A Compendium of Information for Use in CONTROLLING RADIATION EMERGENCIES

**Including Lecture Notes
from a Training Session
at Idaho Falls, Idaho,
February 12-14, 1958**

United States Atomic Energy Commission
Office of Technical Information

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Printed in USA. Price \$1.00. Available from the Office of
Technical Services, Department of Commerce, Washington
25, D. C.

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**Including Lecture Notes from a Training Session at
Idaho Falls, Idaho, February 12-14, 1958**

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G. VICTOR BEARD**
Compilers and Editors

Issuance date: September 1960

**Office of Health and Safety
U. S. ATOMIC ENERGY COMMISSION**

Notice: The material in this report reflects only the opinions of the authors and editors and does not necessarily represent the policies of the Atomic Energy Commission.

INTRODUCTION

On February 12, 13, and 14, 1958, a training course was held at Idaho Falls, Idaho, to familiarize members of radiological assistance teams from various parts of the United States with the origin and nature of situations that might, by the event of an unusual accident, release radioactive materials to a populated environment. Some 80 persons attended this course, representing AEC Operations Offices, AEC-contractor installations, Armed Forces radiological units, and Federal and State health agencies. The course consisted of a series of lectures and a tour of some of the radiation monitoring, source handling, and transportation facilities at the National Reactor Testing Station.

A summary of the lecture material presented in the course is reproduced here as a reference for members of radiological assistance teams. Several of the topics discussed have been given only in abstract form, but an attempt has been made to include most of the essential information. In addition, Appendixes A and B include some data and references that may be useful in organizing or conducting radiological assistance. No effort has been made to make the coverage of this report complete, but only to present some information in a concise form, to be used only by trained personnel in exercising rapid and professional judgment during the immediate period following an unexpected incident. Many more comprehensive references and much data have been omitted that are well known and otherwise available to the members of the radiological assistance teams.

ACKNOWLEDGMENTS

Much appreciation is due Mr. John R. Horan, Director, Health and Safety Division, Idaho Operations Office, USAEC, and his staff for providing facilities and helping with the arrangements for the course. We are also indebted to the speakers listed on the agenda for providing very interesting presentations.

Thanks are also due Dr. James Turner for his aid in reviewing the manuscript, and Mrs. Mary Howes and Mrs. Louise Johnson for their patience and skill in typing.

G. VICTOR BEARD

U. S. Atomic Energy Commission
Washington, D. C.
October 1959

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AEC EXPERIENCE IN RADIATION ACCIDENTS

By D. F. HAYES

U. S. Atomic Energy Commission, Washington, D. C.

Data on accidents, particularly those of a catastrophic nature, are still limited. We hope this will continue to be so. But, as might be expected, there are a few accidents each year, and undoubtedly there will be more as the number of plants and other activities using fissionable materials and radioisotopes increases. To make this talk valuable to you, it is probably best for me to talk about the kinds of accidents that have occurred that would require your services on an emergency basis. To do this it is probably best to start by reviewing those accidents which occurred under circumstances approximating those of the Windscale accident, and then to look at some others.

The significant accidents from 1944 through 1958 are collected in report TID-5360, *A Summary of Accidents and Incidents Involving Atomic Radiation in Atomic Energy Activities, June 1945-December 1955*, Supplement to TID-5360 (1956), and TID-5360, Supplement 2 (1957-58). Report AECU-3613 contains a compilation of accidents involving materials in transit.⁸⁶ This document does not deal exclusively with radiation accidents but covers other types as well and was prepared because we have other needs for transportation accident data. It includes all those reported to Washington through 1956.

These summaries are as complete as we can make them, based upon reports of investigations that come into Washington from the Operations Offices. They describe the incidents that fall within the reporting requirements of the Commission. Because it is difficult to define exactly what is reportable, or more important, what is significant, these compilations of experience are not complete in any sense. For instance, we have a requirement that a fire does not need to be investigated and reported if the monetary loss is below \$5,000. With this minimum, we miss a great many *significant* accidents simply because the monetary losses did not exceed the required amount.

A very illustrative accident occurred recently at one of the production reactors. The amount of the loss was estimated at \$4,600, and, therefore, was not necessarily reportable. However, the fire in this case was extremely important since it occurred at the face of one of our plutonium-producing reactors. A crew of workmen using a cutting torch allowed sparks to fall into a supply of paraffin (used for neutron shielding). The fire department responded to this fire. While the fire was being suppressed, another fire broke out on a platform on a higher level at the same face in the flammable insulation connected with the shield and into which the same crew had precipitated sparks from the same cutting equipment. The resulting confusion was understandable, since some people were reporting and talking about one fire while some others were reporting and talking about another. This circumstance added importance to the incident, but the main fact was that paraffin, a highly flammable material with a high Btu content, was (and is) used so frequently without the most elementary care against burning. The other significant feature of the incident was that any good safety program today will require that standard operating procedures be written for welding and cutting, and that permits be required from an authority before this kind of work can be done. The requirements for permits bring about the assignment of a fireman or a responsible person to stand by in case anything

happens, as occurred in this incident. The point here is that the definition of reportable incidents can never be sufficiently comprehensive to bring in all of the experiences.

Only recently, on the occasion of the fire at another plant involving plutonium, did we become fully aware that—to quote the operators—“These fires happen all the time.” Although we have been able to gather some data on this experience, this statement shows that there have been many fires involving plutonium that remained unreported because the monetary loss was not considerable, those involved did not recognize or realize the importance, or the experimenters felt that the data were not an important part of the experiment.

No two accidents are *exactly* alike in every detail. However, when I look at our own experiences, I find that we have had several cases where the circumstances were surprisingly alike. For instance, the fire in the plutonium line in Rocky Flats in October 1957 was caused by the spontaneous ignition of plutonium; the heat that ignited the Plexiglas hoods and burned out the line could very well have been of the same order of magnitude as the Windscale accident. There were a couple of very fortunate differences. The material that burned at Windscale involved spent uranium fuel rods. One of the fission products was iodine, which, in the heat of the fire, passed through the Windscale filters as a gas. Iodine sublimates directly from a solid to a gas. Iodine, escaping through Windscale filters, was the only radioactive material that gave the British a serious problem. At the Rocky Flats incident, fresh plutonium burned and its particulate oxides were contained by air filters.

Incidentally, the similarity between these two incidents suggests a comment. It should be noted that in both the Windscale incident and the Rocky Flats incident, the basic cause was a fire. The accident at the National Research Experiment (NRX) in 1950 originated in a nuclear runaway, but there also, fear of a fire created a good deal of the trouble and damage. I am sure you are well aware that the great worry has been that there would be a nuclear incident and an explosion with the undesirable spread of contamination, but fires have actually been more frequent. It is my observation and firm conviction that the peculiarities of an incident that might involve you and might require your services will more likely come from a fire. Moreover, if the circumstances are just right, the fire itself may be of an inconsequential nature. For example, a recent fire in a small experiment at the Materials Testing Reactor at Idaho Falls resulted in a fire loss amounting to only \$400. It cost the contractor \$23,000 to clean up the mess.

Again, to go on with similarities and relatively small dissimilarities, consider the two railroad accidents described in TID-5360.⁸⁷ One occurred between St. Maries and St. Joe, Idaho, on March 24, 1956; the other was at Western Montana on April 20, 1956. These two accidents happened within less than a month of each other, occurred in the same part of the country less than 75 miles apart, and involved freight cars in transport. One case involved drums of uranium compound; the other involved bars of uranium metal (slugs). Each incident occurred on the banks of an important river. In the first case, the train was derailed because of a rock and snow slide; the second was derailed as a result of a broken rail. In the first case, boxes of uranium were spewed along the right of way; but in the second case the cars carrying radioactive material were precipitated into the river at a time when the water level was increasing at the rate of about one foot an hour. Both incidents occurred on busy main lines. Although these accidents occurred in a remote place and involved only uranium compounds of low toxicity, it is not difficult to imagine this problem in terms of liquid high-level wastes. Quick action would be necessary to determine the extent of damage; the probability that it might extend further; the precautions to be taken to protect the public; and—perhaps most exacting of all—the manner in which the public is informed of the real extent of the damage or danger.

At Chalk River, Canada, on December 13, 1950, there was an explosion involving a relatively small amount of material, killing one man and injuring and contaminating five others. Only through good luck, 45,000 pounds of the same material in a room adjacent to the room in which the explosion occurred did not detonate. This accident is not widely known because there was much stricter security at the time the incident occurred. It is possible now to relate more about this accident and a few significant lessons that were learned from it. In fact, it may interest some of you to know that this accident helped initiate action on the part of the

National Committee on Radiation Protection to produce Handbook 56, *Safe Handling of Cadavers Containing Radioactive Isotopes*.*

Many lessons were learned from this accident, and it should be of interest to quote the following from AEC 43/345,⁸⁸ dated February 9, 1951:

The relative urgency of different kinds of medical treatment became immediately complicated by radiation, since, in cases where shock would normally have been given priority, the doctors hesitated to make hypodermic injections through contaminated skin. The decontamination of wounds was complicated by the lack of a probe sufficiently directional to tell from what part of the wound the radiation was coming. The volume of contaminated materials became a sizable problem. As a result of the explosion, four truckloads of materials required decontamination, while two more truckloads had to be scrapped. For two days after the accident, most of the medical facilities were tied up as a result of contamination. After preliminary treatment at the plant hospital at Chalk River, a new kind of decision had to be made concerning the extent to which the patient should be decontaminated before transfer to the townsite hospital could be allowed.

Identification of the personnel was difficult because badges had been blown off or destroyed. One of the casualties had about 200 mr per hour over the face and chest, and this further inhibited work. Unprecedented problems arose in connection with transferring the contaminated body to an undertaker.

The medical group at Chalk River had been planning for some time to rebuild the medical set-up at the plant, and, until this accident, their thinking had been almost entirely concerned with facilitating routine decontamination. The accident exposed shortcomings in the present set-up, and the experience of handling the contaminated and wounded patients dictated revision of their plans for rebuilding.

What lessons might one draw from the actual experience that we have had? Certainly, each accident has its own lessons, and all together the conclusions leading to actions that one might take are numerous. Again, probably some of the most difficult problems raised by accidents in atomic energy operations are those involving the public reaction. I think all of you should have as a part of your library, "Lessons Learned from A Pyrophoricity Incident," by Merrill Eisenbud.⁸⁹ One of the most important lessons of this incident is that frankness with the public, and particularly with the press, is absolutely necessary.

It seems that the interest of reporters is in inverse proportion to the ease with which we make information available to them, or equivalently, directly proportional to the difficulty they have in getting the information. When an accident occurs, you cannot avoid newspaper accounts. It is best to give a straightforward, honest, and complete story and to trust the responsible papers to treat the story as you gave it to them. There is nothing you can do about irresponsible newspapers or "headline seekers." But you do have the satisfaction, if you have told as much as you can, as truthfully as you can, that you can deny the lies with justifiable righteousness.

In this connection, I might remind you that the Commission has recently stated that the amounts of contamination that have resulted from accidents by AEC contractors or by AEC licensees is not classified information.

It may not be enough to give the straight story to the press agents, who really may not understand some of the technical details that are being given to them. Only a few of the larger newspapers can afford to have technical writers on their staffs. It would be well, therefore, to recognize that the AEC and its contractors have an obligation to educate the public in the simple facts to the greatest extent possible. The material in Mr. Eisenbud's article is the basis of a program which has been initiated by the Safety and Fire Protection Branch, Atomic Energy Commission, Washington, D. C. The objective of this training program is the reduction of simple concepts of radiation to the understanding of the layman. The events with the Sylvania explosion show conclusively that the first people to arrive at the scene of an accident, i.e., police and firemen, can or cannot do great harm, depending on how well they understand radiation phenomena. In the Sylvania case,⁸⁹ they understood very little about it, with the result that the firemen transmitted their fears and hesitation to the surrounding community. Furthermore, there was a lack of pre-planning for a major plant accident. Because of lack of

*Superseded by Handbook 65.¹³²

training or indoctrination, the plant personnel added to the public relations problem that came about. Another observation that I would like to put forward is that at the time of an emergency, interest is intense. The desire to learn immediately what happened is at a high pitch, and the atmosphere is ripe for the dissemination of rumors. Therefore proper communication, or the lack of it, can affect, and has affected, the outcome of many of these accidents. The lack of communication in the Sylvania case shows how the people concerned carried the wrong stories home and spread them throughout the community. Mr. Eisenbud pointed out the need for a quick way to inform the people.

An obvious conclusion is that under circumstances involving contamination of a building, a neighborhood, a river, or anything else, you will be looked to as the expert, and your attitude must be above reproach. It will be necessary for you to act calmly, without exaggerating or minimizing what you find when you make your surveys.

RADIOACTIVE SHIPMENTS FROM THE MTR

By E. H. DAVIDSON

Idaho Operations Office, U. S. Atomic Energy Commission

I am surprised that the meeting has proceeded this far without some remark having been made on the name of the meeting place. This establishment is called the "Odd Fellows" Hall. I am not sure who the term was meant to refer to in this case, but I want to make it clear in the beginning that I am not a member of the Health and Safety Division. I am attached to a group of normal fellows, and girls, called the Operations Division. I point this out also since the subject matter, if not the subject to be discussed, cuts across these two organization lines. Consequently, my remarks about the functions of the Health and Safety Division can be interpreted as objective and unbiased.

The discussion on radioactive shipments from the Materials Testing Reactor (MTR) is divided into two general categories: The first is related to shipments which I will classify as *ordinary*. This category includes shipments which do not contain significant amounts of nuclear fuel in the sample being returned to the sponsor; it also includes radioactive isotopes, such as cobalt and iridium, as well as other samples which may have become radioactive as the result of some change, other than fission, occurring in the sample while in the reactor.

The second category I will classify as *unusual* shipments. These shipments are of samples which are radioactive largely by virtue of the fissions which have occurred in the sample itself while in the reactor. Included, as well, in this unusual category are shipments of spent MTR fuel elements. We will spend the greater portion of time this morning discussing the relatively few, but, I think, fairly unique shipments of this type. We will conclude with a brief discussion of the procedure necessary to get a shipment off-site.

First, the ordinary shipments. The MTR experimental program has grown to include many types of irradiations which, I am sure, were never anticipated by its designers. The primary experimental facilities which were built into the MTR originally, were the six horizontal beam holes and the 62 vertical graphite holes. Yet, last year the MTR averaged 576 separate samples at one time, primarily as a result of the drilling of holes into the beryllium reflector to accommodate additional experiments. The operators of the MTR sometimes describe its core as being experiment-reflected as well as beryllium-reflected. Even one of the shim rods has been removed to accommodate experiments.

Of the samples which are shipped in the ordinary category, the cobalt ones are the most intensely radioactive. A few months ago, a cobalt sample was shipped to the Argonne Cancer Research Hospital. This particular piece of cobalt possessed a specific activity of 222 curies per gram and a total activity of 2640 curies. This represented the highest contained activity of any cask which has left the MTR in the category of ordinary shipments. The lower limit of contained activity in the ordinary shipments approaches zero since some of the irradiated samples do not become significantly radioactive.

In moving into the category of unusual shipments, I might first say that I am calling these unusual because of the curie content. Most of these have become quite routine.

Casks which contain 100,000 curies are often shipped at about three-week intervals. Most of these casks contain a number of samples whose original U^{235} content was in the neighborhood

of 40 to 50 g and which have experienced almost complete burn-up. In April of 1958, a single sample will be shipped whose activity has been estimated at about 820,000 curies. The total number of shipments containing material which had been irradiated in the ordinary and unusual categories during calendar year 1957 exceeded 2500.

In moving into a discussion of spent fuel element shipments from the MTR, it will first be necessary to examine another type of irradiation service which is not so well known but which is widely used at the MTR site. This service is gamma irradiation.

Each fuel element used in the MTR contains a total of about 200 g of U^{235} when initially charged into the reactor. This U^{235} is distributed through 19 adjacent curved plates which form an active assembly about 24 in. long and approximately $3\frac{1}{2}$ in. $\times$ $3\frac{1}{2}$ in. in cross section. About 60 g of U^{235} per fuel element is consumed in operating the reactor and the fission products created within the plates have been used for some time to provide intense gamma sources. There is a need to know the effect of gamma rays on many materials, without introducing these materials to the more confusing environment which exists in the reactor.

Because of the mode of operation of the MTR, the full 60-g burn-up for a single element is not usually attained in one cycle. The elements are periodically discharged and later reinserted into the reactor from the canal. In 1954, a small facility was constructed in the MTR canal to take advantage of the gammas available from the inventory of partially burned fuel elements. The configuration used can provide a gamma source as high as 10^7 r/hr to a multiplicity of samples. This facility is designed to utilize full-length elements, since these elements are destined for reinsertion into the reactor.

About a year later a separate building was provided to utilize fully burned elements prior to chemical reprocessing. These elements have had the aluminum ends removed and, consequently, are only about 25 in. in length. Over 1450 shipments of samples, which had been irradiated in the gamma facility, were made during calendar year 1957. These samples included everything from potatoes to automobile tires, and I am being serious here. These shipments, of course, were not measurably radioactive.

In addition to supplying its own needs for government and commercially-sponsored gamma irradiations, spent MTR fuel elements are supplied to other sites for use as gamma sources. At six-week intervals, the Army Proving Grounds at Dugway, Utah, is routinely shipped a supply of eight MTR fuel elements between the ages of 60 and 80 days cooled, measured from time of last reactor discharge. These eight fuel elements are shipped in a single cask. It is difficult to estimate the curie content of this cask with its eight fuel elements because of the wide range of fission product isotopes represented, which have different decay rates. However, if all disintegrations are considered, a rough estimate of 148,000 curies is obtained for the average shipment of this type. The dose rate from each of these fuel elements is about 300,000 r/hr as measured underwater 3 in. from the fuel element sideplate at mid-plane.

Other spent fuel elements are shipped to the Argonne National Laboratory at six-week intervals. Though only four fuel elements are contained in the Argonne cask, these fuel elements are shipped about 14 days after discharge from the reactor and are estimated to have a disintegration rate in excess of a quarter of a million curies. The dose rate, measured in the same manner, from each of these fuel elements, is about 10^6 r/hr. Both the Dugway and the Argonne shipping containers are designed to provide rapid heat transfer during transit.

In September 1956, the Commission issued a press release making three announcements:

1. That the Commission, through Idaho Operations Office, would rent limited numbers of spent fuel elements to interested commercial users.
2. That the age of these fuel elements would be approximately 90 days cooled.
3. That the Commission would furnish a shipping container for the transportation of these fuel elements.

Somewhat to the temporary embarrassment of the Commission, two things occurred after the press release:

1. The workload at our own gamma facilities increased drastically — primarily as a result of an expansion in food irradiation for the Quartermaster Food and Container Institute.
2. The Argonne National Laboratory announced that the shipping container it had hoped to rent for this civilian program could no longer be released because of increased demands for its use. These two developments left the civilian-spent fuel programs with an availability of

140-day-old elements (about half as strong gamma-wise as the 90-day-old elements originally announced available) but without a means of transportation.

From the civilian point of view, the lack of a shipping container was the more stringent limitation since it appeared that the shortage of 90-day-old elements was a temporary one.

During a period of about six months after the press release many inquiries were received from industrial organizations who were considering the construction of their own gamma facilities utilizing spent MTR fuel elements. The lack of a shipping container was pointed out in our replies and each interested organization was advised to consider the use of a cask that could transport either spent MTR fuel elements or spent fuel elements from the Engineering Test Reactor (ETR), which was then still under construction. The active section of the ETR fuel elements is about 50 per cent longer and slightly smaller in cross section, so that a cask designed to transport MTR fuel elements could not double as a transporter of ETR fuel elements, though it appeared likely that a cask designed to carry ETR fuel elements could be easily modified internally to handle MTR fuel elements. *

Out of the splash of initial interest came the granting of five licenses to civilian users for the possession of four MTR or four ETR fuel elements with possession of up to eight fuel elements allowed when the fuel elements were being replaced. The plans of three of these companies have materialized and all three companies now possess MTR fuel elements in their privately constructed gamma facilities. These companies are the Sinclair Research Laboratory in Harvey, Illinois; the Minnesota Mining and Manufacturing Company near St. Paul, Minnesota; and the Continental Oil Company in Ponca City, Oklahoma. Both Sinclair and Conoco had seriously considered the purchase of a shipping container, but the entry of another company into this program altered their plans.

The Tri-State Warehousing and Distributing Company of Joplin, Missouri, had for a long time specialized in the transportation of high explosives and was interested in expanding into the field of transportation of radioactive materials. Its problems were many. Perhaps the foremost was obtaining insurance, which had never been issued for this type of shipment. Still another was the opposition from other trucking companies who opposed the operation of a Tri-State truck through their own territories. Much of this opposition arose as a result of misunderstanding about the intentions of Tri-State to provide a specialized shipping service in its own container, whose empty weight was about 13 tons, and whose contained "payload" was less than 50 lb. In any event, Tri-State has been duly licensed in the past for four shipments from the MTR, two shipments from ANL, and two return shipments to the Idaho Chemical Processing Plant (ICPP) from the licensees.

The first shipment of 90-day-cooled fuel elements from ICPP to a licensee was made last August, with Tri-State furnishing the cask and transportation service. The contained activity was estimated at about 80,000 curies. At 90 days cooled, the approximate dose rate from each element was 250,000 r/hr as measured underwater with an ion chamber 3 in. from the fuel element sideplate. This shipment was delivered to Sinclair. During the second shipment last November, interesting problems arose both before and after shipment.

The before-shipment problem was the desire of Tri-State to remove the aluminum paint which coated the cask during the first shipment to Sinclair and to grind the stainless steel surface to a smoother finish. This was to be done at a Pocatello machine shop as a further aid to the MTR in decontaminating the cask in future shipments. The owner of the machine shop was concerned about this operation, since he realized that the cask had at one time been contaminated by MTR canal water. Though it had been thoroughly decontaminated prior to the Sinclair shipment, he wanted assurance from the AEC that no hazards were involved in breathing the dust from the smooth grinding operation. An AEC health physicist was dispatched to Pocatello, monitored the grinding operation, and became convinced that no airborne activity was released as a result of this grinding.

The second problem was related to weather in November. The inside of the cask contains water to surround the spent fuel elements. In the case of the shipments to Argonne and to

*EDITORS' NOTE: Additional requirements in shipping irradiated fuel elements are now specified in reference 90.

Dugway, previously mentioned, we had never worried about freezing the water during cold weather shipments because of the amount of heat generated by the shorter-cooled, and more intense, gamma-emitting elements. But in the case of elements cooled as long as 90 days, the problem of protection against freezing was raised by the Health and Safety Division. This was a nasty question.

We knew these 90-day elements generated heat, but were not sure how much. Moreover, we knew neither the heat transfer rate through the cask nor what sort of minimum outside temperature to assume as an aid to calculating heat transfer rates. Rather than try to answer complicated questions of this type, it appeared simpler to replace the water in the cask with an antifreeze solution. We were not unmindful of the peculiar tricks which can be played by gamma rays on all sorts of materials. A literature search was made and indication was received that ordinary permanent-type automobile antifreeze, which has an ethylene glycol base, would be satisfactory, from a radiation damage standpoint, for the short time it would be exposed to gamma rays during transit.

Therefore about half the water in the cask was replaced with glycol prior to shipment, giving a 50-50 mixture of glycol and water expected not to freeze above -30°F . The Minnesota Mining and Manufacturing Company (MMM), which was to receive this shipment, had been alerted to the contents of the cask and had arranged to drain it and flush the antifreeze prior to unloading. This precaution had been taken both to keep the MMM water canal free of this antifreeze solution, and to permit examination of the solution, since no shipment of this type had been made before.

At about 11:30 p.m. on Thanksgiving night, I received a call from the MMM Co. The cask had been delivered, but the MMM canal into which the elements were to be placed had sprung a leak, and it was not felt advisable to unload the cask until the canal leak could be repaired. Arrangements were made with Tri-State to leave the elements in the cask until the canal leak had been fixed. In the meantime, the cask was to be drained of its antifreeze, filled with water and protected from freezing. The next week I received another call pertaining to the draining of the cask. It appeared, upon draining, that a portion of the antifreeze solution had turned to a molasses-like substance.

The reasons for this unexpected radiation damage are not yet well known. Samples of the same mixture have been exposed to 200 million roentgens in the MTR gamma building without showing any evidence of physical change except for discoloration. Apparently, either the dose rate given to the antifreeze between the fuel element plates was much higher than anticipated, or else the temperature of the elements served to accelerate this damage.

Three other shipments have been made since that time using a methyl alcohol and water mixture as the antifreeze with no evidence of radiation damage to the mixture.

In our concluding discussions on shipping procedure, it may be of interest to you to follow an example of the transfer of radioactive material from the MTR. The example chosen here is a truck shipment, since roughly 70 per cent of our shipments are by highway, with about 20 per cent by rail and 10 per cent by air.

After being unloaded from the truck by an overhead crane at the MTR, the empty transfer cask is lowered into the canal in the basement. The lid of the cask is removed and the radioactive material is placed into the container under about 15 ft of water, which serves as a shield. The top of the cask is replaced, underwater, and the cask is removed from the canal by the crane. Since the canal water is contaminated, the outside surface must be cleaned. It is first rinsed by an uncontaminated, demineralized water supply and then scrubbed with a detergent solution. After a second rinse, the cask is allowed to dry. In the meantime, the cask has been surveyed with an instrument to determine that the gamma emission from the cask is below acceptable levels. After drying, the container is wiped or smeared with small pieces of paper. The activity of the paper is counted as a final test to determine that the exterior of the cask has been properly decontaminated.

Once the cask has been approved by the Phillips Petroleum Company Health Physics representatives, it must pass an inspection by the AEC Health and Safety Division. This inspection is a two-pronged one. Representatives of that Division must become assured, first, that the container is properly tied down to the truck. It is apparent that many casks have been designed, which are perfectly adequate from a shielding standpoint, but have not been designed

with a means of adequate tie-down. I am told that of 38 truck shipments last month, 22 were delayed because of inadequate means for securing the container to the truck.

The second check made by Health and Safety Division personnel is another radiation level examination, including additional smear tests.

Thus prior to the approval for any off-site radioactive transit, each outgoing radioactive shipment is checked first by contractor representatives and again by AEC representatives to assure that decontamination is adequate, radiation levels are below acceptable limits, and the method of tie-down is secure.

The adequacy of this procedure may be illustrated by a few examples of traffic accidents that have occurred while hauling radioactive shipments. In 1954, in Pennsylvania, a long-haul government truck was involved in an intersection accident with a passenger car. The truck snapped off a telephone pole and wound up in a private front yard. The radioactive material container was not displaced. In 1954, in Illinois, a long-haul truck was struck head-on by a private vehicle. No displacement of the load occurred. At this site in 1956, a locking connection between the tractor and trailer failed and the trailer was dropped with no displacement of the load. A similar failure occurred in Tennessee in 1955 with no load displacement.

Although the carriers themselves have been involved in accidents, there has been no incident involving the release of radioactivity during transit from this site. I think you will agree that this is a mutual tribute to the Bureau of Explosives for its container requirements, to the Phillips Petroleum Company, and to the Idaho Operations Office Health and Safety Division.

Thank you.

FUEL ELEMENTS AND CORE SHIPMENTS

By DALE GRAY

Westinghouse Naval Reactor Facility

The fuel elements used in the majority of the reactors in the Naval Reactor Program are the flat plate design. The number of fuel plates per element varies between the different sizes of reactors, the largest being the Pressurized Water Reactor (PWR) cluster which measures approximately 8 in. square $\times$ 76 in. in length. In addition to plate type elements, the PWR core utilizes pellet assemblies in the blanket region.

The naval reactor plants may be refueled by either removing the complete cartridge as is done during refueling S1W-S2W plants, or by removing the individual elements as in the more advanced designs. The cores that can be refueled by removing individual fuel elements require more removal equipment and more specialized containers; however, in many instances, it is felt that this method will be faster because it can be accomplished without first removing the reactor vessel head.

The only complete expended core cartridge which has been shipped across country to Idaho was the one removed from the USS *Nautilus* submarine. The *Nautilus* refueling program commenced February 16, 1957. The core was removed and in the shipping container, ready for transfer in 15 days. Because of decay heat, however, the core was held in its shipping container at Electric Boat Company for approximately two weeks to allow the heat to decay to an acceptable level. The core left Groton, Connecticut, by special train and arrived at Central Facilities Area—National Reactor Testing Station (NRTS) four days later. The core cartridge coffin and shipping shield assembly were removed from the special rail car and transported to the Naval Reactor Facility (NRF) site via truck. During the transportation of the *Nautilus* core, there was at no time any serious danger from decay heat and/or radiation. Health Physics personnel and complete disaster gear for five people accompanied the shipment from Groton, Connecticut, to NRTS in Idaho.

At the present time the Naval Reactor Program is sponsoring the design of three standardized shipping containers. The designation of these containers is L-130 (large), M-130 (medium), S-130 (small). The large and small containers are being designed by Westinghouse, the medium container is being designed by General Electric. The largest of the containers consists of three major assemblies: the shipping container, the shipping shield, and the secondary cooling system. The secondary cooling system is capable of removing 450,000 Btu/hr decay heat. The medium container is a hollow right circular cylinder surrounded by a series of vertical finned heat exchangers extending the length of the container. The decay heat removal system is capable of dissipating 45,000 Btu/hr by convection or 140,000 Btu/hr by forced circulation. The smaller container is designed as a shield into which fits a removable pressure vessel supporting the fuel element and containing primary coolant. Decay heat is removed by natural convection at the maximum rate of 45,000 Btu/hr. Design of all the containers is based upon a maximum radiation level of 200 mr/hr surface and 10 mr/hr at one meter. All containers are transported by standard railway equipment.

The different naval reactor plants are identified as follows:

S1W: Prototype plant of the *Nautilus* submarine located at the NRF

S2W: Submarine *Nautilus*

PWR: Pressurized Water Reactor plant located at Shippingport, Pennsylvania

S3W-S4W: Submarine plant smaller than, but similar to the *Nautilus*, such as the *Skate*, *Sargo*, and *Swordfish*

S5W: Submarine with the "Football" shaped *Albacore* design hull

A1W: Prototype plant of the aircraft carrier located at the NRF

S1G: Prototype plant of the submarine *Seawolf* located at West Milton, New York

S2G: Submarine *Seawolf*

S3G: Prototype submarine plant under construction at West Milton, New York. Same core design will be used in the *Triton* submarine presently under construction at Electric Boat Company

D1G: Destroyer class reactor plant presently under design at General Electric

S1C: Killer type submarine prototype presently under construction by Combustion Engineering at Windsor, Connecticut.

The expended core cartridges from all of these plants will be shipped to the ECF for disassembly and inspection.

HIGH-LEVEL WASTE SHIPMENTS

By R. E. HAYDEN
Phillips Petroleum Company

(Abstract)

Past and planned future shipments of highly radioactive waste from the Idaho Chemical Processing Plant to Oak Ridge National Laboratory were described. The curie content, volume, chemical, and isotopic composition were reviewed. The schedule, radiation, contamination, loading and handling problems, as well as, cask design and previous difficulties, were outlined. The following summary will indicate the types of problems that may arise with these waste shipments:

Routine shipments of radioactive waste from Idaho to Tennessee have been accomplished without incident. Maximum shipments have been about 500 gal. and 12,000 curies per transfer cask. The casks are lined with Type 347 stainless steel and have $3\frac{1}{2}$ in. of lead and $1\frac{1}{8}$ in. of steel shielding. They are cylindrical, approximately 8 ft 10 in. high $\times$ 6 ft outside diameter. They weigh about 16 tons each. Future shipments will consist of six casks on two gondolas.

One cask failed before shipment. The radioactive liquid apparently leaked into a very small space between the lead shielding and the outer steel plate. There was no leakage of liquid, but shipment was impossible owing to radiation, averaging about 1200 mr/hr at 3 in. from the shell with a maximum of 4700 mr/hr at 3 in.

Solutions have a specific gravity of about 1.27 and are basic in pH. Casks are shipped at about 7 psi less than atmospheric. Solutions are calculated to form a heavy sludge at -4°F . About 10 per cent of the volume is allowed for expansion. No freezing or pressure build-up problems have been encountered.

Somewhat less than 5000 curies of radioactive gas per hydrogen cylinder have been shipped in improvised shields at a maximum of 200 psi. Cylinders have $\frac{5}{8}$ in. of iron and $\frac{3}{4}$ in. of lead shielding. The combined weight is about 1500 lb per cylinder and shield. Apparent adsorption on foreign material in cylinders produces a non-uniform source of radiation. Radiation problems during bottling and handling are important. The dose rate in a cloud containing one curie per cubic meter would be a few rem/minute, according to calculations.

Radiation survey meters and automatic personnel or area monitors readily detect radioactive gases, even in very dilute form. Available constant air monitors are practically useless as emergency detectors for high-level waste surveillance since they depend on filter collection, have a low gas flow rate, and are shielded from external radiation.

It is not uncommon for radioactive gases intentionally released through hood ducts or vacuum lines to find their way back into buildings through the ventilation system or other unpredictable paths.

CONTAINERS AND SHIPPING REGULATIONS

By L. C. J. WHITE

*Supervising Inspector, Bureau of Explosives
Association of American Railroads*

Gentlemen: I consider it a privilege to be with this group today, and to be able to contribute to the discussion about a very important subject.

This is probably the first time that some of you have come into direct contact with the Bureau of Explosives, and for that reason I will give you a short history of the organization and its purpose.

At the beginning of the century, railroads were suffering huge losses. A great number of personal injuries were inflicted and lives were lost as a result of the transportation of explosives. The explosives at that time consisted primarily of dynamites, black powder, and blasting caps. After several attempts were made by individual railroads to exert some control measures of their own, chaos prevailed as each road prescribed its own conditions under which it would accept shipments. As time went by, it was realized by responsible people that a central agency was necessary if safety and uniformity were to result. Fifty years ago last June, the bureau for the safe transportation of explosives and other dangerous articles opened their office in New York. This bureau was named in a revised Act of Congress as an agency which could be utilized by the Interstate Commerce Commission (ICC) in the execution of its regulations. Then, again to be brief, an orderly procedure was established for making changes in, or additions to, the regulations. I need not mention that certain penalties are provided by the Act of Congress for failure to comply with the requirements set forth by the Interstate Commerce Commission.

The bureau has District Inspectors located at many points throughout the country, with two in Canada. They are on duty and available at all times for emergencies that affect the carriage of dangerous goods by railroads. Presently, we have 23 District Inspectors who are authorized by Interstate Commerce Commission regulations to work with carriers and shippers in the fulfillment of the regulations and to consider packaging and transportation methods that will provide for reasonable safety in the shipment of dangerous articles.

Let us go back to the regulations for a moment. Any recommendations for changes in the regulations are generally submitted by the Bureau to the ICC after substantial agreement among interested parties has been obtained. I am not here to discuss changes in the regulations that may be considered by some to be desirable. Any changes of this nature should be requested by the General Manager of the Atomic Energy Commission or other designated official of that agency and be handled in the manner which has been established by the very inception of the regulations relating to the shipment of radioactive material. I am here only to acquaint you with the ICC regulations and what they require in the preparing of packages for shipment and in the handling of such packages by carriers.

As you probably know, the ICC regulations prescribe methods for packaging many kinds of explosives and dangerous materials. I will talk only about radioactive material. The principles for ICC control of radioactive materials were established by a Committee having representatives from the AEC, National Research Council, Bureau of Mines, several uni-

versities, and the Bureau of Explosives. These principles were translated by the Bureau of Explosives into the language of the Federal code, were presented to the ICC, and were incorporated into their regulations January 21, 1948.

Radioactive materials were divided into three groups under the new regulations designated "Radioactive Material, Class D Poison," and were defined as any material or combination of materials that spontaneously emit ionizing radiation. For the purpose of the regulations, radioactive materials were divided into three groups according to the type of ray emitted at any time during transportation.

Group I materials are those emitting gamma rays or electrically charged corpuscular rays.

Group II materials are those that emit neutrons and either or both types of radiation characteristic of Group I.

Group III materials are those that emit electrically charged corpuscular rays only, such as alpha, beta, or any other that is so shielded that the gamma radiation at the surface of the package does not exceed 10 mr for 24 hr at any time during transportation.

Two types of labels are prescribed for these packages. One is the red label that states, "No person shall remain within three feet of this container unnecessarily." It further states, "Do not place undeveloped film within 15 feet of this container." There are three lines provided on this label that must be completed by the shipper. One is the principal radioactive content; this must not be shown by symbols since railroad personnel or others handling the shipment could not be expected to know the chemical symbols for the various isotopes. The activity of contents must be stated and this may be shown in curies or disintegrations per second, whichever is desired. The number of radiation units from the package should be shown. A unit is defined by the regulations as one milliroentgen per hour at one meter (mrhm) for hard gamma radiation, or the amount of radiation which has the same effect on film as one mrhm of hard gamma rays of radium filtered by $\frac{1}{2}$ in. of lead. It will be noted that the label also states: "Not more than 40 units of radioactive material can be loaded in one car or one motor vehicle or held in one location." The matter of loading more than 40 units in a car or truck is beginning to cause some concern. For example, with a great number of small packages constituting a truck load, the number of units could be exceeded although the hazard in transportation probably would not be increased substantially. Of course, the only way this could be done at the present time would be by a special permit issued by the Interstate Commerce Commission.

The other label is for Group III material, and here you will note that it is necessary merely to state the name of the contents of the package.

Not more than 2000 mc of radium or members of that family, and not more than 2700 mc of any other radioactive substance, may be packed in one outside shipping container, except that up to no more than 300 curies of solid cesium-137, cobalt-60, gold-198, or iridium-192 may be packed in one outside Specification 55 container, unless special arrangements and conditions are approved by the Bureau of Explosives. I will come back later to conditions required by the Bureau of Explosives for approval of containers in transporting more than curie quantities.

The specification for a 55 container is relatively brief and consists essentially of a lead or uranium shielded container with an outside steel jacket at least $\frac{1}{8}$ in. thick for containing shielding of thickness up to 6 in., and $\frac{1}{4}$ in. steel for shielding thicker than 6 in. The depth of shielding would be measured from the inside cavity to the nearest point on the outer steel container. The container must be designed with an offset opening to prevent beams of radiation from escaping; the closure must be secured in place to withstand severe impacts or jolts incident to transportation; and the closure must be equipped with a seal of a type that must be broken for the container to open for any purpose. If the container weighs over 500 lb, it must be mounted on a suitable skid or cradle designed to provide for a greater distribution of weight over the areas of car or truck floors. We learned that this was necessary for heavy objects by our experience in World War II with the transportation of heavy bombs or shells, where in some cases we found that they broke through car floors and were dropped along right of ways. The Specification 55 container should be designed to withstand severe impacts during a fire, without losing its shielding efficiency. This is the primary object of the specification.

The regulations provide certain exemptions for packages containing radioactive material. I think that I probably shall pass over some of these. However, I do think that radioactive materials such as ores and residues of low activity should be discussed. The pertinent paragraph of the regulations states:

Radioactive materials such as ores, residues, etc., of low activity packed in strong tight containers are exempt from specification packaging and labeling requirements for shipment in carload lots by rail freight only, provided the gamma radiation or equivalent will not exceed 10 milliroentgens per hour at a distance of 12 feet from any surface of the car and that the gamma radiation or equivalent will not exceed 10 milliroentgens per hour at a distance of 5 feet from either end surface of the car. There must be no loose radioactive material in the car and the shipment must be braced to prevent leakage or shift of lading under conditions normally incident to transportation. The car must be placarded by the shipper as provided by the regulations with the radioactive Class D Poison placard. Except when handling is supervised by the Atomic Energy Commission, shipments must be loaded by consignor and unloaded by consignee.

Certain exemptions are also permitted for detonating fuzes having radioactive components, but I will not dwell on this subject.

Radioactive materials that present special hazards due to their tendency to remain fixed in the human body for long periods of time (e.g., radium and members of that family) must, in addition to the packaging prescribed, be packed in an inside metal container, Specification 2R, or some other container approved by the Bureau of Explosives. A Specification 2R container must be of stainless steel, brass, or malleable iron, and must be fitted with threaded closures. Flange closures with gaskets are authorized for certain sizes as stated in the specification. The container must not be more than 6 in. in diameter and 16 in. in length, and must have a minimum wall thickness of $\frac{3}{32}$ in. for diameters up to 2 in., and not less than $\frac{1}{8}$ in. for diameters of 2 to 6 in. I am of the opinion that most of the containers in which plutonium or other elements of long effective half-life are shipped are at least as efficient as the Specification 2R container.

All radioactive materials must be packed and shielded so that the degree of fogging of undeveloped film under conditions normally incident to transportation, 24 hr at 15 ft from the package, will not exceed that produced by 11.5 mr of penetrating gamma rays of radium filtered by $\frac{1}{2}$ in. of lead. This requirement was put in the regulations to protect shipments of film.

The design and preparation of the package must be such that there will be no significant radioactive surface contamination of any part of the container. The word "significant" as used here would limit the contamination to not more than 10 mr for 24 hr.

The smallest dimension of any outside shipping container for radioactive materials must be not less than 4 in. This is required to prevent someone from throwing the package or putting it into a person's pocket.

All outside shipping containers must be of such design that the gamma radiation will not exceed 200 mr/hr or equivalent at any point of readily accessible surface. Containers must be equipped with handles and protective devices when necessary in order to satisfy this requirement.

The regulations permit the Bureau of Explosives to approve containers other than those prescribed. I will not describe the containers that are prescribed by the regulations but will outline the details which the Bureau of Explosives must consider when approving containers. Actually, the conditions set forth in the memorandum that has been passed to some of those present follow closely the requirements of a Specification 55 container.* These are:

1. A shield of lead or other material of equal efficiency should be provided, encased in steel so that it will not flow away or lose its efficiency if involved in a fire. One-eighth-inch thick steel is required for 6 in. of shielding or less. One-fourth-inch thick steel is required for more than 6 in. of shielding.

* EDITORS' NOTE: Additional requirements in transporting irradiated fuel elements are set forth in Reference 90. Reference 91 describes some past experience in transporting high-level radioactive materials.

2. The shield is to be supported in the outer container so that it cannot change position or open under any ordinary conditions. Parts of the shield must be designed so that radiation cannot be beamed at points where sections join. Radiation at any surface of the package must not exceed 200 mr/hr and radiation at one meter from the axis of the package must not exceed 10 mr/hr. The container must be designed so that it can be properly braced in a car. Containers weighing more than 500 lb must be fitted with skids or otherwise designed so they will not create excessive pressures on small areas of the car floor. Heavy containers must be provided with hooks, handles, and skids, or any other device necessary to facilitate normal handling. The outer container when practicable must be of metal and must be securely closed by a positive fastening device capable of withstanding severe impacts without failure. A seal must be applied to indicate readily whether or not the container was opened for any purpose. For containers of more than 4000 mc or the equivalent the name "Radioactive Material" and the Bureau of Explosives permit number assigned for the approval must be stamped, embossed, or otherwise permanently applied so that it will not be obliterated by fire. Since we are concerned primarily with the shipment of "over-curie" quantities of radioactive material, and since under section 73.391 of the regulations, this Bureau can approve these shipments, it is necessary that our office be notified on or before the date of shipment of the date of shipment, origin and destination, type of carrier employed, the assigned Bureau of Explosives Permit number marked on outer shipping container, the number of curies or disintegrations per second of material in the container, and the principal type of radiation.

3. Radioactive Materials, Group I (liquid, solid, or gaseous) must be packed in suitable inside containers completely surrounded by a shield of lead or other suitable material of such thickness that at any time during transportation the gamma radiation at one meter from any point on the radioactive source will not exceed 10 mr/hr. The shield must be designed so that it will not open or break under conditions incident to transportation. The minimum shielding must be sufficient to prevent the escape of any primary corpuscular radiation to the exterior of the outside shipping container.

4. Radioactive Materials, Group II (liquid, solid, or gaseous) must be packed in suitable inside containers completely shielded so that at any time during transportation the radiation measured at right angles to any point on the long axis of the shipping container will not exceed the limits hereafter stated. The shielding must be designed to maintain its efficiency under ordinary transportation conditions, and must provide personnel protection against fast or slow neutrons and all other ionizing radiation originating either in the radioactive materials or any part of the aggregate constituting the complete package. The limits that cannot be exceeded are:

- a. Gamma radiation of 10 mrhm.
- b. Electrically charged corpuscular radiation which is the physical equivalent of 10 mrhm of gamma radiation.
- c. Neutron radiation which is the physical equivalent of 2 mrhm of gamma radiation.
- d. If more than one of the types of radiation named are present, the radiation of each type must be reduced by shielding so that the total does not exceed the equivalent of the amounts stated.

5. Liquid Radioactive Materials, Groups I, II, or III, must, in addition to packaging prescribed, be packed in tight glass, earthenware, or other suitable inside containers. The inside container must be surrounded on all sides by an absorbent material sufficient to absorb the entire liquid contents and of such nature that its efficiency will not be impaired by chemical reaction with the contents. Where the use of shielding is necessary to reduce radiation to limits prescribed by this section, the absorbent cushioning material must be placed within the shield. However, if the inside container is packed in a metal container Specification 2R, or other container approved by the Bureau of Explosives, the absorbent cushioning is not required.

6. Radioactive Materials, Group III (liquid, solid, or gaseous) must be packed in suitable inside containers completely wrapped and/or shielded with a material that will prevent the escape of primary corpuscular radiation to the exterior of the shipping container. The secondary radiation at the surface of the container must not exceed 10 milliroentgens per 24 hr at any time during transportation.

For the purpose of measuring radiation from packages, the Bureau of Explosives uses a Landsverk meter; there are other meters of this type that are just as useful for this purpose.

You were previously furnished with a one-page pamphlet detailing Bureau of Explosives instructions which have been distributed to carriers as a guide in the handling of radioactive materials involved in leakages and fires. You will note that this pamphlet is worded simply and does not deal with specific types of radiation. We must consider the people who will be the first ones at a wreck; they probably will be railroad train conductors, brakemen, engineers, and firemen—until other persons arrive. We must present them with something that is simple to read, complete, and specific in its purpose, since these people are not familiar with all of the aspects of radiation. It would have been much simpler for us to have prepared a lengthy article dealing with this material. In fact, it was most difficult to be concise and "to-the-point" in issuing these instructions.

The Bureau of Explosives has, for a considerable time now, conducted semiannual staff meetings during which our Inspectors have been instructed in the handling of radioactive materials that might be involved in wrecks, fires, or derailments. Undoubtedly, any railroad encountering a situation of this nature would advise our District Inspector, and he would proceed as instructed until competent help arrives. Many of our men have been furnished with radiation detectors for surveying a wreck area to determine whether radioactive material is present. After qualified help equipped to handle an emergency of this type arrives, the Inspectors will cooperate with them and follow their advice. Where railroad operations are affected, the Inspectors are under instruction to contact the New York office.

That is all I have to say, gentlemen, and I will be happy to answer any questions to the best of my ability.

FISSION PRODUCT REACTOR POTENTIAL

By D. J. NELSON, JR.
U. S. Public Health Service

(Abstract)

The hazard resulting from the release of fission products accumulated in a reactor depends upon three basic factors: power level, power history, and mechanism of release. Though the mechanism of release is difficult to predict, there are reliable methods for estimating the effect of power level and power history on fission product inventory. Gross fission product beta activity can be estimated by the integrated form of the Way-Wigner relation where the beta activity in curies is given by

$$A_B = 1.4P [t_c^{-0.2} - (t_c + t_0)^{-0.2}] \quad (1)$$

Power (P) is expressed in watts, cooling time (t_c) and operating time (t_0) are expressed in days.

Neglecting neutron capture by fission products, one obtains the build-up of a specific nuclide in a reactor by the equation

$$\text{Curies} = 0.84 PY (1 - e^{-\lambda t_0}) \quad (2)$$

where P is the reactor power in watts; Y is the fission product yield expressed as a fraction (so $\sum Y_i = 2$); λ is the radioactive decay constant in units of $(\text{time})^{-1}$; and t_0 is the operating time in units reciprocal to those of λ .

The release of fission products from a reactor may result in an ingestion and inhalation hazard in addition to the external gamma dose from the released cloud of radioactive material. The passage of the radioactive cloud over grazing or agricultural areas will result in the contamination of milk or food products. Where this route for uptake of the radioactivity exists, it is expected that reduction of the radiation dose to the population can be most effectively carried out by restricting the consumption of the contaminated food product. In other cases the maximum dose delivered to a critical organ will be to the thyroid from the iodines, or to the bones from the bone-seeking nuclides. Calculations have shown that for power histories of greater than sixty days, the bone dose will predominate.

The determination of approximate fission product inventories for reactors can indicate to the emergency monitoring team the most probable hazard and help in their selection of initial dose control procedures.

Table 1 gives the approximate activities of several of the more prevalent fission products produced after various operating times at 50 Mw and several cooling times. Table 2 gives some useful data for calculating internal doses of the more hazardous isotopes.

TABLE 1—SIGNIFICANT REACTOR ACTIVITIES

Isotope	Curies for various cooling times		
	6 hr	24 hr	48 hr
Reactor operation—1 day, 50 Mw			
(1) Xe ¹³³	57.5×10^4	52.0×10^4	45.6×10^4
(2) Ba ¹⁴⁰	24.8×10^4	23.5×10^4	22.2×10^4
(3) La ¹⁴⁰	24.8×10^4	23.5×10^4	22.3×10^4
(4) Pr ¹⁴³	24.0×10^4	23.0×10^4	21.8×10^4
(5) I ¹³¹	17.2×10^4	16.1×10^4	14.8×10^4
Reactor operation—21 days, 50 Mw			
Xe ¹³³	44.3×10^5	40.0×10^5	35.0×10^5
Ba ¹⁴⁰	33.2×10^5	31.8×10^5	30.5×10^5
La ¹⁴⁰	33.2×10^5	31.8×10^5	30.5×10^5
Pr ¹⁴³	32.2×10^5	30.0×10^5	28.4×10^5
Ce ¹⁴¹	19.4×10^5	18.8×10^5	18.4×10^5
Reactor operation—6 months, 50 Mw			
Ba ¹⁴⁰	49.0×10^5	47.0×10^5	45.0×10^5
La ¹⁴⁰	49.0×10^5	47.0×10^5	45.0×10^5
Ce ¹⁴¹	48.6×10^5	47.2×10^5	46.2×10^5
Pr ¹⁴³	48.0×10^5	46.2×10^5	43.7×10^5
Xe ¹³³	47.0×10^5	42.6×10^5	37.5×10^5
Reactor operation—1 year, 50 Mw			
Ba ¹⁴⁰	49.0×10^5	47.0×10^5	45.0×10^5
La ¹⁴⁰	49.0×10^5	47.0×10^5	45.0×10^5
Ce ¹⁴¹	48.6×10^5	47.2×10^5	46.2×10^5
Pr ¹⁴³	48.0×10^5	46.2×10^5	43.7×10^5
Y ⁹¹	47.4×10^5	46.8×10^5	46.2×10^5
Reactor operation—1½ to 3 years, 50 Mw			
Ba ¹⁴⁰	49.0×10^5	47.0×10^5	45.0×10^5
La ¹⁴⁰	49.0×10^5	47.0×10^5	45.0×10^5
Ce ¹⁴¹	48.6×10^5	47.2×10^5	46.2×10^5
Pr ¹⁴³	48.0×10^5	46.2×10^5	43.7×10^5
Y ⁹¹	47.8×10^5	47.3×10^5	46.7×10^5

TABLE 2—CONSTANTS FOR THYROID AND BONE DOSE CALCULATIONS

Thyroid organ (wt, 20 g), $b = 4.45 \times 10^{-8} \text{ sec}^{-1}$, $fa = 0.15$					
Isotope	Half life	Decay constant, sec^{-1}	Effective decay constant, sec^{-1}	Yield	Effective energy, Mev
I^{131}	8.14 days	9.86×10^{-7}	1.03×10^{-6}	0.031	0.22
I^{132}	2.4 hours	8.02×10^{-5}	8.02×10^{-5}	0.045	0.50
I^{133}	20.5 hours	9.44×10^{-6}	9.44×10^{-6}	0.051	0.40
I^{134}	52.5 min	2.88×10^{-5}	2.88×10^{-5}	0.063	0.65
I^{135}	6.7 hours	2.20×10^{-4}	2.20×10^{-4}	0.062	0.31
I^{136}	86 sec	8.06×10^{-3}	8.06×10^{-3}	0.034	2.2
Bone organ (wt, 7000 g)					
Isotope	Decay constant, sec^{-1}	Effective decay constant, sec^{-1}	Yield	Effective energy	Fraction to critical organ
Sr^{89}	1.52×10^{-7}	1.54×10^{-7}	0.047	0.55 Mev	0.22
Sr^{90} , Y^{90}	8.8×10^{-10}	2.97×10^{-9}	0.052	1.00 Mev	0.22
Y^{91}	1.41×10^{-7}	1.57×10^{-7}	0.056	0.57 Mev	0.14
Zr^{95} , Nb^{95}	1.24×10^{-7}	1.7×10^{-7}	0.067	(0.2)	0.12
Mo^{99}	2.82×10^{-6}	2.86×10^{-6}	0.062	0.22	1.8×10^{-4}
Ba^{140} , La^{140}	6.26×10^{-7}	6.69×10^{-7}	0.066	1.06 Mev	0.20
Ce^{141}	2.43×10^{-7}	2.58×10^{-7}	0.059	0.16	0.10
Pr^{143}	5.81×10^{-7}	7.3×10^{-7}	0.061	0.31	0.063
Ce^{144} , Pr^{144}	2.92×10^{-8}	4.46×10^{-8}	0.059	1.29 Mev	0.10
Pm^{147}	5.5×10^{-9}	5.73×10^{-8}	0.026	0.067	0.09
Sm^{151}	2.23×10^{-11}	2.06×10^{-10}	0.005	0.02	0.05
Eu^{154}	4.07×10^{-9}	9.8×10^{-9}	0.0085	0.366	0.09

FIRE POTENTIAL OF ATOMIC REACTORS

By OTTO F. KARG

Idaho Operations Office, U. S. Atomic Energy Commission

The exact fire potential of atomic reactors cannot positively be stated at this time. Many unknown factors still remain to be analyzed before all the potentials can be known. Our Safety and Fire Protection Branch has long felt that fire prevention and fire protection start on the drawing board with the initial designs in the development of reactors.

First considerations, of course, must deal with adequate protection of operating personnel from the standpoint of radiation hazards. Secondly, confinement of contamination is of utmost importance. Ventilation systems properly designed remove the off-gases and contamination when reactors are operating under normal conditions. When a fire occurs, of course, the situation is no longer normal and any number of unusual situations may arise. This is the condition that exists when the fire protection personnel arrive on the scene.

Fire-fighting officers are responsible for seeing that all of their personnel are properly equipped with self-contained breathing masks; film badge and dosimeter; shoe covers, gloves, and protective clothing; and firemen's turn-out suits.

Reactors may be classified according to three different types for the purposes of this discussion: (1) power reactors, which have a low fire potential in ordinary operations; (2) test reactors, which have low fire potential when testing reactor components; and (3) research and development reactors, which have a relatively high fire potential as a result of the types of experiments conducted. The fire protection problems associated with these types of reactors, and some of the methods we have for dealing with them are outlined below.

CLASSIFICATION OF REACTORS BY FIRE HAZARD

Research and Development Reactors

These are machines to test materials at high neutron fluxes—tools to provide neutrons. This type of reactor is relatively high in fire potential, principally as a result of the varied experimentation conducted with it. Many experiments are extremely complex and unusual.

The use of considerable amounts of wax and wood for neutron shielding presents a serious fire hazard. The use of flammable blotting papers in large quantities can be a problem. There are additional fire hazards connected with the cleaning of spills, such as the use of rags and absorbent paper. Even the large quantities of protective clothing stored around the reactor plant create a fire hazard.

Power and Test Reactors

Power reactors operate on a production basis and are used as a source of heat to generate steam for producing electricity. They are brought up to power and operate at a normal level. This type of reactor should present only minor fire potentials, which depend on the coolant used.

Test reactors are designed for the purpose of testing either the reactor itself or its components. The possibility of a major accident is more likely for this type than for the power reactor, but the likelihood of an ordinary fire is about the same.

Power and test reactors may be of various types:

1. Water or nonflammable gas-cooled—considered a low hazard.
2. Air-cooled—considered a moderate fire hazard, attributed to the oxygen content of the air stream.
3. Those using liquid, pyrophoric metals, or organic liquids above the flash point, are definitely a fire hazard. Leaks in the cooling systems are the most frequent hazards. Liquid metals such as NaK ignite spontaneously on contact with air and are very difficult to control. They must be fought by confinement. Spills of NaK two inches or more in depth are very difficult to handle with any known agent. Small fires have occurred in leaks of organic material.

OUTLINE OF HAZARD EXPERIENCE

Research and Development Reactors

At the Materials Testing Reactor (MTR), we have actually experienced fires from the following causes:

1. Fires in shielding may be caused by allowing high temperatures or pressures. In one case, paraffin encased in plywood boxes was stacked against high temperature piping in a shielded cubicle. Lack of ventilation caused the paraffin to liquify and the wood to char. The wax-impregnated wood apparently ignited spontaneously.
2. Charcoal filters. Fires may be caused by oily vapor in an air stream at high temperature and pressure, or by the accumulation of pyrophoric materials. In a fire that occurred this year, activated powdered charcoal, used to filter high pressure air from an experiment, ignited as in a dust explosion and contaminated charcoal was blown throughout the building. The actual loss was only \$200, but the reactor down-time and the cleanup of contamination resulted in a \$22,000 loss. The cause of this fire was unknown, but may have been oily vapors, ozone, or uranium dusts.
3. Minor organic fires occur in experimental test loops, resulting from the leaking of organic materials after they have been heated above the flash point.
4. From our knowledge of materials that have been used around these types of reactors, we consider the following factors important to fire safety:
 - (a) Combustible materials should be held to a minimum in reactor areas. Where large amounts of flammable materials are used for shielding (such as the 15 tons of wax at the MTR), it is important that air space be provided and that a positive supply of cooling air be induced into the shielded area.
 - (b) The possibility of leaks must be guarded against in cooling systems using organic liquids at temperatures above the flash point.
 - (c) NaK and sodium react violently with water, and ignite spontaneously on contact with air at temperatures normally reached in reactor operations. Provisions for adequate damage control must be made where such situations are possible.
 - (d) Radioactive and pyrophoric metals such as some of the fissionable elements may present a combination radiation-and-fire hazard, which must be anticipated—especially when the material is finely divided and at high temperatures.

Power and Test Reactor Experience

1. Water or nonflammable gas-cooled reactors have exhibited a low fire hazard; but possible problems may arise from auxiliary equipment. Depending on the reactor power and the purity of the water, there may be a measurable amount of hydrogen produced. A minor incident at the Naval Reactor Facility was believed to have been caused by hydrogen accumulation. Measurements show that under conditions of design and operation of the MTR, about 80 cu ft of hydrogen is generated per minute.
2. An air-cooled reactor incident occurred at Windscale, England, in a uranium-loaded, graphite-moderated reactor. The release of stored "Wigner" energy during the annealing of

the reactor caused the graphite to swell, obstructed the coolant airflow, overheated sections of the reactor, and caused the fuel element jackets to split. Uranium is pyrophoric and is subject to possible burning if the cladding fails and allows the uranium to come in contact with air. Also, graphite can ignite if the temperature is high enough. After some hesitation about the possible induction of a chemical explosion, the reactor was finally flooded with water to control the interior fire. This successfully cooled the reactor down after a period of time, but not before there had been wide contamination of the countryside and ruination of the plant.

3. Reactors cooled and/or moderated with liquid, pyrophoric metals, or organic liquids above their flash points, have a high chance of producing a fire if a leak develops in the system. Radiation, smoke, and toxicity problems are very acute in a fire of this nature. Safety devices must be built in to handle this type of incident.

In general, any large accumulation of combustible material such as a large quantity of protective clothing or rags and absorbent paper used to mop up spills can contribute to a serious fire condition.

Filter systems should use the newer types of fire resistant filters manufactured by some of the prominent filter manufacturers.

METHODS OF EXTINGUISHING REACTOR FIRES

First, the type of material involved in the fire must be determined; then the appropriate fire-fighting technique can be selected. For pyrophoric or liquid metal fires, a water spray can be used only if the material is in a safe, isolated location designed to withstand any subsequent violent reactions. Otherwise, MET-L-X (resin and salt fused over burning metal), GI powder (graphite), or soda ash may be used to smother the fire. For paraffin or organic liquids, CO₂, dry chemical or water fog may be used. For "Class A" materials, such as wooden blocks, or blotting paper, a high or low pressure water fog may be used. Water can be used on most reactor fires, provided the criticality or pyrophoricity hazards are not too great. A 400-psi high-pressure water fog, finely divided, at a rate of 30 gal/min, would be effective in most cases and would be driven off as steam leaving very little runoff to contaminate or damage facilities.

PREVENTING CONTAMINATION

Firemen should be provided with proper protective clothing and shoe covers, as well as personnel dosimeters and film badges. Men working in the contaminated areas or buildings should not be allowed to return to the trucks to remove equipment, but should have the equipment removed and passed to them across an imaginary boundary or rope by the pump operators.

SUMMARY

In summary, it should be pointed out that the preliminary planning of the plant firefighters and plant designers is the most important phase of the fire protection program. In fire-fighting where radiation is no problem, the principal objectives are: saving life; preventing excessive heat or chemical exposure; confining the fire; extinguishing the fire; and overhauling and salvaging the remains. Where radiation is a problem, the objectives are similar except that the use of water or other extinguishing agents must be done with an eye on the possible spread of contamination, and the additional exposure of personnel to radioactive materials must be controlled.

INSTRUMENTATION IN AIRCRAFT FOR RADIATION MEASUREMENTS*

By F. J. DAVIS

*Health Physics Division,
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(Abstract)

The results that are described were obtained with the more recently developed NaI scintillation detector that has been used by the U. S. Geological Survey in making radiation surveys throughout the United States and Canada. These scintillation detectors have been found to be superior in sensitivity to the Geiger-Müller counters and air conductivity apparatus used previously.

The equipment installed in the Douglas DC-3 U. S. Geological Survey aircraft includes a detector unit of six NaI(Tl) crystal detectors, each 4 in. in diameter by 2 in. thick and shielded on the sides by $\frac{1}{2}$ in. of lead. The crystals are each in contact with a 6364 photomultiplier tube operating on approximately 1500 volts obtained from an Atomic Instrument Supply Company Model 306 high voltage supply. The signals from the photomultipliers are fed to a mixing preamplifier and from there through cables to the linear amplifier. The crystals, photomultipliers, and preamplifier are situated below the floor and toward the rear of the aircraft. A Frahm resonant-reed controlled oscillator of frequency 500.2 cycles per second can be switched into the circuit to calibrate the rate meter. A discriminator allows only pulses above a certain pulse height to be accepted by the rate meter. This pulse-height level is usually chosen for normal operations to be the height of pulses originating from 50-kev gamma radiation. The rate meter signal is fed to two vacuum-tube voltmeters that record the counting rate on Esterline-Angus recorders. One of the recorders has a variable series resistance that is controlled by the radar altimeter servomechanism to compensate the counting rate for altitude fluctuations. Another recorder records the count rate uncompensated for altitude, and a third recorder records the height of the aircraft above the terrain.

The method of determining the plane's position was developed by the U. S. Geological Survey for use in aerial magnetic surveying. A gyro-stabilized continuous strip-film camera photographs the ground directly below the aircraft. Comparing this strip film with aerial photomaps gives an accurate record of the path of the aircraft. When crossing recognized points on the ground, the operator of the drift sight manually operates an edge marker switch that places edge marks on all recorders and operates simultaneously a shutter to correlate any point on the record with a corresponding point on the film.

Since the equipment must operate over a wide range of temperature, and since the equipment is temperature sensitive, provisions for adjusting and calibrating the equipment using a cesium-137 source are provided in the aircraft. The source is manipulated by a pneumatic tube that blows the source from behind a lead shield to a position below the crystal detectors. To adjust the over-all amplification of the system, a curve is taken of count rate versus dis-

* Much of the data and technical details can be found in an article by F. J. Davis and P. W. Reinhardt, *Nuclear Sci. and Eng.* 2: 713-727 (1957).³

criminator setting. At higher discriminator settings, the count rate will begin to decrease rapidly and go through an inflection point that corresponds to the 0.661 Mev photopeak of the differentiated spectrum. The amplifier is then adjusted so that the same counting rate obtained at the inflection point is obtained with the discriminator knob set at 661. This combination of count rate and discriminator setting can then be used under each situation to adjust the equipment to a standard response, which ensures that the discriminator setting reads directly in kev. An additional check can be made with the cesium source in position and the discriminator set at the usual operational setting of 50 kev, to ensure that the system is counting the usual number of pulses above 50 kev. The rate meter itself is periodically calibrated with the oscillator.

With the discriminator set at 50 kev, the sensitivity in counts per second/microroentgen per hour varies with gamma ray energy from zero below 50 kev to about 3000 at 50 kev, a peak of about 3400 at 75 kev, 550 at 410 kev (Au^{198}), 280 at 661 kev (Cs^{137}), and 120 at 1250 kev (Co^{60}). The count rate for a dose rate of one microroentgen per hour from radium gamma rays is 225 counts per second (cps). When flying over a 1 g radium source at 2000 ft, an increase of 50 cps over a background of 220 cps was noted.

The sensitivity ranges are 10, 20, 50, 100, and 200 cps per division where 100 divisions is the full-scale deflection. The time-constant choices are 10/S, 20/S, 40/S, and 100/S seconds, where S is the sensitivity setting in counts per second per division. However, the time constant is limited by the recorder to not less than 0.4 second.

The error in readings is largely due to the uncertainty in altitude. The radio- and barometric altimeters have uncertainties of plus or minus 50 ft. Where accurately spaced markers can be placed on the ground, this error can be reduced by the use of photographs. If the source to be measured is on land, fluctuations in background will be reduced if there is water or snow on the ground to absorb some of the background gamma radiation. Finally, there is a statistical fluctuation in the rate meter of standard deviation $(R/2t)^{1/2}$, where R is the counting rate in counts per second and t is the time constant in seconds. However, if the source is constant for time T, where T is much longer than the time constant t, then the standard error in the average reading over time T will be $(R/T)^{1/2}$. The choice of a long time constant t might reduce the statistical fluctuations in measuring a fairly constant source of radiation; however, t must be kept short in order to have the instrument respond to small localized sources on the ground. In measurements made at low altitudes, a faster responding recorder was used to obtain time constants less than 0.1 second.

Variation of Radiation With Altitude

The background radiation received by a scintillation detector is composed of the components given in Table 1, as measured near Oak Ridge, Tennessee.

TABLE 1—BACKGROUND COUNT OF AERIAL SCINTILLATION DETECTOR

	On ground	500 ft above ground
Air (radon products)	50 cps	80 cps
Ground	900 cps	300 cps
Cosmic rays	200 cps	200 cps

The radon concentration builds up during the night and is usually highest in the morning. The cosmic ray background is fairly constant up to altitudes of 10,000 ft, and since most of the cosmic background spectrum is below 3 Mev it cannot be "gated" out electronically.

The variation of radiation intensity I with distance above a point source can be expressed by the relation,

$$I = \frac{I_0 e^{-\mu h} B(\mu h)}{h^2} \quad (1)$$

where μ is the absorption coefficient for air and $B(\mu h)$ is the build-up factor due to radiation scattered by the air. The function $B(\mu h)$ can be expressed empirically in a power series as

$$B(\mu h) = 1 + a\mu h + b(\mu h)^2 + \quad (2)$$

where a , b , etc., are constants. When close to the source, $B(\mu h)$ approaches unity, but in the neighborhood of 500 ft and over, the second term becomes the controlling term, so that the intensity expression can be approximated by

$$I = \frac{I_0 e^{-\mu h} a\mu h}{h^2} = \frac{a\mu I_0 e^{-\mu h}}{h} \quad (3)$$

The value of a generally falls between 4 and 7, depending on the primary gamma-ray energy and the discriminator setting. For cobalt-60 gamma rays the values of a using the discriminator settings of 50 kev and 100 kev are 5.4 and 4.2, respectively. These values were computed from tables by Goldstein and Wilkins.²¹ The value of a when the build-up factor is expressed as energy or dose build-up, rather than gamma quanta build-up, is of the order of unity.

Experimental measurements of intensity versus altitudes up to 2000 ft were in agreement with Eq. 3 above 400 ft in that the points fell on straight lines when intensity times height was plotted on a semilog scale versus height on a linear scale. Measurements using point sources of Co^{60} , Ra , Cs^{137} , and Au^{198} yielded half-thicknesses in air of 350, 310, 240, and 190 ft, respectively, where the half-thickness is given by $T_{1/2} = 0.693/\mu$. Note that the exponential attenuation calculated from these half-thicknesses must be corrected for altitude and build-up factor as given in Eq. 3.

Note also that these half-thicknesses are in terms of air at standard temperature and pressure, so that readings taken at high elevations or temperatures much different from normal would have to be interpreted using corrected values of half-thickness.

When Eq. 3 is integrated over a plane to obtain the attenuation of radiation from a broad source on the ground, the intensity becomes

$$I = k_1 I_0 e^{-\mu h} \quad (4)$$

where k_1 is a constant. Interestingly, the same form of equation is obtained when Eq. 3 is integrated over a semi-infinite space with a plane surface, except that k_1 is replaced by a different constant k_2 . In our own case, a modification of Eq. 4 was used, obtained by integrating Eq. 3 only over the area seen by the detector crystal through the cone defined by the $1/2$ -in. lead side shield. This modified theory gave an excellent fit to the data measured from fall-out fields from altitudes ranging from 100 to 1200 ft. The half-thicknesses for radiation from fall-out fields, normalized to a detector altitude of 1000 ft, representing fall-out from Nevada weapons tests as detected at Oak Ridge, Tennessee, Casper, Wyoming, and Fruita, Colorado, are given in Table 2.

TABLE 2—HALF-THICKNESS IN AIR FOR FALL-OUT AND NATURAL GROUND ACTIVITY

Sample	Half-thickness in air (measured at 1000 ft)
Fall-out 2 days old	270 ft
Fall-out 13 days old	230 ft
Natural activity of ground	310 ft

This fall-out followed the $I = I_1 t^{-1.2}$ decay law for gross fission products.

Measurements of intensity minus cosmic background were taken as a function of altitude above the natural ground activity of a flat mesa near Fruita, Colorado, using several different discriminator settings. The resulting data plotted as logarithm-intensity versus height yielded four parallel lines, one for each discriminator setting. This indicates that the energy distribution of the scattered gamma rays from natural ground activity has essentially reached equilibrium, and the spectrum does not change much from 100 to 1000 ft in altitude.

In normal survey operations, the aircraft is flown at 500 ft above ground to give optimum coverage for minimum loss in sensitivity. Smaller planes with less sensitive equipment would ordinarily choose lower elevations. For a point source of radium on the ground, at a height of 500 ft the intensity of radiation will be one-half of the intensity directly over the source when the plane is 480 ft horizontally away from the source. For lower altitudes, the distance at half intensity varies almost linearly with height, while for high altitudes, the half-intensity distance varies with the square root of the height. Since the aircraft travels at a speed of about 220 ft/sec, a point source on the ground is seen at 500 ft altitude about 2 sec before and after the plane passes over the source. The time constant of the rate meter generally used is about 1 sec, and the time constant of the recorder is 0.4 sec.

Records of a flight from St. Louis, Missouri, to Moline, Illinois, show the variations of background with terrain. The average count of 500 counts per second decreases to about 200 counts per second (cosmic only) when crossing a large body of water, and decreases noticeably when approaching areas covered by several inches of snow.

[EDITORS' NOTE: For fission product radiation, Dr. Davis' curves indicate a detector sensitivity of about 300 counts per second/microroentgen per hour and an attenuation factor from ground to air of about three. Thus fission product radiation from a broad source on the ground giving one microroentgen per hour near the ground should give a count rate of about 100 counts per second in the sodium iodide detectors at 500 ft.

For I^{131} radiation, the response of the detector should be slightly higher, but the attenuation factor should also be higher, thus giving the same order of sensitivity. During the Windscale incident, it was found that about one microcurie of iodine per square meter of grazing area would result in about 0.1 microcurie per liter of milk a few days after the incident, which was the conservative level at which milk was rejected after the incident. It can be calculated that one microcurie of iodine per square meter over a broad area will give an intensity of about 3 microroentgens per hour near ground level. This would result in a count rate from the sodium iodide detectors at 500 ft of about 300 counts per second.

Thus it can be seen that the method of aerial survey presented by Dr. Davis would be quite sensitive enough for rapid surveys after an incident involving the widespread release of fission products.]

WINDSCALE INCIDENT

By J. W. HEALY

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(Abstract)

The Windscale accident occurred during a release of Wigner energy from the graphite of the Number One reactor. The situation was not recognized until visual observation of the glowing uranium indicated major difficulties. Water was used to cool the reactor starting on the morning of October 11, 1957, and by 1510 on Saturday, October 12, the reactor was cold.

The winds were light on Thursday, October 10, and mainly offshore. During the night they changed direction, and throughout Friday a 10-knot wind blew down the coast. There may have been an inversion at a few hundred feet above the ground during part of the period of release.

A portion of the radioactive materials emitted was caught on the filters at the top of the stack. Surveys in the region indicated that external radiation and inhalation were no problem. The limitation for ingestion was based on milk consumption, due to I^{131} , and a limit of 0.1 μc /liter in milk was used based on a dose of 20 rads to the thyroids of children. Measurements of other isotopes indicated no problem. Restrictions were placed on milk at 2100 on October 12, and by October 14 they were in force over most of the area of concern.

It was concluded that it is extremely unlikely that any harm has been done to the health of any person. Certain weaknesses in organization were apparent to the individuals concerned after the event, but the measures adopted were essentially right, and once they were brought into play they were applied decisively and were adequate. The review committee has recommended certain organizational changes including the integration of the health and safety management into the over-all plant management.

NRTS REACTOR EXPERIMENTS

By P. GRIFFITHS

Idaho Operations Office, U. S. Atomic Energy Commission

During tests at the National Reactor Testing Station (NRTS), several reactors have provided effluent releases that were detectable by field monitoring procedures.

Contrary to some opinion, these releases have not made measurable exposure or contamination of off-site personnel or environments; in one case minimal exposure was calculated to be possible. We are satisfied with the record of exceptional control that we have achieved in connection with these operations, but do not minimize the potential hazard surrounding these tests, especially in view of the fact that the NRTS is located upwind of a large agricultural area. A number of factors have been responsible for the success of this radiation protection program. To enumerate a few:

1. Cooperation between operators and the Idaho Operations Office (IDO)
2. Communications
3. Meteorological control
4. Monitoring network

MONITORING PROGRAM

Very close liaison is maintained between the contractors, Idaho Operations Office, and other offices of the AEC. This is accomplished through weekly briefing meetings and freedom of communication between groups and individuals regardless of affiliation. During operations, direct communication is maintained between the operating contractor, IDO operation control, the U. S. Weather Bureau, and all field monitoring units by means of an excellent two-way radio network. When any doubt arises regarding the safety of a particular operation (for instance, wind shifts or increased activity) reactor shutdown can be, and has been, accomplished immediately. The Division of Security has evolved an emergency evacuation plan.

Meteorological control has been one of the greatest assets of these operations. Unfortunately, some of the test areas are in locations that are extremely unstable from a meteorological standpoint. Very extensive coverage is essential during any test in these areas.

The monitoring network developed for NRTS operations is complex—consisting of a 23-unit telemetering system, supplemented by conventional grids and supporting sampling devices. Mobile monitoring teams are placed downwind at distances ranging from 1 to 20 miles during operations.

SPECIFIC OPERATIONS

During a test operation on February 22, 1956, site survey monitoring two miles downwind detected airborne radioactive material at levels of approximately 1.5 mr/hr as measured with Geiger-Müller type instruments. Reactor shutdown was effected at the time of detection. Subsequent surveys identified an area of approximately 12 square miles contaminated to levels

averaging 20 mr/hr. Since this release occurred over the unoccupied area of the NRTS, with the exception of the immediate test area, no exposures were incurred to personnel other than those in this area. Owing to the short half-lives of the isotopes involved, levels decreased to normal within a few days.

On February 24, 1956, IDO approved operation of the damaged reactor under intensified monitoring and meteorological surveillance. At this time the operation was conducted with the wind direction over an off-site agricultural area with results similar to those described above, occurring in the area between the reactor and the Site boundary. Gamma radiation levels of 3 mr/hr were observed in air at 5.5 miles; 1.5 mr/hr at 8.5 miles. Fall-out at distances greater than 5.5 miles was negligible.

Subsequent tests to December 1957 have not provided activity or dosages of consequence. An experiment currently in progress has provided radioactive effluent release at levels that are detectable in the field. With the rapid decay of the isotopes involved, detection has been extremely difficult at moderate distances. Extensive aerial cloud tracking has been undertaken during this operation, and the results of decay are quite apparent. As a side light, we are preparing a presentation of the results acquired on this aerial monitoring endeavor. Milk and vegetation are being analyzed routinely. We have detected no increase of activity over normal background levels. Jack rabbit thyroids show increases ranging in magnitude by factors of 2 to 10; we are confident that this increase has resulted from the present operation.

RALA MONITORING EXPERIENCE*

By E. R. EBERSOLE

Idaho Operations Office, U. S. Atomic Energy Commission

(Abstract)

The dissolution of short-cooled fuel elements (2 to 4 days) in the RaLa Process produces gaseous fission products which are difficult to monitor with conventional instruments. The principle gaseous fission products of concern are the radioactive iodines. Dissolution of the aluminum cladding and of the fuel element were carried out under conditions controlled by meteorological data, and construction personnel were evacuated from the area during the early RaLa runs. A sky-scanner aimed at the top of the off-gas stack indicated that the time and extent of gaseous fission product release coincided with the dissolution operation.

Jack rabbit thyroid glands collected on the National Reactor Testing Station following each RaLa run indicated that radioactive iodine was being released from the stack after the dissolution process was complete and meteorological control had been lifted. Twenty-four hour monitoring of the stack by the sky-scanner following the dissolution verified this fact.

The quantity of radioactive iodine released following dissolution of the element was reduced in subsequent RaLa runs by delaying the cleanup of equipment until just prior to the next run.

An inexpensive continuous-monitoring device employing activated carbon for trapping the gaseous iodine was developed and put into use in June 1957. Continuous monitoring of the air by this device made it unnecessary to evacuate personnel during the operation and thus saved many man-hours of construction time.

*Presented before Emergency Monitoring Group, Feb. 13, 1958.

FUEL ELEMENT BURN TEST

By G. VICTOR BEARD*

U. S. Atomic Energy Commission, Washington, D. C.

INTRODUCTION

For evaluating reactor hazards, the per cent of various fission products that will escape from a damaged or melted fuel element must be known. Before 1957, very little experimental information was available on which to base estimates of fission product escape. In the author's opinion, this resulted in occasional hazard estimates that assumed catastrophies beyond physicochemical possibility.

To demonstrate the behavior of a fuel element engrossed in a large fire and to provide some initial data on the percentage release of fission products to the environment, the Idaho Operations Office decided to run field tests to measure the radioactive nuclides released from fuel elements destroyed by fire. Two field tests were run at the National Reactor Testing Station:

Phase A

A 90 per cent enriched fuel element clad in stainless steel was supported in an airframe and surrounded by other reactor materials to simulate conditions of a nuclear airplane crash. The fuel element contained about 5000 curies of fission products. A pool of jet fuel underneath the airframe was ignited and the fire burned for about 2 hr. After the fire died, no activity was measurable by portable survey equipment except inside a 24-sq ft area around Ground Zero. Levels averaged 200 mr/hr at 3 ft above ground within this area. Low-activity levels were detected with air samplers and fall-out plates downwind to 5000 ft, indicating that only an extremely small fraction of the fission products had been released in this experiment. The fuel element was still intact after the fire except for one small puncture, although it had reached a temperature of about 2250°F.

Phase B

A fuel element was heated to about 5000°F by an oxygen-fed fire of thermite, steel wool, and iron filings. The fuel element containing 10,000 curies of fission products was supported upright in a cylindrical furnace and surrounded by the combustible materials. Oxygen was fed into the bottom of the furnace through a side tube near the lower part of the element. An arrangement of feeders around the fuel element dispersed the oxygen throughout the furnace at a steady rate during the burning period. The burning continued for about four minutes. Most of the fuel element was melted and dispersed within 90 seconds. Some of the molten uranium and fission products spurted out the top of the furnace in Roman candle fashion, but most of the molten element ran out the side tube to the ground.

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After the fire was out, analysis of filter samples and fall-out plates showed that ruthenium-103 and -106, cesium-137, and halogens (mostly iodine-131) were the principal activities released. These nuclides contributed greater than 50 per cent, sometimes 80 per cent, of sample activities. Only a minor amount of the activity released (less than 2 per cent) was strontium-90. Assuming that 100 per cent of the cesium, ruthenium, and iodine content of the fuel element had been released in the fire, it was estimated that not more than 10 per cent of the fission product activity had been released.

Measurements of gamma radiation from the fuel element, before and after burning, indicated that the per cent of fission products released was probably much less than 10 per cent. Even though the molten element was spread on the ground after the fire, measurements at several places in the field indicated almost the same radiation levels that were obtained when the element was intact. Reductions at other points could be attributed to absorption by the terrain. Fission product samplers indicated that practically all of the fission products had deposited within a distance of 2000 ft, and a great majority of the material had deposited within 300 ft. These results were obtained with 15-mile-per-hour winds varying in direction to within about 45 degrees. An inversion condition up to several hundred feet and the shifting winds caused the loss of much of the data that instruments had been set up to obtain. However, the main objectives of the experiment were accomplished.

Further details and discussion are given in the sections below.

PROCEDURE

Phase A—Airframe Crash Mock-up

Five trials were run without the element or airframe in place to determine optimum conditions for heating the fuel element. The experiment was then set up as follows: A 90 per cent enriched fuel element containing approximately 5000 curies of 70-day-old fission products was supported by a four-legged stand. The stand was designed to keep the element in fixed position throughout the experiment. This would have allowed immediate estimates of fission product release to be made from gamma dose-rate measurements. (Unfortunately, in the final assembly a brass nut was mistakenly substituted for a damaged steel one and the support collapsed. This prevented an early estimate of results by gamma measurements.)

The fuel element was surrounded by other reactor materials and a section of a real airplane fuselage to simulate conditions of an airplane crash. The skin of the airframe was made of magnesium, and the supporting members were aluminum. Holes were put in the fuselage to enhance burning. The entire assembly was placed above a teardrop-shaped pool of jet fuel. The teardrop shape had been suggested by trials to keep the fuel element in the hottest part of the flame under expected wind conditions. Trials had shown also that thermite ignition of the jet fuel was required. All experimental conditions were checked before the fire was begun but the brass nut was overlooked.

The fuel was ignited at 1419 on March 20, 1957, and the fire burned until 1615. Automobile tires placed above the fuel element provided a dense black smoke trajectory that could be located visually. Within six minutes, the brass nut on the fuel-element support melted, and the element was dropped onto a platform slightly above the burning jet fuel. However, it was later found that the element had remained in the hottest part of the flame and had reached a temperature of 2250°F, about 100 degrees higher than any temperature obtained in the dry runs.

The field instruments for measuring fission product release consisted of portable survey instruments for walk-in measurements, high-volume air sampler with MS-2133 filter paper, fall-out papers (1 sq ft of surface area) on stands 18 in. above ground level, and electrostatic precipitators and millipore filters for collecting particles for particle-size measurements. Blimps tied to trucks contained some of the air sampling equipment for portability.

Meteorological data were obtained from a network of stations. Winds up to 6000 ft, as well as surface winds, were obtained using balloon ascents and tower instruments. The vertical rise and spread of the plume from Phase A were measured by pairs of theodolites on each side of the plume. Two sky scanners (collimated scintillation detectors) were operated 2000 ft downwind to detect radioactive material passing overhead.

Dutch rabbits were positioned in a network out to one mile from the airframe.

Further details of these measurements are given in References 101 to 103. Also, an unclassified motion picture¹⁰⁴ has been prepared, giving a view of the general procedures and results of these experiments.

Phase B—Oxygen-Furnace Meltdown

In the Phase B burning experiments, a 10,000-curie element was melted in a furnace designed to attain a temperature of 5600°F. The fuel element was loaded in upright position into a cylindrical furnace lined with 3 in. of magnesia. The element was surrounded by 9³/₄ lb of thermite, 2 lb of steel wool, 1³/₈ lb of galvanized iron filings, and 6 squid ignitors. Ten preliminary mock-ups had determined a design that would ensure the melting of the element. Oxygen was fed from a 600-lb gas cylinder through a pipe into the bottom of the furnace and was dispersed throughout the furnace by feeders to assure maximum and uniform heating of the fuel element.

At 1847 on March 20, 1957, the furnace was ignited and it burned for about 1¹/₂ min. The entire fuel element was melted, most of the melt running out the bottom of the furnace onto the ground and some of it sputtering out the top of the furnace. Despite careful planning, the first puff of smoke from the furnace went sailing in a more northerly direction than the rest of the smoke. Small amounts of activity were detected from this first puff by air monitors at the Naval Reactor Facility (NRF) 5 miles away. The major part of the fumes, however, were directed within a quadrant of 45°, intersecting some of the sampling stations.

The same detection methods were used in Phase B as in Phase A. A fresh change of filter papers and a new set of rabbits completed preparations for detection in Phase B.

RESULTS

Again, we refer to References 101 to 104 for complete details. This report presents a summary of the essential information.

Phase A—Airframe Crash Mock-up

1. *Original Fission Product Content.* The activities of this element were calculated to be about 5000 curies, assuming an operating time of 120 hr and a cooling time of 70 days (Table 1).

2. *Final Condition of the Element.* Slides are available* showing the final condition of the fuel element.¹⁰⁵ Although temperatures up to about 2250°F were attained, the element was intact after the experiment except for one small triangular puncture of the stainless steel cladding. The element was found to have remained in the area of the jet fuel pond designed for maximum temperature. Flames had surrounded the element for almost 2 hr. After the experiment, the element was pried loose from a mass of resolidified magnesium and aluminum. It appeared that little magnesium or aluminum had actually burned during the experiment.

3. *External Gamma Readings.* Isodose lines were plotted before and after the fire.¹⁰⁸ The large decrease in dose rate at some points of the terrain at first gave the impression that there had been about a 50 per cent release of fission products. However, further measurements gave readings in some areas that were almost as high as the initial readings, and inspection of the debris showed that the element support had collapsed. Thus the lower gamma readings were attributable to shielding by the terrain. The fact that some of the gamma readings were about the same indicates that only a very small fraction of the fission product activity was released. The collapse of the support prevents a more accurate estimate of the fission product release from gamma dose-rate measurements.

4. *Meteorological Conditions.* Since the fuel element did not burn and there was a very small release of radioactivity, the experiments did not determine whether or not most fission products released in such a fire would follow the smoke trail. Detailed measurements of wind

*Not presented with this paper.

TABLE 1—FISSION PRODUCT COMPOSITION, PHASE A*

Nuclide	Fraction	Curies	Half life
Nb ⁹⁵	0.18	900	35 days
Zr ⁹⁵	0.12	600	65 days
Sr ⁸⁹	0.11	550	53 days
Sr ⁹⁰ —Y ⁹⁰	0.002	10	19 years
Y ⁹¹	0.11	550	61 days
Ce ¹⁴¹	0.10	500	33 days
Ce ¹⁴⁴ —Pr ¹⁴⁴	0.05	250	280 days
Ru ¹⁰³	0.06	300	40 days
Rh ¹⁰³	0.06	300	54 minutes
Ru ¹⁰⁶	0.005	25	1 year
Cs ¹³⁷	0.002	10	33 years
Ba ¹⁴⁰ —La ¹⁴⁰	0.06	300	12 days
Pr ¹⁴³	0.03	150	14 days
Nd ¹⁴⁷	0.006	30	11 days
I ¹³¹	0.005	25	8 days
Xe ¹³³	0.004	20	5 days
Te ¹²⁹	0.006	30	33 days
Miscellaneous plus error	0.09	450	
Total	1.00	5000	

*NOTE: Based on 20 Mw elements at 120 hr operation plus 70 days immediate decay. Assuming a gamma energy of 0.7 Mev and a gamma "curie" content of 2500, the radiation field would be 12,000 r/hr at 1 ft from a point source.

speeds and direction are given in Reference 102. Reference 110 shows the sector of possible contamination and the most probable path of the plume as calculated from mean wind speeds and directions. The times given on the trajectory show when the plume arrived at each perimeter, assuming material was first emitted at 1415 MST. The slide shows that the plume traveled about 25° north of the sampling sector most of the time.

The top of the cloud rose to a measured mean height of 250 ft at 1000 ft downwind, and to 675 ft at 4000 ft downwind. The bottom of the cloud was at mean heights of 65 ft and 160 ft at 1000 ft and 4000 ft, respectively. This shows that the upper winds were more influential than the lower winds in the movement of the cloud. The 15-min average wind directions at 250-ft altitude, at point CF¹¹⁰ from 1415 to 1600 MST, ranged from 200 to 218° (coming from the southwest); the average wind velocities were 8 to 12 miles per hour. The 20-ft altitude winds ranged from 200 to 210° at 13 to 14 miles per hour. Thus the cloud moved 20 to 35° to the northwest of the sampling sector most of the time, but short-lived direction changes did bring the cloud directly over the sampling sector at times.

The vertical temperature stratification during the burning period showed temperature decreases of 3 to 4°F from ground to 250 ft, indicating optimum diffusion conditions for this phase. See Table 2 for temperature measurements from 1400 to 2400 MST.

5. *Environmental Measurements.*¹⁰¹ After the element had been removed, surveys showed that no contamination levels above background could be detected with portable monitoring instruments except within an area of about 24 sq ft at the test location. Here the readings averaged 200 mr/hr at 3 ft above ground.*

High-volume air samples and fall-out plates located downwind produced some low-level activities that were measurable by radioautographic and analytical methods. No sample beyond 5000 ft gave results above background. Reference 109 presents all fall-out data from Phase A.

6. *Animal and Vegetation Samples.* Dutch rabbits were exposed to inhalation of material at sampling arcs 100, 300, 500, and 1000 ft from Ground Zero on both sides of the 55° north-

*EDITORS' NOTE: An infinite area source of 1 $\mu\text{C}/\text{m}^2$ of fission products gives 10.6 $\mu\text{r}/\text{hr}$ at 3 ft. On this basis an average reading of 200 mr/hr over 24 sq ft would indicate a deposition of about 0.1 curie.

TABLE 2—VERTICAL TEMPERATURE VARIATION
AS A FUNCTION OF TIME

(15-Minute Intervals, March 20, 1957)

Time (MST)	5 ft	100 ft	250 ft
1400	52.4	49.9	49.0
1415	55.4	50.6	49.9
1430	53.5	51.0	50.3
1445	54.4	51.2	50.3
1500	53.0	50.5	49.8
1515	53.5	50.6	50.0
1530	53.9	51.0	50.1
1545	54.9	51.1	51.0
1600	53.4	51.0	50.1
1615	54.9	52.1	51.0
1630	53.2	51.2	50.4
1645	53.8	51.6	50.5
1700	52.6	51.3	50.2
1715	52.1	51.0	50.0
1730	51.9	51.0	50.0
1745	51.5	50.6	50.0
1800	51.0	50.4	49.8
1815	50.5	50.0	49.5
1830	49.3	49.2	49.1
1845	47.6	48.5	48.6
1900	47.0	48.0	48.5
1915	45.0	47.6	47.6
1930	45.0	47.9	48.1
1945	45.0	46.5	46.7
2000	43.7	45.2	46.1
2015	44.1	44.9	45.8
2030	43.8	44.3	46.5
2045	43.2	44.1	47.1
2100	42.0	43.5	46.5
2115	42.2	45.6	45.6
2130	42.4	45.5	45.9
2145	42.9	46.2	46.2
2200	42.5	46.7	46.5
2215	43.1	47.0	46.5
2230	43.7	47.0	47.0
2245	44.2	47.0	46.9
2300	42.6	44.4	46.9
2315	41.5	44.4	46.6
2330	39.4	44.9	47.0
2345	40.6	43.0	46.1
2400	39.0	41.5	45.4

TABLE 3—FISSION PRODUCT COMPOSITION, PHASE B*

Nuclide	Relative yield (activity)	Half life
Kr ⁸⁵	0.3	10 years
Sr ⁸⁹	4.8	53 days
Sr ⁹⁰	5.9	20 years
Y ⁹⁰	5.9	65 hours
Y ⁹¹	5.9	61 days
Zr ⁹⁵	6.4	65 days
Nb ^{95m}	0.06	90 hours
Nb ⁹⁵	6.4	35 days
Ru ¹⁰³	2.9	40 days
Rh ^{103m}	2.9	54 minutes
Ru ¹⁰⁶	0.4	365 days
Rh ¹⁰⁶	0.4	30 seconds
Te ^{127m}	0.06	90 days
Te ¹²⁷	0.06	9 hours
Te ^{129m}	0.34	33 days
Te ¹²⁹	0.34	72 minutes
I ¹³¹	2.9	8 days
Xe ¹³³	6.5	5 days
Cs ¹³⁷	5.9	33 years
Ba ^{137m}	5.4	3 minutes
Ba ¹⁴⁰	6.3	13 days
La ¹⁴⁰	6.3	40 hours
Ce ¹⁴¹	6.0	32 days
Pr ¹⁴³	6.2	14 days
Ce ¹⁴⁴	6.1	280 days
Pr ¹⁴⁴	6.1	18 minutes
Nd ¹⁴⁷	2.6	11 days
Pm ¹⁴⁷	2.6	2.6 years

*Based on 6.16×10^{21} fissions and immediate 250-day cooling. Calculated from individual nuclide properties.

east sampling line.¹¹⁰ Three rabbits were on the 100-ft arc, four each on the 300-, 500-, and 1000-ft arcs. Each rabbit was 2½ ft above ground and facing Ground Zero. None of the rabbits exposed on Phase A showed significant activity in the lungs or on the fur of the head.

Radioautographs of sagebrush on the 300-, 500-, and 1000-ft arcs gave no indication of an increase in radioactivity after Phase A.

Phase B—Oxygen-Furnace Meltdown

1. *Original Fission Product Content.* Table 3 presents the calculated fission products content of the Phase B element, totaling about 10,000 equivalent curies of beta and gamma activity. The activities of this element are based on 6.16×10^{21} fissions and immediate 250-day decay.

2. *Final Condition of the Element.* The entire fuel element melted and ran out the bottom of the furnace to the ground. Part of the element sputtered out the top of the furnace to the immediate area, most of the material remaining within 50 ft. The element was melted within 1½ min and the fire was over in about 4 min at 1851 MST. The maximum temperature was estimated to be about 5000°F.

The radiation level from the fuel element dispersed on the ground was 1.25 r/hr at 40 ft, compared to readings before the experiment of 2.1 r/hr at 40 ft from the element in the furnace, and 3.2 r/hr at 80 ft from the unshielded element hanging just above the furnace.

3. *External Gamma Readings.* References 111 and 112 show isodose curves taken before and after the fire and radiation survey levels around the immediate test area. The levels at some distant points in Reference 111 were almost the same after the fire as before the fire.

This indicates that only a small fraction of the fission product activity was released beyond 50 ft from the furnace.

4. *Meteorological Conditions.* By the time Phase B was ignited (1847 MST), a temperature inversion had set in as shown in Table 2. The wind had shifted from the former 55° angle to an average of about 31°, which caused the trajectory to miss all sampling stations except Numbers 109 and 110 (Ref. 109). The trajectory was calculated from wind observations as described in Ref. 102, and is shown in Ref. 113.

The 20-ft winds at the test site for the first two 15-min periods after 1846 MST averaged 7 miles per hour from 215° and 6 miles per hour from 210°, with a total swing from 195 to 235°. This shows that just downwind of the test site, the cloud moved 20 to 25 degrees to the north-west of the center line of the sampling sector. The vertical temperature variation had changed to an inversion condition, with temperature increases of 1 to 2 degrees Fahrenheit from the ground to the 250-ft level. This inversion prevented the cloud from rising to higher levels and gave poor diffusion conditions.

Thus the smoke stayed low and remained roughly within a 25° plume along the trajectory, except for the first puff of smoke from the furnace. This initial puff trailed slightly to the west of the later trajectory and carried a small amount of activity to the NRF facility 5 miles north.

The two scintillation-type sky scanners, operated at 2000 ft downwind and 500 ft off the sampling area center line, could not be tilted low enough to detect the unexpected trajectory. To be detected by the western-most scanner, the 211° plume would have had to rise above 1000 ft.

None of the blimps were in a position to get a sample, since the trajectory was too low and out of their paths. This was disappointing, since great pains had been taken to plan the experiment. Meteorological conditions had been studied in advance and ten thousand curies were used based on the experience that under average conditions a release of 1000 curies per mile will yield easily measurable activity out to 10 miles. Nevertheless, some activity did deposit on Stations 109 and 110 of the ground network, allowing an analysis of the type of fission products carried off in the plume.

5. *Environmental Measurements.*^{101,103} Reference 112 shows the results of the radiation survey by portable instruments. These measurements, together with Ref. 111, show that most of the fission products remained in the molten material that was spilled on the ground near the furnace.

Field surveys were made with Geiger-Müller survey meters, supplemented by a deep-well counter for measuring vegetation samples. Measurements at Stations 109 and 110 were as follows:

	Distance from furnace, ft	Hi-volume air sampler paper, counts per minute	Fall-out paper, counts per minute
Station 109, 31° east of north	450	1×10^7	-
Station 110, 33° east of north	550	3×10^6	3.5×10^5

Surveys indicated that fall-out had occurred in significant amounts along the 31° trajectory out to 2000 ft. Beyond 2000 ft, ground measurements were not significant. Reference 114 gives intensity readings and vegetation activity readings along the 31° radial. Table 4 shows the Geiger-Müller readings and vegetation activities along perpendiculars to the 31° radial.

The only activity detected beyond 2000 ft that could be attributable to Phase B was that obtained at NRF from the first puff (as mentioned previously); and a small increase in radio-activity on an air monitor at Reno Ranch, about 34 miles from the test site. At 1925 MST, air monitors inside and outside the main NRF building showed a sharp increase in activity by a factor of ten, remaining above background for about 40 minutes. Filter samples removed from air collectors in and out of the main building indicated air concentrations of 16 to 21 disintegrations per minute per liter from 1940 to 1958 MST, and 1.5 disintegrations per minute per liter outside the main building at 2018 MST.

An autoradiograph of the sample tape from the air monitor at Reno Ranch confirmed the collection of small amounts of particulates from about 2115 to 2230 MST.

TABLE 4. GEIGER-MÜLLER AND VEGETATION READINGS, PHASE B

(A=Geiger-Müller readings mr/hr; B=vegetation c/m/g)

	550 ft		750 ft		1000 ft		2000 ft		3000 ft		5000 ft	
	A	B	A	B	A	B	A	B	A	B	A	B
500											0.025	411
480									0.03	30		
460												
440												
420									0.035	25		
400							0.035	30			0.02	191
380												
360							0.03	30	0.03	35		
340												
320							0.03	40				
300									0.03	110	0.03	200
280							0.045	40				
260												
240							0.03	100	0.03	80		
220												
200					0.04	78	0.02	45			0.03	126
180	0.06				0.04	163			0.03	120		
160	0.04		0.05	145	0.04	90	0.04	40				
140	0.07		0.04	150	0.05	160						
120	0.04		0.06	45	0.04	59	0.035	41	0.035	150		
100	0.04		0.07	30	0.05	128					0.045	81
80	0.08		0.07	130	0.06	488	0.04	110				
60	0.1	345	0.09	520	0.12	1038			0.035	450		
40	0.2	2248	0.16	1500	0.06	2943	0.05	390				
20	0.4	7350	0.6	4380	0.3	3780						
Center	0.6	8608	0.7	4260	0.5	3840	0.07	870	0.04	400	0.05	115
20	0.5	2220	0.6	4260	0.5	580						
40	0.4	531	0.16	440	0.1	426	0.065	470				
60	0.1	458	0.09	220	0.09	301			0.05	300		
80	0.04	74	0.2	70	0.06	129	0.1	430				
100	0.2	152	0.11	15	0.04	75					0.015	117
120	0.04		0.07	40	0.05	14	0.04	220	0.045	740		
140			0.07	100	0.05	158						
160			9.0	110	0.06	31	0.04	100				
180					0.04	71			0.08	220		
200					0.04	73	0.035	60			0.03	77
220												
240							0.04	90	0.04	90		
260												
280							0.035	80				
300									0.035	50	0.025	61
320							0.035	40				
340												
360							0.03	41	0.03	80		
380												
400							0.03	25			0.03	80
420									0.035	35		
440												
460												
480									0.04	80		
500											0.03	65

Laboratory analyses were done on the air sampler filter papers of Stations 109 and 110. These were the only stations of the sampling network that collected activities sufficient for analysis. The filter papers were 4-in. diameter MSA 2133 type, which had sampled air at about 4 cu ft per minute pulled by Motoair pumps. Station 110 also had a sticky-paper fall-out tray.

Gamma spectra of the filter papers and the fall-out paper are presented in Reference 103. The spectra of the filter papers from Stations 109 and 110 were identical, showing peaks due to I^{131} , Ru^{103} , Ru^{106} , and $Cs^{137}-Ba^{137}$. The fall-out paper showed peaks due to these isotopes, plus peaks at 0.16 Mev and 0.76 Mev. In addition, filter paper samples from the NRF facility showed $Cs^{137}-Ba^{137}$, and Ru^{103} , Ru^{106} as the principal gamma emitters. Thus the time of collection and the spectra indicate that the initial puff from Phase B was detected by the NRF air monitors.

TABLE 5—ANALYSIS OF FILTER PAPERS, PHASE B

Constituents analyzed	Activity on total paper (dis/min)		Per cent of total activity	
	Station 109	Station 110	Station 109	Station 110
I^{131}	-	-	1-3*	1-3*
$Ru^{103,106}$	3.2×10^7	8×10^6	18	17
$Cs-Ba^{137}$	4×10^7	1×10^7	22	21
Sr, Ba fraction	4×10^6	-	2	-
Rare earths	2.5×10^6	-	1-2	-
Alpha activity	$< 8 \times 10^2$	8×10^2	0	-
Unidentified	9.6×10^7	-	53	-
Total activity	1.8×10^8	4.7×10^7	100	100

* Estimated from gamma spectra.

Samples of the filter papers were analyzed by radiochemical procedures, and the fall-out paper was saved for autoradiographs and particle-size studies. After appropriate chemical separations, beta end-window counting, and corrections for counting efficiency and aliquot factor, the analyses were found as given in Table 5. The principal activities were from ruthenium and $Cs^{137}-Ba^{137}$, each on the order of 20 per cent of the beta activity. Small percentages of I^{131} , strontium-barium, and rare earth activity were present, with about 53 per cent of the activity remaining unidentified. The total activities on the filter papers were 4.7×10^7 disintegrations per minute at Station 11 and 1.8×10^8 disintegrations per minute at Station 109.

By referring to Tables 4 and 5, the maximum amount of fission products released is estimated to be about 10 per cent. However, as mentioned above, gamma radiation survey readings indicate that a much smaller fraction than this was released, probably much less than 10 per cent.

The material collected at Stations 109 and 110 was extremely fine—no particles could be detected. On the 300-ft arc, the ecologists found deposits that were either extremely fine particles or gases. At 500 ft, large spots were found indicating that some particulate matter had deposited.

6. *Animal and Vegetation Samples.*¹⁰⁶ Dutch rabbits were located along arcs at 300, 500, 1000, and 2000 ft from Ground Zero. Only the outermost specimens on the northwest edge of the sampling sector showed significant activity. These were the specimens 100 ft northwest of the 55° radial on the 300-ft arc; 175 ft northwest on the 500-ft arc; 330 ft northwest on the 1000-ft arc; and 560 ft northwest on the 2000-ft arc. The count rates as measured in a well-type scintillation counter are given for several total organs of these animals in Table 6.

The samples adjacent to each of the above showed less than 10 per cent of these activities, and were at background along the 2000-ft arc.

Six wild jack rabbits collected about 2000 ft downwind of the test area 36 hr after the burning showed small amounts of contamination. Stomach contents of two were 2000 counts per minute and 7000 counts per minute—compared with a background of about 1600 counts per minute. The fur of one had been increased from a background of about 500 counts per minute

to 1100 counts per minute; the fur of another contained 14,000 counts per minute after the experiment. The lungs of one had increased from 500 counts per minute background to 1600 counts per minute. In no case did muscle or liver show any increase.

Radioautographs of sagebrush leaves collected after Phase B show traces of radioactivity at the 300- and 500-ft arcs. The activity on the leaves at the 300-ft arc seemed generally diffused over the entire leaf surface, while that at 500 ft occurred as single hot spots.

TABLE 6—ACTIVITY IN RABBITS, PHASE B

	Activity in total counts per minute			
	300-ft arc 100 ft NW of 55°	500-ft arc 175 ft NW of 55°	1000-ft arc 330 ft NW of 55°	2000-ft arc 560 ft NW of 55°
Fur of scalps and ears	73,000	45,000	26,000	400, slightly above background
Lungs	4,700	4,400	4,600	Background

CONCLUSIONS

The Phase A experiment showed that under conditions approximating the greatest possible heating of a stainless steel clad fuel element in an airplane crash, only a small percentage of the 5000-curie fission products (probably less than a few tenths per cent) was released to the environment. Most of this activity was deposited in a small (24-sq ft) area near the test site. Very little activity was detected outside of this area, and none beyond 5000 ft.

In the Phase B experiment, a 10,000-curie fuel element was completely melted down within 1½ minutes, reaching temperatures around 5000°F. The fractional activity released was estimated from field measurements and filter paper analyses to be a maximum of 10 per cent, probably much less. About 40 per cent of the activity found on air filters was Ru¹⁰³, Ru¹⁰⁶, and Cs¹³⁷—Ba¹³⁷. About 1 to 3 per cent was I¹³¹, and 2 per cent or less was in the Sr—Ba fraction.* Most of the activity deposited was within 300 ft, and practically all was within 2000 ft of the burning site.

Although all of the data hoped for were not obtained, the major objectives of these experiments were accomplished—to demonstrate that fractional release of fission products from fuel elements subjected to fire would probably be much less than some previous hazard estimates had assumed. A convenient summary of other measurements of fission product release from nuclear fuels is given in Ref. 107. Other information is available in unclassified Refs. 115 to 126 and 177.

The results of fission product release experiments to date,† as well as the field tests described here, are instructive. Further full-scale field tests, as well as laboratory experiments, are needed to better define the probabilities and conditions of fission product release from fuel elements subjected to heat and destruction.

*EDITORS'NOTE: A recent report¹⁷⁷ concludes that 30 per cent of the cesium and about 1 per cent of the strontium was released in a meltdown experiment. Iodine was found to be the only aerosol that had more than 5 per cent penetration through pleated filters.

†April 1959.

HEALTH PHYSICS ASPECTS OF THE BORAX-1 DESTRUCTION

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Argonne National Laboratory, Idaho Division

This presentation includes collected data gathered jointly and individually by the Argonne National Laboratory and the Health and Safety Division of Idaho Operations Office, AEC.

The BORAX-1 reactor was operated at the Argonne National Laboratory site at the National Reactor Testing Station near Arco, Idaho. The reactor was located 2730 ft northwest of the Experimental Breeder Reactor (EBR) area entrance. It was operated remotely from a control point located approximately one-half mile southeast of the reactor near the EBR area entrance.

The reactor installation consisted of a reactor tank, 4 ft in diameter, inside the shield tank of 10-ft diameter. This was sunk partly into the ground, with earth piled around the installation to provide additional shielding. The reactor core consisted of a number of MTR-type fuel elements, each of which contained 18 fuel plates. The combined U^{235} content was about 140 g per element.

During the year 1953, experimental excursions were produced in the following manner: The reactor was equipped with a number of control rods, one of which could be rapidly ejected downward out of the reactor core by springs and gravity. The reactor was made critical at low power with this rod inserted at a pre-chosen point in the core and the remaining rods fully withdrawn. The control rod was then suddenly ejected and the reactor power allowed to rise rapidly. The reactor would produce sufficient steam to over-compensate for the excess reactivity and to reduce the power level. The remaining control rods were then programmed to be dropped into the core after a preset time interval. These excursions were routinely performed with periods as short as 5 milliseconds. Under these conditions, the ejections of water from the reactor were rather violent, and water was thrown out of the reactor tank to a height of about 30 ft. The decision was then made to make one final short excursion that would result in a possible partial melting of fuel elements with a release of fission products. This experiment was performed on July 22, 1954.

Generally speaking, the radiological aspects were to be taken care of by the previously-mentioned AEC group. With this thought in mind the following operations were prepared and put in readiness for the experiment:

1. Regular emergency monitoring equipment was supplemented with additional instruments and supplies; this included stationary and mobile equipment. All vehicles were previously equipped with two-way radios. A transmitter was installed in the field office.
2. The emergency monitoring teams consisted of five two-man teams with instruments and radio-equipped vehicles. These teams were shifted according to U. S. Weather Bureau recommendations.
3. Mobile smoke generators were located approximately one mile from the reactor installation, were operated in conjunction with remotely-detonated smoke pots in the reactor area, and were fired by ANL personnel.
4. Central maintenance crews were on stand-by with gravel trucks and loading equipment prepared to deliver and spread gravel over roads and highways, if needed. AEC security was on stand-by to effect roadblocks and evacuation, as it turned out to be required.

Fall-out plates and beta-gamma and neutron film were placed in the general downwind direction from the reactor, as well as within the reactor area proper. This was in addition to the regular site monitoring system. All observers in the control area were issued film badges.

Shortly before 0800, the U. S. Weather Bureau notified ANL officials that conditions were favorable for beginning the experiment. This information was relayed to observers and health physics teams.

The reactor was then made critical, and the central control rod was fully inserted. After a final check to determine that no personnel were in the restricted area, one of the remote smoke pots was fired. Ten seconds later, at 0820, the central rod was ejected from its fully inserted position.

In a very short time after the ejection of the rod, a column of what appeared to be dark gray smoke was ejected from the reactor to a height of approximately 80 ft. Shortly thereafter, the sound of a medium sharp detonation reached the control trailer and a slight tremor was felt. In a few seconds the smoke cleared away and it was apparent that the entire superstructure of the reactor had been carried away.

Monitors at the control point (approximately one-half mile away) recorded an instantaneous exposure rate in excess of 500 mr/hr. This was indicated on a cutie-pie type ionization chamber. Within 30 sec the rate was established at 25 mr/hr and was rapidly declining. Within 5 minutes, exposure rates had decreased to less than 1 mr/hr.

Construction personnel working in the vicinity of the present Zero Power Reactor (ZPR), which is about 0.6 mile from the BORAX reactor, were immediately evacuated pending an evaluation of the situation. At the same time, traffic control was established on U. S. Highway 20 and on Van Buren Avenue leading to the EBR.

It was soon evident that no significant exposure had been received at the control point. With a northeast wind prevailing, the smoke from the smoke pots continued to rise to a considerable height before becoming highly diffused. At 0936, after determining that the exposure rates on Van Buren Avenue and the EBR area were less than 0.5 mr/hr, restrictions were lifted to allow traffic and construction personnel to continue normal operations. Subsequent examination of film badges worn by observers at the control point gave no evidence of exposure received at this location.

Monitoring teams that were dispatched shortly after the excursion had by 0900 determined the trajectory to have been roughly in a southwesterly direction, passing over Adams Boulevard approximately $1\frac{1}{2}$ miles from the Van Buren-Adams junction and continuing over the Union Pacific Railway in the same direction. Within 45 min after the excursion, the bulk of the airborne material had passed beyond the railroad right-of-way and the initial fall-out surveys were completed. The following readings of the significant trajectory fall-out summarize the effect of the cloud. Readings were taken in a standard manner, that is, in air at 3 ft above ground.

Across Adams Blvd.	0835	5 mr/hr
Across Union Pacific Railroad	0845	2 mr/hr

The following readings were taken at 1 in. above ground:

Across Adams Blvd.	0915	6 mr/hr
Across Adams Blvd.	1430	0.5 mr/hr
Across Union Pacific Railroad	0920	6 mr/hr
Across Adams Blvd.	1030(July 23)	0.05 mr/hr

After approximately 1 hr, it had been determined that no widespread hazard existed and a cooperative maximum effort was exerted to salvage film and records from the reactor area.

The first attempt to enter through the main gate failed when personnel encountered a field of 40 r/hr at approximately 100 ft from the gate. However, at 0930 an entry was made by cutting through the rear fence and utilizing existing earth shielding. Film were successfully removed while personnel were exposed to a maximum dose rate of 1 r/hr. The following read-

ings were taken to establish emergency time limits for the removal of essential material. All readings were taken at 3 ft above ground:

75 ft NE of reactor	0930	0.1 r/hr
75 ft SE of reactor	0930	0.8 r/hr
60 ft SE of reactor	0930	1.5 r/hr
West end of reactor pit	1400	10.0 r/hr
Over reactor tank hole	1400	8.0 r/hr

Further investigation showed that an area 350×200 ft and lying south of the reactor area fence had been covered with radioactive debris. On August 2 the area was divided by a grid and measured. A number of the side plates of fuel elements were found holding clusters of end fragments of fuel plates, the remaining plates having been melted or torn away. Some elements were found with no fuel plates remaining.

RESULTS

The positioning of the fall-out plates did not coincide with the trajectory center, so worthwhile information was not obtained from the trajectory plates. The plates located within the blowout area were grossly contaminated and could not be used as representative fall-out samples. The trajectory plates were all radioautographed. No significant increase in activity was noted in the locations of these plates.

Film Packets

Film packets placed inside the reactor were highly overexposed and interpretations could not be made. A number of film packets placed in conjunction with the outer area fall-out plates gave the following results:

Distance from the reactor, ft	mr
1450	50
1500	50
1900	30
2250	20
5300	0

Since the fall-out plates in the same locations showed no contamination, it is safe to assume that this recorded exposure resulted from the initial release.

Analysis for Alpha

Over 50 samples were analyzed for alpha activity. These included filter paper samples, samples from the continuous air monitors located on the site, and gummed paper fall-out samples. No unnatural alpha activity of any significance was detected in any of the samples tested.

Analysis for Beta Activity

A section of plastic shoe cover worn in the BORAX area a few hours after the excursion was counted and a decay curve was developed.

Cleanup and Salvage

On July 26, an attempt was made to reduce the radiation background by physically removing all visible pieces of radioactive material that had been scattered south of the reactor. Many such fragments, some reading as high as 50 r/hr, were collected; however, their removal did not reduce the background by very much. It was then necessary to rope off an area of approximately 11,000 sq ft. Within this roped-off area another area of approximately 5,000

sq ft was covered with a 6-in. depth of gravel. Dosage rates were reduced to less than 5 mr/hr over the entire area, compared to rates of 500 to 800 mr/hr before laying gravel.

The salvage operations were conducted jointly by AEC and ANL. Decontamination facilities were set up at the reactor site and much of the material was successfully decontaminated. Exposure rates ranged up to 5 r/hr. Salvaged equipment and materials were utilized in the construction of BORAX-2 at a nearby site.

In conclusion, it can be safely assumed that at a distance of one-half mile there was no evident hazard. The airborne activity could have presented a hazard to anyone in the direct trajectory, but would not have caused chronic exposure from long-lived isotopes. Therefore the radiological hazard that we should be primarily concerned with is obtained in the immediate area affected by the blowout material.

1944-1945

APPENDIXES

Appendix A

RADIOACTIVE CONTAMINATION OF THE ENVIRONMENT FROM NORMAL REACTOR OPERATIONS*

By G. VICTOR BEARD

The number of disintegrations per second per fission for U^{235} , U^{233} , U^{238} , or Pu^{239} atoms is given in Eq. 1.

$$\frac{dn}{dt} = 3.9 \times 10^{-6} t^{-1.2} \text{ dis/sec/fission}, \quad (1)$$

where t is in days. This value of dn/dt is probably accurate within a factor of two. (See also Ref. 174.)

Equation 2 gives the number of curies after time λ (in days after reactor shutdown), where T (days) is the time that the fuel element has been in the pile and P is the average power level of the pile in watts.

$$C = 1.5 P \left[\frac{1}{(\lambda - T)^{0.2}} - \frac{1}{\lambda^{0.2}} \right] \quad (2)$$

Equations 1 and 2 are valid in an interval of about 15 seconds to 5 years.

The number of atoms disintegrated per unit time may be expressed by Eq. 3, when these atoms are being formed at a constant rate and disintegrating at a rate proportional to the number present.

$$\frac{dn}{dt} = R - \lambda n \quad (3)$$

When dn/dt equals zero, R is equal to λn which means that the rate of formation is equal to the rate of decay. This is the steady state. This equilibrium condition will not be attained for any given isotope until a time of about 6 half lives has elapsed for that isotope. In the case of Sr^{90} this is about 150 years; in the case of material with 5 days half life this is 30 days.

Chemical processing plants now release the Kr^{85} (11-year half life) and Xe^{133} (6-day half life) to the atmosphere. If we were to assume 100 reactors, each operating at a power level of 500 Mw (a commonly projected estimate for the year 1975), the released Kr^{85} and Xe^{133} to the atmosphere, uniformly distributed, would result in a concentration such that an individual immersed in an infinite cloud of the material would receive approximately 0.006 rad per year skin dose from the Kr^{85} beta radiation. This would not constitute a significant health hazard, but would present an instrumentation background problem.

* Presented at the 65th meeting of the Advisory Committee for Biology and Medicine, November 9, 1957.

It should be pointed out that the value of 0.006 rad from Kr^{85} was derived from Eq. 3 assuming that a steady-state condition would exist if the 100 reactors were in operation continuously and the entire Kr^{85} production were discharged continuously from the associated chemical processing plants to the world's atmosphere.

In Eq. 4, n is the number of neutrons present at time t , and n_0 is the number of neutrons present at $t = 0$ in a reactor after a step-function change in reactivity:

$$n = n_0 e^{\Delta K t / L}, \quad (4)$$

where ΔK is the given amount of excess reactivity in the reactor, and L is the lifetime of a generation of neutrons.

Excess reactivity is the amount of reactivity above that necessary to maintain a cold reactor in a critical condition. The symbol L represents the average time in seconds from one neutron generation to the next: L is of the order of 0.1 second for thermal, water-moderated reactors operating on delayed neutrons; L is of the order of 0.001 second for thermal, water-moderated reactors operating on prompt neutrons; and L is of the order of 10^{-8} second for fast, water-moderated reactors.

All reactors are built in such a manner that ΔK is zero for the reactor when the shim rods are not inserted, the excess ΔK being always contained in the shim rods. Reactors are always built in such a manner that the shim rods cannot accidentally fall in. They are driven into the reactor by mechanisms which can be controlled but whose rate of drive cannot exceed a certain preselected value. Consequently, the existing reactor ΔK can be controlled precisely. In this way it is virtually impossible for a reactor to have a period that does not depend upon the delayed neutrons, i.e., $L = 0.1$ second. In normal operations the reactor period is of the order of minutes. This means that it takes several hours for a reactor to go from virtually zero power to full power.

When the reactor is at full power, the available ΔK is virtually zero owing to the fact that reactors lose reactivity as they heat up. This means that a reactor operating at full power cannot have a transient or reactor period that is very fast. It is possible to get fast reactor periods only when the reactor is virtually at zero power and cold. Consequently the most critical time in any reactor operation is the start-up period. However, since reactors are designed so that the rate at which ΔK can be added is very slow, it is virtually impossible even during start-up periods to get the reactor power up in so short a period that it can explode. Where attempts have been made to insert the shim rods by violent means (springs, explosions, etc.), starting at zero power, it has proved to be exceedingly difficult to destroy the reactor. This type of procedure (i.e., inserting the shim rods at zero power by forceful mechanical means) did result in the destruction of BORAX-1.* The energy released in this reactor excursion was calculated to be equivalent to about 10 lb of dynamite. In the many attempts at the SPERT reactor to duplicate the BORAX experiment, only water has been boiled out.

In Eq. 5, W is the work obtainable, Q is the heat energy available, and T_2 is the temperature at which the heat is produced:

$$W = Q \frac{T_2 - T_1}{T_2} \quad (5)$$

T_1 is the temperature at which the heat is rejected. This equation is a statement of the second law of thermodynamics. This equation states that heat can be converted into work completely only in case T_1 is at absolute zero (Kelvin) temperature. If we were to produce heat energy at 546°C and were able to take heat out down to the freezing point of water, 273°K , we would be able to convert 50 per cent of available heat into work. In actual current practice, operating under the highest practical temperatures and rejecting heat under the best practicable conditions, the amount of heat converted into work is somewhere between 25 and 30 per cent. The

* EDITORS' NOTE: The radiological results of the BORAX-1 experiment are described in the final paper of the training lectures presented in this document.

over-all mechanical efficiency is about 80 per cent. This means, roughly, that under current practices approximately one quarter of the heat generated in the reactor is being converted to useful work. The other three quarters are lost as heat. Reactor engineers are constantly striving to find cladding material for fuel elements which will permit a higher temperature, T_2 . In pursuing this line of research one usually heats the fuel element to the point at which the reactor cladding fails in order to actually test the efficiency of the cladding. However, if the cladding fails either in a water-cooled, a liquid-metal-cooled, or an air-cooled reactor, cooled fission products usually escape into the cooling medium.

In the case of water-cooled reactors, extensive experience at MTR and at the STR has shown that fission products stay largely in the water and practically none are released to the air. The cooling water can then be decontaminated by ion exchange resins. In the case of air-cooled reactors, the fission product material stays in the air streams. Present filter material seems to be adequate for removing the fission products from the air stream for all elements except iodine. Most experimental reactors, which deliberately push temperatures to the melting point of the cladding, have been located in isolated areas and it has been possible to control the iodine problem without the use of costly filters.

The significant hazards due to ruptured fuel elements caused by operating at very high temperatures will for the time being be limited to experimental reactors. None of the power-producing reactors will be approved to operate at temperatures high enough to cause significant hazards resulting from fuel element rupture. If this does happen, however, most of the released fission products will be trapped in the primary water coolant. In the case of the British gas-cooled reactor of the Calder Hall type, the cooling gas (CO_2) will remain in a closed circuit and thereby keep the released fission products under control.

TABLE A.1—NUMBER OF INDIVIDUALS PER YEAR ESTIMATED TO RECEIVE
VARIOUS EXPOSURES FROM REACTOR ACCIDENTS

Exposure level	Power reactors		Experimental reactors		Military reactors	
	Controlled areas	Uncontrolled areas	Controlled areas	Uncontrolled areas	Controlled areas	Uncontrolled areas
0.5 rem per year	<5,000	0	<5,000	0	?	<10,000
15 rem per year	<100	0	<100	<1	<1,000	<100
Lethal dose	<1	0	<10	0	<100*	<1

* Including crash deaths.

The Windscale incident occurred in a plutonium-production reactor designed approximately eight years ago. This reactor was not designed to utilize the heat energy produced by the reactor. It was using graphite as a moderator and air as a cooling agent. The graphite temperature was kept so low by design that considerable Wigner energy was stored in the graphite. In attempting to release this energy, fuel element capsules were ruptured, releasing radioactive material directly to the air stream. This is detailed in a paper entitled "Accident at Windscale No. 1 Pile on 10th October, 1957," published by Her Majesty's Stationery Office.

If we assume that there are 100 reactors, each operating at 500 Mw power level at some future date (say 1975), calculation plus estimation indicate that the potential annual radiation hazards will be about as shown in column 1 of Table A.1. Column 2 shows the hazard level calculated for current experimental reactors. The figures given in column 3 are partly based upon estimates furnished by the Strategic Air Command based upon the use of one wing (30 planes) operating on nuclear power within three to five years and assuming an accident rate of two crashes a year.

Appendix B

INFORMATION FOR USE IN EMERGENCIES (OR IN EMERGENCY PLANNING)

1 AERIAL SURVEY SERVICES, CONTACTS

- (1) Chief, Airborne Survey Section, Geophysics Branch, U. S. Geological Survey, Department of the Interior, Washington 25, D. C.
- (2) Kirtland Air Force Base, Albuquerque, New Mexico.
- (3) The Heinen Brothers, Dallas, Texas.
- (4) Exploration Branch, Division of Raw Materials, USAEC, Washington, D. C.
- (5) Division of Biology and Medicine, USAEC, Washington, D. C.

2 ALPHA PARTICLES

Range

The range in air of an alpha particle is

$$R = 0.318E^{3/2},$$

where R is the range in centimeters of air at 1 atmosphere and 15°C, and E is the energy in Mev.

Self-absorption

The fraction of alphas emitted in the outward 2π solid angle that is absorbed is:

$$f_a = \frac{T}{2R},$$

where T is the thickness of the deposit and R is the range of the alpha particle in the same units as T. The range of U^{238} alpha particles in U_3O_8 is 1.2×10^{-3} cm.

3 ATMOSPHERIC DIFFUSION

Some useful equations from AECU-3066 are: For isotropic diffusion,* the distance downwind for maximum concentration, and the maximum concentrations are, in consistent units,

* EDITORS' NOTE: A recent report¹⁷⁷ concludes that anisotropic diffusion parameters are needed to more adequately predict airborne concentrations.

Instantaneous point source

$$d_{\max} = \left(\frac{2h^2}{3C^2} \right)^{1/(2-n)}$$

$$\text{Maximum concentration at } d_{\max} = \frac{2Q}{(2e\pi/3)^{3/2} h^3}$$

Continuous point source

$$d_{\max} = \left(\frac{h^2}{C^2} \right)^{1/(2-n)}$$

$$\text{Maximum concentration at } d_{\max} = \frac{2Q}{e\pi \bar{u} h^2}$$

Integrated dose from instantaneous point source

$$d_{\max} = \left(\frac{h^2}{C^2} \right)^{1/(2-n)}$$

$$\text{Total integrated dose at } d_{\max} = \frac{2Q}{\pi e \bar{u} h^2}$$

(Note that the integrated dose from an instantaneous source is identical in form to the concentration from a continuous source.)

In the above equations,

Q = source strength (curies) or emission rate (curies/sec)

C = generalized diffusion coefficient (meters^{n/2}) for isotropic turbulence

n = nondimensional parameter associated with stability

h = height of source, meters

e = 2.718

$\bar{u}$ = mean wind velocity, meters per second

For representative parameters n and C, see Table B.1.

TABLE B.1—C² (metersⁿ) AS A FUNCTION OF HEIGHT (h) AND STABILITY PARAMETER (n)*

Conditions	C ²				
	n	25 meters	50 meters	75 meters	100 meters
Large lapse rate	0.20	0.043	0.030	0.024	0.015
Zero small temp. gradient	0.25	0.014	0.010	0.008	0.005
Moderate inversion	0.33	0.006	0.004	0.003	0.002
Large inversion	0.50	0.004	0.003	0.002	0.001

*Material taken from report AECU-3066.

4 BACKGROUND RADIATION (Average)

- (1) 4.3 r to age 30 from natural background.
- (2) 3.0 r from medical X-rays (0.04-1 r per single chest picture; 10-20 r/min in fluoroscopy).
- (3) 0.1 r to age 30 from weapons test fall-out at present rate of testing (1957).

5 BETA-RANGE

$$X = \frac{1}{\rho} [0.54E_m - 0.13(1 - e^{-3.2E_m})],$$

where X = range in cm of medium having density ρ in gm/cm³, and E_m is the maximum energy of the beta ray in Mev.

6 BIOLOGICAL EFFECTS OF RADIATION^{30,32,58,71-80,92}

Acute LD₅₀

The estimates (80, others) of acute whole-body X- or gamma-ray dose that will result in the mortality within 30 days of 50 per cent of those exposed fall within the ranges:

"larger" adult mammals - 250 to 650 rem

"smaller" adult mammals - 400 to 800 rem

The LD₅₀ for man has been estimated to be in the 250 to 650 r dose range.

In the embryonic development of certain mammals, visible damage has occurred at doses as low as 25 r.

Shortening of Life Span⁸⁰

The shortening of life span is estimated at 1 to 4 days per roentgen. This is not agreed on by all authorities. The extent to which repeated small exposure to X-rays will shorten life expectancy is not yet well established. There has been some indication of a shortening of the life span of radiologists relative to that of other medical specialists. This is not definitely proved. It is known, however, that the incidence of leukemia is increased among radiologists.

Experiments have indicated a life shortening in mice and rats of 10 per cent following a single whole-body exposure of 25 per cent of the LD₅₀. This percentage has not been proved to be linear with dose. There have been some experiments, at low repeated dose levels (less than 1 rem per day), where an increased survival time of the exposed animals has been indicated.

Cancer

If the assumption is made that natural background radiation is responsible for the spontaneous incidence of tumors among the general population, the frequency of tumor production might be about 4 to 8 cases per million individuals exposed to one rem in 70 years. It has not been proved that any increase at all would be obtained from such an exposure.

Based upon statistical studies of persons occupationally exposed to radiation, leukemia incidence has been estimated to be 1.5×10^{-6} per person per roentgen per year as an upper limit, with zero as a lower limit at small dose rates.

Cataracts

More than 1000 rads required for X or γ ; more than 100 rads for neutrons.

Genetic Effects

Estimates of the magnitude of genetic effects are not completely agreed upon. The following statement taken from Ref. 92 is quoted here only to provide an immediate idea of the possible magnitude of the genetic effects of an incident exposing many people: "... if 10,000 individuals were exposed to ... (on the order of 200 roentgens) then there would be perhaps 100 handicapped children of these exposed individuals, in addition to the number resulting from other causes."

7 CLOUD DOSE

The external dose at the center of an infinite cloud of activity is

$$\begin{aligned}\text{Rads/day} &= 50 \times \text{microcuries/gram of air} \times \text{energy per disintegration in Mev} \\ &= 4.0 \times 10^4 \times (\text{microcuries/ml}) \times (\text{Mev/dis})\end{aligned}$$

8 CONSTANTS AND CONVERSION FACTORS

$$\begin{aligned}1 \text{ curie} &= 3.7 \times 10^{10} \text{ dis/sec} \\ 1 \text{ r} &= \text{esu}/0.001293 \text{ g of air} \\ &= 6.77 \times 10^{10} \text{ Mev/m}^3 \text{ (air at } 0^\circ\text{C and 760 mm Hg)} \\ &= 83.8 \text{ ergs/g air} \\ 1 \text{ mile} &= 1.6093 \text{ kilometers} = 5280 \text{ ft} \\ 1 \text{ ft} &= 0.3048 \text{ meter} \\ 1 \text{ meter} &= 6.214 \times 10^{-4} \text{ mile} \\ 1 \text{ mile/hour} &= 0.4470 \text{ meter/sec} \\ 1 \text{ Btu} &= 252 \text{ calories} = 1,054 \text{ joules} \\ 1 \text{ cfm} &= 4.72 \times 10^{-4} \text{ m}^3/\text{sec} \\ 1 \text{ g} &= 0.0022 \text{ lb} \\ 1 \text{ g/cm}^3 &= 62.43 \text{ lb/cu ft} \\ 1 \text{ liter} &= 0.0352 \text{ cu ft} \\ 1 \text{ Mw-sec} &= 10^6 \text{ joules} = 2.5 \times 10^5 \text{ calories} \\ &= 10^{13} \text{ ergs} \\ &= 6.24 \times 10^{18} \text{ Mev} = 3.25 \times 10^{16} \text{ fissions} \equiv 2.5 \times 10^{-7} \text{ kilotons TNT-equivalent} \\ &\quad \text{energy release.}\end{aligned}$$

Fission product decay energy, β and $\gamma = 3.2t^{-1.21}$ Mev/sec/fission

Infinite cloud dose, $1 \text{ curie/m}^3 = 0.547E \text{ roentgens/sec}$, with E in Mev/dis.

9 CONTAMINATION

Beta Surface Dose Rates from U and Pu

$\text{U}^{238} \sim 250 \text{ mrep/hour}$

$\text{Pu}^{239} \sim 900 \text{ mrep/hour}$

(Beta-gamma readings up to about 100 mr/hr may be expected from unfissioned fissionable material spread over a surface.)

TABLE B.2

Nuclide	Half life	Millicuries/ft ³ of water
N ¹⁶	7.3 sec	$1.63 \times 10^3/\text{M watt}$
Na ²⁴	14.9 hour	$0.36/\text{M watt}$
Mg ²⁷	9.6 min	$0.50/\text{M watt}$
Al ²⁸	2.3 min	$1.63/\text{M watt}$

Cooling Water Induced Activities (Oak Ridge Research Reactor)

Table B.2 is not necessarily representative of other reactors, but it gives an idea of the order of magnitude of cooling water activities that might be expected.

TABLE B.3—EMERGENCY FOOD AND WATER LEVELS^{56,57}

Time duration of ingestion	Microcuries per cm ³	dis/min/m ³
Preferable 10 days	3.5×10^{-3}	7.7×10^3
Acceptable 10 days	9×10^{-2}	2×10^5
Preferable 30 days	1.1×10^{-3}	2.6×10^3
Acceptable 30 days	3×10^{-2}	7×10^4

Resuspension factor. $\frac{\text{dis per min per cubic meter}}{\text{dis per min per 100 cm}^2} = 10^{-2} \text{ to } 10^{-4} \text{ for removable activity.}$

Thus the values of removable activity per 100 cm² below which no immediate inhalation or ingestion hazard exists can be obtained by multiplying the maximum permissible concentrations (MPC) in air in dis/min/m³ by 100. For fresh fission products spread on the ground, however, the internal dose to any organ is usually a factor of 1/100 below the external whole-body dose received from staying in the area an appreciable time.

Food Crops. Acceptable emergency limit is about 0.1 millicurie per square foot, at 24 hr after a nuclear explosion (short build-up time), which is equivalent to about 500 mr/hr at H + 1 hr. (Or one megacurie-equivalent of gamma activity per square mile yields about 4.1 r per hour at 3 ft above ground.)

Iodine

It was found after the Windscale incident that iodine deposited on grass was quickly carried into milk. There was no International Commission on Radiological Protection (ICRP) level for iodine in milk but an emergency level of 0.39 microcurie per liter had been suggested by Scott Russell in ARC/RBC Report No. 5, based on ingestion by infants. An emergency committee decided on a limit of 0.1 microcurie/liter of milk, based on a dose of 20 rads to an infant thyroid of 1.5 g. It was found that a grazing area contamination level of 1 $\mu\text{C}/\text{m}^2$ on grass $\rightarrow$ 0.1 $\mu\text{C}/\text{liter}$ of milk.

This amount of iodine spread over a large area would produce a gamma dose rate of about 3 microroentgens per hour near ground level. These levels are detectable* above normal background with a sodium iodide detector biased at 50 kev. The highest amount of thyroid iodine found in a local inhabitant after the Windscale incident was 0.28 μC . (See also Chadwick, AERE-HP/M-24).

Assuming a 30 g adult thyroid, the dose in rads to the thyroid is approximately equal to the number of microcuries of I¹³¹ ingested. (See ORO Report 125, September 1953).

Plutonium (Spec. Act. = 1.4×10^5 dis/min per microgram)¹

Controlled Area: Greater than 35 $\mu\text{g}/\text{m}^2$

Primary Danger: Greater than 3500 $\mu\text{g}/\text{m}^2$
Greater than 1 r/hr of gamma

Air Force Zones: (See also Reference 138.)

0 to 100 $\mu\text{g}/\text{m}^2$ —releasable to controlled area for continuous residence.

1000 $\mu\text{g}/\text{m}^2$ —safe for continuous occupancy

4000 $\mu\text{g}/\text{m}^2$ —safe for 8 days or so

5000 $\mu\text{g}/\text{m}^2$ —safe for 3 to 4 days

*See Editors' Note after article by F. J. Davis, "Instrumentation in Aircraft for Radiation Measurements."

Juno Sensitivity:

350 alphas per minute through 3 in. $\times$ 4.2 in. sensitive area of window indicates 1 microgram of Pu per square meter of rough ground, or, approximately,
Reading in mr/hr $\times$ 2 = micrograms Pu/m² on broomed concrete
mr/hr $\times$ 2 $\times$ 1.4 = micrograms Pu/m² on gravel or rough ground
mr/hr $\times$ 2 $\times$ 0.8 = micrograms Pu/m² on smooth concrete

AN/PDR-10 Alpha Survey Meter Sensitivity:

At $\frac{1}{2}$ in. from surface, 5000 c/min = 100 micrograms Pu/m²

Air Concentrations: A concentration of 100 μ g/m² of loose Pu contamination on the surface may be assumed to produce an air concentration on the order of 100 dis/min/m³.

Long-term Isotopes from Fall-out

The principal isotopes remaining in the Marshall Islands two years after the Operation Castle detonation were Sr⁹⁰, Cs¹³⁷, and Ru¹⁰⁶. Some potable water was still above the MPC for Sr⁹⁰.

10 CONVERSIONS FOR CALCULATING RADIATION INTENSITIES

Gamma Dose

(a) $1 \text{ r/hr} = \frac{5.6 \times 10^5}{E} = \text{photons/cm}^2/\text{sec}$, for E from 0.07 to 2.0 Mev, good within ± 12 per cent, assuming $W_{\text{air}} = 34 \text{ ev/ion pair}$.

The above identity is equivalent to the equation:

$$I = \frac{5.64C \sum_i E_i}{D^2}$$

where I = intensity in r/hr at D ft from a point source

C = curies of activity

$\sum_i E_i$ = sum of gamma energies in Mev per disintegration, if more than one gamma is emitted per disintegration

D = distance in feet

(b) For any energy E in Mev, and its corresponding total minus Compton Scatter absorption coefficient - $(\mu - \sigma_a)$,

$$1 \text{ r} = \frac{7.1 \times 10^4}{(\mu - \sigma_a)E} \text{ photons/cm}^2$$

Beta Surface Dose in Air

In air of density 0.001293 g/cc,

$$1 \text{ rad/hour} = \frac{6.1 \times 10^5}{S_a} \text{ betas per cm}^2 \text{ per sec,}$$

where S_a is the average number of ion pairs produced per centimeter of path in air. Accurate to ± 6 per cent from 0.01 to 2.0 Mev. Values of S_a are given in Table B.4.

11 CRITICALITY BURSTS

Criticality bursts of 0.1, 3, and 15 Mw-seconds produce 400 r at 1.5, 9.5, and 22 ft, respectively.* The greatest number of fissions in a criticality incident (up to 1959) was 10^{19} to 10^{20} fissions. There are 3.25×10^{10} fissions per watt-second from U²³⁵.

* Report Y-897.

TABLE B.4—SPECIFIC IONIZATION OF ELECTRONS

E(Mev)	S _a ion pairs/cm	Range in Air	
		(mg/cm ²)	cm
0.05	250	3.9	3.02
0.10	175	14	10.8
0.20	96	42	32.5
0.30	76	77	59.6
0.50	60	158	122
1.00	53	400	310
1.50	47	680	526

12 DECAY AND BUILD-UP

The number of daughter atoms, Q , having a decay constant λ_2 , building up in time t from a number of atoms, P_0 , of a parent having decay constant λ , is given by

$$\text{Activity} = \lambda_2 Q = \frac{\lambda_2 \lambda_1 P_0}{\lambda_2 - \lambda_1} (e^{-\lambda_1 t} - e^{-\lambda_2 t})$$

13 DECONTAMINATION

Personal

If persons become contaminated with radioactive materials, they should shower or wash immediately with mild soap and cold water. Washing should be repeated several times until the activity has been reduced as much as possible. If necessary, wash with synthetic detergent and sequestrant in a ratio of 1:2. Suitable sequestrants are 3 per cent sodium citrate, sodium acetate, or sodium tartrate. The individual should be monitored after each washing, paying particular attention to hairy and secluded parts of the body. Harsh reagents and stiff brushes should be avoided since they may tend to open the skin and allow radioactive materials to enter the body. Other reagents for skin decontamination include detergents and water, special commercial reagents, half-and-half tartaric-citric acid, and citric acid paste.¹³¹ Do not use oxalates. Specific areas of the body may be scrubbed gently with surgical brushes after the first few initial washings have removed most of the loose contamination. Radioactive materials in the nostrils should be removed by repeated nose swabs. Inhaled radioactive particulates should be washed from the mouth, throat, and nasal passages. Where possible, the contaminated wash water and brushes should be retained for disposal by trained radiological personnel. While waiting for medical assistance, measures should be taken to avoid entry of radioactive substances into the body through any open wounds. Names of all persons exposed to contamination should be referred to the physician for possible further examination. For further details, see Refs. 76 and 132.

Contaminated clothing should be isolated and held until trained personnel determine proper disposition. In an emergency, where the levels of contamination on the clothing are not too high, ordinary washing machines and detergents are usually effective in reducing contamination to a safe level. It is better, where possible, to await disposition of contaminated clothing by local health authorities. Care should be taken to avoid mixing contaminated clothing with clean clothing. Further information on clothing decontamination may be found in Refs. 39 and 163.

Buildings, Equipment, and Other Surfaces

The decision to undertake immediate decontamination will depend upon circumstances. One must determine whether the benefits are sufficient to justify any further exposure to personnel that may be involved. Unless emergency measures for the saving of life and health dictate otherwise, decontamination measures should be planned and carried out only by trained

radiological teams. Particularly for short half-life materials, it may be better to wait several generations of radioactive decay before attempting decontamination, even though it has been shown¹⁰⁰ that if a contaminant is allowed to remain on a surface it might be harder to remove later.

For liquid contaminants, it is best to first remove as much contaminant as possible and keep the surface to be decontaminated moist (unless necessary delays prevent this). The surface should then be leached several times with a minimum amount of suitable reagent. For each leaching, the reagent should be left in contact with the surface for several minutes. For emergency purposes, most ordinary household detergents in water will provide decontamination factors of at least 100 on most non-porous surfaces. Complexing agents such as oxalates, carbonates, and citrates are safe and effective. If further decontamination is needed after the first few leachings, wet scrubbing may be tried. Special complexing or chelating agents, such as the salts of ethylenediamine, tetra-acetic acid (EDTA), have been devised for general fission product decontamination. For the specific nuclides P^{32} , Ba^{140} , and I^{131} , the reagents 3N HNO_3 -3N H_3PO_4 , 6N HNO_3 , and 56 per cent HI, respectively, have been found effective on non-porous surfaces.¹⁰⁰ When using liquid decontaminants, care should be taken to prevent the spread of contaminated washings to unwanted areas. Contaminated washwater should be wiped up or collected for proper disposal, if possible. Water-repellent, non-porous surfaces are most easily decontaminated by chemical reagents.

For dry powdered contaminants on porous surfaces, it is more efficient to first try to remove as much contamination as possible by sweeping or vacuuming, provided this will not produce an inhalation hazard or spread contamination to unwanted areas. Appropriate masks or respirators and protective clothing should be provided to decontamination teams using these methods. Also, it is preferable to use vacuum cleaners having exhaust filters or liquid impingers that are efficient enough to retain the radioactive dusts concerned. These methods should be carried out only by trained radiological teams.

Wet methods on porous surfaces such as concrete or transite will probably be effective only on the very loose contaminant on the surface. Strong agents and excessive scrubbing will probably not be effective beyond a certain point, but in fact may tend to wash the contaminant farther into the porous material. Complete decontamination of porous materials, if necessary, can usually be attained only by removing layers of surface by abrasion or erosion. Care should be taken in these measures to protect against the spreading or the inhaling of the material being removed. Surface erosion should be avoided when possible, since this leaves the treated surface more susceptible to future contamination. Coating porous surfaces with varnishes, lacquers, or paints, may increase future decontaminability by three or four orders of magnitude.

Adhesive tapes or strippable paints may also be used to decontaminate surfaces without causing air contamination or the spread of contamination to other areas. Steam jets are effective for removing contaminated grease from the underside of vehicles, if proper facilities are available to collect the run-off.

Bulldozing the top few inches of soil is effective for decreasing radiation levels over contaminated fields.²² For fission product gamma radiation, at 3 ft above the ground, roughly 50 per cent of the dose rate received in the center of a large, flat, uniformly contaminated area comes from distances greater than 25 ft away, and about 25 per cent comes from distances more than 50 ft away. Complete removal of the activity inside a 50-ft radius circle would decrease the dose rate at the center to one-fourth the original value. Removal of the activity to the perimeter of the 50-ft circle would decrease the dose rate to something greater than one-fourth. Burying activity under 1 ft of soil decreases the fission product gamma dose rate to approximately one-tenth the dose rate when the activity is on the surface.

For further information on decontamination methods, see Refs. 10, 22, 39, 76, 96, 97, 100, 131-134, 150, 151, 163, and 179.

14 DOSES FROM VARIOUS RADIONUCLIDES, INTERNAL AND EXTERNAL

(See Table B.5.)

15 FILTERING EFFICIENCY OF COMMON PERSONAL ITEMS²⁹

The filtration efficiencies for a 2.1-micron mass median diameter aerosol, with 95 per cent of the particles between 1.0 and 5.0 microns, was measured for a number of articles as described in Ref. 29. The results for those items having filtration efficiencies greater than 85 per cent were:

man's cotton handkerchief, 16 layers	94.2 per cent
toilet paper, 3 layers	91.4 per cent
man's cotton handkerchief, 8 layers	88.9 per cent
man's cotton handkerchief, crumpled	88.1 per cent
bath towel, 2 layers	85.1 per cent

The manufacturers of the items listed, and additional conditions, are given in Ref. 29.

16 FISSION PRODUCT AREA CONTAMINATION (BURST)

A fall-out dose rate of 1000 r/hr at one hour after an explosion releasing fresh fission products represents the following surface densities of activity.

TABLE B.6—MICROCURIES PER CM² PER 1000 R/HR AT H + 1 HOUR²⁵

Isotope	Time after detonation (days)			
	1	7	28	365
	$\mu\text{C}/\text{cm}^2$	$\mu\text{C}/\text{cm}^2$	$\mu\text{C}/\text{cm}^2$	$\mu\text{C}/\text{cm}^2$
Cs ¹³⁷	0.0035	0.0035	0.0035	0.0030
Sr ⁹⁰	0.0045	0.0045	0.0045	0.0043
Sr ⁸⁹	0.68	0.63	0.48	0.0060
Ba ¹⁴⁰	3.7	2.7	0.83	-
Ce ¹⁴⁴	0.15	0.15	0.14	0.060
Ru ¹⁰⁶	0.011	0.011	0.011	0.005
I ¹³¹	2.8	1.8	0.27	-
I ¹³²	6.8	2.2	0.023	-
I ¹³³	22	0.27	-	-
I ¹³⁵	13	-	-	-
I Total	44.6	4.3	0.29	-
Gross activity	290	30	7.6~	0.23

17 FISSION PRODUCT ACTIVITY IN 500-MW REACTOR

TABLE B.7—PRINCIPLE ACTIVITIES IN 500-MW REACTOR
(180 days operation, one-day shutdown³⁵)

Activity	Curies
Total	4.1×10^8
Volatile	8.4×10^7
Iodine	5×10^7
Noble gases	3.4×10^7
Sr ⁸⁹	1.7×10^7
Sr ⁹⁰	3×10^5
Ce ¹⁴⁴	8×10^6

TABLE B.5

Single intake that will give 5 rem to critical organ within one year ^a								
by								
Isotope	Radiological half life	Inhaling soluble material		Injecting soluble material		Inhaling insoluble (based on lung dose)	Inhaling insoluble (based on GI tract dose)	Activity of point source giving 100 mr/hr at 1 cm ^b (μ c)
		(μ c)	Critical organ	(μ c)	Critical organ	(μ c)	(μ c)	
H ³ (HTO or H ₂ O)	12.5 years	5.42 × 10 ⁴	T. body	4.14 × 10 ⁴	T. body	8.6 × 10 ²	2.0 × 10 ⁵	>1000
Be ⁷	53.5 days	1.27 × 10 ⁴	Bone	3.18 × 10 ³	Bone	3.86 × 10 ²	7.0 × 10 ³	315
C ¹⁴ (CO ₂)	5568 years	1.02 × 10 ³	Fat	7.34 × 10 ²	Fat	9.56 × 10	2.33 × 10 ⁴	>1000
F ¹⁸	110 minutes	2.13 × 10 ⁵	Bone	5.10 × 10 ⁴	Bone	8.92 × 10 ⁴	2.83 × 10 ⁴	1000
Na ²⁴	15.1 hours	4.8 × 10 ³	T. body	2.86 × 10 ³	T. body	3.86 × 10 ²	9.50 × 10 ²	5.3 ^c
P ³²	25 days	2.55 × 10 ²	Bone	1.65 × 10 ²	Bone	6.06 × 10	3.17 × 10 ²	>1000
S ³⁵	88 days	1.81 × 10 ³	Skin	9.54 × 10 ²	Skin	2.01 × 10 ²	2.67 × 10 ³	>1000
Cl ³⁶	4.4 × 10 ⁵ years	1.27 × 10 ³	T. body	9.54 × 10 ²	T. body	2.01 × 10	4.86 × 10 ³	400 ^c
K ⁴²	12.5 hours	5.42 × 10 ³	Muscle	3.18 × 10 ³	Muscle	7.97 × 10 ²	1.63 × 10 ³	51 ^d
Ca ⁴⁵	180 days	2.13 × 10	Bone	1.46 × 10	Bone	9.56 × 10	6.66 × 10 ³	>1000
Sc ⁴⁶	84 days	2.13 × 10 ²	Spleen	5.08 × 10	Spleen	8.29	1.53 × 10 ²	9.8 ^d
Sc ⁴⁷	3.40 days	1.50 × 10 ²	Liver	2.93 × 10	Liver			
		2.32 × 10 ³	Spleen	4.46 × 10 ²	Spleen	7.97 × 10 ²	3.50 × 10 ²	189 ^e
Sc ⁴⁸	44 hours	1.69 × 10 ³	Liver	3.18 × 10 ²	Liver			
		1.37 × 10 ³	Spleen	3.18 × 10 ²	Spleen	2.01 × 10 ²	1.52 × 10 ²	5.4
V ⁴⁸	16 days	7.64 × 10 ²	Liver	1.49 × 10 ²	Liver			
Cr ⁵¹	28 days	1.37 × 10 ³	Bone	3.50 × 10 ²	Bone	2.13 × 10	1.18 × 10 ²	6.1 ^d
Mn ⁵⁶	2.6 hours	3.18 × 10 ⁴	Kidneys	9.23 × 10 ³	Kidneys	2.13 × 10 ³	8.34 × 10 ³	13.7 ^c
Fe ⁵⁵	2.6 years	1.91 × 10 ⁴	Kidneys	2.20 × 10 ³	Kidneys	6.70 × 10 ³	1.330 × 10 ³	10.6 ^d
		2.64 × 10 ⁴	Liver	6.04 × 10 ³	Liver			
		1.59 × 10 ³	Blood	1.02 × 10 ³	Blood	9.24 × 10 ²	5.16 × 10 ⁴	No γ (5 μ c for 6.5 kev X-ray)
Fe ⁵⁹	45 days	3.82 × 10	Blood	2.51 × 10	Blood	2.01 × 10	1.13 × 10 ³	15.3 ^{f,d,e}
Co ⁶⁰	5.2 years	2.8 × 10 ³	Liver	9.55 × 10 ²	Liver	3.51	1.83 × 10 ²	7.6 ^f
Ni ⁶³	10 ⁵ years	4.14 × 10 ⁴	Liver	1.43 × 10 ³	Liver	1.02 × 10 ²	1.57 × 10 ³	>1000
Cu ⁶⁴	12.9 hours	1.37 × 10 ⁴	Liver	4.45 × 10 ³	Liver	5.74 × 10 ³	1.83 × 10 ³	88 ^f
Zn ⁶⁵	245 days	5.09 × 10 ³	Bone	1.53 × 10 ³	Bone	2.13 × 10	8.18 × 10 ²	12.3 ^c
Ga ⁷²	14 hours	3.82 × 10 ³	Bone	7.64 × 10 ²	Bone	5.42 × 10 ²	1.83 × 10 ²	6.4 ^g
Ge ⁷¹	11 days	1.05 × 10 ⁵	Kidneys	2.61 × 10 ⁴	Kidneys	5.42 × 10 ³	6.50 × 10 ³	91 ^e
As ⁷⁶	26 hours	7.00 × 10 ³	Kidneys	1.69 × 10 ³	Kidneys	4.15 × 10 ²	9.00 × 10	45 ^d
Rb ⁸⁶	19 days	1.05 × 10 ³	Muscle	7.95 × 10 ²	Muscle	4.46 × 10	2.17 × 10 ³	172 ^c
Sr ⁸⁹	51 days	1.49 × 10	Bone	8.27	Bone	2.65 × 10	2.84 × 10 ²	>1000
Sr ⁹⁰ + Y ⁹⁰	28 years	1.81	Bone	9.54 × 10 ⁻¹	Bone	5.10	3.50 × 10 ²	>1000
Y ⁹¹	58 days	2.32 × 10	Bone	6.04	Bone	2.45 × 10	1.10 × 10 ²	>1000
Zr ⁹⁵ + Nb ⁹⁵	65 days	1.69 × 10 ²	Bone	4.46 × 10	Bone	1.37 × 10	2.67 × 10 ²	24.7
Nb ⁹⁵	35 days	5.74 × 10 ²	Bone	2.73 × 10 ²	Bone	3.82 × 10	7.67 × 10 ²	23.4
Mo ⁹⁹	66 hours	1.27 × 10 ⁶	Bone	8.28 × 10 ⁵	Bone	3.82 × 10 ²	1.20 × 10 ³	71.4 ^c
Tc ⁹⁶	4.2 days	6.68 × 10 ³	Kidneys	3.82 × 10 ³	Kidneys	8.92 × 10	4.00 × 10 ²	7.2
Ru ¹⁰⁶ + Rh ¹⁰⁶	1.0y-30 sec	7.64 × 10 ⁻¹	Kidneys	1.910 × 10	Kidneys	4.15	4.50 × 10	74
Rh ¹⁰⁵	37 hours	2.23 × 10 ²	Kidneys	8.28 × 10 ²	Kidneys	8.28 × 10 ²	3.67 × 10 ²	1100
Pd ¹⁰³ + Rh ^{103*}	17 days	1.59 × 10 ³	Kidneys	6.36 × 10 ²	Kidneys	4.78 × 10 ²	2.00 × 10 ³	330 ^e
Ag ¹⁰⁵	40 days	2.77 × 10 ⁴	Liver	9.23 × 10 ³	Liver	9.56	1.57 × 10 ²	16.5
Ag ¹¹¹	7.6 days	7.64 × 10 ⁴	Liver	2.45 × 10 ⁴	Liver	2.13 × 10 ²	1.83 × 10 ²	600
Cd ¹⁰⁹ + Ag ^{109*}	470 days	1.91 × 10 ²	Liver	4.77 × 10	Liver	8.60 × 10	2.17 × 10 ³	205
Sn ¹¹³	119 days	1.59 × 10 ³	Bone	4.14 × 10 ²	Bone	3.09 × 10	7.34 × 10 ²	46
Te ¹²⁷	105d-9.4 hours	2.77 × 10 ²	Kidneys	9.23 × 10	Kidneys	2.870 × 10	3.00 × 10 ²	890
Te ¹²⁹	33d-72 minutes	1.18 × 10 ²	Kidneys	3.82 × 10	Kidneys	1.50 × 10	1.02 × 10 ²	77
I ¹³¹	8.1 days	5.42	Thyroid	4.14	Thyroid	1.69 × 10 ²	4.84 × 10 ³	40 ^c
Cs ¹³⁷ + Ba ^{137*}	27 years	5.73 × 10 ²	Muscle	4.45 × 10 ²	Muscle	9.24	3.67 × 10 ³	28 ^f
Ba ¹⁴⁰ + La ¹⁴⁰	13 days	4.77 × 10	Bone	1.30 × 10	Bone	2.45 × 10	1.17 × 10 ²	6.5 ^h
La ¹⁴⁰	40 hours	1.05 × 10 ³	Bone	2.55 × 10 ²	Bone	2.23 × 10 ²	1.10 × 10 ²	10.5
Ce ¹⁴⁴ + Pr ¹⁴⁴	285d-17 minutes	5.42	Bone	1.37	Bone	4.78	4.84 × 10	50 ⁱ
Pr ¹⁴³	14 days	4.46 × 10 ²	Bone	1.08 × 10 ²	Bone	1.50 × 10 ²	2.17 × 10 ²	>1000
Pm ¹⁴⁷	2.6 years	1.81 × 10 ²	Bone	4.78 × 10	Bone	8.30 × 10	9.34 × 10 ²	>1000
Sm ¹⁵¹	93 years	3.82 × 10 ²	Bone	9.24 × 10	Bone	2.55 × 10 ²	3.17 × 10 ³	>1000
Eu ¹⁵⁴	16 years	1.91 × 10	Bone	5.09	Bone	6.37	1.83 × 10 ²	14.3 ⁱ
Ho ¹⁶⁶	27 hours	2.13 × 10 ³	Bone	6.04 × 10 ²	Bone	7.66 × 10 ²	1.67 × 10 ²	79
Tm ¹⁷⁰	129 days	2.86 × 10	Bone	7.32	Bone	2.77 × 10	2.00 × 10 ²	2500
Lu ¹⁷⁷	6.8 days	3.09 × 10 ³	Bone	7.64 × 10 ²	Bone	5.10 × 10 ²	4.67 × 10 ²	690
Ta ¹⁸²	115 days	5.73 × 10	Liver	1.47 × 10	Liver	1.18 × 10	2.00 × 10 ²	16.78 ⁱ
W ¹⁸¹	145 days	1.05 × 10 ⁴	Bone	3.18 × 10 ³	Bone	7.97	2.67 × 10 ²	>1000
Re ¹⁸³	71 days	2.33 × 10 ⁴	Thyroid	1.21 × 10 ⁴	Thyroid	1.50 × 10	8.34 × 10 ²	69
Ir ¹⁹⁰	11 days-3.2 hours	7.63 × 10 ⁴	Skin	1.11 × 10 ⁴	Skin			
		2.33 × 10 ³	Kidneys	7.96 × 10 ²	Kidneys	2.77 × 10 ²	1.35 × 10 ³	10.9
Ir ¹⁹²	74 days	2.45 × 10 ³	Spleen	1.81 × 10 ²	Spleen			
		1.27 × 10 ²	Kidneys	4.79 × 10	Kidneys	1.18 × 10	2.00 × 10 ²	19.6 ^f
Pt ¹⁹¹	3.0 days	7.96 × 10	Spleen	5.74	Spleen			
		5.73 × 10 ²	Kidneys	1.62 × 10 ²	Kidneys	1.59 × 10 ²	3.00 × 10 ²	33 (?) decay scheme uncertain
Pt ¹⁹³	1.8 × 10 ⁵ days	4.46 × 10 ²	Kidneys	1.31 × 10 ²	Kidneys	1.28 × 10 ²	3.50 × 10 ²	417
Au ¹⁹⁶	5.6 days	5.42 × 10 ²	Liver	1.37 × 10 ²	Liver	2.77 × 10 ²	8.50 × 10 ²	50.8
		4.46 × 10 ²	Kidneys	1.30 × 10 ²	Kidneys			
Au ¹⁹⁸	2.7 days	4.46 × 10 ²	Liver	1.05 × 10 ²	Liver	3.82 × 10 ²	2.34 × 10 ²	40 ^c
		2.87 × 10 ²	Kidneys	7.97 × 10	Kidneys			
Au ¹⁹⁹	3.1 days	9.56 × 10 ²	Liver	2.36 × 10 ²	Liver	7.96 × 10 ²	6.66 × 10 ²	113
		6.69 × 10 ²	Kidneys	1.99 × 10 ²	Kidneys			
Tl ²⁰⁰	26 hours	6.69 × 10 ³	Muscle	2.58 × 10 ⁴	Muscle	4.46 × 10 ²	5.17 × 10 ²	45.5
Tl ²⁰¹	72 hours	1.91 × 10 ⁴	Muscle	7.96 × 10 ⁴	Muscle	1.18 × 10 ³	3.67 × 10 ³	108 (?)
Tl ²⁰²	12 days	5.42 × 10 ³	Muscle	2.29 × 10 ⁴	Muscle	2.33 × 10 ²	2.00 × 10 ³	40.8
Tl ²⁰⁴	3.6 years	1.91 × 10 ³	Muscle	7.96 × 10 ³	Muscle	2.14 × 10	4.50 × 10 ²	No γ (100, K-X-ray)
Pb ²⁰³	52 hours	2.01 × 10 ³	Bone	6.06 × 10 ³	Bone	5.42 × 10 ²	7.67 × 10 ²	61.7
Pb ²¹⁰ + dr	19 years	8.28 × 10 ⁻²	Bone	2.71 × 10 ⁻²	Bone	9.23 × 10 ⁻²	6.50 × 10 ²	449
Po ²¹⁰	138 days	1.18	Spleen	2.29 × 10 ⁻¹	Spleen	1.50 × 10 ⁻¹	1.07	>1000
At ²¹¹ + dr	7.5h-8.0 years	1.69	Thyroid	8.92 × 10 ⁻¹	Thyroid	3.50 × 10	4.00 × 10	>1000
Ra ²²⁶ + 55% dr	1600 years	4.46 × 10 ⁻¹	Bone	1.56 × 10 ⁻¹	Bone	2.01 × 10 ⁻²	8.5 × 10 ⁻¹	11.9 ^f
Ac ²²⁷ + dr	21.2 years						(includ. all dr)	
		1.69 × 10 ⁻²	Bone	4.14 × 10 ⁻³	Bone	1.50 × 10 ⁻²	1.83 × 10	29.8
Th (nat)	1.4 × 10 ¹⁰ years	4.78 × 10 ⁻³	Bone	1.27 × 10 ⁻³	Bone	1.37 × 10 ⁻²	1.53 (Th ²³² + dr)	472
Th ²³⁴ + Pa ²³⁴	24 days	2.35 × 10	Bone	5.73	Bone	3.09 × 10	7.34 × 10	12
U (nat)	4.5 × 10 ⁹ years	8.92 × 10 ⁻²	Kidneys	2.16 × 10 ⁻²	Kidneys	5.42 × 10 ⁻²	1.7 (based on U ²³⁸)	498
U ²³³	1.6 × 10 ⁵ years	2.23 × 10 ⁻¹	Bone	3.50 × 10 ⁻²	Bone	1.02 × 10 ⁻¹	1.25	2800
Pu ²³⁹	2.4 × 10 ⁴ years	3.82 × 10 ⁻²	Bone	9.86 × 10 ⁻³	Bone	9.56 × 10 ⁻²	1.18	59
Am ²⁴¹	458 years	1.18 × 10 ⁻¹	Bone	3.06 × 10 ⁻²	Bone	9.24 × 10 ⁻²	1.12	256
Cm ²⁴²	162 days	2.13 × 10 ⁻¹	Bone	5.42 × 10 ⁻²	Bone	1.27 × 10 ⁻¹	1.00	>1000

* The daughter products in these cases are isomers in an excited state.

^aCalculated from tables by K. Z. Morgan, Walter S. Snyder, and Mary R. Ford, Maximum Permissible Concentration of Radioisotopes in Air and Water for Short Period Exposure, copy of 1955 Geneva Conference paper, dated June 2, 1958.

^bIntensities are not corrected for self-absorption by the source or its container.

^cL. Slack and K. Way, Radiations from Radioactive Atoms in Frequent Use, USAEC Report M-6965, National Research Council, February 1959.

^dL. D. Marinello, E. H. Quimby, and G. J. Hine, Am. J. Roentgenol. Radium Therapy (February 1948).

^eAll values not referenced were calculated by A. Brodsky from tables of nuclear data by D. Strominger, J. M. Hollander, and G. T. Seaborg, Revs. Modern Phys. 30, No. 2, Part II, April 1958. Low energy (below 10 kev) X-rays and beta rays were neglected. Position annihilation radiation was included and adjustments were made for variation of absorption coefficient with energy from 10-60 kev. From 60 kev to 3 Mev, the approximate relation, mr/hr at 1 cm per microcurie = 5.58 E, where E is the average gamma energy in Mev per disintegration, was used.

^fSimon B. Kinsman (Ed.), Radiological Health Handbook, Report PB-121784, p. 139, Robert A. Taft Sanitary Engineering Center, Cincinnati, January 1957.

^gPrice, Horton, and Spinney, Radiation Shielding, Pergamon Press, 1957.

^hA. Brodsky and R. Vogt, unpublished report, NRL, 1952.

ⁱNational Bureau of Standards Handbook 54, p. 42.

18 FISSION PRODUCT ACTIVITY (FAST REACTOR)*

The gross fission product beta activity from the fast fission of U^{235} is

$$\text{Curies/watt} = 16 \{T_S^{-0.2} - (T_S + T_0)^{-0.2} + 0.18 [(T_S + T_0 + 10^7)^{-0.2} - (T_S + 10^7)^{-0.2}]\},$$

where T_S is the cooling time in seconds, and T_0 is the irradiation time in seconds.

19 FISSION PRODUCT INVENTORIES (EQUATIONS)

Assuming 3.25×10^{10} fissions per watt-second and 3.7×10^{10} quanta per second is defined as one curie, the following equations apply: The activity of a fission product beta emitter is

$$\text{Curies/megawatt} = 8.46 \times 10^5 \gamma (1 - e^{-\lambda_a^* T}) e^{-\lambda_a t},$$

where γ = the fractional number of fissions yielding the isotope in question ($\sum \gamma_i = 2$ for all isotopes, since two are produced per fission)

$\lambda_a^* = \lambda_a + \sigma_a \phi$, where σ is in cm^2 and ϕ is in neutrons/ cm^2/sec

T = reactor operating time

t = fission product cooling time

λ_a = radiological decay constant for the isotope in reciprocal units of time

(See Table B.8 for the inventories of the more important fission products.)

For a *short burst from a critical assembly*

$$\text{Beta Activity} = 2.94 t^{-1.2} \text{ curies per watt-sec}$$

$$\text{Gamma Activity} = 1.82 t^{-1.2} \text{ curies per watt-sec, for } t \text{ in seconds, from 10 sec to 100 days}$$

For the *gross beta activity* in the fission products of a reactor,

$$\text{Curies of beta activity} = 1.4P [t_c^{-0.2} - (t_c + t_0)^{-0.2}],$$

where P is the power in watts, t_c is the cooling time in days, and t_0 is the operating time in days.

For *long operating times*, t_0 , compared to the half lives of most fission products, and cooling times $t \ll t_0$.⁵³

$$\text{Beta activity} = 15.4 t^{-0.2} \text{ curies per watt}$$

$$\text{Gamma activity} = 9.1 t^{-0.2} \text{ curies per watt,}$$

where t is in seconds. (In 32 seconds the activity is about one-half of that in 1 sec.)

20 FISSION PRODUCTS (U^{235}) INVENTORY† AS A FUNCTION OF OPERATING AND COOLING TIME

(See Table B.8 on the following page.)

* Dillon and Burriss, Nuclear Sci. and Eng. 2(5) (September 1957).

† Unpublished calculations of A. Brodsky, Health Protection Branch, Division of Biology and Medicine, USAEC, 1958. Correction has been made for the contribution to each nuclide from the decay of fission-product precursors. Nuclides were selected as those of highest activity and biological effectiveness. Fission yields, decay constants, and cross sections were selected from data in the AIP Handbook of Physics, edited by D. E. Gray, McGraw-Hill, 1957; NBS Handbook of Nuclear Data, No. 499, with supplements; and other sources. For further data on total element activities, see Refs. 61 and 62. The total activity of each element, as given in Reference 61 for 1-day cooling time, was no more than 2-3 times the total activity of the isotopes of the corresponding element included in Table B.8, except for iodine. The total iodine activity in Ref. 61 was about 6 times the I^{131} plus I^{133} activity given here, which can be accounted for by the many shorter-lived iodine isotopes included in Ref. 61.

TABLE B.8

Activity* after 2-day cooling time for various operating times, in megacuries of beta activity for 100-Mw operation						
Nuclide	1/2-year operation	1-year operation	1 1/2-year operation	2-year operation	2 1/2-year operation	3-year operation
Strontium-89 (54 days)	3.31	3.59	3.62	3.63	3.63	3.63
Strontium-90 (20 years)	0.084	0.158	0.241	0.306	0.38	0.46
Yttrium-90 (21 hours)	0.084	0.158	0.241	0.306	0.38	0.46
Strontium-91 (9.7 hours)	0.15	0.15	0.15	0.15	0.15	0.15
Yttrium-91 (61 days)	4.08	4.60	4.67	4.68	4.68	4.68
Zirconium-95 (65 days)	4.50	5.20	5.30	5.3	5.3	5.3
Niobium-95 (37 days)	4.29	5.15	5.25	5.48	5.48	5.48
Iodine-131 (8.14 days)	1.85	1.85	1.85	1.85	1.85	1.85
Iodine-133 (21 hours)	1.11	1.11	1.11	1.11	1.11	1.11
Xenon-133 (5.27 days)	4.69	4.69	4.69	4.69	4.69	4.69
Cesium-137 (33 years)	0.083	0.156	0.25	0.31	0.40	0.48
Barium-137 (2.6 minutes)	0.083	0.156	0.25	0.31	0.40	0.48
Barium-140 (12.8 days)	6.00	6.00	6.00	6.00	6.00	6.00
Lanthanum-140 (40 hours)	6.45	6.45	6.45	6.45	6.45	6.45
Cerium-141 (32 days)	3.54	3.55	3.55	3.55	3.55	3.55
Cerium-143 (33 hours)	1.81	1.81	1.81	1.81	1.81	1.81
Praesodymium-143 (138 days)	4.73	4.73	4.73	4.73	4.73	4.73
Cerium-144 (280 days)	1.54	2.52	3.13	3.51	3.77	3.93
Praesodymium-144 (17 minutes)	1.54	2.52	3.13	3.51	3.77	3.93
Neodymium-147 (11.6 days)	1.84	1.84	1.84	1.84	1.84	1.84
Promethium-147 (2.6 years)	0.26	0.49	0.68	0.86	1.01	1.14
Activity after 7-day cooling (Megacuries of beta/100 Mw)						
Strontium-89	3.10	3.37	3.40	3.41	3.41	3.41
Strontium-90	0.084	0.158	0.24	0.306	0.38	0.46
Yttrium-90	0.084	0.158	0.24	0.306	0.38	0.46
Strontium-91	0.000	-	-	-	-	-
Yttrium-91	3.88	4.38	4.45	4.45	4.45	4.45
Zirconium-95	4.25	4.98	5.00	5.00	5.00	5.00
Niobium-95	4.29	5.12	5.22	5.43	5.43	5.43
Iodine-131	1.25	1.25	1.25	1.25	1.25	1.25
Iodine-133	0.02	0.02	0.02	0.02	0.02	0.02
Xenon-133	2.50	2.50	2.50	2.50	2.50	2.50
Cesium-137	0.083	0.083	0.083	0.083	0.083	0.083
Barium-137	0.083	0.083	0.083	0.083	0.083	0.083
Barium-140	4.58	4.58	4.58	4.58	4.58	4.58
Lanthanum-140	5.17	5.17	5.17	5.17	5.17	5.17
Cerium-141	0.75	0.75	0.75	0.75	0.75	0.75
Cerium-143	0.15	0.15	0.15	0.15	0.15	0.15
Praesodymium-143	3.83	3.83	3.83	3.83	3.83	3.83
Cerium-144	1.52	2.49	3.09	3.47	3.72	3.88
Praesodymium-144	1.52	2.49	3.09	3.47	3.72	3.88
Neodymium-147	1.37	1.37	1.37	1.37	1.37	1.37
Promethium-147	0.26	0.49	0.68	0.86	1.01	1.01
Activity after 30-day cooling (Megacuries of beta/100 Mw)						
Strontium-89	2.31	2.51	2.53	2.53	2.53	2.53
Strontium-90	0.084	0.158	0.241	0.306	0.38	0.46
Yttrium-90	0.106	0.196	0.302	0.382	0.474	0.58
Strontium-91	0.000	-	-	-	-	-
Yttrium-91	3.09	3.49	3.54	3.55	3.55	3.55
Zirconium-95	3.34	3.85	3.92	3.94	3.94	3.94
Niobium-95	4.14	4.88	4.97	5.12	5.12	5.12
Iodine-131	0.205	0.205	0.205	0.205	0.205	0.205
Iodine-133	-	-	-	-	-	-
Xenon-133	0.123	0.123	0.123	0.123	0.123	0.123
Cesium-137	0.083	0.156	0.25	0.31	0.40	0.48
Barium-137	0.083	0.156	0.25	0.31	0.40	0.48
Barium-140	1.53	1.53	1.53	1.53	1.53	1.53
Lanthanum-140	1.49	1.49	1.49	1.49	1.49	1.49
Cerium-141	-	-	-	-	-	-
Cerium-143	-	-	-	-	-	-
Praesodymium-143	1.20	1.20	1.20	1.20	1.20	1.20
Cerium-144	1.44	2.35	2.92	3.27	3.52	3.66
Praesodymium-144	1.44	2.35	2.92	3.27	3.52	3.66
Neodymium-147	0.346	0.35	0.35	0.35	0.35	0.346
Promethium-147	0.26	0.48	0.67	0.84	0.99	0.99

* Calculations can be relied on only to within ± 20 per cent. Variations with neutron flux were neglected.

21 FISSION PRODUCT RELEASE

The releases in Table B.9 were obtained from a UO_2 stainless steel dispersion type fuel element melting.

TABLE B.9—FISSION PRODUCT RELEASE*

Type	Per cent released	
	in air	in steam
Gamma activity	4.5	2.3
Iodine activity	33	11
Rare gases	48	38.5
Strontium	0.025	0.4

* Parker and Creek, ORNL-CF-57-6-87; see also WAPD-173 and WAPD-183; Refs. 115 to 126 and 177; and talk on "Burn Test" in this report.

22 FUEL ELEMENT REPROCESSING WASTES

Elements of main hazard after 10-day decay:

After 1000-day decay:

23 GAMMA RADIATION

Relaxation Length

The relaxation length of prompt gamma radiation (4.5 Mev) in air is 338 yd. (See also talk by F. J. Davis.)

Linear Absorption Coefficient for Gamma Rays*

Table B.10 gives absorption coefficients including Compton, photoelectric, and pair production interactions. Table B.11 gives half-value layers. Appropriate build-up factors must be applied for thick absorbers. (Table B.12 gives some build-up factors for air.²² Reference 22, page 380, gives build-up factors as a function of atomic number.)

24 HEAT TRANSFER COEFFICIENTS

Zirconium cladding to water—10,000 $\text{BTU}\cdot\text{hr}^{-1}\cdot\text{ft}^{-2}\cdot^\circ\text{F}^{-1}$ for boiling, 170 for non-boiling
 UO_2 fuel to cladding—1000 $\text{BTU}\cdot\text{hr}^{-1}\cdot\text{ft}^{-2}\cdot^\circ\text{F}^{-1}$
 UO_2 thermal conductivity—1.2 $\text{BTU}\cdot\text{hr}^{-1}\cdot\text{ft}^{-2}\cdot(^\circ\text{F}/\text{ft})^{-1}$
 UO_2 specific heat—0.073 $\text{BTU}\cdot\text{lb}^{-1}\cdot^\circ\text{F}^{-1}$

* From "The Effects of Nuclear Weapons," Samuel Glasstone, ed., U. S. Atomic Energy Commission, June 1957.

TABLE B.10— LINEAR ABSORPTION COEFFICIENTS

Gamma ray energy (Mev)	Linear absorption coefficient in cm^{-1}				
	Air	Water	Concrete (2.3 g/cc)	Iron	Lead
0.5	1.11×10^{-4}	0.097	0.22	0.66	1.7
1.0	0.81×10^{-4}	0.071	0.15	0.47	0.80
2.0	0.57×10^{-4}	0.049	0.11	0.33	0.52
3.0	0.46×10^{-4}	0.040	0.088	0.28	0.47
4.0	0.41×10^{-4}	0.034	0.078	0.26	0.47
5.0	0.35×10^{-4}	0.030	0.071	0.25	0.50
10	0.26×10^{-4}	0.022	0.060	0.23	0.61

TABLE B.11— APPROXIMATE HALF-VALUE LAYERS FOR
GAMMA RAYS FROM FISSION PRODUCTS²²

Material	Density (lb/cu ft)	Half-value thickness (in.)
Steel	490	0.7
Concrete	144	2.2
Earth	100	3.3
Water	62.4	4.8
Wood	34	8.8

TABLE B.12— BUILD-UP FACTORS IN AIR

No. of relaxation lengths, μx	Build-up factor B(x)	
	4 Mev	Fission products
1	1.5	1.8
2	1.9	3.9
4	3	5.5
7	5	10
10	7.5	16
15	12	26
20	17	40

25 INHALED DOSE FROM FISSION PRODUCT CLOUD (from IDO-16425)

Cloud Volume

$$V = [\pi^{1/2} C X^{(2-n)/2}]^3,$$

where V = cloud volume in cubic meters

X = distance downwind from the point origin of the cloud in meters

n = Sutton's stability parameter (Table B.13)

C = Sutton's diffusion coefficient $m^{n/2}$ (Table B.13)

TABLE B.13—SAMPLE METEOROLOGICAL CONSTANTS

	Av. conditions	Inversion conditions	Strong wind
Stability index, n	0.25	0.50	0.25
Diffusion parameter, C	0.20 (meters ^{1/2})	0.05 (m ^{1/2})	0.20 (m ^{1/2})
Wind velocity, meters/sec	3 m/sec	1	15

Concentration of Activity

$$C(r) = \frac{A}{V} e^{-(r/r_0)^2}$$

where C(r) = concentration of radioactivity in the cloud in millicuries/meter³

r = distance from the cloud center in meters

$r_0 = CX^{(2-n)/2}$ = the mean radius of the cloud with X in meters

A = total number of curies in the cloud

Activity Inhaled During Cloud Passage

$$I = 2 \int_0^\infty J C(r) dt = \frac{JA}{vV^{2/3}},$$

where I is the inhaled activity in millicuries, v is the cloud velocity in meters/sec, and J is the inhalation rate in liters per sec. Assuming $J = 17$ liters per min $\times 1.67 \times 10^{-2}$ min/sec, then

$$I = 2.84 \times 10^{-1} \frac{A}{vV^{2/3}}$$

Cloud Activity, A

For a specific isotope, the activity of fission product beta emission (neglecting contributions from parental decay) from a thermal reactor incident is:

$$A = (8.4 \times 10^8) P y (1 - e^{-\lambda t_0}) e^{-\lambda(X/v)} \text{ millicuries}$$

where P = reactor power in megawatts

y = fraction of fissions yielding the isotope (see Table B.14)

λ = radioactive decay constant in sec⁻¹

t_0 = operating time prior to release of fission products in sec

X = the distance from the point of release in meters

v = the cloud velocity in meters per second

TABLE B.14—ISOTOPE DATA FOR INHALED DOSE CALCULATIONS

Isotope	Critical organ	Organ weight (grams)	f_c	f_a	E (Mev)	y	λ (sec ⁻¹)	λ_e (sec ⁻¹)
I ¹³¹	Thyroid	20	1	0.15	0.22	0.031	9.86×10^{-7}	1.03×10^{-6}
I ¹³²	Thyroid	20	1	0.15	0.50	0.048	8.02×10^{-5}	8.02×10^{-5}
I ¹³³	Thyroid	20	1	0.15	0.40	0.051	9.44×10^{-6}	9.44×10^{-6}
I ¹³⁴	Thyroid	20	1	0.15	0.65	0.063	2.88×10^{-5}	2.88×10^{-5}
I ¹³⁵	Thyroid	20	1	0.15	0.31	0.062	2.20×10^{-4}	2.20×10^{-4}
I ¹³⁶	Thyroid	20	1	0.15	2.2	0.034	8.06×10^{-3}	8.06×10^{-3}
Sr ⁸⁹	Bone	7×10^3	0.2	0.22	0.55	0.047	1.52×10^{-7}	1.54×10^{-7}
Sr ⁹⁰ —Y ⁹⁰	Bone	7×10^3	0.2	0.22	1.00	0.052	8.80×10^{-10}	2.97×10^{-9}
Y ⁹¹	Bone	7×10^3	0.2	0.14	0.59	0.056	1.41×10^{-7}	1.57×10^{-7}
Zr ⁹⁵ —Nb ⁹⁵	Bone	7×10^3	0.2	0.12	0.20	0.067	1.24×10^{-7}	1.70×10^{-7}
Ba ¹⁴⁰ —La ¹⁴⁰	Bone	7×10^3	0.2	0.20	1.06	0.066	6.26×10^{-7}	6.69×10^{-7}
Ce ¹⁴¹	Bone	7×10^3	0.2	0.10	0.16	0.059	2.43×10^{-7}	2.58×10^{-7}
Pr ¹⁴³	Bone	7×10^3	0.2	0.063	0.31	0.061	5.81×10^{-7}	7.30×10^{-7}
Ce ¹⁴⁴ —Pr ¹⁴⁴	Bone	7×10^3	0.2	0.10	1.29	0.059	2.92×10^{-8}	4.46×10^{-8}
Eu ¹⁵⁴	Bone	7×10^3	0.2	0.09	0.37	0.0085	4.07×10^{-9}	9.80×10^{-9}

Internal Dose Rate

The internal dose rate to an organ is given by

$$\frac{dD}{dt} = \frac{f_a I E}{f_c g} \times \frac{3.7 \times 10^7 \times 1.6 \times 10^{-6}}{100 \text{ ergs/gm/rad}} e^{-\lambda_e t} \text{ rads/sec,}$$

where f_a = fraction of inhaled activity deposited in the critical organ (Table B.14)

f_c = fraction of the organ available (0.2 for bone, 1 for thyroid)

g = mass of the critical organ in grams

E = effective energy per disintegration in Mev/dis

$\lambda_e = \lambda_r + \lambda_b$ = the effective decay constant in sec⁻¹

t = time after inhalation in sec

I = inhaled activity in millicuries

Total Dosage

The total dose received from a specific isotope is thus:

$$D = \int_0^\infty \frac{dD}{dt} dt = \frac{1.41 \times 10^8 P f_a E y (1 - e^{-\lambda t_0}) e^{-\lambda(x/v)}}{v V^{2/3} f_c g \lambda_e} \text{ rads}$$

This derivation, of course, gives an estimated upper limit to the dose under isotropic turbulence—not a likely dose to be received. Appropriate multiplying factors must be selected on the basis of the situation, such as per cent of fission products released, fraction of activity falling out of cloud, etc. Some of the parameters needed in the above equations are given in Table B.14.

26 JUNO CALIBRATION

Roughly twice the alpha open window reading on the mr per hour scale gives the number of micrograms of plutonium per square meter on broomed concrete (within a factor of two). For gravel or rough ground, the number of micrograms per square meter to give the same reading would be 40 per cent higher; for smooth concrete it would be 20 per cent lower.

27 MILK CONTAMINATION

A level of 0.1 microcurie of iodine per liter of milk a few days after the Windscale Incident was calculated to give a dose of 20 rad to an infant thyroid of 1.5 g. Weights of 20 to 30 g are usually assumed for the adult thyroid. In therapeutic work, an ingested dose of one microcurie of I^{131} is estimated to give a dose of about one rad to a thyroid of 30 g. (See also Refs. 57, 65, and 155.)

28 MAXIMUM PERMISSIBLE CONCENTRATIONS, NORMAL OCCUPATIONAL EXPOSURE^{51,93}

Body Burden of Radioisotopes for Which Skeleton is Critical Organ

The maximum permissible skeletal content is generally derived by comparison with radium, for which the recommended maximum is 0.1 microcurie. On the basis of assumptions made by the National Commission on Radiation Protection (NCRP)-ICRP, such comparison leads to the equation,

$$q = \frac{11}{f_2 \Sigma(E \times RBE \times N)} \text{ microcuries,}$$

where q = body burden in microcuries

f_2 = fraction of the amount in the body that is in the critical organ (bone)

E = effective energy released to the organ in Mev per disintegration, weighted according to the Relative Biological Effectiveness (RBE)

N = non-uniform distribution factor, taken as 5 for elements other than radium, and as 1 for radium*

Body Burden to Give 0.3 Rem per Week to Critical Organ

$$q = \frac{8.4 \times 10^{-4} m}{f_2 \Sigma(E \times RBE \times N)} \text{ microcuries,}$$

where m = mass of the critical organ in grams

f_2 = fraction in critical organ of that in the total body

$E \times RBE$ = energy absorbed in Mev per disintegration weighted by RBE

N = non-uniform distribution factor which is usually taken as $N = 1$ for organs other than bone

Maximum Permissible Concentration, Based on Body Burden

$$MPC_{air} = \frac{3.5 \times 10^{-8} q f_2}{T f_a (1 - e^{-0.693t/T})},$$

where MPC_{air} = concentration in microcuries/cc that will eventually build up to a dose rate of 0.3 rem/week if breathed continuously for a time t

T = the effective half-life of the radionuclide in the body in days

t = period of inhalation in days

q = body burden in microcuries

f_2 = fraction of material in the body that is in the critical organ

f_a = fraction inhaled that arrives in the critical body organ

$$MPC_{water} = \frac{3.1 \times 10^{-4} q f_2}{T f_w (1 - e^{-0.693t/T})},$$

* NOTE: A factor $N = 5$ is now also used for Strontium-90, based on recent animal experiments. An RBE of 1 is used for beta particles, and an RBE of 10 is used for alpha particles. The value of $\Sigma(E \times RBE \times N)$ for Ra^{226} is now⁹³ taken as 110 Mev-equivalent per disintegration, assuming $N = 1$ for radium and a 30 per cent retention of Ra^{226} daughter products in bone.

where MPC_{water} is the concentration ingested for t days that will build up a burden of qf_2 microcuries in the critical organ, giving a dose rate of 0.3 rem/week to the critical organ at time t , and f_w is the fraction ingested that reaches the critical organ.

Maximum Permissible Concentrations, Based on Gastrointestinal (GI) Tract

(These equations have been superseded but will serve for making early approximation.^{51,93})

$$(MPC)_{\text{water}}^{\text{GI}} = \frac{7.6 \times 10^{-7} m^1 G}{H f_w \Sigma(E \times RBE)} \text{ microcuries/cc,}$$

and

$$(MPC)_{\text{air}}^{\text{GI}} = \frac{8.4 \times 10^{-11} m^1 G}{H f_a \Sigma(E \times RBE)} \text{ microcuries/cc,}$$

where

$$G_1 = 0.693H / (e^{-0.693h_0/T_r} - e^{-0.693h_1/T_r}) T_r$$

m^1 = mass of contents of the GI tract in the section considered

H = time (fraction of day) the radioactive material remains in the GI section considered ($H = h_1 - h_0$)

f_w and f_a = fractions of radioactive material arriving at the section considered ($f_w = (1 - f_1)$ and $f_a = 0.62(1 - f_1)$)

f_1 = fraction absorbed through the wall of the section

h_0 = time of arrival in the section considered following inhalation or ingestion

h_1 = time of elimination from section following inhalation or ingestion

Necessary parameters are given in Table B.15.

TABLE B.15—CONSTANTS FOR GI TRACT CALCULATIONS

Critical tissue	m^1 (grams)	H (days)	h_0 (days)	h_1 (days)	Value of T_r for $G = 1.00$ to 1.01
Lower large intestine, $T_r > 0.5$ day	150	18/24	13/24	31/24	$T_r > 64$ days
Upper large intestine, 0.08 day					
< $T_r < 0.5$ day	135	8/24	5/24	13/24	$T_r > 26$ days
Stomach, $T_r < 0.08$ day	250	1/24	0	1/24	$T_r > 1.5$ days

MPC of Noble Gases, Based on Submersion in Cloud⁹³

The 40-hour-per-week occupational MPC_{air} based on the dose within a semi-infinite cloud is

$$MPC_{\text{air}} = \frac{0.024W}{\Sigma E} \rho_a \frac{P_a}{P_t} \text{ microcuries/cm}^3,$$

where W = dose rate allowed in rem/week

$$\rho_a = 0.0012 \text{ g/cc}$$

P_a/P_t = stopping power of air relative to tissue, or 1/1.13 for betas or secondary electrons

ΣE = energy in Mev per disintegration of the isotope in question

Combining parameters, the MPC becomes for a dose rate of 0.1 rem in 40 hr,

$$MPC_{\text{air}} = \frac{2 \times 10^{-6}}{\Sigma E} \text{ microcuries/cm}^3$$

MPC for Insoluble Compounds in Lung⁵¹

$$\text{MPC}_{\text{air}} = \frac{2 \times 10^{-9}}{\Sigma E(\text{RBE})N} \text{ microcuries/cm}^3,$$

to give a dose rate of 0.3 rem per week to the lung, assuming continuous exposure with an equilibrium concentration in the lung and a biological half life of 120 days.

29 MAXIMUM PERMISSIBLE LEVELS FOR EMERGENCIES

No definitive numbers can be given to cover all emergency situations. Each situation will have to be judged with the objective of balancing a maximum saving of life and property against a minimum exposure to toxic materials. Some articles listed in references, and some of the material compiled here, will show how to calculate doses. No numbers given in the report are to be construed as indicating that exposures or concentrations exceeding these amounts in specific instances will produce harm or that smaller exposures are free of risk. In situations where highly desirable emergency operations cannot be performed without exposures of personnel to unusually large amounts of radiation, the NCRP¹⁸⁰ has suggested that, "An accidental or emergency dose of 25 rems to the whole body or a major portion thereof, occurring only once in the lifetime of the person, need not be included in the determination of the radiation exposure status of that person."

There may be, of course, situations in which much higher exposures are justified. The undesirability of any dose short of that producing nausea and other symptoms of acute sickness depends upon its contribution to the probability of such effects as genetic mutations, leukemia, etc. Doses of 100 to 200 rem or more should be assumed potentially capable of producing serious impairment to health if not acute sickness and possible death.

For purposes of comparison, estimates of radiation doses to various portions of the body, as a result of ingestion of various materials, are compiled below.

TABLE B.16 — EMERGENCY LEVELS OF SEVERAL NUCLIDES

Nuclide	MPL for ingestion at one day (microcuries per milliliter)
Co ⁶⁰	2×10^{-1}
Sr ⁹⁰	2×10^{-2}
Pu ²³⁹ (sol)	6×10^{-2}
I ¹³¹	1×10^{-2}
Ce ¹⁴⁴ — Pr ¹⁴⁴	2×10^{-1}
Fe ⁵⁵	2

Based on 25 rad to Critical Organ³⁸

On the basis of an integrated dose of 25 rad to the critical organ, the levels in Table B.16 were among those presented for exposure following a nuclear explosion.

As a simple rule of thumb, levels to ensure a 25 rem limited internal dose would be: 10^{-2} microcuries per milliliter for water ingested during the first day; 10^{-6} microcuries per cc of air for beta-gamma emitters breathed during the first day; and 10^{-9} microcuries per cc of air for alpha emitters breathed during the first day. (See also Refs. 7, 13, 27, 28, 30, 33, 35, and 36.)

Based on Dose to Lower Large Intestine¹³

The dose to the thyroid from ingesting fall-out following a nuclear detonation may be greater than the dose to the GI tract.¹³ However, the greater radiation sensitivity of the GI tract may make it a controlling consideration in selecting emergency levels of ingestion for

adults during the first 20 days after detonation. Table B.17 presents the calculated fission product activities to produce a one-rad dose to the lower large intestine, starting intake at various times.

TABLE B.17—FISSION PRODUCT ACTIVITIES TO PRODUCE ONE RAD TO INTESTINE*

Duration of ingestion (days)	Microcuries per milliliter to give one rad to lower large intestine for various initial times of intake						
	Start of intake						
	1 hr	24 hr	36 hr	48 hr	60 hr	10 days	20 days
1	0.35	0.025	0.019	0.017	0.014	0.011	0.010
2	0.24	0.017	0.011	0.0089	0.0081	0.0062	0.0053
3	0.15	0.013	0.0082	0.0065	0.0056	0.0041	0.0037
4	0.13	0.010	0.0065	0.0053	0.0046	0.0033	0.0029
5	0.12	0.009	0.0057	0.0044	0.0039	0.0028	0.0022
10	0.092	0.0064	0.0040	0.0029	0.0025	0.0017	0.0013
15	0.078	0.0053	0.0033	0.0026	0.0021	0.0013	0.00097
20	0.075	0.0049	0.0029	0.0021	0.0018	0.0011	0.00079

* Activities are given at the start of the intake period, and are based on 2200 ml or grams of water and food per day for adults. Relative doses and effects for children would be greater.

It may be noted that the 10-day intake value, beginning ingestion at 1 hr, is almost identical to the 10-day acceptable level given in Table B.18. Note that these levels may not apply to reactors where long-lived nuclides may have built up to greater concentrations. In this case, assessments based on the more hazardous nuclides must be used.

TABLE B.18—ACCEPTABLE EMERGENCY BETA-GAMMA ACTIVITIES
IN DRINKING WATER*^{56,57}

Consumption period	Microcuries/cc	Disintegrations per second per cubic centimeter
10 days	9×10^{-2}	3×10^3
30 days	3×10^{-2}	1×10^3

* Determined by balancing the risks of ingesting radioactivity against the risk of doing without food and water for a prolonged period of time. Where contaminated food is expendable, the food should not be used.

Emergency Levels for Gross Fission Product Contamination^{7,22,52}

The values in Table B.18 are based on a maximum dose to the bone of 0.7 rep/day with a maximum of 50 rep as the dose which bone can receive without injury, assuming that all of the fall-out activity is Sr⁸⁹. Although Dunning's calculations¹³ assume the GI tract dose to be dominant until 20 days after detonation, Table B.17 shows that the numbers in Table B.18 would be safe for use in any event for material ingested during the first week following detonation, if we assume an allowable dose of 25 rad to the lower large intestine. (See also Ref. 55.)

Comparison Standards for Testing Food and Water

The University of Rochester Atomic Energy Project has developed for the AEC a suitable emergency comparison standard for use with the ordinary Geiger counter.^{54,56,57} The amount of uranyl acetate necessary to give a response on a Geiger counter equal to that given by mixed fission products from a nuclear detonation was experimentally determined at the 10-day

TABLE B.19— PROPOSED MAXIMUM ALPHA ACTIVITIES FOR INGESTION*^{7,56}

Duration of ingestion	Safe		Acceptable risk	
	Microcuries per cc	dis/min per cc	Microcuries per cc	dis/min per cc
10 days	2×10^{-4}	440	5×10^{-3}	1.1×10^4
30 days	6.7×10^{-5}	145	1.7×10^{-3}	3.7×10^3

* Determined by balancing the risks of ingesting radioactivity against the risk of doing without food and water for a prolonged period of time. Where contaminated food is expendable, the food should not be used.

acceptable risk concentration. This standard should also be roughly applicable to the measurement of fission products from most reactor operations, since, as shown in Ref. 27, p. 79, the doses due to longer-lived fission products such as Sr^{90} would, for a reactor operating time of 180 days with a decay time of 24 hr, not likely be more than a factor of two greater than the Sr^{89} dose on which the comparison standard is based.

The beta comparison standard is prepared as follows:⁷

The standard is prepared in the lid of an ordinary 4-ounce ointment tin (7.9×2.3 cm). The base is used to hold the sample to be tested.

Each standard can be prepared by thoroughly mixing 3.14 g of finely divided (60-mesh) uranyl acetate, $\text{UO}_2(\text{C}_2\text{H}_3\text{O}_2)_2 \cdot 2\text{H}_2\text{O}$, in 5.0 g of liquid casting plastic. Other uranium compounds such as oxides and salts can be employed in amounts proportional to the desired uranium content. The amount of uranium required can be obtained from ordinary chemical supply houses without an AEC license. Both the plastic and catalyst, used to hasten its setting, can be obtained in small quantities from hobby shops. Ward's "Bioplastic" is one material that has proven useful. In a crisis, asphalt might even be used.

When thoroughly mixed with a glass stirring rod, add 6 to 8 drops of the catalyst and stir the mixture for 5 min. After swabbing the inside surfaces of the ointment tin with CCl_4 to remove any adhering grease film, then weigh 8.14 g of the mixture (or other appropriate weights for salts other than uranyl acetate) into the lid of the standard 4-ounce tin. Heat the lid at low heat on a hot plate until the plate begins to harden, then remove. Overheating will cause the plastic to separate from the lid.

Comparison between the standard and food and water samples may be achieved by filling the ointment tin with a sample to within the same distance from the top as the uranium standard in the lid.

For fresh fission products from a nuclear detonation, food and water samples reading less than the standard can be ingested for 10 days, starting within the first 30 days after detonation. For fission products from a reactor accident, a sample reading less than the standard could be ingested for 10 days starting within a few days after the accident.

For testing food and water for alpha contamination, a Landsverk Electrometer calibrated with a Po^{210} source is satisfactory. The 10-day acceptable risk level for alpha emitters in food and water for 10-day consumption is 11,000 dis/min/cc.

Canada Criteria

For emergencies only, the British²⁸ have set ingestion limits of 150 microcuries of Ba^{140} and 30 microcuries of Sr^{89} in a fresh mixture of fission products, ingested over a 10-day period to give 5r to the bones; 120 microcuries of separated Sr^{89} and 200 microcuries of separated Ba^{140} can also be used. The level for Sr^{90} was 1 microcurie, but for reactors of short operating times the low relative concentration of Sr^{90} in fresh fission products makes Sr^{89} and Ba^{140} concentrations the dominant considerations.

To give levels for drinking water that may be easily measurable with a portable survey meter, G. W. C. Tait^{20,28} has obtained the values in Table B.20 for emergency drinking for a 10-day period after an explosion releasing fission products, based on the British levels of intake. The levels in Table B.20 take into account all modes of internal exposure.

TABLE B.20—EMERGENCY DRINKING LEVELS (10 DAYS)
(within two weeks after detonation)

Body of water	Time since burst			
	12 hr	1 day	2 days	10 days
On lake, far from shore	100 mr/hr	50 mr/hr	25 mr/hr	12 mr/hr
On lake, close to surface, water over 2 ft deep, arms length from shore	50	25	12	6
150 to 1000 gallon tank, in contact with center of one surface	50	25	12	6
Water can, 2 to 4 Canadian gallon, at side	25	12	6	3

30 NEUTRON DOSIMETRY FROM HOMOGENEOUS REACTOR ACCIDENT^{37,152,158}

The following estimates of neutron dose and gamma dose were made from mock-up experiments after the Y-12 criticality accident. The source of the accident was a 55 gallon drum, 21.75 in. inside diameter, filled to a height of about 9 in. with 180 liters of liquid containing 2.1 kg of U²³⁵, mostly as uranyl nitrate. The estimates of dose for the highest-exposed employee were:

Na ²⁴ (microcuries/cc blood)	First collision neutron dose (rad)	First collision gamma dose (rad)	Total estimated RBE dose (rem)
5.8×10^{-4}	96	269	461

An RBE of 2 was assumed for acute fast neutron dose. Gold foil measurements showed that the thermal neutron dose was less than 1 per cent of the fast neutron dose. The spectrum measured by threshold detectors on the mock-up is given in Table B.21.

TABLE B.21—NEUTRON SPECTRUM FROM HOMOGENEOUS
REACTOR ACCIDENT

Energy range	Neutrons/cm ²
Thermal	0.90×10^{11}
Total fast	2.0×10^{11}
5 kev to 0.75 Mev	0.25×10^{11}
0.75 Mev to 1.5 Mev	0.70×10^{11}
1.5 Mev to 2.5 Mev	0.58×10^{11}
2.5 Mev and up	0.50×10^{11}

31 RARE EARTHS

Lanthanum through samarium concentrate in the liver, europium through gadolinium concentrate in the liver and skeleton, and terbium through lutetium concentrate mainly in the skeleton. The biological half-life in liver is about 15 days; in bone it is greater than 2.5 years.

TABLE B.22—REACTOR ACCIDENT LEVELS

Category	Full fission product release (curie-sec/m ³)	Volatile fission product release (curie-sec/m ³)
Cloud activity exposure		
A—Lethal exposure	> 400	> 350
B—Illness likely	400-90	350-80
C—Injury unlikely, but may incur some expense	90-10	80-10
D—No injury or expense	< 10	< 10
Ground deposit exposure		
	(curie/m ²)	(curie/m ²)
I—Urgent evacuation necessary (within 12 hr)	> 0.2	> 0.1*
II—Evacuation necessary	> 10 ⁻²	> 0.1
III—Severe restrictions on land use, possible tem- porary evacuation, re- strictions on outdoor work	10 ⁻² -10 ⁻³	0.1-10 ⁻²
IV—Probable destruction of standing crops, restric- tions on agriculture for first year	10 ⁻³ -10 ⁻⁴	10 ⁻² -10 ⁻³
V—No expense likely	< 10 ⁻⁴	< 10 ⁻³

* Unless adequate shelter is available.

33 RULES OF THUMB

Alpha Particles

- (1) An alpha particle of 7.5 Mev just penetrates the 0.07 mm minimum protective layer of skin.
- (2) Plutonium has an alpha activity of about 140,000 alphas/minute microgram; natural uranium activity is 1.5 alphas/minute per microgram; U²³⁸ emits about 0.741 alpha/minute per microgram; Th²³² emits 0.247 alpha/minute per microgram; and Np²³⁷ emits 1519 alphas/minute per microgram.¹⁵⁷
- (3) Self-absorption of a thin layer of alpha-emitting deposit may be estimated by the equation

$$f_a = \frac{T}{2R},$$

where f_a is the fraction of alphas emitted in the upward 2π direction that are absorbed within the deposit, T is the thickness of the deposit, and R is the "effective" range of the alpha particles in the deposit (in the same units as T). The "effective" range is the total range minus the residual range necessary to produce a count. Values of R used for counting alphas in a methane-flow proportional counter are: 8.6 mg/cm² for U²³⁸ alpha particles of 4.2 Mev in U₃O₈; 8.6 mg/cm² for Th²³² alpha particles of 4.1 Mev in ThO₂; and 10.6 mg/cm² for Np²³⁷ alpha particles in NpO₂.¹⁵⁷

Beta Particles

- (1) A beta particle must have at least 70 kev energy to penetrate the 0.07 mm protective layer of skin.

(2) The range, R , in grams/cm² is roughly equal to the maximum energy, E , in Mev divided by 2:

$$R = \frac{E}{2}$$

(3) The range of beta particles in air is about 12 ft per Mev..

(4) The air dose rate at 1 ft from a beta point source is about 200 C rads/hr neglecting self-absorption and air absorption, where C is the number of curies. Variation with energy is small for most beta emitters.

(5) Geiger-Müller tube survey meters calibrated with gamma radiation in mr/hr must be multiplied by two to measure beta radiation in mrad/hr with the window open.

(6) Beta-ray surface dose rates for several materials are given in Table B.23.

TABLE B.23—BETA-RAY SURFACE DOSE RATES¹³¹
(through 7 mg/cm²)

Material	mrads/hour
Thorium (4 to 5 yr after separation)	40
Tuballoy (D-38)	200
Oralloy (40%)	180
(93%)	140
Plutonium-239 (Ni-coated)	360
(uncoated)	440
U ²³³ (1-month U ²³² build-up)	7,000
(1-yr U ²³² build-up)	58,000
Uranium slug (natural)	233
UO ₂ (brown oxide)	207
UF ₄ (green salt)	179
UO ₂ (NO ₃) ₂ · 6H ₂ O	111
UO ₃ (orange oxide)	204
U ₃ O ₈ (black oxide)	203
UO ₂ F ₂	176
Na ₂ U ₂ O ₇	167

(7) The fall-out beta dose to the skin from contact with deposited fall-out within 200 days after detonation is up to 150 times the gamma dose rate in rad/hour at 3 ft above ground. (Project 37.2, Operation Teapot, May 1955.)

(8) The dose to a 30-g adult thyroid from an intake of 1 microcurie of I¹³¹ is about 1 rad delivered within several weeks after intake.

(9) The beta activity from fission products produced in a short criticality burst is: Beta activity = $2.94 t^{-1.2}$ curies per watt-sec, where t is the time since the burst in seconds from 10 seconds to 100 days. For a weapon in the kiloton-TNT range, this becomes: Beta activity = $1.17 \times 10^{13} t^{-1.2}$ curie/kiloton-TNT.

Bremsstrahlung

(1) The energy radiated as x-radiation per beta-ray absorbed is:¹⁶¹

$$B = 1.23 \times 10^{-4} (\bar{Z} + 3) E^2, \text{ Mev/beta,}$$

where E is the maximum beta energy in Mev, and $\bar{Z}$ is the effective atomic number given by

$$\bar{Z} = \frac{\sum f_a Z_a^2}{\sum f_a Z_a},$$

where f_a is the fraction of the number of atoms of atomic number Z_a . The bremsstrahlung spectrum is given in Table B.24.

Table B.24—BREMSSTRAHLUNG SPECTRUM FROM BETA ABSORPTION¹⁶¹

Photon energy intervals in fractions of the maximum beta energy	Per cent of total intensity contributed by photons in energy intervals
0 to 0.1	43.5
0.1 to 0.2	25.8
0.2 to 0.3	15.2
0.3 to 0.4	8.3
0.4 to 0.5	4.3
0.5 to 0.6	2.0
0.6 to 0.7	0.7
0.7 to 0.8	0.2
0.8 to 0.9	0.03
0.9 to 1.0	< 0.01

(2) When beta particles from a 1-curie source of Sr^{90} - Y^{90} are absorbed in aluminum, the bremsstrahlung intensity is approximately equal to the gamma intensity from 12 milligrams of radium. The average bremsstrahlung energy is about 300 kev (Haybittle, Phys. in Med. Biol. 1(3) 270 (1956)).

(3) The bremsstrahlung from a 1-curie P^{32} aqueous solution in a glass bottle is about 3 mr/hour at 1 meter.

TABLE B.25—DOSE RATE VERSUS DISTANCE FROM 100 CURIES Co^{60}

Distance (ft)	Dose rate (r/hr)
1	1,500
10	15
50	0.60
100	0.15
400	0.0075
800	0.0012
1000	0.0006

Gamma Radiation

(1) The gamma-ray dose rates from various radioisotopes are given in Table B.5.

(2) As a rule of thumb, accurate to ± 12 per cent from 0.07 to 2.0 Mev, the dose rate at 1 ft from a point source of gamma radiation is: r/hr at 1 ft = 5.64 CE , where C is the number of curies of the parent nuclide, and E is the total gamma energy in Mev emitted per disintegration of the parent. Rounding off, the dose-rate to within ± 20 per cent from 0.07 to 4 Mev is r/hr at 1 ft = 6 CE .

(3) The dose rate versus distance from a 100-curie Co^{60} source, taking into account inverse-square attenuation, air absorption, and build-up factor for air at normal temperature and pressure, is given in Table B.25.

(4) The gamma activity of fission products produced in a nuclear burst may be expressed in equivalent curies according to the equation:

$$\text{Gamma activity} = 1.82 t^{-1.2} \text{ curies per watt-sec,}$$

where t is the time in seconds since the burst. This equation holds approximately between 10 seconds and 100 days following the burst. For kiloton-TNT size bursts, this equation becomes

$$\text{Gamma activity} = 7.3 \times 10^{12} t^{-1.2} \text{ curies/kiloton-TNT}$$

(5) A deposition of 1 microcurie per square meter of I^{131} on grass gives a gamma radiation level of about 0.003 mr/hr near ground level and results in about 0.1 microcurie per liter of milk from cows grazing in the area. A deposition of 1 megacurie of fission products per square mile gives a gamma dose rate of about 4 r/hr at 3 ft above ground. A deposition of 1 microcurie of fission products per square meter gives 10.6 microrentgens per hour at 3 ft above ground.

(6) The dose rate in a foxhole from air-scattered radiation from a fall-out field is 2 per cent of the open field dose rate per steradian of the sky viewed from the foxhole.¹⁷⁶

(7) The back-scattered intensity of Co^{60} gamma radiation from a thick wall per square meter of the wall visible at both source and detector may be approximated by the equation

$$\text{milliroentgens per hour per square meter for 1 kilocurie } Co^{60} = \frac{4 \times 10^3}{D^2 d^2}$$

where D is the distance in meters of the source from the wall and d is the distance in meters at which the back-scattered dose rate is being measured. It is assumed that D and d are large compared to the dimensions of the irradiated area of the wall and that the gamma rays are scattered at angles greater than about 140° . For example, if a 10-kilocurie Co^{60} source in well-shielded collimator 10 meters from a thick concrete wall irradiated the wall with a beam area at the wall of four square meters, the dose rate at a point beside the collimator would be about 16 mr/hr from radiation scattered back from the wall.

34 SHELTER SHIELDING

The added shielding against fall-out radiation to give an additional factor of five to a nominal basement shelter factor of twenty, or a total factor of one hundred, is given in Table B.26.²⁵ The dose rate in a foxhole due to air-scattered radiation from a fall-out field is 2 per cent of the open field dose rate per steradian of sky viewed from the foxhole.¹⁷⁶ For attenuation factors in various shields see Figs. 1 to 4 and the section on Shield Transmissions.

TABLE B.26—SHELTER SHIELDING MATERIALS FOR ONE-FIFTH REDUCTION

Material	Density (lb/ft ³)	Thickness (in.)
Wood (birch, oak, maple, etc.)	40	17.5
Earth (loose)	75	9.3
Sand (dry)	100	7.0
Brick (common)	110	6.4
Concrete block (solid)	140	5.0

35 SHIELD TRANSMISSIONS

Gamma Rays

Tables B.27 and B.28 (broad-beam transmissions) have been abstracted from curves or data in Refs. 94 to 97, and 131. See also Figs. 1 to 4.

Neutrons

Figures 3 and 4 are representative curves of broad-beam dose attenuation of neutrons by soil and concrete.¹⁴² The attenuated dose includes contributions from secondary gamma radiation.

For crude approximations, the half-thickness of concrete for attenuation of pile-neutrons or fast neutrons up to 15 Mev may be taken to be 3 in. The half-thickness of water is about 10 per cent less than that of concrete. Paraffin half-thicknesses for 4 to 5 Mev (e.g., Po-Be, Ra-Be, or Pu-Be), and 1-Mev neutrons (fission), are 2.73 in. and 1.26 in., respectively. (See Refs. 97-99, 134, 139, 142, and 160.)

TABLE B.27—SHIELD THICKNESS VERSUS GAMMA DOSE TRANSMISSION

		Shield thickness (inches)			
Broad-beam Transmission	Concrete* (147 lb/cu ft)	Iron	Lead		
Radium (11 principal gammas, 0.24 to 2.20 Mev)					
0.1	10	3.1	1.6		
0.01	19	6.2	3.5		
0.001	28	9.1	5.5		
0.0001	38	12	7.8		
0.00001	47	15.3	10.2		
Cobalt-60 (1.33 + 1.17 Mev per disintegration)					
0.1	11	3.2	1.7		
0.01	19	6.0	3.3		
0.001	27	8.8	4.8		
0.0001	35	11.4	6.5		
0.00001	43	14.6	8.1		
Cesium-137 (0.66 Mev)					
0.1	8.5	2.6	0.85		
0.01	15	4.7	1.7		
0.001	22	6.8	2.5		
0.0001	28	8.9	3.4		
0.00001	34	11	4.2		
Iridium-192 (Gammas from 0.13 to 0.87 Mev, averaging --: 0.3 Mev)					
0.1	7	--	0.48		
0.01	13	--	1.1		
0.001	18.3	--	1.9		
0.0001	24	--	2.6		
0.00001	30	--	3.5		
Gold-198 (Au ¹⁹⁸) (0.41-, 0.68-, and 1.1-Mev gammas)					
0.1	6.6		0.35		
0.01	12		0.83		
0.001	17.4		1.7		
0.0001	22.6		2.8		
0.00001	28		4.3		
Barium-140 + Lanthanum-140 (0.030 to 2.5 Mev, averaging about 1.6 Mev) ¹³¹					
	Water	Aluminum	Iron	Lead	Uranium
0.1	25	9.8	3.4	1.6	0.87
0.01	44	18	6.4	3.4	2.0
0.001	64	27	9.2	5.2	3.0
0.0001	81	35	11.8	7.1	4.1
0.00001	104	44	14.8	9.0	5.2
10 ⁻⁶	--	51	--	--	6.3
Iodine-131 (0.08 to 0.723 Mev, predominantly 0.36 Mev)					
Concrete					
0.1	6				
0.01	12				
0.001	18				

*NOTE: After several mean free paths, each ten inches of concrete reduces the radiation by another factor of ten.

Fig. 1 — Transmission of fission product gamma radiation in several shield materials.

Fig. 2—Transmission of prompt gamma radiation in several shield materials.

Fig. 3— Transmission of total dose rate from neutrons incident on soil.

Fig. 4—Transmission of total dose rate from neutrons incident on 10 per cent moist soil and concrete.

TABLE B.28 — HALF-THICKNESS VERSUS PHOTON ENERGY⁸⁵
FOR SEVERAL MATERIALS

Photon energy (Mev)	Half-thickness (inches)			
	Water	Concrete	Iron	Lead
0.2	2.0	0.8	0.28	0.06
0.5	3.0	1.1	0.45	0.17
1	4.1	1.7	0.63	0.35
1.5	4.7	2.0	0.70	0.46
2	5.7	2.3	0.82	0.53
2.5	6.8	2.7	0.88	0.57
3	7.7	3.0	0.92	0.59
4	8.4	3.3	1.02	0.59
5	9.2	3.7	1.10	0.58

36 SPECIFIC ACTIVITIES AND HALF LIVES

TABLE B.29

Isotope	Specific activity	Half life
Natural uranium	1.4 dis/min/microgram	
Uranium-238	(0.741 dis/min/microgram)	
	3.36×10^{-7} curies/g	4.49×10^9 years
Uranium-235	2.14×10^{-8} curies/g	7.13×10^8 years
Plutonium-239	6.17×10^{-2} curies/g	2.44×10^4 years
Carbon-14	4.61 curies/g	5568 years
Cesium-137	79.4 curies/g	33 years
Strontium-90	2.00×10^2 curies/g	19.9 years
Tritium, H ³	9.60×10^3 curies/g	12.46 years
Cobalt-60	1.14×10^3 curies/g	5.27 years
Ruthenium-106	3.39×10^3 curies/g	1.0 year
Cerium-144	3.22×10^3 curies/g	282 days
Praseodymium-144	7.48×10^7 curies/g	17.5 minutes
Polonium-210	4.50×10^3 curies/g	138.3 days
Sulfur-35	4.28×10^4 curies/g	87.1 days
Strontium-89	2.77×10^4 curies/g	53 days
Phosphorus-32	2.85×10^5 curies/g	14.3 days
Barium-140	7.30×10^4 curies/g	12.8 days
Iodine-131	1.23×10^5 curies/g	8.14 days
Gold-198	2.44×10^5 curies/g	2.7 days
Yttrium-90	5.72×10^5 curies/g	61 hr
Lanthanum-140	5.61×10^5 curies/g	40 hr
Sodium-24	8.69×10^6 curies/g	15.06 hr
Potassium-42	6.01×10^6 curies/g	12.44 hr
Nitrogen-16	9.60×10^{10} curies/g	7.35 sec

37 STANDARD MAN

TABLE B.30 — CHARACTERISTICS OF THE STANDARD MAN⁸³

Water intake in food	1000 cc/day
Water intake in food and fluids	2200 cc/day
Oxidation	300 cc/day
Total water intake per day	2500 cc/day

TABLE B.30—(Continued)

Water excretion from lungs	300 cc/day
Water excretion by faeces	200 cc/day
Water excretion by urine	1400 cc/day
Water excretion by sweat	600 cc/day
Total water excretion per day	2500 cc/day
Air inhaled during 8-hr work day	10^7 cc/day
Air inhaled during 16 hr not at work	10^7 cc/day
Total air inhaled per day	2×10^7 cc/day
Vital capacity of lungs	3 to 4 liters (men) 2 to 3 liters (women)
Total surface area of respiratory tract	70 m^2
Total water in body	43 kg
Average life span of man	70 years
Occupational exposure of man: 8 hr/day, 40 hr/week, 50 weeks/year	
Minimum thickness of epidermis—0.07 mm (7 mg/cm^2), and depth of blood-forming organs—5 cm.	

TABLE B.31—MASSES OF CRITICAL ORGANS⁵¹

Organ	Mass (g)
Total body	70,000
Muscles	30,000
Fat	10,000
Bone	7,000
Blood	5,400
Skin	2,000
Liver	1,700
Lungs	1,000
Kidneys	300
Spleen	150
Thyroid	20

TABLE B.32—PARTICULATE RETENTION IN THE RESPIRATORY TRACT^{51,93}

Distribution	Readily soluble compounds, %	Other compounds, %
Exhaled	25	25
Deposited in upper respiratory passage and swallowed	50	50
Deposited in the lungs (lower respiratory passages)	25 (this is taken up into the body)	25*

*Half of this is eliminated from the lungs and swallowed in the first 24 hr, making a total of $62\frac{1}{2}$ per cent swallowed. The remaining $12\frac{1}{2}$ per cent is retained in the lungs with a half life of 120 days; it being assumed that this portion is taken up into body fluids.

38 STRONTIUM-90 IN SOIL

It is estimated that the average concentration of Sr^{90} in the U. S. diet in 1958 was about 5 to 10 μc per kilogram of food (grain, vegetable, or milk).¹⁸¹ The corresponding average concentration in soil in 1958 was about 30 mc per square mile.¹⁸² However, this comparison

exaggerates the contribution due to Sr^{90} actually in the soil, since some of the Sr^{90} in food was due to retention of direct fall-out on the vegetation itself. The annual calcium intake is about 400 grams per person. The total annual Sr^{90} intake for 1958 was 3500 to 6000 $\mu\mu\text{c}$ per person.

For comparison, the NCRP¹⁸⁰ gives a concentration of 1000 $\mu\mu\text{c}$ Sr^{90} per kilogram of water as a permissible level in water for continuous occupational exposure during a 50-year period, based on the total daily intake of water in food and drink, 2200 ml.

39 WEAPON-PRODUCED RADIATION²²

Total Gamma Activity^{22, 129}

The total gamma radiation activity of fission products from a 1-megaton (1000-kiloton*) explosion is given in Table B.33.

TABLE B.33—TOTAL GAMMA ACTIVITY OF
FISSION PRODUCTS VERSUS TIME
FOR A 1-MEGATON EXPLOSION

Time after explosion	Activity (curies)
1 hour	3×10^{11}
1 day	6.6×10^9
1 week	6.4×10^8
1 month	1.1×10^8
1 year	5.5×10^6

NOTE: As a rough rule of thumb, an increase in time since the explosion by a factor of seven, allows decay by a factor of ten in fission product activity from an explosion or short-duration criticality burst.

Residual Contamination from Fall-out

An idea of the magnitude and possible extent of contamination from a nuclear explosion in the kiloton range may be obtained from Table B.34, which gives the parameters of an idealized fall-out isodose rate pattern assuming 15-mile-per-hour uniform winds and extrapolating back to the dose rates that would have existed at 1 hr if all of the material to be deposited had fallen

TABLE B.34—IDEALIZED 1-HR REFERENCE DOSE RATE CONTOURS
FOR 20-KILOTON SURFACE BURST AND 15 MPH WIND

Dose rate (r/hr)	Radius of GZ circle (miles)	Downwind displacement of center of GZ circle (miles)	Maximum downwind distance (miles)	Maximum crosswind width (miles)
3,000	0.10	0.08	1.0	0.3
1,000	0.22	0.14	2.3	0.7
300	0.41	0.22	5.3	1.2
100	0.66	0.28	11.5	1.8
30	0.95	0.36	22.0	2.8
10	1.4	0.42	50.0	5.1

*NOTE: A 1-kiloton TNT-equivalent fission bomb releases energy equivalent to: (a) The energy released by the fission of about 0.11 lb or 50 g U^{235} , i.e., 1.25×10^{23} fissioned atoms (1.34×10^{23} in Ref. 129); (b) About 2.4×10^{25} Mev energy released; (c) 4×10^6 Mw-seconds of energy.

out by that time. (Of course, at 1 hr the fission products would not yet have fallen out at distances greater than 15 miles.) (These are average representative values for planning purposes only, since the actual extent of contamination in a given circumstance must be determined by surveys.)

The dispersal of fall-out from megaton-size weapon bursts is determined by the direction and speed of winds up to 60,000 ft and above. The directions and speeds may vary considerably with altitude. However, for planning and hazard-estimating purposes, idealized 1-hour-reference contours based on a uniform 15-mile-per-hour wind are given in Table B.35 for a 1-megaton surface burst.

TABLE B.35—IDEALIZED 1-HR REFERENCE DOSE RATE CONTOURS
FOR 1-MEGATON SURFACE BURST AND 15 MPH WIND

Dose rate (r/hr)	Radius of GZ circle (miles)	Downwind displacement of center of GZ circle (miles)	Maximum downwind distance (miles)	Maximum crosswind width (miles)
3,000	0.43	0.60	22	3.1
1,000	1.4	0.80	40	6.8
300	2.8	1.02	70	11.8
100	4.7	1.24	114	16.7
30	7.5	1.46	183	22.8
10	11.0	1.65	317	34.1

Initial Gamma Radiation Dose Versus Distance²²

For explosions of 1 to 10 kilotons TNT-equivalent, the initial gamma dose in roentgens as a function of distance in yards from the explosion is given by

$$I_0 = \frac{1.4 \times 10^9 W}{D^2} e^{-D/338} \text{ roentgens,}$$

where W is the yield in kilotons TNT. For higher yields than 10 kilotons, the rarefaction of the atmosphere by the blast wave increases the range of the gamma rays. Thus for yields greater than 10 kilotons W must be replaced in the above equation by Wk, where k is given as a function of yield in Table B.36.

TABLE B.36—SCALING FACTOR k FOR
INITIAL GAMMA RADIATION DOSAGE

Energy yield	k
100	1.6
1,000	5
10,000	17

The initial gamma radiation is emitted mostly within one second after the explosion and has an average photon energy of roughly 4.5 Mev.

Initial Neutron Dose Versus Distance²²

Since the neutron dose just as the gamma dose depends on weapon design, it is possible to give only a representative equation as an example for planning and hazard-estimating purposes. The neutron dose as a function of distance and yield is

$$N = \frac{8.4 \times 10^9 W}{D^2} e^{-D/242} \text{ rems,}$$

where W is the explosion yield in kilotons and D is the distance from the explosion in yards. In this equation, as in the one for initial gamma radiation dose, the dose build-up factor has been taken into account approximately by selecting a relaxation length representative of actual dose attenuation. The initial neutron radiation is also emitted mostly within the first second following the blast. Most fission neutrons have energies of 1 to 3 Mev before being slowed down. Neutrons produced by the fusion reaction between deuterium and tritium have 14-Mev energy.

Dose Rate Versus Height Above Contaminated Land or Water²²

Table B.37 gives the dose rate versus height above a plane surface uniformly contaminated with 1 megacurie per square mile. Note that the dose rate per megacurie per square mile is roughly proportional to the photon energy.¹²⁹

TABLE B.37—DOSE RATE VERSUS HEIGHT ABOVE CONTAMINATED SURFACE

Height (feet)	Dose rate in roentgens per hour for photon energy of		
	0.7 Mev	1.5 Mev	3.0 Mev
3	4.1	8.5	14.2
6	3.5	7.2	12.3
12	2.7	5.8	10.2
18	2.3	5.1	9.0
24	2.0	4.5	8.2
27	1.8	4.3	7.8

The altitude dose attenuation factors for fission product radiation spread on the ground over an area of dimensions large compared to the altitude are given in Table B.38. (These dose attenuation factors may be compared with the count-rate attenuation factors for the scintillation detector discussed in the paper by F. J. Davis, "Instrumentation in Aircraft for Radiation Measurements.")

TABLE B.38—ALTITUDE ATTENUATION FACTORS FOR FISSION PRODUCT RADIATION

Altitude above ground (feet)	Attenuation factor
	$\frac{(\text{Dose rate at ground})}{(\text{Dose rate at altitude})}$
250	4.5
500	9
1,000	29
1,500	90
2,000	280
2,500	800
3,000	2,600
3,500	7,000
4,000	24,000
4,500	65,000

The dose rates of gamma radiation of several energies above a large body of water having a uniform contamination density of 1 curie per cubic yard are given in Table B.39. The dose rate several gamma mean-free paths within a large volume of water contaminated with 1 curie per cubic yard is slightly more than twice the dose rate at the surface given in Table B.39.

TABLE B.39—DOSE RATE VERSUS HEIGHT ABOVE WATER
CONTAINING 1 CURIE PER CUBIC YARD

Height above surface (feet)	Roentgens/hour for photon energy of		
	0.7 Mev	1.5 Mev	3.0 Mev
0	0.94	2.0	3.7
25	0.85	1.8	3.5
50	0.77	1.7	3.2
100	0.63	1.4	2.7
150	0.54	1.2	2.5
200	0.47	1.1	2.2
250	0.43	0.97	2.0
300	0.38	0.90	1.9

Shielding of Initial and Residual Weapons Radiation²²

For rough estimates of the attenuation of fast neutrons emitted from a fission burst, the following equation may be used:

$$N = N_0 e^{-sx},$$

where N/N_0 is the fractional neutron dose transmitted by the shield, x is the shield thickness in cm, and s is the mean macroscopic cross-section in cm^{-1} . Some representative values of s , which are approximations that take into account dose build-up in broad beams transmitted through thick shields, are given in Table B.40.

The attenuations of initial gamma radiation (average energy about 4.5 Mev per photon) and fission-product residual gamma radiation (average energy about 0.7 Mev) in several shielding materials are given in Figs. 1 and 2. Curves of total dose transmission for fast neutrons incident on concrete and soil are given in Figs. 3 and 4.

TABLE B.40—EMPIRICAL MACROSCOPIC
CROSS-SECTIONS FOR FAST NEUTRONS

Material	s (cm^{-1})
Water	0.1
Concrete	0.09
Iron Concrete	0.16

Radionuclides of Biological Interest

Klement¹⁴⁴ presents data on the abundances of various weapon-produced fission products and induced radionuclides in air, soil, and water. He estimates that the important fission product radionuclides after one-month decay include Sr^{89} , Sr^{90} — Y^{90} , Y^{91} , Zr^{95} — Nb^{95} , I^{131} , Cs^{137} — $\text{Ba}^{137\text{m}}$, Ba^{140} — La^{140} , Ce^{143} — Pr^{143} , Ce^{144} — Pr^{144} , Nd^{147} — Pm^{147} , and possibly isotopes of Co, Zn, Fe, and Mn, including Fe^{55} , Fe^{59} , Co^{57} , Co^{58} , and Co^{60} . Also, residual Pu^{239} is potentially of interest. (The fission products selected agree with those included in Table B.8.)

C^{14} is the only radionuclide produced by neutron absorption in air that appears to be of long-term significance.

Radionuclides produced in soils include P^{32} , Ca^{45} , Fe^{55} , and Fe^{59} . Most of the radionuclides induced in seawater are relatively short-lived. The only ones that appear to have possible significance are S^{35} and Zn^{65} . Klement¹⁴⁴ estimates that weapons detonated near bodies of seawater produce less than 1 curie of Zn^{65} per megaton TNT-equivalent. (See also Ref. 47.)

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