MASTER COPY

OAK RIDGE NATIONAL LABORATORY

operated by

UNION CARBIDE CORPORATION

for the

U.S. ATOMIC ENERGY COMMISSION

ORNL-TM-110

75541

14

DRAFT

TECHNICAL PROGRESS REVIEW

NUCLEAR SAFETY

Volume 3, No. 3

UNITED STATES
ATOMIC ENERGY COMMISSION
OAK RIDGE, TENNESSEE

This document has been reviewed and it has been determined that it does not contain Restricted Data or other classified defense information as defined in the Atomic Energy Act of 1954 H. F. Carroll/ae
Chief, Declassification Branch
2/8/62

NOTICE

This document contains information of a preliminary nature and was prepared primarily for internal use at the Oak Ridge National Laboratory. It is subject to revision or correction and therefore does not represent a final report. The information is not to be abstracted, reprinted or otherwise given public dissemination without the approval of the ORNL patent branch, Legal and Information Control Department.

ORNL-TM-110

Contract No. W-7405-eng-26

DRAFT

TECHNICAL PROGRESS REVIEW
NUCLEAR SAFETY

Volume 3, No. 3

Prepared for U. S. Atomic Energy Commission
by Oak Ridge National Laboratory

DATE ISSUED

FEB - 2 1962

OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
operated by
UNION CARBIDE CORPORATION
for the
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
C. G. BELL, E. E. GROSS, C. E. GUTHRIE,
W. de LAGUNA, A. W. SAVOLAINEN, and
C. S. WALKER, Assistant Editors, Oak
Ridge National Laboratory

TECHNICAL PROGRESS REVIEW

Nuclear Safety

Foreword

I. General Safety Criteria

Second Report of the Federal Radiation Council 2

State Control of Radioactive Materials 11

IAEA Nuclear Safety Activities 17

Administration of Radiation Protection at the Oak Ridge National Laboratory 25

References 40

II. Accident Analysis

Power-Excursion Experiments in Spert I 43

Reactor Stability Experience 47

Nuclear Reactor Stability Theory 52

U^{235} Fission-Product Heating 65

References 69

III. Control and Instrumentation

Criticality Excursion Detection in Processing Plants 76

Reactor Neutron Detection 84

Calculation of Control Rod Effectiveness 92

References 98

IV. Plant Safety Features

Radioactive Fluids Leak Detection 104

Leak Tests of Containment Structures 110

Seventh AEC Air Cleaning Conference 118

Glove Box and Hot Cell Design Considerations 128

References 140

V. Consequences of Activity Release

Analytical Requirements for Environmental Surveys 148

Waste Disposal by Incineration 157

Development of Environmental Survey Programs 161

References 178

VI. Current Events

Homogeneous Reactor Experiment No. 2 Operating Experience 185

The SL-1 Incident 186

Radiation Incidents at the MTR and ETR During 1959 187

Changes in Licensing Regulations 196

Action on Reactor Projects by Licensing and Regulating Bodies 203

Safeguards Reports and Selected Reading 216

References 219

I. GENERAL SAFETY CRITERIA

SECOND REPORT OF THE FEDERAL RADIATION COUNCIL

The Federal Radiation Council is the official government advisory body with respect to radiation matters directly or indirectly affecting health. The establishment of the Council and a summary of its first report[1] were discussed in an earlier issue of Nuclear Safety.[2] The second report[3] of the Council provides background material for a memorandum, entitled "Radiation Protection Guidance for Federal Agencies," which was approved by the President of the United States on September 20, 1961.[4] The memorandum consists essentially of four recommendations which are quoted here in some detail:

"It is recommended that:

1. The following Radiation Protection Guides be adopted for normal peacetime operations.

TABLE I – RADIATION PROTECTION GUIDES FOR CERTAIN BODY ORGANS IN RELATION TO EXPOSURE OF POPULATION GROUPS

Organ	RPG for individuals	RPG for average of suitable sample of exposed population group
Thyroid- - -	1.5 rem per year- - -	0.5 rem per year.
Bone marrow	0.5 rem per year- - -	0.17 rem per year.
Bone- - - - -	1.5 rem per year- - -	0.5 rem per year.
Bone (alternate guide).	0.003 micrograms of Ra-226 in the adult skelton or the biological equivalent of this amount of Ra-226	0.001 micrograms of Ra-266 in the adult skelton or the biological equivalent of this amount of Ra-226

. . . . . .

"2. The radiological health activities of Federal Agencies in connection with environmental contamination with radioactive materials be based, within the limits of the agency's statutory responsibilities, on a graded series of appropriate actions related to ranges of intake of radioactive materials by exposed population groups.

". . . The general character of these actions is suggested in the following table.

TABLE II – GRADED SCALES OF ACTION

Ranges of transient rates of daily intake	Graded scale of action
Range I----------	Periodic confirmatory surveillance as necessary.
Range II----------	Quantitative surveillance and routine control.
Range III----------	Evaluation and application of additional control measures as necessary.

. . . . . .

"3.(a) The following guidance on daily intake be adopted for normal peacetime operations to be applied to the average of suitable samples of an exposed population group:

TABLE III – RANGES OF TRANSIENT RATES OF INTAKE (MICROMICROCURIES PER DAY) FOR USE IN GRADED SCALE OF ACTIONS SUMMARIZED IN TABLE II

Radionuclides	Range I	Range II	Range III
Radium-226----	0-2	2-20	20-200
Iodine-131[1]---	0-10	10-100	100-1,000
Strontium-90--	0-20	20-200	200-2,000
Strontium-89--	0-200	200-2,000	2,000-20,000

[1]In the case of iodine-131, the suitable sample would include only small children. For adults, the RPG for the thyroid would not be exceeded by rates of intake higher by a factor of 10 than those applicable to small children.

"(b) Federal agencies determine concentrations of these radionuclides in air, water, or items of food applicable to their particular programs which are consistent with the guidance contained herein on average daily intake for the radionuclides radium-226, iodine-131, strontium-90, and strontium-89. Some of the general considerations involved in the derivation of concentration values from intake values are given in Staff Report No. 2.

. . . . . .

"4. For radionuclides not considered in this report, agencies use concentration values in air, water, or items of food which are consistent with recommended Radiation Protection Guides and the general guidance on intake.

"In the future, the Council will direct attention to the development of appropriate radiation protection guidance for those radionuclides for which such consideration appears appropriate or necessary. In particular, the Council will study any radionuclides for which useful applications of radiation or atomic energy require release to the environment of significant amounts of these nuclides. Federal agencies are urged to inform the Council of such situations."

It is apparent that the RPG values recommended in Table I memorandum are somewhat different than corresponding guides recommended by ICRP[5] or NCRP,[6] and it is instructive to review the considerations which led to these differences. In the case of the thyroid, the values recommended by the Council are half the corresponding values suggested by the ICRP. (In its 1958 report the ICRP did not directly recommend limiting doses for exposure of the population but did suggest 1/30 of the occupational MPC as a guide for exposure to radionuclides which did not give a significant genetic dose. It indicated these values were provisional and that the problem was under study. In the Munich supplement issued in 1958, the ICRP further indicated that this figure of 1/30 applied to an average in the population but that 1/10 might be considered as applicable for exposure of individuals in the population. The Munich Statement also indicated that these recommendations should apply to the doses received and not merely to the environmental levels.) The second FRC report

indicates that the reason for this lowering is the increased incidence of thyroid carcinoma in children given x-ray treatment for enlargement of the thymus. Several articles are available that give more details of this work.[7,8,9] Although the dosimetric data are not very precise, they indicate that the child's thyroid may not be more radioresistant than other body tissues, as is thought to be the case for the adult. Thus the FRC has recommended a value in line with limits set for other soft tissues. Probably the preponderance of informed opinion would now support this change.

Neither the memorandum nor the report[3] clarifies the intent of the FRC in giving two RPG values for bone. The alternative RPG value of "0.003 μgm of Ra^{226} in the adult skeleton or its biological equivalent" is used in the discussion on Ra^{226}, while the RPG value of 1.5 rem/yr is used in the discussion of Sr^{90}. It is not clear whether the FRC recommends both these as applicable to all bone-seekers or not. Since the NCRP and the ICRP have used Ra^{226} as the standard for all bone-seekers, it may be that the FRC is giving agencies the option of continuing this. The two RPG values are not equivalent, however, in terms of average rem dose to the skeleton. In fact, 0.003 μg of Ra^{226} in the skeleton delivers an average dose to the bone of about 0.87 rem/yr, whereas the RPG value based on dose is about 50% higher than this. This discrepancy is not a significant one in view of the large measure of uncertainty concerning the effects of such doses and body burdens. It will be noted that the FRC has decreased the values recommended by ICRP and NCRP for Ra^{226}. Instead of taking 1/10 of the occupational body burden as the criterion

for individuals in the population, it has adopted 1/30 for the occupational level.

If the RPG value for bone is applied to other radionuclides, the differences noted above become larger. The NCRP and ICRP have used biological data on relative toxicity of Ra^{226} and other bone-seekers in attempting to obtain safe skeletal burdens of these radionuclides. The experimental data available are rather limited, especially for Sr^{90}. Report No. 2 indicates, however, that long-term experiments now under way may provide a better assessment of the relative hazard. At present the ICRP and NCRP use a relative damage factor of 5 for bone-seekers other than radium, and this factor applies to all dose to bone, except that from photons; that is, a dose delivered to bone by alpha particles, beta rays, etc. (but not photon dose) is multiplied by 5 before the comparison with radium is made. In essence, this usage implies that these radionuclides are about 5 times as toxic as Ra^{226} when compared on the basis of equal rem dose to bone. Thus the ICRP and NCRP effectively limit an individual's skeletal dose from Sr^{90} to 0.58 rem/yr averaged in bone, whereas the FRC is using 1.5 rem/yr for this purpose. Although the FRC has chosen a lower RPG value for Ra^{226} than that of the other groups, it has in fact chosen a rather larger RPG value for Sr^{89} and Sr^{90}. The increase is by a factor a little less than 3. There is considerable discussion in the report[3] of the uncertainties involved in comparing bone dose from Sr^{90} and Ra^{226}, but there is little discussion of the reasons for preferring the higher RPG value of 1.5 rem/yr. It seems a bit strange that the FRC has based the RPG value for Sr^{89} and Sr^{90} on the rather arbitrarily chosen dose rate of 1.5 rem/yr rather than on a comparison with the Ra^{226} standard

with which we have 50 years of human experience. This is true especially in view of the extensive studies that have been under way for some time which are aimed at determining the relative toxicity of Ra^{226}, Sr^{90}, and various other radionuclides.

For bone marrow the FRC has chosen an RPG value of 0.5 rem/yr. This is in line with the dose the bone marrow would receive from external irradiation according to the exposure limits suggested by NCRP and ICRP for whole-body exposure. However, ICRP and NCRP suggestions would limit bone marrow dose of individuals in the population from Sr^{90} to 0.19 rem/yr, and thus the FRC also is recommending an increase in bone marrow dose by the same factor used for Sr^{90} in the case of bone. This increase in the bone marrow dose that is permitted from Sr^{90} or Sr^{89} is in accordance with the limits used by all these bodies for total-body exposure. Once the decision is taken to abandon the comparison with radium, there is little else to turn to except the comparison with external dose. The rather extensive and long-term experiments now under way which will more directly determine the relative toxicity of Sr^{90} and Ra^{226} in the skeleton should do much to clarify the significance of the increased dose permitted by these recommendations.

Having developed these basic recommendations, the FRC devoted much attention to secondary standards. In addition to having standards, every practicing health physicist will use a number of "active points" which serve to alert him to the possible need for corrective action and provide some lead time for such action to be effective. The choice of these action points may vary widely with the type of problem under consideration and with the means available for remedial action. Thus the choice of

these action points is always left to the agency responsible for the local program. In the memorandum and the report the FRC recommends a policy involving three ranges of daily intake and undertakes to recommend measures which are appropriate to each of these levels. (The upper boundary of Range II, Table III, is to correspond to the RPG.) These daily intakes are considered to apply to a suitable sample of exposed individuals, as specified in the first report of the FRC. It is curious that the FRC feels impelled to issue these directives even though the report[3] states that the intake values which define these ranges are not the only factors which should be considered in determining the appropriate action to be taken or the timing of such action. Probably those responsible for radiation protection programs will continue, as in the past, to base their action on as complete a knowledge of the problem as they can obtain. Not only a single level of intake will be considered but also the trend in time, the available knowledge concerning the source, the means available for meeting the problem, etc., before determining their action. One may hope the recommendations so strictly coupled here to the intake level will not cause any health physicist to overlook these other factors.

In recommending numerical levels for these ranges of intake for I^{131}, Ra^{226}, Sr^{90}, and Sr^{89}, the FRC has made several departures from previous practices. It has considered the infant or young child as the critical individual in the population when exposure to I^{131} is in question. This is in accord with British practice following the Windscale incident and is a major correction to previous practice. While the ICRP had indicated that such corrections were to be made, it has never implemented the recommendation by giving specific numerical values. Thus the intake values

for I^{131} are markedly lower than have been generally used. In fact, the lower range of values presents the Health Physicist with a not inconsiderable problem; he is directed to provide confirmatory evidence that the situation is within Range I, but it will be difficult to obtain this evidence by most technical means now available.

The intake levels of Sr^{90} and Sr^{89} are based on a comparison with calcium in the diet. British data on stable strontium obtained within the last few years[10] indicate that this is an acceptable basis for coupling the average daily intake with the average bone burden of strontium. There are fewer data available to indicate the extent of the perturbations to be expected as a result of the competition of other dietary elements, physiological state, calcium deficiency in the diet, etc. However, this is probably a sound enough approach to a population average, and the ranges are defined only in terms of averages over suitable samples of the population.

In the case of Ra^{226} the FRC has used data from populations using water supplies with relatively high radium content. The rule that the adult body burden represents about 50 daily intakes is taken as the basis. According to the report there is no marked age dependency of Ra^{226} concentration in bone. Perhaps such population studies are the soundest basis for population exposure when adequate data are available, since the conditions of the study are quite close to the exposure situation that is envisaged.

Like the first report of the FRC, report No. 2 is not documented, and thus it is not possible to indicate specific references for the many facts which are alluded to in the course of the argument. This is the

more surprising, as these reports are intended to fulfill an educational function and are not merely administrative orders to Federal agencies. As with the previous report, there is extensive discussion and consideration of the risks of radiation exposure. There is little discussion of the benefits or how these were balanced against the risks. Section 1.10 of the Report reads:

> "1.10 In order to consider as completely as possible the many factors involved in establishing radiation protection standards for the general population, the Council solicited comments from interested organizations and individuals. For this purpose, the Council prepared and transmitted widely a paper stating major policy issues involved in the development of radiation protection guidance in connection with the radionuclides considered in this report. Among these policy issues is the question as to the appropriateness of specific radiation protection standards from the point of view of their social and economic impact. Questions of this sort do not lend themselves to exact quantitative treatment. They are matters of judgement on which the best available information is brought to bear."

There is no indication of the specific policy issues raised, the organizations or individuals polled, or the response that was obtained. Thus only one pan of the balance is offered to public view. The act of balancing the benefits of radiation usage against the risk of radiation exposure remains as elusive as the Eleusinian mysteries of the past.

(W. S. Snyder)

STATE CONTROL OF RADIOACTIVE MATERIALS

Since the beginning of the "Atomic Age" the responsibility for the control of radioactive material has been vested in an agency of the Federal government. During the intervening years, in which the use of atomic energy has become quite diverse and widespread, many of the states have become concerned about the use of potentially hazardous materials within their borders. These states have not only established competency for dealing with some of the radiation problems within their own State Health Departments, but also have agitated for the Federal government (i.e., the AEC) to relinquish its regulatory control over many functions involving radioactive and special nuclear material.

In response to public desire, Congress in 1959 amended[11] the Atomic Energy Act of 1954 to permit state control of such activities. Kentucky has now become the first of the states to submit a formal proposal[12] for the assumption of certain authority over the regulatory control of by-product, source, and limited quantities of special nuclear materials. The Commonwealth of Kentucky is to be commended for its forward action in attempting to take full advantage of the provisions of Section 274 of the Atomic Energy Act. Their action will undoubtedly serve to set the pattern for other states to follow in assuming local responsibility in a field heretofore limited by federal control.

In conjunction with the proposed agreement with Kentucky, the Commission has also issued proposed regulations[13] authorized by Section 274c of the Atomic Energy Act for exercising certain controls over byproduct source and special nuclear materials in states with which it has agreements.

These controls involve the disposal of such materials and the transfer by the producer of products containing such materials. Some comments on these proposed controls are included in this Review in Section VI, under "Comments on Proposed Regulations."

Governor Combs of Kentucky made the initial presentation of his state's program to the Atomic Energy Commission on September 20, 1960. Subsequently modifications were made in consultation with the Commission's staff. After public hearings by the Commonwealth on June 30, 1961, the completed proposed regulatory program was submitted July 6, 1961, by Attorney General John D. Brechinridge to Dr. Glenn T. Seaborg, Chairman of the Commission. Until approved by the Commission, the agreement still has no force or effect. If and when approved and published for four consecutive weeks in the Federal Register, as provided by statute, Kentucky will be the first state to assume regulatory control over byproduct, source, and special nuclear materials in quantities less than a critical mass pursuant to Public Law 86-373, the Atomic Energy Act of 1954, as amended in 1959.

The Chairman of Kentucky's Advisory Committee describes the document as "an important forward step in the preparation of the Commonwealth for a leading role in the peace-time development of atomic energy for the benefit of our state," and recites the Commonwealth's intention to continue to take the initiative in doing all it can to develop sources of ionizing radiation for the benefit of its citizens in order to realize the full industrial, agricultural, and medical advantages of nuclear energy, while at the same time recognizing the vital importance of proper protection of its citizens' health and welfare.

James N. Neel, Jr., is the designated Coordinator of Atomic Activities for the Commonwealth. In the foreword of the publication, he refers, among other things, to the desirability of decentralization of licensing and regulatory functions from federal to state levels. He foresees the over-all effect as being a stimulation of the development and utilization of these materials by science and industry. His expressed belief is that this will attract nuclear and nuclear-related industries to locate within the state, since industry will prefer the time- and cost-saving advantages of dealing with a decentralized agency in obtaining and amending licenses for the possession and use of the materials involved.

The state's regulatory program proposal is described as comprised of: (1) Kentucky's amended Nuclear Energy Act authorizing the Governor to execute the Federal-State Agreement and conferring the licensing power over sources of ionizing radiation on state agencies; (2) executive orders creating a state task force on atomic energy and the position of Coordinator of Atomic Activities, and designating the Division of Radiological Health of Kentucky State Department of Health as the agency primarily responsible for licensing and regulating the possession and use of by-product, source, and special nuclear material; (3) twelve regulations based on Title 10 of the Code of Regulations[14] parts 2, 20, 30, 31, 40, and 70; and (4) other data relating to personnel, administration, and equipment.

The Coordinator refers to a careful effort to comply with the specification of Public Law 86-373 that a state's regulatory program must be compatible with the federal program. In addition, the state's radiation protection standards are said to have been prepared in cooperation with

the AEC and are designed to conform with the Commission's regulations for protection against radiation. The Coordinator also concludes that if another regulatory system of control were devised, even if deemed compatible, AEC licensees who have grown accustomed to the AEC procedure would be burdened with ascertaining the compatibility of the new system, and, finally, that the health and safety of the public would tend to be better secured by the fostering and adoption of a uniform and compatible system based on the federal program.

The Division of Radiological Health, referred to as the "Agency," proposes the following regulations designed to protect the health and safety of the public:

RH-1 Scope of Radiological Health Regulations

RH-2 Definition of Terms

RH-3 Rules of Practice

RH-4 Permissible Dose Levels, Concentrations, and Precautionary Procedures

RH-5 Waste Disposal

RH-6 Records, Reports, and Notifications

RH-7 Exemptions or Additional Requirements

RH-8 Licensing of Byproduct Material

RH-9 Radiation Safety Requirements for Radiographic Operations

RH-10 Licensing of Source Material

RH-11 Licensing of Special Nuclear Material

RH-12 Transportation of Radioactive Material

General and specific licenses are to be issued. Persons who use, transfer, possess, or receive source, byproduct, or special nuclear material

in quantities not sufficient to form a critical mass within Kentucky, pursuant to a license issued by the USAEC or any state having an agreement with USAEC pursuant to Section 274 of the Atomic Energy Act of 1954, as amended, may be exempted from the requirement for a license under Kentucky's regulations, provided that, such persons notify the Agency immediately of the presence of such materials within the state; and, the AEC, as well as its installations, are specifically exempted. Under these Kentucky regulations persons who possess byproduct, source, or special nuclear materials in less than a critical mass pursuant to a general or specific license issued by the AEC shall be deemed to possess such material pursuant to a license issued under the provisions of Kentucky's regulations and which will expire 90 days after receipt from the State Agency of a notice of expiration of such license, or on the date specified in the federal license, whichever is earlier.

Under the proposed regulations it shall become a state offense for any person to manufacture, construct, produce, transfer, acquire, or possess any special nuclear material, byproduct material, production facility, or utilization facility within the Commonwealth in the absence of compliance with the Federal Atomic Energy Act of 1954 and amendments and the rules and regulations promulgated by the USAEC. The Attorney General is empowered, whenever in his opinion there is or is about to be a violation of any of the Commonwealth's rules and regulations, to seek a restraining order or injunction or any other appropriate order.

Other provisions of the proposal include the following:

1. Nothing in the proposed regulations shall be interpreted as limiting the intentional exposure of patients to radiation for the purpose of medical diagnosis or medical therapy.

2. The Agency may, upon application by any licensee or upon its own initiative, grant exemptions it determines are authorized by law and will not result in undue hazard to life or property. The Agency may also, by rule, regulation, or order, impose upon any licensee requirements, in addition to those established, that it deems appropriate or necessary to protect health or to minimize danger to life or property.

3. No license is transferable or subject to assignment.

4. Although the custom is provided to hold public hearings thereon, in case of an emergency, the Governor is empowered by the Act to promulgate by executive order any additional rules and regulations.

(W. L. Harwell)

IAEA NUCLEAR SAFETY ACTIVITIES

The international Atomic Energy Agency which just passed its fourth anniversary was created primarily to facilitate the supply and use of fissionable materials. It was intended that the so-called "nuclear" powers would place at the disposal of the "have not" countries materials necessary for operating power reactors and that the agency would play the role of the broker in this transaction. In addition to reactors the widespread use of radioisotopes for medical, industrial, agricultural, and research purposes was also strongly supported.

The Agency has made great progress during the past few years, although all the aims originally set have not yet been reached. It soon became obvious that radiation safety measures would play an important role in the activities because relatively easy availability of fissionable materials represents an ever increasing risk in the hands of persons not accustomed to handling them. The degree of technical competence of personnel of the various recipient nations varies greatly, and measures which appear obvious under a certain set of circumstances must be spelled out in detail in other circumstances in order to avoid accidents. While individual scientists, even in the under-developed countries, are usually highly capable persons, difficulty arises from the lack of the carefully trained supporting personnel who in most cases must directly handle these materials.

It is not surprising, therefore, that the agency turned its attention to developing the necessary safeguards at an early stage. In view of the fact that development and operation of power reactors represents an advanced

stage of technical progress that has not been reached by the majority of the member countries of the Agency, it is quite natural that the first set of safeguards spelled out in detail is devoted to radioisotopes. While nuclear power is only a promise of the future, radioisotopes are already in full use in a large number of laboratories and it is obvious at an early stage that their handling and transportation must be carefully controlled.

In establishing the standards, the Agency had to overcome several difficulties of nonscientific nature. Although in principle a scientific body, the Agency is a political organization associated with the United Nations and, as such, it may occasionally get "bogged down" in politics. When the safeguards provisions and procedures for international control of nuclear materials supplied to member nations by the agency was adopted by the Board of Governors, the adoption of the provisions was strongly opposed by the Soviet-block nations. This opposition was primarily political rather than technical: the representatives of these nations insisted that a simple pledge on the part of the recipient nations stating that they would not divert any materials received to nonpeaceful applications would be satisfactory; they proposed to make consideration of the provisions dependent on the resumption of the Geneva Atomic Weapons Control discussions. The Soviet position was that inspection would be a violation of national sovereignty.

Radioisotope Handling

It is interesting to note that in spite of the political problems and the many organizational difficulties experienced during the first year

of its existence, in 1958 the Agency published the first of its safety manuals entitled "Safe Handling of Radioisotopes."[15] The foreword of this book outlines the broad philosophy of the Agency:

> "Under its Statute the International Atomic Energy Agency is empowered to provide for the application of standards of safety for protection against radiation to its own operations and to operations making use of assistance provided by it or with which it is otherwise directly associated. To this end authorities receiving such assistance are required to observe relevant health and safety measures prescribed by the Agency"

There is an important legal difference between this manual and the rules and regulations imposed by a national government. In the absence of legislative power and a police force, the safety problems remain largely a responsibility of the national government of the country in which the material is used; however, guidance must be provided for the benefit of the law-enforcing agencies of those countries.

This first publication of the series is primarily a statement of general safety principles without spelling out detailed standards. On occasions, as in the case of the maximum permissible levels for exposure to external radiation and to radioactive contamination, reference is made to pertinent recommendations of the International Commission for Radiological Protection (ICRP).

The manual also treats related subjects, such as the medical supervision of workers, determination of radiation exposure of personnel, monitoring, and record keeping. Turning to more specifically radiochemical studies, it discusses sealed and unsealed sources, their storage, the transportation of radioactive materials, accidents, and decontamination procedures. Finally, special attention is paid to radioactive waste control and disposal. The maximum permissible levels mentioned above are

given in the appendix on the basis of data supplied by other international organizations.

Health Physics and Medical Aspects

The publication of the manual on the handling of radioisotopes was soon followed by two brief books, numbered 2 and 3 of the Safety Series, entitled "Health Physics Addendum"[16] and "Medical Addendum."[17] There is a great difference between the original publication and these adenda: the first one[15] was prepared by an international panel not only as a technical but also as a legal guide. On the other hand, the "Health Physics Addendum" was prepared by two scientists and consists of an excellent short introduction to nuclear physics and, in particular, to health physics. In presenting a very abbreviated description of nuclear physics fundamentals it treats the interaction of radiation with matter, in particular with the living cells, and then discusses more in detail radiation units, dose rates, and maximum permissible doses. Measures of radiation, monitoring instruments, and control measures receive detailed treatment. The main part of the book is its chapter dealing with radiation shielding, basic design features of working places, and description of handling equipment for radiation sources, which are treated in detail. The practical points of radiochemical work, such as containers for radioactive materials, protective clothing for the workers, transport of the radioactive substances, decontamination, and waste disposal procedures, are all discussed.

Of special interest are the two short chapters on radiation accidents and emergency procedures and on administration of radiation protection. The radiation industry can be proud of its excellent safety record up to now, although there have been some potentially dangerous events. This is

a tribute to the careful work of the individuals involved in investigations in this novel field of science. However, this very absence of accidents and lack of experience in dealing with them makes it difficult to evaluate how the various installations would fare should a major emergency occur involving a large number of people. Therefore it is imperative to give thought to the emergency organization in the various laboratories. In view of their excellent radiation safety record, this phase of the safety program might be neglected, even in the more advanced countries. The only way to ensure that an accident will not become a major catastrophe is to prepare for it by carefully organizing emergency procedures and by holding test evacuations and simulated accident drills as often as possible without interfering with the work of the installation. Many weaknesses of a point become obvious only at the time of a realistic field test. The IAEA book deserves credit for attracting attention to these matters.

The administration of radiological protection is another sometimes neglected field which deserves attention. The general principles to be considered are outlined in the last few pages of the book. It is emphasized that the person in charge of the organization and administration of the radiation protection program must be responsible for the design of working areas and the specification of the permissible types of radiation sources to be used there. He must be aware of the environmental factors related to the disposal of radioactive material and of the problems presented by neighboring occupied areas. In the opinion of the authors, this indicates that the radiation protection officer of an installation must have management responsibilities rather than being a staff member who only takes care of the monitoring and reporting duties.

Practices in the administration of radiation safety vary considerably in the various countries and often change even in the nuclearly advanced countries. It would be highly desirable if the Agency or a similar international body or technical society would undertake the correlation of the management techniques used for radiation control.

As indicated by its title, the "Medical Addendum"[17] to the "Safe Handling of Isotopes" is primarily devoted to the medical aspects of the program. It not only discusses the basic concepts of radiobiology and radiopathology and the incorporation of radioisotopes, but it also discusses in great detail the health supervision of radiation workers, including personnel monitoring, medical care, and the keeping of health records. The required minimum medical facilities and equipment are indicated.

It should be emphasized that these booklets are relatively brief and thus do not give detailed instructions on every phase of this somewhat complicated problem; in addition, not being primarily technical publications, they do not give detailed bibliographies on the subjects treated. They represent a very important step in an international agreement on the minimum standard of a new industry which up to now has an excellent safety record but which carries within it the potentiality of causing a tremendous amount of damage.

Transportation of Radioactive Materials

If there are many problems connected with handling of radioactive materials in a given installation placed under the jurisdiction of a national government it is obvious that transportation of such materials

involving two or more governments and international bodies presents even more serious problems. Carrying radioactive materials across state boundaries necessitates their handling by a large number of persons untrained in these specific hazards. Therefore their mode of transportation and packaging must be such that no accident is liable to arise even if the shipping containers are mishandled or are broken in transit.

The regulations should be applied uniformly by all countries, but this can be achieved only by an international agreement on the basic safety standards. It is expected that the regulations framed by the Agency for its own licensing practices will be accepted by the member countries as the basis of an international convention. In view of the different technical aspects involved, two separate panels have been convened by the Agency to draft regulations for the safe transportation of all types of radioactive materials: the first one studied the transportation of radioisotopes and radioactive ores of low specific activity, while the second one was assigned the task of developing safety measures for the transportation of large radioactive sources and fissile materials. As the basic principle, the dose rate of external radiation from shipment of radioactive materials was limited to values considered safe in the light of maximum permissible radiation levels. In addition, the quantities of material per package and per shipment were limited in order to reduce the likelihood of contamination in the event of an accident. The permissible quantities are dependent largely on the nature of the materials and their levels of activity.

The problem of transporting radioactive materials is complicated by legal considerations. The questions of liability and compensation for

damages suffered by third parties must be solved, and agreement on the solution is far from being achieved. In the first version[18] the Agency paid primary attention to technical and administrative factors in an attempt to find a compromise between the various national regulations.

Conclusions

In these attempts to codify nuclear safety procedures and thus prepare the way for a truly uniform international practice in this field, the Agency makes good use of the experts from its member countries. Whenever the need arises, advisory panels on safety subjects are brought together for evaluating conditions and formulating recommendations in various fields. To mention only a few cases, such a group of experts was empaneled for investigating the Yugoslav reactor accident in order to determine the factors that caused it and to provide safeguards for preventing similar occurrences in the future. At the request of the Dutch Government, an advisory panel was organized for evaluating the safety of a reactor project in the Netherlands. Another panel of experts from nine countries has been called together for devising methods of determining the level of radioactivity in the biological environment. A similar group started work recently on drafting a manual on low-level-waste disposal techniques.

In summary, it can be stated that, although up to now the main achievement of the Agency consisted in the coordination of existing national regulations on the field of radiation safety, it can render a still greater service by establishing uniform standards throughout the world and by developing new methods for detection and control of radioactivity in all its forms through mutual collaboration. (Francois Kertesz)

ADMINISTRATION OF RADIATION PROTECTION AT THE OAK RIDGE NATIONAL LABORATORY

(Editor's Note: This article was originally intended to be a survey of radiation protection administrative practices in the United States. It quickly became apparent, however, that too little had been published on the subject to make such a review possible. Therefore this is the first of a series of invited articles which will describe the administration of radiation protection at the various atomic energy installations.)

Introduction

The Oak Ridge National Laboratory presents a wide variety of radiation safety problems. The Laboratory population (about 4200 people) is engaged in a spectrum of activities ranging from fundamental research to routine production. The work of the Laboratory is broadly divided into seven fields. In reactor technology, the problems associated with in-pile experiments, the examination of irradiated samples, and the operation of reactor pilot plants are encountered. In the chemical technology program, small-scale laboratory development and hot-cell activities culminate in the pilot-plant demonstration of separation processes for highly irradiated uranium and plutonium and for the transplutonium isotopes. The Laboratory is active in the development of processes for production of stable and radioactive isotopes. In this program, most of the fission products are separated, some in multikilocurie quantities. Many isotopes not produced in fission are prepared by reactor or cyclotron irradiation of special targets. In basic research, a wide variety of radioisotopes is employed in the biology, chemistry, physics, metallurgy, and health physics programs. These activities also make extensive use of x-ray

machines, cyclotrons, and other accelerators for the production of ionizing radiation. In the education program of the Laboratory, students engage in training activities which employ source and fissionable material and radioisotopes.

During 1959, when unusually large quantities of radioactive materials were being employed in several of the Laboratory's programs, three accidents were experienced.[19] Early in 1960, Laboratory management consequently instituted a new radiation safety and control program which had the dual objective of upgrading Laboratory facilities[19] and improving operating practices. The ORNL Radiation Safety and Control program, to be described here, was developed to achieve the desired goals without imposing so many restrictions that the Laboratory's primary mission – research and development – would be compromised. Under this program, a progressive decrease in both the frequency and seriousness of radiation accidents at the Laboratory has been observed.

The essential elements of the ORNL program are: (1) organizing for coping with radiation protection problems; (2) assigning specific responsibilities for safety to the various Laboratory groups; (3) maintaining each facility in a safe condition; (4) requiring safe operating procedures; (5) profiting from the experience offered by minor accidents; (6) personnel training; (7) holding the interest of Laboratory personnel in radiation safety and control problems; and (8) planning for radiation emergencies.

Organizing for Radiation Safety

The radiation protection organization which has evolved at the Laboratory is shown in Fig. 1. Under the Director and Deputy Director of the Oak Ridge National Laboratory there is a Director of Radiation Safety and

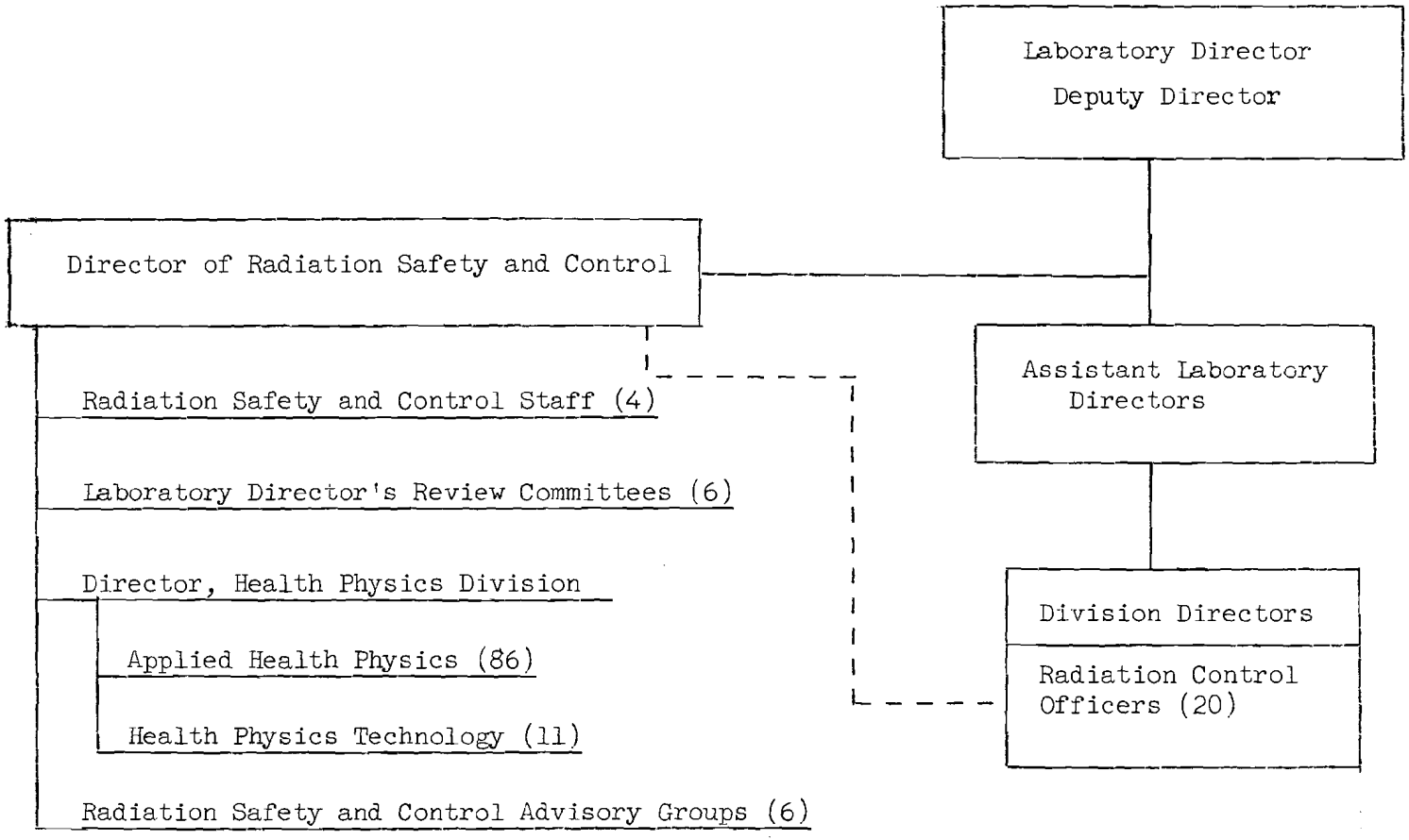

Fig. 1. Organization for Radiation Safety at the Oak Ridge National Laboratory

Control who is a member of the Laboratory Director's staff. Close and frequent contact is maintained between the Laboratory Director, Deputy Director, and the Director of Radiation Safety and Control. In addition to the Director, the office of Radiation Safety and Control includes four staff members — two chemical engineers, a chemist, and a physicist. Each staff member is assigned responsibility for following closely the activities of certain of the 20 Laboratory divisions which handle significant quantities of radioactive materials. Each one also specializes in certain key elements of the radiation safety program: for example, criticality, safe design criteria, training, waste disposal, etc. One staff member serves as full-time executive secretary of the Laboratory Director's review committees.

The six review committees are regarded as the Laboratory Director's "eyes and ears" on radiation safety matters. These committees consistently make valuable suggestions for improving the safety of the Laboratory. Although most committee business is handled by the Radiation Safety and Control office, direct contact between the committees and the Laboratory Director is encouraged and left to the discretion of the committee chairmen. Contact between the Director and the committees is assured by an annual meeting with each committee. On this occasion, the highlights of the past year's activites are reviewed, particular problems of concern to the committees are discussed, and plans for the coming year are described. The office of Radiation Safety and Control is responsible for informing divisions of all committee recommendations affecting their operations and for seeing that the committee's recommendations are, in fact, implemented.

The Health Physics Division at ORNL has two rather distinct functions: health physics research and health physics services in direct support of the Laboratory's research and development activities. Health Physics services may be further broken down into Applied Health Physics and Health Physics Technology. In the areas of Applied Health Physics and Health Physics Technology, the Director of the Health Physics Division reports to the Laboratory Director through the Director of Radiation Safety and Control.

Six advisory groups both supplement the staff of Radiation Safety and Control and make available specialized knowledge to it.

The Laboratory Director established a post within each division for handling radioactive materials. The individual holding this post is known as the Radiation Control Officer. Depending on the nature of the division's activities, he spends between 20 and 80% of his time on radiation safety matters. The Radiation Control Officers, in monthly meetings with the Director of Radiation Safety and Control, are informed of changes in radiation safety policy and are made aware of those areas where increased attention should be focused. In general, the radiation safety program is implemented in each division through the Radiation Control Officer. Experience has shown the Radiation Control Officer function to be one of the most important elements of the Oak Ridge National Laboratory's radiation safety program.

Radiation Safety Policy

The radiation protection program at the Oak Ridge National Laboratory has the following broad objectives:

1. To carry out all operations with the lowest reasonable personnel exposure to radiation and contamination; in no case should internal and external exposures exceed the recommendations of the Federal Radiation Council and the National Committee on Radiation Protection.

2. To perform all work in such a manner that losses resulting from contamination are minimized; such losses may include research, development, and production time; facility or equipment abandonment; and the cost of cleaning up contamination.

3. To maintain environmental contamination at as low a level as possible consistent with sound operating practice; in no case should the atmospheric and water contamination outside the controlled area exceed the maximum permissible concentration values for the neighborhood of an atomic energy installation.

In order to carry out these objectives, specific responsibilities and authorities are delegated by the Laboratory Director.

Assignment of Responsibility

Laboratory Director and Deputy Director. Both the Laboratory Director and the Deputy Director show deep personal interest in all matters of radiation safety. This is manifested by personal participation in the translation of AEC regulations into Laboratory operation, by vigorous investigation of radiation accidents, and, most importantly, by strong enthusiastic support of the radiation safety and control programs. The importance of top management interest in, and support of, the radiation safety program cannot be overemphasized.

Director of Radiation Safety and Control. The Director of Radiation Safety and Control establishes, on behalf of the Laboratory Director,

policy with respect to radiation protection and ascertains that Laboratory policy with regard to radiation safety and control is met at all times.

Laboratory Divisions. It is essential to the Laboratory's program that each Laboratory division assume the primary responsibility for attainment of the objectives which have been set with regard to personnel exposure, minimizing losses due to contamination, and environmental contamination.

Radiation Control Officers. The Radiation Control Officer represents the Division Director on matters of radiation safety. He actively and aggressively keeps under continuous surveillance all operations of the Division that could lead to radiation incidents. He, under the Division Director, assumes full responsibility for radiological safety in the Division. In spite of the great responsibility for radiation safety which is borne by the Radiation Control Officer, it is made perfectly clear that the Division Director, himself, bears the primary responsibility and will be held accountable for radiation safety within his Division.

Applied Health Physics. A major role in the radiation safety program is played by the Applied Health Physics Section, which advises the Laboratory divisions on the health physics aspects of their programs. It also carries out the personnel and environmental monitoring programs. The Applied Health Physics section includes 86 people, 6 of whom are administrators, 13 are engaged in the personnel monitoring program that assesses the exposure which Laboratory employees receive, 47 perform the radiation survey needed to establish radioactivity levels in Laboratory facilities and areas where ionizing radiation is present, 11 are engaged in the bioassay and instrument calibration program, and 9 carry out area monitoring

for the purpose of determining the environmental contamination resulting from ORNL operations.

Health Physics Technology. In many cases, existing "know-how" is inadequate to assess radiation hygiene problems which arise at the Laboratory. In order to cope with these, a Health Physics Technology Section was established to perform the necessary research and development in support of the Laboratory's radiation protection program. This Section has six people who are engaged in activities such as developing analytical techniques suitable for detection of environmental iodine in concentrations that are approaching the newly established limits, developing methods for determining in body fluids radioisotopes unique to the ORNL program, and putting into operation a newly acquired whole-body counter.

Instrumentation and Controls. The Instrumentation and Controls Division is responsible for specifying the health physics instruments to be employed at the Laboratory, with the advice of the Applied Health Physics Section. In making such judgments, cognizance is given to past experience with similar instruments, the existence of spare parts, and maintenance problems. This Division also performs the necessary maintenance work on health physics instruments.

Laboratory Shift Supervisor. Under the ORNL operating pattern, a Laboratory Shift Supervisor is present at all times. He immediately and automatically assumes responsibility for an emergency following an incident and remains in charge until he is relieved by Laboratory management.

Radiation Safety and Control Advisory Groups. Standing and "ad hoc" groups are used to supplement the Radiation Safety and Control staff and to cope with specialized problems which arise. The standing groups are:

1. The Advisory Group for Fixed Radiation Control Instruments, which specifies the numbers and types of fixed instruments required for new and existing facilities and advises on the necessary replacement schedule.

2. The Advisory Group for Portable Radiation Control Instruments, which specifies the numbers and types of portable instruments required for new and existing facilities and advises on the necessary replacement schedule.

3. The Radiation Incident Advisory Group, which gives advice on the action to be taken following an incident, including health physics procedures, cleanup techniques and standards, staffing the cleanup effort, and work restrictions to be imposed on exposed personnel.

4. The Analytical Chemistry Coordinator, who arranges procedures and priorities for the analysis of samples during an emergency.

5. The Design Criteria Review Group, which approves (for conformance with radiation safety and control policy) the design criteria for facility changes and for new facilities.

6. The Radioactive Materials Transport Advisory Group, which establishes the criteria for shipping containers for radioactive materials and recommends procedures to be employed for shipments in the Laboratory and to other Oak Ridge plants.

Safety of Facilities

In recognition of the importance of safe facilities for the handling of hazardous materials, criteria paralleling those already established for reactors have been set up for assessing the adequacy of Laboratory facilities, hot cells, and chemical plants. These criteria were described

earlier.[19] Existing facilites either have been or are being modified to meet these standards.

Preliminary design criteria for all new facilities are presented to the office of radiation Safety and Control by the responsible division for review to determine compliance with safety criteria. Radiation Safety and Control approval is required before detailed engineering design is started. Before construction is started, the appropriate Laboratory Director's review committees must also approve the design. Appropriate drawings, a hazards evaluation, emergency plans, and operating procedures are presented at this time for inspection and evaluation. Acting on the recommendations of the review committee, the Director of Radiation Safety and Control gives written approval of the facility.

Review of Operating Practices

Laboratory Director's Review Committees. For several years the Laboratory Director has called upon review committees to advise him on the safety of the Laboratory's operations. Early in 1960 the number of such committees was increased and, at present, totals six. Each of these committees consists of a chairman, five or six members, and an executive secretary in common. The executive secretary is a member of the Radiation Safety and Control staff. Only senior personnel are appointed to the Laboratory Director's committees; their terms of office are two or three years. Although the number of meetings held by each committee varies, the total number of committee meetings in 1960 was 86. During that period, which was typical, approximately 250 technical man-weeks of effort were devoted by the Director's committees to the review of safety aspects of

Laboratory operations. Committee review may be initiated by Laboratory management, the committee chariman, the Director of Radiation Safety and Control, or upon request of a Division. In general, each facility is reviewed by a committee at least annually.

Audit of Operating Practices. It is required that all nuclear or radiochemical operations involving 1 g or more of plutonium, or material of equivalent hazard, be formally reviewed by the appropriate Laboratory Director's review committee. Reviews of operations with less activity may be made at the discretion of the Laboratory Director's review committees and the executive secretary. Compliance with the recommendations of the committees is effected through the office of Radiation Safety and Control.

The executive secretary requests from the responsible Division a hazards report, operating procedures, and emergency plans covering the operation for committee review prior to the formal meeting.

In these audits, particular attention is given to the following points:

1. Kind and quantity of material to be handled.
2. Containment features of the facility.
3. Condition of instrumentation and controls.
4. Adequacy of startup, operating, maintenance, and shutdown procedures.
5. Adequacy of emergency procedures.
6. Operator training and qualifications.
7. Adequacy of supervision.
8. Unusual occurrences since previous review.

9. Radiation exposure record.

Staff members of the office of Radiation Safety and Control also conduct frequent informal audits of all Laboratory operations.

Unusual Occurrences

In the belief that seemingly insignificant occurrences frequently presage serious accidents, all unusual occurrences at the Laboratory are carefully examined. If a violation of Laboratory rules is entailed, or if the accident could have been serious under less fortunate circumstances, the responsible Division is required to investigate it, to prepare a report describing the accident, and, finally, to make recommendations for preventing recurrences. If it seems indicated, suitable Laboratory-wide restrictions or procedural requirements to preclude repetition of similar accidents are instituted.

Training

The Oak Ridge National Laboratory's radiation safety training program has several facets. New employees, upon arrival, are given general orientation, including an introduction to radiation safety. This is followed by on-the-job instruction by the employee's immediate supervisor in specific radiation protection practices, procedures, and responsibilities. The radiation safety program is discussed with the scientific staff and supervisory employees in Staff Conferences. In 1960, approximately 132 people participated in such Staff Conference discussions.

In 1961, two radiation safety training courses were instituted and presented to approximately 800 employees. The first, which will be

repeated once each year, consists of 12 hr of instruction and covers such topics as ORNL radiation safety policy, radiation units, permissible exposures, radiation damage to man, instruments for radiation detection, chemical toxicity, containment criteria, initial hazards evaluation, the Laboratory Director's review committees and operating procedures, Applied Health Physics Section procedures, radiation emergency procedures, intra-laboratory transfer, handling, and storage of radioactive materials, and waste disposal. As a text for the training program, a 117-page Radiation Safety and Control Training Manual was prepared that serves as a basis for the lectures. In 1961, approximately 1300 copies of the Training Manual were distributed to Laboratory personnel.

An abridged training course was presented to basic research workers handling relatively small quantities of radioactivity. In addition, a 3-hr training course was presented to approximately 70 technical summer employees.

Radiation Safety and Control Publications

Quarterly Reports. A Radiation Safety and Control quarterly report is distributed to Laboratory management, Division Directors, and Radiation Control Officers. It provides a running assessment of the state of radiation safety within the Laboratory and emphasizes those areas where greater divisional attention is needed. An Applied Health Physics Section quarterly report, prepared by the Health Physics Division, summarizes personnel exposures during the quarter and evaluates local and environmental contamination problems stemming from Laboratory operations.

Advisory Bulletins. Advisory Bulletins are published periodically by Radiation Safety and Control. These bulletins describe selected unusual occurrences involving personnel exposure or spread of radioactive contamination. Their purpose is to maintain Laboratory awareness of radiation safety and to effect improvement of specific hazardous conditions or practices which are brought to light.

Radiation Safety and Control Handbook. To provide a convenient reference source, a Radiation Safety and Control Pocket Handbook was prepared and 1200 copies were distributed. This Handbook serves the dual purpose of providing a simple introduction to radiation safety and health physics for nontechnical personnel and of providing a convenient source of reference material for scientists and engineers.

In addition to the preceding publications prepared by Radiation Safety and Control, in 1961 the Health Physics Division published the Health Physics Manual, which constitutes a "bible" of radiation protection procedures and practices at the Laboratory. It is widely distributed, and all Divisions are required to carry out their operations in conformance with the management-endorsed recommendations contained therein.

Planning for Emergencies

The Laboratory Shift Supervisor, who represents the Laboratory Director, is responsible for the handling of emergencies. An Emergency Manual has been prepared and given wide distribution to assure expeditious, effective handling of emergencies.

Basic elements of the emergency plan are:

1. Evacuation. Personnel are evacuated from the accident area, and contaminated equipment, buildings, or areas are isolated.

2. Delineation. Radiation or contamination zones are closed off. When contamination is the primary consideration, the affected area is sealed off if possible.

3. Containment. If air contamination is present, air conditioning and ventilation systems are shut off if conditions permit. If liquid wastes are involved, their release to the environment is prevented.

4. Sampling. Samples of air, liquid, and surface contamination are obtained for immediate analysis to identify and determine the concentration of potentially hazardous materials.

5. Decontamination. As soon as possible, steps are taken to remove contamination to a sufficiently low level that normal Laboratory operations may be resumed.

6. Monitoring. Personnel are surveyed on exit from the Laboratory and those who are contaminated or have been exposed are segregated for decontamination or medical treatment.

Definite responsibilities for specific phases of the emergency plan have been assigned, and about twice a year an unannounced Laboratory-wide evacuation, premised on a plausible situation, is held. The practice evacuations are monitored by a group of judges, and their comments are used to improve both the emergency plan and the response of the emergency groups. (F. R. Bruce)

References

1. Federal Radiation Council, Background Material for the Development of Radiation Standards, Report No. 1, May 13, 1960.

2. First Report of the Federal Radiation Council, Nuclear Safety, 2(2): 6-7 (December 1960).

3. Federal Radiation Council, Background Material for the Development of Radiation Standards, Report No. 2, September 1961.

4. "Radiation Protection Guidance for Federal Agencies," An Executive Order signed by the President of the United States on September 20, 1961, Federal Register, 26: 9057-8, September 26, 1961.

5. Recommendation of the International Commission on Radiological Protection, Radiation Protection, adopted September 9, 1958, Pergamon Press, New York.

6. Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and in Water for Occupational Exposure, NBS Handbook 69 (June 5, 1959).

7. C. Lenore Simpson and L. H. Hempelmann, The Association of Tumors and Roentgen-Ray Treatment of the Thorax in Infancy, Cancer, 10: 42-56 (1957).

8. C. L. Simpson et al., Neoplasia in Children Treated with X-Rays in Infancy for Thymic Enlargement, Radiology, 64: 840-845 (June 1955).

9. C. Lenore Simpson, Carcinogenesis from Irradiation of the Thymus Gland, Acta Union Int. Contre Le Cancer, 16: 448-451 (1960).

10. Maximum Permissible Dietary Contamination after the Accidental Release of Radioactive Material from a Nuclear Reactor, Brit. Med. J., I: 967-9, (1959).

11. Atomic Energy Act of 1954, Public Law 703; as amended in 1959, Public Law 86-373.

12. Policies and Procedures of the Commonwealth of Kentucky for the Regulation and Licensing of Byproduct, Source and Special Nuclear Materials in Less Than a Critical Mass, prepared by The Select Kentucky Task Force on Atomic Energy, June 1961, available from The Kentucky State Department of Health.

13. "AEC Regulatory Authority in Agreement States," Title 10, Code of Federal Regulations, Part 150; proposed regulation printed in the Federal Register, September 29, 1961, 26(188).

14. Rules and Regulations, Title 10, Code of Federal Regulations, Chapter 1, Law Library of the USAEC, Washington, D. C., 1958, plus subsequent amendments.

15. Safe Handling of Radioisotopes, International Atomic Energy Agency, Vienna 1958, by a panel of international experts.

16. G. J. Appleton and P. N. Krishnamoorthy, Health Physics Addendum, International Atomic Energy Agency, Vienna, 1960.

17. Dr. F. Hervik and Dr. H. Jammet, Medical Addendum, International Atomic Energy Agency, Vienna, 1960.

18. Safe Transport of Radioactive Materials, International Atomic Energy Agency, Vienna, 1961 (STI/PUB/40).

19. Radiation Safety Improvements to ORNL Facilities, Nuclear Safety, 2(3): 40-43 (March 1961).

II. ACCIDENT ANALYSIS

POWER-EXCURSION EXPERIMENTS IN SPERT I

Numerous light-water moderated reactors, among them Dresden and Yankee, employ slightly enriched UO_2 as fuel. Other similar reactors, such as the Consolidated Edison reactor, employ fuel rods that contain enriched UO_2 mixed with a large amount of ThO_2. In these oxide-fueled reactors, the dimensions of the fuel elements are usually such that the time constant for heat transfer from the fuel to the moderator is of the order of several seconds. Under such circumstances, the reactivity effects of temperature or density changes of the moderator are much too slow to effectively stop excursions initiated by large, rapid, reactivity additions. Safety calculations then rely on the prompt Doppler effect to counteract such reactivity additions. Therefore the safety of bulk-oxide-fueled reactors depends on the mechanism entirely different from the shutdown mechanism that controls excursions in plate type reactors and which has been studied extensively.

The first systematic investigation of power excursions, in a reactor with fuel rods of the type discussed above, was carried out in the summer of 1961 at the SPERT I facility at the National Reactor Testing Station.[1,2,3,4] The core used was composed of the 6-ft-long stainless-steel-clad UO_2 fuel pins employed earlier in the NS Savannah critical experiments.[5] A quarter-section of the core consisting of 592 of these pins on a 0.663-in. pitch is shown in Fig. 2. In its original version, hereafter referred to as the "unconstrained core," the fuel rods were fastened at both ends but were free to move elsewhere. (This freedom does not exist in the actual reactor of the NS Savannah or in other power reactors.)

ORNL-LR-Dwg. 65365
Unclassified

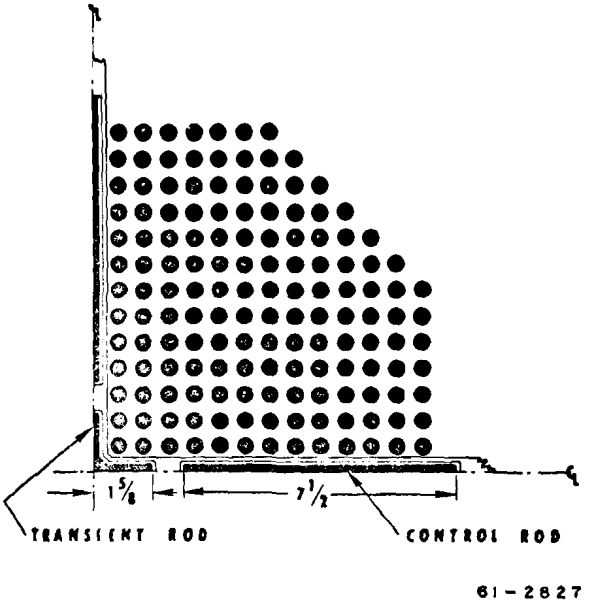

Fig. 2. Quarter-Section of Spert I Oxide-Core Containing 592 Fuel Pins (Reprinted from USAEC report IDO-16693).

In a later version, the "constrained core," spacers were inserted between the fuel rods to restrain sideways displacement. The initial periods of the excursions investigated ranged down to 3.2 msec, corresponding to an initial, stepwise reactivity addition of approximately $2.

At the present time (November 1961) the experiments have barely been finished, but the following general conclusions have been reported:[1,4]

1. A prompt, negative reactivity effect, presumably largely Doppler, indeed counteracts the reactivity addition.

2. In the unconstrained core, a positive reactivity effect appears, with a time constant of the order of 10 to 100 msec. This reactivity effect may reach almost $1 and has to be compensated by additional Doppler effect (as well as some moderator density effect), requiring a larger temperature rise of the fuel and an increase in the energy generated during the excursion by a factor of about 2 or 3 at the most. In the excursions, with initial periods below 40 msec, the original peak is over before the positive reactivity effect appears and thus there is a second peak in the power trace.

3. The positive reactivity effect mentioned under item 2 has been identified with the effect of bowing of the fuel rods as a result of the gradient in the power density. This identification was accomplished by motion pictures of the fuel rods during the excursion, and also by the fact that the constrained core did not show the positive reactivity effect.

4. The fuel rods can be expected to bow into the regions of higher power density, or higher flux. For this bowing to result in a positive reactivity effect, the high-flux regions must also be the regions where the importance of a fuel rod is high. This condition exists, in the

reactors in question, at the interfaces between fueled and unfueled regions, such as the outer surface of the core, and possibly the sides of empty control-rod channels.

5. The power traces for the constrained core, and also the initial parts of the power traces for the fast transients of the unconstrained core, are, in first approximation, in agreement with a prompt, negative, reactivity coefficient of about 3 cents per Mw-sec. For the fastest transients investigated, however, a systematic deviation toward lower reactivity coefficients seems to appear.

The positive reactivity effect of the fuel-rod bowing was somewhat of a surprise. Such effects have been investigated extensively for fast reactors, but they were widely believed to be unimportant for water-moderated thermal reactors, as the reactivity effects due to water-temperature or density changes would be far greater. After these experiments, it is clear that the bowing effects can make themselves felt anyway, because they act before the water temperature or density changes appreciably.

Positive reactivity effects of fuel-element deformation also manifested themselves surprisingly in the Sodium Reactor Experiment,[6,7] and some modification of the SRE fuel element was indicated.

To what extent actual power reactors are subject to positive reactivity effects from fuel-rod bowing, or, more generally, from mechanical deformation, will have to be determined on an individual basis. Hopefully, it will be possible to design reactors so that the power-induced bowing or other mechanical deformation will give a negative, rather than a positive, reactivity effect. (W. K. Ergen)

REACTOR STABILITY EXPERIENCE

A common feature of recent papers on experimental observations of reactor stability characteristics is the emphasis on the effects of "minor" phenomena which could easily be ignored in the design. While much of the information reported is quite specialized, it must not be assumed that experience which has been gained with a particular reactor is restricted in usefulness to reactors of the same type; the little things which have caused trouble in an aqueous system may have their analogs in a molten-salt reactor. A further impression is one of confidence that stability problems can be understood and dealt with rationally.

Experimental Breeder Reactor

EBR experience provides perhaps the strongest support for these contentions. Smith et al.[8] have reported their work on instability studies with the Mark III core of the EBR-I. While the EBR-I operated smoothly enough at full power and steady state with the Mark II core, departures from the steady state often caused power oscillations. Further, a prompt positive power coefficient of reactivity was evident on varying the coolant flow, and a partial meltdown eventually occurred because of it. Specifically, the power would increase following a reduction in flow, pass through a maximum, and eventually decrease to some lower equilibrium value; or, conversely, the power would decrease following a rise in flow, pass through a minimum, and eventually increase to some higher equilibrium value. The power coefficient was considered to have a large but slowly acting negative component and a smaller but prompt positive component. The authors summarized[8] the work of others which showed that fuel rod

bowing was the principal source of the prompt positive power coefficient and that the Doppler-effect hypothesis could be rejected. The delayed negative temperature coefficient is explained by the authors as the result of delayed structural expansion.

The Mark III core was designed to eliminate instabilities characteristic of previous cores. The results of the studies reported show that the fully ribbed and rigid Mark III loading was thoroughly successful in this respect. No positive reactivity effects were noted for this core. Extrapolation of test data showed that the reactor, which normally operates at 1.2 Mw, could be made unstable only by exceeding 1000 Mw; but, while the power coefficient is strongly negative above 200 kw, it is quite nonlinear. The origins of this nonlinearity are complex, and the authors were unable to construct a unified mathematical model which would embody the involved relationships between the clearances and power and temperature. Empirical methods were reasonably successful, however.

Tests run with the stabilizing ribs removed gave data which, when extrapolated, showed that the power at which resonance would occur was drastically reduced from something in excess of 1000 Mw to 11 Mw; however, the Mark III core would still be governed by a small prompt negative power coefficient even if all the ribs were sheared. It is believed that the prompt positive rod-bowing component was the result of conditions of fuel rod restraint characteristic of the Mark II core; conditions that are not easy to duplicate. The Mark III core has contributions to the prompt negative power coefficient from both axial and radial expansion, whereas in Mark II radial expansion contributed little. The authors indicate[8] that the prompt negative effects in Mark III would be enough

to overcome the prompt positive contribution from rod bowing, and inclusion of the stabilizing ribs eliminates the bowing. An additional delayed negative component was found in the Mark II core to be due to delayed expansions in the lower shield plate. This characteristic was eliminated in Mark III.

Sodium Reactor Experiment

Griffin and Lundholm[9] describe experimental measurements in the SRE of power coefficients and reactor parameters by oscillating a shim rod. In this report the authors present individual transfer functions of fuel, moderator, and coolant, which in turn were used to obtain the separate temperature coefficients of reactivity. For the fuel this was $-1.1 \times 10^{-5}/°F$ and $+1.3 \times 10^{-5}/°F$. The writers were able to show that "the SRE has excellent dynamic stability over all frequency ranges and operating conditions of practical importance." Measurements were conducted at substantial power levels ranging between 5.25 and 18.5 Mw in the neighborhood of 600°F. It was found that the stability was due to the large and relatively prompt negative power coefficient. The positive contribution from the graphite was sluggish and therefore easily controlled. Good agreement was generally found between measurement and theory.

Homogeneous Reactor

Rosenthal, Jaye, and Tobias[10] examined the power traces from the HRT (Homogeneous Reactor Test), when the core flow was upward. The data treated generally covered the period of operation ending in September 1958. During periods when the HRT was operated at power, large positive power excursions occurred at a rate of about two per hour. These excursions were rarely observed at power levels below 1 Mw, but above 1 Mw the frequency

of occurrence increased with power level. Negative excursions were also observed, but with much lower frequency. The power increased in most of the excursions to less than 50% above its initial value. Three excursions were noted that resulted in power increases between 100 and 150%, and once the power rose from 3.7 to 25 Mw in 8 sec. Recovery from these excursions was rapid, and no great pressure surges were noted. It is generally believed that these excursions were the consequence of uranium precipitation. At the same time, the HRT power trace displayed fluctuations of a different character that occurred at much higher frequency, 8 to 16 peaks per minute, with an amplitude usually less than 3% and on the average about 1% of the total power. These fluctuations showed no relationship with fuel feed, letdown-valve operation, or core pressure. The authors point out[10] that the response of the HRT to the added reactivity illustrates the rapid and effective compensation afforded by the large negative temperature coefficient. Even though the reactivity addition responsible for the largest excursion exceeded 2% $\Delta k/k$, the reactor was never prompt critical and the net excess reactivity was quite small. The pressure in the core never increased appreciably over that in the pressurizer.

The Homogeneous Reactor Program progress report[11] for the period December 1, 1960 to May 31, 1961 throws further light on the origin of the power fluctuations in the HRT. When the HRT was operated with downward flow through the core, the large power excursions did not occur. The frequency of the low-level power fluctuations increased markedly to about four times that observed in upward flow operation. These fluctuations were duplicated in a full-scale model of the core into which salt was injected. Using the fluctuations in salt concentration as an analog of the

temperature level, an analog computer circuit embodying the reactor kinetic equations was used to produce a fluctuating "power trace" which resembled the real reactor power trace in many important respects. Statistical analyses of the power trace were then performed with the aid of digitizing equipment and IBM-7090 codes. In this way many hours of reactor operating data could be objectively examined. The results of these studies strongly supported the hypothesis that the fluctuations originated hydrodynamically. That is, the flow pattern was constantly undergoing changes that produced slight variations in the core average temperature. These, in turn, produced the power fluctuations.

Another stability problem peculiar to aqueous circulating-fuel reactors was also briefly discussed in the progress report.[11] The fuel concentration was different in different parts of the system, and the equilibrium between these concentrations was easily upset; such changes frequently led to a transient in operating conditions. An analysis of a temperature transient on startup is described in the report.[11] It is stated that when operating at a power level just sufficient to make up heat losses, the core, the heat exchanger, and the inlet and outlet lines to the core were all at about the same temperature. On increasing the power to 5 Mw, the average core temperature remained the same, but the inlet and outlet temperatures fell below this temperature. Liquid would therefore have to be added to the core circulating system to make up for the contraction. The resulting minor disturbances to the flow streams led to a rise in temperature of about 0.7°C followed by a gradual fall over a 2-hr period to an equilibrium point 1.3°C below the original temperature. (Melvin L. Tobias)

NUCLEAR REACTOR STABILITY THEORY

Stability studies of nuclear reactors are concerned with the investigation of system power behavior following reactivity disturbances. If the influence of a disturbance significantly affects the eventual time behavior of the power, regardless of how small the reactivity disturbance may be, the system is termed unstable. In addition, instability can be associated with "threshold" values of reactivity disturbances, that is, the power level may tend to increase indefinitely with time only if the initial reactivity change exceeds a critical value. Conversely, a system is stable if, following a reactivity change, the reactor power eventually returns to a steady-state condition. The theory of stability is concerned with the determination of criteria by which to judge whether a system is stable or unstable; such criteria help ensure safe-reactor designs.

In order to examine the stability of a reactor system in a general fashion, the appropriate equations of motion are required. These involve equations such as the time-dependent Boltzmann equation, continuity, momentum and energy equations, and the equation of state. Under certain conditions these equations can be greatly simplified to facilitate stability analysis; specifically, the assumption is often made that the space-independent kinetic equations are valid and that system stability can be judged from examination of the linearized equations of motion. The validity of these and similar simplified procedures is uncertain in many cases, but information is gradually being developed which helps clarify their applicability. However, a great deal of work remains to be done before a general understanding of reactor stability will be obtained.

A discussion of approaches used in studying reactor behavior is given by Kemeny.[12] In linearized analysis, transform-function representation is very useful and convenient. If nonlinear equations are considered, use of topological methods gives insight into stability for certain cases. With the more complicated mathematical models involving many variables and nonlinearities, practical integration of the equations requires use of digital or analog computers. In such studies, specific cases are considered, rather than general cases.

In investigating reactor stability, reactor kinetics equations have to be considered, as well as equations involving reactivity feedback. The majority of work has been done using the conventional space-independent kinetics equations and a feedback mechanism that is a linear function of the power history. Even on these bases, the equations are nonlinear; however, since the linearized equations approximate the nonlinear ones for relatively small changes in power, they should give useful information if small reactivity disturbances are involved and should provide guidelines to actual-system stability.

Linear Analysis — Transform-Function Method

With the linearized equations, general information can be obtained concerning system stability by examining the transfer function of the system (either by means of a Nyquist plot, a root-locus plot, or a Bode diagram) and essentially determining the conditions under which reactor power oscillations (initiated by a reactivity disturbance) decay with increasing time. St. John[13] illustrates the various methods used to examine stability of linear reactor systems.

The conventional linearized kinetic equations can be written in the form:

$$\frac{dx}{dt} = \left(\frac{\delta k_e(t) - \beta x}{l}\right) + \sum \lambda_i c_i \quad , \tag{1}$$

$$\frac{dc_i}{dt} + \lambda_i c_i = \frac{\beta_i}{l} x \quad , \tag{2}$$

where

c_i = relative latent power associated with delayed-neutron precursors of ith group, normalized to power at time zero,

$\delta k_e(t)$ = reactivity,

l = prompt neutron lifetime,

t = time,

x = relative reactor power, normalized to its value at time zero,

β = fraction of fission neutrons which are delayed,

β_i = fraction of fission neutrons in ith group of delayed neutrons, $\sum_i \beta_i = \beta$,

λ_i = decay constant associated with ith group of delayed-neutron precursors.

Stability is determined by the time behavior of x following a reactivity disturbance. As reviewed by Soodak,[14] the stability of the above equations is dependent on the reactivity feedback mechanism associated with $\delta k_e(t)$. Specifically, let

$$\delta k_e(t) = \delta k_{ext} + \delta k_f \quad , \tag{3}$$

where

δk_{ext} = external or applied reactivity changes,

δk_f = feedback or inherent reactivity changes.

Designating W_0 as the zero-feedback transfer function associated with equations 1 and 2, and W_f as the feedback transfer function associated with δk_f, the transfer function becomes

$$W(s) = \frac{W_0(s)}{1 + W_0(s)\ W_f(s)} \quad , \tag{4}$$

where s is the Laplace-transform parameter. The condition for stability requires that none of the roots, s_j, have a positive real part, where the s_j satisfy the equation

$$1 + W_0(s_j)\ W_f(s_j) = 0 \quad . \tag{5}$$

If δk_f can be considered a linear function of power history and steady-state operation exists for $t < 0$, δk_f can be written as

$$\delta k_f(t) = -\int_0^t K(t - \tau)\ [x(\tau) - 1]\ d\tau \quad , \tag{6}$$

where K(t) is the reactivity decrease at time t following a net energy addition at time zero. Under these conditions,

$$W_f(s) = -\overline{K}(s) \quad , \tag{7}$$

where $\overline{K}(s)$ is the Laplace transform of K(t). As pointed out by Brooks,[15] $\overline{K}$ can be obtained in principle from reactor oscillator measurements (as long as the essential information contained in the Nyquist criterion corresponds to s traversing the imaginary axis). Thus, from Eqs. 4 and 7, $\overline{K}$ is given by

$$-\overline{K}(s) = \frac{1}{W(s)} - \frac{1}{W_0(s)} \quad . \tag{8}$$

Steady-state oscillator measurements can be performed at several reactor powers over a large range of impressed reactivity frequencies, and Eq. 8 can be used to predict $\overline{K}(s)$ as a function of power level. Stability as a function of power level can then be examined by means of the Nyquist criterion.

The feedback mechanism is mot always a linear function of power history; under such circumstances the reactivity feedback is often represented by several differential equations. This was done by Smith and Stenning[16] in studying the stability of nuclear rocket systems, in which reactivity effects associated with core temperature changes, gas pressure change (hydrogen gas coolant), pump characteristics, and turbine power input were considered. By linearizing the resulting equations and neglecting delayed neutrons, stability criteria were obtained. Stability was also examined using the nonlinear equations, by means of an analog-computer study. In general, the specific results obtained from the analog study were in agreement with those predicted by linearized analysis.

Other illustrations of linearized analysis are given below. For example, Agresta[17] used linearized equations in stability studies of fast reactors having conductive and convective delayed-reactivity coefficients. It was shown that appearance of roots with positive real parts corresponds to appearance of an infinite resonance in the transfer function. Most of the studies were done considering no delayed neutrons. In cases where one group of delayed neutrons was treated, its presence caused the threshold power of instability to be displaced slightly higher than when no delayed

neutrons were present. A similar analysis was performed by Miida and Suda[18] for cases where the feedback transform function has two terms, each containing a single time constant. Kochenov[19] also used linear equations to examine the stability of a water-moderated-and-cooled power reactor system, in which equations covering operation of the steam generator, turbine, and delay elements were included. The effects of various parameter values on stability under these conditions were discussed.

The use of linear equations to study reactor system stability was discussed by Gyftopoulos,[20] who points out that experimental data appear to be in good agreement with theoretical predictions based on transfer-function analysis.

Based on linearized analysis, instability is usually associated with operation of a reactor above a critical power level. As discussed by Brooks,[15] stability against small oscillations does not guarantee stability against large oscillations if the power level during an oscillation exceeds the critical power level (even neglecting nonlinear effects).

Usually stability is investigated with the reactor considered to be oscillating about some average power level; however, stability can also be studied with the average power increasing with time. Using linearized equations, Potter[21] showed that a reactor is less stable when reactivity oscillations accompany a positive stable period rather than an infinite period.

Linear Analyses — Describing-Function Method

In the conventional treatment, the kinetics equations are linearized with the power level as the dependent variable. As shown by Smets,[22] however, linearization can also be done by treating the logarithm of the

power as the variable. Under such circumstances a sinusoidal variation in δk_{ext}, with no reactivity feedback, produces a sinusoidal variation in $ln\ x$. Thus, $ln\ x$ is given by

$$ln\ x = A(\omega)\ k_1 \sin(\omega t + \phi) \quad , \tag{9}$$

where

A = parameter determined by ω,

k_1 = magnitude of δk_{ext} variation,

ω = frequency of δk_{ext} variation,

ϕ = phase angle.

Thus, if the reactivity oscillation were a sine wave which averaged zero, and the reactor were operating at low power, the neutron density would tend to increase with time. In practice this particular tendency can be negated by a small adjustment in reactivity.

Analysis of system stability using the zero-feedback kinetics equations indicated above (linear in $ln\ x$) can be performed using the describing-function method. This method applies the transfer-function method of analysis and is valid only when the harmonics generated in the nonlinear components are strongly attenuated by the linear components of the system.

To illustrate the describing function, consider first the conventional linearized kinetics equations; the equation corresponding to Eq. 9 would be

$$x = 1 + A(\omega)\ k_1 \sin(\omega t + \phi) \quad . \tag{10}$$

As defined previously, the conventional transfer function associated with Eq. 10 is $W_0(s)$. Comparison of Eqs. 9 and 10 shows the describing or "transfer" function of the low-power nonlinear system to be given by the

conventional transfer function followed by a nonlinear amplifier. Thus, with insertion of a linear feedback mechanism whose transfer function is $W_f(s)$, the system describing function analogous to Eq. 4 would be

$$W(s) = \frac{W_0(s)\ C}{1 + W_f(s)\ W_0(s)\ C} \quad , \tag{11}$$

where C is associated with a nonlinear amplifier having an exponential characteristic.

Because of C, the value of W(s) in Eq. 11 is dependent on the magnitude of k_1; for a specific case,[22] increasing k_1 caused the gain at the resonant frequency to increase and the frequency of maximum gain to increase. Thus, increasing k_1 appears to decrease the margin of stability.

The use of the describing function to study stability of the above nonlinear system was also discussed by Franz;[23] he points out that in practical reactor systems there are thermal and hydraulic time constants which tend to attenuate the harmonics generated in the nonlinear element.

Sandmeier[24] also found the stability of the nonlinear system to be dependent upon the input reactivity, and he states that experimental results in the EBR-1 and EBWR indicate a decrease in stability with increasing k_1. This was found by obtaining steady-state oscillator results and the average power as a function of frequency and amplitude of reactivity disturbance, analyzing the wave form by Fourier analysis, and treating the fundamental flux component as if it were the response from a linear system (transfer-function method).

Nonlinear Analysis

The use of linearized equations as an approximation to nonlinear systems is in general an incorrect procedure, although for specific cases this approach may give valid stability criteria. In order to examine the validity of such criteria, it is necessary to investigate the nonlinear equations.

The conventional nonlinear kinetics equations are given by

$$\frac{dx}{dt} = \left[\frac{\delta k_e(t) - \beta}{l}\right] x + \sum_i \lambda_i c_i \tag{12}$$

and Eq. 2. Using these and the linear feedback mechanism given by Eq. 6, Welton[25] obtained a general stability criterion. It was found that a sufficient condition for stability is

$$\int_0^\infty d\tau \left[\frac{-K(\tau)}{K(0)}\right] \omega \sin \omega\tau > 0 \quad , \tag{13}$$

where ω is the frequency of ln x and K has its previous meaning. Gyftopoulos and Devooght[26] give a mathematical proof of Welton's criterion and use the results to evaluate linear stability criteria for two cases. In one, stability was predicted by both the linear and nonlinear criteria for some postulated operating conditions; in the other, the linear stability criteria were less restrictive than the nonlinear criteria.

Recently Akcasu and Dalfes[27] performed an analytical study of nonlinear stability criteria for stationary reactors and obtained a general form of solution applicable to both linear and nonlinear feedback mechanisms.

Specific cases have been calculated in which the stability criteria based on the conventional linearized equations have been compared with results obtained with the conventional nonlinear equations. For example, Chernick[28] has studied the dynamics of a xenon-controlled reactor and found conditions where the nonlinear equations gave stable oscillations, while the linear equations predicted divergent oscillations; conversely, conditions were also noted where the linearized equations predicted stability, while the nonlinear equations predicted instability. Gyftopoulos[29] also compared results obtained from the nonlinear reactor equations with those from linearized analysis. In this study, geometric theory was used to find the stability properties of the nonlinear system. He found that the linearized equations do not necessarily contain all the information required to predict the stability conditions associated with relatively large power changes. They are useful, however, within a limited range of amplitude variation of the dependent variables.

Although often used, the conventional nonlinear reactor equations do not apply in general. Gyftopoulos and Devooght[30] have recently studied the range of validity of the nonlinear equations. If the spatial modes are higher than the fundamental decay quickly following a reactivity disturbance, the nonlinear form of the equations should be correct. However, the very condition necessitating the nonlinear equations, namely, a large reactivity disturbance, tends to activate the higher-order spatial modes. The specific range of validity depends on the particular case at hand.

Under conditions where spatial oscillations are important, a complex time variation of the power distribution can occur. Spatial oscillations

tend to be initiated more readily in large reactors than in small reactors, and so stability studies of physically large reactors should consider such variations. Some of the difficulties associated with stability analyses of large reactors are discussed by Fonda,[31] who notes that the mathematical models used in formulating kinetics problems are considerably less detailed than those used in statics calculations. Hitchcock[32] gives a detailed method for studying stability under conditions of spatial power variation. The space and time portions are assumed to be separable, and model analysis is used to determine flux shape. The variables are expanded in suitable modes and integration is performed; time dependent calculations are then made for the different model amplitudes and coupling parameters. Little in the way of general results have been obtained in this area.

Influence of Delayed Neutrons and Heat Diffusion on Stability

The effect of delayed neutrons on reactor stability has been a subject of investigation for many years. With no delayed neutrons and a negative temperature coefficient of reactivity, a series of undamped power oscillations follows a step change in reactivity; addition of a constant source of delayed neutrons leads to a single power excursion. Also, as discussed by Soodak,[14] a constant source of delayed neutrons and a ramp reactivity addition leads to a damped sequence of power oscillations.

Addition of the decay characteristics of delayed-neutron precursors makes analysis more difficult. Again with a negative temperature coefficient of reactivity, it can be shown that for the conventional linearized equations, delayed neutrons increase the stability of the system. It appears that this is also true for the nonlinear system; Akcasu and Dalfes[27] have shown that

the effect of delayed neutrons is to make the stability condition slightly less restrictive. For the nonlinear system, Gyftopoulos and Devooght[33] have also shown that if the system is stable without delayed neutrons, it is stable with delayed neutrons.

If complex reactivity feedback mechanisms exist in a reactor system, it is possible that delayed neutrons may actually decrease stability. For example, Welton[25] considers a hydrodynamic system in which the delayed neutrons may supply a feedback mechanism exciting a hydrodynamic oscillation which, in turn, influences reactivity. In the case considered, friction was required to compensate for this anti-damping tendency. Thus, the influence of delayed neutrons upon system stability appears dependent upon the physical reactor system, inasmuch as delayed neutrons can damp one kind of motion and excite another.

There are a number of other factors which can be looked at individually in restricted studies and investigated with regard to their influence on reactor stability. One of these is the diffusion of energy from the point of generation. For this case, Ergen[34] showed that the effect of heat diffusion was to increase system stability.

Status

In actual reactor systems involving many variables, stability analysis is normally performed using linearized equations and the transform function method, in combination with computer studies of the nonlinear equations. This approach will undoubtedly continue. Based on the work of Welton, Akcasu, Smets, Gyftopoulos, and others, a much better understanding of nonlinear kinetics has been obtained, and further work on nonlinear feedback mechanisms, such as that of Akcasu and Dalfes, should provide

additional insight. The associated results should be used wherever possible to complement, expand, or correct results from other stability studies.

Investigations concerning the effects of spatial oscillations on stability have been initiated, but a great deal of work remains before these effects are understood in a general fashion. Future studies should emphasize areas involving the effects of spatial oscillations and non-linear feedback mechanisms on reactor response; results from such studies should permit better interpretation of experimental data. (P. R. Kasten)

U^{235} FISSION-PRODUCT HEATING

In the design and operation of nuclear reactors, consideration must be given to the heat generated in the various components following a shutdown or a reduction in the operating power level. This is necessary in the sizing of emergency cooling systems, in the evaluation of the potential hazards, and in the development of necessary protective devices. As mentioned earlier by Binford,[35] approximately 6.9% of the U^{235} fission energy results from the decay of the fission products produced during the fission process. For the computation of the reactor shutdown heating rates, knowledge of the following parameters as a function of time is required:

1. The energy distribution and the energy release rate of the fission-product gamma rays.

2. The energy release rate of the fission-product beta rays.

A number of investigations have been made to provide this required information. Summaries of the results of these investigations have been prepared by Keagy,[37] Knab and Putnam,[36] and Avery et al.[38] Keagy indicates that the early relation developed by Way and Wigner[39] for predicting beta and gamma afterheat production rates probably is accurate within a factor of 2 for shutdown times ranging from 10 sec to several weeks. The more recent investigations were for the purpose of being able to predict these heating rates more accurately, particularly for shutdown times of less than one day. (Experimental determinations of the short-lived fission products are difficult to obtain because of the time required to perform the necessary chemcial separations.)

All three summaries mentioned above indicate that the most reliable data for the calculation of the beta and gamma decay energy and the gamma energy spectrum for shutdown times greater than 1 hr are those presented by Perkins and King.[40] For decay times in the range of 1 to 1550 sec, the data obtained by Zobel and Love[41] and by Zobel et al.[42] appear to be the most reliable at the present time. Knabe and Putnam combined the data presented by Perkins and King with those obtained by Zobel et al. to obtain the decay energies and the gamma energy spectrum for shutdown times between 1550 sec and 1 hr. The Knabe and Putnam results[36] are presented in tabular and graphical forms and formulas are given with which these results can be applied to a reactor. Recently, Zigmon and Mackin[43] made experimental measurements on low-energy gammas given off during the early time decay of U^{235} fission products, and they made refinements to the gamma energy and spectrum data presented by Knabe and Putman for times from 1 sec to 5 hr after an instantaneous thermal-neutron fission of U^{235}. For completeness, mention should be made of the work of Stehn and Clancy,[44] which Avery et al.[38] suggested for the use of calculating the afterheat in an infinite reactor. Their results appear to be in fair agreement with those presented by Perkins and King. Speigler[45] also computed the afterheating rates in a reactor using the data of Perkins and King, with some refinements. His results are presented in the form of charts for a reactor which had operated for an infinte length of time.

In addition to the contribution by the fission-product betas and gammas, other sources such as activation products must also be considered in calculating the shutdown heat-generation rates. For example, Al^{28} in the High Flux Isotope Reactor (HFIR) fuel region makes a significant

addition to the heating rates during the first 1000 sec of shutdown.[46,47] Another source of shutdown heat somewhat unique to the HFIR is that from the spontaneous fission of Cf^{252} and Cf^{254} in the reactor's target rods.[46] This source could increase the after-shutdown heat-generation rate in the HFIR target rods 50% for cooling times of 10^5 sec.

The delayed gammas can also cause heating in the reactor by an indirect mechanism such as increasing the rate of oxidation of graphite in the Experimental Gas-Cooled Reactor (EGCR) in case of a rupture of a gas coolant line.[48] Such a reaction possibly could be initiated by the after shutdown heat in the reactor and be accelerated by the gamma rays. Of course, the heat resulting from the graphite-oxygen reaction would also add to the afterheat produced in this reactor.

In some reactors, such as the ORNL Graphite Reactor and the Low Intensity Test Reactor (LITR), the intrinsic heat removal mechanisms are sufficient to remove the afterheat produced in these reactors.[35] In other reactors, however, particularly the ones which normally operate with higher heat fluxes, emergency cooling equipment, such as auxiliary pumps or battery-powered pony motors attached to the main circulating pumps, must be provided. As mentioned by Binford,[35] it may be possible to take advantage of the fact that the pumping equipment will continue to operate for a short time after a power outage. Methods of estimating flow rates after an electrical power outage have been presented by Burgreen[49] and Love.[50]

For very short shutdown times, heat resulting from delayed neutrons should be considered in addition to that produced from the betas and gammas. The standard kinetic equations[51] could be used in estimating the after shutdown neutron fluxes from which the heating rates could be estimated.

These equations are somewhat difficult to use, however, and Zubarev and Sokolov[52] suggested the use of an empirical relation which will give sufficiently accurate results for all the negative reactivity values of general interest.

There is sufficient information available to predict the shutdown heating rates in a reactor with a fair degree of accuracy, except for very short cooling times. This information should permit the designer to size emergency cooling equipment, if required, for most reactors with an appreciable amount of confidence. (H. A. McLain)

References

1. A. H. Spano and L. A. Stephan, Experimental Investigation of the Kinetic Behavior of a 4%-Enriched UO_2 Reactor in SPERT I, Trans. Am. Nuclear Soc., 4(2): 312-313 (November 1961).

2. F. Schroeder, ed., Quarterly Technical Report, SPERT Project, January, February, March, 1961, pp. 3-6, USAEC Report IDO-16693, Idaho Operations Office, June 30, 1961.

3. F. Schroeder, ed., Quarterly Technical Report, SPERT Project, April, May, June, 1961, pp. 3-13, USAEC Report IDO-16716, Idaho Operations Office, September 15, 1961.

4. F. Schroeder, ed., Quarterly Technical Report, SPERT Project, July, August, September, 1961, pp. 3-7, USAEC Report IDO-16726, Idaho Operations Office (to be published).

5. R. M. Ball et al., Marty Critical Experiments Summary of 4% Enriched UO_2 Cores Studied for NMSR, USAEC Report BAW-1216, Babcock & Wilcox Company, May 1961.

6. L. E. Glasgow, Sodium Reactor Experiment Operating Experience, Trans. Am. Nuclear. Soc., 4(2): 202-203 (November 1961).

7. C. W. Griffin, Analysis of the Sodium Reactor Experiment Prompt Power Coefficient, Trans. Am. Nuclear Soc., 4(2): 314 (November 1961).

8. R. R. Smith, J. F. Boland, F. D. McGinnis, M. Novick, and F. W. Thalgott, Instability Studies with EBR-I, Mark III, USAEC Report ANL-6266, Argonne National Laboratory, December 1960.

9. C. W. Griffin and J. G. Lundholm, Jr., Measurements of the SRE Power Coefficients and Reactor Parameters Utilizing the Oscillation Techniques, USAEC Report NAA-SR-3763, North American Aviation, Inc., November 1, 1960.

10. M. W. Rosenthal, S. Jaye, and M. Tobias, Power Excursions in the HRT, USAEC Report ORNL-2798, Oak Ridge National Laboratory, February 15, 1960.

11. R. B. Briggs, Homogeneous Reactor Program Progress Report for Period from December 1, 1960 to May 31, 1961, USAEC Report ORNL-3167, Oak Ridge National Laboratory, August 30, 1961.

12. L. G. Kemeny, Reactivity Fluctuations in Reactor Systems, Nuclear Engineering, 6(60): 208-213 (May 1961).

13. D. S. St. John, Reactor Stability, USAEC Report DP-517, E. I. du Pont de Nemours & Co., September 1960.

14. H. Soodak, Problems of Reactor Kinetics, in Proceedings of Symposia in Applied Mathematics, Vol. XI – Nuclear Reactor Theory, pp. 233–255, Am. Math. Soc., Providence, Rhode Island (1961).

15. H. Brooks, Temperature Coefficients and Stability, ibid, pp. 289–308.

16. H. P. Smith, Jr., and A. H. Stenning, Open Loop Stability and Response of Nuclear Rocket Engines, Nuclear Sci. and Eng., 11: 76-84 (1961).

17. J. Agresta, Stability Studies on Fast Reactors with Delayed Reactivity Coefficients, USAEC Report NDA-2147-7, Nuclear Development Corporation of America, August 15, 1961.

18. J. Miida and N. Suda, General Stability Criteria for Nuclear Reactor with Two Feedback Paths of Single Time Constant, Nuclear Sci. and Eng., 11: 55-60 (1961).

19. A. S. Kochenov, The Stability of a Nuclear Power Plant, Reactor Science, 12(4): 194-200 (1960).

20. E. P. Gyftopoulos, Transfer Function Representation of Nuclear Power Plants, in Proceedings of Conference on Transfer Function Measurement and Reactor Stability Analysis, May 2, 1960, USAEC Report ANL-6205, p. 18, Argonne National Laboratory.

21. R. Potter, Transfer Function of a Steadily Diverging Reactor, J. Nuclear Energy, 6: 291-299 (1958).

22. H. B. Smets, The Describing Function of Nuclear Reactors, IRE Transactions on Nuclear Science, NS-6(4): 8-12 (1959).

23. J. P. Franz, Reactor Transfer Functions, in Proceedings of Conference on Transfer Function Measurement and Reactor Stability Analysis, May 2-3, 1960, p. 9, USAEC Report ANL-6205, Argonne National Laboratory.

24. H. A. Sandmeier, Nonlinear Treatment of Large Perturbation in Power Reactor Stability, Nuclear Sci. and Eng., 6: 85-92 (1959).

25. T. A. Welton, System Kinetics, in Proceedings of Symposia in Applied Mathematics, Vol. XI – Nuclear Reactor Theory, pp. 309-326, Am. Math. Soc., Providence, R. I. (1961).

26. E. P. Gyftopoulos and J. Devooght, Boundedness and Stability in Nonlinear Reactor Dynamics, Nuclear Sci. and Eng., 7: 533-540 (1960).

27. A. Z. Akcasu and A. Dalfes, A Study of Nonlinear Reactor Dynamics, Nuclear Sci. and Eng., 8: 89-94 (1960).

28. J. Chernick, Dynamics of a Xenon-Controlled Reactor, Nuclear Sci. and Eng., 8: 233-243 (1960).

29. E. P. Gyftopoulos, Applications of Geometric Theory to Nonlinear Reactor Dynamics, Nuclear Sci. and Eng., 10: 254-268 (1961).

30. E. P. Gyftopoulos and J. Devooght, On the Range of Validity of Nonlinear Reactor Dynamics, Nuclear Sci. and Eng., 10(4): 372-376 (1961).

31. A. Fonda, The Instability Problem in Gas-Cooled Reactors, Nuclear Power, 5(49): 93-96 (May 1960).

32. A. Hitchcock, Nuclear Reactor Stability, Temple Press Ltd., London, (1960).

33. E. P. Gyftopoulos and J. Devooght, Effect of Delayed Neutrons on Nonlinear Reactor Stability, Nuclear Science and Engineering, 8: 244-250 (1960).

34. W. K. Ergen, Rearrangement Inequalities and Nonlinear Stability Criteria, in Proceedings of Symposia in Applied Math., XI – Nuclear Reactor Theory, pp. 327-329, Am. Math Soc., Providence, R. I. (1961).

35. F. T. Binford, Decay Heat Removal from Research Reactors, Nuclear Safety, 1(3): 34-35, (March 1960).

36. W. E. Knabe and G. E. Putnam, The Activity of the Fission Products of U-235, USAEC Report APEX-448, General Electric Co., October 31, 1958.

37. W. R. Keagy, Evaluation of Fission Product Inventory, Activities, and Decay Power Levels in Nuclear Reactors, Neue Technik, 2(5): 41-47, (May 1960).

38. A. F. Avery, D. E. Bendall, J. Butler, and K. T. Spinney, Methods of Calculations for Use in the Design of Shields for Power Reactors, British Atomic Energy Authority Report AERE R-3216, February 1960.

39. K. Way and E. P. Wigner, The Rate of Decay of Fission Products, Physical Review, 73: 1318-1330 (1948).

40. J. F. Perkins and R. W. King, Energy Release from the Decay of Fission Products, Nuclear Sci. and Eng., 3: 726-746, (1958).

41. W. Zobel and T. A. Love, Time and Energy Spectra of Fission-Product Gamma Rays Measured at Short Times After Uranium Sample Irradiations. Part III of Applied Nuclear Physics Division Annual Report, USAEC Report ORNL-2081, Oak Ridge National Laboratory, pp. 95-101, November 5, 1956.

42. W. Zobel, T. A. Love, G. M. Estabrook, and R. W. Peele, Characteristics of Fission-Product Gamma Rays Emitted from 1 to 1800 sec After Thermal Fission. Part 4 of Neutron Physics Division Annual Progress Report, USAEC Report ORNL-2609, pp. 50-51, Oak Ridge National Laboratory, February 1958.

43. P. Zigman and J. Mackin, Early Time Decay of Fission Product Mixture – II, γ Energy Release and Ionization Rates Following Thermal Neutron Fission of U-235, Health Physics, 5: 79-84 (1961).

44. J. R. Stehn and E. F. Clancy, Fission Product Radioactivity and Heat Generation, Second United Nations International Conference on the Peaceful Uses of Atomic Energy, 13: 49-54 (1958).

45. P. Speigler, Energy Release from the Decay of Fission Products, USAEC Report NAA-SR-Memo-4126, North American Aviation, Inc., July 31, 1959.

46. N. Hilvety, Gamma and Beta Heat Generation Rates in the HFIR Core, USAEC Report ORNL CF-60-4-110, Oak Ridge National Laboratory, April 29, 1960.

47. H. A. McLain, After Shutdown Heating in the HFIR, USAEC Report ORNL CF-60-12-118, Oak Ridge National Laboratory, December 29, 1960.

48. M. H. Fontana, Status Report on Oxidation Analyses for the EGCR, unpublished report, Oak Ridge National Laboratory, June 15, 1961.

49. D. Burgreen, Flow Coastdown in a Loop After Pumping Power Cutoff, Nuclear Sci. and Eng., 6: 306-312, (1959).

50. W. J. Love, Hydraulic System Flow Decay Relations During Loss of External Power, USAEC Report HW-62846, Hanford Atomic Products Operation, February 11, 1960.

51. S. Glasstone, and M. Edlund, The Elements of Nuclear Reactor Theory, Van Nostrand, New York, (1952).

52. T. N. Zubarev and A. K. Sokolov, Calculation of the Heat Release in a Shut-Down Reactor, J. Nucl. Energy, Part A: Reactor Science, 13: 72 (1960).

III. CONTROL AND INSTRUMENTATION

CRITICALITY EXCURSION DETECTION IN PROCESSING PLANTS

The detection of accidental critical nuclear reactions has been of primary importance in the field of personnel protection since the advent of large-scale atomic energy programs. Although much work has been done in the past on instrumentation systems to accomplish this end, this work has not been given nearly as much attention as the before and after phases of the problem, that is, nuclear accident prevention and procedures for coping with the effects of a radiation excursion. This lack of comprehensive assessment of criticality monitoring in nuclear processing facilities has been remedied in the past few years, however. The Y-12 incident[1] in mid-1958, followed by accidental excursions at Los Alamos[2] and the Idaho Chemical Processing Plant[3] within a period of 16 months, has spurred action on the detection problem.

Recent Developments

Notable work has been done by the Area Monitoring Subcommittee of the Radiation Emergencies Steering Committee, which was formed after the Y-12 accident and was composed of representatives from the three Union Carbide Nuclear Company facilities at Oak Ridge, Tennessee, the Union Carbide plant at Paducah, Kentucky, and the Goodyear Atomic Corporation works at Portsmouth, Ohio. The subcommittee was charged with the responsibility for redefining the criteria for an alarm and monitoring system suitable for use in the AEC production plants involved. Consequently, intensive reviews of existing monitoring systems within the affected facilities and at other AEC operations were initiated.

It was evident from these investigations that all systems then in operation were of a similar nature. They utilized gamma radiation rate-measuring instruments installed proximate to areas considered as possible accident points. The radiation-rate detectors at the various sites employing monitoring systems were of diverse operational design; e.g., among others were the ionization-chamber-type Argon-Gammagraph,[4] the Victoreen Model 350 Geiger tube instrument,[5] and the Rocky Flats scintillation detector.[6] The essential purpose of the monitors was to immediately indicate the need to evacuate personnel in the event of an excursion, especially those people who may have been subjected to an appreciable radiation dose from the reaction, in order to protect them from exposure to fission-product radiation or subsequent bursts. The alarm levels were rather low (0.1 to 10 mr/hr), having been selected so that an area would be evacuated if the dose rate anywhere in the area exceeded a minimum allowable amount.

It was the subcommittee's conclusion that these systems could not provide the most effective surveillance of large area nuclear processing plants. The response of rate instruments to critical reactions is somewhat unpredictable because the time constant of the usual instrument of this type is not short enough to allow presentation of the true radiation rates. This requires sensitive alarm-rate settings that will evacuate much larger areas of an installation than is desirable, since instruments far from the source of the excursion and out of the danger area will respond. Unnecessary or too hasty evacuation may be dangerous in continuous production plants where operator action at a safe distance could conceivably avert further damage. Furthermore, following evacuation, the processing of large numbers of people from safe areas can seriously complicate

the finding of those persons from the vicinity of the excursion who received significant radiation exposures.

The question of a minimum expected excursion from a processing plant accident has received considerable attention. A number of experienced persons have expressed the belief that a lower limit for the size of an accidental radiation burst will be on the order of 10^{16} fissions. This is borne out by reported studies[7] that the "average accident" in uranium processing operations is expected to yield between 4×10^{16} and 4×10^{17} fissions in the initial prompt-critical burst. It is felt that the addition of reactivity to a subcritical configuration in an accidental manner will be at such a rate as to preclude a smaller energy release within the first minute, or less, of the criticality.

The Radiation Emergencies Steering Committee report[8] lists the design criteria developed to define a dose-sensitive radiation alarm system suitable for production plants. Numbered among the requirements are a system detection sensitivity sufficient to sound an alarm in the event of a burst of radiation from a minimum of 10^{16} fissions, response of the monitors to both gamma rays and neutrons, locked in audible and visual alarms, maximum reliability, use of emergency electrical power, and display of the alarmed monitors in a central control facility. The criteria are being followed in the radiation alarm systems presently being installed in the gaseous diffusion plants at Oak Ridge, Tenn., Paducah, Ky., and Portsmouth, Ohio.

The minimal excursion of a radiation burst from a critical incident expressed in terms of number of fissions may be predicted; however, the time duration cannot. The time may be from several microseconds to several

seconds for the initial burst. Under these conditions, the reaction of rate-sensitive instruments cannot be determined as closely as that of integrating instruments; in addition, the "rate" in such an instrument has no predictable relation to the exposure received by an individual in a critical accident. In view of the shortcomings of rate-measurement monitoring devices, the direct relationship between burst size and dose, and the biological significance of radiation dosage, it was concluded that dose-sensitive instruments will provide a more satisfactory criticality detection and alarm system than dose-rate instruments.

It is interesting to note that the British have adopted, quite independently, the concept of using dose-sensitive criticality-detection instruments in their plants at Capenhurst and Windscale. Wachter[9] indicates that this system is very similar to that described below, with the exception that a discontinuous timer-controlled method of recharging the monitor ion chambers is utilized to annul the effects of background radiation. Such a method is not considered as safe as continuously rate-compensated excursion detectors because of the instrument inactivity during the recharging period. Although such periods can be made quite small, they become significant when it is considered that a prompt burst can occur within a few microseconds.

Dose Type of Monitor

The ideal dose-sensitive radiation-excursion detector embodies two mutually exclusive attributes, perfect integration and insensitivity to background radiation. In practice, the time constant of the instrument must be much larger than the longest expected burst if good integration

is to be obtained. These requirements were met by the Radiation Alarm Instrument[10] developed through the mutual efforts of Union Carbide Nuclear Company and Goodyear Atomic Corporation engineers. A unique method of continuously compensating the integrating ionization chamber of the instrument for incident radiation below a predetermined threshold rate enables the monitor to function in normally expected radiation fields with undiminished sensitivity to radiation bursts.

The rate-compensated dose-type excursion detector includes a constant current source supplying a radiation detector, which is shunted by both a unidirectional current integrator and a compensating element possessing a current-voltage relationship with an infinite slope over a portion of its characteristic. In a background radiation field, the detector current, which is proportional to the radiation rate, and the compensating element current automatically apportion themselves to maintain a nearly constant potential across the integrator at a value to prevent integration. However, in the presence of radiation rates exceeding a threshold limit, the compensating element no longer is able to maintain a constant output voltage or supply the additional current demanded by the rate detector. This current is provided by the integrator. Thereby, the instrument performs as a dose-sensitive detector in high radiation fields.

The key component in the rate compensation scheme is the compensating element. In practice there are a number of devices having operating characteristics that approach the ideal. Examples are a Zener diode, a normal diode with low forward resistance, and an electrometer tube operating in the high, positive, grid-current region of its characteristic. The

latter was coupled with an ionization chamber, which doubled as the rate detector and integrator in the final instrument configuration.

Reliability of performance was paramount in the over-all Radiation Alarm Instrument design and in the choice of individual components. Noteworthy among the steps taken to ensure a high degree of dependability are: selection of the highly reliable ionization chamber as the radiation detector element; the use of a two-out-of-three unit alarm matrix; provision of an instrument cluster trouble alarm in the event of failure of a single unit from practically any cause; supplying power to each instrument from its own trickle-charged nickel-cadmium battery pack, which is capable of at least 8 hr of operation in the absence of line power; and the use of degeneration in the electronic circuitry of the instrument to reduce sensitivity drift with component aging.

Prototype instruments were subjected to fast radiation bursts of approximately 10^{16} fissions from the Godiva II reactor[11] at Los Alamos to study the instrument response. As a result of these tests the instrument alarm sensitivities were independently adjusted to the equivalent of either 10 mr Co^{60} gamma or an estimated 6 mrem (RBE = 2) neutron radiation. The threshold level, below which the instrument does not accumulate background radiation, is approximately 50 mr/hr. Instruments exhibiting these sensitivities can be expected on the average to cover as much as a 225-ft radial process area in the event of a minimum burst, no matter what the nature of the criticality, be it aqueous, dry chemical, or metal.

Rate Type of Systems

Criticality monitoring in nuclear processing facilities that are small or where the evacuation of the entire plant will create no serious problems can be satisfactorily accomplished by the use of radiation-rate detectors. Independent tests of various types of rate detectors on the Godiva II reactor by personnel from the Y-12 Plant[12] and Hanford[13] have shown that radial in-plant coverage of 400 ft or more may be assigned to a single instrument with a low (3 mr/hr or less) alarm-rate setting. Systems operating in this fashion are much more susceptible to false alarms, however, than systems composed of rate-compensated integrating instruments.

Several planned excursion-alarm systems were discussed at the Hanford Criticality Instrumentation Meeting[14] in 1959. Among these was a rate type of system designed to provide full coverage of the 400- by 800-ft plutonium-processing area involved in the 1958 accident at Los Alamos. This system has since been placed in operation.[15] It consists of 15 modified Victoreen Instrument Company, Model 712 Series, Remote Area Monitors and 81 alarm horns interconnected through a single, remotely located, control console. The ion-chamber monitors are gamma-sensitive only. Two or more detectors must be actuated before a general evacuation alarm is sounded.

A hybrid system has been described by J. N. Wilson, Savannah River Plant, which includes both rate and integrating action.[16] The basic criticality monitor developed for the system is a rate detector with a Neher-White ionization chamber and a contact-meter relay alarm device. The instrument is simple, compact, reliable, and shows positiveness of

operation in high radiation fields. The unit is operated from mercury cells and the alarm system is powered by a storage battery under trickle charge. The excursion alarm system utilizes the instrument in two fashions. The first unit is adjusted to alarm at 1 r/hr of gamma radiation and is intended to initiate evacuation of a local area, say within 100 ft to 200 ft of the accident. The second instrument is an auxiliary box set to alarm if a dose rate of 10 r/hr persists for at least 5 sec, in which case the entire building or area is evacuated. By using a relatively high primary alarm rate and the rate-time dependent auxiliary instrument, the Savannah River system is somewhat akin to the dose type of system.

Conclusion

The papers which have been cited here are thought to present a comprehensive picture of past and contemporary information on criticality monitoring in the nuclear-processing industries. It must be kept in mind that the systems and philosophies examined are intended solely to detect a radiation excursion and to provide an evacuation warning subsequent to the accident. The instrumentation discussed cannot anticipate a prompt critical burst and sound an alarm for prior clearing of the danger area. Proper process system design, with due regard for nuclear safety practices, and strict administrative control of operating procedures are mandatory for accident prevention. Additionally, intensive emergency training is required to fully realize the value of an excursion detection system. (D. M. Papke)

REACTOR NEUTRON DETECTION

Neutron detectors comprise one of the more important types of sensors in reactor safety and control. Since a reactor may operate at power levels from a few microwatts to several megawatts, a rapid and accurate means of sensing the neutron population over a wide range is essential in order to maintain surveillance over the reactor kinetic behavior. Counting techniques are usually employed at very low power levels, and excellent discrimination against gamma photons is possible with pulse chambers. Ion chambers provide a continuous signal at higher values of neutron flux; however, gamma compensation is necessary in order to obtain a signal proportional to neutrons alone in the presence of a strong gamma field.

At the Sixth Tripartite Instrumentation Conference[17] at Chalk River, April 1959, papers pertinent to neutron monitoring were presented that covered counting and integrating ion chambers and the usual neutron-detection materials, B^{10} and U^{235}. These papers are reviewed here, together with a number of current documents on the same subject.

Compensated and Uncompensated Ionization Chambers

The development of three direct-current (d-c) ionization chambers was reported by Lones[18] of BNL. All three chambers are neutron sensitive, and one of the three is also used uncoated as a gamma chamber. The neutron-reacting material is B^{10}. The chambers are constructed with aluminum walls and standard steatite insulators with Kovar-seal feedthroughs. A differential, or compensated, chamber that departs from common design was one of the three detectors described by Lones. In this chamber

the collector electrode is surrounded by a cylinder axially split into halves. One of the half-cylinders, coated with B^{10}, is connected to the positive terminal of the high-voltage supply. The other uncoated half-cylinder is connected to the negative terminal of the supply voltage. The proposed method of compensation is to rotate the chamber in its hole until the gamma signal goes to zero. Also, by rotating the chamber, the sign of the direct current can be changed. Reversal of the electrical polarity of the coated and uncoated half-cylinders will also accomplish this.

The second and third detectors that were described consist of a boron-coated chamber (uncompensated for gamma rays) and its uncoated twin that can be used as a pair to obtain gamma compensation. The two signal electrodes are connected, and voltages of the opposite polarity are impressed on the high-voltage electrodes. Compensation is achieved by moving one chamber with respect to the other.

In a paper by Henderson,[19] boron-coated ion chambers designed at AERE, Harwell, were described. These chambers are simple in design. For the gamma-compensated chamber, concentric right cylinders, open at both ends, serve as electrodes. For the uncompensated chamber, the electrodes are concentric right cylinders closed with hemispherical end caps at the sensitive-volume end. The chambers are 3.5 in. in diameter and 13 in. long. The construction materials are ceramics, aluminum, and some steel. The lead-through insulators are commercially available ceramic-to-metal seals. The compensated chamber is of the fixed variety but is adjusted during manufacture for a 30:1 compensation in a uniform gamma field to assure a better than 10:1 compensation in actual use. The chambers are

filled with hydrogen gas to give an optimum neutron-to-gamma ratio. With a higher filling-gas pressure to prevent collapse of the outer aluminum case, the chambers may be operated at a temperature up to 300°C. Some parameters of these chambers are listed in Table III-1.

Managan[20] of ANL reported that one goal is to design high-current output chambers with saturation voltages from 100 to 300 v. Experimental data were presented on boron thickness vs current output with filling gases of argon, nitrogen, and hydrogen; saturation voltages for different current densities; and saturation voltages for the three gases at various pressures. Some parallel-plate chambers that make use of these data were described. The results showed that an uncompensated chamber having a parallel-plate spacing of 0.093 in. and a gas filling of nitrogen at 1 atm will collect 95% saturation current at 150 v for 100-μa current output or at 300 v for 1-ma output.

Sani and Zanchi[21] have discussed some of the problems of neutron detectors in reactor applications. Data for specific BF_3 counters are plotted that give gas amplification, count rate vs counter voltage at different pulse-height settings, count rate vs pulse height at different counter voltages, and characteristics for different amplifier time constants. A liberal translation of the summary of their paper follows:

> "One needs to know the magnitude of the neutron flux for the control of a nuclear reactor at the first phase of startup and at the moment of shutdown. For this reason, it is indispensable to have a system with the capacity to discriminate between neutrons and gamma radiation.... The boron trifluoride proportional counter responds excellently to this requirement when connected to a scaler and amplifier with pulse height discriminator.... Besides having a high sensitivity for neutrons when used with an amplifier of short time resolution, the boron trifluoride counter works well at a maximum rate of 10^5 counts/sec and at the same time discriminates well against the pulses due to a gamma dose of some hundreds of roentgens per hour."

Table III-1. Parameters of AERE Ionization Chambers

	Type RC-6[a]		Type RC-7[b]	
Hydrogen filling-gas pressure, cm Hg	25	150	25	150
Coated area, cm^2	640	640	450	450
Coat[c] thickness, mg/cm^2	0.4	0.4	0.4	0.4
Sensitivity per unit neutron flux, amp	8.4×10^{-15}	2.7×10^{-15}	6.4×10^{-15}	1.9×10^{-14}
Gamma sensitivity, amp/r·hr	7.6×10^{-13}	4.5×10^{-12}	5.0×10^{-13}	3.0×10^{-12}
Voltage for 90% saturation at 10^{-4} amp, v	23	125	52	300

[a]Compensated ion chamber.
[b]Uncompensated ion chamber.
[c]Enriched boron coating.

Fission Chambers

An interesting paper by Salmon[22] from AERE, Harwell, described three fission counters (pulse devices) that are designed to operate at a temperature of 500°C or higher. The electrodes are stainless steel, and the insulators are alumina. The filling gas is argon at 50 psi. These devices have 15-ft rigid extensions and are for flux scanning and reactor control (see Table III-2 for chamber characteristics). Processing and cleaning techniques required to assure that the detectors will work at high temperature were discussed.

A counting system employing a fission counter that is capable of processing input pulses at an average rate which exceeds 3×10^5 events/sec was developed[23] in order to monitor lower neutron levels for startup of

Table III-2. Characteristics of AERE Fission Counters

Chamber diameter, in.	1.75[a]	0.4[b]	0.125[c]
Coated[d] area, cm^2	500	8	0.8
Coating thickness, mg/cm^2	1	1	1
Sensitivity per unit neutron flux, counts/sec	0.5	8×10^{-3}	5×10^{-3}
Operating temperature, °C	500	500	600
Operating voltage, v	400	400	300

[a]For reactor control.

[b]For reactor control and flux scanning; a chamber of this type has been operated successfully in a laboratory for eight months at 500°C and has been in low-temperature use for several years.

[c]For flux scanning; this type of chamber was in an experimental stage at the time of the report.

[d]All chambers coated with enriched U^{235}.

the EBR-II reactor than could be done by current-measuring devices (compensated ion chambers). In order to obtain a decade overlap with the compensated chamber, the upper operating level was set at 2.2×10^5 neutrons/cm^2·sec. At the lower operating level, the detector had to be capable of detecting a flux of 2 neutrons/cm^2·sec in a 10^3-r/hr background.

The detector is a fission counter (Westinghouse WX4245) which is coated with 1 mg/cm^2 of U_3O_8 enriched to 90% U^{235}. The coated area is 1020 cm^2; the electrode separation is 0.15 in.; the filling gas is an argon-nitrogen mixture at 1 atm; and the electrical capacity is 200 μμf. The sensitivity per unit neutron flux is 0.45 counts/sec. Thus at the upper level (2.2×10^5 neutrons/cm^2·sec) the detector output is approximately

10^5 counts/sec. Measurements made on rise time of the detector showed a charge collection time of 7×10^{-8} sec.

The system design allows a 0.5-μsec time span for detection, amplification, and discrimination, and an additional 0.5 μsec for recovery. This allows 1 μsec/event. The preamplifier-amplifier gain is in excess of 5×10^4. The rise time of the cascaded preamplifier and amplifier is less than 0.1 μsec. The amplifier is not recommended for severe overload.

Results of tests showed that the actual count rate decreased by 10% from the true count rate at 3.3×10^5 counts/sec and that the measured losses of neutron sensitivity as a function of gamma-ray background were the following (losses are based on the sensitivity obtained in gamma fields of 10^4 r/hr or less):

1. At 10^4 r/hr no loss of sensitivity was encountered.
2. At 10^5 r/hr the sensitivity decreased to 60%.
3. At 10^6 r/hr the sensitivity decreased to 10%.

These losses come about from the necessity to increase the discriminator setting to a higher level as the gamma-ray intensity increases. (This is the problem of pulse pileup.)

Self-Calibrating Thermal-Neutron Counter

Preliminary work has been done by Eisenacher at KAPL to test the feasibility of operating a neutron-sensitive counting device with an internal alpha source in order to make a detector that could be checked without an external neutron source.[24] This is proposed as a desirable alternative to using a neutron source and suitable mechanical equipment

to check a neutron detector in position prior to reactor startup. A BF_3 counter was fabricated with an internal alpha source of 1.7×10^{-9} g of Am^{241}. The source was micropipetted into the cathode cylinder as an aqueous solution and then evaporated to dryness.

The counter performed as expected. The largest pulses from the alpha source were greater than the pulses from the B^{10} reaction by a factor of approximately 2; however, the alpha source also gave rise to pulses in the amplitude range of the neutron-produced pulses. The use of an internal alpha source appears to be a reasonable way to check the performance of the detector. However, the detector must always be operated with a background if all pulses above a fixed bias are counted.

Eisenacher suggests that alpha source pulses falling in the neutron pulse distribution could be minimized by a design that would achieve greater energy resolution of the alpha source and that alpha-source pulses exceeding the maximum neutron pulses could be eliminated electronically. The minimum detectable neutron flux with the 100-count/sec alpha background is 20 counts/sec using the 95% confidence level as $\pm 2\sqrt{100}$. The detector sensitivity per unit neutron flux is 5 counts/sec.

Threshold Detectors for Fast-Neutron Flux Measurements

Ricabarro[25] notes that several years ago it was suggested that the (n,p) threshold reaction on P^{31} yielding Si^{31} (2.6 hr) could be used to measure fast-neutron yields from accelerators. He states that this process for measurement of fast neutrons around a reactor is about 70 times more sensitive than the S^{32} (n,p) P^{32} (14.3 d) process, and, based on evidence available on cross section and extrapolated threshold energy,

the two processes should be equivalent over the neutron spectrum of interest. To check this conclusion, samples in pairs (sulphur and phosphorus) were irradiated in several positions in a reactor to expose the samples to fast fluxes that varied by a factor of 300. Ricabarro concludes that the activities of P^{32} and Si^{31} showed the same ratio (within the limits of accuracy) in all cases. A flux of roughly 100 fast neutrons/$cm^2 \cdot sec$ or a single burst of about 2×10^6 neutrons/cm^2 can be observed.
(R. K. Abele)

CALCULATION OF CONTROL ROD EFFECTIVENESS

Calculations of control rod worth play an important role during the design of a nuclear reactor. Control rods and the associated control rod drives are mechanically complicated and quite expensive, and therefore there is considerable incentive to reduce the number of control rods to a minimum. Additional incentive is present if the control rods give rise to difficult structural design problems in the reactor core and pressure vessel.

On the other hand, the safety of the reactor hinges on the provision of sufficient Δk in the array of control rods to maintain the core in a subcritical state for the highest reactivity condition achievable. Except for times when a number of control rods are removed for repair, core access, etc., or when special fuel management schemes are being applied, the maximum reactivity condition of the reactor will normally occur at the beginning of an operating cycle with a fresh core. Additional incentives for reduction of the shutdown margin thus arise from the desire to obtain a long core lifetime in order to reduce fuel fabrication expenses.

Although sufficient shutdown margin must be available for safe reactor operation, its achievement does not depend solely on accurate calculations of control rod worth. Initial critical experiments at reactor startup provide the final information on control rod effectiveness, and the discovery at that time of overestimates of control rod worth allows the development of special steps to assure adequate margin. Such steps would, in general, involve either incomplete loading of the core or the insertion of additional auxiliary poison elements. This available check

and the opportunity for corrective action prevent inaccuracies in calculations of control rod worth from becoming a direct cause for a hazardous situation, although a very important indirect connection may exist.

Methods of Calculation

The traditional difficulty in doing accurate control rod calculations stems primarily from the inapplicability of diffusion theory in systems with the high degree of flux anisotropy characteristic of control rod lattices. The most direct means of combating this difficulty is through the application of transport theory to all aspects of the problem. Here the difficulty arises that the equations of most interesting systems can be solved only through the applications of numerical methods for the direct integration of the transport equation. The modern computing facilities now available provide the means for doing this, but the simplicity and economy of diffusion theory calculations have tended to prevent widespread use of this method. As noted above, the diffusion theory is inadequate near a control rod, but this shortcoming may be removed by imposing an accurate boundary condition at the control rod surface. In this way the well-known accuracy of diffusion theory in the asymptotic region is continued up to the control rod surface.

The classic work in the description of this boundary condition for a black rod is that of Davison, Kushneriuk, and Seidel.[26-29] In this series of reports the flux extrapolation length of transport theory is derived for spheres and cylinders by examining the solutions of the transport equation for these systems in the limits of both large and small rods. For slab systems this extrapolation length is obtained as the solution to

the famous Milne problem, which was first solved in connection with problems of stellar atmospheres. Once the flux extrapolation length is known at the control rod surface, a complete description of the boundary condition is possible, since the inverse extrapolation length is equal to the logarithmic derivative of the flux at the surface.

The boundary condition at the surface of a control rod with some transmission of neutrons has been studied by Davison and Kushneriuk[30] and by Royston.[31,32] Here again, the resulting quantity is the flux extrapolation length. In this case it is convenient to note that

$$\frac{D}{\phi}\frac{d\phi}{dr} = \frac{D}{l_{ext}} \quad ,$$

where $d\phi/dr$ is the radial derivative of the neutron flux, D is the diffusion coefficient in the medium outside of the rod, and l_{ext} is the flux extrapolation length. From this case, the extrapolation length may be seen to provide a measure of the current per unit flux at the surface of the rod, which is a relative transmission factor for neutrons.

The above discussion covers a method for describing the rod that replaces the actual rod region with a boundary condition at the physical surface of the rod. This method is equally satisfactory for rods that are completely absorbing (black) or partially absorbing (gray). An alternate method which has been useful primarily for black rods of irregular shape is the effective radius model of Wheeler.[33] The effective radius, r_{eff}, is defined as the radius of a cylindrical rod that has the same effect on the asymptotic neutron density as the actual rod under the condition that the neutron density vanishes at the surface of the equivalent

rod. The relationship between this model and the previously described model based on specifying the extrapolation length is most apparent for large cylindrical rods where the effective radius is just the actual radius minus the extrapolation length.

An analogous method for handling bodies of irregular shape is the absorption area method of Hurwitz and Roe.[34] Here, the absorption area is defined as the ratio of the rate of neutron absorption by the body to the source intensity in the surrounding medium. Any two bodies with the same absorption area may be considered equivalent in their effects on the system.

The usefulness of the effective radius and absorption-area methods rests mainly on their ability to provide geometrical simplification where the shape of the control element is complicated. For example, the reduction of a cruciform rod to an equivalent cylindrical rod reduces the entire problem from two dimensions to one.

The methods described above provide the basis for an accurate description of the control element itself, which is the starting point for calculations of the reactivity worth of a control rod or an array of control rods. The problem most likely to cause trouble at this point is the need to provide accurate treatment of rod interaction effects and the effect of neutron leakage. These effects cause the total worth of various arrays of rods to be less than the sum of the worths of the separate rods when inserted individually. Precise theoretical studies of this effect have been made by Nordheim and Scalettar.[35] They suggest synthesizing the neutron flux distribution by a weighted superposition of two or more distributions, each of which contains a characteristic singularity at the

position of a control rod. For systems with a high degree of heterogeneity, the small source and sink method of Horning,[36] Feinberg,[37] and Lanning[38] is accurate. The characteristic of this work is the replacement of each control rod and fuel element by an equivalent source or sink. The entire heterogeneity of the core may thus be maintained in the calculation in an approximate way.

A more direct procedure for use after the rod boundary condition has been determined is the use of a high-speed computer in solving the diffusion equations over a one- or two-dimensional description of the system. This method is applicable to singly inserted rods and arbitrary arrays of rods inserted into a core. The effects of rod interactions and leakage are automatically taken into account and the accuracy of the small source or Nordheim-Scalettar methods will be obtained.

In summary, we may note that a number of different methods are available for the calculation of control rod worth. Except for those involving the direct application of transport theory, usually numerically, these methods require the provision of a boundary condition at the control rod surface or its equivalent. This introduction of a boundary condition from transport theory results in an over-all calculation with the accuracy of transport theory.

Applications of the Methods

In essentially every instance some version of the methods outlined above is being employed in the control rod calculations currently being done for the practical design of nuclear reactors. In recent years the application of high-speed digital computers has been especially common, and two examples are given here by way of illustration.

Extensive use of numerical transport theory has been made by Colston, Gross, and Winton[39] in the analysis of control rod effectiveness for the NS Savannah reactor. In this study transport theory was applied to a complicated control rod geometry and the boundary condition at the control rod surface was determined. This boundary condition was then used to represent the control rod, and the effects of various complex arrays of control rods were studied. Since this later calculation was done in a two-dimensional lattice geometry, the reactivity effects of control rod interactions and the effect of neutron leakage on control rod effectiveness were obtained directly. The authors show that the use of inaccurate methods of control rod analysis can result in considerable error in the prediction of operating characteristics of the core as well as in control rod effectiveness.

A second illustration of the methods employed in control rod calculations is provided by the analysis which has been done for the design of the Experimental Gas-Cooled Reactor.[40] In this analysis the simple cylindrical geometry of the control element allows the use of boundary conditions from the previously discussed basic treatments[26-34] over the range of neutron energies for which the absorber is "black." Diffusion theory is applied over the energy range for which the absorber is only "gray." Since the fuel for the EGCR is in a regular lattice array, the authors also compared the results for homogeneous approximations to the lattice with those for the fully heterogeneous treatment. In all cases the calculated values agreed with the experimental values within the quoted experimental uncertainty. (C. A. Preskitt)

References

1. Accidental Radiation Excursion at the Y-12 Plant, June 16, 1958, Final Report, USAEC Report Y-1234, Y-12 Plant, July 1958.

2. H. C. Paxton, R. D. Baker, W. J. Maraman, and Roy Reider, Nuclear-Critical Accident at the Los Alamos Scientific Laboratory on December 30, 1958, USAEC Report LAMS-2293, Los Alamos Scientific Laboratory, February 1959.

3. Idaho Chemical Processing Plant Criticality, Nuclear Safety, 1(3): 75–77 (March 1960).

4. G. A. Smith, Improvements to The Argon-Gammagraph — A Wide Range Gamma Monitor, USAEC Report K-1284, K-25 Plant, May 1956.

5. E. L. Olson and J. A. Harter, Process Monitor Analysis, Victoreen Model 350, USAEC Report AECD-4082, May 1947.

6. H. C. Miller, Scintillation Gamma Alarm, USAEC Report RFP-61, Rocky Flats Plant, May 1955.

7. H. T. Williams et al., Safety Analysis of Enriched Uranium Processing, USAEC Report NYO-2980, New York Operations Office, March 1960.

8. K. W. Bahler, A. F. Rupp, R. H. Lafferty, Jr., J. W. Wachter, L. S. O'Rourke, and C. R. Milone, Prevention and Handling of Radiation Emergencies, USAEC Report K-1436, K-25 Plant, December 1959.

9. J. W. Wachter, Union Carbide Nuclear Company, Y-12 Plant, January 1960, unpublished information.

10. S. E. Groothuis, G. L. Mustard, and D. M. Papke, A Dose Sensitive Radiation Alarm Instrument with Rate Compensation, Instrument Society of America Fall Instrument-Automation Conference and Exhibit: Preprint 44-NY60, September 1960.

11. T. F. Wimett and J. D. Orndoff, Applications of Godiva II Neutron Pulses, Second United Nations International Conference on the Peaceful Uses of Atomic Energy, 10: 449–460 (1958).

12. J. W. Wachter, B. J. Youngblood, and S. E. Groothuis, Response of Radiation Monitors to Prompt-Critical Bursts, USAEC Report Y-1182, Y-12 Plant, April 1958.

13. J. L. Stringer, General Electric Company, Hanford Laboratories Operation, July 1959, unpublished data.

14. C. L. Brown, Minutes of Criticality Instrumentation Meeting, Hanford Atomic Products Operation, August 17 and 18, 1959, USAEC Report HW-61931, Hanford Atomic Products Operation, September 1959.

15. J. B. Deal, Jr., Gamma Radiation Detector and Warning System, USAEC Report LA-2490, Los Alamos Scientific Laboratory, September 1959.

16. J. N. Wilson, E. I. du Pont de Nemours and Co., Savannah River Plant, unpublished information.

17. J. E. Woolston and S. J. Townsend (eds.), Proceedings of the Sixth Tripartite Instrumentation Conference, Chalk River, 1959, AECL-805, Part 5 (August 1959).

18. W. Lones, Ionization Chambers for Reactor Operation, ibid., Part 5, pp. 11–17 (1959).

19. R. P. Henderson, RC6 and RC7 Ionization Chambers, ibid., Part 5, pp. 20–24 (1959).

20. W. W. Managan, Ion Chambers for Reactors, ibid., Part 5, pp. 1–10 (1959).

21. L. Sani and F. Zanchi, Some Aspects of the Control of Nuclear Reactors – Experimental Investigation of Properties of Boron Trifluoride Proportional Counters, l'Energia Electrica, 37(7): 642-654 (1960).

22. P. G. Salmon, High-Temperature Fission Chambers, op. cit., AECL-805, Part 5, 37-41 (1959).

23. R. J. Epstein and D. C. Thompson, A Low Dead-Time Neutron-Counting System, USAEC Report ANL-6292, Argonne National Laboratory, January 1961.

24. P. L. Eisenacher, Self-Calibrating Thermal Neutron Counter, USAEC Report KAPL-M-PLE-3, Knolls Atomic Power Laboratory, November 14, 1960.

25. B. Ricabarro, Measurement of Fast Neutron Flux in a Reactor by Means of Threshold Detectors, Reactor Science, 12(4): 177 (August 1960).

26. B. Davison, Influence of a Large Black Sphere Upon the Neutron Density in an Infinite Non-Capturing Medium, USAEC Report MT-93, November 1944.

27. B. Davison, Influence of a Small Black Sphere Upon the Neutron Density in an Infinite Non-Capturing Medium, USAEC Report MT-88, November 1944.

28. B. Davison, Influence of a Large Black Cylinder Upon the Neutron Density in an Infinite Non-Capturing Medium, USAEC Report MT-135, April 1945.

29. W. P. Seidel, B. Davison, and S. A. Kushneriuk, Influence of a Small Black Cylinder Upon the Neutron Density in an Infinite Non-Capturing Medium, USAEC Report MT-207, February 1946.

30. B. Davison and S. A. Kushneriuk, Linear Extrapolation Lengths for a Small Black Cylinder with a Small Air-Gap and a Small Imperfectly Absorbing Cylinder with and Without an Air-Gap, USAEC Report TPI-45 (AECL-124), July 1947.

31. R. J. Royston, The Linear Extrapolation Length at the Surface of an Imperfectly Absorbing Cylinder, J. Nuc. Energy, 1(3): 194 (1954–1955).

32. R. J. Royston, The Behavior of a Flux of Neutrons in the Neighborhood of a Control Plate, British Report AERE T/R 2211, March 1957.

33. J. A. Wheeler, Principles of Nuclear Power, USAEC Report N-2292, June 1944, Chapter 22.

34. H. Hurwitz, Jr., and G. M. Roe, Absorption of Neutrons by Black Control Rods, USAEC Report KAPL-1336, Knolls Atomic Power Laboratory, March 1955; or J. Nuc. Energy, 2(2): 85 (1956).

35. L. E. Nordheim and R. Scalettar, Theory of Pile Control Rods, USAEC Report MDDC-42, June 1946.

36. W. A. Horning, Small Source Model of a Thermal Pile, USAEC Report HW-24282, Hanford Atomic Products Operation, February 26, 1957.

37. S. M. Feinberg, Heterogeneous Methods for Calculating Reactors: Survey of Results and Comparison with Experiment, in Proceedings of the International Conference on the Peaceful Uses of Atomic Energy (Geneva, 1955), Vol. 5, p. 484, United Nations, New York, 1956.

38. D. D. Lanning, Heteorogeneous Reactor Critical Conditions Using Small Source Theory, Reactor Control Meeting Held in Los Angeles, Calif., March 6–8, 1957, USAEC Report TID-7532 (Oct. 1957), Part I, p. 61.

39. B. W. Colston, E. E. Gross, and M. L. Winton, Heterogeneous Control Rod Studies, in Proceedings of the ANPP Reactor Analysis Seminar, October 11–12, 1960, MND-C-2487, p. 131, January 1961.

40. Gas-Cooled Reactor Quarterly Progress Report for Period Ending December 31, 1959, p. 9, USAEC report ORNL-2888, Oak Ridge National Laboratory.

IV. PLANT SAFETY FEATURES

RADIOACTIVE FLUIDS LEAK DETECTION

The high levels of contamination and potential hazards that may result as the consequences of a leak in a system containing radioactivity have imposed upon the nuclear industry a leaktightness requirement not generally found outside the high-vacuum industry. The actual leaktightness requirement depends on the specific system or container, as well as the level of the activity therein, and the probability and consequences of escape of the activity. This article is principally concerned with leakage of radioactive fluids from dynamic systems. Some recent articles on liquid-metal and heavy-water leak detection are also reviewed. Some aspects of radioactive-fluid leak detection have been covered in previous articles in Nuclear Safety, including fuel element leak detection[1,2] and leak tests of containment structures.[3] The latter subject is reviewed further in the succeeding articles. In addition, techniques of monitoring liquid[4] and gaseous wastes[5] have been reviewed that are applicable to detection of leaks of radioactive fluids into either a liquid or gaseous stream.

Preoperational Leak Testing

A prime step toward eliminating the need for radioactive fluids leak detection is thorough preoperational leak testing of all equipment and piping. Several articles summarizing current methods have been published.[6–9] The difference of opinions noted in refs. 7 and 8 on the usefulness of a radioactive tracer gas as a leak-detection method is of interest. On the one hand Winter[7] reports that P^{32} would have been too

expensive, too difficult to obtain and dispose of, and would have required unduly elaborate safety precautions. On the other hand, Manley[8] reports the favorable results obtained with Kr^{85} as compared with the results of the use of a halogen leak-detection method. Mass spectrometry, infrared, and vacuum techniques were rejected because of their cost and the weight of the required equipment. The difference of opinions on the merits of tracer leak-detection methods may have been the result of comparing the use of an inert gas with the use of a liquid, although the use of the P^{32} in a liquid was not specifically discussed.

Chemical preoperational leak-testing methods can be very useful in certain systems, particularly where one of the test reagents is also used during operation of the system. Such was the case with the leak testing of the Volatility Pilot Plant at Oak Ridge National Laboratory, for which the fluorine-starch-KI reaction was used.[10] As a final leak test (after the soap-bubble test showed no leaks), a fluorine-nitrogen mixture was introduced into the system at a pressure higher than would be encountered during operation. Then each joint was wrapped with cleansing tissue and wet with a starch-KI water solution from a wash bottle. The resultant darkening of the wet tissue disclosed all leaks which were great enough to be of any process significance.

Operation Leak Detection

A common method for the detection of leaks of radioactive gases during actual operation is radiation monitoring of a stream into which the gas has leaked. A change in pressure is another useful way of detecting gas leaks. The two methods were combined in the leak-detection

scheme for the heat exchanger of the Circulating-Fused-Salt-Fuel Irradiation Test Loop located in the Materials Testing Reactor.[11] The loop is composed of a pump and a hairpin-shaped length of 1/8-in. sched.-40 pipe. The heat exchanger consists of a 1 1/4-in.-o.d. tube concentric around the fuel pipe. It carries air at 30 to 90 psi inlet pressure at flows up to 2500 lb/hr to remove heat from the fuel pipe. The fuel pipe through the heat exchanger is of double-wall construction to prevent release of fuel and fission products to the external unshielded cooling air stream if the primary fuel tube should rupture. In order to provide for continuous monitoring of the integrity of each wall, the outer tube of the double-walled fuel pipe has both longitudinal and helical grooves cut on its inner surface. Gas flow is maintained through these grooves with provision for sensing both pressure change and radioactivity resulting from a leak in the primary fuel pipe.

The radiation monitoring of a small air sweep through its containment vessel was the leak-detection method employed with the in-pile corrosion test loops for aqueous homogeneous reactor solutions.[12] The air was routed through a charcoal trap to adsorb radioactive gases picked up by the air and thus provide a concentration factor for the radiation monitor located near the charcoal trap.

Leaks of radioactive gases from the equipment in the current version of the ORNL Fluoride Volatility Pilot Plant will be detected by pressure changes at specially modified ring-joint flanges,[13] by fluoride leak detectors, and by radiation monitors.[14] Leak detection at ring-joint flanges is accomplished as follows: a small hole is drilled into one flange of each pair and another hole is drilled through the ring joint to make a

continuous connection between the cavity on both sides of the ring joint to a small tube leading away from the pair of flanges. Nitrogen pressure is applied through the tube to the opening in one flange of each pair. Thus a leak in either of the two inner, or process, lines of contact between the gasket and the flange grooves will force nitrogen into the process stream if the nitrogen pressure is higher than the process pressure; a leak in either of the two outer lines of contact will allow nitrogen to escape to the atmosphere. Nitrogen pressure is supplied from a manifold which can be used as either a dynamic or static system. If total leak rates are high enough, the nitrogen flow rate to the buffered flanges will be measured by a rotameter. As the leak is decreased, a static system will be used; the manifold will be pressurized, the gas supply stopped, and the leakage will be measured by rate of pressure drop. This technique was adopted from a liquid system devised for use on the Aqueous Homogeneous Reactor[15] and its chemical processing plant at ORNL.[16] The fluoride analyzers previously mentioned as being used in the ORNL Volatility Pilot Plant for leak detection are based on the current flow method and were developed by Howard and Weber of the Oak Ridge Gaseous Diffusion Plant.[17]

Two liquid leak-detection methods from the nuclear process field that do not rely on radiation monitoring have been reported recently. As a part of a discussion on liquid-metal instrumentation practice at Pratt & Whitney Aircraft, Bliss described techniques for detecting liquid metal leaks.[18] Even slight leakage from uncovered vessel walls or piping can usually be detected visually, either by sputtering, smoke, or surface deposits. Covered sections can be wound, prior to covering, with a

conducting tape consisting of stranded No. 18 Chromel wires held in place by weaving in Fibrefrax. Liquid metal leakage is detected by electrical conductivity between the wires. Part of the tape disintegrates at 1600°F, but the wires are held in place by the thermal lagging. The problem of detecting liquid-metal leaks from a radiator into the air stream of a jet engine is much more difficult and still not solved satisfactorily. A smoke detector has been used with some success on test radiator sections.

The problem of heavy-water leakage detection in certain pressurized-water-cooled power reactors falls within the scope of this discussion since the D_2O becomes radioactive from formation of tritium, N^{16}, O^{19}, and activated corrosion products. Two studies related to the problem were made recently in connection with the design of the CVNPA, a pressurized-D_2O-moderated and -cooled tube power reactor.[19,20] The problem was to minimize the loss of expensive D_2O by detecting as quickly and at as low concentration as possible leaks of D_2O from the primary to the secondary circuits of the steam generator. The detection methods reviewed were the infrared spectrometer, which is an on-stream monitor for the isotopic analysis of D_2O in light water, the on-stream radiation detector for monitoring the activity in the secondary system, and the laboratory batch-sample type of spectrometer for measuring the tritium content of the secondary system. The infrared spectrometer was determined by analysis to be the best on-stream detector, since smaller leak rates can be detected than with the on-stream radiation detector. The lower capabilities of the latter instrument are due mainly to the short half lives of the nitrogen and oxygen isotopes and to the low specific activity of the fission products produced from the U^{238} contamination in Zircaloy.

The analysis further indicated that the laboratory tritium monitor would serve as a very good backup instrument because of its much greater sensitivity, despite the longer time cycle required. An on-stream instrument of greater sensitivity than afforded by the infrared spectrometer is needed to lower the heavy-water leak rate that can still be detected. Secondary system losses must be kept to a minimum to reduce losses which might occur when the D_2O loss was at a level below the infrared spectrometer limit of detection. (R. P. Milford)

LEAK TESTS OF CONTAINMENT STRUCTURES

Containment structures are usually provided to enclose nuclear facilities so that in the event of an accident the uncontrolled release of radioactivity will be limited. Although a steel pressure vessel is the most common type of containment structure on power reactors in the United States,[21] containment structures in general have assumed many forms, i.e., concrete, underground buildings, etc., and in each case the potential activity release is combined with some limiting exposure value and other phenomena[22] to determine the maximum allowable leakage rate. With the acceptable leakage thus established, it is then necessary to prove the nonexistence of a higher leakage rate. This article updates a previous similar article in Nuclear Safety[23] and includes a review of several current reports dealing with various aspects of containment leakage.

Integrated Leak Tests of Pressure-Containment Structures

The techniques used in the leak testing of pressure-containment structure have been developed and used in the leak testing of conventional process and steam power vessels. The comparatively large size of "gas-tight" containment structures and the need to establish, in general, a very precise over-all leakage rate are, however, factors which distinguish containment-structure leak testing from the leak testing encountered in ordinary industrial practice. As a result of these factors, a pressure-drop observation has generally been used to determine the over-all leakage rate.

Many environmental variables such as a change in barometric pressure, a heat loss or gain resulting from the daily temperature cycle, and a change in the absorbed water content in the contained air contribute to the difficulty in determining a precise over-all leakage rate. In an effort to combat the effect of temperature variations of the containment structure and the contained air, in particular, on the pressure-drop measurement, some fabricators have employed a technique that is often called the "reference method." Preleak-tested interconnected reference vessels are interspersed throughout the containment shell, and pressure drop measurements are made by comparing the pressure differential between the reference vessel system and the containment shell. Since the temperature of the reference vessels should follow closely the temperatures of the containment air in which they are immersed, the effect of temperature transients may be largely eliminated if proper distribution of the reference vessels has been made.

The expense of providing a reference system for leak testing cannot always be justified. Some gas-tight containment structures have been satisfactorily leak tested by direct observation of all the variables which may affect the contained pressure. The choice between the two techniques can be based primarily on economic factors. There does not appear to be any clear technical advantage of one method over the other, although the majority of gas-tight containment structures have been tested by the reference method.[21] Petersen and Vinther[24] report a preference for the direct-observation method in the leak testing of the containment buildings for DR-2 and DR-3. In the leak testing of the DR-3, both the reference and direct observation methods were used with apparently comparable

technical results. The preference expressed by Vinther and Petersen for the direct observation was based on the complexity of the reference system test setup and the favorable cost experience with the direct observation method.

The results reported by Vinther and Petersen for the DR-2 and DR-3 leak tests point up a characteristic problem of all tests by pressure-drop observation. The data scatter has such great magnitude that a reliable leak test requires days, rather than hours, to run. Another interesting and obvious generalization based on these tests is that the effect of temperature transients on data scatter is considerably lessened with an increase in containment building compartmentation and with insulation of the building shell.

Operation Leak Tests of Containment Vessels

Concern and even doubt have been expressed about the ability to maintain the initial leaktightness of containment shells,[25] but the ability to continuously demonstrate the integrity of containment has been provided at a number of installations. Notable among these is the use at Yankee of a system of reference vessels to monitor leakage of the containment shell.[26] A slight positive pressure continuously exists in the containment shell during operation at Yankee, and maintaining the positive pressure differential by comparison with the reference system by means of gas flow through a common household gas meter permits continuous leakage monitoring of the shell. Operational leak tests at Hallam will be performed continuously by tieing into the system a 220-scf inert gas supply through a pressure regulator to maintain 2 1/2-psig pressure in the cell.[27] The upper limit for leakage rate for the cell is 200 scf per day.

Leak Tests of Container Penetrations

Increasing concern has been manifested with respect to the role that penetrations have in the over-all leakage rate of a containment shell. Baurmash et al. have conducted leakage tests on butterfly valves, roof access hatches, dogging bulkhead doors, quick-acting bulkhead doors, louvers, and hollow metal doors.[28] The formulas established for evaluating the leakage of the various openings permit estimates of attainable leaktightness and thereby enable cost estimates to be made of containment structures, including the openings, for the particular degree of leaktightness required. An extension of the tests for evaluation of other types and sizes of openings, including penetrations for instrumentation and electrical leads, and to higher ranges of pressures would be worthwhile endeavor, since the range of interest in the nuclear power industry is considerably beyond the scope of the report.[28]

The conventional approach to sealing electrical penetrations has been to remove the cable insulation in order to enclose the cable in a pothead or special conduit fitting and to seal it with a potting compound, such as the epoxy resins.[24,29–32] Leak tests of such seals are usually performed by one of the following methods: (1) soap bubble tests, (2) pressure tests using halide leak detectors, or (3) pressure tests using helium mass spectrometer leak detectors. With these methods of leak testing, leak rates of the order of 10^{-5} to 10^{-6} cm^3/sec can be sensed.[33] Hermetically sealed terminals capable of leakage rates as low as 10^{-8} cm^3/sec have also been used for low-voltage electrical penetrations.[34] The contribution of electrical and instrumentation penetrations to the containment leakage rate is usually very low, even though hundreds of electrical

and instrumentation penetrations may be involved. Nevertheless, double-sealed electrical penetrations have been incorporated in the NASA Plum Brook Reactor Facility.[32] Any leakage through the seals is measured and pumped from a void between the seals back to the containment shell. Although this scheme involves considerable complexity, any serious deterioration of the seals will be immediately detected.

Air locks and isolation valves are usually the prime contributors to the over-all container leakage rate, both initially and throughout operation. Regularly scheduled maintenance and tests have been shown to be required at Shippingport to maintain an acceptable leak rate for these openings.[35,36]

Leak Tests of Caverns

Economic and site considerations present attractive and sometimes even overriding advantages for underground containment compared with conventional gas-tight steel buildings for many reactor facilities in Europe.[37] Siting problems have not been generally so constricted in the United States. Shippingport is a notable exception. Nevertheless, underground containment will undoubtedly become increasingly more attractive in the United States, so a consideration of the leakage characteristics is in order.

Tests reported by Leardini and Cadeddu of leaktightness of underground excavations in cracked-rock strata indicate a tightness better than that of the conventional building structure but considerably below that for gas-tight containment.[21,38] Leardini and Cadeddu propose that tightness for underground containment be defined as a half-time ($T_{1/2}$), the

period for which the pressure decays to half of its initial value, as expressed by the following formula:

$$T_{1/2} = \frac{\ln 2}{\alpha} ,$$

where α is the constant of containment.[39] The cracked-rock strata was found to have very poor tightness, so various treatments of the walls were also tested. The treatments and the corresponding half times are listed below:

Treatment	Initial Pressure (cm Hg)	Half-Time
Original lining made of 6 in. of unreinforced concrete		~0
Floor paved with 12 in. of good-quality concrete; walls plastered with good cement mortar	7.2	5.05 min
Space between lining and rock filled with cement grout	7.8	108 min
Cement mortar injected at 30 to 75 psig into rock strata to fill cracks within 10-ft distance from lining	8.0	21 hr

Tests run on the diffusion velocity through the microscopic pores of a concrete test specimen gave a half-time of 25 hr. The half-time is based on an observed diffusion velocity of 0.02 $m^3/m^2 \cdot hr$ at an overpressure of 1 kg/cm^2. Leakage tests conducted with condensation occurring on the cavern lining indicated a marked increase in the half-time. It would therefore be expected that the presence of water in the rock strata or a containment accident involving the release of water would materially improve the tightness of underground containment.

The foregoing tests indicate that containment systems utilizing untreated underground excavation in cracked strata or utilizing untreated

concrete should be used for reactor types requiring the tightness of semi-gas-tight buildings or conventional buildings. Tightness approaching that of a gas-tight building will require special treatment or installation of linings. Further leak tests are being conducted[39] to determine the effectiveness of various linings, including special compact cements, epoxy-resin coats, and sheet-steel liners.

Leak Tests of Other Containment Structures

An important development from a cost standpoint for the boiling-water reactor is the containment concept designed for the Humboldt Bay nuclear unit.[40] A loss-of-coolant accident vents the steam from a dry well enclosing the reactor to a suppression chamber where the steam is quenched in the suppression pool and where noble gases are absorbed by the pool. The cost saving is largely effected by dramatically reducing the usual critical container boundary from a reactor plant outside envelope to one enclosing just the primary coolant. The leakage rate for the dry-well container is very stringent, being 0.05% of the contained volume per day, but there should be little difficulty in performing the leak testing and maintaining the integrity because the vessel is very small.

Bellas[41] has reported the completion of leak testing by the reference vessel technique of the HWCTR containment structure, a composite steel and concrete structure. An over-all leakage rate of 0.58% of the total free volume per day was observed, which was well within the 1% leakage rate permissible for the structure. The leak testing of the HWCTR confirmed that plastic-coated, prestressed concrete structure can be constructed and tested by conventional techniques for leakage rates of 1%

or less of the free volume per day. Prior to the construction of the HWCTR, the use of steel vessels or liners had generally been considered necessary to attain leakage rates of the order of 1%. The prestressed concrete structure approach may present very significant cost savings compared with the conventional steel structure, so the HWCTR structure completion is an outstanding milepost in the pursuit of safe and economic containment.

Nongastight containment structures at ORNL[42] have been tested for integrated leakage rates of the order of 1% of the structure volume per minute by determining the velocity profile in the single exhaust path from the structure. Hot-wire anemometers have been used to determine the leakage rate to an accuracy of ±10%. Since contamination prevents access to the exhaust duct for retesting with hot-wire anemometers, it is anticipated that future installations will have the exhaust duct orificed and instrumented to permit continuous monitoring of the leakage rate. (G. C. Robinson)

SEVENTH AEC AIR CLEANING CONFERENCE

The Seventh AEC Air Cleaning Conference was held at the Brookhaven National Laboratory on October 10–12, 1961. The agenda included 40 papers covering many facets of the various air and gas cleaning problems experienced by companies and agencies in the nuclear field.

Papers presented at this Seventh AEC Air Cleaning Conference are scheduled to be published, together with reports of the discussions from the floor, in the near future. A special effort is being made to keep the time interval between the conference and issuance of the proceedings to a minimum. For those interested in additional information on topics discussed, reference to the proceedings of the Sixth AEC Air Cleaning Conference is recommended.[43] Also of interest would be past reviews in Nuclear Safety.[44–54]

Filtration of Micron and Submicron Particles

Considerable attention is still being given to the problem of efficient removal of micron and submicron particles from air and gases emanating from nuclear installations. This problem may be divided into three parts: (1) procurement of suitable filter units in operating condition, (2) efficient installation of such filters, and (3) inspection and testing of final installation to assure that the desired air-filtration efficiency will be achieved.

At the Sixth AEC Air Cleaning Conference,[43] held in July 1959, reports from several large users of high-efficiency filters indicated that procurement of large numbers of filters in a usable condition was sometimes

difficult. Rejections of filters due to damage to the filter media, separators, or wooden frames ran as high as 100% in some cases. Even if the filter unit passed visual inspection, there was no assurance that the filter would produce the filtration efficiency and pressure drop stated in the purchase specifications. Gilbert of the AEC stated during that conference that the AEC was establishing two filter testing stations, complete with equipment to determine dioctylphthalate (DOP) aerosol penetration and pressure drop values for filters of up to 1000-cfm capacity. One was to be located at Hanford and the other at the Army Chemical Center, Edgewood, Maryland. The services of these stations were to be available to users of high-efficiency filters within the atomic energy program.

At the Seventh AEC Air Cleaning Conference, Gilbert reported that the two filter testing stations are now in operation and that large numbers of filters have been tested. It is interesting to note that the quality of the filters is much better since the inception of this testing program, and it is now possible to obtain high-efficiency filter units of tested quality and in usable condition with much less difficulty.

Progress on the second phase of the problem has been less impressive. Damage during installation of filters into the gas-handling ducts or filter pits seems to have become the major factor in low efficiency. The high efficiency of a filter can be seriously impaired by poor or careless installation methods or improper handling. Persons not familiar with the construction of such filters sometimes do not realize that marred surfaces, split filter media, damaged separators, and small holes in the filter face can impair efficiency. To aid in reducing this problem, a filter handbook is being published by the AEC specifically to inform people

concerned with the installation and inspection of high-efficiency filters as to the proper and accepted methods for storage, inspection, handling, and installation.

In-place testing of filter installations was the subject of papers presented by A. J. Breslin, AEC, New York; J. A. Young, Naval Research Laboratories; R. W. Schneider, Oak Ridge; G. Wehmann, AEC, Idaho; and L. Silverman, Harvard. The in-place testing is accomplished by introducing either DOP or uranine aerosol into the gas ducts upstream of the filter bank. By sampling before and after the filter bank, an over-all efficiency determination can be made. Young of the Naval Research Laboratories described a method for locating small leaks in a large multifilter installation. The test equipment for measuring the DOP aerosol was taken into the gas plenum at the downstream face of the filter bank. By surveying the downstream face with a sampling probe while DOP aerosol is being introduced upstream, the exact location of leaks can be determined. For example, the inspection of one filter installation indicated that each individual filter was functioning properly and that the leakage was through and around the gasketing material used to seal the individual filters into the filter bank. Another leak was found in the welded juncture between the filter bank hardware and the walls of the gas duct.

After initial installation and testing of a high-efficiency filtering system, it is important that the performance of the system be checked continuously or, at least, frequently under actual operating conditions with the particulate matter and aerosols peculiar to the operation present in the air or gas to be cleaned. A report presented by S. Posner, AEC, New York, was concerned with a study of various types of air-sampling

filter papers using 0.025- and 0.3-μ uranine aerosols. Some papers tested indicated quite different efficiencies for the two particle sizes, while other papers exhibited excellent efficiencies for both particle sizes. Among the more efficient papers are Gelman, type AM-4; Whatman, Fibroglas, types A and C; Millipore, types AA and RA; and Mine Safety Appliances, type 1106B. When completed the study will supplement and bring up to date an earlier study by A. D. Little, Inc.[55]

The problem of sample loss in sampling lines from gas ducts or stacks was discussed by L. C. Schwendiman, Hanford. Deposition of zinc sulfide particles of 2- or 4-μ-diam under turbulent conditions has been studied. The degree of turbulence and the particle size were found to be important parameters. Sampling line losses can be significant, as evidenced by losses as high as 87% for 4-μ particles in a vertically mounted 1-in.-diam pipe 72 ft in length. J. F. Manneschmidt, Oak Ridge, reported on research and development at the Oak Ridge National Laboratory on the problem of monitoring gas-disposal stacks. The monitoring devices include a step-moving filter paper collector for particulate detection, a charcoal trap for adsorbable gas detection, and an in-place ion-chamber detector for gross activity monitoring.

Environmental Air Monitoring and Dispersion

Several aspects of environmental air monitoring were covered. C. Distenfeld, Brookhaven National Laboratory, discussed the environmental air control, alarm, and monitoring system of the BNL Research Reactor. Particulate monitoring in the reactor area is by moving filter-tape devices, with the alarm system being activated by high activity detected

on the tape. A. W. Western, U. S. Public Health Service, Idaho, reported on the development of a high-volume electrostatic sampler for the detection of Sr^{90} and Cs^{137} in the atmosphere. Because of the very low concentrations in air, sampling rates of 1000 to 2000 cfm are necessary. Preliminary results on collection of test aerosols are encouraging. Large collector plates are used, and the problem of sample recovery may become annoying. Air monitoring following the SL-1 reactor accident at the NRTS was the subject of the paper presented by G. Wehmann, AEC, Idaho. An existing network of air-sampling stations at the NRTS was involved. Air sampling by filter paper and activated-charcoal units indicated that the air concentration of I^{131}, off site, did not exceed the recommended MPC. The utility of an established area air-monitoring system was vividly illustrated by this unfortunate accident.

The atmosphere is the final disposal medium for decontaminated gases released from nuclear installations. An understanding of the meteorological parameters related to atmospheric disposal is important to both the designer and user of air-cleaning devices. D. Pack of the U. S. Weather Bureau described meteorological tests being conducted at various locations in the United States to study atmospheric dispersal and transport of contaminants released into the environment. It is now possible to predict, with fair accuracy, using readily obtainable meteorological information, the resulting air concentration of a material at various distances from a gas-disposal stack. W. M. Culkowski, U. S. Weather Bureau, Oak Ridge, reported on the development of an analog device for obtaining local meteorological information directly and providing a continuous estimate of atmospheric dispersion constants.

A program is under way at Brookhaven National Laboratory to study the diffusion and deposition of radioactive particles released near ground level. M. Smith of Brookhaven discussed the facility for performing these studies. A 100- by 100-meter field is outfitted with 100 sampling stations with sampling heads at three heights above ground level for diffusion measurement. Radioactive copper spheres, 9.5, 4.6, or 2.7 μ in diameter, are dispersed at one corner of the field to simulate an aerosol. Fallout-plate collection, together with direct counting of the grass, is used to obtain deposition data.

Activity Control in Nuclear Accidents

Accidents involving nuclear reactors or nuclear fuel processing plants where releases of large quantities of fission products could occur must be handled in a manner different from that used for normal processing of low-level contaminated air. In many instances gas-tight metal containment vessels are used to enclose a reactor and prevent release of activity. This is a very expensive solution. Particle filters are used as part of the reactor confinement system at Hanford according to H. W. Heacock. This confinement system consists of (1) a dense, finely atomized spray over the rear face of the affected reactor, which will remove some halogen vapors and particulates as well as provide some cooling and pressure control, (2) a filtering facility using high-efficiency filters, and (3) activated charcoal units for final removal of iodine vapor. J. B. Hays, Savannah River Laboratory, reported on the performance of high-efficiency particulate filters subjected to steam-air mixtures such as those that could be present in the event of an accident involving

water-moderated or -cooled reactors. Exposure of five units to steam-air mixtures for periods of 10 days at rated air flow resulted in severe damage to only one unit. Double-containment criteria for chemical processing pilot plants at the Oak Ridge National Laboratory were discussed by J. P. Nichols. All plant operations at ORNL capable of producing a radioactive hazard have been evaluated, and corrective action has been taken in the air handling and control system if deemed necessary.

Two entirely different methods of confinement or containment of radioactive material emanating from a reactor accident are under study at the Harvard Air Cleaning Laboratory. L. Silverman gave an encouraging progress report on these concepts. One concept concerns use of a diffusion board for reactor containment. This material would replace the metal plate in a containment vessel and would be pervious to expanding gases but impervious to particulate matter and iodine vapor. Other necessary properties include noncombustibility and resistance to steam, shock waves, and nuclear radiation. Preliminary tests on materials meeting some of these requirements indicate that the concept is, indeed, feasible. The other concept, termed "foam encapsulation," involves the use of an aqueous, plastic, or rigid foam to encapsulate halogens and particulates evolved within an existing containment vessel. Studies conducted with 0.2- and 0.07-μ uranine aerosols and with I^{127} vapor have demonstrated that encapsulation and retention of up to 95% of an aerosol is possible in laboratory-size systems.

Control of Specific Isotopes

Laboratory studies of methods and equipment for the control of specific hazardous isotopes were presented. Radioactive iodine vapor is one of the more hazardous volatile fission products and may be removed from a gas stream by activated charcoal, silver-plated copper mesh, copper mesh, and other materials with efficiencies as high as 99.99+% in some cases, according to P. Dejonghe, CEN, Mol, Belgium; R. Dennis, Harvard University; and R. E. Adams, Oak Ridge. The physical form of the iodine in the gas system is considered to be of primary importance in iodine-removal problems. The behavior has been shown to differ depending upon whether the iodine exists as a vapor, as a complex compound, or adsorbed on a small solid particle. Future studies on the removal of iodine vapor, in very small concentrations, from gas streams were indicated. Dejonghe presented results on the removal of iodine through use of a fluidized-bed technique. The activated charcoal is moved in a fluidized state through a "closed" loop. Removal of the charcoal granules from the gas stream is accomplished through the use of a cyclone particle remover. Final filtration of the exhaust air was indicated for removal of charcoal fines penetrating the cyclone. In practice, it is thought that use of the fluidized bed would produce sufficient decontamination with less heating of the charcoal by decay of adsorbed iodine, since adsorption of iodine vapor on the individual charcoal granules is more uniform than in a fixed charcoal bed. If this occurs a significant decrease in the fire hazard would be produced. F. Salzano, Brookhaven, summarized the fission-product adsorption work at BNL covering the adsorption of krypton and xenon on activated charcoal

over the temperature range 25 to 100°C. Preliminary data on the behavior of iodine and xenon on graphite at elevated temperatures were also presented.

Other Gas Cleaning Problems

One major problem facing workers in the field of air and gas cleaning has been and still is the lack of information as to the identity, amount, and physical characteristics of radioactive material released into off-gas systems serving nuclear plants and reactors. Efforts to alleviate this void of information are being made. W. J. Martin reported on studies under way at the Oak Ridge National Laboratory designed to identify and study the characteristics of fission products released from a variety of preirradiated nuclear fuel specimens under melting temperatures. Information is also being collected on the identity and physical characteristics of submicron particles evolved during melting of the specimens.

The disposal of contaminated liquid and combustible waste materials also involves air-cleaning problems according to W. T. Kyritsis and R. Dennis of the Harvard Air Cleaning Laboratory. An evaluation of the performance of a liquid-jet fume scrubber as a means of processing contaminated gas streams resulting from the evaporation and calcination of radioactive liquid wastes is in progress. Removal of iron oxide fume, fly ash, nitric acid vapor, iodine vapor, and nitrogen dioxide gas has been studied. Incineration of combustible solid wastes is an attractive process to reduce the volume so that over-all disposal costs may be reduced. Dennis reviewed incineration experience at several AEC sites.

Failure of most incineration devices to compete economically with the bulk burial of contaminated wastes was due to the high cost of providing adequate gas cleaning systems. Incomplete combustion released sufficient ash and particulate matter to overload high-efficiency filters installed in the gas exhaust ducts. (R. E. Adams)

GLOVE BOX AND HOT CELL DESIGN CONSIDERATIONS

Glove boxes and hot cells are used throughout the nuclear industry for handling and processing radioactive materials. Because of the diversification of research and development work, the designs and the materials of construction of these facilities vary widely. The handling of high levels of radioactivity makes it important that safety be given primary consideration in the design of these facilities.

In considering design criteria the operational aspects of safety must be considered. Procedures for normal operations and all foreseeable emergencies should be in writing. Regular inspections should be conducted to determine the condition of each facility and to ensure that approved procedures are being followed. The value of remote manipulations, containment, and other safety provisions in even the most carefully designed cells or glove boxes can easily be nullified by inexperienced or careless people who fail to work and think safely.

While the basic purpose of both glove boxes and hot cells is to permit certain operations to be accomplished with safety to personnel and property, the construction, operation, and safety aspects differ sufficiently to warrant separate discussions of safety design criteria for glove boxes and for hot cells.

Glove Boxes

Glove boxes are used to permit operations within a controlled atmosphere of a particular nature either for the benefit of the operation or the operator.[56] The welding and casting of highly reactive metals, such

as titanium, zirconium, tantalum, and niobium, in a pure inert atmosphere and the handling of materials requiring an extremely dry atmosphere, such as lithium, sodium, potassium, rubidium, and yttrium, are examples of operations which require a controlled environment. On the other hand, glove boxes are used to protect the experimenter, other persons, and property from radioactive or other types of toxic material. Glove boxes are sometimes used for both reasons, as in the case of the handling of plutonium metal in an inert atmosphere. A previous article in Nuclear Safety contained a detailed discussion of glove box design for plutonium handling.[57]

The design of glove boxes and the associated equipment and systems varies considerably throughout the many Atomic Energy Commission installations.[58] This variance is partly related to the type of operation (research or production), and the kind, form, and quantity of material handled. Another cause of the variance is that safety design criteria for glove boxes have not been definitely or uniformly established. The individual nature of glove boxes accounts for some of the variation in the selection of materials for construction. Plywood, coated mild steel, stainless steel, and aluminum are all partially justified, according to the circumstances and the designers point of view, on the basis of economy. There is a difference of opinion on the fire resistance of aluminum used for floors in glove boxes containing plutonium. Furthermore, there seems to be no one window material that satisfies all the requirements, and consequently compromises must often be made. Safety glass and several plastic materials are used.

Experience with fires and explosions in glove boxes has focused attention on the safety aspects of these facilities.[59–62] An engineering study of the designs and the procedures for the use of glove boxes was conducted by the Atomic Energy Commission in order to establish recommendations for the development of safe and rational design, construction, and operation of glove boxes and associated facilities, especially facilities for handling plutonium or other pyrophoric metals.[58] A general conclusion was that a more unified effort should be made to improve the design of glove boxes with respect to fire and safety features. From the safety aspect, facilities which handle plutonium or other pyrophoric metals are of chief concern. Facilities that are safe for handling plutonium are, in general, safe for work with other alpha emitters and toxic, nonradioactive materials.[56]

The principle hazards arising out of glove box operations are criticality, penetrating radiation, contamination, and fire and explosion. Criticality is of constant concern when more than the "always-safe" amounts of material are being handled. There are two general methods of guarding against critical incidents: one involves the always-safe mass and the other the always-safe shape. These methods are often combined, but the always-safe method is recommended for glove boxes. Precautions must be taken to prevent flooding glove box systems and operating floors. Overhead sprinklers are generally not used in areas containing glove boxes if the quantities of material present could create a criticality problem.

Penetrating radiation (beta and gamma) is not of great concern in the majority of glove-box operations. The addition of shielding material and the use of remotely operable equipment are frequently employed in

order to cope with penetrating radiation. Thin sheets of lead over the glove box and extra glass (safety or lead) are used when the radiation levels are moderately low. Lead-impregnated gloves are often employed; however, such gloves are stiff and do not permit delicate manipulations. For higher levels of radiation, thick sheets of lead or steel are added to the box; lead glass windows to provide comparable shielding are installed; and manipulating devices, such as the Castle manipulator (cylinder and disk type) or the ball-socket type of manipulator, are used. For very high levels of radiation, glove boxes may be placed in hot cells and the remote manipulating facilities of the hot cell used.

Contamination (not aggravated by fire or explosion) is the most prevalent glove-box hazard because there are so many ways for it to happen. The highly toxic nature of alpha emitters requires essentially complete confinement, and this is especially difficult when the contaminant is in the form of finely divided dry particles.

The most common technique of handling alpha emitters is to confine them in a box held at a slightly negative pressure with respect to room pressure. The pressure in the room where the glove box is located is also held at less than atmospheric pressure. While some workers feel that adequate containment can be achieved by holding a negative pressure within the glove box, most feel that the glove box must also be as tight as practical, and the majority of glove boxes are designed to obtain tightness. It should be noted that not all glove boxes are held at negative pressures. In inert atmosphere boxes where no toxic material is involved the glove box may be operated at a slightly positive pressure.

Glove materials are susceptible to cracks, ruptures, and punctures. Sharp objects are sometimes prohibited in glove boxes to avoid puncturing gloves or plastic transfer pouches. Where they are required, it is necessary to use them with care. As protection against incipient cracks, gloves are often changed periodically (a preventive maintenance measure) and some contamination is likely during this operation.

Filtering systems are required. The glove-box atmosphere must pass through one or more high-efficiency filters before being exhausted. In many cases it is desirable to filter the glove-box atmosphere at the glove box before passing the stream on to other filters serving the room or building exhaust system.

Fires and explosion are major glove-box hazards, since gloves, plastic transfer pouches, plastic windows and glove-box panels, filter boxes (perhaps containing large quantities of radioactive particles), and the combustible portions of buildings may be destroyed. The spread of contamination invariably follows the destruction of any of the containment elements. Scattering forces, such as the high thermal currents from the heat of the fire, wind, and water used to control the fire, spread activity, which is likely to contaminate sizeable areas and injure and endanger people. Therefore, effort must be given to preventing their occurrence. Means for detection and indicating dangerous and potentially dangerous conditions should be provided.

Fires may be ignited in a number of ways. The important sources of ignition are the pyrophoric metals (especially in dust, chip, or shaving form); glove-box services and equipment, such as open-flame devices, heaters, tools, motors, electric wiring, etc.; and alkali metal reactions.

The chief methods of fire prevention commonly reported are the use of dry inert atmospheres and very dry air atmospheres. Humidity sensors and oxygen analyzers may be employed to detect and indicate undesirable conditions.

Insofar as possible, noncombustible materials of construction should be used. Except where metal recovery is required, most glove boxes are being equipped with noncombustible filters. Glove boxes should have individual filters. Fire breaks should be used where glove boxes fabricated of combustible materials are joined. Metal covers for glove ports should be provided. Buildings, classified as noncombustible (preferably fire resistive), should be selected to house glove-box facilities. Where criticality is not a problem, combustible buildings should be protected by an automatic sprinkler system. Electrical installations should meet accepted safety codes.

The chief measure to avoid explosions is to limit the use of explosive materials, such as solvents, and to limit dust collections. Explosion venting, that is, unique devices such as rupture disks, blow-out panels, blow-off valves, etc. is not in general use. Gloves and transfer pouches serve the purpose; however, they are not ducted to prevent contamination to the operators and to the room in which the glove box is located. Glove boxes are also vented through their filters and ventilation systems, although again this may not prevent damaging results.

Hot Cells

The basic design considerations for hot cells are similar to those discussed above for glove boxes. The major differences are that most hot

cells are designed for the remote handling of large amounts of beta and gamma activity, although hot cells have been designed for handling alpha and neutron emitters. Hot cells consist primarily of shielding walls, cell liner, services, drainage system, and ventilating system. Special-purpose cells may also have viewing windows, manipulators, hoists, conveyors, and access ports. A number of recent reports discuss the design of hot cells for various applications.[63-66]

Explosions and fires present the greatest hazards in hot cell operations, as is evidenced by several incidents which have occurred.[67-69] Quantities of radioactive materials may be released which could create major problems. Four of the most common explosion hazards that may be encountered are (1) incompatible chemicals, (2) flammable solvents, (3) flammable gases, and (4) excessive heat from critical amounts of fissionable materials. Fires may result from pyrophoric metals and combustible materials. Other hazards include inadequate shielding, loss of viewing window shielding, transfer of radioactive solution to the outside of the cell through service piping, contamination of water supply system, manipulator removal for repair, inadequate cell ventilation filters, and improper handling of radioactive solid waste.

Cell and equipment design and subsequent operating procedures must minimize explosion possibilities in facilities where radioactive materials are handled. The use of flammable gases, flammable solvents, incompatible chemicals, and fine powders must be controlled. For some operations, sensing instruments should be installed in the hot cells with alarm systems to warn of explosive conditions.

Fire-protection equipment should be installed in facilities where the possibility of a fire hazard exists.[70] This equipment may consist of fog nozzles, CO_2 systems, dry powder, or foam, depending upon the nature of the operation. The system should provide for automatic operation by thermal detectors inside the cell and manual control from outside the cell. At the same time, precautions must be taken to prevent pressurization of the hot cells during the discharge of the fire-protection systems. This may be an emergency ventilating duct complete with filters and equipped with a damper actuated by the fire-protection system.

Two methods employed in hot-cell operations to guard against unexpected chain reaction incidents are the always-safe mass and the always-safe shape. Wherever possible, the always-safe mass should be used. However, if the operation requires amounts of fissionable material in excess of the always-safe mass, attention must be given to the design of the equipment, product flow, and equipment arrangement to provide for safe-shape conditions at all times, making it impossible for critical amounts of fissionable materials to accumulate in one place.

The cell walls must provide protection from the radioactive material handled and must withstand forces resulting from the maximum credible accident that could occur inside the cell. Ordinary concrete, with suitable reinforcement, is used for constructing most cells, but high-density concrete,[71,72] steel, lead, and combinations of these materials are also used.

All piping that penetrates the cell walls should be arranged in a manner that results in a minimum loss of shielding and must withstand the maximum pressures that can occur. Access ports or plugs must be stepped

or flanged to eliminate straight paths for escape of radiation from the cell. All cell penetrations including plugs, access ports, and doors should be equipped with tight seals and secured to prevent leakage from the cells under all operating conditions.

Viewing windows should provide shielding equivalent to that provided by the cell walls. Windows filled with zinc bromide solution (density, 2.5 g/cm^3) are usually used with normal concrete walls. Nonbrowning lead glass (density, ~3.3 g/cm^3) windows are used with high-density concrete walls. Cells with walls of steel or lead are equipped with glass of 6.2-g/cm^3 density or combinations of glass of 3.3-g/cm^3 and 6.2-g/cm^3 density. Cover panels of nonbrowning safety glass are recommended for the in-cell side of the windows. Viewing windows utilizing a liquid for shielding must be protected from accidental loss of the shielding liquid. In addition to safety-glass cover panels, additional protective covers of plastic or glass should be used if the operations are such that the integrity of either of the glass panels is endangered.

All piping should be designed to prevent the transfer of radioactive solutions to unshielded areas outside the cell. A satisfactory system for service lines is to enclose the control valves in a service tunnel and install a check valve in each line at the cell wall. Solution addition lines may be equipped with check valves or may be vented inside the cell to prevent back pressure. Steam lines to jets should be provided with vacuum breakers.

All radioactive waste solutions should be discharged into a hot waste system. Solutions that are normally nonradioactive, such as cooling water and steam condensate, should be monitored for radioactivity and discharged

into a process waste system. Provision should be made for automatic diversion of the process stream to the hot-waste system if excessive radiation is detected.

A process-water supply should be used in all hot-cell areas to eliminate the possibility of contamination in the potable-water system.

Electrical outlets should be of the vapor-proof type to prevent deterioration from corrosive atmospheres. Cell lighting should be accessible either directly or remotely for replacing bulbs. All manipulators should be grounded to protect operating personnel from electric shock. Manipulators that must be removed from the cell into the operating area should be equipped with slave-end booting. It is recommended that the boot be remotely removable and that it be discharged into the cell for disposal. The boots should be examined regularly and should be changed as soon as a failure is detected. Various types of boots are available from manipulator suppliers. It is also recommended that all Model 8 master-slave-type manipulators be equipped with air restricters in the mounting tube. This device is available from manipulator suppliers and greatly reduces the transfer of air into or from the cell through the manipulator installation.

Consideration must be given to the problem of removal and disposal of radioactive solid waste from hot cells. This waste includes manipulator boots, plastic bags, wipes used for cleaning, and discarded equipment. Provisions must be made for bagging or canning the material, monitoring for excessive radiation, and removing it in shielded containers to a disposal area. Shielded casks with truck-dumping features and special casks for canned waste are used.

The escape of radioactive gases and airborne particulate matter from the cells is accomplished by maintaining a negative pressure on the cell interior. A minimum flow of 100 ft/min through all openings is desired. An alarm should be provided to indicate loss of cell pressure. Primary and secondary filtering systems prevent the discharge of excessive amounts of radioactivity to the atmosphere. The primary system should be located in or near the cell and should consist of roughing and high-efficiency filters. The secondary system should be located downstream from the cell and may be of adequate size to serve an entire processing facility or area. The filters must be continuously monitored for pressure drop and radiation accumulation. If either becomes excessive, the filters must be changed. Contaminated filters should be removed in plastic bags or metal cans enclosed in shielding to permit safe transport to a disposal area.

Conclusions

While the above discussion has covered the major points of concern with regard to hot-cell and glove-box design requirements, it remains for generalized criteria to be evolved for the radioactive facilities. Some tentative criteria for hot-cell design at Oak Ridge National Laboratory were developed[73] after an incident there. These criteria required that a vacuum equal to or greater than 1.0 in. H_2O be established to assure a positive flow of air from the building into the cell. Cell exhaust capacity equal to or greater than 0.1 cell volume per minute should be provided to eliminate any explosion hazard in cells handling flammable solvents or gases. A leakage rate not to exceed 0.01 cell volume per

minute at a pressure differential of 2 in. H_2O is specified to assure that contamination escaping the cell will be at a minimum in the event of cell pressurization.

Similarly, the generalization of criteria for glove boxes is difficult because of the highly individual nature of the facilities and their operation. On the other hand, the general safety principles discussed herein provide the basis for arriving at a safe design. Items which must be considered in evaluating glove box operations involving highly toxic radioactive materials are listed below. Specific criteria must be developed for each facility and operation.

1. Design glove boxes for pressure and leaktightness.
2. Use noncombustible and fire-resistance materials of construction.
3. Provide for the movement and filtering of the glove box atmosphere.
4. Use inert dry atmospheres where pyrophoric materials are involved.

(L. N. Howell and E. E. Pierce)

References

1. Fuel-Element Leak Detection, Nuclear Safety, 1(3): 33–34 (March 1960).

2. Fuel-Element Leak Detection, Nuclear Safety, 2(3): 27 (March 1961).

3. Leak Tests of Containment Structures, Nuclear Safety, this issue, p. 110.

4. Radiation Monitoring of Liquid Wastes, Nuclear Safety, 1(4): 58–61 (June 1960).

5. Monitoring of Gaseous Wastes, Nuclear Safety, 2(2): 25–27 (December 1960).

6. J. P. Verkamp and S. L. Williams, Testing Nuclear Plant Leaktightness, Nucleonics, 14(6): 54–57 (June 1956).

7. P. Winter, et al., How to Find Microscopic Leaks in Process Equipment, Ind. & Eng. Chem., 50: 53A–55A (May 1958).

8. T. R. Manley, Leak Detection Methods, Ind. & Eng. Chem., 51: 73A (August 1959).

9. W. Steckelmacher, Leak Detection, Nuclear Engineering, 4: 450–3 (December 1959).

10. F. W. Miles and W. H. Carr, Engineering Evaluation of Volatility Pilot Equipment, USAEC report ORNL-CF-60-7-65, pp. 324–9, Oak Ridge National Laboratory, September 1960.

11. D. B. Trauger and J. A. Conlin, Jr., Circulating Fused-Salt Fuel Irradiation Test Loop, Nuclear Sci. and Eng., 9: 346–56 (1961).

12. H. C. Savage and W. D. Reel, In-Pile Corrosion Test Loops for Aqueous Homogeneous Reactor Solution, ORNL-2977, p. 46, November 1960.

13. R. P. Milford et al., Volatility Pilot Plant – Hazards Review Report, unpublished report, Oak Ridge National Laboratory, July 26, 1960.

14. R. P. Milford and W. H. Carr, Hazards Evaluation for Bldg. 3019: Pt II. Fluoride Volatility, unpublished report, Oak Ridge National Laboratory, May 1960.

15. J. E. Kuster, HTR Leak Detection System, USAEC report ORNL CF-56-4-105, Oak Ridge National Laboratory, April 1956.

16. W. L. Carter, HRP-CP: Design of Leak Detection System for HRT Chemical Processing Plant, unpublished report, Oak Ridge National Laboratory, December 1955.

17. O. H. Howard and C. W. Weber, A Portable Continuous Analyzer for Gaseous Fluorides in Industrial Environments, Archives of Ind. Health, 19: 355–364 (March 1959).

18. P. Bliss, Liquid Metal Instrumentation Practice, Nuclear Process Instrumentation and Controls Conference, USAEC report ORNL-2695, pp. 86–101, Oak Ridge National Laboratory, April 1960.

19. W. R. Snyder, Carolinas-Virginia Tube Reactor – Steam Generator Heavy Water Leakage Study, Carolinas-Virginia Nuclear Power Associates, Inc., CVNA-57, Westinghouse Electric Corporation, Atomic Power Department, June 16, 1960.

20. A. Andonedis and R. A. Wiesemann, Carolinas-Virginia Tube Reactor – Heavy Water Leak Detection Technique and Development, Carolinas-Virginia Nuclear Power Associates, Inc., CVNA-68, Westinghouse Electric Corporation, Atomic Power Department, June 1960.

21. Containment of Power Reactors, Nuclear Safety, 2(1): 55–71 (September 1960).

22. Reactor Siting Trends and Developments, Nuclear Safety, 2(4): 1–3 (June 1961).

23. Containment Vessel Leak Detection, Nuclear Safety, 1(1): 29–30 (September 1959).

24. J. S. Petersen and F. H. Vinther, Leak Tests of the Buildings for the DR-2 and the DR-3 Reactors, RISO-12, March 1960.

25. T. J. Thompson, Power Reactors and Reactor Safety, Energia Nucleare, 8(1): 62–70 (January 1961).

26. Yankee Nuclear Power Station Technical Information and Final Hazards Summary, Part B, License Application, YAEC-167, 1959.

27. Additional Safeguards Evaluation for the U-Mo Fueled Core of the Hallam Nuclear Power Facility, NAA-SR-Memo-4067, July 7, 1959.

28. L. Baurmash et al., Leakage Characteristics of Opening for Reactor Housing Components, NAA-SR-Memo-5137, June 20, 1960.

29. E. T. Witt et al., Electrical Features of the Yankee Atomic Electric Plant, Part IIIA, AIEE Transactions, pp. 94–107, April 1959.

30. J. R. Jacobsen, Supplement I to the Preliminary Safeguards Report for the Piqua Organic Moderated Reactor, Revised, NAA-SR-Memo-4048, July 1, 1959.

31. Halden BHWR, Nuclear Engineering, 4(36): 110–111 (March 1959).

32. NASA Plum Brook Reactor Facility, Parts I, II, and III, Final Hazards Summary, NP-8326, December 1959.

33. W. Steckelmacher, Leak Detection, Nuclear Engineering, 4: 50–3 (December 1959).

34. R. H. Logue, Electrical Engineering Aspects of the Enrico Fermi Atomic Power Plant, Part IIIA, AIEE Transactions, pp. 71–76, April 1959.

35. Test Results Reactor Plant Container Integrity Test Core I Seed 2, DLCS 3570201 (T-643733), TID-5842, May 25, 1961.

36. Test Results Reactor Plant Container Integrity Test Core I Seed 2, DLCS 3570101 (T-643733), June 23, 1960.

37. J. Kagi, Some Observations Upon the Influence on Public Safety of Underground Containment in Nuclear Power Plants, Congresso Scientifico, VI Rassengna Internazionale Elettronica e Nucleare, 1: 69–86 (1959).

38 T. Leardini and M. Cadeddu, Caverns as Nuclear Power Reactor Containers, Energia Nucleare, 8(2): 93–98 (February 1961).

39 T. Leardini, Research on Containment Feature of Cavern for Underground Location of Reactors, Congresso Scientifico, VI Rassengna Internazionale Elettronica e Nucleare, 1: 89–101 (1959).

40. C. C. Whelchel, Pressure Suppression Approved for Humboldt Bay, Electrical World, 154(21): 68–72 (Nov. 21, 1960).

41. H. W. Bellas, Heavy Water Components Test Reactor, Savannah River Plant, Composite Concrete-Steel Containment Vessel Engineering Considerations, DP-599, June 1961.

42. Personal communication, A. B. Fuller to G. C. Robinson, November 6, 1961.

43. Sixth AEC Air Cleaning Conference, TID-7593 (October 1960).

44. Removal of Fission-Product Activity from Gases, Nuclear Safety, 1(3): 40–46 (March 1960).

45. Removal of Fission-Product Activity from Gases, Nuclear Safety, 1(4): 47–53 (June 1960).

46. Fission-Product Release Tests, Nuclear Safety, 1(4): 77–82 (June 1960).

47. Monitoring of Gaseous Wastes, Nuclear Safety, 2(2): 25–27 (December 1960).

48. Particulate-Filter Testing and Inspection Program, Nuclear Safety, 2(2): 37–38 (December 1960).

49. High-Temperature Fission-Product Traps, Nuclear Safety, 2(2): 39–42 (December 1960).

50. Atmospheric Dispersion Calculations Using the Generalized Gaussian Plume Model, Nuclear Safety, 2(2): 56–59 (December 1960).

51. Removal of Radioiodine from Reactor Gases, Nuclear Safety, 2(3): 35–38 (March 1961).

52. Removal of Contaminants from Gas Streams, Nuclear Safety, 2(4): 39–41 (June 1961).

53. Hazards Associated with Waste Treatment, Nuclear Safety, 2(4): 45–47 (June 1961).

54. Use of Routine Meteorological Observations for Estimating Atmospheric Dispersion, Nuclear Safety, 2(4): 47–51 (June 1961).

55. Media for Air Cleaning and Air-Assay Purposes, Arthur D. Little, Inc., AECU-3119 (October 1955).

56. C. J. Barton, A Review of Glove Box Construction and Experimentation, USAEC Report ORNL-3070, Oak Ridge National Laboratory, May 1961.

57. Plutonium Handling, Nuclear Safety, 1(3): 49–52 (March 1960).

58. C. K. Beck, Report of Engineering Study of Glove Boxes, Washington, D. C. (July 15, 1959).

59. Staff Paper AEC 344/43, 1957 Summary of Fire and Property Damage in AEC Operations (October 3, 1958).

60. Report of Investigation of Serious Incident in Building 71 on September 11, 1957, The Dow Chemical Company, Rocky Flats Plant, Denver, Colorado (October 7, 1958).

61. USAEC pamphlet, Serious Accidents, Issue No. 130 (November 27, 1957), Small Metallic Plutonium Fire Leads to Major Property Damage Loss.

62. USAEC pamphlet, Serious Accidents, Issue No. 180 (November 20, 1961), Glove Box Explosion.

63. J. R. Farmakes, R. A. Blomgren, and J. H. Schraidt, Design Problems of a Proposed Radiochemistry Cave, Fourth Hot Laboratory and Equipment Conference, Washington, 1955, pp. 121–128, USAEC report TID-5280.

64. Technical Function and Operation of a High Radiation Level Experimental Laboratory, unpublished report, Oak Ridge National Laboratory, January 31, 1961.

65. J. R. Berrith and R. P. Schuman, A Chemistry Hot Cell for Handling Alpha-Gamma Activities, Ninth Hot Laboratory and Equipment Conference, Chicago, 1961, pp. 22–26.

66. B. B. Klima, Transuranium Development Facility, Ninth Hot Laboratory and Equipment Conference, Chicago, 1961, pp. 27–34.

67. Plutonium Release from the Thorex Pilot Plant, Nuclear Safety, 1(3): 78–80 (March 1960).

68. BNL Incident of May 1957, Nuclear Safety, 2(1): 51–53 (September 1960).

69. ORNL Incident of April 1960, Nuclear Safety, 2(1): 53–55 (September 1960).

70. T. W. Hungerford, Fire Protection in Radiochemical Facilities, Sixth Hot Laboratories and Equipment Conference, Chicago, 1958, pp. 327–333.

71. H. M. Glen, The Manipulator Hot Cells Above the Oak Ridge Research Reactor Pool, Sixth Hot Laboratories and Equipment Conference, 1958, pp. 75–80.

72. H. M. Glen, An Engineering Approach to a High Level Manipulator Hot Cell, Fifth Hot Laboratories and Equipment Conference, Philadelphia, 1957, pp. 299–306.

73. Radiation-Safety Improvements to ORNL Facilities, Nuclear Safety, 2(3): 40–44 (March 1961).

V. CONSEQUENCES OF ACTIVITY RELEASE

ANALYTICAL REQUIREMENTS FOR ENVIROMENTAL SURVEYS

The recent resumption of weapons testing has once more directed public attention to radioactive contamination of the biosphere and may result in increased interest and concern with the proximity of reactors to populated areas, site selection, and operation of nuclear facilities in general. Actually the need for more and better data on levels of environmental contamination has been recognized for several years, as illustrated by an order issued by the President in August, 1959. This order gave the Secretary of Health, Education, and Welfare the primary responsibility within the Executive Branch for the collation, analysis, and interpretation of data on environmental radiation levels.[1] The general problem assigned the Secretary is well stated in the first report of his department,[1] p. 6: "The determination of the degree and nature – type of radionuclides – of environmental contamination [which probably will be everchanging], the study of movement of radionuclides in the biosphere, the estimations of intake, ... constitute sets of problems that will hold the attention of scientists in many fields for years to come."

The analytical requirements of an environmental survey are influenced by many factors. The intensity of the survey will depend on the purpose, and whether it is preoperational or continuing. The kinds of radionuclides for which analyses are made, the environmental media which may be contaminated by waste effluents, and the kinds of biological material requiring analysis are among the primary factors which will govern the organization of the work. The relative importance of the various elements and stages of the survey will depend on the nature of the nuclear

installation, the character and composition of its waste releases, and the extent to which surrounding environments are utilized as sources of food and for recreation. It is obvious then that the objective design and operation of an environmental survey is dependent on the evaluation of a number of complex interrelated factors, and to these there must be added such subjective considerations as public relations and the legal requirements for third-party liability. In practice, however, the resolution of these many requirements results in a general design of a survey which is relatively simple.

The primary purpose of most surveys[2,3] is to see that the radioactivity of the surrounding area is maintained at or does not exceed some predetermined tolerable level and that the quantities of radioactive materials released to nearby environments do not exceed levels recommended by the appropriate regulatory agencies. The Federal Radiation Council[4] has recently established new limits for certain radionuclides, such as Sr^{90} and I^{131}, in terms of a permissible rate of intake both for the general population and for the population in the immediate vicinity of a nuclear installation (see Sec. I, this Review). In order to relate these new limits to observed concentrations in environmental media, more data on uptake and concentration in biological materials are needed. In other words, sampling and subsequent interpretation of environmental data should now be designed to relate quantitatively the movement of radionuclides through ecological food chains in order that environmental contamination can be related to human intake.

The analytical methods for determining the quantities of individual radioisotopes in various media are being constantly improved. Simple

determinations of gross beta, gamma, and alpha activity are still very useful, provided these are augmented by other more specific measurements. The low cost of such gross measurements is a material advantage, especially in a continuing survey. A gross beta analysis is also useful for background measurements in preoperational surveys. The disadvantage of the gross beta analysis for operational monitoring is that all activity detected must be assumed to be due to the limiting nuclide, which is usually Sr^{90}. Under these conditions, if the total beta activity exceeds the maximum permissible concentration for Sr^{90}, a radiochemical analysis for that isotope becomes essential. In general, measurements of specific nuclides are absolutely required only when the source of the activity is uncertain, but they may be of interest in many cases where a degree of uncertainty exists regarding the species and quantities of radionuclides which may have been released.

Sr^{90} Analysis

A great deal of attention has been directed toward the problem of Sr^{90} in global fallout, but until recently relatively little effort has been applied to the investigation of the behavior and distribution of Sr^{90} in limited areas. In assessing the contribution of Sr^{90}, or any other radionuclide, from a local installation, it is necessary to be able to distinguish between the contamination from weapons fallout and from local sources. This separation of sources is exceedingly difficult for several reasons. New weapons testing may give rise to new materials whose composition and rate of deposition will vary not only with meteorological conditions but also with the type, size, and altitude at which

the device was exploded. Consequently, the assumption that the deposition rate will be nearly constant over a small study area,[5] while perhaps true during the testing moratorium, is likely to be invalid during the next few years. Moreover, as Olson[6] and others have pointed out, such local features as topography, rainfall, and surface vegetation may affect the movement and subsequent distribution of both locally derived and fallout contamination. The types and accuracy of the analyses required for an environmental survey therefore may be controlled by the more complex problems of selection and collection of samples and of the interpretation of the results, rather than by the complexity or accuracy of the analytical methods.

Many analyses of radiostrontium are complicated by the need to separate the strontium from large quantities of chemically similar calcium. This is true, for example, where the survey requires that this radionuclide be measured in soil, water, vegetation, milk, bone, and other biological materials. Unfortunately, because Sr^{90} is a pure beta emitter, the rapid and simple spectrographic techniques that can be used to detect and measure many of the alpha and gamma emitters cannot be employed.

In September, 1958, the World Health Organization met to review existing analytical procedures for Sr^{90} and adopted[7] as standards the methods used at the British Atomic Energy Research Establishment at Harwell[8] and at the U. S. Atomic Energy Commission's Health and Safety Laboratory (HASL).[9,10] These methods are, basically, modifications of the original method of Glendenin[11] which employed an initial isolation of barium and strontium by precipitation as nitrates with fuming nitric acid. A new

method[12] has recently been developed in which the final Sr^{90} assay is obtained by separating Y^{90} from dilute acid solution by extraction with di-2-ethylhexyl phosphoric acid.[13] This is a relatively simple procedure requiring fewer manipulations and chemicals than the fuming nitric acid method and is applicable to virtually all types of samples taken in environmental surveys. Accuracy may be checked by addition of Sr^{90}-Y^{90} tracer to identical samples or by interlaboratory comparisons, and both checks have shown excellent agreement.[12] None of these radiochemical methods can, however, avoid the need for the difficult and time-consuming separation of radiostrontium from calcium in biological materials. A single radiostrontium analysis of bone, soil, or vegetation, including sample preparation and low-level counting, requires some 4 man-hours of effort, plus the delay required to permit the growth of Y^{90}, if this is being measured. The relatively high cost of an individual radiostrontium analysis can be reduced somewhat by simultaneously processing large numbers of samples. The long half-life of strontium makes frequent samplings of soil or vegetation unnecessary because rates of change would be small, and some installations currently sample on an annual basis.[12] On the other hand, milk and water, which are direct pathways of ingestion, must be analyzed more frequently. Fortunately, the preparation time for these samples is less than for soil or bone, and a new method[12] in which Dowex 50W resin in the sodium form is used to remove strontium and calcium from milk permits a considerable saving of time as compared with the ashing process.[7]

Mixed-Fission-Product Analysis

The complications introduced by recurrent fallout of a mixture of fission products can be partially offset in the case of gamma emitters by the use of pulse-height analyses. Although quantitative analysis of mixed gamma emitters is difficult, many of the components of a mixture can be qualitatively identified, and particular radioisotopes can be singled out for more refined analyses. With careful calibration and correction for Compton scattering, however, the gamma spectrometer can also be used for quantitative analysis of simple mixtures. Recent developments in computer techniques and electronic data processing offer considerable promise for almost complete automation of this analytical tool. Currently such programs, involving the solution of the requisite number of simultaneous equations, are under development and in partial use at several installations.[14] This possibility, in addition to the well-established advantages of minimum processing of samples, ability to accommodate relatively large-sized samples, and little or no need for self-absorption corrections, make the multi-channel gamma spectrometer an indispensible tool in environmental radiation studies.

Iodine Analysis

Radioiodine (I^{131}) is another nuclide that is of special concern and which has been considered by the Federal Radiation Council as meriting particular attention in studies of environmental contamination.[4] The new limits recommended by the Council are considerably lower than those previously recommended by the NCRP and the ICRP. The problems involved in

relating meteorological conditions to I^{131} deposition on vegetation and particularly on tree foliage have recently been reviewed.[6] As the reviewer and others have pointed out, relating rate of uptake to rate of deposition is dependent upon a good knowledge of local meteorological conditions and accurate stack monitoring equipment.[3]

Because radioiodine is concentrated in the thyroid, the sampling and analysis of animal thyroids can provide an index of the extent of radioiodine contamination and of the degree to which the environmental contamination is making its way into the biota. Herbivorous mannals, which feed directly upon vegetation are the obvious choice. Less obvious are the criteria for choosing between domestic and native mammals. Here selection must draw upon knowledge of the feeding sites of the animals in question. Although thyroids are usually readily available from local slaughter houses, the animals from which they were obtained may not be a representative sample of local conditions, and the previous history of many of the animals may not be readily obtainable. In such cases, use of rabbits or other native herbivores may be advisable. We may quote the following from Horton:[3] "A program [regional survey] that will be satisfactory for all locations cannot be designed without the aid of a biologist who is well acquainted with local conditions." The short half-life of radioiodine makes it highly desirable that its analysis be done by means of gamma spectrometry in all cases where such instrumentation is available.

Cesium Analysis

Cesium^{137} is another nuclide that can be determined by gamma spectrometry. Its presence can be determined by direct comparison of its energy peak at 0.661 Mev with that from a standard source. If interfering nuclides, such as Zr^{95}, are present in the mixture, it may be necessary to allow the samples to decay before the final determination of radiocesium. Ruthenium-106, whose characteristic photopeak (0.51 Mev) is close to that of cesium, may interfere and, for critical quantitative determinations, a radiochemical separation may be necessary. In general it is easier to separate the cesium from the mixture than to try to separate the ruthenium. In one of two common methods, Cs^{137} is precipitated with cesium carrier and phosphotungstic acid;[12] this method is shorter and simpler than to precipitate the cesium as cesium aluminum sulfate and then as cesium chloroplatinate.[7]

Analysis for Alpha Emitters

The analysis for certain alpha emitters requires a chemical separation, but under favorable circumstances much simpler methods employing pulse-height analysis may be used. Alpha pulse-height analysis is not troubled by the Compton effect, and, if care is taken to reduce self-absorption, quantitative determinations are more easily made than with gamma emitters. Recent interest in natural radioactivity in water supplies has directed attention to the analytical requirements of large-scale field-sampling programs.[15] The standard techniques for analysis of radon and rhodium are designed for determination of low levels,[7] and, as Higgins et al[15]

have pointed out, these are too time-consuming when there are many samples. These authors have developed a procedure which involves driving the Ra^{222} out of a solution and collecting the gas in an Erlenmeyer flask coated with a screen of silver-activated zinc sulfide, followed by alpha-scintillation counting. Radium-226 was also analyzed with the use of a simple modification of this procedure.[15]

Stable Element Analyses

Many environmental surveys require analysis for stable elements. For example, the uptake and accumulation in organisms of radioisotopes is related to the uptake and accumulation of stable isotopes of the same or chemcially related elements. The maximum concentration of the stable form of an element in a marine or aquatic organism, or in some part of such an organism, is also the maximum concentration that can be expected of a radioactive isotope of this element. Because complete replacement of a stable isotope is extremely improbable, this limit is not likely to be attained. Nevertheless, a knowledge of the relation between the stable isotope content of the medium and that of the organisms of special interest in that medium provides a means for predicting levels of contamination in these organisms. If such analyses are possible, they are strongly recommended, particularly if there are aquatic environments in the areas subject to radioactive contamination. (S. I. Auerbach)

WASTE DISPOSAL BY INCINERATION

Disposal of radioactive solids has long been recognized as a problem in all nuclear activities. One of the early approaches for combustible waste was incineration to reduce the original volume to a relatively small volume of ash. Investigation of this possible use of incinerators was in progress by 1948.[16] In 1950, Argonne, Los Alamos, Knolls, and Mound Laboratories had incinerators in operation, under construction, or had investigated them.[17] Also, at about this same time, the Combustion Research Section, U. S. Bureau of Mines, Pittsburgh, Pennsylvania, designed a special incinerator[18] for the disposal of radioactive waste materials from hospitals and research laboratories.

During the decade 1950 to 1960, considerable work on incinerators was reported from a number of sites. In many cases the documents were completion reports for activities begun during the early years. As an alternative solution to incineration, however, one contractor[19] found that by careful sorting to eliminate miscellaneous solids and to separate the small amount of highly radioactive material, the remaining "paper" could be safely baled without serious air contamination. Baled material was shipped to the Lake Ontario Disposal Site for storage in AEC custody, at a cost of $1.56 per cubic foot, which included collection, baling, and disposal. Since incineration gives a net volume reduction in comparison with baling of only 25% when nonburnable items are included, the controlling factor is the large capital requirement for incineration ($50,000 compared with $4550 for baling). Consequently, incineration of combustible radioactive wastes has never been widely practiced, particularly at relatively small installations.

The Air Cleaning Laboratory of the Harvard School of Public Health designed a new air-cleaning system for the incinerator developed by the Bureau of Mines for the AEC. Tests of the incinerator revealed that operational and leakage problems made design changes necessary for practical application of the unit.[20] A small incinerator, designed to process waste containing >200 μc/kg and incorporating the required design changes, was then built and tested, and costs for commercial production are now being developed. The new design utilizes the tangential overfire air-supply system of the Bureau of Mines system, but a number of new features have been added to overcome disadvantages which came to light in the tests. For example, side charging, with predrying in the combustion chamber, allows continuous operation, and firebrick lining of the combustion chamber and of an afterburner permits higher temperatures and more complete combustion. An incinerator incorporating this advance design is to be built for the Nuclear Defense Laboratory, Army Chemical Center, at Edgewood, Maryland.

There have been few new incinerator installations at nuclear plants within recent years. An installation at the UKAEA Copenhurst Research Establishment has been described[21] that processes 550 gal of mixed radioactive oil wastes and 15 tons of general factory refuse per week (about 14 gal of liquid and 750 lb of refuse per hour). The furnace is lined with firebrick to permit a relatively high combustion temperature.

An incinerator has been used to destroy uranium-contaminated paper, polyethylene bags, and other containers, and paper clothing worn by workers in a fuel-element-fabricating plant.[22,23] This installation uses three combustion chambers, followed by a spray chamber, prefilters, and absolute

filters. Large savings are reported for on-site incineration in comparison with handling by AEC waste disposal licensees. Segregation of plastic material, rubber gloves, tape, and similar items from the waste greatly increases filter life and thus reduces cost of operation.

An incineration unit at the Rocky Flats Plant of the AEC also used the three-chamber design.[24] The offgas cleaning system consists of a slag-wool roughing filter, a tar trap, and absolute filters. The waste handled consists of rubber, plastics, rags, paper, and even ion-exchange resins containing 20 to 50% water, packed in polyethylene bags, 3 lb per bag. The entire operation is glove-box enclosed. Eight AEC filters are required for each 16 hr of operation (approximately 1000 lb of waste), but the system is reported to be feasible because of plutonium recovery from the ash and reduced disposal costs of the ash as compared with unburned waste.

Simple destructive incineration is carried on at a number of locations. The wastes burned are not considered to have sufficient possibility of contamination to justify a complete air-cleaning system.[25] These incinerators are typically installed at fuel-element-fabrication sites and at small, experimental reactors.

At the Chalk River Laboratories of Atomic Energy of Canada Limited, there is being installed a 200-lb/hr incinerator to handle contaminated combustible waste.[26] This is a two-chamber incinerator, ceramic lined, that operates at 2000°F. The exhaust air is cleaned by a bag house followed by absolute filters.

It may be concluded that incineration has so far proved to be of limited usefulness in the disposal of low-level solids. Where the solids

are contaminated with uranium or plutonium, incineration may be useful because it makes possible recovery of fissionable material from the ash. Where no material is to be recovered and surficial burial is available at reasonable costs, the capital investment and operating costs of an incinerator and associated air-cleaning equipment make incineration less desirable than competitive methods of disposal. (F. M. Empson)

DEVELOPMENT OF ENVIRONMENTAL SURVEY PROGRAMS

In retrospect, present programs of radiation control have been developed in the face of unprecedented problems. When the atomic energy project was initiated in 1942, "some experts believed that nuclear reactors would create so much powerful radiation that it would be impossible to protect the operators by any means." There was similar uncertainty and concern about the methods by which the escape of radioactive materials to the environment could be prevented and the public protected against the hazards of the unique industrial wastes. Past experience with the management and disposal of toxic wastes was not directly applicable, and experience with radiological hazards had been mainly in connection with medical usage of x-rays and radium.[27]

In spite of these problems, the Manhattan Engineer District (MED) established a tradition of radiological safety which has been passed on to the U. S. Atomic Energy Commission (AEC) and is now accepted by virtually all the agencies that are concerned with applications of nuclear energy. Conservative standards of permissible radiation exposure were developed early and enforced. Despite a lack of exact knowledge of radiation hazards and of the behavior of radioactive materials in the environment, the AEC adopted and made its contractors responsible for these standards, which were believed to ensure against harm from radiation.[27]

The original policy of the MED, the specific requirements of the Atomic Energy Act of 1946, with later amendments, and the very real public concern with radiological hazards have all demanded safety in the atomic energy program. They have committed both the AEC, as well as its

contractors and licensees, to the very greatest care in the control of radiation inside plants and laboratories and, especially, in the release of potentially dangerous materials outside these installations. There is evidence that the AEC and its contractors have accepted and met these responsibilities admirably. Semiannual (now annual) reports by the AEC to the Congress have given details of the radiation exposure and accident record in the MED project and in Commission-sponsored operations, and surveys by outside authorities have substantiated the claim that the record is "without parallel in industrial history."[27-29] For example, up to 1950, injuries of all kinds in AEC operations, including construction, averaged less than half the rate for all American industries,[27] and the radiation safety record was better than that enjoyed by any other organization.[29]

Early Environmental Monitoring Programs

Dispersal of contaminants in the environment — water, air, soil, vegetation, and animals — may result from deliberate routine discharges of radioactive wastes or from accidental releases of dispersable radioactive materials. Broadly speaking there are two methods of control of radioactive wastes: (1) concentration and storage so that dispersal is prevented and (2) dispersal and dilution by mixing the material with enough nonradioactive diluent (for example, air or water) to ensure that concentrations in the environment will always be below harmful levels.[27]

The prevention of excessive contamination of the environment depends upon control of radioactive materials at the source. Environmental monitoring serves as a check on the effectiveness of the control measures

and as a basis for evaluating the degree of potential hazards from environmental exposures. The nature and extent of environment monitoring that is necessary at a particular site are influenced by the quantities of radioactive materials being handled or in storage, certainly of the ability to retain the materials at the source and to control both routine and accidental discharges, and the number and distribution of the population that might be affected by excessive releases.

In the early years of the atomic energy program, first attention was given to measurement and evaluation of the materials handled at each particular installation and to inventories of the materials in storage and released routinely. From these data, probable concentrations in the environment and potential hazards to people were estimated. Where the quantities of material and potential hazards were large, as at the Hanford Works, environmental monitoring of the environs, in some cases to distant points, was developed rapidly; but, where relatively smaller amounts of materials and potential hazards existed, as at the Oak Ridge National Laboratory, environmental monitoring beyond the plant boundaries was less extensive and developed more slowly. Examples of the early programs at Hanford and ORNL are summarized below.

During the period, 1947 to 1950, the operating contractor at Hanford was expanding the environmental program with emphasis on three major aspects: (1) area monitoring to discover as precisely as possible where radioactive materials discharged from the plants were going, (2) research to determine as precisely as possible the composition and behavior of the radioactive wastes and their effects on plant and animal life, and (3) installations of facilities to reduce or eliminate the release of

radioactivity to the environment. By 1950 seven site surveyors were continuously in the field in the Hanford regions, some with mobile laboratories, measuring radioactivity and collecting samples of air, water, earth, vegetation, and animal life. They traveled scheduled routes, some covering points 100 miles away. Their travel amounted to 10,000 miles or more per month with more than 5000 measurements. A typical month's record showed 600 samples of river and well water, 4000 air samples and readings, 500 samples of vegetation, soil, and mud, and 15 surveys of special areas with portable instruments. At site survey headquarters, maps and charts were kept current to show the month-by-month distribution of radioactive materials in the air, water, and earth around the Hanford reservation and also natural background and dispersed radioactivity at sources of selected points outside the reservation.[27]

Sampling of the water of the Columbia River, which receives large volumes of reactor cooling water containing induced radioactivity, was begun in 1945. In 1950 the site survey group was collecting weekly samples from eleven points on the Columbia River, five of the Yakima, and less frequently from points as far downstream as the mouth of the Columbia. In addition to monitoring, intensive field studies and biological laboratory experiments were developed. These studies were designed to gain knowledge of the concentrations and effects of specific radionuclides in aquatic food chains, particularly in salmon and other important food fishes.[27,30] It was found, for example, that plankton could accumulate phosphorus in concentrations 200,000 times that of the surrounding water, algae 50,000 times, and fish over 100,000 times.[27] During a two-year period, 1951–1953, an extensive survey of the Columbia River to define

its hydrological, physical, chemical, and biological characteristics in relation to Hanford operations and waste disposal was made by the U. S. Public Health Service.[31]

Water-monitoring and air-monitoring procedures at the Oak Ridge National Laboratory (then Clinton Laboratories), prior to 1946, were reviewed in two reports by Parker. The report on water monitoring deals mainly with methods of measurement and interpretation of radioactivity in the waters and muds of White Oak Creek and White Oak Lake and contains calculations of the probable concentrations in the Clinch River from routine discharges or from releases at the Laboratory, as well as those from floods that might scour out the activity in the sediment in White Oak Lake. A maximum permissible discharge at White Oak Dam of 5 curies per day was considered reasonable.[32] In later years, permissible discharges were decreased, and water monitoring was extended to include routine collection and analysis of water samples near the mouth of the Clinch River. Since 1952 the ORNL area-monitoring group has made annual surveys to evaluate the reactivity in the bottom sediments of the Clinch and Tennessee Rivers, employing both underwater radiation measuring instruments and laboratory measurements of concurrently collected mud samples.[33,34]

The 1946 report by Parker on air monitoring shows that even prior to that time an extensive system had been developed for measuring the concentrations of beta, gamma, and alpha activity in the discharges from the Laboratory stacks and in the atmosphere at distances of up to about 15 miles from the site. At that time, also, continuous air monitors (integrons) were being operated in buildings, and ionization chambers,

electrostatic precipitators, film meters, and standard gas-sampling instruments were in regular use at numerous places on-site and off-area to monitor routine releases of stack gases and to make spot checks or special investigations of areas suspected of temporarily increased activity.[35] Since 1946 the procedures for air monitoring have been improved and the area studied has been extended to include more distant points, but there has been no basic change in the program. The many problems of contamination of the air have been studied in principal and in detail to gain more exact knowledge of the phenomena involved and to refine the monitoring techniques.[28,33]

A program for monitoring the amounts and kinds of radioactivity present in the environment is essential for assurance that the environment is protected and that control at the source is operating effectively; such programs have been established at all major installations. The programs vary from installation to installation according to the problems present in each case. At the younger installations, monitoring systems were established in advance of plant operation to provide data on the pre-existing background, which varies according to local conditions. Consultation and cooperative assistance for these programs have been provided by various agencies and individuals, including the U. S. Weather Bureau, the U. S. Geological Survey, the U. S. Public Health Service, and specialists from universities. Advisory boards also gave assistance to the AEC in particular aspects of certain of these programs.[28]

Extension of Programs of Environmental Control

Since 1948 the programs for protection against radioactivity in the environment have been greatly expanded and strengthened by (1) extension

of radiological monitoring to include the basic elements that experience has shown necessary in other fields of environmental sanitation and (2) extension of older established programs for control of nonradioactive hazards to include the hazards of radioactivity. Prior to 1943, well-defined sanitary engineering programs had been developed for environmental control of various health hazards; for example, organisms of disease, toxic industrial materials, and physical conditions, such as defective lighting, inadequate heating and ventilation, noise, and vibration. The basic elements of these programs included (1) field and laboratory investigations to define environmental conditons, hazards, and effects of exposure; (2) research; (3) development of control measures; (4) establishment of standards; and (5) promotion of control measures backed by mandatory regulations when necessary. These studies employed the knowledge of various professions – sanitary engineers, chemists, biologists, medical scientists, hydrologists, and others – in a coordinated program. Field investigations and surveys alone, although of primary importance, were ineffective unless combined with the other basic elements outlined above.

A pattern of environmental control suited to the wartime operation was developed in the MED project and continued during the early years of the AEC program. Safe methods were devised and used, whatever the cost, in order to ensure safety for workers and the public. There was little opportunity for research and long-range planning. Environmental measurements were almost solely for monitoring the releases of radioactivity; and, because of security restrictions, there was a minimum of intercommunication between health physicists in the atomic energy program and workers in the other health professions.[29]

Some sanitary engineers, and others in the nuclear energy program, saw the need to extend the programs for control of radiation in the environment to include, on the one hand, more basic and applied research and, on the other hand, to increase the participation of those established Federal, state, and local public health agencies which have responsibility for protection of the public health. At a seminar on radioactive waste disposal, sponsored by the AEC in 1949, Wolman summarized these viewpoints.[36] He pointed out the peculiarities and similarities of radioactive waste as compared with other waste and emphasized the gaps in the program of radiological control up to that time. Addressing his remarks particularly to the sanitary engineering profession and the AEC, he recommended that sanitary engineers familiarize themselves with the nuclear energy industry, develop research and special investigations in the entire field of radioactive wastes and of water and sewage treatment, atmospheric dilution, and disposal of liquid and solid radioactive materials, and extend their normal responsibilities into the areas of nuclear research and of commercial nuclear power production.[36]

A significant stimulus in the development of the programs for the control of radiation in the environment was a survey of all major AEC installations made by the Safety and Industrial Health Advisory Board in 1947 and a report to the AEC in 1948.[29] This Board of 12 members, selected from fields related to health and safety, made specific recommendations for improvements in such AEC programs as fire protection, employee hygiene and health, and public health. The Board recommended that environmental radiation monitoring be continued and extended and emphasized the prime importance of developing a comprehensive program, including field surveys

and laboratory research. For example, one of its five principal recommendations was the following:

> "Promptly study and correct radiation and other hazards to public health, both immediate and long range. Make greater use of the cooperation of public health authorities, and of outside consultants generally."

Under the heading of "Environmental Sanitation" the Board listed 11 "problems demanding immediate investigation." These included: laboratory and field study of the problem of the disposal of radioactive and toxic wastes; determination of the extent to which such wastes have traveled in surface water courses and have deposited on the banks or beds of waterways; determination of the air travel of such wastes from vents or stacks and their effects on plant and animal life in the area affected; development of equipment and sampling techniques capable of maintaining and recording tests in air, soil, water, and exposed surfaces with a minimum of labor and time; study of the biology, geology, meteorology, and hydrology of areas within and surrounding present and proposed AEC operations; and study of the effects of wastes on the biological processes common to sewage treatment and disposal and the self-purification of streams. The Board further recommended the establishment of a health protection division in the AEC and the immediate employment of a sanitary engineer with responsibility for initiating, developing, and administering, as summarized above, an expanded program of environmental sanitation at the operational atomic energy sites.[29]

The detailed recommendations of the Board regarding public health and environmental sanitation provided a well-conceived plan which, in general, was accepted and followed by the AEC and which has also served

as a guide for programs of other agencies. A staff sanitary engineer was employed by the AEC in November 1947, a position which evolved into the Environmental and Sanitary Engineering Section in the AEC Division of Reactor Development. The primary goal of this section has been the advancement of research and development and the appraisal of practices in the disposal of radioactive and toxic wastes.[37] From the first, this work included the cooperation of the U. S. Geological Survey, the Weather Bureau, and the Public Health Service, whose staffs have long experience with similar problems in other industries. Research programs were developed in nearly all the major AEC operations, including Hanford, ORNL, Los Alamos, KAPL, BNL, and, in later years, the National Reactor Testing Station and other AEC installations. Noteworthy among the university programs is that of the Harvard Air Cleaning Laboratory, which has done pioneer work in research, development, and testing to improve air-cleaning systems and which has assisted in evaluating and solving air-cleaning problems at AEC installations. Numerous other research projects and consultant contracts were initiated.[37,38] During the period, 1948 to 1955, the research and development program was primarily concerned with high-volume low-level liquid wastes, air-borne particulate wastes, and the assessment of the effects of the relatively low levels of radioactivity in these wastes on public sewage treatment and water-supply facilities. From about 1954 to the present time, the program has been expanded to include studies on problems of high-level radioactive wastes, particularly those anticipated from future development of a nuclear power industry.

As described by Gorman,[37] the guiding policy in the AEC sanitary engineering program was to appraise the problem first and then attack

it in as direct a manner as possible. Appraisal of waste disposal problems at operating sites required extensive meteorologic, geologic, and hydrologic programs to define environmental parameters and special field surveys of radioactivity in the environment to indicate the nature and importance of particular problems. Environmental-monitoring data were useful, but intensive and specialized field surveys were necessary to guide and supplement laboratory research development.

A schematic outline presented in 1959 by H. M. Parker at the Congressional hearings on radioative waste disposal summarized the principal pathways of exposure of humans or other organisms to radioactive wastes in the environment. "The situation around any particular site must be determined experimentally for that site."[39] The major components is this outline (omitting detailed paths of exposure) are the following:

A. Radioactive Wastes in Rivers, Streams, Lakes, or Oceans
   1. Hazards related to use as drinking water
   2. Hazards arising from immersion in the water
   3. Hazards arising from close approach to the water
   4. Hazards arising from irrigation
   5. Hazards arising from biological chains in fresh water
   6. Hazards arising from biological chains in the ocean
   7. Hazards related to industrial processes
   8. Hazards related to sewage disposal

B. Radioactive Wastes Released to the Atmosphere
   1. Hazards related to inhalation
   2. Hazards from radioactive material in ambient air
   3. Hazards related to deposition of materials

C. Ground Disposal of Liquid Wastes

1. Hazards at the disposal site

2. Hazards resulting from underground travel

Types of Environmental Surveys

The general types of surveys that have been developed to aid in the evaluation and control of radioactivity in the environment include: (1) surveys of the environs of nuclear energy installations and surrounding areas; (2) background and radiological surveillance surveys by state or local agencies (most often state health departments); (3) intensive surveys for evaluation of particular features of the environment or special problems of waste disposal; and (4) regional or nation-wide network surveys to evaluate over-all conditions or special hazards (for example, fallout) as a guide for determining the type and magnitude of the radiological health programs that are needed. A more detailed summary might include many other special types of surveys and variations of these general types according to particular circumstances. Examples of the four types outlined above will be mentioned and references given to sources of more detailed information.

A good example of the first type of survey is given by Nehemias et al[40] in the description of a recent preoperational survey of the radiologic environment of a nuclear power plant being constructed near Monroe, Michigan. Plans call for continuing the measurements of environmental radioactivity after routine operation of the plant has begun. Both the preliminary survey in 1958-1959 and the subsequent continuing investigations were conducted for the power company by the National Sanitation

Foundation of Ann Arbor, Michigan. The purpose was to provide background information which, in conjunction with continuing measurements of environmental radioactivity, will provide a sound basis for evaluating the impact of reactor operation on the radiologic environment. During 1959, 1221 samples were collected from the reactor site and surrounding communities including samples of surface water, ground water, sediment, aquatic plants and algae, rain water, milk, mammals, fish, earth, dust, grass, trees, fruits, and vegetables. Analyses included gross-beta activity of all samples, radiochemical procedures to determine levels of specific radionuclides, with particular emphasis on Sr^{90}, and identification of principal radionuclides by gamma spectrometry. The results are summarized in tables and charts. Collection and analyses of environmental samples for gross-beta activity are also made in the same general area covered by this survey by several government agencies, including the Detroit city health department, the Michigan state health department, the Ohio state health department, and the Michigan Water Resources Commission.

One of the earliest extensive preoperational surveys was initiated in June 1951 at the Savannah River Plant. This work, as well as a design for a regional survey program, has been reported by Horton.[41,42] In general, the nature and scope of the Savannah River survey and of other well-designed surveys for large operations are very similar to the program for the Michigan plant described above. Many annual reports on continuing postoperational surveys of the environs of sites in the United States and foreign countries are available in the literature. Typical is the 1958 report from Hanford.[43] In addition to listing sampling locations, analyses, and results, this report includes an estimate of the annual

exposures of the population from various features of the environment so that these may be compared with maximum permissible exposure values. A report by Mawson et al.[44] presented at the Monaco Conference on Waste Disposal in 1959 includes four papers describing waste management and monitoring at Chalk River, Canada, including a description of sampling devices and techniques.

With regard to surveys of the second type, it is a long-established principle that the basic legal responsibility for the protection of the environment rests with the various states. In recent years an increasing number of state health departments, and a few other state agencies and city health departments, have developed radiological health programs, including environmental surveys for background and surveillance of radioactivity. The earlier and more elaborate programs were developed in the larger states with major nuclear energy problems, such as Illinois, New York, Pennsylvania, and California. The Illinois program was described recently by Klassen and French.[45,46] Since the early 1950's, Illinois sanitary engineers have maintained contacts and developed cooperative working relationships with the AEC, the Argonne National Laboratory, and other nuclear energy agencies. State-wide control has been established over the use and disposal of radioisotopes by nearly 300 users licensed by the AEC, and over 8000 x-ray machines registered in the state are being surveyed and checked for potential hazards. Determinations of radioactivity have been added as a regular part of the analyses of samples from over 250 stations on Illinois streams from which water is regularly collected as part of the water quality control program. Additional special samples and analyses are taken as considered necessary in the environs of nuclear

energy sites. A program of continuous air monitoring and sampling of particulate matter in selected areas has been begun, and plans have been made for future extensions.

An extensive program of background surveys and environmental surveillance has been carried out by the New York State Health Department since 1953.[47] This program has provided for the collection of samples of air, water, precipitation, and milk. The purpose of the program is to collect basic background data, record fallout radioactivity in rainfall and dust, and follow the level of radioactivity in public-water supplies and milk supplies. Prior to 1959, gross-beta analysis was used. Since February 1959, radiochemical analyses have been used to determine alkaline earths, some of the rare earths, and radium. Water-sampling points and supply systems have been selected, and the number is being increased so that the samples will represent state-wide conditions. Milk samples are taken monthly from seven stations selected to represent different geographical parts of the state.

An example of surveys of the third type, surveys for evaluation of special problems, is provided by a recent comprehensive investigation of radioactive pollution of the Animas River by refinery wastes of uranium ore, as reported by Tsivoglou.[48] These studies lasted a full year, including a two-month period of intensive study and an additional ten months of less intensive sampling. Samples were obtained of river water, muds, and biota along a 60-mile stretch of the Animas River, of locally grown crops, and of local milk supplies. Drinking water supplies and various other possibilities of environmental exposure were investigated. Estimates were made of the possible radiation exposures from Ra^{226} and Sr^{90}

in the environment. The survey results showed a need for pollution abatement and other protective measures, and these were quickly installed. Subsequent surveys showed that satisfactory protection had been achieved.

A comprehensive study of the Clinch River was initiated at the Oak Ridge National Laboratory in February 1960, with participation by five state and federal agencies in addition to the Laboratory. The purpose of this survey study of the Clinch River below ORNL is to obtain fundamental information on the physical, chemical, and biological dynamics of this stream, which receives contamination from discharges of large-volume low-level radioactive wastes from several sources. The area monitoring program on the Clinch River is being continued and amplified to aid in this special study.[49]

An example of the fourth type of survey is the nationwide radiation surveillance program of the U. S. Public Health Service, sponsored jointly by the Public Health Service and the AEC. The actual sampling is done cooperatively by the Public Health Service and state and local health departments. The survey was designed as a primary alert system to detect and evaluate fallout and also to indicate levels and changes in environmental radioactivity from other sources. It now includes an established network of locations for sampling the atmosphere, surface water, and milk. The basic program is occasionally suplemented by additional samples of air, water, and milk or of other materials by certain "quick-check" special determinations and by radiological analyses of samples collected for other purposes. Reports of the results are included in monthly and quarterly issues of Radiological Health Data.[50]

Atmospheric samples for gross-beta analysis are taken daily from continuous sampling at 58 stations in the United States (including Alaska) and Canada. Also, radiological determinations are included in analyses of weekly or monthly air samples from about 200 stations in the general air pollution sampling program. Surface-water samples are collected on a weekly schedule from 93 established water quality stations and analyzed for specific radionuclides. Twelve of these stations represent metropolitan water supplies. Samples of market milk are obtained from approximately 60 established stations, 18 on a daily sampling schedule. These daily composites represent at least 90% of the milk sold in the particular community. The milk and food program is supplemented by sampling 10 raw milk sources, by grab samples for spot checks of fresh foods (leafy vegetables, etc.), and by quick checks of milk for radioiodine before beginning complete analyses.[51] (Roy J. Morton)

References

1. U. S. Dept. Health, Education and Welfare, Radiological Health Data, 1(1): 5-6 (April 1960).

2. A. Pelletier, Environmental Surveys for Nuclear Facilities, Nucleonics, 17(1): 56-59 (1959).

3. J. H. Horton, Jr., Design of a Regional Survey Program, USAEC Report DP-253, E. I. du Pont de Nemours & Co., November 1957.

4. Federal Radiation Council, Background Material for the Development of Radiation Protection Standards, Staff Report No. 2, September 1961.

5. W. B. Silker, Strontium^{90} Concentrations in the Hanford Environs, USAEC Report HW-55117, Hanford Atomic Products Operations, May 20, 1958.

6. J. S. Olson, Ecological Sampling and Meteorological Calculation of Fallout on Forests near Oak Ridge, USAEC Report ORNL-3181, Oak Ridge National Laboratory, September 6, 1961.

7. World Health Organization, Method of Radiochemical Analysis, W. H. O. Technical Report Series 173, 1959.

8. F. J. Bryand, A. Morgan Chamberlain, and G. S. Spicer, Radiostrontium Fallout in Biological Materials in Britain, British Atomic Energy Authority Report AERE HP/R 2056, 1957.

9. J. H. Harley and I. B. Whitney, Health and Safety Laboratory Analytical Methods, USAEC Report NYO-4700, New York Operations Office, 1957.

10. J. B. Hursh (ed.), Chemical Methods for Routine Bioassay, USAEC Report AECU-4024, Rochester, New York, 1958.

11. L. E. Glendenin, Radiochemical Studies: The Fission Products, 3: 1460-64, edited by C. D. Coryell and N. Sugarman, McGraw-Hill Co., Inc., New York, 1951.

12. F. E. Butler, Sr^{90} Monitoring at the Savannah River Plant, USAEC Report DP-473, E. I. du Pont de Nemours & Co., May 1960.

13. D. F. Peppard, G. W. Mason, and S. W. Moline, The Use of Dioctyl Phosphoric Acid Extraction in the Isolation of Carrier Free Y^{90}, La^{140}, Ce^{144}, Pr^{143}, and Pr^{144}, Jour. Inorg. Nucl. Chem., 5: 141-146 (1957).

14. G. G. Manov and D. L. Eads, Techniques of Large Scale Processing of Counting Data, in Symposium on the Metrology of Radionuclides, pp. 103-115, International Atomic Energy Agency, Vienna, Austria, 1960.

15. F. B. Higgins, Jr., W. N. Grune, B. M. Smith, and J. G. Terrill, Jr., Methods for Determining $Radon^{222}$ and $Radium^{226}$, Jour. Amer. Waterworks Assoc., 53(1): 63-74 (1961).

16. R. J. Schauer, Pilot Plant Work on Solid Burnable Waste Disposal for Mound Laboratory, USAEC Report MLM-262, Mound Laboratory, November 1, 1948.

17. A. T. Rossano, Jr., L. Silverman, and M. W. First, A Preliminary Field Survey of Air Cleaning Activities at AEC Facilities, September–November 1950, USAEC Report NYO-1580, New York Operations Office, 1951.

18. R. Dennis, L. Silverman, C. E. Billings, D. M. Anderson, W. R. Samples, H. M. Donaldson, Jr., and P. Drinker, Air Cleaning Studies Progress Report, July 1, 1954 to June 30, 1955, USAEC Report NYO-4611, New York Operations Office, October 15, 1956.

19. R. E. Larson and R. H. Simon, Solid Waste Disposal at the Knolls Atomic Power Laboratory, USAEC Report KAPL-936, Knolls Atomic Power Laboratory, June 15, 1953.

20. R. Dennis, F. L. Muller, E. Kristal, and L. Silverman, Sixth AEC Air Cleaning Conference, July 7-9, 1959, USAEC Report TID-7593, pp. 344–364, October 1960.

21. R. F. West, Disposing of Radioactive Waste by Fire, Industrial Review Africa, Supplement, Atomics and Energy, 11(10): 98 (April 1960).

22. R. C. Thorburn and R. J. Chandler, An Incinerator for Uranium Contaminated Wastes, Industrial Water and Wastes, 6(2): 46-48 (1961).

23. R. C. Thorburn and R. J. Chandler, Incinerator Reduces Volume of U-Contaminated Paper, Nucleonics, 19(5): 86-87 (May 1961).

24. R. Dennis and L. Silverman, Seventh AEC Air Cleaning Conference, October 10-12, 1961, Brookhaven National Laboratory (Proceedings to be published).

25. P. Loysen, Seventh Air Cleaning Conference, October 10-12, 1961, Brookhaven National Laboratory (Proceedings to be published).

26. J. Neil, Atomic Energy of Canada Limited, personal communication to F. M. Empson, Oak Ridge National Laboratory (September 29, 1961).

27. Control of Radiation Hazards in the Atomic Energy Program, Eighth Semiannual Report to Congress, U. S. Atomic Energy Commission, (July 1950).

28. Radiation Safety and Major Activities in the Atomic Energy Programs, July–December 1956, U. S. Atomic Energy Commission, January 1957.

29. Excerpts from the Report of the Safety and Industrial Health Advisory Board, USAEC Report ORNL-CF-48-5-160, Oak Ridge National Laboratory, April 2, 1948 (Declassified, June 17, 1957).

30. Richard F. Foster, Research and Development Programs Related to the Disposal of Reactor Effluent to the Columbia River, Industrial

Radioactive Waste Disposal, Hearings before the Special Subcommittee on Radiation of the Joint Committee on Atomic Energy, Congress of the United States, Vol. 2, pp. 1018-1047, January 28, 29, and 30; February 2 and 3, 1959.

31. G. G. Robeck, C. Henderson, and R. C. Palange, Water Quality Studies on the Columbia River, U. S. Public Health Service, Robert A. Taft Sanitary Engineering Center, Cincinnati, Ohio (1954).

32. H. M. Parker, Review of Water Monitoring Procedures at Clinton Laboratories, U. S. Atomic Energy Commission, Oak Ridge, Tennessee, USAEC Report MDDC-401 (Declassified, October 28, 1946).

33. K. Z. Morgan, Waste Management Program at Oak Ridge National Laboratory, Industrial Radioactive Waste Disposal, Hearings before the Special Subcommittee on Radiation of the Joint Committee on Atomic Energy, Congress of the United States, Vol. 1, pp. 445-460, January 28, 29, and 30; February 2 and 3, 1959.

34. W. D. Cottrell, Radioactivity in Slit of the Clinch and Tennessee Rivers, USAEC Report ORNL-2847, Oak Ridge National Laboratory, November 18, 1959.

35. H. M. Parker, Review of Air Monitoring Procedures at Clinton Laboratories, U. S. Atomic Energy Commission, Oak Ridge, Tennessee, MDDC-471 (Declassified November 19, 1946).

36. Abel Wolman, Appraisal of the Sanitary Engineering Problems in the Atomic Energy Program, Proceedings of Seminar on Disposal of Radioactive Wastes, January 24-25, 1949, U. S. Atomic Energy Commission, Washington, D. C.

37. Arthur E. Gorman, Sanitary Engineering Objectives in the Atomic Energy Industry, Proceedings of Seminar on Sanitary Engineering Aspects of the Atomic Energy Industry, Cincinnati, Ohio, December 6-9, 1955, USAEC report TID-7517, Part Ia, pp. 4-9, October 1956.

38. J. A. Lieberman, AEC Waste Disposal Research and Development Program, Proceedings of Seminar on Sanitary Engineering Aspects of the Atomic Energy Industry, Cincinnati, Ohio, December 6-9, 1955, USAEC Report TID-7517 (Part 1a), pp. 10-18, October 1956.

39. H. M. Parker, Exposure Pathways of Released Radioactive Wastes, Industrial Radioactive Waste Disposal, Hearings before the Special Subcommittee on Radiation of the Joint Committee on Atomic Energy, Congress of the United States, Vol. 3, pp. 2359-2366, January 28, 29, and 30; February 2 and 3, 1959

40. John V. Nehemias, G. H. Whipple, and N. C. Kothary, Environmental Radioactivity Prior to Nuclear Power Plant Operation, Journal American Water Works Association, 52(10): 1291-1302 (October 1960).

41. J. H. Horton, Radioactivity in the Environs of the Savannah River Plant, January to July 1954, USAEC Report DP-92, E. I. du Pont de Nemours & Co., November 1954.

42. J. H. Horton, Design of a Regional Survey Program, USAEC Report DP-253, E. I. du Pont de Nemours & Co., November 1957.

43. B. V. Andersen, Hanford Environmental Monitoring Annual Report — 1958, USAEC Report HW-61676, Hanford Atomic Products Operation, August 27, 1959.

44. C. A. Mawson et al., Waste Management and Monitoring at Chalk River, Atomic Energy of Canada Ltd., Chalk River, Ontario, CRL-59, November 1959, (AECL No. 987).

45. C. W. Klassen, Environmental Aspects as Applied to the State of Illinois, Proceedings Second Sanitary Engineering Conference, Radiological Aspects of Water Supplies, January 27 and 28, 1960, University of Illinois, Circular No. 69, pp. 16-19.

46. Robert R. French, Monitoring Experience of the Illinois Department of Public Health, Proceedings Second Sanitary Engineering Conference, Radiological Aspects of Water Supplies, January 27 and 28, 1960, University of Illinois, Circular No. 69, pp. 62-64.

47. Sherwood Davies, Radiological Monitoring of the Environment in New York State, Health News, 37(10): 4-13 (October 1960), New York State Department of Health, Albany.

48. E. C. Tsivoglou et al., Estimating Human Radiation Exposure on the Animas River, Journal American Water Works Association, 52(10): 1271-1290 (October 1960).

49. R. J. Morton (ed.) et al., Status Report No. 1 on Clinch River Study, USAEC Report ORNL-3119, Oak Ridge National Laboratory, July 27, 1961.

50. Radiological Health Data, Monthly and Quarterly Reports, PB-161371, U. S. Public Health Service, Washington, D. C.

51. Personal communication from U. S. Public Health Service to R. J. Morton, October 26, 1961.

VI. CURRENT EVENTS

HOMOGENEOUS REACTOR EXPERIMENT NO. 2 OPERATING EXPERIENCE

(Material held over from draft of Vol. 2, No. 4 to be inserted here; see draft pages 223 through 238 and references 1 through 6 on page 274.)

THE SL-1 INCIDENT

(Material held over from draft of Vol. 3, No. 2 to be inserted here; see draft pages 170 through 186 and references 20 through 28 on pages 211 and 212.)

RADIATION INCIDENTS AT THE MTR AND ETR DURING 1959

Unplanned events which occurred in 1959 at the MTR and ETR and which caused evacuation or severe restrictions of entry of significant reactor operating areas were reported in the hazards evaluation of the ETR II (later designated ATR) proposed site.[1] Incidents occurring in nuclear energy operations are of particular interest to everyone concerned with nuclear safety, and as information of this nature becomes available it is discussed in Nuclear Safety.

Both the MTR and the ETR are water-moderated, beryllium-reflected, enriched-uranium reactors using plate-type fuel elements. They are operated and maintained for the AEC by the Atomic Energy Division of the Phillips Petroleum Company. The MTR, which has a design power level of 40 Mw, was started up in 1952, while the ETR, with a design level of 175 Mw, was started up in 1957. The reactors provide space to carry out experimental programs with thermal-neutron fluxes varying from 10^{12} to 5×10^{14} neutrons/cm^2·sec. The engineering complexity of the experiments many times approaches or exceeds that of the reactors. In addition to the prime objective of developing radiation damage information from these experiments, the reactors also produce neutrons for solid-state studies and isotope production. It has been stated that[2]

> "The basic philosophy governing the operation of the MTR and ETR is to conduct operations in as safe and expeditious a manner as possible consistent with the overall objectives for which the reactors were constructed. This philosophy requires the acceptance of some calculated risks but the probabilities of major incidents are kept below what could be considered the taking of chances with respect to the reactors or their associated experiments. This method of operation is made mandatory by an ever increasing tendency to work at higher

and higher temperatures and pressures and to subject materials to stresses under conditions not hitherto explored."

Reactor Buildings

The MTR is located in a 130-ft^2, 80-ft-high building that contains a 17-ft-high basement. In addition to the reactor structure, the building contains experimental facilities and a storage canal. All experimental facilities that are located outside the reactor are in shielded cubicles in the basement. The reactor structure, including shielding, is approximately 35 ft by 35 ft by 25 ft high and is located in the center of the building. Most irradiated material leaves the reactor tank downward through a discharge mechanism to the canal. The head of the reactor is removable for access from above for fuel and experimental equipment handling. Samples for irradiation in the horizontal beam hole are handled by pulling them out into shielded containers.

The ETR building is 112 ft by 136 ft by 58 ft high above ground, and it has two complete floors below grade. The main floor of the building is relatively clear of equipment, with the top of the reactor rising 8 ft above floor level. All heavy and radioactive experimental equipment is housed in shielded concrete cubicles in the lower basement. The upper basement houses all experiment control panels. The reactor dome can be removed to move fuel elements and small experimental equipment underwater to the storage canal. Large experimental equipment is usually pulled from the experimental tube into heavily shielded carriers.

Incident Record

While none of the events that occurred at the reactors in 1959 were considered to be serious, some were of such nature that they required

evacuation or severe restrictions of entry to major operating areas of the reactor buildings and facilities. If the affected area was less than one operating floor of a reactor building, the event was classified as minor. Only the major events are enumerated here.

The major events which interrupted operations at the two reactors in 1959 are listed in Table VI-1. All the events originated at the MTR, but they all caused evacuation of the ETR, with the longest evacuation time at the ETR being 1.8 hr.

The major events at the MTR, which affected only that reactor, are listed in Table VI-2. A series of ruptures of fuel-bearing structures in the MTR in October and early November caused so many air-activity problems that the whole period of time can be classified as a major event. In addition to the major events, there were also 21 events at the MTR that caused severe restrictions to entry and occupancy of small sections of operating areas. These events are not enumerated.

The ETR major events are listed in Table VI-3. In addition, there were 18 events that were classified as minor, since they affected only small sections of operating areas.

Study of the events described in the three tables reveals that 80% of them were directly due to releases of activity from experimental assemblies in the reactors. The releases were due to leaks or ruptures or occurred as the experimental assembly was being removed for disassembly.

Conclusions

Each of the reactor sites has had radiation levels for short periods of time which have exceeded desirable limits. These events have not,

Table VI-1. MAJOR EVENTS AT MTR THAT AFFECTED OPERATION AT BOTH THE MTR AND ETR IN 1959

Date and Time 1959	Description and Cause	Location	Radiation Level (rad/hr)	Contamination Level on Smear (dis/min)	Units of Air Activity, One Unit = 3×10^{-9} mc/cm^3	Major Isotopes or Half-Life	Restricted Entry or Evacuation	Reactor Down Time	Remarks
January 8, 1115-2200	MTR experimental assembly ORNL-44-1 fission-gas purge line plugged and caused diffusion of gases up feed line and out through a leak in the line	Gas feed line from HB-3 to NW corner of MTR	15 000 at surface of line	Short-lived	>1000	Xe^{133}, Xe^{135}, Kr^{88}, Kr^{89} and daughters	Total Evacuation of MTR reactor building and complex for 8 hr and ETR for 1.5 hr	MTR down 52 hr for removal of experimental equipment and evacuation; down time of the ETR was extended by 1.5 hr	
March 3, 2100	Two walls on the container of MTR experiment ANL-43-1 ruptured and allowed a NaK-water reaction which caused a vent line to split and release fission gases into the reactor building	MTR top	0.5 (max) around reactor main floor	6×10^6 (max) on reactor main floor	>1000	Na^{24}, I^{131}, I^{132}, I^{133}, I^{135}, I^{139}, Te^{132}, noble fission gases and daughters	Evacuated MTR for 27 hr, except for personnel wearing respiratory protection; evacuated ETR for 1.8 hr	MTR down 48 hr for evacuation and for removal of experimental equipment; extended shutdown of ETR for 1.8 hr	Short half-life material decayed off rapidly; air activity removed by normal ventilation after experimental equipment was discharged
October 24	Air activity from ruptured capsules at MTR	MTR tank	Minor	Negligible	>1	~30 min	Evacuated ETR, TPP, and maintenance building for 45 min	Extended ETR shutdown for 45 min	General activity rise over entire area
November 1, 0420	Process water containing iodine leaked through a packing gland in a lead-irradiation MTR experiment onto the reactor top and caused air activity as the iodine vaporized	MTR top	Minor	2000, small area	6 to 8 on reactor top and main floor	I^{131}, I^{132}, I^{133}	Evacuated MTR top and main floor for 40 min; evacuated ETR for 45 min	Reduced MTR power for 3.2 hr while repairing leak; extended ETR shutdown by 45 min	Personnel wore positive air-supply respirators while repairing leak; ETR had intermittent air activity all day

Table VI-2. MAJOR EVENTS AT THE MTR IN 1959 THAT AFFECTED OPERATION

Date and Time 1959	Description and Cause	Location	Radiation Level (rad/hr)	Contamination Level on Smear (dis/min)	Units of Air Activity, One Unit = 3×10^{-9} mc/cm^3	Major Isotopes	Restricted Entry or Evacuation	Reactor Down Time	Remarks
February 24, 1730	High gamma heating caused the evolution of radioactive gases from GE-ANP-1 which escaped into the reactor building as the experimental equipment air lines were cut prior to removal of the experimental assembly	HT-1 north and south reactor faces	Minor	None	3 to 10 in reactor building	I^{131}, I^{132}, I^{133}	Evacuated the reactor building for 1.2 hr; respirators worn by those who reentered	Increased down time by ~1 hr	The element was moved to a high gamma heat zone in order to cut the air lines
April 3, 2000	Fission gases from the ruptured GEH-4-36 experimental assembly were released into the reactor building through a faulty isolation valve as the experimental equipment was being discharged	Reactor tank	230 (max)	4700 on reactor top	~900 on reactor top	Fission gases, Rb^{88} and particulate matter	Evacuated MTR reactor building for 4.5 hr	Down 19.1 hr because of evacuation and removal of GEH-4-36 and WAPD-22.2, which had also ruptured	Contamination level decayed rapidly
May 15, 2400	Fission gases from ANL-35 rupture escaped into reactor building when reactor tank was opened	Reactor top	Minor	None	5 (max)	Short-lived	Evacuated reactor building for ~15 min	Increased down time by ~1 hr	Activity removed by normal ventilation and decay
June 12, 1100-1400	Experiment GEH-4 ruptured and gave three separate releases of noble fission gases; water samples confirmed fission break	Reactor tank and top	Minor	Negligible	285 (max) in tank; 25 on the south reactor face, and 7.6 in basement	Noble fission gases, Rb^{88}, Cs^{138}	Main floor evacuated 40 min; basement evacuated 70 min; ETR critical facility evacuated 21 hr	Down time 7.1 hr; reduced power 0.7 hr	Personnel re-entered building wearing positive air-supply equipment
September 16, 1730-1830	The GEH-4 defect test (operated with reactor at power for 1.5 hr) evolved fission-product gases into the experimental loop which were released into the reactor building when the loop was opened	Reactor top	Minor	None	10-15 at reactor top 5-8 at floor	Kr^{88}, Rb^{8}	Partial evacuation of reactor top and main floor for 1 hr	Down 8.5 hr	
October 12, 0150-0315	Noble fission gases from a ruptured experiment (KAPL-2-95 and perhaps others) diffused up through the process water after shutdown and escaped into the reactor building when the reactor tank was opened	Reactor top	Minor	Negligible	5-25 in reactor building	Kr^{88}, Rb^{88}, Xe^{138}, Cs^{138}	Evacuated reactor top and main floor for 2.5 hr, except for those wearing respiratory equipment	None	Reactor tank vented to the stack to remove activity

Table VI-2 (continued)

Date and Time 1959	Description and Cause	Location	Radiation Level (rad/hr)	Contamination Level on Smear (dis/min)	Units of Air Activity, One Unit = 3×10^{-9} mc/cm^3	Major Isotopes	Restricted Entry or Evacuation	Reactor Down Time	Remarks
October and November	During October and early November a number of ruptured capsules caused so many air activity problems that the whole period can be classed as a major event				Usually <10 but on one occasion >100; duration of release usually 1 hr; but on occasion longer				The activity was usually released either from the MTR stack over whole area or from process water in process-water building
November 6, 0855-1100	Fission gases were released into building from reactor when it was opened immediately after a scram in an effort to locate which experiments were ruptured and giving off bubbles	Reactor top	0.027	4000 on reactor top	270 on reactor top, 70 on floor, 2 in wing	Cs^{138}, Rb^{89}	Evacuated building and wing for 2.1 hr	Down 22.8 hr because of removal of suspect capsules and evacuation	When the reactor was restarted, activity was again detected; on November 8 experiment NAA-47-3 was found to be ruptured
December 9, 1315	Numerous radiation alarms were triggered by direct radiation from the removal of a thermocouple lead on the reactor floor north coincident with direct radiation from the upper grid of the reactor being removed to the canal	Reactor top and main floor	20	None	None	None	Evacuated entire MTR complex for ~10 min	Down time increased by ~0.1 hr	Total evacuation signal was inadvertently tripped instead of reactor building signal

Table VI-3. MAJOR EVENTS AT THE ETR IN 1959 THAT AFFECTED OPERATION

Date and Time 1959	Description and Cause	Location	Radiation Level (rad/hr)	Contamination Level on Smear (dis/min)	Unit of Air Activity, One Unit = 3×10^{-9} mc/cm^3	Major Isotopes or Half-Life	Restricted Entry or Evacuation	Reactor Down Time	Remarks
March 17, 1730	Evacuated reactor building because of air activity caused by a flux run; activity stopped when flux run ended	Reactor tank	Unknown	Unknown	70 (max), 30 (avg)	34 min	Building evacuated for 4.5 hr	None	Personnel remaining in building wore respiratory equipment
March 23, 1900	Air activity was released when packing was removed from an ANP experiment in preparation for pulling reactor head	Reactor top	Unknown	Low	1200 over tank, 50 on reactor floor	18 min	Reactor main floor evacuated for 4 hr	None	Personnel in area of air activity wore respiratory equipment
March 31, 2015	Air activity released from under reactor dome when removed from tank	Reactor top	Low	Low	16	18 min	Evacuated building for 45 min	Extended shutdown by 45 min	Activity source was unknown
April 10, 1215	Air activity came from experiment GEEL after a pipe housing thermocouple leads broke; air entered the primary water system	Reactor tank	Low	Low	14	18 min	Evacuated building for 3.8 hr	Caused shutdown; power reduced 3.3 hr	Evacuated except for personnel wearing respiratory equipment; repaired leak with welded cap over lead connection
April 10, 1615	Air activity released on removal of reactor dome following shutdown	Reactor top	Low	Low	17	18 min	Evacuated building for 2.3 hr	Extended shutdown by 2.3 hr	Shutdown was caused by previous item
May 17, 0630	Fission break in experiment GEH-10 caused radiation field and intermittent air activity until discharged to canal	Outside experiment cubicle	1	Unknown	79 in basement, 3 on main floor	15 min	Evacuated basement for 1 hr	Work other than cleanup done during shutdown; estimated extended time, 24 hr	Scram caused by rupture; power reduced 37 min before scram
June 10–11	Intermittent leaks from the tank access nozzle of KAPL-33; air activity fluctuated with leaks	Reactor tank	Low	Low	10	20 min	Evacuated main floor several times for a total of 45 min	None	Wore respirators to repair leak with remote tool

Table VI-3 (continued)

Date and Time 1959	Description and Cause	Location	Radiation Level (rad/hr)	Contamination Level on Smear (dis/min)	Units of Air Activity, One Unit = 3×10^{-9} mc/cm^3	Major Isotopes or Half-Life	Restricted Entry or Evacuation	Reactor Down Time	Remarks
June 14, 2130	Air activity and contamination from water leaking from GEH 6 × 9 cubicle	Basement floor	0.5	2×10^5	19	63 min	Evacuated basement for 4 hr	None	For an indefinite period entry to the basement was limited to those wearing respirators
June 15, 0030	Air flush of reactor dome caused air to be blown out of hot catch tank and drain system into building	Reactor building	Low	Unknown	>1000	Kr^{88}, Rb^{88}	Evacuated building for 8 hr	Extended shutdown by 8 hr	
July 23, 0235	Air activity caused by lowering reactor tank level	Reactor building	Low	Low	3.5	Kr^{88}, Rb^{88}, Te^{132}	Evacuated building for 1 hr	Extended shutdown by 1 hr	Evacuated because of long half-life of air activity components
September 8, 0130	Air activity and smoke from ANP Sutorbilt filters	Reactor building	Low	Low	20	32 min	Evacuated building for 20 min	Extended shutdown by 20 min	
November 30	GEH 6 × 9 cubicle ruptured allowing water to be forced into the air annulus and then into instrument air lines and finally onto floors in the basement and console levels; water was drained on floors in attempt to empty lines	Reactor console and basement levels, and then main floor	25 at 1 ft in air	5×10^5	Slight	Mn^{56}, Na^{24}, Cu^{64}, Ag^{110}, Ag^{111}	Evacuated basement for 6 hr	Scrammed reactor to start scheduled shutdown early; cleaned effected areas up in about 1 week	Water in air lines gave erroneous indications on instruments and made loop operation very difficult; jogged canned rotor pumps during shutdown to prevent burnout and release of fission products; clothing contaminated and confiscated from a large number of people; in the future bottled gas will be used for process application
December 1	Air activity caused by draining of the GEH 6 × 9 cubicle air annulus	Reactor	Unknown	Low	>1000	~18 min	Evacuated basement for 10 min	None	

however, prevented complex experimental programs from being completed without significant loss of time. Further, there have been no serious incidents in nine years of reactor operation. This record is based not only on the good engineering design of the reactors but also on the experienced operating organization and sound safety practices which are used to constantly survey the reactors and their associated experiments. (J. R. Buchanan)

CHANGES IN LICENSING REGULATIONS

Amendments to the AEC rules and regulations, as found in Title 10, Code of Federal Regulations, Chapter 1, are frequently proposed. Among those now being considered for change and discussion herein 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 of Production and Utilization Facilities; Part 55, Operation Licenses; Part 72, Protection Against Accidental Criticality and Radiation Exposure in Shipment of Fuel Elements; Part 95, Safeguarding of Restricted Data; Part 140, Financial Protection Requirements and Indemnity Agreements; Part 150, Transfer of Regulatory Power.

In addition to the above, a proposed amendment to Title 41, Part 9, entitled "Public Contracts" was directed to the operators of AEC-owned nonlicensed reactors, including critical facilities, and is discussed herein.

Title 10, Part 9, Public Records

On October 9, proposed amendments to Part 9 were published in the Federal Register.[3] The amendments would require that all reports of nuclear incidents filed by AEC licensees be included in the public records of AEC. The names of persons exposed to radiation will not be made a part of the public records because of the personal nature of the information.

By the end of October two comments had been received on the proposal; both opposed the change. One felt that the reporting could serve no useful purpose and would create confusion, since they doubted that the public

had enough understanding of radioactivity to properly use the information. The other commentator felt that the amendments would tend to inhibit the reporting of incidents by Commission licensees. It was felt that licensees would withhold reports where they felt that no harm had been done rather than see every report in the public record.[4]

Title 10, Part 20, Standards for Protection Against Radiation

The Federal Register of August 9 contained a proposed amendment to Part 20 "to provide additional ways to determine a permissible concentration limit for a mixture of radioisotopes contained in effluents released into air or water."[5] One comment was received that expressed the opinion that the amendment was a desirable improvement.[6]

Title 10, Part 30, Licensing of Byproduct Material

The AEC continues to receive both favorable and unfavorable comments on the requirements for exemption of tritium from licensing for use in automobile-lock illuminators that were set forth in the Federal Register of April 26, 1961.[2,6-8] The Commission on September 12 proposed an additional amendment to Part 30 that would authorize possession and use of luminous safety devices not containing more than 4 curies of tritium each in aircraft.[9] Favorable comments were received both from the airlines and from possible manufacturers of such devices. There were suggestions that some of the qualification tests were too stringent and control samples too numerous.[2,6,10]

Title 10, Part 40, Licensing of Source Material

A proposed amendment was published in the Federal Register of August 9 which would extend exemption from certain licensing requirements, that is,

those dealing with uranium in the form of aircraft counter weights, and would also exempt certain shipping casks made of or incorporating uranium as a shielding material.[5]

Title 10, Part 50, Licensing of Production and Utilization Facilities

Change Without Specific Authorization. Comments continue to be received on the "change" procedure rules published in the Federal Register of April 8, 1961.[7] One commentator felt that proposed amendments to Part 50 were inconsistent with each other and could have consequences not intended by the Commission.[11] Another commentator, while generally favoring the proposal, felt that some of the approval and reporting requirements should be eliminated.[12]

The Federal Register of October 10 published an amendment to Part 50, effective immediately, that gives AEC consent to creation of mortgages or other liens on nuclear reactors and other production and utilization facilities licensed by AEC. No creditor may take possession of the facility, however, without a license from AEC authorizing the possession of transfer.[13]

Title 10, Part 55, Operator Licenses

AEC published in the Federal Register of October 10 proposed amendments on operator licensing. They provided for licensing of supervisory operators of reactors and for a more detailed description of the qualifications of applicants and of the examinations to be administered. Examinations to be given to operators and to supervisory operators were differentiated. Conditions were established for re-examination of those who fail the written or operating tests and for license renewals. Individuals who "manipulate" controls for a research or training reactor

as students of nuclear engineering courses were exempted from licensing provided they are in the presence of an operator or supervisor who is licensed.[13]

Title 10, Part 72, Protection Against Accidental Criticality and Radiation Exposure in Shipment of Fuel Elements

Comments were received on Part 72, proposed regulations which were issued initially on March 17, 1960, in the Federal Register and recently reissued on September 23, 1961. It was felt that the AEC had obviously given a great deal of thought to the shipment of fuel elements and that in some respects the regulations were quite satisfactory. Some commentators feared, however, that too much attention had been given to specification of what they felt was a preconceived shipping cask design instead of setting forth performance principles.[5,6]

Title 10, Part 95, Safeguarding of Restricted Data

Amendments to the AEC regulations on security data access permits were proposed in the Federal Register of August 12. The permit program allows private business and research access to restricted data which can aid in developing the peaceful uses of atomic energy. The proposed changes related to the following:[5]

> "New technology; new address for communications and reports; technical changes in right of access; termination of and limitations on the right of access; clearance by other agencies under certain circumstances; making of documents; provisions for accounting for restricted data upon expiration, suspension or revocation of an access permit; new reports, particularly year-end reporting of use made of an access permit and number of access authorizations received; and miscellaneous technical changes."

A comment was received desiring clarification of four of the points.[3]

Title 10, Part 140, Financial Protection Requirements and Indemnity Agreements

AEC adopted amendments to its indemnity agreements by having them published in the Federal Register of August 19. The amendments concern the definitive form of the agreement which the Commission will execute with those licensees who are required to maintain public liability insurance or other financial protection. Several editorial changes in the amendments were made as suggested by the Atomic Industrial Forum.[14]

On August 3 the Division of Licensing and Regulation sent out a form letter to various licensees relating to defining the location of a site to be indemnified.[15]

At the close of the first session of the 86th Congress, the Joint Committee on Atomic Energy had pending before it an item of legislature (H.R.-9244) which would amend the indemnity provisions of the Atomic Energy Act to extend coverage to incidents occurring outside the U.S. The matter will be considered during the recess and the next session of Congress.[16,17]

Title 10, Part 150, Transfer of Regulatory Power

Terms of a proposed agreement with Kentucky for partial transfer to the state of the AEC's regulatory authority over radioisotopes, the source materials uranium and thorium, and small quantities of special nuclear materials (U^{233}, U^{235}, and plutonium) were approved for public comment and published in the Federal Register of September 29. Also published was a radiation control program proposed by Kentucky (see article in the first section of this issue). The state proposes to assume all

regulatory responsibility allowed under the law, including disposal of low-level radioactive wastes other than into the sea or ocean.[18,19]

Numerous comments have been received by the Commission, primarily from various other states. They were virtually unanimous in the opinion that the regulations would set up a potential duality of jurisdiction between the states and the federal government, especially as regards sections 150.8 (d) and (e), which deal with waste disposal and transfer of or control of radioactive materials.[3,6,10,20]

Title 41, Part 9, Public Contracts

On August 29 a nuclear reactor safety clause for AEC-owned nonlicensed reactors was released by the AEC Division of Contracts for publication in the Federal Register as an amendment to Part 9-7, Contract Clauses; Title 41, Public Contracts. The new regulations became effective immediately upon publication in the Federal Register. It is stipulated that the clause shall be incorporated in all contracts involving the startup or operation of AEC-owned nonlicensed reactors, including critical facilities. The clause will have the effect of requiring of nonlicensed reactors the essentials required of licensed reactors.

Among points covered in the clause[21] are:

1. Acknowledgement that the contractor will exercise a degree of care commensurate with the risk of a nuclear incident which is involved.

2. Agreement by contractor to comply with all nuclear reactor safety requirements (including reporting and instructions) of the Commission.

3. Agreement by contractor to prepare a hazards summary report, plans, check lists, and procedures to assure safe operation and maintenance

of the reactor. The contractor will also carry out personnel training programs and obtain contracting officer approval prior to startup of the reactor. Significant deviation from the provision of the hazards report will require that all the conditions of this item (hazards report, training, approval, etc.) be repeated.

4. The contractor agrees to see that all operations and maintenance activities are performed by qualified trained personnel (unless otherwise specified in writing) under qualified supervisory personnel. Plans for procedures for nuclear safety must be followed strictly. A system of approved inspections must be instituted to provide frequent checks on reactor and personnel performance. Any change which might affect the safe operation of the reactor should be promptly reported. The reactor should be shut down when the risk of a nuclear incident endangering persons or property warrants. Plans to minimize the effects of a nuclear incident on the health and safety of the operating personnel and public must be prepared. (J. R. Buchanan)

ACTION ON REACTOR PROJECTS BY LICENSING AND REGULATING BODIES

The Atomic Energy Act requires the AEC to protect the public from undue hazards resulting from exposure to radioactivity. 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 the construction permit or the operating license for a facility. It also requires authorizations for changes in facility equipment or operation which contain an element of hazard not previously reviewed or approved. The license application record of the various facilities is reported in Table VI-4.

The ACRS on September 11 recommended to Chairman Seaborg of AEC that five points be considered to help ensure adequate mechanical design of the primary loop of any nuclear reactor system. Since the primary loop is the principal and, sometimes, the first line of containment for radioactive fission products, adequate design of the loop components is important to the integrity of the reactor and the health and safety of the public.[22]

The points recommended were:

"(1) Except in special units, all nuclear vessels and piping be designed, built, and inspected in accordance with one of the recognized codes and that no deviation be permitted except as provided by the code used;

(2) In selection of vendors for supplying equipment for use in nuclear installations, recognition be given to the need for care beyond code requirements in

Table VI-4. Calendar of Legal Steps in Licensing U. S. Power Reactors

(to be provided separately)

selection of materials, control of fabrication, and testing;

(3) In the Hazards Summary Report, (a) the code used be specified, and (b) assurance be given that adequate concomitant material selection, manufacturing procedure, and testing will be used;

(4) Every effort be made to expedite the laboratory and field study of radiation damage of metals as related to nil ductility temperature;

(5) The Division of Licensing and Regulation continue to pursue vigorously the determination of the status of present reactor vessels, the necessary steps to avoid failure from brittle fracture, and definition of the basis for preventive design of future installations."

Recent actions on specific reactors are described below.

BORAX V

On September 7–9, the Advisory Committee on Reactor Safeguards considered the 35.7-Mw(t) experimental boiling-water, internal nuclear superheat, BORAX V reactor,[23] which is nearing completion at the National Reactor Testing Station. Since the facility is testing a new concept, the Committee felt it would be prudent to provide duplicate period circuitry and to provide that the reactor be shut down in the event that one of the two power level scrams is inactivated for any reason. The ACRS also suggested that the Argonne National Laboratory do some additional accident analyses.

The ACRS concluded that the "facility may be operated without undue hazard to the health and safety of the public and site personnel" with the addition of suggested instrumentation and evacuation plans in incorporating the results of suggested accident studies.

Dresden Reactor (Docket 50-10)

On September 28, Commonwealth Edison of Chicago was advised by the AEC's Division of Licensing and Regulation that it could conduct a high core-void content test in the 630-Mw(t) Dresden boiling-water power reactor. The test is to be conducted as described in Edison's proposal of September 6 and four supplementary submittals, subject to additional conditions set by AEC.[24] It is planned to increase the voids in the core from a normal value of 12 wt % to about 30 wt % in gradual steps. Technical personnel from General Electric will supervise the test, which will take only about 12 hr. The results of the test are expected to contribute to the advancement of boiling-water reactor design.[25]

Edison was advised, on October 4, that its July 11 request for exemption from equipping high-radiation areas with visible and audible alarms as required by part 20 of the Code of Federal Regulations could not be justified. It was felt that sufficient information had not been presented to warrant accepting the proposed administrative and mechanical safeguards as suitable substitutes.[24]

On August 1, Edison was advised[26] that there was no objection to a change in the design of fuel assembly PF-12. During the month of September, Edison proposed three additional changes to reactor specification components. They were:

1. To increase the primary steam flow by 12% and secondary flow by 18%.

2. To change the hydraulic control blade drive to reduce operational testing time.

3. To increase the quantity of emergency liquid poison storage so that the concentration of the borate in solution could be lowered.

On October 3, the AEC notified Edison that the three changes should not be made until authorized in writing by the Commission. Additional information was requested, since the Commission felt that possible hazards associated with the changes could not be properly analyzed using the information previously submitted.[24]

Edison has requested suspended credit on its insurance indemnity fee covering the recent shutdown period.[26,27] The insurance associations, Nuclear Energy Property Insurance Association and Mutual Atomic Energy Reinsurance Pool, have the refund request under consideration, but nothing firm on the formula to be followed has been announced, as yet.[25]

Elk River Reactor (Docket PP-1)

The AEC, on September 27, submitted a letter to Allis-Chalmers with respect to the pressure vessel difficulties of the 58-Mw boiling-water Elk River reactor.[27] The vessel had developed many pin holes and cracks in the weld overlay cladding near the head flange. Destructive examinations revealed thousands of additional internal defects in the cladding which were not apparent from surface examinations. The AEC feels that improper material or techniques were used in applying the cladding.[24]

The head of the vessel was returned to the manufacturer so that the defective base metal could be removed to its full depth and new cladding applied. Although the AEC had received information on the repairs, it felt that the data were not adequate and that information on quality control techniques, results of chemical and metallurgical examinations of numerous sample plugs, and other techniques of examinations must be

forthcoming. The AEC also requested a comprehensive examination of the vessel other than the head flange and repair if defects are found.[24]

In a report of August 18, Allis-Chalmers advised the AEC that during vessel fabrication additional metal was welded on the outside surface of a vessel flange because of a mismatch between the flange and shell.[28] In view of this, the AEC feels that the stress analysis completed during the vessel design is no longer adequate and accordingly has requested a more complete analysis.

On July 10 and 18, in letters to Alphonso Tammuro, assistant general manager of AEC's Division of Reactor Development, Edward E. Wolter of Rural Cooperatives Power Association protested against what he felt were unnecessary delays of the reactor project by the action of AEC's Division of Licensing and Regulation. In a reply of July 25, Tammuro assured that AEC is aware of the sense of urgency for the Elk River startup but that the concern for safety justified any time lost in clarifications and remedy of the reactor's problems.[29]

The ACRS, on August 3, advised that on the basis of its subcommittee study the Elk River vessel "meets the minimum requirements and is acceptable from a safety viewpoint if no further detrimental information develops." They felt that the AEC staff was cognizant of the situation and should continue to follow it closely.[30]

Enrico Fermi Atomic Power Plant (Docket F-16)

On September 10, the Power Reactor Development Company submitted to the AEC the seventh volume of a seven-volume hazards report on the 200-Mw(t) (maximum power in authorization request) Enrico Fermi

fast-breeder reactor.[25] The other volumes had been previously submitted.

Hallam Reactor (Docket PP-3)

Atomics International requested, on September 25, that its prior application for operating authorization for the 240-Mw(t) Hallam Reactor be amended to cover the Dry Zero Power Experiments (fuel in reactor, but no coolant). They hope to begin the experiments in January 1962.[24]

The AEC, in a reply of September 29, advised that very close scheduling was needed in order to hold a hearing and get authorization for the experiments to begin as soon as planned. Before the hearing, which was to be held on November 30, the AEC needed proposed technical specifications from Atomics International,[30] the ACRS had to review the case, and the staff's hazards analysis on the reactor had to be available to the public at least 20 days prior to the hearing date.[24]

Atomics International's course of action will require another hearing on authorization for wet operations after plant completion and possibly still another hearing to consider full-power operation.[24]

Humboldt Bay (Docket 50-133)

The Pacific Gas and Electric Company, on September 1, submitted to the AEC the Final Hazards Summary Report for the 163-Mw(t) Humboldt Bay reactor.[25] On October 4, Pacific Gas and Electric released a revised working schedule for the plant, and, if the schedule is met, construction will be completed, fuel loading commenced, and initial criticality attained in August 1962.[31]

Pacific Gas and Electric submitted to AEC, on October 23, a requested report concerning AEC's survey of reactor vessels for induced brittleness from the effects of neutron dose.[32]

Indian Point Reactor (Docket 50-3)

On August 24, Consolidated Edison Company received permission to extend the completion date for its 585-Mw(t) pressurized-water Indian Point Reactor to December 15, 1961 (from October 1, 1961). The extension had been requested because of delays caused by labor disputes and because of extensive testing programs.[33] During September, Consolidated Edison submitted the AEC technical specifications, a preoperational environmental survey of the reactor vicinity, and additional information on the suitability of 17-4 PH stainless steel for use in the control rod drive mechanisms.[27,28,34,35]

Consolidated Edison asked AEC, on October 19, for a license amendment establishing safety limitations beyond which the facility should not operate without AEC authorization.[36]

LaCrosse Boiling-Water Reactor

On May 18–20, the Advisory Committee on Reactor Safeguards considered the Dairyland Power Cooperative site near Genoa, Wisconsin, for the proposed 165-Mw(t) LaCrosse improved-cycle boiling-water reactor.[37] The same site had been considered, and found suitable, by the Committee on April 6–8 for a 200-Mw(t) organic-cooled prototype reactor. The Committee had been presented a preliminary description of the improved-cycle boiling-water reactor in 1960 for siting in California. The ACRS concluded that

in its opinion, the Dairyland site was "suitable for a reactor of this general type and power level."

Nuclear Merchant Ship NS Savannah

Early in November, a rigorous 4-hr "dry run" in a test loading of the core of the 69-Mw(t) pressurized-water reactor of the NS Savannah was completed. The test loading prepared shipyard personnel for actual loadings which were expected to follow soon.[38] Prior to the dry run, final integrated tests of the cooling, control, purification, radiation monitoring, emergency shutdown, and all other systems had been completed. During these tests, the reactor control rod drive system initially failed to perform up to specifications. The difficulty was eliminated by adjustments in the hydraulic portion of the system.[39]

Pathfinder Reactor (Docket 50-130)

On August 8, the Northern States Power Company advised the AEC that the schedule submitted (in September 1960) for review of the 203-Mw(t) Boiling-Water Pathfinder Reactor[30] cannot be met. Events are generally being postponed by two to three months, with initial criticality expected in June 1962. The Company expects to issue a new schedule soon.

Peach Bottom Reactor (Docket 50-171)

On August 7 and October 20 Philadelphia Electric Company submitted data to the AEC on the 117-Mw(t) High-Temperature Gas-Cooled Peach Bottom Reactor.[29,36] The information included a plant description, safeguards analyses (including an initial hazards report), site and environmental data, and specifications on the proposed reactor pressure vessel.

The final hazards report is expected to be issued about October 1963. The AEC has set a hearing for December 18 to consider issuance of a construction permit for the facility.[40] The earliest completion of the plant is estimated to be March 1, 1964, the latest December 31, 1964.

Piqua Nuclear Power Facility (Docket PP-2)

In a letter of October 20, Atomics International recommended that a public hearing for an operational license not be held on the 46-Mw(t) Organic-Cooled and -Moderated Piqua Reactor[32] in December because the comtemplated fuel-loading date did not necessitate a hearing that early and, also, they were preparing for a hearing on their Hallam facility. Early 1962 was suggested as a possible hearing date for the Piqua reactor.

Plum Brook (NASA) Reactor (Docket 50-30)

On July 24, NASA submitted details of a change on the 60-Mw(t) Light-Water-Cooled and -Moderated Plum Brook Reactor.[26] The change concerned installation of beam room facilities so that experiments with neutrons in the epithermal energy range could be run. NASA reported that their safeguards committee had reviewed the change and that it was not considered an unreviewed safety question.

Saxton Reactor (Docket 50-146)

AEC's Chief Hearing Examiner, Samuel W. Jensch, recommended, on October 11, that a provisional operating license to cover the startup period be granted to the 20-Mw Pressurized-Water Saxton Reactor. Included in the proposed license was a change procedure patterned after that in

the license of the Vallecitos Boiling-Water Reactor with a modification representing "an initial endeavor to assist in the determination of the authorized change procedures." This was done by including in the definition of the term "facility" only those plant systems and components having a significant bearing on the safety of the plant.[41] The hearing on which Jensch's decision was based was held September 6.[42,43]

The Saxton Nuclear Experimental Corporation, on October 23, supplied information to the AEC concerning a survey of reactor vessels for induced brittleness from the effects of integrated flux. The submitted information was obtained by Westinghouse and A. D. Smith from studies on the Saxton vessel.[32]

Vallecitos Boiling-Water Reactor (Docket 50-18)

On September 1, the AEC issued to the General Electric Company an amendment to the license of the 50-Mw(t) (authorized power) Vallecitos Boiling-Water Reactor

> "setting forth new provisions on reactor component inspections, visual monitoring for indication of control rod sticking or binding, withdrawal of disconnected control rods, reactor shutdown procedures and precautions, and a new reporting requirement on significant variations from predicted values of reactor operating characteristics which might affect nuclear safety."

In addition, the amendment provides that GE should maintain adequate nuclear control instrumentation and qualified supervision during shutdown periods when core reactivity can change.[34]

On September 29, GE objected to the fact that the license amendment had incorporated, by reference, a GE letter of January 25, 1961, since the letter was prepared without knowledge that the information in the

letter would become part of the facility license technical specifications. While GE has no objections to the substance of the additional requirements, they felt that they should be specified in the amendment instead of merely referenced.[24]

GE was advised on September 21 that permission to perform a control rod material test was granted according to filings of May 16 and July 14.

Vallecitos Experimental Superheat Reactor (Docket 50-183)

On August 10, the AEC gave the General Electric Company a permit authorizing construction of the 12.5-Mw(t) Vallecitos Experimental Superheat Reactor.[30] The earliest completion date is set for April 1, 1962, and the latest for February 28, 1963.

Westinghouse Test Reactor (Docket 50-22)

On August 17, the AEC granted Westinghouse Electric Corporation exemption from requirements for audible alarm devices for a specific area surrounding part of the primary coolant loop of its 60-Mw(t) test reactor.[29] The AEC felt that the Westinghouse proposed high-radiation control procedures were an acceptable alternate to the requirements in part 20 of the Code of Federal Regulations.

Yankee (Docket 50-29)

The Yankee Atomic Electric Company was advised by the AEC, on July 21, that its proposed change (No. 9) which provided for access to the vapor containers while the 485-Mw(t) Pressurized-Water Yankee Reactor is at power was acceptable (as modified by AEC).[29]

On August 21, Yankee advised the AEC that it was adding undervoltage protection to the four 2400-v circuits supplying the main coolant pumps to positively prevent restarting a pump by re-energizing a 2400-v bus following an interruption in power to that bus. They noted that prior authorization from AEC was not required.[33]

On September 12, AEC notified Yankee that approval to perform a test demonstration of the use of boric acid at higher concentrations while operating the power at full load was granted. The license restrictions on boric acid concentrations were varied for the period of the test, that is, two to six weeks. "The proposed test is intended to demonstrate feasibility of using boric acid as a means of extending the lifetime of future core loadings."[25]

On October 23, Yankee submitted an answer to AEC concerning irradiation of its reactor vessel.[32] (J. R. Buchanan)

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.

1. L. R. Amyot et al., Safeguards Report for the MGCR Critical Experiment, GA-1100, General Atomic, December 18, 1959.

2. Application to U. S. Atomic Energy Commission for Reactor Construction Permit and Operating License, Final Safeguards Report from Saxton Nuclear Experimental Corporation at Reading, Pennsylvania, NP-10094, 1961.

3. Army Gas-Cooled Reactor Systems Program, GCRE-I Hazard Summary Report, IDO-28506, July 8, 1959; GCRE-I Hazard Summary Report, Addendum I; GCRE-I Hazard Summary Report, Addendum II; GCRE-I Hazards Summary Report, Addendum III.

4. R. L. Ashley et al., Final Report, SRE Fuel Element Damage, of the Atomics International Ad Hoc Committee, NAA-SR-4488 (Supplement), June 30, 1961.

5. P. Balligand, Operational Accidents in EL 2 and EL 3 between 1.1. 1957 and 1.7. 1959, CEA-1397, 1960.

6. D. W. Battles and E. G. Joki, Summary Report on the Hazards of the UTR Test Reactor, ATL-D-619, June 7, 1961.

7. Belgian Engineering Test Reactor, BR-2, Safety and Design, Final Report, BLG-59, May 1, 1961.

8. W. E. Bost, Radiation Protection Standards, A Literature Search, TID-3551, rev. 1, June 1961.

9. J. C. Couchman, Graphic and Tabular Aids for Reactor Hazards Evaluation, FZM-227, June 9, 1961.

10. P. W. Davison et al., Yankee Critical Experiments Hazards Summary Report, YAEC-31, Supplement 2, July 1, 1958.

11. D. R. deBoisblanc et al., Hazards Evaluation of Proposed Site for ETR II, IDO-16623, Phillips Petroleum Company, August 18, 1960.

12. Enrico Fermi Atomic Power Plant Revised License Application, Part A, General Information and Request for License; Part B, Technical Information and Hazards Summary Report, Volumes 1 through 6, NP-10458, July 1961.

13. F. L. Fillmore, Analysis of SRE Power Excursion of July 13, 1959, NAA-SR-5898, Atomics International, Division of North American Aviation, September 15, 1961.

14. Final Safeguards Summary Report for the Piqua Nuclear Power Facility, NAA-SR-5608, Atomics International, Division of North American Aviation, Inc., August 1, 1961.

15. W. Haass and J. Seig, PM-1 Nuclear Power Plant Addendum to Hazards Summary Evaluation, MND-M-1853, Addendum I, February 1961.

16. W. Haass, PM-3A Nuclear Power Plant, Hazards Summary Report, Site Description and Safety Evaluation, Vol. III, MND-M3A-2496-II, June 1961.

17. Hazards Summary Report for the University of Puerto Rico Nuclear Reactor, ORO-402, March 1960.

18. E. U. Kauer, Metal-Water Reactions, A Literature Survey, TID-12746, March 17, 1961.

19. L. V. Koch et al., EBR-II Dry Critical Experiments, Experimental Programs, Experimental Procedures, and Safety Considerations, ANL-6299, February 1961.

20. A. L. MacKinney, Hazards Summary Report Advanced Test Reactor Critical Experiment at Lynchburg, Virginia, TID-13696, August 1961.

21. R. L. Menegus and H. F. Ring, Reactor Safety Determination Savannah River Plant, TID-5584 (DPW-56-106), February 1956.

22. R. C. Paulus et al., Nuclear Incident at the Idaho Chemical Processing Plant January 25, 1961, Report of the Investigating Committee, IDO-10036, June 1, 1961.

23. H. D. Raleigh and R. L. Scott, Nuclear Instrumentation, A Literature Search, TID-3550, rev. 1, June 1961.

24. N. G. Wittenbrock, Final Safeguards Analysis, Plutonium Recycle Test Reactor, Supplement 3, Effluent Activity Limits, HW-61236, Supplement 3, April 1961.

References

1. D. R. deBoisblanc et al., Hazards Evaluation of Proposed Site for ETR II, Phillips Petroleum Company, IDO-16623, August 18, 1960.

2. J. P. Lyon, Management Aspects of MTR-ETR Operations, Symposium on High Flux Material Testing Reactors Held in Brussels, Belgium, September 21-26, 1959, TID-7584, pp. 148-158.

3. Proposed Rule Making, Atomic Energy Clearing House, 7(41): 31-35 (October 9, 1961).

4. Comments on Proposed AEC Rules, Atomic Energy Clearing House, 7(44): 25-26 (Oct. 30, 1961).

5. Coverage on Proposed Amendments to AEC Regulations, Atomic Energy Clearing House, 7(33): 18-20 (Aug. 14, 1961).

6. Comments on Proposed AEC Regulations, Atomic Energy Clearing House, 7(42): 18-22 (Oct. 16, 1961).

7. Changes in Licensing Regulations, Nuclear Safety, 3(2): ____ (December 1961).

8. Opposition to License Exemption for Lock Illuminators, Atomic Energy Clearing House, 7(32): 14-15 (Aug. 7, 1961).

9. A Proposed Regulatory Amendment, Atomic Energy Clearing House, 7(38): 26-27 (Sept. 18, 1961).

10. Comments on a Proposed AEC Regulation, Atomic Energy Clearing House, 7(43): 7-12 (Oct. 23, 1961).

11. Two Memoranda Setting Forth Comments, Atomic Energy Clearing House, 7(32): 23-27 (Aug. 7, 1961).

12. A Comment Generally Favoring a Proposed Regulation, Atomic Energy Clearing House, 7(34): 30-31 (Aug. 21, 1961).

13. AEC Consent to Creation of Mortgages, Atomic Energy Clearing House, 7(42): 26-28 (Oct. 16, 1961).

14. AEC Has Approved Regulation Amendments, Atomic Energy Clearing House, 7(35): 19 (Aug. 28, 1961).

15. Facility Licensing, Atomic Energy Clearing House, 7(33): 9-13 (Aug. 14, 1961).

16. A Bill (H.R.-9244) to Expand the Coverage, Atomic Energy Clearing House, 7(38): 24-25 (Sept. 18, 1961).

17. Discussing Its Plans for 1962, Atomic Energy Clearing House, 7(40): 4-6 (Oct. 2, 1961).

18. AEC Has Approved for Public Comment, Atomic Energy Clearing House, 7(35): 17-19 (Aug. 28, 1961).

19. The Proposed Agreement with Kentucky for Discontinuance, Atomic Energy Clearing House, 7(40): 17-18 (Oct. 2, 1961).

20. Comments on AEC's Proposed Part 150, Atomic Energy Clearing House, 7(44): 13-21 (Oct. 30, 1961).

21. Nuclear Reactor Safety Clause in Contracts, AEC Memo, August 31, 1961, available at AEC Public Document Room.

22. Letter from T. J. Thompson to Hon. Glenn T. Seaborg, September 11, 1961, subject: Nuclear Reactor System Mechanical Design, available at AEC Public Document Room.

23. Letter from T. J. Thompson to Hon. Glenn T. Seaborg, September 11, 1961, subject: Report on BORAX V, available at AEC Public Document Room.

24. Facility Licensing, Atomic Energy Clearing House, 7(42): 7-13 (Oct. 17, 1961).

25. Facility Licensing, Atomic Energy Clearing House, 7(39): 9-13 (Sept. 25, 1961).

26. Facility Licensing, Atomic Energy Clearing House, 7(33): 9-13 (Aug. 14, 1961).

27. Action on Reactor Projects by Licensing and Regulatory Bodies, Nuclear Safety, 3(2): _____ (December 1961).

28. Facility Licensing, Atomic Energy Clearing House, 7(37): 4 (Sept. 11, 1961).

29. Facility Licensing, Atomic Energy Clearing House, 7(35): 4-10 (Aug. 28, 1961).

30. Facility Licensing, Atomic Energy Clearing House, 7(34): 4-7 (Aug. 21, 1961).

31. Facility Licensing, Atomic Energy Clearing House, 7(43): 4 (Oct. 23, 1961).

32. Facility Licensing, Atomic Energy Clearing House, 7(45): 4-8 (Nov. 6, 1961).

33. Facility Licensing, Atomic Energy Clearing House, 7(36): 9-15 (Sept. 4, 1961).

34. Facility Licensing, Atomic Energy Clearing House, 7(38): 7-11 (Sept. 18, 1961).

35. Facility Licensing, Atomic Energy Clearing House, 7(40): 10-14 (Oct. 2, 1961).

36. Facility Licensing, Atomic Energy Clearing House, 7(44): 4-5 (Oct. 30, 1961).

37. Letter from T. J. Thompson to Hon. Glenn T. Seaborg, May 20, 1961, subject: Report on Dairyland Power Cooperative Site for Improved Cycle Boiling Water Reactor, available from AEC Public Document Room.

38. The Savannah Passes Major Test, The New York Times, S-15 (Nov. 12, 1961).

39. Final Integrated System Tests of the N.S. Savannah's Reactor, Atomic Energy Clearing House, 7(41): 35-36 (Oct. 9, 1961).

40. Hearing to Consider Issuance of a Construction Permit, Atomic Energy Clearing House, 7(47): 3 (Nov. 20, 1961).

41. A Provisional Operating License, Atomic Energy Clearing House, 7(42): 22-25 (Oct. 16, 1961).

42. Hearing to Consider Issuance of a Provisional Operating License, Atomic Energy Clearing House, 7(33): 16-17 (Aug. 14, 1961).

43. Hearing on a Provisional Operating License, Atomic Energy Clearing House, 7(37): 11-16 (Sept. 11, 1961).

INTERNAL DISTRIBUTION

1. C. G. Bell
2–6. W. B. Cottrell
7. E. E. Gross
8. C. E. Guthrie
9. W. L. Harwell (K-25)
10. W. de Laguna
11. H. G. MacPherson
12–16. E. J. Murphy
17. A. W. Savolainen
18. M. J. Skinner
19. J. A. Swartout
20. C. S. Walker
21. A. M. Weinberg
22–27. Laboratory Records Department
28. Laboratory Records Department, RC

EXTERNAL DISTRIBUTION

29–30. R. Anderson, Patent Branch, AEC, Washington
31. H. A. Blair, University of Rochester, Rochester, N.Y.
32. B. F. Boardman, Phillips Petroleum Co., Idaho Falls, Idaho
33–34. E. T. Bond, OTIE, Oak Ridge
35–44. T. W. Hazard, Jr., Office of Technical Information, Washington, D.C.
45. T. S. Chapman, Dow Rocky Flats, Denver, Colorado
46. B. J. Colley, TISE, Oak Ridge
47. F. E. Croxton, Goodyear Atomic Corporation, Portsmouth, Ohio
48. F. L. Cuthbert, National Lead Company of Ohio, Mt. Healthy Station, Cincinnati, Ohio
49. James Dodd, Babcock & Wilcox, Lynchburg, Virginia
50. W. E. Dreeszen, Ames Laboratory
51. K. A. Dunbar, Chicago Operations Office
52. W. E. Elliott, Patent Group, Livermore, California
53. E. E. Hall, Office of Ind. Dev., Washington, D.C.
54. C. D. Harrington, Mallinckrodt Chemical Works, St. Charles, Mo.
55. Henry Heyman, Los Alamos Scientific Laboratory
56. D. W. King, Albuqerque Patent Group
57. S. Lawroski, Argonne National Laboratory
58. G. H. Lee, Chicago Patent Group, Lemont, Illinois
59. Max Linn, Sandia Corporation, Albuquerque, New Mexico
60. F. R. Long, Atomics International, Canoga Park, California
61. W. K. Woods, Hanford Atomic Products, Richland, Washington
62. W. A. Minkler, Bettis Atomic Power Laboratory
63. J. W. Morris, Savannah River Laboratory, Du Pont, Aiken, South Carolina
64. J. C. Nevenzel, University of California, Los Angeles, California
65. H. S. Potter, New York Patent Group, Brookhaven National Laboratory
66. Dennis Puleston, TID, Brookhaven National Laboratory
67. Helen Redman, Los Alamos Scientific Laboratory

68. F. A. Robertson, Savannah River Patent Group, Augusta, Georgia
69. H. M. Roth, ORO
70. H. W. Russell, Battelle Memorial Institute, Columbus, Ohio
71. E. M. Simmons, Battelle Memorial Institute, Columbus, Ohio
72. Delia F. Vroman, Knolls Atomic Power Laboratory
73. R. K. Wakerling, Lawrence Radiation Laboratory, Berkeley, California
74. H. D. Young, Argonne National Laboratory
75. W. H. Zinn, General Nuclear Engineering Corporation
76–90. DTIE, AEC