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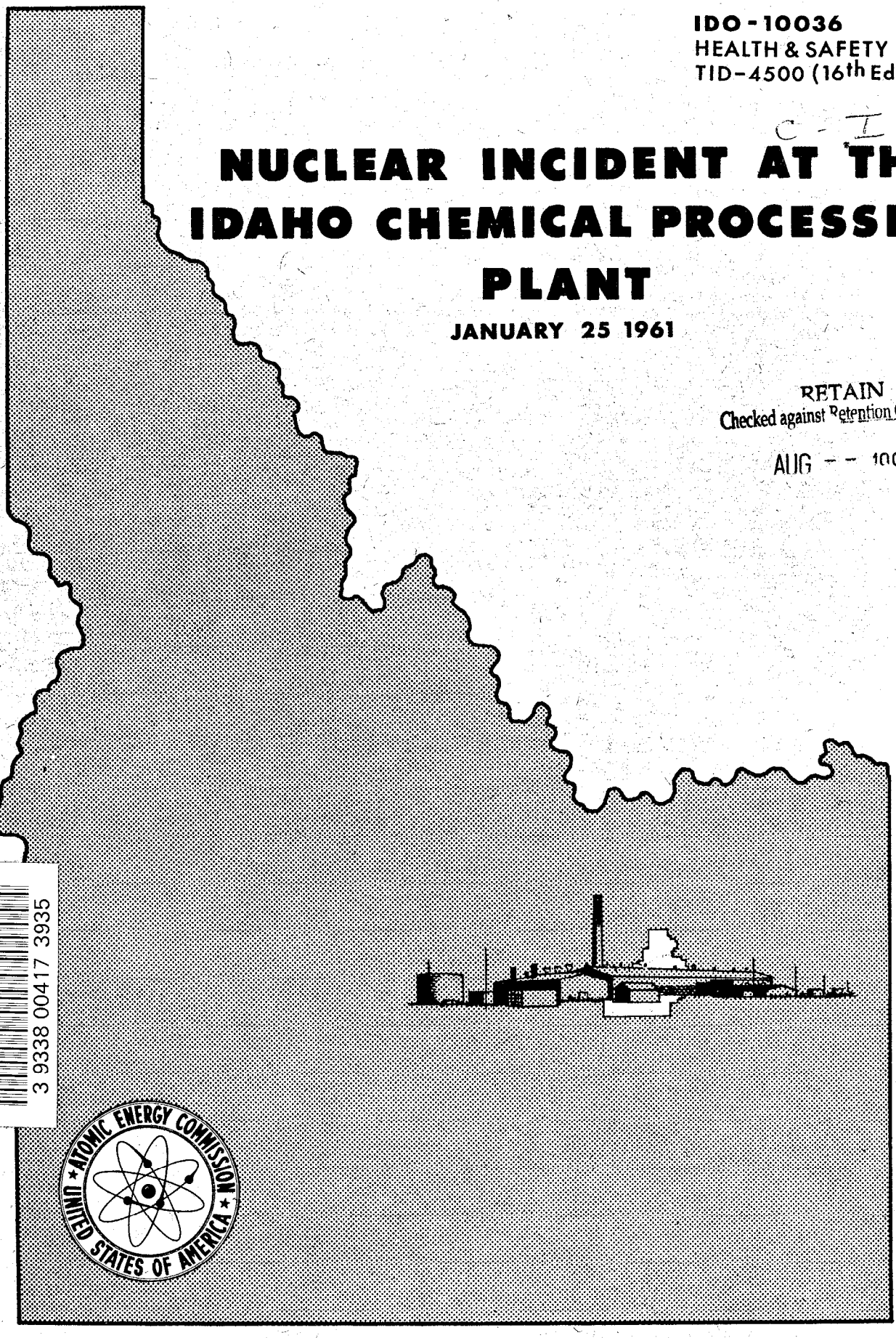
NUCLEAR INCIDENT AT THE IDAHO CHEMICAL PROCESSING PLANT

JANUARY 25 1961



UNITED STATES OF AMERICA ATOMIC ENERGY COMMISSION

NATIONAL REACTOR TESTING STATION

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NUCLEAR INCIDENT
at the
IDAHO CHEMICAL PROCESSING PLANT
on
JANUARY 25, 1961

REPORT OF THE INVESTIGATING COMMITTEE

by

R. C. Paulus, Chairman
A. O. Dodd
K. K. Kennedy
F. H. Tingey
F. M. Warzel

PHILLIPS PETROLEUM COMPANY - ATOMIC ENERGY DIVISION
U. S. ATOMIC ENERGY COMMISSION - IDAHO OPERATIONS OFFICE
Idaho Falls, Idaho June 1, 1961

ABSTRACT

A nuclear incident involving an enriched uranium solution occurred in a first cycle product evaporator at the Idaho Chemical Processing Plant, National Reactor Testing Station, at approximately 0950 on January 25, 1961. The reaction resulted from an accidental air lifting of a quantity of solution from a geometrically safe region of the evaporator into the 24-inch diameter vapor disengaging chamber. An energy release of 20 megawatt seconds (6×10^{17} fissions) apparently occurred as a single burst. No significant radiation exposure to personnel, contamination of facilities or environs, property damage, or product loss resulted. Details of events leading to the incident, recommendations and corrective measures are reported.

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NUCLEAR INCIDENT AT THE IDAHO CHEMICAL PROCESSING PLANT JANUARY 25, 1961

A. SUMMARY

At approximately 0950 Wednesday, January 25, 1961, a nuclear excursion occurred in a first cycle product evaporator at the Idaho Chemical Processing Plant (ICPP), National Reactor Testing Station. Radiation alarms were actuated by the prompt gamma radiation produced at the time of the nuclear excursion and from radioactive gas subsequently released through the exhaust stack to the atmosphere. All personnel, including construction workers, in the ICPP area evacuated to a region approximately 500 feet west of the Process Building, where the radiation level was determined to be less than one mr/hour.

Available evidence indicates that criticality resulted from the accidental lifting of a solution of enriched uranyl nitrate (200 gU/liter) from the lower, geometrically safe section of the evaporator into the upper, 24-inch diameter, critically unsafe, vapor-disengagement section. The most likely cause of the lift appears to have been a sudden burst of air inadvertently introduced into the bottom of the evaporator. The mechanism by which this could have occurred is well defined; however, no definite proof either from instrument records or direct testimony has been obtained to make this premise conclusive.

Of the 251 individuals present in the ICPP area at the time of the incident, none received significant radiation exposure. The highest exposure as determined from film badge readings did not exceed 55 millirem of penetrating radiation. Essentially no beta radiation was detected. No significant neutron exposures or internal contamination from inhalation were found. The absence of significant exposures is attributable to the extensive shielding provided by the process cell in which the event took place and the control of the fission gases by the equipment.

B. FINDINGS

Operational Background

On January 20, 1961, the ICPP (Appendix I) began processing highly enriched uranium-aluminum fuels, after a lengthy shutdown. Initial extraction was of salvage material from previous runs. Simultaneous with the startup of the extraction equipment, dissolution of ETR fuel was started on a planned 20-day run. This was the first hot processing in the equipment involved in the incident in approximately 12 months. Previously, a cold run of 12 days' duration had been made using synthetic solutions. Some unirradiated enriched uranium scrap was also processed during this time for check-out of equipment modifications.

Two process cells, G and H, are routinely involved in dissolution and first cycle solvent extraction. Equipment in G-Cell includes two continuous dissolvers, two input measurement tanks, a feed adjustment tank, a feed evaporator, a waste evaporator, two waste collection tanks, a rework tank, and the first cycle A (extraction) column. H-Cell equipment

includes the first cycle B (scrub), C (strip), and D (solvent wash) columns, a steam stripper, three mixer-settlers for solvent clean-up, a solvent storage tank, a scrub collection tank and the first cycle product evaporator.

The nuclear excursion occurred in the product evaporator (H-110), which, together with associated equipment, is illustrated in Figures 1 and 2. Photographs of the evaporator, taken during the installation of the equipment in H-Cell, are shown in Figure 3. The uranium shown entering from H-103 in Figure 1 is contained in an aqueous solution recovered from the first cycle strip column. The normal evaporator feed concentration is four grams of uranium per liter; however, at the time of the incident the actual concentration was approximately 0.1 gram per liter.

The uranium solution passes downward through the 1D solvent wash column (H-105), in which it is contacted with fresh extraction solvent diluent. This packed column was designed to remove most of the dissolved tributyl phosphate extractant (TBP) from the aqueous uranium stream, thereby preventing its accumulation in the product evaporator with several undesirable effects. From the bottom of the 1D column the aqueous uranium solution is air-lifted to the top of the steam stripper (H-109). Steam is admitted to the bottom of the packed section of the steam stripper and passes upward, stripping entrained or dissolved solvent diluent, as well as additional TBP, from the descending liquid before it reaches the evaporator. The overhead steam is condensed and routed to the process equipment waste (PEW) system for re-evaporation and disposal.

The aqueous solution flows by gravity from the steam stripper to approximately the mid-point of the cold leg of the product evaporator, in which it is normally concentrated about 50 fold, to 200 grams per liter. Since, at the time of the incident, the evaporator feed was very dilute, the concentration factor in the H-110 evaporator was on the order of 2000, to give the 200 grams per liter product. Because of the low evaporator product rate, intermittent rather than continuous pumpout had been scheduled. The evaporator product is normally transferred by remote head diaphragm pump PA-238 or PA-239 to either P- or Q-Cell for second cycle extraction or by PA-239 to N-Cell for temporary storage, from which it can go to second cycle extraction. Off-specification product may be transferred to salvage equipment (J-Cell) by pump PA-238 or PA-239 or through steam jet H-526.

As indicated in Figures 1 and 2, H-110 is a continuous thermosiphon evaporator. Feed to the evaporator mixes with material already concentrated and application of heat to the tube bundle vaporizes part of the liquid in the tubes, discharging a mixture of liquid and vapor into the 2-foot diameter vapor-disengaging space atop the cold leg. Entry to the disengaging space is tangential as an aid in vapor-disengagement. The lower part of the vapor-disengaging space, immediately above the 4 7/8 inch (inside) diameter cold leg, contains an anti-swirl baffle consisting of two crossed plates set on edge, one inch high by eight inches long. Steam from the top of the evaporator flows to the bottom of H-109 stripper; additional steam is introduced in H-109 to give the desired vapor-liquid ratio.

It was recognized during design that the two foot diameter vapor-disengaging space was not geometrically safe. As a safety measure, a 1 1/2 inch diameter overflow line was provided below the two-foot diameter section. Overflow material is collected in two geometrically safe vessels (H-111 and H-112). Should these vessels fill, additional material overflows to the cell floor rather than being allowed to back up into the critically unsafe expanded section of the evaporator.

Pumps PA-238 and PA-239 have never performed as desired. Considerable maintenance has been required due to erratic pumping rates. The decontamination system (Figure 4 and vessel and piping at top of Figure 1) was installed as a means of purging the pumps and piping upon shutdown. It has also proved useful for opening plugged lines, cleaning out the pump check valves, etc., in order to keep the pumps working.

The decontamination system located in the process makeup area (Figure 4), consists of a 10-gallon tank with sight gauge, funnel for water addition, air supply lines, pressure gauges, and associated valving. The usual procedure for water flushing with this equipment is to introduce a quantity of water into the tank, build up the pressure in the tank with air to 20-50 psig, and discharge the water to the decontamination lines by opening appropriate valves. This procedure frequently allows air to enter the process line after the water is discharged. In steam flushing, 35 psig steam is introduced through a manifold bypassing the 10-gallon vessel.

Events Leading to the Incident

During the 0000 to 0800 shift on January 25, the specific gravity of the solution in the evaporator approached 1.28, indicating a uranium concentration of approximately 200 grams per liter and an attempt was made to pump material from H-110 to N-Cell. Pump PA-239 required for transferring this material to storage (N-Cell) was started but failed to deliver. An attempt was made to clear the lines. This included flushing the lines with water and air, using the decontamination equipment. The discharge line was apparently cleared as noted by a response on the liquid level recorder on the storage vessel in N-Cell. Also, the pump suction line from H-110 to PA-239 was air-purged at this time. The pump still would not deliver. All attempts to start the pumpout of H-110 to N-Cell failed. During this period the columns were on recycle feed and there was only very slight buildup of uranium in the H-110 evaporator.

The day shift continued in the effort to make pump PA-239 operate. Several attempts to pump evaporator product to N-Cell failed. In order to preclude further dilution of first cycle product in N-Cell by the water used for flushing and testing, pumping to J-Cell was attempted. Also, instrument response to transfers to J-Cell is more rapid than to N-Cell because of shorter piping runs. In addition, differences in sizes of the receiving vessels make incremental volume changes in J-Cell more discernable. The discharge lines of both pumps to J-Cell were flushed with water. The suction lines were purged from the pumps back into the evaporator with steam. An attempt to pump through PA-239 failed. Flow was established through PA-238 for a short period, indicating that that pump was operating satisfactorily and the suction line from the evaporator

was not plugged. Consequently, either the discharge line from PA-239 was plugged or that pump was not functioning properly. However, since PA-238 discharge is not routed to N-Cell, further attempts were made to make PA-239 operate.

Immediately prior to the time of the incident, an operator in the process makeup area was instructed to put a "couple of funnel fulls" (approximately four liters) of water into the 10-gallon flushing tank and use 40 psig air pressure to move it into the decontamination line. The process operator at the control panel, one level below the makeup area, opened PA-239 discharge valve to J-Cell. A buildup of liquid in the J-Cell storage vessel, as reflected by the liquid level recorder, equivalent to approximately four liters was noted. The operator in the makeup area was instructed, from the operating corridor through a pipe chase, to close the valve from the decontamination tank. A reply was noted; however, the words were not distinguishable. (The makeup operator later stated that the pressure in the decontamination tank had dropped to 5 or 10 psig before he closed the valve.) Pump PA-239 was then started and momentarily ran with the suction valve from H-110 closed. The process operator "had hardly taken his hand off the valve (PA-239 suction) control" after opening this valve when radiation alarms sounded throughout the plant. Immediately the operations superintendent went to the Health-Physics office (some 50 feet away) and found the master panel for radiation detectors almost completely covered with alarm lights. Without further delay he proceeded to the end of the corridor and actuated the plant evacuation alarm within no more than a minute after the area alarms had sounded.

The Nuclear Incident

From a detailed examination of pertinent instrument charts it appears that H-110 contained approximately 40 liters of uranyl nitrate solution immediately prior to the incident. The estimated concentration was 200 grams of uranium per liter, 90% enriched in U-235. At the time of the incident both the liquid level and the density instrument traces for H-110 showed an abrupt change indicative of an air surge, a pressure transient, or both (Figures 5 and 6). A multipoint temperature recorder on a three-minute scan interval indicated a rise of six degrees centigrade in the H-110 vapor head and a rise of twenty-eight degrees in the condensate downstream of the condenser (H-300), the latter returning to normal within six minutes (two recorder printout cycles) as indicated in Figure 7. The H-110 pressure recorder was heavily snubbed and did not respond (Figure 8).

The calculation of liquid volume around the evaporator and the overflow tanks before and after the incident indicated an increase of at least 11 liters (Appendix II). This liquid could have come from the decontamination vessel and associated lines and/or the steam stripper. It is not uncommon for the stripper to "dump" about 10 liters when disturbed. Also, it appeared from H-110 instrumentation charts that the uranium content in the evaporator was reduced by approximately two kg during the incident. After shutdown and flushing of the evaporator system a material balance well within measurement uncertainty was obtained.

An over-all uranium material balance for the plant based on measured input to process and material collected and measured after shutdown indicated that no appreciable amounts of uranium were spread through the system.

The magnitude of the excursion has been estimated at 6×10^{17} fissions (20 megawatt seconds) with an error not to exceed 25% (Appendix III). This estimate was based primarily upon radiochemical analyses for Mo-99 and Ce-143 in samples of the solution involved in the incident. Integrated thermal neutron flux, as determined by scintillation spectrometer counting of activated indium foils obtained from various operating areas in the plant, ranged from below the limits of detection (5×10^{6} n/cm^2) to 3.3×10^{7} n/cm^2. The integrated thermal neutron flux at the evaporator was determined from activation analyses to have been 2.9×10^{12} n/cm^2.

From detailed examination of instrument charts and radiation detector traces it is believed that the nuclear excursion was of short duration (Appendix IV). No evidence exists that any gaseous or air-borne contamination entered the operating areas of the process building. Since sounding of the alarms was almost simultaneous throughout the process building, prompt gamma radiation from the excursion appears to have actuated the majority of the alarms. The fission gases were conducted through the vessel off-gas system and cell vent tunnel to the atmosphere via the plant exhaust stack. No hazard to personnel or environment is considered to have resulted from this release (Appendix V).

Post Incident Activities

The evacuation of the building and outside working areas by Phillips and other personnel was orderly (Appendix VI). Evacuation was complete within five to seven minutes, except for one construction employee who was working near an air compressor and did not hear the evacuation signal. This man left the area about ten minutes after evacuation had been initiated. (His film badge indicated no detectable exposure.)

Two health physicists made a final check of the process building before evacuating. No radiation above normal background was detected with survey instruments having a range of 0-250 mr/hr.

Shortly after roll calls had been completed at the assembly area and it had been ascertained that all personnel had evacuated, a team of operating and health physics personnel re-entered the plant and shutdown all process equipment. At the same time processing systems were checked for possible indication of a source of activity release. The first re-entry was made approximately twenty minutes after the evacuation had been signalled.

A filter from the stack gas monitor was recovered for analysis and the presence of Cs-138 (T 1/2 = 32 minutes) was reported at about 1035, approximately the same time that Cs-138 was identified from a high volume air sampler filter in the Central Facilities Area, some three miles south of ICPP. Thus, the first proof of a criticality incident was received approximately forty-five minutes after the incident. Observations of

process instrument records and knowledge of the location of significant quantities of uranium indicated that H-Cell was the most likely site of the excursion (Appendix VII). After the identification of Cs-138, the solution in the H-110 evaporator was transferred to J-Cell where it was sampled and subsequently shown to contain short-lived fission products.

Health Physics' approval for the return of all employees, based upon the absence of external contamination and of radiation levels in excess of normal, was given at approximately 1345. An examination of instrument charts and a review of potentially hazardous situations resulting from the incident were made prior to allowing the full plant staff to return to the ICPP area at 1445.

Radiation Exposures

Shortly after evacuation, a general check of pocket dosimeters worn by process operating personnel indicated that radiation exposures were probably less than daily administrative limits (50 millirem). Following identification of short-lived fission products, selected film badges considered representative of the entire plant area were scanned for neutron activation of indium and analyzed for beta and gamma exposure. Subsequent analyses of badges from 65 individuals indicated a maximum exposure of 55 mrem gamma, 0 beta (Appendix VIII). The maximum thermal neutron exposure detected in the 65 badges analyzed was less than 10 millirem. Analyses of nuclear accident dosimeters indicated that there was negligible fast neutron flux associated with personnel exposures.

Selected personnel were subjected to bioassays and whole body counting. No blood sodium activation was found, further indicating that there were no significant neutron exposures. Whole body counting supported this conclusion and indicated that there were no significant internal exposures.

C. CONCLUSIONS

1. A nuclear excursion of approximately 6×10^{17} fissions occurred in a first cycle product evaporator at about 0950 on January 25, 1961. Available evidence indicates that criticality resulted from accidental lifting of uranyl nitrate solution containing about 200 grams of uranium per liter (90 percent U-235) from the lower geometrically safe section of the evaporator into the upper 24-inch diameter critically unsafe vapor-disengagement section. The cause of the lifting is not certain; however, the sequence of events which occurred just prior to the incident suggests that a burst of air was inadvertently introduced into the bottom of the evaporator. Since testimony of operating personnel did not completely support this proposed mechanism, other theories of less credibility were evaluated in the light of available evidence.

2. Communications, both with respect to detail of instructions and voice contact between individuals, was inadequate and may have contributed to the incident.

3. Personnel response to radiation alarms and the evacuation signal

was prompt and orderly. Re-entry into the plant and identification of the location and nature of the incident was performed with efficiency and dispatch.

4. The use of direct reading pocket dosimeters proved to be an effective tool for immediate estimation of personnel exposures.

5. No costs were incurred which were attributable to contamination cleanup, product recovery, or equipment repair as a result of the incident.

D. RECOMMENDATIONS

The Committee recognizes that absolute protection against criticality incidents in processing enriched uranium fuels is neither possible nor practical. Safeguard reviews of process equipment at the ICPP have been standard practice since first startup in March of 1953. It is worth noting that in 1955 the criticality hazard presented by the product evaporator (H-110) was considered by the ICPP Safeguard Committee. Minutes of meetings on this subject, report that that committee felt the likelihood of operational circumstances resulting in a criticality incident in the equipment was small. Also, it was concluded that if criticality should occur, it would probably be accompanied by limited physical damage. The correctness of both of these predictions in a large measure has been demonstrated. Also, the Safeguard Committee had previously considered the use of air and steam under pressure in cleaning process lines. Administrative controls had been established to cover this operation. Operator training and orientation has been a continuing concern. Standard Operating Procedures and Operations Instruction Bulletins have been prepared for every major operation. Startup after appreciable down time is preceded by cold runs for equipment check out and to re-orient personnel with procedures and equipment.

The fact that criticality did occur, however, indicates that a strengthening of certain areas of safeguards is needed. The following recommendations are directed toward this end:

1. Limitations should be placed on the amount and manner in which expansible fluids are introduced into any part of the process system. Contingencies associated with such usage should be carefully evaluated and supplemental control systems devised.

2. Communication between safeguard committees, operating supervisors and operating personnel should be improved. Suggestions for consideration are:

   a. codification of safeguard committees' actions,

   b. chairmanship of the ICPP Safeguard Committee be a full-time assignment, and

   c. improve direction and follow-up in operator training, particularly with respect to infrequent and emergency operations.

3. Engineering safety reviews should always consider complete systems rather than single items of equipment. In particular, the potential for maloperation of misuse of any part of the system should be considered.

4. Consideration should be given to the redesign of the evaporator system to increase physical protection against criticality.

5. Attention should be given to improving intercommunication between work areas both in person-to-person communications and data telemetering.

6. As a matter of policy, all personnel badge dosimeters should be collected and analyzed immediately following a criticality incident or a release of appreciable quantities of radioactive material to operating areas.

E. ACTIONS ON RECOMMENDATIONS

Corrective measures instituted are as follows:

1. The use of steam or compressed air to clear obstructed process lines has been forbidden except with the specific approval of the Assistant Manager for Operations. Only water delivered by low volume, controlled pressure pump is permitted for this service. The decontamination system piping has been modified to reduce the invitation for unauthorized use and operating procedures have been revised accordingly.

2. Each shift has been supplemented by additional technical personnel to provide increased surveillance over processing activities.

3. Additional instrumentation has been added to both the evaporator (H-110) and the decontamination system. Pressure readout for the decontamination system has been extended to the operating corridor so that the process operator will have complete knowledge of the system.

4. In addition to installing a new tube bundle on the evaporator, it is planned to install fixed nuclear poison in the 24-inch disengaging head. Also, it is planned to lower the tube bundle to permit flooded operation and hence smoother control.

5. An experimental mixer-settler unit will be installed to remove residual TBP from the stream entering the evaporator. This unit will replace both the 1D scrub column and the steam stripper (H-109). This is expected to reduce pump maintenance problems as well as reduce the potential for a critical incident in the H-110 evaporator system.

6. Air lifts have proven satisfactory in many parts of the plant. Additional installations will be made either to replace or serve as alternate means of transfer for the diaphragm pumps (PA-238 and PA-239).

7. Cell floor drain valves will be sealed and transfer jets installed to preclude leakage from process cells into the waste system.

8. Positive means of communication between operating areas will be established with emphasis on instrument readout in the main operating area.

9. Operator training and review has been given increased emphasis. Particular attention will be given nonroutine and emergency procedures.

In order to complete the processing run that was in progress at the time of the incident, a number of safety measures were initiated. These included administrative restrictions, some piping and instrument modifications to the decontamination system, and flowsheet changes both to poison the solution with boron and limit the uranium concentration in the evaporator. However, during restart of the plant after the incident, leakage from the evaporator tube bundle was detected and shutdown for repairs became necessary. Leakage was due to stress corrosion at the tube sheet joints. This was apparently due to excessive cold work during fabrication. There is no evidence that pressures from the nuclear reaction contributed to the leakage.

F. ACKNOWLEDGEMENT

The Committee wishes to express its appreciation to the personnel of the Idaho Operations Office and the Phillips Petroleum Company for their cooperation and assistance during the Committee's investigation. We are particularly grateful to the ICPP Safeguard Committee for analyses of various possible reaction mechanisms; to W. H. Burgus of Phillips for the material presented in Appendix III; to D. G. Olson of Phillips for the neutron field intensity data in Appendix IV; to A. L. Ayers of Phillips for the material balance data in Appendix II; to W. G. Morrison of Phillips for the criticality calculations; and to personnel of the IDO Health and Safety Division for the environmental data presented in Appendix V and the personnel exposure data in Appendix VIII.

G. SIGNATURE OF THE COMMITTEE

This report represents the combined efforts of the members of the investigating committee and the findings, conclusions and recommendations are concurred in by the members as witnessed by their signatures below:

R. C. Paulus

R. C. Paulus, Chairman
Inspector
Licensee Compliance Division
Idaho Operations Office, USAEC

A. O. Dodd

A. O. Dodd
Nuclear Safety Engineer
Health and Safety Division
Idaho Operations Office, USAEC

F. H. Tingey

F. H. Tingey
Division Technical Consultant
Phillips Petroleum Company
Atomic Energy Division

K. K. Kennedy

K. K. Kennedy
Director, Chemical Processing
and Development Division
Idaho Operations Office, USAEC

F. Morgan Warzel

F. Morgan Warzel
Manager, Process Development Branch
Phillips Petroleum Company
Atomic Energy Division

APPENDIX I

OPERATIONAL ASPECTS

Introduction

The Idaho Chemical Processing Plant was constructed in 1950-51 to recover enriched uranium from aluminum-uranium alloy fuel assemblies and unalloyed fuel pins. The plant is located on a 97-acre fenced plot in the south central part of the National Reactor Testing Station. Original facilities utilized batch dissolvers and three hexone extraction cycles for the processing operations. In 1954-55, continuous dissolvers and tributyl phosphate first cycle extraction equipment were installed in spare cells G and H. At about the same time, facilities were added in other cells for processing zirconium and stainless steel fuel assemblies and for isotope recovery (RaLa). Each addition tied into existing facilities and increased the versatility of the overall processing complex.

The continuous dissolution system for aluminum fuels contains one solvent extraction cycle as part of the headend. Partially decontaminated product solution from the headends is concentrated and stored until enough is accumulated to permit most efficient operation of second and third extraction cycles. The accident herein described involved equipment used in the concentration of the first extraction cycle product and the associated valving and piping required to transfer the concentrated solution to temporary storage. Figure 9 shows the relative positions of the cells and the operations carried out in each cell. A section view of the main processing building (CPP-601) is shown in Figure 10.

Criticality Control Practices

Geometrical Control

Wherever feasible and practical, equipment that handles significant quantities or concentrations of uranium is built to such dimensions that is is impossible to reach a critical configuration.

Concentration Control

Aluminum alloy fuels processed at the ICPP have an indirect concentration control for the first few processing steps provided by the presence of the aluminum, i.e., upon concentration of the aluminum-uranium fuel solution, aluminum nitrate will crystallize and prevent further concentration long before the minimum critical uranium concentration is reached.

Concentration control is used in certain areas, such as salvage operations, where precise analytical determinations of the uranium content are possible and the possibility of precipitation or other chemical reaction is very remote. Wherever possible, when relatively pure uranium solutions are involved, safety is further assured by the additional backup of mass control.

Mass Control

In certain processing and salvage operations it is necessary to handle pure uranium solutions, solutions partially separated from diluent metals or solutions in which there may be nonhomogeneity, in equipment that is not geometrically safe. In these cases the maximum quantity of uranium handled at any one time is limited.

Administrative Control

In order to minimize the possibility of human error, numerous procedures have been devised so that decisions which could lead to loss of uranium or to a dangerous condition are checked by two or more persons. Detailed run sheets providing check points and guide limits require the approval of the shift foreman at critical steps. For salvage operations and other transfers not in the normal processing chain, special detailed procedures are provided. The foreman is required to check all analytical determinations and approve all movements of uranium-bearing solutions. Process alarms are used throughout the system to warn of abnormal conditions of specific gravity, solution flow, tank volumes, etc.

Solution transfers are accomplished by steam jet ejectors, pumps, air lifts and gravity. In cases where uranium might be misdirected from normal channels, pertinent controls are red. If a criticality incident is possible in any equipment involved in the transfer, the control is orange. As further assurance that the consequences of such a transfer have been considered, such normally used valves are locked so that the operator is required to get permission and the specific key from supervision before the transfer can be made. Some such valves, used only during system decontamination are sealed; the seals are broken only by express permission of supervision.

APPENDIX II

MATERIAL BALANCE

To ensure that uranium was neither lost from the system nor residing in unidentified locations within the processing system, solutions were accumulated in vessels equipped with samplers and accurate volume measurement instrumentation. Material balances were made around both the evaporator system and the total process. Material balance data are given in Tables 1 and 2 below.

TABLE 1

Evaporator System Material Balance

(Concentrations Estimated From Instrument Specific Gravity)

	H-110		H-111, 112		Total	
	Volume Liters	Uranium Kg	Volume Liters	Uranium Kg	Volume Liters	Uranium Kg
Before Incident	40.1	8.0	13.8	0.5	53.9	8.5
Immediately After Incident	40.9	4.7	23.7	1.9	64.6	6.6
After Shutdown	56.8	6.7	24.0	1.9	80.8	8.6

TABLE 2

Estimated Material Balance of Uranium in Processing Equipment
After the Incident as Related to Plant Input

			Uranium-Kg
Input Analyses:	Feed		38.4
	N-Cell Recycle		4.3
	Total Plant Input		42.7
Post Incident Inventory:		Vessel	
		G-115-116	0.2
		J-102	3.9
		J-105	8.6
		G-108	0.7
		G-106	0.2
		G-115	0.3
		G-116	negligible
		H-110	0.7
		D-106	0.3
		D-156	0.8
		D-155	0.1
		D-152	0.1
		D-102	0.1
		WG-101	1.3
		WH-100	negligible
		WG-100	negligible
		WH-101	negligible
		N-100) N-130)	26.3*
	Total Uranium in Plant		43.6

*Estimated from volume and specific gravity of solution in N-100 and N-130

APPENDIX III

MAGNITUDE OF THE EXCURSION

The radiochemical determination of the number of fissions involved in the nuclear excursion is, in principle, quite simple and straightforward. From an aliquot of the entire volume of solution containing all the non-volatile fission products, a single fission product nuclide is isolated and its concentration at the time of the incident determined (in terms of atoms per unit volume of solution). The number of atoms is obtained by radiochemical separation and purification of a selected nuclide, followed by determination of its absolute disintegration rate. This involves an accurate knowledge of the abundance (branching ratio) of the particular radiation chosen for counting. It further requires accurate knowledge of the efficiency of the counter used, and an accurate knowledge of the radioactive decay constant. Corrections are made for losses encountered in the isolation and purification procedures and for radioactive decay between the time of the incident and the time of counting. From the number of atoms per unit volume of solution and the known fission yield of the chosen nuclide, the corresponding number of fissions in the aliquot and consequently, the total number of fissions associated with the incident may be calculated. The preceding assumes that (1) the entire solution is uniform in composition so that the aliquot withdrawn for analysis is representative, (2) the volume of the solution is known, and (3) there has been no loss of the specific fission product isolated.

Following the ICPP nuclear excursion in H-Cell, all of the solutions which possibly could have been involved were transferred to banks J-102 and J-105. These vessels are in reality sets of four and nine infinitely safe cylinders, each set being manifolded at the bottom. Sparging can be accomplished in each cylinder but circulation between cylinders in each bank is not possible. However, the vessels of the J-105 bank can be filled only by way of the J-102 bank through the manifold at the bottom of the J-102 bank. Consequently, mixing is accomplished in the transfer between banks. Samples can be obtained only from one vessel in each bank. Prior to sampling, each bank was sparged for 30 minutes to ensure uniform mixing of contents at least within each tube. Duplicate samples were then withdrawn by use of the standard ICPP sample recirculation technique which provides for recirculation in this instance of about 100 milliliters per minute. The first samples withdrawn had approximately 15 minutes of recirculation and the second samples had approximately 10 minutes additional recirculation. That the contents within the two sampled cylinders were quite uniform is shown by comparison of the total uranium contents of the samples and by comparison of the gross beta activities of small but equal aliquots evaporated and beta-counted. Comparisons are shown in Table 3.

TABLE 3

Uniformity of Solutions Analyzed

Sample	Vessel J-102	Vessel J-105
Sample Bottle 1	U = 54.63 g/l	U = 58.86 g/l
Sample Bottle 2	U = 54.63 g/l	U = 59.07 g/l
Sample Bottle 1	Gross Beta = 5.44×10^5 c/m	5.08×10^5 c/m
Sample Bottle 2	Gross Beta = 5.39×10^5 c/m	5.26×10^5 c/m

Samples were all crystal clear, quite acidic, and homogeneous. No solids were visually discernible.

The measured total volumes of solutions after transfer were 77 liters for J-102 and 153 liters for J-105. The errors in measurement are conservatively estimated to be ± 1 liter in each case. In view of other larger errors involved in the calculations, the small error associated in total volume measurement is not significant.

Because of early availability of representative samples (within 8 hours after the incident) it was possible to select several different fission product nuclides for measurement of the total number of fissions. In selecting these it was necessary to choose rather short-lived nuclides which would be indicative only of fissions which occurred in the incident and which could not have remained (after first cycle decontamination) from fission which had occurred during reactor operation. Consideration also had to be given to the question of measurement of a chosen short-lived nuclide over a possible high background of a longer-lived isotope of the same element which might have come through first cycle extraction or possibly had remained as contamination in the vessels in which the excursion occurred or to which solutions were later transferred.

The most satisfactory fission product, and one generally used for measurement of total fissions, is Mo-99 (T 1/2 = 67 h). It has no rare gas ancestory, has a very well known fission yield and decay scheme, and has no longer-lived isotopes to interfere with its measurement. Mo-99 was therefore chosen for measurement. Values of total fission based on Mo-99 are probably the most accurate of all values obtained.

Notable among long-lived fission nuclides which are not well removed in the first cycle of the process is Zr-95 (T 1/2 = 65 d). A generally used nuclide for measurement of a relatively low number of fissions is Zr-97 (T 1/2 = 17 h). The measurement of this nuclide over a relatively high Zr-95 background is possible however, because the Zr-97 has a 60 minute half-life Nb-97 daughter which can be extracted from Zr-97 and measured independently. Zr-95 has a Nb-95 daughter also but it has a half-life of 35 days. Niobium-95 was expected to contribute only slightly to the total Nb fraction activity when the differences in all parent and

daughter half-lives were considered. (Scintillation spectrometric analyses verified this expectation.) Because considerable quantities of rare gas fission products escaped in the incident and because Zr-97 is known to have no rare gas parentage, it is ideal from this standpoint. Its decay scheme and the decay scheme of its daughter are well known. Likewise, its fission yield is well established. A disadvantage, however, is that daughter extractions are necessary thus introducing further possible error. A further disadvantage is that zirconium tends to hydrolyze and is often partially lost by deposition on vessel walls. Nonetheless, Zr-97 analyses were run.

Other nuclides which could also serve for measurement of total fissions are Ba-140, Ba-139, Sr-91, Sr-92 and Ce-143. For each of these nuclides, decay of a rare gas ancestor provides a portion of the total fission yield. Ordinarily the use of these nuclides for measurement of fissions is justified in situations where gaseous parents cannot and do not escape (e.g., as in the case of a solid uranium fuel, tightly clad). However, in cases where all or a portion of the gaseous ancestor may escape, measurements based on such nuclides may be expected to yield a low result. The degree of error is clearly dependent on the fraction of the chain which comes via a rare gas and upon the fraction of rare gas which may have escaped. In the ICPP incident extensive escape of rare gases was expected since a boiling solution was involved. That this occurred is supported by the observation of short-lived rare gas decay products in the ICPP stack monitor. Despite the expectation that measurements based on Sr-91, Sr-92, Ba-139, Ba-140 were expected to yield low results, these measurements were made to provide order of magnitude confirmation of the number of fissions involved. Table 4 lists the available pertinent information on rare gas ancestory of these measured nuclides. The data in this table may be used to calculate maximum expected errors if all the gaseous parents were to escape. It should be noted that since only a very small fraction of Ce-143 comes through a gaseous parent, only a very small error would be involved in gas escape in this case. Therefore, fission calculations made from Ce-143 measurements may be considered very reliable in this respect.

TABLE 4

Rare Gas Precursors of Selected Fission Product Nuclides

Nuclide	Precursor	Fission Yield at Rare Gas In the Decay Chain	Total Fission Yield at the Measured Nuclide	% of Measured Nuclide Coming Through Rare Gas
Sr^{91}(9.7 hr)	Kr^{91}(10 sec)	3.45%	5.81%	59%
Sr^{92}(2.7 hr)	Kr^{92}(3 sec)	1.87%	5.3 %	35%
Ba^{139}(83 m)	Xe^{139}(41 sec)	5.4 %	6.55%	82%
Ba^{140}(12.8 d)	Xe^{140}(16 sec)	3.8 %	6.35%	60%
Ce^{143}(33 hr)	Xe^{143}(1 sec)	0.051%	6.0 %	<1%

The actual fission product analyses were carried out by personnel of the Analytical Section at ICPP. Standard radiochemical procedures were used in making the analyses. Determination of absolute disintegration rates

were made by counting each sample on a scintillation spectrometer to measure a specific gamma ray. All samples were counted through a standard polystyrene absorber (1.25 g/cm^2) to absorb beta rays. The parameters used to convert measured gamma counting rates to absolute disintegration rates were taken from IDO-16408. The values of other constants used in the calculations are given in Table 5. There was available no sound indication that the nuclear excursion continued over a period as long as a few minutes. Therefore, in lieu of any better information, the excursion has been considered as nearly instantaneous, taking 0950 on January 25, 1961, as the time of the incident. Since the shortest half-life isotope measured was 82.9 minutes Ba-139, it is believed that no significant error resulted from assuming a very short burst.

TABLE 5

Values Used in Calculation of Number of Fissions

Nuclide	Half-Life	Energy of γ-ray Measured	Branching Ratio (γ abundance)	Fission Yield
Sr-92	2.6 hr	1.39 Mev	0.90	5.3 %
Sr-91	9.67 hr	0.551 Mev	0.60	5.81 %
Ba-139	82.9 min	0.164 Mev	0.231	6.55 %
Ba-140	12.8 d	0.54 Mev	0.250	6.35 %
Mo-99	66.5 hr	0.75 Mev	0.14	6.06 %
Ce-143	33 hr	0.29 Mev	0.43	6.0 %
Zr-97	17.0 hr	0.66 Mev (Nb-97 daughter)	1.00	5.9 %

As mentioned previously, two samples were withdrawn from each bank (J-102 and J-105). For the Mo-99, Sr-91, Sr-92, and Ce-143 determinations duplicate analyses were run on each sample (four analyses per bank). For the Zr-97, Ba-140 and Ba-139, single analyses of each of the four samples were run. The Zr-97, Mo-99 and half of the Ce-143 analyses were run by one group of individuals. The other half of the Ce-143 analyses, the Ba-140, Ba-139, Sr-91 and Sr-92 analyses were run by a second group of individuals.

Table 6 presents a summary of the results. Values of the number of fissions represented in each bank are the average of the four (or in some cases two) analyses run on samples from that particular bank.

TABLE 6

Summary of Determinations of Total Number of Fissions

Nuclide Measured	Vessel J-102	Vessel J-105	Total Fissions
Mo-99	2.1×10^{17}	3.7×10^{17}	5.8×10^{17}
Ce-143	2.2×10^{17}	4.1×10^{17}	6.3×10^{17}
Zr-97	1.4×10^{17}	2.3×10^{17}	3.7×10^{17}
Sr-92	1.6×10^{17}	3.3×10^{17}	4.9×10^{17}
Sr-91	1.3×10^{17}	2.5×10^{17}	3.8×10^{17}
Ba-140	0.9×10^{17}	1.7×10^{17}	2.6×10^{17}
Ba-139	1.2×10^{17}	2.0×10^{17}	3.2×10^{17}

From preceding discussions on the choice of fission products to measure, it was expected that Mo-99, Ce-143 and Zr-97 would all indicate the same number of fissions unless there had been some zirconium loss. Table 6 shows that indeed the number of fissions based on Zr-97 measurements is low. It is apparent that somewhere Zr-97 had been lost. The number of fissions based on Mo-99 and Ce-143 are considered to be quite accurate and are estimated to be better than $\pm$ 25 per cent on an absolute basis. Together they provide the best estimate of the number of fissions, 6×10^{17} (20 megawatt seconds). As expected, the results based on the strontium and barium isotopes indicate a low number of fissions due to gaseous precursor escape. When the half-lives and fission yields of the rare gas ancestors are considered, these data appear to be quite consistent.

APPENDIX IV

NUCLEAR ASPECTS

Review of plant operations and fuel inventory at the time of the incident led to the early suspicion that a major release of activity or a criticality event had occurred in or near H-Cell. Shortly after plant personnel were permitted to return to the operating areas, a brief visual and photographic inspection of the interior of H-Cell indicated that no apparent physical damage to, nor material loss from, the evaporator system had resulted. These observations, plus the desire to complete the processing run, postponed collection and analysis of metal samples from the system to determine the precise location of the reaction. Subsequently the system was emptied, decontaminated, and metal samples obtained from suspected points where criticality conceivably could have occurred.

Stainless steel nuts were retrieved from the flanges supporting the H-110 disengaging head and the H-110 thermal leg. In addition, a marking tag was removed from the steam stripper (H-109). Gamma spectral analyses and subsequent calculations showed the neutron field intensities as given in Table 7.

TABLE 7

Relative Neutron Activations

Location	Indicating Gammas	
	0.32 mev Cr^{51}	0.82 mev Co^{58}
H-110 Disengaging Head	100 (thermal neutrons)	100 (fast neutrons)
H-110 Thermal Leg	20 " "	7 " "
H-109 Steam Stripper	50 " "	35 " "

An integral neutron flux measurement was made on a portion of the nut removed from the H-110 disengaging head flange. The segment nearest the reaction vessel was cut from the nut, dissolved in aqua regia, and analyzed for Cr^{51} and Co^{58}. The stainless steel was typed by X-ray analysis to be approximately 18% Cr - 10% Ni.

Cross sections used in the calculations were 15 barns for the Cr^{50} (n, γ) Cr^{51} and 91 millibarns for the Ni^{58} (n,p) Co^{58} reactions. It is assumed that the first reaction proceeded predominantly with thermal neutrons while the latter required fast neutrons. The integral neutron flux seen by the nut segment is shown in Table 8.

TABLE 8

Neutron Flux Determinations

Nuclide	nvt
Cr^{51}	2.94×10^{12} thermal n/cm^2
Co^{58}	2.92×10^{12} fast n/cm^2

From these data it appears certain that the reaction did occur, as was originally postulated, in the vapor disengaging head of H-110 evaporator.

Admittedly a descriptive reconstruction of the nuclear event must be consistent with the general facts comprising the over-all incident. The pertinent facts in this regard are: (1) approximately 40 liters of uranyl nitrate solution were in the evaporator system H-110; (2) the solution was at a concentration of about 200 grams of uranium (about 90% U-235) per liter; (3) during the incident H-110 showed no significant solution gain, while the overflow vessels H-111 and H-112 gained about 10 liters total; (4) with the gain in solution there was no measurable gain in uranium in the evaporator system; (5) H-110 liquid level and density instrument traces (Figures 5 and 6) show an abrupt disturbance at the time of the incident indicative of an air surge and/or pressure transient; (6) the solution temperature throughout the evaporator was at or near the boiling point; (7) significant temperature changes occurred in the evaporator system though the recorder print-out interval (three minutes) was large enough to introduce uncertainty as to maxima (Figure 7); (8) the most accessible vessel having critical geometry was the 24-inch diameter vapor disengaging head; (9) radiation alarms sounded coincident with the opening of valve RVC-28H; (10) the valve provided a connection between the cold leg of the evaporator and an air pressure system which could have been charged with water; (11) gamma-sensitive radiation monitors recorded responses indicative of a prompt burst, while neutron-sensitive devices showed absorption activation; (12) there was no measurable beta dose to either fixed or personnel film dosimeters in the plant operating areas; (13) stack air samples and evaporator system solution samples showed fresh fission products; (14) neutron activation of metal samples from the evaporator system verified the original assumption that the excursion occurred in the 24-inch diameter vapor disengaging vessel of the H-110 system.

Reference to Figure 11 indicates that for the stated fuel concentration, mass, and solution volume, criticality could occur in the 24-inch diameter vapor disengaging vessel of the H-110 system. It would appear quite unlikely that the entire 40 liters in the evaporator just before the excursion could have been transferred almost instantly into that particular part of the system. (Testimony indicated that radiation alarms sounded virtually at the same instant in which valve RVC-28H was actuated.)

An alternative hypothesis could be that 10 to 20 liters from the cold leg were suddenly forced up into the expanded portion, producing a geyser-

like effect momentarily. Assuming a somewhat spherical geometry of 10 to 12 inches diameter and a U-235 concentration of 125 to 150 grams per liter, criticality could have occurred with as little as 10 to 20 liters. Experimental data* indicate the minimum critical mass for a homogeneous solution of uranyl fluoride, (water reflected, spherical geometry) to be 800 grams of U-235. This would have a diameter of about nine inches, a volume of 5.90 liters, an H/U-235 ratio of 185, and a U-235 concentration of 135 grams per liter. This reference also shows the minimum critical mass for an unreflected sphere to be 2.13 kg U-235. This system would have an H/U-235 ratio of 200, a U-235 concentration of about 125 grams per liter, a diameter of 12.6 inches, and a volume of 17.2 liters. Nuclear parameters are considered almost identical for uranyl nitrate and uranyl fluoride solutions.

Another conjecture as to the geometry of the reacting system depicts the hot leg of the evaporator bumping a quantity of solution over into the vapor disengaging head, as a result of a perturbation associated with the opening of the valve. Such a sudden transfer of solution from a geometrically safe region of the evaporator tangentially into the 24-inch diameter vessel conceivably could have formed an annular ring or vortex configuration capable of transient criticality.

Along with these conjectures, it is possible that shortly before the incident the steam stripper dumped feed solution into the upper region of the evaporator cold leg, as a result of disturbances from line purging operations. Consequently, when material was abruptly lifted into the 24-inch diameter vessel, it may have been momentarily at a concentration considerably less than the 200 gU/liter indicated from instrument readings. It can be seen from Table 9 that approximately 40 liters of solution can become critical in a 24-inch diameter cylindrical vessel even at concentrations as low as 125 grams of U-235 per liter. However, if the configuration were a sphere or a vortex, then criticality could be achieved with a volume much less than 40 liters at similar uranium concentrations. These conditions, however, are postulated here more to indicate the variety of conceivable critical configurations and concentrations than to indicate any relative probability of one with respect to another.

* K-1380 Studies in Nuclear Safety

TABLE 9

Calculated Critical Conditions For Uranium Solutions In Bare 24-Inch Diameter Right Cylindrical Stainless Steel Vessels By The Modified One-Group Method

Uranium-235 Concentration (g/l)	H : U-235 Ratios	Reflector Savings (cm)	Critical Conditions		
			Height (cm)	Volume (liters)	Mass (kg U-235)
740	30	2.7	14.1	41.3	30.6
580	40	2.8	14.1	41.0	23.8
475	50	2.9	14.0	40.7	19.3
325	80	3.1	13.9	40.7	13.2
252	100	3.2	14.0	40.7	10.3
158	160	3.4	14.3	41.7	6.6
128	200	3.5	14.5	42.4	5.4
85	300	3.8	15.5	45.3	3.9
51	500	4.0	18.3	53.5	2.7
34.5	750	4.2	23.4	68.2	2.4
26.0	1000	4.3	31.6	92.1	2.4
17.6	1500	4.5	118	345	6.1

Traces on recording radiation monitoring instruments have been examined in detail in a further effort to define the nuclear characteristics of the reaction. Records are available from three types of instruments: (1) Geiger-Muller tube (comprising the continuous air monitors); (2) ionization chamber (general area radiation monitors); (3) scintillator crystal detector (also as general area monitors). None of the instruments in the building where the excursion occurred showed any evidence of air-borne activity in the plant operating area. This is supported by the absence of beta exposures on film dosimeters, either those worn by personnel or those at fixed locations throughout the operating areas. The only instrument in the ICPP area which collected air-borne activity was at Building CPP-603, located about 2500 feet south of Building CPP-601 where excursion occurred (Figure 12). Building CPP-603 was in line with movement of the radioactive cloud emitted from the ICPP exhaust stack.

Figure 13 shows a typical response of the continuous air monitors to direct radiation rather than to air-borne radioactivity. This instrument was about sixty feet straight-line distance from the reaction, with approximately twelve feet of concrete intervening. The following interpretation appears plausible: the first abrupt rise, from about 36 to 54 percent of chart range, and return to 36 percent, was the response to prompt gammas from the excursion; a few seconds later an abrupt rise to off-scale and change-of-scale resulted from movement of fission gases through the cell vent corridor and vessel off-gas system at high velocity, passing within about fifteen feet of the monitor and with only two or

three feet of concrete shielding; immediately the radiation level dropped to a point on the chart which, for the less sensitive (10X) scale, was about equal to background on the previous (2X) scale; the following trace rise was the response to the radioactive gas emission from the ICPP main exhaust stack, which this instrument readily saw because of its location on the top floor of Building CPP-601, with only transite walls and roofing separating it from the outside--(this same cloud response was shown by all other instruments of similar orientation; Figure 14); the trace returned to normal background for 10X scale of sensitivity after an interval of five minutes from the first response and remained at this level until manually reset at about 1020 hours.

Figure 15 shows the response of a scintillator-photomultiplier detector located on the operating corridor level about seventy-five feet line-of-sight away from the reaction and having at least twelve feet of concrete intervening.

The trace on this chart could be interpreted as follows: the abrupt rise off-scale was due to prompt fission gammas and decay gammas from fission gases released; the precipitous decline began to retard between 40 and 30 percent of chart as the fission gases cleared the cell vent and vessel off-gas systems, moving in a direction away from this detector; the rise from 10 to 20 percent which followed was the response to the radioactive cloud as it emerged from the plant stack and cleared the area. Although this instrument is more sensitive to radiation than those in Figures 13 and 14, the response to the activity emerging from the stack was considerably less, due to the larger amount of concrete between this detector and the cloud. A significant deduction from Figure 15 is that there was only one super critical power rise in this reaction.

Figure 16 presents the response of an ionization chamber detector located in the waste gas treatment building (Building CPP-604-5) at a point near the vessel off-gas line. This monitor has a thin-walled chamber sensitive to beta as well as gamma radiation. It appears to have seen radiation above background (approximately 0.5 mr/hr) beginning at about 0920, rising to about three times background, holding there for some 15 to 20 minutes, abruptly peaking at a point equivalent to about eight mr/hr, then falling back immediately to a level representing three or four mr/hr, from where it trailed off to original background over the next 10 minutes. There is no evidence of malfunction of the instrument. Inspection of the chart over a period of days before and after the incident indicates no other significant deviation from normal background. No satisfactory interpretation of this unique trace has been developed.

In summary, it appears plausible to postulate that transient criticality occurred from a sudden lift of fissile solution into a geometrically favorable region of the evaporator system. Also, it is thought that the reaction occurred in something less than the 40 liters normal operating volume of the evaporator and in a geometry different from that of the full diameter of the vapor disengagement vessel.

Resolution of the uncertainties characterizing this reaction could hardly be accomplished without careful mock-up of the system. Such would require the services of a facility for experimental solution criticality, which is not available at the NRTS. In the absence of personnel overexposures, which might have required more precise radiation dosimetry, there did not appear to be any compelling reason for repeating the reaction elsewhere.

APPENDIX V

ENVIRONMENTAL DATA

Radiation levels were measured at several points downwind from the ICPP area. After passage of the radioactive cloud, the maximum radiation level at the perimeter fence of the ICPP was 0.7 mr/hr. Radiation levels were measured from U. S. Highway No. 20 north to a point approximately two miles south of the ICPP. The radiation levels varied with time, undoubtedly reflecting the passage of the radioactive cloud. The maximum radiation levels measured at various points in the Central Facilities Area are shown in Figure 17. As noted on the Figure, the maximum radiation level detected was 30 mr/hr at a point approximately one-quarter of a mile north of the intersection of Portland Avenue and Lincoln Boulevard. (Approximately two miles south of ICPP.) This level was measured at 1015, approximately 25 minutes after the initial release of activity at ICPP.

The period following the nuclear incident was characterized by light northerly winds and a temperature lapse up to 500 feet above the surface at 1100. The wind direction and speeds recorded at Central Facilities and Grid No. 3 (about one mile north of ICPP) are listed in Table 10. From the indicated winds, a trajectory passing 0.1 mile to the west of the junction of Lincoln Boulevard and Portland Avenue and on toward the Big Southern Butte (Figure 18) was computed. The measured wind speeds indicated that an average cloud movement of three to four miles per hour should be expected for any radioactive material traveling within the first 250 feet above the ground. Somewhat higher wind speeds at greater elevations were measured from pilot balloon runs at 0902 and 1240. The interpolated wind at the base of the inversion (500 feet height) was from 020° at seven miles per hour for the hour following the release. This probably represented the upper height of the cloud. Aerial monitoring, which was restricted to elevations greater than 500 feet above the ground, reported a mean cloud travel of near six miles per hour.

Since the surface winds as shown in Table 10 were approximately 20° more from the northwest than the 250 foot winds, and since all wind directions tended to shift toward the north-northwest about 30 minutes after the incident, some fanning of material toward the east of the main trajectory was anticipated. A smoke bomb release north of Central Facilities at 1030 showed this drift to the south-southeast and also showed diffusion features associated with a strong temperature lapse. A plume width of at least one mile would be likely at the surface in the Central Facilities Area.

The movement of the released radioactive material, as measured by the surface monitoring teams and the aerial monitoring team, was in good agreement with meteorological calculations. The aerial monitoring team first detected radiation levels above background at the junction of U. S. Highway No. 20 and Main Street at 1135 (about 50 counts per second above the background of approximately 200 counts per second) and followed the path of the cloud in a straight line to the northeast side of the Big Southern Butte, where the highest readings were detected at about 1145

(approximately 200 counts per second above background). The time of travel for the cloud to reach the Big Southern Butte, nearly two hours, was in good agreement with the winds measured at near the 500 foot level by pilot balloon runs. The distance to the butte from ICPP is 12 miles and wind speed at 500 feet was seven miles per hour. The released radioactive material appeared to stagnate against the Big Southern Butte at 1145 and showed little movement for almost one hour. At 1240, the cloud had split and was moving around both sides of the butte. Peak radiation levels at this time were 100 counts per second above background. When the cloud appeared to be breaking up near the butte, another survey was made just south of U. S. Highway No. 20 and revealed that radiation levels in that area had returned to background.

TABLE 10

Winds After ICPP Incident
January 25, 1961

	Central Facilities				Grid No. 3	
	250' Level		20' Level		140' Level	
Time*	Direction degrees	Speed mph	Direction degrees	Speed mph	Direction degrees	Speed mph
0950	016	06	355	03	011	01
1000	010	04	351	03	012	02
1010	009	03	357	02	005	02
1020	005	04	349	02	349	02
1030	360	04	343	02	345	01
1040	360	03	330	02	345	00
1050	340	03	332	02	353	01
1100	342	03	325	02	360	01
1110	320	02	330	00	010	01
1120	333	00	339	00	010	01
1130	346	04	345	02	020	02
1140	035	05	005	03	035	02
1150	037	05	015	03	040	03
1200	044	06	033	03	050	03
1210	042	08	045	05	045	03
1220	045	07	042	05	045	04
1230	040	08	042	06	040	04
1240	039	07	036	05	030	02
1250	042	09	033	05	030	03
1300	039	10	033	07	030	04
1310	036	11	033	09	045	05
1320	055	10	042	09	040	06
1330	066	10	060	08	050	07
1340	066	09	072	07	050	06

*Values for 10 minute averages for period ending at the indicated time

APPENDIX VI

EVACUATION OF PERSONNEL

Radiation process alarms in the CPP-601 building sounded almost as soon as the valve to H-110 was opened. The shift supervisor and plant superintendent went immediately to the Health Physics field office where they determined from the radiation alarm display panel that radiation alarms throughout the plant had been actuated. The plant superintendent then proceeded directly to the evacuation alarm control located in Building CPP-602, just north of the Building CPP-601 operating corridor, and signaled a general evacuation. It is estimated that the time interval between the initial radiation alarms and the evacuation signal was from 30 to 45 seconds.

At the time of the incident, there were 251 people in the ICPP area, consisting of the following personnel:

Phillips Petroleum Company Employees		188
Atomic Energy Commission Employees		3
Construction Company Employees		60
H. K. Ferguson Company	57	
Fluor Corporation, Ltd.	3	
Total		251

Evacuation of Phillips Personnel

All of the Phillips employees evacuated through the main guard house area to the parking lot located just west of the ICPP fenced area (Figure 12). Many people in Buildings CPP-601 and CPP-602 heard the radiation alarms and anticipated the evacuation signal. Thus, when the evacuation signal was sounded, several individuals were already outside the buildings and some were nearly to the west guard gate. At an estimated 3 1/2 minutes after the evacuation signal had been sounded, all but four Phillips employees had reached the parking lot. (Two health physicists were making a final check for personnel in Buildings CPP-601 and CPP-602, and two employees were evacuating from Building CPP-603, approximately 2,500 feet south of the main processing building.) The estimates of time for complete evacuation of Phillips personnel ranged from five to seven minutes. This compares with evacuation times of approximately four to five minutes noted in practice evacuations.

The first health physicist to arrive at the assembly area made a survey of the area immediately with radiation monitoring instruments from the emergency kit located in the guardhouse. The only radiation level detected above normal background was one of approximately one mr/hr in the vicinity of the guardhouse.

At approximately 1000, the personnel monitor in the guardhouse stopped alarming. This monitor is normally set to alarm at radiation levels of less than one mr/hr and resets automatically when the radiation level falls below the alarm level.

Buses from the Central Facilities Area arrived at the evacuation area between 1010 and 1025. At approximately 1030, Phillips employees who were not part of the re-entry teams, boarded the buses. Health physicists from the MTR-ETR area arrived and checked personnel on the buses for contamination. Employees who were wearing "hot" area clothing and shoes were given "cold" area laboratory coats and shoe covers. Film badges were collected on the buses from 20 individuals representing 13 areas of the ICPP. These badges were scanned for neutron activation of the indium foil and subsequently analyzed for beta and gamma exposure. At approximately 1140, the personnel on the buses were transferred to the Central Facilities Cafeteria where they remained until approximately 1445, at which time they returned to the ICPP area.

Evacuation of AEC and Construction Personnel

At the time of the incident, 57 H. K. Ferguson Company employees were working in various locations in the ICPP fenced area. In addition, three Fluor Corporation, Ltd. and three AEC employees were working in an office just north of the ICPP fenced area. When the evacuation alarm sounded, the AEC and Fluor employees with approximately 45 of the H. K. Ferguson employees evacuated through the north gate, traveled around the perimeter fence, and assembled at the north end of the parking lot in front of the main guardhouse. The remainder of the construction personnel, with the exception of one man, evacuated through the main guard gate and joined the other construction employees at the assembly area. Approximately five minutes after the evacuation had been initiated, a count of construction workers was made by the area superintendent and it was believed that all employees had evacuated. This group then moved to the junction of Cleveland Boulevard and Lincoln Boulevard to await further instructions.

One construction employee remained inside the fenced area for approximately 10 minutes. This man was working on an air compressor near the ramp leading to the south end of Building CPP-601 and did not hear the evacuation signal over the noise of the compressor.

At approximately 1120, the construction employees were notified that they would not be allowed to return to the ICPP area for the remainder of the day. All construction personnel and automobiles were monitored and released.

APPENDIX VII

IDENTIFICATION OF LOCATION OF INCIDENT

The specific location of the incident was not immediately apparent. All uranium in the plant was believed to be contained either in geometrically safe equipment or at solution concentrations below the minimum for criticality. No unusual operations were known to have been performed and instrument records were in themselves not indicative of a significant upset.

Two processes in the plant handle gaseous and volatile fission products, viz., a radiobarium separation system (RaLa process) and the dissolver off-gas handling system. A RaLa processing run had been completed two days earlier and no operations were being performed in that equipment. However, post-run activities in the past had resulted in some release of Iodine-131. Chemical changes in the process had essentially eliminated this problem, but the associated high concentrations of short-lived fission products cause this system to be suspect in any unidentified radiation incident.

Initially it was believed that the release of air-borne radioactive materials might have been the result of a failure in the krypton recovery system. Consequently, the first re-entry team consisted of operators particularly familiar with the RaLa and dissolver off-gas systems, process operators to shut down all operating equipment, and health physics personnel. All instrument charts were scanned for evidence of abnormal transient pressure and volume changes. In addition to monitoring for radiation fields, the health physicists collected radiation detection film packets from various areas. The only unusual instrument record noted at this time was an apparent overflow from H-110 to H-111 (Figure 19).

After it had been established that a criticality incident had occurred, instrument records of all equipment containing uranium were again inspected in detail. H-Cell was suspected as the location of the incident. This cell contains several vessels that are not geometrically safe. The disengaging sections of the solvent extraction columns present a potential criticality hazard if flow of certain streams is interrupted for two hours or more. Also, failure to strip uranium from the organic stream for a prolonged period could lead to an unsafe condition in the solvent purification mixer-settlers. An examination of liquid flow records however, quickly verified that this equipment most likely was not involved. The evaporator (H-110) and steam stripper (H-109) remained as possible locations in which criticality could occur. Other instrumentation (and later sample analyses) indicated that no significant volume of solution had left H-Cell equipment.

To preclude recurrence of criticality, while investigating the exact location and cause of the incident, the contents of H-110 were transferred to the salvage equipment in J-Cell. The material balance computed after measurement of solution in J-Cell (Appendix II) indicated that all uranium, within limits of measurement uncertainties, was accounted for.

From a detailed review of all available evidence and questioning of personnel in the plant at the time of the incident, it was concluded that criticality could only have occurred in the vapor disengaging section of the H-110 evaporator. Several conceivable mechanisms for getting a critical mass of uranium into the vapor head were considered; however, only one appeared to be reasonably consistent with all the evidence.

This mechanism required some amount of air under pressure to have been in the pump decontamination line (Figure 1) at the time that the valve between Pump PA-239 and H-110 was opened, causing a large portion of H-110 content to be air-lifted into the geometrically unsafe vapor disengaging chamber. This mechanism would be much more readily acceptable if there had been indications from operator testimony that the 10-gallon decontamination pot had still been open to the system at the time, thus providing a larger volume of air for expansion. However, the limited number of witnesses and the imperfect communication between them preclude reconstruction of the exact sequence of events.

APPENDIX VIII

PERSONNEL EXPOSURES

The first indication that personnel exposures probably were not excessive was received when direct reading pocket dosimeters worn by process operating personnel were checked. This general check indicated that exposures probably did not exceed 50 millirem. Later, when it had been established that a nuclear excursion had occurred, film badges of 65 individuals from various locations in the plant area and the construction area were collected and analyzed. The highest exposure detected was one of 55 mrem gamma, 0 beta. The maximum thermal neutron exposure as measured by threshold detectors in the 65 badges analyzed was less than 10 millirem. A total neutron dose (fast and thermal) was calculated from data obtained from film badges worn by the three individuals in the access corridor near H-Cell and from monitors located in the same area. It was concluded that the maximum total neutron dose that could have been received by any individual was 55 millirem.

The film badges worn by Phillips employees had been changed on Friday, January 20, 1961. Thus, at the time of the incident, these film badges had been worn for periods of three to five days, depending on whether the individual had worked during the weekend. Except for those film badges read on the day of the incident, Phillips personnel continued to wear the same film badges until February 16, 1961, at which time the regular four-week badge change was made. With the reading of film badges worn by all individuals in the ICPP area at the time of the incident, it was established that the highest personnel exposure received for the four-week period of January 20, through February 16, 1961, by any Phillips employee in the ICPP at the time of the incident was 240 mrem gamma, 310 mrem beta. It was also determined that 20 Phillips employees in the ICPP area at the time of the incident received four-week total exposures greater than the 55 mrem noted in the initial 65 badges analyzed on the day of the incident. Although film badges for all personnel in the ICPP at the time of the incident were not analyzed immediately, it is very unlikely that any personnel exposure in excess of 55 mrem could be attributed to the excursion. The 65 personnel badges, direct reading pocket dosimeters, fixed area monitors and recording radiation instrumentation are considered to present sufficient evidence to support this conclusion.

APPENDIX IX

ILLUSTRATIONS

Figure No.	Caption
1.	Process Flowsheet - First Cycle Product Wash and Concentration - Cell H - Continuous Aluminum Process
2.	H-Cell Evaporator System
3.	H-Cell Interior, Evaporator, H-110
4.	Decontamination System in Process Makeup Area
5.	H-110 Liquid Level Chart - January 25, 1961
6.	H-110 Specific Gravity Chart - January 25, 1961
7.	H-110 Temperature Chart - January 25, 1961 Point 6 - Top of Thermal Leg Point 7 - Vapor Outlet Point 8 - Product Outlet
8.	H-110 Pressure Chart - January 25, 1961
9.	Process Building Arrangement
10.	Section of Process Building CPP-601
11.	Critical Volume vs Critical Mass - Bare Stainless Steel Right Cylindrical Vessel - $UO_2 F_2$
12.	CPP Area Plot Plan
13.	CAM Chart - Bldg. 601 - West Side of PM Area
14.	CAM Chart - Bldg. 603
15.	Background Monitor Chart - Bldg. 601 - Health Physics Field Office
16.	Monitron Chart - Bldg. 604 - Operating Corridor
17.	Maximum Radiation Levels Measured at Ground Level January 25, 1961
18.	Estimated Cloud Trajectory - January 25, 1961
19.	H-111 & H-112 Liquid Level and Specific Gravity Chart - January 25, 1961

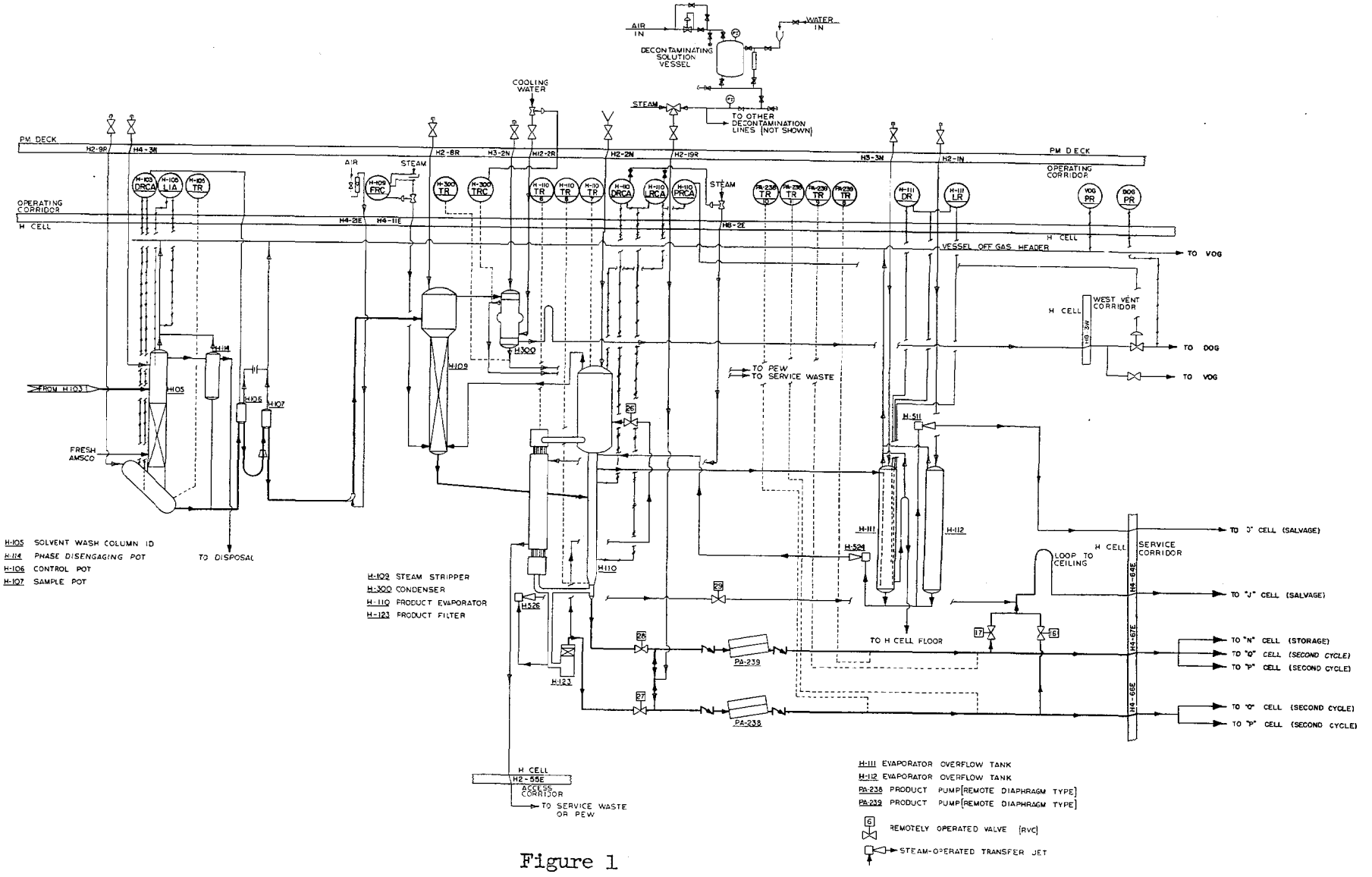


Figure 1

PROCESS FLOWSHEET
FIRST CYCLE PRODUCT WASH AND CONCENTRATION CELL H
CONTINUOUS ALUMINUM PROCESS

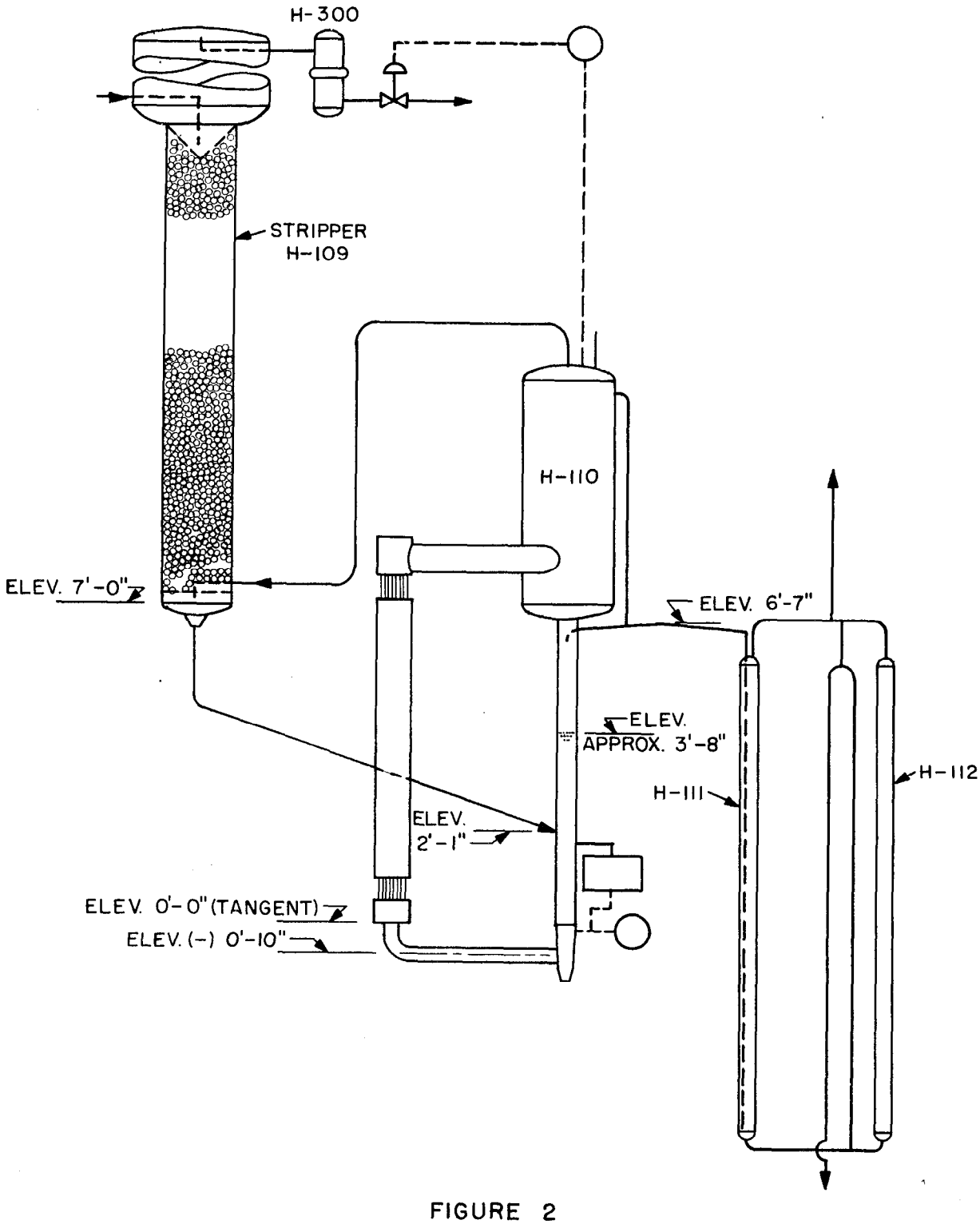


FIGURE 2

"H" CELL EVAPORATOR SYSTEM

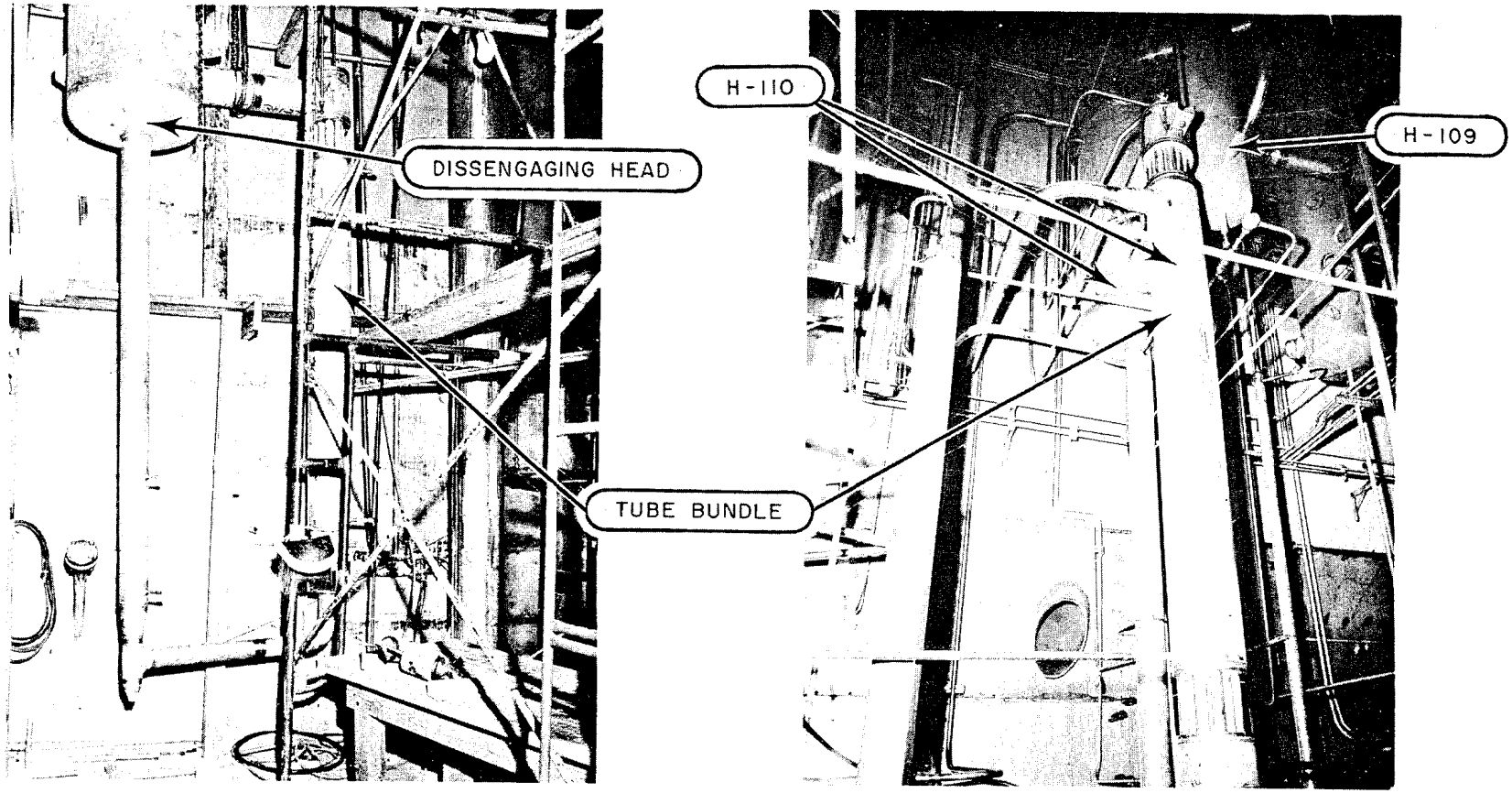


FIGURE 3

H-CELL INTERIOR, EVAPORATOR, H-110

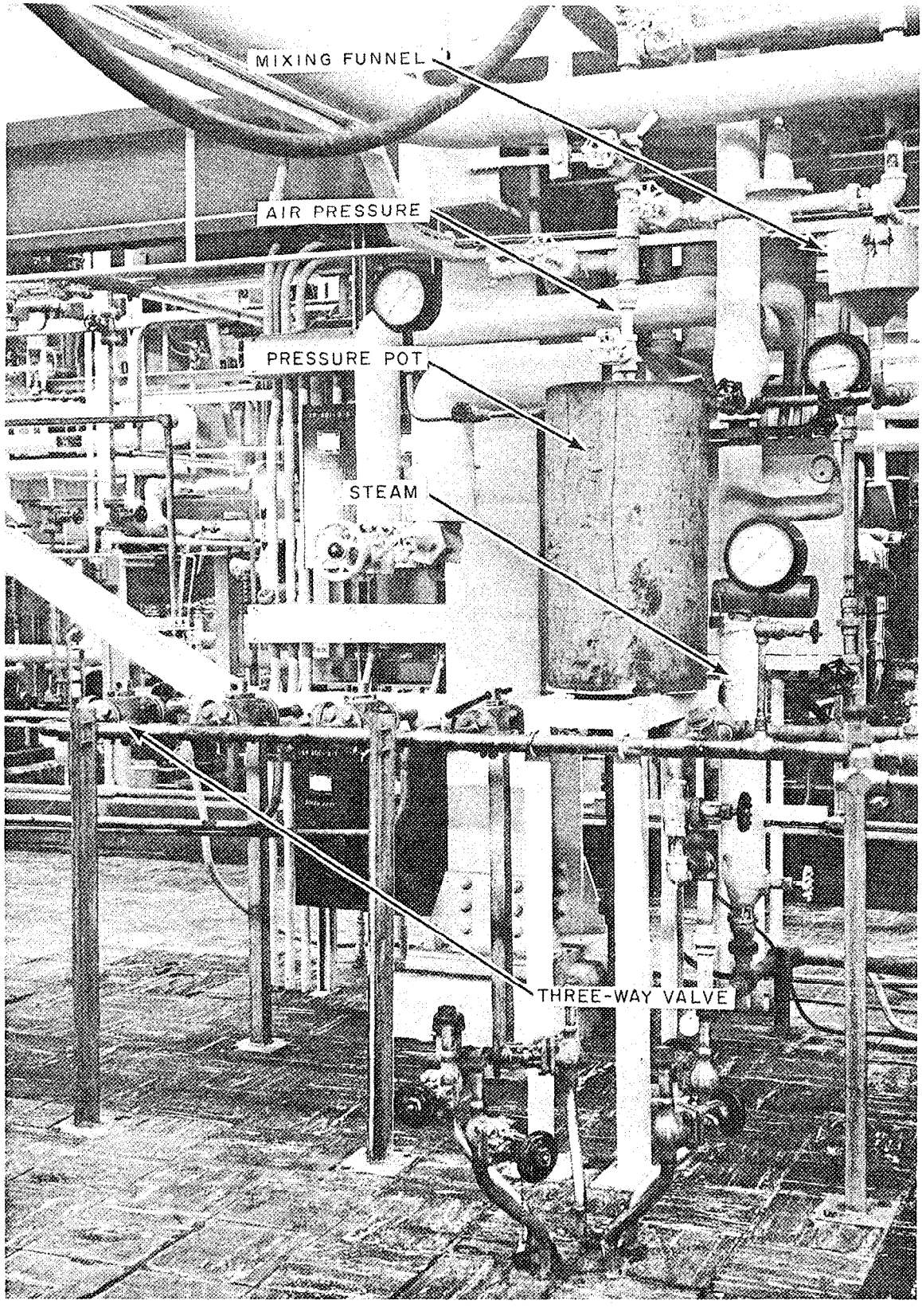


Figure 4

DECONTAMINATION SYSTEM
IN PROCESS MAKEUP AREA

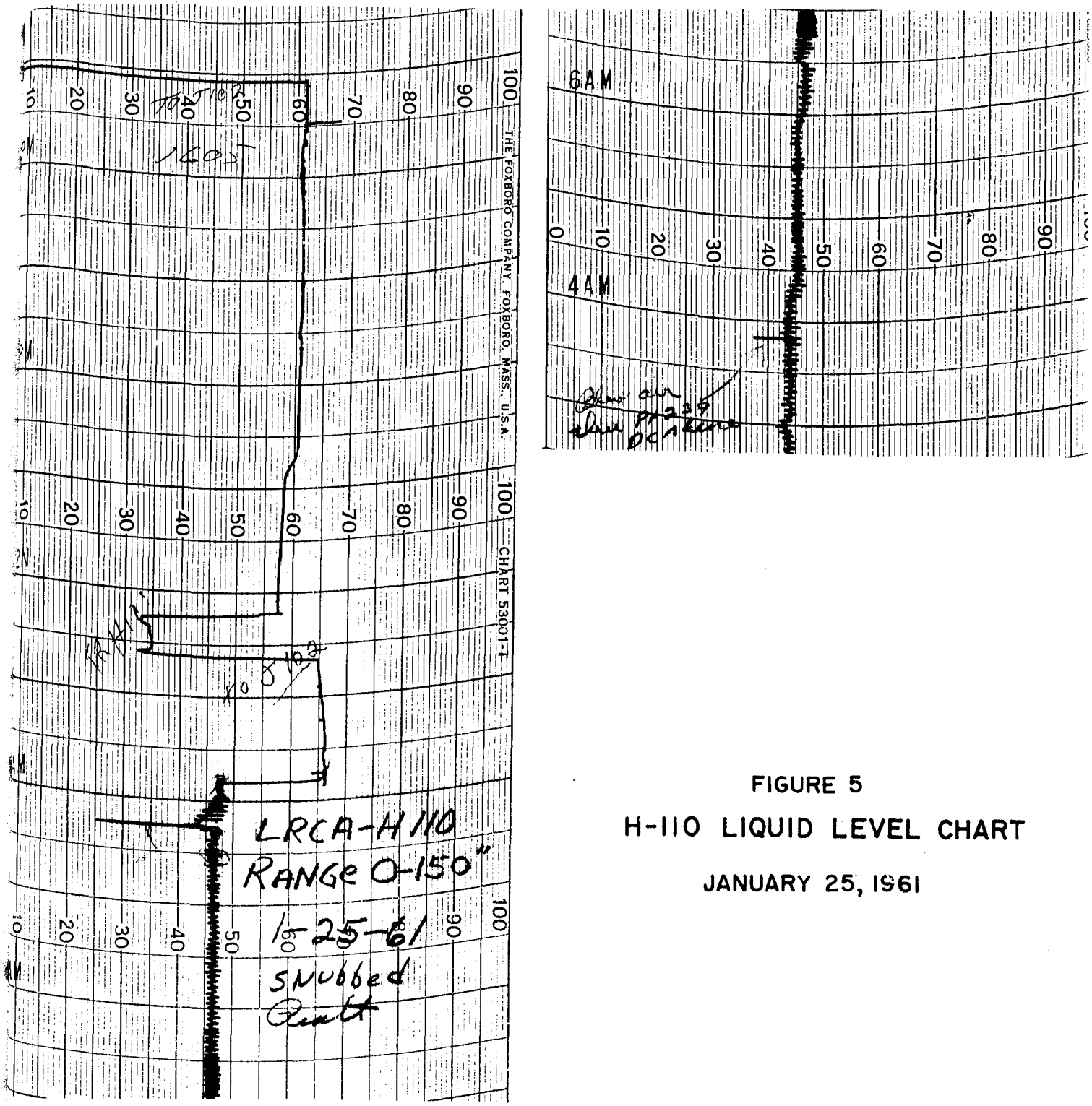


FIGURE 5

H-110 LIQUID LEVEL CHART

JANUARY 25, 1961

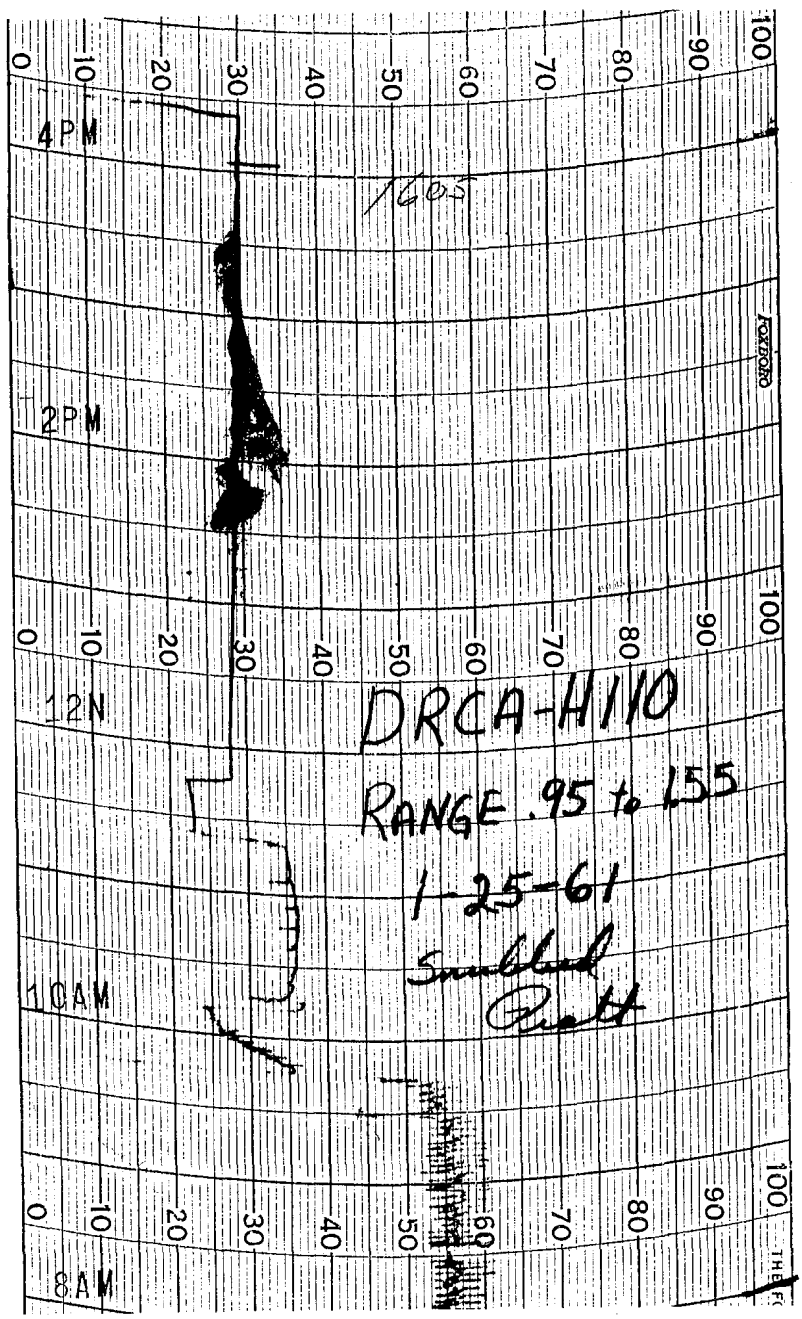


FIGURE 6

H-110 SPECIFIC GRAVITY CHART

JANUARY 25, 1961

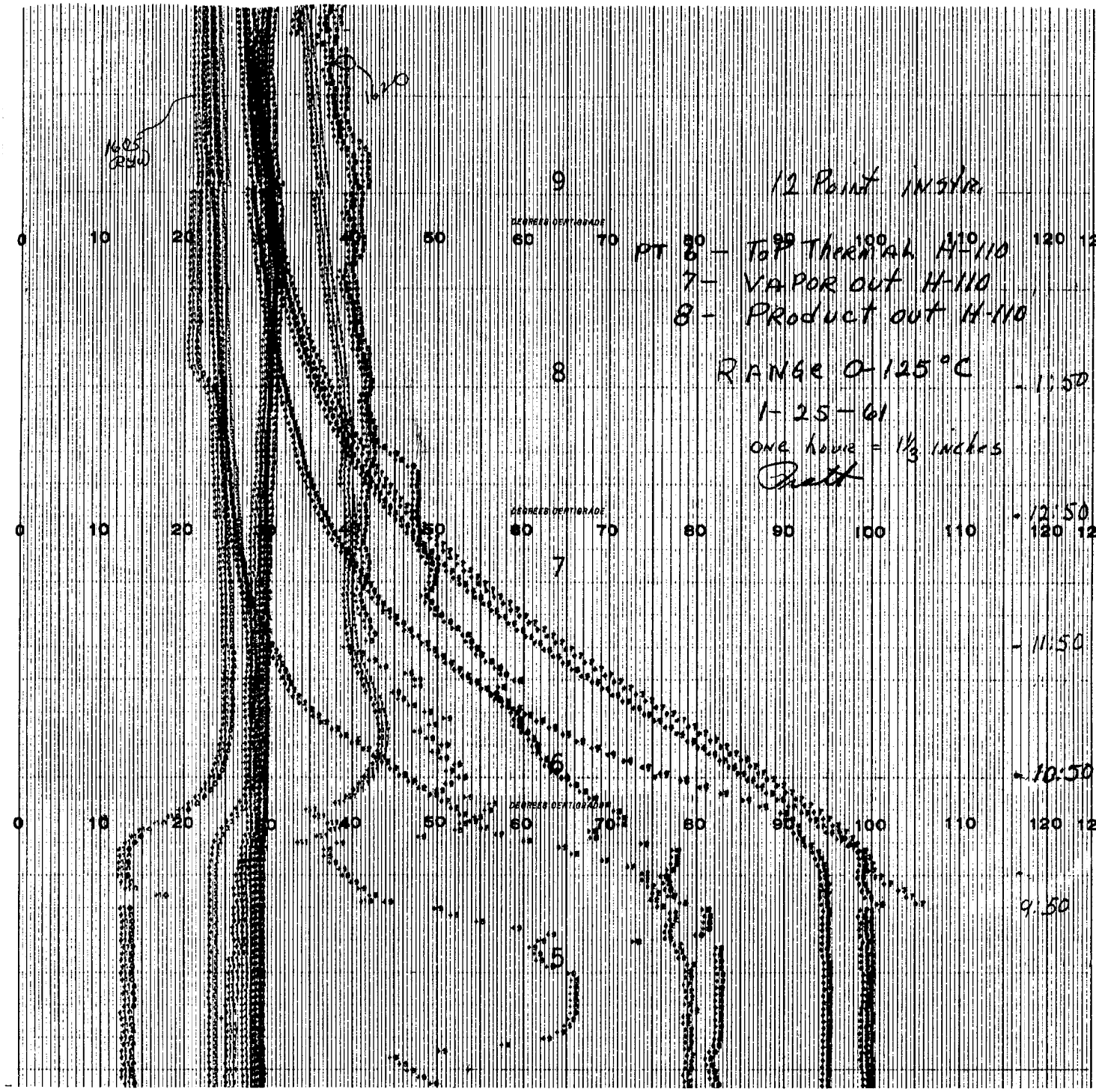


FIGURE 7

H-110 TEMPERATURE CHART

JANUARY 25, 1961

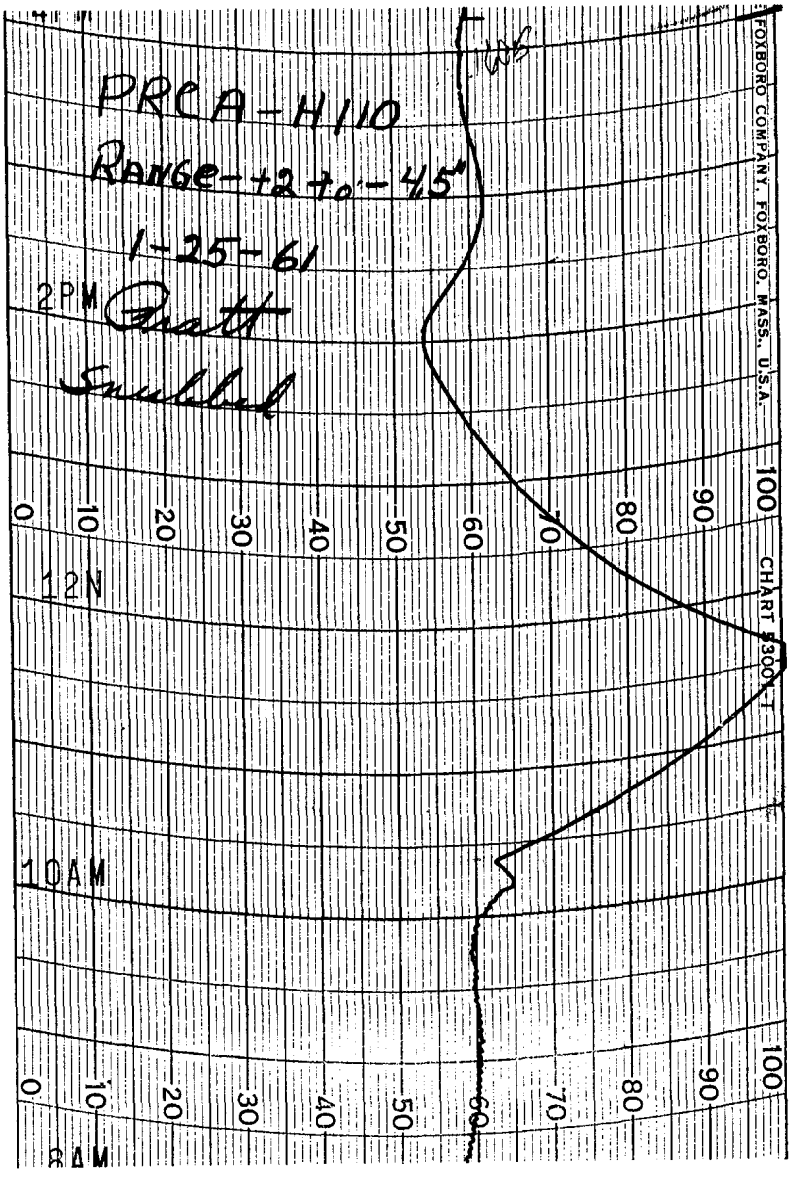
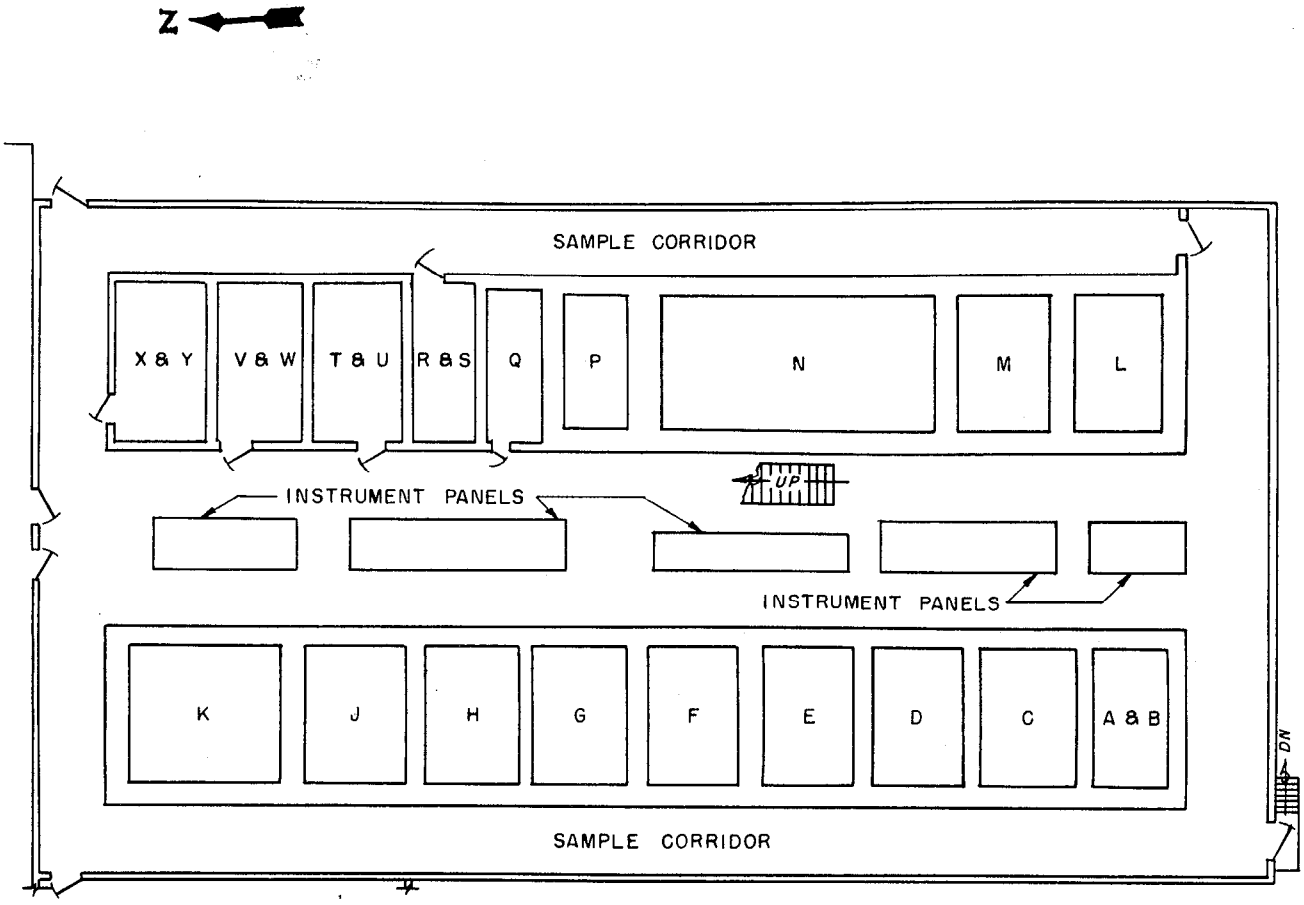


FIGURE 8

H-110 PRESSURE CHART

JANUARY 25, 1961



CELL DESCRIPTION

A EBR DISSOLUTION	M SPARE
B UNH FEED STORAGE	N FEED STORAGE
C BATCH Al DISSOLUTION	P 1st CYCLE EXTRACTION
D BATCH Al DISSOLUTION	Q 2nd CYCLE EXTRACTION
E SS - Zr DISSOLUTION	R & S 3rd CYCLE EXTRACTION
F SS - Zr TBP EXTRACTION	T SOLVENT FEED
G CONTINUOUS Al DISSOLUTION	U 1st CYCLE AQUEOUS RAFFINATE
H Al TBP EXTRACTION	V H P OFFICE
J SALVAGE	W 1st CYCLE ORGANIC RAFFINATE
K SOLVENT RECOVERY	X SAMPLE DILUTION
L RALA	Y 2nd, 3rd CYCLE RAFFINATE

Figure 9
PROCESS BUILDING ARRANGEMENT

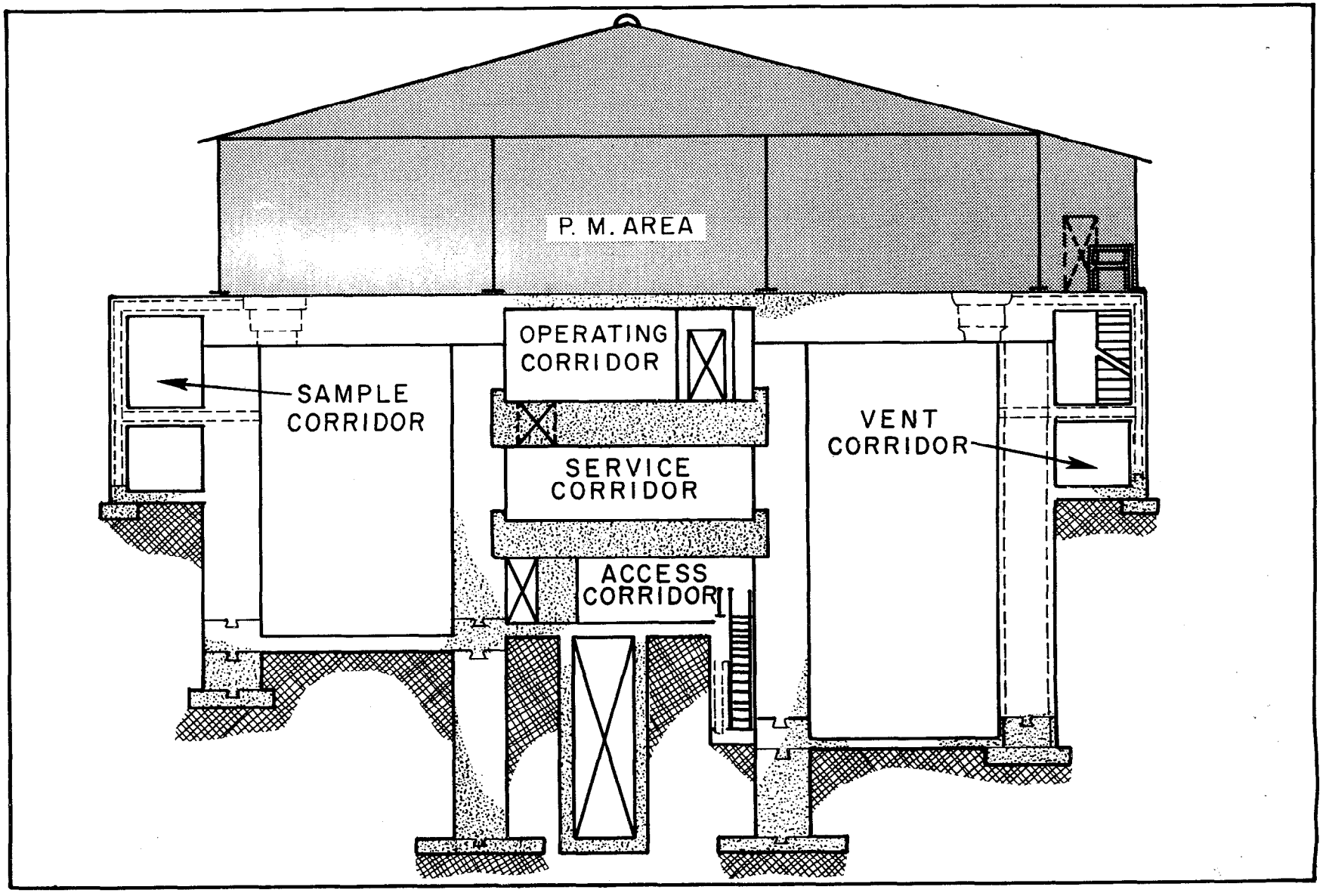


Figure 10

SECTION OF PROCESS BUILDING CPP-601

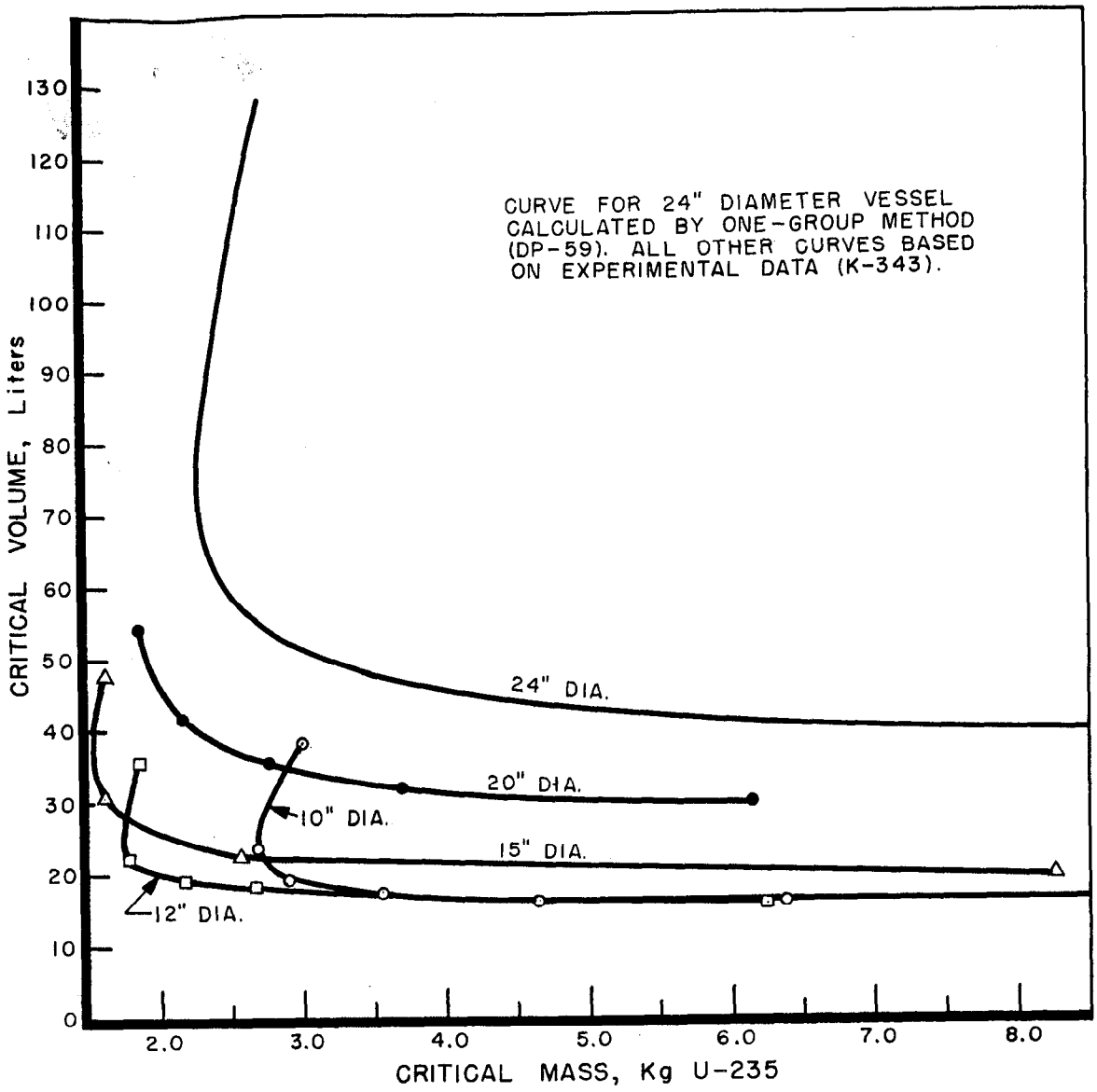


FIGURE 11

CRITICAL VOLUME vs. CRITICAL MASS

BARE STAINLESS STEEL RIGHT CYLINDRICAL VESSEL

UO_2F_2

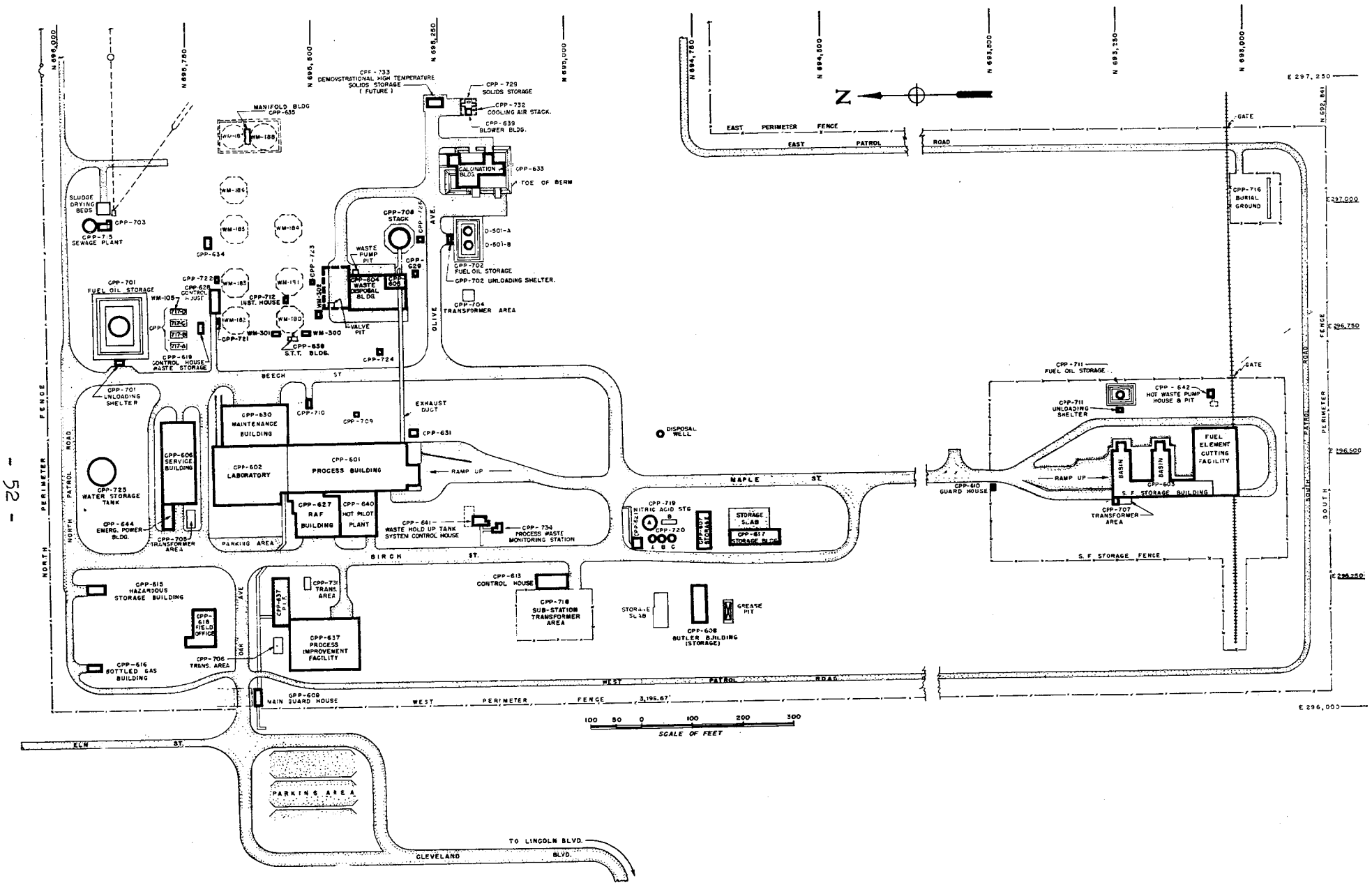


Figure 12

CPP AREA PLOT PLAN

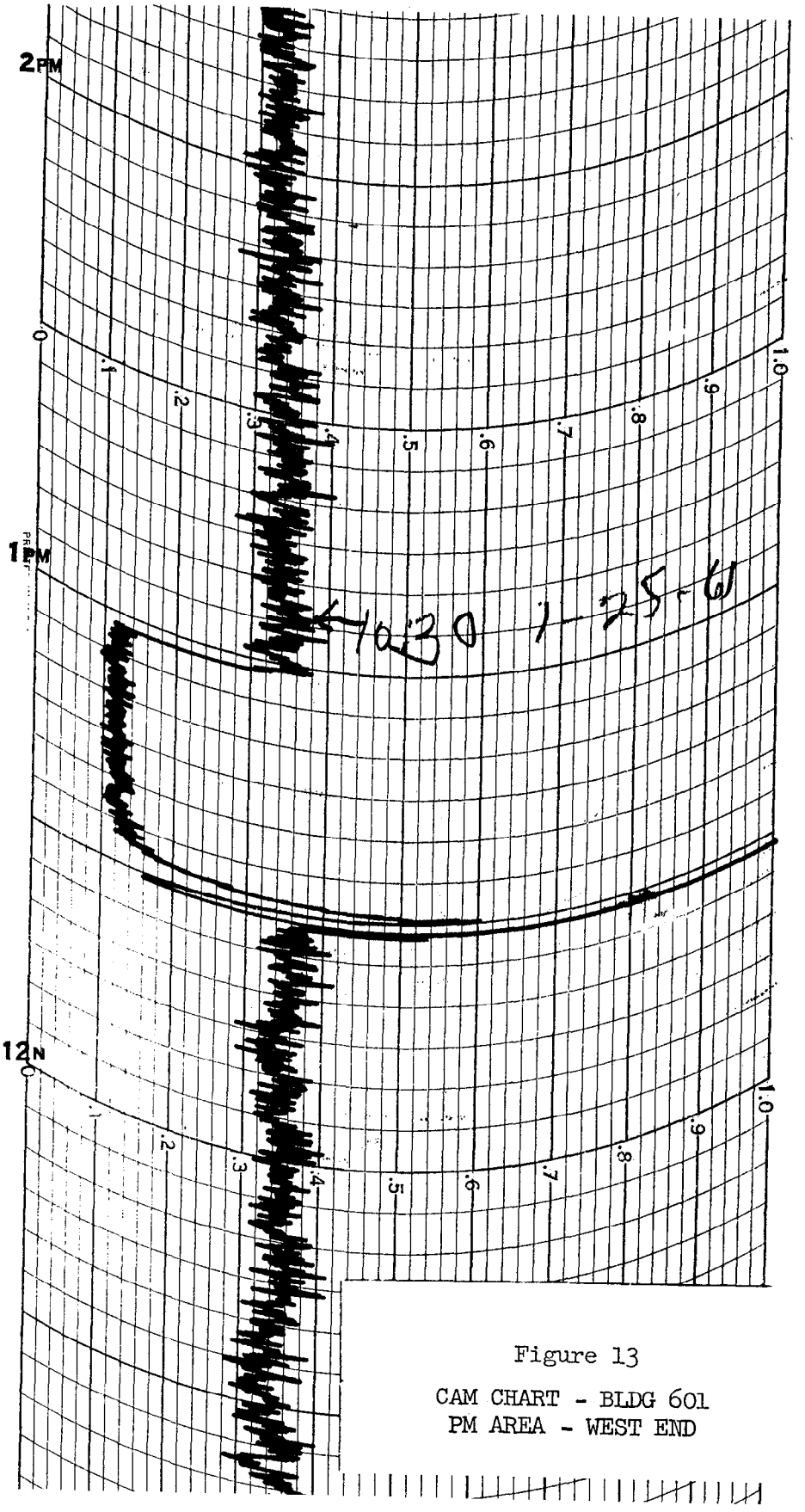


Figure 13

CAM CHART - BLDG 601
PM AREA - WEST END

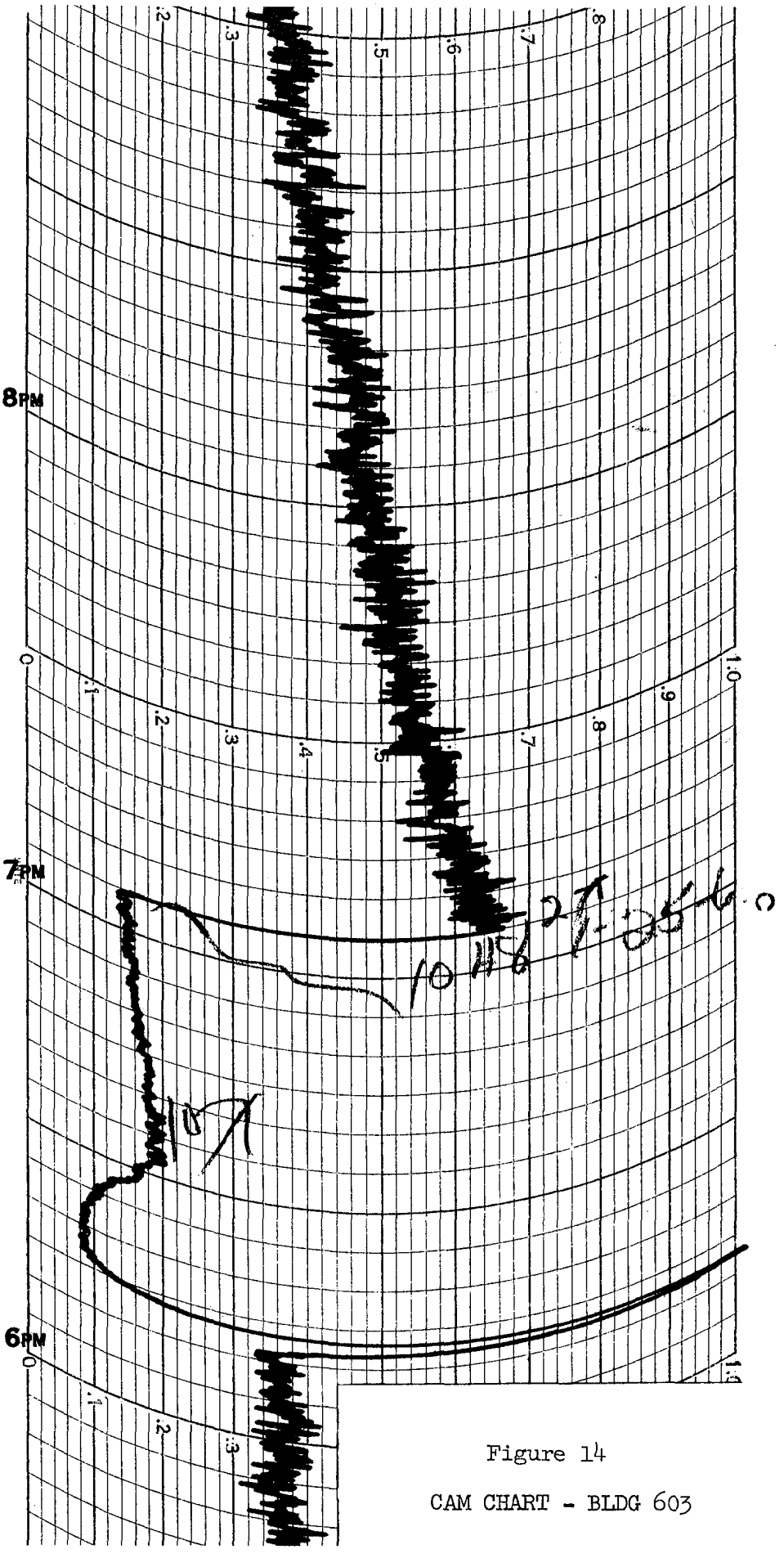


Figure 14

CAM CHART - BLDG 603

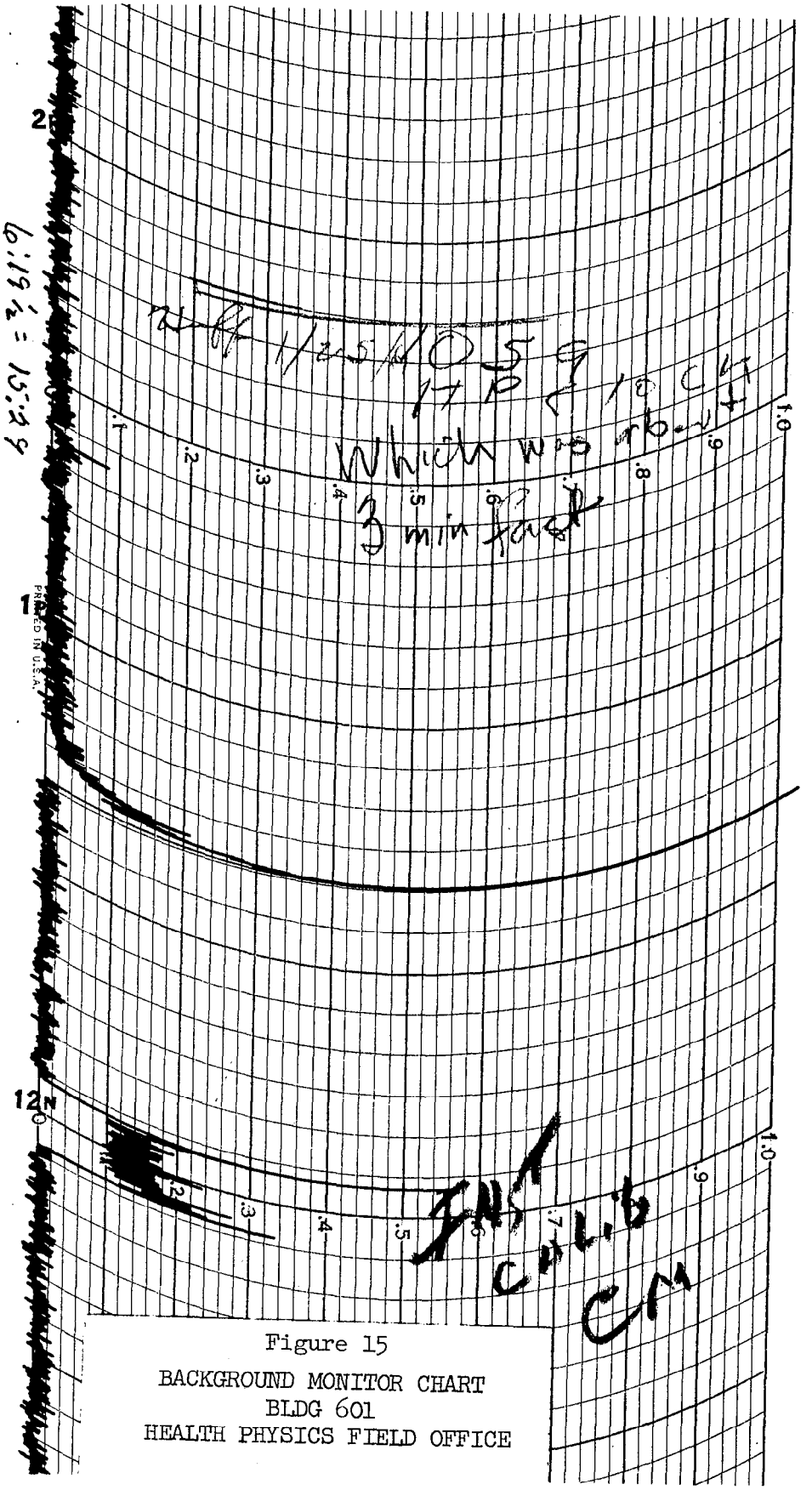


Figure 15

BACKGROUND MONITOR CHART
BLDG 601
HEALTH PHYSICS FIELD OFFICE

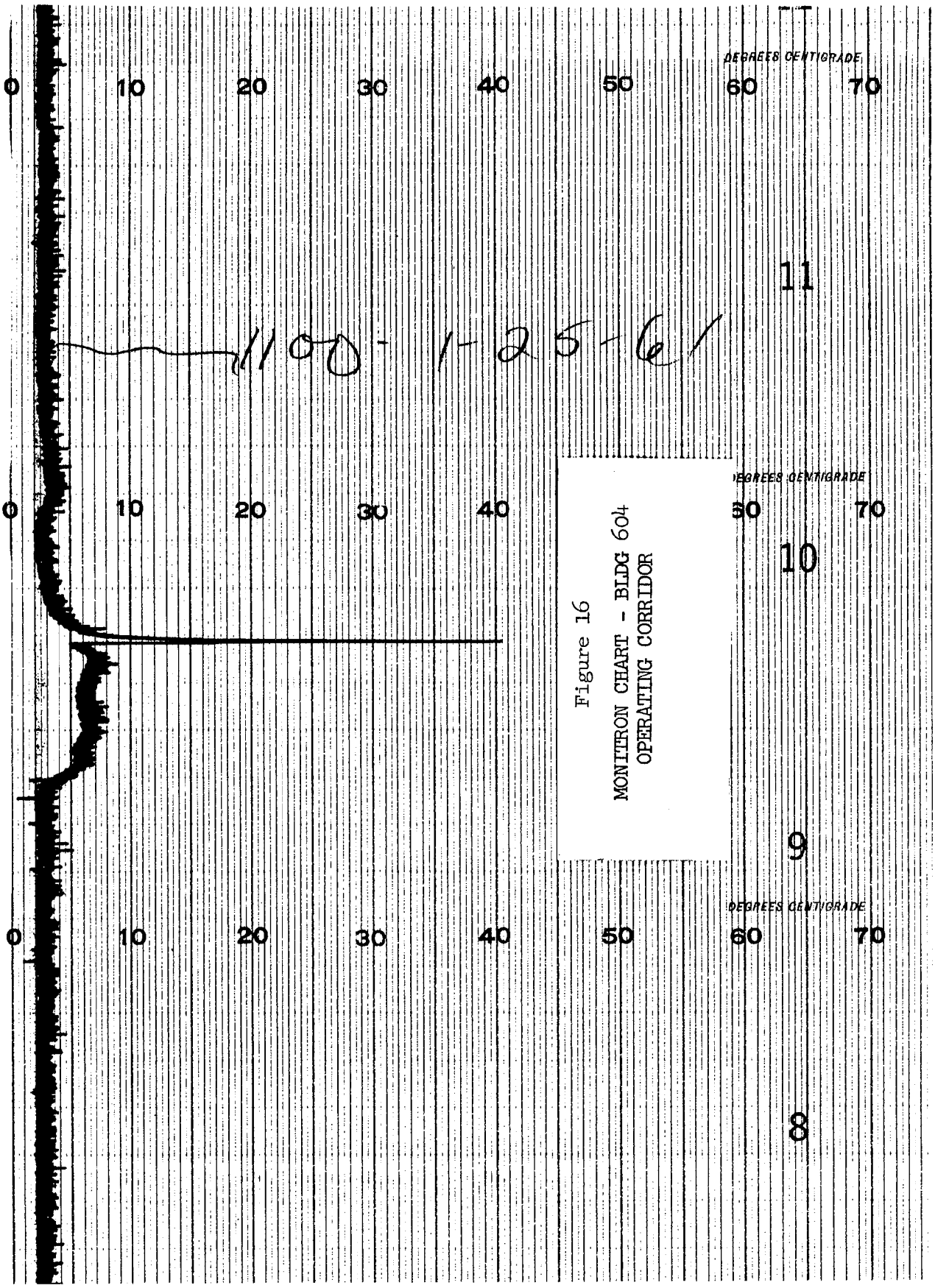


Figure 16

MONITRON CHART - BLDG 604
OPERATING CORRIDOR

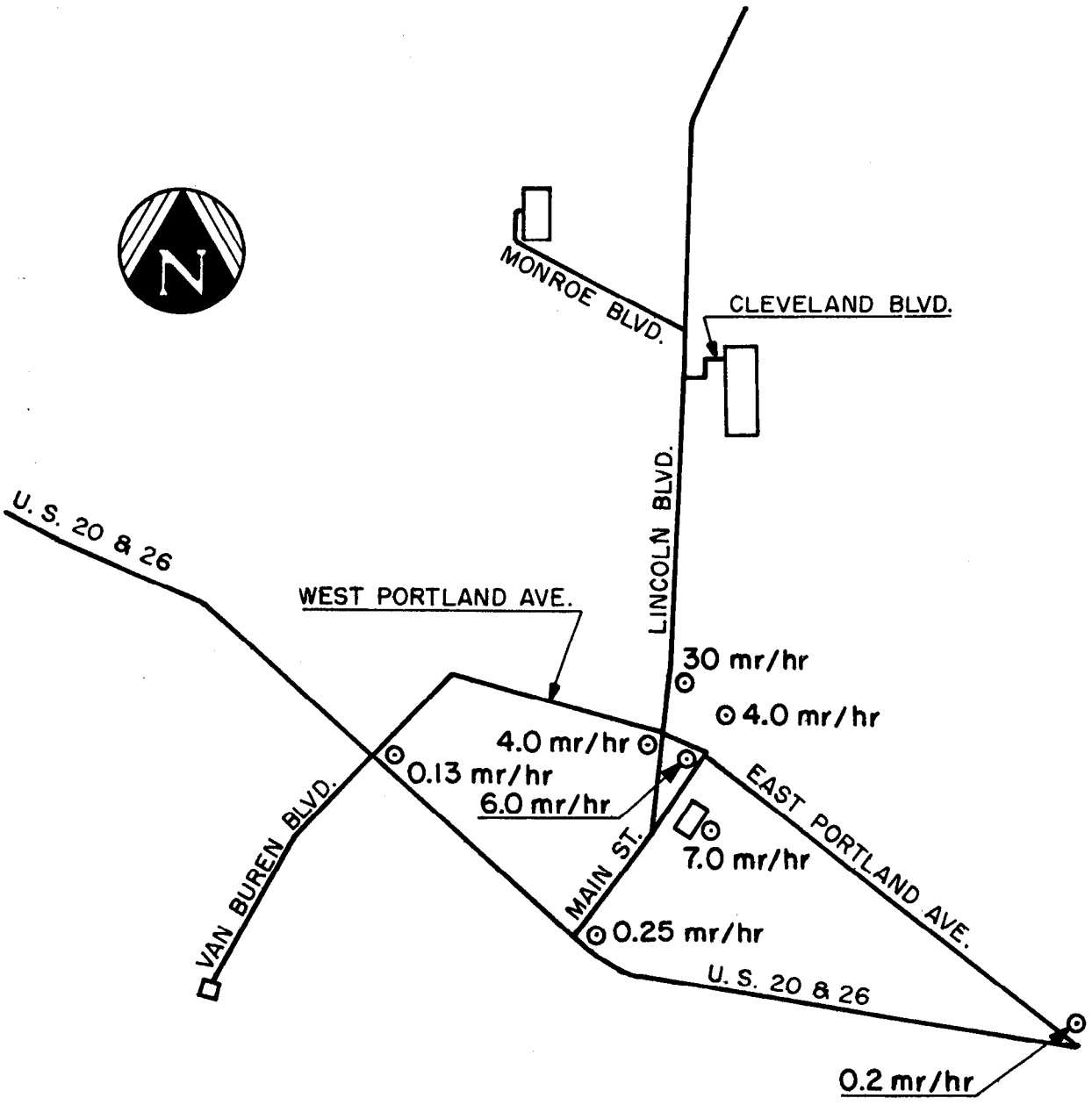


FIGURE 17

MAXIMUM RADIATION LEVELS MEASURED AT GROUND LEVEL
JANUARY 25, 1961

NATIONAL REACTOR TESTING STATION

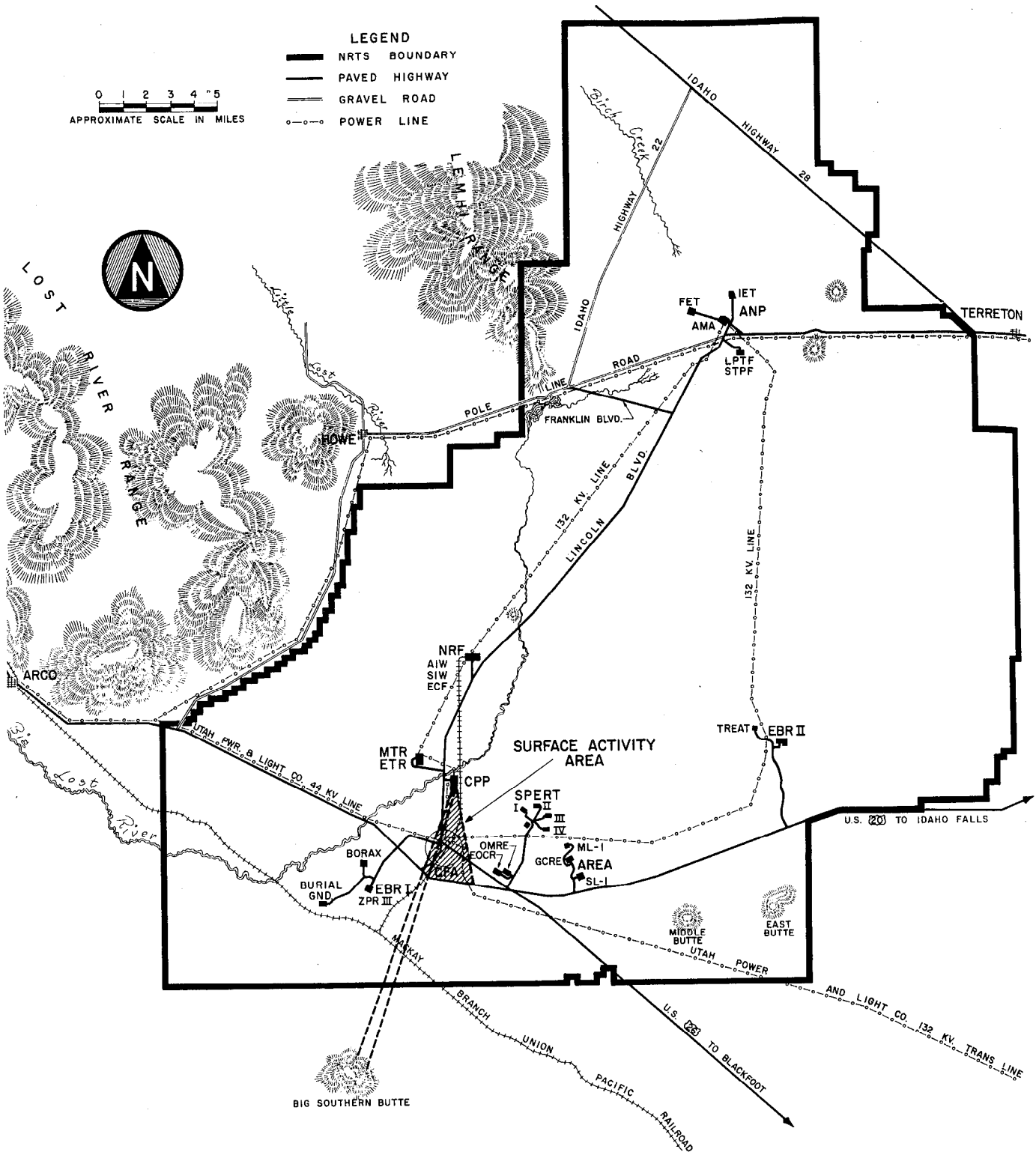


FIGURE 18

ESTIMATED CLOUD TRAJECTORY - JANUARY 25, 1961

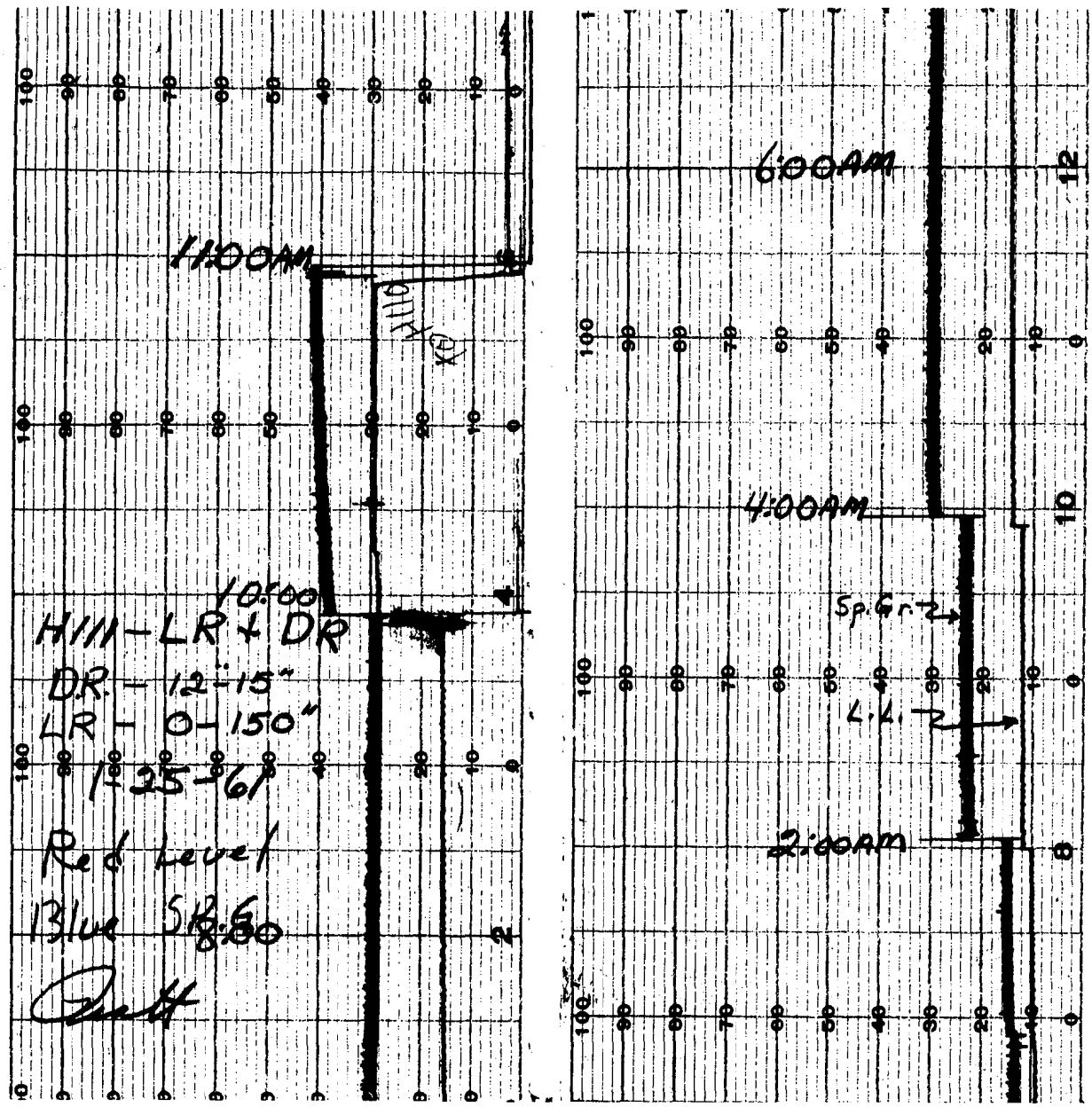


FIGURE 19

H-111 & H-112 LIQUID LEVEL AND SPECIFIC GRAVITY CHART

JANUARY 25, 1961