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TECHNICAL PROGRESS REVIEW

NUCLEAR SAFETY

VOLUME 5, NO. 1

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A draft copy of Nuclear Safety, Vol. 5, No. 1, 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, Mr. H. F. Carroll of Declassification, and to me. Inasmuch as the final draft of this issue is scheduled to go to press early in September, your comments will have to be received by August 10 in order to be considered for inclusion in the final publication. Mr. Hazard's and Mr. Carroll's addresses are as follows:

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Yours truly,

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TECHNICAL PROGRESS REVIEW
NUCLEAR SAFETY

Volume 5, No. 1

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by Oak Ridge National Laboratory

Date Issued

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OAK RIDGE NATIONAL LABORATORY
Oak Ridge, Tennessee
operated by
UNION CARBIDE CORPORATION
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U.S. ATOMIC ENERGY COMMISSION

FOREWORD

This quarterly journal is one of a series of Technical Progress Reviews prepared by the Oak Ridge National Laboratory at the request of the Division of Information Services, U.S. Atomic Energy Commission. This Review is intended to assist those interested in keeping abreast of significant developments in the field of nulcear 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.

Three distinctly different types of articles may be found in this issue of Nuclear Safety. In addition to the usual reviews of current

literature and special review articles on specific topics, this issue contains a feature article. The feature articles are solicited monographs by nationally recognized experts on particular topics of interest selected by the editors.

The special articles permit discussion of pertinent subjects that 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. With the possible exception of the invited articles, 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.

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TECHNICAL PROGRESS REVIEW

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I. GENERAL SAFETY CRITERIA

RADIATION HAZARDS IN SPACE

W. E. Kinney

It was thought prior to the launching of the first artificial earth satellites that man had little to fear from the radiation he would encounter in space. Data obtained with rockets had indicated that the electromagnetic radiation outside the earth's atmosphere lay mostly in the visible portion of the spectrum and that the intensities of ultraviolet and x radiation were so low as to be of no concern. The available data likewise indicated that the intensity of cosmic radiation in interplanetary space was low enough to be well within tolerance limits. Although Forbush, as early as 1946, suggested the sun as a possible source of large bursts of energetic particles,[1] the evidence supported the conclusion that such solar bursts were a rarity.

This period of cosmic euphoria was ended abruptly by Van Allen in May of 1958 when he announced the discovery of the great radiation belts girdling the earth. From data taken with the artificial earth satellites Explorers I and II, he concluded that large numbers of charged particles were trapped in the earth's magnetic field. As if this were not enough, data obtained by satellites and by continuous monitoring devices, such as the riometer,[2] were bringing about the realization that Forbush solar bursts were by no means so rare as had been thought. They were found to be occurring on the average of once a year and to contain large numbers of particles of such energies as to pose a very serious threat to the safety of man in space.

In the following, the radiations found in interplanetary space — electromagnetic, primary cosmic, trapped, and solar cosmic — will be

described and the hazard associated with each will be indicated. The author has relied heavily on an excellent review by Newell and Naugle[3] and upon information presented at the symposium on "Protection Against Radiation Hazards in Space," which was held November 5–7, 1962, at Gatlinburg, Tennessee.[4-10]

Electromagnetic Radiation

Most of the electromagnetic radiation present in space comes from the sun. In the earth's orbit the total solar energy flux amounts to about 0.14 w/cm^2, 93% of which lies in the visible portion of the spectrum, with most of the remainder occupying the ultraviolet region from 2000 to 4000 A. Intensities fall off very rapidly as wave length decreases below 2000 A, so, by the time the soft x-ray region is reached, intensities of the order of 10^{-8} w/cm^2 are found. Gamma-ray intensities are so small as to be negligible.

Near a planet there is infrared radiation from the planet itself, which, in the case of the earth, amounts to 0.06 w/cm^2 on the average. A planet also reflects some of the solar radiation falling on it, the amount being determined by the albedo, which, for the earth, is equal to 0.4.

It appears that the electromagnetic radiation presents no serious problem in space flight, because the cooling requirements are well within the capabilities of current technology. Only when missions close to the sun are contemplated will it become a major problem.

Primary Cosmic Rays

The most energetic particles in space are the primary cosmic rays, there having been detected particles with energies as high as 10^{19} ev.

About 86% of the rays are protons, 13% are alpha particles, and the remaining 1% consists of nuclei of heavier elements up to, as least, iron. These primary cosmic rays originate outside the solar system, and the flux in interplanetary space is about 2 particles per square centimeter per second. The flux is distributed in energy as shown in Fig. I-1, but near a planet one-half the particles are removed because of shielding by the planet itself. It is fortunate that the fluxes are low, because shielding space vehicles against particles of so high an energy is quite out of the question.

The total dose from the primary cosmic radiation[3] is between 5 and 12 r/yr. While the total integrated dose from the heavy nuclei cosmic-ray components is a small faction of this total, there has been concern, in the absence of data, with respect to the damage caused by the extremely dense ionization tracks produced by these particles. Recently, Curtis,[4] based on a simulation of the heavy particle ionization tracks with a deuteron microbeam, concluded that the heavy component need be of no special concern.

The primary cosmic rays give rise to secondary particles by interacting with the nuclei in a planet's atmosphere, as evidenced by the cosmic-ray showers recorded on the earth's surface. While the secondary protons and mesons slow down and may be neglected, the secondary neutrons, by escaping to high altitudes and there decaying into protons and electrons, provide a mechanism for replenishing the particles in the Van Allen belts.[11] So it is seen that Forbush's early concern regarding the danger associated with electromagnetic and primary cosmic radiation in space was essentially correct and that later data have only reinforced his view.

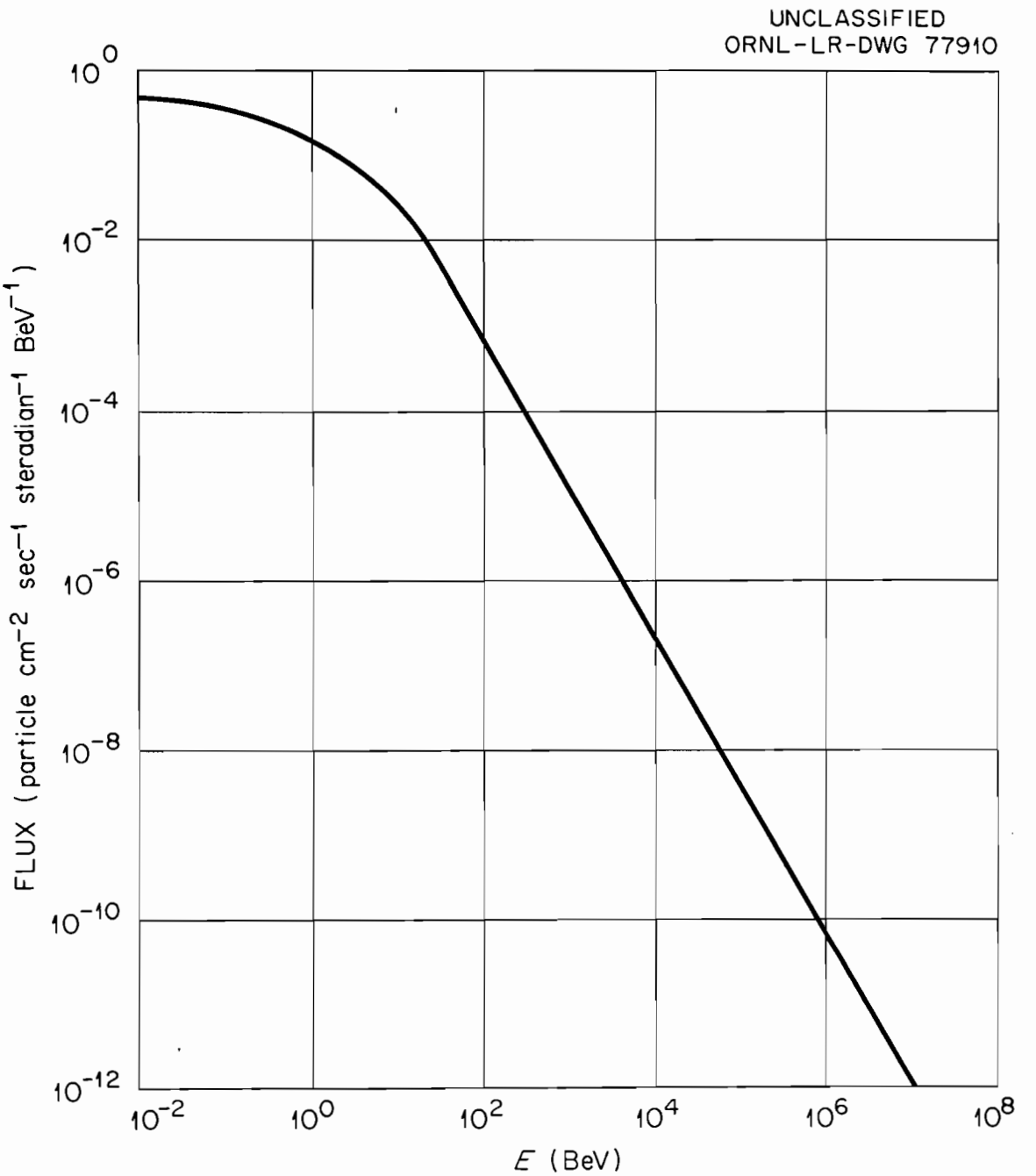

Fig. I-1. Cosmic Ray Energy Spectrum (see ref. 3).

Trapped Radiation

The Van Allen belts are made up of electrons and protons caught in a prison whose bars are the earth's magnetic field lines. They spiral about magnetic lines of force and bounce rapidly (periods ranging from micro- to milliseconds) back and forth from pole to pole while drifting slowly (periods ranging from minutes to hours) around the earth.[12] The combined effect of all these motions is to produce a swarm of particles in the shape of a figure of revolution whose axis, being coincident with the axis of the earth's dipole magnetic field, is offset 360 km from the center of the earth and whose cross section is rather like that of a kidney bean. The displacement of the field axis causes the height of the inner edge of the belts to vary longitudinally from a minimum of about 400 km over Santiago to a maximum of about 1500 km over Australia. Particles are constantly being lost from the belts by various means, such as slowing down and charge exchange, while new ones enter through the decay of the cosmic-ray secondary neutrons.[11]

The trapped radiation is separated into two zones, the first of which has its outer boundary at 1.8 earth radii, the second at 12 earth radii. The inner zone contains protons having energies as high as 700 Mev, as well as electrons, and is quite stable. The outer zone, on the other hand, is made up of electrons and low-energy protons and is subject to large intensity variations about a more or less standard state. Recent compilations by Van Allen[5] giving sample structure functions for the two types of particles are illustrated in Figs. I-2 through I-5. Three-dimensional distributions are obtained by rotating the figures about the vertial axis. In Figs. I-2 through I-5, J_o is the omnidirectional flux.

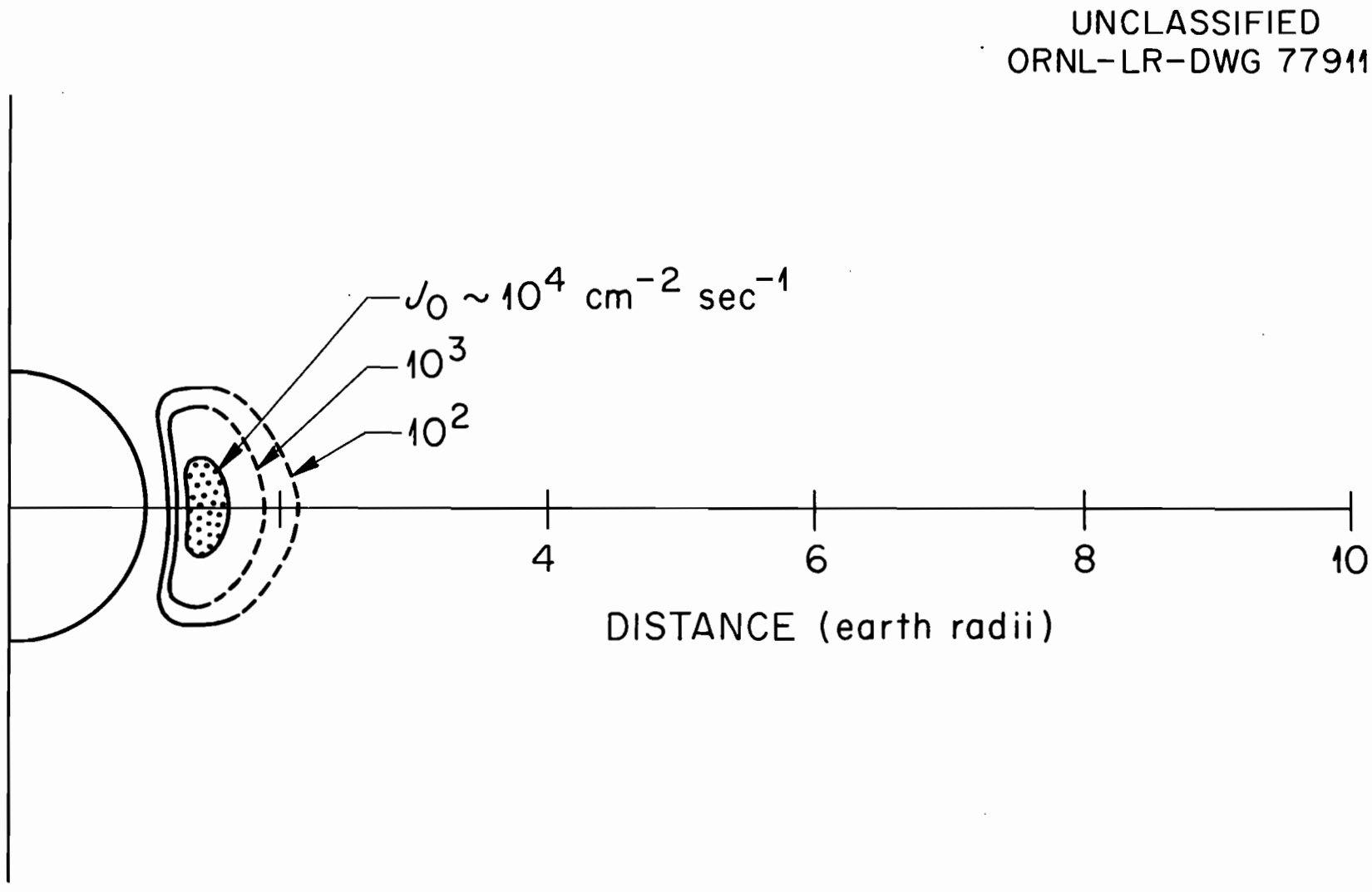

Fig. I-2. Distribution of Trapped Protons with Energies Greater Than 30 Mev.

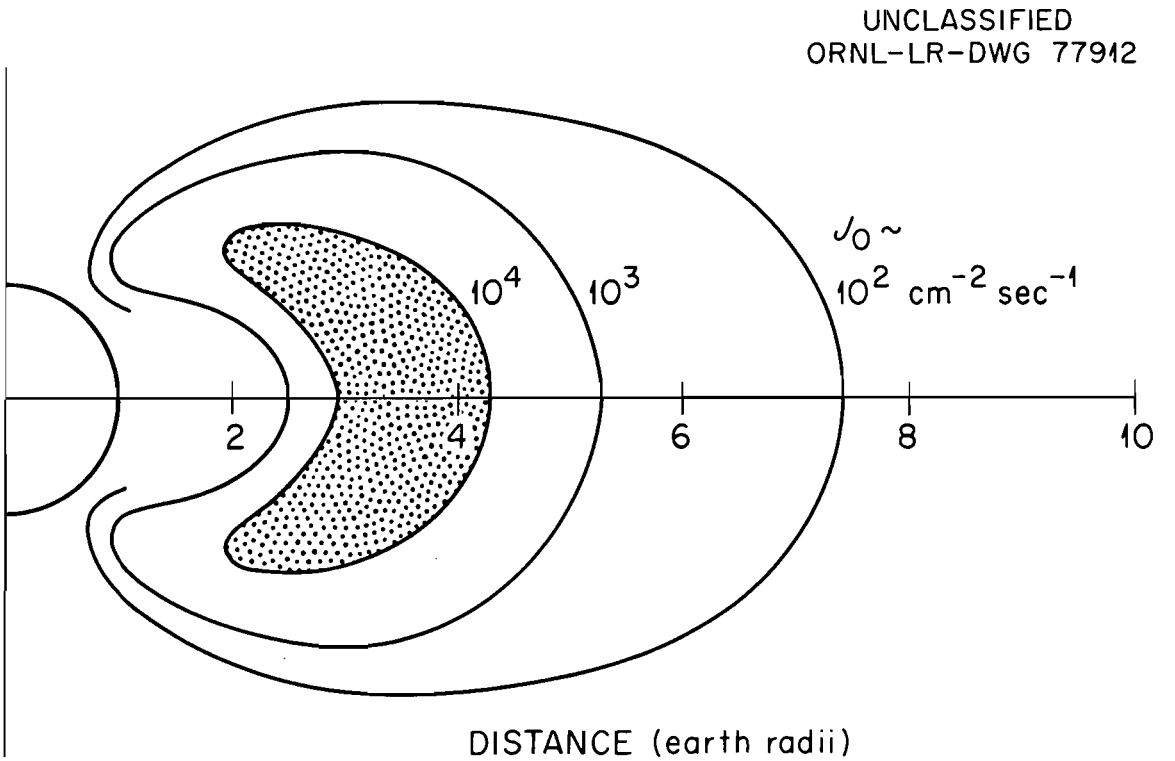

Fig. I-3. Distribution of Trapped Electrons with Energies Greater Than 1.5 Mev.

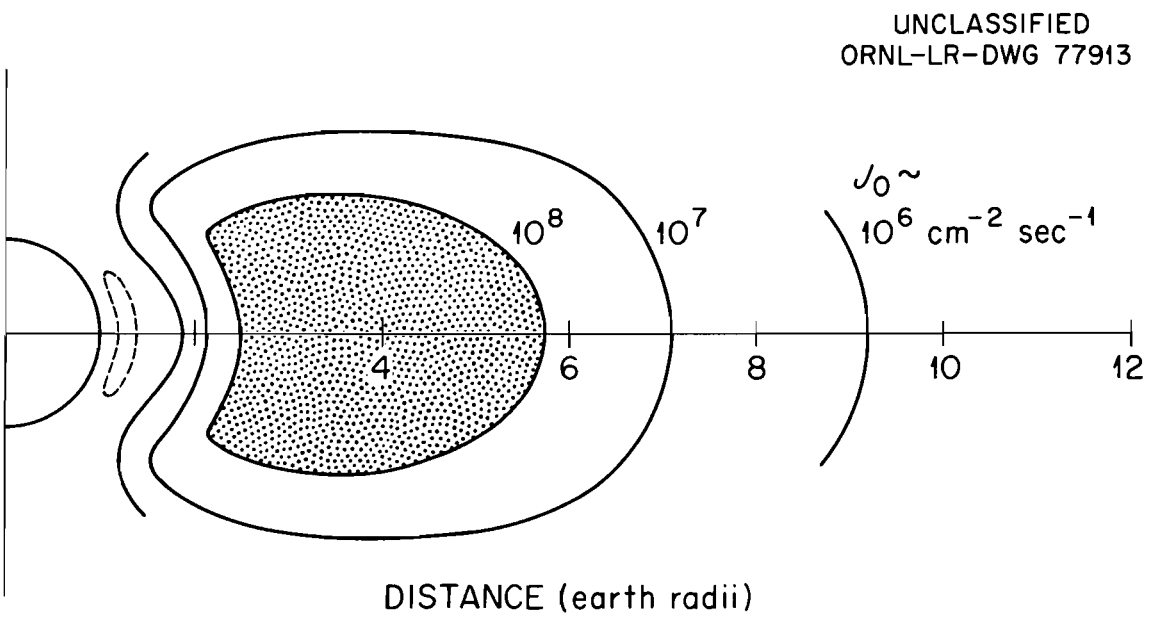

Fig. I-4. Distribution of Trapped Protons Having Energies Between 0.1 and 5 Mev.

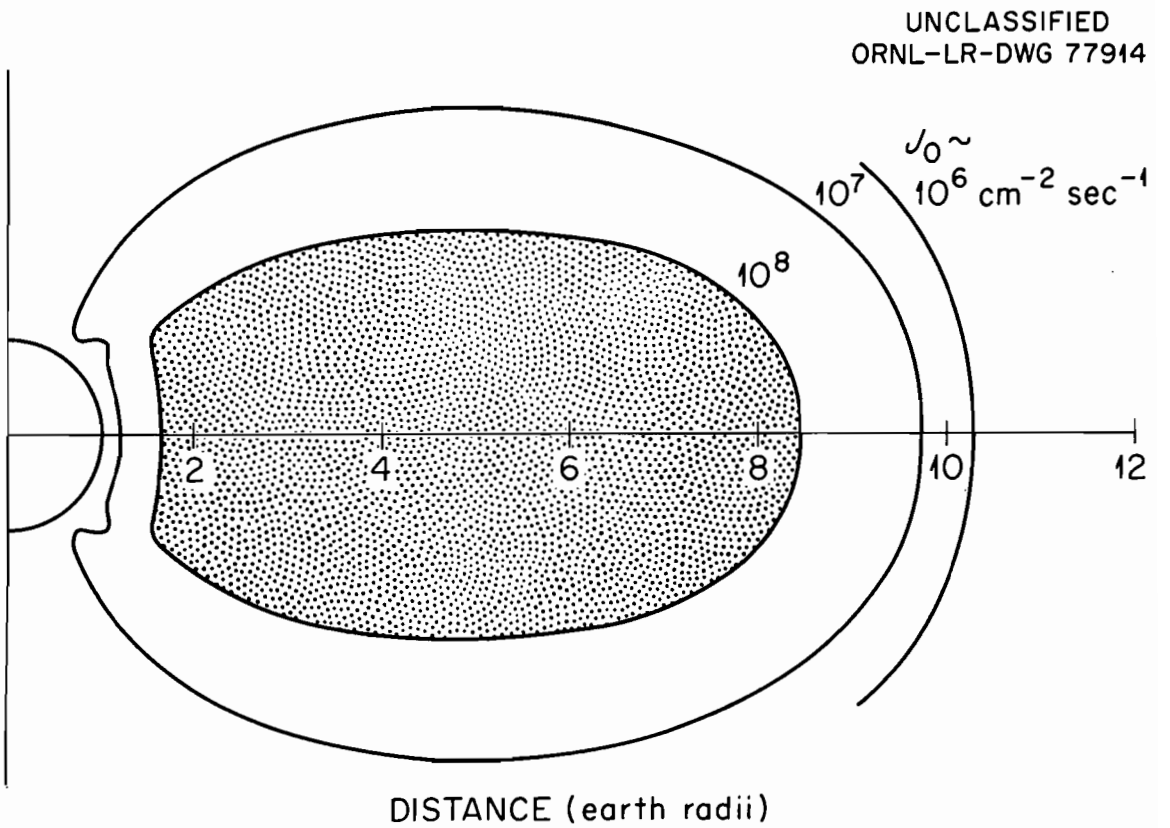

Fig. I-5. Distribution of Trapped Electrons Having Energies Greater Than 40 kev.

The distribution of protons with energies greater than 30 Mev is given in Fig. I-2. The flux of the electrons having energies greater than 1.6 Mev is shown in Fig. I-3 and is intended to be representative of the standard state. The distribution of protons in the energy range from 0.1 to 5 Mev is given in Fig. I-4, and Fig. I-5 shows the flux of electrons having energies greater than 40 kev.

The protons and electrons upon interacting with matter, such as the wall of a space vehicle, give rise to secondary radiation. The protons undergo nuclear interactions and produce secondary neutrons, protons, mesons, and photons. For thin shields, this proton secondary radiation is of little importance relative to the primary high-energy protons, which penetrate the walls. The relative importance of the emerging secondary radiation increases, however, with shield thickness, since the primary radiation is attenuated by the very process that creates secondary radiation. The secondary radiation produced within the body of the astronaut by the interaction of the leaking radiation with tissue is quite another matter and cannot be neglected. Electrons are slowed and stopped in the wall, and it is the resulting x radiation that presents the hazards. The dose as a function of distance from the earth along the trajectories of Pioneer III and IV is shown in Fig. I-6 (see ref. 3). The contributions of the protons in the inner zone and the electron-produced x-rays in the outer zone are clearly separated.

Any planet with a magnetic field would be expected to have trapped about it bands of radiation similar to the Van Allen belts. This type of radiation may well be encountered elsewhere in the solar system, although apparently not on a mission to Venus, where Mariner 2 found a very weak magnetic field and consequently no significant radiation belts.[13]

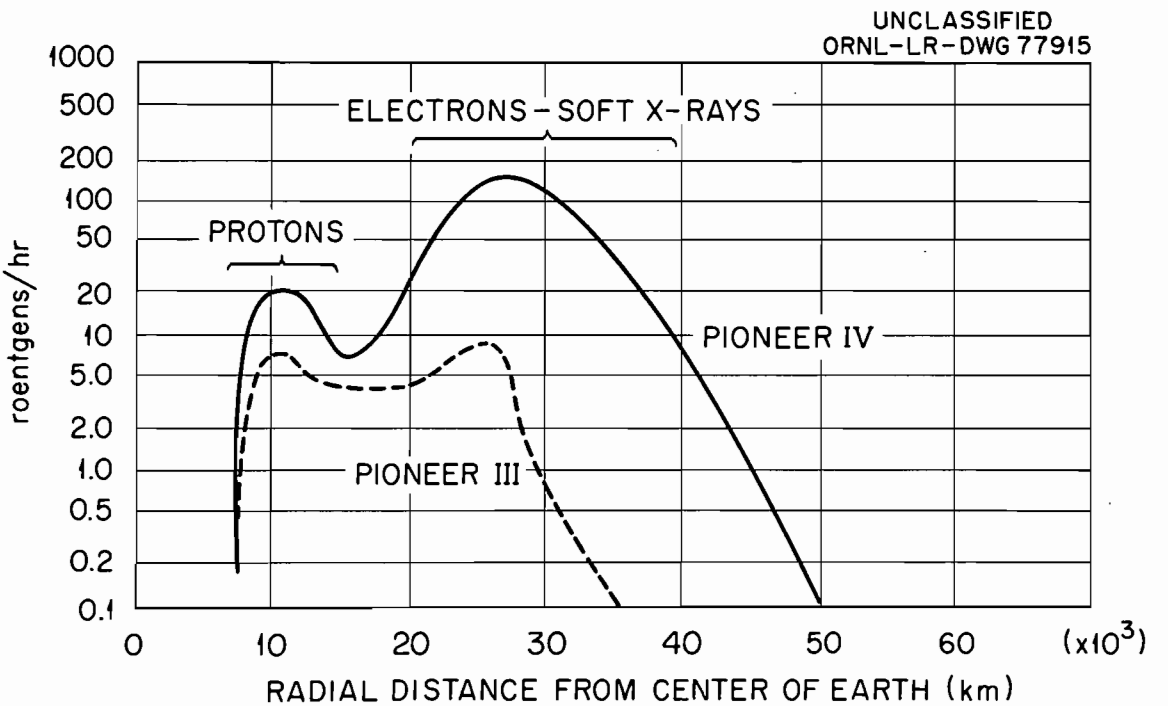

Fig. I-6. Variation of Radiation Level Along the Trajectories of Pioneer III and Pioneer IV (see ref. 3).

Solar Cosmic Rays

The most serious radiation problem in space appears to be presented by the solar cosmic rays. These, like the primary cosmic rays, are composed of particles, with protons being generally the most plentiful. They originate in severe magnetic storms on the sun, known as solar flares, and travel out into space with various energies under the influence of expanding magnetic fields.[6] Each event is highly individual, with its own peculiar intensity, temporal, spectral, directional, and charge characteristics.[7] For example, the total intensity may vary from 10^6 to several times 10^9 particles per square centimeter. An event may last from a few hours to a few days, and the proton-to-alpha particle ratio may range from 1 to 20. Even so, there are a few similarities among the events. The time to reach peak intensity at the earth can be expected, however, to increase with decreasing particle energy. Most of the flare particles arrive at the earth isotropically, even though frequently there is a short-lived component coming from a preferred narrow region of space.

Bailey[14] has constructed a "typical" flare that illustrates the history of a solar event and is useful for the comparison of dose calculations. The time behavior of the integrated proton flux above various energies for Bailey's flare is given in Fig. I-7, and the time-integrated proton and energy spectra are given in Fig. I-8. From these data, Keller[8] has computed the primary proton dose as a function of shield thickness, as shown in Fig. I-9.

The problem of secondary radiation is the same for solar protons and Van Allen protons. The importance of the secondaries relative to the primaries increases with shield thickness so that as shields get thicker

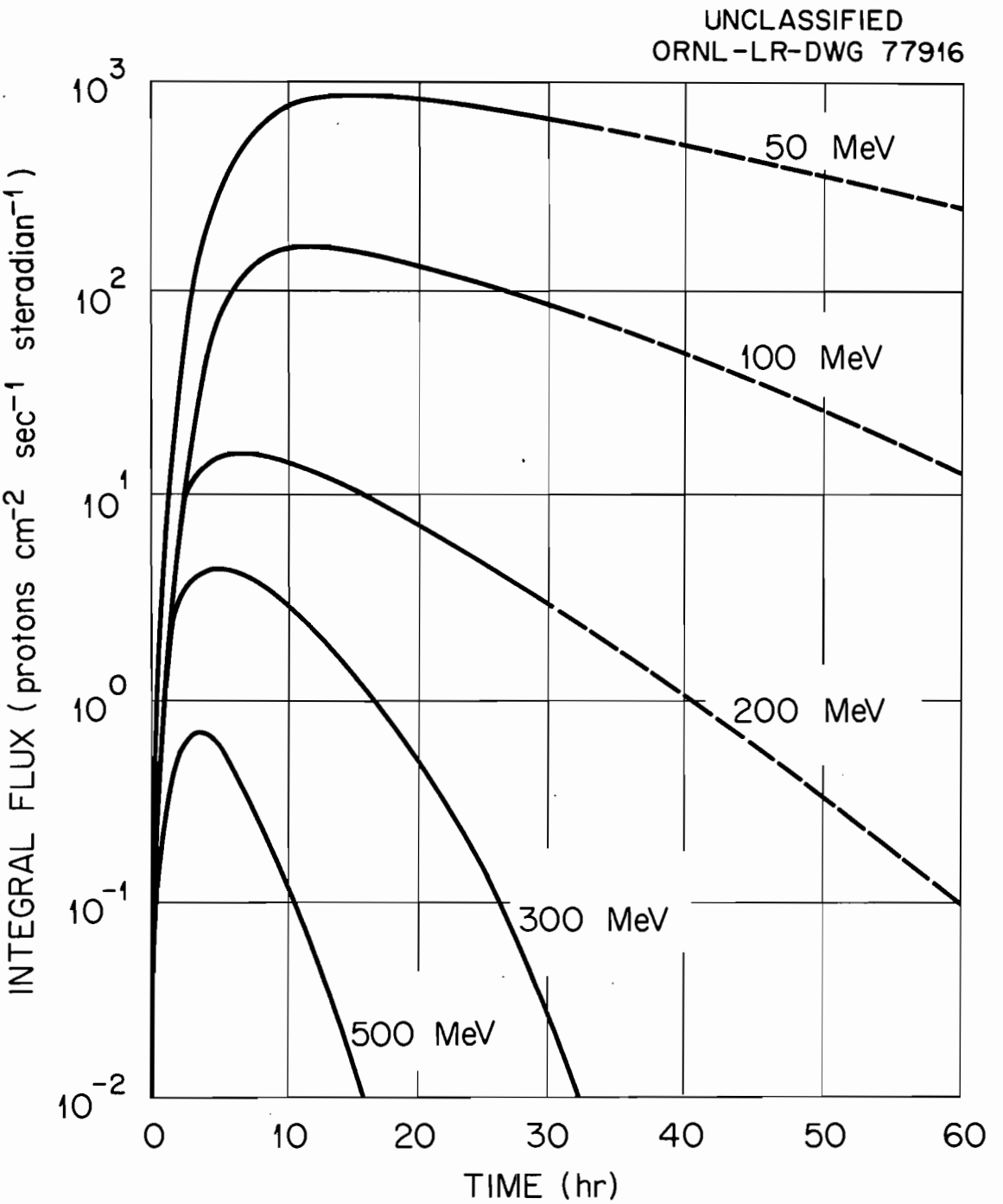

Fig. I-7. Time Variation of the Integral Proton Flux from a Bailey Flare (see ref. 8).

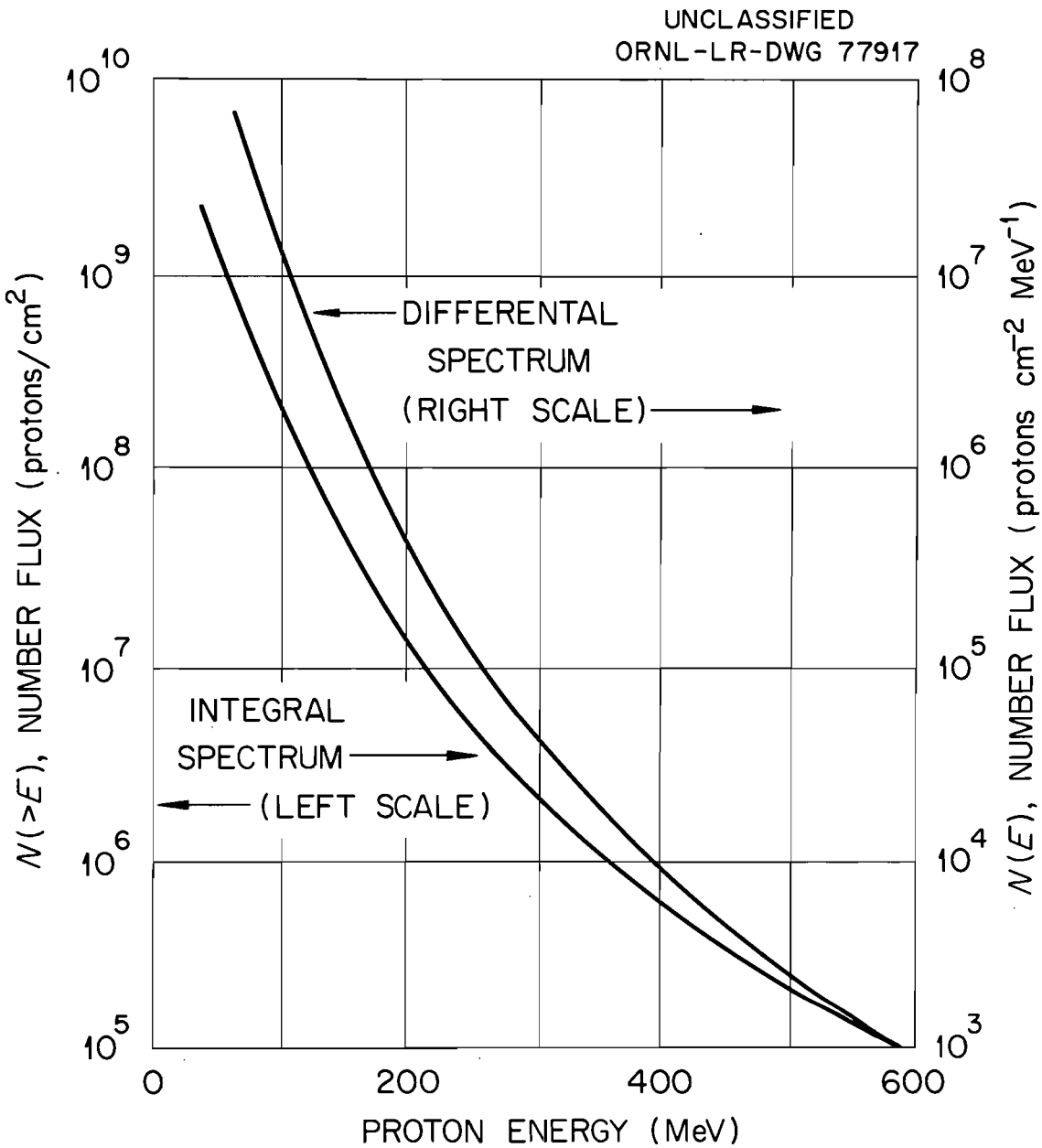

Fig. I-8. Time Integrated Proton Spectra from a Bailey Flare (see ref. 8).

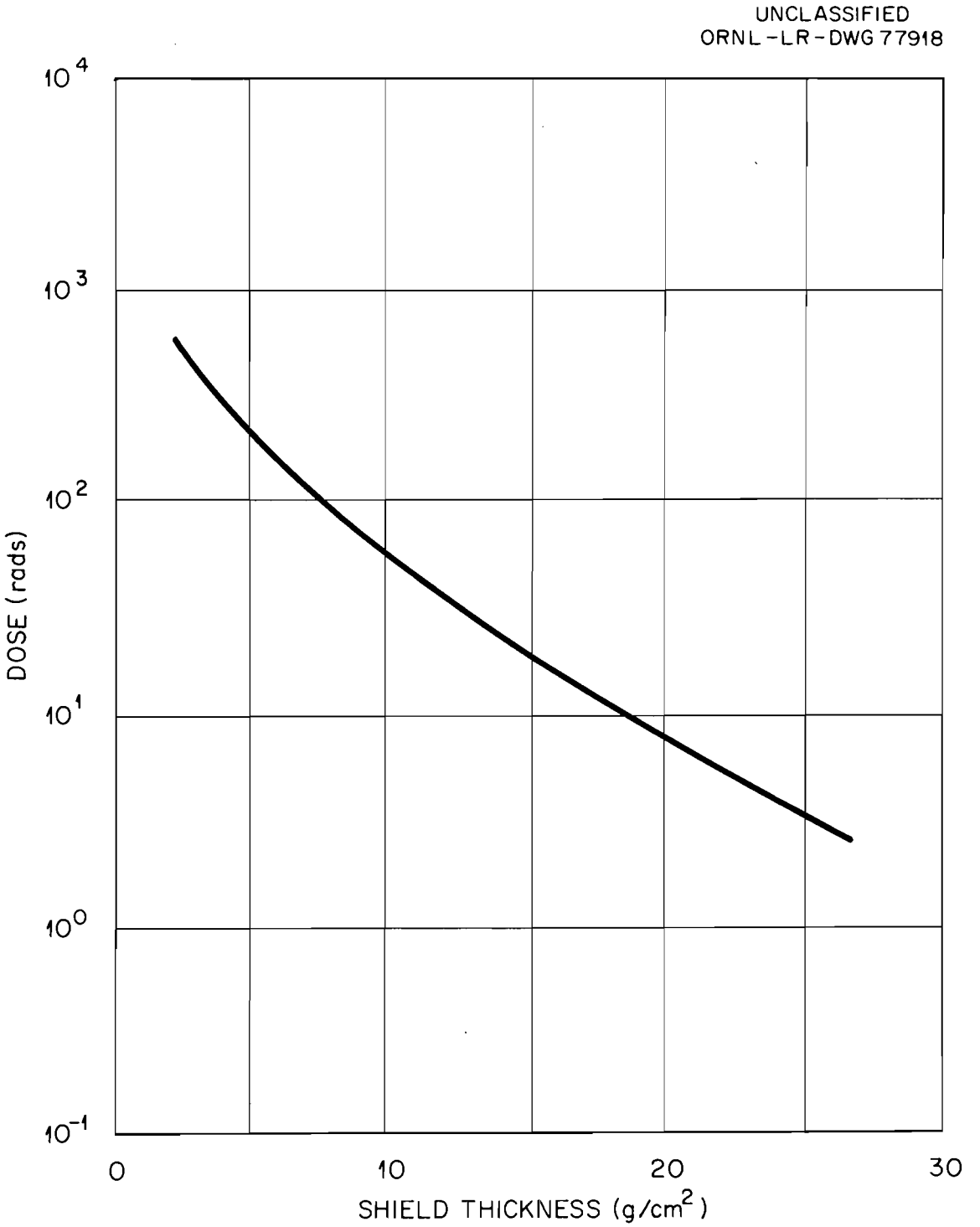

Fig. I-9. Bailey-Flare Proton Dose as a Function of Shield Thickness (see ref. 8).

to allow missions of longer duration, the secondaries could become the sole contributor to the dose resulting from solar events.

If a solar weatherman could predict with a high degree of certainty the occurrence of a flare event, he would in large measure solve the solar cosmic-ray problem for missions in the vicinity of the earth. The missions could then be delayed until after the storm passed or terminated before it started. Work is proceeding in the correlation of certain conditions on the sun with the occurrence of solar particle events.[9]

Comparison of Components

The relative importance of the types of radiation described above may be inferred from a comparison of the typical exposure levels given in Table I-1. It is obvious, however, that the relative importance depends very much on the type of mission to be undertaken, that is, on the

Table I-1 A COMPARISON OF SOME TYPICAL EXPOSURE LEVELS (TAKEN FROM REF. 3)

Radiation	Typical Exposures
Normal radiation at sea level	0.001 r/day
Cosmic rays in undisturbed interplanetary space	5–12 r/yr
Protons at middle of inner Van Allen belt	24 r/hr
Soft x rays at middle of outer Van Allen belt	~200 r/hr
Solar proton events	$10–10^3$ r/hr

time of exposure to each type. Keller[8] has compared doses from solar protons, primary cosmic rays, and Van Allen belt particles as a function of shield thickness for a two-week lunar mission and for a two- to three-year interplanetary mission. Results for the lunar flight, assuming that one Bailey flare occurred, are given in Fig. I-10. For this type of mission, the Van Allen cosmic radiation is seen to be of little importance. The expected doses for the long interplanetary trip, where one flare per year was assumed, are given in Fig. I-11. The solar proton and cosmic-ray doses have naturally increased in relation to the Van Allen dose.

Apollo Radiation Tolerances

The effort of the United States to land men on the moon is designated Project Apollo. To maximize the likelihood of success of the mission requires a balancing of risks, of which radiation is but one. It is unrealistic to apply atomic industry tolerances to this pioneering effort, although they have served as guidelines in the setting of the tolerances. Briefly, Apollo has taken the recommended maximum permissible integrated lifetime doses and allowed them to apply to a flight career of 5 years.[10] The tolerances are listed in Table I-2. The Apollo capsule will have wall thicknesses of about 3 to 6 g/cm^2, so, as Fig. I-10 shows, if no solar flare occurs, the mission dose will be well within tolerance. Moreover, even if a Bailey flare does occur, the whole-body emergency dose will not be exceeded.

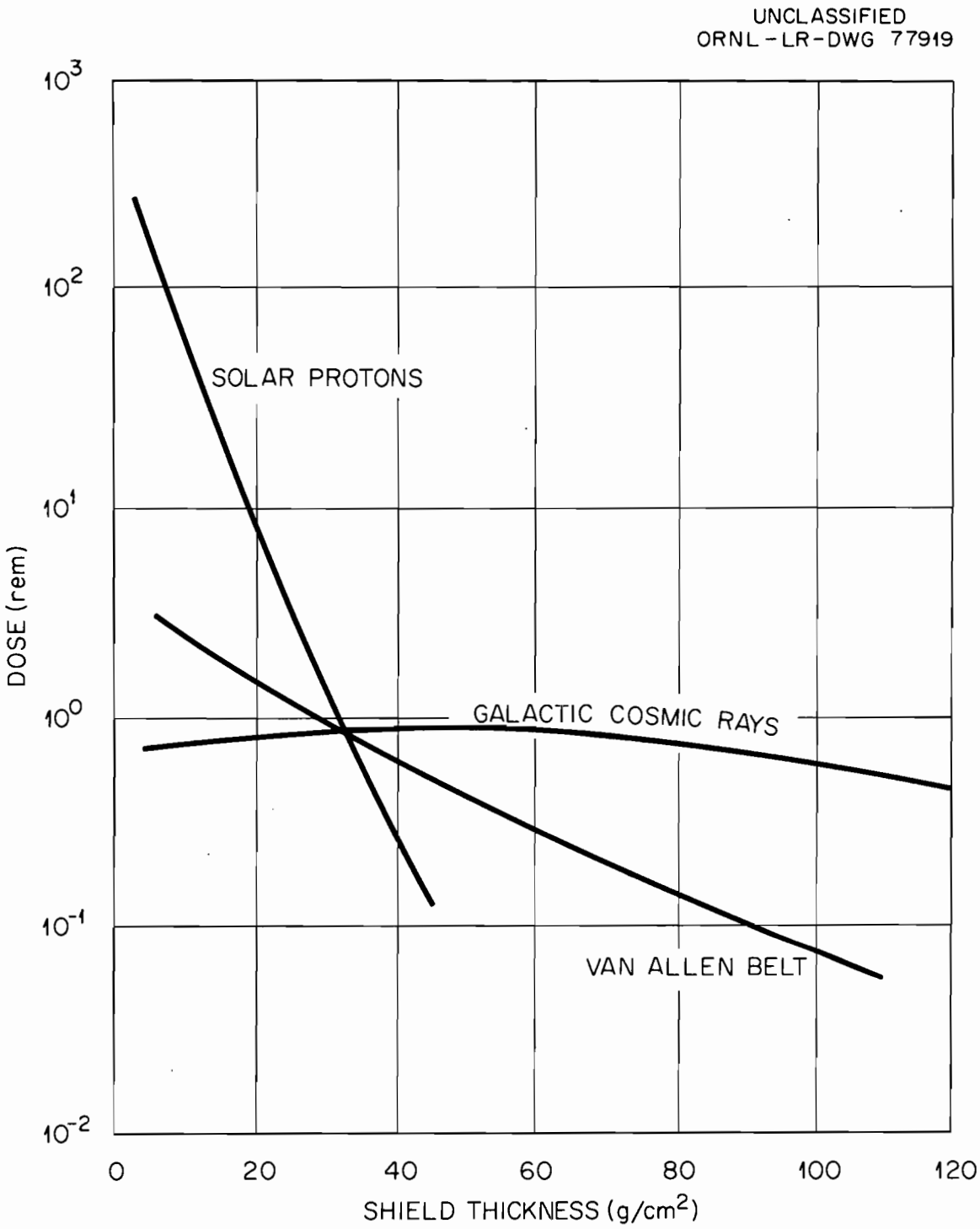

Fig. I-10. Partial Doses as a Function of Shield Thickness for a Two-Week Lunar Mission (see ref. 8).

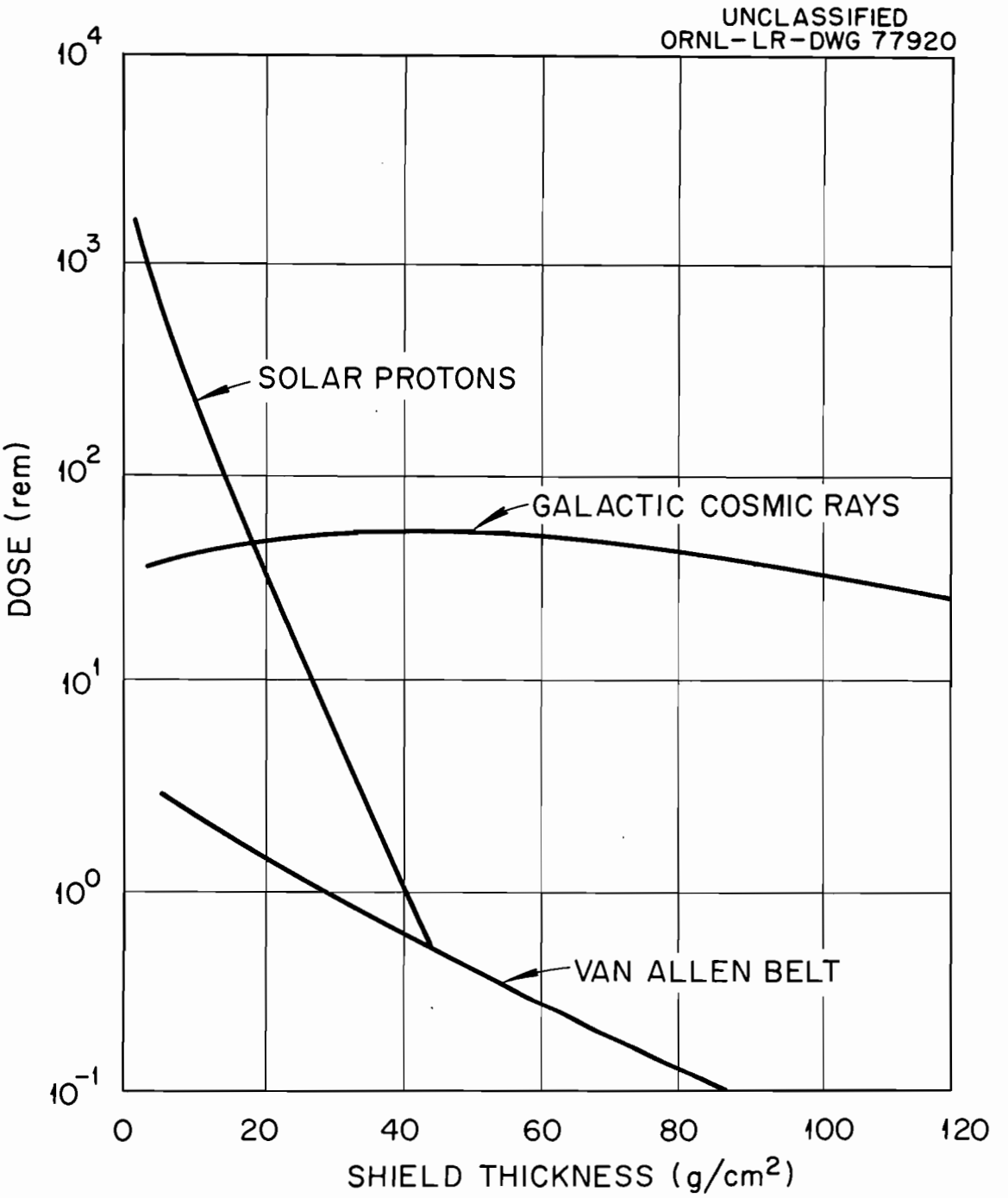

Fig. I-11. Partial Doses as a Function of Shield Thickness for an Interplanetary Mission (see ref. 8).

Table I-2 SUMMARY OF SPACE RADIATION EXPOSURE DOSE LIMITS

Critical Organ	Maximum Permissible Integrated Dose (rem)	RBE, Relative Biological Effectiveness (rem/rad)	Average Yearly Dose (rad)	Maximum Permissible Single Acute Emergency Exposure (rad)
Skin of whole body	1,630	1.4	233	500[a]
Blood-forming organs	271	1.0	54	200
Feet, ankles, and hands	3,910	1.4	559	700[b]
Eyes	271	2[c]	27	100

[a]Based on skin erythema level.

[b]Based on skin erythema level; these appendages believed to be less radiosensitive.

[c]Slightly higher RBE assumed because eyes are believed more radiosensitive.

IAEA SYMPOSIUM ON SITING NUCLEAR FACILITIES

Wm. B. Cottrell*

A symposium on "Criteria for Guidance in the Selection of Sites for the Construction of Reactors and Nuclear Research Centers" was held in Bombay, India, March 11-14, 1963. About 120 scientists from 12 countries and five international organizations took part in the four-day symposium, which was sponsored by the International Atomic Energy Agency with the cooperation of the Government of India.

The symposium was in reality an outgrowth of the more limited panel meeting on "Reactor Siting"[15] which was held in Vienna in the fall of 1961 and which was jointly sponsored by IAEA and ISO (International Standards Organization). The Bombay symposium was divided into four sessions which were devoted to a consideration of the various factors that have to be taken into account in the choice of atomic center sites and the criteria that can be applied in the evaluation of particular reactors. For convenience in the discussion of the information presented, a different division of the materials is employed here, and a single paper may be referred to in several sections. This review is also limited to those articles that were generally available at the meeting; it is recognized that some significant material may be missing, although copies of 30 of the 37 or more papers presented at the meeting were examined.

As announced by the IAEA:[16]

> "The Bombay meeting...[discussed] the criteria for guidance in the selection of sites for the construction of reactors

---

*W. K. Ergen, who attended the symposium, assisted in this review by discussion of some of the papers.

> and nuclear research centres. Because of the possible hazards associated with atomic energy operations, the siting of a reactor centre is often a matter of great public interest. In addition to the radiation hazards from the normal operation of a reactor or arising from a reactor accident, an important consideration relates to the suitability of the site for the disposal of radioactive wastes. Besides, there are major considerations concerning engineering requirements or arising from the desirability of having a nuclear research centre reasonably close to supporting facilities or educational institutes or having a power reactor near enough to the consumers to make it economically viable. The requirements are often complicated by many local factors and are sometimes mutually conflicting. The aim of the Bombay meeting [was] to examine the principal considerations and the criteria which can be applied in dealing with the relevant factors in a specific case."

In general the considerations involved in siting low-power research reactors and nuclear research centers are the same as those for siting high-production power reactors, but the requirements for the latter are more severe and the relative importance of various factors may change. Although none of the papers presented in this session included all the factors listed below, each of the following factors enters into site selection to some degree:

1. Labor and building costs
2. Living conditions for employees (economic, social, cultural)
3. Proximity of universities and research centers
4. Capability for future expansion
5. Land costs
6. Utilities costs
7. Transportation costs (roads, railroads, etc.)
8. Proximity of manufacturing firms
9. Site development costs
10. Public relations

11. Surrounding land use
12. Distance from densely populated areas
13. Interference with other important installations
14. Surrounding population
15. Reactor type and power
16. Containment provisions
17. Floods or earthquakes
18. Meteorology, hydrology, and geology
19. Contamination of water supplies
20. Availability of cooling water
21. Means of waste disposal
22. Capability of storing radioactive materials
23. Local or national zoning
24. Local or national building restrictions

Power Reactor Siting in the United Kingdom

British philosophy with respect to the siting of power reactors is rather thoroughly described in three papers.[17-19] The first two are concerned with many nonnuclear aspects of site selection, and the third with the evaluation of power reactor sites. This latter evaluation is the same as that described in the 1961 siting meeting.[15] Of particular interest in the recent paper are the two tables reproduced here as Tables I-3* and I-4.** Table I-3 lists the total doses to the general population that are considered acceptable in major accident conditions

*Table II of ref. 17.

**Table III of ref. 17.

Table I-3 TOTAL DOSES TO GENERAL POPULATION ACCEPTABLE IN MAJOR ACCIDENT CONDITIONS IN THE UK

1. External Radiation
   a. Children up to age of 16 and pregnant women - 20 r.
   b. Other persons - 30 r.
2. Beta and Gamma Dose Combined in any Superficial Tissue
   a. Children up to age of 16 - 75 r.
   b. Other persons - 150 r.
3. Internal Radiation from Inhalation and Ingestion

Isotope	Dose to Critical Organ
I	25 r to the thyroid
Sr^{89}	15 r to the bone
Sr^{90}	1.5 r/year to the bone
Cs^{137}	10 r whole body

Table I-4 RELATIVE LIMIT RANGES FOR A VOLATILE RELEASE

	Nature of Hazard	Relative Range
1.	Radiation from cloud	1
2.	Radiation from ground deposition Inhalation of strontium	2
3.	Inhalation of mixed fission products to lung	4
4.	Control of exposed crops (Ca, Sr, deposition)	7
5.	Inhalation of iodine	50
6.	Iodine on surface crops and in eggs	60
7.	Iodine in milk	350

in the United Kingdom. These doses may be compared with the 25-rem whole body and 300-rem thyroid doses similarly employed in the United States. Table I-4 lists the relative ranges (distances) for limiting exposures from a number of different exposure sources following an accident. It is apparent from Table I-4 that ingestion exposures must be given serious consideration in the evaluation of accident consequences.

Examination of the nonnuclear considerations reveals that, in addition to the economic considerations of power needs, water supply, site conditions, employment, population, transportation, etc., the British are especially concerned with the "need to avoid profligacy in the use of natural resources" and a "desire to preserve the amenities of suitable sites." These considerations have resulted in the withdrawal from consideration of over 40% of the land area of England and Wales. "Planning" legislation exists for the national and local coordination of developments that would involve a change of land use or large (>5000-ft^2 floor area) industrial construction, nuclear or othewise. The selection of suitable sites on this basis for further technical evaluation not only involves conformity to these national plans but also provides for a public inquiry, similar to the Public Hearings[20] in the United States, at which objectors have the opportunity to be heard. In pursuing these site policies, the United Kingdom Atomic Energy Authority endeavors to cultivate an informed public opinion to achieve objectivity and realism in assessing nuclear hazards.

The site requirements for nuclear power stations are, collectively, so exacting that it is geographically rare to find the desired conjunction of physical characteristics in England or Wales. Although all

power reactor sites chosen so far carry a Class I rating,[15] the number of such sites that appear even remotely acceptable technically for future stations is quite limited; however, relaxation in siting standards may well follow improvements in reactor design and engineered safety features that will result in increased safety.

Power Reactor Siting in Other Countries

Several other countries, including India, Poland, Sweden, France, and Mexico, were represented by papers that described in varying detail the specific procedures and criteria for siting power reactors. As might be expected, the procedures in none of these other countries approached those of the British or the United States in sophistication, although they frequently contained many common elements.

The practice of the Indian Atomic Energy Establishment is a curious combination of United States[21,22] and British practices, with greater emphasis on the latter. Use is made of the United States concepts of activity release from the maximum credible accident and multiple zones around the site; however, the British concept of grading sites on a relative rather than absolute basis is used, and thus the exposure limits are more comparable with the British limits. On such a basis, the Indian Atomic Energy Establishment selected Tarapur, Kalpakkam, and Rawatbhata as sites for the first power stations in western, southern, and north central India, respectively. As has been the experience of nuclear siting authorities in the United Kingdom, the best of several sites as far as accident exposures are concerned must frequently be eliminated for other reasons.

The philosophy of Poland regarding power reactor site selection[23] was not presented in detail, but it was stated that, in addition to the usual economic and other nonnuclear considerations, there were the following objectives. Power reactor sites should (1) have no inhabitants in the immediate vicinity, (2) be located in rarely (sparsely) inhabited areas, (3) not be nearer than 20 to 30 km from a big town, which should not lie in the line of prevailing winds, and (4) be downriver from nearby towns. Sites for the first nuclear power stations in Poland will be in the northeast along the shores of the Big Bug River or the coast of the Baltic Sea.

In Sweden[24] the siting problem of interest relative to a power reactor was that for the 65-Mw (thermal) Agests reactor, the primary function of which is the production of process heat, rather than electrical power generation. Since both the installation costs and the operating costs for heat transmission are high, the reactor was to be sited as close as possible to the consumer area, which was to contain at least 30,000 people. This is in direct contradiction to the general siting objective of large separation distances and is compensated for by the containment features which were provided with the installation. Although the reactor is only 15 kilometers (approximately 9 miles) from the center of Stockholm, it is underground in a steel-lined rock chamber[25] that has a prescribed leakage rate of only $2 \times 10^{-4}\%$ of the chamber volume per hour at 1.5 atmospheres over-pressure. Since only 10% of the iodine is assumed to be available for leakage, the specified leakage would give an exposure of 25 rem to children of the most sensitive age at a distance of 2 kilometers, which by definition becomes the restricted zone around

the reactor. Although the siting policy indicated here resembles that of the United States to the extent that a postulated maximum credible accident is used to determine exposures in the surroundings, the Swedish paper[25] indicated concern with the wide range of probabilities associated with several of the factors in the exposure calculation and stated that it would be more consistent and justifiable to employ "meteorological parameters connected with average meteorological conditions as long as the probability for the MCA [could] be considered as very low."

France likewise presented a report[26] on the site selection for a specific reactor, the gas-cooled reactor finally located at Chinon. The economic or safety features of the site were discussed in detail, but no claim was made that this siting set a policy. Mexico similarly presented, as a case history, its procedure in selecting a site for a power reactor, the construction of which is quite some time in the future.[27] Mexico also presented a site rating index based on the total population exposure.[28]

Siting Considerations for Nuclear Research Centers

At least six papers were concerned completely or in part with siting problems peculiar to nuclear research centers.[23,24,29-32] Some of the reactors in the various research centers discussed below could be considered power reactors, inasmuch as they have power levels ranging up to 30 Mw (thermal), but they are discussed here because of the siting factors unique to research centers that were considered. The extent to which many of the previously identified siting considerations enter into the evaluation of the location of a particular nuclear center at a

particular site is singularly dependent upon the type and power level of the reactor to be located at the center. Bell and Chicken[29] of the United Kingdom identified three classes of research reactor, as described below:

Class I. Reactors that are safe against all accidents to the coolant and in which excess reactivity is positively restricted so that damaging excursions cannot occur are included in Class I. This class of reactor would include "swimming-pool" reactors with plate-type fuel elements cooled by natural circulation and generally limited to powers less than 100 kw.

Class II. Reactors that are safe against all loss-of-coolant accidents followed by reactor shutdown and in which reactivity excursions may be prevented by limitation of fuel loading or administrative control of step changes in reactivity are in Class II. This class of reactor would include low-power (≤ 1 Mw) reactors, such as the Argonaut, Triga, or AGN reactors.[33]

Class III. Reactors that are subject to meltdown following a coolant accident but in which no significant pressure is generated are considered to be Class III. This class of reactor usually employs forced cooling with no significant pressurization of MTR-type fuel elements. Although Bell and Chicken[29] limited the power level in this class to 3 Mw, there is no apparent reason why higher power reactors with the same safety performance characteristics should not be considered as being in this class.

Bell and Chicken[29] concluded that there

> "[is] no safety reason why research reactors up to 3 Mw power cannot be built on suitable sites near urban districts.

> By proper design of reactor and building the release from a research reactor of less than 3 Mw power can be kept to a level at which the inhalation hazard should not extend beyond ~500 yards."

Although not specifically stated, it may be inferred that at ~500 yards the ingestion exposure may be readily controlled and that the "proper design" is capable of limiting I^{131} release in any accident to <10 curies. Obviously the low-powered research reactors are readily capable of meeting this criterion, even when located in a conventional building structure; the General Atomic paper[31] listed several such reactors located in populated surroundings.

Of particular interest were two papers on studies for siting major nuclear research centers, one by Sweden[24] and the other by the Netherlands.[30] The two centers are comparable in that large integrated installations are envisioned at both sites that should be compatible with reactor power levels in the 100-Mw range. The major Swedish considerations, which led to the selection of the site at Studsvik on Tvaren Bay on the Baltic Coast, were the following:

1. Situated on the coast
2. At least 1 km^2 of ground
3. Sparce population within 3 km
4. No population center with more than 1000 within 15 km
5. A university town within 100 km
6. Town with good schools in the vicinity

Only two places appeared to meet all the criteria, and Studsvik was chosen because of favorable road and railroad connections with Stockholm, which is 108 km away. While the details were not presented, it appears

that the third criterion serves as a limit on the ingestion iodine exposure from the 10% of the iodine inventory released to the building and subsequently filtered and discharged, whereas the 15-km zone was established for the same exposure but on the assumption, no longer considered valid, that the building could be damaged by an accident.

The considerations in the selection of the Rector Centrum Nederland site in Holland were necessarily less stringent because of the much higher population density and land use in Holland as compared with Sweden. The safety of the population in the vicinity of the center was, however, of primary concern, so the exclusion areas were established around all population centers in which nuclear centers were not to be located, as indicated below:

Population	Exclusion Area (km)
10,000–25,000	10
25,000–50,000	12
50,000–100,000	14
>100,000	16

In this manner approximately 75% of the country was excluded. In the remaining areas 30 reference sites were chosen for detailed examination. In this examination the surrounding population was of particular concern. Sites were considered acceptable if no more than 10,000 people lived within 7 km and the number of inhabitants at greater distances did not exceed that determined by the following relationship:

$$N = 30R^3 ,$$

where N is the number of inhabitants and R is the distance from the source in kilometers. Of the 30 sites considered, 18 were rejected on the basis of the population analyses, two because of the need to safeguard

important objects or installations, 26 because of technical requirements, and 11 because of social requirements. The RCN site was selected because it fulfilled all the requirements.

Mexico evaluated six sites for a nuclear research center containing a 1-Mw reactor and a 12-Mw Van de Graaff generator.[32] All sites were within 20 to 50 kilometers from Mexico City. The proximity to Mexico City appeared to be a necessary condition because of the requirement for qualified personnel. On the basis of meteorology, seismology, soil mechanics, hydrology, and population distribution, Sulazov and Parres appeared to be the most favorable locations.

Consequences of Calculated and Actual Activity Releases

The ultimate concern in the siting of a nuclear installation is the subsequent exposure of the surrounding population as a consequence of a major activity release. There is not, at the present time, an internationally accepted basis for the determination of the potential quantity of such a release, and there has been no agreement as to the exposure levels that should be employed for the calculation of control zones. Although the United Kingdom has been concerned with ingestion exposures for some time, particularly since the Windscale incident,[34] such considerations have not entered directly into United States siting criteria. The ingestion exposures have recently, however, been receiving significant attention. In a comprehensive analysis by Watson and Gamertsfelder[35] of Hanford, all the exposure paths from an atmospheric release were examined, as indicated in Table I-5.*

*Table I of ref. 35.

Table I-5 EXPOSURE PATHS - ATMOSPHERIC RELEASE

I. Airborne Contaminants

A. External Irradiation by:

1. Ambient air contamination[a]

a. whole body[b]
b. lung (contained air)
c. skin

B. Internal Deposition by:

1. Inhalation[a]

a. whole body[b]
b. thyroid[b]
c. bone
d. lung - GI tract

II. Deposited Contaminants

A. External Irradiation by:

1. Contamination of ground,[a] buildings, and clothing

a. whole body[b]
b. skin

B. Internal Deposition by:

1. Ingestion of contaminated fresh leafy vegetables or water

a. whole body[b]
b. GI tract[b]

2. Ingestion of milk[a] and other food chain products

a. thyroid[b]
b. bone[b]

[a]Predominant paths.

[b]Organs of primary importance.

Watson and Gamertsfelder calculated doses for all the exposure paths based on the assumptions listed in Table I-6,* and, on the basis of these assumptions, they calculated doses from (1) exposure to a cloud

*Data taken from ref. 35.

Table I-6 PERTINENT ASSUMPTIONS IN EXPOSURE CALCULATIONS

A. Fission-Product Fraction Escaping Containment	
Noble gases	1.0
Halogens	0.25
Volatile solids	0.15
All others	0.003
B. Atmospheric Dispersion — Moderate Stability: Modified Diffusion Equations	
a	97 m
b	0.33 m^2 sec^{-1}
k^2	2.5×10^{-4} sec^{-2}
h	0
u	1 m/sec
C. Deposition Rates	
I^{131}	3.4×10^{-3} $\bar{u}_0$ m/sec
Sr^{89}, Sr^{90}	2.2×10^{-4} $\bar{u}_0$ m/sec
Cs^{137}	2.2×10^{-4} $\bar{u}_0$ m/sec
D. Retention of Contamination by Herbage (Uptake Through the Root System is Ignored)	
I^{131}	100%
Sr^{89}, Sr^{90}	40%
Cs^{137}	20%
E. Amount of Herbage Cover	1 kg/m^2
F. Cow — Intake of Succulent Pasture Grass	
Milk production	40 kg/day
Fraction of ingested contaminant secreted in milk at peak output:	
I	10%
Cs	10%
Sr	1%

Table I-6 (continued)

G. Man – Only Those Assumptions Which Differ from ICRP Recommendations are Listed	
Milk consumption rate	1 liter/day
Thyroid mass	2 g
Fractional uptake (ingestion) for a child's thyroid	0.3

containing fission products and (2) exposure to ground-deposited fission products as a function of a unit release, as shown in Figs. I-12 and I-13, and respectively.* The paper[35] stated further that

> "The relative importance of the cloud and ground-contamination exposure paths depends upon both the release rate and the decay time. It is not feasible here to make a general comparison which examines all possible combinations of these two parameters. A limited comparison is possible by assuming appropriate values for release parameters... . The values selected assume that the exposure to the cloud is the same as the duration of release. The dose from [the] cloud is maximized by ignoring radioactive decay during cloud passage. The estimated doses at two distances for several exposure paths assuming [moderate inversion] conditions are listed in Table [I-7]**... .
>
> "The dose estimates [of] Table [I-7] are based on the quantity released as measured at one day [for a moderate inversion. However], different decay time [or meteorological conditions] would not change their relative magnitude."

The doses listed in Table I-7 were obtained from analytical calculations and appropriate numerical constants. The validity of the use of the mathematical model to calculate exposures was indicated by an adaptation of a portion of this model to the calculation of the area of ground contamination resulting from a number of actual ground-level

*Figs. 5 and 6, respectively, from ref. 35.

**Taken from Table VII of ref. 35.

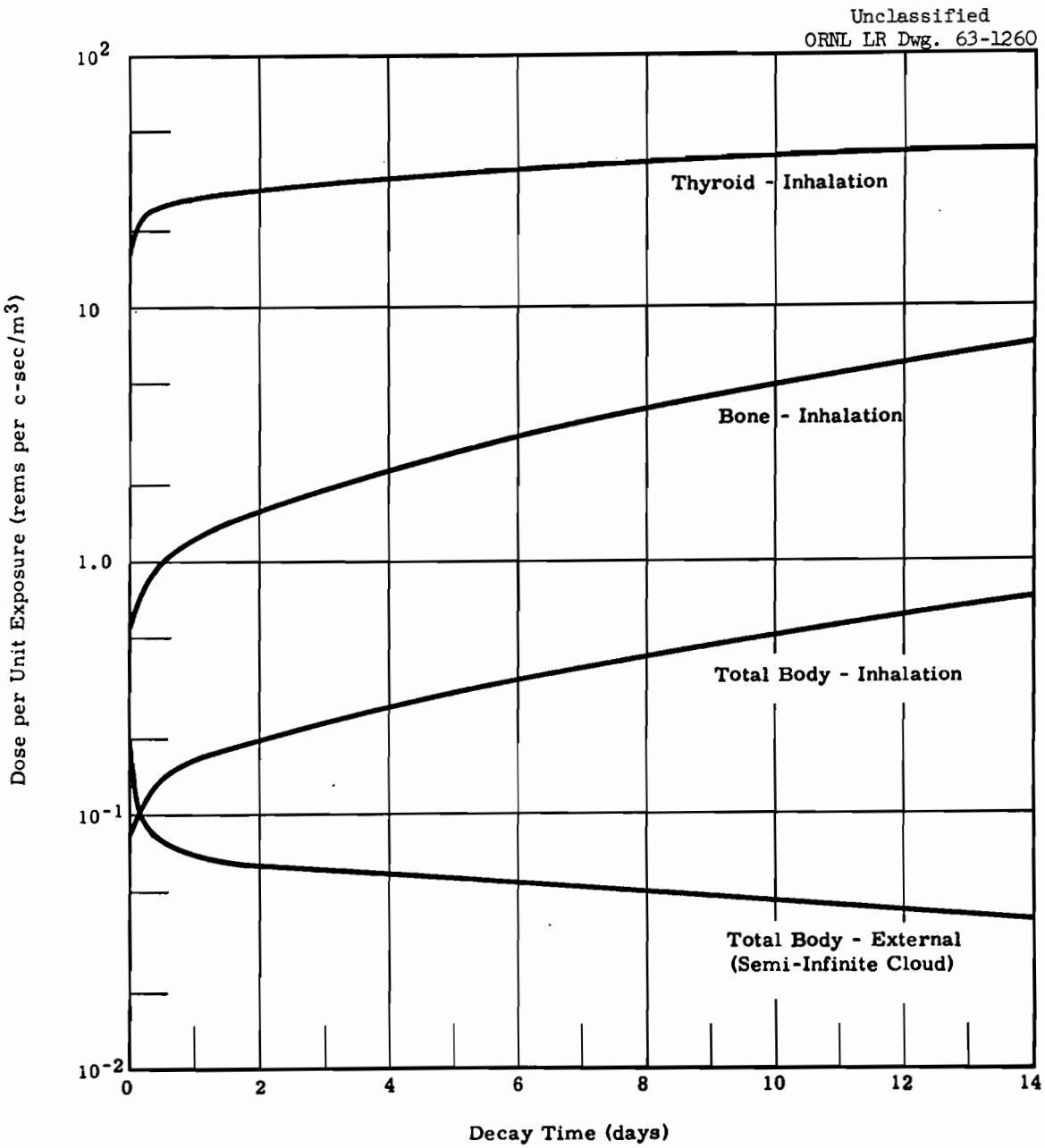

Fig. I-12. Dose from Exposure to Cloud of Released Fission Products versus Decay Time.

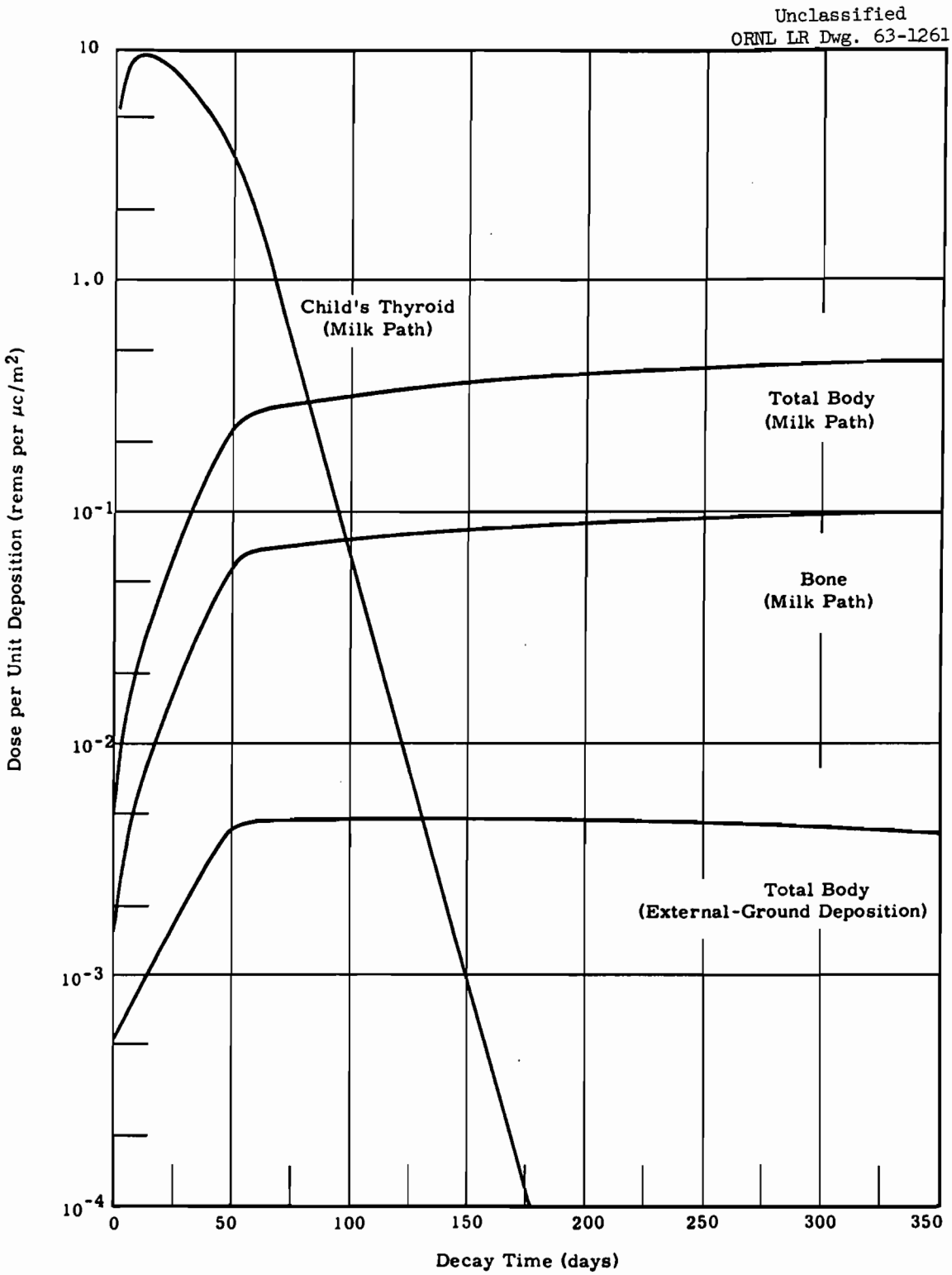

Fig. I-13. Dose from Ground-Deposited Fission Products Versus Decay Time.

releases and comparing these with experimental data. The data are presented in Table I-8.* The paper[35] states that

> "The experimental data are based on Hanford[36] and the Project Prairie Grass experiments.[37] The material used in the experiments was zinc sulfide (Hanford) and sulfur dioxide (project Prairie Grass). The agreement between the model and experimental data is good, especially if it is assumed that zinc sulfide, the material used for the larger areas, has a deposition coefficient equal to $\sim 3 \times 10^{-3}$."

Table I-7 COMPARISON OF ENVIRONMENTAL PATHS OF EXPOSURE (MODERATE INVERSION)

Distance (meters)	Source of Exposure[a]	Organ Dose (rems per 1000 curies of mixed fission products released at 1 day)			
		Total Body (External)	Total Body (Internal)	Bone	Thyroid
1,000	Cloud	3.2×10^{-2}	7.9×10^{-2}	0.60	11
	Ground	5.8×10^{-2}	0.65	0.18	7.8×10^{3}
	Cloud	1.3×10^{-2}	3.1×10^{-3}	2.5×10^{-2}	0.32
	Ground	6.3×10^{-3}	7.1×10^{-2}	1.9×10^{-2}	5.3×10^{2}
100,000	Cloud	9.2×10^{-5}	1.8×10^{-4}	1.4×10^{-3}	1.9×10^{-3}
	Ground	1.3×10^{-4}	1.5×10^{-3}	4.1×10^{-4}	1.3

[a]The internal dose from ground exposure includes the milk path only.

A somewhat complementary report was presented by Horan, Wehmann, and Schmalz[38] on the monitoring of radioactive effluents at the National Reactor Testing Station (NRTS). Included in this report was (1) a review of plant-locating practices at the site, where several facilities are grouped together to share common utilities but each group is isolated from other groups, (2) the surrounding population distribution, and (3) the physical characteristics of the site, such as meteorology, hydrology, seismology, and geology.

*Taken from Table A-IV of ref. 35.

Table I-8 COMPARISON OF MATHEMATICAL MODEL[a] WITH EXPERIMENTALLY DETERMINED AREAS

E/Q	$V_\Delta/\bar{u}_o$	Area (square miles)	
		Calculated Values for Moderately Stable Conditions	Experimental Data for Isothermal to Stable Conditions
10^{-3}	2.2×10^{-4}	0.0025	0.0012 – 0.0086
	3.4×10^{-3}	0.0020	
10^{-4}	2.2×10^{-4}	0.074	0.012 – 0.0066
	3.4×10^{-3}	0.029	
10^{-5}	2.2×10^{-4}	2.0	0.14 – 0.51
	3.4×10^{-3}	0.37	
10^{-7}	2.2×10^{-4}	270	18 – 31
	3.4×10^{-3}	4.9	

[a] V_Δ = deposition velocity, m/sec
$\bar{u}_o$ = average wind speed at ground level, m/sec
$V_\Delta/\bar{u}_o$ = the deposition coefficient, dimensionless
E = time-integrated concentration at ground level, $(\mu c/sec)/cm^3$
Q = quantity emitted, curies.

The atmospheric monitoring program at NRTS was developed to assure that no releases from an operation would result in doses at other sites or to the off-site public that would be in excess of the Federal Radiation Council recommendations.[39] Further, as a result of extensive meteorological research over the last six years, reliable calculations of the dispersion of radioactive material in air can now be made for distances of at least 5 miles during most meteorological conditions for releases lasting several hours.

The experience at NRTS was used in the evaluation of data from destructive tests, as well as nuclear incidents.[38] In particular, it was concluded from field data obtained following the SPERT I excursion on November 5, 1962 that:

"1. Release of radioactive materials to the atmosphere was less than 1 percent of total fission product inventory.

"2. Sutton's diffusion equations fit very closely the field data.

"3. The cloud gamma dose was negligible beyond 1500 feet from the reactor.

"4. No radioiodines were detected in field samples.

"5. Deposition velocity of gross fission products on a grass surface along cloud axis was 0.1–0.2 cm/sec and varied little with distance.

"6. Particle size data indicate that along the cloud axis 80 percent of the radioactive material was less than 1.5 microns in size."

In the case of the SL-1 accident, however, the vegetation and milk sampling gave the most sensitive indication of cloud trajectory and low-order contamination.

Extensive measurements have also been carried out at NRTS on the movement of radioactivity in the lithosphere, primarily as a means of monitoring the movement of liquid wastes discharged to the soil. In this program the water level in the soil, as well as the rate and direction of movement of isotopes of concern has been plotted.

An over-all review of the impact of activity releases, both routine and accidental, on agricultural interests surrounding nuclear sites was presented by Wortley of the Food and Agriculture Organization of the United Nations.[40] Of particular interest in this paper is the following plan for the surveillance and control of waste effluents:

"Where dairy cattle are involved then a milk sampling and assaying program will give the required information for adequate protection. This in turn involves the selection of farms likely to be in permanent production suitably oriented around the atomic energy plant and at varying distances from the plant. For example a suitable sampling plan could well include eight to twelve farms located on two rings of the order of one mile and five miles radius from the site. The samples need to be taken at regular

> intervals, bulked as conditions permit and assayed as frequently as analytical resources will allow. For I-131 with the short half-life of eight days it is necessary to assay at least fortnightly, while for longer lived Sr-90 and Cs-137, analysis of bulked samples at three-monthly intervals would be adequate.
>
> "In order to ensure that unexpected hazards have not been overlooked it is advisable to assay soil and herbage for a complete fission product range and alpha-active material such as Pu-239 at least once per year. These samples should be taken from the same farms as those which provide the milk samples... . If milk is not involved then a similar procedure should be adopted for the food at next greatest risk."

Wortley[40] also presented a comparison of the risks from atomic energy operations with those in other fields, including (1) other radiobiological exposures, (2) natural radiation sources, (3) labor statistics from the mining industry, (4) air pollution (smog) casualties, and (5) chemical-induced mutation. From this comparison it was concluded that

> "a proper perspective would cause much less public alarm about radiation genetic hazards, noting the relatively small risk involved and the thoroughness of the safety measures which will ensure that this situation can, and will, be maintained."

The general meteorological and waste disposal problems connected with site selection were treated in two French papers.[41,42] A paper[43] by Lieberman, Pack, and Simpson of the United States discussed the environmental dispersion and transport systems, specifically meteorology and geohydrology, as they enter into site evaluations.

Effect of Engineered Safety Features on Reactor Siting

Several papers presented descriptions and evaluations of engineered safety features of reactor installations.[44-47] Particular emphasis was given to containment provisions which, when coupled with a capability for reducing the concentration of activity within the contained atmosphere,

can be extremely effective in reducing the amount of activity released to the atmosphere.

Edwards and O'Neil[44] of the United Kingdom not only reviewed the essential assumptions involved in the analytical evaluation of various containment systems for water reactors but also presented results on the performance of single- and double-containment systems both with and without iodine filtration. The assumptions used in these calculations are listed below and the results are presented in Figs. I-14* and I-15:**

Initial release	
I^{131}	10^7 curies
Xe^{133}	2×10^7 curies
Deposited material	
I^{131}	With half-life of 1 and 3 hr (maximum of 6 1/2 half lives)
Xe^{133}	None
Recycle filtration effects	
I^{131}	Decontamination factor of 3×10^3
I	Filtration rate as specified (2, 2.4, 6, and 24 changes per hour)
Xe	None
Containment features	
Inner volume	10^6 ft^3
Outer volume	Not given
Inner or single containment vessel leakage	1%/day
Outer containment vessel leakage	10%/day

*Taken from Fig. 1 of ref. 44.

**Taken from Fig. 2 of ref. 44.

UNCLASSIFIED
ORNL-DWG 63-1702

SINGLE-SHELL CONTAINMENT VESSEL

INITIAL RELEASE TO CONTAINMENT ATMOSPHERE: 10^7 curies I^{131}, 2×10^7 curies Xe^{133}

ASSUMED LEAK RATE: 1% PER DAY

CURVE	ISOTOPE	HALF-LIFE	RECIRCULATION CLEANUP
①	I^{131}	3h	NONE
②	I^{131}	1h	NONE
③	I^{131}	1h	6 CHANGES PER HOUR
④	I^{131}	3h	2 CHANGES PER HOUR
⑤	I^{131}	3h	6 CHANGES PER HOUR
⑥	Xe^{133}		

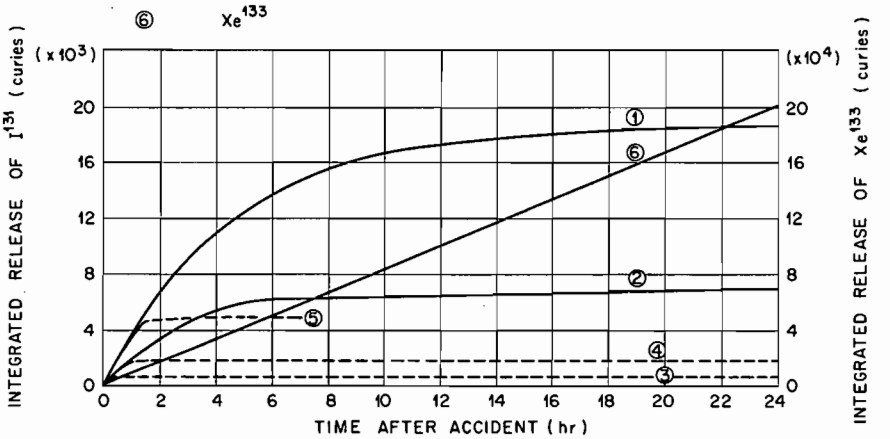

Fig. I-14. Performance of Single-Shell Containment Vessel.

UNCLASSIFIED
ORNL-DWG 63-1703

DOUBLE-SHELL CONTAINMENT VESSEL

INITIAL RELEASE TO CONTAINMENT ATMOSPHERE: 10^7 curies I^{131}, 2×10^7 curies Xe^{133}

ASSUMED LEAK RATES: INNER SHELL 1% PER DAY, OUTER SHELL 10% PER DAY

CURVE	ISOTOPE	HALF-LIFE	SECONDARY FILTER RATES
①	I^{131}	3h	NONE
②	I^{131}	1h	NONE
③	I^{131}	1-3h	ONE CHANGE PER DAY
④	Xe^{133}		NONE OR 10% PER DAY
⑤	Xe^{133}		ONE CHANGE PER DAY

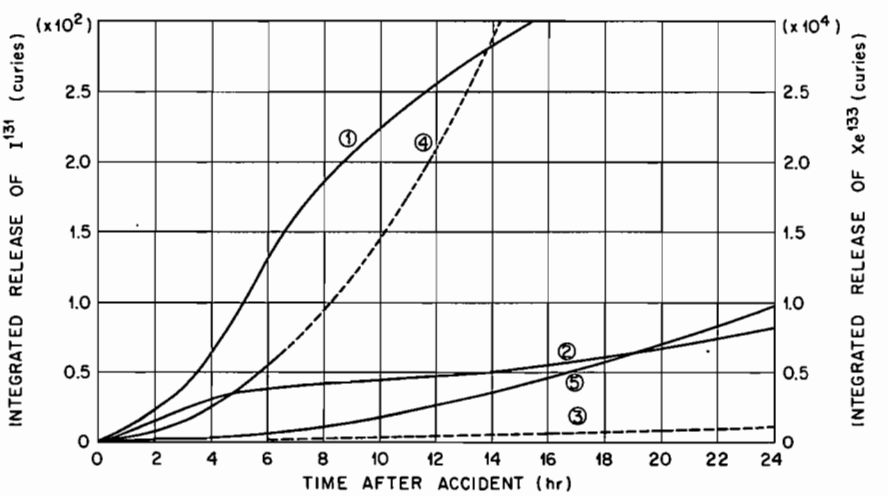

Fig. I-15. Performance of Double-Shell Containment Vessel.

It is apparent from Figs. I-14 and I-15 that in the cases examined the integrated iodine release was reduced by one to two orders of magnitude through the use of double containment, depending upon the time in question and the assumed deposition and recirculation, etc. It is further apparent that while deposition and filtration may be effective for reducing iodine concentration, they do not affect the noble gases, which may become controlling in situations where the over-all iodine decontamination factor is in excess of 10^3. A supplementary paper by Rushton and Edwards[45] presented a more extensive investigation of the performance of double-containment systems with illustrative calculations based upon the N.S. SAVANNAH design.

Two United States papers[46,47] indicated a trend toward dependence on some engineered safety features in the siting of power reactors. In particular, the paper by Ergen[46,48] cited several reactors, some already approved, in which engineered safety features (namely double containment and iodine filters) have been employed to justify site distances that are substantially less than those suggested by the existing siting guide.[21] In view of the incentives for locating power reactors even closer to the load centers, the power industry is quite anxious that sufficient confidence be developed regarding the reliability of engineered safety features to permit even greater reliance on them in the near future. Davis, Schmitz, and Smith[47] concluded that

> "The role of engineered safeguards has only indirectly been recognized to date. Our basic concept of the role of engineering safeguards needs considerable clarification. Research programs on nuclear safety are the first place that the recognition of the importance of engineered safeguards should be reflected."

Other Considerations

A paper on waste management[49] described the implications of waste disposal considerations with respect to site selection but concluded that, although of economic concern, it was not controlling. Binford[50] summarized in a straightforward manner the calculation of external and inhalation exposures from stack releases and discussed qualitatively the relative degree of conservatism in several of the elements considered. Smith[51] extended these considerations and advocated a site rating index that was similar to the British scheme but which would include probability factors for wind speed and direction, as well as population distribution.

Another paper by Edwards[52] reported on a survey of

> "experimental measurements and theoretical methods of calculating the maximum discharge from pipes containing high temperature water... . Recommendations are made for estimating the limiting flow rates to be expected in three broad divisions; (a) orifices, (b) short pipes ($9L/D < 50$), (c) long pipes."

Menoux, Bovard, and Doury[53] presented a review of environmental radioactivity monitoring.

Conclusions

The preceding discussion may have tended to emphasize the differences rather than the similarity of the nuclear siting practices in many countries. It is certainly not surprising that those countries with few nuclear installations and a limited nuclear technology are inclined to adopt conservative siting practices in lieu of a well-developed siting policy. The United States initially located many of its facilities at

the isolated National Reactor Testing Station in Idaho; however, nothing has happened that would contradict the conclusion that "most of the plants could have been located in a metropolitan area of any part of the country."[38]

Further, even the superficially divergent siting criteria of the United States and the United Kingdom, as interpreted by the appropriate licensing authorities, can and have produced siting practices that will become more similar as the industry grows. In particular, with limited land area in a more densely populated country, the British use relative site ratings. This system permits them to locate uncontained gas-cooled reactors on sites conceivably unsuitable for water-cooled reactors. This is possible because gas-cooled reactors are less subject to severe accidents than water-cooled reactors if the belief is accepted that their maximum activity release is reasonably well defined. In the United States a similar end has been obtained by the containment of water-cooled reactors that could not reasonably be located at remote sites, even this country with its vast land area. Hence an attempt has been made in the United States to establish criteria of a more absolute nature, such as distance versus power level. Licensing authorities in both countries are now under pressure to relax siting restrictions on the basis that the nuclear industry has a proven safety record and that engineered safety features which can be provided on nuclear facilities in populated surroundings can more than compensate for the additional risks involved. The reasons most frequently given for proceeding cautiously are that the total number of operating years of nuclear reactors is not great and that it is difficult to prove the reliability of engineered safety features.

In considering the port entrance requirements for the N.S. SAVANNAH,[54] licensing authorities in the United States were compelled to evaluate the reduction of the distance requirement and the acceptability of engineered safeguards with respect to reactor operation in populated areas. Application of such considerations was inevitable in any event. The extrapolation of similar features to stationary power plants in populated areas must likewise be acceptable on the basis of comparable risks. A trend toward the acceptance of safety features as justification for higher potential though less probable population exposures may be evidenced in the siting of recent large power reactors in the United States. Exposures from comparable accidents at these reactors would be much less drastic than in the case of the N.S. SAVANNAH, however. The real question is not whether the location of a power reactor with suitable safety features in a metropolitan area can be justified on the basis of comparable risk with other facilities, but whether the United States Atomic Energy Commission would agree that there was sufficient justification to warrant the greater potential exposure at this time, even though the probability of such exposure might be orders of magnitude less than in existing approved facilities. As stated by Davis,[47]

> "confidence in the engineered safety features is today the most crucial issue in the field of reactor safety and siting. There is a very great need at the present time for large-scale tests on a number of these features to increase confidence in their use.
>
> "For the present, those faced with site selection must be guided mostly by the precedents which have been established throughout the world and the attitude of the government agency which must make the final review on a given project. The judgment on the type of containment and the safety features which must be used at alternate sites to meet the government criteria for safety, whether the criteria is

> [sic] stated or unstated, is a crucial factor in the economic comparison of possible sites on a utility system."

There are several interesting anomalies in site evaluations, at least as performed in the United States and the United Kingdom. Although these anomalies may not appear to be significant when considered in light of variance of the other parameters accepted in such evaluations, each tends to assume more significance when absolute site evaluations are desired. Of especial concern in this regard, since it played a role in about half the papers presented, is that of iodine exposure both in terms of acceptable limits and exposure routes. There was some discussion of the iodine dose to the thyroid of an adult; the United States site criteria mention a value of 300 rem, whereas in a superficially similar context the United Kingdom uses 25 rem. Edwards and O'Neil[44] pointed out that the 25-rem dose does not enter into United Kingdom site criteria, as such, but it constitutes an "action limit," above which some remedial action definitely would be intended. They commented that United States authorities also would not sit idle until exposures reach 300 rem but would take action at much lower levels. They also expressed the opinion that the advance calculation of iodine exposure is much more uncertain than the advance calculation of whole-body exposure to rare gases, because the latter involves no reliance on retention by fuel elements, plateout, filtration, etc. Hence he felt that iodine exposure limits should be more conservative than whole-body exposure limits. This is accomplished by assigning to both limits the same numerical values, in spite of the lower damage potential of a given number of rem of iodine. On the other hand, by ignoring the

iodine ingestion exposure potential, presumably on the grounds that there is the option of controlling such sources of exposure after an accident, the United States practice overlooks what could be the most significant consequence of an accident, and incidentally, one which could be somewhat independent of an urban or a rural location. The United Kingdom hazards analysis, however, is concerned with ingestion exposures to iodine which, as indicated by Watson and Gamertsfelder,[35] may be orders of magnitude greater than inhalation exposures at the same distance from the point of release.

In closing the symposium, Mr. Pierre Balligand, Deputy Director General of IAEA, said that

> "the symposium had provided an opportunity for an exchange of information on past experience and the current practice in nuclear site selection as well as for an exchange of views on possible formulation of more widely, or even universally, applicable criteria. At the present stage, the Agency was not trying to frame any rules but rather to stimulate the widest possible exchange of information. The meeting had shown that before universal rules could be framed the main task was to collect the maximum possible information and study individual cases with the help of small groups of experts. IAEA had supplied such expert groups to advise on specific problems at the request of interested parties. [In addition] the meeting had confirmed the impression that the security of nuclear sites from the point of view of public health and safety was at present extremely well supervised all over the world."[55]

MEDICAL ASPECTS OF BERYLLIUM POISONING

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(Held over from Vol. 4, No. 3)

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II. ACCIDENT ANALYSIS

SHUTDOWN HEAT GENERATION

J. G. Delene

The energy release when a reactor is shut down following an extended period of operation is initially approximately 7% of the operating power and is in the form of radiation from decaying fission products. The amount of this energy release must be known as a function of time and prior operating history in order to ascertain the requirements for emergency shutdown cooling for the reactor core and for assessing the consequences of a loss-of-cooling accident. Knowledge of the gamma-energy release rate is also important in determining the shielding requirements for spent fuel. Two previous Nuclear Safety articles[1,2] have dealt with this subject. The purpose of this article is to explore further the existing information on shutdown heat generation and to review some of the more recent work on the subject.

If all information concerning the yields, half lives, and decay schemes of the fission products were known, it would be a straightforward task to sum the contributions from all fission products and arrive at a final determination of fission-product behavior. All information is not known, however, and the different extrapolations and interpolations that have been made have resulted in significant differences in shutdown heat generation rate as a function of time. Nevertheless, recent studies by several authors based on experimental measurements of total energy and more detailed data on additional fission products, particularly those with half lives greater that 10^3 sec, are in remarkably close agreement.

Total Fission-Energy Release and Time Behavior

The total fission-energy release is the sum of the associated gamma and beta energies. There is wide agreement[3-5] that the amount of energy involved in gamma release is approximately 7 Mev per fission. The agreement on the total beta-energy release is not so good, however. Estimates range from approximately 6 Mev per fission (ref. 5) to 8 Mev per fission (ref. 1), but a value of approximately 7 Mev per fission seems to be preferred. Since the total sensible heat from fission is about 200 Mev (ref. 6), approximately 7% of the operating power of a reactor is available after shutdown in the form of decaying fission products.

Estimates of the time behavior of fission-product decay rely for the most part on experimental information for times less than 1/2 hr after shutdown and on summation studies for times greater than these. Experimental determinations of the gamma energy spectra that were made at ORNL and were reported by Maienschein et al.[7] or Zoble et al.[8,9] are widely used for times ranging up to 1800 sec. The calorimetric measurements of Day and Cannon[10] provide an excellent comparison with summation studies in the range from 1 hr to several days. A comparison of these measurements with several other experimental determinations may be found in an article by Perkins and King,[3] who also present what is probably the best summation study to date. Since Perkins and King did not consider any isotope whose half-life was less than 1 min, their results are not considered accurate at times less than about 500 sec. Speigler[11] used Perkins and King's input data, with some refinement, and obtained results for infinite reactor operation, in contrast to the maximum of 1000 hr considered by Perkins and King. Speigler separated the decay

gamma spectra into 16 groups, as compared with the 7 groups of Perkins and King. This finer grouping is somewhat better for use in shielding calculations.

Stehn and Clancy[4] combined a partial summation study of their own with experimental information for short decay times to obtain a total energy-release-rate curve that is widely accepted. Perkins and King's total energy curve is very reliable for shutdown times greater than about 1000 sec and shows close agreement with Stehn and Clancy's results when compared for the same operating times.

There is virtually complete agreement in the more recent studies as to what are the best sources of information on the gamma-energy decay rate. Avery,[12] MacBean,[5,13] and Knabe and Putnam[14] all rely on the ORNL data reported by Maienschein or Zoble for times less than 1000 sec after shutdown and on Perkins and King for times greater than 1000 sec. Even though they rely on the same basic data there is a slight difference in the results of these authors. This comes from different methods of adjusting Perkins and King's data to infinite operation and adjusting the magnitude of the ORNL results.

With close agreement among the various authors, it is difficult to determine which work is most reliable. Shure [15-17] has reviewed rather completely the work of other investigators and compared their results. His analysis is the most recent and probably the most accurate. He recommends Stehn and Clancy's total energy results for times less than 10^3 sec and Perkins and King's results for times greater than 10^3 sec. In the case of gamma decay, Shure recommends MacBean's data for time periods from 1 to 10 sec after shutdown, Avery's results for times from

10 to 10^3 sec, and those of Perkins and King for times greater than 10^3 sec. Perkins and King's data are scaled up to infinite operation for each spectra. Shure presents recommended composite curves for infinite reactor operation. The total fission-product decay power following infinite operation, as given by Shure's analysis, is listed below and plotted in Fig. II-1 for a fission energy of 200 Mev per fission:

Time After Shutdown (sec)	Fission-Product Decay Power as Percentage of Full Power
10^2	3.3
10^3	1.87
10^4	0.97
10^5	0.48
10^6	0.268
10^7	0.121
10^8	0.0545

The decay power for finite reactor operation may be found from Fig. II-1 by using the following relation:

$$M(T_o,T_s) = M(\infty,T_s) - M(\infty,\ T_o + T_s)\ , \tag{1}$$

where $M(T_o,T_s)$ is the heat generation rate at shutdown time T_s following an irradiation period of T_o. The expressions $M(\infty,t)$ and $G(\infty,t)$ (the total gamma energy) are presented in the form of analytical expressions by Shure:

$$M(\infty,t) = At^{-a} \tag{2}$$

and

$$G(\infty,t) = Bt^{-b}\ . \tag{3}$$

Tables II-1 and II-2 give the values of the constants of Eqs. (2) and (3), along with the appropriate time intervals and the maximum deviations from the "correct values."

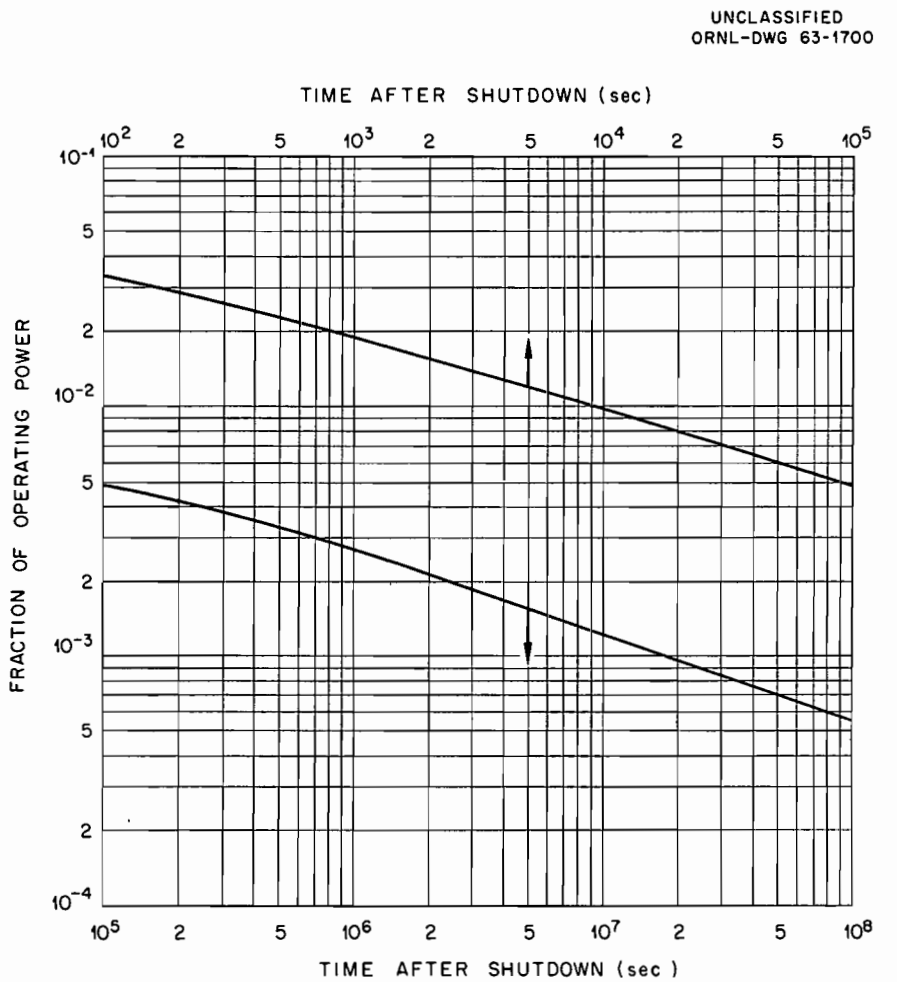

Fig. II-1. Total Absorbable Energy Released by Fission Products Following Infinite Reactor Operation.

Table II-1. Constants for Eq. (2) for Total Power Emission

Applicable Time Interval (sec)	A	a	Maximum Positive Deviation	Maximum Negative Deviation
$10^{-1} \leq t < 10^{1}$	12.05	0.0639	4% at 1 sec	3% at 10 sec
$10^{1} \leq t \leq 1.5 \times 10^{2}$	15.31	0.1807	3% at 1.5×10^{2} sec	1% at 3×10^{1} sec
$1.5 \times 10^{2} < t < 4 \times 10^{6}$	26.02	0.2834	5% at 1.5×10^{2} sec	5% at 3×10^{3} sec
$4 \times 10^{6} \leq t \leq 2 \times 10^{8}$	53.18	0.3350	8% at 4×10^{7} sec	9% at 2×10^{8} sec

Table II-2. Constants for Eq. (3) for Total Gamma Emission

Applicable Time Interval (sec)	B	b	Maximum Positive Deviation	Maximum Negative Deviation
$10^{0} \leq t < 1.5 \times 10^{2}$	6.71	0.1316	7% at 1.5×10^{2} sec	7% at 10^{1} sec
$1.5 \times 10^{2} \leq t \leq 10^{6}$	14.77	0.2919	7% at 1.5×10^{2} sec	6% at 4×10^{3} sec
$10^{6} < t < 3.5 \times 10^{7}$	536.4	0.55	4% at 2×10^{7} sec	5% at 4×10^{6} sec
$3.5 \times 10^{7} \leq t \leq 4 \times 10^{8}$	0.4285	0.1405	2% at 4×10^{8} sec	1% at 3.5×10^{7} sec

Other Sources of Shutdown Energy

There are sources of shutdown heat other than fission products, and the nature and relative importance of these sources vary for different reactors. For example, Np^{239} and U^{239} decays contribute some afterheat, as do the decays of some metals that may be in the reactor core, such as aluminum. The spontaneous fission of some of the transuranic elements could also be significant, and at short times after shutdown the heating from delayed neutrons is also important. Zubarev and Sokolov[18] suggest the use of the following equation for delayed-neutron heating rates:

$$P(t) = P_o \frac{\rho}{\rho + \beta} \exp\left(-\frac{\rho + \beta}{T} t\right) + \sum_{i=1}^{m} \frac{\beta_i}{\rho + \beta} \exp\left(-\frac{\rho}{\rho + \beta} \lambda_i t\right), \quad (4)$$

where

$P(t)$ = power at time t after an instantaneous reactivity reduction to the value of reactivity ρ,

P_o = operating power,

β = fraction of total neutrons that are delayed neutrons,

β_i = fraction of total neutrons belonging to the ith delayed neutron group,

T = lifetime of neutrons in the reactor,

λ_i = decay constant of fission fragment nuclides in the ith group,

m = total number of delayed groups.

Equation (4) should give reliable heating rates for most negative reactivity values.

With the growing interest in plutonium as a fuel and the importance of the thorium-U^{233} cycle in breeder reactors, it would be of interest to know how closely the fission products from Pu^{239} and U^{233} behave like

those from U^{235}. Petrov[19] studied the gamma radiation from U^{235} and Pu^{239} fission fragments at times ranging from 0.6 sec to 11 hr after a pulsed irradiation. He found nearly the same mean gamma energy and total yields of gamma energy per fission for both isotopes. Levockkin and Sokolov[20] analyzed the decay rates of U^{235} and Pu^{239} fission products for times from 2 hr to 1 year after irradiation. They found identical rates up to 200 days. At that time, a difference appeared that grew to 60% toward the end of the year.

More complete fission yield data on U^{235} and Pu^{239} are becoming available, but a comparison by Katcoff[21] of U^{235}, U^{233}, and Pu^{239} fission yields shows that the data are still incomplete. A recent survey by Ferguson and O'Kelley[22] on U^{233} fission yield data seems to place U^{233} at about the point where a summation study could be justified. At present, however, the shutdown heat curves for U^{235} should be used for all fissionable isotopes.

REVIEW OF KEWB PROGRAM

J. H. Marable

The Atomics International program of kinetic experiments on water boilers (KEWB) has been completed. The objectives of the KEWB program were, as described previously in Nuclear Safety,[23] to provide design data for ensuring maximum safety and to develop reliable analytical methods for predicting the dynamic behavior of solution-type reactors. In these respects the program has been highly successful. It is this reviewer's opinion, however, that the need remains for a deeper understanding of the phenomena involved in order to put the analytical methods on a firmer foundation. These methods are now based largely on semi-intuitive models.

As may be recalled,[23-25] a "water boiler" reactor is a small (10–30 liters), low-power (<100 kw), homogeneous reactor with a core of enriched uranium in aqueous solution; the core usually has a graphite reflector. In normal operation the temperatures are below the boiling point, and the absolute pressures are below one atmosphere. In the KEWB experiments, the gases evolved in the core were processed by means of an external plumbing system.

These reactors are of interest for research and training purposes because of their simplicity of design and their inherent safety characteristics, that is, large negative temperature and void coefficients. Since such a reactor is likely to be located in a highly populated area, it is important to understand the kinetic behavior in order to prevent core vessel failure and subsequent release of fission products to the environment.

Description of the Experiments

Two core vessels, A and B, were used for the KEWB experiments.[23-25] Core vessel A was a 12.3-in.-ID stainless-steel sphere, and core vessel B was a cylinder 12 in. in diameter and 36 in. long. Five core loadings were tested: (1) vessel A 85% full, (2) vessel A completely filled, (3) 18 liters of fuel solution in vessel B, (4) 26 liters of fuel solution in vessel B, and (5) 24 liters of fuel solution in vessel B. These first four cores were completely reflected by graphite, and the fifth was unreflected.

Instrumentation was developed and installed to give an accurate record of reactor period, reactor power, and pressures during each transient. The following data for each transient with a step reactivity input are tabulated in ref. 24: (1) reactor period, (2) peak power release, (3) total energy release, (4) energy release to peak power, (5) initial temperature, (6) initial pressure, (7) peak pressure at bottom of core, and (8) final temperature. The acceleration of core vessel B was also measured. The physical events of a typical large power burst in a KEWB core are described below in a quotation from ref. 24:

> "Following a large reactivity injection, the reactor responded with a power burst that reached power densities up to 180 Mw/liter of fuel solution and maximum fluxes up to 5 to 10 $\times$ 10^{16} nv. Radiation intensities in the room were sufficient to light the reactor room brilliantly through air ionization, even with the 2-ft graphite reflector in place. Plastic and water-filled objects in the room yielded intense Cerenkov self-images. Argon-41 production from each burst resulted in the reactor room atmosphere being at least ten times the recommended breathing tolerance and the building had to be thoroughly ventilated after each experiment.
>
> "For reactivity insertions up to about $1.25 the only effects of the burst were temperature rise, bubbling, and splashing of the fuel solution. These compensate reactivity

and are responsible for the self shutdown of the reactor. When the reactivity insertion was large enough to yield reactor periods of about 20 msec, however, in addition to the above effects, pressures in the reactor core were observed during the shutdown. This inertial pressure, resulting from the rapid production of radiolytic gas, increased with decreasing reactor period. The inertial pressure is the factor which limits the maximum amount of reactivity which may be safely released. In the KEWB reactor, allowing a safety factor of at least four from yield of the core vessel, reactivity releases producing pressures as high as 400 psi were studied. Typically, following the burst, the reactor power level would decay to a level between 50 to 60 kw and then decrease less rapidly due to heating with no coolant flow. For a few special runs the maximum reactivity was installed and the reactor left to its own devices for a period of half an hour. In these cases the reactor power decreased to about 4 to 6 kw in the first 5 min and then maintained this low power for the rest of the test. The reactivity installed was consumed by neutron leakage due to steam bubbles. When full cooling was applied to the core, the reactor behavior was similar, except the final power level at equilibrium was from 50 to 100 kw.

"The reactor was found to be stable for all loadings under every condition investigated. That is, no uncontrolled or divergent oscillations were encountered. During the "A" core tests it was discovered that with the core completely filled with fuel solution, chugging occurred when the reactor was operated at steady power levels greater than a few kilowatts. This was probably caused by the limited free surface area at which the liquid and gas could separate. The inability of the gas to escape freely would cause fuel solution to be carried up into the central tube of the overflow chamber, and therefore decrease the reactivity. As the power dropped with a corresponding decrease in gas formation, the fuel would again fall back into the core and the power would rise. The process would repeat every few seconds."

The experiments[24] indicate that the initial temperature, pressure, and power had only a small effect on burst magnitude and peak pressure. The burst was governed by the amount of reactivity and the rate at which it was inserted. Inertial pressures were produced only when the initial periods were somewhat less than 20 msec. The shortest period, 0.56 msec, occurred in unreflected core 5 with a corresponding peak power of 4000 Mw

and a peak inertial pressure of 400 psi. Possible explosions caused by the recombination of radiolytic hydrogen and oxygen[24,26] should be considered, but they do not present a serious threat to safe operation.

A Model for Kinetic Behavior

A kinetics model has been reported[24,27] by Dunenfeld that gives a fairly good prediction of the kinetic behavior of the fastest transients measured by the KEWB experiments. The model was developed to facilitate computation of the kinetic characteristics of designs differing substantially in configuration from that of the KEWB systems with greater assurance of accuracy than would be obtained from simple extrapolation of the existing data. In this respect Dunenfeld claims considerable success, although he is careful to point out weaknesses in the model.

The characteristics of the model are listed below:

1. The usual space-independent time-dependent kinetics equations are valid for the model, but a seventh group of delayed neutrons is added to simulate the neutrons that slow down in the reflector.

2. Reactivity feedback occurs by two mechanisms; first, a temperature dependence with a temperature coefficient of reactivity that varies linearly with the temperature, and second, a dependence on the void volume characterized by a constant void coefficient.

3. All power removal is neglected in the calculation of the temperature rise, and the heat capacity is assumed to be constant.

4. A threshold energy density below which the evolved gas does not produce void formation is assumed to exist. (This threshold energy density is indicative of a certain high degree of dissolved hydrogen gas

supersaturation in the fuel solution.) When the threshold energy density is reached at some point in the reactor, void formation begins.

5. The inertial force at the bottom of the core is assumed to be proportional to the solution density, height, and the second time derivative of the void volume.

6. Instead of giving a model for the details of void formation, the inertial pressure at the bottom of the core is made to fit an empirical curve-form that has a peak occurring 1.5 times the initial period after the threshold is reached. This peak pressure is assumed to be proportional to the average power density just when the threshold is reached.

In making the last assumption, Dunenfeld apparently forsakes attempts at a complete model of void formation. Such attempts are discussed in ref. 27. If the last assumption is simply a curve-fitting technique, it certainly should be viewed with suspicion when extrapolating to other systems. The limitations of the void model are exhibited by the fact that it was not applicable to core 2 because of core neck restrictions that limited the free surface of the solution and hence resulted in pressure higher than those found in corresponding transients with the other cores. For the cores in which the model was applicable, the range of validity of the void model was given[24] to be from periods of about 10 to 0.6 msec. These short periods form, of course, the range of greatest concern, and for these the model will predict the power and pressure with good accuracy up to peak power, with some error possible after peak power. Adequacy of the model is demonstrated by the comparison of theoretical and experimental results for a 1-msec burst in Figs. II-2 and II-3. The void model does not apply to the relatively mild burst induced by ramp reactivity inputs. The basis for the void model is presented in ref. 28.

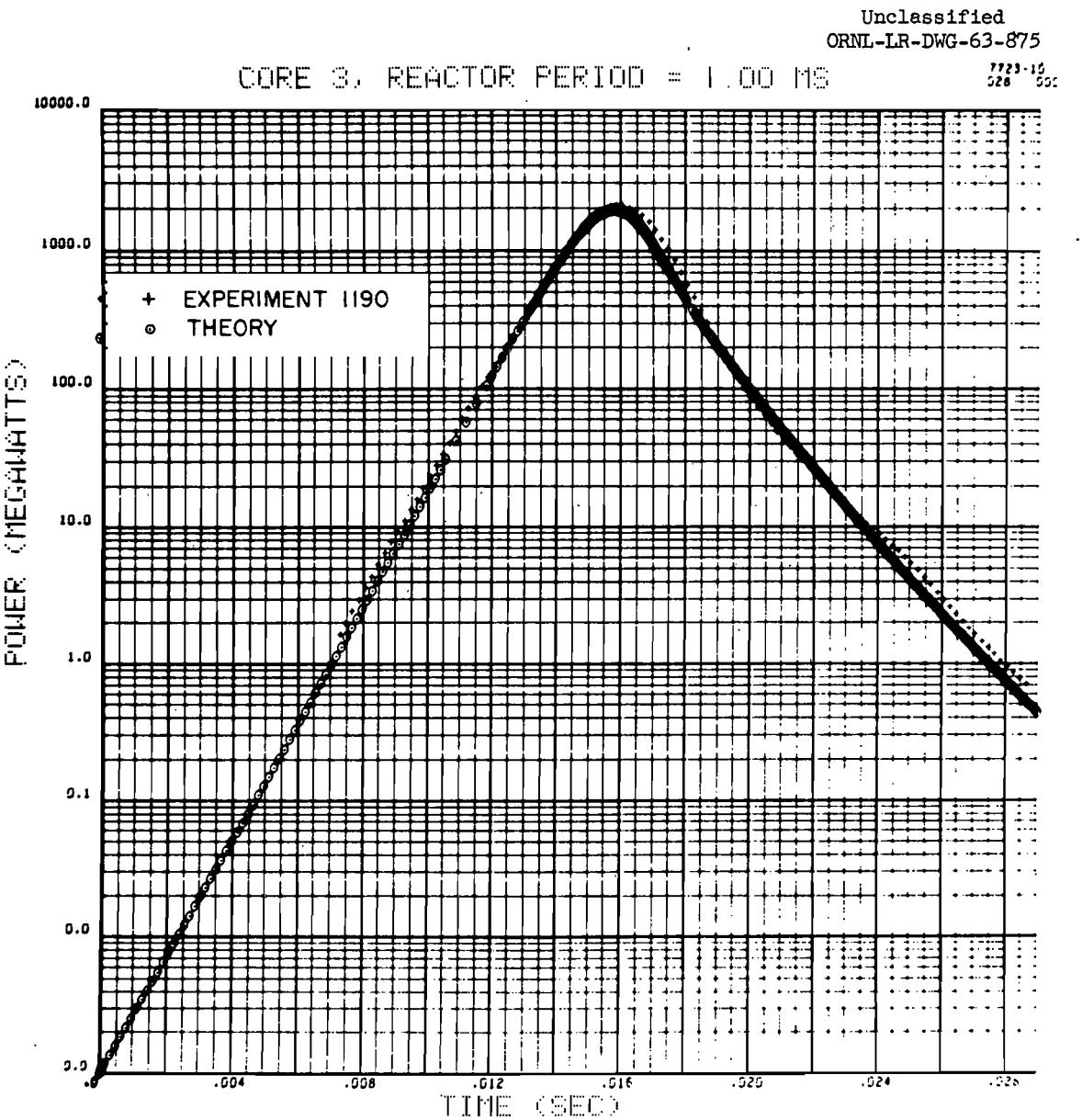

Fig. II-2. Experimental and Theoretical Power for a Reactor Period of 1.0 msec, Core 3 (from ref. 24).

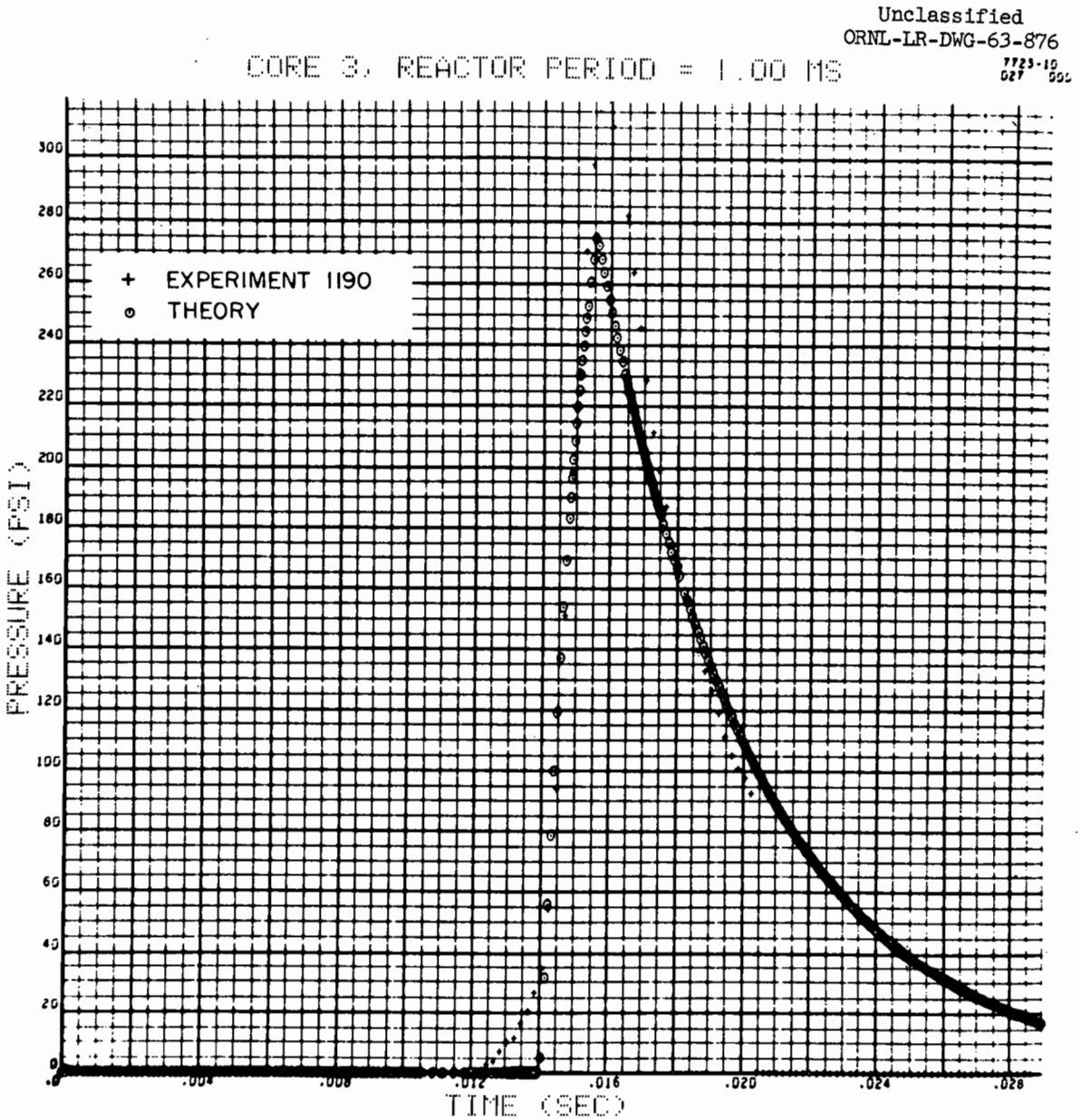

Fig. II-3. Experimental and Theoretical Pressure for a Reactor Period of 1.0 msec, Core 3 (from ref. 24).

Instructions are given[24] for applying the model to other systems, and a computer program AIREK has been developed[24,27,29] which solves the kinetics equations for the model. In addition to providing increased reliability in safety studies and hazards evaluations of existing and future aqueous homogeneous research reactors, Dunenfeld suggests[24] that the model "should prove a valuable tool in the study of chemical processing plant accidents."

ACTIVITY BUILDUP IN PRESSURIZED-WATER REACTOR SYSTEMS

C. F. Baes, Jr.

The buildup of activity is a matter of continuing concern in the safe and efficient operation of nuclear reactor systems. In a previous review article on this subject[30] the nature of the transport processes involved, the consequent buildup of activity in out-of-flux primary system components, and the methods of activity buildup control were discussed. The present review is intended only to indicate current significant trends in similar studies rather than to be a complete discussion of the abundant but diffuse and often non-quantitative information available.

Activity Transport Processes

The principal agents of activity accumulation in the primary systems of pressurized-water reactors are corrosion and wear products that originate at primary system surfaces, both in the flux and out of the flux, and are released to and subsequently deposited from the coolant. The particulate material produced is called "crud." Its principal constituent is magnetite (Fe_3O_4), the stable oxide of iron, under pressurized-water reactor conditions (i.e., 500°F, with dissolved H_2 present).[31] While the level of water-borne particulate crud is often low (<0.1 ppm), its specific activity can be high (e.g., $>10^7$ dps per gram of Fe^{59}, Co^{58}, Co^{60}, and Mn^{54}).[32] In systems operated at neutral pH, the major portion of the water-borne activity is sometimes found to be nonfilterable and is thought by some to be present as dissolved ionic material.[33]

Attention is presently being given to the exchange processes between these materials and system surfaces. The preferential deposition of crud

on in-flux surfaces has long been known. This results in loss of heat-transfer at fuel element surfaces, in the clogging of flow channels, and in the unreliable operation of mechanical systems. This effect has recently been discussed in several articles.[34-36] In particular, it has been hypothesized that preferential deposition on in-flux surfaces is caused by short-range beta particles, which induce electrostatic precipitation of crud particles.[37]

Studies of deposition, exchange, and release processes on out-of-flux surfaces are also being made. In general, the release rates are low[38] and difficult to measure accurately and unambiguously.[39] Such parameters as release rates and deposition rates, both in the flux and out of the flux, as functions of system structural materials and operating conditions are needed to account for and to predict behavior in reactor systems.[40,41] Perhaps the most fundamental process of all is the dissolution rate and dissolution equilibrium of magnetite, which probably proceeds by the reaction

$$Fe_3O_4(s) + 6H^+ + H_2 \rightleftharpoons 3Fe^{++} + 4H_2O$$

in neutral pH systems at 500°F. Judging from the levels of dissolved activity and from the conductivity, the equilibrium Fe(II) ion concentration in such water is of the order of 10^{-7} M. Attempts to measure directly the solubility of magnetite have thus far indicated only that the upper solubility limit is in this range.[42] This solubility is low, but if equilibrium is attained sufficiently rapidly, this process can account for much of the observed activity transport in neutral pH systems.[41]

Fission-product activities have, in general, not been as important a problem, but the potential hazard of sudden and extensive fuel release to the coolant has encouraged continued study. For practical purposes, fission-product releases have been distinguished[43] on the basis of the observed relationships among the activity level A, the fission yield y, and the decay constant λ of a given fission product, as indicated below.

Event	Activity Relationships
Recoil	$A \propto y\lambda$
Diffusion	$A \propto y\lambda$
Equilibrium	$A \propto y$

The first event, recoil, is caused by tramp uranium present on in-flux surfaces. The second, diffusion, evidently results from a relatively large cladding defect that provides good communication between the fuel and the coolant. The third, equilibrium, occurs with any release mechanism when release is delayed sufficiently that equilibrium between production and the decay of fission products in the fuel is maintained.

Activity Buildup

The accumulation of activity in out-of-flux system components is largely attributable to the accumulation of crud, either loosely or adherently deposited, in "crud traps," that is, dead legs, regions of low flow rates, or regions of change in flow direction. It is manifested by the presence of relatively high radiation fields after reactor shutdown.

The radiation levels encountered in several reactor systems were recently reviewed.[44] Selected data, presented below, indicate that activity levels after shutdown can be significant in piping, heat exchangers, and pumps of pressurized-water systems, whereas the steam piping, turbines,

and condensers of boiling-water systems show much lower radiation intensities:

	Intensity (mr/hr)
Shippingport 231-Mw pressurized-water reactor	
Boiler chambers, general area	2–30
Heat exchangers on contact	200
Inside heat exchanger plenum	3000
Purification system regenerative heat exchanger	400–800
Experimental Boiling-Water Reactor, 100 Mw, direct cycle	
Steam dryer	0.5
Main steam piping	0.2
Condenser steam dome	0.2
Condenser hot well	2.0

The lower radiation intensities of the boiling-water reactor are presumably the result of decontamination of the coolant as it passes from the liquid to the steam phase.[45,46]

The consequence of activity buildup in a pressurized-water reactor are well illustrated in the following quotation describing maintenance operations on the Shippingport reactor:[44] "Two major maintenance jobs involving removal of a reactor coolant pump and volute and the inspection of heat exchangers for tube leaks were successfully completed. Both cases required the rotation of personnel due to dosage accumulation. The radiation level inside the heat exchanger was about 2.5 r/hr. In both cases, the actual mechanical maintenance job was minor compared with the problems created by radioactivity. Many man-hours were spent in clean-up, shielding, precautionary measures to avoid or remove contamination, constant radiation control and personnel rotation."

Activity Control

The principal methods of activity buildup control are

1. Use of corrosion-resistant structural materials containing, where practical, low concentrations of elements that yield energetic, long-lived nuclides

2. Maintenance of desired water conditions by continuous side-stream purification and pH control

3. Decontamination.

The commonly specified materials of construction are stainless steel (in which it is highly desirable that the cobalt content be as low as possible), Zircaloy-2, Inconel, and carbon steel. The latter has been intensively investigated because of its inherently lower corrosion resistance. Preconditioning by exposure to high-temperature water has been found to reduce the crud release rate of carbon steel to acceptable levels.

The efficiency of ion-exchange purification is low when the activity removed is compared with that present on out-of-flux surfaces.[30] Inorganic ion-exchanger columns suitable for use at elevated temperatures offer the prospect of higher side-stream purification flow rates. A recent study[47] indicated that hydrous ZrO_2 is promising in this application. Magnetite beds have been shown to be effective as high-temperature mechanical filters,[48] and magnetite is also being examined as a high-temperature ion-exchange material.[41]

System decontamination is the last resort in the control of activity buildup because (1) it is difficult to find reagents and devise treatments that are compatible with the varieties of materials and conditions

found in pressurized-water systems and (2) the subsequent reactivation of the system with resultant accelerated corrosion during film formation would produce a new batch of crud. Common procedures include the use of alkaline permanganate solution to oxidize the chromium-containing protective oxide of stainless steel to soluble chromate or the use of buffered acidic ammonium oxalate—hydrogen peroxide solution to dissolve corrosion-product oxide and UO_2.[49] The latter treatment has an advantage in that the solutions resulting from decontamination may be evaporated to a much lower weight of solid residue prior to disposal than is the case with permanganate solutions.[50]

CHEMICAL REACTIONS OF GRAPHITE

S. Peterson

The combustibility of graphite is of recognized importance to the safety of nuclear reactors. Serious destruction could result from excessive temperatures in air-cooled graphite-moderated reactors and from air inleakage in high-temperature reactors cooled by other gases. As a consequence, there is a growing body of information on the reaction between oxygen and graphite.

Investigations of the oxidation of graphite have been reviewed both in Nuclear Safety[51,52] and elsewhere.[53] At a given temperature and partial pressure of oxygen, the reaction rate shows considerable variation among graphite samples with different histories, although both the rate and the variation in it decrease with decreasing impurity content in the graphite. The attack on the graphite is often uneven, being concentrated at deposits of catalytic impurities. Since in massive graphite structures, such as reactor moderators, conditions of temperature and gas flow are possible under which the heat of combustion is not adequately removed, the oxidation can become uncontrolled.

Other gases that can react with graphite include carbon dioxide, steam, and hydrogen. These may be present in graphite reactors as coolants, as a result of an accident, or as inevitable contaminants in inert-gas coolants.

Observations of Oxidized Surfaces

The random fine-grained structure and complicated porosity of reactor-grade graphite have made the interpretation of observed oxidation

effects difficult. Improved insight into the nature of oxidation is being obtained by oxidation of single crystals of graphite, followed by optical- or electron-microscopic observation of the oxidized surfaces.[54-60]

Oxidation of freshly cleaved surfaces of specially purified natural graphite crystals was studied by Hennig.[54] At 500 to 650°C he found no attack on the graphite in the cleavage planes when using any of several colloidal metal catalysts deposited on the graphite. On the other hand, in the neighborhood of metal particles deposited on other crystal surfaces, attack parallel to the cleavage planes produced channels.

Oxidation frequently produces pits, usually hexagonal, on the cleavage planes of graphite.[58,60] Hennig[54] observed this pitting on graphite that had been prepared from the nonpitting crystals by introduction of boron impurity or by introduction of imperfections by irradiation, deformation, or suitable heat treatment. Similarly, Presland and Hedley[60] found uniform pitting during oxidation at 550°C of irradiated coarse-grained natural graphite that before irradiation pitted only at grain boundaries. Hennig[54] found that the pitting was markedly increased by the presence of catalysts and that the size and shape of the pits were strongly temperature dependent. He deduced an activation energy 10 kcal/mole greater for the catalyzed penetration perpendicular to the cleavage planes than for the uncatalyzed burning of the pit walls. Related to this, Lang and his co-workers[61] found, in oxidizing specimens of pyrolytic graphite in air at 600°C, much greater attack on prism faces than on the basal planes. Hughes and Thomas[57] found that pretreatment with hydrogen at 1000°C markedly reduced the pitting in graphite oxidized at 850°C. Ryan[62] found that pitting in reactor-grade graphite, caused by agglomeration of

catalytic impurties, is markedly decreased by additional purification and aggravated by treatment with iron and vanadium.

Oxidation in Reactor Channels

Experiments simulating reactor channels and anticipated conditions were reviewed previously.[52] Oxidation rates of some 1-ft-long channels heated in the range 550 to 650°C are reported by Lewis and his co-workers[63] for various air flow rates and inlet temperatures. Kolb and Cottrell[64] have developed a computer code for determining the emergency cooling requirement after accidents in gas-cooled reactors. They have determined temperature distributions after conceivable accidents in the Experimental Gas-Cooled Reactor. The oxidation kinetics used in the calculation were measured for EGCR graphite by Dahl.[65]

Manganaro and his co-workers[66] have developed a computer code for calculation of oxidation transients in a bed of graphite spheres. New kinetic data on the oxidation of specimens of reactor-grade graphite are presented by Lang,[67] by Jacquet and Guérin,[68] and by Lang and Magnier.[69]

Reaction with Other Materials

Many gases can attack graphite. Of particular interest are steam, which could enter a coolant stream from a failure in a steam generator, and carbon dioxide, either as the coolant or an impurity in an inert-gas coolant. Fortunately these substances react endothermically with graphite, so the reactions cannot accelerate uncontrollably. The reaction products, however, are combustible and therefore may be hazardous where they meet the atmosphere. In addition, carbon deposited by reversal of these

reactions at lower temperatures may adversely affect heat transfer in the reactor systems.

A number of recent reports[67-72] give kinetic data on the reaction

$$C + CO_2 \rightarrow 2CO$$

for reactor graphite. Jacquet and Guérin[68] found the rate at 800°C about the same as the graphite-air reaction rate at 525°C. For temperatures from 650 to 870°C and pressures from 1 to 30 atm, Blackwood[70] expresses the rate by

$$\text{rate} = \frac{k_1 p_{CO_2}}{1 + k_2 p_{CO}} ,$$

where p_{CO_2} and p_{CO} are the partial pressures of CO_2 and CO, respectively, and k, and k_2 are constants. Below about 700°C, the reaction appears to be accelerated by radiation considerably more than the graphite-oxygen reaction. Several reports[73-78] have treated various aspects of the radiation-induced reaction.

Zumwalt and his co-workers[79] have considered the reactions that take place between graphite and impurity gases in a helium coolant. They have reviewed the reactions with hydrogen, nitrogen, steam, and carbon dioxide and have measured the rates of the reactions

$$2C + H_2 + N_2 \rightarrow 2HCN ,$$

$$C + H_2O \rightarrow CO + H_2 ,$$

and others. Effects of irradiation and catalysis by fission-product and corrosion-product elements were considered. The amount of carbon that would be transported as a result of these reactions is sufficiently small that levels of carbon monoxide, hydrogen, and nitrogen as high as 100 ppm can be tolerated in the coolant of the high-temperature gas-cooled reactor.

Graphite in contact with refractory oxides can be oxidized at sufficiently high temperatures. Klinger and his co-workers[80] measured the kinetics at various temperatures for the solid-state reactions of graphite with alumina, magnesia, spinel, beryllia, thoria, and titania. From the results, the authors calculated that 0.1% reduction would take place in 15 min for alumina at 1316°C, magnesia at 1404°C, spinel at 1532°C, and beryllia at 1707°C.

Summary

Continuing kinetic studies of the reaction between reactor-grade graphite and air or oxygen serve mainly to provide data on specific batches of graphite for particular reactors; they contribute little to the understanding of the reaction. Microscopic observations of the attack on single crystals, particularly the work of Hennig,[54,55] provide the most recent steps toward understanding graphite oxidation.

Increasing information is being obtained on the attack of graphite by other materials, particularly carbon dioxide, as operating temperatures sought for gas-cooled graphite-moderated reactors increase. It appears that these reactions would not cause a serious reactor hazard.

POOL BOILING BURNOUT

W. R. Gambill

The most important aspect of boiling heat transfer with regard to nuclear safety is the peak, critical, or burnout heat flux. At this heat flux a transition is made at the heated surface from efficient nucleate boiling to high-thermal-resistance film boiling that can cause the surface temperature to exceed the melting point of the solid. The general status of this problem for several boiling systems was reviewed in a previous issue of Nuclear Safety.[81] The current review deals with recent developments on pool boiling burnout only. Other systems will be reviewed in the future.

Influence of Heat Source Materials and Geometry

In an experimental investigation of the influence of heater material, diameter, and wall thickness on the critical heat flux for saturated pool boiling of demineralized water at atmospheric pressure, Ivey and Morris[82] obtained approximately 650 burnout values. The test section wires, tubes, and flat strips of nickel, Chromel, stainless steel, silver, and platinum were oriented horizontally, heated with direct current, and protected from physical burnout by a Wheatstone-bridge burnout detector. The study was motivated by the limited amount of data which have been taken under constant external system conditions while systematically varying the material and dimensions of the heater. The 109 pertinent burnout values already published, most of which had been surveyed earlier by Bernath,[83] were compared by Ivey and Morris with their own data.

Comparison of the average critical heat fluxes for 20-mil-wires of the five materials tested showed no significant variations in the critical flux with material. The maximum variation in the average critical fluxes of ±5.5% was similar to the experimental scatter characteristic of each material. The literature data cited show, however, that oxide layers give significantly higher critical fluxes than those associated with the bulk metal. The same observation was made by Bernath[83] with regard to aluminum heaters and was substantiated by Ivey and Morris with check tests made on 62-mil aluminum wire heaters. Bernath suggested that the higher critical fluxes with aluminum were related to the incremental energy required to drive off water from the hydrated aluminum oxide. It is concluded that while the effect of the bulk material is negligible, surface phenomena related to changes in the material at the surface can effect the critical flux.

Two materials were used to investigate the effect of heater diameter, 0.75- to 80-mil platinum wires and 36- to 104-mil-OD stainless steel tubes. Although the scatter in critical fluxes at a given diameter was too large for each type of test section, the average results for the tubes were in reasonable agreement with the curve proposed by Bernath,[83] which showed the critical flux to be unaffected by diameter, d, for $d > \sim 100$ mils, but to decrease with decreasing diameter for smaller diameters. The platinum-wire data were in approximate agreement with Bernath's curve at the intermediate diameters but diverged above Bernath's curve at both the smaller and larger diameters. This difference was attributed by Ivey and Morris to mechanical bubble dislodgment from the small-diameter wires ($d > 3$ mils) because of continuous vibration and to large induced

convection velocities in the region of the test section for the largest wire (80 mils).

The effect of heater wall thickness was investigated with 5- to 40-mil-thick Zircaloy and stainless steel strips (flat side horizontal) and 2- to 17-mil-thick stainless steel tubes. Although there was again considerable experimental scatter of the data, the results were in reasonable agreement with Bernath's suggested curve[83] of critical flux versus wall thickness. A lower thickness limit exists (~10 mils for the data of Ivey and Morris) above which there is no detectable variation of critical flux with increased wall thickness. It is suggested that very thin heater walls give lower critical fluxes because their small heat capacity allows larger temperature excursions as the bubbles grow on the surface before departure and consequently an earlier entry into the film-boiling regime.

Influence of Surface Conditions

The influence of surface conditions on the critical flux was further discussed by Costello (University of Washington), McEligot (Stanford University), and Aladyev (USSR) at the Second International Heat Transfer Conference[84] held in 1961-1962 in Boulder, Colorado, and in London. All three investigators cited instances of the apparent dependence of the pool-boiling critical flux on surface conditions. Aladyev's statements that "it has been experimentally found [that] the critical heat flux depends on the properties of the heating surface" and that the hydrodynamic analysis of burnout (ref. 85, e.g.) "does not in principle allow for the effect of the heating surface and thus of wetting,...whose effect on boiling is unquestionable" are representative. Aladyev cited the data of Labuntsov[86] in support of his contention.

In an extensive reply, Zuber, Tribus, and Westwater[84] reviewed the history of the hydrodynamic analysis of burnout and restated their conviction that both the critical heat flux and the minimum heat flux (at the Leidenfrost point) were determined by hydrodynamic conditions, that is, by the instability of the liquid-vapor interfaces, without regard to the nature of the surface. They made an attempt to classify "surface effects" into those of mechanical, thermal, and chemical nature. These investigators cited the substantial evidence that pool-boiling burnout is essentially insensitive to mechanical surface treatments, such as polishing and roughening, and rightly noted that the use of additives is a chemical process that may change not only the contact angle at the surface but also the characteristic of the vapor-liquid interface. Regarding changes in the surface material itself, however, they stated only that this is "a thermomechanical effect," without further comment on its specific influence on the burnout heat flux.

Costello and Frea[87] obtained interesting results pertaining to the surface effect with 1/8-in.-diam, type 304 stainless steel, horizontal, direct-current-heated semicylinders. Their test sections were pretreated with boiling tap water before the critical flux was determined in distilled water. The mineral deposits formed on the tubes in this fashion increased the critical flux by ~50%, and in two tests during which the water level was held a maximum of 1/8 in. above the heater in order to deposit a heavy coating on the surface without liquid available to wash it off, the burnout fluxes were approximately double "normal" values, that is, 1.16×10^6 and 1.18×10^6 Btu/hr·ft^2. These results, obtained at the University of Washington, are in substantial agreement with similar data obtained recently in tests conducted by Gambill.[88]

Although the issue regarding surface influences on the pool-boiling critical flux is still unsettled, it may be tentatively concluded that, although the hydrodynamic instability theories of burnout undoubtedly predict the most important experimental trends, surface variability does appear to constitute a significant second-order influence, as shown increasingly by recent comparative investigations.

Subatmospheric Pool Boiling

Patten[84] has briefly reported saturated pool data for the boiling of water from 18-mil horizontal wires under conditions for which very few critical-flux data presently exist, that is, at subatmospheric pressures. The Kutateladze-Zuber equation

$$(\phi_{bo})_{pool,sat} = KL_v\rho_v \left[\frac{\sigma g_c a(\rho_\ell - \rho_v)}{\rho_v^2}\right]^{1/4} \tag{1}$$

was found to correlate the data quite satisfactorily over the full pressure range of 0.7 to 14.7 psia, with the maximum deviation (14%) occurring at the lowest pressure at which ϕ_{bo} was 1.6×10^5 Btu/hr·ft^2. In Eq. (1), ϕ_{bo} is the burnout heat flux, K is a constant (taken as $\pi/24$ by Patten[84]), L_v is the latent heat of vaporization of the coolant, ρ_ℓ and ρ_v are the liquid and vapor densities, respectively, σ is the surface tension, g_c is the force-to-mass conversion constant, and a is the local acceleration.

It might be noted relative to Eq. (1) that Kutateladze pointed out[89] that, although the equation gives values that coincide closely with most experimental data when K is taken in the range 0.12 to 0.16, there are some apparent discrepancies, and he cited fused salts as a class of coolant that gave small values of K (<1). Similarly small K values, in

the neighborhood of 0.08, have been derived for some cryogenic liquids on the basis of the limited critical-flux data available.

Burnout with Cryogenic Fluids

The recent increase in interest in boiling heat transfer to cryogenic fluids is related to their use both as propellants in regeneratively cooled nuclear rockets and in cryogenic electromagnets. A timely survey has been prepared by the National Bureau of Standards[90] that covers the literature from 1940 to 1960. Data are tabulated and plotted as heat flux versus film temperature difference for helium, hydrogen, oxygen, and nitrogen separately for both natural-circulation and forced-convection heat transfer under nonboiling, nucleate boiling, and film boiling conditions. The plots reveal a considerable divergence between various sets of experimental data and between the data and predictions made with various correlations. The critical heat fluxes of these liquids are relatively quite small, probably because of the large vapor formation rates associated with their small latent heats of vaporization, and the critical fluxes are significantly overestimated by most correlations. Examination of the plots of experimental data indicates that the maximum nucleate-boiling heat fluxes that have been attained experimentally with helium, hydrogen, nitrogen, and oxygen are ~3,800, 35,000, 38,000, and 70,000 Btu/hr·ft^2, respectively. Considerably larger fluxes have been transferred in the fully established film-boiling region at much higher temperature differences.

METAL-WATER REACTIONS

E. G. Bohlmann

Most of the metals utilized in the construction of fuel elements for water reactors, that is, aluminum, zirconium, and uranium, have been shown by experience and experiment to undergo explosive reactions with water under certain circumstances. Such circumstances can be produced in a reactor core through loss of coolant or a power excursion. The calculated contribution of a metal-water reaction to the total energy release is indicated in Table II-3, which is from the work of Liimatainen and his co-workers.[91]

Table II-3 CALCULATED ENERGY RELEASE FOR A 135-Mwsec (128,000-Btu) BURST[a] INITIATING A METAL-WATER REACTION INVOLVING 100 lb OF METAL

Metal	Energy Release (Btu)	
	For 10% Metal-Water Reaction	For 100% Metal-Water Reaction
Al	75,500	755,000
Zr	28,000	280,000
U	10,700	107,000
Type 304 stainless steel	2,560	25,600

[a]The final destructive Borax I burst (ref. 92).

Except in the case of stainless steel, as may be seen from Table II-3, the energy from the chemical reaction could equal or exceed that from a nuclear excursion. Consequently a knowledge of the parameters that would produce or promote such a reaction and the kinetics of the reaction is of substantial importance to the safe design and operation of a reactor. As a result, a substantial amount of work has been done on

these problems by a variety of investigators and using various techniques. The investigations range from laboratory to in-pile experiments in specially designed and constructed reactors. Schaffer[93] recently presented a general review of the subject, and Weir[94] has reported on the TREAT in-pile experiments.

Condenser Discharge Experiments

Baker and his co-workers[95,96] adapted a technique in which condenser discharge through wires is used to produce shock waves, plasmas, and colloidal dispersions of metals in fluids to the study of metal-water reactions. The condenser bank used permitted delivery of energy to a maximum of 296 calories at 5000 volts to the circuit containing a 1-in.-long specimen wire. The nominal time for delivery of the energy to the circuit was 0.3 msec. Calibration of the energy delivered to the specimen by two different electrical measuring methods made it possible to calculate the metal temperatures achieved to within 100°C. This capability was confirmed by experiments in which the energy input required to partially melt a specimen was calculated. The physical changes observed were consistent with the predicted degree of melting. Separate reaction cells provided for both low- and high-temperature experiments. Information concerning the reactions was obtained from high-speed photography, pressure-transient measurements, residue examination, and hydrogen evolution, based on stoichiometric oxidation with no side effects, such as hydride formation.

Studies of the reaction of 30- and 60-mil zirconium wires in water were carried out with initial metal temperatures from 1100 to 4000°C and water temperatures from room temperature to 315°C. The experimental data

obtained indicated that several variables influenced the metal-water reaction. Wire size, particle size, water temperature, and initial metal temperature seemed to be important parameters.

The data were analyzed with the help of an analog computer. In developing the equations for formulating the simplified mathematical model used in the computations, the following physical picture of a reacting metal droplet in a water environment was used. The high-temperature droplet almost instantaneously becomes enveloped by a hydrogen-steam film diffusion barrier — reaction of 0.2% of the zirconium in the droplet would produce a gas-vapor envelope more than seven times the volume of the original droplet. As the reaction continues, loss of bubbles from the envelope is assumed to lead to a more or less stable volume or film thickness. Diffusion of water through this film thus becomes the initially limiting rate in the over-all reaction. As the oxide formed on the particle or droplet thickens, however, the parabolic rate law for the oxidation becomes limiting, with subsequent rapid quenching of the reaction. The formation of an adherent protective oxide under the conditions described is somewhat surprising, but metallographic examination of the particulate residues confirmed its presence so long as the temperature did not exceed that of the melting point of ZrO_2, that is, about 2600°C.

In developing the equations describing the reaction for programming on the computer, the following assumptions were made:

1. The simple parabolic oxidation rate law holds for spherical geometry.

2. The particle radius was unchanged throughout the reaction.

3. Specific heat and density were constant throughout the reaction.

4. An approximation of the temperature drop across the oxide film was adequate.

In addition, the Nusselt number describing both the gaseous diffusion rate and the rate of convective cooling was given the theoretical minimum value for spheres, that is, Nu = 2 in the absence of violent motion of the particle; and the emissivity of the oxide was given the theoretical maximum value of one. The remaining adjustable constants in the computer solutions, the pre-exponential factor, and the activation energy in the parabolic oxidation rate equation were obtained empirically from the computer solutions and by reference to previous zirconium-water studies.[97,98] The rate law deduced was

$$W^2 = 33.3 \times 10^6 t \exp(-45{,}500/RT) ,$$

where W is the milligrams of zirconium reacted divided by the surface area (in cm^2); t is time in seconds, R is the Boltzmann gas constant, and T is the absolute temperature.

Comparison of the computed and experimental results for zirconium revealed reasonable agreement for the extent of reaction; however, while the computations gave pressure-rise curves similar to those observed for the slow reactions, the explosive pressure rises were much more rapid than the computed rates. This inconsistency was explained on the basis that the assumption of a minimum Nusselt number equal to 2 was erroneous for the fast reactions. The Fastax motion pictures of experiments in which rapid pressure rises occurred showed high-speed "streaks," suggesting rapid (20 to 50 ft/sec) particulate motion. Such motion could be expected to remove the hydrogen envelope from around the blanket, with a resultant increase in the rates of the diffusion and cooling processes and a higher

Nusselt number. Evaluation of the total effect of such rate changes led, however, to the conclusion that only the rate of reaction, not the extent of reaction, would be increased. It should be pointed out that oxide film and droplet breakup from collisions with parts of the apparatus would affect both the rate and extent of the reaction. No attempt to include consideration of these effects in the computations was made.

On the basis of the experimental and theoretical (computer) studies, Baker and his co-workers concluded that the extent of the zirconium metal-water reaction was primarily dependent on particle size produced and water temperature, as shown in the Fig. II-4. The interesting dependence of the extent of the reaction on the water temperature, that is, the materially increased reaction that occurred when the temperature was raised from room temperature to ~100°C compared with the lack of effect of a further increase to 315°C, was explained on the basis that the primary parameter affecting the rate of diffusion of water vapor through the gas film surrounding a particle is the ratio $\Delta P/P$, where ΔP is the water vapor pressure and P is the total pressure. Since the experiments were all performed without significant overpressure, the pressure ratio was approximately equal to 1.0 in the case of the experiments with heated water, and the computer studies indicated that a value of 0.5 gave the best fit for the room-temperature data. The latter "was interpreted physically to mean that during reaction in room-temperature water the water surface facing heated particles is a dynamic mixture of water at the boiling point and water at room temperature." Explosive reactions were attributed to formation of particles of less than a critical size and the ability of the evolving hydrogen to propel such particles through the water at high

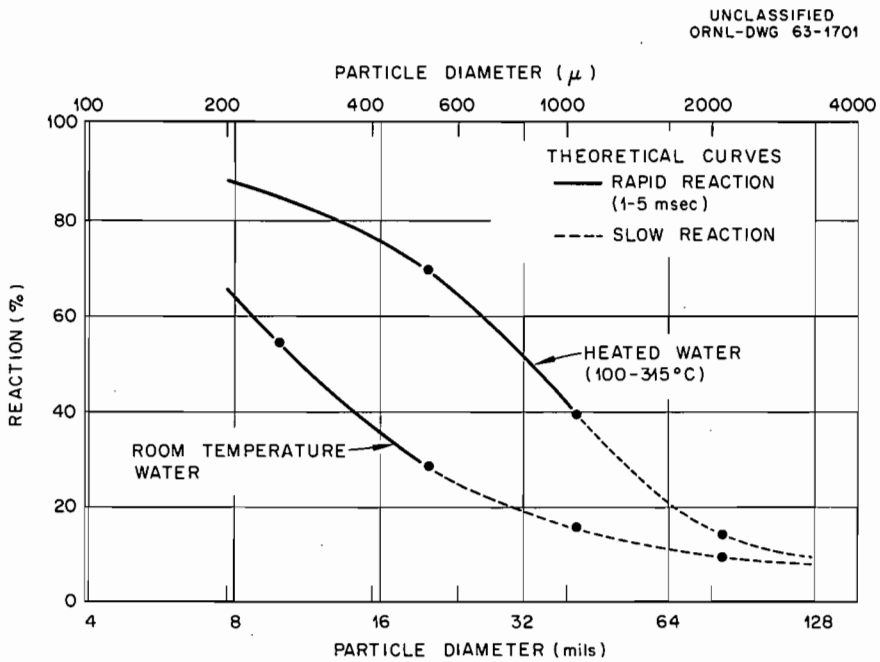

Fig. II-4. Extent of Reaction as a Function of Particle Diameter for Molten Zirconium Spheres Formed in Water.

speed and thus greatly speed the reaction through the diffusion regime. The critical size estimated from the residue examinations differed for room-temperature and for heated water, being ~1 mm in the latter and 0.5 mm in the former. Initial metal temperatures of 2600 and 1900°C were required to produce explosive reactions in room-temperature and heated (100–315°C) water, respectively.

The occurrence of explosive reactions in heated water with initial metal temperatures as low as 1900°C is substantially at variance with the critical temperature hypothesis proposed by Epstein,[99] which resulted from the observation that violent reactions for aluminum, uranium, and zirconium occurred when the initial temperature of the metal was approximately equal to the temperature at which the metal vapor pressure is 0.15 mm. For zirconium this temperature is 2983°C.

The method of estimating the extent of the reaction from residue particle size was applied to four TREAT experiments. Reasonable agreement between the calculated and experimentally measured results was obtained, as shown in Table II-4.

Table II-4 ZIRCONIUM-WATER REACTIONS IN TREAT

TREAT Experiment No.	Metal Reacted (%)	
	Experimental Values	Calculated Values
28	4.1	8.1
29	8.0	8.3
30	14.0	13.1
49	24.0	14.4

Similar condenser discharge studies[100] were carried out with uranium metal, stainless steel, Zircaloy, and zirconium-uranium alloys. The results obtained were similar to those for zirconium and consistent with the model described above.

Studies of aluminum-water reactions are also in progress.[100] The condenser discharge technique was unsuccessful for use with aluminum wires. Slow pressure rises indicative of burning propagation from a hot spot were observed at water temperatures of 200°C. In room-temperature water the wires broke in a few places, and the discharge energy was dissipated as arcs. Other metal heating techniques, such as levitation melting, use of an induction-coupled argon plasma, and use of an intense beam from a high-energy laser, are under development.

SL-1 Incident

The difficulty in producing significant metal-water reactions experienced by many investigators[101] was also demonstrated by the SL-1 incident. The nuclear excursion (130 Mwsec on a 4-msec period, peak power 19,000 Mw) which terminated operation of that reactor destroyed 47% of the central 16 fuel elements. Most of the destruction was attributed to normal melting, but 5% of the fuel (80.5 wt % Al, 2 wt % Ni, 17.5 wt % U) in these elements was estimated to have reached temperatures in excess of 2060°C, the temperature of vaporization of aluminum at one atmosphere. On this basis, in excess of 30 kg of aluminum and aluminum-uranium alloy was melted and somewhat more than 10% of this amount was vaporized. In spite of this and the probable violent disintegration of the fuel plates, no evidence for an extensive metal-water reaction could be found.[102] The

best estimate, based on analysis of the fines found in the bottom of the pressure vessel, was that 1.3 ± 0.5 kg of aluminum reacted. The energy generated in such a reaction is equivalent to 24 Mwsec or about 20% of the estimated nuclear energy release. The bulk of the material recovered from the bottom of the pressure vessel had the sponge-like appearance that results from the relatively rapid cooling of molten metal in contact with water or water vapor.

The relatively negligible metal-water reaction that occurred in the SL-1 incident, in spite of substantial provocation, taken in conjunction with the work of Baker and his co-workers[100] and that of Zelezny,[103] speaks well for aluminum alloys as materials of construction for water reactor fuel elements.

Curiously, considering its obvious significance, there appears to be no published examination of the implications of the negligible reaction encountered in the SL-1 incident with regard to metal-water reactions and nuclear safety. This seems a surprising omission, particularly since this incident has provided data on what Epstein in his excellent review[104] classed among the unknowable parameters, that is, how a nuclear incident disperses the reactants and initiates the reaction.

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III. CONTROL AND INSTRUMENTATION

SAMPLING CONSIDERATIONS IN MONITORING PROCESS-WASTE EFFLUENTS

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Sampling techniques are used in the monitoring of process-waste effluents from plants handling radioactivity for the determination of average concentrations of radioactive components in effluent gas and water, both on long-term and short-term bases. The average concentration (usually expressed in units of microcuries per cubic centimeter) must be known to fulfill the legal requirement of an accurate record of nuclides released to the public domain. The long-term trend is also useful in the planning of plant activities, since it furnishes information about the over-all cleaning efficiency and the amount of waste being generated. On the other hand, knowledge of the average concentrations on a short-term basis, that is, from minute to minute or hour to hour, is useful for process control. Short-term peaks coordinated with process runs allow analysis of the adequacy of the cleanup for specific processes or operations. In the event of a massive breakdown of cleaning and protective equipment that results in a nuclear incident, the short-term concentration record provides for early detection and furnishes information useful in detecting the source of the radioactivity that may be helpful in controlling the size of the incident. After a release large enough to require evacuation and subsequent cleanup, the record and collected samples should prove useful in analyzing the history of the release in order to prevent a recurrence.

Since the monitoring systems are concerned with large volumes of air or water, the means for taking samples is one of the first aspects to be considered in the design of a practical monitoring system. If a sample

is to be dipped out of a waste stream,[1] certain questions immediately arise:

1. Is the sample representative of the average or composite flow?
2. Was any material lost in the withdrawal?
3. How accurate must the sample be?
4. How often must the stream be sampled?
5. Is the total fluid flow of interest?

An attempt is made in the following paragraphs to supply some guideline for answering these questions.

Representative Sampling

A sample taken from a flowing stream may be expected to have both dissolved and suspended material in it. For the sample to have all components present and in the same proportions as the whole (a representative sample), the sampling method must not discriminate against the particle size, density, or physical state of the contaminants. For a flowing gas stream, the basic sampling rules suggest that (1) the stream flow should be well mixed at the sampling point; a rule of thumb is that the stream will be well mixed from five to ten pipe diameters downstream from the last tributary entrance to the stream; and (2) that the opening of the sampling pipe should face the gas stream.[2] The best condition would be to have the gas flow laminar at the sampling point. If it were laminar and the velocity of the air entering the sampling pipe were the same as the velocity of the air stream being sampled (isokinetic sampling), the suspended material would be collected as a representative sample. In order to attain an even more ideal condition, the sample should be taken

in a vertical pipe run where the flow is upward to avoid any loss of mixing from gravity dropout. In the practical case, however, flow is turbulent in 99% of the cases and the sampling conditions far from ideal,[3,4] therefore compromises are necessary. (Turbulent flow is defined as flow having a Reynold's number greater than 2000.)[5] If there is mismatch between sample flow and stream flow, there is some bias for or against the larger particles because of gas stream disturbance at the mouth of the sample pipe. Particles less than 5 μ in diameter tend to behave as a gas,[6] but the larger particles tend to move in straight lines because of their inertia and thus do not conform to the flow patterns at the sampler.

The opening of the sampling pipe should be as large as possible. Large fluffy particles, few in number but high in individual radioactivity, may get past any sampler and it is therefore best to have the largest practical opening. A sample opening larger than 3/8 in. in diameter[3] pointing into the gas stream and removing within ±20% of the average flow in the duct will probably collect a sample biased no more than ±25% in favor of the heavy particles.[6]

These considerations do not cover the case of a stack monitor where the flow at the collection point is gusty and variable. Attempting to maintain an isokinetic condition by automatically controlling the sampling flow soon gets out of hand, and the equipment and maintenance problems outweigh the possibility that the sample will not be exactly representative.[4] The size of the opening of the sampler is usually determined by the size of the pump available and is adjusted to give the best match between the average duct flow and the sampling-line flow.

Sampling for water monitoring follows the same rules, except that the water is almost always flowing in the horizontal direction at a slow rate and it is probably more loaded with uninteresting dirt. There is always a possibility that the inclusion of a screen or coarse filter on the water intake may keep out not only mud and leaves but also some appreciable portion of the total radioactivity. Screens, filters, and periodic back-flushing are necessary evils, and if a reasonable doubt exists that radioactivity in interesting amounts is getting past the monitor, no solution exists except to do additional mud and ecological sampling downstream. Such programs are already in existence at the national laboratories.

The only practical way of arriving at the operating characteristics of the sampling portion of a monitoring system is to perform spiking tests on the final installation in order to obtain numbers that define how representative and how proportional the sample is.[3,6]

Losses in Sampling System

Losses in conveying a sample from the point of withdrawal to the collector are most severe in the sampling of gas streams.[7] A sample taken by means of a pipe leading from the center of a duct or stack that takes at least one right-angle turn and runs several feet in the horizontal direction to a piece of filter paper will lose up to 75% of the sample in transit.[6] Since about 36% of this loss is due to the particulate matter being thrown against the wall of the pipe during a 90-degree turn,[4,6] the designer must keep the total number of degrees turned between duct and filter paper to a minimum and must also keep the horizontal runs at a

minimum length. It seems difficult to meet these rules in most cases, and 180 or 270 degrees of turn may be necessary, as well as 6 to 10 ft or more of horizontal pipe run. The net effect is the loss of sample and discrimination against the larger or heavier particles. This leads to the conclusion that the most sensitive and the most representative sample must be taken in the air stream itself. Where it is not practical to do this, some method should be provided for determining what the losses are in a given sample collector. As mentioned above, losses in the sample pipe may mean an error by a factor of 4 if not taken into account.

Sampling Accuracy

An attempt to set hard limits on necessary accuracy leads to some speculation. A superficial look at typical monitoring stations may lead to an over-all estimated confidence zone of ±40% or more. In terms of hazards to the general public, this is probably adequate considering the allowances provided for in the maximum permissible concentration levels;[8] however, with some care in design and by calibrating the monitoring stations in controlled experiments,[4,6,9] it should not be difficult to narrow the limits to ±20% or better, which is certainly adequate.

Even when the sampling accuracy is known, there is still the problem of interpreting the monitored data. As a rule of thumb, an alarm point is safely set at three times the normal operating maximum. This implies a detailed knowledge of operating experience and that the alarm is merely a demand to investigate. If the alarm never rings, the record of concentration must nevertheless be kept. A continuously collected sample withdrawn once a day or once a week in the absence of an alarm will satisfy the last

requirement. The maximum permissible concentration levels are based on a yearly exposure to the internal organs, with a restriction on the amount received during any 13-week period, which allows the averaging of samples over at least 13 weeks to determine whether the maximum has been exceeded.[10]

Sampling Frequency

The frequency and duration of sampling are functions of the total flow, the concentration in the stream, and the rate of change of the concentration.[3] If the activity in the stream is essentially constant, there is no need to sample at short intervals; however, since short-time bursts of activity are of interest for process control, even though the normal average concentration may be very stable, continuous withdrawal of a sample is indicated. This attitude may differ for water and for gas streams in terms of what can be done if a sudden rise in concentration takes place. A radioactive water sample discovered at the end of a shift would call for impounding water in a settling basin. A similar sample in a gas stream might indicate that an incident had been in progress for several hours. This leads to the suggestion that continuous monitoring, as well as continuous sampling, is necessary. An exception to the rule of continuous sampling occurs when a sample proportional to a very low water flow is desired and it becomes practical to devise a sampler that collects a fixed volume of water and then takes a fixed volume of sample before sending the batch of collected water on.[11]

Total Flow Measurements

Measurement of the total flow in an air or water stream is necessary almost without exception. The sampling rate should be proportional to

the total flow rate in order to avoid a bias caused by large changes in the flow rate. Water streams may be subject to rather wide variations in flow as one process or another goes on or off the line. If the flow rate is measured and controls the sampling rate, the correction for flow variations is automatically made. The total amount of activity (in curies) emitted during a given time interval can also be calculated from this information. Air streams do not normally vary over long intervals, except as they may be off or on. The measurement of flow rate is necessary to attain some velocity match in the sampler, and the total amount of air passed through the sampler to collect a given weight of particulate sample must be known to calculate the concentration.

The consideration of total flow measurements also introduces a question as to how large a sample is needed. If the sample is accurately proportional to flow, then almost any convenient amount is the answer to the size of the sample. Two gallons a day from a 10,000-gal/day stream might be convenient, with a test tube filled from the well-mixed 2-gal sample being adequate for the laboratory. Since early detection of changes in the activity is necessary to allow time for taking corrective action, continuous monitoring is again indicated and the determination of how much to sample will depend on the sensitivity of the available monitor. A given air monitor, for example, will take a known time to reach a given alarm point for a given air concentration of particulate nuclides. If this time is too long, the air flow in the sampler must be increased to meet the time-to-alarm requirement.

Conclusions

A philosophy of effluent monitoring is apparent from consideration of the facts stated above. Effluent streams, liquid or gaseous, need two channels of information at each monitoring point. One channel that takes a continuous sample and provides a continuous record of the concentration of radioactivity in the stream should be provided to give short-term information for alarm and analysis of cyclic variations. A second channel that also takes a continuous sample but consists of only a sample-collecting system should be provided for the long-term averaging of concentrations and for collecting material for detailed analysis.[11,12]

It should be apparent that much thought should be given to the design of the sampling portion of a monitoring system, even though it may consist of only a short piece of curved pipe, since there exists the possibility of orders of magnitude errors as a result of losses in collection and in biasing the sample in the process of collection so that it is no longer representative. The design philosophy should therefore also cover the inclusion of methods for testing permanent monitoring installations by spiking the monitored gas or water with the materials expected in the stream or good models of them.

A final word concerns the interpretation of the maximum permissible concentration term. The maximum permissible concentrations listed in the Bureau of Standards Handbook[10] pertain to the ingestion of radioactivity in water or air. A stack monitor or a water stream monitor may therefore indicate activities several decades above the permissible values, unless the additional dilution before the effluent is actually available for use by people is considered. Under these conditions, effluent monitors fall

in the category of the first line of defense, with environmental monitoring systems at ground level some distance from the plant as the second line. A definitive determination of whether the published maximum permissible concentration levels are truly being met is obtained from the ground-level monitors. The setting of alarm levels in effluent monitors must therefore take into consideration the additional dilution to be expected and the efficiency of the detector, as well as the deposition of activity in the sampling apparatus.

DIGITAL COMPUTER CONTROL OF POWER REACTORS

C. W. Ricker

The general view expressed in the papers presented at an American Nuclear Society Symposium sponsored by the Pittsburgh Section on February 24, 1962 on "The Role of Computers in Nuclear Power Plants" was that digital computers have the potential for providing (1) economic benefits by improving operating efficiencies and control and (2) greater safety. Although digital computers are at present only used for monitoring and data logging, many reactor engineers consider that completely automatic computer control of nuclear plants will be attained in the near future. Toward this end, the Canadians will install, in the fall of 1963, an on-line computer on the CANDU reactor at their Douglas Point plant.[13] This computer will monitor the outlet temperature, control the temperature and output power of the plant, control chemical reactor poisons, control flux tilt, monitor for fuel failure, and control turbine runup. Another step in this direction is the digital computer being installed in the Big Rock Point Nuclear plant at Charlevoix, Michigan.[14,15] In addition to performing data-processing functions, this computer will provide control-rod scheduling for maximum burnup and reactor calculations that will lead to increased power density. This computer installation is considered by its designers to be the forerunner of a completely automatic control system for future nuclear plants.

There seems to be no question as to the economic benefits to be derived from automatic data logging and processing. The economic benefits to be derived from completely automatic computer control of nuclear plants, however, are not immediately obvious, at least to this reviewer.

As an example, Lewis[16] stated that the completely automatic computer control system envisioned for CANDU will not remove the human element; rather, it is anticipated that the operators will be replaced with highly trained personnel to observe and monitor computer operations. This statement will probably be true for reactor plants other than CANDU. It would appear, therefore, that if significant economic benefits are to be derived from completely automatic computer control, such control must allow for a closer approach to the physical limits of the plant than is now possible with more conventional control systems. This move to higher performance will, of necessity, require adherence to established safety criteria for nuclear plants. If computer control can accomplish this, economic benefit will be derived.

Although the general view is that completely automatic control of nuclear plants will be realized, there is considerable difference of opinion as to how this may best be accomplished. Some people argue that both nuclear power plants and computers are in the experimental stages of development and that to combine them at this time would lead to disproportionate complexity, which would only delay the attainment of completely automatic computer control. On the other hand, there are those who believe that the experience gained with computers on nonnuclear steam-electric generating plants has provided the necessary information to allow a straightforward application of computers to nuclear plants.[17] For example, it has been stated that the greatest difficulty associated with computer control on conventional plants has been that the engineers do not have sufficient understanding of how the plant works to program the computer.[18] It is argued, however, that programming control computers

for power reactors is not a problem, since an adequate description of the reactor system can be made by the reactor engineer who has been forced, by strict safeguard requirements, to develop a sensitive understanding of his system.

Consider first the argument that application of computer control to nuclear plants is premature. The question can logically be asked, "How can the end of the experimental development of computers or reactors be determined." Recognizing the increase in complexity, it seems to this reviewer that simultaneous development along both lines will, in fact, lead to an earlier attainment of completely automatic control in that it will more quickly become apparent what degree of understanding of the reactor is required and what limitations are imposed on the reactor by the computer. No matter how the end of experimental development is defined, these questions will remain unanswered until the computer and reactor are actually brought together.

Consider next the argument that the reactor engineer knows more about his system than the nonnuclear plant engineer. If this is so, is this better understanding on the part of reactor engineer really adequate? In general, the reactor control engineer has been pressed in those areas which involve personnel and plant safety but has not been pressed for highly precise and accurate control. Since most reactor systems are able to tolerate large margins of error, there has not been the need to provide and process highly accurate signals from the reactor. The questions now are whether these same signals are adequate for computer calculations and whether, on the basis of these signals and calculations, plant performance can be safely improved. It is the opinion of the

reviewer that computer control can provide better and more efficient control and thereby improve plant performance if, and only if, the information provided to the computer is a complete, high-quality representation of the system. It follows therefore that it is necessary to critically evaluate the fidelity of (1) the sources of information for the computer, that is, the sensors, and (2) the processing of the information provided by the sensors. Further, it is necessary to precisely define the amount and type of information required for the computer to perform its proper function. It is not obvious that the sensors or sensing techniques presently used on nuclear plants are suitable for economically beneficial computer-controlled systems.

Nuclear plants such as CANDU and Big Rock Point have a multiplicity of sensing elements in or around the reactor core in order to obtain the necessary amount of information for computer control. On the other hand, the reactor core designer whose objective is also to design high-performance systems cannot tolerate more than a few sensors in or near the reactor core. Thus, high-performance reactor computer control systems requiring a multiplicity of sensing elements are incompatible with high-performance core design. If, however, a high-performance reactor computer control system could be designed which would require only a few sensing elements, this incompatibility might possibly be eliminated. Therefore, the input information required by the computer and the availability of this information from the reactor should be very carefully evaluated with the objective of reducing the number of sensing elements in mind. For instance, in the past, only one or two bits of information have been extracted from a signal where, quite possibly, many bits of

information were available. As an example, the fluctuations of the signal from an ionization chamber positioned near the core may contain a great deal of valuable information. It may prove profitable to analyze these fluctuations rather than to reject them with smoothing filters. Also it may be possible to effectively use the signals initiated by gamma radiation, which have been considered objectionable and useless for control purposes.

FUEL-ELEMENT FAILURE DETECTION

J. L. Kaufman

Several significant contributions to the "state of the art" of detection of burst fuel elements have appeared in the published literature within the last 18 months; they are reviewed here in a continuation of the series of reviews on fuel-element failure detection.[19-21] Of particular value are the photoneutron monitor developed by Lapsley and Dexter[22] and the fast gamma scanner described by Paul.[23]

The photoneutron monitor developed at Savannah River[22] samples the reactor gas, whether the primary coolant or the blanket gas, and holds it for N^{16} decay. The gas is then passed through a pipe surrounded by a tank of heavy water. Immersed within the tank is a thimble containing a BF_3 proportional neutron counter. High-energy gamma rays from fission-product gases react in the D_2O to release photoneutrons. Common background activities from the irradiation of reactor structural materials are not involved, since the threshold energy for (γ,n) reactions in D_2O is 2.2 Mev. The ultimate in simplicity is achieved with this device; there are no moving parts, and the BF_3 counters require no elaborate discriminators for background rejection.

In 1960, Paul of Hanford filed an invention report[23] describing a rotating gamma-ray spectrometer. The scintillation detector was mounted on a large turntable to observe the activity in 100 or more shielded water-sample chambers. Each scan was made at a rate of 1 rps, and a second spectrometer running at a much lower speed was used for verification or continuous observation of a suspected channel. The details of the final

design and operation are given in a report by Stringer and Sheen.[24] While rather complicated, this dual scanning system offers one of the best solutions to the surveillance of multichannel water-cooled reactors.

Baum has suggested a system for the monitoring of fission products exhausted from direct-cycle reactors in nuclear ramjets or rockets.[25] Two collimated gamma detectors, spaced along the exhaust line, are used to monitor the rate of increase in delayed coincidence. Variations in reactor power and radiation from the reactor are rejected, and only the flow of increased activity past the two detectors signals a fuel and fission-product release into the atmosphere. A paper[26] by Melgard of Tracerlab, prepared for the Air Force Special Weapons Center, gives details of methods for analyzing sample filters for uranium, beryllium, and fission products released into the atmosphere near nuclear rocket test sites.

A progress report[27] from Savannah River on the Heavy Water Components Test Reactor describes the proposed use of delayed-neutron monitors on the main coolant and two test loops because of their fast response. Use of an improved multiport valve of the type used for the Shippingport pressurized-water reactor was also announced.[28]

Several short reports on the performance of the failed-element detection and location (FEDAL) system[29-31] have been issued by the Shippingport reactor personnel. Experience was summarized in a report[32] covering the time from its installation through Core 1 Seed 3 loadings. The delayed-neutron monitors have decreasing signal-to-background ratios as the Pu^{239} and Pu^{241} build up from the irradiation of natural uranium in the Zircaloy cladding. Two small failures have been detected and located,

but only by verification during startup when the fission products were released at a higher than normal rate as the waterlogged elements were being boiled out. Close comparison with the flux plot is required to detect a failed element. A statistical analysis of delayed-neutron detection for water-cooled reactors was reported by Dudziak and Schmucker.[33] After reviewing many papers on the subject of delayed-neutron detection in water-cooled systems, the reviewer found that the performance of such systems was, at best, marginal; that is, these systems require too much time and effort for the confirmation of a failure. The reviewer also found repetition[34] of previously published and reviewed studies on this subject.[19]

More complete details of the design of the precipitation monitors used on French gas-cooled reactors, previously reviewed in Nuclear Safety,[21] are outlined in a recently translated paper by Megy and Roguin.[35]

REACTOR SELF-SHUTDOWN MECHANISMS

R. S. Stone

Requirements of economy for power reactors and flexibility for research reactors have led to the evolution of a number of reactor systems which, because of large reserve fuel supplies, undermoderation, overmoderation, or unusual modes of operation, possess the potential for large, rapid additions of reactivity. To assure termination of accidental excursions in such reactors, electromechanical safety systems have been developed with ever shorter response times. Within the limitations of a given speed of response, confidence in the system is raised through the use of multiple rods and redundant information channels. Although the speed of action of safety systems has improved, mechanical inertia and demagnetization delays provide certain lower limits on the interval of time within which neutron poisons can be added to a reactor. In order to extend the range of the mechanical device, work has been done on power-sensitive fuses that inject poison into the reactor in an explosive manner.[36,37] While such devices might, in certain instances, have useful protective features, they are very little faster than the best available electromechanical safety-rod systems and are decidedly harder to deactivate after resolution of the emergency.

In this situation there is justifiable interest in developing self-limiting principles for reactor design that would penalize a rise in power by a large and immediate loss of δk. Negative Doppler coefficients have received increased attention,[38,39] and the demoderating behavior of the TRIGA reactor[40] when subjected to a rise in temperature is a particularly good example of designed-in protection. The laws under which such

shutdown mechanisms work are well understood, and the magnitudes of the effects can, at least in principle, be calculated.

Great interest has been shown in self-shutdown effects developed in water-moderated reactors of various types. A continuing program of SPERT experiments[41-45] has clearly demonstrated that such reactors do have self-limiting mechanisms that arise from fuel expansion, moderator expansion, and steam void formation. These mechanisms have been shown to be capable of reversing a 10- or 20-msec period before core damage occurs, thereby providing the prompt shutdown effect sometimes necessary to back up a relatively slower electromechanical safety system. However, since the faster electromechanical systems do somewhat better than the self-shutdown mechanism in such experiments,[44,46] it appears that optimized mechanical safety systems must remain as the ultimate source of fast-transient protection until self-shutdown effects can be more efficiently exploited. Nevertheless, major advantages would accrue from the development of inherent protection in these water-moderated reactors or from the enhancement of self-shutdown mechanisms already present.

Self-Shutdown Phenomena in SPERT

The enhancement of existing self-shutdown mechanisms is hampered by an imperfect understanding of their mode of operation. Steam void formation is conceived to be the dominant mechanism in water reactors, but steam-generation rates coupled with measured void coefficients do not yield sufficient negative reactivity to account for the observed shutdown potential.[41,46] Experiments with pressurized systems in which no steam is permitted to form also show more power compensation effects than can

be accounted for by fuel-plate expansion.[47] Such results are most pronounced in the shorter period ranges and suggest the operation of one or more unidentified mechanisms. One approach to this problem is to accept self-shutdown as one of the conveniences of life and to attempt to describe it purely in terms of its effects. This approach bypasses the necessity for finding a theoretical model, although in the event a workable reactor-excursion equation is developed, it may be possible to use its form to deduce the type of physical phenomena responsible for its terms. Such an approach can also have real value in hazards evaluations for a particular core. Since a theoretical basis is lacking, this approach is questionable when extrapolated to a different type of reactor and does not help in attempts to improve reactor shutdown characteristics.

Certain equations developed at the SPERT facility in Idaho[41, 43] have been successful to a degree in predicting the magnitude and form of excursions in a particular type of core. The equation that seems most successful in matching experimental performance is the so-called "long-delay model." In this model there is a fixed time delay between the start of energy release and the initiation of reactivity compensation effects, after which the reactivity compensation is assumed to be proportional to some power of the energy released. With proper choice of delay time and constants in the power term, this model can be made to fit an experimental-burst shape with impressive fidelity. The fitting constants change somewhat with the magnitude of the reactivity addition so that care must be exercised in extrapolating results. The delay term suggests steam formation as the prime shutdown mechanism, and attempts have been made to fit steam generation terms to reactor shutdown data. Unfortunately, water

and metal expansion terms are not large enough to account for reactivity compensation observed below the boiling point.[41] Above the boiling point, steam void generation accounts for the observed reactivity compensation very nicely if empirical data are matched at one point by an arbitrary choice of the fraction of released energy that goes into steam formation. Since this released energy could also go into other shutdown mechanisms, the relative role of steam formation in reactor shutdown is still in question. It appears that some other shutdown mechanism is prominent, particularly for long period and very short period excursions.

Investigations of Potential Shutdown Mechanisms

Many programs have been conducted in attempts to understand the physical basis for shutdown effects.[42,48-52] Space Technology Laboratories have built an electrically heated analog of a reactor fuel channel[50] which, when programmed according to reactor kinetic equations, displays more moderator displacement than conventional heat transfer equations predict. Attempts to explain these large moderator displacements have recently centered around the expansion of small gas "microbubbles."[50,51] The mechanism for supplying the original bubbles has been variously identified as the entrapment of air in small crevices in the surface of the cladding during fuel loading, the radiolytic decomposition of water, or chemical reactions between metal and water.[50] Other sources go back to localized steam formation.[51] An objection advanced to radiolytically produced gas as a shutdown mechanism has been based on the contention that only very small amounts of such gas would be produced during a reactor transient. The bubble dilation model depends, however, on a

pre-excursion population of small gas bubbles, and this could arise from weeks or months of radiolytic production. Moreover, there are some calculations which suggest that in-transient production of gas could account for shutdown effects of the magnitude observed.[52]

Expansion of a population of microbubbles already in existence at the start of the transient seems a currently credible mechanism for explaining reactivity compensation. If such bubbles consist of fuel-element-trapped air or metal-water-formed hydrogen, they can be assumed to be attached to the fuel plates to begin with. If they consist of radiolytic gas from previous operation, they can be assumed to migrate to fuel surfaces through convection currents or Brownian motion.[50] At the fuel surface, such bubbles are in the optimum position to receive energy from radiation and thermal conduction. Moreover, this is the place at which a void can be expected to remove the most reactivity.

One question must still be answered, however, before accepting the explanation offered by these small bubbles. This concerns the bubbles' continued existence in the face of surface tension effects or the solubility of the gas in the surrounding water or both. For a spherical gas bubble, the pressure due to surface tension is $2\gamma/r$, where γ is the surface tension and r is the radius of the bubble. If r is 10^{-5} cm, the pressure is over 200 psi at room temperature. If r is 10^{-6} cm, the pressure exceeds 2000 psi. To exist at all, therefore, a microbubble must consist of gas at very high pressure. The volume of such "hard" bubbles would be relatively unaffected by fairly large variations in reactor tank pressure, in accord with the behavior exhibited by the not-yet-identified shutdown mechanism. On the other hand, gas at such high pressure is far

from being in equilibrium with the surrounding water, and it should quickly dissolve. The fact that at least some such bubbles can be demonstrated to exist for extended periods of time has been explained on a number of grounds. For bubbles which exist in crevices in metal surfaces or attached to small irregular bits of microdebris, the bubble surface in contact with the water can be concave toward the water. In this event, the internal pressure will be less than the pressure of the surrounding water, and no solubility problems arise. Collection of a protective coating of scum on the surface of a bubble is also postulated as a possible means of keeping the gas out of solution.

A population of small hard bubbles can be shown to have a net volume that is quite sensitive to the ambient temperature, particularly as this affects the surface tension.[50] Gas may be added by evaporation[50] or radiolytic[52] production as the power level goes up, and considerable amounts of negative reactivity may be explained. Such effects are quantitatively dependent upon the number and sizes of bubbles initially present, and experimental demonstration of such initial populations would add a sounder basis upon which to base calculations. It seems evident that the mechanism must be understood before it can be augmented.

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IV. PLANT SAFETY FEATURES

PROCEDURAL CONTROL OF SAVANNAH RIVER RADIOCHEMICAL OPERATIONS

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The operation of any manufacturing process imposes many responsibilities on supervison and management. Safety of personnel and preservation of the integrity of investment in buildings, equipment, and material are high in importance on any list of these responsibilities. In discharging its obligations, the management of a typical chemical plant contends with the more common hazards of falls, moving machinery, fire, and explosion, as well as normal wear and tear on equipment. In a radiochemical processing plant, it must, in addition, contend with those hazards that arise from radiation, the spread and ingestion of radiochemicals, and the potential for uncontrolled nuclear reaction. The principles used to control radiochemical processes and their attendant hazards are basically the same as those applied in the ordinary chemical plant. These entail (1) eliminating or reducing hazards by the application of process

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engineering fundamental knowledge and sound design and (2) controlling the actions of personnel.

Among the processes used at the Savannah River Plant are the Purex process[1] for the separation and isolation of plutonium from irradiated natural uranium and a process for reclaiming U^{235} from depleted enriched fuel elements.[2,3] Each of these processes involves solvent extraction using tributyl phosphate and kerosene solvent in mixer-settler equipment. The processing equipment is located between thick concrete walls and is serviced by remote maintenance techniques. Chemical flowsheets are adjusted so that the concentrations of fissile materials are normally a factor of 2 or 3 below the "ever-safe" concentrations[4] (<7.8 g per liter of plutonium and <12.1 g per liter of U^{235}). After the enriched uranium has been purified, it is shipped off site as a dilute solution. The Purex process product, a dilute solution of plutonium nitrate, is first concentrated by passing it through cation exchange columns. The plutonium is then precipitated as the triflouride (PuF_3) and finally reduced to metal. In the final processing steps the plutonium solutions exceed the "ever-safe" concentration. Hence, in order to maintain adequate reactivity control, limits have been established for batch size and concentration, and procedures have been prepared to ensure operation within these limits.

This paper describes the application of written procedures at the Savannah River Plant to the problem of controlling the actions of personnel. The system in use has been applied successfully for many years in the operation of widely varied processes. It has been expanded from the area of pure process control to radiation and contamination control,

equipment cleanup and repair, etc. The aim is to ensure that work is carried out in accordance with a preplanned, prearranged, and well thought-out method in order to reach the objective in a safe, efficient manner.

Organization

The Savannah River Plant, located near Aiken, South Carolina, is operated by the Atomic Energy Division (AED) of the Explosives Department, E. I. du Pont de Nemours & Company, for the United States Atomic Energy Commission. Some knowledge of the organization of Du Pont's Atomic Energy Division is essential to an understanding of the assignment of responsibilities for procedural control to various agencies, and therefore an abbreviated organization chart is shown in Fig. IV-1. The agencies concerned with radiochemical processing are the Technical Division, the Separations Production Department, and the Works Technical Department.

The Technical Division develops new processes and undertakes improvement in existing products and processes where major research is involved. The Technical Division also supplies the Production Department (through the Manufacturing Division) with:

1. Basis technical information for design

2. Technical standards or limits within which the process must be operated

3. Technical manuals that give the basis of development and provide the origin for the details of the technical standards

4. Product specifications

Unclassified
ORNL-Dwg. 63-1885

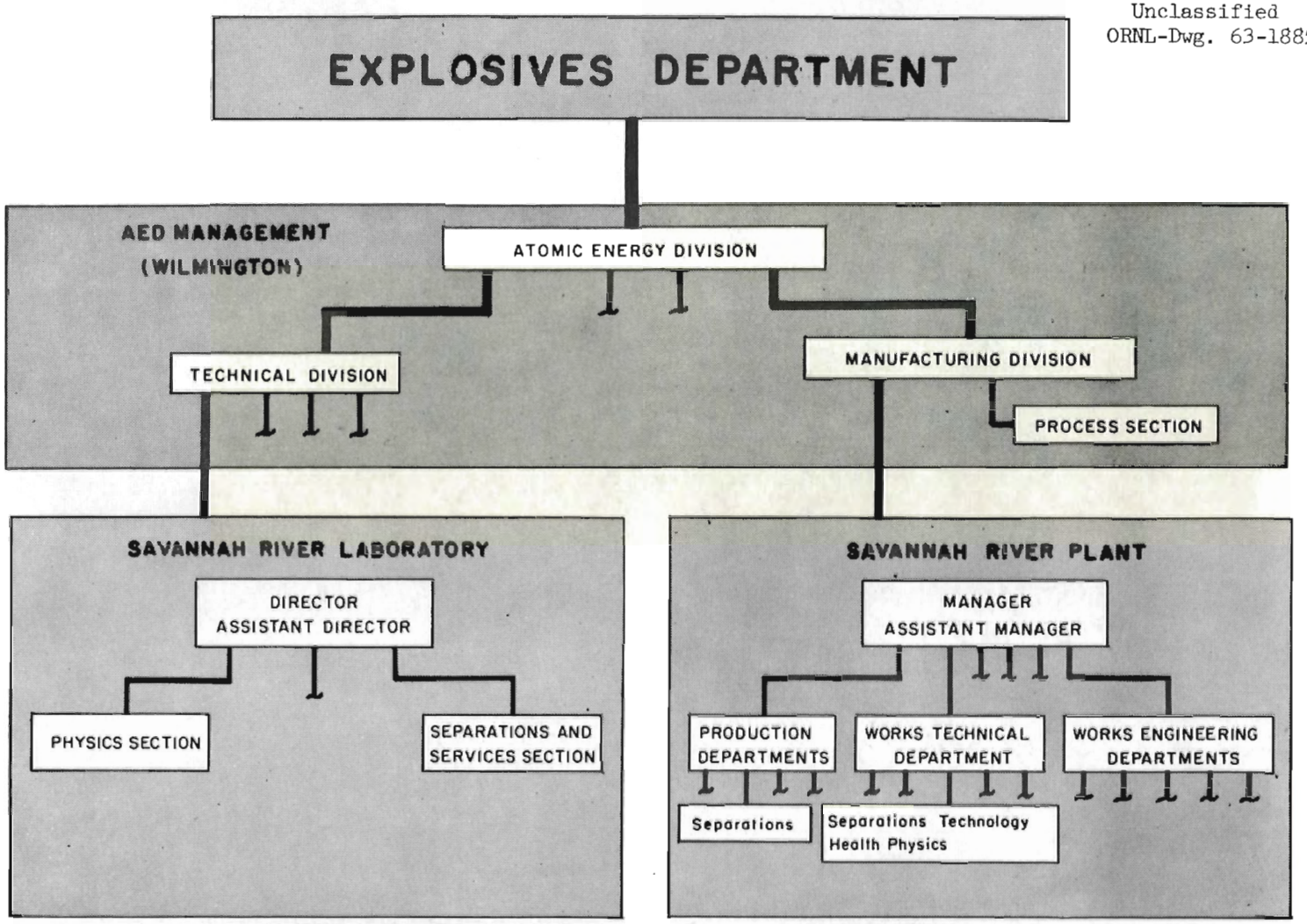

Fig. IV-1. Organization of the Atomic Energy Division of DuPont's Explosives Department.

The Technical Division also maintains liaison on technical matters with other AEC installations and outside organizations.

The Production Department has the responsibility of

1. Maintaining and operating the plant, buildings, and equipment to assure the safety and security of both equipment and personnel

2. Operating production facilities in accordance with the detailed procedures they have prepared and which have been approved by the Works Technical Department

3. Participating in process and equipment improvement programs

The Works Technical Department translates technical standards into detailed procedures for use by the Production Department. Works Technical Department personnel

1. Prepare standardized operating "runbooks" within the framework of the technical standards

2. Prepare test authorizations that permit deviations from technical standards

3. Maintain liaison between the Technical Division and the Production Department

4. Approve all operating procedures

5. Provide process assistance to the Production Department

6. Engage in process and equipment improvement programs

Procedural Controls

Procedures used in the control of personnel and processes are based on technical limits defined in technical standards, on basic design criteria documented in technical manuals, and on safety rules and principles

based on common sense and experience. The procedural controls are developed in technical standards, mechanical standards, technical manuals, operating procedures, and runbooks. The procedures actually used in operating the plant are based on the technical standards, mechanical standards, technical manuals, and other sources of information, such as safety rules and engineering blueprints. There are two general types, the operating procedures and the runbooks. A third type, the "job plan," is prepared for nonroutine work not covered by existing procedures.

The technical standards are documents that set forth the physical and chemical limits within which each major process step is to be taken. These standards, both process and nuclear, are normally based on laboratory or pilot-scale experience and permit enough safe deviations from a given flowsheet to allow sufficient latitude for productive operations. Operating limits on temperature, reagent concentrations, flow ratios, etc., are defined so that changes in variables within these limits are not expected to affect the quality or yield of the product, the safety of the process, or the discharge of waste to the environs.

The mechanical standards define physical installations, including buildings, process and service lines, and equipment. They are approved by the Works Engineering groups and authorized by the Works Engineer or Process Manager.

Technical manuals are prepared for each process operated at the Savannah River Plant. These manuals give detailed descriptions of the chemistry or physics involved in each step, together with the design basis for and complete description of the equipment used to carry out

the processes. It is the responsibility of the Technical Division to see that these manuals are prepared.

The operating procedures give detailed instructions for the performance of steps in a given process or operation. Production processes, safety and radiation control practices, and emergency measures are each covered by separate formal procedures which are prepared and revised by qualified personnel within the group that is to apply them. Each procedure that affects process variables is approved by a line organization of the Production and Works Technical Department and is authorized by the Production Department Superintendent. Wherever possible, procedures are prepared in the form of logsheets that give a step-by-step description of the manipulations required for a broad process step. These procedures are used as checksheets for the performance of the process step, and they provide, where so desired, a written record that the specified actions were carried out. Completion of the logsheet is required each time a process step is carried out on an operation that is nonroutine or which involves a real potential for injury, material loss, or equipment damage.

Job plans are developed for the cases that frequently arise and for which no procedure has been prepared. These cases may involve such nonrepetitive events as preparing equipment for maintenance, decontaminating buildings or equipment, or dismantling equipment. In these cases a job plan is prepared by the group responsible for seeing that the work is carried out, assisted usually by representatives of other groups involved in the assignment (e.g., Health Physics, Maintenance, etc.). Approval of procedures developed as job plans, when they do not affect

process variables, is at the level of area supervision in the Production and Works Technical Department. In all cases, procedures so prepared and used are updated and filed for future use.

Runbooks are a sequence of operating instructions which, when filled in, serve as written records of operating data and of the fact that the operating steps have been carried out in the specified sequence. They make reference to operating procedures that describe the required manipulations, and these log sheets may become part of the assembled package of runbook data. Runbooks are designed to provide a means of process control. Completed copies are reviewed by both the Production and Works Technical groups who analyze critical data to determine process performance and efficiency and to verify that the operations are being carried out as specified. Since runbooks are used to establish the proper sequence of process steps, they also provide an effective means of administrative control of criticality hazards and are thus considered basic to criticality control.

Changes in Procedural Controls

In the administration of any procedural system, some provision must be made to accomplish changes in an approved and controlled manner. Continued experience in operating a chemical process dictates frequent, numerous changes. At the Savannah River Plant, if such changes require operation outside the existing technical standards, test authorizations may be issued after thorough consideration of safety and process consequences. These documents provide a definite procedure for trial deviations from existing standards, assign responsibility for the test, and

specify that a final report of the results and recommendations for future action be made in the form of a test conclusion.

Test authorizations are also used to document agreed-upon changes in process conditions which are within the broad technical standards but which call for significant deviation from long-accepted operating limits. Test authorizations covering operations outside technical standards are approved by Plant and Wilmington management and authorized by the Director of Manufacturing. This means that they are approved by the personnel who had the responsibility for preparing the original technical standard. Test authorizations covering operations within standards require authorization at the plant level.

Specific Control Measures

The preceding discussion has described the development of operating procedures for radiochemical operations at Savannah River. The procedures provide for the various specific control measures that are employed at the appropriate points in the procedure. These specific control measures may be divided into four groups: primary controls, secondary controls, system audits, and procedural deviation reports. The primary controls are those exercised by the operators in the course of following the operating procedures and runbooks for the conduct of a given operation. Secondary controls are regarded as monitoring instrumentation and other techniques for reducing the potential for nuclear incidents. System audits and procedural deviation reports are employed to ensure conformity with the procedures available. Each of these measures is discussed in detail below.

Primary Controls. At points in the process where it is evident that additional attention is required because of a nuclear safety prolem, attention is focused on the particular step involved by means of "nuclear safety control points." These points are warning notices that appear in runbooks and procedures (high-lighed by having been blocked off and typed in block letters) and which restate the limits that must not be exceeded. Because of the high hazard potential involved, the signature of the supervisor is required at such points to show that the specified conditions have been met.

The concept of nuclear safety control points is applied as follows in procedures for process control of batch operations. Material balances are used to maintain process control in the final processing of plutonium. A batch of plutonium is carried through successive steps of concentration, precipitation as the trifluoride, and reduction to the metal. Nuclear safety is assured when it is know that

1. A portion of the batch being processed has not been lost
2. A portion of the previous batch was not left in the equipment
3. The size of the batch does not exceed the technical standard

A material balance is calculated for each step (or in some cases, across each tank) in the process. The mass of plutonium from a previous step, as determined by the material balance, is used to check the mass obtained in a subsequent step, as determined by measurement and laboratory analysis. If discrepancies appear, these are resolved by a resample or by recalculation of the material balance, or both. The mass must be known and must be within the technical standard before the plutonium is transferred to the next step. In obtaining this mass, the sample analyses

reported by the laboratory are screened against what is expected and against previous material balances. Screening of sample analyses and the calculation of material balances for the product are performed by an operator. Supervisory checks are made where warranted. In those cases where the quantity of fissile material must be known for nuclear safety control, the runbooks require that analytical data be obtained before proceeding to the next processing step.

The processing of fissile materials in continuously operated equipment, such as mixer-settlers, cannot be adequately controlled by means of material balances because of the dynamic situation which exists, that is, the constant blending of batches in tanks and processing equipment and the time lag between sampling and receipt of analytical data. Nuclear safety control points in these operations center around proper makeup of batches of feed materials, together with very rigid flow control. As an example, in the purification of enriched uranium fuels, loss of acidity in the scrub stream to an extraction mixer-settler, or a significant drop in the flow of the strip stream in another mixer-settler, will permit U^{235} to reflux and accumulate, and a critical configuration can be reached. Nuclear safety is therefore provided by flow instruments in duplicate, with alarms to sound in the control room. Continuous conductivity measurement,[5] with alarms, is provided to ensure that acidity of the scrub stream is not lost. Enough runbook entries are required to ensure that the control instruments are read on a sufficiently frequent basis.

Secondary Controls. The secondary controls include backup instrumentation, which warns of increased U^{235} accumulations, a "scram system,"

and other safety precautions. Neutron monitors[6] with duplicate detectors and transmitters are provided at critical locations to warn of accumulation of U^{235}. Futhermore, solution from specific stages in the mixer-settler is continuously withdrawn to a sample area where the uranium content is measured by a flow-through colorimeter.

In addition to instrumentation, a "scram system" is provided which, when activated, immediately closes all stream control valves, stops feed jets, and shuts down agitation. This system is activated if a stream flow fails and cannot be corrected within 15 min, if the acidity of the scrub stream is found to be low, or if both channels on the neutron monitor indicate an increase in U^{235}. Although the above description is based on U^{235} operations, similar process control philosophy is used in plutonium separation (Purex process) operations.

Other specific controls have been developed and defined by procedure to reduce the potential for nuclear incidents. Among these is the use of nuclear safety blanks. At numerous locations throughout the processing plants, piping may exist that is not required by the process being operated but which may be difficult or undesirable to remove. When it is found that this piping presents a potential hazard through misuse, a nuclear safety blank is installed. This blank is designed so that it cannot be removed from between pipe flanges unless a cutting torch is used on its collar, which is welded around the pipe connecting to the flange. Piping is not damaged by installation of this blank. Typical applications of this principle are the blanking of caustic lines to prevent concentration of fissile materials by precipitation and the blanking of steam lines to the lower coils in enriched-uranium dissolvers to

prevent concentration of the solutions by evaporation. Approvals by the Separations and Separations Technology Superintendents are required for removal of these blanks. Orders for removal specify replacement when the condition for which removal is necessary no longer exists.

System Audits. To fulfill its purpose, a control system such as the one described must provide for an independent review by parties not directly concerned with the operation. Audits should be carried out to ensure that the system is up-to-date, is operable as intended, and is operated as intended. Three independent auditing groups have been established at the Savannah River Plant.

The Plant Criticality Audit Committee, established by the Plant Manager, audits any aspects of plant operations that present a potential for a criticality incident. The present membership of the committee includes the General Superintendents of the Production and Works Technical Departments and the Director of the Physics Section of the Savannah River Laboratory. Audits are usually planned in connection with some significant process or equipment change.

The Separations Department Criticality Audit Committee is a three-man committee, reporting to the Separations Department Superintendent, which provides an independent survey of conditions, practices, and procedures in current use for Separations Department criticality control. Quarterly reports of their audits are required. Members are selected from higher supervision of the Separations, Separations Technology, and Health Physics groups. Members of this committee are also available to act in an advisory capacity on criticality considerations when equipment design and process changes are contemplated.

The Separations Department Procedures Audit Committee surveys on a quarterly basis each process for compliance with approved procedural control systems. Reports, with recommendations, are sent to the Separations Department Superintendent. Recommendations made by any of these three independent auditing groups must be answered in writing. If compliance with recommendations is not intended, reasons must be given and the differences resolved. Recommendations that cannot be complied with at once are carried from audit to audit until their intent is met, or their status is reported on a quarterly basis, whichever comes first.

Procedural Deviation Reports. All known deviations from approved procedures receive immediate attention from operating area supervision. Those of a more serious nature result in a written Operating Incident Investigation, which is circulated within the originating department or throughout the Plant if it has wider application. Reports on deviations that could affect radiation safety practices or criticality control measures receive further attention; they are ultimately reported to the Plant Special Hazards Committee, which is appointed from the Plant staff by the Manager. The investigation report is reviewed and issued as a Special Hazards Investigation Report to applicable Plant groups. Written followup on the recommendations is required from all groups within whose province similar incidents could occur.

Conclusions

The systematic use of written procedures prepared by line supervision, certified as technically correct by staff technical groups, and authorized by management assures control of the actions of personnel

carrying out the various operations at the Savannah River Plant. From the viewpoint of some, the system appears to diminish the authority of the first-line supervisor and to limit his freedom for making decisions in the realm of process control. It is true that the supervisor's responsibility is more closely restricted than in a normal chemical plant, but when the tremendous potential for injury, or death, or complete loss of exceedingly expensive buildings and equipment is considered, some limitation is to be expected. It should be emphasized that considerable supervisory responsibility still remains under any system of procedural control, no matter how rigid, particularly in the area of nonrepetitive operations. Before such operations are undertaken, consideration must be given to existing conditions of radiation and contamination and to the effect on subsequent processes, such as waste disposal. Detailed planning, based upon sound knowledge as to how such conditions may change as the job progresses, is used. Judgment must be made as to whether or not existing procedures cover the work to be undertaken in its entirety. If they do not, new or modified procedures must be prepared.

Under a system of broad procedural control, management participates with first-line supervision in decisions that have an important bearing on safety and on the integrity of the investment. Under this system there is, and always will be, a need for first-line supervisors of sound technical background, with creative and analytic minds, and good common sense.

REACTOR CONTAINMENT SYSTEMS

W. R. Gall

Reactor containment systems are designed to control the release of radioactive materials to the environment in the event of a gross failure of the primary reactor system regardless of how remote the possibility of such a failure may be. The containment vessel used most often today is a welded hermetically sealed shell, designed, constructed, and tested to ensure the lowest leakage rate that can be achieved and measured accurately. Other types of containment have been proposed, however, to reduce cost or to fit siting considerations that require more complete containment that can be assured by the minimum leakage rate attainable with the sealed shell. These other types include various forms of multiple containment, such as pressure-suppression containment, building containment, and underground containment.

Although the concept of surrounding a reactor system with an impervious envelope as a precaution against the ultimate release of fission products to the public domain seems simple, it poses severe and costly engineering problems. In particular, when the engineering details are completed, it is always found that the envelope has upward of several hundred penetrations, which in order for the containment structure to perform its function in the event of an accident, must be sealed by a number of valves operating and seating perfectly. It is of more than passing interest to note that, after some discussion of these problems, Farmer[7] concluded that "...development of new reactor systems with improved fuels should reduce the possibility of releasing fission products

from the reactor even after severe faults. If such a release could be positively prevented by simple emergency procedures, no containment would be necessary." (See also, pp. ____, this Review)*. In this connection it should be noted that the United Kingdom central power stations using gas-cooled reactors are contained within conventional building structures on the basis that adequate emergency procedures are available to prevent a serious release of activity from those reactors.

Containment is of such vital interest to the nuclear power program, both in terms of cost and siting, that it has been the subject of frequent previous articles in Nuclear Safety. The present article is a review of developments during the past year. In addition to a review of the performance of various systems, it includes the results of a recent comparative cost study.

Multiple Containment

Multiple containment connotes any one of a number of systems in which there is more than one complete barrier surrounding the primary system. Although the retentive capability of such systems is readily apparent, it was only with the advent of the N.S. SAVANNAH[8] and its porting requirements that the potentialities of such a system were fully exploited. Subsequently, another version of double containment was proposed for the city of Los Angeles reactor, and it has received preliminary approval.[9] Although in a general sense pressure suppression constitutes one form of double containment, in view of the directed

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*See addendum to this draft.

discharge of material from the inner to the outer container, it is reviewed as a separate system.

The most spectacular site proposal for a nuclear power plant to date is that for Consolidated Edison's 1000-Mw (electrical) pressurized-water reactor in the heart of New York City at Ravenswood in Queens County on the bank of the East River just across from East 72nd Street in Manhattan.[10] A new type of containment has been proposed for the reactor that has a four-layer composition consisting of a pair of 0.25-in.-thick carbon steel shells (one within the other and 2 ft apart), a 2-ft-thick layer of pervious concrete between the shells, and 5 1/2 ft of normal-density reinforced concrete outside the outer steel shell. The outer layer provides the bulk of the shielding from internal radiation sources and the strength needed in the event of pressure buildup from a serious reactor accident. The pervious concrete layer between the two steel shells forms a negative pressure zone to assure inward flow at any leakage point, even in the event of the worst conceivable accident. Pervious concrete is solid and strong enough to transfer all internal pressure to the outer reinforced concrete shell and yet sufficiently permeable to permit the flow of fluids. The negative pressure zone is filled on completion with an air-freon mixture and "sniffed" for leakage. The final leakage rate will not be more than the equivalent of less than 0.1% of the containment volume per 24-hr period at 15 psig. The design pressure is 40 psig. Experience at the Yankee nuclear power plant, the proposal says, shows that the leakage rate can be held to 0.03%. In the "worst conceivable accident" the entire contents of the primary coolant system would escape instantaneously into the containment vessel, and, simultaneously, complete core meltdown

would release fractions of the accumulated fission products; both these eventualities are considered highly unlikely. The containment design is such that under these conditions "all the water, air, steam, and fission products are contained within the containment and no leakage to the outside environment occurs." Sufficient cooling is provided to offset the effects of residual and decay heat, as well as any heat generated by chemical reactions.

A method of analysis of multiple containment has been developed by Anderson[11] for a three-barrier system consisting of the core surrounded by two concentric containment shells. A step release of activity from the core is assumed. The relative leak rate, L, is plotted (Fig. IV-2) against the relative time, θ, from the equation

$$L(\theta) = \frac{K}{K - l} \left(e^{-\theta} - e^{-K\theta}\right) ,$$

where

$$\theta = l(t - t_0) ,$$

$$K = \frac{m}{l} ,$$

$$L(\theta) = \frac{mU(t)}{lfN(t)} ,$$

l = leak rate constant for first containment vessel,

m = leak rate constant for second containment vessel,

f = fraction of the total quantity of the isotope that has escaped from the core as a gas,

U(t) = quantity of the isotope held up in the second containment vessel,

N(t) = total quantity of precursor isotope existing at time t,

t_0 = time elapsed from the start of the accident to the initial fuel element failure.

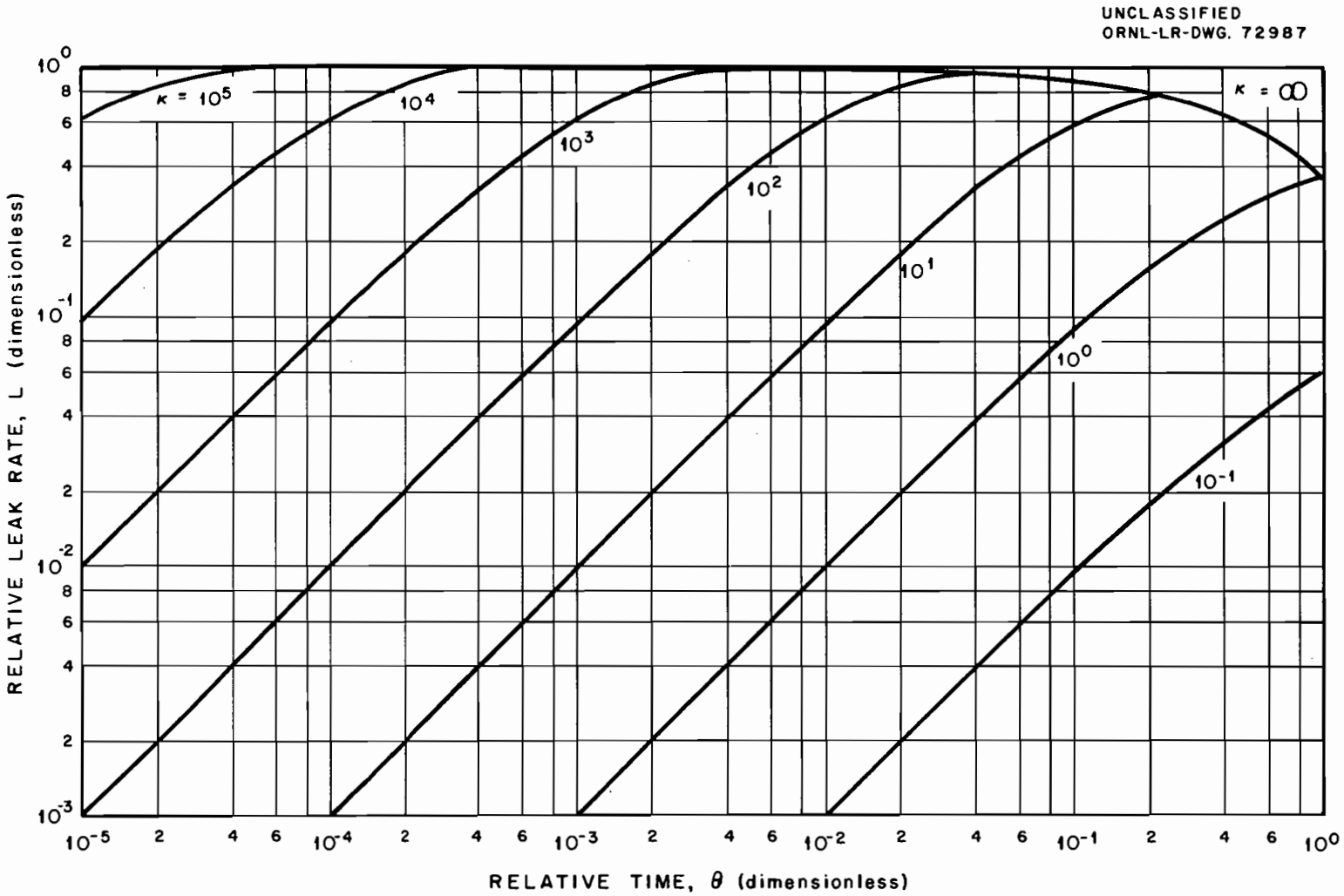

Fig. IV-2. Leak Rate to Atmosphere from Double-Containment System with Step Release from Core.

The equation takes into account decay of the isotope in the interval t, as well as the buildup from decay of its precursors. The chart (Fig. IV-2) can be used to determine

1. The release rate at any time after the initiation of an accident:

$$\text{Release rate} = mU(t) = L(\theta)\ell fN(t)$$

2. The amount of any gaseous isotope held up in the second containment vessel:

$$U(t) = \frac{\text{release rate}}{m}$$

3. The release rate of any given isotope when a filter is used;

$$\text{Release rate with filter} = mU(t)\ (1 - E)$$

where E is the efficiency of the filter for removal of the isotope under consideration.

The Dragon Reactor under construction at Winfrith, England, is contained in a system of multiple containers.[12] Two containment system barriers to the release of fission products are provided in addition to the primary coolant system, which consists of the main coolant circuit and a fuel element purge circuit. The activity level in the purge circuit is of the order of 10^4 times that in the main coolant circuit, and the maximum leakage rate specifications for the two systems are 0.5 cm^3/sec for the main circuit and 2×10^{-3} cm^3/sec for the purge system. Both the primary and purge systems are surrounded by a steel containment shell designed to withstand an internal pressure of 10 psig, at which pressure the leakage rate shall not exceed 0.1% per day. A second container, which is provided by a concrete structure surrounding the steel shell,

is designed to withstand an internal pressure of 3 psig and will have a leakage rate of not more than 5% per day for a differential pressure of 1 psi. This outer shell also serves the purpose of shielding the fission products contained within the inner containment vessel. A closed-circuit filter system is provided in the outer container for cleanup of the gas in the space between it and the inner container. Calculations of the integrated activity release indicate a range of 5 orders of magnitude, depending upon whether one or two containment barriers are employed and whether both, one, or neither containment zone is filtered. The performance of containment systems of this general nature was also examined by Rushton and Edwards[13] in a paper presented at the IAEA Symposium on Criteria for Guidance in the Selection of Sites for the Construction of Reactors and Nuclear Research Centers (see pages ____through____, this Review).

Pressure-Suppression Containment

Pressure-suppression containment[14] seems particularly suitable for application to boiling-water reactors of the direct-cycle type because of the compact arrangement of the primary system equipment. The primary system is contained within a dry well, and the primary steam is piped outside the containment shell through double valves. This type of containment is being used in the Humboldt Bay nuclear plant[15] of the Pacific Gas and Electric Company, which is now under construction, and is the type proposed for use in the larger plant to be constructed at Bodega Bay.[16] The Humbolt Bay plant has a 50-Mw boiling-water reactor with natural circulation and internal steam separation, and the Bodega Bay

plant has a 325-Mw (electrical) boiling-water reactor with forced circulation and internal steam separation.

The pressure-suppression containment system provides complete enclosure of the reactor and primary coolant system up to double isolation valves in a primary containment vessel, which is the so-called dry well. The dry well is a pressure vessel that is designed for the maximum credible internal pressure and which is vented to a pool of water by large-flow-area vent piping. The water pool is located in a vapor-tight suppression chamber, which also contains a large air volume. A refueling building, which may be used in conjunction with the pressure-suppression containment to collect system leakage and to provide containment during refueling, is built over the primary reactor plant. The refueling building has a limited-volume gas exhaust that is processed through a gas treatment system and then released by way of a tall stack.[17] Advantages cited include (1) rapid reduction of the primary containment pressure, which reduces leakage, (2) a high degree of fission-product retention in the pool water, (3) adaptability to below-ground-level placement, which eliminates direct radiation from the containment following an accident, (4) low leakage rates because of compactness of the dry well and suppression chamber and the multiple barrier nature of the containment system, and (5) postaccident core cooling by flooding the dry well.

Three test facilities were built and operated to obtain performance data in support of the Humboldt Bay design. The first was a large tank in which a steady flow of steam was used to investigate condensation. These tests proved the effectiveness of straight pipes discharging downward, and all subsequent work was done with this arrangement. The second

facility was a reduced-scale model of a reactor and pressure-suppression system for exploring transient operation, measuring transient pressures for comparison with analytical results, and studying the ability of the water pool to retain fission products. This system was approximately 1/1000th scale by volume, as compared with the Humboldt design. Results of the tests were in substantial agreement with analytical calculations, except that the flow rates from the reactor vessel were lower. Condensation was complete and effective, and the measured separation factor for fission products between the reactor vessel and the enclosure above the pool was of the order of 10^{-8} for small solid particles and about 10^{-6} for halogens and soluble salts; more than half of the noble gases were retained initially.

The third test was a full-scale demonstration of one of the 48 segment vent pipes for the Humboldt design. In this facility, under conditions simulating the Humboldt maximum credible accident, the maximum dry well pressure was in the range of 25 to 36 psig (the Humboldt dry well design pressure is 72 psig), and the test flow rate was 54% of that predicted. The results indicate that the estimates of rupture flow rates, static pressure rise in the dry well and suppression chamber, and jet and other dynamic forces are conservative.

A method for predicting the maximum pressures has been reported.[18] The maximum pressure in the suppression chamber occurs at the end of the venting period when all the dry well air has been transferred and the temperature in the chamber is at its maximum. It is assumed that all the air initially in the dry well is transferred to the suppression chamber during venting. The design pressure of the chamber is then determined

by a weight and energy balance based on the assumption of complete condensation.

Containment by Building Structures

Nuclear facilities in which the stored energy is small are sometimes contained in buildings that are specially designed to be gas tight. Cleanup systems are provided for the gases that may be released into the buildings. Examples of this type of facility are the Oak Ridge Research Reactor[19] and the production reactors at Hanford.[20]

Extensive leakage measurements of various building components, materials, and complete structures have been reported by Koontz, Nelson, and Becker[21] for various applications of building containment. Components of metal and concrete structures representing possible leakage paths and doors, louvers, butterfly valves, caulking materials, and paints were tested. Detailed data and methods of analysis were subsequently published in a design manual.[22]

In the course of obtaining the design data several prototype building models were fabricated and tested after being constructed according to rigorously specified rules and designs. Nevertheless the error of the predicted building leakage rate was approximately ±50% of the measured leakage rates. When buildings were designed and constructed for minimum leakage by the methods developed during these tests, however, the reliability of the predictions increased so that the predicted leakage rate was within approximately ±10% of the measured values.

Table IV-1, taken from reference 22, gives some of the results of component tests of concrete buildings, metal buildings, and penetrations.

Table IV-1 RESULTS OF TESTS OF BUILDING COMPONENTS

Component	Leakage Coefficients, cfm/(in. H_2O) (unit of D)		Units of D, Length or Area	Pressure Range of Test
	A	B		
Concrete with a compressive strength of 3500 to 5000 psi after 28 days; No. 3 gravel; ratio of cement to water 0.43 ± 0.03; slump, 2–4 in.	2×10^{-5} to 3×10^{-6}	0	Inches of thickness per square foot	10 psi
Cold concrete joint after 3-day set	2.4×10^{-4}	0	Per foot of joint	5 psi
Epoxy-coated concrete joint	4×10^{-5}	0	Per foot of joint	5 psi
Concrete crack, 12-mil mean size	0.069	0.1	Per foot of 4-in.-deep crack	2 psi
Concrete expansion joint with Thiokol (1/2-in. separation of two pieces filled 2 in. deep)	4.6×10^{-6}	0	Per foot of joint	
Frame cast in concrete with anchors	1.3×10^{-4}	0	Per foot of joint	
Industrial metal skin panels, 18-gauge, insulated				
End laps, Robertson RP-545 caulk or equivalent	1.6×10^{-4}	0	Per foot	10 in. H_2O
Edge laps, male and female with double caulking in female joint	4.2×10^{-5}	2.8×10^{-6}	Per foot	10 in. H_2O
Intersection of end joint and edge joint	1.6×10^{-2}	2.2×10^{-3}	Per intersection	10 in. H_2O

Table IV-1 (continued)

Component	Leakage Coefficients, cfm/(in. H_2O) (unit of D)		Units of D, Length or Area	Pressure Range of Test
	A	B		
Typical hollow metal personnel access door, 3 × 7 ft, sound insulated with weather strip	0	35	Per opening in door	10 in. H_2O
Marine, quick acting, 28 × 62 in.	1×10^{-4}	2.6×10^{-2}	Per opening in door	15 psi
Louvre with 30-lb closure, 15 × 20 in.	6.7	28	Per louvre	10 in. H_2O
Electrical fittings in bulkhead	10^{-9}	No detectable leak	Per fitting	10 psi
Hatches, 30 × 36 in.	0	20	Per hatch	5 in. H_2O
Butterfly valves	3×10^{-8}	0	Per valve 12 in. in diameter	No detectable leakage at 14 psi

The data of Table IV-1 may be used to establish the leakage rate of a full-scale building constructed in a normal fashion but using the specific techniques and precautions established in the laboratory. The following relations are applicable:

$$V_T = C_1 \Delta P + C_2(\Delta P)^{1/2} ,$$

where V_T is the total volumetric leakage rate, ΔP is the pressure differential, and C_1 and C_2 are constants determined as follows:

$$C_1 = \sum_{i=1}^{n} D_i A_i$$

and

$$C_2 = \sum_{i=1}^{n} D_i B_i ,$$

where D_i represents the total length or area of the leakage path of the ith component, and A_i and B_i are the leakage rates per unit length or area per unit pressure difference, as given in Table IV-1 and the tables of ref. 22, for the ith component.

In addition to standard specifications recommended by the American Institute of Architects, the National Association of Architectural Metal Manufacturers, and the American Concrete Institute, the specifications should be especially definitive and should require a final proof test of the building leaktightness. It is also essential that during the construction process a superior program of construction supervision be effected if conventional structures are to be made leaktight.

Underground Containment

European reactor designers have given considerable thought to the possibilities of underground containment. A previous article[23] in Nuclear Safety covered investigations and experience through 1960 for this type of containment. This review is concerned with more recent reports describing the Swiss installation at Lucens near Lausanne, where underground confinement is used for a 30-Mw (thermal) experimental nuclear power plant[24] that includes a D_2O-moderated reactor cooled with CO_2.

The plant site is approximately 30 kilometers north of Lausanne in a rocky escarpment bordering on the valley of the Broye. The population within a radius of 10 kilometers is 25,000. The reactor, the power conversion equipment, and the irradiated fuel element storage facilities are located in three underground caverns. The reactor cavern is lined with concrete. The control room, shops, laboratories, and administrative offices are located in an external service building at the surface. The postulated worst accident consists of sudden interruption of cooling and loss of coolant when the reactor is operating at full power. This accident would result in rupture of the primary CO_2 loops and secondary steam and water loops and in total melting of the reactor core, with the release of 100% of the noble-gas fission products, 50% of the halogens and volatile solids, and 1% of the nonvolatile solids. It was assumed that the reactor fluids would expand adiabatically and fill the reactor cavern to an equilibrium pressure in excess of 2 atm. After a period of approximately 30 min, during which the temperature and the pressure would decrease as the water vapor condensed, gates would be

opened in the wall of the reactor cavern leading through a filter to the high-porosity rock that contains the caverns. Flow through the filter would be by the influence of the pressure in the cavern, or it could be accelerated by the use of CO_2 compressors.

The specifications call for a maximum permissible leakage from the reactor cavern of 60 m^3 (STP)/hr, a maximum leakage toward the other caverns of less than 10^3 curies of I^{131}, and a maximum permissible total leakage of 15 m^3 (STP). The results of an analysis of the tightness and leakage from the reactor cavern are given in Table IV-2.

General methods of analysis of the diffusion and retention of radioactive products in the rock chambers surrounding underground power plants have been described by Binggelli and Verstraete.[25] They state that although the tightness is poor, it is adequate for ensuring that leakage is slowed down over a period of several hours. The gases that flow through the concrete are stored in the pores of the rock to a shallow depth. The progression of the leakage gases through the rock is by diffussion at a very low rate. The concrete walls and the rock behave as filters that pass only the fission gases and halogen vapors. The fission gases whose disintegration products could lead to contamination of long duration are short lived, and the duration of retention in the rock is such that the radioactivity becomes negligible before the gases reach the external atmosphere. The underground placement of a nuclear power plant in a judiciously chosen mass of rock permits the confinement of the gravest accident by the utilization of procedures current in engineering practice.

Table IV-2 ESTIMATED LEAKAGE FROM REACTOR CAVERN OF THE SWISS EXPERIMENTAL POWER PLANT AT LUCENS FOLLOWING POSTULATED WORST ACCIDENT

Accident Phase[a]	Leakage Parameters	Parameter Values Based on Pressure Relief	
		By Expansion Through Filter Into Rock Chamber	By Flow Through Filter Accelerated with Compressed CO_2
I	Permissible flow rate of leakage from reactor cavern for specified $\Delta p = 1\ kg/cm^2$	60 m^3 (STP)/hr	60 m^3 (STP)/hr
I	Volume of leakage from reactor cavern during phase I	13.4 m^3	13.4 m^3
II	Duration of lowering of pressure	46 hr[b]	3 hr
II	Volume of leakage from reactor cavern during lowering of pressure	1023 m^3	27.2 m^3
I + II	Total volume of leakage from reactor cavern	1036 m^3	40.6 m^3
I + II	Permissible flow rate of leakage toward adjacent caverns for specified $\Delta p = 1\ kg/cm^2$	4 m^3 (STP)/hr	23 m^3 (STP)/hr
I	Volume of leakage toward adjacent caverns during phase I	0.9 m^3	4.5 m^3
II	Volume of leakage toward adjacent caverns during lowering of pressure	14.1 m^3	10.5 m^3
I + II	Specified total volume of leakage toward adjacent caverns	15 m^3	15 m^3

[a] I. A period of approximately 30-min, during which the temperature and the pressure in the cavern decrease following the peaking of the equilibrium accident pressure.
II. A period following Phase I, in which the gases in the cavern are discharged through filters.

[b] Time necessary for lowering the excess pressure to 0.003 kg/cm^2, or 1% of the value at the start of the second phase.

Economic Analysis of Boiling-Water-Reactor Containment

In a study[26] previously reviewed,[27] Sargent and Lundy Engineers made a technical and economic comparison of four types of containment system for boiling-water power reactors. The four types were (1) the standard steel containment structure that is vaportight to accident pressures greater than 15 psig; (2) a pressure-relief containment system that is leaktight but relieved by a burstable diaphragm or valve in a vent duct that can be closed after a short period to prevent release of the products of a core meltdown; (3) a pressure-suppression containment system such as that described in the previous section, and (4) a low-pressure containment structure consisting of a vaportight shell of large enough volumes to enclose the entire plant and to prevent the equilibrium pressure from exceeding 5 psig in the event of a loss-of-coolant accident. In a supplement to the report,[28] a new type of containment system referred to as "total containment" was added to the four basic types covered in the original study, and a 300-Mw (electrical) direct-cycle plant was considered for all five types of containment system.

The total-containment system was described as being similar to the low-pressure containment system; however, the structure which completely enclosed the plant would not be restricted to a 5-psig maximum design pressure. The total-containment structure for the four types of plant considered consisted of a concrete base surmounted by a steel cylinder with a hemispherical dome. In the technical comparison it was considered to be equal to the low-pressure containment system in over-all protection, although the higher design pressure tended to increase the leakage from the building. The net difference in the amount of leakage is probably

not significant. Technical and radiological data on all five containment systems for the 300-Mw (electrical) direct-cycle plant are given in Table IV-3. The boundary of the exclusion area was established by assuming that the plant land area of 1200 acres was arranged in a square with the plant located in the center. No low-population zone was established.

Cost comparisons from both Sargent and Lundy studies are presented in Tables IV-4 and IV-5. The comparisons indicate that the pressure-relief method shows an appreciable saving over standard containment; however, an extensive research effort would be required to verify the design calculations. There is a significant saving using pressure-suppression containment for the direct-cycle plant in which the reactor primary system can be contained in a relatively small volume. Low-pressure containment was found to be uneconomical for all the plants studied. The large coolant inventory together with the maximum 5-psig design pressure results in an oversize structure in which the space is not utilized efficiently. Total containment makes efficient use of building space, and the costs are of the same order of magnitude as those for standard containment.

Summary and Conclusions

Multiple containment, the newest development, is now proposed as a basis for locating the world's largest power reactor in the heart of the largest city in the United States. The advantages of this power plant over conventional coal-fired plants from the standpoints of atmospheric pollution, power costs, and reliability provide strong incentives for its construction. If approved and built, it may represent a breakthrough

Table IV-3 CONTAINMENT STUDY DESIGN DATA FOR A 300-Mw DIRECT-CYCLE BOILING-WATER REACTOR

Electrical rating, Mw	300
Thermal rating, Mw	871
Reactor system pressure, psia	1250
Reactor system temperature, °F	572
Steam volume, ft^3	1935
Water volume, ft^3	1532
Total weight of water and steam, lb	199,850

Containment scheme	Standard shell	Pressure-relief system	Pressure-suppression system	Low-pressure system	Total containment system
Container configuration	Cylindrical with hemispherical top	Rectangular	Cylindrical with hemispherical top	Cylindrical with hemispherical top	Cylindrical with hemispherical top
Container dimensions, ft	84 (diam)	50 x 68	21 (diam) 29 (bottom)	286 (diam)	200 (diam)
Gross containment volume, ft^3	908,400	525,000	34,100	9,020,000	3,670,000
Net containment volume, ft^3	750,000	420,000	25,300	8,400,000	2,940,000
Containment design pressure, psig	49	5	127	5	14
Containment wall thickness, in.	0.75 (top) 1.50 (cylinder)	18 (upper wall) 24 (lower wall)	0.75 (top) 1.0 to 1.35 (cylinder)	0.26 (top) 0.52 (cylinder)	0.59 (top) 1.18 (cylinder)
Containment materials	Steel	Concrete	Steel	Steel and concrete	Steel and concrete
Assumed leakage rate, % of total volume in 24 hr	0.1	0.25	0.05	0.25	0.25
Radiation dosage at exclusion boundary, rem					
Direct radiation	0.0934	0.00179	2.1×10^{-6}	0.225	0.196
Internal dose from					
I^{131} (a)	0.01252	0.1146	1.11×10^{-4}	0.0225	7.86
I^{133} (a)	0.00665	0.0606	5.89×10^{-5}	0.0117	4.15
Sr^{89}	0.00067	0.00612	5.95×10^{-6}	0.0118	0.420
Sr^{90}	0.00250	0.0228	2.22×10^{-5}	0.0044	1.562
Fission-product deposition					
Dry	1.58	3.78	0.82	3.94	3.94
Wet	6.31	15.12	3.28	15.75	15.75
Whole-body-dose from fission product plume	0.303	0.756	0.126	0.693	1.071
Total whole-body-dose from dry deposition	1.981	4.581	0.946	4.877	17.189
Total whole-body-dose from wet deposition	6.403	15.122	3.28	15.975	15.946

[a] Iodine dose can be converted to equivalent whole-body radiation by multiplying by a factor of 0.0833, since a total dose of 300 rem to the thyroid is equivalent to 25-rem whole body.

Table IV-4 COMPARISON OF CONTAINMENT COSTS IN DOLLARS PER KILOWATT

Plant Cycle Type	Electrical Output (Mw)	Containment Cost ($/kw)				
		Standard Shell	Pressure Relief System[a]	Pressure Suppression System	Low-Pressure System	Total Containment System
Direct	44	31.05	12.90	17.86	71.59[b]	26.83[b]
Dual	180	30.63	6.46	38.59	88.36	28.19
Dual	300	21.07	4.62	25.14	59.35	20.73
Direct	300	8.91	2.57	4.80	19.10[b]	10.35[b]

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

[b]Cost of moat not included.

Table IV-5 COMPARISON OF CONTAINMENT COSTS IN MILLS PER KILOWATT-HOUR

Plant Cycle Type	Electrical Output (Mw)	Containment Cost (mills/kwhr)				
		Standard Shell	Pressure Relief System[a]	Pressure Suppression System	Low-Pressure System	Total Containment System
Direct	44	0.63	0.26	0.36	1.46[b]	0.55[b]
Dual	180	0.63	0.13	0.79	1.80	0.58
Dual	300	0.43	0.09	0.51	1.21	0.42
Direct	300	0.18	0.05	0.10	0.39[b]	0.21[b]

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

[b]Cost of moat not included.

in the development of nuclear power in the near future. Other examples of multiple containment already in use or under construction are the N.S. SAVANNAH and the UKAEA high-temperature reactor, Dragon. While such systems do not offer the simplification desired by Farmer, they do provide a reasonable and effective containment system designed to comply with the USAEC site criteria.[29]

Although hailed as a means of making nuclear power plants acceptable for populous sites, multiple containment represents another level of conservatism added to a system already conservative by virtue of superquality construction and engineered safety features. Whereas the use of a single container implies lack of confidence in the safety of the nuclear system and its safety features, the use of multiple containment implies lack of confidence in both the nuclear system and the single container. The next step obviously is triple containment. A possible consequence of this trend is to relieve the reactor builder of the necessity for building high-integrity primary systems, since failure of the nuclear system is a postulate of the design and since the cost of the nuclear system must be reduced to pay for the extra layers of container. Then, when failure of a nuclear system occurs, it will be taken as corroboration of the need for the ultraconservative containment system. A far better trend would be for nuclear systems, including their safety features, to demonstrate a level of reliability such that the time will come when containment can no longer be justified.

Pressure-suppression containment, although to some appearances a modified form of double containment, is dependent upon the discharge of fluid from the dry well to the suppression chamber, which, although

underground, is not further contained. Although pressure suppression appears to be particularly suited to the direct-cycle boiling-water reactor because of the compactness of the primary system of that reactor, this compatibility is the direct consequence of a regulatory decision that accepts the operation of double valves in the primary coolant lines to and from the reactor as the basis for not requiring the entire primary system to be contained. There is no reason, in principle at least, why pressurized-water reactor systems may not employ pressure-suppression containment.

Experience and data on the use of "conventional" building structures support continued use of such buildings for containment of reactors where the stored energy is small and adequate gaseous cleanup facilities are provided. At some future date the improvements in reactor design and the reliability of safety features may reduce the probability of activity release to such a level that it will be acceptable to house most large nuclear power plants in "conventional" buildings, as the British have seen fit to do with their gas-cooled reactors that use Magnox-clad fuel.

From analyses of the information available from the Swiss and other European applications, it appears that underground containment also is feasible. Cost data are not available for construction of this type of containment in the United States.

HAZARDS OF HYDROGEN HANDLING

C. J. Barton

Current interest in the large-scale use of hydrogen in reactor technology stems from the desirability of employing this gas as a coolant in advanced gas-cooled reactors and from its application as a propellant for nuclear-powered space rockets. The scale of hydrogen use in the NERVA stage of the nuclear rocket development at the National Rocket Development Station near Las Vegas was indicated by a recent news release.[30] The facilities at the Station include a propellant tank that holds 70,000 gal of liquid hydrogen, a 250,000-gal spherical dewar for liquid-hydrogen storage, and storage capacity of 4,000,000 scf for high-pressure gas.

A previous review[31] discussed the hazards resulting from inadvertent exposure of the graphite in a gas-cooled reactor to water at high temperatures but gave little consideration to problems arising from use of this gas as a coolant in high-temperature reactors. An incident involving a hydrogen fire in a pressurized-water reactor was also reviewed.[32] Reports dealing with hazard aspects of the use of hydrogen as a coolant and with the handling of liquid hydrogen will be reviewed here.

Hydrogen-Cooled Reactors

The advantages of hydrogen compared with other gases proposed for use as coolants in gas-cooled reactors have been discussed in a number of reports[33-35] and need no elaboration here. Possible reactions of hydrogen with structural materials in a high-temperature, hydrogen-cooled reactor cause some concern, but the possibility of hydrogen-oxygen explosions is the principal deterrent to this otherwise highly desirable

use of hydrogen. A hydrogen explosion at the KIWI B-1A the week of November 5, 1961 demolished the shed housing the reactor and injured four men;[36] the accident resulted from hydrogen leaking through a faulty valve into the system and then into the shed. The hydrogen-air mixture in the shed was detonated by a spark from an unknown source, but the reactor itself was not damaged.

Michelson, Culp, and Neill[35] have considered the explosion problem in some detail in connection with the design of a hydrogen-cooled loop for the EGCR. Possible methods of preventing the accumulation of an explosive atmosphere around a hydrogen system include ventilation (dilution with air), maintaining a low pressure in the immediate environment of the system, and blanketing the system with a nonreactive atmosphere. The first method, which is employed in a large-scale hydrogen liquefication plant,[37] was considered[35] unacceptable for protecting a radioactive system because an adequate ventilation rate is usually incompatible with the safe confinement of radioactivity. The low-pressure method was deemed impractical for reactor containment vessels, leaving the blanketing method as the most practical way of preventing the accumulation of an explosive mixture.

In addition to displacing the oxygen surrounding the system, the heat capacity and thermal conductivity of the nonreactive gases are said to play important roles in inhibiting the H_2-O_2 reaction. As the concentration of nonreactive gas in a gas-air-hydrogen mixture increases, the flammability limits converge and ultimately coincide, after which the addition of more nonreactive gas assures that the mixture cannot propagate a flame. Reactor containment vessels are constructed with low leakage

rates in order to minimize escape of radioactive fission products to the environment in the event of a reactor accident. It therefore should be easy to maintain a nonexplosive mixture in the environment of a gas-cooled reactor.

Helium, nitrogen, and carbon dioxide were the nonreactive gases considered[35] for use in a containment shell. A considerable amount of information has been reported[38-41] on the flammability of hydrogen in CO_2-air and N_2-air mixtures and also on the flammability of CO_2-H_2 and N_2-H_2 mixtures in air. Only limited data on helium-hydrogen mixtures (undocumented) were found. The published data indicate that an air-CO_2 mixture containing more than 61% CO_2 cannot be made flammable by any addition of hydrogen. Also, if a H_2-CO_2 mixture contains more than 10.1 times as many CO_2 molecules as hydrogen molecules, it will not become flammable regardless of the quantity of air that might be added. Carbon dioxide was found to be superior to the other gases in both regards, and it was also said to be preferred over the other gases because it is cheaper and easier to handle and store and its high density would expedite flushing air from a containment vessel. Use of a high concentration of CO_2 or other nonreactive gas in the containment vessel would necessitate supplying maintenance workers with air or oxygen while they were inside the structure, but this would not be a severe problem.

Liquid-Hydrogen Handling

Recent books on cryogenic engineering[37,42] cover quite well the hazards of cryogenic systems. The safety measures employed in a National Bureau of Standards hydrogen liquefication plant[37] include making all

equipment air-tight that will contain hydrogen, purging the system with nitrogen until the purge gas contains less than 0.5% O_2, providing six air changes per hour outside the hydrogen-containing equipment to dilute hydrogen leaking from the system to a concentration below its explosive limit, and installing a hydrogen-detector system equipped with an alarm and "explosion-proof" electrical fixtures in the cell containing the equipment.

A French report[43] gives results of some experiments on the electrostatic charge of solid crystals in liquid hydrogen, results of some explosion tests on mixtures of solid oxygen and liquid hydrogen, and a survey of incidents involving hydrogen liquefiers in England, Scotland, Holland, the U.S.S.R., Canada, and the United States. The author[43] excluded from consideration those accidents attributable to "remarkable folly or ignorance" which, he says, lead to accidents everywhere. Conclusions based on the experimental studies and on the results of the accident survey include the following:

1. Liquid hydrogen itself presents no hazard while boiling under constant pressure. No accident has been reported involving liquid hydrogen boiling at a pressure of one atmosphere.

2. The mere presence of solid oxygen in liquid hydrogen is not enough to cause an explosion, but it may cause delayed boiling, which can destroy the container and result in secondary explosions.

3. If hydrogen flows very rapidly over solid oxygen, it is probable that the oxygen crystals become electrostatically charged and that discharge of these electrostatic charges initiates explosion.

4. All the known explosions that were unpredictable occurred in free-expansion liquefiers working at pressures between 100 to 150 atmospheres. They were generally attributed to the presence of solid oxygen.

5. Primary explosions in hydrogen-liquefier apparatus, in general, produce serious effects only if they rupture the container and produce secondary explosions. The author states[43] that "this secondary explosion can become catastrophic if the expanding hydrogen immediately encounters some liquid air or is in an enclosed and insufficiently ventilated space, where it forms an explosive mixture." It is recommended that the liquid hydrogen reservoir be constructed of metal possessing sufficient elasticity to absorb the energy produced by the combination of hydrogen and oxygen. A tin-soldered copper reservoir was reported to have survived about 10 explosions with some distortion.

6. It is further recommended that a liquefier be set up in a spacious room, that no enclosure be used around or near the apparatus, that only fireproof electrical materials be employed, and that liquid oxygen or liquid air <u>never</u> be used in or near hydrogen plants.

In connection with the recommended use of a metal container possessing some elasticity, it is interesting to note that, on the basis of both theoretical and experimental studies,[44] the conclusion was reached that a reactor vessel constructed of a ductile metal, such as type 347 stainless steel, can withstand a maximum transient pressure approximately twice the amount indicated by static tests. Several reports[45,46] have been published dealing with hydrogen hazards connected with bubble chambers and similar equipment.

A considerable effort has undoubtedly been expended on development of methods and equipment for the safe and economical handling of liquid hydrogen. Many studies have been conducted by Arthur D. Little, Inc., personnel under Air Force contracts, and three reports[47-49] have been published summarizing the results of these investigations. One report[48] is especially concerned with handling equipment, but it has some safety information. The hazards[47] associated with liquid-hydrogen handling include the freezing hazard to people who come in contact with unprotected fluid or equipment, explosions and detonations, and leaks and spills. The exposure hazard can presumably be minimized by care in designing and operating equipment. Detonations in hydrogen-oxygen mixtures occur much less frequently than explosions, but the consequences are much greater because peak pressures several hundred times the initial pressures may be reached in detonations, whereas the maximum pressure reached in an explosion is usually only 10 to 20 times the initial pressure. A deflagration can change into a detonation when the flame velocity exceeds the velocity of sound because of the existence of favorable gas compositions and container dimensions. The detonation limits for confined mixtures of hydrogen and air are 18.3 to 58.9% H_2 by volume, that is, well within the ignition limits for such mixtures. Ignition tests[50] with solid air and solid O_2-N_2 mixtures in liquid hydrogen showed that the mixture would not ignite on exposure to a hot wire when solid air of normal composition was present, but when the solid mixture contained more than the normal concentration of oxygen, ignition would sometimes occur. When pure oxygen crystals in liquid hydrogen were subjected to impact, the mixture detonated, but the addition of nitrogen to the oxygen reduced the sensitivity

to impact quite sharply. It was not possible to detonate solid air in liquid hydrogen with the apparatus employed in these tests.[51]

The leakage of hydrogen does not appear to present unusual hazards because the gas is light and tends to dissipate rapidly.[47] Precautions needed to reduce leakage hazards include ventilation of enclosed spaces and elimination of ignition sources. Spilled liquid hydrogen[52] evaporates at a rate of 5 to 7 in. of pool depth per minute, initially, and then more slowly until a constant rate of about 1 1/2 in./min is reached. If the pool ignites, the evaporation rate is in the range of 4 to 9 in./min. The danger from combustion of spilled hydrogen is believed[47] to be no greater than the danger from similar incidents with jet fuel or gasoline. Liquid hydrogen is different from these fuels, however, because it vaporizes rapidly and produces large clouds of combustible vapors. The Bureau of Mines has considered this phenomenon in some detail.[53,54] Their tests, conducted in unconfined spaces, indicate that ignition of the flammable mixtures formed above a pool of liquid hydrogen results in hazards to equipment and personnel from direct exposure to the fireball formed and hazards to personnel from exposure to the radiant energy liberated by the fireball. On the basis of these studies, the recommended distances from storage tanks to inhabited buildings and from storage tank to storage tank shown in Table IV-6 were selected. The values in the table are conservative for spills under quiescent conditions, but the authors state that the factor of safety that they afford may be eliminated by adverse weather conditions.

An investigation was also made of the hazards resulting from electrostatic charges produced during storage and handling of liquid hydrogen.[55]

Table IV-6 QUANTITY VS DISTANCE FOR LIQUID HYDROGEN

Distance to Inhabited Buildings		Distance Between Storage Tanks	
Quantity (lb)	Distance (ft)	Quantity (lb)	Distance (ft)
0-200	100	0-2,000	50
200-1,000	150	2,000-10,000	100
1,000-5,000	200	10,000-20,000	150
5,000-20,000	250	20,000-40,000	200
20,000-40,000	300	40,000-60,000	250
40,000-100,000	350	60,000-100,000	300

It was concluded that significant and long-lasting electrostatic charges can be generated in well-grounded systems, especially during two-phase flow conditions, but the charge buildup in all tests was insufficient to cause spark ignition.

Hazards can result from low-temperature embrittlement of commonly used materials of construction. This problem was considered by Eden,[56] who states that the engineer cannot rely on his ordinary experience as a guide to the choice of materials for low-temperature service. Austenitic stainless steels and similar materials having the face-centered-cubic structure seem to maintain their ductility at low temperatures and are consequently recommended as container materials for liquid hydrogen.

SAFETY IN GLOVE BOX DESIGN AND OPERATION

L. N. Howell

The increasing importance[57-59] of glove boxes is evident in a number of ways. Operations with specific materials[58,60] in glove boxes have increased, and special committees[57] and working parties[61] have been formed to study, evaluate, and make recommendations pertaining to glove boxes, the auxiliary equipment, and the operating techniques. The subject of glove boxes has received much attention at recent Hot Laboratories and Equipment Conferences,[58,62,63] and much literature has been published in recent years by those interested in glove boxes.[57-90]

The tremendous potential for widespread contamination arising from glove-box accidents, which could endanger or injure people and threaten or destroy sizeable investments in materials and equipment, makes glove-box safety important.[64] Accident experience[65,91,92] keeps concern for safety in constant focus. Several excellent reports dealing thoroughly and extensively with glove-box design and operation have been published recently.[57,61,66,67] Particular designs, materials, facilities, and operations are also well covered in recent reports[63,68-86] of more specific scope.

General Design Considerations

Glove-box design is influenced by the use for which the glove box is intended, that is, the materials to be handled and the operations to be performed, by the user's experience, by economic considerations, and by safety requirements. Although glove-box designs vary widely at AEC sites and somewhat less at British establishments, there is a tendency

toward standardization. General opinion, the natural tendency to choose or adapt from designs proved successful, and economic considerations help advance standardization, whereas vastly different end uses, unfamiliarity, and inexperience tend to result in diversification.[57,87]

Materials of Construction. The diversity in designs is demonstrated by the variety of materials of construction used. Box frames of all-welded construction made from stainless steel or carbon steel suitably painted are most common.[57] Box frames built up from aluminum extrusions are in use at Argonne National Laboratory.[73,74] Fiberglas-reinforced plastic boxes have been developed and are in use at a number of sites.[57,76,84] A unique glove box that costs only a few dollars, has a "Tinker-Toy" framework and a Masonite floor, is covered with a transparent polyethylene bag, and uses embroidery hoops for glove ports has been reported by Shore.[83]

Stainless steel is chosen for its good mechanical strength, fire resistance, and resistance to alkalis and organic solvents. It has moderate resistance to most acids; however, it has poor resistance to all concentrations of hydrochloric acid.[67] The comparative chemical resistance of several plastics and metals (including stainless steel) to hydrochloric, nitric, and sulphuric acids was tested under three different conditions and reported by Hughs and Jastrab.[76] Some users report stainless steel to be easy to decontaminate,[57] but others report it to be difficult to decontaminate.[67] These opinions obviously are based upon and reflect different experiences, since the degree of ease or difficulty of decontamination depends upon the condition of the surface, the specific contaminant, and the process of decontamination. The bare metal surface readily adsorbs activity.

Mild steel, suitably painted, is frequently used because the cost of mild steel fabrication is less than stainless steel fabrication, and the paint, properly chosen, can largely overcome the difficulties of adsorption and subsequent decontamination.[67] The cost of fabrication of extruded aluminum is less than that of welded stainless steel, and leak-tightness is easier to obtain.[74] An aluminum alloy is also used as floor material because of its high thermal conductivity. A protective coating is required when the process conducted in the glove box would expose the aluminum to acid or alkali.

Fiberglas-reinforced plastic construction offers superior resistance to chemical corrosion, ease of decontamination, and flexibility in repair and modification during operation.[67,76,84] Hughs and Jastrab have discussed the decontamination characteristics, the fire resistance, and the fire retardant characteristics of Fiberglas-reinforced plastic glove boxes.[76]

From a safety point of view, the window is one of the weakest parts of a glove box, and no single material has been found that possesses all the desired properties, such as adequate light transmission, shock resistance, abrasion resistance, chemical resistance, fire resistance, ease of fabrication, ease of alteration, and ease of repair. Glass and a number of plastic materials are used for window construction (refs. 57, 58, 63, 66, 67). Safety glass and acrylic plastic materials (Lucite, Plexiglas, Perspex) are the most widely used.[57,67] Safety glass is recommended[57,67] when the fire risk is high, although when compared with some plastic materials its impact resistance is low and it is relatively inflexible, difficult to seal, and difficult to fabricate. Since glass is difficult to fabricate, it is necessary to determine the location and make holes

for glove ports before the glass is attached to the box. After the box has been used, it is very inconvenient to modify the window.

Ventilation. The first requirement in ventilation, filtration, and atmospheric control is that the entire system must have a low leakage rate. Roberts and his co-workers[61] have reported leakage rates used to test glove boxes for leaktightness, as follows:

1. For air-filled boxes, in-leakage should be less than 0.5% of the box volume per hour at -4-in. H_2O gage with all port bungs in place.

2. For inert gas-purged boxes, in-leakage should be less than 0.05% of the box volume per hour at -4-in. H_2O gage with all port bungs in place.

Care must be taken in the design of filter mounts, which must be carefully installed to avoid leakage around the filter. Individual filters located so that they can be changed inside the box are preferable. Prefilters are required to protect high-efficiency filters from rapid clogging. For air-purged boxes, inlet filters are required, and it is preferable that the building air be filtered. Filters should be of the fire-resistant type, and flame and spark arresters may be employed. Ventilation, filtration, and atmospheric control systems are discussed thoroughly in refs. 57, 61, 66, and 67.

General Operational Considerations

The chief immediate cause of accidents in any field is human error, which may come about as a result of forgetting, being mistaken, carelessness, ignorance, or some other human characteristic in an individual or group of persons. The value of proper administrative attitudes and controls, including written operating and emergency procedures, is widely

recognized. Although the human element in safety is beyond the scope of this review, it may be worthwhile to emphasize that the advantages of properly designed equipment can easily be lost by inexperienced or careless people who fail to think safety and to work safely.[64]

Procedures. According to the "Report on Glove Boxes and Containment Encloses,"[57] operating procedures should be prepared in detail by the person responsible for the operations and should be reviewed by supervision and appropriate hazards control groups. The procedures should deal with how to avoid puncture wounds and how to avoid ingestion or inhalation of toxic materials, and they should provide instructions for repairs and disposal. Operating procedures must be updated as operations change, and appropriate measures must be taken to ensure that those who must follow the procedures are thoroughly familiar with them.

Emergency procedures[57] must deal not only with emergencies that arise within the glove-box working area but also with emergencies that may spread beyond the glove-box area. These procedures must be coordinated with the operating procedures and other similar procedures in effect at the site, and they must be understood and accepted by the groups comprising the emergency organization, including fire-fighting, rescue, medical, health physics, and criticality control groups. Measures for dealing with glove-box emergencies have been discussed by Barton.[66]

Practices. Safe operations are achieved by employing proper techniques and sound operating principles. A partial listing of successful techniques and sound practices, which are discussed in detail in the literature,[57,60,66,67] follows:

1. Glove boxes should be comfortable to the operator. Stands should be provided for operators whose height is less than average. Cotton gloves may be worn to advantage by operators with small hands and short fingers.

2. It is necessary that surgical gloves be worn and that hands be monitored when they are removed from glove-box gloves.

3. The operator should be protected from thermally hot surfaces.

4. Heat-producing equipment (soldering irons, heaters, torches, etc.) should be shut off when not in use. Gas flames should not be used in glove boxes.

5. Plastic bag-out material should be clean and dry when seals are made. Cuts should be made through the middle of a welded seam, not between two welded seams.

6. Sharp objects should be avoided. Sharp edges should be rounded or taped. Sharp objects and special tools should be kept in a separate box within the glove box.

7. The supply of liquids should be limited. Liquid-level warning systems should be used, if necessary, and drains should be available. The electrical supply should be shut off if a liquid-flow fault occurs. Liquid wastes should be disposed of promptly.

Good housekeeping is recognized by experienced personnel as one of the most effective practices for minimizing radiation exposure and unsafe occurrences.[57] Briefly, good housekeeping requires prompt action in cleanup after spills; it avoids the accumulation of waste; and it assumes order and cleanliness within the glove box on a continuous basis. Only essential materials and equipment should be kept within the glove box. The radiation dose to the hands can be reduced by keeping gloves and

glove-box surfaces clean. The importance of clean surfaces becomes clear when it is realized that gram quantities of plutonium can accumulate on glove-box surfaces.[61] Good housekeeping has been discussed thoroughly by Barton.[66]

Combating Fires and Explosions

Fires and explosions receive top billing in glove-box literature[57,61,66,75] and are generally regarded as the most serious glove-box hazards. Glove-box equipment and operating techniques are designed to minimize the occurrence of accidental fires and, when accidental fires do occur, to successfully fight them. Minimizing combustible materials of construction, excluding or minimizing flammable materials within the glove box, minimizing sources of ignition, maintaining the purity of inert gas systems, and controlling humidity and temperature are means commonly employed to avoid accidental fires. Successful fire fighting requires early detection, prompt and proper action by trained personnel, and the use of the proper extinguishant.

The avoidance of explosions is imperative, since means for venting and limiting or suppressing them have not been perfected to a satisfactory degree. Explosions are avoided by excluding flammable solvents and gases, by maintaining air flows sufficient to prevent the buildup of gas concentrations within the flammable range in air-purged boxes where flammable solvents or gases are used, by maintaining the purity of inert atmospheres in inert-atmosphere boxes, and by excluding sources of ignition. Where the possibility of an explosion exists, some means of venting is advisable to minimize the damage that may occur to the containment and atmospheric control system.

Ignition. The fire risk resulting from the use of conventional motor-driven stirrers within a glove box may be eliminated by the use of an electromagnetic stirrer[85] that has no moving part other than the polyethylene-encased rotor. Under certain conditions, static electricity may be a potential hazard.[57] A material[86] has been developed for use in coating machine tool equipment components to aid in preventing accidental detonation of high-explosive materials during machining operations. A graphite-filled, plasticized polyvinyl chloride composition produces a resilient coating with the capability to leak off or dissipate charges of static electricity. The coating is tough, abrasion resistant, and capable of being repaired if torn or otherwise damaged in use.[86]

Fire and Explosion Tests. Considerable work has been done in the United States and at British establishments on testing and evaluating the fire and explosion resistance of glove boxes. In a continuing effort,[88] a group at Harwell conducted tests in which several extinguishants were used on solvent fires in both open and closed glove boxes. In all these tests the fires were created by 100 ml of n-hexane. Water, methyl bromide, chlorobromomethane, and Pyromet powder were tested as the extinguishing agents. The powder extinguishant was most favored, although difficulties of application, believed to be not insurmountable, were pointed out.

At the Oak Ridge National Laboratory, Mackey[89] performed several tests in which approximately 3 liters of ortho-xylene was ignited on the floor of a glove box. The objective of the tests was to assess the seriousness of the fire and explosion hazard and to determine satisfactory means of prevention. These tests conducted in a glove box proposed for use in the transuranium program demonstrated the necessity for promptly

extinguishing the fire to prevent containment loss as a result of possibly burning through the rubber gloves or an explosion from carbon monoxide or solvent vapor.

In some of the tests, it was found that the fire would smother because of lack of oxygen, even though an extinguishant was not used. Surprisingly, the fire died sooner when purge air was allowed to continue to flow through the box than when the purge air was shut off. This result was probably due to cooling, since this condition also resulted in a lower maximum temperature.

Both CO_2 and water-spray extinguishing systems were tested. The fires were extinguished almost immediately upon activation of the CO_2 extinguishing system. The water spray was found to be effective under what was considered to be the most difficult test condition; that is, the condition of an open box. A fixed-volume water system released automatically with a reliable sensing and pressuring mechanism was recommended.

Fire and explosion tests were also conducted at the Argonne National Laboratory[90] on one of the plutonium metallurgy facility glove boxes to determine the effect of fires and explosions, to find the lowest oxygen concentration that will support combustion, and to test the effectiveness of standard fire extinguishers and a fused salt mixture for extinguishing glove-box fires. It was concluded that the plutonium metallurgy glove boxes are quite rugged. For small metal and alcohol fires, the fires could be safely left to burn out, if necessary, providing the gloves were not involved. In the case of an explosion, less damage was done if glove-port covers were not used. Over 10% oxygen was required for an alcohol fire in the glove box. The metal chips burned in 1% oxygen if some

additional heat was supplied. The dry chemical, Met-L-X, and CO_2 extinguishers were all excellent for alcohol fires in the glove box. A fused salt mixture was excellent for metal fires.

Summary

Along with an increasing need for glove-box operations, glove-box technology is growing. Recent literature, often including large bibliographies, helps those in the field share experiences and provides the newcomer with a broad source of reference material. Improved designs, materials, and techniques continue to be developed. The safety aspects of glove-box operations are significant and are commanding attention from an increasing number of people. In particular, more effort is being expended on studying the prevention and control of fires and explosions in glove boxes.

Safer boxes at lower costs will evolve due to the sharing of experiences through personal contact and published reports; however, the present state of wide diversity of glove box designs is likely to persist, even in the presence of tendencies to standardize, since the requirements of operations and safety will continue to vary widely.

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V. CONSEQUENCES OF ACTIVITY RELEASE

WORLD-WIDE RADIOACTIVE FALLOUT FROM NUCLEAR TESTS — PART II

L. Machta*

Radioactivity created by or during the testing of explosive nuclear devices may be found embedded entirely within the earth's crust (underground explosion), partially in the earth and partially in the air, or entirely in the air. This series of two articles discusses only those tests from which the debris entered the atmosphere, and an attempt is made to follow the history of the debris as it returned to the earth's surface or decayed. In Part I of this series,[1] it was noted that the spectrum of sizes of particles from a surface or near surface burst permitted a fraction of the radioactivity to be deposited locally. Details of the atmospheric features, such as the local wind structure, determine the location and nature of this close-in fallout. The radioactivity remaining airborne, whether particulate or gaseous, may be conveniently classified as tropospheric or stratospheric, according to the part of

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the atmosphere in which it is found after a few hours following the nuclear explosion. This division follows from the nature of the earth's gaseous envelope.

The Atmosphere

The permanent gases of the atmosphere are largely transparent to the incoming short-wave solar radiation. Water vapor in the lower atmosphere absorbs about 15% of this radiant energy. The remaining 85%, which "drives" the atmosphere, is absorbed at the earth's surface, where it either heats the soil and sea or evaporates the water. Thus, although the sun's energy from outside the earth "drives" the atmosphere, it is principally the real and latent heat sources near the ground that link the sun to the atmosphere's motions.

The atmosphere below about 300,000 ft is turbulent; eddy exchange coefficients, for example, exceed molecular diffusivity. The bulk vertical movement of air parcels involves a change in pressure that causes expansion for rising motion and compression for sinking motion. Because of the relatively slow radiant exchange between air parcels and their environment, the expansion or compression may be considered to be an adiabatic process. It is found that vigorous vertical motion in the atmosphere produces a characteristic vertical temperature distribution (an adiabatic lapse rate of temperature change) consisting of a 10°C decrease with each kilometer (3,200 ft) of increase in altitude.

On the average, the combined effect of convection on all levels in the atmosphere only approximates an adiabatic lapse rate, the average decrease in temperature being about 6.5°C per kilometer. This lapse

rate characterizes the lower layer of the atmosphere, that is, the troposphere. Even within the troposphere, the lapse rate may depart significantly from its average over deep layers. The presence of a temperature inversion (increase of temperature with altitude) signifies little vertical movement; for example, that near the ground on quiet, cloud-free nights.

Abruptly, at a certain altitude that is dependent on the season, the latitude, and the weather situation, the lapse rate of the temperature changes to a constant or increases with height. The point at which the discontinuity of the temperature gradient occurs is called the tropopause. The atmospheric layer above it is called the stratosphere (see Fig. V-1). The height of the turbulent troposphere is related to the distribution of the solar heating of the ground, that is, the troposphere is higher near the equator than the poles, and higher in summer than in winter in the temperate zone. The reason for the sharpness of the temperature gradient change at the tropopause is unknown. There are also large changes in the lapse rate within the troposphere and the stratosphere, as, for example, the overlapping tropopauses in the midlatitudes of Fig. V-1. The uncertainties in identifying the tropopause can be largely, but not entirely, resolved by the professional meteorologist.

It follows from the association of vertical motions and lapse conditions that there should be little vertical turbulence in the isothermal stratosphere. This idea of a stratified and only slightly turbulent stratosphere prevails with most meteorologists and is well supported in fact. There are real or apparent exceptions, however, when the turbulence may be large, as evidenced, for example, by aircraft bumpiness and special layers of lapse conditions in the polar winter night.

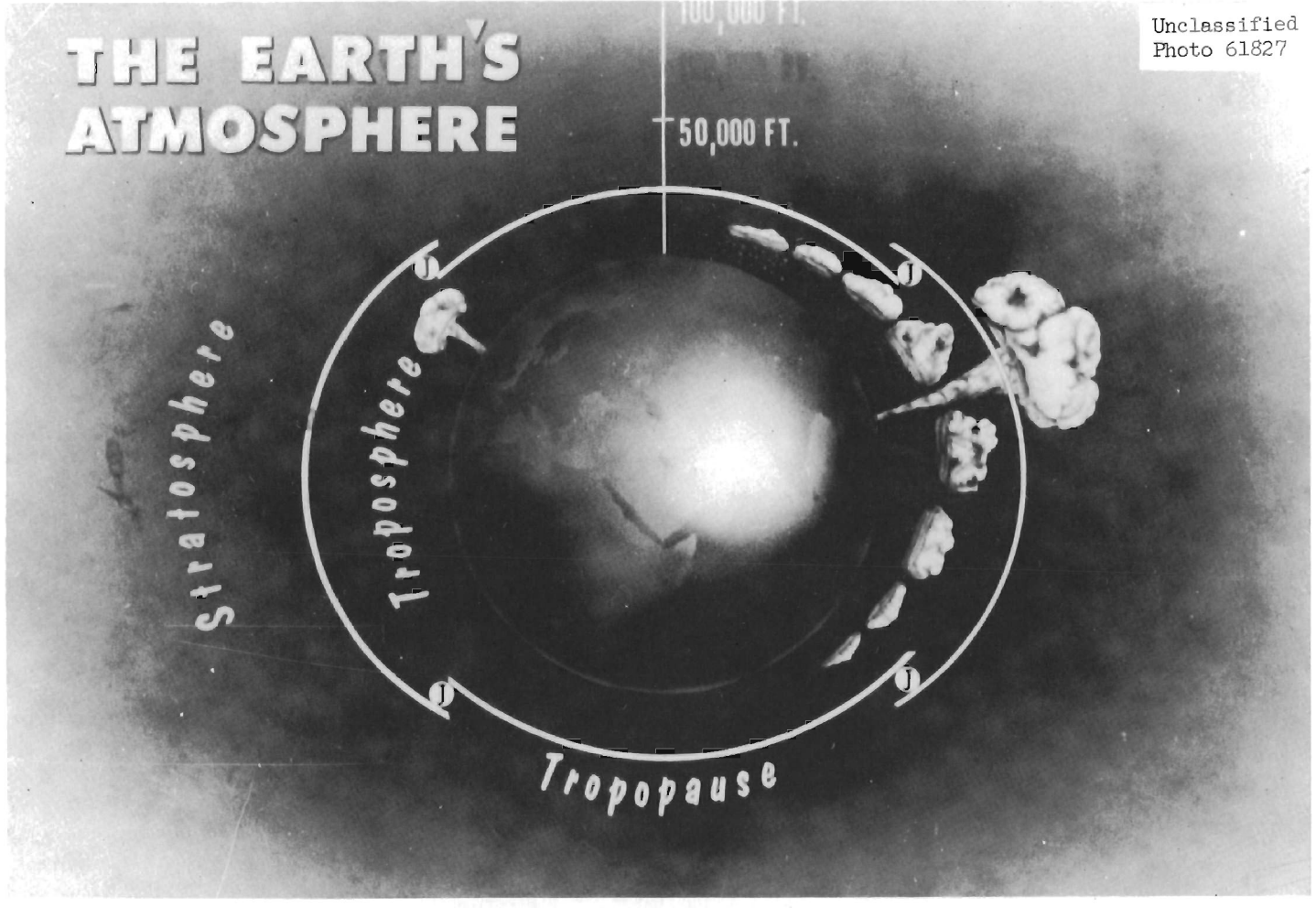

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Fig. V-1. A Cross Section of the Earth's Atmosphere Showing the Troposphere and Its Weather, the Tropopause, and the Stratosphere. The mushroom clouds illustrate the heights to which kiloton and megaton yield nuclear explosions rise.

Weather in the sense of clouds and precipitation is almost exclusively confined to the troposphere. Special unique clouds of the stratosphere, nacreous or mother-of-pearl and noctilucent clouds, are of little concern here. But the possibility of penetrations of the stratosphere by intense cumulonimbus activity presents a more significant stratospheric cloud problem. In fact this has been suggested as a prime means of extracting stratospheric aerosols and soluble gases from the stratosphere, particularly in the spring season. Below many hundreds of thousands of feet the same pressure, Coriolis, and eddy-viscous forces of the lower atmosphere control the motions of the stratosphere. The horizontal component of stratospheric airflow appears to be similar to that within the troposphere, even to a high-speed jet stream (the polar night jet). Winter winds blow from the west, while summer winds blow from the east, in general, in the midstratosphere. A strange band of winds lying within about 5° of the equator alternates between east and west wind directions with about a 26 rather than 12 month periodicity.[2]

The polar region wind patterns may be related to the mechanism for the exchange of air between the troposphere and the stratosphere. As the winter sunless night sets in, cooling creates a circumpolar wind vortex located at or near each pole. This intensifying west-to-east wind regime breaks down into several cells in January or February in some years, only to become a single vortex a few weeks later. A later breakdown into several cells, frequently but not always taking place abruptly, converts the wind into the summertime airflow from the east. The rapid changes from circumpolar to multicellular flow are associated with a warming of the polar stratosphere. This observed warming cannot

be accounted for by solar heating, especially in midwinter when the polar cap lies in darkness. It is often attributed to a compressional sinking motion. The same subsidence may then be invoked to bring higher altitude fission products and other radioactive tracers closer to the tropopause.

The troposphere, on the other hand, possesses a wide spectrum of weather phenomena. Climatologically, the main scavenging by precipitation occurs below about 10,000 ft, but, as already noted, some active clouds penetrate the entire depth of the troposphere. The amount of precipitation has one maximum in the temperature latitude that is due to extratropical storms and another in the equatorial belt that is due to the convergence of trade winds into the meteorological equator. Certain limited regions, such as orographic barriers, are notably wet, while others are arid. Experience has indicated that the amount of deposited debris correlates well with differences in rainfall for stations lying in the same latitudinal belt. The intensity of both horizontal and vertical turbulent exchanges varies greatly with time and space, but over-all they are believed to be greater than in the stratosphere. Wintertime storminess produces the greatest horizontal mixing in the temperate and subpolar zones, but convective activity over summertime-heated continents creates the most intense vertical mixing. In addition to the transport along the gradient of concentration by turbulent mixing, there is at least one organized circulation cell. The trade winds below about 10,000 to 20,000 ft converge into the intertropical convergence zone (the meteorological equator), rise, spread poleward, and sink in the horse latitudes located near 25° north or south latitude to rejoin the trade winds and close the loop.

Fallout Characteristics

Two features of the time and space distribution of stratospheric fallout have become well established during the past several years. These two features, the seasonal trends and the geographical distribution, are described and interpreted here in terms of the known behavior of the atmosphere.

Seasonal Trends. The northern hemisphere curve of both the deposited Sr^{90} and the average ground level Sr^{90} concentration in the air display a seasonal trend. A maximum appears each spring and a minimum each autumn,[3] as indicated in Fig. V-2. Historically, the meteorological cause of the seasonal variation was not universally accepted until 1959 or 1960. The reason for such doubts in a meteorological explanation lay in several directions. First, it was only in 1959 that every northern hemisphere station showed a spring maximum; the trends in Sweden and Japan, for example, were confused in earlier years. Second, many tests releasing tropospheric Sr^{90}, which is deposited within a month or so, were conducted in the spring season. Third, the Soviet Union had conducted high altitude stratospheric injections in the autum of each year between 1955 and 1958, inclusive, and it was speculated that the heavy spring fallout might simply reflect about a six-month delay between injection time and appearance at ground level. In 1959, 1960, and 1961, there were no spring tropospheric injections (the small French tests may be neglected), and yet maxima are apparent in Fig. V-2. Further, there were no Soviet tests in the autumn of 1959 or 1960, and yet, as can also be noted in Fig. V-2, there are spring maxima in the succeeding years.

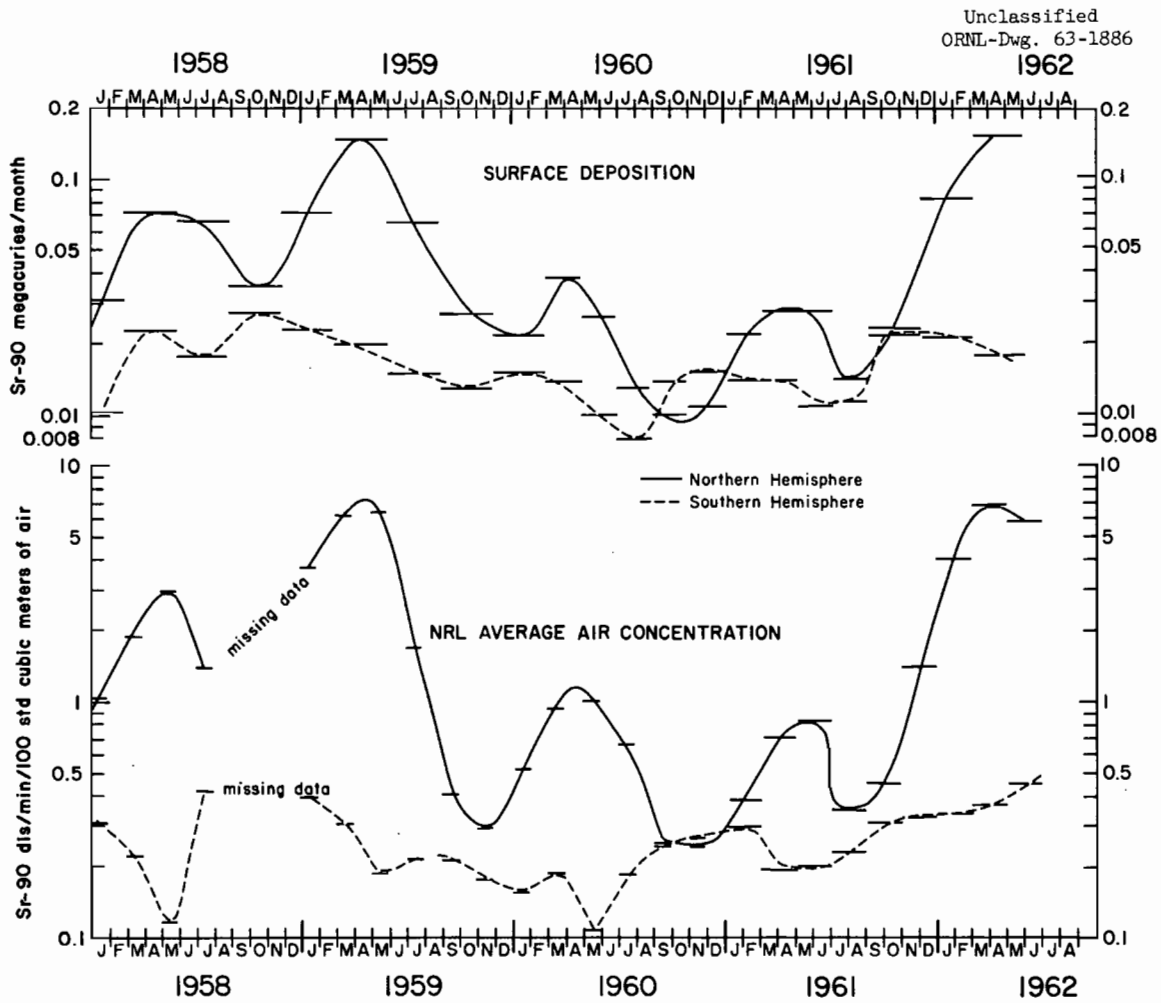

Fig. V-2. The Time History of Deposited and Average Air Concentrations of Sr^{90} in Each Hemisphere. The deposited radioactivity was derived from rainwater analyses over much of the globe; the air concentration was obtained at ground level along the 80°W meridian.

It may be observed that the corresponding seasonal trends do not appear in the two southern hemisphere curves prior to 1960 and any appearing after that date are weak. It is to be noted that the southern hemisphere seasons are displaced by six months from their calendar month of the nothern hemisphere. The reason for the absence of a southern hemisphere trend is unknown. It is conceivable, but unlikely, that the meteorology of the two hemispheres differs enough to produce such a profound discrepancy. This writer believes, however, that there were two sources of bomb debris for the southern hemisphere compared with one (the stratosphere) for the northern hemisphere. The stratosphere, it is suggested, transfers bomb debris into the troposphere with about the same seasonal regularity into both hemispheres. Since the sources of debris are principally in the northern hemisphere, much more debris is transferred to the northern hemisphere troposphere and thence into the sourthern hemisphere via a tropospheric route by turbulent exchange across the equator. It has been pointed out that the heavy equatorial rainfall scavenges debris in its vicinity; however, it has also been established that fission products from 11° north have found their way into the sourthern hemisphere across the meteorological equator.[4] Figure V-2 shows that the movement of debris along the concentration gradient because of eddy diffusion will be southward. By 1960 the hemispheric concentrations almost equalized, and only the stratospheric source became significant. Thus a seasonal trend may be found from the station records.

The amplitude of the seasonal variation depends on the stratospheric burden. As far as the period in Fig. V-2 is concerned, the heavy testing in 1958 and 1961 was followed by particularly large spring maxima.

Qualified predictions of a 1962 peak twice that observed in 1959 on the basis of twice as much Sr^{90} from the Soviet 1961 tests have proved to be in error. The reason lies in a need to carefully specify that portion of the stratosphere which will empty in the first spring following injection. Thus two of the 1961 Soviet tests with multimegaton yields apparently inserted their debris to such great heights that re-entry into the troposphere was delayed for at least a year. As of this writing, it appears that the altitude division between stratospheric fallout deposited the first spring after a polar injection and that delayed a year or more is about 55,000 to 60,000 feet, the altitude of the continuation of the tropical tropopause which can sometimes be traced by the thermal structure to the pole.

Seasonal variations in ozone (O_3), a nonradioactive gas, similar to fallout have been observed in the troposphere and lower stratosphere for many years.[5] In the past several years, measurements of tropospheric Be^7, a cosmic-ray-produced radionuclide that has its main (but not exclusive) source in the stratosphere, also contain a similar spring maximum at most stations.[6] In 1959, a seasonal variation of the ground-level C^{14} had its northern hemisphere maximum in July rather than in April or May. Carbon-14 is measured as gaseous CO_2 and does not have the rapid tropospheric sink of the particulates. Rather, its loss from the northern hemisphere troposphere is to the southern hemisphere troposphere on a time scale of months or to oceanic uptake on an even longer time scale.

Since so many of the stratospheric tracers contain a seasonal trend in the troposphere, it is most plausible to hypothesize a cycle in the exit rate from the stratosphere. In the northern hemisphere, the cycle

consists of an increase to about April, followed by a sharp decrease to zero or very low until about October. All the previous tropospheric findings appear to be consistent with this kind of history in the exchange of air with the stratosphere.

Why does it occur? One explanation is the change of the mean altitude of the tropopause, which has a minimum in winter and a maximum in summer. Thus the rate at which stratospheric air may be ingested into the troposphere reaches a peak in spring. The sense of the change in tropopause altitude can qualitatively explain the seasonal variations in stratospheric tracers in the troposphere. Quantitatively, the importance of the meteorological process is uncertain because tracer inventories in both the lower stratosphere and the troposphere are too poorly established. Gustafson, for example, has suggested that the tropopause height change can account for the Be^7 seasonal cycle in air concentration at Argonne, Illinois. It has never been established that the changing of the mean tropopause altitude actually results in stratospheric air of the same hemisphere being left behind in the troposphere, although this must be considered highly likely. It is this writer's view that the changes in the temperate and polar tropopause height contribute to but do not play a major role in explaining the seasonal trends.

The ozone and Be^7 concentrations[5,7] in the lower stratosphere of the north temperate zone appear to also contain a seasonal trend similar to that of the troposphere but with maxima displaced several months earlier. This suggests a meteorological process contolling both downwind transport within the lower stratosphere (the concentration increases upward) and through the tropopause. Finer analyses of the seasonal trend in ground-

level fission-product concentration reveal a peak appearing first in subtropical latitudes (say, Miami, Florida) and a month or two later at sub-polar latitudes (say, Ft. Churchill, Canada). Again this suggests a more complex process than mere elevation of the tropopause.

The late winter and early spring season is the period of temperate latitudinal storminess, with the storm tracks generally shifting northward with advancing spring. Associated with the intense weather systems are marked perturbations of the tropopause. There is indirect evidence, based on dynamical and humidity considerations, that these storms bring stratospheric air deep into the troposphere. A study of one particular mode[8] for explaining this process was initiated in the spring of 1963. Weather processes in this category include the possibility that transport takes place through the gap in the tropopause (see Fig. V-1). This writer considers the assorted dynamical mechamisms connected with winter and spring large-scale storms as the most important contributors to exchange of air between stratosphere and troposphere.

Another weather process that can bring stratospheric gases and aerosols into the troposphere or to the ground is the thunderstorm that penetrates the tropopause. Such storms are believed to be most frequent in the spring season over continental areas. Radar reports of the tops of weather clouds strongly suggest that thunderstorms can penetrate into the dry stable stratospheric air. Certain instances of intense spring rainout of radioactivity are suspected as being in this category.[9] As concluded with respect to the rising tropopause in the spring season, this writer has doubts whether thunderstorms or other cloud scavenging of stratospheric air masses can quantitatively account for the spring fallout maximum.

In summary, there are several processes that reach a maximum frequency in the spring and which potentially can account for the spring fallout peak. Of those noted above, the dynamical processes associated with major storms hold the most promise of quantitatively transporting enough of the stratospheric tracers into the troposphere. It must be admitted at this time, however, that a definitive explanation of the spring fallout maximum does not exist.

Geographical Distribution. Ground-level activity concentrations in air and the deposition of fission products appear to show slightly different north and south profiles, but the main features are common to both. Both show a maximum lying between about 30° and 50° and a minimum at the equator, and both probably have a secondary minimum in polar latitudes, although the 80th meridian air-concentration data, at times, suggest a renewed increase northward from Ft. Churchill, Canada, to Thule, Greenland. The main difference between air concentration and accumulated deposition of long-lived fission products in soil is the northward displacement of the peak of the soil profile. The 80th meridian air concentration peak appears at 25° or 30° north,[10] whereas the world-wide Sr^{90} content in soils has a maximum at about 45° north.[11] The southern hemisphere data are too sparse to define a difference, but the suggestion of a poleward displacement of deposition versus air concentration can also be detected. In part, the latitudinal difference between the peaks in the north and south profiles results from the global sampling networks. Soils are collected virtually world-wide, but the best air data derive from the 80th meridian network. In fact, it has been suggested that the air concentrations might be expected to be shifted equatorward on the

east side of the two major continental masses, since certain meteorological features have their southernmost displacement here.

There is a difference in the meteorological mechanisms responsible for the final transit from the stratosphere to ground or filter paper. The deposited radioactivity is primarily brought down in precipitation; rainy regions may have over 90% of the particulates in rain or snow, whereas arid regions which have less fallout, in general, may have less than 50% of the fallout in precipitation. The rest deposits as dry fallout by impaction, etc. As already noted, precipitating clouds originate many thousands of feet above the ground. Further, there is evidence that many, if not all, of the fission products attach to the precipitating elements during their growth from the surrounding water vapor. Thus, the distribution of deposited radioactivity is related to the concentration in the lower or midtroposphere and to the amount of precipitation. On the other hand, the ground-level air concentration depends on the downward mixing from the same lower or midtropospheric levels. This mixing process may possess a different geographical distribution than that of precipitation types or amounts.

The midlatitude (30° to 50°) peak in the stratospheric fallout has suggested to many that the exit from the upper atmosphere occurs exclusively at about the latitude of maximum ground deposition or air concentration. Other processes within the troposphere must be contributary, however, to the location of the peak. For example, the polar regions are both arid and, in general, are shielded from the midtroposphere by a cold stable layer near the surface that prevents downward mxing. These

processes probably produce the poleward decrease in ground-level fallout and air concentration.*

Air concentrations in the upper troposphere over the polar latitudes appear to be at least as great as in the temperate zone. This suggests that the exit from the stratosphere is not limited to the temperate zone, although a word of caution is in order. Horizontal mixing in the middle and upper troposphere is intense in the winter half of the year. If air from the stratosphere entered the upper troposphere in a narrow latitude band, it might approach within 90% or so of its long-term uniform distribution within a matter of days to weeks between 25° and 80°. Thus it is not at all clear whether penetration through the temperate or polar stratosphere or both control the exit into the troposphere. The upper air measurements appear to rule out the equatorial stratosphere as the source of tropospheric air masses, since there is a decrease of tropospheric radioactivity concentrations approaching the equator. The horse latitudes between 20° and 30° represent an arid region of the globe. This helps to account for the sharp equatorward decrease in deposited fallout. The temperate latitude industrial and agricultural zones bear the brunt of the stratospheric fallout. The bulk of the stratospheric fallout originating in one hemisphere deposits in the same hemisphere. The exception is the fallout from very high-altitude injection.

In summary, the midlatitude maximum in deposited fallout and ground-level air concentration can probably be explained by the following features. First, the exit from the stratosphere occurs in the temperate or polar

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*It must be noted, however, that existing methods for collecting deposited fallout in pots or from soils are unsatisfactory in the polar regions.

latitudes or both rather than in the tropical or subtropical latitudes. Strong horizontal mixing produces a nearly uniform north-south distribution in the middle and upper troposphere between about 25° and the pole. Less intense horizontal mixing in the tropics combined with other factors accounts for the decrease in fallout toward the equator. The small winter precipitation and shielding of the ground explains the uncertain decrease of fallout toward the poles.

Tropospheric Fallout

The prime concern with tropospheric fallout lies in short-lived radioactivity; such radionuclides in stratospheric fallout will decay harmlessly before re-entering the troposphere. Another concern with tropospheric fallout is the possibility of intense localized deposition in precipitation or by other mechanisms. This means that tropospheric fallout can enter into the food chain or man's environment early in its history before significant dilution or decay has taken place.

Precipitation Scavenging. There have been several instances of the coincidence of a fresh cloud of debris from a nuclear test intersecting a thundershower with a resultant heavier-than-usual fallout in the limited area. Probably the most well-known case took place in the Albany-Troy region of New York State on April 26, 1953. An intense thunderstorm intercepted the seventh cloud of the Upshot-Knothole Nevada test series after 36 hr at about 40,000 ft, according to meteorological inferences.[12] Albany reported a 24-hr gross beta deposition of over 10^7 dpm/ft^2 on gummed paper; this "hot spot" was confirmed on May 1st by aerial reconnaissance, which found gamma doses of over 0.2 mr/hr extrapolated to 3 ft

above ground. Six other gummed film stations were located within 150 miles of Albany, but none reported values in excess of 50,000 dpm/ft^2, a value found frequently in the network. Hourly precipitation rates reflecting thunderstorm scavenging superimposed on the meteorologically reconstructed 40,000-ft cloud trajectory pin pointed the localized time and region "hot spot." No radioiodine measurements were made in the Albany-Troy area in 1953, but Lapp[13] has estimated that the thyroid dose from this event alone might have been as great as 30 rads to infants 6 to 18 months of age.

It is apparent from the small dimensions of the affected area that some "hot spots" may go undetected. The probability that a high-reaching thunderstorm will coincide with a still undiffused nuclear cloud is small, so the number of undetected cases must be equally small. It is not believed at this time that other properties of the nuclear cloud, such as its nucleating (i.e., serving as condensation nuclei) or ionizing effects, help induce precipitation scavenging.

Dry Fallout. As opposed to the localized thunderstorm scavenging, dry fallout covers a broader area during the passage of a low altitude cloud. Nuclear clouds from detonations near the ground with yields of several kilotons or more generally rise to over 10,000 ft; those above about ten kilotons will rise to the temperate tropopause. The clouds from small-yield tests that remain near the ground are carried slowly out of the controlled area, but the initial weak source strength, the dilution from vertical mixing, and the dry fallout result in low offsite contamination. On June 13, 1962, however, a vented, underground detonation in Nevada produced I^{131} levels in milk in the Settle, Washington,

area during the following weeks in excess on 100 μμc per liter of milk.[14] Since routine offsite surveillance was not as extensive prior to the 1959 moratorium, it is uncertain what levels may have passed unnoticed downwind from the Nevada test site from low-altitude cloud passage.

In late September 1962, a case of marked dry fallout came to light.[15] High ground-level gross-beta air concentration measurements moved successively around a high-pressure system over the eastern United States, appearing first along the middle Atlantic coast, then the southeastern United States, and then showed the highest concentrations in the lower Mississippi Valley. The dating of the debris near New York City and Washington, D.C., strongly suggested an origin date corresponding to two Soviet tests, which took place about six days earlier on September 10, 1962 at Novaya Zemlya, U.S.S.R., in the Arctic. The meteorological back trajectories from the contaminated region in the United States headed northward towards Greenland but became unreliable because of the loss of weather data. Forward trajectories beginning on September 10th from Novaya Zemlya were also uncertain in passing through the Artic and near Greenland, but those that were computed fell short of reaching the United States east coast by September 16th. The meteorological conditions producing the extremely high ground-level air concentrations in the midwestern United States (one station reported in excess of 1000 μμc of gross beta activity per cubic meter) can be only pieced together from indirect evidence. The upper tropospheric air over Novaya Zemlya was contaminated by a kiloton-range device and the lower portion of a megaton detonation. Any debris left in the lower Arctic atmosphere could not reach the United States in six days. An infrequently occurring trajectory carried the

debris southwestward, initially in the upper troposphere, in a subsiding air mass. The subsidence of a large equatorward moving air mass results from the spherical shape of the earth and continuity considerations. Thus the layer of contaminated air that usually remains at roughly 30,000 ft in the course of a normal west-to-east trajectory was brought to much lower elevations. Over the United States land mass, the layer may have subsided to 10,000 ft or lower, where convection of the heated land mass mixed the debris to ground level. This convection and continued sinking appears to be the main explanation for greater ground concentrations in the Mississippi Valley than in the southeast. Wide areas of the southeast and midwest United States reported milk concentrations of I^{131} in excess of 100 μμc per liter following this contamination episode.

In November 1962 another case of dry fallout over the United States from a Soviet nuclear test took place. Again the meteorological reconstruction of the path from the source to the United States was unreliable or confusing. The subsidence of the air mass on the southern side of a high-pressure system paralleled that of the September 1961 case, however, as did the increasing ground-level radioactivity concentration as the air moved westward over the southern United States.

It may seem surprising that two cases were documented in 1961 and 1962, and yet none appear to have taken place prior to 1959. In large part the explanation lies in better surveillance in recent years. Previous cases may have gone unnoticed. The weather does change from year to year, however, and there were not enough years of testing before 1959 to establish a climatology or any precise measure of expectation.

Unique Bomb Tracers

The same fission products, in general, are released to the atmosphere by all nuclear tests. The interpretation of the fallout history often requires a designation of the altitude or latitude source of the fallout. This identification can be made from isotope ratios which provide an age. Unfortunately tests have been conducted with such frequency that the age identification becomes ambiguous. To this must be added the uncertainty in the age calculation resulting from fractionation, use of differing fuels, and the accuracy of the method. For these reasons, emphasis has been placed on the interpretation of atmospheric transport derived from two special tracers added to the atmosphere by the U.S. in 1958.

Tungsten-185 Tracers. Tungsten-185 was inserted into the atmosphere by nuclear tests during the summer of 1958 at the Eniwetok Proving Grounds. The tests were conducted so that an unknown amount of local fallout took place and the W^{185} was largely limited to altitudes below approximately 80,000 ft. By the beginning of 1959, almost all the W^{185} left in the troposphere could be expected to be washed out, and only that portion initially injected into the stratosphere would represent the atmospheric burden.

The time and space distribution of the ground-level air concentration and deposition of W^{185} in 1959 confirmed the two features already noted earlier, that is, a seasonal trend with a spring maximum (in the northern hemisphere) and a peak in the temperate latitude. In addition, it confirmed another spatial distribution hypothesis, that is, the unequal partitioning of material added to the lower stratosphere at 11° north. A 2:1 partition favoring the northern hemisphere was deduced from the

Sr^{90} fallout, but it was subject to uncertainty because of contributions from tropospheric sources almost entirely located in the northern hemisphere and United Kingdom tests at Christmas Island. The rough 2:1 split found in the Sr^{90} observations was supported by the W^{185} distributions.

Probably the most important contribution of the W^{185} experiment lay in its stratospheric distribution. One theory of stratospheric transport (supported, among others, by this writer) called for a slow upward air current through the equatorial tropopause and lower stratosphere followed by a poleward drift and sinking back into the troposphere. This meant that the peak concentrations at 11° north between the tropopause (about 55,000 ft) and the nuclear cloud tops (about 80,000 ft) should be transported upward and poleward. The peak concentration of W^{185} was, however, at or near 11° north at the maximum sampling altitude of 65,000 to 70,000 ft for as long as measurements were made (at least 2 years). The conclusion of this finding rules out any rising current through the 65,000-ft level in the equatorial stratosphere. The results of the W^{185} experiment are not inconsistent with, but do not support, a rising air motion through the equatorial tropopause up to about 60,000 ft.

Rhodium-102 Tracer. About 3 megacuries of 210-day Rh^{102} was introduced into the upper atmosphere on August 12, 1958 during a nuclear test aboard a rocket fired at Johnston Island (17° north).[16] The altitude of detonation was about 100,000 ft, but the extent of the rise of the cloud from the megaton device is uncertain, the best opinion being that the stabilized cloud reached about 300,000 ft. About 0.3 megacuries of Rh^{102}, unfortunately, were also added to the lower stratosphere by some tests

in the summer of 1958. Thus, the source of 1958 and early 1959 Ru^{102} concentrations was ambiguous. In October 1959, aircraft sampling high in northern latitudes encountered increasing concentrations that grew in the succeeding months and were believed to represent the leading edge of the main high altitude Rh^{102} cloud.

The patterns of the distribution became clear with time. The greatest concentrations could be found at the highest aircraft sampling altitudes (65,000 to 70,000 ft) and closest to the poles. Further, although data from the southern hemisphere were more sparse than those from the northern hemisphere where they could be obtained at comparable altitudes, latitudes, and times, the concentrations were roughly the same. Since Johnston Island lies at 17° north, the appearance of debris at about 70,000 ft and in greatest concentrations close to both poles presents the meteorologist with a puzzle. One suggested explanation calls for rapid horizontal mixing because of molecular diffusion (requiring the cloud to have stabilized at perhaps 500,000 ft), followed by fairly quick settling of the fine particles to about 250,000 ft, and downward mixing from about 250,000 ft to 175,000 ft. At this point preferential mixing over the poles brings the Rh^{102} to the lower stratosphere in the winter season of the year where it finds its way into the troposphere to participate in the spring maximum.

Spring maxima were observed at ground level since 1960 wherever measured in north temperate latitude stations. Since the Rh^{102} was found in equal amounts in the stratosphere of each hemisphere, a spring maximum might be expected to appear in the Rh^{102} ground-level measurements of the southern hemisphere, there being no net flux between hemispheres.

In fact, the Rh^{102} ground-level air concentration measurements in 1959 and 1960 at Santiago, Chile, display a marked seasonal variation of Rh^{102}, with a peak in December.

Conclusion

Atomspheric transport of bomb-produced debris represents but one link in the chain connecting testing to possible human hazards from world-wide fallout. Much must still be learned by the meteorologist; some data present unresolved puzzles, not all of which have been described above, but the predictions in terms of the generalized rather than detailed times and areas of the amount of fallout to be deposited are at least as good as those produced by other scientific disciplines.

The U.S. Atomic Energy Commission, U.S. Department of Defense, U.S. Public Health Service, the Weather Bureau, and other government and private agencies are hard at work collecting and interpreting data to better document and understand the characteristics of world-wide fallout. There is promise of improvement. As in the past, the improvement in predictions may result from an empirical extrapolation of analogous cases rather than from a true appreciation of the atmosphere's behavior.

CLOTHING FOR USE IN CONTAMINATION ZONES

H. M. Butler

Radioactive contamination in the form of dust or other uncontrolled particles remains one of the major potential hazards in the atomic energy industry today. Its existence in this expanding industry is as natural as smoke is to the steel industry and, consequently, has dictated a continuously vigilant awareness of its presence. This awareness makes possible, through protective and control methods, the safety of personnel within the industry who are potential victims of such a health hazard.

Protection against radioactive materials that might enter the body through the mouth or wounds or by inhalation or skin absorption is achieved largely by adequate ventilation to keep fumes, dust, and vapors at low concentrations in areas where work is done; by enclosing the sources; by strict industrial housekeeping to prevent or control contamination of surfaces with which workers may come in contact; by vigorous training and discipline; by vigilant radiation detection controls; and, where necessary, by special protective masks, clothing, and other apparel.[17] The principal purposes for all radiologically protective apparel are to provide the wearer with protection from contaminants and to control the spread of contamination. Protection from contaminating materials is of prime importance because radioactivity in the human body can be reduced only by biological turnover or radioactive decay, except in a very few cases.[18]

Most protective apparel currently in use throughout the atomic industry is, in a general sense, of two types: Category I, apparel to protect personnel from loose radioactive material or low or moderate

concentration in a chemical and physical state that presents a potential for radiation exposure by uptake through external body surfaces or by becoming affixed to external body surfaces; and Category II, apparel for protection against loose radioactive material in high concentrations or for protection against material that has the ability to diffuse through and counteract the effectiveness of clothing belonging in Category I. Common to both these categories are devices, such as masks, hoods, and respirators, which offer protection against radioactive material in a chemical and physical state that presents a potential for radiation exposure through inhalation or ingestion.

In evaluating the protective quality of any garment or portion of apparel, the state of the contaminating material becomes a basis determinant.[19] The contaminant, although comprised of various radioactive elements, is always found in either a wet or a dry state and is transmitted as a solid, a liquid, or an aerosol. Garment fabrics may, therefore, be selected to accommodate the wet or the dry state. Standard closely woven cotton fabrics may be utilized for protection against dry contaminants, whereas rubber or plastic apparel is required for wet contaminants. The garment should, of course, be designed to be put on and taken off quickly and easily and in such a fashion that contaminant transfer to the body is minimized or eliminated.

Category I Protective Clothing

Various and sundry components make up the complete unit of protective clothing. The major component of this unit is the coverall. A cotton type of coverall appears to be the most widely used, but limited

use is made of vulcanized and canvas materials. Studies made at the Savannah River Plant revealed that 7.8-oz drill, a closely woven cotton fabric with a 72/60 thread count, is the most satisfactory coverall material.[20] The basis selected for comparison in the study was the air flow rate at a constant pressure drop through a specific cross-sectional area of a sample swatch of material. Coveralls made of the selected fabric provide twice as much protection against penetration of particulate matter as the previously used herringbone twill. In addition, drill is more durable, lighter in weight, and costs less than herringbone twill.

Work conditions often require that personnel wear two pairs of coveralls. This procedure is not always necessary for protection of the person but is, rather, a technique that helps limit the spread of contamination beyond desired boundaries. The outer or most-contaminated clothing can be removed, leaving the inner clothing for use in a buffer, or regulated, zone that sometimes surrounds a contamination zone. Where this technique is practiced, it is advantageous to have the coveralls segregated and marked, because the inner clothing will require less attention at the decontamination center. A survey at a decontamination center revealed that 90% of the laundered coveralls that were rejected for reuse because of excessive residual contamination were contaminated to a level of 1 to 3 mrad/hr. Since the residual contamination becomes fixed in the garment through laundering, coveralls with activity in this range can be reused as the outer garment.

Some less important features of suitable coveralls are listed below:

1. They should be made of a material that has low shrinkage, since most coveralls are laundered and reused many times.

2. They should have a sturdy strap across the front so that personnel monitoring devices can be firmly attached.

3. They should have buttons of material that will not deteriorate during laundering. Brass is a satisfactory material for buttons.

4. They should not have openings that are difficult to seal against leakage of contamination. Many times it is necessary to seal pockets or other openings with tape.

5. They should have a minimum of flaps or other dangling attachments that increase the possibility of the suit being snared or torn.

Laboratory coats may sometimes be substituted for coveralls as protective clothing. Their use in areas of mild contamination is practiced, and they are widely used when personnel are entering a mildly contaminated area for observation purposes only.[21] The coat serves mainly to prevent contamination of personal clothing, since entry into these zones is usually made without changing clothes.

Gloves and shoe covers, when added to coveralls, complete the unit described as Category I apparel, except for devices to protect the head and respiratory system. Several different types of materials are used for protective gloves, with emphasis placed upon features such as length of life, permeability characteristics, and design. Closely woven cotton fabrics are commonly and successfully used where the contamination involved is in a dry solid state and of low concentration. Rubber, plastic, or neoprene are utilized for protection against liquids and materials of high contamination. Natural rubber is not satisfactory for heavy-duty applications or for use where extremely high concentrations of contamination are present because of its high rate of deterioration, and butyl

rubber is undesirable because of its high percentage of breaks and its manufacturing difficulties.[22] Milled neoprene, although it has high permeability, appears to be the most universally suitable material available for protective gloves. The permeability of neoprene is insignificant when considered in the light of the low probability of breaks that would permit the introduction of hygroscopic material. The use of low permeability as a criterion for material choice should be resorted to only after it is concluded from thorough analysis of the intended glove application that permeability is the limiting factor.

Glove thickness is a prime consideration and is usually determined on the basis on the required suppleness versus safety. A balance must be achieved that will permit the needed dexterity of the individual and yet provide safety from the hazard present. For most production applications, a glove having a nominal thickness of 0.030 in. is recommended. Where protection against contamination by isotopes emitting low-energy gamma and beta radiations is desired, gloves with heavy-element additives are a significant means of reducing exposures to the hands.[23] Gloves of this type are fabricated by sandwiching a layer of heavy, leaded neoprene between two layers of standard neoprene. Gloves with a lead equivalent of 0.5* decrease the exposure to the hand from x-rays or gamma radiation of less than 0.1 Mev by several orders of magnitude.[24]

The important factor in determining the length of the glove is the opinion that the hand of the glove should seat at the ends of the fingers rather than at the web between fingers.[22] This permits small objects to

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*This indicates that a thickness of leaded rubber of 1 mm protects to the same degree as a lead plate of 0.5 mm.

be picked up and handled with much more ease and dexterity than would otherwise be possible. A 3-in. middle finger provides this condition for most users without sacrificing comfort in the glove. A standard size of 9 3/4 has been suggested. It is best for the glove to fit freely enough to permit the use of a thin inner glove when necessary.

Shoe covers do not need to offer quite as great a protection factor as gloves, but they should be made of a material that has fairly low permeability and they should be ruggedly constructed. The pressure and abuse applied to the shoe covers during use is normally much greater than that applied to gloves. Polyvinyl chloride appears to be the most popular shoe-cover material, but rubber, cloth, and paper are also widely used. The plastic material is the least permeable, but it is the most easily torn, and it is susceptible to breakage during cold weather. A cotton fabric cover offers good protection against most types of dry particulate contamination. From an extensive study made at a reactor during shutdown operations, it was learned that the particulate contamination level on the inside of used cloth shoe covers averaged 1/400 the magnitude detected on the outside.[25] Paper shoe covers of the overshoe type are adaptable to some operations where numerous personnel require protection from moderate concentrations of dry particulate contamination.[26] The paper covers can be discarded by removal to the burial ground or burned after use.

Laundering of cotton and plastic shoe covers and gloves is an effective way of reducing operating costs where large numbers of these items are used. In most cases they can be restored to normal use without creating an additional hazard to personnel. A garment containing as

much as 1 to 2 μc of mixed fission products, or the equivalent, can be laundered safely without special precautions.[27] In general, however, most contaminated garments, even after being washed, do show some residual contamination, but in many cases the contamination is not readily detectable by using an ordinary GM survey instrument. Effectively laundering the rubber-type surgeons gloves is difficult, and special techniques must be employed because of the tendency for them to conglomerate.[20] Moderately hot water should be used, and the gloves should be treated with talcum powder before being placed in the water.

Many sites that handle radioactive materials have special laundries for protective clothing. These laundries not only permit the reuse of contaminated clothing but, also, by the use of good safety and radiation practices, have accumulated many years of satisfactory operation, as described in a recent article in Nuclear Safety.[28]

The clothing described above is not universally adequate for protection against all contaminants that might be encountered and is definitely not adequate for protection against any contaminant in sufficiently high concentration. No effort is made here to outline what levels of radioactive contamination would make the Category I clothing inadequate, because the conditions of each use of protective clothing must be individually evaluated.

Category II Protective Clothing

A whole-body air-pressure-ventilated suit of some form is the major component of Category II clothing. The pressurized suits vary in design and material, with numerous combinations available. The suits are most

commonly one or two piece, with the air for respiratory purposes being used to inflate the suit and to provide a positive-pressure region in which to enclose and protect an individual.[29] Where maximum personnel protection against contamination is desired, the one-piece suit is recommended. Two-piece suits provide adequate protection against most contamination conditions if the air flow rate is sufficient; they are more comfortable to wear; and they are less costly to replace or to maintain. The most commonly used suit materials are polyvinyl chloride, rubber, nylon, Saran plastic, and neoprene. Polyvinyl chloride is the most preferred material in that it is light, relatively impervious, easily decontaminable, and inexpensive; further, the seams can be electronically welded.[30] Rubber and neoprene offer many advantages over polyvinyl chloride and are especially adaptable to some applications where rugged construction is a requirement, but their weight is a severe disadvantage. The weight of the rubber suit (approximately 30 lb) is three times that of most polyvinyl chloride suits, and at low air flow rates, the air pressure inside the suit is insufficient to support the suit.[31] Nylon-reinforced vinyl sheeting is a desirable material for suits to be used for heavy work within a severely contaminated area. The nylon mesh material reduces the danger of large tears which would seriously contaminate the inside of the suit.[32] Small tears are of little concern, since the inside of the suit should be positively pressurized when exposed to conditions which could cause internal suit contamination.

Since the whole-body suits, when performing properly, resist entry of loose contamination through cracks and openings, the permeability characteristics of the suit are important. Rubber and neoprene are the

least permeable of the materials commonly used, and the Saran-type plastic material is the least permeable of the plastics.[33] A test of a vinyl plastic suit with an H_2^3O* atmosphere showed that the maximum suit air activity reached 0.49% of the maximum ambient air activity during exposure.[34] The suit was somewhat less permeable, however, to atmospheres containing other contamination.

The whole-body air-pressurized suit loses its effectiveness if the suit air supply is not adequate to maintain the inside of the suit at a pressure that is positive with respect to its contaminated surroundings. Hence, the air flow rate is a critical factor. Not only does a low air flow rate increase the possibility of contamination entering the suit through openings, but it also allows the CO_2 concentration to increase above a desirable level. Air flow rates for most suits range from 3 to 8 cfm, depending on factors such as the surrounding temperature, the type of work to be performed, and the length of enclosure. A medium flow rate of 5 cfm has been found to be ideal for most conditions.[31]

Decontamination of the whole-body suit is a routine function when it is found that the contamination level is low enough for safe processing. Unpigmented polyvinyl chloride can be decontaminated more easily than pigmented material of the same type[30] and repeated decontamination does not appear to affect the over-all integrity of the plastic.[33] Rubber suits are harder to decontaminate than neoprene, especially if the characteristic deterioration of rubber has set in.

Accessories that may be used in conjunction with the pressure-ventilated suit are numerous. Rubber or neoprene surgeon-type gloves are

*Tritium penetrates, to some degree, all materials commonly used in the fabrication of protective equipment.

usually worn over the suit gloves, and some suits are designed so that the glove is a necessary addition. To eliminate unnecessary decontamination, the foot portions of the suit can be covered with standard plastic shoe covers. Provisions may be made for an intercommunication system using earphones and a throat microphone. Some suits can be equipped with filters that permit the escape of exhaust air while under pressure but ensure a temporary supply of air if the main air supply fails.

Respiratory-Protection Equipment

Category II apparel, because it completely encloses the individual and prohibits him from being exposed to the surrounding atmosphere, provides, in most cases, an adequate respiratory protection system. There are some cases, however, where additional respiratory-protection devices must be added to the whole-body suits, especially if contaminants such as H_2^3O which can penetrate (physically, not radiologically) the suit are to be encountered. Category I apparel must always be supplemented with respiratory-protection devices if the air activity is of a level that dictates protection.

The fundamental aim of respiratory protection is to prevent internal exposure from inhalation of radioactive materials. In some cases it is essentially an emergency measure and is employed when containment is inadequate and contamination becomes airborne, whereas in other cases it is a routine measure practiced where entry into extremely contaminated areas is necessary. Respiratory-protection devices become necessary in any case where the average weekly concentration of radionuclides in the air exceeds 1/10 of the maximum permissible concentration for a 40-hr

week[35] and should also be worn during operations that produce radioactive sprays, active volatile products, or dust of any kind.

Respiratory-protection devices currently used in conjunction with the previously described apparel may be grouped into four general classes: (1) mechanical-filter respirators, (2) air-purifying respirators, (3) self-contained breathing apparatus, and (4) supplied-air respirators. The first two classes of equipment process the air that is breathed by the worker from his surroundings. These devices will provide protection from airborne particulate radioactive material at levels up to 100 times the maximum permissible concentration. For higher concentrations, each worker should wear respiratory-protection equipment for the self-contained or supplied-air types.

The mechanical-filter respirator is usually a rubber face mask equipped with fibrous particulate filters. The mask, which covers only the nose, mouth, and part of the chin, offers no protection to the eyes or the remainder of the face. Each filter consists of fine fibers loosely packed into a canister. These filters are effective for removing dusts, fumes, and mists from the air and have efficiencies of 99.99% for particles with diameters as small as 0.05 μ.[36] In emergencies, items such as men's cotton handkerchiefs, toilet paper, and other fibrous materials can be used for filtering particulate activity. These items, when folded in layers, have filtration efficiencies which approach 90%.[37]

Air-purifying respirators are usually a combination of mechanical and chemical filters housed in a small canister. In addition to removing particulate matter from the air, these devices can clean air of certain types and remove some gases from the air. Two air-purifying

respirators commonly used for internal exposure control are the "assault" mask and the commercially available universal type of gas mask. The assault mask (military type) consists of one or two canisters attached to a partial or full face mask. Asbestos-fiber filter paper for mechanical filtering and impregnated charcoal for absorbing gases or vapors are contained in each canister. Charcoal filters are not effective, however, for all radioactive gases and should not be depended upon for full protection.

The universal type of gas mask utilizes a separate canister which is strapped to the individual's chest and connected to the full face mask by a length of flexible rubber hose. The canister is somewhat larger than those used for combat masks and is more effective in removing acid gases and organic vapors.

The self-contained and supplied-air types of respiratory-protection devices should be used where the airborne radioactive material is in concentrations greater than 100 times the maximum permissible concentration or where radioactive gases in concentrations greater than the permissible level exist. They should be used also where the adequacy of the first two classes of respirators is unknown or questionable or where there is an oxygen deficiency in the surrounding air.

Self-contained breathing apparatus may be classified into three general types: self-generating, demand, and oxygen cylinders. The self-generating type is a closed system that features a canister containing potassium tetroxide. By breathing into the canister, the wearer is supplied with oxygen for a period of approximately 45 min, which is the life of the canister. Carbon dioxide is removed from the exhaled air by another chemical reaction. The demand type of self-contained breathing apparatus

consists of a compressed-gas cylinder filled with air or oxygen and is connected to a full face mask by a length of corrugated rubber hose. A regulator valve on the tank acts to reduce the pressure in the cylinder to that required for breathing. The oxygen-cylinder rebreathing apparatus is similar to the demand type, except that the cylinder contains only oxygen and there is recirculation in the system. Oxygen present in the exhaled breath is allowed to be recirculated after it has been diverted through a chemical that removes the carbon dioxide.

The hose mask and air-line respirator are two types of the supplied-air respirator group. The hose mask utilizes a long flexible hose which has a relatively large bore through which clean air from another area is forced or drawn into the hose. A hose length of 150 ft should not be exceeded, since air resistance tends to increase with hose length.[38] The air-line respirator utilizes a hose of smaller diameter than the hose mask, but the supply air is under much greater pressure. When a pumping mechanism is used with supplied-air respirators, it should be of a diaphragm type so that carbon monoxide and other intoxicants cannot contaminate the air supply. In areas where there is a possibility of the air supply becoming radioactively contaminated, care should be taken to ensure that the air is prefiltered.

If any respiratory-protection device is to be effective, it must fit tightly to the face. A cleanly shaved face makes possible a closer fit and consequently more absolute protection. The manner in which a device is utilized is often more important than the type of device utilized.

Specialized protective clothing for personnel who are exposed to multiple hazards would also be of value. A welder, for example, when

working in a contamination zone, is exposed not only to a radiation hazard but to thermal and safety hazards as well. At present, the welder must wear the normal protective clothing used by all persons in a contaminated area. In addition, a welder should wear leather gloves, a jacket, a hood, and, if hot metal is being expelled, a leather apron. It would be extremely advantageous from a protective standpoint and would also increase the efficiency of the welder if specialized clothing for protection against these multiple hazards were available for use.[39] It should be of interest to protective clothing manufacturers to explore this area of specialized protection and develop a wardrobe of clothing for multiple purposes.

If the clothing and equipment described in this paper are used properly, personnel can be effectively protected from the hazards created by loose radioactive materials. It should be emphasized, however, that protective clothing is never a substitute for sound operating practices and good facility design.

MOVEMENT OF RADIONUCLIDES THROUGH THE GROUND

D. G. Jacobs

Unless complete containment is provided, any method for treatment or disposal of radioactive wastes finally involves discharge of some of the long-lived radiocontaminants into the environment. The liquid effluent thus discharged may constitute the bulk of the activity or the decontaminated portion of the original waste stream. Evaluation of the consequences requires a knowledge of the further reactions that will occur between the hazardous radionuclides and the earthen materials they will contact. These earthen materials may be deep disposal formations, shallow ground disposal formations, river and ocean sediments, or agricultural soils contaminated by radioactive fallout or by irrigation with contaminated waters. A wide spectrum of conditions exists under which the radionuclides may react with the minerals in the ground, and the variability of these conditions makes precise predictions of radionuclide migration quite difficult.[40] In addition to the natural variation of conditions, the chemical and physical conditions of the system may be greatly altered by the disposal operations. Upon termination of the effluent discharge, the system will tend to revert to its natural state of behavior.

A qualitative description of the rate and direction of ground water movement can often be obtained from permeability data and the hydraulic gradients. When this information is combined with laboratory investigations of the specific radionuclide interactions with the formation materials, a first approximation of the rate and direction of radionuclide migration is obtained. Because of the anisotropy of most geologic formations, many of the factors affecting the rate of water movement and the

sorptive properties of the medium vary considerably over a relatively short distance. In addition, many of the laboratory studies of the interactions of radionuclides with formation materials have been conducted under static conditions using finely ground materials. Such studies are an invaluable step in interpreting radionuclide behavior, but direct extrapolation of the quantitative information to migration of the radionuclides under field conditions should be made with caution.

Fortunately, the interest in radionuclide movement in the ground has led a number of groups to conduct studies that should lead to more reliable means for making quantitative predictions in the field. The four general approaches for comparing the migration of radionuclides in the ground with the movement and dispersion of the transporting water involve its correlation with (1) a qualitative description of water movement, (2) a detailed classical hydrodynamic description of the receiving formation, (3) direct measurement of hydraulic dispersion using an appropriate water tracer to obtain the solution velocity distribution, and (4) a quantitative description of the various factors affecting water movement and dispersion. The primary concern is to describe the variation of activity in the solution phase with time and distance. A separate, but related, aspect of the problem concerns the buildup of activity on the variable solid phase with time and distance.

Qualitative Description of Water Movement

Bovard and Grauby made laboratory and *in situ* studies of the downward movement of radionuclides in undisturbed soils.[41] The retention of Sr^{90}-Y^{90} was determined by analysis of the recovered solution and by autoradiography of the exposed block of soil after percolation of a fixed

volume of solution. As a result, these measurements provided direct field data on the distribution of radiostrontium with distance in the soil but yielded little information on changes in the solution and solid phases with time. The results of the experiments presented a rather qualitative description of the nature of the soil and the hydrodynamics of the system. When the soil was originally saturated with water, the volume of the eluate was nearly the same as the volume injected; however, when the moisture content of the soil was less than the field retention capacity, a considerable portion of the liquid was retained by the soil mass. This method appears to be suitable for obtaining a quick field estimate of the soil retention capacity for soluble radiostrontium deposited by fallout or by an emergency release of liquid, but it would be less desirable for predicting the movement of radionuclides resulting from continued discharge of radioactive liquids to the ground. Refinement of the method could be accomplished by including measurements of the exchange capacity of the soil, chemical analyses of the local percolating waters, and evaluations of other parameters that have a direct bearing on the reactions of specific radionuclides with the formation. A quantitative description of dispersion for short-term experiments involving only the surface layer of soil does not seem to be feasible.

Detailed Classical Hydrodynamic Description of the Formation

Parsons has made a detailed correlation of the continued movement of radionuclides through the soil with a classical description of the geology and hydrology of the disposal formation.[42-44] The radionuclide movement resulted from four separate liquid discharges which were

sufficiently separated in distance that the path of radionuclide movement from one discharge did not overlap with those of the other discharges. The predicted paths of radionuclide movement were obtained from water-table contour data. Transverse cross sections for soil sampling showed that the paths of movement could be rather accurately delineated and that the general paths of movement coincided with those predicted from water-table data. "Undisturbed" soil samples of unconsolidated sands were subjected to laboratory testing to determine porosity, permeability, mechanical analysis, and mean and maximum pore velocity. The laboratory pore velocity measurements, determined by the minimum and average transit times of electrolyte through an "undisturbed" sample, were compared with *in situ* maximum pore velocity determinations using tritium as a tracer. The field determinations showed a greater maximum velocity, by a factor of 1.72 and 1.87 for two tests, than the laboratory tests. The difference was considered to be due to the fact that vertical movement was measured in the laboratory, whereas horizontal movement was measured in the field. This is a possible explanation, but it seems that small amounts of compaction of the laboratory sample or differences in the sensitivity of detecting the first breakthrough of tritium compared with the first breakthrough of electrolyte could also be responsible. A correction factor of 1.8 was applied to all points in the formation. Although this assumption seems to be dangerous, being based on only two checkpoints, the predicted rate of movement of the radionuclides agrees rather well with the actual rate of movement. Use of the complete range of solution pore velocities, instead of just the mean and maximum velocities, might have permitted even greater quantitative correlation of the movement and dispersion of the radionuclides with the movement and dispersion of the water.

Parsons also refers[43] to the effects of the electrolyte concentration of the transporting solution on the interactions of the radionuclides with the formation, but there is a lack of quantitative data on the exchange properties of the formation and the changing electrolyte concentration of the percolating waters. Continued discharge to these formations would require consideration of the increased hydraulic gradients and the effects of the increased electrolyte concentration that would result from discharge of large volumes of liquid. It is also quite likely that continued discharge of acid effluents would result in significant lowering of the pH of the percolating water, although the buffering action of the formation and the diluting power of the formation water showed this effect to be inconsequential for the major path of movement for the single discharges. In fact, the "anomalous" high (31.5 to 43 ml/g) range of strontium distribution coefficient (K_d) obtained from the discharge of unneutralized acid waste (2.5 M HNO_3) compared with the strontium K_d (16.8 to 25.1 ml/g) for the neutralized stream (1.5 M HNO_3, 3.0 M NH_4NO_3 neutralized by addition of lime and limestone to the disposal hole) is probably due to the buffering action of the formation. (The distribution coefficient is defined as the amount of solute sorbed per gram of sorbent divided by the amount of solute remaining per milliliter of solution.) Calculations based on electrolyte content, assuming complete buffering, indicate that the results are consistent.

Other examples of correlation of radionuclide movement with a detailed description of the geology and hydrology of the disposal formation have been presented by Jones[45] and Schmalz.[46] Jones has described a wide spectrum of borehole geophysical methods that were used at the

National Reactor Testing Station in conjunction with lithologic and hydrologic investigations to provide detailed information on the zones of water movement in relatively deep formations. Schmalz has shown that the movement of the radionuclides in these zones follows the same paths of movement as the water and the stable sodium of the injected waste stream. Attempts to measure the dispersion of water in the field, using fluorescein dye, were not successful, and quantitative rates of radionuclide movement and dispersion compared with that of the transporting water were not presented.

Measurements of Solution Dispersion

According to Ionue and Kaufmann[47] the need for separate evaluation of each of the factors contributing to solution dispersion (for example, lateral dispersion, longitudinal dispersion, formation heterogeneities, geometry, etc.) can be reduced by considering only the net effect on the distribution of travel times of solution between two points in the system under conditions of steady flow. They have presented results of an experiment designed to give a direct quantitative correlation of the radionuclide travel time distribution with that of the transporting water. Comparisons were made between radiostrontium and tritium breakthrough curves for laboratory columns and, in the field, for a two-well system and an inverted five-spot system (one center injection well and four corner relief wells). The tritium breakthrough curve reflects all the dispersive factors influencing the movement of the water. Strontium breakthrough curves were predicted by applying a constant time transformation to the tritium breakthrough curve, which was based on the assumptions

that the sorption isotherm for strontium is linear for the range of concentration considered, that the strontium sorption properties of the formation are constant or can be averaged in space and time, and that the exchange is sufficiently close to equilibrium at the pore velocity of the solution that no further dispersion results from the rate of the exchange reaction. Except for extremely slow exchange reactions or for very high flow rates, it is reasonable to assume that equilibrium conditions are met for radiostrontium exchange. In order for the strontium exchange properties to remain constant with space and time, Ionue and Kaufmann point out that it is necessary for the stable-chemical composition of the percolating solution to remain constant. In actual disposal systems this may not be true. The work of Parsons[43] indicates that the radionuclide front may extend beyond the point of equilibrium of the macroconstituents of the formation and the waste stream; this is reflected in a K_d that is variable in time and space and is more pronounced when the injected solution and the original formation solution are greatly different. Since the rate of movement of a radionuclide compared with the movement of water is lower with increasing K_d,[47] the movement of radionuclides at the leading edge of the sorption front will be slower than at the point of injection, and the sorption front should be sharpened when the injection solution has a higher electrolyte concentration than the original formation solution. When the waste stream has a significantly lower electrolyte concentration than the disposal formation water, the radionuclides at the leading edge of the sorption front should move more rapidly than those at the point of injection, thus leading to a greater dispersion than predicted from equilibrium studies.

For systems where ion exchange is the major interaction between the radionuclide and the formation, the use of a constant time transformation of the solution dispersion curve is probably the most satisfactory means for predicting radionuclide arrival times that is now available. For precipitation or filtration reactions, such as the calcite-phosphate reaction[48] or the removal of ruthenium in the ground,[49,50] there is no apparent saturation capacity of the solid phase for the radionuclide, and the nonconstancy of the distribution coefficient with time and space does not permit direct time transformation of solution dispersion curves to give a reliable prediction of radionuclide breakthrough.

The necessity for K_d to remain constant with both time and space and the necessity for maintaining conditions of "steady flow" make it difficult to apply this method to formations having a high degree of heterogeneity, especially when changing flow conditions are likely to be encountered. Its greatest application would be in an isolated formation or a situation where the flow conditions remained constant. The problems encountered in expressing quantitative effects of formation heterogeneities have been discussed by Theis,[51] who points out the impossibility of extrapolating the results of laboratory dispersion studies directly to predict the field conditions.

The method employed by Ionue and Kaufmann does not provide information on the distribution of radionuclides in the soil phase. Baetsle and Dejonghe[52] have considered this adjunct problem and have presented a method for describing the distribution of radionuclides resulting from the downward percolation of contaminated solution through an anisotropic soil. They assume that the extent of the radionuclide front is quite short and

use a step function to introduce differences of sorptive properties in the various layers of soil.

Quantitative Description of the Various Factors Affecting Water Movement and Dispersion

A group at Hanford[53-55] is presently engaged in studies for determining the quantitative effects of: (1) the vertical component of ground-water movement, (2) the quantitative dispersion of ground-water contaminants, (3) the effect of aquifer anisotropism, and (4) the distribution of ground water velocity about the mean velocity. The studies involve field investigations, as well as laboratory and theoretical approaches. Although the description of the ground-water system would be quite complete, the method would have the same limitations as the direct measurement of velocity distribution in regard to the time transformation for prediction of radionuclide movement and dispersion. Differences of sorptive properties with time and space can be incorporated into the description, however, and, if the disposal site has been adequately described, changes in operating conditions could be reflected in the predictions of radionuclide behavior. The major disadvantage of the method is that detailed quantitative information on the site is required. This limits the application of the method to actual disposal area rather than providing a useful means for preoperational site analysis.

Summary

The approach of Bovard and Grauby[41] seems to be adequate only for gross comparison of the radionuclide retention properties of surface soils. The use of a water tracer to obtain the distribution of ground-

water velocities, combined with a time transformation for radionuclide sorption, seems to be ideally suited for quantitative site evaluation, provided heterogeneities and changes in sorption properties are not too severe and the flow conditions remain reasonably constant. In any case, an extensive evaluation of a disposal site requires knowledge of the geology, hydrology, and chemistry of the disposal formations and their boundaries. Obviously, the more adequately the system can be described in terms of quantitative effects on the movement and dispersion of water and the sorption of the specific radionuclides, the more reliable are predictions of changes that will occur upon altering the disposal conditions.

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VI. CURRENT EVENTS

ACCIDENTS IN SHIPPING RADIOACTIVE MATERIAL

L. B. Shappert

Shipments of radioactive materials are being made more and more frequently, and the number and size of the shipments will undoubtedly grow in the future. Thus the necessity for reasonable safeguards against accidents is obvious. Shipping containers are now being designed to conform to proposed government regulations, which are still in the process of being written and revised to preclude grave consequences in the event of serious accidents. Much effort is going into impact testing of casks and cask models to provide a sound engineering basis for the proposed regulations.[1]

Data have been obtained and analyzed on the type and frequency of accidents involving the shipment of both nonradioactive and radioactive materials. The data are quite valuable in pointing up possible methods of reducing accidents, and they provide information that will aid in the safer design of radioactive material containers and their "tiedowns." It must be noted, however, that the shipping experience to date is statistically small and that megacurie shipments, which may become common in the future, have been infrequent. Undoubtedly the advisability of such shipments will be affected by accident experience, both past and future.

There are now three summary reports that discuss the accident experience of the AEC and its contractors. The first[2] covers the period from 1949 to 1956 and discusses the circumstances surrounding 15 accidents which occurred during that time. Accident experience for the period 1957

to 1961 is presented in a report by Patterson and DeFatta.[3] The most recent summary[4] is for the 6-month period from October 1, 1961 through March 31, 1962. It includes a discussion of licensee shipments and a breakdown and analysis of all the AEC shipments of radioactive material. This analysis indicates that during the 6-month period the AEC and its contractors made a total of 23,346 nonexempt shipments of radioactive materials and that private licensees made a total of 85,687 shipments. The AEC shipping data are given in Table VI-1 and are broken down by package type and curie ranges above the maximum 2000-curie limit for most materials in the type B package, as set forth in the Draft Technical Standards reported by Patterson.[4] The total number of shipments listed in Table VI-1 exceeds the 23,346 shipments noted previously because it also includes the small percentage (4.3%) of shipments that were exempt from packaging requirements. Similar information on the shipments by licensees are given below according to curie limits:

Activity	Number of Shipments
<1 mc	48,942
1–10 mc	21,790
10 mc–2 c	13,726
2–20 c	415
20–200 c	548
200–2000 c	89
2000–5000 c	85
5000–50,000 c	7
>50,000 c	5
Total	85,607

From the foregoing data, it appears that the AEC's current annual shipping rate is approximately 47,000 shipments per year, and about four times that many shipments are made by licensees; however, the AEC ships

Table VI-1. NUMBER OF SHIPMENTS OF RADIOACTIVE MATERIALS BY TYPE OF PACKAGE REQUIRED FROM OCTOBER 1961 THROUGH MARCH 1962

	Shipments Requiring Type A Package[a]	Shipments Requiring Type B Package[b]	Shipments Exceeding Curie Limits for Type B Package			Total Shipments (Exempt Plus Nonexempt)
			Up to 5000 Curies	5 to 50,000 Curies	More than 50,000 curies	
Number of shipments	11,878	10,459	1700	67	113	24,217
Percentage of total shipments	49.0	43.2	7.0	0.3	0.5	100

[a]Type "A" package is adequate to prevent loss or dispersal of its radioactive contents and retain its shielding under normal conditions of transport.

[b]Type "B" package is adequate to prevent loss or dispersal of its radioactive contents and retain its shielding efficiency under normal and severe impact conditions.

a greater number of large curie sources. A breakdown of the mode of transport used by the AEC, its contractors, and it licensees is presented in Table VI-2. It is evident from Table VI-2 that the AEC and its contractors send a much lower percentage of their shipments by air (24%) than do the licensees (58%). This reflects the fact that the licensees make many shipments of short half-lived isotopes that require the speed of air transportation.

During the 5-year period from 1957 to 1961, a total of 47 accidents occurred in shipments by the AEC or its contractors.[3] Six were on-site accidents and were eliminated for the purpose of establishing an accident frequency rate. Averaging the remaining 41 accidents over the 5-year period gave a rate of 8.2 accidents per year. Based on 47,000 shipments per year as a basis, even though the reporting periods were different, the accident frequency rate is 0.17 accidents per thousand shipments. If it is assumed that the distance of the shipments averages between 100 and 1000 miles, the accident frequency may be converted to a range of 1.7 to 0.17 accidents per 10^6 miles of travel. This estimate will be low because the annual number of shipments is increasing and the 47,000 shipments per year basis covered a 6-month period from October 1961 to March 1962. This figure may be compared with the average accident frequency rate of 3.64 per 10^6 miles reported by Leimkuhler, Karson, and Thompson[5] for shipments in trucks that might be used to transport radioactive materials. Even if the assumption is made that the calculated accident frequency range (1.7 to 0.17 per 10^6 miles) is a factor of 2 low because of its method of calculation, the calculated frequency rate still could be a factor of 10 lower than that presently experienced by the trucking industry.

Table VI-2. AEC AND LICENSEE SHIPMENTS BY MODE OF TRANSPORT

Mode of Transport	Percentage of Total Shipments	
	AEC	Licensees
Truck	34.3	20.0
Rail freight	15.0	3.0
Railway express	24.0	14.0
Air	24.0	58.0
Regular mail	0.2	2.0
Other	1.5	3.0

In the report by Patterson and DeFatta,[3] the accidents involving only the AEC and its contractors are classified in accordance with their consequences, following the method suggested by Morgan, Knapp, and Thompson.[6] The classes are:

Class I. The vehicle has been involved in an accident or package damage is suspected. The shipment is delayed or stopped. No radioactive material is released, and there is actually no loss of integrity of the package.

Class II. The package integrity is breached; however, there is no release of radioactive materials.

Class III. Radioactive material is released but is confined to the package or the vehicle.

Class IV. Radioactive material is released to the ground or trafficway, with no runoff or aerial dispersal.

Class V. Radioactive material is released, and there is aerial dispersal.

Class VI. Radioactive material is released and enters a watercourse, either directly or after spilling to the ground or trafficway.

The 47 accidents reported[3] for the 5-year period (1957–1961) were broken down according to the class as well as where or in what location the accident happened, the accident experience by month and by year, the primary cause of the accident, and the type of material involved. This information is presented in Tables VI-3 through VI-8, which pertain only to shipments by the AEC and its contractors. For comparative purposes the accidents reported in ref. 2 were broken down in a similar manner by the reviewer as much as it was possible to do so from the descriptions. There was a definite lack of information as to the type of material carried.

Table VI-3. ACCIDENT EXPERIENCE BY CLASS

Class	Number of Accidents	
	1949–1956	1957–1961
I	6	29
II	2	7
III	0	7
IV	3	3
V	1	1
VI	1	0
Nonradioactive material	2	0
Total	15	47

The following observations are based on the information presented in Tables VI-3 through VI-8. There was only one accident within the 12-year reporting period that resulted in release of radioactive material to a watercourse. Only two accidents resulted in aerial dispersal of

Table VI-4. ACCIDENT EXPERIENCE BY MODE OR LOCATION

Mode or Location	Number of Accidents	
	1949–1956	1957–1961
Truck accidents	10	21
Railway accidents	2	11
Terminal accidents	1	15
Airlines accidents	2	0
Total	15	47

Table VI-5. ACCIDENT EXPERIENCE BY YEAR

Year	Number of Accidents	Year	Number of Accidents
1949	1	1957	6
1950	1	1958	5
1951	2	1959	15
1952	2	1960	12
1953	0	1961	9
1954	2		
1955	2	Total	47
1956	4		
Total	15		

Table VI-6. ACCIDENT EXPERIENCE BY TYPE OF MATERIAL INVOLVED

Type of Material	Number of Accidents	
	1949–1956	1957–1961
Source material	6	14
Radioisotopes	0	10
Irradiated fuel elements	0	8
Special nuclear material	2	4
Radioactive wastes	0	3
Empty containers	0	3
Contaminated machinery	0	3
Nonradioactive or unknown material	7	2
Total	15	47

Table VI-7. ACCIDENT EXPERIENCE BY TYPE, IRRESPECTIVE OF MODE

Type	Number of Accidents	
	1949-1956	1957-1961
Handling (in terminals)	0	15
Impact (collision)	6	12
Fire	3	5
Vehicle or equipment failure	4	5
Leakage of containers	0	4
Derailment (rail)	2	3
Tiedown failure	0	2
Aerial dispersal	0	1
Total	15	47

Table VI-8. ACCIDENT EXPERIENCE BY MONTH

Month	Number of Accidents		Month	Number of Accidents	
	1949-1956	1957-1961		1949-1956	1957-1961
January	0	4	July	2	9
February	2	2	August	1	2
March	1	2	September	2	4
April	1	5	October	2	2
May	1	3	November	1	6
June	0	5	December	2	3
			Total	15	47

radioactive material. Fifty-five of the sixty-two total accidents did not release any radioactive material from the transporting vehicle.

The largest percentage of the AEC's shipments are made by truck (34.3%) and, as might be expected, the greatest majority of accidents noted in Table VI-4 are truck accidents (47%). It is interesting, however, that even though the AEC transports 24% of its radioactive shipments by air, air accidents accounted for only 3% of the accidents in the 12-year reporting period.

The total number of annual accidents reported continued to rise with the increasing number of shipments until 1960, when this trend reversed. It is difficult to say whether this is significant, although it may reflect an increasing emphasis on safety in the shipping of radioactive material by the AEC. At approximately the same time a large effort was started to produce uniform safety criteria and regulations covering such shipments. Most of the accidents were caused by impact or collision. This is important, since the number of turnpike miles is increasing annually, and if the radioactive transporting vehicles are allowed to use them, the chance of collision accidents will be reduced.

The accident experience noted in Table VI-8 appears to indicate that, for the accidents reported, the number per month is about constant. Information on the number of shipments per month was not available, so the accident frequency could not be correlated with the shipping frequency on a monthly basis. Leimkuhler, Karson, and Thompson,[5] as mentioned above, studied the accident experience of trucks that might be used to transport radioactive material. This study covered a 4-year reporting period from 1956 through 1959, during which 111,120 accidents were experienced by approximately 2500 large motor carriers in more than 30.5 billion miles of intercity travel. The average quarterly frequency of these accidents per million vehicle miles is noted below:

	Average Accident Frequency Per Million Vehicle Miles in 1956 through 1959
First quarter	4.28
Second quarter	3.28
Third quarter	3.34
Fourth quarter	3.65
Average	3.64

This tabulation indicates significant differences in the quarterly accident frequencies. The quarterly or seasonal variation in accident frequency followed a cyclic pattern throughout the 4-year period, falling in the summer and rising in the winter. It was also found that fire occurred in about 1% of all accidents, but, in overturn accidents, fires occurred with twice that frequency.

The number of shipments of radioactive material is increasing each year, and many of the sources being shipped are large. Soon spent fuel from power reactors will be shipped regularly to processing sites. Such shipments could easily contain millions of curies. More information of the type discussed in this paper is needed on all shipments of radioactive material now being made. Such reporting creates an increased awareness of the hazards on the part of the shipper and provides a basis for the statistical analyses required as a basis for establishing insurance rates. There will always be shipping accidents, but with the effort being put forth by the regulatory bodies, cask designers, and transporters, the safety record has been, and should continue to be, extremely good.

ACTION ON REACTOR PROJECTS BY LICENSING AND REGULATING BODIES

J. R. Buchanan

The Atomic Energy Act requires that the AEC protect the public from undue hazards resulting from exposure to radioactivity. Regulations that 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 that the AEC make certain specific findings regarding the safety of the public before issuing either a construction permit or an operating license for a facility. It also requires authorization for changes in facility equipment or operations that contain an element of hazard not previously reviewed or approved. The license application records of various power reactors are reported in Table VI-9. Recent actions and activities relative to specific reactors and a fuel reprocessing plant are described below. Some information is also included on reactors that are not docketed for legal action.

Big Rock Point Reactor (Docket 50-155)

On March 29, 1963 the Consumers Power Company advised the AEC that the power operation test program of the 157-Mw (thermal) Big Rock Point boiling-water reactor would be completed in early April.[7] Following completion of that program the reactor was shut down so that developmental fuel could be inserted in the core and the initial phase of the research and development program started. The program will work toward the achievement of higher power densities in boiling-water reactor systems.

Table VI- 9 CALENDAR OF LEGAL STEPS IN LICENSING U. S. POWER REACTORS

REACTOR INFORMATION					CONSTRUCTION PERMIT					OPERATING LICENSE				CURRENT STATUS
Code No.	Name and Location Owner and Operator	Reactor Type	Authorized Power Mw(t)	Authorized Power Mw(e)	ACRS Action	Hearing Board Approval	Construction Permit No.	Construction Permit Date	Actual or Scheduled Completion Date	ACRS Action	Hearing Board Approval	Authorization to Operate No.	Authorization to Operate Date	As of June 1, 1963
50-155	BIG ROCK POINT, Charlevaix County, Mich. Consumers Power Co.	Boiling water	157	47.8	Mar. 1960	May 1960	CPPR-9	5-31-60	1962	June 1962	Aug. 1962	DPR-6	8-30-62	Produced first electricity in Dec. 1962
50-205	BODEGA BAY, Bodega Bay, Calif. Pacific Gas & Electric Co.	Boiling water	1008	325	Apr. 1963				Jan. 1966					Construction permit applied for Dec. 1962
115-4	BONUS, Rincon, Puerto Rico USAEC Puerto Rico Water Resources Authority	Boiling water Nuclear superheat	50	16.3	Mar. 1960	June 1960	*	6-28-60	1963					Applied for operating authorization in Mar. 1963
None	BORAX-V, NRTS, Idaho USAEC Argonne National Laboratory	Experimental boiling water Nuclear superheat	35.7	2.7			Not required					Not required		Achieved criticality in Feb. 1962
50-144	CVTR, Parr, S. C. Carolinas-Virginia Nuclear Power Associates	Pressurized D_2O Moderator and coolant	44.3		Dec. 1959	Apr. 1960	CPPR-7	5-4-60	Dec. 1962	July 1962	Oct. 1962		11-7-62	Initially critical in Mar. 1963
	CITY OF LOS ANGELES, Corral Beach, Calif. City of Los Angeles Department of Water and Power	Pressurized water		490 gr 462 net	Oct. 1962 Nov. 1962				Jan. 1968					Preparing to apply for construction permit
50-206	COAST NUCLEAR STATION, Camp Pendleton, Calif. Southern California Edison Co.	Pressurized water	1210	395					June 1966					Construction permit application submitted in Feb. 1963
	CONNECTICUT YANKEE, Haddam Neck, Conn. Connecticut Yankee Atomic Power Co.	Pressurized water	1473	500	Feb. 1963				1967					Preparing to apply for construction permit
50-10	DRESDEN, Grundy County, Ill. Commonwealth Edison Co.	Boiling water	700	192 gr 180 net			CPPR-2	5-4-56	Sept. 1959	May 1959	Sept. 1959 Nov. 1959 Oct. 1960 May 1961 Sept. 1962	DPR-2	6-2-60 10-14-60 6-9-61 11-30-61 9-19-62	Initially critical Oct. 1959
115-1	ELK RIVER, Elk River, Minn. USAEC Elk River Rural Cooperative Power Association	Boiling water Thorium converter	58.2	22	Aug. 1959	Dec. 1959	*	12-18-59	1963	May 1962	Nov. 1962		11-10-62	Achieved criticality in Nov. 1962
None	EBWR, Argonne, Ill. USAEC Argonne National Laboratory	Experimental boiling water	100	4.5			Not required					Not required		Operating since Dec. 1956
None	EBR-II, NRTS, Idaho USAEC Argonne National Laboratory	Experimental breeder Fast-neutron spectrum Sodium cooled	62.5	16.5			Not required					Not required		Startup planned in 1963
None	EGCR, Oak Ridge, Tenn. USAEC Tennessee Valley Authority	Experimental Graphite moderated Helium cooled	84.3	22.3	May 1959		Not required		Dec. 1962			Not required		Under construction
50-16	FERMI, Monroe County, Mich. Power Reactor Development Co. Detroit Edison Co.	Fast-neutron spectrum Plutonium breeder Sodium cooled	200	60			CPPR-4	5-26-59 11-18-60 7-14-61 11-29-61 8-6-62	May 1963	Oct. 1962	Apr. 1963			Hearing Board recommended issuance of operating license in Apr. 1963
	HALLAM, Hallam, Nebr. USAEC Consumers Public Power District	Graphite moderated Sodium cooled	240	76	July 1959 Feb. 1960	June 1960	*	6-6-60	July 1962	Feb. 1962 Feb. 1963	July 1962	DPRA-1	1-2-62 8-9-62	Provisional operating authorization permits operation up to 15% of full power

50-133	HUMBOLDT BAY, Buhne Point, Calif. Pacific Gas & Electric Co.	Boiling water	163	48.5	June 1960 July 1960	Oct. 1960	CPPR-10	11-9-60	Aug. 1962	Apr. 1962	Aug. 1962	DPR-7	8-28-62	Produced first electricity in Apr. 1963
50-3	INDIAN POINT (CETR), Buchanan, N. Y. Consolidated Edison Co. of New York	Pressurized water Thorium converter	585	275 gr 255 net		Final design June 1961	CPPR-1	5-4-56 8-4-61	Apr. 1962	Nov. 1961	Feb. 1962	DPR-5	3-26-62	Reached full-power operation in Jan. 1963
115-5	LA CROSSE, Genoa, Wis. USAEC Dairyland Power Cooperative	Boiling water	165	50	Dec. 1962	Mar. 1963	Not required		1965			Not required		Construction authorization issued in Mar. 1963
None	N. S. SAVANNAH, Constructed at Camden, N. J. USAEC States Marine Lines	Pressurized water	69	22,000 shp			Not required		Dec. 1961	Dec. 1960	June 1961	Not required		Criticality achieved in Dec. 1961; full power in Apr. 1962; inactive due to labor difficulties
50-130	PATHFINDER, Sioux Falls, S. Dak. Northern States Power Co.	Boiling water Nuclear superheat	203	66	Feb. 1960	Apr. 1960	CPPR-8	5-12-60	June 1963					Initial criticality tentatively scheduled for 1963
50-171	PEACH BOTTOM, York County, Pa. Philadelphia Electric Co.	High temperature Unclad fuel Graphite moderated Gas cooled	115	40	Nov. 1961		CPPR-12	2-23-62	Mar. to Dec. 1964					Under construction
115-2	PIQUA OMR, Piqua, Ohio USAEC (Reactor) City of Piqua Municipal Power Commission	Organic cooled and moderated	45.5	11.4	July 1959	Jan. 1960	*	1-7-60	Oct. 1961	May 1961	June 1962	DPRA-2	8-23-62	Operating authorization is provisional; core loading not yet authorized
None	PWR, Shippingport, Pa. USAEC Duquesne Light Co.	Pressurized water	231	68 gr 60 net	Jan. 1954		Not required		Nov. 1957	Nov. 1957		Not required		Operating since Dec. 1957
50-204	RAVENSWOOD, New York City Consolidated Edison Co. of New York	Pressurized water	2030	1000					Oct. 1969					Construction permit application submitted to AEC in Dec. 1962
50-146	SAXTON, Berks County, Pa. Saxton Nuclear Experimental Corporation	Pressurized water	23.5	3.2	Sept. 1959	Jan. 1960	CPPR-6	2-11-60	1962	July 1961	Oct. 1961 Aug. 1962	DPR-4	11-15-61 10-3-62	Achieved full-power operation in Jan. 1963
None	SRE, Santa Susana, Calif. USAEC Atomics International and Southern California Edison Co.	Graphite moderated Sodium cooled	20	6			Not required					Not required		Operating since Apr. 1957
50-18	VBWR, Alameda County, Calif. General Electric Company	Test reactor Boiling water	50	5			CPPR-3	5-14-56				DPR-1	8-31-57	Operating since Sept. 1957; operating restrictions required as of Aug. 1962
50-183	VESR, Alameda County, Calif. Empire States Atomic Development Associates General Electric Company	Light water moderated Separate steam superheater	12.5		May 1961	July 1961	CPPR-11	8-10-61	Apr. 1962 to Feb. 1963	Apr. 1963				Hearing on issuance of provisional operating license planned for 1963
50-29	YANKEE (YAEC), Rowe, Mass. Yankee Atomic Electric Co.	Pressurized water	540	155	Sept. 1957	Financial qualifications May 1959	CPPR-5	11-4-57 6-10-59	July 1960	Feb. 1960 May 1960 June 1960	April 1960 July 1960 Dec. 1960 June 1961	DPR-3	7-9-60 7-29-60 1-12-61 6-23-61	Operating since Aug. 1960; granted license amendment in Oct. 1962 for operation at 540 Mw(t)

*Reactors classified as parallel procedure cases (code numbers 115-1 to 115-5) do not require construction permits, but their construction and operation are authorized by AEC.

Bodega Bay Nuclear Plant (Docket 50-205)

The Pacific Gas and Electric Company on December 28, 1962 filed with AEC a preliminary hazards summary report to support its application for a license to construct and operate a nuclear plant at Bodega Bay.[8] On April 4 they submitted an amendment to the application, at the AEC's request, in order to supply additional information needed in the hazards report.[9]

On April 18, 1963 the ACRS filed with the AEC a report on the reactor.[10] Portions of the report follow:

> "The site of the proposed nuclear power unit is on the Pacific coast, about fifty miles northwest of the City of San Francisco. Areas nearby are used for camping and picnicking, agriculture, and some commercial fishing. An adjacent tract of land is being acquired by the University of California as a field station for marine biology and other scientific studies. The reactor site proposed is about one thousand feet west of the San Andreas fault zone. The surrounding population density is relatively low. The plant will incorporate engineered safeguards.
>
> "The 1008 MW(t) reactor plant proposed is of the forced-circulation, boiling water type. Pressure suppression confinement is to be used, with the reactor located in a dry well that is rated at 62 psi internal pressure, and is connected to a toroidal suppression chamber rated at 35 psi internal pressure. As proposed, the turbine and some associated primary piping as well as some other parts of the primary system, such as feedwater pumps and the primary water cleanup system, are not to be confined. All piping penetrations through the dry well wall are to be provided with isolation valves. The applicant states that all major lines that penetrate the dry well wall and open into the dry well or the nuclear steam supply system are to be provided with two valves, one being remotely operable. Consideration should be given to the inclusion of additional backup isolation valves in some of the primary system piping and to inclusion of a strainer before the main steam line isolation valve to prevent foreign matter from interfering with proper valve action.
>
> "Further consideration should be given to providing the emergency cooling system with pump back-up beyond that provided by the auxiliary feedwater pump.

"The Committee believes that the dry well and the suppression pool should be designed and built to permit leak testing at design pressure after the installation of all penetrations, and that design and construction should permit periodic leak testing at the suppression chamber design pressure. In the initial tests of the dry well, the leak rates should be determined as a function of the internal pressure, so that the results of subsequent tests at lower than dry well design pressure can be extrapolated with confidence.

"In the unlikely event of a reactor accident, the emergency ventilation system is relied on to reduce the consequences. The Committee believes that this system should also be designed and constructed to permit frequent testing of the ability of the filtration and iodine removal units to meet their specified efficiency levels.

"The requirements that are imposed on plant design because of location in an active seismic area have been considered by the applicant, and [they have furnished] recommendations of seismologists who have been consulted on this question. Tentative exploration indicates that the reactor and turbine buildings will not be located on an active fault line. The Committee believes that if this point is established, the design criteria for the plant are adequate from the standpoint of hazards associated with earthquakes. Careful examination of the quartz-diorite rock below should be made during building excavation, to confirm this point. Furthermore, the Committee suggests that, during design, careful attention should be given to the ability of emergency shutdown systems to operate properly during and subsequent to violent earth shocks, and to the stress effects that might be introduced because the reactor building and the turbine building are to be anchored in different geological formations. The need for earthquake-induced shutdown and isolation of the primary system can be considered at a later time. A meteorological tower has been erected on the site and analyses of the observations will be used to establish atmospheric dispersion coefficients, and as one of the bases for determining the height of the off-gas stack.

"The applicant has submitted evidence to establish that the anticipated temperature changes in the ocean near the condenser coolant discharge point will not be great enough to cause appreciable influence on the local marine life. The expected release of radioactive effluent by the condenser coolant water will be at rates well below the 10CFR20 limits. In addition, the applicant has made arrangements to obtain environmental data on marine and land life in the neighborhood of the proposed plant both before and after reactor operation. These measures seem adequate

to insure that the plant can be operated so as not to alter the local ecology or cause undesirable concentration of radioactivity in marine life.

"The ACRS believes that, subject to the above conditions, the proposed reactor can be designed and built at the proposed location on Bodega Head with reasonable assurance that it can be operated without undue hazard to the health and safety of the public."

BONUS Reactor (Docket 115-4)

The Puerto Rico Water Resources Authority and the Nuclear Engineering Corporation on March 13, 1963 submitted to the AEC an application for authorization to start up and operate the 50-Mw (thermal) BONUS boiling-water superheater reactor. The authorization would provide for initial operation of the reactor up to full power under the direction of the General Nuclear Engineering Co. and further operation over a period of 10 years by the Authority.[7]

CVTR (Docket 50-144)

The Carolinas-Virginia Nuclear Power Assoicates, Inc., received a provisional operating license for the 44.3-Mw (thermal) heavy-water-moderated tube reactor on November 7, 1962.[8] Initial criticality was achieved on March 30, 1963.[11]

Dresden Reactor (Docket 50-10)

In March 1963 information was made public on difficulty that was encountered in June 1962 with one of the control rod drives of the 700-Mw (thermal) boiling-water Dresden reactor. It was concluded that the cause was galling of the index tube in the drive. None of the other 75 drives exhibited sluggish behavior. Consequently, the affected drive was left

in the fully inserted position, and plant operation was resumed five days after the initial difficulty was encountered.[12]

The AEC on February 1, 1963 authorized Commonwealth Edison to relax the requirements in the technical specification pertaining to a "cocked rod criterion" as applied during irradiated-fuel transfer.[13] The request for the change was reviewed in the previous issue of Nuclear Safety.[8]

Commonwealth reported that a recent inspection and overhaul of the Dresden turbine generator, which has logged 12,325 hr of operation and generated over 2,000,000,000 kwhr of electricity, showed it to be in excellent condition.[14]

Elk River Reactor (Docket 115-1)

The Allis-Chalmers Manufacturing Company on February 5, 1963 requested authorization from the AEC to make two changes in the technical specifications of the 58-Mw (thermal) Elk River boiling-water reactor.[15] The changes are described below:

> "With respect to maximum permissible rate of reactivity change (specification 8.1.1(5)), the revision indicates that during loading, testing and power operation and such operations as may be conducted in accordance with Sec. 14.4(9), the rate at which reactivity may be increased by movement of control rods shall be limited by administrative control so as not to exceed 0.001/sec in any core with k_{eff} greater than 0.98. Maximum capability of the control system in any core with k_{eff} greater than 0.98 shall not exceed 0.002/sec. ... [T]he control rod calibration work performed as part of the startup program shows that a maximum control system capability of 0.0018 can be achieved under abnormal control rod configurations. These configurations are such that operating practice, as covered in the operating manual, will prevent them. Under normal operating procedures, as covered in the operating manual, the rate of reactivity increase is limited so as not to exceed 0.001/sec.
>
> "As regards the stack monitor detectors, ... [it was noted] that the requirement that the ranges covered by the

immediate particulate and immediate gaseous stack monitors be extended and be covered continuously (without switching) requires use of detectors with shorter resolving time than the GM tubes used in the original system. The shorter resolving time was obtained by using scintillation detectors. Commercially available scintillators used for continuous particulate monitoring beta activity, and those used for continuous gaseous monitoring detect gamma activity."

On March 15 the AEC advised Allis-Chalmers that further information was desired concerning the requested changes, as follows:[12]

"A. Regarding maximum permissible rate of reactivity change:

"1. A Statement as to the reactivity insertion limitation which would be in effect at times other than 'during initial loading, testing and power operation and during such operations as may be conducted in accordance with Section 14.4(9)' or an indication of whether you believe that this phrase should be deleted from the wording of the authorization you are requesting.

"2. The reactivity insertion rates which would be in effect when k-effective is less than 0.98.

"3. An analysis of the results of a reactivity insertion accident caused by control rod withdrawal when the reactor is operating at power. As a part of this analysis, you should:

"a. State the maximum fuel element cladding temperature which would result.

"b. Indicate the pressure which would be present in the fuel elements if the accident occurred near the end of core life and discuss the ability of the fuel element cladding to withstand this pressure.

"B. Regarding the stack monitor detectors:

"You are requested to indicate the relative efficiencies of the stack monitors for detecting beta radiation and gamma radiation in the energy range of interest at the Elk River Reactor. It appears that the scintillation detectors which you are using are, to some degree, capable of detecting both types of radiation. It is, therefore, our opinion that the proposed revision regarding the stack monitor detectors may not be required. We request your comments regarding this matter."

Enrico Fermi Atomic Power Station (Docket 50-16)

In March 1963 the Power Reactor Development Company advised the AEC that initial fuel loading of the Enrico Fermi fast-breeder reactor, which had been scheduled for March 25, had been delayed because of difficulties at the reactor.[16] A portion of a pressure probe that was in use during nonnuclear testing became detached in a control rod drive tube. If the detached piece (a stainless steel tube about 10 in. long and 1/4 in. in diameter together with its connecting nut and washer) became lodged in the drive tube or its orifice section, it would interfere with control rod operation. Therefore, a program was undertaken to insure that no part of the probe remained there. This added several weeks to the time interval needed to complete the nonnuclear testing program.

The company also reported that a vibrational test of a tube bank from its No. 2 steam generator opened six stress-corrosion cracks. This particular tube bank was repaired after stress cracks were discovered in it in 1961. The vibration tests were done in such a manner as to simulate the vibration expected in the No. 1 generator when sodium is circulated. As a result of the test, the No. 1 tube bundle was removed for inspection. (No results are available.) The steam generators are not, however, needed for the proposed initial nuclear test program that is to be conducted at power levels not in excess of 1 Mw (thermal).

On April 16 the AEC Atomic Safety and Licensing Board recommended issuance of a provisional operating license that would authorize loading to criticality and operation of the Fermi plant at power levels not to exceed 1 Mw (thermal).[17] Intervening AFL-CIO unions on April 22 petitioned for a stay, pending filing by the intervenors of exceptions to

the decision.[18] The Commission on April 29 denied the request on the grounds that the motion for a stay was improperly addressed to the board, since the board's jurisdiction had terminated on issuance of the decision.[19] Additionally, the motion did not state with "particularity" the grounds on which it was made.

Humboldt Bay Reactor (Docket 50-133)

The Pacific Gas and Electric Company's Humboldt Bay boiling-water reactor generated its first electricity on April 18, 1963. The first chain reaction was sustained on February 16.[20]

The AEC on March 29 authorized the company to change the technical specifications of the reactor with respect to (1) the set points of the pipe tunnel temperature sensors, (2) the location of a suppression chamber leak test reference system, (3) the water supply for the suppression pool cooling system, and (4) four minor connections in the specifications.[7] On April 3 the company was authorized to make additional changes[9] related to removal of peripheral control rods, repositioning of 16 fuel assemblies in the core, relocation of the core operating sources, and the specified sensitivity of compensated ion chambers.

Indian Point Reactor (Docket 50-3)

The Consolidated Edison Company of New York in a November 15, 1962 report to the AEC advised them of some control rod drive difficulties at the 585-Mw (thermal) Indian Point pressurized-water reactor.[21] The company noted:

> "Difficulties in operating the drive mechanisms have arisen primarily from the tendency of the pressure tight housings which contain the drive mechanisms, to retain

moisture which is carried down along the seal shafts past the buffer seal housings. Examination has revealed that this moisture collects in droplets in various areas within the drive housings causing corrosion. The resulting malfunctions have been (1) faulty disengagement switch performance, (2) faulty insert limit switch performance, (3) faulty angle actuator assembly performance.

"Other problems have arisen from (1) the inadequate design of the angle actuator assembly which has caused intermittent indication of disengagement because of slight bowing of the angle iron member and resultant spurious operation of the disengagement switch, and (2) blower motor bearing failures due to loss of lubricant."

The Consolidated Edison Company also noted that corrective changes were planned.

La Crosse Reactor (Docket 115-5)

On March 29, 1963 Allis-Chalmers Manufacturing Company received a construction authorization for the 165-Mw (thermal) La Crosse boiling-water reactor.[7] An AEC Atomic Safety and Licensing Board had ordered the authorization in a March 18 decision.[22] The board concluded:

"1. The applicant has submitted sufficient information to provide reasonable assurance that a facility of the general type proposed in the application can be constructed and operated at the proposed location without undue risk to the health and safety of the public.

"2. There is reasonable assurance that technical information omitted from and required to complete the application will be supplied.

"3. The applicant is technically qualified to design and construct the proposed facility.

"4. The issuance of a construction authorization for the facility will not be inimical to the common defense and security or to the health and safety of the public."

Nuclear Fuel Services Reprocessing Plant

Nuclear Fuel Services, Inc., proposes to build a plant to process reactor fuels in Cattaraugus County, New York, about 30 miles southeast

of Buffalo. On October 11, 1962 the ACRS reported on the facility as follows:[23]

> "The proposed plant has a nominal capacity of 1000 kg. per day of nuclear fuels of various types. It will employ a number of chemical processes, most of which have already been used on a substantial scale at other locations. The Committee has not reviewed these processes or the equipment design in detail.
>
> "The applicant estimates that for typical operating condition none of the general public will be exposed to an integrated radiation dose in excess of the limits of 10 CFR Part 20. The most severe accident postulated would not cause exposures in excess of the guides suggested in 10 CFR Part 100. While these limits would not be exceeded, even if releases several times those estimated by the applicant were to occur, such increases in load is not visualized.
>
> "It is the opinion of the Advisory Committee on Reactor Safeguards that the site selected may be considered as suitable for a fuel reprocessing plant of the type and capacity proposed with reasonable assurance that it may be operated without undue hazard to the health and safety of the public."

The ACRS on December 26 issued an additional report on the facility as follows:[24]

> "The process to be used is a batch dissolution of fuel elements which are usually chopped into small pieces. The plant will be designed to handle a nominal throughput of 1000 kilograms of uranium per day. A variety of types of fuel with varying exposure time histories are to be processessed, the upper limit of which is approximately represented by the following parameters:
>
> | | |
> |---|---|
> | Burnup | 20,000 mwd/ton |
> | Specific power | 27.5 mw/ton |
> | Irradiation time | 2 years |
> | Load factor | 85 percent |
> | Cooling time | 150 days |
>
> "Similar chemical processing operations have been conducted on a production basis at various Commission-owned plants for several years. A prototype fuel element chopping operation has been carried on at Oak Ridge for about three years. This experience furnishes an adequate basis for plant design.

"The Committee believes that this plant can be designed and constructed with reasonable assurance that it may be operated without undue hazard to the health and safety of the public."

An AEC Atomic Safety and Licensing Board on April 20, 1963 recommended that a provisional construction authorization be issued for the facility.[25]

Nuclear Merchant Ship SAVANNAH

On April 27 the ACRS issued a report on the N.S. SAVANNAH after reviewing its status two weeks prior to that date.[26] Some of the ACRS comments were as follows:

"Numerous significant changes to the ship, its propulsion system, and its methods of operation are in progress or completed. The Committee considers that most of these changes improve operability or safety of the ship. Since there have been no reported malfunctions of the present control rod drives, the Committee believes that the continued temporary use of the present drives and the use of inert gas in the containment is acceptable pending installation of the new drives.

"The Committee believes that the question of pressure relief and scram settings for operation at the proposed 80 MW(t) power level can be resolved by the AEC staff and the Joint Group, so as to insure that transient pressures do not exceed design values.

"Operation of the N.S. SAVANNAH in crowded metropolitan areas can only be reconciled with current views on reactor siting by the incorporation of both assured engineered safeguards and assured ability to remove the ship rapidly in event of an accident. The Committee has given considerable attention to assuring itself that the engineered safeguards and mobility of the N.S. SAVANNAH are established and demonstrable beyond any reasonable doubt at all times when the ship is in such areas.

"The two principal engineered safeguards that provide protection against release of radioactivity in the unlikely event of an accident are (1) the reactor containment vessel and (2) the reactor compartment with its air cleaning system.... The containment vessel is believed to be structurally

adequate, although its leak rate has increased since initial testing.... Since leakage may increase [with] the plant operation, the Committee is of the opinion that 'as-is' leakage tests must be made at the proposed regular intervals....

"The reactor compartment is to be maintained by below-atmospheric pressure so that any leakage will always be into the compartment....

"The air cleaning system is relied upon to remove substantially all particulate and halogen radioactivity from the air exhausted from the reactor compartment.... The effectiveness of the air cleaning system depends on the integrity of its structure and on its installation and maintenance.

"The Committee believes that as a factor of conservatism, the assumed performance of the air cleaning system should be lower than the test values due to (1) uncertainties in the parameters controlling the air cleaning process and (2) the possibility of faults in the installation or of deterioration during operation, particularly due to vibration. Furthermore, the failure of a penetration seal or the opening of a moderate size leak in the containment in case of an accident might produce sufficient pressure to rupture the filters of the cleaning system with substantial loss of filter protection.

"The Committee has suggested installation of on-board testing equipment for frequent air cleaning efficiency determinations. It believes that such measurements should be made prior to each port entry....

"In addition to engineered safeguards, the N.S. SAVANNAH depends upon its mobility to provide adequate protection in populated areas. This mobility in event of an accident may be achieved either by an adequate auxiliary power system or by prompt availability of tugs....

"If, due to any of a variety of reasons such as fog, pier blockage, or wrecks, the mobility of the N.S. SAVANNAH cannot be assured, the Committee believes that the reactor should be shut down and depressurized when at dock, unless the site meets the guide lines of 10CFR Part 100 as modified by permissible credit for the engineered safeguards and by the recent reactor operational history....

"In summary, the Committee has reviewed the operating history of the N.S. SAVANNAH up to the present period of overhaul at Galveston. It has reviewed the significant changes being made during this overhaul, and considers that these generally represent improvement in operability

> and safety of the ship. The Committee believes that, subject to the points specified in the above paragraphs, the N.S. SAVANNAH can continue to be operated and visit ports under the interim criteria of August 1, 1962,[27] without undue hazard to the health and safety of the public."

A public hearing to consider proposed technical specifications for future operations of the N.S. SAVANNAH was held at AEC headquarters in Germantown on May 17.[26] Pending any changes in the specifications, operations will continue under the present operating authorization, which is consistent with the recommendations of ACRS.

The U.S. Department of Commerce on May 17 notified States Marine Lines that its contract for operation of the N.S. SAVANNAH was being canceled. The move followed a long series of labor disputes. The latest of these was with the ship's engineers, who refused to resolve their wage dispute through arbitration. The ship will be idled for four to six months while new engineers are trained. The Commerce Department has arranged to place the ship in the custody of the Babcock and Wilcox Company while the government decides how best to operate it in the future.[28]

Peach Bottom Reactor (Docket 50-171)

On February 20, 1963 the Philadelphia Electric Company submitted to the AEC the design of gas and liquid waste disposal systems for the 117-Mw (thermal) gas-cooled Peach Bottom reactor, which is under construction.[29] Included with the submittal was a second semiannual report on the reactor and the duties, qualifications, and training programs for the operating personnel.

Piqua Nuclear Power Facility (Docket 115-2)

On February 21, 1963 the city of Piqua, Ohio, requested that the operating license for the 45.5-Mw (thermal) organic-cooled and -moderated Piqua reactor be a five-year authorization and that it coincide with the terms of the contract under which the city will assume responsibility for the operation of the facility on August 1 of this year.[30] A provisional operating authorization for the reactor was initially issued on August 23, 1962.[31]

Saxton Reactor (Docket 50-146)

On April 5, 1963 the AEC authorized a change in the requirements concerning the charge pumps for the pressurized-water Saxton Reactor.[13,21] The specifications originally required reactor shutdown in order to perform any minor maintenance on the pumps that charge borated or purified water to the main coolant system. This made it difficult, if not impossible, to obtain the extended operating runs of four to six weeks that are required for the experimental program. The change will allow one of the pumps to be rendered inoperative for up to 4 hr for maintenance, such as replacing packing, while the reactor is in operation.

During the review period Saxton requested permission to make some small changes in the reactor specifications.[6,13,21] These changes had to do with administrative safeguards, additional shielding of the purification surge tanks, and a change in the set point for scram on loss of the main coolant pump.

Vallecitos Boiling-Water Reactor (Docket 50-18)

The AEC has criteria that require that a reactor remain subcritical with any one of its control rods fully withdrawn. The General Electric Company would like to delay the effective date for these criteria to apply to the 50-Mw (thermal) Vallecitos Boiling-Water Reactor (VBWR). Temporary restrictions were placed on the reactor in September 1962, since physical removal of a single control rod could cause the reactor to become supercritical when cold and xenon free.[31] The AEC felt that "good cause" existed for delaying the effective date of the criteria from March 10 to October 10, 1963 because of delays in the programs at the reactor.[32] The General Electric Company was reminded, however, that its license (DPR-1) expired on May 14, 1963 and that they should apply for a renewal pursuant to Section 50.30 of 10 CRF 50 if the VBWR was to be operated after the expiration date. The company was also reminded of Section 2.109 of 10 CFR 2 regarding the filing of timely applications for renewal. On April 5 GE applied for a 10-year renewal of its license.[9]

Vallecitos Experimental Superheat Reactor (Docket 50-183)

On March 5, 1963 the General Electric Company submitted an amendment to the operating license application for the Empire State Atomic Development Associates, Vallecitos Experimental Superheat Reactor that is intended primarily for the development of superheat reactor fuel elements. Included were some design modifications and a revised plan for fuel tests in the reactor that reflect AEC's request for more rigid and restrictive experimental limits as an alternate to the plant set forth in the proposed technical specifications.

On April 18 the ACRS filed a report with the AEC concerning the reactor.[10] The Committee concluded that there is reasonable assurance that the reactor "can be operated at power levels up to 12.5 Mw(t) with steam from [its] gas-fired boiler without undue hazard to the health and safety of the public."

Yankee Reactor (Docket 50-29)

The Yankee Atomic Electric Company on March 19, 1963 sought authorization from the AEC to install a vent line in the vapor space of the pressurizer for the 540-Mw (thermal) pressurized-water Yankee reactor.[33] Recent operations during boric acid tests (to allow higher core enrichments) revealed that oxygen was temporarily present in the main coolant system during pressurizer level changes. The vent was felt to be needed to provide a desirable degree of control over the accumulating of noncondensables in the pressurizer.

CHANGES IN LICENSING REGULATIONS

J. R. Buchanan

Amendments are frequently proposed to the AEC rules and regulations published in the Code of Federal Regulations, Title 10, Chapter 1. Among those pertinent to nuclear safety which are being considered for change are Part 2, Rules of Practice; Part 9, Public Records; Part 50, Licensing of Production and Utilization Facilities; Part 115, Procedures for Review of Certain Nuclear Reactors Exempted from Licensing Requirements; and Part 140, Financial Protection Requirements and Indemnity Agreements. The various changes are discussed below.

Part 2, Rules of Practice

A proposed amendment to Part 2 was published in the Federal Register of April 3, 1963.[34] The proposed amendment provides a procedure by which an initial decision for issuance or amendment of a construction permit or construction authorization can be made effective 10 days after issuance provided no substantial question of fact, law, or discretion has been presented. The purpose of the change is to expedite the effective dates of permits and authorizations or amendments to them without the necessity of recourse to more complex procedures. Comments were accepted for a period of 30 days following publication in the Federal Register.

Part 9, Public Records

On February 20, 1963, a proposed amendment to Part 9 was published in the Federal Register.[35] The proposed amendment permits the AEC General Manager or Director of Regulations to disclose, on request,

to Federal and non-Federal government agencies information not included in the public records.

The amendment was endorsed by the states of California and Indiana.[36,37] It is felt that the rule change will enhance the cooperative effort between the states and the AEC to achieve the best possible program for the health and safety of the public.

Part 50, Licensing of Production and Utilization Facilities

A proposed amendment to Part 50 was published in the Federal Register of April 3, 1963.[34] The proposed amendment modifies existing provisions of 50.57(e) that presently permit immediate effectiveness of initial decisions granting provisional operating licenses and authorizations by providing, instead, that they may become effective 10 days after issuance. Thirty days were allowed for comments on the proposed change.

Part 115, Procedures for Review of Certain Nuclear Reactors Exempted From Licensing Requirements

A proposed amendment to Part 115 was published in the Federal Register of April 3, 1963.[34] In line with the change proposed in Part 50 above, the proposed change in Part 115 modifies existing provisions of 115.45(e) that presently permit immediate effectiveness of initial decisions granting provisional operating licenses and authorizations by providing, instead, that they may become effective 10 days after issuance. A period of 30 days was allowed for comment on the proposed change.

Part 140, Financial Protection Requirements and Indemnity Agreements

On April 20, 1963, a proposed change in Appendix A of Part 140 was published in the Federal Register.[34] The change covers the form of

liability insurance policies issued by Nuclear Energy Liability Insurance Association (NELIA) and Mutual Atomic Energy Liability Underwriters (MELIA) and includes an industry credit-rating plan. A period of 30 days was allowed for comment on the change.

SAFEGUARDS REPORTS AND SELECTED READING

J. R. Buchanan

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.

Hazards and Safeguards Reports

1. J. R. Coombe, L. D. Stephenson, and F. G. Moote, Hazards Report - SM-1A Core II, USAEC Report ACNP-62832, Allis-Chalmers Manufacturing Company, Oct. 18, 1962.
2. Babcock and Wilcox Company, Consolidated Nuclear Steam Generator for Merchant Ship Application, a Conceptual Design, USAEC Report BAW-1243, August 1962.
3. L. M. Arnet et al., Final Hazards Evaluation of the Heavy Water Components Test Reactor (HWCTR), USAEC Report DP-600, E. I. du Pont de Nemours and Company, Inc., December 1962.
4. G. H. Hanson and F. K. Clements (Eds.), Experimental Organic Cooled Reactor Safety Analysis Report, Volumes I and II, USAEC Report IDO-16820, Phillips Petroleum Company, Nov. 30, 1962.
5. Fluor Corporation, Ltd., Experimental Organic Cooled Reactor Preliminary Hazards Report, and Add. I and II, USAEC Report IDO-24034, April 1, 1960.
6. W. A. Poley and P. M. Finnegan (Eds.), Mock-Up Reactor, NASA Plum Brook Facility, Final Hazards Summary Report, USAEC Report NP-12379, Lewis Research Center, September 1962.

7. Consolidated Edison Company of New York, Inc., Ravenswood Nuclear Generating Unit A. Preliminary Hazards Summary Report, Volumes I, II, and III, 1962, USAEC Report NP-12466.

8. Pacific Gas and Electric Company, Bodega Bay Atomic Park, Unit Number 1, Exhibit C. Preliminary Hazards Summary Report, December 28, 1962, USAEC Report NP-12476.

9. Georgia Tech Institute of Technology and General Nuclear Engineering Corporation, Final Safeguards Report for the Georgia Tech Research Reactor, USAEC Report NP-12512, January 1963.

10. Southern California Edison Company, Preliminary Hazards Summary Report on Southern California Edison Company Nuclear Station at Camp Pendleton, California, Unit No. 1, Part B, USAEC Report NP-12513, January 1963.

11. United Aircraft Corporation, Hazards Summary Report on Critical Experiment Program No. 1 at Canel, USAEC Report PWAC-170, May 24, 1957.

12. Oak Ridge National Laboratory, A 750-Mw(e) Gas-Cooled Clad-Fuel Reactor Power Plant, A Joint Design Study, USAEC Report ORNL-3353, February 4, 1963.

Operations Reports

1. Westinghouse Electric Corporation, Carolinas Virginia Nuclear Power Associates, Inc., Research and Development Program Quarterly Progress Report, July-September 1962, USAEC Report CVNA-165.

2. K. W. Reid, The Cobalt Spill at NRX, July 1958, Canadian Report AECL-1665 (also IOI-158), reprinted January 1963.

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Accidents in Shipping Radioactive Material

1. K. W. Haff and L. B. Shappert, Standards and Tests for Radioactive Shipping Casks, Nucl. Safety, 4(4): _____ (June 1963).
2. A Summary of Transportation Incidents in Atomic Energy Activities, 1949-1956, USAEC Report AECU-3613, December 1957.
3. D. E. Patterson and V. P. DeFatta, A Summary of Incidents Involving USAEC Shipments of Radioactive Material, 1957-1961, USAEC Report TID-16764, November 1962.
4. D. E. Patterson, Types and Quantities of Materials Being Shipped and AEC Accident Experience, p. 3 of Summary Report AEC Symposium on Packaging and Regulatory Standards for Shipping Radioactive Material held in Germantown, Maryland, December 3-5, 1962, USAEC Report TID-7651.
5. F. F. Leimkuhler, M. J. Karson, and J. T. Thompson, Statistical Analysis of the Frequency and Severity of Accidents to Potential Highway Carriers of Highly Radioactive Materials, p. 20, USAEC Report NYO-9771, July 1961.
6. J. M. Morgan, J. W..Knapp, and J. T. Thompson, A Study of the Possible Consequences and Costs of Accidents in the Transportation of High-Level Radioactive Materials, USAEC Report NYO-9772, August 1961.

Action on Reactor Projects by Licensing and Regulating Bodies

7. Facility Licensing, Atomic Energy Clearing House, 9(15): 16-20 (April 15, 1963).

8. J. R. Buchanan, Action on Reactor Projects by Licensing and Regulating Bodies, Nucl. Safety, 4(4): _____ (June 1963).

9. Facility Licensing, Atomic Energy Clearing House, 9(16): 30-34 (January 22, 1963).

10. Reports by the Advisory Committee on Reactor Safeguards, Atomic Energy Clearing House, 9(18): 18-20 (May 6, 1963).

11. The Carolinas-Virginia Reactor Went Critical, Nucleonics Week, 4(14): 2 (April 4, 1963).

12. Facility Licensing, Atomic Energy Clearing House, 9(13): 22-25 (April 1, 1963).

13. Facility Licensing, Atomic Energy Clearing House, 9(8): 9-14 (February 25, 1963).

14. Company Notes, Personnel, Etc., Atomic Energy Clearing House, 9(8): 41 (February 25, 1963).

15. Facility Licensing, Atomic Energy Clearing House, 9(9): 15-19 (March 4, 1963).

16. In a Supplementary Statement, Atomic Energy Clearing House, 9(10): 22-24 (March 11, 1963).

17. Issuance of a Provisional Operating License, Atomic Energy Clearing House, 9(16): 6-12 (April 22, 1963).

18. The Initial Decision of an AEC Licensing Board, Atomic Energy Clearing House, 9(17): 8-9 (April 29, 1963).

19. Denial of Request by Intervenors, Atomic Energy Clearing House, 9(18): 20 (May 6, 1963).

20. Organization Notes, Personnel, Etc., Atomic Energy Clearing House, 9(18): 31-32 (May 6, 1963).

21. Facility Licensing, Atomic Energy Clearing House, 9(17): 24-27 (April 29, 1963).

22. Authorization for Construction, Atomic Energy Clearing House, 9(12): 20-22 (March 25, 1963).

23. The Approval by the ACRS, Atomic Energy Clearing House, 8(43): 5-7 (October 22, 1962).

24. Belief That Nuclear Fuel Services, Inc., Atomic Energy Clearing House, 9(2): 2 (January 14, 1963).

25. Issuance of a Provisional Construction Permit, Atomic Energy Clearing House, 9(17): 30-33 (April 29, 1963).

26. A Public Hearing to Consider, Atomic Energy Clearing House, 9(18): 9-14 (May 6, 1963).

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28. George Horne, Contract Ended on the Savannah, New York Times, May 18, 1963.

29. Facility Licensing, Atomic Energy Clearing House, 9(10): 18-20 (March 11, 1963).

30. Facility Licensing, Atomic Energy Clearing House, 9(11): 12-16 (March 18, 1963).

31. J. R. Buchanan, Action on Reactor Projects by Licensing and Regulating Bodies, Nucl. Safety, 4(3): 90-97 (March 1963).

32. Facility Licensing, Atomic Energy Clearing House, 9(12): 29-30 (March 25, 1963).

33. Facility Licensing, Atomic Energy Clearing House, 9(14): 14-17 (April 8, 1963).

Changes in Licensing Regulations

34. AEC Regulations, Atomic Energy Clearing House, 9(16): 37 (April 22, 1963).

35. Various Amendments to AEC Regulations, Atomic Energy Clearing House, 9(9): 24-25 (March 4, 1963).

36. Comments on Proposed AEC Regulations, Atomic Energy Clearing House, 9(12): 25-26 (March 25, 1963).

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