UNION CARBIDE NUCLEAR COMPANY
DIVISION OF UNION CARBIDE CORPORATION

Y-12 PLANT

Contract No. W-7405-eng-26
with the U. S. Government

ACCIDENTAL RADIATION EXCURSION AT THE Y-12 PLANT
June 16, 1958
Final Report

Oak Ridge, Tennessee
July 28, 1958

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PREFACE

This report discusses the radiation accident which occurred at the Y-12 Plant on June 16, 1958. To the extent that information is available, it describes the circumstances leading to the accident, attempts to reconstruct the nuclear reactivity conditions, and reviews the dosimetric means and results which were used to help determine the exposure of affected employees.

Clinical findings and the medical progress of the individuals receiving significant radiation exposures are not included and will be presented by appropriate medical authorities in a separate report. It is appropriate, however, to preface this report with the news that these eight men have been released from the hospital and have resumed their normal activities.

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INTRODUCTION

On Monday, June 16, 1958, an accidental nuclear excursion occurred in an enriched uranium salvage recovery area of the Y-12 Plant. In accordance with the requirements of the Atomic Energy Commission, a committee was appointed to investigate the incident.

This report presents specific information on conditions prior to, during, and following the radiation incident and general information on the prevention of a re-occurrence.

This regrettable accident is believed to be the first nuclear excursion to have occurred in a uranium processing facility. In the hope that it may benefit others engaged in enriched uranium processing, a considerable treatment, above and beyond the Atomic Energy Commission's minimum requirements for such an investigative report, is given.

As might be expected, the events and circumstances associated with an incident of this nature are complex. A principal motivation in many phases of the investigative work was concern over the persons exposed to nuclear radiation and the desirability of arriving at accurate estimates of the radiation doses received. While a considerable effort has been made by the committee and by those who donated their time and talents to develop information considered pertinent, no pretense is made that all questions which might arise have been answered.

INVESTIGATING COMMITTEE

Patton, F. S. - Process Analysis, Y-12 (Chairman)

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CONSULTANTS TO THE COMMITTEE

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SUMMARY

An accidental nuclear excursion occurred in the Y-12 Plant at approximately 2:05 p. m. on Monday, June 16, 1958. The following remarks summarize information obtained by the committee appointed to investigate the accident:

1. The site of the accidental nuclear excursion was a 55-gallon stainless steel drum located in the C-1 Wing of Building 9212. Figure 1 is a photograph of this drum taken approximately 24 hours after the incident. Its location is referenced in progressively greater detail in Figures 2, 3, and 4.

2. On the basis of the available data, the following sequence of events is postulated as leading to the incident:

    A portion of enriched ($\sim$90% U^{235}) uranium-bearing solution, containing approximately 50 gm U^{235}/liter, flowed through a valved pipeline from an extraction product "safe" tank in B-1 Wing into C-1 Wing and partially filled "safe" tank 1-2 as well as the piping connecting tanks 1-2, 6-1, and 6-2. This piping arrangement is schematically presented in Figures 6 and 10.

    Subsequent to this inadvertent transfer, tanks 6-1 and 6-2 were partially filled with water for purposes of routine leak testing following the monthly inventory clean-out.

    When the valve on the drain line leading to the drum shown in Figure 1 was opened, the enriched uranium solution in tank 1-2 and the connecting piping preceded the water from tanks 6-1 and 6-2 into the drum causing the incident.

3. Following the initial nuclear burst, which did not discharge the contents of the drum, the nuclear system appears to have oscillated. The reaction was ultimately stopped by the additional water flowing into the drum. Based upon an examination of the chart taken from a recording monitor located in another building and other indicative information (Figure 13), it is believed that the nuclear reaction lasted approximately twenty minutes.

4. Upon the sounding of the radiation monitor alarm siren, plant emergency procedures were put into effect. Descriptions of the evacuation and the activities of UCNC emergency personnel from ORNL, ORGDP, and Y-12 are presented in Exhibit III.

    By 5:00 p. m. of June 16, radiation survey teams established that the incident had in fact taken place in a drum located in C-1 Wing of Building 9212. At approximately 9:30 p. m., the drum was poisoned by the insertion of a cadmium scroll. Clean-up of all Building 9212 areas except C and C-1 Wings was begun during the night of June 16. During the night of June 17, a "safe" tankage facility was fabricated and installed in one of the Building 9212 shielded radiograph cells (see Figure 3), and the contents of the drum were transferred to this improvised storage site during the afternoon of June 18. The empty drum was then transported to ORNL for analysis.

    Clean-up activities were continued, and by the morning of June 19, all recovery facilities with the exception of those in the central and east portion of C-1 Wing were put back in operation.

    In the afternoon of June 20, a team consisting of members of the investigating committee, UCNC operations, and development supervision moved into C-1 Wing and

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carried out a program of dismantling, sampling, inspection, and hydraulic testing. As of June 23, after all available raw data had been gathered to the satisfaction of the investigating committee, all recovery facilities were returned to normal operations.

5. Eight Y-12 employees were in the vicinity of the drum at the time of the incident. The five men exposed to what has been described as a medium dose of radiation by Dr. Marshall Brucer, Chairman, Medical Division, Oak Ridge Institute of Nuclear Studies, were:

> Employee "A", Chemical Operator - Age 40; 365 rad - 461 rem*
>
> Employee "B", Electrician - Age 32; 270 rad - 341 rem*
>
> Employee "C", Maintenance Mechanic - Age 39; 339 rad - 428 rem*
>
> Employee "D", Electrician - Age 51; 327 rad - 413 rem*
>
> Employee "E", Maintenance Mechanic - Age 35; 236 rad - 298 rem*

The positions of these men and their routes of exit from the area of the incident are portrayed in Figures 4 and 5.

The three men exposed to a lower dose of radiation were:

> Employee "F", Welder - Age 41; 68.5 rad - 86.5 rem*
>
> Employee "G", Maintenance Mechanic - Age 56; 68.5 rad - 86.5 rem*
>
> Employee "H", Chemical Operator - Age 25; 22.8 rad - 28.8 rem*

Following the accident, these men were hospitalized at the Oak Ridge Institute of Nuclear Studies where specialized medical attention was provided. Employees "F", "G", and "H" were released from the hospital on June 26, 1958, and allowed to resume their normal activities. Employees "A", "B", "C", "D", and "E" were released on July 30, 1958.

6. The neutron and gamma radiation of personnel whose indium foil badges indicated significant exposure was determined by measuring the Na^{24} in the bodies of those exposed. This was done in two ways: (a) by counting blood samples, and (b) by counting the total body in a whole body counter. The neutron and gamma doses measured in a mock-up of the excursion, carried out in the ORNL Critical Experiments Laboratory on June 18, provided necessary data to which the Na^{24} values could be related.

The evaluation of evidence pertaining to the exposure of personnel is presented in Exhibit V.

7. Although it is unlikely that any future accidental nuclear excursion would exactly duplicate the incident sustained at the Y-12 Plant, there are certain aspects which would be common to all incidents. In the interest of attaining an adequate coverage of such items, a number of appendices which support the main body of the report have been incorporated.

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* Estimates taken from Table X, "Sodium Activation and Dose Estimates for Exposed Personnel," first collision total dose in rads and estimated RBE dose in rem, with an assumed RBE = 2 for fast neutron dose.
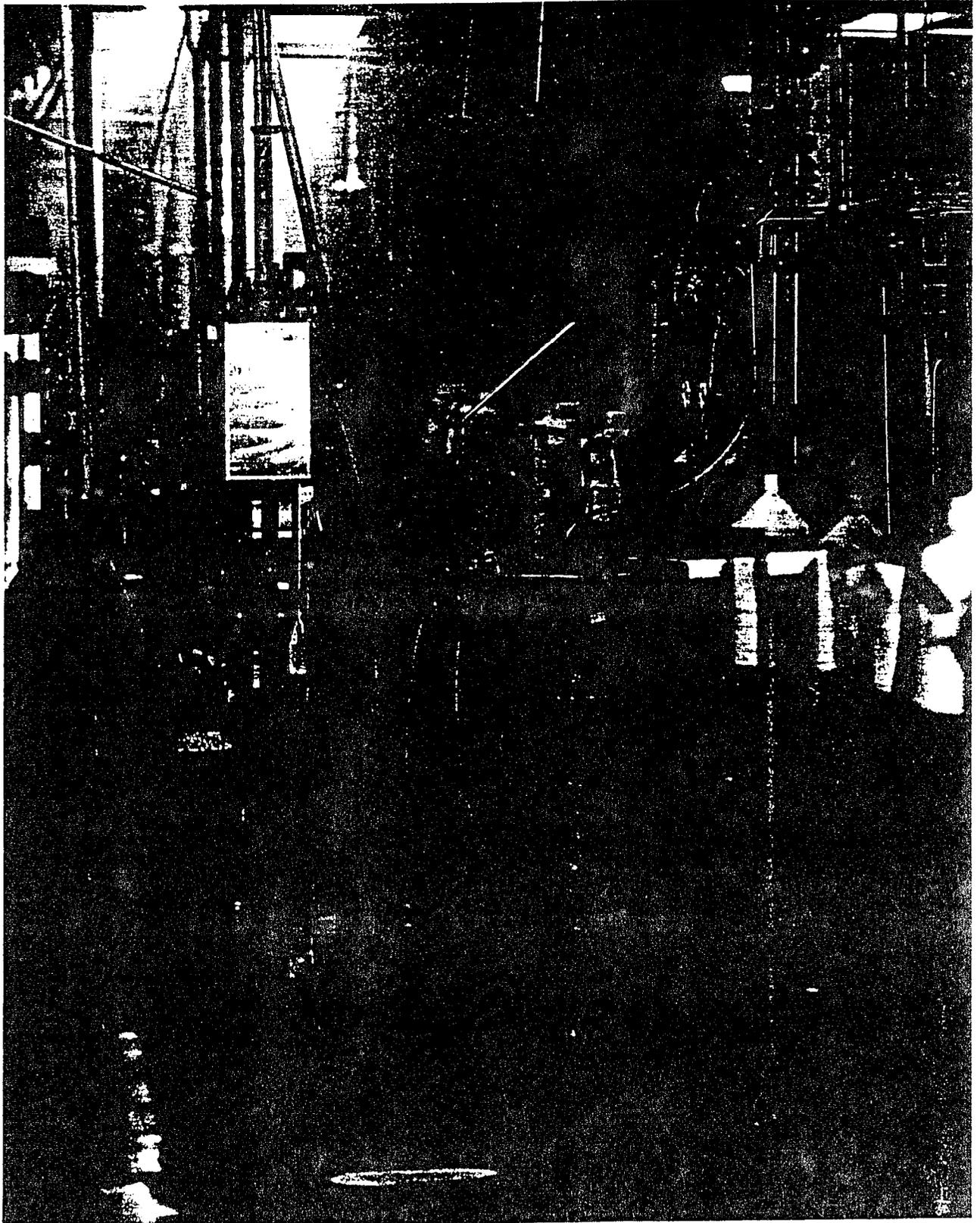
Figure 1
ACTUAL 55 GALLON DRUM IN WHICH THE CRITICAL INCIDENT OCCURRED
(Photograph Taken Approximately 24 Hours After Critical Incident)
Looking West

GENERAL
WIND DIRECTION →

LOCATION OF
NUCLEAR
EXCURSION

WEST PORTAL

CENTRAL PORTAL

NORTH PORTAL

TRUE NORTH

BEAR CREEK
PORTAL

EAST PORTAL

ANALYTICAL
LABORATORY

BEAR CREEK ROAD

9212

9207

SCC
9723-19

FIRST STREET

9706-2
DIS

SCC
9723-24

9764

9204-3   9204-1

0' 100'   500'   1,000'

LEGEND

DIS - DISPENSARY
● - AIR SAMPLERS
—·— PERIMETER FENCE

★ - EMERGENCY CONTROL CENTER
Ⓧ - PRIMARY ASSEMBLY AREAS
SCC SECONDARY CONTROL CENTER

/--/--/ INITIAL DELIMITATION OF RADIOACTIVITY
---- SECONDARY DELIMITATION OF RADIOACTIVITY
▬▬▬ TERTIARY DELIMITATION OF RADIOACTIVITY

Figure 2
LOCATION OF Y-12 PLANT BUILDINGS PERTINENT TO INCIDENT

REFERENCE ILLUSTRATION: FIGURE 2
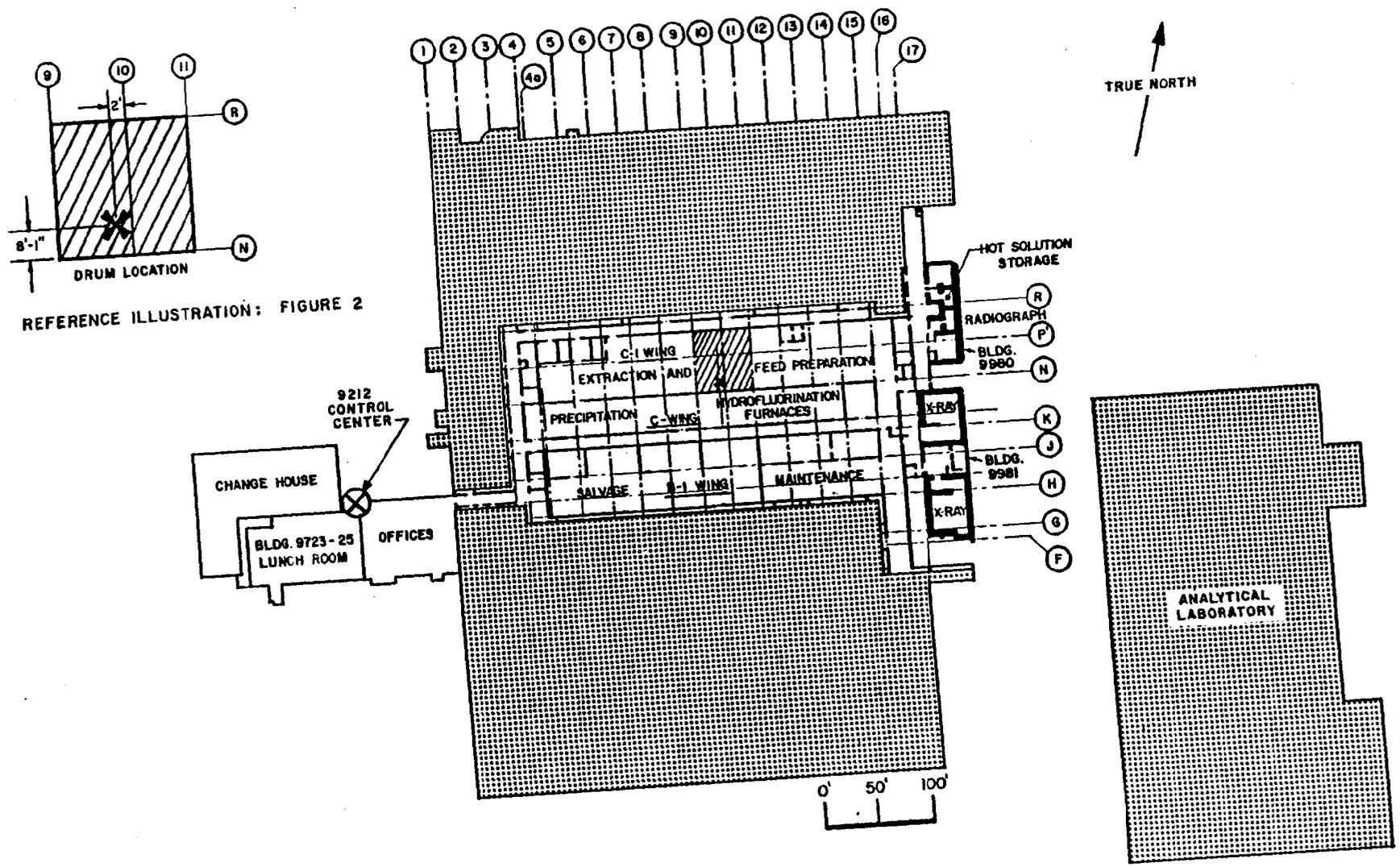
Figure 3
LOCATION OF NUCLEAR EXCURSION

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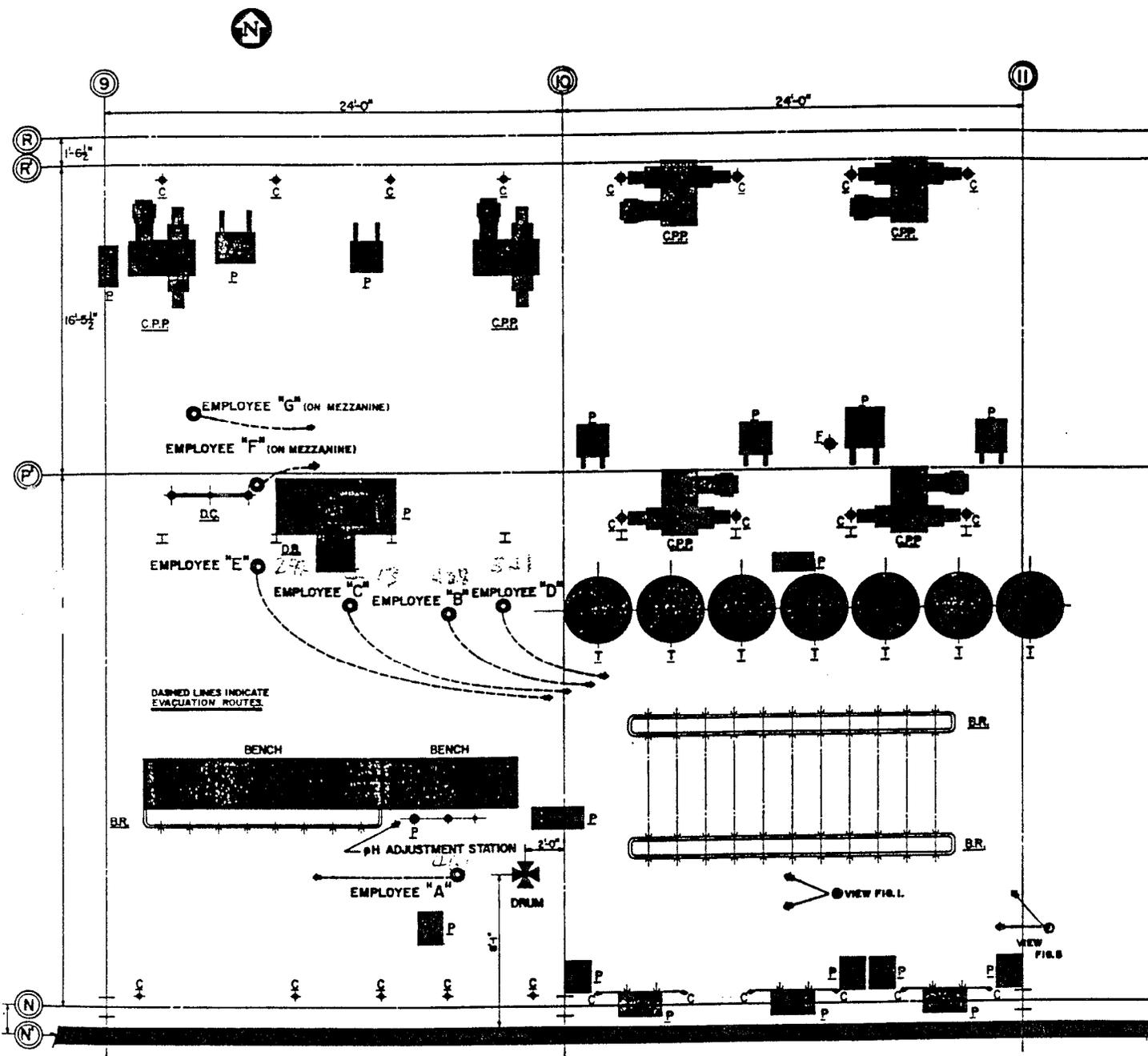
MAIN FLOOR ELEVATION 1015 ft.
MEZZANINE ELEVATION 1027 ft.

NOTE:

EMPLOYEE "H" WAS APPROXIMATELY
50 FT. EAST OF THE DRUM NEAR COLUMN
LINE 12 WHEN THE INCIDENT OCCURRED
AND HE EVACUATED EAST.

LEGEND

C. LIQUID-LIQUID EXTRACTION COLUMN.
T. COLUMN LIQUID STORAGE TANKS.
B.R. ~5" DIA. PLASTIC—BOTTLE RACKS.
F. EXTRACTION COLUMN FILTER.
P. PUMP.
C.P.P. COLUMN PULSE PUMP.
D.B. DRY BOX.
D.C. DISSOLVER COLUMN.
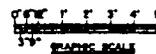
Figure 4
EQUIPMENT LOCATION PLAN IN AREA OF NUCLEAR EXCURSION

DISTANCE FROM CENTER OF REACTANT
SOLUTION TO EMPLOYEE'S BELTLINE
(AT TIME OF FIRST CRITICALITY)

EMPLOYEE		DISTANCE
"A"	365	6'
"B"	270	15'
"C"	339	17'
"D"	327	16'
"E"	236	22'

CENTER OF
REACTANT SOLUTION
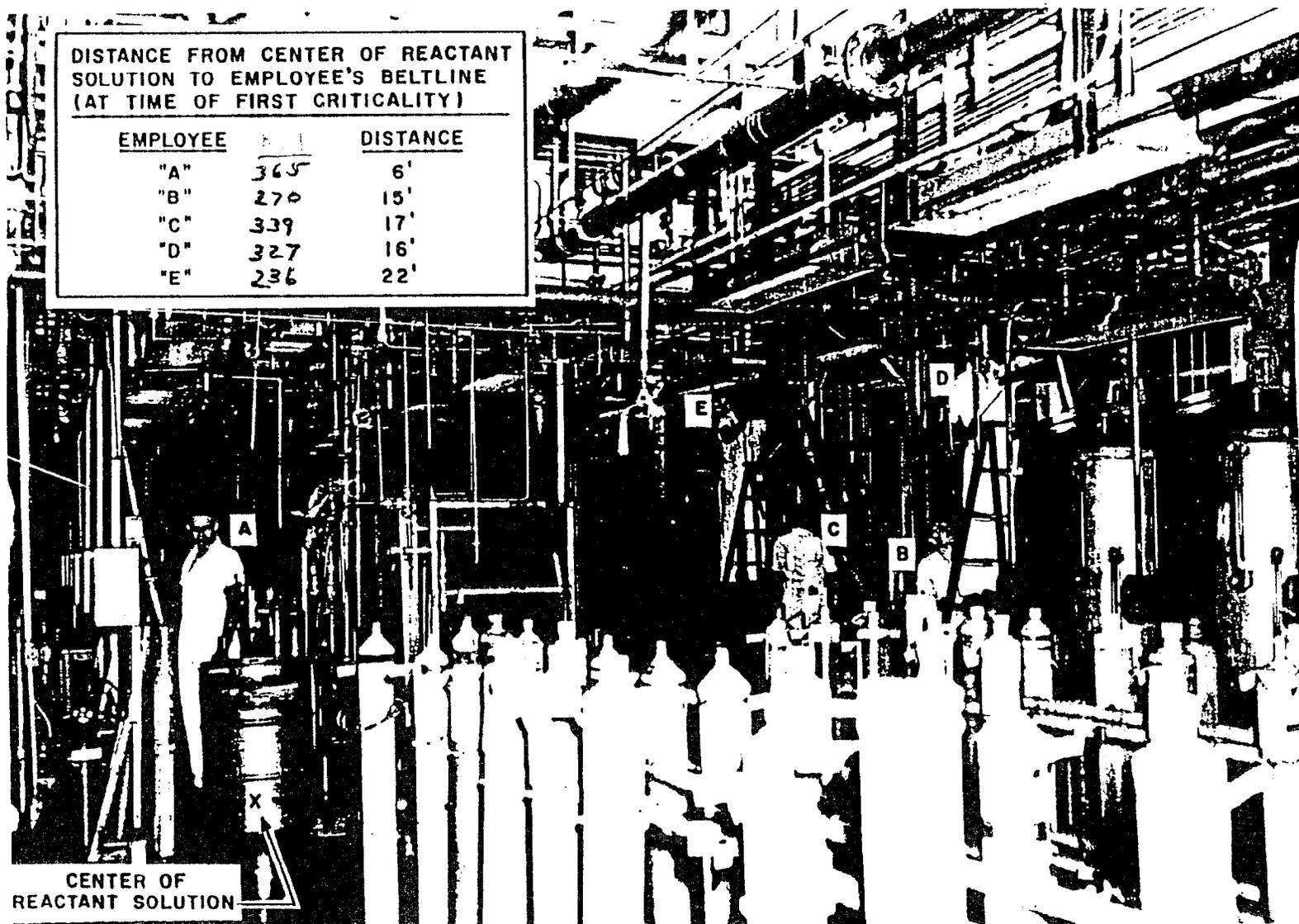
Figure 5
RE-ENACTMENT OF POSITIONS OF EMPLOYEES IN THE PROXIMITY OF THE DRUM
AT THE TIME OF THE INCIDENT
Looking West

FINDINGS

CAUSES OF THE INCIDENT

It is believed that this accident was caused by a number of interdependent contributing circumstances. Although of uneven weight, no single happening can be said to be a principal contributor. Accordingly, these items are listed with no special emphasis on the order of enumeration, commencing with the general and proceeding to the specific.

1. The process phase in which the accident occurred was a temporary arrangement encompassing portions of a new installation in the startup stage (B-1 Wing), and an old installation in the shutdown stage (C-1 and C Wings). This arrangement was necessitated by delays in the activation of new facilities in B-1 Wing for the conversion of uranyl nitrate solution to uranium tetrafluoride.

   This temporary arrangement of old facilities combined with part of a new installation was a compromise between the customary detailed design planning of valving, instrumentation, and other safeguards, and a requirement for maintaining production during this interim phase. Also, the responsibility for the uranyl nitrate to uranium tetrafluoride operation was thereby split among three different supervisors in three physically separated areas, instead of being under a single supervisor as would be the case in the completed B-1 Wing. Communications were considerably complicated by this situation.

2. At the time of the incident the uranium processing areas had been concerned with the required monthly accounting of uranium in inventory, which necessitated a stoppage of operations. However, all operations were not stopped or started at the same time due to the complexity of the installation. The method of taking inventory varied with the form and concentration of the uranium. For example, where equipment contained dilute homogeneous solutions of uranium, a satisfactory accounting could be made by taking samples and computing the contents of known volumes.

   In the process phase wherein the accident occurred, because of the high concentration of the uranium and the tendencies of the solutions to deposit uranium-bearing solids, more precise accounting is obtained by processing the contents of the 5"-diameter "safe" geometry tanks to uranium tetrafluoride just prior to the inventory period. In addition, it was recognized procedure to wash, dismantle, and swab out these 5"-diameter "safe" tanks, collecting the washings in portable plastic "safe" bottles.

   Certain routine duties, such as the mopping of floors and the checking of equipment that has undergone minor maintenance, due to their simple nature and the many variations involved, have not been explicitly detailed in procedures. Instead, over the fourteen years of operation, general rules have been formulated and the task of seeing that routine applications conform to these criteria has been assigned to the process foreman.

   As reassembled "safe" tanks were prone to leak at the tank ends when placed back in service after the monthly inventory cleanup, leak testing of reassembled tanks by filling with water, checking and draining prior to their return to operation, was practiced. Leak testing with water was among the previously mentioned routine duties that were not formalized and were carried out under the discretion and supervision of the process foremen.

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Although this leak testing had considerable utility, as practiced it deviated from the intent of two mandatory area procedural rules by the incorporation of a 55-gallon drum to collect water drained from "safe" tanks after the leak testing. These rules are:

a. Process liquids are never to be transferred from a geometrically "safe" container to a geometrically "unsafe" container.

b. "Unsafe" containers used to collect dilute liquids (such as mop water) must contain a charge of cadmium nitrate (a nuclear poison).

An unfortunate interpretation of the above rules was that they did not apply to the leak testing of the 5"-diameter "safe" tanks, since the tanks were clean and only water was used in the operation.

The significance of the foregoing, with regard to the accident, is that it furnished the mechanism whereby an "unsafe" geometry container (i.e., the 55-gallon drum) was separated from concentrated uranyl nitrate solutions by only a single valve (V-1).

3. The dismantling, cleaning, reassembly, and subsequent leak testing of the C-1 Wing "safe" tanks involved a number of different employees, including both maintenance personnel and chemical operators, and usually required several eight-hour shifts for completion. Under these circumstances, it is evident that good communications were necessary.

The leak testing practice included the following pertinent routine safeguards:

a. The process foreman in charge assures himself, by reference to the operating log and by discussion with the preceding shift foreman, that the tanks to be tested have actually been disassembled, cleaned, and reassembled.

b. The process foreman, either personally or through instructions to his operators, checks all valves connecting the tanks to be tested with other process areas and determines that their position is correct. In addition, the pneumatic liquid level indicators are checked to determine that the tanks are empty.

c. During the draining of the leak test water from the "safe" tanks into a container (i.e., in this case a 55-gallon drum), an operator is stationed adjacent to the container to observe the flow of water, and safeguard against any unusual development.

A simple schematic of the piping arrangement involved in the incident is shown in Figure 6.

Early during the shift preceding the accident (11:00 p.m. Sunday, June 15, to 7:00 a.m. Monday, June 16), the process foreman (Foreman "Y") in charge of C-1 Wing noted that solution (wash water) was present in the 6" glass standpipe of the C-1 Wing pH adjustment station and directed one of the chemical operators to drain this liquid. At 5:00 a.m. Foreman "Y" again noted liquid in the glass standpipe and questioned the forementioned operator as to whether his previous order had been carried out. This operator stated that the standpipe had been drained. Upon investigation, Foreman "Y" found that solution was slowly leaking through valve V-2. Foreman "Y" tightened this valve, stopping the leak. (Figure 7 is a photograph of this standpipe as found after the accident.) Foreman "Y" was aware at the time that the B-1 Wing secondary extraction systems were in operation producing uranyl nitrate product, but believed that the leak testing of the 6-1, 6-2, and 1-2 tanks had been completed on the previous Friday.

The closing of valve V-2 allowed the uranyl nitrate solution, which had been leaking into the pH adjustment station standpipe, to back up into the C-1 Wing "safe" geometry storage tanks (see Figure 6).
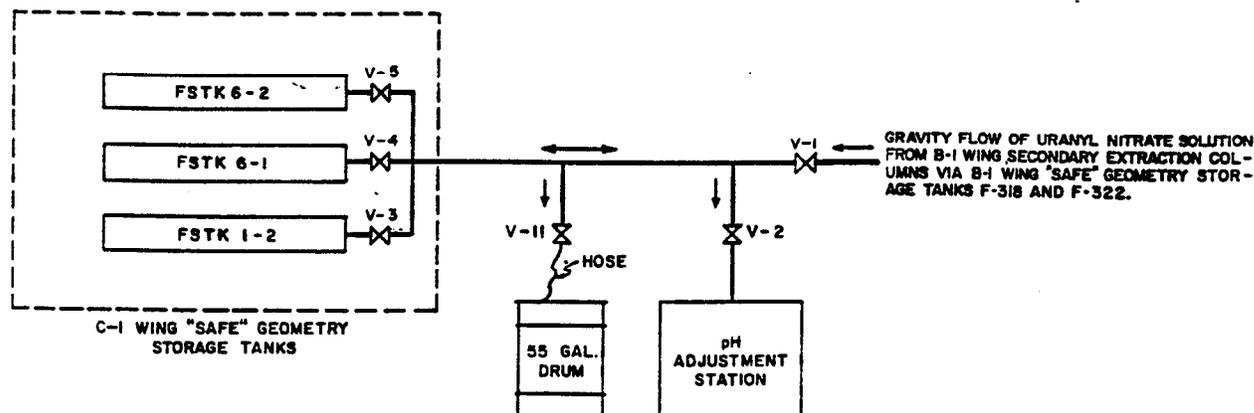
Figure 6
SIMPLIFIED SCHEMATIC OF PIPING INVOLVED IN THE ACCIDENT

At 7:00 a.m., June 16, Foreman "X" relieved Foreman "Y". The accounts of whether Foreman "Y" notified Foreman "X" of the above mentioned uranyl nitrate leakage are conflicting. In any event, no mention was made of it in the operating log.

At 8:00 a.m., Foreman "W" came on duty. One of his jobs was to complete the leak testing of the C-1 "safe" tanks including tanks 6-1, 6-2, and 1-2. He assigned Operators "A" and "J" to this work. Foreman "W" was completely unaware of the circumstances of the uranyl nitrate leakage observed on the previous shift. He was, however, quite certain that the "safe" tanks 6-1, 6-2, and 1-2 had been dismantled and cleaned during the previous week and that no operations had been started in C-1 Wing since that time. This information had been logged and had also been given him on the preceding Friday by Foreman "U".

On the basis of this previous knowledge, Foreman "W" did not deem it necessary to check the tank level indicating panel nor did he attach any significance to the open or closed condition of valve V-3 at the bottom of tank 1-2 during his piping check. Being aware of the fact that B-1 Wing was in operation, he did, however, instruct Operator "J" to check valve V-1 in the line from B-1 Wing. Furthermore, Operator "A" was stationed at the 55-gallon drum during the "safe" tank draining operation.

Subsequent investigation indicated that valve V-3 at the bottom of tank 1-2 was open and that this tank contained a substantial quantity of concentrated uranyl nitrate solution. This solution had leaked from B-1 Wing through valve V-1 between Sunday night and 1:30 p. m. Monday when Operator "J" checked valve V-1 and applied pressure to the handle to assure positive closure.

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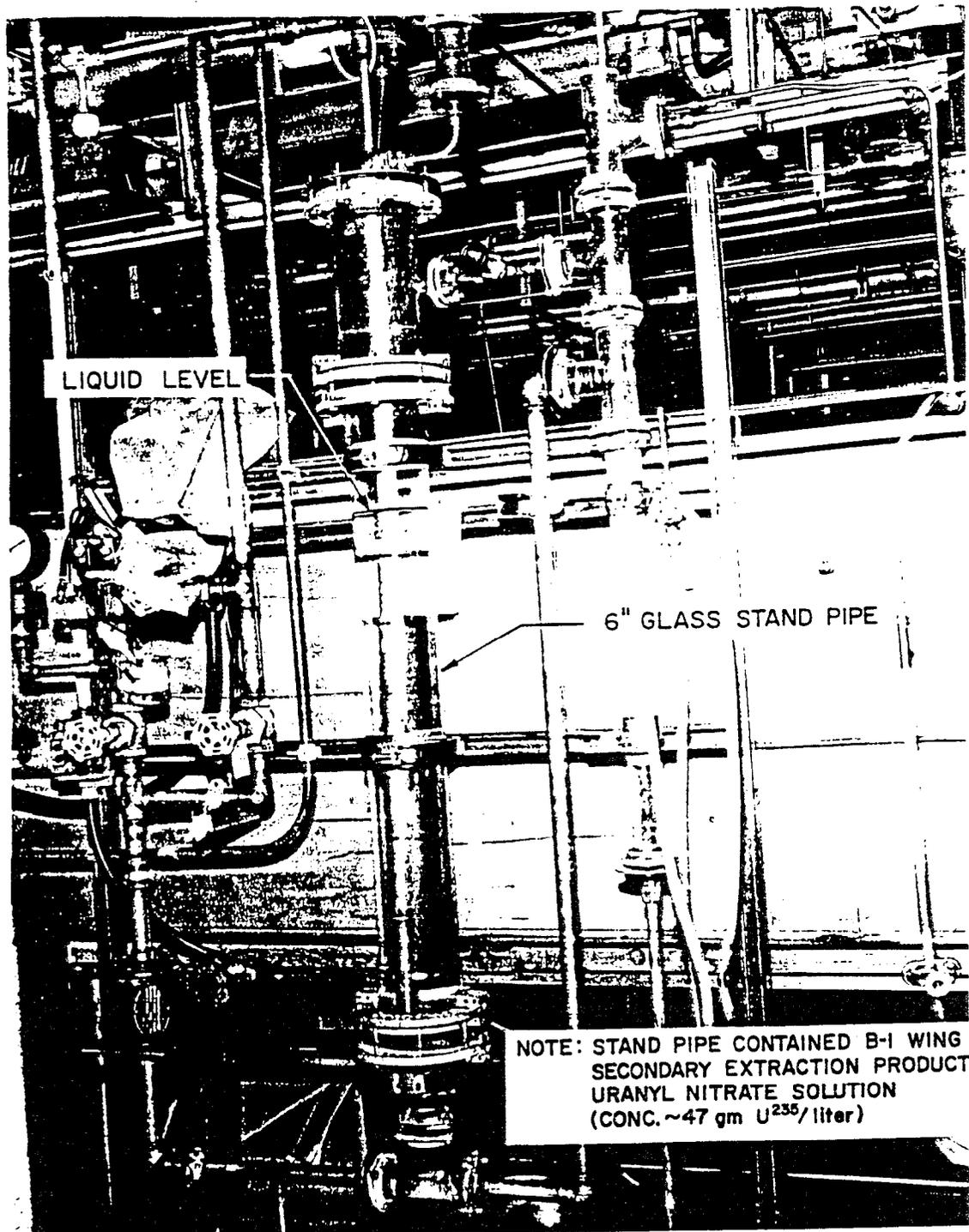
Figure 7
ph ADJUSTMENT STATION C-1 WING
(Photograph Taken Approximately 24 Hours After the Critical Incident)

4. Shortly before 2:00 p. m., the leak testing of tanks 6-2 and 6-1 having been performed by Operators "A" and "J", Operator "J" opened drain valve V-11 to empty these tanks into the 55-gallon drum and temporarily left the C-1 area. Operator "A" remained by the drum. At 2:05 p. m., an accidental nuclear excursion took place in the drum. Subsequent investigation has established the following facts:

   a. The excursion took place after the concentrated solution in the drum had reached a height of 9 inches.

   b. It appears that this solution came from tank 1-2 into which it had previously flowed from B-1 Wing. This was indicated by hydraulic tests (see Appendix D) which showed that liquid drains from tank 1-2 in preference to liquid in tanks 6-1 and 6-2; it was supported by chemical analysis (see Appendix M) which showed the liquid in tanks 6-1 and 6-2 to have contained a negligible amount of uranium while a sample of residual solution removed from tank 1-2 contained approximately 35 g U^{235}/liter.

   c. The leak test water from tanks 6-1 and 6-2 followed the concentrated solution from tank 1-2 into the drum and approximately twenty minutes after the beginning of the excursion, when the level in the drum had reached a height of 14 to 16 inches, this additional water caused the nuclear reaction to subside.

5. Operator "A", an experienced man (one year of college training, six years in uranium processing operations), was adjacent to the 55-gallon drum observing the slow flow of liquid. The previously mentioned hydraulic experiments, performed after the accident, established that approximately a quarter of an hour was required for the liquid in the drum to reach the level at which it became critical. In addition, the yellow color of concentrated uranyl nitrate is distinctive and was well known to Operator "A". It would thus appear that Operator "A" had an opportunity to shut off the flow of solution prior to the accident.

RADIATION ALARM SYSTEM

The utility of radiation detection instruments can be summarized by stating that they are important after an accident in indicating the radiation hazard then prevailing, but in general, they have no value in predicting that a nuclear excursion is imminent.

There were six radiation alarm monitors in the general area of Building 9212 which encompassed the site of the accident. These monitors actuated alarm sirens when the dose rate at the instrument exceeded 3 mr per hour. However, in tests subsequent to the accident, it was determined that a period of 3 to 5 seconds was required, after actuation of the radiation monitors, for the alarm sirens to reach audible speed. The first several seconds are the period of greatest danger in a criticality accident.

Since the emergency procedure specifies that personnel should leave by the nearest building exit and since the radiation monitors are not capable of pinpointing the site of an accident, the possibility exists that personnel could receive serious additional exposure if the source of radiation were near an exit.

EVACUATION OF EMPLOYEES "A", "B", "C", "D", and "E"

The positions of Employees "A", "B", "C", "D", and "E" at the time of the incident are portrayed in Figures 4, 5, and M.1. With regard to Operator "A" (height 5 ft., 11 in.), it is to be noted from Figures 4 and M.1, that the 5 ft., 9 in. high by 20 ft. long stainless steel laboratory bench limited his view of the positions and movements of other employees.

Evacuating to the west when the alarm sounded, it appears that by the time Operator "A" reached a point from which his view was unobstructed, the other nearby employees had initiated their evacuation to the east. It thus appears doubtful that Operator "A" had an opportunity to inform the others present of his observations at the 55-gallon drum.

The decision of Employees "C" and "E" to evacuate to the east rather than the west was unfortunate, in that this route actually led them closer to the 55-gallon drum. It does not appear from the position of Employees "B" and "D" that the path taken would have made any significant difference.

That all of these employees heeded the alarm and instantly evacuated the building must be emphasized. It can be stated unequivocally that fatalities in this incident were prevented by the rapid and orderly exit of the employees. Their action in this manner, to which at least one (Employee "A") owes his life, is evidence of an effective indoctrination in safety practices.

NUCLEAR SAFETY EDUCATION PROGRAM

The most recent nuclear safety training, prior to the accident, consisted of a program prepared by the Y-12 Radiation Control Department which was presented in a series of sessions in February and March, 1958. Nearly all supervisors in the plant, as well as all personnel in the plant who handle uranium, attended at least one session of one-and-one-half hours duration. About 550 supervisors and approximately 500 employees, including all chemical operators and foremen referred to in this report, attended.

This training included the following topics:

1. Nuclear safety; the nuclear chain reaction, its prevention and results.

2. Nuclear safety in Y-12.

3. The responsibility for nuclear safety.

4. Methods of nuclear safety.

This session included detailed information on the recognition and consequences of a nuclear accident quite similar to the actual occurrence of June 16, 1958.

Plant personnel involved in uranium processing were given a lecture on the same material approximately one year earlier.

In all of the above discussions, the primary emphasis was on the prevention of nuclear accidents.

CONCLUSIONS

CAUSES OF ACCIDENT

This accident is not attributable to the action of any single individual, but rather, it arose out of a combination of circumstances involving the character of the facilities as well as the behavior of individuals.

An abstract, yet significant, contributing circumstance was the interim status of the enriched uranium recovery facilities as discussed in the section entitled *FINDINGS*. For example, the fact that the facilities for converting concentrated uranyl nitrate into uranium tetrafluoride were spread over three areas seriously compounded the communications problem. Futhermore, C-1 Wing had for years been operated under the principles of administrative batch control of nuclear safety. The extensive use during these years of equipment not of "nuclearly safe" dimensions due to its size and shape had previously conditioned plant personnel to the unchallenged acceptance of a 55-gallon drum in the leak testing of the C-1 Wing "safe" tanks with water.

In addition, the complete exchange of significant information among personnel was not assured, nor was the potential significance of several observations, now recognized as highly pertinent to the occurrence, adequately appreciated.

It is highly likely, if not certain, that the accident would not have occurred in the absence of any one of several factors. Among these are the use of the 55-gallon drum, the inadvertent flow of unidentified solution between areas, and the subsequent drainage of this solution into the 55-gallon drum without recognition of its composition.

It seems reasonable to conclude that the accident resulted largely from an accumulation of observable physical conditions which, though unknown in full to any individual at the time, should have prompted preventative action.

The committee also concludes that, although the environment in which this event took place and the performance of some individuals might have been improved, a nuclear accident will always be within the realm of possibility whenever potentially critical quantities of fissionable material are being handled.

NATURE OF ACCIDENT

The accident took place as a result of the inadvertent introduction of concentrated uranyl nitrate solution into a 55-gallon drum. The energy release concomitant with the accident occurred during an interval of minutes in which the effective reactivity and the power level oscillated a number of times. The nuclear reaction was ultimately stopped by the additional flow of water into the drum. No solution was forcibly expelled from the drum during the power evolution, other than an aerosol. It is evident from a review of the accident that very slight differences in any one of several controlling factors could have resulted in an energy release several orders of magnitude greater than that observed. The energy release was however, about ten times greater than that resulting from previous accidents of this type.

EMERGENCY PROCEDURES

The emergency procedures previously established to provide for incidents of this nature and magnitude are considered to have been adequate. The number of people involved over large areas, as might be expected, introduced a degree of confusion, causing some delay. However, work progressed, information was obtained and coordinated, and the basic principles of the emergency plan (that is, personnel evacuation, personnel monitoring, medical assistance, and radiation area isolation), progressed in a satisfactory manner.

DOSIMETRY

The sodium activation of the blood provided the best estimate of the radiation dose received by exposed personnel. The indium foil in the badges carried by the Y-12 employees enabled health physics personnel to quickly and efficiently identify highly exposed employees and make preliminary estimates of the magnitude of the doses.

RECOMMENDATIONS

It is recognized that extensive study and evaluation are required to improve existing radiation control practices and procedures if such action is to be taken without (a) establishing unduly rigid controls which would seriously interfere with operating efficiency, or (b) embarking on large expenditures for equipment and facilities which might be of only minor assistance in preventing or coping with a similar incident in the future. Accordingly, a study group, composed of representatives from AEC installations operated by the Union Carbide Nuclear Company and the Goodyear Atomic Corporation, has been established. Its mandate is to develop detailed recommendations regarding means of avoiding the occurrence of radiation emergencies and of providing adequate preparation for handling such emergencies if they do occur. Subjects being considered include: equipment design philosophy, operating procedures, nuclear safety education, radiation detection and warning devices, dosimetry, and emergency planning.

Nevertheless, the committee feels that, in keeping with the purpose of this investigation, the following general recommendations should be made at this time in the hope that they may be applicable and of value to other processors of fissionable materials.

EQUIPMENT DESIGN PHILOSOPHY

Nuclear safety often can be enhanced without compromising economy by the extension of present control methods and, perhaps more significantly, by the utilization of other well-known nuclear concepts which thus far have not been extensively applied to production operations. Examples of these methods are included in the following recommendations:

1. Within the bounds of economic practicability, nuclear safety should be incorporated in the design of the equipment, taking full advantage of the characteristics of the material and process.

2. Within the same bounds of economic practicability, if materials of different isotopic enrichment are to be processed simultaneously or in campaigns in a single facility, the entire facility should be designed for the highest level of enrichment.

3. Transfers from a processing train which relies for nuclear safety on equipment construction to one which relies on administrative control should be avoided unless no practical alternative is available. These transfers, if made, must be conducted under extremely rigid control conditions. For example, no single analytical determination should be depended upon for the limitation of a batch size.

4. An investigation of the use of fixed neutron absorbers in process equipment to implement nuclear safety should be actively pursued. The properties to be investigated should include the necessary configuration and concentration of the absorbers and their mechanical and chemical stability. Information from such tests will allow future design decisions to be based on economic and technical considerations.

OPERATING PROCEDURES

The use of portable unsafe containers in operating areas incorporating "safe" processing equipment should be held to an absolute minimum.

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The means of communication between shifts, between operating and maintenance groups, and between production and staff groups should be more highly formalized than is customary in the chemical industry.

NUCLEAR SAFETY AND HEALTH PHYSICS EDUCATION

It is recommended that the importance of nuclear safety in fissionable materials processing plants be restated and re-emphasized periodically to all personnel working in the processing areas. Although primary dependence for nuclear safety lies in equipment or procedural restrictions, it is clear that only by creating a constant awareness of nuclear safety can unusual and unexpected circumstances be viewed in terms of their possible nuclear hazard.

Likewise, management and all plant personnel should be reinstructed periodically in the health physics aspects of potential nuclear emergencies.

DOSIMETRY AND RADIATION DETECTION

The incident has underlined the urgent need for personnel dosimeters at installations which handle fissionable materials. Records of dosimetric findings should be kept for each individual. Only by requiring that the best dosimetry available be employed routinely can one insure that accurate dose values will be obtained in case of accidents. It is recommended that a single personnel dosimeter packet be used.

1. The personnel dosimeter should be capable of measuring both the gamma and neutron dose. A film type badge dosimeter which fulfills these requirements is available. It contains the following:

    a. A film sensitive to gamma energies ranging from a few milliroentgens to thousands of roentgens.

    b. An NTA film pack and approximately 1 gram of sulfur for fast neutron detection.

    c. Indium foil for rapid identification of individuals who received appreciable neutron doses.

    d. Bare- and cadmium-covered gold foils for slow neutron detection (the gold permits scanning over several days).

    Where economically feasible, Hurst threshold detectors in addition to appropriate gamma detectors should be located at the various danger points. The threshold detectors would be used to establish the spectral distribution of neutrons in the neighborhood of an accidental excursion and the gamma detectors would aid in establishing the ratio of the gamma and neutron yields.

2. Sampling procedures should be established to determine neutron activation of the persons and possessions of exposed individuals. The activation of blood sodium, as discussed in Exhibit V, is particularly valuable in this connection. A whole body counter should be used for the scanning of large numbers of people and for the rapid assay of large volumes of low level liquids.

3. A competent, well-informed health physics group, vested with a reasonable degree of authority, is vital in properly coping with the aftermath of a nuclear accident.

EMERGENCY PLANNING

Any facility concerned with the processing of fissionable materials should have a detailed emergency plan. This plan should closely coordinate all plant emergency activities and, in applicable areas, close interplant coordination should exist. Trained local and plant emergency squads should be maintained, and the emergency plan should be given thorough testing and periodic review to maintain its adequacy.

As a minimum, this plan should ensure that adequate provisions are made for the following points:

1. Immediate alerting and evacuation of personnel.

2. Adequate communications including an information control center.

3. Prompt location of the affected area.

4. Location, monitoring, decontamination, and medical treatment of personnel involved in the incident.

5. Control of re-entry to the affected areas.

6. Adequate identification for prompt access of emergency personnel.

7. Mobilization of adequate transportation facilities.

APPROACH OF NEAR CRITICAL SOLUTIONS BY PERSONNEL

The following recommendation is made governing the approach of a near critical solution of U^{235} by personnel. The recommendation is based on the analysis of the effect, on the solution reactivity, of the neutron reflection by a simulated human body which is presented in Appendix L. A vessel containing solution in which a nuclear accident has recently occurred should be approached no nearer than five feet, and the number of persons at this distance should be limited to one. This person should be equipped with both neutron and beta-gamma survey meters, the former of a type which is operative in a high-level gamma-ray field. If only a gamma monitor is available, a person should remain at the 5-foot distance a maximum of 10 seconds to avoid possibly incurring significant radiation exposure. This exposure is in addition, of course, to that from the delayed gamma rays which may impose additional limitations on the minimum approach distance. It is emphasized that this recommendation is applicable only to incidents stemming from nuclear excursions in aqueous solutions of fissionable materials. It does, however, include a safety factor of more than two on the result of the analysis.

INVESTIGATING COMMITTEE

UNION CARBIDE NUCLEAR COMPANY
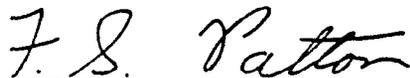
Patton, F. S., Chairman
Process Analysis, Y-12

Jasny, G. R.
Chemical Operations, Y-12
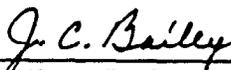
Bailey, J. C.
Health Physics, ORGDP
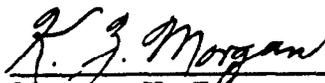
Morgan, K. Z.
Health Physics, ORNL
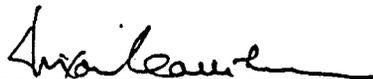
Callihan, A. D.
Critical Experiments, ORNL

Sullivan, C. R., Jr., M. D.
Medical, Y-12

Googin, J. M.
Development, Y-12
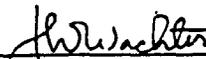
Wachter, J. W.
Radiation Control, Y-12

UNITED STATES ATOMIC ENERGY COMMISSION

McAlduff, H. J.
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EXHIBIT I - BACKGROUND TO INCIDENT

A discussion of Y-12 philosophy and policy with respect to nuclear safety prior to the accident is presented below in order that the reader might be better able to visualize the circumstances and causes of the incident in proper perspective.

The term nuclear safety connotes freedom from accidental and unscheduled nuclear chain reactions. A nuclear chain reaction may occur when a certain "critical" quantity of fissionable material has accumulated. In the Y-12 facility for the recovery of enriched uranium from fuel fabrication scrap and other salvage, the feared consequence is not a high order nuclear explosion, but rather the lethal radiation accompanying an uncontrolled nuclear chain reaction. Such a non-explosive nuclear chain reaction corresponds to the reaction carried out in a controlled manner in an atomic reactor installation.

It is in order to distinguish between chemical processing facilities for the preparation of "cold" enriched uranium, which has relatively little radioactivity, and those for the recovery of uranium from "hot" irradiated reactor fuel by separation from the highly radioactive fission products. The latter process must be constructed behind adequate shielding to protect personnel from exposure to the ever present radiation. Although enriched uranium is processed in both types of facilities and both are, a priori, vulnerable to nuclear accidents, the consequences to personnel in the vicinity of a nuclear excursion are likely to be far more serious in a "cold" processing facility (such as Y-12) than those from a corresponding accident in a well shielded "hot" processing plant.

Initiation of a nuclear chain reaction is dependent upon the favorable disposition of such variables as: mass of uranium, shape and size of system, reflection, interaction, chemical composition, concentration, nuclear poisons, isotopic enrichment. In practice, in a uranium processing facility, the physical form of the uranium and the isotopic enrichment principally control the extent of the processing restrictions which must be imposed. Figure 8 presents minimum critical gross masses of uranium (i.e., $U^{235} + U^{238}$) at various U^{235} enrichments.[8]

From Figure 8 it is seen that only about 0.8 kg of gross uranium of 90% U^{235} content is required to achieve criticality in aqueous solution under optimum conditions, while nearly fifty times this amount of gross uranium ($U^{235} + U^{238}$) would be required for criticality with a 5% U^{235} enriched uranium solution. It is also of interest to note from Figure 8 that over fifty times as much uranium of approximately 90% U^{235} content is required to achieve criticality when the uranium is in the form of unreflected massive metal than when the uranium is in a homogeneous aqueous solution (water reflected and with optimum moderation).

It is also shown in Figure 8 that the critical quantity of U^{235} increases very rapidly as the U^{235} enrichment decreases below 5%, a region of interest in reactor development. Indeed, unmoderated massive metallic uranium containing no more than 5% U^{235} by weight cannot be made critical.

In general, two approaches to nuclear safety have been employed at Y-12. They are as follows:

1. Administrative Control

   Administrative control of nuclear safety implies a principal reliance upon operations

personnel and their line supervision to prevent an accidental nuclear excursion. Radiation control procedures specifically define allowable operating parameters such as the amount of uranium which may be batched in a given container, or the number and weights of metallic items which may be stored in a given array. In arriving at values of operating parameters, account is taken of the possibilities of human errors and production accidents by the application of safety factors which, though essentially arbitrary, are sufficiently large to cover certain 'events that are recognized as being possible under the processing conditions. In general, these safety factors are adequate to maintain safety despite the independent occurrence of two contingencies such as the insertion of twice the allowable limit of uranium in a container and an inadvertent placement of two containers side by side. It is obvious that heavy emphasis must fall upon accuracy in sampling and analytical procedures under the administrative control approach.

2. Geometric Control

The intent of the geometric control approach to nuclear safety is to so design processing equipment, including storage vessels for solutions, that no critical accumulation can occur regardless of other factors such as the quantity of material in process, its chemical composition, or the proximity of neutron reflecting bodies. Such systems are most applicable and indeed are most economical for handling free flowing, highly enriched (in U^{235}), highly concentrated uranium where the form of the desired product does not often change. In the aqueous chemical processing of highly enriched uranium, common applications of the geometric approach are pipes of 6 inches, 5 inches, and lesser diameters, and pans of 1.5 inches and lesser depths.
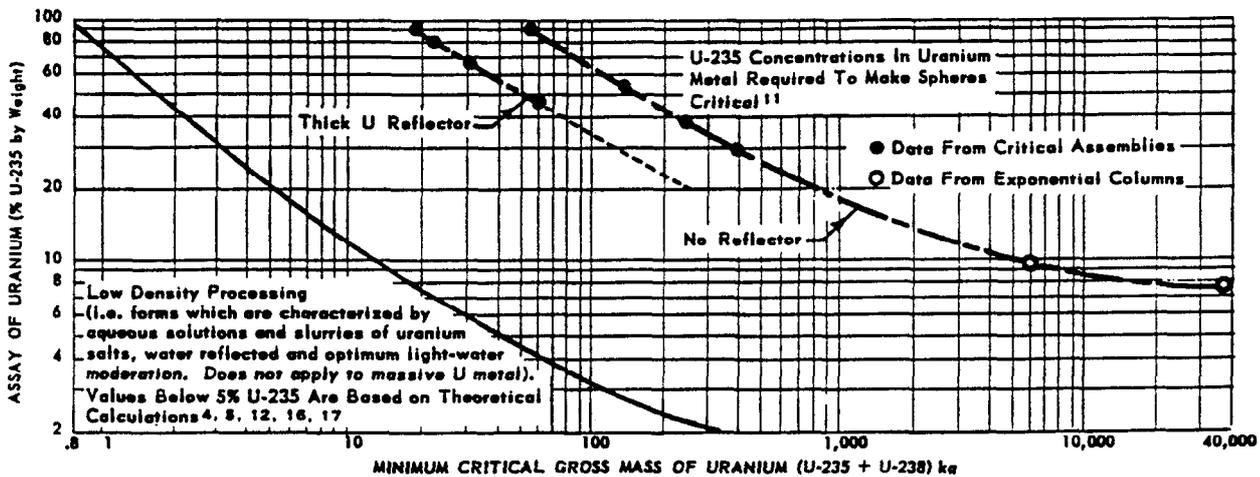
Figure 8
ASSAY vs MINIMUM CRITICAL GROSS MASS OF URANIUM

In metalworking operations the preponderance of measures must be based on administrative control, while with salvage operations, wherein dissolution and solvent extraction are employed, there is a considerable option as to the control approach.

In the early years of the present decade, the administrative control approach was dominant in the highly enriched uranium (> 75% U^{235}) recovery facilities at the Y-12 Plant. Uranium salvage recovery operations were located in Wings C-1 and C of Building 9212.

The processing steps involved and the approximate uranium concentrations at the several steps are presented in the lower portion of Figure 9.

Unlike the situation in industrial safety, the effectiveness of a nuclear safety program cannot be estimated by the frequency and severity of minor incidents. However, "negative statistics" provide some indication of the adequacy of the program. Such statistics are acquired through study of limit violations in which through human error, process failure, or other means, one of the "at least two" contingencies prevailing from the safety factor employed in the administrative control approach is violated. Such situations are carefully studied by nuclear engineers as an indication of the soundness of the control program.

In the Y-12 Plant, in the C-1 Wing and C Wing areas and in other areas, a total of twenty-five such limit violations were recorded in a five-year span commencing in 1952. The vast majority of these violations did not involve a close approach to criticality and were caused by a variety of actions ranging from analytical errors to the inadvertent transfer of enriched uranium from one area to another. However, one incident in 1956, involving the pouring of enriched uranium solution into an "unsafe" container, was of a serious nature, as it was computed that a critical excursion could have occurred if the depth of liquid in the container had been slightly greater.

The above experiences, coupled with the necessity for an expansion of facilities arising from increased throughputs and attendant larger inventories, brought into focus the desirability of a shift in dependence from administrative control to geometric control in the uranium chemical recovery operations.

These considerations resulted in efforts aimed at the evolvement of a continuous equipment train wherein uranium salvage, from point of entry to issuance as a pure uranium compound, would be processed in equipment that was "geometrically safe". The nearly completed B-1 Wing facility, Building 9212, is the culmination of these developments. (See Figure 9).

It should not be supposed that such transition to geometric control is easy of accomplishment or can be purchased without incurrence of disadvantages. As geometrically "safe" equipment is usually equipment of small cross-section, flow rates must be high and reaction times must be short, which circumstances are sources of mechanical difficulties. In addition, solutions are concentrated early in the processing train to eliminate extraneous bulk as quickly as is feasible, with the net result that considerable liquid volumes, containing uranium of sufficient concentration for criticality are constantly in process.

In brief, it may be stated that the principal change (from a nuclear safety standpoint) resulting from the transition from the C-1 Wing facility to the new B-1 facility is as follows:

In the C-1 Wing facility, the administrative control approach to nuclear safety prevailed. Equipment that was not geometrically safe was extensively used. However, as regards the individual equipment items, contained process solutions were routinely dilute, and/or uranium inventories were small. Many chances for human error in weighing, chemical analysis, transcription, etc., existed, but at a minimum; several such errors were required simultaneously for a critical incident to occur. Thus, it is seen that rigid adherence to batching procedures and duplication of measurements and analyses were the principal control responsibilities.

In the B-1 Wing, the chances for human error are vastly reduced. The principal contingency meriting concern is the inadvertent transfer of concentrated uranium solution from "safe" geometry equipment to an "unsafe" container. At a number of points in the

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SOLID
SALVAGE MATERIAL
<1% U

CRUSHER-GRINDER

CALCINER

DISSOLVER
0.2 kg TOTAL

0.8 gm/liter

5 gm/liter

CENTRIFUGE

LIQUID SALVAGE
AS URANYL NITRATE SOLUTION
15 gm/liter

SAFE STORAGE
1.5 kg

SAFE STORAGE
9 kg

30 gm/liter

"THIN-FILM"
EVAPORATOR
1 kg

PRIMARY EXTRACTION SYSTEM
3 kg/COLUMN SET
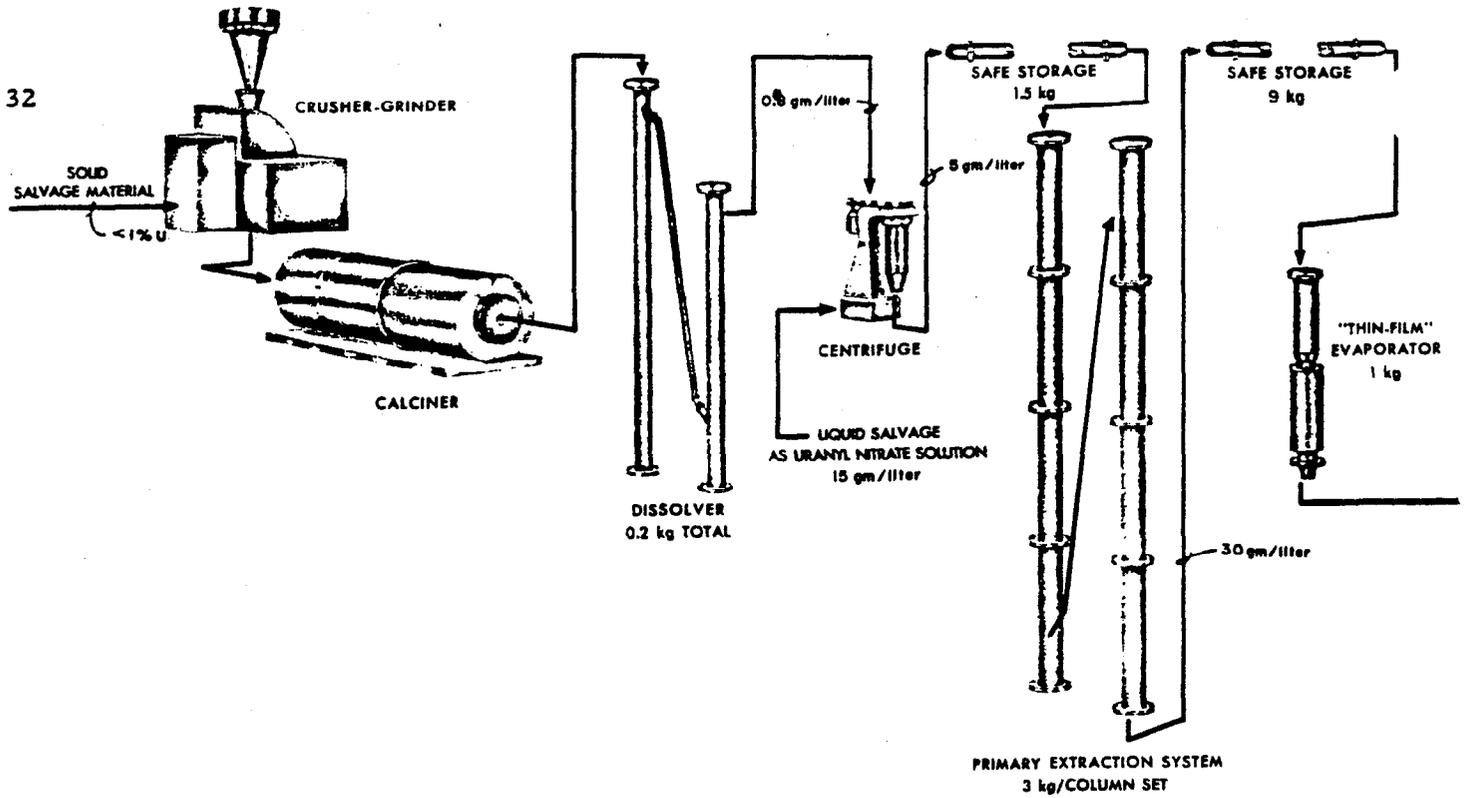
B-1 WING CONTINUOUS URANIUM RECOVERY PROCESS

(DUAL PROCESSING TRAINS)

C-1 WING SEMI-CONTINUOUS URANIUM RECOVERY PROCESS

(MULTIPLE PROCESSING TRAINS)

SOLID & LIQUID
SALVAGE MATERIAL
<1% U

DISSOLVER
0.2 kg

0.3 gm/liter

FILTER PRESS

UNSAFE
GEOMETRY
STORAGE TANK
0.2 kg

0.8 gm/liter

SAFE STORAGE
0.1 kg

SAFE EVAPORATOR
0.2 kg
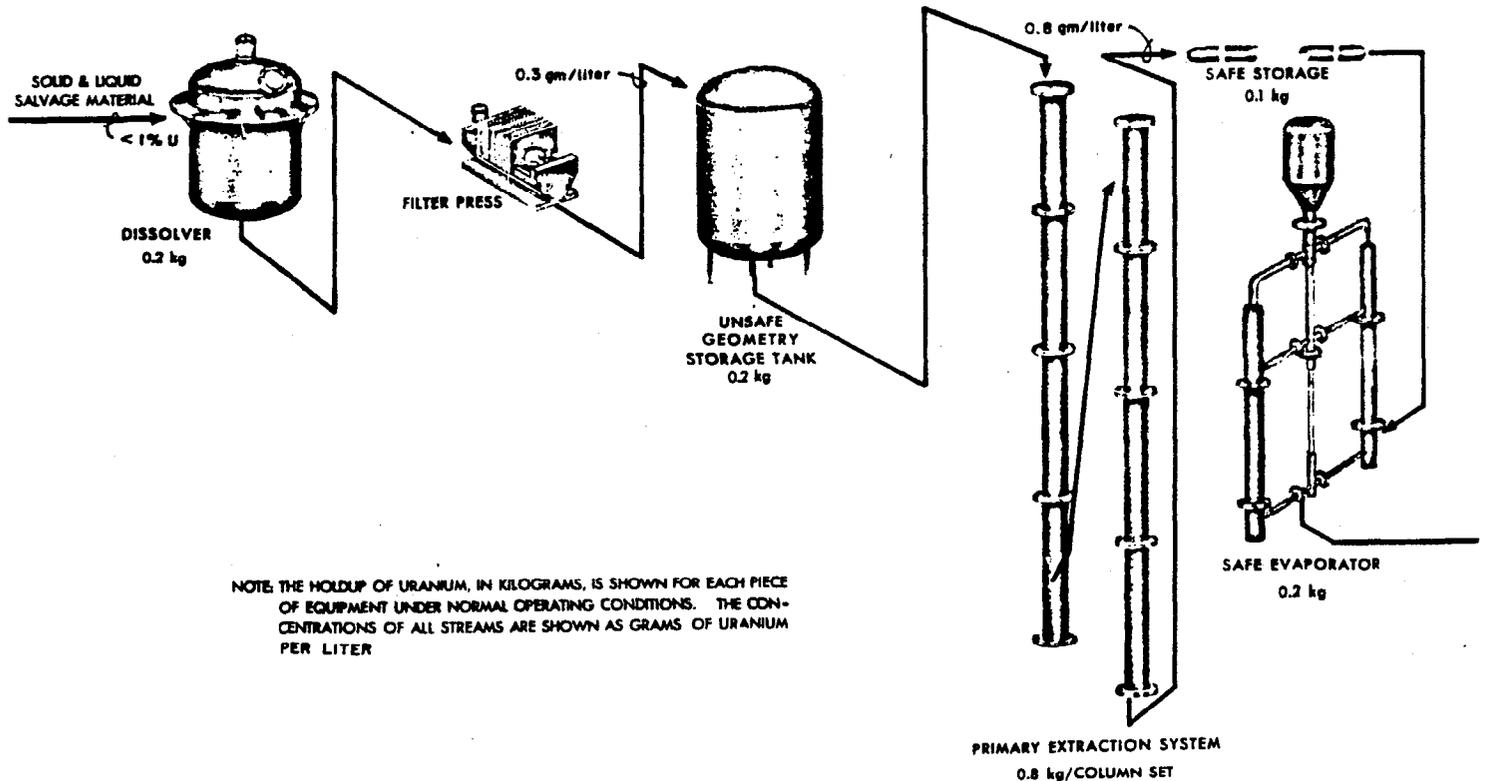
NOTE: THE HOLDUP OF URANIUM, IN KILOGRAMS, IS SHOWN FOR EACH PIECE
OF EQUIPMENT UNDER NORMAL OPERATING CONDITIONS. THE CON-
CENTRATIONS OF ALL STREAMS ARE SHOWN AS GRAMS OF URANIUM
PER LITER

PRIMARY EXTRACTION SYSTEM
0.8 kg/COLUMN SET
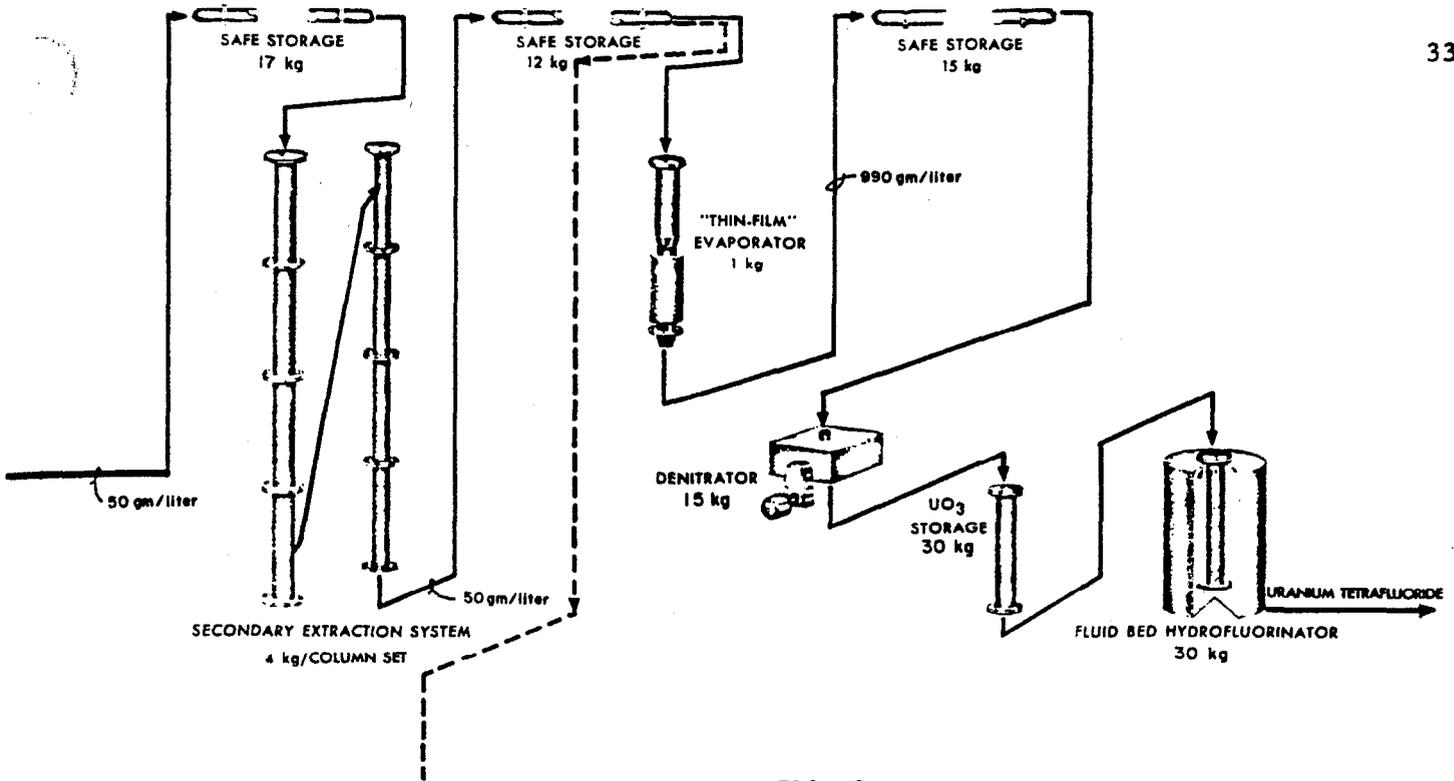
FIG. 9

CONCEPT OF SALVAGE PROCESSING
IN B-1 AND C-1 WINGS
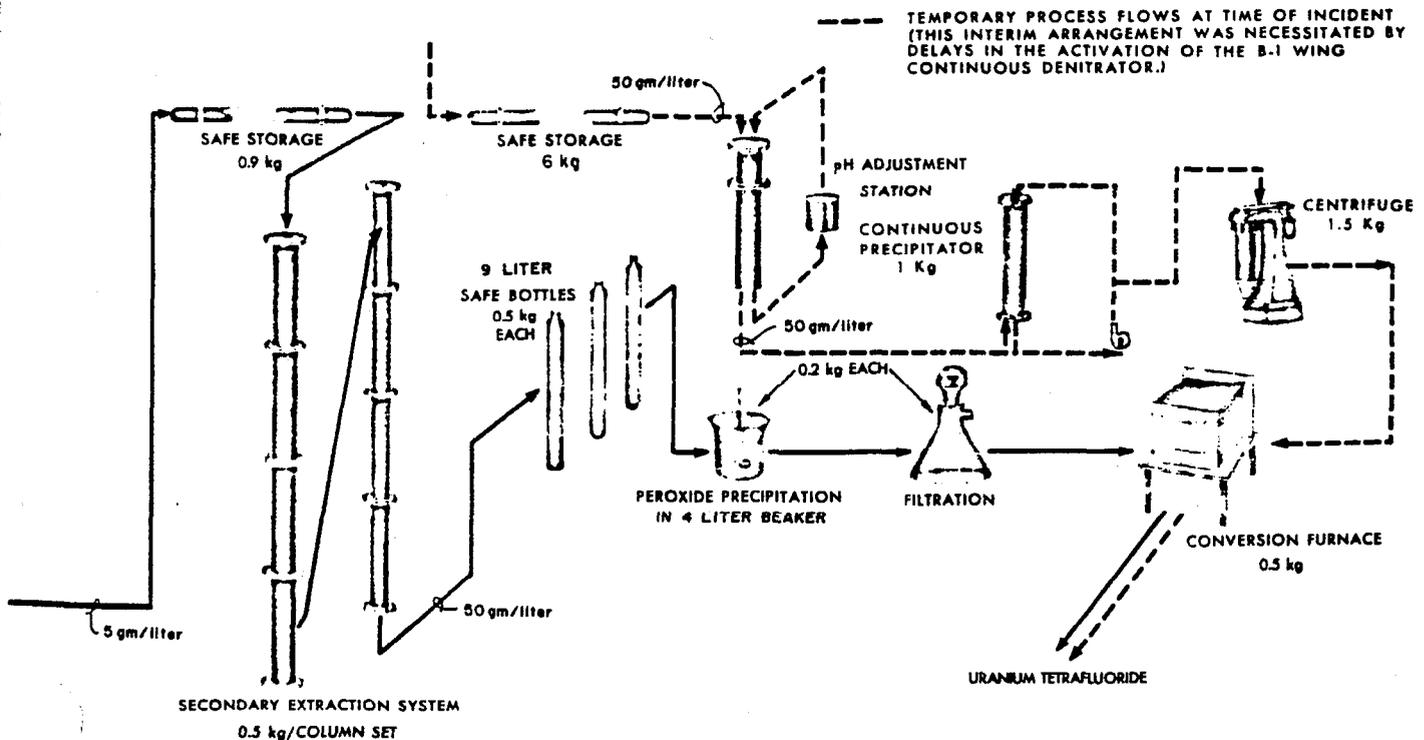
B-1 equipment train, concentrations are such that there is a high probability that a nuclear excursion would be incurred if solution were to fill by leakage, or other means of inadvertent transfer, such mundane objects as a waste basket, a mop bucket, a desk drawer, or a workman's tool box.   Thus, it is seen that a principal control responsibility is the exclusion of "unsafe" containers from the process area.

At the time of the nuclear incident (June 16, 1958), and as is illustrated in Figure 9, the denitration and hydrofluorination sections of the B-1 facility were not yet in operation. In consequence, a temporary arrangement was made which encompassed a transfer pipeline from the B-1 Wing secondary extraction product "safe" tanks to three C-1 Wing "safe" tanks and the subsequent operations portrayed in Figure 9. This temporary arrangement in C-1 Wing had the same characteristics as B-1 Wing as regards concentrations of solutions and uranium inventories in individual equipment items. In consequence, the same nuclear safety approach as in B-1 Wing was required; i.e., exclusion of "unsafe" containers from the process area.

EXHIBIT II - DETAILED INVESTIGATION
OF OPERATIONAL EVIDENCE*

On June 16, 1958, the Building 9212 chemical area was in the latter stages of the required uranium inventory for the month of May. This accounting is expected to balance within a few tenths of one percent. The inventory is not the mere counting of a large number of discrete objects, but involves rather the determination of the amount of uranium contained in a large variety of complex mixtures. In order to improve the chances of obtaining an accurate measure of the uranium, the salvage section of the plant has been used as an analytical tool for the quantitative conversion of all of these complex mixtures to purified solutions and compounds for which the uranium content can be accurately determined.

After the treatment of the salvage materials generated during the inventory of the rest of the plant, the salvage facilities have to be inventoried for their residual uranium content before the over-all plant balance can be closed. It was this latter operation which was being performed at the time of the nuclear incident.

In the interest of achieving a closer inventory balance in the salvage facilities, it was deemed advisable to wash and dismantle for swabbing some of the equipment, especially the safe geometry tanks used for the storage of concentrated uranium solutions. In the past, inventory errors had resulted from the undetected accumulation of solid uranium compounds.

As they were prone to leak after reassembly, some of the older tanks were tested prior to reuse. Simply filling the tanks with water proved to be an adequate method. The procedure involved bringing a 55-gallon drum of water into the area which was closed down for inventory. The drum was equipped with a bail so that the existing hoisting equipment could be used to elevate it to the mezzanine floor from which the required water (about 42 gallons per tank, in this instance) could be siphoned into the tank to be tested. In C-1 Wing the safe tanks are suspended just beneath the mezzanine floor. After filling the "safe" tank to overflowing, the drum was lowered to the floor below so that the water could be collected after the inspection for leaks. The water was customarily reused in a number of tanks because of the possibility of recovering small quantities of uranium (value = 15,361 $/kg) which otherwise might have been lost.

Normally the B-1 and C-1 recovery areas were started up at the same time after the inventory period. After the May inventory, the B-1 leaching and extraction equipment (see Figure 9) was ready for operations before the C-1 area which received the B-1 product under the temporary arrangement described in Exhibit I. However, in this instance, since there were adequate storage facilities in B-1 Wing (tanks F-318 and F-322) for the expected product, the B-1 area was placed in operation before the C-1 area. This was done in an attempt to minimize equipment downtime.

As illustrated in Figure 10, two identical secondary extraction units, known as systems 1300 and 2300, are contained in B-1 Wing. It is to be noted that the uranyl nitrate product transfer piping from the B-1 Wing "safe" tanks (F-318 and F-322) had no shut-off valve in B-1 Wing. This condition was allowed in the interest of minimizing air locking of the

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① YELLOW URANYL NITRATE SOLUTION NOTICED AT pH STATION ABOUT 5:00 A.M.
② V-2 CLOSED AT ABOUT 5:00 A.M.
③ V-I CHECKED CLOSED ABOUT 1:30 P.M.
④ SAFE TANKS FSTK 6-I AND 6-2 WERE LEAK TESTED BY FILLING FSTK 6-2.
   WITH WATER THROUGH FUNNEL ON MEZZANINE.
⑤ CRITICAL EXCURSION OCCURRED IN 55 GALLON DRUM AT 2:05 P.M.
⑥ V-3 WAS FOUND OPEN AFTER INCIDENT.
⑦ V-4, V-5, V-7, V-8 AND V-II WERE FOUND OPEN AFTER INCIDENT.
⑧ V-9, V-IO, V-I2 AND V-I3 WERE FOUND CLOSED AFTER INCIDENT.
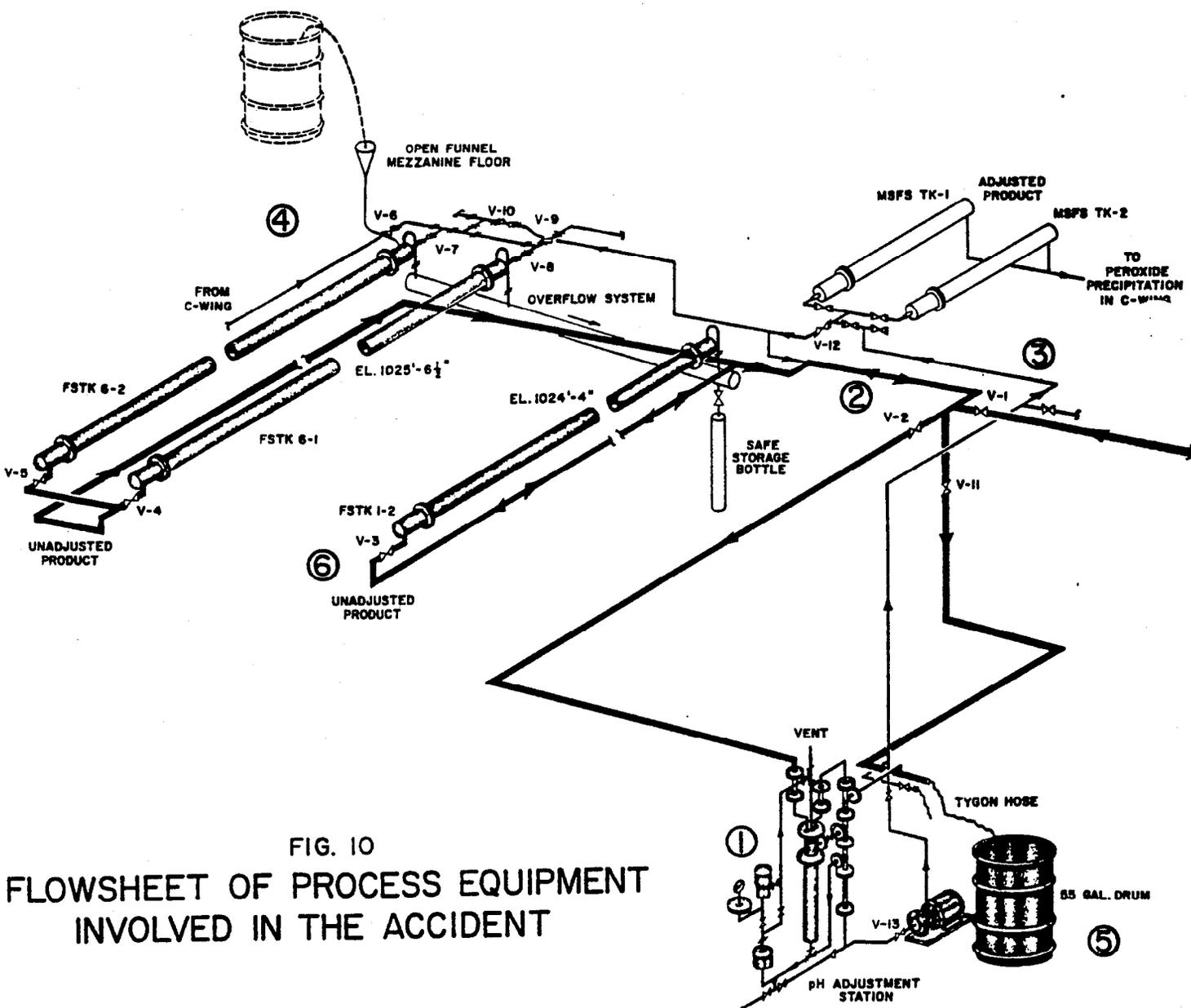
FIG. 10

FLOWSHEET OF PROCESS EQUIPMENT
INVOLVED IN THE ACCIDENT

C-I WING

ACIDIFIED WATER
BACKWASH

SECONDARY
EXTRACTION
FEED

VENT
HEADER

1" VENT

EL. 1045'-2½"

CLEAN
CARBITOL
STORAGE

AMMONIUM
NITRATE
STRIP

SAFE STORAGE
BOTTLES

FROM SYSTEM 2300

TO FUTURE EVAPORATOR
AND DENITRATOR

F-318

URANYL
NITRATE
PRODUCT
TO C-1

AQUEOUS
BACKWASH

F-322

RAFFINATE

175 ft. ¼" O.D.
TUBING
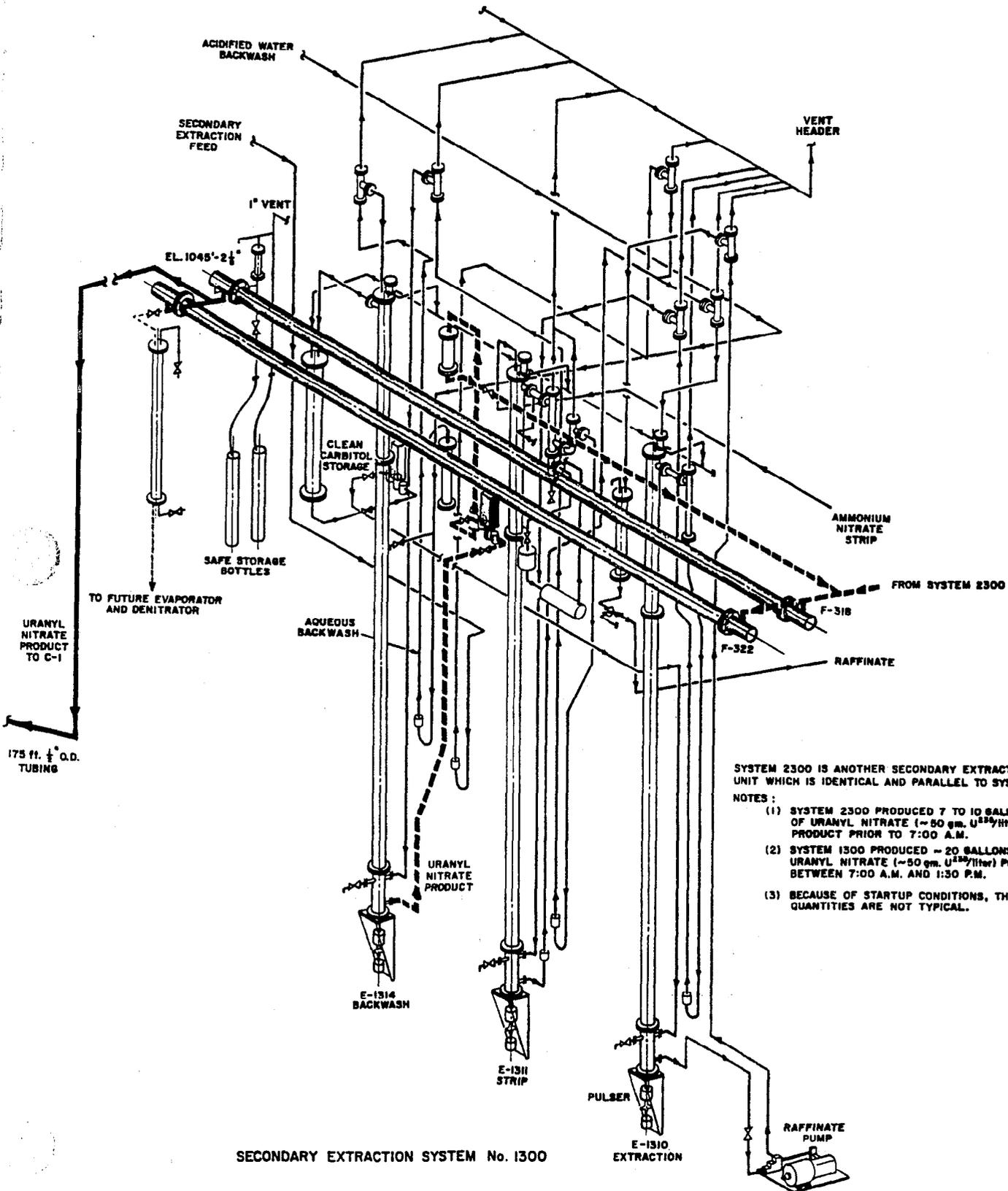
SYSTEM 2300 IS ANOTHER SECONDARY EXTRACTION
UNIT WHICH IS IDENTICAL AND PARALLEL TO SYSTEM 1300.

NOTES :

(1) SYSTEM 2300 PRODUCED 7 TO 10 GALLONS
OF URANYL NITRATE (~50 gm. U^{235}/liter)
PRODUCT PRIOR TO 7:00 A.M.

(2) SYSTEM 1300 PRODUCED ~20 GALLONS OF
URANYL NITRATE (~50 gm. U^{235}/liter) PRODUCT
BETWEEN 7:00 A.M. AND 1:30 P.M.

(3) BECAUSE OF STARTUP CONDITIONS, THE ABOVE
QUANTITIES ARE NOT TYPICAL.

URANYL
NITRATE
PRODUCT

E-1314
BACKWASH

E-1311
STRIP

PULSER

RAFFINATE
PUMP

SECONDARY EXTRACTION SYSTEM No. 1300

E-1310
EXTRACTION

B-1 WING

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transfer line. The primary control valve (V-1, in Figure 10) was located in C-1 Wing and was controlled by the C-1 Wing process foremen.

An examination of records and interviews with operating personnel discloses the following sequence of events in the pertinent sections of B-1 and C-1 Wings prior to and at the time of the accident:

B-1 WING - MIDNIGHT SHIFT

(11:00 p.m., June 15, 1958 - 7:00 a.m., June 16, 1958)

1. "Safe" tanks F-318 and F-322, which collect the secondary extraction product in B-1 Wing, had been previously cleaned and were empty at the beginning of the midnight shift.

2. During this shift, secondary extraction system 2300 produced between 8 and 10 gallons of uranvl nitrate product (at approximately 50 g U^{235}/liter) which was fed to the B-1 Wing "safe" tanks, F-318 and F-322.

3. System 1300, during this period, was on recycle and no uranyl nitrate solution was pumped from this equipment to F-318 and F-322.

B-1 WING - DAY SHIFT

(7:00 a.m. - 3:00 p.m., June 16, 1958)

Shortly after the beginning of the day shift, system 2300 was shut down and the pumping of uranyl nitrate product from system 1300 to F-318 and F-322, at a rate of approximately 3.5 gallons per hour, was initiated. At 1:30 p.m., system 1300 was shut down. Operations during this period, including maintenance downtime, etc., were such that approximately 20 gallons of uranyl nitrate product were pumped to tanks F-318 and F-322 from system 1300.

C-1 WING - MIDNIGHT SHIFT

(11:00 p.m., June 15, 1958 - 7:00 a.m., June 16, 1958)

1. No equipment had been started up after the inventory break.

2. "Safe" storage tanks 6-1, 6-2, and 1-2 had been cleaned and reassembled and were to be leak tested on the day shift (Monday) prior to reuse.

3. C-1 personnel were engaged in clean-up work and sampling of miscellaneous solids batches, prior to the resumption of routine operations.

4. Foreman "Y", at approximately 1:00 a.m. (June 16), observed wash solution in the 6" glass column of the pH adjustment station and told a chemical operator to drain the solution.

5. At approximately 5:00 a.m., Foreman "Y" again noted solution in the pH adjustment station and asked the chemical operator whether his instructions had been carried out.

Upon receiving an affirmative answer, this supervisor observed that solution was leaking through valve V-2. (See Figure 10.) Foreman "Y" then closed valve V-2 which leads to the pH adjustment station. <u>Valve V-1, which controls the flow of concentrated uranyl nitrate from B-1 Wing, was not checked.</u>

6. Foreman "Y", in a subsequent interview, stated that he was aware that the secondary columns in B-1 Wing were operating.

C-1 WING - DAY SHIFT

(7:00 a.m. - 3:00 p.m., June 16, 1958)

There is conflicting testimony as to whether or not the information concerning the leakage of concentrated material into the pH adjustment station was passed on to supervision on the day shift. In any event, no entry to that effect is found in the operating log.

The sequence of events occurring on the day shift on June 16, 1958, has been reconstructed as follows:

1. Foreman "X" came on duty at 7:00 a.m. Monday, June 16, 1958, and proceeded to complete the sampling of inventory material.

   a. Foreman "X" did not start up any equipment in C-1 Wing.

   b. Foreman "X" later stated that he <u>was not aware</u> that the secondary extraction columns in B-1 Wing were operating at that time.

2. At 8:00 a.m., Foreman "W" came on duty in C-1 Wing. He was assigned to a straight day shift and was in charge of certain specific C-1 operations which are carried out on the day shift only. One of these operations, on this day, was the completion of the leak testing of the C-1 "safe" tanks. This leak testing had been started on the day shift by Foreman "U" who left for one week's vacation at the completion of his shift on the previous Friday.

3. Foreman "W" then assigned Chemical Operators "A" and "J" to the task of completing the leak testing of the remaining safe tanks. Operator "J" had leak tested a number of tanks on Friday, June 13, 1958, with another operator who had also gone on vacation at the end of the Friday shift. On Monday morning, Operator "A" was substituted for the employee on vacation.

4. By late morning on June 16, 1958, Operators "A" and "J" were ready to begin the leak testing of "safe" tanks 6-2 and 6-1.

5. Valves at both ends of tanks 6-1 and 6-2 were closed (Valves V-4, V-5, V-7, and V-8).

6. A 55-gallon drum, containing water which had been used in leak testing other tanks, was hoisted to the mezzanine level, and its contents were siphoned into "safe" tank 6-2 by Operator "J" until water was observed by Operator "A" to overflow at the high end of tank 6-2 and drain into a "safe" bottle connected to the overflow header.

7. A leak at the end cap flange at the high end of tank 6-2 was observed. In the interest of lowering the level in tank 6-2 below the site of the leak, valves V-4, V-5, V-7, and V-8 at the ends of 6-1 and 6-2 were opened to allow half the contents of 6-2 to flow into

tank 6-1. (The supposition being that all other valves in lines to and from these tanks were closed.)

8. While the water in 6-1 and 6-2 was equalizing, Foreman "W" and Operator "J" checked out the piping connecting tanks 6-1 and 6-2 to the drain point adjacent to the pH adjustment station. The actual extent to which associated valves in the piping system were checked could not be determined from interviews with the personnel involved.

9. The empty 55-gallon drum (mentioned in 6, above) was lowered to the main floor and positioned under the drain valve located near the pH adjustment station.

10. Prior to draining the test water from tanks 6-1 and 6-2 into the 55-gallon drum, Operator "J" was instructed by Foreman "W" to check the valve in the line from the B-1 product tanks (V-1). This was done and Operator "J" reported finding valve V-1 closed. However, to be certain that the valve was fully closed, Operator "J" (a large and powerful man), applied vigorous pressure to the valve handle.

11. At approximately 1:45 p.m., Operators "A" and "J" began draining the leak test water into the 55-gallon drum, whereupon Operator "J" left the C-1 area. (Foreman "W" was occupied in the office.) Just prior to 2:05 p.m., the situation in the immediate vicinity of the 55-gallon drum was as follows:

   a. Operator "A" was checking the draining of the water into the drum, standing approximately three feet from it. His position was as is shown in Figures 4 and 5.

   b. Maintenance Mechanics "E" and "C" were engaged in installing ductwork and were in the positions shown in Figures 4 and 5.

   c. Electricians "B" and "D" were engaged in removing conduit and were in the positions shown in Figures 4 and 5.

   d. Welder "F" was working on the C-1 mezzanine approximately above Maintenance Mechanic "E".

   e. Operator "H" was in the process of starting up an evaporator approximately 50 feet due east of the 55-gallon drum.

   f. Maintenance Mechanic "G" was working on a filter house on the mezzanine about 6 feet northwest of Welder "F".

At approximately 2:05 p.m., the following events took place:

Operator "A" looked into the 55-gallon drum and noticed yellow-brown fumes (associated with carbitol and nitric acid) rising from the liquid. He stepped back and within a few seconds noted an odd bluish flash, the origin of which he was unable to determine. Almost immediately thereafter the radiation evacuation siren was heard and he started to run west. (See Figure 4.) The liquid continued to flow into the drum. While running west, Operator "A" looked back and noticed a yellowish fog behind him. Upon reaching the west end of C-1 Wing (approximately 100 feet away from the drum), he slowed to a walk. With regard to the drum, he had observed that it was about one-third full of yellow solution when he left it.

Maintenance Mechanic "E" noticed a strange odor just before the radiation monitor alarm siren was audible. Upon hearing the siren, he immediately stepped off his ladder and evacuated east at a walking pace. (See Figure 4.)

Maintenance Mechanic "C" noticed that the air had a "smoky look" and left with Maintenance Mechanic "E" when he heard the siren.

Electrician "B" saw a blue flash, like a welding flash, reflected in the white ceiling overhead, and smelled a peculiar odor. Upon hearing the radiation monitor alarm siren, he left C-1 Wing going east at a fast walk. Electrician "D" observed nothing unusual. He left with Electrician "B" and evacuated east. Welder "F", wearing his welder's hood, noticed a blue flash inside his hood just before the siren was heard. He jumped off his ladder and evacuated east. Neither Operator "H" nor Mechanic "G" observed anything unusual.

Mechanic "G", upon hearing the siren, evacuated east, somewhat behind Welder "F".

Operator "H" mistook the siren noise for that of a super centrifuge operating nearby. Upon observing others leaving eastward, he shut off the evaporator and evacuated east.

RECONSTRUCTION

The above sequence of events, when coupled with evidence obtained after the accident covering valve positions, solution inventory and analysis, and hydraulic data, are considered sufficient to allow a reasonable reconstruction of the incident. Pertinent items are presented below:

1. Valves

    a. V-1 was found closed after the accident and was found not to leak in the closed position at pressures substantially in excess of those encountered in normal operation.

    b. V-2 was found closed and also did not leak under pressure.

    c. V-3 at the low end of tank 1-2 was found open.

    d. V-4, V-5, V-7, and V-8, at the ends of tanks 6-1 and 6-2, were found open.

    e. V-6 was found to be 5/8 of a turn open, but this circumstance proved to have no connection with the accident.

    f. V-9 and V-10 were found closed.

    g. V-11, the drain valve on the "safe" tank system, was found open.

    h. V-12 was found closed.

2. Uranyl Nitrate Volumetric Balance

    a. Approximately 14 gallons of uranyl nitrate product were found in B-1 "safe" tanks F-318 and F-322. This solution analyzed approximately 40 grams U^{235}/liter.

    b. About 4 gallons of solution, containing approximately 47 g U^{235}/liter, with a chemical impurities composition identical to the solution contained in F-318 and F-322, were found in the 6-inch glass standpipe of the pH adjustment station.

    c. A small sample of aqueous solution obtained from tank 1-2 analyzed about 35 g U^{235}/liter and contained carbitol (extraction solvent). It also contained a quantity of the aluminum impurity found in the other B-1 samples.

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d. Analysis of the slight amounts of residual liquid found in the low end of tanks 6-1 and 6-2 indicated a uranium content of less than 1 gram U^{235}/liter.

e. The analysis of the contents of the 55-gallon drum, after the event, disclosed the total quantity of U^{235} to be about 2.5 kg. Due to the nature of operation of the B-1 extraction system during this period, this quantity of uranium could have been contained in approximately 10 - 12 gallons of B-1 uranyl nitrate product. (It is to be recalled that estimates indicated that 28 - 30 gallons of uranyl nitrate product were produced in the B-1 Wing secondary extraction systems between 11:00 p.m. Sunday night and 1:30 p.m. Monday.)

3. Hydraulic Data

a. The elevation of tanks F-318 and F-322 in the B-1 Wing is approximately 20 feet higher than tanks 1-2, 6-1, and 6-2 in the C-1 Wing.

b. Hydraulic tests (Appendix D) indicate that solution contained in tank 1-2 will precede solution contained in 6-1 and 6-2 when drained. These data also indicate that some mixing occurs within the system between solutions of differing concentrations.

Based on the foregoing information, it appears that the uranyl nitrate solution, produced by system 2300 in B-1 Wing on the midnight shift, started flowing at a low rate into the C-1 area "safe" tank storage system between 1:00 and 5:00 a.m. on June 16, 1958. This flow is evidenced by Foreman "Y"'s observation of solution in the pH adjustment station at 5:00 a.m. after the pH adjustment station glass standpipe had been previously drained on his orders. His closing of valve V-2 allowed all of the flowing uranyl nitrate solution to back up into the C-1 Wing "safe" storage tanks. No evidence was obtained during the investigation which established that valve V-1 had been manipulated for any reason prior to its being checked by Operator "J" at approximately 1:30 p.m. on June 16, 1958. Accordingly, the assumption must be made that valve V-1 was open to a sufficient extent to allow the flow of the approximately 4 gallons of uranyl nitrate solution found in the pH adjustment station glass standpipe and an additional flow of approximately 10 to 12 gallons of uranyl nitrate product from B-1 Wing tanks F-318 and F-322 which entered the transfer piping system and partially filled tank 1-2. The fact that valve V-1 was partially open rather than fully open during the period from 1:00 a.m. to 1:30 p.m. appears to be substantiated by the presence in B-1 Wing tanks F-318 and F-322 of 14 gallons of uranyl nitrate product which is about half of the total produced by systems 1300 and 2300 on the midnight and day shifts.

The introduction of the large quantity of leak test water into tank 6-2 and the opening of valves V-4, 5, 7, and 8, to allow this water to enter tank 6-1, provided the necessary mechanism for some dilution and mixing to occur in the system. Since the hydraulic data indicate a preferential flow from tank 1-2, the opening of valve V-11 to drain the system allowed uranyl nitrate solution to flow into a configuration (the 55-gallon drum) at a concentration optimized at some finite point for a nuclear excursion to occur.

EXHIBIT III - MEASURES TAKEN FOLLOWING THE INCIDENT *

Shortly after the alarm, assistance was requested from Oak Ridge National Laboratory and the Oak Ridge Gaseous Diffusion Plant. Accordingly, it should be recognized that the description of health physics activities and other emergency measures represents the combined and joint efforts of personnel from the three Oak Ridge installations.

PRELIMINARY SURVEYS

At approximately 2:05 p. m., June 16, 1958, sirens of the radiation monitoring system sounded the alarm in Building 9212. (See Figure 2.) Persons in the enriched uranium salvage processing facility evacuated to primary assembly areas according to plan. Immediately following the incident, process supervisors equipped with radiation survey meters, assembled at the control center in the 9212 office building hall. (See Figure 3.) The radiation intensity at this location was found to be in excess of 100 mr/hr. The Plant Emergency Director, who was present in 9212 when the alarm sounded, acted to put the plant emergency procedure into effect, and arranged to activate the emergency control center in Building 9764. This group of supervisors then evacuated the office building and made a quick survey of the west and north ends of Building 9212, observing readings of from 50 to 100 mr/hr. During this immediate post incident period, radiation was detected by laboratory supervisors at the north end of the analytical laboratory (see Figure 2), at first fluctuating in intensity up to ~1,000 mr/hr and shortly thereafter up to 500 mr/hr. Those persons who had evacuated to that point moved on to the south assembly area. (See Figure 2.)

As had been anticipated in the event of a true nuclear incident, considerable radiation (50 to 175 mr/hr) was detected at the primary assembly areas. Personnel assembled in those areas were instructed immediately to move to Change Houses 9723-19 and 9723-24, the secondary control centers. (See Figure 2.)

The radiation detected up to this point made it clear that the incident had occurred within Building 9212, with the precise location yet to be determined. Consequently, road blocks were established to prevent inadvertent entry into Building 9212. Concurrently, a survey was made along the outside of the perimeter fence and readings of up to 50 mr/hr were observed until about 2:25 p. m., after which time it was noticed that they had dropped to 5 to 10 mr/hr.

Health physicists and supervisory personnel deployed to the secondary control centers for monitoring and interrogation purposes, and steps were taken to assemble other available personnel for plant-site surveys and monitoring of employees at portals.

During the period from 2:20 p. m. to 2:40 p. m., radiation surveys were made of the plant area to obtain an over-all evaluation of conditions. These surveys indicated that there was no direct radiation or significant contamination in the areas south of First Street or east of the Dispensary Building, 9706-2. The area outside the initial delimitation boundary, shown on Figure 2, was cleared for re-occupancy by approximately 3:00 p. m.

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* Principally prepared by G. R. Patterson, Health Physics, Union Carbide Nuclear Company, and H. J. McAlduff, Production Division, ORO, USAEC.

IDENTIFICATION OF PERSONS EXPOSED TO RADIATION

Since 1955, strips of indium foil (approximately 1 gram each) have been included in the security badges of all employees at Y-12. The purpose of these foils is to provide a quick, positive means for segregating employees who receive a significant radiation dose in the course of a nuclear reaction. This determination is accomplished by the measurement of beta and gamma radiations from the radioactive In^{116} isotope which is produced by neutron irradiation of the stable In^{115} isotope in the foil.

By 2:45 p. m., the checking of personnel in the two secondary control centers, for indications of neutron activation of the indium foil in their badges and for evidence of personal contamination, was under way. Interrogation of persons assembled in these centers was begun in an attempt to establish the exact location of the incident. At approximately this time, very high readings were detected from the indium foil in the badge of Chemical Operator "A" who worked in C-1 Wing. A process supervisor questioned Employee "A" at this time and concluded that the excursion must have occurred in C-1 Wing of Building 9212. The radiation from the indium foil in this person's badge was positive evidence that he had been very close to a neutron source.

All persons at the secondary control centers were checked for contamination and their badges were examined for indium foil activation. Those persons whose badges gave evidence of possible high neutron doses were directed to the Y-12 Dispensary for further tests and medical attention, and by 3:00 p. m., the first of these individuals was received. All significant badge foil readings were recorded for further evaluation. By 4:00 p. m., twelve persons, out of approximately 1,200 surveyed, had been sent to the dispensary. During the period from 2:05 to 4:30 p. m., all personnel leaving the plant were checked by health physics teams for clothing contamination, and every effort was made to make a second check for possible cases of activation of the foil in the badges. This procedure was time consuming, and the major day shift change at 4:30 p. m. produced a situation whereby hundreds of people would be delayed several hours if the procedure were continued. Since it was considered that all individuals with a significant exposure had been detected by this time and that no significant fall-out contamination had occurred, all badges were collected as personnel passed through the plant gates, but personnel monitoring was terminated.

It is not intended that the forementioned description of activities should convey the impression that all actions were carried out with military precision. An independent observer could state, with some justification, that the several hours following the incident were a period of some confusion. The following circumstances contributed to the situation:

1. The radiation burst energized evacuation sirens in a number of buildings in which no actual radiation hazard existed.

2. The 3:00 p. m. shift change involved large numbers of people who were not allowed to leave the plant prior to being monitored, and large numbers of people who were not allowed to enter the plant until the situation had stabilized.

3. The mass exodus of day employees at 4:30 p. m. compounded the above situation.

4. The large number of people involved contributed to some disruption of communications and the necessary flow of information.

It is believed, however, that much of the confusion was more apparent than real, because during this period work was progressing, information was being obtained and passed on,

and the basic principles of the plant emergency plan, i.e., evacuation of personnel, provision of necessary medical assistance, isolation of radiation area, and monitoring of personnel, were being accomplished.

FOLLOW-UP SCREENING OF PERSONNEL

Badges left at the plant gates were taken to the secondary control center for the purpose of obtaining activity readings. In the course of these and prior badge checks, about 4,500 badge activity readings were made. Film badges turned in at the gates were subsequently processed for dose determinations.

Those persons who had been sent to the Y-12 Dispensary — because of preliminary badge surveys — were checked for personal contamination, interviewed briefly, and their badges rechecked. Individuals showing evidence of beta-gamma body contamination were scrubbed at the dispensary decontamination facility with soap and water, mild acids, etc., until it could be assumed that their body counts resulted from sodium activity in the blood. Body survey results are summarized in Table I.

Samples of blood and urine were collected for complete blood count and urinalysis. Samples were also sent to ORNL for sodium activation analysis and bio-assay procedures. The eight persons with the highest indium foil activation were given clean clothes and sent to be checked in the whole body counter for neutron activation of body sodium.

The first determinations of estimated individual doses based on indium foil readings were undertaken, and all badges with significant indium foil activation were sent to ORNL for more precise counting in a gamma-ray scintillation counter.

During the night of June 16 and the early morning of June 17, all indium foil readings were re-evaluated, and a list was compiled of 31 persons whose foil activities indicated a possibly significant neutron dose. At 8:00 a.m., June 17, those persons were instructed to report to the Y-12 Dispensary. In addition, investigations were made to ascertain the location of all persons in Building 9212 at the time of the incident. Throughout June 17 and 18, persons who might have sustained a significant dose, by reason of their reported proximity to the site of the incident, were routed through the medical test routine. Concurrently, indium foil readings were tabulated in order of decreasing badge activity and individuals high on this list also were sent to the dispensary in an effort to make certain that every person for whom any presumption of significant dose could be made was checked through the test routine.

Of the eight employees receiving the highest radiation doses, seven were referred to the dispensary on the basis of the initial measurement made on security badges at the secondary control centers. The activated foil in the badge of the eighth man was detected during the checking of badges collected at the gates. This employee received the same medical attention given the other seven.

The use of indium foil in the security badges made possible the early identification of employees who had been in the immediate vicinity of the reaction and facilitated their segregation from those employees who had not received sufficient radiation exposure to warrant concern. A possible unmanageable flood of employees to the dispensary was thereby forestalled.

SURVEY OF ENVIRONS AND DETECTION OF RELEASED ACTIVITY

Concurrently with the previously described activities, efforts were made to survey the environs and detect the release or subsequent fall-out of fission product activity. High-volume air samplers were set up outdoors at the points shown on Figure 2. Sampling was

begun at approximately 3:00 p. m. Unfortunately, this was about 50 minutes after the incident and 20 minutes after the ventilation supply and exhaust fans for the C-1 Wing had been cut off. Counting determinations of the activity collected on these samples indicated a maximum concentration of 2.5×10^{-11} μc/cc beta-gamma activity as of the time of collection. This is well below the 10^{-9} μc/cc permissible level of air-borne activity suggested by the National Committee on Radiation Protection (Table 2, NBS Handbook 52).[28] It should be noted that this conservative limit is intended to apply to an unknown mixture of long-life contaminants; its application to concentrations of the very short-life particulates and noble gases in this case is ultraconservative. As would be expected, no significant air-borne alpha contamination was detected.

Some indication of air-borne contamination released to the atmosphere prior to this may be obtained from two continuously recording beta-gamma air monitors which were located in Building 9207, about 3,000 feet downwind of the incident, and in Building 9204-1, about 1,400 feet south of the incident. Both of these instruments detected the initial direct gamma radiation from the actual excursion, and both detected subsequent increases in the level of atmospheric beta-gamma contamination (see charts, Figures 16 and 17). From these charts it can be seen that the level of initial direct radiation reaching Building 9204-1 was higher than that reaching Building 9207 because of the distance. The air-borne contamination, however, reached Building 9207 much sooner and in higher concentrations, since it was directly downwind. Because of the very short half-lives of these fission products (demonstrated on the monitor charts), and the relatively short length of exposure of any persons to the contaminated atmosphere, the levels of concentration detected constituted no particular hazard. It can be stated, with a high degree of confidence, that no significant concentrations of these activities reached any nearby populated areas.

Between 3:00 p. m. and 4:00 p.m., survey teams checked the parking lots along Bear Creek Road for evidence of contamination on the ground, paved areas, or automobiles. No evidence of beta-gamma contamination was detected and the automobiles were released from the parking lots.

RADIATION SURVEY AND RE-ENTRY OF THE BUILDING 9212 URANIUM RECOVERY AREA

At about 3:30 p. m., June 16, teams of health physicists began approximating the site of the incident by a series of perimeter radiation surveys. Radiation measurements observed were on the order of 0.2 mr/hr. These teams then entered the area and surveyed change houses and the south processing area of Building 9212. Radiation and contamination levels were such that employees were permitted to enter all areas except those within the secondary delimitation boundary shown in Figure 2.

At approximately 5:00 p. m., an emergency team made a "preliminary approach" survey of the C-1 Wing area. The radiation dosage rate at the southwest door of the salvage area (approximately 100 feet from the drum) was 60 mr/hr. When these men emerged from the area at approximately 5:10 p. m., the canisters of the gas masks they had worn read from 10 to 15 mr/hr, indicating that significant concentrations of air-borne contamination still existed in the area. Subsequent readings of these canisters indicated the expected rapid decay pattern of fission products.

Within a few hours after the incident, personnel were allowed to re-occupy all areas within Building 9212 other than those areas north of column line H. Control stations, manned by health physicists and stocked with the necessary items of protective equipment, were set up in the hallways to prevent unauthorized entry into the tertiary delimitation area shown in Figure 2. Authorized persons were permitted to enter the controlled zone only

in teams of two or more. Each team carried at least two radiation survey instruments. Required protective equipment included coveralls, shoe covers, stocking caps, rubber gloves, and either an MSA "All Service" mask (ultra filter) or a U. S. Army assault mask M-9 (with M-11 canister). Each person wore direct-reading pocket dosimeters and a film badge, and each was surveyed for personal contamination upon return from the controlled zone.

At 5:20 p. m., an extensive survey was made of C-1 Wing. Radiation readings ranged from 60 mr/hr, 100 feet away, up to 1,400 mr/hr about 15 feet from the drum.

By early evening of June 16, after joint evaluation of available evidence by management, radiation control specialists, and process supervision, the decision was reached that, in this instance, the most appropriate measure to be taken as a safeguard against a re-occurrence of criticality in the 55-gallon drum, was the insertion of cadmium metal to "poison" the contained solution. Accordingly, a scroll of cadmium sheet, 18" long, 14" in diameter, and weighing 8.9 kilograms was fabricated. At 9:30 p. m., this scroll, ma-nipulated with a ten-foot piece of pipe, was dropped into the drum. The vigorous reaction of the nitric acid with the cadmium resulted in the stripping of fission product gases from the solution, significantly raising the air-borne activity level.

This had been anticipated, however, and required no change in the planned operations. C-1 Wing and the adjoining C Wing were isolated, and a small fan, exhausting through a CWS filter, was started in C-1 Wing to maintain this area under negative pressure and prevent the spread of the air-borne contamination to other parts of the building.

At 10:00 p. m., a sample of solution was removed from the drum by remote handling tech-niques and transferred to Oak Ridge National Laboratory for fission product analysis. Nothing further was done with the drum or its contents for about 36 hours.

The results of a detailed survey of C-1 Wing, undertaken at 10:30 p. m., June 16, were as follows:

Position	Reading* (r/hr)
20 ft east of drum	1.1
12 ft east of drum	4.7
8 ft east of drum	9.8
12 ft west of drum	3.6
8 ft west of drum	8.0
12 ft north of drum	3.6
8 ft north of drum	9.8
2 ft north of drum	81.0

The immediate nuclear hazard having been disposed of, the problem of removing the highly radioactive solution from the operating area was approached. Several possibilities, in-cluding (1) removal of the drum with its contents, and (2) direct transfer of the solution into portable, shielded, safe bottles were considered. After deliberation, it was decided to install the necessary "safe" geometry tankage in an available radiographic cell (see Figure 3) east of C-1 Wing and to vacuum transfer the solution via stainless steel tubing

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* All readings were taken with an ionization chamber (cutie pie) approximately 3 feet above the floor with the exception of the last reading. The reading 2 feet from the drum was made at a height above the floor of approximately 1/2 the height of the drum.

into this tankage, where it would be allowed to decay prior to reprocessing. Accordingly, this safe tankage was fabricated and installed during the night of June 17.

Monitoring services were provided throughout the night of June 17 and morning of June 18 during the preparations for the transfer of the solution to the shielded, safe storage vessels. Dosimeters worn by persons installing the vacuum transfer line and others indicated that no person received any appreciable gamma dose while making these preparations. Additional surveys of the drum, made during these two days, give some indication of the decay of the radioactivity as shown below.

Date	Time	Reading (r/hr)	Position
June 16	10:32 p.m.	81	at 24 in. - middle of drum
June 17	10:30 a.m.	100	at 3 in. - middle of drum
June 18	10:00 a.m.	48	at 3 in. - middle of drum

"Wipes" or "smears" of the floor near the drum, at the time of the 10:30 a.m., June 17 survey, indicated that surface contamination had been confined to the immediate vicinity of the drum. The highest contamination detected on these smears was up to 250 mrad/hr from the smear itself and up to 16,000 d/m alpha contamination (normal 500 - 1000).

On June 18, vacuum transfer of the solution from the drum to the safe containers in the radiograph cell was begun by process personnel. Radiation detected during transfer varied from 52 mr/hr to 80 mr/hr at 3" from the transfer line. Radiation at the transfer line during water flushing of the drum and line was 38 mr/hr during the first 5-gallon flushing and 5 mr/hr during the second 5-gallon flushing. The empty line read 1 mr/hr at the exterior surface, the top of the empty drum read 5 r/hr, and the exterior surface, near the bottom, read 30 r/hr due to sludge in the bottom of the drum. The drum was removed to a shielded truck for transfer to Oak Ridge National Laboratory.

After the drum was removed, the boundaries of the controlled area were moved in to include only C-1 Wing itself. A slightly relaxed control was maintained over this area until necessary investigations had been completed. Spot air samples taken in C-1 Wing indicated that ventilation of the area would not contaminate surrounding areas. The supply and exhaust ventilation fans for C-1 Wing were turned on at 1:45 p.m., June 18. Subsequent air samples indicated air contamination was within permissible levels, and by 3:00 p.m., June 18, personnel were allowed to enter C-1 Wing itself without respiratory protection.

Decontamination of C and C-1 Wings started at 9:30 p.m., June 18, and continued intermittently for the next several days as portions of the area were released for decontamination by the investigation committee. Radiation monitoring and smear surveys were made to help direct and evaluate the decontamination efforts.

On June 20, a team, consisting of members of the investigating committee and UCNC operations and development supervision, began the physical investigation of the cause of the accident; valve positions were noted, and samples of solution were removed from pertinent piping and vessels. On June 22 and 23, the hydraulic tests described in Appendix D were carried out.

On June 23, all uranium recovery facilities were returned to normal operations.

EXHIBIT IV - NUCLEAR PHYSICS ANALYSIS*

It is the purpose of this section of the report to review the Y-12 radiation excursion as a study in reactor physics. A specification of the manner and rate of establishment of the neutron chain reaction system, the determination of the time which elapsed between its first becoming critical and its final return to subcritical together with the power pattern within this interval, and the mechanism by which the nuclear reaction was ultimately terminated would constitute a minimal description of the event. Although the process of transfer of liquid from one vessel to another is fundamentally simple, it is correct to infer from the earlier description of the present operation that many of the details of this transfer are not known even after some careful attempts at reconstruction with non-reactive solutions. It should be pointed out, parenthetically, that although the liquid transfer can certainly typify chemical operations in which accidents of this kind may be expected to occur, it is not believed that this same series of events would ever again ensue, thereby duplicating the consequences. This discussion has been undertaken with the prime intention of supporting other measures of personnel exposures by establishing the power-time pattern. Any value of a detailed analysis to the field of reactor physics is doubtful for the reasons given above. There is no evidence, however, of any basically unexpected physical phenomenon. A complete analytical description of the critical event, agreeing with the observations, would be gratifying and would satisfy the scientific curiosity of many readers. Such a description is not possible at this time.

DESCRIPTION AND ANALYSIS

A quantity of enriched uranium solution, sufficient to become critical, was accidentally drained from a bank of cylinders into the 55-gallon drum during an operation when only water was believed to be in the cylinders. This solution was followed by a relatively larger volume of water (or more dilute solution) which ultimately reduced the concentration below the value which would maintain a nuclear chain reaction in the geometry of the drum. A great many observations have been combined to present here a qualitative description of the course of events with some of the details being recorded in appendices.

Chemical analysis has shown 50 g U^{235}/liter to be the most concentrated solution available for transfer to the drum and 2.5 kg U^{235} as the mass transferred. A plot, Figure 11, of a short extrapolation of measured critical dimensions of $U^{235}O_2F_2$ solutions ($\sim 90\%$ U^{235}) gives critical masses as a function of critical volumes in a 21.75 in. diameter unreflected steel cylinder. It is seen that the above quantities set 7.6 and 17.2 inches as the lower and upper limits on the critical height. Since both the sequence of valving operations postulated and data from the hydraulic reconstruction experiments stipulate some dilution of the original solution as it flowed into the drum, a volume of 56.2 liters containing 2.10 kg U^{235} standing at a height of 23.45 cm (9.23 inches) is selected as the initial delayed critical configuration. This selection is justified by three factors - the reactor analysis, based on the initial conditions, yields time intervals consistent with what is believed to be the observed duration of the excursion; the assumed critical height agrees with both the liquid level estimated in the drum by the individual standing nearby at the time of the first indication of a reaction and with the distribution of induced activity in the walls of the drum described below. If it is assumed that the concentration of the solution subsequently added to the drum is uniform and that the volume in the drum reached 180 liters when the entire

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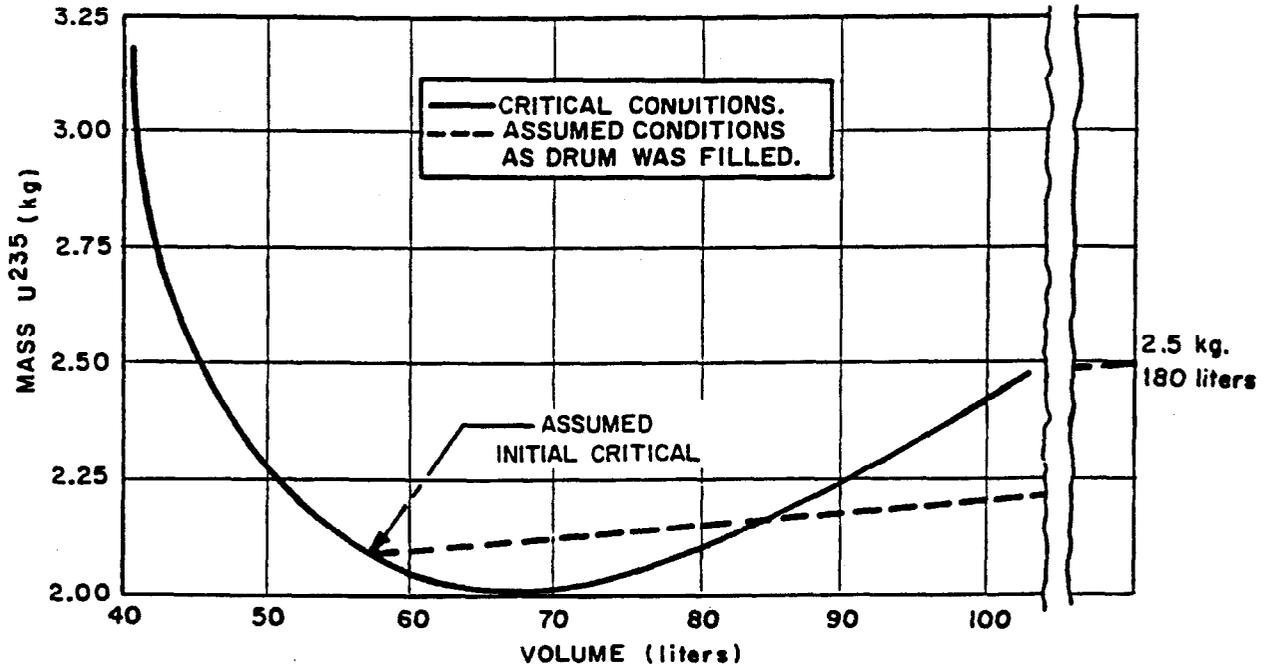
Figure 11
MASS-VOLUME RELATIONS IN 55 GALLON DRUM DURING RADIATION EXCURSION
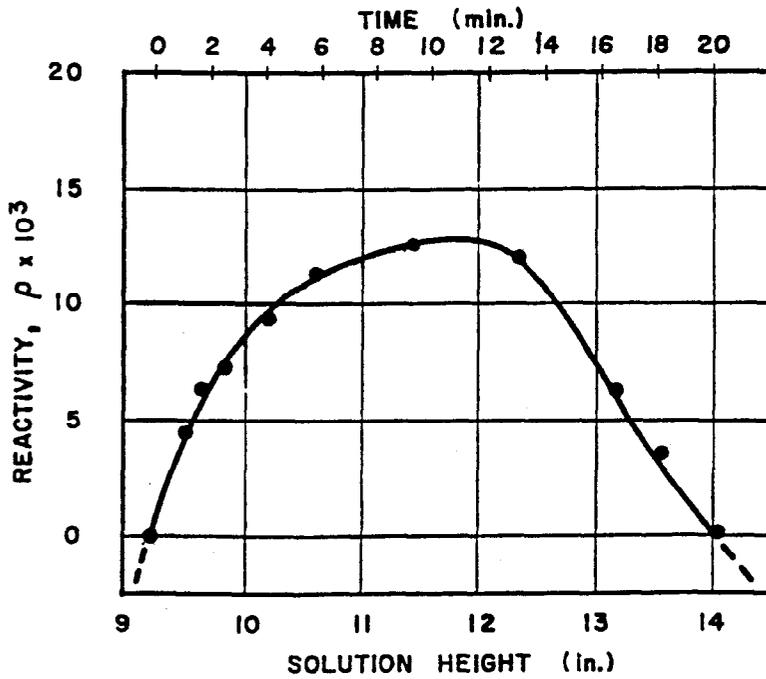
Figure 12
CALCULATED REACTIVITY IN DRUM DURING RADIATION EXCURSION
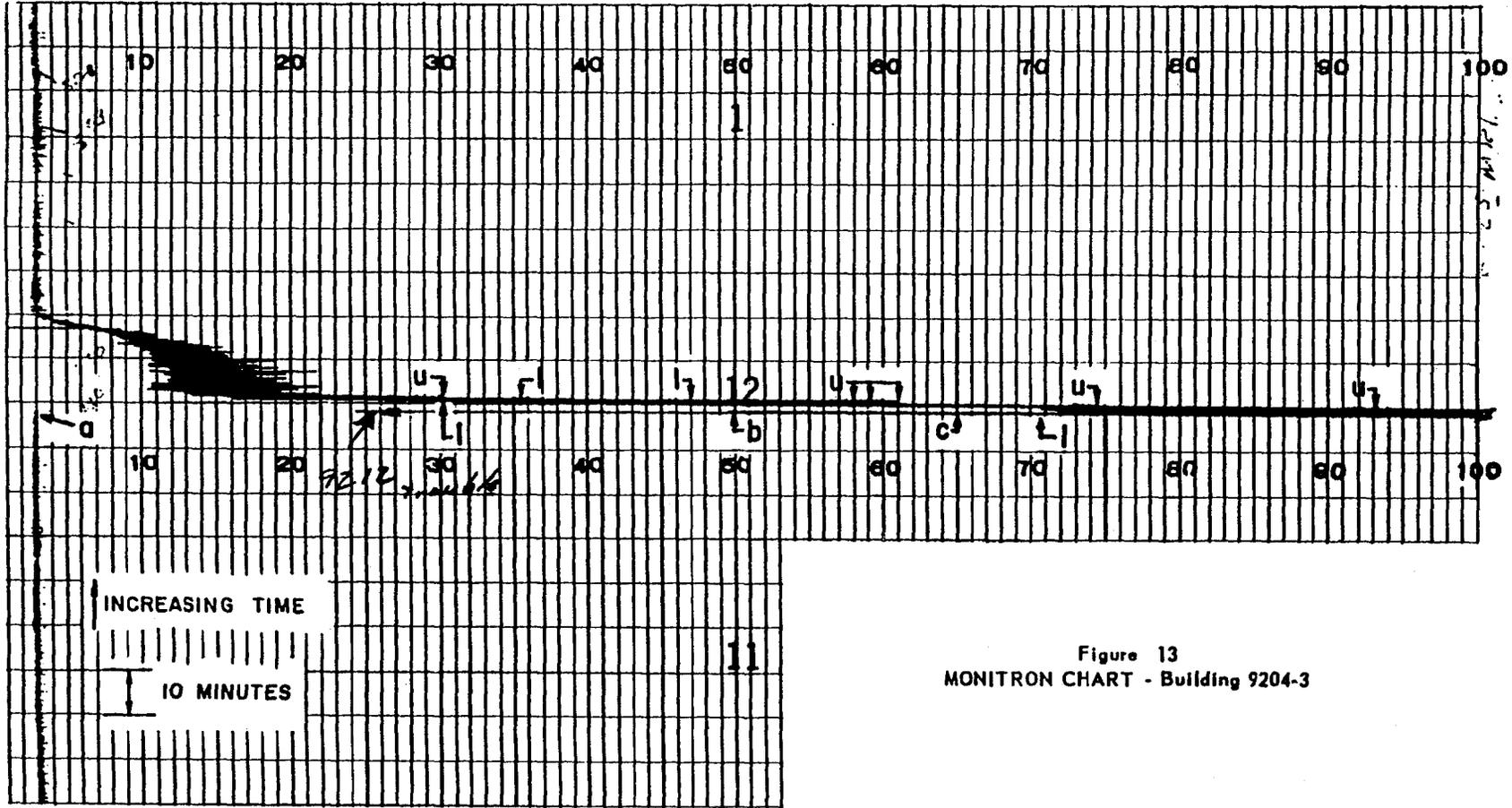
Figure 13
MONITRON CHART - Building 9204-3

2.5 kg U^{235} had been transferred, the mass-volume relation in the drum is described by the straight line on the plot. A comparison of this relation with the critical mass-critical volume curve allows an estimation of the reactivity at subsequent times. The details of the analysis are given in Appendix B and the results are in Table II. The reactivity as a function of the solution height in the drum and of the time after delayed critical is shown in Figure 12. The time scale was derived from some of the post-accident hydraulic measurements reported in Appendix D, particularly the rate of flow shown in Figure D. 1. The duration of the excursion, by this analysis, is 20 minutes. The effects of the neutron absorption by the nitrogen and of the neutron reflection by the concrete floor, located approximately 3 inches below the drum, are somewhat compensating and have been neglected. The bases for, and the results of, the above analysis are also not inconsistent with the following additional significant observations:

<div align="center">Table II</div>

<div align="center">CALCULATED REACTIVITY DURING RADIATION EXCURSION</div>

| Time* (min) | Solution | | | | Critical Mass** U^{235} (kg) | Reactivity, $\rho \times 10^3$ |
| | Height | | Volume (liters) | Mass U^{235} (kg) | | |
	(cm)	(in.)				
14.6	23.45	9.23	56.2	2.10	2.10	0
16.4	25.07	9.87	60.1	2.11	2.04	7.5
20.0	27.12	10.68	65.0	2.12	2.02	11.4
23.6	29.20	11.50	70.0	2.13	2.03	12.4
27.1	31.29	12.32	75.0	2.15	2.07	11.8
29.9	32.82	12.92	78.7	2.16	2.15	7.5
34.6	35.67	14.04	85.5	2.18	2.18	0

*The drum begins to fill at zero time.

**This mass in the volume shown at the left will be critical.

1. Records from Radiation Monitors

During the excursion a radiation detection instrument, sensitive to both neutrons and gamma rays, was operating in Building 9204-3 some 1400 ft. distant and cross wind from C-1 Wing of 9212. (See Figure 2.) The characteristics of the detector and its associated equipment are described in Appendix C. Figure 13 is a reproduction of its recorded trace during that time. The following discussion is based on a 7x enlargement of this record although most of the points are discernible on the reproductions shown here. Figures 14 and 15 are two parts of the enlargement illustrating some of the detail in the record. The radiation intensity is observed to first increase extremely rapidly from "a", Figure 13, driving the pen off scale, then decrease to "b", repeating to "c", all in about 15 seconds determined by the chart drive speed. During the next interval, the signal oscillated an indeterminate number of times, finally decreasing to about five times background, 2.8 minutes after the first rise in level. The upper and lower limits of some of these pulses, discernible on the enlarged trace, are indicated by "u" and "l", respectively. This (average) high-intensity field is then followed by a slowly decreasing level of some 18 minutes duration again characterized by pulses. One peak, at 61 on the scale, is separated inordinately in time from adjacent portions of the trace and may be due to a peculiarity of this detector, particularly since it is not readily identified on the charts from either of the air monitors referred
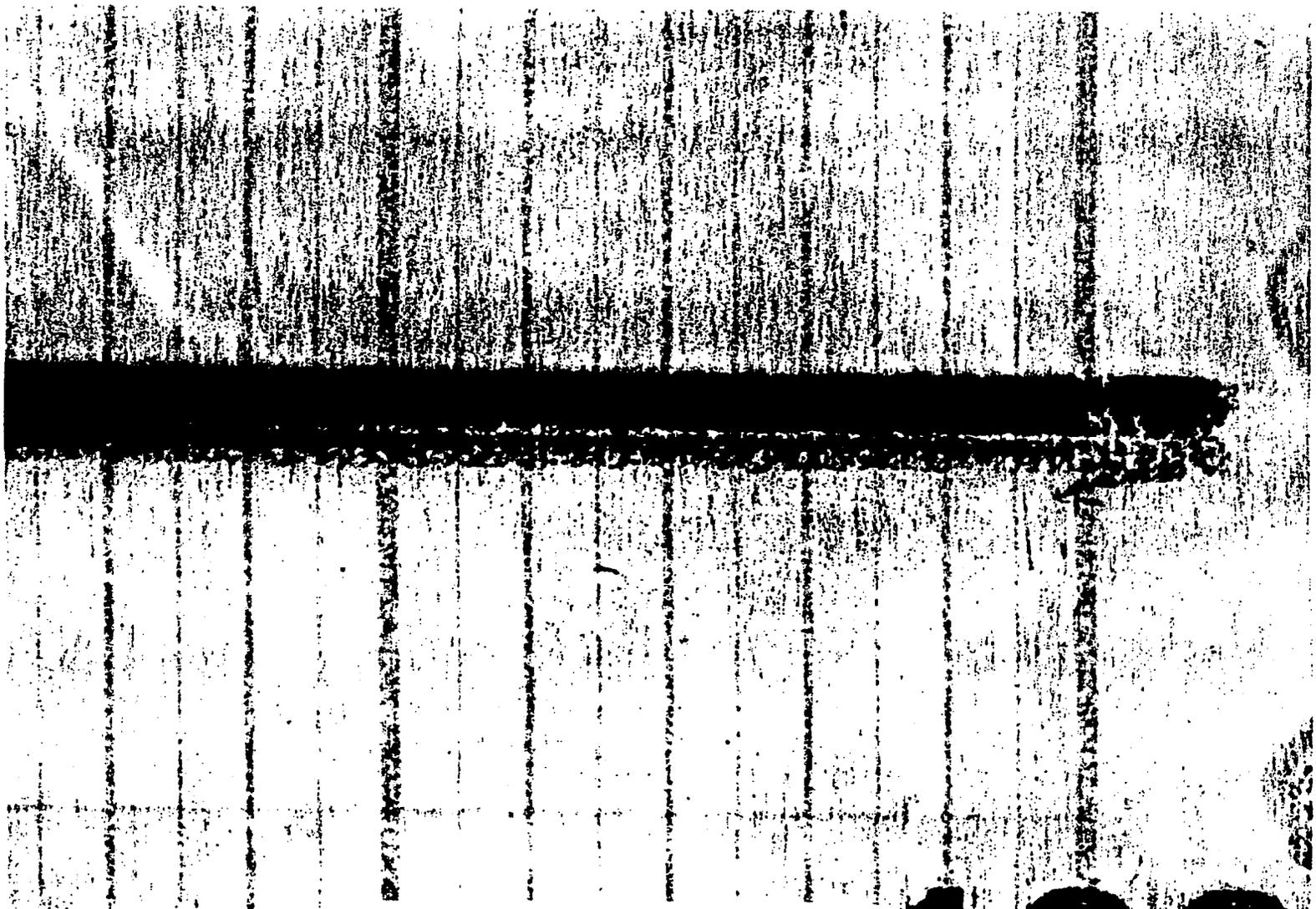
Figure 14
TOP SECTION OF MONITRON CHART
Enlarged Approximately 9.5 Times
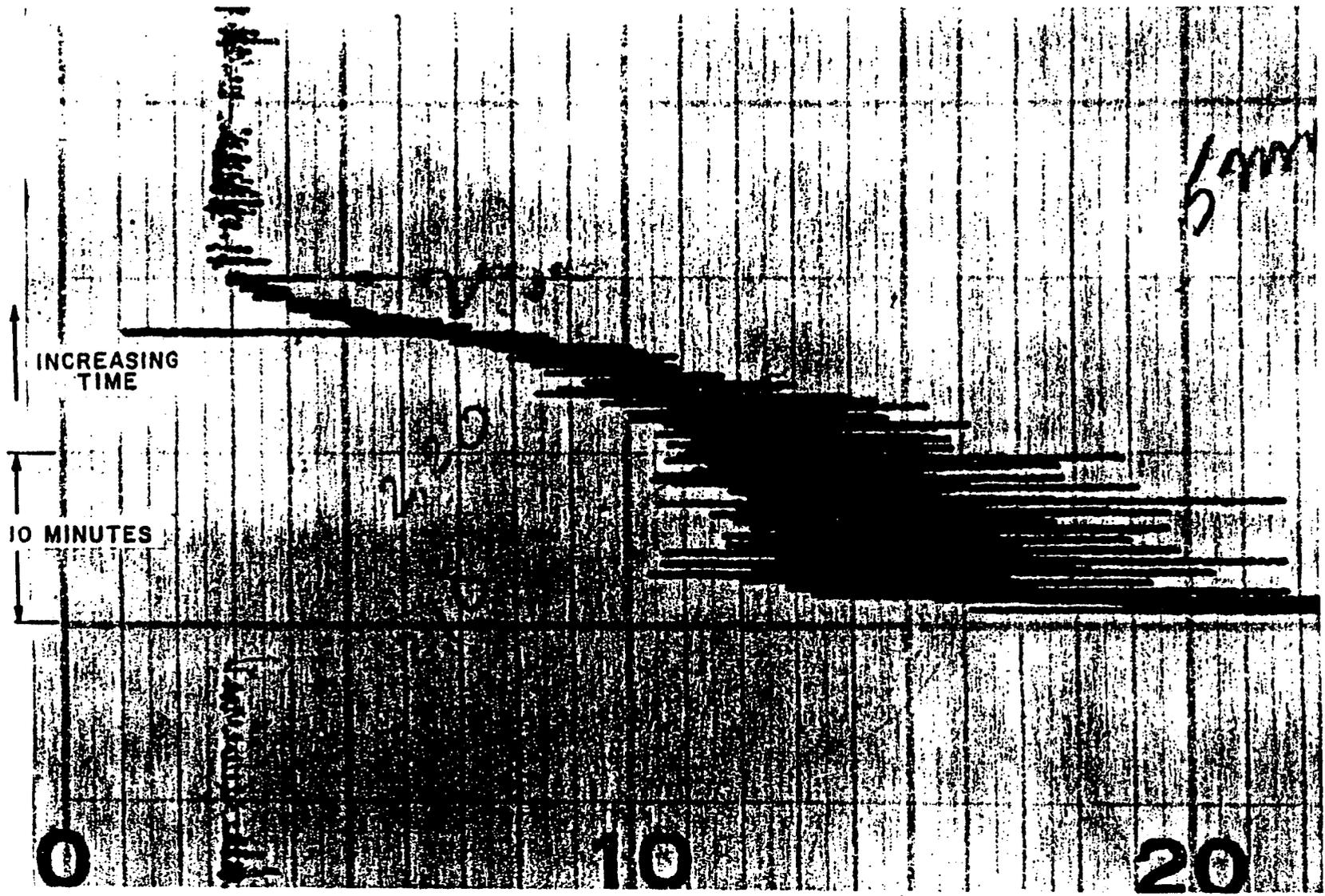
Figure 15
BOTTOM SECTION OF MONITRON CHART
Enlarged Approximately 4 Times

to below. Although this neutron detector is equipped with two sensitivity ranges (25 and 125 mr/hr, full scale, respectively), it is believed to have remained on the more sensitive scale during the entire period discounting the inference that some of the discontinuities are due to scale changes.

The enlargements, Figures 14 and 15, of the chart reproduced in Figure 13 were selected to illustrate qualitatively the power pulses which occurred during the excursion. They show the structure at full scale and during the extended period of relatively lower activity, respectively.

The over-all time of the excursion is shown by this trace to be 21 minutes. The absence of a strong neutron field within the drum as it initially became critical probably means that the critical height was reached prior to the initial energy release; that is, even though the system was critical, it did not manifest itself until it was "triggered" at a low power level, in a statistical manner, by ambient neutrons. This dormant period may have been a few tens of seconds, well within the accuracy of the above estimate.

Two additional radiation monitoring instruments were operating during the time of interest, both being air samplers which detect the gamma radiation from particulates collected on a filter surrounding a Geiger tube. Figures 16 and 17 are copies of the records from these instruments which were located in Buildings 9207 and 9204-1. Each chart shows the direct radiation from the excursion and, subsequently, the arrival of the air-borne activity. The differences in the interval between the detection of these two activities at the two locations, about 12 minutes and 48 minutes, respectively, can be qualitatively correlated with the recorded wind direction at that time. Building 9204-1 is adjacent to Building 9204-3, the site of the detector discussed above, so the delay in the arrival of air-borne activity at the two locations is expected to be comparable and equal to about 3/4 hr. This observation is presented as evidence favoring the interpretation of the extended, low-level activity in Figure 13 being direct radiation. In addition, of course, Figure 15 does not typify a radioactivity decay curve. No other quantitative interpretation is made of Figures 16 and 17. Figure 18 shows one of several traces recorded in the control room of a cyclotron in Building 9204-3. These traces also indicate the duration of the excursion to be approximately 20 minutes.

There are a number of undocumented observations made with portable radiation detection instruments in the vicinity of Building 9212 to the effect that the radiation level remained constant for times of 5 to 15 minutes which is at least supporting evidence that the source of radiation was extended in time.

2. Analysis of Induced Activity in the Drum Wall

Activity was induced by neutrons in the components of the stainless steel of which the drum was constructed. Analyses of these activities yield at least relative values of the neutron exposure and, hence, of the neutron flux at various elevations along the side of the drum. The fast neutron measure was derived from the activity of Co^{58} arising in the Ni^{58} (n, p) Co^{58} reaction assuming an 80 mb cross section. The thermal neutrons were evaluated from the Cr^{51} activity from the Cr^{50} (n, γ) Cr^{51} reaction with a cross section of 15 barns. The steel analyzed 17.99 and 11.84 wt% chromium and nickel, respectively. The data are recorded in Table III and are plotted in Figure 19. The results from additional samples from peripheral locations at three elevations showed no significant asymmetry in the flux pattern in horizontal planes. It is interesting to note that the maximum activation occurred between 3 and 5 inches from

AIR-BORNE ACTIVITY

SCALE CHANGE

DIRECT RADIATION
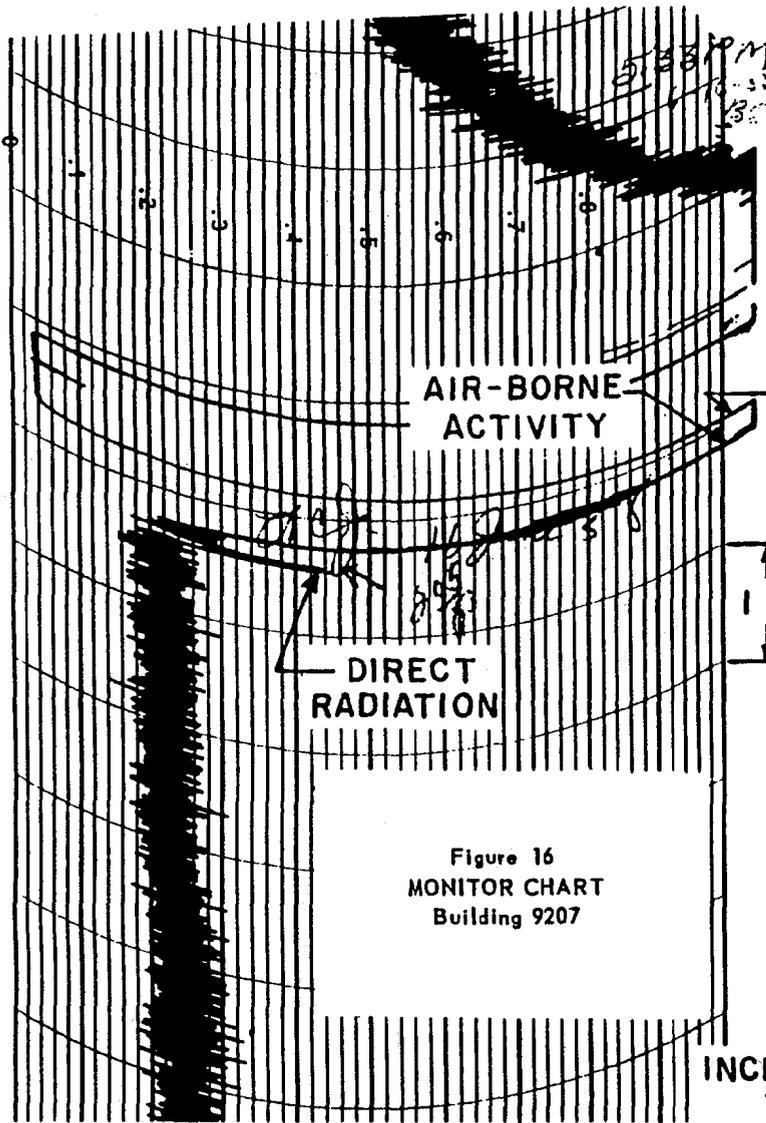
Figure 16
MONITOR CHART
Building 9207

AIR-BORNE ACTIVITY

SCALE CHANGE

DIRECT RADIATION

1 HOUR
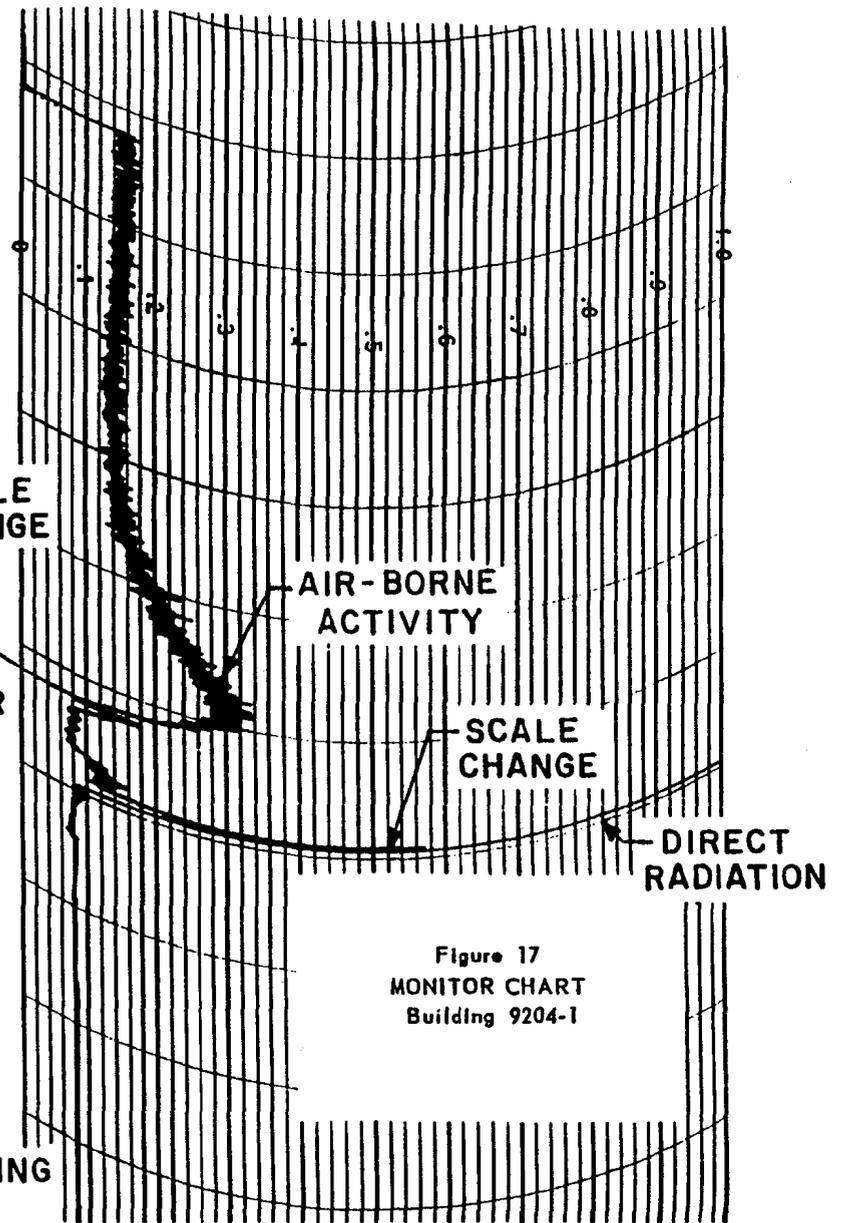
Figure 17
MONITOR CHART
Building 9204-1

INCREASING TIME
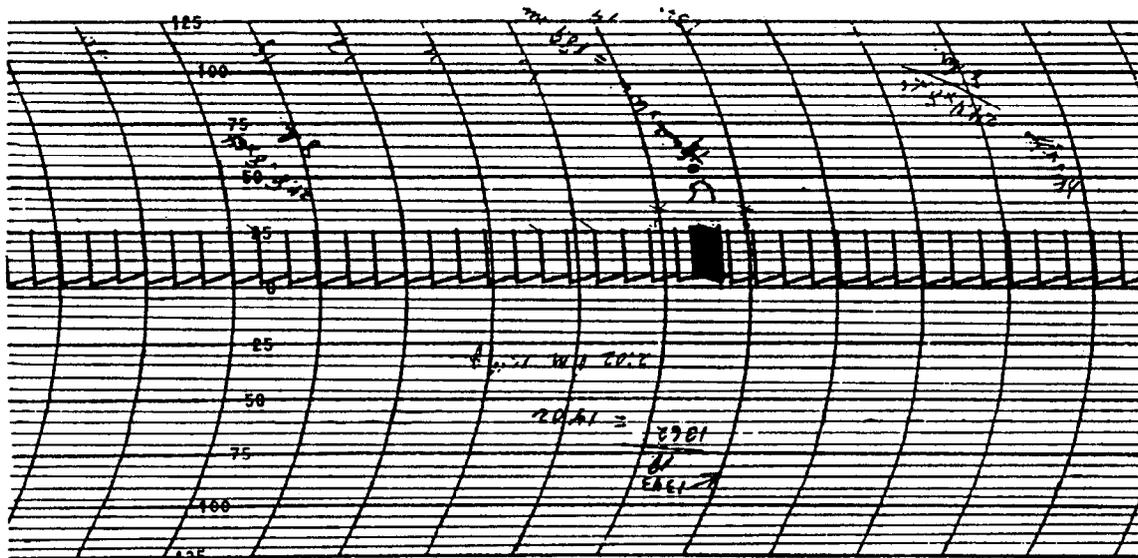
| I hr. |

◄— INCREASING TIME

Figure 18
MONITOR CHART
Building 9204-3

60
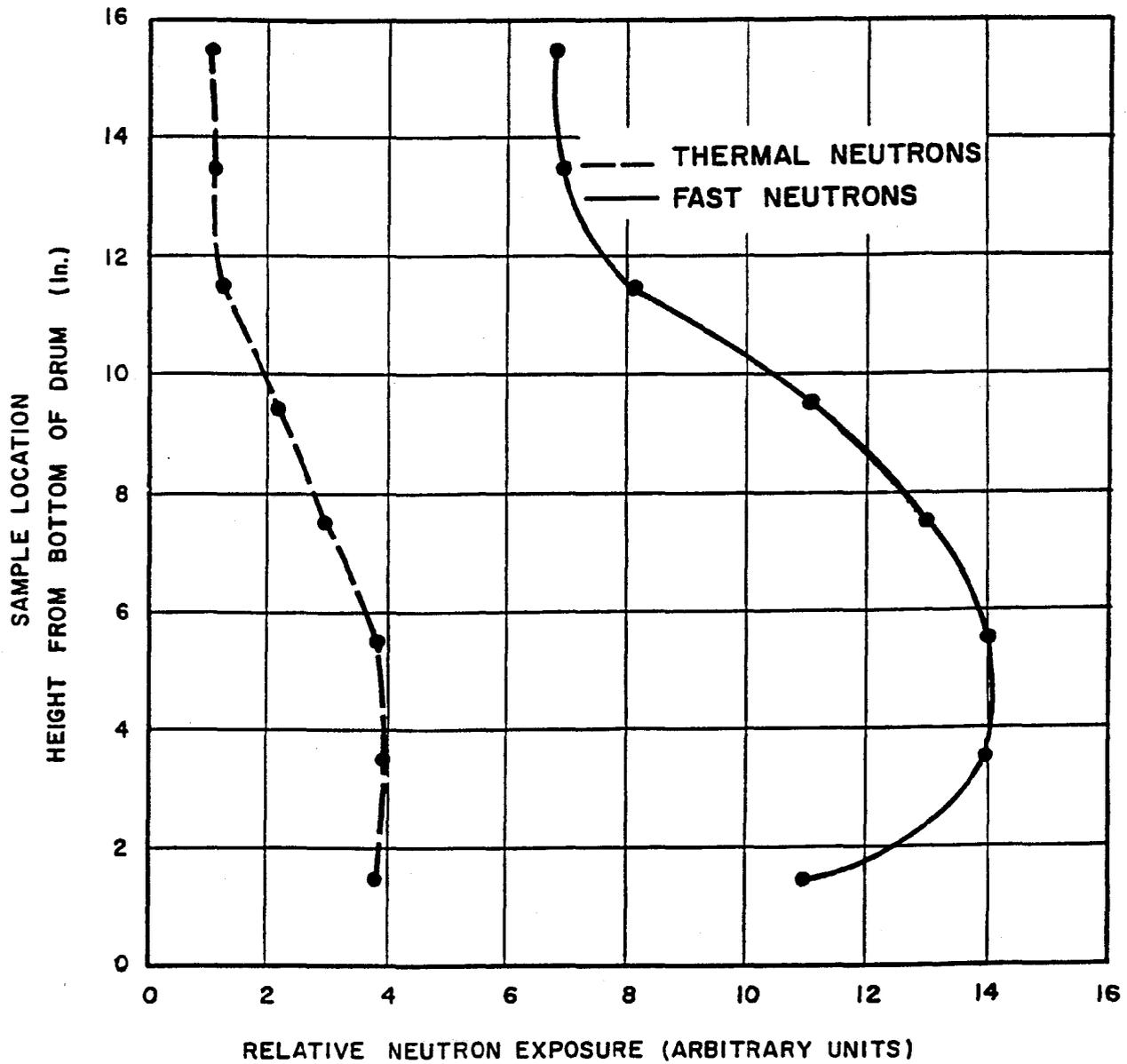
Figure 19
ACTIVITY OF STAINLESS STEEL SAMPLES FROM Y-12 DRUM

the bottom, and that there is some evidence of asymmetry in the thermal neutron distribution, implying an effect of the stainless steel covered concrete floor as a reflector. If the peak activity is associated with some weighted center of reactivity of the supercritical system, an effective reactor height of 10 inches is not inconsistent with the assumptions in the above analysis. No estimate of the energy in the excursion has been made from these values of the steel exposure.

Table III

RELATIVE ACTIVITY OF STAINLESS STEEL SAMPLES FROM DRUM

Height From Bottom of Drum (inch)	Thermal Neutron Activation	Fast Neutron Activation
	(arbitrary units)	
15½	1.0	4.8
13½	1.1	4.9
11½	1.2	6.1
9½	2.1	9.1
7½	2.9 .	13
5½	3.8	14
3½	3.9	14
1½	3.8	11

Center of bottom: Thermal neutron activation = 18; fast neutron activation = 28.

NOTE: The above values were obtained by γ - spectrometry; radiochemical analysis of three typical samples gave fast neutron activations 5 to 15% lower.

3. Chemical and Radiochemical Analyses; Energy Release

The number of fissions which occurred during the power excursion, and hence the energy release, has been determined from radiochemical analyses of samples of the activated uranium solution. A sample of limited size was taken from the top of the liquid in the drum on June 16, about eight hours after the accident. Since this sample may not have been representative of the entire volume of the solution, a pair of samples was taken on July 15 from the well-homogenized solution as it was then stored in shielded containers. It must be pointed out that some dilution occurred upon transfer from the drum which accounts for the differences noted in the specific activities and the solution volumes in the data tabulation. This, of course, in no way invalidates the method, provided the volume is measured at the time of sampling. From the concentration of appropriate fission products obtained by measuring their characteristic radiation, the decay constants and the fission yields of the isotopes, and the elapsed time since the excursion, the number of fissions which occurred per unit volume of the solution was obtained. All of the analytical results and a weighted "best value" of the energy released in the excursion, 2.6×10^{20} Mev from 1.3×10^{18} fissions, are given in Table IV.

It will be noted that large discrepancies exist in the data of Table IV. A partial explanation lies in the existence of noble-gas precursors of most of the nuclides measured in the analysis. A list of these precursors is given in Table V. Gases of longer half-lives obviously have higher escape probabilities from the liquid than those of short half-lives. Further confirmation of this explanation is obtained from observations on samples of solutions in which the fission concentrations have varied; the difference between the fission concentration values derived from Ba^{139} and Mo^{99} increases with

Table IV

RADIOCHEMICAL AND CHEMICAL ANALYSES AND ENERGY RELEASE ESTIMATE

Nuclide	Method*	June 16 Sample	July 15 Samples	
			I	II
Mo^{99}	βct.	7.7×10^{12} Fissions/ml		
Ba^{140}	βct.	6.0×10^{12} "	2.8×10^{12} Fissions/ml	3.0×10^{12} Fissions/ml
Ba^{140}	γS**	6.5×10^{12} "		
La^{140}	γS	4.6×10^{12}*** "		
Ba^{139}	γS	2.2×10^{12} "		
Ce^{141}	γS		5.8×10^{12} "	5.6×10^{12} "
Ce^{144}	βct. #		4.1×10^{12} "	4.0×10^{12} "
Zr^{95}	γct.		3.5×10^{12} "	3.6×10^{12} "
Cs^{137}	γS		0.6×10^{12} "	0.6×10^{12} "
Sr^{89}	βct.		0.5×10^{12} "	0.5×10^{12} "
"Best Value"		7×10^{12} Fissions/ml	5×10^{12} Fissions/ml	5×10^{12} Fissions/ml
Uranium		14.0 gU^{235}/liter	9.6 gU^{235}/liter	9.6 gU^{235}/liter
Volume of Solution		180 liter	252.8 liter	
Total Fissions		1.3×10^{18} Fissions	1.3×10^{18} Fissions	
Energy		2.6×10^{20} Mev = 11 kw hr.		
Uranium Mass (From Average Analysis)		2.5 kg U^{235}		

* Activities were measured by β or gamma counting (β ct. or γ ct.) or by scintillation spectrometry (γ S).
**Assuming intensity of 0.54 Mev$_\gamma$ = 21.5% [15]
***After several hours growth in separated Ba.
#With absorber

Table V

PROPERTIES OF NUCLIDES

Nuclide	Fission Yield[2] fraction	Gas Precursor
Sr^{89}	0.048	3.2 m Kr
Zr^{95}	0.064	"short" Kr
Mo^{99}	0.062	- -
Cs^{137}	0.059	3.9 m Xe
Ba^{139}	0.063	41 s Xe
Ba^{140}	0.061	16 s Xe
Ce^{141}	0.060	1.7 s Xe
Ce^{144}	0.061	$\sim$ 1 s Xe

increasing concentration, i.e., increasing heat output. A qualitative confirmation is also furnished by the knowledge that gases escaped from the drum. The "background" of the In Vivo counter at Y-12, for example, was due largely to Cs^{138} at 5:30 p.m., June 16, and there was evidence of Rb^{88} earlier; both of these are daughters of gaseous isotopes. In Table IV, the apparently low values of the fission concentration in the July 15 sample, based on Zr^{95} and Ba^{140}, may be explained by the well-known hydrolytic behavior of Zr and possible similar loss of Ba due to traces of sulfate (in addition to the loss of 16 sec Xe^{140}). Disagreements between values from Ce^{141} and Ce^{144} have not been explained.

4. Hydraulic Reconstruction Experiments

Considerable effort has been expended in attempts to reconstruct the flow patterns of the several volumes of liquids as they were added separately to a somewhat complex system of piping, partly mixed therein, and finally drained into the 55-gallon drum in a stream of variable uranium concentration. An aqueous solution of cadmium nitrate, adjusted in concentration to approximate the fluid properties of the mislocated uranium solution, together with the volume of water believed appropriate, were used in these tests. Flow rates into the drum were measured and frequent samples were obtained from both the top of the liquid in the drum and from the line as the drum was filled. Although, in principle, the analyses of these samples allow estimation of the uranium inventory and concentration in the drum as a function of time, it is not certain they are truly representative of the conditions in the drum on June 16. This uncertainty may be due, for example, to irreproducible mixing conditions, particularly since the first emission of nuclear energy caused at least local turbulence. The fill rate was used in the above reactivity analysis, but it has not been possible to correlate the time-uranium inventory data with the uranium concentrations required for criticality. The results of these reconstruction experiments are, however, recorded in Appendix D.

5. General Observations

There are two additional observations which should be recorded for consideration.

There was no strong ambient neutron field at the scene of the accident, the most likely source being the $O(\alpha, n)Ne$ reaction between the U^{234} alpha particles and the oxygen in the water. As a consequence, the system may have become well above delayed critical before the power level increased from zero.

There was no evidence of the rapid production of large quantities of gas or vapor. There was, for example, no liquid on the floor, under or adjacent to the drum, nor was there an inordinate amount of localized fission product contamination on the fill tube (see Figure 1) except where it was in contact with the liquid. The nature of the process in the area precluded any meaningful alpha particle contamination survey for dispersed uranium. These observations minimize any assumption of vigorous boiling of the solution. There is no clear explanation of why the solution was not dispersed outside the drum, although speculation can relate the violence of the turbulence to the rate and mode of the approach to critical, to the characteristics of the first power surge, and possibly, to the geometry of the vessel. Comparison of experiences with other critical accidents[24] with solutions shows that large as well as insignificant discharges of liquid have been observed in events with the same energy release.

DISCUSSION

An attempt has been made in the preceding paragraphs to record and interpret a rather wide variety of observations made in connection with the Y-12 radiation accident on June 16.

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It is believed, unquestionably, that sufficient enriched uranium solution was added to a 55-gallon drum to become critical, that the concomitant energy release occurred during an interval of a few minutes in which the effective reactivity and the power level oscillated a number of times, and that the chain reaction was ultimately stopped by the addition of water to the solution, a very fortunate circumstance of the valve, through which the solution was admitted, being left open as personnel evacuated the area. The quantity of uranium involved and the energy developed in the reaction are moderately well known; the uncertainty in the duration of the excursion and the fluctuation in the reactivity have not allowed an evaluation of the peak power. The potential personnel hazard from the ionizing radiation generated in the observed number of fissions is developed elsewhere in this report and is compared with the exposures experienced by employees in the vicinity of the accident.

As pointed out earlier, it is impossible to reconstruct the reactivity-time pattern and there are, no doubt, several combinations of events which can account for the observations. It is intended to outline very briefly here one possible sequence.

With reference to the power-level relation, indicated by the radiation monitor record described in Figure 13, the following sequence of conditions is suggested. In the absence of a source of neutrons, this system was prompt critical before any energy was emitted. Once started, however, the power level rose quite rapidly to a high value. The energy from these fissions produced gases by dissociation,[3] reducing the density and driving the solution subcritical. Exit of these gas bubbles once more made the system prompt critical and, with the delayed neutrons as a source, the power level again rose. This cycling persisted for an estimated 2.8 minutes, during which, of course, the temperature of the solution increased. Boiling* finally ensued, causing a sharp decrease in density and a concomitant return to subcritical indicated by the decrease in the instrument deflection to about scale reading "20", Figure 13. Following this steep descent, the system settled into an equilibrium condition somewhere in the delayed critical range where it was controlled for about 18 minutes by vapor formation and, to a lesser extent, by decomposition gases. The system remained delayed critical until the inflow of water reduced the concentration to a final subcritical value.

In previous experiences with accidental critical assemblies, described in Appendix E, which have been limited to a single burst by some reactor shutdown mechanism, the energy release has been from 10^{16} to 10^{17} fissions, a not unreasonable estimate of the first of the several pulses in this case.

It is appropriate to consider, briefly, other courses the reaction may have taken and the consequences which could have resulted.

A shutdown mechanism for a supercritical solution, alternate to a dilution, is the removal of sufficient water to increase the chemical concentration beyond that which will support a nuclear chain reaction under the other existing conditions. This removal would be by dissociation and vaporization. In this particular instance, the above analysis shows, in Figure 11, the limit to be about 54 g U^{235}/liter with 2.5 kg of U^{235}, a value, incidentally, not much different from that of the original solution. Had no water been added in the operation, the excursion might not have been as severe as the one experienced.

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* The permanent deformation of a polyethylene liner, present in the drum during the excursion, into the convolutions of the drum is evidence that the temperature of the solution at least approached the boiling point. The energy release obtained from the fission product analyses was adequate to boil the solution.

A third shutdown mechanism is a dispersal of the fissionable material, the causes of which are difficult to predict from past experiences.

It is believed that the Y-12 incident is a point of departure for predicting the causes and effects of possible future accidents. It does not set an upper limit to the consequences to be expected for, as pointed out above, there were associated with it a number of unique, fortunate circumstances which reduced the problem significantly. A study of this type of accident has been made,[8] which is supported in part by the findings reported here, and which, in the absence of externally applied shutdown mechanisms, predicts, much more severe results.

EXHIBIT V - DOSIMETRY*

There are three sources of evidence from which information may be obtained relative to the magnitude of exposure and absorbed dose of ionizing radiation received by personnel as a result of radiation that emanated from the critical mass of U^{235} solution accidentally collected in a 55-gallon drum in C-1 Wing of Building 9212 in the Y-12 area at approximately 2:05 p.m. on Monday, June 16, 1958. These sources of evidence are as follows: (1) the critical assembly or radiation source, (2) the dosimetry, and (3) the clinical symptoms of the exposed individuals. Of these, (1) and (2) will be discussed in this section.

THE CRITICAL ASSEMBLY OR THE RADIATION SOURCE

Some of the earliest and most convincing evidence of the exposure of personnel to a large burst of ionizing radiation was obtained from the study of the critical assembly (the 55-gallon drum) and from observations in the C-1 Wing of Building 9212. Three of the workmen, Employee "A", Employee "B", and Employee "F" saw the blue glow known as Cerenkov radiation. They did not at the time necessarily associate this blue glow with a serious event or an accident, partly because welding had been going on in the area that same day, but its occurrence a few seconds prior to the alarm - in retrospect at least - indicates they must have been close to a large source of ionizing radiation.

The second evidence of exposure was the sounding of the automatic radiation alarm system. This system consists of building monitors which actuate an alarm when the dose rate at the detectors exceeds 3 mr per hour. There are six monitors in the C, D, and E Wings of Building 9212, and one of these was only about 50 feet from the 55-gallon drum in C-1 Wing.

The third evidence of radiation exposure, that was apparent at the time, consisted of a fog and an odor which may have been produced by the high flux of ionizing radiation. The fog may have been the result of fumes from the carbitol and nitric acid (see Table M. 1) in the 55-gallon drum, or it may have been an illusion caused by direct action of ionization on the ocular systems, and the odor may have been caused by radiation produced nitrous oxide commonly associated with ozone. The fog was observed by Employee "A", and the odor was detected by Employee "A", Employee "B", and Employee "E".

The above (the blue glow, the sounding alarms, the fog, and the odor) were the only immediate evidences of exposure to ionizing radiation. Shortly afterwards, as the Y-12 employees were evacuated, their security badges were checked with survey meters. Each of these badges contained a small imbedded indium foil, and this activated foil read from 50 mr per hour to 8 mr per hour in the case of the eight individuals who were exposed in the C-1 Wing of Building 9212. Later a Geiger counter was placed against the throat of each of these individuals and the counter read from 0.5 to 0.2 mr per hour on these same individuals. About three hours later, when the health physics surveyors entered Building 9212, they obtained high readings on their survey meters as they entered C-1 Wing and as they approached the 55-gallon drum from several different directions. By this time, there was little doubt concerning the location of the source of high levels of radiation in

* Principally prepared by K. Z. Morgan, G. S. Hurst, R. H. Ritchie, and L. C. Emerson, Health Physics, Oak Ridge National Laboratory.

the Y-12 area. An examination cf the source, i.e., the contents of the 55-gallon drum, the next day furnished unequivocal evidence that indeed a nuclear reaction had taken place. An analysis of the contents of the 55-gallon drum indicated that at that time (24 hours after the excursion) the drum contained 180 liters of liquid in which were dissolved and suspended 2.5 kg of U^{235}, mostly in the form of uranyl nitrate. In addition, there was a small amount of solid material on the bottom of the drum. The height of the critical assembly at the time of the excursion was determined as described on page 59 et. seq. and was found to be ~9 inches. The inside diameter was 21.75 inches (both measurements were made with reference to the inside of a plastic liner that had been fitted inside the 55-gallon drum).

A fission analysis of the fuel (see Table IV) indicated that about 1.3×10^{18} fissions took place during the excursion.* From a study of the rate at which the solution entered the 55-gallon drum and from an examination of the radiation exposure charts from various parts of the Y-12 area (see Figures 13, 16, 17, and 18), it was evident that the reactor in the 55-gallon drum had gone through not one but several excursions, and these altogether may have lasted over a period of several minutes. Criticality is believed to have been reached when a uranyl nitrate solution flowing into the drum from the plastic tubing (see Figure 1) reached the critical mass. In any event, there was no violent reaction and the solution may have cycled several times while the exposed personnel were nearby and after they had left the area. Film badges and pocket meters of the health physics surveyors and others who entered the C-1 Wing of Building 9212 about two hours after the excursion but before the cadmium safety curtain was dropped into the 55-gallon drum do not indicate any exposure except that received from the fallout contamination, and charts continuously recording the radiation level in several nearby Y-12 buildings do not indicate that any other radiation excursions occurred more than a few minutes following the initial burst of activity.

Crude estimates were made of the fast neutron dose received by the exposed personnel on the basis of the total number of fissions in the drum, neutron leakage from the drum, and the inverse square law. The values, shown in Appendix H, are unreasonably large and in no way consistent with early clinical observations of the exposed personnel.

DOSIMETRIC EVIDENCE FOR RADIATION EXPOSURE

The individuals accidentally exposed were wearing security badges which contained a small piece of indium metal. This metal has a strong absorption resonance for neutrons with energy near 1.44 ev. It becomes radioactive and is often used to monitor slow neutron flux. However, it was considered that the dose measurements, as indicated by the indium foils, were not of sufficient accuracy to use as a final basis for determining medical treatment. Thus, since body fluid analysis with calibration appeared to provide the most reliable method for determining exposure, this method was utilized for quantitative evaluation of the exposures.

Both urine and blood samples were counted with sensitive scintillation counters adjusted so that gamma rays of energy above 0.66 Mev and above 2.0 Mev were indicated on separate registers. Sodium-24 decays primarily by emitting a beta particle and two gamma

* An earlier calculation applying the Way-Wigner relation and based on a radiation survey measurement of 23 r/hr at 2 feet from the surface of the drum and taken at 20.5 hours following the incident yielded an estimate of 2.2×10^{18} fissions and a similar calculations using unpublished data by Spencer and Hubble yielded an estimate of 3.7×10^{18} fissions (see Appendix G).

rays of energy 2.75 Mev and 1.38 Mev. The 2.0 Mev adjustment of the counters discriminated against activities other than sodium-24.

Fast neutrons from the reacting assembly entered the body and were moderated by the hydrogen, oxygen, and carbon of which the proteins, fats, body water, and other biochemicals are chiefly composed. As the neutron energy was degraded, the neutrons were distributed throughout the body in a way which, according to the calculations of Snyder[21] was largely independent of the initial incident energy. Most of the neutrons were captured by hydrogen, but some were captured by the normal Na^{23} to form the radioactive isotope Na^{24}.

That part of the body sodium which is in the blood becomes mixed as it is pumped about by the heart so that eventually the concentration of radioactive sodium assumes a constant value and thus serves as an indicator of the original total flux of fast neutrons averaged over the whole body. This indicator is substantially independent of body orientation and of the time spent in the fast neutron field around the reacting assembly.

The amount of radioactive sodium in the blood was, then, the datum which related the experiment discussed below with the exposures to gamma rays as well as fast neutrons during the accident.

MOCK-UP EXPERIMENT

The mock-up experiment described in Appendix I was designed to determine the relationship between the blood sodium activity and radiation dose. This experiment consisted of first determining the gamma dose (D_γ) to neutron dose (D_n) ratio: D_γ/D_n, then determining the relationship between blood sodium activity and fast neutron dose. To accomplish the latter, a large animal (burro) was exposed to the mock-up reactor which was assumed to give the same neutron spectrum as the accidental excursion. The amount of blood sodium per ml of blood is remarkably constant for a number of animals (see Table VI).

Table VI

SODIUM IN BLOOD[22]

	Na in mg/ml
Dog (plasma)	3.45
Horse (serum)	3.43
Monkey (plasma)	3.33
Man (serum)	3.17

The burro was chosen as the experimental animal because its thickness and size approximates man far better than any of the other animals mentioned in Table VI. No correction was attempted for the geometrical distribution of the neutrons as they are moderated since it was felt that the magnitude of the correction was insignificant compared to other errors in the analysis. A mock-up experiment was conducted on June 18, 1958 (see Appendix I), at a critical assembly which closely approximated the excursion.

The ratio of blood weight to total body weight is nearly the same in man as in the burro. Measurements have been made of the blood volume of the burro using radiophosphorus and radioiodine tracer techniques. The body of the burro is 6.7% blood. This is to be compared to the same figure for the standard man of 7.7%. It is felt that no correction is required since, as it was explained above, the blood is essentially a well-averaged tissue sample. It is believed that, for these and other reasons, the burro is a reasonable substitute for a man and may be expected to give results well within the requirements of this particular incident without corrections. Measured blood sodium values for those exposed and for the burro used in the June 18, 1958, experiment are shown in Table VII.

TABLE VII
BLOOD SODIUM FOR INDIVIDUALS EXPOSED AND FOR EXPERIMENTAL BURRO

Employee	mg/ml Serum
"A"	3.2
"B"	3.2
"C"	3.2
"D"	3.1
"E"	3.2
Experimental Burro	3.1

At about 5:00 p.m., June 16, 100 ml of blood was drawn from the eight men at the medical dispensary. This blood was placed in beakers and the sodium-24 activity was counted in the manner indicated above. On the morning of June 17, a second 100 ml was taken. An anticoagulant (heparin) was added to preserve the blood in liquid form. The specific activity of the two samples is in substantial agreement, but the second set seems to be the better. The dose values are based on the second set.

The burro blood was withdrawn and placed in the same size polyethylene bottle, the anticoagulant added, and the sample counted in exactly the same way as the human blood. Only the ratio of the two counts is needed.

In the mock-up experiment, the gamma-ray dose was measured using carbon CO_2 ionization chambers which are standard for shielding work. The neutron dose was measured using the absolute proportional counter which is also standard for dose measurements. Using these techniques, it was found that the gamma-neutron ratio was 3.3. However, the gamma dose should be corrected by a multiplication factor which is dependent on the manner in which the delayed gamma contribution is treated (see calculational method below).[*] The results are given in Table VIII.

A second part of the experiment, at an approximate power level of 300 watts, was run to activate the sodium in the blood of the experimental burro. It was exposed at the same position at which the gamma and neutron doses were determined in the first experiment. A neutron dose of 48 rads was given. Blood samples were drawn and counted using exactly the same procedures as were used in the case of the exposed personnel. Thus, the

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[*] This factor is due to the fact that the mock-up reactor was run at a constant power level, whereas the personnel were exposed to a burst or series of bursts.

neutron dose to the personnel could be determined by means of the relation,

$$D_{n_H} = \frac{Na_H}{Na_B} \times 48 \text{ rads}$$

and the gamma doses by the product of this neutron dose and the factors given in Table VIII.

Table VIII
CORRECTED GAMMA NEUTRON DOSE RATIO FOR EXPOSED PERSONNEL

Employee	Corrected Gamma-Neutron Dose Ratio D_{γ_H}/D_{n_H}
"A"	2.62
"E"	3.03
All Others	2.80

By this means the neutron dose, gamma-ray dose, and total dose in rads for the exposed individuals were obtained. (Table X.)

Some information on the spectrum of neutron was obtained through the use of threshold detectors. The relative flux density of neutrons for various energy regions are shown in Table IX.

Table IX
RELATIVE FLUX DENSITY OF NEUTRONS FOR VARIOUS ENERGY REGIONS

Energy Range	Neutrons/cm2
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	0.50×10^{11}

This information shows that the neutrons are predominately in the fast region, thus adding validity to the concept of Na activation as a measure of fast neutron exposure. The last column in Table X gives the RBE dose in rem assuming a value of 2 for the RBE of fast neutrons.

Table X

SODIUM-24 ACTIVATION AND DOSE VALUES FOR EXPOSED PERSONNEL

Name	Na^{24} (microcuries/cc)	First Collision** Neutron Dose (Rad)	First Collision Gamma Dose (Rad)	First Collision Total Dose (Rad)	Estimated RBE Dose (rem)*
"A"	5.8×10^{-4}	96	269	365	461
"B"	4.3×10^{-4}	71	199	270	341
"C"	5.4×10^{-4}	89	250	339	428
"D"	5.2×10^{-4}	86	241	327	413
"E"	3.7×10^{-4}	62	174	236	298
"F"	1.1×10^{-4}	18	50.5	68.5	86.5
"G"	1.1×10^{-4}	18	50.5	68.5	86.5
"H"	0.36×10^{-4}	6.0	16.8	22.8	28.8
Exptl. Burro	2.9×10^{-4}	48 rad			

*With an assumed RBE = 2 for fast neutron dose.
**Gold foil measurements indicated that the thermal neutron dose was about 1% of the fast neutron dose and thus can be neglected.

CALCULATIONAL METHOD

Independent calculations of the total doses received by the individuals were made. These doses were also based on the sodium activations observed in those exposed. The dose and flux penetration factors for the assembly were made by assuming a spherical reactor of radius r_0 with a neutron-gamma source distribution proportional to

$$\frac{\sin \pi r/r_0}{r}$$

the fundamental mode of critical operation of a spherical reactor. The penetration of the radiation resulting from fission was calculated from point source attenuation data in water obtained by the moments method.[10] The neutron dose times $(4\pi R^2)$ at a distance $R \gg r_0$ from the reactor was found to be 1.08×10^{-9} rad cm^2/fission by this procedure. It has been assumed that $R \gg r_0$ in all calculations described below.

The total flux of neutrons escaping from the assembly was calculated, yielding a flux leakage factor of 0.35 neutrons escaping per neutron born in the assembly. The ratio of neutron dose to the integrated neutron flux (in units of neutrons/fission cm^2) in the escaping beam was calculated to be 1.22×10^{-9} rad cm^2/neutron. The assumption was made that the spectrum of neutrons did not depend upon the distance from the reactor.

The sodium activation of the blood of an exposed individual was related to the incident neutron flux by assuming that neutrons striking the body are captured therein with a probability which does not depend on their incident energy. The capture probability (f_c) was estimated roughly from some work of Snyder and Neufeld to be 0.4.[21] Then one may write for the fast flux ϕ_f which results in the activation of N atoms of Na^{24} per cm^3

$$\phi_f = \frac{t\,(\Sigma_a)}{f_c\,\Sigma_{Na}} \times N = 3.51 \times 10^4 \times N$$

where t is the average thickness of the body ($\sim$30 cm), Σ_a is the total macroscopic absorption cross section of the human body for thermal neutron capture, and Σ_{Na} is the activation cross section of Na^{23}.

For an observed activation of A μc of Na^{24} per cm^3 in the blood (assumed to be the same as in the whole body), one may infer that the fast neutron dose D received by that individual was approximately

$$D\ (rad) = 1.24 \times 10^5 \times A$$

Table XI gives neutron dose values for the exposed personnel calculated from this equation.

Table XI

DOSE VALUES FOR EXPOSED PERSONNEL BASED ON CALCULATIONS

Employee	Na^{24} (microcuries/cc)	First Collision Neutron Dose (Rad)	First Collision Gamma Dose (Rad)	First Collision Total Dose (Rad)	Estimated RBE Dose (rem)*
"A"	5.8×10^{-4}	72	211	283	355
"B"	4.3×10^{-4}	53	156	209	262
"C"	5.4×10^{-4}	68	199	267	355
"D"	5.2×10^{-4}	64	189	253	317
"E"	3.7×10^{-4}	46	135	181	227
"F"	1.1×10^{-4}	14	40	54	68
"G"	1.1×10^{-4}	14	40	54	68
"H"	0.36×10^{-4}	5	13	18	23

*With an assumed RBE = 2 for fast neutron dose.

The gamma dose incurred at the time of the incident and within the few seconds following was due, principally, to three sources: (1) prompt gamma radiation resulting directly from the fission process, (2) capture gamma radiation resulting from neutron capture within the assembly, and (3) delayed gamma radiation from the fission products within the assembly. Both the prompt gammas and the capture gammas are emitted within a very short time interval following the fission events and are distributed in the same geometrical pattern as are the fission neutrons. This permits a calculation of the gamma-neutron dose ratio from these two sources. The contribution to the dose delivered by the delayed gamma radiation must be treated in a different manner for two reasons: (1) convection currents and the turbulence caused by bubble formation tend to distribute the fission products evenly throughout the solution, and (2) delay gamma source strengths are strongly time dependent in the few seconds following the fission events so that any subsequent motion of the exposed persons affects the gamma-neutron ratio.

The manner in which the delay gamma contribution is obtained is, of necessity, determined by the transient behavior of the assembly. For the purpose of this analysis, it was assumed that the employees were exposed to a single excursion, but that their individual actions in the few seconds following the excursion resulted in an altered gamma-neutron dose ratio. Such an assumption results in a delay gamma dose estimation which is slightly larger than that obtained by assuming a constant power assembly. Three separate cases were treated.

1. Case I (Employee "A") - This employee was exposed to the full complement of prompt and capture gamma radiation, but, as he apparently left the area first, it was assumed that only the first five seconds of the delay gamma spectrum contributed to his dose.

2. Case II (Employee "E") - This employee was also exposed to the prompt and capture gamma source, but the gamma-neutron ratio was adjusted to take into account an assumed 15-second exposure to the delay gamma spectrum and, further, that his exit took him to within 10 feet of the assembly. This case gives the largest gamma-neutron ratio used in the analysis and reflects the fact that this employee was apparently the last to leave the area, and, consequently, saw more of the delay gamma spectrum. This procedure results in an 8% increase in the total gamma dose over the constant distance - 15-second exposure case, and is thought to represent the maximum increase that could have occurred for any of the personnel.

3. Case III (Employees Other Than "A" and "E") - These employees were assumed to have received the full contribution of prompt and capture gamma radiation, but like Case I, the delay gamma were received at a constant distance and, like Case II, the first 15 seconds of the delay spectrum was effective.

The differences in exposure, due to the presence of delayed gamma rays, were found to be small (see Table VIII), and the results for Case III were used in calculating the gamma doses for the whole group. This study shows that the actions of the exposed people, immediately after the excursion was initiated, probably did not affect materially the delayed gamma dose received.

The spectral distribution for the prompt and delay gammas and the time depending for the delays were obtained from recent data of Peelle, Zobel, Love, and Maienschein,[20] while the spectral distribution for the capture gammas was obtained from Glasstone.[9] The prompt and capture gamma sources were assumed to be distributed in the same manner as the fission neutron source, i.e., spherical with a $\frac{1}{r}$ Sin π $r/_{ro}$ distribution. The leakage of delayed gammas from the assembly was calculated by assuming a homogeneous source distribution and by applying standard build-up factors to the exponential absorption terms determined for each of the sources.

A critical concentration of 50 grams of uranium per liter was assumed to have existed at the time of the incident. The corresponding atom density was then used to determine an effective absorption coefficient for the solution. When this atom density is combined with the calculated value for the non-leakage neutron flux, the capture gamma source term may be determined. The total gamma dose is not greatly altered by assuming other concentrations since the effect on the absorption coefficient is small and, although the capture source term varies linearly with concentration, the capture gamma dose represents only some 20% of the total gamma dose from all sources.

The calculated doses agree reasonably well with the values derived from the experiment. It is felt that the largest uncertainty in the calculations described lies in the determination of the probability of capture of incident neutrons in the body.

In order to make a valid comparison between the gamma dose received by the burro and that received by those involved in the accident, one must take into account that the relative contribution of fission product gammas was different in the two cases. The gamma dose - neutron dose ratio in the radiation field to which the burro was exposed was determined by separate measurement of the gamma and neutron dose rates during steady state operation of an experimental water-moderated assembly. The ratio was determined at approximately 3 minutes after the beginning of operation. A calculation of the gamma doses per fission in the two cases yielded correction factors to be applied to the gamma dose - neutron dose ratio for burro exposure to obtain the gamma dose - neutron dose ratio for the accident. The corrected gamma-neutron ratios are given in Table VIII. Since the differences were small, the value 2.8 was assumed for the whole group.

COMPARISON OF MOCK-UP EXPERIMENT AND CALCULATIONAL METHOD

It should be emphasized that both the experimental and calculational approaches to establishing the dose are based on experimental values of Na^{24} activity in the blood of the persons involved in the accident. They differ in the manner in which the Na^{24} activity is translated into neutron and gamma dose.

The assumption is made throughout that the neutrons spectrum in the neighborhood of the critical assembly does not depend upon distance from the assembly. Wall and floor scattering of neutrons may invalidate this assumption to some extent, but unpublished data by Hurst and Mills, obtained at the Lady Godiva assembly at Los Alamos, tends to substantiate it for the distances which are of interest here.

OTHER METHODS

Two other less quantitative methods of estimating the dose are reported in the Appendices J, "Estimates of Dose Based on In Vivo Body Counter," and K, "Estimates of Dose Based on Indium Foils Measurements."

COMPARISON WITH 1946 LOS ALAMOS EXPOSURES

On May 21, 1946, a criticality accident occurred in one of the facilities of the Los Alamos Scientific Laboratories. Eight persons were exposed to the ionizing radiation in amounts comparable to those received during the Y-12 incident. The dosimetry problem was similar in that film badge dosimeters were not worn in either case. The area in Y-12 in which this accident occurred was one in which only insignificant amounts of radiation are normally present. Accordingly, film badge dosimeters were not worn by the employees.

Thus, as in the Y-12 case, the Los Alamos investigating committee used the blood sodium activity in their estimation of the neutron dose received by each of the eight persons exposed. In essence, the technique consisted of using Na^{24} activity to measure the integrated thermal neutron flux and from this to compute the associated fast flux. This fast flux is then converted into dose units by assuming an average energy for the leakage spectrum. To account for the effect of the higher energy neutrons, a factor of 3 was arbitrarily applied to the estimated doses. This final figure is considered to be the "most probable" dose.

While estimates of the neutron component of the Y-12 doses may be obtained by comparison of the Na^{24} activity of the Y-12 employees to that of the Los Alamos personnel, the

estimates so obtained will contain all of the assumptions and inaccuracies inherent in the calculations made 12 years ago.

The technique of comparison consists in averaging the neutron dose per unit blood serum activity for the Los Alamos cases and multiplying this into the blood serum activities found for the Y-12 cases.

The ratio of neutron dose per unit volume activity of Na-24 for the Los Alamos cases was 0.84 rad per disintegration per second per cc, while for the Y-12 cases the ratio was 2.0. There are several possible reasons for this difference; e.g., (1) the capture cross section used by Los Alamos was 23% higher than the value used for the Y-12 cases; (2) an estimated energy was used for the leakage neutrons for the Los Alamos cases rather than a calculated value.

A gross underestimate of the total dose would be obtained by applying the above technique to the gamma plus neutron dose instead of just the neutron dose since the gamma to neutron dose ratio was considerably different for the two accidents. The difference is due to the fact that the Y-12 incident involved a moderated homogeneous system whereas the Los Alamos incident involved a beryllium reflected metal system.

ACKNOWLEDGEMENTS

APPENDIX A

Y-12 EMERGENCY PLAN AS APPLIED TO NUCLEAR ACCIDENTS

The following are the Y-12 emergency procedures as applied to a nuclear accident. With the exception of Part II, which is the 9212 building radiation alarm procedure, this information is contained in the Y-12 Emergency Manual*.

      I Plan for Coping with a Critical Reaction

     II 9212 Building Radiation Alarm Procedure

    III Emergency Philosophy

    IV General Emergency Plan

     V Responsibilities of Line Organization

    VI Responsibilities of the Plant Emergency Director

   VII Responsibilities of Y-12 Emergency Service Units

PLAN FOR COPING WITH A CRITICAL REACTION

Operational Philosophy

If a critical reaction occurs, the following things should be done in the order named:

1. Evacuate personnel from the area involved, and make any necessary operational changes.

2. Rope off or mark unsafe areas to avoid unintentional re-entrance and exposure of personnel.

3. Study the situation and plan for further action, including determination of the extent of exposure received by personnel.

Radiation Exposure

Exposure during radiation emergency work should be held to a minimum and no one engaged in such work should be permitted to receive a dose in excess of 25 r without explicit instructions from the Plant Emergency Director.

Procedure

All plans to cope with a critical reaction should include:

1. Means of recognizing a critical reaction.

    a. A critical reaction might be recognized by one or more of the following phenomena followed by positive verification with a high range gamma meter:

       (1) A blue glow or haze.

       (2) Fuming and steaming of a vessel containing quantities of fissionable material.

       (3) Audible signal initiated by recording gamma meters.

       (4) Observance of unusually high gamma readings over a wide area, where not routinely expected.

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* A pocket notebook which is issued to every supervisor in the plant.

2. Procedures concerning personnel.

   a. All personnel should be promptly evacuated to a point at least 500 feet from the suspected critical accumulation or until the radiation level is less than 12.5 mr/hr.

   b. Personnel selected from the emergency phone listing should be notified immediately so they can assemble at the emergency control center.

   c. The Health Physics Departments of ORNL and ORGDP should be contacted to lend assistance.

   d. The security badges of personnel in or near the vicinity of the reaction should be monitored with an Eltronics or equivalent gamma survey meter and readings recorded along with names and badge numbers. Dosage should be obtained by use of the calibration table supplied by the Plant Emergency Director*.

   e. All individuals whose badges show evidence of radiation exposure, and those in the vicinity of the accident, should be taken to the dispensary for medical attention.

   f. The Medical Departments of ORNL and ORGDP should be notified to stand by if needed. The Oak Ridge and ORINS Hospitals should be notified to be prepared to receive casualties.

   g. Personnel engaged in emergency work in areas of 125 mr/hr. or greater should wear a high range dosimeter or be accompanied by a man using a survey meter, and be relieved of emergency work after receiving a dose of 25 r.

3. Procedures concerning the radioactive area.

   a. Trained personnel should monitor toward the area of the reaction and establish boundary lines at intensities of 12.5 mr/hr. and 125 mr/hr. Guards should be posted so as to permit only designated personnel to pass within the 12.5 mr boundary.

   b. Rehabilitation and decontamination operations should not be commenced until direct authorization is given by the Plant Emergency Director, who will give such directions only after a thorough study of the situation by qualified plant personnel.

4. Desirable information to be obtained by the Plant Emergency Director.

   a. A Plot plan of the emergency area, showing locations and movements of personnel involved and dosages as determined from security badges, should be prepared for use of medical personnel, the Plant Emergency Director, and for future reference.

9212 BUILDING RADIATION ALARM PROCEDURE

General Information

The purpose of this outline is to describe the procedures necessary for the protection of personnel in the event of a monitor alarm in Building 9212.

1. Monitor Alarm - A monitor alarm is defined as an emergency condition, possibly resulting from an excess of radiation within an operating area. Such an emergency is indicated by the activation of the sirens and bells of the monitoring system.

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* All personnel badges contain an indium foil for the detection of neutrons.

It is emphasized that a condition giving rise to a monitor alarm can be serious. The only "protection" provided by the alarm system is a warning that an emergency exists so that evacuation can be initiated immediately. In case of a monitor alarm, quick orderly evacuation by the most direct route away from the processing area is the best path to follow. Evacuation route charts showing the various evacuation routes are posted throughout the building. A copy of this chart is included as part of the procedure.

2. Local Emergency Director - The C-Wing foreman will be the Local Emergency Director. All other shift foremen will report to and aid him during an emergency. The Plant Shift Superintendent will be the Plant Emergency Director and, during the time of emergency, the Local Emergency Director will report to him.

3. Monitor System - The monitor system is composed of six permanently mounted radiation detection instruments, a siren system audible throughout the building, and an annunciator system which indicates the location of an activated monitor. The locations of the monitors are shown on the drawing. The annunciator lights are located in the headhouse corridor, and the annunciator panel in the west end of the office wing corridor. Monitors are set to activate the alarm when the radiation level exceeds 3 mr/hr.

All monitors are checked daily and activated weekly using a radium source. Any monitor failing to act normally is reported to the Electrical Department at once, and the instrument is immediately replaced. Each monitor is checked monthly by the Electrical Maintenance Department and removed for repair as they see necessary. Whenever a monitor is removed, it is immediately replaced by a spare so that the area it serves does not go unprotected.

All monitor checking is to be done under the supervision of the special processing foreman.

During the period of the test, persons with portable detection instruments should be constantly surveying the area. Should these persons note a reading above 100 mr/hr., they should have the sirens unmuffled at once.

During this test a person should be stationed in the headhouse corridor to see that the annunciator lights come on in the correct order, and another person should be stationed in the office wing corridor to similarly check the annunciator panel.

4. Portable Detection Equipment - The radiation detection instruments used to survey are located in the following places:

a. Men's changehouse - two Radectors

b. North end Analytical Laboratory-two Radectors

c. Northwest assembly point - two Radectors

These survey meters are to be serviced in the same manner as the monitors. The C-Wing foreman will be responsible to see that the routine servicing of the portable survey meters is accomplished.

5. Push Button Alarm System - The north or the south processing areas siren systems may be activated at any time by the use of the push button switches in the main corridor near the entrance of the office building; thereby, it is possible to evacuate the respective areas for any reason.

Evacuation Areas

The building is divided into two areas; the north has sirens that are activated by the monitors, and

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the south has auto-call bells that are also activated by the monitors.

1. Siren Area -. The sirens are used to notify personnel in these areas that they are to evacuate <u>at once.</u>

2. Bell Area - The bells are used to notify personnel in areas not to be evacuated that there is an activated monitor in the building. There is a silencing button which will silence the bell system. This button is located below the annunciator panel at the west end of the office wing hall. These bells may be silenced just prior to the "All Clear" signal by the Local Emergency Director or someone authorized by him.

Under no circumstances are persons from the bell area to go into the siren area while the bells are sounding.

Responsibility of Personnel

1. The Local Emergency Director:

   a. To notify the Plant Shift Superintendent that there is a monitor alarm and to station an employee to direct the Shift Superintendent to the affected area.

   b. To survey or have surveyed, the area covered by the activated monitor and notify the Plant Emergency Director of the results of the survey.

   c. To have defective monitors replaced immediately.

   d. To maintain an efficient shift emergency squad whose duty during a radiation alarm will be to guard the entrance to the building and see that no person enters the building until the "All Clear" is sounded. The emergency squad members should place themselves at least 100 feet from the building while guarding the entrance. This function is to be arranged in advance.

   The C-Wing foreman, in cooperation with the other foremen on the shift, will see that these assignments are made.

   e. While surveying the building any reading in excess of 100 mr/hr. will be sufficient to suspect that a radiation accident has occurred. In this case the Local Emergency Director should immediately leave the building with his survey crew and proceed as follows:

      (1) Evacuate the south processing areas by pushing the manual alarm button.

      (2) Survey the assembly area, moving personnel as necessary to keep them in a radiation zone of less than 5 mr/hr.

      (3) Survey the personnel in the assembly areas as rapidly as possible. Check both the clothing and the person for gross contamination using the available portable survey instruments. (Gross contamination will give readings above 1 r/hr.) Contaminated persons should have their contaminated clothing removed on the spot and be sent to the shower without delay. Minutes count in these cases.

2. Plant Emergency Director - Upon obtaining the results of the survey will determine that the building is safe for re-entry and give his permission to sound the "All Clear".

   . If, in his opinion, a radiation accident has occurred, he will apply the plant emergency plan

as outlined in the Y-12 Emergency Manual.

EMERGENCY PHILOSOPHY

The following underlined statements and supplementary discussion highlight some of the fundamental concepts upon which the Y-12 emergency plan is based:

1. Line organizations have major emergency responsibilities. The responsibility for emergency planning and execution must rest with the line organizations which are responsible for anticipating potential emergencies with their areas, arranging to avoid the occurrence of such emergencies, and making adequate preparation for handling these emergencies if they do occur.

2. Prompt local action is needed. Because of the variety of potential hazards and conditions encountered, the most effective emergency control is supplied through action of the local emergency group. This group, consisting of a local emergency director and his squad, must be prepared to effectively cope with any eventuality and to obtain and direct the efforts of the various plant emergency groups. Therefore, the Plant Shift Superintendents must see that well-trained local emergency directors and squads are maintained in all hazardous areas.

3. Plant-wide emergency direction is necessary. Provision for plant-wide direction of emergency efforts is necessary, in cases of serious emergency, to insure that all emergency groups involved function as a team. Such direction is supplied by the Plant Emergency Director, who is the Plant Shift Superintendent on duty. It is recognized that in combating emergencies he must rely heavily upon the performance of trained local and plant-wide emergency groups.

4. Shift organizations must handle emergencies. In order to insure clear-cut responsibility for the direction of activities involved in actual handling of emergencies, responsibility must be fixed with the shift organization. Therefore, where day supervisors participate in emergency work, they are expected to function as staff to the Local or Plant Emergency Director, keeping well in mind the tie of responsibility which the Local Emergency Director has to the Plant Emergency Director. When day supervision chooses to take over local emergency responsibilities, the change must be made only with the full knowledge of the Plant Emergency Director.

GENERAL EMERGENCY PLAN

1. The Local Emergency Director goes to the scene of the emergency and does the following:

   a. Estimates the magnitude of the problem.

   b. Ascertains that emergency groups have been summoned.

   c. Promptly notifies the Plant Emergency Director.

   d. Arranges for necessary further evacuation of personnel.

   e. Directs emergency groups as they arrive.

2. The person who meets the emergency group directs them to the scene of the emergency and the Local Emergency Director.

3. The emergency group leaders follow the instruction of the Plant Emergency Director in

bringing the emergency under control.

4. The Plant Emergency Director estimates the extent of the emergency, assumes general direction of emergency activities to the degree required, arranges for establishing necessary road blocks, and as necessary, arranges for additional service groups, manpower, and/or equipment, including possible invocation of any of the existing mutual aid agreements.

5. Building personnel will be notified of an existing emergency by means of sirens which are activated by radiation monitors.

6. The Plant Emergency Director, after consultation with the Local Emergency Director, declares the termination of the emergency and arranges for the "All Clear" signal.

RESPONSIBILITIES OF LINE ORGANIZATION

1. Anticipating potential emergencies within their facility and arranging to avoid their occurrence.

2. Making adequate preparation for handling emergencies.

3. The above basic responsibilities include, but are not limited to, the following detailed responsibilities:

   a. Joint responsibility, along with the Plant Shift Superintendent, for the appointment and training of Local Emergency Directors and necessary local emergency squads for all hazardous areas.

   b. Arrangement for the following in the event of a serious plant emergency:

      (1) Dispatching of necessary emergency units upon request of the Plant Emergency Director.

      (2) Holding in readiness at predetermined points such emergency units.

      (3) Reporting to the emergency area command post.

      (4) Serving in a line position subordinate to the Plant Emergency Director and assisting in the coordination and direction of emergency units.

RESPONSIBILITIES OF PLANT EMERGENCY DIRECTOR

1. Joint responsibility, along with Divisional Superintendents, for plant-wide emergency planning, training, and evaluation consistent with plant-wide policies and plans.

2. Providing overall direction of emergency efforts as required, including the following:

   a. Providing direction for plant emergency groups, the plant and local emergency squads.

   b. Giving adequate attention to such matters as necessary plant operational changes.

   c. Determining the necessity for and scope of any large-scale evacuation or dispersal, and where necessary, arranging for the announcing and direction of such evacuation or dispersal.

   d. Arranging for procurement of additional manpower and/or emergency equipment as required, including:

(1) Notification of the proper emergency personnel.

(2) Invoking mutual aid agreements.

Local assistance may be requested directly between Oak Ridge installations, and the assisting group will normally operate under the direction of the organization being assisted.

3. Determining the termination of a state of emergency, and arranging for announcing the "All Clear" signal.

RESPONSIBILITIES OF Y-12 EMERGENCY SERVICE UNITS

All Emergency Service Units

1. Be prepared to accomplish the assigned responsibilities of the unit.

2. Promptly report to predetermined or other specified assembly points when called.

Industrial Relations Units

Fire Prevention and Control Department

1. Dispatch ambulance service in response to ambulance calls.

2. Provide and maintain adequate personnel protective equipment as required.

Guard Department

1. Provide radio communication between the local emergency area command post and the plant command post (Plant Shift Superintendent's Office), and other locations as directed.

2. Direct traffic over plant roads.

3. Arrange entrance for authorized personnel who appear without badges.

4. Operate a courier service as required.

5. Provide necessary mobile radio communications between the Plant Shift Superintendent's Office and assembly points as required.

Safety Department

1. Determine that adequate steps are being taken to protect life and property at scene of emergency and keep Plant Emergency Director advised.

2. Procure additional personnel protective equipment as required.

Medical

1. Provide for first aid and handling of the injured.

2. Mobilize area-wide medical first aid personnel as required.

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Personnel Department

1. Provide emergency food and shelter.

2. Handle public relations aspects of the emergency.

Maintenance Division Units

Emergency Electrical Maintenance Personnel

1. Report to the scene of the emergency for instructions.

2. Determine the need for and obtain necessary outside electrical maintenance assistance.

Transportation

Provide necessary motor vehicle operations.

Shift Superintendent and Utilities Division Unit

Utilities

Arrange necessary curtailment of utility services.

Technical Division Units

Health Physics Department

1. Mobilize additional radiation monitoring teams necessary for the protection of personnel engaged in rescue work involving penetrating radiation.

2. Audit radiation hazard activities, advising the operating groups and the Plant Emergency Director as required.

3. Provide additional radiation survey instruments required.

4. Collaborate with medical director in the treatment and calculation of dosage of radiation exposure cases.

5. Assist the Plant Emergency Director and operating groups in establishing decontamination procedures and decontamination centers for personnel and equipment.

6. Provide laboratory sampling and analysis services required for determining air contamination and degree of radiological hazard in the field.

Radiation Control Group

The Radiation Control Group will serve in an advisory capacity in emergencies where a critical reaction has occurred or is likely to occur.

Finance and Materials Division Unit

Direct procurement and issuance of stores material as required.

APPENDIX B

METHOD OF CALCULATION OF REACTIVITY*

The critical mass in an unreflected 21.75 inch diameter stainless steel cylinder was determined as a function of the critical volume by equating its geometric buckling to that of a similar cylinder of 20-inch diameter for which the critical parameters are known. Once the variation of critical mass with critical volume of the larger cylinder is known, an initial critical point on the curve, commensurate with facts observed after the excursion, is chosen. This point, A, in Figure B.1, represents 2.1 kg U^{235} in 56.2 liters of solution. The subsequent masses and volumes, as additional solution enters the drum, are represented by line AB, assuming that the concentration of the incoming solution remains constant. It is further assumed that the final contents are 2.5 kg U^{235} in 180 liters.

In a two neutron-energy group analysis, the effective reactor multiplication factor, k, of critical and near critical assemblies is related to the material and geometric properties of the assembly by

$$k = \frac{\eta f}{(1 + L^2 B^2)(1 + \tau B^2)} \qquad (1)$$

where

η = number of fission neutrons produced per neutron absorbed by U^{235}

f = thermal neutron utilization

L^2 = square of the thermal diffusion length

B^2 = geometric buckling of the reactor

τ = neutron age

Along the critical curve in Figure B.1, the equation has the value unity, of course.

As the cylinder continues to fill, the mass and volume increase to point E which describes a different (supercritical) combination of geometry and material. The nuclear properties of the latter are the same as those of the solution critical at point D, since a line through the origin represents a particular chemical concentration, and the values of ηf at D and E are, therefore, equal. Since the geometric buckling at conditions C and E are the same and L^2 and τ are essentially constant over this concentration range, the multiplication constant at E is given by

$$k_E = \frac{(\eta f)_D}{(\eta f)_C}$$
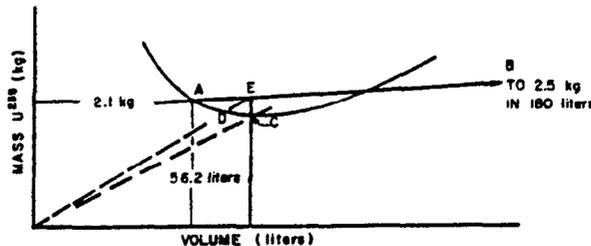
Figure B.1
SCHEMATIC DIAGRAM OF CRITICAL MASS IN AN
UNREFLECTED 21.75" DIAMETER STAINLESS STEEL
CYLINDER AS A FUNCTION OF CRITICAL VOLUME

APPENDIX C

COMMENTS ON RADIATION RECORDER CHARTS*

MONITRON

The Monitron is an ionization-chamber type instrument, utilizing a boron-lined ionization chamber for the detection of gamma and neutron radiations. Ions produced in the chamber gas by gamma rays are collected by a central electrical conductor and the chamber walls, and the resulting current constitutes the input signal for an amplification circuit. The detection of neutrons is accomplished primarily by their nuclear reaction with the boron of the ionization-chamber lining to produce alpha particle emission; the ions formed in the chamber gas by these alpha particles are then collected to constitute the input signal.

Full scale readings are 25 mr/hr. for gamma radiation on the low range scale and 125 mr/hr. on the high range, the selection of scales being manually controlled. The full scale readings for slow neutrons are estimated to have about the same numerical values with the dose rate expressed in mrem/hr. The time constant for the electronic circuit is estimated to be about 0.5 second, and the maximum recorder speed corresponds to approximately 1 second for a full-scale traverse of the recorder. The instrument is reported to be linear to within 2%.

The range of statistical fluctuations in instrument response for constant radiation fields of various levels are illustrated in Figure C.1 for both neutron and gamma radiation, the sources for these radiations being a 1 millicurie radium gamma source and a 6 curie polonium-beryllium neutron source.

In order to determine whether the peaks noted on the chart for the period during the nuclear excursion could be attributed to such statistical fluctuations, the form of this trace was approximately reproduced by use of the neutron source, the distance between the source and the instrument being varied to control the instrument reading. Pulses were simulated by alternately decreasing and increasing the distance between the source and the ionization chamber. Such pulses were simulated for the period representing the highest radiation levels, and a peak, corresponding to a similar peak noted on the original chart, was simulated at 3 minutes; other portions of the trace were made with a gradually increasing source-to-detector distance, and in these sections the fluctuations noted represent statistical variations only. This trace is compared with that for the critical excursion in Figure C.2.

Since some of the radiation detected during the excursion would have been gamma, for which the statistical fluctuations are much less than are those for neutrons, and since the fluctuations noted on the trace made during the excursion are noticeably larger than those obtained with a neutron source, it appears that at least part of the variation in instrument reading represented actual momentary variations in the power level of the reaction.

A careful comparison of the chart of the Monitron with other records of the radiation levels produced by the excursion, which are described below, indicated that the peak at 3 minutes may not have been the result of a radiation burst· from the reaction, and the possibility exists that this peak was produced by a scale change of the instrument. However, no such change could be verified. An abrupt drop to zero and return, noted toward the end of the excursion, may be associated with a check of the instrument zero adjustment.
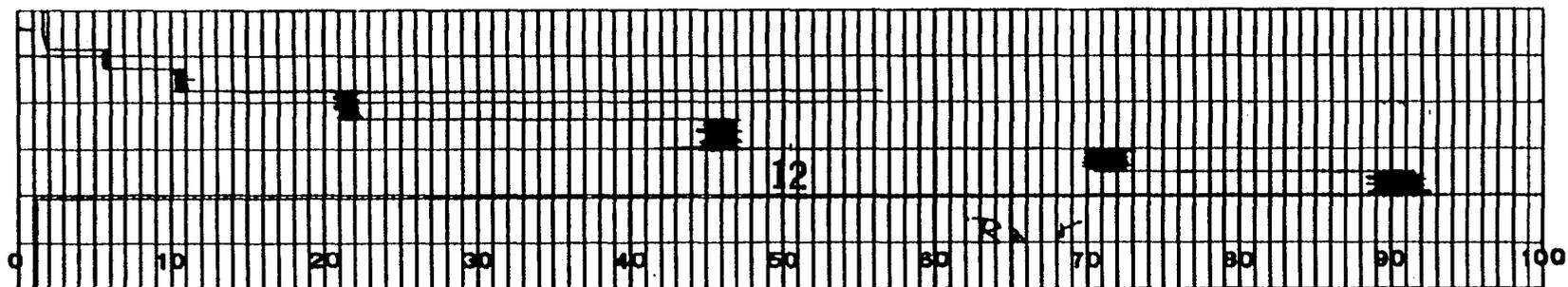
Radium Gamma Radiation
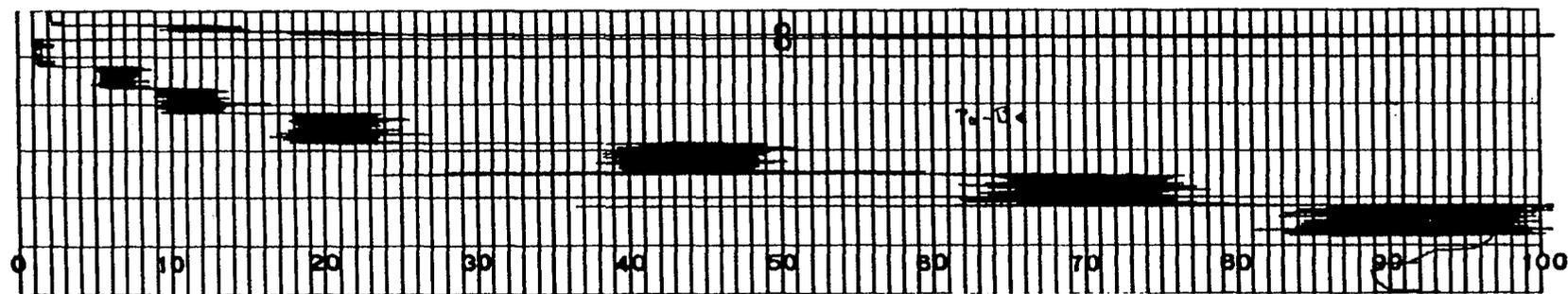
Neutrons

Figure C.1

MONITRON TRACES OF CONSTANT RADIATION LEVELS
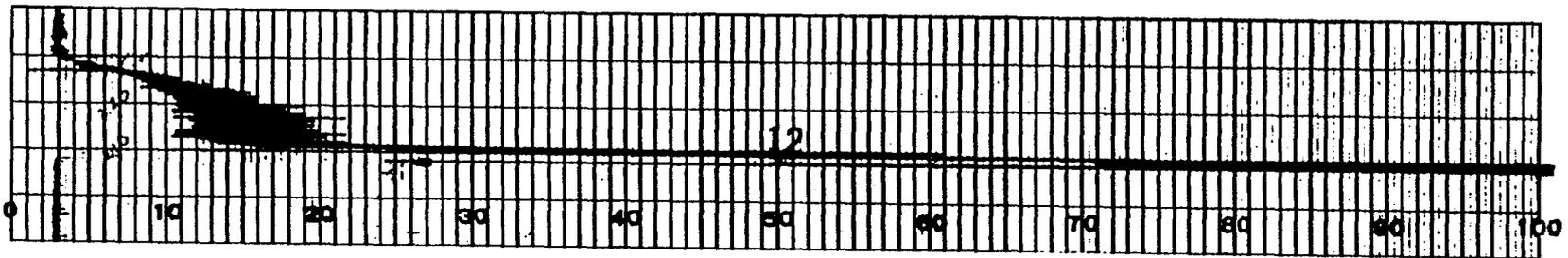
Critical Excursion
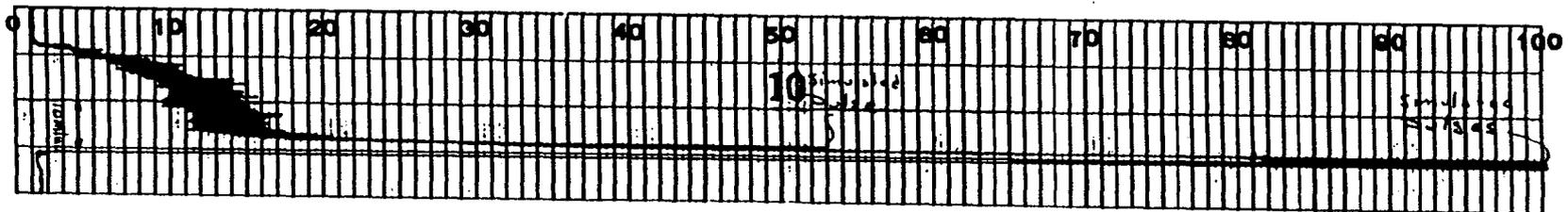
Reconstruction With Po-Be Neutron Source

Figure C.2
MONITRON CHARTS

CONTINUOUS AIR MONITOR CHARTS,
BUILDINGS 9207 AND 9204-1

The continuous air monitors are Geiger-Muller tube instruments counting the beta-gamma emission from particulates collected from an air stream on a fixed filter paper. The tubes are shielded by 1½ inches of lead.

Both of these instruments detected the initial rise in radiation, and the one in 9204-1 automatically changed to a scale with $^1/_5$ the sensitivity of that utilized in normal operations. The trace, shown in Figure 17, then indicates a period of approximately 2 minutes duration with a relatively high radiation field, showing at least one oscillation, followed by a sharp drop with a slower decline in radiation, the total radiation period being of approximately 20 minutes duration. At about 40 minutes after the start of the excursion, the chart indicated a rise in airborne beta-gamma emitting radioactive material; shortly after this rise the instrument was manually returned to the more sensitive scale, and the chart indicates an increase in the amount of radioactivity on the filter paper, this material subsequently decaying to a value slightly above background after a period of about 2 hours.

The chart from the Building 9207, shown in Figure 16, also indicates the initial rise in radiation and the 2 minute high-level period. The low level period, however, is obscured by the detection of airborne activity, a definite increase in such activity being indicated 10 minutes after the start of the excursion. The subsequent drop from off scale to below background with a return to off scale was reported to be associated with operating checks of the instrument during the off-scale period. Decay of the radioactive material by the decreasing trace is subsequently indicated.

LABORATORY-TYPE COUNTER, 9204-3

The trace from end-window Geiger-tube instrument, shielded by 1½ in. of lead, is shown in Figure 18. This is one of about 12 such traces recorded by laboratory counters in the control room for the 63-inch cyclotron in Building 9204-3. Counts are accumulated through scaling circuits and activate the pulse recorder upon each accumulation of a preset number of counts. The individual register pulses were not discernible for any of these instruments except at the end of the excursion. However, the duration of the excursion was indicated by all of these instruments to be approximately 20 minutes.

APPENDIX D

HYDRAULIC EXPERIMENTS*

An effort was made to duplicate the actual filling of a 55-gallon drum using plant equipment involved in the incident and substituting a solution of cadmium nitrate in place of uranyl nitrate. The specific gravity of the solution was adjusted to 1.19, approximately the same as that of the uranium solution found in the line from the B-1 to the C-1 areas, up stream of valve V-1. The cadmium nitrate was added to the C-1 system from B-1 tanks F-318 and F-322, via the process lines. (See Figure 10.)

In a preliminary experiment, it was determined that at least one-half hour was required for the solutions in the C-1 tanks to come to equilibrium levels if the 6-2 tank was filled with water and the solution allowed to distribute between tanks 6-1, 6-2, and 1-2. The point of equilibrium was determined by the stability of the levels in the tanks as indicated by the pneumatic liquid level indicators connected to the tanks. It was found that the level gauges on tanks 6-1 and 6-2 were inaccurate. They failed to read until the tanks were one-third to one-half full. The gauge on tank 1-2 appeared to be reasonably accurate.

In the experiment, which is believed to most closely approximate the actual incident, 11.7 gallons of the cadmium nitrate solution were added from B-1 tanks F-318 and F-322. One hundred minutes were allowed for the levels to equalize in the C-1 system. Water was added to tank 6-2 after closing the discharge valve (V-5). The volume required to give an overflow was 39.3 gallons. Since the "safe" tanks hold 42 gallons, approximately 2.7 gallons of the cadmium solution had entered this tank prior to the closing of valve V-5 and the subsequent water addition.

This total of 52 gallons is about 4.5 gallons more than was recovered in the drum after the nuclear incident. The difference is not well defined. Some of the original water was lost in the detection of the leak in tank 6-2. Leaks are not usually allowed to go to such a volume because of the clean-up work which is required.

Based on the estimate of the volume of cadmium solution in tank 6-2, the assumption that a like amount entered tank 6-1, and the level gauge readings on tank 1-2, it can be assumed that more than half of the cadmium solution remained in tank 1-2 even after the time allowed for the system to come to equilibrium. The level gauge indicated that only a few gallons of this water were transferred to tank 1-2. A time period of about thirty-five minutes was allowed for this latter transfer.

A 55-gallon drum, equipped with a polyethylene liner of the same kind that was found in the drum involved in the incident, was placed in the same location. The position of the drum was easily located because of the activation of the stainless steel floor by the neutron flux from the original. A tygon tube, equivalent to the one involved in the incident, was put on the drain line and supported as nearly as possible in the manner of the original tube. The drum was held on a dolly at the required distance off the floor.

As the solution entered the drum after the opening of the drain valve, samples were taken from the tube and from the surface of the liquid in the drum, and concurrent measurements of solution heights in the drum were recorded. Plots of the data are presented in Figures D.1 and D.2.

D.2

In Figure D.1 the rate of filling of the drum shows a sharp break at a level of 9.5 inches and a time of 15 minutes. This corresponds to the time at which the level indicator on tank 1-2 indicated empty. The break has been duplicated in two experiments which involved different quantities of both cadmium solution and water.

The break in the flow curve is duplicated by the break in the concentration of the solution entering the drum. This further indicates that the uranium solution entered tank 1-2 for the most part and stayed there. The concentration axis in the plots has been given on the basis of the 11.7 gallons of solution added and the measured inventory of uranium of 2,500 g U-235, as if the original solution had been 57 g per liter of U-235.

In Figure D.2 there is added a calculated curve for the concentration of the solution in the drum derived from the concentration of the entering solution under the assumption of complete mixing in the drum. It can be seen that there is an increasing difference in the plots of the calculated and the actual concentrations after about thirty minutes. It appears that the assumption of complete mixing is justified up until that time.

An integration of the curve of the samples entering the drum gives rise to the estimated uranium inventory of the drum as a function of time. Again the axis corresponds to an initial solution of 57 grams per liter of U-235.

The calculated curves of concentration and inventory have been replotted in Figure D.3 as a function of the height of the solution in the drum liner.

In the curves of the concentrations of the samples entering the drum there appears a sharp initial rise, (see Figure D.2). This appears to be a result of the filling of the pipe lines with water, or at least dilute solution, when the water in 6-2 was brought into equilibrium with tank 1-2. This initial rise was not found in any experiment in which equilibrium with 1-2 was not allowed during the filling cycle.

A determination of the minimum opening at which valve V-1 would pass the solution required was made using the cadmium nitrate solution. The required flow of about 140 ml per minute can be obtained in the plant configuration if the valve is 1.2 turns opened or within slightly less than one turn of being closed.

Because of the small driving forces involved in the establishment of equilibrium between the tanks and of the tendency of the system to develop gas locks, the distribution of solutions obtained in the experiment may not duplicate that which occurred prior to the incident. Since there was good evidence of the flow of the solutions to all of the tanks involved in the experiment, it is believed that it constitutes a limiting case and that any other circumstances would have resulted in the uranium being added to the drum sooner because of failure to undergo dilution in the tanks. The rates of flow of the solutions from the tanks in the incident and the experiment should be very similar in any case.
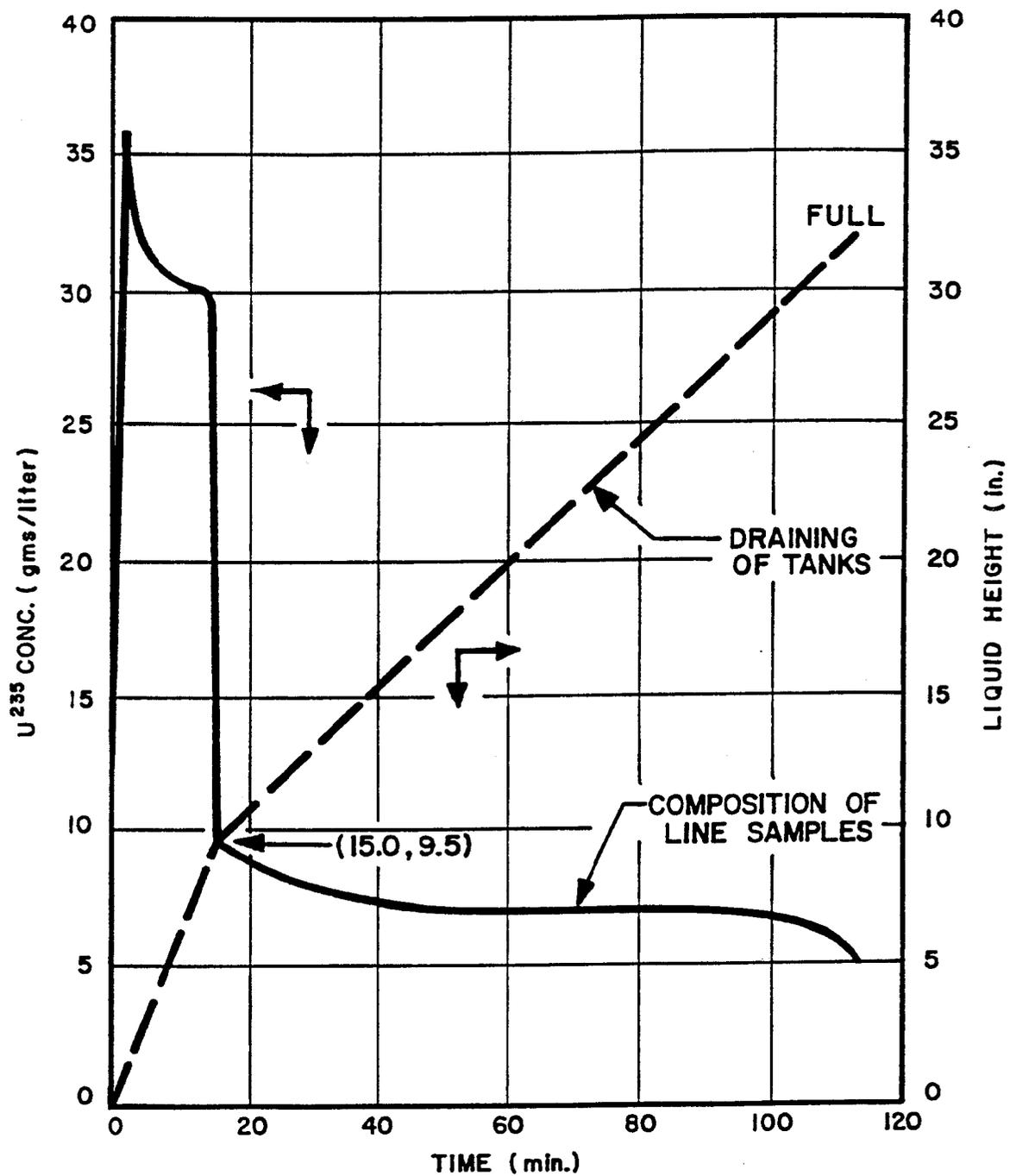
Figure D.1
LIQUID HEIGHT AND CONCENTRATION OF
SOLUTION IN DRUM AS FUNCTION OF TIME
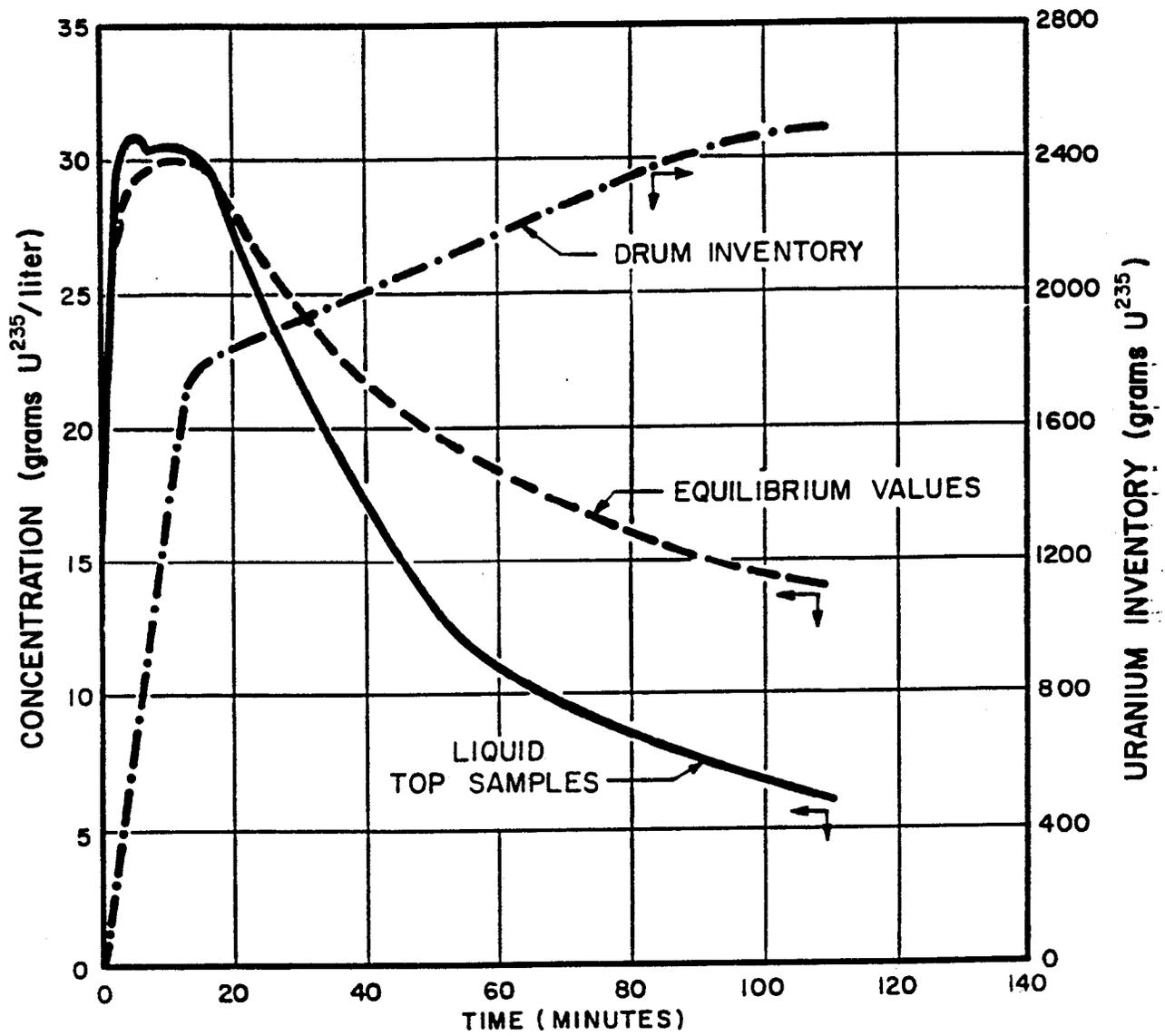
Figure D.2
URANIUM INVENTORY & CONCENTRATION OF SOLUTION
IN DRUM AS FUNCTION OF TIME
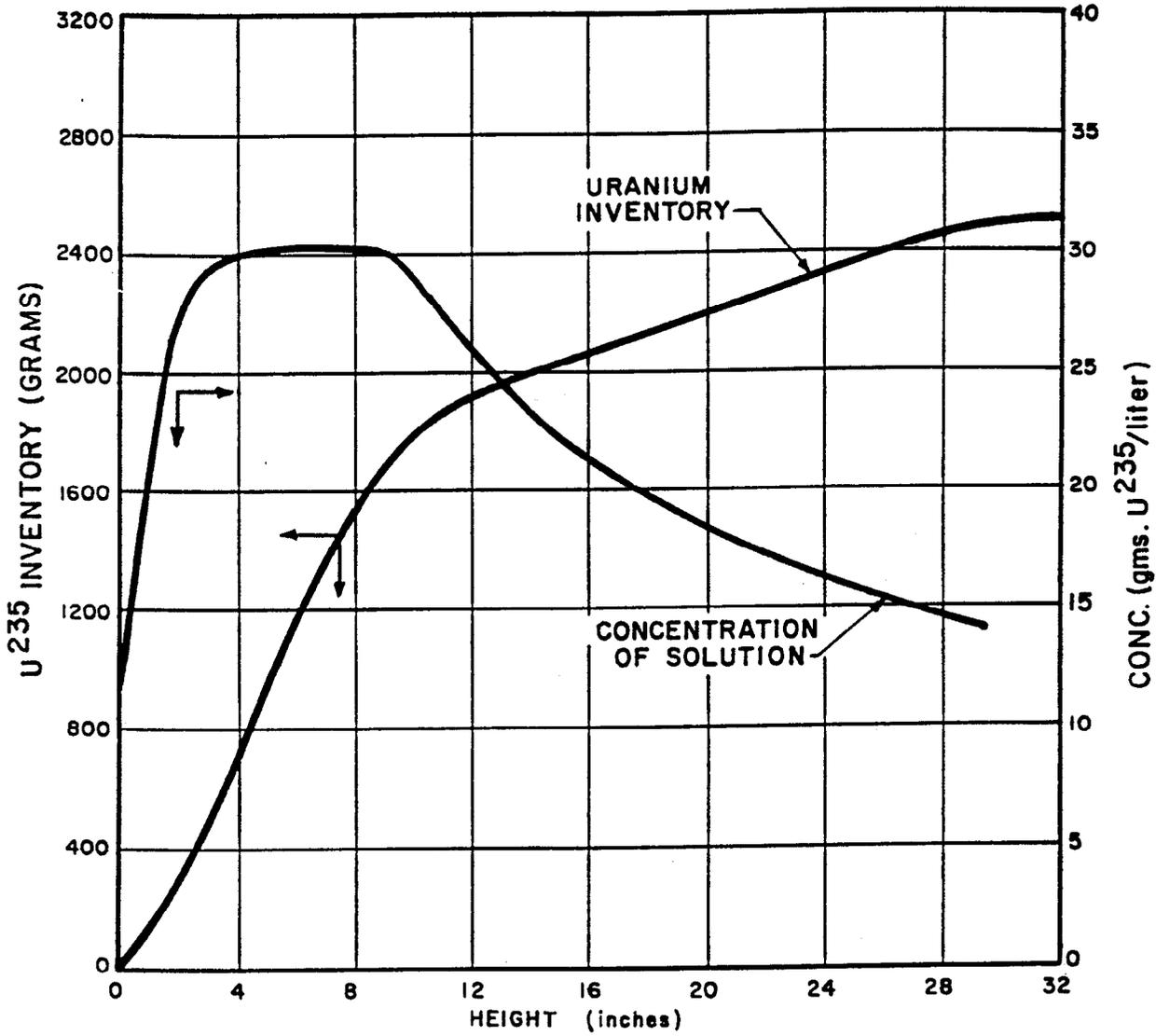
Figure D.3
URANIUM INVENTORY AND CONCENTRATION OF SOLUTION
IN DRUM AS FUNCTION OF HEIGHT

APPENDIX E

COMMENTS ON PREVIOUS NUCLEAR ACCIDENTS*

Previous experiences with unscheduled prompt neutron chain reacting assemblies have been in critical experiment laboratories [14, 24, 19] where, in most instances, at least the following conditions differ significantly from those present in the power excusion under consideration.

1. Inherent in the apparatus of critical experiments are devices for reducing the reactivity of an assembly from well above prompt critical to below delayed critical in the order of a hundred milliseconds. Although it is conceded that these devices may not act in a time sufficiently short to prevent a power excursion, they do minimize recurrences of power surges which may themselves have been previously terminated by some change in an intensive property of the assembly.

2. Critical experiments are made chain reacting in a fairly intense neutron field so the approach to critical, or to a condition in which it is exceeded, is made readily apparent through the radiation resulting from the fissions produced by the ambient neutrons and their progeny. In the absence of a source of neutrons and with a continuing addition of reactivity, an assembly may be well into the critical range before this condition manifests itself. In one controlled laboratory experiment, for example, a ten-second interval elapsed between the addition of 0.07% in reactivity to a critical sphere of solution, with no source, and an observable signal on the sensitive control instruments.

3. Most accidents in critical experiments have occurred with adequate separation or shielding protecting personnel so few radiation exposures have occurred. In two instances where this condition did not exist, fatalities resulted.

The energy release preponderating these accidents has been that originating from 10^{16} to 10^{17} fissions. That is, in the range of about a tenth to one kilowatt-hours.

The cause of some of these accidents has been sufficiently well understood to permit their reconstruction in order to analyze their behavior.

There have also been studies of scheduled prompt critical power excursions in both pure fissionable material (U^{235} enriched uranium metal) and in aqueous solutions of a U^{235} salt [6, 23]. The transients in these experiments, initiated by operation of the reactor controls, were observed to be suppressed by a change in fuel characteristics before the action of the reactor shutdown mechanism. These fuel alterations result, of course, from density variations due, in turn, to temperature and phase changes. In these pulses also, the energy corresponded to a range of 10^{16} to 10^{17} fissions and their duration was 100 msec or less. It is apparent therefore, from these observations, that nuclear power excursions in homogeneous fissionable materials have been self-quenching with an energy release of about 10^{17} fissions in a fraction of a second.

APPENDIX F

STATEMENT BY MEDICAL DIVISION STAFF, ORINS, JULY 30, 1958

Following the accident, Employees "A", "B", "C", "D", "E", "F", "G", and "H" were hospitalized at the Oak Ridge Institute of Nuclear Studies where they received specialized medical attention. This included consultations and visits by leading specialists in the field of radiation medicine. The medical status of the individuals involved is described in the following statement released on July 30, 1958, by the Medical Division Staff of ORINS:

"Of the eight men who were exposed to radiation in the Y-12 accident, three (Employees 'F', 'G', and 'H') did not have a sufficient dose to require prolonged hospital care. They exhibited mild changes in blood elements characteristic of radiation, but they showed no symptoms. They were released from hospitalization as soon as it was evident that their radiation exposure was small.

"Five (Employees 'A', 'B', 'C', 'D' and 'E',) of the eight men showed significant decreases in blood elements and other clinical and laboratory findings that were characteristic of more severe, but sublethal, radiation damage. During the first two days there was an early period of mild nausea accompanied by some vomiting. This was followed by a period of about three weeks during which they felt quite well and were almost completely free of symptoms. During this period the men left the hospital and returned to their homes for most of each day. During the fourth and fifth weeks following the accident, the platelet counts were decreased to levels that indicated the possibility of serious hemorrhagic complications. The blood platelets are one of the elements that control the ability of the blood to clot. At this time the men stayed in the hospital full time so that they could be watched by the medical personnel for possible bleeding. Except for a few almost unnoticeable events, such as 'pink toothbrush,' this bleeding did not occur. By the sixth week the laboratory studies indicated that all the men were showing unmistakable signs of recovery. Now (July 30, 1958) that this recovery phase is clearly established, they have been released from the hospital. There will be a long period of observation but this will be done on an out-patient basis."

APPENDIX G

ESTIMATION OF THE TOTAL NUMBER OF FISSIONS BASED ON A RADIATION SURVEY MEASUREMENT; A METHOD INDEPENDENT OF CHEMICAL ANALYSIS OF THE FUEL*

Knowledge of the time dependent behavior of the gross fission products formed from the thermal fission of uranium-235 may be used to estimate the total number of atoms of uranium which have undergone fission. Such a calculation relates the dose rate measured at a known time and distance to the rate of energy emission from the source resulting from fission events occurring at an earlier time.

A necessary assumption in the calculation is that all of the measured dose rate is due to homogeneously distributed fission products within the reaction vessel. While such an assumption is necessary, it results in an estimate which is obviously too high since other sources contribute to the observed dose rate. The most likely contributions result from fission product contamination of the surrounding area and the neutron induced activities within materials in the vicinity of the measurement.

The magnitude of the error introduced by these unwanted radiation sources was reduced by experimentally measuring the dose rate at a point near the source contained within the 55-gallon drum. Such a technique tended to increase the gamma dose rate from the desired source relative to that from the undesired sources.

METHOD 1

This calculation was made using a form of the Way-Wigner relationship

$$\Gamma(t) = 0.9\, t^{-1.2}\ \frac{Mev}{sec\text{-}fission}$$

where t is the time in seconds since fission.

Assumptions: (1) 47.5 gallons of liquid in drum
(2) Effective gamma energy = 1.0 Mev
(3) Density of solution = 1.1 gm/cm^3

The dose rate used in the calculation was measured on the mid-line of the drum at a point 2 ft. from the outer surface at 20.5 hours following the incident. This was found to be 23 r/hr.

This calculation indicates an excursion of 2.2×10^{18} fissions.

METHOD 2

This calculation was made using the decay spectrum from some unpublished data of Spencer and Hubble of the National Bureau of Standards. This technique eliminates the necessity of assuming

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G.2

an effective energy for the gross fission products. The energy spectrum was broken up into groups with an average energy for each group chosen to permit more reliable self-absorption calculations for the source. The dose rate measurement made at 20.5 hours was converted to 23.8 hours to correspond to the Spencer and Hubble data by assuming that the energy emission rate varies as $t^{-1.2}$.

The assumptions made are the same as previously indicated with the one exception of the effective energy.

This technique resulted in an upper estimate of 3.7×10^{18} fissions.

APPENDIX H

CRUDE ESTIMATES OF NEUTRON DOSE BASED ON RADIATION SOURCE*

The neutron flux leakage of 0.28 was calculated from a knowledge of the dimensions of the fuel in the 55-gallon drum assembly. These data were used to estimate the first collision and multiple collision neutron dose received by the eight exposed individuals assuming there had been a single excursion of very short duration, that the dose dropped off according to the inverse square law, and that there was negligible scattering and attenuation of the radiation. Table H.1 lists the crudely calculated absorbed dose and the RBE dose of neutrons received by the five individuals who accumulated the highest absorbed dose of ionizing radiation. The multiple collision neutron dose is given for the peak dose inside the body. The values are given for several energies, since at the time of these calculations the effective neutron energy was unknown. In this case the values of the RBE dose are undoubtedly too high because the functional relationship between RBE and specific ionization as given in the NBS Handbook 59 was used and, although this relation may apply to chronic exposure, it is known to be 3 to 5 times too large for neutrons in this energy range when the dose rate is very large. None of these values checks with the experimental and theoretical values determined by dosimetry, and they in no way confirm the clinical observations on the patients. In fact, the data in Table III are not given to aid the reader in estimating the dose to the exposed individuals, but rather to warn of the serious errors that result from this type of crude approximation. There is no doubt that the estimates of dose discussed in Exhibit V are far more accurate than these estimates which are based primarily on the energy released from the source.

Table H.l
APPROXIMATE DOSE CALCULATIONS FOR FAST NEUTRONS**

| | | | First Collision Dose | | | | | | | |
| | | | Absorbed Dose (rad) | | | | RBE Dose (rem) | | | |
Employee	Distance r from 55-gal drum (feet)	Neutron Flux (n/cm2)	2.5 Mev	1.0 Mev	0.5 Mev	0.1 Mev	2.5 Mev	1.0 Mev	0.5 Mev	0.1 Mev
"A"	6'	60.0×10^{10}	2,080	1,470	999	350	15,100	16,100	10,600	5,000
"B"	15'5"	10.09×10^{10}	349	247	168	60	2,540	2,720	1,780	840
"C"	17'8"	7.69×10^{10}	266	188	128	46	1,940	2,070	1,350	640
"D"	16'4"	9.00×10^{10}	312	220	150	53	2,270	2,420	1,580	750
"E"	22'3"	4.85×10^{10}	168	119	81	29	1,220	1,300	856	404
"F"	31'0"	2.50×10^{10}	86	61	42	15	630	673	440	210

| | Multiple Collision Dose | | | | | | | |
| | Absorbed Dose (rad) | | | | RBE Dose (rem) | | | |
	2.5 Mev	1.0 Mev	0.5 Mev	0.1 Mev	2.5 Mev	1.0 Mev	0.5 Mev	0.1 Mev
"A"	2,640	2,280	1,320	630	20,400	23,400	13,200	6,000
"B"	444	383	222	106	3,430	3,935	2,220	1,010
"C"	338	292	169	81	2,614	3,000	1,690	769
"D"	396	342	198	94	3,060	3,510	1,980	900
"E"	213	184	107	51	1,650	1,890	1,070	485
"F"	110	95	55	26	850	975	550	250

* Principally prepared by K. Z. Morgan, Health Physics, Oak Ridge National Laboratory.

** Based on 10^{18} fissions, $r^{1/2}$ corrections, an escape fraction of 0.28, and W. Snyder's Dose Curves in NBS - HB 63. The RBE values used were functions of specific ionization as described in NBS - HB 59. It was later determined that an escape fraction of 0.22 is more accurate, but the values above have not been revised.

I.1

APPENDIX I

EXPERIMENTAL APPROXIMATION OF THE Y-12 ACCIDENTAL
NUCLEAR EXCURSION OF JUNE 16, 1958*

The biological effect of the ionizing radiation emitted by a volume of fissionable material in which a nuclear power excursion is occurring is a strong function of the distribution of this energy between its neutron and gamma-ray components and the spectral distribution of the energy in each of these components. These characteristics of the radiative energy are dependent upon the physical properties, such as the shape, dimensions, chemical concentrations, etc. of the volume in which the excursion occurs.

In order that post-accident measures might be representative of the radiation exposures received by the Y-12 personnel, it was deemed necessary to generate a radiation field by a chain-reacting system having characteristics at least similar to the solution in the C-1, 55-gallon drum while it was critical, and measure pertinent physical and biological properties. Accordingly, a series of experiments were planned and performed in the Oak Ridge Critical Experiments Laboratory, Building 9213, on June 18 and 19.

The critical conditions of an aqueous solution of U^{235} salt were predicted from best estimates of the quantity of the uranium which became critical in the drum located in C-1 Wing in Building 9212 and the dimensions of the volume it then occupied. A critical system was then constructed from which, it is believed, the emitted radiation was similar to that from the C-1 55-gallon drum. In this experiment the cylinder diameter and height were 20 inches and 15 inches, respectively, and the critical concentration was 25.9 g. U^{235}/ liter. The critical mass was 2.00 kg U^{235}.

Tests were made during two operations of this critical system. During the first of these, which was operated for eleven minutes at a power of about 6.5 watts (a total of 1.3×10^{14} fissions), a comparison was made of the gamma-ray and neutron yields and some measurements of the spectral distributions of the energies were made. The second run lasted 42 minutes at a power of about 300 watts, yielding 2.4×10^{16} fissions. In this test additional detectors and some animals were irradiated.

The results of these tests are given in Exhibit V of this report.

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* Principally prepared by A. D. Callihan, Critical Experiments, Oak Ridge National Laboratory.

APPENDIX J

ESTIMATES OF DOSE BASED ON IN VIVO BODY COUNTER*

All Y-12 personnel with significant activity indication from the indium foil security badge were analyzed in the body counter. The eight men with highest badge activity were counted between 6:30 p.m. on June 16, 1958, and 1:00 a.m. on June 17, 1958. A direct measurement of the Na^{24} activity was obtained by counting the gamma energy band between 2.76 and 2.98 mev.

These measurements were compared with that of a burro which was exposed to the simulated incident in an experiment conducted at the criticality laboratory. The product of the ratio of Na^{24} activity in the humans to that of the burro per unit weight and the determined neutron dose of the burro gave a neutron dose for each of the eight employees.

Two sources of error in this analysis have been examined. These are: (1) the different counting geometry between a burro and a human, and (2) the variation in equipment counting loss at high gamma activities. Subsequent Na^{24} phantom experiments revealed variations in the dead time loss of the counting equipment, which imposes the necessity of reporting a revised estimate as shown in Table J.I.

Additional experiments are planned to evaluate the difference in the blood and total body counting of Na^{24}.

Table J. I
DOSE MEASUREMENTS FROM IN VIVO COUNTER

Employee	Activation ($\mu c\ Na^{24}/kg$)	Neutron Dose* (rads)	Total Dose*** (rads)
"A"	.682	139	528
"C"	.652	133	505
"D"	.524	107	407
"E"	.506	103	391
"B"	.382	78	296
"F"	.198	40	152
"G"	.191	39	148
"H"	.111	23	87
Burro	.236	48**	--

* Neutron Dose (rad) = $\dfrac{Na^{24}\ \text{in human/kg}}{Na^{24}\ \text{in burro/kg}}$ x Neutron dose of burro

** Burro Dose = 48 rad as determined by ORNL

*** Total Dose = Neutron Dose + Neutron Dose x gamma/neutron ratio (2.8) as provided by ORNL

APPENDIX K

ESTIMATES OF DOSE BASED ON INDIUM FOIL MEASUREMENTS*

As indicated under Health Physics Activities, the indium foils in the employees' security badges proved to be entirely adequate for the purpose for which they were intended, this being to permit the rapid detection of those employees who had received a significant radiation exposure from an inadvertent critical reaction and to permit a rapid estimate of the exposure levels involved. However, the foils obviously have some of the same deficiencies with respect to accurate dose measurements as other personnel dosimetric devices. Specifically, the human body absorbs an appreciable fraction of the radiation resulting from a critical reaction which is incident upon the body, and, thus, the response of any dosimetric device will vary according to whether its location on the body is toward or away from the source of radiation.

However, in the absence of other calibrated means for determining the actual exposures, efforts were made to estimate these exposures on the basis of the indium foil activities, using an existing approximate calibration[1] obtained with a similar well-moderated critical assembly. For this calibration, the computed doses were based upon an RBE value of 10 for fast neutrons, and upon a neutron spectrum in which 1/3 of the neutrons were of energy greater than 1000 ev and 2/3 of energy less than 1000 ev. The gamma-to-neutron dose ratio was based upon the probabilities of escape of these radiations from the critical assembly.

The activities of the indium foils of the persons receiving the highest exposures were determined by a gamma scintillation counter having an estimated geometry of 15%. A correction for the RBE of fast neutrons was applied to the neutron dose upon the advice of Dr. K. Z. Morgan, who indicated that with radiation levels of the type involved the applicable RBE value would be 3 or less. Since the indium foils were expected to yield only approximate dose values, however, no immediate effort was made to include corrections for individual foil weights, a relatively small geometry correction, or variation of the neutron energy distribution from that assumed for the early calibration determination.

The doses as determined as of 2:00 A.M. on June 17, 1958, by this method are listed in Table K.I.

In subsequent evaluations of the indium foil data, the relationship between the dose per neutron/cm^2, as given as a function of neutron energy in the National Bureau of Standards Handbook 63, was utilized in conjuction with a theoretical spectrum**, calculated at the ORGDP, to determine the average dose per neutron/cm^2. This average value R, is represented by the ratio

$$ R = \frac{\int_0^\infty \phi(E) R(E) \, dE}{\int_0^\infty \phi(E) \, dE} $$

$\phi(E)$ = neutrons per unit energy interval in the neutron spectrum

$R(E)$ = rads per (neutron/cm2).

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* Principally prepared by J. Bailey, Health Physics, Oak Ridge Gaseous Diffusion Plant.

** Neutron energy spectra for the different degrees of moderation as used in this evaluation were developed by J. R. Knight.[13]

Table K.I
DOSES FROM INDIUM FOIL MEASUREMENTS
INITIAL ESTIMATES

Employee	Gamma Dose (rad)	Neutron Dose (rad)	Total Dose (rad)	Total Dose (rem) RBE ≈ 3	RBE = 1.5
"A"	167	62	229	353	260
"B"	125	46	171	264	194
"C"	209	77	286	441	325
"D"	138	50	188	289	213
"E"	282	104	386	595	438
"F"	68	25	93	143	106
"G"	55	20	75	115	85
"H"	37	14	51	78	58

The integrals indicated were evaluated numerically to obtain the value of R, and this value was also computed from the energy spectrum as determined by fission foils in the mock-up experiment, utilizing the first-collision dose as a function of energy as given by G. S. Hurst of ORNL. The value of 1.89×10^{-9} rad/(n/cm^2) given by this latter method was in close agreement with the value 1.93×10^{-9} rad/(n/cm^2) as determined theoretically, although differences were noted as to the distribution of dose among the various energy ranges; in particular, the first-collision dose calculation indicates a lower fraction of the dose for thermal neutrons than does the theoretical calculation, which includes the total dose.

The percentage of flux and percentage of dose for various energy ranges, as computed by the theoretical method, are given in Table K.II.

Table K.II
CALCULATED NEUTRON ENERGY AND DOSE DISTRIBUTION

Energy Range	% Total Neutrons	% Total Dose
Thermal (0 - 0.04 ev)	35.8	13.1
0.04 ev - 5000 ev	9.2	0.4
5000 ev - 0.75 Mev	14.0	8.0
0.75 Mev - 1.5 Mev	8.4	11.5
1.5 Mev - 2.5 Mev	10.9	18.0
2.5 Mev - 10 Mev	21.8	49.0

An evaluation of the effect of variations in the neutron spectrum on foil activation indicated that this activation was due primarily to neutrons in the thermal region for both the original calibration experiment and for the neutron spectrum currently under consideration; the fraction of neutrons in the thermal region, according to the calculated spectra, was twice as great in the moderation range existing at the time of the accident as for the calibration experiment; and accordingly, the neutron activation per neutron/cm^2 was considered to be twice that determined in the earlier experiment, this earlier value being 4.6×10^{-3} (disintegrations/min/g)/(n/cm^2). The factor applicable to the personnel-foil activations was therefore considered to be 9.2×10^{-3} (disintegrations/min/g)/(n/cm^2). The ratio of neutron dose to indium foil activity, with foil activity being corrected for radioactive decay subsequent to exposure, was determined from this value and from the value for rad/(n/cm^2), previously discussed, to be 2.1×10^{-7} rad/(disintegrations/min/g).

The ratios for gamma to neutron dose, as given in Exhibit V, were used to determine the gamma dose. The radiation doses for the individuals receiving the highest exposures were calculated with these factors and are tabulated in Table K.III, along with the foil activations corrected for radioactive decay following exposure. These values are considered to represent the best dose estimates currently available from the indium-foil measurements.

Table K.III
INDIUM-FOIL DOSE DETERMINATIONS

Employee	Foil Activation (dis/min/g)	Neutron Dose (rad)	Gamma Dose (rad)	Total Dose (rad)	Total Dose (rem)*
"A"	2.28×10^8	48	126	174	222
"B"	1.48×10^8	31	94	125	186
"C"	2.40×10^8	50	140	190	240
"D"	1.92×10^8	40	112	152	192
"E"	2.66×10^8	56	157	213	269
"F"	0.71×10^8	14.7	41	56	71
"G"	0.73×10^8	15.3	43	58	73
"H"	0.42×10^8	8.9	25	34	43

* With an assumed RBE = 2 for neutrons

In may be noted that, although some indium-foil data were obtained during the mock-up experiment, this experiment was designed primarily to calibrate the blood-sodium dose determinations, and information to permit a reasonably adequate evaluation of the indium-foil activations experienced in the accident was not obtained.

However, if the activations of indium foils located on the side of the burro directly toward the reactor, and thus exposed to the maximum incident and reflected fluxes, were utilized to determine the personnel doses, the indicated doses would be lower by a factor of approximately 2.4 than those shown in Table K.III. If the activation of foils on the side of the burro directly away from the reactor were utilized to provide a calibration, the personnel doses would be higher than those indicated in the table by a factor of about 3.2, these differences reflecting an approximate eightfold reduction in activation effected by the body of the burro.

APPENDIX L

DANGER ASSOCIATED WITH PHYSICAL APPROACH TO A NEAR-CRITICAL REACTOR*

The criticality accident discussed in this report has raised a question concerning the advisability of physical approach to the "reactor" following the nuclear excursion. This question is associated with the possibility that the "reactor" is virtually critical, and that the approach of an individual may add sufficient reactivity so as to cause another nuclear burst, resulting in dangerous exposure to the individual concerned. Based on the results obtained here, no danger appears associated with neutron instrumented approach to within about 2 feet of the "reactor"; however, extreme danger can be associated with approaches less than about a foot. Clearly, inadvertent approach is hazardous.

If the reactor is critical, the physical approach of a person will decrease the effective neutron leakage, and result in a reactivity addition. The amount of neutron reflection associated with the approach of an individual is a function not only of the distance of the individual from the reactor but also of the fraction of generated neutrons which leave the reactor. The latter quantity will decrease with increasing reactor size and will be smaller for H_2O moderated systems than for comparable D_2O systems.

Since the amount of neutron reflection is very dependent upon the distance a person is from the reactor, and increases with decreasing distance, near the reactor it may be possible to add relatively large reactivity additions by physical movements toward the reactor.

If the "reactor" were operating at a very low but detectable fission-power level, an individual carrying appropriate neutron-detection equipment, while yet an appreciable distance away, would observe a relatively low rise in neutron level as he approached the reactor. However, if the neutron source were not detectable (i.e., if the neutron source strength were extremely low or masked by residual activity from a previous excursion), the approaching individual may approach significantly closer to the reactor before detecting the rise in neutron level (assuming that in coming closer he causes the reactor to become supercritical). Under these circumstances, it is important that the rise in neutron density be detected before the reactor period becomes too short; this will permit the individual to move away from the reactor before he receives a harmful exposure.

In the study it is assumed that the initial neutron source strength is not detectable, and that the important quantity is the amount of reactivity addition associated with physical movement toward the reactor. It is further assumed that the person detects the reactor power when it reaches a level of ten watts. He is then exposed for a period of ten seconds during which the reactor power is rising on a period associated with the reactivity addition. After 10 seconds it is assumed that the individual has removed himself from the vicinity of the reactor and receives no more exposure. In this study it was assumed that the reactor and the person can be replaced by equivalent parallelepipeds; one-half of the neutrons striking the person were assumed to be reflected back through the entering surface; an appropriate geometric factor (based on the assumed geometries) was used in calculating the number of these neutrons which are returned to the reactor. The reactivity addition associated with these reflected neutrons was then calculated.** The power as a function of

$$k\infty = \frac{\text{NEUTRONS GENERATED}}{\text{NEUTRONS ABSORBED}}$$
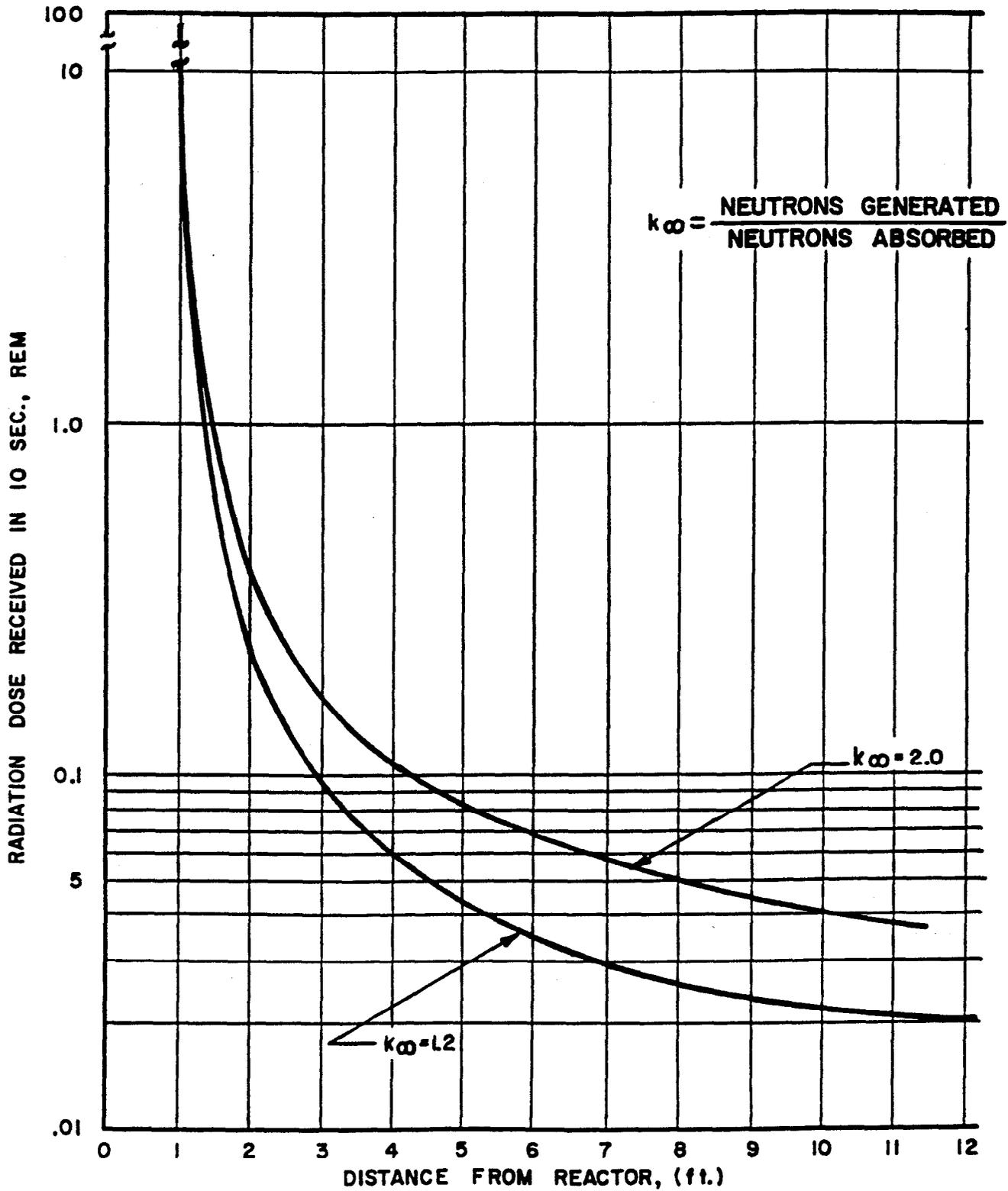
Figure L.1
RADIATION DOSE AS A FUNCTION OF
DISTANCE FROM REACTOR

time was determined from the stable period associated with the reactivity addition; the radiation exposure was then calculated on the basis of a ten-second exposure during which the power rose from its initial power of 10 watts. Figure L.1 gives the results of these calculations, and plots exposure in rems (assuming a RBE of 2 for neutrons) as a function of the closest distance between the person and the reactor surface. The fraction of generated neutrons leaking from the reactor was assumed to be 15% or 50%; this variation appears to cover cases of interest. (The neutron lifetime would be significantly different for these different neutron-leakage levels; this was considered in the calculation.)

In this study the reactor was assumed to be essentially critical under initial conditions. If the reactor were subcritical initially, the distance associated with a given exposure would be smaller, and would decrease to zero if the reactor were sufficiently subcritical. As indicated in Figure L.1, no dangerous exposure appears associated with physical approach within about 2 - 3 feet of the system; however, closer approaches could cause a reactor excursion leading to extreme over-exposure to the individual concerned, and also to personnel within the immediate vicinity. In order to include a reasonable safety factor, approach should not be closer than within 5 feet of the reactor.

APPENDIX M

SUPPLEMENTARY INFORMATION

RE-ENACTMENT PHOTOS

Presented herein are several figures, M.1 through M.4, which depict the additional details of the actions of Employees "A", "B", "C", "D", and "E" in C-1 Wing, Building 9212, as well as other aspects of the enriched uranium salvage facilities.

Table M.I presents the results of significant chemical analyses of samples taken from the system following the incident.
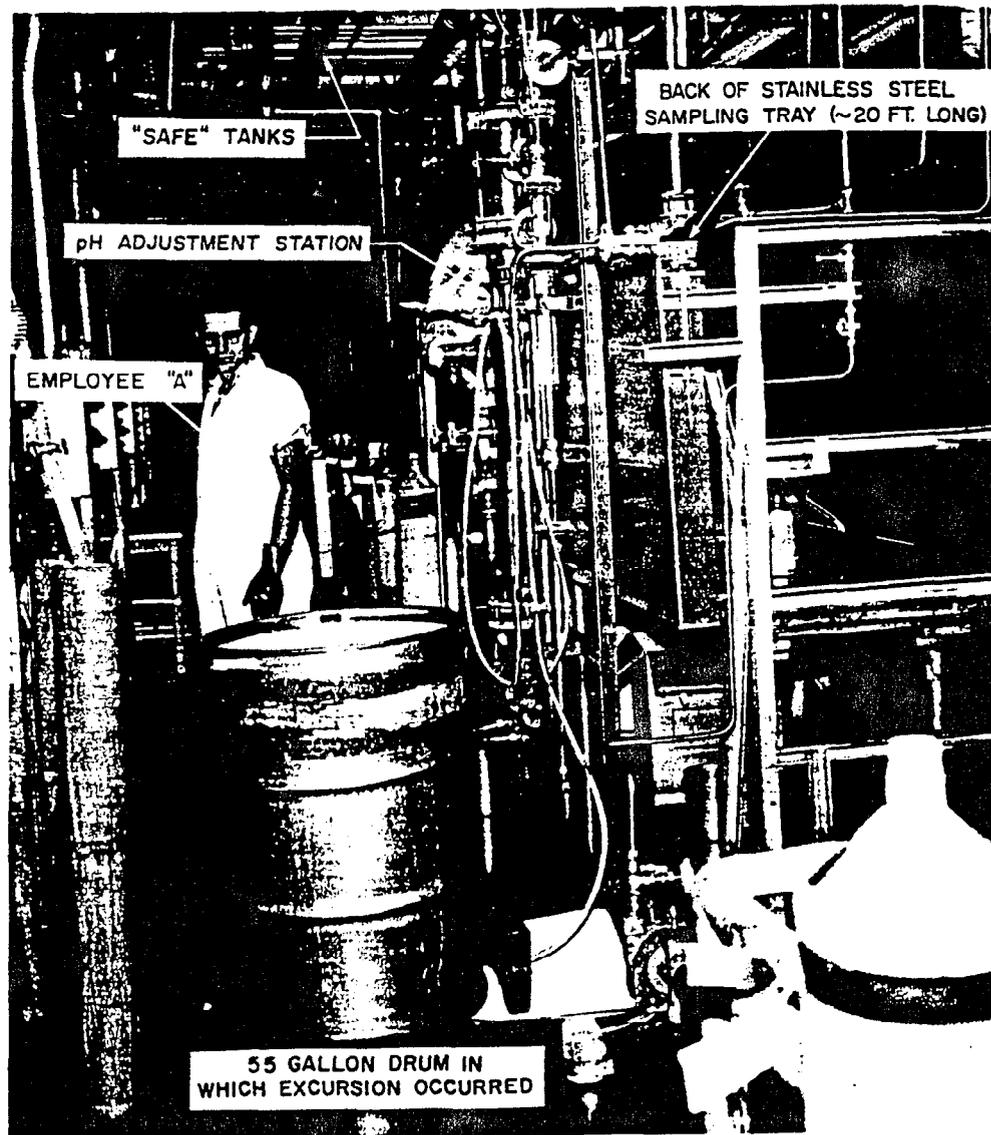

Figure M.1

POSITION OF EMPLOYEE "A" AT THE TIME OF THE INCIDENT

Note: The high back of the sampling tray prevents Employee "A" from observing the actions of Employees "B", "C", "D", and "E". (See Figure 5)

M.2
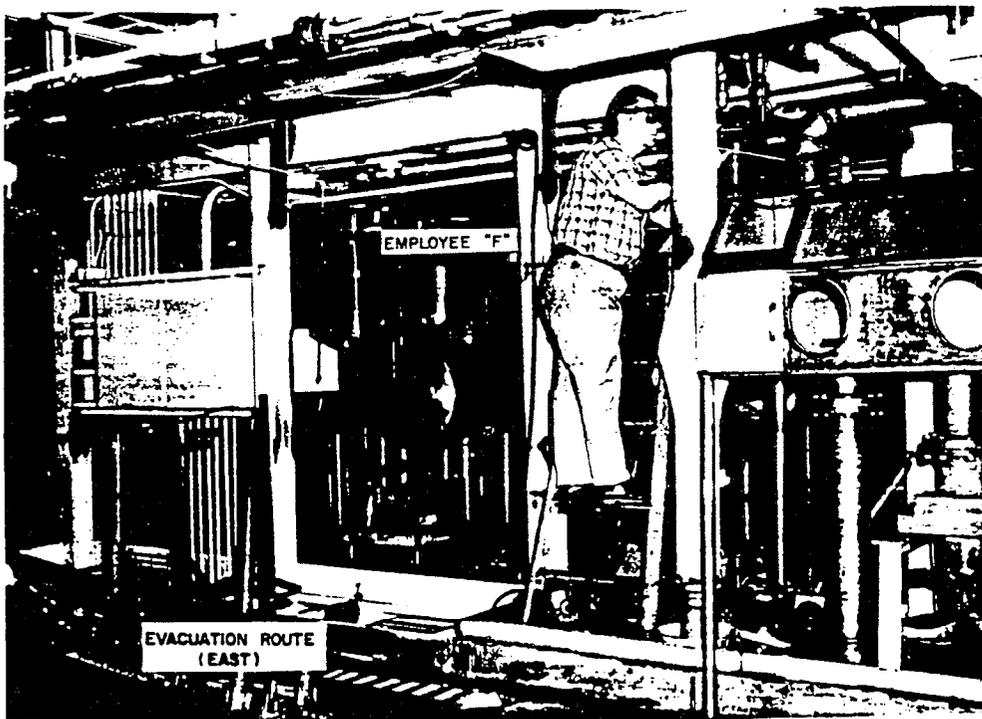
Figure M.2
CLOSE-UP OF THE POSITION OF EMPLOYEE "F" ON MEZZANINE DIRECTLY
ABOVE EMPLOYEE "C" AT THE TIME OF THE INCIDENT (Looking South)
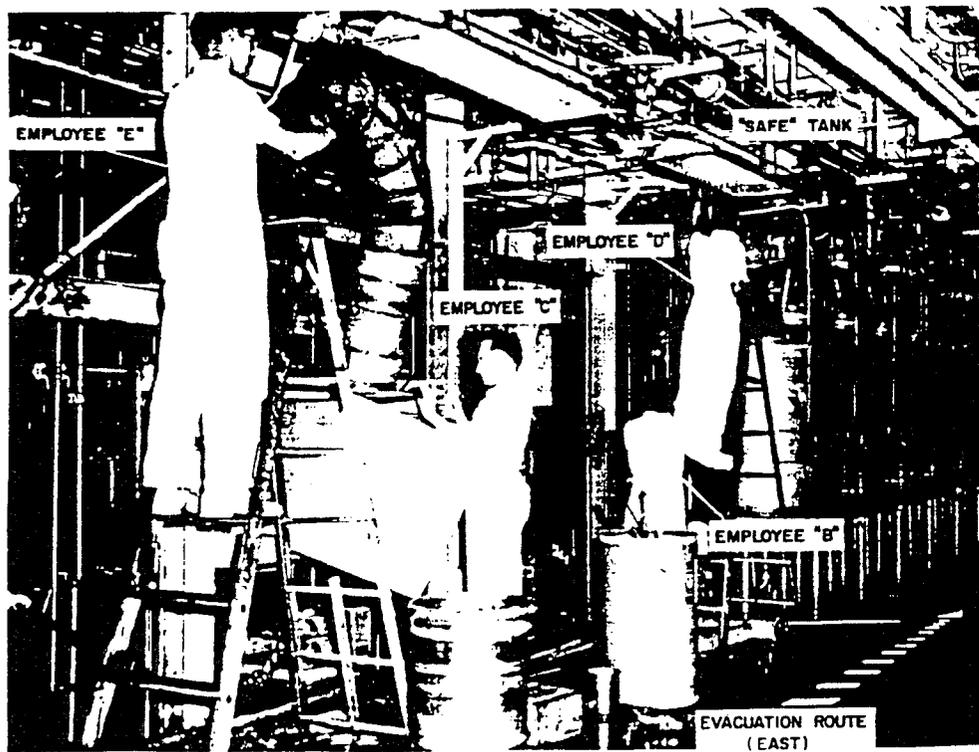

Figure M.3
CLOSE-UP OF THE POSITIONS OF EMPLOYEES "E", "C", "B", AND "D"
(Looking North East)
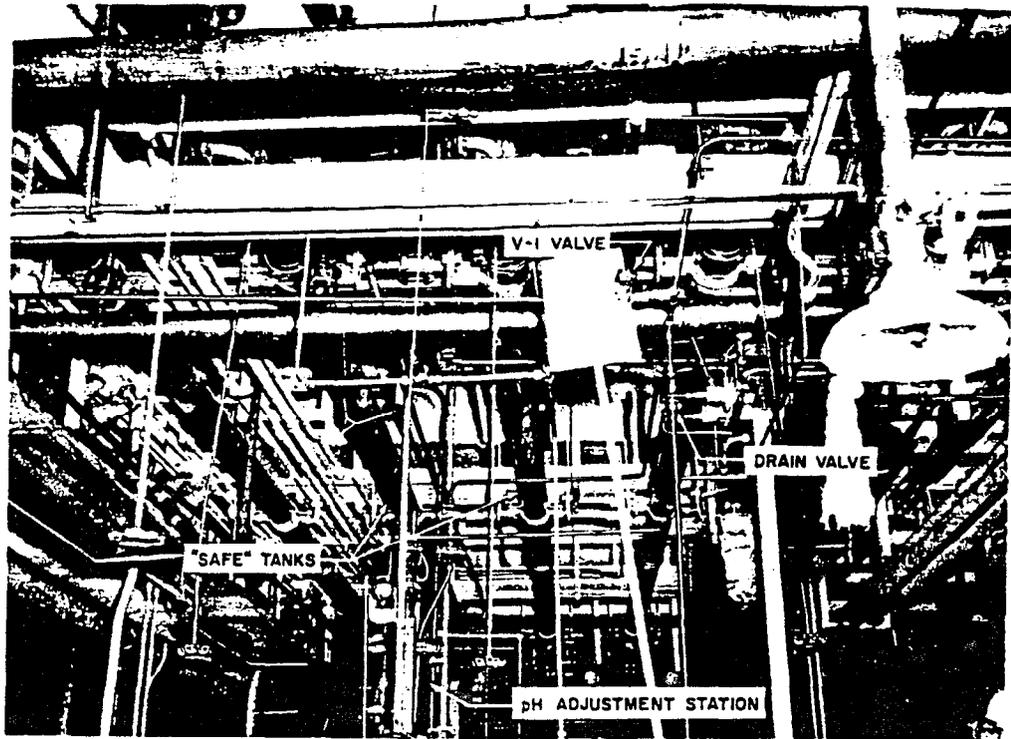
Figure M.4

LOCATION OF V-1 VALVE AND THE DRAIN VALVE IN C-1 WING

SUPPLEMENTAL CHEMICAL ANALYSES

Table M.I

ANALYSES OF SIGNIFICANT SAMPLES REMOVED FROM
THE SYSTEM AFTER THE INCIDENT

Sample Source	Conc. of U (g U^{235}/g Solution)	Sp. Gr.	Nitric Acid Conc. (%)	Carbitol Conc. (%)	Significant Impurities, U_3O_8 Basis (ppm)				
					Al	Fe	Ca	Na	Cd
Product left in B-1 Product Tanks	0.0406	1.186	18.85		2050	180	150	175	150
B-1 to C-1 Transfer Line Upstream of Valve V-1	0.0476	1.188	17.79		2600	775	225	150	150
pH Adjustment Tube	0.0469	1.138	10.76		2200	200	70	140	125
Low End of Tanks 6-1 and 6-2*	0.00082	0.995	0.13	9.53					
Low End of Tank 1-2	0.0352	1.042	8.12	32.4	360	220	225	145	35
Overflow Safe Bottle at High End of Tanks 6-1 and 6-2*	0.0005								

* The Spectrographic analyses of these samples were not significant because of the use of tap water in the leak testing procedure and the low uranium concentration.

M.4

POST-ACCIDENT DESCRIPTION OF THE DRUM

Following the transfer of the irradiated solution from the drum to storage, the drum and its liner were removed to ORNL for examination. There it was found to contain, in addition to the cadmium scroll, some liquid with suspended solids. Although rather detailed analyses were made, only the results will be summarized here.

The liquid was an aqueous-organic mixture, not unexpected since carbitol is used in the B-1 Wing extraction columns. The solids were largely uranium with a few percent cadmium and iron, consistent with the B-1 Wing process, the stainless steel vessels, and the addition of cadmium to the solution shortly after the accident. The liquid and the solids contained a total of about 25 grams uranium. Adhering to the cadmium scroll were yellow crystals which analyzed 35% uranium. Figure M.5 is a photo showing the sludge at the bottom of the drum liner, and Figure M.6 is one of the cadmium scroll after removal. Figure M.7 is a side view of the polyethylene liner and shows the distortion of the wall resulting from molding it into the convolutions of the drum, an indication of pressure and temperature conditions during the accident. Infrared analysis of microtome sections of polyethylene samples showed some degradation of the plastic due to chemical rather than radiation effects. There is no information on the possibility that the chemical reactions were induced by radiation. The liner material was estimated to contain 50 grams uranium.

Radioisotopic analyses were made of the stainless steel from various locations on the drum, yielding the relative neutron exposures recorded in Table III and Figure 19.
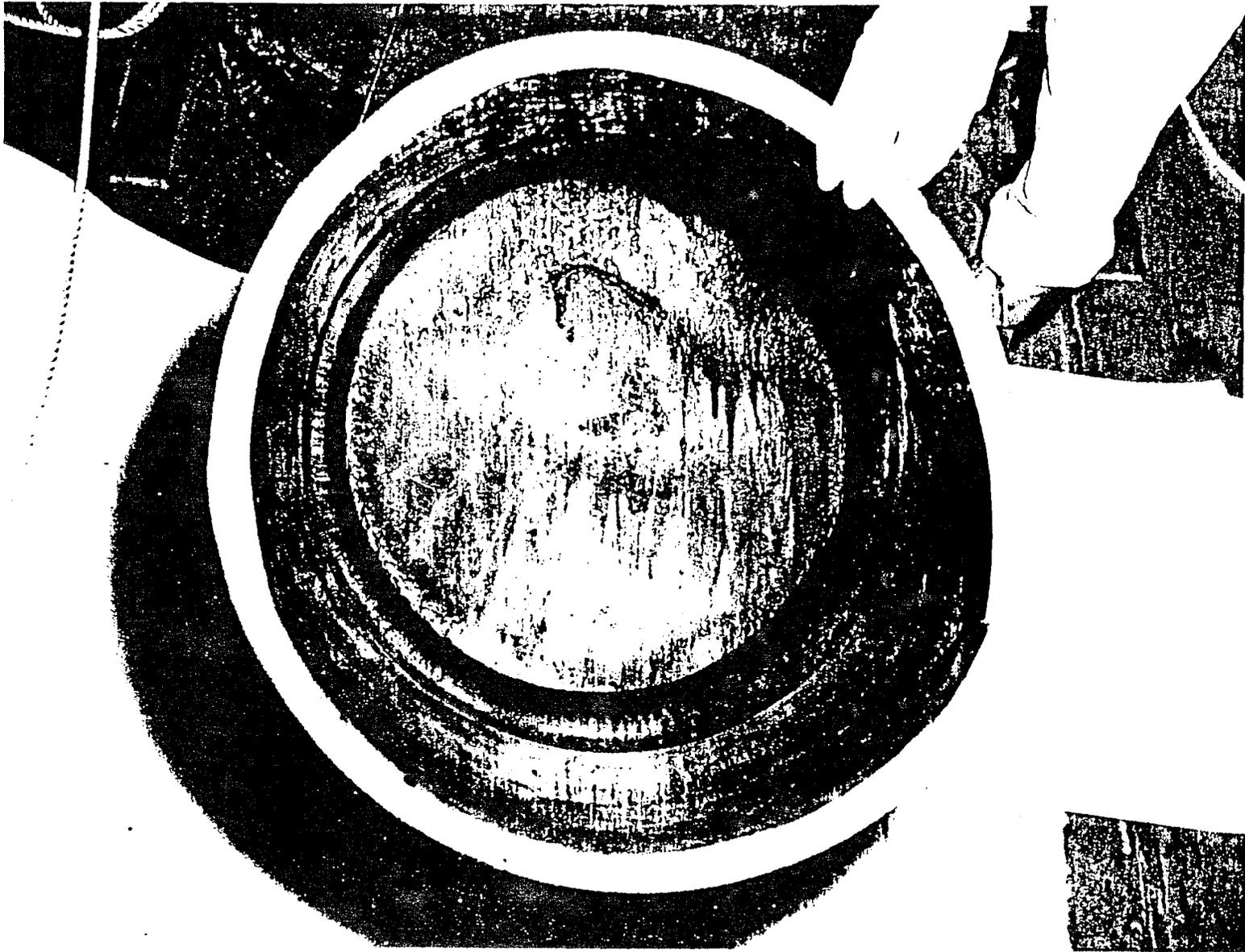
Figure M.5
PRECIPITANT IN POLYETHYLENE LINER

Figure M.6
CADMIUM SCROLL AFTER REMOVAL FROM 55-GALLON DRUM

Figure M.7
OUTSIDE SURFACE OF POLYETHYLENE LINER

GLOSSARY

(a, n) reaction — the capture of an alpha particle by a nucleus which results in the emission of a neutron.

Alpha radiation (α) — doubly charged helium ions, He^{++}, which are emitted from radioactive nuclides.

Beta radiation (β) — electrons emitted from radioactive nuclides.

Cerenkov radiation — electromagnetic radiation (in this report a visible blue glow) emitted during the interaction of radiation with matter.

Critical mass — minimum amount of U^{235} required to maintain a nuclear chain reaction in a particular set of physical and chemical conditions.

Critical reaction — a situation in which a nuclear chain reaction is self-sustaining; just as many neutrons are produced as are absorbed and lost.

Cross-section (neutrons) — a measure of the probability that a nucleus will capture a neutron. The cross-section is a function of the neutron energy and the structure of the target nucleus.

Curie (c) — 3.7×10^{10} disintegrations per second.

Delayed critical — the condition of a reactor whereby the nuclear chain reaction is maintained by both prompt and delayed neutrons.

Delayed neutrons — neutrons emitted by radioactive fission products during their decay.

dpm — disintegrations per minute.

dps — disintegrations per second.

Dosimeter — a device from which the exposure of personnel to radiation can be determined.

Electron volt (ev) — the energy acquired by any particle carrying a unit charge when it passes, without resistance, through a potential difference of one volt.

Fission — the disintegration of a heavy nucleus, made unstable by neutron absorption, into two or more nucleii of intermediate mass accompanied by neutrons and other radiation; e.g., U^{235} may capture a neutron and split into Ba^{144} and Kr^{89}, plus 3 neutrons, plus gamma radiation.

Fission product — a nuclide which results from the fission or splitting of an atom of a heavy element, such as U^{235}; e.g., U^{235} may be split into 3 neutrons plus Ba^{144} and Kr^{89} which are fission products.

Gamma radiation	- photons (electromagnetic energy) emitted from radioactive nuclides.
Geometric buckling	- a property specified by a reactor's size and shape which determines the neutron leakage from the reactor.
Godiva reactor	- an unreflected U^{235} metal critical assembly located at the Los Alamos Scientific Laboratory.
Gross activity	- total activity of unseparated fission products.
Half life	- the time required for one-half of a given number of radioactive atoms of the same element to decay.
Isotope	- one of a group of nuclides having the same atomic number, but various atomic weights; e.g., O^{16}, O^{17}, and O^{18} are all isotopes of oxygen.
Kev	- one thousand electron volts.
Mev	- one million electron volts.
Microcurie (μc)	- one one-millionth of a curie, or 3.7×10^4 disintegrations per second.
Moderator	- a material, such as water or beryllium, which will effectively slow neutrons to thermal energy without capturing a significant number of them.
mr	- one one-thousandth of a roentgen.
(n, γ) reaction	- the capture of a neutron by a nucleus which results in the emission of gamma radiation.
(n, p) reaction	- the capture of a neutron by a nucleus which results in the emission of a proton.
Neutron	- a fundamental atomic particle carrying no electrical charge. Its mass is slightly greater than a hydrogen atom or 1.00897 atomic mass units.
Neutron flux	- the number of neutrons passing, per second, through an area of one square centimeter (equals number of neutrons per cubic centimeter times neutron velocity).
Neutron leakage	- the escape of neutrons from a reactor.
Nuclear poison	- a material, such as cadmium, having a high neutron absorption cross-section which, if present in a reactor, reduces the neutron flux.
Nuclide	- an atomic specie characterized by the composition of its nucleus; i.e., the numbers of protons and neutrons it contains.
ORGDP	- Oak Ridge Gaseous Diffusion Plant.

ORINS	- Oak Ridge Institute of Nuclear Studies.
ORNL	- Oak Ridge National Laboratory.
Power excursion	- a nuclear chain reaction in which a relatively large amount of energy is produced in a short period of time.
Prompt critical	- the condition of a reactor whereby the nuclear chain reaction is maintained by prompt neutrons alone.
Rad	- that amount of ionizing radiation which imparts 100 ergs of energy per gram of irradiated material.
Radiation dose	- a quantity of ionizing radiation.
RBE (relative biological effectiveness)	- a constant for converting radiation dose expressed in physical units (rad) to its biological effect; e. g. , one rad of fast neutrons (with an RBE of 2) does twice as much damage to a living organism as one rad of gamma rays (with an RBE of 1).
Reflector	- a material which scatters neutrons back into a nuclear reactor.
Rem (roentgen equivalent man)	- defined by: Dose in rems = (Dose in rads) x (RBE).
Rep (roentgen equivalent physical)	- dose of any nuclear (or ionizing) radiation that results in the absorption of 93 ergs/gram of tissue.
Roentgen (r)	- that quantity of X- or Y-radiation producing, as a result of ionization, one electrostatic unit of electricity in 1 cc of dry air at 1 atmosphere and 0° centigrade.
"Safe"	- a term describing equipment for processing fissionable materials in which nuclear safety is imposed by geometry alone.
Subcritical	- a condition in a reactor whereby neutrons are absorbed and lost at a greater rate than they are produced; subsequently, the chain reaction dies out.
Supercritical	- a condition in a reactor whereby neutrons are produced at a greater rate than they are absorbed and lost.
Target nuclide	- a nuclide which captures incident radiation.
Thermal fission	- fission induced by thermal neutrons.
"Unsafe"	- a term describing equipment for processing fissionable materials in which nuclear safety is not imposed by the geometry of the equipment.
Whole body (In Vivo) counter	- a highly sensitive gamma counter, located inside a shielded room, which is used to determine, from the gamma ray spectrum, any radioactive nuclides which are present in a patient's body.

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