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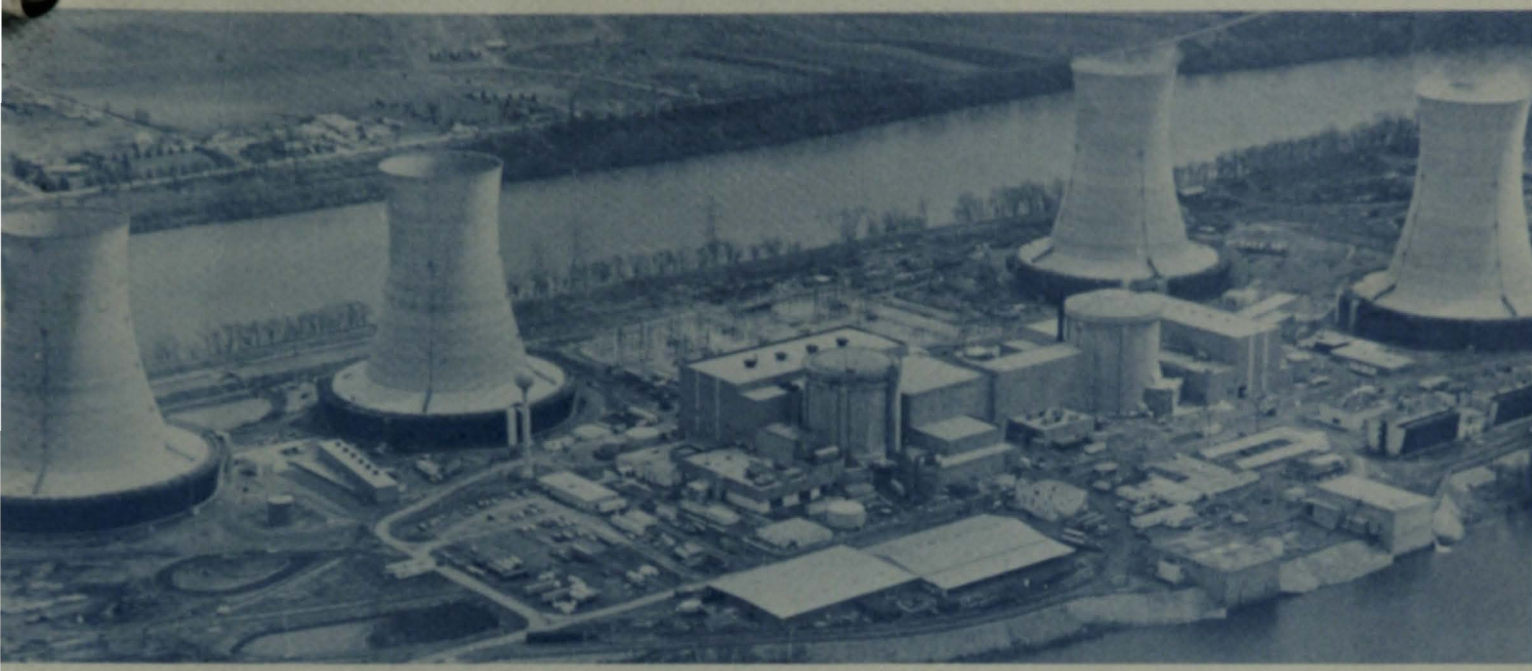

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General Public Utilities • Electric Power Research Institute • U.S. Nuclear Regulatory Commission • U.S. Department of Energy

Facility Decontamination Technology Workshop
November 27-29, 1979
Hershey, Pennsylvania

Sponsored by
Department of Energy
and
Electric Power Research Institute

October 1980

Prepared for the
U.S. Department of Energy
Three Mile Island Operations Office
Under DOE Contract No. DE-AC07-76IDO1570

GEND-0002

FACILITY DECONTAMINATION TECHNOLOGY WORKSHOP
NOVEMBER 27-29, 1979
HERSHEY, PENNSYLVANIA

Sponsored by
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Published October 1980

Technical Integration Office
TMI-2 Technical Information and Examination Program
Middleton, Pennsylvania 17057

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FACILITY DECONTAMINATION TECHNOLOGY WORKSHOP

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Abstract

Art Carson (EPRI), Herb Feinroth (DOE) and Bill Hopkins (Bechtel) provided the welcome, introduction and opening remarks. The purpose of the meeting was stated as:

1. Provide a record of experience at other facilities of events and incidents which have necessitated decontamination and dose reduction activities.

2. Furnish GPU, and others involved in the TMI-2 cleanup, with the results of that decontamination and dose reduction technology.

Jack Devine (GPU) described plant layout and design. Units 1 and 2 are 820 MWe and 960MWe respectively. Unique plant features include a flood dike around the island and design of all Class 1 structures for 200,000 pound aircraft impact due to proximity to Harrisburg airport. It was noted that the plant model used during plant construction will be refurbished for use in clean-up activity planning.

Bill Hopkins (Bechtel) described the results of containment radiation measurements to date. Collimated instrument reading indicate that most of the contamination is on the floor level surfaces where plate out is estimated at about 20_{μ} Ci/cm^2. Collimator measurements have established containment water level at 6 to 7 ft. Airborne β dose rate is approximately 200R/hr due primarily to the Krypton-85 concentration of 0.8 $_{\mu}$Ci/cc. It is estimated that these may be about 240,000 ft.^{2} of exposed surface subject to contamination in the 2 million cubic foot containment.

George Kulynych (B&W) described the Nuclear Steam Supply System. TMI-2 is a B&W 177 fuel assembly core similar to Oconee, Arkansas Nuclear 1, Rancho Seco and Crystal River-3. Internal surfaces exposed to primary coolant are 280,000 ft^2 inconel (Steam Generators), 57,000 ft.^{2} Zircaloy (core), 130,000 ft.^{2} stainless (piping, vessel and component internals). External surfaces are generally aluminum paint covered carbon steel. Particular decontamination problem area are expected in the thermal insulation, equipment supports, motors and fuel handling equipment.

Jack Daniels (GPU) discussed the chemical and radiological analysis of the containment sump water. No normal means existed for sampling of the approximately 270,000 gallons of water released to the sump during the accident. Samples of sump water were eventually obtained and found to have an activity of about 180 Ci/cc (essentially all due to Cesium). There were extremely small concentrations of uranium and transuranic elements (parts per billion). This supports earlier predictions that the fuel fragments have not dissolved.

Ed Walker (Bechtel) described the results of measurements taken through a 9" penetration. A beta/gamma dose rate ratio of 100 was observed in containment. Video tape views inside containment were also obtained. It was noted that a continuous "Raining" process appears in progress due to simultaneous evaporation and condensation in containment.

Mike Morell (GPU) discussed preparations for containment reentry. The four alternatives for handling the Krypton gas in containment are: cryogenic processing, gas compression, charcoal absorption and atmospheric dispersion by venting. Means for control of contamination, personnel clothing, communications, personnel breathing and containment lighting were discussed.

Paul Ruhter (GPU) discussed the Health Physics Program at TMI-2. He noted that there is really no one portable instrument satisfactory for use in the radiation fields encountered at TMI-2. The high beta field also has a complicating effect in the interpretation of film badge readings. Use of a lead impregnated rubber suit (such as Beta-guard) to protect from the beta field was being considered for use in containment entry.

Ed Gupton (ORNL) discussed problems of personnel dosimetry in the Auxiliary Building during clean up operation after the TMI-2 accident.

Tom Block (GPU) discussed decontamination experience at the TMI-2 Auxiliary and Fuel Handling Building. Methods used include dry vacuuming with HEPA filters, manual wiping, Radiac wash, wet vacuuming and use of strippable coating

to "lift" contamination. Overall results are a reduction in Iodine from 10^{-6} to 10^{-12} μCi/cc, surface from 10^{7} to 10^{3} dpm/100 cm^{2} and dose rate from 1R/hr to 1mr/hr.

Rick McGoey (GPU) discussed liquid/solid waste processing experience at TMI-2. In excess of 50,000 gallons of containmnent water entered the Auxiliary building. The assessment of water on site requiring processing was 15,000 gallons less than 1 μCi/cc, 360,000 gal from 1 to 100 μCi/cc and 530,000 gal greater than 100 μCi/cc. Processing to date has been basically by filtration and demineralization with DF's on the order of 10^{8}.

Bud Arrowsmith (Battelle) described the equipment decontamination system (EDS) which will be used for decontamination of some of the containment equipment. This system uses advanced decontamination techniques such as electropolishing, vibratory finishing and high pressure freon cleaning. A unique feature of this system is that it allows reprocessing of acids and freon during the decontamination process.

Frank McDougall (Bechtel) described plans for containment recovery. Presently, many options are being considered for containment decontamination. As detailed information concerning the chemical, radiological and structural condition of the containment becomes available, however, some options will disappear and detailed plans can be prepared. A plan to use the reactor building spray system for initial decontamination was discussed. Other options include use of hydro-lasers, steam lances, and local chemical decontamination.

C. Wayne Bills (EG&G Idaho) discussed the SL-1 recovery. About 5% of the core was washed out of the vessel during the SL-1 accident. Beta exposure was limiting. He noted that steam cleaning was effective for surface decontamination and that about 85% of the Anti-C clothing was recycled. He stressed the importance of planning, training, rehearsals, and debriefing of recovery teams. Sufficient lighting should be provided. Documentation of the recovery effort, including live movies, videotapes, etc., should be planned for.

John Logie (Chalk River Nuclear Lab) discussed recovery of the NRX 1 and NRU reactors at Chalk River. Thorough planning was emphasized. In the NRU

recovery, for example, decay heat removal was not provided for the removed fuel, and a fuel fire was initiated. The NRX 1952 accident resulted in an exposure of 2,600 man-rem for 1,100 people. The 1958 accident exposure was 700 man-rem for 800 people.

Paul Pettit (AIF) discussed the chemical processes used for decontaminating stainless steel and carbon steel reactor coolant systems in Canada. A solution of demineralized water and oxyalic acid, citric acid or EDTA was used. The solution was pumped through the RC loop, to a filter-demineralizer combination and returned to the loop. DF's were generally less than 3.

Paul Bacca (Argonne-West, Idaho) discussed decontamination of the hot cell of the Idaho fuel cycle facility. Initial efforts using Turco followed by vacuuming reduced gamma by a factor of 6 and beta by 2. Other methods tested whose effectiveness has not been fully evaluated to date include: spray and strip of strippable coating, high pressure water spray and high pressure Freon 113 spray.

John Johnson (Exxon Nuclear, Idaho) discussed decontamination of the Idaho Chemical Processing Plant. He noted that they had good success with Methyl chloroform for removal of organic films. Radiac wash was found useful for painted concrete surfaces, baked oxide films require use of caustic permanganate followed by strong acid and then sand blasting for removal. Water and chemical spray systems are used for removing external deposits in fuel cells in addition to water lances and long handle brushing.

Ray King (Battelle) discussed decontamination of a plutonium storage facility at Hanford. Strippable coatings (with cheese cloth for vertical surfaces) were used to prevent spreading of plutonium contamination. DF's greater than 100 were achieved for non-fixed contamination. A concrete spaller was used to decontaminate bare concrete surfaces with a removal rate of about 100 ft^2 per hour. Backup procedures were used throughout.

Lyle Perrigo (Battelle) discussed cleanup of a loop containing ruptured fuel particles. Effective decontamination was achieved using a 5% OPG solution (composed of H_2O_2, oxalic acid, oxalates, gluconic acid, gluconates and a peroxide stabilizing reagent).

Wes Lewis of Nuclear Fuels Services discussed decontamination at the West Valley Reprocessing Plant. The following points were made: All floors are lined with stainless steel fo facilitate later decontamination. Fuel cells are decontaminated by means of systems which spray the decon solutions over the entire cell surface. The chemical components of all solutions must be analyzed (including proprietary brands) to assure they are not flammable or possibly explosive. Concrete surfaces should be painted. Contaminated concrete surfaces must be removed. Chemical solutions drive the contamination deeper into the concrete. A list of the possible chemical solutions available for decontamination of various materials was presented.

Bob Brooksbank (ORNL) discussed decontamination processes at Oak Ridge. Certification of decontaminants was stressed. During chemical decontamination of a system, for example, a proprietary chemical flush was followed by Nitric Acid and the system exploded. It is also important to plan for the processes that follow the decontamination. For example, phosphates used for cleaning will destroy the resins for later Cesium recover.

Mark Rohner (Philadelphia Electric) discussed the recent chemical decontamination of six non-regenerative heat exchangers at Peach Bottom Units 2 and 3. A system which pumped the heated chemical solution through the heat exchangers was used. Although water testing of the system indicated no leaks, leaks developed once the hot chemical solution was used. The process lasted 48 hours. The decontamination factor was 10.

A. L. (Butch) Parrish III (VEPCO) discussed decontamination of the Surry Unit 2 plant. In this operation, reactor coolant piping, which was removed in conjunction with steam generator replacement, was decontaminated using an electropolishing process.

Arden Bicker (REECO) submitted a paper covering decontamination of the Nevada Test Site. Water is used whenever possible for decontamination. Other agents used include Alcohol (Ethanol), Freon-22 for electrical equipment, caustics for oxidized ferrous materials, acids for spot removal, petroleum derivatives, hydrocarbon digesters, chelating agents and abrasives.

SESSION A

WELCOME AND INTRODUCTION

Art Carson

EPRI

Good morning and welcome to the EPRI-DOE workshop on Decontamination and Dose Reduction Technology. This is one of the first activities to be implemented under the TMI-2 Information and Examination Program being jointly sponsored by DOE, EPRI, GPU and NRC to provide initiative and support primarily for acquisition of generic technical information of value which otherwise might not be obtained in the course of TMI-2 cleanup but also for provision of generic technical information which is not immediately or readily available to GPU but which could be made so with appropriate initiative and support. This workshop is being held for the latter purpose, of course. Decontamination and dose reduction are areas in which substantial amounts of technical information of generic value have been generated but not uniformly well documented, particularly all the difficult lessons learned from prior real-life experiences. It also is an area in which TMI-2 cleanup efforts can benefit from fullest possible access to existing and potentially available technology. It was agreed among the joint program participants that exchange of technical information between those directly knowledgeable in this area and those responsible for planning and conduct of TMI-2 recovery operations should be supported and that a workshop might be the best way to accomplish it. So we have invited GPU and their representatives to present a picture of the situation at TMI-2, concentrating on the reactor building and its contents, and have tried to assemble an audience that includes people from the locations and organizations which have been involved in some of the more significant facility decontamination and personnel exposure control programs to date. After their briefing on the situation at TMI-2, we will give them a chance tomorrow to describe their related experiences, particularly their "lessons learned" in planning and implementing efforts of this kind. In regard to Thursday's discussion sessions, we have tried to set up a situation where we can have maximum interchange of information between those who have the background and those who have the responsibility for developing a specific program plan for TMI-2 cleanup.

In addition to the direct participants in our workshop sessions, on behalf of EPRI I would like to welcome the various utility representatives here today. Among our

sponsors there is a great deal of interest in TMI-2 recovery operations as a source of information on what their future action requirements might be. A number of people I'm sure are here mainly to observe rather than to get involved directly in the exchanges between GPU representatives and those with particular experience in this area. We certainly encourage that kind of participation. Our only concern is that the small group discussions on the third day do not get so large that we can't have really effective exchange. When we are setting up for those discussions, we will try to make sure that those who have the most information to bring have ample opportunity to do so. We will be requesting those who are there to observe to keep their role to just that, but that is a third-day problem. The next step in our first day program is for me to introduce Herb Feinroth of DOE who will present their perspective on this workshop.

SESSION B

OPENING REMARKS

H. Feinroth

DOE

W. Hopkins

Bechtel

HERB FEINROTH

Thank you Art -- I would like to add to Art's welcome a welcome to all of you from the Department of Energy, Nuclear Regulatory Commission and our public utilities, people who are also sponsoring this workshop. Before I give you a very brief perspective of the Department of Energy's view of this activity there seems to be a little bit of confusion in the media as to what the Department of Energy's role is in the general subject of recovery of TMI II; I thought I should clarify that. The Department of Energy and its predecessor the Atomic Energy Commission have always had a fundamental role in the technology of nuclear power and in furtherance of the objective. This past summer we started to explore activities with NRC and GPU, those kinds of activities which would basically further the development of nuclear technology specifically in the area of recovery of plants in action. In addition, the other objective of our activities in this area is to learn as much as we could that would be of value to the safety of future reactors. We are in the process of entering into an agreement with both the GPU and NRC research arms to carry out a series of activities in the interim--this being one of them. This function here has two basic purposes. One is to record for future owners of nuclear plants those aspects of recovery technology which the value should further accidents occur. As a general rule, when an emergency occurs or an accident occurs, or clean-up problems occur, in the heat of the moment you generally don't record those good things or those bad things that you learn as you go through the operation. So it's our intent and EPRI's intent as your sponsors, to try to end up from this three-day meeting with a pretty good record of the technology of recovery as being learned by many others. We feel that it will be a valuable contribution to the business.

The second basic purpose is to allow the GPU Company direct access to that information during the course of these three days, so that they can benefit now when they need the information to the extent that is possible to implement some of the lessons learned in their own planning operation. We do not have, as some in the media have represented, any intent whatsoever of shouldering the responsibility of the recovery operation. GPU has that responsibility and has accepted that responsibility and that is the way it is. So that is sort of the background of our involvement in this particular meeting. I did want to mention there are two senior NRC people here who are involved in trying to assist in learning as much as we can from this activity.

In addition, GPU and their contractors are co-sponsors of this activity. I would now like to introduce Bill Hopkins from Bechtel Corporation who will chair today's session.

BILL HOPKINS

On behalf of GPU, Met Edison, I want to welcome you all this morning. The members of this audience represent the best minds in the country. To those minds I would like to propose a challenge. As a technical specialist myself, I'm always intrigued by the ultimate challenge. I think you will find as we go through the program this morning that for those of you involved in decontamination and dose reduction, that TMI represents the ultimate challenge to the technical specialist. I hope as we go through the program, the unique problems of TMI will intrigue your imagination and represent a challenge to all of you. Getting right along, I would like to go into a general review of the TMI II Plant's overall layout and design and introduce to you Jack Devine who is the recovery engineering manager for GPU for TMI II. Jack.....

SESSION C

GENERAL REVIEW OF TMI-2 PLANT

LAYOUT AND DESIGN

Jack Devine

GPU

Good morning - Since I am the first GPU speaker, let me take this opportunity for GPU to welcome you all here. I hope that it will be a profitable time for everyone. I think it is a rare opportunity to get everyone together to face the problem and look at it specifically in detail. We repeat to you that we are certainly going to try to be as helpful as we can about telling you what is going on. I know that Bill has done a lot of work trying to orchestrate the whole thing and getting our presentations together so I hope that they will be of value. I have been asked to give a brief orientation to TMI site. I don't want to insult anyone's intelligence. I think we probably have a random group here in terms of familiarity with our plant design. Rather than spending a couple of hours giving you a full indoctrination to the TMI site, I think it would probably be more profitable just to spend a few minutes to describe the site in overview, describe both Units I and Units II and give you some picture of what the site looks like, what some of the unique features about Three Mile Island are, and then we will proceed into the other speakers who will go into specific details.

Slide 1 shows Three Mile Island, Units 1 & 2. North is towards you. The Three Mile Island site and the generating station occupy I would say roughly one half of Three Mile Island itself. The background of the property is along the line that I'm showing with my pointer right here. It is a two unit site. Newspaper reports about the incident have described the units as identical. That is not true at all. Both units have Babcock and Wilcox Nuclear Steam Supply Systems. However, they were designed at different times by different architect engineers. For the most part, components such as the turbine-generators and condensers, etc., were built by different people. The two units are also largely independent, however, there are a few shared facilities.

First is the fuel handling building contained in adjacent but separate buildings. They share the same air space and the same loading area. Secondly, a number of radioactive waste handling facilities are currently shared. Thirdly, there are some secondary facilities like administration which are shared. And at the moment we're trying to establish separate and independent radwaste facilities so that we can expect to clean up TMI Unit 2 without interrupting preparations for operations of Unit 1.

For orientation purposes when you come into the Island, I think you probably all will be visiting the Island, you'll probably be coming in through the North Gate, which is the work area where most of us have trailers with access to the plant on the West side.

Slide 2 shows the Three Mile Island Two Reactor Building, a 130 ft. diameter, 200 ft. tall post-tensioned reinforced concrete structure. The auxiliary building for Unit 2 where a great deal of the decon work has been done are also shown on the slide. The control and service building is to the east of the plant and is where the plant control room is along with some supporting services. The Turbine Building is this large structure to the south of the Reactor Building. The main access to the Reactor Building is to the east. The Generator Building for Emergency Services is adjacent to the Auxiliary Building where the Fuel Handling Building is also shown. Three Mile Island Unit 1 as we pointed out in the photograph is this direction to the north.

That is a brief layout of the plant. Let me give you a little overview information about the plant. I think I mentioned that the two units are separate. Unit 1 has been on the line since 1974, it's rated as about an 800 megawatt electric plant. It's had one of the highest productivity factors of any nuclear power plant in the country; it's been an extremely successful unit. Unit 2 has been on the line, in commercial operation status, since Dec. 30, 1978, so it was on the line substantially less than a year at the time of the accident. It has a thermal megawatt capacity of 2,770 thermal megawatts. The electrical capacity is about 960 megawatts electric.

A few features about the plant that are unique to the Three Mile Island site. First, the entire site is surrounded by a dike which is a major licensing feature and which as a matter of interest has already been pressed into service. Substantial floods in 1972 came within a few feet of the top of that dike which is designed as a 1,000 year flood dike. Even though the dike was not then completed, there was no flooding on the site.

Secondly, the proximity of the site to the Harrisburg Airport was a significant licensing problem during the design of the plant. All Class 1 structures in the plant were designed to be aircraft proof. They are designed for direct impact of a 200,000 pound aircraft, which is a large commercial aircraft at 200 knots directly in the plant. That includes the Reactor Building, the Control Building, the Auxiliary Building,

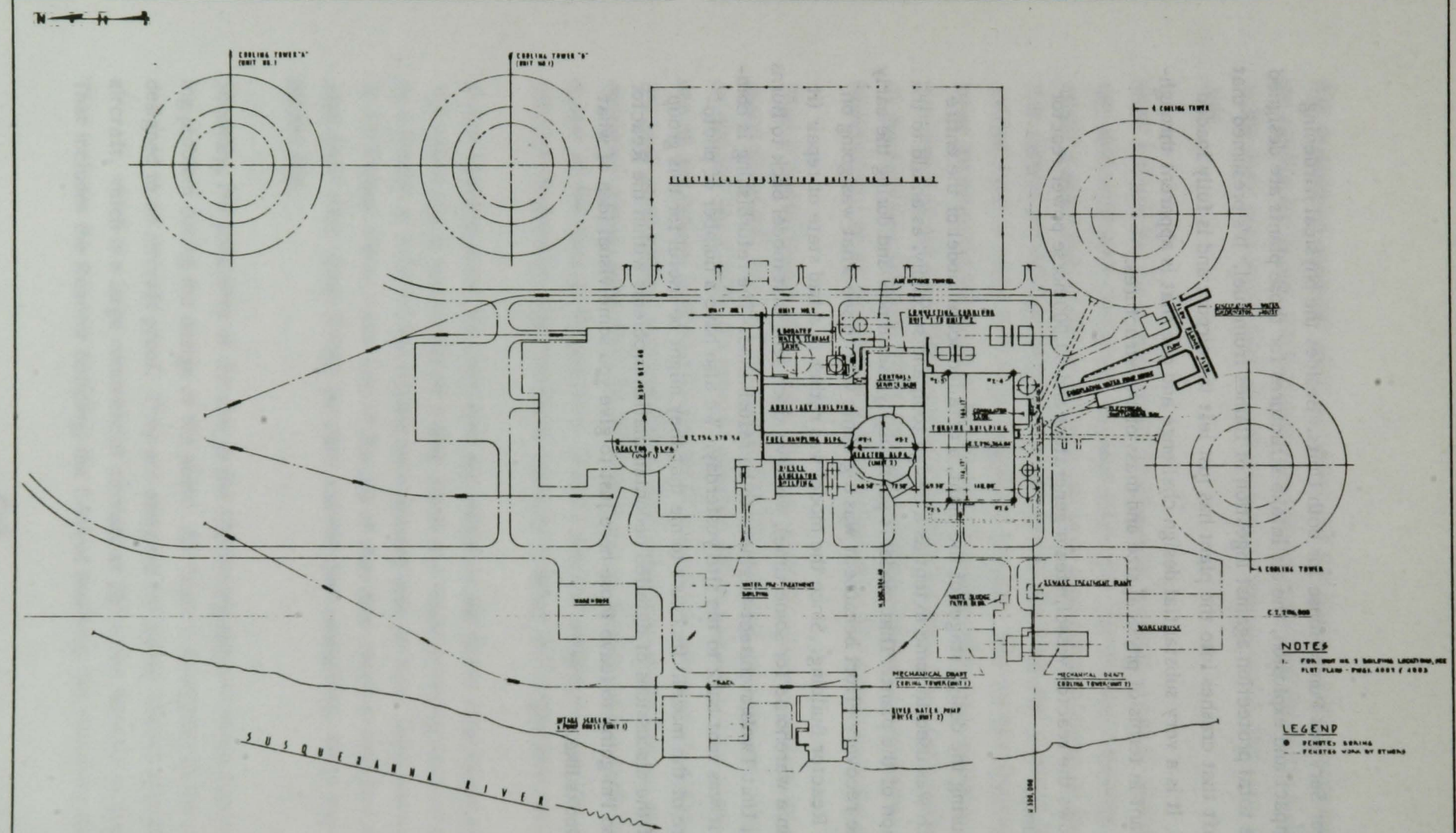
the Nuclear Services Pump House for both Units. Besides the physical hardening against impact and explosion, the air intake structures for those plants are designed to provide total protection against ingestion of flames from fuel. It's presumed that the aircraft that crashes into the plant has just left the airport and is fully loaded with fuel. It is a very substantial design challenge and one that is apparent throughout the plant in terms of physical size and mass of the structures.

Slide 3 shows the reactor vessel, steam generators and 69,000 horse power reactor coolant pump.

We had, during the design stages of the plant, a detailed scale model of the entire plant which was used to some extent as a design tool; and secondly, as an aid to the construction of the plant. That model is particularly useful now and during the early days of the recovery effort because it was a graphic way to see what was going on inside the Reactor Building. Since the model was rather in a bad state of repair (it had been in a warehouse for some time), we have since sent that model back to Burns and Rowe, Unit Two's architect-engineer to be refurbished. The refurbishing is complete and it was sent back to the site yesterday. We also have a number of photographs here of the model. It occurs to me that they might be useful for this group in some of the discussions of detailed decontamination problems within the Reactor Building, so I'm going to leave those here just to give you some visual idea of what the plant looks like.

SLIDE 1

C-4



BURNS AND ROE, INC.
ENGINEERS AND CONSTRUCTORS
HEMPSTEAD, N.Y. LOS ANGELES, CALIF.

SITE PLAN

JERSEY CENTRAL POWER & LIGHT CO.
THREE MILE ISLAND STATION UNIT-2

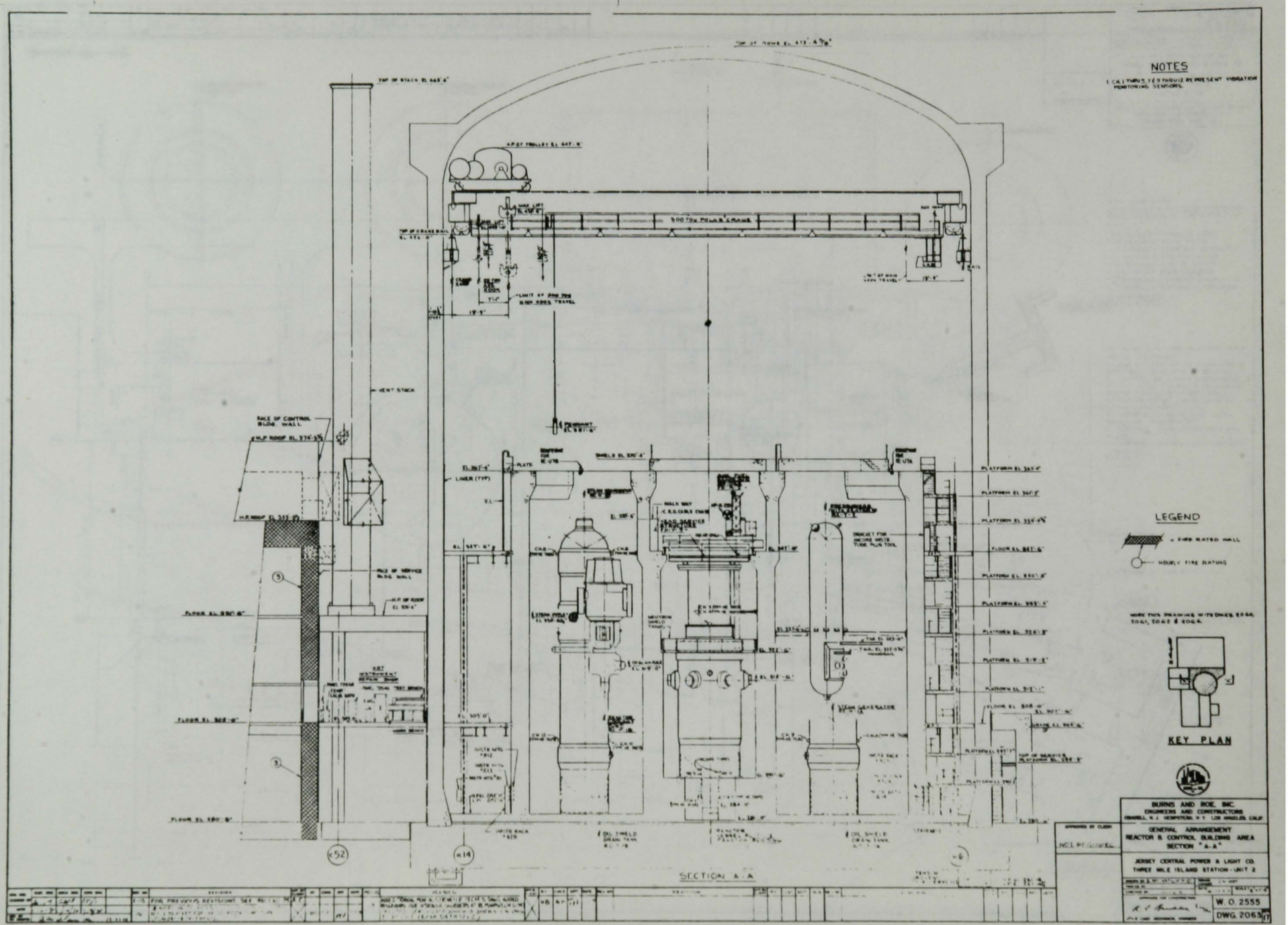
W. O. 2555

DWG. 4001

SLIDE 2

C-5

SLIDE 3
C-6


NOTES
1 (4,) THRU 5 (9 THRU 12 REPRESENT VIBRATION MONITORING SENSORS.
LEGEND
= FIRE RATED WALL
HOURLY FIRE RATING
KEY PLAN
SECTION A-A
BURNS AND ROE, INC.
ENGINEERS AND CONSTRUCTORS
GENERAL ARRANGEMENT
REACTOR & CONTROL BUILDING AREA
SECTION "A-A"
JERSEY CENTRAL POWER & LIGHT CO.
THREE MILE ISLAND STATION - UNIT 2
W. O. 2555
DWG. 2063

SESSION D

THE CONTAINMENT DESCRIPTION AND RADIONUCLEAR STATUS

W. Hopkins
BECHTEL

What I want to discuss is the containment general nuclear status as we estimated it about two or three weeks ago. Later on, Ed Walker in the 11:15 presentation will give you the hot-off-the-press numbers as we know them to be. These slides show our estimates about two or three weeks ago as to what we would find in the containment. I'd like to mention to you that Jack Daniels, Ed Walker, Mike Morell, Paul Ruhter and I will essentially all be speaking in behalf of the containment assessment task force. This task force was put together early in the summer. Its primary function was to put a systematic and planned experimental program into operation that will recover data necessary for planned re-entry and assessment of the Radio-nuclear status and the physical status of the containment. The containment assessment task force periodically issues TDR's, Technical Data Reports. Right now I know there have been two issued and one is in press and the idea is that these are journal article quality reports that will be made available after they have been reviewed internally within GPU and Bechtel for distribution to the public.

What I am going to do is take you through a walking tour from the basement on up and try to acquaint you with some of the jargon that people will be using this morning as to what is used in containment to locate different penetrations. Jack has already described the containment in general as to its size; I'll have another slide that will give you actual dimensions in terms of cubic feet. Slide 1 shows the basement, (it's the 282 elevation). The elevation signifies the highest level. What is shown is the reactor coolant drain tank, also called a Quench Tank. This is where the rupture disk blew. The rupture disk goes through a penetration in the wall in this compartment so we expect some of the highest contamination to be centered roughly in this area right around the quench tank. There is also an open stair well that transcends all three elevations of the containment that we expect where some chimney convection effects that sent the contamination to the upper elevations. On one side there is a closed stair well in reinforced concrete and an elevator. Going around the building, here's the reactor cavity where the pressure vessel is; it is inside the primary biological shield. The outer walls of the steam generator compartment are called the "D-Rings." Next are the letdown heat exchangers and some of the electrical equipment along the bottom floor. Also, to give you a general idea on the next level, elevation 305, is personnel airlock No. 2. On the west side of the building at 305 is the equipment hatch with another

personnel airlock through it. The reason I point out to you these particular landmarks is that we have been using penetrations in the containment, also called the Reactor Building, to do experiments with. There was one penetration, No. R605 located about 9 o'clock at el. 292' that we did an experiment in. There was another one R401 that was located about 1 o'clock where the sump sample was taken. These are the two penetrations we got to first with our experimental packages. There is one thing that I would like to clear up that I have read in different technical journals and the trade magazines. They have reported that we had taken a measurement and obtained a sump sample by means of drilling the containment. The containment is four feet of concrete reinforced structure and we did not drill the containment. There are penetrations, which are piped sleeves. They vary anywhere from roughly a foot to some as large as two feet in diameter through which the electrical instrumentation cabling and mechanical penetrations for pipe and steam go through the reactor building into the control room or the turbine building. So what we, in fact, did was not drill or core drill a reinforced concrete rebar containment; we actually used one of the spare penetrations. All it had was a pressure cap to permit testing the integrity of that particular penetration. A device was designed that enabled us to maintain the containment integrity and do the experiments that I will talk later on about.

Slide 2 shows elevation 305. This is the equipment hatch where we did our first GE(Li) experiment. This is the personnel airlock which penetrates the equipment hatch. Over here is the No. 2 personnel airlock. Probably one of the major contributors in the distribution of the contaminants in the containment are the containment air coolers; there are five of them. The stairwell that I talked about before spans all three elevations. Jack Daniels will be talking about our sump water, which is really like a flooded basement, and is on the order of six to seven feet deep. If one then were standing here at the edge, you could look down and look into the water.

Here's the D rings of the steam generators, the pressurizers over here. Along the axis of this personnel lock is our R-605 penetration which it was down below on 282 level. The R-401 penetration from which one of the sump samples was taken is over in this vicinity. There are several radiation monitors that feed to a strip chart. One of these was HPR 212; it's a GM-tube monitor. It was turned off early on in the accident to preserve quench gas. We reactivated this monitor back in

late August and it's been tracking and we believe the results are reasonably accurate. It is reading around 800 to 1,000 MR per hour, in this general vicinity. It's located on the 305 ft elevation.

Slide 3 shows the 347 foot elevation also called the operating deck level. I guess for those of you who kept track of the accident, on top of the elevator and the concrete enclosed stair well is the HPR214 "Dome Monitor" at elevation 374'. Over here is one of the equipment floor hatches in which you can lower equipment down through, and these are two refueling bridges on each end of the refueling canal. The refueling canal is stainless steel lined. There are two radiation monitors on the refueling bridges. Early on, these radiation monitors did track the radiation in this area.

The R626 penetration is where we did the B & W Peep Show and we'll be talking about that this morning. R626 is another blind flanged penetration that was originally designed to access the incore instrumentation seal table area right here. It's through this penetration that we've done other experiments also besides the B & W Peep show.

Slide 4 shows the 282 ft elevation and we're roughly flooded this far with water up to about the center line on the reactor coolant drain tank. This is the personnel airlock No. 2 at elevation 305'. Here is the 347 ft elevation. The R626 penetration is at 11 feet off the floor which is at 358' and overlooks this general area. For those of you who aren't familiar with the way the plant is laid out, you have a seal ring that would go in here at the elevation of the RPV flange. One would normally flood this refueling cavity with water during fuel transfer and the fuel would then be transferred into the fuel handling machine and through these fuel transfer tubes.

Slide 5 is just another elevation view looking from another direction and it happens to pick up the pressurizer and also picks up the stair cases that one would use to traverse between the different operating levels 305, 347 and 282. Up here also is the polar crane.

Slide 6 gives a rundown of some of the surface areas that we are now looking at in the containment. We divide it by elevation, by vertical and horizontal areas. We expect the contamination to be preferentially laying on the floor. The overall

grand total for our first cut on all the different surfaces (this could be roughly 50% low) is around 240,000 square feet to decontaminate. The overall containment free volume is about 2,000,000 cubic feet. So this gives you a feel of the magnitude of the problem you're faced with. We'll be talking later on this afternoon about the different types of surfaces. We did have epoxy paint on most of the containment walls. The refueling liner is stainless steel, so in terms of the type of decontamination techniques to be used, different surfaces will use different decontaminants.

Slide 7 shows a brain-child of several people. In actual practice it was a collimator that was fabricated by Jim Cline and his crew. This was the device we used along with a Teletector to scan the various penetrations that I talked about before: the ones that overlook the sump water, the one in front of the equipment hatch. There was another smaller NaI(Tl) device that was put in the R626 penetration that was used on the operating deck. This collimator, however, housed a germanium lithium drifted crystal. It was designed to have the Dewar flask below and it was hooked up to a portable multi-channel analyzer with in-situ spectrum stripping capability. We took that data back from the various experimental penetrations points and reduced that into plate out-dose-rate estimates.

Slide 8 shows a cross section. You can see that this is some part of a hexagon shape; it's lead and it's heavy. It was uniquely designed to have a "watermelon like" plug that can be pulled out, and that plug had different diameters for collimation effects. They were from "wide open" to as small as a pencil lead. We used primarily the one centimeter opening and the "wide open" opening. When this device was first envisioned, we were afraid of saturating the electronics on the counter but with the collimation that we were afforded here, we were able to get good counting statistics.

Slide 9 shows the equipment hatch where we first did our first Ge(Li) scan with the device you saw in the previous two slides. This is at elevation 305. Remember the equipment hatch was up in the about 2 o'clock area when you looked at the plan view of elevation 305. These numbers were readouts from an Eberline GM probe and this cross here indicates where we had the collimator angled to look into the equipment hatch. The equipment hatch, if you take a typical section here, is only an inch and five-eighths steel. So we knew we had a shot at looking through it; the question was whether or not we would saturate the electronics on the multi-channel analyzer.

The numbers shown are the dose rates for around the flange bolt circle. The hatch is about 20 feet in diameter and the various dose rates were taken on the inside of the flange area where it bolts on the outside. We normalized all of our data for the equipment hatch to the GM tube probe reading.

I mention in passing that for those of you who have seen the initial planning study of the Bechtel report, most of these figures come out of there and the rest come out of the TDR's, Technical Data Reports, that we also issued.

Slide 10 shows the different positions that we scanned with the Ge(Li) detector set up from the equipment hatch at 305'. Using these scans we can see the photo-peaks drop down and come up. This gave us an idea, for instance, if we were looking towards the containment air coolers, we could see exactly whether or not we were picking anything up that way. At that time, we didn't realize that the cooling banks on the air coolers instead of being at about 308' (3 or 4 feet off the ground) were actually about 15 feet up in the air. Therefore what we plan to do before we go in with our inital entry team is go back and take another shot. We have a table now that will tilt that Ge(Li) detector up in the air and we're going to take another shot on the containment air cooler cooling coils because we've done some theoretical calculations that predict very high dose rates possibly from plate-out on the air cooling coils. We did take several scans and the nature of the scans lead us to believe that most of the plate-out is laying on the floor, not on the D ring walls or on the equipment hatch itself which is a curved piece of steel.

We were having trouble with determining the exact level of the water in the containment by the Heise Gauge. Somebody got the brainy idea to see if we can tell how deep the water is with this device using the Cs-137 photo-peak, and with a little mathematical manipulation, we came out with the height of the water at this time was 6 1/2 feet and Heise Gauge was measuring something about 6.7 feet. We thought our measurement was just as good, if not better, than theirs was because we didn't have to worry about the atmospheric pressure. If I had a plant, I think I would have one of these little devices around. This, then, gives you a general idea of the way we used the collimator set on; Tom Menzel of GPU rigged a new table at the site shop. This new table could be tilted very carefully, and you could measure the angles at which it could tilt. So essentially we had something like a surveying instrument that could give you exact locations, a "nuclear surveying instrument."

After massaging the photo-peak data and doing all the shielding calculations, the slide 12 gives estimates of plate-out levels that we observed on 305' through the equipment hatch. We did a similar estimate of the plate-out levels on elevation 347'. However, different apparatus was used. A sodium iodide (thallium doped) crystal that was inserted through the R626 penetration, the same penetration of the B & W Peep Show went into. We used that photo-peak information to back-out these plate-outs on 347'. The technique that we used and the analytical methods were described briefly in the Bechtel Planning study. We're issuing these technical data reports and plan to give some papers at some of the upcoming conferences on these techniques. As you note in Slide 13, we made some wild stabs because you can't measure the betas off the strontium with a Ge(Li) detector. We assumed the same proportions of the plate-out we had on measurements from Oak Ridge back in August on the isotopic concentration of the various isotopes in the water at the 282' level. We just assumed that had been the same proportions for plate-out just in order to get a general feel for the beta dose fields. But overall you notice that Cs-137 predominates with its Ba-137M daughter. It's the major isotope we're faced with. The real question is, and maybe Mr. Walker has more information on this, what is the chemical nature of that isotope? This will in a large part determine your decontamination techniques.

I haven't cranked through the total curies but I think with this 10 to 15 maximum micro curies per square centimeter implies something on the order of 3 or 4 thousand curies total, that's just on 305' and 347'. We've been keeping the containment at high humidity, I believe that will be discussed in some of the unusual effects when Mr. Walker gets on. We did that to sweat the containment particularly on the underside of the surfaces, so we're getting some decontamination for free ongoing already.

Slide 14 shows an up-to-date estimate of the gamma dose rates on all elevations from the different components. This is from the contamination estimates which you say in the previous table for 305' and 347'. This 102R per hour represents the dose rate right over the water in elevation 282' with 180 micro curies per cc of Cs-137 dominating the total dose in the basement. Up on the 305' level the dose rate is dominated by that 120R per hour diminished by the effects of geometry for approximately 15 feet and attenuated by a 7 to 9 inch floor.

All the floor in there is "Q" decking if you're familiar with that terminology. It's a type of standard decking used in nuclear power plant architecture that runs between 7 to 9 inches thick because it has an up and down sort of a wave form. So if you take the top of the water at approximately 289, that gives you 16 feet distance and about a 10th value layer of concrete, that knocks the dose rate through the floor down to a 2.3 R/hr and gives the rest (0.3 R/hr) of your dose rate components from the plate-out sources that were already listed. We assumed in these analyses that we had half as much plated-out on the walls as we did on the floor. The airborne is from the krypton 85 gamma. This gives an all over dose rate of about 2.6 R/hr. These numbers are tending to run high, but I'd rather run a little bit high than low. For those of you who saw the initial planning study, you remember the 347' level numbers were anywhere from 300 to 3,000 R per hour. That was because at the time when the planning study was issued, we normalized all of our numbers to the readout on the 214 "dome monitor" which at the time was inside a lead collimator reading 40R per hour. That was the only piece of data that we had directly from the 347 elevation. The monitor had not gone into its characteristic failure mode which is oscillation of the control room read-out dial. Victoreen had seen that type of failure experience before in hot cells. Since we hadn't seen that failure indication at the time that the planning study was issued, we decided to err on the conservative side and safe side when it came to exposures. So we normalized to the 40R per hour that the monitor was indicating. Sure enough, as fate would have it, six weeks later the oscillation did evidence itself and we knew at that time that the 40R per hour value was questionable again. There is a lot of discussion going on that 214 dome monitor and I know it's going to be one of the pieces of equipment that will be studied in depth as to what it really meant.

These numbers here were synthesized with the sodium iodide experiment in the R626 penetration normalized to the Teletector reading. For all of these penetration experiments before we put in the sodium iodide or the Ge(Li) detector, we put in a Teletector. This is a direct current GM device, it's like a fishing pole. We always normalized to that GM dose rate which double checked the calibration on the collimator. Massaging that data, we ended up with around 700 MR per hour as a gamma dose rate around 347. If you remember the polar crane that was in one of previous elevations, we must have access to the polar crane to move equipment around in that containment. This, is the dose rate at the elevation of the polar crane cab which a person has to go up and crawl into. Because he's farther away

from the floor, he's getting a little less dose rate via the geometry effect. Again, these dose rates are with the sump not drained and no purge done from the containment.

Slide 15 gives beta dose rates. You are now essentially in an infinite beta cloud in most locations in the containment. We roughtly have about 0.8 micro curie per cc of krypton 85 which represents around 50,000 curies total of krypton 85 in the containment. That gives you for all cases an infinite airborne dose rate of 210 R per hour. In calculating the dose rates which I showed you on previous slides, the plate-out sources were without the strontium levels that were estimated based upon the proportions in the sump. The plate-out is like 42 R per hour. You throw in the Strontiums and the Yttriums; it bumps that up to about 140 which has been reflected in the total dose rate. On 305 before we were to enter the containment initially through personnel air lock No. 2, this gives you from 250 to 350 R per hour beta. The 347 is essentially the same figure we show here but because it has about a 50% higher plate-out estimate, again, than the 305 level.

Slide 16 shows gamma dose rates assuming the sump is drained. Using our estimates of plate-out coefficients and a 180 μCi/cc of CS-137, we'd have about an 18 microcurie per square centimeter Cs-137 plate-out along with an 18 microcurie/ cm^2 of Ba-137m. This gives you roughly 30 to 40 microcuries per square centimeter plate-out for "the ring around the collar" or "bathtub ring" on the basement walls and floor. With the airborne krypton this gives an overall beta dose rate of roughly 700 R per hour. Again, this is assumed that the containment had not been purged. If you do a quick and dirty range calculation, you can find that you can stop the krypton beta with roughly about an eighth to a 1/4 inch rubber. I don't think it would not be, at least in my estimate, impossible to get a person into the containment with a good scuba suit and several layers of anti-C-suits on. That will be one of the topics for discussion at the workshop seminar. If you were going into that type of field, what would be your best estimate; it sounds like a good certification question for a health physicist to me.

That concludes the radionuclear status of the containment. Are there any questions?

Question:

Those readings on the hatch, were those direct?

Answer:

They were direct readings with an Eberline Probe. We have a guy who is very gymnastic and he crawled around there and took those readings with a probe. I think we cross-checked the calibration on it but I would have to refer to my experimental person, Mr. Walker, as to exactly how good that calibration was.

Question: Illegible on the tape.

Answer:

Yes, those were, if you take a survey of the field in that area, and we've been able to notice the decay. At that time, that field was dominated by barium/lanthanum. There's lot of barium/lanthanum 140 that came out along with the cesium. Those dose rates that were reported in the Bechtel initial planning study report, Chapter 2, that dose rate was overall dominated at least, I think, 50 or 60% of that total dose rate was barium/lanthanum. So now it's decayed away and I think we're seeing something around 30 MR per hour. Is that right, Ed? That's roughly where we're standing right now in terms of the equipment hatch. We're going to go back in with the Ge(Li) detector, take a scan on the banks of the containment air coolers, and we'll get another reading on that particular location right now.

Question:

You mentioned the two radiation detectors on the crane bridge, can you go into a little more detail about those?

Answer:

Well, they have failed. They got zapped out early on. The only ones that are currently activated, I failed to mention, are on the 305' level. There is one near personnel air lock No. 2 where we plan to do the initial entry, its quench gas is

shot. The other hatch, where we did the first Ge(Li) detector. We feel it's working, the question is just, in that type of mixed field, it being a GM tube, how good is that 800 MR per hour; plus or minus 50% on the efficiencies and calibrations? The dome monitor is still on and if you go into the control room, you see it sit there and oscillate with the period of about 1 second, about the 40 R per hour reading. When I reported all the results in the initial planning study, it is the dose rate inside the lead housing. The 214 Dome Monitor is an Ionization Chamber. And we also discovered about three or four weeks ago, there is an eighth inch ID hole right opposite the Ionization Chamber's active region that allows that Ionization Chamber sensitivity to low energy around 80 KEV. In June, we weren't aware of that and when we started unfolding some of the plated-out spectrums through that from early on, we were using an inch and a half of lead. That's why we are issuing a supplement to the initial planning study. It will have the actual calibrations that were furnished by Victoreen for that monitor. Some of the people I know have already been contracted by EPRI/NSAC to unfold that data with an estimate of the radioisotopic releases in the containment. That's very important. Those are the only monitors I know that are out.

Question:

On the surface area, was the 240,000 ft^2 concrete, steel etc.?

Answer:

That's everything. I was told by the people that run the numbers that could be low, but this like our first detailed cut in going over the layouts.

Question:

You mentioned that the containment walls were covered with epoxy paint, does that include the D ring wall also or just the containment itself?

Answer:

The containment has a liner plate, 3/8 inch steel liner plate, and has an epoxy coating.

Speaker again:

That reinforced masonry stairwell I told you is a bare concrete block, so that has to be like a sponge effect. That's pretty well crapped up. We did make an estimate in Chapter 2 of that planning study on what we called "hot spots" and that's one of the things we tried to model.

Question:

Do you have a diagram showing the ratio of concrete vs. metal according to the elevations?

Answer:

The best I have right now is that one figure that we had. Now what we can do for the purpose of the workshop, we could Xerox that and use that in the workshops. Don, you know more about the table than I do. Does that have it broken down to necessarily steel vs. concrete?

Don's Answer:

It's got each one identified. It's broken down to horizontal and vertical areas. You can see there's an estimate for some of the ventilation duct work, for example.

Speaker again:

For instance, the refueling and cavity walls are stainless steel. The air coolers are sheet metal. The containment dome and the steam generator walls are epoxy covered surfaces.

Question:

Are these things before or after the peep show?

Answer:

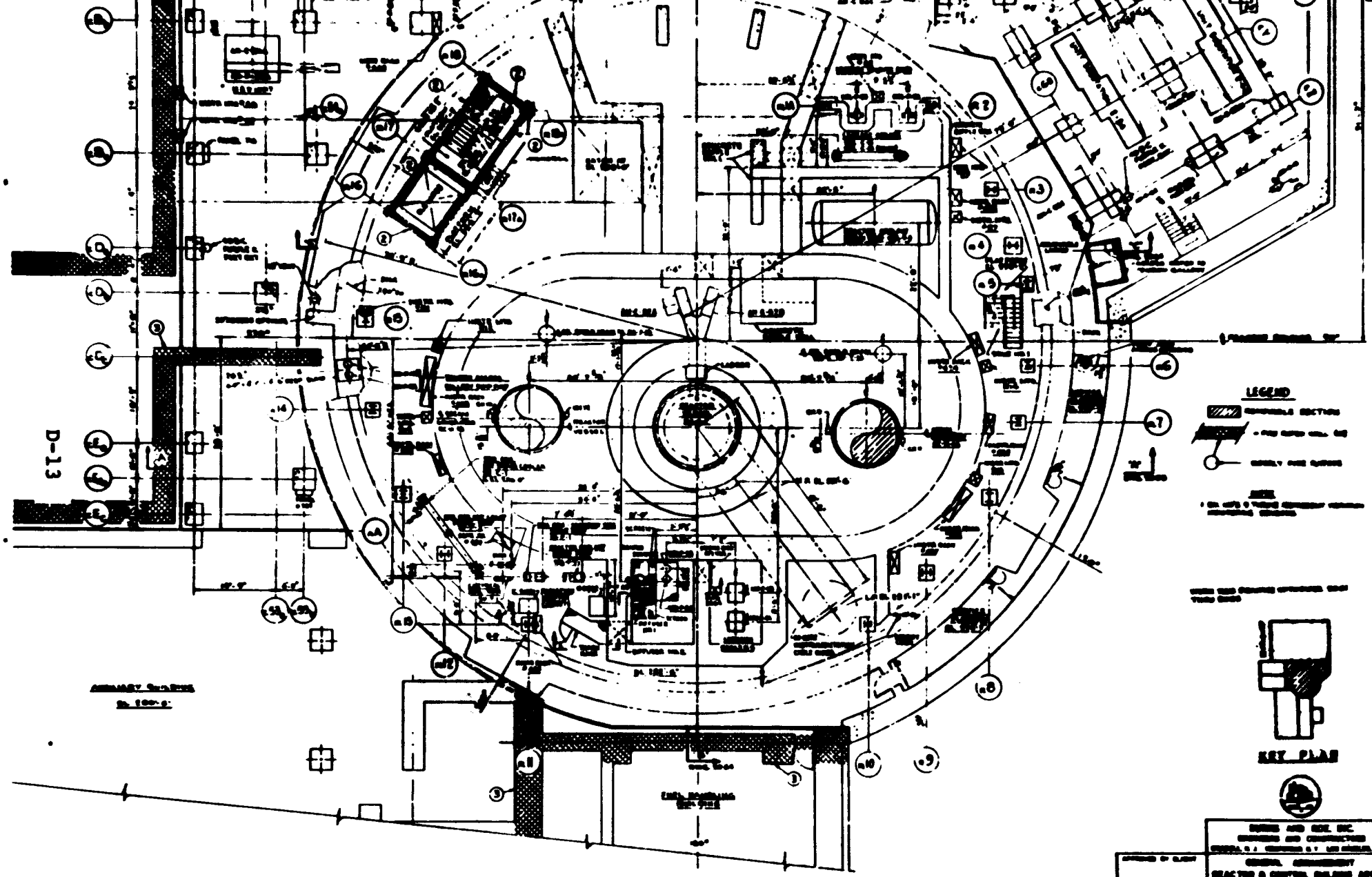
These are before the peep show and like I say Mr. Walker has some of the newest data and I think if you want to talk about numbers and how they are modified, Ed can give you a clue on exactly what we measured. I know a lot of you gentlemen have played around with experimental apparatuses. And that you know if you're going to get accuracy less than a factor of two or get down to what we ought to, i.e., 10 to 20%, it usually takes a combined bootstrap method of where you do some analysis and then recalibrate your instrumentation to the field which you expect. That's what we're in the process of doing now. Once we've got in there, we know what we've got. We've done some theoretical estimates and we're now trimming up the instrumentation which we put into the peep show.

Question:

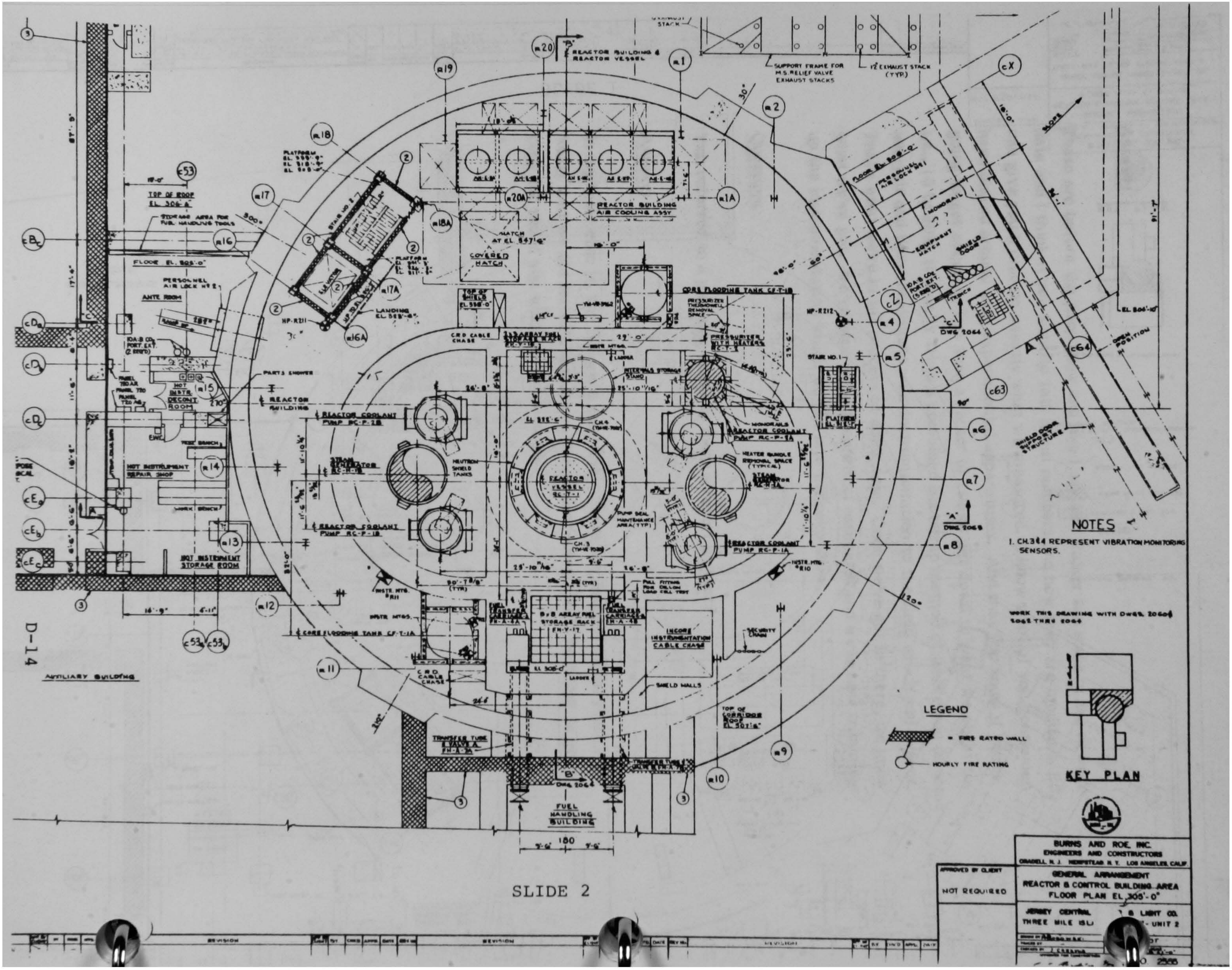
You referred to a planning study, is there a report on that?

Answer:

In my brief case is, the Bible you might say, that's the Bechtel Initial Planning Study that we did for GPU and it's in the public record. We'll be happy to get a copy for you if you will leave your name.

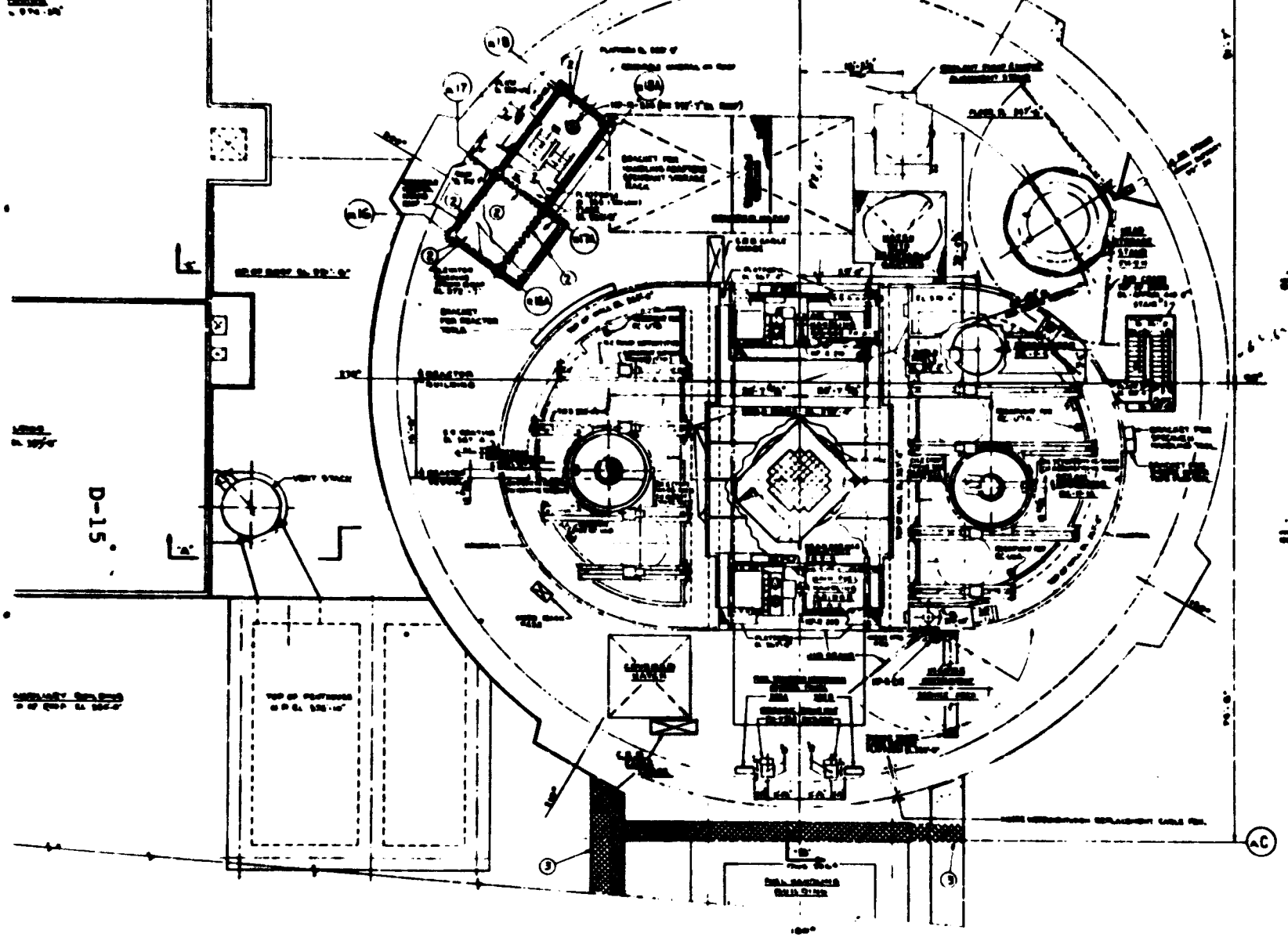


SLIDE 1

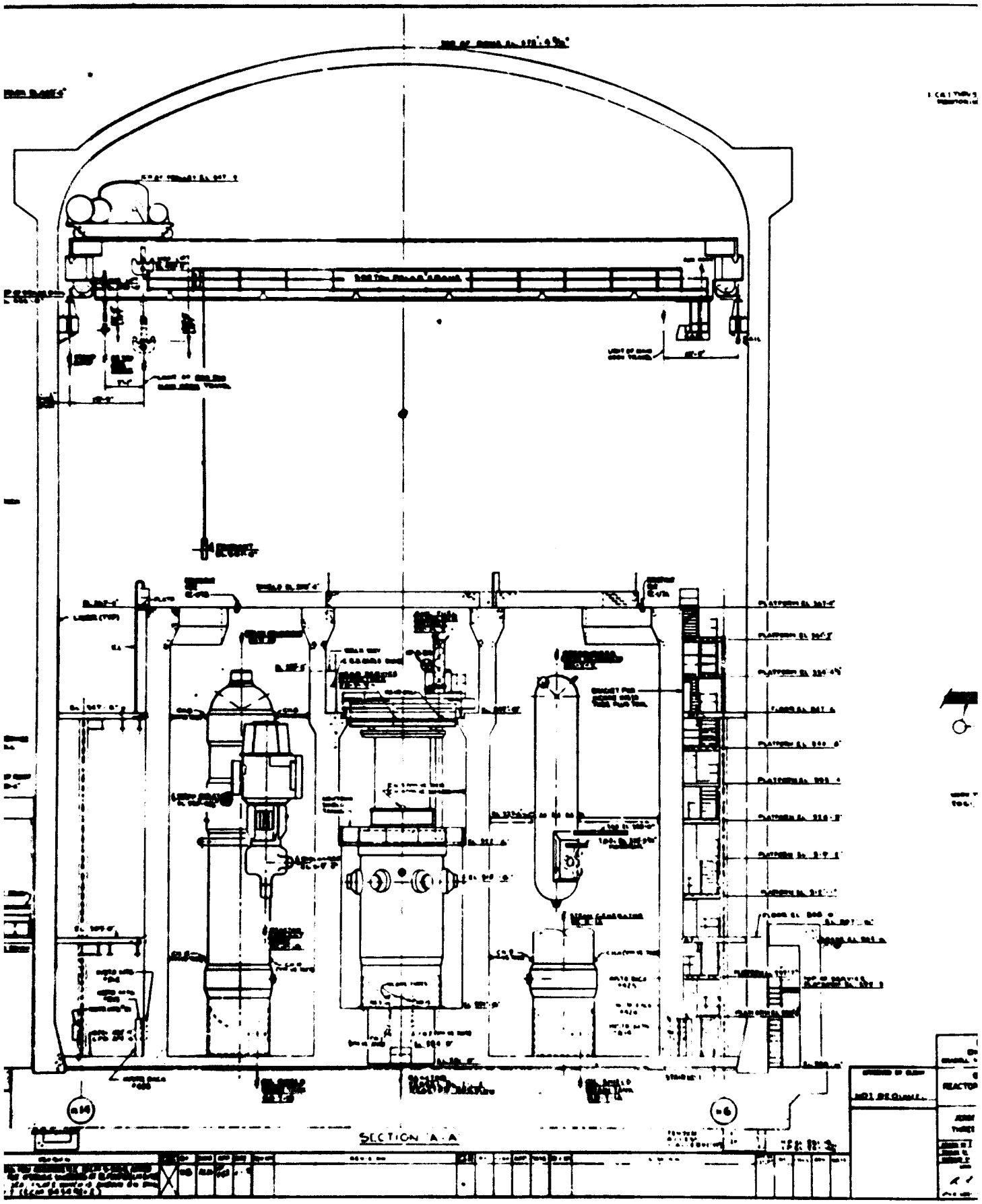

NOTES
1. CH.3&4 REPRESENT VIBRATION MONITORING SENSORS.
WORK THIS DRAWING WITH DWGS. 2060 & 2062 THRU 2064
LEGEND
= FIRE RATED WALL
HOURLY FIRE RATING
KEY PLAN
REACTOR VESSEL RC-T-1
REACTOR COOLANT PUMP RC-P-1A
REACTOR COOLANT PUMP RC-P-1B
REACTOR COOLANT PUMP RC-P-2A
REACTOR COOLANT PUMP RC-P-2B
REACTOR BUILDING AIR COOLING ASSY
CORE FLOODING TANK CF-T-1B
CORE FLOODING TANK CF-T-1A
INCORE INSTRUMENTATION CABLE CHASE
HOT INSTRUMENT REPAIR SHOP
HOT INSTRUMENT STORAGE ROOM
ANTE ROOM
AUXILIARY BUILDING
FUEL HANDLING BUILDING
SHIELD WALLS
COVERED HATCH
SUPPORT FRAME FOR M.S. RELIEF VALVE EXHAUST STACKS
12" EXHAUST STACK (TYP)
BURNS AND ROE, INC.
ENGINEERS AND CONSTRUCTORS
ORADELL, N.J. HEMPSTEAD, N.Y. LOS ANGELES, CALIF.
GENERAL ARRANGEMENT
REACTOR & CONTROL BUILDING AREA
FLOOR PLAN EL 305'-0"
APPROVED BY CLIENT
NOT REQUIRED
THREE MILE ISL

D-14

SLIDE 2



SLIDE 3



SLIDE 5

Surface Areas in the TMI-2 Containment

Elevation 347'-6" and above

Surface Areas

(Vertical)		
1	Steam Generator Walls	6,188 ft^2
2	Containment Walls	41,303 ft^2
3	Elev. Stairs Vertical Enclosures	2,150 ft^2
4	Sides of Fuel Transfer Pit	1,326 ft^2
5	Ventilation Ducting (most is vertical)	1,873 ft^2
(Horizontal)		
6	Floor Elev. 347'-6"	9,793 ft^2
7	Roof of Elev. and Stairs	325 ft^2
8	Containment Dome	28,000 ft^2

Elevation 305' to 347'-6"

Surface Areas

(Vertical)		
9	Elev. and Stairs	3,596 ft^2
10	Steam Gen. Compt. Walls (Outside)	11,288 ft^2
11	Containment Walls	17,098 ft^2
12	Air Coolers (Vert) Sides & Fan Assemblies	3,138 ft^2
13	Sides of fuel Transfer Pit	1,363 ft^2
14	Inside Surface of "D"s	5,320 ft^2
15	Outside Reactor Cavity S.G. Walls	6,193 ft^2
16	Refueling & Cavity Walls	3,943 ft^2
(Horizontal)		
17	Ceiling Elev. 346'-6" (3 times flat area)	28,416 ft^2
18	Flor Elev. 305'	9,472 ft^2
19	Refueling Floor	1,193 ft^2

Elevation 282'-6" to 304'

Surface Areas

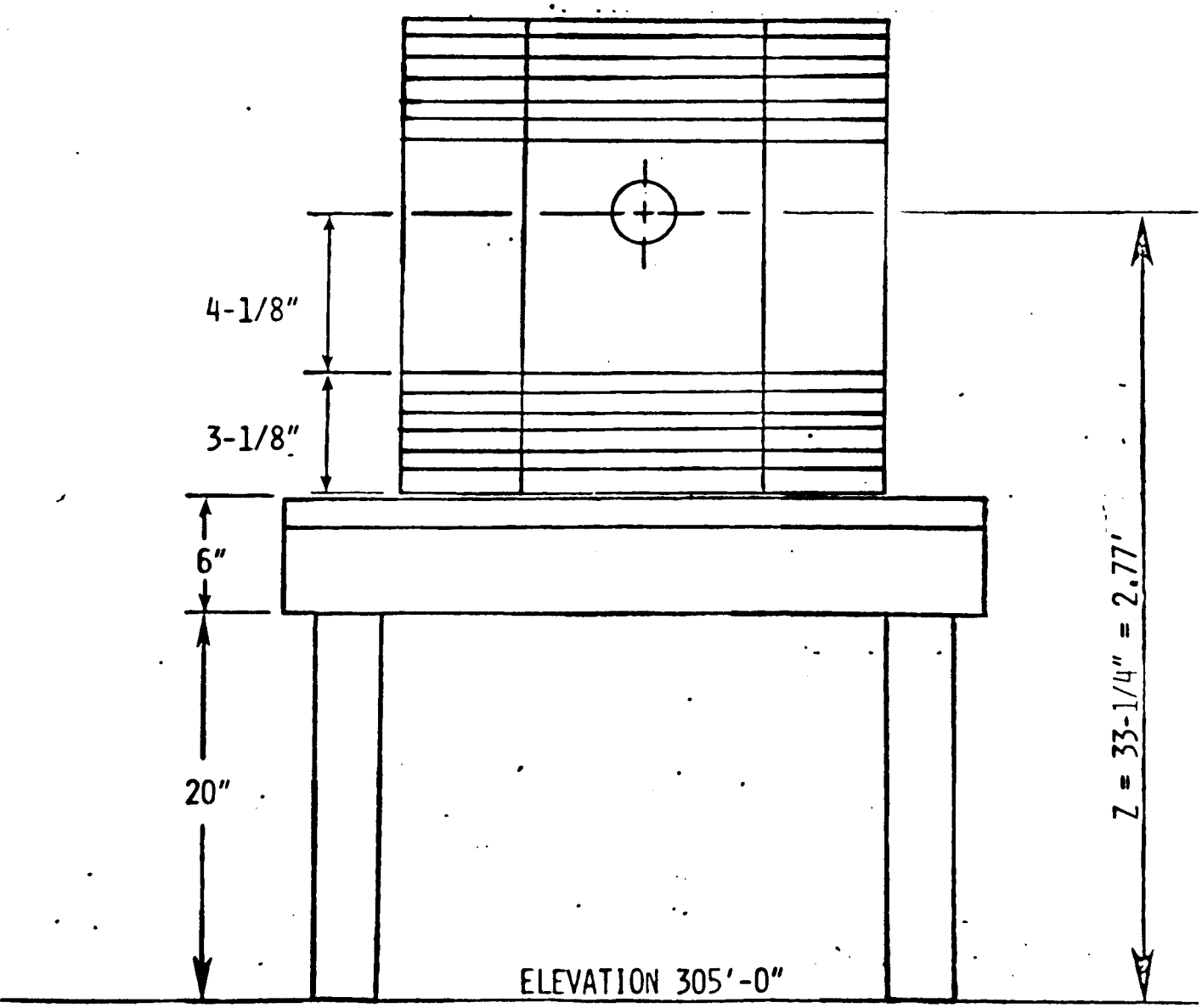
(Vertical)		
20	Walls	35,712 ft^2
(Horizontal)		
21	Floor	9,803 ft^2
22	Ceiling	8,969 ft^2

Grand Total - All Elevations 236,462 ft^2

Total Containment Free Volume 2.05×10^6 ft^3

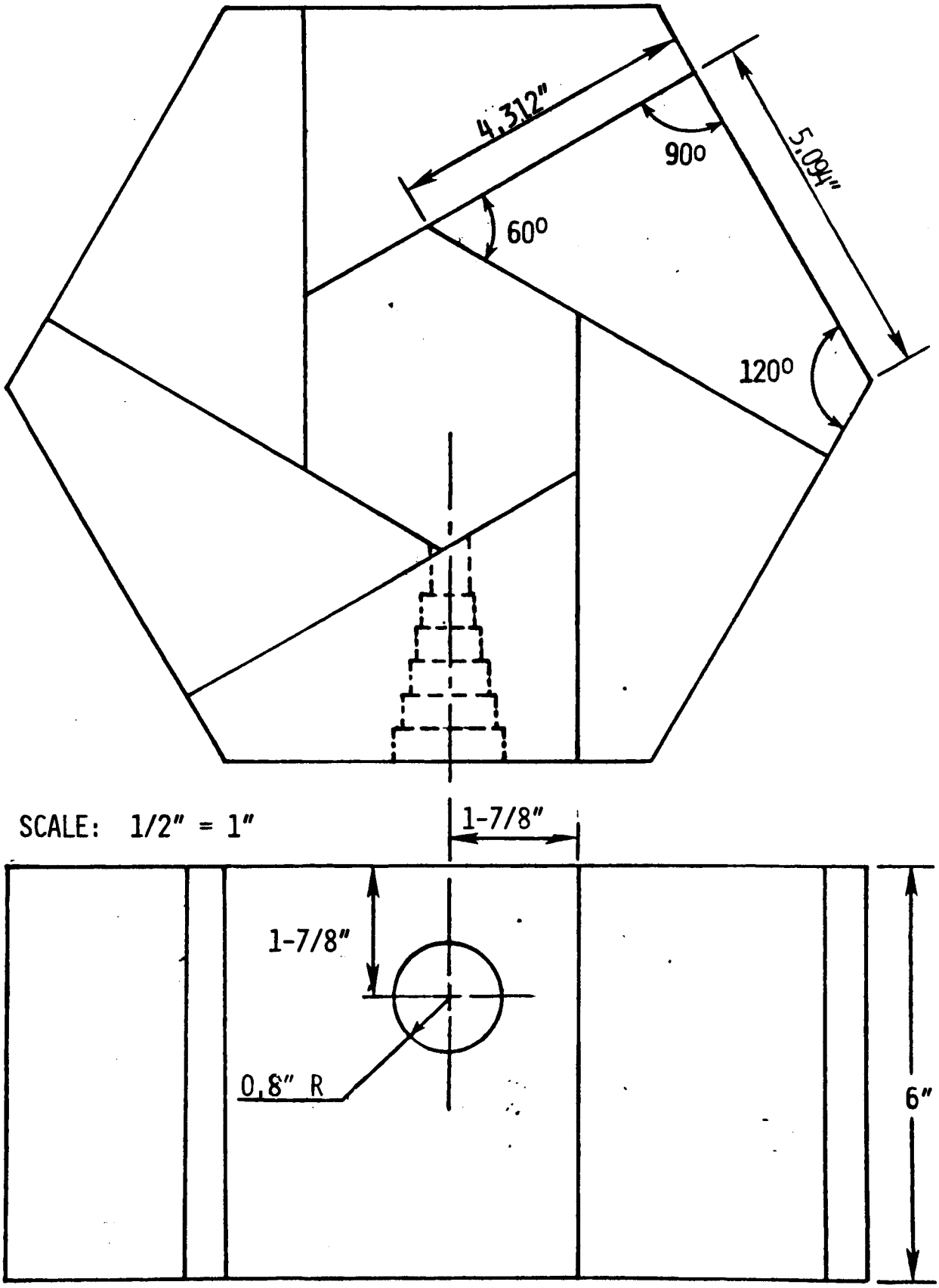
SLIDE 6

FIGURE A-6 COLLIMATOR ASSEMBLY ON SUPPORT TABLE

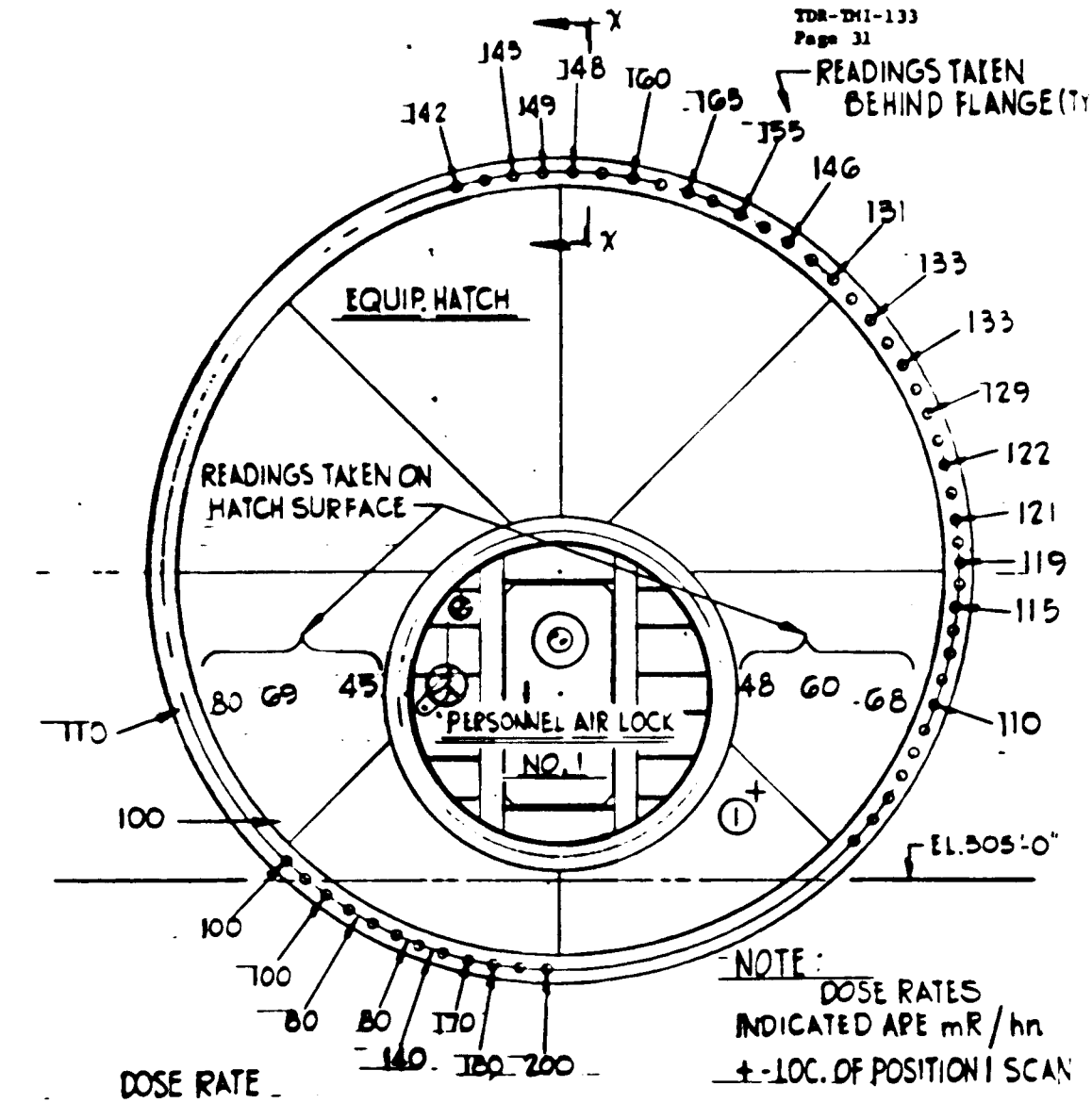


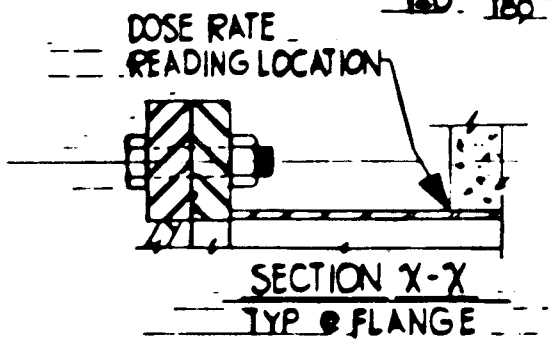
SLIDE 7

FIGURE A-2 COLLIMATOR BLOCKS - LEAD

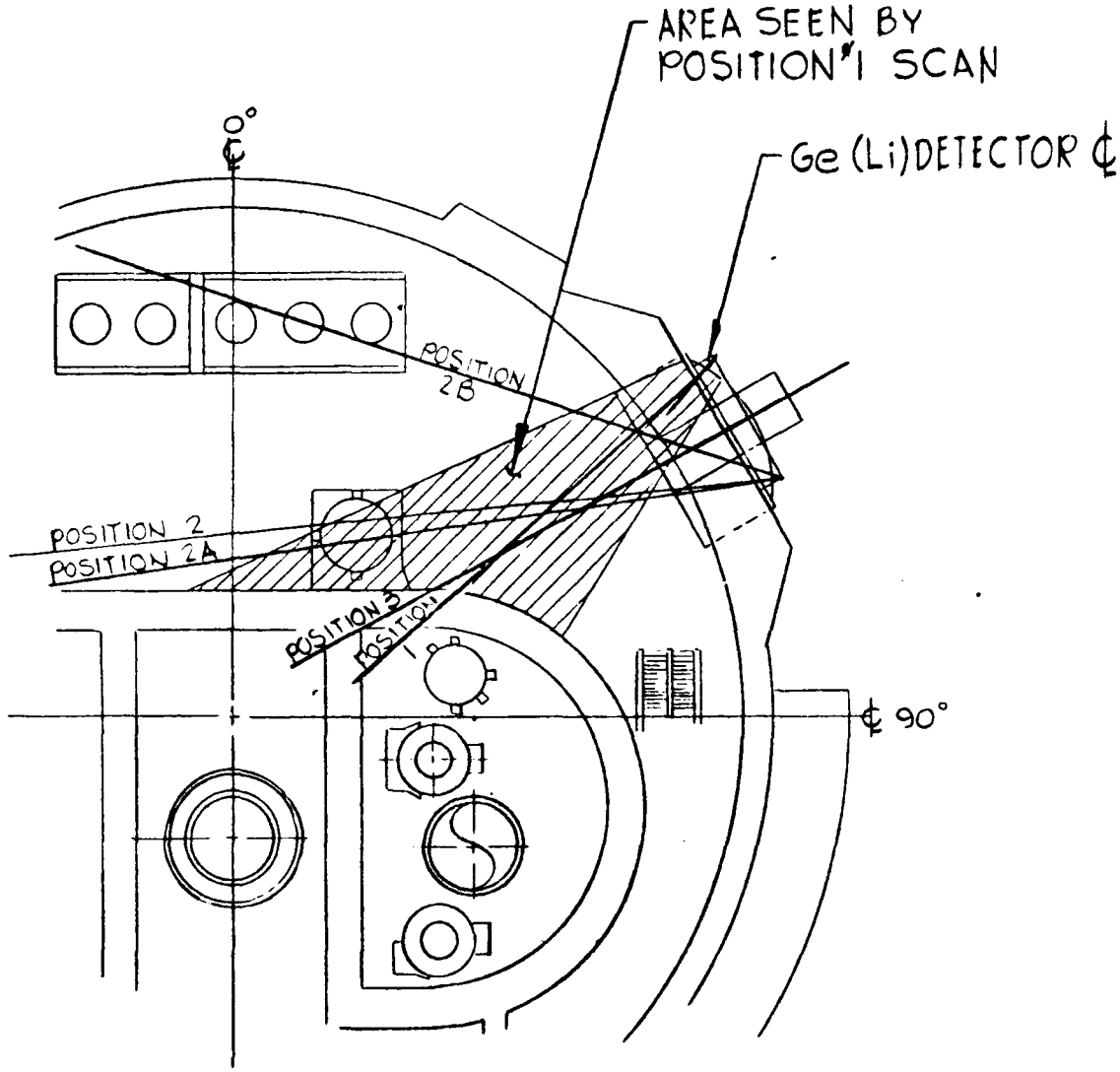


SLIDE 8





DOSE RATE MEASUREMENTS
@ EQUIPMENT HATCH
TAKEN ON JUNE 1, 1979 WITH
EBERLINE E-520 STD. GM PROBE
FIG. C-1

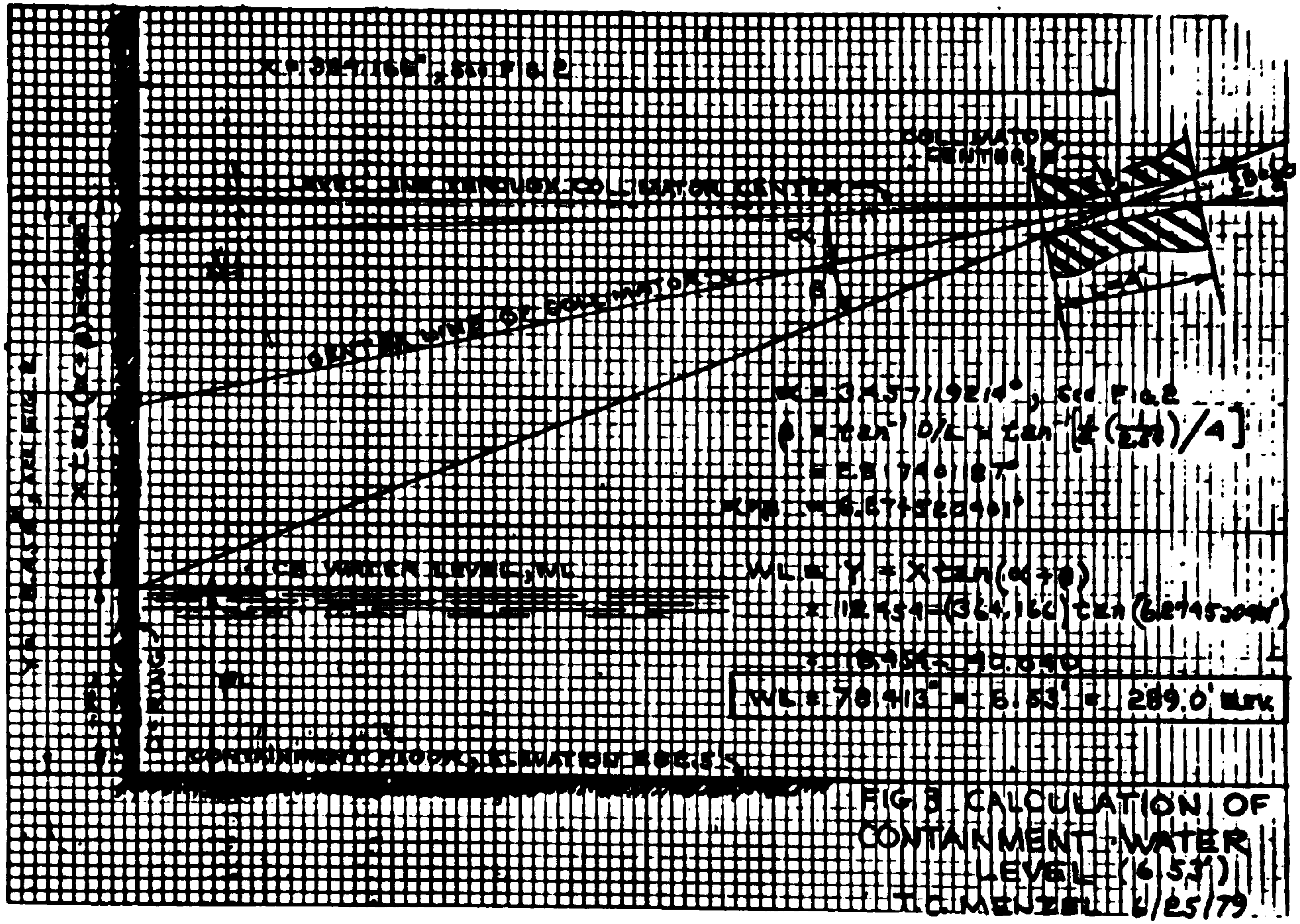


PARTIAL CTMT. PLAN
@ EL 305'-0"

AREA OF EL 305'-0" FLR. SEEN BY
Ge(Li) DETECTOR FROM POSITION 1

SLIDE 10

FIG. 2-2


COLLIMATOR CENTER
WL = 78.413" = 6.53' = 289.0' ELEV.
FIG 8 CALCULATION OF CONTAINMENT WATER LEVEL (6.53')
T.C. MENZEL 6/25/79

SLIDE 11

TABLE 2-5

EXPECTED ACTIVITY[2] AND GAMMA DOSE RATES[3] AT EQUIPMENT HATCH ASSUMING ALL ACTIVITY IS ON FLOOR ELEVATION 305'

Isotope	Major Photon Energy (Mev)	Attenuation Factor[1]	Dose Rate on Floor in Front of Hatch (mr/hr)	Activity on Floor In Front of Hatch ($\mu Ci/cm^2$)
Cs-137	0.662	3.3	95	4.0
La-140	1.596	2.5	240	4.8
I-131	0.365	4.9	79	5.8
Cs-134	0.796	2.7	29	1.1
Cs-136	1.048	2.2	4	0.13
TOTAL	-	-	443	16.

NOTES:
(1) Attenuation through 1.5" of steel

(2) Based on Position 1, Uncollimated Ge(Li) Experiment by SAI on June 1, 1979

(3) Dose rates normalized to 60 mr/hr reading from Eberline 520 Standard GM probe

(4) All numbers listed to two significant figures

CAUTION: These dose rates are due to plateout only. See Table 2-12 (A) and (B) for additional dose rate components. For hot spots see Table 2-17.

TABLE. 2-5 Estimate of Plateout Activity on Elevations 305' and 347'

Isotope	El. 305'	El. 347'
	$\mu ci/cm^2$	$\mu ci/cm^2$
Cs-137	4.0	5.8
Cs-134	1.1	1.8
Ba-137m	4.0	5.8
Sr-89(1)	0.97	1.4
Sr-90(1)	0.060	0.087
Y-90(1)	0.060	0.087
Grand Total	10.	15.

Notes:
1. Plateout activities for Sr-89, Sr-90, and Y-90 synthesized by assuming that they exist in the same proportions to Cs-137 as that observed in the ORNL analysis of the sump samples taken on 8/28/79.
2. All values listed are for two significant figures only.
3. All plateout activities keyed to 9/7/79.

SLIDE 13

TABLE 3: SUMP NOT DRAINED

GAMMA DOSE RATES AT ALL ELEVATIONS

(Rads/hr)

Elevation	Dose Rate From Floor Plateout	Dose Rate From Wall Plateout	Airborne[3] Dose Rate	Dose Rate Through Floor	Dose Rate Through Ceiling	Total Dose Rate	Post Purge Total Dose Rate
282	120.	---	0.046	---	0.0075	120.	120.
305	0.40	0.14	0.077	2.3	0.0077	2.9	2.1
347	0.6	0.14	0.22	---	---	0.96	0.7
Polar Crane	0.11	0.30	0.22	---	---	0.63	0.4

NOTES:
1. All dose rates are for 12/1/79 assuming sump is not drained, and are listed for two significant figures only.
2. Dose rate is immediately over centerline of sump, not a plateout dose rate based on ORNL analysis of specific activity.
3. Airborne dose rates assume containment has not been purged.

SLIDE 14

TABLE 2: SUMP DRAINED

BETA DOSE RATES AT ALL ELEVATIONS

(Rads/hr)

Elevation	Contact Dose Rates	Airborne Dose Rates(2)	Dose Rates Total	Post Purge Total Dose Rates
282	510.	210.	720.	510
305	42.	210.	250.	42
347	42.	210.	250.	42
Polar Crane	---	210.	210.	

NOTES: 1. All dose rates are for 12/1/79 assuming sump is drained, and are listed for two significant figures only.

2. Airborne dose rates assume containment has not been purged.

SLIDE 15

TABLE 1: SUMP DRAINED

GAMMA DOSE RATES AT ALL ELEVATIONS

(Rads/hr)

Elevation	Dose Rate From Floor Plateout	Dose Rate From Wall Plateout	Airborne[3] Dose Rate	Dose Rate Through Floor	Dose Rate Through Ceiling	Total Dose Rate	Post Purge Total Dose Rate
282	(1.9-19.)[2]	0.27	0.046	---	0.0075	(2.2-22)[2]	(2.2-22)
305	0.40	0.14	0.077	0.073	0.0077	0.70	0.6
347	0.6	0.14	0.22	---	---	0.96	0.7
olar Crane	0.11	0.30	0.22	---	---	0.63	0.4

D-28

NOTES:
1. All dose rates are for 12/1/79 assuming sump is drained, and are listed for two significant figures only.
2. Dose rates in () represent possible ranges due to uncertainty of chemical solubility of isotopes drained from the sump.
3. Airborne dose rates assume containment has not been purged.

SLIDE 16

SESSION E

NSSS DESCRIPTION AND STATUS

George Kulynych
B&W

My assignment this morning is to give you a brief overview of the reactor system highlighting some of the areas that might result in significant or unique decontamination problems. In this discussion, I intend to concentrate primarily on the external areas of the NSSS as opposed to the internal decontamination and cleanup.

I want to describe the physical description of the NSS System, the layout of the NSSS and the reactor building. I would also like to describe the materials and the surfaces that are key considerations for the cleanup.

The TMI-2 NSS System is a B&W 177 fuel assembly reactor plant. This reactor is essentially an identical NSSS to the units at Oconee, Arkansas Nuclear 1, Rancho Seco and Crystal River-3. As seen in Figure 1, the NSS consists of a reactor vessel, two once through steam generators with each steam generator loop having two reactor coolant pumps. Mounted on the reactor vessel is the head service structure with 69 control rod drive mechanisms. The other major component is the pressurizer. Not shown in Figure 1 but included as part of the NSSS are two core flooding tanks that are located in the reactor building. Figure 2 gives some idea of the size of these components and the overall system.

The steam generators are mounted at the basement floor, level 282. The overall height to the top of the candy cane pipe is about 80 ft. The steam generator is approximately 12 ft. in diameter, and 72 ft. high. It has a number of appurtenances connected to it and is insulated with metal reflective insulation. The reactor vessel is also skirt mounted and is approximately 40 ft. high and 14 ft. in diameter. Mounted on the top head is the service structure which supports the control rod drive mechanisms. The reactor coolant pumps are manufactured by Bingham and the motors by Allis Chalmers. The complete pump-motor assembly is in excess of 30 ft. high and approximately 8 ft. square. In one of the steam generator cavities is the pressurizer. The Pressurizer is approximately 8 ft. in diameter by 40 ft. high.

Looking at a plan view of the NSS System in Figure 3, it is approximately 30 ft. from the center line of the reactor to the center line of the steam generators and about 23 ft. between the center line of the two reactor coolant pumps.

There are four major areas of the reactor building:

- the shield cavity
- the D-ring cavitiies
- the outer annulus around the D-ring cavities
- refueling canal

Figure 4 is a section view of the reactor cavity and the D-rings. The reactor cavity is the area below the reactor flange and encloses the reactor vessel.

One D-ring cavity enclose a steam generator and two RC pumps. The other D-ring cavity has the steam generator, two RC pumps and pressurizer. The area outside the D-ring cavities is called the annulus area.

Figure 5 is a plan view of the refueling canal area, and laydown area for some of the equipment. Also shown are the steam generators, the four reactor coolant pumps and pressurizer. Shown outside the D-ring are two core flooding tanks.

Next, let us review the materials of the reactor coolant system that may be significant in this decontamination effort. Of the internal surfaces, approximately 80% of the surface that's exposed to reactor coolant during normal operations is Inconel. It's almost entirely the steam generator tubes in the two steam generators; in excess of 280,000 square foot of Inconel surface. Approximately 16% of the surface is the zircaloy clad fuel with the 177 fuel assemblies, each with a 15 X 15 matrix of zircaloy tubes. Approximately 8,000 square ft. or 2½% of the surface is machined stainless. This is primarily the reactor internals surfaces. About 1½% of the surface, or approximately 5,000 square ft., is the weld deposit stainless cladding material in the coolant piping, the reactor vessel, pressurizer and steam generator heads.

The external surfaces are covered by insulation and are primarily carbon steel which have been painted in the shop with aluminum paint for protection during transportation and storage. That paint is still on the components although prior to the unit going into service, there was indication of peeling of that paint. The reactor coolant pumps, control rod drive mechanisms, parts of the fuel handling masts are stainless. The reactor coolant pump motors, the fuel handling bridges and control rod drive support structure are painted with an epoxy coating.

The internal surfaces have been subject to normal PWR primary coolant water chemistry prior to the accident with boron, lithium used to control the pH, very low chlorides and essentially zero oxygen. As shown in Figure 6, since the accident, the boron level has ranged between 2,800 and 3,900 ppm boron with an average of about 3,400. There is approximately 1,000 ppm sodium in the coolant system because of the sodium hydroxide additions to control pH. pH ranged between 7.3 and 8.4 and is in the order of 7.9 most of the time. There are indications of chlorides up to approximately 4 ppm in the coolant system. Excess dissolved hydrogen has been maintained in the coolant system since the incident over the range between 10 and 45 standard cc's per kilogram with the average being on the order of 20 cc/kg. Dissolved gas measurements have not indicated any presence of dissolved oxygen. It is not expected that there is any significant amount of oxygen in the coolant.

There are several key areas that may be unique problems in decontaminating the reactor coolant system.

- the insulation on the component external surfaces
- the reactor vessel head and control rod drive mechanism service structure
- the reactor vessel supports and the incore piping
- the various component supports and restraints
- the reactor coolant pump motors
- the fuel handling equipment

The following is an illustration of the types of geometry that will be involved in the decontamination effort.

The reactor vessel insulation shown in Figure 7 is a metal refective type made up of very thin panels of stainless steel. There are approximately 20 thin layers in an assembly of about 4 inches thick. These are pre-assembled panels that are installed in the field. At TMI-2, there are stand-off's between the insulation and the reactor vessel so that you can get between the insulation system and the vessel for inservice inspection. The complete NSS loop and some of the smaller piping is covered with this insulation. Depending upon the degree of contamination of the insulation this may involve significant decontamination problems.

Figure 8 shows the reactor vessel head and the control rod drive service structure. This cylindrical structure encloses 69 control rod drive mechanisms. Figure 9 shows the nests of mechanisms including the motor tube; and the water jackets around the stators. These mechanisms are mounted on a reactor vessel nozzle that extends above the reactor vessel head. The flange has six hold-down bolts to mount the control rod drive mechanism. Depending upon the degree of contamination in that area it may require some special decon methods. It is necessary to work above this structure to uncouple the control rod drive mechanisms prior to removing the head. Therefore, it is necessary depending upon what the activity levels, to at least decontaminate this structure to some extent prior to head removal.

The reactor vessel is skirt mounted and sits on a concrete pedestal near the basement floor. The insulation system covered part of the skirt and then went under the vessel. The skirt is attached to the concrete with hold-down bolts, both on the inside and the outside of the reactor support skirt. In addition, the 52 incore penetrations at the bottom of the vessel that go out along the outside of the D-ring for the incore detectors. The water level currently is part way up to the cavity, not quite touching the bottom of the vessel. It is not up to the level of the reactor vessel skirt flange yet and it is not anticipated that it will reach that level before the water is removed.

Another major area of decontamination concern is inside the D-rings shown in Figure 12. The major components are located in the D-rings, but also located there are various piping systems such as the steam piping, the feedwater piping, and of more significance are the supports and restraints. Figures 13 and 14 illustrate simple schematics of some of the supports and restraints located in the D-rings.

There are column type supports for each of the reactor coolant pumps in Figure 13. In addition, Figure 14 shows the large supports for the steam generator that extend from the steam generators out to the D-ring walls. Not shown are supports for the reactor coolant piping and the support structure for the pressurizer. In addition, there are various work platforms around the reactor coolant pumps.

Figure 15 is an illustration of the reactor coolant pump motors. The motor along is about 20 ft. high. They are 9,000 HP, totally enclosed aircooled motors, with an integral cooling jacket. Each has an oil reservoir, an oil lift pump and other accessories. There are two of these pumps in each of the D-ring cavities.

The fuel handling equipment is shown in Figure 16. Within the canal there are two bridges in the reactor building plus the fuel transfer station that's located in the reactor building refueling canal.

Figure 17 is an illustration of the bridge part of the main fuel transfer mechanism showing the hydraulic power supplies, the mast supports and the various hose reels and cable reels associated with the fuel handling equipment. Again, there are two of these machines in the reactor building at this time.

The NSS system internal surface areas will also present significant decontamination problems. One of the more unusual ones may be reactor internals with the amount of fuel damage that's been postulated. It's quite likely that loose fuel particles are trapped in the areas between core barrel and the former plates.

If the internals are to be reused or removed, it is necessary to decontaminate and remove the loose fuel. These assumptions are based upon inspections that still have to be made once the reactor is defueled.

Other areas that will require decontamination are the internals of the control rod drive mechanisms.

That is an overview of the major areas of the NSS System involved in this decontamination effort, I would like to accept any questions at this time.

Question: What is the primary systems volume?

Answer: The total volume is about 11,800 cubic ft. including all major components.

Question: Can you show us where the stuck open relief valve is and when it discharges?

Answer: That power operated relief valve is mounted on top of the pressurizer and then it is vented with piping to a quench tank located at the basement level. The quench tank was protected with a relief valve and a rupture disk that ultimately blew and discharged the effluent out through the rupture disk.

Question: Is there a chance that some of the pumps will have to be replaced instead of decontaminated?

Answer: That's a possibility certainly; we don't know what kind of debris has been dislodged and how it's been circulated. I wouldn't think that small pieces of zircaloy tubing or small pieces of fuel would mechanically damage the pump internals. With respect to the ability to decontaminate those internals, that could be another question. Filtered water has been on the pump seals thru seal injection continuously from the time of the accident to this day. Therefore, there is flow of filtered water coming into the pump seals down into the pump cavity throughout this period. That should serve to minimize the amount of debris that would get into the close fitting portions of the pump.

Question: What about steam generator damage?

Answer: The only indication of potential steam generator damage to date is one suspected generator tube leak that showed up during the incident, although after the first day there was no indication that it continued to leak. Whether or not the thermal transient that the system underwent was enough to damage the steam generator will require analysis have not been completed yet. Other than that, we have no indication of steam generator damage.

Question: Is the primary piping insulated?

Answer: Yes it is. It's mirror insulation.

Question: What is the material on the supports and restraints?

Answer: They're all high strength carbon steel and for the most part they're painted with epoxy.

Question: Why is it necessary to decontaminate the insulation.

Answer: I'm not absolutely sure that it is necessary; it may be desirable. The program for recovery has not been worked out, but if you get into a major program of inservice inspection you may be handling those quite a bit, and so you might want to decontaminate it.

Question: Is it gamma type radiations you're trying to get rid of here or would it be beta type contamination?

Answer: Again, we're not sure what's in those panels. If it's deposition and if it's only on the outside, say it's cesium, that's both beta and gamma radiation. Depending upon how much is down inside the panels that would be the real problem.

Every time you try to remove these panels to do inservice inspection or even just with the ventilation blowing by them, you can get resuspension of the airborne radioactivity. They may be contaminated internally.

Question: Weren't the pumps shut off very early in the accident, at least two or three of them.

Answer: The pumps were shut off during the incident but one of the pumps was started some 15 hours into the accident and it continued to operate for approximately 2 weeks and then that one was shut off and another one was operated. One pump A1 or A2) operated until April 27th. The B pumps have not been operated since the incident.

FIGURE 1

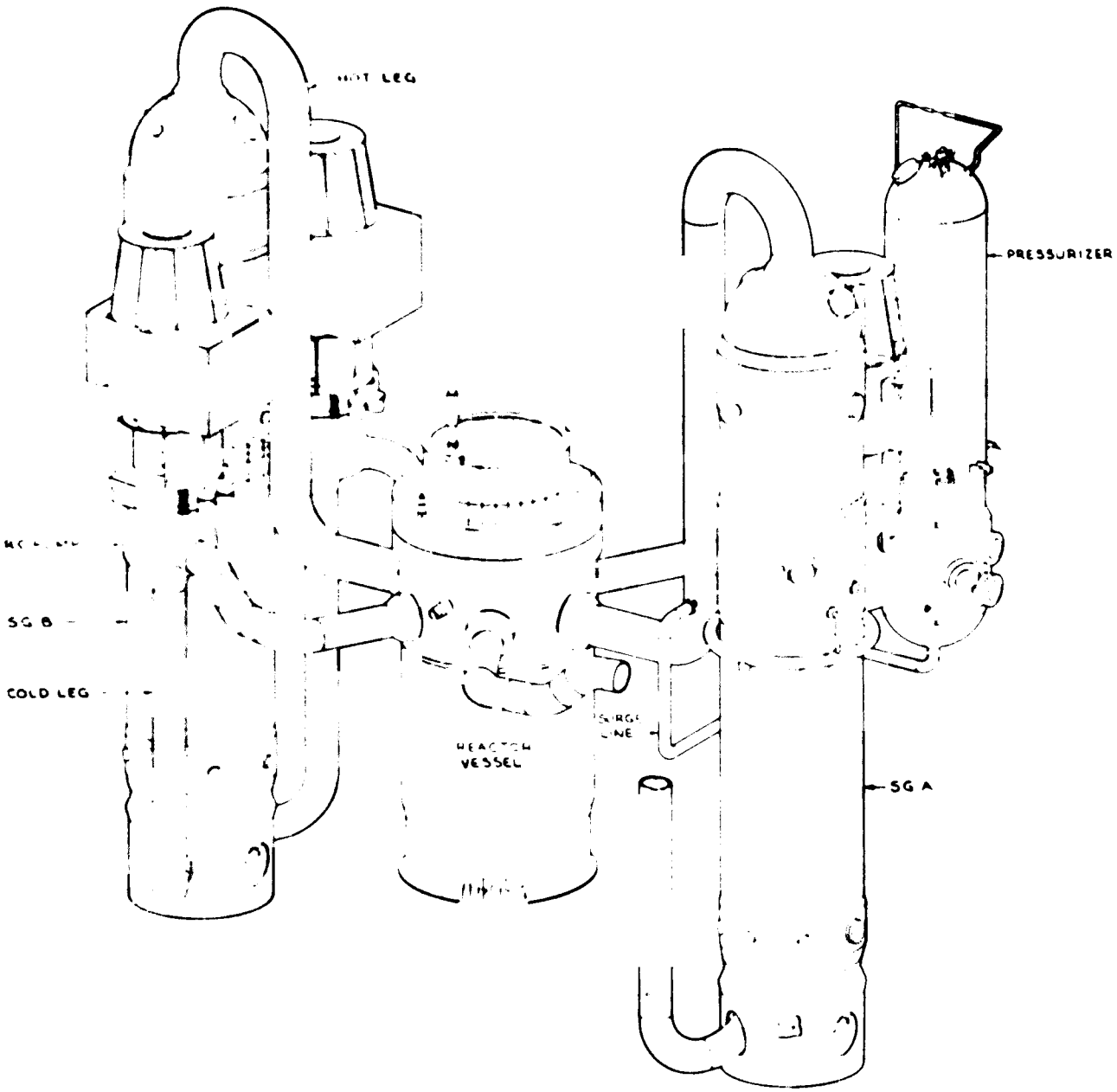


FIGURE 2

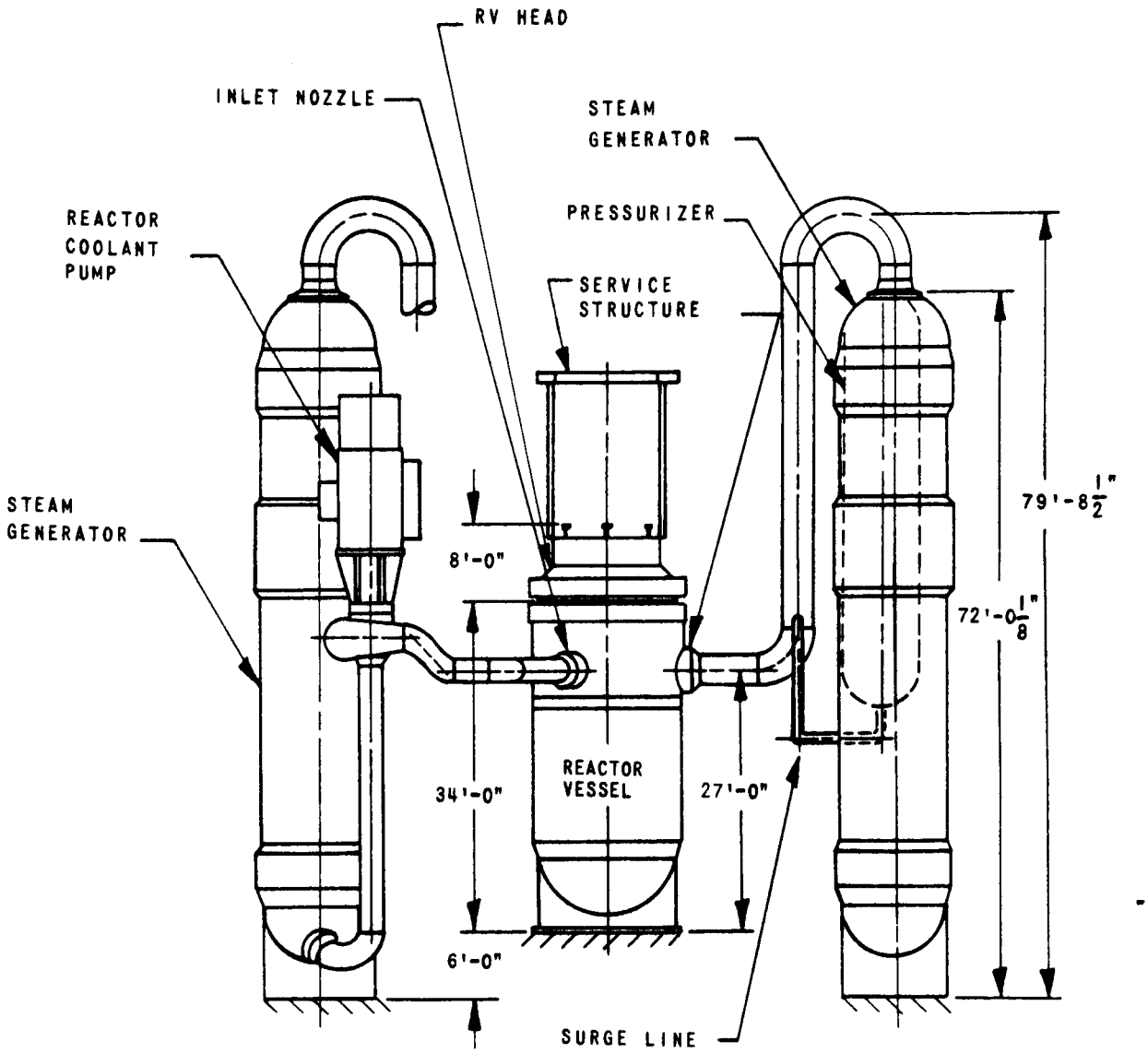
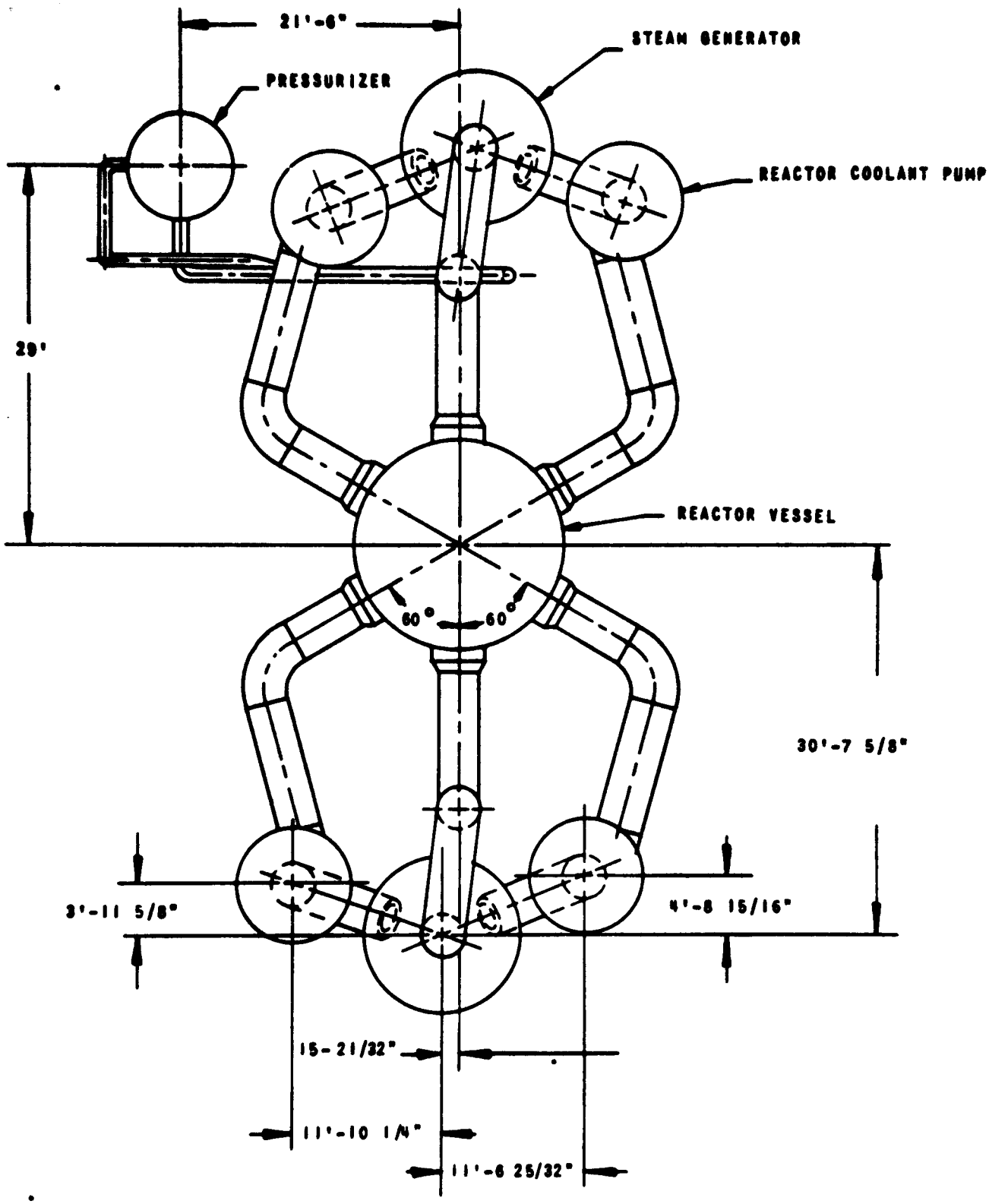
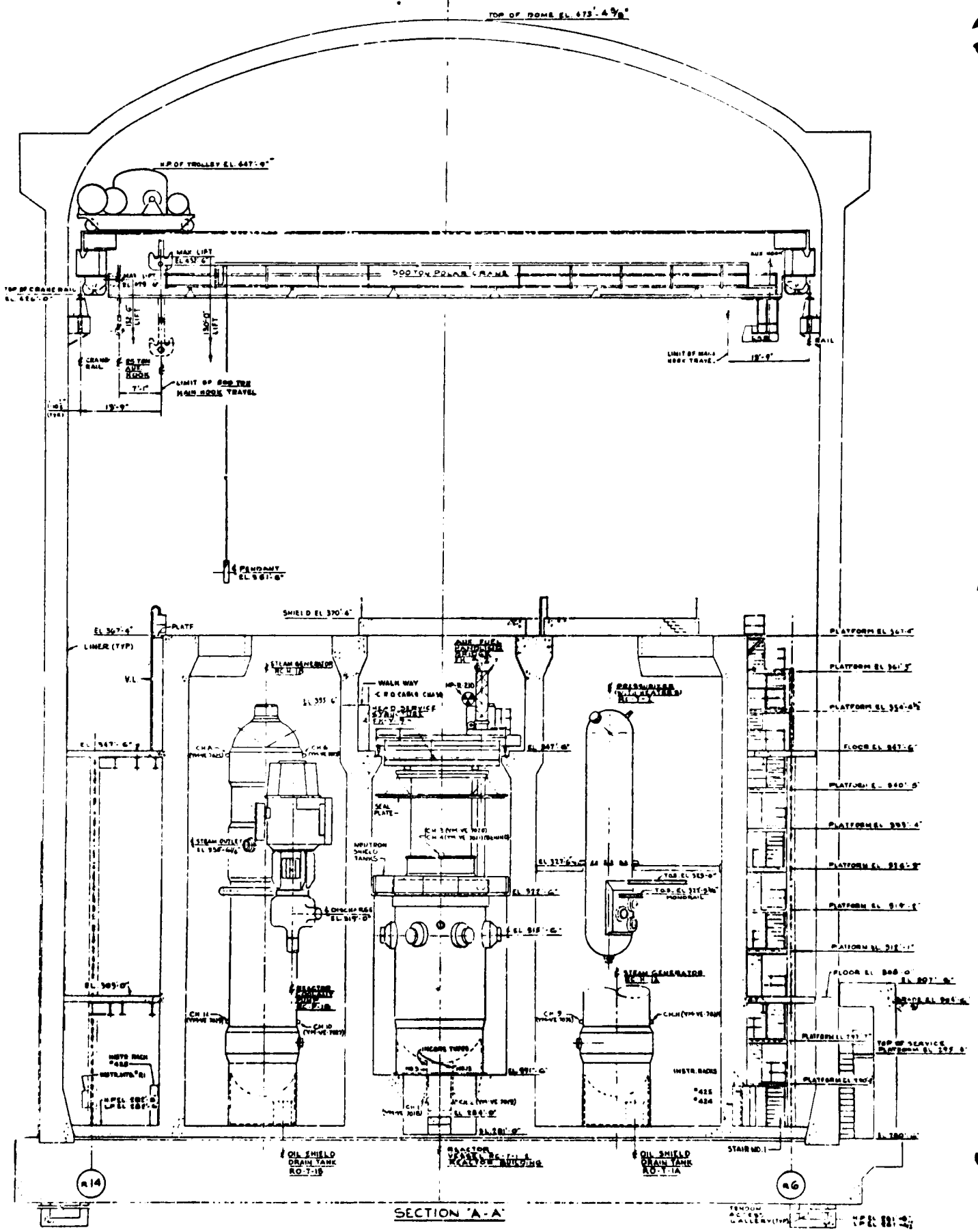

RV HEAD
INLET NOZZLE
STEAM GENERATOR
REACTOR COOLANT PUMP
PRESSURIZER
SERVICE STRUCTURE
STEAM GENERATOR
8'-0"
34'-0"
REACTOR VESSEL
27'-0"
6'-0"
SURGE LINE
79'-8 1/2"
72'-0 1/8"

FIGURE 3

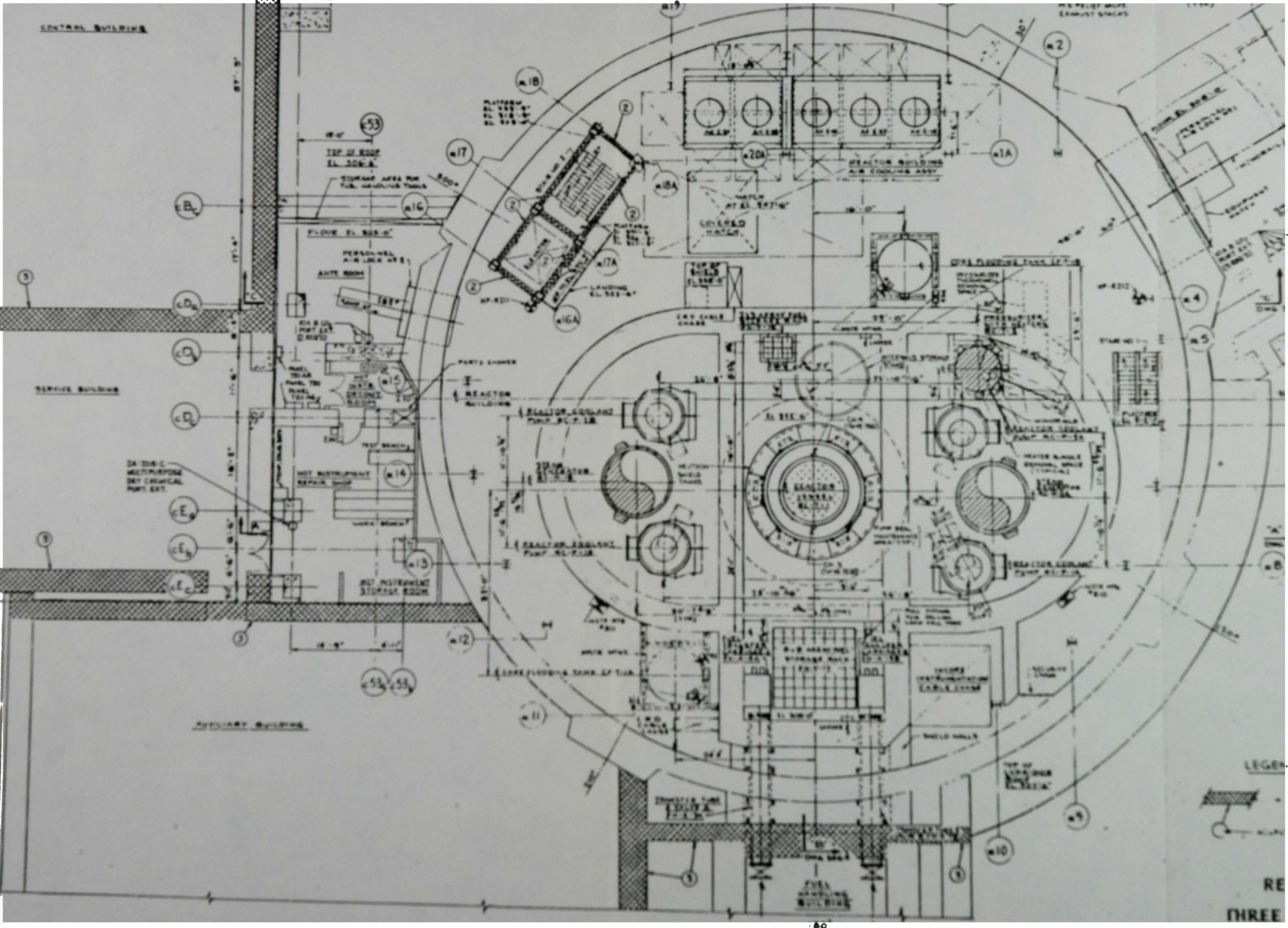

21'-6"
STEAM GENERATOR
PRESSURIZER
REACTOR COOLANT PUMP
29'
REACTOR VESSEL
60°
60°
30'-7 5/8"
3'-11 5/8"
4'-8 15/16"
15-21/32"
11'-10 1/4"
11'-6 25/32"

FSAR-FIG 1.2-6

FIGURE 4



SECTION 'A-A'



CONTROL BUILDING
SERVICE BUILDING
AUXILIARY BUILDING
REACTOR BUILDING AIR COOLING ASSY
COVERED HATCH
PERSONNEL AIR LOCK NO. 1
ANTE ROOM
HOT INSTRUMENT REPAIR SHOP
HOT INSTRUMENT STORAGE ROOM
REACTOR BUILDING
REACTOR COOLANT PUMP RC-P-1B
REACTOR COOLANT PUMP RC-P-1A
INCORE INSTRUMENTATION CABLE CHASE
SHIELD WALLS
FUEL HANDLING BUILDING
LEGEND

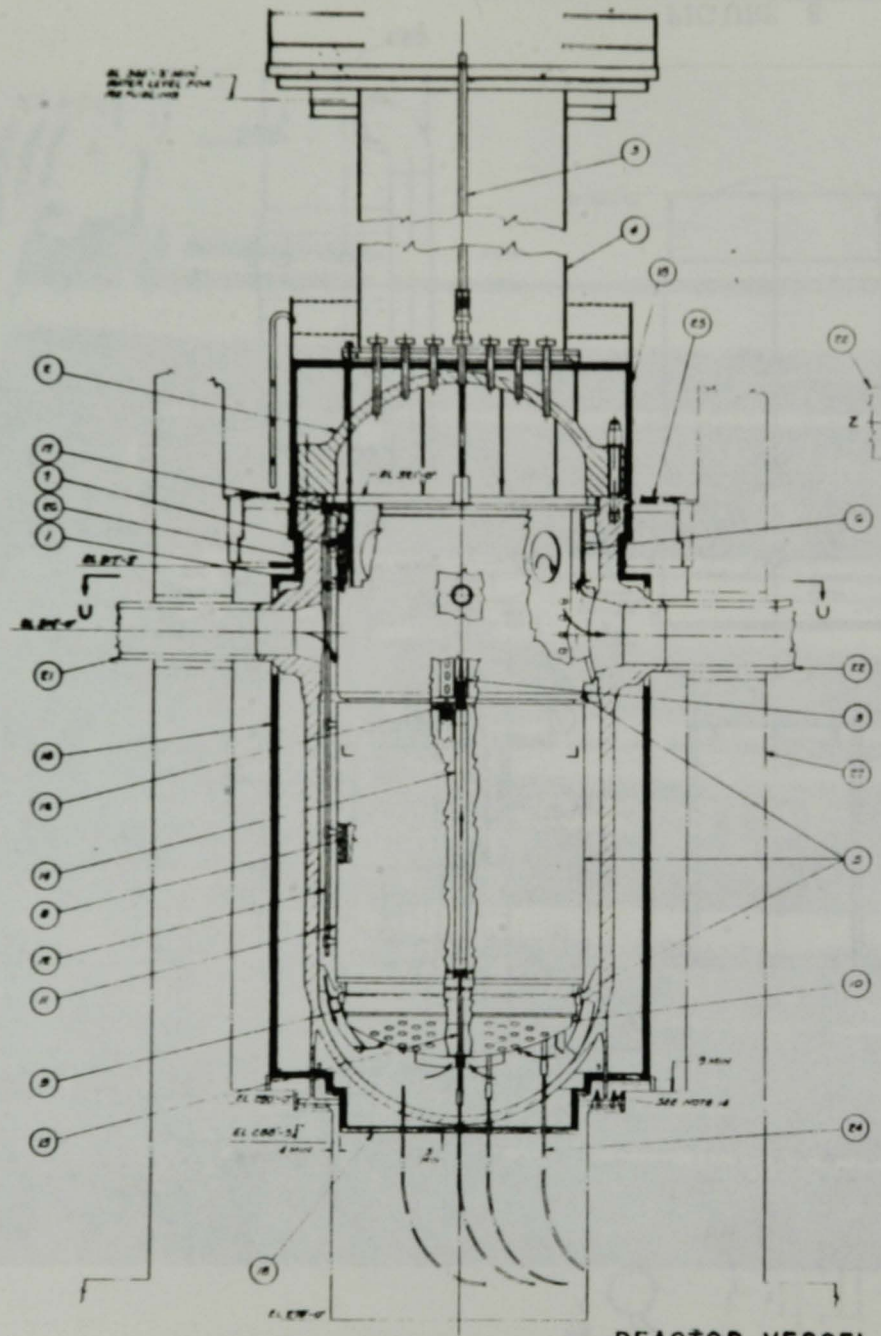
FIGURE 6

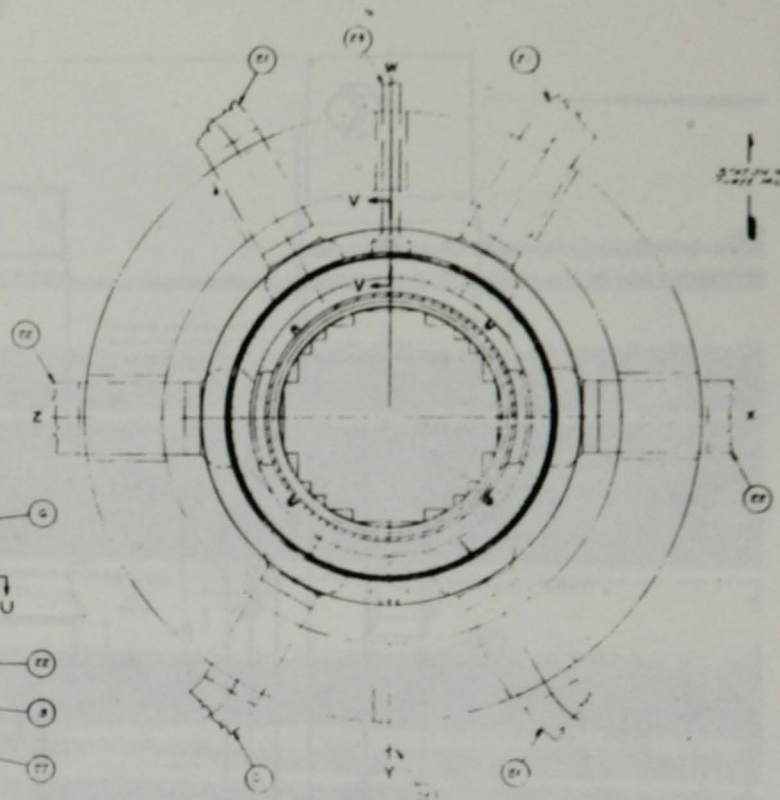
TMI-2

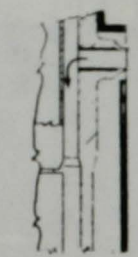
REACTOR COOLANT WATER QUALITY

(since 3-28-79)

	Average	Range
Boron (ppm)	3400	2800-3900
Sodium (ppm)	1120	350-1650
pH @ 77F	7.9	7.3-8.4
Chlorides (ppm)	3.8	1.6-6
Hydrogen (std cc/kg)	19	10-45
Oxygen	ND	ND







REACTOR VESSEL ARRANGEMENT

FIGURE 7

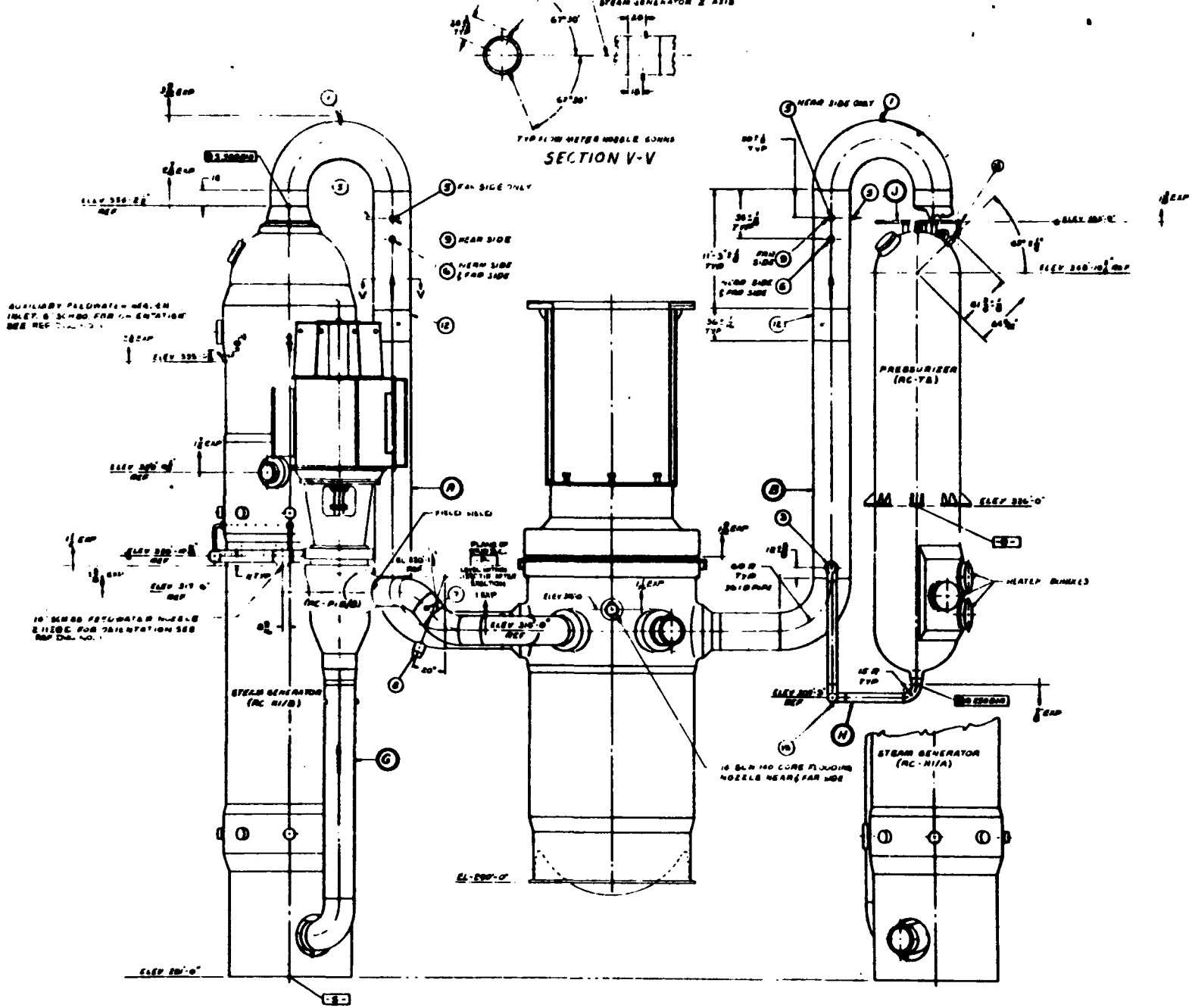
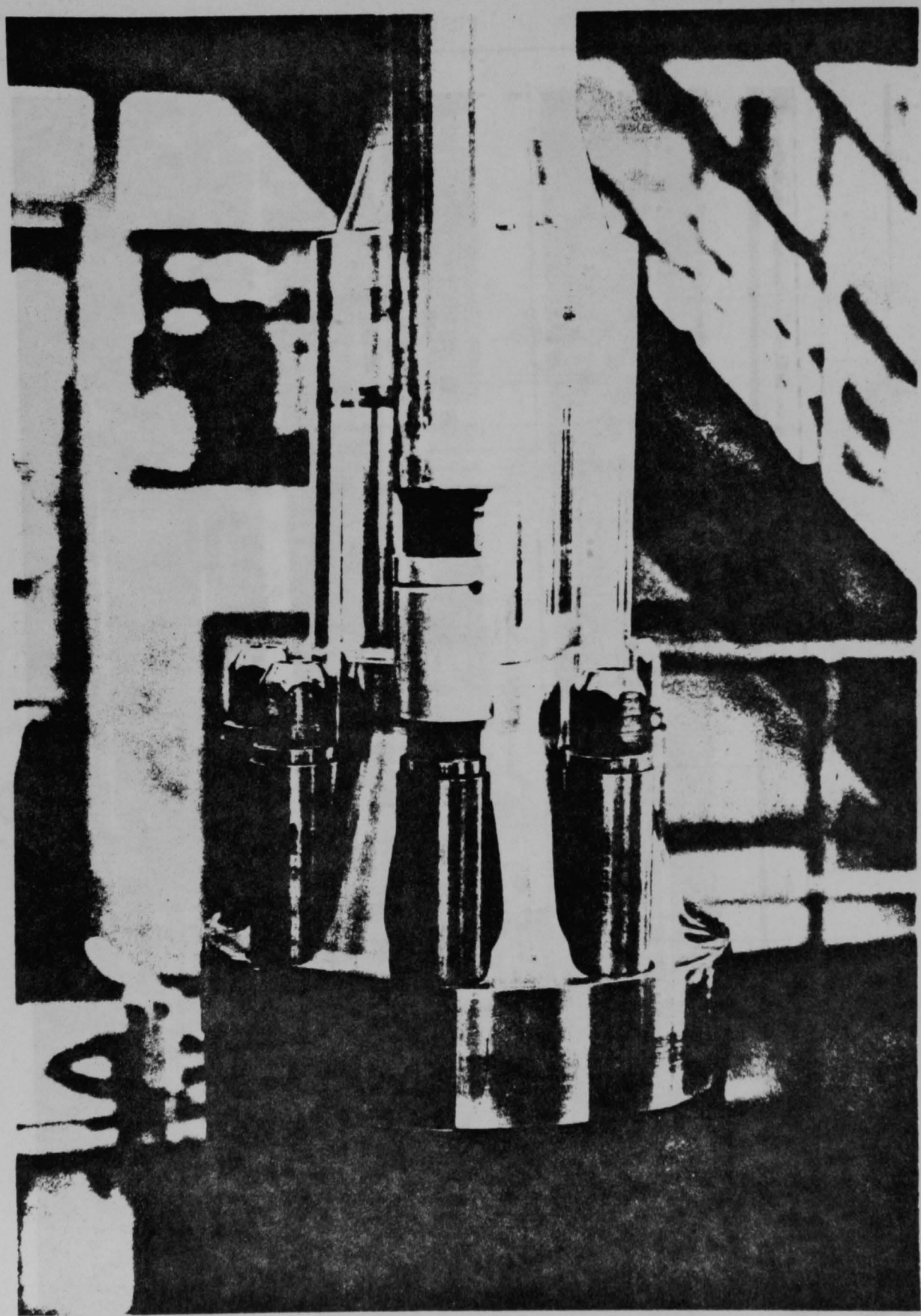

SECTION V-V
PRESSURIZER (RC-T1)
STEAM GENERATOR (RC-H1/B)
STEAM GENERATOR (RC-H1/A)
HEATER BUNDLES
FIELD WELD

FIGURE 8

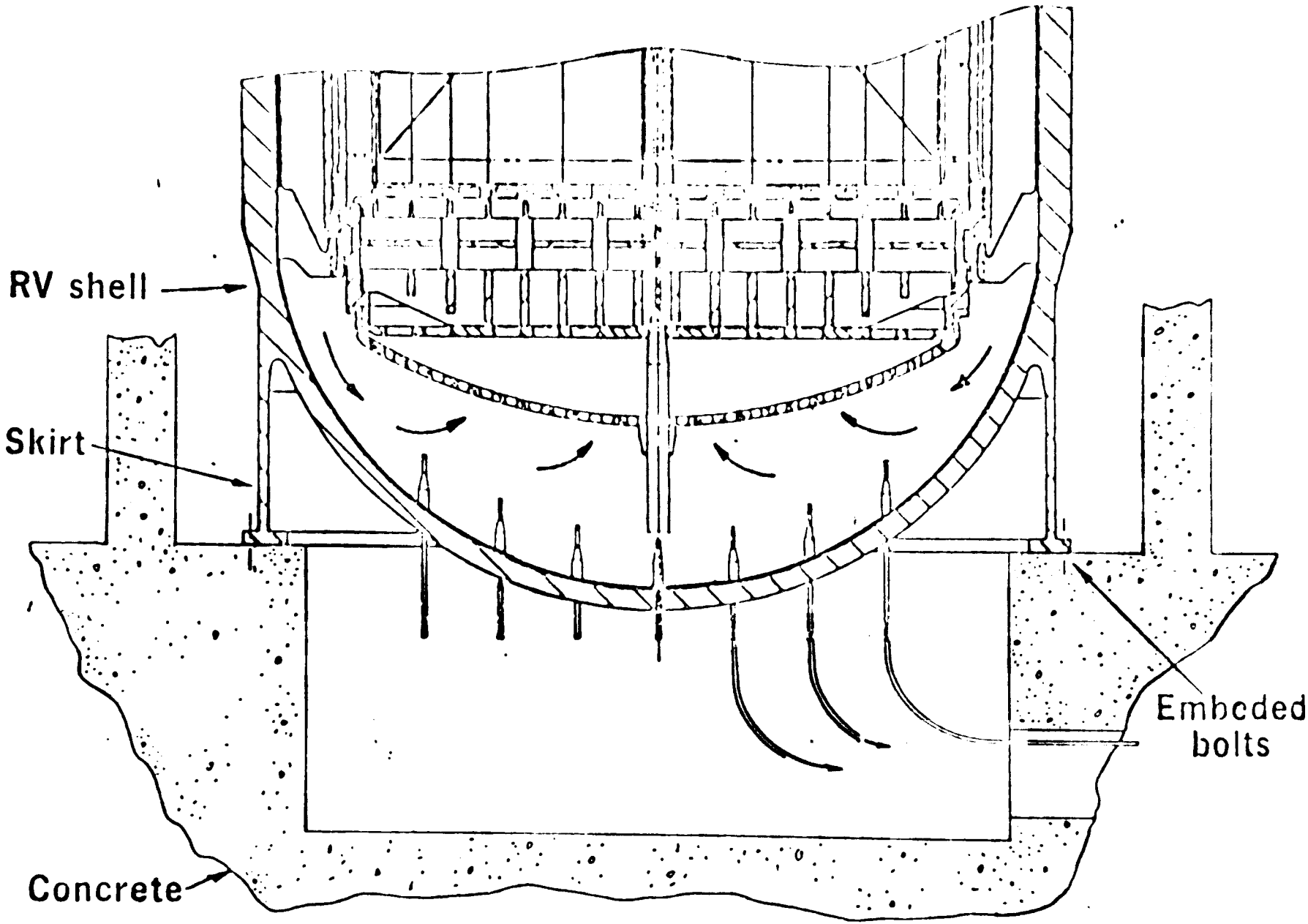
FIGURE 9

FIGURE 10



HOLDOWN BOLT REMOVAL

FIGURE 11



REACTOR VESSEL SKIRT SUPPORT

Reactor coolant system

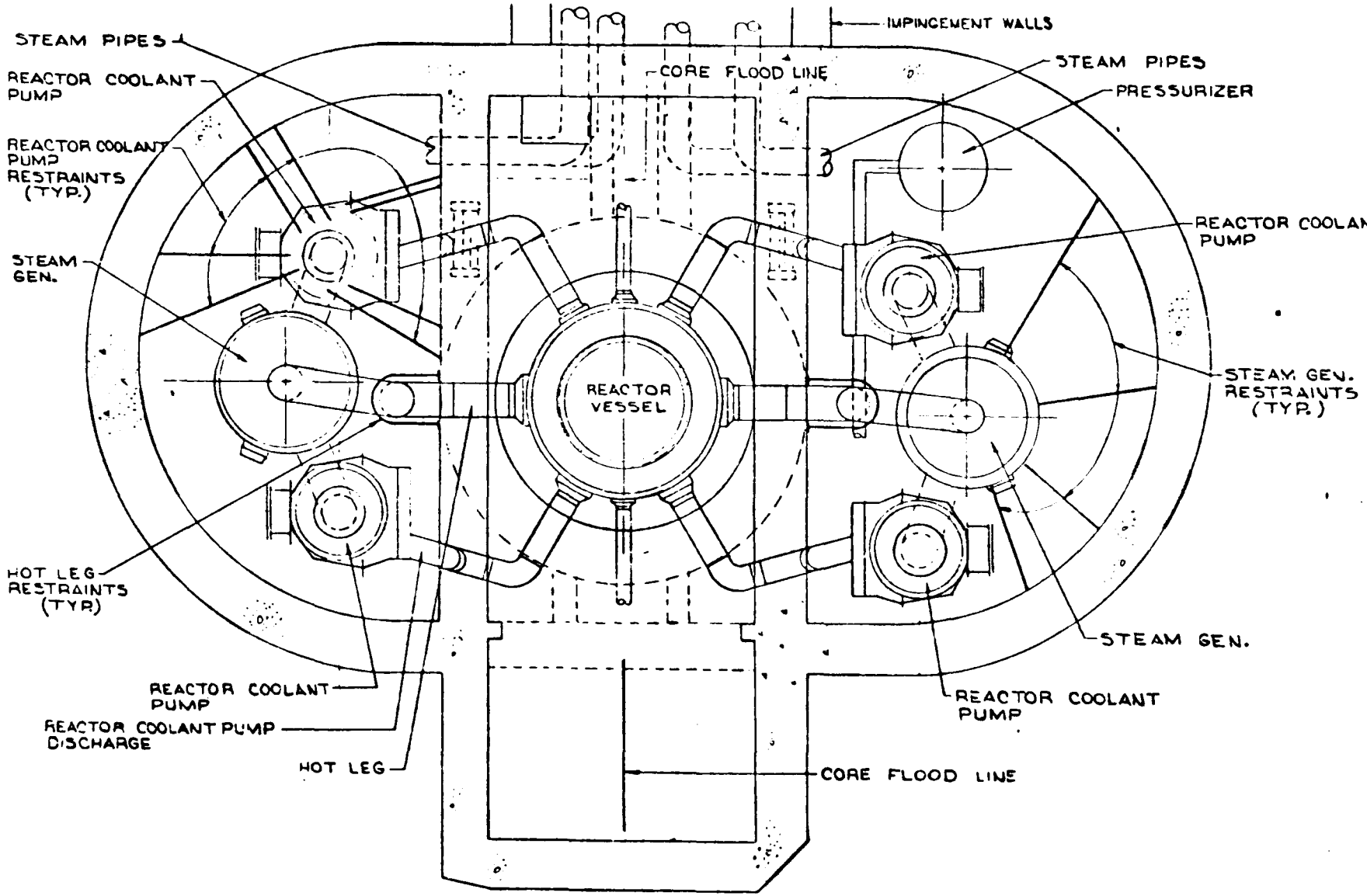


FIGURE 13

Pump supports

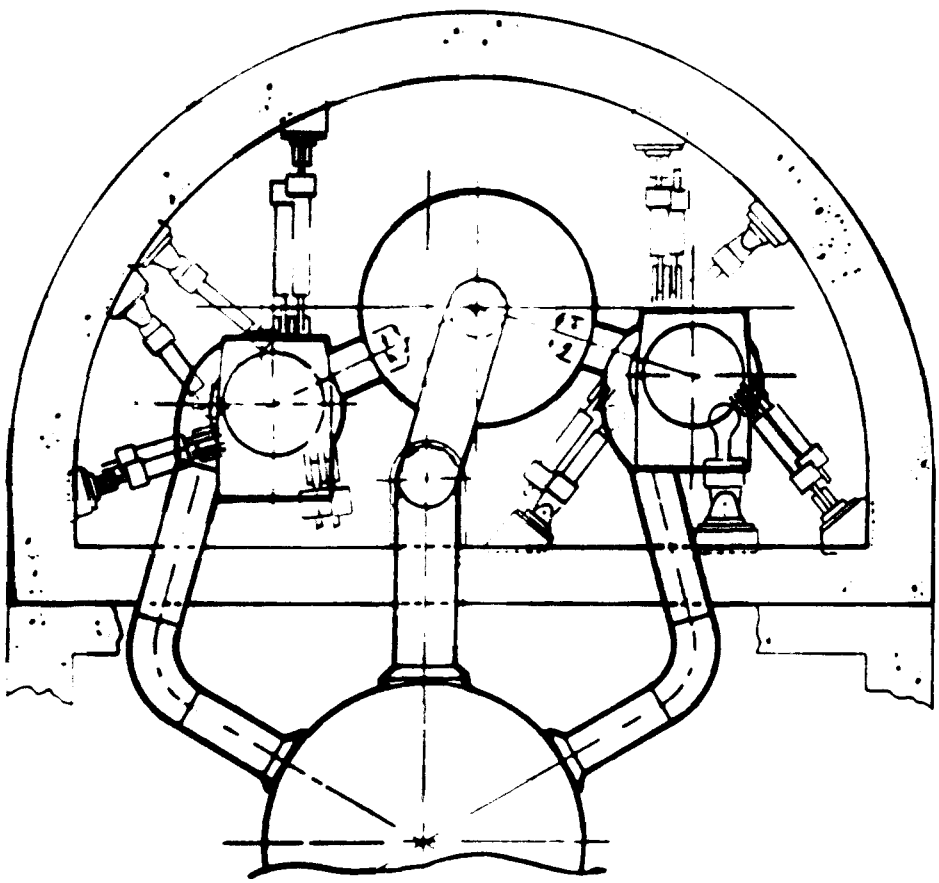


FIGURE 14

OTSG supports

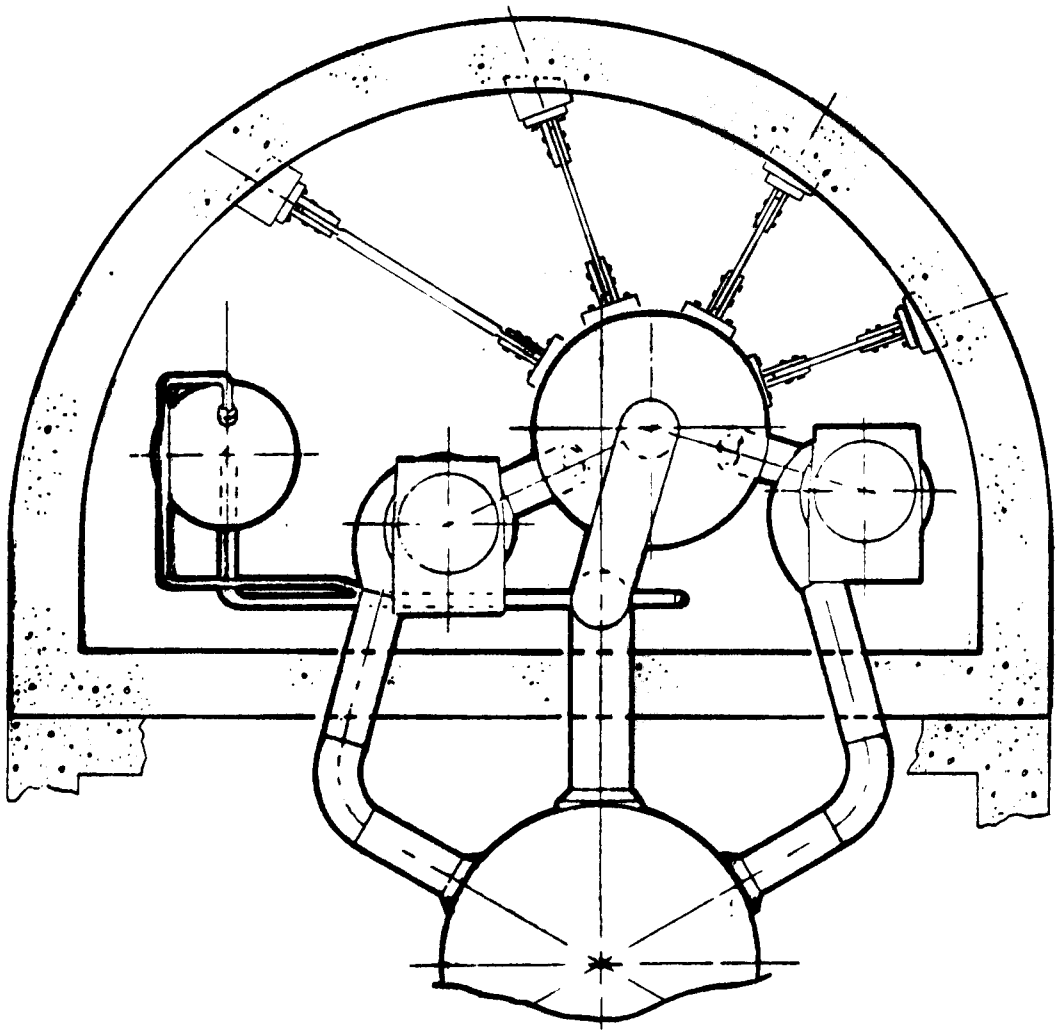


FIGURE 15

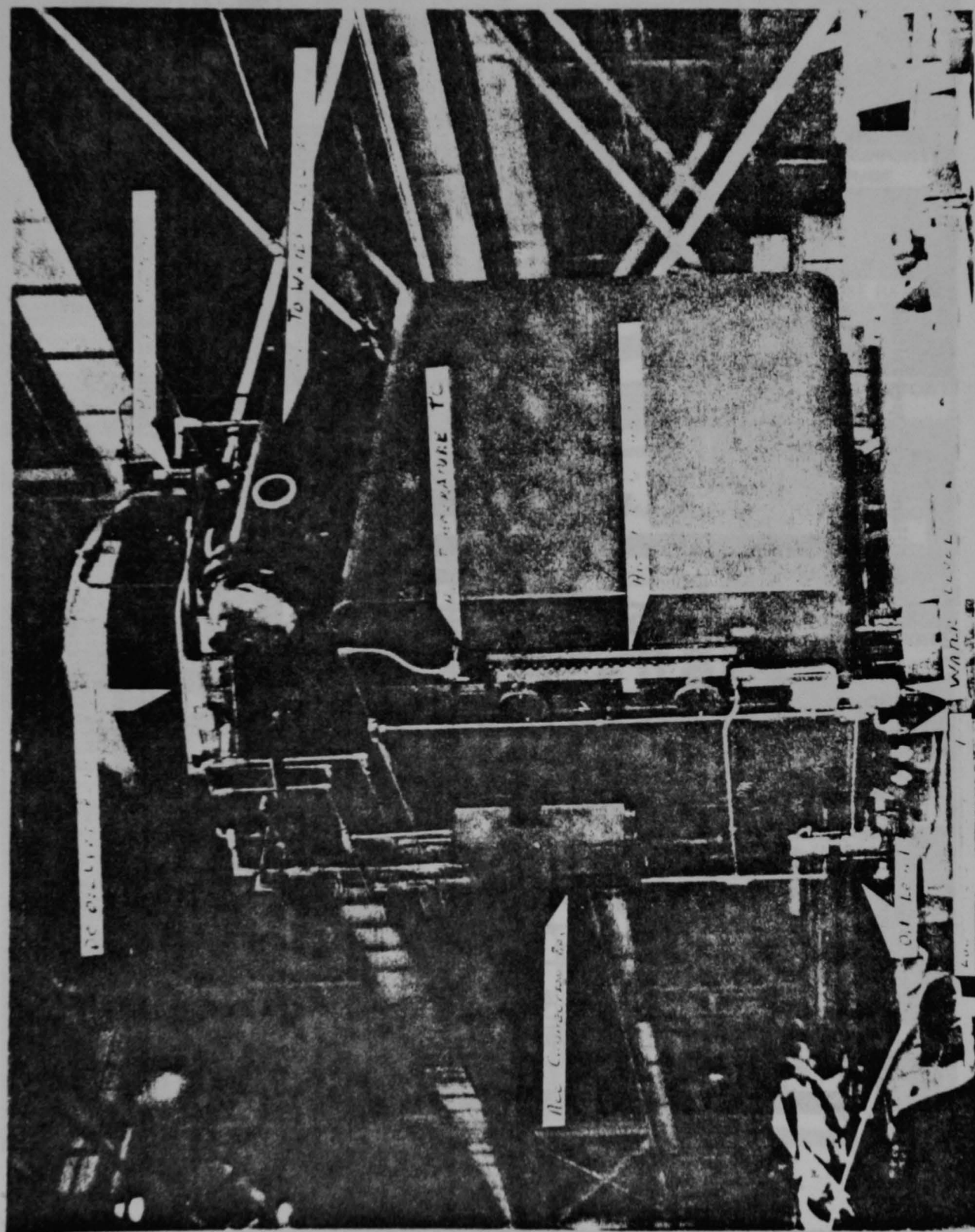


FIGURE 16

Fuel Handling Operations

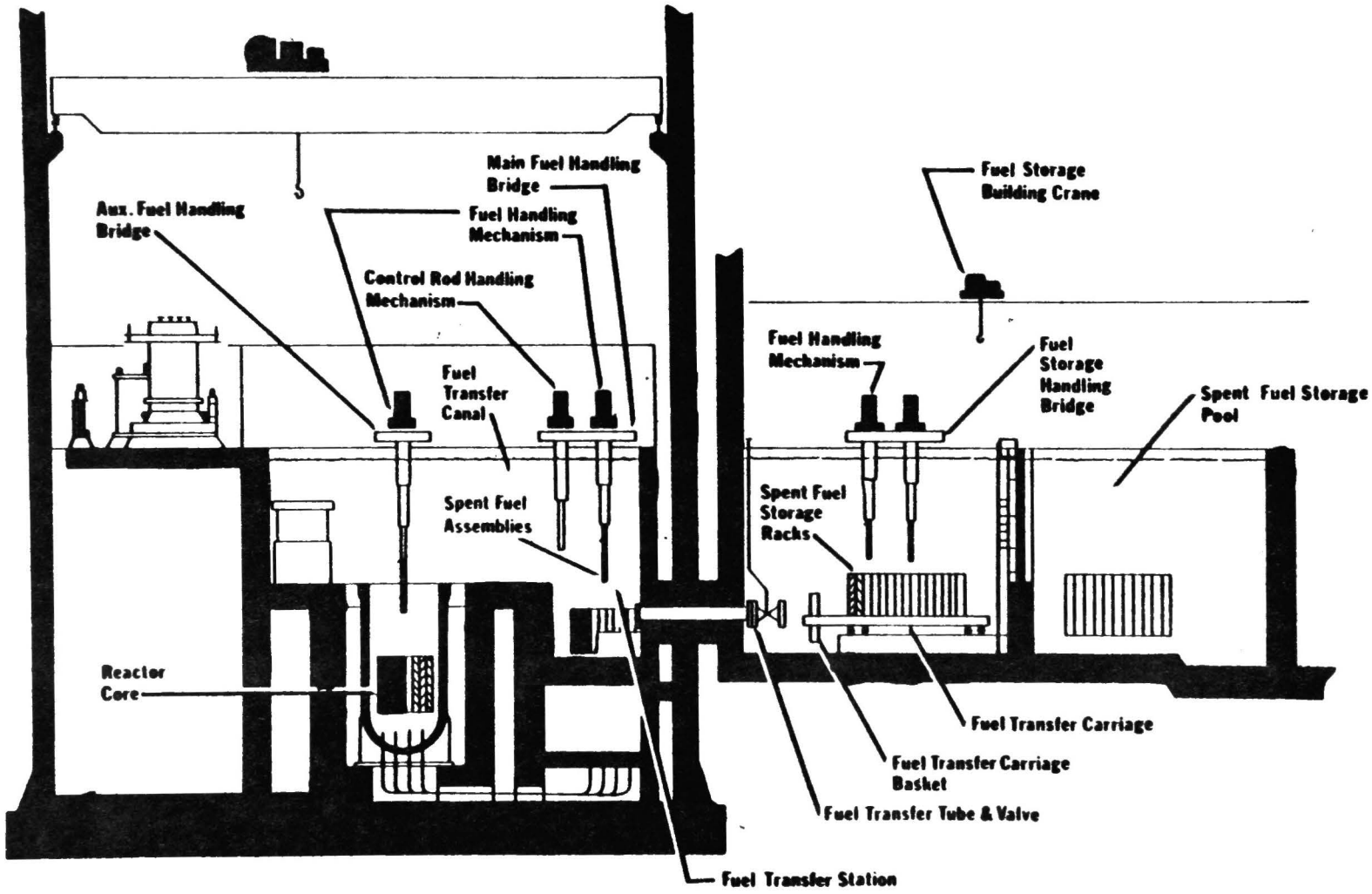


FIGURE 17

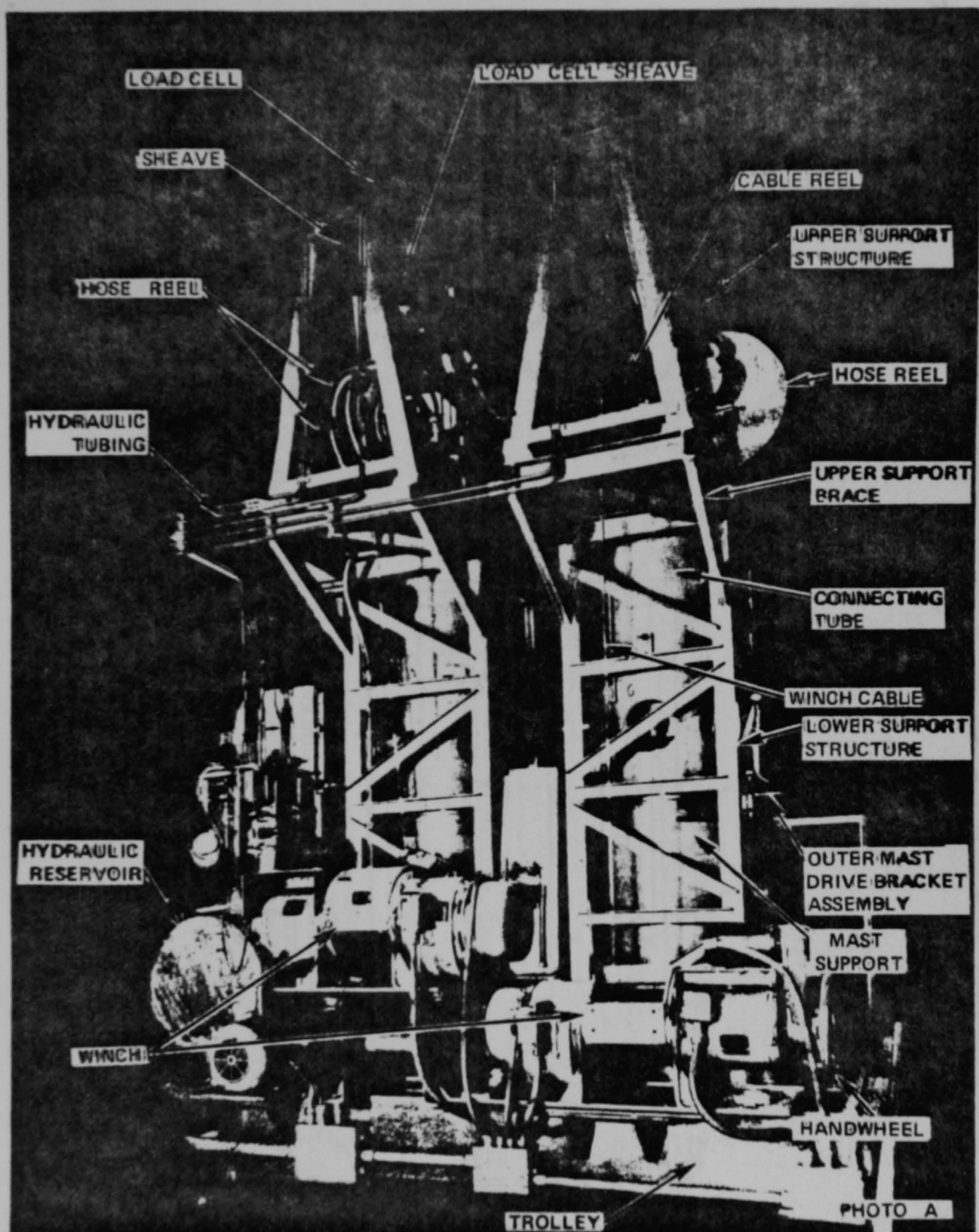
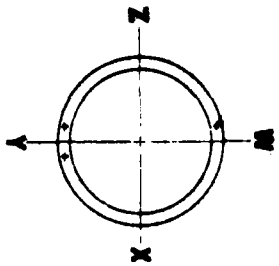
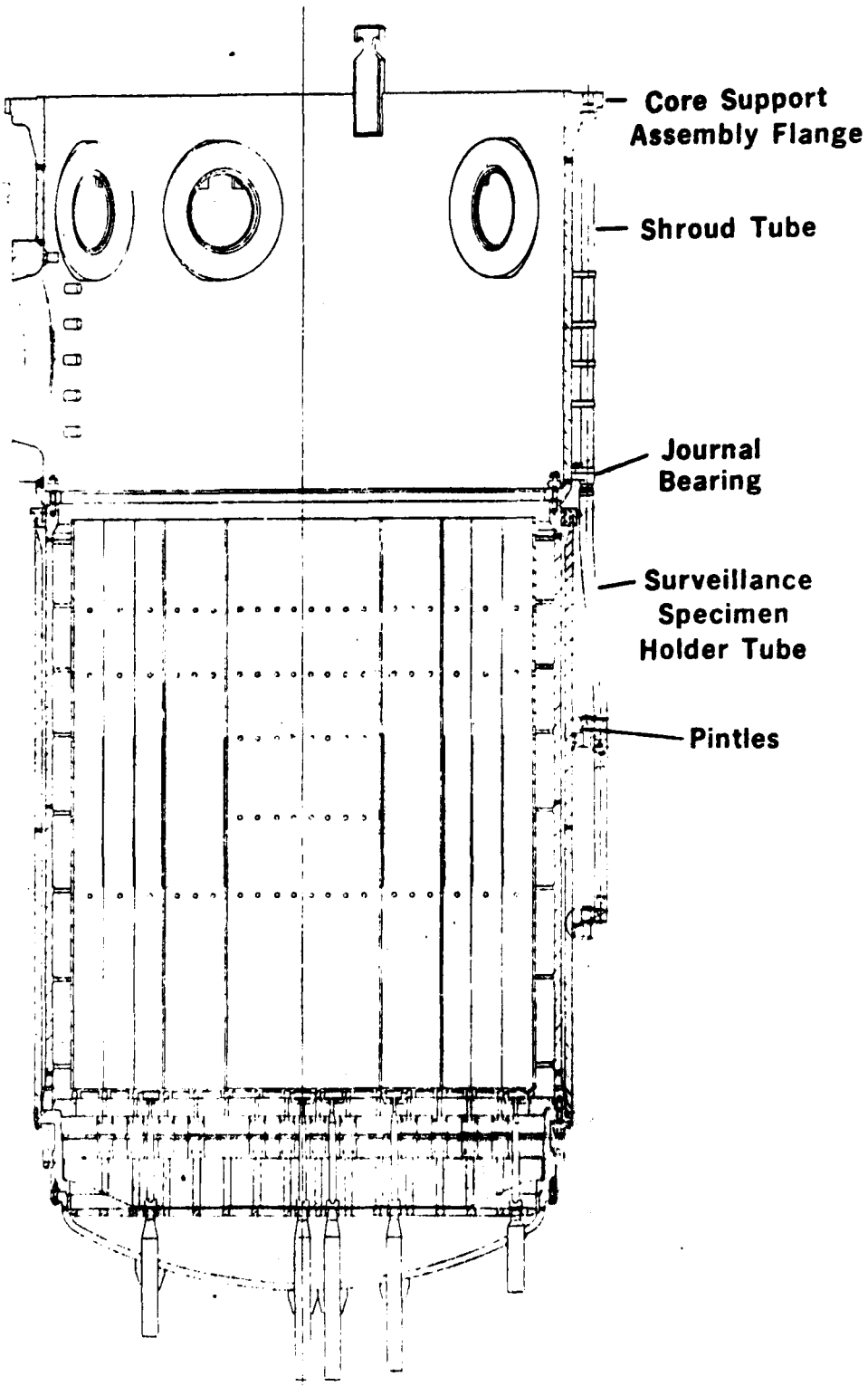

LOAD CELL
LOAD CELL SHEAVE
SHEAVE
CABLE REEL
UPPER SUPPORT STRUCTURE
HOSE REEL
HOSE REEL
HYDRAULIC TUBING
UPPER SUPPORT BRACE
CONNECTING TUBE
WINCH CABLE
LOWER SUPPORT STRUCTURE
HYDRAULIC RESERVOIR
OUTER MAST DRIVE BRACKET ASSEMBLY
MAST SUPPORT
WINCH
HANDWHEEL
TROLLEY
PHOTO A

FIGURE 18

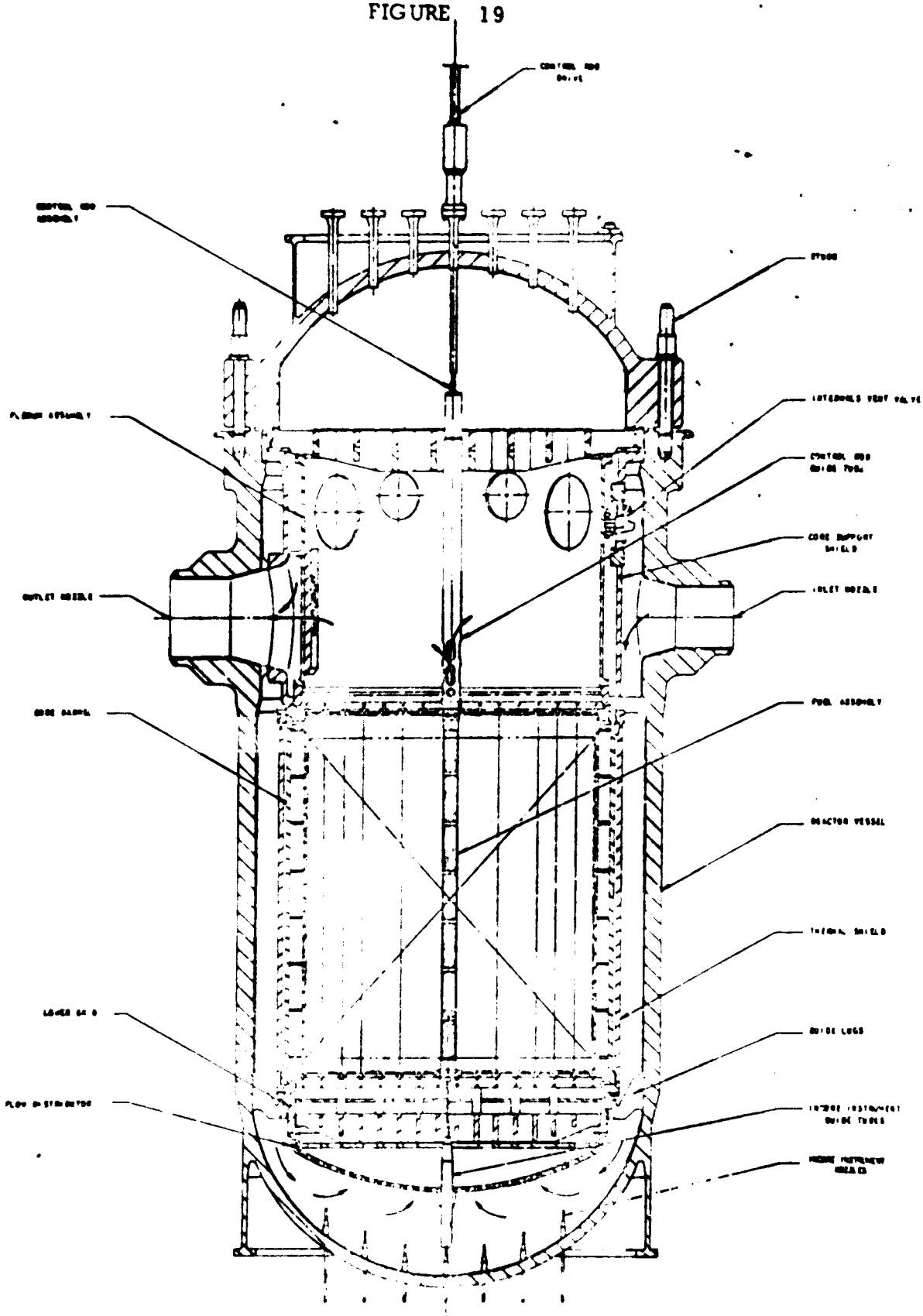
Key Plan





FSAR-FIG-4.2-3

FIGURE 19



SESSION F

CONTAINMENT ASSESSMENT TASK FORCE
EXPERIMENTAL PROGRAMS

CONTAINMENT SUMP ANALYSIS

Jack Daniels
GPU

Subsequent to the accident of March 28, numerous calculations were performed to determine the amount of fission products released from the fuel and thereby obtain a preliminary assessment of actual core damage. These isotopic inventory calculations were based on sample results from the Reactor Coolant Bleed Tanks and various other tanks in the Auxiliary Building. During the summer, a significant amount of leakage from the Reactor Coolant System occurred and was discharged to the building basement through equipment located within the containment, such as Reactor Coolant Pump seals. As engineering design of cleanup systems progressed, it became increasingly apparent that sampling the water in the Reactor Building basement was necessary in order to assess the performance characteristics of the various process systems. This paper will describe the method by which the sample was obtained while still maintaining containment integrity.

Method of Taking Samples

After several weeks of assessing the safest method to obtain a representative sample, it was determined that a direct access through a spare electrical penetration was the optimum technique. For these reasons, Penetration R-401, located at elevation 292', proved to be ideally suited for such an operation. Figure 1 shows the location of the penetration with respect to the equipment hatch and the Reactor Coolant Drain Tank, which was the pathway of the initial discharge. The Reactor Building Sump, located on the East side of the building, will be the point at which the process systems will take suction for the cleanup of the approximately 600,000 gallons of water currently in the Building.

A remote boring device was designed, built and tested for access to the building. Figures 1 and 2 show the method used for the boring operation while maintaining containment isolation. Before the boring operation, the outboard end of the penetration was fitted with a twelve inch isolation valve and four connections for pressure testing to verify the integrity of the apparatus during key points in the procedure. Figure 2a shows the cutter assembly in the retracted position with the isolation

valve closed. Figure 2b shows the arrangement of the cutter assembly in the drilling position. An air-actuated "steady rest" located approximately 6 inches behind the cutter head, was used to center the boring bar. An air driven motor was used to power the cutter. A 3/4 inch pilot hole was first drilled through the inboard plate to help support the larger (3 inch) trepan cutter. The final cut was completed at 3:30 A.M. on August 12, 1979. A section of tygon tubing, shown in Figure 2, was used to monitor for inleakage of water into the penetration in the unlikely event that the water level was above the elevation of the penetration.

When boring was completed, the steady rest was deactivated and retracted such as in Figure 1, allowing the isolation valve to be closed and the cutter assembly to be removed.

The actual sampling operations was conducted on August 24, 1979, with the apparatus shown in Figures 3 and 4. A sample guide tube and its support was attached to the outer flange with the isolation valve closed. After performing leak rate testing, the isolation valve was opened and the sample guide tube inserted into the penetration. The end of the guide tube extended approximately 3 inches beyond the penetration to avoid catching the sample tube on the end of the penetration.

A flexible tygon tube with a weighted end on the containment end and an in-line double check valve quick disconnect on the other was lowered at 6 inch intervals until water was reached in order to obtain water samples from the top, middle and bottom of the Reactor Building.

The sample piping arrangement shown in Figure 5 was utilized to take the samples. A roto-flex pump was used to pump the water into the sample bomb, and an overflow tank provided to catch any fluid inadvertently pumped through the sample container was filled in order to clear the lines before drawing another sample. Double hydraulic seals were used to seal the tygon inside the sample guide tube with the guide tube sealed at all times within the "top-hat" attached to the end of the penetration.

RADIOISOTOPE RESULTS

Tables 1 through 7 presents the results of the analyses as performed by Oak Ridge National Laboratory. Of particular interest is the bottom sample which contained a greenish, gelatinous precipitate. The presence of the precipitate was not surprising and, in fact, had been predicted several months earlier. Tables 5 and 6 show analytical results of the precipitate based on the total volume of the sample. The predominant element was copper, which is the reason for the greenish color, and is thought to come from grounding cables in instrument racks at the 282 ft. elevation.

A second point of significance is the concentration of cesium in the three samples. An average concentration of 176 μCi/ml of Cesium-137 represents a total curie content of 3.8×10^5 curies, based on an estimated 570,000 gallons of water in the basement, or approximately 43 percent of the total core inventory. An additional 17 percent can be accounted for in the Reactor Coolant System and tankage in the Auxiliary Building bringing the total Cs-137 release fraction to 60 percent.

In comparison, the Strontium-90 concentration found in the basement represents less than 1 percent of the core inventory.

Table 8 presents the results of the Babcock and Wilcox analyses. B & W received a ml solution from the original 30 ml sample sent to ORNL. The discrepancy between the Cesium concentrations reported by the two laboratories has not been satisfactorily resolved at this time, but is believed to be associated with the transfer of 5 ml from the 30 ml sample.

Table 9 shows the total activity found on the steel plug removed from the penetration during the boring operation. The surface area of the plug is approximately 35 cm^2. There appears to be an unusually high concentration of Te-129 m found on the plug, which is posing some questions to those of us involved in understanding fission product transport.

In conclusion, the sampling operations through penetration R-401 represented the first direct access to the building itself and was done with minimum personnel exposure. The information gained by this project has confirmed that a sound basis exists for the design of recovery systems as well as providing data to the industry as a whole. This operation and others yet to come will help provide the basis for understanding the releases from design basis accidents and fission product transport in particular.

Table 1. Solution Characteristics

	Top	Middle	Bottom
Color	Light yellow	Light yellow	Greenish with precipitate
Visible organic	None	None	None
Radiation level, side (mR/hr)	580	500	530
Radiation level, bottom (mR/hr)	740	780	800
Precipitate	None	None	Yes[a]
Volume (ml)	30	30	30

[a]Flocculent in appearance, gelatinous, dirty green color, 10% by volume, centrifuged to 4% by volume.

Table 2. Radiochemical Analyses of Three Solutions
(μCi/ml at 0800, 8/28/79)

Isotope	Top	Middle	Bottom
^{137}Cs	176	179	174
^{134}Cs	40	40	39.6
^{140}La	0.09	0.078	0.14
$^{89}+^{90}Sr$	46.3	43.5	44.9
^{3}H	1.03	1.05	1.01
^{129}I	0.079[a]	0.080[a]	0.076[a]
^{131}I	0.012	0.012	0.013
^{90}Sr	2.70	2.90	2.83
Activity in scavenging precipitation with $Pr(OH)_3$			
^{95}Zr	—	0.0030	0.0025
^{95}Nb	0.0021	0.0030	0.0099
^{103}Ru	0.005	0.0050	0.0071
^{106}Ru	0.0039	0.0072	0.0099
^{113}Sn*	—	—	0.0016
^{125}Sb	0.012	0.015	0.017
^{129}Te	—	—	0.035
^{134}Cs	0.0066	0.0059	0.0042
^{137}Cs	0.029	0.028	0.0175
^{141}Ce	—	0.00047	0.0019
^{144}Ce	—	0.0046	0.0080
^{140}La	0.036	0.028	0.052
^{140}Ba	—	0.0038	—
Gross α	3.4 ± 1.6[b]	1.2 ± 1.3[b]	5.4 ± 2[b]

[a]Units are μg/ml.

[b]Units are dpm/ml.

*Tentative identification.

Table 3. Spark Source Mass Analysis of Three Solutions (ppm)

Element	Top	Middle	Bottom
Ag	<0.5	<0.2	<0.3
Al	3	3	3
As	<0.2	<0.05	<0.1
B	1950	2200	1900
Cl	10	15	8
Ca	10	10	8
Cd	<0.2	<0.2	<0.2
Co	<0.1	<0.1	<0.1
Cr	0.7	0.7	0.7
Cs	0.6[a]	0.7[a]	0.7[a]
Cu	≤0.2	≤0.2	10
Fe	0.58	1.1	1.8
I	<0.5	<0.5	<0.5
In	<0.1	<0.1	<0.1
K	4	4	4
Li	1.61[b]	1.55[b]	1.44[b]
Mg	≤3	≤2	≤1
Mn	≤0.1	≤0.05	≤0.1
Mo	≤0.5[c]	≤0.5[c]	1[c]
Na	1080	1200	1200
Ni	≤0.2	≤0.2	3
P	0.3	0.3	0.2
Rb	0.3	0.3	0.3
S	9	8	7
Sr	≤0.1	≤0.1	≤0.1
Te	<0.2	<0.5	<0.4
Ti	≤2	≤1	≤1
V	≤0.2	≤0.1	<0.1
Y	≤0.4	≤0.1	<0.1
Zn	0.5	0.5	0.4

[a] Fission product Cs.

[b] >99% ^{7}Li

[c] Stable Mo, not fission product.

Table 4. Solution Isotopic Analysis

Sample	Top	Middle	Bottom
U, ppb	7	13	28
234, %	0.021	0.014	0.021
235, %	1.98	1.34	2.04
236, %	0.058	0.036	0.066
Pu, ppb	0.010	0.011	0.033
239, %	89.1	89.4	89.8
240, %	8.5	8.4	8.1
241, %	2.3	2.1	2.0
242, %	_____	____	Assume 0.1

Table 5. Solids From Bottom Sample (μCi/ml at 0800, 8/28/79, Based on Total Volume of Bottom Sample)

Isotope	Sample 1[a]	Sample 2[a]
^{58}Co	0.0055	0.0079
^{60}Co	0.0011	0.0015
^{95}Zr	0.037	0.061
^{95}Nb	0.104	0.162
^{103}Ru	0.042	0.078
^{106}Ru	0.035	0.051
^{110m}Ag	0.0015	0.0025
^{113}Sn*	0.015	0.021
^{125}Sb	0.022	0.033
^{129m}Te	0.277	0.514
^{131}I	0.0108	0.016
^{134}Cs	0.018	0.011
^{137}Cs	0.078	0.049
^{140}Ba	0.041	0.047
^{140}La	0.106	0.122
^{141}Ce	0.0034	0.0097
^{144}Ce	0.0134	0.0446
$89 + 90Sr$	2.78	

[a]Two samples were taken at different times; they were centrifuged, washed, and re-scanned.

*Tentative identification.

Table 6. Solids from Bottom Sample, Neutron Activation Analysis (Units are μg/ml, Based on Total Volume of Bottom Sample)

^{235}U	0.00459
In	0.16
^{129}I	0.07
Cu	54
Mn	0.62
Al	7
Ca	≤ 2

Table 7. Spark Source Mass Analysis of Solids From Bottom Sample (ppm) Based on Total Volume of Bottom Sample

Ag	8*	Li	≤ 0.3
Al	8	Mg	7
B	3	Mn	1
Ca	2	Mo	≤ 1[b]
Cd	≤ 0.5	Na	≤ 1
Co	≤ 0.1	Ni	10
Cr	2	P	0.4
Cs	≤ 0.5	Rb	≤ 0.3
Cu	54[a]	S	5
Fe	10	Sr	≤ 0.2
I	0.7	Te	≤ 0.2
In	0.3	Ti	0.5
K	1	Zn	2
U[c]	0.106	Pu[c]	0.00016
^{234}U	0.022 (At %)	^{238}Pu	≤ 0.1 (At %)
^{235}U	2.35 (At %)	^{239}Pu	91.13 (At %)
^{236}U	0.065 (At %)	^{240}Pu	7.57 (At %)
^{238}U	97.56 (At %)	^{241}Pu	1.10 (At %)
		^{242}Pu	0.1 assumed

*May be some memory.

[a]Internal standard from NAA.

[b]Stable Mo; not fission product.

[c]Thermal emission mass resin bead analysis.

Table 8

Reactor Building Sump Samples

Unfiltered (μCi/gm)	Top	Middle	Bottom
Cs-134	30	29	30
Cs-137	145	144	148
La-140	0.05	0.04	0.10
I-131	0.06	< 0.5	< 0.5
Sn-113	0.11	< 0.4	< 0.4
Ce-141	0.4	.06	< 0.4
Filtrate (μCi/gm)			
Cs-134	28	27	28
Cs-137	138	135	138
H-3	0.92	0.96	0.94
H	8.6 ± 0.2	8.6 ± 0.2	8.6 ± 0.2
Na (ppm)	850 ± 85	850 ± 85	850 ± 85
Cl (ppm)	14 ± 2	13 ± 2	14 ± 2
B (ppm)	1780 ± 50	1740 ± 50	1750 ± 50
Gross β (μCi/gm)	175	190	200
Gross α (μCi/gm)	< 1.E-6	< 1.E-6	< 1.E-6
Separable Solids (μCi/g solution)			
Mn-54	< 4.8E-4	< 5.2E-5	3.2E-4
Co-58	< 5.7E-4	< 5.7E-5	< 3.0E-3
Zr-95	< 1.3E-3	1.2E-4	1.2E-2
Nb-95	6.8E-4	3.6E-4	5.0E-2
Ru-103	3.4E-4	6.3E-4	3.8E-2
Ru-106	1.4E-3	4.0E-4	1.5E-2
Sn-113	< 1.7E-3	< 1.0E-4	9.0E-3
Cs-134	0.12	6.7E-3	0.26
Cs-137	0.56	3.1E-2	1.2
La-140	4.0E-4	1.0E-4	0.89
Ce-141	2.8E-4	6.6E-5	3.7E-3
Ce-144	< 1.0E-4	2.8E-4	1.3E-2

Table 9. Painted Steel Plug (uCi Total at 0800, 8/29/79)

Isotope	uCi
^{58}Co	0.032
^{60}Co	0.01
^{95}Zr	0.09
^{95}Nb	1.7
^{103}Ru	0.58
^{106}Ru	0.42
^{110m}Ag	0.080
^{113}Sn	0.24
^{124}Sb	0.005
^{125}Sb	0.45
^{127m}Te	7.8
^{129m}Te	23.6
^{125m}Te	0.5
^{131}I	0.33
^{134}Cs	0.47
^{137}Cs	2.07
^{140}Ba	
^{140}La	0.019
^{141}Ce	0.057
^{144}Ce	0.24

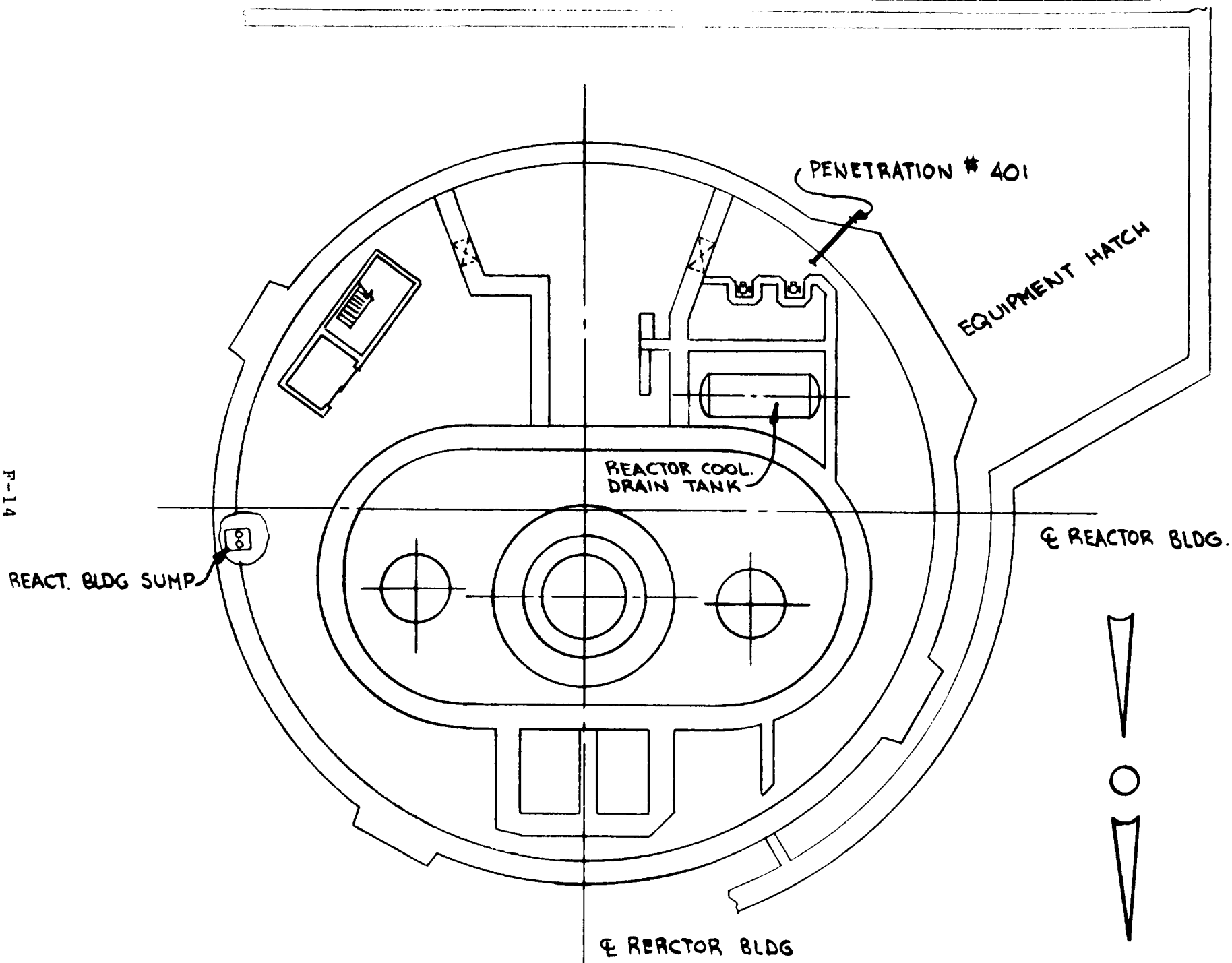

PENETRATION * 401
EQUIPMENT HATCH
REACTOR COOL.
DRAIN TANK
℄ REACTOR BLDG.
REACT. BLDG SUMP
℄ REACTOR BLDG

FIGURE 1

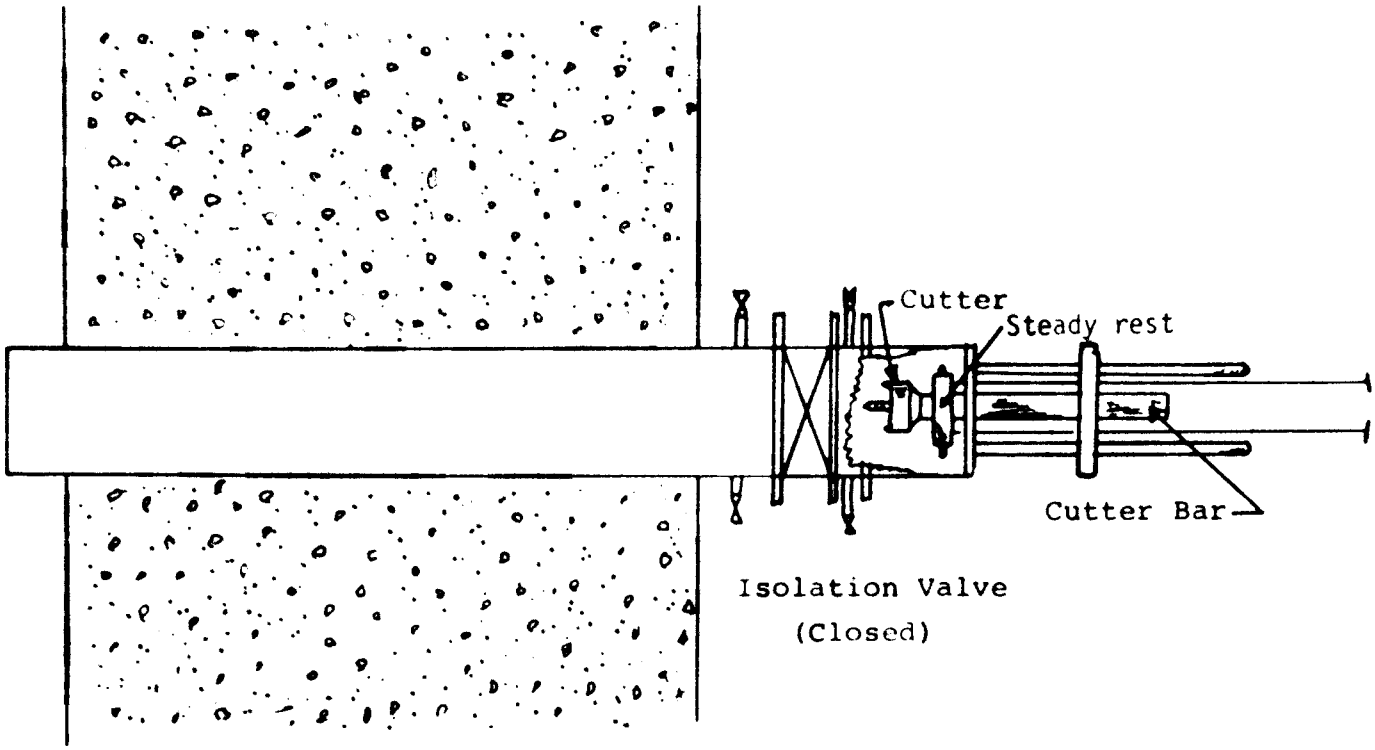


Figure 2a
Cutter & Bar in Retracted Position

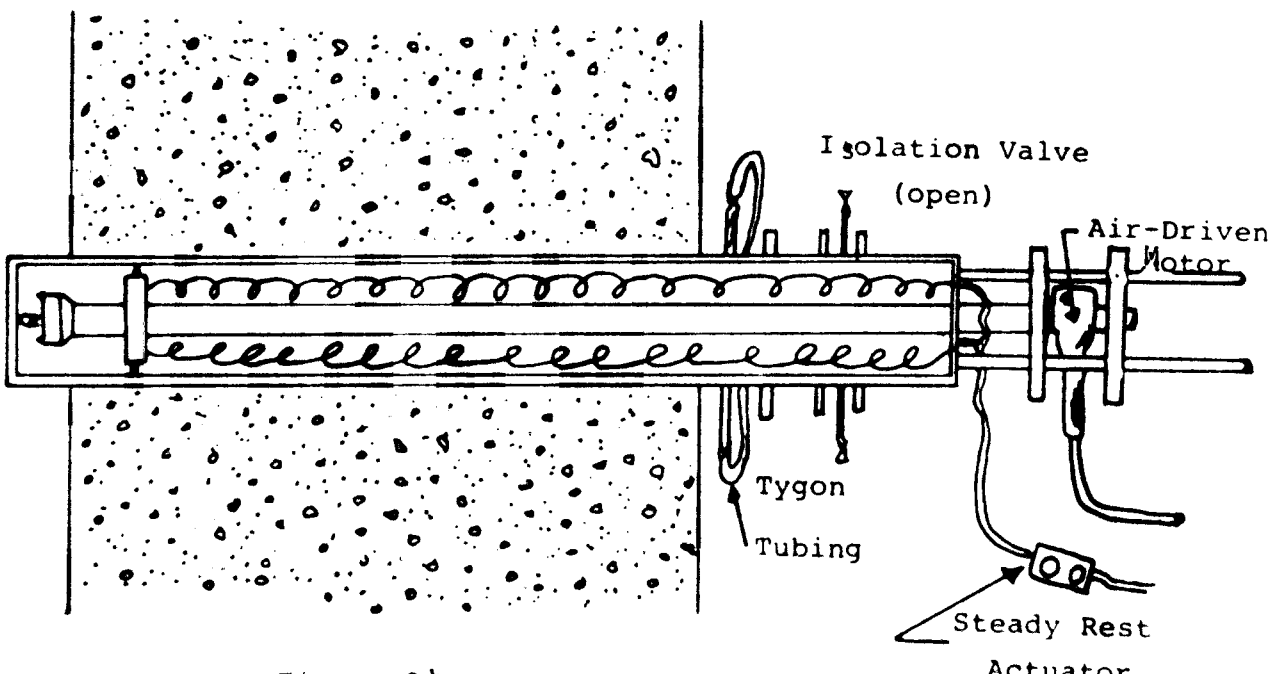


Figure 2b
Cutter & Bar in Drilling Position

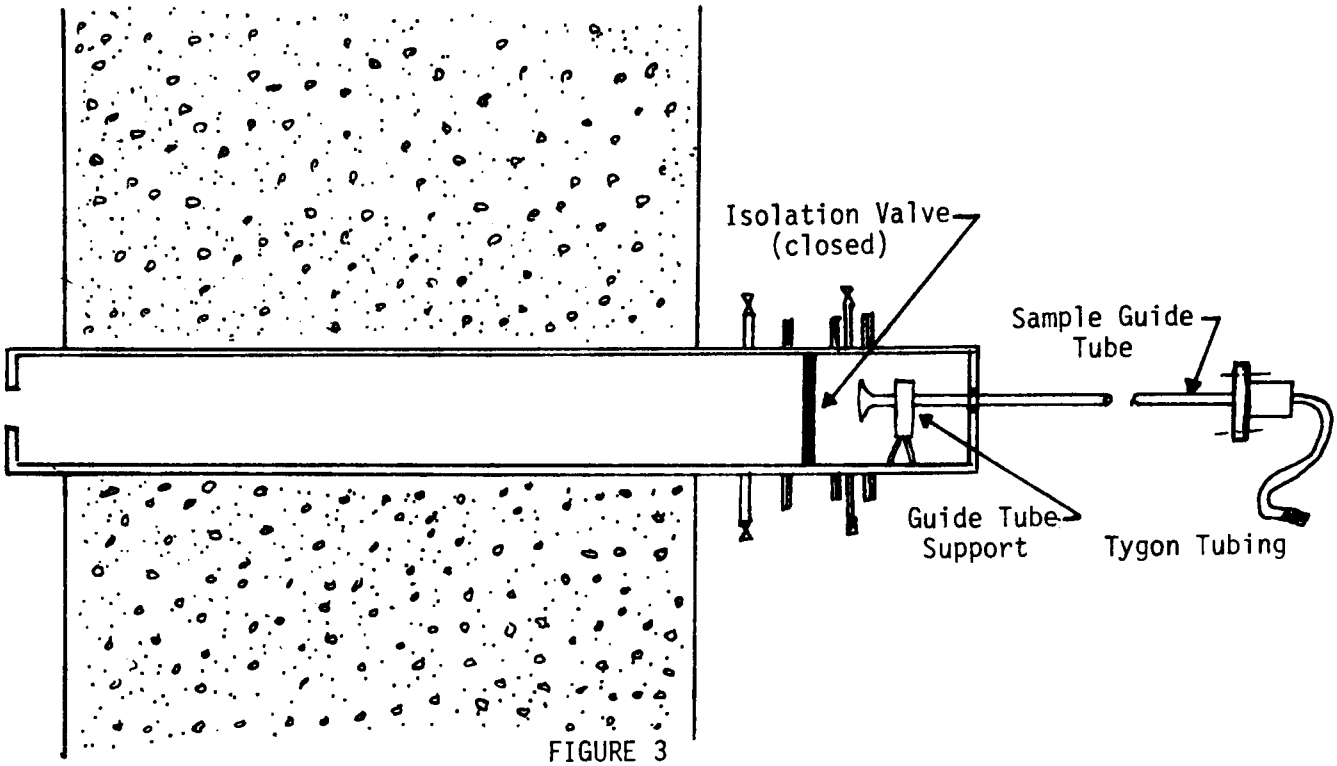


FIGURE 3

SAMPLE TUBE IN RETRACTED POSITION

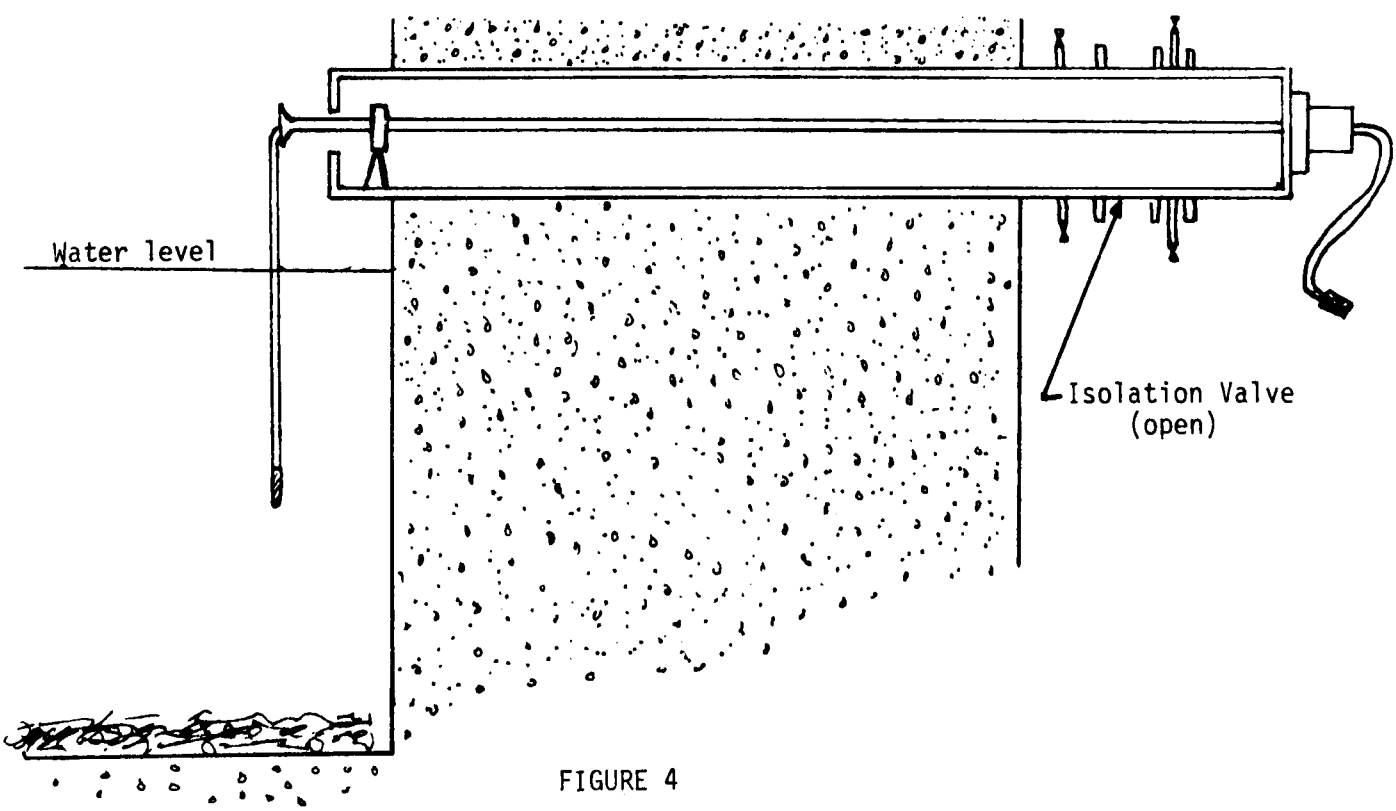


FIGURE 4

SAMPLE TUBE IN SAMPLING POSITION

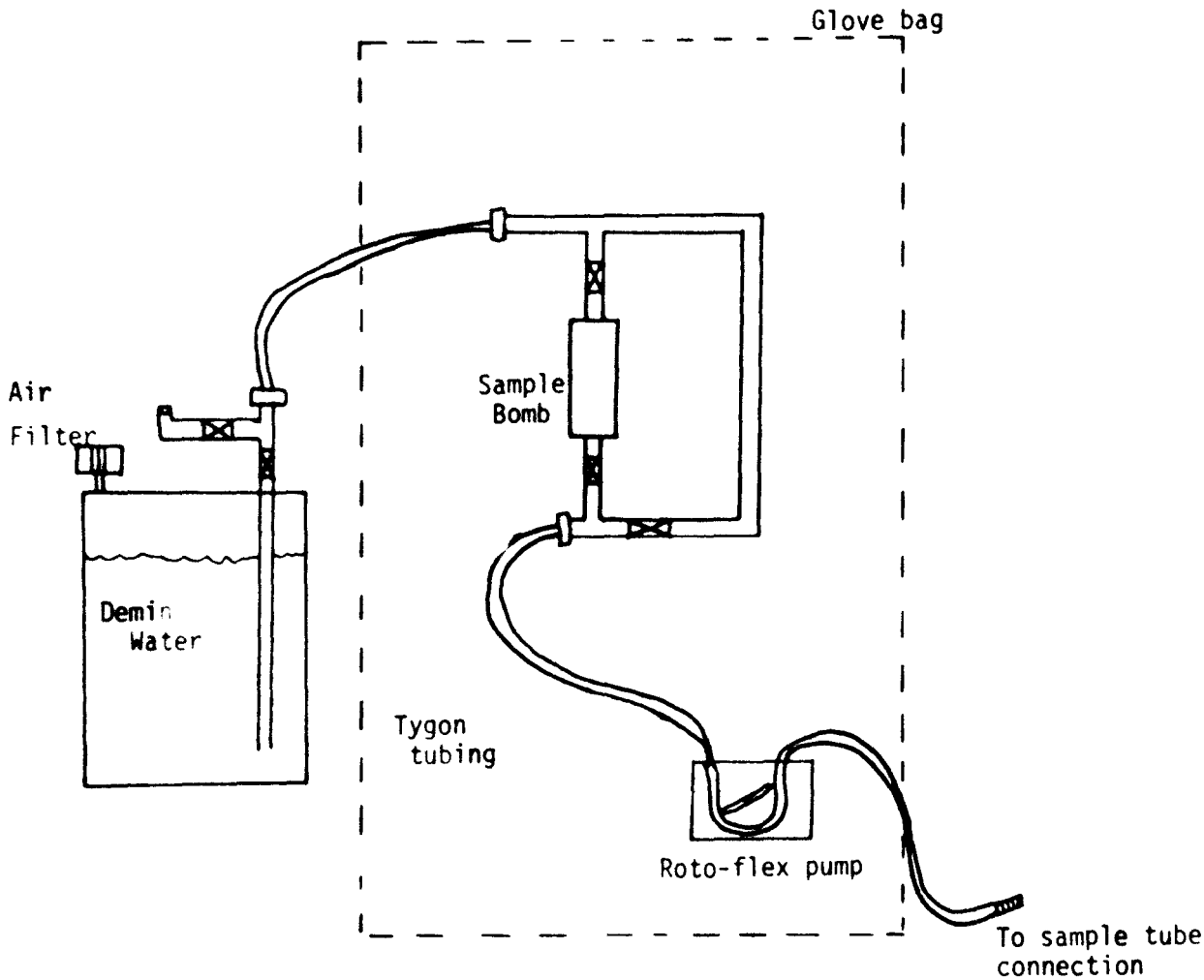


FIGURE 5

SAMPLING APPARATUS

SESSION G

CONTAINMENT ASSESSMENT TASK FORCE EXPERIMENTAL PROGRAMS

INITIAL ENTRY EXPERIMENTAL PROGRAM

Ed Walker
Bechtel

This presentation will cover the experiemntal program that has been conducted to provide an assessment of environmental conditions inside the Unit-2 containment. This data is to provide a basis for initial entry dose assessments and to provide input into decon and recovery planning. Bill Hopkins, in his paper, has discussed the results of experiments conducted from May to September. This discussion will include the most recent work, namely the penetration R-626 "Peep Show"; and will discuss future plans.

The initial radiological data provided by penetration R-626 was the 9-inch diameter cut out from the inner flange. This cutout is an unpainted stainless steel disk. Of particular interest were indications of the beta activity levels inside containment because of the high beta activities already encountered in areas of the Auxiliary Building. These areas have experienced reactor primary system coolant leakage and have been measured in the 10 R/hr to 100 R/hr gamma range with associated beta dose rates in the 1000 Rad/hr to 10,000 Rad/hr range. Using an Eberline RO-2A ionization chamber to survey the cutout, the readings was 2 mr/hr with the beta shield closed and 42 mr/hr with the shield opened. If you apply a factor of 4 for beta efficiency, the beta to gamma ratio, as indicated by the ionization chamber, was approximately 100. So, we are starting to see evidence of the same beta dose rate problem inside the containment that we have already seen in some of our support buildings and this is probably going to be what limits our planning and decon activities inside the containment.

One of the first experiments conducted through the penetration was inserting the TV camera into containment. At this time we'll make a little pitch for B & W. They came with a 15 minute edited film. There was recorded about 2 hours of film, but it becomes tedious for anyone to sit for 2 hours watching the camera focus on a dial or ventilation duct. There is about 15 minutes of the more exciting material, and B & W has set it up outside

on a small screen. It will have to be viewed in small groups. Basically, the views look like an operating containment with the lights turned out. One of the more interesting features is that it appears that the containment is raining ..... in other words, the water in the basement evaporates, goes up to the top of the dome, condenses on the top of the dome, and is falling back down. So, we are already starting to get Mother Nature helping us with our remote decon plans. It will be interesting to see what, if any effect, the "rain" has had on certain contamination levels inside the containment. Another condition noted was that there appears to be deposits of boron crystals. Again, this is consistent with high boron levels in the water and condensate dripping off components. There was also sodium hydroxide introduced into containment through the spray header which operated for approximately 5 minutes and sprayed approximately 500 gallons into the containment. Another 4500 gallons went into the primary system. This accounts for the high sodium level in the primary system.

The penetration was used to obtain a beta and a gamma dose rate map inside the containment. The results of that mapping are inconclusive to date. It is not certain whether there is a compatability problem between the instrument and the containment environment. The instruments that went inside the containment were air ionization chambers. We are currently evaluating response of these instruments to a noble gas cloud. Theres just not much to conclude from the mapping effort. Since the mapping was done, another gamma probe, a GM detecter, was inserted through the penetration with the probe 3 feet inside containment. The reading was 700 mr/hr. In Bill Hopkins' presentation, there was a chart with a predicted dose rate of 730 mr/hr at the operating deck.

Next, a series of air samples were obtained to measure noble gas, particulate, halogens, and airborne tritium. We are currently evaluating the data, which indicate that the krypton

level is in the range of 0.8 uCi/cc. The other sample results may be subject to interpretation because the air sampling system that was used was not designed to deal with the high humidity that currently exists in the containment. The results could be distorted by water depositing in the sampling lines.

There was also a series of smears taken. A special device like a shephards hook with a masolin cloth on it was used. Three smears were obtained on top of the flange thats on the inside of the containment, and another smear was taken off the wall directly behind the penetration. All that can be concluded from these smears in an indication of the smearable contamination levels that might be there. This can not be assumed to be the total contamination level. For example, the smearable activity on the wall behind the flange was roughly twice the activity on the flange itself. This is contrary to what we mormally expect with deposition activity where most of it should be on horizontal surfaces. Also, in attempting to correlate the activity level on the flange with that on the floor by making assumptions for smear area, the indicated smearable activity on the flange is two orders of magnitude lower than we are predicting on the floor level itself.

Some of the future experiments planned with this penetration include, inserting a TLD "tree" into the containment. This will include tests to determine the beta protection factor for protective clothing that is intended for the re-entry team. This will be accomplished by taking film badges and TLD badges that we currently use for dosimetry on the island and using them in pairs, one unchanged and the other covered with the protective clothing, These badge pairs will be mounted at different elevations on a wire frame and inserted into containment down to the floor. The badge pairs will be mounted on the frame at one foot intervals, for about six feet. This experiment will provide an unshielded and a shielded beta dose rate plus gamma doses at levels above the floor up to six feet.

An Eberline RMS-2 was inserted into containment above the TV camera, to obtain gamma dose rate readings. However, there is a compatability problem between the scalar readout and the dose levels in containment. The scaler used has a bottom range of 1 R/hr. When the detector was inserted, there was no reading because indications now are that the dose rate is less than 1 R/hr. This scaler is currently being adapted to the detector to provide a range from 1 mr/hr to 10 R/hr which should cover the range of any gamma readings expected.

There are additional experiments planned to obtain more information on conditions inside containment. Already discussed was the problem with the beta measurements. We're looking at more compatability between instrumentation and noble gas, and maybe the work groups can come up with ideas on how to better measure the beta fields that are in containment. It appears that the chambers will have to be sealed.

Also needed is a more refined way to obtain air samples if a better determination on particulate, halogen, and airborne tritium is desired. These activities are affected by water deposition in sample lines from the high humidity. Also of interest with future air samples in an ozone determination. The instrumentation people at the site have indicated that ozone in the containment atmosphere attacks any electronics with silver solder that is uncoated or un-waterproofed The ozone oxidizes the silver producing electrical resistance where you want conductance. It could be a problem getting electronics to survive when put inside containment.

Several experiments are either in the planning stages or actually underway. One experiment, for which the procedures have all been completed and we are awaiting the completion of the support equipment, is removal of a spool piece from the hydrogen recombiner. This experiment may give an indication as to what activity was in the air during the periods that the hydrogen recombiner was operated. Analysis planned for the spool piece include gross beta, gross gamma, and a beta and a gamma isotopic analysis.

Also in the planning stage is a Ge(Li) scan on the equipment hatch using a tilt table with a collimator. This will permit changing the angle of vertical scan and will be used to get an indication whether the air coolers inside containment contain a high level of plateout activity.

Another program under consideration is called "Peep Show 2". This involves drilling through another spare penetration that comes into the containment above the 305' elevation approximately 30 feet from airlock number 2. Activities proposed would include experiments similar to what has been done with R-626 penetration in terms of radiological data, air samples, contamination data on surfaces, etc. Another benefit from this experiment would be installation of a TV camera to obtain live coverage of the initial containment entry. It might be possible to negotiate with CBS for a couple of million dollars and get back some revenue for GPU. That program is currently on hold and the benefit of data that would be obtained being analyzed.

That summarizes the containment assessment program that has been completed to date.

Q. How high above the floor is the penetration with the TV camera?

A. The penetration is 11 feet above the operating deck. The operating deck is at elevation 347 feet 6 inches, and the penetration centerline is at 358 feet 6 inches.

Q. Why was the penetration flange of stainless steel?

A. I don't know whether there was a material compatability concern or what. If you go to Unit-1, the spare incore instrument penetration is a painted carbon steel flange. The Unit-1 penetration pipe is like R626, a carbon steel pipe, but R-626 has unpainted stainless steel flanges.

Q. Why are the lights out in containment?

A. The lights were turned out. We really don't know whether or not the lights survived the initial transient. About 3 or 4 weeks after the initial transient, it was suggested

that if the lights had survived for whatever reason, we should turn the switches off just to preserve them in case we wanted to reactivate them later. However, the design of this facility is such than in order to turn them back on again, you have to manually push a re-set switch which is inside containment. The TV camera was focused on one of the working lamps on the top of the crane rails and the view is inconclusive. As best as we can tell, the glass does appear to be intact.

Q. The level of the airlock is located at what station level?

A. The airlock is at 305 foot elevation which is essentially ground level.

Q. What is the krypton inventory inside the containment?

A. I think Bill mentioned earlier, it is somewhere in the range of 50,000 curies.

Q. What type of air samples did you take and what were the results?

A. We used an air pump type system with sample bombs for gas analysis. We also used particulate and iodine canisters. In fact, the iodine was set up to do a species analysis where three different types of absorbent are used. The sample bomb was used for krypton and diatomic gas analysis. The iodine analysis indicated that Iodine - 131 levels are below MPC inside containment.

SESSION H

CONTAINMENT ASSESSMENT TASK FORCE EXPERIMENTAL PROGRAMS

INITIAL ENTRY TEAM PROGRAM

Mike Morrell
GPU

Good afternoon. Shortly after the accident at Three Mile Island, GPU set up this containment assessment task force which I have been involved in. The objectives of the assessment task force are shown in Figure 1. We had basically three goals; the ultimate goal was to go into the reactor containment building to break down the technical and psychological barrier that's keeping us from recovering the plant. Before that, we hoped to purge the reactor building, however, we do not necessarily consider this to be a prerequisite for reactor building entry. It is definitely desirable, but we believe that reactor building entry is possible without reactor building purge. The initial part of it, however, was to establish without actually going into the building what the re-entry environment was.

The real objective of our efforts to determine environment and to re-enter the reactor building is to obtain technical data on the contamination and radiation in the building; we want to know what kind it is, beta, gamma, alpha. We want to know what chemical form it is and where it's at. The information is needed so that we can adequately plan recovery and decontamination. It is also important to you as representatives of the industry, so that you can understand the mechanisms that go on inside a reactor building when an accident similar to that at Three Mile Island occurs. So there's a lot to be gained for the recovery effort, but there's a lot to be gained just in terms of accident dynamics which will be important to each of you. We hope also in our entry to obtain some material for some decontamination studies in addition to what we already have. As you know, right now we have two cookies from the R401 and R626 penetration. That is essentially all we will have until we send people into the building to bring back something for decontamination studies. We hope then to get a preliminary

visual assessment of the damage. The camera through the peep show indicated there was no damage, but that was only one level. We believe that is what we will find when we actually walk into the building, but the visual assessment of the damage is important, for our planning of the recovery. When we enter, we may decide to establish some kind of permanent radiation monitoring in addition to the permanent monitor which has been installed through penetration R626.

We have gone over each one of the experiments that are listed in Figure 2, so there's no need to cover each one of them. It's just important to realize that we did sit down and , with a little bit of forethought back in May when we set up the containment assessment task force, listed the things we could do to establish the re-entry environment and set out on a program to determine exactly what that re-entry environment was. These consist of routine things such as air samples to relatively exotic things such as the "peep show".

As part of our re-entry program, we hope to purge the reactor building (see Figure 3). As you know, the krypton 85 concentration is approximately .8 micro curies per milliliter in the reactor building. There are other isotopes present but they're importance is minimal compared to the krypton 85. We did a pretty extensive study to determine exactly what the best method should be for treating the air in the containment building and our recommendation to the NRC, which we submitted on 13th of November is that we vent the reactor building atmosphere to the environment in a controlled manner. Our study included a study of cryogenic processing, gas compression, charcoal absorption and atmospheric dispersion by controlled venting. The basic difference between the first three and the last one is that atmospheric dispersion solves the krypton 85 problem.

The other three only transfer the krypton 85 problem from one storage vessel into another storage vessel. The other reason for selection of atmospheric dispersion is that the other methods require a very long period of installation. The least time estimate for any of the alternatives is for use of the cryogenic processing system. This is because there is a cryogenic processing system available from the Limerick Nuclear Station. This system was to be discarded. The estimate for installing and making the system operational is 20 to 30 months. The direct cost estimate for the system is about 10 to 15 million dollars. If allowance for funds used during construction and replacement power due to the 2 to 4 year delay were factored into the estimate, the total cost is somewhere between 210 and 300 million dollars. Cryogenics is also the least cost estimate of the four, by the way.

In contrast, the atmospheric dispersion essentially can be done now with some minor modifications to the hydrogen control system that we have installed for our post loca use and the system that we would intend to use will use a meteorological feed back system which would take advantage of favorable meteorology to disperse the krypton 85 in a controlled manner over a 30 to 60 day period. Our studies show that the maximum skin dose from releasing that approximately 44,000 curies of krypton 85 would be five millirems and the maximum whole body dose would be 0.1 millirems. This is in comparison with the 10 CFR 50 Appendix I ALARA limits of 15 millirems skin dose and 5 millirems whole body dose. These doses are for the fifty mile radius around Three Mile Island.

I've gone over purge briefly, but what I want to talk about for the remainder of this presentation is the preparation for doing the actual re-entry of the reactor building.

We've broken that re-entry into two parts which are support facilities and administrative support. (see Figure 4) The support facility essentially included the control envelope around the No. 2 personnel air lock of which you'll get a sketch in a minute, plus deciding just what the people should wear and carry when they go into the building. We will go into those in a little more detail. Administrative support includes radiation mapping, that's the determining of the re-entry environment which is under way. We will do the experiments, Bechtel will do most of the analysis and then Bechtel will aid us in coming up with a radiation map which will be used to select the path to be taken by the re-entry team. We also have to go through an entry team selection, procedure preparation film training and determination of exactly what data we want to retrieve from the building.

We have selected the re-entry teams for re-entering the reactor building. The criteria that we used is knowledge of containment layout, health physics knowledge, plant operational understanding and physical fitness. (see Figure 5) We believe that the first team we have selected meets these criteria. We have real problems finding people that are available and meet the criteria to make up these teams. We do intend to send in three members with a back-up team standing by outside and also a decon team, I'll show you how these work on one of the future slides.

Our re-entry training (see Figure 6) includes familiarization with the procedures, a review of radiological conditions inside the building and a review of normal radiological rules of thumb and health physics rules that can be used to ensure that they understand exactly what they're getting into, so they understand the doses that they may receive, and so they understand

how to minimize their dose when they go into the building. There will also be some containment building familiarization training which will include use of the model. The model has been refurbished as stated earlier, however, that refurbishment was essentially designed only to aid in the re-entry preparation for now. That model was only refurbished outside the D ring and from the 305 elevation up, because we believed the re-entry team will be restricted to that area during the initial entry. The yellow strips in that model, are called the yellow brick road. This is the preliminary path that we've laid out for the re-entry team members to take when they enter the building. We will also go through a clothing and equipment use test and hopefully run through a complete dry run in a darkened TMI Unit 1 containment building prior to the entry into Unit 2.

Our entry plan uses No. 2 airlock with the ante-room outside (see Figure 7). That anteroom will be maintained at a negative pressure and the negative pressure will be maintained by exhausting the air through a filtered system having air flow into the ante-room so that any contamination brought out during the entry or after the entry is handled through those filters. We'll have a decon team which will meet the team members when they come out to aid in the disrobing in this area and also to check them for contamination. Theoretically, when they reach the step off pad and go into the next room they should be uncontaminated. Should they be contaminated, they will go through a door to a shower facility which is located in the support building. The back-up team is available to go in in case of some kind of emergency where they might be needed to aid the team members inside the building. We'll have a command center right here which will have a communications center so that we can record all of the communication between the command center and the people inside. Plus, we'll record all the communication between team members.

The criteria for items to be worn or carried by the members are shown in Figure 8. We've been trying to select some of these items, and we have made some selections; we'll go over those briefly in just a moment. It is noted that the radiation levels in the building are not as high as we originally predicted.

As stated earlier, the containment lights are inoperable. One of the questions was, why is that. There is a good reason for some of them being inoperable and that is that the panels which control the lights are on the 282 ft. level and they're covered with water. So probably 3/4 of the lights are shorted out by that water on the floor. The rest may or may not be shorted out, but as you know, we have to reset the switch inside the building so there is no chance for energizing the lighting prior to entering the building. As a result, some primary and secondary lighting will be used by the team members (see Fig. 9) and also the light at the airlock door will be set up and hopefully it will illuminate all of the 305 elevation. There's a good possibility that the team members may be restricted to the 305 elevation during the initial entry. That's possible because we're not really sure of hot spots; although according to our estimates the 305 elevation is probably a higher radiation field that the 347 elevation due to the fact that there is still water on the 282 ft. level.

The breathing air system has been selected (see Fig. 10) and that system will be an enclosed system that recycles the air that's breathed, removing the CO_2 and supplementing the oxygen from an installed oxygen bottle in the system. In addition, there will be a back-up oxygen system which is not a very sophisticated system; it will require pulling the mask that they are wearing away from the face and inserting the tube from the oxygen bottles into the system to give a flushing flow of oxygen while the people leave the building.

That system is not very sophisticated and one of the reasons why we really want to get the building purged is that we can shield against the beta dose from the krypton 85 as long as they're completely covered. However, as soon as you would pull away from the face then the lens of the eyes are exposed and the lens of the eye with that beta field would probably be limiting. As a matter of fact, with a 3 to 4 hundred rad beta dose rate in that building, the stay time to exceed the 7½ rem lens of the eye skin dose limit is probably on the order of 1 to 2 minutes.

The communications system has also been selected. (see Fig. 11) We looked at a lot of different systems, some more sophisticated than others and we tried to pick a relatively unsophisticated system made by Motorola. This will be a wireless system so that there will be no wires trailing behind the members. We will have to use an antenna which will be installed through penetration R626, that's the peep show penetration up on the 347 elevation. In addition, another antenna will be taped to the glass window on the outer door of the airlock through which they will enter. That will allow communications while they are in the airlock and the antenna through R626 will allow communications while they are in the building. This will be a two channel system and all communications will be recorded, as I said before. In addition, we're also considering telemetering the dosimetry that is worn by members out to the command center; that has not been decided on definitely though.

The protective clothing has been chosen for two cases, the no-purge and the purge situation. We haven't actually placed that on order, however, pending some experiments which have already been described by Ed Walker. Basically, we want to do some shielded and unshielded TLD measurements through R626 to make sure that the things that we have chosen would be effective in shielding against the beta dose we see in the building right now.

So, refering to figure 13, assuming we determined the re-entry environment, we selected our re-entry personnel, we've evaluated all the data from the previous tasks and we've trained the re-entry team, we intend to enter the reactor building in late January. When we enter the building, we will do radiation surveys, swipe surveys, visual assessment, install some TLD's to do some other time exposures to determine better the radiation environment, obtain some materials for samples and also obtain photographs. Our current plan is not to carry in video tape or movie camera type equipment since the weight of that is just too much for the re-entry team.

That sums up our re-entry team program and our plans for re-entering the reactor building. We believe this program is very important; as a matter of fact, purging the reactor building and re-entering the containment building are probably the first major psychological barriers that we have to overcome in recovering Unit 2. As a result, we're trying to put as much emphasis as we can on it.

I'll entertain some questions now if there are any.

Question: Will you tell us a little bit about (1) the internal quality assurance program for making sure the re-entry procedure is the best possible and (2) what your're relationship will be with NRC relative to re-entry.
Answer: The re-entry procedures are being written by GPU personnel. One of the main people writing the procedure is Ed Walker, who had had access to extensive experience in this area through Bechtel, through his connections through Bill Hopkins and the various committees existing with the DOE. We believe we'll obtain some very good input from previous experience; we're trying to obtain that as much as possible. In addition to obtaining input from past experience, however, there's a review procedure which included review by the plant operations

review committee, an NRC sign-off which includes an extensive review by them and a review by an ALARA group which is headed by Paul Ruhter who will speak next. So, using industry experience reviewed by the plant committees and then having to obtain approval by the NRC, we believe that adequate assurance will be obtained to ensure that we really have a good procedure and we understand that we are getting into when we go into the building. I think I explained the interface with the NRC, but basically our philosophy since March 28th is that the NRC approves essentially everything we do. So we will not enter the building without NRC approval.

Question: How long do you actually envision the stay time for the initial entry and what do you possible predict now for the second entry; are we going to wait about a month?
Answer: The stay time will probably range between 15 min. and one hour: there's a big discrepancy there because although we know the general area radiation level, hot spots will be a problem. For instance, when you walk into the No. 2 personnel airlock, the containment air coolers are just to the left, just past the stairwell, and that is known to be a hot spot. I think it's realistic to say that probably no more than 15 min. on the initial entry. We'll gain a little bit of information, but the main thing is that we will have broken down the psychological barrier, we'll have a procedure that has been tried and tested and then we should be able to make subsequent entries for further the reconnaisance. No second entry is actually planned, however I think it's fair to say your time frame of about a month is a reasonable time frame that we would re-enter the building to do some more reconnaissance. My problem is to get into the building, find out what is there and esentially I will have worked myself out of a job at that time. At that time, Bechtel takes over the recovery and engineering group and they will then do whatever

they need to do to further determine the environment and plan the recovery and decontamination.

Question: What's your planned exposure limit?
Answer: The people will not be allowed to exceed the 3 rem per quarter dose. Planned exposure is as low as reasonably achieveable.

Question: Do you have authoritative representatives on the site to approve the procedure?
Answer: Yes. We've had NRC representatives on the site since the accident. I believe there are about 20 on site right now. John Collins is the home site representative and he or his designated representative signs off every procedure on sight, so the answer is yes they will sign off and the review team from the NRC is readily available to us.

Question: Are there any plans to rehearse the re-entry into the containment on Unit 1 or any other place?
Answer: Yes. We do hope to rehearse the entry in the Unit 1 containment building with the lights out so those guys will know really how dark it can be in a building like that.

Question: Can you describe the apparel that will be worn?
Answer: I can basically say that it will be some kind of material like betaguard, if you know what that is. If you don't we could leave that question until Paul Ruhter gets up here as the next speaker. He can expound on it a little bit more, but it has not been selected. I'm thinking in terms of a layer of beta-guard and perhaps a layer of anti-C's and plastic anti-C's over the top. That's really about as far as we can describe it now.

Question: How will you protect your instrumentation from the high humidity; from getting erratic reading on the portable instrumentation your team takes in?
Answer: If we purge the building there won't be a high humidity environment theoretically. If we don't purge the building, we believe that the high humidity won't harm the most of the instruments we carry in for the short time that they're in. There is some evidence, however, that submerging them in a field of krypton 85 may alter the readings of some of the beta and the gamma probes that we've used so far. The best answer to that is it needs to be investigated.

Question: Will an NRC employee be a member of the team?
Answer: There was talk of that originally but our plan now is that three member of GPU have been selected; there will be no NRC representative.

Question: Mike, I'm surpresed you're not going to put a remote TV camera in there on your initial entry so that you can leave it there to gain further information as you exit.
Answer: We're not sure there's anything to be gained by further pictures, even on the 305 elevation. We are considering that experiment through the R508 penetration which is on the 305 elevation just to the right of the No. 2 personnel airlock. But, if we go in take direct radiation measurements and if we already have the pictures on the 347 elevation, we're not necessarily convinced that there's anything to be gained by taking further pictures. There may be some public relations and some historical value to filming the initial entry; I guess it could be compared to putting a man on the moon or something like that, however, I don't think it's going to receive the wide media coverage that an event like that would have. Therefore, we're not convinced that movie cameras taken in will be a benefit to GPU or our R & D program.

Question: What is the logic of a three person team as opposed to two persons?
Answer: That can certainly be debated a long time. As a matter of fact in Bechtel's initial study they recommended a three man team and there was wide disagreement among their task force that finally made the recommendation.
However, the real logic is that we essentially want to use the buddy system. The buddy system would require two people to be together, however we believe it is safe for one person to stand at the airlock door and perform some experiments himself, take some radiation measurements, take some extensive swipes around the airlock door, set up the beacon light, etc. Rather than having somebody with him, we believe it would be safe to leave one man at the door and have the other two go together on the 305 elevation and/or up to the 347 elevation for reconnaissance. That really forms the basis for our three man estimate. Do you want to add something to that Ed? If you didn't hear that, what he basically said was that we want to take so much equipment in and gain so much from the initial entry that two people might not be able to accomplish it or carry in the equipment.

Question: Why did you not go with a fresh air system? I understand the system is going to have oxygen. Fresh air generally adds a lot to the cooling of the suit?
Another voice: For one thing, it's heavy.
Mike: That's part of the answer right there. We looked at a lot of different systems and the system was selected partly by our safety personnel on site and we're trying to keep the amount of the equipment carried in terms of total weight down to some minimum, preferably below 50 lbs., and the system we selected has a fairly low weight. Do you know the weight of it, Jim?
Jim: I think the total system is going to be below 80 lbs.

Mike: So you know we're already talking above 50 lbs. which was our initial goal, and this system is just lighter than most other systems available, and that's one of the reasons we selected it. We're not using any kind of suit cooling, I think that was part of your question. We did look into NASA suits and things like that, but we finally rejected that idea, I think permanently.

Question: Who manufactured your oxygen system?
Answer: I don't know the answer to that, but I can find out and let you know if you want me to.
Another voice: Currently, the only oxygen systems on the market have protection factor of only 50 and the only one I know of that's in testing is Bio-Marine and they haven't come out with a number for us yet.
Mike: I'm told that it is a Bio-Marine system, I don't know the protection factor of it and that raises a point which obviously I have to look into to make sure there is an adequate protection factor.
Paul pointed out that we thought that the air in the containment building is oxygen deficient, we've had quite a few measurements that say that. So we needed to have some oxygen supplement when we went into the building. So we do have to carry our own supply of oxygen for the people to be able to breathe if we go in without purging.

Question: Does your physical fitness include psychological?
Answer: Yes. The entry team members, the primary team has already had physicals at the Hershey Medical Center, and it looks to me to be a very extensive physical at least comparable to that given to a professional football player. It also includes psychological questions and psychological profiling by the doctors at Hershey Medical Center.

Question: The reading of the radiation monitoring equipment undoubtedly is going to pose a problem, how do you intend to handle that?
Answer: Are you talking about reading what they hold. It'll pose a problem but we believe that the lights carried on the miner's lamp type of thing will allow the guy to read. Say he's carrying a monitor, when he looks down at it, we're pretty sure that adequate lighting will be available to read that. If it's not we'll find that out during our proof test in Unit 1.

Question: What's the temperature inside the containment?
Answer: 75 degrees and about 100% humidity and .5 PSI negative pressure.

Question: NOT LEGIBLE ON THE TAPE
Answer: No I don't, however, as part of the dynamic analysis that was done to determine where the alleged detonation that occurred one or two days after the accident, try to determine whether a detonation occurred and if so, was it localized or was it generalized. Extensive data was taken on all the temperature sensors, do you recall, Frank or Bill, what the highest reading was? It was in the 140, 150, 160 degree range.
Another voice: 155, I think
Another voice: Mike, that might have been as high as 175.
Mike: I wouldn't dispute that. There have been a lot of different numbers thrown about in that analysis.
Thank you.
Ed Walker: I guess there's a couple of points I want to sort of clarify here, when Mike mentioned the 5 millirem skin and the .1 millirem for the gamma, for those who are involved in these Appendix I calculations that really is for the guy at the site boundary. When you average that over your total sector of say out to a 50 mile radius, the dose drops down tremendously.

So actually it's the guy sitting out there in his birthday suit next to the river getting all of it as it comes past him. That's the way you do the Appendix I calculations.

I guess one of the things I really want to clear up is the fact that when the hydrogen detonation occurred, you had the 28 PSI peak that turned on the containment spray system for approximately five minutes. I haven't had a chance to look at the films, but you would expect to see some of the boric acid and sodium hydroxide crystal that's probably washed the upper operating deck, the 347 deck, and cascaded down to the 305 level and even into the 282 basement level. So this is one of the things the initial entry team has to look for. Also, you have to look at the associated chemical reactions that go along with that type of chemical mixture of sodium hydroxide and boric acid.

Now, what we would like to get into is the current Unit 2 health physics program, Paul Ruhter is head of the ALARA group and he will be talking about their current procedures and practices and dosimetry program that they've had ongoing and some of these high beta fields that we mentioned previously. So, Paul.

OBJECTIVES

- OBTAIN TECHNICAL DATA ON CONTAMINATION AND RADIATION
  - MAGNITUDE
  - IDENTITY
  - DISTRIBUTION
  - CHEMICAL FORMS
- OBTAIN MATERIAL FOR DECONTAMINATION STUDIES
- PRELIMINARY VISUAL ASSESSMENT OF DAMAGE
- ESTABLISH PERMANENT MONITORING

FIGURE 1

ESTABLISHING THE RE-ENTRY ENVIRONMENT

- WEEKLY AIR SAMPLES
- EQUIPMENT HATCH GAMMA SCAN
- GAMMA SCAN THROUGH PENETRATION R605
- SUMP WATER SAMPLE
- GAMMA SCAN THROUGH PENETRATION R626
- PERSONNEL AIRLOCK RADIATION SURVEY
- PERSONNEL AIRLOCK AIR SAMPLE
- PEEP SHOW
- HYDROGEN RECOMBINER SPOOLPIECE ANALYSIS
- AIRLOCK ENTRY

FIGURE 2

REACTOR BUILDING PURGE

- KRYPTON 85 CONCENTRATION $\approx 0.8_4$ Ci/ML
- FOUR ALTERNATIVE TREATMENTS CONSIDERED
  - CRYOGENIC PROCESSING AND STORAGE
  - GAS COMPRESSION AND STORAGE
  - CHARCOAL ADSORPTION AND STORAGE
  - ATMOSPHERIC DISPERSION BY CONTROLLED VENTING
- REQUEST TO VENT SUBMITTED TO NRC 11/13/79
- OFF-SITE DOSES DUE TO VENTING
  - MAXIMUM SKIN DOSE ≈ 5 MREM
  - MAXIMUM WHOLE BODY DOSE ≈ 0.1 MREM

FIGURE 3

ENTRY PREPARATION

I. SUPPORT FACILITIES
   - A. CONTAINMENT CONTROL ENVELOPE
   - B. PROTECTIVE CLOTHING
   - C. COMMUNICATIONS SYSTEMS
   - D. BREATHING AIR
   - E. LIGHTING

II. ADMINISTRATIVE SUPPORT
   - A. RADIATION MAPPING
   - B. ENTRY TEAM SELECTION
   - C. PROCEDURAL SUPPORT
   - D. ENTRY TEAM TRAINING
   - E. ENTRY DATA RETRIEVAL

FIGURE 4

RE-ENTRY TEAMS

- MEMBER SELECTION CRITERIA
  - KNOWLEDGE OF CONTAINMENT LAYOUT
  - HEALTH PHYSICS KNOWLEDGE
  - PLANT OPERATIONAL UNDERSTANDING
  - PHYSICAL FITNESS
- PRIMARY TEAM: 3 MEMBERS
- BACKUP TEAM: 3 MEMBERS
- DECON TEAM: HEALTH PHYSICISTS

FIGURE 5

RE-ENTRY TEAM TRAINING

- PROCEDURES
- RADIOLOGICAL CONDITIONS
- PHYSICAL
- HEALTH PHYSICS
- CONTAINMENT BUILDING FAMILIARITY
- CLOTHING/EQUIPMENT USE
- DRY RUN/PROOF TEST IN TMI UNIT 1

FIGURE 6

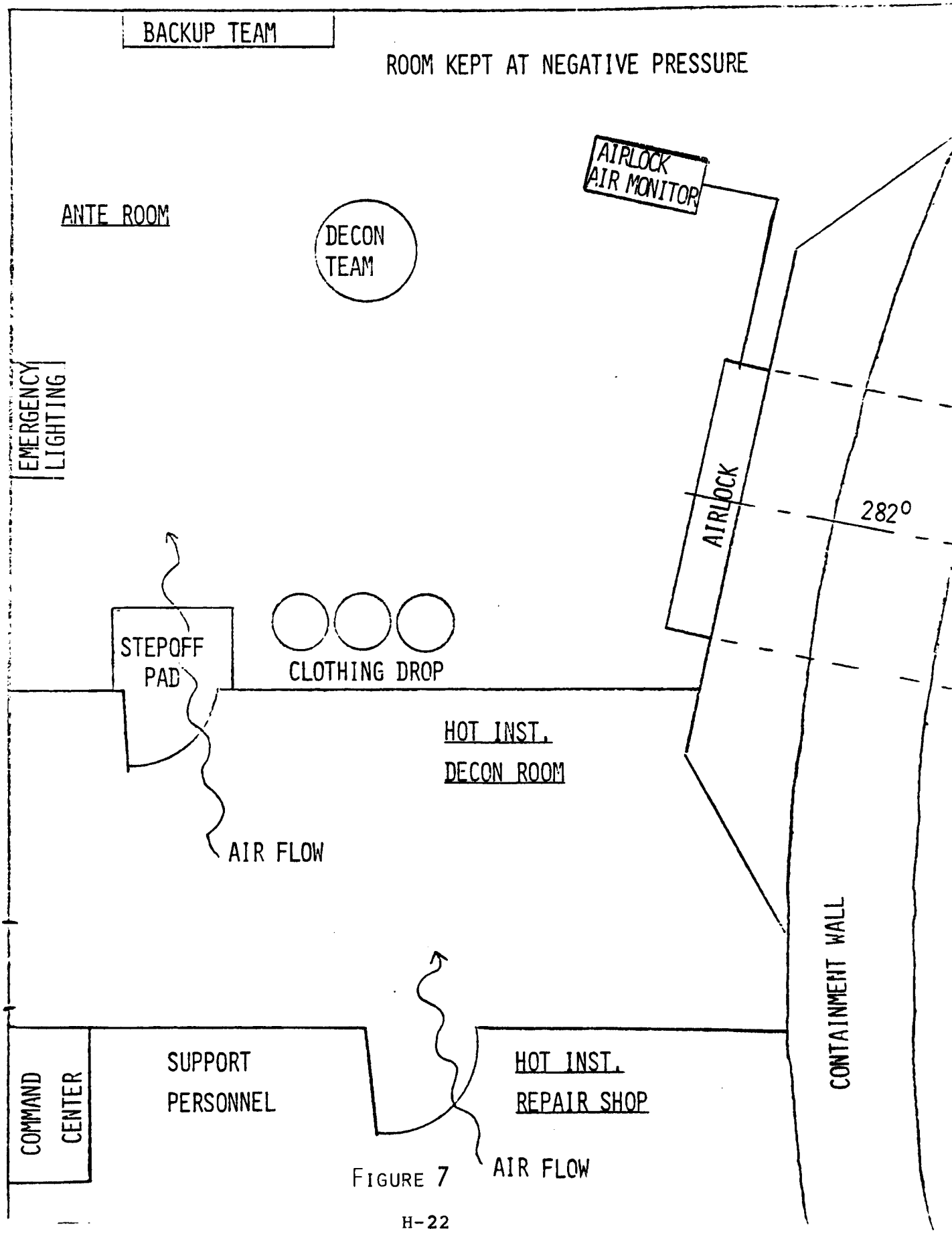

BACKUP TEAM
ROOM KEPT AT NEGATIVE PRESSURE
AIRLOCK AIR MONITOR
ANTE ROOM
DECON TEAM
EMERGENCY LIGHTING
AIRLOCK
282°
STEPOFF PAD
CLOTHING DROP
HOT INST. DECON ROOM
AIR FLOW
CONTAINMENT WALL
COMMAND CENTER
SUPPORT PERSONNEL
HOT INST. REPAIR SHOP
AIR FLOW

FIGURE 7

CRITERIA FOR ITEMS WORN/CARRIED BY ENTRY TEAM

- MOBILE
- LIGHTWEIGHT
- COMPATIBLE WITH OTHER EQUIPMENT
- RESISTANT TO CONTAINMENT ENVIRONMENT
  - HIGH HUMIDITY
  - HIGH RADIATION
- DECONTAMINABLE/DISPOSAL

FIGURE 3

LIGHTING

- CONTAINMENT LIGHTS ARE INOPERABLE
- PRIMARY LIGHTING: MINER'S LAMPS, BATTERY OPERATED
- SECONDARY LIGHTING: BELT MOUNTED, BATTERY OPERATED
- BEACON LIGHT AT AIRLOCK DOOR

FIGURE 9

BREATHING AIR

- PRIMARY SYSTEM HAS BEEN SELECTED
  - RECYCLE AIR
  - CO_2 SCRUBBING
  - OXYGEN SUPPLEMENT
- BACKUP SYSTEM
  - OXYGEN BOTTLE: 5-15 MINUTE SUPPLY

FIGURE 10

COMMUNICATIONS SYSTEM

- WIRELESS
- ANTENNA THROUGH R626
- TWO CHANNELS, 450 MHz FM
- RECORDED AT COMMAND CENTER
- TELEMETERED DOSIMETRY
  BEING CONSIDERED

FIGURE 11

PROTECTIVE CLOTHING

- MUST SEAL TO PREVENT CONTAMINATION
- MUST SHIELD AGAINST BETA RADIATION
- FINAL SELECTION DEPENDENT ON ADDITIONAL EXPERIMENTS THROUGH R626

FIGURE 12

CONTAINMENT REENTRY PLAN

- o SELECT REENTRY PERSONNEL
- o EVALUATE ALL DATA FROM PREVIOUS TASKS
- o TRAIN REENTRY TEAM (USE UNIT 1, MODEL, AND/OR MOCKUP)
- o INITIAL ENTRY
  - RADIATION SURVEY (GENERAL AREA, HOT SPOT, AND BETA)
  - SWIPE SURVEY
  - VISUAL ASSESSMENT
  - INSTALL TLD'S (OR OTHER MONITORS) FOR TIME EXPOSURE
  - OBTAIN MATERIAL SAMPLES IF POSSIBLE
  - OBTAIN PHOTOGRAPHS

FIGURE 13

SESSION I

TMI-2 HEALTH PHYSICS PROGRAM

P. Ruhter, GPU

To date, most of our decon activities have been in the fuel handling and the auxiliary building. The beta fields we've experienced aren't the krypton fields we're speaking of or concerned about in the reactor building, but are fields that involve mostly cesiums and strontiums. I'd like to discuss our current practices in three different areas: the field instrumentation we're using, the personnel dosimetry we're using and the protective clothing we're using.

The field instrumentation is basically the Eberline standard line of instruments; the R01 which is a Cutie Pie type instrument, the R02 and 2A which are box type instruments and a Teletector. The R02A and Teletector are used for most of our field surveys.

TLD personnel monitoring, is done with a Harshaw TLD badge, it has two chips; one under about 230 mg/cm^2 and one under an open window which is about 34 mg/cm^2.

Our protective clothing is cotton PC's in multiple layers. If necessary, paper anti-C's over the top, and wet suits if a wet environment is being encountered; booties, surgeons gloves and heavier gloves for hands. We're using a surgeon's hood for hair control and then a regular cotton hood that comes down over the shoulders for contamination on the head level. Initially we were in Scott Air Packs in the fuel handling/auxiliary building after the accident. As air activity decreased, we went to full face respirators. In the last month or so, for routine work, no respiratory protection has been required since the air activity is to background in these two buildings. In the cubicles, the air activity is higher and of course we're still using respirators and Scott Air Packs. Since decontamination, there has been a significant reduction in the air activity.

Figure 1 shows the mix of isotopes that have been discussed earlier today, but I would just like to refresh your mind on the concentrations of cesium 137, cesium 134, strontium 90, and strontium 89. The information shows the concentration in the primary cooling system as of September 5th and on October 15th. The mix here respresents better than 99% of the activity in the primary system. The reason this is significant is because everywhere we've had beta problems in cubicles and elsewhere, it has been where primary cooling system liquid has leaked out on the floor or on to the valves and piping. The November 15th data is hot off the press and we did not have time to incorporate it in the slide but the concentration of cesium 137 is 59 uci/cc, of cesium 134 is 11uci/cc, strontium 89 is 91 uci/cc and strontium 90 is 24 uci/cc. If you look at this information and the November 15th information which basically confirms that, you see that the cesium numbers are decreasing in time, the concentrations of strontium 90 which has a 30-year half life is strying steady if not increasing. The concentration of strontium 89 which has a 54-day half life actually reflects a 66-day half life. Now what that tells us is that the cesium concentrations are decreasing as you'd expect it because of the dilution, yet the strontium concentrations are increasing; we're getting more strontium in the primary system. Now that impacts our protection problems from the standpoint that it changes isotopic mix, changes the beta field mix and complicates things as time goes on. You can't go on the September data, or the October data or the November data when you're making an entry into the field in December, if there's been a recent leak.

The cesium isotopes have betas with energies in the 0.5 Mev range, the strontium-89, which has the highest concentration, has a maximun energy of about 1.5 Mev, yttrium-90 (the daughter of strontium 90) has a 2.3 Mev beta. We are looking at some rather high energy betas; we have not seen cerium and ruthenium with 3 Mev betas; however, we're looking for those.

You saw them on the coupons and in the solid materials discussed by Jack Daniels so we suspect we'll see more of them in time, but in the primary coolant in the Aux Building we have not seen them yet.

The mix here has a penetrating capability which is fairly astounding in the normal power plant industry beta health physics problems. Figure 4 depicts this on a table. This table is calculated mathematically from the data obtained in early September. In other words, we ran it through a computer program which considers the concentrations of the isotopes, the relative rations of all the different betas, the different fractional yields, etc. Those two columns compare milligrams per square centimeter on the left vs. fractional transmission on the right. For example, a shielding material of 13 milligrams per centimeter square will transmit 86% of the beta particles, 86% of the beta field penetrates 13 milligrams. At 300 milligrams per square centimeter, approximately 20% of the beta field will pass through and at 500 milligrams per centimeter square transmission is about 5%.

How does that impact us from a health physics standpoint? Let's look at the instruments we're using, the R01 is a Cutie Pie type instrument, it's wall is 300 milligrams per cm^2 thick. At 300 milligrams per square centimeter, we're still getting about 20% of the beta field penetrating into the sensitive chamber of the instrument; consequently, if we take a closed window reading with an R01 you get a tremendous over-response; you're not reading gammas, you're reading something that penetrates 300 milligrams per square centimeter. The R02 has about 500 milligrams, so at a 5% transmission through 500 milligrams, we're down to where that does not significantly affect the closed window reading. In other words, you really are looking at mostly gamma type exposure. The teletector on the very high range is very shielded

and doesn't have any sensitivity to betas, so when we're in high enough fields to use a teletector, we can't use the same instrument to measure the beta fields so you have to use a different instrument. Consequently, the HP must use two instruments which complicates things very severely.

Relative to trying to determine skin exposure vs. whole body exposure, we don't have a field instrument that can really tell us what skin exposure is vs. the penetrating exposure. In other words, according to the I.C.R.P./ N.C.R.P. type definitions, the skin exposure is the surface dose which penetrates 7 mg/cm^2 while the whole body dose is that which penetrates 1,000 mg/cm^2. The R02 measures those radiations which penetrate 500 milligrams per centimeter squared. So there's a little discrepancy, but as indicated above at 5% that's not a big deal.

On the dosimeter we're using, the Harshaw dosimeter, the deep chip is 230 mg/cm^2. Making reference at the chart above, at about 230 mg/cm^2, the deep chip filter transmits 30-35% of the beta energies through to the deep chip, which would normally be interpreted as the gamma component or the penetrating radiation. Obviously when working in the high beta fields, we get significant penetration of beta radiation into the deep chip. That really complicates the interpretation of that badge since you normally use the deep chip value as the gamma component and subtract that from the open window chip to obtain the beta dose. Well that doesn't work here because it's over-responding to the beta, consequently you have to interpret those badges differently than the normal badges in the rest of the plant where you see a more routine situation. So again, considering the fact that the standard is 7 milligrams and 1,000 milligrams for the differentiation between penetrating and non-penetrating exposure, we must make some adjustments in that badge or in our way of doing personnel monitoring before we can accurately and reasonably interpret personnel exposures.

Now, protective clothing; trying to protect from beta radiation or skin exposure that penetrates 300 to 400 milligrams per centimeter squared is a tricky problem. You can't do that with PC's of cotton coveralls unless you severly overdress him. So that's why we're looking at other material. Mike mentioned the material called beta guard; it's basically a rubberized suit with some lead impregnated into it. It appears to be equivalent to about 300 milligrams per square centimeter. Now that still gives us a transmission factor of only 20%. Against krypton 85 which has a much lower beta energy, it should give us substantially greater reduction factors than say a factor of 80% here. This beta guard would be made into something like a diver's suit so it essentially encloses the whole individual. We are procuring one of these to see how it really works; our exposures on small swatches looks like it might work very effectively but we have yet to have a full suit of it made. Presently, we're using heavy rubberized suits for any work that involves entering into high beta fields or cubicles with primary coolant system leaks on to the floor. We are using fireman's boots that are in excess of 500 milligrams per square centimeter on the bottom and sides. We are using heavy rubberized coats. One other problem is that the TLD dosimeter that we're using is a fairly flat two dimensional instrument. It's designed to detect beta radiation that's coming directly at it. If the dosimeter is oriented in a non-perpendicular fashion to the radiation, you get self-shielding by the dosimeter itself and sometimes this can be a factor of 10 to 100 reduction that must be accounted for. Typically, the way we have been accounting for it is by putting multiple badges on an individual; putting badges front and back, inside and outside the masks on the head, putting them on the wrists, ankles, thighs, just trying to measure everything we can without weighting him down with dosimeters.

Our current ALARA approach is to not enter into any of these fields until we absolutely have to. We are going to have to soon in terms of decontaminating the cubicles and getting them cleaned up so we can carry on with the program. But we're putting that off until we're absolutely sure we can get in and out without having any unusual exposures.

I think that is really the gist of what I wanted to cover. Are there any questions?

Question:
Are the general walkways the only place to be cleaned up?
Answer:
The decon efforts have concentrated in the hallways and the general open areas in the fuel handling and auxiliary building, there has been some decon in the cells and other cubicles.

Question:
This question is a follow-up of one that another gentleman asked an earlier speaker it involves budgeting radiation dose. When you are approaching the task, do you use a method of assigning a preplanned radiation dose any more narrowly than just quarterly limits?
Answer:
Oh yes. With this particular task of reentry, that's a major job. The individuals involved with the reentry are not routinely involved in radiation work so that allows us to think in terms of the whole quarterly dose.

However, each job is planned out from the standpoint of how much exposure do we think this job will take, what can we do to reduce it and seeing if we've complied with that or not. But it's basically a case by case job.

Question:

Are you doing your own on-site official dosimetry and if you are when you got all of these, on which one are you finally calling your official skin badge?

Answer:

We do our own dosimetry and we're taking a composite of the results. We had an over-exposure in August, for example, where it looked like a significant area of the leg was higher. The skin exposures elsewhere on the body were not significant but the leg happened to be close to a valve that was hot so we have to assume that the skin exposure basically. It's identified as being on the left leg or the right leg or whatever, but it's a skin exposure and it goes down in the record as such.

Question:

Your fractional penetration I presume will be based on a maximum beta energy so if that is true the real penetration will be quite a bit lower.

Answer:

No, that curve is based on the non-uniform spectrum of energies. By the way, it does bear out in terms of what we've done in field measurements by putting different shielding layers in front of badges and doing field measurements in cubicles with TLD arrays. We have plated a sample of the primary coolant system onto a plaque that we are using for calibrations in a little more controlled environment and basically the information confirms the data.

Question:

Isn't your beta problem primarily in the hands-on type decontamination?

Answer:

Yes, Sir.

Another voice:

I assume it would be a good while before you would be on the hands-on type decontamination.

Paul:

In the fuel handling auxiliary building that's what we're doing right now on a day-to-day basis because they were very heavily contaminated.

Question:

Then the major contamination would be to the hands.

Answer:

If you're not wearing something on your feet, you're walking around in it. And since it's on the walls and on pipes -

Question:

Is it localized?

Answer:

When a valve fails and sprays water everywhere, it is not very localized in this case. In some cases it is; if you're working on a specific valve that's isolated from the system it's going to be localized in there and that's not a big problem to control. But where you have a valve leaking as they did earlier, it gets sprayed all over the room and you get large areas that reads 100's or 1,000's of Rad/hr beta.

Question:

Do you hands-on decontaminate that type of level?

Answer:

Not any more we don't. At one point they didn't appreciate that a problem existed. This resulted in several individuals receiving inadvertent exposures on August 28, 1979. While repairing leaky valve in the makeup system.

Question:

(Illegible on tape)

Answer:

Okay. When we go in to decon those areas we're looking at using sprays to wash the floors and piping down. Okay, when you're talking hands-on, you're probably thinking of swabbing things down; we're not doing that kind of hands-on work. We're talking hands-on where you use vacuum cleaners and sprays that are working a few feet in front of you, but you still have to protect for it.

Question:

What has been the total occupational dose to date and also were there whole body exposures on August 28th or just skin?

Answer:

Just skin.

Question:

Have you calculated those numbers down to show the whole body exposures?

Answer:

Yes. The whole body exposures were less than 3 rem, more like .6, .8 rem.

Question:

What was the total occupational exposure to date?

Answer:

Total occupational exposure to date runs in the neighborhood of 1,000 man rem since the day of the accident. Now, in the last couple of months where we're to a non-emergency situation, the man rem exposures for August and September were 63 man rem apiece, October was about 55 man rem.

These valves are for both Unit I and Unit II. So for a plant that has several hundred people doing the kind of work they're doing that's not an unusual man rem. There were several hundred per month right after the accident, but those were extreme conditions and extreme operations.

Question:
You mentioned that plastics were not good absorbers for beta.
Answer:
No, I meant to indicate that they are good.

Question:
Has there been a safety analysis or hazard assessment made on this whole reentry operation and is it available for review?
Answer:
An assessment. It is being done and will be reviewed at the end of December when we submit the reentry procedure.

Question:
Have you considered the use of lead-loaded aprons, lead-loaded gloves?
Answer:
Yes, we looked at that type of material, but we've opted for the beta guard at the moment from the standpoint that it's lighter, more flexible, protects the back and front both and it looks like it will do the job. A lead-lined apron is something that's in the background for the moment; certainly that is a reasonable thing to do but it does get a little heavy and cumbersome. We have used the lead-lined gloves in a few operations that didn't require tactile sensitivity to speak of.

Question:

Can you help me out with the interpretation as to when you feel it necessary to discount doses and call emergency doses vs. 10CFR limit doses, have you tried this yet?

Answer:

No, with the exception of the exposures that occurred the day of the accident and I guess I can't answer whether we classed them as emergency or not. In the interim we've not classed any exposures as emergency.

FIGURE 1

Reactor Coolant System Activity Concentration are Dominated by Sr-89 and Cs-137

Concentration in uCi/ml

	9/15/79	10/15/79
Cs-137	80	69
Cs-134	16	13
Sr-89	188	106
Sr-90	20	27

FIGURE 2

Fractional Beta Transmission for Composite Beta Spectrum of Primary Coolant

Absorber Thickness mg/cm^2	Fractional Transmission
13	0.86
42	0.72
78	0.59
119	0.48
164	0.40
211	0.34
259	0.27
309	0.21
360	0.15
412	0.11
464	0.07
517	0.04
570	0.02
624	0.01
677	0.009
731	0.006

SESSION J

ORNL EXPERIENCES IN PERSONNEL MONITORING DURING HOT CELL RECOVERIES

E. D. Gupton
OAK RIDGE NATIONAL LABORATORY

I'm going to reiterate some of the things that already have been said. In an accident of this sort, you will have these sources of radiation (Fig. 1). Of course, you won't have the activation products, perhaps, outside the cooling system or containment building. You'll have a whole bunch of garbage as far as radiation is concerned, beta, gamma, x-ray, scattered photon, etc. and this can be a mess.

We have developed estimates shown in Figure 1, not knowing anything about the age of the TMI fuel, i.e., the megawatt days of operation. I tried to speculate what the relative abundance of the various activities were. I think your fuel was not as old as this fuel. This is fuel that's been irradiated somewhat more than a 100 days of operation and has since decayed 250 days, which is the approximate time since the TMI accident. You'll see that most of what you have here is beta radiation. Of course, krypton won't be in the auxiliary building and the other facilities outside the containment. I'm surprised that the last figure of the previous speaker showed that we had very little cerium 144. I don't know whether it's because it's volatile or it doesn't get into the auxiliary area. But in the Figure 1 case that would be the governing radiation at this time. Of course, these shorter lived activities, which already have decayed from a very high relative abundance, will be gone in a short period of time.

On the basis of this speculative mixture of isotopes that I showed, there are approximately six betas for each gamma ray. In terms of the dose in tissue for unit fluence, you'll get about 50 dose equivalent units per incident beta ray for each dose equivalent unit from an incident gamma ray. Beta then is definitely the controlling hazard for any external

radiation exposure. The beta to photon dose ratio (Ref. Fig. 2), particularly without much absorber between the skin and the surface that's contaminated, can be much greater than 100 to 1. Again, this demonstrates that beta radiation is the primary problem. As you have been told already, the commonly used survey instruments are inadequate for estimating these beta ray dose equivalents. The personnel dosimetry is definitely not simple, and if you have gaseous activity such as the krypton, not only do you have the problem of estimating the dose equivalent from both surface beta radiations and the gaseous beta beta radiations, but the gaseous beta radiations will affect most personnel dosimetry instruments in a way that the apparent dose equivalent is higher than what it actually was. This is because of beta particles from the gas being intimately mixed in the same environment with your dosimetric material. If you assume, as was said earlier that there may be 250 milligrams, 50 milligrams, i.e., some absorber between the external surface of the dosimeter and the sensitive system within the dosimeter, that absorber will not be effective for the beta particles emitted within the sensitive system. As was said, you may have to monitor various parts of the body, and/or definitely shield certain parts of the body from these radiations.

If we must work in the krypton environment, about 50 milligrams per centimeter square is a half-value layer for krypton betas. It was earlier questioned about the atomic number or the kind of milligrams per centimeter square for shielding betas. For beta radiation, the absorption is a very slowly changing function of the atomic number of the material that is used, and quite different from shielding x-rays or gamma rays, for which 1,000 milligrams per centimeter square of lead would be a very effective shield compared with a 1,000 milligrams per centimeter square of plastic. That is not the case with betas, the 1,000 milligrams of lead would be a very slightly better shield than the 1,000 milligrams of plastic.

Another concern in your personnel dosimetry is not only the thickness of the badge material, or whatever contains your sensitive device, but also what the thickness of the sensitive device itself might be. The Harshaw TLD chip is 345 milligrams per centimeter square thick. 345 milligrams per centimeter square is a bit more than a half-value layer for the most energetic beta that you have, and it's certainly a number of half-value layers for some of the lower energy betas that you might encounter. The dosimetric environment within the thickness of this chip is quite different from front to back, so it's not only important how the chip is oriented, but of very much concern about how you're going to do the dosimetry after you get a reading. The chip gives you a light output reading; in order to get dose you may have to know many other things.

As important as the personnel dosimetry, of course, is the type of monitoring instrumentation that one might use. In many cases, it is almost necessary to have an instrument that is a device that is hand held, for example, and that can measure the dose equivalents both to the 7 to 10 milligrams per centimeter square depth and also to a depth on the order of 1,000 milligrams per centimeter square. Then expose some of your personnel dosimeters on a phantom and relate the readings of those dosimeters to the measurements that you made with your instruments. I don't think that the regulations require us to measure beta dose per se, for example, vs. gamma dose. We are constrained to monitor or measure as best we can the dose equivalent to the superficial live tissues of the body at a nominal 7 milligrams per centimeter square depth, and to measure a dose equivalent to the deeper organs of the body, which can very well be assumed to be a 1,000 milligrams per centimeter square, minimum, with exception to the lens of the eye. Thus, your personnel dosimeter need not be, and perhaps should not be a device that measures beta and gamma or the difference between beta and gamma, but rad dosimeter for whatever the radiations (Fig.3).

There are many photons that have as little or less penetrating power within tissue than some of these betas that are listed. So one needs to measure the superficial dose without regard to the kind radiation and a so-called depth dose, again without regard whether some betas penetrated to 1,000 milligrams per centimeter square and be included in the so-called depth dose.

With regard for internal exposure, particularly in the auxiliary facilities, the far overriding concern is the inhalation of strontium 90, The relative hazard of the strontium 90 compared with the other activities present is on the order of 30 to 50 to 1, so that if you keep the strontium intake below the so-called regulatory levels, you need not be concerned about the intake of the other isotopes. The fission products will throughout almost all of this subsequent recovery operation outweigh any potential hazard from the alpha emitting activities that might be present. This is fact already discovered, particularly at the lower levels of the facility where water has stood. The majority of the activity is going to be on the floor and when one can do something to remove or shield most of the radiation coming from the floor he perhaps will have reduced by 50% or more the total radiation. That's all I have to say at this time; I'll be glad to try and answer any questions any of you may have.

Question: On the strontium you say it's 30 to 50 to 1 importance, is that specifically related to the Three Mile Island situation?

Answer: Well, once your shorter lived activities decayed away, I think most of those will be gone away before much is going to be done in this area, you'll have-although we were just told how the ratios of cesium to strontium seem to be varying - regardless of the age of the fuel and so forth, approximately one cesium isotope for each strontium 90 isotope. As to other activities, the so-called permissible air concentrations or the permissible amounts of activity in the body, the strontium is 30 to 1,

approximately more hazardous than all the other isotopes combined under these conditions.

Question: Ed, I wonder if you could tell us how the krypton beta might affect the thin wall, thin window ionization chambers and things like that.

Answer: Unless the monitoring instrument or personnel dosimeter is gas tight, and I would say it is all but impossible to make a good beta instrument that is gas tight, then as soon as you put that instrument into the atmosphere that has a partial pressure of krypton it's only a matter of a very short time certainly, at most a few minutes, and maybe only a minute or two until whatever the relative abundance of krypton in the atmosphere that you enter, that relative abundance will have occurred in the gas, which is usually air, in the device that you are using. Let's say, for example, that you had an ionization chamber with air filling and at atmospheric pressure. You want that chamber to breathe, because you want it to stay at atmospheric pressure, you don't want it to bulge or deflate as the pressure changes. The krypton betas have a maximum energy a bit under 0.7 MEV, 50 milligrams per square centimeter half-value layer. Let's say you had an instrument that had a wall thickness of 100 milligrams per square centimeter because you're going to suit your people out and all of the things you're going to put on them is 100 milligrams per centimeter square. Therefore, you'd like to measure the kind of dose they're going to get. So you enter this atmosphere properly suited out and you have a positive pressure in the suit and there will be no krypton inside that 100 milligram per square centimeter suit. So, all of the betas have to go through the suit. But that's not true of your instrument unless you also put the instrument in some sort of a gas tight bag or something. The krypton gets into the chamber and a lot of betas are emitted within the chamber and the instrument is going to read significantly higher than if it didn't have the gas in it. A similar thing can happen to your personnel dosimeter.

Question: Do you think the krypton will be diffusing through the poly bag that we would wrap an instrument in to carry it in?

Answer: No. That takes a much, much longer time; it will certainly if you leave it there for days. It will diffuse, but not within the stay time these people would normally be in these environments. You take the bag off that you used and put another one on before you go in there next time.

Question: Would it be very difficult to seal the bag adequately?

Answer: No, not if you were going to leave your instrument on a given range. If you had to get hold of something to make a range change or some check, but you can seal it adequately. We did some studies in a krypton atmosphere and the poly bag should be at least three mills thick. It will not allow a significant amount of the gas to diffuse for the times you plan to spend in initial reentry.

FISSION PRODUCT RADIATION

(PLUS 250 DAYS)

ISOTOPE	T½	BETA	GAMMA	RA
SR-89	51 D	1.46		1
Y-91	59 D	1.54		2.6
ZR-95	65 D	1.90	0.75	3
CE-144	284 D	0.32 2.97	0.13(11)	10
RU-106	1 Y	3.54	0.51(11) 0.62(11)	1
PM-147	2.6 Y	0.22		2.8
KR-85	10.7 Y	0.67		
SR-90	28.5 Y	0.54 2.28		1
CS-137	30 Y	0.51	0.66	1

FIGURE 1

BETA RADIATION CONCERNS

* BETA IS THE CONTROLLING RADIATION HAZARD DURING DECONTAMINATION WORK.
* BETA TO PHOTON DOSE RATE RATIOS MAY BE GREATER THAN ONE HUNDRED TO ONE.
* COMMONLY USED SURVEY INSTRUMENTS ARE INADEQUATE FOR ESTIMATING BETA DOSE EQUIVALENTS.
* PERSONNEL DOSIMETRY IS NOT SIMPLE, AND IS FURTHER COMPLICATED BY RADIOACTIVE GASES.
* MONITORING AND/OR SHIELDING OF VARIOUS BODY PARTS MAY BE REQUIRED.

FIGURE 2

GOVERNING RADIATIONS

SOURCES:

- FISSION PRODUCTS
- ACTIVATION PRODUCTS

RADIATIONS:

- BETA
- GAMMA
- BREMSTRAHLLUNG
- X RAY
- SCATTERED PHOTONS

FIGURE 3

SESSION K

AUXILIARY BUILDING DECONTAMINATION
WASTE PROCESSING EXPERIENCE

DECONTAMINATION EXPERIENCE

Tom Block
GPU

We've been invloved in the decontamination of the Auxiliary and the Fuel Handling Building on all elevations in areas that are accessible to us for decontamination. We also have responsibility for contamination control and collection and packaging of radioactive waste which is generated during the decontamination activities. We also support construction and operations during the recovery phase on Unit No. 2.

We're doing the decontamination in a multi-stage plan. Our first step is to get the accessible areas on all the levels in the Aux and Fuel Handling Building, down to 100,000 DPM/100cm^2. The second step is to get it down to 10,000 DPM/100cm^2 and then less than 1,000 DPM/100cm^2 down to design tolerances. The methods which we have employed so far in the decontamination effort have been to remove all the non-essential equipment which was generated during the initial recovery phase in late March and early April. This involved removing the equipment, staging, tools, barrels and boxes which are brought into the building for the recovery stages. The methods of decontamination which we have been using have been dry vacuuming with a HEPA filter; this is on non-wet floors and on pipeing and cable trays, and wet vacuuming after scrubbing down the floor with Radiac wash.

We use manual wiping of piping, components, either using disposable paper towels or Maslin wipes. Strippable coating has been used in areas where we have no coating on concrete. We also used the strippable coating on portable shields which we use for shielding access into certain areas. We have an electro-con unit set up which we're using for decontamination of tools and small equipment such as drain caps, small open and wrenches and other equipments which were used in the decontamination process. We have freon cleaning unit set up which we are using for decontaminating electrical tools. We can put them right into the unit

and decontaminate them and bring them out and they come out clean and can be placed in service after rewbescating. We have a Hydro-lasser which we've had limited use due to the restrictions on the use of water. We have used it on occasions in areas such as the annulus between the fuel handling building and the reactor building. We have a steam cleaner which to date we have not employed in the decon effort. We have it standing by though in case we need it.

The building atmosphere inside the Auxiliary and Fuel Handling building causes rapid exhaustion when decontamination is performed while a man is completely suited out.

During June, July and August and September we were running temperatures greater than 90 degrees in these buildings, so you can understand what the working conditions were. We have found bubble suits to be effective but they limit the access into areas for decontamination due to air supply problems and limited hose lengths. We have used Scott air packs, but the weight of the air pack causes exhaustion. The working time is restricted by the limited air supply and the working capabilities are also hampered by the bottles on the back. The care and cleaning of the respirators is very critical due to the suceptability of the plastic in the lens being scratched with resulting loss of visibility. The respirators also have a tendency to fog and anti-fogging agent must be used on the inside. We have used Scott respirators and MSA respirators. We find that the MSA breathes easier and adopts to eyeglass use easier but of course, we have to use a Scott on certain occasions in decontamination work because of the better protection provided. With regard to protective clothing, we have used cloth cover-alls and have found it very critical that the proper size cover-all be used on the individual doing the physical decontamination work. If it's too tight, it restricts them; if it's too baggy it tends to get caught and snag and prevent his mobility during his decon efforts. We have found cotton coveralls absorb sweat better than the nylon type. We've used plastic suits, but

there is a body heat build-up while working and they tear easy which causes problems. We've used paper suits and found that they also tear easily. We've found it critical that the proper size rubber gloves be used so that a man has the proper movement of his hands while performing his decon activities. It's very critical that the proper equipment and the sizing for the existing conditions is used during an effort such as this to allow greater effectiveness by the individuals performing the decontamination work. As shown on Slide 1, on April 27, 1979, when we commenced the decon, we found that the iodine was up to 5.4×10^{-6} µc/cc. Smears taken on the 281 level in the Aux Building were reading 15×10^{-6} DPM/100 cm^2. The general radiation level was 1 R/hr; there were some areas that were higher that, as we went up to the 305 level we ran into the iodine being 2×10^{-7} µc/cc. Smears were again 700,000 DPM/100cm^2 and the radiation level at the 305 level was 80 mR per hour. On the 328 level we had iodine concentrations of 1×10^{-7}; smears of 4000,000 DPM/100cm^2 and radiation levels of 10 mR/hr. This is what faced us when we entered to start the gross decontamination activities in the building. As shown on Slide 2, thirty-three days later, after decontamination, we had the iodine level down to 2×10^{-9} µc/cc (some of it by natural decay but a lot of it by the decontamination efforts). The smears were down to 350,000 from 15,000,000 DPM/100cm^2, the radiation levels in general were down from 1 R/hr 10 to 12 mR/hr by the decon efforts which we used. On the 305 level, the iodine dropped from 2×10^{-7} to 1.8×10^{-9} µc/cc; the smears dropped from 700,000 to 200,000 DPM/100 cm^2. The radiation levels dropped from 80 mR/hr to 10 mR/hr. On the 328 level, the iodine was down to 3.3×10^{-9} DPM/100 cm^2, the radiation levels down to 2 mR/hr from 10 mR/hr. This is after thirty-three days of decon which showed that by our decon efforts, we had reduced the levels on the 281 level of the Aux and the Fuel Handling Building by a factor of 1,000 for the iodine, and a factor of 100 for the smears and by a factor of 100 for the rad levels. You can see the reduction factors on the 305 and 328.

If you allow a natural decay of iodine 131 and an 8.02 day half life, this would have resulted in 7.5 x 10^{-9}µc/cc during this period. Without decon, the iodine would have taken additional 45 days to reach 2 x 10^{-9}µc/cc which we reached in the 30 day decon operation.

Slide 3 shows the building status as of July 1, after 60 days of decontamination. We now had the levels on the 281 level down by a reduction factor of 30 for the DPM on the 281 and 2 for the rad level. The reduction factor on the 305 level was 25 for the DPM, and 3 for radiation, and on the 328 level the reduction factor was 20 for the DPM and 10 for the rad level. During this period of time, the major decon effort shifted from the Aux Building to the elevator pit; we wanted to get the elevator back in service so that we could haul materials from the 280 level on up to the 328 level. We wanted to get the model room completely decontaminated so we would have a working and staging area for our recovery efforts. The rad waste panel on the 281 level was an area that the operators had to get into hourly. Having to fully suit-up with Anti-C's was slowing down their operation drastically so we put a full effort on getting the rad waste panel back into a street clothes area.

Slide 4 shows the building status on August 1st. The major effort during this period was on the building floors and the major equipment in the buildings. On the 281 level, the radiation was down to 4mR/hr; the smears at this time were showing 5,000 DPM, the iodine concentration was 3mR/hr, the smears were down to 2,000 DPM and the iodine was down to 9 x 10^{-12}.

On the 328, the iodine was MDA, the smears were giving us 2,000 DPM and the radiation level was 1 mR/hr. There was a very extensive decontamination effort that went on all of these levels in the accessible areas during this first 90 days of recovery.

During July, the reduction factor for radiation was 0 on all levels. The DPM reduced by 2 on the 281 level, 4 on the 305 level and 9 on the 328 level.

The status as of September 1st is shown in Slide 5. We had the building down to where all non-essential items were removed from the building; that is, any equipment that was in there in the form of staging, etc. that was brought in for the recovery program. We then commenced our overhead decon work. Overhead decon work included sending decontamination teams up into the overhead areas in the cable trays, wiping pipes, equipment and lights, and anything that is off the floor in the building at the different elevations.

Slide 6 shows that we started with a radiation level greater than 1,000 mr/hr on the 281 level back in April. By May, we were down to approximately 50 mr/hr, on the 305 level we were down to approximately 10 mr/hr and on the 328 level we were down to about 5 mr/hr. Today the levels in the accessible areas in the Aux and the Fuel Handling Building are down to less than 1 mr/hr in all areas which are accessible.

Slide 7 shows you a curve of the results of our swipe surveys. As you notice, when we started this back in April, we were at 15 x 10^{-6}DPM/100cm^2 on the 281. The 305 level at the time was running about 900,000 DPM and the 328 level was running approximately 300,000 DPM.

By the decontamination efforts, we brought down the levels to in the middle of June as the curves show; we had made an exerted decontamination effort to knock down the levels. In the middle of June, we had some back-up of our floor drains from our Aux sump. As you see, there is a spike which appears in early July. The back up in the drains caused some contamination, not to a great extent that it couldn't be decontaminated, but it did slow down the decontamination efforts in these areas. It backed up through the drains in the 281 and the 305 levels. We now have the Aux Building levels all down less than a 1,000 DPM in all of the accessible areas.

A typical decontamination process was done on the evaporator condensate test tanks. The results are shown in Slide 8. We went in to do the initial decon on this area with four men. The initial survey is shown on Slide 9. For the initial pass of decontamination, we used Radiac wash and had scrubbed the floors in the area. We wet vacuumed the area and the total dose which was accumulated during this initial entry to decontaminate this area was 1,200 mr. The survey results after initial decon are shown on Slide 10 which indicates that the levels had significantly decreased. We went in and did our second decon using four men, this time on respirators vice air packs. We did a single pass scrubbing the area with Radiac wash and wet vacuum. The total dose expended on this second pass for the four individuals was 160 mR. Our third pass again was done in respirators with four men. At this time, if you notice on Slide 11 that the floor drain strainer shows no dose rate. We removed this floor drain to get rid of this source of radiation. On the third pass, the four men were in respirators and again it was had scribbed with Radiac wash; the total dose expended at this time was 50 mR. The levels are shown on Slide 12. On the fourth and last pass the 4 men used respirators again. We Maslin wiped all the equipment in the evaporator condensate test tank room and again hand scrubbed with Radiac wash. This time the total exposure dose taken by the individuals was 40 m/R. The final levels are shown on Slide 13.

Slide 14 shows that after 33 days of deconing, we reduced the iodine on the 281 level 131 to $2x10^{-9}\mu c/cc$. The natural decay to reach this level would have taken an additional 45 days, if we would have let it decay off naturally. We used charcoal canisters during the first 33 days of deconing the area. Using approximately 300 charcoal canisters per day, the cost of the canister being $12.50 each, runs $3,750 a day for canisters alone for the decon teams entering. Used for 45 days beyond this 33 day decon effort would have expended an additional $168,750 in charcoal canisters. By going in and deconing this area, getting the iodine down rather than letting it decay off naturally, we were able to go on particulate canisters and through this effort we had a saving of approximately $95,000 by going in and physically doing the deconing and getting off the charcoal canisters and on the particulate to canisters.

Slide 15 shows the iodine levels over the decontamination period. It also shows the natural decay of iodine. We were using supplied air up until we got the levels down to $10^{-7}\mu c/cc$; we then went on charcoal canisters back in July and have been on respirators ever since. As of November 4th we have the levels down to where we don't need respirators anymore, except in areas and cubicles where we're working and there's a possibility of ingesting.

The Auxiliary and Fuel Handling exposure for the decon shown on Slide 16 takes us from April 27th through November 20th. From April 27th to June 30th, you can see that our decontamination supervision expended 13,424 man rem; thirty seven people were invloved in the operation. It averages out to be 362 mR per man. July 1st to September 30th, we expended 6,985 MR there were 28 supervisory decon people involved which is an average of 249 October and November we've used 5,658 mR; a total of 15 decon supervisory type people for an average exposure of 377 mR which gives us a total 26,067 mR expended by our decon supervision within the Aux and the Fuel Handling Building. In the early stages we were using Catalytic Construction Company to do the decon work for us and they picked up an average of 192 mR from April 27th to May 17th.

May 17th is when we entered into our contract to have the Aux and the Fuel Handling Building done by sub-contractors. Met-Ed, the personnel doing the decon work, the hands on decon work in the building, are all Met-Ed volunteer type people. The people have had no experience in working around radiation and working in a nuclear plant. When we brought them in, they were linemen, meter readers, etc.; they came from all of our different areas. We gave them a very extensive oral indoctrination and a hands on indoctrination using the equipment prior to sending them into an area which is contaminated. This has paid off extremely well in exposure to personnel; we've used 170 volunteers from our GPU companies and we have instilled in the people that the work can be accomplished and can be accomplished safely and we've got about a 98% return coming back in to do the down stream decontamination work. On the Met-Ed personnel, these are the people doing the wiping, scrubbing and moving the trash and so on. May 30th to June 30th we had 9,919 mR exposure this was among 118 personnel; the average exposure per individual was 84 mR per quarter. July 1st to September 30th the average exposure was 79 mR per quarter; October 1st to November 20th we've expended approximately 63 mR during this quarter. The total exposure for all personnel involved in the decontamination has been 63,187 mR.

Question: Is that mR or Man Rem?
Answer: That is mR.

We have not had an over exposure of any of the personnel performing the decontamination work in the Auxiliary and Fuel Handling Building. This is-there's not been any over exposure by any of the individuals who have been engaged in the decontamination work.

The net results of our decontamination efforts up to date as of November 20th is the iodine decreased from 10 to the minus six to 10 to the minus twelve, DPM went from 10 to the 7th to the 3rd and the dose rate which was greater than one R/hr is now decreased down to one mR.

The efforts continuing on the decontamination, we're presently establishing a program to go into cubicles which we have had isolated due to not wanting to go into them due to radiation exposure which we would have to take. We are setting up a program for getting into these areas, into these cubicles and high rad areas. We expect to complete our decon efforts in the Aux and the Fuel Handling Building sometime at the end of next year. Do you have any questions?

Question: How much surface area in the decon is there?
Answer: Corridors, passageways, lay-down areas.

Question: Do you have any idea of square feet?
Answer: I'd say 60 to 70% of the surface area.

Question: Did you use any incentive pay to obtain volunteers?
Answer: Our volunteers are all paid the standard scale of wages plus per diem.

Question: You mentioned that when the iodine went down to ... to the minus 7th you went to charcoal canisters. How did you determine the expectant life of the canister and what percent rise did you see in body burden the iodine on the people?
Answer: We only used it one time.

Question: Just one time?
Answer: One time, right.

Question: Did you figure MPT hours?
Answer: That was all figures out by the HP people who allowed us to go in.

Question: Did you monitor the thyroid for iodine during the period of time you were using the canisters?
Answer: Did we monitor them? We did whole body counts

Question: Did you do thyroid scans?
Answer: Yes, we did the entire body.

Question: Do you have any idea of how many gallons of Radiac wash you've used so far?
Answer: Yes. We have stored in the building right now approximately 1,600 gallons of Radiac wash. This is diluted Radiac wash which we have used in scrubbing down the floors and so on. We will solidfy that Radiac wash; we plan on starting solidification of it some time next month, but there's a total of about 1,600 gallons right now of water which we have used to accomplish this decon effort.

Question: .....activity on it?
Answer: Well, it has been decaying off, I can't tell you exactly what it is. It's maybe-the hottest stuff we've got might be 2 R, it's not that bad.

Question: You reported personnel exposure, are those are whole body gamma doses?
Answer: Yes

Question: Any larger percentage of the limit on the beta dose numbers?
Answer: No I don't have that number right now, no.

Question: Have you developed any square foot costs as to what these approaches are costing?
Answer: For decontamination? No I haven't.

Question: What would you do differently if you ever have to do this type of thing again?
Answer: I don't think I'd make any different approach whatsoever. I would go in the same way on a very controlled mode of decontamination. I would decon the same areas that we've deconed to date for accessibility and do it all on the methodical way which we have done up to this point. I don't think I would change it whatsoever.

Question: Did you engineer any systems for deconing with the goal of reducing the exposure, that is, long-handled tools, etc.
Answer: Oh yes. We've used long-handled tools, we used portable lead shields that we fabricated and wheeled them around in the building. We've mainly designed long-handled tools to stay away from the high rad level areas until we knocked them down.

Question: Wouldn't you expect the extremity exposures data to be very important to you as opposed to the overlying data you have there?
Answer: Oh yes, they're important to us.

Question: Why did you choose iodine as a criteria instead of other isotopes?
Answer : Well what we've got is we've got the iodine - I used the iodine here on my charts because we knocked down the iodine, we knew the decay on the iodine and we knew that it would have decayed over a certain period of time. What I was trying to show here was the deconing efforts which we put into this decon program. Got the iodine level way down before it would have taken for it to decay off naturally.

Question: Did you coat the surfaces after you got them down to a reasonable level?
Answer: We've done no coating whatsoever, except on some bare concrete areas which weren't epoxy coated. We have used a strippable coating like in the elevator pit, the elevator pit which had no coating whatsoever, it was just raw concrete, we went in and coated the elevator pit. We have done some coating with strippable coating in the diesel generator building. We've done some coating in the annulus area between the containment and Fuel Handling Building where again it was bare concrete. We have coated our portable lead shields which we have fabricated for ease of deconing; we've coated those with a strippable coating. That is the only coating that we have done to date.

Question: To what extent are the plans to document this story together with the basis for selecting your strategies?
Answer: We'll knock radiation levels down in certain areas and they will come back up again because of the construction; some construction work that is going on, some cross contamination problems. As far as moving around through the ventilation systems? We've picked up some of it in some other areas that we've already deconed, yes.

Question: Were the decon workers given iodine before they went in?
Answer: Were they given iodine? No.

Question: Did you do any coreing of any non-coated surfaces to find out whether or not you had any.....
Answer: We haven't as yet. We have scheduled in the very near future to go into the elevator pit which is an uncoated unsealed area and take core bore samples out of it.

Question: If the costs are not related to a square foot basis, are the costs related to the efficiency of the decon?
Answer: To efficiency of decon? Yes they are.

Question: So you would have, it would cost so much to occur such and such removal, and do you have that data?
Answer: Yes and I don't have that data with me, but it is available.

Question: And then when you have to go into a particular area, the method that you use to go into a particular area when you have to make that selection, is the priority of the criteria on the reduction of the personnel exposure or the effeciency of the method?
Answer: The personnel exposure is always the prime consideration to us.

Question: Do you use dollar value on man rem.....
Answer: Have I used dollar value? No I haven't.

SLIDE 1

April 27, 1979 - Building Status When Commence Decon

281:	Iodine 131 μc/cc	5.4×10^{-6}
	Smear DPM/100 cm^2	15×10^{6}
	Radiation Level	1R/hr.
305:	Iodine 131 μc/cc	2×10^{-7}
	Smear DPM/100 cm^2	700K
	Radiation Level	80mr/hr.
328:	Iodine 131 μc/cc	1×10^{-7}
	Smear DPM/100 cm^2	400K
	Radiation Level	10mr/hr.

SLIDE 2

June 1, 1979 - Building Status (33 days Decon)

281:	Iodine 131 μc/cc	2×10^{-9}
	Smear DPM/100 cm^2	350K
	Radiation Level	12mr/hr.
305:	Iodine 131 μc/cc	1.8×10^{-9}
	Smear DPM/100 cm^2	200K
	Radiation Level	10mr/hr.
328:	Iodine 131 μc/cc	3.3×10^{-9}
	Smear DPM/100 cm^2	150K
	Radiation Level	2mr/hr.
281:	Reduction Factor	1000 for I^{131}
		100 for DPM
		100 for Rad Level
305:	Reduction Factor	100 for I^{131}
		3 for DPM
		8 for Rad Level
328:	Reduction Factor	100 for I^{131}
		2.5 for DPM
		10 for Rad Level

NOTE: Allowing Natural decay of I^{131} with 8.02 day half life would have resulted in 7.5×10^{-7} μc/cc at this period. Without Decon I^{131} would have taken an additional 45 days to reach 2×10^{-9} μc/cc.

SLIDE 3

July 1, 1979 - Building Status (60 days)

281:	Iodine 131 μc/cc	2×10^{-10}
	Smear DPM/100 cm^2	11K
	Radiation Level	5mr/hr.
305:	Iodine 131 μc/cc	1.4×10^{-10}
	Smear DPM/100 cm^2	8K
	Radiation Level	4mr/hr.
328:	Iodine 131 μc/cc	8×10^{-11}
	Smear DPM/100 cm^2	18K
	Radiation Level	1mr/hr.
281:	June Reduction Factor	30 for DPM
		2 for Rad Level
305:	June Reduction Factor	25 for DPM
		3 for Rad Level
328:	June Reduction Factor	20 for DPM
		10 for Rad Level

Major Decon effort shifted to Aux Bldg. Elevator Pit, Model Room, Trash Removal and Radwaste Tent Construction.
NOTE: I^{131} Natural Decay

SLIDE 4

August 1, 1979 - Building Status (90 days)

281:	Iodine 131 μc/cc	2 x 10^{-11}
	Smear DPM/100 cm^2	5K
	Radiation Level	5mr/hr.
305:	Iodine 131 μc/cc	9 x 10^{-12}
	Smear DPM 100 cm^2	2K
	Radiation Level	3mr/hr.
328:	Iodine 131 μc/cc	MDA
	Smear DPM 100 cm^2	2K
	Radiation Level	1mr/hr.
281:	July Reduction Factor	2 for DPM
		0 for Rad Level
305:	July Reduction Factor	4 for DPM
		2 for Rad Level
328:	July Reduction Factor	9 for DPM
		0 for Rad Level

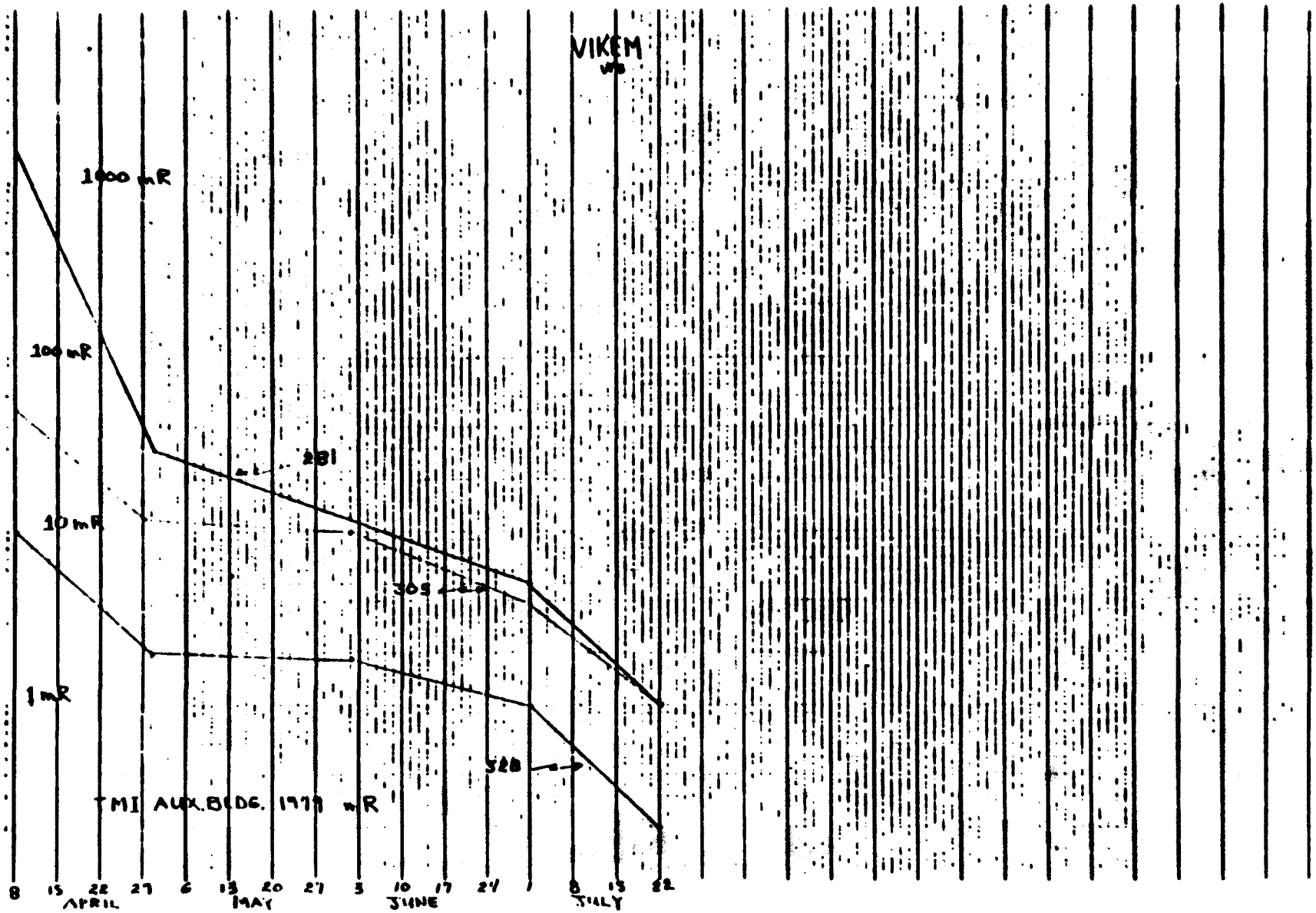
Major Decon effort on building floors and major equipment.

SLIDE 5

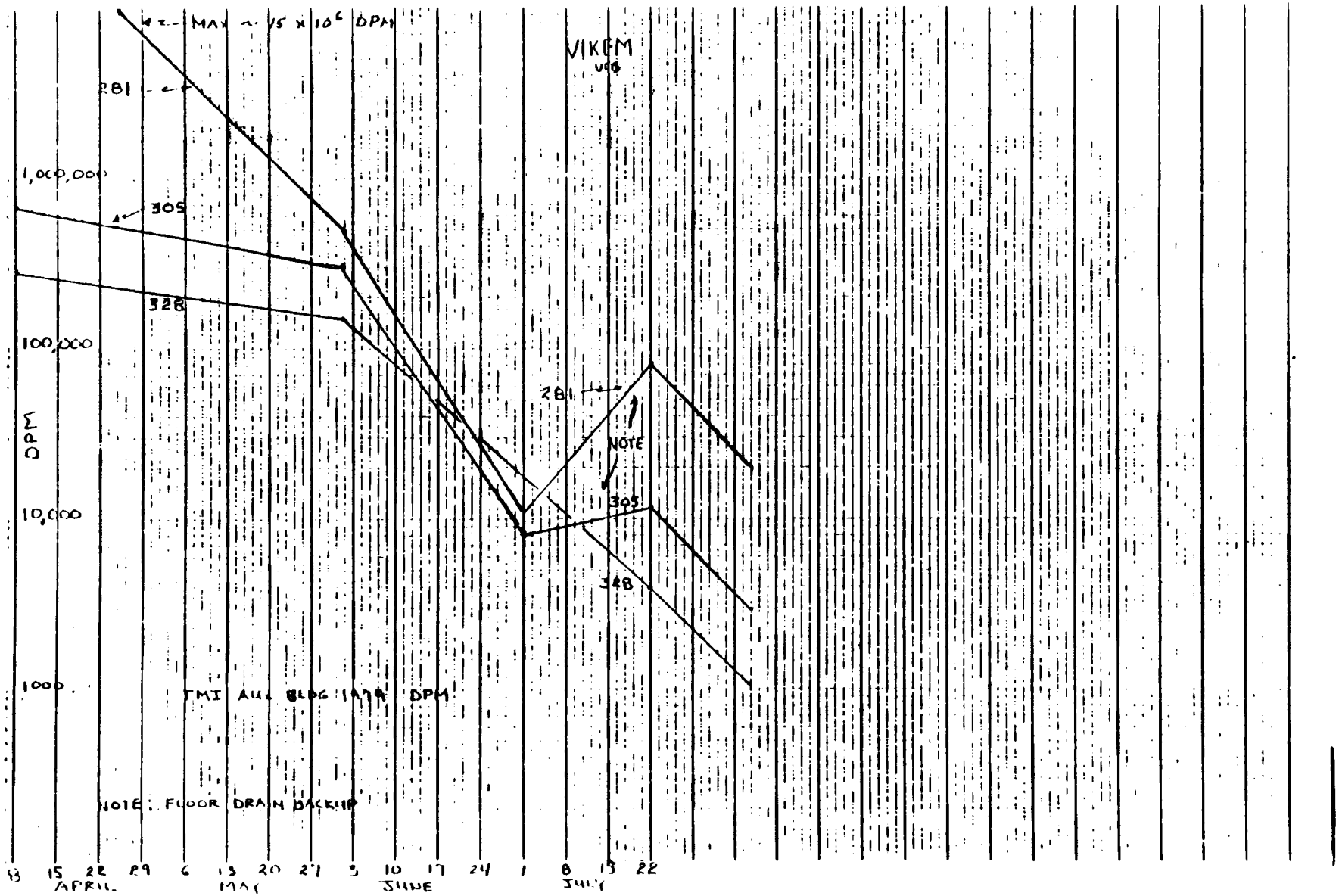
September 1, 1979 - Building Status (120 days)

281:	Iodine 131 μc/cc	2×10^{-11}
	Smear DPM/100 cm^2	2K
	Radiation Level	3mr/hr.
305:	Iodine 131 μc/cc	2×10^{-11}
	Smear DPM/100 cm^2	1K
	Radiation Level	1mr/hr.
328:	Iodine 131 μc/cc	2×10^{-11}
	Smear DPM/100 cm^2	1K
	Radiation Level	$<$ 1mr/hr.
281:	August Reduction Factor	2 for DPM
		2 for Rad Level
305:	August Reduction Factor	2 for DPM
		3 for Rad Level
328:	August Reduction Factor	2 for DPM
		2 for Rad Level

All non-essential items were removed from building.
Commenced overhead Decon.


VIKEM
1000 mR
100 mR
10 mR
1 mR
281
305
328
TMI AUX.BLDG. 1979 mR
8 15 22 27 6 13 20 27 3 10 17 24 8 15 22
APRIL
MAY
JUNE
JULY

SLIDE 6


MAX ~ 15 x 10⁶ DPM
VIKEM
281
1,000,000
305
328
100,000
DPM
281
NOTE
10,000
305
328
1000
TMI AUX BLDG 1A7A DPM
NOTE: FLOOR DRAIN BACKUP
8 15 22 29 6 13 20 27 3 10 17 24 1 8 15 22
APRIL
MAY
JUNE
JULY

SLIDE 7

SLIDE 8

EVAPORATOR CONDENSATE TEST TANK

Initial Decon - 4 Men

Scott Air Pak's

1 Pass

Radiac Wash - Hand Scrub

Wet Vacuum

Total Dose 1200mr

Second Decon - 4 Men (Respirators)

1 Pass

Radiac Wash - Hand Scrub

Wet Vacuum

Total Dose 160mr

Third Decon - 4 Men (Respirators)

1 Pass (Strainer Removed)

Radiac Wash - Hand Scrub

Total Dose 50mr

Fourth Decon - 4 Men (Respirators)

1 Pass

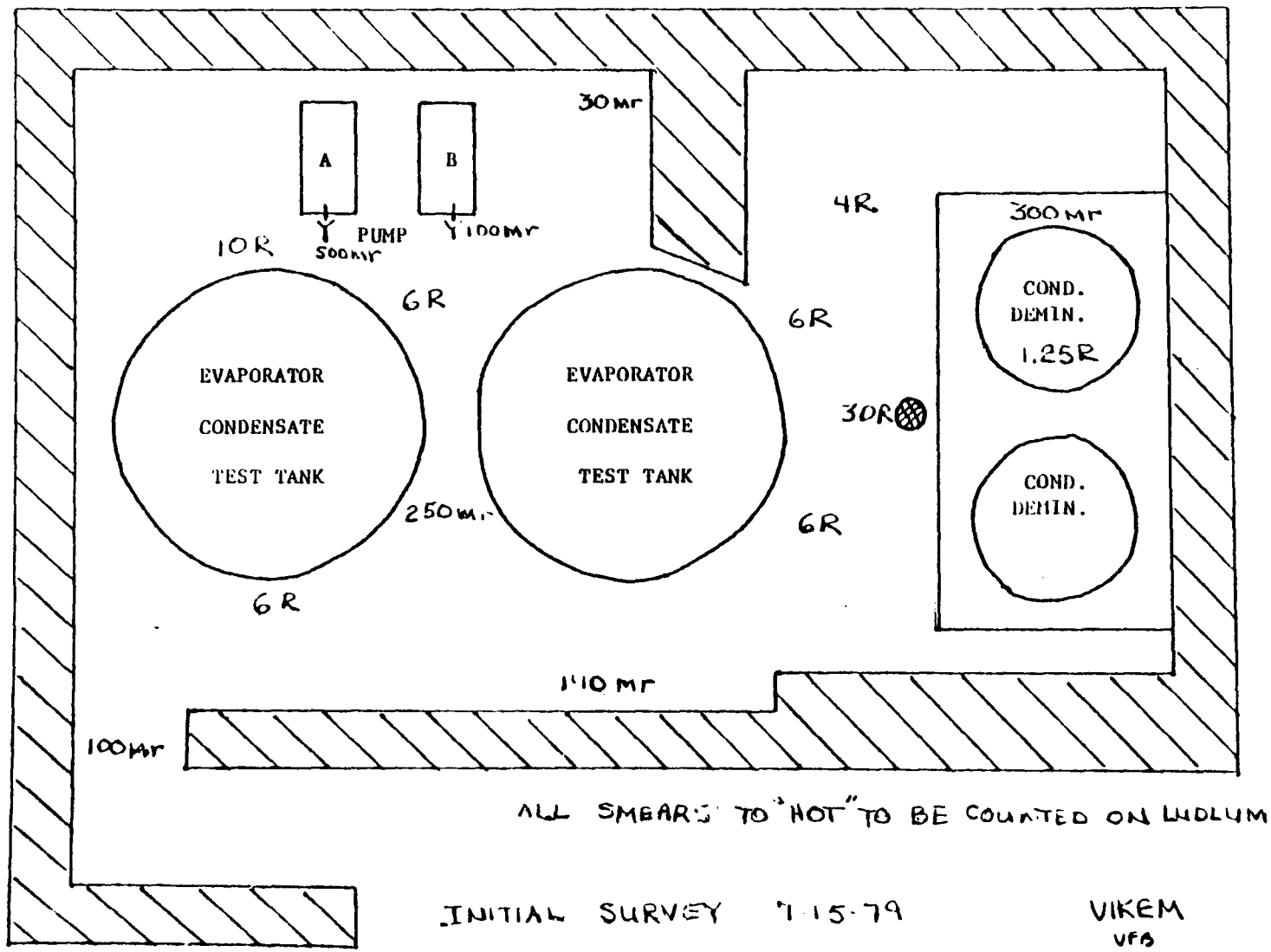
Masslin Equipment

Radiac Wash - Hand Scrub

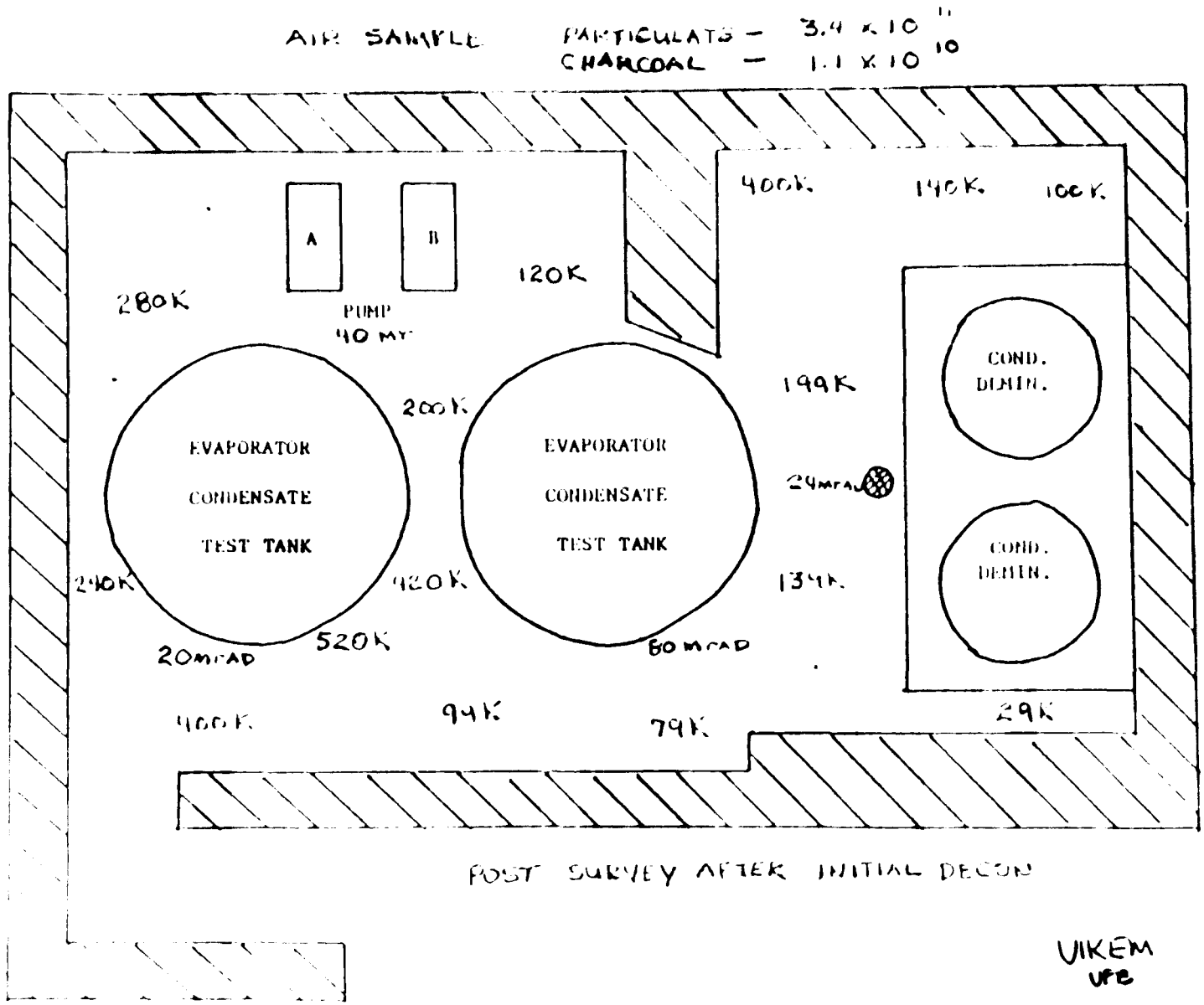
Total Dose 40mr

Total Dose - 1450mr

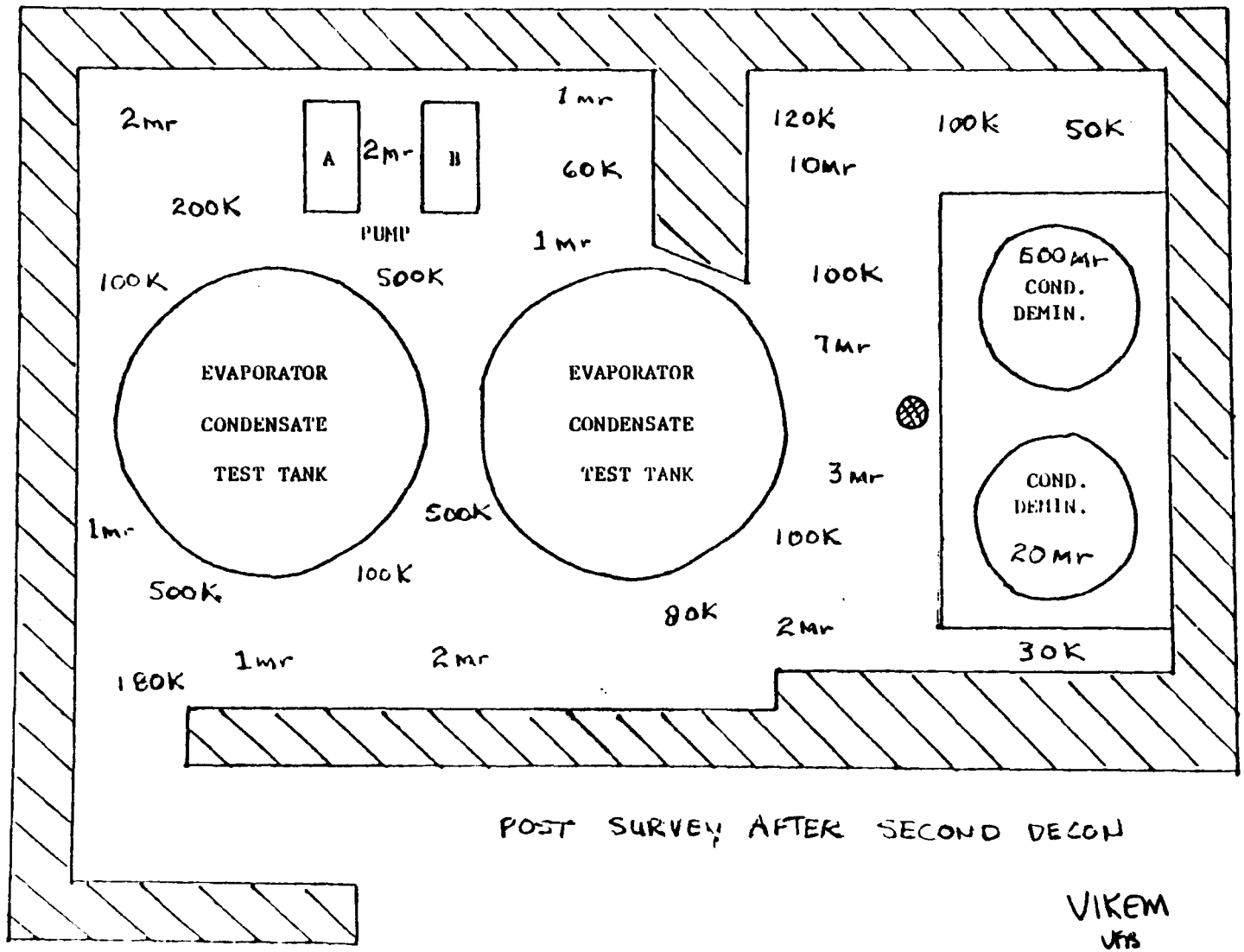
AIR SAMPLE PARTICULATE — 2×10^{-8}
CHARCOAL — 6×10^{-9}



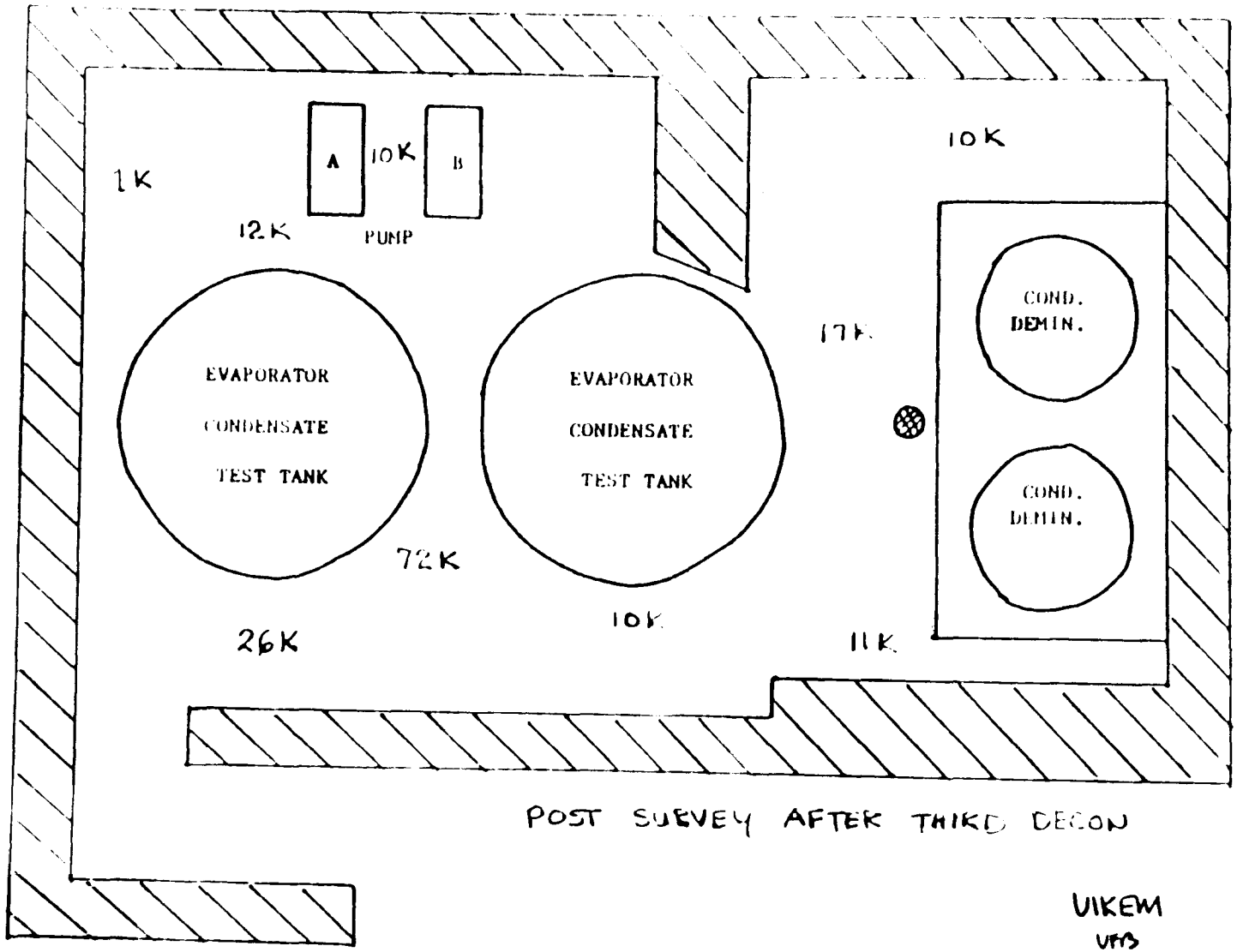
SLIDE 9


AIR SAMPLE
PARTICULATE – 3.4 x 10 11
CHARCOAL – 1.1 x 10 10
A
B
PUMP
40 MY
280K
120K
400K.
140K.
100K.
EVAPORATOR
CONDENSATE
TEST TANK
200K
240K
420K
520K
20MRAD
EVAPORATOR
CONDENSATE
TEST TANK
60 MRAD
199K
24MRAD
134K.
COND.
DEMIN.
COND.
DEMIN.
400K
94K
79K
29K
POST SURVEY AFTER INITIAL DECON
VIKEM
UFE

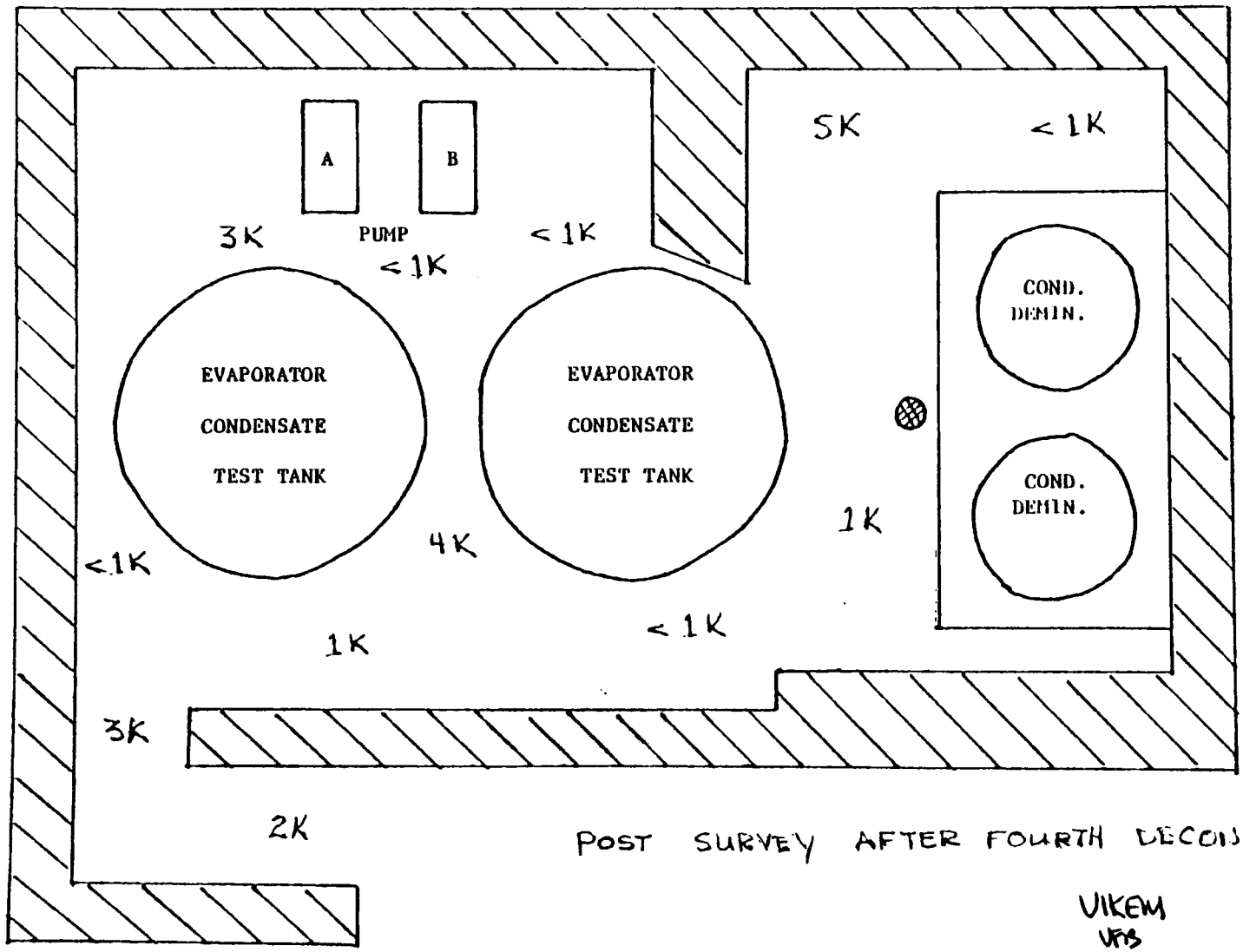
SLIDE 10


2mr
A
2mr
B
1mr
60K
120K
100K
50K
10Mr
200K
PUMP
1mr
100K
500K
100K
7mr
EVAPORATOR
CONDENSATE
TEST TANK
EVAPORATOR
CONDENSATE
TEST TANK
500Mr
COND.
DEMIN.
COND.
DEMIN.
20Mr
3mr
1mr
500K
100K
500K
100K
80K
2mr
30K
180K
1mr
2mr
POST SURVEY AFTER SECOND DECON
VIKEM

SLIDE 11


A
10K
B
1K
12K
PUMP
10K
EVAPORATOR
CONDENSATE
TEST TANK
EVAPORATOR
CONDENSATE
TEST TANK
17K
COND.
DEMIN.
COND.
DEMIN.
72K
26K
10K
11K
POST SURVEY AFTER THIRD DECON
UIKEM
UFS

SLIDE 12


A
B
PUMP
3K
<1K
<1K
5K
<1K
EVAPORATOR
CONDENSATE
TEST TANK
EVAPORATOR
CONDENSATE
TEST TANK
COND.
DEMIN.
COND.
DEMIN.
<1K
4K
1K
1K
<1K
3K
2K
POST SURVEY AFTER FOURTH DECON
VIKEM
VBS

SLIDE 13

SLIDE 14

VIKEM DECON AND IODINE^{131} VS. COST

33 Days Decon Reduced I^{131} to 2 x 10^{-9} c/cc.

Natural decay to reach this level would have taken an additional 45 days.

Charcoal Canisters

Use 300/day

Cost $12.50 each - $3750/day

45 day use - $168,750

Shipping

First 15 days 50 cannister/drum

6 drums x 15 days - 80 drums

80 drums x $67/drum - $4020

168,750
5,360
4,020
$178,130

Particulate

Use 300/day

Cost $6.00 each - $1800/day

45 day use - $81,000

Shipping

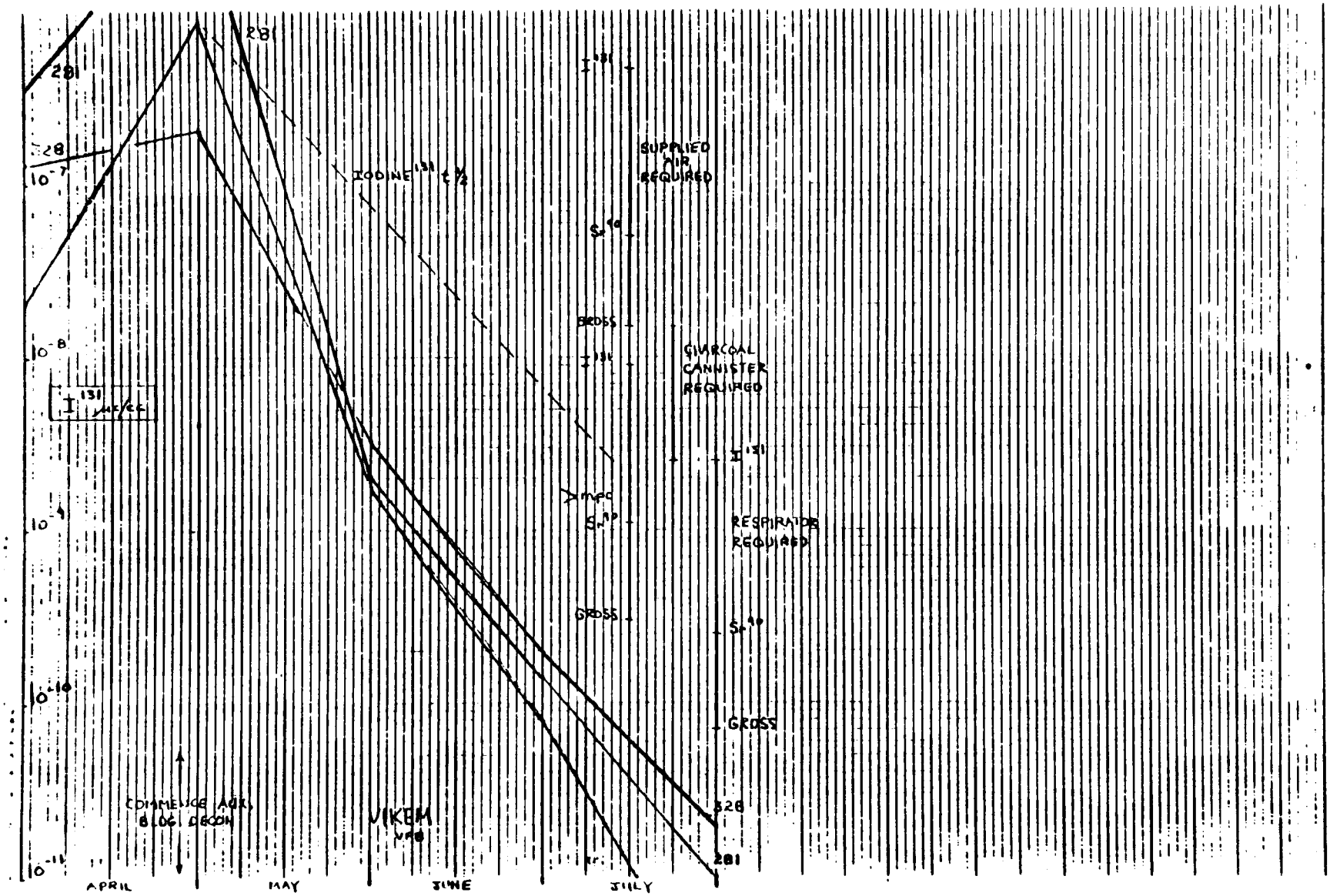
300 canisters/drum - 1 drum/day

45 drums x $67 - $3015

81,000
1,800
$82,800

178,130
82,800
$95,330 Savings

Packaging does not include labor costs


28l
28
10^{-7}
10^{-8}
10^{-9}
10^{-10}
10^{-11}
IODINE 131 t ½
I^{131} μc/cc
SUPPLIED AIR REQUIRED
S_M^{40}
GROSS
I^{131}
CHARCOAL CANNISTER REQUIRED
I^{131}
MPC
S_M^{89}
RESPIRATOR REQUIRED
GROSS
S_M^{40}
GROSS
328
281
COMMENCE ADDL BLDG DECON
VIKEM
APRIL
MAY
JUNE
JULY

SLIDE 15

SLIDE 16

NUKEM STAFF

AUX & FHB EXPOSURE FOR DECON

NUKEM	EXP. (MR)	PERSONNEL	AVG. EXP.
April 27 - June 30	13,424	37	362 mr
July 1 - Sept. 30	6,985	28	249 mr
Oct. - Nov. 20	5,658	15	377 mr
	26,067		

CATALYTIC	EXP. (MR)	PERSONNEL	AVG. EXP.
April 27 - May 17	5,371	28	192 mr

MET-ED	EXP. (MR)	PERSONNEL	AVG. EXP.
May 30 - June 30	9,919	118	84 mr
July 1 - Sept. 30	12,982	164	79 mr
Oct. 1 - Nov. 20	8,848	139	63 mr
	31,749		

TOTAL EXPOSURE 63,187 mr NOTE: NO OVEREXPOSURES

NET RESULT

I 131	10^{-6}	Decreased to 10^{-12}
DPM	10^{7}	Decreased to < 1K
Dose Rate	> 1R	Decreased to 1mr

SESSION L

AUXILIARY BUILDING DECONTAMINATION WASTE PROCESSING EXPERIENCE

LIQUID/SOLID WASTE PROCESSING EXPERIENCE

Rick McGoey
GPU

As the result of the accident at Three Mile Island, we had large quantities of water generated in the Auxiliary and Fuel Handling Building. It was my responsibility as well as other people on the Island to manage the water in terms of storage and processing. I'm sure a lot of you are familiar with the fact that contaminated water entered the auxiliary building from the reactor building. That estimated quantity of water is greater than 50,000 gallons of water. We had five to six inches of water on the bottom of the Auxiliary Building that people had to walk through. It was part of my task to get that off the floor and do something with it. The source of water came from the Reactor Building sump pumps, over pressurization, lifting the reliefs and the make-up purification system over pressurizing tanks, putting water in our waste gas system, excessive seal leakage on pumps as well as a lot of other multiple sources. The inability of having full access to the Auxiliary and the Fuel Handling Buildings prevented us from going in and locating and isolating all the sources and having a good management of that water. The primary objective the first day into the accident was just get it off the floor so people could walk around with a little more ease and not worry about excessive contamination. So, the first thing we did was take the water that was in the Unit 2 radwaste storage tanks and pump it to Unit 1. That was essentially low activity water (pre-accident) passed to Unit 1, from there we could process it and release it. Now with a large volume of water produced at the start of the accident, we realized immediately that we were to need supplemental waste processing equipment. Therefore, the day of the accident we called in outside contractors, Capolupo & Gundal, Inc. and Epicore to provide some radwaste processing services. These people had been on the Island earlier to assist us in supplementing our radwaste systems. We had had trouble with our waste evaporator and had used a demineralization

process in the past, so with the accident with a large generation of water we called them in immediately. They arrived on the Island and within a week they were processing water through TMI Unit 1. As Tom Block pointed out in an earlier paper, we had repeated back-ups of water coming into the Auxiliary Building sump that upset the decontamination plan, caused increases in the radiation and airborne levels, etc. This was due in part to the inability of being able to put water into storage tanks, lack of access to the Auxiliary Building to monitor what was going on.

We did not have monitoring capability in the control room of monitoring tank and Auxiliary Building sump levels. We, therefore, went through extraordinary efforts as Tom also pointed out in building a tunnel to the radwaste panel in order to allow people to get to that panel to monitor the radwaste systems.

My talk is going to centralize on the liquid radwaste characteristics, the Epicore 1 radwaste system, the Epicore 2 radwaste system and touch upon a third radwaste system, a Submerged Demineralizer System.

First off, the radwaste liquid characteristics. We categorize the water into three types, low, medium and high activity based on the specific activity. Low is less than 1, medium was 1 to 100 and high was greater than 100 micro curies per milliliter. That provided a convenient means of segerating it into separate tanks to facilitate processing. The low activity water is the type of water that we sent TMI Unit 1, some was processed in the TMI 1 radwaste system, predominantly it was this water that was processed by Epicore 1. We presently have a small amount of water, 15 to 20,000 gallons. We have processed 470,000 gallons since the accident started, we process it and then release it to the river. Now don't be confused, that is not just Unit 2 water, it's Unit 1 and Unit 2 water. I'll get into a little explanation of how and why we ended up processing Unit 1 by this system.

The Epicore 1 radwaste system is simply a demineralization system. It makes use of a pre-filter and a 130 cubic foot mixed bed demineralizer. We called the contractor in the day of the accident. Within a week of the accident we had fully approved waste processing procedures, a system review and approved by the NRC and we were processing. It is not a sophisticated system, it was something that was used to allow greater control over the liquid waste problem. We continue to use this system of processing water.

The predominant isotopes in the water presently are cesium 137, cesium 134, strontium 89-90. There is also some low level cobalt 58 and cobalt 60 present. Water is still coming in at the rate of about 200 gallons a day. We're still sending it to TMI 1 through the Epicore 1 system and releasing it to the river. We generated, also as a result of the accident, another level of radwaste, "medium level," 1 to 100 micro curies per milliliter. We've realized that the Epicore 1 radwaste system was not designed to process medium level water. Therefore, early on we commenced to develop a radwaste system that was especially tailored for the processing of this water. We termed this system Epicore 2, "Epicore" by the way is the name of a contractor. This system was placed in the chemical cleaning building which is an existing structure on the Island. I am going to centralize on this system since it is operational and we have some pretty good experience on it.

Presently we have 360,000 gallons of wastewater; we've processed 48,000 gallons through the system. While we're developing a system through April and May, the city of Lancaster has obtained an injunction against the NRC to prevent us from releasing the water to the Susquehanna River. The response that the NRC took to this was to issue an order to us requiring that an environmental impact assessment be prepared, issued to the public. This would give the public a chance to comment, NRC would resolve

the comments and then give us permission to process. That environemntal impact assessment was essentially needed for three particular purposes; No. 1 - to gain permission to process intermediate level water with the Epicore 2 system, No. 2 - environment impact assessment that once we processed the water to release it to the river and No. 3 - the processing of high activity water.

While this paper was being prepared, went out for public comment, etc., our in-leakage continued at the rate that we were losing storage capacity at TMI 2. That forced us to put some water in the TMI 1 radwaste systems and it also forced us to build a tank farm of 110,000 gallons of storage capacity in our spent fuel pool. Luckily, the fuel pool was uncontaminated, never had any spent fuel in it and it was accessible for installing tankage. I'm sure you all realize that at the time of the accident we had enough tanks on the Island to take care of the water forever. Not all the tanks were well qualified, but we had ample access to tanks on the Island that were shipped in from all over the United States. So we had the tanks here; it was a matter of installing them and hooking them up to the radwaste systems. Those tanks were installed. With the in-leakage continuing, we went through exhaustive efforts to reduce the in-leakage as much as possible. Now, most of the water leaking is was coming from non-contaminated systems, river water, demineralized water, etc., going to a common sump and it took just a small amount of primary coolant leaking into the sump which is coming from the make-up and purification system to contaminate the water to the intermediate level or medium level activity. We went through great efforts to try to reduce that. However, due to the radiation levels, inaccessibility of various locations in plant, we just couldn't stop that in-leakage completely. Therefore, approximately five to six weeks ago when we were running out of storage capacity completely, we contacted the NRC

and through various efforts we gained permission to start processing via the Epicore 2 radwaste system.

That system takes water from tanks in the Auxiliary Building to a separate building through a series of three demineralizer beds, a pre-filter demineralizer, a mixed cation bed and a mixed cation-anion bed, to a clean water receiving tank from here it is sampled. If it is clean we can put it back into the existing tankage in the building for final disposition or if it's off spec, we sent it to another tank and then we can return it back through the process.

Although we gained permission to process with Epicore 2, we have not received permission to release the water. What we are now doing is to put the processed water in tanks for storage. By gaining permission to operate this system we opened up, approximately 220,000 gal worth of storage capacity. Once a demineralizer liner is expended, we have built a transfer shield. We also have a especially designed HVAC system for the building which makes use of HEPA filters and charcoal filters. It keeps the building at a negative pressure.

The Epicore 2 building is outside the confines of the Auxiliary and Fuel Handling Buildings. The chemical cleaning building which houses the Epicore 2 process was originally intended for the cleaning of our steam generators. It was never used, but the building was designed for the storage of contaminated water; it is seismically designed and has a Butler building super structure that is actually well suited for the containment of a radwaste system such that if should any leaks occur, we have confinement. We build a separate control room to permit monitoring and control of the process inside the building remotely without radiation exposure.

We make use of quick disconnects and hoses on the demineralizer beds. We have five inches of steel and three inches of lead shielding on top of the canister, around the canister we also have approximately four inches of lead and about one inch of steel. The radiation levels in the building have proved to be very low when we had 1,000 curies of radio nuclides in the first liner. The radiation levels on top of the shield were roughly 25 mR/hr and adjacent to the liner on the floor on the order of 5mr/hr. So we put a tremendous amount of shielding in this system in order to handle the high level of radio nuclides we expected to remove in this system. The system was especially designed for this purpose and it has performed pretty much as we expected and personnel exposures have been very low.

Air operated positive displacement pumps are used. We do use quick disconnects and hoses in order to allow for rapid removal of the pumps should it be necessary. The entire building, is painted with Imperial strippable coating.

We have processed 40,000 gallons through the system and you can walk into the building just as you are dressed now; there are no contamination levels, radiation levels are quite low walking around this area. The integrity of the system has been essentially leak proof; we have had minor seepage but nothing of significant concern.

We do have a monitoring capability in the control room; pH radiation level, etc., so that we can monitor the process in the building without having to go inside the building. We have seven TV monitors by which we can monitor the whole operation. Now one of the problems obviously is once we get all these curies deposited in demineralizer bed what do we do with them, how do we handle them. At present, we plan to deposit up to 1,300 curies in each demineralizer bed. Right now, we've evaluating taking that further, but we had a significant concern

for handling the beds out of this building. We designed a transfer bell. It's a cylindrical configuration such that it can be placed over a demineralizer bed and through various rigging operations, the liner can be drawn inside of the transfer bell and moved outside.

Doors on the bottom are opened and the liner demineralizer bed is brought up inside of the transfer bell and then the doors are closed. Once it is in the bell, it reduces the radiation levels significantly. When we had 1,200 curies in one demineralizer, the radiation levels along the bell were reading on the order of 35 mr/hr. This is approximately four inches of lead, it's a significant construction. When we initially went into this we thought it was going to be a mechanical nightmare of opening doors, closing doors, picking up a liner that we couldn't even see, etc. This whole operation has gone extremely well. We've had minimal problems with the bell, minimal handling of liners and minimal exposure. Once the liner's brought inside of the bell, these are doors that will open.

We have a transporter which is used strictly for the use of moving liners around the Island. It does have additional shielding. This is approximately 16 inches of concrete which the bell and the liner is put inside. This shielding is also used for holding the liner and the bell on the transporter during movement. This is a three inch lead plate that is put on top of the liner for shielding. We also have three stand pipes to which the hoses are connected.

Now one of our problems that I'm sure just about everybody here is aware of, the inability of shipping and burying radwaste. Because of the TMI 2 accident, Barnwell was closed to TMI and Met-Ed for the burial of it's radwaste. We had to ship all of

our radwaste to Washington for burial there. We realize that we're going to be generating liners at a pretty high rate and that we had to build some type of an interim staging facility for the storage of liners. This is mainly for two purposes: 1. For the availability of shipping casks, we will produce liners faster than we had shipping casks available to ship out to Washington and get it back and also because of the opening and closing of the various burial grounds. With this realization, we sought off immediately to develop what we call a waste station facility, it's an interim storage facility. It's a little bit hard to see here but this is our storage facility. It's essentially cylindrical steel cylinders surrounded by compacted dirt with a layer of concrete on top. The liner is placed inside the storage facility and a large concrete block is placed on top of it. Now, as you can see here, the transporters arrived at the station facility, the bell and the liner are pulled out together, moved over the storage location, the bell sits on top of the storage location, the doors are open and the liner's lowered into it's final storage location. The liner is removed and then the large cement block is placed on top. The cement block is roughly 3 1/2 to 4 feet thick. Now this is only an interim storage facility; it was something we needed right away. Again, we used steel cylinders with compacted dirt; that only allowed us time to build a more sophisticated facility up here which is solid concrete with some reinforcing, re-barring it, a water catching facility, a sump, a monitoring capability, etc., just for the storage of these liners. One module holds roughly 60 liners and we're in the process of building two more.

We have yet to ship any of our resin liners off the Island either from Epicore 1 or Epicore 2. Epicore 1 has generated roughly 14 liners. We anticipate shipping those liners hopefully starting sometime next week. As a result of the interface with the NRC

and the concerns of the city of Lancaster and the public, etc., about the handling of TMI waste, when we went through the environmental impact assessment on operation of Epicore 2 and we gained permission to operate Epicore 2. One of the orders that we received is that we must solidify all resin used in the Epicore 2 process. Now with that, our storage facility is extremely valuable. We have no capability on the Island for the solidification of resins. We are initiating a crash program to come up with some concept for solidification of the resins with a longer range program in development. We anticipate storing these resins from Epicore 2 until we have the solidification process in operation which could take six months to a year.

The water we process is about 7.2 uCi/ml. The clean water passing through the system is 7.3×10^{-8} uCi/ml. The rough decontamination factor is 10^{8}. During the processing of the Epicore 1 water, we found that there was a recalcitrant species of cerium and strontium which was causing some problem in the removal of those isotopes from the water. Through various efforts of resin column tests and support tests performed by Oak Ridge National Laboratory, we got a better feel for these recalcitrant species. It looks like they are on the order of .3 to .5 hundredths of a percent. It does offer a specialized problem to the processing of the water. I don't want to fool you by the term recalcitrant, essentially it's a species which is difficult to remove. What we feel is that the cesium and strontium exist in the ionic as well as the colloid stages and you have to go about the removal of those species in a different way. The Epicore 2 system as you can see has done that very well. To try to put that a little bit in perspective the 10 CFR 20 limit is 2×10^{4} pico Ci/liter. You can see the water is an order of magnitude below what is required for release. The EPA drinking water limits were below that by an order of magnitude. If we throw in dilution, this is dilution prior to

the water reaching the Susquehanna River, our 73.7 number drops down to .019. So if we release this water we are significantly lower than our tech spec limits as well as EPA drinking water limits. Now that's true for cesium and strontium. The tritium, as you all know, can't be removed by a demineralization process. This shows here that the number is dropped, that really a matter of analytical accuracy; we really aren't removing tritium, that is our predominant isotope of concern. The number 1×10^{7} is slightly higher than our 10 CFR 20 release rates but if we include dilution for that we are lower than 10 CFR 20, and we're also lower than EPA drinking water limits. So we've had great success with the operation of the system and as we go into the next phase with the NRC and the environmental impact assessment, we're hoping we're going to gain permission to release the water into the river. We've had a lot of people interested in this, we've had Maryland taking samples of water, Maryland Department of Health, I couldn't name all the various agencies. We are going through a more detailed study of things we can do with this water other than releasing it. Such as recycle, we plan to use it for recycle into the primary make-up system, possible introduction for use for make-up in the secondary system, and a lot of uses like that and that study - we're really in the middle of it and we haven't reached any conclusions, but because of it's cleanliness we are hoping to gain permission for release.

Brian, can you go back to the second slide. That pretty much covers the Epicore 2 process it's the intermediate level water. I just want to touch upon the high activity water. Again, this is water that is greater in concentration of 100 micro curie per milliliter; we have roughly 530,000 gallons of that, that is the water that is in the reactor building floor, it's also in the primary coolant system. We have not processed any; we are intending to use a submerged demineralizer system. It is going

through final engineering stages; we have started a small amount of construction. What it is essentially comprised of is inorganic beds of zeolites, three inorganic zeolite beds going to a large cation bed and going to a final mixed cation anion resin bed. That work is being performed by various agencies, Chem Nuclear, Oak Ridge National Laboratories, Savannah River; a lot of different organizations are assisting us in the clean up of that water. I can give you a little more details if necessary but we really aren't at the stage of finalizing exactly how we're going to process that water. Again, I included the cesium levels; these are essentially the same numbers Jack Daniels gave you earlier in the day. I only put them up there for comparison purposes. That's about it, are there any questions.

Question:
That inventory of high activity water is that what is sitting inside the containment building?
Answer:
Seven feet worth, right.

Question:
How much solid waste has come out of the Epicore 2 system....?
Answer:
Epicore 2. We have produced three - the first demineralizer beds.... three of these, two of these and one of these. Essentially six beds. The first two beds are 35 cubic feet each, the last bed is 120 cubic feet.

Question:
(Illegible)

Answer:

Yes. As I said before, early in the accident with realization of the generation of all this water of higher activity than you normally have in the power plant, we looked at all the alternatives that were available to the industry. We immediately brought in the Epicore 2 system because it was quick and easy, that was a demineralization system and it did the job very well. For the intermediate level waste we went with the Epicore 2 system essentially second generation system, we had a building already there that could be used. For the high activity water we looked at may alternatives; we looked at demineralization, we looked at evaporation, we looked at volume reduction, many options. As a result of that effort we proceeded actually with two paths and I haven't touched upon one of them. One path is the submerged demineralizer system making use of demineralizer beds. The second path is the development of evaporation. The evaporation option was initially started out to be the back up to the submerged demineralizer system and now it's evolved into a system for the processing of high solids waste such as the decon waste. That option is still being pursued for that purpose. One of our problems at TMI 2 is that TMI 2 relied upon TMI 1's miscellaneous waste evaporator. TMI 2 does not have a miscellaneous waste evaporator. It is important for us to segregate Unit 2 so the accident does not impact Unit 1 operations. Therefore, we needed that evaporator for two reasons, for the high solids content that could not be used and removed very efficiently in the demineralization process and also to supplement the existing radwaste system. So we are proceeding with an alternative evaporation.

Question:

Have you run into any problems licensing or otherwise in increasing your on site storage or radwaste?

Answer:
No. Well, realize it's only staging not storage. It's only an interim staging until we gain a shipping cask. It's not a storage facility, that's a very important point, it's just staging until we ship it.

Question:
Do you have any scheduled target dates for this activity, this operation?
Answer:
Well the Epicore 2 system is not operational. The question is what is our time frame for this water processing. We expect to have all the medium level waste processed probably by February or March and that's pretty rough. The submerged demineralizer system, that is the system that we are putting in to remove the water from the reactor building and you heard all of the importance of doing that, get people in the building. We are hoping that that will be operational in the fall of next year.

Question:
And how long do you think that operation will take?
Answer:
I'll give it two to three months, maybe four months. A lot of it depends upon the final disposition; if we have to store the water, it's a matter of just the water management, where do you put it once you clean it. Do you put it here, do you put it there, and what do you do with it then. We're hoping to clean up this water then use it for additional decon efforts for example in the reactor building. But if we can release it that offers us greater flexibility in processing and in water management and hopefully that should facilitate moving water around and ending up with a higher processing rate.

Question:

How do you keep your transfer bell and your staging vaults clean when you are handling wet resins, etc. Don't you ever run into a problem with cross contamination?

Answer:

Okay, I guess that's a fair question. The system in the chem cleaning building and the quick disconnects we've selected, the way we make, break, the hydro testing, the quality control, all that was designed in the system to insure that we don't contaminate the outside of the liners. We de-water the resins; we have gone through an extensive de-watering testing process by which we insure that we remove all the water from the liners prior to removing it out of the building. So hopefully if all goes well we won't have any leaks and we won't contaminate the outside of the liners. Now let's say it does for some reason, and I think that is your question, we could contaminate the inside of the bell, the bell is fully coated with the paint easy for decon. Moving that down to the storage facility, the storage facility is cleaned before we put anything in it. If we do put it in it, let's say we put a contaminated liner in it, there's no problem with that. You know, necessarily you can contaminate the storage facility but it's not going to affect anybody or anything. If we move the liner out, we'll go in and decontaminate that storage facility, or staging facility, thank you.

We're running a little behing schedule, I'm going to try to pick it up. Our next presenter is Bud Arrowsmith. He's going to be talking on the demonstration of the alternative decontamination techniques that heoped to be used at TMI. We got the slides back there so we'll get Bud going here.

THREE MILE ISLAND UNIT II

LIQUID RADWASTE CHARACTERISTICS

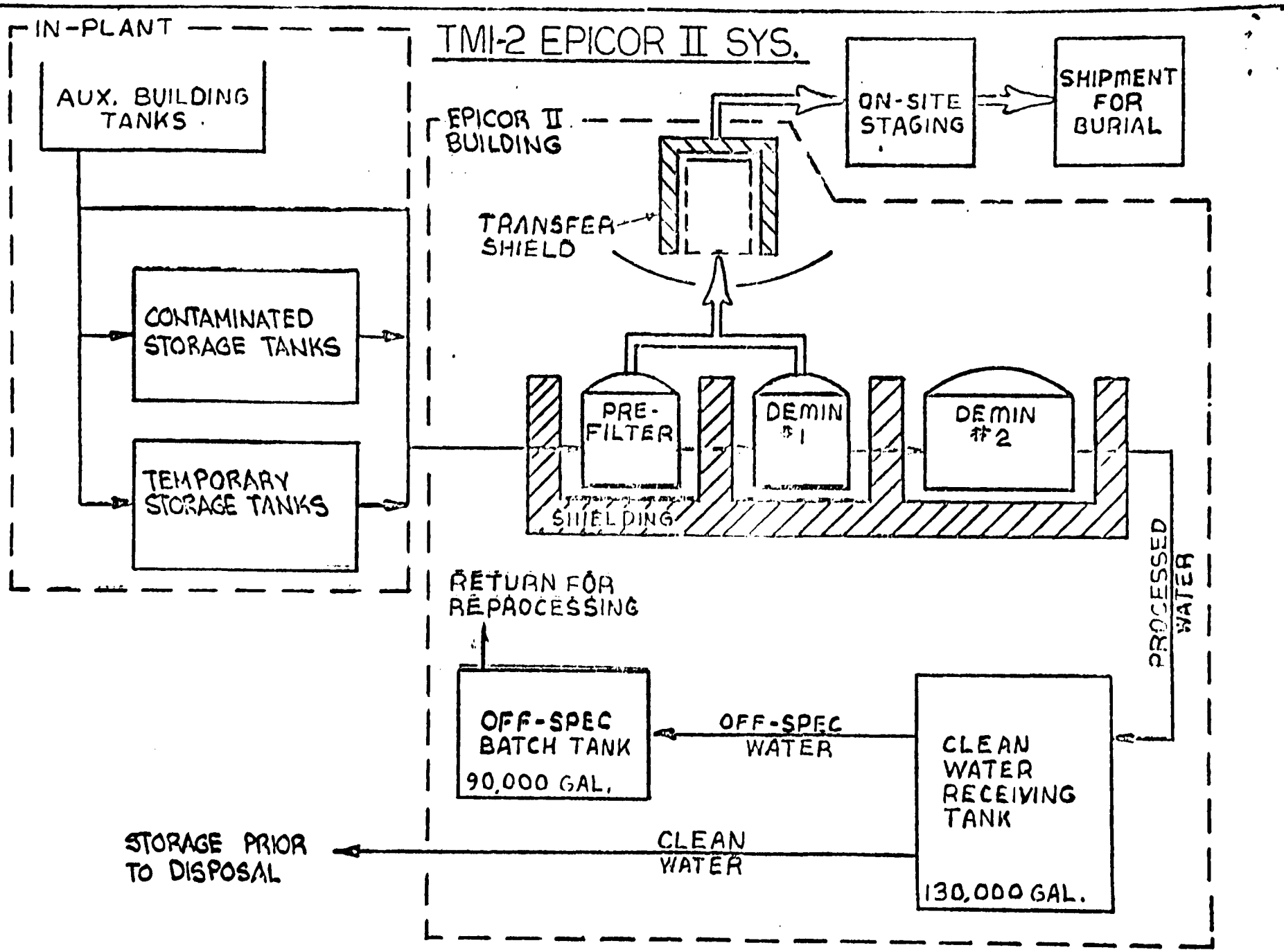
AS OF NOVEMBER 27, 1979

ITEM	LOW	MEDIUM	HIGH **
CRITERIA (μc/ml)	< 1.0	1.0 TO 100	> 100
INVENTORY (GALLONS)	15,000*	360,000	530,000
PROCESSED (GALLONS)	470,000*	48,000	NONE
PROCESSED BY (SYSTEM)	EPICOR I	EPICOR II	SUBMERGED DEMIN SYSTEM
RADIONUCLIDES (μc/ml)			
Cs 137	$1 \times E^{-2}$	7 TO 35	180
Cs 134	$2 \times E^{-3}$	2 TO 7.5	40
Sr 89	$4 \times E^{-5}$	9 TO 12	41
Sr 90	$1 \times E^{-5}$	2 TO 3.0	3.0
TRITIUM	$2 \times E^{-3}$	0.02 TO 0.5	1.0

* INCLUDES UNIT 1 AND UNIT 2 WATER

** In Reactor Building & Primary System

FIGURE 1


IN-PLANT
TMI-2 EPICOR II SYS.
AUX. BUILDING TANKS
EPICOR II BUILDING
ON-SITE STAGING
SHIPMENT FOR BURIAL
TRANSFER SHIELD
CONTAMINATED STORAGE TANKS
TEMPORARY STORAGE TANKS
PRE-FILTER
DEMIN #1
DEMIN #2
SHIELDING
PROCESSED WATER
RETURN FOR REPROCESSING
OFF-SPEC BATCH TANK
90,000 GAL.
OFF-SPEC WATER
CLEAN WATER RECEIVING TANK
130,000 GAL.
STORAGE PRIOR TO DISPOSAL
CLEAN WATER

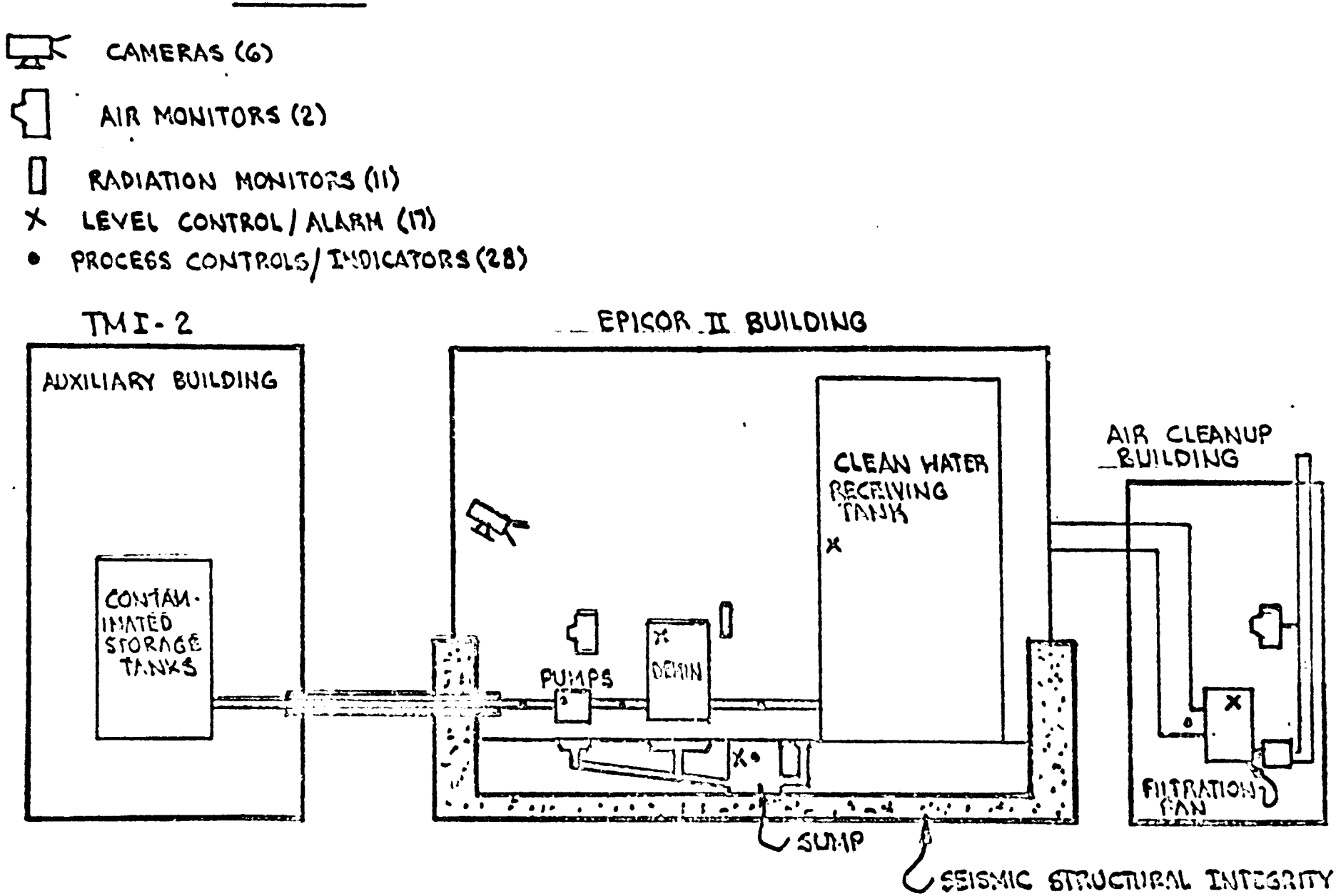


FIGURE 3

EPICOR II SAFETY PROTECTION

EPICOR II PROCESSING

RADIONUCLIDE CONCENTRATIONS (pc/l)

Radionuclide	System Performance · Influent	Cleaned Water	10CFR20 MPC Conc.	EPA 570 Drinking Water	Calculated Concentration If Discharged
Cesium 137	7.2×10^9 = 7.2 μc/cc	73.7	2×10^4	2×10^2	0.019
Cesium 134	1.55×10^9	26.8	9×10^3	2×10^4	.007
Strontium 89	9.4×10^9	96.2*	3×10^3	20	0.025*
Strontium 90	1.8×10^9	18.4*	3×10^2	8	0.005*
Tritium	2×10^7	12.6×10^6	3×10^6	2×10^4	3.3×10^3

* Estimated Values

Actual Values will soon be available

FIGURE 4

SESSION M

DEMONSTRATION OF ALTERNATIVE DECONTAMINATION TECHNIQUES AT TMI

H. W. Arrowsmith

PNL

(slide 1) I work in the Materials Department at the Pacific Northwest Laboratory (PNL) in Richland, Washington. Over the last three years we have been working on DOE sponsored programs to develop electropolishing, vibratory finishing and other associated metal cleaning and finishing techniques into an integrated large-scale decontamination system capable of decontaminating large volumes of TRU and other surface contaminated solid waste.

I would like to describe to you a program that is jointly funded by the DOE (Division of Nuclear Power Development) and General Public Utilities (GPU). The object of this jointly funded project (slide 2) is the transfer decontamination technology from the other DOE funded programs for an in-plant demonstration of advanced decontamination processes capable of significantly reducing occupational radiation exposure to workers in nuclear power plants. The project will utilize the Three Mile Island nuclear power station as the test facility and the demonstration activities will be conducted in conjunction with the on-going TMI-II recovery operations. PNL's main role will be 1) design and procurement of all decontamination equipment, 2) provide technical

support for GPU's design of the decontamination facility and the installation of the decontamination equipment, 3) provide technical guidance to GPU's operating staff, 4) collect, analyze and evaluate exposure reduction data and disseminate results via progress reports and 5) promote transfer of the successfully demonstrated decontamination technology to the nuclear industry.

This slide (slide 3) illustrates the broad areas of decontamination technology that are being developed by Richard Allen and myself at PNL for the DOE Division of Waste Products. The objectives of this and other programs are to develop integrated decontamination systems for taking care of surface-contaminated radioactive waste generated by ongoing DOE operations and by the decommissioning of surplus DOE nuclear facilities.

On the left are the technologies we have been working on, the bar graph in the center shows you what we think our percentage of completion is and on the right side is when we expect to have each of those technologies completed and documented so that the various DOE facilities can use those technologies to solve their site specific decontamination problems.

This slide (slide 4) illustrates the decontamination facility we have in operation at Pacific Northwest Laboratory as part of the effort to develop an integrated decontamination system. Basically, you'll find

a solution processing facility which allows us in one half to clean up the contaminated acid used in the electropolishing systems and the other half is a waste evaporator used to clean up the other liquid effluents from the system. The next major part of the facility is what we call the pretreatment/sectioning facility. This room is used for the testing of various kinds of sectioning and disassembly of equipment. Here you see depicted an automated electropolishing system and various pumps and filters. In this room is a manual electropolishing system, perhaps many of you visited this facility and had a chance to look at it. Without going into a lot of detail, we have developed this integrated facility in which we're able to decontaminate large volumes of material and produce only small volumes of solid waste as a waste product. The emphasis of this program is to take transuranic contaminated material and decontaminate it to less than 10 nanocuries per gram which allows you to bury that material in shallow land fill, and eliminate, of course, the very expensive geologic disposal. But, by the same means, the technologies developed for this project are directly applicable to the problems that you face in your nuclear reactors because it turns out that the decontamination technologies for TRU contamination are generally directly transferable across the board to your problems.

This is a view of the electropolishing room in our facility (slide 5) where we have an unpackaging glove box, an electropolishing tank, which holds 400 gallons and is 6-ft long by 4-ft deep by about 3-ft across, two rinse tanks the same size as the electropolishing tank, and the control panels. You will notice we've made extensive use of highly

polished stainless steel so that we can maintain the radiation level in this facility at very low background levels. The reason for doing that is that you can't get anything any cleaner than the surroundings in your decontamination facility. Even the floor plates are made so that if they become contaminated we can simply pick them up and put them in an electropolishing tank and decontaminate them.

This is a graphic depiction (slide 6) of the pretreatment/sectioning facility at PNL, and we will be putting in something like this at Three Mile Island, not for sectioning but primarily for the disassembly of components which are too large to be decontaminated as one piece. Here you can see manipulator arms reaching in for disassembly. In this view you see a robot arm with a plasma torch for sectioning glove boxes, but of course that's for the TRU end of our business.

Now I would like to discuss the transfer of the decontamination technology we have just reviewed to the decontamination demonstration program at Three Mile Island. This slide (slide 7) shows the possible location of the Decontamination Demonstration Facility (DDF) at the Three Mile Island nuclear plant (TMI). The selection of a location for the DDF is being made jointly by GPU, Betchel, and Battelle. This particular site is being considered because it allows easy access in and out of the Containment Recovery Building located here and because it is compatible with other construction programs now being planned at TMI. The planned flow path of components and equipment from the containment is as follows:

1. equipment will be removed from the reactor containment facility and moved through the air lock into the Containment Recovery Building seen here,

2. the equipment will be given a preliminary decontamination treatment and then,

3. moved in containers to the DDF located here. I would like to emphasize that since this is a demonstration, components to be processed through the DDF will be selected to give the widest possible challenge to the decontamination techniques and equipment being evaluated.

This slide (slide 8) is an elevation view of the facility that we plan to put in. Basically, the demonstration facility begins here with a room much like our pretreatment/sectioning room where you can disassemble material. For those of you who are familiar with glove box structures, that's what you see. The round circles are gloves so that you will be able to reach in and will still have good beta shielding. The windows are lead impregnated windows; overhead cranes allow you to pass material through. We plan to do a preliminary decontamination in here and final decontamination by immersion in tanks outside the enclosure. We expect to have 95% of the contamination removed before the overhead crane here picks up the material and passes it on.

This is a plan view (slide 9) of the same piece of equipment. You have the disassembly area, the glove boxes for pretreatment and a vibratory finisher, which we'll talk about later, for doing small components. I expect that the vibratory finisher will take over the majority of the load of decontaminating tools that are used in the decontamination work. Here are the electropolishing tanks we talked about.

This slide (slide 10) shows the decontamination techniques to be evaluated during the demonstration decontamination program. Immersion electropolishing will be used with both acid electrolytes and basic electrolytes. Many of you are familiar with the use of acid electrolytes from the work done at Battelle and the commercial applications of that technique. A new electrolyte we're planning to try out at the DDF is one that we call a basic electrolyte. It is essentially a sodium nitrate solution. The reason for considering it is that as you electrochemically process something you form a precipitate immediately, from the metals being dissolved, and that precipitate carries the contamination to the bottom where you can clarify the electrolyte by filtration or by processing with a centrifuge. After clarification you have non-contaminated electrolyte, free of solids, and ready to be reused. Probably one of the other really important transfers of technology will be what we call in situ electropolishing and we'll talk about that in detail later. Other techniques will include barrel electropolishing, vibratory finishing, high pressure sprays and Freon® cleaning technology. For those of you who are not familiar with electropolishing as a decontamination technique, it is an electrochemical process. This slide (slide 11) illustrates the essential components: the part to be

decontaminated is made anodic and is surrounded by a conductive electrolyte and cathodes that can either be the tank walls or separate sheets of metal insulated from the tank walls. To decontaminate a part, you place the part to be decontaminated into the tank and apply a positive voltage. The time required for a typical stainless steel object to be decontaminated usually ranges from 5 - 15 minutes. This slide (slide 12) illustrates what happens to an as-received #1 stainless steel bar. After five minutes of electropolishing, you'll notice that you have removed all the microporosity and the little bumps and humps. Based on the laboratory studies that we have done, the contaminatability of the surface is related to these micro features which you see on the surface and not these large rolling hills. So even five minutes of electropolishing removes the microporosity and also removes most of the contamination. Of course the length of time required to decontaminate something depends a lot on what the surface looked like to begin with before it was contaminated and secondly, how it was contaminated.

Electropolishing has been used to decontaminate a great variety of equipment at Hanford. This slide (slide 13) shows a carbon steel valve which Tom Hall from UNC Nuclear Corp. gave us to decontaminate as a test of the electropolishing technique. This slide (slide 14) shows the same valve after being decontaminated to background. Using special in situ techniques, it was possible to decontaminate even down in the bottom areas of this valve, around the seal surfaces and even inside the pipes attached to the valve. With careful engineering, you can decontaminate

valves and other sensitive components without destroying the integrity of the component. In the last year we have decontaminated three valves from UNC Nuclear Corporation's N-Reactor. After decontamination, the valves were remanufactured and returned to service.

I expect in situ electropolishing techniques to be of great interest to those of you who are associated with reactor operations. This slide (slide 15) lists the four major categories of in situ electropolishing. The categories are; pumped stream, contact, brush/swab, and internal cathode devices. In view of the short presentation time, I would like to give you one example of how we have used each in situ technique. If any of you are interested, we can talk in greater detail later.

The schematic diagram shown in this slide (slide 16) illustrates what we call a pumped stream arrangement. Remember that in the electrolytic operation we always have to maintain the cathode/anode relationship. The part we are trying to clean is made anodic and to complete the circuit we need a cathode. In this case we use a pumped stream like this to do irregular surfaces. For example, we recently did some work for the Navy decontaminating a RLW tank where they had many pipe flanges and other irregular surfaces inside and wanted it decontaminated to the backgorund. Well, we had a choice, we could take the flanges apart and take the bolts out and really suffer trying to decontainate it by a swab or various other techniques or we could use this pumped stream technique. You can imagine that it doesn't make too much difference what the geometry of the surface is as long as you can pump a stream against it. So in

this case, pumped stream techniques worked very well for cleaning out between flanges where you have the flanges separated by the width of the gasket.

This view is looking down into this 5,000 gallon Radioactive Liquid Waste (RLW) Tank which is approximately 6 ft in diameter and 24-ft long. (Slide 17.) This slide shows you the business end of the pumped stream device being used in the RLW tank. If, for example, I'd already decontaminated most of the surface of the tank and I had just one small contaminated spot left, it would be a logical choice to use the pumped stream technique to decontaminate the "hot" spot. The disadvantage of the pumped stream technique is that the electrolyte runs uncontrollably from the area being decontaminated down to a collection point. Fortunately, phosphoric acid, which is generally used as the electrolyte, has a low recontaminatability factor and seldom recontaminates areas previously decontaminated by electropolishing.

A contact in situ device shown schematically in this slide (slide 18) has been developed to overcome the problem of free-flowing electrolyte. In this case, we actually circulate the electrolyte in a closed, sealed system. For example, if you had a fuel pool liner or a refueling cavity liner that needed to be decontaminated and you could not allow the electrolyte to escape, the contact in situ devices would be an excellent way to do it. By sealing the contact in situ device against the surface to be decontaminated, it is possible to circulate the electrolyte without any leakage. The hydrogen and oxygen generated during the

electrolytic operation is released into the reservoir and escapes. A dc power supply capable of supplying up to 10 Vdc is used to operate the contact in situ device. This slide (slide 19) shows a polished spot that is 2 ft in diameter. The polished spot was produced by sealing the contact in situ device against the sheet, filling the in situ device with electrolyte, supplying approximately 10 Vdc at 250 A for 10 minutes. Not only is this device useful for decontaminating refueling cavities, it is also useful for improving the surface finish on those liners after they have been fabricated and put in place.

This slide (slide 20) shows a schematic diagram of a brush in situ device. A simple version of this called a "Johnny Mop" has been used for many years by the nuclear industry. We have modified the original deisgn to include the addition of a pump to circulate electrolyte through the porous insulator. The porous insulator prevents accidental shorting between the cathode inside the in situ device and the anode or part being decontaminated.

A magnetically coupled swab in situ device is shown in this slide (slide 21). This device was used to decontaminate 85% of the inner surface area of a 5000 gallon radioactive liquid waste tank using only 50 gallons of electrolyte. What you see is a frame with two strong samarium cobalt magnets, one on each end, and the swab in situ device in the center. To operate the device, acid is recirculated through this tube and electrical current is provided through this electrical lead. A current density of 4 A per square inch provides a decontamination rate of approximately 3 square feet per minute. This magnetically coupled

swab in situ device is moved by moving a set of samarium cobalt magnets on the outside of the tank which are magnetically coupled to the magnets attached to the frame of the in situ device. We expect to adapt this device in many ways to assist in the decontamination of the Three Mile Island Plant.

This slide (slide 22) shows a schematic diagram of an internal cathode device used to do the internal surfaces of pipes. In this device, we use a cathode which can slide along the inside diameter of the pipe to be decontaminated. In some cases we actually fill the pipe with electrolyte and let it drain out through a drain. In other cases we only fill the annulus between the anode and cathode without filling the pipe. For example, if you had a 29 in. diameter pipe you would not want to fill it because of the large volume of electrolyte required. This slide (slide 23) shows an internal cathode device being inserted into a contaminated 4-inch-diameter diffuser pipe. Using this cathode design and flexable leads, we were able to decontaminate 24 ft of the diffuser line even though it made two 90^{o} turns.

Use of electropolishing as a decontamination technique has been increasing over the last three years. Commercial applications in the nuclear power industry have been rapidly increasing since the DOE sponsored public seminar on electropolishing techniques in April of 1978.

Up to present time, electrolytes have been used until they were either lost by drag-out or become too contaminated to allow continued

use because of personnel exposure problems. Techniques to purify the electrolytes are being developed as part of the decontamination programs sponsored by the DOE. A schematic diagram shown in this slide (slide 24) illustrates how the reciprocating acid adsorption system functions. Contaminated acid is forced under pressure into the bottom of the resin column. As the column is filled, the acid is adsorbed onto the resin and the water and impurities including the contamination are passed through the resin and out of the column as waste. After the resin column is saturated with acid, the input process is stopped and a flow of water from the top of the resin column elutes the acid from the resin. One pass through the resin column has removed up to 70% of the contamination present in the electrolyte. The removal of the contamination and the subsequent extension of the electrolyte life is significant in terms of radiation exposure reduction and waste volume reduction.

This slide (slide 25) is a photograph of an acid purification system which can purify approximately 10 gallons of 70% H_3PO_4 acid per hour. The acid purification system we plan to install at Three Mile Island will be able to process approximately 30 gallons per hour, which will allow the removal of a significant amount of contamination from the phosphoric acid bath and out into the waste stream.

Vibratory finishing techniques (slide 26) used in the metal finishing industry for the preparation of surfaces and the deburring of edges have been adapted for use as a decontamination technique. This technique is

capable of removing most of the gross surface contamination and produces surfaces that are usually nonsmearable. The decontamination of hand tools that are going to be reused in a radiation zone is a good application of vibratory finishing technqiues. In this application, you also want to reduce the dose to near background from the tools and you want to remove all the smearable contamination from the tool to prevent contamination of the worker or nonradioactive equipment used in the same area. The significant advantage of vibratory finishing as compared to electropolishing is that vibratory finishing can be used as a mass production tool whereas electropolishing is a labor intensive batch process capable of decontaminating components to background.

Vibratory finishing techniques are excellent for running tools like a hammer that has a wooden handle and a metal head or a hammer that has a plastic handle and a metal head. This slide (slide 27) illustrates a vibratory finishing system that has been modified for use as a decontamination system. The vibratory finishing process combines mechanical scrubbing action with chemical cleaning action. The process takes place in a vibratory tub of loose ceramic or metal shapes (media) through which flows a liquid chemical compound. The vibrating tub is powered by an electric motor which drives a system of eccentric weights. The energy of the tub causes the medium to scrub the surface of the parts to be decontaminated while the liquid compound flushes away the material removed by the scrubbing action. The material flushed out by the flowing water or sodium hydroxide end up here in the sludge tank. Liquid from the sludge tank is filtered and recirculated through the vibratory finisher.

This slide (slide 28) is a photograph of the vibratory finisher we plan to install at TMI. This top view of the vibratory finisher shows the machine without the media in the tub. You can throw your tools, nuts from the reactor head and any number of metallic and nonmetallic components into this machine and let them go around and around. After an hour of processing, the parts are discharged across this screen and the media falls back into the tub.

This slide (slide 29) shows two of the three different types of vibratory finishing media that we have evaluated. The larger pieces are ceramic media with aluminum oxide impregnated into the ceramic binder. The cone shaped pieces are plastic media with aluminum oxide. One type of media not shown on this slide is metal burnishing media. The metal media is made out of case hardened carbon steel which produces a media that retains its shape and produces almost no secondary waste from media wear.

This slide (slide 30) shows the flow diagram for the vibratory finisher. We circulate liquid from the sludge tank through a pump and filter and then back into the vibratory finisher. In some cases we use recirculated solution for 45 minutes of an hour cycle and then use clean solution for the last 15 minutes of the cycle. Flow rates are generally 20 gallons per hour for the 12 cu. ft vibratory finisher being installed at TMI. This slide (slide 31) shows a before and after of some carbon steel pipe clamps that were contaminated with fission products and were

heavily rusted from years of use. After one hour of vibratory finishing, the pipe clamps were decontaminated to a low level with essentially no smearable contamination remaining.

Freon as a decontaminating solution is the final decontamination technique I would like to discuss with you. Many of us have used Freon in ultrasonic cleaners and vapor degreasers for many decontamination tasks with mixed success. At one time ultrasonic cleaner/vapor degreasers were the thing to have at every reactor or nuclear installation. Recent experiences in the nuclear industry have probably decreased the popularity of this kind of equipment probably since they were expected to be a cure-all for our decontamination problems.

This slide (slide 32) shows a new commercial machine which uses Freon as a working solution. This machine is capable of pumping Freon under high pressure against the surface of the part to be decontaminated. One of the things this system is very useful for is the decontamination of electrical components. A good example of equipment that has been decontaminated using this technique is shown in this slide (slide 33). Here you can see an electric drill motor inside the washing chamber of the Freon decontamination system. Freon under 2000 psi is pumped from the hand held nozzle and used to blast the loose contamination from the surface of the drill motor. The Freon used in the system is type 113 with several additives. One of the interesting things about Freon is that through the use of bonding agents, it is possible to put up to 30% water in Freon or water plus soap or water plus an acid. The Freon can

be used as a solvent and the carrier of a variety of strong cleaning compounds. We believe that Freon will be a significant part of our decontamination demonstration at Three Mile Island.

The objective of this demonstration of alternative decontamination techniques is to take all the decontamination techniques that we have developed under DOE sponsored programs and bring them to Three Mile Island for field testing. The idea is to put in place a fully integrated system capable of decontaminating components and processing the wastes. This slide (slide 34) shows the waste treatment systems flow diagrams. We are putting in a small evaporator to recycle our rinse waters, an acid purification system to purify our electrolytes, a centrifuge to separate the solids from the liquids and finally a solidification unit to make a solid out of our waste streams.

We hope that by transferring these techniques out of a program that you in the reactor business wouldn't normally see, and putting them in what we'd hope will be a showcase decontamination system, we will be able to demonstrate how effective or ineffective they are in the Three Mile Island setting. One of the most important objectives of this project is to document just how good and how bad these various techniques are, especially when they are used in combination with each other. With that, I think I'm finished; any questions?

Question: What kind of DF's order of magnitude can you get with each of these .....

Answer: Electropolishing, the greatest, 10,000 to 1 as a conservative number. Freon and vibratory finishing are probably at the other end of the spectrum, maybe 100 to 1 with vibratory finishing.

Question: Unintelligible?

Answer: Well, the hydrogen comes off the cathode and oxygen off the part we are trying to clean. The evolution of oxygen from the part we are cleaning would promote cleaning. Actually in electropolishing you are removing the surface that the contamination is sitting on. In our experience at Hanford, it does not make much difference what the surface is composed of or what the contamination is, electropolishing will be effective as long as the part to be decontaminated has a conductive surface.

Question: You stated that 95% of the contamination would be removed prior to going into the electrochemical bath as a pretreatment stage.

Answer: By that I meant in that enclosure, that glove box like enclosure, we intend to remove most of the contamination so that we don't have a problem, airborne or otherwise, when we take it out into those tanks which are cleaner and get it completely clean.

Question: How much is the cost of the system for Three Mile Island?

Answer: The actual equipment itself, just the hardware, is estimated to be $250,000.

Question: During your electropolishing are you within the polishing region or off in the etching region?

Answer: We have a slide that shows that, but we are operating in the polishing region or even above it.

Question: I notice that you didn't metion ultrasonics at all, is that nonflamable?

Answer: We've done a good deal of experimentation with ultrasonics and traveled a good deal in the nuclear industry and we find almost a universal sadness about ultrasonic cleaning and our tests verify that it is not effective, particularly if you use Freon in ultrasonics.

Question: Are you .....the compatibility of these waste and the different solidification .....exist. Have you investigated that in detail.

Answer: We have two things going at the same time. No. 1 - We have to have TMI tell us what we are allowed to do here because they obviously must operate under a different set of rules than anybody else in the whole world. At the same time, at Hanford we are undergoing a testing program of several different materials including Dow, there are any number of them which you can add as either a liquid or powder to your affluents and solidify them in 55 gallon drums. That is what we intend to do here, but we have to wait for Three Mile Island people to tell us which would be acceptable.

Question: Do you consider the chemistry, the long term effects of the chemicals that are made, being present in the waste and how it might affect .....

Answer: As of right this minute we do not have, as you might have guessed, a really - no one knows for sure just what we're going to have when we have components coming out of the containment and so the answer to your question is no, but it will be under consideration.

Question: Nothing was mentioned about Can-Decon or Adell process, why not?

Answer: As a way of solidifying?

Questioner: No, as a way of decontaminating.

Answer: Well those are chemical decontamination solutions and we are not into those kinds of things; there are a lot of people who are. And I would assume that the people sitting in this room, for example, are going to talk about that in the workshops as to where they should be applied and as to where these kinds of decontamination techniques I talked about should be applied. They don't always compete. For example, if you could possibly do it, you'd much rather decontaminate the primary system with a chemical rather than take it apart and do it by components as much of the technology I've talked about fits into. So one has to choose what options you have.

Question: How about compatibility with the manufacturer's materials, the chemicals used, etc.

Answer: That's an interesting question. I think that the best example that I could give you very quickly is that NRC permitted one of the vendors to electropolish the 29 inch primary steam lines at Surrey in the refabrication and the reinstallation of the steam generators there. Now that is not a complete answer and I'd be glad to go into a long discussion with you if you are interested. It is a problem and sooner or later we have to deal with it because electropolishing is a very good tool when used in the right place, but sooner or later we have to deal with just where can we use it and where can't we.

Question: Have you looked at the long term effects on materials in the system?

Answer: We have looked at it to some extent, but keep in mind most of the technology that this was developed from is a junk man business. So that wasn't a problem because all we're trying to do is get it from very expensive disposal to very inexpensive disposal. The transfer of this technology to Three Mile Island brings up the question of when you use technologies like this, whichever technology it is, it doesn't make any difference if it's chemical, electropolishing or what; can we reuse that system once we've used that. And that's going to be one of the most important discussions, I think, later on in Three Mile Island.

Question: .....project manager.....first place question, I didn't hear it initially.

Answer: He wanted to know how safe are these various decontamination techniques in terms of if you go ahead and electropolish it is that pipe going to last for 30 years or is it going to fail because of electropolishing.

Question: Well there was a study done and it was okayed.

Answer: That's right, NRC permitted it.

Question: We removed a couple of mils ..... by accident and we had a decontamination factor of about 5,000, I just sat here and figured it. We were running about 25 hours internally and were able to go down to around 5 .....

Answer: Now the in situ techniques that I talked about, had they been available when you did your steam generator pipe, this would have eliminated the necessity of having that very large bath with 1,600 gallons of acid in it. We could have done that pipe with - for example, we did a 5,000 gallon tank with 50 gallons of electrolyte. So there are improvements to be made in technology and Three Mile Island is going to be a place that forces the developments because I think in situ electropolishing techniques are going to be one of the most important contributions that are made.

DEMONSTRATION OF ALTERNATIVE DECONTAMINATION TECHNIQUES AT THREE MILE ISLAND

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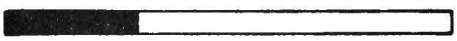
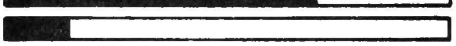
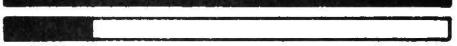
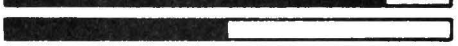
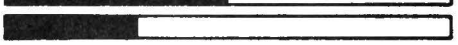
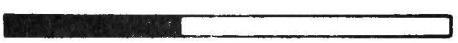
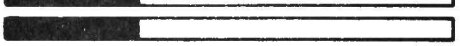
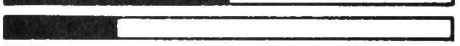
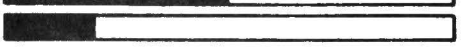
SLIDE 1

THE OBJECTIVE OF THIS JOINTLY-FUNDED GPU/DOE PROJECT IS THE DEVELOPMENT AND IN-PLANT DEMONSTRATION OF ADVANCED DECONTAMINATION PROCESSES CAPABLE OF SIGNIFICANTLY REDUCING OCCUPATIONAL RADIATION EXPOSURE TO WORKERS IN NUCLEAR POWER PLANTS

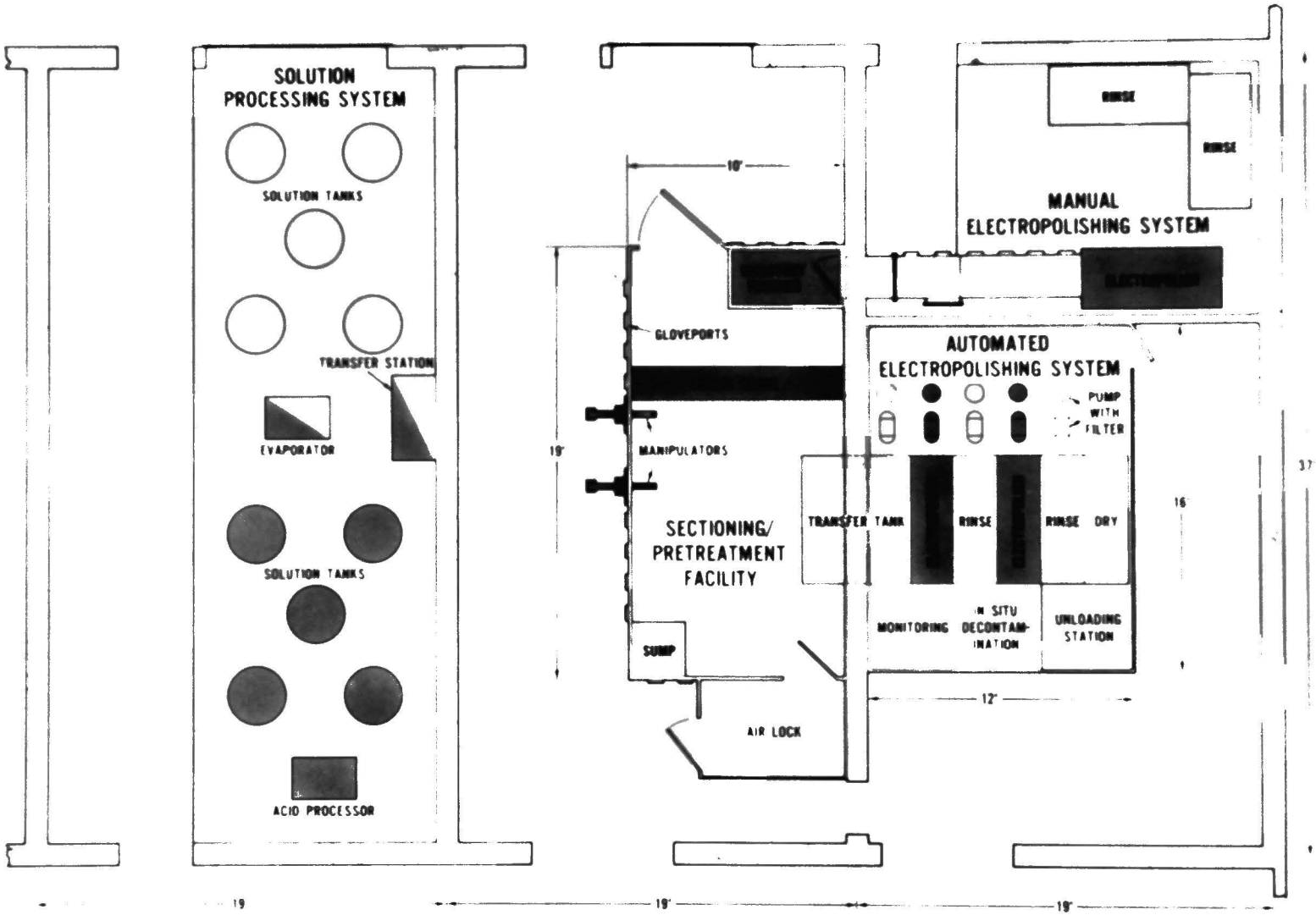
SLIDE 2

SLIDE 3

TRU WASTE DECONTAMINATION

TASK / ACTIVITY	STATUS AT END OF THIRD YEAR	COMPLETION DATE
PRECLEANING		12/80
DISASSEMBLY		6/80
SECTIONING		
• PLASMA TORCH		3/81
• SAW / NIBBLER		9/80
• SHEAR / PUNCH		12/80
PRETREATMENT		
• MECHANICAL		9/80
• HIGH-PRESSURE SPRAY		3/81
• ULTRASONIC CLEANING		12/79
• WET / DRY BLASTING		COMPLETE[
VIBRATORY FINISHING		12/81
ELECTROPOLISHING		
• IMMERSION TECHNOLOGY		6/80
• RACKING TECHNOLOGY		3/81
• BARREL SYSTEMS		6/82
• SYSTEM AUTOMATION		-----
• IN SITU TECHNIQUES		12/81
SOLUTION PROCESSING		
• ELECTROLYTE PURIFICATION		3/82
• AQUEOUS SOLUTION RECYCLE		9/81
CRITICALITY SAFETY		12/82
CONTAMINATION MEASUREMENT		3/83
PROCESS DEMONSTRATION		
• SECTIONING		9/81
• PRECLEANING / PRETREATMENT		12/81
• VIBRATORY FINISHING		12/81
• ELECTROPOLISHING		6/82
• SOLUTION PROCESSING		3/82
• INTEGRATED OPERATION		6/82
• TEST SYSTEMS & FACILITIES		12/82
TECHNOLOGY ADAPTATION		3/83
TECHNOLOGY TRANSFER		6/83
MATERIALS BALANCE & ECONOMICS		6/83

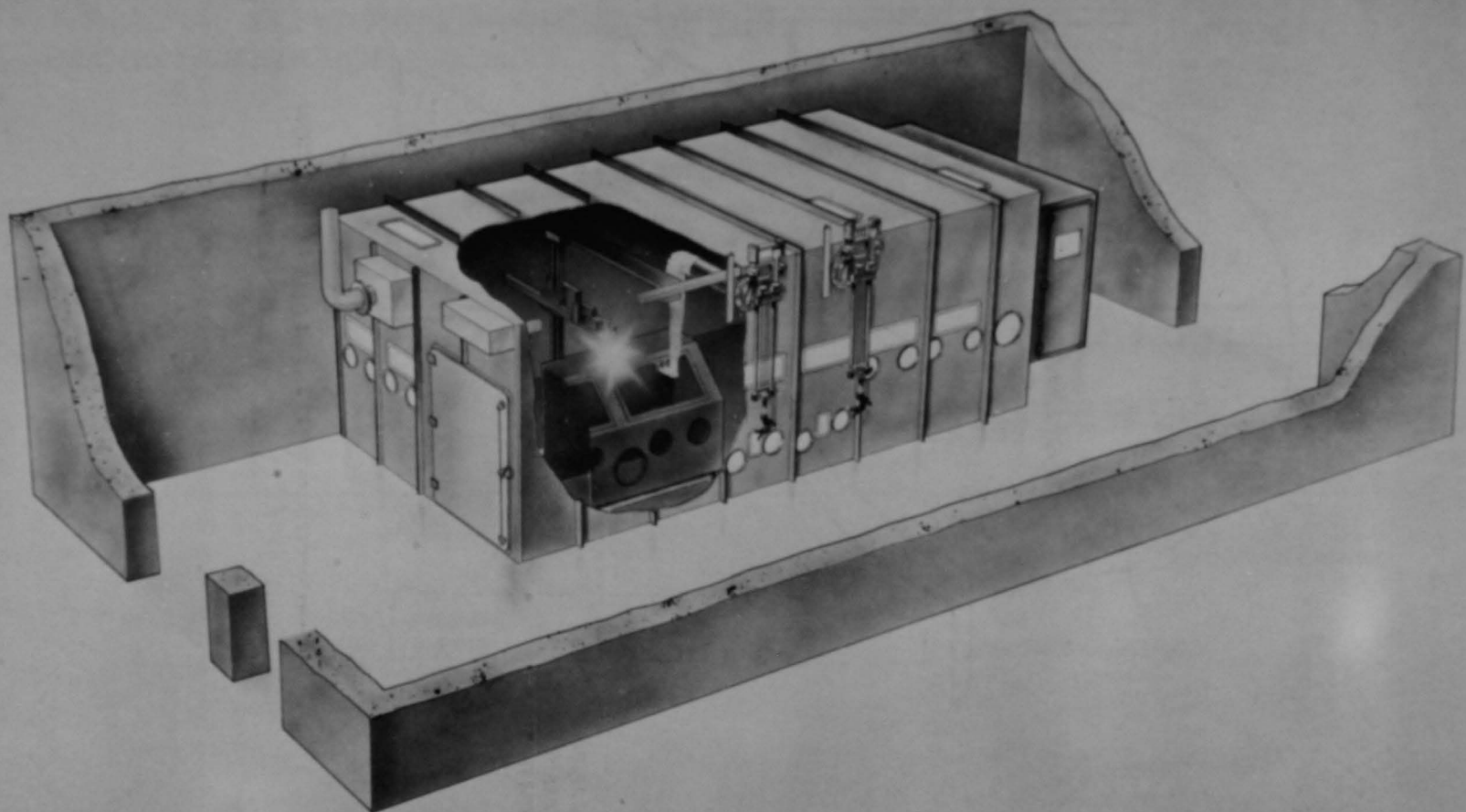
PROTOTYPE
DECONTAMINATION FACILITY



SLIDE 4

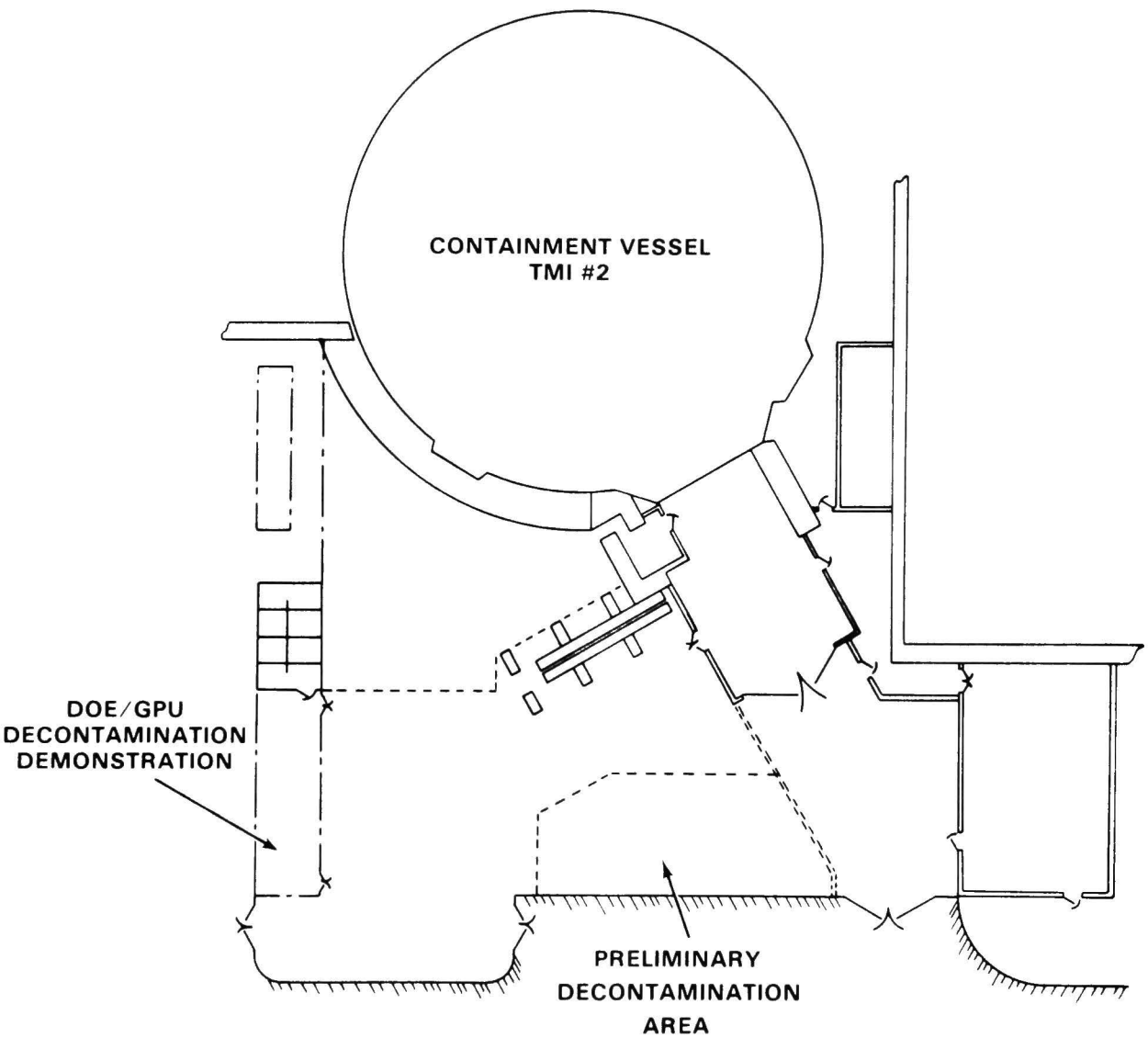
SLIDE 5

SECTIONING/PRE-TREATMENT FACILITY

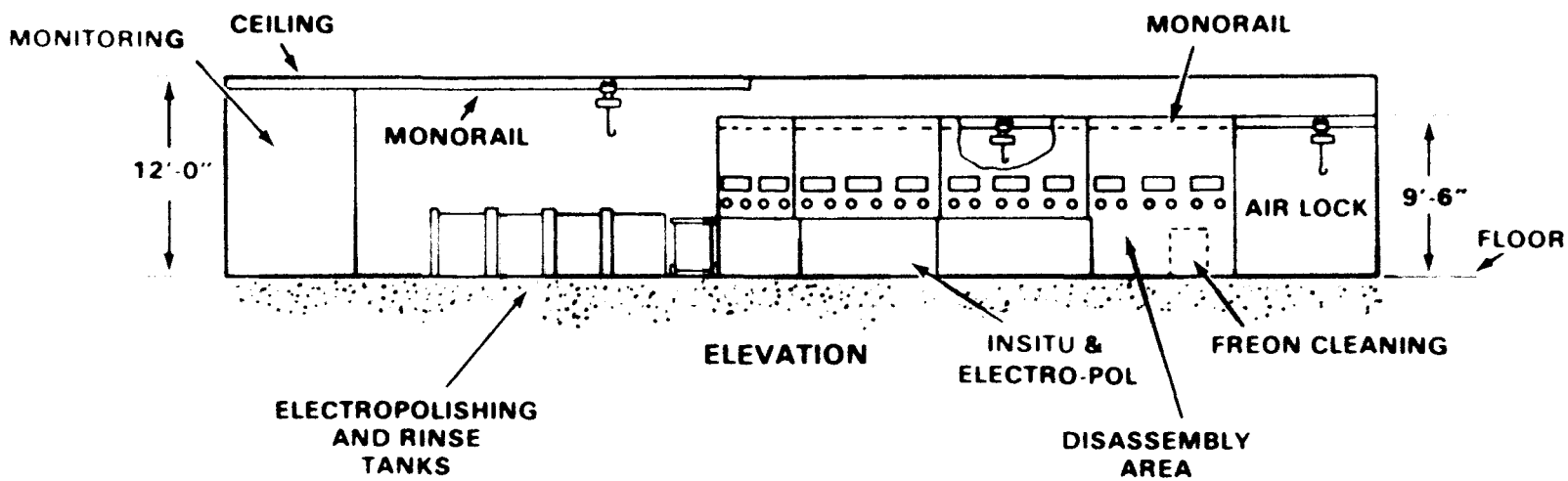


SLIDE 6

SLIDE 7

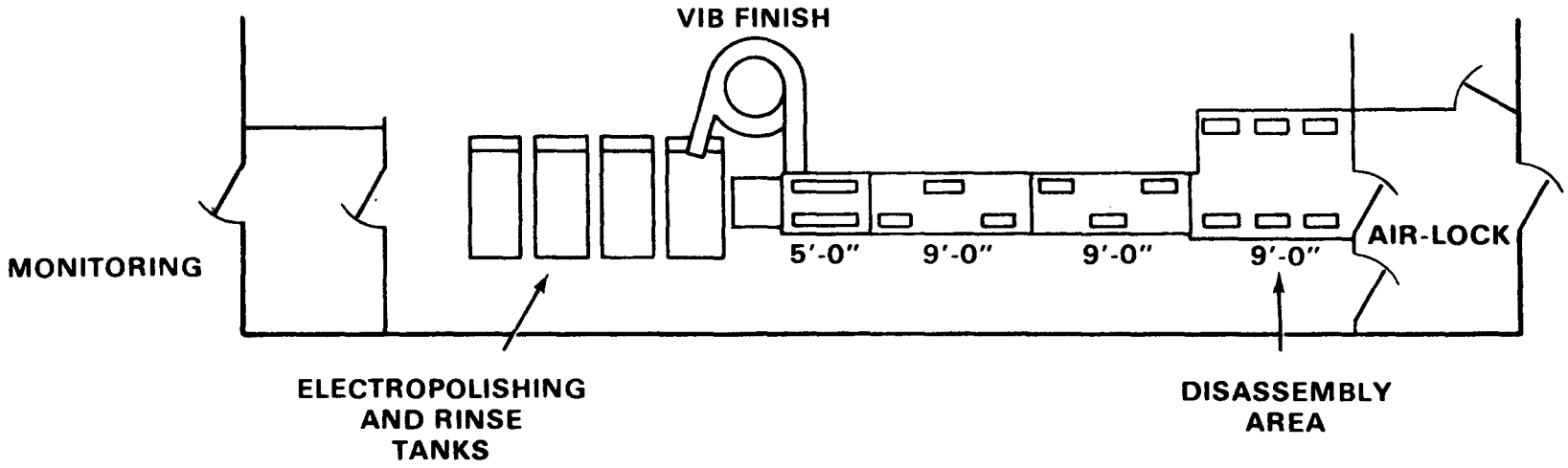

CONTAINMENT VESSEL
TMI #2
DOE/GPU
DECONTAMINATION
DEMONSTRATION
PRELIMINARY
DECONTAMINATION
AREA

A PARTIAL ELEVATION DRAWING OF THE DECONTAMINATION FACILITY



SLIDE 8

A PARTIAL PLAN VIEW DRAWING OF THE DECONTAMINATION FACILITY



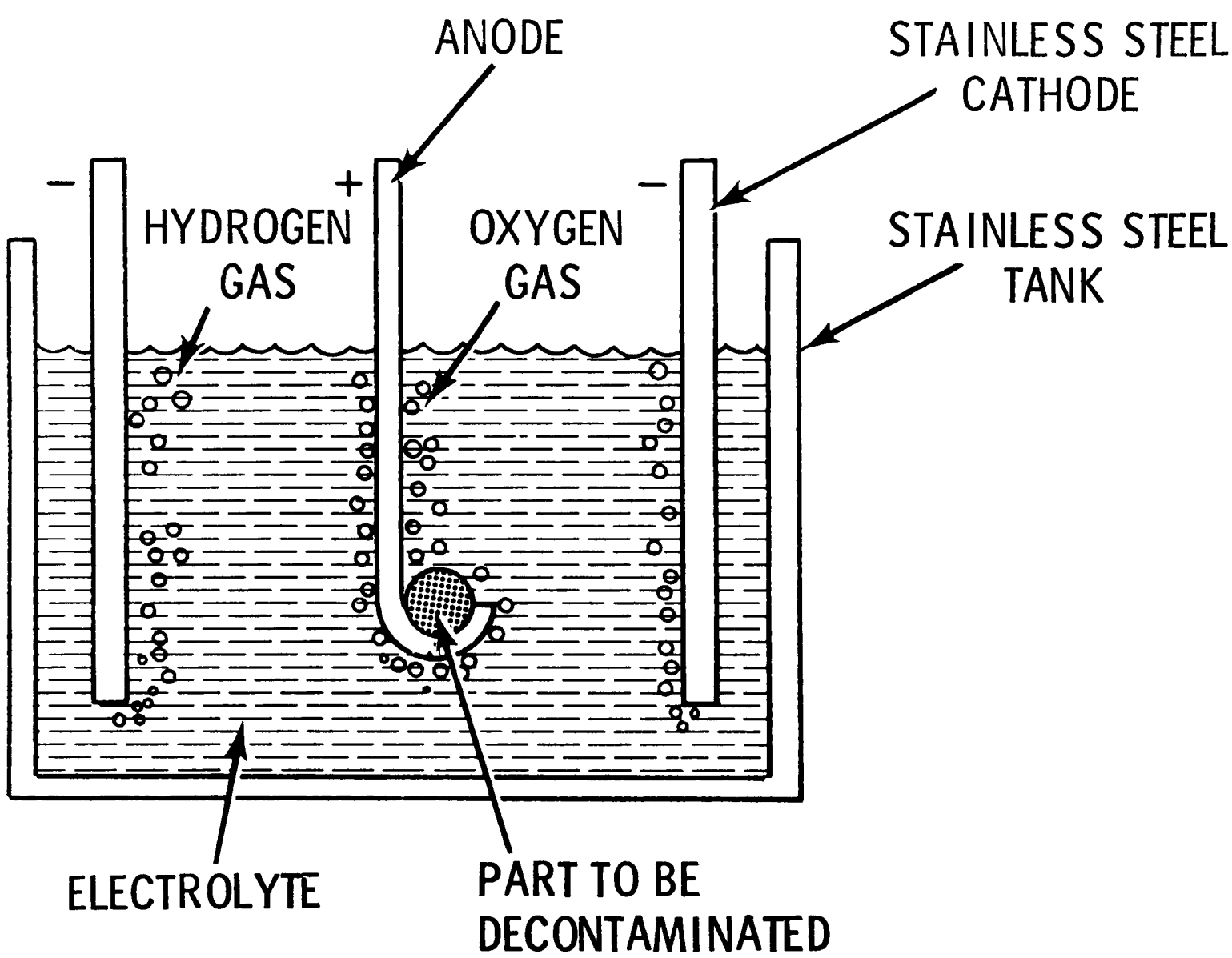
PARTIAL PLAN VIEW

SLIDE 9

DECONTAMINATION TECHNIQUES TO BE EVALUATED AND COMPARED

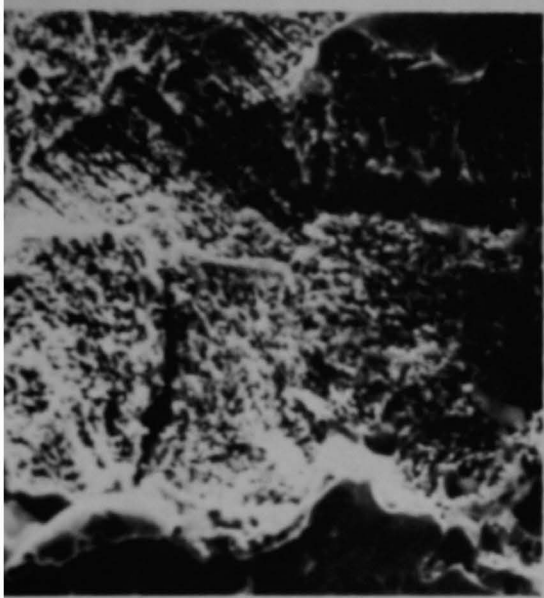
- IMMERSION ELECTROPOLISHING
  - A. ACID ELECTROLYTE
  - B. BASIC ELECTROLYTE
- IN SITU ELECTROPOLISHING
- BARREL ELECTROPOLISHING
- VIBRATORY FINISHING
- HIGH PRESSURE SPRAY
- FREON CLEANING
  - A. IMMERSION
  - B. SPRAY

SLIDE 10


ANODE
STAINLESS STEEL CATHODE
–
+
–
HYDROGEN GAS
OXYGEN GAS
STAINLESS STEEL TANK
ELECTROLYTE
PART TO BE DECONTAMINATED

SLIDE 11

ELECTROPOLISHING OF 304 L STAINLESS STEEL IN PHOSPHORIC ACID

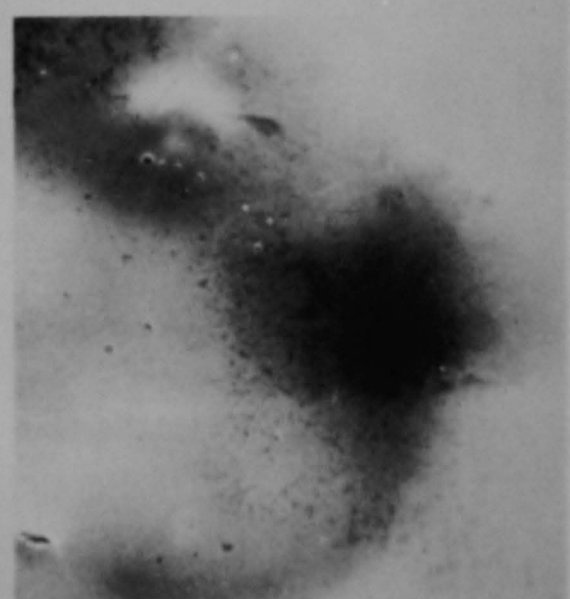


AS RECEIVED

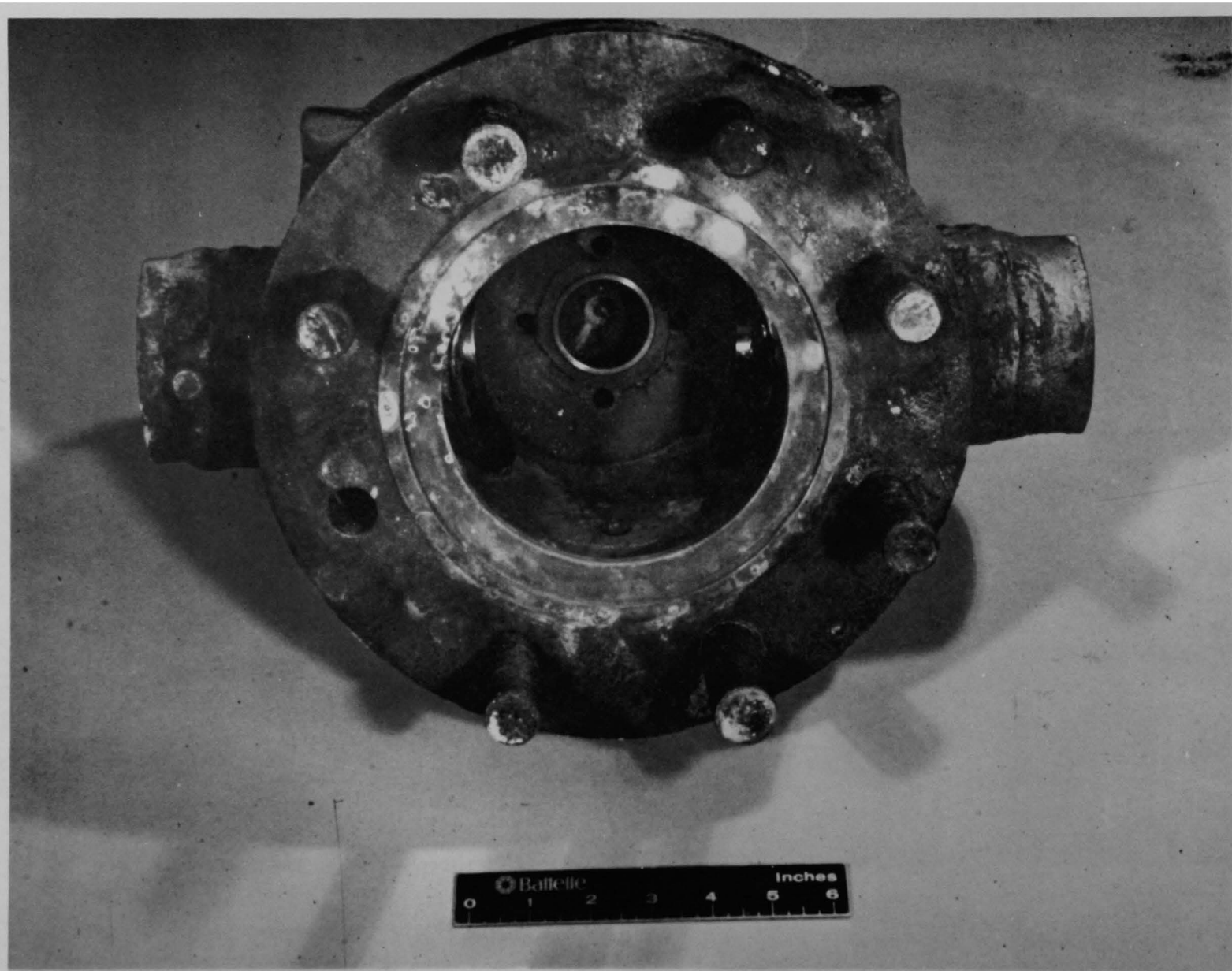
5 MINUTES OF ELECTROPOLISHING

25 μ

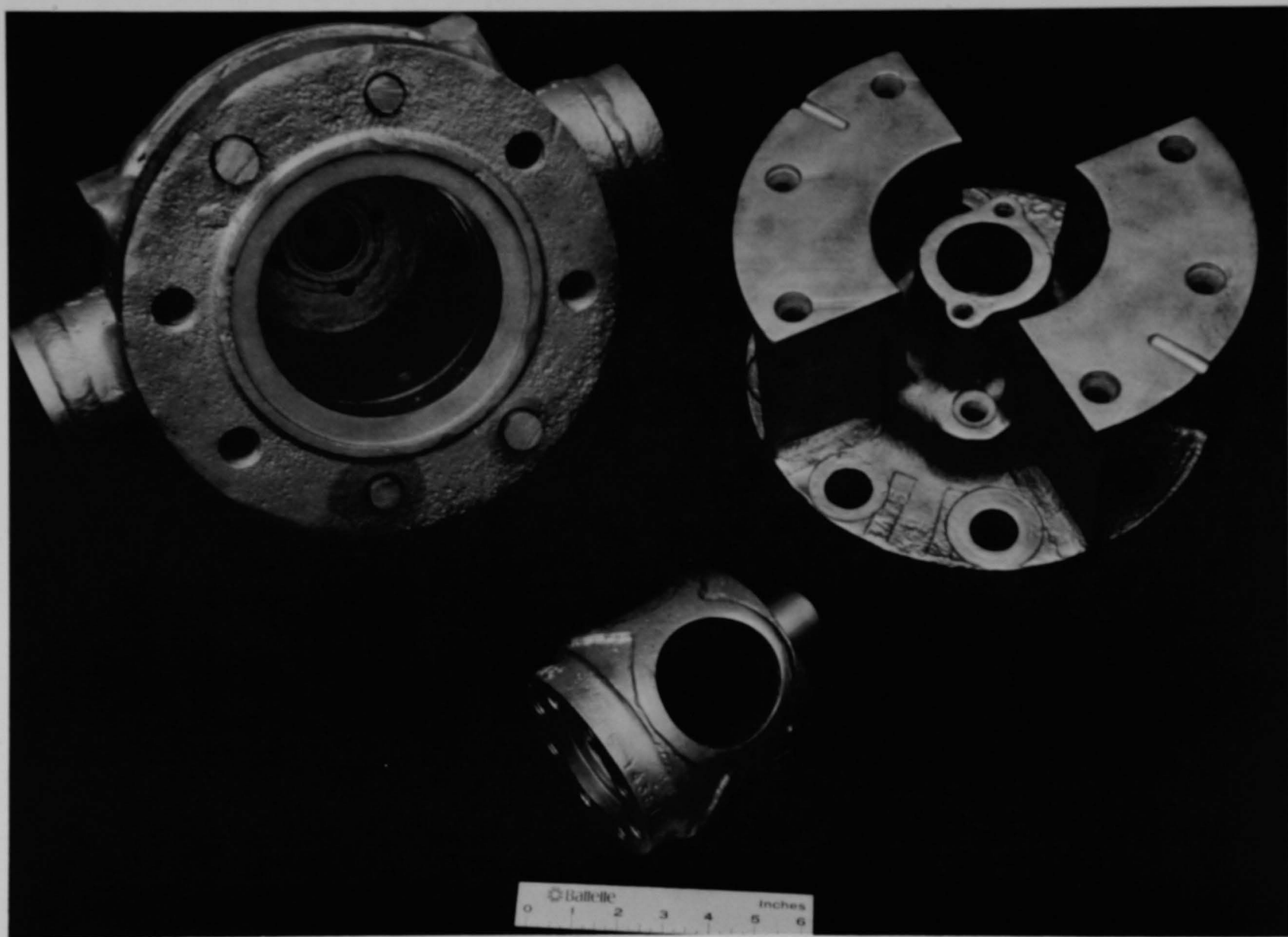
10 MINUTES OF ELECTROPOLISHING



15 MINUTES OF ELECTROPOLISHING



SLIDE 13



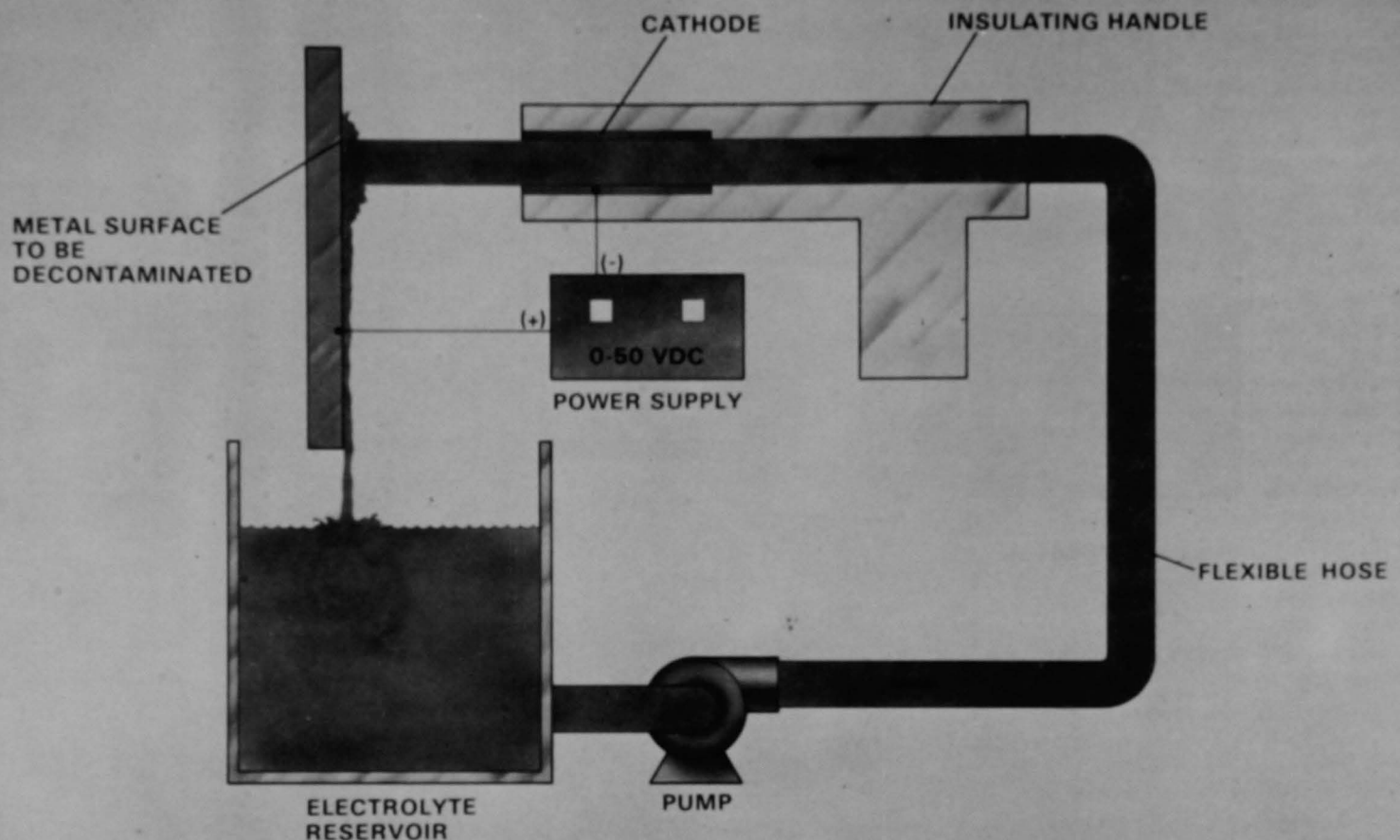
SLIDE 14

IN SITU TECHNIQUES

- PUMPED STREAM
- CONTACT
- BRUSH/SWAB
- INTERNAL SURFACES

SLIDE 15

PUMPED STREAM TECHNIQUE

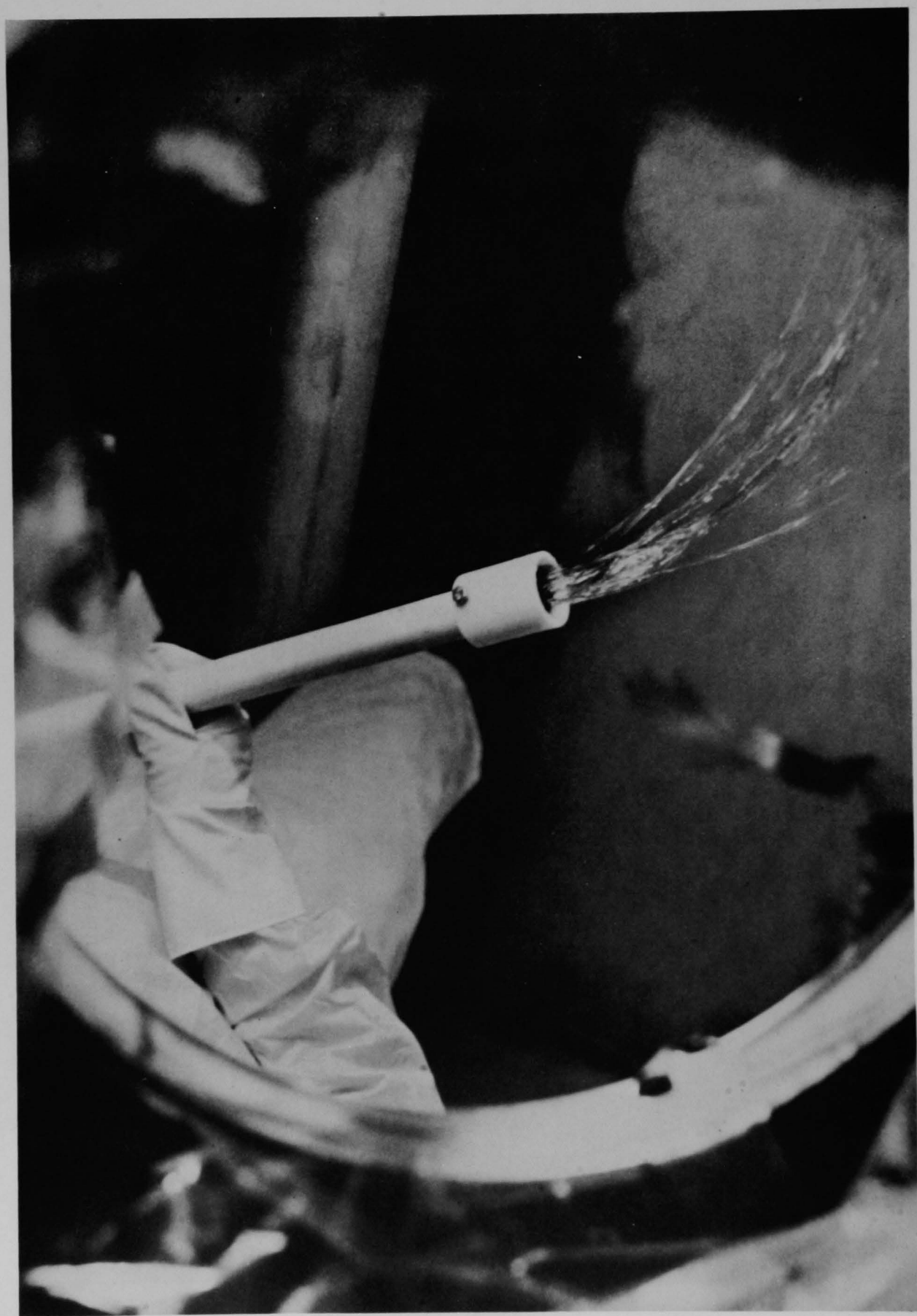


FEATURES:

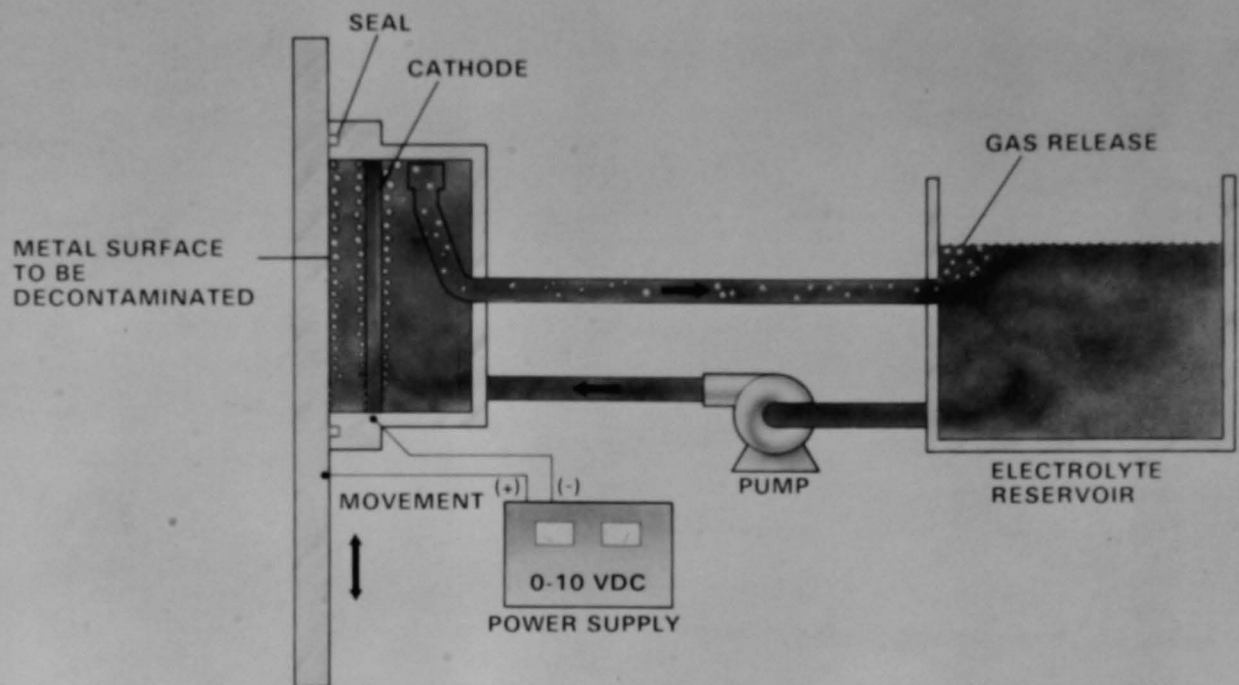
- DECONTAMINATE IRREGULAR SURFACES WITHOUT CONTACT
- PORTABLE SYSTEM IDEAL FOR FIELD APPLICATIONS
- SPOT DECONTAMINATION

SLIDE 16

SLIDE 17



CONTACT TECHNIQUE

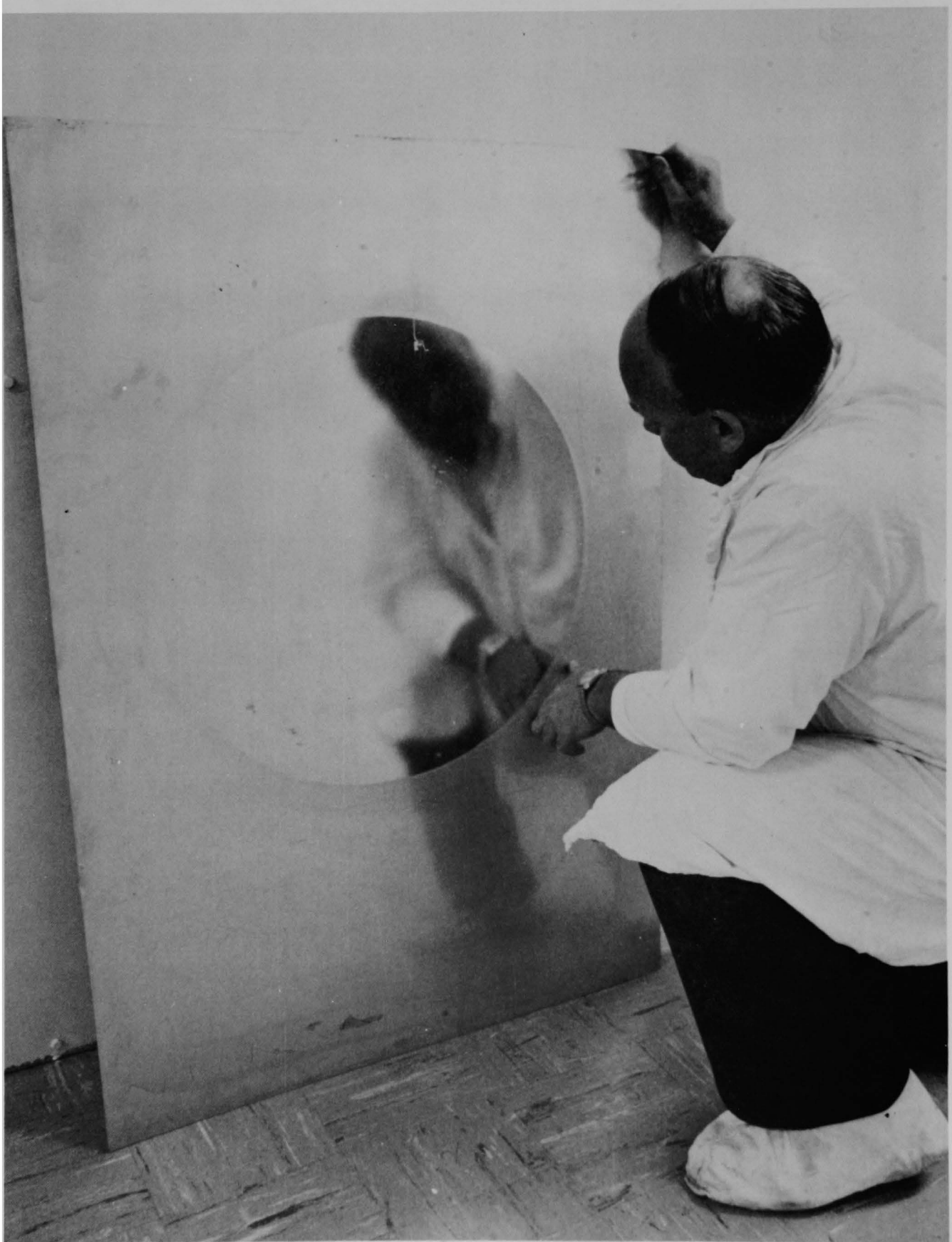


FEATURES:

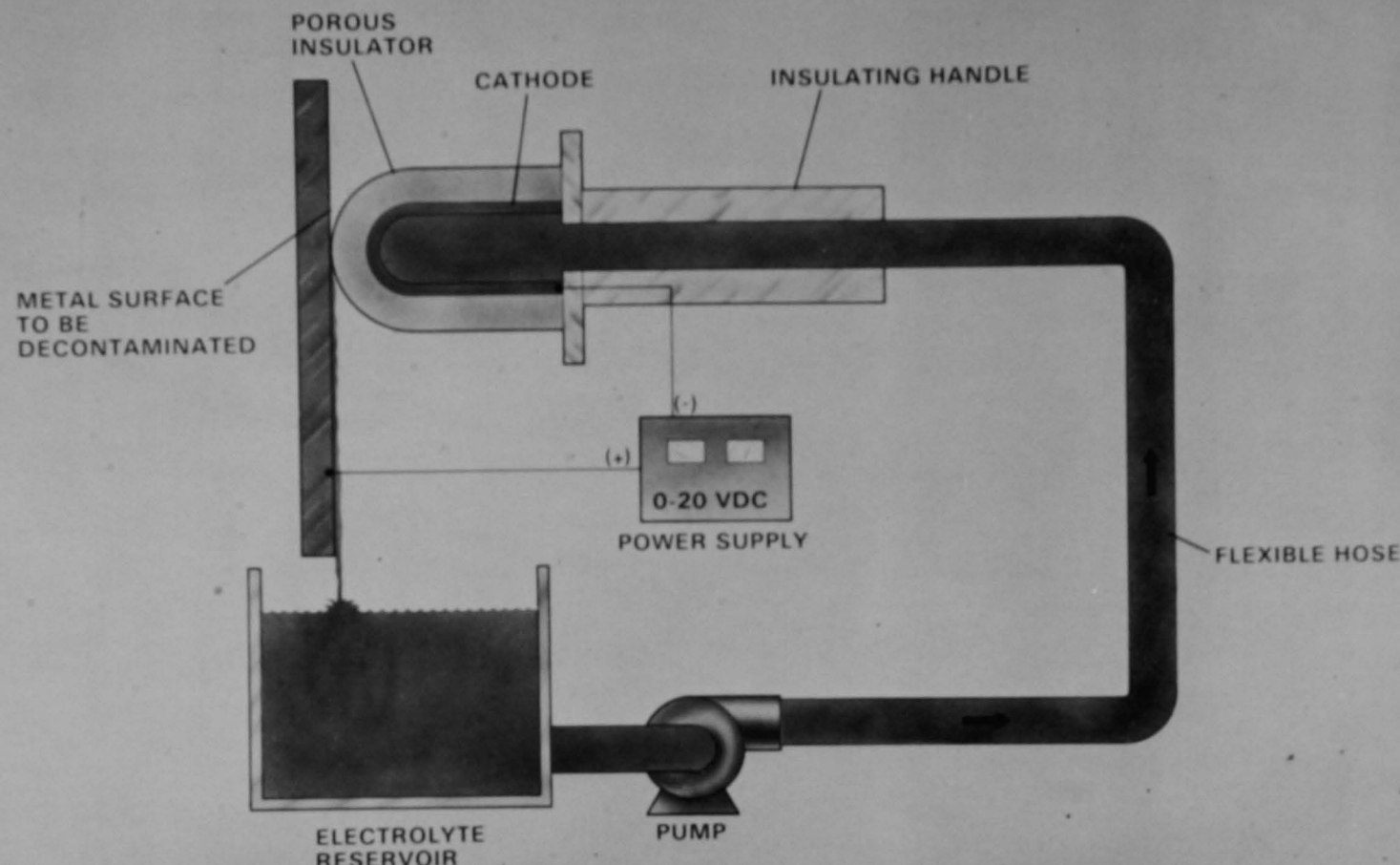
- DECONTAMINATE PLANAR SURFACES
- COMPLETE CONTAINMENT OF ELECTROLYTE
- USEFUL FOR LARGE AREAS

SLIDE 18

SLIDE 19



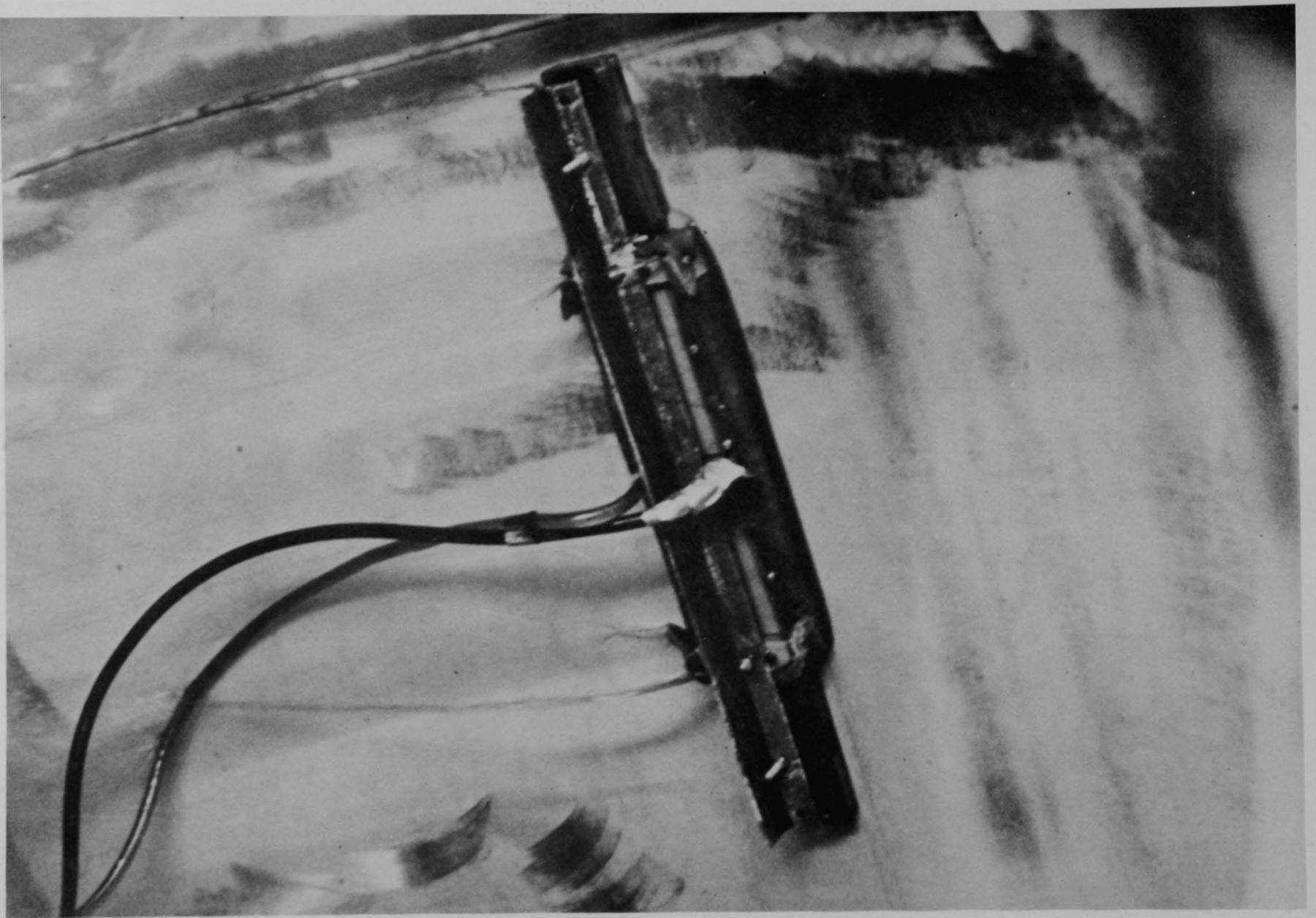
BRUSH TECHNIQUE



FEATURES:

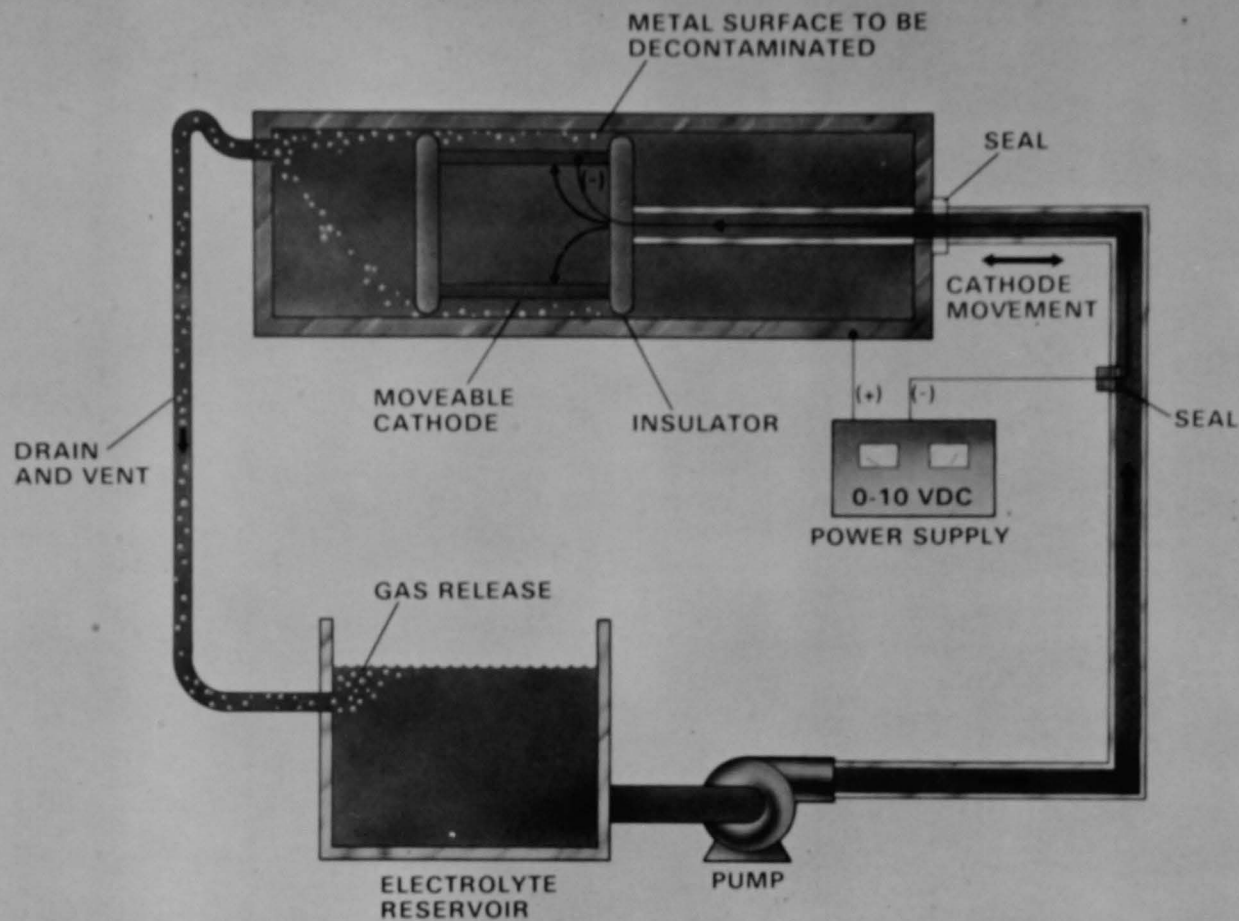
- PRECLEANING
- SPOT DECONTAMINATION

SLIDE 20



SLIDE 21

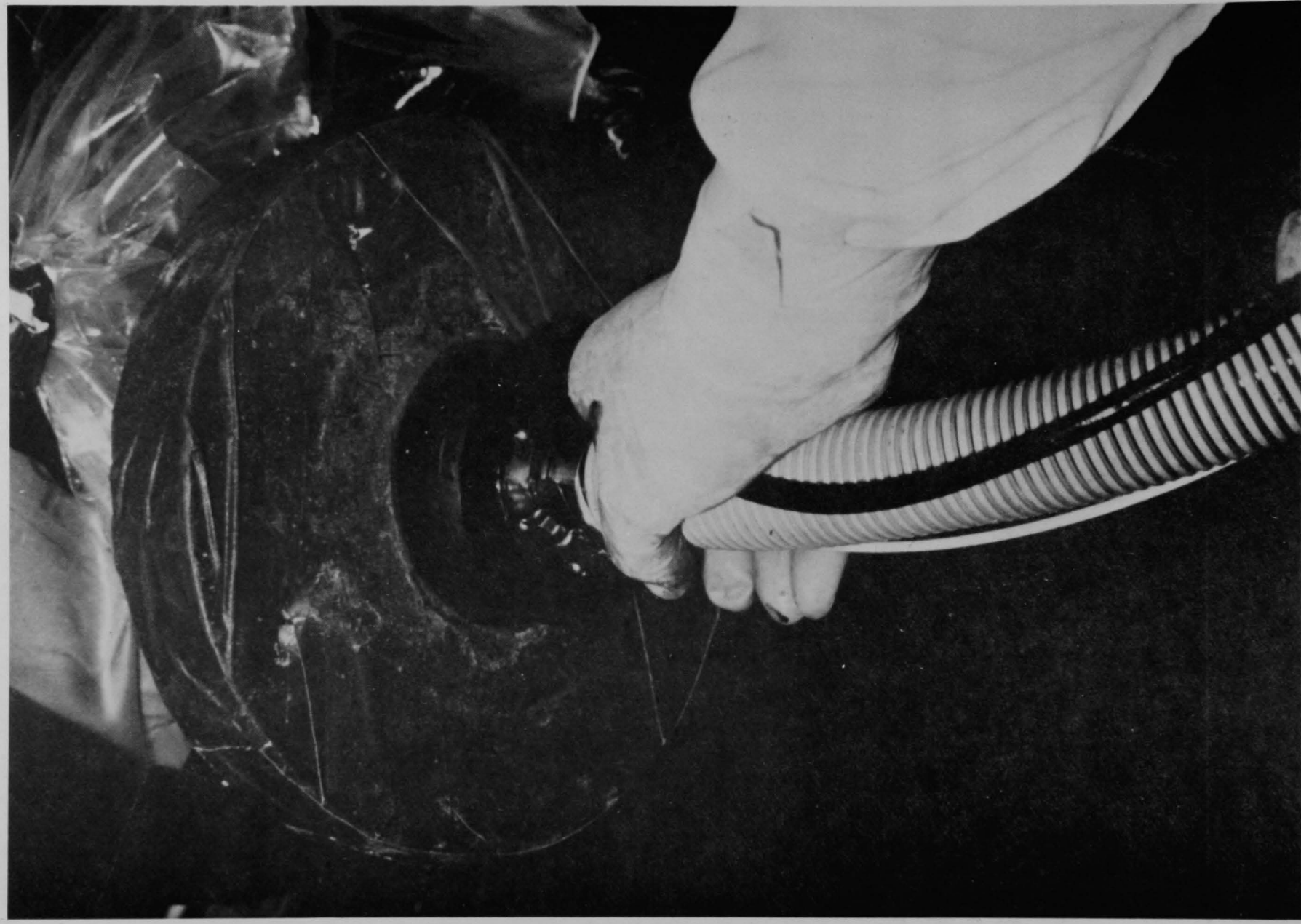
INTERNAL CATHODE TECHNIQUE



FEATURES:

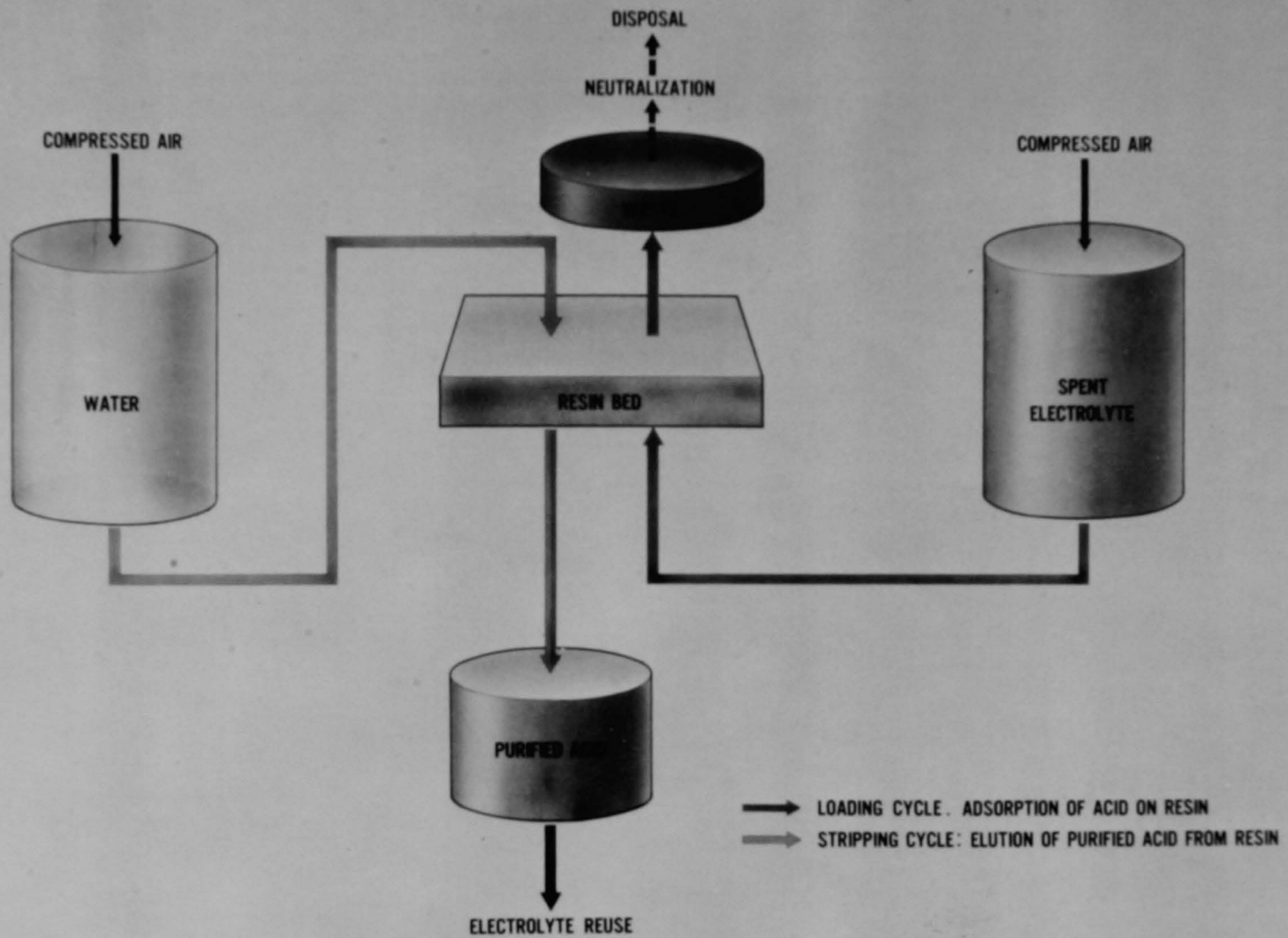
- DECONTAMINATE INTERNAL SURFACES OF PIPE
- DECONTAMINATE LONG LENGTHS
- DECONTAMINATE CURVED SECTIONS

SLIDE 22



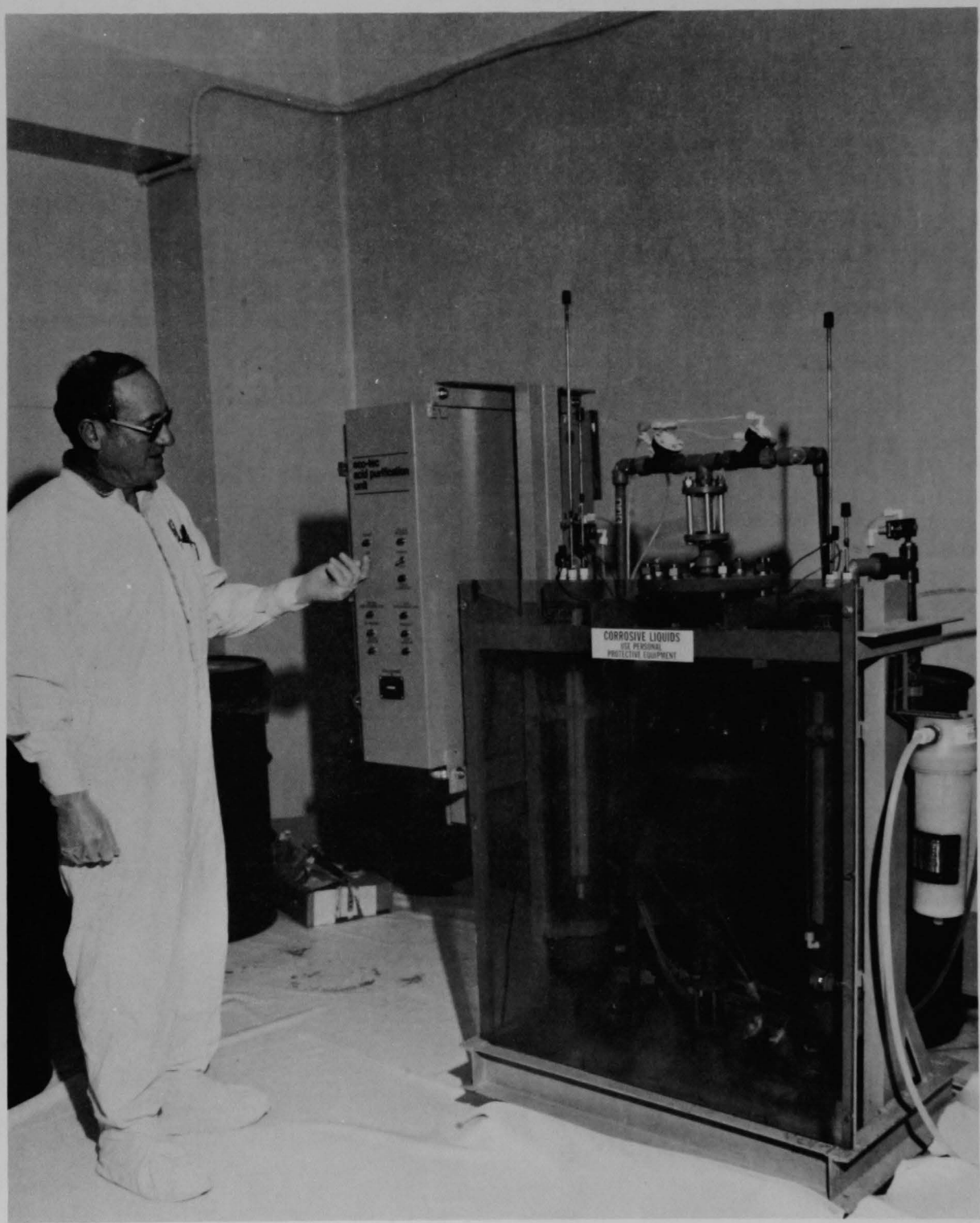
SLIDE 23

SPENT ELECTROLYTE PURIFICATION BY RECIPROCATING ACID ADSORPTION

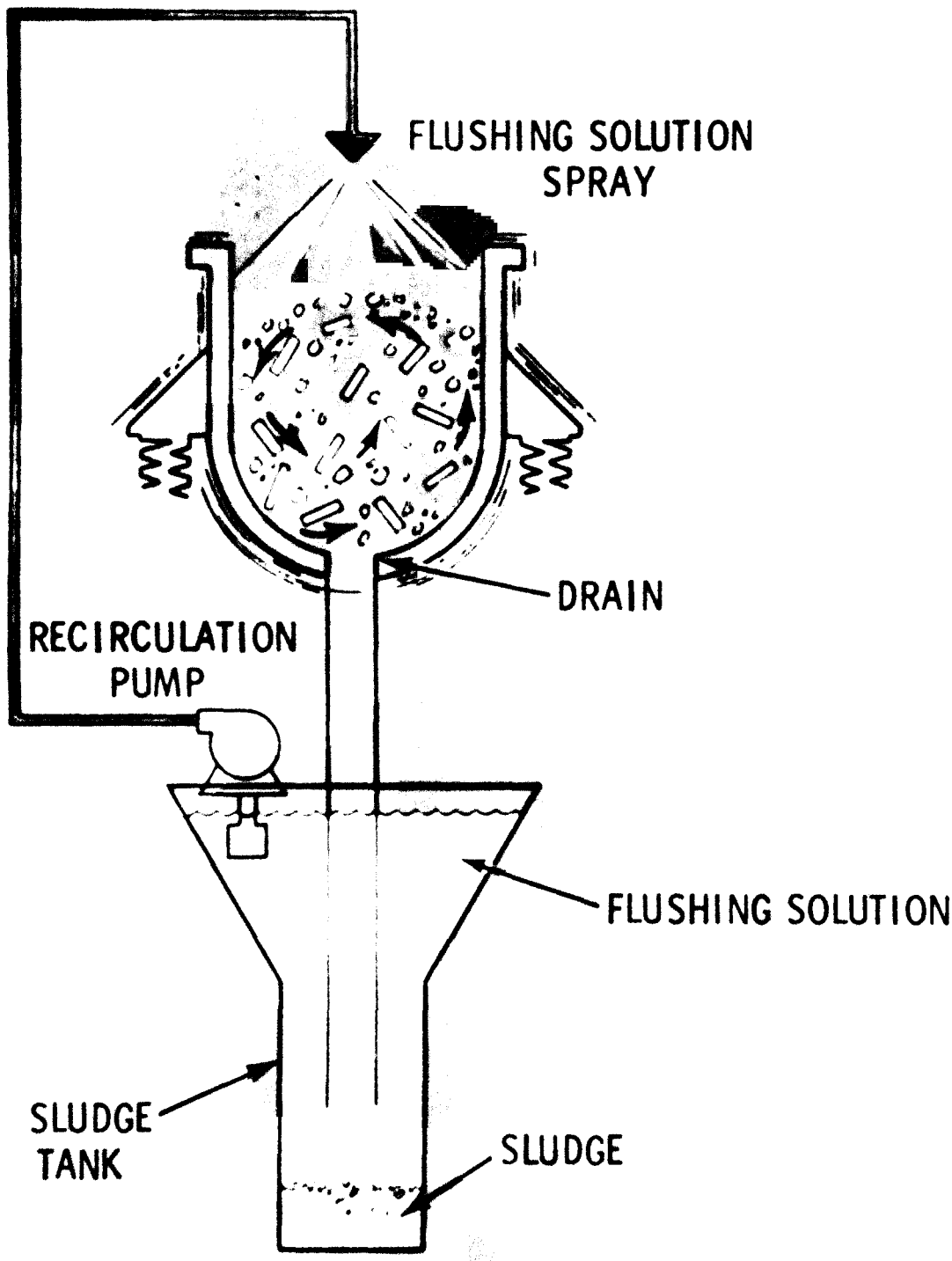


SLIDE 24

SLIDE 25



VIBRATORY FINISHING SYSTEM

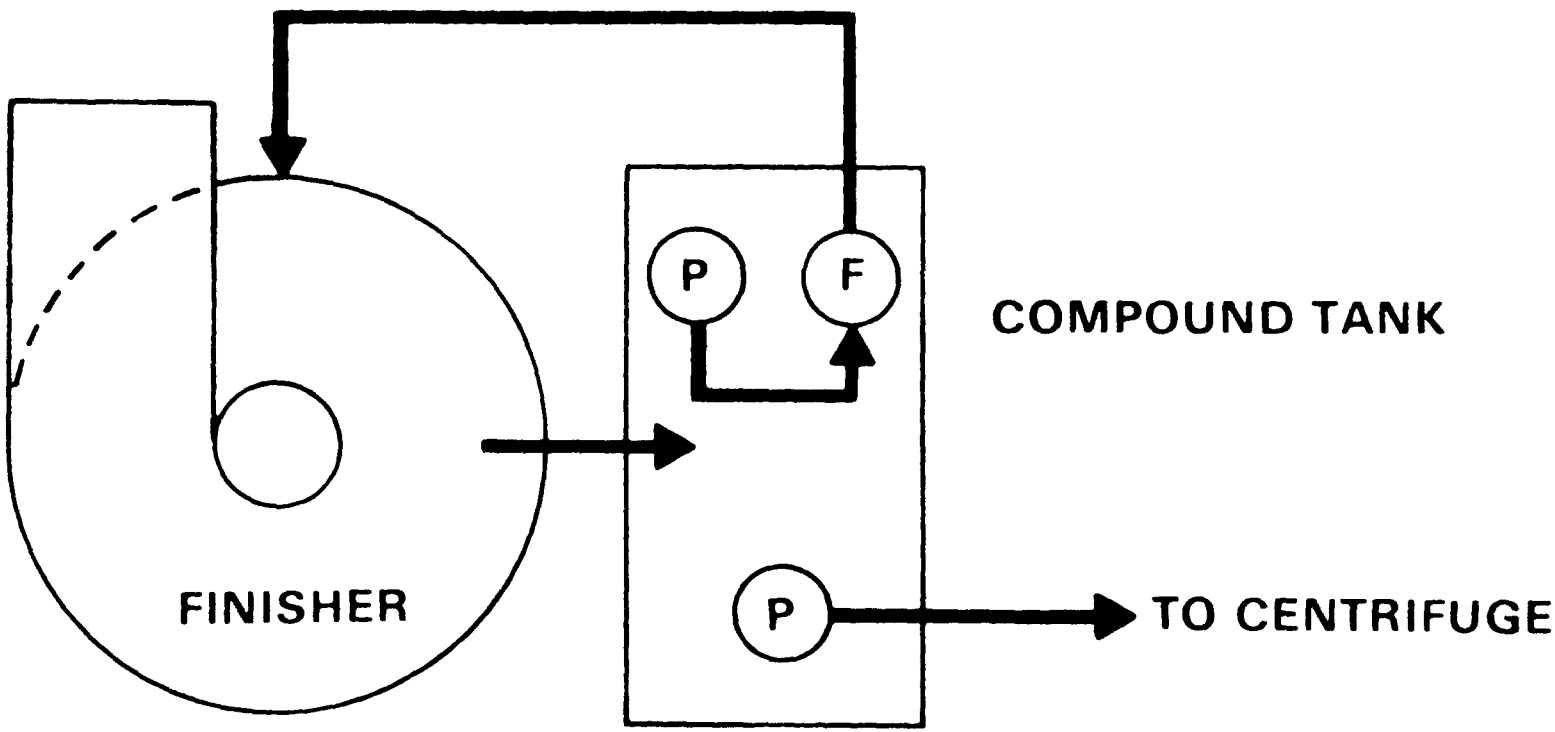


SLIDE 28

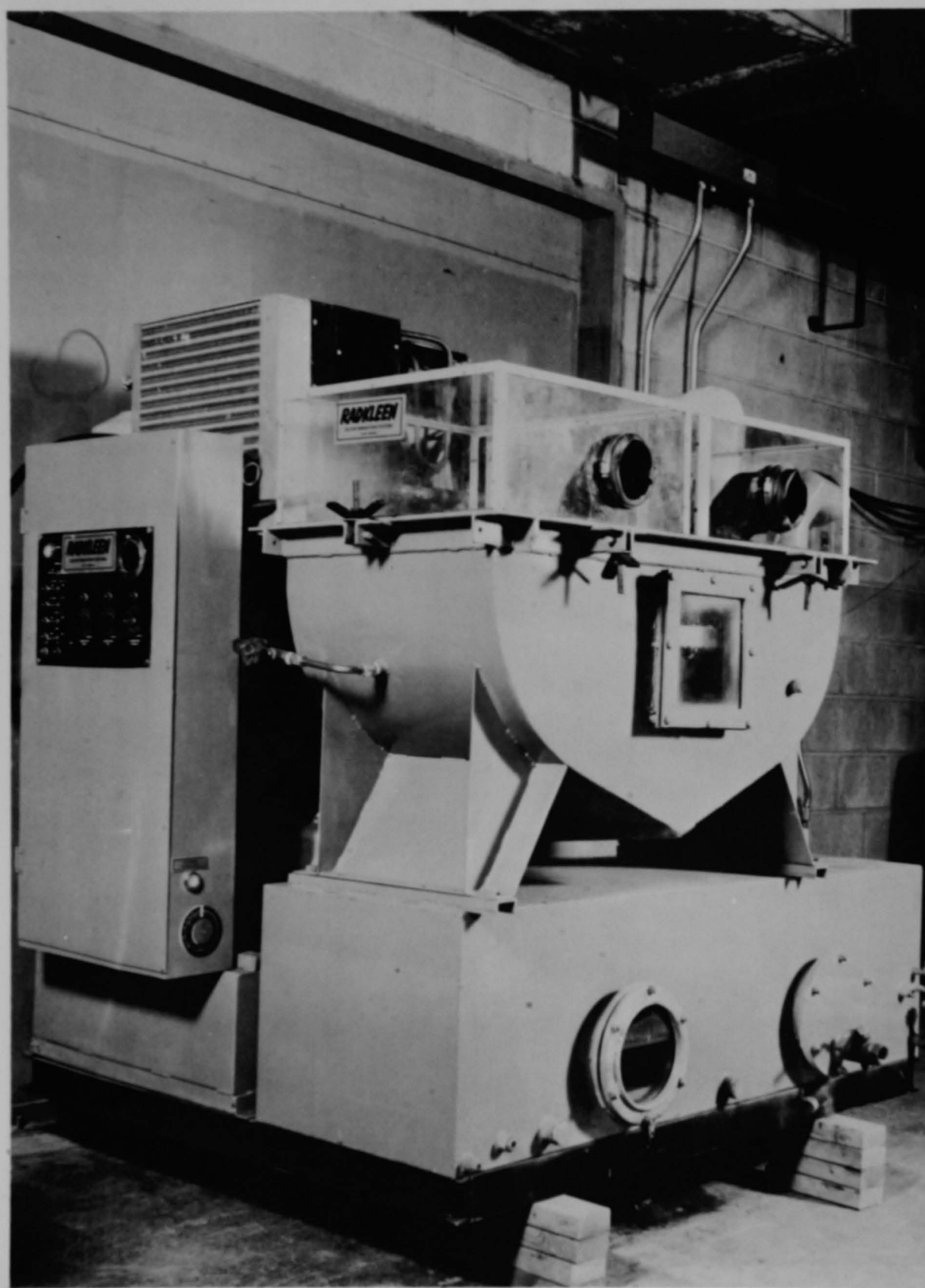
M-50

SLIDE 29

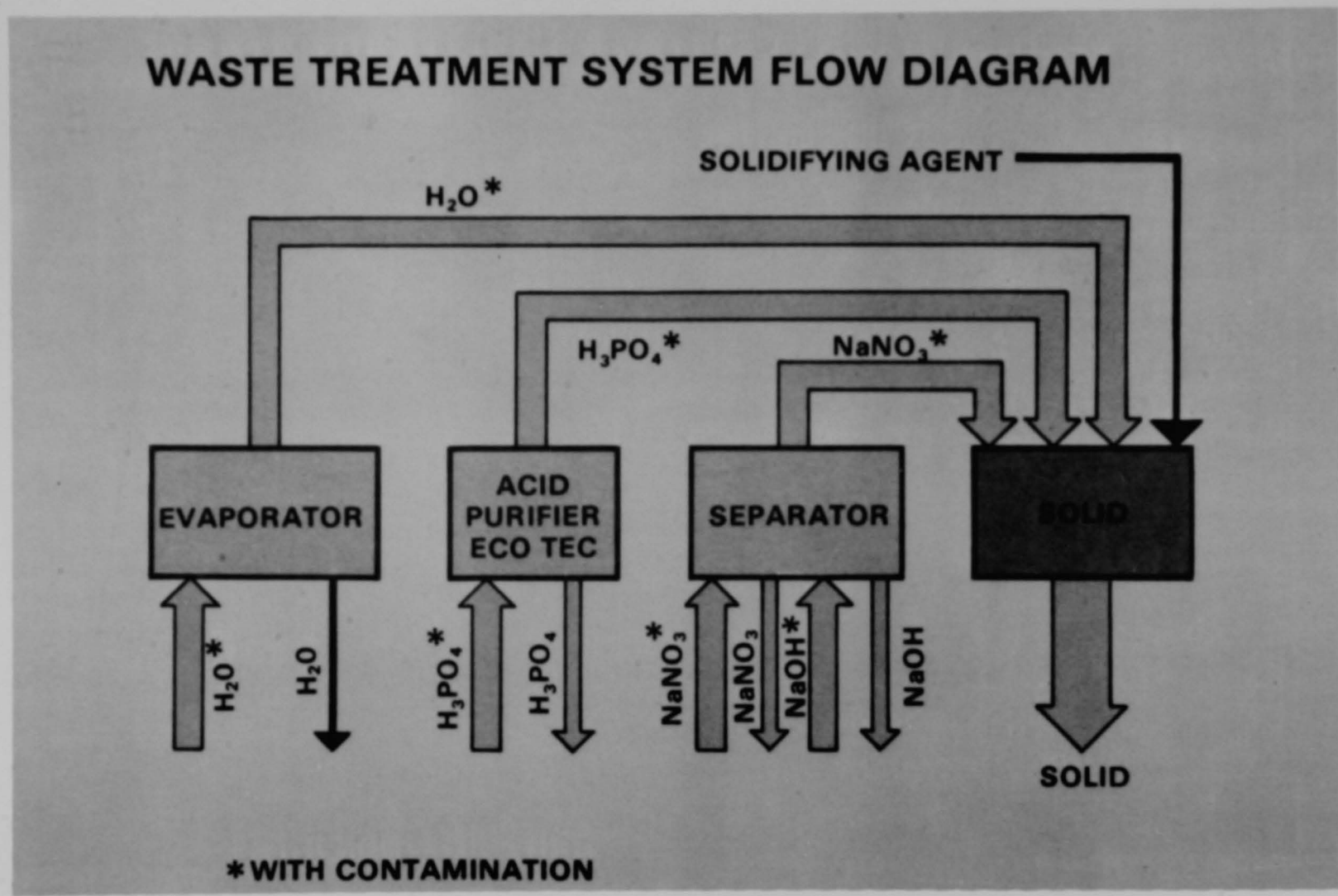
FLOW DIAGRAM FOR VIBRATORY FINISHING SYSTEM



SLIDE 31


RANKLEEN

SLIDE 33


WASTE TREATMENT SYSTEM FLOW DIAGRAM
SOLIDIFYING AGENT
H_2O*
H_3PO_4*
$NaNO_3$*
EVAPORATOR
ACID PURIFIER ECO TEC
SEPARATOR
SOLID
H_2O*
H_2O
H_3PO_4*
H_3PO_4
$NaNO_3$*
$NaNO_3$
$NaOH$*
$NaOH$
SOLID
*WITH CONTAMINATION

Session N

TMI-2 CONTAINMENT DECONTAMINATION PLANS

Frank McDougall

Bechtel

Because of other priorities such as containment entry and the purge program and engineering for recovery facilities, containment decontamination is only in the preliminary planning stages. However, in the Bechtel initial study effort, a planning study was completed for containment decontamination. It is my intention today to summarize that study with emphasis on the remote decontamination techniques and hopefully to obtain some feedback from you later this week to help us evaluate remote decon and other methods of gross decontamination.

First let me emphasize again the preliminary nature and the fact that we have not yet factored in the experience of the Auxiliary Building and the Fuel Handling Building decon efforts. It's essential at this time, because of the lack of specific knowledge of what's inside the containment, that all of the planning be flexible and contain as many options as possible. As more information is gathered some of these options can be closed out and the plans can become firm. The cost of the various options and the man rem assessments associated with all of them are two of the primary factors to be evaluated. However, I won't be covering them at all today.

To put containment decontamination in the perspective of the overall recovery schedule, we are now, as I mentioned, in the preliminary planning stage. Containment reentry is the next important phase because of what information we hope to gain. It's going to be very important to gain information on radiation mapping of the containment to identify hot spots and any residual damage that may have occurred. It's also going to be very important to gain information as to the chemical nature of the contamination deposition. Hopefully this will provide more specific input to decision making on the use of remote decon and

other options later. After this information is obtained and evaluated, some final decontamination plans can be made. Then obviously the next step before actual decontamination proceeds would be to process the water in the containment basement.

I characterize containment decontamination into three phases or three levels. First, there's gross decontamination for which we have the remote decon option and gross manual decon which has to be done whether or not you use the remote decon option, only to different degrees. Then, there's local hands-on decon and I don't really know how tc separate the terms except that I tend to think of local hands-on decon as more people closer to the work as opposed to gross decon being fewer people further away and using different techniques. And finally there is special equipment component decon. I won't be covering much today on the latter two phases simply becuase they have been covered before and also because we've done less detailed thinking about them at this point in time.

Looking now more specifically at remote decontamination; it is an option for the gross decontamination level of detail. The objectives would be to reduce the likelihood of significant personnel contamination and to reduce the general radiation levels in the containment to allow longer personnel stay times. The concept itself involves utilizing the existing containment spray system. In case some of you don't know what that is, in every pressurized water reactor containment there is a containment spray system which probably consists of at least two loops of spray headers located near the containment dome which are designed primarily to reduce pressure and radio iodine levels following an accident. So, it is an existing system, capable of delivering 1,500 gallons per minute through over a hundred spray nozzles near the dome.

Four basic concepts have been considered in the evaluation. The use of deionized water flushes (using processed water) is recommended as a first concept primarily because it's use would avoid combining high specific activity with off-normal radwaste processing. After, or as part of, the evaluation of the effectiveness of the deionized water flush we would consider using a detergent solution flush similar to Radiac wash and, this is not necessarily in sequence, but followed by some steam condensation cycles. The remote concept using the containment sprays would be relatively effective in terms of coverage of area above the operating deck because that is the way the system was designed. There are penetrations between the floor levels which were designed to allow water to flow down to the containment sump level. However, obviously below the operating deck much of the equipment would not be sprayed directly and probably not a significant amount of the equipment surface area would be even washed by the sprayed water. So the potential effectiveness is something that needs to be evaluated in more detail. It's even possible, for example, that you would just relocate some of the contamination from the operating deck level to a level below. As a last resort, and after evaluation of other flushes, the possibility exists that chemical solutions could be used. Many of these have been evaluated; they're not preferred. They're not preferred on this kind of level of processing simply because of the off-normal radwaste problems.

Since the Bechtel Initial Planning Study was completed several months ago, estimates of the radiation levels in the containment have decreased significantly. For this reason remote decon has become less desirable. It does have the inherent potential advantage of reducing the over-all man rem in the recovery effort. Because of this potential it can't be fully discounted or eliminated at this time. Because of the difficulty in estimating it's effectiveness

this potential advantage could be, more or less, an imagined advantage, whereas there are some real and numerous disadvantages. However, it's too early to tell if the disadvantages are overriding. So, additional evaluations of remote decontamination are being considered.

At this time, I'll summarize some of the advantages and disadvantages. Basically it is felt that the use of remote decontamination will extend the recovery schedule. If remote flushes of the containment are used then the waste from each flush must be processed, effectiveness evaluated, the next step decided, and an iterative process continued. These steps are all probably going to end up being on the critical path to recovery so there is an inherent disadvantage. The volume of radwaste liquid is expected to be much greater using remote decontamination even if recycled water is used. The original remote decontamination concept involved estimates near 250,000 gallons per flush. The potential use of any detergents and/or chemicals would cause the need for more or larger capacity sophisticated radwaste processing (or again an extended schedule). Use of remote decon via containment sprays would not allow other activities to be ongoing concurrentlv in the containment. Support systems would have to be designed, purchased and installed thus adding to the cost of recovery. Finally, the effectiveness of this method will probably never be truly quantified until it's done.

Evaluating the potential for flushing the containment with smaller volumes, performing tests in laboratories, and possibly devising a method for insitu testing in the containment itself are now being considered as ways to reduce the negative impact of remote decon.

Also being considered is the evaluation, in more detail, of the effective coverage of the spray system and the wash down. So at this time we are

embarking on further evaluating these factors and that's going to be one of the main thrusts of one of the workshop sessions later this week.

As I mentioned, remote decon is just an option and even if it is used, gross decontamination of the containment by manual means will still be required. One method preferred for doing this is a detergent solution wash down using a mild, chloride-free, detergent through nozzleshooked up to the existing fire protection system in the containment. This technique may allow personnel to keep away at distances up to 50 feet. It may allow the decontamination of the polar crane without the necessity of scaffolding and it would deliver large quantities of water directly aimed at the hot spots or at the greatest areas of loose contamination.

Another method for performing manual gross decon that seems to be preferred, compared to the use of hydro lasers, would be the use of saturated steam at low pressure with, again, a mild, chloride-free, detergent using hand held steam nozzles. This doesn't mean, of course, that the other techniques we've heard about today, such as water lances, flared nozzles, fire hose nozzles, etc. would not be used but in our initial evaluations these are the techniques that were deemed preferable.

It's been mentioned several times today that there are expected to be many specific hot spots in the containment; for example, the block wall structure around the elevator-stairwell shaft, which is uncoated, is expected to be a high source of contamination. The polar crane might well be another. Some of these areas will have to be accessed early in the containment decon program in a manner that will allow decontamination from the top of the containment downward. Some areas because of priority may just be shielded away so that more meaningful work can be done.

Other techniques evaluated to date include the use of local chemical decontamination techniques, because it is felt strongly by some that the right application of a chemical solution minimizes contact time and potentially man rem. However, strong reagents such as strong caustics and acids are not at all desirable because of the potential corrosion to the NSS system and again because of off-normal radwaste processing considerations. Mechanical decontamination techniques are also being considered but appear not to be desirable unless absolutely necessary. In that case, impact tools were considered preferable over abrasive tools due to the potential airborne generation problem.

We really don't have a good idea how the coatings have held. Coatings could be decomposing because of the radiation exposure, and as they decompose, create craters and trap contamination and possibly would then have to be removed. So needle guns and such methods to remove the coatings used in conjunction with a wet-dry vacuum to remove the chips is a recommended method.

The containment decontamination program is so extensive that, first we're trying to get a good handle on where it fits in the overall recovery program, what the major options are and how the right evaluations are to be performed to close those options or to clarify them, and then proceed.

We're talking about purchasing massive amounts of equipment to do the job. We're not just talking about a few hundred sets of Anti-C's, but about hundreds of thousands of sets that have been estimated to date that would be required. So it's essential in the preliminary planning stages to at least identify the techniques that may be used, the special requirements that those techniques would have on equipment and personnel needs so that

these things can be scoped out and plans can be made for purchasing and using them. So as I mentioned earlier, I hope that this august body can function in November to provide us with some good input later this week. Thank you - any questions?

Question:

Has the option of doing nothing been looked at and if so what are the economics of that option?

Answer:

The options of doing no containment decontamination? I'm sure they have or I think they have because I've heard discussions but I'm not aware of them specifically; I can't answer the question directly.

Question:

It seems to me that the one of the most important parts of the decontamination program is how are you going to handle your waste. What is being done to make a decision on how the waste is going to be handled. I think I'm talking about a chicken and the egg approach here.

Answer:

Your point is well taken; the criterion that says everything we consider in containment decontamination should minimize either volume of radwaste or off-normal radwaste processing certainly is the governing criterion at this point in time probably until it clarifies. Now studies are being done to estimate for all of the various options how much radwaste will be generated, what is the trade off in radwaste generation, even in man rem, potential man rem savings from decon.

Question:

What's the possibility of running into uranium oxide in the bottom of that containment vessel?

Another voice:

It hasn't been sampled.

Frank:

I can't answer that question.

Another voice:

It would have a lot to do with how you handle your radwaste.

Question:

Frank, one of the points you talked about right in the beginning which we don't know the answer to and we'd have to find out as soon as people go into the containment. We don't know how that stuff has deposited itself on the containment walls and surfaces and it doesn't do us any good to speculate on how to remove this stuff with all this remote decontamination if it has in fact gone through some ion exchange with the paint and it's going to stay there no matter how many times we spray, so we can't talk about that and, therefore, we can't talk about how we're going to do our waste processing until we know how it's sticking itself on the surface.

Frank:

Well we can't ignore those topics either, we have to at this time plan for many options.

Questioner:

I realize that but we've got to get into the containment and take some samples and find out what that stuff has done. It's going to be there a good year before we start.

Frank:

Well maybe I didn't make the point strong enough that the initial entry or entries and the data gathered from those will to a large extent dictate the direct path of containment decon, especially remote decons.

Question:

What kind of chemical solution for flushing have you considered?

Frank:

Well there was in the planning study a list of about 10 solutions that were evaluated and listed in order of preference; preference being weak to mild to strong.

Question:

Were they dilute solutions or not?

Frank:

Let me just read you some of them, okay. Morphaline which is a mild base.

Question:

Pure solution or mixed with the water?

Another voice:

Solution and water.

Frank:

Yes, all of these are delivered with water through the containment spray system. Some of the others are disodium or trisodium phosphate, sodium hydroxide, boric acid, hydrogen peroxide, peracetic acid; these are some that we evaluated and that's generally in order of preference.

Question:

In the TV shots of the internals of the containment there appeared that there was some condensation action occuring inside, do you think that's the primary solution for remote decon to the walls?

Frank:

That's a good question; it's been debated and I've heard it mentioned that that's probably an advantage and that might be worth enhancing. It's also possible that just that little bit of moisture has allowed for some of the contamination to creep into crevices and that type of thing and be harder to remove later.

Session O

SL-1 RECOVERY EXPERIENCE

C. Wayne Bills

EG&G Idaho

January 3, 1961, or TMI minus 18 years and I have been asked to reach back all those years and talk about some of the activities and things we've learned that were relevant to the decontamination and recovery here. I'm sure that my recall is not going to be nearly as complete as it was 18 years ago. It might be useful, however, for the people who do not have too much knowledge of that accident to emphasis some of the dissimilarities with respect to the size of the containment, the isolation of the facility, the fission product inventory involved in the three recovery phases.

We had three phases to our recovery, the first phase or the emergency phase involved recovering the three bodies. The second phase had to do with determining the nuclear status of the reactor core. We had to find out whether it could go critical again, whether there was water in the vessel and so forth. Then the third phase was gathering and evaluating the accident data, removing the hardware and building and decontaminating the area and renovating the site.

The SL 1 site (Figure 1) was very remote in the desert of Idaho. The reactor was in a corrugated tin structure and it was only a confinement building not a containment building. The reactor floor being some 20-25 feet above ground and the access was through a freight door on the back side of that building. The other buildings were just support buildings.

The entry was through the door up on the side of the reactor building and, therefore, much of our operation had to be done very much on the blind side with remote operations. We used a cherry picker with multiple booms to do most of the recovery operations in the early stages. The radiation

involved initially were in excess of 1,000 R per hour and that was gamma plus whatever beta recorded by the instruments. Although at that range they were estimated values. In phase 2 we got down into the range of 500 R per hour instruments and worked in 200 to 300 R per hour fields, and in the final phase went from about 200 R per hour down to where we were using the 1 to 10 R per hour range when we had the reactor and its head fairly well shielded. This was in the last phases before we lifted the reactor vessel out and took it to a hot shop some 28 miles away. I won't speak too much about the beta ratios although we did have about a 15 to 1 ratio there and it did cause us some concern, but not the kind of problems I heard about yesterday. The high gamma radiation fields was caused from about 5% of the core being washed up and outside of the reactor vessel and it was estimated that there was about a half a million curies involved in the core at the time of the accident. And as we come here seven months after this TMI event probably many things that we'll be mentioning here today were touched on in yesterday's discussion, and so what we will be talking about today may only bring reinforcement or rejection to some of those ideas. And it may be just the nucleus of the idea that's planted today rather than what's said here that might be of value to the ongoing operations.

While the operations were going on we did have a radiation shield set up for the health physicist and the people who were not directly involved in that operation could get behind a shield and remain out of the direct shine from the radiation off the top level (Figure 2). We knew from our experience from SL 1 that beta exposure would be limiting at TMI. The fission products have undergone process in their evolution and disbursion around the building, we found that their physical and chemical properties in part determined where we found them. Considering there was quite a difference in the

spatial distribution on equipment and buildings, we found that as we cleaned up some of the fission products migrated back to cleaned up areas. I thought perhaps out of the data yesterday maybe more can be learned about the physical and chemical properties and what's going on but it may take additional samples out of there to get you accurate enough data to try to make some sense and project what could be helpful in decontamination.

In the SL 1 area it took multiple entries really to establish this spatial distribution. We had our entries limited to from 30 seconds to 1 or 2 minutes, and so it took several entries to really find out where the fission products were located and where the fuel was located. I think it should certainly be the goal here to solve that early on because we wasted a lot of time trying to work around things while not having the spatial distribution well pinned down. I was pleased to note about the collimated germanium detector yesterday. Figure 3 shows a little data we got out of a pin-hole camera. The pin-hole camera is a box with a pin-hole in it. We shielded three sides so it was unidirectional and put a piece of visual film in it, and also a piece of gamma-sensitive film. We exposed the visual film for about an hour and then we closed the hole and exposed the gamma film for another 24 hours. This picture is an artist's concept made about five months after the accident showing the high radiation zones up in the fan loft or attic of that building. It was a year later that we found out that this was pretty accurate and we did find those pieces up there and they did constitute a radiation source that was well above the operating floor. So we should have probably given this more credibility than we did at the time; it was pretty rough data working with a pin-hole camera and I'm sure there must be state of the art directional gamma detectors and so forth today which can be of value here.

Still photography certainly played a major role in our training for the reentry crews at SL 1. Figure 4 is one of the first three pictures that was taken and it gave us immediate evidence that the shielding above the reactor had given way and all those metal pellets that you see around are punchings that had been used for shielding on top of the reactor. They turned into shrapnel and gave us a great deal of problems during recovery because we found them everywhere. I know that the television today is much better than what we worked with in 1961; we didn't have an instant replay, but we found that the radiation and lighting conditions were difficult to set up for television. The time to set it up, the picture resolution, and so forth made it somewhat more difficult to work with than still or motion picture that we used.

One of the television cameras was rigged on the crane or boom out of the cherry picker. We had a light hanging below the camera and we had to go fishing down the holes looking for evidence inside the reactor to find out whether we had water in the vessel during the second phase of the recovery.

The photographers were always getting the highest radiation exposures and we had to rotate them frequently because they spent too much time trying to get a good picture instead of just getting working pictures. (A short 2-minute movie.) What you'll see here is trying to penetrate - there goes the light down one of the nozzles as it's swinging down with the camera above it, and occasionally you'll get bright flashes of core; you can see some of the spray rings and so forth hanging in the way. You'd always like to see more, you're always vulnerable, you never have enough light. We used high quality quartz lamps. You can see some fuel and end boxes off of some fuel down there; you can see the control rod crushed up against the side there. That one flash showed you a flattened spray ring at the top which began to give us evidence of the 10,000 PSI type of pressures we had in the reactor.

It takes a lot of effort to man that type of a short exposure on a remote reentry run. We did find that the optics and the film was much more durable than we had thought and that the browning of optics and the fogging of film were not a continuous problem with reasonable care and shielding. And also it may be important to have before and after photos as things are removed from the operating floor or any one of your three levels.

We certainly had a lot of dry runs in the recovery operation. We found that models and mock-ups were invaluable for our training and Figure 5 shows a mock-up using one of our fire towers out at the fire training station where we put things at the proper level and then used the cherry picker to reach in and drop the camera down through holes in plywood to simulate the reactor top. You will notice that the cherry picker has a lot of lead on the front of it, that's shielding that was put on to protect the operator who had to move in very close to the building.

Again, up on the fire tower we did have a mock-up as you can see in Figure 6 the simulated nozzles and you can see that the camera has a guide on the front of it so that it would guide down the holes, and we only had one or two holes that were accessible, the others had rods and racks broken off in them, so out of the nine holes, we probably had entry only through two holes. Clear at the top you will see a movie camera that is shielded in a plywood box with just the shielding to protect the film with the optics being exposed through a hole in the bottom.

Figure 7 shows the type and quality of photo that we were able to get with that type of apparatus. Again you will see the lead slugs of metal, the punchings, were thrown all over. This was our first opportunity to see that the plugs had been

thrown out and that some of the holes did have things in them that limited the access to looking down in at the core. Certainly, step by step procedures are necessary; this is to insure that the equipment works, that the operator's questions get worked out ahead of time, that there's nothing in the way and that the timing is developed to minimize the time personnel are to be involved in exposed conditions. I find that just in commenting on yesterday's 15 minute planned first entry, it's going to be difficult to plan 15 minutes let alone an hour to make it very efficient. You should have a back-up plan in the event the first mission has to be aborted. Adhoc deviations from the plan except for life saving actions should not be allowed and even there the broad guide lines should be laid out for the people from the beginning.

I think, as noted yesterday, lights and lighting are probably going to be more trouble than cameras and I've already commented on that. We've found out that due to the explosion involved, things were not understood as we saw them and they needed to be confirmed by modeling to get some answers. I would guess that you will find that you will need to answer some things before decontamination operations go ahead. We found that the nozzles were bulged and we had to do core modeling and testing at the Army Ballistics Research Lab in Aberdeen Proving Grounds to determine the pressure impulses that were involved.

We did not know the cause of the accident until many months after the accident and, therefore, we had to very carefully take data even though it involved working in radiation fields before we did some of the decontamination work.

Figure 8 is a picture of what we found inside the pressure vessel. The streaking down the sides seen in this picture is probably from the boric acid solution that had been dumped in

there. You can see the flattened spray ring, that upper spray ring was up at the top. Some of those things lead us to look for the high pressure pulse that we got up near the top of the vessel. The water was down 2 feet at the time of the accident, so it was a free moving piston and when the nuclear explosion went off down in the bottom of the vessel it had about 2 feet of drive before it hit the top head.

As you bring things out of the building and perhaps if they've been misplaced by water, shock or other forces, you may want to reconstruct their location at some cold lay down area after the pieces have been decontaminated to better understand the distortion or displacement. You will undoubtedly find certain things out of place inside the containment and in the primary system later.

We've learned a lesson that will probably be more intensive today than it was during the time of SL 1 and that is <u>not to make waste.</u> I think in terms of a system to minimize the decontamination solution, the solid waste, and the scrap, we found that we could use liners for boxes, we could use liners in our casks and we could use other things on transport vehicles so that when we got through we didn't have to decontaminate the whole cask or decontaminate the whole vehicle. So that looking for a secondary box or something to put things in can be very helpful. It seems to me that yesterday I heard some of these things being talked about. Particularly the sprayed on removable paint which can be cleaned up quite easily. We've found that steam cleaning was very effective and that certainly minimizes the volume of liquids. We recycled a lot of laundry even to the extent that we used very low level contaminated clothing. We discharged right after the entries, about 10% of the coveralls, about 15% of the head covers, about 15% of the shoe covers and about 5% of the rubber boots. Everything else was recycled. There certainly must

be a lot of up date on disposable clothing. I know that John Johnson from the direct maintenance in the Idaho chemical processing plant will add to this data later.

Let me get back to the personnel exposures for a minute. We found out that the shield shown in Figure 9 is the kind that we needed to do some of the cutting. It was shielded box with a lead shielded window with glove ports and we could put a craftsman in there with a torch and do some cutting to get access to the building and to cut away some of the things that had to be taken out of the way in order to gain access for some of the remote operations. This was swung from the boom of a crane.

Figure 10 shows a welder doing the welding operation. We did mockup and training on all of these beforehand. Some modification like this might be used if you have real high sources as a result of the gross decontamination where it might have seeped into pipe insulation or somewhere you may have to get to it and still protect your craftsman.

Figure 11 shows a picture of that cherry picker with the lead added to it. That's a lot of weight that was never intended for that type of vehicle. You have to be concerned about that, but you also have to be concerned about things like that little curved part at the top to make sure to get shadow protection for your driver. There was nothing better than the innovative gadgeteer that figured out how to do some of these tasks. In our case we had a group of construction workers that happened to be based on site, H.K. Furgeson Company, and these practical fellows really translated all the technical and engineering jargon into things like special booms, mockup shields and without a lot of paper work. One example is the movie camera box shown in Figure 12.

We ended up putting cloth booties on the cherry picker tires; we found out we had to decontaminate that cherry picker about every time we turned around so we made some cloth booties and we'd run it in and then we'd strip those off and a lot of the decontamination of the equipment was avoided - like using gloves. You might find that there are different ways of protecting your equipment from having to be decontaminated everytime you come out of the building. We had to work, as I mentioned, predominately outdoors and, therefore, not in a very controlled environment. In fact it was 10 below zero all the month of January; that had it's advantages and disadvantages.

While the people are being exposed from being right inside the source, you have this elusive beta gamma field dosimetry to deal with, we found out that we not only used badges from top to bottom on a person but around the individual to measure the integrated dose. We had some standard systems for putting badges on and then we augmented this for the job that was in the specific procedure that was going on. Now the one I'll mention here is using a vacuum cleaner, for example. As you pick up the material and the material was starting out to be at the 200 R per hour level, it was concentrated during the flow up the hose so that the exposure to the hands, the arms and across the hip line was quite excessive, and we could obviously tell a right-handed person from a left-handed person by the exposure of the film around their middle. Also when contamination was being collected by vacuuming techniques, we found out we piled up 2,000 R per hour source right behind our rear end; we had forgotten to shield the tank for that at the moment. So you have to provide for shielding of your collection buckets or your collection tank. One of the things I recall, one of my personal experiences up in the reactor building, was to clean up lead shot and we had used lead shot

to shield the reactor head but in the process some of the bags had gotten snagged. If you think sodium hydroxide is slick on the floor, you ought to put ball bearing on it and try to walk around on it. My job was to go up there in a minute and a half and clean up as much of that lead shot as I could.....in full protective clothing...and then move the bucket over to the door where it could be picked up and taken out of the building. Working like mad for a minute and a half, I turned around the bucket weighed 400 pounds and it wasn't going to go anywhere. So you get some rather quick experiences that way.

From your description of your entry, I'd be a little concerned of having that inside airlock door closed if I were in there; I might make a hole in it coming out. We had a stairway that went up the side of this building and it rattled pretty good so we got a big club and we'd hit that and it would jar the whole building. The health physicist would stand down at the bottom and he would hit that once with about 10 seconds to go and then he would hit it a lot and that was when you were supposed to come out of there; by the monitor staying outside, we tried to minimize the exposure to our health physics people. There must be some sophisticated things like that; I'd use the wireless but I think I'd have a big gong or something right inside to use as a back up system.

I though one of the things we ought to quickly talk about is challenging the obvious. It turned out that we had a lot of things that happened to us that should have been obvious but we rather ignored them at the moment. One I recall and I'll move ahead here rapidly. We took some pictures down through one of the nozzles looking for water and then we had to send the film off to Salt Lake City a couple hundred miles

away and it was dictated that we send that film on to Washington for review over the weekend and, therefore, with the time difference, Washington ended up looking at it two hours before we did. On Monday morning they called out and said there was water because they saw bubbles then we ran the film and we saw bubbles too. About two days later we found out that we were running the film backwards because it had been spooled backwards down at the processor and instead of bubbles, when the light went in, it had knocked some of the boric acid flakes in and when the flakes floated down in the light it looked very much like bubbles coming up. The vessel was dry and we should have really challenged ourselves to look at that. Knowing that it had been opened in the after heat of the accident there shouldn't have been any water in there.

The first thing we had to do was face up to the fact that we had the wrong film on the market. We had to get the candid answer out so people understood what was going on. You always took the brunt of working in a fish bowl, and everybody is saying how stupid could you be to run that film backward.

I think I'll just talk for a moment about two or three things that I didn't hear anything about yesterday and one was good log entries and records are essential to those entries. I didn't hear anything said about the debriefing, but you must debrief those entry crews and make that record as soon as possible because the memory can be very short and lose important details in just a matter of hours. You can pick volunteers for those entry crews that will really pay dividends in experience. We used management, public relations people, firemen, security guards, administrative personnel and much of the fear and mystery was removed as these people experienced the care taken on the job. They gained and we gained a lot of grapevine PR by them telling their co-workers how it really

is and taking some of the mystery out of it. I didn't hear anything about documentary movies. Certainly it's much easier to document, with some movies, as you go along than to go along after and try to reconstruct, particularly in our case after the building was gone, try to reconstruct some of the first entries. The idea of having an independent review of your procedures and justifications before the major entries is very important. While this is very valuable, it can be very frustrating if it gets on the critical path. I heard yesterday about some of the reentry reviews and approvals. Certainly that has to be far in advance so that it isn't right in the middle of your operation. I think with this I'll just close knowing that there's probably a lot of questions and in the interest of time would be available to being button-holed somewhere and try to recall some of the things even after 18 years. Thank you.

FIGURE 1

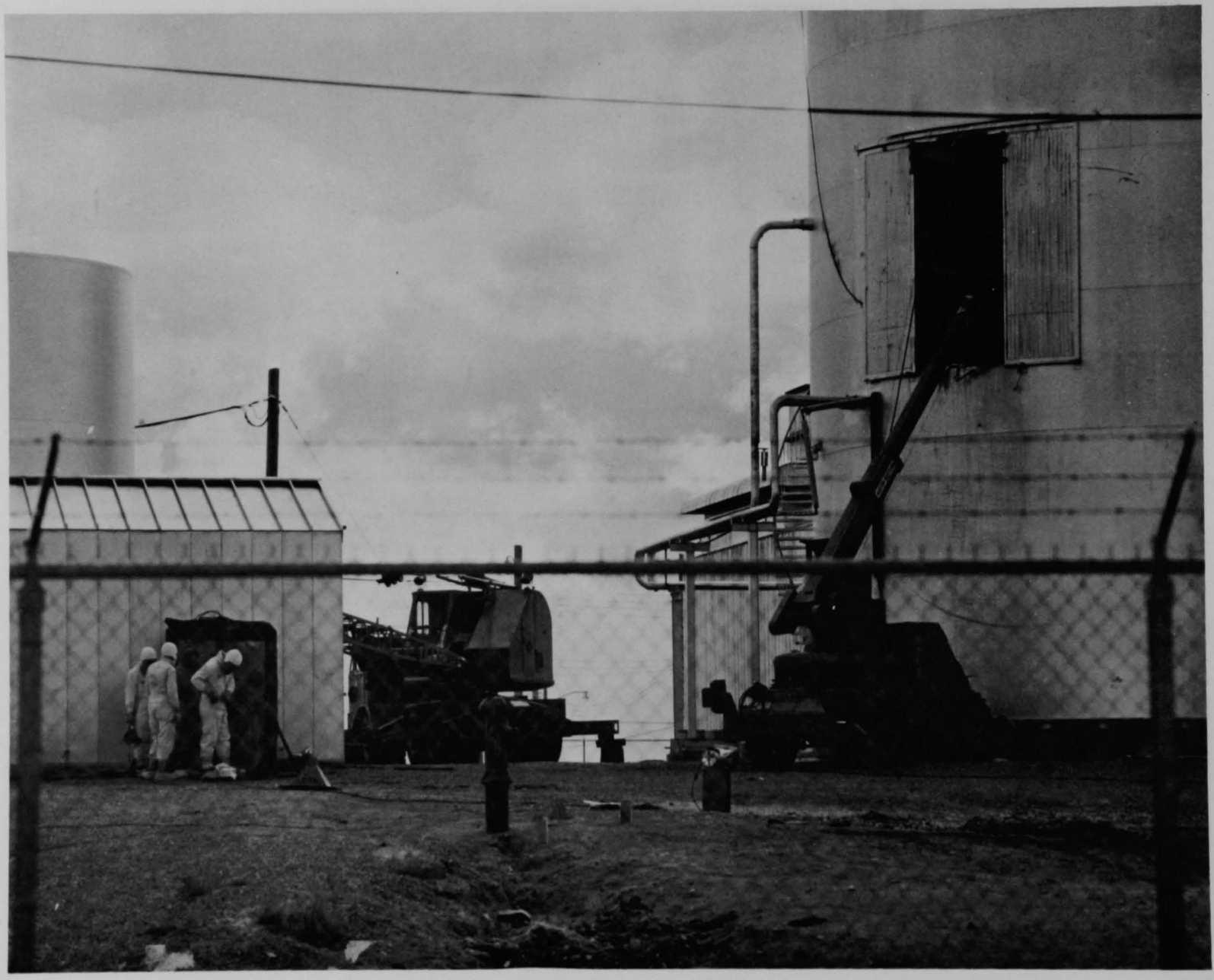


FIGURE 2

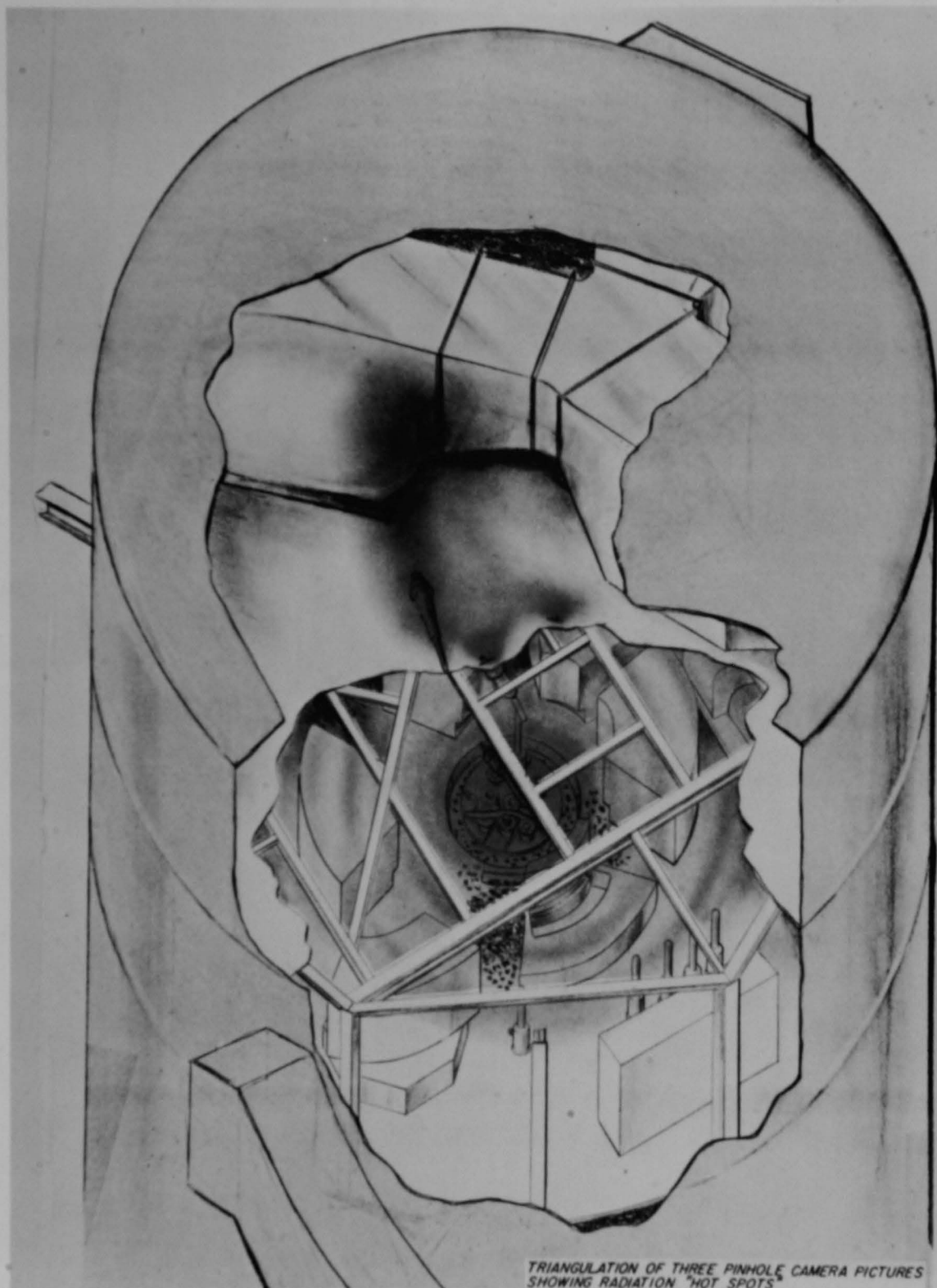

TRIANGULATION OF THREE PINHOLE CAMERA PICTURES
SHOWING RADIATION "HOT SPOTS"

FIGURE 3

FIGURE 4

FIGURE 5

FIGURE 12

Session P

CANADIAN DECONTAMINATION EXPERIENCE

J.H. Collins
J.W. Logie

Chalk River Nuclear Laboratories

The Canadian experience starts on December 12, 1952 at Chalk River Nuclear Laboratories. The NRX Reactor is a 30 megawatt, heavy water moderated light water cooled engineering test reactor. During a series of critical height measurements, a power surge occurred due to design errors, mechanical failure and, of course, personnel errors. This is the same kind of thread that runs through the majority of nuclear incidents. The power surge damaged 22 fuel rods, and leaked highly contaminated light water coolant into the lower header room beneath the reactor.

The first thing that occurred was that personnel heard a rumble from the reactor, reported to the control room this fact and also that they saw water bubbling up over the top of the reactor. Water was also found pouring out of the bottom of the reactor and into the lower header room. The initial flood rate into the lower header room was something in the order of 1,500 liters per minute and personnel went down immediately to get a quick sample. The leakage was light water with a bit of tritium involved. The first problem to be overcome of course was the flooding of the lower basements. A series of operational moves reduced this flood rate from 1,500 to 64 liters per minute over the two week period following the incident. Accumulating water was stored in outside temporary storage tanks while a pipe line was hurriedly constructed to a clay-sand site some 2 kilometers inland from the Ottawa River. This is a far cry from what we hear about TMI,

of course, but we had lots of room to play with and it was early in the game and of course things have changed since then.

In 1973, a very intensive survey was carried out of the ground water leading to a small stream and lake system very close to where we stored this water, and there was no evidence of any radionuclide entering the small lake and stream system from this source. This situation is being continuously monitored and to this time no evidence of such activity has been seen.

As a result of the incident the radiation level was 100 mr/hr at the entrance to the control room. The main floor levels varied from 200 to 1000 mr/hr and beneath the reactor at waist level fields of approximately 10 R/hr were found. Loose swiping showed contamination levels of the order of 50 mr/hr. The flow of light water was not completely turned off until February 3rd of the following year, and decontamination efforts in the building and particularly in the bottom header room were effectively restricted until this time. A period of trial and error followed with the result that we did a flush with light water initially and pumped the drainings through the existing piping system to the inland storage area. We flushed again using a high velocity stream of hot water with or without detergents depending upon the surface, and at the same time we moved equipment to allow decontamination.

Larger components were removed to a large room beneath the reactor on the main floor where they were decontaminated. Again, the flush solutions were sent to the disposal area. Before we got at the lower header room itself, the water covered about half a bank of instruments, very similar to the TMI situation. We were able to reclaim the instruments by removing them from their sites and sending them, properly encased in plastic, of course, to a site decontamination center. We flushed once more with light water and then we surveyed for hot spots which we either removed by elbow grease or we shielded. In particular, the bare concrete surfaces of the original reactor structure gave us a fit. In some areas there was as much as 200 R/hr in the concrete; we chipped, we ground, we sandblasted, all of which resulted in difficulties with cleaning up the grit that resulted. We flame primed and once we got down to the aggregate we found we just couldn't go any further. We then covered what was remaining with a layer of 15 centimeters of fresh concrete. At that point we had achieved a reduction to about 20 mr/hr. We found that stainless steel responded better to wiping with acid or detergent-dampened rags than to normal flushing and scrubbing.

At the end of the procedure we were able to remove the NRX Calandria from its site. We hauled it out and put it in a canvas bag that was on an upturned skid; the skid was then turned to the horizontal and we towed the unit to the disposal area where it was buried. Planning, mock up work, intensive training, all these things

put together gave us a fairly quick and safe procedure. From the time of first hooking on to the time the unit was going out of the building was something on the order of 30 minutes.

Perhaps we can leave the NRX initial episode there and move on in the interest of time to NRU 1958 which I've entitled "burning fuel on top of reactor." The NRU reactor is a heavy water moderated, heavy water cooled engineering test facility that went critical in late '57 and had operated at power levels up to 200 megawatts thermal prior to May 23, 1958. On that day several linear rate trips occurred, the last of which was coincident with very high gaseous fission product activity in the heavy water system. There were other indications that were later related to a high pressure transient in the core, presumably due to the violent failure of a natural uranium metal fuel rod. Each rod had in the past been connected to an individual GFP monitoring system, but instrumentation had been desensitized by previous cladding failures at the time of the incident, and was saturated by the burst. Three suspect fuel rod defects were noted by examining radiation levels on top of the reactor.

The fuel rod changing flask in NRU is an essentially self-contained machine that needs only electricity and air. When a reactor site has been opened up, the flask is positioned and a flexible extension, called the snout, goes into the top of the position, an extractor comes down, unlocks the rod, and then locks onto a gripper in the

rod itself. The rod is then pulled up at a programmed speed. The flask is equipped with two barrels so that you can interchange barrels, once you have the first rod up, and install the second one. The first of the suspect rods was pulled out without difficulty, but the second one wasn't. An untried procedure was then used. Although this procedure should have worked, in this case it did not because one step was omitted, and it resulted eventually in a three foot section from the middle of the fuel rod being deposited in a flask maintenance pit, where it started to burn. When they moved the flask off the top of the reactor, operating personnel were desperate to get to a station where a light water hose could be attached to the flask to cool the bit of fuel remaining in the flask itself. Another piece of uranium was later seen to have burned on top of the reactor. We were able to restrict the uptake of radiation to 5 Rem individual maximum during this first part of this episode Personnel had put on particulate respirators prior to removing the rod so that gross inhalation was not a problem. The flask was quickly parked over an elevator shaft leading to long rod bays and was later connected to the shaft so that all the water from the emergency hose still cooling what was left in the flask went to the rod bays.

The first thing people did, wearing full face Army respirators, was to come up the stairwell at the end of the building with buckets of sand and cover the burning bit of fuel in the center maintenance

pit. When you are wearing respirator equipment, hard work is extremely difficult. I would very strongly advise those who get into this kind of a situation to be very aware of that. We had people half collapsing on the stairwell. The adrenalin bursting through your system is a bit more than what the oxygen intake can keep up with at time, and you can get into some very serious situations if you are not very careful. Radiation fields within the maintenance pit just after the incident were in the order of 50,000 R/hr; that's an estimate of course. Vertical surfaces in the building, 5 to 400 mr/hr; horizontal surfaces, 200 to 2500 mr/hr, excluding the top of the reactor. On top of the reactor anywhere from 10 to 1,000 R/hr fields existed, and the reactor hall air activity some 12 days after the accident was 200,000 DPM/m^3.

The decontamination effort was carried out in two initial phases. Overnight, a special wooden pallet was constructed and was lowered into the center pit by a crane. Operators using long handled rakes, very long handled rakes, then manuevered the bit of uranium onto the pallet, covered it with more sand, and got out of the way. The pallet was then removed from the pit by the crane into a waiting heavily shielded float. The main crane operator was changed in two minute intervals to restrict radiation uptake.

The second phase on May 25th concerned the removal of much of the sand from the center maintenance pit. Personnel not normally

involved in radiation work were hurriedly brought in from the yacht club and the golf course and put to work in teams of two, again with long handled rakes. Sand was scooped into buckets which were then moved to an elevator at the end of the building and eventually wound up in the waste disposal area. The highest individual radiation doses of the incident were received during this episode, ranging from 5 to 19 R. The comment here I guess is that perhaps this fast reaction should have been delayed to allow further decay.

The next few days were spent in controlling the spread of contamination from the building and in organizing the main decontamination effort. Fields from the main pit by June 6th had been reduced to 1 R per hour by use of properly shielded vacuum equipment. Armed Forces personnel from Camp Petawawa, a military base next door to Chalk River, were brought in, and they did the bulk of the swabbing and mopping. They were on hand from about June 2nd to July 7th, at which time radiation levels were well down. Walls and ceilings in awkward locations, however, had to be handled by commercial operators and steeple jacks. They were only brought in after the bulk of the heavy decontamination work had been carried out. We encountered a number of problems. The building ventilation system has been operating at the time of the incident and was very heavily contaminated. We had to go in through shafts and mop things out, and final cleaning was on a semi-permanent basis by circulating fresh air through the systems and through filters on the discharge points. Electronic equipment

was very sensitive and difficult to clean; dampened rags were found effective. Large quantities of what should have been recoverable equipment found their way into waste disposal bags; perhaps because of people being too cautious, perhaps because of a lack of planning on that particular point. At any rate, it was becoming very costly and we had to appoint qualified personnel to attend the disposal bags and sort recoverable equipment from the non-recoverable. This resulted in an enormous increase in workload on the decontamination staff in the laundry. Housekeeping services become very important in situations like this and a lot of planning has to go into the staffing of them, the material supply and the organization. It quickly became apparent that simple existing procedures, the kinds of things people do every day, could no longer be used, and written instructions were required to ensure not only minimum exposure to radiation and contamination, but to prevent recontamination in recently cleaned areas. I might mention here that in the first episode, the "NRX 52", a total 2,600 man rem were accumulated by 1100 people. In those days, the allowable for Canada was 15 Rem/year and we had one person who barely exceeded that level; things are different now, of course. In the second episode, the "NRU 58", approximately 700 Rem were accumulated by 800 personnel who worked in the building during the two months it took to clean it up and to return the reactor into proper operation.

The two other major areas that are perhaps worth mentioning

here are the 1970 NRX calandria replacement and the 1972 NRU vessel change. In 1970 the NRX reactor was again torn apart, the calandria was removed, and the reactor was reconstructed. When you decontaminate in situ and vacuum as much as possible, you have a fair chance of not contaminating areas external to the work site. Radiation accumulation during the NRX vessel change of 1970 was in the neighborhood of 200 R and this was over a period of about six months and certainly more than 250 people were involved.

The NRU vessel was changed in 1972 and 73. Long screws extending all the way to the top of the reactor were used to lower the bottom header and the vessel onto a carriage in the basement. The carriage was then moved to the other end of the basement and the vessel was picked up, using a lifting adapter, into a nylon shroud, put on a skid, and moved out to the disposal area. 360 man-rem were accumulated by the 300 workers in NRU over the two-year period involved.

A number of the following points are obvious; but they bear repeating as we have found, and a lot of these items, a majority of them, were generated following the'58 incident, which I'm sure is one of the reasons we had very little trouble in either 1970 or 1972.

1. It is noted that when working with nuclear reactors, force leaders, subordinates and the workers must all be dedicated to the job. Normal duties must be turned over to other members of their organization.

2. Only with the greatest reluctance should departure from the safest approach to a situation be considered. Planners should always search for the inherently safest procedure.

3. Deviations from authorized procedures must have leader approval. The leader must satisfy himself that the change is justified, bearing in mind the criterion of item two particularly.

4. Careful training of the entire operation should be done before embarking on any phase: this is the critical path structure. Full scale rehersals using mock-ups should be done on any difficult operation to uncover unforeseen problem areas and to write accurate procedures.

5. Where heavy decontamination is planned, a large non-nuclear body of personnel is required to spread the radiation load and to ensure that trained station staff do not receive exposures that would prevent them from carrying out tasks requiring their expertise. Intensive training programs are mandatory and instructions to the non-nuclear workers must be simple and explicit. Again, trained station staff are invaluable in this role.

6. Make-do tooling and make-shift operations simply are not good enough. Special tooling must be designed, built and proved out on mock-ups where hazardous, time consuming or difficult tasks are concerned. Obviously, special tooling requirements should be identified early in the planning phase in order to avoid needless hold up later on.

7. One item that we have found to be very important is to appreciate the slightly increased background radiation level that occurs during these kinds of operations. This is probably more important, in the long run, in establishing how long people can work in a given area, than a one time calculated exposure to a high field.

8. Continued review of progress is very valuable, as we found during weekly group meetings to discuss previous work, the following week's projected schedule and problems encountered. It is also important to get down in written form all work, ideas and problems that come up. The meetings can also be used as brainstorm sessions.

9. The mobile personnel decontamination center, equipped with electrical, water and drain connections, can be located close to the scene of action for initial decontamination work or it may replace the normal site center if the latter itself is badly contaminated.

I could possibly make one final point on beta - we assume a β/γ ratio of 100:1 in general down to about six inches from the source. Beyond that we run scared because things are not quite the way we'd like them when you get down very close to a high beta source.

Table I Selected Radiation Levels NRX 1952

Immediately Following Accident		
Control room door to top of reactor	100	mR/hr
Main floor around reactor	200-1000	mR/hr
Foot of stairs into basement	5-10	R/hr
Directly under reactor at waist level	10	R/hr
During Dismantling		
Bottom of second thermal shield	4	R/hr (c)
Top of third thermal shield	5	R/hr (c)
Bottom of third thermal shield	200	R/hr (c)
During Fuel Removal		
At 10 cm above rotatable lead shield	1-3	R/hr
Over certain holes at top of shield	20	R/hr
Certain holes at top calandria tube sheet	200	R/hr
During Calandria Removal		
Top tube sheet after removal	20	R/hr (c)
Vessel Wall at 3.05 m	65	R/hr

NOTE: (c) indicates contact measurement

Table II

Selected Vessel Removal Data

	NRX 1	NRX 2	NRU 1
First Criticality	1947	1954	1957
Planning time - years	0	7	12
Service time - years	5	16	15
Reactor Flux max. thermal	10^{14}	10^{14}	3×10^{14}
S/D to removal - days	161	28	236
Unload time - days	125	7	17.5
Vessel material - Al	ALCAN 2S	ALCAN 6056	ALCAN 6057
Vessel diameter m	2.58	2.58	3.51
Vessel height m	3.35	3.35	3.60
Vessel weight kg	3,540	3,540	11,560
Max. rad. at 3.05 m	65 R/h	18 R/h	40 R/h
Rem cost (max. individual)	17	7	5
Rem cost (total)	2600	117	176.5

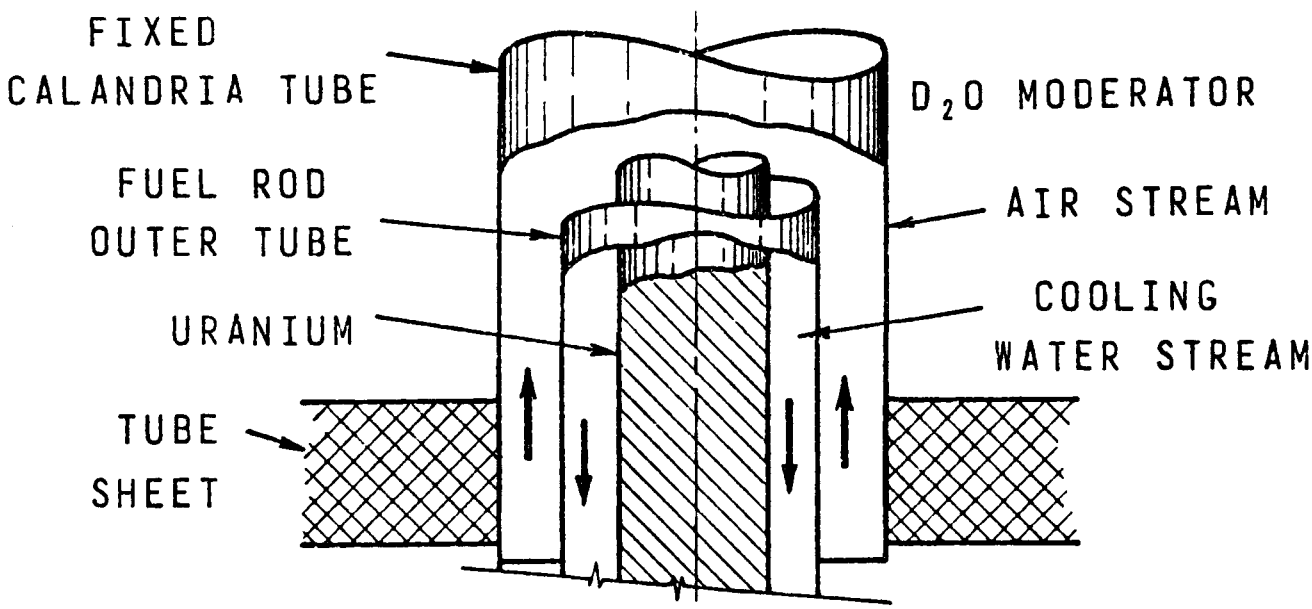


FIG. 1: NRX FUEL 1952

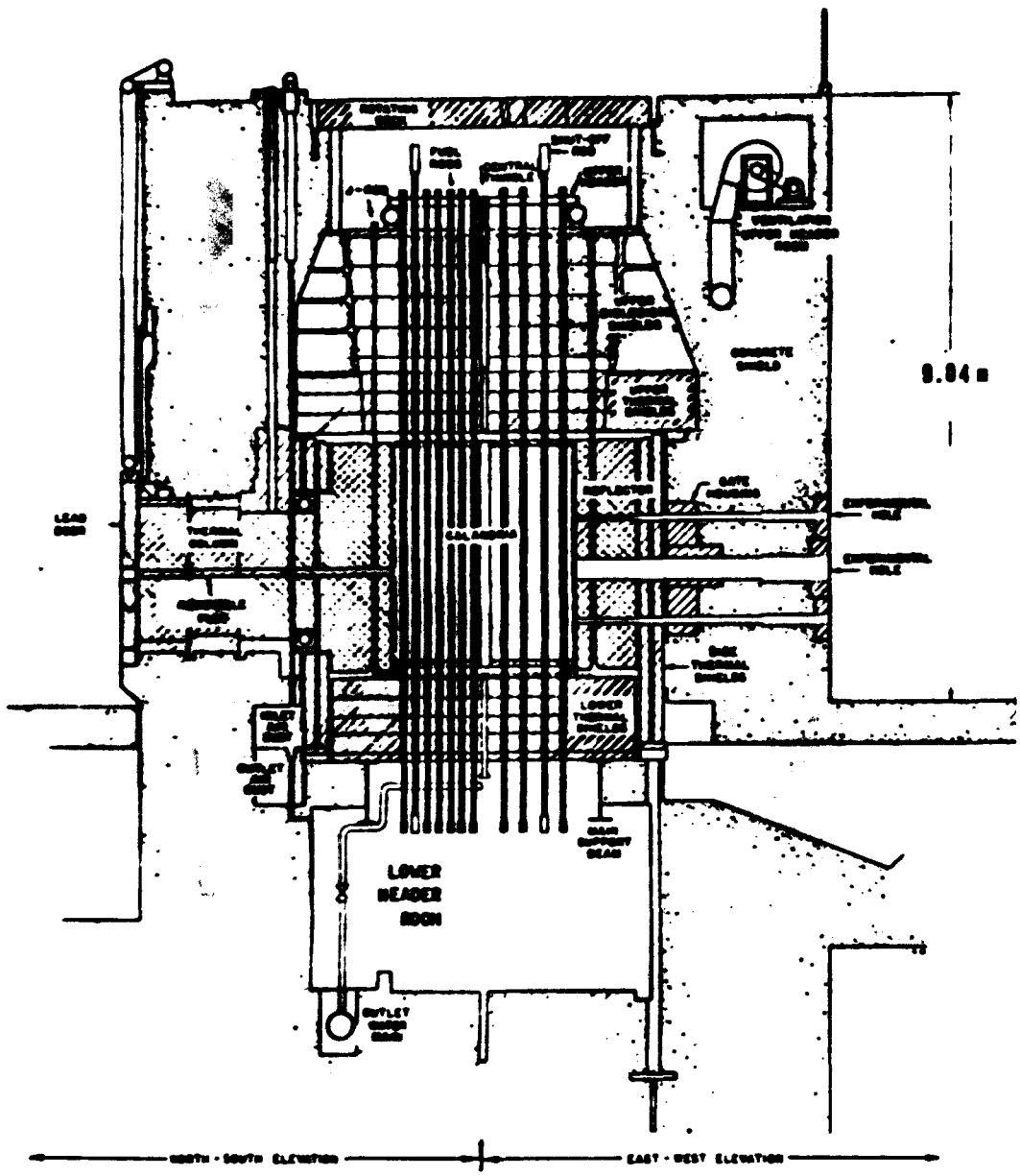


Fig 2: NRX Elevation

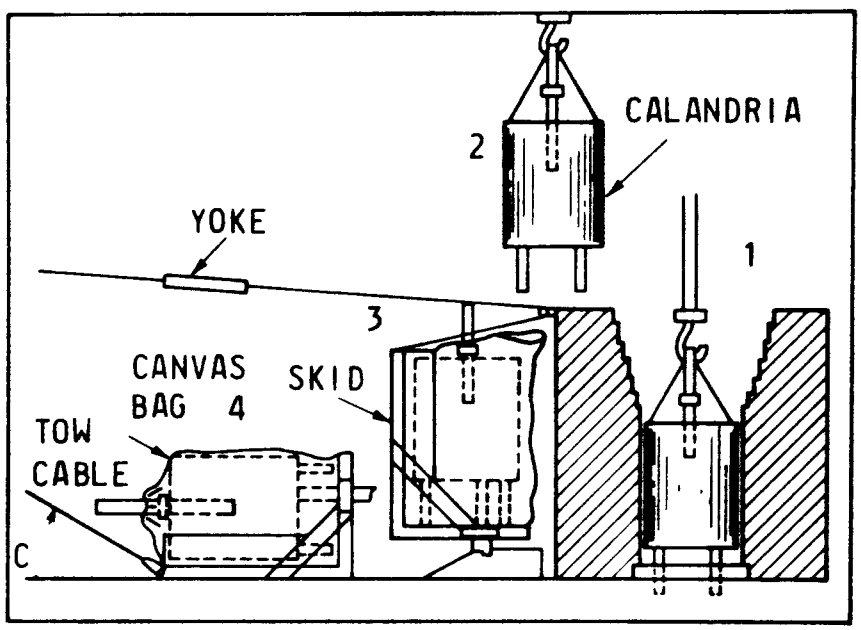


Fig. 3 **NRX 1 Calandria Removal Sequence**

FIGURE 5

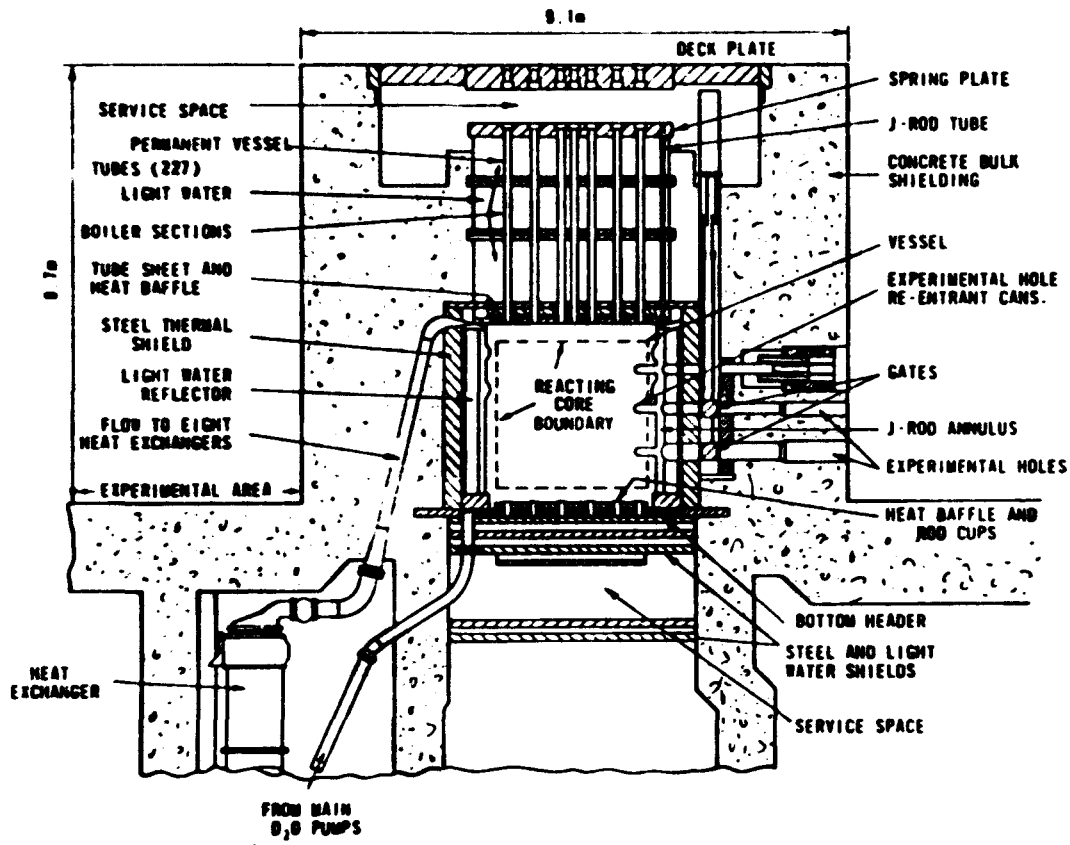


Fig 8: NRU Elevation

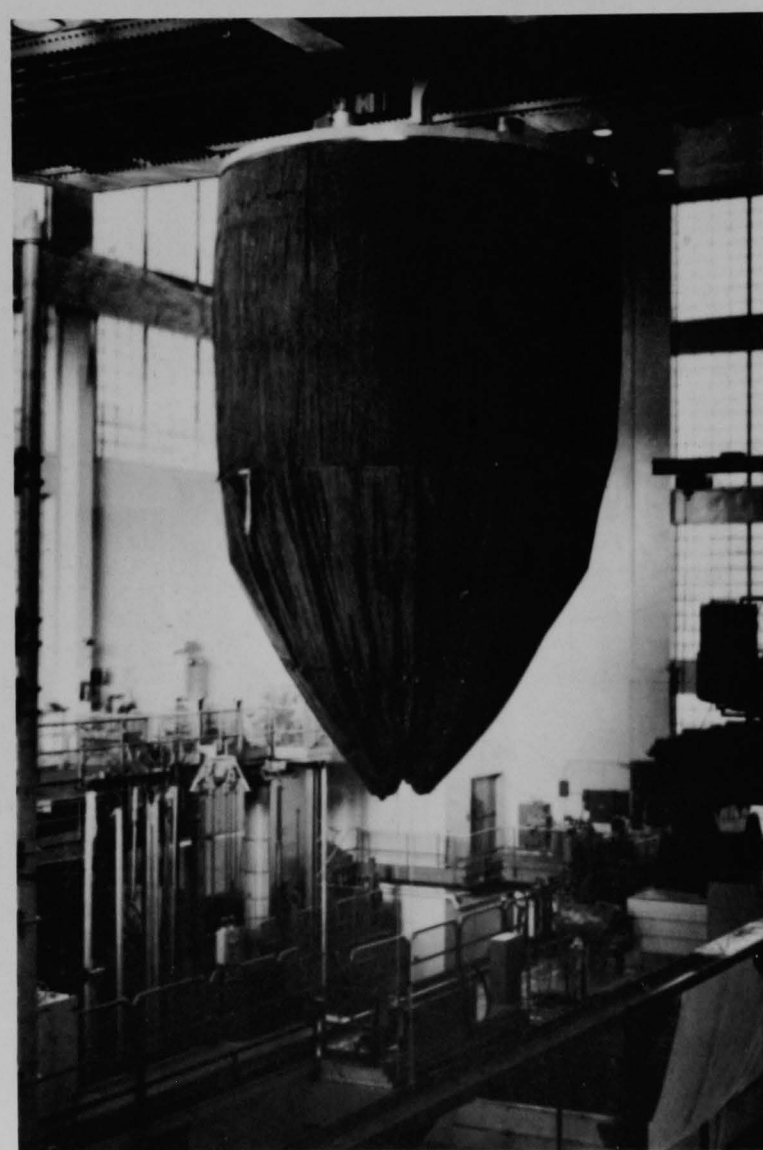
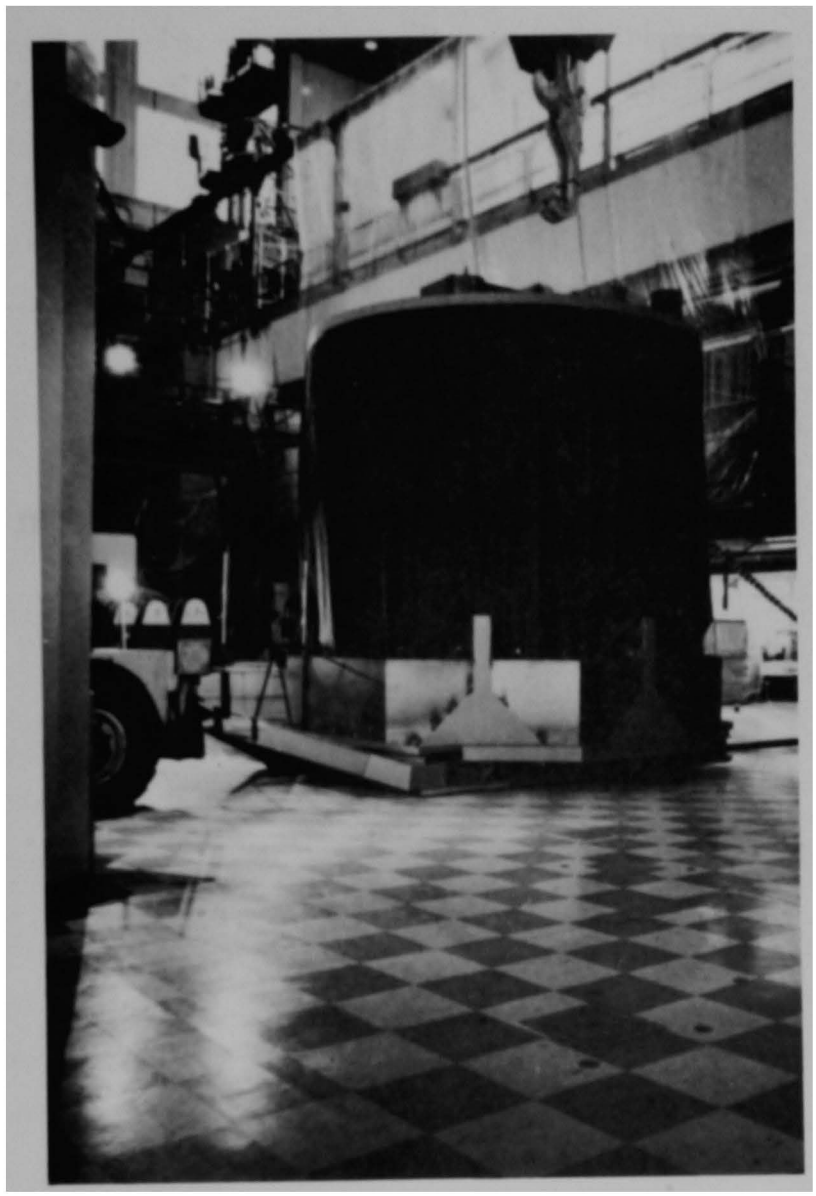


FIGURE 7



Session Q

INTERNATIONAL DECONTAMINATION EXPERIENCE

Paul J. Pettit

Atomic Industrial Forum, Inc.

I've been asked to relate some of my experiences with decontamination in the hope that they might be of some use in either the effort to recover TMI-2, or the effort to expand knowledge about reactor cleanup.

The experiences I can share with you were all gained while I was employed by Atomic Energy of Canada, Ltd., and appropriate credit is due many individuals in AECL, Cntario Hydro, and Hydro Quebec who were part of the effort. All the information I shall present has been previously discussed in various public forums.

What I want to talk about pertains to the use of chemicals to remove the activated corrosion products from the primary systems of water-cooled reactors.

About a decade ago, the designers and operators of Canadian reactors found themselves facing a situation involving rapidly increasing Co-60 contamination in the primary systems and rapidly increasing radiation exposure to operating and maintenance personnel.

To control exposure, a comprehensive, multiorganizational development program was undertaken. One of the several pursuits of that comprehensive program was the development of a means of using chemicals to remove the activated corrosion products from the piping and equipment in the primary coolant systems of the reactors.

I won't cover the details of the development program, but I would like to describe to you the product of the effort and some of the highlights of approaches used and lessons learned along the way. Perhaps these lessons could have some relevance to the problems at hand.

The process was developed with special needs and features as mind.

- o First it had to be safe and effective.
- o Next, there was a strong need to minimize the volume of waste produced.
- o It was desirable to avoid having to remove fuel from the reactor.

The process development involved an immense amount of laboratory testing of chemistry and materials, but the successful development of an integrated process hinged almost entirely on tests in reactors.

A series of trials was needed of increasing scope in progressively larger parts of real reactor systems. Laboratory simulations were helpful, but were inadequate for the full development of engineering data needed to achieve chemistry control and assure process effectiveness.

The process that resulted from the effort is as follows:

- o To apply the process the reactor is shut down and the coolant is kept circulating at about $90^{o}C$. The coolant is than purified. Mixed bed ion exchange is used to remove additives (Figure 1) and neutralize the coolant.
- o Next, a small amount of chemical is added directly into the circulating coolant, that is, the coolant itself becomes the decontaminating solu-ion, the chemicals circulate through the system attacking deposits and releasing contaminants from the walls of the piping.
- o Once the contaminants are suspended in the liquid they can be removed from the reactor by purifying the liquid (Figure 1). Cation ion exchange resin is used to remove the dissolved metals, like iron and cobalt. The cation resin also has another important function; it converts the spent, contaminated solution into a cleaned, reusable form. This is called

"regeneration". The regenerated stream is recirculated to the reactor to be used over and over again and continue the process for a long time.

- o Decontamination is terminated by replacing the cation resin (Figure 1) with a mixture of anion and cation resin which, together with the filter, removes everything from the coolant.

Clearly, the heart of the process is the use of a chemical and the purification. The process is independent of any particular chemical, and will operate with several different chemicals or mixtures of chemicals. The chemicals used were weak organic acids like citric acid. Only enough chemical was added to make a concentration of about 0.1% in the coolant.

Now I want to tell you how this process was applied to some big systems, describe the results and talk about some of the lessons I think we learned.

One of the systems decontaminated was the Douglas Point reactor. Douglas Point is a 200 MWe PHWR that began operation in 1967. The piping is constructed mostly of carbon steel, and the steam generator tubes are monel. Figure 2 shows some of the radiation fields around the maintenance areas before Douglas Point was decontaminated in August 1975. The unbracketed numbers represent the radiation fields befoere decontamination.

For the decontamination, special temporary, high flow, high capacity purification equipment was used. This equipment was prepared in advance and installed just after the reactor was shutdown.

Figure 3 shows what happened when the chemical was added. The concentration of contaminants immediately jumped to very high values. The water went black with fine particles.

Figure 3 also shows Co-60 in the main coolant and in the effluent from the purification systems. The high purification rate drew down the high Co-60 at about the expected rate, then a balance occurred between what was coming off the piping and what was removed by purification.

The "regeneration" phase was about 10 hours long. The cation resin approached its useful lifetime, and then the cleaning was started with the mixed bed resins.

The decontamination was completed and the system restored ready for startup after only 72 hours from the time of shutdown (Figure 4).

Sampling indicated that a total of 210-260 Ci Co-60 were removed (Figure 5), two-thirds was removed by the cation resin. About 90% of the contaminants were removed from the fuel and about 30% of the boiler contamination was removed.

Figure 2 shows how the fields decreased in the primary system.

Another of the systems decontaminated was Gentilly-1, a 250 MWe boiling light water cooled reactor consisting of many individual pressure tubes which carry the steam-water mixture to elevated steam drums. (See Figure 6). After the Gentilly-1 system had operated for only about 150 EFPD total, work was needed in the feeder area and in the steam drums. Decontamination was undertaken in 1973 to reduce exposures.

With Gentilly we had a special concern: Potential sedimentation in low-flow areas of the steam drum of high activity particles of corrosion product released from the fuel cladding. The way we approached this was to do the job in two phases: First, we attacked the high contamination using only an abbreviated portion of the system shown by the hatched flow path. Second, we treated the whole normal flow path using the primary pumps for circulation.

Figure 6 shows the results of the decontamination of Gentilly-1. Fields were reduced in both the steam drums and the feeders. No significant corrosion occurred, as indicated by coupons.

Conclusions:

- o In big equipment, significant portions of the contamination that resides in the activated corrosion products can be quickly and easily removed.
- o Transfer of contamination was avoided when a stepwise process was used to extricate the highest contamination first from a limited part of the system.
- o To develop a process, tests in large size, real equipment are essential.
- o When developing a process, it is important to understand the performance capabilities of and response of the system to be treated.
- o Training is essential.

Question:

Have you noticed any long term effects from the chemical cleaning on these plants?

Answer:

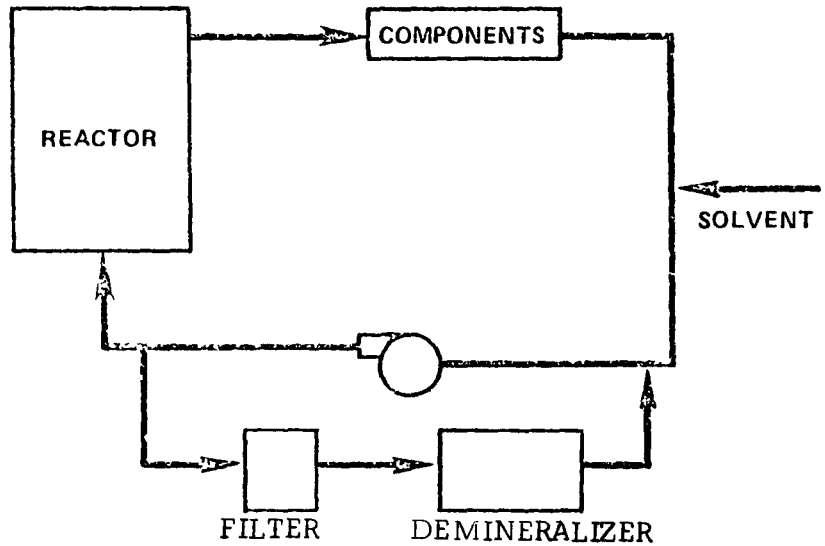
None. The Douglas Point reactor that I referred to has been successfully operated without any problems since 1975. There was an immense amount of effort put into the development and the testing of possible long term effects and none were found.

Question:

Was a passivation technique used on these plants?

Answer:

No passivation technique was used and no rapid rise in contamination was observed afterward.



SIMPLIFIED FLOW DIAGRAM FOR
RAPID DECONTAMINATION PROCESS

FIGURE 1

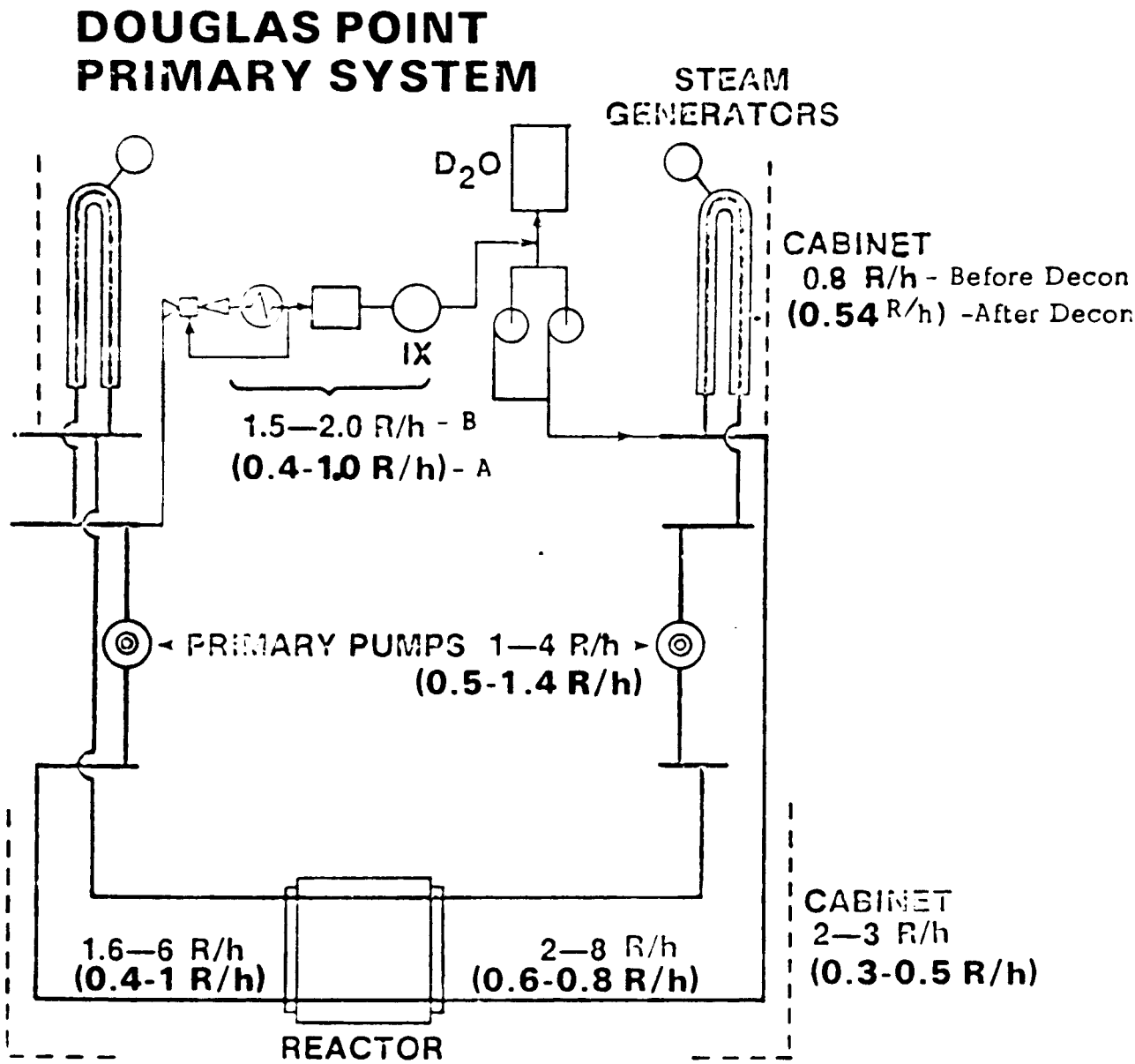
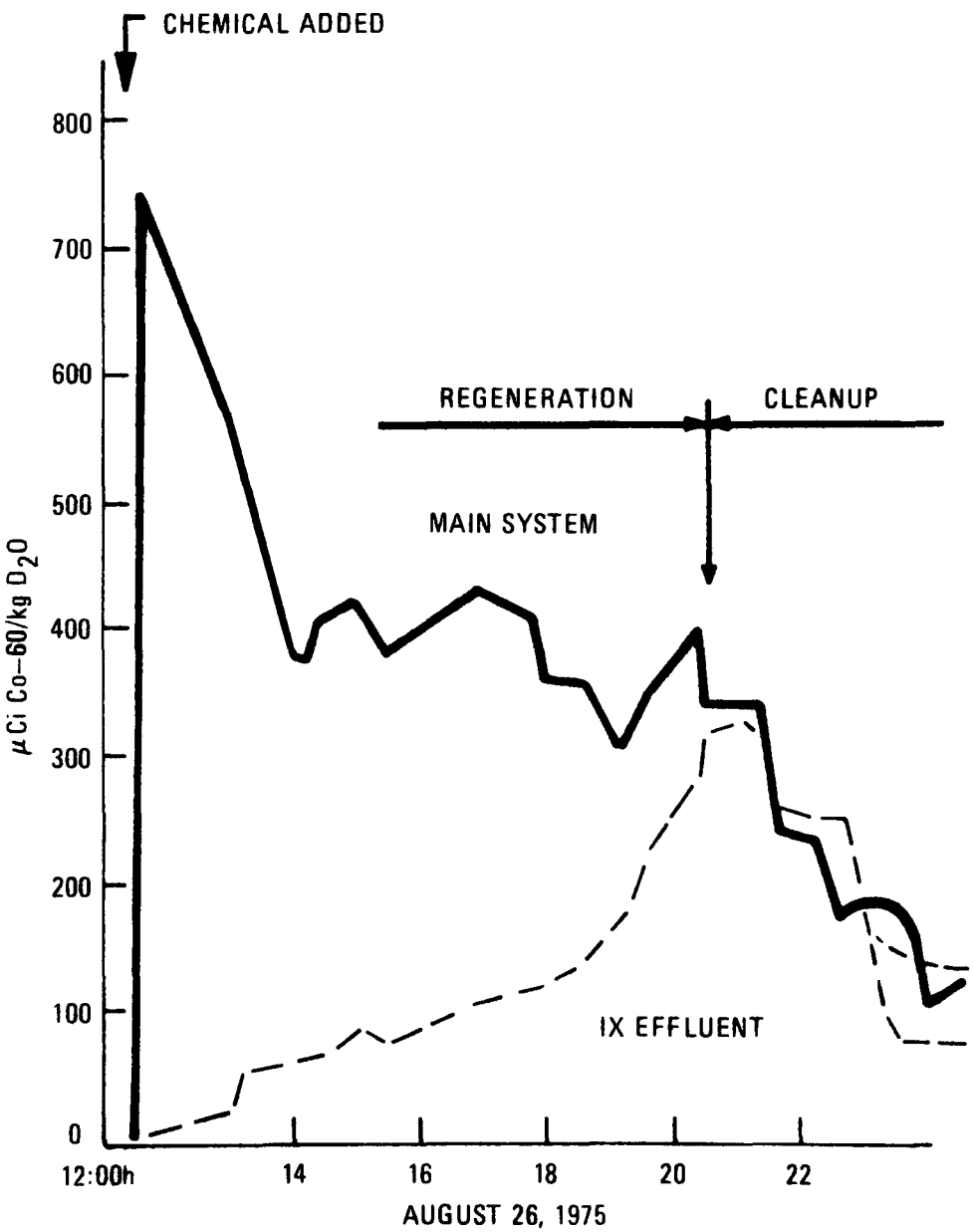

DOUGLAS POINT
PRIMARY SYSTEM
STEAM
GENERATORS
D_2O
IX
1.5—2.0 R/h - B
(0.4-1.0 R/h) - A
CABINET
0.8 R/h - Before Decon
(0.54 R/h) -After Decon
PRIMARY PUMPS 1—4 R/h
(0.5-1.4 R/h)
1.6—6 R/h
(0.4-1 R/h)
2—8 R/h
(0.6-0.8 R/h)
REACTOR
CABINET
2—3 R/h
(0.3-0.5 R/h)

FIGURE 2



Co-60 IN DOUGLAS POINT COOLANT

FIGURE 3

Co-60 REMOVED FROM DOUGLAS POINT

	Ci Co-60	
REMOVED BY IX	140-170	210-260
BY FILTERS	70-90	
FROM FUEL	34	(90%)
FROM BOILERS	110	(30%)
FROM PIPING	75-115	

FIGURE 4

DECONTAMINATION OF DOUGLAS POINT

TIME REQUIRED	72 h
Ci Co-60 REMOVED	210-260
DECONTAMINATION FACTOR	2.8
MAN-REM SAVED	170
CORROSION AND DAMAGE	INSIGNIFICANT

FIGURE 5

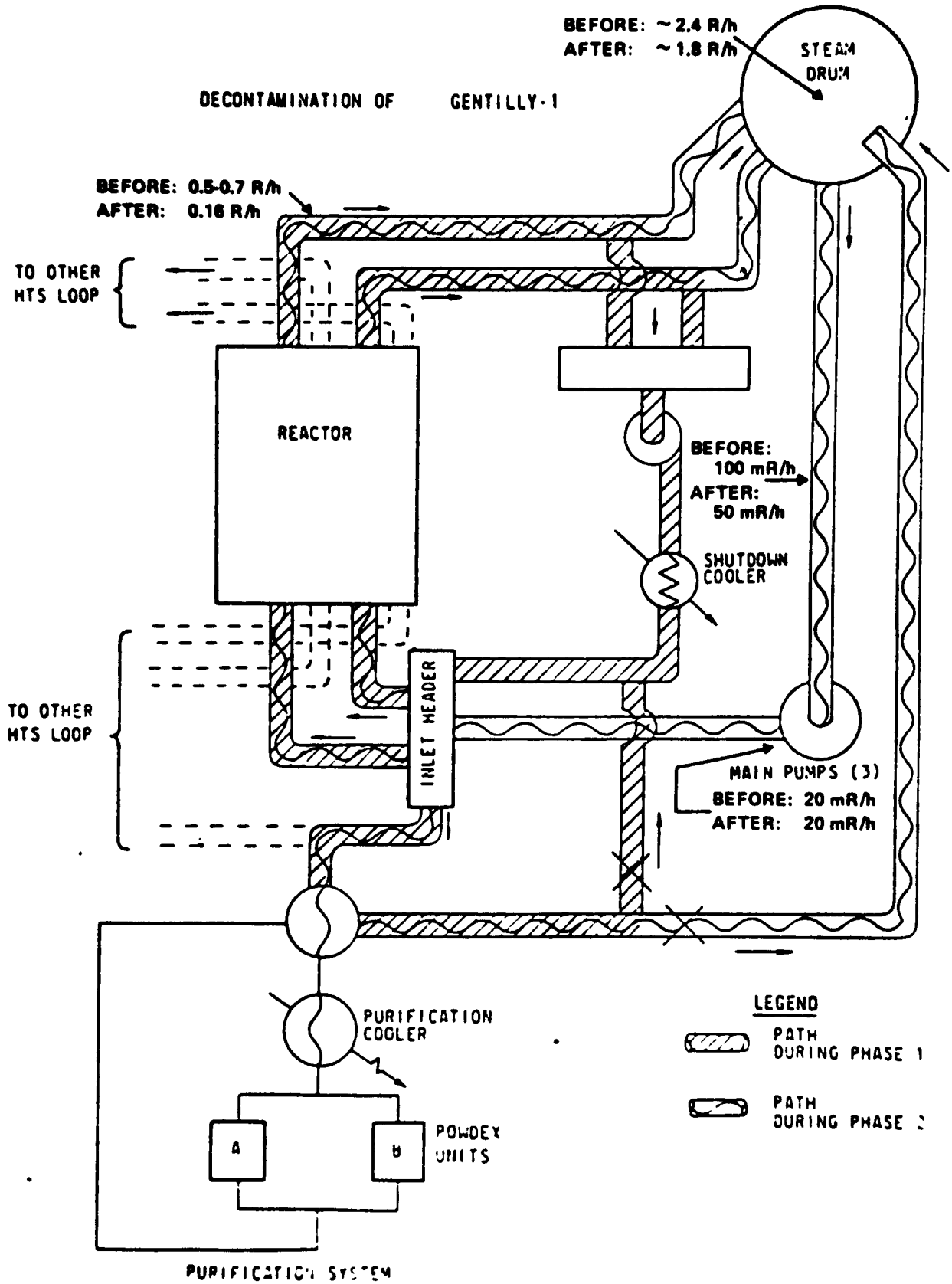

BEFORE: ~2.4 R/h
AFTER: ~1.8 R/h
STEAM DRUM
DECONTAMINATION OF GENTILLY-1
BEFORE: 0.5-0.7 R/h
AFTER: 0.16 R/h
TO OTHER HTS LOOP
REACTOR
BEFORE: 100 mR/h
AFTER: 50 mR/h
SHUTDOWN COOLER
TO OTHER HTS LOOP
INLET HEADER
MAIN PUMPS (3)
BEFORE: 20 mR/h
AFTER: 20 mR/h
PURIFICATION COOLER
A
B
POWDEX UNITS
PURIFICATION SYSTEM
LEGEND
PATH DURING PHASE 1
PATH DURING PHASE 2

FIGURE 6

Session R

DECONTAMINATION OF THE ARGON CELL OF THE IDAHO FUEL CYCLE FACILITY*

J. Paul Bacca
Argonne-West, Idaho

*Now termed Hot Fuel Examination Facility/South.

Thank you, Mr. Chairman. My presentation concerns a major effort currently underway to decontaminate a large hot cell such that systems in this hot cell can be modified, repaired, and/or upgraded. I would like to rapidly take you through a summary of events and experiences we have had including lessons we have learned in this difficult, but very real, activity. All of these elements, although addressed here for a hot cell, are the same as those necessary for the TMI containment entry, radiation surveying, and eventual decontamination and refurbishment. These elements strike familiar tones to me after having listened to Ed Walker, Mike Morrell, Paul Ruther, and others in yesterday's presentations. Because of the limited time that I have, I refer you to the complete paper that I do have, the manuscript of which will be available to you in the Proceedings of the American Nuclear Society Conference entitled, Decontamination and Decommissioning of Nuclear Facilities, which was held at Sun Valley, Idaho, the week of September 16, 1979. Those proceedings, I understand, will be published in the early part of calendar year 1980. Should you have immediate interest in my paper, I will be very happy to obtain a copy for you, if you let me know.

Figure 1 shows the Hot Fuel Examination Facility/South (HFEF/S) (formerly called Fuel Cycle Facility) which is a large hot-cell facility immediately adjacent to the EBR-II power plant. This facility is comprised of an air-atmosphere hot cell and an argon-atmosphere hot cell. The argon-atmosphere hot cell is the cell for which I intend to describe our remote contact decontamination efforts and experiences. This cell has a volume of about 60,000 cubic feet and an internal surface area of about 12,000 square feet. Inside the cell are two 5-ton cranes and six electromechanical manipulators which

rotate around a central pivot post. The interior surface of the cell is zinc-metallized carbon steel, clad to 5-ft-thick high-density concrete. The cell incorporates 18 viewing windows. Our purpose for entering this hot cell and its decontamination is to carry out major overhaul and refurbishment on the overhead handling systems, to upgrade the in-cell lighting systems, and to modify and improve the viewing window systems.

Figure 2 includes a plan view of the argon cell. The cell has the shape of a 16-sided polygon. It measures some 62 ft across opposite flats, is 22 ft high inside, and has 18 viewing windows. The distance across the annulus of the donut (the interior of the hot-cell) measures 16 ft.

The argon cell was used in the period 1964 to 1968 for the remote pyrometallurgical reprocessing and refabrication of uranium fissium* metal driver fuel for the EBR-II. Although the fuel was uranium-based, fast reactor burnup resulted in some buildup of Pu-239. We realize that this buildup was a reasonably low level, but it was an important consideration in our planning and conduct of all decontamination activities to date. The major radioactive contamination in the argon cell is believed to have resulted from the pyrometallurgical reprocessing furnaces and operations which allowed the oxides of the fuel to move around the hot cell as carried by the recirculating argon gas stream therein. It is believed that use of the hot cell subsequent to the remote reprocessing demonstrations (that is, for nondestructive and destructive examinations of breeder reactor fuels and materials irradiation experiments) did not contribute significantly to the contamination-radiation environment inside the cell. As you would expect, the significant long-lived fission products to be contended with are Sr-90, Y-90, Cs-137, and Ba-137m.

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*Fissium (Fs) is a mixture of fission-product alloying elements, principally molybdenum and ruthenium.

From a very clean, empty hot cell at startup, it evolved after some 14 years of operation without personnel entry to a cell nearly completely filled with all sorts of reprocessing, fabrication, and examination equipment. Our first activity, obviously, was remote removal of this equipment and its packaging, disposal, decontamination, or storage as appropriate. Following the removal of the equipment, the cell was subjected to dry methods of decontamination using remote means. The cell floor was brushed, swept, and vacuumed using the electromechanical manipulators and master-slave manipulators. Special procedures were developed to assure nuclear criticality and safety and fissile materials accountability during these operations. These procedures included sweeping, segregation, weighing, sieving, and limiting the quantities of materials collected in vacuum cleaners to a safe weight of 2.5 kg. I noted with interest during yesterday's presentations that nothing was mentioned with respect to fissile materials and criticality hazards controls being preplanned in the entry and decontamination activities, but I'm sure consideration is being given to these subjects. Following remote dry decontamination activities, remote wet activities using Turco 5865*, a foam-type decontamination agent, was used.

The Turco agent was spread over the floor areas, vacuumed and collected in drums, solidified using Safe-T-Set**, a solidifying agent, and disposed of as dry waste. Before dry vacuuming, the general radiation level in the cell was about 6 R/hr penetrating and 30 R/hr nonpenetrating at about 2 ft above the floor. Dry vacuum cleaning reduced the penetrating radiation by a factor of 6 (that is, from 6 R/hr to about 1 R/hr), but the nonpenetrating was reduced by only a factor of 2 (that is, from 30 R/hr to about 15 R/hr). Wet decontamination which followed reduced the penetrating radiation by a factor of 2;

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*Turco Products Division, Purex Corp., Carson City, Calif.
**Oil Center Research Inc., Lafayette, Louisiana

that is, to about 500 mR/hr level, and the nonpenetrating by even less. The nonpenetrating radiation was reduced from about 15 R/hr to approximately 10 R/hr.

Because the penetrating radiation levels (500 mR/hr) and the nonpenetrating radiation levels (10 R/hr) were near our criteria goals for remote decontamination, we prepared to enter the cell for hands-on assessment of the radiation levels in the cell. The initial survey when entering showed numerous hot spots for which the sum of the penetrating and nonpenetrating radiation ranged from 1 to 4 R/hr at 1 ft. The average personnel exposure accumulated for a 30-min. stay in the cell during the initial survey entries was about 0.3 Rem/hr penetrating and 2.6 Rem/hr nonpenetrating to give skin doses which totalled 2.9 Rem/hr. Because these levels were considered too high, we repeated the remote wet decontamination operations, but this procedure proved to be relatively ineffective in reducing the penetrating and nonpenetrating radiation. We, therefore, prepared for further decontamination using what we call "contact" means; that is manual hands-on decontaminations.

In preparation for the contact decontamination, elaborate, disciplined measures were carried out to assure positive contamination-control and personnel safety. All operations are controlled by written, approved procedures which assure that contamination-control and personnel safety hazards are considered and adequately addressed. Each step of the operation is carefully preplanned to develop techniques so as to minimize personnel exposure. Mandatory administrative controls and procedures detail specific actions to be taken prior to and during each entry. Responsible persons-in-charge are clearly delineated and considerations such as worker's medical history and current physical health, personnel rescue responsibilities, etc, are clearly addressed. In the area of training, when using non-ANL personnel, workers are given ANL radiation-worker training which

includes radiation-exposure control, contamination control, and information concerning plutonium safety. General training in the use of specialized hot-cell equipment is given, as well as is training in the specific task to be performed. To control radioactive contamination and air flow, temporary rooms were built inside and outside of the hot cell at a window port as shown in Fig. 3. The room outside the cell is used for personnel ingress and egress. This room is 8 ft by 30 ft and is a dry-wall construction attached to metal studding; the interior is lined with reinforced plastic. Inside the cell are two rooms; room No. 1 measures 6 ft by 8 ft, and room No. 2 measures 6 ft by 7 ft, also lined with reinforced plastic sheeting. The hot-cell gas-circulation system (now circulating air instead of argon gas) maintains an inward flow of about 600 cfm at the window entry-exit location and has proven to be very instrumental in excellent contamination control.

Personnel protective clothing is elaborate and comprehensive. For cell entries, personnel wear the following anticontamination clothing (listed in order from the body outward): (1) shorts, T-shirt, socks, and safety shoes provided by the Laboratory; (2) a pair of sack-type cotton coveralls (the first of three pairs); (3) two pairs of low-quarter polyethylene shoe covers; (4) one pair of high-top shoe covers; (5) two pairs of low-quarter shoe covers; (6) one pair of cotton glove liners; (7) one pair of rubber gloves; (8) one TYVEK* surgeon's cap; (9) one pair of safety glasses; (10) one pair of TYVEK coveralls; (11) a polyethylene supplied-air breathing hood; (12) a second pair of rubber gloves; (13) a two-piece plastic wet suit; (14) one pair of rubber boots; (15) a lead-loaded apron (0.5 mm Pb); and (16) one pair of lead-loaded gloves (0.35 mm Pb).

Before use, breathing air hoods are modified to include a short section of 25 mm (1 in.) dia. polyvinyl chloride (PVC) tubing clamped to a Scott HEPA

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*TYVEK is a fabric by DuPont Corp., Wilmington, Delaware.

filter approved for use in radioactive mists and fumes. The filter is covered with a piece of duct tape folded over to provide a pull tab. If the air supply to the worker's hood were to be interrupted, the worker would bite down on the stub of the PVC tubing inside his hood, pull the tape from the HEPA filter, and breathe through the filter while making an emergency exit from the cell. Integrity of the HEPA filter and its installation in the breathing air hood is pretested with stannic-chloride fumes. Each in-cell worker carries a pair of heavy-duty shears with which to sever his air-supply hose should it become entangled.

Certified breathing air for the in-cell worker is provided from redundant sources. In order of priority and backup, the sources are: (1) a large-capacity, two-compressor plant air system in the EBR-II facility; (2) a standby bank of breathing-air cylinders; and (3) a low-capacity breathing air system in the HFEF Complex. At 30-day intervals, certification of these breathing-air systems is reviewed. The operational readiness of each system is confirmed prior to every cell entry. The breathing-air supply is connected out-of-cell to a NIOSH-approved* breathing-air manifold** that can provide about 6 cfm air to each of four workers. From the manifold, breathing air is supplied through a continuous length of about 50 ft of 200 mm OD heavy-walled hose which passes through a cell-wall penetration without intermediate connections. A quick disconnect fitting, protected from contamination by a plastic sleeve connects the air hose to the worker's hood.

Each in-cell worker wears three pairs of thermoluminescent dosimeter (TLD 700) chips. One chip in the pair is unshielded and the other is shielded with 2 mm of aluminum. One TLD pair is included in the standard Idaho National Engineering

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*National Institute of Occupational Safety and Health
**Mine Safety Appliances Co., Evans City, Pennsylvania

Laboratory badge worn along with a self-reading pocket dosimeter in the breast pocket of his cotton coveralls. The second TLD pair is taped to the worker's forehead, and the third pair is taped to the back of the worker's thigh. TLD finger rings are worn on the middle finger of both hands. All TLD rings and chips are processed, and the dosimetry data are reported by the DOE Radiological Environmental Services Laboratory at the Idaho National Engineering Laboratory. The worker's whole body exposure to penetrating radiation is based on the highest reading of the three shielded TLDs. Skin exposure is based on the sum of the whole body penetrating radiation exposure and the highest nonpenetrating radiation reading of the three TLD pairs. Nonpenetrating radiation for a given TLD pair is determined by subtracting the shielded TLD reading from the unshielded TLD reading. Exposure to extremities is monitored by the TLD finger rings.

During entry, the digital dosimeter taped to the front of the worker's lead-loaded apron is used to monitor his exposure to penetrating radiation. By radio communication, the worker in the cell is periodically requested by the person-in-charge to read his digital dosimeter, and it is recorded by the radiation monitoring technician who logs this accumulated exposure. When a worker's exposure approaches a preestablished control value, the person-in-charge of the entry is advised that the worker should start to exit the cell. In-cell working times are now limited by the worker's whole-body skin exposures. The present 100 mR control value has limited exposures to the skin and extremities below 2000 mRem, which is the ANL-West administrative limit for any four-week period. This is how we maintain control with respect to personnel exposures times; it is a dynamic system and allows us to keep tabs on the worker as he is accumulating exposure.

Now I'll proceed to describe the contact decontamination of the hot cell; again, "contact" meaning hands-on decontamination. We were interested in

using dry methods, or those methods which require none or only very limited quantities of water. Our reasons were heavily motivated by the size and complexity of the hot cell and the susceptibility of a large number of in-cell systems which could potentially be damaged by water. Additionally, at that time we did not have facilities at ANL-West for processing large quantities of radioactive, plutonium-contaminated water or liquid mixtures. Therefore, our initial contact methods consisted of sweeping and vacuuming the cell floor, much as we had done in the remote activity, and other areas that were inaccessible during the remote cleanup. The floor of the hot cell was scrubbed with a water-RADIAC* solution using brushes. Powered floor scrubbers were used with Turco foaming agent 5865. Overhead, the hot-cell cranes and electromechanical manipulator systems were wiped down. Using what we call a "Howda", a personnel carrier, comprised of a box with wheels which spans the bridge of the cranes or electromechanical manipulators, was used to enable workers to wipe down the rails of the overhead cranes and manipulators. Because the "Howda" is provided with a hand-wheel drive, the worker merely moves along the bridge via his own hand power.

To develop more effective methods applicable to the cell walls and floor, a strippable coating was also tested. Turco water-based latex strippable coating No. 5931 was applied using an airless spray gun. Although the test results were encouraging, several operational problems discouraged us from using this method. These problems included overspray, thickness control, and especially the great difficulty we encountered in stripping the coating from the numerous projections of the cell's surface. We concluded that use of the coatings would be very appropriate should you have only plain surfaces; should you have protrusions, penetrations, etc, coatings use becomes quite questionable. During our testing activities, the Turco coating was sprayed on the hot-cell floor.

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*Atomic Products Corp., Center Moriches, N.Y.

Because the hot-cell interior surface is carbon steel metallized with zinc, it has to be one of the worst surfaces to decontaminate and certainly reflects that design of the facility in 1955-1960 gave very little consideration to the fact that one day the facility would have to be decontaminated.

Because of the problems that we encountered with the strippable coating, decontamination tests using high-pressure water-spray methods were begun despite the liquid waste disposal problem inherent in such an operation. These methods have been researched by others to be successful. The tests showed that high-pressure water-spray method was practical and operationally effective and efficient. Accordingly, we selected this method to be used for general in-cell decontamination. To minimize the volume of contaminated liquids generated, tests were made to determine maximum nozzle-to-surface distances of the spray gun at which the spray would still be effective for decontamination. Based on these tests, a high-pressure water pump and spray system capable of delivering approximately 4 gal./min. at 2000 psi was selected for use. The very sensitive and expensive glass viewing windows of the cell are protected using a window cover plate and an air purge system which slightly pressurizes the volume between the cover plate and the window. In this manner, decontamination water is essentially prevented from coming in contact with the windows. Covers are also provided to protect the electrical lights, electrical outlets, manipulator penetrations, etc, from damage by the water.

For worker positioning, a pneumatic lifter is used to allow the worker to easily get up and down the 22-ft-high hot cells. The individual can control his vertical position anywhere along the hot-cell wall up to the ceiling.

The in-cell system that we elected for removal of the water is a vacuum system piped to a collection drum and pump. We batch-collect the water in 55-gal.

batches, then pump it to a 1000 gal. mobile tanker which is parked outside of the facility. The tanker is shielded with lead to protect personnel traffic nearby. With respect to the spray gun, we are very interested in protecting the worker from the water spray and in keeping the water volume to a minimum. That is why we selected the 4-gal./min. high-pressure unit as opposed to the much higher-flow systems. We were interested in getting water to the activity for its dilution of the activity. We are fairly confident that about 80% of the Cs-137 and Sr-90 activity is soluble, and effectively getting water to every unit area of the hot-cell wall and floor is the way we can get the most effective decontamination operation.

The pumping appartus is a McCormick unit that delivers 2000 psi water. We utilize a standard commercial spray gun, outfitted with a custom-made shroud as shown in Fig. 4. It is outfitted with cam rollers such that the worker can place these rollers up against the surface of the hot cell, move them along in a very disciplined fashion, and thereby guarantee that one has unit coverage over the entire area and does not haphazardly spray and get water to that point, water to this point, but miss the intermediate point.

At this point, I think I'll wind up by saying that diminishing in-cell radiation intensity as indicated by personnel dosimetry confirms the progress of the decontamination process to date. The ratio of nonpenetrating to penetrating exposure has ranged from 4 to 7 and averages about 6. Figure 5 summarizes our progress through the month of August 1979. To date we have used the high-pressure water-spray method over about 90% of the surface areas of the hot cell. Very recently we have employed Freon 113 in the same high-pressure water delivery system for crane trolleys, electromechanical manipulator carriages, and the bridges decontamination. The Freon 113 is used, as opposed to water, with the incentive of protecting the electrical motors, wiring, etc. I think this use

is consistent with the information described yesterday by the Pacific Northwest Laboratory speaker.

After completion of the high-pressure spray-down of the cell, it is our plan to comprehensively map and assess the cell radiation and contamination levels to determine whether: (1) it will be necessary to spray down the cell roof of the hot cell because of its significant contribution to the radiation and contamination levels to personnel exposure, and (2) there is a need to conduct a second and possibly additional complete decontamination spray-down of the cell. Our goals are to reduce the radiation intensities to about 5 mR/hr penetrating and less than 20 mR/hr nonpenetrating plus penetrating. Whether we will be able to achieve these goals remains to be seen. To date, we have made something like 344 team entries.

In summary, we've learned a great deal in this large-scale decontamination experience which is apparently still a fairly unsophisticated science. We hope that our experiences at HFEF-Idaho may be relatable to you at TMI and to others involved in similar activities now or in the future.

Thank you for your interest and your attention.

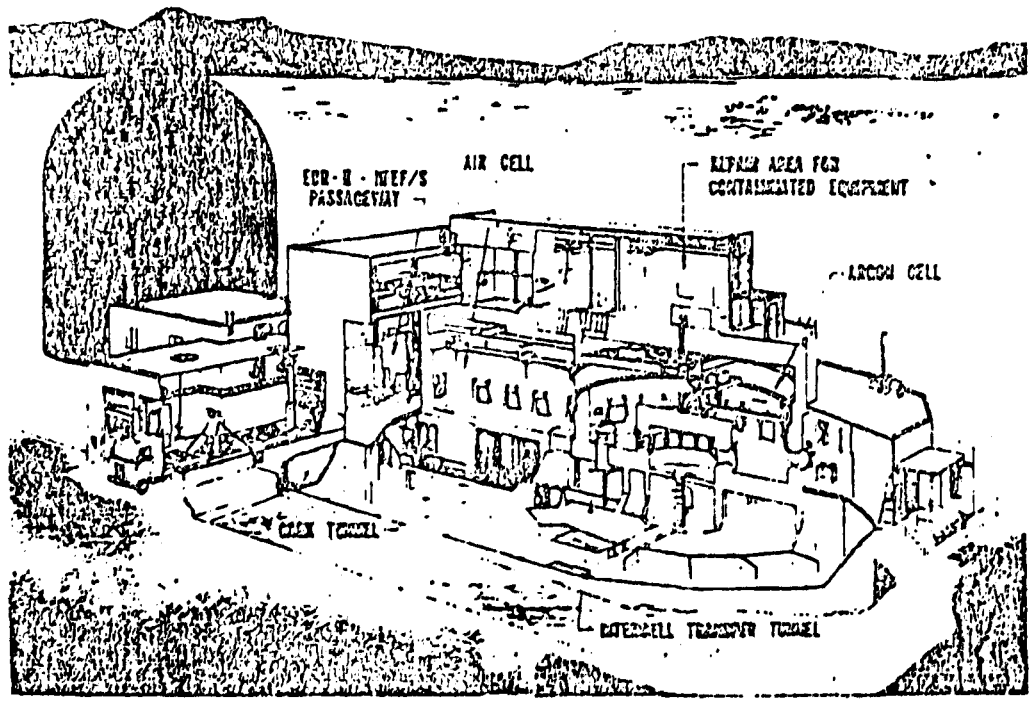

Fig. 1. HFEF/South Facility

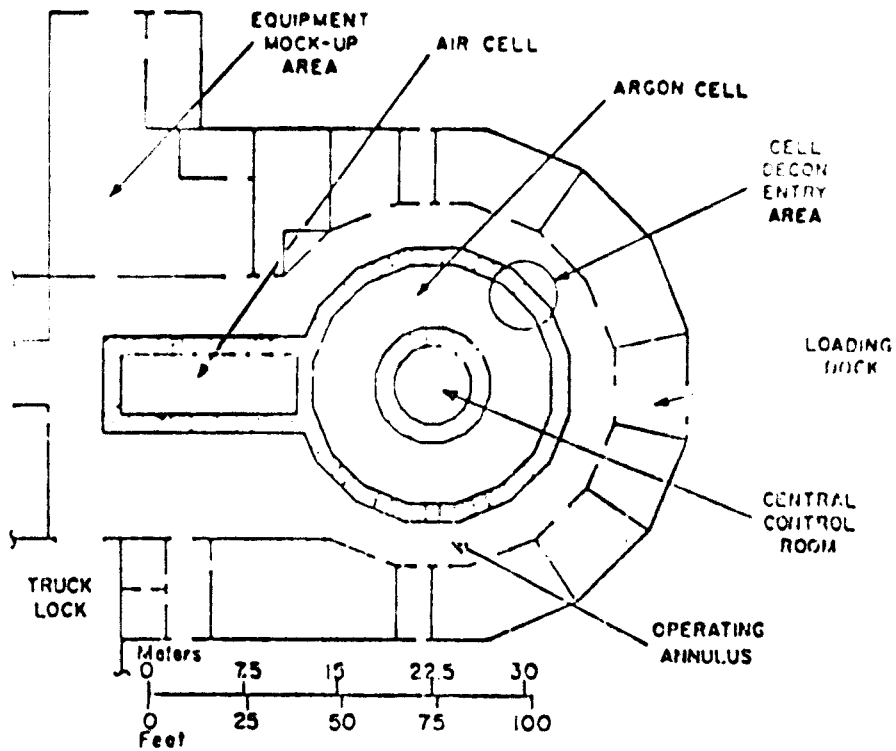

Fig. 2. Plan View of HFEF/South Hot Cells

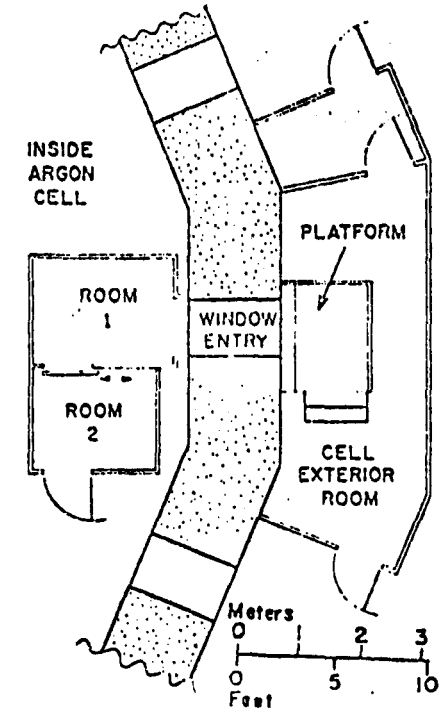

Fig. 3. Argon Cell Entry Rooms

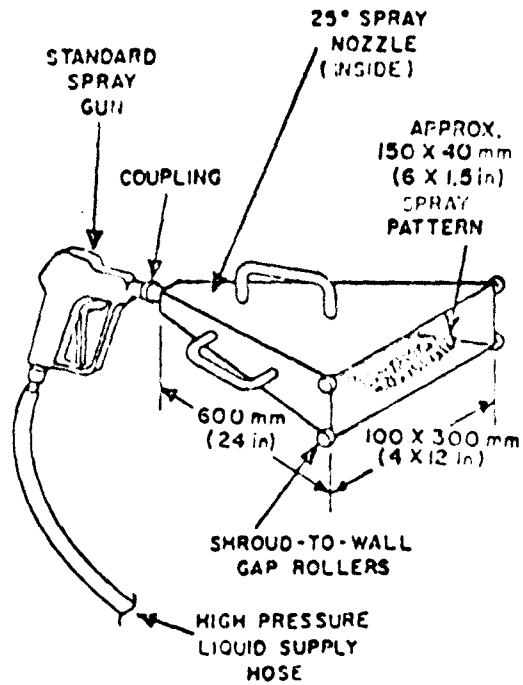

Fig. 4. High-pressure Spray Gun and Shroud

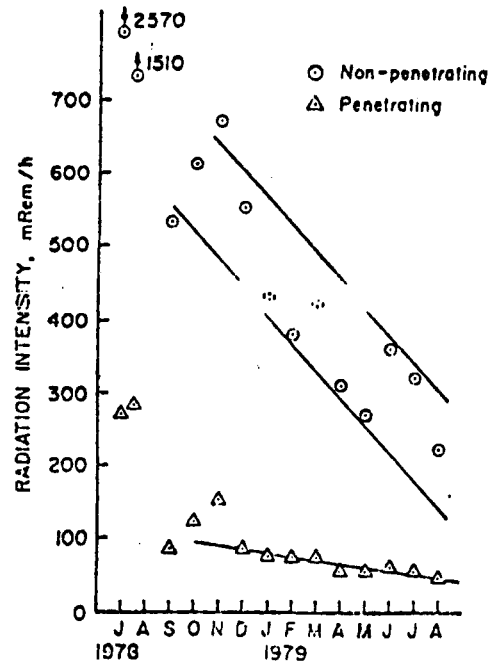

Fig. 5. Radiation Intensities During Contact Decontamination

Session S

DECONTAMINATION EXPERIENCE AT THE IDAHO CHEMICAL PROCESSING PLANT

John Johnson

Exxon Nuclear Idaho

The Idaho Chemical Processing Plant is operated by Exxon Idaho Company for the DOE principally for the processing of highly enriched uranium reactor fuels for return of this valuable enriched uranium commodity to fuel cycles and also for management of the resulting radioactive waste generated during these activities. The plant was built from about 1949 to approximately 1951; it started up in the 1951-1952 period. Originally, the plant was built for principally processing aluminum clad test reactor fuels such as the Materials Testing Reactor at the Idaho site and the Engineering Test Reactor which followed shortly thereafter. Since that time additional processes have been developed and added to our plant for processing fuels clad with zirconium, stainless steel and also for processing of graphite matrix fuels.

The fuel storage building consists of three pools approximately 20 ft. deep for underwater storage of metal clad fuels. A new building has been added for dry storage of graphite matrix fuels. At the present time, we're storing Peach Bottom Core 2 fuel in there and we're very shortly to receive first shipments of the Fort St. Vrain reactor fuel for storage in this facility. There are underground storage silos for storage of Peach Bottom Core 1 fuel. The dissolution of aluminum fuels is accomplished with nitric acid, catalyzed by mercuric nitrate catalyst. Zirconium fuels are dissolved in hydrofluoric acid; the excess fluoride remaining is complexed by aluminum nitrate solution for processing of the resulting solution through stainless steel equipment with minimal corrosion. Stainless steel fuels, principally EBR 2 reactor fuels, are dissolved electrolytically in nitric acid solution. Graphite fuels are processed by burning in fluidized bed burner. This is actually a two stage burning process followed by leaching of the resulting ash to dissolve uranium oxides for further processing. These various solutions are sent through our

first cycle of solvent extraction which is a tributyl-phosphate hydro-carbon solvent extraction cycle. The uranium product from that cycle is further purified by two cycles of solvent extraction by a hexane type system. The final uranyl nitrate product is converted to the uranium-trioxide product in a fluidized bed reaction vessel and shipped off site as a solid uranium-trioxide product. The radioactive waste solutions generated are stored in large stainless steel underground storage tanks that hold approximately 300,000 gallons each. There are about 23 tanks. After some 3 to 5 years of storage, these solutions are converted to a solid granular waste product in the waste calcining facility. This is also a fluidized bed process. The solids from the waste calcining facility are stored in large underground silos. These silos are approximately, 11 feet in diameter and about 50 feet tall. The fuel storage basin has overhead hangers that hold fuel down under the water. The water is approximately 6 inches below the grating level. The process building consists of 25 process cells in two rows of about 12 each. These cells are approximately 20 feet square by about 40 feet tall. In the remote analytical facility, highly radioactive samples are analyzed and/or diluted for further analysis. Some analysis has to be done on raw samples and analysis can be done on some solutions that are diluted quite a bit. A portal monitor is used for contamination control. These monitors we feel are state of the art. Each of these consist of several chambers and gas proportional detection chambers, that are highly efficient. I believe they use propane as the purge gas. When traffic is not traveling through them, they are in a background counting mode. The frisker can only be entered sideways to make sure the detectors come very close to personnel clothing. An interrupted beam switches to a mode where radioactivity on the clothing is counted and background is subtracted off. So these units can detect very low levels of contamination. The alarm points are set at about 1.4 times the standard deviation of the background counting level.

I will now talk briefly about some contamination problems that we encounter, our general decontamination approach in the plant, decon provisions on our process equipment, anti-C clothing we use and special problems and techniques we use.

In our storage basin, leakage of radioactivity into the water complicated fuel transfers. During the worst periods of water activity, the activity in the fuel storage basin water was as high as 0.2 μCi/ml.

This was primarily cesium 137, strontium 90, cerium-praseodymium 144 and strontium 89. At times we have had short periods of barium-lanthanum 140 which indicated leakage of fairly fresh fuel. The cerium 144 interestingly enough is essentially all absorbed onto solids in the basin water. Solids in our water were a problem until recently when most of the sludge on the bottom of our fuel storage basin was removed by Chem Nuclear under a sub-contract. This sludge consisted primarily of general dirt plus a fair amount of colloidal matter. At the bathtub ring level we had radiation fields on the order of 1 to 5 R per hour penetrating plus non-penetrating. Getting an accurate beta to gamma ratio is very difficult at that point because of the relatively high gamma background from the activity in the water itself but it's approximately 10 to 1.

For decontamination of the casks as they are brought out of the water, we have found degreasing agents such as methanol chloroform to be useful indicating most likely the presence of organic films on the cask. This could be from exhaust fumes of trucks that pass through the fuel storage building that eventually deposit organic films on the surface of the water. Radiac wash has been found very effective for decontamination of some casks and painted concrete surfaces.

In the fuel processing systems, one of the biggest problems is undissolved fuel solvents that collect in various places in our process equipment such as in the bottom of solvent extraction columns and bottoms of the process solution tanks. In the zirconium fuel processing system, there is a lot of zirconium oxide that does not dissolve and tends to absorb fission products from solution. We've had fields as high as 100 R per hour gamma fields at the bottom of process vessels and solvent extraction columns. In processing EBR 2 fuels by electrolytic dissolution, there is small fraction of the uranium fission products that do not dissolve. These fission products consist of ruthenium, promethium, zirconium, elements such as this. A fair fraction of the ruthenium 106 tends to follow this material. There is also a fair amount of colloidal meterial in these solutions that tends to act to glue the solid particles together to form plugs. We have found it necessary to use caustic solutions to break up these deposits. We do find organic cruds in various places from use of organic solvents in the solvent extraction business. We do form bathtub ring type deposits at the tops of our solvent extraction volumes which require the use of detergents for removal. Of course, in all of our processes, corrosion failures, gasket failures,etc. cause external contamination in our process cells. These result in rather high beta fields which we will discuss more later.

The waste calcining facility presents a few unique contamination problems. The fluidized bed calcination process takes place at about 500 degrees C. At this temperature there is a significant amount of ruthenium volatilization. The species we're concerned most with is ruthenium 106 in these aged process solutions. This ruthenium plates out at a fairly high temperature in our off gas clean up equipment and becomes incorporated in a very high temperature oxide film on the stainless steel surfaces which cannot be removed by normal film stripping techniques such as alkaline permanganate followed by oxalacetic acid solutions. We find it necessary in laboratory studies to go

to more aggressive treatments such as use of caustic permanganate solutions followed by oxalic acid and techniques such as wet sand blasting to actually blast off these high temperature oxide films. In one instance in which we found it necessary to open up the waste calcining vessel, we encountered a very, very high non-penetrating field of about 330 R per hour. The general gamma background in the same area was down to about 2 R per hour giving you a beta gamma ratio in that instance of about 150 to 1.

Another serious problem that we have encountered in the waste calcining process is leakage of process solutions and decontamination solutions from the calciner vessel during decontamination through nozzle holes in the side. The calcining process involves inbed combustion of kerosene and oxygen sprayed into the fluidized bed through nozzles and these don't always fit very tightly; they're spring loaded when they're attached, and we do get leakage through these nozzles. We've had considerable contamination in the vessel insulation which of course does not lend itself to easy removal. We've actually pumped nitric acid solutions into these insulating material in an effort to leech some of the contamination out and we've removed as much as 50% of the contamination by using the technique. The insulation itself did not dissolve in the nitric acid.

The general decontamination approach in our process systems involves as much remote decontamination as possible. This is accomplished by injection of decontamination solutions through decon lines and instrument lines that run from operating areas into the process cells agitating, heating the solutions and draining them to the waste system. Water flushing is used as much as possible since the evaporation of water solutions give us minimal amounts of waste that have to be stored. Chemical solutions are ultimately required for further reductions of radiation fields. We've used fairly standard

solutions; alkaline permanganate, oxalic acid solutions, nitric acid solutions, sodium hydroxide, tartaric acid mixtures, some detergent solutions for removal of organic cruds. For removal of external contamination in process cells we have installed spraying systems in our process cells for the first cut at removing external deposits. Generally, it's required that the hatches be removed from the tops of these cells and additional directed water sprays from the process cells are required for large deposits which sometime form. After remote work is done, cell surveys are performed by HP technicians. In some cases, when specific hot spots are identified, we do some gamma spectrometry. At this point, we have a sodium iodide detector and a shielded container with a window at one end for determining the principal radionuclides in specific hot spots. We're working towards the capability of using a germanium-lithium detector in the same type of a system. However, the sodium-iodide detector is considerable smaller and more mobile than the germanium-lithium detector housing will be. This is principally because the jelly detector requires being kept at liquid nitrogen temperatures for good resolution of the gamma spectrum. When very, very hot spots are identified, we install the remote monitor heads in our cells with readouts outside of the cells for monitoring the effectiveness of various treatments that are employed. Most of these heads are plastic scintillator type detector mounted on a photo multiplier tube in a contamination proof housing. Eventually, after the gamma background has been reduced as much as possible, considerable hands-on external clean-up is often required in those cells where failures have resulted in substantial amounts of external contamination. Generally, spraying of water and chenical solutions through things such as Turco barrel pumps or Grayco barrel pumps and spray lances are used along with scrubbing with long-handled brushes. In most of these operations to minimize non-penetrating field exposures, we try to keep a little bit of water on the process cell floors because the cell floors are generally the collection point of most external contamination in the process cells.

Let me run down quickly the regalia of anti-C clothing that we wear. Generally we wear cloth coveralls, skull cap, over this Ty-Vac coveralls with hood. For foot protection, we wear shoe covers over safety shoes and this inside of latex boots. For hand protection, we generally wear two pairs of latex gloves. In more cases we use full face respirators for respiratory protection. As you probably noticed from the view of that process cell, it would be somewhat difficult to get around in one of those cells with an air-supplied suit (or a bubble suit), so we have gone almost exclusively to full face respirators. In cases where we're going to be spraying decontamination chemical solutions, we'll generally put a disposable acid protection suit on the outside of the protective clothing to prevent any chemical burns and that sort of thing. In the past we have recycled things like shoe covers, latex boots, latex gloves, but the problems of these things being still residually contaminated and/or damaged by the laundering process has forced us to go to complete discarding of this type of thing. We do launder cloth items and respirators for reuse. Personnel contamination has not generally been a serious problem; we find that good technique in undressing in staging areas after emerging from a hot area has resulted in a complete success in minimizing personnel contamination. Since we are a waste handling facility, in general, we have the luxury of a system for evaporating only radioactive wastes. Our process equipment waste system is a thermal siphon evaporator. The condensate from this evaporator is tested for activity; if it meets certain specifications, then it is combined with what we call our service waste for injection into the ground. If it is above set limits, it's recycled through the evaporator again. The concentrates from our waste evaporator are stored in our tank farm. Eventually, these waste are combined with the high level fuel reprocessing waste for conversion into granular solids in the waste calcining facility.

I'd like to say a few things quickly about the clean up of the ventilation tunnel which has recently been completed. This ventilation tunnel runs along the row of cells of either side of the plant, and it collects ventilation air from the process cells and conducts it to the south end of the plant where the ventilation air is transferred through an overground duct to our atmospheric protection system or the HEPA filtration system. This ventilation duct has also been used for locating various process in off gas piping and contains considerable numbers of valves and pipes and so on. So it has been entered on occasion for maintenance work on some of these process pipes. We have had leaks of process solutions, acidic solutions in this corridor which soaked into the concrete and resulted in very high radiation fields in the corridor. Due to the expediancy of operating the plant and meeting production schedules, in many cases these contamination spots have simply been covered with sheets of lead. This is to enable subsequent maintenance work to be done on the process pipeing. Radiation fields in the tunnel as high as 50 R per hour gamma have resulted from acidic solutions that soaked into the concrete floor, perhaps several inches. This tunnel is a concrete structure; I believe the walls were painted with an epoxy paint. The floor that we have in the tunnel at this point is an unpainted concrete floor. I believe that it has been built up from the previous floor which suffered similar problems of contamination and simply is an additional layer of concrete over old concrete. Cleaning techniques used in this ventilation tunnel involved spraying with an installed remote spray header. There are floor drains in this tunnel which conduct the waste to our process equipment waste collection system. These installed spray headers were not positioned in the most optimum places thus eventually personnel entries were necessary to do hands-on type spraying of cleaning solutions and water. There were a lot of contaminated tools; reminants of pipeing which were left by previous maintenance operation and a considerable amount of contaminated lead which had to be hauled out of the tunnel. Access into this

tunnel is extremely difficult. The tunnel is about 300 feet long, 6 feet high and about 8 feet wide. Eventually, we found it necessary to use techniques such as high pressure water spraying with a 10,000 PSI spray pump to spall off from a 1/2 inch to 2 inches of concrete in some areas. As I mentioned, we had a few areas with gamma fields as high as 50 R per hour. This actually covered only 100 to 150 square feet total area. Background fields in the tunnel prior to the clean up ranged from as low as about 500 mR per hour penetrating plus non penetrating up to the 50 R per hour. After approximately 5 to 6 weeks of cleaning, we have reduced the general background in about half of the tunnel to 100 to 200 mR per hour level. In a few of the small areas where we did have acidic solutions soaked into the concrete we were not able to remove enough concrete to lower the field below about 30 R per hour gamma. The actual decision on whether to excavate this material or simply cover it up with lead and additional concrete is being made right now. This clean-up operation over a 5 to 6 week period involved entries by approximately 150 people. Total skin exposure for this clean up operation was 80 man-rem skin exposure and about 40 man-rem penetrating exposure. Approximately 20 to 25% of the exposure was absorbed by health physics technicians. They are required to be present during any clean up operations which could possibly result in an overexposure of personnel.

No questions.

PNL-SA-8560

SESSION T

DECONTAMINATION EXPERIENCE AT HANFORD

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April 1980

Pacific Northwest Laboratory
Richland, WA 99352

NOTE

This paper has also been issued as a separate report, prepared for the Department of Energy, April 1980, under contract DE-AC06-76RLO 1830.

My plan today is to share with you the details of two recent decontamination projects at Hanford that have not been, as yet, widely recorded. The two projects that I'm going to describe both involved plutonium contamination. I think I will win the prize for low radiation exposure because we really didn't have the exposure problem others have commented about this morning. The scale of our projects was different, of course, from Three Mile Island, but I think the principles involved in recovering from a plutonium contamination incident are similar to your problems.

The first problem occurred earlier this year when a container of plutonium oxide ruptured in our storage facility. The entire interior surface of the facility became grossly contaminated. As other speakers have commented, few facilities are designed for decontamination, and this facility was no exception. The problem was further complicated by the minimal containment features and the location of the facility, the 303-C building in the center of the 300 area of the Hanford project (Figure 1). In addition to the Battelle facilities, United Nuclear Corporation and Westinghouse Hanford operate various research and production complexes in the 300 Area.

Our general approach was to first prepare the site for safe repetitive entry. The second step was to clear the floor area, and the final step was to decontaminate the structure.

Immediately following the incident, recovery and investigating personnel built a two stage greenhouse at the major entrance to the facility for the initial entry to investigate the accident (Figure 2). This greenhouse later became our base of operations for entry and decontamination activities. If some emergency were to occur in the greenhouse area when we had staff working inside the facility, there would be no safe way for them to get out. The staff inside would be grossly contaminated and would carry plutonium contamination outside. For this reason, we built a second greenhouse on the other side of the building. As a general rule, we had two to four decontamination staff and

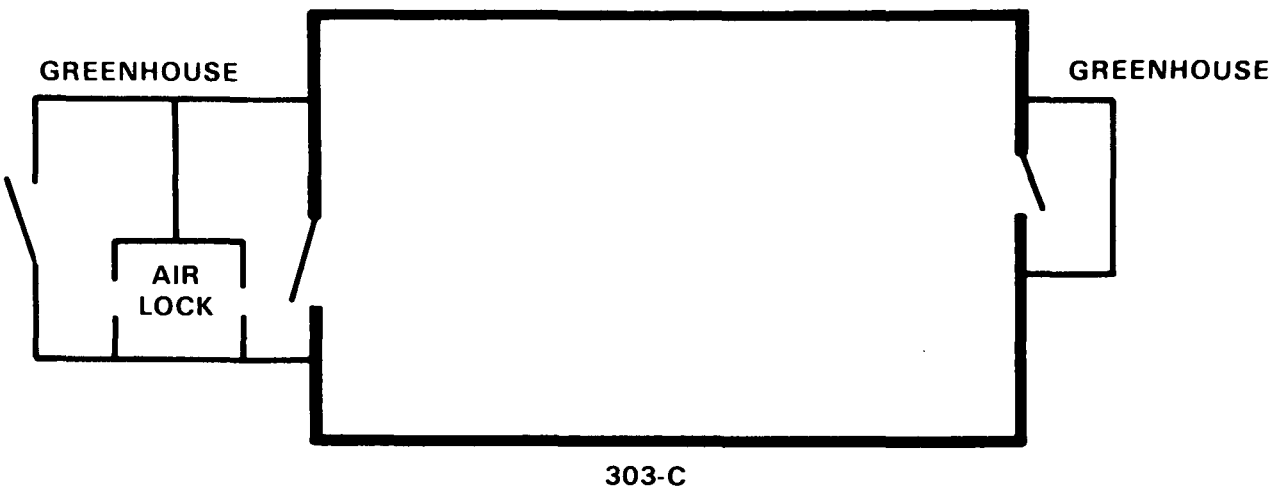


FIGURE 2. Greenhouse Construction for Decontamination Access to 303-C Building

one radiation monitor in the facility at all times. We had emergency staff, undressing staff, and radiation monitoring staff in both stages of the main greenhouse.

There were many operations that had to be managed and there was no place in the greenhouse complex to use as a control center. We acquired a trailer (Figure 3), which was outfitted to provide a base of operations (Figure 4). The trailer included an office area for radiation monitoring operations. The opposite end of the trailer was the control center for the project. One area of the trailer was used to complete final dressing for entry into the facility and to conduct briefing and debriefing operations.

We did have, as was mentioned by several other speakers, the advantage of having television in the area where we were working. The control center closed circuit television monitored all operations inside the facility (Figure 5).

This turned out to be a tremendous advantage, speeding up the operations and improving communications between the people doing the decontamination and those involved in directing activities.

The temperatures in June, July, and August when the recovery operations took place frequently reached or exceeded 100°F, and the greenhouse and facility temperatures were unbearable. We did add air conditioning to the greenhouse area and coupled the greenhouse to the facility for some measure of cooling (Figure 6), providing the recovery and undressing staff with livable conditions. It was not practical to add air conditioning to the facility proper.

The original HEPA-filtered ventilation system was inside the facility. It consisted of a pre-filter, two stages of HEPA filtration, and the blower that discharged the air outside. This turned out to be a problem because the most negative portion of the ventilation system contained penetrations and flexible joints. There was air leaking in at these points, and some contaminated air was being discharged out of the building directly instead of passing through the filter system. It was necessary to add a HEPA filter system outside the building to assure air being discharged was within release limits.

The decontamination technicians wore two pair of standard cotton overalls underneath the outer plastic wear (Figure 7) and respirators. The working conditions were extremely difficult because of the temperature and the dress, and the technicians who worked in the facility were limited to one hour entries because of these working conditions rather than radiation exposure. At the end of an hour, they were exhausted, and it was not unusual to lose several pounds in water loss during the one hour entry.

I want to comment on the powered air purifying respirator (PAPR) that we used. The PAPR was new to the Hanford experience, and this was the first extensive use of it. This light weight unit is worn strapped around the waist and consists of a battery-operated blower and two rectangular shaped HEPA filters, one on each side of the pack. The battery drives the blower that draws air through the filters and delivers air to the mask at 4 cubic feet a minute. The PAPR is NIOSH approved and has the same protection factor as a

fresh air system. A fresh air system has the disadvantage of dragging air hoses around and resuspending the contamination when working inside a grossly contaminated facility. Another major advantage of this particular mask is that if for some reason the battery happens to fail, you still have a standard canister mask and you can evacuate the facility without panicking. If you are in a facility with a fresh air supplied system and you lose the fresh air or your hose disconnects, it can be a very difficult, hazardous, and sometimes panicking situation.

Attesting to the reliability of the PAPR, we took nasal smears from every technician who entered and performed decontamination functions. In the entire project, we never had a positive nasal smear. Also, the technicians received baseline lung and whole body counts prior to the decontamination effort and close-out lung and body counts following the decontamination effort, and there was no positive indication.

There is a disadvantage to using the PAPRs; they require an above average amount of maintenance (Figure 8). The battery is only approved for four hours of operation; therefore, the mask has to be disassembled every day, the battery recharged, and the mask reassembled for use the next day. During this operation, the mask facepieces are decontaminated, if needed, and sanitized. Fresh filters are put on the assembly, and it is tested to make sure that the battery is charged and the blower is delivering the proper amount of air through the facepiece hose. Because the mask was new, it was an optional feature for the volunteers who worked on the decontamination effort, and some elected, at least in the beginning of the project, to continue using fresh air. However, by the midpoint of the decontamination effort, everyone was using the PAPR system.

The team that entered the facility immediately following the accident to recover the ruptured package and to investigate the accident became grossly contaminated and resuspended material that was on the floor. All of the horizontal surfaces in the room were grossly contaminated to 4.5 to 5 x 10^6 d/m/100 cm^2. All the vertical surfaces were contaminated, too, but not nearly to that level. A view of the facility from the front door is shown in Figure 9. We elected to use a strippable fixative to tie the contamination to the floor so we could make entries into the room. The fixative is milky

appearing when applied and dries to a clear finish. We applied about an 1/8-in. thick layer of the fixative to the floor, it dried overnight, and the next day we were able to enter the facility and walk the length of the building and back out to the greenhouse with contamination levels between 500 to 1,000 d/m/100 cm^2 on the shoecovering.

We were very impressed with the effectiveness of the fixative both as a contamination control feature and as a decontamination method itself. When we removed the strippable material from the floor, we removed a great deal of the loose contamination too. The floors, after the first stripping, were down from 4.5×10^6 to about 1.0×10^4 d/m/100 cm^2 smearable. There was higher fixed contamination, but the smearable contamination was significantly reduced. We ended up using the strippable fixative as a contamination control measure, as a decontamination measure, and as a protective measure as the decontamination project advanced to prevent recontamination of areas that we had already cleaned.

As shown in Figure 9, the facility was extremely cluttered with conduit, duct work, and other devices mounted on the wall. When we first entered the facility our approach was to decontaminate the movable items and to clear the floor area before we attempted to clean the structure. This went reasonably well; we were able to clean the movable items and safely remove them from the facility. When we started to decontaminate the wall area, we had a very serious airborne contamination problem. We were resuspending plutonium oxide trapped between the fixtures and wall. Air concentrations in the room approached 10^{-8} μCi/cc. We were mislead a little bit by our early equipment decontamination successes and felt this was not going to be too difficult of a project. This turned out to be a false impression and the project was much more difficult than we originally anticipated. We did successfully decontaminate all the movable material. We were able to clean the file cabinets and the other movable objects that were on the floor down to very low level of fixed contamination, but not to the point where they could be released and reused. The remaining nonsmearable contamination was covered with a fixative

to further tie the contamination to the object, and the item was wrapped in plastic, removed from the facility, and packaged in radioactive waste burial boxes for disposal (Figure 10).

The strippable fixative was applied using a low pressure paint spray delivery system to minimize resuspending contamination. We used the strippable coating on vertical surfaces but not with nearly as much success as we had on horizontal surfaces. The main problem was getting a thick enough coating to strip the material from the vertical surface. We tried to make that a little easier by hanging cheese cloth from the vertical surfaces and then applying the fixative (Figure 11). It worked quite well on smooth surfaces. As you sprayed the cheese cloth, it would cling to the smooth surface. If there were perturbations, it did not cling, and we had to go in and do a lot of hand decontamination in those areas. But, basically, the system worked quite well, and we were able to get very decent decontamination factors.

We also used the cheese cloth-strip coat technique to clean inaccessible areas. We had a blind ledge on the top of the storage array that we were barely able to reach; it was about 9 in. high and 7 ft deep. We were able to use strippable fixative to decontaminate that area reasonably well. We attached the cheese cloth to a section of pipe (Figures 12 and 13), placed it on the barely accessible ledge areas, sprayed it with the strippable material, let it dry over night, and rolled the pipe along the edge to roll up the cheese cloth (Figure 14). We were able to do a pretty decent job of decontaminating such areas. The contamination levels in the ledge area when we began were probably 10^6 d/m/100 cm^2, and we were able, with several attempts, to get it down into the thousands of d/m/100 cm^2.

It was extremely difficult, and I think many of you have experienced the same thing, to decontaminate the block walls. We were not able to completely decontaminate the block surfaces; there were low-level spots that just could not be completely decontaminated (Figure 15). Rather than demolish the block wall, we painted over the remaining low-level contamination with a yellow base coat (Figure 16) and a finish coat (Figure 17). The yellow paint, when it wears through, is a warning to those occupying the facility that they should

be alert to possible contamination. The areas that were painted over were well documented, and any modifications to the facility will refer to that documentation for guidance.

We also had a portion of the concrete floor in the facility that we could not clean. We did not want to leave any detectable contamination on the floor because of foot traffic and the heavy shipping containers and drums that are moved across the floor. We used a mechanical concrete spalling technique to remove the floor contamination (Figure 18). The technique involves drilling a pilot hole in the concrete about an inch in diameter and 1-1/2 to 2 in. deep (Figure 19), inserting an expanding bit into the hole (Figure 20), and hydraulically advancing a mandrel into the bit, which causes the bit to expand, grip, and spall out the concrete. We used this technique in the facility to decontaminate a floor area that we could not decontaminate any other way.

The second project involves, for those of you who are familiar with the Hanford project, the 231-Z buidling (Figure 21) in the 200 West area, a structure roughly 150 feet square. The facility had been in continuous operation

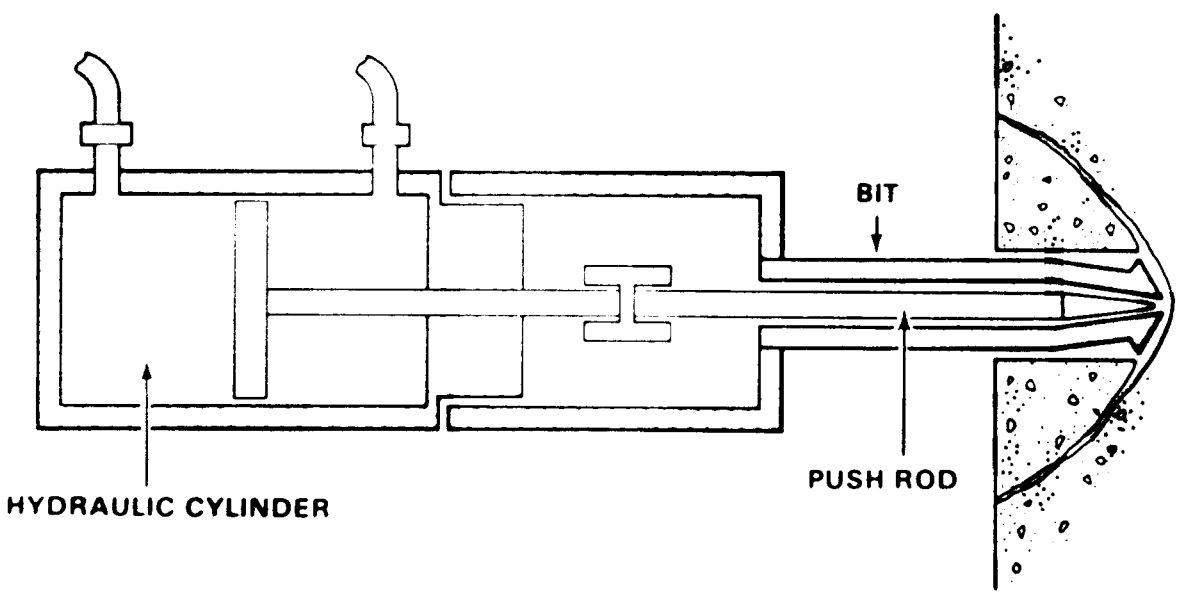


FIGURE 18. Concrete Spaller

as a plutonium facility for 30 years. The plutonium programs were phasing out, and the major sponsor agreed to decontaminte and restore the facility to put it back into useful service in the Department of Energy complex. Glove boxes in the facility had to be removed, along with their associated piping, ventilation, duct work, and accessory equipment. We measured the residual plutonium hold-up in each item removed from the facility in preparation for retrievable storage burial at Hanford (Figure 22). After the glove boxes were removed, the facility was further stripped, surveyed, decontaminated, and restored as a modern materials research laboratory (Figure 23).

Considerable piping, duct work, and ventilation equipment was removed from the facility as part of the operation (Figure 24). We successfully used electro-polishing to decontaminate much of the material (Figure 25). The equipment had been in service many years and had been grossly contaminated with plutonium. The final figure (Figure 26) shows the 16,000 ft^2 of material removed from the facility, packaged for retrievable storage. We had one tremendous advantage; the Hanford burial site was about 150 yards away.

FIGURE 1. The 300 Area of the Hanford Reservation at Richland, Washington

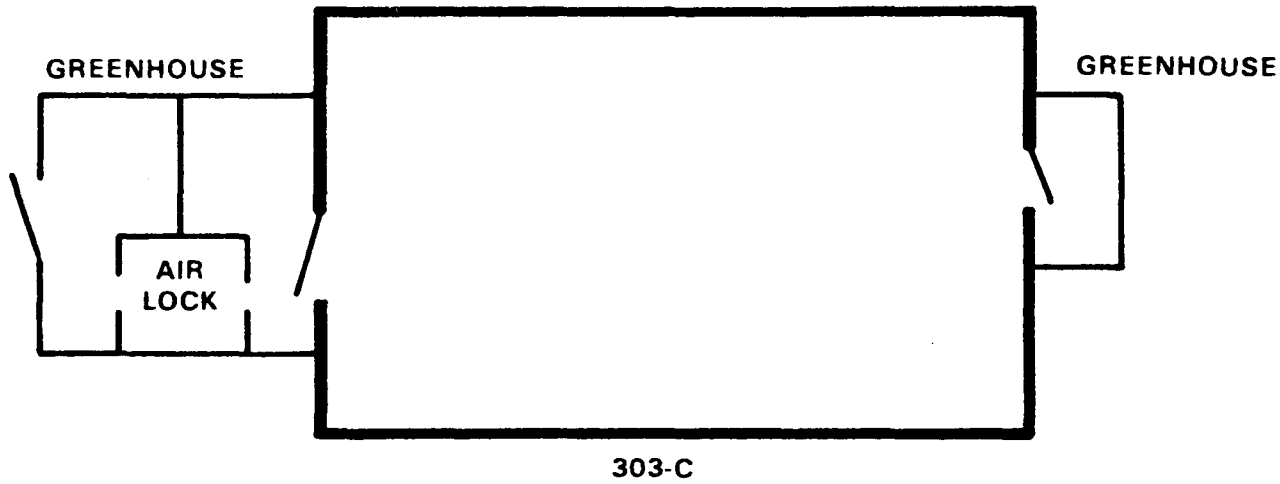


FIGURE 2. Greenhouse Construction for Decontamination Access to 303-C Building

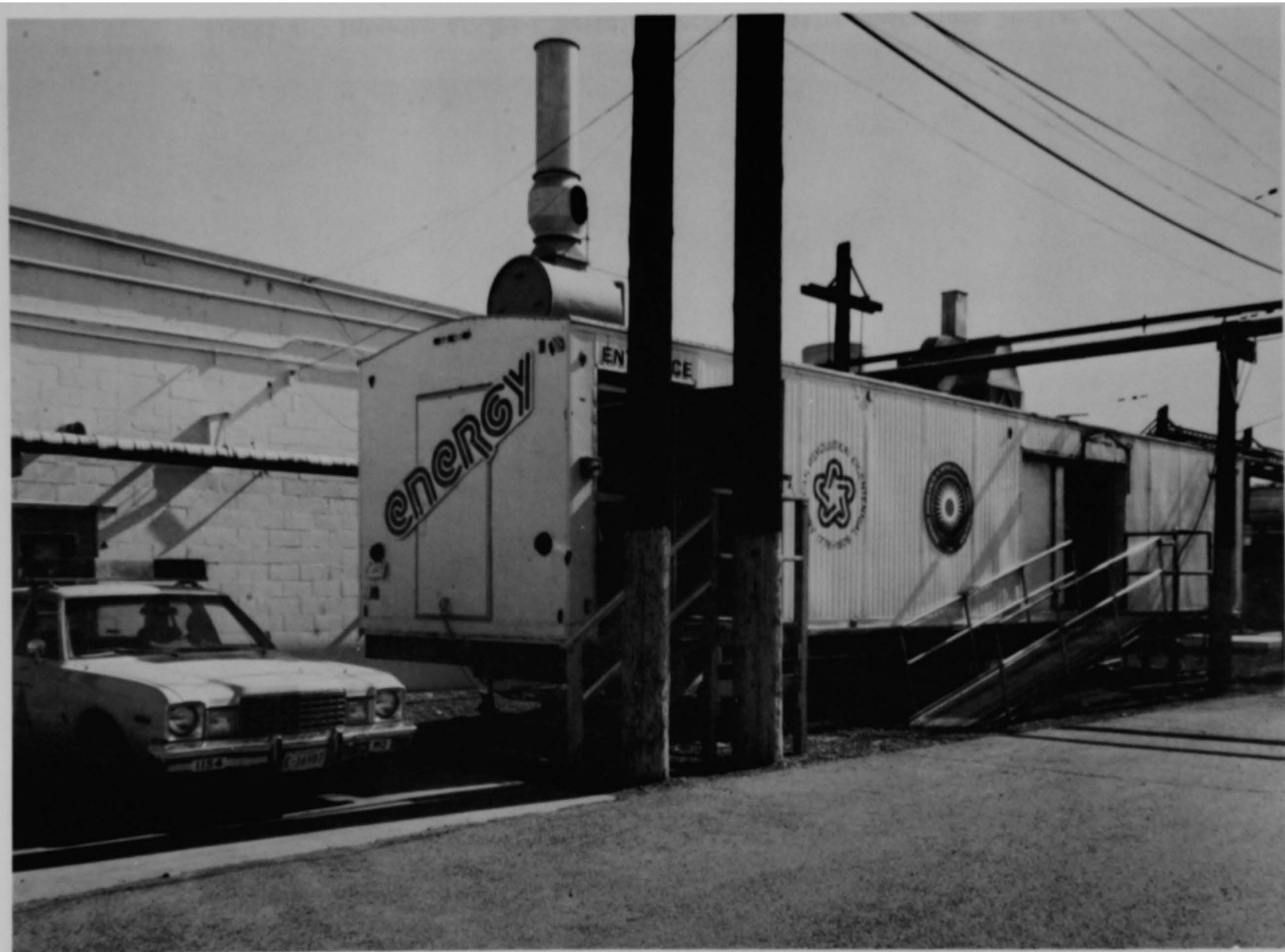
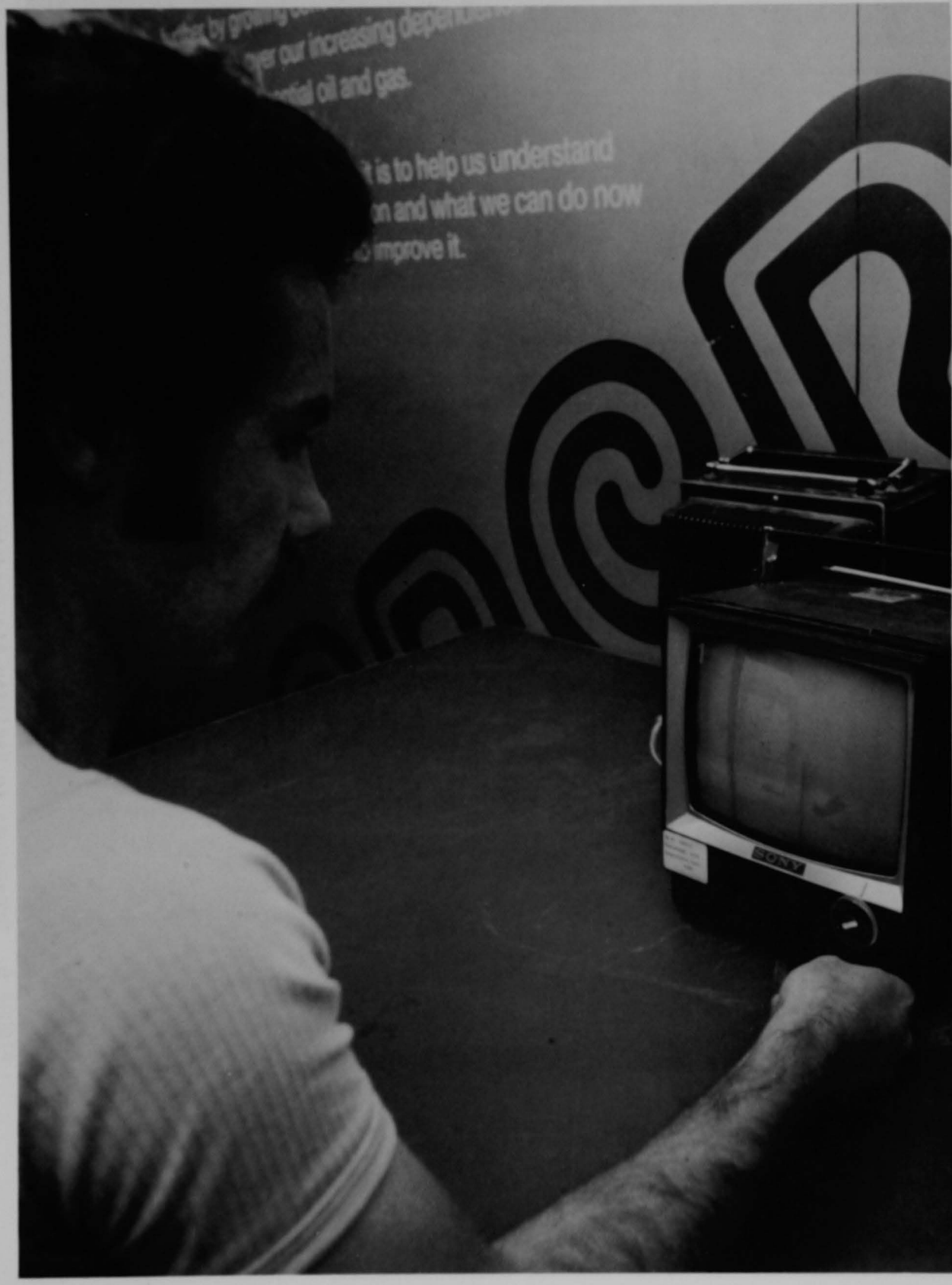


FIGURE 3. Trailer Used as Base of Operation for Decontamination of 303-C Building

FIGURE 4. Interior of 303-C Building Decontamination Operations Trailer


it is to help us understand
on and what we can do now
improve it.
SONY

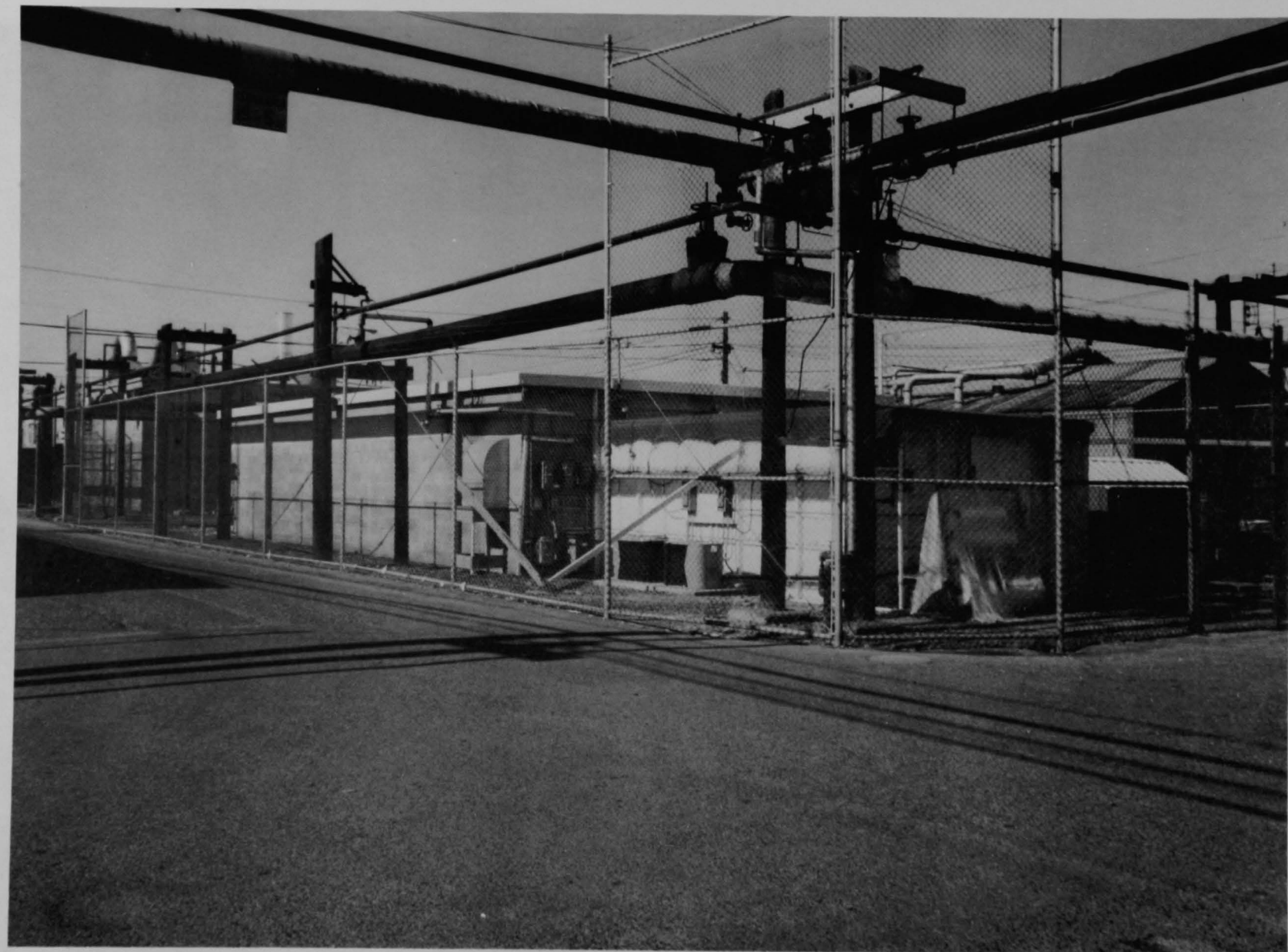
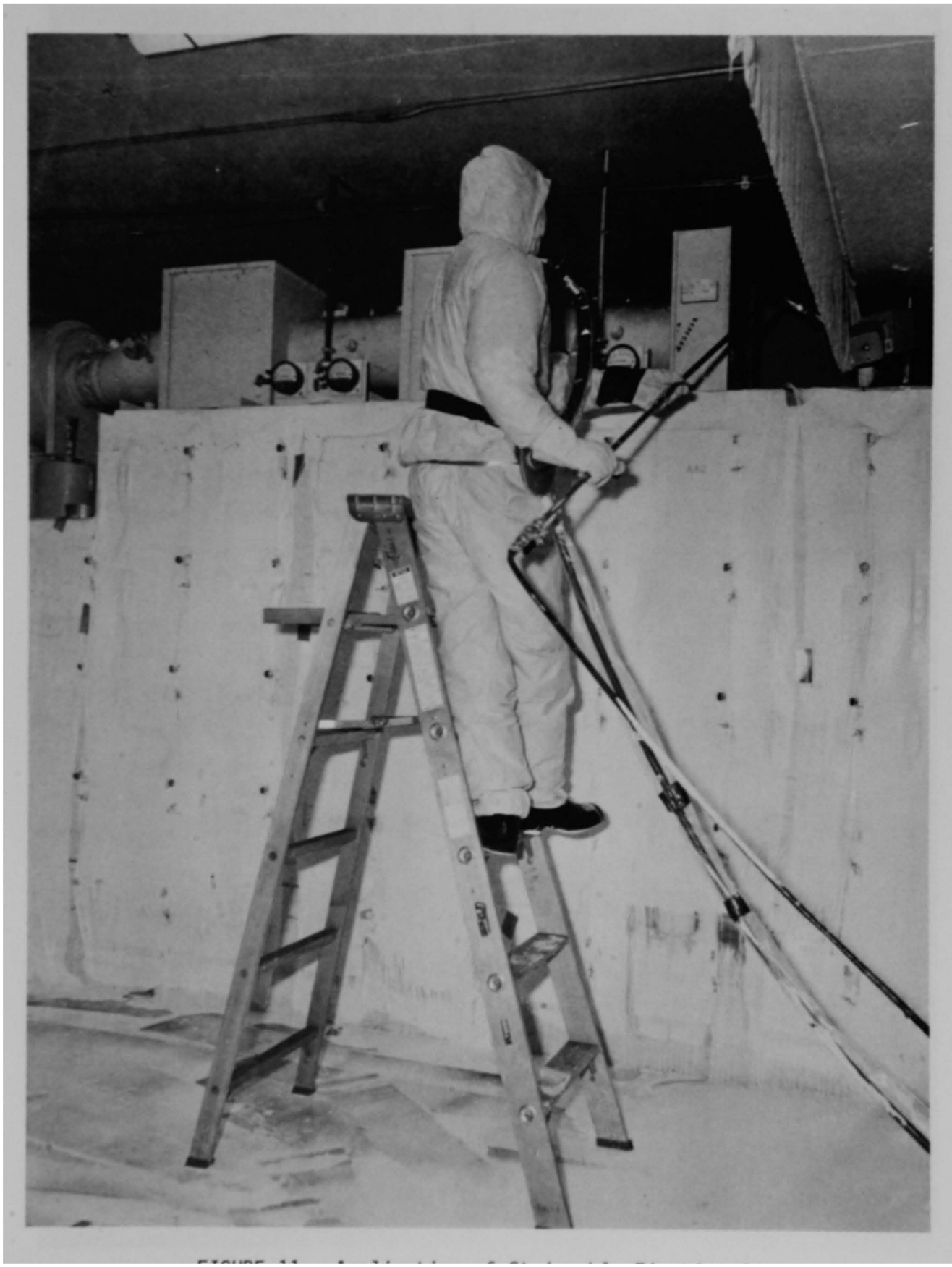


FIGURE 6. Greenhouse to Facility Air Conditioning Coupling at 303-C Building

FIGURE 8. Personnel Respirator Disassembly and Maintenance

FIGURE 9. Front Door View Into 303-C Building



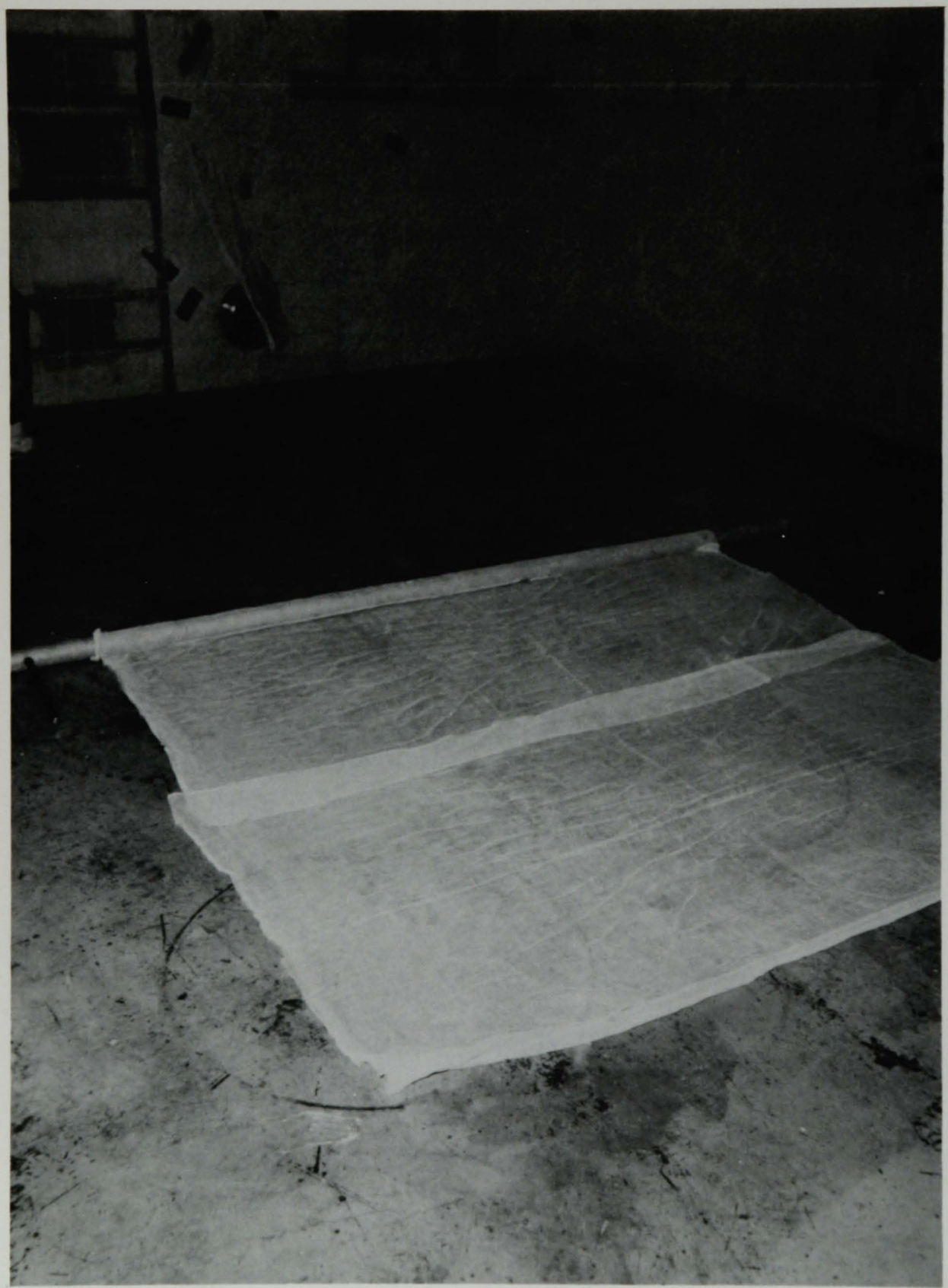
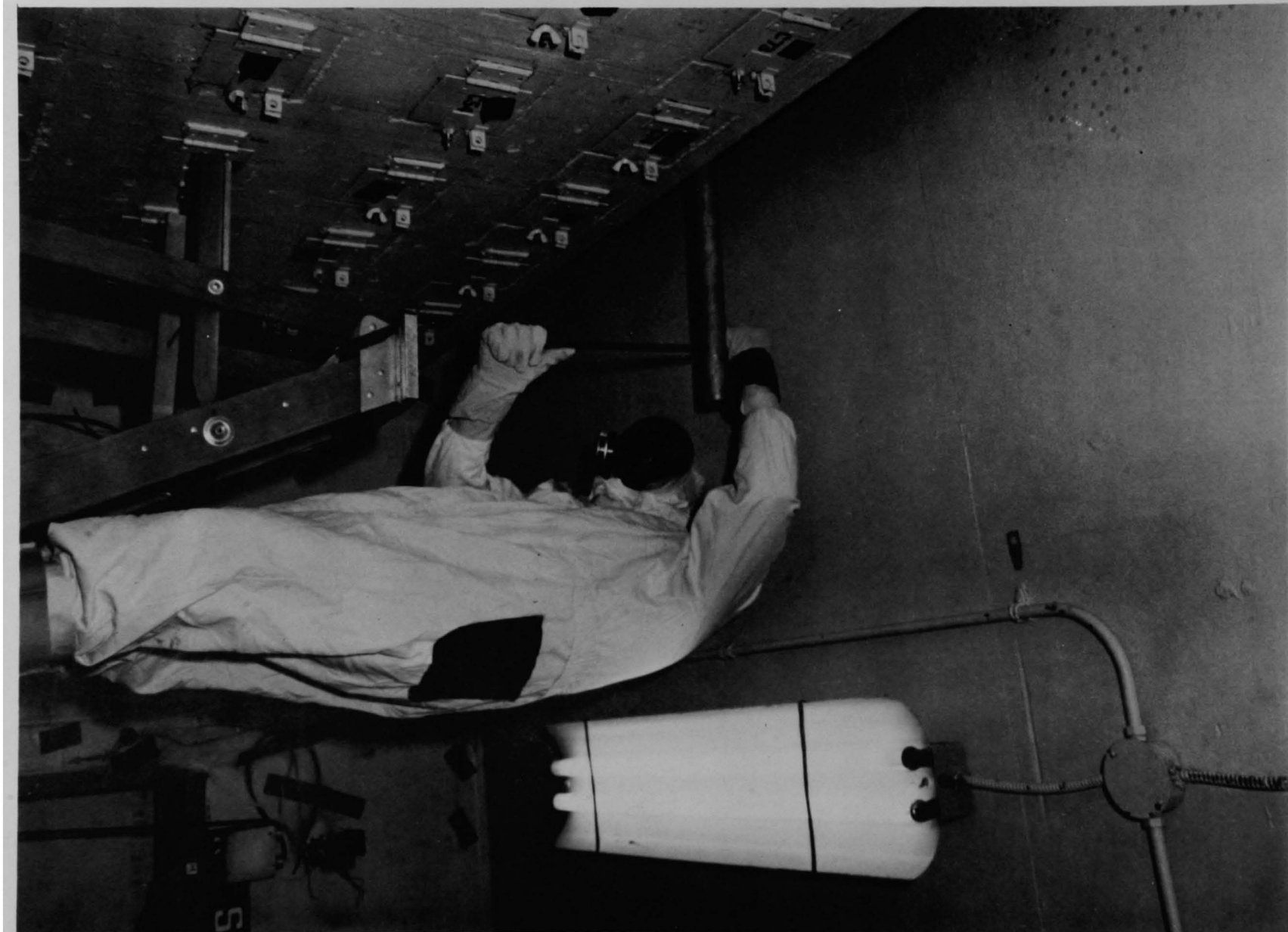
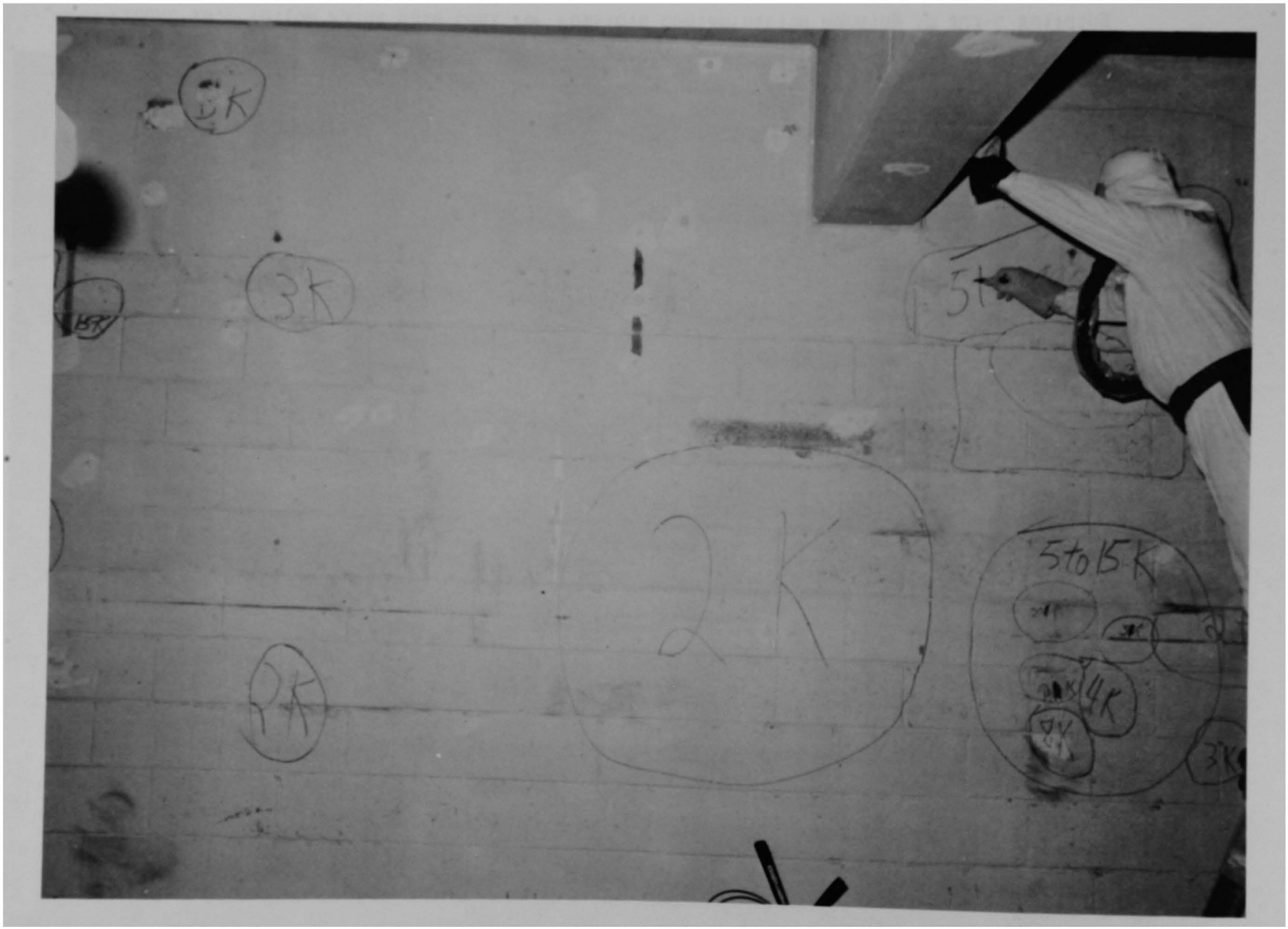


FIGURE 12. Cheese Cloth and Pipe Assembly for Decontamination of Difficult Access Areas in the 303-C Building




1K
3K
2K
5 to 15K
4K
3K

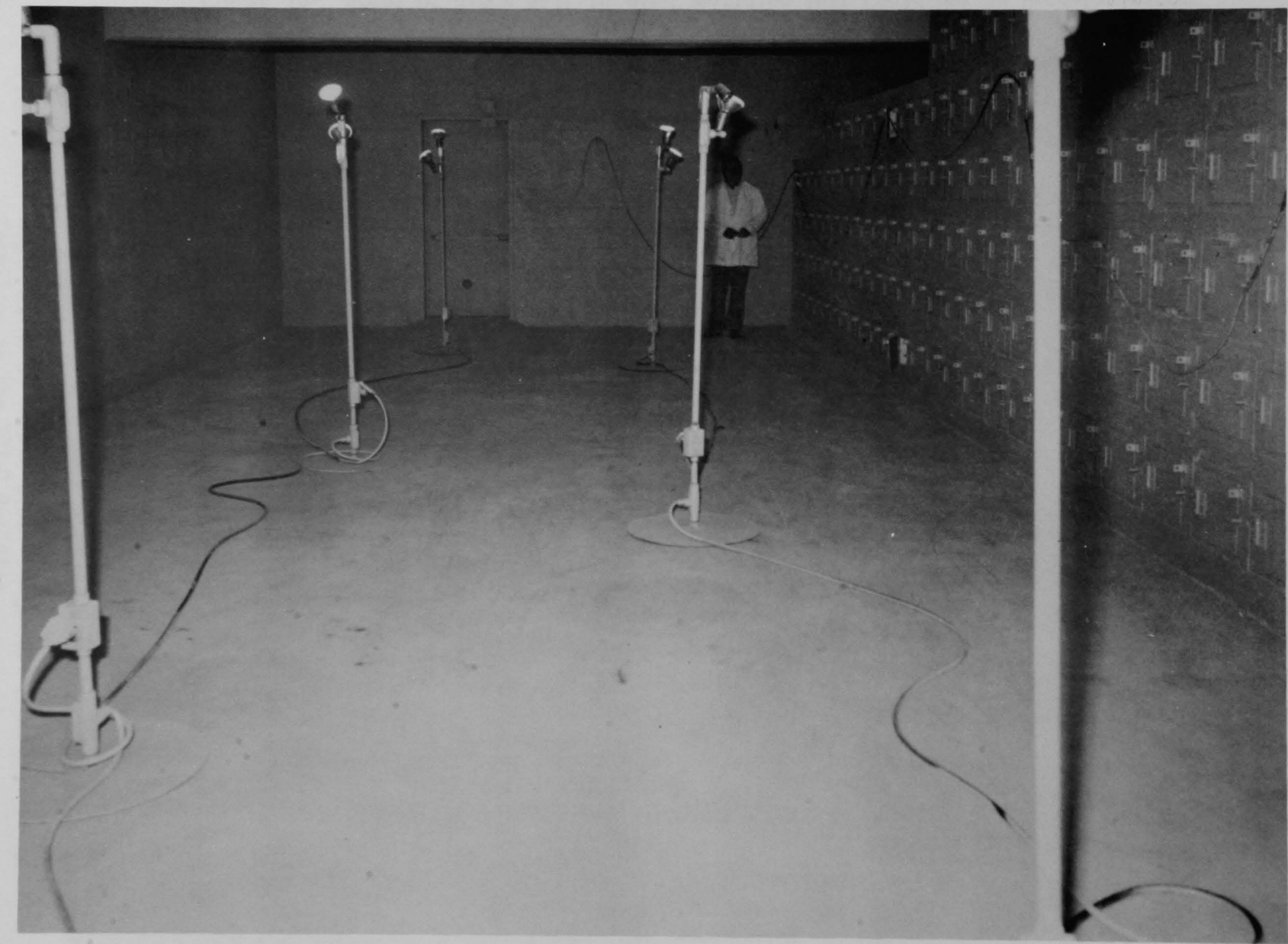


FIGURE 16. Yellow Paint Base Coat for Possible Contamination Warning in 303-C Building

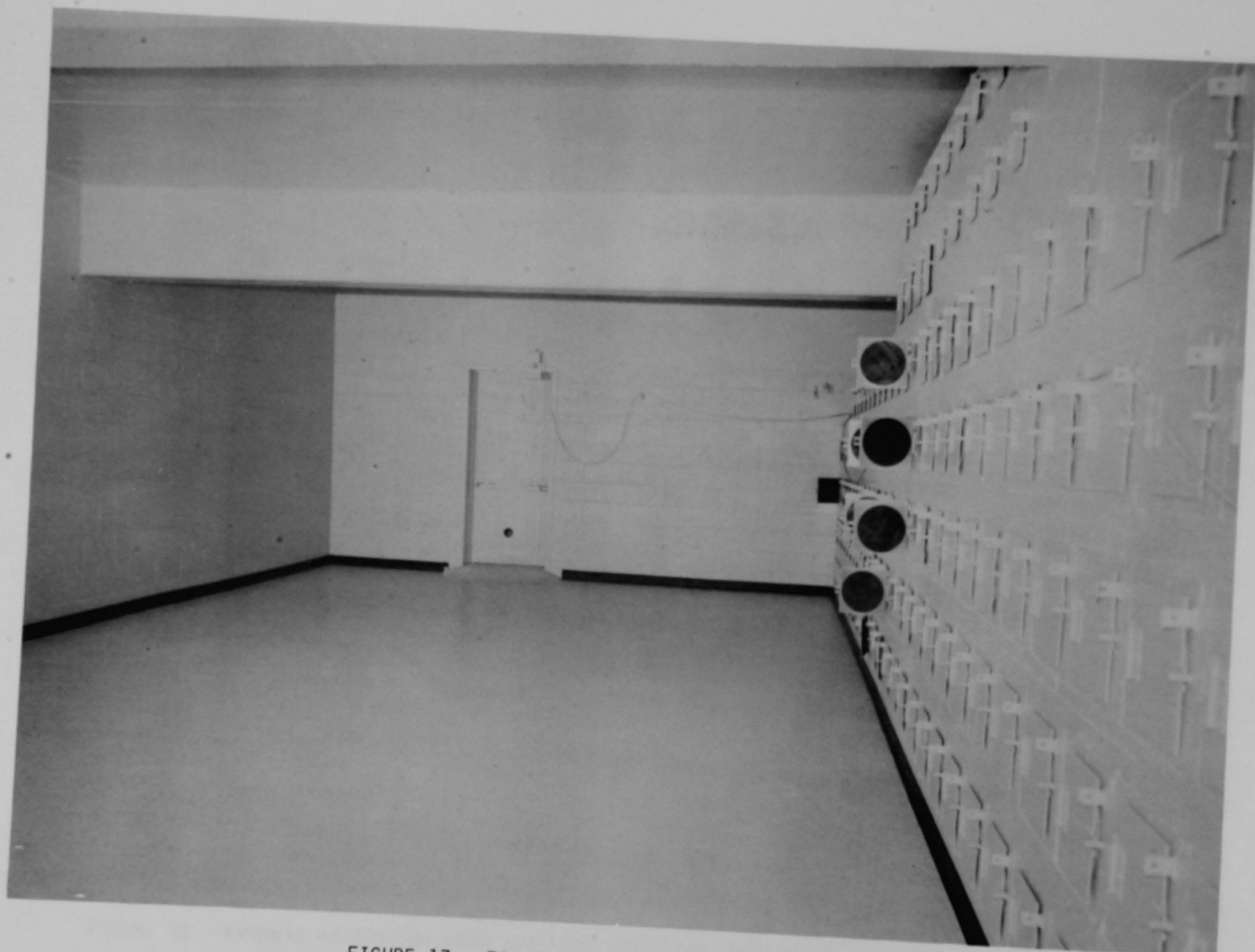


FIGURE 17. Final Painting of 303-C Building

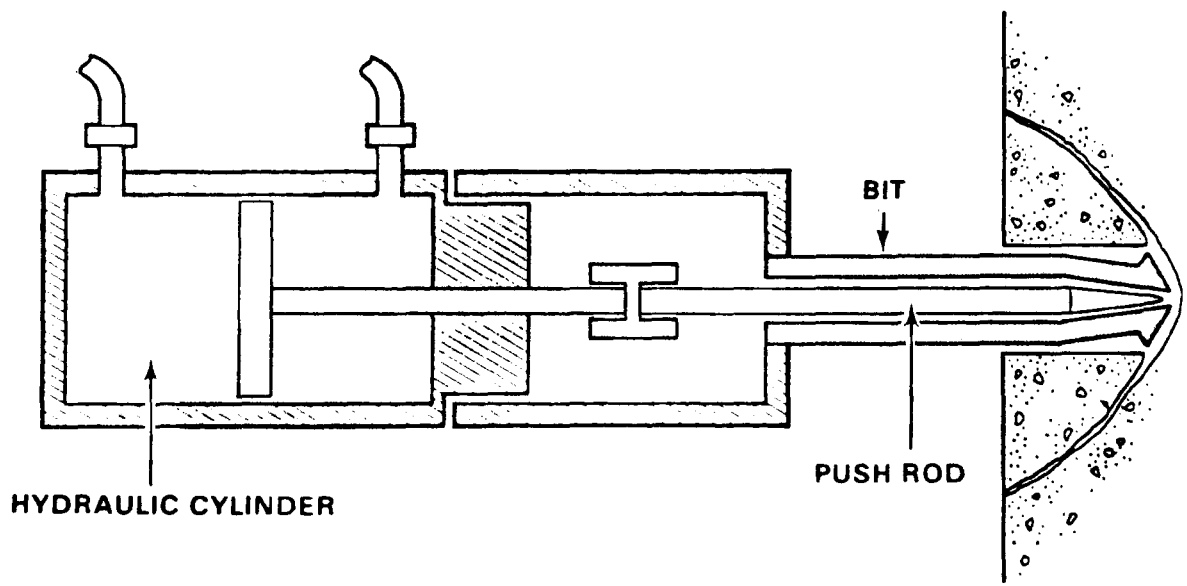


FIGURE 18. Concrete Spaller

FIGURE 19. Example of the Concrete Spalling Technique Used in 303-C Building Decontamination

FIGURE 20. Expanding Bit Used in Concrete Spalling Technique

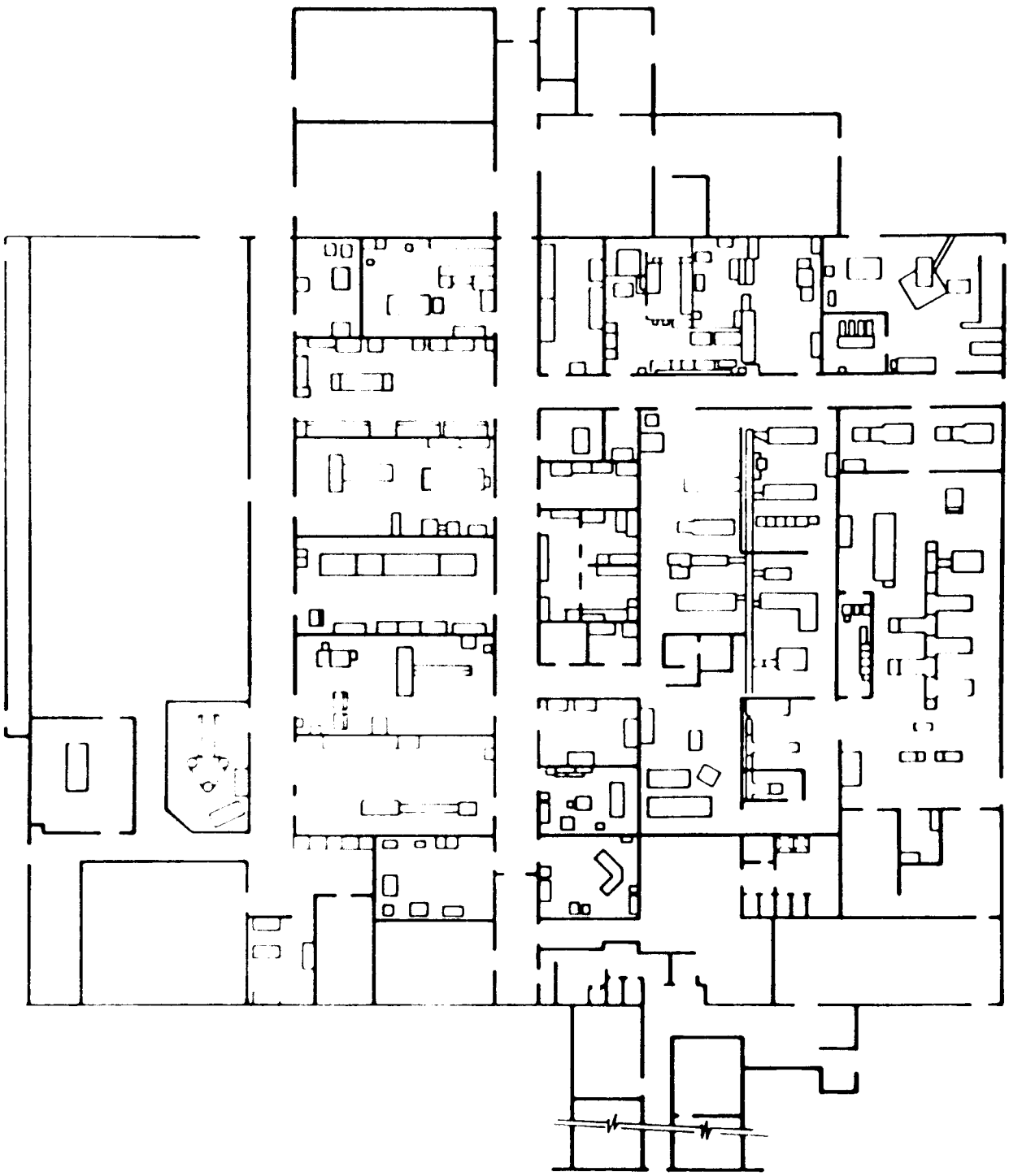
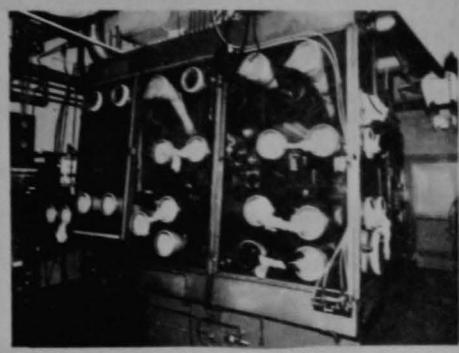
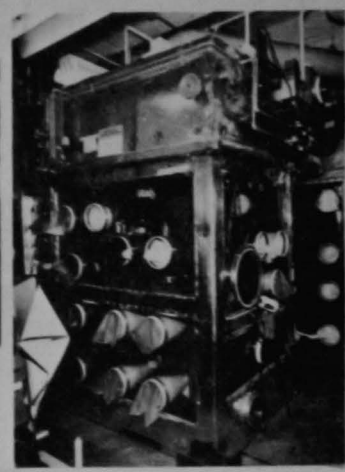
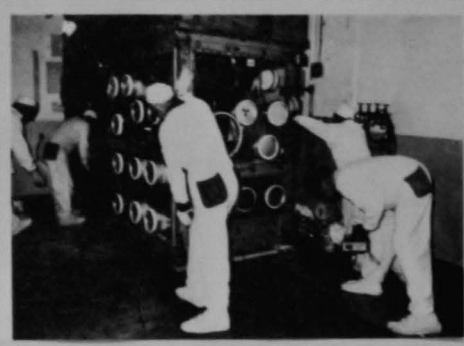


FIGURE 21. First Floor Arrangement of 231-Z Building

231-Z FACILITY DECONTAMINATION AND RESTORATION







REMOVE GLOVEBOXES

PACKAGE FOR RETRIEVABLE STORAGE

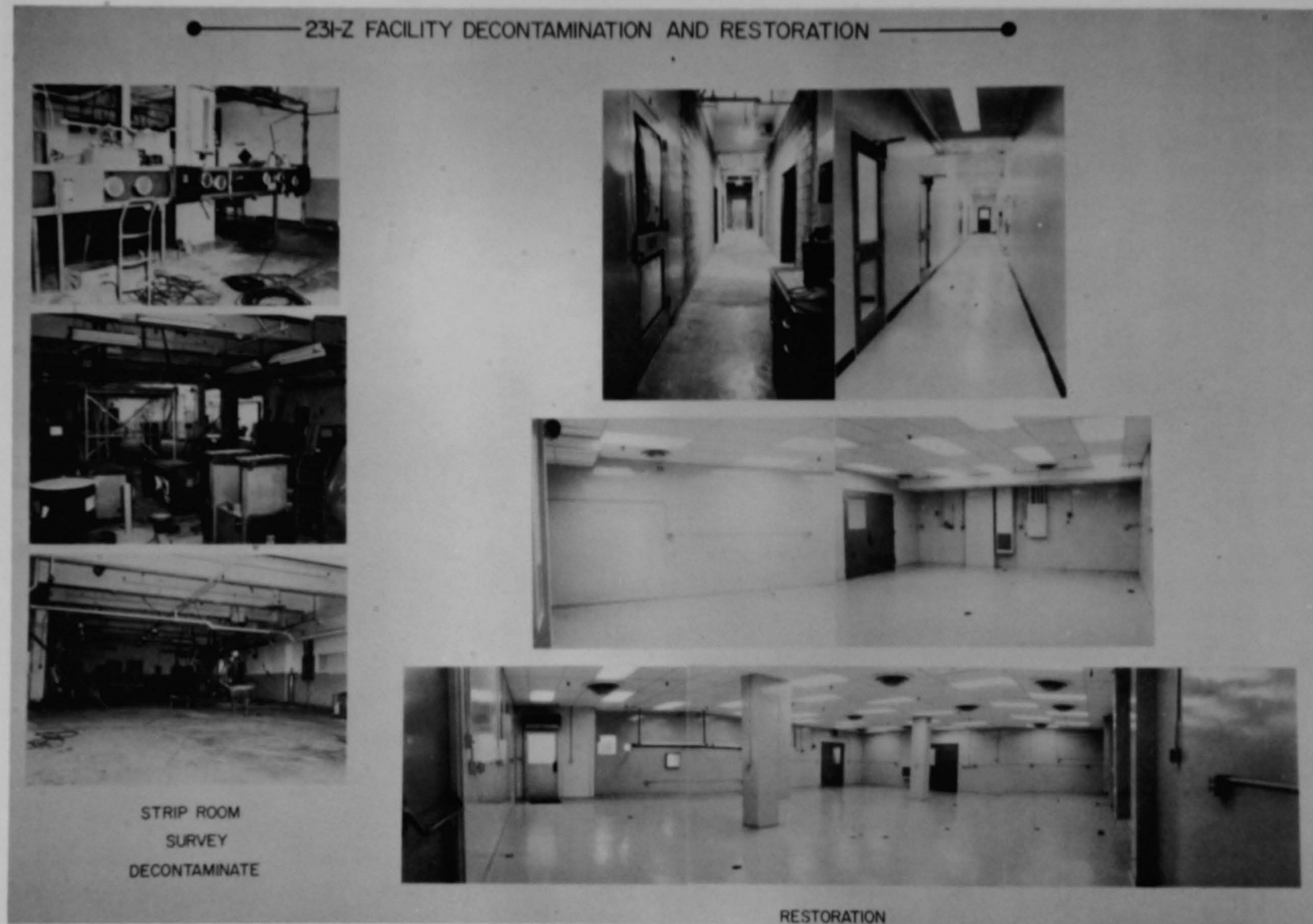


FIGURE 23. 231-Z Building Decontamination and Restoration

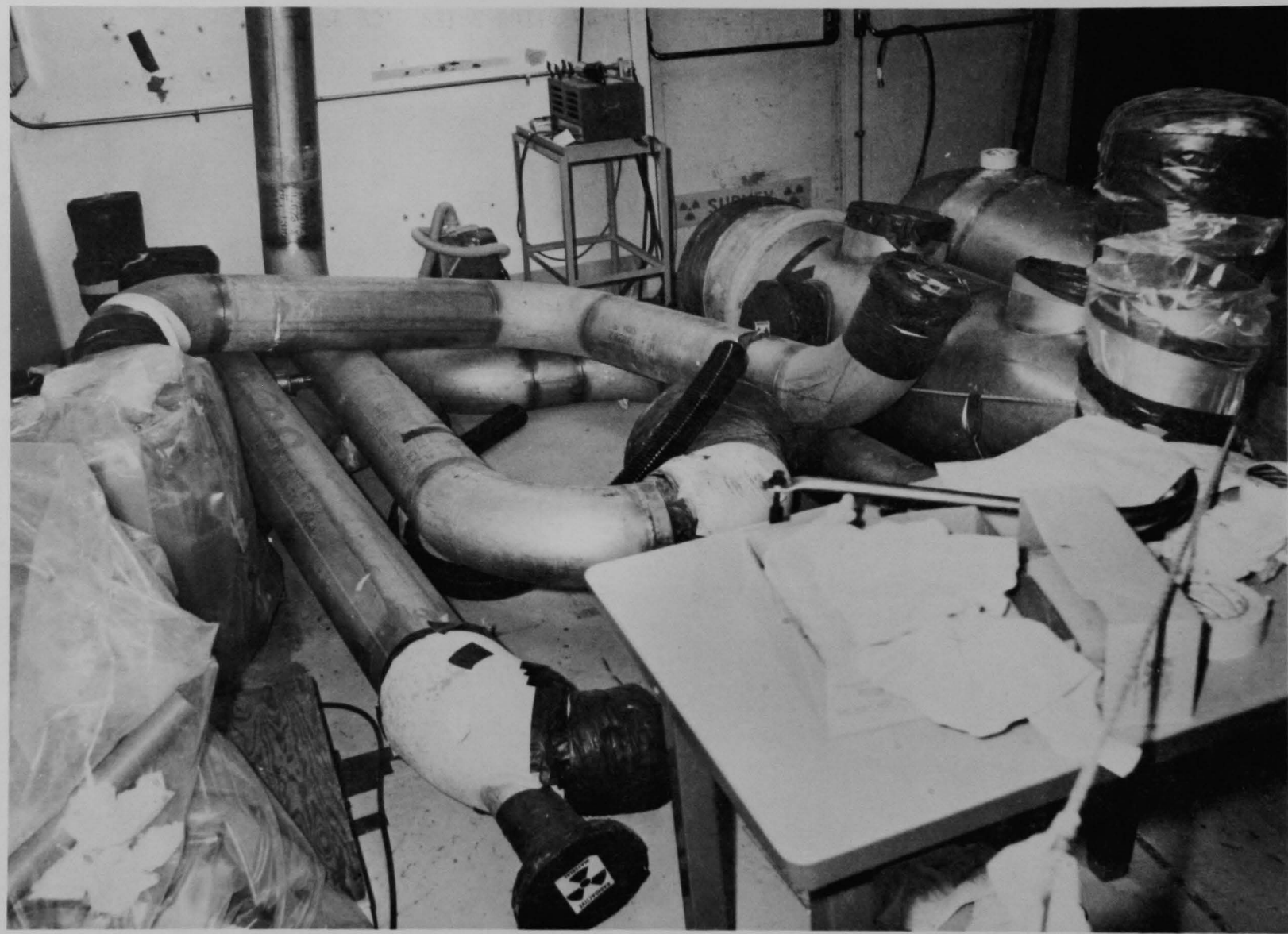


FIGURE 24. Contaminated Piping, Ducting, and Ventilation Equipment from 231-Z Building

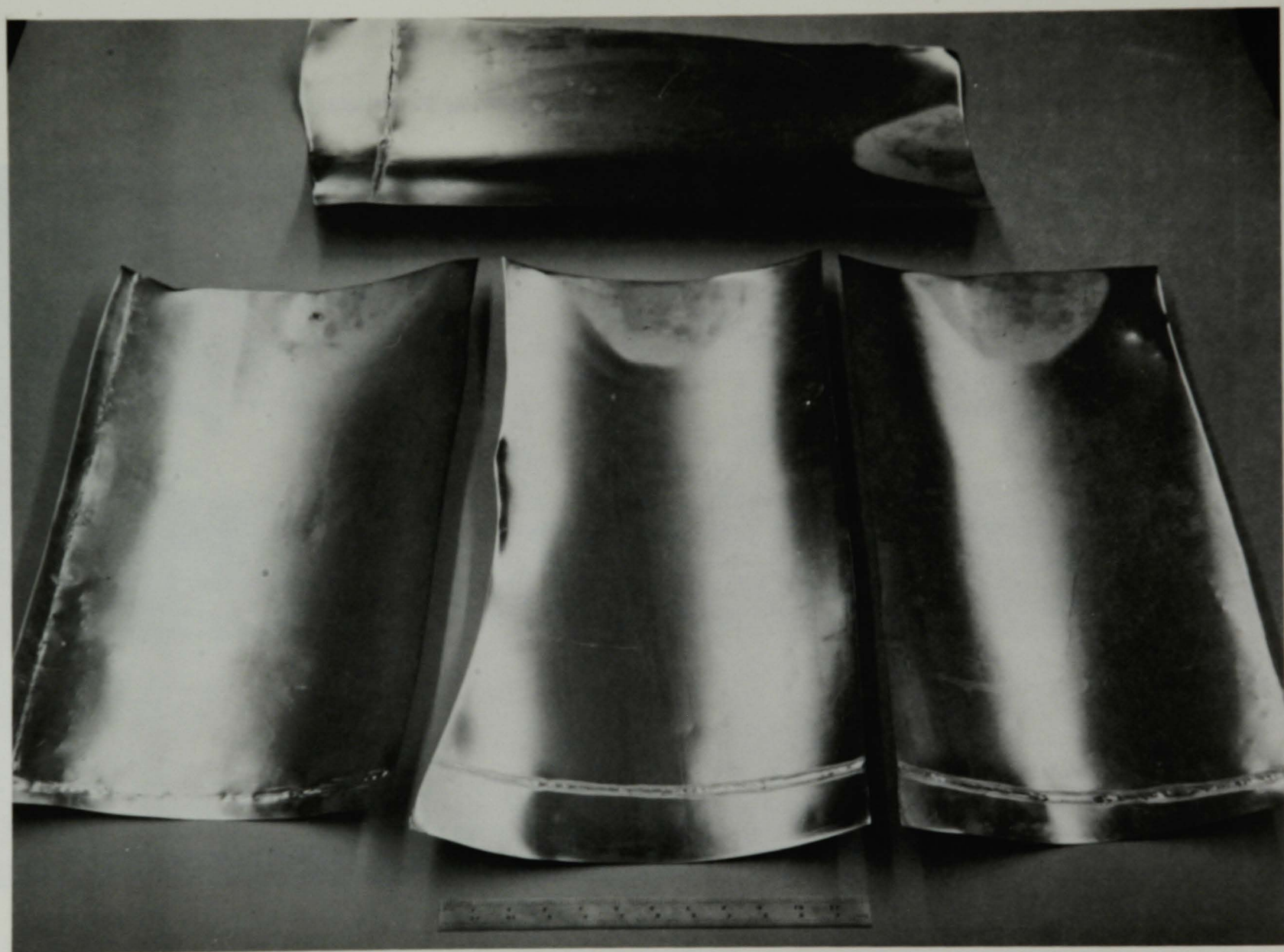


FIGURE 25. Electro-Polished Material from 231-Z Building

FIGURE 26. Contaminated Material from 231-Z Building Packaged for Retrievable Storage

Session U

PRTR RUPTURE LOOP DECONTAMINATION

by

Lyle D. Perrigo

Battelle Pacific Northwest Laboratories

I want to talk to you very briefly about a situation we encountered at the Pacific Northwest Laboratory (PNL) in 1965 in the operation of the Plutonium Recycle Test Reactor (PRTR). This experience is relevant to our discussions about reactor decontamination.

One of the parts of the PRTR was a rupture loop with a volume of about 200 gallons. It was used to test the rupture behavior of experimental fuel. On one occasion in 1965 as we started up the reactor with a mechanically defected and previously irradiated UO_2-4% PuO_2 fuel element, we experienced a sizable rupture. At the time of failure the centerline portion of the fuel was molten. The rupture proceeded rapidly and the molten fuel cut through the Zircaloy pressure tube containing the fuel element and loop coolant. About 1 kilogram of rupture debris was discharged into the loop and other parts of the reactor through the pressure tube opening.

I won't discuss the cleanup of the containment system and any parts of the reactor other than the rupture loop. Those other systems were cleaned up by procedures similar to those described by others at this workshop. However, it should be of interest to discuss briefly what was done to remove the UO_2-PuO_2 rupture debris from the loop.

We found that most of the rupture debris in the loop was located on the baffles of 10 vertical tube heat exchangers. Some was found in deadlegs and other low spots in the loop. The debris was finely divided and could be compared in size and texture to rough sand. The characteristics of this material may be of interest to those concerned with the decontamination of the TMI-2 if significant amounts of rupture debris were discharged from overheated fuel into the primary system.

Our first attempts at decontamination were to mechanically dislodge the rupture debris by manipulation of loop flow. Screens and filters were installed to collect the debris. All of these efforts were unsuccessful. These efforts paralleled or followed cleanup orerations elsewhere in the reactor and the decontamination of the primary system that was undertaken during this extended outage to reduce high radiation levels resulting from the buildup of activated corrosion products.

About a year after the rupture, the loop was chemically decontaminated using an OPG solution. The exact formulation and details of these cleanup operations were described in the book by Ayres entitled Decontamination of Nuclear Reactors and Equipment. Basically the OPG solution was composed of hydrogen peroxide, oxalic acid, oxalates, gluconic acid, gluconates and a peroxide stabilizing reagent. The solution was about 5% by weight OPG and was used at $80^{\circ}C$.

The decontamination was undertaken in such a way that parts of the loop were isolated from each other. This approach was found to be extremely useful in limiting the amount of material to be dissolved during any particular step of the cleanup operation. During the first part of the operation a greater volume of the loop was filled with OPG that had been intended. The activity in the solution rose dramatically. The solution was removed from the loop quickly to avoid waste disposal problems. This event could be significant at TMI-2 if rupture debris with similar characteristics is found in the primary system. Finely divided debris with a large surface area will dissolve rapidly in OPG. This type of problem is to be contrasted with the concern often encountered in decontamination of the rate being sufficient to avoid long solution treatment times.

The chemical decontamination of the rupture loop was completed in less than 48 hours. The "hottest" spot before treatment was 200 R/hr. Following decontamination, readings at that site were 25 mR/hr; other post decontamination radiation readings were of comparable levels.

There were several lessons that we learned from the rupture loop decontamination or similar but earlier cleanup operations at PRTR. These were:
Different types of operations and people are required for decontamination. Operation shifts from power generation to chemical processing. Procedures, organization, safety and control processes must be shifted accordingly.

Successful decontaminations are the result of meticulous planning and training. Even experienced personnel must go through the cleanup procedure step by step in mock runs prior to decontamination to keep problems to a minimum and ensure a successful and efficient operation.

Good communications are mandatory for successful decontamination operations. Thorough records should be kept on each activity so that the factors that lead to successes and difficulties can be quickly identified and exploited or avoided.

Remote TV monitors are an excellent means for following operations inside containment.

Session V

DECONTAMINATION EXPERIENCE AT WEST VALLEY, NEW YORK

W.H. Lewis, NFS

My subject for today is decontamination experience at the West Valley reprocessing plant. However, before I discuss decontamination, I would like to make a few comments about the plant. The West Valley plant was the first industrial reprocessing plant to be built in the United States. The plant employs the so-called chop-leach process for head-end treatment of fuel. In this process, fuel is shared into small segments of one to two inch lengths and then the fuel is dissolved from the cladding with nitric acid leaving the fuel cladding as solid waste. After the dissolution, the nitric acid solution containing the uranium and plutonium is submitted to solvent extraction for purification and separation of the uranium and plutonium from the fission products. The plant produced uranyl nitrate and plutonium nitrate solutions. Waste was concentrated and stored as a neutralized waste solution.

The maintenance philosophy used at the West Valley plant was a combination of remote maintenance and direct maintenance. If you are familiar with reprocessing, there's two concepts in this country. I believe the Idaho Chemical Plant was the first completely direct maintained type plant. Most of the other production plants operated by the government are remotely maintained

plants. In a direct maintenance plant, you're betting that you can decontaminate your equipment that fails and replace it and have a good operating continuity. Therefore, decontamination became very important in the planning stages of the West Valley plant. The solvent extraction portion of the plant was a direct maintenance type plant. The head-in facility (fuel shearing), the dissolution and the waste evaporation was a remote type plant, where if equipment failed it could be replaced remotely using cranes and power manipulators. Since a large portion of the plant (solvent extraction section) was a directly maintained plant, it was necessary to develop our decontamination program during the design of the plant.

Figure 1 shows some of the things that were considered important in the design of the plant especially in the direct maintenance portion of the plant. No valves or pumps were located in the process cells. Cell floors were lined with stainless steel. Cell walls were lined with stainless steel up to 18 inches from floor. Process piping was provided so that decontamination solutions could be added to each tank to facilitate rapid decontamination. Solution transfer from one tank to another was accomplished primarily by installed steam ejectors. Heating and cooling was installed on all all tanks in the process to facilitate decontamination. In our origi design concept in 1962, each cell was equipped with a spray nozzle system which permitted remote washdown of the cells and equipment, thereby minimizing personnel exposure to radiation during plant decontamination.

Planning a decontamination program is the most important part of decontamination because it generally takes more time to plan a good program than it does to do the work. Every detail has got to be planned so that there is no loss of time once a man enters the cell, because in all cases, he's going to be working under restricted radiation conditions. Figure 2 shows a list of some important onsiderations for planning a good decontamination program. I think everyone has discussed radiation control and I will not elaborate. We are concerned with external exposure; this is determined by film dosimetry. Internal exposure is controlled mostly by supplying the worker with fresh air for breathing. We believe a fresh air purged face mask or an air-purged plastic suit affords the best protection against internal exposure.

The next item to consider in organizing a decontamination program is decontamination methodology to be used. A decision must be made on the feasibility of performing decontamination by either hands-on (contact) or hands-off (remote) techniques. Generally the radiation background in the area where work is to be done determines to a large extent the methodology to be used. The hands-on or contact method is usually applicable to areas where low radiation level exist, and the remote method is used in areas of high radiation background.

Another important part of a decontamination program is selection of decontaminating reagents. One of the important considerations is the compatibility of the solutions with the

materials of construction. The reprocessing plant is a nitric acid base system. Therefore, all of the equipment in our plant is constructed of stainless steel. The solutions must also be compatible with the waste treatment system. If decontamination waste is to be evaporated, you should be sure that there are no chemicals in the waste solutions that would cause an explosion.
We have a policy that before we'll use a commercial product in our decontamination program, we must have the chemical composition of that solution from the vendor. If we cannot get this information, we don't use the product because we will not take the risk of an explosion in our plant. If you use an ion exchange system for recovering fission products from decontamination waste, you must be sure the decontaminating reagents do not contain chemical complexing agents which interfere with ion exchange recovery. Occasionally, a very small amount of chemical complexing agents in a waste solution will affect drastically the recovery of fission product by ion exchange technology.

We have found high pressure spray systems to be very effective in removing large amounts of contamination. Generally, our first approach is to use a high pressure water spray system for grass decontamination. If necessary, a high pressure chemical spray is used next. After removing as much as possible of the contamination with high pressure spraying, then decontamination is usually finished using contact methods. In a chemical plant all waste solutions are generally concentrated by evaporation, which produces a condensate that can be released and a concentrate containing the radioactivity that is stored as liquid waste until converted to a solid form for disposal.

Concrete surfaces are most difficult to decontaminate. Generally, all concrete surfaces which are likely to become contaminated should be covered with a chemical resistant paint. Once contamination gets into concrete, it becomes necessary to remove a layer of the concrete surface by some technique such as chipping or sand blasting.

Equipment removal and packaging is something overlooked in most planning programs. It becomes very important to plan exactly how you are going to disconnect plant equipment and remove the equipment from the cell. The method selected depends on the radiation background in the area where work is to be done and on whether or not like-equipment is to be used as a replacement. Pipe cutting can be done manually or it can be done remotely. We've done both and we prefer the remote operation in cells that exceed 500 mr/hr. Of course, use of the cutting torch is usually the quickest way but sometimes complicates reinstallation of equipment.

During any decontamination program, it is necessary to provide a plan for determining progress on a timely basis. In a chemical plant where it is necessary to remove radioactivity from a tank, the decontamination progress is usually monitored by radiochemical analyses of the decontamination reagents. After the radioactivity in the decontamination solution indicates a leveling-off, radiation surveys are usually made with gramma instrument to determine local high radiation areas. Surface contamination outside of vessels can be monitored by air sampling

and by smear techniques once the radiation level in the area is reduced to permit personnel entry. You need analytical facilities to analyze samples for specific fission products. It is desirable to know what radionnuclides the decon-solution has removed because it may be necessary to use specialized decontamination reagents for specific radionuclides.

The next slide (Fig. 3) shows a list of standard decontamination reagents that are used in the plant and specific application of each chemical reagent.

These are "homemade" solutions and the recipe for the makeup of each chemical solution is shown in the next slides (Fig. 4A and 4B) We do use commercial decontaminants under controlled conditions. For decontamination of stainless steel vessels, the nitric acid fluoride solution is used only in special cases and after other treatments have failed to achieve the desired result.

The most frequently used decon-solutions are types I and II as shown in Figure 4A. We highly recommend that all radiochemical plants have a list of approved decontamination reagents which can be used as required. Approval of this list of chemical solution for use in the plant should be the responsibility of the plant safety committee and no deviation should be permitted without the safety committee's approval. The advantages in having an approved list of decon-solutions is that it saves time and it prevents the use of harmful and dangerous chemicals without due consideration by management.

During the next few minutes I would like to discuss briefly the results achieved on the two major decontamination programs that have been completed at West Valley. The two programs in (1) the clean up of the Fuel Storage Pool and (2) the decontamination of the solvent extraction areas of the plant to permit personnel entry for major equipment modifications.

In the late 1960's the water in the Fuel Storage Pool became contaminated excessively with radioactive cesium and an investigation revealed the source of radioactivity to be caused by the leaching of cesium from uranium metal fuel that had oxidized while in storage. To reduce the radioactivity in the water to a tolerable concentration, it was necessary to remove all of the oxidized uranium from the pool. This was accomplished by suspending the uranium fines into the water with high pressure spray agitation and pumping the resulting slurry through filters to remove the solids. Ion exchange was used to remove the radioactivity from the water after the solids had been removed by filtration. Some algae containing radioactivity had collected on the walls of the pool and this material was removed semi remotely using long handled scrub brushes and high pressure spraying. Clean up was accomplished in three months and the pool has been in use for seven years since the cleanup and the concentraion of radioactivity in the pool water has remained 10^3 lower than before cleanup.

The other major decontamination program which I would like to mention briefly, involved the decontamination of all the solvent extraction equipment in the Plant to permit major equipment modifications using direct maintenance technology. This decontamination

program was completed in about six months and the radiation reading in the cells were reduced from about 5000 r/hr to 50-100 mr/hr. The decontamination was accomplished using water and nitric acid solutions to flush the loose activity from the equipment. The residual radioactivity remaining after the flushing operation was then removed using the chemical agents which was mentioned earlier.

Since a detailed discussion of the decontamination program would be too time consuming for this meeting, I have decided to discuss briefly a typical decontamination procedure for a process tank and cell. The procedure (Fig. 5) for the decontamination of a process tank is designed primarily to remove radioactive contamination from the inside surfaces of the tanks. Gross removal radioactivity was accomplished by flushing the tank with dilute nitric acid and water until the fission product concentration in the solution levels off. After this happens, then a chemical solution, Type I is added; heated and refluxed, if possible, until repetitive radiochemical analyses indicate no further removal of fission products. After removing the Type I solution, the tank is rinsed with water and Type II solution is added, heated and held in the tank until radiochemical analyses indicates a leveling-off of radioactivity in the solution. After removing Type II solution from the tank and rising with water, a radiation survey is made to determine if further treatment is required. If further treatmen is required, the same procedure is usually tried at least once again before using other chemical decontamination agents.

The decontamination of a process cell is designed to remove radioactivity from the outside surfaces of the tanks and walls of the process cells. Figure 6 shows our stepwise approach again, water is applied (batchwise) using the in-cell spray system and after a batch of water is generated, the water is sampled, analyzed and evaporated to reduce the waste volume. This procedure is repeated until the radiochemical analyses of the wash water indicates no further removal of radioactivity. At this point surfaces are allowed to dry and smears are taken to determine the effectiveness of the treatment. Chemical decontamination agents can also be used if necessary, however, you must be sure that all materials of construction in the area are compatible with the chemical agent to be used.

FIGURE 1

DECONTAMINATION PLANNING IN DESIGN

* NO VALVE OR PUMPS IN PROCESS CELLS.
* CELL FLOORS LINED WITH S.S.
* CELL WALLS LINED WITH S.S. UP TO 18 INCHES FROM FLOOR.
* ENTIRE CELL LINED WITH S.S. IN SPECIAL AREAS.
* DECONTAMINATION SOLUTION PROVIDED TO EACH TANK FROM MAKE UP AREA.
* SOLUTION TRANSFER FROM TANK TO TANK.
* HEATING, COOLING AND SAMPLING.
* CELL SPRAY SYSTEM.

FIGURE 2A

DECONTAMINATION PLANNING CONSIDERATION

* PERSONNEL RADIATION EXPOSURE CONTROL:
  - -EXTERNAL EXPOSURE
  - -INTERNAL EXPOSURE
  - -TRAINING

* DECONTAMINATION METHODOLOGY:
  - -HANDS ON - CONTACT
  - -HANDS OFF - REMOTE

* SELECTION DECONTAMINATION REAGENTS:
  - -SOLUTION MUST BE COMPATIBLE WITH MATERIAL OF CONSTRUCTION.
  - -SOLUTION MUST BE COMPATIBLE WITH WASTE TREATMENT SYSTEM.
    - (a) EXPLOSION
    - (b) COMPLEXING AGENTS

FIGURE 2B

* EQUIPMENT SELECTION:
  - -HIGH PRESSURE - LOW VOLUME SPRAY.
  - -HIGH PRESSURE - HIGH VOLUME.
  - -CONCERTE SURFACE REMOVER.
    - *SCABBER
    - *SAND BLAST
    - *CHIP HAMMER

* EQUIPMENT REMOVAL AND PACKAGING:
  - -PIPE CUTTING
    - *REMOTE SAW
    - *TORCH
    - *SHEAR

*MONITORING PROGRESS:
  - -SAMPLES
  - -RADIATION READING
  - -SMEARS

GENERAL DECONTAMINATION AGENTS

Surfaces		Solution	Remarks
Stainless Steel Vessels	High Radiation Levels (10-200 R/hr) MFP	Sodium Tartrate	Used heated to 150-170° F. Good in vessels that had contained solvents.
Stainless Steel Vessels	High Radiation Levels - MFP	Nitric Fluoride	To remove plated-out contamination.
Stainless Steel Vessels	Medium to High Radiation Levels. 1-50 R/hr. MFP.	Citric/ Nitric	Sample results showed major isotopes as Co & Cs, minor SbTe. With residence of 8-12 hrs heated, SbTe major, Cs & Co minor.
Stainless Steel Vessels	High Radiation Levels - 10-200 R/hr. MFP.	Type 1 & Type 2 Solutions	Type 1 showed high tendency to plug lines. Used at 1/2 strength was still effective. Less plugging.
Stainless Steel Floors, Carbolene-Coated Floors & Walls, Unpainted Concrete	MFP. 1-50 R/hr Radiation Levels.	Sodium Tartrate	Used with hydrobrush and as a high volume spray. Good scrubbing solution. Used heated to ~150° F. Good on dirty, greasy surfaces.
Carbolene-Coated Concrete	MFP. 1-10 R/hr Radiation Levels.	Citric/ Nitric	Alternated with Sodium tartrate.

FIGURE 3

FIGURE 4A

SOLUTION MAKEUP PROCEDURE

Type I Solution

For each 1,000 liters of decon solution to be made up proceed as follows:

a) Add 500 liters of H_2O at 180°F.

b) Turn on agitator.

c) Add 200 liters of NaOH (Slowly).

d) Add 100 lbs of Potassium Permanganate (Optional).

e) Add 55 lbs of Potassium Dichromate (Optional).

f) Add H_2O to bring final level to 1,000 liters. Add the final H_2O at as high a temperature as possible in order to bring the final temperature to between 180° - 200°F.

Type II Solution

For each 1,000 liters of decon solution to be made up proceed as follows:

a) Add 700 liters of H_2O at 150°F.

b) Turn on agitator.

c) Add 180 lbs of oxalic acid (Slowly).

d) Add 10 lbs of citric acid (Optional).

e) Add 10 lbs of tartaric acid (Optional).

f) Add 8 lbs of "NTA".

g) Add H_2O to bring the final level to 1,000 liters. Add the final H_2O at a temperature sufficient to bring the final solution temperature to 150° - 170°F.

FIGURE 4B

Nitric Acid/Fluoride Solution

For each 1000 liters of decon solution to be made up proceed as follows:

a) Add 870 liters of H_2O at approx. 150°F.

b) Turn on agitator.

c) Add four (4) pounds of NH_4F.

d) Add 130 liters of 12M nitric acid.

Sodium Tartrate Solution

a) 880 liters of water

b) 120 liters of 18M HaOH

c) 20 kgs of tartaric acid
Heat to 150°-170°F

Citric/Acid Solution

a) 800 liters of water

b) 108.5 lbs of citric acid

c) 20 liters of 15M nitric acid (Optional)
Heat to 150°F

FIGURE 5

DECONTAMINATION PLAN FOR S.S. PROCESS VESSEL

* FLUSH TANK WITH NITRIC ACID SOLUTIONS UNTIL F.P. CONC. IN TANK LEVELS OFF.
* RINSE TANK WITH WATER.
* ADD TYPE II SOLUTION - HEAT TO 200°F.
* RINSE WITH WATER.
* ADD TYPE II - HEAT AND SAMPLE.
* REVIEW PLAN DATA AND DECIDE.

FIGURE 6

DECONTAMINATION PLAN FOR REMOVAL OF IN CELL SURFACE CONTAMINATION

* USING IN CELL SPRAY SYSTEM -
  SPRAY UNTIL 1000 GALLONS IS RECORDED ON SUMP LEVEL DETECTOR.
* SAMPLE AND TRANSFER WASTE TO EVAPORATION.
* REPEAT UNTIL SAMPLE INDICATES LITTLE OR NO FURTHER REMOVAL.
* LET CELL SURFACES DRY - SMEAR.
* DECIDE IF CHEMICAL TREATMENT IS REQUIRED.

Session W

DECONTAMINATION EXPERIENCE AT THE OAK RIDGE NATIONAL LABORATORY

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*Operated by Union Carbide Corporation under contract W-7405-eng-26 with the U.S. Department of Energy.

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Introduction*

We have selected two major facilities at the Oak Ridge National Laboratory (ORNL) as a basis for this discussion on decontamination: the Radiochemical Processing Pilot Plant (Building 3019), and the Multicurie Fission Product Pilot Plant (Building 3517), wherein large quantities of fission products were handled in the course of our work.

It is interesting to note at this point that our experiences at ORNL in this area of decontamination seem to follow the same general pattern as outlined by other speakers in this session.

Radiochemical Processing Pilot Plant Experience

Over the past 30 years, the Radiochemical Processing Pilot Plant has been demonstrating flowsheets employed in the reprocessing of irradiated fuel. As flowsheets were adequately demonstrated, systems would be redesigned and the equipment fabricated and inserted within the remote processing cells. Accordingly, decontamination programs were undertaken to allow operating personnel and maintenance forces to enter the directly maintained cells to prepare for the new flowsheet. Flowsheets that were demonstrated with irradiated fuel in this facility included Purex, Thorex, Volatility, and numerous other processes. Currently, the facility serves as a warehousing and dispensing station for the major inventory of ^{233}U, and operations

*This presentation was condensed at the request of the Session Chairman in order that it not delay a scheduled tour of The Three Mile Island (TMI) facility.

continue to be done remotely because of the unique nature of this isotope. This planning of the decontamination experience program is time consuming. Therefore, as systems have been modified, we have found it necessary to build into our designs such systems that could be easily decontaminated.

The Building 3019 facility is composed of a series of seven processing cells, each 20 x 20 x 27 ft high, that contain the remotely operated equipment. Surrounding the cells is a 5-ft-thick layer of concrete biological shielding. (A cross section of the facilities, coupled with the contamination levels experienced during an incident which took place on November 20, 1979 is presented in Fig. 1.) Because there may be some phases of the cleanup operations which may be applicable to TMI, perhaps some discussion of the events leading up to the incident which caused this contamination level may be beneficial.

Following the completion of a program phase, a decontamination procedure was initiated where proprietary decontamination reagents were being used to reduce the internal fission product inventory within the process vessels. Suitable water and mineral acid flushes were also utilized as required. In the course of these operations, an intercycle evaporator formerly used to boildown the uranium/plutonium solution containing fission products, was being decontaminated using the flush procedures. Following the use of the proprietary reagent, a water flush was inadvertently eliminated, and a nitric acid rinse was made at elevated temperature. Because phenol was used in the reagent, the subsequent reaction with nitric acid caused an explosion

in the evaporation vessel. The vapor separators contained in the evaporator system were completely shattered (Fig. 2) and thrown to the floor. Residual plutonium and fission products were then distributed throughout the process cell and the building. Also, because a cell door was blown open, some activity was released to the immediate area outside of the facility. Figure 3 is a schematic representation of the area in and around Building 3019 showing the extent of contamination as determined by a radiation survey.

The decontamination program employed many of the same procedures that have just been outlined by Mr. W. H. Lewis of Nuclear Fuel Services in their West Valley, New York, facility.

Our original step was to place a plexiglass "greenhouse" in the cell doorway to spray down the surfaces and remove gross contamination. Because plutonium was present, a potential critically problem existed, requiring the use of boron as a soluble nuclear poison in all process solutions. Following this intial step, cell entries were made to remove large pieces of debris which represented high sources of radiation. The radiation background in the cell following this step was in the range of 10 to 20 R/hr. Contact flushing was then accomplished with planned exposures and various reagents.

A telescoping elevator was found to be useful in the removal of debris and high-density block shielding from the site of the accident (Fig. 4).

The major problem in the cleanup was the air activity level within the cell as the result of dispersed plutonium. Figure 5 shows the trend of α-activity over the five-month decontamination program.

The lessons learned from this decontamination experience are rather obvious. First, it is imperative that the chemical analysis of decontamination reagents be known well in advance of their use. The use of the term "proprietary" is not sufficient to eliminate the need for understanding the makeup of reagents in an expensive hardware system in need of decontamination. Second, one should be well aware of the downstream effects of the use of decontaminants. We have found in our waste studies that the effectiveness of ion-exchange resins for ^{137}Cs removal is greatly dependent on the ionic phosphate contained in solutions. As is well known, phosphate is a vital constituent in detergents, which are sometimes used in random fashion in decontamination programs. Therefore, in the TMI case, all liquids generated in the decontamination work will require subsequent treatment prior to ultimate disposal.

Multicurie Fission Product Pilot Plant Experience

Since 1948, the Multicurie Fission Product Pilot Plant has been used to produce large quantities of cesium, strontium, and promethium for space and other isotopic power programs. Over this time span, a total of 10 MCi of fission product material has been handled in this facility. Because two of the isotopes handled in this plant are also being encountered in TMI, perhaps a discussion of the decontamination experience in this facility is desirable.

The Multicurie Fission Product Pilot Plant (Fig. 6) is composed of 25 process cells containing equipment for isotope processing. In one-half of the cells, manipulators are used to operate equipment such

as vacuum and pellet presses and other small-scale equipment. Also present is a series of large remotely operated cells that contain solvent extraction, ion-exchange, and crystallizing equipment. The systems that have been used in this facility have been decontaminated and removed, and the entire facility has been decontaminated to an acceptable level.

In this decontamination effort, the major source of activity was removed (500,000 Ci ^{90}Sr) during the early phase of the program. Once this was done, the manipulator cells were decontaminated by flushing the small equipment items with chemical lances guided by the manipulators. The decontaminated items were then bagged into 55-gal drums for disposal. If a component was too large for the disposal package, it was disassembled and cut with remote tools. When the in-cell activity reached <10 R/hr, the upper cell plugs were removed. A 3-in.-thick steel plate was then placed in the plug locations along with 2-in.-thick plexiglass windows that contained a series of hand-holes through which high-pressure steam jets could be utilized. The original radiation level in these cells was in excess of 1000 R/hr. The treatment outlined above reduced the level to less than 1 R/hr.

Information on TMI-Penetration R-401

Although not specifically related to this discussion, I have been requested to give a few comments on the plug cut from TMI Penetration R-401, which is currently being analyzed at ORNL.

The plug cut from TMI penetration R-401 measured 2.8 in. in diameter and was 1.1 in. thick. The surface of this specimen was

covered with a Phenoline-300 series coating within a 10-mil-thick specification.

Following the completion of the analysis of the intact plug, the specimen was cut with eight pie-shaped wedges for a series of experiments involving decontamination procedures. Some of the tests considered of radiation surveys, x-ray spectural analysis, decontamination, dry-film thickness, and scanning electron microscopy on the paint surface. An artist's sketch of the plug and the corresponding sections is presented in Fig. 7.

Radiation surveys of the surface of the plug as measured by TMI and ORNL are compared in Table 1. X-ray fluorescence analysis

Table 1. A comparison of the radiation measurements of the surface of plug from TMI penetration R-401

Type of radiation	Radiation level (mR/hr)
TMI	
Gamma, 4 in. from plug	1.2
Beta-gamma, 4 in. from plug	100
ORNL	
Beta-gamma, shielded at 2 in.	1.5
Beta-gamma, unshielded at 2 in.	600

of the plug surface indicated that the major constituents were aluminum, copper, iron, potassium, silicon, titanium, and zinc.

Dry-film thickness measurements are reported in Table 2. Basically, these measurements were well within the thickness specifications applied during the construction of the reactor containment vessel.

Table 2. Dry-film thickness of paint on plug from TMI penetration R-401

Method	DFT (mils)
Nordson film gauge	10.5 ± 0.5
Tubular micrometer	10.5
Edge photographs	10-11

In performing the decontamination experiments, the Bechtel CP-952 procedure was used. This procedure called for scanning the activity on the surface, washing with water for 10 min at 25°C, washing with oxalic acid for 10 min at 25°C, followed by an elevated oxalic acid temperature treatment (10 min, 80°C). All specimens were air-dried following each step and scanned for activity. Results of these tests (Table 3) indicated that the use of water did not remove activity (DF = 1). Acid treatment with oxalic was also ineffective. Basically, therefore, we conclude that the machining steps used to cut the plug from the penetration resulted in localized high temperatures that caused fusion of the fission products on the surface.

The spectrum of activity on the surface of the plug (Table 4) indicates that the predominant isotopes are ^{129m}Te, ^{127m}Te, ^{137}Cs, and

Table 3. Decontamination factors (DFs) obtained for plug from TMI penetration R-401, using Bechtel Procedure CP-952

Contaminant	DF Water at 25°C	DF acid at 25°C	DF acid at 80°C	Total DF
Ag-110m	1.0	1.02	1.16	1.18
Ce-141	1.0	2.04	2.82	5.74
Ce-144	1.06	2.37	2.25	5.65
Cs-134	1.0	1.24	1.83	2.28
Cs-136	1.0	1.08	2.28	2.46
Cs-137	1.0	1.24	1.85	2.30
I-131	1.08	1.01	1.07	1.17
Nb-95	1.03	1.28	2.34	3.07
Ru-103	1.0	1.20	1.24	1.50
Ru-106	1.33	1.30	1.24	2.15
Sb-125	1.02	1.21	2.76	3.43
Sn-113	1.0	1.83	1.54	2.82
Te-129m	1.02	1.37	2.54	3.43
Zr-95	1.15	1.25	2.46	3.53

Table 4. Isotopic content of painted steel plug from TMI penetration R-401 (as of 0800, August 29, 1979)

Isotope	μCi
^{58}Co	0.032
^{60}Co	0.01
^{95}Zr	0.09
^{95}Nb	1.7
^{103}Ru	0.58
^{106}Ru	0.42
^{110m}Ag	0.080
^{113}Sn	0.24
^{124}Sb	0.005
^{125}Sb	0.45
^{127m}Te	7.8
^{129m}Te	23.6
^{125m}Te	0.5
^{131}I	0.33
^{134}Cs	0.47
^{137}Cs	2.07
^{140}Ba	----
^{140}La	0.019
^{141}Ce	0.057
^{144}Ce	0.24

^{95}Nb. A graph showing the decay of the various fission products plated on this plug as a function of time is offered (Fig. 8) to indicate the residual contamination that will require removal at a future date.

Finally, the preliminary gamma scan of the 9-in. "cookie" recently removed from TMI-R626 penetration is presented (Fig. 9) for comparison purposes.

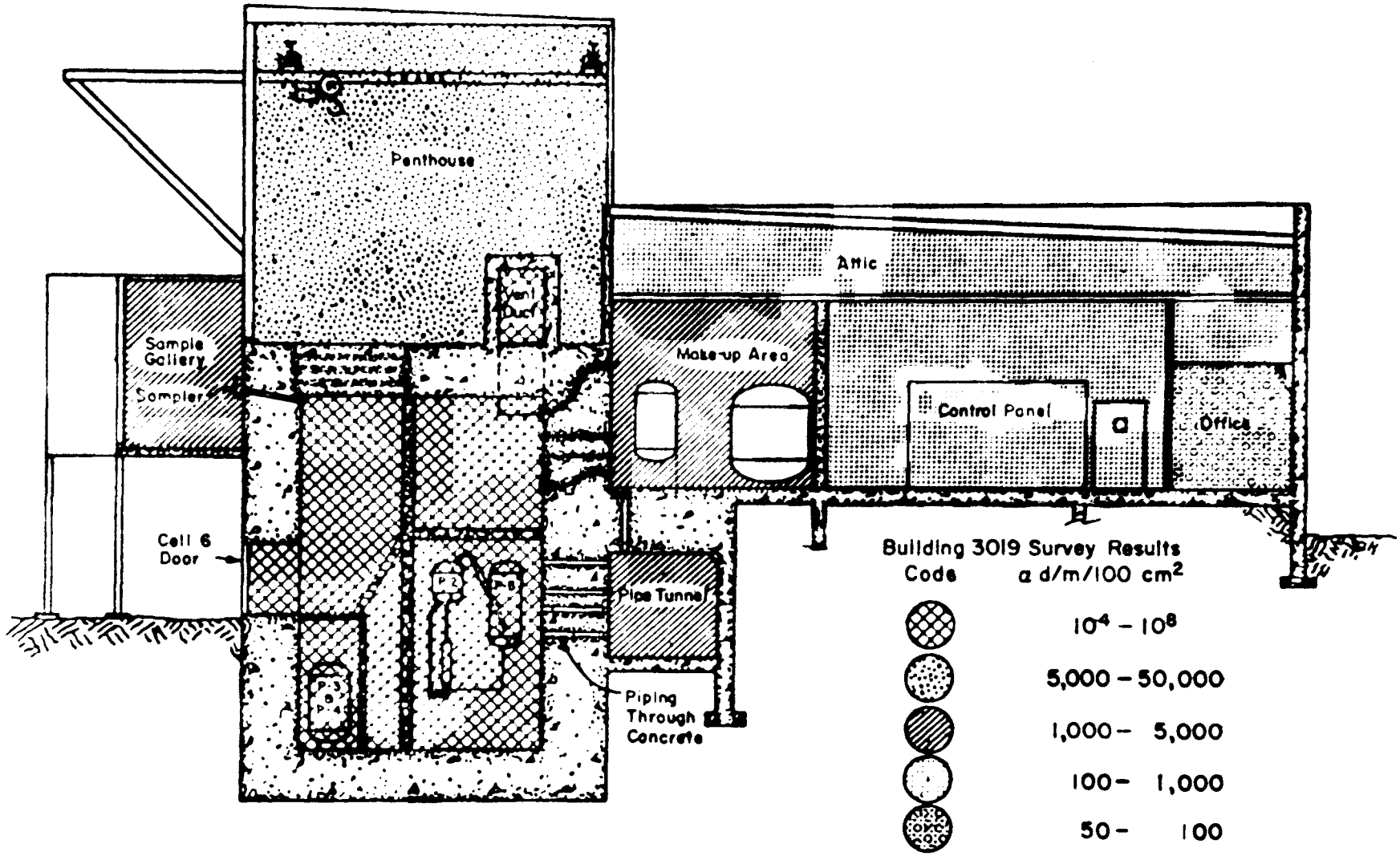


Fig. 1. Sectional elevation through cell 6, Radiochemical Processing Pilot Plant, showing inside contamination levels after explosion.

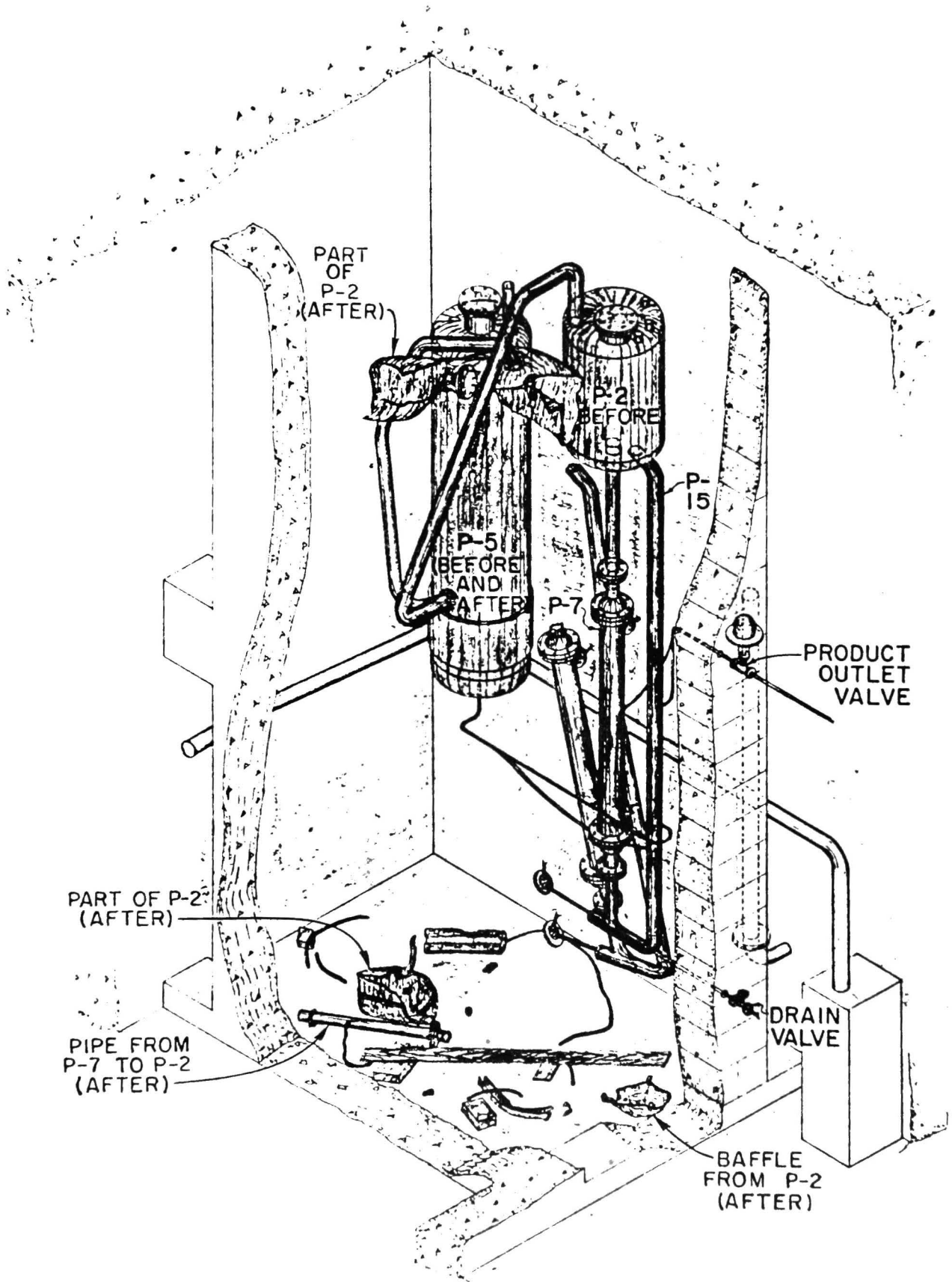


Fig. 2. Intercycle evaporator before and after explosion in Radiochemical Processing Pilot Plant.

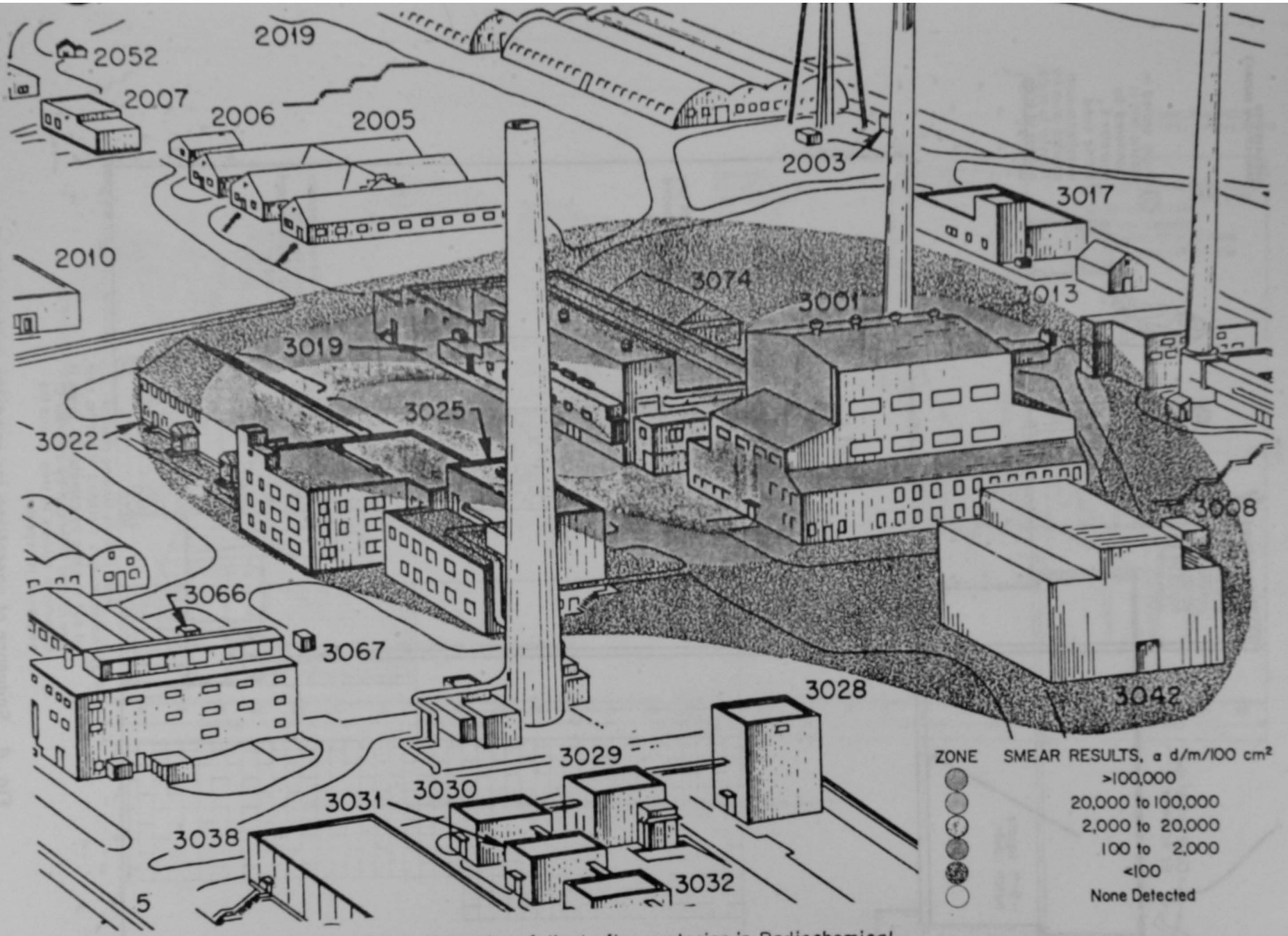


Fig. 3. Plutonium fallout after explosion in Radiochemical Processing Pilot Plant.

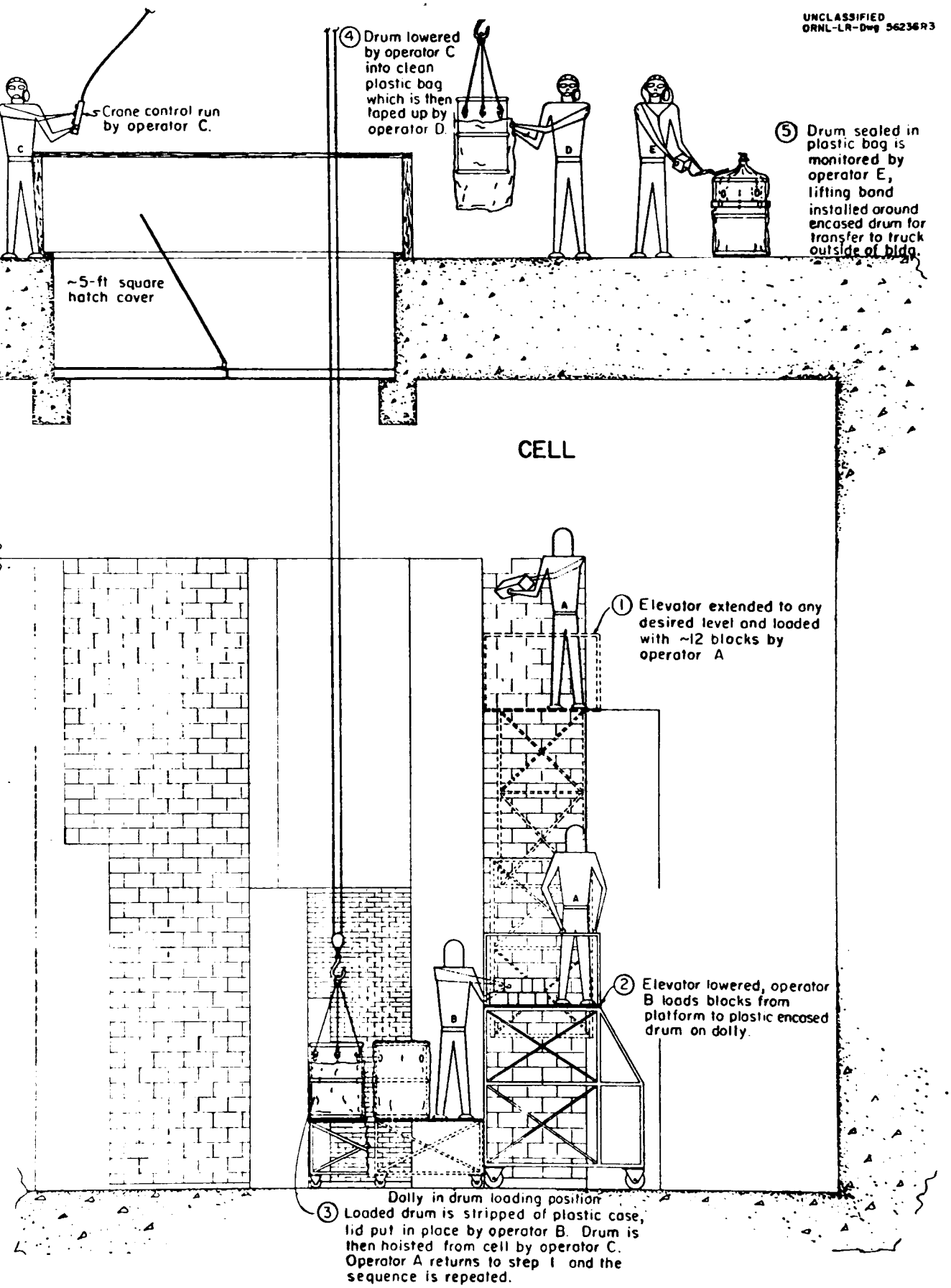


Fig. 4. Sequence of operations for removing contaminated concrete blocks from cells 6 and 7.

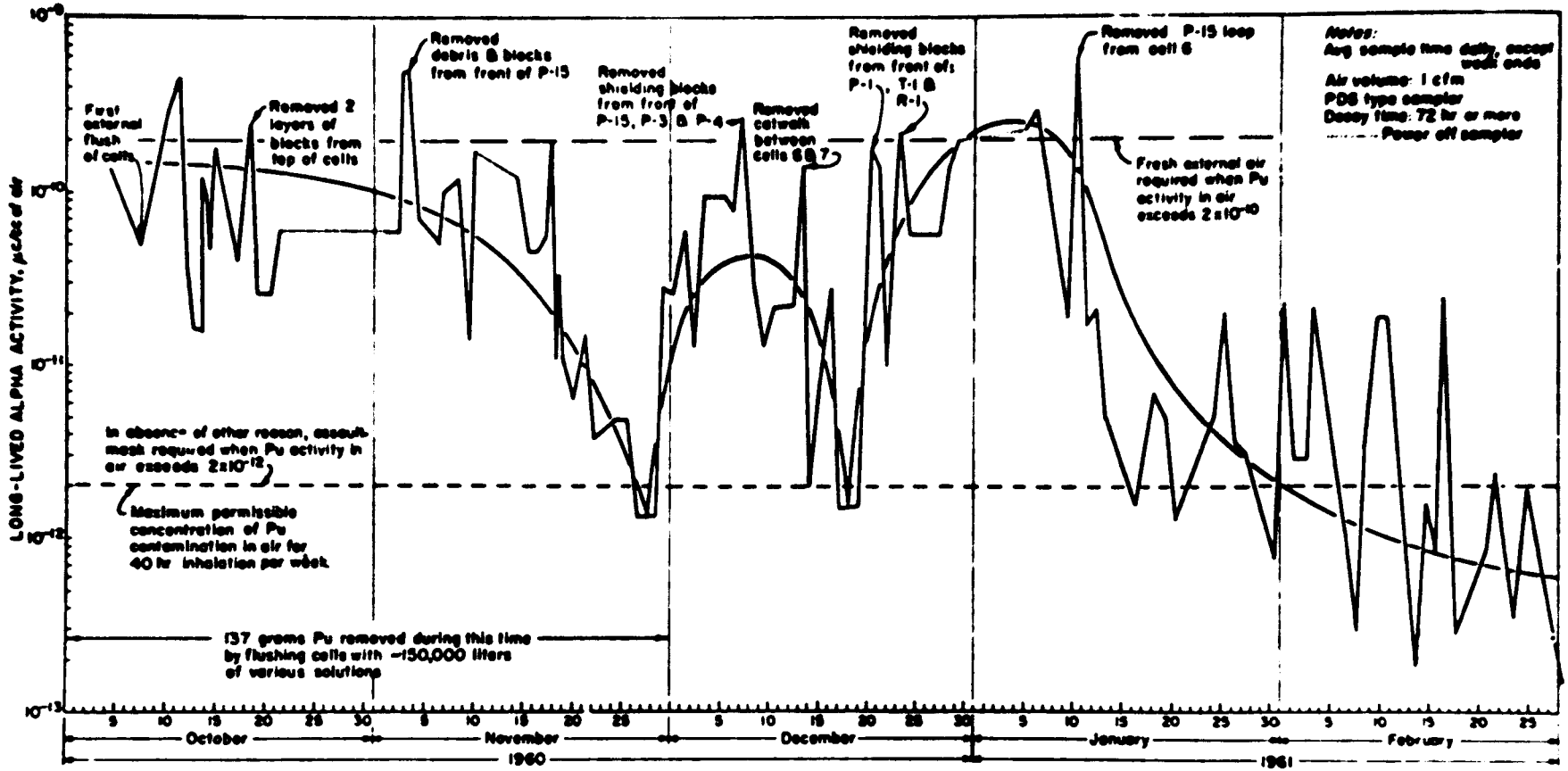


Fig. 5. Long-lived alpha activity during decontamination of cells 6 and 7.

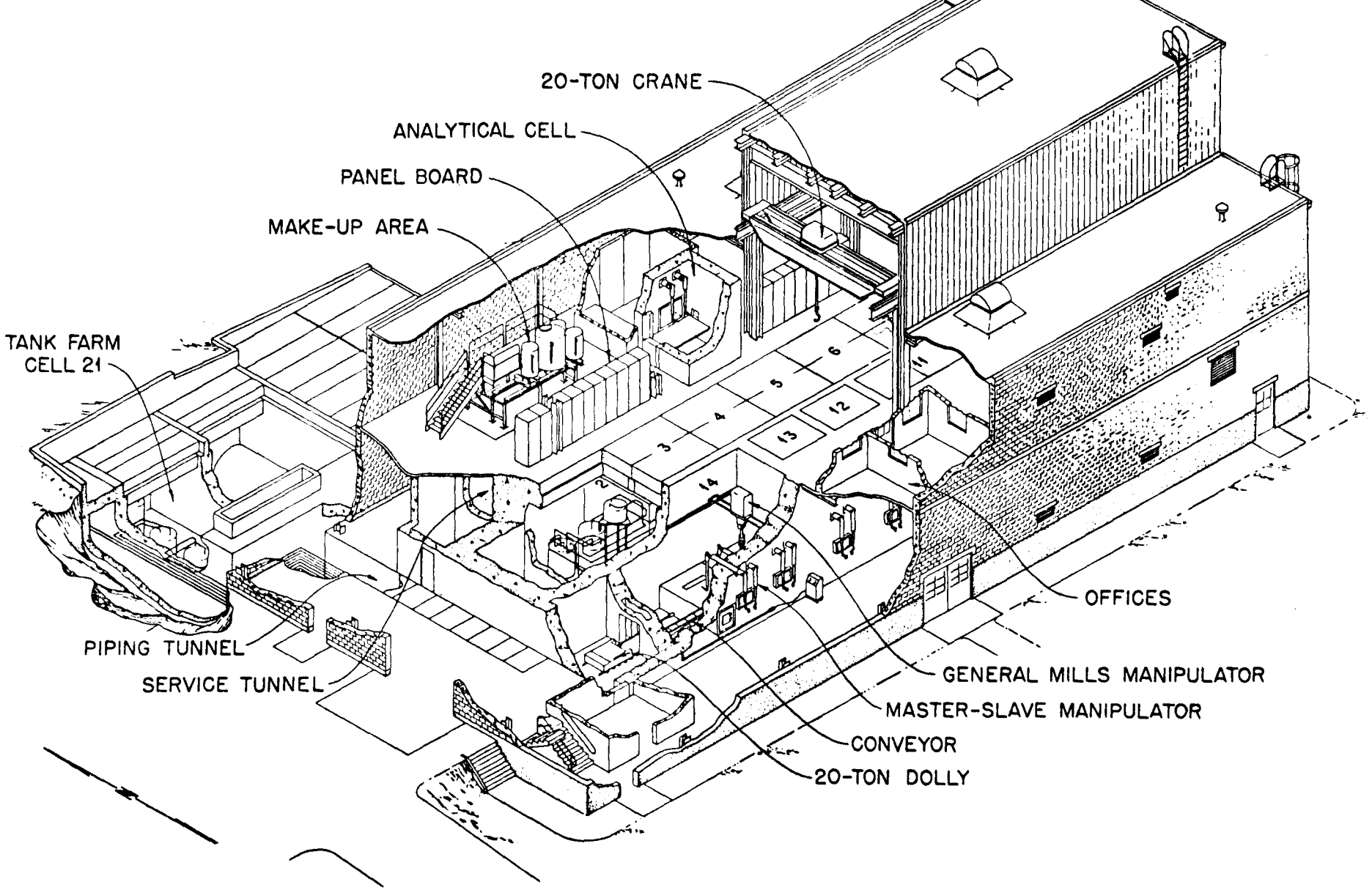


Fig. 6. A schematic of the Multicurie Fission Products Pilot Plant at Oak Ridge National Laboratory.

ORNL Dwg 79-19557A

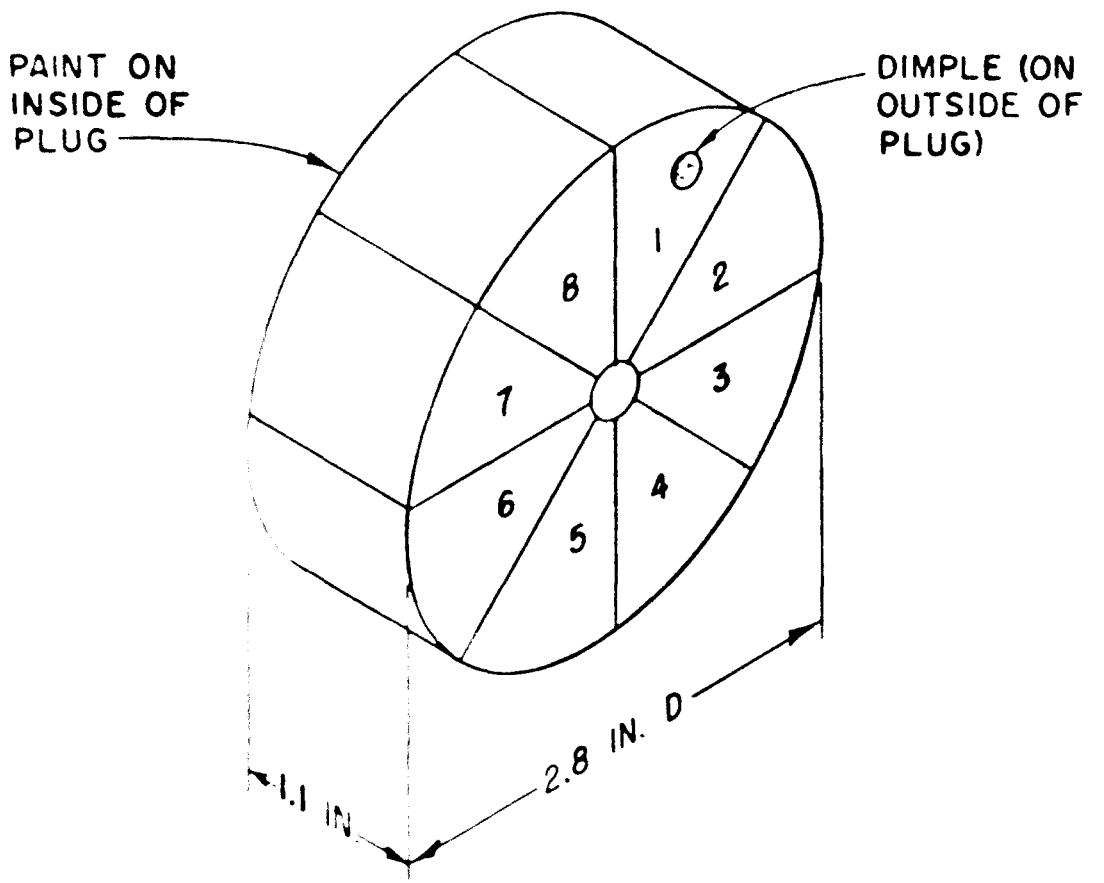


Fig. 7. Sectioning diagram of plug from TMI penetration R-401. Outside surface of containment wall.

ORNL Dwg 79-15895

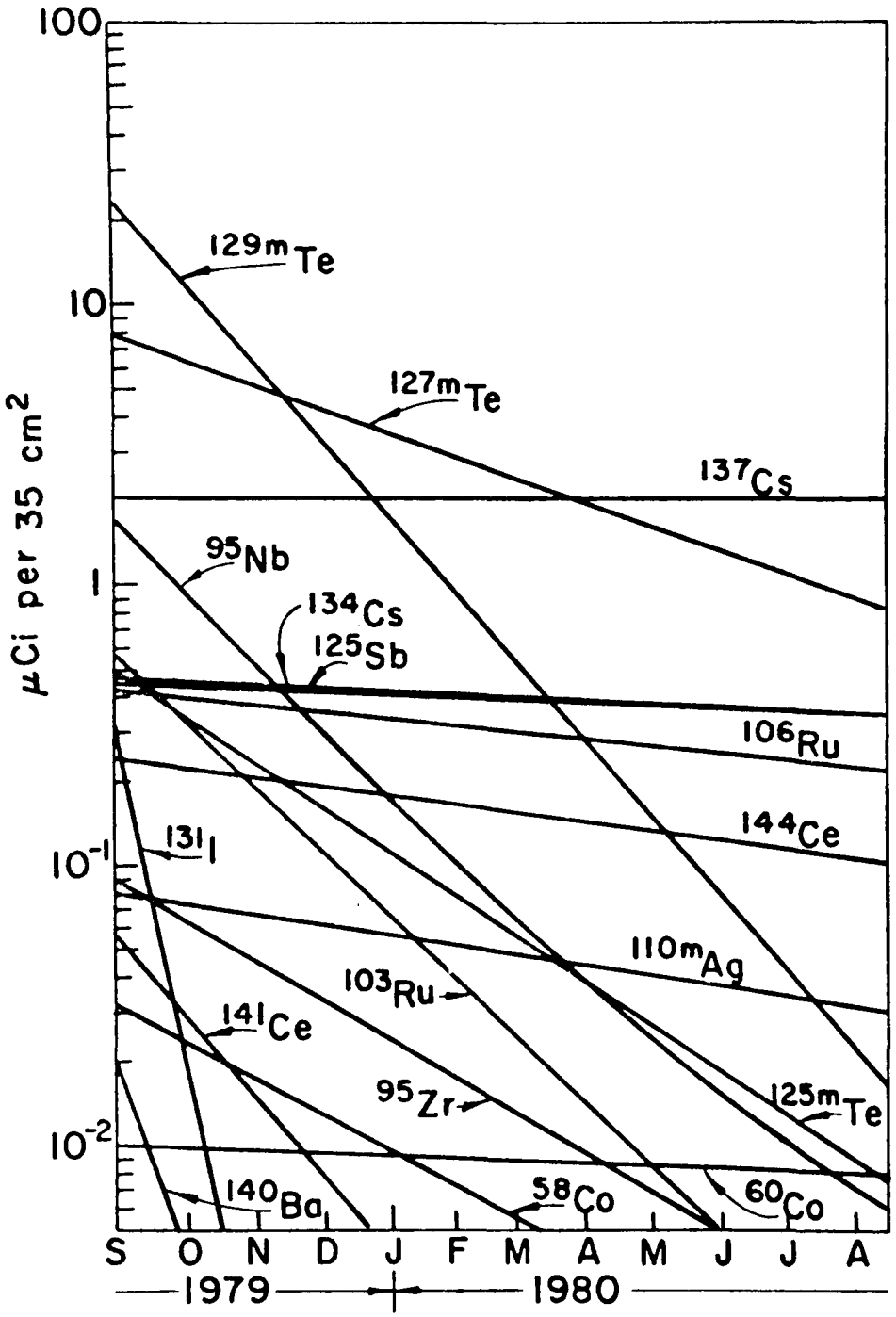


Fig. 8. Decay of radioelements on painted steel containment wall.

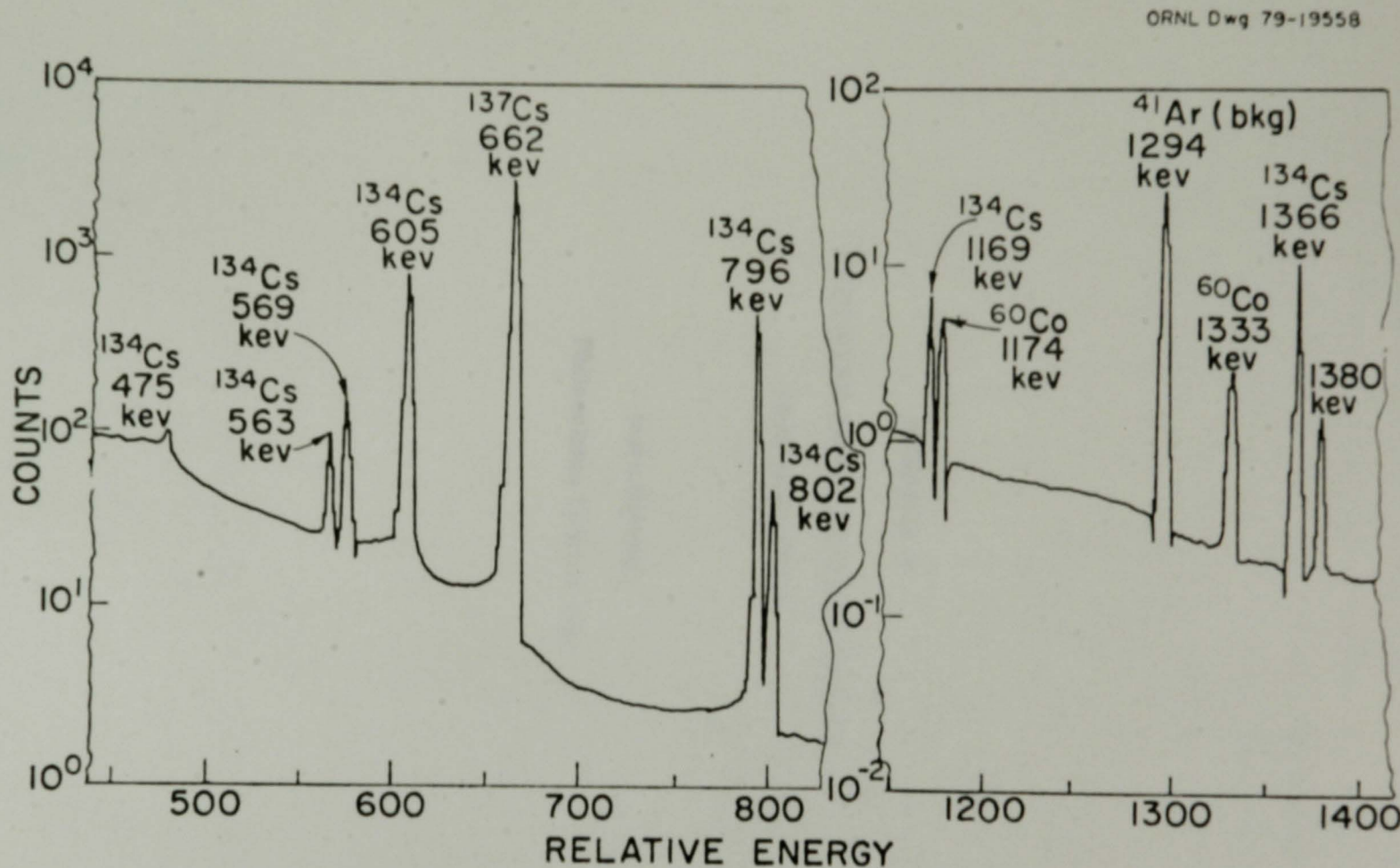


Fig. 9. Preliminary relative gamma scan of segment from TMI-R626 penetration.

SESSION X

DECONTAMINATION EXPERIENCE AT PEACH BOTTOM

Mark Rohner

Philadelphia Electric Co.

The Mark Rohner presentation at the Hershey Meeting was a condensation of "Peach Bottom 2 & 3 Regenerative Heat Exchangers, Chemical Decontamination and Seal Ring Repair" by Mark M. Rohner, Philadelphia Electric, 6/6/78 and "Peach Bottom 2 & 3 Regenerative Heat Exchangers, Chemical Decontamination and Solidification" by Gregory E. Casey, Dow Nuclear, 2/10/78. The original papers are reproduced here in their entirety.

PEACH BOTTOM 2 & 3

REGENERATIVE HEAT EXCHANGERS

CHEMICAL DECONTAMINATION

AND

SEAL RING REPAIRS

by:

Mark M. Rohner
Philadelphia Electric Company
June 6, 1978

ABSTRACT

In 1977 and early 1978, Philadelphia Electric Company chemically decontaminated and installed seal rings into the shell to channel joints of all (6) Reactor Water Clean-Up Regenerative Heat Exchangers located in Units 2 & 3 at Peach Bottom Station. The cost to perform this work was approximately $[illegible],000. The radiation exposure accumulated during chemical decontamination and repairs of all (6) heat exchangers was approximately 215 man-rem. This exposure was spread among approximately 300 individuals with individual exposures ranging from .5 to 7 rem over a one year period.

Problems with the Regenerative Heat Exchangers date back to 1974 when Unit 2's heat exchangers began to leak. In 1975, Unit 3 was placed into commercial service and its Regenerative Heat Exchangers also began to leak. Retorquing of the shell to channel bolts was performed with little success. Furmanite compound was injected into the flanged joints of (5) of the (6) heat exchangers during 1976. This temporarily stopped leakage and associated iodine releases. However, continual reinjection of (2) of the heat exchangers became necessary after the Reactor Water Clean-Up System was cycled. Continuing difficulties led to the installation of a bypass line around the Regenerative Heat Exchangers in 1976 and 1977 as an interim solution. Seal ring repairs were then performed.

This report contains the details of the background and history leading up to the repairs including:

1. The Reactor Water Clean-Up System description
2. Sealing the Regenerative Heat Exchangers with Furmanite
3. Installation of a bypass

4. Seal Ring design

5. Radiation exposure analysis

6. Seal Ring installation details

The chemical decontamination which was performed for Philadelphia Electric is detailed in a separate paper by The Dow Chemical Company.

I. HISTORY AND BACKGROUND

A. Introduction

The Regenerative Heat Exchangers form an integral part of the Reactor Water Clean-Up (R.W.C.U.) System. They are located in the reactor building just outside the drywell. Their purpose is to cool reactor water before it enters the demineralizers and then reheat it on its way back to the reactor. This regeneration recovers approximately 4.4 MW's worth of thermal energy. Because this system is the reactor's "kidney", removal of the system for more than 48-72 hours cannot be performed without seriously effecting reactor water chemistry. The absence of a clean-up system for this period usually causes the reactor water conductivity to approach limits which require shutdown. Figure 1 shows the relationship of the Regenerative Heat Exchangers to the R.W.C.U. System.

PEACH BOTTOM UNITS 2&3
REACTOR WATER CLEAN-UP SYSTEM

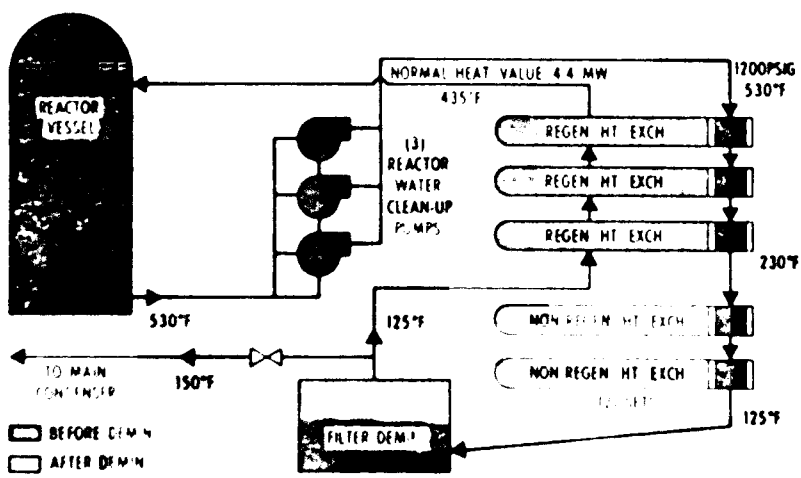


FIG. 1

In August 1974, leakage was observed on one of Unit 2's R.W.C.U. Regenerative Heat Exchangers. Investigation revealed that the stainless steel clad asbestos gasket in the shell to channel joint was leaking. Recommendations from Perfex (the Manufacturer) were that the bolting on all the Regenerative Heat Exchangers be retorqued, including the three heaters in Unit 3 which had not yet been placed in service. Torquing was performed and the leakage in Unit 2 was reduced. In December 1974, Unit 3's reactor was placed into commerical service. Shortly after this, leakage was observed on one of Unit 3's Regenerative Heat Exchangers. During the next 15 months (March 1975 to June 1976), leakage developed in all six Regenerative Heat Exchangers. Retorquing of the shell to channel joint bolting was performed with little success.

B. Sealing with Furmanite

Through conversation with other utilities, it was learned that Vermont Yankee was having a Company called "Furmanite" inject compound into their leaking shell to channel joints. As a result of these conversations, five Regenerative Heat Exchangers during an eleven month period (November 1975 to September 1976) were injected and sealed. Several of the heat exchangers required reinjection almost every time the R.W.C.U. System was cycled. Others held tight or developed only slight leakage. Although this was not as successful as Vermont Yankee's endeavor (they were reinjected yearly), it did reduce leakage from the heat exchangers. The injection of each heat exchanger required 8 to 10 craftsmen who received radiation exposures of 2.4 rem each after 8 hours of work. This occurred because radiation levels were approximately 2,000

to 3,000 MR/HR on contact with the heat exchanger flanges. The cost to prepare and inject one heat exchanger with Furmanite was about $15,000. Approximately $130,000 was spent over an eleven month period to keep Units 2 & 3 heaters sealed. Travel time and Health Physics training represented a high portion of this expense due to a turnover rate of 2-3 men/shift. Figure 2 illustrates the positioning of injection fittings and a caulking ring used in the Furmanite injection process.

PEACH BOTTOM UNITS 2 & 3
REGENERATIVE HEAT EXCHANGER
INJECTION OF SHELL TO CHANNEL JOINT
WITH FURMANITE

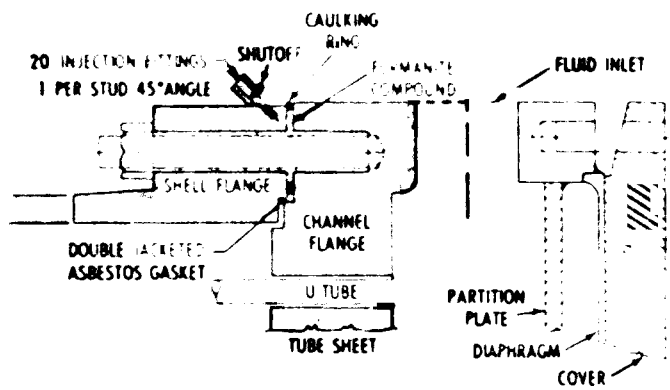


FIG. 2

During 1976, Plant Hatch (Georgia Power & Light) and Brunswick (Carolina Power & Light) developed similar leaks. Plant Hatch had pulled one tube bundle and installed a flexitallic gasket in the early part of 1975. This was done during the first few months of operation

when radiation levels were still low. In 1976, both of these Plants had their heaters Furmanited including the one which had a flexitallic gasket installed, as it was found to be leaking also.

C. Installation of a Bypass

Because of the failure of Furmanite compound at Peach Bottom to act as a permanent seal, repair alternatives were studied and a bypass line was installed around the Reactor Water Clean-Up pumps and the Regenerative Heat Exchangers. Mechanical seal problems on the R.W.C.U. pumps necessitated their inclusion in the bypass scheme. Figure 3 illustrates this bypass. The energy loss, due to the loss of regenerative heating, amounted to 4.4 MWt.

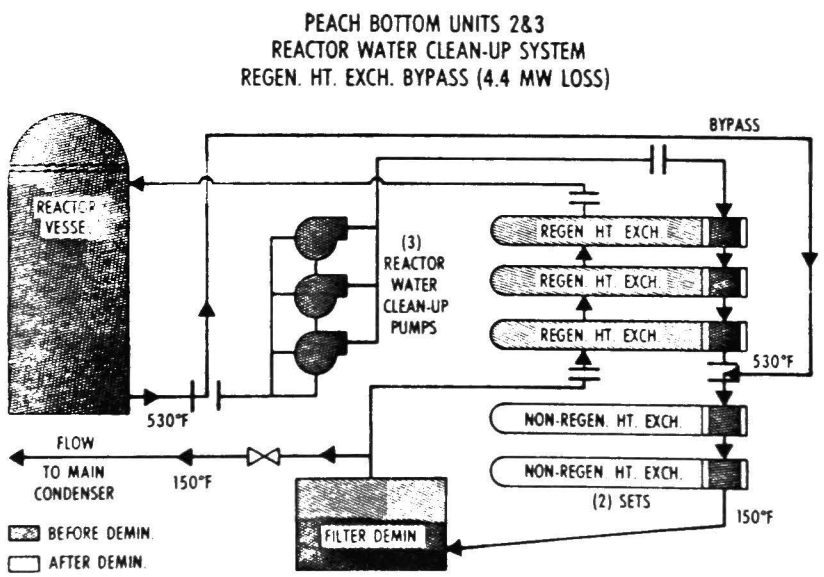


FIG. 3

D. Seal Ring Design

Consultation with Perfex, the heat exchanger designer and General Electric, the system designer, resulted in a recommendation to remove the tube bundles and install flexitallic gaskets. An alternate repair consisting of seal ring installation in place of a gasket was agreed upon by Perfex. This design was proposed by Philadelphia Electric because of previous successes at Fossil Generating Stations. Some of the advantages of this design, which involves the replacement of a gasket with a weldable seal ring, are as follows:

1. It does not have the limitations that a gasket has in thermal cycling applications where "gasket fatigue" can occur.
2. Its installation eliminates the need to remove certain piping and obstructions that are usually removed to change a gasket. In this particular installation, it eliminated the removal of a 48" thick wall and cutting of (2) 4" pipe loops which would have required radiography after rewelding (4 welds). It also eliminated removal of certain 1" connections to which there was limited access.

Perfex indicated that during the original design stages, they tried to eliminate the gasketed shell to channel joints by designing these heaters with welded joints as was done in the case of Non-Regenerative Heat Exchangers. It was found, however, that a difference in code requirements between building the Regenerative Heat Exchangers to Section III and the Non-Regenerative Heat Exchangers to Section VIII

were enough to prevent welding of the shell to the channel joint on the Regenerative Heat Exchangers. Radiography would have been required if the Regenerative Heat Exchangers were welded and physical obstructions prevented this.

Based on Perfex's positive response, Maintenance recommended that a seal ring repair be employed. The seal ring design provides a welded joint exempted from the radiography required by code on butt joints. The bolting used for this joint provides the closure strength normally afforded by a butt weld. The seal ring was designed to comply with 1974 ASME Section III, Class ND code requirements. A design change submittal was sent to the Pennsylvania Department of Labor and Industry for a "Pennsylvania State Special" authorization number, to perform the modification as detailed. This was required since modifications were to be made to a National Board vessel by someone other than a "stamp" holder. This design also included the installation of stainless steel bolts in place of the original carbon steel bolting. Stainless steel bolts were specified to help stabilize the clamping force in the joint between hot and cold situations, since it had been determined that carbon steel bolts would be overstressed when the Unit was hot. It is believed that the differential expansion that existed in this joint may have caused the original gasket to fatigue. Calculations indicate that a differential expansion of .015" between the heater flanges and the originally installed carbon steel bolting existed over the change in temperature encountered. Figure 4 illustrates the position of the seal ring in the shell to channel joint.

PHILADELPHIA ELECTRIC COMPANY
PEACH BOTTOM UNITS 2&3 REGENERATIVE HEAT EXCHANGER
SHELL TO CHANNEL JOINT REPAIR-USING SEAL RING

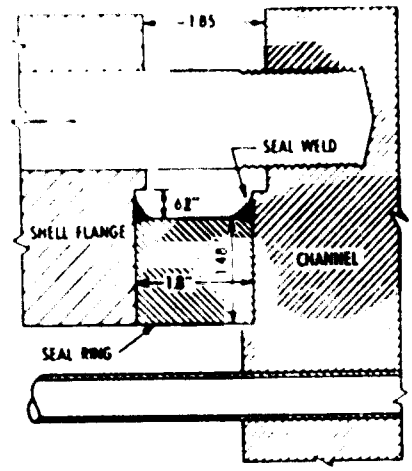


FIG. 4

II. RADIATION EXPOSURE ANALYSIS

Prior to making repairs, Maintenance and Health Physics personnel performed an analysis to predict the radiation exposure and the amount of labor required to perform repairs. The repairs themselves were estimated to take a minimum of 90 shifts/unit. Calculations based on actual radiation exposure data obtained from experience with previous work indicated approximately 1100 man-rem would be required to repair all six heat exchangers. It was estimated that a total of $250,000 would be spent for Welder Qualification Testing ($700/welder) and Health Physics Training. Review showed that it would require 3 days to train and qualify a welder, to the requirements of the ASME Section IX code, only to have him work for 4 hours and then have to be dismissed from the site until the next calendar quarter. These figures indicated that approximately

500-700 craftsmen would be required to perform repairs and that a majority of these individuals would receive radiation exposures equal to 2.5 rem/quarter. This analysis clearly indicated that an alternative arrangement for performing this work was essential.

Review of the various methods available to reduce radiation exposure and manpower requirements lead to chemical decontamination as the only alternative. None of the usual methods of reducing radiation exposure (shielding, time and distance) could be employed since it was the heaters themselves which were the principal radiation source in the room and to make repairs, shielding and distance could not be employed. Even with shielding, general area dose rates in the rooms ranged from 200 to 400 MR/HR. With the heaters opened, it was expected that dose rates would have been 1000 to 1500 MR/HR in the area where work was required to be performed. Figure 5 shows an area adjacent to the heaters where a field of 400 MR/HR exists.

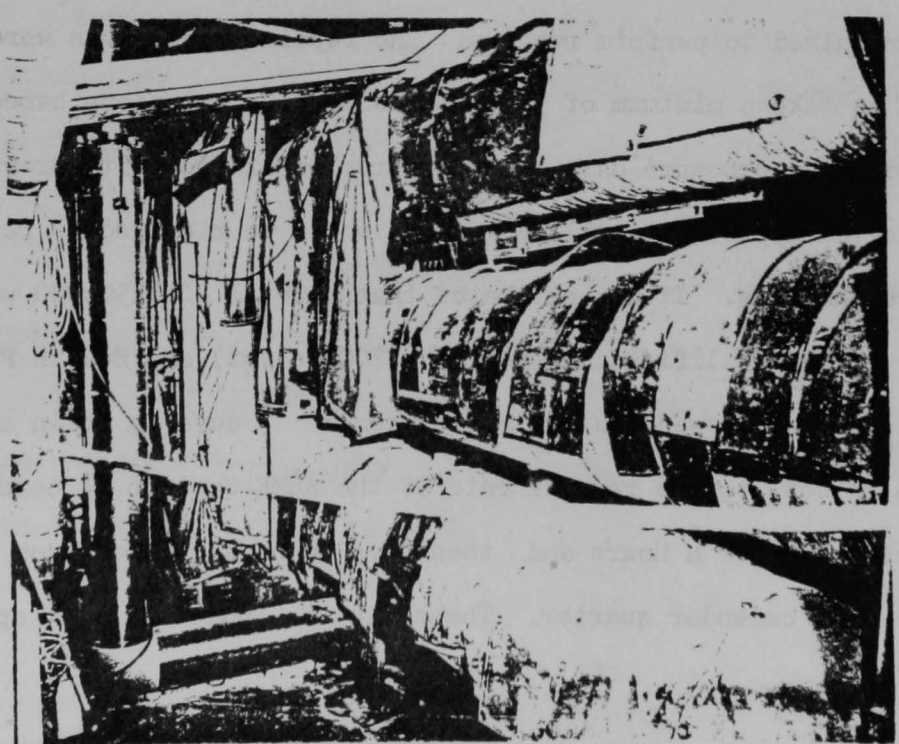
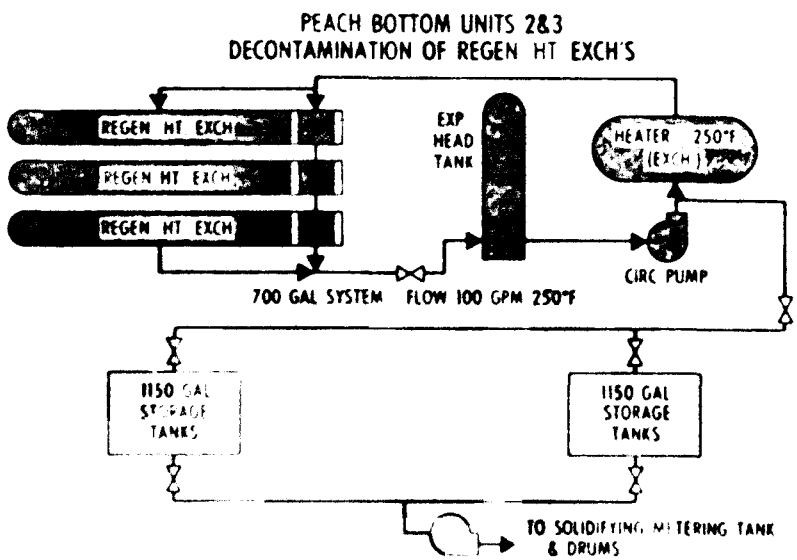


FIG. 5

Contact with Dow Nuclear Services revealed that a solvent was available that could be circulated through the heaters and would chemically remove the radioactive deposits which had plated out on the 3600 sq. ft. of heat transfer surface.

After testing samples taken from Units 2 & 3 (pipe removed during installation of the bypass line), Dow indicated that their solvent (NS-1) would remove a very large percentage of the radioactive contaminants in the Regenerative Heat Exchangers. A proprietary agreement was signed and detailed information regarding the chemicals and their effects on the reactor, piping and valves, etc., was obtained. After reviewing these, a decision was made to contract Dow Chemical to perform decontamination of the heat exchangers. Safety reviews on the process particulars were made and flow diagrams were used to develop piping sketches and drawings necessary for the placement of equipment, etc. Figure 6 shows the simplified flow diagram.



SIMPLIFIED FLOW DIAGRAM

FIG. 6

III. CHEMICAL DECONTAMINATION WITH DOW NS-1

Dow Chemical performed chemical decontamination of Unit 3 Regenerative Heat Exchangers in April, 1977 and Unit 2 heat exchangers in September, 1977. The total cost to perform decontamination of both Units was approximately $450,000. The chemical decontamination and solidification processes required approximately 25 shifts of work, utilizing (4) Dow personnel/shift. Preparation for Unit 3 required two-three months. Unit 2 preparation required approximately one and one-half months.

A description of the process (including solidification) is described in a separate paper prepared by The Dow Chemical Company.

Dow's role at Peach Bottom was that of providing; 1) Engineering & Health Physics expertise for the equipment and piping designs, 2) Chemicals and labor to perform chemical decontamination and solidification.

Catalytic Construction Co. was retained to provide necessary labor and equipment needed for the installation of the chemical piping. This included procedures and drawings necessary to effect complete isolation of the heaters from the Reactor Water Clean-Up System and installation of chemical piping.

IV. SEAL RING REPAIRS

Seal ring repairs consisted of removal of all vent and drain lines, relief valves, piping and piping supports. Shown on Figure 7 is one-half the piping.

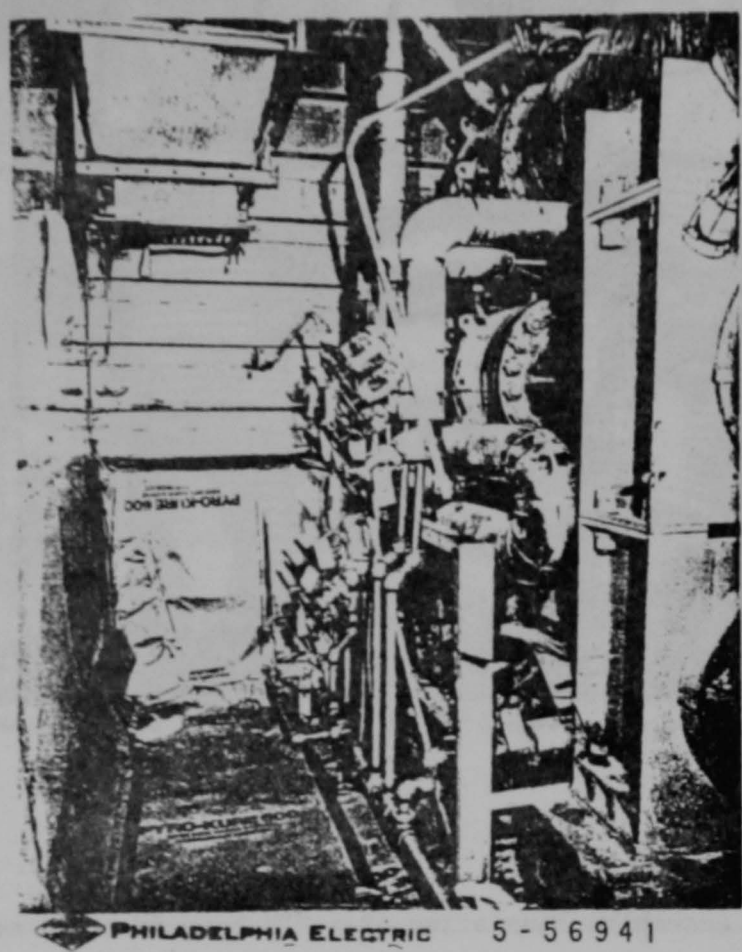


FIG. 7

Approximately 25 shifts utilizing 10 craftsmen/shift were required to remove 60 (1-5/8") flange bolts and to remove Furmanite from the flange faces and bolt holes. Figure 8 illustrates Furmanite adhering to the bolting.

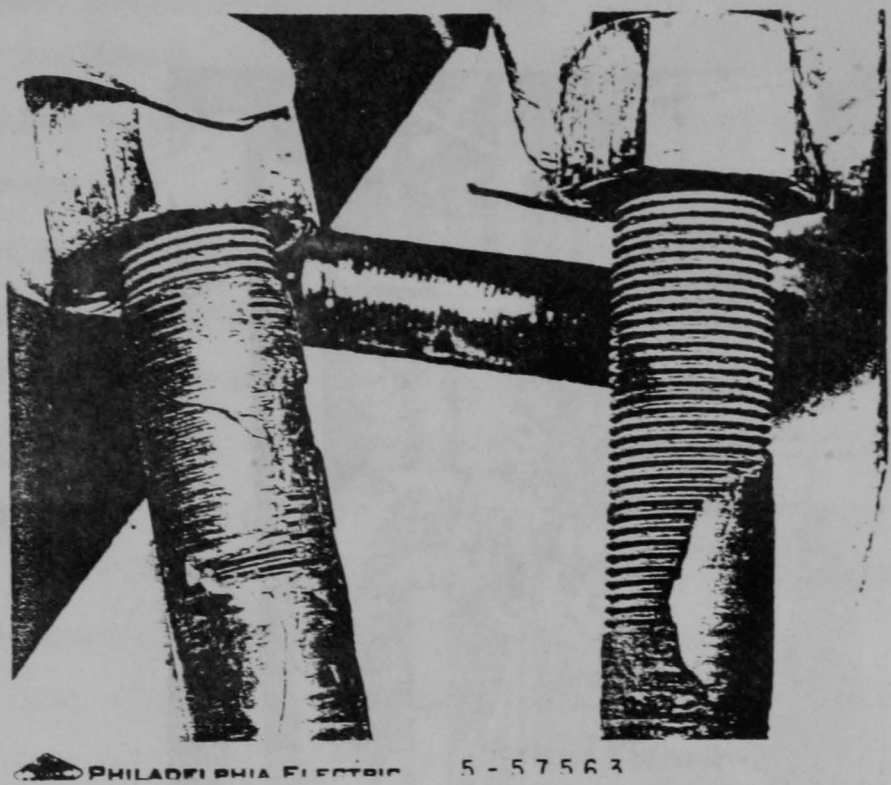


FIG. 8

A jacking assembly, consisting of a "T" beam fastened across the three channel heads and two 9 ton jacks, capable of jacking all three heat exchanger bundles (20,000 lbs.) apart simultaneously was utilized. This was done by mounting the jacking assembly around the middle shell and jacking the middle channel forward. Double acting jacks were used so that opening and closing operations could be performed with minimum set-up time. Jacking in this manner permitted repairs to be made without cutting the loop piping (2 loops) between heaters. The jacking collar and one jack is shown on Figure 9.

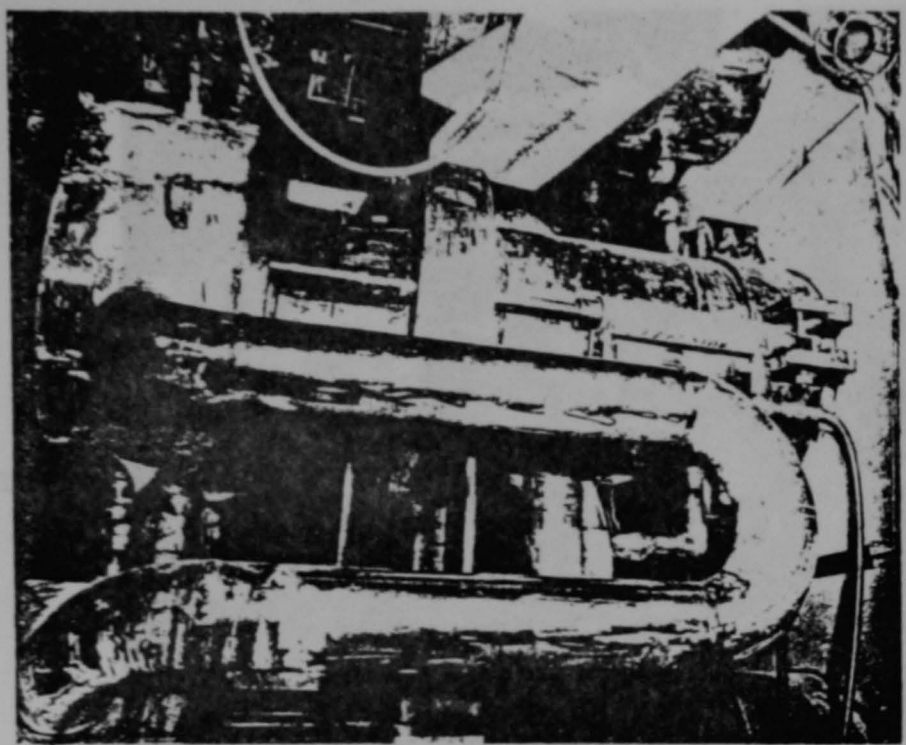


FIG. 9

Once the heaters were apart, split seal rings were mounted on each channel flange. A copper ring was temporarily used to protect the flange face. A stainless steel clamping ring with six clamps was used to prevent warpage during welding. See illustration in Figure 10.

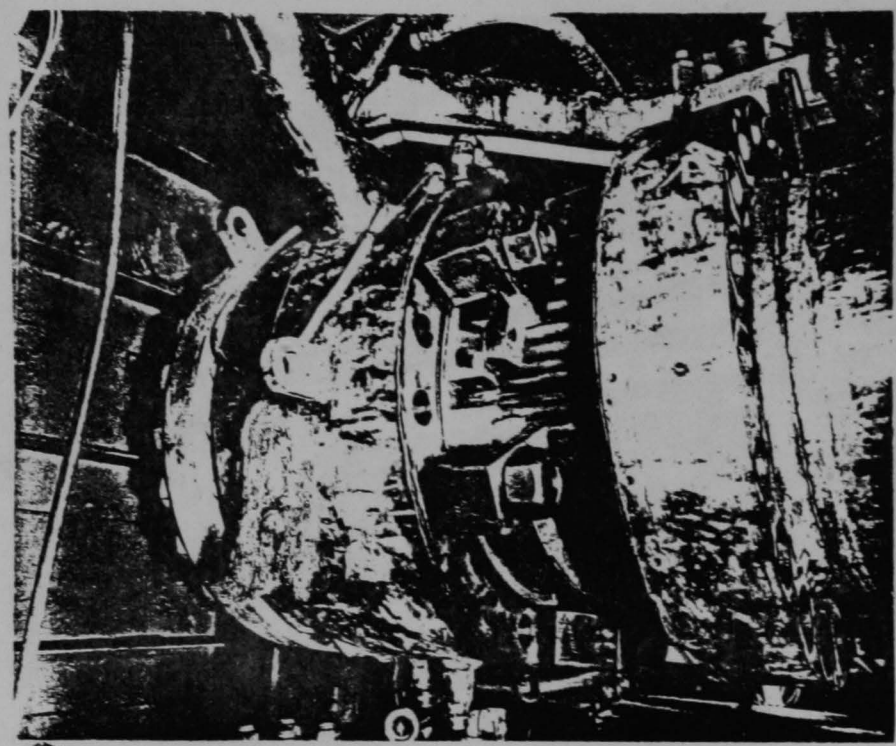


FIG. 10

Butt welding of the rings was performed using both the tig and electric arc welding processes. Fiberglass backing tape was used as a backing band during root welding. Surface grinding and penetrant testing of all welded surfaces (including the root I.D.) was performed. During the joining process, the welder alternated between each of the three rings so that the 300-350°F maximum interpass temperature limit required for 304 stainless steel would not be exceeded. Distortion during welding was controlled by utilizing a peening process between weld passes. The performance of the six butt welds required approximately 30 shifts, utilizing 10 craftsmen/shift. Figure 11 shows the ring with a partially made butt weld.

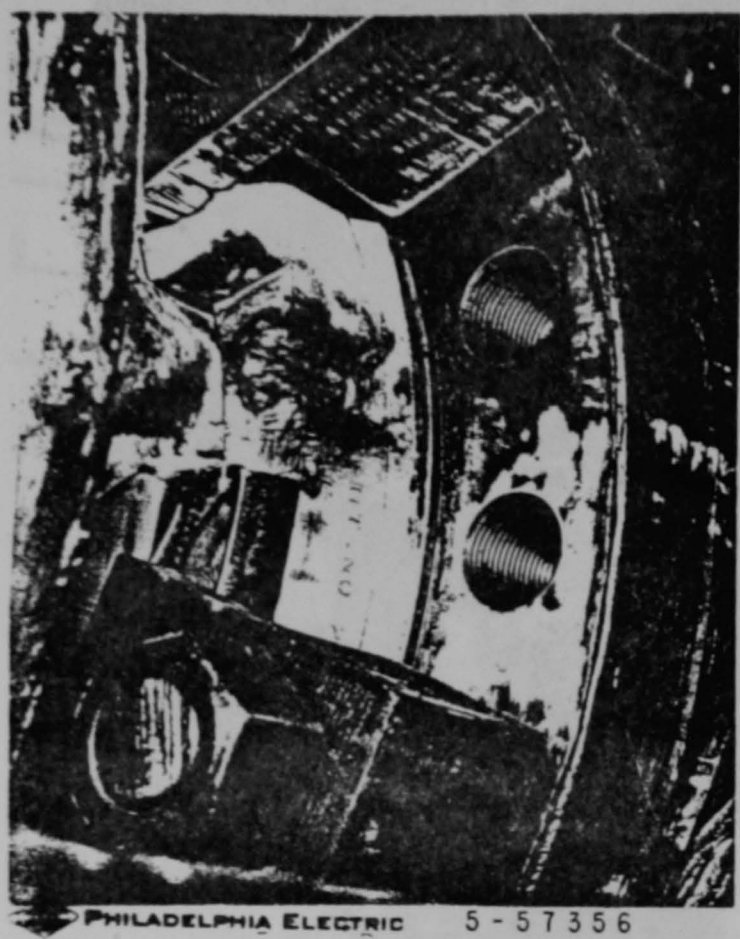


FIG. 11

A ½" fillet weld was utilized to seal weld the rings to the shell and channel flanges. Accurate positioning of the rings against the flanges was required due to the limited clearance that existed between the ring I.D. and existent steps on the flange faces. New SA 453 GR 660 stainless steel bolting was installed and torqued. This bolting was designed to hold the ring in compression at all times. Seal welding of three rings to the shell and channel flanges required approximately 15 shifts, utilizing 10 craftsmen/shift. Figure 12 shows the finished joint.

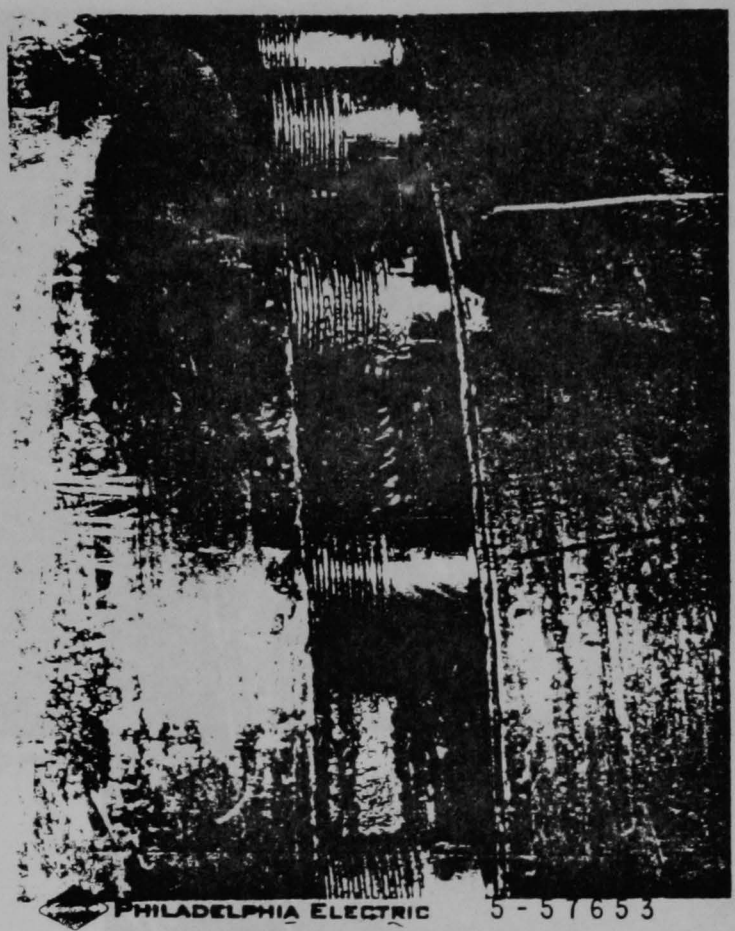


FIG. 12

Following completion of the job, all drain and vent piping was re-installed and insulated (See Figure 13). Approximately 40 shifts, utilizing 10 craftsmen/shift, were required for piping and insulation work. Repositioning of some of the piping was required to compensate for the 1-3/4" change in length caused by substituting a seal ring for a gasket. Prior to this, all valves (approximately 42) were repacked and repaired. Inspection of the tube sheet and channel boxes indicated all internal parts to be in good condition with the exception of an internal weld between the channel box and the channel pass cover which was cracked. This was repaired. Hydrostatic testing to 2180 PSIG was performed and performed and witnessed by an Insurance Inspector.

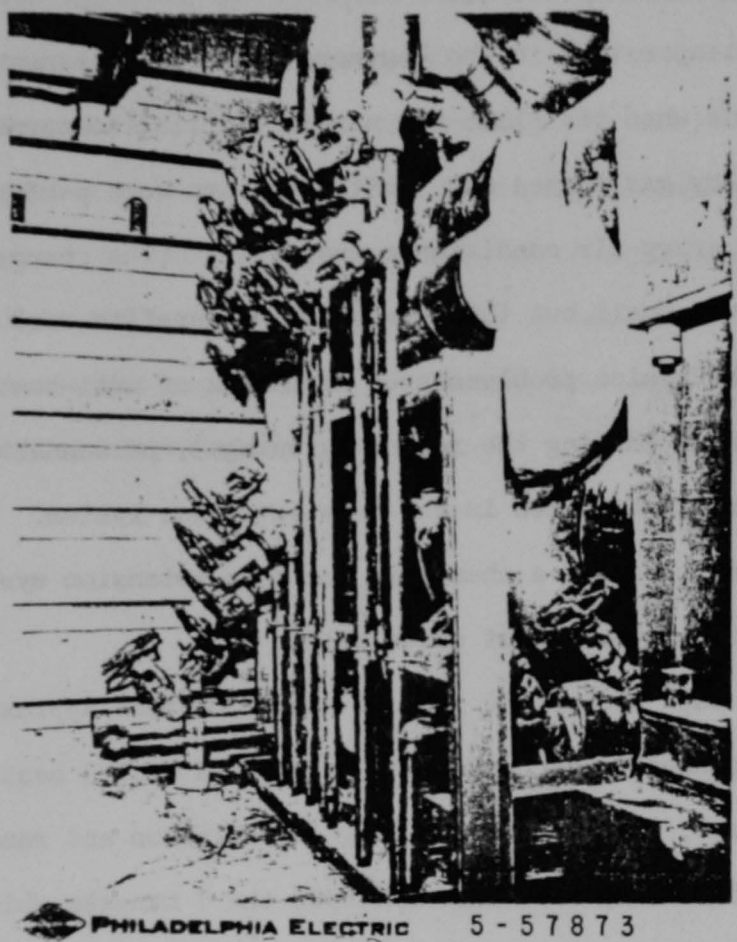


FIG. 13

V. SUMMARY

Installation of seal rings into each of the shell to channel joints was in some ways easier than chemical decontamination. Approximately 135-150 shifts/unit were spent to perform the repairs, which included 25 shifts for channel box inspection and tube testing that had not originally been planned. This was about 30% more time than was estimated for the planned work.

For the most part, 10 men/shift were used on a 2 shift/day, 5 day per week basis. The principal problems that seemed to exist which caused reductions in labor output were:

1. High temperature in the Regenerative Heat Exchanger room during periods when the Plant had normal ventilation turned off and stand-by gas turned on. Unit 2 repairs were performed with a temporary air conditioner installed. (The change in temperature was small but the psychological benefits were large.)
2. Health Physics problems such as a lack of anti-contamination equipment (during the refueling outage), personnel contamination and inflexabilities in the dose extension system.

Since these repairs, a change in our dose extension system has been implemented and has worked out quite well.

From data dept during the job, it was found that approximately 110 man-rem was expended to decontaminate Unit 2 and Unit 3 heat exchangers. For the most part, this includes piping installation and removal, plus Engineering and Testing. It also includes the 7 man-rem which Dow Company Personnel received during the decontamination process. An additional 105 man-rem (extrapolated from data taken during work on one unit) was expended to install the seal rings. The total radiation expenditure was approximately 215 man-rem, as opposed to the originally estimated 1100 man-rem without decontamination. Thus, an estimated total of 900 man-rem of radiation exposure was saved by chemical decontamination.

If chemical decontamination had not been available, it is estimated that an additional $350,000 would have been added to the repair cost due to the increased crew size, welder qualification and Health Physics training

that would have been necessary. Thus, the estimated cost to reduce radiation exposure by chemical decontamination was approximately $115/man-rem after applying the $350,000 potential increase in the repair cost had decontamination not been performed.

In retrospect, had chemical decontamination not been available at the time repairs were performed, the only viable alternative available would have been to scrap the Regenerative Heat Exchangers and purchase replacements without gasketed joints. It is estimated that the cost for these installed replacements would have been approximately $1,000,000 and 300 man-rem of radiation exposure.

PEACH BOTTOM 2 & 3

REGENERATIVE HEAT EXCHANGERS

CHEMICAL DECONTAMINATION

AND

SOLIDIFICATION

by

Gregory E. Casey

DOW NUCLEAR SERVICES

February 10, 1978

ABSTRACT

In 1977, Dow Nuclear Services, under contract to Philadelphia Electric Company, chemically decontaminated the regenerative heat exchangers at the Peach Bottom 2 and 3 Atomic Power Station. The purpose of the decontamination was to reduce the radiation levels associated with the subsequent heat exchanger repairs to be performed by PECO maintenance. Samples of piping from the regenerative heat exchangers were analyzed at Dow Chemical, Midland, Michigan, and solvent testing and selection was performed. Nuclear Solvent-1 was selected. Temporary equipment, piping and radiation shielding was installed to perform all necessary functions safely. All designs and procedures were approved by the Peach Bottom Plant Operations Review Committee. The chemical decontamination removed 10.6 curies of radioactive material in the case of Peach Bottom 3 and similarly at Peach Bottom 2, 6.3 curies of material was removed. Radioactive waste generated by decontamination that could not be treated by existing facilities, was successfully solidified by the Dow Solidification process.

Overall, chemical decontamination proved to be a very cost-effective method of radiation reduction at the Peach Bottom regenerative heat exchanger repairs.

In December 1976, Dow Nuclear Services was contacted by Philadelphia Electric Company with questions as to the feasibility of chemically decontaminating the regenerative heat exchangers at Peach Bottom 2 and 3. At this time, Pete Frauson, Dow Nuclear Services, made the initial site visit, ultimately requesting samples to be cut and sent to Midland, Michigan for analysis and solvent testing. The samples were received and surveyed by Dow Health Physics in January, 1977. Warren Strom, Sr. Research Chemist for Functional Products and Systems, R&D, examined and identified the samples with reference to the shipping papers as follows:

Sections from Peach Bottom 2

Peach Bottom 2, Section I - 1 piece, 4 inches diameter by 30 inches length, from V-2 RWCU region, heat exchanger outlet before demineralizer.

Peach Bottom 2, Section II - 1 piece, 4 inches diameter by 9 inches length, inlet to heat exchanger shell side from demineralizers.

Peach Bottom 2, Section III - 1 piece, 4 inches diameter by 14 inches length from the demineralizer bypass line.

Sections from Peach Bottom 3

Peach Bottom 3 Section I - 1 piece, 4 inches diameter by 30 inches in length from V-3 RWCU region, heat exchanger outlet to demineralizer.

Peach Bottom 3 Section II - 1 piece, 4 inches diameter, 18 inch by 18 inch elbow from the heat exchanger shell inlet from the demineralizers.

Small samples of approximately one square inch were cut from the larger sections of pipe. Each sample was appropriately labeled.

Sampling & Preliminary Analysis

The radioisotope identification and quantification was performed by a high resolution Germanium-Lithium crystal gamma ray spectrometer. The standards used for calibration were ^{133}Ba at 0.356 Mev, ^{137}Cs at 0.662 Mev, and ^{60}Co with peaks at 1.173 and 1332 Mev. Table I lists the isotopes identified and quantified. The data shows that the major isotopes present in Peach Bottom 2 to be ^{60}Co and ^{65}Zn, whereas the scale from Peach Bottom 3 has a much higher ratio of ^{65}Zn to ^{60}Co.

Next, the samples were exposed to NS-1 at 250°F for different periods of time. Tables II and III record the results of the Peach Bottom 2 and Peach Bottom 3 samples, respectively. Although other selected solvent systems were tried, none were found to be more effective than the NS-1 Solvent system.

After the timed solvent experiments had been completed, the solvent was chemically analyzed for Iron, Chromium, Copper, Nickel, and Zinc. The results are summarized in Table IV. Finally, the amount of sloughed material was determined for four samples. The used NS-1 solutions were passed through tared Millipore® filters of 0.45 μ pore size. The filters were dried and then weighed. The results are shown in Table V. The activity remaining on the filters was determined by a Germanium-Lithium spectrometer. The percent of activity was calculated by comparison to the original activity of the sample. In all

cases, the undissolved sloughed material was less than 2 percent of the original. Table VI shows this data.

The analytical test data was transmitted to Philadelphia Electric Co., with conceptual flow diagrams, procedural outlines, and contractual agreements. After due considerations and review, Philadelphia Electric decided to proceed with the chemical decontamination and subsequent solidification of generated waste with the Dow solidification process.

Planning, System Modification & Equipment Design

After review of the isometric drawings of the regenerative heat exchangers and piping, a visit to Peach Bottom Station was arranged. The regenerative heat exchanger room was inspected with Mark Rohner, Philadelphia Electric Co., Maintenance Division. The heat exchangers had been isolated from the reactor system by cutting the inlet and outlet piping on both the tube side and the shell side with necessary spool pieces and blanks put in place to allow the reactor water clean-up system to be operated. The open inlet and outlet pipes on the heat exchangers would be utilized as connections for the chemical decontamination. The flow of the NS-1 was to be the opposite of the normal path to act as a back flush and to facilitate circulation in the low flow areas under normal flow conditions. The normal flow path is from the reactor to the top heat exchanger channel inlet through the tube side of all three exchangers and on to the non-regenerative heat exchangers from the lower regenerative heat exchanger channel outlet. The water returns from the cleanup demineralizers to the lower exchanger shell inlet passing through the middle and top heat exchanger and

exiting through the top regenerative exchanger shell outlet returning to the reactor.

The parameters for using NS-1 Solvent decontamination were an operating temperature of 250°F to 260°F at a flow rate of approximately 100 gpm to 125 gpm. The working pressure of the system was calculated to be 30 psig vapor pressure plus 40 psig pump head. The total solvent contact time was to be determined by analytical tracking of solvent chemical parameters. These parameters were total Iron, total activity, and percent NS-1 capacity available. All equipment and piping was specified to operate safely within these given conditions. The temporary circulation pump was a stainless steel 3" x 2" x 6" centrifugal pump rated at 100 gpm at 100 ft. T.D.H. The discharge of the pump was piped to the tube side of a 75 sq. ft. single pass, stainless steel tube and head, carbon steel shell temporary heat exchanger. The fluid was then piped with 2 inch schedule 40 304 stainless steel pipe to the lower regenerative heat exchanger's normal channel outlet. With the concept of reverse flow in mind, this channel outlet became the temporary solvent inlet. The solvent flowed upwards through the tube side and channels of all three regenerative heat exchangers. The normal channel inlet, which now is the channel outlet for the solvent, was connected to the normal shell outlet with a temporary cross over line. The NS-1 passed through all three shell sides and exited through the normal shell inlet on the lower regenerative heat exchanger. From this point the solvent returned to the head tank. The head tank was constructed from six inch stainless steel pipe with sight glasses

attached for level indication. The three inch suction of the temporary circulation pump was drawn from the bottom of the head tank. The pump was protected by an in-line stainless steel strainer. Two large waste collection tanks were constructed and installed. These tanks of approximately 1,150 gallons each were multi-purpose units. They were to act as condensate/cooling water holdup tanks, storage tanks for spent NS-1, and contaminated rinse water to be solidified later and to provide a tank for emergency dump-quenching safety procedure. A small pump with necessary piping was installed between the two temporary waste storage tanks to allow mixing of the two tanks individually or simultaneously. This pump was also used to charge the metering tank to be used in the solidification process. Each tank was individually vented to the hall area through a manifold of six Iodine canisters with their check valves removed. This allowed the tanks to breathe as needed. During the actual decontamination the waste holding tanks were isolated from the pressurized system by a single valve.

The pressurized portion of the chemical decontamination system was protected by a one inch stainless steel relief valve set at 55 psig while a vacuum relief valve was also installed to protect against a negative pressure. Both of these relief valves were located on the top of the head tank and piped to the waste storage tanks. A nitrogen line was also connected to the top of the head tank to allow the system to be kept under a blanket between stages as well as to assist in the draining operations.

Instrumentation to monitor the system were thermocouples, used with thermowell thermometers as a backup, pressure gauges and an ultrasonic flow meter. The temperature was monitored at the temporary heat exchanger solvent outlet, the suction head tank, the cooling water supply and the steam supply to the temporary heat exchanger shell side. Pressure gauges were used on the shell side of the temporary heat exchanger. The solvent circulation pump discharge and the section head tank also had pressure indicators. The ultrasonic flow meter was attached to the two inch pump discharge to monitor flow. This meter was used to confirm flow with the hot water test run but would not function properly with the solvent stage. Flow in the system was then judged on the basis of the differential between the suction head and discharge pressure.

During all phases of this project, the safety of the personnel was the prime consideration. The system was checked, rechecked, and reviewed by Philadelphia Electric Company, Catalytic, Inc., and Dow Nuclear Services for maximum safety and minimal radiation exposure. Work areas were designed to allow as open area as possible while providing measures to contain a "worst case" spill or accident. Floor drains were plugged; the floors protected with layers of plastic and dams erected on each end of the hallway to contain a maximum spill.

Existing radioactive hotspots were mapped out and new high radiation areas to be generated due to the decontamination were projected and considered. Lead shielding was erected where ever practicable. The

working crews were monitored continuously by Health Physics. In addition to personnel radiation protection equipment, the work area was surveyed, wipe-tested and air sampled on a regular basis. A daily exposure record was maintained, attached to the radiation work permit at the Health Physics desk.

From the flow sheet and isometric drawings, procedures were developed to regulate the operations from the testing stages through the solidification of the wastes generated. The procedures can be broken down into four basic sections. The first area to be addressed was pre-operational testing of the temporary system to assure all design criteria had been satisfied. These tests include hydrostatic tests for leaks, filling the system with deionized water in much the same way the solvent would be injected; running the circulation pumps and testing the temporary heat exchanger. The test water was heated to the operating temperature of the solvent and cooled at a controlled rate. An emergency dump with hot water was performed to test the calculations of the necessary amount of quench water in the waste tank to handle safely the quick removal of the hot liquid in the pressurized system. The over pressure and vacuum relief valves were also tested to assure their proper responses. The temperature, pressure, and flow rate of the system was monitored and recorded in a permanent record.

The next major section was concerned with the solvent injection and circulation. The procedures gave step by step directions on filling, venting, and controlled heat up of the cleaning system. The solvent

chemistry was to be periodically sampled, checked and recorded. The third portion described the cooling, draining, and rinsing of the system. Controls were imposed as to the proper disposal of liquids and minimum acceptable rinse water standard to allow the return of the units to Philadelphia Electric Company.

Finally, the solidification of wastes that could not be handled by Peach Bottom's existing radwaste system was detailed. These procedures were submitted and approved by the Peach Bottom Plant Operations Review Committee.

The testing stages as described earlier for both Peach Bottom 2 and 3, were completed approximately one week before the Dow work crews were scheduled to arrive. With minor exceptions such as valve packing leaks, unlabeled valves and last minute adjustments, all systems performed well.

The crew arrived three days before the NS-1 was scheduled to be injected into the system. This lead time was necessary for Health Physics requirements, security badges, full body counts, system inspection by the work crew, and a final briefing with the necessary crews and support personnel.

Solvent Addition

The NS-1 Solvent, which was packaged and shipped in polyethylene lined 55 gallon barrels, was moved to the work area. The solvent was then moved to a radioactivity clean area near the temporary cleaning

equipment. A self-priming air powered barrel pump was used to inject solvent at the rate of approximately 15 gpm until the system was filled. All high points were vented and NS-1 was injected to assure a full system. The calculated volume needed to fill was 650 gallons. The volume of NS-1 used to fill the system was approximately 625 gallons at Peach Bottom 3, and 605 gallons at Peach Bottom 2.

Circulation

Circulation was then established and heating of the solvent began. The Peach Bottom normally allows a heat up rate of 100°F per hour. As a safety margin, the procedures for the chemical decontamination limited the heat up/cool down rate to 50°F per hour. The solvent steam pressures & temperatures were monitored and recorded on data log sheets for a permanent record. A sample tap was located on the discharge pipe of the circulation pump. Samples were taken at 30 minute intervals for the first 6-8 hours of NS-1 Solvent contact. The samples were then taken on an hourly basis for approximately the next 12 hours and then on a two hour sample time for the rest of the chemical decontamination stages. Residual NS-1 capacity, dissolved Iron and Cobalt 60 were analyzed. Figure 4 and 5 is a composite graph of selected analytical data generated on Peach Bottom 3, in April, 1977 and on Peach Bottom 2 from September 22 to September 25, 1977 respectively. The final data for the solvent is as follows:

PEACH BOTTOM 3 SOLVENT CONTACT
(From April 15, 1977 to April 19, 1977)

Total hours solvent contact at 250°F----------48 hours.
Residual NS-1 capacity at termination----------78%
Iron concentration (maximum detected)----------600 µg/ml

Peach Bottom 3 Solvent Contact - con't.

Cobalt 60 activity (maximum detected)--------1.4 μCi/ml
Total Iron removed----------1153 gms
Total Radioactivity removed ----------10.6 curies

Table VII gives an isotopic breakdown of the activity removed.

PEACH BOTTOM 2 SOLVENT CONTACT
(From September 22 to September 25, 1977)

Total hours of solvent contact at 250°F---- 44 hours
Residual NS-1 capacity at termination------- 70.3%
Iron concentration (maximum detected)------- 900 μg/ml
Cobalt 60 activity (maximum detected)------- 1.6 μg/ml
Total Iron removed---------- 2100 gms
Total Radioactivity removed---------- 6.3 curies

Table VIII gives an isotopic breakdown of the activity removed.

During the solvent run, the piping system was inspected approximately every two hours. Any unusual or abnormal conditions were noted in the engineer's log book and corrected when feasible. The problem of leaks was addressed and planned for during the design phases. The heat exchanger gaskets themselves were leaking and could not be sealed off. These existing leak points had spray belts wrapped around them with any liquid directed to an installed temporary drip pan. Many other small drip pans were made and placed in the area for use in the event of small unexpected leaks such as valve packings, flanged gaskets, or threaded connections. These pans were emptied during the inspections and the waste placed in a lead shielded waste drum to be solidified at a later time. It is also important to note that while leaks were

experienced on both Peach Bottom 2 and 3, no airborne activity was generated.

The termination of the NS-1 stage was based on the relative stability of the previously mentioned analytical parameters. If the residual NS-1 concentration was not decreasing nor the Iron concentration, and Cobalt 60 activity increasing for an eight to twelve hour period, the chemical contact stage was considered completed and the cooling sequences initiated. As mentioned before, at Peach Bottom 3, the solvent stage was of 48 hours duration. In the case of Peach Bottom 2, the solvent was in contact for 44 hours before a weld failure in the solvent return lines forced an emergency dump to the quench tank. From the analytical data it can be seen that the NS-1 solvent conditions had been relatively stable for the final 18 hours of the run indicating that the majority of the deposit had been removed.

Drain and Flush

The solvent was cooled and drained under a nitrogen blanket to the Decon Waste Storage Tank #1 for later solidification. The rinse cycles were basically filling the system with demineralized water, circulating the water, sampling the rinse water for purity and then draining the system in much the same way as the solvent was handled. Of course the purpose of rinsing and flushing the system was to remove any residual NS-1 left in the equipment or piping. The rinses were tested for residual NS-1, pH, conductivity, radioactivity, and Iron. Criteria for rinse water quality were set forth in the procedures. From the laboratory results it was then decided if the rinse water should be

barreled and/or allowed to go to the floor drains at a limited rate. In the above case, the rinse water was diluted with large volumes of water in the existing radwaste treatment system to eventually be processed through the demineralizers. If the rinse water was outside the criteria stated in the procedures, it was to be drained to the Decon waste storage tanks to be mised with the solvent for eventual solidification. In the case of both Peach Bottom 2 and 3, the rinses were of sufficient quality to be treated by the Peach Bottom radwaste system.

The system was rinsed until the conductivity of the water being circulated was no more than 20 μmhos/cm. The final rinse of Peach Bottom 3 was 6.2 μmhos/cm and similarly, at Peach Bottom 2, 5.35 mhos/cm. The system was turned over to Philadelphia Electric Company at this point.

Any waste that was now to be treated by the Dow solidification system was located in Decon Storage Tank #1. The valve line-ups were checked and Decon Pump No. 2 was used to circulate Tank #1 for three to four hours to mix the waste and to blend in a small amount of an antifoam agent.

Waste Solidification

The solidification system was comprised of the Waste Storage Tank, Decon Pump No. 2, a metering tank and an air powered mixer. By a remote switch, Decon Pump No. 2 was energized drawing suction from Decon Tank #1 and discharging to the metering tank. At a pre-deter

mined volume, the waste liquid would overflow, as observed by a liquid flow through a section of clear tubing, the excess returning to the waste tank. A 55 gallon drum that had previously been filled with the prescribed amount of binder and promotor was locked into position at the mixer. The air powered mixer was lowered and the mixing began. The valve on the metering tank was cracked open and the waste slowly blended into the barrel. A shroud had been attached near the top of the barrel. This shroud was connected by a flexible hose to a portable HEPA Filter to eliminate any vapors or airborne particles generated during the mixing. After the metering tank had emptied, the catalyst was injected into the barrel and mixed. The air motor for the mixer was shut off and the mixing head raised. With a drip pan moved under the shaft of the mixer, the full barrel was rolled out from under the mixing unit to a curing area. Another "prepped" drum was placed under the mixer and the process continued. The mixed drums were allowed to cure for approximately one hour and then checked for hardness. With Philadelphia Electric Health Physics approval, the lids were sealed and bolt rings installed. Each barrel was wipe tested and surveyed by Health Physics. This information was recorded in a permanent record. After the tests, the barrels were removed to a temporary storage area to be properly disposed of by Philadelphia Electric Company. At Peach Bottom 3, a total of 34 barrels were solidified with a surface radiation dose ranging from 1,000 to 1,200 mr/hr. At Peach Bottom 2, 38 barrels were solidified with surface radiation dose ranging from 350 mr/hr to 800 mr/hr.

Table

RADIOISOTOPE IDENTIFICATION AND QUANTIFICATION OF PEACH BOTTOM DEPOSITS

ISOTOPE	ENERGY (Mev)	HALF-LIFE	PEACH BOTTOM II Section I ($\mu Ci/cm^2$)	PEACH BOTTOM II Section II ($\mu Ci/cm^2$)	PEACH BOTTOM II Section III ($\mu Ci/cm^2$)
^{65}Zn[1]	1.115	245d	1.32[1]	0.026[1]	1.10[1]
^{60}Co[2]	1.173	5.62y	1.15	0.003	0.24
^{60}Co[2]	1.332	5.62y	1.20	0.003	0.25
^{58}Co	0.810	71.3d	0.18	0.002	0.10
^{51}Cr	0.320	27.8d	N.D.[3]	0.008[4]	0.04
^{54}Mn	0.835	303d	N.D.	N.D.	N.D.
^{95}Nb	0.765	35d	N.D.	N.D.	N.D.
^{137}Cs	0.662	30.0y	N.D.	0.001	N.D.
^{137}Cs	0.606	2.05y	N.D.	N.D.	N.D.
^{134}Cs	0.606	2.05y	N.D.	0.005	N.D.

[1]Values are corrected for 50% efficienty for 1.116 Mev gamma rays of ^{65}Zn.

[2]Two gamma rays per disintergration.

[3]N.D. - Not determined, may have been present in small amounts.

[4]Values are corrected for 9% efficiency for 0.320 Mev gamma rays for ^{51}Cr.

Table I

RADIOISOTOPE IDENTIFICATION AND QUANTIFICATION OF PEACH BOTTOM DEPOSITS

ISOTOPE	ENERGY (Mev)	HALF-LIFE	PEACH BOTTOM III Section I ($\mu Ci/cm^2$)	PEACH BOTTOM III Section II ($\mu Ci/cm^2$)
^{65}Zn[1]	1.115	245d	12.68[1]	2.58[1]
^{60}Co[2]	1.173	5.62y	4.13	0.10
^{60}Co[2]	1.332	5.62y	4.95	0.10
^{58}Co	0.810	71.3d	1.24	0.03
^{51}Cr	0.320	27.8d	1.32[4]	0.33[4]
^{54}Mn	0.835	303d	0.61	0.01
^{95}Nb	0.765	35d	0.14	0.01
^{137}Cs	0.662	30.0y	0.11	0.03
^{137}Cs	0.606	2.05y	0.11	0.03
^{134}Cs	0.606	2.05y	N.D.	N.D.

[1]Values are corrected for 50% efficiency for 1.116 Mev gamma rays of ^{65}Zn.

[2]Two gamma rays per disintergration.

[3]N.D. - Not determined, may have been present in small amounts.

[4]Values are corrected for 9% efficiency for 0.320 Mev gamma rays for ^{51}Cr.

TABLE II

DISSOLUTION OF PEACH BOTTOM II DEPOSIT USING NS-1

	γ Energy (Mev)	Isotope	Original c/sec	After Cleaning c/sec	DF	% Removed
A. 1.	Section I Sample #1 - 24 hours at 250°F					
	0.69 to 0.88[1]	^{58}Co	90.4	0.75	120	99.2
	0.99 to 1.40[1]	^{65}Zn + ^{60}Co	167	1.31	127	99.2
2.	Section I Sample 2 - 69 hours at 250°F					
	0.69 to 0.88[1]		104	0.40	260	99.6
	0.99 to 1.40[1]		190	0.69	275	99.6
3.	Section I Sample #2 using Ge(Li) system - 69 hours at 250°F					
	0.512[2]	^{58}Co+$^{65}Zn(^{+}\beta)$	13.3	0.03	490	99.8
	0.812	^{58}Co	12.1	0.02	600	99.8
	1.115	^{65}Zn	30.2	0.14	216	99.5
	1.173	^{60}Co	52.3	0.22	237	99.6
	1.332	^{60}Co	46.9	0.20	235	99.6
	SURFACE AREA OF SAMPLE = 4.86 cm^2					
4.	Section I Sample #3 - 93 hours at 250°F					
	0.512	^{58}Co+$^{65}Zn(^{+}\beta)$	9.94	0.03	330	99.7
	0.812	^{58}Co	9.54	0.02	380	99.8
	1.115	^{65}Zn	22.4	0.14	160	99.4
	1.173	^{60}Co	39.5	0.21	188	99.5
	1.332	^{60}Co	35.0	0.20	175	99.4
	SURFACE AREA OF SAMPLE = 4.94 cm^2					

[1] NaI (Tl) detector

[2] Ge(Li) detector

Table II (Continued)

γ Energy (Mev)	Isotope	Original c/sec	After Cleaning c/sec	DF	% Removed
5. Section I Sample #4 - 118 hours at 250°F					
0.512	$^{58}Co+^{65}Zn(^{+}\beta)$	13.3	0.03	440	99.8
0.812	^{58}Co	13.1	0.02	655	99.8
1.115	^{65}Zn	30.7	0.14	219	99.5
1.173	^{60}Co	34.2	0.22	250	99.6
1.332	^{60}Co	48.5	0.19	255	99.6
6. Section I Sample #5 - 48 hours at 250°F					
0.512	$^{58}Co+^{65}Zn(^{+}\beta)$	11.8	0.04	289	99.6
0.812	^{58}Co	11.4	0.02	590	99.8
1.115	^{65}Zn	27.8	0.15	186	99.5
1.173	^{60}Co	48.9	0.21	232	99.6
1.332	^{60}Co	43.2	0.20	216	99.5
B. 1. Section II Sample #1 - 71 hours at 250°F					
0.321	^{51}Cr	20.7[1]	<.5	>40	>99%
0.812	^{58}Co	1.37	0.065	21	95.3
1.115	^{65}Zn	11.7	0.250	47	97.9
1.173	^{60}Co	2.78	0.270	10	90.3
1.332	^{60}Co	2.48	0.290	8.5	88.3
2. Section II Sample #4 - 48 hours at 250°F					
0.321	^{51}Cr	17.6	0.15	117	99.2
0.812	^{58}Co	1.31	0.06	22	95.4
1.115	^{65}Zn	10.5	0.19	55	98.2
1.173	^{60}Co	2.42	0.42	5.8	82.6
1.332	^{60}Co	2.27	0.35	6.5	84.6

[1] Different sample counting position than Section I, same as Section III.

Table II (Continued)

γ Energy (Mev)	Isotope	Original c/sec	After Cleaning c/sec	DF	% Removed
B. 3. Section II Sample #2 - 48 hours in NS-3					
0.321	^{51}Cr	2.24	0.06	37	97.3
0.511	$^{58}Co+^{65}Zn(^{+}\beta)$	1.17	<.05	>23	>96%
0.812	^{58}Co	1.87	0.03	62	98.4
1.115	^{65}Zn	3.43	0.11	31	96.8
1.173	^{60}Co	4.05	0.11	37	97.3
1.332	^{60}Co	3.58	0.14	26	96.1
C. 1. Section III Sample #1 - 71 hours at 250°F					
0.321	^{51}Cr	164[1]	0.27	607	99.8
0.511	$^{58}Co+^{65}Zn(^{+}\beta)$	130	13.8	9.4	89.4
0.812	^{58}Co	114	4.88	23	95.7
1.115	^{65}Zn	728	99.7	7.3	86.3
1.173	^{60}Co	317	91.4	3.5	71.2
1.332	^{60}Co	290	79.8	3.6	72.5
2. Section III Sample #2 - 44 hours at 250°F					
0.321	^{51}Cr	71.3[1]	0.40	180	99.4
0.511	$^{58}Co+^{65}Zn(^{+}\beta)$	70.6	12.9	5.5	81.2
0.812	^{58}Co	53.0	3.25	16.3	93.9
1.115	^{65}Zn	346	90.5	3.8	73.8
1.173	^{60}Co	153	72.5	2.1	52.6
1.332	^{60}Co	138	60.1	2.3	56.4
3. Section III (14") Sample #5 Dry Cut, 48 hours at 250°F					
0.321	^{51}Cr	127	0.75	170	99.4
0.511	$^{58}Co+^{65}Zn$	91.8	15.3	6.0	83.3
0.812	^{58}Co	74.1	5.52	13.4	92.6
1.115	^{65}Zn	517	131	3.95	74.7

Table II (Continued)

γ Energy (Mev)	Isotope	Original c/sec	After Cleaning c/sec	DF	% Removed
C. 3. Section III (14") Sample #5 Dry Cut, 48 hours at 250°F (continued)					
1.177	^{60}Co	184	56.5	3.26	69.3
1.332	^{60}Co	162	50.6	3.20	68.8
4. Section III Sample #3, 48 hours in NS-3					
0.321	^{51}Cr	108	0.60	180	99.4
0.511	$^{58}Co+^{65}Zn$	81.4	6.33	12.9	92.2
0.812	^{58}Co	71.4	2.16	33.1	97.0
1.115	^{65}Zn	449	44.9	10.1	90.0
1.177	^{60}Co	185	43.8	4.22	76.3
1.332	^{60}Co	164	39.6	4.14	75.8

[1] Different sample counting position that Section I, same as Section II.

Table III

DISSOLUTION OF PEACH BOTTOM III DEPOSIT USING NS-1

	γ Energy (Mev)	Isotope	Original c/sec	After Cleaning c/sec	DF	% Removed
A. 1.	Section I, Sample #1 - 48 hours at 250°F					
	0.321	^{51}Cr	19.5	0.33	59.1	98.3
	0.512	$^{58}Co+^{65}Zn$	64.5	6.31	10.2	90.2
	0.812	^{58}Co	62.1	2.50	24.8	96.0
	0.834	^{54}Mn	31.0	0.33	93.9	98.9
	1.115	^{65}Zn	217	33.0	6.58	84.8
	1.173	^{60}Co	136	6.12	22.2	95.5
	1.332	^{60}Co	122	5.68	21.5	95.3
2.	Section I, Sample #1 - 120 hours at 250°F*					
	0.321	^{51}Cr	19.5	.341	56.5	98.2
	0.512	$^{58}Co+^{65}Zn$	64.5	5.55	11.6	91.4
	0.812	^{58}Co	62.1	2.55	24.4	95.9
	0.834	^{54}Mn	31.0	0.49	63.3	98.4
	1.115	^{65}Zn	217	31.0	7.00	85.7
	1.173	^{60}Co	136	4.95	27.5	96.4
	1.332	^{60}Co	122	4.90	24.9	96.0
3.	Section I, Sample #2 - 70 hours at 250°F					
	0.321	^{51}Cr	12.5	0.27	46.3	97.8
	0.512	$^{58}Co+^{65}Zn$	45.3	3.07	14.8	93.2
	0.812	^{58}Co	42.9	1.58	27.2	96.3
	0.834	^{54}Mn	20.6	0.23	89.6	98.9
	1.115	^{65}Zn	143	20.7	6.91	85.5
	1.173	^{60}Co	91.4	3.30	27.7	96.4
	1.332	^{60}Co	80.9	2.93	27.6	96.4

Table III (Continued)

γ Energy (Mev)	Isotope	Original c/sec	After Cleaning c/sec	DF	% Removed
A. 4. Section I, Sample #2 - 94 hours at 250°F*					
0.321	^{51}Cr	12.5	0.46	27.0	96.3
0.512	$^{58}Co+^{65}Zn$	45.3	3.33	13.6	92.6
0.812	^{58}Co	42.9	1.22	35.2	97.2
0.834	^{54}Mn	20.6	0.25	82.4	98.8
1.115	^{65}Zn	143	18.7	7.65	86.9
1.173	^{60}Co	91.4	3.25	28.1	96.4
1.332	^{60}Co	80.9	2.80	28.9	96.5
B. 1. Section II (elbow) Sample #1 - 48 hours at 250°F					
0.321	^{51}Cr	3.34[1]	0.80	4.18	76.0
0.512	$^{58}Co+^{65}Zn$	3.34[1]	1.59	2.10	52.4
0.812	^{58}Co	0.89	0.26	3.38	70.4
0.834	^{54}Mn	0.17	----	>10	----
1.115	^{65}Zn	28.4	12.93	2.20	54.5
1.173	^{60}Co	2.22	0.83	2.67	62.6
1.332	^{60}Co	2.00	0.70	2.86	65.0
2. Section II (elbow) Sample #1 - 120 hours at 250°F					
0.321	^{51}Cr	3.34	0.27	12.4	91.9
0.512	$^{58}Co+^{65}Zn$	3.34	0.86	3.88	74.2
0.812	^{58}Co	0.8	0.18	4.89	79.6
0.834	^{54}Mn	0.17	<.05	>3.40	----
1.115	^{65}Zn	28.4	5.74	4.95	79.8
1.173	^{60}Co	2.22	0.44	5.05	80.2
1.332	^{60}Co	2.00	0.37	5.41	81.5

*Enough NS-1 Solvent used.

Table III (Continued)

	γ Energy (Mev)	Isotope	Original c/sec	After Cleaning c/sec	DF	% Removed
B. 3.	Section II (elbow) Sample #2 - 70 hours at 250°F					-
	0.321	^{51}Cr	3.12	0.48	6.50	84.6
	0.512	$^{58}Co+^{65}Zn$	3.32	0.98	3.39	70.5
	0.812	^{58}Co	0.89	0.18	4.94	79.3
	0.834	^{54}Mn	0.25	0.004	62.5	98.4
	1.115	^{65}Zn	27.6	7.95	3.47	71.2
	1.173	^{60}Co	2.25	0.52	4.33	76.9
	1.332	^{60}Co	2.02	0.46	4.39	77.2
4.	Section II (elbow) Sample #2 - 94 hours at 250°F					
	0.321	^{51}Cr	3.12	0.09	34.7	97.1
	0.512	$^{58}Co+^{65}Zn$	3.32	0.99	3.35	84.3
	0.812	^{58}Co	0.89	0.08	11.1	91.0
	0.834	^{54}Mn	0.25	0.06	4.17	76.0
	1.115	^{65}Zn	27.6	6.23	4.43	77.4
	1.173	^{60}Co	2.25	0.40	5.63	82.2
	1.332	^{60}Co	2.02	0.42	4.81	79.9

Table IV

CHEMICAL ANALYSIS OF PEACH BOTTOM DEPOSITS

SAMPLE	µg Fe/cm^2	µg Cu/cm^2	µg Ni/cm^2	µg Cr/cm^2	µg Zn/cm^2
A. PEACH BOTTOM II					
1. Section I, Sample #3	330[1]	3.85[1]	40[1]	53[1]	
Section I, Sample #3 after Decon	20	N.D.[2]	N.D.[2]	N.D.[2]	
Net	310	3.85		53	
		4.5[3]			
2. Section II, Sample #1	275	Not determined	14	30	
3. Section III, Sample #1	700	Not determined	42	44	
B. PEACH BOTTOM III					
1. Section I, Sample #1	612[1]	25[1]	57[1]	175[1]	181[1]
Section I, Sample #2	563	22	49	142	141
2. Section II, Sample #1	341	<1	21	58	42
Section II, Sample #2	228	<1	10	66	37

[1] Atomic Absorption Analysis

[2] N.D. - Not detedted, may have been present in very small amounts

[3] X-ray fluorescence

Table V

INSOLUBLE MATERIAL AFTER DECONTAMINATION

Weight of sloughed-off and undissolved material in Peach Bottom II Samples.

SAMPLE	Inner Surface Area (cm^2)	Wt. of Residue(g)	Wt. per cm^2
Section I, Sample #5	5.00	0.032	0.006
Section II, Sample #2	4.68	0.033	0.007
Section III, Sample #3	4.23	0.32	0.008
Section III, Sample #5	3.51	0.038	0.011

Table VI

Radioactivity of undissolved material in Peach Bottom II Samples

γ Energy	Isotope	Original c/sec	Residue c/sec	% Activity on Filter
1. Section I, Sample #5				
0.321	^{51}Cr	M.D.	0.20	-----
0.512	$^{50}Co+^{65}Zn$	354	0.83	0.2
0.812	^{58}Co	342	0.65	0.2
1.115	^{65}Zn	834	4.55	0.6
1.173	^{60}Co	1467	10.8	0.7
1.332	^{60}Co	1296	9.77	0.7
2. Section II, Sample #2				
0.321	^{51}Cr	67.2	----	<0.1
0.512	$^{58}Co+^{65}Zn$	35.1	0.06	0.2
0.812	^{58}Co	56.1	0.02	<0.1
1.115	^{65}Zn	103	0.43	0.4
1.173	^{60}Co	122	0.57	0.5
1.332	^{60}Co	107	0.61	0.6

Table VI (Continued)

γ Energy	Isotope	Original c/sec	Residue c/sec	% Activity on Filter
3. Section III, Sample #3				
0.321	^{51}Co	108	--	<0.1
0.512	$^{50}Co+^{65}Zn$	81.4	0.25	0.3
0.812	^{58}Co	71.4	--	<0.1
1.115	^{65}Zn	449	0.52	0.1
1.173	^{60}Co	185	0.65	0.4
1.332	^{60}Co	164	0.67	0.4
4. Section III, Sample #5				
0.321	^{51}Cr	127	0.78	0.6
0.512	$^{58}Co+^{65}Zn$	91.8	0.51	0.6
0.812	^{58}Co	74.1	0.41	0.6
1.115	^{65}Zn	517	3.40	0.7
1.173	^{60}Co	184	3.50	1.9
1.332	^{60}Co	162	2.81	1.7

Table VII

RADIOSOTOPES REMOVED FROM PEACH BOTTOM III

REGENERATIVE HEAT EXCHANGER

Radioisotope	μ Ci/ml	Error	Ci/System (625 ga
^{60}Co	1.25	±2%	2.94
^{65}Zn	2.72	±2%	6.39
^{134}Cs	0.07	±15%	0.16
^{137}Cs	0.09	±10%	0.21
^{58}Co	0.18	±6%	0.42
^{54}Mn	0.15	±7%	0.35
^{51}Cr	0.06	±17%	0.14
Total Ci/system			10.6

Table VIII

RADIOSOTOPES REMOVED FROM PEACH BOTTOM II

REGENERATIVE HEAT EXCHANGER

Radiosotope	μ Ci/ml	Error	Ci/System (605 gal)
^{60}Co	1.48	±3%	3.38
^{65}Zn	1.22	±5%	2.79
^{54}Mn	$2.9X10^{-2}$	±5%	0.07
^{58}Co	$2.5X10^{-2}$	±5%	0.06
^{137}Cs	$1.1X10^{-3}$	±50%	0.002
^{57}Co	$8X10^{-4}$	±20%	0.002
Total Ci/system			6.30

TABLE IX

REGEN. HEAT EXCHANGER DOSE RATES mR/HR.

NO.	Contact Readings- LOCATION	Before H_2O Flush	Before NS-1 Flush	After NS-1 Flush
1.	Channel Drain	600	400	75
2.	Channel Drain	800	400	15
3.	Channel Vent	2000	500	150
4.	Channel Vent	2000	550	100
5.	Channel Drain	6000	500	100
6.	Channel Drain	5000	550	80
7.	Channel Vent	600	300	130
8.	Channel Vent	800	500	75
9.	Channel Drain	8000	350	200
10.	Channel Drain	15000	350	75
11.	Channel Vent	300	200	75
12.	Channel Vent	300	200	50
13.	Bottom of Channel Hd.	---	200	15
14.	Bottom of Channel Hd.	---	250	15
15.	Bottom of Channel Hd.	---	200	15
16.	Shell to Channel Joint	---	200	50
17.	Shell to Channel Joint	---	300	50
18.	Shell to Channel Joint	---	250	50
19.	Shell Flange	2000	700	100
20.	Shell Flange	2000	600	140
21.	Shell Flange	2000	500	150
22.	Channel Outlet	500	600	75
23.	4" Crossover (Channel)	---	400	60
24.	4" Crossover (Shell)	---	400	100
25.	4" Crossover (Channel)	---	400	80
26.	4" Crossover (Channel)	---	350	150
27.	4" Crossover (Channel)	---	350	100
28.	Channel Inlet	1000	200	50
29.	Channel Outlet	---	400	100
30.	Shell Inlet	---	200	50
31.	4" Crossover (Shell)	---	100	30
32.	End of Shell	---	70	30
33.	End of Shell	---	350	100
34.	End of Shell	---	1000	75
35.	Shell Drain	---	1000	700
36.	Midsection of Bottom Shell	---	500	175
37.	Midsection of Middle Shell	---	500	200
38.	Midsection of Top Shell	---	2000	325
GENERAL AREA DOSE RATE MR/HR AVERAGE		265	250	35*

*After removal of 11 Curies

(Co_{60}, Zn_{65}, Mn_{54})

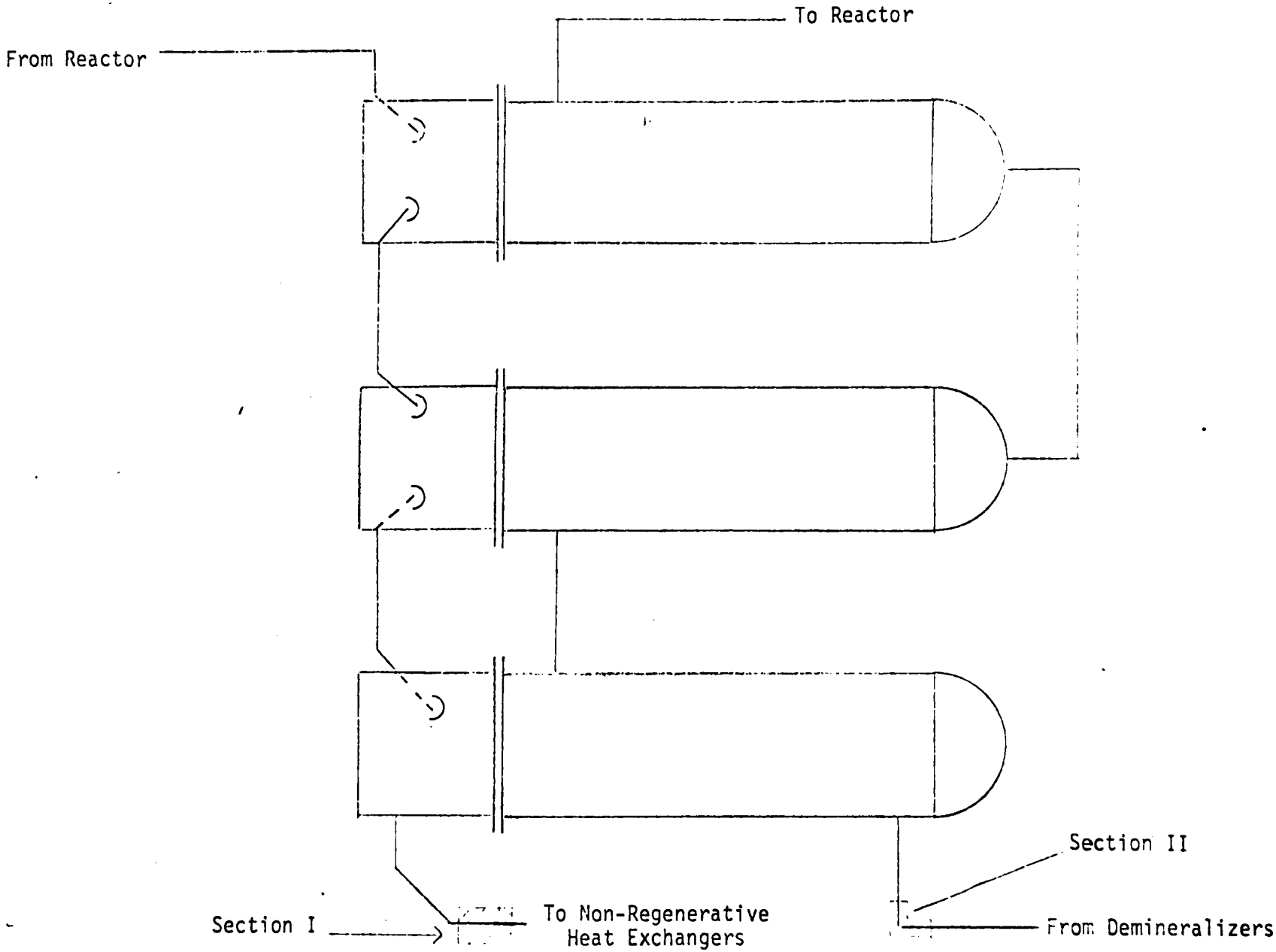
TABLE X

REGEN. HEAT EXCHANGER DOSE RATES mR/HR.

NO.	Contact Readings - LOCATION	Before H_2O Flush	Before NS-1 Flush	After NS-1 Flush
1.	Channel Drain	2000	1500	600
2.	Channel Drain	2500	1500	500
3.	Channel Vent	7000	6000	200
4.	Channel Vent	2500	3500	250
5.	Channel Drain	2500	2200	275
6.	Channel Drain	2000	2000	300
7.	Channel Vent	3000	2000	400
8.	Channel Vent	2500	3000	350
9.	Channel Drain	2800	5000	500
10.	Channel Drain	2500	2000	400
11.	Channel Vent	1500	800	350
12.	Channel Vent	1500	1000	350
13.	Bottom of Channel Hd.	---	500	125
14.	Bottom of Channel Hd.	---	500	110
15.	Bottom of Channel Hd.	---	400	200
16.	Shell to Channel Joint	500	350	150
17.	Shell to Channel Joint	600	150	125
18.	Shell to Channel Joint	2500	250	150
19.	Shell Flange	600	---	220
20.	Shell Flange	700	---	280
21.	Shell Flange	2500	---	400
22.	Channel Outlet	500	600	150
23.	4" Crossover (Channel)	500	500	180
24.	4" Crossover (Shell)	700	400	200
25.	4" Crossover (Channel)	700	1800	150
26.	4" Crossover (Channel)	1500	800	125
27.	4" Crossover (Channel)	2000	1000	150
28.	Channel Inlet	500	600	200
29.	Channel Outlet	500	300	100
30.	Shell Inlet	150	50	40
31.	4" Crossover (Shell)	75	75	50
32.	End of Shell	75	50	50
33.	End of Shell	150	75	150
34.	End of Shell	100	75	80
35.	Shell Drain	100	3000	600
36.	Midsection of Shell (Bottom)	150	100	150
37.	Midsection of Shell (Middle)	300	300	300
38.	Midsection of Shell (Top)	2800	700	125
GENERAL AREA DOSE RATE MR/HR AVERAGE		350	300	60*

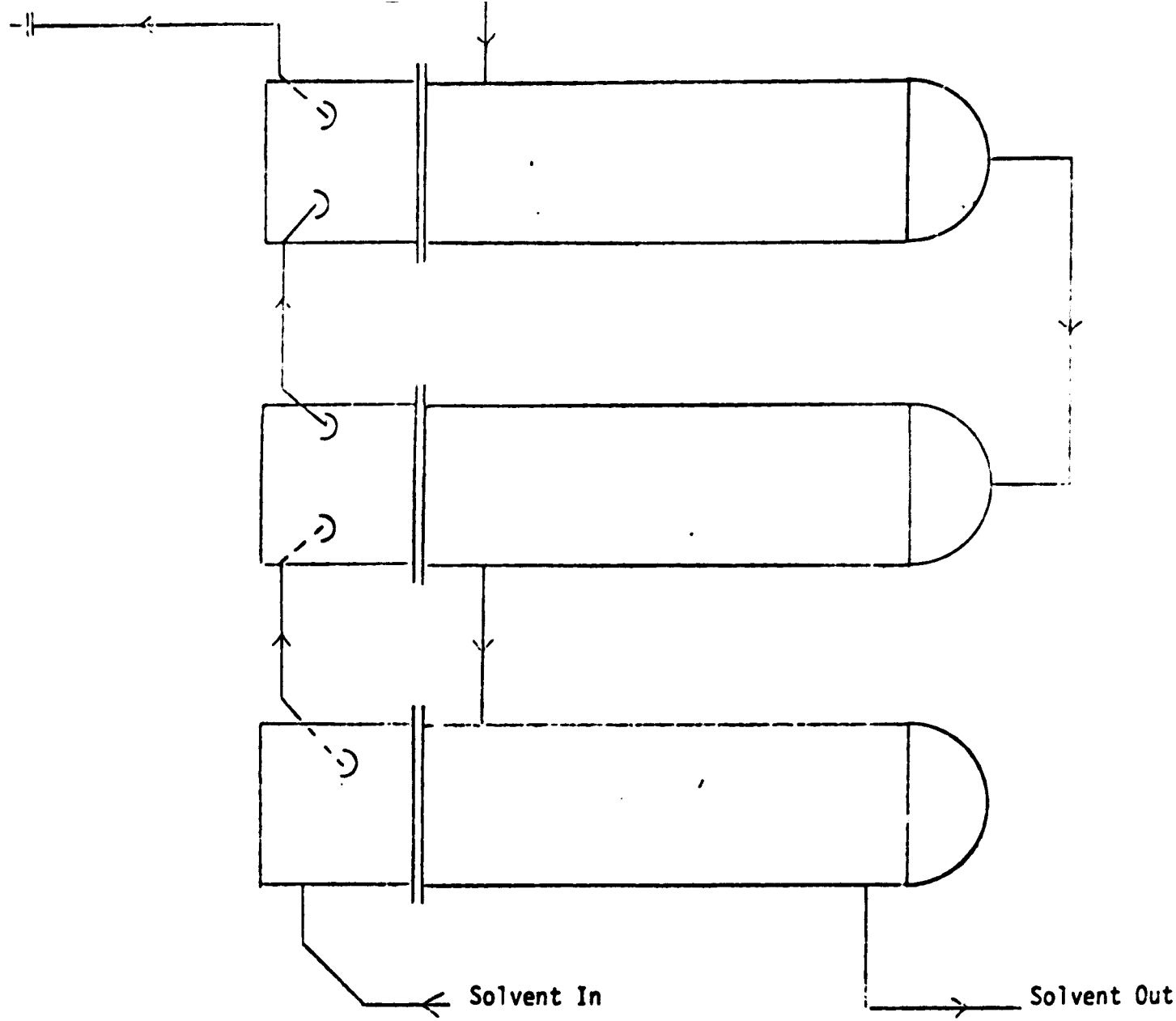
*After removal of 7 Curies

(CO_{60}, ZN_{65}, MN 54)



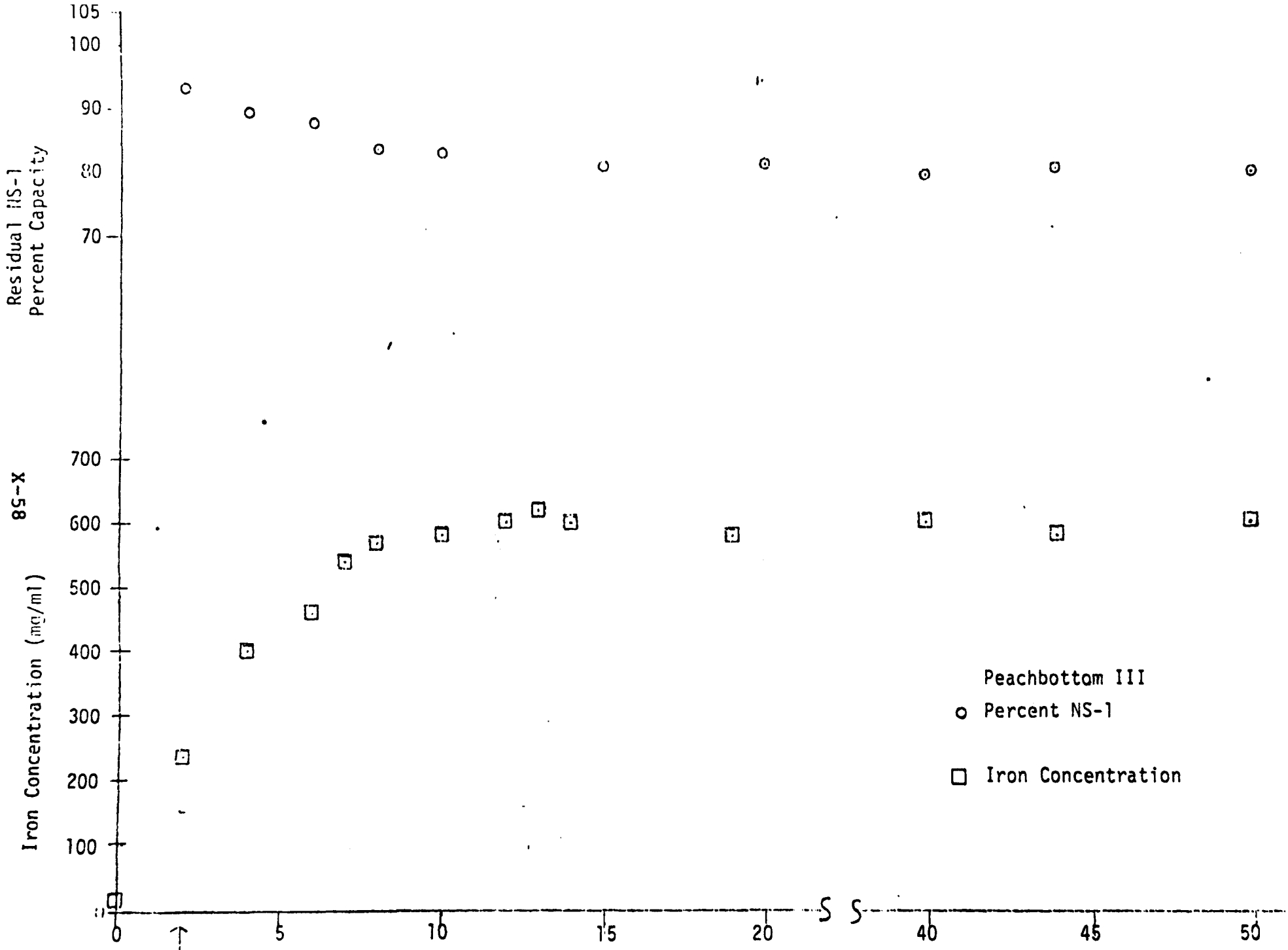
Peachbottom III

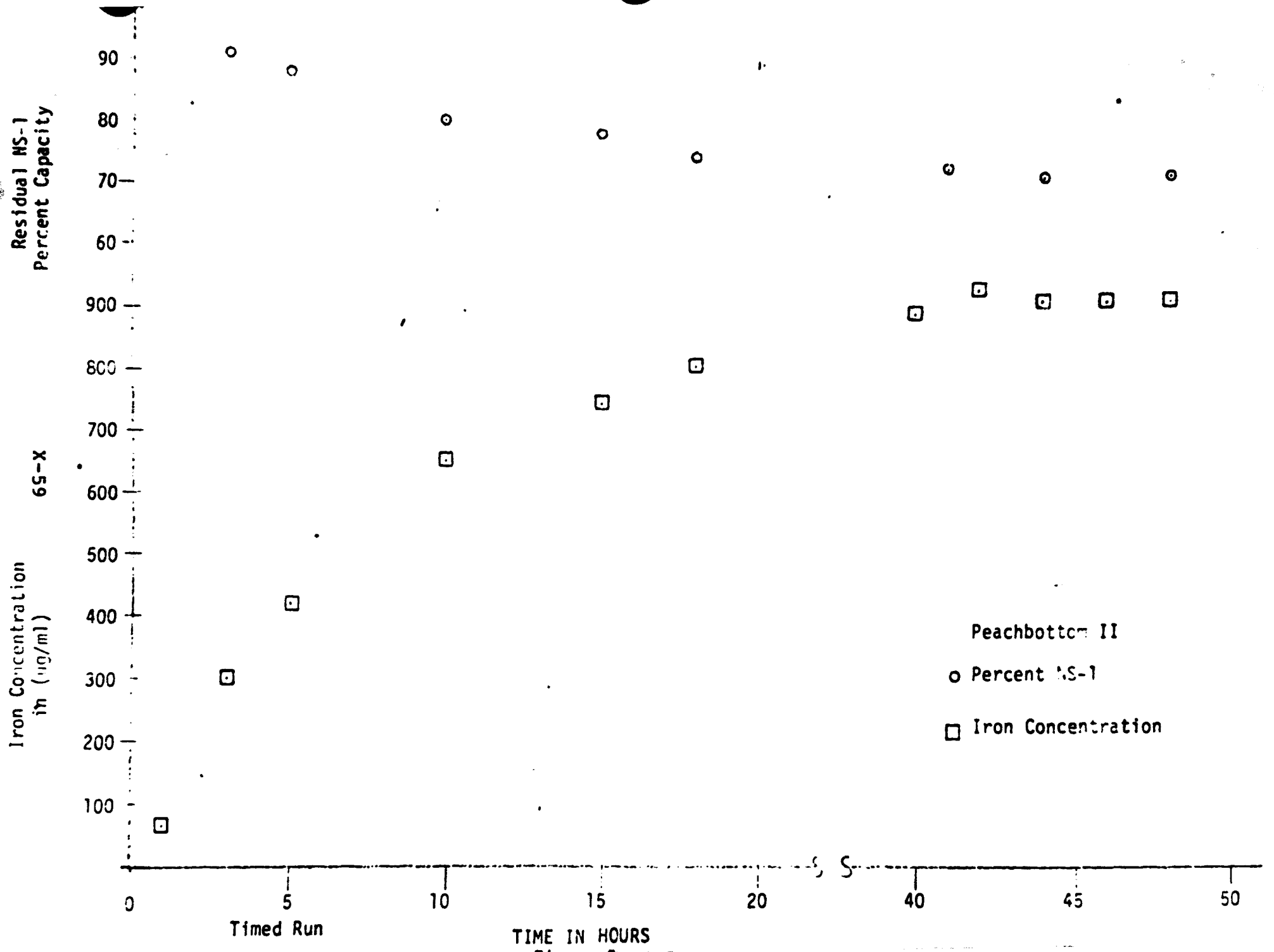
Figure 2

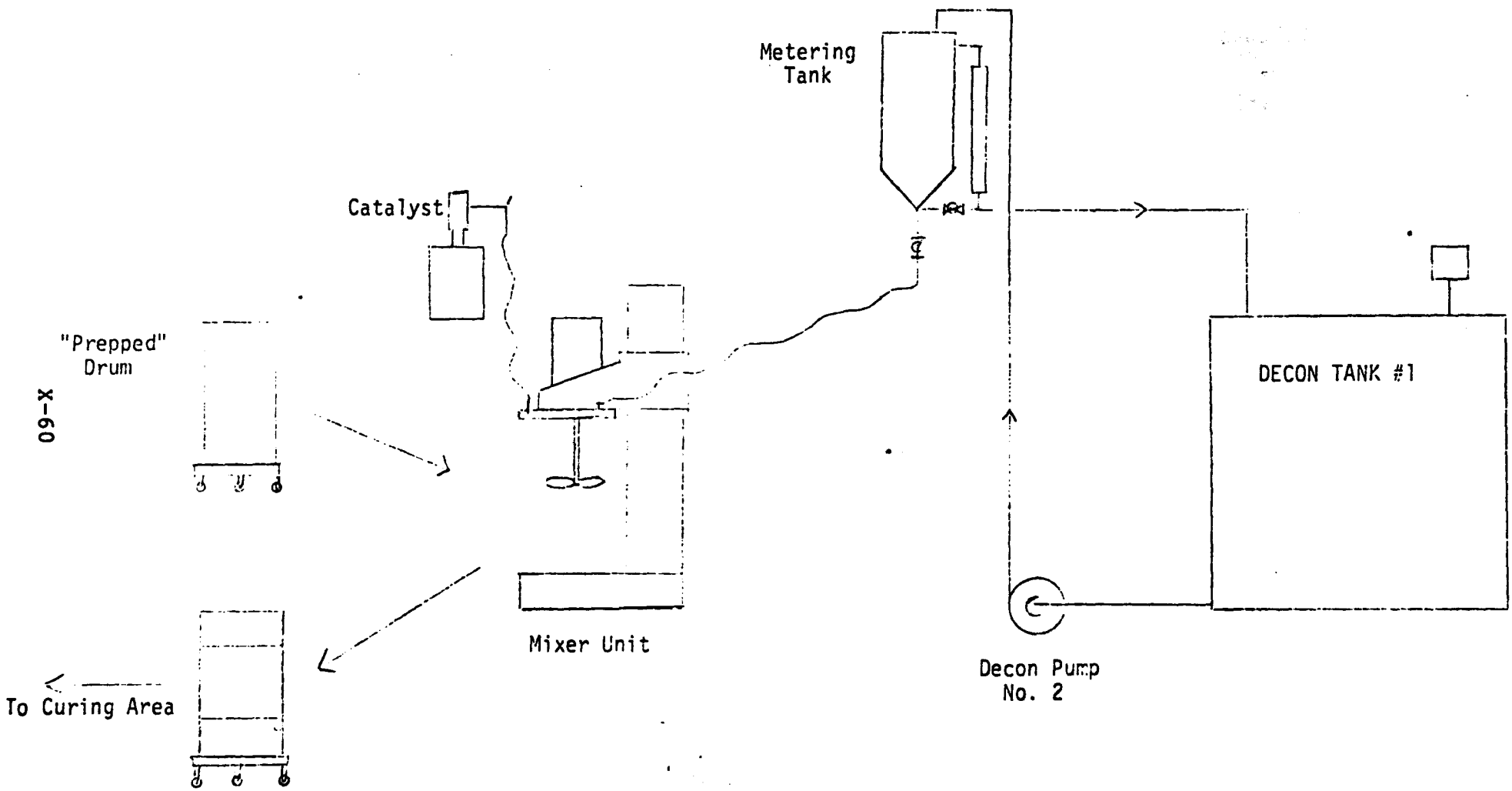


SOLVENT FLOWPATH

Figure 3


Residual NS-1
Percent Capacity
105
100
90
80
70
Iron Concentration (mg/ml)
700
600
500
400
300
200
100
0
5
10
15
20
40
45
50
Peachbottom III
Percent NS-1
Iron Concentration


Residual NS-1 Percent Capacity
Iron Concentration in (ug/ml)
65-X
90
80
70
60
900
800
700
600
500
400
300
200
100
0
5
10
15
20
40
45
50
Timed Run
TIME IN HOURS
Peachbottom II
Percent NS-1
Iron Concentration



SOLIDIFICATION FLOW DIAGRAM

Figure 6

ACKNOWLEDGEMENTS

Acknowledgement is made to the following departments for their contributions to this report.

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Health Physics & Chemistry, Philadelphia Electric Company

SESSION Y

DECONTAMINATION EXPERIENCE AT THE SURRY PLANT

A.L. Parrish III
VEPCO

Surry Plant is located approximately 50 miles southeast of Richmond and 17 miles upriver from Newport News on James River at Hog Island, Surry County, Virginia. It is a 2 unit 822 MW 3 loop Westinghouse NSSS Power with common auxiliary and T/G Buildings. Stone & Webster was the A-E and constructor. The 2 units initially went on the line in 1972 and 73.

I'll take a couple of minutes and relate some of the projects particulars and key philosophies, then address specifics relating to this workshop. With Unit #2's S/G's replaced and primary hydro's in progress (hydro's have just been completed) Surry Unit #2 becomes the 1st commercial plant of what is a growing line of Westinghouse plants that will replace S/G's due to denting. Almost 3 years ago the plant operations engineering group evaluated steam generator plugging trends and started to plan and engineer for steam generator replacement.

A project in the PSE&C division was established approximately 2½ years ago to handle all aspects of the replacement.

After several schedule slips caused initially by late delivery of Westinghouse steam generators, outage started February 6, 1979.

VEPCO staff of approximately 225 at peak has acted as general contractor. We handle all purchasing, accounting, warehousing, provide all key supervision, all planning and scheduling and cost, and provide all support functions such as first aid, document control. Realizing that no one contractor is best in all areas we split up the work and contracted these specialty packages to the best qualified contractors available.

Some 28 contractors were on site with a peak employment of about 600 on the steam generators and 600 on additional work such as retubing condensers, and erecting a condensate polishing system and building. This other work also allowed us to rotate people for dose considerations.

With regard to the ALARA Program, the initial estimate was 2070 manrem; the present status is 1981 manrem as of 11/21/79. We will finish this 1st unit about even with our estimate. The philosophy is to use totally separate facilities from operating plant; only the laundry is common, a permanent change room was built outside the R/C and ran clean and dirty personnel walkways to the equipment hatch. A permanent hot shop was also built to to handle refurbishment of hot reactor coolant loop piping.

The fuel was removed and the R/C area was declassified from a vital area. A point concerning insurance, even though nuclear insurance must be carried by us and provides nuclear coverage for the contractor, the non-nuclear exposure is being covered by requiring each contractor to carry his own non-nuclear insurance and workmen's compensation. Contractually, each contractor must stand legally and financially responsible for meeting applicable state and federal laws. With the current growing concern about effects of low level radiation, we feel this whole issue is best dealt with in this manner.

The Project basically consists of four (4) phases:
Phase I - Shutdown and Preparatory Activities - Which consist of items such as:

(1) Defueling,
(2) Removal of RC pump motors,
(3) Protection of containment components,
(4) Disassembly and removal or storage of plant equipment in the way of steam generator removal operations,
(5) Installation of temporary R/C ventilation system,
(6) The decon cleanup,
(7) General shielding of the R/C,
(8) Installation of S/G handling equipment.

Phase II - Removal Activities - Consisting of items such as

(1) Removal of insulation and miscellaneous piping,
(2) Cutting of steam generator girth welds and removal of steam generator upper shell,
(3) Cutting and removal of reactor coolant piping,
(4) Refurbishment of steam generator upper shell,
(5) Disassembly of steam generator supports and removal of steam generator lower assemblies.

Phase III - Installation Activities - Consisting of items such as

(1) Installation of steam generator new lower shells,
(2) Refurbishment and reinstallation of R/C piping,
(3) Installation of steam generator upper shell on the new lower and performing the girth weld,
(4) Installation of miscellaneous piping,
(5) Steam generator support system and
(6) Insulation.

Phase IV - Post Installation and Startup Activities - consisting of

(1) Removal of those items installed to support S/G exchange,
(2) Replacement of plant items removed,
(3) Flush and hydro of systems
(4) Preparation and turnover of systems involved to plant operations personnel for startup.

Conversations with Mr. Williams and Mr. Carson concerning the various activities included in this project indicate that the relative priority for purposes of this workshop should be:

1. Initial decon of reactor containment.
2. Shielding of the reactor containment and personnel protection.
3. Decon of R/C piping.

To cleanup the containment, start project in a clean condition and then contain the produced contamination at its point of origin

through the use of tents, gloveboxes and exclusion areas. This would minimize lost time and/or reduced productivity and minimize holdups/backups of personnel at frisking stations at lunchtime, breaks, and end of shifts all of which convert immediately into dollars.

The initial job of deconning took approximately 2 weeks. These were 2 12 hr. shifts 7/day/week. Approximately 25 experienced decon people per shift were used, not average jumper type personnel rather navy ELT types. Cost was approximately $200/man per day or roughly $140,000 total. Contamination levels started at an average of 100,000 DPM's (DPM/100 CM^2) with high areas of 500,000 DPM's average. This was reduced to average of less than 1000 DPM's with 3 areas (cubicles) of approximately 2000-3000 DPM's. We started up in the dome and cleaned down thru the plant to the basement. Each grating level was removed and cleaned; the grating was replaced and covered with herculite. This provided for easy future decon by mopping and prevent circulation of air currents and therefore prevent circulation of any airborne contamination that occurred.

Regarding cleaning technique, a number of cleaning compounds were tried; I guess about everything on the market. Since this is a workshop I'm going to name brands however this is not to be taken as an endorsement by VEPCO. It simply means that we found that certain items worked better than others in our applications and we want to pass along all useful information. Due to amounts of grease (snubber oil) tracked around, a degreaser was initially used (3M floor stripper half & half with water to cut the worst). Once this was used it was solidified in 55 gallon drums for disposal mixed 20 gallons with 5 bags cement in a 55 gallon drum and mixed with a 1 HP drill motor and paddle. The vast majority of deconning was then done with Spic-N-Span, water, Scotchbrite scrubbing pads and a lot of elbow grease. A full

time decon crew of approximately 6 per shift was used throughout the outage in order to continuously cleanup as we proceeded. For certain operations such as welding of RC piping, local decon was continuously performed in order to eliminate the need for welders to be in respirators.

In final analysis did not have a single occurrence of airborne that stopped all work. Had levels of 8×10^{-9} mc/ml in localized tents.

We actually lost more time to bomb threats than airborne activity. We evaluated performing a primary system's chemical decon but due to the extra work plus the additional dose that would have been expended handling the decon than without decon, we decided against it.

Shielding and personnel protection were a very important phase of SGR since our Amendment to Operating License tied us to a formal ALARA program that required that a personnel radiation exposure estimate be formulated, and reported against with actual exposures throughout the project.

Actions to reduce/minimize personnel exposure came basically from two sources:

First - The conceptual frame work of the methods to be utilized in accomplishing the task at hand was thoroughly critiqued,
- In some cases actually mocked-up full scale
- Then detailed to provide the step-by-step work packages used.
- Our intention was to perform the Project on paper; in many cases using the actual craftsmen.

and Secondly
- Innovations on the spot by on the job personnel.

- VEPCO had been able to hire/staff and have available more hot work experience than any contractor we reviewed. Again our intention was to do at least a 500% planning job and then have as much VEPCO expertise as possible on the job to make on the spot decisions.

There is no replacement for either of these - both are absolutely necessary.

Since most of work was in the 3 loop cubicles our philosophy was to start shielding with the hottest spots and work down in levels while monitoring general area decrease. Backscatter caused lot of problems in 1st cubicle. We reduced the general area levels in cubicles from 300-500 MR/HR to approximately 35-50 MR/HR. We used leadwool blankets, molded pipe halves, lead sheets, and strips, bricks and bags. We found through trial and error that the best instrument for quickly doing this type survey and shielding work to be Eberline E530N with a peanut (10450-B9) shielded (HP 220A) probe. Has 20R range, this could immediately detect the true hot spots. Iodine and Xenon had decayed off by the time we started. The radiation was mainly due to Gamma's, (70%) Cobalt 60 and 58 with some (30%) Cesium - 137 and a little Cesium 134. A shop on site produced all the shielding blankets, gloveboxes, containment tents, molded shielding, etc. The philosophy was to have total control of design as well as production of same on site around clock basis.

We expended less manrem shielding 2nd and 3rd cubicles combined than we did on the 1st one.

In the 1st cubicle, our philosophy had been to shield all the miscellaneous piping in the overhead individually while in the 2nd and 3rd cubicles we simply laid large grating on the floor attached to chainfalls, loaded sheet lead on the

grating and hoisted this shielding layer up into the overhead just under the hot piping.

As I mentioned earlier this Project is being performed under an amendment to our operating license. This amendment requires that a report of our actuals vs. estimates to sent to the NRC every 60 days. Aside from this and the other so called normal requirements we have personally felt that one of the worst occurrances that we could have had would have been one involving personnel overexposure or worse yet, personnel injuries in a highly contaminated or highly radioactive area. This 500% planning effort helped immensely in this area also.

Several things that came out of this planning:
1) Continued to use the stations step approval program which requires supervision and/or management approvals at various steps as quarterly exposure increases. For instance our programs 1st step is a 1250 mP/Qtr. signoff by H.P. Shift Supervisor and goes in steps to a 2400 mR/Qtr. signoff by our V.P. Operations. This along with a new project computer system that provides up to date shift by shift person by person accumulated doses provided a real awareness of doses to all concerned. It became evident that to provide the best in customed designed tents, gloveboxes and shielding that we must have this capability on site. We setup an around the clock shop facility to handle this. We have designed and fabricated all containment devices and shielding on site since. With this ability the personnel in containment knew that they never had to make do with less than desirable.

Realizing that a number of personnel would not have a hot work background we entered into a fairly extensive training program, in may cases using full size mockups. For instance,

our coolant loop welding was done completely with Diametrics automatic welding machines as best as I can determine this is also a first in the industry. We started about a year before the outage started with this program. We were on shifts welding mockups several months before the outage. The idea was not only to completely qualify the process and the welders but to have a complete experience of machine failure rates and why and to be able devise and perfect methods resulting in lower manrem and good welds.

Right now a laborer coming in off the street spends almost a week in orientation and training before being put to work. This includes:

A) 1 day H.P. School
   Written test - 70% score to pass - flunk twice you're out.
B) Whole body count - for our protection we've committed to ourselves to whole body count all personnel on the front end and at termination if possible. Not NRC commitment just insurance on our part.
C) Video tape orientation - Due to being an around the clock 7 day a week operation we've made extensive use of video tapes for both orientation and also specialized training.
D) Training - again may be just videotapes but for higher classification personnel includes actual performance of operations on mockups.

At present a qualified TIG welder takes about 10 days to get ready to go to work.

Refurbishment of many of the valves was performed with lower personnel exposure by simply quick cutting the valve out of the system and taking it to a shielded area in the basement for rebuilding. In most cases the associated piping was replaced with new. I'm talking about systems such as RTD bypasses, blowdown and low point drains. We figure a dose saving of a factor of 10 for this operation.

An emphasis of low as possible rework, attempting to do it right the first time results in eliminating that exposure for rework. Several of our contracts have bonus/penalty clauses based on percentage of rework experienced.

An initial extensive photographic entry into the areas in R/C where work was performed. We shot thousands of black and white pictures and I emphasize black and white so they can be blown up and retain maximum definition. The details could then be studied and plans formulated in a zero radiation area.

In-containment "rest" areas for dressed out personnel to stand by in when not actually needed in the work area for a short time were provided. These were very low radiation and/or shielded areas very close to work areas to encourage use.

Secondary water was kept in the S/G for shielding throughout the cutting apart. We then drained thru the blowdown after rigging the S/G for lift.

The reactor coolant piping cut out initially read as high as 20 R/Hr; it was deconned to 5-10 mR/Hr thru electropolishing. Electropolishing is an electrochemical process used in both laboratory and industrial applications to produce a smooth, polished surface on a variety of metals and alloys. The object to be decontaminated serves as the anode in an electrolytic cell. The passage of electric current results in the anode in an electrolytic cell. The passage of electric current results in the anodic dissolution of the surface material and, for proper operating conditions, a progressive smoothing of the surface. Any radioactive contamination of the surface or entrapped within surface imperfections is removed and released into the electrolyte by this surface dissolution process. The production of a polished surface also facilitates the removal of residual electrolyte by rinsing to leave a contamination-free surface.

The pipe was cut out with plasma arc in 20 minutes. We installed shield caps and rigged to the decontamination tank in the basement. We deconned by electropolishing and by hand. We cut off old heat affected zone in the pipe refurbishment shop.

Session Z

DECONTAMINATION AT THE NEVADA TEST SITE

Arden E. Bicker

REECO

INTRODUCTION

The Nevada Test Site is devoted in large part to the testing of nuclear explosive devices. One result of sample recovery and other experimental work is the radioactive contamination of facilities and equipment. An adjunct activity in past years was the now moribund nuclear rocket development program. During the years of peak activity, numerous experimental reactors were operated and subsequently disassembled for examination. The contamination from fission and activation products was considerable, and periodic decontamination of facilities and equipment was required. More recently, a facility built for the rocket program has been used for such activities as disassembly and fuel-crushing of the nuclear ramjet reactor, and for the dry handling and storage of spent commercial reactor fuel bundles.

The points I shall make which hopefully will have relevance to TMI are for the most part qualitative in nature and will be limited to the areas of personnel control, decontamination agents and techniques.

PERSONNEL CONTROL

The importance of <u>experience</u> in performing decon work of highly contaminated structures cannot be over-emphasized. It has been our practice that initial early entries are manned only by personnel with experience in their specialty and with the particular facility. If the latter is not available, their experience with key elements is a must.

Training and procedures are of course required. Establishment of appropriate guides regarding actions and decision points are necessary. Dry runs are advisable in high-exposure situations.

Protective equipment in use at the NTS, including anti-contamination clothing and respiratory protective gear, are standard. We do currently place considerable emphasis on respirators and their use, and we have found that the effort is worth it. All personnel who may use such devices are given training, and qualitative and quantitative fitting on an initial and periodic basis. The fitting is done for the four full-face masks in inventory and individual use limit factors are assigned based on test results. In practice a maximum limit of 50 times MPC is imposed for air purifying respirators. Airline respirators are used for high concentrations or where the operation permits, and air supplied directly by compressors is filtered and is monitored for CO. Communications between entry team members and supervisors must be clear and reliable. We have had the best experience, when using respirators and full dress-out in our situations, with sound powered hardwire systems. Decontamination of personnel exiting highly-contaminated areas involves use of sufficient step-off and monitoring stations. Washdown facilities are necessary, and their multiple use per person should be anticipated. Accidents, especially those involving skin penetration, should be anticipated and appropriate monitoring devices available.

DECONTAMINATION AGENTS

Our experience has shown that, except for certain materials or configurations, a standard approach is to use water in several steps. First cold water, which will almost never cause fixation, is used. When the decontamination factor is

no longer sufficiently large (a subjective judgement), progression to hot water with detergent is done. Steam may be used, although our experience shows that steam generally has no great advantage over pressurized hot water (180°).

Alcohol in the ethanol form is used where water may harm the object. Ethanol is preferred over isopropyl in most cases because of less residue. Freon-22 has been used successfully in the decontamination of items such as electric motors and for inaccessible locations where other solvents are inappropriate. Caustics are used to remove oxidized layers of ferrous metals. Acids, including nitric, hydrochloric, sulphuric and phosphoric, are used for spot removal or as a last resort. Other agents include petroleum derivatives, hydrocarbon digesters, and chelating agents. Abrasives of various kinds are used, including a wet sand blast which can effectively clean large structures with little contaminant suspension.

TECHNIQUES AND EQUIPMENT

The Decontamination Facility at the NTS may be the largest in the world. One bay is used for structures or large equipment components. Pressurized solvent delivery systems are available as well as systems to properly locate operating personnel. The other bay is used for smaller parts and has soaking, agitating, and ultrasonic cleaning facilities in addition to the pressurized solvent systems. Personnel decon areas, monitoring equipment and counting systems are located adjacent to the bays. All drainage is to an evaporative pond.

It should be noted that field decontamination is used where and when feasible to minimize contamination spread during transit and to speed the pad process. This field work is done with mobile decon equipment utilizing pressurized

delivery systems.

In some facilities located in underground tunnel complexes, the need to have routine equipment and personnel passage through contaminated areas was met with covering and fixation techniques. Physical walls coupled with appropriate ventilation characteristics were successful in effecting sufficient containment. Fixation techniques utilizing various agents including water glass (sodium silicate) have also been successfully used on large structures of different materials.

The rocket reactor disassembly work was accomplished in large hot-bay facilities. One such structure is the E-Mad, or Engine Maintenance and Disassembly building. Experimental work would routinely contaminate the entire interior with all its systems. Decon work on the bay began with water washdowns except where this solvent would be inappropriate. Freon systems were used on items such as the bridge crane and others with exposed electrical components. Freon was found to be more effective than alcohol, but the cost differential dictated the use of alcohol on many systems. Hand work utilizing kotex soaked with solvents on low-level contamination was successful in obtaining a satisfactory product.

In summary, experience at the Nevada Test Site has shown a number of agents and techniques to be viable for controlling the radioactive contamination on a wide range of objects and materials. We have also determined that experienced, trained, equipped personnel operating with adequate guidance are mandatory when dealing with high-level radioactive contamination.