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A draft copy of Nuclear Safety, Vol. 3, No. 2, is being transmitted to each Technical Information Panel member, as well as to others concerned with the review for patent, classification, and administrative matters. If you have any comments or criticisms concerning the contents of this issue, please send copies to Mr. Thomas W. Hazard of DTI and Mr. H. F. Carroll of Declassification and to me. Inasmuch as the final draft of this issue is scheduled to go to press the latter part of November, your comments will have to be received by November 15 in order to be considered for inclusion in the final publication. Mr. Hazard's and Mr. Carroll's addresses are as follows:

Mr. Thomas W. Hazard
Industrial Information Branch
Division of Technical Information
U. S. Atomic Energy Commission
Washington 25, D. C.

Mr. H. F. Carroll
Declassification Branch
Division of Classification
U. S. Atomic Energy Commission
Oak Ridge, Tennessee

Yours truly,

Wm. B. Cottrell AWS

Wm. B. Cottrell
Editor, Nuclear Safety

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DRAFT

TECHNICAL PROGRESS REVIEW
NUCLEAR SAFETY

Volume 3, No. 2

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

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OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
operated by
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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, 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

Reactors, Sites, and Safety 2

Nuclear Space Systems 13

JCAE Hearings on Radiation Safety and Regulation 18

Breeder Reactors and Nuclear Safety 29

*Safety in Source Handling 30

References 31

II. Accident Analysis

Fission-Product Release from UO_2 35

Activity Buildup in Pressurized-Water Reactor Systems 49

Studies in Reactor Kinetics 55

References 56

III. Control and Instrumentation

The Karlsruhe Symposium on Criticality Control 61

Measurement of Control Rod Effectiveness 72

In-Core Flux Monitoring 79

Nonnuclear In-Core Instrumentation 82

References 89

IV. Plant Safety Features

Deposition of Radioactive Materials from Gases 94

Effects of Blasts on Containment Vessels 102

Beryllium Hazards in Nuclear Operations 110

References 117

V. Consequences of Activity Release

Chemical and Radiation Induced Genetic Exposures 123

Dependence of Mutation Rate on Radiation Intensity 129

River Surveys 136

Recent Studies Related to Determination of MPC Values 149

References 150

VI. Current Events

EBR-1 Operating Experience 155

*Idaho Chemical Processing Plant Criticality Incident of January 25, 1961 164

*The SL-1 Incident 170

Comments on Proposed Reactor Site Criteria 187

Changes in Licensing Regulations 194

Action on Reactor Projects by Licensing and Regulating Bodies 197

Safeguards Reports and Selected Reading 208

References 210

I. GENERAL SAFETY CRITERIA

REACTORS, SITES, AND SAFETY

By M. C. Leverett*

The subject of reactor safety has been a live one during the past year. A set of fairly definitive reactor location criteria has been issued.[1**] Our first fatal reactor accident has occurred.[2] The safety of a major reactor project has become a matter of litigation in our highest court.[3] The Atomic Energy Commission has given new recognition to the importance of reactor safety by reorganizing and expanding its staff.[4] The intention here is to discuss this

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*This article is a condensation of the address by ANS President M. C. Leverett at the June 4–8, 1961, meeting of the American Nuclear Society in Pittsburgh, Pennsylvania. Dr. Leverett is Manager of the Development Laboratories Section, Aircraft Nuclear Propulsion Department, General Electric Company, Cincinnati, Ohio. He received the B.S. degree from Kansas State College in 1931, and the Ph.D. degree in chemical engineering at Massachussets Institute of Technology, 1938. He worked for the Phillips Petroleum Company from 1933–1935 and joined the Humble Oil Company in 1938. On loan to the University of Chicago, he took part from 1942 to 1943 in the initial work on the Hanford reactors. In 1943, he was transferred to the Clinton Laboratory at Oak Ridge, where he was in charge of engineering of the MTR, which was later built in Idaho. In 1948, he returned to Humble Oil Company, but he left after one year to become Technical Director of the NEPA (nuclear aircraft propulsion) Project at Oak Ridge. In 1951, he transferred with the project to the General Electric Company at Cincinnati, in charge of all technical activities. Since 1956, he has held his present title there. He is chairman of ASA Nuclear Standards Board Project N6, "Reactor Safety Standards," and past chairman of the ANS Nuclear Standards Board. He led the U. S. Delegation at the meetings of the International Organization for Standardization on reactor safety standards in 1958 and 1960. He is a past Vice President (1959–1960) and past President (1960–1961) of ANS. He is a Fellow of the American Nuclear Society and an Associate Fellow of the Institute of Aerospace Sciences. He is a member of A.I.Ch.E., A.I.M.M.P.E., A.P.S., and the Engineering Society of Cincinnati.

**See also article in this issue reviewing comments on the site criteria, p. ____.

highly technical subject in a rather broad fashion, taking up three principal aspects of reactor safety: first, the philosophy of reactor safety; second, reactor siting criteria; and third, a description of some technical matters which need attention in the field of reactor safety.

Safety Philosophy

The seemingly simple question, "How safe do you want to be?" has as many answers as there are experts who have addressed themselves to the question. It is interesting to consider the following answers:

1. The answer "as safe as other industries" may be put on various bases. One basis is the usual one of frequency of lost time accidents in disabling injuries per million man hours. The atomic energy industry, with a record of about two such accidents per million man hours, is already safer than many industries reckoned as adequately safe. Another basis is the probability that an "average reactor" will cause a death during a year, which may be computed to be about 1 chance in 75 (although the statistics are very poor). Comparing this, for example, with 1 in 26 for a railway locomotive, 1 in 6 for a transport aircraft, and 1 in 2000 for an individual automobile indicates that reactors are safer than a good many devices which we accept as being adequately safe.

2. Another answer is that "no substantial increase in risk to individuals" should be caused. This criterion has been adopted by some responsible groups, including the Advisory Committee on Reactor Safeguards. In the entire nuclear energy industry, the risk of accidental

death due to all causes is about 1 in 40,000, as compared with the present risk of 1 in 2000 to an individual from all causes, and 1 in 7000 in the home. Certainly the nuclear energy industry is adequately safe by this criterion. A possible defect in this approach is the fact that none of the large-scale accidents which have been calculated to be possible have occurred in the nuclear industry. One such accident could change the statistics markedly.

3. The answer that "The probable damage over a long period should not exceed some set level" expresses an entirely different approach. For example, it has been suggested for a given industry that the public injury or damage which might be acceptable should be proportional to the contribution of that industry to the public welfare — possibly as measured by the dollar value of the services or products provided. This is a very objective approach and in a theoretical sense has considerable attractiveness. However, it involves estimating the probability that an accident will occur which is at present practically impossible either a priori or from experience. Use of such a criterion, therefore, is at present impractical; also, it is probably a little bit cold-blooded for general public acceptance.

4. A fourth type of answer which has received some publicity is, "No matter how safe it is; if you can make it safer, do so!" This criterion is at the opposite end of the objectivity scale from the one just previous and is essentially emotional and unreasoned. It is unfortunate that some of this view has crept into our thinking with respect to safety in nuclear matters. Obviously, this kind of

criterion is uneconomic and very difficult to apply, since there is no logical limit to the extent to which one might be driven in the name of increasing safety.

A somewhat different approach is to observe that about 350 decisions have been made to build and operate various reactors in the United States, thereby accepting a certain level of risk. These have been thoughtful decisions, in some cases agonized over at length by many conscientious and informed people. If a pattern can be found in them and codified, the code can be used as a standard in judging future proposed reactors. The recently announced AEC "site selection guide"[1] is a good example of the emergence of such a pattern. Other examples of patterns will doubtless emerge on further study. Insofar as possible such criteria should be expressed in terms of performance or objective to be achieved, rather than in details of design or construction of a reactor. For example, a desirable type of criterion would specify the off-site dosage in event of an accident, together with needed assumptions in the calculation of such dosage. It would not simply specify, for example, a relation between reactor power and distance. It is important to remember that such standards are evolutionary developments based on many difficult considerations. The standards follow and are changed by practice, instead of preceding and determining practice. Sizable departures may very well be proper in individual cases, thus providing more background for modification of the standards. This approach would leave room for reduction of cost, without compromising safety, through ingeneous design; allow comparison with past practice

only, rather than with the credibility of an accident or with some absolute level of risk; offer a basis for relaxation from unnecessarily stringent and pessimistically based prior practice.

In summary, it is proper that safety criteria be derived from practice instead of from "first principles," and that the criteria evolve with experience, being regarded as indicators rather than laws, and meant to be departed from when the justification is suitable. This is the basis on which many of our "standards" have been developed — speed limits for automobiles, factors of safety in structural design, and standards of purity in milk, to mention a few. The use of such comparative criteria involves difficult problems, such as judging the relative probability and seriousness of accidents between two reactors of different design, but is easier than judgments of the absolute probabilities and intensities of the same accidents. In many cases a qualitative judgment will suffice, if a quantitative one cannot be made.

Site Criteria

Decisions on the acceptability of a site for a reactor depend on many considerations. Much attention has been given to site criteria in the past year, and there is reason to think that comparative judgments of the safety of sites will become much easier in the future.

The possibility and desirability of evolving an index of merit for sites has been considered. It has been the basic practice to judge the suitability of a reactor site by the probable damage a large accident might cause. There has been some discussion as to whether the index of

damage should be human injury or property damage. Probably either is satisfactory as a relative index, but the absolute numbers resulting should not be taken too seriously. Too often, the site evaluation is required to state the worst possible accident at a site, regardless of its probability. The compounding of improbable combinations can produce frightening damage numbers, but they tax one's credulity. Such calculations are useful only in a relative way and should not be used to determine the required dollar assets of the reactor owner, as sometimes appears to be done.

The proposed guide published by the Atomic Energy Commission a few months ago for use in siting reactors essentially codifies existing siting practice.[1] The nuclear industry appears to consider the criteria to be in the main sound and workable, but with room for improvement. One weakness in the guides is that, if taken literally, they compensate only for possible total size of accident but not for the probability, which is logically unsupportable and surely not intended. Another weakness is that they do not allow for a realistic reduction of pressure in the containment vessel following an accident — a factor that can in many cases make an important difference in the maximum damage probability. Another difficulty is in the definition of population zones. This could be obviated by prescribing the total man-rem dose as the severity criterion, which would remove the risk of defining "population center" and would limit the total radiation injury to the population. Meanwhile, the guides should be accepted until something

better appears, with whatever modifications may be made as a result of the recent intensive discussions of them by the industry.

Codification of siting standards in the United States should not imply, as some will object, that we are setting standards for other countries to follow. We may indeed, as necessity demands, tend toward the practice of other countries ourselves.

Desirable Safety Programs

It seems that neither industry nor government is doing as much as it should in the area of making the fruits of experience of others available to those working on reactor designs and hazards evaluation. There are four categories of things that ought to be done, as set out below.

1. More work should be done in the area of reducing the probability of accidents. While the statistics on large reactor accidents are zero, there have been several small accidents, and even more near-accidents. Standards based on these and on other experiences must be evolved and, moreover, subscribed to. Better standards of design, operation, organization, or procedure could have prevented all the mishaps mentioned of which the author knows. The standards will have to be developed piece-meal, a little at a time, until we are able to write comprehensive standards on all types of reactors, operations, and circumstances. We must not fail to start simply because we can't do a finished job now.

2. More work should be done in the reduction of severity of accidents. Ingenious and resourceful designers have come up with a

number of devices for reducing the severity of an accident should it occur. As examples one might mention pressure suppression and containment sprays. However, these are as yet only partially evaluated and, as a result, the designer does not know what credit he can take in terms of decreased severity of the maximum supposed accident or whether his evaluation of the effectiveness of his particular safety device is going to be acceptable. Other possibilities have not yet been evaluated at all. In addition, the entire process of escape of fission products from within the fuel to the outside environment needs more investigation. This includes more investigation of release from molten or overheated fuel elements, plate-out on surfaces, removal by condensing steam or water sprays, reduction of pressure with time, the fraction likely to escape even through an open (e.g., ruptured) pressure vessel, filtration of escaping gases and vapors, and removal by rain-out in the immediate vicinity. Fortunately, there is an AEC program of this type, although it does not seem as aggressive as might be desired. After a suitable series of small-scale tests and analyses, large-scale experiments should be carried out, perhaps culminating in intentional reactor runaways with measurements of fission-product release, deposition factors, and other parameters. The cost would probably not exceed that of the average reactor experiment program, and the results could be of great value.

3. Additional work should be performed on the evaluation of the magnitude of the risk involved in possible accidents. The magnitude of the risk of off-site damage is obviously a direct function of fission-

product release, which has just been discussed. In addition, some work remains to be done in the meteorological field, such as determining reasonable probabilities for specific meteorological conditions and their durations at particular sites, rather than assuming either the worst conditions or improbably long durations.

4. More work should be done on the further study and evaluation of potential sources of accidents. Hazards reports frequently state that all conceivable accidents have been examined and that the particular reactor cannot experience an accident of significant severity. However, potentialities of accidents not previously envisioned turn up from time to time. An example is the ejection of a control rod from a reactor due to the pressure of water and steam within it, which could conceivably result in an excursion-generated shock wave. It would seem that experimental work on the resulting effects on the pressure vessel and other components of the reactor would be justified. Perhaps intentional chemical explosions, planned to simulate reactor excursions, could be used in test vessels. Shock might cause brittle failure of the vessel wall rather than ductile failure, with a resulting increase in accident severity. In addition, it has been found in tests of some alloys that the time to rupture under a given stress at high temperature may be reduced by up to several orders of magnitude if they have been subjected to irradiation and that this kind of radiation damage dose not always anneal out at temperatures as high as 1200°F. At present there is almost a complete lack of data on the effects of radiation on high-temperature structural alloys,

particularly in their long-term properties. The collection of such data is difficult and expensive, but is important to reactor safety. It is understood that such a program is beginning.

Conclusion

We can now begin to see the general dimensions of the technical problems of reactor safety. Work classifiable as reactor safety research presently is going on in this country at the rate of about $12 million per year. Our total reactor development program is approximately $400 million per year. From the point of view of balance, it would not seem unreasonable that expenditures for reactor safety research should be stepped up. Perhaps for a few years a doubling of these expenditures would not be out of order. Such expenditures are trivial when compared with the estimated magnitude of the damages for a severe reactor accident. More realistically, such expenditures might produce very substantial savings (perhaps of the order of $10 million per year when we are constructing five new power plants annually) just in the shortening of distance between the generating capacity and load centers. Additional less tangible savings are likely also. This work should be a part of the program of the Atomic Energy Commission.* Simultaneously, there must be the development of standards of reactor design, operation, organization, and procedure by industry that will assure safety without undue economic penalty. There must also be the collection of standard practices in the assessment of effects of nuclear

*In this connection, it is noted that the direct expenditure on nuclear safety studies by the Reactor Safety Branch has increased from 12 million dollars in fiscal 1961 to million dollars in fiscal 1962.

accidents in reactors and the methods of calculating such effects. The total program will involve considerable radiation damage work, metallurgy, meteorology, and engineering analysis and experimentation. Some of the last should be essentially full-scale using specially constructed reactors and containment vessels. We need to move ahead in this program as rapidly as possible. (Condensed by L. A. Mann)

NUCLEAR SPACE SYSTEMS

Preparations are currently being made to fly a number of different space nuclear energy sources which range from the radioisotope-fueled sources of a few watts in power up to the kilomegawatt fission reactor for the nuclear rockets. Safety analyses of these nuclear sources follow the usual lines of reasoning that the general population should be protected from unreasonable hazards. In addition, since these energy sources will fly over other than our own territory, an additional step of considering the diplomatic impact of the flight must be taken. It appears that initially each flight will require direct Presidential approval.

On June 28, 1961, a significant step in the utilization of nuclear energy sources in space was taken when the Navy's Transit IV-A navigational satellite went into orbit carrying a SNAP 3* radioisotope-powered auxilary power unit. The flight had originally been prepared for June 6 following Presidential approval, but the State Department requested reconsideration because of the world situation at that time, and the flight was delayed for the additional period.

The general safety considerations in the use of nuclear energy sources in space were discussed in a previous review[5] and at the Space-Nuclear Conference[6] of the American Rocket Society in May 1961. The periods considered were (1) the prelaunch phase, (2) the launch phase, and (3) the return phase. Prelaunch is defined as the period from construction of the source until lift-off from the launch pad. The launch

---

*The SNAP (Systems for Nuclear Auxiliary Power) units are given an odd number if they have a radioisotope energy source and an even number if they have a fission energy source.

phase proceeds from lift-off until a successful launching is completed. The return phase refers to that time, planned or unplanned, when part or all the nuclear package returns to the vicinity of the earth. Branch and Conner[7] expect that the radioactivity which might be credibly released over the next 20 years, if distributed on the global scale, will cause only a minor change in the existing levels. They have reviewed all the nuclear space systems from the standpoint of nuclear safety.

Radioisotope Energy Source

In general, the radioisotope systems present their greatest hazard during the prelaunch and launch periods, since at this time the isotopes are at their greatest power output of the entire mission. Some of the problems of radiological safety analysis for the 125-watt radioisotope-powered SNAP 1A have been discussed by Dick,[8] including core loading in a hot cell, transportation of the fueled core, transfer of the fueled core to the generator, generator shielding, launch abort procedures, vehicle failure on the launch pad, vehicle failure during ascent flight, and re-entry from extraterrestrial flight. Safety analyses of all radioisotope energy sources follow this general outline, since these sources differ only with respect to power level and the specific isotope used.

The alpha-emitters Po^{210}, Pu^{238}, and Cm^{242} are favored[9-11] for the small (few-watt) SNAP 3 sources, since they have a high power density and present essentially no gamma shielding problem. The 125-watt SNAP 1A uses Ce^{144} and the accompanying Pr^{144} beta-emitters and gamma emitters for a power source and requires a drainable mercury shield at the launch pad. The use of Sr^{90} or Ce^{144} in the larger units is dictated primarily by cost considerations. Both are produced from fission products.

The radioisotope is in each case tightly encapsulated in a high-temperature-resistant high-strength material capable of containing the materials under the conditions of a launch abort. The launch abort tests in the altitude range from 0 to 1000 ft have been described by Dick.[8] The extreme condition would occur following a booster failure in which the core could be subjected to a 6000°F fireball and explosion with shockwave overpressures of 8000 psi as the booster exploded, followed by 8-min immersion in a high-temperature fire of from 2000 to 4500°F. Tests to approximate such conditions were made at Aberdeen Proving Ground on Inconel X, Hastelloy B, and type 316 stainless steel cores. Inconel X was also tested by immersion in liquid oxygen to demonstrate low-temperature integrity and resistance to thermal shock. A study of abort conditions from 1000 ft to 300 miles indicated that the core would impact at 500 ft/sec and be at a temperature of 1500°F. Tests conducted against consolidated earth showed that the core would retain the radioisotope, but granite impact test results indicated that the core should be redesigned. Subsequent tests of redesigned cores with impact-absorbing caps gave satisfactory results.

As a general criterion for re-entry, the requirement for complete burnup above 100,000 ft has been generally accepted. This means that the debris would be above the usual air travel lanes and would be retained for some time if the particle size were small enough. In re-entry considerably more energy is liberated in a short time from the re-entry drag heating than in the abort conditions, and it appears feasible to burn up the cores for both SNAP 1A and the various SNAP 3 units. Breakup

of the core aids in the re-entry burnup. Analysis of SNAP 1A indicated complete burnup to particles of sizes less than 10 μ above 120,000 ft.

Nuclear Reactor Energy Sources

The safety analyses of the SNAP reactor systems[12,13] as energy sources in space parallel closely the analyses of the radioisotope systems, except for the additional hazard of unplanned criticality. Shipment problems for the reactor are somewhat more difficult than for the radioisotope systems, although this problem is simplified when the fission-product inventory is kept low by limiting the preshipment testing of the reactor. Shipping provisions have to guard against accidental excursions of the reactor during shipment. The development of safety criteria and examination of hazardous conditions for the launch period are necessary for each of the SNAP reactors. Once at the launch pad, the integration of the power package with the rest of the payload is also made simpler if the fission-product inventory in the reactor has been kept low during the preflight period, since only minimum shielding of the reactor is necessary under those conditions.

If orbital startup of the reactor is used, the launch phase problems will be minimized. Zero-power testing of the reactor can take place without an appreciable increase in the rate of fission-product inventory. If the reactor is operated at full power on the pad and during launch, the gamma dose rates will be within safe limits up to a thermal power of almost 500 kw. Remote removal of the reactor because of malfunctioning or long delay on the launch pad does not present difficult problems, but a reactor excursion on the launch pad or an abort followed by an excursion

could present hazardous conditions. Analyses of the SNAP 2 (50-kw thermal) reactor indicate that a reactor excursion would release fission products equivalent to those present after operation for 30 min. Personnel evacuation radii and blockhouse protection appear adequate for radiological protection.

Re-entry problems for the SNAP reactors can be minimized by placing the package in a long-lived orbit so that a long decay period is available after reactor shutdown. A 700-mile orbit which would allow the reactor about 300 years to decay would provide for safe, intact re-entry. On the other hand, re-entry analyses[14] of the fuel elements have indicated that if the elements are released above 280,000 ft they will burn up and disperse above 100,000 ft. These analyses are still subject to experimental verification, and the long decay period is favored. Some combination of the two may be the ideal answer. (P. G. Lafyatis)

JCAE HEARINGS ON RADIATION SAFETY AND REGULATION

The Joint Committee on Atomic Energy (JCAE) held public hearings in Washington June 12–15, 1961. In a press release,[15] Committee Chairman Holifield announced that the hearings were to cover two major topics: radiation safety and regulatory problems. The testimony on radiation safety covered general safety problems, the Commissions reactor safety program, and recent developments in reactor safety, with particular emphasis on the SL-1 incident (see discussion of SL-1 incident, p. 170, in this Review). The testimony on regulatory problems presented not only formal and informal aspects of the regulatory process but also possible further improvements in each.

As stated[15] by Chairman Holifield, "the Commission's regulatory operations have been the object of increasing concern and study. On March 20, the Joint Committee released[16] a two volume print containing studies by the AEC, the University of Michigan Atomic Energy Research Project and the Joint Committee staff. The AEC recommended and has since adopted[17] certain organizational changes including creation of a Director of Regulation reporting directly to the Commission while the University of Michigan study recommended creation of a separate agency to handle the commission's regulatory functions. The Joint Committee staff adopted an intermediate position, recommending creation of an internal Atomic Safety and Licensing Board."

The hearings lasted for four days, during which many personages from government, scientific institutions, and business presented testimony on the various topics considered. All testimony will be incorporated in

the record of the hearings, copies of which will be available.[18] In addition, the Joint Committee had solicited the views of labor and industry representatives, as well as a number of legal scholars, on the alternative organizational arrangements and other suggested improvements in the AEC regulatory process. These statements will be published in an appendix to the record of the hearings.

The record of the hearings is voluminous and it is hardly possible to summarize fairly the various positions of different individuals and organizations on the issues discussed. Hence, the reader is encouraged to refer to the Committee report for background information, specific data, and details; however, some of the apparent highlights and trends presented in the testimony are discussed here (see also "Comments on Proposed Reactor Site Criteria," p. 187, this Review).

General Reactor Safety Problems

Most of the testimony on reactor safety was a restatement of the present status of the business of nuclear safety. The importance of each phase of the program, that is, design, construction, and operation, in the over-all safety was duly recognized, and the continuing need for government review and surveillance to assure no undue hazard to the public was expressed. Although no great changes in the subjective basis of safety assessment was anticipated in the near future, McCullough of the ACRS noted that "attempts [were] being made to review all the information [on reactor technology] and to summarize the criteria contained."

Perhaps the most informative testimony on this subject was that of Beck, Assistant Director for the Facilities License, Division of Licensing

and Regulation, who discussed the present status of the assessment of reactor safety and the Commission's experimental program for providing pertinent information where known inadequacies exist. In the area of safe design, he identified the following general safety objectives:

> "a. The reactor must be stable in its operation. That is, there should be no self-generated oscillatory, erratic or unpredictable variations in power level and any perturbations likely to be imposed on the system should not lead to such unstable behavior.
>
> "b. Insofar as possible there should be inherent, self-limiting characteristics which would cause the reactor's power level to remain at or return promptly to a given condition when any perturbing condition tends to cause it to diverge from that condition.
>
> "c. If a reactor is not fully self stabilizing, its dynamic response to perturbations likely to occur should be sufficiently slow that control systems having reasonable characteristics should be able to restrain the power level within desired bounds.
>
> "d. The temperature of fuel elements, the pressure in the reactor vessel, the heat flux on fuel surfaces, the stress on mechanical components, and other similar process variables must have a sufficient margin of safety before failure that likely perturbations will not cause the failure point to be reached.
>
> "e. Routine effluents from the facility to the environment must be within permissible limits with adequate monitors to determine, and facilities for controlling, effluents levels if they are above tolerance.
>
> "f. Design should be so adjusted that failures, malfunctions and misoperations that could occur would not lead to serious accidents.
>
> "g. There must be adequate safeguards against release to the environment of fission products which might be accidently released from their normal confining barriers within the reactor facility. The safeguards may include, among others, a high integrity containment building, a washdown spray system, or a filtration system on the building exhaust."

Regarding the operation of a nuclear facility, the following basic principles were identified by Beck as fundamental to safeguarding against the likelihood of misoperations.

"a. It is important that the lines of authority and responsibility in the operating organization be clearly defined.

"b. There is usually provided some system of checks and balances, internal to the operating organization, by which some competent technical group separate from the line operating staff, periodically examines reactor operations and performance, reviews proposed changes in the facility or in the operating procedures, and examines important revisions in the experimental program, to insure that hasty or ill considered revisions are not made, and generally exercises a second management judgment on operations, independent of the operating staff.

"c. It is important that there be definite, written procedures by which all normal operations are performed and by which all forseeable abnormal operations and contingencies are carried out. Deviation from these procedures should only be permitted after alternate procedures are approved.

"d. The supervisory and operating staff should have adequate experience and training for the tasks assigned.

"e. There should be well defined boundaries and limits within which the facility may be operated."

The importance of the choice of materials and the quality of workmanship in the safe construction of the reactor were noted. As an example of the problems with materials the experience of 17-4 PH stainless steel in control rods was cited.[19] Beck also presented a brief review of the Commission's safety research program. This program was reviewed in an earlier issue of Nuclear Safety,[20] but it has since been expanded somewhat.

Specific Reactor Safety Problems

Most of the testimony on specific problems was related to the SL-1 incident. Inasmuch as this issue of Nuclear Safety includes a feature article (see Current Events section, p. 170, The SL-1 Incident) that subject per se will not be discussed further here. As an outgrowth of the SL-1 incident, however, the AEC conducted a survey[21] of all licensed

critical facilities by requesting that specific information on the reactivity worth and shutdown capabilities of the reactors under various conditions be submitted. Reporting on this survey of the hearings, Price, Acting Director of Regulation, summarized the conclusions of the survey as follows:

> "1. No licensed reactor was found in which removal of a single control rod could make the reactor critical when other control devices were in operation.
>
> "2. In no licensed reactor was it judged that the shutdown margin was inadequate.
>
> "3. In no licensed reactor was it found necessary to require any changes in the arrangement or condition of the control system.
>
> "4. For some licensed facilities it was concluded that nuclear instrumentation should be used during more non-operating activities than had been the practice, that written procedures should be extended to cover such activities, and, in a few cases, that more direct supervision should be required during non-operating activities. However, none of these situations appeared to require any immediate action."

At the same time that licensed reactor operators were asked to undertake the above survey, investigating teams were established at each of the commission-owned reactor sites to review the safety of commissioned-owned reactors. The review at the Brookhaven National Laboratory led to the shutdown of the Brookhaven reactors. Their subsequent startup and adsolvement of Laboratory management was reviewed by Tape, Acting Director of the Brookhaven National Laboratory. He stated that "In the case of the Brookhaven action, every improvement could have been achieved without shutting down the reactors and without the accompanying, damaging effects resulting from misconceptions by the public, effects on staff morale, and the undue amount of time required of senior responsible personnel both in the AEC and at Brookhaven to rectify a situation which was made worse

by the order. The public is best served by improvements in our operation rather than by unnecessarily causing alarm and casting doubt upon the responsibility and technical ability of the operating contractor and his employees."

So far as has been reported, no other reactors were shut down as a consequence of these investigations.

Problems in Regulatory Processes

The testimony presented on the problems in regulatory processes included review by the AEC staff personnel of the operation of the regulating procedures, as well as testimony by several others directing attention to alleged shortcomings of the existing procedures. Those aspects of the existing procedures which were criticized by one or more of the participants included (1) the Public Hearings, (2) the number of repetitive reviews, (3) the need for standard criteria, and (4) the function of the ACRS.

Although there was obviously no opportunity to resolve different attitudes with regard to each of these subjects the most frequently expressed attitude on each is summarized below.

1. While the virtue of public documentation was everywhere defended, the value of public hearings was seen as limited only to construction approval.

2. The apparent cumbersomeness of the review procedure, including reviews by the Division of Licensing and Regulation, the ACRS, the

Hearing Examiner, and the AEC Commissioners, was generally noted. In addition to the curtailment of the hearing step as suggested above, others commented on the use of conventional administrative procedures, the overlapping of the reviews by the Division of Licensing and Regulation and the ACRS, and the fact that the AEC Commissioners make the final review.

3. The need was cited for the formulation of some general criteria, in the form of a preliminary guide, or merely a statement of existing practice.

4. The discussion of the ACRS was principally directed toward consideration of optimum utilization of this group by restricting the area of their review and, in another instance, by limiting it to the development of criteria only.

Suggested Improvements in Regulatory Processes

As previously noted,[17] the March 1961 reorganization of the Commission was an attempt to separate promotional and regulatory functions. Prior to this action, the regulatory processes were studied by the AEC, the Joint Committee Staff, and the Atomic Energy Research Project at the University of Michigan.[16] Much of the discussion at this hearing was concerned with various recommendations resulting from these studies. As stated by AEC Commissioner Olson, "In line with suggestions made in both the JCAE staff study and the AEC report, we have adopted new procedures in reactor and waste disposal cases which will provide more complete and meaningful information to local

officials and the public prior to hearing. In connection with applications for construction permits and operating licenses for power and test reactors, we are attempting to make public a staff hazards analysis and the ACRS report well in advance of hearing. Notice is being served on the chief executive of the municipality or county in which the project is located. A similar procedure will be followed in waste disposal proceedings."

Olson further suggested that "The Joint Committee might well conclude that the mandatory hearing requirement of 1957 legislation has accomplished its purpose and that the time has come to consider a relaxation or some modification of the mandatory hearing requirements. Such a relaxation would make it possible, without violating the spirit of this legislation, to reduce the number of hearings without prejudice to the public's right of access to full and timely information; and without prejudice to any interested party's right and opportunity to intervene. Any relaxation of the mandatory hearing requirement should also provide some flexibility in referring cases to the ACRS so they will need to consider only safety questions of real significance in connection with power and test reactors and any other the AEC is interested in."

Commenting on the recent AEC reorganization and the fact that it did not completely conform to the recommendations of the Joint Committee Staff, Olson concluded "Although we do not believe the Commission's regulatory and other responsibilities should be divided between separate agencies at this time, the reorganization which the Commission

has adopted is certainly consistent with, and a logical step towards, ultimate separation. The Commission has considered the independent Licensing Board recommended by the staff study and has concluded that it would be unwieldy. The proposed Licensing Board would leave divided responsibility for regulatory functions between the Board and the Commission."

The testimony was concluded by the presentation of three important points of view: that of ACRS by its 1960 Chairman, Leslie Silverman, that of the Atomic Energy Research Project by W. H. Berman, and that of the JCAE Staff by Wm. Mitchell. The JCAE proposal would create a separate "Atomic Safety and Licensing Board" whose members would be appointed by the president. This Board was proposed to accomplish three principal purposes: " ... to separate the final responsibility for decision-making in licensing from the responsibility for operating and promoting; to combine in a single expert body the functions now performed at the initial hearing by the examiner, who necessarily has no expertise in scientific and technical matters, with the functions now performed on appellate review by the Commissioners; and at the same time to preserve the close working relations which are necessary between the regulatory divisions of the AEC staff, on the one hand, and the operating divisions and field laboratories of the AEC, on the other hand."

On the other hand the Atomic Energy Research Project proposal was principally concerned that any separated licensing board also possess rule-making authority. The Project report concluded that the combination

of licensing and rule-making authority in a single group was essential but Berman explained that this did not necessarily involve the complete separation of promotional and regulatory functions by the creation of a regulatory agency outside the AEC.

The ACRS position, as attested to by Silverman, was that any of the proposals could be made workable but that the proposal by the Joint Committee, i.e., the creation of an "Atomic Safety and Licensing Board" was the middle course and was favored by the ACRS, since it would establish "a sound framework for the future growth of the hearing and licensing processes." Regarding the scope of ACRS activities, Silverman stated that "it is not essential ... that [ACRS] review every power and test reactor case in the same detail in the case of those reactors above 10 megawatts thermal where there are features similar to those already evaluated in depth in past cases." On the other hand, Silverman did not think that ACRS review of even power reactors should be restricted to the construction permit phase because many important changes with respect to safety features can take place after the construction permit has been issued.

Subsequent Legislation

Two months after the hearing, JCAE Chairman Holifield introduced a bill[22] in the House of Representatives which would amend the Atomic Energy Act of 1954 by providing for the establishment of a three-man Atomic Safety and Licensing Board, which would fulfill the functions of the hearing examiner. A second provision of the bill also dispensed with hearings on the granting of amendments to construction permits, as well as hearings on the issuance of operating licenses and, under certain circumstances, the 30-day notice and publication of such pending action.

Another section of the amendment provided that certain reviews which have been a routine function of the ACRS would be required only when specifically referred to ACRS by the Commission. Included in this category are applications under section 104c of the Act for construction permits or operating licenses for testing facilities and any applications for amendment to construction permits or amendments to operating licenses under section 103 or 104a, b, or c. (W. B. Cottrell)

BREEDER REACTORS AND NUCLEAR SAFETY

(Material held over from draft of Vol. 3, No. 1 to be inserted here; see draft pages 15 through 20 and references 30 through 37 on page 46.)

SAFETY IN SOURCE HANDLING

(Material held over from draft of Vol. 3, No. 1 to be inserted here; see draft pages 30 through 43 and references 64 through 76 on pages 48 and 49.)

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FISSION-PRODUCT RELEASE FROM UO_2

A major problem associated with fuel elements containing bulk UO_2 is the release of volatile fission products from the UO_2. In a previous review article,[1] it was pointed out that two important areas of uncertainty in predicting fission-gas-release rates are the temperature distribution in the UO_2 and the rate of release of volatile fission products from a given batch of UO_2 at a given temperature. Since the publication of the last review article on fission-product release from UO_2, a number of papers have appeared which help to reduce these uncertainties.

Temperature Distribution in UO_2

Postirradiation examination of some of the earliest UO_2 fuel elements tested revealed that regions of columnar grains were present at the centers of the fuel rods. Since these structures were reminiscent of the structure of metal castings, it was concluded that the UO_2 had been molten at the center of the fuel rod during irradiation and that the observed structure developed upon solidification. By assuming that the edge of the columnar region was at the UO_2 melting point, 5000°F, Eichenberg et al.[2] estimated the average thermal conductivity of irradiated UO_2 between 750°F and the melting point to be 1 Btu/hr·ft·°F. This value has been used widely in the design of reactors with UO_2 fuel elements.

Further studies revealed that the centers of fuel rods where columnar grain growth had occurred had probably not been molten after all. Robertson[3] observed that outside of the columnar grains there was a narrow band of relatively large, equiaxed grains. Outside these, the

structure of the pellets was unchanged. From a knowledge of the grain-growth characteristics of UO_2, an approximate temperature was assigned to the outer limit of the equiaxed-grain region. An average thermal conductivity of the UO_2 was then computed assuming that the edge of the columnar-grain region was at the melting point of UO_2. An improbably low value for the thermal conductivity was obtained.

In a series of out-of-pile tests, MacEwan and Lawson[4] duplicated the columnar-grain-growth region without central melting and postulated the following mechanism of columnar grain growth. In a steep temperature gradient at temperatures near 3600°F, where the vapor pressure of UO_2 is appreciable, UO_2 sublimes on the hot side of small pores and condenses on the cold side. The pores then move toward the center of the rod and leave behind a single, oriented grain. Volatile fission products in the path of a pore are collected in the pore and released to the central void. Subsequently, pores at the periphery of the columnar region move through the same region and continually sweep out the volatile fission products in the affected region.

Additional confirmation of this mechanism was provided from examination by Robertson et al.[5] of fuel elements irradiated at much higher heat ratings than those of the earlier tests. It was observed that within the region of columnar grain growth, there was another sharply outlined region. After short-time irradiations (less than 1 min), this innermost region was observed, but the region of columnar grain growth was greatly reduced. Thus, it was believed that the inner region, whose extent was not time dependent, was the molten zone. By assigning a melting point of

5070°F to the boundary of the innermost region and estimating the surface temperature to be 480°F, Robertson et al.[5] calculated that

$$\int_{480°F}^{5070°F} k(\theta)\ d\theta = 75\ w/cm \quad .$$

This corresponds to an average thermal conductivity between 750°F and the melting point of about 1.7 Btu/hr·ft·°F for UO_2. Below 1800°F, the thermal conductivity values of Robertson et al.[5] were in agreement with the data of Scott.[6] Above 1800°F, Robertson's results indicated a constant value of 1.62 Btu/hr·ft·°F for the thermal conductivity of UO_2.

More recently, Bates[7] has published evidence that the thermal conductivity of UO_2 increases with increasing temperature above about 2000°F. This evidence was based on postirradiation examination of an element operated at a high heat flux and the correlation of the observed structures with the known temperatures at which equiaxed grain growth and columnar grain growth occur.

From postirradiation examination and data on unirradiated UO_2, Bates gave the following equation for the thermal conductivity of UO_2:

$$k = \frac{54}{18 + T}\ \frac{\rho}{\rho_0} + 4.37 \times 10^{-13}\ T^3 \quad , \qquad (1)$$

where

k = thermal conductivity, Btu/hr·ft·°F,

ρ = fuel density, g/cm^3,

ρ_0 = UO_2 theoretical density, g/cm^3,

T = absolute temperature, °R.

The first term in Eq. 1 is in accord with the normally observed decrease in thermal conductivity with increasing temperature for ionic solids. The second term is the contribution arising from the radiant heat transfer in porous solids. The agreement between observed and calculated values confirms the postulate that radiant heat transfer causes an enhancement of UO_2 thermal conductivity at high temperatures. Equation 1 predicts a value of 2.1 Btu/hr·ft·°F for the average thermal conductivity of UO_2 between 750 and 5000°F.

Although the discrepancy in the results of Bates[7] and Robertson et al.[5] is unresolved, it is clear that the average thermal conductivity of UO_2 is higher than the 1 Btu/hr·ft·°F previously supposed. Further, the discrepancy is minor if the fuel element heat ratings are modest and the central temperature of the UO_2 is not above 3000°F.

The above data were obtained with pressed and sintered UO_2 pellets and probably do not apply for the recently developed swaged and vibratory-compacted fuel elements. The recent preliminary data of Lamartine[8] indicate that the thermal conductivity of swaged UO_2 is about one-half that of pressed and sintered pellets with the same density.

Rates of Release of Fission Products

Numerous investigators[9-18] have established that the primary mechanisms of fission-gas release from UO_2 are diffusion and recoil below about 2700°F. The amount released by recoil is a function of the shape and density of the fuel body, but it is independent of temperature. Under the normal

operating conditions of most fuel elements, the recoil release is negligible compared with the amount released by diffusion. Thus, most studies to date have been concerned with the rate of diffusion of fission products in UO_2. Above 2700 to 2900°F, the processes of grain growth, columnar grain growth, and sublimation occur. The effects of these processes have not been assessed quantitatively, but they may account for the abnormally high rates of release observed while structural changes were occurring.[12,14]

A comprehensive review of work on fission-product release from UO_2 prior to the fall of 1959 was published by Cottrell et al.[19] A compilation of values of the diffusion coefficients of fission products reported by various investigators was given as well as the equations for predicting the fractions released under various conditions. The diffusion coefficients for inert gases in UO_2 showed considerable uncertainty, even in regions where the equations appeared to be applicable. The scatter in the data was about five orders of magnitude at 2550°F. Reported values of the activation energy were between 34 and 81 kcal/mole.[11] In view of these uncertainties, Cottrell et al.[19] concluded that any particular fuel material should be characterized by a measurement of the diffusion rate constant, $D' = D/a^2$, where a is the radius of a sphere whose volume is that of the average solid fuel particle. The quantity, D', arises naturally in the solution of the diffusion equation in spherical geometry and can also be used with success to characterize release from nonspherical particles. The wide scatter in D' values at a given temperature for UO_2 fabricated by different processes was concluded to result from lack of understanding of the effects of important parameters affecting release, such as thermal

conductivity, density, distribution of open and closed pores, grain growth, oxygen-to-uranium ratio, material discontinuities, and temperature structure.

More recent investigations have cleared up a number of these uncertainties. Long et al.[14] have shown by neutron-activation studies that there is good correlation between the diffusion rate constant and the BET surface area in the range 1500 to 2900°F. The UO_2 compacts varied in surface area from 7 to about 23 000 cm^2/g. Over this wide range, the following equation was found to apply:

$$\log_{10} D'_{2550°F}\ (sec^{-1}) = -13.55 + 2 \log_{10} S.A.\ (cm^2/g) \quad ,$$

where S.A. is the BET surface area.

From D' values and surface areas, the diffusion coefficient D was calculated and found to follow the equation

$$D\ (cm^2/sec) = 7.8 \times 10^{-6} \exp\ (-71\ 000/RT) \quad ,$$

where T is in °K, in the temperature range 1500 to 2900°F. In the temperature range 2900 to 4000°F, Long et al.[14] observed that the rate of evolution of fission products was higher than that expected from diffusion alone and that the slope of the fraction released versus the square root of time increased with time. The effect was attributed to grain growth, which was also observed to occur in this temperature range.

Independent studies of Toner and Scott[12] on the release of Xe^{133} from UO_2 using the neutron-activation technique confirmed the results

of Long et al.,[14] and the following relationship between the BET surface area and D′ was found to apply:

$$\log_{10} D'_{2550°F}\ (\mathrm{sec}^{-1}) = -14.0 + 2 \log_{10} \mathrm{S.A.}\ (\mathrm{cm}^2/\mathrm{g}) \quad .$$

Above 2900°F, the rate of release was higher than would be predicted on the basis of diffusion. This was attributed to sublimation of UO_2 and grain growth. Contrary to early work by Eichenberg et al.,[2] no correlation between D′ values and UO_2 density in the range 90 to 96% of theoretical was found. The differences between the results of the two investigations can probably be explained as differences in the samples used; all the measurements of Eichenberg et al.[2] was made on pellets fabricated by a single method, whereas the data of Scott and Toner were obtained on pellets fabricated by a number of commercial vendors and by a variety of methods. In the density range studied, the ratios of the BET surface area to the total surface area varied by two orders of magnitude for pressed and sintered pellets of the same density but fabricated by different techniques.

Stevens et al.[13] have observed that a correlation exists between UO_2 density in the range 93 to 97% of theoretical and D′ values for pellets made by a given method but that there are exceptions to the general behavior. These exceptions have been related to three microstructures which were anomalous for the density concerned. Stevens et al.[13] also found evidence which suggests that the initial high rate of release normally observed during neutron-activation tests,[9,12,14] is due to an oxygen-rich surface layer in which the xenon diffusion rate is much higher than in the bulk of the oxide.

As a result of the recent work, the scatter in the values of the diffusion coefficient for Xe^{133} in UO_2 has been considerably reduced from the 5 orders of magnitude reported by Cottrell et al.[19] to about 1.5 orders of magnitude. A compilation of recent values obtained at different laboratories is shown in Table II-1. The reasons for the remaining scatter are not known.

Table II-1. VALUES OF THE DIFFUSION COEFFICIENT FOR Xe^{133} in UO_2 AT 2550°F

Laboratory	D (cm^2/sec)	Type of UO_2	Reference
ORNL	7.5×10^{-16}	Sintered	12
Harwell	4.0×10^{-15}	Sintered	14
Battelle Memorial Institute	1.1×10^{-16}	Fused	17
Chalk River	5.1×10^{-16}	Fused	13

Values of D at 2550°F may be extrapolated to other temperatures by use of the Arrhenius equation once the activation energy is established. An average value of 65 kcal/mole was used by Cottrell et al. in making calculations. More recently, Long[14] reported a value of 71 ± 5 kcal/mole. Since this latter value was obtained in a hydrogen atmosphere, the effects of oxygen on the activation energy should have been negligible. Toner and Scott[12] and Stevens et al.[13] observed a range of activation energies for different types of UO_2. The former work was done in a vacuum and the latter was done in helium, so trace amounts of oxygen may have affected these results. Further work is needed to clarify present uncertainties.

As discussed previously, Long et al.[14] and Toner and Scott[12] observed that rates of release were higher than those predicted on the basis of

diffusion above about 2900°F. Until further information is obtained on the mechanisms responsible for the effect, it seems safest to anticipate that all the fission products are released from those portions of the UO_2 which are above 2900°F.

Studies of Fission-Gas Release from UO_2 During Irradiation

Two important limitations of the neutron-activation technique for studying fission-gas release are that the burnups are low, and the effects of fissioning cannot be evaluated. These limitations are removed when the rates of release are measured in in-pile capsules through which a carrier gas is passed. By monitoring the amounts of short-lived krypton and xenon isotopes collected in suitable traps, much information can be obtained on rates and mechanisms of release. A major difficulty with such sweep capsules is a temperature limitation set by the materials of construction. Additional problems arise in temperature measurement and control, flux monitoring, and maintenance of the purity of the sweep gas.

The rates of release of fission products from UO_2 in sweep capsules have been measured by Carroll[18] and by Melehan and Rough.[17] Additional studies are being made by Stubbs and Webster[20] at Harwell. Carroll's measurements were made over the temperature range 800 to 1800°F. In the range 800 to 1000°F, Carroll observed the temperature-independent release that is characteristic of recoils. At higher temperatures, the rates of release increased markedly with increasing temperature in a manner characteristic of diffusion. The activation energy for the diffusion of Kr^{88} from UO_2 was found to be 30 kcal/mole. It was also observed

that iodine diffused in UO_2 at about the same rate as xenon and krypton and had the same activation energy.

The low-activation energy suggested that the UO_2 was oxidized during the test. This was confirmed by postirradiation examination, as well as by observations of the release of stored energy from the UO_2. Carroll concluded that trace amounts of oxygen in the helium sweep gas were sufficient to oxidize the UO_2 during the test period. Oxidation was eliminated in later tests by the use of 3% H_2–97% He as the sweep gas. In both atmospheres, Carroll observed bursts of activity on heating and cooling of the sample.

In a second series of tests,[21] Carroll and Baumann used platelets of ThO_2 outside of the UO_2 to trap recoils. Since the diffusion rates of volatile fission products in ThO_2 could be expected to be about the same as in UO_2, it was hoped that the ThO_2 would eliminate direct recoils. However, results showed no significant differences between the recoil contributions in the presence and absence of the ThO_2; this observation suggests that the "recoil" contribution does not come solely from direct recoils.

Melehan and Rough[17] helped to clarify this point with studies of two types of UO_2 in the temperature range 500 to 1500°F. One sample was a single crystal of fused oxide and the other was a pressed and sintered platelet of UO_2. The surface areas for direct recoil (geometric surfaces) were the same in the two samples; however, the BET surface area of the pressed and sintered material was about 100 times that of the fused oxide sample. In the temperature range studied, only a very small temperature dependence of the release was observed,

but surprisingly the rates of release were found to be proportional to the total surface area and not the geometric area.

Melehan and Rough[17] explained the anomalous release in these low-temperature experiments in terms of a knock-out mechanism originally proposed by Lewis.[22] According to this theory, gas release occurs by localized evaporation of the fuel surfaces during the passage of fission recoil fragments near the surface. The energies of these vaporized fission gases would be much lower than primary recoils and thus would not re-enter the fuel. Since some of these fission gases would be retrapped by the condensing UO_2, the amounts of fission gases would be dependent upon the efficiency of recapture. However, in neither the work of Carroll[18,21] nor that of Melehan and Rough[17] is there any evidence that the diffusion of volatile fission products is greatly increased by simultaneous fissioning or that there is any increase in release which could be directly attributed to burnup.

Regarding the work with sweep capsules, it is concluded that a new mechanism of release has been demonstrated. However, the temperature range studied has been too limited for use of the data to predict the contribution of this mechanism to the release of fission products in fuel elements that normally operate at considerably higher central temperatures.

A limited amount of information is available on the release of fission products from UO_2 at high burnups. On the basis of data obtained from in-pile capsules, Toner and Scott[12] concluded that there was no marked effect of burnup on D' values up to a burnup of 16 400 Mwd/MT. At higher burnups at temperatures below 1500°F, however, Bleiberg et al.[23] found that

the crystal structure of UO_2 was destroyed and that a large fraction of the fission gas was released. The effect was found to be initiated at a burnup of 60 000 Mwd/MT. Such an effect is not surprising in view of the change in composition associated with this burnup.

Estimates of Fission Product Release in Fuel Capsules

A relatively simple method for predicting the amounts of stable fission gases released in fuel elements was given by Cottrell et al.[19] It was proposed that the temperature distribution could be computed by conventional heat transfer equations. An average temperature is then computed by use of the equation

$$\bar{T} = \frac{2T_{max} + T_{surface}}{3} \quad . \tag{2}$$

A D' value associated with $\bar{T}$ is then selected on the basis of neutron-activation test results, and the fraction released is calculated by use of the equation

$$F = \frac{4}{\sqrt{\pi}} \sqrt{D't} \quad , \tag{3}$$

where t is the irradiation time. Equation 3 applies for low values of the fraction released and when the structure of the UO_2 is unchanged by irradiation. For a given set of D' values, the fractions computed by the above method were found to agree well with fractions computed by summing the release from a number of incremental volumes in the fuel elements.

Scott[24] used the method described above to estimate the fractional release of Kr^{85} from prototype fuel elements for the Experimental Gas-Cooled Reactor (EGCR). These elements were irradiated in the Oak Ridge National Laboratory Research Reactor (ORR) and actual fractions released were measured by postirradiation puncture tests at the Vallecitos Laboratories of the General Electric Company. Maximum burnups were of the order of 2000 Mwd/MT. In making estimates, a value of 1.62 Btu/hr·ft·°F for the thermal conductivity of UO_2 in the temperature range 1300 to 2900°F was used, and D' values were calculated from measured BET surface areas by using the neutron-activation test results of Toner and Scott.[12] The data were analyzed for the case of a 3-mil helium-filled gap and no gap between fuel and cladding and for the case of no gap. The results showed that the measured values of the Kr^{85} released generally fell within or close to the limits set by the 3-mil helium-filled gap and no gap. Further, most of the measured values were close to those predicted for a 3-mil helium-filled gap when the type 304 stainless steel cladding temperature was about 700°C. For those elements whose cladding temperatures were about 800°C, the measured release values corresponded more closely to the condition of no gap. These results are reasonable, since the external pressure on the capsules was 320 psia and creep with subsequent cladding collapse would be more likely to occur at the higher temperatures.

Robertson[25] used a similar method of analysis to predict the dependence of fission-gas release on heat flux. These predictions were tested for a series of capsules that were irradiated for the same time at different heat fluxes and which contained specimens from the same batch of sintered

UO_2. Predicted values were compared with the amounts of free gas collected by puncturing the cladding after irradiation in an evacuated enclosure. Close agreement between predicted and observed release was found. Robertson[25] asserts that "these tests and their interpretation have shown that it is unnecessary to assume anything like 100% release in even the most highly rated [high heat flux] fuel elements currently designed." He also remarks that "designing for 100% release constitutes a deplorable waste."

Concluding Remarks

Although much work is still needed to clarify mechanisms of fission-gas release from UO_2, reasonable estimates may now be made of the amounts released in in-pile capsules, at least for pressed and sintered UO_2 at modest heat fluxes. The major areas of uncertainty requiring further investigation are the fuel-cladding interaction and the in-pile behavior and thermal conductivities of nonsintered UO_2 (e.g., vibratory-compacted and swaged oxides). Emphasis should also be placed on the long-term effects of structural changes on release in fuel fuel elements operating at high heat fluxes with both sintered and nonsintered oxides. (D. F. Toner and J. L. Scott)

ACTIVITY BUILDUP IN PRESSURIZED-WATER REACTOR SYSTEMS

The corrosion of in-core metals and the transport of "crud" by the coolant in a pressurized-water system results in the buildup of activity throughout the primary system. As experience has been gained with operating reactors, it has been observed that this activity buildup produces radiation fields that are a safety hazard to personnel. The radiation levels encountered may only limit accessibility during routine operation, but they pose more serious problems during extensive maintenance operations. This article brings up to date a previous article on this subject.[26]

The term "crud" is used to designate the corrosion and wear products produced in the circulating system. Studies of the behavior of the crud indicate that a portion of it continuously migrates from one part of the system to another. At any particular time, only very small amounts (<1 ppm) of dissolved and suspended material are present in the circulating coolant,[27] but the rapid accumulation of a large number of coolant cycles results in effective transport. In the course of this transport, the crud is in neutron-flux regions a portion of the cycle time. Deposition in high flux regions has been attributed to two mechanisms,[28] a hydraulic effect that operates mainly on the particulate matter and radiation effects that operate on colloidal or soluble matter. The activated nuclides which reenter the coolant stream are distributed through the crud of the system both by particulate- and ion-exchange processes that are usually irreversible.[27] Table II-2 lists some of the important radioactive nuclides which have been found in the circulating streams of pressurized-water reactors.[29]

Table II-2. REACTIONS IN COOLANT THAT PRODUCE ACTIVATION DURING NORMAL REACTOR OPERATION

Reaction	Radiation	Half Life
$Fe^{58}(n,\gamma)Fe^{59}$	Beta, gamma	45 days
$Ni^{58}(n,p)Co^{58}$	Gamma	71 days
$Co^{59}(n,\gamma)Co^{60}$	Beta, gamma	5.2 years
$Cr^{50}(n,\gamma)Cr^{51}$	Gamma	27 days
$Fe^{54}(n,p)Mn^{54}$	Gamma	300 days
$Ta^{181}(n,\gamma)Ta^{182}$	Gamma	112 days

The importance of the various nuclides in crud activity buildup is indicated by data given in Table II-3 for the Shippingport Pressurized-Water Reactor (PWR) and the Army Package Power Reactor (SM-1).[27]

Table II-3. INDUCED ACTIVITIES FOUND IN PRESSURIZED-WATER-REACTOR CRUD

Nuclide	Activity (dpm/mg of crud)		Crud Activity as Percentage of Total Activity	
	PWR	SM-1	PWR	SM-1
Co^{60}	5.8×10^6	3.9×10^5	39.2	10.8
Co^{58}	2.5×10^6	21.0×10^5	16.9	58.0
Fe^{59}	1.8×10^6	5.0×10^5	12.1	13.8
Cr^{51}	2.2×10^6	4.9×10^5	14.8	13.5
Mn^{54}	0.8×10^6	1.4×10^5	5.4	3.9

The distribution of the activity among the several nuclides is, of course, dependent on the specific reactor. Thus the relatively high Co^{60} activity in the PWR crud shown is attributed to the fact that the control rod mechanisms are located above the reactor core.[30] As a result, corrosion and wear products from Stellite (~50% Co) control

rod components may drop down and lodge in high-flux regions with resultant high production of activated cobalt. Similarly the relatively high quantity of Co^{58} in SM-1 is attributed to the fact that the SM-1 fast-to-thermal flux ratio is seven times that of the PWR; this, combined with the stainless steel cladding, favors the high energy (n,p) reaction[27] on Ni^{58} in the SM-1 relative to the PWR, which uses Zircaloy cladding.

The buildup of activity and its nature, then, is a function of many variables, including primary system materials, core materials, reactor characteristics, coolant chemistry, primary system flow, and primary system geometry. Some of these variables may apply generally to reactors of given type, whereas others may be specific to a given reactor, with characteristic changes in the activity buildup encountered. In general, of course, the long-lived activities are the most important, since they will predominate after long periods of power production. Fission-product activity has not been found to be a significant contributor to activity buildup so far.[27]

The radiation levels that result from buildup of activated crud range from 0.1 r/hr or less to several r/hr. Cutie pie measurements of the SM-1 primary system[27] gave the following radiation levels:

1. An elbow in the pressurizer line reached 4 r/hr, but the level dropped to 50 mr/hr when the system was drained. The geometry of the elbow was such as to provide a trap for loose material which presumably flushed out.

2. Primary piping radiation levels ranged from 0.2 to 0.6 r/hr, depending on geometry factors.

3. The impeller and shroud of one of the 4000-gpm primary coolant pumps read 2 r/hr at contact.

4. The radiation levels were 0.4 and 1.4 r/hr on the vaporizer and superheater sides of the steam generator, respectively. This compares with a level of 2 r/hr "as close as practical" to the tube sheet of a PWR steam generator.

5. The dose-rate buildup at all points was approximated by the equation $y = cx^2$, where y is the dose rate in mr/hr, x is the number of months since reactor startup, and c is a constant ranging from 0.2 to 1.0, depending on the particular point in the system.

Similarly, the work area radiation field during removal of the rod drive mechanisms of the PWR in the first refueling operation was 150 to 400 mr/hr with shielding and 1 r/hr without shielding. Individual components of the mechanisms ranged in activity level from a few tenths to a few r/hr as a result of deposited crud.[30]

The presence of such radiation fields results in an appreciable increase in the hazards to personnel. This is particularly true during maintenance operations when the plant is, of necessity, in an abnormal condition. The radiation levels and possibilities for spread of contamination during such periods require the application of stringent measures for the protection of personnel. Health physics supervision of radiation dosages, protective clothing, and equipment, as well as facilities for decontamination and active waste disposal, are needed, not only for safety, but also for efficient work progress. Thus there is material incentive for the reduction of such buildup from both the safety and economic viewpoints.

As experience with operating reactors has directed attention to the importance of this phenomenon, studies have been undertaken to provide means of controlling it. The most obvious means of reducing the crud concentration in the coolant is use of the following design features:

1. Elimination of crud traps and accumulation areas, such as cracks, crevices, and points of abrupt change in flow.

2. Minimization of moving part wear and location of such parts so that wear products are least likely to get into the coolant stream.

3. Wherever possible, use of materials which are not a source of undesirable nuclides, e.g., minimize cobalt content of stainless steel fuel cladding[31] and substitute noncobalt-containing alloys, particularly in the core region.

The mathematical equations developed by several investigators[32,33] are useful in developing special materials specifications. Experience in operating reactors indicates[34] that activity buildup is reduced when the coolant pH is raised from 7 to 10. This is attributed to a reduction in transportable crud as a result of lower corrosion rates and the formation of more tenacious films at the higher pH's. Lithium hydroxide is the recommended reagent for pH control for a variety of reasons: it does not cause radiolytic dissociation, it is the least aggressive reagent with respect to caustic stress-corrosion cracking, and there is a considerable backlog of proof-testing of it in operating reactors. Careful exclusion of oxygen from the coolant system has also been demonstrated to be important, presumably because increased corrosion due to the presence of oxygen results in higher crud levels.[35]

Fixed-bed mixed-ion exchangers are conventionally used for coolant purification; however, in the KAPL-37 Test Facility and the S3G Reactor[32] their effectiveness is rather low (0.10 to 0.15) when judged on the basis of the ratio of the activity removed by the purifier to the activity of the out-of-flux system. Other purification means, including evaporation, hydrocloning, and centrifugation, have been examined,[36] but they were discarded on the basis of higher cost. It is conceivable, however, that another system, such as evaporation, which could give very high decontamination[37] factors for all constituents of the recycled water, might give lower over-all costs when activity buildup costs are completely evaluated.

Considerable effort has gone into the development and testing of decontamination reagents and techniques. Several reagents and techniques have been tested in the laboratory and in loops, but none have been tested in a reactor. (E. G. Bohlman)

STUDIES IN REACTOR KINETICS

(Material held over from draft of Vol. 3, No. 1 to be inserted here; see draft pages 74 through 93 and references 24 through 60 on pages 111 through 115.)

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THE KARLSRUHE SYMPOSIUM ON CRITICALITY CONTROL

A symposium on Criticality Control in Chemical and Metallurgical Plants was sponsored by the European Nuclear Energy Agency for the Organization for European Economic Cooperation and was held in collaboration with the Kernreaktor Bau- und Betriebsgesellschaft mbh (Nuclear Reactor Construction and Operation Co.). Meetings were held May 2–5, 1961, at the Nuclear Reactor Station in Karlsruhe, Germany.

This was the first international meeting devoted entirely to this topic, and its stature was far greater than that of any like discussions in the United States. Attendance at the symposium was limited to approximately 100 persons representing the Western European Countries, the United Kingdom, the United States, and international organizations with nuclear interests. The participants were primarily criticality specialists, but engineers having responsibilities for safety in operations with fissile materials were among those present.

Twenty-one papers were presented at the symposium: six from the United Kingdom, five from France, one from the International Atomic Energy Agency, and nine from the United States. The proceedings of the symposium, including the discussion of each paper, are to be published by the Organization for European Economic Cooperation in the fall, and the resumés below refer primarily to the written papers rather than to the abbreviated versions which were presented verbally at the meetings. Professor Werner Heisenberg, Director of the Max Planck Institute for Physics and Astrophysics and Honorary Chairman of the Symposium, gave the opening address, followed by Dr. Leandre Nicolaidis of the European Nuclear Energy Agency.

Theoretical and Calculational Methods

A comprehensive introduction to the subject of nuclear safety, including both the physics of criticality and its application to practical problems was presented by E. R. Woodcock (UK—Risley). The following paper, by R. T. Ackroyd (UK—Risley) and E. D. Pendlebury (UK—Aldermaston), surveyed in relatively detailed fashion the theoretical calculation methods employed to determine criticality. This review gave the mathematical bases for diffusion theory and its variational and perturbation techniques and for the spherical harmonics, S_n, and Monte Carlo methods. Detailed applications of these techniques were not discussed, but adequate literature references to their use were made. The expositions of each method were sufficiently complete to indicate clearly the characteristics of each. Further, the authors commented on the conditions under which each calculation can be properly applied, and made recommendations regarding cheapening the cost of extensive calculational programs by the use of relatively simple computational techniques taken in conjunction with the more advanced methods. The paper concluded with a note on cross-section data requirements.

A paper by P. Benoist, R. Caizergnes, Clouet d'Orval, and B. Duchemin (France—Saclay) furthered the discussion of theoretical methods. These authors placed particular stress on data requirements, the availability and characteristics of various computer programs, and the application of these calculational methods to specific types of critical systems.

Experimental Results

A review of criticality data was presented by A. D. Callihan (US—ORNL). Emphasis was placed largely on experimental data obtained recently,[1-6] particularly data from interaction tests and experiments for determining

critical parameters at low U^{235} isotopic enrichments. Of particular interest was an extensive presentation of data relating critical mass, volume, and thickness to moderation and density. Much data had been assembled and correlated covering a wide range of isotopic compositions, materials, and degrees of heterogeneity. In addition, the survey included results from experiments on U^{233} and plutonium, as well as from a variety of rather specialized experiments involving U^{235} of various geometries and a number of different moderators.

Various correlations between experimental and theoretical critical data were presented by H. C. Paxton (US—LASL). Topics such as the adjustment of empirical data using relatively simple theoretical models and the use of computation as a substitute for experiments were discussed. Many comparisons of the results of experiments with computed results were given, and extensive computational studies by Mills[7] and Roach[8] were reported. A comprehensive survey of the minimum critical dimensions of aqueous solutions of U^{235}, U^{233}, and Pu^{239}, based on S_n calculations, was presented. Paxton concluded with a discussion of the reliability of the computed results and the selection of safety factors when such data are to be used for designing nuclear systems and operating procedures.

A session of the Symposium dealt with the facilities for experimental criticality research. The equipment, methods, organization, and types of experiments under way were discussed by Clouet d'Orval (France—Saclay), by J. G. Walford (UK—Dounreay) and A. F. Thomas (UK—Aldermaston), and by A. D. Callihan (US—ORNL).

Criticality Control in Plant Design

The application of criticality control principles to plant design was discussed by J. N. Hill and H. F. Parker (UK—Risley), who presented the concepts involved in safe-by-shape controls and the determination of safe geometric parameters from criticality data. Examples of process and mechanical design problems which may arise in such treatment were considered, and practical applications of the safe-by-shape approach at the chemical separations plant of the Dounreay fast reactor were discussed.

Concentration control as a means of effecting nuclear safety was discussed by J. Gunther, (France—Dijon). Considerable attention was given to the selection of safe concentration limits by using suitably conservative cross sections for calculation. The relationship between the process chemistry and criticality problems was discussed in connection with equipment such as extraction columns in which fissionable material may appear in more than one chemical phase.

Mass batch control and its applications were described by A. J. Mallett, (US—UCNC). The safety factors for such mass batches, the hazards of batch operation, and the problems of sampling analysis and administrative controls were discussed. In addition, the use of mass batch controls in combination with volume and moderation controls was considered and examples were quoted.

The somewhat controversial question of the use of nuclear poisons for nuclear safety controls in an industrial milieu was discussed in two papers. Potential homogeneous and heterogeneous poisoning methods for controlling criticality were reviewed by J. W. Wachter (US—UCNC),

who cited various experiments performed to evaluate poisons for nuclear safety purposes. The use of poisons as additional, primary, or contingent controls was discussed, distinctions among the various types of control being drawn according to the degree of reliance placed on the presence of the poison. A specific example of the use of fixed nuclear poisons was provided by R. Caizergues (France—Saclay), who described a theoretical study of a vessel poisoned by radial fins containing boron or cadmium. From these papers, it appeared that where the use of poisons has been thoroughly established as being mechanically, chemically, and nuclearly safe, poisoned equipment can be considered equivalent in safety to the corresponding geometrically safe vessel. To establish this safety may require a costly experimental or theoretical study, but economic savings may be very great.

Specific applications of the various control principles were discussed in papers by J. W. Wachter and W. T. Mee (US—UCNC) dealing with a typical plant for manufacturing enriched UO_2 fuel elements and by E. D. Clayton (US—GE) concerning nuclear safety in the chemical and metallurgical processing of plutonium. In both papers the flow of fissionable material was traced through its various processing steps and the nuclear safety problems of each step were discussed at some length. Although such descriptions are highly specific, they serve to illustrate that practical nuclear safety remains something of an applied art.

The problems involved in the storage and transport of fissionable material were considered in the next session. A paper by R. T. Ackroyd (UK—Risley), F. R. Charlesworth, and A. F. Thomas (UK—Aldermaston) dealt with methods of calculation of interacting arrays, with special reference to

storage problems. The most frequently used theoretical and empirical approaches to the problem were discussed in considerable detail. The well-known solid-angle method was described, as well as two unpublished methods: the "density analogue approach" of H. C. Paxton and the "interaction rules" of D. Kenyon. Greatest emphasis was placed, however, on the unified interaction theory developed by Ackroyd et al.[9] and the interaction parameter method of Scriven and Thomas.[10] The authors conclude that the empirical methods of solid angle, density, etc., suffice where plant operations require relatively open spacing of fissile material to enable loading and unloading of storage areas. Where tighter spacings are of advantage, they could be obtained through the greater accuracy of more sophisticated calculations. The latter approach should be adopted, however, when considering a plant having a wide variety of vessels of fissile materials of differing moderations. In such a case, it is necessary to examine in detail the effects of such factors as spacing, geometry, and interspersed moderators, and the cost of elaborate computation becomes worthwhile.

The problems which arise in considering such interacting arrays were brought out in a discussion of regulations to be proposed by the International Atomic Energy Agency to govern the international transport of radioactive and fissionable material. The proposed regulations were prepared by R. F. Barker and were presented by C. Erginsoy (IAEA). Much of the discussion dealt with the proposals themselves, which have since been published,[11] but the technical bases for the proposals were also considered in some detail, and the safety factors were discussed.

The organizational structure by which criticality control is effected was described for their several countries by P. LeCorche and R. Caizergues (France—Saclay), H. Howells and J. T. Daniels (UK—Risley), and C. L. Schuske (US—Dow). These papers dealt with the administrative aspects of such controls and, in addition, considered such general methods of control as instrumentation, inventory accountability, batch limitations, and the like.

Criticality Accidents

A review of criticality accidents was presented by W. R. Stratton (US—LASL) that was based on an earlier review.[12] Several accidents were described and discussed. Tables III-1 and III-2, taken from this paper, summarize the accidents on which the discussion was based. Conclusions were drawn regarding the characteristics of prompt excursions and the quenching mechanisms which terminate the reactions. Such practical considerations as the use of a neutron source as a means of limiting the burst intensity were also discussed.

Conclusions

The discussions which took place following each paper were of great interest. It appeared that, whereas installations in the United States have utilized the experimental approach, greater efforts have been applied elsewhere to the calculation of critical conditions, primarily because of limited fissile material. The result has been the application of correspondingly greater safety factors. Moreover, the relative cost of establishing the nuclear safety of a production plant in comparison

Date	Location	Active Material	Physical Damage
December 1949	Los Alamos Scientific Laboratory, New Mexico	~1 kg of U^{235} in $UO_2(NO_3$ in 13.6 liters of water	None
November 16, 1951	Hanford Works, Richland, Washington	1.15 kg of Pu in $PuO_2(NO$ in 63.8 liters of water	None
May 26, 1954	Oak Ridge National Laboratory, Tennessee	18.3 kg of U^{235} in UO_2F_2 55.4 liters of water	None
February 1, 1956	Oak Ridge National Laboratory, Tennessee	27.7 kg of U^{235} in UO_2F_2 58.9 liters of water	Warping of bottom of cylinder
June 16, 1958	Y-12 Processing Plant, Oak Ridge, Tennessee	2.5 kg of U^{235} in $UO_2(NO$ in 56 liters of water	None
December 30, 1958	Los Alamos Scientific Laboratory, New Mexico, Pu Processing Plant	3.27 kg of Pu in $PuO_2(NO$ in ~168 liters of water	None
October 16, 1959	Chemical Processing Plant, Idaho Reactor Testing Area	34.5 kg of U^{235} in ~800 ters of $UO_2(NO_3)_2$ and w	None
January 25, 1961	Chemical Processing Plant, Idaho Reactor Testing Area	8 kg of U^{235} in $UO_2(NO_3)$ 40 liters of water	None
August 21, 1945	Los Alamos Scientific Laboratory, New Mexico	6.2 kg of δ-phase Pu	None
May 21, 1946	Los Alamos Scientific Laboratory, New Mexico	6.2 kg of δ-phase Pu	None
April 18, 1952	Los Alamos Scientific Laboratory, New Mexico	92.4 kg of uranium metal riched to 93% U^{235}	None
February 3, 1954	Los Alamos Scientific Laboratory, New Mexico	53 kg of uranium metal riched to 93% U^{235}	Slight warping of pieces
February 12, 1957	Los Alamos Scientific Laboratory, New Mexico	54 kg of uranium metal riched to 93% U^{235}	Warping, oxidation, near melting close to center
June 4, 1945	Los Alamos Scientific Laboratory, New Mexico	35.4 kg of uranium enric to ~83% U^{235}, 1/2-in. c	None
February 1, 1951	Los Alamos Scientific Laboratory, New Mexico	2 cylinders of uranium, and 38.5 kg enriched to U^{235}	Slight oxidation
July 6, 1952	Argonne National Laboratory, Illinois	6.8 kg of U^{235} in oxide particles in plastic	Plastic destroyed
December 12, 1952	Chalk River, Canada	Normal uranium	Core ruined
July 22, 1954	Reactor Testing Area, Idaho	U-Al plates clad with Al	Reactor destroyed
October 15, 1956	Vinca, Yugoslavia	3996 kg of normal uraniu	None
January 3, 1961	Reactor Testing Area, Idaho	U-Al plates clad with Al	Not yet determined
February 11, 1945	Los Alamos Scientific Laboratory, New Mexico	UH_3 pressed in Styrex	UH_3-Styrex cubes swollen and blistered
1953	U.S.S.R.	Unknown	Unknown
November 29, 1955	Reactor Testing Area, Idaho	1/2-in. U^{235} rods	Core molten
July 3, 1956	Los Alamos Scientific Laboratory, New Mexico	58 kg of uranium enriche 93% U^{235}, 2- and 5-mil	None

*W. R. Stratton, A Review of Criticality Accidents, Proceedings o the Symposium on Criticality Control in Chemical and Metallurgical Pla Karlsruhe, Germany, May 2-5, 1961, to be published.

Table

Incident	Observed Fission Yield	Spike Yield (fissions)	Platea (fiss	Initial Plateau Power (fissions/sec)	Probable Quenching Mechanism
Water boiler reactor control rods removed by hand, December 1949	$3–4 \times 10^{16}$	$\sim 3 \times 10^{15}$	2.7–3.	$2–3 \times 10^{16}$	Rise of neutron temperature
Plutonium solution assembly, cadmium rod removed too rapidly, November 16, 1951	8×10^{16}	8×10^{16}	0.0	0.0	Microbubbles, neutron temperature, and poison rod
Uranium solution assembly, central poison cylinder tilted to less favorable position, May 26, 1954	10^{17}	5×10^{16}	5×10	4.8×10^{16} to 1.4×10^{17}	Microbubbles and neutron temperature
Uranium solution assembly, wave motion created by falling cadmium sheet, February 1, 1956	1.6×10^{17}	1.6×10^{17}	0.0	0.0	Microbubbles and neutron temperature
Uranium process solution combined with wash water in 55-gal drum, June 16, 1958	1.3×10^{18}	$\sim 10^{16}$	1.29 ×	No estimate	Microbubbles, neutron temperature, and boiling
Separated phases in plutonium process tank, December 30, 1958	1.5×10^{17}	1.5×10^{17}	0.0	0.0	Microbubbles, neutron temperature, and stirring action
Uranium process solution siphoned from safe to unsafe geometry, October 16, 1959	4×10^{19}	$\sim 10^{17}$	~4 × 1(	No estimate	Microbubbles and neutron temperature
Uranium process solution pumped from safe to unsafe geometry, January 25, 1961	6×10^{17}	No estimate	No est:	No estimate	Microbubbles and neutron temperature
Plutonium core reflected with tungsten carbide, August 21, 1945	10^{16}	$\sim 1.8 \times 10^{15}$	~8.2 ×	$\sim 1.2 \times 10^{16}$	Thermal expansion
Plutonium core reflected with beryllium, May 21, 1946	3×10^{15}	$\sim 1.8 \times 10^{14}$	~2.86 >	$\sim 1.2 \times 10^{16}$	Thermal expansion
Jemima, a cylindrical unreflected U^{235} assembly, April 18, 1952	1.5×10^{16}	$\sim 10^{15}$	~1.4 ×	$\leq 10^{17}$	Thermal expansion
Godiva, a bare U^{235} sphere, February 3, 1954	5.6×10^{16}	5.6×10^{16}	0.0	0.0	Thermal expansion
Godiva, a bare U^{235} sphere, February 12, 1957	1.2×10^{17}	1.2×10^{17}	0.0	0.0	Thermal expansion
Psuedosphere of uranium cubes, water reflected, June 6, 1945	$3–4 \times 10^{16}$	$\sim 3 \times 10^{15}$	2.7–3.7	No estimate	Boiling
Critical separation experiment, two large U^{235} masses in water, February 1, 1951	10^{17}	$\sim 6 \times 10^{15}$	~9.5 ×	$\sim 10^{17}$	Boiling

Incident	Observed Fission Yield	Spike Yield (fissions)	Platea (fiss	Initial Plateau Power (fissions/sec)	Probable Quenching Mechanism
UO_2 particles in plastic, water moderated, July 6, 1952	1.22×10^{17}	1.22×10^{17}	0.0	0.0	Plastic bubbles
NRX reactor, normal uranium in D_2O, graphite reflected, December 12, 1952	1.2×10^{20}	Not prompt critical	Not pro critic	0.0	D_2O drained from core
Borax reactor, Al-U alloy, water moderated, July 22, 1954	4.68×10^{18}	4.68×10^{18}	0.0	0.0	Boiling
Critical assembly of normal uranium rods in D_2O, unreflected, October 15, 1958	2.5×10^{18}	Not prompt critical	Not pro critic	0.0	Cadmium rods inserted in core
SL-1 reactor Al-U alloy, water moderated, January 3, 1961	1.5×10^{18}	1.5×10^{18}	0.0	0.0	Boiling
The Dragon Reactor, UH_3 pressed in Styrex, February 11, 1945	$\sim 6 \times 10^{15}$	$\sim 6 \times 10^{15}$	0.0	0.0	Thermal expansion
Experimental reactor, U.S.S.R., 1953	Unknown	Unknown	Unknown	Unknown	Unknown
EBR-1, fast breeder reactor, U^{235} rods in NaK, November 29, 1955	4.7×10^{17}	Not prompt critical	Not pron critica	0.0	Reflector dropped from core
The Honeycomb, U^{235} foils moderated with graphite, July 3, 1956	3.2×10^{16}	3.2×10^{16}	0.0	0.0	Motion of safety rods

*W. R. Stratton, A Review of Criticality Accidents, Proceed of Criticality Control in Chemical and Metallurgical Plants, Ka May 2–5, 1961, to be published.

with the over-all initial investment appears, at the present time, considerably greater in the United Kingdom than in the United States.

An important benefit of these discussions was the direct exchange of specific experimental data. The cost of both theoretical and experimental data acquisition has become so great that the opening up of channels of communication among the participants will be of continuing mutual benefit.

The proceedings of the Symposium, when published, will be the most comprehensive treatment of the theoretical and practical aspects of nuclear safety that has yet appeared. As such the proceedings will be of great benefit for those entering the field of fissionable material processing and a valuable reference work for those presently engaged in nuclear safety problems. (J. W. Wachter)

MEASUREMENT OF CONTROL ROD EFFECTIVENESS

The potential nuclear hazard arising in the operation of chain-reacting systems can be directly related to the amount of excess reactivity available and the method of its control. The worth of a control rod in a reactor is therefore a measure of the reactivity capable of being suppressed, as well as an indication of the margin of safety in the reactor operation. The prominence of control rod effectiveness in the evaluation of nuclear safety warrants this presentation of the current status of methods of its measurement. Reactors in which the active core is not large are amenable to most of the experimental procedures practiced today; however, with very large reactor cores, aside from calculational techniques, the absolute determination of control rod worth has remained a "gray" area. Recent application to this problem of the PCTR-k_∞ measuring technique that is available through the use of the Physical Constants Testing Reactor at Hanford appears to be promising.

Description of Measurement Techniques

The rod-oscillation technique is an application of pile-oscillator[13,14] measurements to a reactor in which an absorbing rod, being displaced periodically, plays the role of an oscillating absorber. In practice, the associated oscillating signal from a suitable detector is recorded. The relation between changes in amplitude and reactivity increments is experimentally established by associating a change in amplitude with a known reactivity change. This method is advantageous because the time required for each measurement is short (~5 min), and the reactor power need not

be constant, since an alternating component of the power is being measured rather than the power itself. The small deviation from an average power level during the course of a measurement obviates the need for equilibrium of the delayed-neutron emitters. Although comparison of the amplitudes of periodic signals derived from successive measurements on several rods yield only relative values of their effectiveness, Jankowski, Klein, and Miller[15] have presented an absolute evaluation by solving the kinetic equations and expressing the results in terms of a Fourier series. This absolute evaluation is applicable only to oscillations of small amplitude.

One of the early standard methods used for rod-worth measurements was the rod-drop technique. It is simple and direct. The rod is rapidly inserted into a reactor operating at a constant neutron level and the decreasing counting rate, C, is observed as a function of time. When the counting rate is extrapolated back to the time of rod insertion, its worth may be expressed as

$$\frac{\rho}{\beta} = \frac{C_1 - C_2}{C_2} , \qquad (1)$$

where the subscripts 1 and 2 denote the counting rate, C, before and after insertion, respectively, ρ is the reactivity, and β is the delayed-neutron fraction. It is assumed that the change in reactivity is instantaneous and that the prompt-neutron lifetime is short compared with the lifetime of the shortest delayed group. Usually, the finite time required for insertion causes uncertainty in the evaluation of C_2. Kolar and Kloverstrom[16] eliminate this difficulty by using the determined stable period, T, and

the inhour equation in the form

$$\rho = \frac{l^*}{T} + \gamma \sum_{1}^{6} \frac{\beta_i}{1 + \lambda_i T} , \qquad (2)$$

where γ is the ratio of effectiveness of delayed neutrons to prompt neutrons, l^* is the prompt-neutron lifetime when the system is critical, and λ_i and β_i are the decay constant and delayed-neutron fraction for each of the six delayed neutron groups, respectively.

In the rod bump-period technique, with the reactor critical and at a constant neutron level, the rod to be evaluated is displaced a small amount and the associated reactor period is determined. The inhour equation then yields the associated reactivity, and this, in turn, is related to the rod insertion distance by means of first-order perturbation theory, which adequately[16] expresses the differential effectiveness relation. The major disturbing feature appears to be the time consumed, although the problem of establishing the constant neutron level between increments can be avoided, thus eliminating an elusive time variable. For example, Williams et al.,[17] by means of a relative measurement of two control rods in the APPR critical experiments, obtained a calibration from successive measurements of positive and negative periods resulting from alternate compensating displacements.

The pulsed-neutron technique has been adequately described by Simmons.[18,19] Succinctly, a subcritical multiplying system is subjected to a burst of neutrons, and the resulting stable decay of the neutron population is observed. The observation begins after the initial transients of the burst

have died out and continues until the delayed neutrons dominate the total neutron population. Over the time of measurement the behavior is exponential. The resulting relation between the decay constant and the reactivity is

$$\alpha = \alpha_0 \left(1 - \frac{\rho}{\gamma\beta}\right) , \qquad (3)$$

where $\alpha_0 = \gamma\beta/l^*$ is the decay constant at delayed critical.

The source-jerk technique is similar to the rod-drop technique, except that the reactor is subcritical and the constant neutron level is that due to the multiplication of the neutron source present. The source is suddenly removed and the procedure outlined in connection with Eq. (1) is followed. The limitations of this method stem directly from source strength and its effect on detector response to a weak neutron field.

The PCTR-k_∞ measurement[20,21] has been extended to the evaluation of negative reactivities and, thus, is directly applicable to measurements of control rod strengths. Briefly, the PCTR is a null reactivity test assembly consisting of three concentric regions: a central test cavity surrounded by a buffer zone which, in turn, is surrounded by the reactor composed of enriched uranium rods in graphite. The procedure is to match the neutron flux or adjoint flux spectrum to that of the cell by means of the buffer region (the material to be placed in the cavity may also be used here). Then the test material is poisoned (usually with copper) until there is no observed change in reactivity with or without the test material in the cavity. When applied to control rod evaluation, a super-cell (made up of unit cells of the material under study) is poisoned until $k_\infty = 1$. Then the control rod is inserted and the necessary amount of poison is removed

until no change in reactivity occurs. The rod may be of such strength that a negative amount of poison is associated with the measurement, but the results remain interpretable.[22]

Comparisons of Techniques

As a desirable step in the AEC Control Rod Materials Program, various dynamic techniques for measuring control rod worth were compared. Ballowe, Morgan, and Russell[23] report a set of experiments in which four methods were chosen for intercomparison: rod drop, rod bump-period, rod oscillator, and pulsed neutron. The results disclosed that the rod-drop and rod-oscillator techniques are dependent on the detector location. The exact location of the detector was not given, although it appears to have been near the reactor core and was varied along the axis parallel to the motion of the rods. The dependence was explained on the basis of a one-group diffusion-theory model which showed the signal arising from a local change in the core to satisfy a differential equation differing from that of the steady-state flux in that the signal is a function of the prompt neutrons alone. Consequently, there is no reason to expect the ratio of signal to steady-state flux to be unity. The rod bump-period and pulsed-neutron techniques do not exhibit this dependence, being, in themselves, eigenvalue determinations. The observed dependence is contrary to the earlier work of Jankowski, Klein, and Miller,[15] who compared the rod-drop, rod-oscillator, and source-jerk techniques. The results reported were all within experimental error, and no dependence on detector location was inferred from the independence of the constant relating power-oscillation amplitudes to reactivities. However, all errors in the

rod-oscillation data were observed to be in the same direction and therefore suggestive of a systematic error.

The pulsed-neutron technique was compared with the rod-drop and rod bump-period methods by Kolar and Kloverstrom[16] of Livermore. They confirm the work of others who have examined the technique; namely, its usefulness in evaluating large negative reactivities, as well as its speed and convenience. Important byproducts of this technique, as illustrated by their work, are the quantities ρ/γ and l^*/γ. When the constants are used with the reactor kinetic equations, a calculated curve for neutron population versus time subsequent to the rod drop is obtained that aids in the evaluation of the data. For the rod-drop measurements these authors positioned the detector "about 8 ft from the reactor in locations where inverse multiplication measurements had been found to be independent of counter location." It is possible that detector dependence did not manifest itself in the data, since comparison of the data, the calculated curve, and the calculated points, which were assumed to have varying error magnitudes, seems to indicate a reactivity resolution (±3%) for the method.

The major disadvantages of these methods remain when they are applied to very large reactors. Further the pulsed-neutron technique, which in this application becomes similar to the source-jerk method, suffers from insufficiency of neutron source strength. In comparison the PCTR-k_∞ measurements appear to be applicable to large reactors. Nichols and Heineman[24] have reported on a set of experiments in which an EGCR control rod was evaluated in the PCTR with the use of a supercell containing only 16 fuel elements and a centrally located control rod. An attractive feature

of the technique was thus forcefully demonstrated, when it is considered that only ~1% of the core was represented. (J. T. Thomas)

IN-CORE FLUX MONITORING

The spatial distribution of the neutron flux in a reactor core is a valuable parameter for the evaluation of fuel element and reactor control system performance. Information regarding fuel enrichment and distribution for large core designs can often be determined in research reactors and critical assemblies. Flux plots are needed for power reactors in order to optimize the fuel loading and burnup cycle through flux flattening by control rod correction, which may also yield more uniform channel heating. Devices for ascertaining the flux distribution in a reactor core are known as in-core flux monitors. A number of different monitors currently in use or under development have been discussed in recent literature.

A standard flux-monitoring practice in research reactors is to insert activation foils or wires into the reactor core and then remove them for counting. The use of threshold detectors and associated counting techniques in the determination of thermal- and fast-neutron flux densities and energy spectra has been discussed tutorially by Price.[25] More recently, an improved technique for the measurement of fast neutrons by foil activation was reported by Riccabarra,[26] who uses phosphorus instead of sulphur as the activated material. The flux required for the same activity is reduced by a factor of 70 when phosphorus is used.

Mechanical arrangements[27] for the insertion of activation wires are in standard use at Savannah River, Hanford, and Shippingport. At Yankee,[22] thimbles are provided for remotely operated devices for traversing activation wires.[27] An example of an automatic, high-resolution

counter for activated ribbon has been described by Mulot.[28] A resolution as low as 0.5 mm is achieved by the use of interchangeable collimators. For most applications that would seem to be more than adequate. Researchers at Savannah River have experimented with neutron- and gamma-sensitive thermopiles, but they indicate that they have reservations about their results.[27]

A recent report by Schilling[29] describes a successful device consisting of a copper-constantan thermocouple welded into a platinum pill filled with a pulverized mixture of 55% aluminum and 45% uranium of 20% enrichment. The thermocouple junction, in series with a reference junction, is enclosed in a tiny tube at the end of a long probe handle through which the signal leads are drawn. The probe is calibrated against foils, and the thermal emf is read with a galvanometer. This device has shown only 1% drift in sensitivity after 300 Mwhr of operation in the FRM at a flux of approximately 1×10^{13} neutrons/cm^2·sec.

At Savannah River and at Dresden, work has been carried out on the use of gamma ion chambers.[27] The gamma chamber yields a signal which the authors state is essentially proportional to local power density. Flux tilts are determined by variations in the chamber signals at various points in the core. Savannah River reactors are employing, on a developmental basis, assemblies that contain three chambers and are placed in thimbles in the reactor core. The results have been quite satisfactory. At Dresden, the control system employs a matrix of 64 chambers at fixed positions in four planes throughout the core. It is intended that the reactor control rods be programmed from these readings and that the safety system be operated from these in-core chambers. In November, 1960, these

chambers, which also check out the operation of the control rods, detected the breakage of a control rod.[30] It appears that this system is effective for flux gradient control and rod checking. A commercial manufacturer is marketing a chamber of this type and developing improved high-temperature cables for use with it.

These recent advances in the state of the art of in-core flux monitoring indicate only that standard techniques have been improved, while relatively little in the way of new equipment has been developed. It has been reported[27] that AEC-sponsored work is in progress on microwave attenuation from ionization and heat-transfer null-balance detectors using neutron heating, and it is hoped that some breakthroughs will be reported in the near future. (J. L. Kaufman)

NONNUCLEAR IN-CORE INSTRUMENTATION

A meeting sponsored by the AEC Division of Reactor Development was held at Washington, D.C., on April 28–29, 1960, to discuss in-core instrumentation for power reactors. The objectives of the meeting were (1) to review the present status of the current development programs for in-core instrumentation; (2) to discuss the major problems in this area; (3) to review present and future requirements for in-core instrumentation, particularly in power reactors; (4) and to provide a free exchange of information and ideas that might lead to improvements in reactor core performance and reactor safety through improvement of in-core instrumentation.[31] The proceedings of the meeting contain numerous references to reports that provide much additional detailed information on the various subjects. The meetings consisted of informal question and answer sessions, and the report of the meetings presents a recording of those discussions. No formal papers were presented.

This review describes the state of the art for nonnuclear instrumentation as it was discussed in the meeting. Flow, liquid-level, differential-pressure, and temperature measuring devices are described, in that order.

Flow

Pitot tubes have been used in-core to measure gas flow in fuel channels of gas-cooled reactors and water flow in boiling-water reactors. Before being installed in the reactor core, the tubes are calibrated out-of-pile in an identical configuration to assure accuracy. Stauschiebe tubes, which are small cylinders placed perpendicular to the direction

of flow, with pressure taps facing upstream and downstream, are also used for measurements of water flow rates; these tubes must be calibrated.

Turbine flowmeters can be used to measure the velocity of water in the inlet and in the outlet of a fuel channel with an accuracy of 1%. By knowing the difference in the inlet and outlet water velocities, the steam void at the outlet of the fuel channel can be calculated. Prior to in-pile use, pairs of turbine flowmeters are calibrated with water and with water-air mixtures in an out-of-pile electrically heated loop. The steam voids calculated for void fractions ranging from 0.24 to 0.71 are within ±10% of the values determined by the gamma-attenuation technique.

Turbine flowmeters have worked well in a flux of 10^{12} neutrons/cm^2·sec. One application was described in which, after exposure to an integrated flux of 4×10^{18} neutrons/cm^2 at 975°F while measuring the flow rate of CO_2, there was no evidence of damage to the flowmeter. The flow rates of molten salts and liquid metals have been measured with experimental turbine flowmeters for periods of more than 1000 hr at 1500°F.

Venturi tubes are commonly used to measure flow in fuel channels. As many as 42 in-core Venturi meters have been manifolded out-of-pile to one differential-pressure transmitter.

Liquid Level

Resistance-type level-measuring elements have been used for measurements of fluids with high electrical conductivity. Such elements have operated satisfactorily for periods of up to 20,000 hr at 1300 to 1500°F

while immersed in sodium and NaK. The accuracy of measurement over a 7-in. span of NaK and of sodium has been reported to be 1%.[32]

Acoustic level-measuring devices are available for use at temperatures up to 950°F. The effect of irradiation on these devices has not been investigated.

Differential Pressure

Differential-pressure transmitters located outside the reactor vessel of a boiling-water reactor and connected to a number of pressure taps inside the vessel have been used to measure the carryover of steam, vapor holdup, downcomer velocities, and average steam voids in risers. The individual pressure tap tubes are contained in a duct. A small stream of cooling water flows over the pressure tap tubes within the duct to keep the liquid in the tubes from flashing. A large flow of cooling water cannot be used, because it would cause undesirable axial temperature gradients in the duct.

Temperature

Chromel-Alumel thermocouples in stainless steel sheaths are the thermocouples most often used as sensing elements to measure in-core temperatures in all types of reactor. The thermocouple insulation is usually magnesium oxide, although aluminum oxide insulation is often used in pressurized-water and boiling-water reactors because aluminum oxide will not destructively swell if water gets into the sheath. The sheathed thermocouple assemblies used most often are 1/16 in. o.d., although assemblies as small as 0.040 in. o.d. have been used.

The guide tube, which is an extension of the reactor pressure vessel and through which thermocouples can be inserted into the core,

is a significant development in core temperature measuring. Such tubes permit the replacement of thermocouples at any time during the life of the core. A 1/16-in.-o.d. sheathed thermocouple can be inserted in a guide tube that is 25 ft long and has many turns.

The reliability of 50 sheathed Chromel-Alumel thermocouples during a period of a year of operation in a pressurized-water reactor core has been studied. The variations of the 50 individual thermocouples gave temperature values that were within 1°F of the average (95% C.L.) and within 2.5°F of an external reference temperature. The irradiation dose to these thermocouples was more than 10^{21} neutrons/cm^2. Experience with sheathed Chromel-Alumel thermocouples in liquid metal at 1500°F has also shown that they are reliable and that they hold their calibration when exposed to an irradiation dose of 10^{21} neutrons/cm^2. If the leads from a small-diameter sheathed thermocouple, about 0.050 in. o.d., pass through a zone having a temperature of 2500°F or more and the junction is in a cooler zone, the emf from the thermocouple will be proportional to the hot-zone temperature. The high temperature causes the insulation resistance to become low enough to form another junction which is in parallel with the true junction.

In order to obtain an accurate measurement of the temperature of a fuel element surface, the sensing element must be calibrated out-of-pile using the same materials and configuration as used in-core. Several techniques have been used to measure surface temperature in flowing gas. One technique is to flatten a 0.030-in.-o.d. sheathed thermocouple to 0.020 in., fasten it to the surface of the fuel element, and put a ramp upstream to give smooth flow and avoid error-causing turbulence. Another technique is to place the junction end of the sheath downstream, with

the thermocouple wires extended from the sheath and spot welded to the surface of the fuel element. With this technique, a surface temperature 1300 ± 50°F has been measured. On the other hand, in an organic coolant, the surface temperature of a fuel element has been measured to within 25°F in the 400–700°F range, based on out-of-pile calibration using 0.010-in.-o.d. thermocouple wires fastened to the surface.

The central temperatures of fuel plates for pressurized-water and boiling-water reactors have been measured by the so-called "blocked channel" technique. A special zirconium plate that has the thickness of two standard plates plus one water channel is fabricated into the fuel element. The water channel space in this special plate is filled with a nonfissionable material in which the thermocouples are embedded. As many as 100 thermocouples have been spaced along a 4-ft plate. If stainless steel were used in place of zirconium, the resulting flux depression would give a temperature measurement error as great as 12%.

There is a need for techniques and materials that will make it possible to measure fuel temperatures up to 4500°F in uranium oxide, uranium carbide, and other possible fuel materials. At the present time central temperatures up to 3800°F in UO_2 pellets have been measured for a week or so and up to 6 weeks in the temperature range of 2500 to 3200°F. In these measurements the thermocouple was tungsten vs. rhenium insulated with beryllium oxide and contained in a swaged tantalum sheath. When the junction where the sheath and thermocouple wires were fused together was grounded, the couple often failed because of differential thermal expansion, whereas thermocouples with

ungrounded junction failed less often. The thermocouple wires entered the capsule through a hermetic electric seal so that the fission gases would be contained. Generally, the tantalum sheath showed deterioration after 6 weeks at temperature and exposure to fission gases. Over the 6-week test period, the thermocouples indicated that the central UO_2 temperatures were decreasing. This may have been caused by changes in the pellets or in the thermocouples or both.

A pneumatic system for the measurement of gas temperatures was discussed that consists of two sonic nozzles in series through which a stream of gas flows. The first nozzle is located where it is desired to measure the temperature; the second is at some convenient location and is connected to the first nozzle by a tube. The pressures upstream of each nozzle and the temperature upstream of the second nozzle are measured. The desired temperature is calculated from the relation

$$T_1 = K \left(\frac{P_1}{P_2}\right)^2 T_2 \quad ,$$

where K is the general constant of the system. This temperature measuring system was discussed in detail in an earlier issue of Nuclear Safety.[33]

There are numerous special problems and techniques associated with the determination of in-pile temperatures. Sometimes it is necessary to use odd combinations of thermocouple wires to match the thermal expansion of wires, sheath, and fuel plates. Further, platinum vs. platinum-rhodium thermocouples have shown errors because of transmutation and, in high temperature applications, because of the migration of rhodium. In another situation, resistance thermometers of platinum will

be used in some pressurized-water reactors to measure fuel channel outlet temperatures and the difference in temperature between the inlet and outlet of a channel.

A number of high-speed scanning systems will be used to monitor in-core temperatures in the near future. Plans are to use 18-channel recording oscillographs, a 100-channel oscilloscope with a mercury-jet switch which can scan 100 thermocouple stations, and an analog-to-digital conversion unit with magnetic-tape recording. Experience with current models of jet switches has shown that they must be cleaned after being in service three months because of contamination by the mercury. (H. J. Metz)

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IV. PLANT SAFETY FEATURES

DEPOSITION OF RADIOACTIVE MATERIALS FROM GASES

The designer of a nuclear reactor must be able to predict reliably the location and form of the entire inventory of radioactive fission products associated with the reactor system if he is to fulfill his obligation to control the hazards produced by these materials. One of the most important processes affecting the location of fission products is transport by gases.

Radioactive fission products may be introduced into a gas system either by accident or by design. For example, if a major reactor accident occurs and the fission products are dispersed in the reactor containment vessel, many of the fission products will remain suspended as gases or aerosols in the air inside the vessel. On the other hand, fission gases are circulated in the normal operation of a gas-cooled reactor that uses unclad fuel elements. Some of the fission products are released continuously into the coolant gas stream and are carried throughout the coolant system. In both these circumstances the fate of the fission products is of paramount importance. In each case there are locations where deposition is beneficial and other locations where deposition is harmful.

Accumulations of fission products by deposition must be avoided in components which may require maintenance or replacement or in parts of the reactor system to which access is necessary to control the consequences of a reactor accident. On the other hand, deposition of fission products from gases in some places can substantially reduce the amounts of fission products delivered to other more sensitive locations and can also reduce the total amount of fission-product

activity which may be carried to the outside atmosphere in the event of an accident. Prediction and control of the deposition of radioactive fission products from gases requires a knowledge of the factors that influence deposition.

Theoretical Considerations

Although there are many factors that may affect the deposition of radioactive materials, they can be divided into three general categories: (1) the concentration of the material and conditions in its history which affect its physical and chemical form, (2) conditions near the site of deposition that influence the transport of the material through the gas to the surface, and (3) conditions which influence the retention of the material when it arrives at the surface.

Radioactive material in a gas may be in gaseous form or in the form of a solid or liquid aerosol. If in aerosol form, the behavior of the material in the gas is affected grossly by the particle size of the aerosol. Particles of sizes ranging from slightly larger than molecular dimensions up to several tens of microns in diameter are common in gases. Which of these forms will be taken by the fission-product materials depends upon the history of the material. Fission products may be dispersed in the gases by evaporation at high temperatures or by mechanical action. Evaporation at high temperatures followed by rapid condensation at low concentrations is conducive to the formation of small-particle-size aerosols that are relatively stable under most conditions. These conditions are common in reactors, and it is to be expected that aerosols will be formed.

Particulate matter composed of ordinary nonradioactive materials can play an important role in a gas system containing fission products. Gaseous materials, including radioactive fission products, will tend to approach adsorption equilibrium with the surface of the particulate matter suspended in the gas. In some cases a fission-product material that would be expected to be gaseous under one set of conditions can be adsorbed completely on particles if the concentration of the gaseous species is low and the duration of exposure of the particles to the gas is long. In such a case the gaseous fission-product material will behave as if it were in particulate form.

There are several processes by which material in a gas is transported to the surfaces where deposition can occur. Turbulent mixing continuously presents gas and its burden to the surface. The conditions favoring this process are high velocity and large channel diameter. Gravitational settling is enhanced by large particle mass, long residence time, and small vertical channel dimensions. Diffusion processes are enhanced by small particle size, long residence time, and small channel dimensions. Molecular dispersions diffuse faster than particulate dispersions. Thermal diffusion is enhanced where high thermal gradients exist in a gas and by large particle diameter and small channel dimensions. Centrifugal separation is enhanced by large particle mass and high gas velocity with a change in direction of flow. Under some conditions electrostatic fields may influence deposition. An extensive theoretical treatment of the processes affecting the form of materials and their transport through gases is available in the literature.[1]

The retention of radioactive products after they have reached the surface may be by several kinds of forces. Massive particles can be held in place by the force of gravity. Electrical attraction can sometimes hold the particles in place. Van der Waals forces are effective for smaller particles, and in the extreme case of a molecular dispersion, these forces manifest themselves as physical adsorption. Certain chemical forms of fission products can react chemically with surfaces of many kinds to form a chemical compound that is retained with great efficiency. The various processes by which fission products are deposited on surfaces have been exploited in the design of filters and adsorbers. This subject was reviewed previously.[2] The subject of the present review is deposition in other parts of a reactor system.

Experimental Investigations

The behavior of radioactive materials in gases has interested experimenters since shortly after the discovery of radioactivity. In 1900 Rutherford[3] discovered that there was "emanation" from certain naturally radioactive materials. His investigation of this radioactive gas, now known as radon, and of the radioactive daughters deposited from it, led to the recognition of the decay relationships between various radioisotopes. It was observed that the daughters of the "emanations" were in the form of extremely small particles that remained suspended in the diluent gas.

Most of the recent experimental work on the deposition of radioactive materials from gases has been concerned with radioiodine. This is because of the volatility of iodine and its very low biological

tolerance. A number of experimenters have observed that iodine vapor exhibits varying properties, depending upon the conditions under which it is studied.[4-7] It is very readily adsorbed on a wide variety of surfaces in varying degrees. Chamberlain[6] showed by experiments with a small wind tunnel that deposition of a "carrier-free" radioactive iodine vapor was controlled by diffusion across the laminar layer of gas flowing over a surface and was analogous to heat transfer by forced convection. Flat surfaces of copper and of filter paper acted as "perfect sinks" for iodine vapor in trace concentrations. Any surface imperfection that interfered with the laminar flow caused increased deposition of iodine vapor on the surfaces immediately downstream of the imperfection.

Once iodine is deposited, the desorption from the surface becomes important. Chamberlain and his co-workers[4] report that the rate of desorption of trace quantities of iodine at room temperature is generally slow, but it may be more rapid in the presence of stable iodine, which can exchange with the adsorbed radioactive layer. The desorption is not appreciable from copper or charcoal surfaces, even in atmospheres containing I^{127} at a concentration of 3 g/M^3. There was appreciable desorption from aluminum, polythene, and painted surfaces in clean air, and the desorption was greatly increased by the presence of I^{127}.

Probably the most important factor concerning deposition of iodine vapor onto surfaces is the presence or absence of other substances in the gas stream. Experiments by Chamberlain and Whiffen[7] indicate that, in the presence of condensation nuclei, the behavior of radioiodine depends upon the amount of iodine carrier present and the concentration of

condensation nuclei. Their experimental system consisted of a large aluminum box (6 M^3) into which iodine vapor and condensation nuclei could be introduced. The condensation nuclei were produced by heating a Nichrome wire white hot by combustion of gas in a Bunsen burner or by passing an arc between lead electrodes. Samples of the gas mixture were drawn from the aluminum box through a sampling system consisting of a filter-paper-lined tube, a high-efficiency filter, and an activated-charcoal trap. By observing the distribution of I^{131} activity between the components of the sampling system, some insight into the behavior of iodine vapor was obtained. The experimenters concluded that (1) I^{131} can behave as iodine vapor, even in the presence of large concentrations of nuclei (20,000/cm^3), provided sufficient carrier iodine is present to provide an air concentration of 10 $\mu g/M^3$, (2) in the absence of carrier iodine, the behavior of trace concentrations of I^{131} ($\sim$0.1 $\mu g/M^3$) deviates from that of iodine vapor and the more nuclei present the greater the deviation, (3) in the presence of lead fumes (10–20 mg/M^3) with particle sizes 0.1 to 1.0 micron, most of the iodine radioactivity is associated with the lead particles, and (4) the effect of condensation nuclei on the diffusivity and adsorption of carrier-free radioiodine may have important implications when methods of removing the activity from gas streams are considered.

Chamberlain and his co-workers[8,9] are also responsible for information concerning the effect of condensation nuclei on the behavior of the decay products of thoron and radon. These decay products, isotopes of polonium, bismuth, and lead, adsorb rapidly on condensation nuclei.

The effect of attachment is to reduce markedly the diffusivity of the radioactive nuclide compared with its diffusivity in the atomic or molecular state. It follows that the rate at which the radioactivity will diffuse across a boundary layer to a surface is greatly reduced once it is fixed to a condensation nucleus. Retention of radon and thoron decay products in the human respiratory system is greater when they are generated in air free of nuclei and inhaled in the atomic state than it is when they are allowed to become attached to nuclei prior to inhalation.

Brewer[10] has discussed the possible or probable fate of various fission products that may be present in the coolant gas of a gas-cooled reactor. In the event of a slight fission-product leak from a fuel element the presence of Kr, Xe, Cs, Rb, Ba, Sr, Sm, Eu, Br, I, Sb, Cd, Sn, In, and Ag should be considered. Upon cooling of the gas stream as it leaves the reactor, the condensation of a mixture of metals, oxides, and carbides would be expected, along with CsI and CsBr. The surfaces of carbon dust and the cool surfaces of the heat exchangers will probably remove the remaining nuclides, with the exception of Kr, Xe, Cs, Rb, I, Br, and Sb. These nuclides may act as permanent gases to a considerable extent.

Information on the deposition of mixed fission-product radioactivity is provided by Conn, Collins, and Trice.[11] Fission products were produced for study by uranium fission and allowed to escape into the cooling air stream by direct recoil. Their deposition was then examined in the ducting of the experimental system. The particular isotopes and corresponding decay chains studied were I^{131}, Ba^{140}, Ru^{103}, Sr^{89}, and Te^{132}.

The test system allowed the air to pass over the fission-product generating source only once. It was then routed through stainless steel ducts, through a large charcoal trap, and into the reactor off-gas system. The percentage of iodine, barium, and strontium plate-out differed in the three tests, as well as the amounts of fission products released. According to the data presented, the fraction of activity deposited in the ducting depended only on the amount of the activity released from the fuel. No dependence of deposition on air flow was noted. The percentage of barium and strontium deposition increased as the amount released increased, while the percentage of iodine deposition decreased. The air supply was not characterized, and no information concerning the presence or absence of condensation nuclei is available.

Conclusion

Deposition of radioactivity from the gas phase as it occurs in gas systems associated with reactors is complex, and probably each such system will behave differently depending upon the concentration and identity of the various isotopes released from the nuclear fuel, the presence or absence of other matter, such as dust, condensation nuclei, and water vapor, the temperature of the system, and the materials of construction used in the gas system. Deposition of radioactivity from the gas phase is becoming increasingly important in nuclear technology, but, as yet, insufficient experimental evidence exists to permit reliable predictions of the behavior under conditions existing in nuclear reactors. (R. E. Adams and W. E. Browning, Jr.)

EFFECTS OF BLASTS ON CONTAINMENT VESSELS

A containment vessel for a nuclear reactor must be a safeguard against the effects of a maximum credible accident. The maximum credible accident has been defined for each of the 13 nuclear reactors built prior to 1959 and is discussed in a report prepared by Van Kessel.[12] This report provides an excellent accumulation of the literature (through the latter part of 1958) on reactor safety in general and on the containment vessels for nuclear plants which were in existence or in the planning stage at that time. It lists over 500 references divided into 24 categories; a few of the references are listed in more than one category.

A maximum credible accident is frequently some kind of explosive reaction, and hence the designs of some containment shells must be based on the response of the structure to dynamic, as well as static, pressure loading. In view of the very limited knowledge and understanding of the cause (the explosion) and the effect (the response), initial studies and investigations had to be made for each specific plant. A series of general studies of this behavior, not associated with any specific reactor design, has been made only during recent years. Some of these studies were reviewed previously,[13-15] and some recent results are discussed here.

Experimental Blast Effects

In some reactors the maximum credible accident can be caused by a sudden loss of coolant which, in turn, may cause a rapid pressure buildup in the container. The pressure buildup in a vapor container

during loss of coolant was studied experimentally by the Mine Safety Appliances Research Corporation.[16] The tests involved venting, through an orifice, water which was at a temperature of 500°F and under a pressure of 2000 psig from a 512-gal pressurizer into a 10,000-gal tank. Seven runs were made using four orifice sizes and two water-to-air volume-expansion ratios. The data from this limited number of tests are presented in the report,[16] and, as the authors of the report recognize, are insufficient to draw any basic conclusions that are sufficiently definite for future design purposes.

Experimental measurements of the air-blast effects caused by the bursting of simulated core vessels with differing rates of energy release have been reported by the Ballistic Research Laboratories (BRL).[17,18] In these tests stainless steel cylindrical pressure vessels were used to simulate the reactor cores. The test vessels were burst by employing one of three methods of internal energy release: (1) suddenly releasing a gas under high static pressure, (2) igniting a confined propellant, and (3) detonating an explosive. The vessels were supported 5 ft above ground, and measurements of the resulting air blast loading were recorded for various distances from the vessel. In the first series of tests the vessels were filled with air. The experimental set of data for this series of tests is presented in the first report.[17] From this set of data it was concluded that the blast loading was relatively independent of the rate of energy release within the vessel. It was established that the test data from the energy source surrounded by the pressure vessel could be reasonably well predicted by assuming that the source was equivalent to a certain weight of bare explosive charge, that is, an explosive which

was not discharged in a vessel. The second report[18] presents the results of additional tests that were similar except that the pressure vessels were filled with water instead of air. The pressure-time histories from these tests varied from those of the bare-explosive tests to the extent that it was not possible to establish an "equivalent weight" of bare explosive, as was done for the previous set of results.

Somewhat similar comparison tests by the Stanford Research Institute (SRI) are discussed in two reports[19,20] which were reviewed previously.[15] Simple, scaled reactor models were used in which scaled energy sources (simulating possible nuclear excursions) were released. In this study, energy sources employing high explosives alone and high explosive—pyrotechnic mixtures were developed to simulate reactor excursions with periods in the 1- to 5-msec range. In most cases the tests[19] were run in pairs with the same or similar energy sources but with varying amounts (75% and 100%) of water in the pressure vessel.

The findings and conclusions from the SRI tests appear to contradict the findings and conclusions from the similar tests made at BRL.[18] The similarity in the two sets of tests is in the comparison of the results from the bursting of containers which are first filled with air and then with water. The findings and conclusions from the series of tests at BRL on water-filled vessels are given in the following quotations from the report:[18]

> "Even though the pressure-time history does vary from those generated by bare explosives, the pressures and impulses measured were never larger than those of the air-filled vessels. The vessel fragments as seen from the fragmentation firings had a very low velocity and little penetrating power. Therefore, it may be concluded that the data for air-filled core vessels, which are applicable to the equivalent weight concept, yield an upper bound of pressure and impulse measurements for its water-filled counterpart and can be used as a method of simulation of nuclear reactor runaway to establish an upper bound for blast loading on containment structures from failure of water-filled core vessels."

On the other hand, the findings and conclusions from the SRI tests were the following:[19]

> "In the models tested with vessels filled full of water, the velocity of the plug rise was considerably greater in every case than in similar tests where the vessels were only 3/4 full. However, in models 3/4 full of water, damage to the lower portion of the cylinder was very extensive and the upper portions broke into relatively large chunks. Concrete fragmentation in the models 100 percent full of water was extensive but more generalized over the whole concrete face. It is evident that the air void in a partly full vessel tends to prevent energy transfer to the concrete surrounding it."

It is considered by Davenport and Huber[21] and by Minor[22] that the tests were not sufficiently similar to justify a direct comparison of the experimental findings. Davenport and Huber[21] claim that the adding of the water in the BRL vessels (the vessels contained only air in the initial tests) lowered the gas temperature sufficiently to lower the air shock wave after it expanded to about 200 times its initial volume. This effect of lowering the gas temperature was not as predominant in the SRI tests,[19] because the vessels were always filled at least 75% with water. They further point out that the time of the explosive event in the two sets of tests was considerably different; in the BRL

tests the entire event was completed in approximately 10 msec, whereas in the SRI tests the event was completed in approximately 200 msec. Minor points out[22] that the effects of the BRL tests[18] were measured at a considerable distance from the bursting vessel. Furthermore, it must be recognized that the containers in the BRL tests were of stainless steel (a ductile material), whereas the containers in the SRI tests were of concrete (a brittle material). Also, the technique which was used for measuring the blast effect was different for each set of comparison tests; piezoelectric air-blast gages were used by BRL, whereas SRI observed the extensiveness of the shattering of the concrete.

Accepting the validity of the experimental data presented in each of these reports, it appears that the design criteria for the containment vessel are a function of the material and the type of construction. Hence, caution must be exercised in extrapolating either of the conclusions to any great degree.

Theoretical Treatments of Blast Phenomena

The development of theoretical methods for the study of pressure in explosive-like waves has a long history, but prior to World War II it had been slow and limited; the need was not great, and the equations which describe such phenomena are too complicated to be dealt with directly. The advent of high-speed computers has opened many possibilities for obtaining approximate solutions to specific problems. The general solution of the complex partial-differential equations which are involved for most cases is still unknown; however, considerable material has appeared in the literature during recent years that provides methods

for obtaining the approximate solution of specific problems. The basic equations and a method for solving problems of pressure waves in ducts have been published by Rudinger.[23] In this text it is assumed that all flow variables are uniformly distributed over the cross section of the duct, and hence time and one space coordinate are the only independent variables. The solution of a problem of this type is obtained by graphical and numerical approximations. Complete wave diagrams for the solution of a number of problems are presented in the text.[23] Because of the complexity of the basic equations, most of the work to date deals only with systems which require only one space coordinate. The numerical solutions of spherical blast waves have been published by Brode.[24]

Another major difficulty inherent in the solution of the basic blast wave equations is the discontinuity at the shock front. In order to overcome this difficulty, Brode[24] assumed an equation for the viscosity; he gives credit to previous authors for the method employed.

The response of structural materials to blast waves is another complicated aspect of the entire problem. The basic theories and problems involved in this study are presented by Rice, McQueen, and Walsh.[25] A detailed discussion of the problems in experimentation with strong shock waves and the various forms of the equation of state which have been assumed are also given in the article.[25]

Reports on theoretical and experimental studies of blast waves and the compressibility of materials have been presented by the Armour Research Foundation.[26-28] A technique for studying the behavior of porous material (such as wood) subjected to compressive impact loads is presented

in the first report of the series.[26] The second report[27] is a theoretical study of the growth of a point source explosion (spherical blast waves) in an infinite solid or liquid medium. An expression relating pressure and specific volume (equation of state) of the material is assumed. This equation contains three constants which must be determined from experimental tests of the material (compressibility data). The basic equations were solved numerically and a parametric study of the influence of these constants was made. The results are presented in the report.[27] The results are for highly idealized physical situations, but they do provide information on the relative effectiveness of the materials as shock absorbers. The third report[28] describes the experimental shock-tube testing of three materials: Celotex, redwood, and sugar pine. Points on the pressure—specific volume curve are obtained from the experimental data for each material and are presented in the report.[28]

A method of analytically determining pressures, radii, and velocities in all regions of an exploding reactor is presented by Levedahl and Howerton.[29] Their solution is obtained by numerical integration of differential equations relating rate of energy input, pressure, volume, acceleration, and velocity of several concentric shells representing the core, reflector, pressure vessel, shield water, and shield tank. Complex reactors are transformed into equivalent spheres for analysis and then the results are transformed back to the true configuration. In this report[29] it is claimed that the velocities of missiles resulting from the metal-water reaction are only fractionally higher than the velocities from an explosion resulting from fatigue failure at ordinary pressures.

Conclusions

The present studies of blast waves and the response of materials thereto are, for the most part, highly specialized and idealized. The additions to the literature during recent years have contributed much to the knowledge and understanding of these phenomena; however, much more work (both experimental and theoretical) must be done before the present ignorance factor in the design of containment vessels for nuclear reactors is reduced to any great degree. (F. J. Stanek)

BERYLLIUM HAZARDS IN NUCLEAR OPERATIONS

Beryllium and its compounds are the most toxic of the nonradioactive components of nuclear reactor systems. These materials require special handling during operations that precede reactor startup if harmful effects to personnel are to be prevented. After beryllium and other chemical poisons become radioactive by exposure to neutrons in a reactor, the chemical poison aspects of materials handling hazards become of secondary importance and can be adequately dealt with by radioactive materials handling procedures, as pointed out previously by Grimes.[30] Recent developments in beryllium containment and monitoring methods are discussed here. The toxic effects of beryllium have been thoroughly discussed by previous reviewers[30,31] and are too well known to require further emphasis. A fairly recent bibliography[32] of articles on beryllium toxicology is also available.

Beryllium Monitoring

One of the limits of air pollution recommended by the AEC is worthy of comment. It is stated that "no personnel should be exposed to a concentration greater than 25 μgms./m^3 for any period of time, however short." It seems safe to say that no installation handling beryllium is in a position to guarantee that it is meeting this requirement, which should be modified to eliminate ambiguity. The American Industrial Hygiene Association[33] has adopted a more meaningful upper limit for transient conditions of 25 μg/m^3 for less than 30 min. Monitoring for airborne beryllium is usually performed by drawing a measured volume of air through a filter paper, determining the total amount of beryllium collected, and calculating the concentration per cubic meter of air by assuming a uniform rate of

collection over the sampling period. If, for example, 50 μg of beryllium were collected from 100 m^3 of air, the calculated average concentration of 0.5 $\mu g/m^3$ would be well below the acceptable upper limit of 2 $\mu g/m^3$ for the average concentration during an 8-hr working day. If, however, all the beryllium had actually been collected from 1 m^3, the concentration would have been 50 $\mu g/m^3$ and thus would have exceeded the AEC limit of 25 $\mu g/m^3$.

Several investigators have reported efforts to develop instruments for continuously monitoring air for the detection of beryllium that would give a quick indication of the presence of an unsafe concentration of beryllium in air and would, by shortening the sampling time, come nearer meeting the "however short" time requirement than the usual routine monitoring methods. The first instrument of this type, reported[34] in 1952, had several drawbacks, including the requirement of a great deal of skilled maintenance, sensitivity to particle size and chemical form of the beryllium, and lack of means for checking the calibration of the instrument. Other investigators[35] later developed a similar continuous spectrographic air monitor which seems to have overcome most, if not all, of the troubles inherent in the earlier instrument. An improved version of this instrument, reported recently,[36] is designed to be sufficiently portable so that it is easy to bring the sample intake close to potential sources of airborne beryllium. Air is drawn through a triggered intermittent arc and the intensity of the 3130 Å beryllium doublet is monitored, together with the adjacent background, and the ratio of those intensities is recorded on a ratio recorder. Provision is made for frequent calibration by means of an aerosol of known

concentration produced by means of an auxiliary spark between beryllium-copper electrodes. The accuracy and sensitivity of the instrument are said not to be materially affected by the chemical form of the beryllium or the particle size. The instrument is fully automatic and rapid in operation. Its limit of detection is 1 $\mu g/m^3$ for the normal 20-sec sampling period, and one result is produced each minute. The electrodes last about 8 hr, so they would only need to be changed once each shift for continuous operation. This type of instrument is now being offered for sale by a British instrument manufacturer.[37] Other commercial continuous beryllium monitors are also available from another British producer[38] and an American manufacturer.[39] The reliability of each of these commercial monitors in continuous operation apparently remains to be established.

The spectrographic determination of the beryllium collected on Millipore filters in routine air monitoring has been discussed by Brasch.[40] His method, which employs an aluminum internal standard and a photographic measuring system, is said to be sensitive to 0.05 μg of beryllium per cubic meter of air with a coefficient of variation of less than 11% in the 0.5- to 2.0-μg range. The spectrographic method appears to be widely employed for this type of determination, and it is susceptible to automation[41] when the number of samples justifies the expense. A fluoresence method reported[42] to be in use at Los Alamos Scientific Laboratory is said to be usable in the 0.05- to 300-μg range. Limits for surface contamination have not been established. The fraction of such contamination which may become airborne is more important than that which may become ingested, and this fraction is a function of particle size and density, surface

condition, and air velocity past the surface. A method for measuring this fraction under controlled conditions was described in a recent report.[43]

Control of Beryllium Hazards

Containment techniques employed at various installations where beryllium work is in progress have been discussed by several authors. Equipment and methods used with beryllium in some locations under the most hazardous conditions involving manipulations with dry, finely divided beryllium or beryllium compounds are almost identical to those used to assure the safety of personnel working with plutonium. This is perhaps not surprising because a number of installations have been involved in handling both materials, with the development of plutonium-handling methods preceding beryllium-handling developments, in general.

Mitchell and Hyatt[42] have described types of enclosures employed at Los Alamos for various machining operations with beryllium, together with their monitoring techniques, air-cleaning system, and housekeeping measures. A review of five years of operating experience is given from the health physics standpoint.

Lindeken and Meadors have discussed[44] the philosophy governing the control of beryllium hazards at the Livermore site. They state that dry machining and grinding operations, beryllium welding, and processing of BeO in the powder form should be performed in total enclosures. They also say that considerably more care is required for partial enclosures where the area of the opening is adjustable than for total enclosures. Their observations indicate that operators tend to leave too large an area open during hazardous operations, with a resulting low face velocity that endangers both the operator and his co-workers. Respirators are worn only

in special cases of work not conforming to usual enclosure standards or in cases of accidental failure of an enclosure. They recommend use of smooth floor coverings, such as linoleum, to facilitate cleaning.

Breslin and Harris discuss[45] in some detail items which they term engineering tools required to control atmospheric contamination. These are (1) the selection of process components that will release the smallest possible quantities of contamination, (2) application of local exhaust ventilation at all points of potential contamination release, (3) careful operating procedures, (4) thorough housekeeping, and (5) provision of suitable personal protective measures. A recent book[46] on beryllium contains a chapter on "Health Hazards and Their Control." Mention is made of exceptions to the usual control limits as follows:

> "If the average in-plant concentration over three months is >2 but <5 $\mu g/m^3$, or if a single short-term sample is >25 but <100 $\mu g/m^3$, work can continue if operators wear approved respiratory equipment but corrective measures must begin at once. If the three-monthly in-plant concentration exceeds 5 $\mu g/m^3$ or if the single sample exceeds 100 $\mu g/m^3$ work must cease until corrective action is completed."

A brief but fairly thorough coverage of principles involved in assuring the safety of personnel handling beryllium is given.

Two recent reports describing laboratory facilities for experimentation with beryllium are available. Thys[47] describes a small enclosure containing a vacuum glove box for beryllium welding and a glove hood for chemical operations with beryllium compounds. This enclosure is maintained under a negative pressure, and both inlet and outlet air filters are provided, as well as filters for the exhaust air from the glove box and hoods.

A clothes-change area equipped with a shower is provided within the enclosure. This facility provides an excellent example of a method for providing secondary containment of hazardous materials and appears to meet the most stringent requirements for a plutonium-handling facility, except for the fact that the gloves are apparently removed from the vacuum glove box when it is evacuated. Most vacuum glove boxes are designed to permit evacuation with the gloves in place, which is highly desirable when the box contains hazardous material.

Aitken[48] describes measures employed to convert an existing laboratory room into a beryllium laboratory. The exhaust capacity of the existing hood was increased by installing a larger fan motor, and both roughing and absolute air filters are provided for the exhaust air. A glove box is attached to the hood through an entrance port so that operations involving the transfer of powdered BeO, weighing, and sintering can be performed in the glove box while other operations of less hazardous nature are carried out in the hood, which has a face velocity of 150 fpm with the hood opening at the working level. Flexible exhaust hoses are provided in the vicinity of sintering furnaces to pick up fumes or dust, and a Lucite canopy equipped with gloves is employed to keep splatter or dust from a metallographic polishing wheel from becoming airborne.

Conclusions

There appears to be a reasonable probability that one or more of the commercially available instruments for continuous monitoring of airborne beryllium will solve the problem of determining the extent of air pollution over short time periods. It seems unlikely that these expensive instruments

will replace the present air-monitoring methods, which are undoubtedly adequate for most situations. Extensive experience in a number of installations indicates that reasonable air pollution limits can be met by application of well-established control methods. While total enclosure of certain types of hazardous operations with beryllium is certainly desirable, and perhaps essential, partial enclosures used with adequate precautions have served satisfactorily for less dangerous work. (C. J. Barton)

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V. CONSEQUENCES OF ACTIVITY RELEASE

CHEMICAL AND RADIATION INDUCED GENETIC EXPOSURES

By Peter Alexander* (with Editor's introduction)

At the request of the Joint Committee on Atomic Energy,[1] a series of Advanced Technological Seminars is being conducted by the Oak Ridge National Laboratory. The purpose of this program is to study the future role of the National Laboratories.

One of the most recent seminars was presented by Dr. Peter Alexander during the course of his visit to the Laboratory. Dr. Alexander is a member of the staff of the Chester Beatty Research Institute of England and is interested in the problem of mutation-producing chemicals. His lecture, entitled "Chemical Contamination of the Biosphere," dealt mainly with such chemicals.

The subject of mutagenesis is not new to persons engaged in the atomic industry. Much attention has been given and continues to be given to the problem in an attempt to determine the mechanism of the mutagenic response as well as the magnitude of radiation dose, dose-rate, and radiation energy required to induce such a response.

During a series of hearings before a special congressional subcommittee on radiation,[2] Dr. Hermann J. Muller, Professor of Zoology, University of Indiana, was called to speak on the subject of radiation-produced mutations. Dr. Muller, winner of the Nobel Prize in physiology

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*Dr. Peter Alexander was graduated in Chemistry at Imperial College of Science, London, in 1941. After work on a variety of wartime problems, he joined a textile firm as Research Manager. Studies on crosslinking agents in the textile field brought him into contact with cancer research because these same agents had radiomimetic properties and were used in the treatment of cancer. In 1950, he joined the Institute of Cancer Research and is now in charge of the Radiobiology Department at the Chester Beatty Research Institute.

and medicine in 1946 as a result of his work on the production of mutations by radiation, stated that evidence has in recent years been accumulating for the broad conclusion that the great majority, if not all, of the damaging effects on life and health evoked by ionizing radiation are results of permanent changes produced in the genetic material: the chromosomes or their contained genes.

Nitrogen analogs of dichlorodiethylsulfide, known as nitrogen mustards, were made during World War II for possible use as chemical warfare agents. Because many of the physiologic actions of the nitrogen mustards resemble those of ionizing radiations, they are called radiomimetic poisons. Research has shown that mutagenesis is one of the actions shared by radiation and this group of chemicals.

The question immediately raised by such findings is: Are there other chemicals, more common and frequently encountered in our environment, which are capable of illiciting radiomimetic actions? Dr. Peter Alexander feels that the answer is in the affirmative although it is not known whether the magnitude of the radiomimetic action from chemicals in the environment represents a real health hazard. His case is presented below, as abstracted from a paper originally published in The New Scientist.[3] (C. E. Breckinridge)

Mutation-Producing Chemicals

For more than thirty years the recommendations for safeguards against radiation hazards have taken account of the fact that small doses or continuous exposure to low intensity radiation can be harmful without giving rise to any injuries that can be detected at the time. The injuries, in

fact, may not become detectable during the lifetime of a person receiving the dose but may be transmitted genetically to succeeding generations. In Britain, the Medical Research Council; in the United States, the National Academy of Science; and internationally, the United Nations have set up committees of independent scientists to assess the possible genetic dangers to the general population from exposure to low levels of radiation, whether from fallout, medical radiography, or any other cause.

Radiation, however, is not the only source of genetic damage. It is now known that genetic effects of exactly the same type can be produced by a large number of chemical substances. Attention has been drawn to this fact by the United States National Academy of Science in a recently published report on the biological effects of radiation. The Academy points out that further information is needed about the genetic effects of "abnormally high temperatures, ultraviolet light, various chemicals, etc."

During the last fifteen years several hundred chemical agents have been shown to be mutagenic, and it is certain that there are many more still to be identified. Nowhere has a systematic search been made, and the addition of new substances to the list is haphazard. Usually a new compound is not tested for mutagenicity unless the investigation of its mode of action requires such a test. Moreover, the number of laboratories all over the world specifically engaged in the study of chemical mutagenesis is small; they are far fewer than those working on genetic effects of radiation. Almost no work has been done on the genetic effects of chemicals in mammals, and proof of mutagenicity is obtained from experiments with micro-organisms, seeds, and insects. Nothing comparable to the hugh mouse "factory" of the United States Atomic Energy Commission at Oak Ridge, in

which a hundred thousand animals are used for just one experiment on radiation mutagenesis, is available for the testing of chemicals.

No clear pattern can be discerned from among the chemicals that are known to be mutagenic. The most potent are the highly chemically reactive substances that have the ability to combine by a process known as alkylation with important cell constituents such as proteins and nucleic acids. Substances which have the ability of alkylate under physiological conditions are the epoxides, which have been shown to be highly mutagenic. Epoxides may be formed when foods are irradiated with atomic radiation in order to sterilize them, and this as well as the presence of peroxides, raises the question whether such foods are really safe to eat, though they pass the general food tests for toxic properties. As yet this problem has not arisen because the radiation doses needed to produce complete sterility are so high that they alter the taste of food.

Besides the alkylating agents, many other mutagens are known, but in their case there is no known chemical activity to which their mutagenic action can be ascribed. There are many ways by which genes can be altered, and this probably accounts for the diversity of mutagenic agents. Among the mutagens studied in most detail are formaldehyde and some peroxides, although the evidence that hydrogen peroxide behaves in this way is not strong. Peroxides that may be dangerous are produced when fats are heated, and it is possible that they are present in the frying-pan. Among substances that have been shown to produce mutations are some antibiotics, some hormones, and components of certain weedkillers and contraceptives. Usually these are tested

with one type of organism only, and it is not clear how far all these substances are universal mutagens as are the alkylating agents.

At the present time it is impossible to calculate whether chemical mutagens introduced by Man into his environment constitute a genetic danger. By this I mean whether they lead to a significant increase over the normal mutation rate inherent in cellular reproduction. Most authorities agree that exposure of large populations to atomic radiation must not be such as to double the normal mutation rate, as this would be reflected in a serious increase in certain diseases in later generations. For radiation we know, thanks largely to the pioneer efforts of the mouse geneticists at Oak Ridge in the United States, the order of dose that would double the mutation rate. No such information is available for chemicals. Not only do we not know all the chemicals in our environment which produce mutations, but we cannot calculate the dose that is received by the germcells as a result, say, of inhaling formaldehyde or eating peroxide. It is obvious that safety measures can never be as rigorously defined as with radiation, but the difficulties do not justify the complete disregard with which the subject has so far been treated.

In industry the situation is particularly unsatisfactory, since the maximum tolerated levels of volatile substances that may be present in the air of workrooms and factories is determined largely by the amount that produces immediate discomfort or obvious symptoms within a relatively short time. The so-called "tolerable levels" that are published in America and used as a guide here for some substances that

are already known to be mutagens (such as ethylene oxide, ethylenimine) would produce in the fruitfly a detectable incidence of genetic damage. Exposure to levels of radiation of comparable mutagenicity would be considered completely unacceptable.

It may be that all the possible embryonic and genetic hazards can, on investigation, be discounted one by one. Let us hope so, but instead of merely hoping, let us find out. At the present time these long-term problems are rarely if ever considered by toxicologists and others responsible for industrial hygiene. This disproportionate preoccupation with immediate effects is in striking contrast with the approach of their opposite numbers in the field of radiation protection.

DEPENDENCE OF MUTATION RATE ON RADIATION INTENSITY

A series of recent papers by Russell and his co-workers[4-7] presented evidence that the radiation induction of mutations is dependent on radiation intensity (dose rate), and new data[8] have been presented that strongly support earlier results showing a lower mutation frequency with chronic gamma irradiation than with acute x-rays. Their new findings are summarized in the following quotation:[8]

> "The possibility, rather remote to begin with, that the results could be explained in terms of radiation quality (x-rays versus gamma rays) rather than radiation intensity now seem to be ruled out by additional data obtained from experiments comparing mutation rates induced by acute gamma irradiation with those induced by chronic gamma irradiation. These additional data were obtained with the genetic criterion (specific locus mutations) and cell stage (spermatogonia) most cogent for the comparison.
>
> "The view that the difference in mutation rates is due to an effect of radiation intensity on the mutation process rather than on cell selection was originally supported primarily by the results obtained with oocytes. The additional data obtained with spermatogonia now also provide some evidence on this point: (1) Where a probable effect of cell selection on mutation rate was found, it worked in the direction opposite to that required to account for the above difference in mutation rates. (2) The mutation rates at 86 r and 300 r, for chronic gamma irradiation of spermatogonia, are not higher than a linear interpolation between the control points and points at higher doses. Thus, there is no evidence of a distortion of mutation rates by cell selection."

Russell et al.[8] re-examined the possibility that cell selection might account for the intensity effect in oocytes. Even when the comparison between the results from chronic and acute irradiation was restricted to follicle stages in which oocyte killing was infrequent, they found that an intensity effect on mutation rate was observed. In re-examining the possibility that the difference in mutation rates obtained with chronic and acute irradiation of oocytes could be due to

differences in oocyte stage, they found that even when the comparison was restricted to oocyte stages that are closely similar, an intensity effect on mutation rate was found.

They concluded[8] that (1) the effect of radiation intensity appeared to be greater in oocytes than in spermatogonia and suggested various possible explanations and that (2) with acute x-irradiation, the mutation rate was higher in oocytes than in spermatogonia.

Irradiation of Spermatogonia

Irradiated and control male mice were mated to females homozygous for seven autosomal recessive visibles. The offspring were then examined for mutations at the seven loci. Details of the experimental procedure have been reported.[6,9] The new data for chronic irradiation are presented in Table V-1. Data on chronic and acute irradiation are compared in Fig. 1, in which the solid points represent results obtained with acute x-rays at 80 to 90 r/min, while the open points show chronic gamma-ray results (triangles and square, 90 r/wk; circle, 10 r/wk). The squares represent mutation rates in females; all other points are mutation rates in males. The point at zero dose is the sum for all male controls. Except for one of the 1000-r acute dose points, it is clear from Fig. 1 that higher mutation rates are indicated from acute doses than from chronic doses.

The authors tested their hypothesis that a 1000-r acute dose was killing more of the sensitive cells than was 600 r and that the lower mutation rate data point at 1000 r was characteristic of the more resistant cells (which were also assumed to have a lower mutation rate

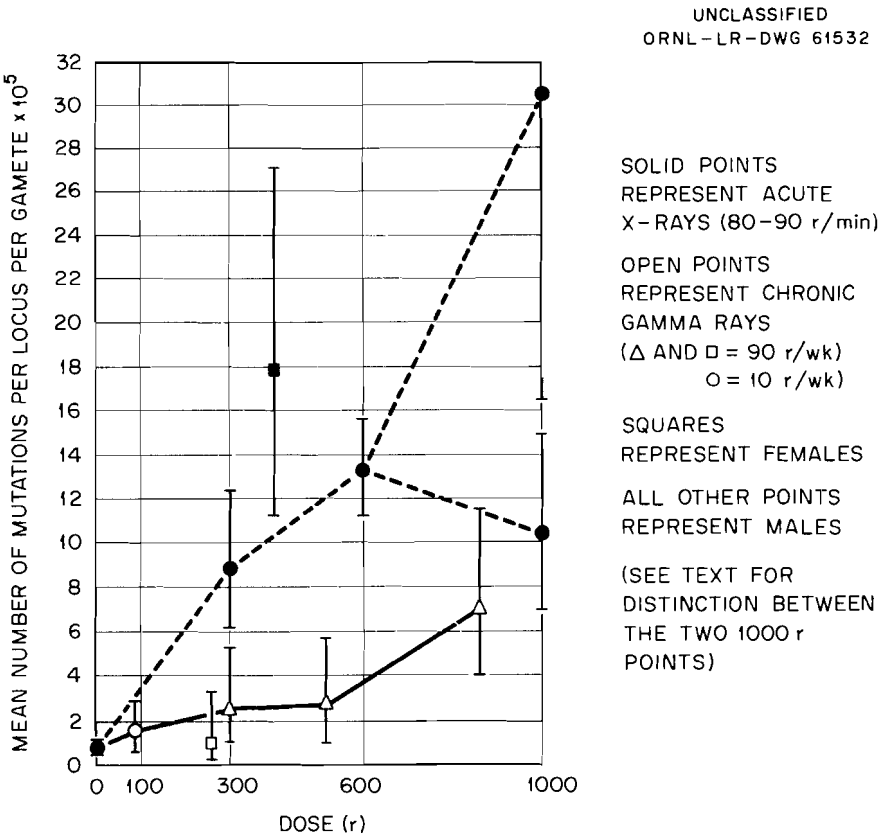

Fig. 1. Mutation Rates of Specific Loci in the Mouse (with 90% Confidence Levels).

Table V-1. MUTATIONS AT SPECIFIC LOCI INDUCED IN SPERMATOGONIA OF MICE BY CHRONIC GAMMA (Cs^{137}) IRRADIATION

Dose (r)	Dose Intensity (r/wk)	Number of Offspring	Number of Mutations at Seven Loci	Mean Number of Mutations per Locus per Gamete ($\times 10^5$)
0	0	166 147	10	0.86
86	10	56 993	6	1.50
300	90	28 170	5	2.54
516	90	26 321	5	2.71
861	90	24 281	12	7.06

than the less resistant cells). The upper point at 1000 r represents a first dose of 600 r, which was followed more than 15 weeks later by a second dose of 400 r. The authors point out the probability that cell selection interfered with the intensity comparison in the case of the single 1000-r dose, instead of accounting for a difference between chronic and acute results, and that the higher divided-dose point is a more valid point for comparison.

It is also pointed out that the chronic doses of 300 and 86 r to spermatogonia do not show evidence of important effects of cell selection on mutation frequencies. Thus, Fig. 1 indicates that cell selection is probably not important in this respect at chronic doses of 861 r or less. A similar conclusion had already been reached in tests of oocytes.[6,7]

Tests of the relative effects of gamma and x-ray irradiation were continued from previous work[6] and indicated that no great difference in mutation rates was caused by the difference in quality of irradiation. In any case, any effect of quality was much less than the effect of intensity.

Irradiation of Oocytes

The results obtained to date on irradiation of oocytes are presented in Table V-2 and in Fig. 1. These data include some obtained in 1959 in experiments[7] in which mutations were measured in the offspring of females exposed to acute x-irradiation. The mutation rate found was almost eleven times that for chronic irradiation, the latest data for which are also included.

Table V-2. MUTATIONS AT SPECIFIC LOCI INDUCED IN MOUSE OOCYTES

Radiation	Dose (r)	Dose Intensity (r/min)	Number of Offspring	Number of Mutations at Seven Loci	Mean Number of Mutations per Locus per Gamete ($\times 10^5$)
None*	0	0	5 845	0	0
X*	400	96	1 729	1	17.8
X	400	92	11 124	15	
None	0	0	47 612	0	0
Gamma (Cs^{137})	258	0.0085	27 174	2	1.05

*Experimental work done in 1950–1951.

The total plot of the data indicates even more strongly than before that the dose-rate effect on oocytes is on the mutation process itself. Two proposed alternative factors that were rejected as causes of the intensity effect were: (1) cell selection and (2) difference in oocyte stage in the chronic and acute radiation experiments.

Cell selection was discarded as an explanation because, finally, there is a significantly higher mutation rate for acute irradiation even when the chronic irradiation data are restricted to the stages almost certainly free of extensive oocyte killing. (It had been found that the

percentage of oocytes killed in certain stages of development was somewhat higher than had been previously noted.[10])

The acute irradiation experiments had been carried out only on dictyate oocytes irradiated in relatively mature follicles, while the chronic irradiations involved follicles in a variety of stages of development. The oocytes in early-stage follicles are the more sensitive to radiation killing.[7,11] Therefore, an analysis was made in which the only litters considered were those conceived within two weeks following the accumulation of the dose.[7] Combining the results with those of Carter,[12] it was found that the mutation rate for chronic irradiation was still significantly lower than for acute irradiation of females; this indicates that the difference between acute and chronic irradiation is not due to oocyte stage of development. It was noted that this comparison is more rigorous than that in the previous paragraph, where the absence of oocyte killing was the sole criterion for comparability. Additional experiments are being conducted at intermediate intensities, offering better comparisons for closely comparable cell stages.

Comparison Between Results in Spermatogonia and Oocytes

The authors discuss possible reasons for the apparently greater effect of irradiation on oocytes than on spermatogonia and suggest that further tests be made to clarify some of the questions raised. The reasons mentioned included (1) somewhat different irradiation "repair" mechanisms, (2) lack of cell selection in oocytes in contrast to the cell selection in spermatogonia which acts to obscure the full effect of radiation intensity, and (3) the existence in

acutely irradiated oocytes of a class of mutations not found either in spermatogonia or with chronic irradiation of oocytes. (W. L. Russell)

RIVER SURVEYS

The wise selection of a site for a nuclear power plant or a fuel fabrication or reprocessing facility and the judicious operation of the plant after it is built require a study of the possible immediate and long-term effects of radioactive materials on the environment. This is particularly true if any radioactive wastes are to be deliberately released, but the ever-present possibility of accidental loss of control of such materials is reason enough for a careful investigation.

River surveys form an important part of almost all attempts to characterize the environment of a power plant or industrial facility. Whether radioactive materials will reach the river accidentally or through deliberate discharge, the investigator must answer the same primary questions: (1) How rapidly will the radioactive materials be moved to points downstream? (2) In what concentrations will the several radioactive contaminants arrive at these points? (3) How long and in what concentrations will they persist? In addition, prudent planning for waste disposal or for emergency measures in case of an accident requires an evaluation of the possible long-term hazards which might result from assimilation and possible reconcentration of the radioactive materials by the total environment, both physical and biological. This long-term hazard is largely a problem in ecology and is not considered in this review.

Types of River Data Required

Data concerning (1) the volume and velocity of the stream flow, (2) the volume and rate of movement of sediment transported by the stream, both in suspension and as bottom load, (3) the chemical quality of the

water, (4) the radioactive-materials content of the water, both natural and from fallout, (5) the temperature of the water, (6) its microbiotic content, and (7) the degree of turbulence, dispersion, and mixing which result from the flow of the river are all required to answer the primary questions. Because all these quantities or properties are in some degree directly or indirectly dependent on the rate of flow of the stream, all must be studied in terms of rate of stream flow and be related, if possible, to variations in this rate. For almost all streams, the studies of variations of stream flow with time and, more particularly, the potential distribution of maximum and minimum flows, present complex problems.

The economic and social importance of rivers has been the motive for many long and penetrating studies, and a variety of valuable methods have been developed for the necessary investigations. It is essential that records and data be available that cover a considerable period of time. A reconnaissance survey must include at least one annual cycle. (The standard "water year" usually begins October 1, which is, in general, the time of low water level and low stream flow.) Comprehensive studies of the distribution and variability of the quantities and qualities of interest require, however, a minimum of five years of data and, for many streams, reliable predictions require a base line of many decades. Sustained periods of low rainfall and low stream flow can limit, importantly, the volume of water available to dilute radioactive contaminants at a time when the need for dilution may be imperative. On the other hand, a single great flood could move and redistribute river bottom and river bank sediment quite beyond the capabilities of any "normal" flood, the sediments thus redistributed would contain concentrated radioactive contaminants.

At Los Alamos, Hanford, the National Reactor Testing Station (NRTS) in Idaho, and the small test-reactor site at Dawsonville, Georgia, relatively little hydrologic information, at least of the type and reliability required, was available prior to the initiation of environmental studies. There had been little prior need for such data at Los Alamos and Dawsonville. The Columbia River, which flows through the Hanford site and the Snake River in Idaho had been studied but not in the detail required. In general, rivers such as these which drain areas of low rainfall are subject to larger variations in both short-term and long-term flow than rivers in more humid areas, so reliable data covering long periods are needed for evaluation of potential hazards.

For the site studies at the Brookhaven National Laboratory on Long Island, some long-term records were available but considerable interpolation and extrapolation were required. The eastern half of Long Island was studied in some detail in the first decade of this century as a possible source of water for New York City, and there are records on some of the streams from environmental studies that were started in 1947; however, the gap in the records between the earlier and the more recent work cannot now be filled.

The most nearly adequate river data are those available for the Ohio and Clinch Rivers. The Ohio River has been studied for many years because of its industrial and transportation importance, and the Tennessee Valley Authority has extensive data on the Clinch River because of its development for power production and flood control.

Special Data Required for Radiological Studies

The environmental studies considered here, like studies of flood control, navigation, water supply, and power development, while in each case requiring certain special data, must all be based firmly on a broad understanding of the river system in question. The general and specialized information required for radiological studies has been discussed by Tsivoglou et al.[13] in a paper that is quite useful to those working in the fields of public health and water-pollution control. They describe the organization, planning, and conduct of stream surveys, the basic equipment required for a modest radiological investigation, and the comparatively extensive requirements of the work which would be required to support a comprehensive study. Some details are given concerning the selection of sampling points, frequency of sampling, determination of background radioactivity, and sample collection equipment. Equal emphasis is placed on determinations of the levels of activity in the water, the biota, and the bottom sediments.

Types of Hydrologic Stations

Even under rather widely differing circumstances there will be a need for three distinct types of hydrologic stations: the areal primary station, the areal secondary station, and the partial-record station. Criteria for establishing such stations, and the details of their operation, have been described by Grover and Harrington[14] and by Corbett et al.[15]

The purpose of the areal primary stations is to provide a long-period record that is as continuous and complete as is feasible, since only from a long base line is it possible to determine the correlations

between the many factors of interest, both within their normal limits of variation and at times of unusually high or low flow. The period of record required is indefinite, but in many cases a record for a full century would be of material value if it were available. Some long-period, but less than full century, records are available at a limited number of stations on some of our rivers, largely as a result of the work of the U. S. Geological Survey. These data, which are neither as complete or as accurate as desirable, are published in the annual series of Water Supply Papers of that organization.

For many environmental studies, however, there will be no areal primary station data available. In general, there will be a station or stations somewhere near the area to be studied, and the records, while falling far short of what might be desired, can provide an invaluable foundation for more detailed local work. It will be a rare and fortunate circumstance when there are detailed long-term data available for a critical point or points within the actual area of interest.

The areal secondary stations provide a basis for correlations between the primary stations and the principle points of interest in the region being surveyed. In almost every case, the secondary stations will be established for the specific environmental survey. Since their value will depend upon the length of time their records overlap the records of a primary station, there establishment should have first priority. The many problems and uncertainties of the required correlations can be materially reduced if the secondary stations are established at points which resemble, as far as possible, the locations of the primary stations with which they are to be compared. Many factors must be considered,

however, and some, perhaps a majority, of the secondary stations will have to be located with the particular needs of the environmental survey as the sole requirement. In this case, no correlation with older primary stations should be attempted. Such secondary stations are actually new primary stations, and, if the eventual accumulation of long-period records is contemplated, the sites should be chosen with this in mind. In general, however, a continuous record is kept at a secondary station only until a correlation with the records at a primary station has been worked out. Usually this will require from five to ten years. After a correlation is established, the equipment may be moved to another site.

At the third type of station, the partial-record station, only a few measurements are made and only a few samples collected. Where possible, the data are collected at times of maximum or minimum conditions rather than at fixed intervals of time, and an attempt is made to relate these extremes to the conditions at one of the primary or secondary stations.

Types of Correlation Required

The general types of data required, as listed above, include stream flow and stream velocity, the chemical quality and radioactivity of the water, and the nature and quantities of sediment transported. Commonly, there is a relation between stream flow and the chemical quality and sediment loads of the stream. If these relations can be established and the variability and distribution of flow can be defined, the total load transported by the stream can be determined from flow measurements alone. The variations of stream flow should also be studied to determine the percentage

of time any given flow rate will be equaled or exceeded, the duration of flow rates, the rate of variation of discharge with time, the magnitude and frequency of floods and of low flows, and the volume of storage required to reduce flood crests or to maintain certain minimum flow rates.

Methods for computing and expressing these relations and for correlating the records of the several types of hydrographic stations have been described by Searcy[16,17] and by Dalrymple.[18] Not only can such correlations be used to predict the volume of water which will normally be available for diluting radioactive contaminants, but also they will make possible estimates of the cost of dams and other structures required to regulate both high and low flows in the stream by storing flood waters and later releasing them. Some estimate may also be possible of the potential frequency of unusual floods which might damage these or other related structures.

Sediment Surveys

A large portion of the radionuclides in radioactive wastes and in the activity that would be released by a reactor accident are cationic and are strongly adsorbed by the clay and silt particles which form the greater part of the suspended sediment and bed load of most rivers. Indeed, the movement and retention of radioactive contaminants in a river system will largely be governed by the transport of the finer grained sediments to which will be attached most of the fission products. Sorathesn et al.[19] have described some of the factors which affect or control this sorption process. These factors include the chemical character of the radioactive material, the chemical composition of the waste stream and the river, and the composition, grain size, and concentration of the sediment.

The relation of sediment load to stream flow and velocity is complex, and although the general principles have been more or less defined by a number of classic studies, their application to specific areas is anything but straightforward.[20-23] Colby and Hembree,[20] for example, have derived a duration curve of suspended-sediment discharge for the Niobrara River in Nebraska from the sediment-flow relationship and the flow-duration curve. Similar relations could doubtless be derived for other streams if the basic data were available and perhaps could be extended to define the relation between flow and the distribution of particle size and particle composition of the suspended sediments. Eventually it may be possible to establish similar relations between stream flow and the movement of the bottom load of the stream.[22] Even where only qualitative relationships can be established they can throw considerable light on an otherwise unmanageable accumulation of emperical observations. Variations in the chemical quality of the river water may also be related to stream flow and to other factors.[24,25] For example, Davis[25] has shown how the quality of the water in a stream is affected by the nature and extent of the geologic formations which underlie the area of its watershed. The natural radioactivity background of the water and sediments in a stream may be similarly controlled by geology; for example, the activity in the sediments in the bed of the Clinch River in Tennessee is highest opposite the mouths of tributaries draining areas of Chattanooga shale, a formation relatively rich in uranium.[21]

Selection of Sampling Sites

Sites for sampling to measure background radioactivity in water and sediments should be established prior to any release of radioactive materials

to the stream. Several of these locations should be at established or potential areal primary stations on streams to which no radioactive wastes are expected to be released. Background readings at these stations should be carefully correlated with the background in potentially contaminated streams prior to their contamination so that later readings may be adjusted for background on the basis of such correlations. The local geology must be carefully considered in the selection of these sites if the correlations are to remain valid.

Other sampling points should be located at the mouths of large tributaries, and at dams, locks, hydroelectric stations, or any other points where stream flows are brought together and channeled so that representative samples may be more easily obtained and the results of the sampling may be correlated with measurements of the stream flow. Where a contaminated waste stream is added to a river, the two most important sampling points are on the waste stream itself, where quantitites are easier to measure because concentrations are higher, and on the river upstream from the point of waste discharge, since any activity found there will be background.

For simplicity in operation, the first station downstream from the point of release should be located where complete mixing of the contaminant is always assured. In the Columbia River, near Hanford, the contaminants are first detected on the opposite bank 7 miles downstream, and complete mixing is probably assured 40 miles downstream.[26] On smaller streams complete mixing may occur within a few hundred feet of the release point. On the other hand, flow in the Ottawa River at the Chalk River Plant is so slow that there is very little mixing, and radioactivity in

the river can only be detected in the general vicinity of the waste pipeline outfall.[27] Another factor that must be considered is the quantity of radioactive material released, for when it is very small, as in the Chalk River, there is little reason to expect to detect it at points distant from the point of discharge. In other cases, however, the contamination may be detected downstream for hundreds of miles, as shown by Nelson.[28] Actually, very little information is available on lateral dispersion in streams and probably the most effective approach for the determination of the location of complete mixing and for the selection of sampling points is to conduct a series of tracer experiments over a range of flow conditions.

Other sampling points should be located where the first major use is made of the water downstream or where the public has access to the water for swimming or fishing. There are no general criteria for the selection of such points because they depend entirely on local conditions.

Samples of bottom sediments should be obtained in the vicinity of abrupt changes in flow conditions and intermediate drainage. Abrupt changes in flow may be expected at constrictions and expansions of the river channel, at points where there are changes in channel roughness, at bends, at changes in slope of the river or river-bottom profile, at the mouths of large tributaries, and at dams.

Sample Collection

In general, the collection of some 8 to 15 samples per year at each of the partial-record and areal secondary stations will be sufficient to describe the full range of variation in flows, in chemical composition, and

sediment-load characteristic, if the times of sampling are carefully chosen to coincide with times of normal flow and with peaks and lows of stream discharge.[29] Care should also be taken to sample the true base flow of the stream. This can hardly be done if the samples are collected routinely as, say, on the first day of each month. The visits to the sampling stations have to be irregular and governed by the flow conditions, which may on occasion change quickly with little prior warning.

In streams contaminated by radioactive wastes, the practice has been to collect "continuous" water samples. Equipment is commercially available which will collect continuous proportional samples from very small streams.[30] This equipment is placed in a pool upstream from a weir. The creation of such a pool may, however, cause deposition of suspended sediments with their adsorbed radionuclides. If a material balance is to be made, the radioactive materials on these and other bottom sediments must be taken into account.

In large streams, the collection of once-daily water samples may be as satisfactory as continuous sampling if the size of the daily sample is made proportional to the daily flow and the changes in the flow of the stream and in the rate of release of radioactive wastes within a day are not large, say, not over 15%. In order to reduce laboratory analytical costs, the daily samples may be composited into weekly, monthly, or even quarterly samples, but if there are many or rapid changes in the stream flow, such composites may not give the true mean concentrations unless the size of the smaller samples composited is carefully adjusted to the flow of the period each small sample is to represent. Indeed, where variations in stream flow or waste discharge are frequent and rapid it may well be

necessary to take samples for compositing more frequently than once a day. Such samples may be taken, up to about four a day, with automatic programmed timers, sequencing switches, and solenoid valves. If a greater number of samples per day is required from a large stream, electronically actuated continuous proportional samplers should be designed. Such electronic equipment is feasible if the relationship between flow and water level for the stream can be determined and if the velocity and flow-area can be measured continuously.

Graphs prepared by Colby and Hembree[20] illustrate the degree to which velocity, sediment concentration, and the chemical quality of the water vary from point to point in a stream cross section. Whether samples are collected manually or by continuously operating equipment, they are, in general, taken from the same point. Therefore, it is desirable, from time to time, to sample the entire stream cross section adjacent to a sampling point to determine adjustment factors between the mean concentration and the concentration at the sampling point.

Most of the available equipment which has been designed to sample such cross sections is intended to collect only one type of sample, either a sample for determining the quality of the water or a sample for studying the biota or the suspended or bottom sediments; there is no one piece of equipment which can handle all phases of the sampling problem. Dissolved-solids samples may be collected in weighted bottles. Dissolved-solids samples, as well as samples of dissolved gases and very fine suspended sediments, can be collected in Kemmerer samplers (also called Guday and Faerst samplers). Special samplers which tend to minimize disturbance to the flow and so provide more accurate samples have been designed for

the collection of suspended sediments, but not all the problems involved in such sampling have been solved.[31] Plankton are sometimes sampled with nets, which tend to clog, and sometimes with Kemmerer samplers.

The greatest difficulty with automatic sampling equipment is to get representative samples of suspended sediments. If the samples are pumped from the river a sufficient velocity of flow in the pipeline must be maintained to prevent their settling out; Durand gives data concerning pipe sizes and pump capacities which will help in avoiding this source of error.[32]

The bottom load moved by the stream and, indeed, any part of the river bottom, whether stationary or in motion, is particularly hard to sample. Pederson and Ekman dredges will reach only the top few inches of the bottom sediments, and although other equipment[32] may reach deeper, the investigator must be prepared to design his own equipment to fit his individual needs. If the river scours its bed deeply in times of flood, the radioactive contaminants in the clay, silt, and sand may be distributed through a considerable depth of material. The bottom sediments in many rivers vary greatly from place to place in composition, texture, and cohesiveness. So-called "undisturbed" core samples, although highly desirable if an inventory or material balance of the radioactive materials is intended, are time-consuming to obtain and all too frequently disappointing. There is no one core sampler or standard sampling procedure that is suitable for all bottom sediment conditions. (P. H. Carrigan, Jr., U. S. Geological Survey)

RECENT STUDIES RELATED TO DETERMINATION OF MPC VALUES

(Material held over from draft of Vol. 3, No. 1 to be inserted here; see draft pages 187 through 190 and references 50 through 55 on pages 197 and 198.)

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VI. CURRENT EVENTS

EBR-1 OPERATING EXPERIENCE

The EBR-1, Experimental Breeder Reactor No. 1, is a liquid-metal (eutectic NaK) cooled, heterogeneous, experimental reactor with a fast-neutron spectrum and a nominal thermal power of 1.4 Mw (0.15 electrical Mw).[1] It was built for, and has been operated under the supervision of Argonne National Laboratory (ANL), an AEC prime contractor. The EBR-1 was constructed at the National Reactor Test Station (NRTS), Arco, Idaho, and operation began in 1951.

The EBR-1 was designed to fulfill two fundamental purposes:[2] (1) to investigate and demonstrate fast reactor breeding and (2) to demonstrate the feasibility of using NaK as a coolant in power reactors, both on a purely experimental basis. It was designed to be quite flexible,[3] both as to loading and operating conditions, in order to facilitate the accomplishment of its aims. Both of these fundamental purposes had been largely accomplished by 1957 during operation with two core loadings (Mark I and Mark II), as reported by Thalgott,[2] and data and corresponding theoretical models on several aspects of liquid-metal-cooled fast-breeder reactors had been developed. The measured conversion ratio was 1.01 ± 0.05, and no significant difficulty was encountered with the liquid metal systems. These developments have been described and followed in ANL progress and topical reports through three core loadings (Mark I, Mark II, and Mark III) and are being employed in the design of a fourth core (Mark IV).[4]

The EBR-1 consists, essentially, of an unmoderated reactor coupled to a heat sink. The heat sink consists of (1) a primary NaK coolant

circuit that removes heat from the reactor, (2) a secondary NaK circuit that removes heat from the primary NaK, (3) a superheater, boiler, steam turbine, condenser, and economizer (water) circuit that removes heat from the secondary NaK circuit, and (4) an electric generator driven by the turbine. The primary NaK flow is from an elevated constant-level tank into the reactor, down through the inner blanket, up through the core, out of the reactor, through the heat exchanger, and down to a receiving tank. From the receiving tank it is continuously pumped up to the constant-level tank, which contains an overflow to the receiving tank. Further details of the heat sink system are described by Zinn,[5] and Trocki et al.[6]

Description of EBR-1, Mark I, Reactor

The EBR-1 reactor, Mark I,[2,3,7] consisted of three principle parts: the core, the inner blanket, and the outer blanket.

The core of the EBR-1 is a hexagon approximately 8 1/2 in. high and 8 1/2 in. diameter (flat to flat). It contains 217 type 347 stainless steel-jacketed elements of 93.2% enriched uranium 3/8 in. in diameter. The stainless steel jackets extend through the upper shield for handling purposes, being approximately 10 ft long. There are slugs of natural uranium both below and above the fuel inside the jacket that form part of the blanket. A NaK annulus forms a thermal bond between the fuel and the jacket. The total inventory of U^{235} in the Mark I core was 55.657 kg, including 3.875 kg added for operating excess reactivity.

The inner blanket[7] consists of elements of stainless steel-jacketed natural uranium slugs 15/16 in. in diameter and 20 1/4 in. long jacketed in 0.020-in.-thick stainless steel tubes. These elements are also provided with extensions projecting through the upper reactor shield for insertion and removal.

An external air-cooled blanket[3,7] surrounds an inner stainless steel tank containing the core and inner blanket. The external blanket consists of natural uranium bricks clad with stainless steel. The blanket is 30 7/8 in. in diameter and 27 in. high and is air cooled. This cylinder rests on a hydraulic elevator platform and acts as a shim control (by raising and lowering it). An 81-kg block of natural uranium is centered below the inner tank. It is supported by air pressure controlled by valves and instruments so that a power failure would cause the block to drop away from the reactor core.

Surrounding the external blanket[7] is a graphite reflector 19 in. thick, followed by 9 ft of concrete shielding. Six experimental beam holes pierced the concrete shield and graphite reflector.[7] Thermal columns and a "rabbit hole" also provide for experiments.

Twelve 2-in.-o.d. control rods of stainless steel-jacketed natural uranium are provided that are designed for vertical motions through holes in the external blanket bricks. Eight are quick-acting safety rods, while four, with controls for more accurate positioning, are for normal operating control. (The relative speeds of these rods figured in the November, 1955, partial meltdown of the Mark II core.) The entire outer blanket, and the shield plug on which it rests, was designed to be dropped "quickly" by the hydraulic elevator in emergencies.

Safety Circuits

Of the 49 safety circuits mentioned by Unbehaun,[7] 26 were designed to provide signals indicating abnormal operation, 8 to provide signals indicating danger (and designed to scram the reactor after 2 min without correction), and 15 were designed to scram the reactor immediately. The immediate scrams were designed to act when any one of the following parameters left its specified range: pile period, fuel temperature, coolant flow rate, coolant temperature, elevator hydraulic pressure, radiation levels at designated points, or position of any of four valves. All safety circuits were duplicated to greatly increase their reliabilities. In addition, periodic checks with check lists are made by the operators, and frequent detailed inspections of all equipment and safety devices must be made and reported to the project engineer.

Fuel Inventory and Reactivity Control

Lichtenberger et al. reported at Geneva in 1955[3] that the operating control rods contained a total reactivity worth of about 1% k, that the safety rods could remove 0.27% k in 100 msec, and the external blanket could remove 8.2% k at a rate depending on an acceleration of a little less than 1 g. They reported that the critical mass was 48.2 kg of U^{235} and that an additional loading of about 4 kg was included to provide for operating control. They also stated that the average temperature coefficient was -3.5×10^{-5} $\Delta k/°C$ from 38 to 200°C and that filling with the coolant added reactivity approximately equal to that added by 2 kg of U^{235} (thus, loss of coolant reduces the reactivity). Separate

instrumentation was provided for startup, for safety, and for steady operation, and all safety circuits were duplicated, as mentioned above.

Operation of Mark I and Mark II Cores Through 1955

Unbehaum has reported[7] that when the Mark I reactor was started up in 1951, it was found that radiation levels around the reactor were higher than had been anticipated. An additional 30 in. of concrete shielding was therefore provided, and operation was resumed. In June, 1952, a NaK-to-NaK leak was discovered in the primary heat exchanger and the reactor was shut down for repair. During the shutdown, 16 fuel elements were removed for examination and were replaced by new elements. The duration of shutdown was not stated, but it was reported that more than 1500 Mwhr (thermal) of operation had been accumulated by April 15, 1953.

The fuel used in the first core was beta-quenched uranium. During operation of the first core, experiments at Argonne National Laboratory showed that an alloy fuel composed of uranium and 2 wt % zirconium was more resistant to irradiation growth and was free of the irradiation-induced surface-roughening characteristic of beta-quenched uranium.[8] Accordingly, in early 1954, a second (Mark II) core was installed with U–2 wt % Zr alloy in both the fuel and the blanket slugs.

Lichtenberger et al. reported at the 1955 Geneva conference,[3] with respect to Mark I and Mark II operation, that "The reactor system has proven to be very reliable and has given very good service during 3 1/2 years of operation .... This plant is inherently quite stable and is largely self-regulating ....."

On November 29, 1955, a partial meltdown of the reactor (second core) occurred during an experiment described by Zinn[9] as a "last experiment" to make measurements and to determine whether fuel rod distortion would occur with the reactor on a fast, positive period. He explained that the reactor was scheduled to be placed in standby in 1956, and that the fuel was to be removed without definite plans for its replacement. So, with the understood risk that some of the fuel might melt, it was decided that the information sought was valuable enough to warrant the risk, and the experiment was carried out. Since a measurement of the temperature coefficient of the fuel alone was desired, the flow of liquid-metal coolant was shut off. Therefore, the experiment was carried out on what amounted to a critical assembly rather than an operating reactor. The period was to be reduced to 0.27 sec and the temperature of the fuel was to be permitted to reach approximately 500°C so that shutdown would have to be initiated at the proper instant and accomplished rapidly. The reactor was equipped with both relatively slow-acting motor-driven control rods and relatively fast-acting shutoff rods. The control operator was to activate the fast shutoff rods on spoken instruction from the observing scientist in charge. Instead, he mistakenly activated the slower acting control rods. In less than 2 sec the scientist pushed the rapid shut-off button, and, simultaneously, the automatic power-level trips responded to activate the shutoff rods. This delay in reactivity reduction "was sufficient to permit the reactor power to overshoot to a point where the alloying of uranium and steel and melting of the uranium took place." It developed that melting had occurred in 40 to 50% of the core.[10]

The core assembly was removed from the reactor and shipped to the Lemont, Ill., Laboratory for examination and disassembly. Observations during disassembly and subsequent simulated meltdown experiments indicated that the porous structure formed in the core could have resulted from the vaporization of entrained NaK.[10]

Operation Since 1955 (Mark III Core)

Thalgott reported[2] in 1957 that, in normal operation, the Mark I and Mark II reactors were quite stable and largely self-regulating. He noted, however, that when the coolant flow through the reactor was changed, a prompt, positive metal temperature coefficient of reactivity was observed. He reported that experiments with the ZPR-III critical assembly indicated a weak Doppler effect in an EBR-1 spectrum but that it was too small to account for the observed coefficient. He proposed another possible explanation; that is, that bowing of the fuel rods toward the core axis occurred because of the increase of the flux and fission rate as a function of nearness to the reactor axis. He proposed that the inward bowing would result in a positive reactivity effect and concluded that appropriate mechanical design could prevent that difficulty. He also reported[2] that a second phenomenon observed in EBR-1 was an increasing tendency toward power oscillation when the ratio of power to coolant flow was increased. He pointed out, however, that a sufficiently large, prompt negative coefficient could override this tendency and so prevent instabilities. He described the features being incorporated in EBR-1, Mark III, that were aimed at prevention of inward bowing of fuel rods and described the design for increased flexibility in coolant flow arrangement and more complete temperature monitoring.

While the design of Mark I included stiffeners intended to keep the fuel elements a set distance apart (prevent bowing), the problem had not been fully appreciated. Mark II included much stronger stiffners to prevent bowing,[11] but it was concluded that radial expansion of the end plates, through which the fuel elements extended, had forced some bowing in the elements. Mark III was designed with great care in this respect to make certain that the fuel elements were rigidly prevented from bowing in under all conditions of operation.

The ANL annual reports for 1957[12] and 1958[13] confirmed the predictions that the EBR-1 with the Mark III core would be stable under normal operating conditions and that "the features of mechanical design responsible for the instability of the Mark II core have been eliminated in the new, more rigid design." Mark III exhibited negative isothermal temperature and power coefficients, and no evidence was found of a positive coefficient associated with an increase in ratio of power to coolant flow rate.

Stability experiments with the Mark III core were continued in 1959.[14] Exchanges of fuel and blanket rods to increase the temperature gradient across the flow separator gave results indicating that feedback effects originating in the radial blanket are not important. Another test with the stiffening ribs removed from a subassembly of blanket rods showed little or no feedback effects, indicating that bowing of the blanket rods is probably unimportant as a cause of feedback. In another test, removal of stiffeners from fuel rods "indicating rather conclusively that there is an effect completely consistent with the

suggestion that the rods bow inward." In all cases studied, the results "tended to substantiate the conclusion that a positive component had been added to the power coefficient" (by the removal of stiffeners).

A mathematical model was developed that showed "good agreement with reactor behavior." Use of the model to investigate extreme conditions showed instabilities at conditions considered most unlikely to be present in the reactor. "The principal conclusion is that stability of fast reactors is not a function of nuclear phenomena but is the result of mechanical phenomena which can be controlled by intelligent design."

A final analysis on the stability studies of the Mark III core was summarized in November 1960.[15] No change was indicated in the conclusion that adequate mechanical design prevented a positive coefficient from fuel rod bowing. It was indicated that a sufficiently large ratio of power to coolant flow would cause instability, but that the thermal power would have to be so high (about 1000 Mw) at full flow (10 Mw at one-third flow) that this is only of academic interest. The power coefficient was found to be nonlinear, but negative at all powers. The effect of inlet coolant temperature on the power coefficient was found to be real, but minor. Low-frequency (e.g., 400 sec/cycle) feedbacks from temperatures of large structures were found. Feedback from the radial breeding blanket, if any, was too small to be detected.

Operation from August 1960 through March 1961 consisted mainly of training reactor operators. The Mark IV core is in various stages of design, development, and fabrication.[4] It is to use fuel elements of Pu–0.125 wt % Al alloy jacketed in Zircaloy-2 tubing in a further series of experiments. (L. A. Mann)

IDAHO CHEMICAL PROCESSING PLANT CRITICALITY INCIDENT OF JANUARY 25, 1961

On January 25, 1961, at 9:50 a.m., a nuclear excursion of approximately 6×10^{17} fissions occurred in an evaporator within a shielded cell of the Idaho Chemical Processing Plant. This was the second criticality incident in the facility; the first incident was reported in a previous issue of Nuclear Safety.[16] The January 25 incident is described in the report of an investigating committee.[17] None of the 251 persons present in the ICPP area at the time of the accident received significant radiation exposure. The maximum radiation dose rate from ground deposition downwind of the facility was 30 mr/hr. No costs were incurred which were attributable to contamination cleanup, product recovery, or equipment repair as a result of the incident.

The accident occurred while operating personnel were attempting to use air pressure to remove an apparent plug in the evaporator product-withdrawal line. The product evaporator is of the thermosiphon type, with a geometrically safe bottom section and a 24-in.-diam nongeometrically safe vapor-disengagement section. It is similar in construction to the evaporator in which the Thorex Pilot Plant explosion occurred.[18]

Approximately 40 liters of solution containing 200 g of 90% enriched U^{235} per liter could not be transferred from the bottom section of the evaporator through a line that extends from the bottom section through a valve and remote head diaphragm pump to a temporary storage tank, either because of failure of the pump or a plug in the pump discharge. An attempt was made to remove a possible plug in the pump discharge by closing the valve between the pump and the evaporator and forcing approximately 4 liters

of water from a 10-gal tank pressurized with 40-psig air through a decontamination line into the suction side of the pump. The 10-gal water tank was located in the plant makeup area. It became apparent that the possible plug had been removed when the liquid-level recorder on the temporary storage vessel indicated an accumulation of approximately 4 liters. The operator in the makeup area was then instructed by voice communication through a pipe chase from the operating area to close the manual valve between the decontamination tank and the pump suction. A muffled reply from the operator in the makeup area was heard, and thereupon the operator in the operating area turned on the pump and opened the valve between the evaporator and the suction side of the pump. Immediately upon the opening of the valve, radiation alarms sounded throughout the plant. At the sound of the alarms, the plant operations superintendent proceeded to the master radiation panel in the health-physics office, found it almost completely covered with alarm lights, walked to the end of a corridor to a plant evacuation alarm station, and actuated the plant evacuation alarm; the alarm was actuated within no more than a minute after the radiation alarms had sounded.

The incident apparently occurred because a burst of air from the evaporator product-withdrawal pipeline lifted most of the fissile solution from the geometrically safe leg of the evaporator into the non-geometrically safe vapor-disengagement section. Neutron activation studies of metal samples from the evaporator system have verified that the excursion occurred in the disengagement section. The operator in the makeup area stated after the incident that the muffled reply he made to the process

operator was that there was still some 5 to 10 psig of air pressure on the line as he closed the valve. It is supposed that the process operator in the operating area assumed that this pressure had previously been vented through the temporary storage tank after transfer of the 4 liters of water. The investigating committee stated that the postulated mechanism by which the solution was lifted in the evaporator would be more readily acceptable if the manual valve between the 10-gal pressurized tank in the makeup area and the pump suction line had not been closed, inasmuch as a larger volume of air would have been available for expansion. The operator in the makeup area testified, however, that the valve was closed. The available radiation instrument charts indicate that the nuclear excursion was of short duration (<2–3 min) but did not necessarily consist of a single burst. Analysis of the fissile solution indicated that a total of 6×10^{17} (±25%) fissions had occurred.

Evacuation of the building and outside working areas to a parking lot was complete within 5 to 7 min after sounding of the evacuation alarm, except for one person working near an air compressor, who did not hear the alarm and evacuated about 10 min after the alarm had been initiated; his film badge showed no detectable exposure. Subsequent analyses of about one-half of the employee film badges and pocket dosimeters and of blood sodium samples, together with bioassays and whole body counting, indicated a maximum gamma exposure of 55-mrem and negligible internal or external beta or neutron exposure.

Analyses made subsequent to the incident indicated that activity was released to the environment only through the vessel offgas system, which exhausts to the local plant stack. The vessel offgas treatment system

consisted at the time of the accident of only a small de-entrainment tower packed with Raschig rings and a long pipeline to the stack. The vessel offgas filters were being bypassed at the time of the incident.[19] The first proof of a criticality incident was received approximately 45 min after the incident by the determination of short-lived Cs^{138} on the stack-sampling filter. The radioactive release from the stack did not cause significant personnel exposures and resulted in a maximum ground radiation reading from fallout of 30 mr/hr.

The investigating committee concluded that absolute protection against criticality incidents is neither possible nor practical in fissile material-handling plants, since it is necessary to use administrative control in some of the operations and since the effects of an incident in an adequately shielded and contained facility are not disastrous. The committee did, however, make the following recommendations to strengthen the safeguards within the plant:

> "1. The use of steam or compressed air to clear obstructed process lines is to be permitted only with the specific approval of the Assistant Manager for Operations after the contingencies have been fully evaluated.
>
> "2. Communications between safeguard committees and operating personnel should be improved by codification of safeguard committees' actions, providing a full-time safeguard committee chairman, and improving the direction and follow-up in operator training, particularly with respect to infrequent and emergency conditions.
>
> "3. Engineering safety reviews should always consider complete systems rather than single items of equipment. In particular, the potential for maloperation or misuse of any part of the system should be considered.
>
> "4. Consideration should be given to the redesign of the evaporator system to increase physical protection against criticality.

"5. Attention should be given to improving intercommunication between work areas.

"6. As a matter of policy, all personnel badge dosimeters should be analyzed immediately following a criticality incident or accidental release of radioactive material.

"7. Additional technical personnel should be available each shift to provide increased surveillance over processing activities."

This incident served to emphasize that dispersive incidents are most likely to occur during periods of nonroutine operations, such as the line-unplugging operation, which may be less thoroughly planned and executed than normal process operations. A recommended policy for such nonroutine operations is that no operation should be performed until a technically responsible employee has developed a detailed step-by-step operating procedure, the procedure has been approved by another responsible technical employee, and the procedure has been reviewed and discussed with the personnel who will perform the operation.

The reviewer would suggest consideration of the use of a fixed poison to provide additional safety in nongeometrically safe sections of existing process equipment. It is often possible to place glass Raschig rings in evaporator disengagement sections, solvent de-entrainment vessels, column end sections, or miscellaneous collection vessels with relatively little inconvenience or cost.

It is apparent from the relatively minor personal exposures which resulted from this accidental criticality that the containment features of the plant functioned satisfactorily. The exposures would have been even smaller, however, had the evacuation signal been given more promptly, as it could have been if it had been possible to actuate the evacuation

alarm from the master radiation panel. Automatic actuation of the evacuation alarm should be considered; this frequently involves the use of coincident signals to avoid unnecessary evacuations.

The reviewer would also like to reiterate the comment that was made in the review of the first ICPP criticality accident,[16] namely, that the consequences of the accident would have been lessened if a more elaborate vessel offgas system had been used. In particular, offgases of radiochemical processing plants should pass through a caustic scrubber and roughing and absolute filters before being exhausted through a stack. It has been calculated that, utilizing such a system, the maximum possible cloud dose and downwind ground contamination resulting from a criticality incident of 10^{18} fissions would be negligible. If such a system had been in use at the time of the incident of January 25, calculations indicate that the maximum ground contamination level would have been less than 0.25 mr/hr, as compared with the 30 mr/hr which was obtained. (J. P. Nichols)

THE SL-1 INCIDENT

The first fatalities as a result of a reactor incident in the United States occurred at the SL-1 reactor facility in Idaho on January 3, 1961. In the time since, the events surrounding the accident have been thoroughly studied by the AEC Board of Investigation. The May 10 report of this Board[20] plus the subsequent testimony before the Joint Committee on Atomic Energy[21] in June provide an excellent insight into the incident and the circumstances surrounding the incident as well as the lessons to be learned therefrom.

In the words of AEC Commissioner R. E. Wilson,[21] "the accident could and should have been prevented." That many of the conditions which directly or indirectly contributed to this accident were allowed to exist is only additional evidence of the fallibility of human beings and their organizations, inasmuch as these conditions repudiate accepted safe practice in reactor design and operations as now recognized and effected at most all nuclear installations in this country. This accident should serve to remind all those not immediately concerned with nuclear safety of the potential dangers of nuclear installations and of the importance of following as closely as possible the intent, as well as the letter, of existing safety practices.

The following review is comprised principally of excerpts from the two sources referenced above. It should be noted that these references are not the final report on the incident, but, pending the completion of the reclamation work now under way at the facility, they constitute an interim status report. It seems unlikely that the conclusions of the

present reports will be significantly altered by the results of the reclamation work. Analyses have established many of the facts surrounding the incident, and many important things have been learned from (or confirmed by) this accident.

Description of the Facility[20,22-24]

The SL-1 is a direct-cycle, natural-recirculation, boiling-water reactor designed for 3000-kw (gross) thermal capacity to produce 200-kw (net) of electricity and 400-kw (net) electrical-equivalent energy in the form of space heat. Work on this plant started in 1955 in response to a request of the Department of Defense for a small nuclear power plant to operate at remote military installations, such as the Dew Line, where minimum operation and maintenance crew size would be important and, hence, ruggedness, simplicity, and long life would be highly desirable. The boiling-water system proposed by Argonne National Laboratory was selected for development and construction on the basis of compactness, simplicity, reliability, technical feasibility, long core life, and minimum plant cost. Construction and operation of the reactor at the National Reactor Testing Station were intended to test the design, to diagnose practical troubles, and to develop corrective measures and improvements.

The site facilities consist of the cylindrical reactor building, an adjoining support building, a training building for the military cadre, an administration building for contractor personnel, and a recently completed air-cooled-condenser building. The majority of the plant equipment is located in the cylindrical reactor building, which is 38.5 ft in diameter and has an over-all height of 48 ft. This building is made

of steel plate, most of which has a thickness of 0.25 in., but it is not a pressure-tight containment vessel.

The reactor vessel, fuel storage well, and water demineralizer are located in the lower third of the building and are shielded with gravel, because that material would be available at the ultimate construction site. A recirculating, air-cooled condenser is located in the upper third of the building. The middle third of the building contains the turbine generator, feedwater equipment, switch gear, and shielding blocks located around the pressure vessel head. The active core is located near the bottom of the pressure vessel; above is the chimney section. Control rod extensions pass through the top head and into thimbles mounted thereon. Over the head is a metal enclosure that is filled with metal punchings and gravel to provide shielding. A top shield cap rests on the shielding blocks.

In the operating reactor, the control rods were driven by a rack and pinion located in a thimble above the upper head of the reactor vessel. The pinion shaft penetrated the thimble wall through a rotating seal, and the shaft was driven by a motor and gearbox through a clutch. De-energizing the clutch coil released the pinion from the gears and permitted the rod to fall. An auxiliary clutch permitted the motor to drive a released rod downward and prevented upward rod motion after release. In the rod-drive mechanism as originally conceived, the downward force of gravity was available at all times to hold the absorber rod in the core. No differential pressure existed across the rod as a consequence of pressurized reactor operation, and the pressure drop across the rotating seal exerted no force on the rod. However, the upper end of the thimble had to be removed for

access to the mechanism and to the core. With the thimble removed, transient pressure rises in the vessel would result in transient upward forces on the rod in the direction of increasing reactivity.

The core structure has provisions for 60 fuel assemblies and nine control rods of which five are cruciform rods and four are T rods. The core now being reclaimed has 40 fuel elements and five cruciform control rods. The control rods are made of cadmium mechanically clad with aluminum. The 40 fuel assemblies are composed of nine fuel plates each. The active portion of each fuel plate is 25.8 in. long and 3.5 in. wide and consists of uranium-aluminum alloy with aluminum cladding. The initial loading of the 40-assembly core was highly enriched and contained 15 kg of U^{235}.

On each of the 16 fuel assemblies in the center of the core, a full-length burnable poison strip was spot-welded to one side plate, and a half-length strip was added to the other side plate. Each of the remaining fuel elements had a single full-length boron plate on one side plate. The strips were aluminum-nickel containing elemental boron. The half-length strips were 21 mils thick, and the full-length strips were 26 mils thick. The core contained a total of 23 g of B^{10} as burnable poison.

Events Preceding the Incident

Argonne National Laboratory performed the initial criticality and startup tests and successfully completed a 500-hr full-power plant performance test in December 1958. In February 1959, Combustion Engineering, Inc., took over the operation of the SL-1. Since startup, the SL-1 has been used to gain operating experience, develop understanding of plant performance characteristics, obtain core burnup data, train military

personnel in plant maintenance and operations, and test components planned for use in subsequent reactors of this type.

As previously noted, boron strips had been incorporated in the core to serve as burnable poison, the depletion of which would compensate for the burning of the fuel and thus extend the operating life of a core loading. By May 1960, it appeared that the core was gaining reactivity faster than predicted. During an August 1960 inspection by Combustion Engineering, it was observed that pieces of the boron strips were missing from some fuel elements, and the fuel elements in the center of the core were extremely difficult to remove by hand. Removal caused pieces of the plates to fall off, and there was flaking of the material. Many flakes were collected from the bottom of the vessel. As a result of these circumstances, it was felt that further removal of fuel elements might cause additional loss of boron, so the remaining fuel elements were not inspected.

The safety significance of the greater-than-expected rate of reactivity gain lies, of course, in the reduced capability of the control rods to render the core subcritical (i.e., a decreased shutdown margin). Because of the reduced shutdown margin resulting from the boron loss, strips of cadmium were inserted in two of the four T-rod control shrouds on November 11, 1960. The offsetting effect of this cadmium at operating level was found to be about 1% reactivity.

Although numerical values for core reactivity, rod worth, and shutdown margin are all subject to various degrees of uncertainty, depending on the assumptions made as to the amount of boron loss, poison burnup, fuel burnup, etc., the available information indicates the following:[21]

"1. The initial shut-down margin for the cold reactor was probably somewhat less than intended — maybe about 3.5% delta k actual margin versus an estimated 4 — 6% design margin. This margin was, nonetheless, considered adequate by Combustion Engineering and the AEC Operations Office.

"2. The design of the reactor was such that withdrawal of the central control rod alone would make the reactor critical.

"3. At the time of shutdown on December 23, 1960, the shutdown margin for the cold reactor was probably 2 to 3%, assuming rod worth was essentially unchanged from earlier measurements and calculations. With this assumption, and a similar one regarding ... the central control rod, it was found that criticality could be produced by withdrawal of this rod approximately 17 inches from the reference zero position."

Testimony before the Board[20] and operating records indicate that in recent months of operation there was increased frequency of sticking of the control rods. On the one hand, it was postulated by several witnesses that the bowing of the boron strips attached to the fuel elements exerted sufficient lateral force to result in reduction of the clearance within the control rod shrouds and restriction of the free motion of the blades. On the other hand, several witnesses felt there was no evidence for such closing of the shrouds but that there might have been some accumulation of dirt or corrosion product on the shroud and blade surfaces. It was also indicated that the higher power operation, that is, operation at 4.7 Mw(t), which took place only after November 1960, and the addition of the cadmium strips required further withdrawal of the control rods than had been previously required. Consequently, the drives were being used in a new region of the mechanical structure where closer tolerances or other differences caused increased difficulties with rod motion.

Events Surrounding the Incident

The reactor was shut down for maintenance on December 23, 1960, after having been operated for slightly more than two years. It was planned to bring it to power on January 4, 1961. On January 3, 1961, 44 cobalt flux-measuring wires were inserted into coolant channels between plates of the fuel assemblies. This work required the removal of the control rod drive assemblies. The disassembly and insertion of the cobalt flux-measuring wires had been completed by the day crew on January 3, 1961. The next crew (4:00 p.m. to midnight of January 3, 1961) consisted of the three military personnel involved in the accident. This crew and the crew that followed had been assigned the task of reassembling the control rod drives and preparing the reactor for startup.

The first indication of trouble at the SL-1 reactor was an alarm at the fire and security stations at about 9:01 p.m. (MST), January 3, 1961. Immediate response to the alarm was made by the fire department and security patrol. They called a health physicist who discovered increasing radiation levels as he approached the reactor building. No entry was made until the arrival of the health physicist. He and others located two of the crewman. One crewman who appeared to be alive was removed at about 11:00 p.m. He was pronounced dead almost immediately after he was removed from the building. The other two were dead when first seen by those who entered the reactor room. The radiation level in the reactor room was approximately 500–1000 r/hr. Of the several hundred persons who assisted in the recovery operation, 22 received radiation exposures in the range of 3 to 27 r total-body exposure. No clinical symptoms have been detected.

Analyses of Incident[20,21]

Postmortem examination of the three deceased crewman showed that all three died as a direct or indirect result of blast damage. There was also evidence of flash burns from steam or hot gas on limited areas of the bodies.

Only minor damage was done to the reactor building. The extent of the damage to the reactor pressure vessel is not known, but it is evident that the core has been damaged extensively. The control-rod shrouds have been greatly distorted, core components litter the top of the core, and the core has been expanded from internal pressure that probably reached several hundred pounds per square inch following the initial shock wave, the force of which is estimated to have reached 7000 psi. The core is in contact with the thermal shield near the wall of the reactor vessel at many points on its circumference. The pressure vented through a number of partially closed nozzles in the top of the head of the reactor and blew out shield plugs, portions of control rods, and some fuel. It has been determined that there is little, if any, water in the vessel.

Although it is not known what caused the explosion, there is good evidence that a nuclear excursion took place. The integrated thermal-neutron flux about the reactor was estimated to have been approximately 10^{10} neutrons/cm^2 and may have been from more than a single burst. Furthermore, the hypothesis of an initial chemical reaction which then induced a nuclear reaction by rearrangement of core components is not supported by any evidence to date. In this regard, the Board[21] has been advised that metallurgical examinations made after the accident probably would

not establish conclusively whether a metal-water reaction, if it occurred at all, could have initiated or would have resulted from the nuclear excursion.

Chemical and radioactivity measurements on a single fragment of reactor fuel ejected by the explosion, if representative of the total fuel, suggest that the reaction may have resulted in 1.5×10^{18} fissions. This would have produced 50 Mw-sec of energy. Other estimates, based on decay of gaseous activity and on analogy with SPERT and BORAX experimental results, give a range from 100 to 500 Mw-sec for the total energy release.

It is possible to conceive of several different items or combinations of items which may have constituted the immediate initiating event. The accident could have occurred with no errors being committed on the part of the crew, although certain errors on the part of the operators also can be visualized as possible initiating events. From the positions of the men after the accident and the injuries they suffered, one or two of them are believed to have been working on the control rod mechanism, but the original hypothesis that they were engaged in lifting the central control rod at the time of the accident is now held to be unlikely. This turn of events is based on the reclamation work[25] now in progress, which has revealed that the nut and washer had been installed on the control rod and that the rod extension was "frozen" by the collapsing guide tube in a position in which the rod still should have extended through the core.

On the basis of existing information on the reactivity worth of the central control rod (prior to shutdown) and the results of BORAX and SPERT experiments, it is estimated that this rod, which weighs 80 lb

immersed in water, would need to be withdrawn 22 to 24 in. at a rate of approximately 24 in./sec in order to produce a nuclear excursion of the magnitude estimated to have occurred. While these actions and conditions appear credible, they do not appear probable without other influences.

Additional factors can be considered that involve the possibility that some changes occurred in the properties of the reactor between December 23, 1960 and January 3, 1961, changes that would minimize the capability of the control rod system to maintain the reactor in the shutdown condition. There is no direct evidence at present that any such changes took place. If loss of cadmium or loss of boron occurred, however, during the shutdown period in question, the shutdown margin of reactivity would have been reduced. With a reduced shutdown margin of reactivity, substantially less withdrawal of the central control rod would have produced criticality.

Other conceivable, although at the present unlikely, initiating events include:

1. A water-metal reaction followed by a hydrogen explosion or other chemical reaction below the reactor core that would drive the central rod or several of the rods up out of the core or which would lift the seal plugs and therefore the attached rods by a general pressure increase.

2. Addition of water to the core which had become dry and otherwise changed.

3. A mechanical shifting of core components.

Although the building housing the reactor was not designed for tight containment, in view of its rather remote site, and the accident and the contamination inside the building were unusually severe, the building

nevertheless proved effective in preventing the spread of contamination and obviating any hazard to people outside the building. Some gaseous fission products, including radioactive iodine, escaped to the atmosphere outside the building and were carried downwind in a narrow plume. Particulate fission products and fuel were largely confined to the reactor building. The limitation of the activity release was further aided by the reactor having been shut down for 10 days prior to the accident. Considerable decay of the volatile fission products had occurred. It was a low power reactor and there was only partial loss of the fuel.

Discussion of Control Aspects

The physics of the reactor core affect the control system principally through the control-rod worth and the reactivity-shutdown margin. These two parameters are of course related, although their choice is dictated by somewhat different considerations. The shutdown reactivity margin is determined by the loaded excess reactivity and the total rod worth. The performance of the reactor — its power, or life, or operating cost — is critically dependent on the excess reactivity available. Thus, the designer may be faced with the dilemna of trading performance (excess reactivity) for safety (shutdown margin). The performance is easily measured; the safety is intangible. The shutdown margin of the SL-1 was inadequate.

No universally recognized standard exists that gives a minimum shutdown margin. Schultz[26] suggests a minimum of 2% and a "usual range" of 5—10%. His reactivity data for 21 power reactors show that their shutdown margins vary widely. A recent compilation by Bates[27] for 19 power reactors in the United States shows two with negative margins

that require auxiliary chemical shutdown; for the other 17 the margins vary between 2% and 8.8%. It is difficult to discern any consistent criterion in these scattered values. Bates has suggested[27] the following criterion for minimum shutdown margin: "No single error in manipulation of rods or fuel in the reactor shall cause it to go critical inadvertently." This criterion thus gives the margin in terms of the maximum change in reactivity which can be produced by a single element; for example, the control rod having greatest worth. It must be concluded that the shutdown margin of the SL-1 as designed and built was inadequate according to Bates criterion.

In addition to the lack of an adequate shutdown margin the principal deficiency in the control rod system was that the rod had to be completely disengaged from the mechanism for access to the mechanism or core. It was necessary to first disassemble the pinion drive shaft components in order to remove the thimble. Thus the rod was left without the protection against upward motion which might have been afforded by the auxiliary clutch on the drive motor, and any transient pressure rise in the core could indirectly result in transient upward forces on the rods in the direction of increasing reactivity. It is not often stated explicitly, even in reactor hazards documents, that the thimbles are essential to the safety of rod drives of this type. This reactor was unknowingly placed in a condition more hazardous than normal every time the thimbles were removed. In other reactors using this type of drive, it would seem prudent to devise operating procedures to minimize the added hazard present when the thimbles must be removed.

The most obvious and most objectionable feature of the SL-1 rod drive system is, of course, the necessity for lifting the rod by hand and holding

it with a "C" clamp while coupling or uncoupling the rod and the drive mechanism. This feature is not essential; other rod drives of this type lack it, and better so. Hand lifting violates several elementary safety criteria:

1. The rod cannot be dropped by the safety system.
2. Rod withdrawal is not supervised by the control system.
3. Rod withdrawal speed and amount are not limited or controlled.
4. Rod position is unknown to the console operator and can only be estimated by the man doing the lifting who is unaware of the readings of the reactor instruments.

Of course, control rods must be removed and replaced because of burnup, but such operations should be carefully programmed and made safe, as necessary, by removing fuel from the core. The routine, casual, and frequent necessity for hand lifting of the SL-1 control elements was an unnecessary danger.

From an operational point of view the decision was unsound to continue operation even though the control rods frequently did not insert on a scram and the core was increasing in reactivity by a mechanism not fully understood and therefore at an unknown rate. It is admittedly difficult to establish criteria for all aspects of reactor safety, but in the case of loss of shutdown ability during operation the rules of the art are straightforward. The safety system is provided to assure the safety of the reactor, and it provides the basis upon which the safety of the reactor is, in part at least, assessed. Any impairment of this system's ability, such as the failure of one or more rods to drop on a scram, or a significant decrease in shutdown margin, is in the direction of danger and the reactor should

be shut down. Before operation should be allowed to resume, a safety review should favorably assess not only the safety of the reactor in its existing condition but also in the event of further losses in performance capability.

Conclusions

The conclusions[21] of the Board of Investigation were supplemented and interpreted by the Chairman of the Board, as well as other responsible members of the Commission, in testimony before the Joint Committee on Atomic Energy in June. Testimony of each of the following will be followed by his last name: Curtis A. Nelson, Chairman of the Board of Investigation; AEC Commissioner Robert E. Wilson; and AEC General Manager A. R. Luedecke.

> 1. Design Features. "A design feature which should be avoided ... is to have the withdrawal of a single control rod cause the reactor to go super-critical. If this feature is deemed unavoidable in a particular design, there must be positive controls not only to prevent its happening in operation ... but to make it impossible in carrying out maintenance work. The failure to make such provisions was a design deficiency in the SL-1." (Wilson)
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> 2. Control Rod Malfunctions. "Where there are repeated malfunctions of a control system such as sticking of control rods, ... the reactor should remain shut down until effective remedial steps have been taken. The decision to continue operation of the SL-1 under such conditions was made at too low a level and without a thorough hazards review by qualified AEC personnel." (Wilson)
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> 3. Reactivity of Core. "Where there are ... obvious structural defects in the core components or a substantial loss of 'worth' in the control system, the reactor should remain shut down until effective remedial steps have been taken. (Wilson) ... It [was] known that progressive deterioration of the SL-1 core ... had taken place. (Luedecke) ... The formal

recommendation of a report on the loss of boron from the reactor core, after intensive review of the problem, was to terminate a previously established inspection routine of the fuel elements and to continue to operate the reactor. It is conceivable that continued inspection of the fuel elements could have led to additional knowledge which would have affected the decision to continue to operate ...."(Nelson)

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4. Safety Reviews. "Deficiencies in the safety procedures and practices of the AEC and contractor organization ... have been found which, if corrected, might have prevented the accident .... I believe that a timely and comprehensive safety reactor review by the contractor would have resulted in an order shutting down the reactor for review and corrective action prior to the incident. (Luedecke) ... Every reactor, licensed or Commission-operated, should have periodic safety reviews by an independent group. The contractor did not make such a review, subsequent to its initial survey in February 1959. The Idaho Operations Office made periodic reviews of fire, radiation hazards, and industrial safety, but did not conduct special reviews of nuclear safety of the SL-1 reactor." (Wilson)

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5. Written Procedures. "Operating personnel were not required to follow a detailed written procedure ...for the disassembly-assembly of the control rod mechanism. (Nelson) ... Even where trained men conduct so-called routine maintenance operations on reactor cores, there should be detailed written procedures to be followed." (Wilson)

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6. Instrumentation. "In order to increase the safety of reactor maintenance operations there should be an operator in the control room and appropriate instruments connected whenever any work is being done on the reactor internals. Had this been the practice at the SL-1, it might have prevented the accident and in any case would have furnished important diagnostic information as to its causes." (Wilson)

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7. Inspection. "Adequate and timely inspection by the AEC would also have revealed conditions which might have caused the reactor to be shut down. (Luedecke) ... The Inspection Division failed to bring this fact [i.e., lack of adequate safety reviews] to the attention of the Reactor Development Division or the General Manager. [This failure was] in part the result of the complex organization." (Wilson)

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8. Operating Organization. "The training and ability of the operating organization appears not to have been entirely adequate, since substandard conditions were allowed to develop in the reactor and its components and yet, reactor operation was allowed to continue. The complexity of the chain-of-command for the SL-1 may explain, in part, the lack of effectiveness of the existing organization in communicating with higher levels of supervision regarding these substandard conditions." (Nelson)

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9. Responsibility. "The lines of responsibility within the AEC for health and safety, from the General Manager down to the operators of the reactors, were not as clear and definite in several respects as they should have been, nor were the levels at which certain safety and operating decisions should be made, spelled out. (Wilson) ... The immediate responsibility for the SL-1 incident ... [however] was that of the contractor, in that the contractor was on site and had immediate responsibility for all reactor operations ... Responsibility for the performance of the contractor is that of the AEC Idaho Operations Office Manager and his staff." (Nelson)

In addition to the above, the Board of Investigation also commented upon actions occurring after the SL-1 accident.

"The recovery operation and the investigatory actions might have been more effective, and more expeditiously carried out had the emergency planning been more extensive .... Examples of what might have been improvements [include] the following:

a) Appropriate choice and placement of suitable incident monitors [in addition to the one present] might have clearly indicated very soon after the incident the nature and extent of the incident.

b) Clearly assigned, and continuing, responsibilities of a 'disaster team' might have improved the execution of early attempts to obtain significant data concerning short-lived activities of various samples."

(An article on the value of analytical procedures in nuclear incidents was presented in an earlier issue of Nuclear Safety[28])

Another suggestion along this line made by the Board was that the "performance of the Board of Investigation, itself, may have been improved

had its organization and assignment been specifically pre-established and described by appropriate AEC procedure." (Wm. B. Cottrell and S. H. Hanauer)

COMMENTS ON PROPOSED REACTOR SITE CRITERIA

Proposed criteria for use as guides in the selection of reactor sites were published in the Federal Register on February 11, 1961, and comments or suggestions addressed to the Division of Licensing and Regulations of the AEC were invited from interested parties. The criteria and previous siting developments were reviewed in a previous issue of Nuclear Safety.[29,30] Even if the proposed criteria were adopted in anything like the present form, the judgment of the designer, the AEC regulatory staff, and the Advisory Committee on Reactor Safeguards (ACRS) remains vital to the selection of a site, since, as acknowledged by AEC, site criteria cannot yet be stated precisely.

Comments were received from 30 sources, including ACRS, the Atomic Industrial Forum, and foreign, as well as American, groups. The Committee on Reactor Safety of the Atomic Industrial Forum on March 17, 1961, discussed the criteria in a special Forum seminar participated in by 17 of its members.[31] In addition to comments, the Forum submitted,[32] as did Atomics International,[33] a proposed complete revision of the criteria. In the discussion which follows selected quotations are presented that were taken from many of the comments in an attempt to indicate both their range and depth. Although it is intended that the original meaning of the comments be maintained, all are, of course, presented here out of context, and persons interested in greater detail should refer to the original comments. It is anticipated that the Division of Licensing and Regulations will embody many of the comments and suggestions into a revision of the criteria which will then be issued in the Federal Register.

General Comments

For the most part the groups who commented welcomed the criteria, since they saw the need for guide lines for the industry. There was, however, much fear expressed that the criteria were too restrictive and that once they were set forth in the Federal Register as regulations and were fixed in the minds of the public and authorities it would be difficult to make site decisions counter to the criteria even though such decisions might be warranted by the particular designs. The use of the calculated example in Appendix A for a serious accident of a hypothetical reactor was a cause of concern. It was felt that too much emphasis was placed on distances to populated centers to the detriment of the engineering safeguards which can be developed. A sampling of the general comments follows:

> "... use of the term 'guides' may lead to some ambiguity as to their intent and purpose ... the proposed 'guides' had been identified by AEC as ... (part) of the Code of Federal Regulation ....
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> "There was an obvious difference of opinion among seminar participants as to the intended purpose of Appendix 'A.'...
>
> "... the guides may, thru inadverted phraseology, have a possible adverse impact on public opinion even though their intent might be clearly understood by reactor licensees and others in the atomic industry."[31]
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> "We believe that the guides outlined in the proposed criteria are most desirable for preliminary evaluation of potential reactor sites....
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> "An effort should be made to establish the trade-offs which can acceptably be effected between engineering safeguards incorporated in the reactor system and the various distances specified in the present Guides."[34]
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> "To my knowledge, there is no definite requirement by the Commission which would prevent an industrial park or surburban housing development from springing up around a reactor, once

it was licensed, and effectively abolishing the 'low population zone' so carefully calculated as a requirement for issuing the license."[35]

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"It is apparent throughout the proposed guide that it was written for a specific class of reactors and a specific type of containment. This is unfortunate."[36]

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"Objective (b) of the Proposed Guides is not clear because the phrases 'not normally considered credible' and 'the number of people killed should not be catastrophic' are subject to a considerable range of subjective interpretation. In addition, the latter could provoke public alarm without need....

"It is recommended, therefore, that the basic objective in the establishment and use of the Proposed Guides be stated as:

> 'Serious injury to individuals offsite should be avoided and the exposure of large numbers of people in terms of total integrated population dose should be low, if any unlikely, but still credible, accident should occur.'"[33]

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"... the document recognizes tacitly that the ACRS and the AEC as a whole are not as yet completely willing to trade distance for engineering design. This is a reasonable position at this stage in the development of the atomic energy industry and in view of the projected future population growth.

"The intent and desire of the ACRS is that this be a very flexible document and that deviations from this document should be allowed in cases where an applicant can show that his design or his siting of the reactor leads to a safe situation. In fact, the ACRS would like to encourage applicants to come in with deviations in those cases in which they believe that a valid reason for deviation exists."[37]

Specific Comments on Guide

Concern was expressed about some special cases in which application of the proposed concept of a "population center distance" would practically exclude power reactors from areas where most authorities feel they will be utilized. Japan feels that its regulation are likely to evolve from the United States regulations, and, in Japan, it would almost be impossible

to find power reactor sites based on the proposed regulations because of the high population density. It was also pointed out by several sources that the regulations created an awkward situation for mobile reactor power plants, such as that of the N.S. Savannah. The situation with respect to process steam reactors would also be clouded. Some of the comments on these special cases follow:

> "The proposed guides on reactor site criteria published by the United States Atomic Energy Commission on February 11 this year [have] brought about a great deal of repercussion among the related quarters of the Japanese industry. We consider that it is now a sort of world wide common sense that in the field of nuclear energy even domestic legislations cannot be made without influencing or being influenced by the situation abroad."[38]
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> "Should these figures of exclusion area, low population zone and population center distance as appearing in the Appendix 'A' be adopted for granted, it would eventually make it practically impossible to find a site in Japan for installing any U. S. type reactor."[39]
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> "I would like to note that the Commission's efforts to develop industrial process steam reactors also appear to raise questions about the consistency of AEC's proposed guides on site criteria. As you know, it is not economically practical to transmit steam for long distances. In most cases, it is necessary that the industrial plants using process steam be located closely around the steam-producing plant. The Commission's distance-from-population criteria for reactors seem to us incompatible with the Commission's expenditures of time and money for the development of process steam reactors....
>
> "There seems to be another inconsistency, as far as remoteness from population is concerned, in the Commission attitude in respect to nuclear powered ships.... I recognize that the [N.S.] 'Savannah' has been designed and built with special features to enhance the inherent safety of the power plant..."[35]

Comments on Multiple Reactor Sites

The question of how to apply the criteria to multiple reactors was introduced. Some of the comments were:

> "Is the power level to be used for determining distances for population the total power level of all nuclear units on site or only the power level of the largest unit?"[40]
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> . . . . . .
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> "Where more than one reactor is located at a particular site, the thermal rating used in hazards calculations pertaining to the total installation should lie somewhere between the thermal rating of the largest single reactor and the sum of all reactors present. If there is negligible possibility that an accident in one reactor could cause an accident in another at the same site, then the thermal rating of the largest single reactor should be the value used in hazards calculations."[34]

Comments on Evaluation Factors

There were several suggestions that the emphasis on distance to population centers as a factor used in evaluating reactor sites should be replaced by an emphasis on the safety of the engineering design. Geological and hydrological factors were discussed in relation to the precautions to be taken to prevent radioactive effluents from running into rivers or reaching underground water tables. Criticism and questions pertaining to the maximum emergency population radiation exposure dose were included among the remarks. Some of the comments were:

> "We view with concern the Site Evaluation Factors.... This approach appears to minimize the possible contributions to safety by advances in design, and to maximize the value of isolation. We believe this approach unrealistically undervalues scientific and engineering skill, both present and future."[41]
>
> . . . . . .
>
> "One fundamental point is that the contribution which site considerations can make to the safety of the public is relatively small, when compared to the contribution made by engineering barriers to the release of fission products....
>
> "We regard the proposed population center distance factor as technically unjustified, and recommend the development of a rating system which factors in meteorology and other environmental factors, in addition to distance. Such an approach provides a much greater degree of assurance that population center distance will make a significant contribution to public safety."[39]
>
> . . . . . .

"It seems very unlikely that future inland sites can be found which have water supplies which are not used at some downstream point.... We would suggest 'The plant design must provide reasonable assurance that radioactive liquid effluents cannot accidentally contaminate usable water supplies and prevent use of the water for long time periods. The provisions to prevent accidental releases as well as the importance of the potentially contaminated water must be considered.'...

"in our opinion, some method of evaluating the risk of exposing large numbers of persons should be developed taking into account the integrated man-rem exposure potential based on the proposed plant including its design features, the meteorology and the size, distance and direction of areas of substantial population."[42]

. . . . . .

"We do not question the 25-rem whole-body exposure, but we do believe that the 300-rem thyroid exposure is too high. The highest value iodine exposure equivalent that we could justify at the present time would be 150 rem.... We suspect that the higher thyroid values were included in the proposed criteria on the basis that exposed persons may be treated in such a fashion as to reduce their thyroid exposure by a factor of 2. However, the use of the higher value would be justified for a particular reactor only if that installation were prepared to, and were capable of, rounding up all the persons which had been evacuated from the low-population zone and getting them to submit to the required treatment. This does not appear to be either desirable or practical."[36]

. . . . . .

"The most important single new development which is embodied in this document is the fact that it contained a set of dosage limits which can be used in the event of a very unlikely reactor accident....

"The site criteria, as published, do not completely define or take care of the problem of either gentic or somatic damage. The problem is recognized and stated in the document, but answers to this problem do not as yet exist.... The Committee [ACRS] believes that the so-called 'man-rem' dose has much to offer as a safeguard for the general protection of the population. We went so far, in December of 1960, as to suggest to the Commission that a dose limit of 4×10^6 man-rems per accident might be reasonable."[37]

Comments on Appendix A

The example in Appendix A which gave calculations for possible reactor siting distances was the most controversial portion of the

proposed regulation. Some of the comments received with respect to the example follow:

> "... the AEC's Division of Licensing and Regulation has clearly demonstrated a tendency on the part of AEC to take the most conservative of alternative interpretations permitted by any rule. Hence, unless there is an unequivocal intent on the part of the AEC not to permit variance from the assumptions used in Appendix 'A' it should not be included as an integral part of the guides."[31]
>
> . . . . . .
>
> "... we also believe that the example at the end should be treated simply as an example and, perhaps, should be either deleted or augmented by the addition of several other examples."[37]
>
> . . . . . .
>
> "The proposed criterion tabulates conversion factors which were used to determine the dose received from breathing a specified quantity of the various iodine isotopes. We believe that the actual conversion factors are appreciably higher and suspect that the lower values in the guide were calculated from an earlier inaccurate value of the uptake of iodine by the thyroid as well as a lower breathing range."[36]
>
> . . . . . .
>
> "We do not agree with the general applicability or the technical validity of the proposed Appendix 'A' and urge that the Appendix be deleted."[39] (J. R. Buchanan)

CHANGES IN LICENSING REGULATIONS

Parts of the AEC rules and regulations, found in Title 10, Code of Federal Regulations, Chapter 1, are frequently amended. Those now being considered for change are discussed here.

Part 2 – Rules of Practice

A proposed amendment to the rules of practice would require that, in requesting reactor licenses and waste disposal licenses, the applicants serve copies of their applications on the chief executive officer of the town or country in which their facility will be located. The requirement would apply to power demonstration reactors constructed under contract with and the account of AEC, as well as to licensed reactors.[43]

Part 30 – Licensing of By-Product Material

Requirements for exemption of tritium from licensing for use in automobile lock illumination were set forth in the Federal Register of April 26, 1961.[44] Both favorable and unfavorable comments have been received since then. Persons in favor usually felt that the exemption should also be extended to other users of similar sources.[45-49] Comments against included statements that (1) an unwarranted precedent would be set and (2) that novelty uses of radioactive materials were being developed when convenient nonradioactive alternates are available.

Part 50 – Licensing of Production and Utilization Facilities

Change Without Specific Authorization. Proposed amendments to the change procedure rules were published[50] in the Federal Register of April 8, 1961. Comments indicated that the general intent of the changes was good,

but it was felt, in general, that there should be additional liberalizations in the rules. The requirement of prompt filing with AEC of a report of every change, test, or experiment was felt to be an unnecessary burden. It was suggested that the AEC technical staff should be able to have ACRS review its judgment without being required to submit the matter to a public hearing. It was felt that operators should be allowed to submit proposed specifications if their licenses did not include such, rather than having the entire hazards summary report taken to be the specifications.[45,50-53]

Special Nuclear Material Processing Facilities. An amendment published in the Federal Register of June 6 would eliminate the requirement of obtaining a production facility license if the material processed by a facility contained not more than 10^{-6} g of plutonium per gram of U^{235}. This change will relieve the Englehard Industries, Inc., from the requirement of obtaining a production license in order to process some fuel and control elements from a Netherlands research reactor.[43]

Part 100 – Site Criteria Guide

The comments on this proposed rule are reviewed elsewhere in this Review (p.).

Part 140 – Financial Protection Requirements and Indemnity Agreements

On March 29, the AEC invited comments on the question of whether processors of unirradiated uranium enriched in U^{235} (not U^{233} or Pu^{239}) should be indemnified and required to furnish financial protection.[50] Since then, many comments have been received. Comments from the nuclear industry have been heartily in favor of the extension, while the insurance

industry has been wholly against it. The insurance companies feel that this is government encroachment into an area in which they are willing and able to serve.[44-47,54] (J. R. Buchanan)

ACTION ON REACTOR PROJECTS BY LICENSING AND REGULATING BODIES

The AEC is required by the Atomic Energy Act to protect the health and safety of the public from undue hazards relative to the operation of nuclear facilities. Regulations which are followed to ensure that this responsibility is fulfilled while furthering the simultaneous responsibility for developing the use of nuclear energy are published in the "Code of Federal Regulations." Title 10 of the Code requires the AEC to make certain specific findings regarding the safety of the public before issuing either the construction permit or the operating license for a facility. It also requires authorization for changes in facility equipment or operation which contain an element of hazard not previously reviewed and approved. The license application record of the various facilities is reported in Table ___ [this table to be prepared after draft is issued].

Early in June 1961, the AEC sent questionnaires to reactor licensees on gaseous effluent monitoring and control methods to obtain appraisals of their respective practices.[55] The objective of this study is to evaluate: "(1) The capability and accuracy of instruments used to measure selected radionuclides in gaseous effluents; (2) the possibility that significant concentrations of other nuclides may escape detection; and (3) the accuracy of the determination of the character and quantities of nuclides actually released." The results of this survey will be reviewed when they become available.

Recent actions on specific reactors are described below:

Dresden Reactor (Docket 50-10)

On June 5, the Dresden Reactor, a 630-Mw(t) boiling-water-power reactor, again started supplying power to the Commonwealth Edison system and was reported to be operating satisfactorily at a power level of 150 Mw(e).[56] The reactor had been shut down on November 15, 1960, because of a control rod separation.[57,58]

Following the shutdown in November, the AEC issued, on February 28, 1961, notice of a hearing to consider whether it should authorize the licensee to resume operation with modified control rod drive mechanisms and whether it concurred in the proposed test program. Subsequently, the licensee filed an application for license amendment to authorize substitution of poison blades of new design, following the observation of cracking of boron-steel blades during modification of the control rod drive. After hearings were held on these points, the AEC granted an interim order on March 31 for limited operation at powers up to 100 kw(t) while review of the new poison blades continued. However, this operation was held up after some cracks were discovered in the reactor core grid structure.[59]

Hearings the last week in April received testimony (from Robert H. Bryant, AEC Division of Licensing and Regulation) which concluded that the grid defects did not significantly impair the immediate usefulness of the structure. Although there was a chance that the defects would increase in number and severity, it was felt that the increase would be gradual. Hence, it was felt that the plant could be safely operated if there were adequate surveillance of the structure. It was concluded that

the probabilities of the fallout of a blade should be made as low as possible by "following determinations" made as early as possible during the course of a startup. Blade movements should be observed by nuclear instrumentation during other operations with the reactor critical.[60]

On May 27, Hearing Examiner Jensch issued a Fourth Supplemental Intermediate Decision which recommended issuance of a license allowing Dresden to resume operation at power levels up to 630 Mw(t). It was held that operation of the reactor with the modified drives could be resumed without endangering the health and safety of the public, subject to considerations involving inspections, tests, and operating procedures. In addition to periodic inspection of the control rod drive mechanisms, the poison blades, and the grid structure, Commonwealth Edison will test its container and mainstream shutoff valves. The containment tests would be made before October 15, 1961, with followup tests to be made at subsequent intervals. Fourteen new fuel assemblies were approved for the reactor subject to the present license restrictions on burnout margin.[61] The new requirements and specification changes were authorized in an amendment to Appendix A of Edison's license due to expire on May 4, 1996.

During startup operations, Dresden found that one of the nine drive rods equipped with thermocouples indicated a somewhat higher temperature than the other eight. On May 31, the approach to power was delayed so that thermocouples could be installed on the remaining drives. While exercising the one drive during the installation period, the condition apparently cleared. None of the drives were equipped with thermocouples during the previous operations, but there is now one thermocouple on each drive.[56]

Elk River Reactor (Docket PP-1)

Specific information was requested by the Division of Licensing and Regulation regarding cracks in the pressure vessel cladding of the 58-Mw boiling-water Elk River reactor.[59] Allis-Chalmers submitted a written report[62] to confirm oral information given to the AEC on June 2. It was judged that the cracks in the cladding would not have any deleterious effects on base material corrosion, strength of the vessel, or in the radioactivity levels in the primary loop. It was felt that the cracks which appeared after the vessel was pressure tested were caused by release of locked-in stresses. Further crack formation under operating conditions is not expected.[63,64]

Allis-Chalmers also filed a report[65] to update an earlier report of May 4 on the 17-4 PH steel in the reactor. A report on control rod blades was also submitted on May 4.[66]

Enrico Fermi Atomic Power Plant (Docket F-16)

The Supreme Court, on June 12, reversed the judgment of the lower court and upheld the legality of the AEC's order which granted a provisional construction permit to the 300-Mw(t) Enrico Fermi fast-breeder reactor.[59] On June 10, the Power Reactor Development Company (PRDC) requested that the construction permit and its license application show December 15, 1961, as completion date for the reactor, rather than July 15, 1961.[67] The extension was granted on July 14 by the Licensing and Regulation Division of AEC.[68]

On July 13, PRDC filed portions of a revised application for an operating license for the reactor. Included in the application were six volumes of a seven-volume hazards report.[68]

FWCNG Reactor (Docket 50-152)

As indicated previously,[59] work on a proposed 153-Mw(t) D_2O-moderated gas-cooled reactor was terminated in early May 1961 by the AEC. On July 12, FWCNG was advised that actions were being terminated on their pending construction license application per their request of June 26.[67]

Indian Point Reactor (Docket 50-3)

A hearing was held early in May 1961 on the final design of the 585-Mw(t) pressurized-water thorium-converter Indian Point reactor in which Edison sought approval relating only to design and construction.[59] In an intermediate decision issued on June 30, Examiner Bond recommended approval of the design. The "ultimate conclusion" of the Examiner was made subject to a condition that the applicant present satisfactory evidence on the metallurgical characteristics of the control rod drive shafts.[69]

On May 15 Edison had their counselors, LeBoeuf, Lamb, and Leiby, file an amendment (No. 4) to the company's amended and substituted license application, presenting as exhibits operating procedures and preoperating and startup test procedures.[70] On July 18, they submitted a tentative regulation schedule that will lead to an anticipated issuance of an initial operating license. The schedule points to a September date for a meeting with ACRS to discuss all matters pertaining to an operating license and completion of plant construction.[71]

Nuclear Merchant Ship N.S. Savannah

The AEC issued in late July 1961 an authorization for the startup and operation of the 69-Mw(t) pressurized-water reactor of the N.S. Savannah through initial sea trials.[72] During March and April, public

hearings were held concerning fueling and operation of the reactor for test and demonstration purposes.[58,59] The AEC staff filed its "Proposed Findings of Fact" accompanied by a "Memorandum for AEC Regulatory Staff" recommending that AEC authorize operation as proposed.[73]

The AEC authorization incorporated the findings and conclusions from the favorable recommendation of Hearing Examiner Jensch[74] but modified it as proposed in part by the Licensing and Regulation staff in a memo of June 29. The conditions of authorization seek to assure that detailed written operating and maintenance procedures for each phase of the reactor testing program will be prepared before commencement of that phase. Startup tests at Camden, New Jersey, are limited to no more than 7 Mw. After completion of tests of Yorktown, Virginia, and initial sea trials, the Licensing and Regulation Division and the ACRS will review all reports on test operations and set down their recommendations in a public hearing before reactor tests are continued or the ship is returned to Camden while pressurized.

Piqua Nuclear Power Facility (Docket PP-2)

The Piqua reactor is a 46-Mw(t) heterogeneous organic-cooled and -moderated reactor being built at Piqua, Ohio.[75] On May 18–20, 1961, the ACRS met to consider the final safeguards report, a supplement, and a proposal for operation submitted by Atomics International. Earlier (on April 20) an ACRS subcommittee had met on the reactor site with representatives of Atomics International and the AEC staff. Based upon information presented and discussed, ACRS felt that the reactor could be operated by Atomics International without undue hazard to the health and safety of

the public. ACRS suggested that at each partial fuel reloading cycle (approximately four- to six-month intervals of power operation) that an inspection of a typical sample portion of the remaining fuel elements be made and that a report be forwarded to the Division of Licensing and Regulation.[76]

Plum Brook (NASA) Reactor (Docket 50-30)

A provisional operating license was issued on March 14 for the 60-Mw(t) light-water-cooled and -moderated Plum Brook Reactor of NASA.[59] Authorization was issued on June 12 for initial core loading. This action followed the finding by the Division of Licensing and Regulation that systems or items needed for full-power operation (but not requiring completion for low-power testing) were built and tested and that modifications required for 100-kw operation were complete.[77]

On June 29, NASA advised of two changes in the reactor. One was a redesigned grid-lifting mechanism and the other was a modification to the manual control section of the regulating rod amplifier.[78]

Radiation Effects Reactor (Docket 50-122)

The Lockheed Aircraft Corporation on June 8 advised the Division of Licensing and Regulation that it agreed to withdrawal of its application for an operating license for the 10-Mw(t) pressurized-water radiation effects reactor but that it anticipated a resubmission at some future date.[79] ACRS had recommended in December 1960 and March 1961 that the reactor be operated at full power in its present form only for work essential to the national defense.[58,59]

Saxton Reactor (Docket 50-146)

The ACRS issued a favorable report on the Saxton 20-Mw pressurized-water reactor based on documents considered in its meeting of July 6–8. They noted that there was the possibility that complete withdrawal of one single control rod could make the reactor critical but that a system involving manually set limit switches is provided to prevent excessive rod withdrawal and to provide an adequate shutdown margin. The committee concluded that the reactor could be operated through its startup without undue hazard to the health and safety of the public.[80]

The Saxton Nuclear Experiment Corporation on June 9 in its sixth license application amendment listed proposed technical specifications, proposed a definition of the Saxton facility, and requested that discussion of Saxton's technical and financial qualifications be considered in a future September public hearing rather than the July hearing.[81]

Vallecitos Boiling-Water Reactor (Docket 50-18)

Permission to resume operation of the Vallecitos boiling-water reactor was granted by the AEC on April 13, 1961, under conditions providing for control rod testing, inspection, and behavior study.[59] The operator, General Electric Company, concurred with the conditions with slight modifications.[82] On May 22, General Electric in an "overdue followup" reported on the Division of Licensing and Regulation's concern over General Electric's apparent failure to report cracks found in one of the BVWR boron–stainless steel control rods during a routine inspection of March 10. General Electric reported that there had been no intent to withhold information since there had been a voluntary report of the

incident to a San Francisco Operations Office inspector on March 15. It was felt by General Electric that the cracks did not constitute a "possible unsafe condition" since structural stainless steel backs up the boron-stainless steel. On May 16, General Electric sought permission to test a new control blade material by fastening it to a standard blade to test its long-time performance in the reactor.[83]

On June 6, General Electric filed with the Division of Licensing and Regulation reports of four proposed changes or experiments. Three had to do with irradiation experiments; the fourth was on a steam-water separator being installed in the reactor pressure vessel to reduce entrainment of water in the effluent steam.[84] On May 4, General Electric reported on a proposal to irradiate a fuel assembly prepared from rejected Savannah Core II stainless steel tubing to study the effects of microscopic leaks and irradiation on the tubing.[82] The Division of Licensing and Regulation on June 15 expressed concern over the fuel assembly and requested a reappraisal with additional information.[85] General Electric submitted an amendment (No. 46) to VBWR's application with respect to a loop called E-SADE on June 26.[86] On July 12, General Electric proposed an experiment to study the causes of a leak in one of the VBWR steam generator tubes.[87]

Vallecitos Experimental Superheat Reactor (Docket 50-183)

The AEC's Chief Hearing Examiner Jensch in an intermediate decision recommended on July 25 that a construction permit be issued for the 12.5-Mw(t) Vallecitos experimental superheat reactor proposed by General Electric.[88] The case had been heard on June 29 when testimony was given

by four representatives of General Electric and AEC. Booth of the Division of Licensing and Regulation had testified that General Electric had "supplied sufficient information to provide reasonable assurance that the facility can be constructed and operated without undue risk to the health and safety of the public."[89]

ACRS has considered the reactor in two full committee meetings and two subcommittee meetings. In a report issued after the latest meeting (May 18–20), the Committee concurred in the desirability of the safety rods proposed by General Electric in addition to nuclear poison rods. The ACRS listed six items which they would like to consider at a later date.[76]

On June 19, AEC's Division of Licensing and Regulation held an informal public meeting at Pleasanton, California, to discuss their regulatory program and its review of VESR in order to give members of the public living in the area a chance to ask questions. Whether or not this practice will be extended to other reactors is not known.[90]

Westinghouse Test Reactor (Docket 50-22)

Westinghouse on April 25 requested an amendment of its 60-Mw(t) Westinghouse Test Reactor license to authorize them to lower the head tank-to-vent effluent discharge ratio by a factor of 2 while increasing the monitor alarm setting by the same factor.[91] The Division of Licensing and Regulation authorized the change on May 17.[92] Westinghouse submitted results on May 18 of a sampling program on the effluent.[93] On July 14, Westinghouse submitted a report requesting authority to modify a high-pressure test thimble.[94]

Yankee Reactor (Docket 50-29)

In a license amendment dated June 23, AEC authorized the Yankee Pressurized-Water Reactor to operate at power levels up to 485 Mw(t).[95] Yankee's provisional license was also converted to a 40-year operating license. Hearing Examiner Jensch authorized the amendment in a decision dated June 20,[96] following a public hearing on June 8.[97] Both ACRS and the Licensing and Regulation staff concluded that the reactor which had been operating at 392 Mw(t) satisfactorily could be operated at the higher power level without undue hazard to the health and safety of the public. The Division of Licensing and Regulation approved several minor modifications to the plant and specifications designed to improve performance and to correspond with changes. (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. R. J. Smith, A Literature Search, Reactor Safety, USAEC Report TID-3525, Rev. 3, July 1961.
2. J. Coombe et al., Hazards Report for the SM-1 Core II with Special Components, USAEC Report APAE-84, March 30, 1961.
3. J. G. Gallagher, Hazards Report for the SM-1 Core II Without Special Components, USAEC Report APAE-84, Add. 1, April 19, 1961.
4. N. G. Wittenbrock and J. Muraoka, Plutonium Recycle Test Reactor Final Safeguards Analysis, Supplement 2. Consequences of a Primary Coolant Leak, USAEC Report HW-61236, Suppl. 2, November 15, 1960.
5. R. W. Clack et al., Hazards Summary Report for the Kansas State University TRIGA Mark II Reactor. Special Report No. 7. Section I – Kansas State University TRIGA Mark II Reactor Hazards Summary Report; Section II – Kansas State University TRIGA Mark II Reactor Operations Manual; Section III – Kansas State University TRIGA Mark II Reactor Administrative Plan, USAEC Report NP-10074, January 1961.
6. Final Hazards Summary Report for the UCNC Research Reactor, Union Carbide Nuclear Company, Research Center, Tuxedo, New York, USAEC Report NP-9674, November 1960.
7. D. Hunter, J. Jedruch, and P. W. Davison, Yankee Critical Experiments Hazards Summary Report, USAEC Report YAEC-31, Suppl. 2, July 1, 1958.

8. J. Carothers, Hazards Summary Report for the LRL Critical Facility, USAEC Report UCRL-6220, March 3, 1960.

9. J. L. Murray, General Electric Vallecitos Boiling Water Reactor. Final Hazards Summary Report, USAEC Report SG-VAL-2, 3rd ed., November 30, 1959.

10. Hazards Summary Report for Diamond Ordnance Radiation Facility, General Atomic Division, General Dynamics Corporation, San Diego, California, USAEC Report NP-10188 (1960).

11. E. Feinauer and R. D. Thomas, Safeguards Report for the Northrop Pulse Radiation Facility, USAEC Report NP-9978, March 22, 1961.

12. A. H. Emmons and W. R. Pearce, Preliminary Hazards Report, Exhibit A, Class 104 Facility License Application with Exhibits B, C, and D, USAEC Report NP-9951, March 1961.

13. P. M. Finnegan, Preliminary Hazards Summary, Mock-Up Reactor, NASA Plum Brook Reactor Facility, USAEC Report NP-9947, January 1961.

14. P. D. O'Brien, Hazards Evaluation of the Sandia Pulsed Reactor Facility (SPRF), USAEC Report SC-4357A (RR), February 1961.

15. C. L. Blackshaw and C. H. Skeen, Safeguards Summary for the AE-6 Reactor, USAEC Report NAA-SR-5499 [no date].

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1. Nuclear Reactors Built, Building, or Planned in the United States as of June 30, 1959, USAEC Report TID-8200.

2. F. W. Thalgott, Fast Reactor Safety; EBR-1, Proceedings of the 1957 Fast Reactor Information Meeting, Chicago, Ill., Nov. 20–21, 1957, pp. 68–76.

3. H. V. Lichtenberger, F. W. Thalgott, W. Y. Kato, and M. Novick, Operating Experience and Experimental Results Obtained from an NaK-Cooled Fast Reactor, Proceedings of the United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1955, Vol. 3, pp. 345–60, United Nations, New York, 1955.

4. Argonne National Laboratory, unpublished report, April 15, 1961.

5. W. H. Zinn, Basic Problems in Central Station Nuclear Power, Nucleonics, 10(9): 8–14 (Sept. 1952).

6. T. Trocki, et al, Sodium and Sodium-Potassium Alloy for Reactor Cooling and Steam Generation, Proceedings of United Nations International Conference on the Peaceful Uses of Atomic Energy, Geneva, 1955, Vol. 9, pp. 241–51, United Nations, New York, 1955.

7. W. E. Unbehaun, History and Status of the EBR, USAEC Report AECD-3712, April 15, 1953.

8. J. H. Kittel and S. H. Paine, Effects of High Burnup on Natural Uranium, USAEC Report ANL-5539, Argonne National Laboratory, May 1957.

9. W. H. Zinn, Letter, Nucleonics, 14(6): 35 (1956).

10. J. H. Kittel, M. Novick, and R. F. Buchanan, The EBR-1 Meltdown – Physical and Metallurgical Changes in the Core, USAEC Report ANL-5731, Argonne National Laboratory, November 1957.

11. F. L. Yaggee, The Manufacture of Centrifugally Cast Reactor Fuel Slugs, Mark II Fuel Loading for EBR-1, USAEC Report ANL-5724, Argonne National Laboratory (no date given).

12. Argonne National Laboratory Annual Report, 1957, USAEC Report ANL-5870.

13. Argonne National Laboratory Annual Report, 1958, USAEC Report ANL-5980.

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

17. R. C. Paulus et al., Nuclear Incident at the Idaho Chemical Processing Plant – January 25, 1961, USAEC Report, IDO-10036, June 1, 1961.

18. Activity Releases at Oak Ridge National Laboratory, Nuclear Safety, 1(3): 77–80 (March 1960), see esp. Fig. 5, p. 78.

19. Personal telephone communication, K. K. Kennedy to J. P. Nichols, July 27, 1961.

20. C. A. Nelson, Chairman, Report on the SL-1 Incident, January 3, 1961 (by the General Manager's Board of Investigation), May 10, 1961.

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30. Changes in Licensing Regulations, Nuclear Safety, 2(4): 63–64 (June 1961).

31. Atomic Industrial Forum, Inc., Summary of Seminar on Reactor Site Criteria – Proposed Guides, March 17, 1961, available at AEC Public Document Room.

32. Atomic Industrial Forum, Reactor Site Criteria (Proposed Revision), June 6, 1961, available at AEC Public Document Room.

33. Site Evaluation, Proposed Revision by Atomics International, W. L. Sequeira, June 7, 1961, available at AEC Public Document Room.

34. Comments on Proposed AEC Regulations, Atomic Energy Clearing House, 7(21): 19–23 (May 22, 1961).

35. Comment from American Public Power Association, A. Radin, June 13, 1961, available at AEC Public Document Room.

36. Further Comments, Atomic Energy Clearing House, Special Issue, 1–6 (July 20, 1960).

37. Comment from ACRS, T. J. Thompson, June 16, 1961, available at AEC Public Document Room.

38. Comment on AEC Regulations, Atomic Energy Clearing House, 7(27): 11-27 (July 3, 1961).

39. Comment on Reactor Site Criteria, Atomic Energy Clearing House, 7(28): 14-17 (July 10, 1961).

40. Comment from City of Los Angeles Department of Waters and Power, I. L. Bateman, March 7, 1961, available at AEC Public Document Room.

41. Comment from Socony Mobile Oil Company, D. H. Clewell, May 29, 1961, available at AEC Public Document Room.

42. Comment from Bechtel Corporation, W. Kenneth Davis, June 9, 1961, available at AEC Public Document Room.

43. An AEC Regulation [illegible] Atomic Energy Clearing House, 7(24): 33 (June 12, 1961).

44. Comments on Proposed AEC Regulations, Atomic Energy Clearing House, 7(21): 19-23 (May 22, 1961).

45. Comments on AEC Proposed Regulations, Atomic Energy Clearing House, 7(22): 13-21 (May 29, 1961).

46. Comments on AEC Regulations, Atomic Energy Clearing House, 7(25): 8-10 (June 19, 1961).

47. Comments on AEC Regulations, Atomic Energy Clearing House, 7(26): 29-31 (June 26, 1961).

48. Comments on AEC Regulations, Atomic Energy Clearing House, 7(27): 11-17 (July 3, 1961).

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50. AEC Actions on Regulations, Nuclear Safety, 3(1): (September 1961).

51. Comments on AEC Proposed Rule for Change Procedures, Atomic Energy Clearing House, 7(29): 27–30 (July 17, 1961).

52. Presented Below Are Further Comments, Atomic Energy Clearing House, Special Issue (July 20, 1961).

53. Two Memoranda Setting Forth Comments, Atomic Energy Clearing House, 7(32): 23–27 (Aug. 7, 1961).

54. Comments on AEC Regulations, Atomic Energy Clearing House, 7(23): 21–24 (June 5, 1961).

55. The Weekly Round-Up of Licensing Material, Atomic Energy Clearing House, 7(24): 8 (June 12, 1961).

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