Reg. Cent.

YELLOW

January 16, 1979

1/16/79

Edward Luton, Esq., Chairman
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. Franklin C. Daiber
College of Marine Studies
University of Delaware
Newark, Delaware 19711

Dr. Emmeth A. Luebke
Atomic Safety and Licensing Board
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

In the Matter of
ROCHESTER GAS & ELECTRIC CORPORATION
(R. E. Ginna Nuclear Power Plant, Unit No. 1)
Docket No. 50-244

Gentlemen:

Copies of NUREG/CR-0400, "Risk Assessment Review Group Report to the U. S. Nuclear Regulatory Commission" (the "Lewis Committee Report"), have been furnished directly to the Licensing and Appeal Board panels for the use of the members of this Board. Under cover of copies of this letter, the NRC Staff is enclosing copies of the Lewis Committee Report for the information of the parties to this proceeding. The Commission is presently in the process of developing a policy statement concerning the report.

Sincerely,

Edward G. Ketchen
Counsel for NRC Staff

cc (w/encl.):

Leonard M. Trosten, Esq.
Mr. Michael Slade
Rochester Committee for Scientific
Information
Jeffrey Cohen, Esq.
Warren B. Rosenbaum, Esq.

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DATE➤						

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June 7, 1978

Edward Luton, Esq., Chairman
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dr. Emmeth A. Luebke
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
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Dr. Franklin C. Daiber
College of Marine Studies
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Newark, Delaware 19711

In the Matter of
Rochester Gas & Electric Corporation
(R. E. Ginna Nuclear Power Plant, Unit No. 1)
Docket No. 50-244

Gentlemen:

Concerns have been raised about the adequacy of neutron dosimetry now being used at nuclear power plants. The subject of neutron dosimetry has been discussed with the Advisory Committee on Reactor Safeguards and the Commission. The enclosed information which is being provided to Licensing Boards and Appeal Boards in pending cases may have relevance to questions relating to occupational exposure in commercial nuclear reactor power facilities.

Sincerely,

Edward G. Ketchen
Counsel for NRC Staff

Enclosure: As Stated in the Attachment

cc w/encl: Leonard M. Trosten, Esq.
Mr. Michael Slade
Robert E. Lee, Ph.D.
Jeffrey Cohen, Esq.
Warren B. Rosenbaum, Esq.
Atomic Safety and Licensing Board Panel
Atomic Safety and Licensing Appeal Board
Docketing and Service Section

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

June 7, 1978

Edward Luton, Esq., Chairman
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

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DOCKETED USNRC JUN 8 1978 Office of the Secretary Docketing & Service Section

In the Matter of
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(R. E. Ginna Nuclear Power Plant, Unit No. 1)
Docket No. 50-244

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Atomic Safety and Licensing Board Panel
Atomic Safety and Licensing Appeal Board
Docketing and Service Section

ATTACHMENT

1. Letter to Mrs. P. M. Allen, North Anna Environmental Coalition, from Brian K. Grimes, Assistant Director for Engineering and Projects, Division of Operating Reactors, and Enclosures 1 and 2.

2. Documents Transmitted to NRR by SD:OHSB Following Receipt of Copy of Letter from Representative John D. Dingell to NRC Chairman Joseph M. Hendrie, FROM: Glenn W. Zimmer, Senior Health Physicist, SD:OHSB. (Only documents circled in index and enclosures to those documents are included.)

3. MEMORANDUM FOR: Director, Office of Standards Development, FROM: G. W. Zimmer, Occupational Health Standards Branch and enclosures.

4. Preliminary Value - Impact Assessment For Task Initiation to Develop an NRC Staff Technical Position on Neutron Quality Factors, and REFERENCES: (1) T. D. Jones, "Radiation Insult to the Active Bone Marrow as Predicted by a Method of CHORD's," Oak Ridge National Laboratory Report, ORNL-TM-5337, 1976; (2) Harald H. Rossi, "The Effects of Small Doses of Ionizing Radiation: Fundamental Biophysical Characteristics," presented at the joint annual meeting of the Health Physics Society/Radiation Research Society," San Francisco, June 29, 1976, and enclosures.

5. Review of NCRP recommendations to date, with enclosures.

6. Origin of Current NRC limits, with enclosures.

7. Preliminary Analysis of Rossi's Presentation Regarding The Risks of Neutron Radiation Exposure.

8. Alternative Actions.

9. Preliminary Value/Impact Appraisal.

10. Recommendation.

11. Activities Under NRC Jurisdiction.

12. Memorandum dated October 22, 1976 from the Director, Office of Standards Development to the Director, Division of Siting, Health and Safeguards Standards, Office of Standards Development.

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

May 12, 1978

Mrs. P. M. Allen
North Anna Environmental Coalition
112 Hallmark North
Briarcrest Gardens
Hershey, Pennsylvania 17033

Dear Mrs. Allen:

I am writing in answer to your letter of March 29, 1978, which asks several questions regarding the neutron exposure issues raised by Mr. Glen W. Zimmer in his January 25, 1978 memorandum to Roger Boyd. I'm sure much of your concern was answered by the presentation of Mr. Seymour Block to the ACRS on April 7, 1978, which I understand you attended. However, we will answer your questions in this letter as well.

There have been several meetings among the NRC staff and several memoranda (enclosed) that summarize the staff interactions and the results of inquiries to Battelle Pacific Northwest Laboratory (BNWL) and to the NRC regional offices. These enclosures include (1) a review of the facts in the case in a memorandum from D. Eisenhut and R. Vollmer to L. Higginbotham of March 6, 1978, which asks the Office of Inspection and Enforcement to review reactor licensee neutron monitoring programs and (2) a memorandum from E. G. Case to S. Levine of April 3, 1978, requesting a research study on effectiveness of neutron dosimetry at operating reactors. Mr. Block's April 7, 1978 presentation before the ACRS indicated that there does not appear to be a significant neutron exposure problem at operating reactors.

In response to your specific questions, we are providing the following responses:

Question 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)?

Response We are not aware of any specific reactors where personnel are receiving neutron exposures which heretofore have been unknown. The Zimmer memorandum was based on limited technical data from a yet uncompleted technical program. We have discussed this matter with Mr. Zimmer and other involved people (see Attachment 1) and have concluded that personnel neutron monitoring inaccuracies (if any) are not significant concerns.

Standard techniques of personnel neutron monitoring require that neutron radiation measurements be made with neutron dose equivalent ratemeters prior to personnel entry into radiation areas where neutrons may be present. Neutron radiation exposures are then controlled by health physics personnel by limiting occupancy time in these areas in accordance with the dose limit requirements of the Code of Federal Regulation, Section 10 CFR 20.101. Additionally, personnel may wear passive monitoring devices in accordance with the recommendation of Regulatory Guide 8.14 to corroborate the radiation survey measurements.

Question 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?

Response No new measurement techniques have been instituted because, at present, the staff feels that they are not warranted. Licensees are using state-of-the-art dosimetry, including neutron/gamma ratio techniques. By the request of the March 6 memorandum (Enclosure 1), NRC Office of Inspection and Enforcement will review neutron dosimetry programs at power reactors, to further ascertain that the programs are in accordance with the aforementioned regulations and Regulatory Guidance. Verification that current regulatory guidance is being followed will provide assurance that neutron exposures are being properly assessed by licensees.

Question 3 If new measurement techniques have not yet been instituted, what measures are being taken to project workers from previously unknown neutron exposure?

Response The enclosure and answers to the above questions answer this question. We do not believe there are "previously unknown neutron exposures" but several checks are being instituted to review existing programs and further research (Enclosure 2) will attempt to determine if guidance can be given that will lead to greater accuracy in neutron measurement.

Question 4 What assessment is being made of potentially inadequate reactor shielding?

Response Assessments of the adequacy of reactor shielding are made at several points in the design and operation of a commercial power reactor. First, the reactor shielding designers perform calculations to estimate the effectiveness of their shield design. During the licensing review process, the NRC reviews the design and the radiation field estimates made by the designers. During start-up testing, extensive survey measurements are made of the reactor shielding to ensure that radiation dose rates (neutron and gamma) are not above design expectations and these are controlled as necessary by adding shielding or restricting access to those areas.

Question 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.

Response With respect to workers being informed of potential neutron exposures, the Code of Federal Regulations, Title 10, Part 19.12 and Part 20.206, requires that all individuals be informed that they are entering an area where they may be exposed to radiation, including neutron radiation, and be instructed in safety precautions associated with radiation hazards. 10 CFR 20.202 requires that licensees supply personnel monitoring equipment to specified individuals. For personnel neutron monitoring, Regulatory Guide 8.14 "Personnel Neutron Dosimeters" was developed to provide acceptable guidance to licensees where exposure to neutrons may occur. All of the above sections of the regulations are inspected for compliance by the Office of Inspection and Enforcement and appropriate enforcement actions are taken as discrepancies are noted. The risks from neutrons per unit dose equivalent (REM) are not different from the risks from gamma radiation for the same dose equivalent. In either case, the permissible limit for occupational radiation exposure is as described in the Code of Federal Regulations Section 10 CFR 20.101.

Data on actual neutron exposures, obtained from licensees when employees terminate, have shown that neutron dose is a small fraction of the radiation exposure received by nuclear power plant workers.

Sincerely,

Brian Grimes

Brian K. Grimes, Assistant Director
for Engineering and Projects
Division of Operating Reactors

Enclosures:
As stated

MAY 12 1978

WASHINGTON, D. C. 20555

ENCLOSURE 1

MAR 0 5 1978

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.

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

ENCLOSURE 2

UNITED STATES
NUCLEAR REGULATORY COMMISSION
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).

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, Acting Director
Office of Nuclear Reactor Regulation

cc: See page 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

Items 5-8
in Enc 3 of
4-4-78
Memo.
Denton to Al

Documents Transmitted to NRR by SD:OHSB
Following Receipt of Copy of Letter from
Representative John D. Dingell to NRC
Chairman Joseph M. Hendrie

FROM: Glenn W. Zimmer, Senior Health Physicist, SD:OHSB

1. Radiation Effects Research Foundation, Organization, Programs and Findings, May 17, 1976.

2. Letter from W. D. Rowe, Ph.D., Deputy Assistant Administrator for Radiation Programs, EPA, to J. G. Speth, Natural Resources Defense Council, Inc., Aug. 10, 1976.

3. COO-3243-5, pages 156-165, "Possible Modification of the Theory of Dual Radiation Action," Y.M.-P Lam and H. H. Rossi, 1976.

4. COO-3243-5, pages 166-173, "Induction of Leukemia by Fast Neutrons," H. H. Rossi, 1976.

5. Memorandum from H. Peterson, ESB, SHSS, SD to R. J. Mattson, Director, SHSS, SD, subject: Merits of Additional Studies by the National Academy of Sciences on Radiation Injury, Nov. 1976.

6. "Leukemia in Atomic Bomb Survivors, Hiroshima and Nagasaki, 1 October 1950 - 30 September 1966," T. Ishimaru, et al., Rad. Res. 45, 216-233 (1971).

7. "Isotropic and Cloud Source Irradiation by Monoenergetic Neutrons and Photons T. D. Jones, and J. W. Poston, ORNL (Pre-publication copy).

8. ORNL/TM 5337, "Radiation Insult to the Active Bone Marrow as Predicted by a Method of CHORDS," T. D. Jones (March 1976).

9. ORNL-5191, "A CHORD Simulation for Insult Assessment to the Red Bone Marrow," T. D. Jones (August 1976).

10. "Leukemia Risk from Neutrons," H. H. Rossi and C. W. Mays (copy of paper submitted to "Science") (1976).

11. "The Effects of Small Doses of Ionizing Radiation: Fundamental Biophysical Characteristics," H. H. Rossi (copy of paper presented to the joint meeting of the Radiation Research Society and the Health Physics Society, San Francisco, 1976).

12. NCRP letter from W. Roger Ney, Executive Director, to Members of the Council, Subject: Proposed NCRP Statement on Reduction of the Maximum Permissible Dose Equivalent for Neutrons, with enclosures except the ballot (March 24, 1976).

13. NCRP Report on Recent Actions of the Board of Directors (1976).

14. Memorandum from Allen Brodsky, Senior Health Physicist, OHSB, SHSS, SD, to files, Subject: Summary of Information Obtained at Health Physics Society Meeting, June 28 to July 2, 1976, on Rossi's Proposal to Lower Permissible Exposure Rates to Neutrons (July 12, 1976).

15. Preliminary Value-Impact Assessment for Task Initiation to Develop an NRC Staff Technical Position on Neutron Quality Factors, G. W. Zimmer (Nov. 1976)

16. Memorandum from R. H. Vollmer, A.D. for Site Analysis, DSE, to Dennis M. Crutchfield, Leader, Technical Support Section, PSB, Subject: Concurrence in OSD Task Initiation (December 1, 1976).

17. Memorandum from B. Grimes, Chief, EEB, DOR, to V. Stello, Jr., Director, Division of Operating Reactors, re: OSD Task Initiation OH 704-8 (Dec. 3, 1976).

18. NCRP Memorandum from Thomas Fearon, Staff Assistant to Members of Scientific Committees 1 and 40, Subject: Rossi Memorandum on Neutron Dose Response (June 21, 1976).

19. Letter from V. P. Bond, M.D., Associate Director, Brookhaven National Laboratory, to Dr. Lauriston S. Taylor, NCRP, regarding "NCRP statement for Reduction of Neutron Dose Limit" (March 16, 1976).

20. Zimmer's comments on V. P. Bond's comments on Rossi in Bond's letter to Taylor dated March 16, 1976.

21. NRPB-R57, "Doses in Radiation Accidents Investigated by Chromosome Aberration Analysis VII, A Review of Cases Investigated: 1976 (received from NRC library, March 15, 1977).

22. "An Analysis of Leukemia Data From Studies of Atomic Bomb Survivors Based on Estimates of Absorbed Dose to Active Bone Marrow," G. D. Kerr, et al., to be published in the Proceedings of the Fourth International Congress of the International Radiation Protection Association, Paris, April 24-20, 1977.

23. "Low-Dose RBE and Q for X-ray Compared to γ-ray Radiations," V. P. Bond, C. B. Meinhold, and H. H. Rossi (submitted to Health Physics Journal, prepar April 1977).

24. "Influence of Dose Rate and LET on Dose-Effect Relationship: Implications for Estimation of Risks of Low-Level Irradiation," Report of NCRP SC # 40, draft of April 5, 1976, pages 1-5 and 116-151.

25. "A Proposal for Revision of the Quality Factor," H. H. Rossi (Pre-print-not for publication) (1977).

26. Letter from R. J. Mattson, Acting Director, SHSS, SD, to Karl R. Goller, A.D. for Operating Reactors, Division of Reactor Licensing, Subject: Standard Review Request: "Neutron and Gamma-Ray Flux-To-Dose-Rate Factors" (September 24, 1975).

27. Letter with copy of Draft ANSI N666, from R. B. Minogue, Director, Office of Standards Development, to Ms. Mary Crehan Vaca, Assistant Program Administrator - Nuclear, ANSI, Subject: ANSI N666 (April 19, 1977).

28. Memorandum from G. W. Zimmer, OHSB, to R. B. Minogue, Director, Office of Standards Development, Subject: Neutron Exposure at Commercial Power Reactors (January 25, 1978).

29. Memorandum from G. W. Zimmer, OHSB, SD, to Roger S. Boyd, Director, Division of Project Management, Office of Nuclear Reactor Regulation, Subject: Neutron Exposure at Commercial Power Reactors (January 25, 1978).

30. PNL-2449/UC-48, Sixth ERDA Workshop on Personnel Neutron Dosimetry, July 11-1 1977, Oak Ridge, Tennessee.

31. Final Draft, "Proposed ANSI Standard N323 Radiation Protection Instrumentatic Test and Calibration" (September 1975 with July 1977 modifications).

32. BNWL-2159, "A Test of the Performance of Personnel Dosimeters," L. L. Nichols (April 1977).

33. Draft "Radiation Protection Based on Risk--No RBE," T. D. Jones (1977).

34. Preprint "Risk of Environmental Cancer Based on Cytotoxicity," T. D. Jones, et al. (1977).

Glenn W. Zimmer

The items circled (copies attached) are in my opinion important enough to receive special consideration by NRR for Transmittal to hearing boards. Items 8 and 11 are references to item 15 and are included for easy references.

Glenn W. Zimmer 3/6/78

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

MEMORANDUM FOR: Robert B. Minogue, Director
Office of Standards Development

FROM: G. W. Zimmer, Occupational Health Standards Branch

THRU: R. E. Alexander, Chief, OHSB
I. C. Roberts, AD for Site and Health Standards
R. G. Smith, Acting Director, SHSS

SUBJECT: NEUTRON EXPOSURE AT COMMERCIAL POWER REACTORS

Recently, because of my work on the Health Physics Society Program Committee, for which I am chairing a session at the forthcoming Health Physics Society meeting on neutron measurement and dose assessment, it has come to my personal attention that personnel at 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.

In accordance with your December 14, 1977 memorandum (Subject: Informing Licensing Boards of New Information), I am submitting for your consideration a memorandum (Enclosure 1) to the Director, Division of Project Management, Office of Nuclear Reactor Regulation calling this to his attention.

Glenn W. Zimmer

G. W. Zimmer
Occupational Health Standards Branch

Enclosure:
As stated

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

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

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.

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.

Preliminary Value-Impact Assessment for Task Initiation to Develop a NRC Staff Technical Position on Neutron Quality Factors

I. The Proposed Action

A. Description

Develop the NRC Staff Position on the need to revise 10 CFR Part 20 to reflect recent scientific findings advanced by Dr. H. H. Rossi, and to assess related regulatory practices if the hypotheses advanced by Dr. Rossi are valid. This task does not include amendment of the regulations if such is warranted.

B. Need for the Proposed Action

Dr. H. H. Rossi of Columbia University, utilizing the predicted dose to bone marrow developed by Dr. T. D. Jones (1) claims (2) to have determined that there is an increase of leukemia at low doses of neutrons, and recommended an increase in the quality factor (Q) for neutrons, thereby reducing the allowable exposure. The Office of Standards Development on June 21, 1976 established a task group to review and analyze Dr. Rossi's paper, and recommend action. The task group report (attachment 1) and supporting data (attachment 2) were reviewed and the recommendation of the task group report that NRC "initiate a thorough review, soliciting comments and discussion from others in the scientific community to arrive at a valid decision, and recommended action" was concurred in by the Director, Office of Standards Development (attachment 3), who directed that activities to reach a concensus NRC staff technical position be initiated.

C. Need for NEPA Assessment

None.

II. Alternatives

A. Alternative actions are enumerated under Tab 4 of attachment 1.

B. Alternative 6 has been judged viable.

III. Probable Value/Impact of the Proposed Action

A preliminary Value/Impact Appraisal is included under Tab 5 of attachment 1. Additional Value/Impact information is enunciated below.

A. NRC

The impact on the NRC staff of this action will be the expenditure of NRC resources to accomplish the task. This allocation of resources is not presently incorporated in the OHSB objectives and is not included in the SHSS five year plan. It is anticipated that due to the complexity of the review and analysis* the "accomplishment of this task within the requested 6-month time-frame will cause a delay of initiation of some work within OHSB. As an indication of the complexity, the NCRP has had two scientific committees reviewing this work for approximately six months. The value of this task is that it will demonstrate that the NRC is an independent agency keeping abreast of current developments in the nuclear area, developing its own independent judgments, and taking action accordingly to assure proper standards for radiation protection of occupationally exposed workers and the public.

B. Other Government Agencies and Industry

This task will not have any impact on other government agencies or industry. However, if the result of this task is a finding that Dr. Rossi is correct in whole or in part there could be impact upon other agencies and industry, i.e., more research on neutron dosimetry may be required, and those agencies and industries that have personnel working with neutron producing equipment or sources may need to revise their radiation protection programs or install additional shielding, thus a greater expenditure of funds. The value of this task to other agencies and industry is that they can be assured that NRC is keeping its standards commensurate with available scientific findings.

C. Public

The impact of this proposed action on the public would be the expenditure of tax dollars to accomplish the task. The value of this task to the public is that they can be assured that NRC is keeping its standards commensurate with available scientific findings.

IV. Relationship to Other Existing or Proposed Federal, State or Local Regulations or Policies

A. Expected Conflict of Conformance

It is not known whether or not other Federal, State or local agencies are evaluating Dr. Rossi's work, as is intended by the initiation of this task.

*See attached Memo.

V. Recommendation

It is recommended that task initiation of the proposed action be approved.

REFERENCES

(1) T. D. Jones, "Radiation Insult to the Active Bone Marrow as Predicted by a Method of CHORD's," Oak Ridge National Laboratory Report ORNL-TM-5337, 1976.

(2) Harald H. Rossi, "The Effects of Small Doses of Ionizing Radiation: Fundamental Biophysical Characteristics," presented at the joint annual meeting of the Health Physics Society/Radiation Research Society, San Francisco, June 29, 1976.

November 5, 1976

Mr. I. C. Roberts

As requested, the attached represents my estimates of the impact on OHSB tasks and the resources required to accomplish the Neutron Quality Factor Task assigned to me. This information is not detailed in the Task Initiation as I did not know if it was needed for concurrence by other Offices or if it was for SD use in the decision process.

Justification for the amount of my time estimated to be required is that due to the complexity of the task I do not believe it can be accomplished in less than six months of essentially full time work. The NCRP has had two scientific committees working on this task for about 6 months and has not yet reached the final decision. Thus it is estimated that to develop an independent NRC position it is not unreasonable to plan on 1/2 of a man year of effort on a task of this importance to radiological safety plus the other associated resources estimated on the attached.

Respectfully,

Glenn

Glenn W. Zimmer

IMPACT ON OHSB TASKS AND ESTIMATE OF RESOURCES REQUIRED

Delay in initiation of the following OHSB tasks

OH 703-1 - Rule Change For Radiation Protection Instrumentation Test and Calibration

OH 702-1 - Inspection Exit Interview Rule Change 10 CFR Parts 30, 40, 50, and 70

OH 705-1 - Occupational ALARA Rule Change 10 CFR Part 20

Delay in issuance of Revision 1 of OHSB task

OH 603-4 - Licensing Guide for Type-A Byproduct Material Licenses of Broad Scope

Presently it is estimated that work can be completed on OHSB task

OH 610-4 - Regulatory Guide for Licensing Laboratory Use of Small Quantities of Byproduct Material

All of the above tasks are assigned to me, and it is estimated that the Neutron Quality Factor Task assigned to me will require virtually full time effort to complete the task within the six month time frame.

Resource requirements in addition to my time are as follows:

Contracts with consultants at an estimated $100 per day
$10,000

Travel for discussions with experts and consultants
$5,000

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

Robert B. Minogue, Director
Office of Standards Development

REPORT OF NEUTRON QUALITY FACTOR TASK GROUP

On June 21, 1976, the Office of Standards Development established a task group to review and analyze a scientific paper by Dr. H. H. Rossi. Dr. Rossi recommended a reduction in the allowable occupational neutron exposure, by a factor of 10.

The task group has completed the assigned task, and the report with alternatives and recommended action is attached.

Glenn W. Zimmer

Glenn W. Zimmer
Chairman
Neutron Quality Factor Task Group

Enclosure

REPORT

OF

NEUTRON QUALITY FACTOR TASK GROUP

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NEUTRON QUALITY FACTOR TASK GROUP

Mr. Glenn W. Zimmer, Chairman, SD

Dr. Allen Brodsky, SD

Dr. John Nehemias, NRR

Dr. Wayne R. Hansen, NMSS

Dr. Shlomo S. Yaniv, RES

Mr. Lemoine J. Cunningham, IE

Mr. Robert E. Baker, Consultant, SD

Dr. J. Kastner, Consultant, SD

OUTLINE

1. Review of NCRP recommendations to date

2. Origin of current NRC limits

3. Preliminary Analysis of Rossi's Presentation Regarding the Risks of Neutron Radiation Exposure

4. Alternative actions

5. Preliminary Value/Impact Appraisal

6. Recommendation

REVIEW OF NCRP RECOMMENDATIONS TO DATE

The principal recommendations of the NCRP with regard to neutron exposures have been contained in:

- NCRP Report No. 20 (Handbook 63), "Protection Against Neutron Radiation up to 30 Million Electron Volts", November 22, 1957;

- NCRP Report No. 38, "Protection Against Neutron Radiation", January 4, 1971.

NCRP Report No. 20, Table 2, pages 15 (Enclosure A), recommended particular QF's (then termed "RBE") and maximum permissible neutron fluxes for neutron energy ranges up to 30 MeV., and recommended a QF value of 10, if sufficiently detailed information on neutron energy is not available.

In NCRP Report No. 38, Table 2, page 16 (Enclosure B), some of these QF's were modified somewhat, although the recommended value for situations involving unknown energies remained at 10. These are the current NCRP recommendations.

With regard to limitations on whole body dose, NCRP representatives took part in a meeting at Chalk River with their British and Canadian counterparts, September 29/30, 1949. At that meeting, it was agreed to reduce the then existing whole body dose limit of 0.1 r/day by a factor of about 2, and to express it as a weekly limit of 0.3 r/week, which was re-affirmed by NCRP Report 17 (Handbook 59), "Permissible Dose From External Sources of Ionizing Radiation," September 24, 1954.. However, in an insert to accompany Handbook 59, dated January 8, 1957, and an Addendum dated April 15, 1958, the NCRP introduced the concept of limiting cumulative career dose, in effect a limit on average annual dose, set at 5 rems/year. This is the current NCRP recommendations, reaffirmed in NCRP Report No. 39, "Basic Radiation Protection Criteria", January 15, 1971.

Enclosure "A"

Table 2. Maximum permissible neutron flux

Time-average flux for 40-hour week to deliver either 100 or 300 mrems.

Neutron energy	RBE	100 mrems	300 mrems
Mev		$n\ cm^{-2}\ sec$	$n\ cm^{-2}\ sec$
Thermal..................	3	670	2,000
0.001....................	2	500	1,500
.005.....................	2.5	570	1,700
.02......................	5	280	850
.1.......................	8	80	250
.5.......................	10	30	90
1.0......................	10.5	18	55
2.5......................	8	20	60
5.0......................	7	18	55
7.5......................	7	17	50
10.......................	6.5	17	50
10 to 30................................		[a]10	[a]30

[a]Suggested limit.

ENCLOSURE "B"

TABLE 2-Mean quality factors, $\overline{QF}$,[a] and values of neutron flux density which, in a period of 40 hours, results in a maximum dose equivalent of 100 mrem.

Neutron Energy	$\overline{QF}$	Neutron Flux Density
MeV		$cm^{-2}s^{-1}$
2.5×10^{-3} (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
40	7	10
60	5.5	11
1×10^{2}	4	14
2×10^{2}	3.5	13
3×10^{2}	3.5	11
4×10^{2}	3.5	10

[a]Maximum value of $\overline{QF}$ in a 30-cm phantom.

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ORIGIN OF CURRENT NRC LIMITS

The principal provisions of the Commission's radiation protection regulations are contained in 10 CFR Part 20, "Standards for Protection Against Radiation".

The Commission implemented the recommendations in NCRP Report 17 (Handbook 59) and NCRP Report 20 (Handbook 63) by including exactly the NCRP values for QF and neutron flux in §20.4(d), see Enclosure "C", and incorporating the new NCRP whole body dose recommendations in §20.101 (a) and (b), see Enclosure "D", both published in effective form on September 9, 1960. The regulations have not been amended to incorporate the current NCRP neutron flux values published in NCRP Report No. 38.

ENCLOSURE "C"

NEUTRON FLUX DOSE EQUIVALENTS

Neutron energy (Mev)	Number of neutrons per square centimeter equivalent to a dose of 1 rem (neutrons/cm^2)	Average flux to deliver 100 millirem in 40 hours (neutrons/cm^2 per sec)
Thermal	970×10^6	670
0.0001	720×10^6	500
0.005	820×10^6	570
0.02	400×10^6	280
0.1	120×10^6	80
0.5	43×10^6	30
1.0	26×10^6	18
2.5	29×10^6	20
5.0	26×10^6	18
7.5	24×10^6	17
10	24×10^6	17
10 to 30	14×10^6	10

ENCLOSURE "D"

§20.101 Exposure of individuals to radiation in restricted areas.

(a) Except as provided in paragraph (b) of this section, no licensee shall possess, use, or transfer licensed material in such a manner as to cause any individual in a restricted area to receive in any period of one calendar quarter from radioactive material and other sources of radiation in the licensee's possession a dose in excess of the limits specified in the following table:

		Rems per calendar quarter
1.	Whole body; head and trunk; active blood-forming organs; lens of eyes; or gonads	1-1/4
2.	Hands and forearms; feet and ankles	18-3/4
3.	Skin of whole body	7-1/2

(b) A licensee may permit an individual in a restricted area to receive a dose to the whole body greater than that permitted under paragraph (a) of this section, provided:

(1) During any calendar quarter the dose to the whole body from radioactive material and other sources of radiation in the licensee's possession shall not exceed 3 rems; and

(2) The dose to the whole body; when added to the accumulated occupational dose to the whole body, shall not exceed 5 (N-18) rems

where "N" equals the individual's age in years at his last birthday; and

(3) The licensee has determined the individual's accumulated occupational dose to the whole body on Form AEC-4, or on a clear and legible record containing all the information required in that form; and has otherwise compplied with the requirements of §20.102. As used in paragraph (b), "Dose to the whole body" shall be deemed to include any dose to the whole body, gonads, active bloodforming organs, head and trunk, or lens of eye.

PRELIMINARY ANALYSIS OF ROSSI'S PRESENTATION

REGARDING THE RISKS OF NEUTRON RADIATION EXPOSURE

Dr. Rossi shows in his paper[1] and in his talk that the incidence of leukemia in Hiroshima appears to be linear with dose to bone marrow (as now re-calculated as described in another paper by Jones[2]) and most of the incidence of leukemia at Hiroshima can be accounted for by the higher biological effectiveness of the neutron exposure component. On the other hand, he points out that the (excess) leukemias in Nagasaki were almost entirely due to gamma radiation, with a negligible neutron component. A particular non-parametric statistical analysis published by Kellerer and Rossi shows (according to Rossi) that the incidence of leukemia in Nagasaki is more likely to be a quadratic rather than linear relationship, resulting primarily from gamma exposure.

An initial examination of Rossi's paper indicates that the resulting risk estimates for neutron irradiation as based on present methods of measuring neutron exposure and leukemia incidence would not be much

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(1) Harald H. Rossi, "The Effects of Small Doses of Ionizing Radiation: Fundamental Biophysical Characteristics" presented at the joint annual meeting of the Health Physics Society/Radiation Research Society, San Francisco, June 29, 1976.

(2) T. D. Jones, "Radiation Insult to the Active Bone Marrow as Predicted by a Method of CHORD'S," Oak Ridge National Laboratory Report ORNL-TM-5337, 1976.

different than originally estimated 20 or more years ago when the occupational exposure limits were established, when proper correction is made for the average depth in bone marrow at which dose is defined. This impression is obtained from the following calculations. According to Rossi's own equations given in his preprint, but not presented in his verbal paper, the neutron and gamma-related leukemia incidence rates obtained from the Hiroshima and Nagasaki data are fitted by Equations (1) and (2), respectively:

$$R = 3 \times 10^{-5} D_n, \qquad (1)$$

where R is the average number of leukemia cases per person-year,* and D_n is in rads to the bone marrow calculated as in the more recent sophisticated computer program.[2,3] For gamma radiation Rossi fits the following relationship to the data:

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(3) G. D. Kerr and T. D. Jones, A Reanalysis of Leukemia Data on Atomic Bomb Survivors Based on Estimates of Absorbed Dose to Bone Marrow (Health Physics Division, Oak Ridge National Laboratory, Oak Ridge, TN 37830), presented at the Twenty-First Annual Meeting of the Health Physics Society, July 1, 1976.

*The unit cases per person-year for R is used to mean that the coefficient ($3 \times 10^{-5} \frac{\text{cases}}{\text{person-yr-rad}}$) is to be multiplied by dose. To obtain the total number of cases, R would be multiplied by the number of persons exposed, and by the average number of years at risk after exposure. To obtain the average incidence of cases/year, R would be multiplied by the dose and the population size.

$$R = 1.8 \times 10^{-8} D_{\gamma}^{2}, \qquad (2)$$

where D_{γ} is the gamma dose at the mean bone marrow depth in rads.

If one assumes a quality factor of 10 for neutrons for occupational exposure, as assumed in the NCRP standards that are incorporated in 10 CFR 20 (the former Handbook 63 values), one should use a factor of 3×10^{-6} (cases per person-rem-year at risk) as a risk factor in the first equation (1), if D_n were to be expressed in rem, since the risk is fixed by the actual observed cases of leukemia. This factor of 3×10^{-6} is not very different from the old factors of 1 to 2×10^{-6}/person-rem-year (PRY)*[(4),(5),(6),(7)] after exposure that we have used for about 20 years, since the rise in leukemia at Hiroshima and Nagasaki was first published in Science in 1958. This is particularly true since Rossi has used dose estimates at a bone marrow depth that includes recent calculations of attenuation showing that the dose at the average bone depth is 1/4 of that near the surface of the

---

* PRY means the risk factor must be multiplied by the population size exposed times the single dose in rem times the "years-at-risk" over which the given disease may appear after the exposures (and following the latent period), in order to obtain the total number of observed cases of leukemia (in excess of those naturally occuring).

(4) The Biological Effects of Atomic Radiation NAS-NRC Summary Reports (1960)

(5) United Nations Scientific Committee on the Effects of Atomic Radiation. Report. General Assembly, 19th Session. Supplement No. 14 (A/5814), p.85, 1974

(6) International Commission on Radiological Protection. Publication 14. Radiosensitivity and Spatial Distribution of Dose, Oxford, Pergamon Press, 1969.

(7) The Effects on Populations of Exposure to Low Levels of Ionizing Radiation, BEIR Report, NAS-NRC, (Nov. 1972).

body, and also about 1/4 of the neutron kerma to a small volume of tissue in free air. With this factor of 4 incorporated in the denominator of the risk estimate obtained from the epidemiologic data, it is no wonder that the risk factor in Equation (1) is somewhat higher than it used to be. The exact derivation of Rossi's risk estimates thus bears further investigation.

If we use the gamma dose at Nagasaki from the highest group (where most of the earlier cases occurred), say about 100 rads, then the gamma risk would also come out consistent with about a 2×10^{-6} risk factor applied under the "linearity" assumption (with a D instead of D^2 dependence). Animal and micro-dosimetric data may show that the neutron response is more likely linear and that the gamma response is probably curvilinear, and that the conclusions are probably well-founded regarding the increase of RBE up to 100 or more as doses decrease to about 0.5 rad or below for neutrons. On the other hand, this increasing RBE at lower neutron doses may be due to a comparison of the neutron risk with the much more rapidly reduced risk as gamma exposures are lowered, rather than a risk of neutron exposures higher than had previously been assumed. This is another question that should be further investigated and clarified.

ALTERNATIVE ACTIONS

1. Do nothing.

2. Wait for NCRP action to determine what action NRC should take.

3. Accept new analysis and issue a Rule Change.

4. Reject new analysis as not valid.

5. Publish a Notice of Consideration of Rule Change and solicit comments.

6. Initiate a thorough review, soliciting comments and discussion from others in the scientific community to arrive at a valid decision, and recommended action.

PRELIMINARY VALUE/IMPACT APPRAISAL

The Nuclear Regulatory Commission as an independent agency should keep abreast of current developments in the nuclear area, develop its own independent judgments, and take action accordingly. This being the case, Alternatives 1 and 2 should not be considered as potential solutions. Due to the complexity, limited time allowable, and the lack of sufficient confirmed information, it is not possible to make a valid scientific decision to accept or reject the new analysis at this time. Thus, Alternatives 3 and 4 should be eliminated from further consideration. The task group is unanimous that additional information must be gathered, reviewed, and discussions held with others in the scientific community to arrive at a valid decision and to be able to recommend action. Since a decision cannot be made at this time, it is considered unwise to issue a public Notice of Consideration of Rule Change as this could create concern which might prove to be unwarranted. Thus, Alternative 6 is considered as the only viable alternative until a thorough review can be completed.

A thorough review and analysis is necessary to assure that valid occupational radiation protection limits exist to protect workers and the public. Although definitive data are not available it is estimated that there may be about 3000 workers in licensees' facilities receiving some neutron exposure, and the number of occupationally exposed individuals in activities not licensed by NRC, but which have chosen to follow NRC regulations for purposes of radiation protection may exceed 30,000 individuals.

It is not intended to fully assess the impact of decreasing the allowable neutron exposure unless it is shown that a reduction is warranted.

RECOMMENDATION

The Neutron Quality Factor Task Group recommends that Alternative 6 be selected as appropriate action. Due to the complexity of the review, the following method would ensure thoroughness and reduce the commitment of resources needed to accomplish the task: assign an individual full time to the review and use the Task Group as a review committee to assure that technical questions/challenges would not be overlooked. It is tentatively estimated that such a review could require 6 months and that travel and use of consultants may be required.

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ACTIVITIES UNDER NRC JURISDICTION

Type of Facility		# of Persons Under NRC Jurisdiction With Potential Neutron Exposure
Cf^{252} (Hosp. & Res. & Schools) (not counting negligible exposure to students)[1]		400
Research Reactors[2]	39 @ 15 people ea.	585
Oil Well logging[3]		200
Accelerators[4]	81 @ 10 people ea.	810
Fuel Fab and Reprocessing[5]		750
Total number of persons with potential neutron exposure under NRC jurisdiction.		2745≈3000

Assuming lic sources are then person exposure is in NRC juris tion.

ACTIVITIES NOT UNDER NRC JURISDICTION

Activity	# of Persons with Potential Neutron Exposure	
ERDA[6]		
Reactor	2730	
Fuel Fab (~1/4 of total monitored)	250	
Fuel Processing (~1/2 of total monitored)	750	
Accelerators	2382	
Weapons	7846	
Irradiation Facilities	24	
	13982	≈ 14000
DOD[7] (Facilities not under NRC license)		
Army	500	
Navy	12000	
Air Force	1500	
	14000	= 1400C
Total not under NRC Jurisdiction		= 280C

Total Number of Persons with Potential Neutron Exposure 310C

1. This estimate was obtained from the Demonstration Centers in Louisiana and San Diego, California. Each center estimated 100-300 individuals receiving neutron exposure from Cf^{252}. Therefore, the median estimate from each center was 200 individuals.

2. There are 70 research reactors in the United States. It is estimated that 39 of them may have either neutron beams or thermal columns. Further, it is assumed that there may be 15 people occupationally exposed due to their work, or their research. Although there are 54 power reactors, 11 experimental reactors, 6 critical assemblies, 2 test reactors and 19 DOD reactors reviewed by NRC, it is assumed that neutron exposure would be minimal even though some occupationally exposed personnel may enter containment during operations.

3. This estimate was obtained from the Radioisotopes Licensing Branch.

4. In 1964 there were 224 accelerators in the U.S. The estimate is based on the fact that some accelerators have ceased operation and others have been built; therefore, it is assumed that at least 81 are in existence that can produce neutron exposure. Further, it is assumed that there are 10 operators and/or experimenters per machine that could be receiving some neutron exposure through the shielding.

5. In 1975 there were 5602 people that received measurable occupational exposure in these facilities. Currently there are probably less than 150 that receive neutron exposure as there is only one plutonium facility active. The estimate of 750 is based on the assumption that other licensees may again become active.

6. This information was supplied by the Health Protection Branch, Division of Safety, Standards and Compliance, and represents personnel with measurable exposure, some of which was assumed to be neutron.

7. Estimated numbers: Does not include any reactor exposure.

REACTORS

14 reactor facilities reported terminations with the personnel exposures shown by type of radiation.

207 people had neutron exposure.

If the quality factor were raised by a factor of ten, then the personnel exposure would be higher by a factor of ten and 6 people would have had a quarterly exposure to neutrons of:

3.1 Rem
4.05 Rem
5.2 Rem
7.4 Rem
10.0 Rem
10.5 Rem

FUEL FABRICATION

5 facilities reported terminations with the exposures shown by type of radiation.

114 people terminated had only neutron exposure.

Total number terminated by the 5 facilities was 143 people..

If the quality factor were raised by a factor of ten, then the personnel exposure would be higher by a factor of ten and 20 people would have had annual exposures to neutrons of:

5.86 Rem	9.88 Rem
6.69 Rem	10.5 Rem
7.32 Rem	10.81 Rem
8.70 Rem	11.06 Rem
9.04 Rem	12.46 Rem
9.13 Rem	12.61 Rem
9.14 Rem	13.72 Rem
9.20 Rem	19.20 Rem
9.51 Rem	22.35 Rem
9.78 Rem	22.40 Rem

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

OCT 2 2 1976

Roger J. Mattson, Director
Division of Siting, Health and
Safeguards Standards
Office of Standards Development

REPORT OF NEUTRON QUALITY FACTOR TASK GROUP

I have reviewed the August 20, 1976 report by Glenn Zimmer and the Neutron Quality Factor Task Group. In addition I have studied and discussed with R. E. Alexander and Zimmer preliminary value impact information prepared by the Occupational Health Standards Branch concerning potential neutron exposures in the United States.

I find this information sufficient basis to initiate activities to reach a consensus NRC staff technical position on the need to revise 10 CFR Part 20 to reflect recent scientific findings advanced by Dr. H. H. Rossi (Alternative six of the Task Force Report). Further, although the number of employees of NRC licensees exposed to neutrons in their work is low and the exposure levels are low, my discussion with Alexander and Zimmer has indicated to me that if the hypotheses advanced by Dr. Rossi are correct, changes of a broad nature to some of our regulatory practices may well be indicated. For example, requirements and practices related to who should be monitored, techniques of neutron dosimetry, storage and handling of neutron dosimeters, operational and procedural controls to reduce unnecessary neutron exposure, and other similar factors may need to be changed where, by current standards, present exposures are at de minimus levels but would take on more significance if the standards were modified by a factor of 10. Your work should be undertaken from this broad perspective and not simply as a potential change in Part 20.

Please arrange for commencement of this work within the other ongoing programs of your Division. The position should be developed within six months. I understand that this can be accomplished without disruption of goals and objectives of the ongoing program of the Occupational Health Standards Branch.

Robert B. Minogue

Robert B. Minogue, Director
Office of Standards Development

cc: Robert E. Alexander
Glenn Zimmer

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ORNL/TM-5337

Contract No. W-7405-eng-26

HEALTH PHYSICS DIVISION

RADIATION INSULT TO THE ACTIVE BONE MARROW AS PREDICTED BY A METHOD OF CHORDS

T. D. Jones

MARCH 1976

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee 37830
operated by
UNION CARBIDE CORPORATION
for the
ENERGY RESEARCH AND DEVELOPMENT ADMINISTRATION

Radiation Insult to the Active Bone Marrow as Predicted by a Method of CHORDS

T. D. Jones

OAK RIDGE NATIONAL LABORATORY

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a small target site or cluster of cells within an organ such as the mandible, or a center such as the central nervous, or active bone marrow system. For some effects, cells or sensitive sites within cells may not be irradiated uniformly because of discrete energy loss events and microdosimetric considerations (Rossi, 1975) may be desirable. On a more macroscopic scale, chronic effects such as bone sarcomas or even leukemia may, in some cases, be directly related to highly localized exposures such as usually encountered in radiotherapy of tumors and the maximum absorbed dose at a particular site (mass of a gram as opposed to an intercellular site) may be more meaningful than the mean absorbed dose to the complete active marrow system (Wilson and Carruthers, 1962; A. R. Jones, 1975). Detailed distribution of photon dose to specific active marrow regions for A-P, P-A, rotational, and side (lateral) incidence have been published and should be readily applied to many situations of interest (Jones et al., 1973; Clifford and Facey, 1970). For radiation protection and risk analyses from acute effects and those chronic effects where risk is thought to be proportional to the insult to the system such as usually assumed for leukemia, it is often not possible or desirable to establish insult-response type correlations on a microscopic level. Therefore, it becomes necessary to assign a "mean" insult or risk to a non-uniformly irradiated "critical organ".

One approach to the dosimetry of a non-uniformly irradiated critical organ, such as the red bone marrow system, is to use a probability density distribution of length, referred to as a CHORD length distribution. Any specific CHORD or $p(\ell)\, d\ell$ distribution is

CRITICAL HUMAN ORGAN RADIATION DOSIMETRY FOR THE ACTIVE BONE MARROW*

Abstract

Critical Human Organ Radiation Dosimetry (CHORD) probability density functions for A-P, P-A, bilateral, rotational, and isotropic incidence, plus simple depth-dose data, permit the rapid estimation of the radiation insult to the active red bone marrow system of the ICRP Reference Man. The CHORD concept follows the variations in the microscopic processes of absorption, attenuation, and scattering on a macroscopic level so that it is not necessary to attempt detailed calculations for each and every case of interest. Similar techniques have been applied to reactor criticality calculations and the general logic of the CHORD process can be applied to any cause-response type situation which can be described in terms of variation with distance in the medium of interest. Doses to active bone marrow from exposures to photons and neutrons are presented and excellent agreement was found with the few available experimental results.

Introduction to the CHORD Concept

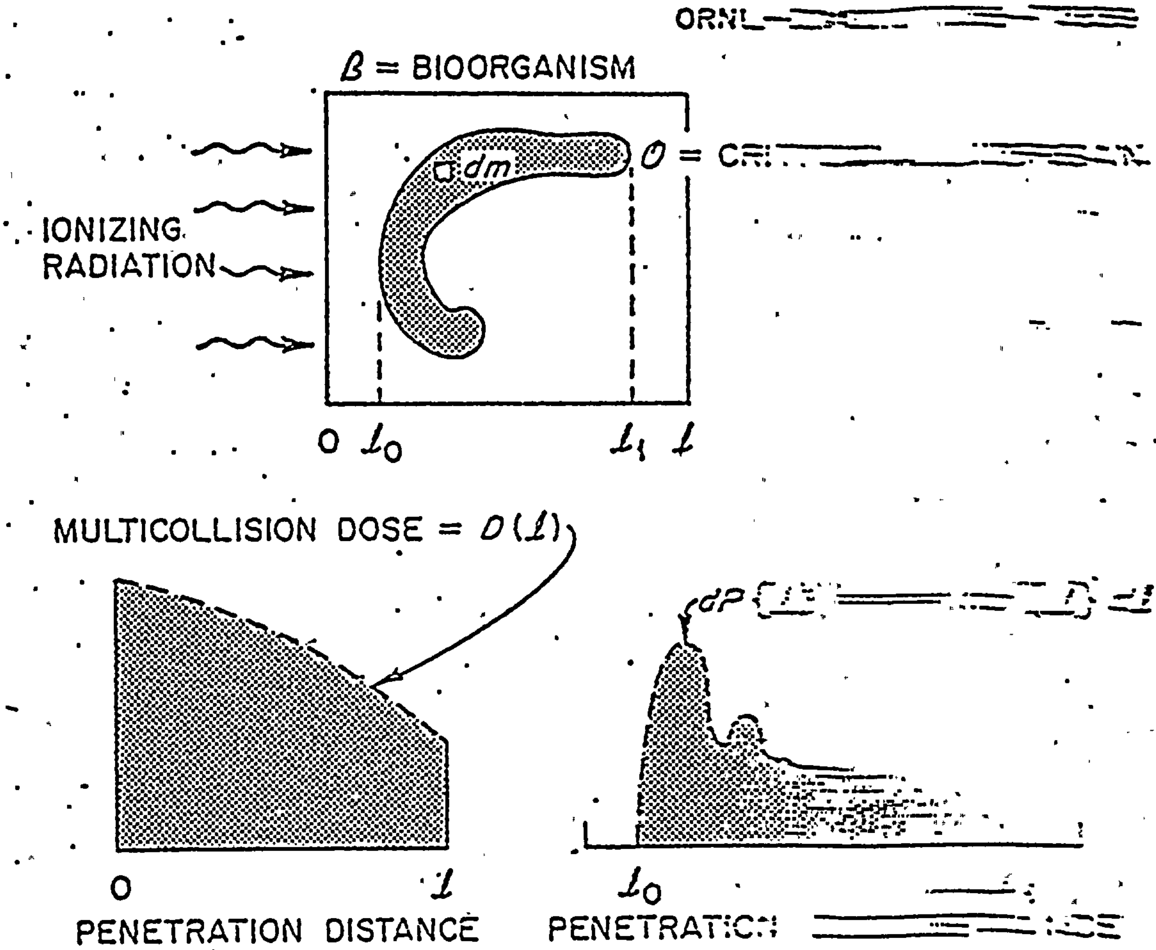
When a bioorganism is subjected to a radiation environment, a critical organ or region of greatest risk usually is irradiated non-uniformly if the linear dimensions of the critical organ are not small or the depth of the critical organ within the bioorganism is not large compared with the mean-free pathlengths of the irradiating particles. Radiation insult specific analyses are usually based on dose to cells,

*Research sponsored by the Energy Research and Development Administration under contract with Union Carbide Corporation.

obtained by assuming that the critical organ is simply a volume of constant density, and for each differential unit of mass dm, chosen by Monte Carlo techniques, the minimum distance ℓ to the closest irradiated air-tissue interface is uniquely determined. This process is continued until $p(\ell)\, d\ell$ is well known statistically. Chord usually implies a straight line through two points on the surface, e.g., the skin; however, in this paper CHORD is an acronym derived from Critical Human Organ Radiation Dosimetry and represents only a specific portion of a "true Chord". The CHORD concept is illustrated in Figure 1 and the CHORD or $p(\ell)\, d\ell$ distribution provides "weighting" factors for an integration over a specific insult such as a "multicollision" depth-dose curve for the source geometry of interest.

CHORD Applications to Red Bone Marrow

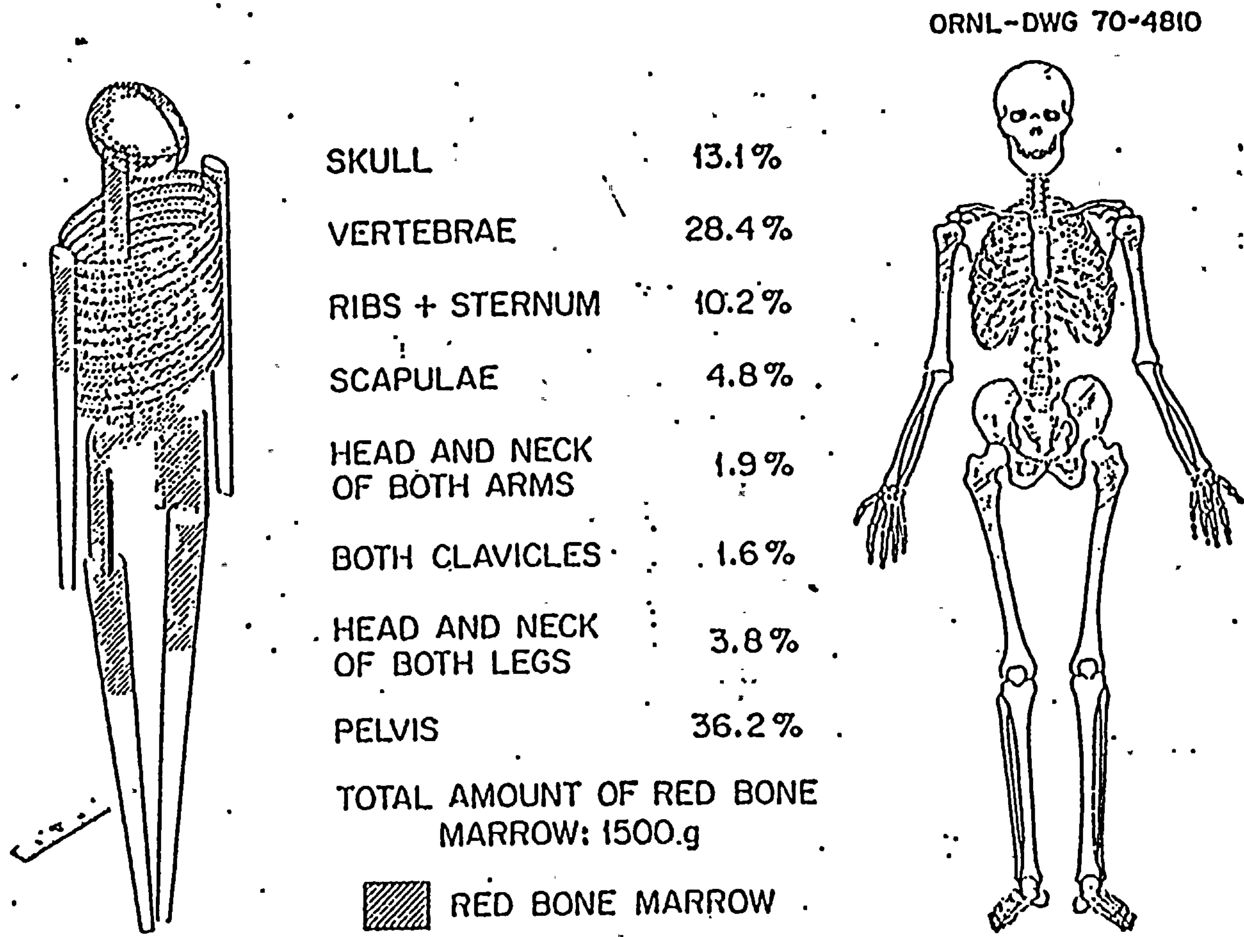
Figure 2 illustrates the distribution of the active red bone marrow in the normal adult and the corresponding analog for our Monte Carlo transport code. In the adult reference man (ICRP, 1975) there are 1500 grams of active red marrow and 1500 grams of yellow marrow which are predominately fat cells. Inactive yellow marrow may be transformed quickly into active marrow by a stimulus such as bleeding or infection; yellow marrow in bone shafts is known to contain some active cells but, in general, the proportion of active cells in adult yellow marrow is usually considered to be small (Spiers, 1966). Thus, for most situations of interest, only the red marrow receives major consideration.



BUT $\ell = \ell(dm)$,

AND $\bar{D} = \int_{0} D(\ell) \cdot p\{\ell\} \cdot d\ell \Big/ \int_{-\infty}^{\infty} p\{\ell\} \cdot$

Fig. 1. Critical Human Organ Radiation Dosimetry ... Concept.



COMPUTER ANALOG OF REFERENCE MAN

A NORMAL ADULT (HASHIMOTO 1060)

Fig. 2. Distributions of the Active Bone Marrow.

The importance of a risk estimate based on radiation damage to the active marrow system cannot be overstated as bone marrow damage usually will be the major mechanism in radiation death and acute radiation sickness stemming from whole body irradiation because it occurs at much lower levels (Facey, 1968; Wald, 1975) than death or incapacitation due to radiation damage of the gut mucosa or the central nervous system. For sublethal criticality accident exposure levels, levels of interest in radiation protection, and population exposure levels, the most demanding recommendations of the ICRP (1964) relate to the maximum permissible doses to the gonads and the blood-forming organs. In radiation protection, the testes are usually considered to be the critical organ of primary interest because of their shallow location and because of the difficulty of estimating the bone marrow insult; however, if the exposure level subjects an individual to considerable risk, then an estimation of the insult to his active marrow system could be advantageous for determining what medical treatment should be administered promptly (Wald, 1975).

The dose at a penetration depth of 5 cm is often chosen to describe the insult to the red bone marrow; however, for photon irradiation the "5 cm rule" is often in error by a factor of two and is expected to be even worse for neutron irradiation. This approximation tends to retain popularity in spite of its inaccuracy, because the red marrow is distributed widely in the skeleton. The skeletal distribution shown in Figure 2 illustrates the fact that, in general, no specific depth can be applied for different exposure geometries and different irradiating particles or even different energies of particles having the same nature.

For internal dosimetry, especially for radionuclides deposited in or near the skeleton, a precise calculational analog of the active marrow system requires some postulations about cavity size variation and the distribution of these marrow cavities within the skeleton. However, for most situations of external exposure, the active marrow may be assumed to be uniformly deposited in certain regions of the skeleton. This simplification is possible because for external exposure, distance versus insult (dose) variation is much less than for internal radionuclide deposition where the insult (dose) usually varies even more rapidly than inversely with the square of the distance. There are two opposing effects that also influence the photon absorbed dose to marrow. These effects are the increased shielding by the bone structure and the enhancement of dose near the higher atomic number bone tissue (Spiers, 1966; Wilson and Carruthers, 1962). As demonstrated later, the net influence of these opposing effects is usually considered to be small for external exposure although such is not always the case for internal emitters.

CHORD Distribution and Marrow Doses

Figure 3 and Table 1 present CHORD density functions for active marrow in the Reference Man Phantom (ICRP, 1975) for A-P, P-A, bilateral, rotational, and isotropic exposure. Due to the nature of the CHORD concept and the general convexness of the Reference Man Phantom, there is no differentiation between 2π and 4π CHORD distributions; however, depth-dose curves will reflect the different

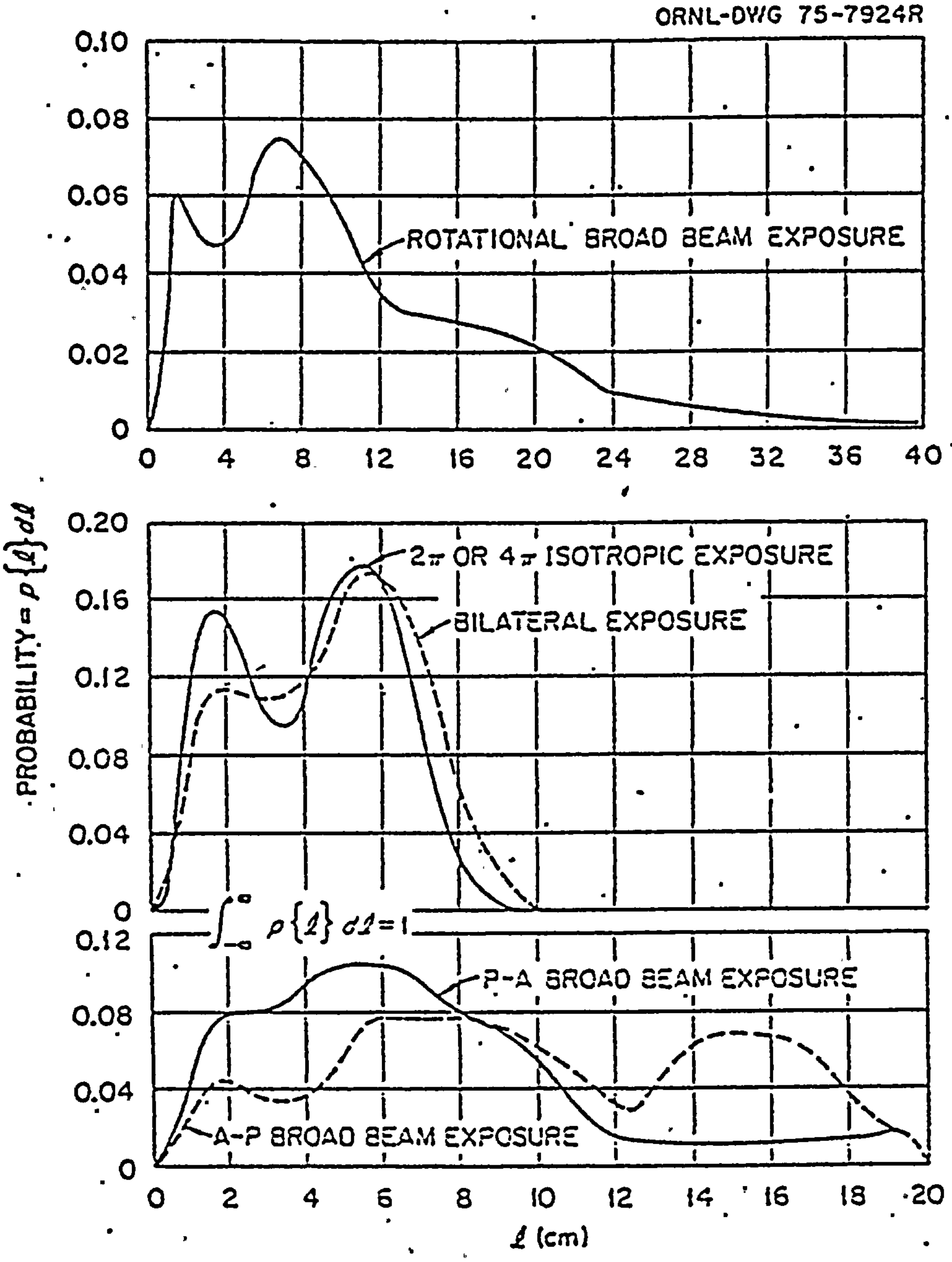


Fig. 3. CHORD Density Functions for Active Marrow in Reference Man.

Table 1. CHORD $p(\ell)\Delta\ell$ Values for Active Marrow in Reference Man.

ℓ (cm)	Rotational	cv	A-P	cv*	P-A	cv	Bilateral	cv	Isotropic	cv
0–0.5	.00515	6	.00626	3	.00718	3	0.0138	2	.0231	3
0.5–1	.0175	3	.0157	2	.0252	1	.0420	1	.0658	2
1–2	.0608	2	.0412	2	.0716	1	.115	1	.154	2
2–3	.0508	3	.0361	2	.0791	1	.114	1	.126	2
3–4	.0465	3	.0340	2	.0850	1	.110	1	.0944	2
4–5	.0505	3	.0442	2	.107	1	.133	1	.152	2
5–6	.0662	2	.0730	1	.126	1	.173	1	.179	2
6–7	.0744	2	.0782	1	.109	1	.160	1	.136	2
7–8	.0705	2	.0748	1	.0806	1	.0966	1	.0586	2
8–9	.0703	2	.0738	1	.0756	1	.0359	2	.0105	6
9–10	.0603	2	.0641	1	.0626	1	.00688	4		
10–11	.0482	3	.0522	1	.0440	2				
11–12	.0380	3	.0364	2	.0207	2				
12–13	.0311	3	.0292	2	.0127	3				
13–14	.0292	3	.0549	1	.0121	3				
14–15	.0282	3	.0658	1	.0119	3				
15–16	.0258	4	.0675	1	.0123	3				
16–17	.0285	3	.0643	1	.0129	3				
17–18	.0283	3	.0492	1	.0130	3				
18–19	.0237	4	.0231	2	.0154	3				
19–20	.0241	4	.0159	3	.0168	2				
20–21	.0218	4								
21–22	.0169	4								
22–23	.0135	5								
23–24	.00985	6								
24–25	.00866	6								
25–26	.00787	7								
26–27	.00672	7								
27–28	.00699	7								
28–29	.00545	8								
29–30	.00562	8								
30–31	.00385	9								
31–32	.00276	11								
32–33	.00194	13								
33–34	.00170	14								
34–35	.00147	15								
35–36	.00184	14								
36–37	.00126	16								
37–38	.00164	14								
38–39	.000988	19								
39–40	.000341	32								

*Coefficient of variation in percent.

exposure geometries. The peak at 2 cm for rotational and isotropic exposure is due to the shorter penetration distances to the side ribs and upper arm bones while the more important peak at about 6 cm is predominantly from the vertebrae and pelvis. The CHORD distributions are influenced strongly by the pelvic region and the thoracic vertebrae which contain about 36% and 28%, respectively, of the total active marrow. In Figure 3, ℓ varies to 40 cm for rotational exposure because it was assumed that rotational CHORD dose estimates will be obtained from broad beam depth-dose data. For bilateral and isotropic exposures, ℓ varies to 10 cm because depth-dose data is expected to be related to the minimum distance to the closest irradiated surface.

The CHORD distributions from Figure 3 were used in conjunction with depth-dose curves (see Figure 1) according to

$$D_{\text{red marrow}} = \sum_{\ell} D(\ell) \cdot p(\ell) \cdot \Delta\ell$$

because all CHORD distributions were normalized to unity. Photon dose to the active marrow as predicted by the CHORD concept is shown in Figure 4; however, bilateral and rotational results are not shown because of close agreement with the results for A-P exposure.

Figure 5 provides active marrow dose relative to exposure at the front of the chest for A-P incidence. Alun Jones' experimental results (1964) are included and the mean deviation between the two methods is only 6% to 1.25 MeV which is high into the Compton range

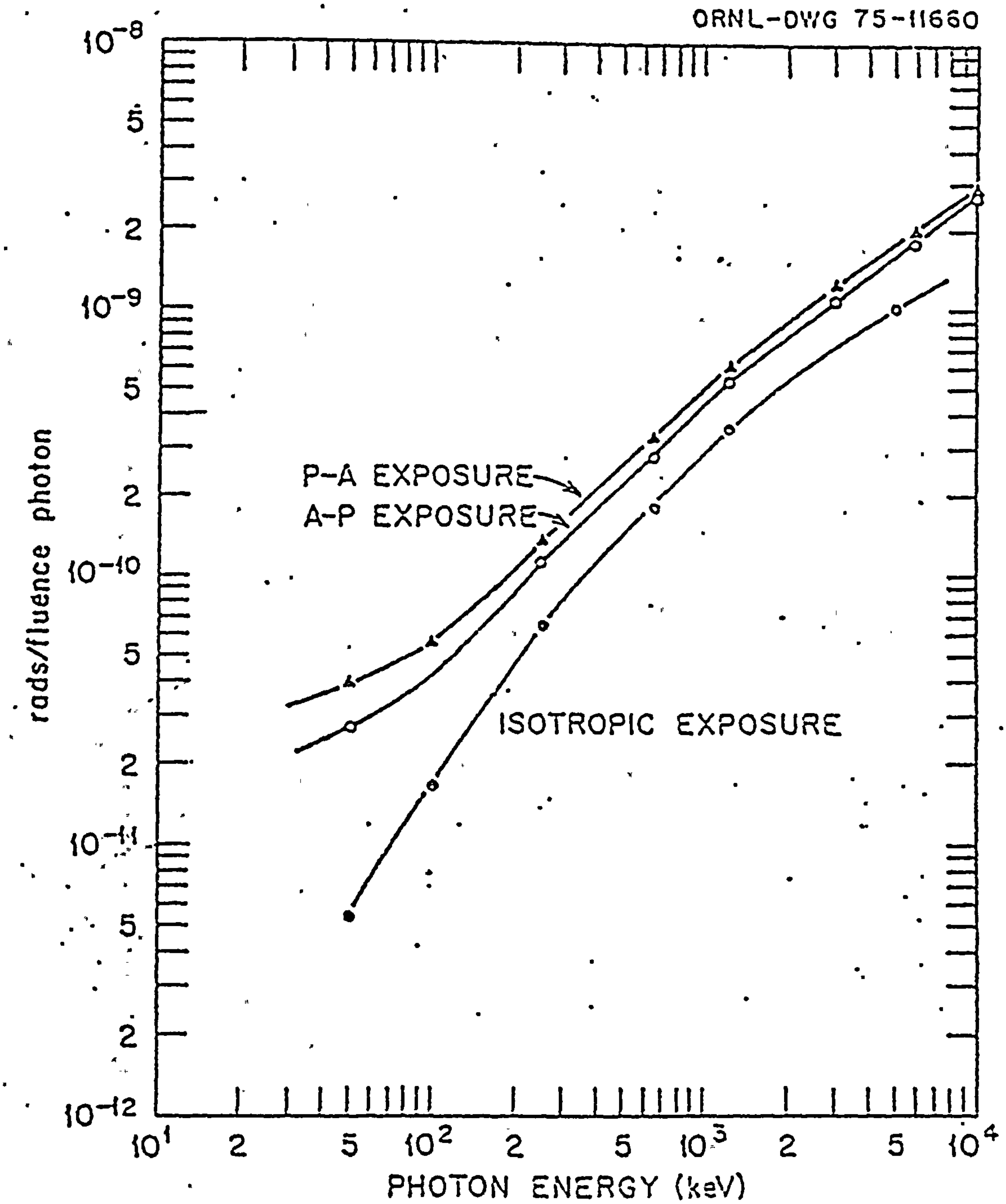


Fig. 4. Dose to Active Marrow as Predicted by the CHORD Concept.

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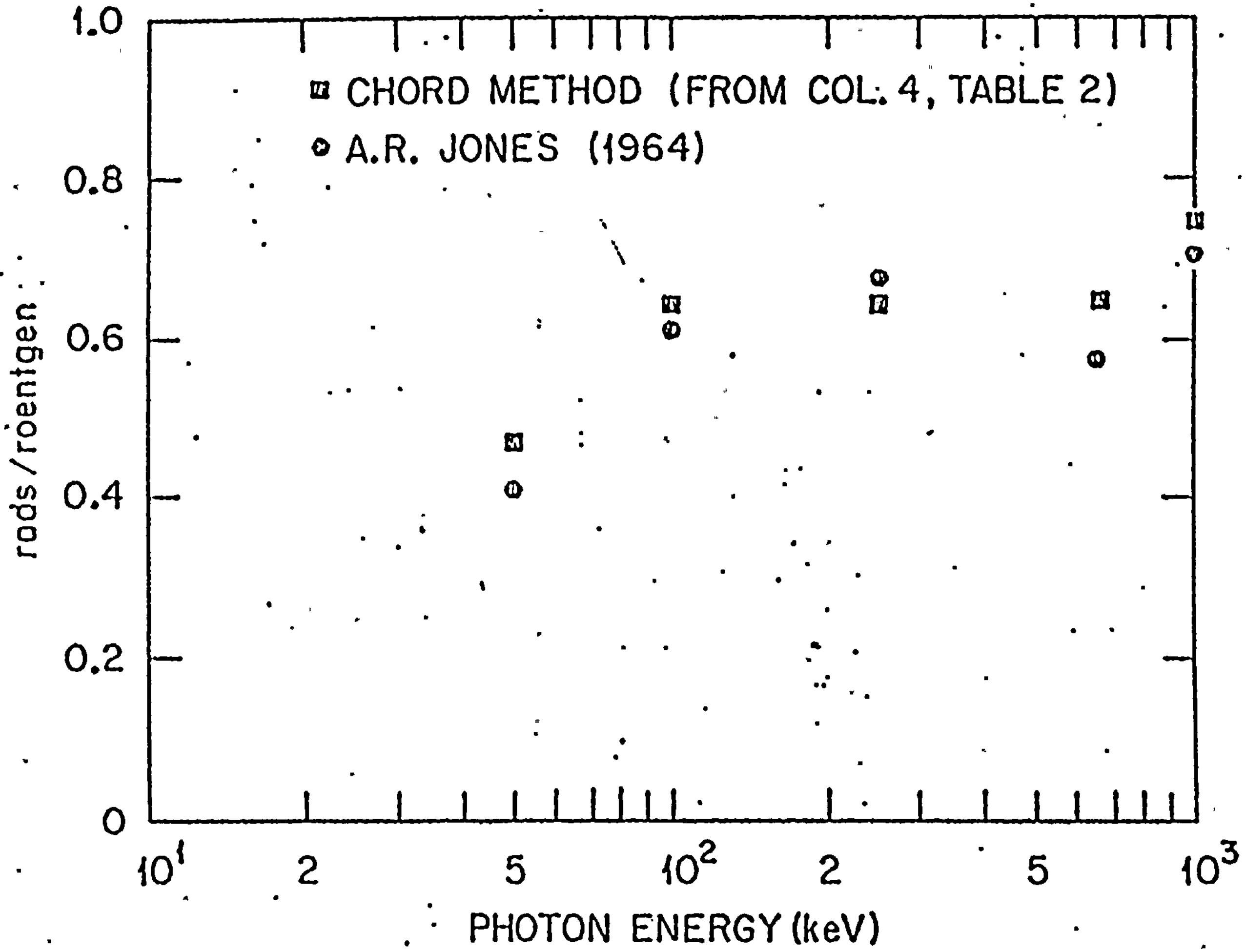

Fig. 5. Active Marrow Dose Relative to Exposure at the Front of the Chest (A-P Exposure).

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shown in Figure 6. Figure 6 is intended to serve as a guideline for applications of the method of CHOROs to critical regions in or near bone tissue. Experimental results were not available for higher energies. Column 4 in Table 2 represents estimates from the CHORD method and column 5 is from our Monte Carlo transport code (Jones, et al., 1973). These values shown in column 5 were calculated at the time of the cited reference but have not been published previously in this form. The Monte Carlo results show excellent agreement in the photoelectric region (see Figure 6) but seem to become increasingly inaccurate in the Compton region. This unexpected characteristic of the Monte Carlo results defies explanation at this time but the effect will be investigated.

The important practical case of dose to the active marrow from broad beam incidence on a constantly rotating phantom is shown in Figure 7. Experimental results from Wilson and Carruthers (1962), Alun Jones (1964), and Facey (1968) may have suffered slight disfigurations due to replotting, but all appear to have been normalized to the same ordinate at 250 keV. Much concern has been expressed (Facey 1968) about whether marrow dose per unit exposure should increase monotonically with energy as noted by Wilson and Carruthers (1962) or whether it should peak at about 100 keV as noted by Alun Jones (1964). The different shapes have been considered due to energy degradation within the phantom and the fact that the detector systems of Alun Jones (1964) and Wilson and Carruthers had energy dependences in opposite directions (Facey, 1968).

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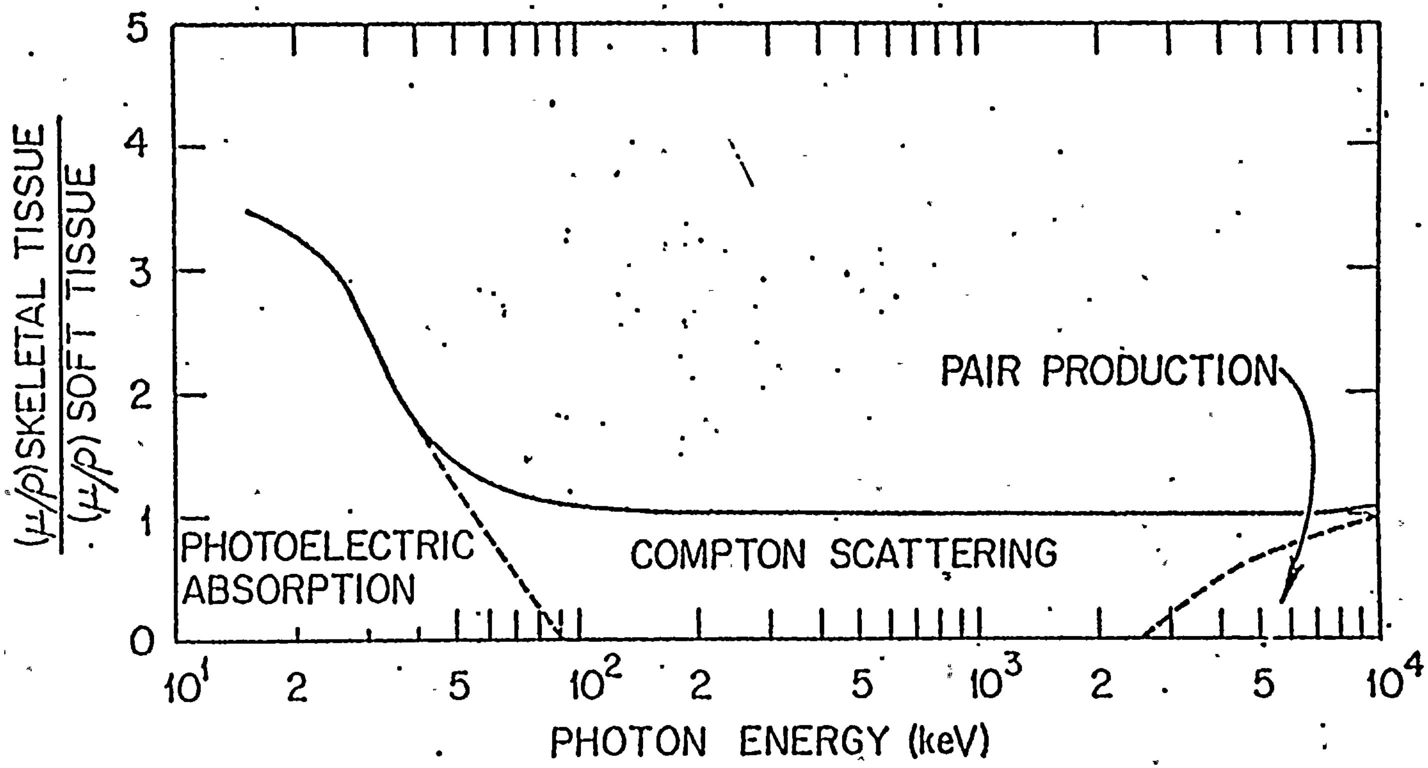

Fig. 6. Photon Attenuation of Skeletal Tissue Compared to that of Soft Tissue.

Table 2. Active Marrow Dose Relative to Dose at the Front of the Chest.

γ-ENERGY	$\bar{D}$ (MARROW)	D^{+} (CHEST)	$\bar{D}/D$ CHORD	$\bar{D}/D$ MONTE CARLO++
50 KEV	.26*	.48	.54	.54
100	.42	.57	.74	.68
250	1.1	1.47	.75	.47
660	2.7	3.61	.75	.50
1.25 MEV	5.3	6.14	.86	.55

*10^{-10} RADS/FLUENCE PHOTON

+T. D. JONES, HEALTH PHYSICS, 1973, VOL. 24, P. 248.

++CALCULATED AT TIME OF HEALTH PHYSICS, VOL. 24, P. 248, 1973, BUT UNPUBLISHED.

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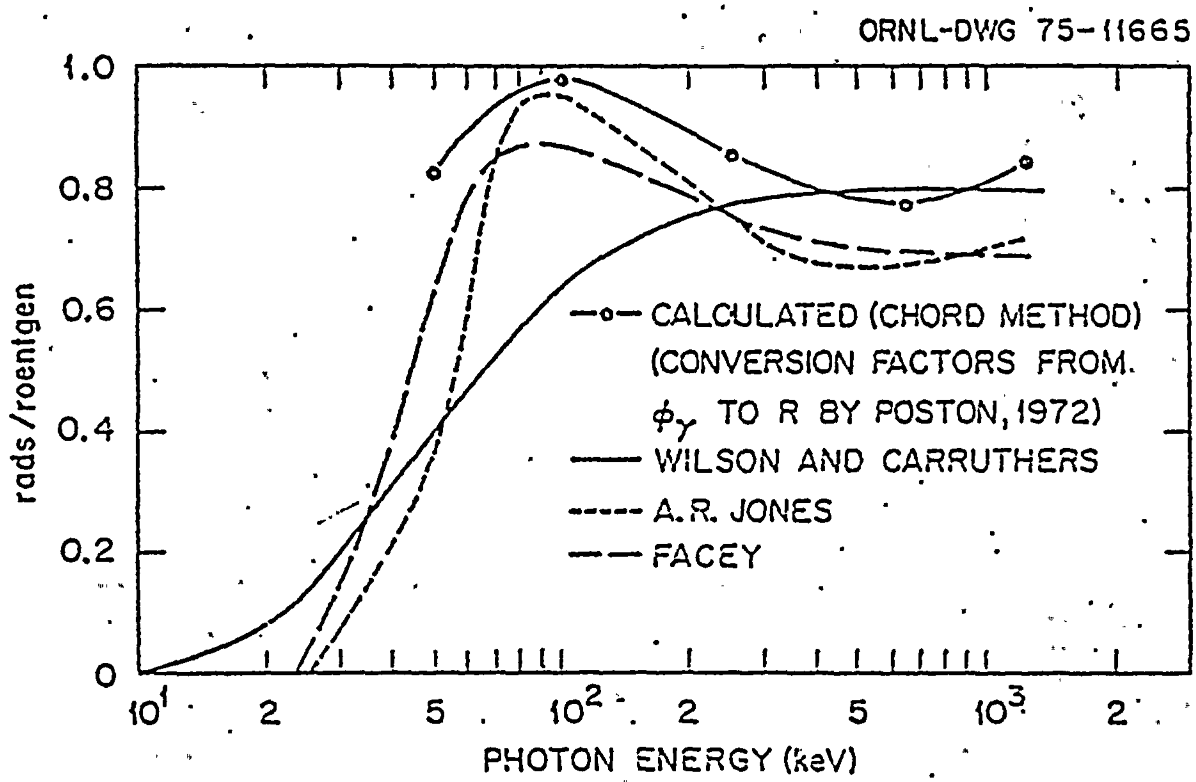

Fig. 7. Dose to Bone Marrow from a Broad Beam Incident on a Rotating Phantom.

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At this time, it seems more probable that the different shapes are due primarily to the fact that if one considers the shape of the curve describing the ratio of the photon fluence per unit exposure as a function of photon energy (Rad. Health Hbk, 1970; Fair, 1967) then the dose response curve must have a shape that peaks about 100 keV because the fluence per unit exposure varies more rapidly with energy than does the absorbed dose to the marrow, and secondarily to the fact that Wilson and Carruthers assumed that 60% of the active marrow received a dose similar to that measured in the thoracic vertebrae and 40% received a dose similar to that measured in the sternum*. The CHORD doses are in excellent agreement with Facey's results.(1968), except for a consistent 12% overestimation. This deviation is attributed to the facts that (a) 13.1% of the active marrow is in the skull (see Figure 2) which Facey did not include, (b) experimental results from Facey appear to have been normalized to other experimental results at 250 keV, (c) experimentally obtained doses to the active marrow system necessitate the assumption of an "effective mass center" of each important marrow region (Clifford and Facey, 1970)**, and (d) the CHORD estimate did not allow for increased attenuation by bone tissue shielding the marrow. As seen in Figure 6,

*This method of averaging would tend to underestimate dose at lower energies because as Facey (1968) points out, the "pelvis dominates dose at higher energies followed by the thoracic vertebrae and sacrum down to 30 keV. There the ribs enter second place and below 30 keV the ribs dominate." Facey (1968) attempted to resolve difficulties in the rotational case and his results are shown in Figure 7.

**For precision, this "effective mass center" would have to be "weighed" proportionally to dose variations in the local volume of interest; however, most experimenters appear to have used the mass centroid.

this effect is not large except for extremely low energies. At the low energies, dose to the shallow marrow becomes increasingly important, as is shown by the rapid attenuation of dose as a function of depth, and most experimental results are expected to be somewhat low because of the method of averaging... CHORD dose values were normalized per unit exposure according to the Rad. Health Hdbk. (1970)*. In spite of factors a, b, c, and d, excellent agreement for A-P estimates (A. R. Jones, 1964) and rotational estimates (Facey, 1968) compared with the method of CHORDs is observed. Figure 4, which shows the dose to the active marrow for exposure to monoenergetic photons, suggests that if one is concerned only about protection of his bone marrow, he should not do the instinctive thing and turn his back, but instead should face the hazard while backing away. The same effect was also observed by Piesch (1968) and holds for the neutron data in Table 3 which illustrates dose to the active marrow from exposure to monoenergetic neutrons. Some of the data in Table 3 are plotted in Figure 8 for ease of application. Bilateral and rotational results are not shown in Figure 8 because of their close agreement with the results for A-P exposure. Absorbed dose from neutron produced recoil ions is usually characterized by the hydrogen atomic density, because about 70% of the absorbed dose is due to interactions with hydrogen atoms for neutron energies below 14 MeV (Auxier, 1968; Jones, 1974). Standard soft muscle tissue contains about 10% by

*Poston's conversion values of fluence per unit exposure for the Reference Man tissue composition are, for all practical purposes, equal to those in the Rad. Health Handbook.

Table 3. Dose to Active Marrow from Neutron Produced Recoil Ions as Predicted by CHORD Distributions.

ENERGY	FREE-SPACE* KERMA	P-A**	A-P	BILATERAL	ROTATIONAL	ISOTROPIC
.025 eV	2.1	2.1	1.2	1.4	1.6	.70
1 keV	1.0	3.3	2.2	2.1	2.3	1.1
10 keV	10.	4.1	2.6	2.6	2.8	1.6
100 keV	70.	12.	7.4	9.4	9.2	5.4
1 MeV	230.	110.	67.	74.	75.	47.
2.5 MeV	340.	240.	180.	150.	190.	84.
14 MeV	690.	590.	520.	420.	540.	330.

* x 10^{-9} ERGS/(GRAM-FLUENCE NEUTRON)

** x 10^{-11} RADS/FLUENCE NEUTRON

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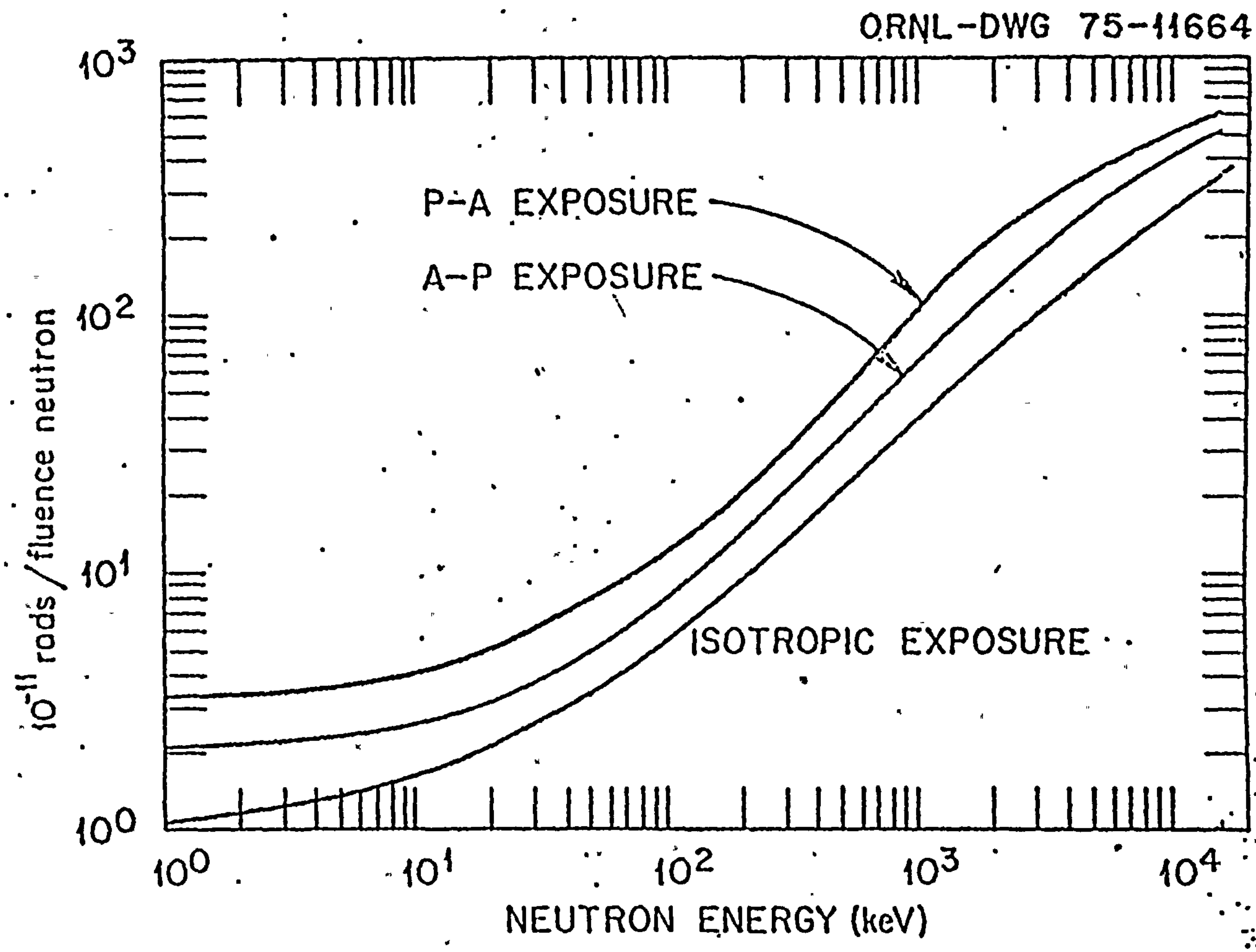

Fig. 8. Dose from Recoil Ions to the Active Marrow as Predicted by the CHORD Concept.

weight hydrogen and has a specific gravity of unity, while bone tissue contains about one-half the weight percentage of hydrogen as does muscle tissue but has about twice the specific gravity of muscle tissue so that the hydrogen atomic density is not very different for the two types of biological tissue. Lung tissue has a specific gravity of only about 0.3 and the hydrogen atomic density, therefore, is quite different; however, most critical organs of interest are either distant from the lung tissue or closer to an irradiated surface so that the penetration distance in grams/cm^2 is less than the other portion of the ray of travel that passes through a section of the lungs. Based on depth-dose curves from some of our previous calculations (Jones et al., 1973), it is believed that most regions of variable specific gravity do not significantly influence the application of the method of CHORDs, unless one is specifically interested in dose to a volume of lung tissue.

Other CHORD Applications

Figure 9 illustrates a proposed dosimeter or "riskmeter" in which the relative settings of the outer two dials select the appropriate CHORD distribution and the inner two dials select the insult (depth-dose) curve for the energy and type of incident radiation. Alun Jones (1966) suggested that dosimetry should be approached by matching variations in dose or risk with scattering, absorption, and attenuation; however, the CHORD method seems to permit this same precision of matching variability on a simplified macroscopic level.

Hopefully a schema such as incorporated into Figure 9 would render the absorbed dose index, D_I, and dose equivalent index, H_I, for

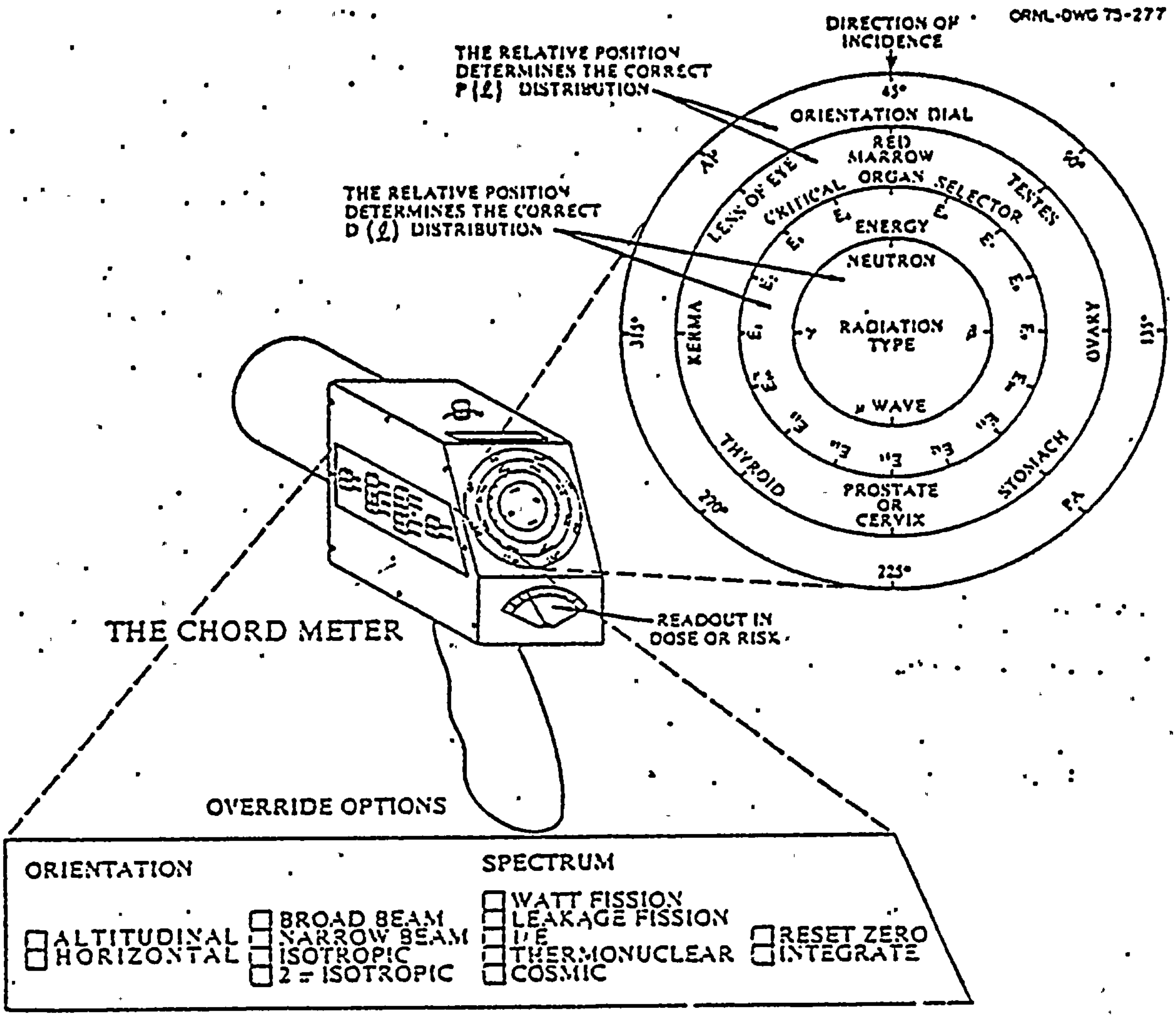

Fig. 9. Critical Human Organ Radiation Dosimeter.

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the standard ICRU 30 cm sphere (ICRU, 1971) even less useful than it already is, because by using CHORD density functions plus standard insult (multicollision depth-dose) curves, a health physicist or medical technician could easily and quickly estimate exposure values to any biological tissue at risk. It is also becoming apparent that significant calculational and experimental efforts will soon be directed to the estimation of tissue risk due to microwave irradiations and the availability of $p(\ell)\, d\ell$ distributions should be helpful.

Conclusions

In summary, the method of CHORDs permits rapid "critical organ" dose estimation and helps to circumvent some of the problems of relating organ dose or risk to readings from meters or film badges. A personal dosimeter measures exposure at the surface of the chest; the measured exposure corresponds neither to the exposure in free space nor to the organ or whole body dose and area dosimeters determine only free space exposure (Piesch, 1967). Alun Jones (1966, 1964) pointed out that a survey meter or personal dosimeter may overestimate the insult to the active marrow by a factor of 10 or underestimate by a factor of 6. In spatially dependant radiation fields, or for exposure to broad beam sources having an orientation other than A-P, it is usually very difficult to have an accurate risk estimate because of normalization to an inaccurate or shielded reading taken at the location of the chest.

Acknowledcments

This paper is heavily dependent upon the experimental work of Wilson, Carruthers, Facey and Alun Jones for the discussion of the results and as a means of estimating the validity of the CHORD concept. It is also necessary to acknowledge the helpful suggestions and data supplied by Alun Jones and J. W. Poston.

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9-89

THE EFFECTS OF SMALL DOSES OF IONIZING RADIATION: FUNDAMENTAL BIOPHYSICAL CHARACTERISTICS

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ABSTRACT

From an application of the concepts of microdosimetry to a wide range of radiobiological data on higher organisms, it has become apparent that the first step in the biological action of ionizing radiation is the induction of subcellular lesions. Two basic characteristics of this process are that it depends only on the first and (sometimes) the second power of the absorbed dose and that the yield of such lesions as well as the magnitude of the domain where energy concentration determines the yield of lesions is relatively constant even for cells and effects that differ greatly in radiosensitivity. These observations have led to the formulation of the Theory of Dual Radiation Action which postulates that the yield of these lesions depends on the square of the specific energy in domains having an effective diameter which differs from 1 μm by much less than an order of magnitude. It has furthermore been deduced that lesions are produced by the interaction of pairs of sublesions which are presumed to be alterations in DNA structure at the nanometer level.

There remain many questions regarding the quantitative relation of lesion production to cellular injury and the dependence of multicellular responses on cellular impairment. While these uncertainties make it frequently impossible to derive explicit dose-effect relations, the existing framework permits a variety of general conclusions and it may be utilized to obtain specific answers in some cases.

An important example are risk estimates for the induction of human leukemia by neutrons. It is concluded that maximum permissible neutron doses must be reduced.

Key phrases: Theoretical radiobiology; Radiation protection; Risk estimates.

Running Title: Fundamental small dose effects

Correspondence to:

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The ultimate objective of radiobiology must certainly be the identification of the intracellular and intercellular alterations that are initiated by ionizing radiation and progress to manifest injury. At present we are still far from this goal and in what is the most important effect for radiation protection - carcinogenesis - this is in a large measure due to our ignorance of the biological changes that underly what is probably a complex of diseases.

Although studies of the action of other carcinogens and biological advances in general may well be important to the achievement of greater insight into the cancer problem, radiobiology has the decided advantage that it can apply an effective agent to the genetic apparatus directly and accurately. This permits quantitative experimentation which is best carried out at low doses where secondary effects are likely to be less important.

The Dose-Effect Relation

An important intermediate goal of theoretical radiobiology would seem to be the establishment of the relations between dose[2] and effect which can not only furnish clues regarding the nature - and especially the kinetics - of radiobiological effects but also constitute the principal objective of radiation protection. In either application the precise values of the relation are of lesser importance than the shape of the dose-effect curve particularly at low values of the dose where extrapolations are of considerable scientific and pragmatic importance.

The shape of the dose-effect curve for individual cells can be stated with a certainty that increases with decreasing dose: It must ultimately become linear regardless of the energetics of cell inactivation and regardless of variations in sensitivity of the individual cells in the irradiated population. This conclusion follows from the simple fact that at sufficiently low doses the traversal of a cell by a charged particle is a rare event and the probability of multiple traversals is negligible. Under these conditions the dose-effect relation must be linear and any effect under consideration must occur with a probability which is in turn the product of the probability that a particle causes the effect and the probability that a cell is traversed by a particle at the dose under consideration. This statement can be formally supported by a very general proof (1) and microdosimetric data (2) may be used to determine traversal frequencies. It can be shown that the linear dependence must for low LET radiation extend up to doses of at least a few hundred millirads. For neutrons of moderate energy this limit is of the order of tens of rads.

These considerations can be applied only to those biological effects which arise from individual non-interacting cells. Thus if carcinogenesis were to require the transformation of a group of contiguous cells (3), one would expect a dose-effect curve exhibiting a positive curvature. While this does not seem to have been extablished at low doses, the reverse condition of negative curvature has been clearly demonstrated for the induction of mammary neoplasms in the Sprague-Dawley rat (4). Thus in at least one instance, carcinogenesis cannot be interpreted in terms of a simple somatic mutation which results in cancer regardless of the irradiation of other cells. Although the argument applied above to single cells can be extended to whatever group of cells might be involved, it loses its practical significance if this group comprises more than a few cells since extremely minute doses are required to limit the collective traversal probability for all cells to a value that is much less than one.

The Dose-RBE Relation

Although collective effects on cells can thus affect the shape of dose-effect curve they are far less likely to influence the dose-RBE curve which is obtained when the RBE of a high LET radiation is compared at various doses of either radiation (i.e. at various levels of effect). The reason for this is presumably that the interaction between cells is the same regardless of radiation quality and that even in complex systems the dependence of RBE on dose primarily reflects differences between the radiations under comparison in their kinetics of the impairment of individual constituent cells.

Figure 1a shows the dependence of RBE on the dose of 0.43 MeV neutrons for the mammary neoplasms in the Sprague Dawley rat. The bars cover RBE values that are excluded with 99% confidence. (The arrowheads correspond to lesser levels of confidence). Figure 1b shows the dose-RBE relation for the same radiation but for opacification of the murine lens over a thousand-fold range of dose (5). The statistical analysis employs non-parametric methods developed by Kellerer and Brenot (6). In either case the RBE increases over a wide range of doses as the inverse of the square root of the neutron dose (as indicated by the slope of -1/2) to values in excess of 100. This indicates that the biological effectiveness of low LET radiation increases as the square of dose (2).

Figure 2 contains the curves in Figure 1 as well as others which are not based on the non-parametric analysis but are nevertheless considered to be of sufficient accuracy (of the order of perhaps ± 30%). The radiations are 0.43 MeV neutrons and "fission neutrons." The latter classification is not very specific since the energy spectrum of the neutrons reaching the biological material must depend on variable and often uncertain moderation of the primary fission spectrum. A calculation of the dose mean lineal energy ($\bar{y}_D$ in a 2 μm tissue sphere) indicates a value of 65.5 KeV/μm for the spectrum in a reactor irradiation facility utilizing a moderated converter.[3] This particular value differs little from that for 0.43 MeV neutrons which is 60.5 KeV/μm (7).

All of the curves in Fig. 2 exhibit the characteristic slope of -1/2 although two indicate constant RBE at lower doses. Curve 2, which represents a somatic plant mutation, levels out near the value postulated by an elementary application of the theory of dual radiation action (8). Higher RBE values may be due to differences in the yield of sublesions (9) while possible levelling at lower RBE values as suggested by curve 3, which relates to survival of cells in tissue culture (10) could be caused by a linear component of radiation action (2). The other curves show no evidence of a change in slope but are limited to comparatively high doses. They are based on the results of two determinations for chromosome aberrations in human lymphocytes (11), (12). The former determination (curve 4) was in vitro at a reactor source, the latter is based on late effects in atomic bomb survivors (curve 5). The RBE values are similar. There is in fact far more variation in RBE with dose than there is between the various systems.

Principal Postulates

The above considerations may be summarized as follows:

1) It can be shown on the basis of elementary microdosimetric considerations that, at extremely low doses, the direct effect of ionizing radiation on individual cells must be proportional to the dose and to the probability that a single charged particle affects the cell. It has been observed that this probability differs between low LET radiation and neutrons

having energies of the order of a few hundred KeV by a factor that is larger than 10 and can exceed 100.

2) At intermediate doses, the RBE declines because the effectiveness of low LET radiation increases as the square of the dose.

3) The dose-effect relation for cell systems can not be deduced from microdosimetric considerations but the dose-RBE relation may be expected to be the same as that for individual cells.

These postulates are insufficient to furnish general answers to most of the primary questions in radiation protection. However, when applied to epidemiological data, they yield significant information.

Application to Radiation Leukemogenesis

One of the principal late radiation effects is leukemia which occurs with clearly increased incidence in the heavily irradiated population groups in Nagasaki and Hiroshima. While in the former city, the radiation consisted almost exclusively of gamma radiation, there was a substantial component of neutrons in the latter.

ordinate is ... by a factor of 10^{-2}

Fig. 3 shows the logarithm of incidence over the period from 1950-1966 (13) after subtraction of the incidence in the group that was assumed to have negligible kerma[4] (less than 5 rad). The abscissa is the logarithm of total tissue kerma (gamma plus neutron) in free air. The uncertainty is such that in either case, a linear relation (i.e. a line of slope 1) can not be rejected (14).

In an application of the non-parametric method illustrated in Fig. 1, it could be shown (15) that a constant ratio of biological effectiveness (as based on total kerma) could be rejected at a significance level of 86% and it was concluded that the Japanese leukemia data do not constitute an exception to the general postulate 3) given above.

The analysis indicates that a total kerma of 10 rad at Hiroshima had approximately the same leukemogenic potential as 70 rad at Nagasaki. The RBE of neutrons relative to gamma rays is much larger because at Hiroshima only one fifth of this total kerma was due to neutrons. In addition, the body tissues surrounding active marrow attenuate neutrons more effectively than gamma radiation, the factors being 0.26 and 0.55 (16). It follows that if the dose to the bone marrow is about 0.5 rad of neutrons and about 4 rad of gamma rays, the same leukemia incidence results as from a gamma dose of about 35 rad. Even if a linear dose-effect relation is attributed to gamma radiation, it is apparent that at low doses, the leukemogenic effect in Hiroshima was almost entirely due to neutrons.

Further study has indicated (15) that if the dose-effect curve consists of a linear and a quadratic term, the former is negligible for gamma radiation and the latter is negligible for neutrons. On the basis of this finding, it has been concluded that if Dn and D_γ are respectively the mean doses of neutrons and gamma rays to the bone marrow, the annual incidence of leukemia (as averaged over 16 years) is about $5 \times 10^{-5} D_n$ and $1.8 \times 10^{-8} D_\gamma^{\,2}$ for intervals of 0.5 to 10 rads of Dn and 2.5 to 50 of D_γ. These relations are plotted in Fig. 4 and the corresponding RBE relation is curve 6 in Fig. 2. The latter appears to be a continuation of the line for chromosome aberrations.

A recent analysis (17), using a somewhat different approach, yields $3 \times 10^{-5} D_n$ for the neutron curve. However, the difference is almost entirely due to the longer averaging period.

Also shown in Fig. 4 are the spontaneous leukemia rates at the two cities and the maximum permissible average annual occupational dose (MPD) for neutron and gamma radiation. If - in line with current thinking -

it is assumed that, at least at low doses, there is little effect of neutron dose rate on biological effect; it becomes apparent that one occupational MPD of neutrons will for some 16 years result in a risk rate that is essentially equal to the natural rate. This seems excessive and a sharp reduction of the MPD for neutrons seems indicated. This might best be accomplished by an increase of the quality factor (Q). On the other hand, no changes seem necessary for the MPD of low LET radiation which is of far greater practical importance.

It will be noted that Fig. 4 applies to the dose in the bone marrow. If limits are applied on the basis of kerma or of the absorbed dose index, a safety factor of the order of 4 applies.

Conclusions

Both theoretical radiobiology and radiation epidemiology are beset by limitations; the former because of the uncertain generalizations which are the essence of induction; and the latter because of the possibility of uncontrolled variables and dosimetry errors. These concerns must be markedly reduced if theory and observation yield concordant results. The validity of the least square fit to the epidemiological data seems more assured because it results in a dose-RBE relation that is similar to that observed in all other systems subjected to analysis, and the accuracy of dosimetry seems to be demonstrated by the fact that the absolute value of the curve parameters is also close to that for the other systems. It seems likely that fundamental biophysical considerations will thus continue to be of practical utility in addition to providing the basis for our comprehension of radiobiological mechanisms.

FOOTNOTES

1. Presentated at the joint meeting of the Radiation Research Society and the Health Physics Society, San Francisco, 1976. This investigation was supported by Contract E(11-1)3243 from the Energy Research and Development Administration and Grants CA 12536 and CA 15307, awarded by the National Cancer Institute, DHEW.

2. Throughout this presentation the shorter term "dose" replaces the more accurate "absorbed dose."

3. Teedla, Peeter: unpublished data

4. In the following "kerma" will, for brevity's sake, replace "tissue kerma in free air."

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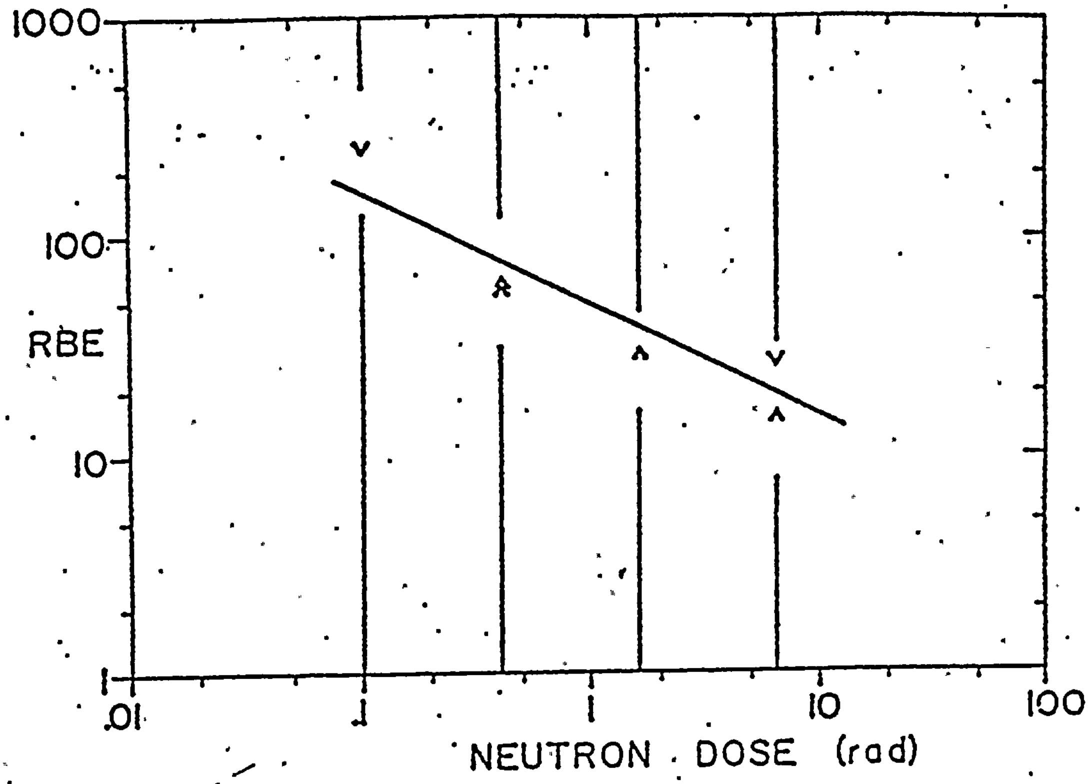
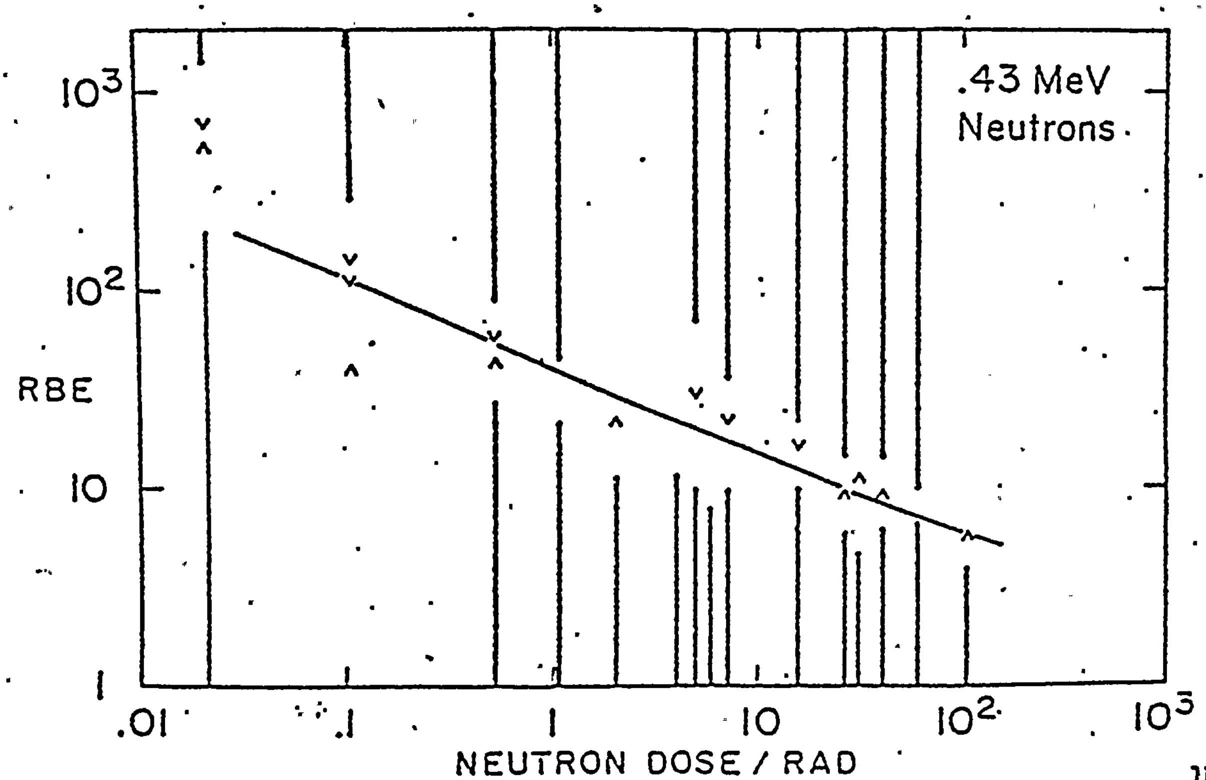
ACKNOWLEDGMENTS

The author is indebted to Dr. Eric J. Hall for helpful discussions and to Mr. Peeter Teedla for calculations.


1000
100
RBE
10
1
.01
.1
1
10
100
NEUTRON DOSE (rad)

1a


.43 MeV
Neutrons
10^3
10^2
RBE
10
1
.01
.1
1
10
10^2
10^3
NEUTRON DOSE / RAD

1b

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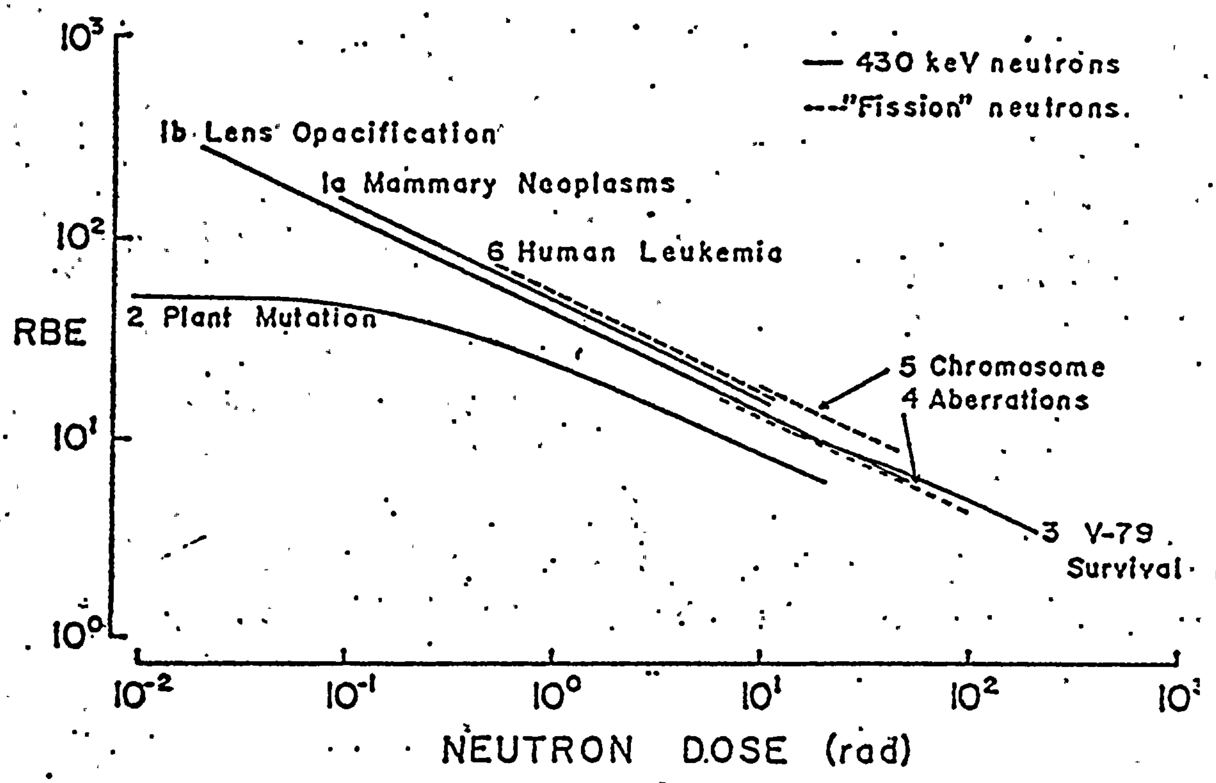
FIGURE 2



— 430 keV neutrons
---"Fission" neutrons.
Ib Lens Opacification
Ia Mammary Neoplasms
6 Human Leukemia
2 Plant Mutation
5 Chromosome
4 Aberrations
3 V-79
Survival
RBE
10^3
10^2
10^1
10^0
10^{-2}
10^{-1}
10^0
10^1
10^2
10^3
NEUTRON DOSE (rad)

Harald H. Rossi

FIGURE 3

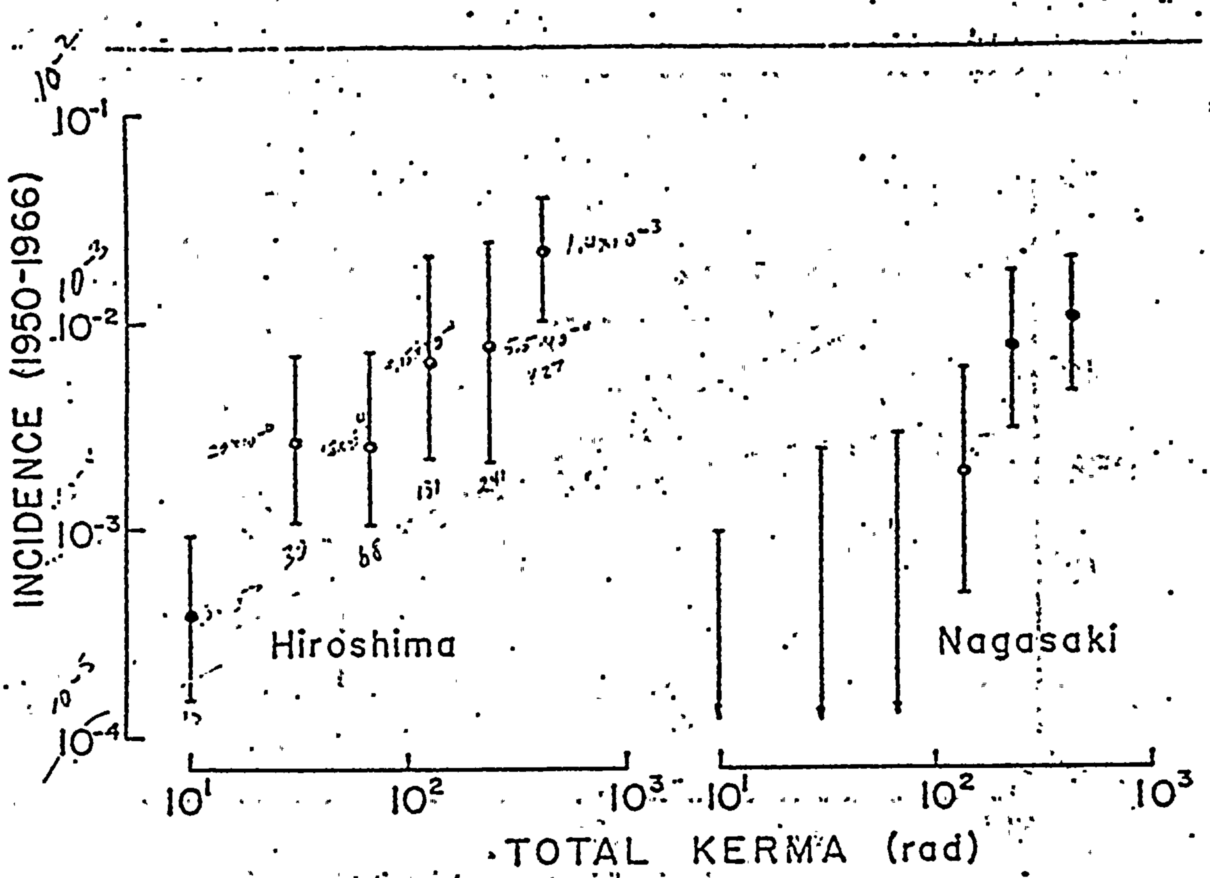
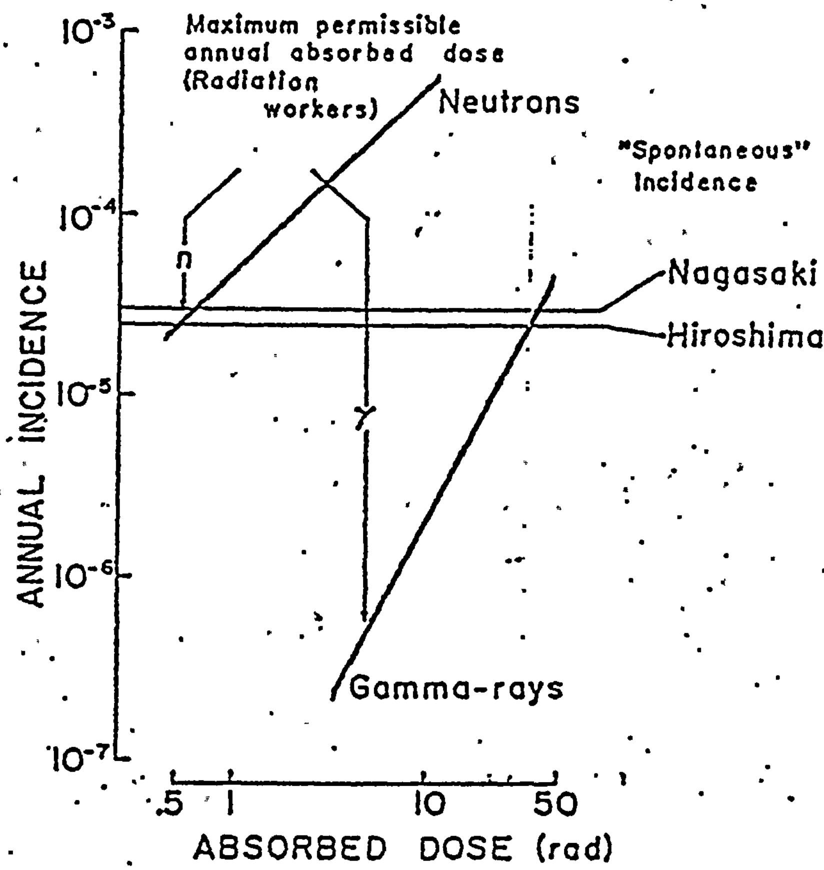


FIGURE 4


Maximum permissible
annual absorbed dose
(Radiation
workers)
Neutrons
"Spontaneous"
Incidence
n
γ
Nagasaki
Hiroshima
Gamma-rays
ANNUAL INCIDENCE
10^{-3}
10^{-4}
10^{-5}
10^{-6}
10^{-7}
.5
1
10
50
ABSORBED DOSE (rad)

LEGENDS TO FIGURES

Fig. 1 Logarithmic representation of RBE of 0.43 MeV neutrons relative to x-rays vs. neutron dose. The bars represent RBE values excluded with 95-99% confidence by a non-parametric analysis (6).

Fig. 1a Induction of mammary neoplasms in the rat.

Fig. 1b Opacification of the murine lens.

Fig. 2 Logarithmic representation of RBE vs. neutron dose 0.43 MeV neutrons and "fission" neutrons for a variety of effects. For details, see text.

Fig. 3 Incidence of all types of leukemia for the period 1950-1966 in Hiroshima and Nagasaki vs. total kerma (13). The incidence in the 0 - 5 rad group has been subtracted.

Fig. 4 Annual incidence of leukemia vs. absorbed dose to the bone marrow as deduced from the Japanese data. The graph also shows the "natural" incidence at Hiroshima and Nagasaki and the maximum permissible annual absorbed doses for radiation workers.

Enclosure 13

NUREG-0179

REGULATORY AND OTHER RESPONSIBILITIES AS RELATED TO TRANSPORTATION ACCIDENTS

UNITED STATES NUCLEAR REGULATORY COMMISSION

Office of Standards Development
U. S. Nuclear Regulatory Commission

9 - 109

May 12, 1978

Edward Luton, Esq., Chairman
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. Emmeth A. Luebke
Atomic Safety and Licensing Board
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In the Matter of
Rochester Gas & Electric Corporation
(R. E. Ginna Nuclear Power Plant, Unit No. 1)
Docket No. 50-244

Gentlemen:

Enclosed are copies of the "Safety Evaluation Report on Interim ECCS Evaluation Model For Westinghouse Two-Loop Plants" (March 1978) and the cover letter of May 1, 1978 to Rochester Gas and Electric Corporation from D. L. Ziemann, Chief, Operating Reactors Branch #2, Division of Operating Reactors.

These documents are submitted to the Licensing Board in keeping with the NRC Staff's policy of keeping Boards informed of matters pending before them. An ECCS contention is presently pending before this Board.

Sincerely,

Edward G. Ketchen
Counsel for NRC Staff

Enclosure: As Stated

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Mr. Michael Slade
Robert E. Lee, Ph.D
Jeffrey Cohen, Esq.
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Atomic Safety and Licensing Board Panel
Atomic Safety and Licensing Appeal Board
Docketing and Service Section

E.K

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[illegible]

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

May 1, 1978

Docket No. 50-244

Rochester Gas and Electric Corporation
ATTN: Mr. Leon D. White, Jr.
Vice President
Electric and Steam Production
89 East Avenue
Rochester, New York 14604

Gentlemen:

Our letter dated December 16, 1977 provided our safety evaluation report on the ECCS evaluation model for Westinghouse two-loop plant. On the basis of that report you were requested to provide within 30 days appropriate bases, including any necessary operating limitations, to justify continued operation of R. E. Ginna Nuclear Power Plant beyond this 30 day period. Your letter dated January 16, 1978 provided a response to this request. By letter dated February 10, 1978, we requested additional information. You responded to this request by letter dated February 15, 1978.

Our attached safety evaluation concludes that the calculations provided by your letter of February 15, 1978 provided an acceptable basis for continued operation of the R. E. Ginna Nuclear Power Plant while long-term efforts continue to develop an ECCS evaluation which specifically treats upper plenum injection. This evaluation demonstrates that, for the Ginna plant, specific consideration of upper plenum injection water interaction with core generated steam, using acceptable modifications of the model described in our November 1977 SER, results in an increase in calculated peak clad temperature of only 15°F (for the 120% ANS decay heat case) over the temperature resulting from prior calculations based on the Westinghouse model.

We acknowledge receipt of your most recent submittal dated March 15, 1978, which responds to that portion of our letter of December 16, 1977, which requested that you provide within 90 days a permanent resolution (and a schedule for its implementation) to staff concerns about upper plenum injection of emergency core cooling system water. Your proposal is consistent with the recommendations contained in the staff's March 1978 SER attached.

We look forward to working with you on the long-range effort to develop an acceptable ECCS model which specifically treats upper plenum injection.

Sincerely,

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosure:
Safety Evaluation

cc: See next page

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SAFETY EVALUATION REPORT

ON

INTERIM ECCS EVALUATION MODEL

FOR

WESTINGHOUSE TWO-LOOP PLANTS

March 1978

Introduction

On December 16, 1977 the licensees of West

plants were sent letters from Mr. Case, wh

an interim basis for continued safe operat

relative to the effectiveness of the two-l...

16, 1978 each licensee provided essentially ...

basis for continued safe operation. The pur...

to provide a safety evaluation of the propose...

analysis as presented in those letters.

Summary of Review

The January 16 licensee letters provided their eval...

effectiveness of the ECCS Upper Plenum Injection ...

postulated LOCA. Use was made of a staff model descr...

"Safety Evaluation, ECCS Evaluation Model for Westin...

Plants," December, 1977. However there were six chan... the

staff model by Westinghouse. The staff SER model acco...

effects of Upper Plenum Injection by estimating the ...

generation; steam condensation; and liquid entrainm... ...ced

rate and the associated change in calculated peak cla... when

the staff model was generated, it was intended to be ...ative

for evaluating Upper Plenum Injection performance d... LOCA

or if necessary, for establishing an interim basis f...

two-loop plants. The staff model, a simplified model ...

as a hand calculation, was an attempt to approximate ... interim

period.. It was not (and still is not) an ECCS Evalu...

fully complied with 10 CFR 50 Appendix K. The staff ...

model with approved changes, could possibly be used as a basis for establishing interim operating conditions. However, neither model is acceptable for long term use. For example, the use of a decay heat model of 1.0 x ANS decay heat might be acceptable on an interim basis (i.e., to determine if there is a safety problem) but is not suitable for a long term evaluation model. Part 50.46 requires that, "ECCS cooling performance be calculated with an acceptable evaluation model...". Appendix K sets forth certain required features of an acceptable evaluation model including the requirement that, "the refilling of the reactor vessel and the time and rate of flooding of the core shall be calculated by an acceptable model that takes into consideration the thermal and hydraulic characteristics of the core and of the reactor system". Neither the Staff model nor the proposed Westinghouse variation is an integral model for the evaluation of a postulated LOCA. Instead, each provides a possible adjustment to be used together with the "incorrect" old LOCA calculation. The documentation and sensitivity studies required of an evaluation model are also absent. Most important is the lack of rigor in the staff approximate method; it was not subjected to the same scrutiny that we demand for long-term generic use.

Each of the six changes proposed by the licensees to the model has been evaluated to determine the acceptability of the Westinghouse calculations. The following description of the first change is taken from the Attachments to the owner's letters.

1. "The clad temperature rise versus flooding rate curve, Figure 24 in the SER, was replaced by a more realistic curve. The new curve was based on the Westinghouse design FLECHT correlation with input more specific to the Westinghouse two-loop plants".

Evaluation:

The SER curve is based on the most conservative data from the reflood rate sensitivity studies presented in the PWR FLECHT Final Report Supplement, WCAP-7931. The Westinghouse calculation takes credit for the calculated pressure, subcooling and linear heat rate in establishing the relationship between peak clad temperature and reflood rate. Based on our review of the actual input values used and the method of implementation, this change is acceptable. The second change is described as follows:

2. "The input was changed to allow transient input for pressure, injection rates, flooding rates and decay heat".

Evaluation:

The most important portion of the reflood transient occurs between 60 seconds and 100 seconds. The time dependent input for decay heat allows approximately a 10% reduction over this time span. The SER model is based on a constant decay heat, with the value determined at the beginning of a reflood. Since this change only involves more detailed input, it is acceptable to the staff. The third change is described as follows:

3. "The carryover fraction, CRF, discussed on page 40 of the SER, was changed from 0.8 in the staff model to 0.7 in the Westinghouse

model. Carryover fractions of 0.7 are more typical of the two-loop plants".

Staff calculation of the carryover fraction, CRF, during reflood for a two-loop plant with upper plenum injection range in value from 0.6 to 0.8 as a function of time.

The carryover rate fraction, CRF, appears in two different forms in the staff model. It appears in the quench front progression equation as (1-CRF) where the value of (1-CRF) is .3. This agrees with the suggested value of CRF = .7 from Westinghouse. The carryover rate fraction is also included as one of the components of the constant which is used to characterize the relationship between changes in bottom quench front steam and water flows, and the flooding rate. In the staff evaluation model the system resistance to flow establishes the total steaming rate out the break during reflood. This steaming rate determines the reflood rate (Vin) according to the following equation:

$$\text{Vin} \times \text{CRF} \times \text{Area} \times \text{Liquid Density} = W_{TOTAL}$$

Changes in the total bottom quench front steam and water flow (W_{TOTAL}), and the reflood rate are therefore related by the following perturbation equation:

$$\Delta \text{Vin} = \Delta W_{TOTAL} / (\text{CRF} \times \text{Area} \times \text{Density})$$

The staff model includes a value of CRF of .8 in this estimate of the system flow resistance.

The use of a constant CRF of 0.7 appears to be consistant with the CRF value for the two-loop plant evaluation model calculations and therefore the use of a 0.7 value is appropriate. The fourth change is described as follows:

4. "The bottom quench front in the staff model was initialized at 0.0 feet. Since this calculation starts some 20 seconds into reflood, the Westinghouse model initiates the bottom quench front at 1.5 feet which is a lower bound value from the Westinghouse ECCS Evaluation Model results".

The SER model was initiated at 60 seconds because this is the time at which the reflood rate calculated with the present evaluation model for the worst break becomes a well behaved and smoothly varying function of time. Prior to 60 seconds the calculated reflood rate varies dramatically as the bottom reflood water first rushes into the core relatively unimpeded and then generates a large amount of steam which causes the reflood rate to drop sharply. The presence of Upper Plenum Injection would significantly alter this initial phase of reflood in a way that the staff's relatively simple, perturbation technique could not accurately represent. Since the upper plenum injection begins at 26 seconds in the evaluation model calculation, for the worst break, significant steam generation from this water would be occuring when the bottom reflood water reached the core. The upper plenum steam generation would lessen the initial rush of water into the core because of the increased steam binding effects.

The initial phase of bottom reflood would therefore be less dramatic in the variation in the reflood rate. The Staff model effectively assumes a smooth and well behaved reflood from the bottom of core recovery. The staff model includes a simple treatment of this initial phase of reflood with upper plenum injection. The proposed change by Westinghouse would not consider any effects of upper plenum injection prior to the reflood level reaching 1.5 feet. Although the staff SER model could be improved in this area, the Westinghouse change does not appear to increase the accuracy of the representation and is clearly in a non-conservative direction. This change is therefore unacceptable at the present time.

The fifth change is described as follows:

5. "The heat transfer model, described on page 37 of the SER, was altered to account for the amount of heat transfer in the unquenched region which is going to the bottom generated steam rather than the top generated steam. This was done by reducing the heat transfer to the top generated steam by 25 percent. This is a conservative lower bound".

The staff SER model assumes two predominant sources of steam:

1. The bottom quench front progression; and,
2. The steam generation due to upper plenum injection water entering the core from above.

The bottom quench front steam was assumed to be carrying a significant amount of water so that the total steam and water from the bottom quench front equalled the carryover function times the reflood rate. Since each pound of steam from the bottom quench front was already carrying on the order of three pounds of water, this steam was not inlcuded in the upper

plenum injection entrainment correlation. The steam generated from the top quench front and from heat transfer to the upper plenum injection water in the unquenched portion of the core was input into the entrainment correlation. The Westinghouse change suggests that three sources of steam provide a better representation of the reflood steam generation. One source of steam is at the bottom quench front; a second source is the top quench front and steam generated by the top injection water entering the core; and the third is the vaporization of water carried up from the bottom quench front. The Westinghouse model therefore proposes to identify two separate sources of steam in the unquenched portion of the core. This is acceptable and in fact may be a more accurate representation. However, the proposed model change does not include the steam generation in the unquenched region from the bottom quench front water as input to the entrainment correlation. The basis for not including the bottom quench front steam in the entrainment is that this steam is already carrying a significant amount of water. No basis has been provided for not including the steam generation in the unquenched region of the core from the bottom quench front water in the entrainment correlation. This proposed change is therefore unacceptable as presented. A modified change which included all non-bottom quench front steam in the entrainment correlation could be acceptable. The sixth proposed change is stated as follows:

6. "The metal heat model was altered to take into account the finite amount of heat stored in the upper plenum metal. The heat capacity of the upper plenum metal is 5930 (BTU/°F). This metal energy is removed in a finite period of time after which no energy is added to the

fluid from the metal resulting in increased subcooling for the remainder of the transient".

The staff SER model uses a simple constant heat input model for the heating effect in the upper plenum. The concept of a finite stored energy model is acceptable. The basis for establishing the initial stored energy and the heat release rate has been reviewed and is sufficiently conservative for use in the interim calculations. This proposed change is therefore acceptable.

Since two of the Westinghouse proposed changes were found to be unacceptable, the staff letters, of February 10, 1978 to the Two-Loop Licensees, formally requesting additional information included a request for new calculations in which the unacceptable proposed changes were removed. Table I presents the results of these calculations for both 100% ANS decay heat and 120% ANS decay heat.

Staff Findings

The following conclusions are based on our review of the information presented by the two-loop plant owner-operators.

First, the calculations performed with the proposed changes 1, 2, 3 and 6 are acceptable as an interim basis for continued safe operations of the Westinghouse two-loop plants. Although some of the calculations result in increases in peak clad temperature, none results in a peak clad temperature greater than 2200°F.

Second, the long term effort to produce an acceptable ECCS evaluation model for treating Upper Plenum Injection should continue unless the two-loop plant owners propose to modify the ECCS hardware to eliminate Upper Plenum Injection.

Table I

Upper Plenum Injection Results

PLANT	CURRENT WESTINGHOUSE EVALUATION MODEL ANALYSIS F_q	PEAK CLAD TEMPERATURE	NEW U.P.I. ANALYSIS* F_q	PEAK CLAD TEMPERATURE 1.0 ANS Decay Heat	PEAK CLAD TEMPERATURE 1.2 ANS Decay Heat
WEP/WIS	2.32	1965	2.32	1945 (-20)	2025 (+60)
RGE	2.32	1957	2.32	1900 (-57)	1972 (+15)
NSP/NRP	2.32	2187	2.32	2110 (-77)	2177 (-10)
WPS	2.25	2172	2.25	2090 (-82)	2162 (-10)

*With Unacceptable Proposed Changes Deleted.

[illegible]

May 10, 1978

Edward Luton, Esq., Chairman
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dr. Emmeth A. Luebke
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dr. Franklin C. Daiber
College of Marine Studies
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Newark, Delaware 19711

In the Matter of
Rochester Gas & Electric Corporation
(R. E. Ginna Nuclear Power Plant, Unit No. 1)
Docket No. 50-244

Gentlemen:

Enclosed are copies of Amendment No. 19 to Provisional Operating License No. DPR-18, and supporting safety evaluations, and an exemption from the requirements of 10 CFR §50.46(a)(1) for the R. E. Ginna Nuclear Power Plant. I have also enclosed a copy of the Safety Evaluation Report, An Interim ECCS Evaluation Model For Westinghouse Two-Loop Plants (March, 1978), which treats upper plenum injection.

These materials are submitted to the Licensing Board in keeping with the NRC Staff's policy of keeping Board's informed. An ECCS contention is presently pending before this Board.

Sincerely,

Edward G. Ketchen
Counsel for NRC Staff

Enclosures
As Stated

cc w/encl: Leonard M. Trosten, Esq.
Mr. Michael Slade
Robert E. Lee, Ph.D
Jeffrey Cohen, Esq.
Warren B. Rosenbaum, Esq.
Atomic Safety and Licensing Board Panel
Atomic Safety and Licensing Appeal Board
Docketing and Service Section

Dist
NRC Central
LPDR(2)
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Engelhardt
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Scinto
Reis
Ketchen
Chron(2)
FF(2)
HSmith
TWamback
ASchwencer
J. Shea

OFFICE➤				OELD	OELD
SURNAME➤				Ketchen/dmr Reis	Scinto Grossman
DATE➤				5/5/78	5/ /78

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

May 1, 1978

Docket No. 50-244

Rochester Gas and Electric Corporation
ATTN: Mr. Leon D. White, Jr.
Vice President
Electric and Steam Production
89 East Avenue
Rochester, New York 14649

Gentlemen:

The Commission has issued the enclosed Amendment No. 19 to Provisional Operating License No. DPR-18 and an Exemption from the requirements of 10 CFR 50.46(a)(1) for the R. E. Ginna Nuclear Power Plant.

The amendment consists of changes to the Technical Specifications in response to your application dated January 6, 1978, as supplemented by letters dated January 10, 1978, March 27, 1978, April 6, 1978, April 17, 1978, and April 25, 1978. We have recently noted that your January 6 application, which was received by the NRC on January 9, 1978, was actually dated January 6, 1977.

The amendment incorporates changes to the Appendix A Technical Specifications to support operation in Cycle 8 with reload fuel by Exxon Nuclear Company (ENC). This fuel has been designed by ENC to be compatible to the fuel supplied previously by Westinghouse. In addition, the amendment allows Technical Specification changes that are required for startup tests.

The Commission has also concluded that your ECCS analysis utilizes upper head fluid (hot leg) temperature and therefore satisfies the provision set forth in the Commission's Order for Modification of License dated August 27, 1976, without changes to the Technical Specifications.

Notice of proposed Issuance of Amendment to Facility Operating License in connection with the license amendment action was published in the Federal Register on February 21, 1978 (43 FR 7275).

In response to your request dated April 25, 1978, we have granted an Exemption from the requirements of 10 CFR 50.46(a)(1) that ECCS performance be calculated in accordance with an acceptable calculational model which conforms to the provisions in Appendix K, without the errors contained in the analyses previously submitted to the Commission. On March 23, 1978, Westinghouse provided the Commission an oral notification related to these errors.

Copies of the Safety Evaluation related to the license amendment, the staff's Safety Evaluation Report dated April 18, 1978, related to the Exemption and Notice of Issuance of License Amendment are also enclosed. The Exemption and the Notice are being forwarded to the Office of the Federal Register for publication.

Sincerely,

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 19 to
   License DPR-18
2. Safety Evaluation
3. Exemption w/Safety Evaluation
   dated 4/18/78
4. Notice

cc w/enclosures:
See next page

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

ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NO. 50-244

R. E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 19
License No. DPR-18

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Rochester Gas and Electric Company (the licensee) dated January 6, 1978, as supplemented by letters dated January 10, 1978, March 27, 1978, April 6, 1978, April 17, 1978, and April 25, 1978, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Provisional Operating License No. DPR-18 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 19 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Darrell G. Eisenhut

Darrell G. Eisenhut, Assistant Director
for Systems & Projects
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance May 1, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 19

PROVISIONAL OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Change the Technical Specifications contained in Appendix A of License No. DPR-18 as indicated below. The revised pages contain the captioned amendment number and marginal lines to reflect the area of change.

Remove	Insert
3.10-2	3.10-2
-----	3.10-2a
3.10-4	3.10-4
3.10-8c	3.10-8c

3.10.1.2 When the reactor is critical except for physics tests and control rod exercises, the shutdown control rods shall be fully withdrawn.

3.10.1.3 When the reactor is critical, except for physics tests and control rod exercises, each group of control rods shall be inserted no further than the limits shown by the lines on Figure 3.10-1 and moved sequentially with a 100 ($\pm$5) step overlap between successive banks.

3.10.1.4 During control rod exercises indicated in Table 4.1-2, the insertion limits need not be observed but the Figure 3.10-2 must be observed.

3.10.1.5 The part length control rods will not be inserted except for physics tests or for axial offset calibration performed at 75% power or less.

3.10.1.6 During measurement of control rod worth and shutdown margin, the shutdown margin requirement, Specification 3.10.1.1, need not be observed provided the reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion and all part length control rods are fully withdrawn. Each full length control rod not fully inserted, that is, the rods available for trip insertion, shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the shutdown margin to less than the limits of Specification 3.10.1.1. The position of each full length rod not fully inserted, that is, available for trip insertion, shall be determined at least once per 2 hours.

3.10.2 <u>Power Distribution Limits and Misaligned Control Rod</u>

3.10.2.1 The movable detector system shall be used to measure power distribution after each fuel reloading prior to operation of the plant at 50% of rated power to ensure that design limits are not exceeded.

If the core is operating above 75% power with one excore nuclear channel out of service, then the quadrant to

3.10.2.4 If the quadrant to average power tilt ratio exceeds 1.02 but is less than 1.12 for a sustained period of more than 24 hours without known cause, or if such a tilt recurs intermittently without known cause, the reactor power level shall be restricted so as not to exceed 50% of rated power. If the cause of the tilt is determined, continued operation at a power level consistent with 3.10.2.2 above, shall be permitted.

3.10.2.5 Except for physics test, if the quadrant to average power tilt ratio is 1.12 or greater, the reactor shall be put in the hot shutdown condition utilizing normal operating procedures. Subsequent operation for the purpose of measuring and correcting the tilt is permitted provided the power level does not exceed 50% of rated power and the Nuclear Overpower Trip "set point is reduced by 50%".

3.10.2.6 Following any refueling and at least every effective full power month thereafter, flux maps, using the movable detector system, shall be made to confirm that the hot channel factor limits of Specification 3.10.2.2 are met.

3.10.2.7 The reference equilibrium indicated axial flux difference as a function of power level (called the target flux difference) shall be measured at least once per equivalent full power quarter. The target flux difference must be updated at least each equivalent full power month using a measured value or by interpolation using the most recent measured value and the predicted value at the end of the cycle life. The target flux difference shall be between +5.0 and -7.5% at the beginning of cycle life and between +2.0 and -7.5% at the end of cycle life. Linear interpolation shall be used to determine values at other times in cycle life.

3.10.2.8 Except during physics tests, control rod excercises, excore detector calibration, and except as modified by 3.10.2.9 through 3.10.2.12, the indicated axial flux difference shall be maintained within $\pm$ 5% of the target flux difference (defines the target band on axial flux difference). Axial flux difference for power distribution control is defined as the average value for the four excore detectors. If one excore detector is out of service, the remaining three shall be used to derive the average.

3.10.2.9 Except during physics tests, control rod exercises, or excore calibration, at a power level greater than 90 percent of rated power, if the indicated axial flux difference deviates from its target band. The flux difference shall be returned to the target band immediately or the reactor power shall be reduced to a level no greater than 90 percent of rated power.

different from those resulting from operation within the target band. The instantaneous consequence of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for flux difference in the range +14 percent to -14 percent (+11 percent to -11 percent indicated) increasing by +1 percent of each 2 percent decrease in rated power. Therefore, while the deviation exists the power level is limited to 90 percent or lower depending on the indicated flux difference.

If, for any reason, flux difference is not controlled within the $\pm$ 5 percent band for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50 percent is required to protect against potentially more severe consequences of some accidents.

As discussed above, the essence of the limits is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished, without part length rods, by using the chemical volume control system to position the full length control rods to produce the required indication flux difference.

The effect of exceeding the flux difference band at or below half power is approximately half as great as it would be at 90% of rated power, where the effect of deviation has been evaluated.

The reason for requiring hourly logging is to provide continued surveillance of the flux difference if the normal alarm functions are out of service. It is intended that this surveillance would be temporary until the alarm functions are restored.

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainity in F_q is depleted. Therefore, the limiting tilt has been set as 1.02. To avoid unnecessary power changes, the operator is allowed two hours in which to verify the tilt reading and/or to determine and correct the cause of the tilt. Should this action verify a tilt in excess of 1.02 which remains uncorrected, the margin for uncertainty in F_q and $F_{\Delta H}$ is reinstated by reducing the power by 2% for each percent of tilt above 1.0, in accordance with the 2 to 1 ratio above, or as required by the restriction peaking factors.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 19 TO PROVISIONAL OPERATING LICENSE NO. DPR-18

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

Introduction

By application dated January 6, 1978, as supplemented by letters dated January 10, March 27, April 6, April 17, and April 25, 1978, Rochester Gas and Electric Corporation (the licensee) requested authorization to operate the R. E. Ginna Nuclear Power Station in Cycle 8 with reload fuel supplied by Exxon Nuclear Company, Inc., and requested a change to the Technical Specifications involving power distribution control limits.

Discussion

The R. E. Ginna Nuclear Power Station has operated seven fuel cycles with fuel supplied by Westinghouse Corporation. Cycle 8 will involve the first use of fuel from a different vendor, Exxon Nuclear Company, Inc. (ENC). The loading for Cycle 8 will consist of 32 new ENC fuel assemblies loaded at the periphery of the core and 89 exposed Westinghouse assemblies scatter loaded in the center of the core. All assemblies are of similar design with the ENC assemblies designed to be compatible with the other fuel assemblies. Reactor power level, core average linear heat rate and primary coolant system temperature and pressure for Cycle 8 will remain the same as for the previous cycle.

The licensee has stated that all technical specification limits for the previous cycle are applicable to Cycle 8, with the exception of one limit involving power distribution control. The licensee also proposed a change to the bases of the specifications involving power distribution control to reflect a revised methodology used in the reactor physics analyses for Cycle 8.

The licensee's analyses for Cycle 8 also include the first use of ENC analytical methods to verify the acceptability of Ginna operating limitations and safety margins.

The staff evaluation which follows, addresses the acceptability of the use of the ENC assemblies in Cycle 8 and the acceptability of the proposed changes in Technical Specification. The evaluation includes the staff's review of nuclear, thermal-hydraulic and accident analyses for Cycle 8 operation.

Evaluation

1. Design of the New Fuel

The new fuel assemblies for the core periphery were designed by Exxon Nuclear Corporation to be compatible with the Westinghouse depleted fuel assemblies that are to remain in the Ginna core.

The Exxon fuel design is similar to the Westinghouse fuel bundle design (References 1 and 2).

The Exxon fuel design criteria and fuel design calculations are discussed in Exxon reports submitted with the application for Fuel Cycle 8 operation. Those aspects of the fuel design important to safety have been reviewed by the staff and found acceptable. Those aspects are: (1) the fuel performance during LOCA; (2) fuel clad collapse and fuel densification; (3) fretting wear; and (4) the effect of fuel rod bowing on the departure from nucleate boiling ratio (DNBR).

The GAPEX code (Reference 3) was used to calculate stored energy for LOCA calculations. GAPEX has been reviewed and approved by the staff for fuel temperature and internal pressure calculations in PWR fuel (Reference 4).

Reference 1 presents calculations which show that the cladding will not collapse during Cycle 8. These calculations utilize the RODEX and COLAPX codes. The RODEX code (Reference 5) calculates the cladding temperature and fuel rod internal pressure while COLAPX (Reference 7) calculates the collapse time using the RODEX input. COLAPX has been reviewed by the staff and found acceptable for cladding collapse calculations. RODEX has not been approved by the staff but the models in RODEX affecting clad temperature and internal pressure are similar to those in the GAPEX code, which has been approved. Moreover, since the clad collapse analyses for the Westinghouse fuel does not predict collapse during Cycle 8, and since the cladding for the Exxon fuel is thicker than that of the Westinghouse fuel (Reference 2) which makes it more resistent to clad collapse, we have concluded, with reasonable assurance, that the results of the RODEX analysis are acceptable.

Exxon tests to determine the magnitude of fretting at the fuel rod axial spacer contact points due to flow induced vibration revealed no active fretting corrosion and negligible difference in wear observed between 500, 1000, and 1500 hours. Based on these test results and the larger diameter - thicker clad of the Exxon fuel rods in the 14 x 14 fuel assemblies for Ginna and therefore greater stiffness, we have concluded that fuel rod integrity with respect to flow induced vibration and fretting wear is acceptable.

The effect of fuel rod bowing on Departure from Nucleate Boiling Ratio (DNBR) has been a subject of continuing discussion between the staff and Exxon. An Exxon analysis considered the fuel rod bowing penalties for the most limiting transients and attempted to show that there is sufficient margin to offset the calculated penalties. These results are presented in Reference 2. The staff has concluded that these analyses are not completely acceptable because the heat flux and pressure used to calculate the bowing penalties were for minimum DNBR conditions and do not represent the worst conditions for calculating the rod bowing penalties. However, Reference 2 shows that there is an 8.5 percent margin to the safety limit which offsets this nonconservatism. On this basis, we have concluded that there is adequate thermal margin to assure safe plant operation without violating the minimum DNBR safety limit.

Based on successful irradiation experience of Exxon fuel assemblies in other PWR cores and the analyses which have been done for Ginna Fuel Cycle 8, we have concluded that the Exxon fuel assemblies for Cycle 8 will perform in a safe and acceptable manner. The licensee has agreed (RG&E telecon 4/14/78) to submit plans for inspection of the Exxon fuel assemblies to NRC for concurrence at least 90 days prior to the end of Fuel Cycle 8 to enable additional NRC review of the fuel prior to its use in Cycle 9.

2. Thermal Hydraulic Design

The new Exxon fuel assemblies are designed to have thermal hydraulic characteristics equivalent to those of the existing fuel. Therefore, there will not be any major differences in the thermal hydraulic behavior of the core.

The licensee has shown that at 118 percent of rated power, the calculated minimum DNBR is 1.47. The corresponding value for the Westinghouse fuel assemblies is 1.43. The fuel and cladding temperature analysis uses Exxon calculational methods (Reference 7), assuming maximum power peaking and engineering tolerances. The calculated maximum fuel and cladding temperatures are well below the design limits. We, therefore, conclude that the Exxon fuel assemblies are compatible with the Westinghouse fuel assemblies in the Ginna core and that the thermal hydraulic criteria will not be exceeded during plant operation.

3. Nuclear Design

The Fuel Cycle 8 loading will consist of 89 fuel assemblies with burnups ranging from 7,178 MWD/MTU to 23,813 MWD/MTU and 32 fresh ENC fuel assemblies.

The licensee has specified new values for the target flux difference. They are between +5.0 and -7.5% for the beginning of cycle life and between +2.0 and -7.5% for the end of cycle life. For the intermediate times the values are obtained by linear interpolation. The licensee has compared the neutronic characteristics of the Cycle 8 and Cycle 7 cores and concluded that they are approximately the same. The reactivity coefficients of the Cycle 8 core are bounded by the coefficients used in the safety analyses and we have concluded that the coefficients are acceptable.

Justification of the assumed total rod worth uncertainty of 10% used in the determination of shutdown margin has not been presented. Confirmatory tests are therefore included in the startup physics tests for fuel Cycle 8.

The physics startup test program for Ginna Cycle 8 presented in the March 27, 1978 submittal (Reference 2), was reviewed with the licensee. Several changes to the rod worth and power coefficient measurements were made. These changes are documented in the Reference 17 submittal. As part of this test program, control rod reactivity worth will be measured for banks D, C, B and A in order to verify that adequate shutdown margin is available. If any one bank worth differs from the predicted value by more than 15% or the sum of the worths of these banks differs from the predicted value by more than 10%, the first shutdown bank should be measured. If the sum of the five measured banks differs from the predicted value by more than 10%, additional shutdown bank measurements will be performed to verify the technical specification shutdown margin.

We have concluded that the total physics startup test program as modified is acceptable. However, there are areas in the licensee's safety analysis that warrant verification in the physics startup test program. Therefore, a summary report as described in the March 27th submittal (Reference 2) will be submitted to the NRC. The licensee has agreed to submit the report within 45 days of completion of the program.

4. Steady State and Load Follow Operation

Compliance with F_Q and $F\Delta H$ limiting conditions for operation is ensured by adherence to previously approved constant axial offset control strategy and core monitoring with incore and excore flux monitors. Incore monitoring is achieved using travelling fission chambers. Data from the fission chambers and calculated coefficients

(Reference 9) are processed by the computer code INCORE to obtain power distribution maps. Extensive comparisons of predicted and measured core power distributions have been performed by Exxon for 3 and 4 loop cores. In general, the results of these comparisons are favorable. However, R. E. Ginna is a two loop plant and there is only a single set of measured and calculated power distributions for R. E. Ginna, Cycle 7, at hot full power, 1000 MWD/MTU. The results of this comparison show good agreement between measurement and calculation and add credibility to the licensee's assertion that an F_Q uncertainty factor of 5% is appropriate for Cycle 8. However, additional data will be obtained during the fuel cycle 8 startup physics tests.

5. Safety Analyses

The licensee has analyzed the anticipated operating occurrences and postulated accidents using the plant transient simulator code PTSPWR (Reference 15). The results of these analyses are presented in Reference 14. Our review of this code has progressed sufficently to allow us to conclude that analyses using PTSPWR provide acceptable margins to peak linear heat generation rate and departure from nucleate boiling design limits. The reactivity coefficients assumed in the safety analyses are to be confirmed during the physics startup tests.

a. Steam Line Break Analyses

The Steam Line Break (SLB) accident analysis (Reference 14) is of particular concern. SLB analysis methods have not been generically approved. The licensee asserts that should a large SLB occur the plant would return to criticality, reaching a peak average core power of 22% of rated power at approximately 90 sec after accident initiation. The minimum DNBR at this condition, using the Macbeth critical heat flux correlation, would be 1.58. Even if DNB were to occur during a steam line break accident, DNB would be restricted to a small region of the core in the vicinity of the assumed stuck rod. It is noted that DNB anywhere in the core is unlikely if all control rods scram as expected. Of the fuel rods which might experience DNB in the vicinity of the stuck rod, some fraction would release their fission gas inventory. The fission gas would have to be transported to the secondary side of the coolant system (primary to secondary steam generator leakage) in order to represent a potential hazard. The potential release to the atmosphere would be significantly less than 10 CFR Part 100 limits. Accordingly, we have concluded that the consequences of a steam line break are acceptable.

b. ECCS Analysis

The licensee has submitted ECCS performance analyses for the Westinghouse (Reference 19) and new ENC fuels (Reference 1). The Westinghouse analysis was performed for Cycle 7 fuel which the staff believes is a conservative evaluation for the Westinghouse fuel during Cycle 8. The ENC analysis was performed for Cycle 8 using the ENC WREM-II ECCS evaluation model (Reference 7) which is described in References 8 and 9. The applicability of the model

to two-loop Westinghouse PWR plants was evaluated by ENC in Reference 10. The ENC evaluation model has been reviewed and approved conditionally by the NRC (Reference 16). The staff has recently considered whether the Westinghouse generic evaluation adequately represented the flow characteristics of the Westinghouse two loop units. The generic evaluation model assumes that all safety injection water is introduced directly into the lower plenum. For the two loop units, the safety injection water is injected into the upper plenum. Thus, the staff was concerned that the Westinghouse model did not consider interaction between UPI water and steam flow. (References 11 and 12). After plant specific submittals by the licensees operating two loop plants were reviewed, the staff concluded that the calculations provided by the licensees (with certain modifications to the staff's model) are acceptable as an interim basis for continued safe operation of the Westinghouse two loop plants, while long term efforts continue for developing a model specifically treating UPI. For the Ginna plant the calculations which specifically considered UPI using the modified version of the staff model, resulted in a change of only 15°F from those using the generic model in which the UPI-core interaction was not specifically considered (Reference 20). In the interim, before these models are developed, the licensee has provided a modification to the current Westinghouse model which accounts for UPI-core interaction (Reference 13). It was demonstrated that the modification resulted in the increase of peak clad temperature by 15°F. Since for the Ginna plant both ENC WREM-II and Westinghouse models predict similar PCT's (1922°F for ENC WREM-II and 1957°F for Westinghouse) it can be expected that the UPI modification, when applied to the ENC WREM-II model, would allow about the same increase in PCT. The licensee has drawn a similar conclusion and agreed to submit within 30 days, calculational results to confirm the validity of this conclusion. (Reference 21).

The ECCS analyses have been performed with the upper head fluid temperature equal to the fluid outlet (hot leg) temperature and assuming 10 percent of steam generator tubes plugged. The analyses included a spectrum of breaks which consisted of guillotine double ended cold leg (DEGCL) breaks with discharge coefficients of 1.0, 0.6 and 0.4 and split breaks with break areas of 8.25, 4.9 and 3.30 ft^2. No small break analysis was performed. The licensee has demonstrated, by showing analogy between the present analysis and the analyses performed previously for other plants, that the small break LOCA is not limiting (Reference 2). The critical break size was determined to be DEGCL with C_D=0.4.

The staff has concluded that although the Westinghouse and Exxon two-loop generic-evaluation models should be changed to consider upper plenum injection (unless the plant is modified), analyses at the specific operating conditions applicable to the Ginna plant demonstrate that the effect of disregarding upper plenum injection interaction on refill and reflood conditions will not be significant (less than 20°F PCT). Therefore, the staff believes that, for the limited range to which

the models are applied for conditions at the Ginna plant, the models do not deviate from the requirements of 10 CFR 50 Appendix K item I.D.3, and the calculations are acceptable.

On March 23, 1978 Westinghouse informed the NRC that an error in the West-ECCS evaluation model had been found which had resulted in incorrectly calculated peak clad temperatures in all LOCA analyses previously submitted by their customers. For several plants preliminary estimates indicated that they would not meet the 2200°F limit of 10 CFR 50.46 at their present maximum overall peaking factor limits. Westinghouse and several of their customers met with the NRC staff on March 29, 1978 in Bethesda to discuss the error and its impact on specific plant analyses. Subsequent to that meeting, Westinghouse provided information through the licensees of operating reactors to justify continued operation at the interim peaking factor Technical Specification limits proposed by the NRC staff on April 3, 1978.

On April 17, 1978 (Reference 19) RG&E submitted a letter indicating that continued operation at their present Technical Specification limit of 2.32 (total peaking factor) was justified on the basis of additional generic Westinghouse analyses. Westinghouse had determined that the impact of correcting the error on the peak cladding temperature for the RE Ginna plant was significant but within the presently existing margin (228°F) to the 2200°F° acceptance criteria limit. The NRC Staff confirmed the conservatism of that and all other plant evaluations and on April 18, 1978 published a Safety Evaluation Report (Reference, attachment to Exemption). Since the Westinghouse and ENC fuels were analyzed using the respective Westinghouse and ENC evaluation models, and since there is no zirconium-water error in the ENC calculational model, the error in zirconium-water reaction in the Westinghouse calculational model has no effect on the Exxon calculations. The Zirconium-water reaction error in the Westinghouse model is the subject of an exemption request by the licensee dated April 25, 1978, (Reference 21) and a separate exemption action by NRC.

6. Technical Specification Changes

The proposed addition to the Technical Specifications restricts the permissable range of the target flux difference i.e. the ratio of the flux in the top half of the core minus the flux in the lower half of of the core to the total flux measured at 100% power, equilibrium conditions. The addition, Technical Specification 3.10.2.7, assures that axial power distributions realized in the reactor will be no more limiting with respect to linear heat generation rate than the axial power distributions used by Exxon to analytically confirm (Reference 18) that, limiting values of linear heat generation vs core height, Technical Specification 3.10.2.2, will not be violated. The restriction has been reviewed and approved on a generic basis and has recently been incorporated in the Technical Specifications of PWR's using Exxon Nuclear fuel.

The change to Technical Specification 3.10.1.4 and the addition of specification 3.10.1.6 are required to permit the physics testing program as discussed in part 3 of our evaluation. The change and the addition are in accordance with the Standard Technical Specifications for Westinghouse PWR's which we have already reviewed and approved.

The changes to the basis of the Technical Specification related to core power distribution are in accordance with the Standard Technical Specification which we have approved and are therefore acceptable also.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: May 1, 1978

REFERENCES

(1) Letter from LeBoef, Lamb, Leiby and MacRae (Counsel for Rechester Gas and Electric Corporation) to E. G. Case (NRC), dated January 6, 1978.

(2) Letter from L. D. White, Jr. (Rochester Gas and Electric Corporation) to D. L. Ziemann (NRC), dated March 27, 1978.

(3) XN-73-25, "GAPEX: A Computer Program for Predicting Pellet-to-Cladding Heat Transfer Coefficients", June 1975.

(4) USNRC Report, "Technical Report on Densification of Exxon Nuclear PWR Fuel", February 27, 1975.

(5) XN-76-8(P), "RODEX: Fuel Rod Design Evaluation Code", February 1977.

(6) XN-72-23, "Clad Collapse Calculational Procedure", November 1, 1972.

(7) XN-NF-77-58, "ECCS analysis for the R. E. Ginna Reactor with ENC WREM-II PWR Evaluation Model", December 1977.

(8) XN-75-41, "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model", Vol I through III, July-August 1975 and Supplements 1 through 7, August-November 1975.

(9) XN-76-27, "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model Update ENC WREM-II", July 1976 and Supplements 1 and 2, September-November 1976.

(10) XN-NF-77-25, "Exxon Nuclear Company ECCS Evaluation of a 2-loop Westinghouse PWR with Dry Containment using the ENC WREM-II ECCS Model - Large Break Example Problem," August 1977.

(11) Letter from E. G. Case (NRC) to L. D. White, Jr. (Rochester Gas and Electric Corporation), dated December 16, 1977.

(12) Letter RG&E to NRC, Development of a New Model to Account for Upper Plenum Injection, dated March 5, 1978.

(13) Letter from L. D. Amish (Rochester Gas and Electric Corporation) to A. Schwencer (NRC), dated February 1978.

(14) XN-NF-77-40, "Plant Transient Analysis for the R. E. Ginna Unit 1 Nuclear Power Plant", November 1977.

(15) XN-74-5, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTSPWR)," Revision 1, May 1975.

(16) USNRC Topical Report Evaluation, Exxon Nuclear Company Report XN-NF-77-25, April 1978.

(17) Letter from L. D. White, Jr. (Rochester Gas and Electric Corporation) to D. L. Ziemann (NRC), dated April 6, 1978.

(18) Exxon Nuclear Power Distribution Control for Pressurized Water Reactors XN-76-40, September 1976.

(19) Letter from L. D. White, Jr., (RG&E) to A. Schwencer (NRC) dated April 7, 1977.

(20) Letter to RG&E dated April 28, 1978 transmitting staff SER of UPI model evaluation.

(21) Letter from RG&E to NRC dated April 25, 1978, related to ENC UPI calculations.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter	)	
	)	
ROCHESTER GAS AND ELECTRIC CORPORATION	)	Docket No. 50-244
	)	
	)	
(R. E. Ginna Nuclear Power Plant	)	

EXEMPTION

I.

The Rochester Gas and Electric Corporation (the licensee), is the holder of Provisional Operating License No. DPR-18 which authorizes the operation of the nuclear power reactor known as R. E. Ginna Nuclear Power Plant (the facility) at steady reactor power levels not in excess of 1520 megawatts thermal (rated power). The facility consists of a Westinghouse Electric Company designed pressurized reactor (PWR) located at the licensee's site in Wayne County, New York.

II.

In accordance with the requirements of the Commission's ECCS Acceptance Criteria 10 CFR 50.46, the licensee submitted on April 7, 1977 and January 6, 1978 ECCS evaluations for proposed operation using 14 x 14 fuel manufactured by the Westinghouse Electric Company and the Exxon Nuclear Company (ENC). These evaluations established limits on the peaking factor based upon ECCS evaluation models developed by the Westinghouse Electric Company (Westinghouse), the designer of the Nuclear Steam Supply System for this facility, and by Exxon, the supplier of the reload fuel. The Westinghouse and ENC ECCS evaluation

models had been previously found to conform to the requirements of the Commission's ECCS Acceptance Criteria, 10 CFR Part 50.46 and Appendix K. The evaluations indicated that with the peaking factor limited as set forth in the evaluations and with other limits set forth in the facility's Technical Specifications, the ECCS cooling performance for the facility would conform with the criteria contained in 10 CFR 50.46(b) which govern calculated peak clad temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry and long-term cooling.

On March 23, 1978 Westinghouse informed the Nuclear Regulatory Commission (NRC) that an error had been discovered in the fuel rod heat balance equation which resulted from the incorrect use of only half of the volumetric heat generation due to metal-water reaction in calculating the cladding temperature. Thus, the LOCA analyses previously submitted to the Commission by licensees of Westinghouse reactors were in error.

The error identified would result in an increase in calculated peak clad temperature, which, for some plants, could result in calculated temperatures in excess of 2200°F unless the allowable peaking factor was reduced somewhat. Westinghouse identified a number of other areas in the approved model which Westinghouse indicated contained sufficient conservatism to offset the calculated increase in peak clad temperature resulting from the correction of the error noted above. Four of these areas were generic, applicable to all plants, and a number of others were plant specific. As outlined in the NRC Staff's Safety Evaluation Report (SER) of April 18, 1978 (attached), the staff determined that some of these

modifications would be appropriate to offset to some extent the penalty resulting from correction of the error. The attached SER of April 18, 1978 sets forth the value for each modification applicable to each facility.

As part of the proposed change to the technical specifications the licensee has submitted information and analyses to permit Cycle 8 operation with reshuffled Westinghouse fuel and with 32 Westinghouse fuel assemblies replaced with fresh fuel assemblies manufactured by the Exxon Nuclear Company (ENC) and loaded on the periphery of the core. Based on an analysis of the information presented by the licensee, the staff has concluded that the new fuel manufactured by Exxon Nuclear Company (ENC) is both similar to and compatible with the fuel previously supplied by Westinghouse. The ENC calculations for the ENC fuel for the Ginna Core are not affected by the Westinghouse error. (Safety Evaluation for the reload application dated May 1, 1978). The staff's evaluation determined that the impact of correcting the Westinghouse Zirconium-water reaction error on the peak cladding temperature for the RE Ginna plant is less than the presently existing margin (228°F) to the 2200°F acceptance criteria limit. The NRC Staff has confirmed that the impact of correcting the error in the Westinghouse ECCS evaluation model as it relates to the use of Westinghouse fuel is conservative, based on the April 18, 1978 Safety Evaluation Report.

Although revised computer calculations correcting the error, noted above, and incorporating the modifications described in the Staff's April 18, 1978 SER have not been run for each plant, the various parametric studies that have been made for various aspects of the approved Westinghouse model over the course of time provide a reasonable basis for concluding that when final revised calculations for the facility are submitted using the revised and corrected Westinghouse model, they will demonstrate that operation will conform to the criteria of 10 CFR 50.46(b), when operated at the peaking factors set forth in the SER of April 18, 1978. Such revised calculations fully conforming to 10 CFR 50.46 are to be provided for the facility as soon as possible.

Operation of the facility would nevertheless be technically in non-conformance with the requirements of §50.46, in that specific computer runs for the particular facility employing revised models with the Westinghouse metal-water error corrected and with the proposed model changes considered, as a complete entity will not be complete for some time. However, operation as proposed in the licensee's application dated January 6, 1978, and at the peaking factor limit specified in this Exemption will assure that the ECCS system will conform to the performance criteria of §50.46. Accordingly, while the actual computer runs for the specific facility are carried out to achieve full compliance with 10 CFR §50.46, operation of the facility will not endanger life or property or the common defense and security.

In the absence of any safety problem associated with operation of the facility during the period until the computer computations are completed, there appears to be no public interest consideration favoring restriction of the operation of the captioned facility. Accordingly, the Commission has determined that an exemption in accordance with 10 CFR §50.12 is appropriate. The specific exemption is limited to the period of time necessary to complete computer calculations.

IV.

Copies of the Safety Evaluation Report dated April 18, 1978, and the following documents are available for inspection at the Commission's Public Document Room at 1717 H Street, Washington, D. C. 20555, and at the Rochester Public Library, 115 South Avenue, Rochester, New York 14627.

(1) Licensee submittals dated April 7, 1977, January 6, 1978, and April 25, 1978.

(2) Amendment No. 19 to License No. DPR-18 and the related Safety Evaluation for the reload application, and

(3) This Exemption in the matter of RE Ginna Nuclear Power Plant.

Wherefore, in accordance with the Commission's regulations as set forth in 10 CFR Part 50, the licensee is hereby granted an exemption from the requirements of 10 CFR §50.46(a)(1) that ECCS performance be calculated in accordance with an acceptable calculational model which conforms to the provisions in Appendix K, without errors discussed herein. This exemption is conditioned as follows:

(1) As soon as possible, the licensee shall submit a reevaluation of ECCS cooling performance calculated in accordance with the Westinghouse Evaluation Model, and approved by the NRC staff and corrected for the errors described herein.

(2) Until further authorization by the Commission, the Technical Specification limit for total nuclear peaking factor (F_Q) for the facility shall be limited to 2.32.

FOR THE NUCLEAR REGULATORY COMMISSION

Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Attached:
Safety Evaluation Report,
dated April 18, 1978

Dated at Bethesda, Maryland
this 1st day of May, 1978

April 18, 1978

Safety Evaluation Report

Error in Westinghouse ECCS Evaluation Model

Introduction

Westinghouse was informed on March 21, 1978 by one of their licensees that an error had been discovered in their ECCS Evaluation Model. This error was common to both the blowdown and heatup codes. Westinghouse determined by analyses that the fuel rod heat balance equation in the LOCTA IV & SATAN VI codes was in error and that the LOCA analyses previously submitted by their customers were incorrect and predicted PCT's which were too low. Westinghouse determined that only half of the volumetric heat generation due to metal-water reaction was used in calculating the cladding temperatures and that an unreviewed safety question existed since preliminary estimates indicated that some plants would not meet the 2200^{0}F limit of 10 CFR 50.46 without a reduction in overall peaking factor limit. Westinghouse notified their customers and NRC on March 23, 1978 while the utilities notified NRC through the regional I&E Offices.

Promptly upon notification by Westinghouse, the staff assessed the immediate safety significance of this information. The staff noted certain points that indicated no immediate action was required to assure safe operation of the plants. First, most plants operate at peaking factors significantly below the maximum peaking factor used for safety calculations. By making safety computations at factors higher than actual operating levels, the facility has a wide range of flexibility, without the need for hour to hour recomputations of core status. The difference between the actual peaking factors and the maximum calculated peaking factors, for most plants, would offset the penalty resulting from the correction of the error. Second, for most reactors there are plant-specific parameters which bear upon aspects of the ECCS performance calculations. Utilities do not generally take credit for these plant-specific parameters, preferring to provide a simpler computation which conservatively disregards these individually small credits. Third, the error in the Westinghouse computations relates to the zirconium-water reaction heat source. This is an aspect of Appendix K, which is generally recognized to be very conservative. New experimental data indicate that the methods required by Appendix K appreciably over-estimate the heat source. Thus, while the error in fact entails a deviation from a specific requirement of Appendix K, it does not entail a matter of immediate safety significance.

Westinghouse continued to evaluate the impact of the error on previous plant specific LOCA analyses and performed scoping calculations, sensitivity studies and some plant specific reanalyses. In addition, Westinghouse investigated several modifications to the previously approved methods which if approved by the NRC staff would offset some of the immediate impact of the error on Technical Specifications limits and plant operating flexibility.

On March 29, 1978, Westinghouse and several of their customers met with members of the NRC staff in Bethesda. Westinghouse described in detail the origin of the error, explained how it affected the LOCA analyses, and how the error had been corrected and characterized its effect on current plant specific analyses. In order to avoid reduction in overall peaking factors (F_q), Westinghouse presented a description of three proposed ECCS-LOCA evaluation model modifications which would contribute a compensating reduction of PCT. They were characterized as follows:

1) Revised FLECHT 15 x 15 heat transfer correlation.

   This new reflood heat transfer correlation which had been recently developed and submitted by Westinghouse (Reference (1) was proposed as a replacement for the currently approved FLECHT correlation. To determine the benefit, the proposed correlation was incorporated into the LOCTA IV heatup code and was found to result in improved heat transfer during the reflood portion of the LOCA.

2) Revised Zircaloy Emissivity.

   Based on recent EPRI data (Reference 2), Westinghouse proposed to modify the presently approved equation for zircaloy cladding emissivity to a constant value of 0.9. The higher emissivity (previously below 0.8) provides increased radiative heat transfer from the hot fuel pin during the steam cooling period of reflood.

3) Post-CHF heat transfer.

   Westinghouse proposed to replace their present post-CHF transition boiling heat transfer correlation with the Dougall-Rohsenow film boiling correlation (Reference 3) which they stated was included in Appendix K to 10 CFR Part 50 as an acceptable post-CHF correlation.

These three model modifications were classified as generic, applicable to all plant analyses. Subsequently, as discussed below, these changes were rejected by the staff as providing generic benefit. However, a portion of the credit proposed by Westinghouse was approved by the staff to certain specific plants, which had provided specific calculations with the new 15 x 15 correlation. During the period March 29 to April 18, 1978, Westinghouse provided the staff with additional sensitivity analyses and plant specific analysis in which they evaluated the effects of some changes to plant-specific inputs in the LOCA analyses. These were as follows:

1. Assumed Plant Power Level

   A reduction of the plant power level assumed in the SATAN VI blowdown analyses from 102% of the Engineered Safeguards Design Rated Power (ESDR) level to 102% of rated power was proposed. Previously, analyses had been performed at approximately 4.5% over the rated power. This change was worth approximately 0.01 in F_Q, and is referred to as ΔFESDR in Table 1.

2. COCO Code Input

   A modification to the COCO code input (Reference 3) to more realistically model the painted containment walls was proposed. Since the paint on containment walls provides additional resistance to heat loss into the walls, the COCO code calculates an increase in containment back pressure, which results in a benefit to the calculated peak cladding temperature of 0 to 40°F, during the reflooding transient. The magnitude of the benefit is dependent on the type of plant and the heat transfer properties of the paint, and results in up to 0.03 benefit in F_Q, and is referred to as ΔF_{CP} in Table 1.

3. Initial Fuel Pellet Temperature

   A modification of the initial fuel pellet temperature from the design basis to the actual as-built pellet temperatures was proposed. In the present LOCA calculations, Westinghouse has assumed margins in the initial pellet temperature. The margin available in plant-specific ranges from 28°F to 55°F. Use of the actual pellet temperature rather than the assumed value results in a reduction in pellet temperature (stored energy) at the end of blowdown, as calculated by the SATAN code, of approximately 1/3 of the initial pellet temperature margin. Westinghouse has provided sensitivity analyses which indicate that a 37°F reduction in fuel pellet temperature at end of blowdown is worth approximately 0.1 in F_Q. This is referred to as ΔF_{PT} in Table 1.

4. Accumulator Water Volume Consideration

Westinghouse has evaluated the effect on ECCS performance of reducing the accumulator water volume, and has determined that for those plants for which the downcomer is refilled before the accumulators are emptied, there is a benefit in PCT. The sensitivity studies have indicated that this benefit in F_Q is plant-specific. This is referred to as ΔF_{ACV} in Table 1.

5. Steam Generator Tube Plugging Consideration

In previous analyses, Westinghouse has assumed values of steam generator tube plugging which were greater than the actual plant-specific degree of plugging. Sensitivity analyses submitted in Reference 4 were used to evaluate the benefit available by realistically representing the plant-specific data. For the plants affected, the benefit in PCT ranged from 7 to 66°F which was conservatively worth from 0.007 to .066 F_Q. This is referred to as ΔF_{SG} in Table 1.

Safety Evaluation

The information provided by Westinghouse was separated into two categories; the generic evaluation model modifications and the plant specific sensitivity studies and reanalyses. The NRC staff reviewed the peaking factor limits proposed by Westinghouse to verify their conservatism.

The metal-water reaction heat generation error in the Westinghouse ECCS evaluation model was evaluated by the staff to determine an appropriate interim penalty. Westinghouse provided two preliminary separate effects calculations which indicated that a maximum penalty of from 0.14 to 0.17 was appropriate to compensate for the model error. As indicated in Reference 5, the staff conservatively rounded up this penalty to 0.20.

As is noted above, Westinghouse had proposed several compensating generic changes in their evaluation model to offset any necessary reductions in peaking factor due to the error. These changes were assessed by the staff and as noted in Reference 5.

1) No credit was given at this time, for the changes in the post-CHF heat transfer correlation and new zircaloy emissivity data.

2) Partial credit (70%) would be given at this time for the use of the new 15 x 15 FLECHT correlation only for plants which had provided a specific calculation demonstrating that such credit was appropriate.

Based on this review the staff developed recommended interim peaking factor limits for all the operating plants and recommended that any other plant specific interim factors (benefits) not related to the generic review be considered separately. In addition, the staff reviewed plant specific reanalyses for DC Cook, Units 1 & 2, Zion, Units 1 & 2, and Turkey Point, Unit 3 which had corrected the error in metal water reaction. In these analyses the Dougall-Rohsenow and zircaloy emissivity credits were not considered, while the new 15 x 15 FLECHT correlation was included. The staff concluded that these reanalyses could serve as a basis for conservatively determining interim peaking factor limits for these plants.

For most of the operating plants the staff's generic review resulted in a lower allowable peaking factor than Westinghouse had proposed. However, in one case, Westinghouse had proposed more limiting peaking factors in order to prevent clad temperatures at the rupture node from exceeding 2200°F. The staff concluded that it would be properly conservative to use the minimum of these values.

Based on plant specific sensitivity studies, performed by Westinghouse, the licensees submitted requests for interim plant specific benefits. The staff reviewed these sensitivity studies and recommended that appropriate credits be accepted. The results of these analyses are shown in Table 1.

We informed each licensee by telephone on April 3, 1978, that he should administratively reduce his peaking factor limit from the limit contained in his Technical Specifications to the interim peaking factor limit contained in the right hand column of Table 1. In those cases where the limit in Table 1 is 2.32, this represents no change from the Technical Specifications limit. The peaking factor limit of 2.32 is generically supported and approved for Westinghouse reactors employing constant axial offset control operating procedures.[6]

For the reactor having an interim peaking factor limit of 2.31, we requested no further justification of the limit. This is because the generic analysis supporting the limit of 2.32 approaches the limit only at beginning of the first cycle. Since the affected reactors have operated past this point, it is clear that the maximum attainable peaking factor will be less than 2.32. While this margin has not been quantified, the staff is convinced it is substantially greater than the 0.01 for which we are requiring no additional justification from the plants with an interim limit of 2.31.

For the reactors with an interim limit less than 2.31, we requested that the licensee furnish administratively imposed procedures to replace Technical Specifications either:

1. To provide a plant specific constant axial offset control analysis of 18 cases of load following which would ensure that the interim limit would not be exceeded in normal operation of the power plant, or, at his option, if such analysis were unobtainable, inappropriate or insufficient,

2. To institute procedures for axial power distribution monitoring of the interim limit using a system designed for this purpose or manual procedures as indicated in Standard Technical Specifications 3/4 2.6 and ancillary Specifications.

We requested the licensees to provide indication that they have adopted the above interim LOCA analyses, interim peaking factor limits and administrative procedures by April 10, 1978, if their reactors were operating, and by April 17, 1978, if the reactors were not operating.

Conclusion

We conclude that when final revised calculations for the facility are submitted using the revised and corrected model, they will demonstrate that with the peaking factor set forth herein, operation will conform to the criteria of 10 CFR 50.46(b). Such revised calculations fully conforming to 10 CFR 50.46(b) are to be provided for the facility as soon as possible.

As discussed herein, the peaking factor limit specified in Table 1, in combination with any necessary operating surveillance requirements, will assure that the ECCS will conform to the performance requirements of 10 CFR 50.46(b). Accordingly, limits on calculated peak clad temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry and long term cooling provide reasonable assurance that the public health and safety will not be endangered.

References

(1) R. S. Dougall, W. M. Rohsenow, "Film Boiling on the Inside of Vertical Tubes with Upward Flow of the Fluid at Low Qualities", MIT Report 9079-26, September 1963.

(2) EPRI Report NP-525, "High Temperature Properties of Zircaloy-Oxygen Alloy", March 1977.

(3) WCAP-9220, "Westinghouse ECCS Evaluation Model, February 1978 Version", February 1978.

(4) WCAP 8986 "Perturbation Technique for Calculating ECCS Cooling Performance", February, 1978.

(5) DSS SER "Metal-Water Reaction Heat Generation Error in Westinghouse ECCS Evaluation Model Computer Programs", Z. R. Rosztoczy to D. F. Ross/D. G. Eisenhut, 4/7/78.

(6) T. Morita, et al., "Power Distribution Control and Load Following Procedures," WCAP-8385 (Proprietary) and WCAP-8403 (Non-Proprietary), September, 1974.

TABLE 1 F_Q Analysis	PCT °F	F_Q OLD	ΔF_T	ΔF_{ZrO_2}	ΔF_{FLECHT}	F_{PCT}	F_{SE}	$F_{Q,MIN}$	ΔF_{ESDR}	ΔF_{CP}	ΔF_{PT}	ΔF_{SG}	ΔF_{ACV}	F_Q LIMIT
2 Loop														
Pt. Beach 1	2025	2.32	.16	-.2	-	2.28	2.32	2.28	.01	-	-	.029	-	2.32
Pt. Beach 2	2025	2.32	.16	-.2	-	2.28	2.32	2.28	.01	-	-	.066	-	2.32
Ginna	1972	2.32	.26	-.2	-	2.32	2.32	2.32	-	-	-	.053	-	2.32
Kewaunee	2172	2.25	.03	-.2	.05	2.13	2.25	2.13	.01	.02	-	-	-	2.16
Prairie Island 1/2	2187	2.32	.01	-.2	.05	2.18	2.26	2.18	.01	.02	-	-	.03	2.24(+)
3 Loop														
North Anna	2181	2.32	.02	-.2	-	2.14	2.32	2.14	-	-	-	-	-	2.14
Beaver Valley	2041	2.32	.15	-.2	-	2.27	2.32	2.27	-	-	.036	-	-	2.31
Farley	1991	2.32	.24	-.2	-	2.32	2.32	2.32	.01	.005	-	-	-	2.32
Surry 1	2177	1.85	.02	-.2	.06	1.73	1.84	1.73	-	.03	.025	.023	-	1.81
Surry 2	2177	1.85	.02	-.2	.06	1.73	1.84	1.73	-	.03	.025	.023	-	1.81
Turkey Point 3	2019*	1.90	.14	0	-.03	2.01	2.05	2.01	-	-	-	.020	-	2.03
Turkey Point 4	2195	2.05	.00	-.2	.05	1.90	1.91	1.90	-	-	-	.01	-	1.91
4 Loop														
Indian Point 2	2086	2.32	.11	-.2	-	2.23	2.23	2.23	.01	-	-	-	-	2.24
Indian Point 3	2125	2.32	.07	-.2	.06	2.25	2.19	2.19	.01	-	.03	-	-	2.23
Trojan	1975	2.32	.26	-.2	-	2.32	2.32	2.32	.01	-	.037	-	-	2.32
Salem 1	2135	2.32	.06	-.2	-	2.18	2.32	2.18	.01	-	.024	-	-	2.21
Zion 1/2	2189**	2.07	-	0	-.03	2.04	-	2.04	-	-	-	-	-	2.04(+)
Cook 1	2161*	1.90	.03	0	-.03	1.90	1.98	1.90	-	-	-	-	-	1.90
Cook 2	2190*	2.10	.01	0	0	2.11	-	2.11	0	0	0	0	0	2.11

ΔF_T - Credit in F_Q for PCT margin to 2200°F limit.

ΔF_{ZrO_2} - Metal Water Reaction penalty on F_Q.

ΔF_{FLECHT} - Credit in F_Q for improvements to 15x15 FLECHT Correlation.

F_{PCT} - Staff estimated F_Q based on 2200°F PCT limit.

F_{SE} - Westinghouse proposed F_Q based on stored energy sensitivity studies.

*Denotes reanalysis at F_Q old value error corrected.

**Denotes reanalyses at F_Q old value, error corrected, accumulator Vol. Change of 100 ft^3, accumulator pressure of 650 psia

(+) These limits are applicable assuming licensee modifies accumulator conditions as appropriate. If not, Prairie Island 1/2 F_Q=2.21, Zion 1/2 F_Q=1.9

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-244

ROCHESTER GAS AND ELECTRIC CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 19 to Provisional Operating License No. DPR-18, issued to Rochester Gas and Electric Corporation (the licensee), which revised the Technical Specifications for operation of the R. E. Ginna Plant (facility) located in Wayne County, New York. The amendment is effective as of its date of issuance.

The amendment changes the Appendix A Technical Specifications to support operation in Cycle 8 with reload fuel by Exxon Nuclear Company (ENC). This fuel has been designed by ENC to be compatible with the fuel supplied previously by Westinghouse. In addition, the amendment allows Technical Specification changes that are required for startup tests.

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.

Notice of proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on February 21, 1978 (43 FR 7275). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

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 Commission's Order for Modification of License dated August 27, 1976, (2) the application for amendment dated January 6, 1978, and supplements thereto dated January 10, 1978, March 27, 1978, April 6, 1978, April 17, 1978, and April 25, 1978, (3) Amendment No. 19 to License No. DPR-18, (4) the Commission's related Safety Evaluation, and (5) the Exemption related to the requirements of 10 CFR 50.46(a)(1) and the Safety Evaluation dated April 18, 1978, attached thereto. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D.C. and at the Rochester Public Library, 115 South Avenue, Rochester, New York 14627.

A copy of items (1), (3), (4), and (5) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, 1st day of May, 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

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

May 10, 1978

5/10/78

Edward Luton, Esq., Chairman
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dr. Emmeth A. Luebke
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dr. Franklin C. Daiber
College of Marine Studies
University of Delaware
Newark, Delaware 19711

In the Matter of
Rochester Gas & Electric Corporation
(R. E. Ginna Nuclear Power Plant, Unit No. 1)
Docket No. 50-244

Gentlemen:

Enclosed are copies of Amendment No. 19 to Provisional Operating License No. DPR-18, and supporting safety evaluations, and an exemption from the requirements of 10 CFR §50.46(a)(1) for the R. E. Ginna Nuclear Power Plant. I have also enclosed a copy of the Safety Evaluation Report, An Interim ECCS Evaluation Model For Westinghouse Two-Loop Plants (March, 1978), which treats upper plenum injection.

These materials are submitted to the Licensing Board in keeping with the NRC Staff's policy of keeping Board's informed. An ECCS contention is presently pending before this Board.

Sincerely,

Edward G. Ketchen

Edward G. Ketchen
Counsel for NRC Staff

Enclosures
As Stated

cc w/encl: Leonard M. Trosten, Esq.
Mr. Michael Slade
Robert E. Lee, Ph.D
Jeffrey Cohen, Esq.
Warren B. Rosenbaum, Esq.
Atomic Safety and Licensing Board Panel
Atomic Safety and Licensing Appeal Board
Docketing and Service Section

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

May 1, 1978

Docket No. 50-244

Rochester Gas and Electric Corporation
ATTN: Mr. Leon D. White, Jr.
Vice President
Electric and Steam Production
89 East Avenue
Rochester, New York 14649

Gentlemen:

The Commission has issued the enclosed Amendment No. 19 to Provisional Operating License No. DPR-18 and an Exemption from the requirements of 10 CFR 50.46(a)(1) for the R. E. Ginna Nuclear Power Plant.

The amendment consists of changes to the Technical Specifications in response to your application dated January 6, 1978, as supplemented by letters dated January 10, 1978, March 27, 1978, April 6, 1978, April 17, 1978, and April 25, 1978. We have recently noted that your January 6 application, which was received by the NRC on January 9, 1978, was actually dated January 6, 1977.

The amendment incorporates changes to the Appendix A Technical Specifications to support operation in Cycle 8 with reload fuel by Exxon Nuclear Company (ENC). This fuel has been designed by ENC to be compatible to the fuel supplied previously by Westinghouse. In addition, the amendment allows Technical Specification changes that are required for startup tests.

The Commission has also concluded that your ECCS analysis utilizes upper head fluid (hot leg) temperature and therefore satisfies the provision set forth in the Commission's Order for Modification of License dated August 27, 1976, without changes to the Technical Specifications.

Notice of proposed Issuance of Amendment to Facility Operating License in connection with the license amendment action was published in the Federal Register on February 21, 1978 (43 FR 7275).

In response to your request dated April 25, 1978, we have granted an Exemption from the requirements of 10 CFR 50.46(a)(1) that ECCS performance be calculated in accordance with an acceptable calculational model which conforms to the provisions in Appendix K, without the errors contained in the analyses previously submitted to the Commission. On March 23, 1978, Westinghouse provided the Commission an oral notification related to these errors.

Copies of the Safety Evaluation related to the license amendment, the staff's Safety Evaluation Report dated April 18, 1978, related to the Exemption and Notice of Issuance of License Amendment are also enclosed. The Exemption and the Notice are being forwarded to the Office of the Federal Register for publication.

Sincerely,

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

Enclosures:
1. Amendment No. 19 to License DPR-18
2. Safety Evaluation
3. Exemption w/Safety Evaluation dated 4/18/78
4. Notice

cc w/enclosures:
See next page

cc
Lex K. Larson, Esquire
LeBoeuf, Lamb, Leiby & MacRae
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Jeffrey Cohen
New York State Energy Office
Swan Street Building
Core 1, Second Floor
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Director, Technical Development Programs - (w/cys of 4/7/77, 1/6/78, 1/10/78, 3/27/78, 4/6/78, 4/17/78, and 4/25/78 filings by RG&E)
State of New York Energy Office
Agency Building 2
Empire State Plaza
Albany, New York 12223

Rochester Public Library
115 South Avenue
Rochester, New York 14627

Supervisor of the Town of Ontario
107 Ridge Road West
Ontario, New York 14519

Chief, Energy Systems Analyses
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U. S. Environmental Protection Agency
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U. S. Environmental Protection
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Region II Office
ATTN: EIS COORDINATOR
26 Federal Plaza
New York, New York 10007

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

ROCHESTER GAS AND ELECTRIC CORPORATION

DOCKET NO. 50-244

R. E. GINNA NUCLEAR POWER PLANT

AMENDMENT TO PROVISIONAL OPERATING LICENSE

Amendment No. 19
License No. DPR-18

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Rochester Gas and Electric Company (the licensee) dated January 6, 1978, as supplemented by letters dated January 10, 1978, March 27, 1978, April 6, 1978, April 17, 1978, and April 25, 1978, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment and paragraph 2.C(2) of Provisional Operating License No. DPR-18 is hereby amended to read as follows:

   (2) Technical Specifications

   The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 19 are hereby incorporated in the license. The licensee shall operate the facility in accordance with the Technical Specifications.

3. This license amendment is effective as of the date of its issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

Darrell G. Eisenhut

Darrell G. Eisenhut, Assistant Director
for Systems & Projects
Division of Operating Reactors

Attachment:
Changes to the Technical
Specifications

Date of Issuance May 1, 1978

ATTACHMENT TO LICENSE AMENDMENT NO. 19

PROVISIONAL OPERATING LICENSE NO. DPR-18

DOCKET NO. 50-244

Change the Technical Specifications contained in Appendix A of License No. DPR-18 as indicated below. The revised pages contain the captioned amendment number and marginal lines to reflect the area of change.

Remove	Insert
3.10-2	3.10-2
----	3.10-2a
3.10-4	3.10-4
3.10-8c	3.10-8c

3.10.1.2 When the reactor is critical except for physics tests and control rod exercises, the shutdown control rods shall be fully withdrawn.

3.10.1.3 When the reactor is critical, except for physics tests and control rod exercises, each group of control rods shall be inserted no further than the limits shown by the lines on Figure 3.10-1 and moved sequentially with a 100 (±5) step overlap between successive banks.

3.10.1.4 During control rod exercises indicated in Table 4.1-2, the insertion limits need not be observed but the Figure 3.10-2 must be observed.

3.10.1.5 The part length control rods will not be inserted except for physics tests or for axial offset calibration performed at 75% power or less.

3.10.1.6 During measurement of control rod worth and shutdown margin, the shutdown margin requirement, Specification 3.10.1.1, need not be observed provided the reactivity equivalent to at least the highest estimated control rod worth is available for trip insertion and all part length control rods are fully withdrawn. Each full length control rod not fully inserted, that is, the rods available for trip insertion, shall be demonstrated capable of full insertion when tripped from at least the 50% withdrawn position within 24 hours prior to reducing the shutdown margin to less than the limits of Specification 3.10.1.1. The position of each full length rod not fully inserted, that is, available for trip insertion, shall be determined at least once per 2 hours.

3.10.2 Power Distribution Limits and Misaligned Control Rod

3.10.2.1 The movable detector system shall be used to measure power distribution after each fuel reloading prior to operation of the plant at 50% of rated power to ensure that design limits are not exceeded.

If the core is operating above 75% power with one excore nuclear channel out of service, then the quadrant to

3.10.2.4 If the quadrant to average power tilt ratio exceeds 1.02 but is less than 1.12 for a sustained period of more than 24 hours without known cause, or if such a tilt recurs intermittently without known cause, the reactor power level shall be restricted so as not to exceed 50% of rated power. If the cause of the tilt is determined, continued operation at a power level consistent with 3.10.2.2 above, shall be permitted.

3.10.2.5 Except for physics test, if the quadrant to average power tilt ratio is 1.12 or greater, the reactor shall be put in the hot shutdown condition utilizing normal operating procedures. Subsequent operation for the purpose of measuring and correcting the tilt is permitted provided the power level does not exceed 50% of rated power and the Nuclear Overpower Trip "set point is reduced by 50%".

3.10.2.6 Following any refueling and at least every effective full power month thereafter, flux maps, using the movable detector system, shall be made to confirm that the hot channel factor limits of Specification 3.10.2.2 are met.

3.10.2.7 The reference equilibrium indicated axial flux difference as a function of power level (called the target flux difference) shall be measured at least once per equivalent full power quarter. The target flux difference must be updated at least each equivalent full power month using a measured value or by interpolation using the most recent measured value and the predicted value at the end of the cycle life. The target flux difference shall be between +5.0 and -7.5% at the beginning of cycle life and between +2.0 and -7.5% at the end of cycle life. Linear interpolation shall be used to determine values at other times in cycle life.

3.10.2.8 Except during physics tests, control rod excercises, excore detector calibration, and except as modified by 3.10.2.9 through 3.10.2.12, the indicated axial flux difference shall be maintained within $\pm$ 5% of the target flux difference (defines the target band on axial flux difference). Axial flux difference for power distribution control is defined as the average value for the four excore detectors. If one excore detector is out of service, the remaining three shall be used to derive the average.

3.10.2.9 Except during physics tests, control rod exercises, or excore calibration, at a power level greater than 90 percent of rated power, if the indicated axial flux difference deviates from its target band. The flux difference shall be returned to the target band immediately or the reactor power shall be reduced to a level no greater than 90 percent of rated power.

different from those resulting from operation within the target band. The instantaneous consequence of being outside the band, provided rod insertion limits are observed, is not worse than a 10 percent increment in peaking factor for flux difference in the range +14 percent to -14 percent (+11 percent to -11 percent indicated) increasing by +1 percent of each 2 percent decrease in rated power. Therefore, while the deviation exists the power level is limited to 90 percent or lower depending on the indicated flux difference.

If, for any reason, flux difference is not controlled within the ± 5 percent band for as long a period as one hour, then xenon distributions may be significantly changed and operation at 50 percent is required to protect against potentially more severe consequences of some accidents.

As discussed above, the essence of the limits is to maintain the xenon distribution in the core as close to the equilibrium full power condition as possible. This is accomplished, without part length rods, by using the chemical volume control system to position the full length control rods to produce the required indication flux difference.

The effect of exceeding the flux difference band at or below half power is approximately half as great as it would be at 90% of rated power, where the effect of deviation has been evaluated.

The reason for requiring hourly logging is to provide continued surveillance of the flux difference if the normal alarm functions are out of service. It is intended that this surveillance would be temporary until the alarm functions are restored.

The quadrant power tilt ratio limit assures that the radial power distribution satisfies the design values used in the power capability analysis. Radial power distribution measurements are made during startup testing and periodically during power operation.

The limit of 1.02 at which corrective action is required provides DNB and linear heat generation rate protection with x-y plane power tilts. A limiting tilt of 1.025 can be tolerated before the margin for uncertainity in F_q is depleted. Therefore, the limiting tilt has been set as 1.02. To avoid unnecessary power changes, the operator is allowed two hours in which to berify the tilt reading and/or to determine and correct the cause of the tilt. Should this action verify a tilt in excess of 1.02 which remains uncorrected, the margin for uncertainty in F_q and $F_{\Delta H}$ is reinstated by reducing the power by 2% for each percent of tilt above 1.0, in accordance with the 2 to 1 ratio above, or as required by the restriction peaking factors.

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

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

SUPPORTING AMENDMENT NO. 19 TO PROVISIONAL OPERATING LICENSE NO. DPR-18

ROCHESTER GAS AND ELECTRIC CORPORATION

R. E. GINNA NUCLEAR POWER PLANT

DOCKET NO. 50-244

Introduction

By application dated January 6, 1978, as supplemented by letters dated January 10, March 27, April 6, April 17, and April 25, 1978, Rochester Gas and Electric Corporation (the licensee) requested authorization to operate the R. E. Ginna Nuclear Power Station in Cycle 8 with reload fuel supplied by Exxon Nuclear Company, Inc., and requested a change to the Technical Specifications involving power distribution control limits.

Discussion

The R. E. Ginna Nuclear Power Station has operated seven fuel cycles with fuel supplied by Westinghouse Corporation. Cycle 8 will involve the first use of fuel from a different vendor, Exxon Nuclear Company, Inc. (ENC). The loading for Cycle 8 will consist of 32 new ENC fuel assemblies loaded at the periphery of the core and 89 exposed Westinghouse assemblies scatter loaded in the center of the core. All assemblies are of similar design with the ENC assemblies designed to be compatible with the other fuel assemblies. Reactor power level, core average linear heat rate and primary coolant system temperature and pressure for Cycle 8 will remain the same as for the previous cycle.

The licensee has stated that all technical specification limits for the previous cycle are applicable to Cycle 8, with the exception of one limit involving power distribution control. The licensee also proposed a change to the bases of the specifications involving power distribution control to reflect a revised methodology used in the reactor physics analyses for Cycle 8.

The licensee's analyses for Cycle 8 also include the first use of ENC analytical methods to verify the acceptability of Ginna operating limitations and safety margins.

The staff evaluation which follows, addresses the acceptability of the use of the ENC assemblies in Cycle 8 and the acceptability of the proposed changes in Technical Specification. The evaluation includes the staff's review of nuclear, thermal-hydraulic and accident analyses for Cycle 8 operation.

Evaluation

1. Design of the New Fuel

The new fuel assemblies for the core periphery were designed by Exxon Nuclear Corporation to be compatible with the Westinghouse depleted fuel assemblies that are to remain in the Ginna core.

The Exxon fuel design is similar to the Westinghouse fuel bundle design (References 1 and 2).

The Exxon fuel design criteria and fuel design calculations are discussed in Exxon reports submitted with the application for Fuel Cycle 8 operation. Those aspects of the fuel design important to safety have been reviewed by the staff and found acceptable. Those aspects are: (1) the fuel performance during LOCA; (2) fuel clad collapse and fuel densification; (3) fretting wear; and (4) the effect of fuel rod bowing on the departure from nucleate boiling ratio (DNBR).

The GAPEX code (Reference 3) was used to calculate stored energy for LOCA calculations. GAPEX has been reviewed and approved by the staff for fuel temperature and internal pressure calculations in PWR fuel (Reference 4).

Reference 1 presents calculations which show that the cladding will not collapse during Cycle 8. These calculations utilize the RODEX and COLAPX codes. The RODEX code (Reference 5) calculates the cladding temperature and fuel rod internal pressure while COLAPX (Reference 7) calculates the collapse time using the RODEX input. COLAPX has been reviewed by the staff and found acceptable for cladding collapse calculations. RODEX has not been approved by the staff but the models in RODEX affecting clad temperature and internal pressure are similar to those in the GAPEX code, which has been approved. Moreover, since the clad collapse analyses for the Westinghouse fuel does not predict collapse during Cycle 8, and since the cladding for the Exxon fuel is thicker than that of the Westinghouse fuel (Reference 2) which makes it more resistent to clad collapse, we have concluded, with reasonable assurance, that the results of the RODEX analysis are acceptable.

Exxon tests to determine the magnitude of fretting at the fuel rod axial spacer contact points due to flow induced vibration revealed no active fretting corrosion and negligible difference in wear observed between 500, 1000, and 1500 hours. Based on these test results and the larger diameter - thicker clad of the Exxon fuel rods in the 14 x 14 fuel assemblies for Ginna and therefore greater stiffness, we have concluded that fuel rod integrity with respect to flow induced vibration and fretting wear is acceptable.

The effect of fuel rod bowing on Departure from Nucleate Boiling Ratio (DNBR) has been a subject of continuing discussion between the staff and Exxon. An Exxon analysis considered the fuel rod bowing penalties for the most limiting transients and attempted to show that there is sufficient margin to offset the calculated penalties. These results are presented in Reference 2. The staff has concluded that these analyses are not completely acceptable because the heat flux and pressure used to calculate the bowing penalties were for minimum DNBR conditions and do not represent the worst conditions for calculating the rod bowing penalties. However, Reference 2 shows that there is an 8.5 percent margin to the safety limit which offsets this nonconservatism. On this basis, we have concluded that there is adequate thermal margin to assure safe plant operation without violating the minimum DNBR safety limit.

Based on successful irradiation experience of Exxon fuel assemblies in other PWR cores and the analyses which have been done for Ginna Fuel Cycle 8, we have concluded that the Exxon fuel assemblies for Cycle 8 will perform in a safe and acceptable manner. The licensee has agreed (RG&E telecon 4/14/78) to submit plans for inspection of the Exxon fuel assemblies to NRC for concurrence at least 90 days prior to the end of Fuel Cycle 8 to enable additional NRC review of the fuel prior to its use in Cycle 9.

2. Thermal Hydraulic Design

The new Exxon fuel assemblies are designed to have thermal hydraulic characteristics equivalent to those of the existing fuel. Therefore, there will not be any major differences in the thermal hydraulic behavior of the core.

The licensee has shown that at 118 percent of rated power, the calculated minimum DNBR is 1.47. The corresponding value for the Westinghouse fuel assemblies is 1.43. The fuel and cladding temperature analysis uses Exxon calculational methods (Reference 7), assuming maximum power peaking and engineering tolerances. The calculated maximum fuel and cladding temperatures are well below the design limits. We, therefore, conclude that the Exxon fuel assemblies are compatible with the Westinghouse fuel assemblies in the Ginna core and that the thermal hydraulic criteria will not be exceeded during plant operation.

3. Nuclear Design

The Fuel Cycle 8 loading will consist of 89 fuel assemblies with burnups ranging from 7,178 MWD/MTU to 23,813 MWD/MTU and 32 fresh ENC fuel assemblies.

The licensee has specified new values for the target flux difference. They are between +5.0 and -7.5% for the beginning of cycle life and between +2.0 and -7.5% for the end of cycle life. For the intermediate times the values are obtained by linear interpolation. The licensee has compared the neutronic characteristics of the Cycle 8 and Cycle 7 cores and concluded that they are approximately the same. The reactivity coefficients of the Cycle 8 core are bounded by the coefficients used in the safety analyses and we have concluded that the coefficients are acceptable.

Justification of the assumed total rod worth uncertainty of 10% used in the determination of shutdown margin has not been presented. Confirmatory tests are therefore included in the startup physics tests for fuel Cycle 8.

The physics startup test program for Ginna Cycle 8 presented in the March 27, 1978 submittal (Reference 2), was reviewed with the licensee. Several changes to the rod worth and power coefficient measurements were made. These changes are documented in the Reference 17 submittal. As part of this test program, control rod reactivity worth will be measured for banks D, C, B and A in order to verify that adequate shutdown margin is available. If any one bank worth differs from the predicted value by more than 15% or the sum of the worths of these banks differs from the predicted value by more than 10%, the first shutdown bank should be measured. If the sum of the five measured banks differs from the predicted value by more than 10%, additional shutdown bank measurements will be performed to verify the technical specification shutdown margin.

We have concluded that the total physics startup test program as modified is acceptable. However, there are areas in the licensee's safety analysis that warrant verification in the physics startup test program. Therefore, a summary report as described in the March 27th submittal (Reference 2) will be submitted to the NRC. The licensee has agreed to submit the report within 45 days of completion of the program.

4. Steady State and Load Follow Operation

Compliance with F_Q and $F\Delta H$ limiting conditions for operation is ensured by adherence to previously approved constant axial offset control strategy and core monitoring with incore and excore flux monitors. Incore monitoring is achieved using travelling fission chambers. Data from the fission chambers and calculated coefficients

(Reference 9) are processed by the computer code INCORE to obtain power distribution maps. Extensive comparisons of predicted and measured core power distributions have been performed by Exxon for 3 and 4 loop cores. In general, the results of these comparisons are favorable. However, R. E. Ginna is a two loop plant and there is only a single set of measured and calculated power distributions for R. E. Ginna, Cycle 7, at hot full power, 1000 MWD/MTU. The results of this comparison show good agreement between measurement and calculation and add credibility to the licensee's assertion that an F_Q uncertainty factor of 5% is appropriate for Cycle 8. However, additional data will be obtained during the fuel cycle 8 startup physics tests.

5. <u>Safety Analyses</u>

The licensee has analyzed the anticipated operating occurrences and postulated accidents using the plant transient simulator code PTSPWR (Reference 15). The results of these analyses are presented in Reference 14. Our review of this code has progressed sufficently to allow us to conclude that analyses using PTSPWR provide acceptable margins to peak linear heat generation rate and departure from nucleate boiling design limits. The reactivity coefficients assumed in the safety analyses are to be confirmed during the physics startup tests.

a. <u>Steam Line Break Analyses</u>

The Steam Line Break (SLB) accident analysis (Reference 14) is of particular concern. SLB analysis methods have not been generically approved. The licensee asserts that should a large SLB occur the plant would return to criticality, reaching a peak average core power of 22% of rated power at approximately 90 sec after accident initiation. The minimum DNBR at this condition, using the Macbeth critical heat flux correlation, would be 1.58. Even if DNB were to occur during a steam line break accident, DNB would be restricted to a small region of the core in the vicinity of the assumed stuck rod. It is noted that DNB anywhere in the core is unlikely if all control rods scram as expected. Of the fuel rods which might experience DNB in the vicinity of the stuck rod, some fraction would release their fission gas inventory. The fission gas would have to be transported to the secondary side of the coolant system (primary to secondary steam generator leakage) in order to represent a potential hazard. The potential release to the atmosphere would be significantly less than 10 CFR Part 100 limits. Accordingly, we have concluded that the consequences of a steam line break are acceptable.

b. <u>ECCS Analysis</u>

The licensee has submitted ECCS performance analyses for the Westinghouse (Reference 19) and new ENC fuels (Reference 1). The Westinghouse analysis was performed for Cycle 7 fuel which the staff believes is a conservative evaluation for the Westinghouse fuel during Cycle 8. The ENC analysis was performed for Cycle 8 using the ENC WREM-II ECCS evaluation model (Reference 7) which is described in References 8 and 9. The applicability of the model

to two-loop Westinghouse PWR plants was evaluated by ENC in Reference 10. The ENC evaluation model has been reviewed and approved conditionally by the NRC (Reference 16). The staff has recently considered whether the Westinghouse generic evaluation adequately represented the flow characteristics of the Westinghouse two loop units. The generic evaluation model assumes that all safety injection water is introduced directly into the lower plenum. For the two loop units, the safety injection water is injected into the upper plenum. Thus, the staff was concerned that the Westinghouse model did not consider interaction between UPI water and steam flow. (References 11 and 12). After plant specific submittals by the licensees operating two loop plants were reviewed, the staff concluded that the calculations provided by the licensees (with certain modifications to the staff's model) are acceptable as an interim basis for continued safe operation of the Westinghouse two loop plants, while long term efforts continue for developing a model specifically treating UPI. For the Ginna plant the calculations which specifically considered UPI using the modified version of the staff model, resulted in a change of only 15°F from those using the generic model in which the UPI-core interaction was not specifically considered (Reference 20). In the interim, before these models are developed, the licensee has provided a modification to the current Westinghouse model which accounts for UPI-core interaction (Reference 13). It was demonstrated that the modification resulted in the increase of peak clad temperature by 15°F. Since for the Ginna plant both ENC WREM-II and Westinghouse models predict similar PCT's (1922°F for ENC WREM-II and 1957°F for Westinghouse) it can be expected that the UPI modification, when applied to the ENC WREM-II model, would allow about the same increase in PCT. The licensee has drawn a similar conclusion and agreed to submit within 30 days, calculational results to confirm the validity of this conclusion. (Reference 21).

The ECCS analyses have been performed with the upper head fluid temperature equal to the fluid outlet (hot leg) temperature and assuming 10 percent of steam generator tubes plugged. The analyses included a spectrum of breaks which consisted of guillotine double ended cold leg (DEGCL) breaks with discharge coefficients of 1.0, 0.6 and 0.4 and split breaks with break areas of 8.25, 4.9 and 3.30 ft^2. No small break analysis was performed. The licensee has demonstrated, by showing analogy between the present analysis and the analyses performed previously for other plants, that the small break LOCA is not limiting (Reference 2). The critical break size was determined to be DEGCL with C_D=0.4.

The staff has concluded that although the Westinghouse and Exxon two-loop generic-evaluation models should be changed to consider upper plenum injection (unless the plant is modified), analyses at the specific operating conditions applicable to the Ginna plant demonstrate that the effect of disregarding upper plenum injection interaction on refill and reflood conditions will not be significant (less than 20°F PCT). Therefore, the staff believes that, for the limited range to which

the models are applied for conditions at the Ginna plant, the models do not deviate from the requirements of 10 CFR 50 Appendix K item I.D.3, and the calculations are acceptable.

On March 23, 1978 Westinghouse informed the NRC that an error in the West-ECCS evaluation model had been found which had resulted in incorrectly calculated peak clad temperatures in all LOCA analyses previously submitted by their customers. For several plants preliminary estimates indicated that they would not meet the 2200°F limit of 10 CFR 50.46 at their present maximum overall peaking factor limits. Westinghouse and several of their customers met with the NRC staff on March 29, 1978 in Bethesda to discuss the error and its impact on specific plant analyses. Subsequent to that meeting, Westinghouse provided information through the licensees of operating reactors to justify continued operation at the interim peaking factor Technical Specification limits proposed by the NRC staff on April 3, 1978.

On April 17, 1978 (Reference 19) RG&E submitted a letter indicating that continued operation at their present Technical Specification limit of 2.32 (total peaking factor) was justified on the basis of additional generic Westinghouse analyses. Westinghouse had determined that the impact of correcting the error on the peak cladding temperature for the RE Ginna plant was significant but within the presently existing margin (228°F) to the 2200°F° acceptance criteria limit. The NRC Staff confirmed the conservatism of that and all other plant evaluations and on April 18, 1978 published a Safety Evaluation Report (Reference, attachment to Exemption). Since the Westinghouse and ENC fuels were analyzed using the respective Westinghouse and ENC evaluation models, and since there is no zirconium-water error in the ENC calculational model, the error in zirconium-water reaction in the Westinghouse calculational model has no effect on the Exxon calculations. The Zirconium-water reaction error in the Westinghouse model is the subject of an exemption request by the licensee dated April 25, 1978, (Reference 21) and a separate exemption action by NRC.

6. Technical Specification Changes

The proposed addition to the Technical Specifications restricts the permissable range of the target flux difference i.e. the ratio of the flux in the top half of the core minus the flux in the lower half of of the core to the total flux measured at 100% power, equilibrium conditions. The addition, Technical Specification 3.10.2.7, assures that axial power distributions realized in the reactor will be no more limiting with respect to linear heat generation rate than the axial power distributions used by Exxon to analytically confirm (Reference 18) that, limiting values of linear heat generation vs core height, Technical Specification 3.10.2.2, will not be violated. The restriction has been reviewed and approved on a generic basis and has recently been incorporated in the Technical Specifications of PWR's using Exxon Nuclear fuel.

The change to Technical Specification 3.10.1.4 and the addition of specification 3.10.1.6 are required to permit the physics testing program as discussed in part 3 of our evaluation. The change and the addition are in accordance with the Standard Technical Specifications for Westinghouse PWR's which we have already reviewed and approved.

The changes to the basis of the Technical Specification related to core power distribution are in accordance with the Standard Technical Specification which we have approved and are therefore acceptable also.

Environmental Consideration

We have determined that the amendment does not authorize a change in effluent types or total amounts nor an increase in power level and will not result in any significant environmental impact. Having made this determination, we have further concluded that the amendment involves an action which is insignificant from the standpoint of environmental impact, and pursuant to 10 CFR §51.5(d)(4), that an environmental impact statement or negative declaration and environmental impact appraisal need not be prepared in connection with the issuance of this amendment.

Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

Date: May 1, 1978

REFERENCES

(1) Letter from LeBoef, Lamb, Leiby and MacRae (Counsel for Rechester Gas and Electric Corporation) to E. G. Case (NRC), dated January 6, 1978.

(2) Letter from L. D. White, Jr. (Rochester Gas and Electric Corporation) to D. L. Ziemann (NRC), dated March 27, 1978.

(3) XN-73-25, "GAPEX: A Computer Program for Predicting Pellet-to-Cladding Heat Transfer Coefficients", June 1975.

(4) USNRC Report, "Technical Report on Densification of Exxon Nuclear PWR Fuel", February 27, 1975.

(5) XN-76-8(P), "RODEX: Fuel Rod Design Evaluation Code", February 1977.

(6) XN-72-23, "Clad Collapse Calculational Procedure", November 1, 1972.

(7) XN-NF-77-58, "ECCS analysis for the R. E. Ginna Reactor with ENC WREM-II PWR Evaluation Model", December 1977.

(8) XN-75-41, "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model", Vol I through III, July-August 1975 and Supplements 1 through 7, August-November 1975.

(9) XN-76-27, "Exxon Nuclear Company WREM-Based Generic PWR ECCS Evaluation Model Update ENC WREM-II", July 1976 and Supplements 1 and 2, September-November 1976.

(10) XN-NF-77-25, "Exxon Nuclear Company ECCS Evaluation of a 2-loop Westinghouse PWR with Dry Containment using the ENC WREM-II ECCS Model - Large Break Example Problem," August 1977.

(11) Letter from E. G. Case (NRC) to L. D. White, Jr. (Rochester Gas and Electric Corporation), dated December 16, 1977.

(12) Letter RG&E to NRC, Development of a New Model to Account for Upper Plenum Injection, dated March 5, 1978.

(13) Letter from L. D. Amish (Rochester Gas and Electric Corporation) to A. Schwencer (NRC), dated February 1978.

(14) XN-NF-77-40, "Plant Transient Analysis for the R. E. Ginna Unit 1 Nuclear Power Plant", November 1977.

(15) XN-74-5, "Description of the Exxon Nuclear Plant Transient Simulation Model for Pressurized Water Reactors (PTSPWR)," Revision 1, May 1975.

(16) USNRC Topical Report Evaluation, Exxon Nuclear Company Report XN-NF-77-25, April 1978.

(17) Letter from L. D. White, Jr. (Rochester Gas and Electric Corporation) to D. L. Ziemann (NRC), dated April 6, 1978.

(18) Exxon Nuclear Power Distribution Control for Pressurized Water Reactors XN-76-40, September 1976.

(19) Letter from L. D. White, Jr., (RG&E) to A. Schwencer (NRC) dated April 7, 1977.

(20) Letter to RG&E dated April 28, 1978 transmitting staff SER of UPI model evaluation.

(21) Letter from RG&E to NRC dated April 25, 1978, related to ENC UPI calculations.

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

In the Matter	)	
	)	
ROCHESTER GAS AND ELECTRIC CORPORATION	)	Docket No. 50-244
	)	
(R. E. Ginna Nuclear Power Plant	)	

EXEMPTION

I.

The Rochester Gas and Electric Corporation (the licensee), is the holder of Provisional Operating License No. DPR-18 which authorizes the operation of the nuclear power reactor known as R. E. Ginna Nuclear Power Plant (the facility) at steady reactor power levels not in excess of 1520 megawatts thermal (rated power). The facility consists of a Westinghouse Electric Company designed pressurized reactor (PWR) located at the licensee's site in Wayne County, New York.

II.

In accordance with the requirements of the Commission's ECCS Acceptance Criteria 10 CFR 50.46, the licensee submitted on April 7, 1977 and January 6, 1978 ECCS evaluations for proposed operation using 14 x 14 fuel manufactured by the Westinghouse Electric Company and the Exxon Nuclear Company (ENC). These evaluations established limits on the peaking factor based upon ECCS evaluation models developed by the Westinghouse Electric Company (Westinghouse), the designer of the Nuclear Steam Supply System for this facility, and by Exxon, the supplier of the reload fuel. The Westinghouse and ENC ECCS evaluations

models had been previously found to conform to the requirements of the Commission's ECCS Acceptance Criteria, 10 CFR Part 50.46 and Appendix K. The evaluations indicated that with the peaking factor limited as set forth in the evaluations and with other limits set forth in the facility's Technical Specifications, the ECCS cooling performance for the facility would conform with the criteria contained in 10 CFR 50.46(b) which govern calculated peak clad temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry and long-term cooling.

On March 23, 1978 Westinghouse informed the Nuclear Regulatory Commission (NRC) that an error had been discovered in the fuel rod heat balance equation which resulted from the incorrect use of only half of the volumetric heat generation due to metal-water reaction in calculating the cladding temperature. Thus, the LOCA analyses previously submitted to the Commission by licensees of Westinghouse reactors were in error. The error identified would result in an increase in calculated peak clad temperature, which, for some plants, could result in calculated temperatures in excess of 2200°F unless the allowable peaking factor was reduced somewhat. Westinghouse identified a number of other areas in the approved model which Westinghouse indicated contained sufficient conservatism to offset the calculated increase in peak clad temperature resulting from the correction of the error noted above. Four of these areas were generic, applicable to all plants, and a number of others were plant specific. As outlined in the NRC Staff's Safety Evaluation Report (SER) of April 16, 1978 (attached), the staff determined that some of these

modifications would be appropriate to offset to some extent the penalty resulting from correction of the error. The attached SER of April 18, 1978 sets forth the value for each modification applicable to each facility.

As part of the proposed change to the technical specifications the licensee has submitted information and analyses to permit Cycle 8 operation with reshuffled Westinghouse fuel and with 32 Westinghouse fuel assemblies replaced with fresh fuel assemblies manufactured by the Exxon Nuclear Company (ENC) and loaded on the periphery of the core. Based on an analysis of the information presented by the licensee, the staff has concluded that the new fuel manufactured by Exxon Nuclear Company (ENC) is both similar to and compatible with the fuel previously supplied by Westinghouse. The ENC calculations for the ENC fuel for the Ginna Core are not affected by the Westinghouse error. (Safety Evaluation for the reload application dated May 1, 1978). The staff's evaluation determined that the impact of correcting the Westinghouse Zirconium-water reaction error on the peak cladding temperature for the RE Ginna plant is less than the presently existing margin (228°F) to the 2200°F acceptance criteria limit. The NRC Staff has confirmed that the impact of correcting the error in the Westinghouse ECCS evaluation model as it relates to the use of Westinghouse fuel is conservative, based on the April 18, 1978 Safety Evaluation Report.

Although revised computer calculations correcting the error, noted above, and incorporating the modifications described in the Staff's April 18, 1978 SER have not been run for each plant, the various parametric studies that have been made for various aspects of the approved Westinghouse model over the course of time provide a reasonable basis for concluding that when final revised calculations for the facility are submitted using the revised and corrected Westinghouse model, they will demonstrate that operation will conform to the criteria of 10 CFR 50.46(b), when operated at the peaking factors set forth in the SER of April 18, 1978. Such revised calculations fully conforming to 10 CFR 50.46 are to be provided for the facility as soon as possible.

Operation of the facility would nevertheless be technically in non-conformance with the requirements of §50.46, in that specific computer runs for the particular facility employing revised models with the Westinghouse metal-water error corrected and with the proposed model changes considered, as a complete entity will not be complete for some time. However, operation as proposed in the licensee's application dated January 6, 1978, and at the peaking factor limit specified in this Exemption will assure that the ECCS system will conform to the performance criteria of §50.46. Accordingly, while the actual computer runs for the specific facility are carried out to achieve full compliance with 10 CFR §50.46, operation of the facility will not endanger life or property or the common defense and security.

In the absence of any safety problem associated with operation of the facility during the period until the computer computations are completed, there appears to be no public interest consideration favoring restriction of the operation of the captioned facility. Accordingly, the Commission has determined that an exemption in accordance with 10 CFR §50.12 is appropriate. The specific exemption is limited to the period of time necessary to complete computer calculations.

IV.

Copies of the Safety Evaluation Report dated April 18, 1978, and the following documents are available for inspection at the Commission's Public Document Room at 1717 H Street, Washington, D. C. 20555, and at the Rochester Public Library, 115 South Avenue, Rochester, New York 14627.

(1) Licensee submittals dated April 7, 1977, January 6, 1978, and April 25, 1978.

(2) Amendment No. 19 to License No. DPR-18 and the related Safety Evaluation for the reload application, and

(3) This Exemption in the matter of RE Ginna Nuclear Power Plant.

Wherefore, in accordance with the Commission's regulations as set forth in 10 CFR Part 50, the licensee is hereby granted an exemption from the requirements of 10 CFR §50.46(a)(1) that ECCS performance be calculated in accordance with an acceptable calculational model which conforms to the provisions in Appendix K, without errors discussed herein. This exemption is conditioned as follows:

(1) As soon as possible, the licensee shall submit a reevaluation of ECCS cooling performance calculated in accordance with the Westinghouse Evaluation Model, and approved by the NRC staff and corrected for the errors described herein.

(2) Until further authorization by the Commission, the Technical Specification limit for total nuclear peaking factor (F_Q) for the facility shall be limited to 2.32.

FOR THE NUCLEAR REGULATORY COMMISSION

Victor Stello, Jr., Director
Division of Operating Reactors
Office of Nuclear Reactor Regulation

Attached:
Safety Evaluation Report,
dated April 18, 1978

Dated at Bethesda, Maryland
this 1st day of May, 1978

April 18, 1978

Safety Evaluation Report

Error in Westinghouse ECCS Evaluation Model

Introduction

Westinghouse was informed on March 21, 1978 by one of their licensees that an error had been discovered in their ECCS Evaluation Model. This error was common to both the blowdown and heatup codes. Westinghouse determined by analyses that the fuel rod heat balance equation in the LOCTA IV & SATAN VI codes was in error and that the LOCA analyses previously submitted by their customers were incorrect and predicted PCT's which were too low. Westinghouse determined that only half of the volumetric heat generation due to metal-water reaction was used in calculating the cladding temperatures and that an unreviewed safety question existed since preliminary estimates indicated that some plants would not meet the 2200°F limit of 10 CFR 50.46 without a reduction in overall peaking factor limit. Westinghouse notified their customers and NRC on March 23, 1978 while the utilities notified NRC through the regional I&E Offices.

Promptly upon notification by Westinghouse, the staff assessed the immediate safety significance of this information. The staff noted certain points that indicated no immediate action was required to assure safe operation of the plants. First, most plants operate at peaking factors significantly below the maximum peaking factor used for safety calculations. By making safety computations at factors higher than actual operating levels, the facility has a wide range of flexibility, without the need for hour to hour recomputations of core status. The difference between the actual peaking factors and the maximum calculated peaking factors, for most plants, would offset the penalty resulting from the correction of the error. Second, for most reactors there are plant-specific parameters which bear upon aspects of the ECCS performance calculations. Utilities do not generally take credit for these plant-specific parameters, preferring to provide a simpler computation which conservatively disregards these individually small credits. Third, the error in the Westinghouse computations relates to the zirconium-water reaction heat source. This is an aspect of Appendix K, which is generally recognized to be very conservative. New experimental data indicate that the methods required by Appendix K appreciably over-estimate the heat source. Thus, while the error in fact entails a deviation from a specific requirement of Appendix K, it does not entail a matter of immediate safety significance.

Westinghouse continued to evaluate the impact of the error on previous plant specific LOCA analyses and performed scoping calculations, sensitivity studies and some plant specific reanalyses. In addition, Westinghouse investigated several modifications to the previously approved methods which if approved by the NRC staff would offset some of the immediate impact of the error on Technical Specifications limits and plant operating flexibility.

On March 29, 1978, Westinghouse and several of their customers met with members of the NRC staff in Bethesda. Westinghouse described in detail the origin of the error, explained how it affected the LOCA analyses, and how the error had been corrected and characterized its effect on current plant specific analyses. In order to avoid reduction in overall peaking factors (F_q), Westinghouse presented a description of three proposed ECCS-LOCA evaluation model modifications which would contribute a compensating reduction of PCT. They were characterized as follows:

1) Revised FLECHT 15 x 15 heat transfer correlation.

   This new reflood heat transfer correlation which had been recently developed and submitted by Westinghouse (Reference (1) was proposed as a replacement for the currently approved FLECHT correlation. To determine the benefit, the proposed correlation was incorporated into the LOCTA IV heatup code and was found to result in improved heat transfer during the reflood portion of the LOCA.

2) Revised Zircaloy Emissivity.

   Based on recent EPRI data (Reference 2), Westinghouse proposed to modify the presently approved equation for zircaloy cladding emissivity to a constant value of 0.9. The higher emissivity (previously below 0.8) provides increased radiative heat transfer from the hot fuel pin during the steam cooling period of reflood.

3) Post-CHF heat transfer.

   Westinghouse proposed to replace their present post-CHF transition boiling heat transfer correlation with the Dougall-Rohsenow film boiling correlation (Reference 3) which they stated was included in Appendix K to 10 CFR Part 50 as an acceptable post-CHF correlation.

These three model modifications were classified as generic, applicable to all plant analyses. Subsequently, as discussed below, these changes were rejected by the staff as providing generic benefit. However, a portion of the credit proposed by Westinghouse was approved by the staff to certain specific plants, which had provided specific calculations with the new 15 x 15 correlation. During the period March 29 to April 18, 1978, Westinghouse provided the staff with additional sensitivity analyses and plant specific analysis in which they evaluated the effects of some changes to plant-specific inputs in the LOCA analyses. These were as follows:

1. <u>Assumed Plant Power Level</u>

   A reduction of the plant power level assumed in the SATAN VI blowdown analyses from 102% of the Engineered Safeguards Design Rated Power (ESDR) level to 102% of rated power was proposed. Previously, analyses had been performed at approximately 4.5% over the rated power. This change was worth approximately 0.01 in F_Q, and is referred to as ΔFESDR in Table 1.

2. <u>COCO Code Input</u>

   A modification to the COCO code input (Reference 3) to more realistically model the painted containment walls was proposed. Since the paint on containment walls provides additional resistance to heat loss into the walls, the COCO code calculates an increase in containment back pressure, which results in a benefit to the calculated peak cladding temperature of 0 to 40°F, during the reflooding transient. The magnitude of the benefit is dependent on the type of plant and the heat transfer properties of the paint, and results in up to 0.03 benefit in F_Q, and is referred to as ΔF_{CP} in Table 1.

3. <u>Initial Fuel Pellet Temperature</u>

   A modification of the initial fuel pellet temperature from the design basis to the actual as-built pellet temperatures was proposed. In the present LOCA calculations, Westinghouse has assumed margins in the initial pellet temperature. The margin available in plant-specific ranges from 28°F to 55°F. Use of the actual pellet temperature rather than the assumed value results in a reduction in pellet temperature (stored energy) at the end of blowdown, as calculated by the SATAN code, of approximately 1/3 of the initial pellet temperature margin. Westinghouse has provided sensitivity analyses which indicate that a 37°F reduction in fuel pellet temperature at end of blowdown is worth approximately 0.1 in F_Q. This is referred to as ΔF_{PT} in Table 1.

4. Accumulator Water Volume Consideration

Westinghouse has evaluated the effect on ECCS performance of reducing the accumulator water volume, and has determined that for those plants for which the downcomer is refilled before the accumulators are emptied, there is a benefit in PCT. The sensitivity studies have indicated that this benefit in F_Q is plant-specific. This is referred to as ΔF_{ACV} in Table 1.

5. Steam Generator Tube Plugging Consideration

In previous analyses, Westinghouse has assumed values of steam generator tube plugging which were greater than the actual plant-specific degree of plugging. Sensitivity analyses submitted in Reference 4 were used to evaluate the benefit available by realistically representing the plant-specific data. For the plants affected, the benefit in PCT ranged from 7 to 66°F which was conservatively worth from 0.007 to .066 F_Q. This is referred to as ΔF_{SG} in Table 1.

Safety Evaluation

The information provided by Westinghouse was separated into two categories; the generic evaluation model modifications and the plant specific sensitivity studies and reanalyses. The NRC staff reviewed the peaking factor limits proposed by Westinghouse to verify their conservatism.

The metal-water reaction heat generation error in the Westinghouse ECCS evaluation model was evaluated by the staff to determine an appropriate interim penalty. Westinghouse provided two preliminary separate effects calculations which indicated that a maximum penalty of from 0.14 to 0.17 was appropriate to compensate for the model error. As indicated in Reference 5, the staff conservatively rounded up this penalty to 0.20.

As is noted above, Westinghouse had proposed several compensating generic changes in their evaluation model to offset any necessary reductions in peaking factor due to the error. These changes were assessed by the staff and as noted in Reference 5.

1) No credit was given at this time, for the changes in the post-CHF heat transfer correlation and new zircaloy emissivity data..

2) Partial credit (70%) would be given at this time for the use of the new 15 x 15 FLECHT correlation only for plants which had provided a specific calculation demonstrating that such credit was appropriate.

Based on this review the staff developed recommended interim peaking factor limits for all the operating plants and recommended that any other plant specific interim factors (benefits) not related to the generic review be considered separately. In addition, the staff reviewed plant specific reanalyses for DC Cook, Units 1 & 2, Zion, Units 1 & 2, and Turkey Point, Unit 3 which had corrected the error in metal water reaction. In these analyses the Dougall-Rohsenow and zircaloy emissivity credits were not considered, while the new 15 x 15 FLECHT correlation was included. The staff concluded that these reanalyses could serve as a basis for conservatively determining interim peaking factor limits for these plants.

For most of the operating plants the staff's generic review resulted in a lower allowable peaking factor than Westinghouse had proposed. However, in one case, Westinghouse had proposed more limiting peaking factors in order to prevent clad temperatures at the rupture node from exceeding 2200°F. The staff concluded that it would be properly conservative to use the minimum of these values.

Based on plant specific sensitivity studies, performed by Westinghouse, the licensees submitted requests for interim plant specific benefits. The staff reviewed these sensitivity studies and recommended that appropriate credits be accepted. The results of these analyses are shown in Table 1.

We informed each licensee by telephone on April 3, 1978, that he should administratively reduce his peaking factor limit from the limit contained in his Technical Specifications to the interim peaking factor limit contained in the right hand column of Table 1. In those cases where the limit in Table 1 is 2.32, this represents no change from the Technical Specifications limit. The peaking factor limit of 2.32 is generically supported and approved for Westinghouse reactors employing constant axial offset control operating procedures.

For the reactor having an interim peaking factor limit of 2.31, we requested no further justification of the limit. This is because the generic analysis supporting the limit of 2.32 approaches the limit only at beginning of the first cycle. Since the affected reactors have operated past this point, it is clear that the maximum attainable peaking factor will be less than 2.32. While this margin has not been quantified, the staff is convinced it is substantially greater than the 0.01 for which we are requiring no additional justification from the plants with an interim limit of 2.31.

For the reactors with an interim limit less than 2.31, we requested that the licensee furnish administratively imposed procedures to replace Technical Specifications either:

1. To provide a plant specific constant axial offset control analysis of 18 cases of load following which would ensure that the interim limit would not be exceeded in normal operation of the power plant, or, at his option, if such analysis were unobtainable, inappropriate or insufficient,

2. To institute procedures for axial power distribution monitoring of the interim limit using a system designed for this purpose or manual procedures as indicated in Standard Technical Specifications 3/4 2.6 and ancillary Specifications.

We requested the licensees to provide indication that they have adopted the above interim LOCA analyses, interim peaking factor limits and administrative procedures by April 10, 1978, if their reactors were operating, and by April 17, 1978, if the reactors were not operating.

Conclusion

We conclude that when final revised calculations for the facility are submitted using the revised and corrected model, they will demonstrate that with the peaking factor set forth herein, operation will conform to the criteria of 10 CFR 50.46(b). Such revised calculations fully conforming to 10 CFR 50.46(b) are to be provided for the facility as soon as possible.

As discussed herein, the peaking factor limit specified in Table 1, in combination with any necessary operating surveillance requirements, will assure that the ECCS will conform to the performance requirements of 10 CFR 50.46(b). Accordingly, limits on calculated peak clad temperature, maximum cladding oxidation, maximum hydrogen generation, coolable geometry and long term cooling provide reasonable assurance that the public health and safety will not be endangered.

References

(1) R. S. Dougall, W. M. Rohsenow, "Film Boiling on the Inside of Vertical Tubes with Upward Flow of the Fluid at Low Qualities", MIT Report 9079-26, September 1963.

(2) EPRI Report NP-525, "High Temperature Properties of Zircaloy-Oxygen Alloy", March 1977.

(3) WCAP-9220, "Westinghouse ECCS Evaluation Model, February 1978 Version", February 1978.

(4) WCAP 8986 "Perturbation Technique for Calculating ECCS Cooling Performance", February, 1978.

(5) DSS SER "Metal-Water Reaction Heat Generation Error in Westinghouse ECCS Evaluation Model Computer Programs", Z. R. Rosztoczy to D. F. Ross/D. G. Eisenhut, 4/7/78.

(6) T. Morita, et al., "Power Distribution Control and Load Following Procedures," WCAP-8385 (Proprietary) and WCAP-8403 (Non-Proprietary), September, 1974.

TABLE 1 F_Q Analysis	PCT °F	F_Q OLD	ΔF_T	ΔF_{ZrO_2}	ΔF_{FLECHT}	F_{PCT}	F_{SE}	$F_{Q,MIN}$	ΔF_{ESDR}	ΔF_{CP}	ΔF_{PT}	ΔF_{SG}	ΔF_{ACV}	F_Q LIMIT
2 Loop														
Pt. Beach 1	2025	2.32	.16	-.2	-	2.28	2.32	2.28	.01	-	-	.029	-	2.32
Pt. Beach 2	2025	2.32	.16	-.2	-	2.28	2.32	2.28	.01	-	-	.066	-	2.32
Ginna	1972	2.32	.26	-.2	-	2.32	2.32	2.32	-	-	-	.053	-	2.32
Kewaunee	2172	2.25	.03	-.2	.05	2.13	2.25	2.13	.01	.02	-	-	-	2.16 -
Prairie Island 1/2	2187	2.32	.01	-.2	.05	2.18	2.26	2.18	.01	.02	-	-	.03	2.24(+)
3 Loop														
North Anna	2181	2.32	.02	-.2	-	2.14	2.32	2.14	-	-	-	-	-	2.14
Beaver Valley	2041	2.32	.15	-.2	-	2.27	2.32	2.27	-	-	.036	-	-	2.31
Farley	1991	2.32	.24	-.2	-	2.32	2.32	2.32	.01	.005	-	-	-	2.32
Surry 1	2177	1.85.	.02	-.2	.06	1.73	1.84	1.73	-	.03	.025	.023	-	1.81
Surry 2	2177	1.85	.02	-.2	.06	1.73	1.84	1.73	-	.03	.025	.023	-	1.81
Turkey Point 3	2019*	1.90	.14	0	-.03	2.01	2.05	2.01	-	-	-	.020	-	2.03
Turkey Point 4	2195	2.05	.00	-.2	.05	1.90	1.91	1.90	-	-	-	.01	-	1.91
4 Loop														
Indian Point 2	2086	2.32	.11	-.2	-	2.23	2.23	2.23	.01	-	-	-	-	2.24
Indian Point 3	2125	2.32	.07	-.2	.06	2.25	2.19	2.19	.01	-	.03	-	-	2.23
Trojan	1975	2.32	.26	-.2	-	2.32	2.32	2.32	.01	-	.037	-	-	2.32
Salem 1	2135	2.32	.06	-.2	-	2.18	2.32	2.18	.01	-	.024	-	-	2.21
Zion 1/2	2189**	2.07	-	0	-.03	2.04	-	2.04	-	-	-	-	-	2.04(+)
Cook 1	2161*	1.90	.03	0	-.03	1.90	1.98	1.90	-	-	-	-	-	1.90
Cook 2	2190*	2.10	.01	0	0	2.11	-	2.11	0	0	0	0	0	2.11

ΔF_T - Credit in F_Q for PCT margin to 2200°F limit.

ΔF_{ZrO_2} - Metal Water Reaction penalty on F_Q.

ΔF_{FLECHT} - Credit in F_Q for improvements to 15x15 FLECHT Correlation.

F_{PCT} - Staff estimated F_Q based on 2200°F PCT limit.

F_{SE} - Westinghouse proposed F_Q based on stored energy sensitivity studies.

*Denotes reanalysis at F_Q old value error corrected.

**Denotes reanalyses at F_Q old value, error corrected, accumulator Vol. Change of 100 ft^3, accumulator pressure of 650 psia

(+) These limits are applicable assuming licensee modifies accumulator conditions as appropriate. If not, Prairie Island 1/2 F_Q=2.21, Zion 1/2 F_Q=1.9

7590-01

UNITED STATES NUCLEAR REGULATORY COMMISSION

DOCKET NO. 50-244

ROCHESTER GAS AND ELECTRIC CORPORATION

NOTICE OF ISSUANCE OF AMENDMENT TO PROVISIONAL OPERATING LICENSE

The U. S. Nuclear Regulatory Commission (the Commission) has issued Amendment No. 19 to Provisional Operating License No. DPR-18, issued to Rochester Gas and Electric Corporation (the licensee), which revised the Technical Specifications for operation of the R. E. Ginna Plant (facility) located in Wayne County, New York. The amendment is effective as of its date of issuance.

The amendment changes the Appendix A Technical Specifications to support operation in Cycle 8 with reload fuel by Exxon Nuclear Company (ENC). This fuel has been designed by ENC to be compatible with the fuel supplied previously by Westinghouse. In addition, the amendment allows Technical Specification changes that are required for startup tests.

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.

Notice of proposed Issuance of Amendment to Facility Operating License in connection with this action was published in the FEDERAL REGISTER on February 21, 1978 (43 FR 7275). No request for a hearing or petition for leave to intervene was filed following notice of the proposed action.

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 Commission's Order for Modification of License dated August 27, 1976, (2) the application for amendment dated January 6, 1978, and supplements thereto dated January 10, 1978, March 27, 1978, April 6, 1978, April 17, 1978, and April 25, 1978, (3) Amendment No. 19 to License No. DPR-18, (4) the Commission's related Safety Evaluation, and (5) the Exemption related to the requirements of 10 CFR 50.46(a)(1) and the Safety Evaluation dated April 18, 1978, attached thereto. All of these items are available for public inspection at the Commission's Public Document Room, 1717 H Street, N. W., Washington, D.C. and at the Rochester Public Library, 115 South Avenue, Rochester, New York 14627.

A copy of items (1), (3), (4), and (5) may be obtained upon request addressed to the U. S. Nuclear Regulatory Commission, Washington, D. C. 20555, Attention: Director, Division of Operating Reactors.

Dated at Bethesda, Maryland, 1st day of May, 1978.

FOR THE NUCLEAR REGULATORY COMMISSION

Dennis L. Ziemann

Dennis L. Ziemann, Chief
Operating Reactors Branch #2
Division of Operating Reactors

April 26, 1978

Edward Luton, Esq., Chairman
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. Franklin C. Daiber
College of Marine Studies
University of Delaware
Newark, Delaware 19711

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Dr. Emmeth A. Luebke
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

In the Matter of
Rochester Gas & Electric Corporation
(R. E. Ginna Nuclear Power Plant, Unit No. 1)
Docket No. 50-244

Gentlemen:

Enclosed for the information of the Licensing Board is an NRC Staff memorandum which discusses certain information concerning behavior of iodine during postulated steam generator tube rupture accidents.

Sincerely,

Edward G. Ketchen
Counsel for NRC Staff

Enclosure
Memorandum fm R. H. Vollmer to D. B. Vassallo
dtd February 22, 1978

cc: (w/enclosure)
Leonard M. Trosten, Esq.
Mr. Michael Slade
Rochester Committee for Scientific Information
Jeffrey Cohen, Esq.
Warren B. Rosenbaum, Esq.
Atomic Safety and Licensing Board Panel
Atomic Safety and Licensing Appeal Board
Docketing and Service Section

OFFICE➤	OELD					
SURNAME➤	~~EKetchen/dkw~~ EReis					
DATE➤	4/ /78					

NRC FORM 318 (9-76) NRCM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1976 — 626-624

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[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]	[illegible]

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

FEB 2 2 1978

NOTE TO: Domenic B. Vassallo, Assistant Director for Light Water Reactors, DPM.

FROM: Richard H. Vollmer, Assistant Director for Site Analysis, DSE

SUBJECT: CONSULTANT REPORT REGARDING NON-CONSERVATISM IN STAFF MODEL

As a result of a technical assistance contract with a staff consultant, a technical report (NUREG-0409) on "Iodine Behavior in a PWR Cooling System Following a Postulated Steam Generator Tube Rupture Accident," by A. K. Postma and P. S. Tam, was published in January 1978.

The report is a theoretical study of the iodine behavior in the primary and secondary coolant systems of a PWR following a postulated steam generator tube rupture. The report concludes that, as a result of such a rupture, primary coolant water containing iodine would be atomized by hydrodynamic forces as it flashed through the leak path into the steam system. The removal of iodine by the secondary water was predicted to be highly dependent upon the primary-to-secondary pressure difference and upon the water depth. Calculations made in the report, and which the report emphasized were designed to yield conservative predictions, indicated that in the early part of the accident less than 50% of the iodine might be removed by the secondary water, whereas in the later phases of the accident, about 99% of the iodine would be removed. Although the report attempted to assess the iodine removal by steam separators it did not examine possible iodine removal due to the proximity of neighboring tubes and other submerged structures in the steam generator.

The present staff model, as outlined in Standard Review Plan 15.6.3, assumes that a constant value of 90% of the iodine transferred to the secondary water is removed and retained in it. Therefore, NUREG-0409 implies that the present staff model may be non-conservative in the early phases of the accident, but may be overly conservative in the later phases.

The overall degree of conservatism or non-conservatism of the staff's present model cannot easily be assessed without a much more detailed examination. However, some perspective regarding the implications may

be gained by observing that the present staff model predicts the radiological consequences of a steam generator tube rupture coincident with a large iodine spike to be about 75 rem to the thyroid for a typical PWR at a site with poor ($X/Q = 1 \times 10^{-3}$ sec/m^3) meteorology. We can conclude from this that even if the staff's model was less conservative throughout the accident by as much as a factor of four, that our conclusions regarding the acceptability of this event would not likely change.

The staff is currently taking action in this matter in two ways. First, the staff is preparing and evaluating a more detailed model to be incorporated in its revised Standard Review Plan in this area which will allow for a time-dependent iodine retention fraction in the secondary water. Second, the staff is planning to have experiments performed, as suggested by NUREG-0409, that will confirm or refute the values indicated by the report.

We believe, in view of the possibility of a non-conservative staff model in this regard, that the licensing boards currently in progress for all PWR plants should be duly informed.

Richard H. Vollmer, Assistant Director
for Site Analysis
Division of Site Safety and
Environmental Analysis

NRC Central

April 26, 1978

Edward Luton, Esq., Chairman
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dr. Franklin C. Daiber
College of Marine Studies
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Dr. Emmeth A. Luebke
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

In the Matter of
Rochester Gas & Electric Corporation
(R. E. Ginna Nuclear Power Plant, Unit No. 1)
Docket No. 50-244

Gentlemen:

During the course of the NRC Staff's continuing studies of ECCS performance characteristics for pressurized water reactors, the Staff has identified certain aspects of accumulator delivery which should be considered further. This matter is discussed in the enclosure to this letter (and in the NRC Staff memorandum attached to the enclosure).

For reasons outlined in the enclosure, the Staff does not believe that this matter has an adverse effect on this proceeding.

Sincerely,

Edward G. Ketchen
Counsel for NRC Staff

Enclosure
"Accumulator Delivery," and attached memo fm
D. F. Ross, Jr., to U.S. Standard Problem Participants

cc: (w/enclosure)
Leonard M. Trosten, Esq.
Mr. Michael Slade
Rochester Committee for Scientific Information
Jeffrey Cohen, Esq.
Warren B. Rosenbaum, Esq.
Atomic Safety and Licensing Board Panel
Atomic Safety and Licensing Appeal Board
Docketing and Service Section

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DATE➤	4/24/78					

Form AEC-318 (Rev. 9-53) AECM 0240 ☆ U. S. GOVERNMENT PRINTING OFFICE: 1974-526-166

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FEB 03 1978

MEMORANDUM TO: U.S. Standard Problem Participants

FROM: Denwood F. Ross, Jr., Assistant Director for Reactor Safety, DSS

SUBJECT: ACCUMULATOR DELIVERY COMPARISONS

RELAP-4 comparisons of LOFT tests L1-3A and L1-4 (U.S. Standard Problem #7) have highlighted certain aspects of accumulator delivery which should be considered in the standard problem program. The RELAP-4 program through version 2 of MOD-6 used an isothermal gas expansion model for nitrogen in the accumulators. Post test analysis of L1-3A by INEL indicated that the actual gas expansion is somewhere between isothermal and isentropic (1.2). L1-4 RELAP analysis used an intermediate value for and after correcting loss coefficients was able to match pressure and delivery driving the early portion of accumulator delivery. After 35 seconds of injection the data shows flow spikes which are not predicted by RELAP. It has been suggested that this is related to nitrogen in the delivery lines and may cause exhaustion of the accumulators sooner than predicted.

We believe that accumulator delivery behavior can have an important effect on ECCS performance. The U.S. Standard Problem suggested list of comparisons includes accumulator delivery. In the past this information has not been provided by all participants. Please provide pressure and flow comparisons for all past and future standard problems where applicable. For L1-4 discuss the comparisons, including the following:

a.) Gas expansion model

b.) Heat transfer

c.) Loss coefficients

d.) Nitrogen ingestion

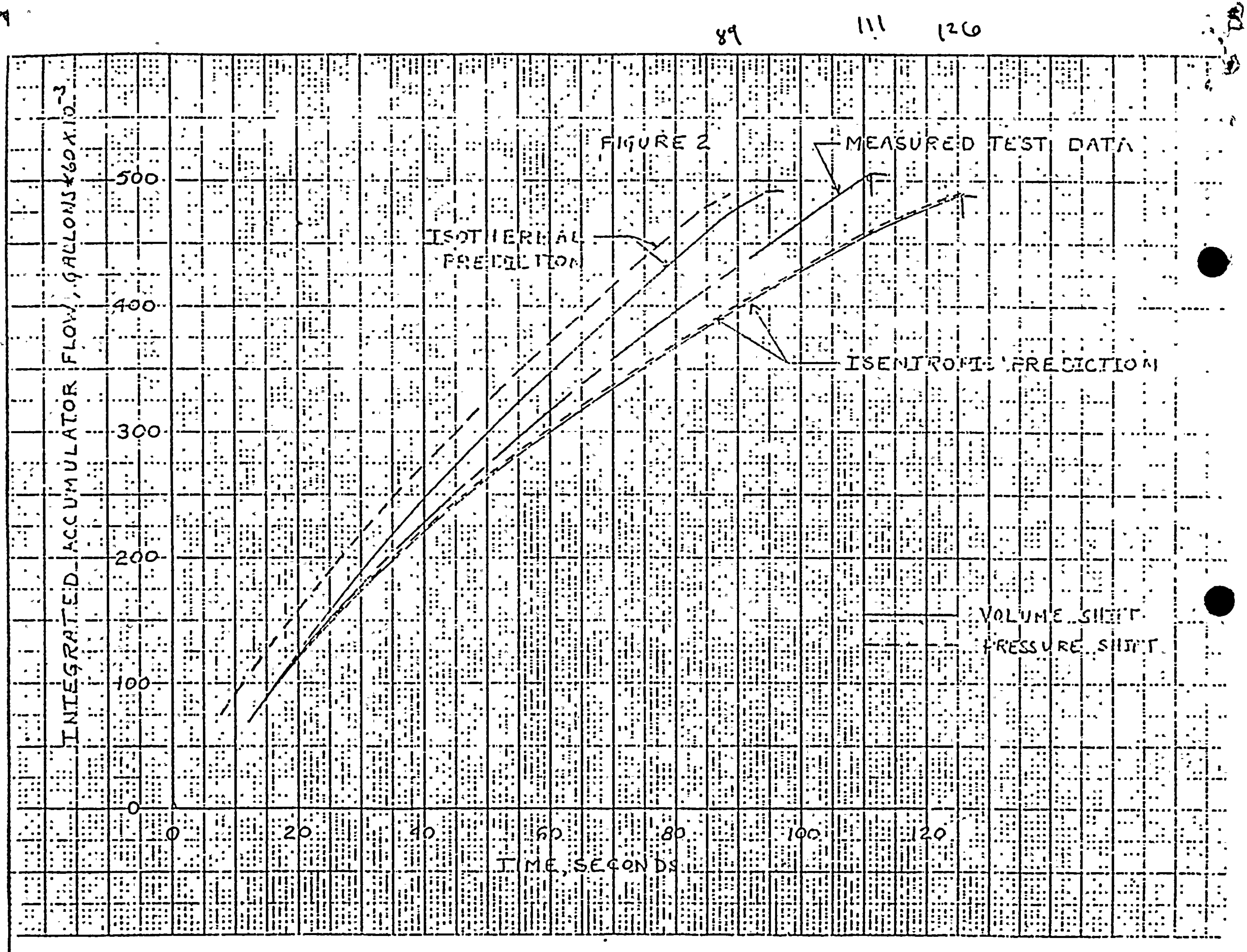
e.) LOFT typicality compared to large scale accumulator data

Participants having approved licensing models are requested to provide any additional comparisons to experiment or other information pertinent to assessing the validity of their accumulator delivery models.

Sincerely,

Original Signed By
D. F. Ross
Denwood F. Ross, Jr., Assistant Director
for Reactor Safety
Division of Systems Safety

bcc: D. Ross
Z. Rosztoczy
L. Phillips
M. McCoy
N. Lauben


89
111
126
FIGURE 2
MEASURED TEST DATA
ISOTHERMAL PREDICTION
ISENTROPIC PREDICTION
VOLUME SHIFT
PRESSURE SHIFT
INTEGRATED ACCUMULATOR FLOW, GALLONS×60×10⁻³
500
400
300
200
100
0
0
20
40
60
80
100
120
TIME, SECONDS

ACCUMULATOR DELIVERY

1. Concern: Actual accumulators may deliver ECCS water to the reactor coolant system faster than is predicted by some computer programs used to predict ECCS performance. This could mean that sufficient accumulator water would not be available at the time it is needed. Attention was focused on this problem when comparisons of accumulator delivery calculations were made between RELAP4 (NRC) and SATAN VI (Westinghouse) as part of the Upper Head Injection (UHI) review. Comparisons to the LOFT experimental data indicated that the Westinghouse model might be underpredicting accumulator delivery flow water. The key factors influencing delivery rates are the gas expansion model and the effective delivery line resistance.

2. Safety Significance: There is no specific reference to our current licensing position. Each reactor vendor proposed a different model in 1974 for compliance with Appendix K. These models are described in the appropriate topical reports. We did not consider this an issue at that time so implicitly accepted each model for accumulator delivery. We do not believe that this issue poses a significant safety problem and can ultimately be handled within the scope of present ECCS design capability. An example of the influence that the gas model can make on integrated accumulator delivery is shown on Figure 2 enclosed. Test data are from full-scale accumulator discharge.

3. Evaluation: We are asking our consultants (Sandia Laboratories) to continue their analytical evaluation of this issue. We have requested Westinghouse to provide comparisons of their model with prototypic accumulator delivery data for UHI plants. As part of the Standard Problem Program we have requested all participants to provide analytic comparisons to available data (see memo Ross to Standard Problem Participants, enclosed).

   It is conceivable that after our review of this issue is complete, changes in some vendor models for some plants may be required. The effect of these changes on calculated ECCS performance is not likely to be large for any plant except UHI plants. In any case, simple adjustments in accumulator water volume could most likely compensate for any model change. This issue should be completely resolved by August of this year.

4. Interim Accounting: It is recommended that no change is required until our evaluation is complete. Since we have notified reactor vendors by mail of the need to do additional calculations, we should consider informing sitting boards in the post-SER space. It is applicable to all such PWRs.

REGCEN.

February 10, 1978

Edward Luton, Esq., Chairman
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dr. Franklin C. Daiber
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Dr. Emmeth A. Luebke
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

In the Matter of
Rochester Gas & Electric Corporation
(R. E. Ginna Nuclear Power Plant, Unit No. 1)
Docket No. 50-244

Gentlemen:

Attached for your information is a copy of a memorandum dated February 3, 1978 from the Executive Director of Operations to the Commissioners which discusses, at page 6, the status of eight "terminal blocks" in use at the Ginna facility. Also attached for your information are the two earlier memorandums (dated January 13 and 27, 1978) to the Commissioners referenced in the February 3, 1978 memorandum.

Sincerely,

Auburn L. Mitchell
Counsel for NRC Staff

Attachments

cc: (w/attachments)
Leonard M. Trosten, Esq.
Mr. Michael Slade
Rochester Committee for
Scientific Information
Jeffrey Cohen, Esq.
Warren B. Rosenbaum, Esq.
Atomic Safety and Licensing Board
Atomic Safety and Licensing Appeal
Board
Docketing and Service Section

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SURNAME	AMitchell/dkw EReis					
DATE	2/10/78 2/10/78					

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

February 3, 1978

MEMORANDUM FOR: Chairman Hendrie
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Bradford

FROM: Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation

THRU: Executive Director for Operations TR for LVG

SUBJECT: UNION OF CONCERNED SCIENTISTS' PETITION

This memorandum supplements the staff reports of January 13, 1978 and January 27, 1978 with regard to:

1) the results of the environmental qualification tests performed for electrical connections used in the Boston Edison Co. Pilgrim Station, and

2) the use of unprotected electrical terminal blocks at Connecticut Yankee Power Company's Haddam Neck nuclear power plant, the results of a preliminary telephone survey of all operating plants concerning the use of unprotected terminal blocks, and a summary of subsequent actions.

Pilgrim 1

On January 13, 1978, the staff reported that: 1) Pilgrim Unit 1 shutdown on January 9, 1978 as a result of unsatisfactory results from preliminary environmental screening tests performed on a typical electrical connector assembly; 2) the licensee planned to conduct LOCA-type environmental qualification tests at the Wyle Laboratories on samples of existing connectors and potential modifications to the connectors; and 3) the staff would follow the licensee's program to ensure acceptable qualification of the electrical connections prior to the return to power operation.

The staff met with the licensee on February 1, 1978 to discuss the results of the program and plans for modifications prior to return to power operation. The licensee had informed the staff of its intention to replace electrical connectors with fully qualified splices similar to those used at D. C. Cook.

During this meeting, the staff and Boston Edison's (BECO) representatives reviewed the tests carried out at Wyle Laboratories to qualify the use of previously qualified Raychem splices with the electrical cable used at Pilgrim Unit 1. The splices successfully passed the test without discernable degradation of insulation resistance or degradation of capability to maintain electrical load. The staff has reviewed BECO's test procedure, test data and test results and inspected the actual test samples. We have concluded that the Raychem splices have been acceptably qualified for use with Pilgrim Unit 1 cables. The tests were witnessed at Wyle Laboratory by the Office of Inspection and Enforcement. These tests taken together with other environmental qualification testing of Raychem splices have qualified these splices to withstand radiation, temperature, pressure, and steam conditions for an accident environment at Pilgrim 1. The licensee has replaced all safety-related electrical connectors with Raychem splices and plans to return Pilgrim Unit 1 to operation on about February 5, 1978.

Unprotected Electrical Terminal Blocks

The staff report of January 27, 1978 advised that 1) the licensee for the Haddam Neck nuclear power plant had voluntarily shutdown the plant that day to replace the Marathon terminal blocks with Westinghouse terminal blocks, 2) the staff would review the complete qualification data on these terminal blocks before the plant was returned to power and 3) the staff would address the generic implications of the failure of unprotected Marathon terminal blocks by completing a telephone survey of all operating plants by January 31, 1978 and issuing an I&E bulletin on January 30, 1978 to require follow-up written responses to the telephone survey and documentation of environmental qualification if unprotected terminal blocks are used.

A. Haddam Neck

On January 27, 1978, the staff requested that the licensee for the Haddam Neck nuclear power plant meet with the staff to discuss its decision to replace terminal blocks. The meeting was held on Sunday, January 29, 1978. During this meeting, the licensee presented information based upon both testing and analysis which demonstrate that the new Westinghouse terminal block would perform as intended during accident conditions. The qualification of the Westinghouse block included the effects of temperature, pressure, humidity, radiation, chemical spray and seismic conditions. Prior to the meeting with the staff, the licensee had considered documentation supporting the plant design change (i.e., replacement of the Marathon terminal blocks with Westinghouse terminal blocks) that only addressed the LOCA environment. However, analysis and data were available to show that the Westinghouse terminal block would also function under the effects of both steam line breaks and spray from other piping breaks.

Following the January 29 meeting with the staff, the Connecticut Yankee Nuclear Review Board and the Plant Operations Review Committee met to specifically address the environmental qualification under both steam line break and spray environments for the Westinghouse terminal board. The review was completed by 2:15 am Monday, January 30, 1978 and the plant was returned to power operation. On February 3, 1978, the staff received documentation supporting the evaluation for Haddam Neck plant which resulted in a decision to replace the Marathon terminal blocks with Westinghouse terminal blocks. (Some proprietary Westinghouse information in the possession of the licensee will be delivered to the staff during the week of February 6, 1978, following approval of Westinghouse to release the data.) The staff has conducted a preliminary review of the supporting documentation which includes the technical review of the Haddam Neck plant change request No. 270 and the documentation regarding the qualification of the replacement Westinghouse terminal blocks. Based upon this review, the January 29, 1978 meeting with the licensee, and on the Franklin Institute tests that were witnessed by the NRC staff we believe that the licensee's decision to resume operation of the Haddam Neck plant was appropriate.

The licensee also used four General Electric terminal blocks, contained in vented boxes, to replace the four original electrical connectors. These were also considered in the meeting with the staff on January 29. As a follow-up to its letter of January 13, 1978, the licensee committed to provide data on the qualification of the General Electric terminal blocks as soon as possible, but not later than 60 days from January 30, 1978, or to replace the General Electric terminal blocks with a continuous run of qualified cable (thus eliminating the four connectors in question) within the same 60 days. As a result of the satisfactory screening test at Franklin Institute on January 26 (which was observed by NRC personnel as indicated in our January 27 report), the licensee concluded, and we have agreed, that 60 days was a reasonable period of time to complete documemtation of the environmental qualification of either the GE terminal blocks or the new cable and to present the data to the NRC.

B. Generic Implications of Unprotected Terminal Blocks

The staff has completed a preliminary telephone survey of all operating plants on the use and qualification of unprotected terminal blocks inside containment in safety-related systems. In response to our telephone survey, three additional licensees indicated the use of unprotected terminal blocks in safety-related systems inside containment and were required to attend meetings with the staff on February 1, 1978. These licensees are the Yankee Atomic Electric Company (Yankee Rowe facility),Sacramento Municipal Utility District (Rancho Seco facility),and Rochester Gas & Electric Corporation (Ginna facility). The following discussion summarizes the results of these meetings.

Rancho Seco

The licensee had stated on January 30, 1978, in response to our telephone survey, that it had unprotected terminal blocks in safety-related circuits inside containment. In the meeting with the staff on February 1, 1978, the licensee corrected this information and stated that a more accurate characterization of terminal blocks inside

containment is as follows:

a. There are no unprotected terminal blocks in use in safety systems at the Rancho Seco facility.

b. There are terminal blocks in use which generally fall in two categories:

(1) Terminal blocks that are used with various pieces of safety-related electrical equipment, for example, motor operated valves. These terminal blocks are enclosed and have been qualified as part of the electrical equipment qualification test program. The licensee considers these terminal blocks to be qualified for their service environments.

(2) Terminal blocks that are in use with containment penetrations. The conductors from the containment penetrations are terminated at terminal blocks from which connecting conductors go to various locations in the containment. These terminal blocks, however, are not exposed. They are enclosed in metal containers physically attached to the containment penetrations.

Although these enclosed terminal blocks on the penetrations were outside the scope of the telephone survey, the licensee could not, in the course of our discussions on February 1, readily demonstrate that the metal enclosures and the terminal blocks were fully qualified. Therefore, the licensee was requested to assemble available information on the enclosed connections at the penetrations, to evaluate the degree to which they assure safety, and, if necessary, to propose solutions for correcting any deficiencies in the area of environmental qualifications. A meeting has been scheduled for February 8, 1978 to discuss this information with the licensee.

Ginna

The licensee stated during the meeting on February 1, 1978 that a total of eight terminal blocks are used in instrumentation circuits that are required to operate in the event an accident. These terminal blocks were mounted in vented cabinets, but the doors had been removed for ease of maintenance. Corrective measures have been developed and evaluated by both the on-site and the off-site plant safety review committees. The corrective measures evaluated were to replace the existing terminal blocks with the same kind of Westinghouse terminal blocks used and qualified by Connecticut Yankee and to reinstall the cabinet doors during the current plant outage.

The licensee described the bases for the safety review committees' evaluation and conclusions regarding the environmental qualification information. This information included test data and analysis for the Westinghouse terminal block which demonstrated that the terminal block would perform as intended during accident conditions, including the main steam line break accident. The qualification of this terminal block included the effects of temperature, pressure, steam, radiation and chemical spray. The licensee also considered submergence of the terminal blocks. An analysis showed that the water level in containment would not reach these blocks until after the instrumentation had completed its safety function. Other instrumentation is available for long-term post accident monitoring that is installed independent of these terminal blocks. After completion of these modifications, start-up of the plant and return to power operation occurred on the evening of February 1, 1978.

Yankee Rowe

The licensee provided information on the corrective measures taken to resolve the unprotected terminal block problem during the current plant outage.

The licensee established that there are a total of 76 safety-related terminal blocks inside the Yankee-Rowe containment. About 6 terminal blocks are installed in the location of the

reactor coolant loops between the primary shield and the biological shield. These terminal blocks are enclosed in boxes and thus are not directly exposed to the containment atmosphere. The balance of the terminal blocks are outside the biological shield. About 17 are at the biological shield wall in enclosures and the remainder are enclosed in boxes at the containment penetration areas. The covers on the boxes at the penetration areas were, prior to the current outage, not installed. The licensee has stated that these covers will be in place prior to return to power.

New Westinghouse terminal blocks that passed CY's screening test were obtained, and the licensee replaced all safety-related terminal blocks inside containment prior to returning the plant to power. The terminal block replacement was reviewed and approved by the plant on-site and off-site safety review committees.

The licensee presented information at the February 1 meeting with the staff on its plant specific evaluation to demonstrate that the new Westinghouse terminal blocks would perform their intended safety functions during accident conditions. For environmental qualification of these new terminal blocks, the licensee based its evaluation, in part, on the records developed by CY. The licensee demonstrated that the screening tests performed by CY envelope the calculated Yankee-Rowe LOCA environment. Further, the licensee determined that because of the high elevation where the terminal blocks are installed, there is no flooding potential. The licensee has considered the potential for radiation damage to the terminal block in the exposed location, and found that there is an acceptable margin for the new terminal blocks as established by Westinghouse (based on resistance characteristics).

The licensee also described considerations of the potential adverse affect of a steam line break on the new terminal blocks. The terminal block enclosures at the penetration locations are in the general vicinity of two of the four main steam lines. The licensee stated that replacing the 1/4 inch thick steel cover plates on these enclosures eliminates direct steam jet impingement on the terminal blocks and that the enclosures have bottom holes to allow for drainage of condensation.

With regard to the concern of superheated steam potentially causing a more severe temperature transient than previously concerned, the licensee indicated that preliminary calculations have shown that the temperatures of the enclosures would exceed 275°F, but only for a short duration. The licensee indicated that this temperature exposure would be less severe than the demonstrated capability of the new blocks to withstand a temperature of 340°F for 5 hours without loss of function. Documentation of the evaluation of all relevant safety considerations and the conclusions of the appropriate review committees were to be completed by the licensee prior to resumption of Yankee-Rowe operation on February 2, 1978.

SUMMARY REGARDING UNPROTECTED TERMINAL BLOCKS

The staff indicated in its January 27, 1978 report that an I&E bulletin would be issued on January 30, 1978 requiring follow-up documentation of whether such unprotected terminal blocks are in use, and if so, their environmental qualifications. A copy of IE Bulletin No. 78-02, Terminal Block Qualification, is enclosed. The responses to this bulletin will be reviewed and evaluated by the staff and appropriate action will be taken as necessary. For those plants with unprotected terminal blocks, documentation of environmental qualification is required. The staff will report to the Commission the results of this evaluation and any further actions that may be required.

Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation

Enclosure:
As stated

cc: Secretary
Public Document Room
Union of Concerned Scientists

January 30, 1978

MEMORANDUM FOR: B. H. Grier, Director, Region I
J. P. O'Reilly, Director, Region II
J. G. Keppler, Director, Region III
E. M. Howard, Director, Region IV
R. H. Engelken, Director, Region V

FROM: Harold D. Thornburg, Director, Division of Reactor Operations Inspection, IE

SUBJECT: IE BULLETIN NO. 78-02 - TERMINAL BLOCK QUALIFICATION

The subject Bulletin should be dispatched for action on January 30, 1978, to all power reactor facilities with an operating license or a construction permit.

The text of the Bulletin and draft letter to licensees are enclosed for this purpose.

Harold D. Thornburg, Director
Division of Reactor Operations
Inspection
Office of Inspection and
Enforcement

Enclosures:
1. Draft Transmittal Letter
2. IE Bulletin 78-02

CONTACT: V. D. Thomas, IE
49-28180

OFFICE	TA:RCI:IE	RCI	XOOS	ROI		
SURNAME	VDThomas sh	KVSeyfrit	ELJordan	HDThornburg		
DATE	1/ /78	1/ /78	1/ /78	1/ /78		

(Draft letter to all power reactor facilities with operating license or construction permit)

IE Bulletin No. 78-02

Gentlemen:

Enclosed is IE Bulletin No. 78-02 which requires action by you with regard to your power reactor facility(ies) with an operating license or a construction permit.

Should you have questions regarding this Bulletin or the actions required of you, please contact this office.

Sincerely,

Signature
(Regional Director)

Enclosures:
1. IE Bulletin No. 78-02
2. List of IE Bulletins
   Issued in 1978

UNITED STATES
NUCLEAR REGULATION COMMISSION
OFFICE OF INSPECTION AND ENFORCEMENT
WASHINGTON, D.C. 20555

January 30, 1978

IE Bulletin No. 78-02

TERMINAL BLOCK QUALIFICATION

Description of Circumstances:

On January 18, 1978, Connecticut Yankee Atomic Company performed a screening test intended to verify previous analyses of the environmental qualifications of unprotected terminal blocks used inside containment. The test was performed at the Franklin Institute Research Laboratories, Philadelphia, Pennsylvania.

The test specimen was a Marathon M-6012 terminal block. It was exposed to a steam environment which was designed to envelope the calculated LOCA environmental conditions in the Haddam Neck containment. The pressure selected for the test was 40 psig for a period of 24 hours.

The temperature profile consisted of a rise from an initial temperature of 100 degrees Fahrenheit to 275 degrees Fahrenheit within ten seconds, followed by a steady state operation at 275 degrees Fahrenheit for four hours. This was followed by a drop of temperature to 140 degrees Fahrenheit within one hour, followed by a repetition of the initial temperature rise to 275 degrees Fahrenheit (within ten seconds). The temperature then remained constant at 275 degrees Fahrenheit for the remaining 19 hours of the test period.

During the initial screening test, 525 volts, single phase, 60 Hertz, ac voltage was applied to two pairs of terminals on the test specimen. Inability of the terminals to hold the voltage was defined before the test as an appropriate failure criterion. The test was initiated on January 19, 1978. The terminal block functioned as intended during the first 5 hours of the test at which time one of the pairs of terminals failed the test.

The cause of failure is still under investigation. The failure occurred during an operator error resulting in a pressure and temperature excursion which is outside the envelope of the intended test. Because of this, the licensee reran the test.

The second screening test was initiated on January 25, 1978. This test included three test specimens: (1) an unprotected Marathon terminal block identical to the one used in the first test; (2) an unprotected Westinghouse terminal block; and (3) a GE terminal block enclosed in a NEMA type 12 box identical to the ones in use in the Haddam Neck plant. The test specimens were exposed to an environment having temperature and pressure profiles essentially the same as those of the first test, minus the inadvertant overpressure transient. All the test specimens successfully operated through the two temperature rise profiles in the test sequence. However, after 21 hours in the test environment, the lower pair of terminals of the unprotected Marathon terminal block failed. The failed terminal points were disconnected and the test was completed. No further failures occurred. The failure mechanism of the terminal blocks during the first and second tests appears to be similar, i.e., the terminal pair that failed in each of the tests was the lower pair on the terminal block. Detailed analysis are in progress to identify the exact cause of failure.

Actions to be Taken by Licensees and Permit Holders:

For all power reactor facilities with an operating license or a construction permit:

(1) Determine whether or not unprotected terminal blocks as used in your facility in systems which must function in the post-accident environment;

(2) If such terminal blocks are used, identify the systems involved and provide the documentation which demonstrates that these terminal blocks have been environmentally qualified; and

(3) If documentation is not available, provide your plans and schedule for achieving full qualification of affected circuits.

For holders of operating licenses, your response to this bulletin is required to be in this office within 14 calendar days of the date of issue of this bulletin.

For holders of construction permits, your response to this bulletin is required to be mailed within 30 calendar days of the date of issue of this bulletin.

Copies of your response are to be provided in the same time frame directly to the Director, Office of Nuclear Reactor Regulation, and to the Director of Reactor Operations Inspection, U.S. Nuclear Regulatory Commission, Washington, D.C. 20555.

Approved by GAO, B180225 (R0072); clearance expires 7-31-80. Approval was given under a blanket clearance specifically for identified generic problems.

LISTING OF IE BULLETINS
ISSUED IN 1978

Bulletin No.	Subject	Date Issued	Issued To
78-01	Flammable Contact - Arm Retainers in G.E. CR120A Relays	1/16/78	All Power Reactors Facilities with an Operating License (OL) or Construction Permit (CP)

JAN 13 1978

MEMORANDUM FOR: Chairman Hendrie
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Bradford

FROM: Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation

THRU: Executive Director for Operations

SUBJECT: UNION OF CONCERNED SCIENTISTS' PETITION

In its report, dated December 15, 1977 the staff indicated that it was continuing review of the responses from licensees of operating reactors to IE Bulletins 77-05/05A and 77-06 concerning the use of electrical connectors and electrical penetrations, respectively. The Office of Inspection and Enforcement has completed a review of these responses and copies of its reports are enclosed.

The reports deal mainly with the adequacy of the documentation to confirm the environmental qualification of the subject equipment. With regard to the use of electrical connectors, the responses either provided or referenced information which indicates adequate qualification test results, which support continued operation for the operating reactors; however, the documentation for Pilgrim and Connecticut Yankee was not satisfactory.

The Connecticut Yankee licensee chose to replace the connectors with qualified terminal blocks inside sealed junction boxes before the deadline for responding to IE Bulletin 77-05. No further documentation is required of Connecticut Yankee.

My memorandum of January 6, 1978 provided information regarding the use of electrical connectors in the Pilgrim Unit 1 plant and described actions that had been taken. As stated in that

memorandum, the plant was to be shutdown by January 21, 1978 for a maintenance outage and to perform additional environmental qualification testing of electrical connectors prior to resumption of power operation. A chronology of the development of the Pilgrim Unit 1 information is attached.

On January 9, 1978, the staff was informed that the licensee began an orderly shutdown of the Pilgrim Unit 1 facility at 9:00 p.m. as a result of the unsatisfactory outcome of preliminary environmental screening tests and diagnostic examination performed on a typical connector assembly obtained from a non-safety system located outside containment. Diagnostic examination of the connector assembly indicated inadequate adhesion of the potting compound which was used in the field installation of the connector. The licensee plans to conduct LOCA-type environmental tests at the Wyle Laboratories this weekend on samples of existing connectors and potential modifications to the connectors. The staff will follow closely the program being undertaken by the Boston Edison Company to ensure acceptable qualification of the connectors prior to the return to power operation.

As indicated in Table 1, Browns Ferry Units 1, 2, and 3 and Nine Mile Point are performing additional testing which will be completed by the end of February. The results of the Nine Mile Point tests will be applicable for Maine Yankee. Oyster Creek is also performing additional testing that should be completed by the end of February. The nature of the additional testing for the connectors in use in these facilities is described in the staff reports of November 18, 1977 and December 6, 1977.

The IE report on electrical connectors states that licensees have not defined specific environmental conditions with regard to accidents other than the LOCA; e.g., a main steam line break. As indicated in the December 15, 1977 staff report (particularly Appendix B of Enclosure B), significant aspects of environmental qualification of electrical equipment are being treated in the Systematic Evaluation Program (SEP) which has now been initiated for the eleven SEP facilities. In addition to this action, the staff is also pursuing the generic subject of equipment qualification as described in Task Action Plan A-24; "Qualification of Class IE Safety-Related Equipment."

The IE report on electrical penetrations concludes that the penetrations used in operating reactors are environmentally qualified for the LOCA condition, based on the IE review of licensee qualification test reports and comparative design analysis. The staff is continuing its review of the need to maintain pressure on those penetrations which would accommodate gas pressures.

Original Signed By
E. G. Case

Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation

Enclosures:
1. Summary of Responses to IE Bulletins 77-05 and 77-05A, dated January 5, 1978.
2. Review of Qualification Test Reports on Electrical Penetrations in Use at Light Water Reactor Power Plants, dated January 6, 1978.

3. Chronology on Pilgrim Unit 1

cc: Secretary
Public Document Room
Union of Concerned Scientists

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D. Davis
P. O'Connor
K. Seyfrit
R. Mattson
J. Scinto
J. Lieberman
E. Case
L. Gossick

*See previous yellow for concurrence

OFFICE	DSS:ADPS	DDOR:DIR	DSS:DIR	DIR:NRR	EDO	
SURNAME	*RTedesco:sl	*VStello	*RMattson	ECase	~~LVGossick~~	
DATE	1/11/78	1/ /78	1/ /78	1/ /78	1/ /78	

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

Enclosure 1

JAN 5 1978

MEMORANDUM FOR: Victor Stello, Director
Division of Operating Reactors, NRR

R. J. Mattson, Director
Division of Systems Safety, NRR

FROM: K. V. Seyfrit, Assistant Director
for Technical Programs, IE

SUBJECT: SUMMARY OF RESPONSES TO IE BULLETINS 77-05 AND 77-05A

The enclosed summary of responses to the subject bulletins is forwarded for your information.

Karl V. Seyfrit

Karl V. Seyfrit, Assistant Director
for Technical Programs
Division of Reactor Operations Inspection

Enclosure: As stated

cc: J. Scinto, ELD
R. L. Tedesco, NRR
D. G. Eisenhut, NRR

SUMMARY OF RESPONSES TO IE BULLETINS 77-05 AND 77-05A

Table B-1 (pg 56) of Mr. Case's memo to the Commissioners dated December 15, 1977, "Union of Concerned Scientists Petition" listed 19 reactors which had reported the use of connectors in safety systems which were required to function in the LOCA environment inside containment. These 19 plants were identified by telephone survey or by initial responses to Bulletin 77-05 issued by the Office of Inspection and Enforcement. The initial responses, as noted in the referenced table, did not include documentation of qualification test results. The complete submittals have now been received. With the exception of the Pilgrim and Connecticut Yankee facilities, all responses have provided or referenced qualification test results which indicate satisfactory performance of the connectors associated with systems required to function in the LOCA environment. In some cases, additional testing is being conducted to demonstrate adequate performance under the additional conditions of aging and integrated radiation exposure.

In the case of Connecticut Yankee, four connectors of the kind described were identified. The licensee chose to replace these with qualified terminal blocks inside sealed junction boxes. In the case of Pilgrim, letters of certification were presented which attested to the acceptability of the connectors, but no evidence of actual testing was provided. As a consequence, the staff held a meeting with the licensee on December 29, 1977. The licensee presented orally a plan of action which was acceptable to the staff. This plan calls for replacement of the connectors associated with the Automatic Depressurization System (ADS) actuation by qualified splices during an outage scheduled for mid-January. Additionally, qualification testing of other connectors will be initiated and will be described to NRR prior to any resumption of power operation following the outage. Summaries of other submittals for the 19 plants are included in Table 1.

The remaining operating plants state in their responses to Bulletin 77-05 that no systems which must function in a LOCA environment contain connectors inside containment.

With respect to other analyzed accidents, specific environmental conditions have not been defined in most cases. For those plants which do not use connectors in safety systems needed for post accident service inside containment, only accidents external to containment need be considered. Licensees indicate that the location of connectors outside containment is such that the connectors are protected from the effects of high energy line breaks or that connectors are not used in systems required to function after such accidents or that whatever safety functions may be required will have been completed before the adverse environment can affect the system function. For the 18 plants (Hatch excluded) which do have connectors in

systems that must function in the post accident environment, and are located inside containment, the most severe accident appears to be a steam line break inside containment. A preliminary evaluation of this event indicates that peak environment temperatures may be considerably higher than those calculated for a LOCA, but the duration of high temperatures will be significantly shorter. It appears likely that equipment qualified for post LOCA conditions would also be qualified for the steam line break inside containment, since with the shorter time at high temperature, the equipment is not likely to absorb heat rapidly enough to exceed the calculated peak temperatures associated with LOCA. This aspect of environmental qualification may require additional review to provide the desired level of confidence in the ability of the connectors to function in accident-environments other than the LOCA.

STATUS OF QUALIFICATION OF CONNECTORS USED INSIDE CONTAINMENTS FOR LOCA CONDITIONS

Facility	Status
D.C. Cook Unit 1	Connectors have been replaced with qualified splices. Splice qualification tests witnessed by NRC representatives.
Browns Ferry Units 1, 2, 3	Partial qualification testing completed and documented. This included short term exposure to accident environment, but without preaging or radiation exposure. Additional testing to be completed by the end of February.
Nine Mile Point	Partial qualification testing completed and documented. Circuits were not energized during testing and preaging and radiation exposure were not included. Additional testing to be completed by the end of February.
Main Yankee	Utilize same connectors as Nine Mile Point. The testing noted for Nine Mile Point will satisfy qualification requirements.
Oyster Creek	Connectors are essentially the same as those tested by Wylie Laboratories for Target Rock Co. Independent review by MPR Associates confirm similarity and concludes that qualification is valid.
Surry Units 1, 2	Documentation received and adequate. The only safety related connectors are those in circuits required for reactor trip. This function would be completed prior to exposure to caustic sprays or high radiation. Qualification test results for other environmental factors are acceptable.
Oconee Units 1, 2, 3	Documentation received and adequate. Testing included preaging at 300°F for 350 hours and irradiation to $1X10^8$ rads absorbed dose.
Hatch	Hatch was listed originally as having connectors inside containment in systems that must function in a LOCA environment, based on telephone contact. The formal response to IE Bulletin 77-05 reveals that there are no connectors in containment which are required to perform in the LOCA environment.
Fort St. Vrain	Documentation received. Connectors are fully qualified for postulated accident environment.

Facility	Status
Pilgrim	Documentation consisted of letters of certification, with no actual test results. Connectors are partially protected from environment. Licensee has proposed a program of qualification testing or replacement of connectors.
Peach Bottom Units 2 3	Documentation received. Connectors are fully qualified for LOCA environment.
Palisades	Response indicates connectors are identical to those used at Oconee, and therefore have been adequately qualified.
Connecticut Yankee	Licensee was not able to provide documentation for the only connectors (4) in circuits required to function in the LOCA environment. The connectors were removed and replaced with qualified terminal blocks enclosed in sealed junction boxes. Licensee plans to install qualified connectors in these circuits during the next refueling outage.

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

Enclosure 2

JAN 6 1978

MEMORANDUM FOR: R. J. Mattson, Director
Division of Systems Safety, NRR

FROM: K. V. Seyfrit, Assistant Director
for Technical Programs, IE

SUBJECT: REVIEW OF QUALIFICATION TEST REPORTS ON ELECTRICAL PENETRATIONS IN USE AT LIGHT WATER REACTOR POWER PLANTS

Qualification test reports on all types of electrical penetration assemblies which are presently in use at operating light water reactors have been received from the manufacturers of these penetrations or from licensees. For Dresden Unit No. 1, Connecticut Yankee, and Yankee Rowe the responses received from NRR's ten (10) day letter of December 2, 1977, together with prototype test reports for the Dresden Unit No. 1 and Yankee Rowe Reactors provided documentation of penetration acceptability. Connecticut Yankee penetration qualification was determined by comparative design analysis and reference to the prototype testing of similar penetrations used at Yankee Rowe Nuclear Station.

LaCrosse, reports that the epoxy compound used for potting is rated at 257°F and begins to soften at 320°F. The maximum temperature achieved during a LOCA is calculated to be 280°F and may remain greater than 250°F for up to one hour. The radiation tolerance of their penetrations is rated at $4X10^9$ rads and maximum radiation exposure during LOCA conditions is calculated to be 10^6 rads. The mineral insulated cable used in the penetrations is rated for continuous operation while immersed in water at a temperature of 150°F. In addition, the cable terminations are capable of withstanding a hydro static pressure of 850 PSIG and is chemically inert. The licensee concludes and we agree, that the electrical penetrations installed at the LaCrosse Nuclear Station will maintain their integrity during a LOCA.

For the remainder of the operating light water reactors, our review of the qualification test reports identified no items which would indicate any lack of functionability during LOCA conditions. Our

CONTACT: V. D. Thomas, TP
49-28180

review included the qualification test reports on penetrations designed and manufactured by Amphenol Sams, Conax, Crouse Hinds, D . G. O'Brien, General Electric, Viking, and Westinghouse. In all cases, the information provided was consistent with information obtained from the earlier telephone survey and the follow-up written responses related to IEB-77-06.

Our review of the test reports included temperatures, pressures, humidity, leakage rates and seismic conditions which are the parameters of concern. Radiation effects were determined in some cases by evaluation of materials used, while others had specific radiation exposure testing for the completed unit.

While the test data verifies that the penetrations have been qualified with and without nitrogen pressure, we believe it would be prudent to maintain nitrogen pressure on those penetrations which would accommodate gas pressures, to provide added protection for the circuits involved. We understand NRR is pursuing this matter with licensees.

On the basis of the information provided, we conclude that all electrical penetration types presently installed at operating light water reactor power plants are environmentally qualified to function as intended during LOCA conditions.

If you should have any questions concerning the above matters, please contact V. D. Thomas on Ext. 28180.

Karl V. Seyfrit, Assistant Director
for Technical Programs
Division of Reactor Operations Inspection

cc: J. G. Davis
H. D. Thornburg
R. L. Tedesco, NRR
D. G. Eisenhut, NRR
V. Stello, NRR

ENCLOSURE 3

CHRONOLOGY

PILGRIM UNIT 1
CONNECTOR QUALIFICATION

November 4, 1977	U.C.S. Petition Filed
November 8, 1977	IE Bulletin 77-05 - Electrical Connector Assemblies Issued
November 15, 1977	IE Bulletin 77-05A - Electrical Connector Assemblies Issued
November 18, 1977	Results of Staff survey of NSSS vendors and A&E firms provided in staff report. No indication received from Bechtel that electrical connectors were used at Pilgrim Unit 1.
November 25, 1977	Further results of survey given in staff report regarding use of connectors in Target Rock relief valves (non-safety function). Pilgrim Unit 1 was one plant identified as having Target Rock valves, but not other connectors.
December 7, 1977	Letter received from GE to D. Eisenhut indicating that other connectors besides those on Target Rock valves were used in Pilgrim Unit 1. Staff decided to wait for licensee's response to 77-05 which was due on December 8. IE notified of GE letter.
December 13, 1977	IE Headquarters received cover letter without nine attachments from Boston Edison Co. (BECO)
December 15, 1977	Staff report to Commission: Table B-1 contains Pilgrim Unit 1 as using connectors in safety systems and awaiting formal documentation of qualification by test.
December 19, 1977	Letter and nine attachments from BECO received in the IE Region 1 Office. Material forwarded to IE headquarters without review by field office staff.

December 27, 1977	IE Headquarters receives full BECO submittal from Region 1. Determines that information did not consist of qualification data but only letters of certification; NRR was notified of the contents of the submittal, i.e., no test results provided by BECO.
December 29, 1977	NRR meeting with BECO. Licensee provided bases for continued safe operation pending complete documentation of qualification. Licensee requested to provide complete documentation to support conclusion that the electrical connectors are environmentally qualified prior to resumption of plant operation following the scheduled maintenance outage.
December 30, 1977	BECO submits confirming letter about plans to qualify electrical connector assemblies.
January 5, 1978	Report from IE to NRR summarizing licensees' responses to IE Bulletins 77-05/05A and indicating that no actual test results were provided for Pilgrim 1.
January 6, 1978	Staff report to Commission discussing actions taken in Pilgrim Unit 1
January 9, 1978	IE informed by BECO of plant shutdown as a result of the unsatisfactory outcome of preliminary environmental screening tests and diagnostic examination performed on a typical connector assembly.
January 10, 1978	Telecon: BECO advises staff of initiation of tests to qualify electrical connector assemblies.

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

JAN 27 1978

MEMORANDUM FOR: Chairman Hendrie
Commissioner Gilinsky
Commissioner Kennedy
Commissioner Bradford

FROM: Edson G. Case, Acting Director
Office of Nuclear Reactor Regulation

THRU: Executive Director for Operations

SUBJECT: UNION OF CONCERNED SCIENTISTS' PETITION

This memorandum provides additional information on electrical connections located inside containment in safety-related systems at Connecticut Yankee Atomic Power Company's Haddam Neck nuclear power plant. In the staff's December 15, 1977 report on the UCS petition we advised that the licensee had replaced four safety-related electrical connectors with environmentally qualified terminal blocks located inside environmentally qualified junction boxes. In our January 13, 1978 report on the results of IE Bulletins 77-05 and 77-05A, we reported that no further documentation of the environmental qualifications of these connections was required. We were relying on the licensee's statement that appropriate qualifications existed.

The staff received telecopied information from the licensee late on January 13, that the environmental qualification data being relied on did not in fact exist for the terminal block and box assembly. We learned on January 16, 1978 that a large number of similar, but unprotected terminal blocks (i.e., blocks without junction boxes) have been in use in safety-related systems inside containment of that facility since it began operation over ten years ago. Since the plant is one of the eleven plants under review in the Systematic Evaluation Program, the broad question of environmental qualification of other electrical equipment, not just connectors, was already under expedited review, as described in our December 15 report.

Although engineering analysis of the design and materials of construction of the terminal blocks convinced the licensee that the unprotected terminal blocks were environmentally qualified, the licensee agreed during a meeting with the staff on January 16 to

perform a screening test of the capability of the terminal blocks without junction boxes to function in an accident environment. The test was called a screening test to differentiate it from a full environmental qualification test. The screening test involved an aged terminal block (10 years old) with steam, high temperature, high pressure, and high voltage conditions. It was to be run on an expedited basis prior to later environmental qualification tests that would include aging and radiation conditions. The premise was that if the unprotected terminal blocks (hereafter called Marathon terminal blocks) could be shown by a combination of analysis and screening tests to be qualified for the accident environment, then time could be allowed for a more deliberately paced, complete environmental qualification testing program of the blocks and their associated terminal boxes.

As indicated in our January 20, 1978 memorandum to the Commission (footnote 8 on page 3), the initial screening test was conducted on January 19, 1978. Details of the test conditions are provided in Enclosure 1. In that test there was a failure in the test rig and its controls (inadvertent application of excessive temperature and pressure during the rise to the second temperature peak of the test sequence). One of the terminal pairs on the Marathon terminal block failed during the inadvertent transient. The other terminal pair survived the test.

The licensee immediately proceeded with steps to repeat the test on additional terminal blocks.

The second screening test was completed on January 26. It included: 1) the General Electric terminal block and box assembly with which the licensee had replaced the four electrical connectors; 2) an aged Marathon terminal block which had been in service for ten years at the plant; and 3) a new unprotected Westinghouse terminal block. The Marathon terminal block functioned normally for 21 hours into the 24-hour screening test, and then one of the two terminal pairs failed. The other pair survived the complete test. The location of the failed terminal pair on the Marathon block was the same as the location of the pair which failed during the inadvertent transient portion of the first screening test. The General Electric terminal block and box assembly and the Westinghouse terminal block passed the screening test.

The licensee voluntarily shut down the plant today to replace the Marathon terminal blocks with the Westinghouse terminal blocks. The staff will review the complete qualification data on these blocks before the plant is returned to operation.

The generic implications of the failure of the Marathon terminal blocks are being pursued by the staff through two parallel channels. First, an NRR telephone survey of all operating plants is being conducted this afternoon to see if there are other plants with unprotected terminal blocks without complete environmental qualifications in use in safety-related systems inside containment. The results of the survey are to be assembled by January 31. Any licensees found to have unprotected terminal blocks in safety-related systems inside containment are to be in the Bethesda offices of NRR on February 1, 1978 with available documentation of environmental qualifications. If unqualified blocks are found, the licensees will be required to show an acceptable basis for continued safe operation. On January 30, IE will issue a Bulletin requiring follow-up documentation of whether such terminal blocks are in use and, if so, their environmental qualifications. Additionally, the subject of environmental qualifications of all electrical connections in safety systems inside containment for the eleven plants in the Systematic Evaluation Program is being studied on an expedited basis relative to the study of other electrical equipment which was already on an accelerated schedule.

Enclosure 1 provides details of the January 18 and 26 screening tests of the terminal blocks. Enclosure 2 is the information phoned to licensees today. We will be reporting on this matter to the Commission by February 3, 1978.

Edson G. Case, Acting Director
Office of Nuclear Reactor
Regulation

Enclosures:
1. Screening Tests of Connecticut Yankee Terminal Blocks
2. Information telephoned to licensees

cc: Union of Concerned Scientists
NRC Public Document Room
SECY

SCREENING TESTS OF CONNECTICUT YANKEE TERMINAL BLOCKS

On January 18, 1978, Connecticut Yankee Atomic Company performed a screening test intended to verify previous analyses of the environmental qualifications of unprotected terminal blocks used inside containment. The test was performed at the Franklin Institute Research Laboratories, Philadelphia, Pennsylvania.

The test specimen was a Marathon M-6012 terminal block. It was exposed to a steam environment which was designed to envelope the calculated LOCA environmental conditions in the Haddam Neck containment. The pressure selected for the test was 40 psig for a period of 24 hours.

The temperature profile consisted of a rise from an initial temperature of 100°F to 275°F within ten seconds, followed by a steady state operation at 275°F for four hours. This was followed by a drop of temperature to 140°F within one hour, followed by a repetition of the initial temperature rise to 275°F (within ten seconds). The temperature then remained constant at 275°F for the remaining 19 hours of the test period.

During the initial screening test, 525 volts, single phase, 60 Hertz, ac voltage was applied to two pairs of terminals on the test specimen. Inability of the terminals to hold the voltage was defined before the test as an appropriate failure criterion. The test was initiated on January 19, 1978. The terminal block functioned as intended during the first 5 hours of the test at which time one of the pairs of terminals failed the test.

The cause of failure is still under investigation. The failure occurred during an operator error resulting in a pressure and temperature excursion which is outside the envelope of the intended test. Because of this, the licensee reran the test.

The second screening test was initiated on January 25, 1978. This test included three test specimens: (1) an unprotected Marathon terminal block identical to the one used in the first test, (2) an unprotected Westinghouse terminal block; and (3) a GE terminal block enclosed in a NEMA type 12 box identical to the ones in use in the Haddam Neck plant. The test specimens were exposed to an environment having temperature and pressure profiles essentially the same as those of the first test, minus the inadvertant overpressure transient. All the test specimens successfully operated through the two temperature rise profiles in the test sequence. However, after 21 hours in the test environment, the lower pair of terminals of the unprotected Marathon terminal block failed. The failed terminal points were disconnected and the test was completed. No further failures occurred. The failure mechanism of the terminal blocks during the first and second tests appears to be similar; i.e., the terminal pair that failed in each of the tests was the lower pair on the terminal block. Detailed analysis are in progress to identify the exact cause of failure.

ENCLOSURE 2

Information Telephoned to Licensees

Recent laboratory tests conducted at the Franklin Institute for the Connecticut Yankee facility have shown that the insulating function of unprotected Marathan Model M-6012 terminal blocks when exposed to the temperature pressure, and humidity conditions which could result from a LOCA inside containment have failed to survive the planned 24 hour duration of an environmental qualification screening test. In the same screening test, in protected Westinghouse Model #542247 terminal blocks survived the planned 24 hour test duration. Unprotected terminal blocks are those that are not installed in sealed or vented metal enclosures. It is requested that you determine whether or not unprotected terminal blocks are located inside containment in your facility and are used in safety system circuits required to function during or subsequent to design basis accidents.

If unprotected terminal blocks are used, you should be prepared to meet with the staff on February 1, 1978 to discuss the environmental conditions for which the terminal blocks have been qualified, including submergence if applicable. You should have available documentation to support your conclusions regarding the qualification of any unprotected terminal block used as described above. If not completely qualified for all expected environmental conditions, you should be prepared to discuss the basis for continued operation of your facility. This information should be reported by telephone to NRR by 12:00 noon January 31, 1978. We plan to issue a bulletin on January 30, 1978, which will document this request and specify written reporting requirements in this matter.

NRC Central

February 2, 1978

Edward Luton, Esq., Chairman
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dr. Emmeth A. Luebke
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dr. Franklin C. Daiber
College of Marine Studies
University of Delaware
Newark, Delaware 19711

In the Matter of
Rochester Gas & Electric Corporation
(R. E. Ginna Nuclear Power Plant, Unit No. 1)
Docket No. 50-244

Gentlemen:

Attached for your information is RG&E's response of January 16, 1978, to the Staff's letter dated December 16, 1977, requiring a submittal dealing with the ECCS model for the Ginna facility. The Staff's letter to RG&E was forwarded to the Board by letter dated December 30, 1977.

Sincerely,

/s/

Auburn L. Mitchell
Counsel for NRC Staff

Attachment
As Stated

cc w/attachment:
Leonard M. Trosten, Esq.
Mr. Michael Slade
Robert E. Lee, Ph.D.
Jeffrey Cohen, Esq.
Warren B. Rosenbaum, Esq.
Atomic Safety and Licensing Board Panel
Atomic Safety and Licensing Appeal Board
Docketing and Service Section

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Ketchen
TWambach
A. Schwencer

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SURNAME	ALM Mitchell/dmr	Reis				
DATE		2/2/78				

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ROCHESTER GAS AND ELECTRIC CORPORATION • 89 EAST AVENUE, ROCHESTER, N.Y. 14649

LEON D. WHITE, JR.
VICE PRESIDENT

TELEPHONE
AREA CODE 716 546-2700

50-244

January 16, 1978

Director of Nuclear Reactor Regulation
Attention: Mr. A. Schwencer, Chief
Operating Reactor Branch #1
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr. Schwencer:

In a letter from Mr. Edson Case dated December 16, 1977, it was stated that the NRC staff has re-evaluated the acceptability of the calculational model used to evaluate the performance of the emergency core cooling system (ECCS) in Westinghouse designed two reactor coolant loop plants. The letter asked that we develop additional bases for continued safe operation of the R. E. Ginna facility and asked that we propose any additional operating limits which might be required.

We have performed analyses to demonstrate the effectiveness of the R.E. Ginna ECCS and have developed bases for continued safe operation in accordance with 10 CFR 50.46 and Appendix K to 10 CFR Part 50. The methods and results of our analysis are presented in Attachment A. Based upon this analysis no additional operating limits are appropriate.

Sincerely yours,

L D White Jr.

L. D. White, Jr.

Att.

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ATTACHMENT A

BASES FOR CONTINUED OPERATION OF WESTINGHOUSE DESIGNED TWO LOOP PLANTS WITH UPPER PLENUM INJECTION

INTRODUCTION

On December 16, 1977, the United States Nuclear Regulatory Commission issued a letter to the owners and operators of Westinghouse designed two loop plants with upper plenum injection. Attached to this letter were Safety Evaluation Reports from both the Analysis Branch and the Operating Reactors Branch of the NRC. The letter requested that an analysis be performed which conservatively accounted for upper plenum low head safety injection in order to provide additional bases for continued operation. The following discussion outlines the interim basis for continued safe operation of the plant.

METHOD

An analysis has been performed to assess the possible safety and operation impact of the NRC conclusions regarding two loop plants with upper plenum injection. The basis for this analysis is the "Staff Model" described in the NRC Analysis Branch "Safety Evaluation Report on ECCS Evaluation Model for Westinghouse Two Loop Plants", November, 1977 (SER).

Westinghouse Electric Corporation wrote a computer program based on the description in the SER. This program was verified as giving results consistent with the staff model by comparing it to a listing and sample output of the NRC staff model. Following this verification, the following changes were made to the model.

1) The clad temperature rise versus flooding rate curve, Figure 24 in the SER, was replaced by a more realistic curve. The new curve was based on the Westinghouse design FLECHT correlation with input more specific to the Westinghouse two loop plants.

2) The input was changed to allow transient input for pressure, injection rates, flooding rates and decay heat.

3) The carryover fraction, CRF, discussed on page 40 of the SER was changed from 0.8 in the staff model to 0.7 in the Westinghouse model. Carryover fractions of 0.7 are more typical of the two loop plants.

4) The bottom quench front in the staff model was initialized at 0.0 feet. Since this calculation starts some 20 seconds into reflood, the Westinghouse model initiates the bottom quench front at 1.5 feet which is a lower bound value from the Westinghouse ECCS Evaluation Model results.

5) The heat transfer model, described on page 37 of the SER, was altered to account for the amount of heat transfer in the unquenched region which is going to the bottom generated steam rather than the top generated steam. This was done by reducing the heat transfer to the top generated steam by 25 per cent. This is a conservative lower bound.

6) The metal heat model was altered to take into account the finite amount of heat stored in the upper plenum metal. The heat capacity of the upper plenum metal is 5930 (BTU/°F). This metal energy is removed in a finite period of time after which no energy is added to the fluid from the metal resulting in increased subcooling for the remainder of the transient.

In addition to these code changes, the input was also changed from the NRC staff model to more accurately match the plant conditions. These changes involve the transient core pressure and decay heat obtained from the Appendix K Analyses of Record, submitted for R.E. Ginna on April 7, 1977. Finally, 100 percent of ANS decay heat was used for upper plenum injection water steam generation. The base case was 120 percent of ANS decay heat. Therefore, the hot rod temperature rise calculation was performed with 120 percent of ANS decay heat. This treatment of decay heat is in accordance with Appendix K to 10 CFR Part 50 since the base case includes the 120 percent of ANS decay heat.

RESULTS

The results for the six units involved are summarized in the attached table. The results for Ginna, identified as RG&E in the attached table, show a reduced peak clad temperature. Thus, the current plant Technical Specifications continue to ensure compliance with Appendix K to 10 CFR Part 50 and to 10 CFR Section 50.46 and no plant operating restrictions are necessary. It should be pointed out that this simple calculation remains overly conservative since 100 percent upper injection distribution and no hot spot cooling by the upper plenum injection water were assumed. Also, a dynamic calculation incorporating all of the hydraulic feedback mechanisms would yield more favorable results.

UPPER PLENUM INJECTION RESULTS

	CURRENT WESTINGHOUSE EVALUATION MODEL ANALYSIS		NEW U.P.I. ANALYSIS	
PLANT	F_q	PEAK CLAD TEMPERATURE	F_q	PEAK CLAD TEMPERATURE
WEP/WIS	2.32	1965	2.32	1872
RGE	2.32	1957	2.32	1852
NSP/NRP	2.32	2187	2.32	2067
WPS	2.25	2172	2.25	2052

NRC Central

January 17, 1978

Edward Luton, Esq., Chairman
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dr. Emmeth A. Luebke
Atomic Safety and Licensing Board
U.S. Nuclear Regulatory Commission
Washington, DC 20555

Dr. Franklin C. Daiber
College of Marine Studies
University of Delaware
Newark, Delaware 19711

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Docket No. 50-244

Gentlemen:

Attached for your information is a letter from Victor Stello, Director, Division of Operating Reactors, NRR to RGE dated December 23, 1977, stating that the first topic of review under the Systematic Evaluation Program (SEP) will be Environmental Qualification of Safety-Related Equipment. NUREG-0413 on that subject is also enclosed. The letter directs RG&E to submit certain identified information to the Staff within 60 days from December 23, 1977.

Sincerely,

/S/

Auburn L. Mitchell
Counsel for NRC Staff

Enclosures
As Stated

cc w/encl: Leonard M. Trosten, Esq.
Mr. Michael Slade
Robert E. Lee, Ph.D.
Jeffrey Cohen, Esq.
Warren B. Rosenbaum, Esq.
Atomic Safety and Licensing Board Panel
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January 11, 1978

Mr. Michael Slade
1250 Crown Point Drive
Webster, New York 14580

In the Matter of
Rochester Gas & Electric Corporation
(R. E. Ginna Nuclear Power Plant, Unit No. 1)
Docket No. 50-244

Dear Mr. Slade:

Enclosed are copies of the Licensing Board decisions and the Denial of Petition for Rulemaking which were intended for enclosure in my letter to you of December 28, 1977. Thanks for your call yesterday advising me of this omission.

Sincerely,

/s/

Auburn L. Mitchell
Counsel for NRC Staff

Enclosures
As Stated

cc w/o encl: Edward Luton, Esq., Chairman
Dr. Franklin C. Daiber
Dr. Emmeth A. Luebke
Leonard M. Trosten, Esq.
Robert E. Lee, Ph.D.
Jeffrey Cohen, Esq.
Warren B. Rosenbaum, Esq.
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