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# DRAFT

# TECHNICAL PROGRESS REVIEW

# NUCLEAR SAFETY

# VOLUME 3, NO. 4

## NOTICE

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# TECHNICAL PROGRESS REVIEW

# NUCLEAR SAFETY

Volume 3, No. 4

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by Oak Ridge National Laboratory

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OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
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U. S. ATOMIC ENERGY COMMISSION

# FOREWORD

This quarterly journal is one of a series of Technical Progress Reviews prepared by the Oak Ridge National Laboratory at the request of the Division of Information Services, U. S. Atomic Energy Commission. This Review is intended to assist those interested in keeping abreast of significant developments in the field of nuclear safety. Nuclear Safety is not a comprehensive abstract of all literature published in this field during a given quarter, but rather a mechanism for presenting concise reviews of selected subjects as prevailing interest and available information warrant.

Coverage of the Review is limited to topics relevant to the analysis and control of hazards associated with nuclear reactors, operations involving fissionable materials, and the products of nuclear fission. Primary emphasis is on safety in reactor design, construction, and operation; however, safety considerations in reactor fuel fabrication, spent-fuel processing, nuclear waste disposal, and related operations are also treated. Safety in the use of radioisotopes in industry, medicine, and research is excluded, as are most topics considered the province of health physics. Even with these exclusions, nuclear safety cuts across such diverse fields as nuclear physics, solid-state physics, mechanics, chemistry, meteorology, geology, seismology, metallurgy, law, and nearly all branches of engineering. The authors will therefore review material from these fields which, in their opinion, has a direct bearing on nuclear safety.

Two distinctly different types of article may be found in this issue of Nuclear Safety. These include reviews of current literature and special

review articles on specific topics. The editors feel that both types of article make a necessary and distinctive contribution to Nuclear Safety. The special review articles permit discussion of pertinent subjects which cannot be adequately considered by reference to only the current literature. The current review articles, however, constitute the major portion of this publication. All incoming literature (including reports, books, American and foreign technical journals, and transactions) is examined for subjects within our area of interest. This material is collected, grouped, and reviewed by experts. Interpretations in any article represent the opinions of the editors, who are employees of the Oak Ridge National Laboratory. Readers are urged to consult the references to original work for more complete information.

It is recognized that the critical evaluation of subject areas leading to the determination of criteria cannot fail to stimulate contrary opinions. This is expected to be particularly true in the area of nuclear safety, since in many instances only preliminary information is available, the ramifications are many and varied, and opinion and judgment must be relied upon so heavily. While the editors do not propose that the pages of Nuclear Safety act as a clearing house for safety correspondence because of the above facts, we have had for some time a policy which would permit the publication of statements of position at variance with those expressed by the editors. Such statements will be published after the editors have ascertained that a real difference exists and that the position is reasonable.

In addition to the invited contributors, many members of the Oak Ridge National Laboratory staff wrote review material, reviewed manuscripts, or

otherwise contributed to this publication. Their contributions are gratefully acknowledged.

W. B. COTTRELL, Editor
W. H. JORDAN, Associate Editor
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Ridge National Laboratory

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# I. GENERAL SAFETY CRITERIA

# STATUS OF ACTIVITIES IN NUCLEAR STANDARDS

(Editor's note: An expansion and updating of information reported previously in this Journal.[1])

The development of nuclear standards in the United States is primarily the responsibility of the various technical societies, scientific organizations, trade associations, manufacturers, and other groups directly affected by these standards. Many areas of the nuclear industry are, however, under regulatory control of the government because of military aspects and the possibility of catastrophic accidents. As technology advances, the extent of regulatory coverage will diminish, and the regulations will be modified into voluntary standards. To be useful, that is, to have wide acceptance, a standard must be approved by all affected organizations. With this in mind the American Standards Association, a federation of technical societies, trade associations, and numerous companies, provides a mechanism for nationwide agreement on standards.[2]

Within the ASA, all nuclear standards work is coordinated by the Nuclear Standards Board under which the N1 through N7 sectional committees and subcommittees are organized. When the need for a standard is determined, the sectional committees try to interest member groups in developing the required information, with the ASA acting in a coordinating or editorial function. This failing, the ASA committee proceeds to develop the standard. Approval of a standard by ASA is indicated by the official designation American Standard. To attain this, the standard, during the course of development, must receive acceptance by a consensus of all concerned.

There are, in addition, many company standards and technical society standards which have never been submitted to ASA for approval. Such standards are nonetheless valid when accepted by those concerned and some are nationally recognized. One of the best examples of this is the Boiler Code, a standard of the American Society of Mechanical Engineers.

An informal cooperative relationship is maintained between ASA and AEC, since compatibility between industry standards and government regulations must exist. Also, the American Standards Association represents the United States in the International Organization for Standardization. Technical Committee 85 of the ISO is concerned with international standardization in the broad field of nuclear energy.

Table I-1 in the previous article[1] presented a list of standards-producing organizations directly involved in nuclear safety; the list is repeated in revised Table I-1 in which significant changes and added information are bracketed. The International Commission on Radiological Protection has been included because of the general acceptance of its recommendations in the establishment of standards and regulations in this country.

The status of work in these organizations is indicated in Table I-2 primarily by listing published documents or citing information under development, i.e., in draft form. For presentation here, pertinent documents generated by standards organizations were categorized as to the stage of development, i.e., published, final draft, and preliminary draft. In many cases the document titles are descriptive of their text and indicate the specific field of endeavor of the issuing organization. Categorization is based on the following: Published documents are those which have been

released to the public. Some of these have been officially designated American Standard, whereas others either require committee action of an editorial nature only or have been developed under organizations other than ASA. Final drafts are documents which have already incorporated comments of committee members and other interested persons and have been submitted for committee or organization approval. At this stage it is unlikely that the technical content of the documents will undergo any further significant changes. Sometimes public comment is solicited prior to final committee action. Preliminary drafts are documents which represent the initial organized ideas on a subject by the responsible working group. These drafts are usually circulated only to committee members and persons with a particular interest in the subject because extensive revision is sometimes required before a final draft is produced.

In areas of more immediate need of standardization, the use of standards is sometimes expedited by case decisions. These case decisions are rendered by groups of qualified and knowledgeable people who have been delegated by the responsible standards organization. The decisions are then transcribed and published for use by industry in lieu of officially changing an existing code, the intention being that the code will be changed or elaborated on when sufficient case decisions have been made to warrant such action. Notification of such decisions by ASA-sponsored groups is given in the ASA publication, The Magazine of Standards; in addition, individual copies are made available to member companies, e.g., Cases N-1 through N-12 pertaining to nuclear applications of the code for pressure piping. The American Society of Mechanical Engineers publishes case decisions concerning the ASME Boiler Code in their publication, Mechanical

Engineering. Of particular interest are Cases 1270N through 1276N pertaining to nuclear applications of the Boiler Code. Of the documents listed in Table I-2, many are of topical interest with respect to the safety aspects of nuclear reactor design, and it is anticipated that in future issues of Nuclear Safety they will be reviewed and discussed. (C. L. Whitmarsh)

Table I-1. LIST OF ORGANIZATIONS PRODUCING STANDARDS DIRECTLY INVOLVING SAFETY IN THE USE OF NUCLEAR ENERGY

---

Ia

American Standards Association, Nuclear Standards Board, 70 East 45th Street, New York, H. G. Lamb, Secretary. This board has 33 organization members and two members at large. President: Morehead Patterson.

ASA Sectional Committee N1. American Standard Glossary of Terms in Nuclear Science and Technology. This committee has already published Glossary of Terms (N1.1, 1957). The committee has been dissolved, and Subcommittee N2-4 will handle all revisions and additions to the glossary.

ASA Sectional Committee N2. General and Administrative Standards for Nuclear Energy. Sponsor: Atomic Industrial Forum, Inc., 3 East 54th Street, New York 22, N. Y. This committee has 17 organization members. It studies standards, specifications, and methods of administration associated with the peaceful uses of nuclear energy, including color codes, symbols, nomenclature, qualifications of professionals, records and reporting systems and procedures, and accountability of materials. The following subcommittees have been established:

N2-1 Color Codes and Symbols

N2-2 Procedures for Industrial Exposure Records

N2-3 Qualifications of Nuclear Professionals

N2-4 Nuclear Terminology (also represents ASA on ISO Technical Committee 85-SC1: Terminology, Units, and Symbols)

N2-5 Model Atomic Energy Legislation

ASA Sectional Committee N3. Nuclear Instrumentation. Sponsor: Institute of Radio Engineers, 1 East 79th Street, New York 21, N.Y. This committee has 21 organization members. It represents ASA on IEC Technical Committee 45, Electrical Instrumentation. The committee studies standards, specifications, and methods of testing for instrumentation in the nuclear field, including instruments for personnel protection, reactor control, industrial processes, analysis and laboratory work, radiation calibration equipment, and components therefor. Subcommittees are as follows:

N3-1 Index of Nuclear Standardization Work (a report has been published: "Index of Nuclear Standardization Work" by R. F. Shea; this report is obtainable from L. G. Cumming, Technical Secretary, Institute of Radio Engineers, Inc., 1 East 79th Street, New York 21, N.Y.)

N3-2 Topical Structure

ASA Sectional Committee N4. Electric Apparatus and Systems for the Nuclear Field. Sponsors: American Institute of Electrical Engineers, 33 West 39th Street, New York 18, N. Y.; Electric Light and Power Group, consisting of Edison Electric Institute, 750 Third Avenue, New York 17, N.Y., and Association of Edison Illuminating Companies, 51 East 42nd

Street, New York 17, N. Y.; and National Electric Manufacturers Association, 155 East 54th Street, New York 17, N. Y. This committee has 12 organization members. No subcommittees are listed by ASA. Standards for electrical power apparatus, systems, and controls used in the nuclear field are considered by this committee.

ASA Sectional Committee N5. Nuclear Fuel Cycle Engineering. Sponsor: American Institute of Chemical Engineers, 25 West 45th Street, New York, N. Y. This committee has 22 organization members. It considers standards, specifications, tolerances, and methods of testing for the chemical engineering aspects of the nuclear field, including the refining, processing, separation, purification, treatment, packaging, handling, and disposal of fuels, radioisotopes, industrial chemicals, pharmaceuticals with isotope tracers, and radioactive wastes (gases, liquids, and solids); the use of radioactive sources for tracers and for processing foods and other materials; the application and use of chemically resistant coatings; and the cleaning of contaminated equipment and facilities. Subcommittees are as follows:

N5-1 Fuel Manufacture and Fabrication

N5-2 Radioactive-waste Disposal

N5-3 Recovery of Irradiated Fuel

N5-4 Use and Handling of Radioisotopes and High-energy Radiation (also represents ASA on ISO Technical Committee 85-SC4, Radioisotopes)

N5-5 Packaging and Transportation of Radioactive Material

ASA Sectional Committee N6. Reactor Safety. Sponsors: American Nuclear Society, 86 East Randolph Street, Chicago 1, Ill., and American Society of Mechanical Engineers, United Engineering Center, 345 East 47th Street, New York 17, New York. This committee has 44 organization members. It represents ASA on ISO Technical Committee 85-SC3, Reactor Safety. It considers codes and standards concerned with the hazards involved in the design, location, construction, and operation of nuclear reactors and of potentially critical assemblies. Subcommittees are as follows:

N6-1 Site Evaluation

N6-2 Containment

N6-3 Fluid Systems of the Reactor and Fuel within the Reactor

N6-4 Reactor Dynamics and Control Requirements

N6-5 Instrumentation and Execution of Control Requirements

N6-6 Operation, Operator Qualifications, Inspection and Maintenance, and Records

N6-7 Failure Probabilities and Maximum Credible Accidents

N6-8 Fissionable Material Outside Reactors (normally subcritical systems)

ASA Sectional Committee N7. Radiation Protection. Sponsors: Atomic Industrial Forum, Inc., 3 East 54th Street, New York 22, N. Y., and National Safety Council, 425 North Michigan Avenue, Chicago 11, Ill. This committee has 22 organization members. It represents ASA on ISO Technical Committee 85-SC2. It studies safety standards for the protection of persons employed in facilities associated with the production and utilization of fissionable materials against the

normal, routine hazards present in such facilities (mines, mills, refineries, separation plants, fuel-element-fabrication plants, critical assembly facilities, processing and reprocessing plants, working areas around reactors, and transportation and purification of radioactive materials). Subcommittees are as follows:

N7-1 Uranium Mines and Mills

N7-2 Uranium and Thorium Refineries

N7-3 Isotopic Separation Plants

N7-4 Health Physics of Fuel-element Fabrication

N7-5 Health Physics for Reactors

N7-6 Transportation of Radioactive Materials

Ib

American Standards Association Committees on standards closely allied with nuclear safety but not under the jurisdiction of the ASA Nuclear Standards Board.

ASA Sectional Committee Z54. Industrial Use of X Rays and Radiation.

Sponsor: National Bureau of Standards. Subcommittees are as follows:

Z54-1 General Provisions: Methods and Materials of Protection

Z54-2 Health Provisions and Monitoring

Z54-3 X-ray Protection for Installations up to 2 Million Volts

Z54-4 Protection for Installations Above 2000 Kv

Z54-5 Gamma-ray Sources for Industrial Radiography

Z54-6 Electrical Protection

Z54-7 X-ray Diffraction, Fluorescence Analysis, and Microradiography

Z54-8 Sealed Beta-ray Sources

Z54-9 Contamination Levels for Industrial Materials

ASA Sectional Committee Z62. Uniform Industrial Hygiene Standards

ASA Sectional Committee B31. Code for Pressure Piping. Sponsor: American Society of Mechanical Engineers. This committee considers piping from the reactor to the power unit. Materials covered to date are primarily stainless steels; future plans include the development of standards for carbon and alloy steels and nonferrous metals.

II

American Meteorological Society, Committee on Air Pollution. Chairman: G. R. Hilst, Hanford Atomic Products Operation, Richland, Wash. The aim of this committee is to produce standards regarding the meteorological phase of waste disposal to the atmosphere. It has no formal connection with other bodies. Findings are promulgated through the Society's official publications.

III

American Nuclear Society Standards Committee. Chairman: C. R. McCullough, 15201 Rosecroft Road, Rockville, Md. This committee is composed of an eight-member steering committee and 10 projects (subcommittees). The ANS standards committees work very closely with the ASA nuclear standards committees N2, N3, N4, N5, N6, N7, and the ASA Nuclear Standards Board. ANS cosponsors ASA Sectional Committee N6, Reactor Safety, together with ASME.

ANS Project 1. Reactor Classification. Chairman: U. M. Staebler, USAEC, Washington 25, D. C. The aim of this committee is to develop a standard classification of reactors by type and use for reference in future reactor-standards work. Copies of a proposed standard have been distributed for comments.

ANS Project 2. Reactor Environment. Chairman: R. O. Brittan, Argonne National Laboratory, Box 299, Lemont, Ill. (See also N6-1 and N6-2). The aim of this committee is to establish standards with respect to population, meteorology, geology, proximity to roads, railroads, waterways, containment system, and other factors pertinent to the selection of suitable reactor sites.

ANS Project 3. Reactor Operator Qualifications. Chairman: M. M. Mann, New York Shipbuilding Corporation, Camden 1, New Jersey. (See also N2-3 and N6-6.) The aim of this committee is to establish standards for evaluating the qualifications of persons as reactor operators of various types and degrees of responsibility.

ANS Project 4. Reactor Operation. Chairman: M. M. Mann, New York Shipbuilding Corporation, Camden 1, New Jersey. (See also N6 and N5.) The aim of this committee is to establish standards with respect to local review boards, inspections, monitoring, maintenance and repair, prestartup checkouts, rules of conduct, organization records, visitors, and other matters which affect the safety of reactor operation.

ANS Project 5. Methods of Estimating Energy and Fission Products Release. Chairman: W. E. Johnson, Atomic Power Department, Westinghouse Electric

Corporation, P. O. Box 355, Pittsburgh 30, Pennsylvania. (See also N6-7). The aim of this committee is to establish standards for estimating probable energy and fission products released in reactor accidents.

ANS Project 6. Reactor Dynamic System Design. Chairman: M. M. Shapiro, Nuclear Development Corporation of America, 5 New Street, White Plains, N. Y. (See also N6, especially N6-3 and N6-4.) The aim of this committee is to establish standards for the general design of the overall reactor complex (cooling systems, safety devices, stability in transient regimes, etc.).

ANS Project 7. Reactor Components. Chairman: R. G. Chalker, Atomics International, Box 309 Canoga Park, Calif. (See also N3, N6, and N7.) The aim of this committee is to establish standards for design, production, and testing of components of reactor systems.

ANS Project 8. Fissionable Material Outside Reactors. Chairman: A. D. Callihan, Oak Ridge, Tenn. (See also N6-8,) The aim of this committee is to establish realistic standards for the prevention of unintentional critical assemblies.

ANS Project 9: Nuclear Glossary Subcommittee. Chairman: J. A. Hunter, Martin Marietta Corporation, Baltimore, Maryland. (See also N2-4,) In cooperation with the ASA N2-4 committee suggested definitions are being prepared for the trilingual nuclear glossary.

ANS Project 10: Units in the Nuclear Industry. Chairman: V. L. Parsegian, Rensselaer Polytechnic Institute, Troy, N. Y. The aim of this

committee is to establish standard units, with definitions, for use in the nuclear energy field.

IV

American Society of Civil Engineers, The Administrative Committee on Nuclear Structures and Materials. Chairman: H. M. Glen, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Task Committee on Hot Laboratories. Chairman: John M. Ruddy, Brookhaven National Laboratory, Upton, Long Island, New York.

Task Committee on Live Loads Due to Heat and Incidence. Chairman: Professor J. E. Goldberg, School of Civil Engineering, Purdue University, Lafayette, Ind.

Task Committee on Structural Aspects of Nuclear Incidents. Chaiman: G. Morris, Oak Ridge National Laboratory, Oak Ridge, Tenn.

Task Committee on Structural Materials in Reactor Design. Chairman: S. H. Fistedis, Argonne National Laboratory, Lemont, Ill.

V

American Society of Mechanical Engineers, Committee on Reactor Safety Standards. Standards Manager: George C. Finster, ASME, United Engineering Center, 345 East 47th Street, New York, N. Y. ASME has representatives on ASA Committees N2, N3, N4, N5, N6, N7, and the ASA Nuclear Standards Board. ASME sponsors the ASA Sectional Committee B31, Code for Pressure Piping, and cosponsors the ASA Sectional Committee N6, Reactor

Safety. ASME prepares and distributes many ASME standards of direct and indirect bearing on nuclear safety, in addition to other standards (e.g., ASME Vessel and Boiler Codes).

Special Committee on Nuclear Power. In 1955 this committee was established by the Boiler and Pressure Vessel Committee to extend the use of the construction rules of the Code to vessels used in nuclear installations, to modify them where necessary, and to develop such additional rules as required for the special construction and uses involved. Adopted rules are presently grouped under seven general headings and bear case numbers 1270N and 1276N. A new section of the Code is now being prepared to cover vessels in nuclear service.

Special Committee to Review Code Stress Basis. This committee plans to use the standard SA specifications in Section II of the Code, but to apply additional requirements, such as ultrasonic and other inspection, heat treatment, and to specify permissible defects, by means of additional specifications after the fashion of SA-300, which gives impact test requirements for plate specifications.

Research Committee on Effects of Radiation on Materials. The objective of this committee is to convert research data to a form useful to engineers and designers, as well as to develop new data where needed.

VI

American Society for Testing Materials, Special Administrative Committee on Nuclear Problems. Chairman: Marcel A. Cordovi, ASTM, 1916 Race Street, Philadelphia, Pa. The following subcommittees are working on standards in the nuclear energy field:

ASTM Committee A-5. Corrosion of Iron and Steel

ASTM Committee A-10. Iron-Chromium, Iron-Chromium-Nickel, and Related Alloys (six groups)

ASTM Committee B-2. Nonferrous Metals and Alloys

ASTM Committee B-7. Light Metals and Alloys Cast and Wrought

ASTM Committee C-9. Concrete and Concrete Aggregates (with special subcommittee on shielding)

ASTM Committee C-21. Ceramic Whiteware and Related Products

ASTM Committee D-2. Petroleum Products and Lubricants

ASTM Committee D-9 and D-20. Joint Subcommittee on Effects of Nuclear and High-Energy Radiation (insulating materials and plastics)

ASTM Committee D-18. Soils for Engineering Purposes

ASTM Committee D-19. Industrial Water

ASTM Committee E-1. Methods of Testing

ASTM Committee E-2. Emission Spectroscopy

ASTM Committee E-3. Chemical Analysis of Metals

ASTM Committee E-7. Nondestructive Methods of Testing

ASTM Committee E-10. Radioisotopes and Radiation Effects

ASTM Committee E-14. Mass Spectroscopy

VII

Atomic Industrial Forum, Committee on Codes and Standards. Chairman: Robert Wells, Secretary: G. Edwin Brown, Jr., Atomic Industrial Forum,

Inc., 3 East 54th Street, New York, N. Y. Nineteen different segments of industry are represented on this committee. The primary objective of the committee is "to provide leadership on behalf of the Forum in the American Standards Association's nuclear standards programs and to review and comment on proposed federal, state, and local legislation affecting safety and other standard areas in nuclear energy." Summaries of the meetings of the committee are available from the Atomic Industrial Forum, Inc.

The Atomic Industrial Forum is a member of ASA and is represented on the ASA Board of Directors, ASA Standards Council, and ASA Nuclear Standards Board. It sponsors ASA Sectional Committee N2 and cosponsors, with the National Safety Council, ASA Sectional Committee N7. It appointed the U. S. delegates to ISO Technical Committee 85-SCI and SC2, the international counterparts of ASA N2 and N7.

VIII

Institute of Radio Engineers. IRE has representatives on ASA Committees N3 and N6 and the ASA Nuclear Standards Board. IRE sponsors ASA Sectional Committee N3, which also represents ASA on IEC Technical Committee 45, Electrical Instrumentation.

Committee 13 on Nuclear Techniques. Chairman: R. W. Johnson, Chief, Radiation Instruments Branch, USAEC, Washington 25, D. C. The scope of the standards work affecting nuclear safety includes (1) the selection of terms and the preparation and maintenance of standard definitions in the fields of nuclear radiation and nuclear instrumentation, (2) the preparation and maintenance of standards covering

methods of measurement in these fields, (3) the compilation of information relating to operating standards in these fields, and (4) coordination with activities of other IRE committees and other professional societies and liaison with technical organizations engaged in allied work.

IX

National Electric Manufacturers Association, 155 East 44th Street, New York, N. Y., J. F. Miller, Managing Director. NEMA has representatives on ASA Sectional Committees N3, N4, and N6 and on the ASA Nuclear Standards Board. NEMA has developed standards for canned motor pumps for nuclear applications.

X

U. S. Department of Commerce, National Bureau of Standards. A. V. Astin, Director. Publications available from the Superintendent of Documents, U. S. Government Printing Office, Washington 25, D. C. NBS prepares handbooks summarizing the recommendations of the National Committee on Radiation Protection and Measurements (NCRP), which also works closely with the International Commission on Radiological Protection (ICRP). The NCRP was organized and continues to operate under sponsorship of the NBS. Chairman of the main committee of NCRP is L. S. Taylor, National Bureau of Standards, Washington 25, D. C. Forty-one main committee members and 18 subcommittees are listed, with their respective chairman, in NBS Handbook 69, Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure, issued June 5, 1959.

The following NBS handbooks that relate directly to safety in utilization of nuclear energy are available:

42 Safe Handling of Radioactive Isotopes

48 Control and Removal of Radioactive Contamination in Laboratories

49 Recommendations for Waste Disposal of $P^{32}$ and $I^{131}$ for Medical Users

50 X-ray Protection Design

51 Radiological Monitoring Methods and Instruments

53 Recommendations for the Disposal of $C^{14}$ Wastes

54 Protection Against Radiation from Radium, $Co^{60}$, and $Cs^{137}$

55 Protection Against Betatron-Synchrotron Radiations up to 100 Million Electron Volts

57 Photographic Dosimetry of X and Gamma Rays

58 Radioactive-waste Disposal in the Ocean

59 Permissible Dose from External Sources of Ionizing Radiation

60 X-ray Protection

61 Regulation of Radiation Exposure by Legislative Means

62 Report of the International Commission Radiological Units and Measurements (ICRU) 1956

63 Protection Against Neutron Radiation up to 30 Million Electron Volts

64 Design of Free-air Ionization Chambers

65 Safe Handling of Bodies Containing Radioactive Isotopes

66 Safe Design and Use of Industrial Beta-ray Sources

69 Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure

XI

Society of Naval Architects and Marine Engineers, Ships Machinery Committee, M-13 Panel. Chairman: Arthur R. Gatewood, American Bureau of Shipping, 45 Broad Street, New York, N. Y. The M-13 Panel prepared Technical Research Bulletin 3-6, "Safety Considerations Affecting the Design and Installation of Water-cooled and Water-moderated Reactors on Merchant Ships," at the request of the U. S. Coast Guard.

XII

United States Committee for the Revision of the International Convention for the Safety of Life at Sea, 1948, Nuclear Power Committee. Chairman: A. R. Gatewood, American Bureau of Shipping, 45 Broad Street, New York, N. Y. A report has been prepared on request of the U. S. Coast Guard for transmission to the U. S. State Department, who presented it to the International Marine Consultative Organization (IMCO) for distribution to the governments who will participate in the 1960 Safety of Life at Sea Conference in London, England.

XIII

Electronic Industries Association, 1721 DeSales Street NW, Washington 6, D. C., J. D. Secrest, Executive Vice President.

Committee TR-19. Committee on Nuclear Instrumentation, Chairman: N. Anton, 1721 DeSales Street NW, Washington 6, D. C.

Committee TR-19.1. Subcommittee on Dosimeters and Chargers.

Committee TR-19.4. Subcommittee on Film Badge Dosimetry.

XIV

International Commission on Radiological Protection, Radiofysiska Institutionen, Stockholm 60, Sweden. Sponsor: The International Congress of Radiology. The Commission has an official relationship with the World Health Organization and the International Atomic Energy Agency. Its main objective is to make recommendations on radiation safety standards. All such recommendations are published by Pergamon Press, Ltd., Headington Hill Hall, Oxford, England. In order to generate this information, the following six committees were established:

Committee I. Advisory Committee on Biology

Committee II. Protection Against Radiation from Internal Radioactive Substances

Committee III. Protection Against X-Rays and Electrons up to Energies of 3 Mev and Beta- and Gamma-Rays from Sealed Sources

Committee IV. Protection Against Electrons and Electromagnetic Radiation above 3 Mev, Neutrons, and Radiation from Heavy Particle Accelerators

Committee V. Handling of Radioactive Isotopes and Disposal of Radioactive Waste

Committee VI. Committee on RBE

XV

American Institute of Electrical Engineers, Nucleonics Committee. Chairman: A. Y. Taylor. AIEE cosponsors ASA Committee N4 and has worked closely with ASA Committees N2 and N6 in developing standard qualifications for personnel.

Standards and Safety Subcommittee. Chairman: L. Horn

XVI

National Fire Protection Association, Committee on Atomic Energy. Chairman: James Troutman, Factory Insurance Association, 85 Woodland Street, Hartford 2, Connecticut. This committee considers applicability of existing fire protection codes to nuclear facilities and the development of new codes for unique situations.

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Table I-2. STATUS OF PROJECTS OF STANDARDS-PRODUCING ORGANIZATIONS

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Ia

American Standards Association, Nuclear Standards Board, 10 East 40th Street, New York 16, N. Y., H. G. Lamb, Secretary.

ASA Sectional Committee N2, General and Administrative Standards for Nuclear Energy.

Subcommittee N2-1, Color Codes and Symbols. The subcommittee's American Standard Radiation Symbol was adopted as an American Standard

(N2.1-1960) and is being considered for an international standard by the ISO Technical Committee 85. This committee has been requested to develop a standard for warning lights to be used on nuclear reactors.

Subcommittee N2-2, Procedures for Industrial Exposure Records. This subcommittee has recently been reorganized.

Subcommittee N2-3, Qualifications of Nuclear Professionals. The subcommittee efforts have primarily been liaison and assistance to ASA Committee N6-6.

Subcommittee N2-4, Nuclear Terminology. The subcommittee is working on a revision of N1.1-1957, "American Standard Glossary of Terms in Nuclear Science and Technology." It is also cooperating with ISO Technical Committee 85 in the preparation of a trilingual glossary.

Subcommittee N2-5, Model Atomic Energy Legislation. No progress was reported.

ASA Sectional Committee N3, Nuclear Instrumentation. Subcommittee efforts have been directed toward internal procedures, e.g., proper committee functions, methods for evaluating standards proposals, and liaison with other standards committees.

ASA Sectional Committee N4, Electrical Apparatus and Systems for the Nuclear Field.

Subcommittee N4-1, Electrical Power Apparatus. The subcommittee is reviewing existing ASA standards for electrical equipment to determine which require modifications for nuclear application. Considerable

commentary on NEMA's canned motor pumps standards has indicated that this document requires some modifications before submission to ASA for consideration as an American Standard.

Subcommittee N4-2, Electrical Auxiliary Power System. This subcommittee is also reviewing existing standards for possible nuclear application.

Subcommittee N4-3, Electrical Control Systems and Apparatus. This subcommittee is still in the formative stage and has indicated that its scope may include transitors, magnetic amplifiers, and limit switches in a study of control system reliability and function.

Subcommittee N4-4, Environmental Classification. This subcommittee is also in the formative stage. Its scope will include the establishment of a standard for environmental classification, e.g., classification of radiation fields and their effects.

ASA Sectional Committee N5, Nuclear Fuel Cycle Engineering.

Subcommittee N5-1, Fuel Manufacture and Fabrication. This subcommittee is currently developing standard specifications for uranium melt stock and nuclear-grade sinterable $UO_2$. Also, analytical techniques of potential use in policing standard specifications are being reviewed. Preliminary drafts of the above specifications have been prepared.

Subcommittee N5-2, Radioactive Waste Disposal. Preliminary drafts of "Site Considerations for Land Burial of Solid Radioactive Waste" and "Categorization of Radioactive Wastes" have been circulated for

subcommittee approval for eventual consideration as American Standards. Although not intended as standards per se, other preliminary drafts of interest are "Current Practices in the Disposal of Radioactive Wastes from Operation of Uranium Mines and Mills" and "A Selected Bibliography on Radiochemical Wastes." Studies have been initiated for site evaluation with respect to "small and probable" as well as "maximum credible" accidents and gas and liquid wastes from accidents at nuclear facilities.

Subcommittee N5-3, Recovery of Irradiated Fuel. The subcommittee has prepared a preliminary draft of "American Standard for Accountability Measurement in Spent Reactor Fuel of Source and Special Nuclear Materials." A Study on siting criteria for reprocessing plants is currently being considered as a future project.

Subcommittee N5-4, Use and Handling of Radioisotopes and High-Energy Radiation. A preliminary draft of "Radioisotope Laboratory Design" has been prepared and circulated for comments. Considerable work on "Unsealed Sources" has been performed in conjunction with this group's responsibility to ISO/TC 85/SC 4.

Subcommittee N5-5, Packaging and Transportation of Radioactive Material. A proposed final draft of "Guide to Practice in Transportation of Source and Special Nuclear Material before Irradiation" has been prepared and submitted for committee action. A preliminary draft of "Design and Operation of Shipping Containers for Irradiated Solid Fuel from Nuclear Reactor" has been circulated for comments and recommendations.

ASA Sectional Committee N6, Reactor Safety.

Subcommittee N6-1, Site Evaluation. This subcommittee's objective has been restricted somewhat because industrial practice has not as yet become sufficiently well defined to permit a meaningful concensus. The initial objective of developing a mathematical procedure for determining the consequences of a reactor accident has been delayed in favor of concentrating on a description of factors to be considered in site evaluation.

Subcommittee N6-2, Containment. A preliminary draft of "Proposed Safety Standard for Design, Fabrication and Maintenance of Steel Containment Structure for Stationary Nuclear Power Reactors" has been prepared and distributed.

Subcommittee N6-3, Fluid Systems of the Reactor and Fuel within the Reactor. A preliminary draft of "Standards for Fluid Systems and Fuel within the Reactor" has been prepared and distributed to committee members for comment.

Subcommittee N6-4, Reactor Dynamics and Control Requirements. A draft of "Proposed Safety Standards for Reactor Dynamic Characteristics and Control Requirements" has been prepared and distributed to qualified people for comments.

Subcommittee N6-5, Instrumentation and Execution of Control Requirements. A performance standard, "Control Rod Drive Systems for Water Moderated Reactors," is in the preliminary draft stage. Future efforts are to be directed towards developing mathematical methods which will allow

comparisons between the reliabilities of reactor safety systems on a numerical scale.

Subcommittee N6-6, Operation, Operator Qualifications, Inspection and Maintenance, and Records. A draft of "Proposed Safety Standard for the Operation and Maintenance of Power and Production Reactors" has been prepared with the cooperation of the AIEE. A rewrite is planned which will incorporate many comments from a thorough solicitation of industry and AEC. Future plans are for preparation of similar standards for other types of reactors.

Subcommittee N6-7, Failure Probabilities and Maximum Credible Accidents. A final draft of "Proposed Standard for Design Pressure and Pressure Decay Requirements for Boiling-Water and Pressurized-Water Nuclear Power Plant Containment Vessels" has been prepared. A number of standards are in a preliminary stage of development: (1) "General Guide for Nuclear Reactor Accident Analysis," (2) "Guide for the Accident Analysis of Boiling Water Reactors," (3) "Guide for the Accident Analysis of Pressurized Water Reactors," (4) "Guide for the Accident Analysis of Organic Moderated Reactors," (5) "Guide for the Accident Analysis of Liquid Metal Cooled Reactors," and (6) "Analysis of Missile Containment." Standards on which future activity is intended are (1) Standards for Evaluation of Failure Probabilities, (2) Guide for the Accident Analysis of Gas-Cooled Reactors, and (3) Guides for the Accident Analysis of Research and Test Reactors.

Subcommittee N6-8, Fissionable Material Outside Reactors (normally subcritical systems). A final draft of "A Proposed Safety Code

for Fissionable Materials" has been prepared and circulated to N6 Committee members for comments.

ASA Sectional Committee N7, Radiation Protection.

Subcommittee N7-1, Uranium Mines and Mills. "N7.1-1960, American Standard Radiation Protection in Uranium Mines and Mills (Concentrators)" was officially designated an American Standard in 1960 and is available from American Standards Association, Inc., 10 East 40th Street, New York 16, New York.

Subcommittee N7-2, Uranium and Thorium Refineries. The subcommittee is working on a preliminary draft of a "Standard for Thorium Metal Production Plants."

Subcommittee N7-3, Isotopic Separation Plants. The document, "Radiation Protection in Isotopic Separation Plants," is in the preliminary draft stage.

Subcommittee N7-4, Health Physics of Fuel-Element Fabrication. A final draft of "Radiation Protection Standards for Nuclear Fuel Fabrication Plants — Natural (or Normal) and Enriched Uranium, Thorium and Plutonium" has been prepared.

Subcommittee N7-5, Health Physics for Reactors. A preliminary draft of pertinent information is being developed.

Subcommittee N7-6, Transportation of Radioactive Materials (Radiation Protection Standards for). The work of this subcommittee has been retarded because of conflicting Governmental regulations in this field.

Ib

ASA Sectional Committee Z54, Industrial Use of X-Rays and Radiation

Subcommittees 1 through 7 have been concerned with the revision of the War Standard Z54.1-1945, "Industrial Use of X-rays." Although the information has not yet been coordinated into draft form, much of it has been published in ICRP publications, NBS Handbooks, and reports of the Federal Radiation Council. Subcommittee 8, Sealed Beta-Ray Sources, completed its work with the issuance on May 28, 1958 of NBS Handbook 66 (see X). Subcommittee 9, Contamination Levels for Industrial Materials, prepared a preliminary draft but was inactive during 1961 because of other activities of members.

ASA Sectional Committee B31, Code for Pressure Piping. Cases N-1 through N-12 concerning nuclear applications of piping have been published in the ASA publication, "The Magazine of Standards," during 1960 and 1961.

II

American Meteorological Society, Committee on Air Polution. No direct work on the establishment of standards has been undertaken.

III

American Nuclear Society Standards Committee

ANS Project 1, Reactor Classification. The standard, ANS 1-61, "Standard for Nuclear Reactor Classifications," was adopted by the ANS Standards

Committee on June 6, 1961, and is available from the Executive Secretary, American Nuclear Society, 86 East Randolph Street, Chicago 1, Illinois.

ANS Project 2, Reactor Environment. Work of this committee is identical with that of the ASA N6-1 Subcommittee.

ANS Project 3, Reactor Operator Qualifications. Information on operator qualifications is in the preliminary draft stage.

ANS Project 4, Reactor Operation. A draft of "Proposed Standards for Operation of Research and Test Reactors — Organization and Procedure" was published in Nuclear News, January 1960.

ANS Project 5, Methods of Estimating Energy and Fission Products Release. Work is identical with that of ASA Subcommittee N6-7.

ANS Project 6, Reactor Dynamic System Design. No progress was reported.

ANS Project 7, Reactor Components. The ANS Standards Committee approved the "Proposed Standard for Plate-Type Uranium-Aluminum Fuel Elements" on June 5, 1961, and it was published in Nuclear News, December 1961. Comments were requested. Various standards in the draft stage include: (1) "Quality Control and Inspection of Pool-Type Fuel Elements," (2) "Leakage Rate Testing of Containment Structures," (3) "Earthquake Design Criteria for Nuclear Applications," (4) "Control and Safety Rod System Design Criteria," (5) "Welding Quality for Aluminum and its Alloys in Nuclear Systems," (6) "Nuclear Pool Facilities," and (7) "Cleaning Nuclear System Components." Future activities are planned in the areas of (1) reliability of valve bodies and pump casings in primary system and (2) safety factors in the design of nuclear reactor components.

ANS Project 8, Fissionable Material Outside Reactors. Activities are identical with those of ASA Subcommittee N6-8.

ANS Project 9, Nuclear Glossary Subcommittee. Work is identical with that of ASA Subcommittee N2-4.

ANS Project 10, Units in the Nuclear Industry. Work on the development of an acceptable system of metric units is in the development stage. Future effort is planned on a booklet of conversion and explanatory tables for science and engineering students.

IV

American Society of Civil Engineers, The Administrative Committee on Nuclear Structures and Materials.

Task Committee on Hot Laboratories. The "Hot Laboratory Design and Criteria Manual" is in final draft form.

Task Committee on Structural Aspects of Nuclear Insurance. "Regulations Governing the Design and Construction of Nuclear Facilities" was published in 1960, and copies are available from Gibson Morris, Oak Ridge National Laboratory, Oak Ridge, Tennessee. This committee has been disbanded.

Task Committee on Structural Aspects of Nuclear Incidents. Committee has been recently formed.

Task Committee on Structural Materials in Reactor Design. "Digest of Nuclear Shielding," "Radiation Effects on Adhesives," and "Design in Steel for Nuclear Reactors" are in the preliminary draft stage.

Future activity is planned on "Selective Placement of Aggregates in Reactor Shield."

Task Committee on Live Loads Due to Heat and Incidence. This committee is planning to develop a bibliography of information on live loads due to heat and incidence.

V

American Society of Mechanical Engineers, Committee on Reactor Safety Standards.

Special Committee on Nuclear Power. Cases 1270N through 1276N have been adopted by the committee and work has begun on the preparation of a section of the Boiler Code to cover vessels in nuclear service.

Special Committee to Review Code Stress Basis. This group is working with the special committee on Nuclear Power in preparing the previously indicated section of the Boiler Code.

Research Committee on Effects of Radiation on Materials. Information is still in the development stage.

VI

American Society for Testing Materials, Special Administrative Committee on Nuclear Problems. A number of specifications pertinent to nuclear standards have been written by the various ASTM committees. The following specifications were approved for publication at the June 28, 1961, ASTM annual meeting. Information as to their availability can be obtained from J. W. Caum, Secretary, American Society for Testing

Materials, 1916 Race Street, Philadelphia 3, Pa. "Tentative Method for Measurement of Gamma Radioactivity of Industrial Water and Industrial Waste Water" (D 1690)

"Tentative Method of Measurement of Beta Particle Radioactivity of Industrial Water and Industrial Waste Water" (D 1890)

"Tentative Methods for Analysis of Zirconium and Zirconium-Base Alloys" (E 146)

"Tentative Methods of Liquid Penetrant Inspection" (E 165)

"Tentative Method of Ultrasonic Contact Inspection of Weldment" (E 164)

"Tentative Methods of Controlling Quality of Radiographic Testing" (E 142)

"Tentative Recommended Practices for Determining Changes in the Chemical Reactivity of Inorganic Materials Exposed to High Energy Radiation" (E 183-61T)

"Tentative Recommended Practice for Effect of High-Energy Radiation on the Tensile and Impact Properties of Metallic Materials" (E 184-61T)

"Tentative Recommended Practice for Surveillance Tests on Structural Materials in Nuclear Reactors" (E 186-61T)

VII

Atomic Industrial Forum, Committee on Codes and Standards. No activities reported.

VIII

Institute of Radio Engineers, Committee 13 on Nuclear Techniques. "IRE Standards on Nuclear Techniques: Definitions for the Scintillation

Counter Field, 1960" 60 IRE 13.S1, has been published and copies may be obtained from Institute of Radio Engineers, 1 East 79th Street, New York 21, New York.

IX

National Electric Manufacturers Association. No direct production of standards is attempted; however, work is performed through other organizations, e.g., ASA Subcommittee N4-1.

X

U. S. Department of Commerce, National Bureau of Standards. The following NBS handbooks have been issued since January 1960. These handbooks may be purchased from the Superintendent of Documents, U. S. Government Printing Office, Washington 25, D. C. Fifty-one main committee members and 18 subcommittees are listed with their respective chairman in NBS Handbook 79, issued on September 1, 1961.

72 "Measurement of Neutron Flux and Spectra for Physical and Biological Applications," issued July 1960.

73 "Protection against Radiations from Sealed Gamma Sources," issued July 1960.

75 "Measurement of Absorbed Dose of Neutrons and of Mixtures of Neutrons and Gamma Rays," issued February 1961.

76 "Medical X-Ray Protection up to Three Million Volts," issued February 1961.

78 "Report of the International Commission on Radiological Units and Measurements," issued January 1961.

79 "Stopping Powers for Use with Cavity Chambers," issued September 1961.

80 "A Manual on Radioactivity Procedures," in press.

XI

Society of Naval Architects and Marine Engineers, Ships Machinery Committee, M-13 Panel. No progress was reported.

XII

United States Committee for the Revision of the International Convention for the Safety of Life at Sea, 1948, Nuclear Power Committee. No progress was reported.

XIII

Electronic Industries Association. No direct standards work is being done and the committees have been dissolved.

XIV

International Commission on Radiological Protection. The Commission's Recommendations are published by Pergamon Press Ltd., Headington Hill Hall, Oxford, England. The Recommendations of Committees I, II, and III from the 1959 Meeting have been published in ICRP Publications 1, 2, and 3, respectively. Reports of Committees IV, V, and VI are being prepared.

XV

American Institute of Electrical Engineers, Nucleonics Committee. Standards and Safety Subcommittee. "Proposed Code for Qualifications of Nuclear Reactor and Accelerator Operators Relating to Nuclear Safety" is in the draft stage. Considerable work was performed on the draft standard developed by ASA Subcommittee N6-6.

XVI

National Fire Protection Association, Committee on Atomic Energy. Booklet No. 802, "Recommended Fire Protection Practice for Nuclear Reactors," was published in May 1960 and copies are available at a charge of $0.75 from National Fire Protection Association, 60 Batterymarch Street, Boston 10, Mass.

## IAEA MEETING ON REACTOR SITING

A panel meeting on reactor siting sponsored jointly by the International Atomic Energy Agency (IAEA) and the International Organization for Standardization (ISO) was held October 30 to November 7, 1961, at IAEA headquarters in Vienna. The subject of reactor siting has been under consideration by a working group of the ISO for some time. At the last meeting of the group in March 1961, the need for a larger discussion under joint IAEA-ISO auspices was recognized in that there was a desire to consider whether it was feasible to make available some general guiding principles regarding reactor siting. About 30 experts from 16 countries participated, with C. K. Beck of the United States Atomic Energy Commission as chairman. Beck is also chairman of the ISO working group dealing with reactor siting. Representatives of the various countries discussed site criteria in their respective countries and also presented data on specific facets of the over-all program.

### General Philosophy of Site Selection

The site selection criteria of a number of countries represented at the meeting were presented. The several approaches listed below differ widely in many respects, and the discussions at the meeting were unable to resolve the major differences. The essence of the approaches presented is discussed below. A comparison of the major points in the British, Canadian, and American criteria is presented in Table I-3.

United Kingdom. Site selection criteria in the United Kingdom, as reviewed by F. R. Farmer at the meeting, were based initially on the work of Marley and Fry,[3] but some changes in policy have been made, particularly

Table I-3. COMPARISON OF REACTOR SITING CRITERIA IN THE UNITED KINGDOM, CANADA, AND THE UNITED STATES

| Country | Reactor Thermal Power (Mw) | Exclusion Area Radius (miles) | Controlled Area Radius (miles) | Population Within Controlled Area |
|---|---|---|---|---|
| United Kingdom | 200–800 | (a) | 1.5 | 500 } $N \propto R^2$ |
| | | | 2.1 | 1,000 |
| | | | 5.0 | 10,000 |
| United States | 200 | 0.21 | 2.2[b] | See notes (b) and (c) |
| | 600 | 0.38 | 7.2[b] | |
| | 1000 | 0.53 | 10.0[b] | |
| | 200 | 0.21 | 4.5[c] | |
| | 600 | 0.38 | 9.6[c] | |
| | 1000 | 0.53 | 13.3[c] | |
| Canada | 1000 | 0.5 | 2.4 | 1,000 } $N \propto R^3$ |
| | | | 5.0 | 10,000 |

[a]Not specified; included in controlled area.

[b]Distance to low population zone. "Low population zone" means the area immediately surrounding the exclusion area that contains residents, the total number and density of which are such that there is a reasonable probability that appropriate measures could be taken in their behalf in the event of a serious accident.

[c]Distance to population center. "Distance to population center" means the distance from the reactor to the nearest boundary of a densely populated center containing more than about 25,000 residents.

in relation to the airborne iodine concentration. The criteria are now based on an expected 25% release of radioiodine, a weather condition consisting of a mild temperature inversion (Pasquill's category "F," see Table I-6, p. 47, this issue) with a frequency of 10 to 20%, a large dispersion angle (30 deg), and a time of release duration of 4 to 6 hr. Permissible exposures are 25 rem whole body and 25 rem to a child's thyroid or 150 rem to an adult's thyroid. The effects of a release of fission products are divided according to source: (1) inhalation (short-period exposure only), (2) consumption of milk, and (3) deposition on the ground.

The expectation of exposure is based on a power law with a weighting factor equal to the square of the iodine concentration multiplied by the number of people in the area. Sites may be compared in this way. Farmer gave one set of conditions for a release of 2500 curies of radioiodine in fair weather in which all persons within 1500 yd in one sector would be evacuated temporarily, and a ban would be placed on the use of exposed crops within a 1-mile radius and on milk produced within 14 miles. He further commented that the available methods of site evaluation give results that vary by a factor of 10 and that if the expected release could be reduced to 250 curies of iodine, the reactor could be set up anywhere in the United Kingdom. Thus the use of exclusion areas per se is avoided, but the possibility of a population increase in adjacent areas must be carefully assessed and excessive housing development must be controlled. A new paper on criteria for siting in the United Kingdom is being prepared, and a preliminary report has been issued by Farmer and Fletcher.[4]

Germany. E. Stauber presented two papers at the IAEA meeting:*

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*Proceedings not yet published.

"Public Safety and Expected Release of Radioactive Materials from Nuclear Reactor Plants" and "Procedure for Reactor Plant Site Selection Based on Nuclear Aspects." Although the studies involved a small power plant (60 Mw thermal) of the pressurized-water type, the limiting parameters were 25-rem total body exposure and 25-rem exposure to critical organs (e.g., thyroid) within an exposure period of 50 years; in this respect Germany's criteria generally follow the United Kingdom's criteria.

In Germany the concept seems to be based on taking complete credit for containment and making conservative allowances for the other factors (i.e., 100% instantaneous release to the containment structure). Germany seems to wish, however, to avoid the concept of exclusion areas even more than the United Kingdom. W. Shröck-Vietor reviewed, at the IAEA meeting, the mechanism of site appraisal in Germany and pointed out that, as yet, no principles have been established in Germany regarding site selection. He also brought to the discussion the following two questions which were not entirely resolved:

1. The United States Atomic Energy Commission proposes in its "Reactor Site Criteria" (issued for review and comments on February 11, 1961)[5] a total permissible dose in case of an accident in excess of 300-rem iodine exposure to the thyroid in constrast to the United Kingdom value reported by Farmer and Fletcher[4] of only 25-rem exposure to the thyroid of children. Which of these opinions is preferred?

2. The ICRP recommendations give values for the maximum permissible concentrations of radioactivity in air for special groups in the vicinity of controlled areas (1/10 MPC for occupational exposures). What time is

reasonable in which to average the radioactive concentration in order to not exceed the allowable exposure?

Japan. The siting of reactors in Japan was discussed at the meeting by Akira Oyama. Japan has based its siting criteria on a containment-with-exclusion-area principle in spite of its population density. Japan has the additional problem of requiring earthquake consideration in the design of containment structures and piping. The Japanese approach resembles the German one in that complete credit for containment is used to ensure that people living outside the exclusion area will not have to be evacuated in the event of the maximum credible accident. Under conditions described as being "more severe than the maximum credible accident," evacuation would be resorted to.

Of interest was a discussion by Oyama on the method of application of probability analysis to reactor safety. According to Oyama a study is being made of component performance reliability, equipment-failure causes of accidents, and the interrelations among facilities in connection with the cause of primary accidents. While this is not as useful as the normal statistical-experience type of analysis, some improvement may result in a type of safety protection and backup system to be added to a facility to decrease the probability of an accident and the extent of the hazards.

Canada. In his review of reactor siting in Canada, G. C. Laurence stated at the meeting that there are no rules or regulations at present governing the location of a reactor. Canada has postponed the writing of any code because it would soon become obsolete. In the actual siting of reactors, however, the isolation policy is now supplemented by the use of a series of siting formulas which classify a site according to its

population. The following excerpt from the report[6] prepared for the Vienna Conference by Laurence is pertinent.

> "Approval of the location for a nuclear reactor in Canada depends on correlation of all safety aspects of its design and operation. Therefore it is not subject to a general code of regulations. In comparing sites, curves are drawn to show the number of persons residing within the circle of radius r about the reactor. Comparison is made with a reference population index which is acceptable for a 1000 MWT power reactor that is constructed and operated according to the principles of safe practice that are normally expected in Canada. In comparing sites in respect to proximity to population, it is suggested that population density be weighed in inverse proportion to the cube of the distance from the reactor. Integration of the weighted population density over the countryside beyond the plant's boundary or exclusion area gives a population index for comparison with that of other sites."

The reference population index referred to by Laurence is given in Fig. 1. He implied that reluctance to make the required assumptions regarding weather and activity release has resulted in preference for the population-index method. Canada had formerly used the following siting formula based on probability:

$$FUV < 10^{-5}/\text{yr for a } Q \text{ of } 10^{5} \text{ curies} \quad ,$$

where

F = frequency of failure,

U = unreliability of a device,

V = unreliability of containment,

Q = quantity of controlling volatile isotope, $I^{131}$.

The inability to obtain meaningful numbers for the terms in this expression has made it of little value.

France. M. Bourgeois related that in hazards analyses of French reactors two types of accident are considered: (1) the technical accident

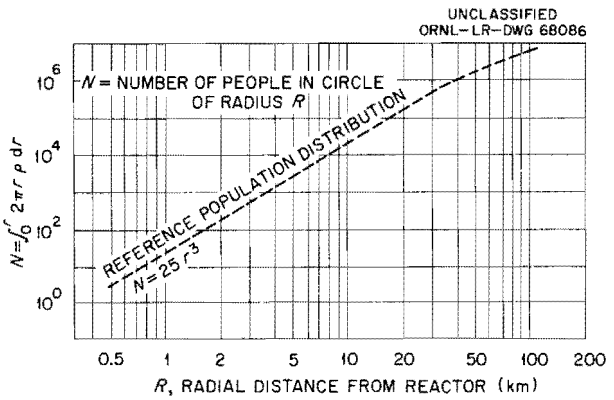


Fig. 1. Allowable Total Population in an Area of Given Radius Around a 1000-Mw (thermal) Reactor in Canada.

and (2) the maximum hypothetical (credible) accident. Although it is not known how the second would occur, the concept is similar to that used in the United States and the calculations closely resemble those used in the United Kingdom.

Czeckoslovakia, USSR. A. Ševčik described a concept of advanced reactor engineering through which sufficiently great reliability would be obtained in the primary system so that containment would not be necessary. He stated that irradiation and mockup tests had demonstrated much greater reliability than that achieved in previous practice, but he did not claim "absolute safety," since absolute is an asymptotic concept. He reported that significant advances had been made in terms of thickness of the shell, welding, and heat treatment. Metallurgical tests of highly irradiated specimens of pressure vessel material failed to indicate changes in properties that might suggest embrittlement. In answer to a question as to whether this applied only to the pressure shell, Sevcik's answer was that they foresee no need for a containment shell.

United States. Tentative reactor site criteria were published last year in the Federal Register[4] for comments, as mentioned above. C. K. Beck summarized the philosophy behind these criteria in the following terms:

> "The hazards of normal operation do not limit the location of a reactor since this is subject to complete containment. Accidents alone seem to be the limiting factor with the probability of accidents virtually nonexistent. No reactor with a likelihood of having an accident in a short while would ever be built. Since in dealing with probability, one cannot distinguish between 1:1000 and 1:1,000,000 this factor is now on no legitimate basis. This leaves the other question — that of the magnitude of an accident. We must recognize that in human events accidents will occur. Then we have two recourses:

1. Isolation
2. Built-in Safeguards, e.g.
   a. Containment as in a sphere
   b. Pressure release through absolute filtration
   c. Spray down
   d. Immersion in water

"An analysis shows that we generally reach a compromise between these two. Therefore a reactor may be built anywhere but in order to protect ourselves we finally choose a site somewhat isolated but not remote. The degree of dependence on safeguards will depend on many things such as (1) material need, (2) national defense, (3) economic needs. Thus one country may reach a different compromise than another country. The U.S. early this year published a set of guides [5] considering separately such factors as (1) type, (2) uses, (3) safeguards, (4) unusual features, (5) hydrology, meteorology, seismology. Permissible radiation dosages were based on (1) a tentative target goal of conservative estimates for calculations and (2) that necessary in case of an emergency. The area around the reactor was divided into zones.

1. An exclusion area where people do not live,

2. An area of low population where protective measures may be taken, and

3. A large area which would be affected only in a substantially larger accident. This might include a population center which would not be evacuated. Examples are given in the guide. The door was left open for review but perhaps not wide enough."

Supporting Experimental Data

The experimental data available as a basis for developing site criteria or, as the case may be, to check the criteria against, include data on activity releases and meteorological studies. The meteorological data have devolved from a science that predates the development of nuclear energy, and they are used extensively in nuclear safety analyses. Data on activity release have been developed from both accidental and intentional releases.

Beattie of the United Kingdom presented copies[7] of a detailed analysis of environmental hazards based on studies of the Windscale accident. From

the analysis it has been possible to assess the prospect of future accidents at normal-uranium gas-cooled power reactor stations. In much detail, Beattie has eliminated almost all assumptions regarding dispersion, fall-out, release ratios, particle sizes, etc., and the exposure values given in Table I-4 have been tediously worked out. Iodine is by far the controlling activity, while tellurium ranks second. Strontium is relatively unimportant. In view of the controlling nature of iodine releases, Beattie also presented a tabulation (see Table I-5) of the maximum permissible cloud dosages of fission-product iodine. In a previous article in Nuclear Safety,[8] the reviewer compared the final average assessment of release values with those postulated by ORNL from laboratory experiments and indicated that the actual releases of iodine and tellurium were about a factor of 3 lower than predicted.

Data on the release of fission products from reactor-grade $UO_2$ were presented at the IAEA meeting by the reviewer. A comparison was made of the extent of fission-product release from $UO_2$-containing reactor fuels by the following mechanisms which might be operative in loss-of-coolant accidents under various conditions: oxidation in air, high-temperature diffusion in the solid, and melting in air, helium, or $CO_2$. Release experiments that demonstrated the effect of irradiation level or burnup were described. The expected pattern was observed; that is, fission-product release increased with irradiation level.

The dispersion of activity in the atmosphere was the subject of a number of papers because of its importance to (1) site selection, (2) routine operation, and (3) accidental releases. Gifford of the United States in discussing present ability to describe atmospheric dispersal

Table I-4. RELEASE PERCENTAGES RECOMMENDED FOR FUTURE ASSESSMENTS BASED ON WINDSCALE ACCIDENT RELEASE

| Element or Isotope | Activity in Damaged Fuel (curies) | Activity Released from Chimney (curies) | Percentage Released from Chimney | Percentage Release Recommended for Future Assessments |
|---|---|---|---|---|
| Krypton | | | | 60 |
| Tellurium-132 | 160,000 | | 7.5 | 15 |
| Iodine-131 | 168,000 | 20,000 | 12 | 25 |
| Strontium-89 | 380,000 | 80 | 0.02 | 0.1 |
| Strontium-90 | 9,200 | 2 | 0.02 | 0.1 |
| Ruthenium-106 | 12,000 | 80 | 0.7 | 4 |
| Cesium-137 | 8,000 | 600 | 7.5 | 15 |
| Cerium-144 | 290,000 | 80 | 0.03 | 0.15 |
| Barium-140 | | | | 0.2 |

Table I-5. MAXIMUM PERMISSIBLE CLOUD DOSAGE OF FISSION-PRODUCT IODINE

| Age of Exposed Person | Maximum Permissible Cloud Dosage (curie·sec/$m^3$ of $I^{131}$) |
|---|---|
| 6 months | 0.007 |
| 3 years | 0.007 |
| 10 years | 0.011 |
| 20 years and over | 0.031 |

problems said that after 15 years of meteorological study, the reactor designer now has recourse to fairly complete dispersion data for distances up to 1 mile. At greater distances this is not so. Recently, the United States Weather Bureau carried out tests at a distance of 1.6 miles, and at Hanford tests have been run at a distance of 16 miles; however, little can be said about dispersion at large distances. Without a new and better

atmospheric tracer technique, no major improvements in diffusion theory can be expected.

The stability categories of the atmosphere were discussed by Pasquill, including the category "G," which was not included in his original classification but has since been added (see Table I-6). The United Kingdom employs the characteristics of this category in their analysis, while the United States employs characteristics representative of category E, i.e., one that represents the worst 20% of the weather.

Table I-6. PASQUILL'S STABILITY CATEGORIES OF THE ATMOSPHERE

| Stability Category | Typical Wind Speed (m/sec) | Description | Frequency* (%) |
|---|---|---|---|
| A | 1 | Very sunny summer weather | 1.8 |
| B | 2 | Sunny | 8.6 |
| C | 5 | Average day | 16.6 |
| D | 5 | Overcast day or night | 38.0 |
| E | 3 | Average night | 12.4 |
| F | 2 | Clear night | 14.3 |
| G | | Cool night with heavy dew | 8.1** |

*At Croydon, England.

**Worst condition (from ref. 7).

## Conclusions

The IAEA prepared and presented an outline for a proposal that might result in a handbook. Reaction was quite unfavorable to this amount of standardization, and the delegates could only agree not to agree for the time being. The attitudes of the representatives were quite varied. Laurence emphasized the "great mass of general agreement" that was reached even though there was little agreement in detail. Joslin of the United

Kingdom and others cautioned concerning the political, as opposed to technical, factors in reactor siting. Beck, chairman of the meeting, concluded, "I think we have to admit that we have to balance need with economics and local conditions. On this basis we may safely proceed to produce a record of the meeting and a useful editorial comment to explain the diversity of the problem."

Accordingly, Mr. Balligand, Deputy Director of IAEA, outlined the final proposal which was accepted:

(1) IAEA would prepare a draft of the proceedings.

(2) An advisory group of three each from IAEA and ISO would be appointed to furnish final editorial comment.

(3) This editorial comment would be sent to all countries for approval by one delegate.

(4) Publication would be deferred until the April (May) meeting at Vienna on reactor safety, since the presence of the advisory group might be necessary. (G. W. Parker)

## SAFETY OF UNIVERSITY REACTORS

At the present time there are 36 university reactors in operation in the United States, an additional 12 are under construction, and four more are in the planning stage. This particular group of reactors is defined by the AEC in the following manner:[9] "Teaching Reactor. Any reactor operated for the primary purpose of training in the operation and utilization of reactors and for instruction in reactor theory and performance. Thermal power level is usually limited to 10 kw."

Thirty-five of the reactors now operating were completed during the period from 1957 to 1960. The acquisition of this relatively large number of reactors by colleges and universities was the result of several factors. The main incentive was the desire of the AEC for colleges and universities to supply trained personnel for the nuclear industry. Sapirie of the Oak Ridge Operations Office expressed the opinion of the AEC in 1956 in an opening address to the Conference on Engineering Education and Nuclear Energy:[10] "The Commission is striving — to the extent that our facilities and personnel permit — to deal with the shortage of nuclear specialists. But a program of correction of a much broader and more far-reaching nature is required." The solution to the problem was also given by Sapirie: "In any long-range planning to solve these basic problems, we must look to the colleges and universities, such as those you represent, for our supply of future nuclear specialists. There is no other practical source."

The initial incentive was enhanced by a second more tangible one. This had to do with the program of financial support offered colleges and universities by the AEC for training personnel for the nuclear industry.

This support consisted of a waiver of use charges on special nuclear materials, fuel, fuel element fabrication, processing of fuel solutions and fuel elements, and neutron sources and financial assistance for the purchase of radiation-detection monitoring and counting instruments, reactor simulators, and equipment for instruction in fields closely allied to reactor technology. Naturally, many schools undertook to establish a graduate program in nuclear energy, and with much of the auxiliary equipment obtainable through AEC grants, some felt the expenditure for a training reactor justifiable.

General Safety Problems

Operation of a reactor on a university campus of course requires USAEC permission. The attitude of the Advisory Committee on Reactor Safeguards toward such an installation was initially one of caution.[11] The manner of evaluating a university reactor was not definitely stated, but the hazards were reviewed to assure the protection of personnel and the public. As in the review of any reactor, the ACRS wanted to be assured that there was a low probability of injury to the public, the smallest possible hazards to the students and the teaching staff who utilize the facility, and sufficient liability coverage by the facility owner in the unlikely event that an accident occurs. The two situations that constituted the main hazards problems were considered to be the possibility of a nuclear runaway and the release of fission products contained in the fuel elements. These two problems have been reduced by appropriate facility design.

Since a relatively large number of teaching reactors are now in operation, it is of interest to examine the current problems. The safety

problems mentioned above have been adequately coped with, and teaching reactors are readily available. There are, however, some remaining problems with respect to the regulation of these facilities. The fact that some safety problems do not originate within the reactor system was brought out by Dr. Clifford K. Beck at the 1960 University Reactor Conference, when he said,[12] "In many ways the inherent nature, organization, and operation of a university are incompatible with the safety requirements of reactor programs." It must be made clear that this is not an indictment of university administrators or those responsible for the reactor programs. It is simply a realistic evaluation of the very difficult problem of applying in an academic atmosphere the stringent disciplines necessary for reactor operation.

In order to illustrate the basic problems, Dr. Beck presented several points for consideration. In many areas, the problems of finance are important. The main difficulty here arises from the allocation of insufficient funds for a truly adequate staff who can devote full time to reactor duties, for maintenance of the facility, and for auxiliary facilities. With the installation of a teaching reactor it is often financially more feasible to reassign work loads within an existing staff than to create a new entity. Sometimes the maintenance of a facility is not carried out as thoroughly as desirable. Since the reactor is usually a self-contained unit, the periodic safety evaluation of the facility may depend on chemical analysis of reactor offgas or coolant. Equipment and procedures for such investigations are not always adequate. Unlike the operation of an industrial reactor, which is based on well-defined lines of authority and is accomplished

by a large staff with specific duties, the operation of the university reactor may be subordinate to other responsibilities and duties. The academic atmosphere does not always lend itself to the required check and doublecheck type of operation. The review and re-review of proposed experimental work in an expeditious manner is often difficult because of the diversity of the duties of the university staff. These problems do not all exist for all university reactors, but most facilities have at least some of these difficulties.

## Safety Features of Specific Reactors

As previously mentioned, teaching reactors of the type desired for university work are now standard items, and their designs incorporate many safety features. Fail-safe, automatic shutdown, low fission-product build-up, negative temperature coefficients, and negative void coefficients are a few of the more often used terms that describe the low hazard level associated with these reactors.

The AEC list of teaching reactors[9] indicates that numerous companies produced the reactors now operating.

One of the first research reactors designed, the Argonne National Laboratory Argonaut, has served as a model for several commercial systems. As stated in the hazards summary report:[13]

> "The design anticipates use of an Argonaut reactor in universities and in other less specialized institutions. Therefore, safety is a prime design feature through choice of a self-limiting system."

The reactor and its basic safety aspects are described in the hazards report as follows:

"Argonaut is a 10-kw (max) thermal reactor moderated by water and reflected by graphite. Plate-type fuel elements are spaced with graphite wedges to form an annular core. The total fissionable material required is about 4 kg $U^{235}$. The experimentally determined void coefficient for the lattice is negative: $-0.25\%$ $\Delta k/k$ per % void. The temperature coefficient is also negative: $-10^{-4}$ $\Delta k/k/°C$. Excess k is less than 0.75%. Rapid insertion of up to 4.75% $k_{ex}$, possible only by deliberate circumvection of procedures, interlocks, trips, and controls, will result in automatic shutdown through a nondestructive BORAX-type process."

The steps necessary to bring about the 4.75% k insertion in the reactor are deliberate and involve some direct physical actions, all of which are contrary to operational procedures. It therefore seems quite improbable that such an insertion can occur, and, even if it does, the system should be able to control the consequences.

A commercially available 10-kw university training reactor (UTR-10) similar to the Argonaut is built by the Advanced Technology Laboratories of the American-Radiator and Standard Sanitary Corporation.[14] This reactor has the inherent safety features of very low built-in excess reactivity (0.5% $k_{eff}$), a negative temperature coefficient [$-0.00683\%$ $(\Delta k/k)/°C$], and a negative void coefficient [$-0.172\%$ $(\Delta k/k)$ per % void]. The control and safety system of the reactor is designed to fail safe on power loss or interruption of control signal. The maximum credible accident introduces all of the excess reactivity (0.5% $k_{eff}$) into the reactor upon failure of the automatic safety system. The analysis of the accident indicates that melting of the fuel plates will not occur. The dose rate at the shield surface as a result of the accident would be approximately 12 mr/hr with the reactor at the stabilized power level established after the initial transient of 164 kw. Other companies manufacturing reactors of this type are General Nuclear Engineering Corporation, Nucledyne Company, and AMF Atomics, Inc.

The Triga Mark II teaching reactor manufactured by the General Atomic Division of General Dynamics Corporation, although different in design, utilizes many of the safety features previously mentioned. The core is made up of solid, homogeneous fuel elements having 20%-enriched uranium combined with zirconium hydride. The fuel elements are submerged in a pool of light water which is used for cooling and shielding.[15] The fuel elements exhibit a prompt negative temperature coefficient of reactivity of $-1.3 \times 10^{-2}\%$ $(\Delta k/k)/°C$. The core void coefficient is $-0.1\%$ $(\Delta k/k)$ per % void, and the initial excess reactivity is 1.6%. The self-limiting characteristics of the fuel keep the reactor in a safe mode even under the maximum reactivity accident postulated as credible. Even reactivity accidents of slightly greater magnitude than those deemed credible were also found to be terminated by the fuel characteristics.

The General Electric Company university reactor is also a pool type that operates at 10 kw.[16] Light water is used for both shielding and cooling. MTR-type fuel elements having a total of 3.4 kg of fully enriched $U^{235}$ give the reactor an initial excess reactivity of 0.5%. The reactor exhibits negative temperature and void coefficients of $-0.5 \times 10^{-2}\%$ $(\Delta k/k)/°C$ and $-2 \times 10^{-1}\%$ $(\Delta k/k)$ per % void, respectively. Several postulated maximum accidents result in possible fuel element failure. No credit was taken for the negative temperature coefficient in evaluating these accidents, and yet the results of the accident were not hazardous to the public. Pool-type reactors are also manufactured by Bendix Aviation Corporation and Lockheed Aircraft Corporation.

On the basis of the number of operating reactors, the Aerojet-General Nucleonics Model AGN-201 appears to be the most popular. This reactor

uses 600 g of $U^{235}$ in the form of 20%-enriched $UO_2$ powder homogeneously dispersed in disks of polyethylene moderator.[17] The core has a graphite reflector. The normal power level is 100 milliwatts, but with the special provisions, the power can be boosted to 3 watts. Operation at 100 milliwatts gives a very small fission-product inventory with a fuel burnup of only 3 μg/yr. The temperature coefficient of reactivity is $-3.6 \times 10^{-2}\%$ $(\Delta k/k)/°C$.

Three concentric aluminum tanks form the containment structure for the system. The inner (core) tank is designed to retain any fission-gas release from the polyethylene. This tank is actually in two sections, top and bottom, each containing half of the core plus top and bottom reflectors, respectively. A nuclear fuse at the reactor midplane would melt in the event of an accidental power surge and allow the lower half of the core to drop to the bottom of the tank and render the system subcritical. An unusual feature of the system is that radiation-stabilized polyethylene has a higher melting point than unirradiated polyethylene and can therefore be depended upon to retain its shape during a transient.

The next two tanks are provided for shielding and containment. The second (reactor) tank, besides containing the lead shielding for the reactor, serves as a secondary container for the core tank. The third tank contains borated water for shielding.

The safety of the system is based on the nuclear fuse and on gravity- and spring-actuated safety rods. However, the referenced Nucleonics article states:

> "If all the fission products created in the greatest conceivable accident were liberated in a room of $10^6$-liter capacity, estimates indicate that the level of the most dangerous isotope I-131 would be just barely at a dangerous level."

Aerojet-General Nucleonics also produces the AGN-211 university reactor, which is very similar to the AGN-201 model in that a plastic fuel element is used.[18] The operating power level is given as 1 w, and the core is operated in a pool to increase its usefulness as a teaching reactor.

The fuel elements are composed of four sections: top and bottom graphite reflectors and two central sections of polyethylene. The fuel is $UO_2$ of the same 20% enrichment as the fuel of the AGN-201 model. There are 16 elements in the 1-ft by 1-ft by 1-ft core.

The prompt negative temperature coefficient $[-3.6 \times 10^{-2}\% \ (\Delta k/k)/°C]$ shuts down the system in the event of a power excursion. Also, since the power level is only 1 w, the equilibrium fission-product inventory will not exceed 10 curies, of which 3 curies is associated with gaseous fission products. The rate of insertion of control and safety rods is limited to a maximum rate of reactivity increase of $2.5 \times 10^{-4}$ $sec^{-1}$. The system excess reactivity is also small, i.e., 0.2%.

## Conclusion

The safety of university reactors seems to be well established on the basis of the inherent safety features and the low power levels. A summary of the various system parameters is given in Table I-7. The fail-safe control features which complement the automatic shutdown inherent in most of the systems have been widely applied. These features include the negative temperature and void coefficients which contribute to the over-all safety. The power levels at which most of the plants operate, 10 kw or less, limit the inventory of fission products so that an accident could not release large quantities of radioactive contaminants. (T. H. Row)

Table I-7. CHARACTERISTICS OF SEVERAL COMMERICALLY AVAILABLE TEACHING REACTORS

| Reactor | Type | Excess Reactivity (%) | Negative Temperature Coefficient [% (Δk/k)/°C] | Negative Void Coefficient [% (Δk/k) per % void] | Power Level (w) |
|---|---|---|---|---|---|
| Argonaut | Water-moderated, graphite-reflected core with plate-type fuel elements | 0.75 | $1 \times 10^{-2}$ | $2.5 \times 10^{-1}$ | $10^{4}$ |
| UTR-10 | Water-moderated, graphite-reflected core with plate-type fuel elements | 0.5 | $6.83 \times 10^{-3}$ | $1.72 \times 10^{-1}$ | $10^{4}$ |
| Triga Mark II | Solid, homogeneous, 20%-enriched uranium—zirconium hydride fuel elements operating in a pool of light water | 1.6 | $1.3 \times 10^{-2}$ | $1 \times 10^{-1}$ | $10^{5}$ |
| GE University Reactor | Fully enriched, MTR-type fuel elements operating in a pool of light water | 0.5 | $0.5 \times 10^{-2}$ | $2 \times 10^{-1}$ | 10 |
| AGN-201 | Graphite-reflected core with fuel elements consisting of 20%-enriched $UO_2$ powder homogeneously dispersed in disks of polyethylene moderator | | $3.6 \times 10^{-2}$ | | $10^{-1}$ |
| AGN-211 | Core similar to that of AGN 201 but operated in a pool | 0.2 | $3.6 \times 10^{-2}$ | | 1 |

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# II. ACCIDENT ANALYSIS

# EQUIPMENT RELIABILITY FOR NUCLEAR POWER PLANTS

A casual perusal of almost any nuclear power plant design report or conceptual design description will disclose a number of statements concerning the reliability of plant components, subsystems, and major operating systems. Normally one of the stated objectives of the design is that a high degree of reliability be achieved for military, economic, or safety reasons. All too often, however, it is not clear exactly what is meant by reliability and what specifically has been done in the design to obtain it. It is particularly important in the design of nuclear plants that the design engineer understand the significance of the term "reliability" and that he be aware of the methodology available for evaluating the design. It is the purpose here to discuss reliability, its meaning, its background, and the necessity for its consideration in the design of nuclear plants.

A reliable component is generally defined as one "suitable or fit to be relied on; worthy of dependence; trustworthy." This definition proves insufficient to the engineer, however, for he is well aware that absolute reliability is an ideal that cannot be achieved. A concise engineering definition of reliability has been given as "the probability that a device will function within its design parameters for a given life."[1] The key word "probability" is included in recognition of the fact that the possibility of failure cannot be eliminated.

Interest in the achievement of highly reliable systems has accelerated in the past 10 years, influenced to a great degree by the demands of the nuclear energy and aerospace industries. In nuclear reactor plants interest

in highly reliable systems has been particularly motivated by the following considerations:

1. The effect of reliability on the evaluation and acceptability of control and safety equipment.

2. The difficulties of maintenance and repair of equipment contaminated with radioactivity.

3. Interest in operation of attended or unattended remote power plants where repair and replacement of reactor components would be sc difficult that they might seriously affect the feasibility of the concept.

4. Interest in operation of unattended reactor power plants, such as auxiliary power plants for the space satellite programs, where maintenance work would be impossible.

The requirement for high plant reliability is easily understood for each of these cases, but reliability is also important in the over-all economic effect it has on the more conventional nuclear plants; it has a direct effect on both feasibility and continuity of operation. It was clearly pointed out by Rickover, some time ago, that one of the biggest stumbling blocks in the development of economic nuclear power is the unreliability of many of the conventional components of nuclear plants.[2]

## The Need for Reliability

Improvements in components and systems in nuclear plants must be preceded by an established need for higher reliability. In recent years the need for reliability has often been specified by the customer, particularly in the case of military reactors, in terms of required life expectancy.[3,4] In these instances, plants were designed to satisfy the requirements of

remote or unattended operation where system reliability is of obvious importance.

The importance of reliability in reducing reactor maintenance in attended power reactors has been discussed by Keetley[5] and Herriott.[6] Keetley[5] pointed out that the maintenance costs at the NRU and NRX reactors are substantially greater than for a normal power plant because of the time required for such common routines as donning protective clothing, the delay time in obtaining permits and radiation surveys before entering reactor equipment areas, and the cost of the special protective equipment required for working in high radiation fields. Further, in special cases where repair work must be done in high radiation fields, large numbers of maintenance personnel may be required to keep individual dose rates within acceptable limits. Equipment is often crowded into a small space to reduce the amount of shielding required and the resultant access difficulties increase maintenance time.

Herriott[6] discussed the direct effect of reliability on reactor safety. For example, the shutdown rods in the NRX were designed with close rod-to-sleeve clearances to facilitate pneumatic raising of the rods during reactor startup. The close tolerance between rod and outer sheath resulted in a number of instances of failure to descend on scram because of rod jamming or seizing. After several years of unsatisfactory operation, including excessive maintenance, a redesigned shutdown rod was installed in the reactor.

Another example of how the reliability of a system affects the operational safety of a plant was brought to light in connection with operation of the ORNL Graphite Reactor. The X-10 Pile, as it was originally

called, was initially designed with a "last ditch" system to shut down the reactor in the event all other measures failed. This system consisted of a hopper filled with boron steel balls. The hopper door could be operated manually to discharge the balls into the reactor. Because of the complications associated with periodic testing of this equipment, such testing was not carried out. After many years of reactor operation the manual trip for opening the hopper door was unknowingly actuated. Operation of the reactor continued for at least two weeks subsequent to the trip of this emergency shutdown system. It was then discovered that the steel balls had self-welded into a solid mass, and the system was completely inoperable.

This case clearly points up the danger of including in the safety system of a reactor components or systems that cannot be periodically checked or which are of a design that discourages checking because of the difficulty of the test or the time required. Components or systems which can fail without operator knowledge are a hazard to safe reactor operation.

Perhaps the most comprehensive single survey of the reliability of nuclear equipment and components was that undertaken by Gilbert Associates for the Rome Air Development Center of the United States Air Force.[3] The report, issued in July 1960, reviews in detail the operating and maintenance records of five nuclear power plants:

| Reactor | Date Operated at Full Power |
|---|---|
| EBWR | December 1956 |
| PWR | December 1957 |
| SM-1 | April 1957 |
| SL-1 | October 1958 |
| VBWR | October 1957 |

The objective was to compile information on the reliability of equipment and components in nuclear plants. Over 1800 events involving abnormal operation of equipment were reviewed in the survey, and nearly 300 were found to have caused unscheduled losses of load by the plant involved. The events were categorized according to type of equipment, nature of the event, and the consequences. Nuclear and conventional equipment were considered separately within the following definitions: Nuclear equipment was defined as "that which is unique to the detection and handling of radiation or radiation effects or that which is unique to the safety and control of the reactor"; conventional equipment was defined as "that which would serve a normal purpose in the main or auxiliary cycles for generation of electrical energy within a power plant (with the reactor supplanting the normal steam generator)."

Of the 1826 events studied, 1354 occurred in the conventional equipment and 472 in the nuclear equipment. All the events studied are listed in Table II-1 according to type of equipment and the consequences. A plant load loss occurred or immediate maintenance work was required to prevent a probable load loss in 284 or 15.6% of the events. In the conventional and nuclear portions of the plants combined, instrumentation and controls were responsible for 914 of the 1826 events or almost exactly 50%. For nuclear equipment alone, instrumentation and controls accounted for 97.7% of all events and for 100% of all failure incidents or events that required immediate corrective action. These rather striking statistics indicate an area to which efforts in reliability improvement may well be directed. It should be pointed out, however, that the very high percentage of failure events attributable to instrumentation and controls in the

Table II-1. SUMMARY OF DATA ON EQUIPMENT MALOPERATION IN NUCLEAR PLANTS

| Equipment Classification | Total Number of Events Involving Abnormal Equipment Operation | Number of Events That Caused Failures or Required Immediate Unscheduled Corrective Action |
|---|---|---|
| Conventional | | |
| Condenser and auxiliaries | 52 | 9 |
| Heat exchanger and vessels | 65 | 6 |
| Instruments and controls | 453 | 16 |
| Piping and piping accessories | 177 | 19 |
| Pumps and compressors | 141 | 4 |
| Turbine generator and accessories | 85 | 32 |
| Valves | 260 | 14 |
| Electrical auxiliary equipment | 121 | 19 |
| Subtotal | 1354 | 119 |
| Nuclear | | |
| Heat exchanger and vessels | 4 | |
| Instruments and controls | 461 | 165 |
| Pumps and compressors | 1 | |
| Valves | 4 | |
| Electrical auxiliary equipment | 2 | |
| Subtotal | 472 | 165 |
| Total events | 1862 | 284 |

nuclear equipment is in part due to factors other than the instrumentation and controls.

In general the reliability obtained in the operation of certain nuclear equipment (i.e., pumps and heat exchangers) has been substantially better

than that obtained with similar conventional equipment. This emphasizes the frequently heard statement that more reliable equipment could be made if designers and manufacturers expended the necessary effort.[2,3,5]

Other studies have been carried out on the reliability of turbine-generator performance in a conventional power plant.[4,7] These studies show that after an initial run-in period for a conventional turbine-generator unit, only five units out of 100 are capable of operating one year under normal conditions without a forced outage. In these studies the operation of nine units over approximately a 14-year period was examined.

It is important in the evaluation of data on reliability of equipment that a thorough understanding exist as to the limitations of such data. In the Gilbert Associates report,[3] the authors are careful to point out a number of such limitations: (1) the thoroughness of the document records on operation and maintenance, (2) the total operating period over which records are kept, and (3) the design and operational objectives of the plant. The five plants included in the Gilbert Associates survey had been operated for different lengths of time, at different power levels, and for different purposes. These differences would be expected to be reflected in the individual plant statistics.

## Factors Affecting Reliability

The problem of designing a highly reliable reactor system requires a full understanding of the factors which may affect the system reliability. Halio[8] and others[1] have suggested a number of steps in the life cycle of equipment which affect the ultimate reliability: (1) planning, (2) design and development, (3) manufacture, (4) inspection, (5) testing, (6) packaging

and transportation, (7) storage, (8) installation, and (9) operation and maintenance.

The designer is normally faced with the problem of obtaining high "inherent reliability." Inherent reliability is defined as "the potential reliability present in the physical article when manufactured to a completely mature design with no degradation due to fabrication or assembly."[1] Once the design of a device is completed, the inherent reliability is established, and subsequent events can only maintain or degrade, but not improve, the inherent reliability.

The planning stage for achieving high reliability involves the determination of the reliability required. It is somewhat of an anomaly that, although a high degree of reliability for reactor plant components is often desired, the designer is unable to say with assurance just what degree of reliability is required. A few of the prime requisites are that (1) the design should be basically simple, (2) parts should be derated in design stress or load, (3) tests of newly designed equipment should be carried out to obtain life-expectancy data, (4) the number of component parts should be minimized to achieve system simplicity, (5) standard designs that have been proved in practice should be employed wherever practical, but universal parts should not be employed under off-design conditions just be keep the number of part types to a minimum, and (6) redundancy systems or circuits should be employed where feasible to improve system reliability.

The possible effects on reliability of the manufacturing, inspecting, and testing process are apparent. The tendency to telescope design, development, and manufacturing time frequently precludes adequate quality

control and inspection. The importance of proper installation, operation, and maintenance is, however, somewhat more difficult to evaluate, because the performance data that are needed as a basis for evaluation are not always adequately recorded.[3]

Determination and Prediction of System Reliability

Lusser[9] has discussed the singular importance of reliability during operation of complex equipment. Many modern systems are dependent on the successful operation of a large number of component parts. The effect of complexity on over-all system reliability is indicated in Fig. 2, which was adopted from the work of Lusser.[9] For simplicity, equal reliability of the parts is assumed in Fig. 2. It can be seen that for a system or subsystem with a great many parts, the individual part reliability must be extremely high to achieve a reasonably high over-all system reliability. In a system where all components are in series, the over-all equipment reliability is a product of the reliability of each part; i.e.,

$$R_{over\text{-}all} = R_1 \times R_2 \times R_3 \ldots R_n \quad .$$

This expression does not include the possibility of reliability interaction among the parts.

Determinations of subsystem and system reliabilities require the determination of the expected lifetimes of the component parts. Wray and Cowper[10] give three principal modes of failure for equipment that determine life expectancy for component parts:

1. Faults that are a direct consequence of malproduction or bad assembly. These are normally revealed during the first few hours of

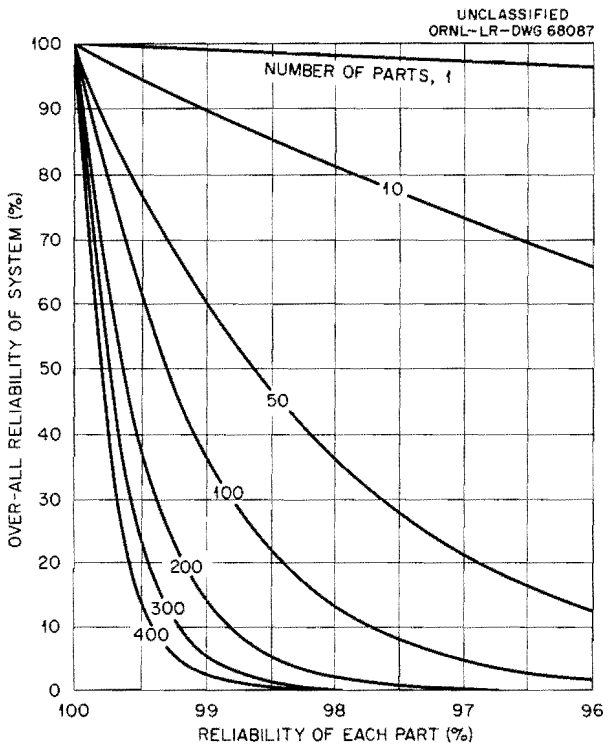


Fig. 2. Over-All System Reliability as a Function of Reliability of the Parts.

operating life. In reactor operation, these faults can usually be found during shakedown operation.

2. Faults that occur at any time during the life of the equipment. These are random failures and have equal likelihood of occurring at one time as at another during operation of the equipment. The failure probability has been termed the "exponential-life curve."[11]

3. Faults that occur as a consequence of wear on some mechanical part. The rate of failure from such faults tends to increase rapidly once the wearout phenomenon appears. Failures of this type can be avoided by proper design and maintenance planning.

A typical statistical curve of the variation of failure rate during the life of equipment is presented in Fig. 3, and the difference between the exponential-life and the wearout-life curves is shown in Fig. 4.[8,11] The prediction of component, subsystem, or system reliability requires knowledge of the lifetime curves for the individual parts making up the system. The scarcity of such information is probably the largest factor in the inability of engineers to predict system reliability accurately. Unfortunately unreliable information and assumptions are often all that is available for use in making calculations of system reliability. Farrier[12] has pointed out that this dearth of data on the life expectancy of parts seriously handicaps the designer. He further states that controlled or semicontrolled field tests should be the prime source of life-expectancy data so that the sampling statistics will be sufficiently large and the environmental conditions can reflect the normal operating conditions to be expected. The deficiencies of such information are also recognized,

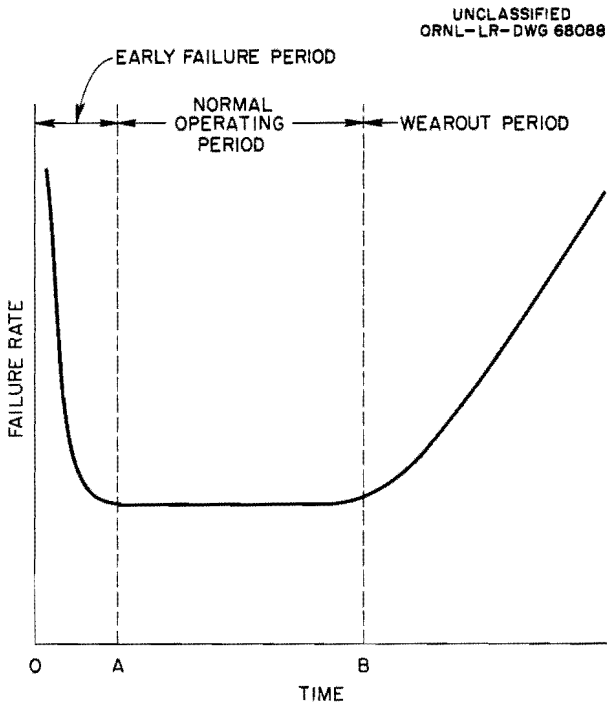


Fig. 3. Failure Rate of Equipment as a Function of Time.

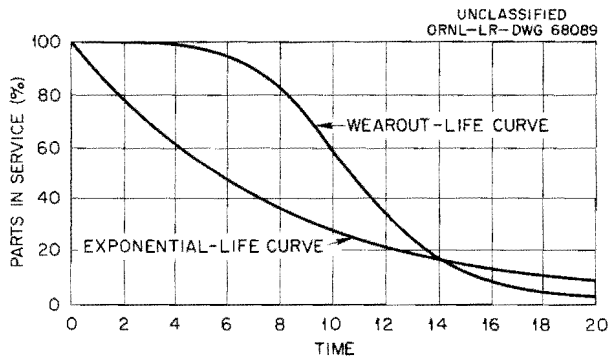


Fig. 4. Curves Illustrating Difference in Shapes of Exponential-Life Curve and Wearout-Life Curve.

however, in that large variations exist in the amount and quality of reported information on failure analyses of parts or systems.

Davis and Holmes[13] have described a program designed to accumulate information in the field on the reliability of complex airborne electronic equipment. The stated objectives of the program are to obtain data from field operation of equipment with which to evaluate the current reliability of the equipment and the causes of unreliability. Such data will permit modification and improvement of the equipment. Their program has been in operation since 1955 and is thought to be highly successful.

In the reviewer's opinion there exists a basic need for a similar program for obtaining and correlating information on nuclear equipment reliability. An information center should be established to serve as a collection point for field data and laboratory test data on life expectancy of reactor parts and components. The information obtained could be reported in an appropriate journal. The present system whereby each design organization bases reliability estimates on information it may happen to possess on life expectancy is not satisfactory. Progress in the direction of improved utilization of field reliability data is indicated by the establishment of a computer at the Air Force Special Weapons Center, Albuquerque, New Mexico,[14] to process field data from the PM-1 and EBWR reactors for the purpose of predicting plant reliability and providing a basis for improving the design of individual components.

If life-expectancy curves for component parts of a system are available, the designer can evaluate system reliability. For example, Kaufmann and Kaufman[1] have incorporated basic design assumptions (i.e., life-

expectancy data) into prediction equations. Their predictor variables and prediction equations have the following bases:

1. Predictor variables should be derived from simple elemental design variables.

2. Prediction equations should forecast malfunctions or other forms of reliability information with statistical assurance of the order of a 0.05 confidence level.

3. Prediction equations should be applicable to various types of devices (electronic, pneumatic, hydraulic).

4. Prediction equations should provide means for evaluating a design at any stage in its evolution.

Three methods for predicting system reliability and mathematical expressions derived from the criteria listed above have been outlined by Kaufmann and Kaufman.[1] The first, or "standard," method consists essentially of using one component as a standard and comparing all other components of the system (or subsystem) on the basis of mean-life ratings or mean time-to-failure rates with the standard. The second method establishes relative complexity factors for each component using a rating for each component established on an agreed upon basis rather than a unit complexity factor for a standard component. The third method is proposed for establishing predictor variables for parts based on cost, weight, and volume. The predictor variables proposed are used to determine a predictor factor in an equation of the form:

$$K = (\text{cost})^a \, (\text{weight})^b \, (\text{volume})^c \quad .$$

Kaufmann and Kaufman[1] applied this third method to 12 subsystems of a ground-to-air missile and were able to obtain good correlation between

the rank order of predictor factors K and the rank order of part failures experimentally charged to the corresponding subsystems.

It should be noted that the concept proposed does not explain system interactions which may affect reliability; rather its usefulness lies in the possibility of predicting the relative value, or as the authors suggest, relative utility of the subsystem. The concept appears to possess particular value in those areas where the incentive for weight and volume minimization are greatest, and it would be well to test the concept for other, perhaps more prosaic, systems.

The problem of assessing reliability of a reactor system may be handled by dividing the system into reliability sections or subsystems.[11] The subsystem would consist of components for which life-expectancy data were either available or for which an estimate of the life-expectancy could be made by assuming, for example, an exponential life curve for the parts. The basic difficulty in predicting system reliability lies in the problem of predicting the part life expectancy and mode of failure under the operating environment. Frantik[15] has pointed out that in studies of reliability at the Sandia Corporation no general procedure has been found for predicting modes of failure, particularly where a combination of environmental factors is involved. It is this imperfect understanding of the environmental factors and their effects on component reliability that creates the greatest difficulty in predicting system reliability. If it is known that wearout phenomena may be encountered, the life-expectancy equations can be modified accordingly. In designing for reliability, however, maintenance or replacement before the wearout phenomena occurred would normally be planned. In unattended plants, on the other hand, the

wearout phenomena should not occur within the desired life of the system.

The problem of predicting system reliability with intermittent component usage, system age, and periodic maintenance as parameters has been studied.[16,17] The effect of periodic maintenance on system reliability is illustrated in Fig. 5 for a system operated continuously except for interruptions caused by system failure accompanied by replacement of the failed component with an identical but new component. The curve parameters are the periodic maintenance intervals. The curve $T = \infty$ indicates no maintenance and thus represents the system reliability for a new system from startup. Increased frequency of system maintenance produces higher reliability, as shown in Fig. 5.

## Reliability Considerations in System Design

There are a number of methods designers employ for improving the reliability of a system or component. One of the more significant schemes involves the use of redundancy in the design. A common example of design redundancy is the use of coincident circuits in the control and safety systems of nuclear reactors, for instance, the use of a system of two out of three signals to initiate the control action on a scram circuit. Such schemes[18] essentially recognize insufficient reliability of available equipment and improve the system reliability by using parallel systems in such a manner that failure of one system will not affect the functional capabilities of the parallel systems.[19,20] Other examples of the use of redundancy for improving reliability are spare pumps and backup valves. Although widely used, the redundancy principle is often wasteful from a

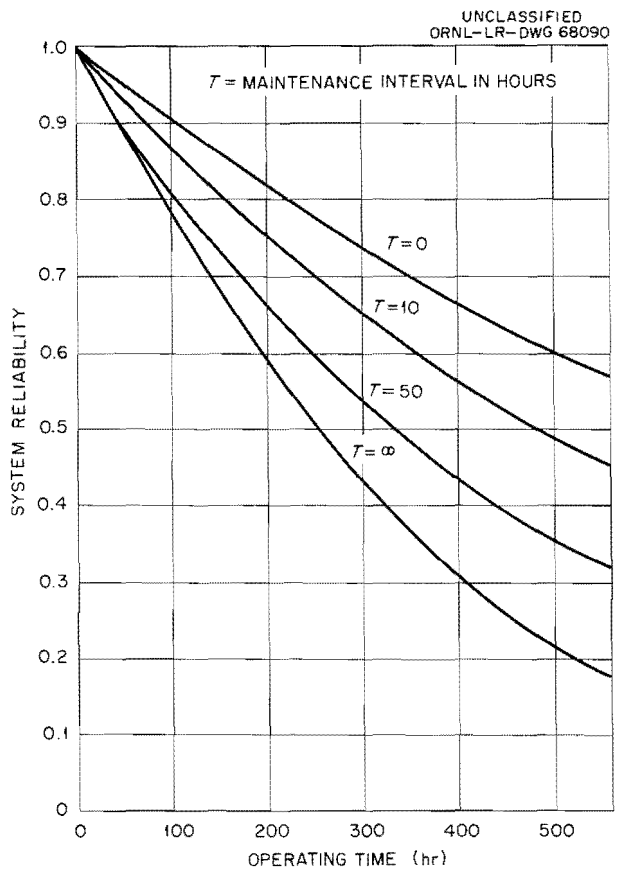


Fig. 5. System Reliability as a Function of Operating Time and Maintenance Intervals.

design economy standpoint and does not attack the basic problem of improving equipment reliability.

In the control and safety systems of a reactor, provisions are often made to sense an incident by the use of an additional independent parameter, since it is, of course, important to assure that no common fault will cause simultaneous failures. The use of independent sensing parameters may, however, result in differences in the time of the response to the occurrence of an incident.

Another rather obvious scheme for improving system reliability is design simplicity. A system designed with a minimum of vital components can have a high inherent reliability. An extreme example of the use of this principle is a natural-convection circulating-fuel reactor that generates electric power by the thermoelectric effect.[4] Such a system eliminates pumps, conventional heat exchangers, and the turbine generator; that is, essentially all moving parts are eliminated from the primary power plant and the resulting system has high inherent reliability.

A word of caution must, however, be injected with respect to a tendency to assess over-all system reliability subjectively on the basis of apparent complexity or number of system components. Subjective reliability is defined as that part of reliability formed in the minds of men; it is a qualitative reliability that is not subject to measurement. As pointed out by Miller,[21] however, it plays a part in system design and cannot be ignored. Subjective reliability determinations are of value when measured reliability data are lacking and when the engineer must assess the potential reliability of a part or component based on extrapolation of his experience with similar parts or systems.

If presently available components are to be utilized and failures are anticipated during the life of the plant, it is important that accessibility and maintainability be considered. If safety equipment is to be provided that will be used only intermittently, the system must be designed so that periodic proof tests can be performed. All possible modes of failure should be considered to assure that failure will not cause an unsafe condition. A technique commonly used to improve system reliability when presently available components are included is to derate the capabilities of the components. For example, electrical and mechanical parts are intentionally subject to loads far under their rated capacity. Again, such design is often uneconomical.

Another factor that affects plant reliability is associated with what is termed "human reliability." This reliability factor affects the adequacy of the design and the quality of fabrication, construction, installation, maintenance, and operation. It is this factor that in practice is the most difficult to evaluate. The Gilbert Associates report[3] points out that approximately 5% of the incidents they surveyed were attributable to human error. The report[3] further states that human error may have been responsible for many other incidents but that this was difficult to determine from the records.

The problem of reducing human error has been discussed by Lincoln,[22] who classified such errors as (1) errors of omission, including errors of memory and attention, and (2) errors of commission, including errors of identification, interpretation, and operation. Lincoln, in summarizing some facets of human reliability engineering, states that equipment design should (1) facilitate the recall of operational sequences, (2) exploit the

effects of pattern in gaining the operators' attention, (3) provide redundant cues to effect identification, (4) eliminate the need for display interpretation, and (5) provide for consistent movement relations.

Improvement in Component Reliability

The long-range solution to the problem of obtaining higher reliability of parts, components, and systems lies in the establishment of research programs for designing and developing reliable components and for testing reliability. The impetus for increased component and system reliability exists, since in the design of reactor components, the required system reliability must be established either by the customer (as in military reactors) or by economic considerations (as in power reactors) and in both cases by safety considerations, at least in systems where plant and human safety are of concern.

Many of the difficulties of attaining high system reliability are inherent in the procurement process. The reliability requirements must be written into the specifications, and the engineering and purchasing organizations must obtain assurance that only qualified vendors have submitted bids. Where equipment reliability is of paramount importance, the bid price should not influence the equipment reliability assessment.

At the present time there is relatively little interchange of information concerning the operation of equipment in nuclear plants from the standpoint of reliability. As more nuclear plants are operated, the sources of reliability information will increase. It seems evident that the establishment of a center for the collection and evaluation of such information could contribute significantly to the solution of the reliability

problems. Incidents involving equipment failure or successful use of parts or special systems could be reported to such a center. Forms for reporting the information required for determining the effects of environmental factors could be made available at all plants in order to alleviate some of the present difficulties in evaluating information from various sources regarding operations under diverse conditions. Many of the components used in nuclear plants are of somewhat unique design, and normal mass production quality control and sampling techniques may not be applicable. Until the effect of quality on the performance of nuclear components is known, meaningful evaluation of reliability will continue to be difficult.

Gilbert Associates, as a result of their survey on nuclear plants, submitted a program for improving over-all nuclear power plant reliability that incorporated the following steps:[3]

1. Build prototype plants for reliability studies and analyze their performance.

2. Define performance requirements, environmental parameters, and other special conditions to be met by the individual components and systems.

3. Review the qualifications of potential bidders to determine their ability to manufacture equipment in conformance with the specifications.

4. Establish a qualified inspection group and a quality control program.

5. Thoroughly train operating personnel for the plant and establish and enforce a program of preventive maintenance.

6. Carry out research and development work for improving designs of specific components and systems based on information obtained from the feedback of operating data.

The improvement of plant reliability must begin with the design concept. Too often systems are designed before the required reliability is determined. In many instances, highly reliable equipment has been developed as a byproduct of extensive industrial experience, rather than as a primary system objective ab initio. In some cases, costly failures have provided the incentive for determining the principal causes of failure and the required remedial measures.

Efforts to institute programs for increasing component reliability might initially increase the already high capital costs of nuclear power plants, and the possible reductions in maintenance costs and increases in operating life have not been sufficient incentive for increased capital expenditure. It is because of this and the fact that nuclear power is still more costly than conventional power that much of the impetus for higher reliability is expected to come from the development of reactors for remote operation at military installations and auxiliary nuclear power plants for space programs.

It has been suggested[16] that a research program for improving component reliability should include (1) design analyses for establishing methods for predicting modes of failure, (2) evaluations of testing programs and field data to obtain reliability information, (3) evaluations of the effect of environment on reliability, and (4) the establishment of methods for specifying reliability.

Ultimately the safety and economy of a nuclear power plant depend on its over-all reliability. A review of present reactor designs indicates that redundancy is normally used to assure system reliability. This design technique has become so firmly established that it is accepted as necessary, and insufficient effort is expended to improve component and part reliability. The establishment of component reliability as a design objective must be done gradually. At present there is apparently little economic incentive for improved reliability and a good deal of hesitancy in the use of this concept. (M. M. Yarosh)

# METAL-WATER REACTIONS

Reactors that utilize water as the coolant, moderator, neutron shielding, or other type of working fluid in combination with metallic core components have an inherent hazard in the possibility of a chemical reaction between the metal and the water. This hazard exists because nuclear systems are potentially capable of overheating to such an extent that melting and, in some cases, vaporization of the metal in the reactor core can occur.

Many nuclear reactors contain large quantities of metallic components which can react exothermically with water. The energy release associated with the reaction can conceivably occur with explosive violence.[23,24] If such a reaction were to take place during a reactor accident and go to completion, the energy output of the metal-water reaction might be greater than the nuclear energy release.[24] Further, since highly toxic fission products build up in the operation of a reactor, if the energy release in a nuclear incident were sufficiently increased by a metal-water reaction to cause violation of the containment structure, widespread dissemination of radioactive materials could occur and cause a catastrophe that would extend far beyond the area adjacent to the reactor. Thus the possibility of the metal-water reaction is a significant factor in the design of the containment structure, since the contribution of the reaction energy to the dispersal of the fission products existing in the reactor system at the time of an accident can be a serious hazard.[23-25]

## Accidents Leading to Metal-Water Reactions

The reactor malfunctions that are most likely to cause a metal-water reaction are loss of coolant and a rapid power rise.[23-26] The loss-of-

coolant accident usually involves a failure in the integrity of the cooling system while the reactor is at power. The resultant interruption of coolant flow should cause a reactor scram. Without cooling of the core, however, the decay heat from the fuel would cause the metal in the immediate vicinity to heat up and possibly to become highly reactive with the surrounding water vapor. Since metal-water reactions are, in general, exothermic, the reaction between the metal in the solid state and the steam would probably accelerate the temperature rise of the core metals until their melting points were reached. The next event in this accident sequence would be the dripping of core metals from the core through the steam-filled interior of the reactor pressure vessel into a pool of water or onto cooler metal surfaces. Obviously the rate and extent of the metal-water reaction must be known in order to evaluate fully the possible hazards associated with this accident.[24,27,28]

The rapid-power-rise accident results when the reactor power increases rapidly and causes the temperature of the fuel and core materials to rise so rapidly that the metals in the core melt before processes such as heat dissipation and coolant flow can occur. Under such circumstances the metal-water reaction could be quite violent. The extent of the reactor power excursion and the rate and extent of the metal-water reaction in relation to the heat dissipated from the core determine the severity of the accident.[23-25]

The mere melting of metal in the presence of water does not necessarily cause a metal-water reaction. This fact is borne out by actual reactor incidents which have resulted in the melting of metal in the presence of water at the NRX, BORAX I, SPERT I, and WTR facilities. These incidents, described

below, show that overheating and melting of core metals can happen without a metal-water reaction.

On December 12, 1952, the Canadian NRX facility underwent a nuclear runaway that involved energy release at a rate of 200 Mw/sec.[29] The accident caused meltdown of part of the aluminum sheaths around the uranium rods, a uranium-water reaction, melting of the uranium, and possibly a hydrogen-oxygen explosion. Whether or not an aluminum-water reaction took place could not be decided.

The BORAX I reactor was purposely destroyed in a reactor safety test[30] on July 22, 1954. The reactor was taken on a very short power excursion of about 30 msec, during which energy was released at a rate of 135 Mw/sec. This test caused melting of most of the aluminum-clad aluminum-uranium alloy fuel plates. Despite the extensive melting, however, there was no evidence that a metal-water reaction had taken place.

In a routine removal of SPERT I Core B on April 24, 1958, a break was noted in one of the removable plates of one of the fuel assemblies. Metallurgical examination[31] of this plate showed that in the course of operation of this reactor, a small area of the aluminum plate on one aluminum-uranium alloy fuel element had failed by melting. This incident did not affect reactor operation and was detected only through routine examination.

On April 3, 1960, the Westinghouse Test Reactor, which is water cooled and water moderated, experienced fuel burnout.[32] Sixty-nine fuel positions were fueled with assemblies in the form of three concentric cylinders composed of uranium-aluminum alloy clad with aluminum. The fuel burnout occurred during a series of low-flow tests. In the postaccident examination it was observed that the fuel elements had melted in the central portion,

and some molten aluminum was found on the upper end of the basket. No indication of a metal-water reaction was reported.

Metal-Water Reaction Kinetics

During a reactor accident, the materials normally used in reactor construction, such as aluminum, beryllium, magnesium, stainless steel, uranium, zirconium, and alloys of these materials, could melt or even vaporize, and, if such melting or vaporization occurred in the presence of water, a violent reaction could result.[23,33] Thermodynamically, nickel would be an exception in that it would not react spontaneously with water at any temperature.[33] At the normal water-cooled reactor operating temperatures the probability of a violent metal-water reaction is essentially zero. The reaction that takes place below the melting point is a corrosion type of process that is relatively slow, the reaction rate being a function of the metal temperature. The reaction rate increases with temperature during the solid metal-water reaction, but rapid and violent reactions apparently do not occur until the metal becomes molten.[23,25] Techniques for experimentally measuring the reaction rates have been established for most materials of interest.[23] Furman[34] has been successful in electromagnetically levitating molten zirconium so that the energy released by the reaction of liquid zirconium with water can be measured.

Once the metal is molten, the subsequent course of events will depend primarily on the amount of the liquid metal, the size of the droplets, the presence of water, the ratio of metal to water, and the temperature of the metal-to-water interface.[23,24,27,28] Vogel et al[35-38] have demonstrated these effects experimentally. In their work, fine wires of uranium

and zirconium were heated under water with the energy from a charged capacitor. The wires were heated to temperatures that were calculated to cause the wire to vary from solid to liquid, and in some tests the wires were partially vaporized. The transient pressure observed was a measure of the reaction rate. Analysis of the average particle size of the residue gave the surface area that was available for reaction. The amount of hydrogen evolved gave a measure of the extent of the reaction. The work with zirconium showed explosive pressure rises only with molten metal at high temperatures, and the reactions produced small-particle-size residues. In addition, the high-temperature experiments with zirconium were the only ones in which large quantities of hydrogen were produced.

The rate and extent of a metal-water reaction depends on the thermodynamic properties of the reacting materials and upon the details of the heat transfer situation. Calculations of the details of a reaction are therefore quite complicated, since all the factors are interrelated and are strong functions of time. The general dependence of the metal-water reaction on each factor is discussed below.

Effect of Metal Droplet Size. Estimation of the average droplet size to be expected in a particular accident remains a problem.[23,33] For the relatively slow loss-of-coolant accident, Lustman[27,28] assumed that the minimum-diameter droplet that could be formed could be estimated by first calculating the weight of a molten sphere of metal forming on a rod of diameter comparable to the minimum thickness of the metal in the core and then balancing this weight with the surface tension forces of the droplet.

Lustman assumed 200 dynes/cm for the surface tension of liquid zirconium. This property has since been measured and found to be 1387 dynes/cm.[39]

There are other incidents, however, that have been conceived as credible and exceptionally violent which would not permit molten metal to form as droplets and be dispersed by dripping from the core. Such incidents would involve a steam explosion or a sudden release of the fission-product gases contained in the fuel elements or both. This action could blow the metal into small fragments or the molten metal into small-diameter droplets. In general, accurate predictions of the extent of dispersion and the metal-particle diameter resulting from a given incident are difficult to make.[23] One solution to this dilemma is to attempt to simulate the accident experimentally on a small scale and then to interpret the results for the accident analysis. The difficulty here lies in knowing exactly what to simulate. Even if this can be done satisfactorily, extrapolating the data from a small scale to actual size of the system is not without problems and uncertainties.

Effect of Metal-Water Boundary. Unless the metal has vaporized, it is realized that the metal-water reaction will most likely be heterogeneous and consequently that a phase boundary will exist between the metal and the water. Therefore, the reaction must occur at or near the phase interface. For this to happen, the two phases must come together, and then the chemical reaction must take place. The reaction involving these two steps can be composed of many processes with rates that vary widely with a particular set of conditions and with time. If one of these steps is particularly slow, so long as it operates in the reaction, it will determine the over-all rate.[24] The transport process can be influenced by such

things as the extent of film or oxide formation on the metal, the stability of the oxide film, and the rate of water-vapor transport. These mechanisms which affect the rate of reaction have been the subject of many studies. Bostrum[40] performed kinetic studies on the reaction of Zircaloy-2 with water, and Lemmon[41] performed kinetic studies on the reaction of Zircaloy-2 with steam. These studies indicated that the reaction rate was approximately parabolic at temperatures approaching the melting point of the metal. Epstein[42] attempted to determine the nature of the fastest possible reaction. From his work it appears that this reaction involves the gas-solid reaction in which water molecules are brought by vapor-phase transport to the metal surface. The rate of this reaction is probably determined initially by the rate of collision of the water molecules on the metal surface. Later the reaction rate is limited by the transport of the water vapor through the hydrogen which evolves as a reaction product. Finally, the process is slowed and controlled by solid-state diffusion.

Effect of Metal-Water Proportions. Metal-water reactions can be significantly limited if the ratio of metal to water does not approach the stoichiometric quantities required and the metal and water are adequately dispersed. If an excess of metal or steam is present, the system temperature will not rise as high as it would if the extra material were not present, since the excess of one reactant acts to cool down the system temperature and the deficiency of the other can limit the reaction that could occur at that temperature. Since the extent and rate of these reactions are very dependent on the temperature of the system and since an unbalance of reactants can limit the temperature, the unbalance can result in limiting the reaction.[23,43]

Effect of Temperature. For each metal there appears to be a characteristic minimum critical temperature, $T_c$, required to produce a significant metal-water reaction. In the work at Argonne[37] and at Aerojet,[26] it was found that the reaction of zirconium with water begins at temperatures near the melting point of the metal, and the extent of reaction increases linearally up to temperatures near, but below, the melting point of zirconium oxide. At this point, the extent of the reaction seems to jump significantly, indicating a critical temperature.

Some data that have been obtained for aluminum indicate that its critical temperature is above the melting point (660°C) of the metal. Higgins and Schultz[26] found a $T_c$ of 1170°C. On the other hand, Elbert and Brown[44] concluded from their tests that the probability of aluminum reacting violently with water is greatest just above the melting point and decreases at higher temperatures. Zelezny[33] in recent studies of the aluminum-water reaction found that the peak reaction rate occurred at temperatures in the neighborhood of 1800 to 1900°F (or 1010°C) and that the rate of reaction declined at higher temperatures.

Higgins and Schultz[26] found that the behavior of aluminum and zirconium could be compared by the expression

$$R = \frac{T_c - T_{mp}}{T_{bp} - T_{mp}} ,$$

where

$T_c$ = critical temperature, °K,

$T_{mp}$ = melting point of the metal, °K,

$T_{bp}$ = boiling point of the metal, °K.

The ratio, R, for both metals was around 0.29.

Epstein[23] hypothesized the following relation for the critical temperature:

$$(T_c - T_w) = (T_{mp} - T_w) + Kr_o^2 \quad ,$$

where

$T_c$ = critical temperature,

$T_w$ = bulk water temperature,

$T_{mp}$ = melting point,

$r_o$ = initial particle radius,

$K$ = constant.

This equation indicates that the critical temperature will approach the melting-point temperature as the particle size approaches zero and, conversely, that the critical temperature will increase as the particle size increases. This concept indicates that, if the metal temperature is below the critical temperature, the reaction will not be significant; however, if the metal temperature is above the critical temperature, the reaction will proceed; and, if $T_c$ continues to rise, the reaction will be accelerated and becomes autocatalytic.

The metal-water reactions of the alkali metals would be expected to be quite different in that the reaction products formed would be water soluble. Higgins[45] found that the critical temperatures of alkali metals in chemical reaction with water were just above their melting points.

Knowledge of the critical temperatures of the various materials is very important. However, from the existing data, it appears that the values of the critical temperatures for the materials that have been studied are not in good agreement. It would seem that the only solution to this

problem is to continue experimentation with all the metals of interest in reactor construction and operation in order to establish or to refine the data.

Estimation of Metal-Water Reactions

In hazards analyses the reactor designer must consider any metal-water reactions that could take place within the system during a credible accident. To do this, the energy released by the metal-water reaction is computed as a function of time, and the possibility of an explosion from the hydrogen formed as a result of the reaction is investigated. Realizing that the metal-water reaction is a secondary process and that it is initiated by the nuclear meltdown accident, the energy release of the chemical reaction is added to the energy release of the initiating accident, and the hazards evaluation is based on the total energy released as a function of time. The problem of evaluating damage resulting from the energy released by chemical reactions is essentially no different from evaluating the damage from an uncontrolled energy release resulting from fission heating and radioactive-decay heating. After the energy release is computed, however, predicting the damage to the reactor system and the release of fission products that will result from a given energy release is not simple in any case.[23]

Several computational procedures have been developed to analyze the extent and rate of energy release. Lustman[27] developed a computational method for calculating the extent of a zirconium-water reaction and applied this method to the analysis of the loss-of-coolant accident for the Shippingport pressurized-water reactor (PWR).[28] This analysis was performed

for the following situation: When the reactor is at power and begins to lose its coolant, the reactor scrams. When the core becomes uncovered, the energy source in the core is primarily from decay heat of the fission products. Losing the coolant greatly reduces the heat transfer from the core, and the fuel elements and core materials begin to heat up. At the same time the zirconium begins to react with the water vapor in the reactor pressure vessel. At about 1000°K the oxidation reaction of the zirconium begins to make a significant contribution to the heat input of the system. If the cooling rate is slow, the metal temperature continues to increase and the reaction becomes autocatalytic. When the melting point of the metal is reached, the zirconium begins to form droplets and finally falls from the core region through steam into water at the bottom of the reactor vessel, where most of the metal-water reaction takes place. Lustman's analysis was made according to the method of LaChatelier (described by Saltsburg[24]), which involves making a heat balance in step intervals of the temperature that balances the decay heat, the heat of reaction, and the heat capacities of the materials involved to determine whether the elements heat up or cool off. The data of Bostrom[40] were used in the reactor kinetics calculations.

The second part of Lustman's analysis involved the molten droplet falling through the steam environment into water. The analysis was made by essentially the same method as that used for the solid-state oxidation analysis, except that the heat balance included the heat loss of the metal droplet by radiation. The rate and extent of reaction while the particle cooled down to 1000°K was estimated.

In studying this heat transfer process of the metal-water reaction, two cases must be analyzed. The first case involves the situation in which the metal is surrounded only by steam. Here the heat transfer can be calculated from physical theory. The problem becomes more complex, however, when the reduction of the metal particle radius as a result of oxidation is accounted for as a function of time. The second case involves the situation in which the metal is immersed in water and the temperature of the metal is significantly higher than the temperature of the water. This situation results in the formation of a steam film around the metal droplet. When the temperature of the particle is determined as a function of time and as a function of the radius of the particle plus the steam blanket, the problem becomes essentially that of the case where the metal is surrounded only by steam. The latter computation is, however, difficult.[23,46] Lustman assumed that, in the temperature region where the metal-water reaction is significant, the heat loss from the metal droplets was by radiation alone. He calculated the extent of the reaction for conditions of emissivity of both $\epsilon = 0.5$ and $\epsilon = 1.0$. He determined from Cubicciotti's data[47] that values of the emissivity in excess of 0.6 were likely. An $\epsilon$ value of 0.67 was found experimentally for oxidized Zircaloy by Wood et al.[48]

Lustman's theoretical calculations of the extent of reaction of Zircaloy agree well with the molten-metal drop-test work of Higgins and Schultz.[26] Hogan et al.[49] considered the convective heat transfer contribution to the heat loss as measured and correlated by Bromley et al.[50] These computations resulted in a prediction that the larger drops would react to a greater extent than the smaller droplets of metal,[28] which is

contrary to the experimental results of Higgins and Schultz. Although the agreement of Lustman's calculations with the Aerojet experiment[26] is striking, it nevertheless should be recognized that the heat transfer mechanism is quite complex and needs to be studied further.

Other calculations and calculational methods for studying metal-water reactions have been reported. Janssen, Cook, and Hikido[25] set up a mathematical model to analyze the nuclear runaway case. Their model includes such factors as the reactor geometry, the hydrodynamics and heat transfer of the system, the steam void formed, and reactor kinetics. Owens[51] applied this model in evaluating three water-cooled and -moderated reactors. These reactors were characterized by one of three types of fuel element: aluminum-clad uranium-aluminum flat plates; zirconium-clad uranium-zirconium alloy flat plates; and zirconium-clad $UO_2$ pellets formed into rods.

Considerable work has been performed, and much work is still in progress in the investigation of metal-water reactions. A large part of the work on metal-water reactions not discussed here may be found in references 23, 28, 33, and 52. (L. D. Schaffer)

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# III. CONTROL AND INSTRUMENTATION

# CONTROL SYSTEMS OF PULSE REACTORS

Under the proper conditions, short high-level bursts of fast neutrons may be predictably generated by subjecting an unmoderated reactor to self-limited supercritical excursions. Such operations originated at the Los Alamos Scientific Laboratory as a supplemental application of the unmoderated Godiva Reactor (originally built for reactivity and fission-rate measurements)[1,2] and was recognized as an end in itself with the construction at that laboratory of Godiva II,[3] a reactor specifically designed for fast-burst operation.

Subsequently there has been considerable interest[4-7] in building such reactors for use as substitutes for bomb tests, as pulsed-neutron sources for time-of-flight measurements, as means of investigating short half-life reactions, and for other applications where a 50-μsec burst of $10^{16}$ to $10^{17}$ fissions (peak power $10^4$ to $10^5$ Mw) would be useful.

## Conventional Startup Criteria

To aid in the evaluation of problems arising from the operation of this type of installation, it is instructive to review the precautions which govern the design and operation of more conventional reactors. The usual reactor is designed to operate for extended periods of time at fixed power. The startup machinery for such a reactor is used relatively infrequently and is designed to insert reactivity slowly to bring the fission rate from the source level to full power in an orderly fashion. This means that the response of the reactor is continually monitored as reactivity is added, and the rise in power is achieved slowly enough to permit reversal should the rate or the level get too high. Monitoring the

effect of rod withdrawal in the subcritical range introduces the requirement for an appreciable neutron source to permit the counting instruments to detect the rise in multiplication as the assembly approaches criticality. The minimum reactor period that can be permitted depends on the speed of the safety system.* Most reactors will not seriously overshoot their design power if the shortest period permitted is made equal to the "dead time" of the safety system. "Dead time" is defined as the interval from the time the reactor reaches a trip point to the time when the safety rods start in. Fixing the shortest reactor period in this way also determines the fastest allowable rod withdrawal rate, since one is a function of the other.

An analysis of the startup accident by Newson[8] relates the minimum value of the reactor period, $\tau$, at the time of a safety trip to the reactivity insertion rate, r, by the following inequality:

$$\tau > \left( \frac{l}{2r \ln \frac{\phi^2}{\phi_1}} \right)^{1/2} ,$$

where $l$ is the prompt neutron lifetime, $\phi_2$ is the flux level at the safety trip point, and $\phi_1$ is the source level. As an example, consider a reactor with a power range of $10^8$ from source level to scram level. Let $l = 5 \times 10^{-5}$ sec and the safety response time be approximately 40 msec. In order to assure that $\tau$ will be more than 40 msec, the allowable reactivity insertion rate becomes

*The term "period" as used here is the reciprocal of $[d(\ln n)]/dt$ at a particular instant.

$$r = \frac{l}{2\tau^2 \ln \frac{\phi_2}{\phi_1}} = 0.85 \times 10^{-3} \ (\Delta k/k)/\text{sec}$$

or 0.085% $(\Delta k/k)$/sec, which is close to the value used in a number of reactors.

If the fast-burst reactor were dependent for safety on its mechanical shutdown system, the short neutron lifetime and large power range of such an assembly would restrict reactivity insertion to

$$r = \frac{6 \times 10^{-9}}{2 \times 1600 \times 10^{-6} \ln 10^{14}} = 5.8 \times 10^{-8} \ (\Delta k/k)/\text{sec} \quad .$$

Operational Problems in Fast-Burst Reactors

It is evident that fast-pulse reactors, whose normal operating procedure may take them to peak powers of up to $10^5$ Mw and which go up to such power levels on 10- to 20-μsec periods, cannot depend on conventional safety systems for the termination of dangerous excursions, since the customary "startup" (or burst) involves times far shorter than the response time of the conventional safety system. The excursion must therefore be self-terminating through a mechanism such as a negative temperature coefficient of reactivity that will bring the reactor subcritical.[6] For fast reactors the Doppler coefficient is slightly positive and must be offset by other components of the over-all temperature coefficient. Thermal expansion of the metal provides such a coefficient and effects the termination of excursions resulting from reactivity insertions of \$1.05 or \$1.10 above delayed critical after $10^{16}$ or $10^{17}$ fissions.

In a thermally isolated system with a large negative temperature coefficient, it is clear that reactivity slowly inserted will give little or no peak in power but, rather, will only raise the temperature of the assembly. It is therefore apparent that to achieve an appreciable power burst from such a reactor the excess reactivity must be inserted rapidly and that to obtain reproducibility of the pulses all the excess reactivity must be inserted before a sustained fission chain reaction is initiated. Once started, such a chain reaction is given time in which to reach a power peak because of the inertial mass of the reactor, which provides a lag in thermal expansion and thus delays shutdown.

The speed of reactivity insertion necessary to ensure that the burst is not initiated before insertion is completed depends primarily on the amount of reactivity inserted and the neutron source level. Hansen[9] gives the following approximate expression for the probability, P, of the first persistent fission chain reaction appearing in the interval $dt_1$ at time $t_1$ after the system becomes critical:

$$P(t_1)\ dt_1 \cong \frac{2rSt_1}{\bar{v}r_2}\, e^{-rSt_1^2/\bar{v}P_2}$$

$$\cong rSt_1\, e^{-rSt_1^2/2} \qquad \text{for } t_1 \geqslant 0$$

where

$r$ = rate of reactivity addition = $\partial k/\partial t$,

$S$ = source strength, r/sec,

$\bar{v}$ = average number of neutrons per fission $\cong 2.5$,

$P_2 = [v(v-1)]/v^2 \cong 0.8$.

If this expression is integrated from time T to $\infty$, the result is

$$\int_T^{\infty} P(t_1)\ dt_1 = e^{-rST^2/2} \quad ,$$

which gives the probability that no persistent fission chain reaction will be initiated before time T after the system becomes critical. If no more than one abortive burst in $10^3$ is desired, $rST^2 \leqslant 2 \times 10^{-3}$. It should be noted that $r \cong \Delta k/T$, so $rST^2 \cong ST\ \Delta k$, and $T \leqslant (2 \times 10^{-3})/S\ \Delta k$, where $\Delta k = k - 1$ and k is the reactor multiplication factor.

Spontaneous fissions in such an assembly[3] produce a source of about 90 neutrons/sec, and if there has been a 20-min wait since a delayed critical run at 1 w, delayed neutrons will boost this to 95 neutrons/sec. A wait of 15 sec from delayed critical to burst-rod insertion permits about a threefold growth in the level,[3] bringing the effective source to 285 neutrons/sec. The $\Delta k$ in the burst rod is about $\$1.08 = 7 \times 10^{-3}$. These values yield

$$T \leqslant \frac{2 \times 10^{-3}}{285 \times (7 \times 10^{-3})} = 1 \times 10^{-3} = 1 \text{ msec} \quad ,$$

and thus a speed of insertion of 333 ft/sec is required for a 4-in. burst rod. Such a speed could perhaps be achieved, but most burst-rod drive mechanisms insert the rod in around 50 msec. Such a time makes the probability for preignition = $[285 \times (7 \times 10^{-3}) \times 0.05]/2 = 0.05$. Hence 5% of the bursts can be expected to give less than the maximum pulse because of initiation before reactivity insertion is completed.

It has occasionally been suggested that the hazard associated with allowing a prompt critical reactor to sit around waiting for a persistent

fission chain reaction (or perhaps for the accidental addition of more reactivity) could be minimized by operating in the presence of a source. In this way the burst would start almost as soon as the rod was in, and the long wait could be eliminated. Moreover, any reactivity inadvertently added between the initial condition and burst-rod firing would make itself known by an immediate rise in power. To make such operations possible, it might be decided that, in the interest of safety, 10% of the bursts could be allowed to abort. This would give $e^{-rST^2/2} \geqslant 0.90$, or $ST\,\Delta k/2 \leqslant 0.105$. If $\Delta k$ is $7 \times 10^{-3}$ and S is $1 \times 10^5$ neutrons/sec, $T \leqslant 0.3$ msec. For a 4-in. burst rod, this is insertion at about 1000 ft/sec, a situation unattractive from the engineering standpoint and productive of 10% bad runs. The source demanded in a conventional reactor is there to prevent "fast-burst" operation, so it is not surprising that to produce fast bursts with a source present requires something like an antitank gun for a control rod drive. In most installations, close procedural control of the reactor during the few seconds wait between assembly and excursion has seemed preferable to such heroic rates of fuel insertion. (There is a mode of operation wherein the source level is so high that burst initiation is assured as soon as the rod starts in. Pulses are low and wide, however, and are not the type of output considered here.)

## Safety Considerations

Instrumentation is employed in a fast-burst reactor to mechanically reduce the reactivity. The over-all response of this system is far too slow, however, for it to be effective in terminating the burst, which is limited by the temperature coefficient. The instrumentation merely serves to hold the reactor subcritical after the completion of the burst.

Since the safety of a fast-burst reactor is dependent upon the thermal expansion of the fuel, it is prudent to inquire how far such protection extends. The fraction of the burst appearing as kinetic rather than thermal energy is almost proportional to the size of the burst. The potential for damage thus goes up in a ratio closely proportional to the square of the energy release. The energy release, in turn, in the region of $1 above prompt critical goes up as approximately the cube of the reactivity increase. Hence the explosive energy goes up as the fifth power of the reactivity addition, and, as expected, this type of reactor proves very sensitive to changes in the burst rod.

A total reactivity insertion of $1.10 above delayed critical can be used in routine burst production; accidental insertion of $1.20 above delayed critical destroyed the original Godiva.[3] Reactor safety thus hinges on precise knowledge of the amount of reactivity added to the system when the burst rod is shot in. Once the transient has started, there is no way of terminating a too-large excursion.[11] The usual sequence of burst production[3] involves first bringing the reactor to delayed critical at some low power. This is done with an adjustable fuel rod in the presence of a source. The reactor is then disassembled, the source is removed, and the neutron population is allowed to decay to close to the spontaneous fission level. The reactor is then reassembled to achieve the same physical assembly which some 20 or 30 min before gave a critical assembly, and the burst is then achieved by shooting in a piece of fuel that was previously accurately calibrated to be worth a dollar and some few cents.

The dangers inherent in such a setup are well recognized.[10] In the time during which the neutron population is allowed to decay, some change

might occur in the reactor environment that would increase its reactivity when reassembled to above what was supposed to be a delayed critical condition. For example, the temperature could fall, an experimental animal could move closer to the reactor, or a broken hydraulic line could pour oil onto a piece of the fuel. Such an occurrence could put the assembly well above delayed critical during the period before the burst rod was inserted, and, indeed, this type of incident was responsible for the end of Godiva I.[11,12]

The importance of administrative control is evidenced by the many facets of the safety of the reactor which cannot be made dependent upon instrumentation. Where changes in the reactor's surroundings are made intentionally, as by the introduction of equipment for a new experiment, the reactivity can, of course, be suitably adjusted. Nonetheless, a subtle source of error is still inherent in the calibration of the burst rod. The worth of the burst rod is a function of the reflecting or moderating materials in the vicinity of the reactor and must be freshly determined for each major change in the environment. Since a previously calibrated shim rod would also be affected, such recalibration of the burst rod can only be done by a critical experiment type of operation in which the burst rod reactivity is restricted to a nominal even dollar, instead of the few cents over a dollar it is usually worth. With the experimental setup in place, the reactor is brought to delayed critical, allowed to decay, and the burst rod is fired, just as in conventional operation. In this case, however, the rod is worth only about one dollar, so the burst will be small and will provide an exact calibration of the worth of the rod because a definite experimental relationship exists between the size of the burst and the worth of the burst rod.[3]

After this calibrating burst, the reactor can be disassembled, allowed to decay, and reassembled in the presence of a source to a configuration some few cents above delayed critical. The exact reactivity used will be such that when combined with the newly calibrated burst rod it will give the desired burst. This small reactivity increase above delayed critical can be adjusted by measuring the value of the slight positive period the reactor assumes when given a few cents of reactivity. In this manner, the worth of the final burst can be programmed with the experimental equipment in place at the expense of minor irradiation of the target prior to the final burst.

The calibration procedures described must be carried out at very low powers in order to avoid temperature effects in the core and to minimize the waiting times for decay of the flux. Measuring reactor periods at such low powers demands extremely sensitive and accurate counting channels. Thus the requirements for pulse-reactor instrumentation are more stringent in some respects than those for conventional reactors, despite the fact that once a burst is under way the instrumentation does not terminate the excursion.

## Conclusion

A fast-burst facility is one that is deliberately operated as a runaway reactor or as a time bomb. Such operation necessarily violates many rules developed for safety in reactors designed for steady-state operation. Recognition of this situation leads to stricter administrative control than is considered necessary for most reactors. The administrative control consists of close surveillance of experimental installations and procedures.

Although it is an inherent characteristic of fast-burst reactors that the burst may be limited by the temperature coefficient, the magnitude of the burst is dependent upon the effectiveness of administrative control.

(R. S. Stone)

# LATCH MECHANISMS FOR CONTROL ROD DRIVES

Latch mechanisms for control rod drives are the product of some combination of mechanical, electrical, pneumatic, hydraulic, or other type of equipment. A safety analysis of the reactor system characteristics establishes the required control rod system performance and therefore the required speed and reliability of the latch.

The latch is the device by which the control rod is attached to or detached from the rod drive. Thus it is the latch that must maintain a firm and reliable attachment of the control rod to its drive during normal control rod insertions and withdrawals and must also assure release of the control rod from the drive when a scram signal is received. For each reactor there is some minimum time following initiation of a signal to scram within which the control rods must start inserting or reach some specified minimum depth of insertion. This so-called "scram time" depends on the type of reactor and its nuclear environment. The allowable time is governed by experimental conditions and the postulated credible accidents that could cause unscheduled or unsafe rates of reactivity increase. The time required for release of the control rod by the latch system can, of course, be only a fraction of the total scram-time allowance. Electrical and other components involved in the scram signal and execution sequence must also be allotted proportionate shares of the maximum allowable scram time within which to function.

In reactor systems that demand exceptionally fast response to a scram signal, the selection of a latch concept and its development into a detailed design requires ingenuity and care. Unfortunately there is little published

information on specific control rod release and insertion times for existing reactors that could serve as a guide for the designer.

For purposes of this discussion, a fast-response time is considered to be an interval substantially less than 1 sec. Examples of reactors that require fast-response control rod systems to afford protection against the reactivity increases which are considered credible [13,14] are the high-flux research and test reactors, including the ETR, MTR, and ORR. Since the scram signal is usually delivered as an electrical impulse, some part of the allowable response time is required for the signal to clear through the electrical system and reach the latch mechanism. This electrical system is frequently elaborate to ensure against misoperation resulting from events such as voltage fluctuation, failure of individual components, such as vacuum tubes, or other disturbances.

Magnetic Latching Mechanisms

One of the simplest latch arrangements and one which adds a minimum of response time to that required for the signal to pass through the electrical system is a direct-lift magnet. In such a system the control rod is attached to the control-rod positioning drive solely by magnetic force, and reduction or interruption of the power supplied to the electromagnet frees the control rod to drop into the core under the action of gravity. If a control rod acceleration greater than 1 g is required, accelerating springs or other loading devices may be used. If the required accelerating force is very large, however, it is likely that some other form of latch will have to be used because it is not generally practical to support large loads with a direct-lift magnet. In order to achieve very short release times

in magnet-supported systems, the magnetic-force field must decay rapidly after the electric power is reduced or interrupted. Therefore materials with low residual magnetism should be used. It is possible not only to design systems in which the magnet current is subject only to on-off control but also to design systems in which some specified magnet current is normally maintained at 100% reactor design power and successively reduced as a function of increasing reactor power to cause a scram at a predetermined level in excess of the reactor design power level.

Mechanical Latches

Some form of mechanical latch may be used if a direct-lift magnet is not suitable or desirable. The design of a mechanical latch requires particular attention to the mass of the moving parts and to the total travel required to execute a scram action. Both relate directly to the time required to release the control rod in response to a scram signal. The force required to accelerate a latch component is proportional to its mass, and as the total travel increases, the acceleration required to make the traverse in a given time increases. When fast release times are required, a sufficient increase in the force actuating the release mechanism could be specified to move relatively massive parts through any reasonable travel distance. In actual practice, solving these two problems of mass and travel distance by using increased force on the latch components is reasonable only to a limited extent. Massive parts accelerated to execute rapid motions must also be decelerated without damage. The high forces involved may require heavy design, which, in turn, requires higher forces. The requirements for the design of a fast-acting mechanical latch system

which emerge from these considerations look, in some respects, very much like a specification for a mouse trap or other hair-trigger device. The additional requirements for reliability demands that the development and selection of a suitable design concept be supported by a thorough analysis of inertia, force, and friction factors. Materials, material hardness, and surface finishes must also be critically evaluated. These analyses must be particularly meticulous for systems in which key latch components cannot be lubricated. In applications in which static contact of latch fingers, balls, or other components under load exists, such problems as inelastic deformation and fretting as a result of vibration, corrosion, and erosion must be considered. Materials must be selected that will ensure against roughening which could result in sticking.

The most rigid reliability criterion is that the latch never misoperate in such a way as to cause failure to scram, and the need to provide a latch which will not cause false scrams is only slightly less critical. Closely associated with the reliability criteria are those which call for testing of the scram performance of the control rod system during reactor operation. Ideally the intent of such tests is to establish proper functioning of the entire scram mechanism without subjecting the operating reactor system to an actual scram. Without a secondary latch or lock on the control rod to hold it in position when the latch is released, it is not possible to test the latch unless it is of such a design that it can be momentarily released and re-engaged before the rod has traveled more than some permissible amount. Since such locks or secondary catches could prevent a scram when it is intended, they are often unacceptable and not permissible as auxiliary latch features to permit

such tests. Whether or not the momentary release and re-engagement arrangement can be used for testing depends upon the control rod system design. It would be difficult to trip a mouse trap momentarily with the intent of recatching it in the cocked position. On the other hand, a control rod suspended on a cable might be momentarily released and stopped if the latch were in the form of a suitable clutch. It is frequently more difficult to provide an operating test for fast-response control rod latch systems than for those with slower characteristics.

Design Analysis

The control rod release mechanism that employs a ball-latch device is examined analytically below to indicate in detail the fine structure of the operating conditions and characteristics of a typical latch. The latch is in the form of a ball locking arrangement in the head of a cylindrical drive tube, as shown in Fig. 6. This drive tube inserts into the tubular lower end of the control rod which has an internal shoulder ring. The drive tube passes freely through this shoulder when the balls are retracted into the drive-tube head. When the balls are displaced radially outward in their cage holes in the head, they engage the shoulder and pick up the control rod. A scram is effected by retracting the balls and allowing the shim rod to drop over the drive tube. The position of the balls in the latch is governed by a tapered cylindrical plunger located inside the drive tube and mounted at the top of a push rod which extends into a region where a release spring is attached. The release spring acts to pull the push rod downward to permit ball retraction and control rod scram. A magnet arrangement opposes and overrides the release spring action, holds the

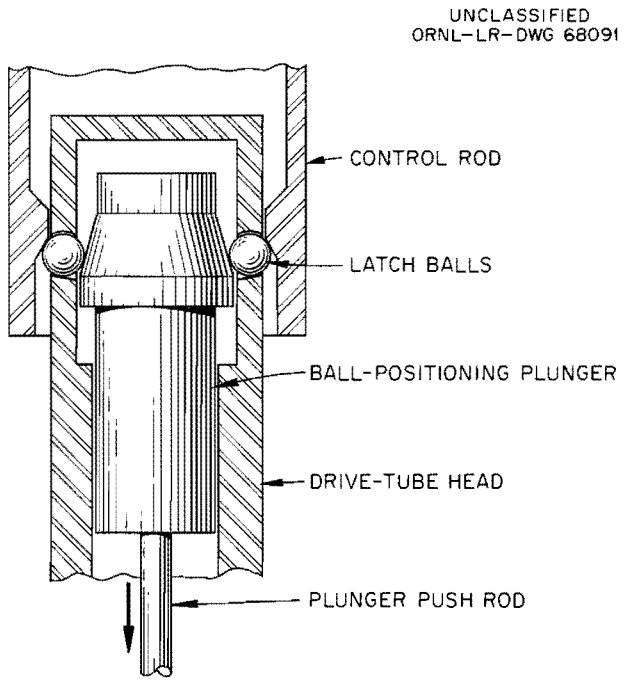


Fig. 6. Schematic Arrangement of Ball Latch for Control Rod Positioning and Release.

push rod up, and keeps the balls engaged to the control rod shoulder until a reduction in magnet current by the control system permits the release spring to lower the push rod and plunger, retract the balls, and release the control rod.

The transmission of both mass and hydraulic loads through the ball latch to the push rod and magnet system has been evaluated by developing and solving a generalized system of equations to define the direction and magnitude of the forces acting at three ball-contact points. Nine quantities, three angles, three forces, and three friction factors were treated as variables to identify the design which minimized the ball-to-plunger force, $P'$. Binding between these components had been identified as a cause of slow operation. The force system around a typical ball is shown in Fig. 7. The symbols used in Fig. 7 have the following meanings:

$F$ = Vertical force imposed on the balls by the control rod, lb

$f$ = Component of $F$ acting perpendicular to ball surface, lb

$f'$ = Resultant of $f$ and $f\mu_f$, lb

$f\mu_f$ = Friction force between control rod and ball, lb

$h$ = Horizontal component of $F$ when $F$ is resolved into $f$, $f\mu_f$, and $h$, lb

$H$ = Force acting perpendicular to ball cage hole in head at point of ball contact, lb

$H'$ = Resultant of $H$ and $H\mu_H$, lb

$H\mu_H$ = Friction force between ball cage and ball, lb

$k$ = Length of perpendicular to $P'$ passing through X, lb

$P$ = Force acting perpendicular to plunger at point of ball contact, lb

$P'$ = Resultant of $P$ and $P\mu_P$, lb

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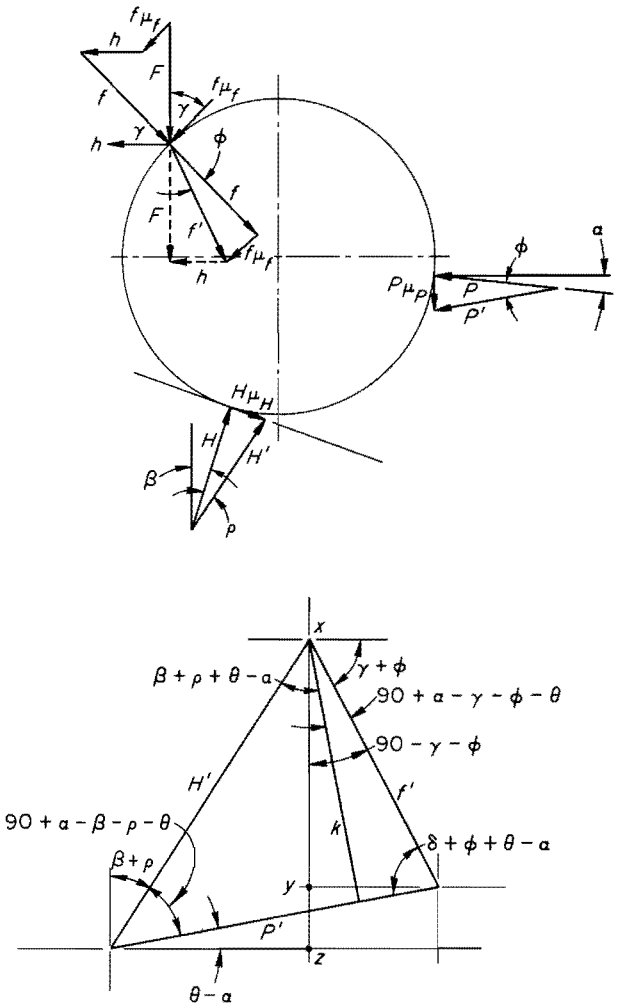


Fig. 7. Force System Around a Typical Ball in a Ball Latching Mechanism. (See text for definition of symbols)

$P\mu_P$ = Friction force between ball and plunger, lb

$\alpha$ = Angle of ball contact surface on plunger from vertical or angle between P and horizontal, deg. $\alpha$ is zero when P is horizontal and increases positively as P assumes increasing negative slopes.

$\beta$ = Angle of ball contact surface on head from horizontal or angle between H and vertical, deg. $\beta$ is zero when H is vertical and increases positively as H assumes decreasing positive slopes.

$\gamma$ = Angle of ball contact surface on control rod from vertical or angle between f and horizontal, deg. $\gamma$ is zero when f is horizontal and increases positively as f assumes increasing negative slopes.

$\phi$ = Arctan $\mu_f$, where $\mu_f$ is the coefficient of friction between ball and control rod

$\theta$ = Arctan $\mu_P$, where $\mu_P$ is the coefficient of friction between ball and plunger

$\rho$ = Arctan $\mu_H$, where $\mu_H$ is the coefficient of friction between ball and ball cage

The analytical relationships can be written as four equations:

$$f' = \frac{F}{\cos\phi\,(\sin\gamma + \mu_f \cos\gamma)}\,, \tag{1}$$

$$P' = \frac{F\,\{\mu_f[1 + \sin(\gamma - \beta)] - \cos(\gamma - \beta)\}}{\cos\theta\,(\sin\gamma + \mu_f\cos\gamma)\{\mu_P[1 + \sin(\beta - \alpha)] - \cos(\beta - \alpha)\}}\,, \tag{2}$$

$$\operatorname{Tan}(\beta + \rho) = \left[\frac{\sin(\alpha - \gamma - \phi - \theta)}{\sin(\gamma + \phi) + \frac{P'}{f'}\sin(\theta - \alpha)} + \cos(\theta - \alpha)\right] \times \frac{1}{\sin(\theta - \alpha)}\,, \tag{3}$$

$$H' = \frac{-f'\sin(\alpha - \gamma - \phi - \theta)}{\cos(\beta + \rho + \theta - \alpha)}\,. \tag{4}$$

Figure 8 displays the solutions of these equations for $\beta = 17.5$ deg as a family of "carpet" plots. Each carpet is the locus of all points corresponding to a given pair of values for $\mu_f$ and $\mu_P$, which are treated as independent variables. The corresponding values of $\mu_H$ are dependent quantities. This analytical study is essential to the understanding of ball latch design and operation, and the relationships presented above are general and applicable to any ball latch of the type shown, regardless of ball diameter.

Performance of ORR Ball-Latch Device

There have been instances in which the release of an ORR control rod has been delayed by more than 100 msec following reduction of magnet current to the ball-latch mechanism. There have also been occasions on which the control rods have dropped when no magnet current reduction took place. In an effort to explain these events, parallel analytical and test programs were carried out. It has been shown analytically and confirmed experimentally that mechanical friction between the balls and plunger surface in the latch can more than offset the force of the release spring and delay release of the control rod following current reduction to the magnet. The foregoing analytical study has provided a basis for selecting new contact surface angles which should eliminate the problem of delayed scram action.

The problem of unscheduled scrams can be explained theoretically by a number of arguments, and one explanation has been demonstrated in a test stand. The long push rod may lead or lag the surrounding drive test in temperature changes and on successive cycles "ratchet" the balls into a locked position in which the push rod builds up a compressive force that

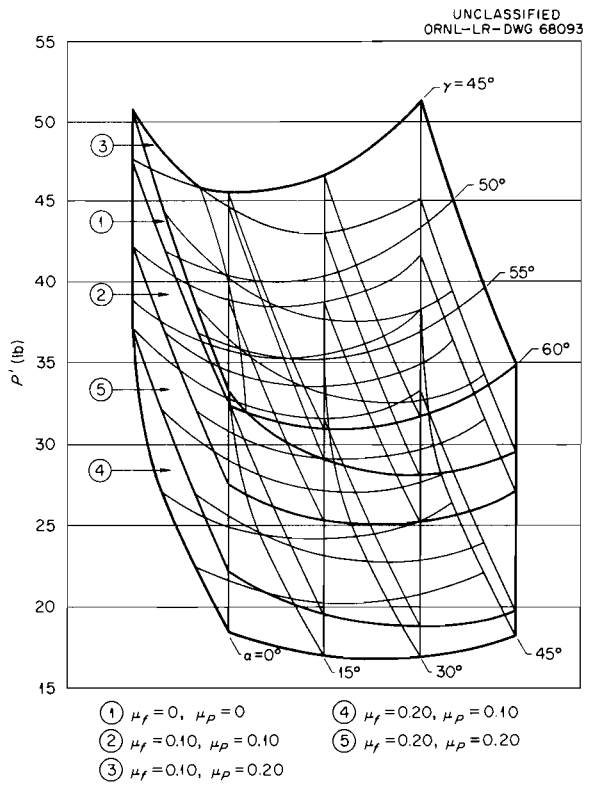


Fig. 8. General Solution to the Force Equations for a Ball Latch with B = 17.5 deg.

eventually exceeds the holding force of the magnet and causes an unscheduled rod drop.

There is a portion of the push rod that passes through a bellows seal assembly and which can be subjected to extreme tensile loads if certain nuts are overtorqued during assembly. The subsequent gradual increase in push rod length as creep relieves this tensile stress acts against the magnet and may also contribute to unscheduled scram action. Significant lengthening of these rods has been demonstrated within the range of torques formerly used in assembling these mechanisms. Work on these mechanisms continues, and test assemblies are being built for further studies of modifications of the present arrangement and some new design features. (J. Foster)

# CONTROL ASPECTS OF THE SL-1 ACCIDENT

Although the course of events which resulted in destruction of the SL-1 may never be determined with certainty, it is now considered probable that the accident was initiated by hand withdrawal of the central control rod.[15-18] If this were the case, the conclusion is inescapable that the reactor control system, which was intended to make reactor operation safe and convenient, was in fact the cause, or at least a contributing cause, of a fatal accident. It is the purpose of the present review, using what is known or can reasonably be conjectured about the SL-1, to identify areas in which its control system caused or contributed to its destruction, and to summarize the lessons in control-system design that have been made evident.

Because of the lack of certain knowledge about the sequence of events, the reviewer has been forced to reconstruct, to postulate, and to guess about the role of some of the items discussed. In order to draw useful conclusions, it suffices to demonstrate that a given sequence of events is plausible. By contrast, investigating boards must stick to proven facts in assigning responsibility and liability.[18] A previous article[19] reviewed the accident in light of the proven facts.

The SL-1 control system will be evaluated here in the light of the state of the control art in 1962, since much has been learned. It is recognized that some of the matters discussed herein are controversial. The opinions expressed are those of the reviewer, based on long-established philosophy and practice at Oak Ridge National Laboratory.

Reactivity Shutdown Margin

It is clear that the SL-1 failed in an essential way to meet requirements regarding shutdown margin, since the reactor did in fact go critical unexpectedly. The shutdown margin of any reactor is determined by the loaded excess reactivity and the total rod worth. The performance of the reactor — its power, or life, or operating cost — is often critically dependent on the excess reactivity available. If the rod worth is fixed by other conditions, the designer may be faced with the dilemma of trading performance (excess reactivity) against safety (shutdown margin). The performance is easily measured; the safety is intangible.

No universally recognized standard exists regarding minimum shutdown margin. Schultz[20] suggests a minimum of 2% and a "usual range" of 5 to 10%. His reactivity data for 21 power reactors show that their shutdown margins vary widely. A recent compilation by Bates[21] for 19 power reactors in the United States shows two with negative margins that require auxiliary chemical shutdown; for the other 17 the margins vary between 2 and 8.8%. It is difficult to discern any consistent set of criteria in these scattered values.

Bates has suggested[21] the following criterion for minimum shutdown margin: "No single error in manipulation of rods or fuel in the reactor shall cause it to go critical inadvertently." Certainly this criterion is a basic one. The shutdown margin must be at least adequate for the maximum change in reactivity that can be produced by withdrawal of a single control rod. But the shutdown margin has more general significance; its function is to preclude criticality during such operations as refueling, control-rod replacement, and core or rod-drive maintenance. Bates' criterion

represents a necessary, rather than a sufficient, condition, and the margin actually required in any reactor must be based on many factors besides the worth of the most important control element.

In the SL-1, as originally designed, the shutdown margin was inadequate even to cover the accidental withdrawal of the central control rod. The original excess reactivity requirements were as follows:[22,23]

| | |
|---|---|
| Steam voids | 1.3–2.0% δk |
| Equilibrium Xe and Sm | ~3 |
| Temperature defect | ~2 |
| Operation | 2–3 |
| Total | 8–10% δk |

The unpoisoned core, as fabricated, had an excess reactivity much greater than this to allow for burnup; burnable poison worth 11% was to be added. The worth of the poison was later measured[22] to be 12.5%. The worth of all the rods was estimated to be 15% in the cold, clean reactor. The design shutdown margin was thus about 5%. A later analysis based on experimental data[24] gave the cold, clean shutdown margin as 4.3% and predicted a minimum margin of 3.1% at the worst point in the burnup cycle.

Although 3 or 4% may seem to be an adequate margin, the great worth of the central control rod must be considered. In the original design (Ref. 22, p. 215) "it was calculated that the cold fresh reactor could not be shut down with the center rod at 30 inches" (i.e., fully withdrawn). This situation was confirmed experimentally (Ref. 22, p. 217) by bringing the cold fresh reactor to criticality upon withdrawing the central rod 19.1 in.

It must be concluded that the shutdown margin of the SL-1 as designed and built was inadequate according to Bates' criterion. The reviewer is

convinced that this criterion is reasonable and that it states shutdown margin requirements in terms that relate correctly to reactor requirements. In the light of this accident, future designs which incorporate inadequate shutdown margin should be demonstrably safe on some other basis.

Rod Drives

The SL-1 rod drives are described best in Ref. 17. The rods were driven by a rack and pinion located in a thimble ("bell housing") above the upper head of the reactor vessel. The pinion shaft penetrated the thimble wall via a rotating seal and was driven by a motor and gearbox through a clutch; de-energizing the clutch coil released the pinion from the gearbox and permitted the rod to fall. An auxiliary clutch permitted the motor to drive a released rod downward and prevented upward rod motion after release.

In the rod-drive mechanism, as originally conceived, the downward force of gravity was available at all times to push the absorbed rod into the core and to hold it there, and no significant upward force opposed gravity. No differential pressure existed across the rod as a consequence of pressurized reactor operation. The pressure drop across the rotating seal exerted no force on the rod. However, the upper end of the thimble had to be removed routinely for access to the mechanism and to the core. With the thimble removed, transient pressure rises in the vessel would result in transient upward forces on the rods in the direction of increasing reactivity. This provided a mechanism whereby an accident could be made worse by the action of the control rods during the accident. Some of the SL-1 rods were in fact forced upward during the accident.[15-16]

It is not often stated explicitly, even in reactor hazards documents, that the thimbles are essential to the safety of rod drives of this type. The reactor was thus unknowingly placed in a condition more hazardous than normal every time the thimbles were removed. In other reactors using this type of drive it would seem prudent to devise operating procedures to minimize the added hazard when the thimbles must be removed. It should be noted that, for absorber rods such as those in SM-1, which are driven from the bottom of the reactor[25] but which are raised to increase reactivity, any unbalanced pressure forces are in the direction of reducing reactivity. This is obviously preferable to the SL-1 design from the hazards standpoint.

In order to remove the thimbles and gain access to the rod drives or the reactor, the pinion drive shaft components had to be disassembled. This left the rod without the protection against upward motion which might have been afforded by the auxiliary clutch. This illustrates the fallacy of attempting to correct a fundamental weakness with an added device — in this case a clutch — which must inevitably be disabled for access to the machinery.

The most obviously objectionable feature of the SL-1 rod drive was, of course, the necessity for lifting the rod by hand and holding it with a "C" clamp while coupling or uncoupling the rod and the drive mechanism.[17,18] This feature was not essential. Other rod drives of this type lack it and are thus better designed in this respect. Hand lifting violates several elementary safety criteria:

1. The rod cannot be dropped by the safety system.
2. Rod withdrawal is not supervised by the control system.

3. Rod withdrawal speed and amount are not limited or controlled.

4. Rod position is unknown to the console operator and can only be estimated by the man doing the lifting. The latter is also unaware of the readings of the reactor instruments and may not understand the danger of the operation he is performing.

Of course, control rods must be removed and replaced because of burn-up, but this is a carefully programmed job that is made safe in most installations by removing the fuel from the core. The routine, casual, frequent necessity for handling lifting of the SL-1 control elements was an unnecessary danger in the reviewer's opinion and denotes unsound design.

The difficulties which were experienced with stuck rods in the SL-1 have been discussed elsewhere.[17,18] At the least, such a succession of troubles makes one wonder whether the design was adequate and also whether the rod drives were performing as designed at the time of the accident.

Conclusion

The possibility that the central rod was stuck (Ref. 18, p. VI), the certainty that it had to be lifted by hand, its excessive reactivity worth relative to an inadequate shutdown margin, the possible unbalanced pressure forces tending to increase reactivity — all weaknesses in the control system of the SL-1 — lead to the conclusion that the accident must be attributed in considerable measure to control system failure. The lessons of this accident should be applied to similar hazards in existing reactors and to eliminate these weaknesses from future designs. (S. H. Hanauer)

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# IV. PLANT SAFETY FEATURES

# REACTOR CONTAINMENT DESIGN

In the design and construction of a nuclear plant, precise analytical, inspection, and testing procedures are followed to ensure the maximum degree of integrity. Consequently, the probability of mechanical failure of most nuclear plants is much lower than that for conventional power plants and chemical facilities. Experience with conventional systems has included enough failures of pressure-containing devices, however, to indicate that there is a significant probability of a failure which could result in the release of large amounts of radioactivity. Hence, it has been required of the nuclear industry that an ultimate, last-ditch means be provided for containment of the radioactive materials which may be released by such a failure. At the present time most reactors are contained in steel pressure vessels which add from $15 to $50 per kilowatt to the capital cost.[1] In efforts to reduce this capital cost, other methods of containment have been proposed and incorporated into the design of several plants now under construction.

The design of an adequate containment system requires selection of the worst credible accident that could occur from the standpoint of release of radioactive products to the atmosphere and evaluation of hazards to the general public associated with that accident. The "maximum credible accident" (mca) commonly associated with most reactor systems is that in which the primary coolant line is ruptured in such a way as to permit full discharge of the coolant from both ends of the pipe. The first effect of the accident is release of the energy stored in the primary cooling system. The containment system must either withstand the energy release and resulting pressure rise that will occur immediately after the accident or it must

release air and vapor to the atmosphere in such a fashion that the release of radioactivity will be limited to the permissible amounts. In this accident, release of the bulk of the radioactive fission products from the reactor would occur unless emergency cooling was provided to prevent fuel-cladding failures after the coolant drained from the core.

Another type of maximum credible accident is that in which a nuclear excursion within the reactor core could cause a pressure surge within the reactor vessel. This might cause brittle failure of the pressure vessel and create missiles of parts of the vessel and the primary shield. Flying missiles might then penetrate the containment vessel and cause uncontrolled leakage to the atmosphere.

Hence, two types of attack on the integrity of the containment system are conceivable: (1) the buildup of pressure and (2) penetration by missiles. These problems have been discussed extensively in the literature, and various research programs have been inaugurated to determine their causes, their effects, and means of controlling the hazards involved.

It was suggested by R. O. Brittain[2] in 1959 that reductions of the order of $10^2$ to $10^5$ in estimates of criteria for reactor containment design could be achieved by (1) increased accuracy of determination of maximum energy release, (2) more accurate appraisal of rate of release, (3) better information on motion of radioactive materials after release and on effects of fallout, precipitation, condensation, and decay, and (4) better information on availability of radioactive materials for release. A review of the literature since that time has indicated that much effort has been spent in attempting to improve the accuracy of determination of mechanical effects of failures of nuclear systems. Experiments have been

reported on measurements of pressure rise caused by release of thermal energy from vessels inside containers[3] and on tests of conversion of blast energy into missiles and shock waves[4] and their damage to containers.[5] Means have been developed for protecting containment vessels against damage from flying missiles,[6] and much theoretical work has been done to increase the accuracy of stress analyses of shells subject to the types of load anticipated.[7] There has been release of radioactivity from one run-away reactor (SL-1, Idaho, January 1961)[8] and several incidents of release of activity from process equipment.[9] The analyses of these accidents have added greatly to the fund of knowledge concerning causes and results of accidents.

Energy Release

A method for determining the energy release and consequent pressure rise in a failure of primary coolant lines for a boiling-water reactor is given by Johnson and Nelson.[1] The following assumptions were made for the analysis:

"1. The sudden expansion of the pressurized coolant causes part of the liquid to vaporize and instantaneous pressure rise is assumed.

2. Adiabatic expansion takes place. Heat is neither added to nor removed from the system.

3. A homogeneous saturated vapor-air mixture exists after the accident.

4. Sufficient time is assumed to have elapsed after the accident so that an equilibrium condition prevails. Energy is confined to the thermodynamic properties of the air and coolant.

5. The validity of the perfect gas law is accepted.

6. The containment vessel is able to withstand the initial pressure shock wave. Calculation is for a thermodynamic steady state.

7. The design of the containment vessel conforms to the ASME Boiler and Pressure Vessel Code, Section VIII Unfired Pressure Vessels and to the applicable case interpretations.

8. Dry air is prevalent before the accident."

The results of calculations of equilibrium pressure and temperature conditions in a containment vessel as functions of initial internal energy of the coolant and of the free containment volume per pound of coolant released are presented in Figs. 9 and 10, which were adapted from the work of Johnson and Nelson.[1] Figure 9 is applicable to pressurized-water and boiling-water reactors with standard containment vessels, i.e., steel containment vessels designed to withstand high pressures. Figure 10 applies when a low-pressure containment structure is used, i.e., a structure designed to withstand pressures of 5 psig or less.

In 1959 Sargent and Lundy conducted tests[3] in an underground containment shell 14 ft in diameter and 32 ft high that was designed for an internal pressure of 100 psig. Inside the containment shell was a drum 42 in. in diameter by 23 ft long that was designed for an internal pressure of 700 psig. The drum was in a vertical position and had a number of 12-in.-diameter openings along the side and one in the bottom. In the tests the drum was filled with various amounts of water heated to a boiling pressure of approximately 600 psig by steam. A rupture disk on one 12-in. opening was triggered at 600 psig. In some tests two diaphragms were burst. Temperature and pressure readings at various points in the containment vessel and the drum were displayed in the instrument house within 0.01 sec. Tests

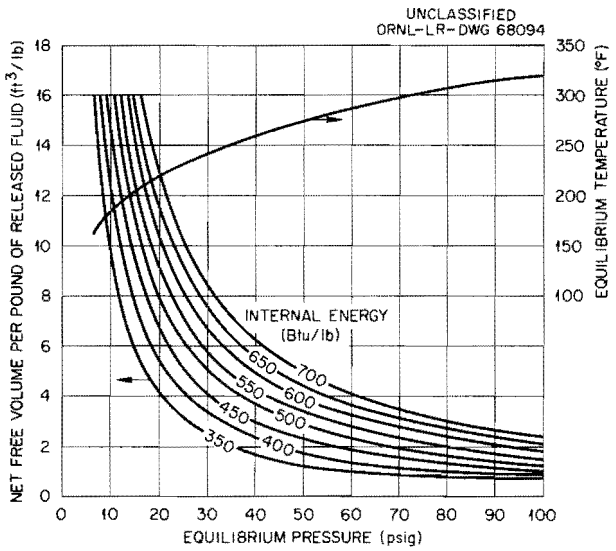


Fig. 9. Calculated Standard Containment Vessel Equilibrium Pressure and Temperature Following a Loss-of-Coolant Accident in a Water-Cooled Reactor as a Function of Internal Energy Release.

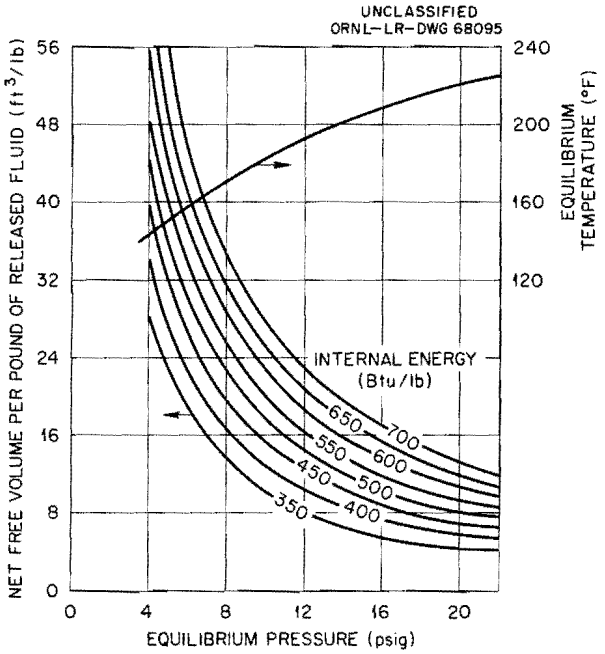


Fig. 10. Calculated Low-Pressure Containment Structure Equilibrium Pressure and Temperature Following a Loss-of-Coolant Accident in a Water-Cooled Reactor as a Function of Internal Energy Release.

were made with various amounts of cold water in the bottom of the containment shell. Of 10 tests reported, five were made with two outlets from the drum open and five with only one. The maximum pressures measured in the tests were in all cases lower than the calculated pressures. The ratios of measured-to-calculated pressures varied from 0.03 to 0.35 with water in the containment shell and from 0.7 to 0.87 with no water in the shell. These results indicate conservatism of from 10 to 30% in the theoretical estimates of pressure rise from the loss-of-coolant accident in a water-cooled reactor.

Reactor Containment Design Studies

A study was also made by Sargent and Lundy Engineers[1] for the purpose of evaluating the merits of four types of containment with respect to technical aspects and from the standpoint of costs for three different sizes of boiling-water reactor. The four types of containment studied were (1) the standard steel containment structure, vapor tight to 15 to 60 psig; (2) a pressure-relief containment system, leaktight but relieved by burstable diaphragm or valve in a vent duct that can be closed after a short period to prevent release of products of core meltdown; example, Canadian NPD-II;[10] (3) a pressure-suppression containment system[11] in which the reactor and piping are contained in a dry well that is vented through larger pipes to a pool of water which serves as a heat sink to condense vapors; example, Humboldt Bay Plant of Pacific Gas and Electric Company;[12] and (4) a low-pressure containment structure consisting of a vapor-tight shell of large enough volume to enclose the entire plant and to prevent the equilibrium pressure from exceeding 5 psig in event of a loss of coolant accident; example, BONUS.[13] The three sizes and types of boiling-water

reactor plant studied were (1) a 44-Mw electrical-output natural-circulation direct-cycle plant, (2) a 180-Mw electrical-output dual-cycle forced-circulation plant, and (3) a 300-Mw electrical-output dual-cycle forced-circulation plant. Table IV-1, adapted from the report,[1] shows the cost comparison of the various containment designs for the three sizes of reactor in dollars per kilowatt. The pressure-relief containment system is the only one that shows a cost advantage in comparison with the standard containment vessel for all three sizes of reactor. The authors explain that "pressure-suppression in the larger sizes was not economically feasible because of the particular type of reactor cycles studied which require a very large dry well to contain the reactor primary equipment."

Table IV-1. CONTAINMENT COST COMPARISONS

| Plant Electrical Output (Mw) | Cost of Containment ($/kw) | | | |
|---|---|---|---|---|
| | Standard Vessel | Pressure Relief[a] | Pressure Suppression[a] | Low-Pressure Structure[a] |
| 44 | 31.05 | 12.90 | 17.90 | 74.85 |
| 180 | 30.11 | 6.46 | 40.65 | 82.95 |
| 300 | 20.82 | 4.89 | 27.04 | 54.46 |

[a]Costs include adjustments for differences in piping, electrical installation, fuel handling, etc., as well as differences in the containment structure.

Plans by Pacific Gas and Electric Company to use pressure-suppression containment at both Humboldt Bay (47 Mw) and Bodega Bay (325 Mw)[14] are based on the assertion that "pressure suppression is saving us money." This conclusion does not seem to be supported by the Sargent and Lundy study. The considerations leading to the high cost estimate were, however, that the reference design used, which consisted of the dual-cycle

high-stream-drum design, required a spherical dry well equivalent in volume to the standard containment vessel and a large suppression chamber. A cylindrical dry well was considered, but only a small reduction in volume was possible and the cost was estimated to be higher than for the spherical shape because of the heavier wall required. Use of a forced-circulation direct-cycle with internal steam separation at Bodega Bay[14] could result in a considerable reduction of the dry well volume and thus effect the saving claimed for it.

A qualitative comparison of the technical aspects of the four types of containment is given in Table IV-2, which was also adapted from Ref. 1. The report states that "all four methods have been found technically feasible and acceptable to the various governmental regulatory bodies concerned." Standard containment has, however, received the widest public acceptance. It is therefore interesting to note that in most respects the other three schemes are judged equal to or better than standard containment.

A pressure-relief type of "confinement" has been approved for the New Production Reactor (NPR)[15] at Hanford on the basis of the ACRS conclusion

> "that the confinement structure proposed for the NPR was not suitable for a less isolated location [although it] will provide a higher degree of protection against the release of fission products than that afforded by the building structure around the other Hanford reactors and therefore becomes acceptable on the basis that its use will not significantly increase the present hazards of the Hanford operation."

This system is based upon the principle of relieving the initial pressure discharge following the mca to the atmosphere through large specially designed vents and the premise that the major portion of the activity

Table IV-2. TECHNICAL COMPARISON OF CONTAINMENT CONCEPTS

| Technical Aspects Considered | Relative Merit of Containment | | | |
|---|---|---|---|---|
| | Standard | Pressure Relief | Pressure Suppression | Low Pressure |
| Containment of fission products | 2 | 3 | 1 | 2 |
| Direct radiation following accident | 3 | 2 | 1 | 3 |
| Containment of released coolant | 2 | 4 | 3 | 1 |
| Containment of primary and auxiliary systems | 2 | 2 | 3 | 1 |
| Accessibility to equipment during operation | 1 | 1 | 2 | 1 |
| Reactor building penetrations | 3 | 3 | 2 | 1 |
| Missile protection | 2 | 1 | 1 | 3 |
| Adaptable to underground design | 2 | 1 | 1 | 3 |
| Cleanup following accident | 2 | 1 | 1 | 3 |

release will not accompany the initial pressure release. The design pressure of the confinement structure is 5 psig. Large vents containing normally open butterfly valves are normally closed by a sealed cap which can be blown free at approximately 2 psig. After release of the pressure the butterfly valve would be closed by a pressure switch. The normal ventilation system is equipped with sliding-gate closure valves in the larger ducts and with water traps in some of the smaller ones. All ventilation air is discharged through filters to the building stack. It would be possible after an accident to maintain a slightly negative pressure in the building by exhausting air through the filters to the stack. This would eliminate possible

buildup of pressure because of decay heat and prevent release of activity through possible leaks which might develop in the building structure. Calculations have shown that the first fission-product release will not occur until about 5 min after the initiation of the mca by the double-ended primary system piping failure. A fog-spray water system is provided to reduce the temperature in the building in the event that it tends to rise above 70°F. All penetrations of the confinement system are doubly sealed, generally with a single mechanical valve with some kind of backup mechanism, such as a balloon seal manually inflated and released or a water trap.

A review of containment capabilities of other reactor building structures in existence at Hanford has been made,[16] along with tests of prototype structures. It was determined that roof tiles should be restrained by using small clips attached to the tile with epoxy adhesives. The concrete block curtain walls of the reactor buildings appeared to be capable of resisting internal pressures of 0.3 psi without additional reinforcement, except at openings such as doors and windows. Caulking and neoprene or Hypalon coating materials were suggested as means for sealing the structures.

The low-pressure containment system proposed for the BONUS Nuclear Power Station consists of a steel dome anchored and sealed to a 42-in.-thick concrete base slab.[17] Since the concrete base slab thus becomes a part of the containment shell, it was considered necessary to investigate the air permeability of the concrete and of the coating materials and joint compounds which might be used. The leakage tolerance specified for the building was 10 in.$^3$/hr·ft$^2$ at a pressure of 6 psig. Tests at the site on 12-in.-long test cylinders showed air permeability of 7.7 to 1.8 in.$^3$/hr·ft$^2$.

Air leakage through specimens of concrete 4 to 42 in. thick in laboratory tests at a constant pressure differential of 6 psi was found to vary from 0.05 to 0.85 in.$^3$/hr·ft$^2$ with no apparent correlation with thickness. The concrete specimens conformed to the mix specifications for the BONUS slab. The membrane system selected on the basis of tests of various materials consists of a specific combination of rubberized-coal-tar elastic joint compound, coal-tar pitch, asbestos felt, and other compounds which, when properly installed, should have a permeability at a 6-psi differential pressure of less than 1% of the tolerable leak rate.

The Heavy Water Components Test Reactor (HWCTR) at the Savannah River Plant is contained in a composite concrete and steel containment vessel.[18] It was found in design studies that the containment structure would be partly below grade, for shielding purposes, and that the part that would be below grade would contain almost entirely concrete structures, such as shielding walls and floors. Therefore, by using prestressed concrete for the containment structure below grade and a steel shell anchored to the concrete above grade, a less costly design with the maximum usable space was achieved. Design data for this structure are presented below:

| | |
|---|---|
| Nominal inside diameter and height | 70 ft × 117 ft 6 in. |
| Acceptable leakage rate at design pressure | 1% of free volume in 24 hr |
| Design pressure | 24 psig |
| Test pressure | 29 psig |
| Concrete thickness below grade | 18 in. |
| Test leakage rate at 24 psig | 0.58% of free volume in 24 hr |

It is estimated that the erected cost of this structure was approximately two-thirds the cost of an all-steel containment shell.

A similar design is proposed by the Swedish Power Board for their R4/EVE 100-Mwe power reactor that is to start operation in 1967.[19]

Radiological Criteria

Criteria for the maximum permissible leakage rates for containment structures depend on the expected activity release into the structure and the distance to the nearest boundary of the exclusion area. Eltham[20] has presented curves, Fig. 11, giving inhalation time as a function of distance and leakage rate to give a dose of 25 rem to an adult thyroid following 20% meltdown of a 100-Mw reactor and the release of 600,000 curies of $I^{131}$ into the containment structure under "average" weather conditions. The average weather conditions are defined by the use of the following values in Sutton's atmospheric equivalents:[21] $C_y = 0.21$, $C_z = 0.12$, $u = 5$ meters/sec, and $n = 0.25$. A leak rate of 1% would yield a dose rate of 25 rem in 1 hr to adults located within 330 yards downwind from the structure. Ground release is assumed. If the leakage is discharged at a height of 70 ft, the hazard is reduced near the structure, but the height of the release has little effect at or beyond the site boundaries. "While these data are of interest, it would be of greater significance if the meteorological condition employed were more representative of the average "worst" (least favorable) weather conditions, as defined in the proposed AEC Reactor Site Criteria.[22]

Relatively high radiation fields might exist around a containment structure for short periods following an accident if the structure were unshielded. The effect of this radiation for the case where 600,000 curies

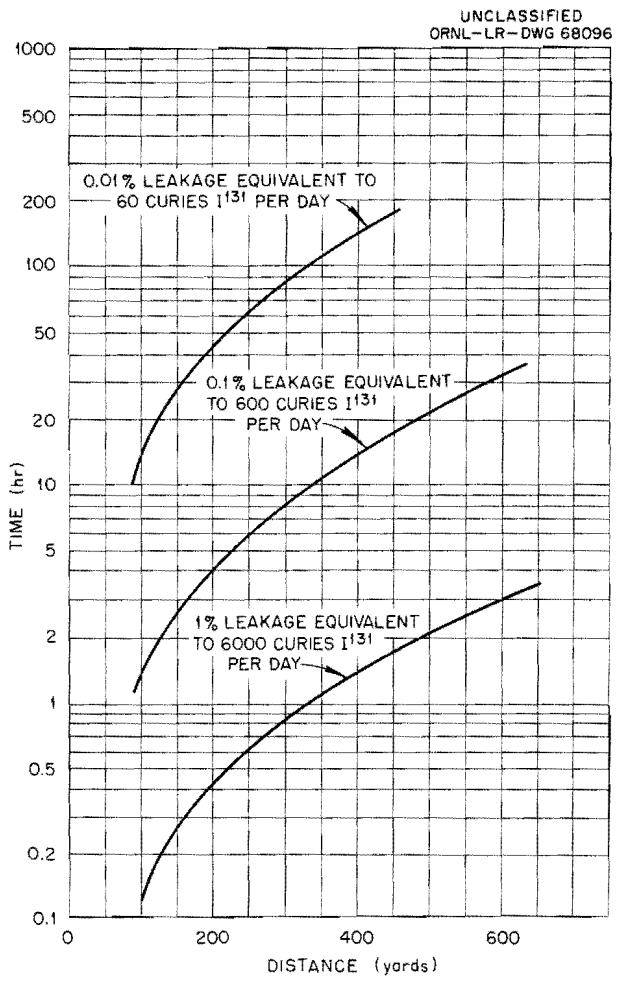


Fig. 11. Inhalation Times to Give Dose of 25 rem to Adult Thyroid Following 100-Mw Meltdown for Various Containment Leakage Rates. (Adapted from Ref. 20.)

of $I^{131}$ is contained within the structure, as reported by Eltham,[20] is shown in Fig. 12. A dose of 10 rem would be received in 1 hr at a distance of 350 yards from a 1-in.-thick steel containment vessel under these conditions.

Despite the stringent leaktightness required of containment structures[23] and the high exposures that may be calculated, the experience provided by the SL-1 accident[8] suggests that existing standards may be overconservative. The SL-1 reactor was contained in a welded steel building of 1/4-in.-thick plate which was not gastight. The nuclear excursion that occurred the evening of January 3, 1961, resulted in irradiation levels inside the building following the accident in the range of 500 to 1000 r/hr. Measurements taken outside the building on January 7 indicated 7 r/hr at the base of the building; at the perimeter of the 350-$ft^2$ enclosure the reading was 156 mr/hr. On the nights of January 3 and 4, measurements showed $I^{131}$ concentrations to be less than $10^{-9}$ μc/ml at Atomic City. Aerial and biological monitoring early on January 4 confirmed these small contamination readings beyond the SL-1 area. The largest activity, 125 counts/sec above a 200-count/sec background, detected by the first aerial monitoring flight was approximately 3 miles southwest of the reactor site. It is evident that in spite of the fact that the containment building was not leaktight, it did serve to confine the radioactivity to a very small area.

## Codes and Standards

The only existing code or standard that provides specific rules for design and construction of reactor containment shells is the ASME Boiler and Pressure Vessel Code through its case interpretations 1270N-5 and 1272N-5.[24] Although the Code does not apply to design of vessels for design

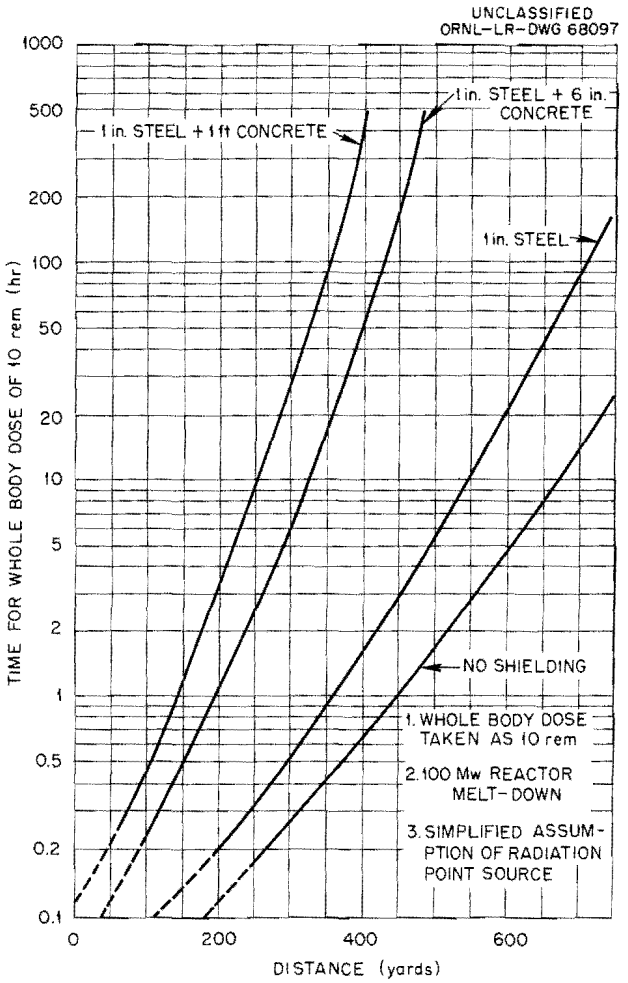


Fig. 12. Effect of Radiation Fields Around Containment Structure for Various Degrees of Shielding. (Adapted from Ref. 20.)

pressures below 15 psig, it may be used as a guide in designing for lower pressures. An alternate guide for the pressure range 0.5 to 15 psig is API Standard 620.[25] The API Code is specifically

> "intended to cover the design and construction of large, low-pressure, welded, carbon-steel storage tanks which have a single vertical axis of revolution and a nominal pressure rating of 0.5 psig or more, up to but not exceeding 15 psig, in their gas or vapor spaces and with metal temperatures from 200°F maximum down to the lowest ambient atmospheric temperature to which such a tank may be subjected."

No nuclear interpretation services are available for this code, as there are for the ASME Code, and therefore the user must provide the necessary nuclear requirements. Neither the ASME Code nor API Standard 620 is applicable to concrete containment structures. A tentative safety code for design, fabrication, and maintenance of containment structures for stationary atomic power reactors[26] was issued in 1959 for comment as a proposed American Standard, but the comments were so extensive that it has not been approved and reissued in final form.

Summary

Criteria for design of containment structures for water-cooled reactors have been fairly well established by precedent and by the experimental studies that have been performed, but there is a trend toward development of new types of containment, such as "pressure-relief" containment, for economic reasons. It appears that pressure-relief containment may eventually offer a satisfactory solution to the problem of containment of large power reactors in which the maximum credible accident would be initiated by failure of the pressure system and release of fission products would occur only after some delay during which the pressure surge could be relieved through the pressure-relief apparatus. Even for a reactor

where the mca consists of failure of the reactor vessel following a nuclear excursion, it is probable that release of fission products will occur after the initial release of pressure. This could explain the evident lack of spread of activity from the SL-1 incident.

The principal unknown factor in the design of pressure-relief containment structures is the time lag between the pressure peak following the accident and the release of fission products. The combination of pressure relief with controlled release of gases through filters and absorbers, as exemplified by Hanford NPR confinement, and controlled containment, as used for the Oak Ridge Research Reactor,[27] offers some improvement over simple pressure-relief containment.

The precedent for containment of gas-cooled reactors in the United States has not yet been established. In Great Britain, large power reactors of the gas-cooled, graphite-moderated type are contained in conventional building structures. The Experimental Gas-Cooled Reactor under construction at Oak Ridge is contained in a standard steel containment shell designed for an internal pressure of 9 psig.[28] The high-temperature gas-cooled reactor of the Philadelphia Electric Company will be contained in a cylindrical steel shell designed for an internal pressure of 5 psig at a temperature of approximately 190°F.[29]

Sutter[30] has pointed out that for water-cooled reactors which operate at a high power density and incorporate a considerable amount of stored energy, the danger of loss of coolant with subsequent core melting is more acute than in some other types. In comparing other types of reactor, the author makes the following statement:

"In Great Britain where some nuclear power plants are being erected within 10 miles of heavily populated areas and in France it is not considered that gas-cooled, graphite-moderated reactors of the actual design should require a containment shell. This is due to the use of natural uranium which allows only small margins of excess reactivity. Due to the relatively low specific power a serious dispersion of fission products could only result from the meltdown of a very large number of fuel elements in the simultaneous rupture of the pressure system. The possibility of such an accident occurring has not been admitted by those who have studied the design in great detail heretofore."

(W. R. Gall)

## SEISMIC CONSIDERATIONS IN NUCLEAR POWER PLANTS

Several important studies relative to the effect of earthquakes on reactor design have been conducted since a previous article on this subject was presented in Nuclear Safety.[31] Contributing factors in the recent interest have been the design and construction of the Japanese power station at Tokai-Mura in an active seismic region with a heavy concentration of people and the general development of the science of hazards evaluation as more and larger nuclear power stations have been constructed.

The design of nuclear power plants to resist earthquakes presents no serious technical problems. However, since an earthquake could initiate failure of vital components and cause conditions that would be hazardous to the public, seismic considerations are extremely important in every phase of design and operation of a nuclear power plant. A penetrating and broad coverage of the subject is to be published shortly by the AEC in a book entitled "Nuclear Reactors and Earthquakes."[32] It is written for the designer, analyst, and operator of nuclear power plants who may have had little or no exposure to the general science of seismology. The nature of earthquakes and the attendant response of specific structures in nuclear power stations are discussed in detail. The text is well illustrated and is documented with several hundred references.

### Site Location

The site location is, of course, the determining factor in establishing seismic design loadings. The probability of an area experiencing an earthquake of a given intensity is impossible to establish because of inadequate data and the general nature of earthquakes. The basis for selection

of a maximum probable earthquake intensity can only be a considered opinion that takes in account past records. A competent engineering seismologist should be requested to make a complete survey of the area being considered as a site for a nuclear plant and he should be retained as a consultant during the design of the plant. Local building codes will not, in general, provide adequate design factors, since they are minimum requirements for protecting structures from excessive damage.

Earthquakes are classified by magnitude, M, as determined by the energy released and the release mechanism and by the modified Mercalli intensity, MM, which indicates the nature and violence of the ground motion in a particular area. A given earthquake has only one magnitude, but it may produce different intensities at different localities depending on the type of soil or rock, the general geology of the area, and the distance from the origin, or focus, of the earthquake. The ground surface location directly above the focus is called the epicenter. The engineer is concerned chiefly with intensity values, seismic-wave periods, and the associated ground accelerations.

A rough relation between the magnitude M and the energy released, E (ergs), as seismic waves is given by:

$$\log_{10} E = 9.4 + 2.14M - 0.05M^2$$

The relation between intensity and maximum ground acceleration and that between magnitude and acceleration at the epicenter is too complicated to be stated in exact form. Estimates of the acceleration, a, may be computed, however, for the location where the MM value is known by either of the following relations:

$$MM = 3 \log_{10} a + 1.5$$

or

$$MM = 7/3 \log_{10} a + 2.1 \quad .$$

Acceleration $a_o$ at the epicenter may be roughly estimated from the magnitude by the equation

$$M = 2.2 + 1.8 \log_{10} a_o \quad .$$

There seems to be an upper limit based on energy consideration to the magnitude of the maximum possible earthquake of M = 8.7. The maximum intensity of ground motion associated with the maximum earthquake has been estimated to be from two to three times that of the El Centro, California earthquake of 1940, which had an MM intensity of 8.2.

For design purposes the assumption is made that seismic waves basically approximate sustained simple harmonic motion. The period is a function of the type of foundation, i.e., soil, bedrock, etc., and the distance from the epicenter. Table IV-3 indicates the relationship between the modified Mercalli intensity, the wave period (in seconds), and the ground acceleration (in cm/sec$^2$).[33]

The Uniform Building Code[34] classifies areas by earthquake damage probability zones, which are roughly related to the modified Mercalli intensity scale as follows:

> "Zone 0 includes areas having extremely small probability of earthquake damage and where the intensity of recorded earthquakes has generally not been greater than 4.
>
> "Zone 1 designates areas where earthquake damage has been minor with intensities limited to 5 or 6 and normally below the threshold of structural damage.

"Zone 2 includes areas of intensities 7 to 8, frequently subject to moderate structural damage. However, in this zone, greater damage corresponding to intensity 9 or higher might be expected at infrequent intervals.

"Zone 3 delineates areas where major destructive earthquakes have occurred in the past and might reasonably be expected at any future time."

Table IV-3. NORMAL GROUND ACCELERATION FOR VARIOUS LEVELS OF MODIFIED MERCALLI INTENSITY AND FOR DIFFERENT WAVE PERIODS

| Modified Mercalli Intensity | Ground Acceleration ($cm/sec^2$) for Indicated Wave Period | | | | | | | | |
|---|---|---|---|---|---|---|---|---|---|
| | 0.33 sec | 0.4 sec | 0.5 sec | 0.75 sec | 1.0 sec | 1.5 sec | 2.0 sec | 3.0 sec | 6.0 sec |
| 1 | 2 | 2 | 2 | 1 | 1 | 1 | 1 | 1– | 1– |
| 2 | 4 | 4 | 3 | 3 | 3 | 2 | 2 | 1 | 1– |
| 3 | 8 | 7 | 7 | 6 | 5 | 4 | 3 | 2 | 1– |
| 4 | 16 | 15 | 14 | 12 | 11 | 8 | 6 | 4 | 1– |
| 5 | 31 | 30 | 29 | 25 | 22 | 17 | 13 | 7 | 1 |
| 6 | 64 | 62 | 58 | 51 | 44 | 34 | 26 | 15 | 3 |
| 7 | 130 | 125 | 119 | 104 | 90 | 69 | 52 | 30 | 6 |
| 8 | 264 | 254 | 241 | 210 | 184 | 140 | 107 | 62 | 12 |
| 9 | 537 | 518 | 490 | 428 | 373 | 284 | 216 | 126 | 24 |
| 10 | 1090 | 1051 | 996 | 869 | 758 | 577 | 440 | 255 | 50 |

The design factor, Z, relative to zone 3, as given in Table IV-4, is proposed[32] for determination of horizontal acceleration forces for specific reactor sites for which adequate seismological data are not available. It should be noted that the only region in the world for which accurate and adequate data on earthquake ground motion exist is the western portion of the United States.[35] However, the El Centro earthquake of 1940 is often considered to have caused the greatest probable ground motion for zone 3. Therefore the maximum horizontal acceleration of this earthquake, which was measured at 0.35 g, is generally taken as the maximum probable acceleration for zone 3, as indicated in Table IV-4.

Table IV-4. MAXIMUM PROBABLE HORIZONTAL ACCELERATION FOR EARTHQUAKE ZONES LISTED IN THE UNIFORM BUILDING CODE

| Zone | Z | Horizontal Acceleration |
|---|---|---|
| 0 | 0.25 | 0.0825 g[a] |
| 1 | 0.25 | 0.0825 g |
| 2 | 0.5 | 0.165 g |
| 3 | 1.0 | 0.35 g |

[a]In this concept g is the gravitation constant.

The Japanese have adopted a policy of extreme safety in specifying earthquake design factors for the 570-Mwt Tokai-Mura reactor.[36] The local code factors were multiplied by 1.5 to give a value of 0.2 g horizontal acceleration for the reactor building and by 3 to give a value of 0.4 g horizontal acceleration simultaneously acting with 0.2 g vertical acceleration for vital equipment such as the reactor, biological shield, steam-raising units, and their supporting structures. The largest earthquake in the Tokai-Mura area over the last ten centuries had an estimated accleration no greater than 0.23 g.

In contrast to the Japanese, reactor builders in the United States have been less conservative with design factors. Vital components of the 165-Mwt Humboldt Bay Plant in California[37] are designed to resist earthquake shocks of MM intensity 8, corresponding to a horizontal acceleration of 0.25 g with periods of from 0.1 to 0.3 sec. The shocks are expected to be repeated several times. Nonnuclear portions of the plant have design factors of 0.2 times the dead load plus 1/2 the live load.

The 20-Mwt Vallecitos boiling-water reactor in a zone 2 area in California[38] is designed for a horizontal acceleration of 0.134 g. The 50-Mwt Boiling Nuclear Superheater (BONUS)[39] power station in Puerto Rico is designed "to resist moderate to severe earthquake activity for zone 2." The 84.3-Mwt Experimental Gas-Cooled Reactor (EGCR) at Oak Ridge, Tennessee, is designed to resist horizontal ground accelerations of 0.05 g simultaneously acting with 0.025 g vertical accelerations and with dominant periods of from 0.1 to 0.5 sec.[40] The Oak Ridge area is generally considered to be in zone 1.

Plant Design

It is impractical to build earthquake resistance onto an existing design. Therefore nuclear plants must be designed from the beginning with earthquake considerations in mind. The need for earthquake resistance will sometimes override other considerations.[41] On the other hand, nonearthquake hazards such as hurricanes may be more important in establishing design requirements, and, as in the case of BONUS, may create adequate safety in the plant during an earthquake if the earthquake and the hurricane do not occur at the same time. Design requirements for minimizing the possibility of damage affect many areas of a plant design, although most important from the hazards viewpoint is the integrity of the final fission-product containment system.

For earthquake resistance the general arrangement of the plant should be such that different components of the plant do not vibrate independently in a manner that will damage each other. Seismic restraints that could act as battering rams should be avoided. Long pipes that are not supported on a

monolithic foundation structure are particularly dangerous. Equipment for removing decay heat should be located near the reactor in order to reduce the danger of pipe rupture during seisms.

The center of gravity of the plant and major components should be near the ground, since seismic activity is amplified above ground level. Equipment running on tracks should be safe from overturning and causing damage to electrical and pneumatic lines or other vital equipment, such as blowers and gas and water lines.

Automatic provisions for plant shutdown should be included in the plant control system since earthquake shocks last only a few seconds, although they may recur in a matter of minutes, hours, or days.[32] Many reactor plants have, as auxiliary equipment, seismometers that automatically scram the reactor at the beginning of an earthquake and then record the ground accelerations and displacements. In the interest of gathering earthquake data which are of direct interest to construction engineers in general, every plant should be equipped with an approved seismometer.[35] Several different designs are currently available; some are discussed in Ref. 1. For more detailed information the U. S. Coast and Geodetic Survey should be consulted.

Damage to structures overstressed as the result of the direct transmission of ground motion is referred to as direct mechanical damages. Such failures may be categorized approximately as follows:

1. Failure of isolated structural units because of bending, buckling, or shear stresses created by their own movements. In particular, long, thin structures, such as control rods, fuel subassemblies, vent stacks, and water towers, may be subject to distortion or breakage.

2. Failure of lines or members attached to structural units that undergo large differential motion, e.g., the coolant pipes joining the reactor pressure vessel to heat exchangers.

3. Failure at the base of structural units. Tie bolts and anchor bolts may be stretched to failure. The concrete of the base may crack or break off.

4. Failure caused by dynamic loads, such as liquid movement, since oscillations of liquids with a free surface may magnify stresses.

5. Failure of materials which have been weakened by chemical attack, radiation embrittlement, stress-corrosion cracking, thermal shocks, or previous overstressing from earthquakes.

Possible indirect damage may result from such things as falling objects, fire, and operator panic. Unprotected control rod drive mechanisms are extremely sensitive to damage from falling objects. Attention should also be given to all heavy masses in elevated positions, i.e., the overhead crane system.

Nuclear damage can be caused by the action of the reactor control system during an earthquake. For example,[32] consider a typical water-moderated reactor, the SM-1, in which a scram is accomplished by gravity-induced control rod insertion. If a scram is initiated during an earthquake which is causing the entire reactor structure to vibrate vertically with a period of 0.1 to 0.2 sec and an acceleration of ±0.1 g, the effects of initial upward velocity of the rod could cause a free fall delay of 0.02 sec. This small possible delay is a significant portion of the normal (0.05 sec) delay time from the scram signal to initiation of the scram.

This additional delay is estimated to cause the reactor power transient to peak at 187 Mw rather than 97 Mw.

A delineation of specific items to be considered is given in Ref. 42. The following summary is taken directly from that reference.

"Elements Particularly Sensitive to Seismic Damage. — Some of the elements of which particular care must be taken when making an aseismic design are listed below. In each case the number in parentheses indicates the classification of the element, that is (1) indicates that damage to the element might lead to the release of fission products; (2) indicates that damage to this element could lead to prolonged shut down of the reactor.

a. Reactor core (1)
   1. The fuel elements (1)
   2. The moderator (1)
   3. Control rods and safety rods (1)
   4. Supporting and positioning members (1)

b. Reactor pressure vessel (1)

c. Primary coolant loop (1)
   1. Piping system (1)
   2. Heat exchanger (1)
   3. Pressurizer (1)

d. Secondary coolant system (2)
   1. Heat exchanger (2)
   2. Condenser (2)
   3. Cooling water system (2)
   4. Turbo-generator (2)

e. Instrumentation (1), (2)

f. Electric power system (2)

g. Safety devices (1)

h. Emergency water system (1)

i. Fuel handling and storage system (2)

j. Containment building or vessel (1)

k. Biological shield (2)

l. Water storage tanks (2)

m. Chimneys (2)

"It will be noted that many of the above items refer not to buildings and such like structures, but rather to machinery, equipment, piping, etc. This means that it is just as important

> to design, say, the piping of the coolant system to resist earthquakes as to design the large pressure vessel that houses the reactor. The engineer who is responsible for the earthquake-resistant design must therefore extend his attention to all the important elements of the reactor as well as to the structures and buildings. The possibility of rupturing connecting pipes by the relative motion of two pieces of equipment during an earthquake is an example of possible seismic damage."

A noncritical category, identified in Ref. 42 as category 3, includes service and support structures not vital to reactor safety and operation for which use of seismic requirements in building codes is appropriate.

The prime consideration for the reactor is that the structure itself may not distort and jam control rods or damage fuel elements. Graphite-moderated cores which consist of large stacks of smaller blocks should be keyed and braced to prevent shifting. The structures of small reactors usually have sufficient strength and rigidity for earthquake resistance if properly mounted. However, large massive reactors which must accommodate larger thermal expansions pose particular problems.

The biological shield is preferably a monolithic structure firmly anchored to the foundation. Free-surface liquid reservoirs, highly elastic and poorly damped structures, heavy wheeled machines, or tall heavy equipment with poor lateral bracing should not be permitted on top the biological shield, since such equipment may impose large stresses on the shield by amplifying the effect of seismic velocities and displacements.

During a typical earthquake, a well-designed foundation in strong soil or a foundation poured on bedrock will follow precisely the motion of the ground without amplification. Amplification is evidenced by fissures or cracks between the foundation and the earth after an earthquake.

Foundations should be sufficiently thick and well reinforced to withstand the shear and bending forces imposed by the passing seismic waves. These forces include dynamic reactions of the biological shield, heat exchangers, pressure vessel, and other massive equipment supported by the foundation.

In the design of the over-all reactor complex, interconnected components should, when possible, be mounted solidly on an adequate unit foundation. This includes all components interconnected with duct work or piping that carry radioactive fluids, or fluids vital for reactor safety. Expansion bends and flexible couplings should be designed to accommodate the maximum predicted motion caused by an earthquake, in addition to the normal requirements. When it is necessary to mount components on separate foundations, e.g., the reactor and the steam generators or the steam generators and the turbine, the foundations should be well separated to prevent battering, and the interconnections should be able to accommodate more than normal relative displacement.

Reference 31 contains an excellent survey of analytical methods for studying the effects of earthquakes and examples extending from the simple one-degree-of-freedom concentrated-mass system to the matrix formulation of large multidimensional problems. In an excellent paper[43] by Leckie and Livesley, the containment vessel for the Tokai-Mura plant is analyzed for lateral loads, such as wind and earthquake forces. Various other papers have been written which treat the general problems of structural response to earth shocks.[44-46]

In some cases it may be desirable to dynamically test critical components or scale models, particularly if the plant is to be built in a seismologically dangerous area and if the part is brittle or loosely tied

together, e.g., a graphite moderator. A one-third scale model of the Japanese reactor graphite core was dynamically tested on a specially built shaker table.[47] The results of the test proved that, if properly supported, a graphite core could survive very severe earthquakes (0.7 g horizontal plus 0.38 g vertical accelerations) without distortion to the point that control rods would be prevented from entering.

Scale-model tests have also been conducted on outer containment structures to check the validity of the scaling laws with respect to rapid transient dynamic loads.[48]

Summary

Reactor plants are ruggedly constructed for many reasons, so the possibility of damage from strong earthquakes is, in general, unlikely. However, an adequate seismic analysis should be part of every nuclear power plant design effort, even though the reactor is to be constructed in an inactive earthquake area. In the event of an earthquake sufficiently intense to require shutdown, the plant should be thoroughly inspected for damage and potentially dangerous conditions before operation is resumed. Damage may be hidden and difficult to detect.

It is apparent that uniform consideration of aseismic design for nuclear power stations has been lacking. The reason is partly that the development of nuclear power has required the solution of many other new and difficult problems and partly that adequate earthquake data are not yet available. The extension of nuclear plant design parameters to include the effects of earthquakes is, however, important to the continued safe operation and development of nuclear power. (S. E. Moore)

## SAFETY IN FUEL-STORAGE CANALS

The design of fuel-storage canals presents several significant problems of safety that require attention. While the hazards are orders of magnitude less severe than those associated with reactor operation, they are significant, particularly since the manipulations of the fuel in the canal are done by more or less direct manual means through only the protective barrier created by the canal water. The hazards associated with fuel storage include direct gamma radiation from the fuel, the possibility of accidental nuclear criticality, and radiation-exposure problems created by the release of gaseous, soluble, or particulate activity to the storage canal. These problems have all been discussed to some extent in more general articles in previous issues of Nuclear Safety.[49-52]

### Canal-Water Contamination

Contamination of the canal water may arise from radionuclides in corrosion films (crud) deposited on the fuel-element surface during reactor operation, from accidental contamination of the fuel surface with fissionable material during manufacture,[49] from failure of the fuel-element cladding during irradiation or storage, or from operations such as dismantling or sawing to prepare the element for shipping or reprocessing. A review of the problems of contamination of shipping-cask and canal water by fuel elements removed from pressurized-water reactors was made by King.[53] The crud film on the elements was estimated to be a source of excessive contamination of the canal and shipping cask unless controlled by prior descaling of the fuel or by water-cleanup methods. Small cladding penetrations were not considered likely to cause significant contamination of the canal, since

$UO_2$ is quite stable in water. The report by King[53] also contains criteria for designing reprocessing canals based on ORNL experience. The criteria recommend the use of impermeable coatings on the canal walls, ion exchange and filtration of the water, and canning of sheared irradiated uranium. To make the preceding criteria generally applicable, the following additional requirements may be desirable: (1) provision for nuclear criticality safety; (2) performance of potentially hazardous operations, such as dismantling fuel by sawing prior to shipment or reprocessing, in areas other than the main body of the canal to minimize the spread of any activity that may be released; and (3) ventilation of the canal area to minimize personnel exposure to gaseous or particulate activity released from the canal.

Underwater shearing of aluminum-clad uranium-metal plate-type experimental fuel elements irradiated to levels of 480, 600, 940, and 1500 Mwd/T has been described.[54,55] The first shearing experiments were a feasibility study and were done in a small water tank in a hot cave with 600- and 1500-Mwd/T fuel that had been cooled four to six months. Approximately 100 μc of activity was released per cut surface per 24-hr storage period in 2.5 liters of water. About 0.17 μc of activity was released to 156 liters of air above the water. The addition of 1% sodium dichromate to the water reduced the activity released to the water and air. Correlations of fuel irradiation level and air and water activity release were made. The larger scale operations[55] were conducted under water in a fuel storage canal with the shearing equipment isolated from the canal by an open-top tank. A 15-gpm recirculating water system in the isolation tank removed activity from the water by filtration and ion exchange. The cut fuel was canned in leaktight cans before removal from the shearing tank. Water

activity during shearing of 480 Mwd/T and 940 Mwd/T long-cooled fuel rose to a maximum of $3 \times 10^{-3}$ $\mu c/cm^3$ in the tank sump and did not increase above the canal background level of $3 \times 10^{-4}$ $\mu c/cm^3$ at the top of the tank where the filtered and deionized water was discharged. The radiation level was 8 mr/hr at 3 in. above the water surface. Air activity did not increase above background during the shearing operation. The shear was not seriously contaminated and could have been serviced by direct means if desired.

Fission-Gas Release

Fission-gas release in a storage canal can occur as a result of fissionable-material surface contamination of the fuel or as result of a rupture. Ruptured metallic uranium fuel is particularly troublesome because oxidation of the uranium by water will release fission gases.

Surface $U^{235}$ contamination of ORR fuel has resulted in detectable, but not hazardous, air activities above the fuel storage canal for a few hours after the fuel was unloaded from the reactor.[49] King reports[53] that freshly discharged ruptured ORNL Graphite Reactor slugs, if not immediately canned, cause sufficient airborne activity to require air masks in the canal room for 24 hr. The activity seems to be predominantly the rubidium and cesium daughters of krypton and xenon. Similar experience has been reported[56] with freshly discharged ruptured slugs in the storage canal at Hanford. The principal gaseous activity was $Xe^{133}$, and it gave general canal-room air activities of as much as 1 r/hr 6 ft from the canal water. Canal-water contamination sometimes rose to 100 to 200 times the normal level. Gaseous activity levels remained high for about 4 hr and, after 8 hr were usually low enough not to interfere with canal operation. Prompt canning of ruptured slugs reduced the difficulties encountered.

A radiation accident[57,58] at the General Electric Test Reactor resulted in the exposure of one person to a thyroid dose of 48 rad; six others received an estimated dose of 1875 to 7060 mrem from gaseous $I^{131}$ and $I^{133}$. A three-day cooled, defective fuel test element unexpectedly gave the gas release when the cask containing the element was being cleaned prior to transfer from the building. Previous experience with similar uranium-aluminum alloy fuel specimens had shown that no activity release would be expected after discharge from the reactor. Surveys of the cask 2 hr after the incident indicated a rise of surface activity from 400 mrad/hr to 50 rad/hr. Preventive procedures adopted as a result of the accident included canning all ruptured fuel and modifying a mobile continuous air sampler for detecting noble gases so that it would also detect halogens. Further checking of air samples and personnel after the incident revealed that a very low level of iodine contamination had existed in the reactor containment vessel prior to the accident. It was apparently coming from a previously undetected leaking fuel element in the storage canal. Methods are now available at Vallecitos that will detect 0.002 μc in the thyroid.

Shippingport Experience

A pressurized-water-reactor fuel-storage canal installation and the operations therein are exemplified by the canal at the Shippingport Atomic Power Station.[59] The fuel-handling canal, which permits underwater refueling of the reactor, is divided into a fuel-storage section, a blanket-inspection and cask-loading section, a core- and mechanical-parts-storage section, and the refueling pool over the reactor. The fuel-storage section includes an underwater tank capable of containing released fission products

and within which damaged fuel elements can be stored. Canal sections are isolated from each other by water-tight gates. A recirculating water system is provided for the canal to remove waterborne radioactivity by ion exchange and to remove decay heat generated by the fuel. The canal surfaces are coated with a polyester paint to prevent activity penetration into the concrete.

Criticality safety is maintained in the fuel-storage area by moving only one fuel element at a time and by storing the fuel in racks on approximately 30-in. centers. The blanket-storage rack would permit storage of fuel on a smaller center-to-center distance than that permitted by the fuel-storage rack, and therefore special care is taken to avoid unintentional storage of fuel in the blanket storage rack. The container for storing damaged fuel differs from that for normal fuel elements in that only the four corner positions of the rack can be used. This arrangement assures safe geometry.

Fuel elements stored in the PWR canal have radiation levels as high as 43,000 r/hr in water at 1 ft from the midpoint of the cluster, but this level is reduced to background level of less than 0.5 mr/hr at working areas immediately above the water by always handling the fuel under at least 10 ft of water. Fuel-handling devices and procedures are designed to prevent lifting fuel or other irradiated components closer than 10 ft to the water surface. Contamination problems arising from the crud film and fission products on the surface of the fuel are, however, less serious than anticipated.[60] For example, a transfer container provided for moving fuel from the reactor to storage was found to be unnecessary because "no fission products and very little crud became dispersed from the fuel ...

into the canal ... and what crud did disperse ... was controllable by means of a demineralizer and by washing the canal walls with fresh water after drainage."

There was concern over the possibility of atmospheric contamination or of high radiation fields over the canal during underwater refueling because of radioactive gases or possible airborne contamination emanating from crud which had dried on canal walls after draining, but these problems did not materialize. The only indication of cladding failure in the entire core was in one blanket assembly, but it should be pointed out that this fact may present a more favorable fission-product contamination picture than may be justified for recently completed pressurized-water reactors that have higher power ratings and burnup of the oxide pellet fuel.

The highest waterborne contamination level in the canal was 270 dpm/ml. The canal walls were carefully washed after draining and gave smears ranging from 4700 to 16,000 beta and gamma dpm/ft$^2$. Smear readings on the canal bottom were not reported.[60] Cracks in the plastic coating on the reactor pit floor were repaired before opening the reactor for refueling; some trouble had also been experienced during initial construction with failures of the fuel-canal coating. To secure a weld specimen for examination, underwater sawing was done on a control rod that had a reading of 1200 r/hr in water 1 ft from the side. The bandsaw was placed in a pan to catch the chips, and the cutting area was isolated from the rest of the canal. No difficulties were encountered with the cutting operation.

A preshipment check of the water activity in a shipping carrier containing a blanket assembly suspected as having a failed fuel element showed that the carrier-water activity did not exceed mpc for unidentified fission

products in water ($1 \times 10^{-7}$ μc/ml). Thus a very low fission-product leach rate was indicated for the defective fuel element. Care was taken in carrier loading to preserve the crud film for hot-cell examination, and thus the possible contaminating effects of the loosely adhering portions of the crud film were not observed in the cask water before shipment.

Fuel Storage Canals at Other Reactors

Fuel canal facilities at other pressurized-water reactors are, in general, similar to the Shippingport installation, except that the fuel-storage canal area is usually located outside the reactor containment vessel.[61,62] Fuel transfer from the reactor pool to the storage pool is made through a water-filled tube provided with valves at each end to maintain the reactor containment. The Consolidated Edison Thorium Reactor[61] storage canal is provided with removable metal covers and has ventilation exhaust ducts slightly above the canal-water surface to minimize contamination of the canal room by activity from the stored fuel. Borated water is required in the reactor pool during fuel exchange to keep the core subcritical and is used in the storage pools to give an additional margin of criticality safety. Fuel is stored in baskets containing four elements separated by a cruciform partition of boron steel, since, in the absence of poisons, four fuel elements could become critical in water. The baskets are 15 in. square and are stored on 24-in. centers.

The fuel storage systems for sodium-cooled reactors are somewhat different from those described above. At the Hallam Reactor[63] the fuel is removed from the reactor by a shielded refueling machine to a wash cell where the exterior is steam cleaned to remove traces of sodium. The fuel

is then stored in fixed, vertical thimbles that extend into a vault. The vault may be flooded, if desired, for better cooling of the fuel thimbles. The thimble completely isolates the fuel from the pond water. A portable gas sampler is used to monitor the storage thimbles for the $Xe^{133}$ activity that would be released if the fuel cladding leaked during storage.

At the Enrico Fermi Reactor,[64] the fuel is allowed to partially decay in the reactor sodium pool before removal to sodium-filled, externally finned pots in a heavily shielded cask car to a steam-cleaning station where the element is removed from the sodium and steam cleaned. The cleaned element is then stored in a water-filled canal for further decay. All handling is by remotely operated machines.

At the Peach Bottom Atomic Power Station,[65] the uranium carbide fuel, which is reactive with water and may release as much as 79% of the gaseous activity on contact with water, is to be stored under water in cans that will be sealed by brazing and leak tested.

A recent news bulletin described the fuel-storage canal proposed for the Eurochemic Fuel Reprocessing Pilot Plant to be constructed at Mol, Belgium.[66] The canal is divided by gates into three sections. These are the fuel-cask-unloading pool, the mechanical-treatment pool (for removal of unwanted inert ends), and the storage pool. The walls and floor in the first two sections are lined with stainless steel; the storage section is lined with a plastic material. The canal is built above grade because of the sandy soil and the high water table (only 6 ft below the ground surface). It is situated in a containment saucer to give positive control over any possible activity escape in water leakage from the canal. The canal water will be purified by ion exchange and filtration. Special air-

exhaust ventilation will be provided at the water surface of the unloading and mechanical treatment pools. Defective fuel elements will be canned before transfer to the storage pond. The general problems of such a facility have been discussed by Rometsch.[67]

The design and engineering problems of a canal-type irradiation facility in the United Kingdom in which spent fuel is used as the irradiation source have been described.[68] The significant operations include underwater shearing to remove the inactive ends from DIDO and PLUTO fuel elements, canning of the sheared fuel in cadmium-poisoned cans to prevent nuclear criticality and to facilitate handling (the cans are not sealed), and underwater-handling. Ion exchange and filtration of the canal water was found necessary to remove activity and to maintain sufficient water clarity for adequate underwater visibility. The irradiation pond water level is kept higher than the water level in the shearing pond, and fuel transfer from the shearing pool is through a double-door underwater transfer port to confine activity to the shearing pool. The transfer port is also used to check an element for suspected cladding ruptures, since the water trapped in the port can be recirculated and sampled. The observations in the report[68] on this facility are applicable to the design of canals for reactor and fuel-reprocessing facilities.

Conclusion

The design and operating experience with fuel-storage canals is now sufficiently developed so that operations such as reactor refueling, fuel storage, and underwater cutting or disassembly of fuel elements can be done safely. This statement must be qualified by pointing out that experience is not available at present with full-size oxide-pellet fuel

elements of high burnup that may have a substantial number of defects as a result of the irradiation. The technique of canning should, however, permit safe handling of such elements if they would otherwise seriously contaminate the storage facility. (E. L. Nicholson)

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# V. CONSEQUENCES OF ACTIVITY RELEASE

# THE ACUTE RADIATION SYNDROME

Gould A. Andrews*

Information on the acute radiation syndrome in human beings is being amassed quite rapidly. Some of this comes as a result of unfortunate accidents,[1-5] and much of it comes as a result of clinical experimentation.[6,7] The recent accidents have occurred in rather bizarre situations, in which the ordinary self-evident hazards had been largely eliminated by careful planning and control. The use of high doses of total-body irradiation as a treatment of disease has increased largely as a result of experimentation sponsored by the United States Atomic Energy Commission which showed the possibilities of tissue transplantation. These studies have suggested to clinicians several situations in which total-body irradiation and graft procedures may be of value. Incidental to these clinical studies, a considerable amount of valuable information on radiation effects has been made available. This review attempts to bring up to date the information on the total-body acute syndrome produced in the human being by penetrating radiation from external sources. No effort will be made to consider superficial deposition of radioisotopes that yield nonpenetrating radiation or internally deposited radioisotopes. These problems are of great importance,

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but are outside the scope of the present discussion. It is true, of course, that in considering radiation accidents, we are seldom dealing with truly uniform total-body irradiation, and this, indeed, may be the explanation for some of the discrepancies in the information.

Work on experimental animals has been of great importance in elucidating the acute radiation syndrome, but no effort will be made to refer directly to animal experimentation here. Many of the phenomena seen in experimental animals are also observable in human patients, but there are important differences. One of the most significant is the difference in time relationship of the response; most experimental animals show more rapid development of signs of injury and more rapid recovery than humans.[8] Failure to recognize these differences in time relationships may lead to very important misinterpretations. More specific differences are also important, and it is with the gradual accumulation of data on human beings that we are able to outline these for the species of greatest interest to us.

Excellent work on the acute radiation syndrome in human beings has been reported by several authors. An article by Gerstner[9] contains an extensive bibliography and serves as a basis for obtaining information up to the date of its publication. Another valuable study was that of Wald and Thoma,[10] which attempted to evaluate all radiation accidents and to relate clinical and laboratory findings to radiation dose. A recent publication from the World Health Organization presents an inclusive and timely consideration of all aspects of acute radiation injury.[11]

The Radiation Syndrome at Various Dose Levels

It is customary to describe the acute radiation syndrome in terms of rather arbitrary classifications based on dose and response. These classifications serve a useful purpose but are subject to very definite reservations because the manifestations overlap at certain dose levels and because uneven distribution of the radiation may cause atypical combinations of effect. The simplest classification would include, first, a subclinical level in which the dose is too low to cause any definite symptoms but may produce laboratory evidences of radiation injury. At somewhat higher radiation doses, there is a hematologic effect, in which the damage is manifested chiefly by the depression of the bone marrow. The patients usually also have early nausea and vomiting and, later, epilation and other clinical manifestations. Third, at still higher dose levels, there is a gastrointestinal syndrome, in which the chief damage is to the mucosa of the gastrointestinal tract and which thus may be characterized by electrolyte loss and nutritional disturbances. Fourth, at very high doses, there is a central nervous system effect, in which damage to the brain is the most obvious manifestation.

Wald and Thoma[10] have extended the classifications to include five clinical groups. They do this by dividing the hematologic classification into a mild and a severe form. It should be pointed out that the time of manifestation of these various levels of injury is of great importance. At exceedingly high doses the central nervous system damage is the first to be clinically important. At slightly lower doses the gastrointestinal features are the first to appear, and at still lower doses, the hematologic. If, however, patients could survive the nervous system damage at high levels,

they would subsequently suffer from the gastrointestinal syndrome, and if they could survive this, they would show the hematologic syndrome. In other words, apparent injuries to these various systems are based on the fact that the first clinical features to appear will dominate the picture, and at very high levels, patients may die before they have an opportunity to show the manifestations of the lower dose features. Figure 13 indicates the time of death at different dose levels and lists the different syndromes and their probable time of occurrence. The data for Fig. 13 were, in part, extrapolated from data on experimental animals, and it may be that the irregularities of the curve would not be borne out by actual data if they were available.

Subclinical Dosage. There is a wide range of dosage above the maximum permissible exposure and below the minimum dose that will produce symptoms of the acute radiation syndrome. The reason that the maximum permissible dose is set so low is, of course, that the objective is not simply to avoid the acute radiation syndrome but to avoid the statistical hazards of delayed radiation effects, which may appear at much lower doses.

An exposure of 25 r has frequently been stated as a tolerable dose for emergency radiation exposure if it occurs only once in the lifetime of a person. Although many times the usual permissible occupational exposure, this dose is still thought to be low enough to avoid the acute radiation syndrome, and it is well below threshold for clinical effects. In the early stages of pregnancy, such an exposure would offer significant risk of serious, direct damage to the fetus. There is nothing unique about the choice of 25 r for this emergency dose, and some other emergency

level might be defended almost equally well. Although any dose above the maximum permissible occupational level is undesirable, there is a rather wide range, probably up to at least 200 rads, at which the dangers of an acute lethal effect are minimal.

The persons with subclinical radiation injury in accidents usually present no important problem, and they would not even be discovered if it were not for special efforts to observe them, based on the knowledge that they might have been exposed. These patients may show some slight degree of change in hematologic values, an early depression in lymphocytes, and possibly some depression in granulocytes and platelets during the third to sixth weeks after exposure. No treatment is needed for the early manifestations, and no way of preventing possible later undesirable effects is known.

Hematologic Syndrome. The hematologic syndrome is, by all odds, the most important clinical group of radiation effects because it is relatively common and because, while life threatening, it is subject to therapy. It will be seen at total-body doses of from around 100 to 200 up to 600 or 800 rads. The clinical picture is characterized by separate phases, which occur in orderly sequence over a considerable time period, as outlined (Fig. 14) by Gerstner.[9] This sequence of events is sometimes spoken of as if it were characteristic of the acute radiation syndrome in general, but actually it is limited to this general dose range. At lower dose levels, some of the features may be completely absent, and at higher dose levels, symptom signs and laboratory changes develop much more rapidly and are not easily separated into specific phases.

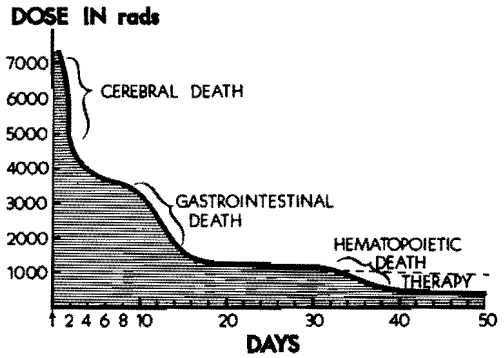


Fig. 13. Survival Period as a Function of Dose (adapted from Gerstner[9]). The dotted line suggests the possible effect of therapy in altering the number of deaths from the hematopoietic syndrome.

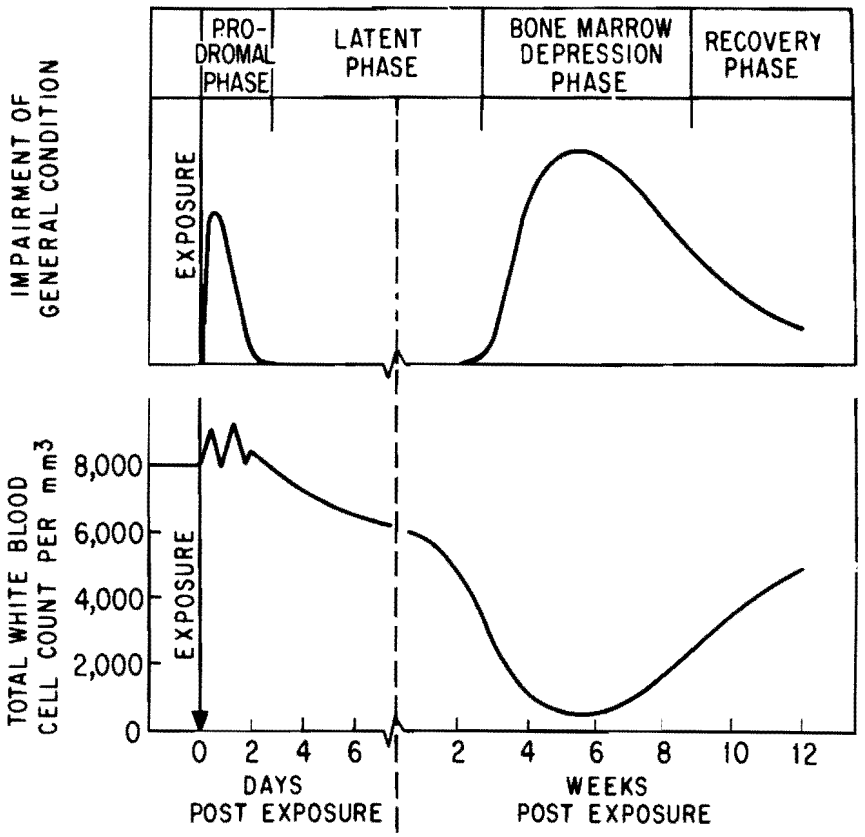


Fig. 14. Schematic Representation of the Hematopoietic Form of the Acute Radiation Syndrome (from Gerstner[9]).

In the hematologic syndrome, there is first a prodromal phase characterized by nausea and vomiting, which may start within 1 to 2 hr after exposure and may last for two days or so, but tends to taper off after the first few hours. This nausea and vomiting have proved to be very valuable indices of serious radiation injury. Absence of nausea and vomiting after an acute exposure is almost, but not quite, certain evidence that no life-threatening dose has been received. Occasionally, however, a patient will fail to vomit even though he has received a very dangerous dose of radiation. The presence of nausea and vomiting is, conversely, an indication that the patient should be given careful attention and that he has very likely received a serious radiation dose. It is sometimes said that nausea and vomiting are very nonspecific reactions that may occur in any frightened or upset person and for this reason are of little value. Although they may be purely psychogenic in origin, the study of most accidents suggests that these symptoms are of great clinical value and in most cases do represent meaningful indications of serious, but not necessarily lethal, radiation injury.

This nausea and vomiting should not be considered as evidence of the gastrointestinal syndrome. This syndrome develops somewhat later, after injury to the mucosa of the intestinal tract has become manifest. Patients who get the gastrointestinal syndrome will, of course, have nausea and vomiting initially, and this may continue without letup into the severe manifestations of damage to the gastrointestinal tract. The term "radiation sickness" is sometimes used for the nausea and vomiting that occur after acute exposure to radiation, but this is a rather poorly defined term. It is very often used in discussing patients who receive clinical

radiotherapy to relatively small parts of the body and who develop gastro-intestinal symptoms during treatment. It has also been used to describe the whole sequence of clinical manifestations of total-body irradiation injury.

In the hematologic syndrome, after the prodromal nausea and vomiting phase is over, there is a latent period during which the patient may seem reasonably well or may complain of some degree of fatigue. Then, at three to four weeks after exposure, he may begin the period of bone marrow depression. This is characterized by profound depression of granulocytes and platelets (the lymphocytes will have been depressed much earlier, within a few hours after the accident, but this is not indicated by any clear clinical symptoms). Figure 15 illustrates average hematologic changes in five persons involved in a radiation accident, who received doses from 236 to 365 rads. The early fall in lymphocytes is not shown because it had occurred before the first values charted on this graph. In the hematologic syndrome, when the granulocytes and platelets become scarce, clinical manifestations develop because of their deficiency. The clinical manifestations are bleeding and susceptibility to infection. Depending on the severity of the deficiency of the granulocytes and platelets, the presence of complicating factors, and the effectiveness of treatment, the patient may survive the period of hazard or may die of hemorrhage or infection or both. At around six to seven weeks after exposure, spontanteous recovery tends to occur; the bone marrow rather rapidly regenerates and for a time produces cells at a rate that is accelerated even above normal. Perhaps because of confusion with chronic irradiation, which produces

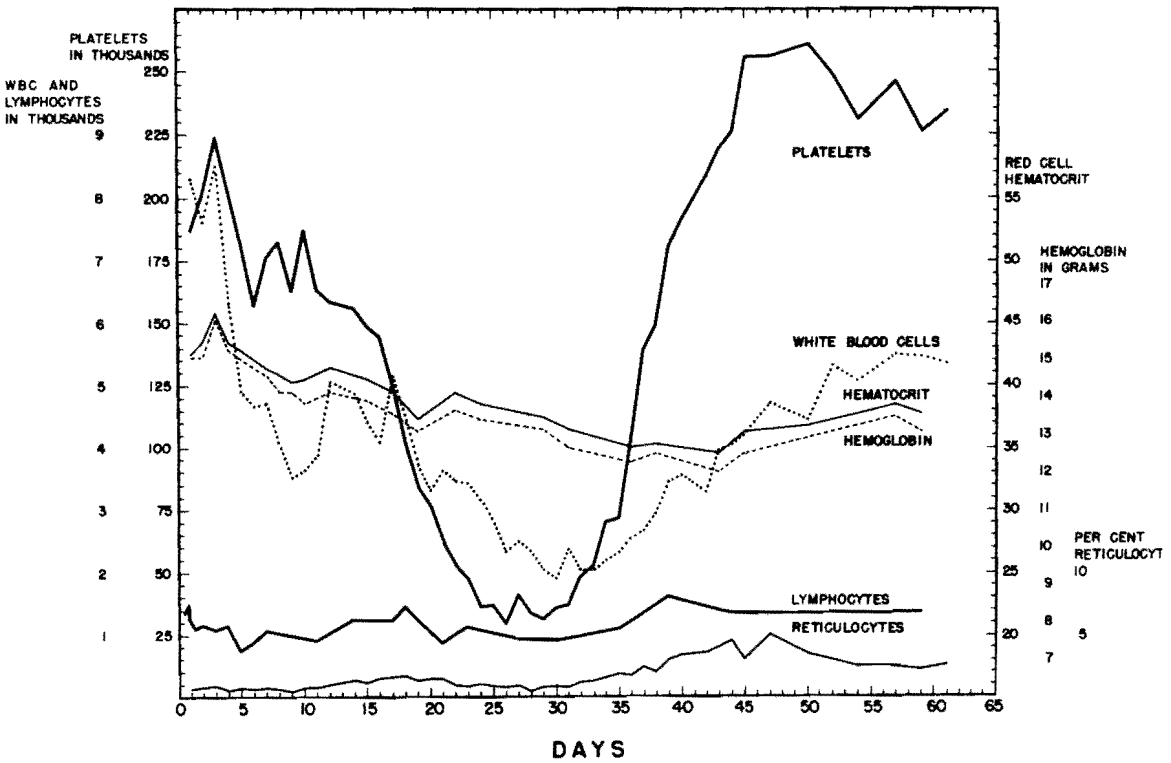


Fig. 15. Average Hematologic Changes of Five Persons Involved in a Radiation Accident. Dose ranged from 236 to 365 rads.

slowly developing, irresversible types of marrow depression, it is sometimes not realized that there is such a striking and rapid spontaneous recovery after acute irradiation. Undoubtedly acute irradiation, if the dose is high enough, can produce irreversible bone marrow damage, but this probably occurs only at dose levels that cause the gastrointestinal or central nervous system syndromes. When a patient dies of hematologic syndrome (that is, of hemorrhage or infection from three to six weeks after exposure to acute radiation), it does not necessarily mean that he had truly irreversible bone marrow damage. It would be more correct, in general, to consider the death as a result of failure to survive the period of temporary marrow depression.

Patients with the hematologic syndrome are likely to develop epilation at around two to three weeks after exposure, most commonly from about the thirteenth to seventeenth day. It may be associated with soreness of the hair-bearing areas. Epilation will of course be more prominent in patients who receive their radiation from low-energy sources that give a relatively higher dose to the superficial tissues. With $Co^{60}$ irradiation of patients, when the duration of exposure extends from one hour to a few hours, it appears that the threshold dose for epilation is around 300 r.

Gastrointestinal Syndrome. The gastrointestinal syndrome is poorly understood in the human being because relatively few patients have shown it. It is generally complicated by development of the hematologic syndrome, although, if severe, it may produce death before hematologic manifestations become very prominent. It is characterized by persistent nausea, vomiting, and diarrhea, by electrolyte imbalance, and sometimes

by infection resulting from invasion of bacteria through the mucosa of the intestinal tract. It was seen in one of the early Los Alamos cases[12] and in the Yugoslav worker who died,[2] although in both these patients the syndrome was complicated by other features. In clinical experiences with the uses of total-body irradiation, gastrointestinal damage has not been very prominent, even at dose levels up to 800 and 900 r. This may suggest that the gastrointestinal tract in the human being is less sensitive than was formerly thought. It has generally been stated that at doses above 500 r, but too low to produce the central nervous system syndrome, the gastrointestinal syndrome would predominate.

Central Nervous System Syndrome. Little is known about the central nervous system syndrome in the human being. Most of the assumptions about it are based on animal experimentation. However, in the latest Los Alamos accident,[4] the patient did have pronounced neurological manifestations, as well as other changes. In that case, because of uneven distribution of the radiation dose, the injury was difficult to classify. There were evidences of severe immediate electrolyte loss, gastrointestinal disturbance, and cardiac damage, the heart having been at a point of very high radiation dosage. It is generally believed that, at doses above about 2000 r, the central nervous system form of the acute radiation syndrome could be expected to develop within a few hours after exposure and that it would produce death in one to four days.

## Diagnosis

The problem of diagnosis of radiation injury usually resolves into an effort to determine the dose or the degree of severity of injury. Occasionally, however, radiation injury occurs under circumstances in

which no one has knowledge that any exposure at all has occurred. In this situation it may be very difficult for a clinician to recognize the nature of the trouble, for the features of radiation injury are not especially characteristic. Nausea and vomiting may be attributed to any number of causes, erythema may be mistaken for sunburn, and even the later developing hematologic changes may be erroneously attributed to a primary bone marrow disease or some other damaging agent. This situation makes it very clear that monitoring instruments, film badges, etc., are of the very greatest importance in signaling the occurrence of radiation injury. Once radiation injury has occurred, it is also desirable to use these physical measures to establish dose, although biological indices will prove of great value, particularly at the higher dose levels.

Changes in the blood count still seem to be by far the best clinical laboratory indicators of radiation injury. This is partly because they have been most extensively studied, but more particularly because the blood-forming tissue is one of the most sensitive, if not the most sensitive, of all tissues, and because the blood values are quite suitable for quantitative determination. The mechanisms by which the changes in the peripheral blood picture are caused are exceedingly complex. The number of any given cells in the circulating blood depends upon several factors: the rate of formation (in bone marrow for most cells; in lymph nodes, spleen, and marrow for lymphocytes), the storage of cells after formation, the rate of release of cells into the blood stream, and the removal of cells by several mechanisms (destruction or passage outside the blood stream into extravascular tissues). Any of these mechanisms may be altered by radiation. Probably the most important effect, however,

is interference with the production of new cells in the bone marrow and lymphatic tissues.

The fall in lymphocytes is believed to be the most valuable index of radiation injury. It is a very sensitive index and develops quite early. It has sometimes been said that, since it is so sensitive, the differences in lymphocyte levels are not very valuable at high radiation doses.

Reticulocytes are of some value as an index of radiation injury. Complete disappearance by four or five days after irradiation suggests a very high radiation dose. However, the failure of the reticulocytes to fall to exceedingly low levels is not necessarily evidence that there has not been a very dangerous dose. The percentage of reticulocytes normally present is quite low, i.e., 0.5 to 1.5%, and when an attempt is made to look at the significance of changes between 1% and zero, it is exceedingly important to count several thousand cells to obtain a statistically valid figure.

The granulocyte levels are of great importance. If they fall very rapidly, a high radiation dose is indicated; however, the granulocyte levels may fluctuate within normal or near normal levels for a period of two weeks and still fall quite low at the point of maximum depression. A pronounced rise in granulocyte levels within a few hours after the accident may be a very unfavorable prognostic sign. In cases where the accident is complicated, however, by local injury or excessive local radiation dose, this may become a less valuable index of total-body effect. The total white count reflects mainly the granulocyte level, although

during the point of greatest depression there may be more lymphocytes than granulocytes present.

Platelet values tend to follow the granulocyte levels, although the rise within a few hours after radiation is not so definite or, at least, not so well documented. During the period of greatest marrow depression, platelet values are of great importance in indicating the severity of injury and the need for treatment.

Red cell values fluctuate early because of dehydration but show their lowest values, related to marrow depression, several weeks after radiation injury. In general, red cell values at this point are an index of dosage, but they do not show the effect early enough to be of much value at the time when one is most interested in making a biological assessment of injury.

It is of interest that the greatest marrow depression occurs a good many days after the radiation injury. Consideration of the known life spans of individual cell types, particularly granulocytes and platelets, in relation to the time of postradiation depression, indicates that the interference with formation is not greatest immediately after radiation but, rather, at a later date. This suggests perhaps that only after several generations of cell precursors is the maximal damaging effect of radiation apparent. Whatever the mechanisms may be, the laboratory findings are useful in indicating radiation injury. Over a rather wide sublethal dose range the intervals from time of radiation to time of greatest depression of the granulocytes, platelets, and red cells tend to remain about the same. With increasing doses within this range, however, the depression begins earlier. Recovery does not begin earlier with increasing

dose, but, if anything, is delayed longer so that the time of maximum depression remains quite stationary over a wide range of radiation dose, but the duration of severe depression increases with dose. The rate of development of marrow depression may seem to be similar in human beings and small animals if higher doses in the human being are compared with relatively lower doses in the small animals. If, however, comparable doses for each species are used, it will be apparent that in the smaller animals the injury and the recovery both develop at earlier time intervals.[8] There are, however, some exceptions to this general rule.

Treatment

Treatment may be exceedingly important in patients who have hematologic manifestations of radiation injury.[13] Most of the efforts to treat the gastrointestinal and central nervous system manifestations are at the present time fruitless, and patients irradiated at very high levels can be given only symptomatic help. Early shock and electrolyte loss may be corrected by replacement therapy. Sedatives and anticonvulsants may be given as needed. The really important forms of treatment, however, which make a difference between life and death, are those directed toward the hematologic syndrome. These forms of treatment either help to correct the bone marrow depression (marrow graft therapy) or help the patient to survive the period of marrow depression until his own regenerative powers come into play. These latter measures have to do with combating hemorrhage and infection.

Combating Hemorrhage. The most important factor, if not the sole factor, in the hemorrhage of radiation injury is the lack of platelets. When the platelet levels become exceedingly low, evidences of hemorrhage

at various sites appear; these are similar to hemorrhages in patients with certain blood diseases and the hemorrhages that occur after marrow-depressing drug therapy. The most important forms of treatment are the use of transfusions of platelets or fresh whole blood handled so as to preserve platelets and, in some cases, the use of bone marrow grafts. It is difficult to know just when platelet treatment should begin, since there may be some disadvantage in starting too early as a result of the development of antibodies against platelets. Whether this is an important problem in irradiated patients, who presumably have a depressed immune mechanism, is not entirely clear. Most workers believe that it is not necessary to start platelet transfusions until some clinical manifestation of bleeding begins to develop; that is, a low platelet count alone is not usually considered a reason for giving platelets. When it does become necessary to give platelets, it is desirable to give relatively large amounts. If patients who have thrombocytosis associated with polycythemia or other disorders are available, they may make ideal donors. Other methods of controlling bleeding are of limited value. It is, however, worthwhile to use antihypertensive drugs if there is high blood pressure, and local measures in the mouth and nose may avoid trauma that could lead to bleeding. A bland diet and avoidance of constipation may also help.

Infection. Infection is a prominent cause of death during the period of pancytopenia. The use of a sterile environment has been developed to a very high degree in the handling of irradiated patients who receive kidney grafts. Such a facility is expensive and difficult to manage but can successfully reduce infections acquired from outside sources. The

danger from infections already present in the irradiated patient must also be given consideration; such danger cannot be avoided by the use of a sterile-environment facility.

The other main treatment is the use of antibiotics, but because of drug-resistant organisms, great problems are presented in proper management of antibiotics. It is generally believed that antibiotic therapy should not be used routinely simply because of a low leukocyte count but, rather, should be withheld until some evidence of infection develops. On the other hand, in patients with severe marrow depression, evidences of infection may be quite obscure; that is, they are not the usual inflammatory and exudative reactions. The presence of high fever may be in itself an indication for antibiotic therapy, although fever has been said to occur in irradiated patients who do not have evidence of infection. It is believed desirable to have very careful bacteriologic studies, including blood cultures, in planning antibiotic therapy. The choice of drugs should be made on the basis of the organism present. When urgent therapy is needed before bacteriologic identification can be achieved, a broad-spectrum antibiotic may be used. Chloramphenical has sometimes been avoided because it may be a marrow depressant, but most workers believe that when there is a real need for this antibiotic it should be used in spite of this possible disadvantage. Drugs directed against fungus infection are of great importance, and there may be more justification for using them prophylactically than there is for using other antibiotics prophylactically.

Bone Marrow Grafts. Extensive work with animals has shown that a graft of bone marrow will carry the animal through an otherwise lethal dose of radiation, and similar treatment of human patients has been tried.

It now appears that most of the therapeutic attempts to use bone marrow grafts in acute leukemia have not been characterized by true "takes," although some patients have benefited and have gone into remissions as a result of irradiation.[6,7] It has been shown that isologous grafts, that is, grafts between identical twins, are successful in human beings.[14] On the other hand, homologous grafts appear to be exceedingly difficult to achieve, and it is difficult to cite clear-cut examples of successful ones. Mathé, however, in treating acute leukemia, has reported grafts followed by what appeared to be secondary disease or foreign bone marrow reaction, that is, an immunologic reaction between the donor and recipient tissues.[15] An isolated report of an apparently successful graft under conditions that would not seem especially favorable for a graft has been published.[16] This graft involved a patient with Hodgkin's disease who had not received total-body irradiation but who had shown evidence of proliferation of donor cells for many months.

The only effort to use marrow graft therapy after a radiation accident was that of Mathé and his co-workers, who treated the Yugoslavian accident victims.[2] This trial of marrow graft therapy was not made until rather late in the period of marrow depression at a point when spontaneous recovery might have been about to develop. There was evidence that a temporary homologous graft was produced in 4 patients,[17] but there has been differences of opinion about the true value of this therapeutic attempt.[18,19]

Most investigators working in this field believe that homologous grafts should be attempted after a radiation accident if the dose is deemed to be probably lethal but not so high as to produce death within the first few days.[11] The severity of injury should be judged mainly by comparison of

clinical and hematologic changes with those in previously reported accidents. It would probably not be of value to give marrow within the first four days or so, but it is probably not wise to wait three or four weeks as was done in the Yugoslavian accident.

## Conclusion

The experiences of recent years have done much to clarify the understanding of radiation injury, and the work in this area has been exceedingly fruitful in yielding information of basic biologic significance. While less is known of the symptoms and effective treatment of radiation exposures in excess of 800 r which produce the gastrointestinal or central nervous system syndromes, the work at radiation levels less than this is most important, not only because it is more common, but also because it is subject to therapy. Treatment of patients manifesting the hematopoietic syndrome (from total body doses of around 100 to 200 up to 600 to 800 rads) involves marrow graft therapy and preventative measures to help the patient combat such secondary effects as hemorrage and infection. It is also significant in view of the use made in radiation accident analysis of once-in-a-lifetime exposures that there is a rather wide range, probably up to at least 200 rads, at which the dangers of an acute lethal effect are minimal.

# SURFACE DEPOSITION OF AIRBORNE MATERIAL

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The removal of pollution from the atmosphere is a biological and geophysical necessity for life. Dust, ash, and gases produced by natural processes and by man would have accumulated, in spite of the diffusive capacity of the atmosphere, to an intolerable extent if a relatively efficient removal process were not continually active. Most of the air's burden of pollution is deposited, and retained, on the earth's surface. This is true even for extremely small particles with negligible gravitational fall rates and even for many gases. In fact the removal rates may be orders of magnitude more rapid than might be expected from purely gravitational settling. This "deposition" phenomenon is particularly significant in relation to environmental radioactivity, since it obviously

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reduces direct exposure from airborne material but creates a comparatively permanent ground-level source of direct radiation and a secondary exposure mechanism through plant uptake and later ingestion by animals and man. Some of the formulas for estimating deposition and determining the effects have been pointed out in other publications.[20-22]

The physics of the actual removal and retention processes for small particles and gases is complex and interesting but not too well understood. From a practical point of view, however, the term "deposition" can be used as a convenient way to characterize the entire complex of physical phenomena which may result in removal of gases or particles at the air-ground interface. Among these are: gravitational settling, adsorption, particle interception (impaction), diffusion, and possibly chemical and electrostatic effects. An excellent summary of the various meteorological analyses of deposition is given by Pasquill.[23] His review concludes with the following straightforward definition of deposition velocity, $V_g$:

$$V_g = \frac{\text{amount deposited per cm}^2\text{ of horizontal surface per sec}}{\text{volumetric concentration above this surface}} , \qquad (1)$$

which was suggested by Chamberlain,[24] who gave some experimentally determined values of $V_g$ that have subsequently been widely used in reactor hazards analyses and environmental studies.

Deposition Measurements

Several important series of deposition experiments have been reported recently that considerably add to the available information on the subject. Chamberlain[25] has summarized a number of Harwell studies. In these experiments, the nature of the surface on which the deposition occurred was

found to be quite important. Four kinds of surfaces were considered: grass taken from a known area of the ground; filter paper placed in a Petri dish on the ground; leaves of dandelion or clover growing in the experimental area; and paper "leaves" attached to dandelion or clover. The deposition to natural and paper leaves was measured in a wind tunnel. Table V-1 summarizes the results of these experiments, in which $I^{131}$ vapor was the material deposited. It may be seen that for all these experiments, $V_g$ is of the order of 1 cm/sec. This is in agreement with the results reported by Healy et al.,[26] who measured deposition of $I^{131}$ to distances of 30 km from a source at Hanford and found an average deposition velocity of 2.8 cm/sec.

Chamberlain[25] has also reported measurements of the deposition of various active nuclei whose diameters lie in the submicron size range on flat plates about 1 $ft^2$ in area that were exposed horizontally. The deposition velocity data presented by Chamberlain are summarized in Table V-2. The small particles (diameters estimated to be 0.02 to 0.2 $\mu$) used had deposition velocities about an order of magnitude below those of $I^{131}$ vapor. It is interesting that they deposited with nearly equal ease on both upward- and downward-facing surfaces. A theoretical discussion of this and other interesting physical aspects of the deposition problem has been presented by Owen.[31]

Simpson[32] at Hanford and Islitzer and Dumbauld[33] at the National Reactor Testing Station have recently studied the deposition of small, fluorescent particles by a material-balance technique. This involves an attempt to measure the total transport of tracer material over a dense air-sampling grid, including numerous towers, which are needed to determine air concentrations aloft. The vertical and horizontal air-concentration

Table V-1. SUMMARY OF RESULTS OF DEPOSITION EXPERIMENTS WITH $I^{131}$ VAPOR[a]

| No. of trial | 1 | 2 | 3 | 4 | 5 | 6 | 7 |
|---|---|---|---|---|---|---|---|
| Date | 5-8-49 | 8-8-50 | 7-3-51 | 8-8-51 | 8-8-51 | 6-2-59 | 7-13-59 |
| Time (G.M.T.) | 1620 | 1500 | 1100 | 1500 | 2000 | 1340 | 1340 |
| Windspeed, cm/sec | | | | | | | |
| At z = 200 cm | 520 | 433 | 515 | 408 | 165 | 230 | 387 |
| At z = 100 cm | 420 | 372 | 442 | 310 | 138 | 164 | 335 |
| Temperature, °F | 50 | 65 | 68 | 69 | 61 | 67 | 61 |
| Relative humidity, % | | 61 | 75 | 51 | 78 | 51 | 63 |
| Weather | Sunny | Cloudy | Sunny | Sunny | Dusk | Sunny | Cloudy |
| Iodine literated, mg | <1 | <1 | <1 | <1 | <1 | 50 | 50 |
| Distance of sampling points from source, m | 15 | 20 | 20 | 20 | 20 | 50–100 | 50–100 |
| Vegetation cover at two sites, $mg/cm^2$ | 50 | 20 | 26 | 42 | 42 | 108 61 | 36 42 |
| Velocity of deposition, cm/sec | | | | | | | |
| To ground | 1.9 | 2.6 | 1.8 | 3.7 | 1.7 | 1.7 1.8 | 1.1 1.1 |
| To leaves | | | 1.3 | 0.9 | 0.5 | | 0.3 0.3 |
| To paper "leaves" | | | 2.0 | 1.5 | 1.0 | | 0.6 0.6 |
| To paper in Petri dish | 0.6 | 0.7 | 0.9 | 0.6 | 0.3 | | |

[a]Adapted from Ref. 24.

Table V-2. DRY DEPOSITION OF ACTIVITY ON NUCLEI ON HORIZONTAL SURFACES[a]

| | 1953 | 1957 | 1959 | 1957 |
|---|---|---|---|---|
| Activity source | Fission products; RaB and RaC | Fission products | $Zr^{95}$ | Fission products in field experiments |
| Surface | Filter paper | Gummed paper | Filter paper | Filter paper and grass |
| Velocity of deposition, cm/sec | 0.1 for fission products<br>0.5 for RaB and RaC | 0.07 | 0.10 | 0.07 on upward-facing surface<br>0.04 on downward facing surface<br>0.20 on grass |
| Reference | Bullas[27] | Stewart[28] | Booker[29] | Megaw and Chadwick[30] |

[a]Adapted from Ref. 25.

distributions are measured and compared at successively greater downwind distances; and any decrease in the mass flux of material is attributed to deposition on the underlying surface. Simpson used zinc sulfide pigment particles having a mean diameter of 2.5 μ. Islitzer and Dumbauld computed an average $V_g$ value of 0.5 cm/sec for the four runs reported by Simpson; all runs were made under stable meteorological conditions. It is remarkable that such a deposition velocity can evidently result in removal after 3200 m of travel of as much as 90% of the material emitted from a source at ground level.

The fluorescent particles used by Islitzer and Dumbauld had a medium diameter of 1 μ. Their computed values of $V_g$ are given in Table V-3. The interesting conclusion to be drawn from this table is that, whereas values of $V_g$ in strong inversions (test O and Q) agree with Simpson's finding for stable situations, the deposition velocities in the remaining cases, which vary from near neutral to highly unstable, range up to more than 20 times

Table V-3. DEPOSITION VELOCITIES COMPUTED FROM TRACER MATERIAL-BALANCE MEASUREMENTS[a]

| Test | Mean Wind Speed at Elevation of 4 m (m/sec) | Temperature Difference Between Elevations of 16 m and 4 m (°F) | Travel Distance (m) | Amount Deposited (%) | $V_g$ (cm/sec) |
|---|---|---|---|---|---|
| C | 6.3 | −0.3 | 400 | 54 | 9.2 |
| D | 4.9 | +0.5 | 400 | 35 | 2.3 |
| E | 6.0 | −2.4 | 200 | 32 | 8.0 |
| F | 4.7 | −1.1 | 200 | 24 | 4.4 |
| G | 6.2 | −1.9 | 200 | 22 | 5.4 |
| I | 6.0 | −1.6 | 400 | 48 | 6.8 |
| M | 3.9 | −0.7 | 400 | 32 | 2.4 |
| N | 4.7 | +0.7 | 400 | 38 | 2.6 |
| O | 2.6 | +3.4 | 400 | 10 | 0.2 |
| Q | 2.5 | +4.1 | 400 | 14 | 0.2 |
| S | 8.4 | −2.3 | 400 | 37 | 8.9 |

[a]Adapted from Ref. 33.

the stable values. Islitzer and Dumbauld also actually measured the deposition of fluorescent material onto flat, sand-covered plates, 0.1 $m^2$ in area. For unstable conditions, these measurements gave values of deposition velocities over an order of magnitude smaller than the values computed by the material-balance technique. This presumably reflects the difference between bare sand and sagebrush-covered desert, as far as removal of material from the air stream is concerned.

Additional experience with deposition of radionuclides has been reported by the U. S. Air Force[34,35] as a result of experiments carried out by Convair. The data in Table V-4 are from the first of these tests.[34] The values of Table V-4 were calculated from measurements of air concentration and deposition at from 100 to 3200 meters from the source. In a later experiment, FRT-II,[35] data was collected over much greater distances of

Table V-4. DATA FROM FISSION PRODUCTS FIELD RELEASE TEST I[a]

| Nuclide | Deposition Velocity (cm/sec) | | |
|---|---|---|---|
| | On Sticky Paper | On $H_2C$ | On Sand |
| $Cs^{137}$ | 0.096 ± 0.055 | 0.88 ± 0.063 | 0.04 ± 0.019 |
| $I^{131}$ | 1.1 ± 0.35 | 1.6 ± 0.65 | 0.74 ± 0.47 |
| $Ru^{103}$ | 2.0 ± 0.98 | 2.3 ± 1.0 | 0.60 ± 0.38 |
| $Zr^{95}$-$Nb^{95}$ | 1.4 ± 0.73 | 5.7 ± 3.4 | 2.9 ± 2.7 |
| $Ce^{141}$ | 0.73 ± 0.63 | | |

[a]Adapted from Ref. 34.

from 1000 to as far as 32 000 meters. This latter work was all performed at night during strong inversions. Except for $I^{131}$ on grass, the data were often widely scattered. Table V-5 shows the averages and ranges of $V_g$ for these experiments.

The Windscale accident has been analyzed in detail in companion papers by Chamberlain[36] and Crabtree.[37] The $I^{131}$ deposition velocities tabulated by Chamberlain range from 0.51 to 0.24 cm/sec, with a mean value of 0.36 cm/sec. Chamberlain originally proposed a vapor-diffusion mechanism for the deposited iodine; but the order of magnitude difference between the deposition velocities observed in comparison with Chamberlain's own experiments, as well as the Hanford and FRT data, indicate that another mechanism may have operated. Since there was an intense graphite fire, and presumably therefore an abundance of small-diameter (<10 μ) carbon particles, it is possible that iodine vapor was first absorbed on the carbon and it then diffused and deposited at the slower rate more nearly characteristic of small particles, as Chamberlain[25] indicates.

Table V-5. DATA FROM FISSION PRODUCTS FIELD RELEASE TEST II[a]

| Nuclide | Deposition Velocity (cm/sec) | | |
|---|---|---|---|
| | On Sticky Paper | On Soil | On Grass In Trays |
| $Cs^{137}$ | 0.21 ± 0.27 | 0.045 ± 0.053 | 0.19 ± 0.17 |
| $I^{131}$ | 0.28 ± 0.39 | | 1.53 ± 0.59 |
| $Ru^{103}$ | 0.34 ± 0.31 | 0.113 ± 0.178 | 0.57 ± 0.42 |
| $Te^{127n-129m}$ | 0.71 ± 1.3 | | |

[a]Adapted from Ref. 35.

The SL-1 accident[38] also resulted in emission of $I^{131}$ at a steady rate for several weeks. Islitzer[39] reports the deposition velocities listed in Table V-6, based on measurements of air concentration and deposition on foliage (sagebrush), that resulted from the extensive field surveys made following this incident. The values of Table V-6 are quite interesting both because of their agreement with the Windscale values and because of the constancy of the values as a function of distance from the source.

A study of the deposition of glass spheres averaging 50 and 100 μ in diameter has been reported by Hage.[40,41] These interesting results were not accompanied by measurements of air concentration (i.e., the denominator of Eq. 1) and so cannot immediately be converted into $V_g$ values. They have so far primarily been used in testing theoretical diffusion-deposition models.

## Conclusions

Several interesting conclusions can be drawn from the preceding data. A clear distinction has been illustrated between the rate of removal of

Table V-6. DEPOSITION VELOCITIES OF $I^{131}$ RELEASED BY THE SL-1 ACCIDENT

| Distance (km) | Date | Concentration (curies/m) | Amount Deposited ($\mu c/m^2 \cdot day$) | $V_g$ (cm/sec) |
|---|---|---|---|---|
| 1 | January 24, 1961 | $6.0 \times 10^{-11}$ | $1.3 \times 10^{-2}$ | 0.25 |
| 8.5 | January 4, 1961 | $3.6 \times 10^{-11}$ | $6.7 \times 10^{-3}$ | 0.21 |
| 67 | January 9, 1961 | $4.0 \times 10^{-12}$ | $7.8 \times 10^{-4}$ | 0.23 |

chemically active material and of inert material of similar size; vegetation, specifically grass and sagebrush, brings about removal at rates an order of magnitude larger than those measured on bare soil or flat-plate collectors. Table V-7 summarizes these conclusions.

Almost equally important is what these experiments do not show. We do not know, for example, what effect a forested or cultivated area would have on the interception of airborne material. The difference between deposition during inversion and lapse conditions is not well defined. Practically no data are available on deposition or retention on buildings and houses. Finally, there is no real classification of important radionuclides by deposition velocity, except for $I^{131}$ and $Sr^{90}$ from bomb debris.

Further experimental work with tracers is planned to obtain some of the desired information. The Weather Bureau program already under way at the National Reactor Testing Station has been mentioned above, and the Hanford fluorescent particle data are being examined further. An extensive deposition research program is also under way at the Brookhaven National Laboratory, as reported by Smith.[42] Uranine dye and radioactive-copper

Table V-7. AVERAGE DEPOSITION VELOCITIES FOR ACTIVE AND INERT MATERIAL

| Material Removed | Deposition Velocity (cm/sec) | |
|---|---|---|
| | On Flat Plates or Bare Soil | On Vegetation (Grass, Sagebrush) |
| Active material: $I^{131}$, $SO_2$, Ru | <1 | 1–3 |
| Inert material: $Cs^{137}$, $Sr^{90}$ | <0.1 | 0.1–0.2 |

particles are being emitted under carefully controlled conditions and sampled, by a variety of techniques, on an extensively instrumented experimental plot. The results of these should greatly augment the available information on deposition.

## IN VIVO RADIOACTIVITY MEASUREMENT

The technique of determining the amount and type of radioactivity within the human body by measurement of the radiation emitted by the internal radioactivity and emerging from the body has been used extensively since 1950. In addition to application to tracer studies of the metabolism of radioisotopes in the human body, in vivo radioactivity-measurement (IVRM) techniques have importance in applied health physics, in the routine monitoring of personnel to ensure that dosage from internal emitters is within the prescribed range, in the evaluation of dosage from accidental inhalation, ingestion, or deposition of radioisotopes, and even in assessing the intensity of neutron activation of the body.[43,44] The purpose of this review is to outline the history, techniques, problems, and limitations of IVRM as an aid and factor in nuclear safety.

In recognition that the determination of amount and type of radioactivity within the human body by direct measurement of the radiation emanating from the body constitutes a "distinct problem in the measurement of radioactivity,"[45] the phrase "in vivo radioactivity measurement" will be applied to the problem in this discussion. This title for the field is perhaps less familiar and popular than other nomenclature, but it seems the most descriptive of the various names used thus far and has been used by others.[46,47] The term "in vivo radioactivity measurement (or IVRM)" indicates clearly that the radioactivity to be measured is within a living body, but it does not erroneously suggest that the radioactivity is of necessity throughout the body, as do the adjectives "total body" or "whole body." Indeed, a most useful feature of some IVRM

techniques is that the measurements may be restricted to certain body portions and organs whose radioactivity contents are of special importance.

By definition IVRM techniques cannot be used to measure internally deposited radioactivity if the radioisotope does not emanate radiation to the exterior of the body. Ultimately, it may be possible to directly measure pure alpha or beta emitters, or at least the radiation dose from such emitters, by placing detectors within the human body,[48,49] but such technology has not yet been developed to a routine, useful state. The most practical application of IVRM methods, then, is to the measurement of isotopes that emit gamma radiation in the process of their disintegration. Beta emitters that do not produce line gamma spectra may be subject to in vivo measurement through detection of the internal or external Bremsstrahlung associated with such emitters.[44,50]

## Historical Review

Recognition of the hazard of radioactivity inside the human body was followed closely by efforts to assess directly the quantities of radioactivity within the living body. Indeed, interest in measuring the amount of radioactivity which had been deposited in the human body probably preceded the awareness of the hazard of internal emitters, as evidenced by a quotation from a 1905 publication: "J. Elster and H. Geitel give the results of testing the breath of Herr F. Geizel, the well known radium physicist. He was asked to blow air into the apparatus some 18 hours after he had left his laboratory. The result was a degree of ionization distinctly above the normal. The authors found that Herr Geizel was himself a radioactive body, the substance having irradiated all of his tissues.[51]

Reports between 1925 and 1931 of the injury or death of numerous persons because of internally deposited radioactive substances, particularly materials used in the self-luminous dial-painting industry,[52] prompted an interest in determining the quantity of radium deposited in humans by measuring the intensity of the gamma radiation emanating from the human body. The use of the Wulf-Hess quartz fiber electroscope to measure the in vivo radium deposits was reported by Schlundt, Barker, and Flinn in 1929.[45] Flinn states that he began such measurements in 1925.[53]

The introductory paragraphs of these early works state facts which still characterize efforts to directly measure in vivo radioactivity. Experienced present-day investigators, faced with the proliferated list of radioisotopes of interest in the human body, would agree with the 1929 statement: "It is clear that the detection of radium in persons suffering from 'radium poisoning' presents a distinctive problem in the measurement of radioactivity."[45]

During the 1930's, direct measurement of body radioactivity was greatly advanced by the work of R. D. Evans with cases of "radium" ($Ra^{226}$ and $Ra^{228}$ sub-series) poisoning.[54] Two aspects of Evans' work are particularly important. First, pulse-counting techniques were substituted for electroscope methods, with an estimated sensitivity increase of 10 to 100. Second, Evans devised the "arc" position, in which the patient's body was curved on the arc of a circle with the detector at its center. This detector-patient orientation greatly reduced the variations in counting geometry due to differences in patient body size.

In the broad field of radium poisoning, Evans has discussed in detail the radiological physics of $Ra^{226}$ and $Ra^{228}$ ingestions in a section

of his exhaustive paper on the effects of radioactive materials (primarily radium isotopes and daughters) in man.[52]

In contrast to this basic approach of inspecting the body with a small detector volume either at localized areas or at a distance sufficient to collect radiation at a low geometric efficiency from all the body, there has been and remains a considerable interest in the alternate basic approach of a large radiation-detecting volume completely enveloping the human body. These divergent but complementary techniques have become quite clearly separated as scintillation detectors have developed, but it should be pointed out that the large-counting-volume scheme preceded scintillation detectors. Multiple high-pressure ionization chambers surrounding the body were developed in Europe and are still listed as in use in Sweden and Australia.[55] Sievert suggested the use of multiple ionization chambers and built a first model with insufficient shielding and just enough sensitivity to detect the natural level of radiation from human subjects.[56,57] Subsequently, Sievert built two ionization chamber systems capable of accepting the human body and used a subway lined with tanks of water as a counting room. Rundo has described an ionization chamber system designed and built at Harwell in England and used in Denmark for study of thorium dioxide poisoning.[58,59]

Ionization chamber systems may be considered for use with large-volume liquid scintillation systems. The important feature of energy discrimination is, however, not available.

Development of Scintillation Detector Systems

The advent of scintillation-type radiation detectors since 1947 has advanced IVRM techniques to their present state and has resulted in two

major IVRM technologies, each having certain desirable characteristics and important limitations. These two technologies are the continuing parallel development of large and small counting-volume systems. The radiation energy spectrometry characteristics of scintillation detectors gives the small-volume crystal detector generally superior radioactivity identification capabilities that counter-balance the higher counting efficiency of the large-volume detectors.

Large-Volume Scintillation Devices. Large liquid scintillation detectors capable of containing the human body (and to which the term "total body" monitor can best be applied) were developed by E. C. Anderson and associates at Los Alamos Scientific Laboratory as an outgrowth of neutrino-detection experiments.[60,61] Subsequently, two other types of device specifically designed for IVRM purposes and described as 2 $\pi$ and 4 $\pi$ counters have been developed.[62,63]

The original Los Alamos human counter is horizontal, with 560 liters of terphenyl-POPOP in toluene liquid scintillator in a 6-in.-thick layer around an 18-in.-diam, 72-in.-long annulus for entry of the body. Shielded by a 5-in.-thick lead cylinder, this counter (Humco I) has a counting efficiency of 8.7% for $K^{40}$ in the human body and an energy resolution for $K^{40}$ (1.45 Mev) of 60%.[60] The vertical 2 $\pi$ model, into which the subject walks, has 280 liters of scintillator in the wall of a half-cylinder. This device (Genco or "Geneva Counter") has about 43% of the counting efficiency of Humco I, but a $K^{40}$ energy resolution of about 36%.[64] The third device (Humco II) returned to the horizontal, annular arrangement, but uses large (16-in.-diam) photomultipliers, a 12-in.-thick scintillator layer, and 1514 liters of liquid scintillator. This counter has a

resolution of about 40% for $K^{40}$. The initial preliminary reports on Humco II indicate that its improved performance over earlier models was not as great as had been expected.[63]

The major advantage of the large-volume liquid scintillation detectors is the short counting time required to measure small amounts of radioactivity in the body. This feature permits a high throughput of subjects. For instance, the natural potassium content of the human body can be determined with a precision of 2% in a 100-sec counting time.[61] The limitations of these counting devices are inflexibility in discriminating between sites of deposition in the body, only fair energy resolution, and poor scintillation and light collection efficiency below 200 kev.[63,65]

Small-Volume, High-Resolution Scintillation Devices. IVRM techniques using small-volume scintillation detectors with good energy resolution have been developed by L. D. Marinelli, C. E. Miller, J. E. Rose, and others in the Radiological Physics Division of Argonne National Laboratory. The broad scope of these important advances in IVRM has been reported in three inclusive papers.[44,46,66] The contributions of the ANL work may be summarized as follows:

1. Substitution of efficient, good-energy-resolution, NaI(Tl) scintillation detectors for the gas-ionization counters previously used.[67]
2. Use of relatively large-volume crystal scintillators to minimize the partial energy-transfer count rate, thus enhancing the spectrometric capabilities.[68,69]
3. Reduction of background counting rates through development of "iron room"-type shielding and detector-background studies.[68]

4. Development of the "tilted chair" counting position as an improvement over the low efficiency 1-meter-arc position.[69]
5. Discovery of $Cs^{137}$ in the in vivo gamma spectrum of humans.[70]

The original IVRM facility at the Argonne National Laboratory, which has become a model for similar systems throughout the world, was a shielded counting room, or "iron room," 7 ft square and 6 ft high, with 8-in.-thick walls, floor, and ceiling. The basic scintillation detector inside this room was an 8-in.-diameter, 4-in.-thick, NaI crystal carrying a single 5-in.-diameter photomultiplier. Pulse-height data from the detector were sorted and stored by a multichannel analyzer outside the iron room. The detector yielded a 9% resolution of the $K^{40}$ photopeak and was sufficiently sensitive to indicate that a total-body burden as low as $10^{-9}$ curies of gamma-emitting isotopes could be detected in 10 min of counting time.[68]

Summary of Facility Types. It is of some interest to note the types of IVRM facilities which have been constructed, as the field has grown rapidly in recent years. Study of the data obtained in a preliminary review of IVRM facilities in International Atomic Energy Agency member states reveals that "iron-room" facilities that utilize NaI(Tl) scintillation detectors outnumber other types.[55] The distribution of types of equipment at the 64 facilities surveyed is summarized in Table V-7.

## Problems of in Vivo Radioactivity Measurement

In vivo radioactivity measurement is simplest if the radioisotope has a high gamma radiation abundance per nuclear disintegration, if the emitted gamma radiation has an energy higher than about 250 kev, and if the gamma energies are different from those naturally in the human body.

Table V-7. Distribution of Types of Equipment Used at IVRM Facility[a]

| | Total Number of Facilities | Number in United States |
|---|---|---|
| Shielding | | |
| Iron room | 48 | 28 |
| Lead room | 9 | 0 |
| Detectors | | |
| Multiple ionization chambers | 3 | 0 |
| Solid plastic scintillators | 4 | 1 |
| Large-volume liquid scintillators | 5 | 4 |
| NaI(Tl) crystal scintillators | 52 | 25 |

[a]Data from Ref. 55.

Of course, these ideals are seldom met. To indicate briefly and generally the problems encountered in IVRM techniques and some solutions to these problems, several broad procedural steps may be outlined as applicable, particularly to NaI(Tl) crystal spectrometry facilities.

1. The in vivo radioactivity must be detected. This requires that the detector be inherently sensitive and oriented so that it receives and registers a significant portion of the radiation from the total body or from the anatomical region of interest. As was noted in the historical review, large-counting-volume systems have been developed to totally enclose the human subject by the detector volume, while high-efficiency, low-volume crystal detectors have been carefully positioned relative to the subject body to achieve the desired counting geometry, i.e., from the whole body or from the body portion of interest.

For many radioisotope systems ($K^{40}$, $Cs^{137}$, bone-deposited radium, etc.), the detector geometry should include the whole body. In crystal spectrometry, these arrangements would include the "arc position,"[54,71]

the "tilted chair,"[69] multiple crystals distributed along the body length,[72] and recent integration-scanning techniques.[73]

For inhalation of insoluble radioactive materials, for radioactivity which accumulates in a single body organ, and for measurement of contamination in wounds, the capabilities of crystal detectors for being positioned very near the site of deposition to provide higher counting geometry and discrimination against radiation from other body sections are very important. Because of the high probability of inhalation of radioactivity and the difficulties of relating excreted radioactivity to insoluble lung deposits, in vivo measurement of lung burdens has been of considerable interest. Evaluation and application of the IVRM technique to a number of radionuclides in the lungs has been carried out at the Y-12 Plant in Oak Ridge.[74-76]

High-resolution scanning, comparable to the high-level tracer studies of nuclear medicine, has not been extensively exploited by IVRM facilities, probably because of the low counting rates from IVRM subjects. Miller has discussed the matter[77] and King and Mitchell have described proposed U. S. Naval Medical Department plans for IVRM facilities utilizing collimated scanning.[78] As noted above, Palmer and Roesch at Hanford have used uncollimated scanning instead of the arc positions for the measurement of low concentration, nonuniformly distributed radioisotopes.[73]

2. The counting rate contribution of environment and equipment must be reduced to a level that will permit measurement of in vivo radioactivity in the desired concentration in the presence of the statistical fluctuations of the background. Massively shielded counting rooms, cosmic ray coincidence "umbrellas,"[68] selected materials in detector packaging and

other equipment, purified atmosphere systems,[79] and careful anticontamination measures have been used to achieve this goal.

3. The radiation detected must be sorted according to energy. As a prerequisite, the detector must have energy resolution capability. Sodium iodide scintillation crystals have demonstrated the best energy resolution characteristics, with resolutions of about 9 to 10% for $Cs^{137}$ (660 kev) being obtained[80] on 8- to 9-in.-diameter, 4- to 6-in. thick crystals by the use of multiple photomultipliers. A multichannel pulse-height analyzer for the sorting and storing of the energy spectrum is almost a necessity because of the low counting rates of in vivo gamma spectrometry.

4. Care must be exercised in preparing the IVRM subject to eliminate surface or external radioactive contamination which can be mistaken for internal deposition. This problem is particularly acute when IVRM is used to monitor personnel continually engaged in handling radioactive material, as in the Y-12 Plant uranium IVRM program.[75] Careful bathing and provision of radioactivity-free clothing is required. Surface contamination meters (i.e., alpha detectors) are used to inspect the chest, hands, and arms of subjects for IVRM examination at Y-12. To a limited extent, the relative heights of two gamma peaks in the in vivo spectrum are indicative of the scattering and absorption effect from passage through body tissue and can be used to determine whether measured radioactivity is internal or external.[80]

5. The requirements above involve obtaining the vivo gamma spectrum. Several additional procedures are generally required before quantitative statements regarding the in vivo activity of the individual can be made.

In the first place, the measured spectrum will contain some activities which are present in all humans (primarily $K^{40}$, $Cs^{137}$, and a very small amount of $Ra^{226}$ and daughters). The $K^{40}$ (1.45-Mev gamma) content of an individual depends on body weight, fat-to-muscle ratio, age, and basal metabolism. The in vivo $Cs^{137}$ (1.660-Mev gamma) content has been a function of atmospheric $Cs^{137}$ contamination and dietary habit. Some of the gamma quanta of both of these emitters, in passing through the human body, undergo some scattering. Thus, the radiation incident upon the detector will not be the monenergetic radiation of $K^{40}$ and $Cs^{137}$. Because of this scattering or energy degradation and, therefore, regardless of the detector efficiency in converting incident gamma rays into a voltage pulse, the in vivo spectrum will contain the two full-energy photopeaks plus a continuum of scatter counts at energies below the 1.45-Mev and 660-kev energies. Obviously, difficulty is encountered if the abnormal gamma activity in the human subject has only an energy indistinguishable from that of $Cs^{137}$ or $K^{40}$. Even if the abnormal activity has an energy different from but lower than the energies of $Cs^{137}$ or $K^{40}$, sensitive IVRM techniques require that a correction be made for the $Cs^{137}$ and $K^{40}$ scatter contribution to the energy region of interest.

To provide this correction for the normal in vivo gamma activity, the ideal technique is to determine the normal in vivo spectrum before an internal contamination or deposition occurs. This method is suitable for tracer and metabolic studies but is usually not practical for health physics utilization because the normal in vivo spectrum is subject to change with time and because accidental contamination cannot be predicted.

However, this technique has been used in health physics personnel monitoring in a thorium-processing program.[74,81]

The usual approach is to subtract from the in vivo spectrum in question the in vivo spectrum of a control subject. Such controls must not have been exposed to internal contamination, must have $K^{40}$ and $Cs^{137}$ contents (as measured by IVRM) essentially equal to those of the examinee, and must have body size and shapes equal to those of the examinee (in order to equalize the scattering structures involved).

A third technique involves calculating the scatter contributions of $Cs^{137}$ and $K^{40}$ to the other spectral energy regions on the basis of the $Cs^{137}$ and $K^{40}$ photopeak amplitudes and body size and shape parameters. This method has been used successfully in estimating normal activity in the $U^{235}$ energy region (186 kev)[75] and deserves more development.

6. If the residual spectrum is complex because of the presence of more than one gamma emitter, it must be further reduced to determine the contributions of each radionuclide.

7. An in vivo spectrum corrected for the normally present activities must finally be related to quantity of activity in the human. The calibration factors utilized must be consistent with the body section chosen for monitoring. Even for the same radioisotope, different calibrations are required for a shallow wound, a lung deposit, a uniform total-body deposit, a uniform total-body deposition in soft tissues, or a skeletal deposit of a bone-seeking radionuclide. In some cases calibration can be achieved by administering tracer doses of a short-lived radionuclide. This technique was used by Miller to calibrate the ANL "tilted chair" for $K^{40}$, using $K^{42}$ as a tracer.[47] For the 1-meter arc, the use of a

Presdwood or untempered Masonite phantom containing a point source is considered standard.[71,82] The thickness of the absorbing-scattering material in front of and behind the source is varied to produce the same spectrum shape (ratio of peaks or area amplitudes in the spectrum) as that of the residual in vivo spectrum. This technique, with the incorporation of open spaces to simulate lung cavities, has been used for calibrations of lung deposits.[75,83] The recent availability of plastic-shelled manikins, contoured to body size and shape, containing organs, and filled with tissue-equivalent solutions, has made possible a quite versatile phantom for calibrating IVRM techniques.[84] At the Y-12 Plant, where lung deposition is a major IVRM interest, considerable study of calibration variability has been done by computer simulation of the human upper torso as a source of radioactivity.[85-87]

Application of IVRM

It is certainly not possible to detail all the applications of IVRM to specific radioisotopes or even merely the health physics application, ignoring tracer and metabolic utilizations. All the applications may only be found by careful perusal of the extensive IVRM literature, evidenced by the lengthy but incomplete bibliography accompanying this discussion and by other bibliographies.[88-90] Nevertheless, a tabulation of some applications has been made and is presented in Table V-8.

The applications listed are generally those for which enough data or description was reported in the reference to warrant the statement that an IVRM technique for the particular radioisotope system had been devised.

Table V-8. REPORTED IVRM TECHNIQUES FOR SPECIFIC RADIONUCLIDE SYSTEMS

| Radionuclide System | Radiation Measured | Deposition Site | Stated Sensitivity | | Reference |
|---|---|---|---|---|---|
| | | | Quantity | Percentage of Maximum Permissible Body Burden[a] | |
| $Na^{24}$ | 2.75-Mev gamma | Total body | 0.0025 μc | | 43 |
| $Co^{60}$ | 1.17-, 1.33-Mev gamma | Lungs | | | 91 |
| | | Lungs | 0.00066 μc | 0.056 | 76 |
| | | Total body | 0.001 μc (calculated) | | 72 |
| $Zn^{65}$ | 1.14-Mev gamma | Total body | <0.1 μc | | 44 |
| | | Total body | <0.0018 μc | | 73 |
| $Sr^{85}$ | 0.51-Mev gamma | Total body | 0.002 μc | | 83 |
| $Sr^{90}$-$Y^{90}$ | Bremsstrahlung | Total body | 0.02–0.01 μc (calculated) | | 50 |
| | | Total body | 0.05 μc (calculated) | | 72 |
| $Zr^{95}$-$Nb^{95}$ | 0.75-Mev gamma | Lungs | 0.001 μc | 0.062 | 76 |
| $Te^{129}$ | 0.46-, 0.68-Mev gamma | Lungs | | | 82 |
| $Cs^{137}$ | 0.662-Mev gamma | Total body | <0.003 μc | | 46,92 |
| | | Total body | 0.0015 μc (calculated) | | 59 |
| $Tm^{170}$ | 0.052-, 0.084-Mev gamma | Total body | <0.001 μc | | 92 |
| $Ra^{226}$ | 1.76-Mev gamma | Total body | 0.00006 μc | | 93 |
| $Ra^{226}$ + 30% daughters | | Total body | 0.008 μc (calculated) | | 72 |
| $Ra^{224}$ + daughters | | Total body | 0.0015 μc (calculated) | | 72 |
| $Th^{232}$ series | 0.24-, 0.58-Mev gamma | Lungs | 9 mg ($Th^{232}$:$Th^{228}$ = 2.2) | 18 | 76 |
| | | | 4 mg ($Th^{232}$:$Th^{228}$ = 1) | 11 | 76 |
| $Th^{228}$ | 0.24-, 0.58-, 2.6-Mev gamma | Total body | 0.00006 μc | | 93 |
| $Th^{228}$ + daughters | | Total body | 0.0015 μc (calculated) | | 72 |
| $Pa^{233}$ | 0.31-Mev gamma | Lungs | 0.00027 μc | 0.006 | 76 |
| U (natural) | 0.09-Mev gamma | Lungs | 0.001 g $U_3O_8$ | | 94 |
| $U^{235}$-enriched (93%) U | 0.186-Mev gamma | Lungs | 0.130 μg | 47 | 75 |
| $U^{238}$ ($U^{235}$-depleted U) | 0.06-, 0.09-, 0.186-Mev gamma | Lungs | 21.1 mg | 42 | 75 |
| $Np^{237}$-$Pa^{233}$ | 0.08-, 0.31-Mev gamma | Lungs | 0.00026 μc | 1.5 | 76 |
| $Np^{237}$ | 0.08-Mev gamma | Lungs | 0.0027 μc | 16 | 76 |
| $Np^{239}$ | 0.105-Mev gamma | Total body | <0.01 μc | | 44 |
| $Pu^{239}$ | 0.17-kev x-ray | Wounds | 0.0004 μc | 0.1 | 95 |
| $Am^{241}$ | 0.06-Mev gamma | Total body | 0.032 μc | | 83,92 |
| | | Lungs | 0.001 μc | | 96 |

[a]Based on Ref. 97.

## Conclusions

In vivo radioactivity measurement has grown into a complex and widely practiced technology since 1950. The application of gamma scintillation spectrometry, particularly with NaI(Tl) detectors, and reduction of environmental background counting rates have been major factors in stimulating this growth. At present, IVRM technology includes large-counting-volume systems which essentially envelope the human body and provide high counting efficiency but rather poor radiation energy discrimination, and solid crystal detectors which have low efficiency for counting radiation from the whole body but good radiation energy discrimination and high capability for measuring radiation from individual body sections or organs.

As a general statement of sensitivity, it may be concluded that quantities as small as $10^{-3}$ μc of radioisotopes which have gamma radiation of energy greater than 50 kev and gamma abundance of one gamma per disintegration can be easily measured in the human body. IVRM has been and can be successfully applied in less ideal specific cases by utilization of more advanced or more complex techniques.

The IVRM method has been applied to a long list of radioisotopes in the total body and in specific body organs. Since IVRM provides a direct measurement of body burden of radioactivity, independent of questions of solubility, excretion rate, etc., the technology is a valuable tool for all aspects of the study of radioactivity in living organisms and, particularly, in applied heath physics. (R. E. Cofield)

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# VI. CURRENT EVENTS

# ACCIDENTS IN NUCLEAR ENERGY OPERATIONS

Information on incidents that occur in nuclear energy operations is discussed in Nuclear Safety as it becomes available because of its particular interest to those concerned with nuclear safety. This article contains a review of three unrelated incidents which occurred in USAEC facilities in late 1961, as well as a recently issued AEC tabulation of like incidents. Incidents that occurred during the period July 1, 1958, to June 30, 1960, which involved nuclear equipment or activity release were reported previously.[1] Table VI-1 continues the chronological listing to December 31, 1960, based on material contained in a recent AEC summary report of all accidents in USAEC facilities.[2] The AEC report also contains a listing of inadvertent criticality situations in USAEC contractor plants that was previously included in Nuclear Safety.[3] All incidents listed that occurred after February 1957 have been separately reviewed in this publication.

A summary[2] of records of radiation exposures of up to 15 rem in Commission operations for the years 1959 and 1960 is presented in Table VI-2. In both years 99.9% of the contractor employees received less than 5 rem and 94.5% received only 1 rem or less. Exposures are kept to a minimum consistent with the benefits to be derived, and they average well within the limits for atomic energy contractor workers.

Three unrelated incidents that occurred at USAEC nuclear facilities in late 1961 are reviewed here because of their interest to those concerned with safety in nuclear facilities. They are (1) the SPERT III pressurizer-vessel failure, (2) an ORNL criticality excursion, and (3) an

Table VI-1. CHRONOLOGICAL LISTING OF ACCIDENTS IN AEC-OWNED NUCLEAR ENERGY OPERATIONS IN THE PERIOD JULY 1, 1960 TO DECEMBER 31, 1960

| Date | Location | Nature of Incident | Extent of Exposure |
|---|---|---|---|
| 7-6-60 | Mound Laboratory, Miamisburg, Ohio | Radioactive material was accidentally discharged into a room because the pressure buildup in a dry box blew out of one of the gloves | Eleven individuals received minor exposure and contamination of clothing, hair, hands, etc. |
| 10-4-60 | Station 311, Nevada Test Site, Mercury, Nevada[a] | Two employees were accidentally exposed to radiation from a 340-curie $Co^{60}$ calibration source | One person received a total body dose of 18 rem; the other received 5 rem |
| 11-8-60 | Sandia Corporation, Albuquerque, New Mexico | Two employees were accidentally exposed to an electron beam emanating from a Van de Graff accelerator | One individual received a total-body dose of 55 rads; the other exposure, while unmeasured, was estimated to be 760 rads to the face |
| 11-8-60 to 11-28-60 | National Lead of Ohio, Cincinnati, Ohio | Slightly enriched $UF_6$ lost up the stack of a dust collector | None |
| 11-10-60 | University of California, Lawrence Radiation Laboratory, Berkeley, California | Fire started by apparent over-heating of an oil bath in a glove box caused loss of entire contents of one room of a curium-processing cave | None |
| 12-21-60 | General Electric Company, Richland, Washington | Build up of pressure in a steam autoclave resulted in a blowoff that sent some of the contents of the autoclave through the building roof | None |

[a]Reviewed in Nuclear Safety, 3(1): 90 (September 1961).

Table VI-2. EXPOSURES OF AEC CONTRACTOR PERSONNEL TO PENETRATING RADIATION AS SUMMARIZED FOR CALENDAR YEARS 1959 AND 1960

| Range of Annual Total Exposure (rem) | Number of Employees Exposed in 1960 | Number of Employees Exposed in 1959 |
|---|---|---|
| 0-1 | 77,522 | 71,600 |
| 1-2 | 2,828 | 2,584 |
| 2-3 | 1,405 | 979 |
| 3-4 | 283 | 236 |
| 4-5 | 113 | 113 |
| 5-6 | 24 | 29 |
| 6-7 | 10 | 16 |
| 7-8 | 3 | 11 |
| 8-9 | 2 | 5 |
| 9-10 | 2 | 5 |
| 10-11 | 2 | 1 |
| 11-12 | 0 | 0 |
| 12-13 | 0 | 1 |
| 13-14 | 0 | 0 |
| 14-15 | 0 | 0 |
| 15 plus | 3* | 1 |

*The three individuals who received in excess of 15 rem exposure were involved in the October 4, 1960, and November 8, 1960, accidents listed in Table VI-1.

ETR fission break incident. No personnel injury or significant radiation exposure occurred in any of the incidents. The first and third accidents did, however, involve significant damage to reactor components. The second case involved an "unexpected" criticality excursion.

SPERT III Pressurized-Vessel Failure

A failure occurred in the pressurizer vessel of the Special Power Excursion Reactor Test III (SPERT III) on October 26, 1961. Nonnuclear tests were being conducted at the time of the failure in preparation for a series of experiments to study cold-water accidents. Investigations of the failure have led to the conclusion that it was due to third-stage

creep at elevated temperature. Most of the information for this article has been derived from a report by R. E. Heffner et al.,[4] on the vessel failure. Enough evidence is presented in this report to permit the reader to make an independent analysis of the cause of the failure.

The SPERT III reactor is part of a project operated by Phillips Petroleum Company for the USAEC at the National Reactor Testing Station in Idaho as part of the Commission's reactor safety program. The project is directed toward experimental and theoretical investigations of the kinetic behavior and safety of nuclear reactors.[5,6] SPERT III has been used for conducting reactor behavior and safety studies under conditions typical of pressurized-water and boiling-water reactors. The reactor has heat removal capacity of 60 Mw, but full-power operation is only necessary for limited times. The major components of the reactor, i.e., the vessel, pressurizer, coolant loops, pumps, and heat exchangers, were designed for operation at pressures up to 2500 psig and temperatures up to 650°C. (A more complete description of the facility has been published.[7])

Pressurizer Vessel Failure. The pressurizer vessel was a 2-ft 9-in.-i.d., 16-ft 8-in.-high, all-welded vessel of ASTM A-264, grade 3, 0.04%-maximum-carbon steel with type 304L stainless steel fittings and internal cladding. The backing plate was of ASTM A-212, grade B, firebox-quality carbon steel. The cylindrical vessel walls were 2.95 in. thick, including 1/8 in. of cladding. Normally submerged electrical heaters in the pressurizer controlled the system pressure. Liquid level was measured by a differential-pressure-type instrument with associated temperature (density) compensation. A low level in the pressurizer actuated a makeup pump, while high level operated a blowdown valve.

Prior to the failure of the pressurizer, a nonnuclear cold-water accident series of tests had been initiated. The test under way on October 26 was for determining the effect on the moderator temperature of the startup of a cold, stagnant-coolant loop. The east loop was to be used as the cold loop at a temperature differential of 20°F. Both of the two coolant loops were initially placed in operation at a temperature of 430°F and a system pressure of 2460 psi. At 1745 hr the east loop was isolated, and the pumps in the loop were stopped. The west loop temperature was raised to 445°F and leveled off to start the test. At that time the east loop temperature was 423°F. The east loop was then placed back in operation, and data were recorded for about 20 min. During this time the system pressure rose about 140 psi and the pressurizer level about 8.3 in. The behavior was a result of the increase in the bulk water temperature when the east loop pumps were turned on and was considered to be normal.

The system pressure was returned to 2460 psi and by 1920 hr was stabilized for the next test. At about 2000 hr, smoke was observed to be coming from the pressurizer vicinity. A normal shutdown of the plant was started immediately. The fire department was called to stand by in case it was needed. Shutdown was faster than normal but still within the cooldown limitation of 100°F/hr. By 2350 hr the plant had cooled to 150°F and the pressure had dropped to 400 psig. The pumps were turned off.

Inspection of the pressurizer after shutdown of the plant revealed that the smoke which had been observed came from the fabric covering of the blowdown-line riser, which was almost directly in front of the steam leak. The cloth had charred for about 24 in. along its length. Aside from rips, the pressurizer vessel insulation and covering were normal

in appearance. The insulation in the vicinity of the leak was removed from the vessel on October 27. It was found that a 3/8-in.-wide, 2 1/4-in.-long hole had opened in the central girth-seam weld metal. It was also noted that a 1-in. bolt tying a stabilizing band around the vessel had broken. It appeared to be an old break, however, as no fresh metal could be seen. The fracture was covered with corrosion products. Later close inspection of the band revealed that it had moved in what appeared to be two separate stages. Indications were that the bolt failed in the first or old stage when the band had moved about 1 in.

Following the failure, many tests were performed to obtain information on possible causes. Various system components which might have been affected by unusually high pressure or temperature or which might have affected the validity of the pressure and temperature records were examined. Metallurgical examinations were made of the pressurizer shell, cladding, and welds.

Conclusions and Recommendations. Seven possible mechanisms, listed below, that could have contributed to or caused the failure of the pressurizer were postulated. Each was then evaluated in light of the available information.

1. Deficient design and materials of fabrication.

2. Fatigue failure as a result of thermal cycling in normal operation.

3. Corrosion of vessel walls.

4. Failure from thermal shock.

5. Damage by transient pressure rises during nuclear testing.

6. Failure from pressure in excess of design at or below design temperature.

7. Failure from temperature in excess of design at or near design pressure.

Examination and analysis eliminated each of the above possible mechanisms, except the last. Overheating of the upper half of the pressurizer was indicated as the probable cause of failure by the results of metallurgical examinations, calorimetric tests conducted on strain-gage materials, and limited temperature data on earlier strain measurements.

Examination of the data indicated that temperatures approaching 1000°F were reached in the upper portion of the vessel. The temperature could not have exceeded about 680°F with the heaters submerged in water as they should have been with the pressurizer level in the operating range. Means of uncovering the heaters were investigated. Using temperature distributions actually measured in the vessel, it was shown by calculations that the indicated water level could be as much as 20% higher than the actual level. This could place one or more heaters above the water. Once the top two heaters, which were employed for pressure control, were exposed to steam, control would be accomplished by adding heat to the steam only. The error in level indication would then increase as the water stratified and cooled.

Failure of the vessel could be expected at 2500 psig and 1000°F. This type of failure occurs in three stages and can occur in times as short as 10 hr. It was felt that the first two stages in this case occurred at an earlier date, as evidenced by the failure of the stabilizing band bolt. On October 26, 1961, the third-stage failure occurred.

Several deficiencies in the design and specifications of the pressurizer and plant instrumentation were brought to light by the investigation. These will be given further consideration.

Several design changes are planned for the new pressurizer vessel. The fabricator will be required to furnish tensile specimens of all vessel materials. The specimens will be subjected to treatment identical to that received by the vessel. The fabricator will also be required to furnish calculations and drawings to reflect all design considerations. This should include specification of the maximum heatup and cooldown rates. Consideration is being given to the use of an A-302, A-387, or A-204B steel in the replacement vessel. The method of cladding application used previously (vacuum brazing) was not suitable for SPERT III service and will not be repeated. Inspection ports will be provided in the next vessel.

Changes in the pressurizer instrumentation are also required. Among these, the pressure-control circuit will be altered so that the heaters cannot be energized unless the pressure recorder-controller is in service. Redesign of the heat control system to require that energy be uniformly applied to all heaters is being considered to prevent a recurrence of the unbearable situation in the old system. Relocation of the control heaters at the bottom of the vessel may be adequate to overcome the difficulty. A more reliable liquid-level sensing device will be installed in the new vessel, since the old system was seriously inadequate. Several less important changes and additions to circuits are also planned.

ETR Fission Break Incident

The Engineering Test Reactor (ETR) on December 12, 1961, suffered a fission break[8] that required a manual scram from a power level of 90 Mw. This reactor, whose design level is 175 Mw, is water moderated and beryllium reflected and has enriched-uranium fuel in plate-type elements. It is located at the National Reactor Testing Station at Arco, Idaho, and is operated and maintained for the AEC by the Phillips Petroleum Company. (The previous issue of Nuclear Safety reviewed some aspects of operation of this reactor.[9]) The fission break was caused by melting of portions of fuel elements in the northeast quadrant of the reactor core where there had been inadequate cooling. Primary cooling water flow had been severely restricted by remnants of a Lucite sight box that had lodged on top of the affected elements. The sight box, which had been used to eliminate visual distortion caused by ripples in the water during in-tank maintenance, had inadvertently been left in the tank when the top head was closed.

The Incident and Its Cause. Work had been performed in the reactor continuously on the two days preceding the incident. A minimum of two individuals had worked over the tank on each shift. The sight box may have been used for 18 or more operations by maintenance people in the course of their work. The box was triangular in shape, about 8 in. on a side and 5 in. deep, and was made of 1/2-in.-thick Lucite, a clear acrylic resin. Initially the boxes were made with 1/4-in. plastic bottoms and metal sides, but they often leaked at the plastic-to-metal joints. All-plastic boxes were procured and placed into use in accordance with a decision made at the Shift Supervisor level. For no known reason the thickness of the plastic was increased to 1/2 in.

When the work over the core was completed, the tank was inspected on the December 11 evening shift by a senior reactor engineer, a reactor engineer, and the shift supervisor, in accordance with the Pre-Dome Check List (which emphasizes a visual examination of the top core structure for foreign materials) and was found to be satisfactory. The inspections were made from a work platform which was approximately 1 ft above the tank water level. After removal of the platform and just prior to the head replacement, the water surface was scanned, and the sight box was not observed. The head was therefore replaced and startup procedures were pursued; criticality was achieved at 2255 hr on the 11th.

The reactor power level had been increased to 90 Mw by 0100 hr on December 12. This level was maintained until 0145 hr, when it was reduced to 70 Mw because of a spike on the $N^{16}$ primary water system recorder. This device was not intended to be a fission-break detector (although that was the indication this time) but was basically a secondary means of determining the power level. A fault in the instrument or its cables had caused intermittent spikes on the recorder in the previous three months, and there had been repeated efforts to eliminate the difficulty. After 2 min at the lower power level the operators restored the power to 90 Mw.

Control instrumentation, including the servo error signal, period meter, and count rate meter then started behaving erratically. A radiation alarm was received from an area monitor at the console floor level on the south side of the reactor. The radiation level around certain primary coolant piping was found to be three to five times normal. Several area monitors showed increasing radiation levels. As a result of all this, the reactor was manually scrammed at 0152 hr.

The following information was determined by checking key indicators after the scram:

1. The primary coolant flow decreased gradually a total of 7% from 0115 hr until the scram.

2. The coolant differential pressure across the core remained normal except for two small blips. These were probably due to controller automatic action, which maintained the desired differential as the flow decreased.

3. Nothing abnormal was noted from the bypass demineralizer system, which maintained coolant water quality.

4. All three neutron level monitors behaved erratically when the power was returned to 90 Mw at 0147 hr. This behavior could have been caused by boiling in the core.

5. The stack particulate activity increased sharply at 0132 hr. The stack gaseous activity increased at about 0145 hr. There was no response on the iodine monitor, however, because of a line that plugged during startup. (It was operable at startup time.)

6. A 10-ml sample of primary cooling water read about 1 r/hr at contact shortly after the scram. Gamma spectrographic analysis confirmed "fresh" fission products.

Postincident Inspection and Operation. After reactor shutdown, the top head was removed and the core inspected. There were significant deposits of foreign material lying on the northeast quadrant of the core with two or three fuel elements appearing to be completely covered. The material was first suspected to be resin from the bypass demineralizer but later was found to be the remains of the Lucite sight box.

The core and reflector were cleaned and inspected for damage. Eight fuel elements appeared to have melted plates from which fission products could have been released. Nineteen elements were replaced. All the remaining elements were inspected in place. Nine control rod assemblies were removed to the canal, disassembled, and inspected. The remaining assemblies were inspected in place. There was no indication of assembly damage. The core grid structure also appeared to be undamaged. After replacement of the head, water was circulated for several hours, and the core was inspected again. After removing a few additional foreign particles the vessel was closed and startup commenced.

The reactor was critical by 0505 hr on December 14 and was at full power by 0835 hr on December 15. The activity level in the reactor coolant started increasing on the 14th and appeared to stabilize on the 16th at about twice the normal level. There was, however, an over-all trend upward for the next two days. The stack gas fission level stabilized at about six to seven times normal. These high activity levels were believed to be due to small quantities of fuel alloy from the overheating of the 12th that were still in the coolant loop and which were impossible to locate and remove.

The reactor was shut down at 1531 hr on December 18 for additional cleaning. Considerable amounts of lint from mops and brushes, used in cleaning the reactor tank a few days earlier, were found on the core. Accessible foreign material was again removed from the reactor. No person received in excess of the administrative exposure limits of 300 mr/week or 600 mr/week skin dose in this or the previous cleanup operations.

Through December 27, about 90 operators and health physics personnel had a total gamma exposure of 12 r, or an average of 133 mr each.

Following the cleanup, operation was resumed on December 23. By the next day, the reactor was at full power. Discharge of radioactive krypton and xenon and particulate activity at the ETR stack is about 20 times higher than before the incident. This represents about 3 curies/Mwd of the particulates. (This excludes emissions attributable to experimental activities.) These emission rates, while significant, are not considered to represent a health hazard on or off the site.

Because of the contamination of the coolant system with fuel element particles and fission-product activity, it is anticipated that downtime will increase at each future shutdown to allow for additional cleanup.

Conclusions and Recommendations. The following was proposed to prevent a recurrence of this type of incident:

1. Operating procedures should be reviewed to assure that they provide for an accounting of all items used in reactor remote operations before the reactor tank is closed. Sight boxes and some other classes of object should be numbered so that "sponge counts" can be kept.

2. Side plates of sight boxes should be made of colored plastic or other material to make them more readily visible when submerged. The boxes should be made of bouyant plastic or with buoyancy chambers. Only 1/4-in. or thinner plastic should be used.

3. New tools or modification of tools (i.e., sight boxes) to be used in the reactor should receive approval by plant management.

4. Use of the type of cotton mop and bristle brush utilized for earlier reactor cleaning should be eliminated. Acceptable substitute cleaning tools or methods should be found.

5. The iodine monitor should have an alarm which would alert the operators to lack of flow through the detector. Installation of an additional or backup monitor should be considered.

6. A policy should be formulated regarding consultation with the ETR Safeguards Committee about reactor operations following recovery from incidents involving core components or experiment damage.

ORNL Criticality Excursion

An unscheduled criticality excursion occurred on November 10, 1961, during the course of some experiments at the Oak Ridge Critical Experiments Laboratory. Personnel exposures were insignificant and there was no property damage. Loss of operating time amounted to less than 1.5 hr. Although occurrences of this kind must not be taken lightly, they are not considered entirely unexpected in light of the many short-range experiments performed in the critical laboratory, and the facility is designed to handle them. The significance of this criticality was due solely to the fact that when it occurred it was unexpected. Nuclear Safety discusses the incident because of its general interest and because of the potential hazard associated with such operations. Most of the information here was derived from a report of the incident by Callihan.[10]

Description of Facility and Equipment. Three critical experimental laboratories or assembly rooms comprise the facility, which is located in a building (designated 9213) about 1/2 mile south of the Oak Ridge Y-12 plant.[11] The topographical location of the site is favorable, since it

is separated from the main plant area by a strip of high ground called Chestnut Ridge. Controlled ventilation systems prevent air from the assembly rooms from reaching other parts of the building, which comprises offices, shops, counting rooms, etc. Adequate shielding (5 ft of concrete between assembly rooms and control areas) is provided to prevent personnel exposure.

The fissile material involved in the unscheduled incident was approximately 93% $U^{235}$, which was neutron-reflected and -moderated with paraffin. The materials contained a mass of about 75 kg of uranium and were divided into two parts.[12] One part was moved vertically upward by a hydraulic piston acting through a magnetic coupling to contact the other part. Air pressure applied to the hydraulic system controlled the initial speed of approach of the parts. The initial speed was reduced by a switch operated by the moving member at a preset separation of the parts. Manual adjustment at the equipment could change the point at which the step change occurred. A manually controlled valve in the hydraulic system also could be used to adjust the speed. There was, however, no means for reducing the speed of approach from the remote control point.

The initial speed during this experiment was 16 in./min. When the parts were separated by 1.94 in., the step change to a slower speed occurred. The location of the moving part of the apparatus during fast motion was not followed properly because of underdesign of the position-indicating device. Limit lights did correctly indicate closure of the parts. Rates of separation for normal shutdown were greater than the closure rate. Even greater rates were used for emergency shutdown, these being 12 in./sec for the first downward inch, 60 in./sec for the next 9 in., and 10 in./sec

for the rest of the travel. About 1 msec was required for collapse of the supporting magnet current after receipt of the scintillation detector signal. An additional time of approximately 50 msec was required to disengage the magnet.

The Incident. The reactivity of the experiment was incrementally changed by adding more uranium or more moderator or reflector. The addition of a large reflector was the final alteration to the assembly which had previously been subcritical under all arrangements.

Unexpectedly the assembly became critical and emitted the usual "blue glow." When the assembly parts had been separated by 2.7 in., it had been in a delayed critical condition, as indicated by tests after the incident. The sensitivity of the system at this point was $8.6/in., while the rate of approach was still at the higher rate and corresponded to an activity increase of $2.3/sec. There is some doubt that the assembly reached prompt critical. A 0.12-in. displacement of the lower assembly section in 435 msec was required to go from delayed to prompt critical. The instrument response time is believed to be shorter than this. The reactivity decreased at a rate of about $100/sec in the first inch of drop of the lower section. It had decreased by $5 only 50 msec after the disengagement of the magnet. (This is about 100 msec after detection of a high radiation level.)

The energy-release yield was estimated to be $10^5$ to $10^{16}$ fissions on the basis of analysis of a threshold detector. No direct evaluation was made because destructive sampling of the assembly was impractical. It also had a complex undetermined power distribution.

Assuming no losses, the average temperature of the uranium would have increased by 35°C from the energy of $10^{16}$ fissions. It could be seen that the surface temperature of the uranium had remained below about 70°C, since the paraffin in contact with the metal showed no indication of having melted. The shiny appearance of the metal and absence of alpha-particle contamination in the area indicated that no uranium oxidation had occurred.

Estimates of the airborne fission-product contamination 30 min after the release indicated that between $2 \times 10^{-7}$ and $1 \times 10^{-6}$ $\mu c/cm^3$ (1.5 mc, maximum, total contamination) evolved from the uncoated uranium. The next convenient observation 15 hr later indicated no air contamination.

It was noted that the delayed gamma-radiation field 20 ft from the source decreased three orders of magnitude during the interval between 2 min and 1 hr after activation. The field at the surface of the uranium was as high as 10 r/hr at 19 hr after the event. If personnel exposures resulted directly from the burst, they were less than 5 mrem. Only insignificant exposures (<100 mrem) were incurred by health physicists in their subsequent surveys of the area.

It is interesting to note that in two periods following this occurrence, pulses of radiation 10 to 100 times greater than those produced in this case were routinely induced in the same laboratory while prooftesting the Health Physics Research Reactor. The significance of this case was that the event was unexpected.

Conclusion and Recommendations. It was concluded that the incident was the result of errors in judgment by those performing the experiment. The rate of approach of the two parts was too fast. The operational

radiation-detection instrumentation was too sluggish, perhaps due to the thick neutron reflector around the assembly.

Prior to reuse of the experimental apparatus certain corrective measures will be taken. These include installation of an improved position indicator and adoption of a procedure that requires measurement of closure speeds as a part of the pre-experiment equipment checkout. (J. R. Buchanan)

## CHANGES IN LICENSING REGULATIONS

AEC rules and regulations, as found in Title 10, Code of Federal Regulations, Chapter 1, are frequently amended. Among these now being considered for change are Part 9, Public Records; Part 20, Standards for Protection Against Radiation; Part 30, Licensing of Byproduct Material; Part 40, Licensing of Source Material; Part 50, Licensing Production and Utilization Facilities; Part 55, Operator's Licenses; Part 72, Protection Against Accidental Criticality and Radiation Exposure in Shipment of Fuel Elements; Part 100, Site Criteria Guide; Part 140, Financial Protection Requirements and Indemnity Agreements; Part 150, Transfer of Regulatory Power. The various amendments are discussed below:

### Part 9, Public Records

Comments on an amendment to Part 9 proposed on October 9 continue to be received by the AEC.[13,14] The amendment which would require that all reports of nuclear incidents filed by AEC licensees be included in the public records of AEC was criticized in all recent comments, as it had been in those received in the previous report period.[15] There is a general fear that the amendment may focus adverse public attention on the nuclear industry that will not be consistent with its fine safety record.

In another and separate proposition, the AEC published, in the Federal Register of January 26, 1962, a proposal which provides that notices to licensees of alleged violations and correspondence between AEC and licensees concerning alleged violations will not be made public until after the licensee has responded to the allegations and AEC has acknowledged the response. Comments were invited.[16]

Part 20, Standards for Protection Against Radiation

An amendment to provide additional ways to determine a permissible concentration limit for a mixture of radiosotopes contained in effluents released into air or water was published in the Federal Register of November 25, 1961, and went into effect 30 days later.[17] No suggested changes or objections to the amendment were received after its proposal on August 9.[15]

Part 30, Licensing of Byproduct Material

Another favorable comment was received by AEC on its September 12, 1961, proposal that would authorize the possession and use in aircraft of luminous safety devices not containing more than 4 curies of tritium each in airplane.[15,18] The commentors felt, however, that there should be modifications in testing procedures used on the devices so as to lower the associated costs. Accordingly, on November 27, 1961, the AEC was asked to consider amending its regulations on equipment leak testing requirements. It was alleged that

> "persons qualified to be only general licensees...(users with minimum qualification)...are obtaining specific licenses... (which usually require that the licensee have trained personnel)...allowing (1) a 3-year leaktest interval on certain gauges, and (2) untrained personnel to perform the actual wipe testing in the area of the source. It is proposed that 10 CFR 30, paragraph 30.21 (c), be amended to allow general licensees a variable time interval for leaktesting certain gauges, and that only properly trained persons be allowed to perform leak tests."[19]

Part 40, Licensing of Source Material

The AEC on November 22, 1961, published and put into effect amendments to Part 40 that would exempt from licensing requirements uranium in the form of aircraft counterweights and also exempt certain shipping containers made of or incorporating uranium as a shielding material.[17] No

objections to the amendments had been received after their proposal on August 9.[15]

Part 50, Licensing of Production and Utilization Facilities

Comments continue to be received on the "Change Without Specific Authorization" procedure rules published in the Federal Register of April 8, 1961.[13,14,20,21] Most of the comments have come from research institutes that objected to portions of paragraph 50.59, which would require "the licensee [to] promptly file with the Commission a report of each change, test, or experiment carried out pursuant to the authorization granted in this paragraph." It was felt that research and development personnel could become overwhelmed with paper work and the result would greatly hinder the ability to do research.

Part 55, Operator Licenses

Proposed amendments to the operator licensing regulations were published in the Federal Register in October.[15] Since that time many comments have been received by the AEC from industry and university groups. There was wide feeling regarding the need for changes in or additions to certain portions of the amendments. The following comments were predominant.[13,14,18,22,23]

1. An operator's or supervisor's background should determine his qualifications rather than a stringent rule that he have two years of prior operating experience. It was pointed out that a person may have had several years of operating experience but, just prior to application for licensing, may have been working on design or construction aspects of a reactor project.

2. The coverage of licensing examinations should be outlined in some manner so that sensible training programs can be carried out. It was felt that examinations could be too theoretical with too high an educational level being required of the operator.

3. The real need for licensing supervisory operators was questioned. It was suggested that there should be a specification defining a supervisor. A labor organization felt that the AEC should relieve itself of the task of having to indicate job descriptions by designating licenses as to class (such as "Class I" and "Class II") instead of operator and supervisor.

4. The three-year waiting period for a third re-examination was felt by some to be too harsh. A one-year wait, as for the previous re-examination, was suggested as being sufficient.

5. There were suggestions that "controls" be defined to be more limited in scope. It was felt that too broad an application could cause hardship in operating auxiliary and secondary equipment. Operation under the directive of a licensed operator was felt by some to be sufficient for certain systems.

6. More definitive information on what constitutes a comparable nuclear facility was felt to be needed by some.

Part 72. Protection Against Accidental Criticality and Radiation Exposure in Shipment of Fuel Elements

Comments continue to be received[15] on Part 72 of the proposed regulations[24] printed in the Federal Register of September 23, 1961. Many modifications have been suggested[13,14,18,21,23,25] that, in general, deal with the need for setting up performance rules rather than standards for design and construction. Therefore the AEC made available, in December 1961,

Draft Technical Standards for Transport of Radioactive Materials[26] which were prepared by the Inter-Agency Committee on Transportation of Radioactive Materials. The Inter-Agency Committee is composed of representatives of the Interstate Commerce Commission, the Federal Aviation Agency, the Post Office Department, the Coast Guard, the Bureau of Explosives (Association of American Railroads), and the AEC. The proposed standards contain many of the principles proposed in 1960 by the International Atomic Energy Agency for safe transport of radioactive materials.

When the standards have been developed in final form, they will serve as a basis for revision of existing federal transportation regulations. Several comments have been received by the AEC on the standards, and, in general, there is agreement with the objectives and a feeling that uniformity is essential for movement of such materials across interstate or international boundaries. Some reviewers have objected, however, to the complexity of the requirements and have required simplifications.[14,23]

The AEC now requires compliance with the shipping regulations of the Interstate Commerce Commission[27] for all off-site irradiated fuel shipments. The ICC regulations cover radioactive material shipments, escorted or unescorted, and do not deal specifically with irradiated uranium shipments. Recognition of the increasing frequency of shipment of high irradiated and enriched uranium fuel elements prompted the proposal now being considered for issuance as Part 72.

When the comments on proposed Part 72 have been correlated and a final draft has been prepared, regulations will be issued governing the loading of fuel elements into a transfer cask, transportation of the cask, and unloading of the fuel elements from the cask. A previously published

edition of these proposed regulations was reviewed in Nuclear Safety.[28] The latest draft of these proposals[24] contains additional material relating particularly to the design of the cask and procedures for packaging and shipping fuel elements.

The proposed regulations[24] require Commission approval of the cask and procedures used for shipping solid, irradiated fuel elements containing more than 2000 curies outside the confines of the licensee's plant. Shipment by air and shipment of nuclear fuel in other than solid form are not authorized by these regulations. The regulations state that each application for a license to be issued under Part 30 and Part 70 of Title 10 of the Code of Federal Regulations must include (1) an evaluation of the ability of the cask and the handling procedures to protect the public against radiation and the release of radioactive contamination, (2) information concerning the design and construction of the cask, and (3) procedures for loading, transporting, and unloading the fuel elements. The hazards evaluation of the cask must contain an analysis of the following features: structural integrity, resistance to fire, shielding, protection against the occurrence of criticality, provisions for the removal of radioactive decay, and the adequacy of cask-handling equipment.

Part 100, Site Criteria Guide

Revised criteria guides for use in the selection of reactor sites are expected to be issued soon by the AEC. The proposed criteria were published early in 1961 and elicited comments from many sources.[29] The guides are being revised to make it plain that the criteria were not meant to be unyielding or indifferent to advances in technology.

Part 140, Financial Protection Requirements and Indemnity Agreements

A proposed amendment to Part 140 was published in the Federal Register of November 22. It took into account enacted legislation (effective September 6, 1961) which amended the Atomic Energy Act of 1954 to eliminate indemnity coverage of public liability for damage to "on site" property.[17]

Part 150, Transfer of Regulatory Power

The AEC on February 2, 1962, announced approval of an agreement to transfer regulatory power to the Commonwealth of Kentucky.[30] The agreement which became effective on March 26, 1962, is the first to be reached with any state. The AEC's present regulatory authority over radioisotopes, the source materials uranium and thorium, and small quantities of fissionable material is involved in the transfer.[31]

The Commission will transfer authority to regulate land burial of low-level waste and authority to regulate the transfer by the manufacturer to the user of industrial-type devices and products (such as thickness gages, etc.). The Commission will, however, retain jurisdiction over the transfer from the manufacturer to the general public of consumer-type products, the use of which could not be controlled after they reached the public. The original proposal on these points was published for comment in the Federal Register of September 29.[15] A total of 51 organizations and individuals commented to AEC on the proposal. The most frequently expressed remark had to do with the fear that a potential duality of jurisdiction between the state and the federal government would be established. There was also a general feeling that responsibilities should

be delegated to the states when they have the capabilities of assuming them.[13,18,21,22,25,32]

In mid-December, California filed a proposed regulator program with AEC for a similar transfer of authority.[33]

## ACTION ON REACTOR PROJECTS BY LICENSING AND REGULATING BODIES

The AEC is required by the Atomic Energy Act of 1954 to protect the health and safety of the public from undue hazards relative to the operation of nuclear facilities. Regulations which are followed to ensure that this responsibility is fulfilled while furthering the simultaneous responsibility for developing the use of nuclear energy are published in the Code of Federal Regulations. Title 10 of the Code requires the AEC to make certain specific findings regarding the safety of the public before issuing either a construction permit or an operating license for a facility. It also requires authorization for changes in facility equipment or operation that contain an element of hazard not previously reviewed or approved. The license application record of various power reactors is reported in Table VI-3.

The ACRS on November 1, 1961, suggested to AEC Chairman Seaborg that clean, critical experiments need not be referred to it for review. An example of this would be the Hallam "dry critical" experiment reviewed by the ACRS in October (see Hallam below). In discharging its responsibilities under the provisions of the Atomic Energy Act, the ACRS felt that "protection of the health and safety of the public should receive the primary attention."

> "Analyses of the consequences of possible reactor malfunctions have shown that the most serious widespread effects are the result of dispersal of fission products. In the case of clean critical experiments in which no significant fission product burden is present, the health and safety of the public is usually not placed in jeopardy. The AEC staff is qualified to judge the adequacy of the precautions taken in critical facilities and has been taking the responsibility for them without formal reference to this Committee. We [ACRS] suggest that this procedure be

Table VI-3. Calendar of Legal Steps in Licensing U. S. Power Reactors

(to be provided separately)

extended to clean critical experiments carried out in facilities which will ultimately be used as power or test reactors."[34]

Big Rock Point Reactor (Docket 50-155)

On December 1, 1961, Consumers Power Company, Jackson, Michigan, sent the Final Summary Hazards Report of the Big Rock Point boiling-water reactor to the AEC, together with a request for an 18-month provisional operating license covering operation of the reactor up to a maximum thermal power level of 157 Mw and conduct of the first phase of a research and development program.[35] Earlier (on November 10) Consumers Power had issued a proposed schedule of activities that would lead to issuance of the provisional license.[36] They expected to file the proposed reactor technical specifications on January 10, 1962, meet with the ACRS in March, participate in a public hearing on May 15, and receive the license by July 15. This would enable them to initiate reactor fuel loading on September 1. The company expects at a later time to seek a license to allow operation up to its full design capacity of 240 Mw(t) and to carry out additional phases of its research and development program.

Dresden Reactor (Docket 50-10)

Commonwealth Edison of Chicago received authorization on November 9, 1961, from the AEC to modify the liquid poison system on its 630-Mw(t) Dresden boiling-water reactor.[37] Edison plans to increase the size of the poison storage tank and decrease the concentration of the solution without changing the amount of boron in solution. This will enable them to eliminate handling difficulties associated with the higher concentration. Edison[38] originally requested authorization for the change in

September 1961 and then submitted additional information, per AEC request, on October 23, 1961.[39]

On October 25, Edison submitted to the AEC a report on the anticipated neutron dose which the Dresden reactor vessel will receive in its lifetime. On October 26 they reported to the AEC on the re-examination of the core support structure. The report stated that the structure was unchanged since the first examination conducted in March and April 1961. The cracks found then had neither lengthened nor widened. A total of 2884 hr of operation had been accumulated since the first examination.[39]

Edison received notification in January that its request for suspended credit on its insurance indemnity fee covering the reactor shutdown period from November 15, 1960, to June 21, 1961, had been consummated. They received a premium credit of $107,580.50 which was 75% of the full premium applicable to the shutdown period.[40]

Enrico Fermi Atomic Power Plant (Docket F-16)

The Power Reactor Development Company on November 29, 1961, was allowed to extend the latest completion data of the 200-Mw(t) Enrico Fermi fast breeder reactor from December 15, 1961, to September 1, 1962.[41] The later data allows the needed flexibility in the reactor program. When PRDC requested the extension on November 10, they reported the following:[42]

1. The graphite shielding blocks around the reactor vessel did not withstand the 1000°F preoperational test undertaken in early summer and will have to be replaced.

2. The offset fuel-handling mechanism became jammed as a result of inadvertent misoperation while checking dummy subassemblies for binding. Several of the assemblies did bind, but this may have been due to the

dummies having been manufactured to less rigid tolerances than those specified for the actual fuel subassemblies. The offset handling mechanism will be removed for examination and any needed repairs.

3. Nonnuclear tests of the fuel and repair facilities are under way.

4. Further work is under way to determine the cause and extent of the stress-corrosion cracking found in a section of the No. 2 steam generator. Retubing of the generator bundle will be done as necessary.

5. Plant construction is essentially complete except for items which must await satisfactory conclusion of the nonnuclear test program.

Indian Point Reactor (Docket 50-3)

Hearings were held by the AEC on December 7, 1961, and January 3, 1962, to consider granting a provisional operating license for the 585-Mw(t) pressurized-water Indian Point reactor.[43,44] Issuance of a license was recommended by the AEC staff, but they differed with Consolidated Edison Company as to the form. In mid-January Consolidated Edison's counsel, LeBoeuf, Lamb, and Leiby, submitted a brief to AEC supporting the form of license desired by them. Four points of difference between it and the Regulatory Staff recommendation were described. They related to:

> "1. The propriety of including material license in and with the facility license.
>
> "2. The proper definition of the physical complex or facility, the operation of which is to be regulated.
>
> "3. The proper definition of the term 'Hazards Summary Report' as the chief referent for determining the scope of the 'change' procedure.
>
> "4. The extent of the mandatory hearing requirement of Sections 189a of Atomic Energy Act."[45]

The AEC has the brief under consideration.

Prior to the hearings, the ACRS in its September and October meetings considered previously unresolved questions concerning design and operation of the reactor. It was the conclusion of the ACRS that the reactor could "be operated without undue hazard to the health and safety of the public."[46]

On December 4, the AEC granted an amendment to Consolidated Edison's construction permit that advances the facility completion date December 15, 1961, to February 15, 1962. The extension was justified because of the extensive and time-consuming preoperational testing of components of the facility.[42]

Hallam (Docket PP-3)

In mid-January 1962 the 240-Mw(t) Hallam reactor achieved self-sustained fission without coolant. This test was part of a "dry critical" experiment to verify physics calculations on the reactor core.[47] A six-month provisional operating authorization to permit these tests was granted by AEC on January 2 following a public hearing on November 29, 1961.[48,49] After reviewing the favorable testimony by all parties, the hearing examiner on December 28 had recommended the authorization to the extent required for the dry zero-power experiments only.[50] The ACRS had reviewed the proposed critical experiments in their October meeting and concluded that they could "be conducted without undue hazard to the health and safety of the public."[51]

Nuclear Merchant Ship N.S. Savannah

Criticality of the 69-Mw(t) pressurized-water reactor of the N.S. Savannah was achieved at Camden, New Jersey, on December 21, 1961. Testing will continue at Camden until the reactor power is increased to a maximum of 7 Mw. The ship will then be transferred to Yorktown, Virginia, under auxiliary power, for tests at higher nuclear power.[52]

Peach Bottom Reactor (Docket 50-171)

At its October 1961 meeting, the ACRS reviewed the 117-Mw(t) high-temperature gas-cooled Peach Bottom reactor. They heard of the extensive research and development program being carried on by the General Atomic Division of General Dynamics to resolve all the health and safety problems of the reactor. The main problems and means of solution are the following:[51]

> 1. "While the hydraulic control rod system remains basically the same, the added rod separation detection system, electrically driven emergency shutdown rods, fusible-link poison rods, and installation of a finger-type holding lock on control rods provide a satisfactory control and backup scheme... .
>
> 2. "The questions raised concerning the inherent shutdown characteristics appear to have been resolved by changes in thorium concentration and addition of rhodium to the core, and recalculation and measurements on the Doppler contribution... .
>
> 3. "In order to prevent reaction between core graphite and moisture, provision has been made for rapid moisture detection, loop isolation, and scramming the reactor if excessive moisture is detected in the primary system. Further protection of the graphite is provided by maintaining the oxygen content of the containment vessel at a level below 5%. An emergency cooling system has been provided around the reactor cavity to remove decay heat after shutdown in the event of loss of coolant circulation... .
>
> 4. "Considerable information has been developed on barriers against fission product release. Pyrolytic coating of fuel particles, the use of an impervious graphite sleeve around the fuel compacts, internal fission product traps on fuel elements, and external fission product traps are proposed as the means of controlling fission product concentration in the coolant. The current results of the fission product research program appear to be favorable. However, should later results indicate that a reliable system can not be obtained by the present approach, alternate methods appear to be available to insure that the fission product concentration in the helium coolant will be kept low."

Since these problems give reasonable assurance of satisfactory resolution, the ACRS concluded "that the proposed reactor can be constructed at

The AEC, on December 28, was advised by Saxton Nuclear Experimental Corporation that it would be necessary to vary some of the operating conditions outside those specified in the Technical Specifications (and Final Safeguards Report) in order to conduct a series of core-parameter measurements. Since they did not believe it was the intent to have the steady-state conditions of the specifications restrict the special core tests, the tests will be conducted without changing the specifications.[55]

Vallecitos Boiling-Water Reactor (Docket 50-16)

The General Electric Company, on November 2, 1961, was granted permission to operate its 50-Mw(t) VBWR with increased outlet steam temperature, increased fuel element cladding surface temperature, and with intentionally defective fuel elements as a Superheat Fuel Demonstration Test Loop.[37]

In October and November, GE submitted to AEC seven reports of proposed changes, tests, or experiments that were not deemed to be unreviewed safety questions as defined in the VBWR license.[36]

On December 20, 1961, GE was authorized to install and operate the Expanded Superheat Advanced Demonstration Experiment (E-SADE) loop facility in the VBWR. This required a partial waiver of the technical specification limitations which state that only one new type of fuel element may be introduced into an in-core test. The waiver was made only for this special test.[56]

Westinghouse Test Reactor (Docket 50-22)

On December 4, the Westinghouse Electric Corporation submitted a report to the AEC on an alternate method of pump operation for test loop

No. 4 of its 60-Mw(t) test reactor. Westinghouse felt that this did not involve hazards greater or differing from those in the final safety report. The AEC, on December 5, 1961, notified Westinghouse that they were in agreement with their conclusions.[34,55]

Yankee (Docket 50-29)

The Yankee Atomic Electric Company on December 8, 1961, requested exemption for their 584-Mw(t) pressurized-water Yankee Reactor from rules in Part 20 of the Code of Federal Regulations which require the equipping of high-radiation areas with visible and audible alarms. They cite their basic policy, which follows, as making installation of an automatic device unnecessary.

> "1. Control of all entry to the plant through attended gatehouse.
>
> "2. Visitor accompaniment in all areas of the plant.
>
> "3. Restricted visitor access to the potentially contaminated area.
>
> "4. Employees educated in radiation protection.
>
> "5. Routine radiation survey and area pre-entry survey.
>
> "6. Use of barriers and posting of each high radiation area.
>
> "7. Issuance of a Radiation Work Permit for entry into a high radiation area."[22]

The AEC has the request under consideration.

During December 1961 Yankee submitted to the AEC six additional reports of proposed changes, tests, or experiments.[55,56] On January 3, 1962, they sought guidance from the AEC on the advisability of using boron in the main reactor coolant with their Core III, which is under design. They requested a Licensing and Regulating staff review and a meeting with the

ACRS so that they might settle on an acceptable design and meet their delivery schedule on the new core.[57]

On January 10, 1962, Yankee advised AEC that it planned to inspect the control-rod drive-shaft-coupling fingers during the first reactor refueling. They requested authorization to replace the couplings and drive shafts with an alternate design if they deem it necessary.[57]

(J. R. Buchanan)

# BOOK REVIEW: ATOMIC ENERGY WASTES

ATOMIC ENERGY WASTE. Its Nature, Use and Disposal. E. Glueckauf, Editor; Interscience Publishers, Inc., New York; Butterworth & Co., Ltd., London; 1961, 420 pp., 109 illustrations, $14.00.

Dr. Glueckauf, a research chemist on the senior scientific staff of the United Kingdom's Atomic Energy Research Establishment at Harwell, has compiled and edited this comprehensive report on research into the utilization and management of atomic energy wastes with contributions from nineteen of his colleagues in the U.K.A.E.A. Many of the contributors are known authorities in particular areas related to the general subject and all have been active in Britain's development of peaceful uses of atomic energy. Some of them are senior staff members at the Authority's establishments at Harwell and Windscale and others are consultants to, or responsible officers in, the Authority itself. Their specialties include research and application in physics, chemistry, chemical engineering, meteorology, biology, medicine, agriculture and health physics. Dr. Glueckauf is recognized for his contributions to present-day theory for understanding and predicting fission-product behavior in waste treatment and waste-processing systems.

Organized in the form of a symposium volume, the book is in six parts. About two-thirds of the text is devoted to a description of the origin and nature of fission-product wastes and wastes containing alpha emitters (Part 1), and the methods of managing low-level liquid effluents, stack effluents and solid residues, of storing high-level liquid effluents from reprocessing plants, and of converting high-level liquid effluents to solid, disposable products (Part 4). Since the recovery of individual fission products from certain waste streams affects the character of such wastes,

these separations processes are also described in this part. Part 2 deals with the physical, chemical, and biological effects of radiations and the movement of radioactive substances through aquatic and terrestrial food chains. Also included is a brief discussion of radiation protection problems in occupational and public health. Other parts deal with legislation on the transport and disposal of radioactive materials in Britain and elsewhere (Part 3), and with the utilization of fission products as sources of radiation in industry, medicine, and agriculture (Parts 5 and 6).

The book summarizes knowledge in these subject areas prior to 1958. Few of the more than four hundred references cited were published later and many were published in the early fifties. Selections of published work outside Britain rest mainly on the proceedings of the 1st and 2nd Geneva conferences and on numerous U.S.A.E.C. reports of that period. This is not a particular criticism, but rather an indication of the "state of the art" in relation to time and space in a rapidly advancing technology. It is an excellent condensation of theoretical and practical knowledge in waste technology to that date, and its place in any good technical library is assured.

This is the third book on radioactive-waste disposal to have been written by staff members or former staff members of the U.K.A.E.A. Several of the contributors to this volume prepared like contributions to one or the other of the first two books.[58,59] This detracts somewhat from the value and importance of these publications since they cover much of the same ground. Discussion of the problems involved and the techniques employed in waste management are essentially the same in all three, which suggests that those interested in the engineering aspects of waste

treatment and waste disposal may make a choice between them. Where ATOMIC ENERGY WASTE differs is in its pointed emphasis and detailed consideration of the fundamental properties and chemical behavior of fission products in relation to their recovery and use in industry, medicine, and research. As the editor puts it in the introduction, " ... it is inconceivable that such valuable sources of radiation energy should be allowed to remain a waste."

That atomic energy wastes contain byproducts too valuable to discard is an unrealized promise often used to suggest that more research into the use of radioactive fission products will eventually simplify the waste-disposal problem. In the opinion of the reviewer, the widespread use of fission products in thousands of applications will only complicate the problem. The fission products likely to be most useful as sources of radiation are the same ones that create difficulties in waste treatment and waste disposal, viz., those having relatively long half-lives. Separating fractions of these fission products from waste streams, and packaging them for distribution and use elsewhere, merely transfers responsibility to others and postpones the date of final disposal. Eventually they will need to be permanently disposed of in some safe place in the environment. (E. G. Struxness)

# SAFEGUARDS REPORTS AND SELECTED READING

The recently issued safeguards reports and selected literature pertaining to hazards of reactors are listed below for reference. Because of the similarity of many reactors (in particular, research reactors), this list is not intended to be all inclusive.

## Safeguards Reports

1. Design and Hazards Summary Report Boiling Reactor Experiment V (BORAX V), USAEC Report ANL-6302, Argonne National Laboratory, May 1961.

2. G. P. Kerr, ed., Summary Hazards Report for Zero Power Tests with the R-1 Mockup Reactor, USAEC Report APEX-110, General Electric Company, October 1952, declassified June 16, 1961.

3. G. P. Kerr, ed., Summary Hazards Report for Critical Experiments with TRA-2 Reactor, USAEC Report APEX-155, General Electric Company, April 2, 1954, declassified June 9, 1961.

4. C. C. Gamertsfelder, ed., HTRE Hazards Report, USAEC Report APEX-180, General Electric Company, December 15, 1954, declassified June 16, 1961.

5. J. N. Renaker, ed., SMR Summary Hazards Report, USAEC Report APEX-200, General Electric Company, June 15, 1955, declassified June 16, 1961.

6. J. A. Hoefer, ed., Summary Hazards Report for Critical Experiments with RSM Reactor, USAEC Report APEX-254, General Electric Company, July 10, 1956, declassified June 9, 1961.

7. C. C. Gamertsfelder, et al., D 102-A Hazards Report, USAEC Report APEX-344, General Electric Company, December 12, 1957, declassified June 16, 1961.

8. C. C. Gamertsfelder et al., Addendum to D102A Hazards Report, USAEC Report APEX-482, General Electric Company, April 15, 1959, declassified June 16, 1961.

9. J. A. Hoefer, Summary Hazards Report for Critical Experiments with the HOTCE Reactor, USAEC Report APEX-345, General Electric Company, November 20, 1957 (declassified by authority of TID 12-62).

10. R. B. Cochran et al., Hazards Summary Report for the Nuclear Science Center, FZK-110 (Suppl. 1), Nuclear Science Center (Texas), n.d.

11. Description and Hazard Analysis of Japan Research Reactor (JRR-2), JAERI-6003, Japan Atomic Energy Research Institute, Tokyo, 1961.

12. D. F. Spencer, Thermal and Criticality Analysis of the Plasma Core Reactor, JPL-TR-32-189, Jet Propulsion Laboratory, January 1, 1962.

13. W. Haass, PM-3A Nuclear Power Plant Hazards Summary Report Plant Design, MND-M3A-2496 (Vol. 1), Glenn L. Martin Company, March 1961.

14. J. M. Duncan et al., Final Safeguards Report for the Northrop Reactor (NOR), NB-61-349, Northrop Corporation, December 1, 1961.

15. Final Safeguards Report for Armed Forces Radiobiology Research Institute, NP-10714, National Naval Medical Center, September 1961.

16. Hazards Summary Report for a TRIGA-I Nuclear Reactor, NP-10964, Texas University, 1961.

17. Final Hazards Summary Report for Big Rock Point Plant. Volume I. Plant Technical Description and Safeguards Evaluation; Volume II.

Part I. Engineering Drawings; Part II. Topical Reports, NP-11153 (Vols. I and II), Consumers Power Company, Jackson, Mich., November 14, 1961.

18. N. J. Becar et al., Critical Experiment Tank (CET) Reactor Hazards Summary, USAEC Report TID-12719, March 31, 1961.

Bibliographies and Miscellaneous Reports

19. E. E. Campbell and H. M. Miller, Comps., Chemical Detectors. A Bibliography for the Industrial Hygienist with Abstracts and Annotations, USAEC Report LAMS-2378 (Vol. I), Los Alamos Scientific Laboratory, October 1961.

20. M. J. Oestmann, Radiation Dosimetry: An Annotated Bibliography, REIC-Memo-23, Battelle Memorial Institute, September 15, 1961.

21. A. W. Klement, Jr., An Atomic Radiation Bibliography. A List of Reports Submitted to the United Nations Scientific Committee on the Effects of Atomic Radiation, USAEC Report TID-3909, March 1, 1961.

22. E. Bock, comp., Insurance Questions Including Third Party Risk and Recovery of Damages in the Application of Atomic Energy. A Selective Bibliography, Swedish Report VDIT-25, 1960.

23. E. Bock, Tube and Fuel Element Rupture, A Bibliography of Selected Literature, Swedish Report VDIT-28, 1960.

24. The Shippingport Pressurized Water Reactor Project Catalog of Document Abstracts, USAEC Report WAPD-PWR-1606 (Rev.), Westinghouse Electric Corporation, December 1961.

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