
22 reactors not ordered*	26,660,000
232	230,822,500

* Since these units have not been ordered and site information is incomplete, there are no further references to them in this document. There are no symbols for units planned but not sited.

Because of space limitations, symbols do not reflect precise locations. See reverse side for site information.

(over)

Energy Research & Development
Administration
June 30, 1977

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COMMERCIAL NUCLEAR POWER REACTORS IN THE UNITED STATES

SITE	PLANT NAME	CAPACITY, NET (MW)	UTILITY	COMMERCIAL OPERATION
[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

*Site not selected

17-324

Table 5. DEATH AND DEATH RATES FOR MAJOR CAUSES OF DEATH FOR THE UNITED STATES, EACH REGION, DIVISION AND STATE; BY COLOR AND SEX FOR THE UNITED STATES: 1972

[By place of residence. Refers only to resident deaths occurring within the United States. Excludes fetal deaths. Rates per 100,000 estimated population in each color-sex group and area. Numbers after causes of death are category numbers of the Eighth Revision, International Classification of Diseases, Adapted, 1965]

Color, sex, and area	Diseases of heart (390-398,402,404, 410-429)		Malignant neoplasms, including neoplasms of lymphatic and hematopoietic tissues (140-209)		Cerebrovascular diseases (430-438)		All accidents (E800-E949)		Motor vehicle accidents (E810-E823)	
	Number	Rate	Number	Rate	Number	Rate	Number	Rate	Number	Rate
United States	755,854	363.0	345,518	166.7	213,344	102.5	115,448	55.4	56,278	27.0
Male	424,634	418.5	188,474	185.7	95,380	94.0	79,752	78.6	40,194	39.6
Female	331,220	310.3	157,144	147.2	117,964	110.5	35,696	33.4	16,084	15.1
White	684,352	376.2	308,236	170.0	187,550	103.0	98,226	54.0	48,764	26.8
Male	385,802	434.1	167,678	188.7	83,210	93.6	67,190	75.6	34,566	38.9
Female	298,550	321.0	141,558	152.2	104,140	112.0	31,036	33.4	14,198	15.3
All other	71,482	271.5	36,382	138.2	25,994	98.7	17,222	65.4	7,514	28.5
Male	38,832	308.4	20,796	165.2	12,173	96.6	12,572	99.8	5,628	44.7
Female	32,650	237.6	15,586	113.4	13,824	100.6	4,650	33.8	1,886	13.7
Regions:										
Northeast	206,334	416.3	93,896	188.4	47,510	95.5	20,516	41.3	9,224	18.5
North Central	222,180	388.8	96,342	167.8	61,866	107.4	30,676	53.4	14,240	26.1
South	221,154	344.7	103,210	159.5	72,324	111.2	42,392	65.2	21,458	33.0
West	105,226	232.0	52,370	145.5	31,644	83.4	21,856	60.7	10,558	29.5
NORTHEAST										
New England	45,830	379.1	21,490	185.3	12,056	99.6	5,325	44.0	2,194	18.1
Maine	4,626	418.9	1,924	187.5	1,108	116.6	558	54.4	222	21.6
New Hampshire	2,762	358.8	1,440	184.0	890	115.0	348	45.0	160	20.7
Vermont	1,884	385.1	788	187.0	432	53.9	208	45.2	100	21.7
Massachusetts	22,754	392.6	11,032	190.3	5,754	99.3	2,786	48.0	1,110	19.2
Rhode Island	4,094	622.5	1,938	200.0	678	90.6	360	37.2	134	13.8
Connecticut	9,988	324.3	5,388	176.0	2,904	94.4	1,068	34.7	470	15.3
Middle Atlantic	160,144	428.7	71,926	193.5	35,474	94.2	15,183	40.4	7,028	18.7
New York	77,834	419.4	34,902	190.3	16,462	89.7	6,850	37.7	3,242	17.7
New Jersey	29,821	408.7	13,582	184.8	6,456	87.8	2,920	39.7	1,234	17.5
Pennsylvania	53,286	447.7	22,572	190.4	12,516	106.1	5,338	44.8	2,492	20.9
NORTH CENTRAL										
East North Central	158,188	380.4	67,478	165.4	41,220	101.1	20,319	49.8	9,828	24.1
Ohio	40,448	377.2	18,088	168.4	10,978	102.4	5,234	48.8	2,460	22.9
Indiana	18,204	355.4	8,482	160.1	6,054	114.5	2,933	55.5	1,500	28.4
Illinois	48,184	421.6	19,453	173.3	11,334	107.8	5,225	46.5	2,384	21.2
Michigan	30,826	338.6	14,030	154.2	7,933	88.7	4,540	50.0	2,288	25.2
Wisconsin	17,136	380.8	7,390	163.3	4,585	107.5	2,512	51.3	1,126	24.9
West North Central	65,182	398.1	28,668	173.7	20,434	123.0	10,346	62.6	5,122	30.9
Minnesota	13,283	337.6	6,352	161.3	4,410	113.7	2,258	59.0	1,072	27.7
Iowa	12,186	423.8	5,118	177.5	3,648	126.5	1,740	60.3	928	32.2
Missouri	20,184	425.4	8,304	128.1	6,274	132.2	2,934	61.8	1,468	30.9
North Dakota	2,182	339.4	886	129.7	722	113.9	430	67.8	190	30.0
South Dakota	2,776	397.9	1,112	163.6	744	109.4	768	112.9	304	44.7
Nebraska	6,186	396.3	2,712	177.5	2,016	131.9	942	61.6	484	31.7
Kansas	8,830	383.3	3,951	174.3	2,621	115.6	1,252	55.6	638	30.2
SOUTH										
South Atlantic	111,788	348.9	52,030	163.1	34,244	107.4	20,344	63.9	10,210	32.0
Delaware	2,124	372.0	894	156.6	434	76.4	264	46.2	132	23.1
Maryland	13,188	325.9	6,386	157.8	2,532	63.3	1,762	43.5	810	20.0
District of Columbia	2,552	339.4	1,534	234.7	722	93.1	416	59.3	112	14.9
Virginia	14,773	313.1	7,210	131.9	4,238	91.5	2,742	57.5	1,328	28.8
West Virginia	8,148	453.9	3,322	183.1	2,100	133.1	1,290	71.9	560	31.2
North Carolina	16,684	323.4	7,032	134.0	5,006	111.2	3,848	71.7	2,062	38.3
South Carolina	8,418	313.2	3,456	118.6	2,394	111.4	2,004	74.6	1,062	39.5
Georgia	14,099	310.5	6,398	134.9	5,040	128.0	3,560	77.3	1,694	33.8
Florida	30,396	418.4	15,617	215.2	9,176	126.3	4,624	63.0	2,420	32.9
East South Central	47,224	359.9	27,623	157.2	17,052	120.4	9,224	70.9	4,844	36.0
Kentucky	12,430	376.2	5,698	172.6	4,066	121.4	2,214	67.3	1,112	33.5
Tennessee	14,850	364.7	6,154	151.3	5,254	129.0	2,670	65.5	1,339	34.3
Alabama	11,234	322.5	5,300	[illegible]	4,700	133.5	2,925	74.0	1,380	39.2
Mississippi	7,790	345.3	3,448	152.6	3,252	125.3	1,824	80.9	954	42.3
West South Central	65,514	327.9	30,462	152.4	20,983	105.0	12,684	63.5	6,424	32.0
Arkansas	9,086	398.7	3,487	172.3	2,344	116.5	1,578	75.1	796	37.6
Louisiana	12,824	333.4	5,726	151.2	3,728	99.7	2,378	63.5	1,102	29.5
Oklahoma	10,123	383.9	4,463	167.7	3,210	125.1	1,700	64.6	828	30.5
Texas	34,884	298.5	16,788	144.3	11,076	95.4	7,108	61.2	3,740	32.2
WEST										
Mountain	22,890	255.5	10,742	121.0	6,510	73.5	6,670	74.0	3,386	38.1
Montana	2,239	319.4	1,130	123.6	692	95.3	612	82.7	350	48.0
Idaho	2,304	291.9	960	127.2	680	92.1	704	93.2	310	41.1
Wyoming	1,068	306.7	422	121.5	362	104.6	314	91.4	174	50.5
Colorado	6,310	265.4	2,704	117.1	1,640	69.4	1,300	57.8	712	30.1
New Mexico	2,068	192.0	1,174	110.1	790	65.1	962	89.4	504	46.9
Arizona	6,042	307.8	2,571	131.1	1,300	63.7	1,441	73.6	773	39.6
Utah	2,404	210.0	1,000	84.9	730	63.0	732	62.3	350	31.1
Nevada	1,725	233.0	737	137.3	449	78.1	478	76.7	224	39.0
Pacific	82,334	313.0	41,628	151.3	25,334	93.3	15,206	54.2	7,250	25.7
Washington	11,071	321.0	5,300	152.7	3,144	117.0	1,988	57.3	820	24.0
Oregon	7,422	310.6	2,510	161.6	2,516	115.1	1,379	63.1	760	34.3
California	62,150	301.4	31,036	152.0	18,984	92.6	11,404	55.3	5,410	25.7
Alaska	310	95.4	225	67.7	74	24.0	276	84.0	63	20.0
Hawaii	1,412	171.4	808	108.5	366	45.1	257	34.3	142	17.4

TABLE 5 (a)

.7325

Table 5. DEATHS AND DEATH RATES FOR MAJOR CAUSES OF DEATH FOR THE UNITED STATES, EACH REGION, DIVISION, AND STATE; AND BY COLOR AND SEX FOR THE UNITED STATES: 1975

[By place of residence. Refers only to resident deaths occurring within the United States. Excludes fetal deaths. Rates per 100,000 estimated population in each color-sex group and area. Numbers after causes of death are category numbers of the Eighth Revision, International Classification of Diseases, Adapted, 1965]

... and area	Diseases of heart (390-398,402,404, 410-429)		Malignant neoplasms, including neoplasms of lymphatic and hematopoietic tissues (140-209)		Cerebrovascular diseases (430-438)		All accidents (E800-E949)		Motor vehicle accidents (E810-E823)	
	Number	Rate	Number	Rate	Number	Rate	Number	Rate	Number	Rate
United States	716,215	336.2	365,693	171.7	194,038	91.1	103,030	48.4	45,853	21.5
Male	379,501	365.2	193,443	192.3	84,286	81.3	72,376	69.8	33,597	32.4
Female	336,714	299.7	169,250	152.1	109,752	100.4	30,654	28.0	12,258	11.2
White	648,037	350.0	325,519	175.0	170,689	92.2	87,760	47.4	39,883	21.5
Male	362,616	401.1	176,108	194.8	73,318	81.1	61,186	67.7	29,292	32.2
Female	285,421	301.3	149,411	157.7	97,371	102.8	26,574	28.0	10,701	11.4
All other	68,178	244.4	40,174	144.0	23,349	83.7	15,270	54.7	5,970	21.4
Male	36,885	277.1	23,335	175.3	10,988	82.4	11,190	84.1	4,505	33.8
Female	31,293	214.7	16,839	115.5	12,361	84.9	4,080	28.0	1,465	10.0
Regions:										
Northeast	183,274	382.7	96,947	195.0	41,633	84.2	18,243	36.9	7,631	15.4
North Central	203,535	352.9	99,700	171.8	54,459	93.1	26,172	45.4	11,633	20.0
South	219,275	321.9	112,001	164.4	66,858	94.2	38,079	55.9	17,612	25.9
West	104,161	275.0	57,653	152.2	30,650	80.9	20,530	54.2	9,074	24.0
NORTHEAST										
New England	43,287	354.9	23,718	194.4	10,378	85.7	4,315	35.5	1,335	16.1
Maine	4,127	399.7	1,939	185.0	1,044	98.6	524	49.5	225	21.2
New Hampshire	2,554	310.3	1,434	182.6	782	95.6	339	43.3	153	19.6
Vermont	1,671	354.9	819	173.9	257	82.2	225	49.9	106	22.5
Massachusetts	21,263	366.5	11,534	198.4	5,113	87.7	2,400	41.2	927	15.9
Rhode Island	3,836	410.5	2,032	216.0	761	82.1	310	33.4	135	14.6
Connecticut	9,932	319.9	5,870	189.0	2,421	80.5	990	32.0	441	14.2
Middle Atlantic	143,937	391.8	73,231	196.5	31,285	83.3	13,428	36.0	3,479	23.1
New York	60,381	332.9	35,530	194.1	13,948	77.0	5,836	32.3	2,370	13.1
New Jersey	27,346	373.8	14,285	195.3	5,812	80.0	2,544	34.8	1,107	15.1
Pennsylvania	49,263	416.5	23,416	198.0	11,253	95.2	5,002	42.3	2,103	18.2
NORTH CENTRAL										
East North Central	142,998	348.7	69,460	169.5	36,940	90.1	17,323	42.3	7,632	18.6
Ohio	37,818	351.5	18,824	175.0	9,340	97.6	4,330	41.3	1,733	16.1
Indiana	17,664	332.6	8,521	161.5	5,401	103.3	2,570	48.4	1,157	21.8
Illinois	43,148	387.2	19,671	176.3	9,484	83.4	4,636	41.5	1,941	17.4
Michigan	23,345	379.8	14,482	158.2	7,289	79.6	3,801	41.6	1,843	20.2
Wisconsin	15,030	340.8	7,630	166.3	4,334	94.8	2,020	43.9	952	20.7
West North Central	60,527	363.1	29,430	177.5	17,519	107.4	8,811	51.6	3,901	23.4
Minnesota	12,411	318.1	6,322	161.8	3,801	99.2	1,823	47.5	799	20.1
Iowa	11,046	384.9	5,300	186.7	3,326	112.4	1,654	55.7	658	24.3
Missouri	18,593	393.4	9,113	191.3	5,236	111.3	2,430	51.0	1,176	25.2
North Dakota	2,109	340.0	922	145.2	579	91.2	410	64.6	176	27.7
South Dakota	2,534	371.0	1,118	163.7	716	104.8	431	63.1	214	31.3
Nebraska	5,510	356.4	2,757	179.0	1,713	110.8	854	55.3	387	25.0
Kansas	8,344	368.1	3,939	176.4	2,381	106.7	1,141	51.2	533	23.5
SOUTH										
South Atlantic	110,316	326.3	55,915	161.2	31,217	92.7	17,760	52.7	8,094	24.0
Delaware	1,953	337.7	935	161.5	348	60.1	238	41.1	128	22.1
Maryland	12,117	323.0	6,922	158.4	2,333	58.4	1,563	39.1	701	17.1
District of Columbia	2,173	313.5	1,583	218.3	554	76.0	312	43.7	94	13.1
Virginia	14,597	293.3	7,483	150.3	3,919	78.9	2,528	50.9	1,037	20.9
West Virginia	7,760	430.4	3,303	183.2	1,831	101.6	1,049	58.2	433	24.1
North Carolina	16,818	304.4	7,944	143.9	5,247	95.3	3,263	59.9	1,129	20.0
South Carolina	9,191	290.7	3,830	135.9	2,726	96.7	1,682	60.0	821	29.1
Georgia	14,401	292.3	6,551	134.7	5,135	104.3	2,930	59.5	1,434	29.5
Florida	31,976	381.3	18,247	213.3	9,471	108.0	4,247	50.8	1,827	21.1
East South Central	45,309	335.2	22,355	165.1	15,717	116.0	8,199	60.5	3,701	27.7
Kentucky	12,761	375.5	5,967	175.7	3,811	113.1	1,984	58.4	934	26.6
Tennessee	14,144	337.7	6,537	163.7	4,835	116.9	2,332	55.7	1,139	27.0
Alabama	10,948	302.9	5,127	141.2	4,180	115.9	2,261	62.6	1,077	29.8
Mississippi	7,856	332.1	3,724	157.9	2,701	119.0	1,588	67.7	646	27.5
West South Central	63,650	306.2	32,723	154.9	19,812	93.6	12,143	58.2	5,767	27.7
Arkansas	7,933	374.9	3,795	179.3	2,725	123.7	1,129	42.8	630	29.4
Louisiana	12,444	323.3	6,141	152.5	3,461	91.3	2,275	60.0	969	25.6
Oklahoma	10,053	373.1	4,654	173.4	3,024	111.3	1,437	53.3	741	29.8
Texas	33,230	271.6	17,934	146.3	10,724	87.6	6,932	56.6	3,417	27.9
WEST										
Mountain	22,773	234.1	12,321	127.7	6,539	68.0	5,910	61.3	2,931	30.4
Montana	2,177	221.0	1,209	145.8	654	81.4	510	77.3	243	32.8
Idaho	2,254	272.4	1,113	133.7	701	86.3	558	68.0	285	34.9
Wyoming	1,026	274.3	486	129.9	317	82.1	313	83.7	164	41.9
Colorado	5,931	231.1	3,073	121.3	1,574	62.1	1,323	52.3	600	23.7
New Mexico	2,015	175.7	1,315	114.7	663	57.8	835	77.0	477	41.6
Arizona	5,515	243.4	3,215	141.6	1,320	60.5	1,237	57.5	673	33.5
Utah	2,443	203.4	1,145	95.2	715	59.9	651	57.2	271	27.7
Nevada	1,379	232.0	874	147.6	401	81.3	380	65.0	185	30.0
Pacific	81,388	291.3	45,332	162.5	24,091	83.3	14,520	51.3	6,143	21.2
Washington	11,012	313.7	5,823	164.5	3,104	89.4	1,793	53.3	833	22.7
Oregon	7,230	323.6	3,944	164.1	2,537	113.1	1,290	53.3	620	24.9
California	61,341	288.7	34,221	162.5	18,119	85.5	12,032	51.6	4,509	19.3
Alaska	243	73.9	213	63.5	74	22.2	410	114.2	120	33.1
Hawaii	1,471	162.0	937	107.2	336	45.8	253	24.2	120	15.0

TABLE 5 (1)

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TABLE 1

Cows milk measurements near Millstone Plant, reproduced from Environmental Statement for 1976.

TABLE 7
DAIRY MILK
(PCI/L)

LOCATION	COLLECTION DATE	SR-89	(+/-)	SR-90	(+/-)	I-131	(+/-)	CS-137	(+/-)
20	2/23/76	0.0	0.7	14.5	0.5	0.01	0.06	22.7	1.1
20	4/12/76	1.6	1.2	7.7	0.4	0.0	0.08	12.4	0.4
20	5/10/76	0.3	1.1	7.9	0.3	0.30	0.09	13.5	0.9
20	6/ 7/76	0.2	1.5	16.2	0.6	0.0	0.08	26.3	0.8
20	7/19/76	0.0	1.7	27.1	0.8	0.05	0.05	32.0	2.0
20	8/ 2/76	0.0	2.0	18.0	0.6	0.09	0.10	27.0	2.0
20	9/13/76	1.1	0.7	15.2	0.5	0.0	0.08	24.2	1.1
20	10/ 5/76	37.0	2.0	14.2	0.7	310.00	6.00	19.5	1.5
21	2/23/76	0.0	0.7	10.2	0.4	0.05	0.08	28.0	2.0
21	4/12/76	1.5	1.2	6.9	0.4	0.0	0.07	17.3	0.8
21	5/10/76	0.4	1.0	7.4	0.3	0.0	0.08	10.3	0.8
21	6/ 7/76	0.7	0.6	9.3	0.4	0.10	0.06	15.8	1.1
21	7/19/76	0.9	0.7	8.0	0.4	0.0	0.06	15.7	0.5
21	8/ 2/76	0.0	1.1	13.1	0.5	0.13	0.09	30.0	2.0
21	9/13/76	0.3	0.9	12.9	0.9	0.0	0.07	31.3	1.3
21	10/ 5/76	44.0	2.0	11.3	0.6	415.00	3.00	15.8	1.4
22	2/23/76	0.0	0.5	5.6	0.3	0.07	0.08	16.7	1.0
22	4/12/76	0.0	0.6	6.3	0.3	0.0	0.05	13.5	0.5
22	5/10/76	1.6	1.3	7.7	0.4	0.0	0.08	16.0	1.0
22	6/ 7/76	0.3	0.9	11.4	0.5	0.0	0.06	21.3	1.2
22	7/19/76	0.0	1.1	19.7	0.6	0.0	0.05	33.3	1.7
22	8/ 2/76	0.0	2.0	16.3	0.6	0.0	0.07	36.0	3.0
22	9/13/76	1.7	0.9	10.8	0.5	0.20	0.09	11.4	1.8
22	10/ 5/75	37.2	1.5	13.6	0.6	217.00	5.00	26.0	2.0
23A	2/23/76	0.0	0.4	4.5	0.3	0.05	0.08	11.8	1.1
23A	4/12/76	1.1	0.9	4.0	0.3	0.0	0.03	8.3	0.4
23A	5/10/76	0.5	0.9	5.2	0.3	0.0	0.07	13.6	1.0
23A	6/ 7/76	0.0	0.6	4.2	0.3	0.40	0.08	4.8	0.8
23A	7/19/76	0.0	0.8	7.4	0.5	0.02	0.06	5.5	0.4
23A	8/ 2/76	1.0	2.0	6.8	1.1	0.04	0.07	10.0	0.8
23A	9/13/76	0.4	0.6	7.0	0.4	0.0	0.08	13.7	1.3
23A	10/ 5/76	4.3	1.3	7.2	0.8	37.90	1.30	6.2	0.7

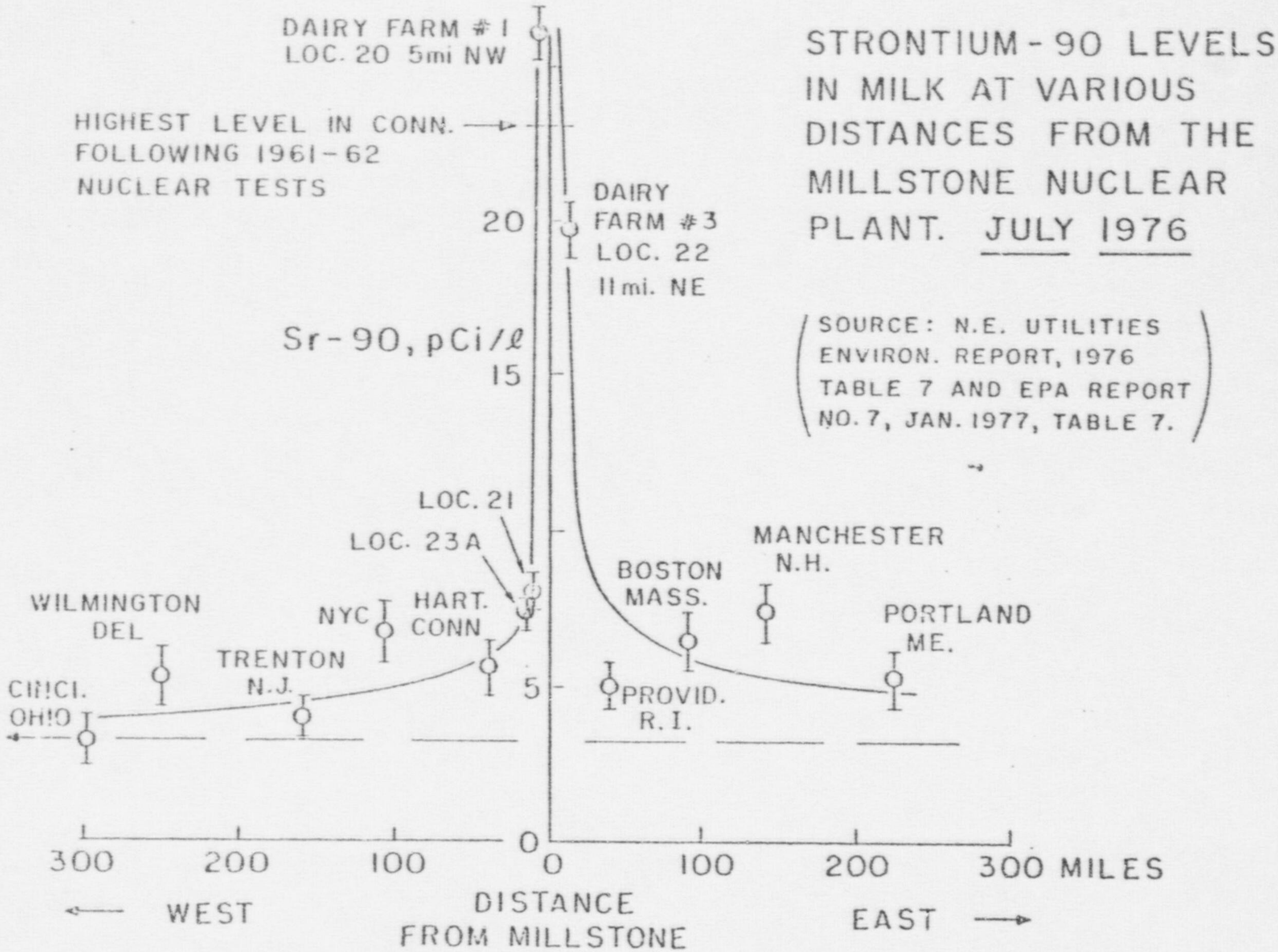

FIGURE 2(a)

APPENDIX I

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APPENDIX I

TABLE 9

Comparison of Sr-90 Concentrations in Milk Near the Millstone Nuclear Reactor With Concentrations in Hartford and the U.S. as a Whole - 1970 to 1976

Year	(a) Av. Daily Milk Sr-90 Concentr. Near Millstone pCi/l	(b) Av. Daily Milk Sr-90 Concentr. In July (Hartford) pCi/l	(c) Av. Daily Milk Sr-90 Concentr. for Year In U.S. pCi/l	Excess Sr-90 In Milk Near Millstone over U.S. pCi/l	%Ex[illegible] Sr-[illegible] Near Mil[illegible] pCi
1970#	9.8	8	8	1.8	—
1971	8.8	9	7	1.8	20[illegible]
1972	9.6	7	6	3.6	3[illegible]
1973	15.0	4	5	10.0	6[illegible]
1974	14.8	Not Avail.	4	10.8	7[illegible]
1975	10.7	3.1	3	7.7	72
1976	13.0	5.7	4	9.0	6[illegible]

\# Millstone Operation began in October 26, 1970
Conn. Yankee (Haddam Neck, 20 miles N.W., Started July 24, 1967.

(a) Three locations within 10-15 miles; $1\sigma = \pm 0.2$ pCi/l
From Millstone environmental reports, annual averages

(b) E.P.A. Measurements (Rad. Data and Reports) in July.
$1\sigma = \pm 1$ pCi/l

(c) E.P.A. Network Average (Rad. Data and Reports)
$1\sigma = \pm 0.2$ pCi/l

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APPENDIX I

TABLE 10

Doses to the Bone of Children Due to Sr-90 in the Milk and Total Diet Near the Millstone Nuclear Plant - 1970 to 1976

Year	Total Diet Sr-90 Intake - Near Millstone pCi/day	(a) Annual Sr-90 Bone Dose For Child-All Sources mrem/yr	(b) Annual Sr-90 Bone Dose For Child Due to Millstone mrem/yr.	Cumul. Sr-90 Bone Dose For Child Due To Millstone mrem	(c) Annual Sr- Bone Dose Child Due Millstone % of Nat
1970#	29.4	185	——	——	——
1971	26.4	266	33	33	47%
1972	28.8	181	69	102	99%
1973	45.0	283	190	292	271%
1974	44.4	279	204	495	291%
1975	32.1	202	145	640	207%
1976	39.0	245	169	809	241%

\# Millstone Operations Began October 26, 1970

(a) Using dose factor of 0.0172 mrem/pCi annual intake from Table A-5, NUREG 1.109 (N.R.C., March 1976), equivulant to 6.28 mrem/yr. per 1 pCi daily intake in total diet.

(b) Using percent excess Sr-90 levels due to Millstone from milk measurements (Table 9).

(c) Natural Radiation background 70 mrem/yr. ((E.P.A. measurements; E.P.A. report on Haddam Neck E.P.A. - 520/3-74-007 ; Sect. 7.7 , page 109

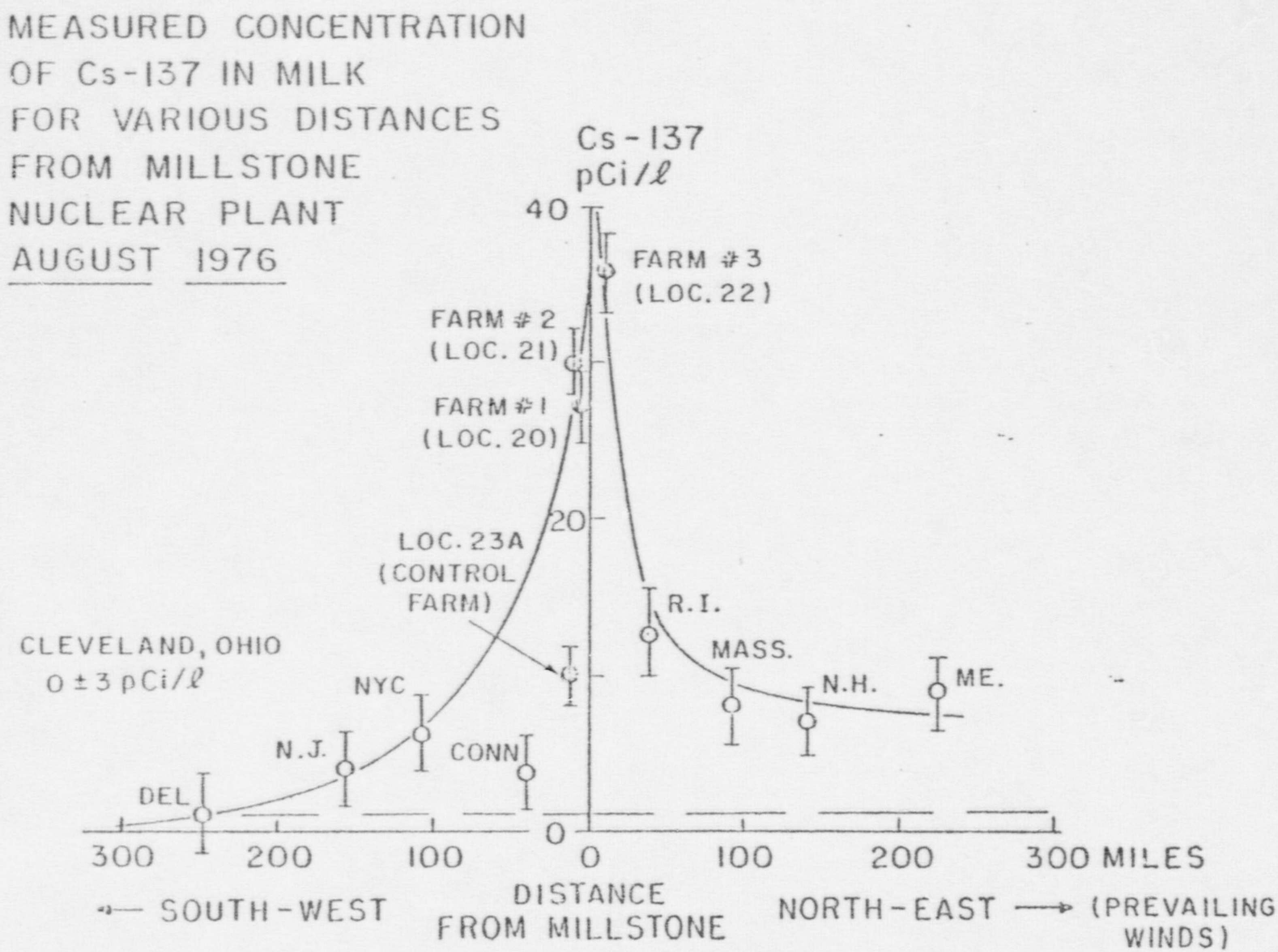
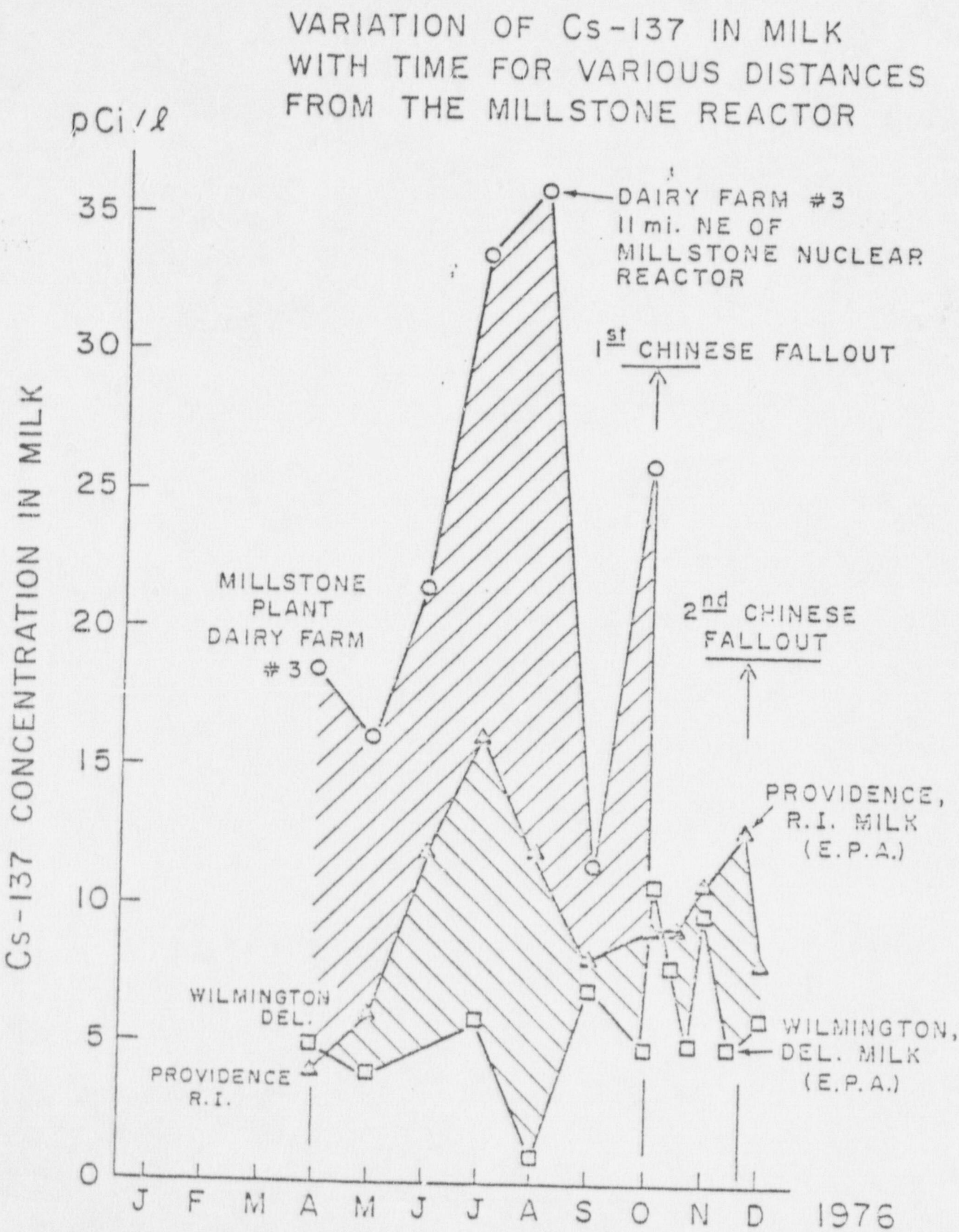


FIGURE 2(b)

17-3-31

APPENDIX I



A-332 FIGURE 4

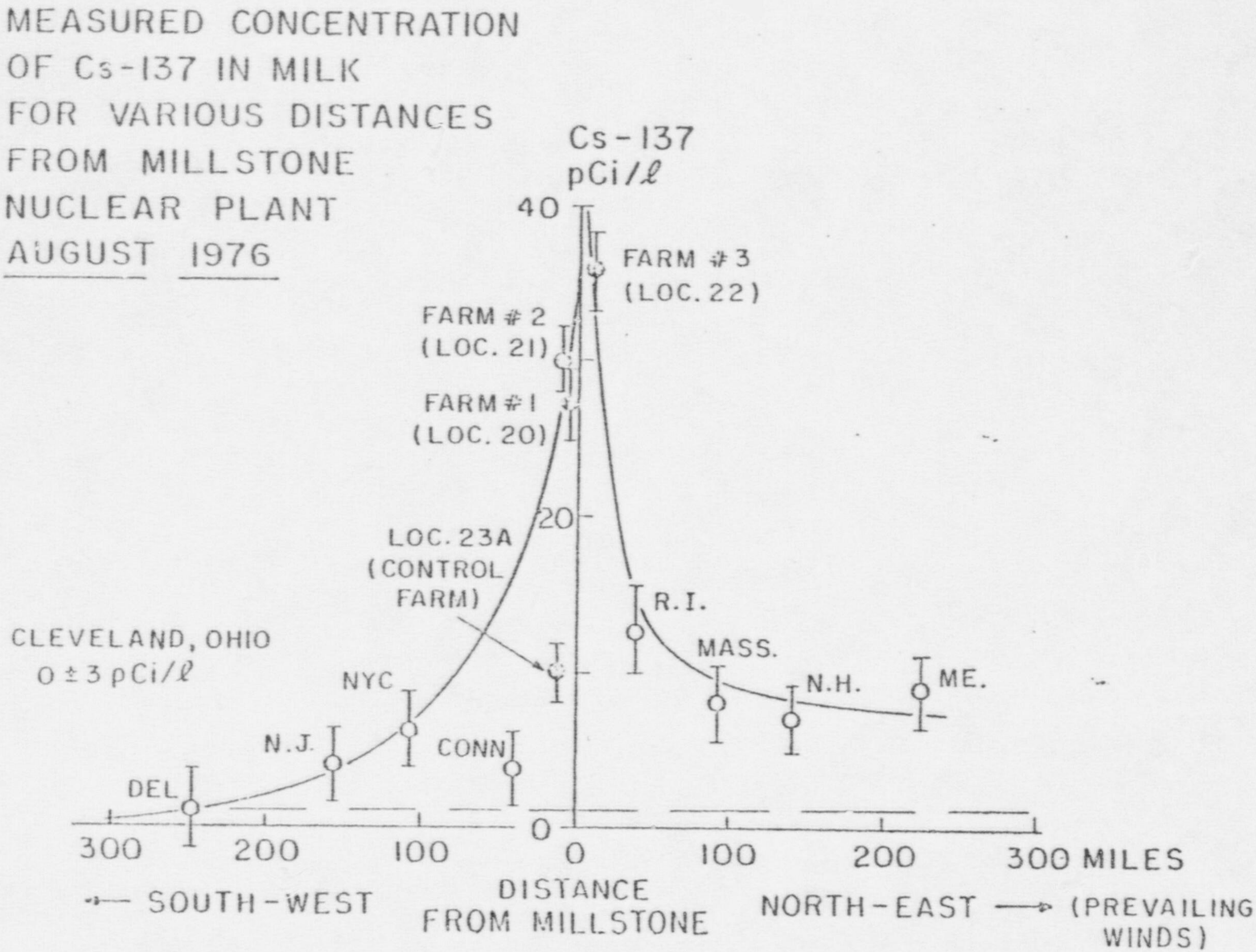


FIGURE 2(b)

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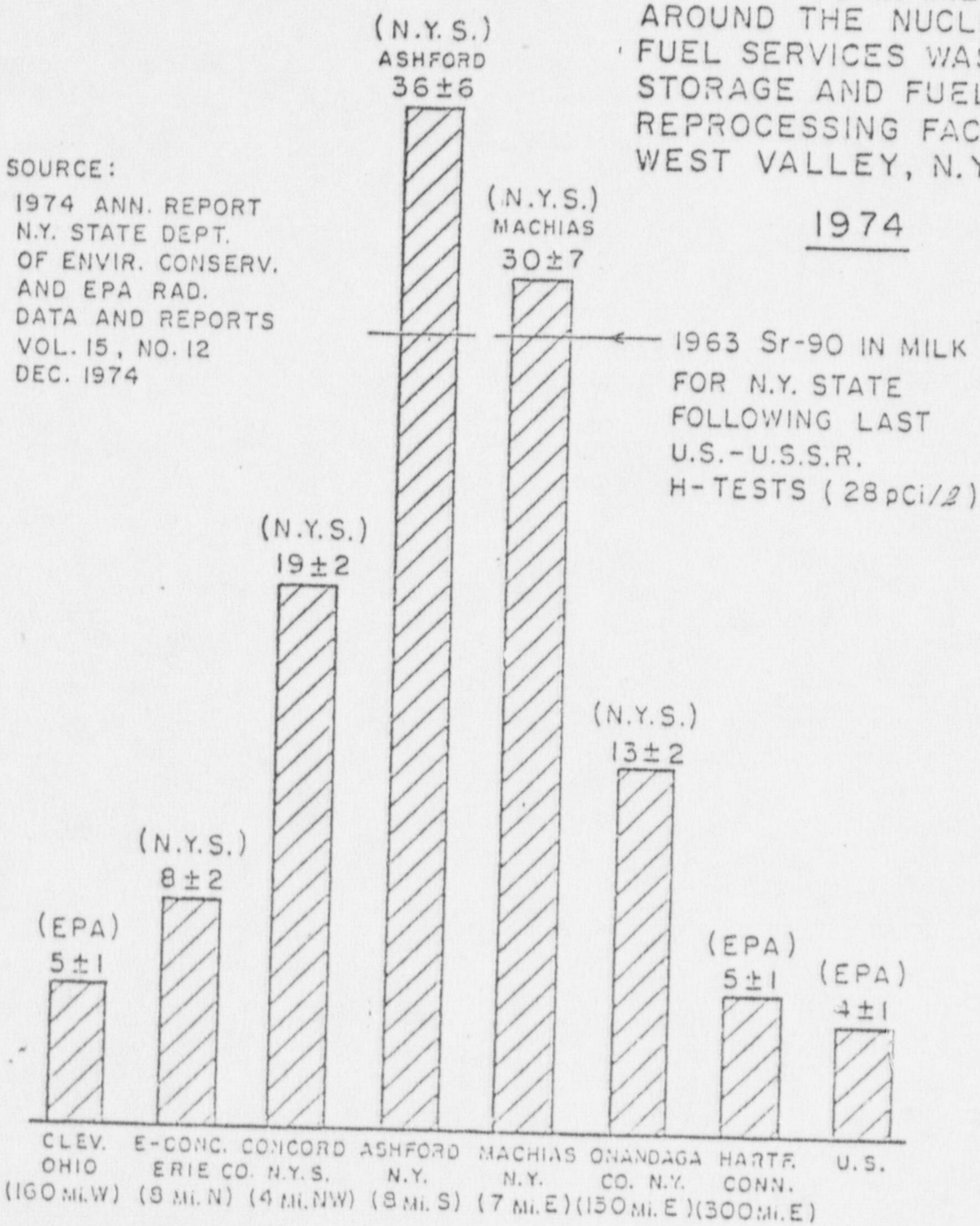
APPENDIX II

PEAK Sr-90 IN MILK AROUND THE NUCLEAR FUEL SERVICES WASTE STORAGE AND FUEL REPROCESSING FACILITY, WEST VALLEY, N.Y.

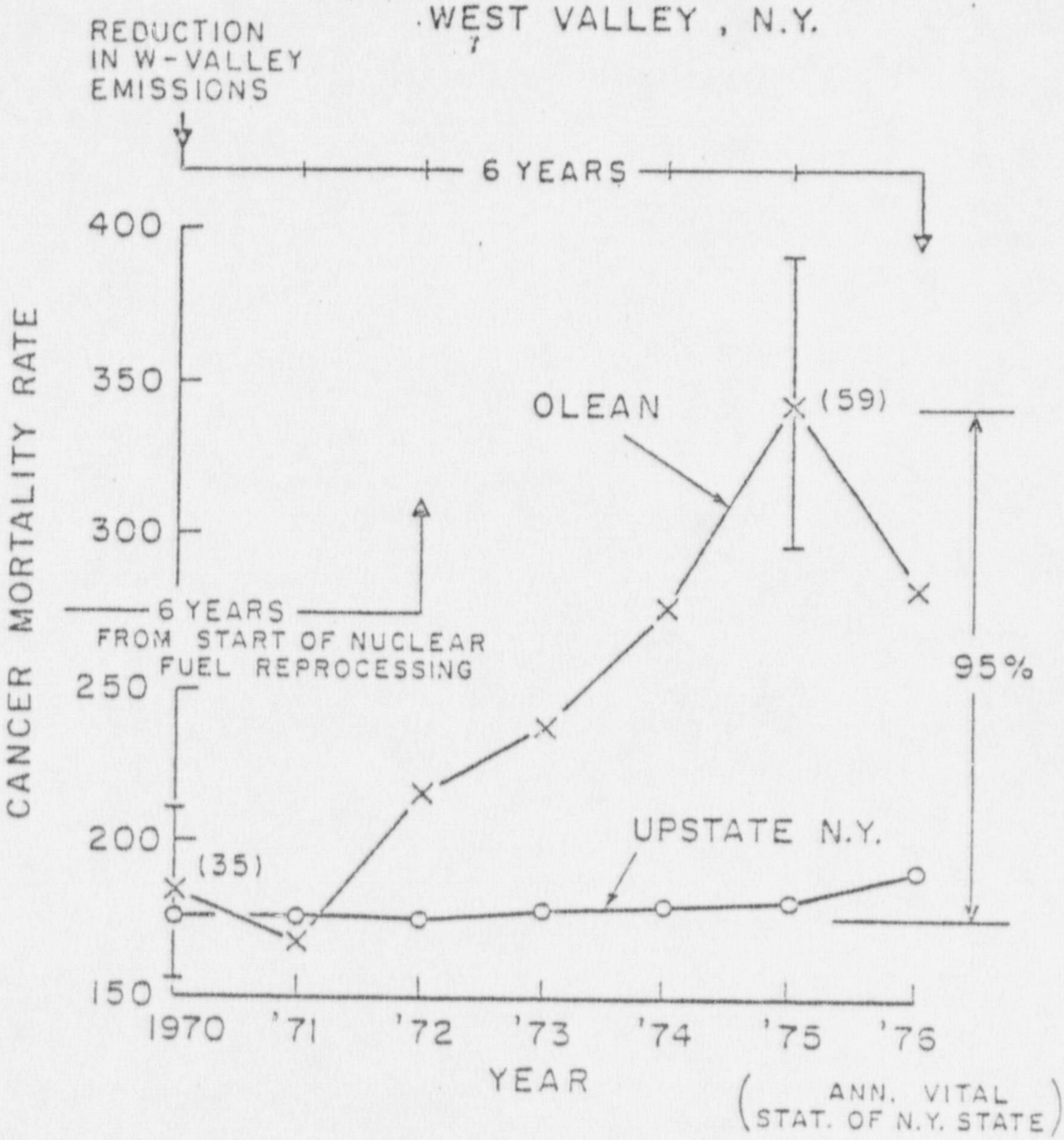
1974

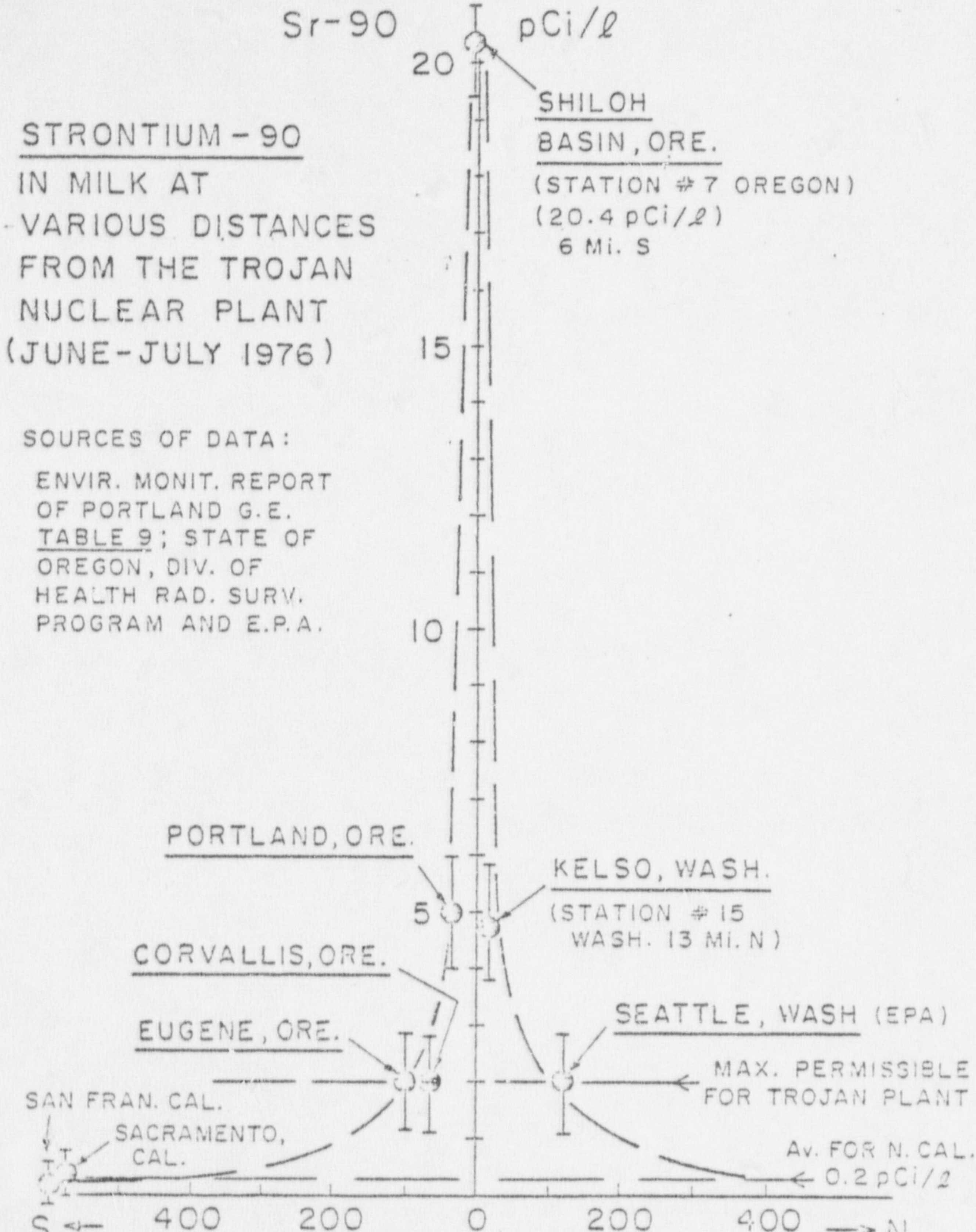
SOURCE:
1974 ANN. REPORT
N.Y. STATE DEPT.
OF ENVIR. CONSERV.
AND EPA RAD.
DATA AND REPORTS
VOL. 15, NO. 12
DEC. 1974



CANCER MORTALITY

FOLLOWING START OF NUCLEAR
FUEL REPROCESSING PLANT
WEST VALLEY, N.Y.




Sr-90 pCi/ℓ
STRONTIUM-90
IN MILK AT
VARIOUS DISTANCES
FROM THE TROJAN
NUCLEAR PLANT
(JUNE-JULY 1976)
SOURCES OF DATA:
ENVIR. MONIT. REPORT
OF PORTLAND G.E.
TABLE 9; STATE OF
OREGON, DIV. OF
HEALTH RAD. SURV.
PROGRAM AND E.P.A.
SHILOH
BASIN, ORE.
(STATION #7 OREGON)
(20.4 pCi/ℓ)
6 Mi. S
PORTLAND, ORE.
KELSO, WASH.
(STATION #15
WASH. 13 Mi. N)
CORVALLIS, ORE.
EUGENE, ORE.
SEATTLE, WASH (EPA)
MAX. PERMISSIBLE
FOR TROJAN PLANT
SAN FRAN. CAL.
SACRAMENTO,
CAL.
Av. FOR N. CAL.
0.2 pCi/ℓ
20
15
10
5
S 400 200 0 200 400 N
DISTANCE FROM TROJAN REACTOR IN MILES

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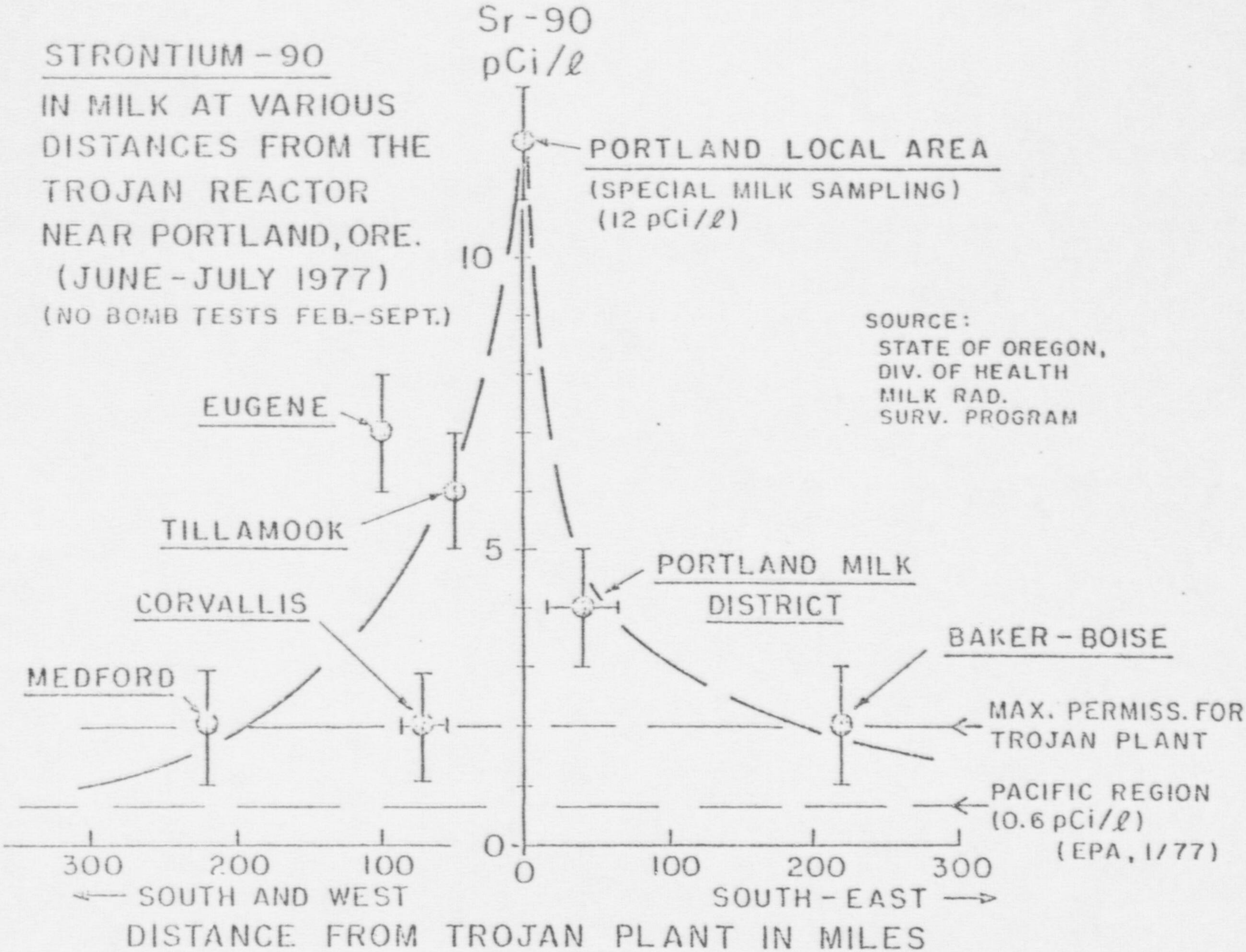

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APPENDIX II

STRONTIUM - 90 LEVELS

IN THE MILK AND DIET NEAR

CONNECTICUT NUCLEAR POWER PLANTS

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For many years, both the general public and the scientific community have believed the assurances of the nuclear industry and government agencies that under normal operating conditions, the radiation doses due to releases from commercial nuclear power plants are negligibly small, and that therefore nuclear power represents much less of a threat to human health than the operation of oil or coal burning electric power stations. (1)

In particular, the nuclear industry , while admitting the atmospheric releases of iodine - 131,has repeatedly claimed that no significant amounts of strontium - 90 and cesium - 137 are released into the air from nuclear power stations, and that therefore the strontium - 90 and cesium - 137 measured in the local milk must be due to fallout from nuclear weapons tests. (2)

However , a detailed examinations of the levels of radioactive strontium and cesium in the air, the soil, the vegetation and the milk around two large nuclear power stations in Connecticut as measured by the utility's own environmental consultants over a period of many years reveals that this claim was valid only for the first few years of operation , and that in the last few years, the levels of these known cancer producing substances have reached or exceeded the levels observed in Connecticut during the height of nuclear weapons testing in the early 1960's. (3)

For instance whereas the highest yearly average following the

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large nuclear tests of 1961 - 1962 was 23 picocuries Sr-90 per liter of cow milk (pCi/l)[1],[(4)] the dairies within a radius of eleven miles around the Millstone Point Reactor near New London showed milk concentrations as high as 27 pCi/l in July of 1976.[(5)] At the same time, the levels measured by the U.S. Environmental Protection Agency (EPA) in Hartford some 40 miles northwest showed only 5.7 pCi/l of Sr-90 in the milk.[(6)] See Fig. 1 and Table 1)

That these high levels of Sr-90 in the local milk could not possibly be explained by fallout from nuclear weapons tests is further supported by the fact that the levels measured in the milk by the EPA for the same period decline in all directions away from Connecticut, reaching values as low as 3 pCi/l in Cincinnati, Ohio and 5 pCi/l in Portland, Maine.[(7)] A similar pattern was found for cesium - 137 in the milk, with peak values near the Millstone plant of 36 pCi/l compared with only 5 pCi/l in Hartford.[(8)] (See Fig. 2 and Table 2)

That fallout from nuclear testing cannot explain the very high levels near the nuclear plants in south-eastern Connecticut is further supported by the fact that both cows milk and goat milk in the nearby farms rose sharply from their values after April and May when the plants were shut down for refueling to peak levels in July and August of 1976, with Sr-90 in goat milk reaching 32 pCi/l near the Connecticut Yankee plant, and 61 pCi/l near the Millstone Nuclear Power Station.[(9)] Yet, the fallout from the Chinese Bomb Test of September 26, 1976 was not detected in Connecticut until October 5th, and when it arrived, the levels in goat milk, as measured by the

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1. One Curie is the radioactivity of one gram of radium. One pCi is one millionth of one millionth of this amount, or 10^{-12} Ci.

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utility company rose by less than 3 pCi/l.[10] (See Fig. 3 and Table 3)

The fact that the month-by-month changes in cesium - 137 in the milk measured by the E.P.A.[11] for the Providence, Rhode Island milk showed a similar but lower peak in July and August for the sampling stations near the Millstone Nuclear Plant in 1976 before the Chinese fallout arrived in October while the rest of the U.S. did not, clearly indicate that the contamination of the local milk can affect distant population centers as far as 50 or more miles away, depending on where the milk is shipped. (Fig. 4 and Table 4)

Since the concentration of strontium - 90 near the two plants was 30 to 60 times the minimum detectable level, there can be no question as to the statistical significance of these large rises in the Sr-90 concentrations around these plants. (See Tables 1 and 3).

That the source of these high levels of radioactivity in the milk is mainly due to the emissions from the Millstone Plant located some 15 miles from the Connecticut - Rhode Island border is further supported by a recent report sent by the Nuclear Regulatory Commission to Congressman C. J. Dodd of Connecticut.[12] In this report it is stated that "During 1975, the liquid and gaseous effluents from Millstone Unit No. 1 were high in radioactivity, in terms of Curie content, as compared to other nuclear power facilities", and that "this was due primarily to defects in the nuclear fuel that was being utilized at that time".

According to the N.R.C.'s report on radioactive releases from Nuclear Facilities issued in March of 1977,[13] the Millstone Plant near New London, Connecticut discharged some 2,970,000 Curies of radioactive gases into the air in 1975, the highest of any commercial nuclear plant ever reported in the United States. That same year the environmental report for the plant also showed an all-time high of 36 pCi/l of cesium - 137 produced in nearby farms, as compared with only 5 pCi/l of CS-137 for milk measured by the EPA in Hartford some 40 miles to the north.[11]

Although the N.R.C. report to Congressman Dodd indicates that the total gaseous emissions were reduced to 500,000 curies in 1976, strontium - 90 in goat milk continued to rise from 37 pCi/l in 1975 to an all time high of 61 pCi/l in the third quarter of 1976, compared to a minimum level of only 5 pCi/l measured in earlier years. Since the accuracy of these measurements is ±1 pCi/l, there can be no question as to reality of these extremely high levels of the most biologically serious of all fission products in the milk and food, a chemical substance which is known to have induced leukemia and cancer in numerous animal studies.

In order to appreciate the seriousness of these levels, it must be realized that the Federal Radiation Council in 1961 set the maximum levels of strontium - 90 for continuous consumption in the total diet at 20 pCi/day(top of Range I)#, and recommended that countermeasures should be taken such as placing cows on stored, uncontaminated feed, removing the milk from the market, or removing the strontium - 90 from the milk by ion - exchange processes when levels exceed 200 pCi/day.[15] As another indication of the seriousness of these levels, it must be noted that the Environmental Protection Agency (EPA) in its newly adopted standards for drinking water which came into effect in June of 1977 requires that the levels of Sr-90 must be less than 8 pCi/l.[16] Since the carcinogenic action of a given intake of Sr-90 is the same whether it occurs in water or milk, levels that are as high as 30 to 60 pCi/l are clearly in violation of presently accepted health standards.

See Table 5. Note that since milk is only 1/3 of the daily dietary intake, the maximum allowable concentration in milk for one l/day should be 7 pCi/l.

For an intake of 200 pCi per day, the Federal Radiation Council (15) and more recently the N.R.C. (17) has calculated a maximum dose to the bone of 1500 to 1800 mrem for individuals in the general population, namely the young infant. This dose may be compared with an annual average dose of about 80 mrem from natural background radiation in the U.S. (18)

Using the latest figures on the dose per unit strontium - 90 intake in a year published by the Nuclear Regulatory Commission in its Regulatory Guide, 1.109, March 1976, (17) it is possible to calculate the maximum dose to any individual resulting from the measured levels of strontium - 90 in the diet for individuals living near the Millstone plant.

For the measured average of 35.1 pCi/l at a goat farm two miles from the plant to the east, (19) and correcting for the background or local minimum due to world-wide fallout of 9.6 pCi/l measured at a more distant goat farm 15 miles away, (20) one obtains an excess due to the releases from the plant of 25 pCi/l for 1976. Since the total intake from all dietary sources as measured by the Connecticut Department of Health is about three times that for cows milk and 2.0 times that ingested with goat milk, one gets a daily intake of about 365 x 2.0 x 25 pCi/l or 18,250 pCi per year. From the N.R.C. Guide Table A-5, (17) one obtains a yearly dose to bone of children of 0.0172 mrem per pCi, giving a dose of 0.0172 x 18,250 or 314 mrem per year, and about half as much to bone marrow (21)

This dose is 450% of the average background radiation of 70 mrem in the area according to the EPA's latest measurements, (22) and some 40 times larger than the maximum dose to any individual of 7.9 mrem calculated by the utility in the summary of its 1976 environmental report, in which no account is taken of the Sr-90 in milk since it is claimed "to be unrelated to plant operations." (23)

These doses are some 50 times larger than bone marrow doses from a typical chest - x-ray (2-4 mrems).[24] Furthermore, these Sr-90 doses are comparable with those known to double the risk of childhood cancers and leukemia for infants following diagnostic x-rays during pregnancy as determined by the large-scale statistical studies of Dr. Alice Stewart at Oxford University[25] and Dr. Brian McMahon at Harvard,[26] which range from about 1200 mrems for exposure of the full-term infant to as low as 80 mrem for a fetus exposed in the first three months of development.[25]

Finally, these large Sr-90 doses from normal operation of nuclear reactors as determined from the detailed measurements of the utility's own monitoring organization must be compared with the recommendation of the National Academy of Sciences in its 1972 report,[27] where it is recommended that the dose from the generation of nuclear power should not exceed a small fraction of the natural radiation background of about 100 mrem/year. In fact, the Academy Report expressed the view that "societal needs can be met with far lower average exposures and lower genetic risk than permitted by the current Radiation Protection Guide."

The dose from the strontium - 90 in the local milk and diet must also be compared with the new (Appendix I) regulations of the N.R.C.,[28] which call for less than 15 mrem per year to any organ as a result of iodine and particulates such as Sr-90 and CS-137 released to the air as given on page 1.109-15 of the Regulatory Guide 1.109. (Table 6)

The average level of 35.1 pCi/l in the goat milk near the Haddam Reactor at two miles, and the level of 17.3 pCi/l Sr-90 in cows milk (five miles north-west, Station 20)[29] inidcates that although the releases from the Boiling Water Reactor (BWR) at Millstone are larger by about four times, the releases of th--e most biologically hazardous of all fission products is by no means negligible for commercial Pressurized Water Reactors (PWR). Comparison with the releases from the much smaller PWR at Rowe, Massachusetts, indicates that it is apparently the effort to operate the new large reactors at higher temperatures and with thinner fuel rod claddings in order to increase their thermal efficiency that has been primarily responsible for the great increase in radioactive releases to the environment.[30] (See Table 8)

These results also indicate that existing environmental statements for nuclear reactors do not reflect the true population doses observed after a few years of actual operation.[31]

For the case of the Millstone plant, the average dose alone from the Strontium - 90 in the milk and other food is some 500 times larger than the value of 0.13 mrem/yr. average dose to the population claimed in the summary of the environmental report.[32] The average bone dose to adults is found to be about 75 mrem from the ingestion of Sr-90 by adults consuming the local milk in 1976, and some five times greater for the total amount of Sr-90 accumulated in the bones of residents in the area since the plant went into operation.

The detailed calculation of the dose is summarized in Tables 9, 10, 11. Table 9 shows the annual average concentration of Sr-90 in the milk around Millstone as measured at three sampling stations located within a radius of about 10 to 15 miles starting in 1970.[1] Also shown in this table are the Sr-90 concentrations for Hartford, Connecticut, located some 35-40 miles to the north-west, and the U.S. average concentrations measured by the E.P.A. in its pasteurized milk network (PMN) for some 50 to 60 locations in the U.S.(33)

It is seen that in general, the Hartford milk concentration is very closely equal to that for the U. S. average between 1970 a 1975, so that the U.S. average, representing world-wide fallout fro nuclear tests, can be used as a control for the concentrations founc near Millstone.

Also shown are the excess average concentrations in the three locations near Millstone over the U.S. average, which are seen to increase from 1.8 pCi/l in 1971 to a maximum of 10.8 pCi/l in 1974. It is seen that since 1973, the excess Sr-90 near Millstone has exceeded that due to world-wide weapons testing by about two times. Thus, for the years 1973 to 1976, about two-thirds of the measured Sr-90 near Millstone must be attributed to releases from the plant, and only about one-third can be attributed to weapons fallout.

Based on these measured milk concentrations, it is now possible to obtain the average annual dose to bone as follows. First, one can obtain the total daily intake of Sr-90 from the total diet by multiplying the amount in one liter of milk consumed per day by three, according to the measurements of the Connecticut Department of Health.(34)

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1. A fourth station (No. 23A) located in a direction away from the two prevailing directions served as "control" after 1972.

This daily intake, when multiplied by 365 then gives the yearly ingestion of Sr-90 which accumulates in bone. One can then utilize the dose factors listed in the N.R.C. Regulatory Guide NUREG - 1.109 Tables A3 to A-6 to obtain the annual dose for a specific organ and a specific age group due to this annual intake of Sr-90.

This has been done in Tables 10-11 for the years following the start-up of the Millstone Reactor in October of 1970, for the case of bone in children and adults.

Inspection of these tables reveals the following important facts:

1. The yearly dose to the bone of children due to the combined amounts of Sr-90 from bomb tests and Millstone releases increased from 166 mrem in 1971 to a high of 283 mrem in 1973, compared with a normal annual background from cosmic rays and other natural sources of only 70 mrem, or some 400% of natural background radiation.

   The cumulative dose to the bone of children living in the area since 1970 from Sr-90 in the milk and food produced near Millstone reached 1,356 mrem by 1976, compared to a dose from natural background of only 420 mrem during the same period.

3. Considering only the portion of the Sr-90 in excess of the fallout levels, the annual bone dose for children reached a peak of 204 mrem in 1974, and was still at 169 mrem in 1976. After six years of operation, the cumulative dose or "dose-commitment" from Millstone had reached 809 mrem to bone.

4. For adults, the bone dose due to all sources of Sr-90 increased from 73 mrem per year in 1971 to 125 mrem per year

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in 1973, decreasing slightly to 108 mrem by 1976.

5. The dose due to the "excess" Sr-90 over that due to world-wide fallout Sr-90, increased from 15 mrem to bone in 1971 to 75 mrem in 1976. This is 577 times larger than the average dose of 0.13 mrem claimed by the utility for that year when Sr-90 is disregarded.

6. The total population exposure in man-rems may be estimated by multiplying the average bone dose by the number of people drinking the milk. Since inspection of Fig. 4 shows that the milk as far away as Providence shows a peak of radioactivity at the same time as the milk near Millstone in 1976 with a peak height of about half that of the nearby milk, one can estimate that the population of Rhode Island receives about half as much radioactivity as the people in New London County where the reactor is located.

Thus, the total population dose can be obtained by taking the population of New London County (240,000) and adding to it half the population of Rhode Island (1/2 of one million, or 500,000), and multiplying the total by the annual Sr-90 bone dose.

The result is shown for each year since 1970 in Table XI The population man-rems to bone due to Sr-90 alone are seen to have increased from 11,100 in 1971 to 55,500 in 1976, while the total cumulative man-rems for the duration of plant operation reached a total of 264,920 man-rem by 1976.

At the valuation of $1000 per man-rem assigned for purposes of cost-benefit studies by the NRC in existing regulations (35), this means that the annual health costs of operating the Millstone plant have reached a level of at least 55.5 million dollars by 1976, with a cumulative cost since start-up of 265 million dollars, ignoring the contribution of Cs-137 and I-131. These must be compared with the values one arrives at when the Sr-90 contribution is left out of consideration, namely 0.13 X 10^{-3} rem X 0.74 X 10^6 or 96 man-rems, with a cost of $96,000. The failure to consider Sr-90 doses therefore underestimates the total man-rems and health costs to society relative to fossil fuel plants by more than 500 times, considering that the actual bone-dose factors for infants, children and adolescents in the total population are significantly greater than for adults.(36)

Not only have the maximum individual and average population doses been grossly underestimated by the failure to take the airborne releases of strontium - 90 into account that enter the milk and food - chain, but also the dose to the whole body or the soft tissue organs has thereby been greatly understated.

This follows from the fact that the N.R.C. Regulatory Guide 1.109 Table A-3 to A-6 gives a whole body dose from the ingestion of Sr-90 which is about one-fourth of the dose to bone. Thus, for the adult, the whole body dose factor in Table A-3 is D.00186 mrem/pCi, compared with 0.00761 mrem/pCi for bone. (See Appendix V)

For the infant, the situation is even more serious since both the bone and whole body doses per unit Sr-90 intake are three times larger than for the adult. Thus, Table A-6 of the F.R.C. Guide gives 0.0251 mrem/pCi for infant bone, and 0.0064 mrem/pCi for soft tissue including the reproductive organs and the glands that control body growth and metabolism.

The result is that the maximum annual whole body dose due to

Sr-90 ingestion alone is one-fourth of the maximum bone dose to a one year old infant of 314 mrem/year for the case of an infant given goat milk, or 78.5 mrem/year. This may be compared to the maximum permissible whole body dose of 5 mrem/year required by the F.R.C.'s Appendix I rules reproduced in Table 6 or the 15 mrem/year to any organ from Sr-90 or other particulates.

Again, the whole body man-rems for the population drinking the milk and eating the food produced in the area are one-quarter of the bone man-rems of 55,000 or 13,750 man-rems in 1976, giving an accumulated whole body dose of 66,230 associated with a health cost of 66.2 million dollars since the Millstone Plant began operation. For a projected 30 year life of the plant, even if the strontium - 90 levels released to the environment were to drop sharply by greatly curtailing the emissions, the already accumulated bone and therefore the whole body concentrations of Sr-90 for adults would diminish only slowly so that at a minimum as a result of excretion and radioactive decay over the remaining 23 years, the annual bone and whole body man-rems would still be one-half to one-third what they were in 1976.

Likewise, the impact on local animal, fish and bird reproduction as well as disease produced by the Sr-90 already accumulated in the soil and sediment will diminish only slowly due to the 28 year half life of Sr-90, thus exacting its environmental toll for decades to come, no matter how successful the efforts might be to lower the additional amounts of Sr-90 discharged annually into the air and rivers of the area.

Conclusions:

1. The evidence on the gradual build-up of Sr-90 in the local cow

and goat milk around both the Haddam Neck and Millstone Point nuclear plants while elsewhere Sr-90 levels were steadily decreasing indicates that the high levels of Sr-90 in the nearby milk cannot be explained by fallout. Furthermore, the evidence for many-fold rises of Sr-90 in the milk within a matter of months following shut-down during 1973-1976 to levels not seen for the U.S. as a whole since the end of large scale nuclear testing in 1963 clearly rules out the possibility that these levels of Sr-90 can be completely explained in terms of fallout from nuclear weapons tests.

2. This conclusion is supported by the existence of a similar pattern in the milk concentrations of Cs-137 which is known to accompany Sr-90 in nearly constant proportions.

3. The evidence of growing accumulations of Sr-90 in the environment around these plants and the need to include it in dose calculations is further supported by the measured pattern of airborne Sr-90 concentrations and soil concentrations that are much larger to the north-east than to the west and north of the Millstone plant as shown in Appendix VII.

4. The detailed environmental measurements carried out by the monitoring organizations employed by the utility show that the Sr-90 concentrations in the milk and diet cannot be left out in calculating either the maximum doses to individuals or the average doses to the population measured in man-rems, as was done by the utility in its reports to the N.R.C. and by the N.R.C. in its report to Congressman C. J. Dodd in September 1977, a copy of which is enclosed as Appendix I. . The reson is that a single strontium unit (pCi) per liter of milk

every day, when the total dietary intake is considered, leads to an annual dose to the bone of a child of 18.8 mrem. This is about 20% of the annual dose from natural background radiation, and exceeds all other sources of radiation in importance. Yet the excess Sr-90 levels were as high as 10 pCi/l in the milk over periods of a year.

5. By either failing to examine the environmental reports in detail, by failing to recognize the crucial importance of strontium - 90 or by not questioning the utility's practice of leaving out the Sr-90 doses in the milk and diet in calculating the doses, the Nuclear Regulatory Commission and the Environmental Protection Agency failed in their primary duty to protect the public health and safety since the population doses and therefore the health effects were under estimated by anywhere from 500 to 2000 times.(37) The resulting health costs to the nation in man-rems per plant, instead of being in the range of 10 to 100 man-rems per year, must now be regarded as in the range of 10,000 to 100,000 man-rems, or in hundreds of millions of dollars for the 5 to 8 years of operation of the two Connecticut plants at the presently accepted health cost of 1,000 dollars per man-rem.

6. Since there is evidence that other nuclear plants have emitted comparable amounts of Sr-90 into the air as Haddam Neck and Millstone,(38) an immediate investigation by the legislatures of the states as well as by Congress is required to end the serious threat to human health that has resulted from the failure of the regulatory agencies of the Federal Government to protect the health and safety of the people living near these plants.

REFERENCES

1. "Health Effects Attributable to Coal and Nuclear Fuel Cycle Alternatives", NUREG-0332 (September 1977 Draft) and references cited therein.

2. See for example the summary of the 1976 environmental report for the Millstone Nuclear Power Station, page 1-1, (Northeast Nuclear Energy Company, Dockets 50-245 and 50-336, March 31, 1977) (Reproduced in Appendix III).

3. See environmental reports for Haddam Neck and Millstone Nuclear Reactors for 1974-1976 for cows milk and goat milk (Northeast Nuclear Energy Company)

4. The average daily levels of Sr-90 in pasteurized fluid milk for Connecticut were 8 pCi/l in 1961, 11 pCi/l in 1962, 23 pCi/l in 1963, and 20 pCi/l in 1964. (Testimony of J. G. Terrill, Jr., U.S. Department H.E.W. page 371 ff, Hearings of the Subcomm. on Research, Development, and Radiation, Joint Comm. on Atomic Energy, 89th Congress, June 29-30, 1965)

5. Reference 3, 1976, Millstone, Table 7. (Reproduced as Table 1 in the present report, and plotted in Figure 1)

6. "Environmental Radiation Data", U.S. E.P.A., Office of Radiation Programs, Monthly Reports prior to January 1975, quarterly reports since January 1975. Report number 7 for July - September 1976 milk levels of Sr-90.

7. Reference 6, Table 12. (Reproduced in Table 2(a))

8. Reference 6, Report number 7, Table 9. (Reproduced as Table 2(b) in present report)

9. Reference 3, 1976 Millstone Plant ,Table 8 (Reproduced as Table 3 in present report)

10. "Radiological Environmental Report for the Millstone Point Site", 1976 (Northeast Nuclear Energy Company) (March 31, 1977) page C-1 ff.

11. Reference 6, quarterly reports numbers 5, 6, 7, and 8 for 1976 Cs-137 in milk. (Reproduced in Figure 4 and Table 4 of the present report).

12. "Evaluation of Radioactive Effluents from Millstone Unit No. 1", Office of Nuclear Reactor Regulation. (September 1977) N.R.C. (Copy inclosed in Appendix).

13. "Radioactive Releases from Nuclear Reactors", N.R.C. Report NUREG-0218 (March 1977)

14. Reference 6, 1975, Report Number 3 for July - September 1975.

15. Federal Padiation Council Reports numbers 1 and 2, May 13, 1960 and September 1961 (U.S. Government Printing Office, Washington, D.C.)

16. U.S. E.P.A. Drinking Water Standards adopted June 24, 1977. (See E.P.A. - 902/4-77-009, Region II, 1975-76 ERAMS Summary Data Report, page 4) Note that the level of 8 pCi/l Sr-90 does not reflect any decrease in allowable levels since the F.R.C. Report Number 2 (1961), representing an infant bone dose of 7.5 mrem from water intake alone or almost 100% of the normal background radiation level, rather than a few percent as recommended by the N.H. Academy of Sciences (Reference 27).

17. "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR50, Appendix I." Regulatory Guide 1.109, N.R.C., March 1976.

18. "Natural Radiation Exposure in the United States", D. T. Oakley, U.S. E.P.A. (ORP./SID 72-1) (June 1972) page 39 gives 84 mrem/year average for the U.S. and 65 mrem/year for coastal plain areas such as Connecticut.

19. Reference 3 1976, Table 8, Farm Number 24 (Reproduced in Table 3).

20. Reference 3, 1976, Table 8, Farm Number 25A (Reproduced in Table 3).

21. Although the 1961 F.R.C. Report Number 2 estimated that the bone marrow dose is about one-third of the bone dose (See Ref. (15) and Table (5)), the 1969 U.N. Scientific Committee Report on the Effects of Atomic Radiation, Table XIX, page 57 gives a revised estimate of 50% (64 mrem vs. 130 mrem bone dose due to Sr-90 for all Nuclear tests prior to 1968).

22. "Radiological Surveillance Study at the Haddam Neck P.W.R. Nuclear Power Station", E.P.A. - 520/3-74-007 (December 1974) Section 7.7, page 109 gives an average of 8.1 μr/hr. offsite locations, or 70 mrem per year in 1970-71. This is in sharp contrast with the claim of the utility company in Reference 2 that the normal background radiation in south-eastern Connecticut was 129 mrem per year in 1976, and indicating a significant rise in background from Cs-137 releases. (See Appendix VI)

23. Reference 2, Paragraph 3 reads as follows: "The observed results indicate that the predominant radioactivity at offsite locations are from nonplant related sources such as fallout from nuclear weapons tests and from naturally occurring nuclides." The N.R.C. report to Congressman C. J. Dodd (Ref. 12) gives an even lower average dose of 0.04 mrem in 1976 for Millstone operations.

24. "A System for Estimation of Mean Active Bone Marrow Dose" R. E. Ellis et al., U.S. Department of H.E.W. (September 1975), Table 4, page 16. (DHEW - FDA - 76 - 8015) (Reproduced as Appendix IV).

25. A. Stewart and G. W. Kneale, Lancet 1, 1185 (1970)

26. B. MacMahon, J. National Cancer, volume 28, 1173 (1962).

27. "The Effects on Populations of Exposure to Low Levels of Ionizing Radiation", National Academy of Sciences (November 1972) Summary and Recommendations, page 1-3.

28. Reference 17, page 1.109-15 (See Table 6 of present report)

29. Reference 3, 1976 Haddam Neck environmental report, Table 8 (Reproduced as Table 7(a) in the present report)

30. "Report on Releases of Radioactivity in Effluents and Solid Waste from Nuclear Power Plants for 1972" Division of Regulatory Operations, U.S. AEC, August 1973. Table 4A gives airborne halogens and particulates for the Yankee (Rowe) Reactor in 1972 as 0.00077 Ci vs. 0.01810Ci for Connecticut Yankee (Haddam) and 1.32000 Ci for Millstone.

31. The Existing Environmental Impact Statements prepared for both Nuclear reactors and Nuclear fuel reprocessing plants under the requirements of the N.E.P.A. act have all been based on the assumption that no significant amounts of Sr-90 escape via atmospheric releases, and not on the actual environmental measurements for large reactors operating more than 3 to 4 years such as Haddam Neck or Millstone. As a result, the calculated health impacts have been seriously underestimated in all such statements.

32. See reference 2, and the N.R.C. dose estimates for Millstone (Ref. 12) which give equally low average doses when Sr-90 in the milk and diet are ignored.

33. Reference 6 for years 1970-1976.

34. Radiation data and reports; E.P.A.; A series of reports, for instance in November 1967, page 646-647 gives 33 pCi/day of Sr-90 in the total diet in June 1967 while Hartford milk was 11 pCi/l in July 1967.

35. Appendix I to 10CFR50, issued in final form April 30, 1975 (Fed. Register May 5, 1975)

36. NUREG-1.109 (Ref. 17), Tables A-3 to A-6. (Population aver. is 1.4 X adult dose)

37. Using the 1976 Millstone environmental report average dose to population of 0.13 mrem per year (Ref. 2) and the N.R.C. report to

Congressman C. J. Dodd (Ref. 12) which gives 0.04 mrem per year,as compared with the annual adult bone-dose due to Sr-90 alone for 1976 of 75 mrem/year.

38. For the year 1974, the annual report of Environmental Radiation for New York State gives maximum Sr-90 concentrations in the milk near a series of nuclear reactors that can be compared with the U.S. average for 12 months ending July 1974 of 5 pCi/l, typical for the eastern U.S. and at sites far from operating nuclear facilities such as East Hampton, N.Y. (5±2 pCi/l).

1. Indian Point Reactor - (Westchester County ,Bedford: 18±3 pCi/l maximum and 14 pCi/l average).
2. 9 - Mile Point Reactor - Oswego County Scriba: 17±2 pCi/l maximum and 12 pCi/l average).
3. Brookhaven National Laboratory (Gas Cooled Testing Reactor) Suffolk County (Site M515101) .20±4 pCi/l maximum and 1[illegible] pCi/l average).

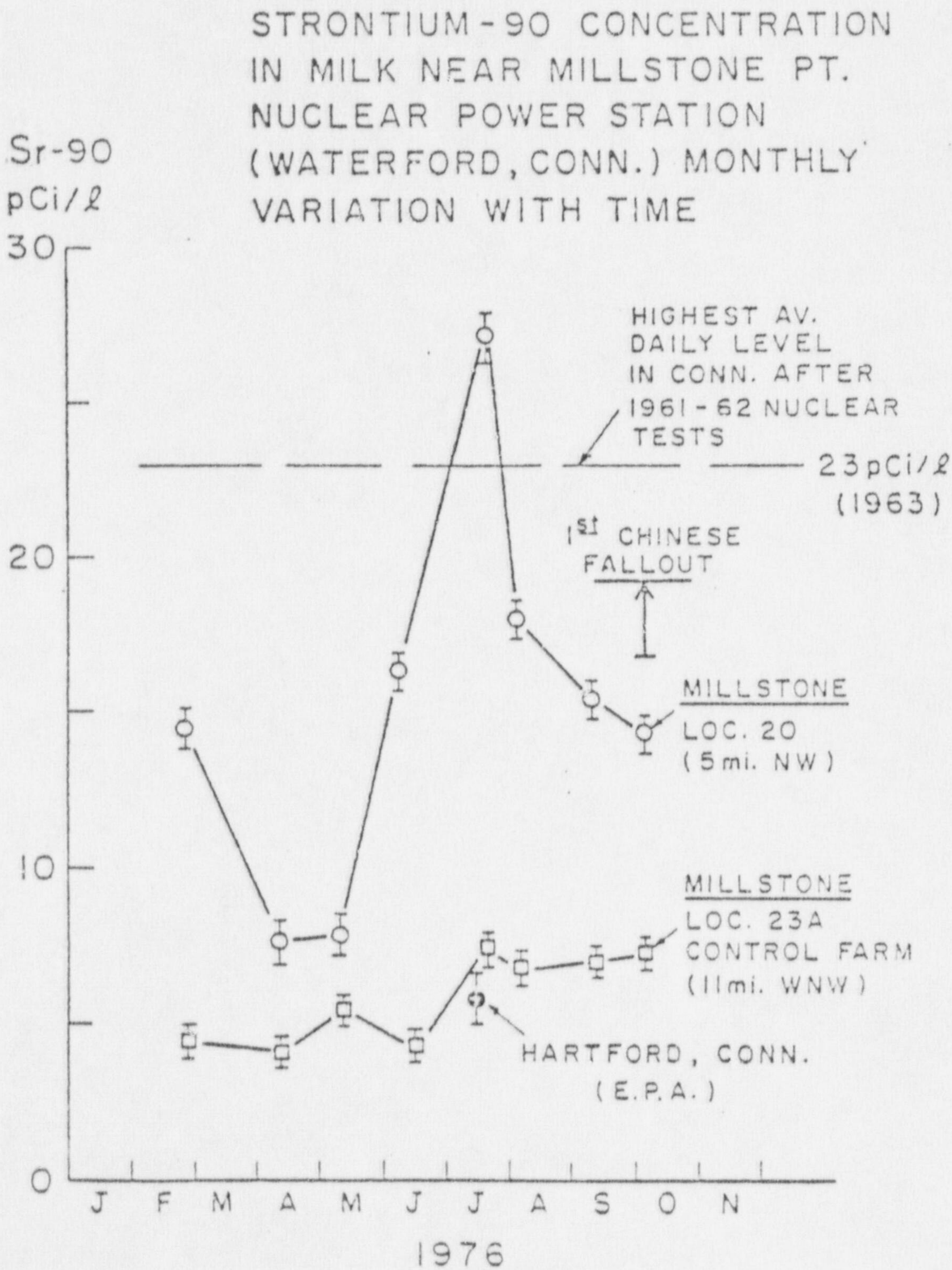

FIGURE 1

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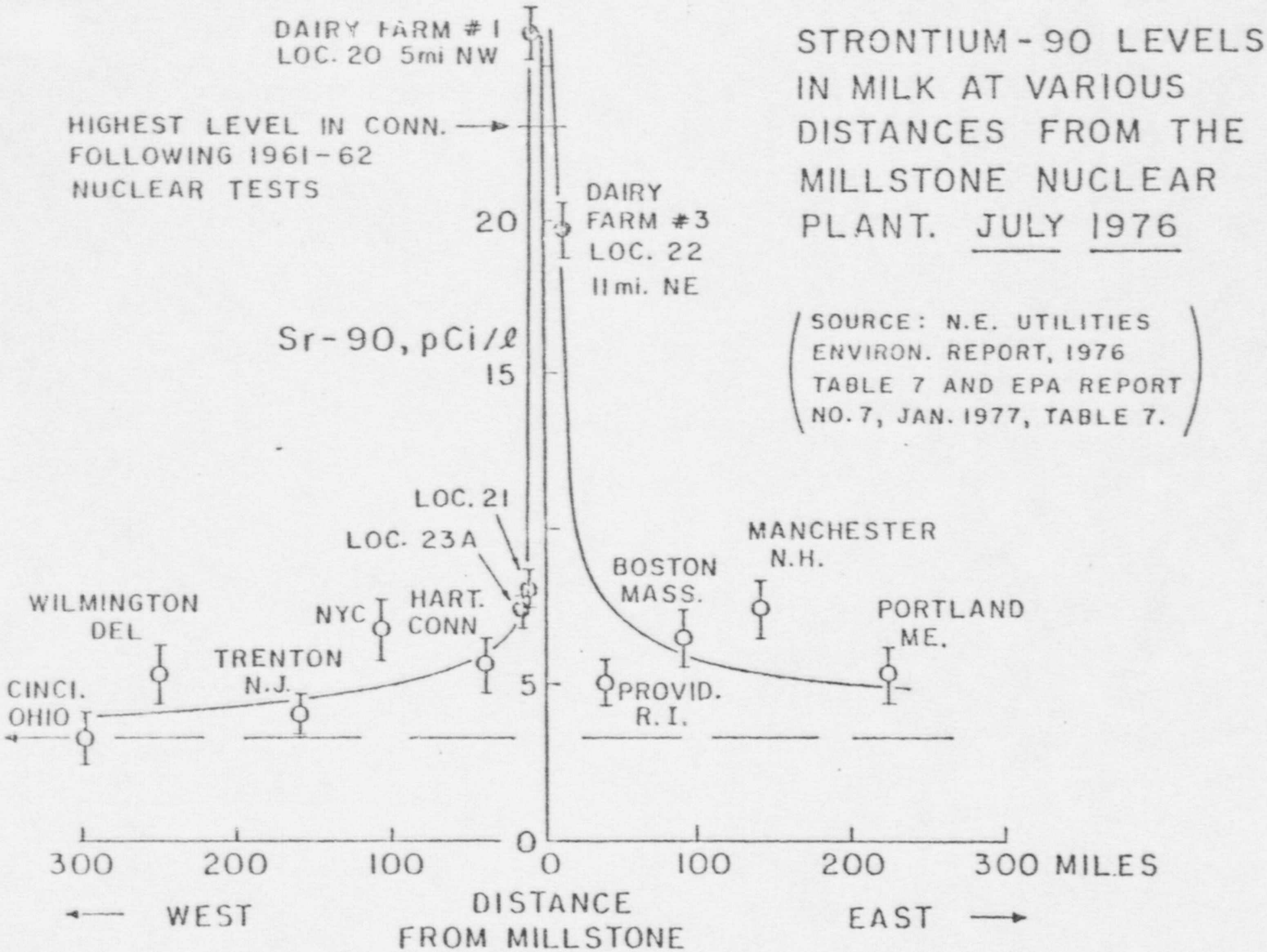

FIGURE 2(a)

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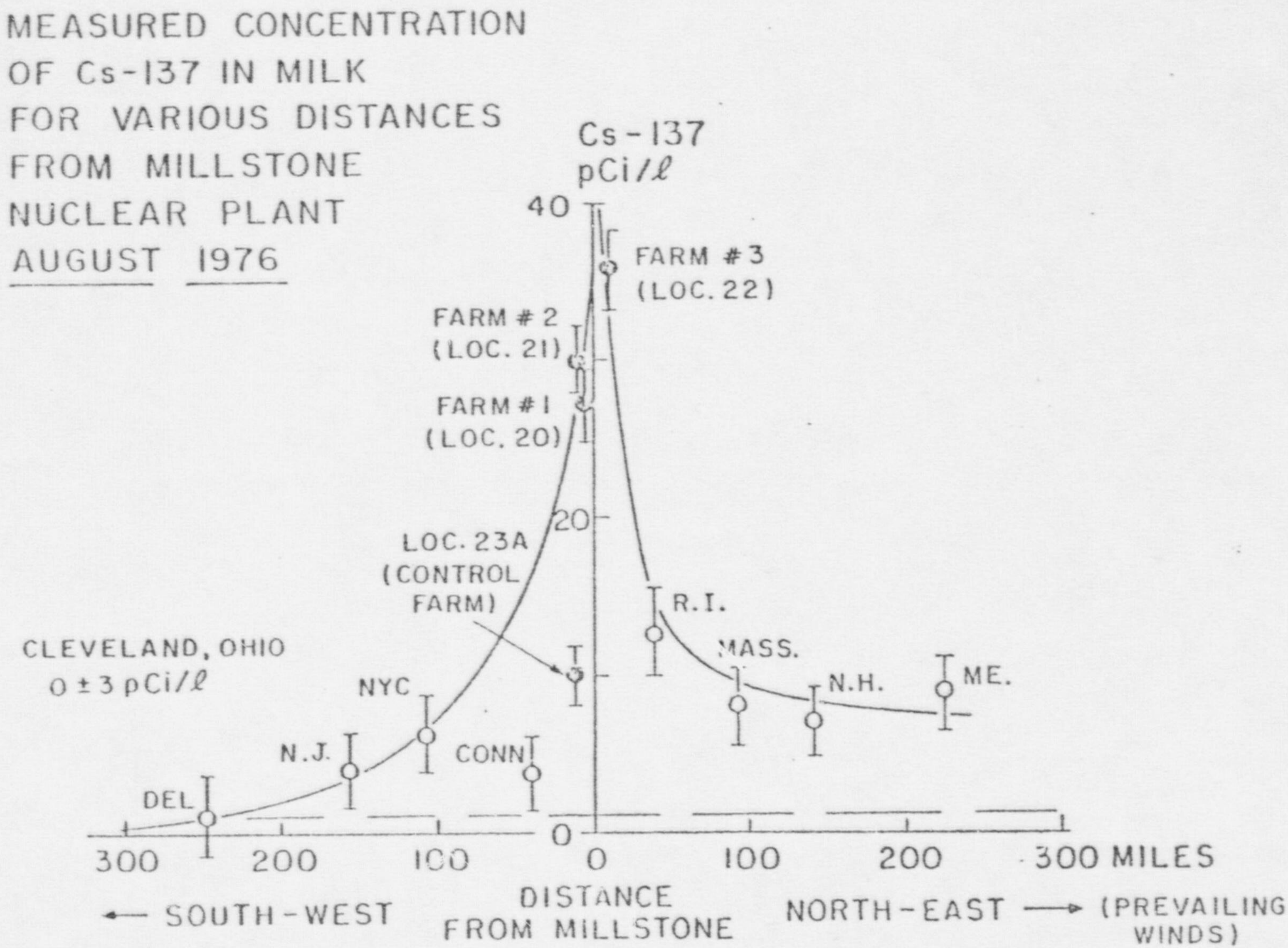

FIGURE 2(b)

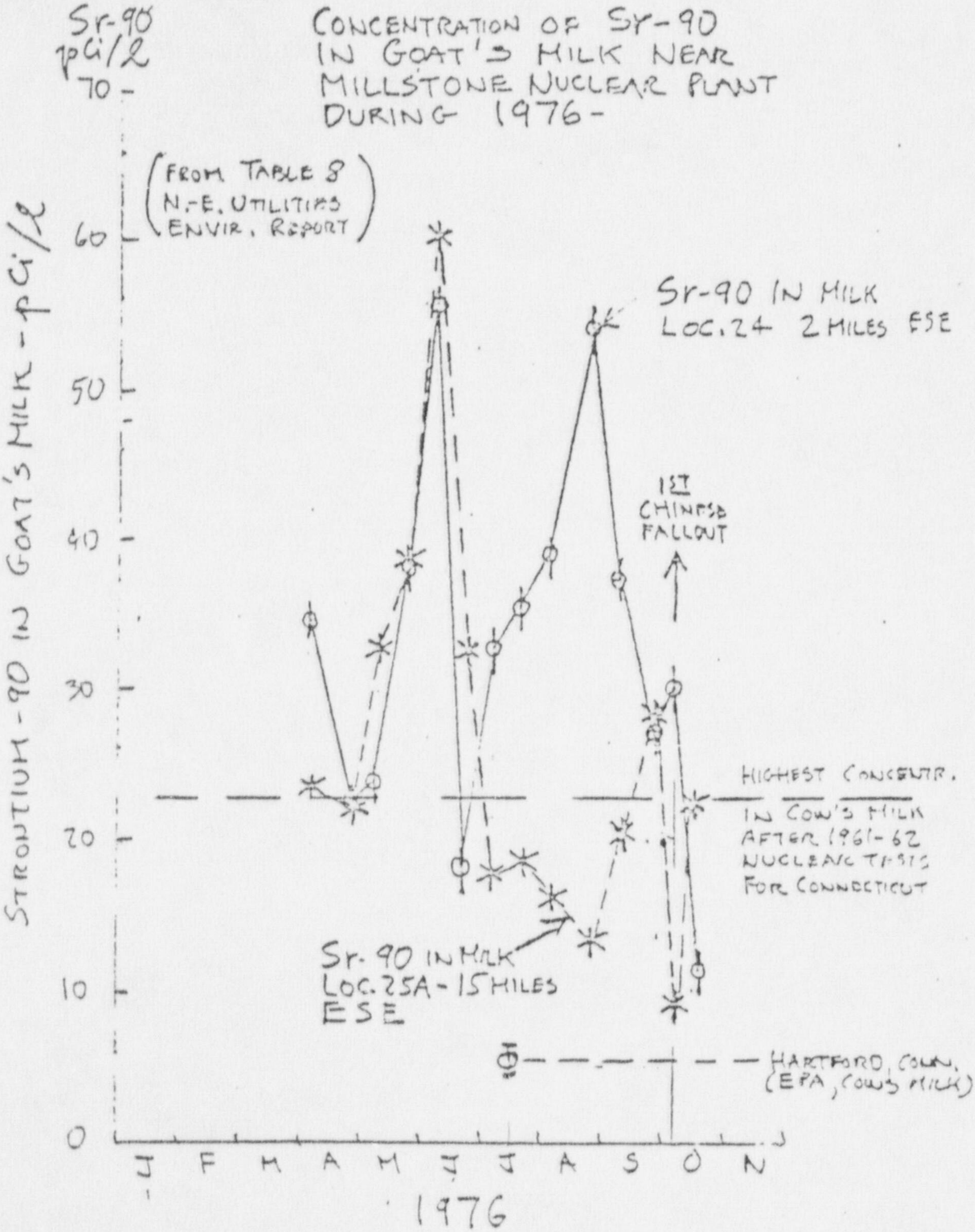

FIGURE 3

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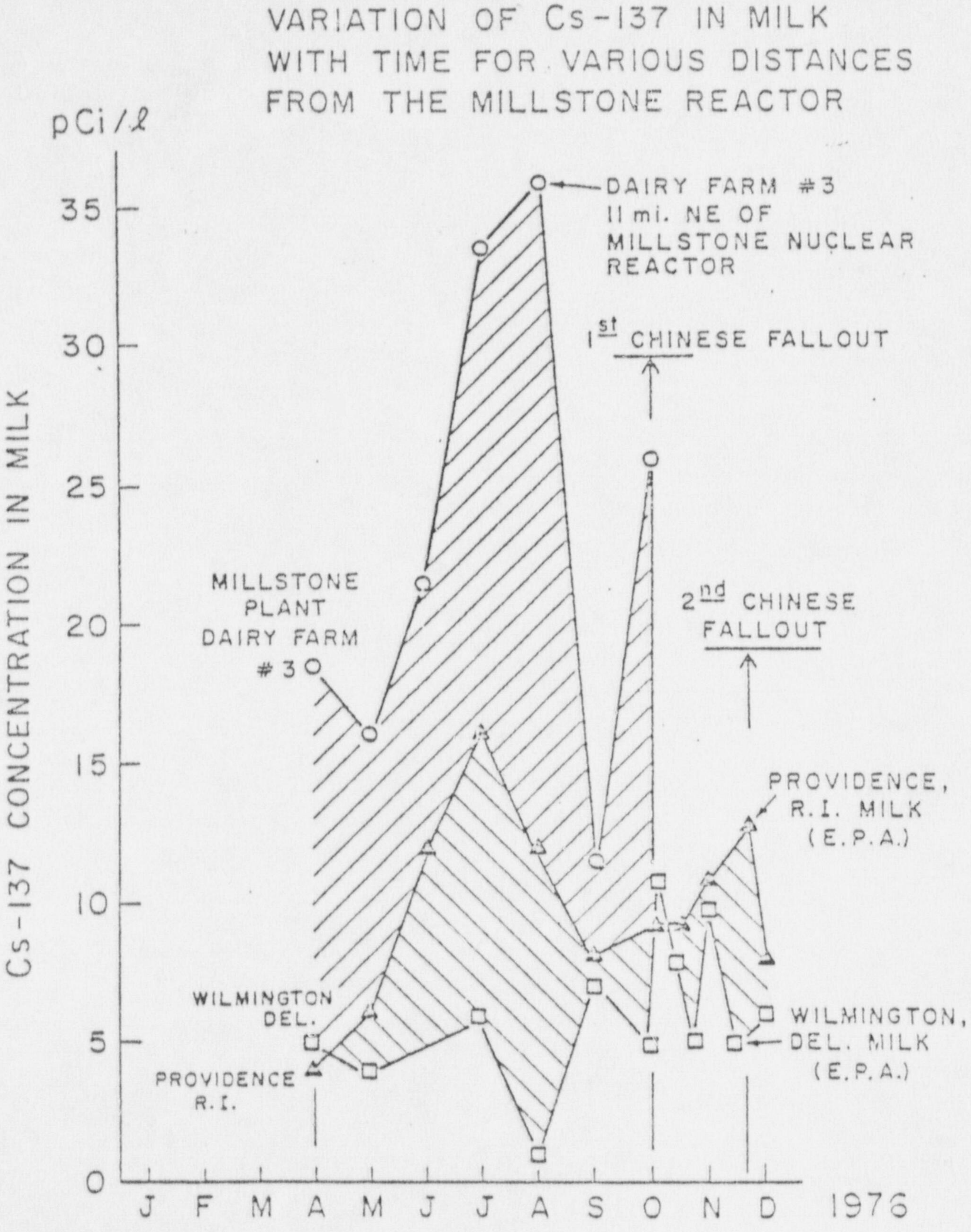

FIGURE 4

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TABLE 1

Cows milk measurements near Millstone Plant, reproduced from Environmental Statement for 1976.

TABLE 7
DAIRY MILK
(PCI/L)

LOCATION	COLLECTION DATE	SR-89	(+/-)	SR-90	(+/-)	I-131	(+/-)	CS-137	(+/-)
20	2/23/76	0.0	0.7	14.5	0.5	0.01	0.06	22.7	1.1
20	4/12/76	1.6	1.2	7.7	0.4	0.0	0.08	12.4	0.4
20	5/10/76	0.3	1.1	7.9	0.3	0.30	0.09	13.5	0.9
20	6/ 7/76	0.2	1.5	16.2	0.6	0.0	0.08	26.3	0.8
20	7/19/76	0.0	1.7	27.1	0.8	0.05	0.08	32.0	2.0
20	8/ 2/76	0.0	2.0	13.0	0.6	0.09	0.10	27.0	2.0
20	9/13/76	1.1	0.7	15.2	0.5	0.0	0.08	24.2	1.1
20	10/ 5/76	37.0	2.0	14.2	0.7	310.00	6.00	19.5	1.5
21	2/23/76	0.0	0.7	10.2	0.[illegible]	0.05	0.08	28.0	2.0
21	4/12/76	1.5	1.2	6.9	0.4	0.0	0.07	17.3	0.3
21	5/10/76	0.4	1.0	7.4	0.3	0.0	0.08	10.3	0.8
21	6/ 7/76	0.7	0.6	9.[illegible]	0.4	0.10	0.08	15.8	1.1
21	7/19/76	0.9	0.7	8.0	0.4	0.0	0.06	15.7	0.5
21	8/ 2/76	0.0	1.1	13.1	0.5	0.13	0.09	30.0	2.0
21	9/13/76	0.3	0.9	12.9	0.9	0.0	0.07	31.3	1.3
21	10/ 5/76	44.0	2.0	11.3	0.5	416.00	3.00	16.8	1.2
22	2/23/76	0.0	0.5	5.6	0.3	0.07	0.08	16.7	1.0
22	4/12/76	0.0	0.6	5.3	0.3	0.0	0.08	15.5	0.5
22	5/10/76	1.6	1.3	7.7	0.4	0.0	0.08	16.0	1.0
22	6/ 7/76	0.3	0.9	11.4	0.5	0.0	0.06	21.3	1.2
22	7/19/76	0.0	1.1	19.7	0.6	0.0	0.05	33.3	1.7
22	8/ 2/76	0.0	2.0	16.3	0.6	0.0	0.07	36.0	3.0
22	9/13/76	1.7	0.9	10.8	0.5	0.20	0.09	11.4	1.8
22	10/ 5/76	37.2	1.5	13.6	0.6	217.00	5.00	25.0	2.0
23A	2/23/76	0.0	0.4	4.5	0.3	0.05	0.08	11.8	1.1
23A	4/12/76	1.1	0.9	4.0	0.3	0.0	0.05	8.3	0.4
23A	5/10/76	0.5	0.9	5.2	0.3	0.0	0.07	13.6	1.0
23A	6/ 7/76	0.0	0.6	4.2	0.3	0.40	0.05	4.8	0.8
23A	7/19/76	0.0	0.8	7.4	0.5	0.02	0.06	5.5	0.9
23A	8/ 2/76	1.0	2.0	6.8	1.1	0.04	0.07	10.0	0.8
23A	9/13/76	0.4	0.6	7.0	0.4	0.0	0.08	13.7	1.3
23A	10/ 5/76	4.3	1.3	7.2	0.8	37.90	1.30	6.2	0.7

TABLE 2(a)

Pasteurized Milk Sr-90 concentrations in the U.S. for July 1976 reproduced from Environmental Radiation Data Report 7, U.S. E.P.A. Table 12.

Strontium-90 and Strontium-89 in Pasteurized Milk

Annual Report - July 1976

Location	^{90}Sr (pCi/l)	^{89}Sr (pCi/l)
AK:Palmer	NS	
AL:Montgomery	4.5±1.0	0±5
AR:Little Rock	6.7±1.5	0±5
AZ:Phoenix	.5± .4	1±5
CA:Los Angeles	1.4±1.1	-1±5
Sacramento	.3± .3	1±5
San Francisco	.06± .3	1±5
CO:Denver	2.8±1.6	-2±5
CT:Hartford	5.7±1.6	-2±5
CZ:Cristobal	1.9±1.0	0±5
DC:Washington	NS	
DE:Wilmington	5.2±1.7	1±5
FL:Tampa	2.8± .9	1±5
GA:Atlanta	6.9±2.0	-4±5
HI:Honolulu	1.1± .8	0±5
IA:Des Moines	3.8±1.1	0±5
ID:Idaho Falls	3.4±1.6	-1±5
IL:Chicago	4.4±1.4	0±5
IN:Indianapolis	4.1±1.0	-1±5
KS:Wichita	3.8±1.1	1±5
KY:Louisville	3.1± .8	2±5
LA:New Orleans	4.1± .9	5±5
MA:Boston	6.5±2.1	-1±5
MD:Baltimore	5.2±1.6	-2±5
ME:Portland	5.2±1.2	1±5
MI:Detroit	3.8±1.2	0±5
Grand Rapids	6.2±1.2	1±5
MN:Minneapolis	4.4±1.3	1±5
MO:Kansas City	3.8±1.4	-1±5
St. Louis	3.7±1.3	0±5
MS:Jackson	6.2±1.4	-1±5
MT:Helena	4.6±1.7	-3±5
NC:Charlotte	4.7± .9	1±5
ND:Minot	NS	
NE:Omaha	2.9±1.3	0±5
NH:Manchester	7.3±1.3	-1±5
NJ:Trenton	4.0±1.0	1±5
NM:Albuquerque	NS	

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Strontium-90 and Strontium-89 in Pasteurized Milk

Annual Report - July 1976 - Continued

Location	^{90}Sr (pCi/l)	^{89}Sr (pCi/l)
NV:Las Vegas	.7± .6	0±5
NY:Buffalo	2.3± .8	2±5
New York	6.9±2.0	-1±5
Syracuse	3.1±1.1	0±5
OH:Cincinnati	3.2±1.2	1±5
Cleveland	4.5±1.4	0±5
OK:Okalahoma City	3.6± .9	0±5
OR:Portland	5.0±1.8	-2±5
PA:Philadelphia	4.7±1.0	0±5
Pittsburgh	7.2±1.9	-2±5
PR:San Juan	1.8±1.0	0±5
RI:Providence	5.0±1.3	0±5
SC:Charleston	5.6±1.4	-1±5
SD:Rapid City	4.7±1.4	-2±5
TN:Chattanooga	5.5±1.3	0±5
Knoxville	4.3±1.2	0±5
Memphis	5.1±1.3	-1±5
TX:Austin	1.1± .9	0±5
Dallas	4.7±1.4	-1±5
UT:Salt Lake City	NS	
VA:Norfolk	3.9±1.0	0±5
VT:Burlington	4.1±1.4	-1±5
WA:Seattle	2.0± .9	1±5
Spokane	3.0± .8	1±5
WI:Milwaukee	2.6±1.2	0±5
WV:Charleston	4.8±1.3	0±5
WY:Laramie	2.5±1.7	-1±5

NS - No sample.

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TABLE 2(b)

Pasteurized milk Cs-137 concentrations in the U.S. for July 1976, reproduced from Environmental Radiation Data Report 7, U.S. E.P.A., Table 9.

Concentrations of Radionuclides in Pasteurized Milk

July 1976

Location	K Conc. (g/l)	Radionuclide Concentration (pCi/l ± 2 Sigma Counting Error) ^{137}Cs	^{140}Ba	^{131}I
AK:Palmer	NS			
AL:Montgomery	1.56± .12	7± 7	-5± 9	-1± 7
AR:Little Rock	1.53± .12	3± 7	-5± 9	4± 7
AZ:Phoenix	1.51± .12	4± 7	1± 9	2± 7
CA:Los Angeles	1.51± .12	4± 7	2± 9	3± 7
Sacramento	1.56± .12	9± 7	0±10	-1± 7
San Francisco	1.57± .12	6± 7	-7± 9	2± 7
CO:Denver	1.41± .11	8± 7	0± 9	2± 7
CT:Hartford	1.42± .11	5± 7	3± 9	-1± 7
CZ:Cristobal	1.36± .11	28± 7	-1± 9	3± 7
DC:Washington	NS			
DE:Wilmington	1.43± .11	6± 7	-5± 9	0± 7
FL:Tampa	1.51± .12	32± 7	-6± 9	5± 7
GA:Atlanta	1.25± .11	10± 7	2± 9	-4± 7
HI:Honolulu	1.42± .11	13± 7	-5± 9	4± 7
IA:Des Moines	1.42± .11	10± 7	-2± 9	-1± 7
ID:Idaho Falls	1.47± .12	6± 7	-5± 9	-1± 7
IL:Chicago	1.38± .11	12± 7	-13± 9	2± 7
IN:Indianapolis	1.46± .11	9± 7	-12± 9	1± 7
KS:Wichita	1.40± .11	9± 7	-2± 9	-1± 7
KY:Louisville	1.46± .12	6± 7	0± 9	1± 7
LA:New Orleans	1.49± .12	10± 7	0± 9	0± 7
MA:Boston	1.47± .12	8± 7	-5± 9	1± 7
MD:Baltimore	1.43± .11	5± 7	-2± 9	1± 7
ME:Portland	1.44± .11	26± 7	-4± 9	0± 7
MI:Detroit	1.40± .11	7± 7	1± 9	-4± 7
Grand Rapids	1.44± .11	7± 7	-6± 9	-3± 7
MN:Minneapolis	1.42± .11	6± 7	3±10	0± 7
MO:Kansas City	1.59± .12	7± 7	-2± 9	-2± 7
St. Louis	1.47± .12	19± 7	-2± 9	-1± 7
MS:Jackson	1.36± .11	15± 7	-6± 9	-1± 7
MT:Helena	1.52± .12	12± 7	-4± 9	-1± 7
NC:Charlotte	1.41± .11	7± 7	-2± 9	2± 7
ND:Minot	NS			
NE:Omaha	1.50± .12	9± 7	-1± 9	0± 7
NH:Manchester	1.51± .12	11± 7	-10± 9	1± 7
NJ:Trenton	1.47± .12	12± 7	-4± 9	1± 7
NM:Albuquerque	NS			

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Concentrations of Radionuclides in Pasteurized Milk

July 1976 (Continued)

Location	K Conc. (g/l)	Radionuclide Concentration (pCi/l ± 2 Sigma Counting Error) ^{137}Cs	^{140}Ba	^{131}I
NV:Las Vegas	1.57± .12	4± 7	-3± 9	1± 7
NY:Buffalo	1.46± .11	12± 7	-12± 9	0± 7
New York	1.51± .12	13± 7	-1± 9	-1± 7
Syracuse	1.50± .12	7± 7	-6± 9	0± 7
OH:Cincinnati	1.47± .11	9± 7	-11± 9	-1± 7
Cleveland	1.45± .11	5± 7	-3± 9	1± 7
OK:Okalahoma City	1.53± .12	7± 7	-1± 9	0± 7
OR:Portland	1.39± .11	6± 7	5± 9	1± 7
PA:Philadelphia	1.44± .11	5± 7	-9± 9	0± 7
Pittsburgh	1.46± .12	4± 7	-7± 9	-2± 7
PR:San Juan	1.47± .12	6± 7	-3± 9	1± 7
RI:Providence	1.52± .12	16± 7	-7± 9	-3± 7
SC:Charleston	1.41± .11	13± 7	2±10	-2± 7
SD:Rapid City	1.46± .11	4± 7	-3± 9	4± 7
TN:Chattanooga	1.47± .12	11± 7	-1± 9	-3± 7
Knoxville	1.43± .11	9± 7	-2± 9	1± 7
Memphis	1.54± .12	20± 7	7±10	-4± 7
TX:Austin	1.49± .12	8± 7	-3± 9	-4± 7
Dallas	1.58± .12	6± 7	-10± 9	-2± 7
UT:Salt Lake City	NS			
VA:Norfolk	1.47± .11	4± 7	-10± 9	0± 7
VT:Burlington	1.46± .12	9± 7	-3± 9	4± 7
WA:Seattle	1.57± .12	18± 7	-3± 9	0± 7
Spokane	1.44± .11	6± 7	-2± 9	-2± 7
WI:Milwaukee	1.52± .12	6± 7	-5± 9	-1± 7
WV:Charleston	1.44± .11	5± 7	-2± 9	-2± 7
WY:Laramie	1.45± .12	5± 7	-5± 9	4± 7

NS - No sample.

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TABLE 3

Goat Milk data from Mill. Plant Environmental Report, 1976.

TABLE 8
GOAT MILK*
(PCI/L)

LOCATION	COLLECTION DATE	SR-89	(+/-)	SR-90	(+/-)	I-131	(+/-)	CS-137	(+/-)
24	4/12/76	1.6	1.4	34.4	0.6	0.0	0.20	51.1	0.8
24	4/26/76	0.0	1.4	22.8	1.3	0.60	0.20	64.0	2.0
24	5/10/76	0.1	1.0	24.0	0.5	0.30	0.20	44.0	2.0
24	5/25/76	0.0	1.2	38.1	0.7	1.77	0.14	80.0	2.0
24	6/ 7/76	0.0	2.0	56.0	0.8	0.44	0.15	171.0	4.0
24	6/22/76	0.0	3.0	18.3	0.9	0.03	0.10	104.0	1.0
24	7/ 6/76	0.0	2.0	33.2	1.4	0.32	0.14	70.3	1.5
24	7/19/76	0.0	1.2	35.8	0.7	2.70	0.20	67.0	2.0
24	8/ 2/76	0.0	2.0	39.3	0.9	2.30	0.20	127.0	4.0
24	8/24/76	0.0	2.0	54.2	1.0	0.53	0.15	65.9	1.5
24	9/13/76	1.2	0.2	37.7	0.3	1.40	0.11	45.4	1.4
24	9/28/76	1.0	2.0	27.8	0.7	3.40	0.20	35.0	2.0
24	10/ 5/76	7.0	2.0	30.1	0.9	484.00	11.00	64.0	2.0
24	10/19/76	8.2	1.2	11.9	0.6	850.00	20.00	53.0	2.0
25A	4/12/76	0.0	1.1	23.5	0.6	0.0	0.20	28.7	0.5
25A	4/26/76	1.1	1.5	22.6	0.5	0.0	0.20	15.4	1.1
25A	5/10/76	0.2	1.8	32.6	0.6	0.10	0.20	21.9	1.1
25A	5/27/76	0.0	1.2	39.1	0.7	0.07	0.11	21.1	0.9
25A	6/ 7/76	0.0	3.0	51.0	0.9	0.20	0.40	25.8	1.3
25A	6/22/76	1.0	5.0	32.9	1.5	0.84	0.15	28.1	1.1
25A	7/ 6/76	1.6	1.3	18.0	0.6	0.0	0.11	17.0	0.7
25A	7/19/76	0.0	1.0	18.3	0.6	0.20	0.20	15.7	1.3
25A	8/ 2/76	0.0	1.2	16.5	0.5	0.10	0.20	23.7	1.1
25A	8/24/76	1.7	1.0	13.7	0.5	0.0	0.12	29.0	1.2
25A	9/16/76[a]	3.0	2.0	20.9	0.8	0.13	0.05	28.8	1.3
25A	9/28/76	0.0	1.9	25.5	0.7	0.29	0.09	17.9	0.7
25A	10/ 5/76	0.0	1.1	9.6	0.6	1.50	0.20	6.7	0.7
25A	10/19/76	0.0	1.5	22.4	0.3	1.50	0.40	20.8	1.2

* No goat milk or pasture grass samples taken in February since none were available.

a I-131 analysis done on 9/13 sample.

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TABLE 4

Pasteurized milk data from E.P.A. Environmental Radiation Data Report 6 (Oct. 1976) Covering April-June 1976. (Data for Oct.-Dec. 1976, milk in report 8 containing the effect of Chinese Nuclear tests is too bulky to be reproduced here and is avail. from the E.P.A. Office of Radiation Programs.)

Concentrations of Radionuclides in Pasteurized Milk

April 1976

Location	K Conc. (g/l)	Radionuclide Concentration (pCi/l ± 2 Sigma Counting Error) ^{137}Cs	^{140}Ba	^{131}I
AK:Palmer	NS			
AL:Montgomery	1.55± .12	10± 7	3± 10	1± 7
AR:Little Rock	1.43± .11	10± 7	± 9	8± 7
AZ:Phoenix	1.54± .12	8± 7	5± 10	-5± 7
CA:Los Angeles	1.38± .11	1± 7	-8± 9	2± 7
Sacramento	1.47± .12	4± 7	-3± 9	0± 7
San Francisco	1.51± .12	4± 7	0± 9	3± 7
CO:Denver	1.45± .11	6± 7	-1± 9	1± 7
CT:Hartford	1.50± .12	4± 7	4± 10	-1± 7
CZ:Cristobal	1.55± .12	8± 7	2± 10	4± 7
DC:Washington	1.50± .12	6± 7	2± 10	0± 7
DE:Wilmington	1.55± .12	5± 7	0± 9	-1± 7
FL:Tampa	1.52± .12	35± 7	-6± 9	-4± 7
GA:Atlanta	1.29± .11	7± 7	-3± 9	-5± 7
HI:Honolulu	1.53± .12	6± 7	-2± 9	-4± 7
IA:Des Moines	1.62± .12	11± 7	3± 10	3± 7
ID:Idaho Falls	1.52± .12	10± 7	1± 10	2± 7
IL:Chicago	1.57± .12	11± 7	-1± 9	3± 7
IN:Indianapolis	1.48± .12	8± 7	2± 10	1± 7
KS:Wichita	1.51± .12	12± 7	2± 10	3± 7
KY:Louisville	1.43± .11	10± 7	0± 9	2± 7
LA:New Orleans	1.52± .12	9± 7	1± 9	5± 7
MA:Boston	1.57± .12	11± 7	0± 10	2± 7
MD:Baltimore	1.50± .12	7± 7	2± 10	1± 7
ME:Portland	1.61± .12	17± 7	3± 10	-1± 7
MI:Detroit	1.45± .12	9± 7	11± 10	0± 7
Grand Rapids	1.49± .12	14± 7	4± 10	-1± 7
MN:Minneapolis	1.54± .12	10± 7	0± 9	-2± 7
MO:Kansas City	1.42± .12	8± 7	0± 9	-2± 7
St. Louis	1.37± .11	10± 7	-2± 9	5± 7
MS:Jackson	1.55± .12	11± 7	-5± 9	2± 7
MT:Helena	NS			
NC:Charlotte	1.57± .12	13± 7	7± 10	0± 7
ND:Minot	1.60± .12	6± 7	0± 9	0± 7
NE:Omaha	1.34± .11	15± 7	0± 9	-1± 7
NH:Manchester	1.64± .12	17± 7	4± 10	2± 7
NJ:Trenton	1.46± .12	10± 7	3± 10	3± 7
NM:Albuquerque	NS			

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Concentrations of Radionuclides in Pasteurized Milk
April 1976 - (Continued)

Location	K Conc. (g/l)	Radionuclide Concentration (pCi/l ± 2 Sigma Counting Error)		
		^{137}Cs	^{140}Ba	^{131}I
NV:Las Vegas	1.55± .12	6± 7	3± 10	7± 7
NY:Buffalo	1.53± .12	8± 7	1± 10	2± 7
New York	1.52± .12	25± 7	1± 10	3± 7
Syracuse	1.49± .12	12± 7	1± 9	-3± 7
OH:Cincinnati	1.49± .12	17± 7	0± 10	6± 7
Cleveland	1.49± .12	11± 7	-1± 9	0± 7
OK:Okalahoma City	1.56± .12	3± 7	1± 9	2± 7
OR:Portland	1.52± .12	18± 7	3± 10	2± 7
PA:Philadelphia	1.49± .12	13± 7	3± 10	0± 7
Pittsburgh	1.45± .12	7± 7	2± 10	2± 7
PR:San Juan	1.42± .11	8± 7	7± 10	-1± 7
RI:Providence	1.49± .12	-4± 7	-9± 9	-3± 7
SC:Charleston	1.52± .12	16± 7	3± 10	1± 7
SD:Rapid City	1.53± .12	10± 7	-6± 9	1± 7
TN:Chattanooga	1.47± .12	10± 7	4± 10	-1± 7
Knoxville	1.42± .11	9± 7	5± 10	3± 7
Memphis	1.63± .12	6± 7	-3± 9	-1± 7
TX:Austin	1.48± .12	9± 7	1± 9	0± 7
Dallas	1.37± .11	4± 7	-6± 9	-3± 7
UT:Salt Lake City	1.58± .12	11± 7	1± 10	1± 7
VA:Norfolk	1.42± .11	10± 7	5± 10	-2± 7
VT:Burlington	1.43± .11	9± 7	5± 10	1± 7
WA:Seattle	1.54± .12	19± 7	2± 10	0± 7
Spokane	1.42± .11	7± 7	0± 9	-4± 7
WI:Milwaukee	1.55± .12	9± 7	0± 9	2± 7
WV:Charleston	1.44± .11	11± 7	6± 10	2± 7
WY:Laramie	1.43± .11	10± 7	0± 9	-1± 7

NS - No sample.

Table 13

Concentrations of Radionuclides in Pasteurized Milk

May 1976

Location	K Conc. (g/l)	Radionuclide Concentration (pCi/l ± 2 Sigma Counting Error) ^{137}Cs	^{140}Ba	^{131}I
AK:Palmer	NS			
AL:Montgomery	1.30± .11	4± 7	-8± 9	-5± 7
AR:Little Rock	1.42± .11	3± 7	-7± 9	-3± 7
AZ:Phoenix	1.42± .11	17± 7	-6± 9	7± 7
CA:Los Angeles	1.44± .11	5± 7	-4± 9	-5± 7
Sacramento	1.44± .11	6± 7	-7± 9	5± 7
San Francisco	1.38± .11	6± 7	-7± 9	-3± 7
CO:Denver	1.44± .11	5± 7	-8± 9	2± 7
CT:Hartford	1.50± .12	8± 7	-10± 9	-2± 7
CZ:Cristobal	1.27± .11	12± 7	-1± 9	1± 7
DC:Washington	NS			
DE:Wilmington	1.45± .11	4± 7	-7± 9	-2± 7
FL:Tampa	1.47± .12	18± 7	-6± 9	2± 7
GA:Atlanta	1.43± .11	11± 7	-7± 9	-4± 7
HI:Honolulu	1.43± .12	4± 7	-9± 9	1± 7
IA:Des Moines	1.49± .12	7± 7	-6± 9	-3± 7
ID:Idaho Falls	1.49± .12	9± 7	-13± 9	2± 7
IL:Chicago	1.38± .11	6± 7	-6± 9	-3± 7
IN:Indianapolis	1.41± .11	2± 6	-6± 9	1± 7
KS:Wichita	1.50± .12	4± 7	-10± 9	2± 7
KY:Louisville	1.51± .12	2± 6	-11± 9	1± 7
LA:New Orleans	1.49± .12	9± 7	-9± 9	-1± 7
MA:Boston	1.44± .11	7± 7	-6± 9	1± 7
MD:Baltimore	1.50± .12	0± 6	-4± 9	-3± 7
ME:Portland	1.52± .12	15± 7	-5± 9	-2± 7
MI:Detroit	1.30± .11	8± 7	-2± 9	0± 7
Grand Rapids	1.41± .11	4± 7	-8± 9	-2± 7
MN:Minneapolis	1.42± .11	4± 7	-1± 9	-2± 7
MO:Kansas City	1.57± .12	5± 7	-11± 9	1± 7
St. Louis	1.49± .12	9± 7	2±10	-2± 7
MS:Jackson	1.44± .11	10± 7	-4± 9	-4± 7
MT:Helena	NS			
NC:Charlotte	1.39± .11	5± 7	-9± 9	-1± 7
ND:Minot	1.42± .11	8± 7	-8± 9	0± 7
NE:Omaha	1.61± .12	3± 7	-4± 9	-1± 7
NH:Manchester	1.55± .12	7± 7	-6± 9	-2± 7
NJ:Trenton	1.49± .12	5± 7	-7± 9	-1± 7
NM:Albuquerque	1.39± .11	9± 7	-11± 9	-1± 7

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Concentrations of Radionuclides in Pasteurized Milk

May 1976 - (Continued)

Location	K Conc. (g/l)	Radionuclide Concentration (pCi/l ± 2 Sigma Counting Error) ^{137}Cs	^{140}Ba	^{131}I
NV:Las Vegas	1.43± .11	18± 7	-5± 9	0± 7
NY:Buffalo	1.48± .12	9± 7	-4± 9	-3± 7
New York	1.48± .12	8± 7	-3± 9	-4± 7
Syracuse	1.37± .11	5± 7	-1± 9	-3± 7
OH:Cincinnati	1.48± .12	6± 7	-7± 9	2± 7
Cleveland	1.42± .11	6± 7	1± 9	-3± 7
OK:Okalahoma City	1.52± .12	6± 7	-6± 9	-4± 7
OR:Portland	1.45± .11	1± 6	-4± 9	-1± 7
PA:Philadelphia	1.52± .12	7± 7	-17± 9	3± 7
Pittsburgh	1.44± .11	3± 7	-2± 9	-6± 7
PR:San Juan	1.52± .12	9± 7	-3± 9	1± 7
RI:Providence	1.51± .12	6± 7	-1± 9	-1± 7
SC:Charleston	1.40± .11	9± 7	-8± 9	-1± 7
SD:Rapid City	1.43± .11	10± 7	-4± 9	0± 7
TN:Chattanooga	1.42± .11	9± 7	-6± 9	-2± 7
Knoxville	1.53± .12	5± 7	-5± 9	-3± 7
Memphis	1.54± .12	9± 7	-12± 9	-2± 7
TX:Austin	1.46± .11	4± 7	-7± 9	0± 7
Dallas	1.56± .12	6± 7	2±10	1± 7
UT:Salt Lake City	1.63± .12	4± 7	-10± 9	0± 7
VA:Norfolk	1.43± .11	3± 7	-6± 9	-2± 7
VT:Burlington	1.44± .11	7± 7	0± 9	-6± 7
WA:Seattle	1.36± .11	9± 7	2± 9	-7± 7
Spokane	1.49± .12	9± 7	-7± 9	-2± 7
WI:Milwaukee	1.45± .11	3± 7	-9± 9	-3± 7
WV:Charleston	1.43± .12	3± 7	-10± 9	-4± 7
WY:Laramie	1.40± .11	2± 7	4±10	0± 7

NS - No sample.

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Table 14

Concentrations of Radionuclides in Pasteurized Milk

June 1976

Location	K Conc. (g/l)	Radionuclide Concentration (pCi/l ± 2 Sigma Counting Error) ^{137}Cs	^{140}Ba	^{131}I
AK:Palmer	NS			
AL:Montgomery	1.43± .11	11± 7	-3± 9	0± 7
AR:Little Rock	1.39± .11	8± 7	-2± 9	-6± 7
AZ:Phoenix	1.52± .12	4± 7	-9± 9	-1± 7
CA:Los Angeles	1.52± .12	6± 7	-5± 9	2± 7
Sacramento	1.41± .11	6± 7	-6± 9	-1± 7
San Francisco	1.50± .12	5± 7	-10± 9	0± 7
CO:Denver	1.46± .12	7± 7	1± 9	0± 7
CT:Hartford	1.53± .12	6± 7	-10± 9	2± 7
CZ:Cristobal	1.52± .12	8± 7	0± 9	-7± 7
DC:Washington	1.53± .12	6± 7	-10± 9	0± 7
DE:Wilmington	NS			
FL:Tampa	1.45± .11	20± 7	-8± 9	-1± 7
GA:Atlanta	1.41± .11	10± 7	-8± 9	-2± 7
HI:Honolulu	1.43± .11	1± 7	-6± 9	-2± 7
IA:Des Moines	1.44± .11	5± 7	-3± 9	-2± 7
ID:Idaho Falls	1.41± .11	3± 7	7±10	0± 7
IL:Chicago	1.45± .11	3± 7	-6± 9	1± 7
IN:Indianapolis	1.48± .12	8± 7	-10± 9	-3± 7
KS:Wichita	1.41± .11	9± 7	-5± 9	-2± 7
KY:Louisville	1.50± .12	9± 7	-9± 9	-1± 7
LA:New Orleans	1.46± .11	8± 7	-4± 9	-2± 7
MA:Boston	SI			
MD:Baltimore	1.47± .12	5± 7	-6± 9	-5± 7
ME:Portland	1 44± .11	9± 7	-3± 9	-5± 7
MI:Detroit	1.35± .11	7± 7	-5± 9	1± 7
Grand Rapids	1.46± .11	6± 7	-5± 9	-2± 7
MN:Minneapolis	1.57± .12	7± 7	-9± 9	-2± 7
MO:Kansas City	1.43± .11	1± 7	2± 9	1± 7
St. Louis	1.92± .73	0± 4	-1± 9	2± 8
MS:Jackson	1.43± .11	7± 7	1± 9	0± 7
MT:Helena	1.43± .11	8± 7	-1± 9	-6± 7
NC:Charlotte	1.52± .12	7± 7	-7± 9	0± 7
ND:Minot	1.48± .12	1± 7	2± 9	-4± 7
NE:Omaha	1.41± .11	2± 6	-2± 9	-1± 7
NH:Manchester	1.74± .12	3± 7	-2± 9	-2± 7
NJ:Trenton	1.38± .11	5± 7	-7± 9	3± 7
NM:Albuquerque	NS			

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Concentrations of Radionuclides in Pasteurized Milk
June 1976 (Continued)

Location	K Conc. (g/l)	Radionuclide Concentration (pCi/l ± 2 Sigma Counting Error) ^{137}Cs	^{140}Ba	^{131}I
NV:Las Vegas	NS			
NY:Buffalo	1.44± .11	3± 7	-6± 9	-3± 7
New York	1.54± .12	4± 7	-11± 9	3± 7
Syracuse	1.46± .12	8± 7	-3± 9	4± 7
OH:Cincinnati	1.45± .11	8± 6	-3± 9	0± 7
Cleveland	1.48± .12	9± 7	-7± 9	1± 7
OK:Okalahoma City	1.50± .12	4± 7	-5± 9	5± 7
OR:Portland	1.39± .11	6± 7	-4± 9	0± 7
PA:Philadelphia	1.46± .11	5± 7	-4± 9	-6± 7
Pittsburgh	1.41± .11	-1± 7	-12± 9	6± 7
PR:San Juan	1.48± .12	6± 7	-5± 9	-4± 7
RI:Providence	1.46± .12	12± 7	-9± 9	-2± 7
SC:Charleston	1.40± .11	11± 7	-4± 9	0± 7
SD:Rapid City	1.43± .11	5± 7	-6± 9	-3± 7
TN:Chattanooga	1.42± .11	4± 7	-5± 9	0± 7
Knoxville	1.42± .11	7± 7	-3± 9	-1± 7
Memphis	1.44± .11	7± 7	-12± 9	0± 7
TX:Austin	1.47± .12	5± 7	-1± 9	4± 7
Dallas	1.44± .11	18± 7	-7± 9	0± 7
UT:Salt Lake City	1.47± .12	11± 7	1± 9	-1± 7
VA:Norfolk	1.51± .12	16± 7	-7± 9	-5± 7
VT:Burlington	1.31± .11	6± 7	-6± 9	2± 7
WA:Seattle	1.42± .12	12± 7	-5± 9	-5± 7
Spokane	1.46± .12	1± 6	-2± 9	-2±
WI:Milwaukee	1.48± .12	6± 7	-5± 9	1± 7
WV:Charleston	1.40± .11	7± 7	-8± 9	-3± 7
WY:Laramie	1.31± .11	5± 7	-3± 9	0± 7

NS - No sample.
SI - Sample inadvertantly lost in laboratory.

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TABLE 5

Federal Radiation Council Radiation Protection
Guides for daily total diet intakes (1961)

Table 1. Radiation Protection Guides—FRC recommendations and related information pertaining to environmental levels during normal peacetime operation

Radionuclide	Critical organ	RPG for individual in the general population (rad/a)	Guidance for suitable samples of exposed population group[a]				
			RPG (rad/a)	Corresponding continuous daily intake (pCi/day)	Range I (pCi/day)[b]	Range II (pCi/day)[b]	Range III (pCi/day)[b]
Strontium-89	Bone	[c]1.5	0.5	[d]2,000	0–200	200–2,000	2,000–20,000
	Bone marrow	[c].5	.17				
Strontium-90	Bone	[c]1.5	.5	[d]200	0–20	20–200	200–2,000
	Bone marrow	[c].5	.17				
Iodine-131	Thyroid	1.5	.5	100	0–10	10–100	100–1,000
Cesium-137[e]	Whole body	.5	.17	3,600	0–360	360–3,600	3,600–36,000

[a] Suitable samples which represent the limiting conditions for this guidance are: strontium-89, strontium-90—general population; iodine-131—children 1 year of age; cesium-137—infants.

[b] Based on an average intake of 1 liter of milk per day.

[c] A dose of 1.5 rad/a to the bone is estimated to result in a dose of 0.5 rad/a to the bone marrow.

[d] For strontium-89 and strontium-90, the Council's study indicated that there is currently no operational requirement for an intake value as high as one corresponding to the RPG. Therefore, these intake values correspond to doses to the critical organ not greater than one-third the respective RPG.

[e] The guides expressed here were not given in the FRC reports, but were calculated using appropriate FRC recommendations.

NOTE: TOTAL DIET ≅ 3 x MILK CONTENT OF 1 LITER PER DAY

TABLE 6

Maximum permissible doses under Appendix I to 10CFR50
N.R.C. Regulations from NUREG - 1.109

TABLE 1

SUMMARY OF STAFF POSITION -

METHODS OF EVALUATING COMPLIANCE WITH APPENDIX I

TYPE OF DOSE	APPENDIX I DESIGN OBJECTIVE	POINT OF DOSE EVALUATION	EQUATIONS TO BE USED
Liquid Effluents			
Dose to total body from all pathways	3 mrem/yr per unit	Location of the highest dose offsite* (see also Table A-1).	1, 2, 3, 4, & 5
Dose to any organ from all pathways	10 mrem/yr per unit	Same as above.	1, 2, 3, 4, & 5
Gaseous Effluents**			
Gamma dose in air	10 mrad/yr per unit	Location of the highest dose offsite.***	6 or 7, as appropriate
Beta dose in air	20 mrad/yr per unit	Same as above.	7
Dose to total body of an individual	5 mrem/yr per unit	Location of the highest dose offsite.*	8 or 10, as appropriate
Dose to skin of an individual	15 mrem/yr per unit	Same as above.	9 or 11, as appropriate
Radioiodines and Particulates† Released to the Atmosphere			
Dose to any organ from all pathways	15 mrem/yr per unit	Location of the highest dose offsite.††	12, 13, & 14

*Evaluated at a location that is anticipated to be occupied during plant lifetime or evaluated with respect to such potential land and water usage and food pathways as could actually exist during the term of plant operation.

**Calculated only for noble gases.

***Evaluated at a location that could be occupied during the term of plant operation.

†Doses due to carbon-14 and tritium intake from terrestrial food chains are included in this category.

††Evaluated at a location where an exposure pathway actually exists at time of licensing. However, if the applicant determines design objectives with respect to radioactive iodine on the basis of existing conditions and if potential changes in land and water usage and food pathways could result in exposures in excess of the guideline values given above, the applicant should provide reasonable assurance that a monitoring and surveillance program will be performed to determine: (1) the quantities of radioactive iodine actually released to the atmosphere and deposited relative to those estimated in the determination of design objectives; (2) whether changes in land and water usage and food pathways which would result in individual exposures greater than originally estimated have occurred; and (3) the content of radioactive iodine and foods involved in the changes, if and when they occur.

TABLE 7(a)

Measurements of radioactivity in cows milk near Haddam Neck reproduced from the Environmental Report, 1976

TABLE 7
DAIRY MILK
(PCI/L)

LOCATION	COLLECTION DATE	SR-89	(+/-)	SR-90	(+/-)	I-131	(+/-)	CS-137	(+/-)
69	2/23/76	0.0	0.7	8.8	0.4	0.0	0.09	27.3	1.1
69	4/12/76	0.0	0.8	9.8	0.7	0.02	0.09	20.1	0.8
69	5/10/76	0.0	0.8	4.7	0.3	0.0	0.08	11.2	1.0
69	6/ 7/76	0.9	1.1	4.1	0.3	0.18	0.07	9.0	1.0
69	7/19/76	0.0	0.6	6.9	0.3	0.0	0.07	13.9	1.7
69	8/ 2/76	0.0	1.4	6.4	0.3	0.0	0.07	27.6	0.8
69	9/13/76	0.3	0.9	5.2	0.3	0.09	0.08	11.6	0.8
69	10/ 5/76	3.0	1.1	4.6	0.4	143.00	2.00	4.1	0.6
70	2/23/76	0.0	0.6	6.6	0.4	0.09	0.08	7.0	0.5
70	4/12/76	0.0	0.6	6.4	0.4	0.0	0.11	6.3	0.5
70	5/10/76	1.3	0.9	6.1	0.3	0.04	0.09	6.3	1.0
70	6/ 7/76	0.5	1.0	5.5	0.3	0.13	0.07	9.2	1.0
70	7/19/76	0.8	0.5	4.0	0.2	0.09	0.07	7.4	0.7
70	8/ 2/76	0.9	0.8	5.9	0.5	0.05	0.10	6.5	0.3
70	9/13/76	1.0	2.0	7.7	0.8	0.10	0.07	6.2	0.7
70	10/ 5/76	16.4	0.9	9.1	0.4	201.60	0.70	5.4	0.8
71	2/23/76	0.2	0.4	5.5	0.3	0.0	0.06	8.6	0.9
71	4/12/76	0.0	0.7	4.8	0.3	0.0	0.09	6.7	0.5
71	5/10/76	0.3	0.9	3.8	0.2	0.07	0.08	5.4	0.6
71	6/ 7/76	1.6	1.2	6.5	0.4	0.04	0.07	15.7	0.7
71	7/19/76	1.0	0.7	8.4	0.3	0.02	0.07	36.0	2.0
71	8/ 2/76	0.0	1.0	11.2	0.7	0.0	0.10	52.0	3.0
71	9/13/76	1.4	0.8	10.5	0.6	0.03	0.09	30.9	1.1
71	10/ 5/76	0.0	0.4	4.8	0.2	243.00	7.00	16.0	1.2
72A	2/24/76	0.0	0.6	7.8	0.3	0.09	0.06	13.8	0.6
72A	4/12/76	0.0	0.6	6.6	0.3	0.0	0.08	9.5	0.5
72A	5/10/76	0.0	0.8	6.5	0.3	0.01	0.10	7.4	1.0
72A	6/ 7/76	0.2	0.9	6.8	0.3	0.06	0.05	7.9	0.6
72A	7/19/76	0.0	0.5	4.8	0.3	0.0	0.06	15.8	1.5
72A	8/ 2/76	0.0	0.4	3.1	0.2	0.0	0.07	13.3	0.9
72A	9/13/76	0.9	0.9	4.8	0.3	0.09	0.12	12.4	0.8
72A	10/ 5/76	8.0	2.0	5.6	0.5	175.00	91.40	12.6	1.4

TABLE 7(b)

Measurement of radioactivity in goat milk near Haddam Neck reproduced from the Environmental Report, 1976.

TABLE 8
GOAT MILK
(PCI/L)

LOCATION	COLLECTION DATE	SR-89	(+/-)	SR-90	(+/-)	I-131	(+/-)	CS-137	(+/-)
73A	4/12/76	0.0	0.9	18.1	0.4	0.0	0.14	20.7	0.8
73A	5/10/76	1.8	1.6	14.7	0.5	0.02	0.12	56.0	2.0
73A	6/ 7/76	0.0	1.9	18.4	0.5	0.40	0.20	37.8	1.3
73A	7/19/76	0.7	0.7	10.6	0.4	0.18	0.15	23.0	2.0
73A	8/ 2/76	0.0	1.1	11.6	0.4	0.0	0.20	52.0	3.0
73A	9/13/76	0.4	0.9	8.8	0.4	0.09	0.08	11.9	0.8
73A	10/ 5/76	8.9	1.4	10.3	0.5	196.00	2.00	28.2	1.3
74	4/12/76	0.0	1.1	19.0	2.0	0.10	0.20	17.5	0.8
74	5/10/76	0.0	1.4	15.6	0.4	0.05	0.14	37.2	1.5
74	6/ 7/76	1.4	2.5	32.6	0.9	0.09	0.15	77.0	3.0
74	7/19/76	0.0	0.8	19.1	0.6	0.0	0.20	23.9	1.1
74	8/ 2/76	1.9	0.7	23.7	0.7	0.0	0.20	37.0	2.0
74	9/13/76	0.5	0.6	5.5	0.4	0.34	0.07	19.7	1.2
74	10/ 5/76	5.9	1.4	12.8	0.5	57.00	0.60	22.7	1.1

No goat milk or pasture grass samples taken in February since none were available.

TABLE 8

Comparisons of Thermal Efficiency and
Radioactive Releases From the Yankee (Rowe),
the Haddam Neck (Connecticut Yankee) and Millstone
Point Plants (Tables Reproduced from "Report on Releases of Radioactive
Effluents", 1972, U.S.A.E.C. Directorate of Regulations, (August 1973)

	Yankee (Rowe) Mass. (PWR)	Connect. Yankee, Haddam (PWR)	Millstone Point-1, Conn. (BWR)
Lic. Power MW (Therm.)	600 MW	1825 MW	2011 MW
Elect. Power MW (el.) 1972	173 MW	568 MW	658 MW
Thermal Efficiency (1972 Data)	28.8%	31.1%	32.7%
I-131, Cs-137, Sr-90 etc. Airborne Release, Ci (1972)	0.00077 Ci	0.0181 Ci	1.32 Ci
I-131, Cs-137, Sr-90 etc. Airborne Release; Ci Per 1000 MW Electrical Power	0.0044	0.3187	4.0367
Relative Gaseous (I, Sr, Cs) Releases per 1000 MW Power	1	72	917
Year of Start-up	1960	1967	1970

Data taken from A.E.C. 1972 report (See tables reproduced in Appendix VIII)
Note the much greater releases of the biologically most hazardous materials
for the more efficient, later reactors.

TABLE 9

Comparison of Sr-90 Concentrations in Milk Near the Millstone Nuclear Reactor With Concentrations in Hartford and the U.S. as a Whole - 1970 to 1976

Year	(a) Av. Daily Milk Sr-90 Concentr. Near Millstone pCi/l	(b) Av. Daily Milk Sr-90 Concentr. In July (Hartford) pCi/l	(c) Av. Daily Milk Sr-90 Concentr. for Year In U.S. pCi/l	Excess Sr-90 In Milk Near Millstone over U.S. pCi/l	%Exc Sr-9 Near Mills pCi/
1970#	9.8	8	8	1.8	—
1971	8.8	9	7	1.8	20%
1972	9.6	7	6	3.6	38%
19[illegible]	15.0	4	5	10.0	67%
1974	14.8	Not Avail.	4	10.8	73%
1975	10.7	3.1	3	7.7	72%
1976	13.0	5.7	4	9.0	69%

\# Millstone Operation began in October 26, 1970
Conn. Yankee (Haddam Neck, 20 miles N.W., Started July 24, 1967.

(a) Three locations within 10-15 miles; $1\sigma = \pm 0.2$ pCi/l
From Millstone environmental reports, annual averages

(b) E.P.A. Measurements (Rad. Data and Reports) in July.
$1\sigma = \pm 1$ pCi/l

(c) E.P.A. Network Average (Rad. Data and Reports)
$1\sigma = \pm 0.2$ pCi/l

TABLE 10

Doses to the Bone of Children Due to Sr-90 in the Milk and Total Diet Near the Millstone Nuclear Plant - 1970 to 1976

Year	Total Diet Sr-90 Intake Near Millstone pCi/day	(a) Annual Sr-90 Bone Dose For Child-All Sources mrem/yr	(b) Annual Sr-90 Bone Dose For Child Due to Millstone mrem/yr.	Cumul. Sr-90 Bone Dose For Child Due To Millstone mrem	(c) Annual Sr-9 Bone Dose F Child Due T Millstone a % of Natura
1970#	29.4	185	----	----	----
1971	26.4	166	33	33	47%
1972	28.8	181	69	102	99%
19[illegible]	45.0	283	190	292	271%
1974	44.4	279	204	495	291%
1975	32.1	202	145	640	207%
1976	39.0	245	169	809	241%

Millstone Operations Began October 26, 1970

(a) Using dose factor of 0.0172 mrem/pCi annual intake from Table A-5, NUREG 1.109 (N.R.C., March 1976), equivulant to 6.28 mrem/yr. per 1 pCi daily intake in total diet.

(b) Using percent excess Sr-90 levels due to Millstone from milk measurements (Table 9).

(c) Natural Radiation background 70 mrem/yr. ((E.P.A. measurements; E.P.A. report on Haddam Neck
E.P.A. - 520/3-74-00[illegible] ; Sect. 7.7 , page [illegible]

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TABLE 11

Average Doses to Adult Bone From Sr-90 in the Milk and Total Diet Near the Millstone Point Nuclear Plant, 1970-1976

Year	Annual Adult Bone Dose From Total Sr-90 Intake Near Millstone mrem/yr.	Annual Adult Bone Dose Due to Sr-90 from Mill-Stone mrem/yr.	Cumulative Adult Bone Dose Due To Sr-90 From Millstone mrem/yr.	Annual Adult Bone Tot. Pop. Dose Due to Millstone Man-Rem	Cumulative Adult Bone Total Pop. Dose Due To Millstone Man-Rem	Cumulative Health Costs of Bone Doses From Millstone Mill. Dollars
1970#	—	—	—	—	—	—
1971	73	15	15	11,100	11,100	11.1
1972	80	30	45	22,200	33,300	33.3
1973	125	84	129	62,160	95,460	95.5
1974	123	90	219	66,600	162,060	162
1975	89	64	283	47,360	209,420	209
1976	108	75	358	55,500	264,920	264

\# Millstone Operation Began October 26, 1970

(a) Based on dose factor of 0.00761 mrem/pCi annual intake from Table A - 3 NUREG 1.109 (NRC, March 1976)

(b) Based on population of New London County plus one-half of Rhode Island (740,000)

(c) Based on N.R.C. and E.P.A. Health cost of $1000 per man-rem

APPENDIX I
Report to Congr. C.J. Dodd, Sept. 1977

EVALUATION OF RADIOACTIVE EFFLUENTS FROM MILLSTONE UNIT NO. 1

OFFICE OF NUCLEAR REACTOR REGULATION

NORTHEAST NUCLEAR ENERGY COMPANY

DOCKET NO. 50-245

MILLSTONE UNIT NO. 1

1.0 Introduction

During 1975, the liquid and gaseous effluents from Millstone Unit No. 1 were high in radioactivity, in terms of Curie content, as compared to other nuclear power facilities. The dose rate due to these effluents to the local population and it's significance is addressed in Section 2.0 below. Our conclusions regarding effluents from Millstone Unit No. 1 are contained in Section 3.0.

2.0 Explanation for Radioactive Effluents from Millstone Unit No. 1 and Significance to Local Population

During 1975, the comparatively high curie content of the radioactivity, in gaseous effluents from Millstone Unit No. 1, was due primarily to defects in the nuclear fuel that was being utilized at that time. The comparatively high Curie content of the liquid effluents was due mostly to the conduct of required plant maintenance. Moreover, the liquid radwaste was diluted and discharged rather than solidified as is currently the practice at Millstone Unit No. 1. The effluent release data is summarized in Table 1. Although the Curie content of the effluents from Millstone Unit No. 1 during 1975 was the highest for any nuclear power reactor in the United States, the concentration (measured as micro Curies per milliliter) of radioactive material in the effluents was controlled and represent only small fractions of the limits specified in Title 10, Code of Federal Regulations, Part 20.

During 1976, as can be seen in the attached Table 1, the radioactive effluents Curie content decreased considerably. The decrease in the Curie content of the gaseous effluents was due partly to removal of some of the defective fuel in the fall of 1975 during the third refueling of Millstone Unit No. 1. In addition, the licensee (Northeast Nuclear Energy Company) adopted a plant operating technique specifically designed to further maintain fuel integrity, thus reducing the radioactive components of gaseous effluents. The decrease in the level of liquid effluents during 1976 was due to the startup of the liquid radwaste treatment system. This system concentrates and solidifies liquid waste for removal from the site to approved disposal areas instead of discharging the diluted liquid effluent from the plant. During the fourth refueling of Millstone Unit No. 1, in the fall of 1976, more of the previous defective fuel was removed thereby causing the Curie content of the gaseous effluent to undergo a further decrease during 1977. During 1978, we expect that the new augmented gaseous radwaste treatment system will become operational and will result in a further decrease in radioactivity in gaseous effluents.

The calculated dose to the population, as a result of operation of Millstone Unit No. 1, is summarized in Table 2. During the comparatively high releases of 1975, it is significant to note that the average individual's dose in the population located within 50 miles from Millstone Unit No. 1 was only 0.2 millirem per year. As can be seen in Table 2, this exposure is very small compared to the natural background (natural radiation) level of 125 millirem per year. Moreover, the 0.2 millirem per year is small when compared to the normal variation in the background radiation level over the state of Connecticut of at least ±15 millirem per year.

3.0 Conclusion

From the information presented in Section 2.0, we conclude that the doses from the operation of Millstone Unit No. 1 to the population within 50 miles of the facility is statistically indistinguishable from the natural background radiation doses. The extremely low levels of exposure (less than 0.2% of natural background radiation dose) precludes distinguishing any health effects that could have even theoretically been produced by the operation of Millstone. Even so, the licensee has taken positive action to further decrease the liquid and gaseous radioactive effluents from Millstone Unit No. 1, so that we expect the Curie content of radioactive effluents to decrease through 1978 with a corresponding decrease in the dose to the offsite population.

TABLE 1

Release of Radioactive Material from Millstone Unit No. 1

	Airborne Releases (Ci/yr)				Liquid Releases (Ci/yr)	
	Noble Gas		Radioiodine 131			
	Release	% of Federal Release Limit	Release	% of Federal Release Limit	Release	% of Federal Release Limit
1975	3,000,000	12	10	10	199	11.
1976	500,000	2	2	2	9.3	0.5
Projected 1977	200,000	1.5	2	2	0.5	0.06
After mid 78	10,000	0.07	0.6	0.6	0.2	0.01

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TABLE 2

Radiation Dose due to Radioactive Materials Released from Millstone Unit No. 1

	Maximum Individual Dose (mrem/yr)				Population Dose within 50 miles of station (man-rem)			
	Total Body	Thyroid	Natural Background	% of Natural Background	Reactor Releases	Natural Background	Avg. Indv. Dose (mr/yr)	% of Na Backgr
1975	30	10	125 ± 15 *	27	670	370,000	0.2	0.2
1976	6	2	125 ± 15	5	112	370,000	0.04	0.0
Projected 1977	5	1.5	125 ± 15	4	93	370,000	0.03	0.0

*This value is the average for the State of Connecticut. The natural background for Millstone Point averages 150 mrem/yr and is higher than the average for the State due to the granitic nature of the site.

APPENDIX II
From EPA Region II ERAMS 902/4-77-009

ERAMS Data   Analysis of Water Samples Under Drinking Water Standards

- Newly-established drinking water standards, effective June 24, 1977, set the following maximum levels of radioactivity in drinking water:

tritium	20,000 pCi/l
Sr-90	8 pCi/l
Sr-89	80 pCi/l
Cs-134	20,000 pCi/l
Cs-137	200 pCi/l
Ba-140	90 pCi/l
Ra-226 and 228	5 pCi/l

- After analysis of the data collected by the ERAMS network during 1975 and 1976 the following information was noted:

  -- Albany drinking water levels were 2% or less of the tritium limit, 3% of the Sr-90 limit, and 4% of the Ra-226 limit.

  --Bayside (NJ) surface water tritium levels were a maximum of 2% of the drinking water limit.

  -- Buffalo drinking water levels were 2% or less of the tritium limit, 14% of the Sr-90 limit, and 6% or the Ra-226 limit. WEST-VALLEY

  -- Ossining surface water samples never exceeded 1.5% of the maximum tritium drinking water limit.

  -- Oswego surface water samples were never more than 2% of the drinking water tritium limit.

  -- Oyster Creek surface water samples were never in excess of 1.5% of the tritium drinking water standard.

  -- Poughkeepsie surface water samples never exceeded 2.5% of the tritium drinking water standard.

  -- San Juan drinking water never exceeded 1.5% of the tritium limit, 6% of the Sr-90 limit, or 8% of the Ra-226 limit.

  -- Syracuse drinking water never exceeded 4%of the tritium limit, 10% of the Sr-90 limit, or 8% of the Ra-226 limit. Average values for the location were 2% of the tritium limit, 5% of the Sr-90 limit, and 4% of the Ra-226 limit.

  -- Trenton drinking water levels were 2% of the tritium limit, 20% of the Sr-90 limit, and 6% of the Ra-226 limit. Average values for the city were less than 1.5%of the tritium limit, 10% of the Sr-90 limit, and 3% of the Ra-226 limit. N.J

  -- Waretown (NJ) drinking water levels were 2% of the tritium levels, 1% of the Sr-90 limit, and 38% of the Ra-226 limit.

- Average levels in New York State were 1.5% of the tritium limit, 5% of the Sr-90 limit, and 4% of the Ra-226 Limit.

APPENDIX III

1.0 SUMMARY

The radiological environmental monitoring program around the Millstone Nuclear Power Station was continued for the period January through December 1976, in compliance with the Environmental Technical Specifications, Section 3.2. This report for 1976 was prepared by the Radiation Assessment staff of the Northeast Utilities Service Company. The laboratory analyses were done by Interex Corporation of Natick, Massachusetts who also assisted in the qualitative interpretation of the laboratory data.

Sampling and radiological analyses were performed on air particulates gamma exposure rates, soil, milk, fruit, vegetables, well water, reservoir water, bottom sediment, sea water, mussels, oysters, clams, scallops, lobster, fin fish, algae and eggs.

The observed results indicate that the predominant radioactivity at offsite locations are from nonplant related sources such as fallout from nuclear weapons tests and from naturally occurring nuclides. Plant related radioactivity above the minimum detectable levels, as set by counting statistics was observed; as gamma exposure rate at three locations within 3 miles of the station; as iodine-131 in goats milk; as manganese-54 and cobalt-60 in bottom sediment collected within 500 feet of the discharge; as cesium-137, manganese-54, cobalt-58 and cobalt-60 in rockweed collected within 500 feet of the discharge; as cesium-137 , cesium 134, cobalt-60 and manganese-54 in mussels and oysters collected within 500 feet of the discharge; as manganese-54 in scallops; and as manganese-54 and cobalt-60 in lobsters. In general the radioactivity in 1976 was less than that observed in 1975 and the levels in aquatic media in 1976 exhibited a rapidly decreasing trend through the year. This is as a result of the operation of the augmented liquid radioactive waste treatment system.

The radiation dose to the general public from the stations discharges have been evaluated by two methods; one using the measured stations discharges and conservative transport models, and the other using the measured concentrations of radioactivity in environmental media. The maximum dose (at the station boundary) that could occur to a member of the general public as a result of the stations discharges was 7.9 millirem and the average dose to an individual residing within 50 miles of the station is 0.13 millirem. These doses are 1.6 percent and 0.08 percent of the corresponding Federal and State standards for annual permissible doses to the public from man-made radiation, which are 500 millirem and 170 millirem respectively. Natural background radiation in Connecticut gives members of the public a dose of 129 millirem per year. Thus the stations effect on the public is minimal.

Table 4. Mean active bone marrow dose for several different examinations: subject No. 16 of the USPHS study

Examination	SSD	HVL (mm Al)	Calc. Skin Exposure (mR)	Mean Active Marrow Dose per film (mrad)
AP Lumbosac. Spine	64	1.8	1705	87.2
Lat. Lumbosac. Spine	50	1.9	4603	56.2
PA Chest	80	2.7	16	2.2
PA Upper GI	74	2.3	712	51.1
PA IVP	80	2.4	125	13.3
PA Chest	80	2.2	31	4.0
AP Cerv. Spine	80	2.2	189	4.9
PA Barium Enema	80	2.2	759	97.1
Lat. Hip	67	2.6	666	26.7
Lat. Skull	80	2.3	243	13.0
PA Sinuses	71	2.3	987	16.5
AP Left Shoulder	80	2.4	28	3.3
Lat. Skull	79	2.3	311	15.9
PFG Chest	79	2.1	380	41.2
Lat. Gall Bladder	60	2.5	546	24.3
PA Barium Enema	74	3.3	608	119.0
AP Urethrogram	74	2.4	55	5.1
PA Ribs	72	2.0	1049	82.5
AP Thor. Spine	72	3.0	482	84.2
AP Mandible	76	2.2	222	12.9
Lat. Chest	80	3.0	218	20.5
Lat. Cerv. Spine	80	1.0	94	2.0
PA Chest	80	2.3	33	4.1
Lat. Dental	20	2.0	1186	1.8
AP Dental	31	2.6	806	3.7

SUMMARY

The exposure measurements, model and computer program for estimation of mean active bone marrow doses formerly employed in the 1962 British Survey of x-ray doses and proposed for application to x-ray exposure information obtained in the U.S. Public Health Service's X-Ray Exposure Studies (1966 and 1973) are described and evaluated.

The method described in this paper is feasible for use to determine the mean active bone marrow doses to adults for examinations having a skin to source distance of 80 cm or less. For a greater SSD, as for example in chest x-rays, a small correction in the calculated dose can be made.

From DHEW-FDA-76-8015; Ref. 24

APPENDIX V(a)

From N.R.C. Reg. Guide 1.109 March 76'

TABLE A-3

ADULT INGESTION DOSE FACTORS
(mrem/pCi ingested)

NUCLIDE		BONE	LIVER	TOTAL BODY	THYROID	KIDNEY	LUNG	GI-LLI
1H	3	0.0	1.34E-07	1.34E-07	1.34E-07	1.34E-07	1.34E-07	1.34E-07
4BE	10	3.18E-06	4.91E-07	7.95E-08	0.0	3.71E-07	0.0	2.6ME-05
6C	14	2.84E-06	5.69E-07	5.69E-07	5.69E-07	5.69E-07	5.69E-07	5.69E-07
7N	13	8.37E-09	8.37E-09	8.37E-09	8.37E-09	8.37E-09	8.37E-09	8.37E-09
9F	18	6.25E-07	0.0	6.93E-08	0.0	0.0	0.0	1.85E-08
11NA	22	1.74E-05	1.74E-05	1.74E-05	1.74E-05	1.74E-05	1.74E-05	1.74E-05
11NA	24	2.26E-06	2.26E-06	2.26E-06	2.26E-06	2.26E-06	2.26E-06	2.26E-06
15P	32	1.93E-04	1.21E-05	7.47E-06	0.0	0.0	0.0	2.17E-05
20CA	41	1.87E-04	0.0	[illegible]	0.0	0.0	0.0	1.84E-07
21SC	46	5.51E-09	1.08E-08	3.11E-09	0.0	1.00E-08	0.0	5.21E-05
24CR	51	0.0	0.0	2.66E-09	1.59E-09	5.87E-10	3.53E-09	6.69E-07
25MN	54	0.0	4.57E-06	8.73E-07	0.0	1.36E-06	0.0	1.40E-05
25MN	56	0.0	1.15E-07	2.05E-08	0.0	1.46E-07	0.0	3.67E-06
26FE	55	6.20E-06	2.79E-05	7.33E-06	0.0	0.0	3.23E-05	1.09E-05
26FE	59	4.34E-06	1.03E-05	3.92E-06	0.0	0.0	2.86E-06	3.40E-05
27CO	57	0.0	1.75E-07	2.91E-07	0.0	0.0	0.0	4.44E-06
27CO	58	0.0	7.46E-07	1.67E-06	0.0	0.0	0.0	1.51E-05
27CO	60	0.0	2.15E-06	4.72E-06	0.0	0.0	0.0	4.02E-05
28NI	59	9.77E-06	3.35E-06	1.33E-06	0.0	0.0	0.0	6.90E-07
28NI	63	1.30E-04	9.02E-06	4.36E-06	0.0	0.0	0.0	1.88E-06
28NI	65	5.29E-07	6.87E-08	3.13E-08	0.0	0.0	0.0	1.74E-06
29CU	64	0.0	8.34E-08	3.92E-08	0.0	2.10E-07	0.0	7.10E-06
30ZN	65	4.85E-06	1.54E-05	6.97E-06	0.0	1.03E-05	0.0	9.70E-06
30ZN	69M	1.70E-07	4.09E-07	3.73E-08	0.0	2.48E-07	0.0	2.49E-05
30ZN	69	1.03E-08	1.98E-08	1.37E-09	0.0	1.28E-08	0.0	2.96E-09
34SE	79	0.0	2.63E-06	4.40E-07	0.0	4.56E-06	0.0	5.38E-07
35BR	82	0.0	0.0	2.26E-06	0.0	0.0	0.0	2.59E-06
35BR	83	0.0	0.0	4.02E-08	0.0	0.0	0.0	5.79E-08
35BR	84	0.0	0.0	5.22E-08	0.0	0.0	0.0	4.09E-13
35BR	85	0.0	0.0	2.14E-09	0.0	0.0	0.0	0.0
37RB	86	0.0	2.11E-05	9.84E-06	0.0	0.0	0.0	4.16E-06
37RB	87	0.0	1.23E-05	4.28E-06	0.0	0.0	0.0	5.76E-07
37RB	88	0.0	6.06E-08	3.21E-08	0.0	0.0	0.0	8.36E-19
37RB	89	0.0	4.01E-08	2.83E-08	0.0	0.0	0.0	0.0
38SR	89	3.09E-04	0.0	8.85E-06	0.0	0.0	0.0	4.94E-05
Sr 90 → 38SR	90	7.61E-03	0.0	1.86E-03	0.0	0.0	0.0	1.02E-04
38SR	91	5.82E-06	0.0	2.56E-07	0.0	0.0	0.0	2.93E-05
38SR	92	2.16E-06	0.0	9.31E-08	0.0	0.0	0.0	4.26E-05
39Y	90	9.63E-09	0.0	2.58E-10	0.0	0.0	0.0	1.02E-04
39Y	91M	9.10E-11	0.0	3.53E-12	0.0	0.0	0.0	2.67E-10
39Y	91	1.41E-07	0.0	3.78E-09	0.0	0.0	0.0	7.76E-05

Note: 0.0 means insufficient data or that the dose factor is <1.0E-20.

TABLE A-5

CHILD INGESTION DOSE FACTORS
(mrem/pCi ingested)

Sr-90

NUCLIDE		BONE	LIVER	TOTAL BODY	THYROID	KIDNEY	LUNG	GI-LLI
1H	3	0.0	2.03E-07	2.03E-07	2.03E-07		2.03E-07	2.03E-07
6C	14	2.26E-06	2.26E-06	2.26E-06	2.26E-06		2.26E-06	2.26E-06
11NA	22	5.89E-05	5.89E-05	5.89E-05	5.89E-05		5.89E-05	2.57E-06
27CO	58	0.0	1.85E-06	5.58E-05	0.0		0.0	1.10E-05
27CO	60	0.0	5.17E-06	1.55E-05	0.0		0.0	2.86E-05
38SR	89	1.38E-03	0.0	3.95E-05	0.0	(USE	0.0	5.15E-05
38SR	90	1.72E-02	0.0	4.36E-03	0.0		0.0	2.29E-04
39Y	90	4.21E-08	0.0	1.13E-09	0.0	ADULT	0.0	1.20E-04
39Y	91	5.85E-07	0.0	1.56E-08	[illegible]		0.0	7.77E-05
40ZR	95	1.04E-07	2.42E-08	2.20E-08	[illegible]		0.0	2.50E-05
41NB	95	1.95E-08	8.32E-09	6.11E-09	[illegible]	DOSE	0.0	1.44E-05
44RU	103	6.78E-07	0.0	2.79E-07	[illegible]		0.0	1.78E-05
44RU	106	1.19E-05	0.0	1.48E-06	[illegible]		0.0	1.65E-04
50SN	123	1.31E-04	1.62E-06	3.22E-06	[illegible]		0.0	6.50E-05
52TE	125M	1.14E-05	3.09E-06	1.52E-06	[illegible]	FACTOR)	0.0	1.10E-05
52TE	127	4.50E-07	1.20E-07	9.65E-08	3.10E-07		0.0	1.92E-05
52TE	129M	4.95E-05	1.38E-05	7.65E-06	1.58E-05		0.0	5.96E-05
52TE	132	1.02E-05	4.50E-06	5.42E-06	6.62E-06		0.0	7.89E-05
53I	129	1.39E-05	8.54E-06	3.81E-05	2.79E-02		0.0	4.29E-07
53I	131	1.63E-05	1.67E-05	1.26E-05	5.43E-03		0.0	1.43E-06
53I	133	5.98E-06	7.38E-06	2.90E-06	1.79E-03		0.0	2.99E-06
55CS	134	2.24E-04	3.77E-04	8.02E-05	0.0		4.19E-05	2.04E-06
55CS	137	3.12E-04	3.02E-04	4.50E-05	0.0		3.54E-05	1.84E-06
56BA	140	8.26E-05	7.25E-08	4.85E-06	0.0		4.32E-08	4.21E-06
57LA	140	1.01E-08	3.52E-09	1.19E-09	0.0		0.0	1.00E-04
58CE	141	3.76E-08	1.88E-08	2.80E-09	0.0		0.0	2.36E-05
58CE	144	2.14E-06	6.70E-07	1.14E-07	0.0		0.0	1.74E-04
63EU	154	2.58E-06	2.08E-07	2.03E-07	0.0		0.0	4.74E-05
92U	232	1.77E-02	0.0	1.26E-03	0.0		0.0	6.91E-05
92U	234	3.57E-03	0.0	2.21E-04	0.0		0.0	6.32E-05
94PU	238	1.24E-03	1.52E-04	3.04E-05	0.0		0.0	7.50E-05
94PU	239	1.32E-03	1.62E-04	3.27E-05	0.0		0.0	6.85E-05
94PU	240	1.32E-03	1.63E-04	3.30E-05	0.0		0.0	6.85E-05
94PU	241	7.12E-07	8.50E-08	1.81E-08	0.0		0.0	1.32E-07
95AM	241	1.42E-03	6.24E-04	9.96E-05	0.0		0.0	7.37E-05
96CM	242	6.74E-05	5.28E-05	4.46E-06	0.0		0.0	8.03E-05
96CM	244	1.12E-03	5.40E-04	6.99E-05	0.0		0.0	7.64E-05

Note: 0.0 means insufficient data or that the dose factor is <1.0E-20.

TABLE A-6

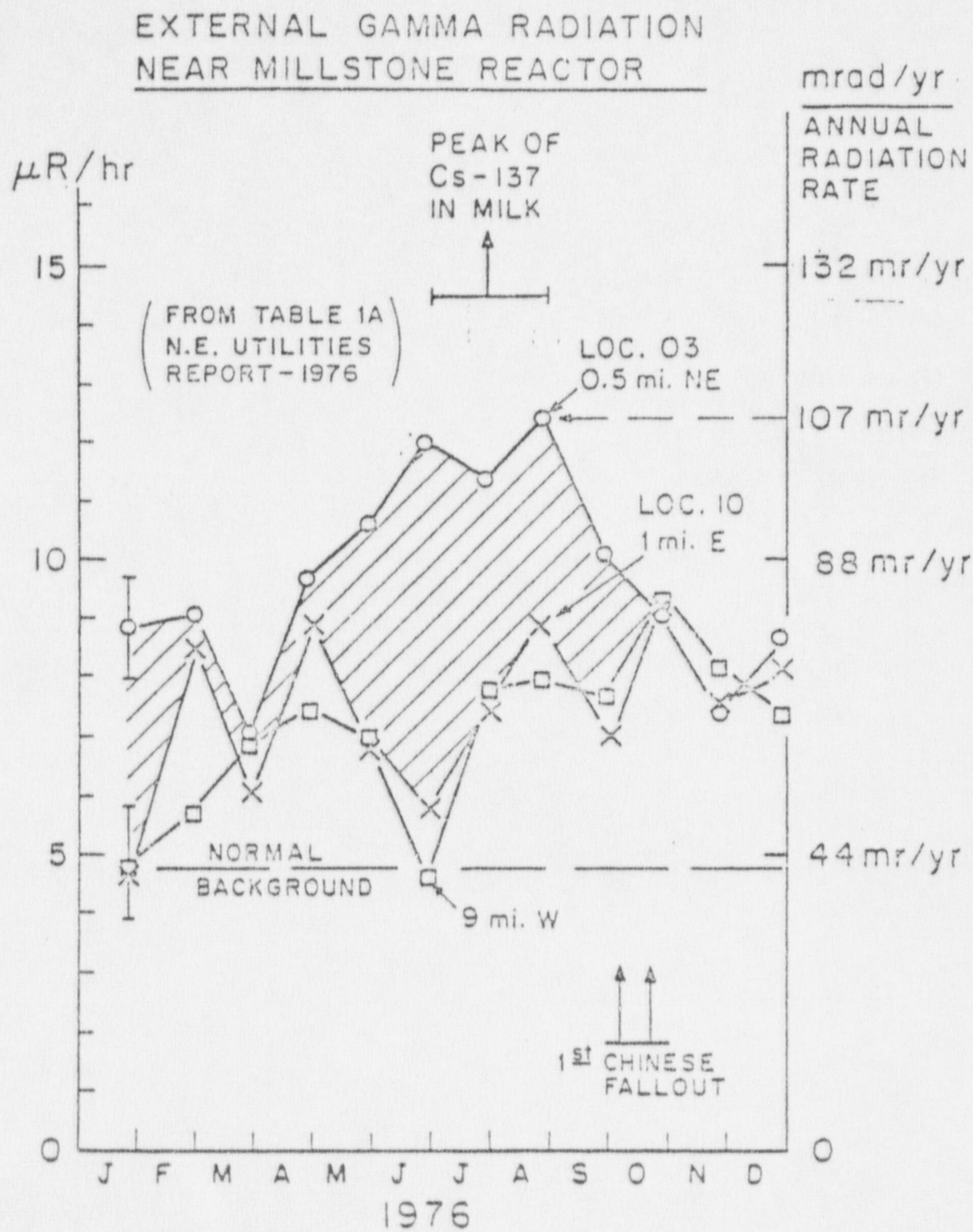
INFANT INGESTION DOSE FACTORS
(mrem/pCi ingested)

Sr-90

NUCLIDE		BONE	LIVER	TOTAL BODY	THYROID	KIDNEY	LUNG	GI-LLI
1H	3	0.0	3.07E-07	3.07E-07	3.07E-07		3.07E-07	3.07E-07
6C	14	4.81E-06	4.81E-06	4.81E-06	4.81E-06		4.81E-06	4.81E-06
11NA	22	1.00E-04	1.00E-04	1.00E-04	1.00E-04		1.00E-04	2.45E-06
27CO	58	0.0	3.78E-06	9.26E-06	0.0		0.0	9.79E-06
27CO	60	0.0	1.07E-05	2.56E-05	0.0		0.0	2.64E-05
38SR	89	2.93E-03	0.0	8.42E-05	0.0	(USE	0.0	5.48E-05
38SR	90	2.51E-02	0.0	6.40E-03	0.0		0.0	2.45E-04
39Y	90	8.97E-08	0.0	2.41E-09	0.0	ADULT	0.0	1.29E-04
39Y	91	1.25E-06	0.0	3.33E-08	0.0		0.0	8.27E-05
40ZR	95	2.11E-07	5.32E-08	3.78E-08	0.0		0.0	2.38E-05
41NB	95	3.89E-08	1.75E-08	1.03E-08	0.0	DOSE	0.0	1.40E-05
44RU	103	1.41E-06	0.0	4.85E-07	0.0		0.0	1.76E-05
44RU	106	2.54E-05	0.0	3.12E-06	0.0		0.0	1.97E-04
50SN	123	2.79E-04	4.33E-06	6.86E-06	4.33E-06	FACTOR)	0.0	6.91E-05
52TE	125M	2.43E-05	8.19E-06	3.24E-06	8.00E-06		0.0	1.17E-05
52TE	127	9.58E-07	3.19E-07	2.06E-07	7.75E-07		0.0	2.27E-05
52TE	129M	1.05E-04	3.61E-05	1.60E-05	3.95E-05		0.0	6.33E-05
52TE	132	2.13E-05	1.05E-05	9.76E-06	1.55E-05		0.0	8.08E-05
53I	129	2.95E-05	2.16E-05	7.76E-05	6.79E-02		0.0	4.46E-07
53I	131	3.42E-05	4.07E-05	2.38E-05	1.31E-02		0.0	1.53E-06
53I	133	1.26E-05	1.84E-05	5.58E-06	4.35E-03		0.0	3.27E-06
55CS	134	4.58E-04	8.24E-04	6.97E-05	0.0		9.42E-05	1.96E-06
55CS	137	6.53E-04	7.31E-04	4.20E-05	0.0		8.81E-05	1.89E-06
56BA	140	1.74E-04	1.75E-07	8.99E-06	0.0		1.07E-07	4.43E-05
57LA	140	2.12E-08	8.37E-09	2.16E-09	0.0		0.0	1.04E-04
58CE	141	8.00E-08	4.91E-08	5.75E-09	0.0		0.0	2.38E-05
58CE	144	4.49E-06	1.77E-06	2.42E-07	0.0		0.0	1.85E-04
63EU	154	4.30E-06	4.84E-07	3.29E-07	0.0		0.0	4.76E-05
92U	232	3.66E-02	0.0	2.68E-03	0.0		0.0	7.34E-05
92U	234	7.40E-03	0.0	4.71E-04	0.0		0.0	6.72E-05
94PU	238	1.71E-03	2.10E-04	4.25E-05	0.0		0.0	7.98E-05
94PU	239	1.78E-03	2.26E-04	4.41E-05	0.0		0.0	7.28E-05
94PU	240	1.78E-03	2.28E-04	4.45E-05	0.0		0.0	7.29E-05
94PU	241	1.00E-06	1.37E-07	2.70E-08	0.0		0.0	1.40E-07
95AM	241	1.93E-03	1.03E-03	1.38E-04	0.0		0.0	7.84E-05
96CM	242	1.43E-04	1.40E-04	9.49E-06	0.0		0.0	8.53E-05
96CM	244	1.64E-03	9.67E-04	1.04E-04	0.0		0.0	6.12E-05

Note: 0.0 means insufficient data or that the dose factor is <1.0E-20.

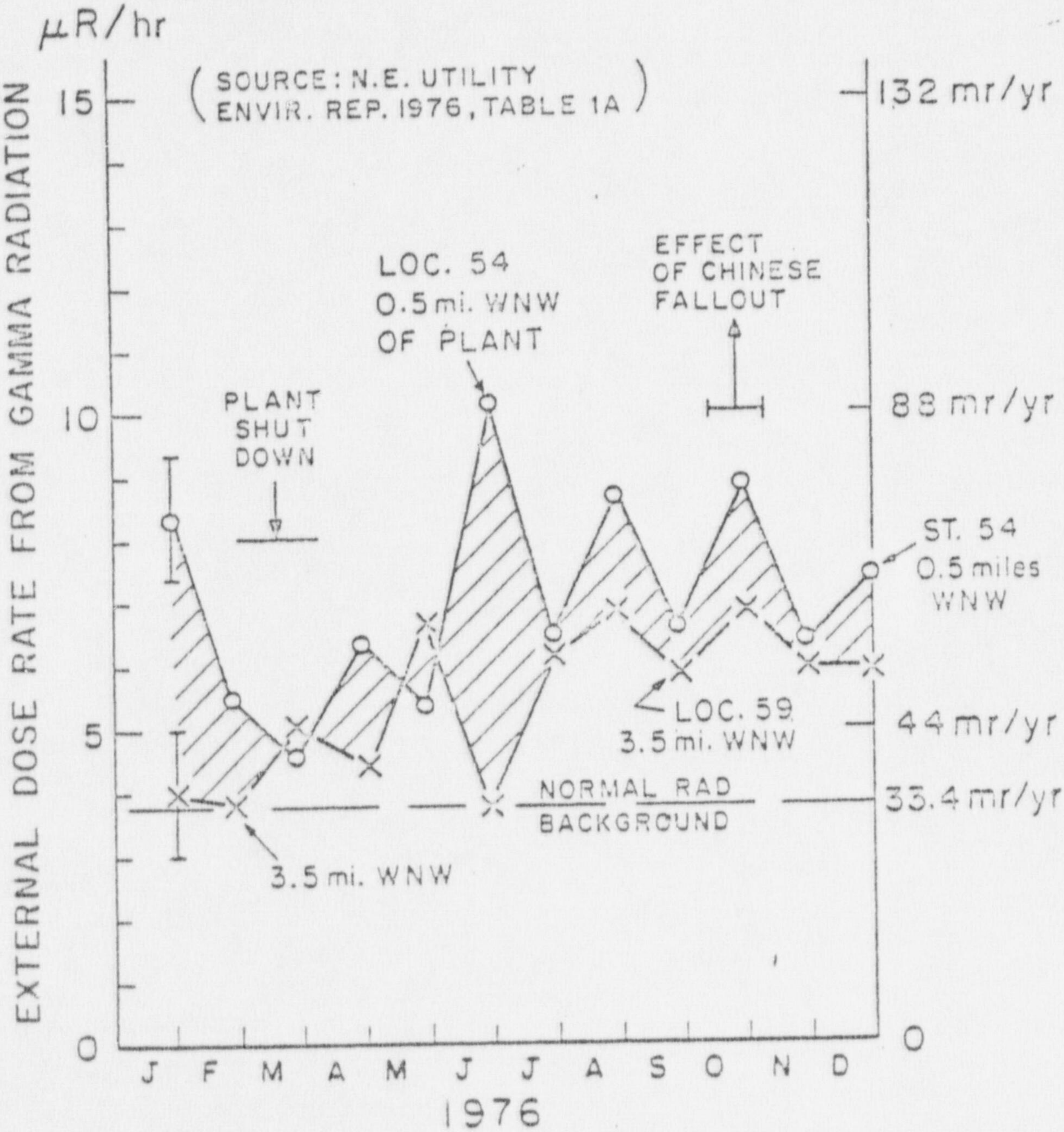
A392



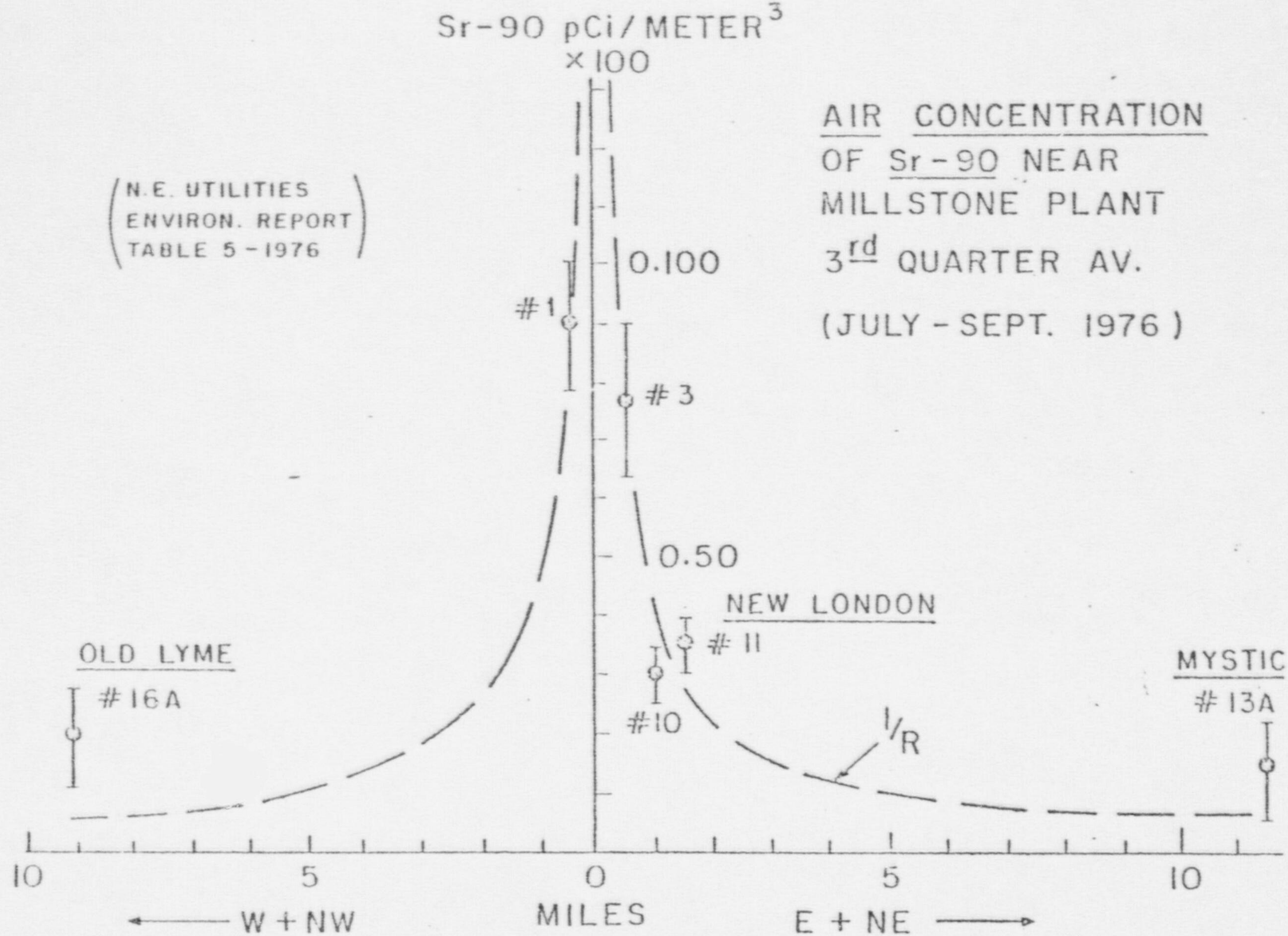
From Table 1(a), Millstone Environmental Report 1976. Note that the minimum values, which are due to the natural radiation background from radioactivity in the soil and cosmic rays are in the neighborhood of 40-50 mrem/year, not 129 mrem/yr. as stated in the utility's report (Appendix III). Furthermore, the external gamma radiation is seen to reach its peak at the same time as the measured peak of Cs-137 in milk as seen in Fig. 4 and Table 1, thus tying it to plant releases.

APPENDIX VI(b)

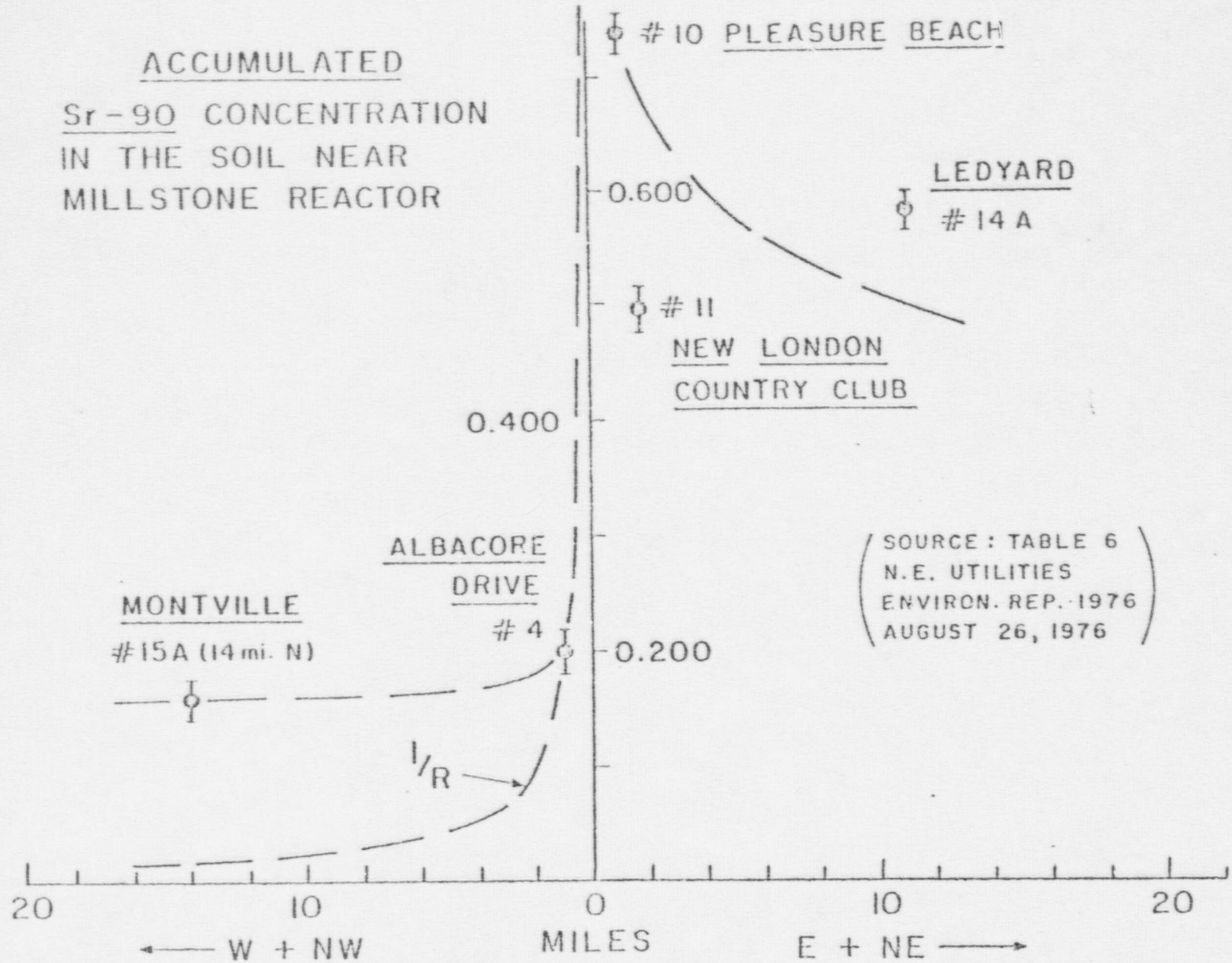
VARIATION OF EXTERNAL GAMMA DOSE RATE WITH TIME FOR VARIOUS DISTANCES FROM HADDAM NECK REACTOR - 1976



From Table 1A, Environmental Report for Haddam Neck Plant, 1976. Note that external gamma radiation is much lower near Haddam Neck than near Millstone, consistent with the lower levels of airborne Cs-137 levols in the local milk as shown in Tables 7(a) and 7(b). Again, the utility's own measurements of external gamma radiation show a level far below the level of 129 mrem per year stated as the natural background level for Connecticut, and much closer to the 1970-71 EPA measurements of 70 mrem per year.



APPENDIX VII(a)
From Environmental Report for Millstone Plant, 1976.



APPENDIX VII(b)

From Environmental Report for Millstone Plant, 1976 (Units are pCi/gm)

P.396

Table 4A

Tables of releases and release limits for Conn. Yankee (Haddam Neck), Mill., and Yankee (Rowe, Mass.) Reactors from A.E.C. Report on Releases of Radioactiv in Effluents " - 1972, Div. of Reg. Oper. Aug. 1973.

AIRBORNE EFFLUENT COMPARISON BY YEAR

Hal jens and Particulates
(Half-life greater than 8 days)

Curies

Facility	1970	1971	1972
Boiling Water Reactors			
Oyster Creek	0.32	2.14	6.48
Nine Mile Point	<0.001	<0.06	0.969
Millstone 1	-	4.0	1.32
Dresden 1	3.3	<0.67	2.75
Dresden 2,3	1.6	0.68	5.89
LaCrosse	<0.06	<0.001	<0.712
Monticello	-	0.052	0.589
Big Rock Point	0.13	0.61	0.148
Humboldt Bay	0.35	0.3	1.78
* Pilgrim	-	-	0.0319
Quad Cities 1,2	-	-	0.747
* Vermont Yankee	-	-	0.171
Pressurized Water Reactors			
* Maine Yankee	-	-	3.71×10^{-6}
Palisades	-	-	9.7 x10-3
Yankee (Rowe)	<0.001	<0.0001	7.77×10^{-4}
Indian Point 1	0.08	0.21	0.928
R.E. Ginna	0.05	0.17	0.035
Connecticut Yankee (Haddam)	0.002	0.03	0.0181
H.B. Robinson	-	None detected	0.0268
San Onofre	<0.001	<0.0001	4.74×10^{-4}
Point Beach 1,2	-	<0.0001	0.0297
* Surry 1	-	-	1.75×10^{-4}
Nonwater Reactors			
Peach Bottom 1	<0.001	<0.003	None
Fermi	-	<0.001	0.001

* Operated less than 1 year

Table 21

PLANT SUMMARY 1972

Licensee: Yankee Atomic Electric Company*

Type: Pressurized Water Reactor

Facility: Yankee Rowe

Docket No: 50-29

Licensed Power Level: 600 MWT ~150 MW(el)

Initial Criticality: 8/19/60

Cooling Water: Deerfield River

Location: 20 miles N.W. Greenfield, Massachusetts

*Owned by 11 utilities.

POWER GENERATION - Megawatt hours

Gross thermal: 2.24×10^{6}

Net electrical: 6.44×10^{5}

Effic. = 28.8%

AIRBORNE EFFLUENTS	Curies released	Percent of limit
Noble gases	1.83×10^{1}	2.63×10^{-2}
Halogens	2.33×10^{-4}	0.0607 1/
Particulates	5.44×10^{-4}	

1/Percent of limit includes halogens & particulates with half life >8 days.

LIQUID EFFLUENTS	Curies released	Average concentration (uCi/ml)	Percent of limit
Mixed fission and activation products	2.06×10^{-2}	1.28×10^{-10}	2.49×10^{-2}
Tritium	803.0	4.97×10^{-6}	1.66×10^{-1}

Volume of liquid waste: 1.13×10^{7} liters

Volume of cooling water: 1.61×10^{11} liters

SOLID WASTE - Shipped offsite

Total curies: 2.31

Total volume: 2.22×10^{2} cubic meters

Table 21A

RADIONUCLIDE SUMMARY - 1972

Facility: Yankee Rowe

LIQUID EFFLUENTS

Nuclides	Curies released	Average concentration (uCi/ml)	Percent of limit
Strontium-90	1.10×10^{-5}	6.83×10^{-14}	2.28×10^{-5}
Iodine-131	1.56×10^{-3}	9.69×10^{-12}	3.23×10^{-3}
Cesium-137	2.33×10^{-4}	1.45×10^{-12}	7.25×10^{-6}
Cesium-134	1.64×10^{-4}	1.02×10^{-12}	1.13×10^{-5}
Cobalt-60	3.3×10^{-4}	2.05×10^{-12}	6.88×10^{-6}
Chromium-51	4.3×10^{-5}	2.67×10^{-13}	1.34×10^{-8}
Manganese-54	5.08×10^{-4}	3.16×10^{-12}	3.16×10^{-6}
Cobalt-58	2.67×10^{-4}	1.66×10^{-12}	1.84×10^{-6}
Carbon-14	1.71×10^{-2}	1.06×10^{-10}	1.33×10^{-5}
Selenium-75	1.24×10^{-4}	7.70×10^{-13}	1.92×10^{-2}
Cesium-144	2.3×10^{-5}	1.43×10^{-13}	1.43×10^{-6}

AIRBORNE EFFLUENTS

Nuclides	Curies released	Nuclides	Curies released
NOBLE GASES		PARTICULATES	
Krypton-85	1.68	Cesium-137	2.0×10^{-6}
Xenon-133	1.12	Strontium-90	1.20×10^{-5}
Krypton-85m	6.0×10^{-3}	Manganese-54	8.5×10^{-5}
Xenon-135	1.94	Cobalt-60	2.23×10^{-4}
Argon-41	1.63	Cobalt-58	1.29×10^{-4}
Xenon-133m	5.4×10^{-2}	Iron-59	8.0×10^{-6}
Argon-37	1.84	Selenium-75	3.9×10^{-5}
Carbon-14	9.79×10^{-1}	Chromium-51	2.4×10^{-5}
HALOGENS			
Iodine-131	2.33×10^{-4}		

A-399

Table 24

PLANT SUMMARY 1972

Licensee: Connecticut Yankee Atomic Power Company

Type: Pressurized Water Reactor

Facility: Haddam Neck

Docket No: 50-213

Licensed Power Level: 1825 MWT

Initial Criticality: 7/24/67

Cooling Water: Connecticut River

Location: 13 miles E. Meridan, Connecticut

POWER GENERATION - Megawatt hours

Gross thermal: 1.38×10^7

Net electrical: 4.3×10^6

> EFFIC. = 31.1%

AIRBORNE EFFLUENTS	Curies released	Percent of limit	(Curies) Limit
Noble gases	6.45×10^2	2.52×10^{-1}	256,000 Ci
Halogens	1.01×10^{-2}	8.7 1/	0.12 Ci
Particulates (Sr, Cs, etc)	8.0×10^{-3}		0.09 Ci

1/Percent of limit includes halogens & particulates with half life >8 days.

LIQUID EFFLUENTS	Curies released	Average concentration (uCi/ml)	Percent of limit
Mixed fission and activation products	4.78	6.20×10^{-9}	2.33×10^{-1}
Tritium	5890.0	7.64×10^{-6}	2.55×10^{-1}

Volume of liquid waste: 3.44×10^7 liters

Volume of cooling water: 7.71×10^{11} liters

SOLID WASTE - Shipped offsite

Total curies: 4.0×10^3

Total volume: 1.07×10^2 cubic meters

A-400

Table 24A

RADIONUCLIDE SUMMARY - 1972

Facility: Connecticut Yankee

LIQUID EFFLUENTS

Nuclides	Curies released	Average concentration (uCi/ml)	Percent of limit	Permiss. Limit Curies
Iodine-131	3.01×10^{-1}	3.9×10^{-10}	1.3×10^{-1}	231
Cesium-137	7.06×10^{-1}	9.16×10^{-10}	2.29×10^{-3}	308
Cobalt-60	1.15	1.49×10^{-9}	4.96×10^{-3}	
Sr^{90} etc → Unidentified	3.75×10^{-1}	4.86×10^{-10}	1.62×10^{-2}	23.1
Iodine-133	5.75×10^{-1}	7.46×10^{-10}	7.46×10^{-2}	
Cobalt-58	9.71×10^{-1}	1.26×10^{-9}	1.4×10^{-3}	
Molybdenum-99	8.96×10^{-2}	1.16×10^{-10}	2.9×10^{-4}	
Cerium-144	2.27×10^{-1}	2.94×10^{-10}	2.94×10^{-3}	
Ruthenium-103	3.04×10^{-1}	3.94×10^{-10}	4.93×10^{-4}	

AIRBORNE EFFLUENTS

Nuclides	Curies released
NOBLE GASES	
Krypton-85	1.09×10^{2}
Xenon-133	4.94×10^{2}
Krypton-88	1.67×10^{-1}
Krypton-87	5.53×10^{-1}
Xenon-135m	1.38×10^{1}
Xenon-135	2.82×10^{1}
HALOGENS	
Iodine-131	1.0×10^{-2}
Iodine-133	6.33×10^{-5}

Nuclides	Curies released
PARTICULATES	
Rubidium-88	2.13×10^{-1}
Unidentified (Sr, Cs)	8.0×10^{-3}

A401

PLANT SUMMARY 1972

Licensee: Northeast Utilities*

Type: Boiling Water Reactor

Facility: Millstone-1

Docket No: 50-245

Licensed Power Level: 2011 MWT

Initial Criticality: 10/26/70

Cooling Water: Long Island Sound

Location: 5 miles S.W. New London, Connecticut

*Affilation of: Conn. Light & Power, Hartford Elec. Light, Western, Mass. Elec.

POWER GENERATION - Mégawatt hours

Gross thermal: 9.69×10^6

Net electrical: 3.17×10^6

> 32.7% THERMAL EFFICIENCY

AIRBORNE EFFLUENTS	Curies released	Percent of limit	Permissible Limit in Curies
Noble gases	7.26×10^5	2.91	24.9 MILLION Ci
Halogens (I-131 etc)	1.23	1.37 1/	90 Ci
Particulates (Sr^{90}, Cs^{137})	9.78×10^{-2}		7.13 Ci

1/Percent of limit includes halogen & particulate with half life >8 days.

LIQUID EFFLUENTS	Curies released	Average concentration (uCi/ml)	Percent of limit	Permiss. Limit (Curies)
Mixed fission and activation products	51.5	8.35×10^{-8}	7.04	731
Tritium	20.2	3.32×10^{-8}	1.13×10^{-3}	

Volume of liquid waste: 1.94×10^7 liters

Volume of cooling water: 6.17×10^{11} liters

SOLID WASTE - Shipped offsite

Total curies: 1.64×10^3

Total volume: 5.68×10^2 cubic meters

A-402

Table 9A

RADIONUCLIDE SUMMARY - 1972

Facility: Millstone

LIQUID EFFLUENTS

Nuclides	Curies released	Permiss Limit (Curies)	Average concentration (uCi/ml)	Percent of limit
Strontium-90	0.216	185	3.5×10^{-10}	1.17×10^{-1}
Cesium-137	16.4	12330	2.66×10^{-8}	1.33×10^{-1}
Iodine-131	11.4	185	1.85×10^{-8}	6.17
Cesium-134	8.84	5560	1.43×10^{-8}	1.59×10^{-1}
Cobalt-60	2.83	308	4.59×10^{-9}	9.18×10^{-3}
Chromium-51	0.854		1.38×10^{-9}	6.9×10^{-5}
Manganese-54	1.09		1.77×10^{-9}	1.77×10^{-3}
Strontium-89	1.30		2.11×10^{-9}	7.03×10^{-2}
Cobalt-58	1.35		2.19×10^{-9}	2.19×10^{-3}
Yttrium-90	0.216		3.5×10^{-10}	1.75×10^{-3}
Unidentified	6.9		1.12×10^{-8}	3.73×10^{-1}
Iron-59	0.109		1.77×10^{-10}	2.95×10^{-4}

AIRBORNE EFFLUENTS

Nuclides	Curies released
NOBLE GASES	
Xenon-133	1.89×10^{5}
Krypton-88	1.16×10^{5}
Krypton-87	7.70×10^{4}
Krypton-85m	4.14×10^{4}
Xenon-138	4.19×10^{4}
Krypton-135	2.07×10^{5}
Other Gases	4.88×10^{4}
HALOGENS	
Iodine-131	1.23 Ci

Nuclides	Curies released
PARTICULATES	
Barium-Lanthanum-140	5.56×10^{-2}
Strontium-90	5.74×10^{-3}
Strontium-89	1.16×10^{-2}
Yttrium-90	5.74×10^{-3}
Cobalt-60	4.22×10^{-3}
Cobalt-58	3.79×10^{-3}
Manganese-54	1.05×10^{-4}
Cesium-134	2.40×10^{-5}
Cesium-137	3.26×10^{-4}
Iodine-131	1.76×10^{-3}
Unidentified	8.86×10^{-3}

March 22, 1978

Dr. Ernest A. Sternglass
Department of Radiology
University of Pittsburgh
Pittsburgh, PA 15261

Dear Dr. Sternglass:

The information you recently provided Dr. Dade Moeller, ACRS Member, dealing with strontium-90 levels and cancer mortality changes around the Haddam Neck and Millstone reactors has been provided to all members of the Committee for their consideration and any action which they may consider appropriate.

Sincerely,

R F Fraley

R. F. Fraley
Executive Director

cc:
K. Z. Morgan, Georgia
Institute of Technology

ATT. 2

A-404

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

March 17, 1978

ACRS Members

REQUEST FOR ACRS REVIEW OF CANCER MORTALITY CHANGES AROUND THE HADDAM NECK AND MILLSTONE REACTORS

The attached material is provided for your consideration. A portion of the 216th ACRS meeting will be set aside to discuss an appropriate course of action regarding this request.

R. F. Fraley

R. F. Fraley
Executive Director

Attachments:
1. Ltr., Sternglass to Moeller dtd. 3/3/78, w/enclosure) SEE ATTACHMENT 1
2. Paper by Sternglass, "Strontium - 90 Levels in the Milk & Diet Near Connecticut Nuclear Power Plants" dtd. 10/27/77) SEE ATTACHMENT 1
3. Ltr., Morgan to Sternglass dtd. 2/14/78 - SEE ATTACHMENT 4

cc:
H. H. E. Plaine, w/att
M. W. Libarkin, w/att
T. G. McCreless, w/att.
R. Muller, w/att.

ATT. 3

A405

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

March 22, 1978

Dr. Albert W. Hilberg
Division of Medical Sciences
National Academy of Sciences
2101 Constitution Avenue, NW
Washington, DC 20418

Dear Dr. Hilberg:

I understand that the BEIR Committee is currently working on Report No. III regarding the effects of low-level ionizing radiation and, in connection with this report, is reviewing the work of several individuals, including the work of Dr. Ernest J. Sternglass.

Dr. Sternglass has provided the attached information for consideration of the Advisory Committee on Reactor Safeguards as indicated in the attached letter to Dr. Dade W. Moeller. In view of the interest and activity of the BEIR Committee in this area, copies are being provided for your consideration and use.

Sincerely yours,

R. F. Fraley

R. F. Fraley
Executive Director

Attachments:

1. Ltr. to D. W. Moeller frm. E. Sternglass dtd 3/3/78) SEE ATTACHMENT 1
2. Rpt. by E. Sternglass, "Strontium-90 Levels in the Milk & Diet Near Conn. Nuclear Power Plants") SEE ATTACHMENT 1
3. Ltr. to Sternglass frm. K. Z. Morgan - SEE ATTACHMENT 4 dtd. 2/14/78

cc: w/o atts.
Dr. Stephen Lawroski, ACRS
Dr. Dade Moeller, ACRS
Dr. Ernest J. Sternglass, Univ. of Pittsburgh

ATT. 4

A-406

See MHS 01-022

SCHOOL OF NUCLEAR ENGINEERING
Georgia Institute of Technology

Atlanta, Georgia 30332 (404) 894-372[illegible]

[illegible] 9 [illegible] 4 47 February 14, 1978

U.S.N[illegible] [illegible]
ADVISORY [illegible] [illegible]EE ON
REACTOR SA[illegible]EGUARDS

Dr. Ernest A. Sternglass
R 406 Scaife Hall, Rad Ctr.
University of Pittsburgh
Pittsburgh, Pennsylvania 15261

Dear Ernest:

I was very interested in the information you gave yesterday (February 10, 1978) at the Conference on Effects of Low-Level Radiation in the Dirksen Senate Building. I think this information in reference to the large increase in Cancer Mortality Rate around Nuclear Facilities in Connecticut and the increase in Respiratory Cancer in the Denver area are evidence for need of pause for concern. These increases seem to relate to the operation of these plants (i.e. Millstone Nuclear Power Plant and the Rocky Flats plant in Colorado). I think this is sufficient evidence to demand further study but not to push the panic button. By not pushing the panic button I mean persons should not be encourgaged to move their homes immediately. By demand further study I do not mean the NRC should try to discredit you or your findings or try to disprove your investigations as our Energy Agencies have done in the past. More specifically, I do not think someone like Dr. E. A. Tompkins should again be employed to show why your data are meaningless. I felt when she attempted this before, she made some of the same errors of which she accused you and did not add much to the credibility of the Agency she represented. More specifically, I think NRC should work closely with you, give you some financial support if you require it, extend the study to more countries and townships, make all appropriate adjustments of the data and see if in fact there is a correlation between radiation exposure from these plants and the observed increase in malignancies. Following such a study I think the findings should be reported to the Advisory Committee on Reactor Safeguards of the NRC. I believe the ACRS wishes to get at the facts, extend and try to explain data such as you have collected. This must be done an impartial way. I have great respect for some members of ACRS and especially my long time friend, Dr. Dade Moeller of the ACRS. With this in mind, I am suggesting you send him a copy of your Congressional testimony of February 10, 1978 and other related information so he can judge for himself its significance and need for action on the part of the Environmental Subcommittee of ACRS.

ATT. 5

A-407

Dade's address is:

Dr. Dade W. Moeller
27 Wildwood Drive
Bedford, Maryland 01730

Sincerely,

KZM

Karl Z. Morgan
Neely Professor

KZM:rs

cc Moeller

This ltr. was provided by D. W. Moeller for Committee info. He does not feel that any follow-up ACRS action is required. - Rff.

A-408

NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

FEB 2 1 1978

MEMORANDUM FOR: Robert B. Minogue, Director
Office of Standards Development

FROM: Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation

SUBJECT: CONGRESSIONAL REVIEW REQUEST: CORRELATION BETWEEN ROCKY FLATS' PLUTONIUM POLLUTION AND INCREASES IN CANCER

In response to your memorandum of February 13, 1978, the paper by S. Davis and E. Sternglass regarding the above subject was quickly reviewed by Dr. F. Congel, RAB. The conclusion that there is a correlation between Pu levels in the environs of the Rocky Flats Nuclear Weapons Plant and the respiratory cancer mortality rates in the surrounding counties is unsubstantiated.

Based on the U.S. Department of Health, Education and Welfare publication, "Cancer Rates and Risks", 2nd edition, 1974, the lung cancer mortality rate for the U.S. population was about 12 per 100,000 people (20/100,000 for men; 4/100,000 for women) in 1952. In 1970, the rate was about 30 per 100,000 people (48/100,000 for men; 10/100,000 for women). Comparing these rates to the rates presented in the Davis-Sternglass paper, Graph I, the following observations are made.

> The lung cancer mortality rates for Adams, Arapahoe, and Boulder counties are less than the U.S. population rates for corresponding time periods. For Denver county, the 1952 rate is greater than the U.S. population rate whereas the 1970 rate is about the same as the U.S. population rate. The Jefferson county rate in 1952 is slightly greater than the U.S. population rate but is significantly less than the U.S. population rate in 1970.

A-409

Therefore, based on national statistics alone, only the Denver metropolitan area has a lung cancer mortality rate that is comparable to the U.S. population rate. All other counties considered in the Davis-Sternglass paper have lower lung cancer rates than the U.S. population as a whole. In addition, while the lung cancer rate for the nation has increased by a factor of 2.5 from 1952 to 1970, all counties considered in the Davis-Sternglass paper showed a rate increase of less than a factor of 2.

Therefore, the cause-effect correlation between increased plutonium in the Rocky Flats environs and the increased lung cancer rates in the surrounding population is unsubstantiated. Many factors which can lead to the observed cancer mortality rate increase are presently under investigation. The most probable causes include increased cigarette consumption and increased concentrations of known carcinogens in the air which may well be important considerations in the Denver area.

The information given above has been communicated to John Hickey and Steve Whitfield by phone on February 15, 1978.

Original Signed By
E. G. Case

Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation

cc: S. Levine
C. V. Smith
H. Denton
E. Volgenau
R. Vollmer
D. Muller
W. Kreger
B. Grimes
F. Congel
C. Roberts
J. Kastner
S. Hanauer

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A-410

OFFICE	RAB:DSE FJC	RAB:DSE	DSE:SA&AS	DSE	NRR	
SURNAME	FJCongel:pc	WEKreger	RHVollmer	HRDenton	EGCase	
DATE	2/15/78	2/ /78	2/15/78	2/10/78	2/ /78	

UNITED STATES
..UCLEAR REGULATORY COMMISSIUN
DVISORY COMMITTEE ON REACTOR SAFEGUARDS
WASHINGTON, D. C. 20555

April 10, 1978

APPENDIX XXXIV
Letter to E. J. Sternglass

Ernest J. Sternglass, Ph.D.
Professor of Radiological Physics
School of Medicine
University of Pittsburgh
Pittsburgh, PA 15261

Dear Dr. Sternglass:

This is in response to your letter of March 3, 1978, by which you provided copies of your reports, "Cancer Mortality Changes Around Nuclear Facilities in Connecticut" and "Strontium-90 Levels in the Milk and Diet Near Connecticut Nuclear Power Plants" for review by the ACRS.

It is our understanding that the Committee on the Biological Effects of Ionizing Radiation (the BEIR Committee) is currently updating its 1972 report, under the auspices of the National Academy of Sciences - National Research Council. This Committee, whose report is scheduled to be issued by the end of this year, is conducting a detailed review of the data available on the effects of low level ionizing radiation, including a number of your reports. We have forwarded the two reports you provided the ACRS to the BEIR Committee to ensure that they are available to them for consideration. Since the BEIR Committee includes a number of recognized experts in radiobiology, epidemiology, etc., we believe their review and report should provide the indepth assessment of your work that you are seeking.

Sincerely yours,

Stephen Lawroski

Stephen Lawroski
Chairman

Attachments:
1) R. F. Fraley, ACRS, letter to A. W. Hilberg, NAS, dated March 22, 1978 w/o attmts.
2) A. W. Hilberg, NAS, letter to R. F. Fraley, ACRS, dated March 28, 1978

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

April 11, 1978

APPENDIX XXXV
Regulatory Guides, ACRS Actions

Mr. Lee V. Gossick
Executive Director for Operations
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REGULATORY GUIDES - ACRS ACTION

Dear Mr. Gossick:

During its 216th meeting, April 6 and 7, 1978, the ACRS approved the following Regulatory Guides:

1. Regulatory Guide 1.68, Revision 2, "Initial Test Program for Water-Cooled Nuclear Power Plants," and

2. Regulatory Guide 1.29, Revision 3, "Seismic Design Classification."

Sincerely yours,

Stephen Lawroski

Stephen Lawroski
Chairman

cc: E. G. Case, NRR
R. Minogue, OSD
G. Arlotto, OSD
S. J. Chilk, SECY

bcc: ACRS Members
J. Jacobs
H. Voress

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

April 18, 1978

APPENDIX XXXVI
Report on ACRS Activities, Dec. 1977 - April 1978

Honorable Joseph M. Hendrie
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON ACRS ACTIVITIES: DECEMBER 1977 THROUGH APRIL 1978

Dear Dr. Hendrie:

This is a brief report of ACRS activities during the period December 1977 - April 1978. Selected topics in this report will be discussed during the next joint NRC-ACRS meeting.

Review of Operations at "stretch" power

During the August, September, and October 1977 ACRS meetings, the NRC Staff discussed with the Committee their plans for approving stretch power operation at the Calvert Cliffs, Palisades, and Maine Yankee nuclear plants.

On November 9, 1977, the Committee requested additional information regarding the scope and nature of the Staff's reviews in this area in order to develop a mechanism for decisions concerning ACRS involvement in similar future reviews. The Director, ONRR responded by memo dated December 21, 1977. A meeting of the ACRS Subcommittee on the Safety of Operating Reactors was held on March 22, 1978, to develop additional information on this subject; the Subcommittee plans to recommend an appropriate course of action to the full Committee.

ACRS Comments Concerning the Establishment of a Statutorily Independent, Quasi-Judicial Board for Evaluation of Reactor Accidents

Congressman Udall has asked for the ACRS' views and recommendations regarding a proposal by Dr. Hal Lewis that a statutory Board, similar to the National Transportation Safety Board, be established to review reactor accidents, determine probable cause, and recommend corrective action to preclude reoccurrence. Thus far, members of the Committee Staff have visited the NTSB to discuss their procedures, a Subcommittee has met to consider the matter, and informal discussions have been held with Dr. Lewis. The Committee will continue to consider its reply, which has been requested by July 31.

New Members

The Commission issued a press release on December 14, 1977 inviting members of the public to nominate individuals to fill two anticipated vacancies on the Committee. In response to the press release, forty-six persons were nominated by various individuals and organizations. The Committee has considered these nominees and is forwarding its recommendations to you separately.

Planned Meetings with FRG Reactor Safety Committee and French Groupe Permanent

The Committee has made preliminary arrangements to meet with members of the Groupe Permanent of France and the FRG Reactor Safety Committee (RSK) in the United States during September and November to discuss matters related to reactor safety.

Members of the Committee have met on previous occasions with representatives of the RSK and the French regulatory organizations including the Groupe Permanent and have found these sessions very informative.

Proposed Use of "Class 9" Accidents in Site Comparisons

The Committee has received a copy of a proposal (SECY 78-137) to use assessment of the relative differences in Class 9 accident risks as an element in the comparison of sites when current criteria call for special consideration of alternative sites in environmental reviews. Since such an approach may have potential impacts on safety reviews, the Committee plans to discuss this matter with the Staff.

ACRS Fellowship Program

Section 6 of the NRC's FY 1978 Authorization Act (PL 95-209) directs the Committee to establish a fellowship program. We have been informed that a request to reprogram funds to initiate the fellowship program has not been approved by the House Committee on Appropriations. The Committee needs guidance regarding compliance with this provision of P.L. 95-209.

Recent Papers by Dr. Ernest Sternglass

Dr. Ernest Sternglass of the University of Pittsburgh forwarded, to an ACRS member, two papers dealing with Strontium-90 levels and changes in cancer mortality near the Haddem Neck and Millstone reactors with a request that the observations reported in these papers be reviewed by the ACRS. These papers were transmitted to the National Academy of Sciences - National Research Council BEIR Committee for their use in their current reevaluation of the effects of low-level ionizing radiation.

A-414

Dynamic Load Combinations

As a result of discussions during several recent ACRS project reviews, the NRC Staff is preparing a study of the rationale and methods for combining or not combining various dynamic loads such as LOCA and seismic loads which are imposed on important nuclear plant structures, systems, and components.

Future Schedule

217th ACRS MEETING
MAY 4-6, 1978

Maine Yankee (Pwr. Increase)

ONRR Safety Evaluation Report	on hand
ACRS Subcommittee Meeting	5/2/78
ACRS Report	5/11/78

Indian Point 3 (Pwr. Increase)

ONRR Safety Evaluation Report	on hand*
ACRS Subcommittee Meeting	4/24/78
ACRS Report	5/11/78

218th ACRS MEETING
JUNE 1-3, 1978

New England 1 & 2 (CP)

ONRR Safety Evaluation Report	5/1/78
ACRS Subcommittee Meeting	5/17-18/78
ACRS Report	6/8/78

Diablo Canyon 1 & 2 (OL)

ONRR Safety Evaluation Report	5/1/78
ACRS Subcommittee Meeting	5/24-25/78
ACRS Report	6/8/78

*A Supplement including revised ECCS calculations is to be provided.

A-415

<u>Davis Besse 2 & 3 (CP)</u>

ONRR Safety Evaluation Report	5/1/78
ACRS Subcommittee Meeting	5/18/78
ACRS Report	6/8/78

Sincerely yours,

Stephen Lawroski

Stephen Lawroski
Chairman

A-416

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

April 12, 1978

Honorable Joseph M. Hendrie
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON ARKANSAS NUCLEAR ONE, UNIT 2 NUCLEAR POWER PLANT

Dear Dr. Hendrie:

During its 216th meeting, April 6 and 7, 1978, the Advisory Committee on Reactor Safeguards completed its review of the application of Arkansas Power and Light Company (Applicant) for a permit to operate the Arkansas Nuclear One, Unit 2 Nuclear Power Plant (ANO-2). The application was also considered at the 214th ACRS meeting, February 9-11, 1978, and was reviewed at Subcommittee meetings on June 24, 1977 in Russellville, Arkansas and February 2 and March 20, 1978 in Washington, DC. Subcommittee meetings were also held on February 28, 1975 and May 20, 1977 in Windsor, Connecticut and on June 30, 1977 and March 20, 1978 in Washington, DC to review the Combustion Engineering designed Core Protection Calculator System (CPCS) which will be employed on ANO-2. A tour of the ANO-2 facility was made by Subcommittee members on June 24, 1977. During its review, the Committee had the benefit of discussions with representatives and consultants of the Applicant, Combustion Engineering, Inc. (CE), Bechtel Corporation, and the Nuclear Regulatory Commission (NRC) Staff. The Committee also had the benefit of the documents listed.

ANO-2 is the second nuclear unit constructed on the Arkansas Nuclear One site which is located on the Arkansas River in Pope County, Arkansas about six miles from the city of Russellville. The two units differ in that Unit 1 utilizes a Babcock and Wilcox Nuclear Steam Supply System (NSSS) which was licensed on May 21, 1974 to operate at 2568 MWt, while Unit 2 is a CE NSSS for which a license to operate at 2815 MWt is sought. The Committee reported on the construction permit application for ANO-2 in its letter of February 10, 1972.

The ANO-2 NSSS is similar to the Calvert Cliffs 1 and 2 and St. Lucie 1 nuclear units which are now operating; however, ANO-2 will be the first reactor to use CE 16 x 16 fuel assemblies. The NRC Staff concluded that the Applicant has acceptably established the basis for this new fuel design. The Committee agrees with this conclusion. The NRC Staff will require that the

Applicant conduct a surveillance program on the new fuel as it is removed from the core. The Committee wishes to be kept informed of the results of this program (Generic Item IIB-2 in ACRS Report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 6," dated November 15, 1977).

The Applicant proposes to make use of the CPCS as part of the reactor protection system. The CPCS consists of four redundant digital computers which acquire data from plant process sensors and from two redundant, computer-based control element assembly calculators which provide control rod position information. This application of the CPCS will mark the first use in a United States power reactor of an online digital computer as part of the reactor protection system. The Applicant has developed an extensive series of tests for determining proper operation of both the hardware and the software that make up the system. The NRC Staff has concluded that, subject to resolution of several issues which appear to have available solutions, the CPCS is acceptable (Generic Item IIB-1 in ACRS Report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 6," dated November 15, 1977).

The NRC Staff has identified six CPCS and a number of other safety related items which remain outstanding. These matters should be resolved in a manner satisfactory to the NRC Staff. The Committee wishes to be kept informed.

Various generic problems are discussed in the Committee's report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 6," dated November 15, 1977. Those problems relevant to the Arkansas Nuclear One, Unit 2 Nuclear Power Plant should be dealt with by the NRC Staff and the Applicant as solutions are found. The relevant items are: II-1, 2, 3, 4, 5B, 6, 7, 10; IIA-2, 3, 4; IIC-1, 3A, 3B, 4, 5, 6; IID-2.

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 Arkansas Nuclear One, Unit 2 Nuclear Power Plant can be operated at core power levels up to 2815 MWt without undue risk to the health and safety of the public.

Sincerely yours,

Stephen Lawroski

Stephen Lawroski
Chairman

Additional Comments by Member William Kerr

I urge the NRC Staff to reconsider its decision to require the Applicant to disconnect the data links from the Core Protection Calculator System to the Plant Computer following initial startup and subsequent refueling startups. The additional information which can be provided by the use of these links could enhance the reliability of both the protection system and of plant control. I find the Staff's arguments against the use of these links unconvincing.

REFERENCES:

1. U.S. Nuclear Regulatory Commission, "Supplement No. 1 to the Safety Evaluation Report (USNRC Report NUREG-0308) by the Office of Nuclear Reactor Regulation in the Matter of Arkansas Power and Light Company Operation of Arkansas Nuclear One, Unit 2," Docket No. 50-368, March 6, 1978.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report by the Office of Nuclear Reactor Regulation Related to the Arkansas Power and Light Company Operation of Arkansas Nuclear One, Unit 2 Nuclear Power Plant, Docket No. 50-368," USNRC Report NUREG-0308, November, 1977.
3. Arkansas Power and Light Company (AP&L Co.), "Arkansas Nuclear One, Unit 2 Nuclear Power Plant Final Safety Analysis Report" with Amendments 1-44.
4. Letter from D. H. Williams, Manager of Licensing, AP&L Co., to J. F. Stolz, Chief, Light Water Reactors Branch No. 1, concerning seismic qualification of a process protective cabinet, dated January 24, 1978.
5. Letter from D. H. Williams, Manager of Licensing, AP&L Co., to E. M. Howard, Director, Office of Inspection and Enforcement (I&E), Region IV, concerning cracking of pump support columns for low pressure safety injection pumps, dated January 16, 1978.
6. Letter from D. A. Rueter, Director of Technical and Environmental Services (TES), AP&L Co., to E. M. Howard, Director, Office of I&E, Region IV, concerning emergency feedwater pump piping, dated November 18, 1977.
7. Letter from D. A. Rueter, Director of TES, AP&L Co., to E. M. Howard, Director, Office of I&E, Region IV, concerning valve motor operators, dated November 7, 1977.
8. Letter from D. A. Rueter, Director of TES, AP&L Co., to E. M. Howard, Director, Office of I&E, Region IV, concerning control room emergency chillers, dated October 17, 1977.
9. Letter from D. A. Rueter, Director of TES, AP&L Co., to E. M. Howard, Director, Office of I&E, Region IV, concerning high pressure safety injection pump flow rates, dated September 30, 1977.

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

April 12, 1978

APPENDIX XXXVIII
Report on McGuire, Units 1 and 2

Honorable Joseph M. Hendrie
Chairman
U. S. Nuclear Regulatory Commission
Washington, DC 20555

SUBJECT: REPORT ON MCGUIRE NUCLEAR STATION, UNITS 1 AND 2

Dear Dr. Hendrie:

During its 216th meeting, April 6 and 7, 1978, the Advisory Committee on Reactor Safeguards completed its review of the application of the Duke Power Company (the Applicant) for a permit to operate the McGuire Nuclear Station, Units 1 and 2. The application was reviewed at a Subcommittee meeting in Charlotte, North Carolina on March 29-30, 1978, and tours of the facility were made on May 17, 1977 and March 28, 1978. During its review, the Committee had the benefit of discussions with representatives and consultants of the Nuclear Regulatory Commission (NRC) Staff, Westinghouse Electric Corporation, and the Applicant. The Committee also had the benefit of the documents listed. The Committee reported on the application for a construction permit for the McGuire Nuclear Station on October 9, 1971.

The McGuire Nuclear Station is located on the southern shore of Lake Norman in Mecklenburg County, North Carolina, about 17 miles north-northwest of Charlotte, North Carolina. Each unit will utilize a four loop pressurized water reactor nuclear steam supply system having an initial power level of 3411 MWt. Each unit employs an ice condenser system enclosed within a free-standing steel containment vessel which is surrounded by a reinforced concrete shield building. The ice condenser system design is similar to that used for the previously reviewed Donald C. Cook Nuclear Plant, but the Applicant has modified the ice condenser system as a result of operating experience gained in the Donald C. Cook Nuclear Plant. The Applicant and the NRC Staff should make plans to monitor the performance of the ice condenser containments at the McGuire Nuclear Station (Generic Item IIA-1 in ACRS Report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 6," dated November 15, 1977).

The McGuire Nuclear Station will utilize 17x17 fuel assemblies. A surveillance program has been developed by the NRC Staff to follow the behavior of these assemblies, and data are being obtained from several plants now in operation which use them. Experience to date has been satisfactory. The Committee wishes to be kept informed of the results of the various 17x17 fuel assembly inspections and test programs now underway (Generic Item IIB-2 in ACRS Report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 6," dated November 15, 1977).

The Emergency Core Cooling Systems (ECCS) for the McGuire Nuclear Station incorporate the Upper Head Injection (UHI) system. The NRC Staff has completed its review of the Westinghouse Electric Corporation ECCS evaluation model for plants equipped with UHI, and the Committee concurs in the Staff's conclusions. The application of the approved model to McGuire should be made in accordance with the Staff's requirements.

The NRC Staff has identified a number of outstanding issues that will require resolution before the issuance of an operating license. These issues should be resolved in a manner satisfactory to the NRC Staff.

Various generic problems are discussed in the Committee's report, "Status of Generic Items Relating to Light-Water Reactors: Report No. 6," dated November 15, 1977. Those problems relevant to the McGuire Nuclear Station should be dealt with by the NRC Staff and the Applicant as solutions are found. The relevant items are: II-2, 3, 4, 5b, 6, 7; IIA-2, 3, 4; IIC-1, 3a, 3b, 5, 6; and IID-2.

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 McGuire Nuclear Station, Units 1 and 2 can be operated at power levels up to 3411 MWt without undue risk to the health and safety of the public.

Sincerely yours,

Stephen Lawroski

Stephen Lawroski
Chairman

REFERENCES:

1. Duke Power Company, "McGuire Nuclear Station, Units 1 and 2 Final Safety Analysis Report," with Amendments 1-48.
2. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report Related to the Operation of McGuire Nuclear Station, Units 1 and 2," USNRC Report NUREG-0422, March, 1978.
3. U.S. Nuclear Regulatory Commission, "Safety Evaluation Report on Westinghouse Electric Company ECCS Evaluation Model for Plants Equipped with Upper Head Injection," April, 1978.
4. Letter from J. L. Riley, Carolina Environmental Study Group (CESG), to the Advisory Committee on Reactor Safeguards, concerning reactor pressure vessel head bolts, dated March 6, 1977.
5. Letter from W. L. Porter, Duke Power Company, to J. L. Riley, CESG, concerning reactor pressure vessel head bolt test data, dated October 4, 1972.

A422

APPENDIX XXXIX
Additional Documents
Provided for ACRS' Use

1. Minutes of the Seismic Activity Subcommittee Meeting, Jan. 27-8, 1978, Washington, DC.

2. Memorandum, R. F. Fraly to B.C. Rusche, ACRS Report on Davis-Besse Nuclear Power Station, Unit 1, dtd. Jan. 14, 1977.

3. Letter, J. Allen, North Anna Environmental Coalition, to R. Muller, ACRS Staff, regarding asymmetric loads on pressure vessel structures, dtd. Feb. 18, 1978.

4. Letter, J. Allen to E.G. Case, Defective Pumps, dtd. Feb. 16, 1978.

5. Franklin Institute Research Laboratories, Interim Report I-C4732-02-1, FIRL General Preliminary Comments of Jan. 23, 1978 on Matters Relating to the Acceptability and Reliability of the VEPCO (North Anna) Low-Head, Safety Injection (LHSI) and Containment Spray (CS) Pumps to Satisfactorily Perform their Intended Functions, dtd. Feb. 2, 1978.

6. Letter, North Anna Environmental Coalition to ACRS, Continuing Settlement Concerns at North Anna's Service Water Pumphouse.

7. Memorandum, R. S. Boyd to D. M. Crutchfield, Comments on Congressman Udall's January 28, 1978 Letter.

8. Memorandum, E. G. Case to NRC Commissioners, SECY 78-137, Assessments of Relative Differences in Class 9 Accident Risks in Evaluations of Alternatives to Sites with High Population Densities, dtd. Mar. 7, 1978.

9. Letter, R. J. Mattson to C. Eicheldinger, Westinghouse Electric Corp., re. Proposed semiscale MOD3 experiments, dtd. Mar. 28, 1978.

10. Minutes of the McGuire Nuclear Station Subcommittee Meeting, Mar. 29-30, 1978, Charlotte, NC.

11. Report of Visit to Japan on Nov. 13-23 of S. Lawroski and M. S. Plesset, dtd. Mar. 30, 1978.

12. Combustion Engineering, Inc., CEN-82-P, Reactor Operation with Guide Tube Wear, dtd. Feb. 3, 1978. PROPRIETARY

13. Combustion Engineering, Inc., CEN-83(B)-P, Calvert Cliffs Unit #1 Reactor Operation with Modified CEA Guide Tubes, dtd. Feb. 8, 1978. PROPRIETARY

14. Combustion Engineering, Inc., Amendment 1 to CEN-79-P, CEN-80(N)-P, CEN-82-P, and CEN-83(B)-P, Responses to Questions from the Nuclear Regulatory Commission on CEA Guide Tube Wear, dtd. Mar. 1, 1978. PROPRIETARY

15. Combustion Engineering, Inc., Amendment 2-P to CEN-79-P, CEN-80(N)-P, CEN-82-P, and CEN-83(B)-P, Additional Information on Guide Tube Wear, dtd. Mar. 8, 1978. PROPRIETARY