R0258

FIFTH REPORT

OF THE

TMI-2 Technical Assistance and Advisory Group (TAAG)

April 1, 1983

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Plenum Cover Plate

page 36

Fifth Report of TAAG

OUTLINE

PAGE

Fifth Report of TAAG

I. Introduction

This report covers the TAAG activities during the period of time from December 1, 1982, to April 1, 1983. In a letter, dated February 9, 1983, Mr. B. K. Kanga, Director, TMI-2, requested that TAAG address the following technical matters:

(1) Prepare for GPU/Bechtel use a draft of the section of the Safety Evaluation Report for Head Removal that addresses the matter of pyrophoricity of core debris.

(2) Comment on the GPU design studies for defueling equipment including cannisters and core sampling that may be required to confirm design requirements.

(3) Resolve the TAAG comments on the plan for Plenum Removal; develop a list of prerequisites for Plenum Removal.

(4) Assist in obtaining analyses of lead screw samples and an evaluation of the results; as a first step, prepare by January 7, 1983, the list of data to be obtained in the laboratory analyses.

(5) Continue to provide assistance related to the TAAG recommendations regarding reactor building decontamination and characterization:

a. Plans for cleaning 282' level.
b. Preparation of Radiological Engineering Plans.
c. Increase air flow and filtration.
d. Use of tunnel concept.

(6) Review documents prepared for design and installation of the fuel transfer system.

(7) Analyze "Quick Scan" results.

(8) Review head removal procedures. (This item was later discontinued).

(9) Provide assistance in technical planning and preliminary engineering related to refueling canal cleanup systems(s). Specific areas of TAAG support are:

a. Establishing water clarity requirements and methods.
b. Establishing basic Refueling Canal Cleanup System(s)/approaches.

(10) TAAG should review the Man-Rem Dose Re-assessment prepared by GPU based upon the latest information on manhours for various activities as well as the anticipated dose in the plant.

The report is organized with section for each of the above items. Attachments and figures are included with each section. An errata statement for the Third TAAG report is included at the end of this report.

II. Pyrophoricity of Core Debris During Head Removal

B. K. Kanga's (GPU/Bechtel) letter of February 9, 1983 requested that TAAG prepare a draft of the section of the Safety Evaluation Report for Head Removal that addresses the month of pyrophoricity of core debris. The following section of this report is the draft prepared by TAAG.

DRAFT

4.3.1.9 PYROPHORIC MATERIALS

Introduction and Summary

Finely divided particles of some metals, metal hydrides, and partially oxidized metals can react with oxygen and, once the reaction is initiated, with water. Initiation can occur due to thermal effects, mechanical shock, or electric spark, and can be very localized, with the reaction spreading from that point. Reaction of thin layers in open systems is not rapid, but wet powders react more vigorously than dry. Reaction is extremely unlikely (or impossible) for the materials of interest when they are completely under water, and this is one of the safest means for collection and storage (Reference 1).

In NUREG-0683 "Final Programmatic Environmental Impact Statement Related to Decontamination and Disposal of the Radioactive Wastes Resulting from March 28, 1979, Accident Three Mile Island Nuclear Station, Unit 2", NRC addressed the potential for zirconium hydride fires. This discussion stated that it was unlikely that a zirconium hydride ignition would occur and that zirconium hydride would not ignite under water. As stated above, zirconium metal fines also are not subject to pyrophoric reactions under water. Based upon these considerations, there has been no potential for a pyrophoric zirconium or zirconium hydride reaction in the pressure vessel environment to date. However, during the head lift operations, the upper surface of the plenum will be exposed to air for a period of time (estimated to be on the order of a few days). It is during this time interval that the potential for a pyrophoric reaction must be evaluated.

As discussed below, a pyrophoric reaction is considered to be unlikely during head lift for the following reasons:

- Flow velocities during and following the accident were such that significant quantities of pyrophoric material would not be expected to have been transferred to the top of the upper plenum assembly. The results of the "Quick Look" and the "Quick Scan" tests tend to confirm this prediction. Those materials which might have been transferred would have been heavily oxidized.

- The dynamics of the accident were such that:

    1. Conditions to form substantial quantities of hydrides were not present.

    2. The fines that might have been carried to the upper plenum would most likely have been completely oxidized.

    3. Fines that might be on the plenum would be diluted with other fully oxidized and non-pyrophoric materials which would tend to inhibit a pyrophoric reaction.

The above conclusions will be supported by the results of additional testing scheduled for completion prior to the head lift. These tests include a remote visual examination that will be made of the upper surface of the plenum. It is expedted that this examination will confirm that this surface does not contain significant quantities of core debris.

The results of this examination will be forwarded to NRC for their information prior to the head lift.

Types of Pyrophoric Material

Of the various materials that may be in the TMI reactor system, the following may be considered potentially (depending on physical form and environment) pyrophoric (capable of initiating and supporting an exothermic reaction with air, water, on some other oxidant):

1. zirconium metal and alloys rich in zirconium;
2. partially or completely hydrided zirconium;
3. partially oxidized zirconium metal;

All other materials will remain relatively innocuous under the TMI-2 accident conditions (Reference 2).

This is a result of:

- The small quantities present (e.g., fission products)
- The stability of the materials (e.g., oxides)
- The formation of stable materials under the accident conditions (e.g., control rod materials)

Pyrophoricity of Zirconium and Zirconium Hydrides

It has long been known that many metallic powders, including zirconium metal and hydrides, can be hazardous to handle in air. This is due to their relative ease of ignition and the amount and rate of energy released during combustion. There is also substantial experience that moist zirconium powders are particularly hazardous to handle because of their reaction with water once ignition does occur.

A number of theories have been developed to explain the various fires and explosions that have occurred with zirconium. It is known that the pyrophoric tendency is greater with particles of increasing surface-to-volume ratios. That is, powder particles in the micron range appear to be the most likely to ignite. Larger particles, however, such as machine turnings can also ignite and burn. Propagation of the

reaction will be prevented if inactive diluents (fully oxidized material) are present in large quantities. An oxide surface coating also impedes the ignition of a pyrophoric material.

In addition to tne tendency for unoxidized metallic zirconium particles to auto-ignite, it has also been shown that zirconium hydride particles are pyrophoric. The hydride is formed when hydrogen is taken up by zirconium in the presence of steam or moisture which is heavily saturated with hydrogen. The uptake of hydrogen is believed to be impeded by a corrosion film on the surface of the metal. Such a coating existed on the outer surface of the TMI fuel rods as they existed in the reactor before the accident. However, under the conditions during the TMI-2 accident, formation of bulk hydride in localized regions of the core cannot be excluded. For example, in the case of a fuel rod, there is a possibility that when the rods ruptured, hydriding occurred on the ID of the cladding since there may not have been a protective corrosion film. This is in contrast to the outside of the cladding which was protected by an adherent protective corrosion film.

The pyrophoric reaction is intiated by reaction of the material with air or other non-inert gases and, possibly in the case of extremely reactive surfaces at elevated temperatures, with water. Initiation causes a localized high temperature on the surface, which can ignite other material (such as hydrogen), propagate (in the case of fine metal particles), or quench (for large particles, as an oxide film forms). Propagation in air is not fast, requiring seconds to spread over a significant area. The reaction is limited by air accessibility to the metal surface.

The reaction is more rapid if the material is wet because the metal-water reaction can follow initiation. Since liquid water contains much more oxygen per unit volume than does air, the oxygen availability is much greater. The reaction generates hydrogen which immediately burns in the surrounding air if it is adequately available, causing a flame. If air is not available, the hydrogen could accumulate and cause and explosion when air is made available. A second effect is generation

of steam due to heating of the water. The result is that burning wet metal powder causes sparks and pieces of metal to be thrown into the air for short distances. The reaction has been described as like a railroad fuzee. However, in all cases in an open system, the reaction requires seconds to extend over an appreciable distance, and there is not a shock wave or semblance of an explosion.

Clearly, there have been serious explosions involving zirconium scrap. Examination of these suggest that in such cases hydrogen could have accumulated in the container and the exploded. In a closed container the reaction of wet scrap with water slowly produces hydrogen and increases the temperature. It is emphasized that no evidence was found for an explosive reaction of scrap-type material in the form of uncontained and reasonably thin layers in open air. In fact, scrap is commonly made harmless by burning in the open. This uncontained environment is similar to that which would exist during the head lift operations. Hence, even if quantities of pyrophoric materials were on the upper surface of the plenum, they would burn harmlessly and would not explode.

The methods used to safely handle, store and ship zirconium material requires the exercise of extra caution. The National Safety Council data sheet (reference 1) lists the safety considerations for shipping, storing and handling of zirconium particles.

Core Debris on the Upper Plenum Surface

Information available to date indicates it is unlikely that there are significant quantities of core debris on the upper plenum surface. This conclusion is supported by:

- The "Quick Look" video tapes show that the top surface of control rod guide assembly first and second support plates have only the light corrosion film typical of that found in normal plants. This suggests that the plenum cover should also be free of debris. The

"Quick Look" tapes also show the bottom surface of each support plate is uncontaminated, which indicates that there is no reason to believe there is any debris on the inside surface of the reactor vessel head.

- These results of the "Quick Look" examination also appear reasonable, based on the flow conditions predicted to exist at the time of the accident. In particular, the principal means by which fuel debris could reach the plenum top cover and inside surface of the vessel head is by entrainment in fluid flowing upward inside of the control rod guide assemblies. It should be note that this bypass flow was a small fraction of the total flow. With one reactor coolant pump running, such as occurred after the March 29, 1979 accident, the vertical velocity within a guide assembly is estimated to be in the order of 0.3 feet per second in the region between support plates, and about 0.45 feet per second at the elevation of the support plates. This velocity is sufficiently low that most of the entrained fuel debris would settle out before it could reach the top end of a guide assembly. Only small particles, on the order of tens of micron in size or less, could reach the upper end of the guide assembly. Because of their small size and the core conditions that resulted in their formation, any particles that did reach the upper plenum surface are likely to be completely oxidized.

- References 3 and 4 evaluated the results of the Quick Scan experiment discussed in reference 5. These studies concluded that the activity levels measured in the upper plenum were likely a result of cesium deposition. Reference 3 calculations show that it is unlikely that the activity levels are a result of debris on the upper surface of the plenum.

Additional Safety Considerations

As stated earlier, the presence of hydrogen is an important factor in determining the severity of a zirconium reaction. During the period of time that the water level will be lowered prior to head removal, air will be circulated throughout the upper head volume at a rate greater than ____ cfm. This will prevent the accumulation of hydrogen from any source in this region.

Prior to uncovering the upper plenum surface, an under head examination will be performed. This examination is expedited to confirm that debris on the surface of the plenum will not constitute a hazard during the removal of the head.

References

1. National Safety Council Data Sheet 382 "Zirconium Powder" dated 1974.
2. "Pyrophoric Reaction Possibilities in TMI-2 Core", J. D. Watrous, Westinghouse Hanford Co., September 23, 1982.
3. "TAAG Evaluation of TMI-2 Quick Scan Results", MPR Associates, Inc., January 25, 1983.
4. D. Cubicciotti memorandum to J. T. A. Roberts dated January 3, 1983; Subj: A Mechanism to Account for the Levels of Radioactivity in TMI-2 Plenum".
5. GPUN Interoffice Memorandum from V. R. Fricke, December 17, 1982, No. 4550-82-067, "Quick Scan" Experiment.

III. Design Studies for Defueling Equipment, and Core Sampling to Confirm Design Requirements

TAAG comments on the design studies to date for defueling equipment and (2) core sampling that may be required to confirm the design requirements for such equipment are as follows.

a. Design Studies for Defueling Equipment

TAAG reviewed Chapter 5 "Fuel Removal" in the Bechtel report "Plenum and Fuel Removal" dated June 1982. This Bechtel study pertained to means for removing solid portions of fuel from the reactor. The results of the review are contained in Attachment III - One to this TAAG report section. In summary:

- The various concepts evaluated by Bechtel involve completelty new manipulators and associated bridge handling systems, as well as the controls which go along with such a system. This new equipment will likely be very costly and take a number of years to perfect based on past experience with similar equipment. It is recommended that an effort be undertaken to develop a concept which uses existing manipulators and bridges, instead of starting with new, complex, undesigned, and untried equipment.

- The concepts evaluated by Bechtel involve a containment barrier tank that runs from the outer edge of the reactor vessel flange up to the water surface. TAAG recommends that the question of a barrier vis-a-vis no barrier be examined in the development of the water quality management and debris removal systems.

- The Bechtel evaluation discusses the need for tools such as driller and end - mills which must take reaction loads to function properly. These types of tools will require development of special devices to handle the associated loads.

Tools which do not involve reaction loads such as plasma are cutters and electric discharge (EDM) devices, could likely be installed and supported by relativley simple hand tools and the existing bridge handling system. It is recommended that such non-reaction tools be considered for required drilling and cutting operations. It must be recognized, however, that the use of these devices, and the high temperatures which necessarily result, can lead to additional fission product releases. This aspect must be examined.

b. Core Sampling to Confirm Design Requirements

TAAG discussed with EG&G the EG&G plan for a core examination and sampling program. Work to date on developing this program has focused on obtaining information which will be of use to the nuclear industry in assessing core behavior under accident condtions. While this program may well also provide information of use for confirming design requirements for defueling equipment and operations, this has not been the focus of the efforts to date. A draft of the program has been prepared and reviewed by an industry-wide group, the Technical Evaluations Group, and will be established in about May of 1983. TAAG intends to review this plan when it is available to assess its applicability for defueling operations, and to define any additional examinations which may be needed.

TAAG near-term recommendations in their regard are as follows.

- Our understanding of the planned examinations is that they do not involve defining the mechanical characteristics of fuel debris which would have to be known for tool design, e.g., such as hardness, etc. "TMI - like" materials are currently being generated during the source of experimental work on accident condition at ORNL, and at the Power Burnt Facility. Such materials could be employed for defining the physical

characteristics needed for tool design. In particular, based on discussion with ORNL, these materials could readily be made available for appropriate mechanical testing. It is recommended that material testing be considered.

- TAAG investigated the feasibility of using sonic testing techniques for characterizing the region of the core below the loose rubble bed seen in the "Quick Look". The two technique investigated were ultrasonic and sonar. Preliminary results indicate that, based on the model used, traditional ultrasonic methods have a low probability of success but that a device known as parametic sonar holds some promise for probing the loose rubble bed and the region below, possibly being able to determine the subsurface structure (e.g., thickness and density). TAAG intends to investigate further the feasibility of using the sonar device as well as clarifying the utilization of the ultrasonic means of thickness measurement.

- Examination plans do not involve assessing the amount of fuel which may have ~~drifted~~ dropped below the region of the lower core support plate to the region of the lower reactor vessel head. Preliminary evaluation by TAAG indicate that a fairly straightfoward approach may be available to obtain such information. In particular, it would involve lowering a TV camera down the annulus between the core barrel and the reactor vessel flange. Such an insertion could also provide information on the condition of the core support assembly bolted connections. Access holes are available in the core barrel flange and have been used at other OTSG type plants for core support assembly inspections. An early inspection after reactor vessel head removal, but prior to plenum assembly removal, would be made possible by drilling holes in the plenum assembly upper flange. It is recommended that the required plans, procedures, and tooling for such an inspection be developed. TAAG intends to pursue this matter further.

ATTACHMENT III - One

MPR ASSOCIATES, INC.

February 18, 1983

Mr. William A. Hamilton
P. O. Box 613
Liggonier, Pennsylvania 15658

Subject: Review of Chapter 5- "Fuel Removal" in Bechtel's Study of Plenun and Fuel Removal, dated June 1982
M

Dear Mr. Hamilton:

Per your request, we have reviewed Chapter 5- "Fuel Removal" in Bechtel's study of Plenun and Fuel Removal, dated June 1982.
M

Section 5.4.1 of the subject report puts forth four basic concepts for defueling TMI-2. The concepts and our comments are as follows:

(1) The first concept is shown on Figure 5.4-1. This concept basically involves the use of a contamination barrier tank that runs from the outer edge of the reactor vessel flange area up to the water surface. The tank has sealable doors on either end to allow fuel and equipment to be passed in and out. There are two new telescoping-type fuel handling manipulators inside the barrier tank area that operate off a new type of trolley/bridge mechanism. (See Figure 4.5-1, Sheet 2 of 2). Basically this concept will involve developing two completely new types of manipulators and their associate bridge-handling systems as well as all the controls that go along with such a system. In addition, the various operating mechanism and tools for the end of the manipulators will have to be developed. In this regard, based on the earlier TAAG work (see the Third TAAG report, page 23), the contamination barrier tank may cause more problems than it is worth. The proposed new telescoping-type manipulators and their bridges are going to be a major development undertaking that will be very costly and take a number of years to perfect based upon past experience with similar manipulators. Also, the development of the grapple devices and the various tools for the lower ends of these manipulators will also be a major undertaking, particularly for a "one-time" job such as TMI-2.

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Also, a duplicate set of equipment will also have to be provided for the spent fuel pool outside of the containment building so the training of people can be done away from high radiation areas. This concept will take many millions of dollars to develop and will involve two to three years of effort before it works satisfactorily and is reliable enough for use in the TMI-2 environment. An item of particular concern with newly developed complex equipment is that it has not been really debugged by actual field experience before it is used in TMI-2.

2. The second and third concepts are basically shown on Figure 5.4-8. These concepts use a different type of containment barrier. Basically it is a cylinder type platform mounted on the reactor vessel flange which has a rotating drive mechanism. This rotating platform has a traveling table with a hole through which the manipulator tool works. By rotating the platform and moving the table, the small access hole in the table can be positioned over any area of the core. The manipulator works down in the core region through the relatively small hole of the moveable table. In these concepts a new telescoping manipulator is also provided and should present the same type of problems as discussed in concept #1 above. Another variation of these concepts involves the use of a manual tools to be used like a manipulator, through the same rotating cylinder concept. Here again, I believe the contamination barrier in this concept will unnecessarily complicate defueling operations and greatly limit the ability to de-fuel this plant. Likewise, the development of a new telescoping manipulator for this one-time application will have the same problems as discussed previously in concept 1 above.

3. Concept 4 can be seen on Figure 5.4-16 (sheets 1 and 2). This concept basically has a telescoping umbrella for a contamination barrier that mounts to the O.D. of the reactor vessel area and to the upper portion of the telescoping mast of a manipulator. Under this umbrella there are also several other remote articulated manipulator-type tools. There is a magnetic trap door in one side of this telescoping umbrella which allows a fuel cannister bucket to be moved in and out of the

umbrella area. Having remote manipulator tools operating under a flexible umbrella contamination barrier will probably create a major problem for maintenance on such tools as well as eliminate any kind of visual access for working with the tools except by remote TV. These articulated underwater tools will be "state of the art" type devices and be an onerous and expensive new development effort for a "one-time" operation such as TMI-2.

It would appear to us, based on what we've learned from TMI-2 since the subject study was prepared, that another effort should be made to develop and prepare a more simple and direct concept for removing the solid parts of the fuel debris. Specifically, we would suggest that an effort be undertaken to develop a concept which uses the existing manipulators and bridges, instead of starting with new, complex, undesigned and untried equipment. It may be possible to develop different types of limited purpose tools which could be attached to the ends of the existing manipulator to remove core debris. Also,it would be prudent to try to look at tools that do not involve reaction loads for cutting and drilling operations to minimize the need to develop special devices to handle loads caused when using saw and drill type of tools. The other advantage of building tools around the existing manipulator is that one of these machines also exists now in the spent fuel building where it could be used to train operators. Accordingly, we recommend that another attempt be made of developing a simpler defueling concept using the existing fuel handling equipment.

In addition to the above general comments, the enclosure to this letter contains detailed comments on fuel removal criteria.

If you have any questions regarding this matter, please do not hesitate to call.

Sincerely yours,

Noman M. Cole

Noman M. Cole

cc: Ed Sise, Newport News
Ed Wagner, Burns & Roe
Harold Burton, EGG

Enclosure

Detailed Comments on Criteria
for Fuel Removal at TMI-2

General Criteria (page 5.2.1)

1. The requirements of Paragraph 5.2.2 may result in an expensive set of tools to cover "any combination" of defined core conditions. Are you sure you don't want to be more selective and obtain just basic tools initially!

2. Do not believe a physical contamination barrier around reactor vessel opening should be required as per Paragraph 5.2.6. Based on earlier TAAG work, not clear that physical contamination barrier is necessary or should be used. (See fourth paragraph on page 23 of the third TAAG report dated August 31,1982).

3. Paragraph 5.2.7: Not sure we should be required all fuel canister loading operations to within envelope of reactor vessel and physical contamination barrier.

4. Paragraph 5.2.8 seems to overstate the requirement for keeping debris from falling through openings in CSA grid plate. (Words in the last paragraph on page 5.3-5 seem more realistic and meaningful.)

5. Paragraph 5.2.12 requires the development of a "Special Nuclear Material Accountability Program" for removal of the TMI-2 Core. It is my impression that the NRC has agreed that this is not required for the special case of the TMI-2 core removal.

6. Section 5.3.2.1 indicates that the vacuum head should be controlled by remote positioners which provides the operator with force feedback information. Based on experience with such equipment, it is not clear that the additional complexity of a force feedback system is warranted or even needed.

7. Section 5.3.2.3 discusses the need for drills, end-mills, core drills, saws, etc. for disassembling the TMI-2 core. All of these remote tools require that they take reaction loads to function properly. Taking of such reaction loads can add considerably to the complexity of these tools, particularly if the tool has to sit on uncontrolled surfaces. Tools such as plasma

ar~~e~~c, ELOX, MDM, etc. can make rough cuts of material and do not involve reacting loads; therefore, they can be roughly positioned by relatively simple handtools which are assisted by a crane that takes the bulk of the weight.

8. Section 5.3.2.5, (page 5.3-5, first paragraph) states "In all cases, the bottom lifting device must have the ability to raise the removed fuel assembly into a bottom loaded sleeve that will eventually discharge its contents into a fuel canister. As the fuel assembly is raised into the sleeve, it passes through a roller system that inwardly deforms broken or bowed fuel . . . ." It is not clear that requiring "rollers" for such devices are warranted and that other device may not be better and more flexible for this case. Why not say what you want accomplished and not "How To"?

9. Section 5.3.2.5, (page 5.3.5, fourth paragraph). You may not be able to lift a fuel assembly with an in-core instrument installed unless you cut the in-core instrument underneath the bottom of the fuel assembly's end-fitting before you lift it. Accordingly, it would be prudent to have such a cutting device that could be made available in a reasonable time period.

IV. Resolution of TAAG comments for Plenum Removal and Development of Prerequisites

TAAG comments on the B&W plan for Plenum Removal have been resolved; however the B&W planning study assumed for scoping purposes that 10% of the Kr-85 remaining in the fuel could be released instantaneously during plenum removal operation. This corresponds to a Kr-85 release of about 1,900 curies. TAAG indicated in the fourth TAAG report that such a release may be unrealistically high, and recommended that a mechanistic evaluation be performed to define more realistically the amount of gas release which could occur.

ORNL has perfomred such a mechanistic evaluation of fission gas release. In particular, the evaluation considered the inventory of Kr-85 present in the outer row of fuel assemblies which might stll be intact because they were cooled by radiant heat transfer. The evaluation also considered the fraction of Kr-85 in these fuel assemblies which could have migrated from the oxide fuel to the fuel rod gas plenum. The evaluation indicates that the calculated amount of Kr-85 available in fuel rod plenum for release is 30 curies, i.e. 1.5% of the release assumed for scoping studies in the B&W report. The greatest uncertainty in calculated value of 30 curies is the number of surviving fuel rods that are leak tight. An examination of the parameter involved in the ORNL evaluation show that the release is most likely between 0 and 100 curies of Kr-85.

In summary, the ORNL evaluation indicates that the amount of Kr-85 available for release from intact fuel rods in calculated to be 30 curies, and could be as low as zero or are high as 100 curies. This is a small fraction of the release assumed in the B&W evaluation, and considered not to represent a significant problem.

Prerequisites for Plenum Removal

A. Introduction

TAAG has continued to review the prerequisites for the various stages of core defueling operations. This section of the report presents suggested prerequisites for the plenum left. Emphasis was placed on the environmental conditions which must exist in the flange region and preparations which should be made in advance for incidents or equipment difficulties.

This Section is divided into the following subsections:

- General Prerequisites and Contingency Planning
- Plenum Removal Operations

Each recommended prerequisite will be discussed individually in the appropriate subsection. The order of the discussions of recommended prerequisites does not relate to their relative importance.

It is assumed that the plenum lift procedures will, where possible, utilize the techniques used in the past, modified as necessary to account for the effects of the accident. Hence, the prerequisites discussed herein address those aspects of the lift that may change due to the unique Unit 2 conditions and do not address the prerequisites for a conventional plenum removal, many of which it is assumed will be used for this lift.

B. General Prerequisites and Contingency Planning

1. A safety evaluation of the plenum lift and associated activities should be prepared under GPU's direction for submittal to NRC.

   If satisfactory to NRC, the report should use the same format at the "APSR" Testing", "Quick Look", and Head Lift Safety Evaluations.

   Maximum use should be made of the APSR testing, Quick Look and Head Lift Safety Evaluations. The prerequisites for the Head Lift Safety Evaluation suggested in the third TAAG Report nave applicability to the plenum removal. A majority of these prerequisites are applicable to the plenum removal operations. It is considered that the actions required to assure safety for the head lift operations will also be required for the plenum removal.

2. The safety of the lifting operation itself from a load drop consideration should be discussed in the Safety Evaluation. It is anticipated that this discussion would show that the 1979 accident did not result in any changes to the safety considerations relating to a load drop other than those actions required to repair and requalify the crane.

3. The possibility of the existence of pyrophoric materials from the core and the safety implications thereof should be discussed in the Safety Evaluations.

4. Potential radiation exposure of operating personnel and the actions taken to minimize such exposure should be addressed in the Safety Evaluations.

5. Any discharges of radioactivity should be identified and quantified in the Safety Evaluations.

6. The plenum may remain in its post lift configuration for an indefinite period of time. The general safety considerations of this static condition should be evaluated. This evaluation should include: 1) the possible effects of the 1979 accident and post accident condition on the plant's long term safety; 2) the effects of the possible environment events (flood, earthquake, etc.) on plant safety. It is anticipated that it can be shown that many of the pre-accident evaluations of these events apply to the post accident condition.

7. The environmental conditions that will exist during the plenum removal operations - radiation fields, special clothing, respirators, special contamination barriers and radiation shields, revised crane controls, communications equipment, etc. - should be considered in revising the existing plenum lift procedures and supporting training program.

8. The training, maintenance and QA programs to support the plenum lift operation should consider the lifting and handling problems that have occurred in the past in industry. GPU should review the problems others have had and assure they have learned from these events.

9. Materials used inside the containment including polyethlene, herculite, etc., should be fire retardant.

10. A detailed training and QA program should be used to support the plenum assembly lift operation.

11. Special tests, if any, during pre-plenum lift or plenum lift operations should be limited to those tests required to support the plenum lift. Tests to obtain accident consequence data should be limited to those tests that will not delay or interfere with the removal of the plenum.

12. The general work area should be under TV surveillance and the operations should be recorded.

13. The procedures and equipment to be used for the plenum lift should provide for contamination control during the lift and transfer.

14. During the plenum lift, transfer and subsequent securing of the pressure vessel after the lift, no other testing or changes should be made to the reactor primary plant or supporting auxiliaries. The containment should be intact and closed.

15. Emergency equipment and procedures should be readily available to respond to the loss of indexing fixture shield water.

16. The source range neutron instrument monitoring limits, established for the APSR test, should be reissued for the plenum lift.

17. Decay heat removal should be reviewed including an evaluation of the need for a backup heat removal path and the methods that will be used to monitor the heatup, the basis for temperature limits, the time available to take corrective action if required.

18. Discuss potential radiation levels with all operating and support personnel.

19. Identify and evaluate, as appropriate, any new short or long term water chemistry considerations.

20. Publish a safety report to cover safe removal of the plenum under essentially normal removal conditions and include an evaluation of contingency removal processes.

21. In order for the continued reactor shutdown, the following conditions should be identified and maintained:

(a) The use of commercially available direct alarming boron monitoring equipment is in place, and is operational to monitor the boron concentration in the primary coolant of the R.V.

(b) Fire fighting procedures have been changed to preclude the use of unborated water in the canal or overhead areas.

(c) Procedures and material are readily available to assure boron control in the event make-up water is added during a loss of coolant accident.

(d) Where possible, the process of positively isolating dilution sources by use of the disconnects and flanges have been extended from those used for the "Quick Look".

22. Detailed step-by-step procedures and emergency equipment of the following operations prepared for closure head removal, should be reviewed for applicability to the plenum removal operations and are available to cover operations.

(a) Crane testing
(b) Primary and secondary water level control
(c) Boron chemistry control
(d) Water chemistry control
(e) Hydrogen monitoring and control
(f) Radiation monitoring
(g) Long-term plenum storage
(h) Long-term layup of the plant after plenum removal

23. Contingency plans should be developed for the plenum lift that recognize the unique conditions that may exist during lifting and transfer operations.

24. The RV cover plate used after closure head removal operations should be readily available for use after plenum removal operations.

25. GPU should review the existing Unit 2 technical specifications to determine if any changes are required. Any changes should be identified promptly. Actions should be initiated by GPU at that time to make any changes required.

26. Radiations levels in the areas where work is to be performed should be controlled to less than 50mr/hr. These levels should be achieved by controlling water activity, through the use of shielding, or some combination of these two factors.

27. The airborne activity within the work area should be monitored by alarming Continuous Air Monitors (CAM's).

28. An enclosed environment with a clean air source should be provided to minimize the need for respirators and simplify contamination control.

29. In order to minimize the possibility of uncontrolled boron dilution, all sources of unborated (below 3700 ppm) water above the canal region should be drained and disconnected. A water exclusion area should be established.

30. In addition to CAM's, gamma alarms and monitors should be located in the working environment during the plenum assembly lift.

31. The plenum assembly removal and transfer procedure should be based upon the use of standard procedures modified as necessary to reflect specific post-1979 accident conditions. The procedures should not attempt to provide contingency steps in the event a significant operation does not go as planned.

The procedure should require that operations be terminated and the plant placed in a safe condition (defined) in the event difficulties are encountered either with the execution of the procedures or with the support equipment. In the event of difficulties, alternate procedures can then be established for the specific difficulties encountered.

32. It is recognized that it is likely that minor changes to the procedures may be desired as the plenum assembly lift operation progresses. To facilitate these changes and at the same time assure proper review, a test group having representatives from the Unit 2 operations, engineering and safety organizations should be established. This group will have the authority to determine that a change is consistent with the original procedure intent, and does not introduce a new or change an existing safety consideration. Under these conditions, and subject to NRC approval, the group (under uranimous agreement) may approve on the spot procedure changes.

B. Plenum Removal Operations

The following conditions exist as pre-requisites to plenum removal:

1. The reactor vessel head is removed and parked on the storage stand at the 347 elevation.

2. The canal seal plate is in place and adequate for canal flood as necessary to support plenum removal activities.

3. The polar crane and rigging has been refurbished and qualified to 170 tons.

4. The canal fill/drain/cleanup systems are operable.

5. The Internal Indexing Fixture work platform will not interfere with operations performed through the 69 control rod guide assemblies and the ID/OD of the plenum cylinder.

6. The Internal Indexing Fixture work platform will support 2" lead shielding and 6 men with tooling.

7. The Reactor Coolant System (RCS) is cleaned to as low as practicable ($\leq$ 0.1 μ ci/ml).

8. Required building services (including service air, electric power, communications, etc.) are available.

9. Canal deep end preparations including transfer equipment modifications have been completed to support canal fill.

10. Inspection has been performed to determine the actual conditions regarding debris deposition and cleaning required on the plenum as well as to identify damage or distortion to the plenum or in the vicinity of LOCA restraints.

11. Video support equipment is available and ready for use for confirmation and trouble-shooting.

12. The keyways in the Internal Indexing Fixture have been removed or modified to support the plenum removal.

13. A safety evaluation of the plenum assembly removal lift and associated activities has been prepared for submittal to NRC. If satisfactory to NRC, this report should use the same format as the "APSR Testing," Quick Look," and "Head Lift" safety evaluations.

14. Maximum use has been made of the "APSR Testing", "Quick Look" and "Head Lift" safety evaluations.

15. A revised discussion of boron dilution, reflecting plenum lift conditions has been included in the safety evaluation report.

16. A revised discussion of water control reflecting the plenum assembly removal conditions has been included in the safety evaluation report.

17. Criticality considerations during plenum removal activities are enveloped by existing or planned criticality evaluations.

18. The Procedure includes the leveling/alignment process to assure successful withdrawal, specifying the cleaning and inspection methods and identifies needed mockup for proof-of-principal testing of tools and techniques as well as training for operations.

19. The lift lugs have been load tested prior to plenum removal.

20. Prior to plenum lift a visual inspection has been made to determine the need to clean the plenum during or prior to the lift. If it is determined they will be required procedures and equipment must be available prior the plenum removal.

V. Lead Screw Samples

Chemical and metallographic analysis of at least one of the lead screws that had been extracted from the reactor vessel could provide valuable insights in four district areas of pertinence to recovery operations. First, identification of the nature of the strongly adherent cesium on the lead screws, by metallorgraphic examination, could yield information of particular utilitiy to the eventual selection of methods for the decontaminating the components underneath the reactor vessel head and other surfaces in the upper plenum region. Second, knowledge of the manner in which the adherent radiocesium is distributed along the length of the lead screw can help in providing a more realistic estimate of radiation fields and dose rates which would be encountered during head and plenum lifts. In addition, a limited amount of information can also be obtained regarding the effect of base material on cesium retention, because of the two differing types of alloy of which the lead screw is comprised. Third, chemical analysis of the debris which had deposited on lead screw surfaces yield further information on the physico-chemical characteristics of material which had been transported away from the core, through the primary system, during the accident. Lastly, chemical analysis of this debris can also assist in deliberations currently underway to assess the likelihood of encountering a problem with pyrophoric materials during head lift.

For the reasons enumerated, TAAG recommends the following actions:

(1) Accurate gamma scans be made of all three lead screws which had been removed from the reactor vessel. These scans could subsequently be used to select regions for detailed metallorgraphic examination.

(2) Metallographic examinations be made of selected lead screw segments in an effort to identify the nature of the adherent cesium. Particular attention should be given to resolution of the two differing viewpoints presented to TAAG, whether the cesium is associated with an oxide layer or is contained within (or by) a boride coating.

(3) Complete chemical analysis of debris collected from the lead screw should be made, and a determination of the size range of the particles comprising the debris should be attempted.

(4) If chemical analysis of the loose debris indicate the presence of unoxidized zirconium, tests for phyrophoricity should be conducted. TAAG has provided guidance concerning the examinations that should be conducted; these are appended as Attachment V-One. Moreover, some of the chemical and metallographic analyses on leadscrew segments have been completed. Although results of these analyses have been presented to TAAG during the latter part of the period covered by this report, the data are preliminary and await more detailed examination before definitive conclusions can be drawn. However, based on the preliminary results it appears that:

- Cesium on the leadscrew surfaces is associated with an oxide film.

- Debris removal from a leadscrew piece contains no unoxidized zirconium and, accordingly, is not pyrochloric.

ATTACHMENT V-ONE

Mr. William H. Hamilton
Consulting Engineer
Post Office Box 613
Ligonier, PA 15658

December 29, 1982

Mr. Richard P. Allen
Corrosion Research and Engineering Section
Battelle Pacific Northwest Laboratory
Richland, Washington 99352

Subject: Three Mile Island Unit 2 Leadscrew Testing

Dear Mr. Allen:

At the request of Mr. B. K. Kanga, GPU Nuclear, the TMI-2 Technical Assistance and Advisory Group (TAAG) have defined test requirements for the section of control rod drive leadscrew which was recently sent to PNL. In particular, TAAG was requested to define tests to determine whether debris present on the surface of the leadscrew piece is pyrophoric in nature.

In addition, measurements at the TMI site indicate leadscrew radiation levels are high due to surface contamination with cesium, and remain high after flushing with clean water. Radiation levels of the reactor internal structures in the region below the reactor vessel head are also high, and may have the same type of cesium contamination as the leadscrews. Accordingly, tests are also needed to characterize the leadscrew contamination, e.g. to determine whether it is present (1) in loose debris on the leadscrew surface, (2) within the adherant oxide film of the leadscrew, or (3) in the base metal.

The recommended testing is defined in the attachment to this letter. The enclosure defines the information to be obtained from the tests, but does not define detailed specific testing techniques which you would employ. I would appreciate the opportunity for TAAG review of specific testing plans once they are developed. In this regard, I have asked

Dr. E. A. Evans of the Westinghouse Hanford Engineering Development Laboratory, and a member of TAAG, to review your test plans, and answer any questions you may have with regard to the enclosed test description.

Sincerely,

David G. Thomson

for/ W. H. Hamilton, Chairman
TAAG

cc: E. A. Evans, HEDL
B. K. Kanga, GPUN
A. Roberts, EPRI

December 29, 1982

TMI-2 LEADSCREW PIECE TESTING AT BATTELLE PACIFIC NORTHWEST LABORATORY

A. Purpose

The purpose of the testing is to:

1. Determine whether the debris on the surfaces of the leadscrew piece contains small size particles of unoxidized zircaloy, and in sufficient concentration, to cause a concern with pyrophoricity.

2. Characterize the surface radiation level of the leadscrew piece, i.e., identify the major isotopes contributing to the radiation level, whether the radiation source is localized in (1) loose surface debris (2) adherant oxide film or (3) base metal, and the chemical form of deposited cesium.

B. Pyrophoric Testing

1. Collect loose debris from the leadscrew surface, and its plastic wrapping material, for use in the testing. The intent is to collect a sample which contains primarily loose debris which might have settled out on the leadscrew surfaces, rather than a sample containing corrosion product film from the leadscrew surface. Accordingly, mild contamination removal techniques should be employed, e.g. brush the surface with a soft brush.

2. Determine the weight percent of the various elements contained in the sample. These should include zirconium, uranium, iron, nickel, and chromium. Possible techniques include (1) atomic absorption of dissolved debris or (2) X-ray fluorescence.

3. Determine the extent of oxidation of the zirconium contained in the sample, and its crystalline structure. Possible techniques include Auger, ESCA, MOLE (Laser excited Ramann spectroscopy) and metallographic (if larger size pieces are available.)

4. Determine the size distribution of the particles contained in the sample. The size ranges of any zirconium particles present should be determined specifically, if possible. The weight fraction in each of the following size ranges should be determined; less than 10 micron, 10-20 micron, 20-50 micron, 50-100 micron, greater than 100 micron. A possible technique is SEM.

C. <u>Surface Contamination Characterization</u>

1. Progressively remove the leadscrew surface deposits by (1) vigorous brushing to remove the loose oxide film and any remaining surface debris (2) electrolytic descaling or severe chemical treatment to remove the adherant corrosion product film.

2. Each decontamination step should be accompanied by leadscrew dose rate measurement at contact with

the surface, plus a gamma scan of the removed contamination and leadscrew surface to determine the significant isotopes and curies of each present.

3. Metallographic examination of the leadscrew surface before and after descaling to identify the structure of the oxide film, and the extent of its removal. In addition, the descaled surface should be examined for evidence of intergranular attack, and for evidence of cesium and iodine associated with chromium in the attack region.

   Localized examinations to determine the cesium and iodine content of the oxide film, and the chemical form of these elements, should also be attempted. A possible technique is Auger analysis. It would also be desirable to sputter the adherant oxide film surface to a depth of several hundred angstroms to determine if contained cesium is highly localized at the surface.

VI. Reactor Building Decontamination and Characterization

A. Plans for cleaning 282' level

GPUN/Bechtel completed a Planning Study and a Plan in January 1983 concerning Sludge Removal from the Reactor Building Basement. The study considered four principal alternatives:

a) Collect sludge, process outside reactor building
b) Collect sludge, process in reactor building
c) Collect sludge, shield and store in reactor building
d) Flush basement to solubilize cesium, remove cesium in SDS

Alternative a) was selected as the preference and was the basis for the plan. In alternative a) a remote, automated collection system was planned. Sludge slurry would be pumped from the reactor building to a sludge processing system located in the auxiliary building. The processing system would include separation, solidification and packaging capability.

TAAG had previously reviewed and commented on the Reactor Building Decontamination and Characterization including the 282' level. Recommendations were reported in the Fourth TAAG Report dated December 1, 1982. In summary the recommendations applicable to the 282' level were:

1. Evaluate effects of leaching of cesium from the basement during fill/pumpout cycles. Use as basis for planning the sludge removal program.

2. Evaluate the use of about 18 inches of shield water in the basement to reduce dose rates above EL 305 floor.

3. Determine cesium penetration into concrete by such methods as taking core samples.

TAAG was requested by B. K. Kanga's letter of February 9, 1981 to provide assistance regarding the plan for cleaning the 282 level. TAAG reviewed the GPUN/Bechtel planning study and plan and presente recommendations to GPUN/Bechtel in a meeting on February 24, 1981. The background for the TAAG recommendations is the evaluation of leaching effects reported in Burns and Roe Technical Memo 3680-7. A copy of revison of this Technical Memo dated April 4, 1983, is Attachment V-A-One.

In Attachment V-A-One the potential sources of cesium and stronitum appearing in the water being pumped out of the reactor building basement were identified. Estimates were made of the quantities of Cs-137 that could have existed in there sources at the time the basement was first pumped-down to EL 281-6. An isotopic balance was then made of the Cs-137 concentration data for all pumpouts from that time to present.

Attachment V-A-1 concludes that Cs-137 and Sr-90 are continuing to appear from a leachable source. The source is postulated to be the sludge and concrete that was flooded for about 3 years with water containing about 160 μ Ci/ml of Cs-137.

Based upon the conclusion of Attachment V-A-One TAAG recommended:

1) Flush and remove cesium from EL 282-6 thru SDS as long as significant cesium removal continues. A target of 1 Ci/day was suggested be attained before terminating this cesium removal process. This flushing process would be the most Man-Rem effective alternative for removing the cesium from the sludge and other leachable sources. It was also noted that the sludge samples taken to date indicate that the 9 percent solids fractions is a gel-like consistency which may also breakdown and be removed with further flushing.

2) Make a determination of the actual gamma dose rate reduction from about 18 inches of shield waer on EL 282-6. If signficant dose rate reduction occurs, use shield water, possible throughout the duration of the defueling if needed. The counterveiling effect on gamma dose rates of cesium leaching into the shield water would cause occasional processing of the shield water to be necessary. A target of 1 μ Ci/ml Cs-137 was suggested to minimize the dose contribution from cesium in the shield water.

3) Base the ultimate physical removal of sludge on the data acquired from leaching and shield water evaluations and additional sludge samples. As the sludge is diplexed of cesium and further breaks-down, the need for a separate sludge removal process may disappear. The remaining sludge may be removed in combination with removal of contaminated surface concrete.

ATTACHMENT VI-A-One
TO FIFTH TAAG REPORT

B&R

TECHNICAL MEMORANDUM

DATE 4/4/83

NUMBER

TM3680-7 Rev. 2

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TO E. J. WAGNER

FROM P. MASSAIA/C. W. HESS

P Massaia/CW Hess

SUBJECT W.O. 3770-01
TECHNICAL ASSISTANCE AND ADVISORY GROUP
TMI-2 RECOVERY
LEACH RATE OF ACTIVITY INTO REACTOR BUILDING BASEMENT WATER

KEYWORDS TMI-2, REACTORS, DECONTAMINATION, LEACHING, CONCRETE

REFERENCE:

(1) B&R TECHNICAL MEMORANDUM #3680-3, REV. 2, "CONTAMINATION INTRUSION INTO CONCRETE," 11/17/82

(2) B&R CALCULATION #3680-15-22, "ESTIMATED VOLUME IN REACTOR BUILDING (RB) SUMP, INSTRUMENT CHASE AND UNDERFLOOR DRAINS," 3/4/83

(3) B&R CALCULATION #3680-15-23, "APPEARANCE AND REMOVAL OF ACTIVITY IN THE RB BASEMENT WATER," 3/83

(4) ORNL TM 7448, 7/80

(5) PRIVATE COMMUNICATION, C. HITZ OF GPU TO S. ROSEN OF B&R, TRANSFER OF ORNL AND GPU RADIOCHEMICAL ANALYSIS OF SUMP WATER

(6) ORNL INTER-OFFICE MEMO, J. A. CARTER TO D. O. CAMPBELL, "TMI-II SLUDGE DATA," 10/27/82

(7) PRIVATE COMMUNICATION, T. LOOKABILL OF NSU TO S. ROSEN OF B&R

(8) BECHTEL NATIONAL, INC., "GROSS DECONTAMINATION EXPERIMENT REPORT (DRAFT)", TMI-II RECOVERY PROJECT, 6/82.

(9) B&R CALCULATION #3680-15-14, "EVALUATION OF EG&G TLD TREES", 11/9/82

(10) B&R CALCULATION #3680-15-16, "CURIES IN RB BASEMENT", 11/15/82

ATTACHMENTS:

(a) TABLE 1 - PUMPOUT DATA FOR PUMPOUTS 16 TO 22
(b) TABLE 2 - ESTIMATED RESERVOIRS OF CS-137 ON EL 282'-6"
(c) FIGURE 1 - EL 282'-6" CESIUM DIFFUSION MODEL 2
(d) FIGURE 2 - EL 282'-6" CESIUM DILUTION MODEL
(e) FIGURE 3 - ELEVATOR SHAFT AND CONCRETE LOADING PLATFORM

SUMMARY

Purpose

High gamma dose rates persist in the basement of the TMI-2 reactor building despite the removal of the contaminated water from the basement. The source of these dose rates is suspected to be mainly the result of cesium (Cs-137) contamination of the sludge 2 on the El. 282'-6" floor and of the long-flooded concrete. As discussed in Reference (1), it may be possible to leach Cs-137 out of these areas by reflooding the basement. This evaluation attempts to determine if radioactive cesium is leaching out of the basement based upon the appearance of Cs-137 in the basement water as it is diluted and pumped out of the reactor building. The evaluation is intended to assist in radiological engineering of the defueling and reactor building cleanup.

Conclusions

1. Reactor building basement water Cs-137 and Sr-90 isotopic balances considering all known volumes of water in communication with basement water indicate that cesium and strontium are leaching into the basement water. Approximately 7,000 curies (Ci) of Cs-137 have appeared from a leachable reservoir and been removed since the initial draindown in February 1982. The rate of appearance con- 2 tinues to provide significant decontamination of El. 282'-6" by removing approximately 25 Ci per day from the reservoir. The leachable reservoirs are presumed to be the sludge on the flooded concrete and the concrete itself.

2. Evaluation of the sludge sample taken on 6/24/82 indicates that the sludge layer on the El. 282'-6" floor contained a reservoir of 3672 Ci of Cs-137 and 1013 Ci of Sr-90.(3) Since 3223 Ci of Cs-137 have appeared and have been removed since the sludge sample was taken, it is probable that not

all of this Cs-137 is from the sludge layer. If all the cesium was from the sludge, it would now be depleted of cesium. The continuing appearance of Cs-137 would then need to be from another reservoir such as the concrete.

3. The available data for Cs-137 activity in the basement water indicates a decrease in activity with successive pumpouts.[5] 2
Figure 1 shows the plot of Cs-137 concentration vs time. A line fit was performed on these data points, which assumed an exponential decrease of concentration with time. A 114-day time constant was obtained [3], which indicates that a 1 uCi/ml concentration level of Cs-137 can be attained by August 1983. This assumes that pumpouts of water will continue at the same rate through August.

RECOMMENDATIONS

1. Washing of building surfaces and consequent dilution of the basement water is effective in reducing activity and should continue as long as it remains effective. Activity concentrations in the basement water and sludge should continue to be monitored as well as dose rates using TLD trees.

   A target of 1 uCi/ml of Cs-137 should be used (based on consideration of using this water as shielding).

2. The major potential reservoirs (concrete fill slab and sludge) are directly affected by the water level being maintained in the basement. Two other locations associated with the enclosed stair and elevator shaft could account for almost 5,500 additional Ci. These are not in communication with basement water and hence are not being treated directly by maintaining water in the basement. Specific action, such as effectively flushing the pit to substantially dilute the water in the pit and further washing down the outside surfaces and the surfaces inside the hollow blocks of the elevator shaft, is recommended. 2

3. At a later date, after continued flushing of the basement, the source of contamination appearing in the basement water should be confirmed. It is noted that this discussion has concentrated on Cs-137 because of its contribution to dose rates. Strontium is much less soluble than cesium and does not have the same potential for removal by leaching. Also it tends to have greater ion exchange propensity with typical concretes. Procedures to enhance the leaching of the remaining Cs-137 and Sr-90 activity out of the sludge/concrete should be explored, and experiments to develop enhanced leaching processes should be designed.

4. The nuclide concentrations in the basement water should be determined during subsequent pumpouts to further define the activity transport from the reservoir and to monitor 2 progress toward a concentration of 1 uCi/ml Cs-137 and an appearance rate of 1 Ci/day Cs-137.

DISCUSSION

Background:

In July of 1980 about 700,000 gallons of water were in the TMI-2 reactor building basement with a Cs-137 concentration of 160 uCi/ml[4]. From July of 1980 until September of 1981 no water was removed from the reactor building basement, and water was added from continuing primary coolant system leakage to bring the total volume in the basement of about 790,000 gallons with a 2 measured Cs-137 activity of 138 uCi/ml[4]. This Cs-137 concentration is consistent considering that approximately 9,500 Ci of Cs-137 were additionally introduced into the basement from the reactor primary coolant system leakage over that same time interval. Between September of 1981 and February of 1982 about 601,000 gallons of water in 16 stages (batches) were pumped out of the basement with some continued input to the basement from reactor system leakage of about 0.1 GPM. During this period the Cs-137 activity remained fairly constant at about 130 uCi/ml[5] indicating an equilibrium situation. During this same period the Sr-90 activity also remained relatively constant at about 5.4 uCi/ml[5] indicating an equilibrium situation for this isotope.

Dilution And Curie Balance to April 1982:

During March and April of 1982 about 14,000 gallons of water were 2 added to the reactor building basement diluting the Cs-137 activity to 118 uCi/ml. This water was principally from decontamination activities and RCS leakage. An activity balance comparing total curies of Cs-137 before and after the dilution was performed on the basement water, including the water volumes in the basement sump, instrument chase and underfloor drains.[3] 2 It indicates that about 2,886 Ci of Cs-137 appeared in the basement water, i.e., the diluted concentration should have been 103 uCi/ml of Cs-137 rather than the measured value of 118 uCi/ml. The additional Cs-137 could have appeared from the sludge on the floor of El. 282'-6", from within the concrete or from a volume of water not in communication with the basement water. This same phenomena occurred with a Sr-90 activity concentration in the basement water as it actually increased slightly to a measured value of 5.82 uCi/ml after the dilution. (See Table 1.)

Sludge Sample Evaluation:

On June 24, 1982, sludge samples were taken from the floor of El. 282'-6" and sent to ORNL for analysis. The results of this analysis indicated the following[6]:

- For both Cs-137 and Cs-134, the supernate contained 74% of the total sample activity for those isotopes and an additional 16% of the activity was easily washed from 2
the solid portion leaving about 9% in the solids portion as insoluble.

- For Sr-90 the supernate contained 12% of the Sr-90 activity and the insoluble portion in the solids was 91% of the total.

Based on the ORNL sludge analysis, the observation that the sludge layer was about 1/2 to 3/4"[7] deep at the time of the sampling, and assuming that the sludge layer was uniform over the entire floor area, the total curies of insoluble Cs-137 and Sr-90 in the sludge layer are 312 and 923 respectively.[3] These are a maximum values based on a 3/4" layer and considering the sludge to be of the same composition as that in the sample sent to ORNL for analysis. In addition to the insoluble Cs-137 a leachable Cs-137 component in the sludge contains 600 Ci and 2760 Ci is dissolved in the supernate for a total of about 3672 Ci of Cs-137 in the sludge layer. Similarly there is a total of 1013 Ci Sr-90 2
in the sludge layer. The supernate portion of the sample contained 150 uCi/ml of Cs-137 whereas a sample of the basement water taken just a few days earlier on June 15, 1982, during pumpout No. 18 showed only 87 uCi/ml[5]. This indicates that the basement water was not in equilibrium with the sludge layer at that time. The same is true for the Sr-90 where the sludge sample supernate contained 6.93 uCi/ml and the basement water 5.4 uCi/ml.[5]

Dilution And Curie Balance From April 1982 to February 1983:

Between the end of May 1982 and the middle of February 1983 an additional 75,050 gallons of water were added to the basement 2
during which time pumpout Nos. 18, 19, 20, 21 and 22 were performed. The measured Cs-137 and Sr-90 activities of the basement water during each pumpout are shown in Table 1. A curie balance for Cs-137 dissolved in the basement water between the 5/9/82 pumpout (No. 17) and each of the subsequent pumpouts indicates that about 4000 Ci have appeared in solution[3]. Although the 2
Cs-137 concentration decreased from 87 to 25.3 uCi/ml between pumpouts 18 and 19 the Sr-90 concentration actually increased

from 5.3 uCi/ml to 5.8 uCi/ml indicating a large source reservoir of Sr-90. Note the sludge layer was estimated to contain 1013 curies of Sr-90 on 6/24/82. A curie balance for Sr-90 between pumpouts 17 and 22 indicates that about 2000 curies have appeared in solution[3]. 2

The above curie balances on the reactor building basement assume that all of the decon water added to the basement does not contain activity. An estimate of the maximum Cs-137 activity in all of the decontamination water added to the basement since June 1982, based on the concentrations in the decontamination water used in the Bechtel Experiment, is 1 Ci.[8] This is small compared with the appearance of over 5000 Ci of Cs-137 in the basement water over the same time period and can be neglected by comparison.

Sources Of Cs-137 and Sr-90:

From the above data and analyses it appears that the additional Cs and Sr activities come from more than one source. The sludge layer was estimated to contain 3681 Ci of Cs-137 and 1013 Ci of Sr-90 on 6/24/82, based on the ORNL analysis. Between the 6/15/82 pumpout (No. 18) and the subsequent ones (Nos. 19, 20, 21 and 22) 3223 Ci of Cs-137 and 2136 Ci of Sr-90 have appeared in the basement water, which is more activity for Sr-90 than was in the sludge layer. It is therefore, reasonable to conclude that the Sr appearing in the basement water was solubilized both from the sludge layer and from within the concrete at the 282' El. 2

Since the mobility of the soluble isotopes is greater from the sludge layer than from the concrete, the sludge layer should be depleted of activity and the concrete now remains as the principle source.

Table 2 shows the major possible sources of Cs-137 and Sr-90 activity at El. 282'-6". These should be the subject of specific decontamination action plans.

Interpretation of Data:

Prior to pumpout 16, the basement water activity was in chemical equilibrium with the basement reservoirs of contamination. The concentration of Cs-137 was stable at about 138 uCi/ml during this time. Since pumpout 16, concentrations have decreased. Two interpretations of this phenomena are discussed here. One treats the period following pumpout 16 as characterizing diffusion from a reservior with a 114 day time constant. The other 2

interpretation treats the concentration decrease as the result of a simple dilution process.

A diffusion or leaching model is represented by Figure 1 which shows the Cs-137 concentrations during each of the pumpouts of the reactor building basement. Zero time in this figure is shown at the time of the 16th pumpout in February 1982. At this time, the Cesium concentration in the reactor building basement water was 138 uCi/ml. Pumpouts prior to this time had not substantially reduced the basement Cesium concentration. A relatively long-term equilibrium is therefore postulated to have existed at time zero. Subsequent to time zero, the reactor building basement was pumped out six times (and is continuing to be refilled and pumped out).

The curve labeled on Figure 1 as "The 114-day time constant" represents the concentrations expected if the pumpout process continues. Some estimate may be made of the time when the Cs-137 concentration may be reducible to approximately 1 uCi/ml. This time is August 1983, assuming that refilling and draining of the basement continue approximately as has been done for pumpouts 17 2
through 22.

The second interpretation is represented by Figure 2 which shows the basement water concentration data plotted against the number of gallons of water that have been added to and removed from the basement since equilibrium existed. If a simple dilution process were operating, the data should approximately fit the line labeled "Dilution - Perfect Mixing". Departure of the data from this line indicates that the process is not a simple dilution. Figure 2 also shows by the dotted line a portion of the process that may have been dominated by dilution.

Two specific large reservoirs of Cs-137 identified on Table 2 do not participate in either of these interpretations. One is the elevator shaft, both the pit and the hollow block, and the adjacent loading platform. (See Figure 3.) The elevator shaft is surrounded by hollow concrete block, and the loading platform is poured, reinforced concrete of 8" thickness. Water in the elevator shaft could have easily seeped through the hollow block wall into the volume underneath the loading platform. The maximum contained water volume in the elevator pit and underneath the loading platform is 918 ft^3; the exact amount is unknown. This reservoir could contain up to 4160 Ci of Cs-137; however, it is not in direct communication with basement water and its effect on activity of the basement water is not considered in the dilution or diffusion models discussed above. The other reservoir is the reactor coolant drain tank volume which is physically separated from the reactor building basement water. It, therefore, does not participate in the contamination processes.

In conclusion, application of the available data suggests that significant activity has been leached out of the sludge, and that it may account for a majority of curies of Cs-137 that have appeared and been removed from the basement. It is also believed 2
that diffusion or leaching from concrete surfaces is ongoing and that continued filling and draining of the basement can achieve about 1 uCi/ml Cs-137 concentration by about August 1983.

TABLE 1

PUMPOUT DATA

PUMPOUT		GALLONS	GAL. ADDED	RCS	UCI/CC	CI FROM RCS		UCI/ML IN SAMPLE		GALLONS	CI REMAINING		CI REMOVED		CI APPEARING (A)	
DATE	NO.	REMOVED	RCS LEAK	CS-137	SR-90	CS-137	SR-90	C -137	SR-90	REMAINING	CS-137	SR-90	CS-137	SR-90	CS-137	SR-90
16	2/28/82	31,948	-	-	-	-	-	138	5.47	36,797	19,222	762	16,689	662	-	-
17	5/9/82	56,960	10,080	14.3	15.4	545.6	587.6	118	5.82	13,795	6,162	304	16,491	813	2,886	-232
18	6/15/82	4,718	5,328	7.1	12.7	143.2	256.1	87	5.32	16,716	5,505	337	1,554	95	754	-129
19	9/28/82	30,031	4,320	2.9	9.2	47.4	150.5	25.3	5.8	16,716	1,601	357	2,876	659	-1,075	539
20	11/8/82	41,681	-	-	-	-	-	12.5	3.8	25,844	1,223	372	1,972	600	1,594	604
21	1/3/83	47,920	-	-	-	-	-	8.5	3.4	15,773*	508	203	1,542	617	826	448
22	2/14/83	41,519	-	-	-	-	-	11.0	3.45	15,773*	657	206	1,729	542	1,878	545
	Totals	234,777											42,853	3,988	6,863	1,775

FOR PUMPOUTS PRIOR TO # 16, TOTAL GALLONS REMOVED = 568,646; @ 138 uCI/ml = 297,050 CI WERE REMOVED

(A) CURIES APPEARING IN THE BASEMENT WATER ARE ADDITIONAL CURIES THAT ARE SOLUBILIZED OUT OF THE CONCRETE OR SLUDGE AND ARE FOUND TO BE IN THE BASEMENT WATER WHEN A CURIE BALANCE IS PERFORMED BETWEEN SUCCESSIVE PUMPOUTS. A NEGATIVE VALUE INDICATE A RETURN OF CURIES FROM THE BASEMENT WATER TO THE RESERVOIR.

* ASSUMED

Attachment a

Attachment b

TABLE 2

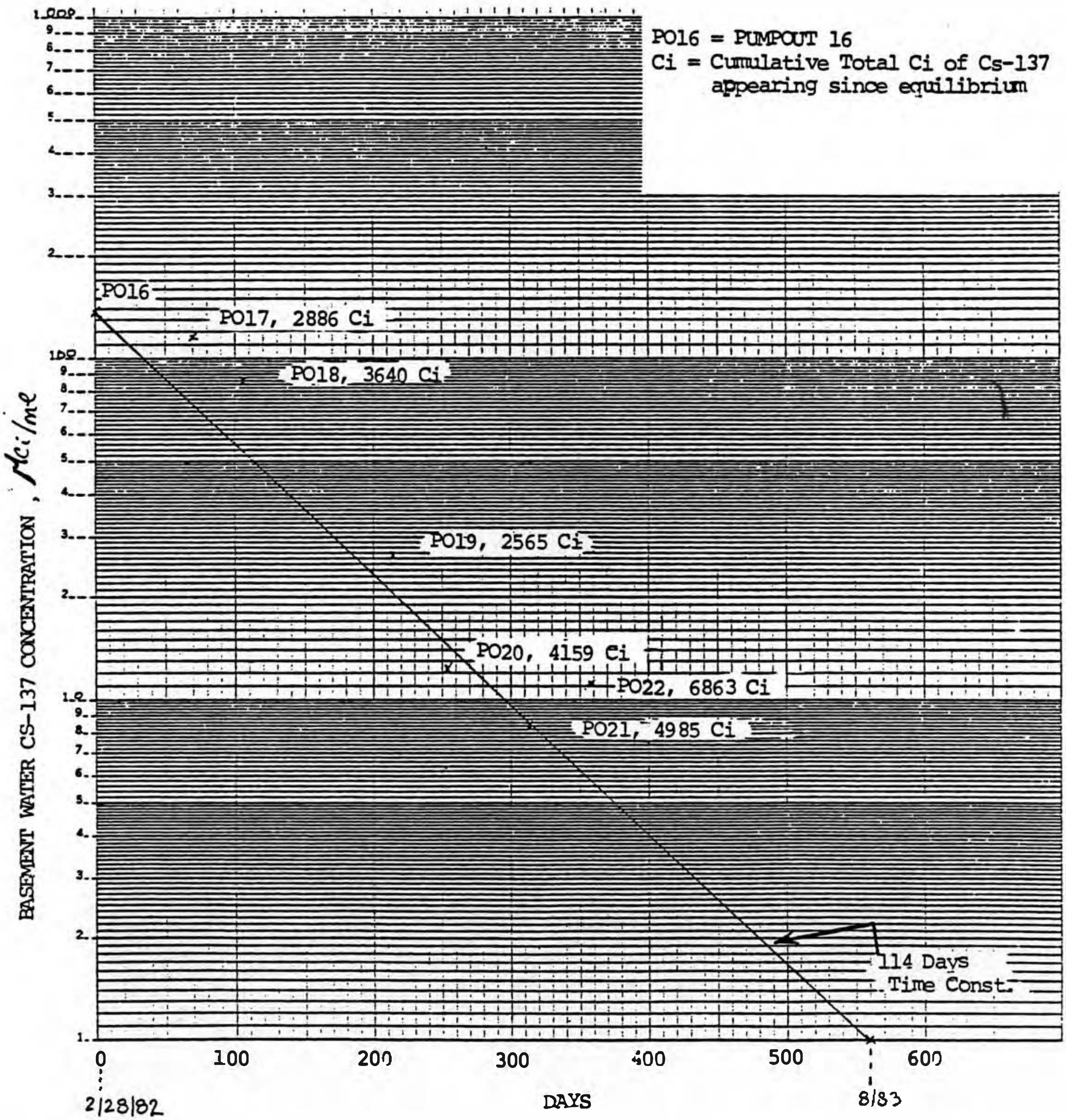
ESTIMATED RESERVOIRS OF CS-137 ON EL. 282-6

Possible Reservoirs	Estimated Cs-137 Conc. uCi/ml (Ref.1)	Estimated Total Curies Cs-137 (Ref.10)
Enclosed Stair Concrete Blocks Fully Flooded to El. 291-10	102	1281
Concrete Fill Slab (Cast) Assume Penetration 2 in. 6 in. 24 in.	 8 8 8	 386 1158 4631
Vertical Concrete Walls (Cast) Assume Penetration 1 in. 2 in.	 8 8	 196 393
Containment Liner	30 uCi/cm^2 (Ref. 9)	106
Sludge Assume Slurry 3/4 in. deep Total Slurry Soluble Solids Insoluble Solids	 204 (Ref.6) 187 17	 3672 3360 312
Reactor Coolant Drain Tank	50 (Ref.9)	1370
Water at Bottom of Elevator Shaft and Concrete Loading Platform	160	4160
Concrete Block Baffle Wall Assume Saturated	33	110
Reactor Building Sump 2,514 gal	160	1522
In-Core Instrument Chase 3,626 gal	160	2196
Underfloor Drains 2,033 gal	160	1231

Attachment c

FIGURE 1

El 282'6" CESIUM DIFFUSION MODEL



Attachment d

FIGURE 2

El 282'6" CESIUM DILUTION MODEL

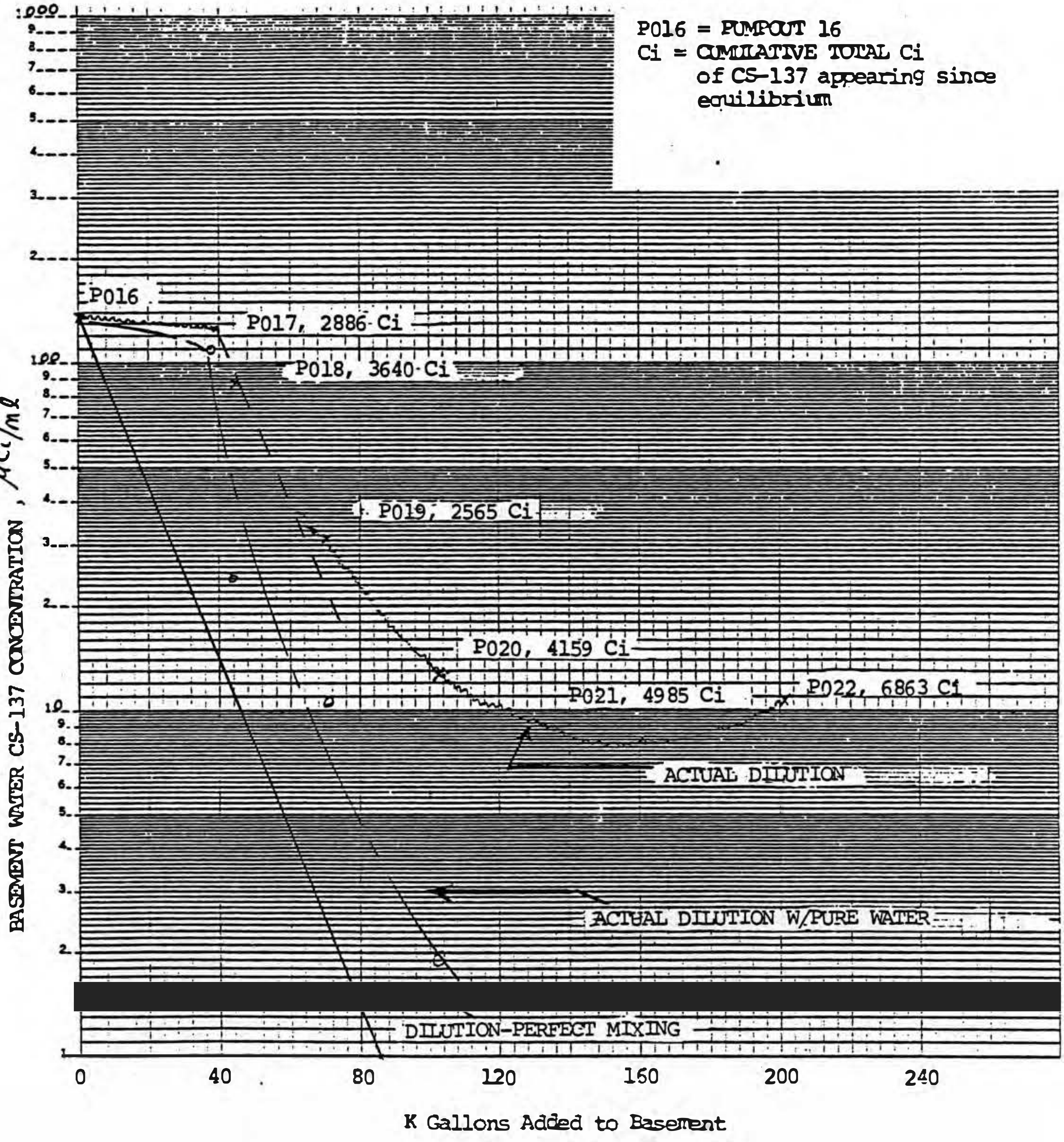
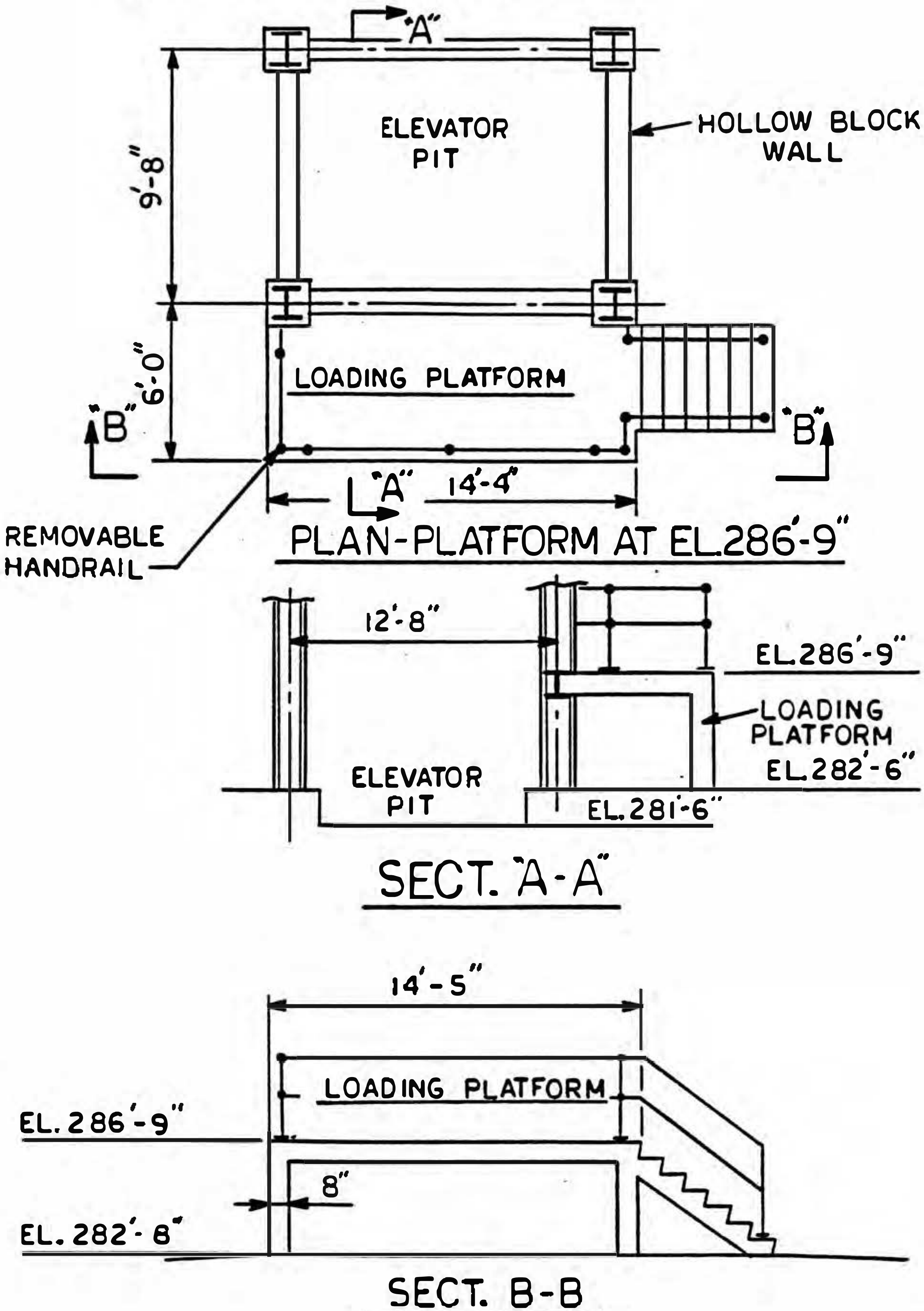


Figure 3

Elevation Shaft and Concrete Loading Platform



B. Radiological Engineering Plans

During this report period, TAAG reviewed numerous tasks with extensive radiological implications. These tasks included items in the initial planning stages, detailed work scoping and reports of completed work, and were as follows:

- Continued Reactor Building Decontamination
- Planning for Plenum Removal
- Head Removal and Underhead Characterization
- Enclosed Pathway Analysis
- Sludge Removal from the 282' Elevation Reactor Building Air Handling
- Dose Reduction Efforts

As requested in the 9 February 1983 letter from B. Kanga to W. Hamilton, TAAG has focused some of its attention on the issue of radiological engineering and how it might best be carried out in order to achieve an optimized approach. The issue is considered important, particularly in light of current man-rem exposure estimates (see section 10) and its potentially high variability. It is considered essential that an integrated and coherent radiological engineering approach exist in all phases of planning and work accomplishment in order to minimize personnel exposure and to achieve the exposure goals established.

TAAG observations regarding the radiological engineering efforts involved in the aforementioned tasks are as follows:

1. There is a great deal of variability in the amount of radiological engineering included in planning studies. In discussions with GPU/Bechtel personnel, it is evident that, depending upon the specific task group, radiological engineering may be factored in the early stages of planning, may be included in the final review circuit (after internal approvals are made) and varies considerably as to the depth of radiological input.

2. Radiological engineering considerations seem to be basically operating on two distinct levels; (a) at the detailed work procedure level, and (b) at a gross level where all sources in the reactor building are being treated, somewhat independent of the operations anticipated. At the detailed work procedure level, the radiological engineering appears to be working well. At the gross level, success in some areas has been good while others have not achieved expectations (e.g., building decon). At this gross level, there does not appear to be a coherent radiological engineering approach which ties building decon, dose reduction and handling of the airborne radioactivity together with mitigating actions to reduce exposure for specific evolutions.

TAAG has made recommendations in previous reports (TAAG report #4, dated 1 December 1982) regarding radiological engineering. Summarized, these were:

a) Develop a radiological engineering plan to support each principle work activity...
b) Direct priorities of the radiological engineering actions to support specific work.
c) Identify work areas and pathways to which radiological engineering need be taken for each principle activity.
d) Perform further characterization of work areas and pathways.

TAAG provides the following implementation recommendations to aid in achieving the recommendations previously made:

1. Assure that an appropriate member of radiological controls is included in any planning task force and that appropriate Radiation Controls management is a signatory of all planning studies.

2. TAAG understands that the Dose Reduction Task Force has been retasked under the name of the Dose Reduction Board, to look at additional activities which affect personnel radiation exposure. It is understood that this Board has been tasked to look at rather limited issues. It is TAAG's recommendation that consideration be given to expand the role of the Dose Reduction Board to advise a method which may be taken to assure a coherent approach to solving the complex and interrelated radiological issues such as dose reduction, building decon, airborne radioactivity handling, exposure resource planning, optimum work sequencing, etc., as these issues affect the overall defueling effort. An example of an area which this task force might undertake would be to determine the best generic approach to assure that work on the 347' elevation in support of defueling was carried out in the best radiological manner. In this area items to be evaluated might include the following:

   a) Best way to handle airborne radioactivity problem to aid in improving productivity through removing respiratory devices for the majority of work in the containment.

   b) Best way to get personnel to and from this work area in order to minimize exposure, e.g., defined pathways through low dose areas, shielding or removal of sources, elevator usage, etc.

   c) Generic dose reduction in the work areas as well as "wait" areas, including source analysis and actions required to reduce dose.

   d) Focus attention on source definition, particularly on the 347' elevation and determining what actions can and should be taken to minimize personnel exposure.

It is noted that the task force action items should be directed towards a more programmatic approach and should not replace the detailed radiological engineering reviews being undertaken on specific work procedures.

C,D. Use of Tunnel Concept and Evaluation of Alternatives

As requested by GPU Kanga Letter 4000-83-059, dated January 31, 1983 and EG&G Letter from H. M. Burton and M. D. Lovejoy to E. F. Sise dated February 7, 1983 an in depth study of alternatives to the pathways and work enclosure methods was conducted along with a cost-benefit analysis of the system of work enclosures. Attachment VI-C-One provides the detailed results of these efforts.

Based on the results of the studies, TAAG recommends that the proposed system of work enclosures be installed and that some of the alternatives which can be implemented for little or no cost, be implemented. None of the individual alternatives to the work enclosure has a probability of success as high as that of the work enclosure. Installation of the proposed enclosures for use during Reactor Disassembly and Defueling would eliminate the need for respirators and wet suits during most of the work and the resulting increase in productivity would generate a calculated net savings of 7,748 man-hours was calculated conservatively to generate a savings of 7700 in-plant man hours, 550 man-rem and $2.5 million.

ATTACHMENT VI-C-One

File: Attachment II C-1

April 5, 1983

B. K. Kanga
Director TMI-2

SUBJECT: The Pathway Approach to Work Area Contamination Control

REFERENCES:
1) NNI Letter, E.F. Sise to W.H. Hamilton, January 24, 1983.
2) GPU Letter #4000-83-059, B.K. Kanga to W.H. Hamilton, January 31, 1983.
3) EG&G Letter, H.M. Burton & M.D. Lovejoy to E.F. Sise, February 7, 1983.

ATTACHMENTS:
1) Study of Alternatives to Work Enclosures, Ed Wagner, Burns & Roe.
2) TMI-2 Work Enclosure Cost Benefit Analysis, Bruce Shiffler, Newport News Industrial Corporation.

Dear Mr. Kanga:

Reference #1 presented a conceptual study and cost estimate for installation of work enclosures in the TMI-2 Reactor Building for the Reactor Disassembly and Defueling effort. In response, GPU requested additional and more detailed information (reference #2). EG&G authorized this work (reference #3) and also requested an additional study of alternatives to the work enclosures which would serve the same purpose (i.e. help meet ALARA and worker productivity objectives). The two attached reports were prepared to provide this information. Attachment #1 is a study by Burns and Roe of the alternative to work enclosures. Attachment #2 is a cost-benefit analysis by Newport News Industrial of the system of work enclosures. Both reports were presented orally to GPU and TAAG on March 23. The written reports contain substantially the same information as the oral presentations, including many of the viewgraphs from the presentations which are used as figures in the texts. A copy of the attachment will also be included as part of the Fifth TAAG Report.

Based on the results of these two studies, TAAG recommends that the proposed system of work enclosures be installed and that some of the alternatives which can be implemented for little or no cost, be implemented.

Very truly yours,

W H Hamilton

W. H. Hamilton
TAAG Chairman

WHH/dms

cc: H. M. Burton

ATTACHMENT I

W.H. Hamilton memo dated April 5, 1983

Study of Alternatives to Work Enclosures
Ed Wagner - Burns & Roe

3/16/83

PARTIAL CONTENT OF REPORT TO GPU/BECHTEL ON A STUDY OF PATHWAYS AND CONTAINMENT OF THE DEFUELING WORK AREA

BACKGROUND:

In B. K. Kanga's, GPUN, letter of January 31, 1983, he requested that TAAG follow up their earlier recommendation concerning the "pathways approach" to establish satisfactory radiological conditions for the TMI-2 defueling. EG&G letter of February 7, 1983 authorized Newport News and Burns and Roe to study this question. The EG&G letter specifically requested that the following aspects should be included - "other alternatives including decontamination of the reactor building air coolers and general filtration of the in-containment air".

This section of the report addresses the alternatives.

CONCLUSIONS:

1. None of the individual alternatives to the tents has a probability of success as high as the tents. This is due mainly to the lack of the positive protective barrier between the contamination and the workers that the tents offer. However, many of the proposed alternatives have the potential for cost effective improvement of the airborne generated radioactivity problems (air particulate activity and surface contamination).
2. Some of the individual alternatives should be performed to improve radiological conditions. It is important to stop the distribution of air into and from the basement. The air flow to the basement should be stopped by closing the accessible balancing dampers, and by blanking off all flow to the D-rings. Improving the supply filtration system and increasing the purge air flow rate by running one train in recirculation and one train in purge should also improve the situation. Painting localized areas to trap smearables is a valuable tool to aid the final decontamination of the upper elevations, but it should not be considered a cost effective general area of decontamination approach. Sealing the basement should only be undertaken after it has been demonstrated that the removal of air flow to the basement has had a significant impact. Decontaminating the RB air cooling units should proceed to facilitate maintenance, to reduce area dose rates, and personnel exposures.
3. A combination of several alternatives would have a good probability of success in creating acceptable radiologic conditions in the defueling work area. However, the assurance of success would only be obtained by a step-wise process of physical testing and evaluation. For example,

a test of airborne activity distribution with the air coolers secured would help confirm whether sealing the basement would reduce airborne generation from the basement sufficiently to eliminate respirators.

4. Regardless which approach is used now to achieve satisfactory radiological conditions for defueling, the source of the radioactivity becoming airborne needs to be identified to enable reactor building cleanup. A coordinated program to identify and treat the airborne source, similar to the dose rate reduction effort, should be mounted.

SUMMARY:

Alternative	Cost: $1000	Cost: ManRem	Benefit: Eliminate Respirators	Benefit: Improve Radiolo Conditi
I. Improve Supply Air Filtration				
a. Modify supply filters	20	0	No	Yes
b. Operate purge filters in purification mode	0	0	No	Yes
II. Increase Purge Rates				
a. Two trains	0	0	No	Yes
b. Supplementary air filtration system	375	<1	Maybe	Yes
III. Decontaminate Air Coolers -	(No incremental costs - being done for dose rate reduction)		No	Yes
IV. Paint to Fix Contamination	1,030	230	No	Yes
V. Seal Basement From Rest of Building	176	92	Maybe	Yes
VI. Combination I, II.a, IV, V above	250-1000	125-300	Probably	Yes

I. INCREASE SUPPLY AIR FILTRATION

A factor in the generation of airborne contamination is the dust loading of the air in association with contaminated surfaces. Fresh air is supplied to the reactor building by the RB ventilation and purge system. As noted in the Technical Memorandum TM3680-8 (attached), the filters installed in that supply system are not adequate to remove all atmospheric dust and increasing the supply filtration efficiency could result in a significant reduction in the airborne contamination in the reactor building.

Replacing the present 85% rated efficiency bag filters with 95% rated efficiency bag filters would improve the efficiency for removing 0.5 micron particles from 55% to 86% and the efficiency for removing 5 micron particles from 98% to 100%.

While this would appear to improve the situation at minimum cost, it ignores a physical problem with the present supply system, i.e. the manner on which the filters are retained in the frame. The frames were installed backwards so that the filters tend to pull away from the frames under flow. This was done so that the filter could be installed back to back with a low efficiency roll pre-filter. However, it allows marginally filtered air to bypass the bag filters. This condition worsens as the pressure drop through the bag filters increases with time.

The best way to improve the supply efficiency of the purge system is to remove the present roll filter, and to reverse the present bag filter frames. This will allow the new higher efficiency bag filters to seat against the frame, and, if used in conjunction with a disposable pre-filter placed upstream of the bag filter, could increase the actual efficiency of supply filtration.

This modification would cost approximately $7-10,000 per train including labor and material costs and would involve no significant man-Rem costs. It will reduce the quantity of outside dust introduced into the building, however, sufficient dust dirt and other airborne particles which can carry radioactive isotopes already exist in the building so that it is not certain that airborne contamination will be significantly reduced. Therefore, the probability of this modification alone eliminating the respirator requirement is very low. We believe that it should be done independent of the pathways and work enclosure approach.

It is noted that the purge system can be run in the purification mode (i.e. recirculation) for extended periods to reduce airborne dust in the reactor building. In the purification mode, all air exhausted from the reactor building is filtered with HEPA and charcoal filtration and discharged back into containment. Also, due to the reduced pressure drop, significantly higher air flow rates can be realized in the existing system over the purge mode, ~25,000 CFM vs ~19,000 CFM. This would increase the turnover rate and would reduce airborne dust more rapidly. If desired, one of the trains could be run in the purge mode to reduce krypton, and tritium concentrations and to supply the

assurance of breathable air, while the other train is operated in the purification mode. This change in operation would introduce virtually dust free air in containment and could be accomplished from the control room at no cost.

II. INCREASE PURGE RATES

A method of reducing airborne concentrations in the reactor building is to increase the purge rates. The steady state airborne concentration is a linear function of the purge rate as described in the Technical Memorandum TM3680-8 (attached). TAAG has recommended that the second train be turned on in the third TAAG report of August 31, 1982. Since two trains would approximately double the purge rate, it would be expected that the airborne contamination would be decreased by a factor of two.

The installed exhaust system has a total capacity of 50,000 CFM. At present, the system is limited by the supply system to around 40,000 CFM. In order to increase the purge rate beyond 40,000 CFM, air would have to be drawn from outside. The filtration system on the containment air control envelope (CACE) could be utilized for a maximum of 50,000 CFM without modification to the existing systems. However, a factor of ten increase in the purge rate would be required to reduce the airborne contamination by enough to remove the requirement for respirators. In order to increase the purge rate by that much, modifications to the existing purge system would need to be performed.

The auxiliary and fuel handling building supplementary air filtration system (henceforth referred to as WG-1) can be utilized for this purpose. A detailed description of the required changes is given in Technical Memorandum TM3680-8. Coupled with a matching supply filtration system in the CACE, this system could deliver a reliable 110,000 CFM of purge flow. This modification would be able to reduce the airborne contamination approximately one order of magnitude.

A modification of this sort would cost approximately $250,000 for the purge exhaust tie-in to WG-1 and $125,000 for the supply system for the CACE. The man-Rem cost of these modifications would not be significant since all of the work would take place outside of the reactor building. Unfortunately, it would only be effective on that function of the airborne activity which is normally airborne. This sort of airborne contamination, as measured by the hi-volume air samplers, is typically an order of magnitude below the BZA data. This is believed to be due to the so called "pig pen" effect; the stirring up at surface contamination on the floor. The purge system will not work directly on the BZA type airborne contamination and may not actually result in the factor of ten reduction in the BZA readings required to be able to remove the requirement for respirators. The recommendation of TAAG's third report to run both purge trains to determine the effects would seem to be a prerequisite to modifications to the purge system in view of the cost and the effort required.

III. DECONTAMINATE REACTOR BUILDING AIR COOLING UNITS

GPU/Bechtel is exploring decontamination of the reactor building air cooling units as a method to significantly reduce airborne contamination. It is suspected by some that the internal contamination of the cooling system is the primary vehicle for recontamination and for airborne contamination. The data supporting this belief are the high gamma dose rates around the building cooling units and the "LOCA" ducts. These average about 1-5 R/hr. on contact with the outside of the unit enclosures near the cooling coils. These sorts of dose rates are understandable for several reasons:

1. The cooling coils in the units consist of copper coils with copper fins. The total exposed surface area is large and the air velocity past the coils is low ($\simeq$500 FPM). Airborne dust would tend to stay on the close spaced fins once deposited.
2. The units are square and have many dust collecting pockets for contamination to accumulate.
3. The units (excluding the coils) are painted with a placite coating system which is supposed to be a high gloss decontaminable epoxy. However, from personal experience, the paint as applied on the fans and the plenum, has a rough, dull finish which would be difficult to decontaminate.
4. All water condensing in the units after the accident would collect on the floor of the unit. The floor of the unit is a flat, placite painted metal surface which has a corner pocket drain. This arrangement would allow most of the water to drain but some would puddle and evaporate leaving behind the containment.

After the accident occured and containment isolation was initiated, the dampers for the LOCA ducts (D-5127A and D-5127B) opened and have remained open ever since. This means that a large portion of containment flow from the cooling units has been directed through these ducts. However, the internal surfaces of these ducts were painted with placite which had the proper gloss finish. It should be receptive to decontamination efforts except near the gaskets. It is not clear why these ducts should have the indicated dose rates (400 mR/hr. on contact with ducts), since they are vertical ducts with relatively high air velocities and smooth internal surfaces.

The planned decontamination of the coils, the units, and any accessible, contaminated duct is a reasonable course of action for dose rate reduction. It would allow maintenance of the units and would reduce general area dose rates. It may, to some extent, reduce airborne contamination but it does not seem reasonable to expect decontamination of the reactor building air cooling units to significantly reduce overall building airborne levels. There are several reasons for this:

1. The recontamination rates observed inside the reactor building seem too high to have their source in a 1-5 R/hr. hotspot.
2. Air distributed from the RB air cooling units is supplied to areas which have high surface contamination, including the D-rings and the basement. All other areas receive ventilation via transfer air from these high contamination areas. While the RB air cooling units are operating normally more than 150,000 CFM of air is supplied to either the D-rings or to the basement. This air can serve as a transport mechanism to redestribute surface contamination in these areas elsewhere in the building.

3. The LOCA ducts are described as being a potential source of contamination. However, the LOCA ducts are not required to be open and can be controlled from the control room. A visual inspection of the EL305' and 345'6" duct work should verify the integrity of the normal ducts (for fire protection function) and eliminate the need for the LOCA ducts to be open. Closing these ducts and monitoring the results should quantify their contribution to airborne.

Decontaminating the RB air cooling units should proceed despite the questionable effect it will have on airborne generation. It will reduce area dose rates around the coolers and allow maintenance to be performed on the units to increase their availability. However, it is not expected that this decontamination effort will be effective in reducing airborne contamination enough to remove the requirement for respirators.

IV. PAINTING

An alternate radiological engineering approach, which does not depend on air handling or filtration systems, relies on the use of painting to fix contamination to surfaces. In general this approach fixes the smearable surface contamination by painting over it, prevents it from becoming airborne, and thus enables work to proceed without the need for personnel respiratory protection. Localized shielding is brought in as required and decontamination is accomplished by removal of the new paint and of any structural material required to reduce dose rates.

This approach has the advantage of decoupling the airborne and the surface contamination problems. Once the contaminants are fixed to the surface, they can no longer contribute to the airborne contamination. Thus, neither worker activity nor decontamination efforts will generate airborne contamination. Also, painting is a fairly quick, easily verifiable process which can be applied with relatively standard equipment and procedures. When dry, the paint and the surface contamination are set up in a composite matrix which prevents the recontamination of surrounding areas even during physical removal of material for decontamination.

While this approach has merits, it seems doubtful that it can be effectively employed throughout TMI-2 reactor building. The smearable contamination contribution to airborne has not been clearly defined and, in fact, it has been demonstrated that smearable surface contamination on the upper elevations (EL305' and 347'6") is mainly the result of deposition from airborne contamination. Also, decontamination efforts

to date have effectively reduced surface smearable contamination on the upper levels without concomitant reductions in airborne concentrations. All of these facts lead to the conclusion that the source of airborne generation is either not on the upper elevations or has yet to be affected by decontamination activities to date.

The dose rates in the basement are too high to permit a thorough painting effort. Also the water logged condition of the walls, floors and paints may reduce the applicability of painting the basement.

This approach would also be costly. It would cost $1,030,000* to paint the two elevations to a height of 10 feet and could entail as much as 230 man-Rem*. Painting the basement would be even more expensive, both in terms of money and man-Rem exposures.

*Assuming $375/manhour, consistent with the GPU/Bechtel TMI-2 recovery estimate, revision 2 of 12/31/82 and 2750 standard manhours of work in containment. Man-Rem estimates based on average of 200 mR/hr. dose rate in the EL305 work area and 75 mR/hr. in the EL347 work area.

V. SEAL-OFF BASEMENT

The basement is the most contaminated of all the elevations in the reactor building. The air flow in the building from the RB air cooling units dumps approximately 25,000 CFM of air into the basement for cooling. This air migrates back up to the upper elevations where the units recirculate at. The bulk of the remaining air flow from the RB air cooling units is directed into the D-rings and much of this flow is also directed into the basement to cool the bases of the steam generators and of the reactor vessel. This air also migrates back up to the upper elevations where the units recirculate it. Somewhere between 50,000 and 100,000 CFM of air is directed into the basement in this manner. Since the airborne and the surface contamination is higher in the basement, it is quite likely that this forced air flow into the basement is the source of the airborne and of the recontamination rates observed on the upper elevations.

Air delivered by the RB air cooling system directly to the basement area is distributed through different ducts than the rest of the building. These ducts have balancing dampers just above the EL305'-0" floor which could be shut to eliminate air distribution to the basement outside of the D-rings. The rest of the air distributed to the basement via the D-rings could be eliminated by blanking off the main discharge duct on elevation 305' (downstream of the smoke detectors) and by routing temporary duct work where desired on the upper elevations. Less than 36,000 CFM are required for fire detection, so temporary duct work need not be sized for full flow. These modifications would eliminate all air flow to the basement under positive pressure. (The reactor building ventilation and purge system, which is functionally

separate from the RB air cooling system, draws air from the "B" steam generator D-ring. This will result in some slight negative pressure in the basement which will result in some small flow into the basement but this will not effect the airborne contamination problem.)

If it becomes necessary to take more direct measures to isolate the basement from the rest of the building, all major penetrations through the 305' floor can be sealed with herculite or facilon sheeting. The major penetrations are: the D-rings, the stairs, the covered hatch, the incore instrument chase, and the seismic gap. This will cost approximately $82,000 for labor and materials and will entail about 92* man-Rem. Modifications to the duct work will cost an additional $28,000 to $94,000* depending on the scope of modifications.

The probability of this approach reducing airborne concentrations on the upper elevations is high. The basement is the obvious place to suspect to be the source of the airborne contamination. However, it is not clear that the BZA data will decrease enough to permit the removal of respirators. Also, the net reduction of air flow would reduce mixing and could result in localized areas of high airborne contamination.

*Assuming $375/manhour, consistent with the GPU/Bechtel TMI-2 recovery estimate, revision 2 of 12/31/82, 220 manhours of work in containment and $1000 for facilon. Man-Rem estimates based on average of 200 mR/hr. dose rate in the EL305 work area and 75 mR/hr. in the EL347 work area.

TECHNICAL MEMORANDUM

NUMBER TM3680-8, Rev. 1

DATE 4/1/83

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TO E. J. Wagner

FROM P. Massaia/C. W. Hess
P Massaia/CW Hess

SUBJECT W.O. 3770-01
Technical Assistance & Advisory Group
Three Mile Island Unit 2
Reactor Building Air Handling Systems
for Airborne Particulate Activity

KEYWORDS TMI-2, Airborne Contamination,

Air Handling, Purge

REFERENCES:

1) "Gross Decontamination Experiment Report", (Draft) TPO/TMI-009, Bechtel National Inc., Nuclear Fuel Operations, June 1982

2) GPU Nuclear Inter-office Memorandum to L. H. Barrett, No. 4400-82M-0061, January 25, 1982

3) Three Mile Island System Description, Number 35A, "Reactor Building Purge System"

4) Flanders Filters, Inc. Catalog

5) Burns and Roe Dwg. 2041, Rev. 26, "Flow Diagram, Reactor Building Ventilation and Purge"

6) Burns and Roe Calculation #36880-15-18, "Airbone Contamination in R.B.", December 15, 1982

7) Burns and Roe Calculation #3680-15-19, "Curies on Concrete Surfaces on El. 305' and 347'6", December 29, 1982

8) Burns and Roe Calculation #3680-15-20, "Curies on Steel Surfaces on El 305' and 347'6", January 5, 1983

9) Burns and Roe Calculation #3680-15-16, "Curies in Reactor Building Basement", November 15, 1982

10) Burns and Roe Calculation #3680-15-15, "Leach Rate of Activity into Reactor Building Sump Water", January 10, 1983

11) GPU Nuclear "TMI-2 Reactor Building Entry Data Table", from S. Frey of GPUN to P. Massaia of Burns and Roe, March 22, 1983 1

ATTACHMENTS:
A) Figure 1 - Reactor Building Purge and Purification System
B) Figure 2 - Airborne Profiles
C) Table 1 - Cesium Burden in Reactor Building
D) Figure 3 - TMI-2 Reactor Building Atmospheric Containment Levels

SUMMARY AND CONCLUSIONS

Recovery efforts in the TMI-2 reactor building are hampered by the airborne contamination inside the building. Airborne contamination measured by Breathing Zone Apparatus (BZA) samples must be lowered approximately one order of magnitude in order to permit the removal of personnel respirators. The methods available to accomplish this are:

1) Surface decontamination - This has been done with some success but appears to have reached a plateau still above MPC.

2) Isolation and purification of the work areas. This is separately being considered by TAAG and GPUN/Bechtel.

3) Aggressive surface decontamination such as paint removal and gross removal of concrete. Not applied as of yet.

4) Fixing loose surface contamination with temporary paint or coatings. Strippable coatings have been used to a limited extent but experienced rapid recontamination.

5) Modifications to the reactor building air handling systems. 1
This technical memorandum considers this method.

The steady state airborne concentration is a function of three factors: 1) the airborne production rate, 2) the airborne redeposition rate, and 3) the purge air flow rate. To date, all efforts to reduce the airborne concentrations have been directed toward the first of these, the production rate. The major activity in this regard has been the decontamination efforts inside the reactor building since March, 1982 (Ref. 1).

The basement is the most contaminated elevation in the reactor building. To date, no systematic decontamination effort has been undertaken in the basement due to the dose rates. Surface contamination in the basement can be transported to the upper elevations by the transfer air caused by the operation of the reactor building air cooling units. These units dump approximately
100,000 CFM of air into the basement. This air sweeps through 1
the basement before returning to the upper elevations through floor penetrations and the D-rings. This process could conceivably be the major production mechanism for airborne contamination in the building. Relatively simple modifications to

the system can eliminate all air flow into the basement and could significantly reduce the airborne production rate. Since the steady state airborne concentration is proportional to the production rate, any reduction in the production rate will result in an equal reduction of the airborne concentration. Also, the recontamination rates observed on the upper elevations are pro- 1
bably a function of this production mechanism and should decrease as the production rate is reduced.

The steady state airborne concentration is inversely proportional to the purge flow rate. If the air flow can be doubled, the steady state airborne concentration will be reduced by a factor of two. Since only one of the two installed purge trains is in operation, a factor of two reduction in airborne can be achieved by simply turning on the idle train. Unfortunately, this effect will be limited to the general airborne concentration. Since the BZA airborne data is driving the requirement for respirators, it is not clear that doubling the purge rate will make a factor of two reduction in the most important component of airborne contamination; BZA.

A potential benefit of the purge system's operation could be the reduction of the atmospheric dust introduced into the reactor building through the purge mode. The present air supply filtration system is not designed to remove all of this dust. Modifications to the air supply filtration system could reduce the dust loading of the containment air and, in conjunction with further decontamination efforts, reduce the BZA airborne concentration.

Another use for the purge system is the prevention of airborne releases from the equipment hatch or the personnel airlock #1 openings. When these hatches have been removed to transfer equipment into the reactor building, the existing air handling units could be used to maintain enough air flow into the opening to assure that no airborne contamination escapes. Such a line-up might also be made to work for personnel airlock #2, but it would be complicated by the restriction imposed by drawing all air through the service building. The service building's air supply system is capable of providing less than 16,000 CFM, which is less than a single turn of the reactor building air handling system (~20,000 CFM). Hence, additional sources of supply air will have to be provided and the entire nuclear air handling system rebalanced. Perturbations in the plant, such as outages of other air handling systems, or the opening or closing of doors in the service building, could cause momentary imbalances in the expected pattern of air flow.

If the equipment hatch is to be removed for long periods of time to facilitate equipment entries, the supplementary air filtration system (WG-1) could be tied directly into the reactor building

ventilation ductwork to supply a maximum of 110,000 CFM air flow through the hatchway which is more than enough to provide a capture velocity to prevent airborne releases. By utilizing an access building, or a restriction in the opening, the existing 50,000 CFM rated purge exhaust would be adequate.

RECOMMENDATIONS

1. Terminate all air flow into the reactor building basement. This can be accomplished by closing the dampers from the air cooling units to the basement and by blanking off the flow to the D-rings. Temporary ductwork, if required, can be 1
routed as needed to maintain the minimum flow rate for the fire detection system (~36,000 CFM).

2. Increase the air clean-up rate in the reactor building by operating both purge trains. Monitor the airborne contamination to document the results. If necessary, the second train could be run in the purification mode and would not affect off-site releases.

3. The importance to airborne contamination of atmospheric dust loading should be considered prior to committing to any action that may affect the reactor building air handling systems. Specifically:

   a) Maintenance of the existing supply filtration units should be adequate to assure their operability.

   b) Increase the efficiency of the supply air filtration by replacing installed "bag" filters with 95% efficient (NBS) units or by modifying the installed system to accept HEPA filters.

   c) Plans to introduce unfiltered air into the building through opened air locks or hatches should consider the consequences of increased dust loadings.

4. Operate the purge system in the purification mode whenever practical. The purification mode has essentially the same effect on the airborne contamination as does the purge mode but has the advantage that it does not introduce atmospheric dust into the reactor building.

5. The fans of the reactor building air cooling units provide air mixing in the reactor building and they should remain in operation to improve the efficiency of the air clean-up systems.

6. Take all steps necessary to demonstrate that borated water is not required for decon solutions. Boron residue on the 1

walls may be a significant contributor to airborne contamination in the building.

DISCUSSION

Current Status

At present, entries to the TMI-2 reactor building are made through personnel airlocks. If required, both doors of either airlock can be opened to facilitate entries into the reactor building and to permit long or bulky objects to be carried into the reactor building. Prior to each entry, a single train of the reactor building ventilation and purge system is operated in the purge mode for at least 24 hours. Due to the current number of reactor building entries, one train of the purge system operates all the time. Samples are taken by using the installed R. B. 1
atmosphere sampling lines (valves AH-V103 and AH-V106), by portable high volume air samplers hand carried into containment and placed at designated locations, and by a BZA on each member of the entry team.

The air cooling units have operated since the accident. Typically, three or four of the five units were in operation during that time. The LOCA ducts have been open since the accident so that the normal air distribution has been perturbed by these additional flow paths. The units were shut off 3/16/83 to permit decontamination of the internals of the units. A minimum flow of 36,000 CFM will be maintained to satisfy the fire protection system's requirements for the smoke detectors.

Current airborne particulate levels, as taken by the high volume air samples, are on the order 1 x 10^{-9} μCi/ml. BZA data typically range on order of magnitude higher than this. Strontium 90 and cesium 137 are currently the major airborne isotopes. Noble gases and tritium contributions to the airborne are sampled using the installed sample points (AH-V103 and AH-V106) which take samples from the containment dome.

Installed Systems (Ref. 3)

The reactor building ventilation and purge system is designed to provide fresh, heated air to containment while purging filtered air to the environment, or to recirculate and to clean the reactor building air. The system consists of two parallel and redundant trains of air supply units and of purge exhaust units (see Figure 1). The system is aided by the operation of the reactor building air cooling unit fans which assure circulation and mixing of air within the reactor building.

The two purge supply units take air from the intake tunnel which is the common air supply source for all nuclear related air

handling systems. Each unit consists of a 25,000 CFM fan, a roll prefilter, a replaceable high efficiency filter and a multi-stage electric heater all mounted in a steel cabinet. The unit's inlet and outlet dampers are interlocked to open with fan start. Each unit will shut down and isolate itself with high reactor building pressure, a fire protection signal, an ES signal, station high radiation, or when the recirculation damper (AH-D-5129D or 5129C) is open. For the purification mode, the purge supply fans are shut down and the purge exhaust fans recirculate and purify air in the containment.

Reactor building air is discharged to the station vent by the two purge exhaust units. Each unit consists of a 25,000 CFM fan, roll prefilter, HEPA filter, (high efficiency particulate air filters), an activated charcoal filter, and another HEPA filter, all mounted in a steel cabinet. The purge exhaust unit will shut down with high reactor building pressure, a fire protection signal, or an ES signal. Purge fan operation requires that either the damper to the station vent be open (Purge Mode) or the supply fan discharge damper is shut (Purification Mode). The reactor building air purification system will automatically shift from the purge mode to the purification mode with a high radiation signal (shut discharge damper and open recirc. damper). Additionally, the exhaust damper to the station vent will shut with high reactor building pressure. Normally, reactor building air is first cleaned in the purification mode, and then discharged to atmosphere in the purge mode. The system may operate with either one (25,000 CFM) or two units (50,000 CFM). (Operational history to date indicates that the installed system line-up operates at less than 20,000 CFM for a single train with an expected 35,000-40,000 CFM for two units.)

All filters in the purge exhaust system are supplied with local differential pressure indicators and all automatic roll filters have differential pressure switches to advance the media on a pre-set differential. Limit switches are provided to energize an alarm when the media is to be replaced.

The system can be placed in operation in the purge mode or the recirculation mode from the control room. In the recirculation mode, all air will be recirculated through ductwork which is rated at 2 psig internal pressure and which has been tested to be bubble tight. There may be some minor amount of discharge to the stack by leakage through dampers D5129A & B but this would be filtered air, and would be a small quantity.

The reactor building air cooling units are part of the reactor building air handling system but are functionally separate from the purge system. The air cooling units, which are located in
the reactor building on EL. 305', are designed to cool the reac- 1
tor building during power operation. The system consists of

five (5) units, each having a variable flow fan and two cooling coils. These units all feed a common plenum. Air is then distributed in the reactor building by ductwork. Air returns to the units by free transfer throughout the building. Under normal conditions, the air cooling units distribute 170,000 CFM throughout the building to remove 5.75 x 10^6 BTU/hr at 96°F. Four of the five units are normally in operation.

In addition to the heat removal function, the air cooling units provide air circulation within the reactor building to reduce temperature stratification and to prevent the formation of gas pockets. Also, the reactor building fire protection system smoke detectors are located in the air cooling units' ductwork.

The LOCA ducts are safety related ducts which tie into the air cooling system ductwork. In the event of a containment isolation signal, the dampers isolating these ducts from the normal air flow open and air flows through the LOCA ducts. These provide a flow path to assure air cooling in the event of an accident which could collapse the normal ductwork.

<u>Airborne Reduction</u>

The basement is the most contaminated elevation in the reactor building. The air flow in the building from the RB air cooling units dumps approximately 25,000 CFM of air into the basement for 1
cooling. This air migrates back up to the upper elevations where the units recirculate it. The bulk of the remaining air flow from the RB air cooling units is directed into the D-rings and much of this flow is directed into the basement to cool the bases of the steam generators and of the reactor vessel. This air also migrates back up to the upper elevations where the units recirculate it. Somewhere between 50,000 and 100,000 CFM of air is directed into the basement in this manner. Since the airborne and the surface contamination is higher in the basement, it is quite likely that this forced air flow into the basement is the source of the airborne and of the recontamination rates observed on the upper elevations.

Assuming that the probability of a particle becoming airborne is equal for all elevations, terminating air flow to the basement should reduce airborne concentration by a factor of 300.

Air delivered by the RB air cooling system directly to the basement area is distributed through different ducts than the rest of the building. These ducts have balancing dampers just above the EL. 305'-0" floor which could be shut to eliminate air distribution to the basement outside of the D-rings. The balancing dampers in the ductwork to the D-rings are located at EL. 337' and would require scaffolding to enable personnel to get to them. An alternative to the dampers could be to blank off the main

discharge duct on EL. 305' (downstream of the smoke detectors) and to route temporary duct work where desired on the upper elevations. Less than 36,000 CFM are required for fire detection, so temporary duct work need not be sized for full flow. These modifications would eliminate all air flow to the basement under positive pressure. (The reactor building ventilation and purge system draws air from the "B" steam generator D-ring. This will result in some slight negative pressure in the basement which will result in some small flow into the basement but this will not affect the airborne contamination problem.)

If it becomes necessary to take more direct measures to isolate the basement from the rest of the building, all major penetrations through the 305' floor can be sealed with herculite or facilon
sheeting. The major penetrations are: the D-rings, the stairs, 1
the covered hatch, the incore instrument chase, and the seismic gap.

The probability of this approach reducing airborne concentrations on the upper elevations is high. The basement is the obvious place to suspect to be the source of the airborne contamination. It contains 99.68% of the total cesium in the building (Table 1). However, it is not clear that eliminating air flow to the basement alone will decrease the BZA concentrations enough to permit the removal of respirators since there are already enough particles on the upper elevations. Also, the net reduction of air flow would reduce mixing and could result in localized areas of high airborne contamination.

An obvious method for reducing airborne levels in containment is to use the installed reactor building air handling systems in the purge or the purification mode. Since the air removed from the reactor building is filtered and monitored, no large releases to the environment should occur. The fresh or filtered air introduced to the reactor building will dilute the airborne contamination and reduce airborne concentrations. In general, larger clean-up air flows will result in larger reductions in the steady state airborne concentration. The relationship between purge rate and airborne concentration is shown on Figure 2. It can be seen from the figure that a steady state equilibrium, which is a function of the airborne production rate, the redeposition rate, and the purge rate, is reached several hours after the initiation of purge. Continued purging past that time maintains this steady state airborne level. Increasing the air flow rate of the clean-up will decrease the steady state airborne concentration in an inverse relationship. Hence, doubling the purge flow rate will reduce the steady state airborne concentration by a factor of two.

However, an order of magnitude reduction in BZA airborne concentration is required to get all samples consistently below MPC.

Increasing the purge airflow by a factor of 10 is not a practical goal. Hence, the reactor building air clean-up systems cannot reduce airborne concentrations below MPC without other airborne reducing activities.

No practical method of increasing the redeposition rate at TMI is obvious. However, the production rate could be reduced by decontaminating all surfaces or by fixing all surface concentration. The air handling system might affect some "decontamination" if it removed significant quantities of surface contamination. Table 1 was prepared to investigate this possibility.

Table 1 shows the cesium 137 burden on or associated with surfaces of various areas and structures within the reactor building. Only a small fraction of the cesium in the building is in the form of airborne contamination. Therefore, the building, especially the basement, serves as an "infinite" source of cesium 1
for airborne contamination. While not all of the cesium is in a form suitable for airborne generation, it seems clear that a large reservoir of radioisotope is available for airborne contamination.

The removal rate of cesium from the building is less than 2 μCi of cesium 137 per minute (@19,000 CFM). A single purge train running at 19,000 CFM would remove the total airborne burden in five hours assuming no production rate and a homogenous distribution of airborne. However, the reservoir is so large that if only 0.1% of the total cesium burden is in a form suitable for airborne generation, it would take 6 years for the purge flow to eliminate this material.

While this is not a major effect, it seems to be possible to identify just such a long-term result. Until recently, the BZA airborne concentration data has dropped with time (see Figure 3). If the number of containment air volume turns per month is compared to the average BZA concentrations over a certain time period, they are found to be inversely proportional (Ref. 6). The variation in the constant of proportionality for the time periods identified on Figure 3 is only a factor of three. This may indicate the presence of a secondary removal phenomenon which reduces the airborne production rate with time. (Note: the production rate for Figure 1 was assumed not to be a function of the air flow rate.) This may indicate that the steady state solution decreases with time for any constant purge flow rate.

However, the BZA data since entry 130 has been rising steadily in spite of a constant purge rate. This could be due to several conditions: 1) Increased work activity in the building; 2) More activities in heavily contaminated areas; or 3) Increased airborne production rate.

Entry #130 corresponds to December 1982. Since this time, the majority of work performed in the reactor building has been on the polar crane. It is unlikely that increased BZA airborne activity is being caused by the work on the polar crane, since contamination in that area is relatively low. Clearly, more worker activity in heavily contaminated areas has not been the reason for BZA increase either. Therefore, an increase in airborne production rate is the probable cause of the increased BZA airborne contamination.

The available data indicates that the depth of water in the basement has been less than 1' since March 1982, and has been as low
as 1/4" (Ref. 11). The surfaces in the basement which were once
covered up to 8' with highly contamination water and sludge, are 1
now drying out. The release of dry, contaminated dust is expected from these surfaces, and could be the source of the recent increase in BZA airborne contamination. Borated water used in decontamination washings of surfaces also has a tendency, once dried, to release dust particle due to the nature of the boron crystals. To reduce the dispersion of the contaminated dust throughout the reactor building, air filteration from highly contaminated dried surfaces, especially from the basement, should be terminated, and the effect of removing boron from the decon water should be investigated.

The purge system might also be used to reduce the airborne production rate by reducing the amount of atmospheric dust introduced into the reactor building. The supply bag filters are rated at 85% using the NBS discoloration test and remove .5 micron sized particles at a 55% efficiency, and 5 micron sized particles at a 98% efficiency (Ref. 3). Since atmospheric dust ranges in size from .01 to 5 microns, this filter would not seem to be adequate to preclude the introduction of dust into the reactor building. Once introduced, this dust will settle onto contaminated surfaces, become contaminated, and serve as a mechanism for airborne generation. It is not clear what fraction of airborne contamination is the result of this mechanism. Increasing the efficiency of the air supply filtration system would reduce the amount of atmospheric dust which may help to reduce the airborne production rate. Intuitively, reducing the amount of dust introduced into the building could not have an adverse impact as long as the air flow is not reduced significantly. As a minimum, maintaining the air supply filtration system should proceed to assure that it is functioning as designed. Operation of at least one of the air handling trains in the purification mode should increase the air flow without increasing the atmospheric dust loading inside the building.

Concepts for Containment Entries

The present access to the TMI-2 reactor building is through the personnel airlocks. Personnel airlock #1 is located in the equipment hatch and is accessible from either the control building or from outside. It is removable and can be pulled out of the equipment hatch if desired. Personnel airlock #2 is located at EL. 305', Azimuth 282°, and is accessible from the service 1
building. It is fixed to the containment building and cannot be removed. Both doors of either airlock can be opened at the same time to permit long objects to be carried into the reactor building. Eventually, a larger opening may be needed to support recovery activities inside containment. At that point, either airlock #1, or the equipment hatch will have to be removed. In order for such an expedient to be permitted, some method must be devised to assure that there will be no uncontrolled airborne releases from the opening.

One method would be to rely on a capture air velocity entering containment through the opening. A common capture air velocity used for radioactive lab hoods is 200 FPM. Only 13,000 CFM is required to achieve this velocity if the removable airlock is removed from the hatch. This can be achieved by using a single train of the existing RB ventilation and purge system which has a capacity of ~ 20,000 CFM.

However, if the entire equipment hatch is removed, 85,000 CFM flow through the hatch would be required to obtain a 200 FPM capture velocity. This flow is not achievable with the existing RB ventilation and purge system (40,000 CFM). In order to achieve the required velocity, the auxiliary and fuel handling building temporary HVAC system (henceforth WG-1) can be utilized. The most favorable way to utilize it would be to utilize WG-1 as a backup to the existing purge system. The following is required to tie in WG-1 to the RB ventilation and purge system (see Ref. 5):

1. Utilize the two containment supply penetrations (R-549 and R-550) for the new exhaust path.

2. Make two ductwork tie-ins: one between damper D5128C and isolation valve AHV-1B, one between damper D5128D and isolation valve AHV-1A. Place an isolation damper at each tie-in. The isolation damper should be a heavy duty low leakage industrial damper similar in construction to D5128 C and D. These dampers should be controlled from the WG-1 control panel.

3. Connect both tie-ins together and enter the auxiliary building penthouse through the existing access opening. Both tie-ins are located close to the existing access

opening. However, the area below the penthouse is congested and routing the ductwork will be difficult. While existing ductwork may not have to be replaced, some ductwork will have to be removed to allow the new ductwork to be installed, then reinstalled. In addition instrument lines, hangers, and conduit will have to be redesigned or relocated. All new ductwork will have to be ductwork designed for 2 PSI positive and negative pressure. It will have to be tested for bubble tightness. This is to make the ductwork similar in construction to the existing purge system ductwork, as well as the WG-1 ductwork.

4. Run the duct through the penthouse and tie in to the existing WG-1 stack connection. There will be difficulties involved with working in the penthouse. The existing access to the penthouse from the surge tank room is partially blocked by ductwork. Any ductwork installed in the penthouse will have to be lifted through the access opening. This opening cannot be left open for long periods because the penthouse is maintained at a high positive pressure. Working in the penthouse with the access closed will necessitate air tanks or respirators for workers. The ductwork run through the penthouse will be as described previously. The tie-in of the new ductwork to the ductwork of WG-1 will be in the transition piece from the penthouse to the stack. A mating flange will have to be welded to the inside of the WG-1 connection, for installation of the new duct.

With this arrangement, the existing system's discharging into the stack will be unaffected by the operation of the WG-1 exhaust fans, since the new exhaust will be separate in the penthouse. When the new installed dampers are closed, the purge supply system can be run normally. The system would be run as follows:

1. Dampers 5128A, B, C, and D would be closed. (See Figure 1.)

2. Valves AH-V-1A, 1B, 2A, and 2B would be opened.

3. The purge supply systems would be off. Actions 1, 2 and 3 would be initiated from the control room.

4. Two of four WG-1 trains would operate for 60,000 CFM. The new isolation dampers would be opened. These actions would be initiated from the WG-1 control panel in the turbine building.

5. One or two existing purge exhaust trains would be operated for a total of 85,000 CFM or 110,000 CFM exhaust. These fans are started from the control room.

These modifications would allow the removal of the equipment hatch because the capture air velocity could be maintained into the reactor building. Hence, components too large to be taken in through the 9' diameter airlock #1 penetration could be brought into containment. Such a modification could provide a high degree of flexibility by placing a total of 110,000 CFM exhaust capacity at the discretion of operating personnel.

All concepts up to this point have assumed a 200 feet per minute capture velocity and the hatch open to atmosphere. While 200 feet per minute is standard for a laboratory, the acceptability of this velocity in an environment subject to wind forces would have to be investigated. A wind blowing directly at the opening would not cause concern. However, wind blowing across the opening may cause eddy currents which may draw particulates out of containment. Air curtains could be installed to negate the effects of wind.

In order to ease fears of an open containment building, a containment access building could be built. This would be a "Butler" type building, which would be kept at a positive pressure with respect to containment, and at a negative pressure with respect to atmosphere. This would be accomplished by a supply air unit supplying filtered, heated air to the access building. Air would enter the access building by means of the supply fan and by infiltration. Air would transfer to containment through the hatch opening. This arrangement would have several advantages:

1. There would be two boundaries between the containment and the outside.

2. Supply air would be filtered, reducing exhaust filter loading, and possibly airborne contamination.

3. Supply air would be heated for minimal temperature control.

4. A laydown area would be available to prepare equipment for shipment in a clean environment.

5. A health physics area would be available.

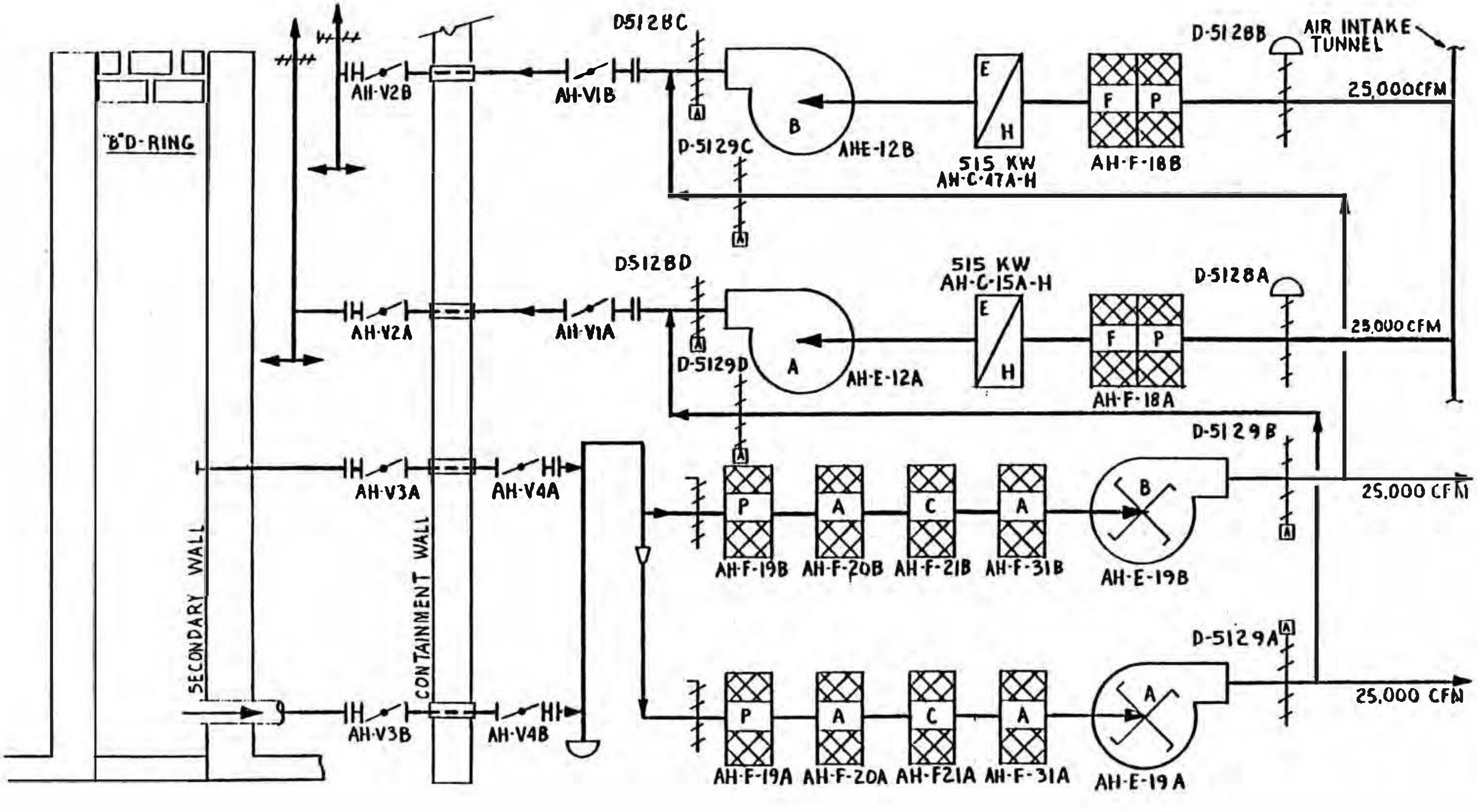


FIGURE 1

REACTOR BUILDING PURGE AND PURIFICATION SYSTEM

FIGURE 2

AIRBORNE PROFILES

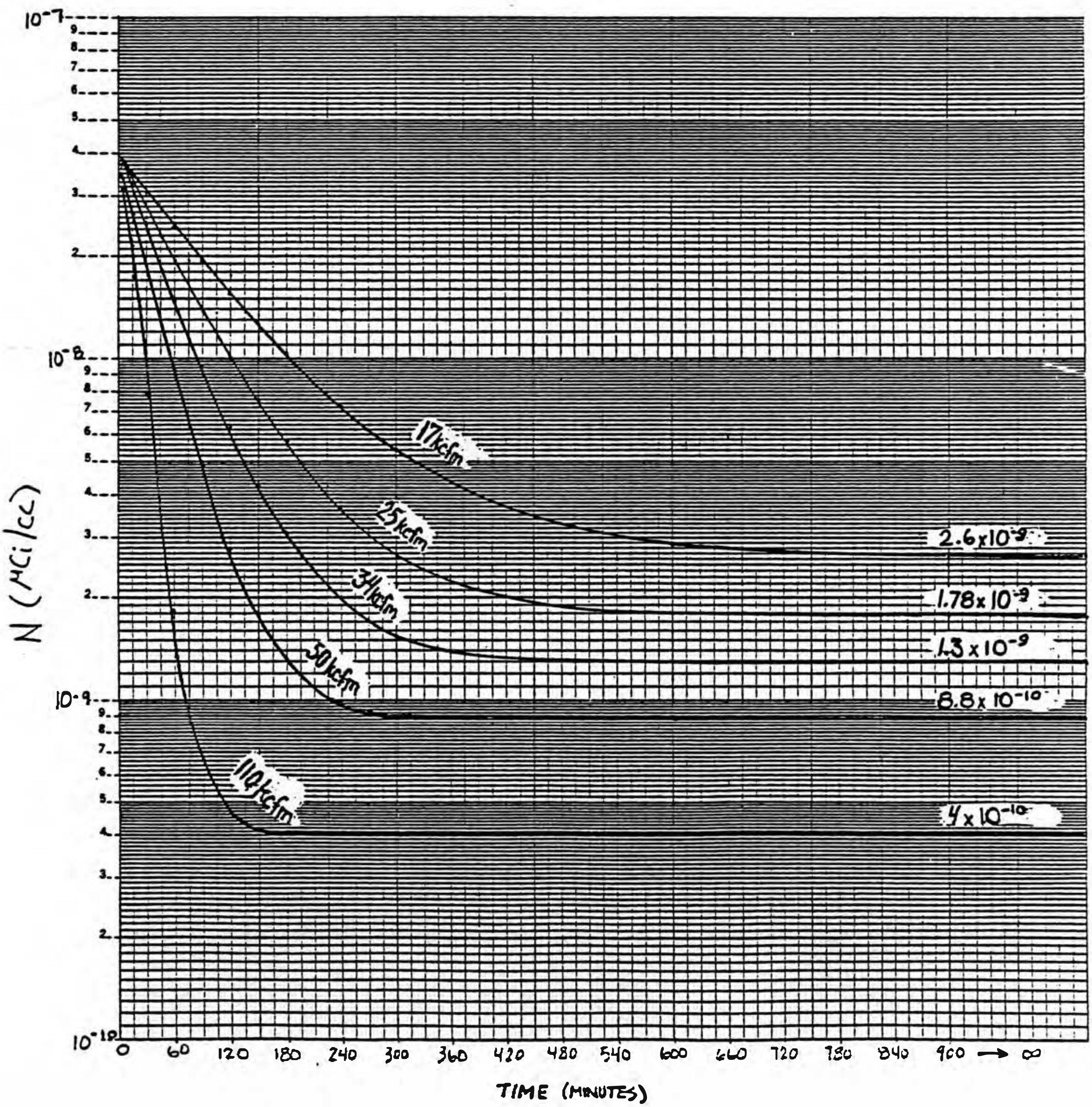


TABLE 1
SURFACE CESIUM BURDEN IN REACTOR BUILDING

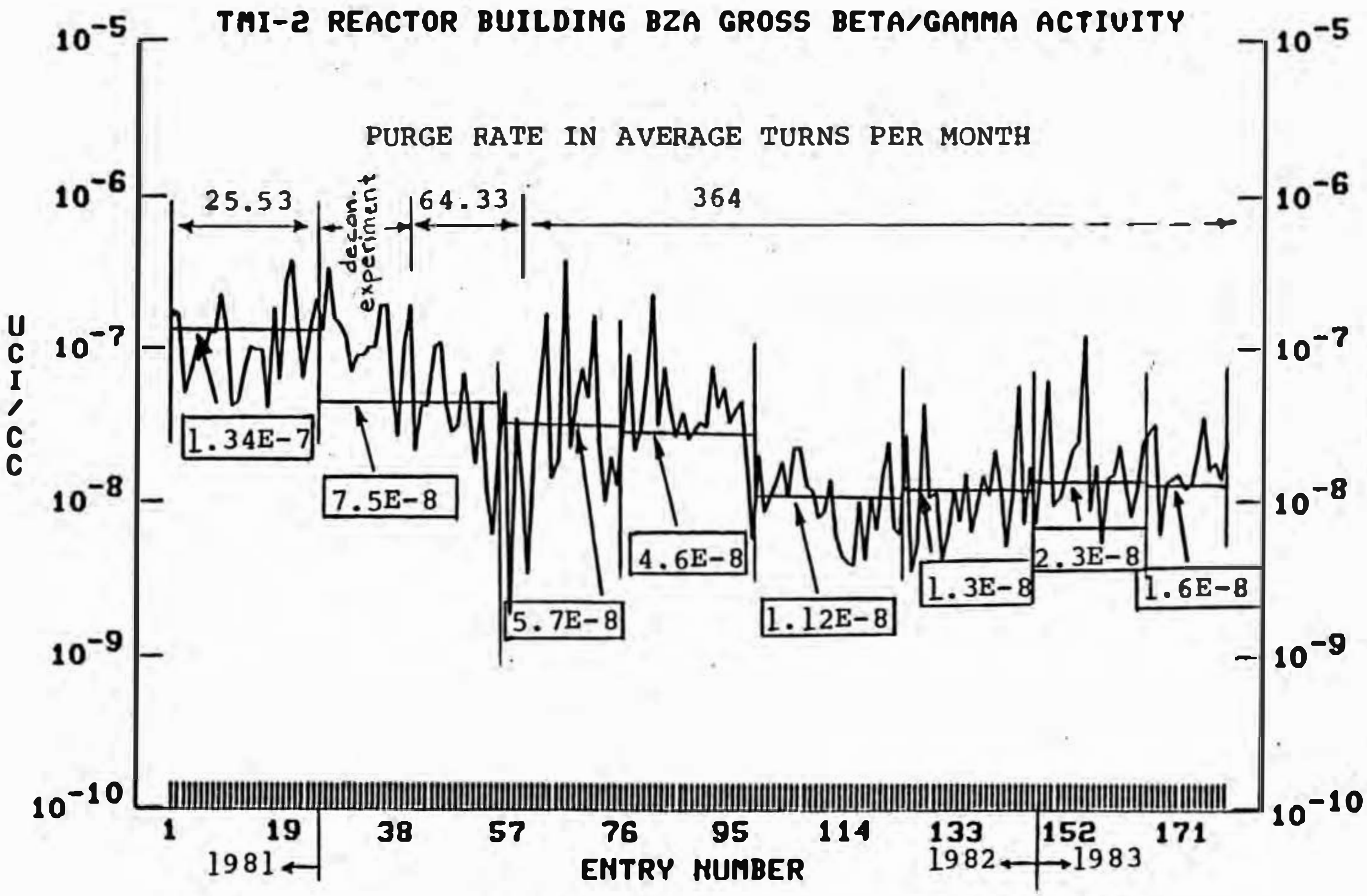
Location	Size	Concentration	Cesium Burden (Ci)	Remarks
Airborne:				
Hi Volume	2×10^{6} ft^3	2.97×10^{-9} uCi/cm^3	1.68×10^{-4}	Ref. 6
BZA	2×10^{6} ft^3	2.5×10^{-8} uCi/cm^3	1.42×10^{-3}	Ref. 6
Air Handling:				
Filters	80 ft^2	2.58×10^{-1} uCi/cm^2	9.58×10^{-4}	
Concrete Surfaces:				
EL 347'6"	2.08×10^{4} ft^2	.09 uCi/cm^2 (V)	4.9	Ref. 7
		.62 uCi/cm^2 (H)		
EL 305'	2.20×10^{4} ft^2	.50 uCi/cm^2(V)	12.47	Ref. 7
		.88 uCi/cm^2 (H)		
EL 282'6"	2.13×10^{4} ft^3	Various	2942	Ref. 9
TOTAL	-	-	2960.3	
Steel Surfaces:				
EL 347'6"	1.02×10^{4} ft^2	5.6×10^{-2} uCi/cm^2	9.8×10^{-4}	Ref. 8
	4.47×10^{4} ft^2	5.3×10^{-3} uCi/cm^2*		
EL 305'	1.94×10^{4} ft^2	2.4×10^{-1} uCi/cm^2	5.25	Ref. 8
EL 282'6"	1.13×10^{4} ft^2	3.0×10^{1} uCi/cm^2	377.5	Ref. 8
TOTAL	-	-	382.75	
Sludge	6.39×10^{2} ft^3	203.5 uCi/cm^2	3681.5	Ref. 10
Sump Water	3.067×10^{3} gal	12.483 uCi/cm^3	144.9	Ref. 10

*Above EL 370'

FIGURE 4

PREPARED BY: STEVEN R. FREY, TMI-2 RTS

February 28, 1983



SOLID LINE : AVERAGE BZA GROSS BETA/GAMMA ACTIVITY

ATTACHMENT 2

W.H. Hamilton memo dated April 5, 1983

TMI-2 Work Enclosure Cost Benefit
Analysis
Bruce Shiffler-Newport
News Industrial Corporation

TMI-2 WORK ENCLOSURE:

COST BENEFIT ANALYSIS

INTRODUCTION

A conceptual study was submitted in January for installation of a system of work enclosures in the TMI-2 Reactor Building to support the Reactor Disassembly and Defueling (RD&D) effort. Following this, in a letter to TAAG, dated January 31, 1983, Mr. Kanga requested additional information and asked several specific questions concerning the proposed system of enclosures. In response to this letter, we have prepared the following report. This report will:

1) Review the concept originally proposed and discuss modifications made based on additional information,
2) Review the RD&D effort including cost in Man-Hours, Man-Rem and Dollars,
3) Detail the estimated work enclosure costs and discuss the benefits,
4) Provide a Cost-Benefit Analysis of RD&D with the Enclosures installed,
5) Answer the specific questions asked in the January 31 letter to TAAG.

SUMMARY

Installation of the proposed system of enclosures in the TMI-2 Reactor Building for use during Reactor Disassembly and Defueling (RD&D) would eliminate the need for respirators and wet suits during most of that work, and the resulting increase in productivity would generate a calculated net savings of 7,748 Man-Hours, 554 Man-Rem and $2,479,000 (see Fig. 14) during the course of the RD&D effort. The analysis used to obtain these results was conservative, and therefore the actual savings is expected to be higher than that calculated.

DISCUSSION

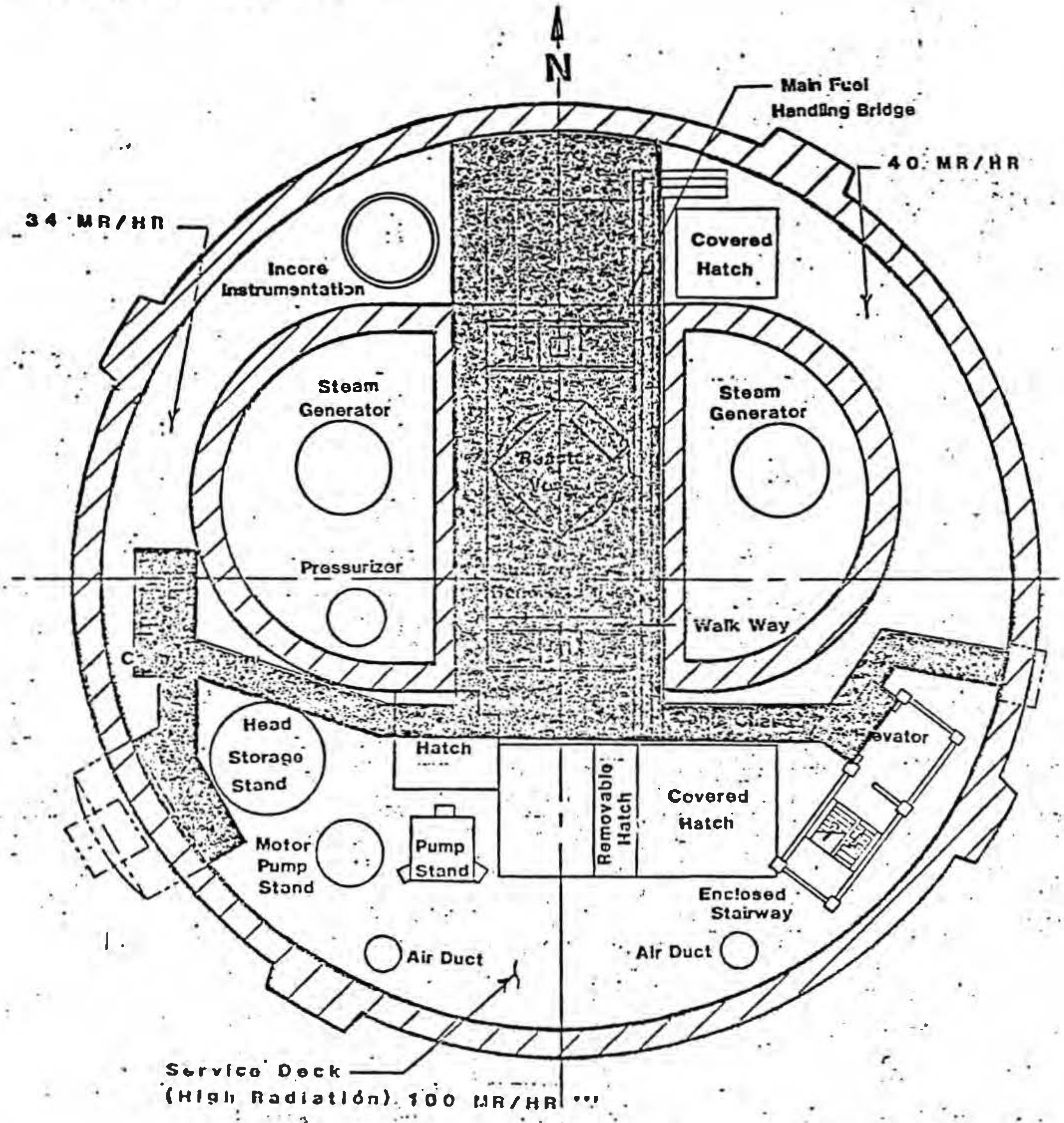
The original concept proposed is shown in Figure 1. These enclosures are designed to isolate personnel from airborne and surface contamination in the Reactor Building by providing covered access and egress routes and work areas and by covering building and equipment surfaces within the enclosures to the maximum extent practical. Traffic through the enclosures would be as follows:

1) Entrance on the 305 Level through Personnel Airlock #2 and up to the 347 level via the elevator,
2) From the Elevator on the 347 level, across the service deck to the enclosure in the canal area, (note that removal of a portion of the CRDM cable chase was assumed),
3) Entrance to the main enclosure, which completely encloses the canal area to the tops of the D-rings (including draping the walls) and extending to the North wall of the Reactor Building, through an anteroom at the South end of the D-rings (note that a hatch would be provided over the Reactor to provide access for the polar crane),
4) Exit from the work area would be through another anteroom, across the service deck and down the open stairway (also enclosed) and out personnel airlock #1 on the 305 level.

This system of enclosures would be air conditioned and maintained at a slight positive pressure. The stated objectives of this system of enclosures are the following:

1) Eliminate Respirators to the maximum extent possible,
2) Solve or reduce the problem of heat stress,
3) Focus the work effort with a managable work boundary,
4) Isolate the RD&D effort from other activities in the Reactor Building,
5) Help to minimize exposure.

*** Per 12/28/82 Survey



ORIGINAL CONCEPT

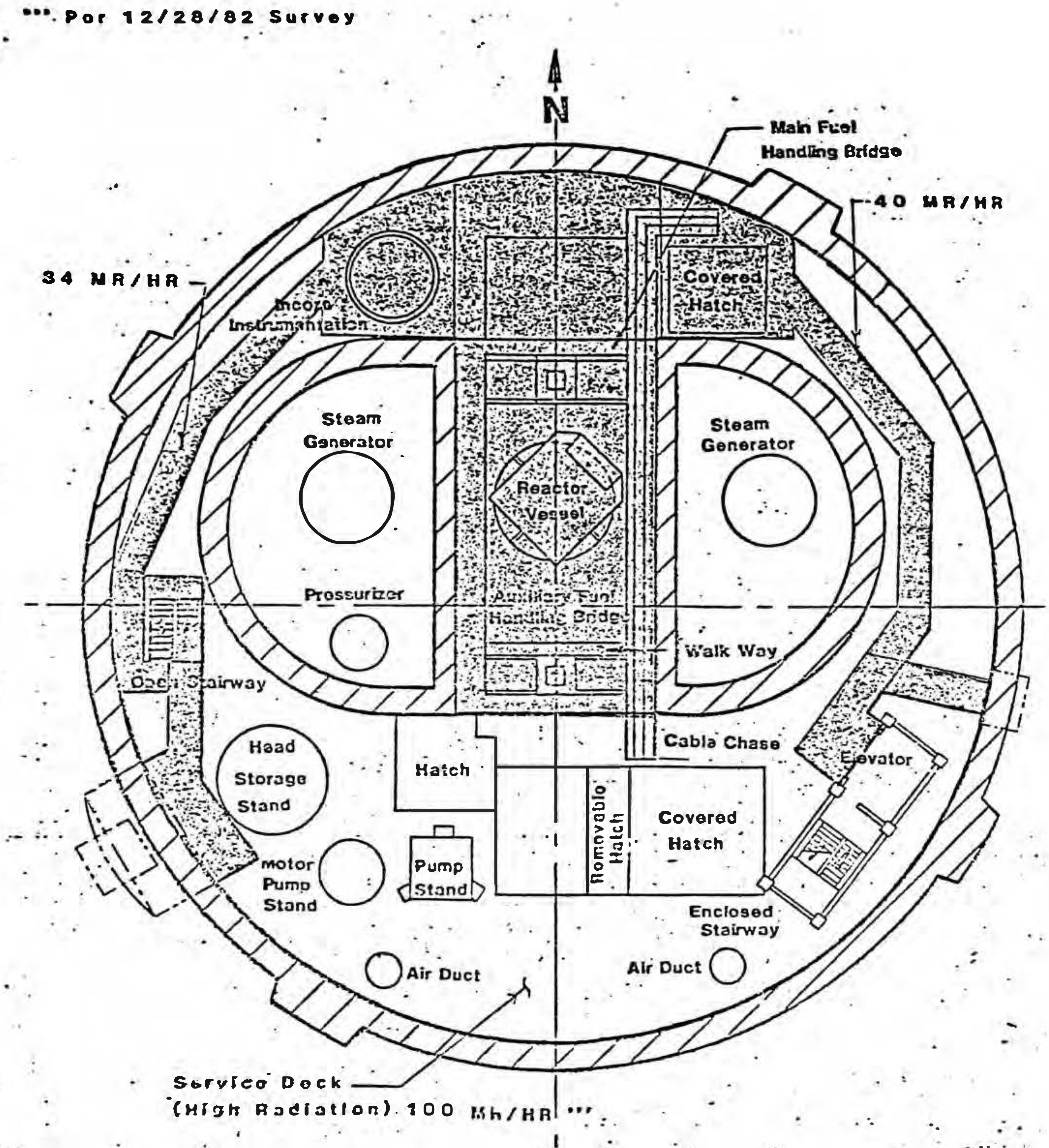
FIGURE 1

Since the date of the earlier submittal the concept has evolved based on additional information obtained including a 12/28/82 Radiological Survey. Figure 2 shows the configuration now proposed. Note that those parts of the system on the 305 level and the main enclosure between the D-Rings remain the same. Only the routing of tunnels and locations of anterooms on the 347 level have changed. These changes:

1) Route transit paths through the lowest radiation areas,(See Figure 2)
2) Make best use of available space by keeping the service deck and equipment hatch clear for equipment laydown and entry.
3) Use the least obstructed routes in and out of the canal area.

Shielding would be installed in the two anterooms located at the North end of the D-Rings. A cap covered with 2 inches of lead would be installed over the in-core instrument table to reduce the current dose rate of 1R/HR to approximately 50 mR/HR. Otherwise the two anterooms would be shielded with 3/8" of lead on the floors and up to 7 feet high on the walls. As in the original submittal, it is assumed that a portion of the CRDM cable chase can be removed. These two anterooms would provide space for laydown of small tools and a shielded waiting/rest area for personnel. It is also assumed that the elevator will be returned to service or that an alternate lift device will be provided. This is an extremely important consideration for stream-lining the movement of personnel and equipment between the 305 and 347 levels.

It was suggested in the original submittal that a partial or staged installation of the enclosures might be of benefit under certain circumstances. The major benefits (i.e. elimination of respirators and wetsuits) are only achievable, however, with the entire system installed, and we recommend that the entire system be installed rather than any separate part or parts of it.



PROPOSED CONCEPT

FIGURE 2

That then is the basic design we are proposing. The important features are:

1) Tunnels and shielded anterooms,

2) Enclosed, top-opening canal area,

3) Air conditioning,

4) Positive pressure ventilation,

5) Elevator or alternate mechanical lift.

The next major point is the scope and cost of the RD&D effort. Figure 3 is a summary of GPU's target cost factor for in-containment work during RD&D. It is based on an estimated 10:1 ratio of total craft hours to craft hours in-containment. The "total craft hours" includes the balance of a man's day prior to and after work in-containment as well as supporting craft work outside containment. Non-manual labor and distributable material (anti-c's, respirators, tools etc.) are estimated as functions of the total number of craft hours as indicated in Figure 3. The calculated factor is $375/in-containment hour. Note that this does <u>not</u> include engineering. The basis for the 10:1 ratio is GPU's target work schedule for the RD&D effort which is shown in Figure 4. Specifically, it calls for individuals to work 2, 2-hour stays in-containment per day, totalling 90 hrs. per day, five days-a-week. Note that this has not yet been achieved. In fact, the current ratio is approximately 25:1 or slightly higher. Based on Radcon's Reactor Building Entry Data, the average stay in-containment through February 16, 1983 has been 1 1/4 hrs. and just under 1-1/2 hrs. for entries in 1983 through that date; and at the rate of 1 entry per man per day. The 25:1 ratio has only recently been achieved and is still a quantum jump from 10:1.

GPU TARGET COST PER

IN-CONTAINMENT HOUR

(CRAFT AND CRAFT SUPPORT)

- TOTAL HOURS TO IN-CONTAINMENT HOURS (CRAFT), 10:1
  @ $25.50/HR.

- NON-MANUAL -- .25 X CRAFT HOURS
  @ $28/HR

- DISTRIBUTABLE MATERIALS -- $5/CRAFT HR

IN-CONTAINMENT HOURS X 10 ($25.50 + .25 X $28 + $5) =
$375/IN-CONTAINMENT HOUR*

* DOES NOT INCLUDE ENGINEERING

FIGURE 3

GPU TARGET WORK SCHEDULE -

2, 2-HOUR STAYS IN-CONTAINMENT/MAN/DAY

90 HOURS/DAY, 5 DAYS/WEEK

= 10:1 RATIO, $375/IN-CONTAINMENT HOUR

CURRENT CONDITIONS (THROUGH FEB. 16, 1983) -

ALL ENTRIES 1-1/4 HRS/ENTRY

1983 ENTRIES 1-1/2 HRS/ENTRY

1 ENTRY/DAY

= 25:1 RATIO, $938/IN-CONTAINMENT HOUR

FIGURE 4

Based on GPU's January, 1983 Program Reassessment, and the cost factor calculated above (Figure 3) a summary of the cost in Man-Hours, Man-Rem and $ of individual phases of the RD&D effort is shown in Figure 5. Note that the man-hours and dose rates accounted for in the Program Reassessment are for "selected" RD&D activities and may not include the total effort estimated for RD&D. Again, note that no engineering is included in these figures. The Man-Hours shown for each phase of the work were further subdivided in the Program Reassessment into hours at different dose rates. The figures shown are totals.

Having reviewed the work scope for which the enclosure are intended, let's look at the cost of the enclosures. Figure 6 shows a breakdown of the estimated dollar costs for the enclosures. The first two rows, "Design" and "Engineering" are based on Man-hours only. "Pre-Installation" is materials and fabrication, including A/C, HEPA'S, blowers and lighting as well as tenting material and scaffolding. "Installation" and "Removal" are in-containment hours including supervision. Note that "Installation" is the biggest item and that the sum of "Installation" and "Removal" makes up over half the cost of the entire system. "Shielding" is a material cost only, including 2" of lead for the in-core instrument cover and 3/8" for the floors and walls of the anterooms. "Maintenance" is based on shipyard experience for enclosures of this type. It was calculated as 10% of the initial installation labor costs per year for 4 years which is the estimated duration of the RD&D work scope. "Disposal" was originally calculated using estimated volumes of the tenting and scaffolding material when taken down, costed at rates provided by GPU for LSA boxes, shipment and burial. The results were then doubled to keep them conservative. Note that costs for enclosing the open stairway are included along with "West Tunnel", but that no cost is included for putting

PROGRAM REASSESSMENT

JANUARY, 1983

TABLE 4 (SUMMARY)

	(CONTAINMENT) MAN HOURS	MAN REM	$(K)
GENERAL RD&D PREPARATIONS	2278	444	$854
REACTOR VESSEL HEAD REMOVAL	2561	487	$960
FUEL TRANSFER CANAL MOD/PREP	5506	278	$2,065
PLENUM ASSEMBLY & FUEL REMOVAL	39462	276	$14,798
CORE SUPPORT ASSEMBLY REMOVAL	5440	101	$2,040
TOTAL	55247	4079	$20,717**

**BASED ON $375/IN-CONTAINMENT HOUR

TMI CONTAINMENT ENCLOSURE

COST BREAK-DOWN

	EAST TUNNEL & ANTEROOM		WEST TUNNEL, STAIRWAY, & ANTEROOM		CANAL AREA		
DESIGN	$4,994		$5,978		$3,528		
ENGINEERING	$3,842		$5,124		$3,024		
PRE-INSTALLATION	$30,160		$36,879		$56,046		
INSTALLATION	$111,949		$148,916		$87,885		
SHIELDING	$9,982		$13,488		-		
MAINTENANCE	$44,779		$59,566		$35,154		
REMOVAL	$54,000		$79,500		$45,000		
DISPOSAL	$10,000		$14,000		$12,000		
TOTAL	$269,706	+	$363,451	+	$236,637	=	$869,794

FIGURE 6

the elevator back in service. Nevertheless use of the elevator is considered to be extremely important for the RD&D effort, with or without the enclosures.

Comparing costs of Man-hours, Man-rem and $ for the Enclosures and for the RD&D effort (See Figure 7) we can see that costs for the enclosures are small particularly when compared to the benefits to be gained from their use. Let us next assess these benefits.

The first of the stated objectives of this proposed system of enclosures are "to get people out of respirators, to the maximum entent possible", and "to solve or reduce the heat stress problem". Currently, however, large penalties are being paid for the required use of respirators and wet suits which together are a major source of the heat stress problem. The first of these penalties is longer suit up time. This time is not in-containment, so it generates no man-rem, but it must be accounted for in the 10:1 ratio which is the basis of the $375/In-containment Hr. Second, is reduced time in containment. The work effort to date has been limited to a large extent by heat stress rather than radiation. The third is reduced worker productivity while in-containment due to physical discomfort, impaired vision and mobility as well as heat stress.

Figure 8 includes a list of the protective clothing which is now generally being worn, depending upon the work being done. The list has recently been stream-lined somewhat but it is still extensive. It should be noted that the effects of heat stress begin to accumulate, not when the individual goes into containment, but as soon as the clothing is put on. Installation of the proposed system of enclosures would eliminate the need for at least the wet suit, respirator and ice vest for most work. This would

TMI CONTAINMENT ENCLOSURE

COST SUMMARY

ITEM	IN-CONTAINMENT MANHOURS	MAN-REM	$1000
EAST TUNNEL	360	22	$ 270
WEST TUNNEL	460	48	$ 363
CANAL AREA	260	23	$ 237
TOTAL	1,080	93	$ 870 (1)
RD&D	55,250	4,080	$20,700 (2)

(1) INCLUDES ENGINEERING, DESIGN, MATERIALS, FABRICATION, SHIELDING, INSTALLATION (BASED ON $375/IN-CONTAINMENT HOUR), REMOVAL AND DISPOSAL

(2) BASED ON $375/IN-CONTAINMENT HOUR, DOES NOT INCLUDE ENGINEERING

FIGURE 7

PROTECTIVE CLOTHING

1) 1 SET ANTI-C'S
2) WET SUIT PANTS PLUS PLASTIC APRON OR WET SUIT SLEEVES
3) UP TO 7 TLD'S
4) RESPIRATOR (MUST BE SMOKE TESTED)
5) ICE VEST
6) 3-4 PAIRS SHOE COVERINGS
7) 3-4 PAIRS GLOVES

PENALTIES: SUIT-UP TIME + PREPARATION, SUIT-UP ASSISTANCE, SOLID WASTE

FIGURE 8

reduce the suit-up time for the individual going into containment and also, as noted in Figure 8, reduce: 1) the time required to prepare this equipment for use, 2) the time spent by individuals assisting during suit-up and 3) the solid waste generated by their disposal. All of these are included in the real cost of time in-containment, although solid waste disposal is not included in the $375/In-containment Hr factor used to compute GPU's costs for this analysis.

"Stay time" in-containment is the second point listed under penalties paid for wearing respirators and wet suits. Figure 9 shows the stay times recommended under various conditions by GPU's Industrial Safety Computer Program. These stay times are currently being extended by use of ice vests, but the weight penalty for the ice vest is considerable, and it doesn't go away when the ice melts. A particularly dangerous aspect of heat stress is the sudden onset of the dizziness and weak-knees which result when limits are exceeded. Even experienced people can experience these effects if they lose track of time. By examining Figure 9 we can see that installing the proposed system of enclosures, so that most of the work can be done wearing only anti-C's, will extend the stay time limits up to and beyond GPU's target 2-hr work period, even without air-conditioning.

The third penalty resulting from wearing respirators and wet suits is reduced worker productivity within the time limits imposed by heat stress. The reduction in productivity results from physical discomfort and impaired vision and mobility associated with respirators and wet suits. Based on shipyard and industry experience, the minimum expected loss in productivity for donning respirators and wet suits, is 33%. Conversely, removal will result in a 50% increase over productivity while wearing them.

GPU INDUSTRIAL SAFETY STANDARD

	STAY TIME (HRS)			
	80°	80°-90°	90-100°	100°-110°
ANTI-C'S	NO LIMIT	3	2	1
ANTI-C'S + RESP	4	2	1.5	1
ANTI-C'S + WETSUIT		1.5	1	.5
ANTI-C'S + RESP + WETSUIT*	1	.75	.5	.25

* GPU TARGET - 2 HRS

FIGURE 9

Improvements in these three factors: Suit-up Time, Stay Time and Productivity would contribute significantly to meeting or reducing the man-hour, man-rem and $ budgets for RD&D, but an additional factor which has even greater potential for savings is the ability to avoid delays. Installation of the propsed system of work enclosures will allow improved controls to be implemented which will avoid delays in completion of the RD&D effort. As diagrammed in Figure 10, improved controls include:

1. Focusing the work effort with a manageable work boundary
2. Isolating the RD&D work from other in-containment activities
3. Better communication in the work area using normal voice communication without radios or shouting through repsirators
4. Better communciation between the actual in-containment effort and outside support ,including engineering, by allowing convenient access to the work site for "gofers" and engineers and others who must clearly grasp the situation in-containment.

Figure 11 summarizes the potential gains to be had from implementation of the proposed system. For the first factor, "Suit-up Time", no quantitative gain has been identified, but recall that included here are support man-hours and solid waste disposal in addition to time spent by the individual actually going into containment (See Figure 8)

For the second factor, "Time In-Containment," a 50% gain is claimed. Note that according to GPU's Industrial Safety recommendations (Fig. 9), there is a 4-fold (400%) increase allowed in stay time if both respirators and wet suits can be removed. However, 400% exceeds the increase that would normally be utilized under GPU's target 4-hour in-containment work day schedule, and

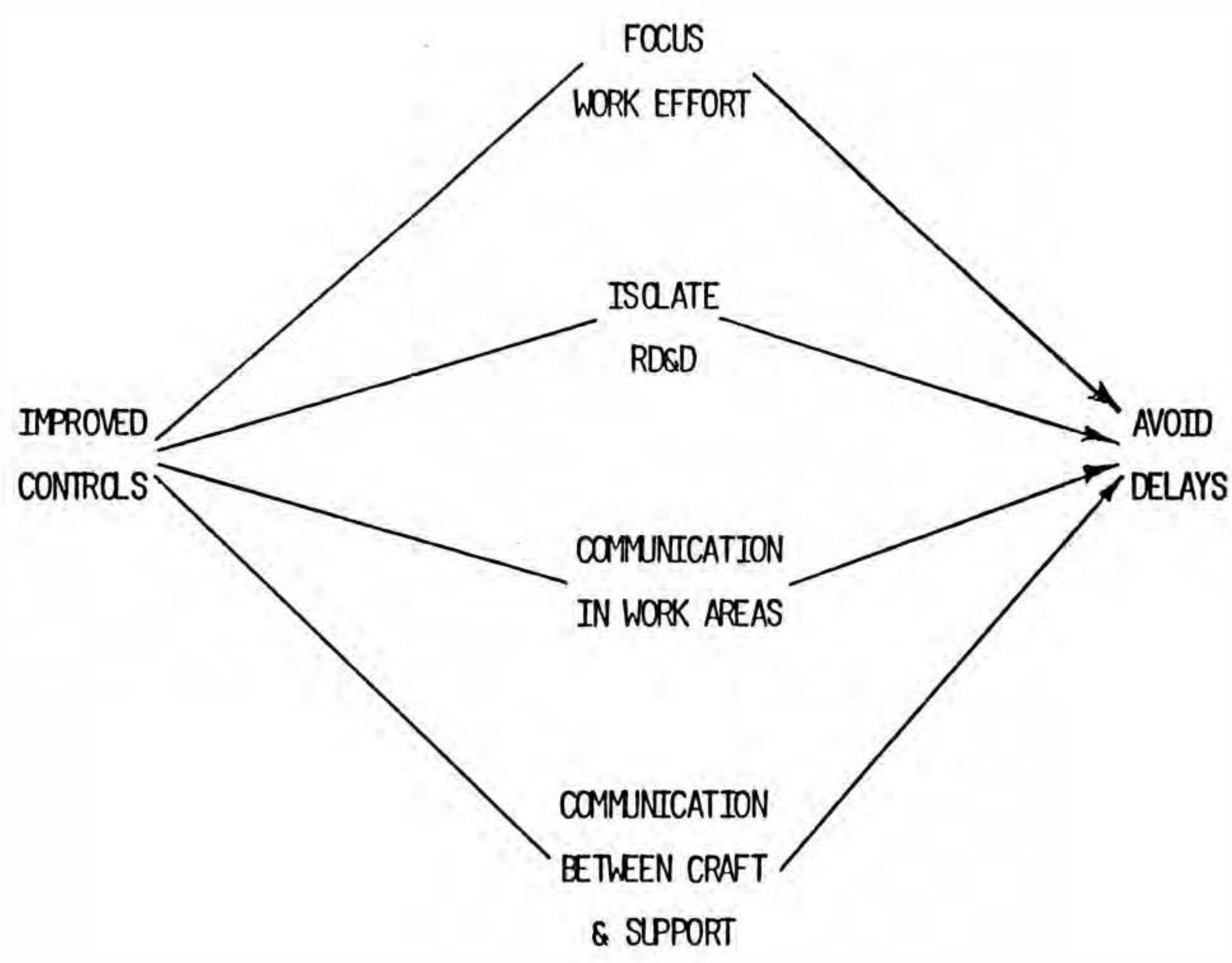

FOCUS
WORK EFFORT
ISOLATE
RD&D
IMPROVED
CONTROLS
AVOID
DELAYS
COMMUNICATION
IN WORK AREAS
COMMUNICATION
BETWEEN CRAFT
& SUPPORT

FIGURE 10

POTENTIAL GAINS

1) SUIT-UP TIME (?)
2) TIME IN-CONTAINMENT (50%)
3) PRODUCTIVITY IN-CONTAINMENT (50%)
4) <u>AVOID</u> <u>DELAYS</u> (?)

USE 15% OVERALL GAIN

FIGURE 11

some of the required gain has already been achieved using ice vests. Therefore a gain of only 50% is claimed, although it is believed to be conservative.

For the third factor, "Productivity In-Containment," the 50% gain referred to for removal of respirators and wetsuits is claimed. Note that this is the minimum gain expected based on industry and shipyard experience.

For the forth factor, "Avoiding Delays" no quantitative gain is identified, but, again, this factor has the greatest potential for savings. Particularly in the "gold fish bowl" environment of TMI the impacts of mistakes and delays are magnified far beyond their actual practical effects.

Having reviewed individual factors contributing to potential gains for installation of the system of enclosures, we must sum these individual factors into a single "Gain Factor" to be used for the cost-benefit analysis. All of the factors identified cannot be applied uniformly to the 3 components of cost (man-hours, man-rem and $) used in the analysis. In addition, all of the work identified as part of RD&D (55,247 man-hours in-containment) may not be able to be done inside the enclosures, therefore the Gain Factor cannot be applied to all of the RD&D cost. For these reasons, and to insure that the results of the cost-benefit analysis are conservative, a Gain Factor of 15% will be used instead of the 50%, 100% or more which might otherwise be deduced from the data presented.

This factor of 15% is used in the cost-benefit analysis shown in Figure 12. The first line, RD&D, is a summary of GPU's estimates from the January,

COST-BENEFIT ANALYSIS

	CONTAINMENT MAN-HOURS	MANREM	$(K)
RD&D	55,250	4080	$20,700 [1]
15% GAIN	8,288	612	$ 3,105
ENCLOSURES	1080	93	$ 870 [2]
NET SAVINGS	7,208	519	$ 2,235

(1) BASED ON $375/IN-CONTAINMENT HOUR, DOES NOT INCLUDE ENGINEERING.

(2) INCLUDES MATERIALS, ENG, INSTALLATION (BASED ON $375/IN-CONTAINMENT HOUR), REMOVAL, ETC.

FIGURE 12

83 Program Reassement (plus the cost estimate) which were detailed in Figure 5. The second line is the result of applying the 15% Gain Factor to each of the RD&D items in the first line. The third line is a summary of all the costs associated with the enclosures which were detailed in Figures 6&7. The "bottom line" is the difference between lines 2 and 3, the potential gains and costs of the enclosures respectively. The Net Savings shown are considerable, particularly in light of the conservatism built into them. 7200 Man-hours makes a total of 16 weeks at the target rate of 90 hrs/day. The Man-rem savings, 500-plus, and the dollar savings, $2-1/4 million, are certainly sufficient justification for implementation of the proposed system of enclosures and would be even if reduced by half.

The enclosures on which this analysis was based were the industry standard "tents" which most people would visualize when referring to the subject. There is, however, another design available which we believe offers considerable advantages for this application. It is a modular, stainless-steel construction which offers in particular, ease of installation and durability. Although it does not lend itself well to enclosing the open stairway or the canal area, this design is very well suited for construction of the tunnels and anterooms. Figure 13, shows a cost comparison of tent (Facilon) and stainless steel for construction of the tunnels and anterooms in the proposed system of enclosure. In both cases, the cost of enclosing the open stairway with Facilon is included. Identical costs are assumed for Design, Engineering and Shielding. For Pre-installation (materials and fabrication) the cost for stainless steel is much higher as might be expected, but for Installation and Removal, which are composed entirely of in-containment hours and make-up the largest item for the enclosures, a

COST COMPARISON
"TENT" VS. STAINLESS STEEL ENCLOSURES
(TUNNELS & ANTEROOMS ONLY)

	FACILON MATERIAL	STAINLESS STEEL PANELS
DESIGN	$ 10,972.00	$ 10,972.00
ENGINEERING	$ 8,976.00	$ 8,976.00
PREINSTALLATION	$ 67,039.00	$ 95,454.00
INSTALLATION	$260,865.00	$156,044.00
SHIELDING	$ 23,470.00	$ 23,470.00
REMOVAL	$133,500.00	$ 25,876.00
DISPOSAL	$ 24,000.00	$ 6,000.00
MAINTENANCE	$104,345.00	$ 62,417.00
TOTAL	$633,167.00	$389,209.00

FIGURE 13

substantial savings is expected from use of the stainless steel construction. Disposal and Maintenance are also considerably less using the stainless steel. Because of their durability and the ease with which they can be cleaned up, they are well suited for disassembly and re-use in different locations and configurations. Therefore, they need not be disposed of. The $6,000 for Disposal under the stainless ateel heading is mainly for the Facilon used to enclose the open stairway. The maintenance costs for both types of construction were calculated as being 10% of the initial installation labor cost per year for 4 years, although this is probably too high for the stainless steel construction. The net result is approximately a quarter of a million dollars saved using the stainless steel. Figure 14 shows a cost-benefit analysis, the same as Figure 12, except that the stainless steel construction is substituted for tent construction for the tunnels and anterooms. Substantially bigger savings would be gained across the board using the stainless steel construction.

The last point to be addressed in this report is the specific questions asked in the January 31 letter from Mr. Kanga to TAAG. These questions are reprinted with answers in Figures 15-20.

COST-BENEFIT ANALYSIS

(STAINLESS STEEL TUNNELS & ANTEROOMS)

	CONTAINMENT MAN-HOURS	MANREM	$(K)
RD&D	55,250	4080	$20,700 [1]
15% GAIN	8,288	612	$ 3,105
ENCLOSURES	540	58	$ 626 [2]
NET SAVINGS	7,748	554	$ 2,479

(1) BASED ON $375/IN-CONTAINMENT HOUR. DOES NOT INCLUDE ENGINEERING.

(2) INCLUDES MATERIALS, ENG, INSTALLATION (BASED ON $375/IN-CONTAINMENT HOUR), REMOVAL, ETC.

FIGURE 14

1. A REALISTIC AND <u>COMPLETE</u> COST ESTIMATE IS NEEDED. MANHOURS AND DOLLARS ASSOCIATED WITH ENGINEERING (INCLUDING PROCUREMENT, SOFTWARE, LICENSING) MATERIAL, INSTALLATION, OPERATION AND MAINTENANCE, DISMANTLING, REMOVAL, AND DISPOSAL MUST BE INCLUDED.

- SEE FIG. 6

FIGURE 15

2. A TRUE MAN-REM BALANCE (COST VS. BENEFIT) MUST BE DEVELOPED. ON THE COST SIDE, MAN-REM ASSOCIATED WITH EACH OF THE COST CATEGORIES IN 1 ABOVE, SHOULD BE INCLUDED. ON THE BENEFIT SIDE, A QUANTITATIVE ESTIMATE IS NEEDED OF THE MAN-REM REDUCTION WHICH COULD BE ACHIEVED.

ON THIS POINT, WE FEEL IT IS IMPORTANT TO BREAK DOWN THE COSTS AND BENEFITS INTO THE MAJOR ELEMENTS OF THE PROPOSED CONCEPTS. SPECIFICALLY:

- SHIELDING (COST VS. MAN-REM REDUCTION DUE TO DECREASED RADIATION DOSE RATES)
- TRANSIT-PATHWAY
- ENCLOSED WORK AREA
- ENCLOSED, SHIELDED REST AREA

(COST VS. MAN-REM REDUCTION DUE TO IMPROVED EFFICIENCY IN THESE AREAS)

- SEE FIG. 7 & 12

FIGURE 16

3. WE NEED A MORE COMPLETE EVALUATION OF THE EXPERIENCE BASE, AMONG SHIPYARDS OR OTHERS, WITH ENCLOSURES OF THE TYPE RECOMMENDED. SPECIFIC QUESTIONS WHICH COME TO MIND ARE:

A) HAVE ENCLOSURES LIKE THIS BEEN USED EFFECTIVELY IN HIGH RADIATION AREAS, AND WITH WHAT RESULTS? (THE ADVANTAGES OF SUCH AN APPROACH IN LOW RADIATION BUT CONTAMINATED AND DIRTY ENVIRONMENTS IS EVIDENT, BUT THIS IS A DIFFERENT SITUATION.)

- YES, CONTAINMENTS OF THIS TYPE HAVE BEEN USED SUCCESSFULLY AT NEWPORT NEWS SHIPBUILDING TO PROVIDE LOCALIZED WORK AREAS IN REACTOR COMPARTMENTS OF MORE THAN 25 SUBMARINES, AS WELL AS NUMEROUS SURFACE SHIPS AND LAND-BASED PROTOTYPES, DURING DEFUELING OPERATIONS. THESE APPLICATIONS INVOLVE HIGH RADIATION AREAS WHERE THE POTENTIAL FOR SPREAD OF CONTAMINATION IS GREAT. THEY HAVE ALSO BEEN USED WITH POSITIVE, INSTEAD OF NEGATIVE, PRESSURE VENTILATION TO ESTABLISH CLEAN ROOM ENVIRONMENTS FOR SOME OPERATIONS.

FIGURE 17

3 B) HAVE SUCH ENCLOSURES BEEN USED EFFECTIVELY FOR EXTENDED PERIODS? THE TMI-2 ENCLOSURE WOULD BE IN-PLACE FOR SEVERAL YEARS.

- YES, TENT ENCLOSURES HAVE BEEN IN USE, EVEN OUTDOORS, FOR 2 YEARS OR MORE. THE STAINLESS STEEL DESIGN WAS ORIGINALLY INTENDED FOR USE ON CONSTRUCTION SITES. AN ENCLOSURE OF THIS TYPE IS CURRENTLY BEING INSTALLED OUTSIDE OF AIRLOCK #1.

FIGURE 18

3 C) WHAT IS THE MAINTENANCE HISTORY ON THESE ENCLOSURES?

MAINTENANCE WORK IN THE TMI-2 CASE WOULD BE DONE IN A HIGH RADIATION ENVIRONMENT.

- THE FACTOR USED IN THE ANALYSIS WAS 10% OF INITIAL INSTALLATION LABOR PER YEAR. THIS IS A CONSERVATIVE ESTIMATE BASED ON SHIPYARD EXPERIENCE. THE SAME FACTOR WAS USED FOR THE STAINLESS STEEL TYPE, BUT THIS IS PROBABLY MUCH TOO HIGH.

FIGURE 19

3 D) WHAT PROBLEMS (AND SOLUTIONS) HAVE BEEN ENCOUNTERED WITH RESPECT TO CONDENSATION INSIDE THE ENCLOSURES?

- NO SUCH PROBLEMS HAVE BEEN ENCOUNTERED. IF THE ENCLOSURES ARE AIR CONDITIONED, CONDENSATION, IF ANY, WOULD OCCUR ON THE OUTSIDE.

FIGURE 20

VII. Review of Fuel Transfer System Design

TAAG was requested by GPU (Kanga letter 4540-82-0039 dated Dec. 3, 1982 and Devine's letter 4500-83-0199 dated March 3, 1983) to perform a parallel review of various items pertaining to the fuel transfer system while Design Engineering proceeded to establish the engineering baseline for the fuel handling system modifications. TAAG's reponse to the various items are noted below:

Item 1: "Provide recommendation on whether one or both mechanisms should be modified"

TAAG's Response: "In order to provide a high degree of reliability, both mechanisms should be modified"

Item 2: "Review the following Bechtel documents on the fuel transfer system - "Functional Requirements" and "Technical Scoope". Please provide endorsement and comments."

TAAG's Response: "Functional Requirements"

1. TAAG's comments on B&W's proposed modifications (contained in TAAG's fourth report) are applicable and should be incorporated as appropriate.

2. The functional requirements should provide a worst case scenario for the vendor to work with.

3. The functional requirements do not include the specific operational and environmental conditions that will be experienced during defueling of TMI-2 (e.g. number of cycles with defueling loads, water chemistry including increased boron and possible higher than normal particulate concentration, duration of immersion in this water chemistry). Such conditions are not obviously within the range of normal 40 year life conditions.

4. With respect to paragraph 4, this requirement could be read to mean that continued use of a modified chain/sprocket drive system is acceptable. Based upon work already done, the requirement should be to replace the chain/sprocket drive with a design that is reliable and not subject to the problems experienced on the chain/sprocket system.

5. Add at the end of last sentence in paragraph 7 "for the removal/installation of all active components".

6. The Functional Requirements should identify that the hydraulic system should use borated water (3500 ppm) or should have a limited volume of hydraulic fluid.

7. Item 11g should indicate that this requirement assumes that an operator or equipment error occurs after the loss of power or switch failure.

8. It is not apparent why it is necessary to design this system to loss of coolant accident conditions. Radiation dose requirements should be that exposure which will be received during normal non-accident conditions over the projected design life.

Fuel Transfer System Modification Technical Specification Rev. A (Specification 15737-2-R-180)

1. Para 4.2.2.2 Operating Instructions

   These instruction should cover at least the first steps to be taken under abnormal or emergency conditions, e.g. if binding occurs, cable breakage, etc.

2. Appendix A

Para 6 - ALARA considerations should be included in maintenance activities as well as installation and operation.

There should be an additional paragraph on operating hydraulic systems. Additionally, if pure water is used, the volume available must be limted and controlled due to boron dilution or water must meet the pool chemistry conditions.

Para 11.d - Limit switches or indicators should be provided to indicate "abnormal" operating condition, e.g. normal carriage cable tension is approximately 150 pounds yet design calls for the limit switch to be set at approximately 3000 pounds (just below breakage). Indications of binding or other problems should be provided.

3. As indicated via reference 3, the specification should indicate that both mechanisms should be modified.

4. It is not apparent that fuel transfer mechanism comments 5, 7, 8 and 10 submitted via the fourth TAAG Report relative to asymetrical loading, limit switches, transfer tube flanges, and embedment integrity of the mechanism rails have been included. These should be evaluated for any necessary changes.

Item 3: "Review the basic B&W proposed plan for Fuel Transfer Tube Modification - RDV - 450101 and endorse as acceptable and proper for defueling TMI-2."

TAAG's Response: TAAG endorses the cable carriage drive (B&W's proposed modification) and provides GPU with the following additional comments:

1. Use of a winch cable drive system will provide a more reliable and less complicated system to move the transfer carriage.

2. The winch should be mounted such that the center line of the cable drums are 90 degrees to the center line of the transfer system as shown on Figures 3 and 4 of reference (b), unless engineering evaluations show that other options are satisfactory.

3. Reeving of the cable should be done so as to ensure the cables are taut.

4. Sheave guards should be provided to prevent the cable from running off the sheave.

5. Additional comments were included in the "Fourth Report of TAAG".

Item 4: "Recommend a test plan on the modified equipment that will assure its long time dependability".

TAAG's Response: Final test plan approval should be deferred pending completion of the system design. Operating components should be selected during system design based on available test results that show dependability under the conditions that are expected during fuel transfer. Vendors should recommend what life cycle testing is required considering their long term field experience with the equipment. In some cases replaceability of components may preclude the need for life cycle testing.

The test plan should include, but not be limited to, performing a reliability operational check of the system utilizing a test load equal to the load to be transported via the fuel transfer system. (See Appendix C of spc. 15737-2-R-180 for Bechtels check-out, testing requirements).

The modified transfer system shall be tested as a minimum, including the following specific system capabilities and features:

a. Emergency carriage retrieval from the reactor building (RB) to the fuel handling building (FHB) should the drive system fail.
b. Emergency basket uprighting/lowering capability should the upender system fail.
c. Drive system cable tensioning method to prevent excessive cable slack. Cable sheaves, pulleys, etc. to retain the capability should the tensioning system fail (if a cable drive system is used).
d. Appropriate control interlocks and limit switches to prevent equipment damage due to operator error (e.g., an interlock to prevent carriage motion with the basket in an upright position).
e. Appropriate control panel indications for remote verification of equipment position (e.g., carriage in FHB, basket uprighted, etc.).
f. Position indicators on the carriage/track suitable for visual confirmation of the carriage position from the RB and FHB operating floor elevations through the canal/pool water.
g. Appropriate fail-safe features to prevent equipment damage due to a power or limit switch failure.
h. Method(s) to minimize the buildup of contamination on above-water drive system components and to minimize the spread of contamination to the FHB atmosphere from these components (e.g., washing systems or enclosures).
i. Maximize the allowable track misalignment while retaining operational reliability of the transfer carriage travel.
j. Threaded fasteners used on underwater installed equipment shall be checked to ensure later removability.

<u>Item 5</u>: "Recommend upender modifications or alternatives to accommodate assymmetrically loaded fuel canisters.

<u>TAAG's Response:</u> It is understood that B&W is now investigating several alternatives to accommodate assymetrically loaded canisters. TAAG has no additional recommendations.

Item 6: Review compatibility of all system components to withstand long term exposure to the borated water in both the canal and the storage pool. This compatibility should include bearing materials and pivot materials in the upender system.

TAAG's Response: Since the design of the equipment is not yet completed, a review of individual components does not seem appropriate at this time. Rather a component by component review can be done when drawings are made. In the meantime the following guidelines can be used in selection of materials:

> The borated water will have very little affect on carbon steel or stainless steel. It will have some effect on nickel base alloys and copper base alloys. It will have a deterimental effect on aluminum alloys. It is also noted that electrolytic characteristics of the components must also be considered.

These guidelines are based on limited research in the following references:

1. Course 14 corrosion, by American Society of metals.

2. Corrosion as Prevention in Water
   Butler & Ison
   Published by Reinhold Publishing Company 1966

3. Milton & Roy Bulletin 200.001 D Material Selection Chart

4. Chemical Resistance of Construction Material
   Bulletin L-907 by Chemical Flow Corporation

VIII. Evaluation of TMI-2 Quick Scan Results

described

TAAG evaluated the results of the Quick Scan experiment which was performed on October 16, 1982. The results of that evaluation are contained in attachments VIII - One and VIII - Two, which are attached to this section of the report. Overall conclusions and recommendations resulting from the evaluation are summarized as follows.

a. Overall Conclusions

(1) The dose rates measured during Quick Scan are most likely due to surface contamination of both vertical and horizontal metal surfaces, rather than to significant core debris on the plenum assembly top cover.

(2) This is consistent with TV inspections during Quick Look of the upper region of the control rod guide assembly, and also with the results of removed leadscrew piece examinations.

(3) The surface contamination is most likely a tightly adherent cesium contamination in either the oxide film or bare metal. The preliminary results of a leadscrew piece evaluation indicate that the cesium is most likely contained in the metal oxide film, and that there has been no intergrannular attack of bare metal.

S

(4) The dose rates from plenum assembly components once the reactor vessel head is removed will be higher than estimated for head removal planning purposes. For example, it was assumed for planning that the dose rate at a point four feet beyond the inside diameter of the reactor vessel and five feet above the flange is 8.5 R/hr. Based on the Quick Scan results, this dose is estimated to be about 40 R/hr. This dose rate can be defined with greater certainty once the results fo Quick Scan 2 measurements become available.

of

(5) The dose rates from the head itself appear to be on the same order of magnitude as currently experienced at other plants. The head itself can likely be handled using procedures currently planned.

b. Recommendations

The following steps are recommended:

- Perform various metallographic and radiological examinations of the leadscrew pieces removed - the core positions B-8, E-9, and H-8.

- Perform further under-the-head examinations to provide further confirmation of the cause and magnitude of dose rates.

- Evaluate the impact on currently planned head removal operations of the higher dose rates measured during Quick Scan, particularly during the short period of time between reactor vessel head removal and filling the indexing fixture with water.

  It is recommended, in addition, that a method be developed to permit the underhead examination to be performed independently of the status of the polar crane, so that these examinations can proceed in spite of the "hold" which has currently been place on operations involving the crane. Specifically, it is recommended that:

  - a tool be obtained for cutting a leadscrew support tube underneath the reactor vessel head, so that access can be obtained to regions above the plenum assembly cover plate without removing a control rod drive mechanism.

- A tool of this type which does not require a reactive force for cutting operations be procured, e.g., the plasma arc cutter which proof-of-principle tests at R&W showed to be promising. This type of tool would likely be very useful in subsequent defueling operations.

ATTACHMENT VIII-ONE

MPR ASSOCIATES, INC.

January 25, 1983

Mr. W. H. Hamilton
Consulting Engineer
Post Office Box 613
Ligonier, PA 15658

Subject: TMI-2 Quick Scan Results

Dear Mr. Hamilton:

Enclosed is a report we prepared for the Technical Assistance and Advisory Group (TAAG) covering evaluations of the TMI-2 Quick Scan Results. This report contains a few revisions to the draft report transmitted to TAAG members at the meeting on January 10-11, 1983.

The major change is that we have included a new section (Page 11) to estimate the doses which would be received from plenum assembly components once the reactor vessel head is removed. This estimate assumes surface contamination on plenum components equal to that inferred from the Quick Scan measurements. This dose rate, at a distance 4 feet beyond the vessel flange and 5 feet above the flange is about 75 R/hr, compared to the planning basis dose of 8.5 R/hr. Accordingly, current head removal plans will have to be re-evaluated to determine if changes are needed to accommodate the higher dose rates.

As discussed at the last TAAG meeting, the results of our evaluations differ in some respects from the preliminary evaluation results of other organizations. These differences need to be resolved. They may be summarized as follows:

- We, as well as GPUN personnel, have concluded that dose rates measured during Quick Scan are most likely due to general surface contamination of stainless steel surfaces, including vertical surfaces as well as horizontal ones. We understand that the preliminary Bechtel evluations indicate, on the other hand, that the dose rates are due to a radiation source on the plenum

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assembly cover plate equivalent to having the debris from five fuel assemblies located on the plate.

- Our evaluation indicates that the dose rate to be expected four feet beyond the inside of the reactor vessel flange and five feet above the flange once the reactor vessel head is removed is about 75 R/hr. The preliminary Bechtel evaluations indicate that the dose rate at this location was about 255 R/hr.

- Our evaluation indicates that the dose rate underneath the reactor vessel head, once the head is sitting on its storage stand, is in the range of 35 to 185 R/hr at a point 18 inches below the centermost leadscrew support tube. The preliminary Bechtel evaluation indicates that the dose rate at this location is about 400 to 600 R/hr.

We also understand that some consideration is being given to repeating the Quick Scan measurements with the reactor water level lowered to the elevation of the plenum assembly cover plate. We have estimated the dose rates for such a measurement, in order avoid having a major suprise if the measurements are taken without water. In particular, we would expect the dose rates to increase significantly if the water level is lowered, because water would no longer be available for shielding. For example, a dose rate of about 550 R/hr was measured at the elevation of the plenum assembly cover plate during Quick Scan with water. With the water level lowered, we would expect this dose rate to increase to the range of 1500 to 2000 R/hr. Similarly, the dose rate of about 50 R/hr measured at the inside surface of the reactor vessel head would likely increase to 500 to 600 R/hr.

Please call me if you have any questions with regard to this letter or the enclosed report.

Sincerely,

David G. Strawson

David G. Strawson

cc: H. Burton, DOE
B. Kanga, GPUN
J. Devine, GPUN

MPR ASSOCIATES, INC. January 25, 1983

REPORT FOR TAAG

EVALUATION OF TMI-2 QUICK SCAN RESULTS

CONTENTS

January 25, 1983

REPORT FOR TAAG

EVALUATION OF TMI-2 QUICK SCAN RESULTS

1. Purpose

The TMI-2 Technical Asssitance and Advisory Group (TAAG) has evaluated the results of the Quick Scan experiment which was performed on December 16, 1982. In summary, Quick Scan consisted of inserting an ion chamber within two of the vacated leadscrew holes, i.e., at core positions B-8 and E-9. At each location, the gamma dose rate was measured at various elevations, from the inside surface of the reactor vessel head down to six inches below the plenum assembly cover plate.

The intent of the TAAG evaluation was to assess the radiation source characteristics underneath the reactor vessel head, based on the Quick Scan results, and to assess the implication of these results on subsequent defueling operations. Information available to date on measurements of the three pieces removed from leadscrew H-8 was also considered in the evaluations.

2. Summary and Conclusions

The results of the Quick Scan experiment are contained in Reference 1, and summarized in enclosed Figures 1 through 3. In particular, Figure 1 identifies the measurement locations and Figures 2 and 3 show the measured gamma dose rates for positions B-8 and E-9, respectively. Based on comparing the measured dose rates to those predicted for various types of sources as discussed in Section 4, and based on an overall review of the measured dose rates, it is concluded that:

a. The dose rates measured at both positions, shown superimposed in Figure 4, are quite similar. This implies that some consistent type of mechanism is causing the dose rates, rather than some localized deposit type phenomenon, such as a pile of debris over a small area of the plenum cover.

b. The measured dose rates are highest in the region of plenum components, i.e., 500 to 600 r/hr from six inches below the plenum cover to about 18 inches above the plenum cover. The dose rates are substantially lower in the region of reactor vessel head components, i.e., about 50 r/hr at the elevation of the inside of the head and up to 200 r/hr at the bottom end of the 2.1 inch ID leadscrew support tubes. Conclusions with regard to head component and plenum component radiation levels are given below.

c. Reactor vessel head component dose rates may be characterized as follows:

- The inside surface of the reactor vessel head is not significantly contaminated. This can be seen from Figure 5, which shows that if the head were contaminated to the same degree as removed leadscrews, (i.e., about 50 r/hr at contact), a significant dose rate peak would have been measured at the inside surface of the head. No such peak was observed.

- The measured dose rates are consistent with having surface contamination on the leadscrew support tubes only, of about the same magnitude as on the removed leadscrew pieces (about 50 r/hr). This is shown in Figure 6, which compares the measured dose rates to those predicted for uniform contamination on the inside and outside surface of the support tube equal to that on the leadscrew. Actual dose rates are lower near the vessel head than would be predicted based on leadscrew contamination. One possibility is that the activity deposition is temperature dependent, so decreases near the head which was likely cooler than the support tubes.

- The dose rate from vessel head components, once the head is removed from the reactor vessel, will likely not be substantially different than the dose rates experienced at an operating plant. For example, Oconee experienced a dose rate of 35 r/hr about 18 inches below the bottom of the leadscrews with the head on its storage stand. This dose rate is included for reference in Figure 6. The dose rate for the TMI-2

reactor vessel head on its storage stand has been calculated assuming surface contamination on the leadscrew support tubes. This dose, at 18 inches below the bottom of the leadscrews, as shown in Figures 7 and 8, has an upper limit value of about 185 r/hr, and is more likely about 35 r/hr.

d. Conclusions with regard to plenum component dose rates are as follows:

- The dose rates from plenum components are high with respect to the leadscrew piece dose rates, and the dose rate assumed for head removal planning. As discussed in Reference 2, this planning basis dose rate, at a point five feet above the plenum cover plate and four feet beyond the inside diameter of the vessel flange, is 8.5 r/hr. As discussed in Section 4, the Quick Scan measurements indicate that the actual dose rate may be to about 75 r/hr, rather than 8.5 r/hr. Accordingly, operations which involve personnel exposure to radiation from plenum components need to be reevaluated.

- The high dose rates are apparently <u>not</u> related to a layer of core debris on top of the plenum cover. This is shown in Figures 9 and 10, which compare the calculated dose rate profile for a layer of debris on the plenum cover plate and the guide assembly first and second support plates, to the actual dose rates at positions B-8 and E-9, respectively. As shown, if the dose rate were due to a layer of debris, a substantial peak would have been observed at the elevation of the plenum cover plate. No such peak was observed at either measurement location.

- The measured dose rates are quite consistent with having surface contamination on vertical plenum component surfaces as well as horizontal ones. This is shown in Figures 11 and 12, for locations B-8 and E-9, respectively. In particular, these figures compare actual dose rate to the dose rate profile predicted assuming surface contamination of the control rod guide assembly 8 inch ID tube plus the top and bottom of the first and second support plates.

- It should be noted that the Quick Scan result, i.e., that doses are not caused by core debris, is consistent with two other results. These include the TV inspection of the control rod guide assembly during the Quick Look experiment, and the leadscrew measurements summarized below.

Results of measurements to date of the three sections removed from leadscrew H-8 are summarized in Table 1, and were obtained from References 3 and 4, as well as from discussions with Mr. J. Daniel of SAI. Behavior of the leadscrews is considered significant, because it may well be similar to that of the stainless steel components of the plenum assembly. Based on these results and further evaluations described in Section 4, it is concluded that:

- The leadscrew dose rates are due primarily to Cs-137 activity, based on the results of a gamma scan of the three leadscrew sections.

- The dose rate at contact with leadscrew surfaces is substantially higher, i.e., by at least a factor of 100, than would be expected based on surface contamination with fuel, or from CS-137 present in a coolant film on the leadscrew surface during leadscrew removal. Cesium has been concentrated on the leadscrew surface by some mechanism, not yet defined.

- The cesium contamination on the leadscrews is tenacious, i.e., most of the cesium contamination remained on a leadscrew piece after 24 hours of soaking in clean water. The cesium is apparently tightly bound into either the leadscrew corrosion product film or the base metal.

- One possible mechanism for the surface contamination is intergranular attack by cesium, and formation of a stable cesium-chromium compound at grain boundaries. This type of attack was experienced during EPRI sponsored work on LMFBR fuel with stainless steel cladding, as discussed in Reference 5.

Overall conclusions with regard to the Quick Scan measurements are as follows:

- The measured dose rates are most likely due to surface contamination of both vertical and horizontal plenum assembly and head component surfaces. This is most likely a tightly adherent surface contamination by cesium in either the oxide film or base metal of the components, similar to that found on removed leadscrews. Based on information obtained to date, it does not appear that this contamination can be removed by flushing or vacuuming under the head.

- The major concern is with plenum component surface contamination, i.e., the components which would remain in place once the reactor vessel head is removed, particularly during the short period of time between head removal and filling the indexing fixture with water.

- The dose rates of the reactor vessel head itself appear to be of the same order currently experienced at other plants. The head itself could likely be handled along the lines currently planned.

- The Quick Scan radiation result (i.e., that radiation is due to surface contamination instead of debris), is consistent with the results of the TV inspection of the control rod guide assembly during the Quick Look experiment, and also with the results of removed leadscrew piece evaluations.

3. Recommendations

TAAG recommends the following course of action:

a. Proceed with examination of the leadscrews which have already been removed. These examinations should include:

    - The 9 inch section of leadscrew H-8 which has already been sent to Battelle PNL. Specific examinations were recommended to Battelle in Reference 6, and include (1) confirming that the surface contamination is cesium, (2) determining the location of the contamination, e.g., in the oxide film or base metal of the leadscrew, (3) determining the chemical form of the contamination, e.g., is

it a stable cesium-chromium compound formed during intergranular attack such as was experienced by the stainless steel cladding of LMFBR fuel.

- Perform a Gamma scan of the lower end of leadscrew H-8, which has already been cut up into three sections, of 4 foot, 4 foot, and about 30 inch lengths. The transition region between 17-4 PH and 304 stainless steel sections of the leadscrew should be examined closely for any evidence of differences in surface contamination. If differences are observed, metallographic examinations similar to those on the 17-4 PH leadscrew pieces should be performed on a 304 stainless steel section of the leadscrew, removed from just below the 17-4 PH threaded section of the leadscrew. The purpose of this examination would be to establish whether the 304 stainless steel and 17-4 PH materials have been contaminated by the same mechanism.

- Obtain contact dose rate measurements along the full length of a removed leadscrew. The purpose of these examinations is to determine if the contamination varies axially in a manner consistent with the course of events during the TMI-2 accident, e.g., is contamination maximized at the interface between reactor coolant and the superheated steam bubble that existed before the reactor coolant pumps were restarted on the evening of March 29, 1979. In addition, measurements of the upper end of the leadscrew could provide confirmation that surface contamination decreases at elevations near the reactor vessel head as implied by the Quick Scan measurements.

b. Proceed with further under-the-head examinations, primarily to confirm by visual inspection that the measured dose rates are <u>not</u> caused by debris on the plenum cover. Specifically, it is recommended that:

- B&W complete development of the plasma arc cutting tool, which initial tests have shown to be promising.

- This tool should be used, following successful mockup demonstration, to cut a 2.1 inch ID leadscrew support tube. This should provide a relatively simple and quick means of obtaining access to the head area with minimum exposure to personnel.

- The TV camera employed for Quick Look be inserted to visually inspect plenum component surfaces for any evidence of debris.

c. Perform evaluations of the impact on currently planned head removal operations, including indexing fixture installation, of <u>not</u> having any water over the plenum components. Specifically, the evaluations should determine whether additional shielding, long-handled tools, etc., could be employed to maintain personnel exposures at an acceptable level during the operations.

d. Perform evaluations of methods for raising the reactor coolant system level as the reactor vessel head is removed so that the plenum can be covered by water for shielding. It is recommended these evaluations include:

- Feasibility of flooding the transfer canal in a time frame consistent with currently planned head removal, i.e., by about June of 1983.

- As an alternate, feasibility of installing a leak-tight sleeve external to the reactor vessel head.

4. <u>Discussion</u>

a. <u>Summary of Quick Scan Results</u>

Enclosed Figure 1 shows the Quick Scan measurement locations. The measured gamma dose rates at leadscrew hole positions B-8 and E-9 are shown in Figures 2 and 3, respectively, and are shown superimposed in Figure 4. The measured dose rates at both locations are quite similar. The dose rates are highest in the region of the plenum assembly components, i.e., 500 to 600 r/hr in the region from six inches below to 12 inches above the plenum cover plate. The dose rates decrease

rapidly in the region from 12 inches above the cover plate to a value of about 200 r/hr at two feet above the cover plate, which coresponds to about the lower end of the 2.1 inch ID leadscrew support tubes. The dose rates decrease smoothly to a value of about 50 r/hr at the inside surface of the reactor vessel head.

Various simple models have been employed to compare the measured dose rates to those which would be predicted for different types of sources. These are discussed further in Sections 4.b and 4.c. Section 4.b covers the dose rates associated with reactor vessel head components, i.e., the 2.1 inch ID leadscrew support tubes and the inside surface of the head itself. Section 4.c covers the dose rates associated with the plenum assembly components, including the plenum cover plate and the eight inch ID control rod guide assembly tube.

Several common assumptions have been applied to the evaluations. These are:

- All radiation, whether from surface contamination or from core debris, is due to Cs-137 gammas at an energy of 0.662 mev.
- Water is present within the vessel at room temperature, ie., its density is 1 gram per cubic centimeter.
- Various physical constants, e.g., macroscopic absorption coefficients, buildup factors, flux-to-dose rate conversions, etc., are obtained from Reference 7.

b. Reactor Vessel Head and Leadscrew Support Tube Dose Rate Contributions

Dose rates due to vessel head components are calculated assuming an amount of surface contamination equivalent to that found on the three sections of leadscrew H-8, i.e., equivalent to a contact dose rate on the leadscrew of about 50 r/hr. Two cases were analyzed.

In the first case, shown in Figure 5, it is assumed that surface contamination is present on the inside surface of the reactor vessel head, and also on the inside and outside surfaces of the

leadscrew support tube. As shown in Figure 5, surface contamination on the head inside surface of the same magnitude as on the support tube would cause a dose rate peak near the head surface. Such a peak was not measured. Accordingly, it appears that contamination of the head inside surface is small with respect to contamination on the support tube surfaces.

In the second case, shown in Figure 6, it is assumed that surface contamination is present only on the inside and outside surfaces of the lead-screw support tubes. As shown, this results in a dose rate of about 160 r/hr from the head surface down to about six inches above the bottom end of the support tube. By comparison with the actual dose rates, the assumed amount of surface contamination is too high over the upper end of the support tube near the reactor vessel head.

The calculated dose shown in Figure 6 is lower than measured at the bottom end of the support tubes. However, this calculated dose does not include any contribution from plenum components. As shown later, if a plenum contribution is added to support tube contamination, calculated doses at the lower end of the support tube are about equal to the measured dose rates.

The dose rates to be expected with the reactor vessel head removed and on its storage stand have also been calculated. In particular, the dose has been calculated at a point 18 inches below the bottom of the leadscrews and at the center of the head, for comparison to the dose rate measured at Oconee-1 at the same locations. The dose rates are calculated by summing the contributions from all 69 leadscrews and their support tubes. The specific geometry and major assumptions employed for the calculations, as well as the calculation results, are shown in Figures 7 and 8. Figure 7 contains an upper limit estimate of the dose rate, while Figure 8 contains a more realistic "best estimate" value. As shown, the calculated dose rate is in the range of 35 to 185 r/hr. It is the same order of magnitude as the Oconee-1 dose rate of 35 r/hr.

c. Plenum Assembly Dose Rate Contributions

Dose rates in the vicinity of plenum components are calculated assuming two different kinds of sources.

In the first case, it is assumed that only the top side of horizontal surfaces is contaminated, which is what one would expect if the dose rates were caused by core debris settling out of reactor coolant. Calculated results, shown in Figures 9 and 10 for positions B-8 and E-9, respectively, are based on assuming debris on the plenum assembly cover plate and on the first and second support plates of the control rod guide assembly. The magnitude of the dose rate is set by normalizing to the peak measured value, i.e., it is assumed that the maximum calculated dose equals the maximum measured dose. The significance of the calculated values is in the shape of the curves. In particular, they show that if dose rates were due only to horizontal surface debris accumulation, a significant peak should have been observed at the elevation of the plenum cover plate. No such peaking was measured.

In the second case, it is assumed that vertical surfaces as well as horizontal surfaces are contaminated. The calculated values, shown in Figures 11 and 11 for locations B-8 and E-9, respectively, assume in particular that the dose is due primarily to surface contamination on the inside surface of the eight inch control rod guide assembly tube, and the top and bottom surfaces of the first and second support plates. Again, the calculated results are normalized to the maximum mesured values, so that the significance of the results is in the shape of the curves. As shown, the measured values are quite consistent with having surface contamination on vertical plenum component surfaces as well as horizontal ones.

The combined dose rates; i.e.the dose rates from plenum components as calculated above and the dose rates from the leadscrew support tubes as calculated in Section 4.b, are shown in Figure 13 for position B-8. The calculated dose rates show fairly good agreement with the measured dose rates.

Finally, the dose rates which would be expected from plenum components once the reactor vessel head is removed have been calculated. In particular, the dose rate has been calculated at a point five feet above the center of the plenum cover plate, and also at a distance four feet beyond the inside diameter of the reactor vessel and five feet above the plenum cover plate. The dose rates are calculated by summing the contributions from the 69 control rod guide assemblies and the plenum cover plate, when no water is present above these components. The specific geometry and major assumptions employed for the calculations, as well as the calculataion results, are shown in figure 14. As shown, the calculated dose rate five feet above the centerline of the plenum components is about 350 r/hr. The calculated dose rate at the "planning basis" location, i.e. four feet beyond the inside diameter of the reactor vessel and five feet up, is about 75 r/hr. Accordingly, the estimated dose rate is about a factor of nine higher than the "planning basis"dose rate of 8.5 r/hr.

d. Leadscrew Piece Evaluations

Results of measurements to date by SAI of the three sections removed from leadscrew H-8 are summarized in Table 1. These results are contained in part in References 3 and 4. Additional results, pertaining to the effectiveness of various decontamination techniques, were obtained from discussions with Mr. J. Daniel of SAI.

In summary, three pieces were cut from the lower half of leadscrew H-8 which was removed for the Quick Look experiment. The elevations of the removed pieces with the leadscrew in its scram position are approximately as follows:

- First nine inch piece:
  - Top end: Seven inches above bottom end of 2.1 inch ID support tube
  - Bottom end: At top of control rod guide assembly
- Second nine inch piece:

- Top end: At top of control rod guide assembly

- Bottom end: One foot above plenum cover plate

* Twelve inch piece:

  - Top end: One foot above plenum cover plate

  - Bottom end: At plenum cover plate

Contact dose rate measurements on the removed pieces showed dose rates of 30, 50 and 60 r/hr for the first and second nine-inch pieces and the 12-inch piece, respectively. Subsequent gamma scans of each piece showed that the dose was due primarily to Cs-137.

In addition, gamma scan results for cerium 144 were used based on a correlation of cerium to plutonium activity in fuel, to estimate an amount of fuel present on the leadscrew surfaces. The amount of fuel varied from about 0.2 grams for the first nine-inch piece to 0.9 grams for the 12-inch piece. It should be noted that this is a small amount of fuel which would not be visible to the naked eye. It corresponds, for example, to a layer of fuel one-tenth of a mil thick (or 0.0001 inches thick) on the top of the leadscrew thread surfaces.

Evaluations were performed by TAAG to determine if the measured dose rates could be explained by a straightforward mechanism. It was first assumed that the dose rate is the result of the measured amount of fuel on a leadscrew, i.e., 0.9 grams. The fuel was assumed to contain all of its initial cesium, i.e., any leaching was neglected, which over-estimates the amount of cesium present. The calculated contact dose rate for these assumptions is about 0.6 r/hr, or a factor of 100 below the actual dose rate.

It was next assumed that the dose rate is the result of cesium contained in water present on leadscrew surfaces when the leadscrew was removed from the reactor vessel. In particular, a 10 mil thick layer of coolant at about 20 microcuries per

ml of cesium 137 was assumed to exist on the leadscrew. The calculated contact dose rate for these assumptions is about 0.06 r/hr, or a factor of 1,000 below the actual dose rate.
It is apparent that the leadscrew dose rates are not due to fuel present on the leadscrew, or due to cesium present in a water film on the leadscrew when it was removed from the vessel. Cesium has been concentrated on the leadscrew surface by some other mechanism, not yet defined.

After the three leadscrew pieces were removed from containment, attempts were made to decontaminate the 12-inch piece using progressively severe techniques. These included (1) soaking the leadscrew for 24 hours in deborated water, (2) soaking for 1-1/2 hours in borated water at 3,700 ppm, (3) soaking for 1-1/2 hours in nitric acid.

A rough measure of the effectiveness of these treatments is given by the dose rate measurements in Table 1. It should be noted that the initial dose rate was a contact value at the leadscrew surface, whereas subsequent measurements were at contact with the PVC container which held the piece. SAI has indicated the reported measurements would be roughly twice the indicated value if actual contact measurements had been made.

The reported results indicate that the initial water soaks had very little effect on the cesium contamination. The soak in nitric acid reduced the contamination by about a factor of five.

It is concluded from these results that the cesium contamination on the leadscrews is tenacious. The cesium may be tightly bound in the leadscrew oxide film or in the base metal.

5. References

1, GPUN Interoffice Memorandum from V. R. Fricke, December 17, 1982, No. 4550-82-067, "Quick Scan" Experiment.

2. Fourth Report of the TMI-2 Technical Assistance and Advisory Group (TAAG), dated December 1, 1982, Section V.B. Underhead Examinations.

3. SAI Interoffice Memorandum from J. A. Daniel, November 24, 1982, No. SAI-82-066, "Observations on Leadscrew Sections".

4. SAI Interoffice Memorandum from J. A. Daniel, December 4, 1982, No. SAI-82-068, "Gamma Scan Results of 9-9-12 inch Leadscrew Sections".

5. Structural Materials in Nuclear Power Systems, J. T. Adrian Roberts, EPRI, Plenum Press, New York and London, Chapter 3, LMFBR Core Materials, Pages 147 and 148.

6. Letter from TAAG (W. H. Hamilton) to Battelle PNL (R. P. Allen) dated December 29, 1982, "Three Mile Island Unit 2 Leadscrew Testing".

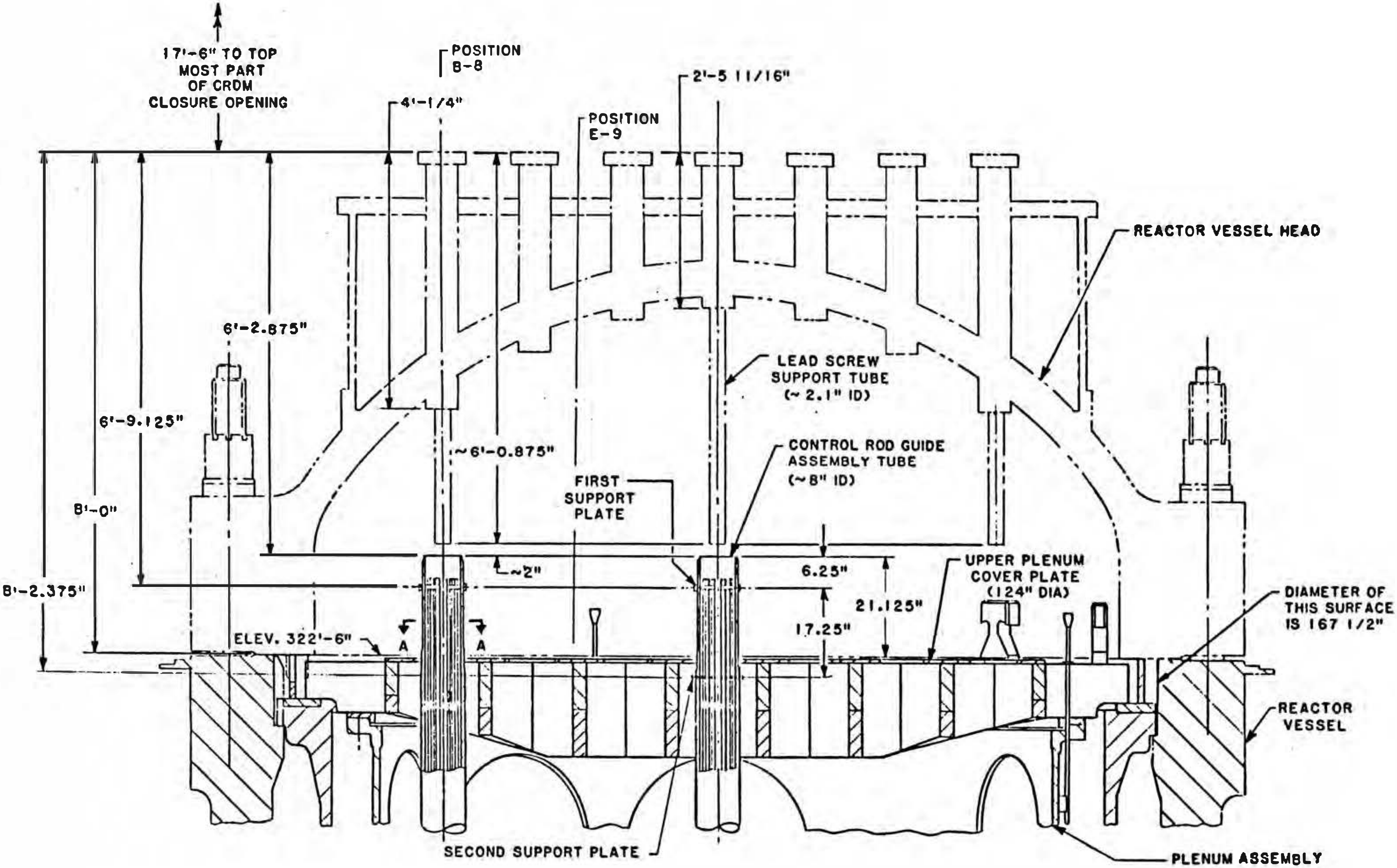
7. Reactor Shielding Design Manual, Theodore Rockwell III Editor, First Edition, D. Van Nostrand Company, 1956.

TABLE 1

LEADSCREW PIECE MEASUREMENTS

Parameter	Value
1. Measured value (by SAI) of leadscrew pieces	
• Contact dose rate	30-60 r/hr
• Cs-137 activity	0.12 to 0.47 Ci
• Fuel on leadscrew pieces (based on Ce-144)	0.2 to 0.9 g (0.9 g corresponds to a 1/10 mil thick layer of fuel on top thread surfaces)
2. Predicted dose rate	
• Assuming 0.9 g fuel on leadscrew, no leaching	0.6 r/hr
• Assuming evaporation of 0.010 inch thick water film, at 20 microcuries/ml, on leadscrew surface	0.06 r/hr
3. Measured contact dose rates on 12-inch piece after various decontamination steps	
• Before decontamination	60 r/hr
• After 24-hour soak in clean water	28 r/hr (1)
• 1-1/2-hour soak in borated water	27 r/hr (1)
• 1-1/2-hour soak in nitric acid	5 r/hr (1)

(1) These measurements were made from the outside surface of the PVC pipe containing the leadscrew piece, rather than at contact. Per SAI, actual contact doses could be about two times as high as these values, e.g., 56 r/hr after a 24-hour soak in clean water.



FOR SECTION A-A
SEE MPR SKETCH
SK-1074-01-519

KEY ELEVATIONS FOR MEASURING UNDER-THE-HEAD RADIATION LEVELS
TO ASSESS THE NEED FOR AN UNDER-THE-HEAD
PHYSICAL INSPECTION

FIGURE 1

MPR SKETCH
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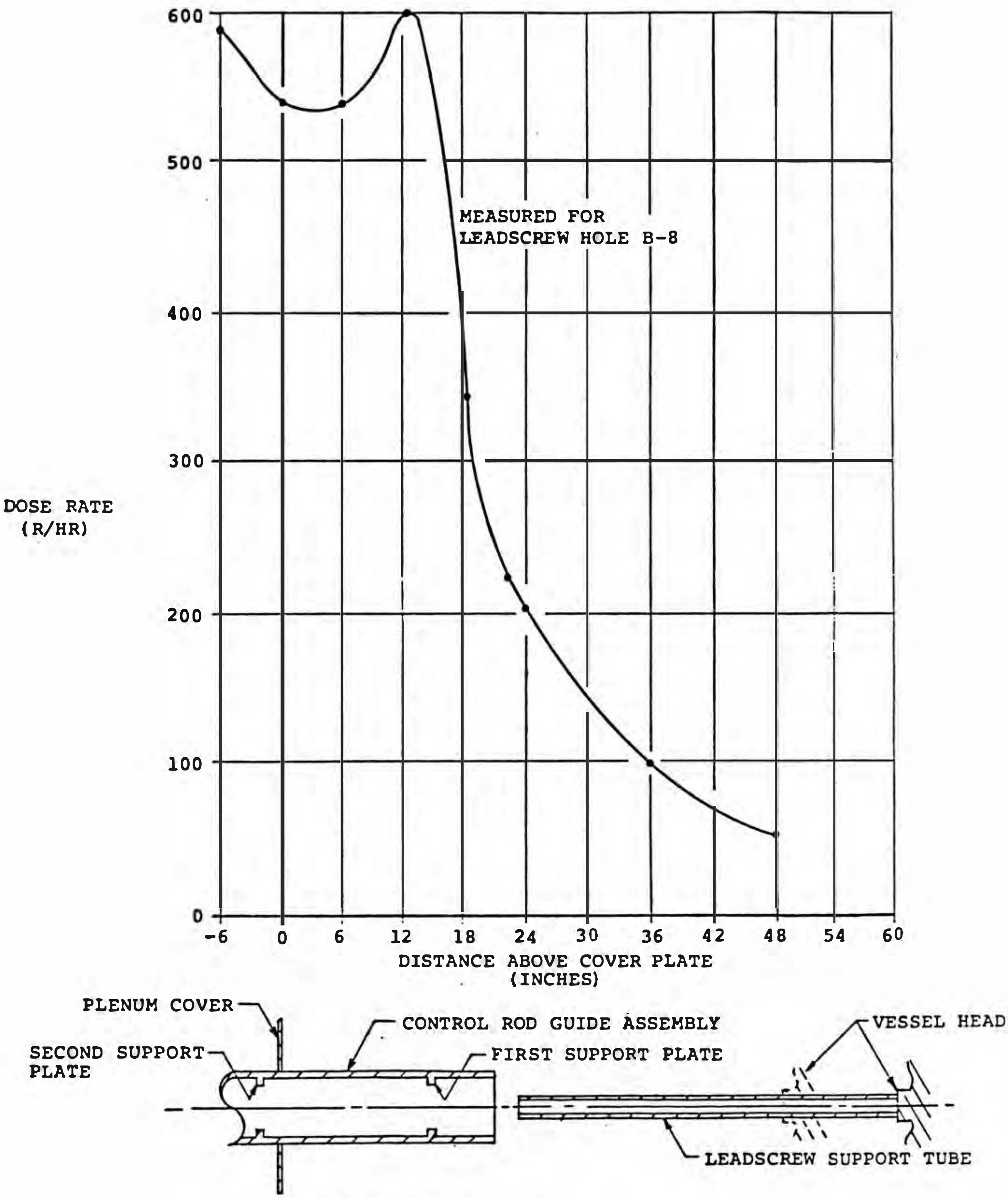
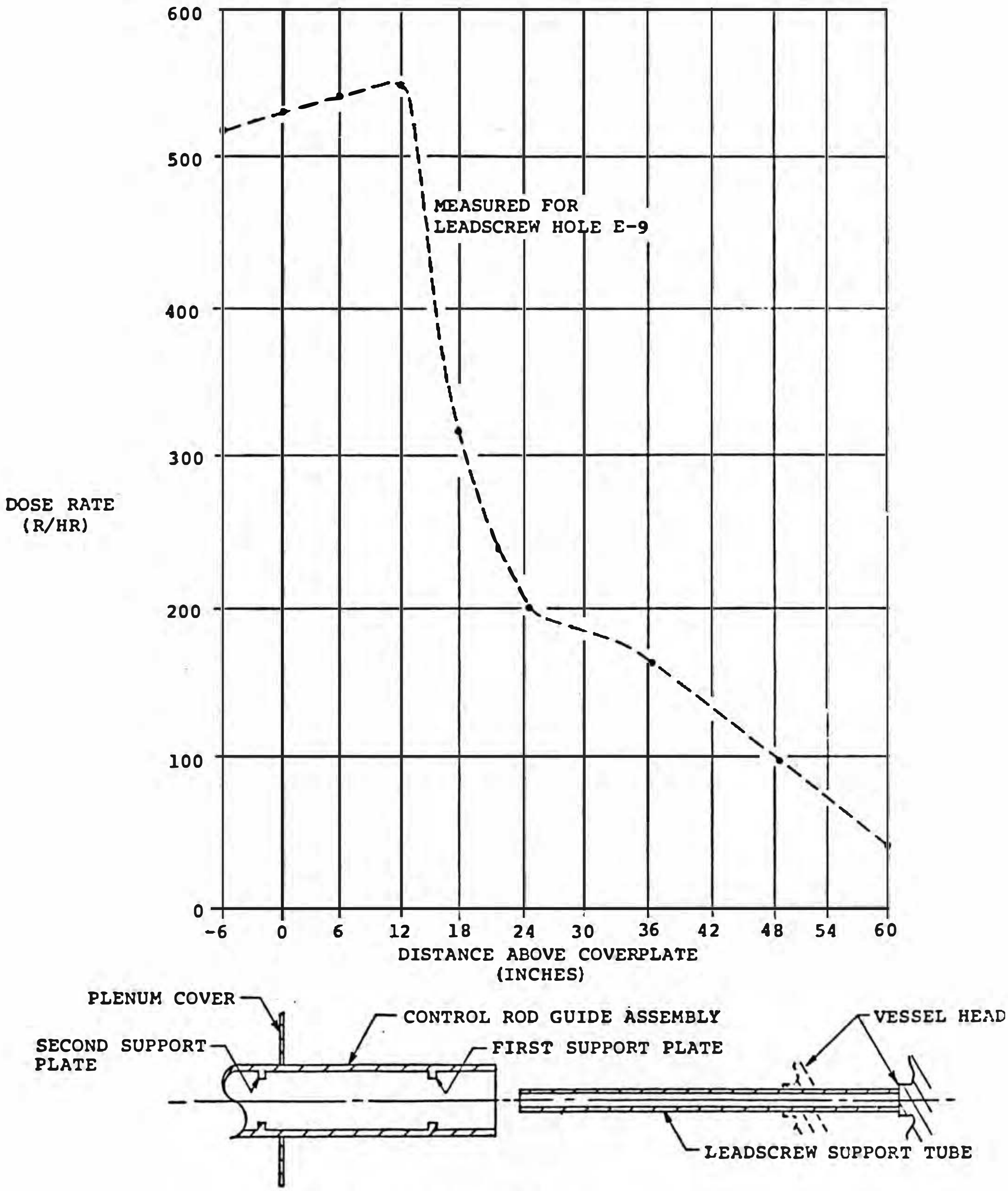
REVISION 1
1/24/83



MEASURED DOSE RATES
AT POSITION B-8

FIGURE 2

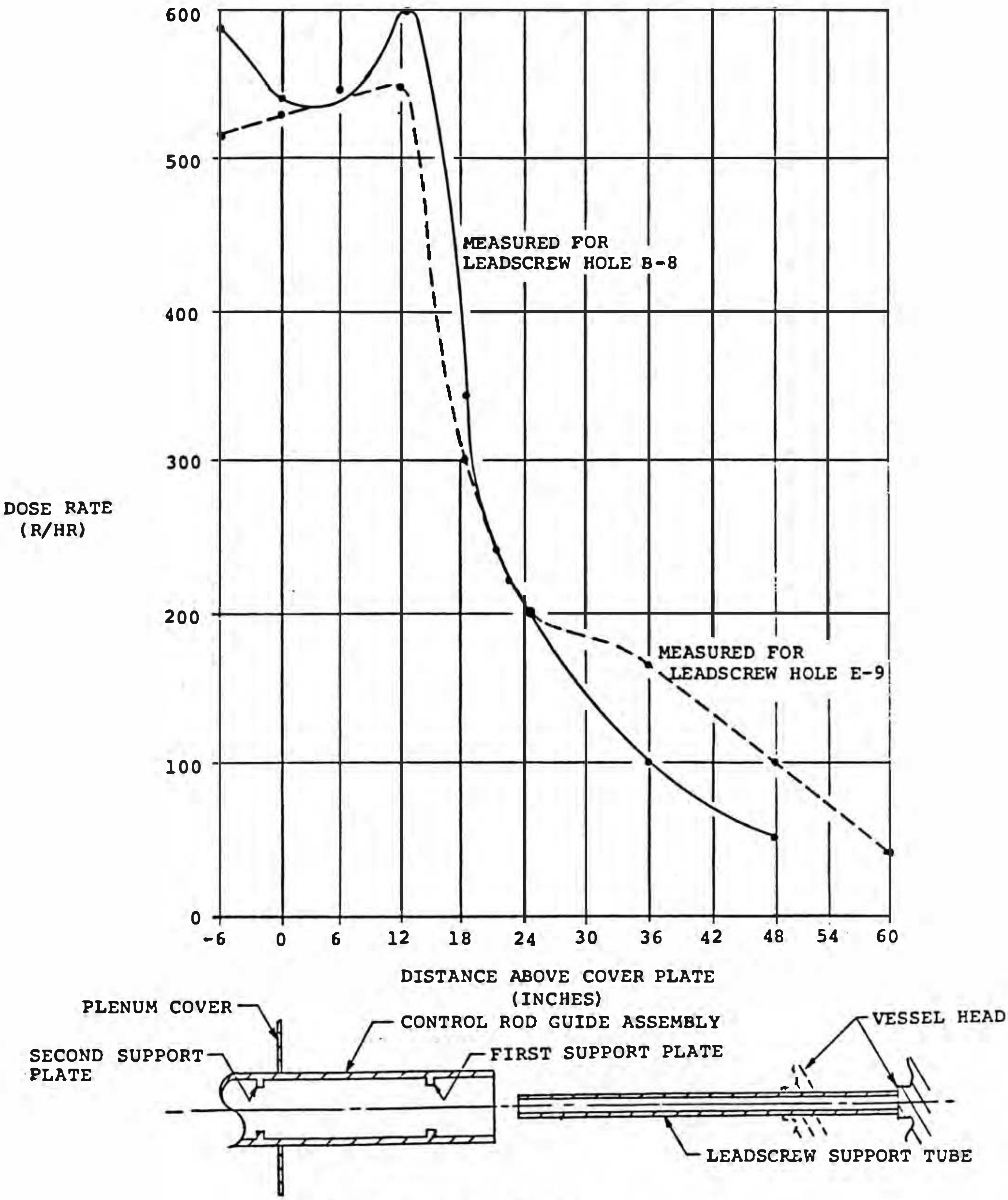
MPR ASSOCIATES, INC
WASHINGTON D.C. 20036
F-74-01-2
REV.1, 1/25/83



MEASURED DOSE RATES
AT POSITION E-9

FIGURE 3

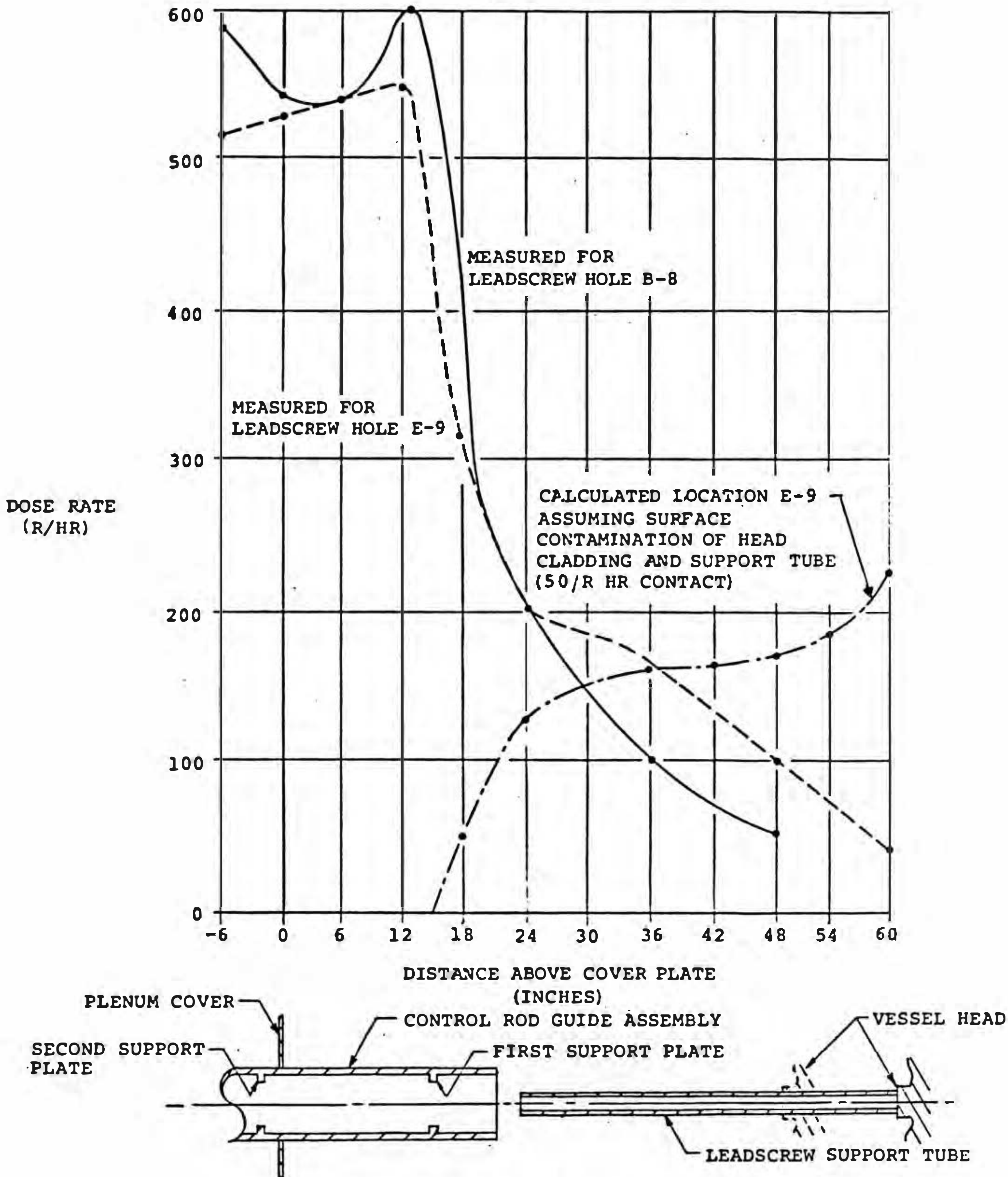
MPR ASSOCIATES, INC
WASHINGTON D.C. 20036
F-74-01-3
REV.1, 1/25/83



MEASURED DOSE RATES
AT POSITIONS B-8 AND E-9

FIGURE 4

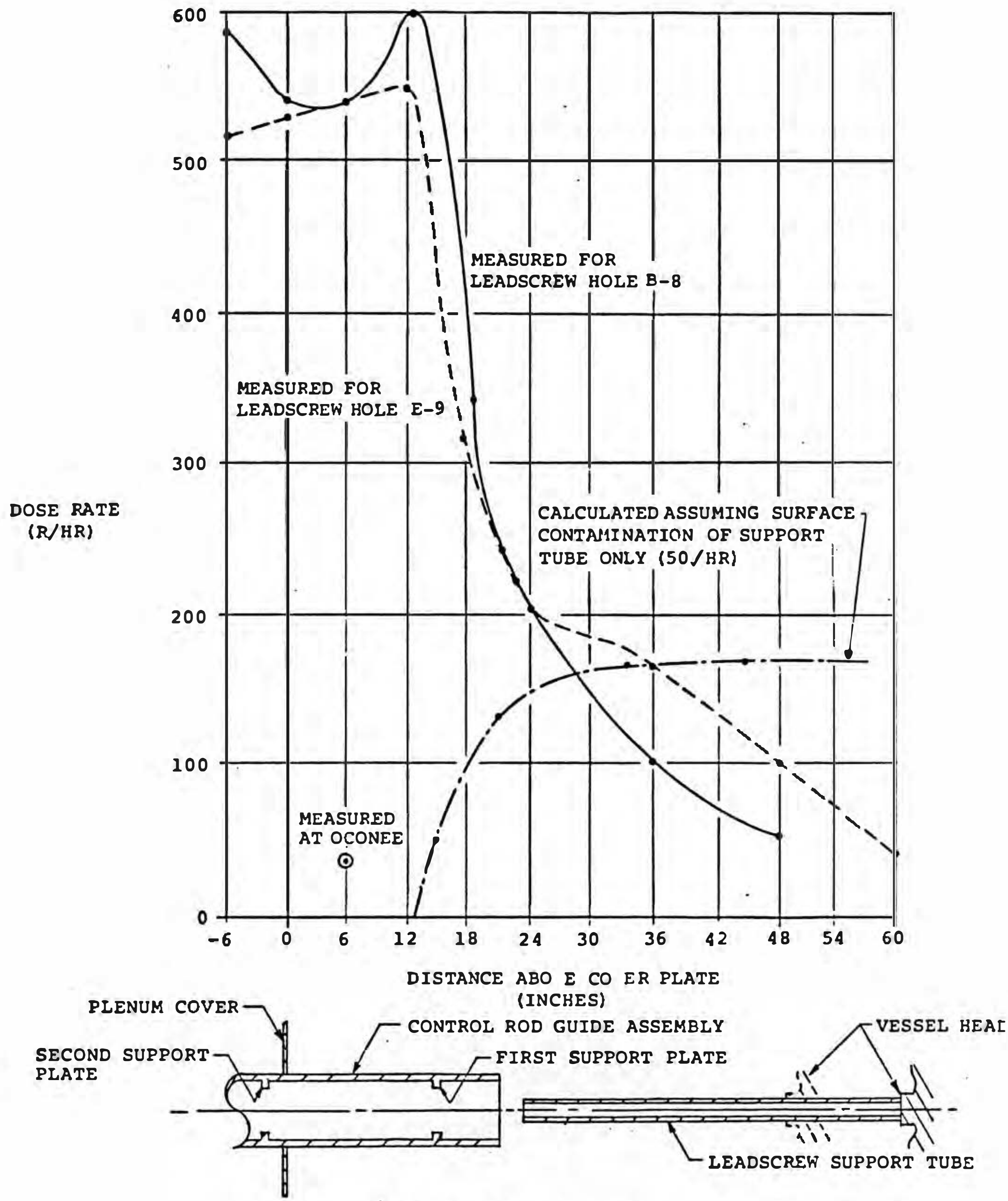
MPR ASSOCIATES, INC
WASHINGTON D.C. 20036
F-74-01-4
REV.1, 1/25/83



CALCULATED HEAD DOSE RATES
UNIFORM CONTAMINATION OF HEAD AND SUPPORT TUBES

FIGURE 5

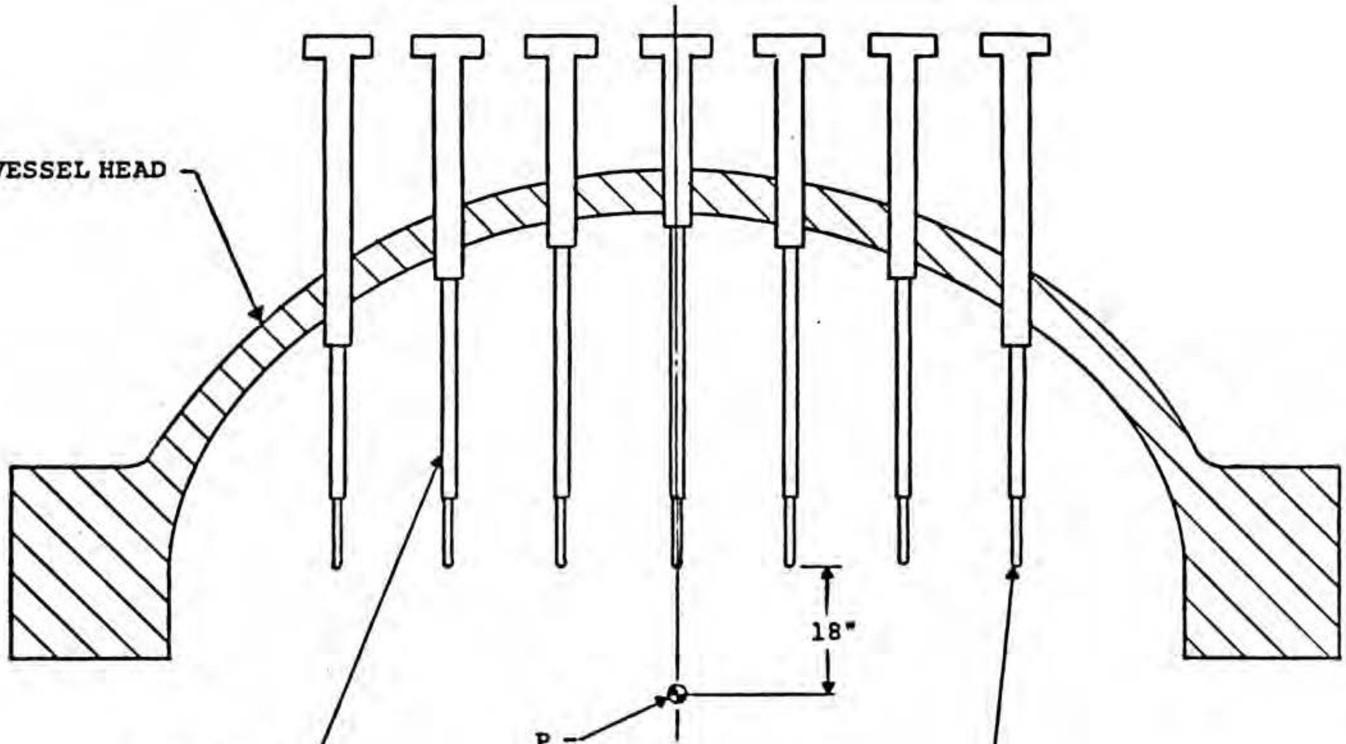
MPR ASSOCIATES, INC
WASHINGTON D.C. 20036
F-74-01-5
REV.1, 1/25/83



CALCULATED HEAD DOSE RATES
CONTAMINATION ON SUPPORT TUBE ONLY

FIGURE 6

MPR ASSOCIATES, INC
WASHINGTON D.C. 20036
F-74-01-6
REV.1, 1/25/83



DOSE RATE AT POINT P ≅ 185 R/HR
ASSUMING:

(1) EACH LEADSCREW AND ITS SUPPORT TUBE INSIDE AND OUTSIDE SURFACES HAVE SURFACE CONTAMINATION EQUIVALENT TO A CONTACT DOSE RATE OF 50 R/HR

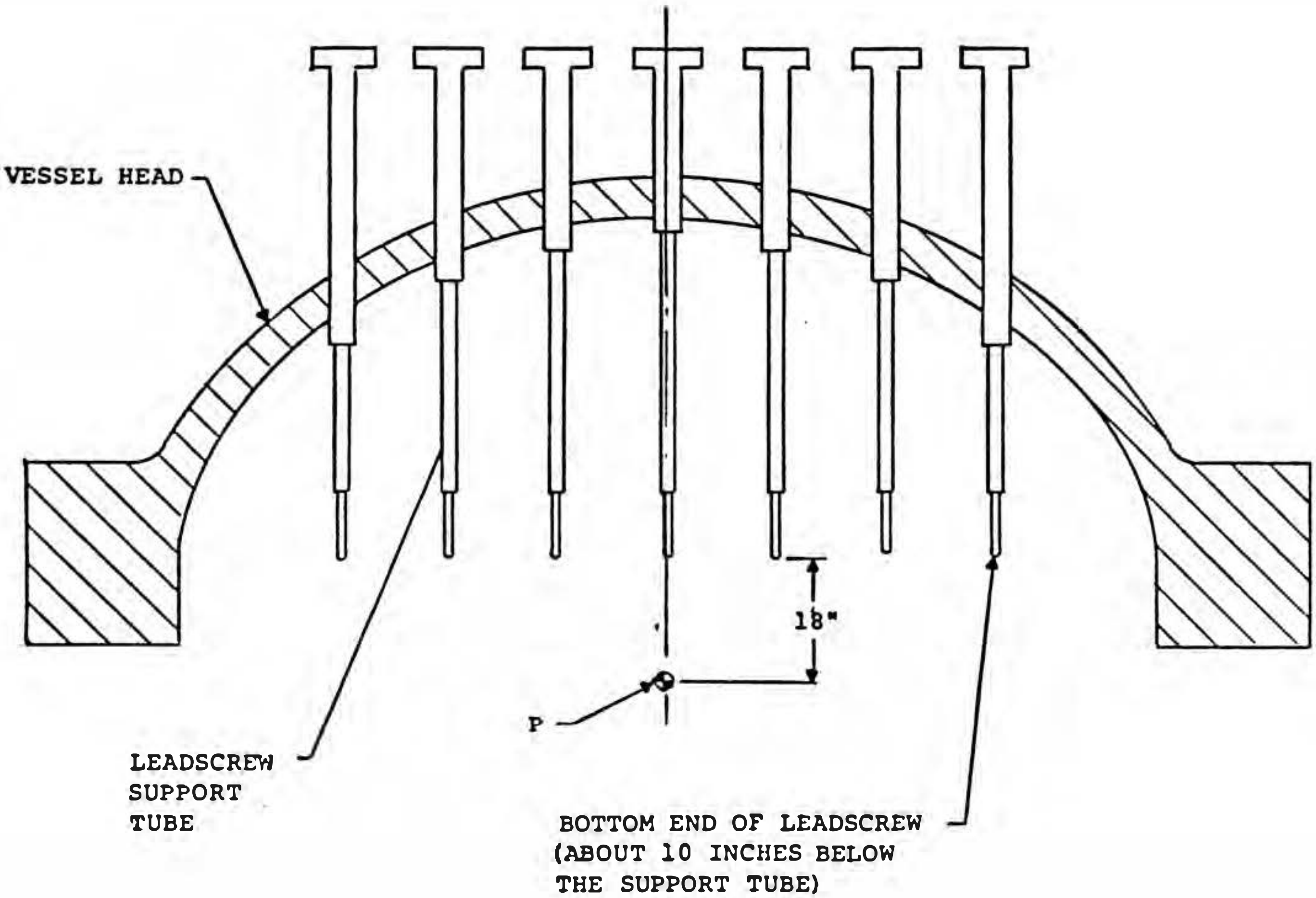
(2) INSIDE HEAD SURFACE DOSE RATE AT CONTACT OF 10 R/HR

(3) NO ATTENUATION BY LEADSCREW METAL OR SUPPORT TUBE WALLS

(4) ALL SUPPORT TUBES ARE SAME HEIGHT AS THE CENTER ONE AND EXTEND TO THE BOTTOM END OF THE LEADSCREW

UPPER LIMIT TO VESSEL HEAD DOSE RATES

FIGURE 7



DOSE RATE AT POINT P $\leqq$ 35 R/HR
ASSUMING:

(1) AVERAGE CONTAMINATION OF THE LEADSCREWS AND INSIDE AND OUTSIDE OF SUPPORT TUBES IS EQUIVALENT TO A CONTACT DOSE RATE OF 25 R/HR

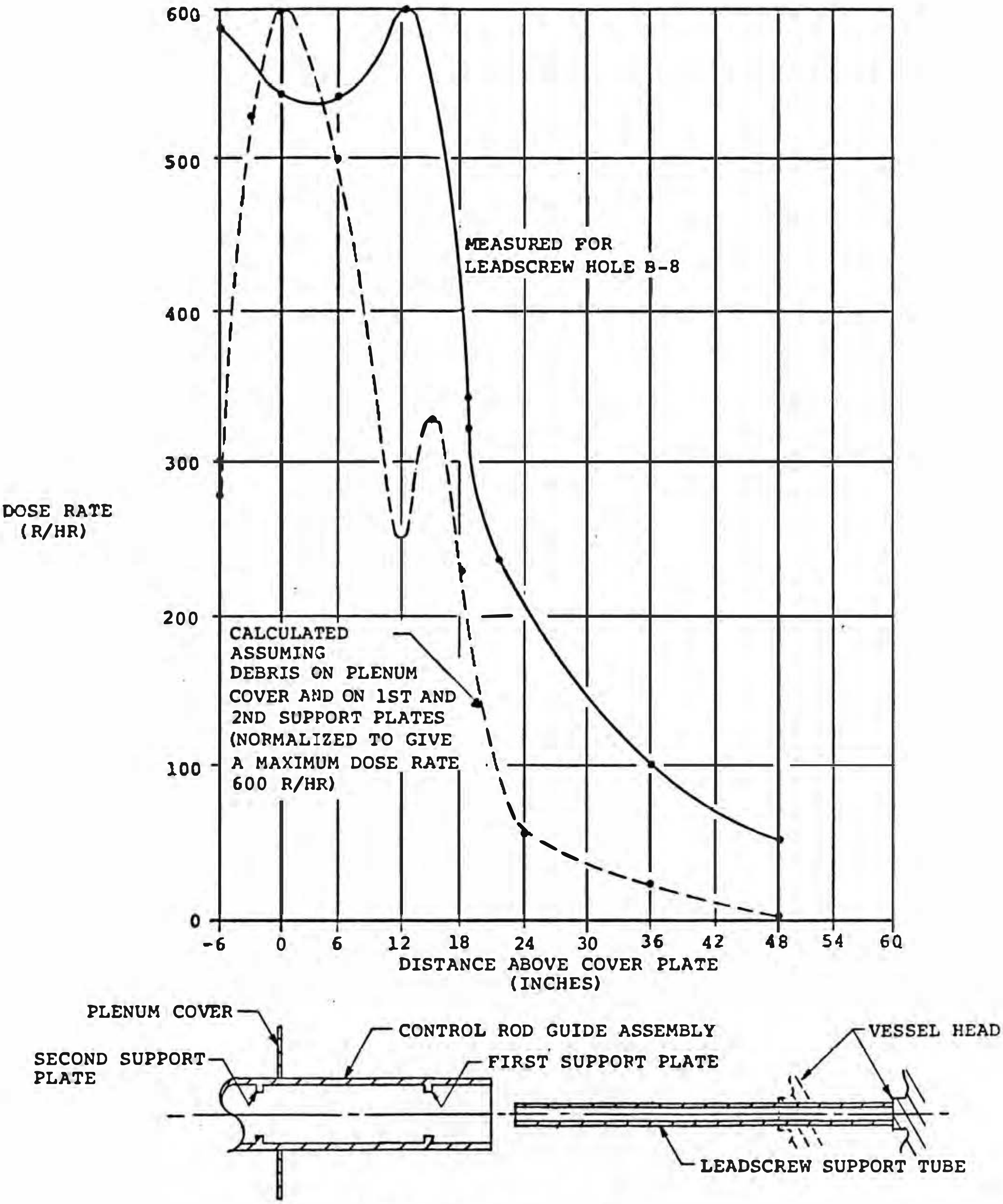
(2) DOSE RATE FROM THE INSIDE OF THE HEAD SURFACE IS NEGLIGIBLE

(3) ATTENUATION BY LEADSCREW METAL AND SUPPORT TUBE WALLS IS ACCOUNTED FOR

BEST ESTIMATE FOR VESSEL HEAD DOSE RATES

FIGURE 8

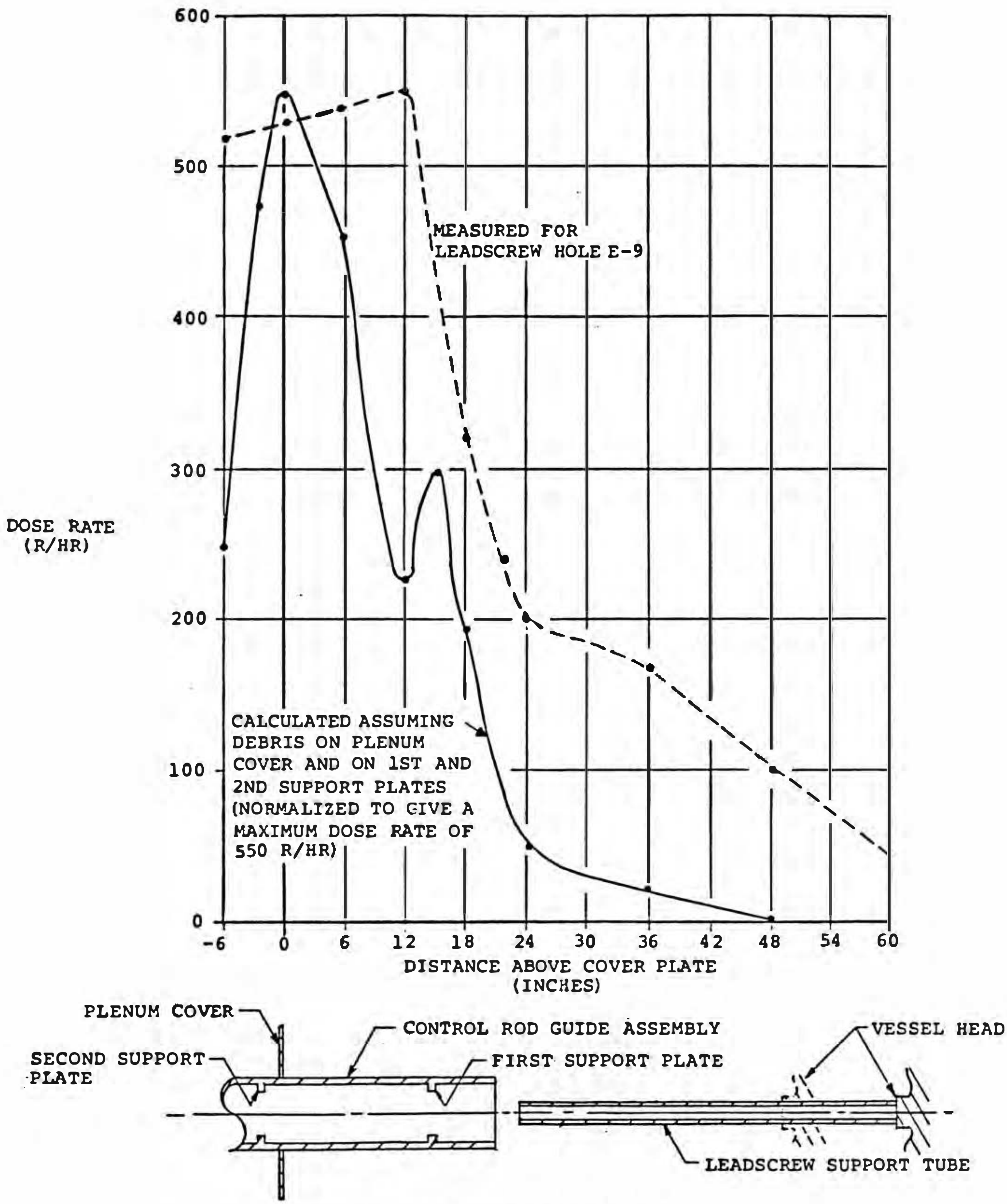
MPR ASSOCIATES, INC
WASHINGTON D.C. 20036
F-74-01-8
REV.1, 1/25/83



CALCULATED PLENUM DOSE RATES
AT LOCATION B-8 ASSUMING DEBRIS

FIGURE 9

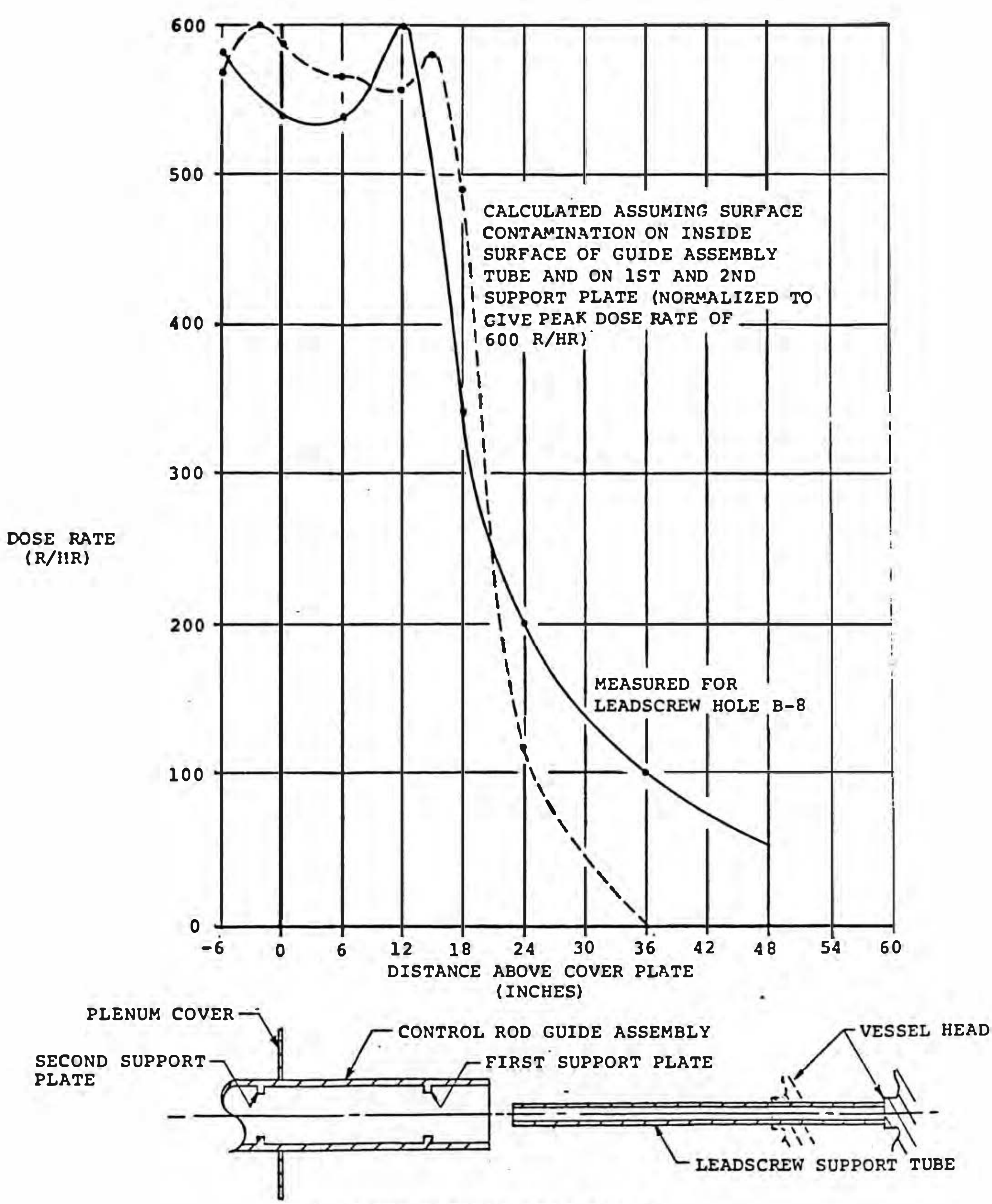
MPR ASSOCIATES, INC
WASHINGTON D.C. 20036
F-74-01-9
REV.1, 1/25/83



CALCULATED PLENUM DOSE RATES
AT LOCATION E-9 ASSUMING DEBRIS

FIGURE 10

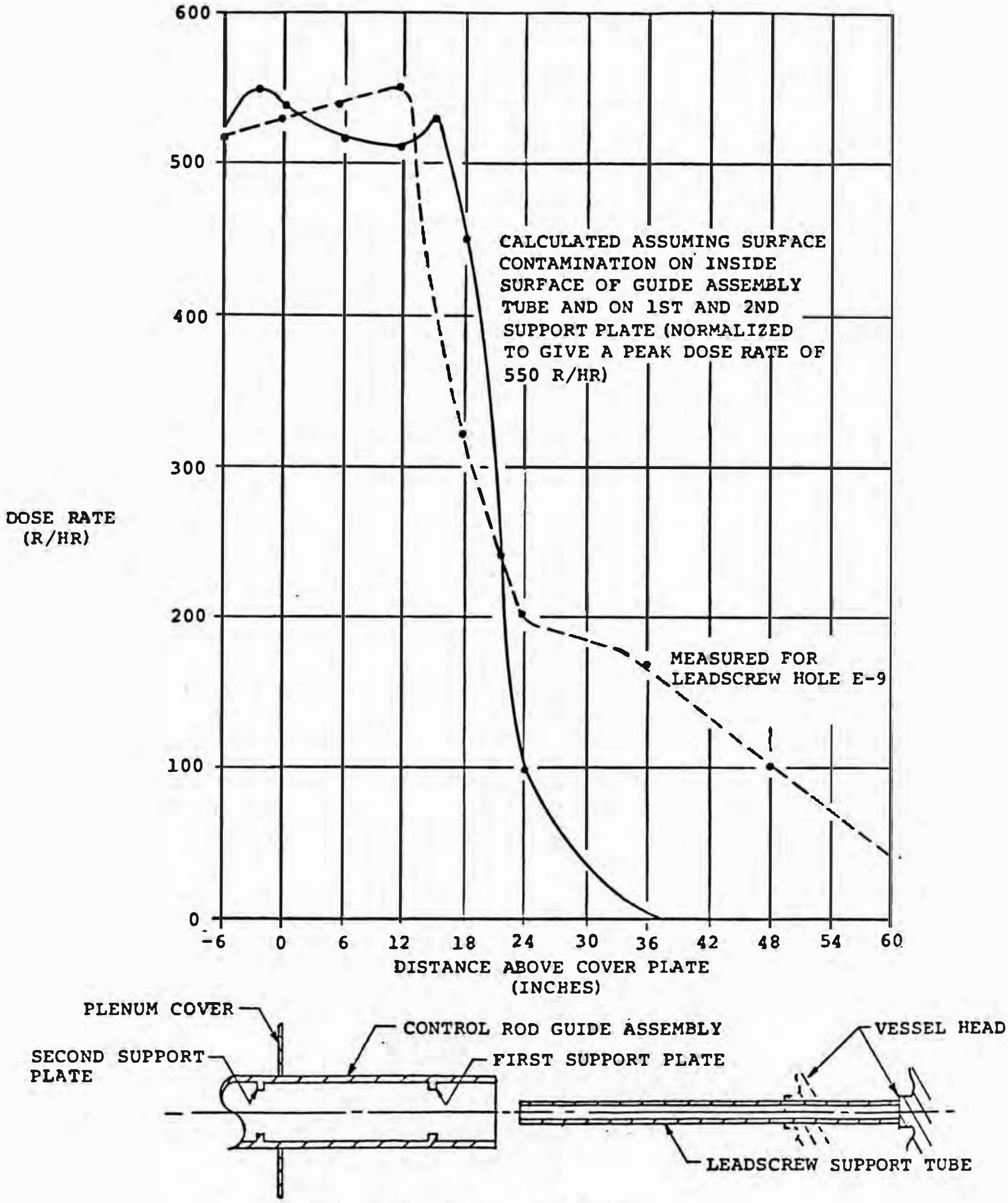
MPR ASSOCIATES, INC
WASHINGTON D.C. 20036
F-74-01-10
REV. 1, 1/25/83



CALCULATED PLENUM DOSE RATES
AT LOCATION B-8 ASSUMING SURFACE CONTAMINATION

FIGURE 11

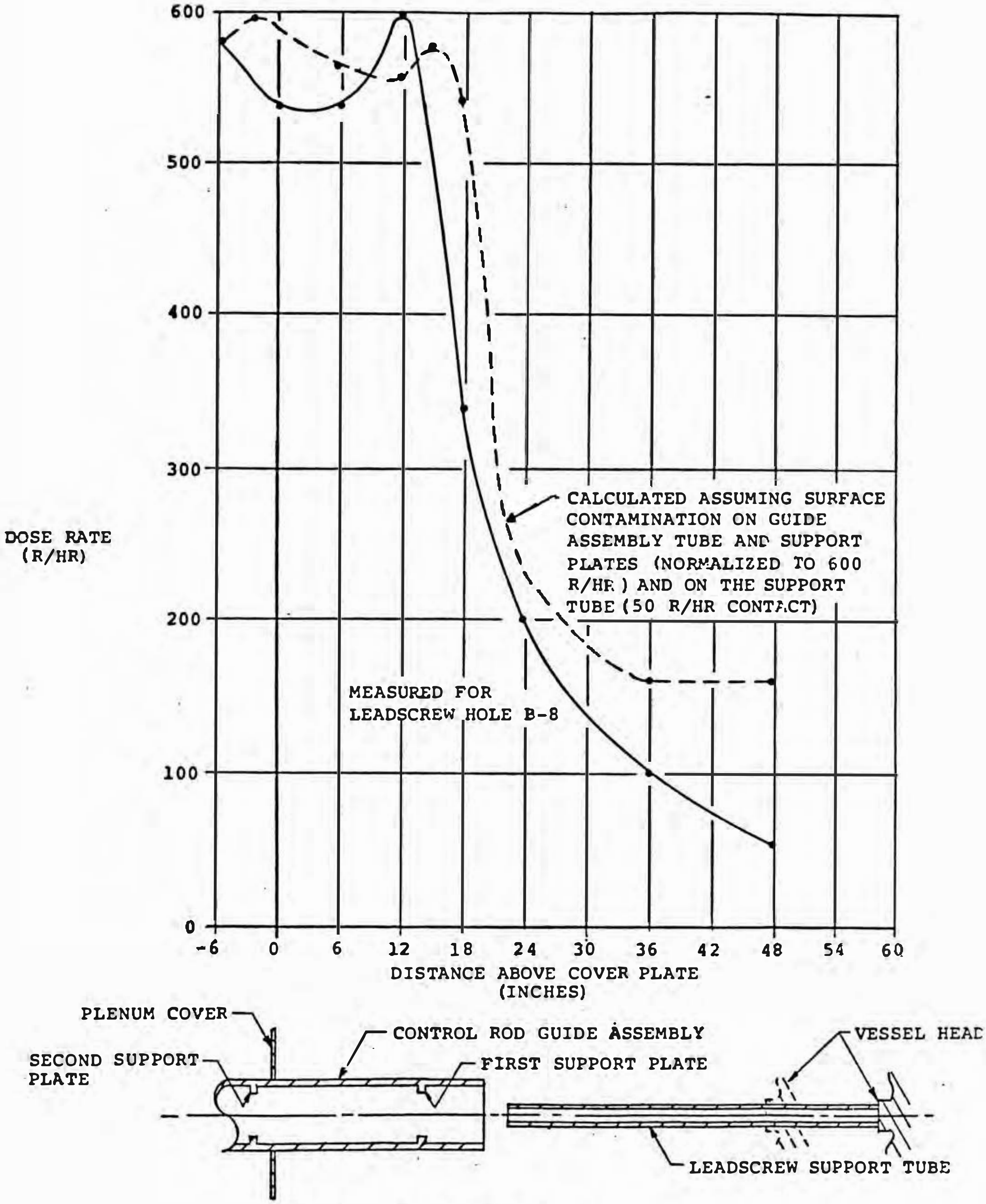
MPR ASSOCIATES, INC
WASHINGTON D.C. 20036
F-74-01-11
REV. 1, 1/25/83



CALCULATED PLENUM DOSE RATES
AT LOCATION E-9 ASSUMING SURFACE CONTAMINATION

FIGURE 12

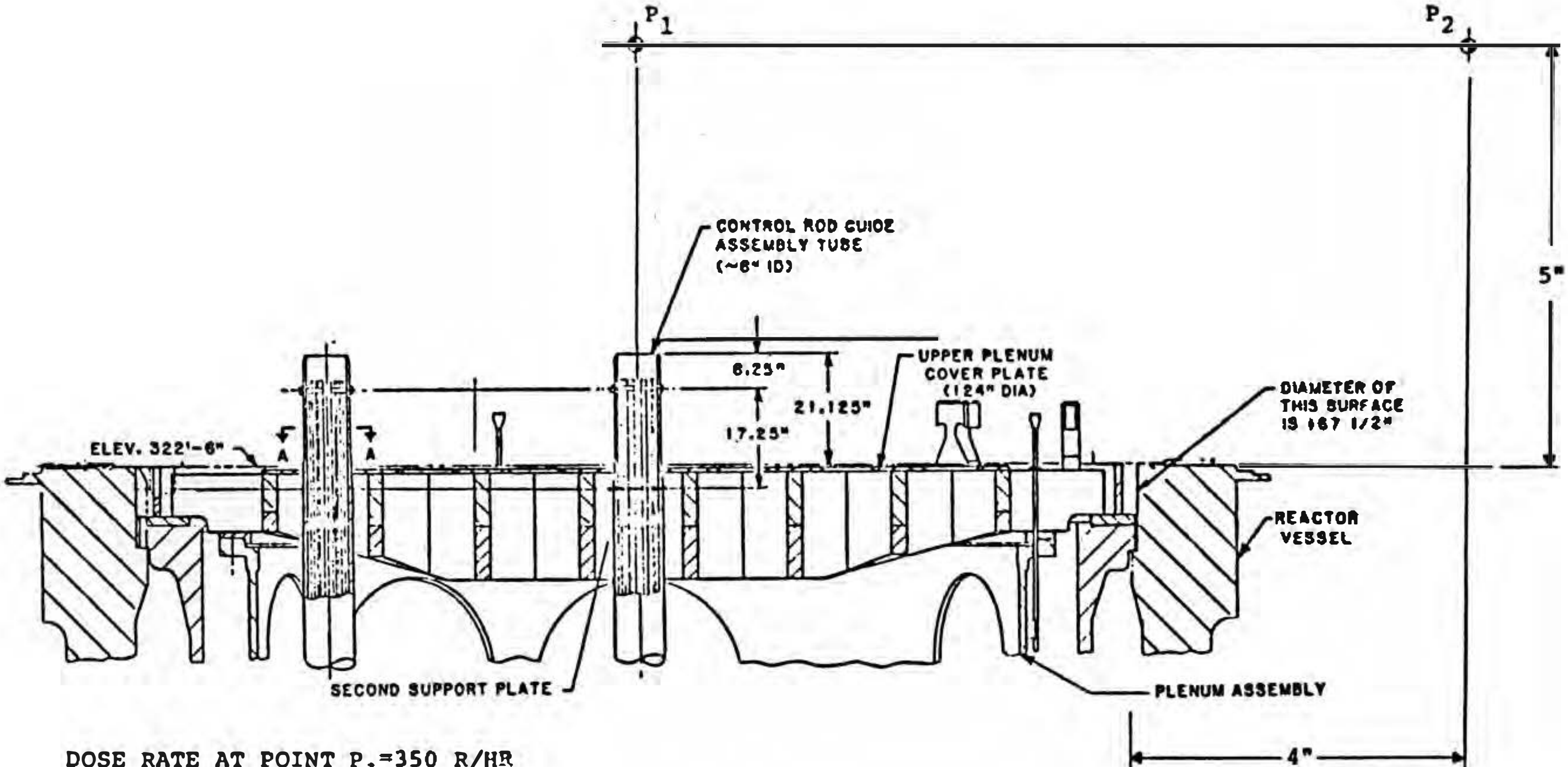
MPR ASSOCIATES, INC
WASHINGTON D.C. 20036
F-74-01-12
REV.1, 1/25/83



CALCULATED DOSE RATES
AT LOCATION B-8 ASSUMING SURFACE CONTAMINATION

FIGURE 13

MPR ASSOCIATES, INC
WASHINGTON D.C. 20036
F-74-01-13
REV. 1, 1/25/83



DOSE RATE AT POINT P_1=350 R/HR
DOSE RATE AT POINT P_2 =75 R/HR

ASSUMING:

(1) PLENUM COVER HAS SURFACE CONTAMINATION EQUIVALENT TO A CONTACT DOSE RATE OF 50 R/HR

(2) CONTROL ROD GUIDE ASSEMBLY COMPONENTS HAVE SURFACE CONTAMINATION EQUAL TO THAT INFERRED FROM QUICK SCAN MEASUREMENTS I.E. A SUFFICIENT AMOUNT TO GIVE A DOSE RATE IN WATER OF 600 R/HR

PLENUM COMPONENT DOSE RATES
WITH VESSEL HEAD REMOVED

FIGURE 14

MPR ASSOCIATES, INC
WASHINGTON D.C. 20036
F-74-01-14
REV. 1, 1/25/83

ATTACHMENT VIII-TWO

MPR ASSOCIATES, INC.

March 4, 1983

Mr. J. C. Devine
GPU Nuclear
TMI Nuclear Station
Post Office Box 480
Middletown, Pennsylvania 17057

Subject: TMI-2 Quick Scan Results

Dear Mr. Devine:

As we discussed during the TAAG meeting of February 23-25, 1983, the MPR and GPUN dose rate estimates for various head removal operations have been quite consistent. The one discrepancy, however, was the "planning basis" dose rate, and the purpose of this letter is to resolve this discrepancy.

The "planning basis" dose rate is at a location four feet beyond the periphery of the reactor vessel inside diameter, and five feet above the vessel flange. Our original estimate for this dose rate was about 75 R/hr, while GPUN estimated this dose rate to be in the range of 6 to 12 R/hr. By comparison, a dose rate of 8.5 R/hr was employed by Bechtel in developing plans for head removal.

We have reviewed the reasons for the discrepancy, and discussed them last week with the GPUN personnel involved in performing the dose rate calculations. Based on these discussions and other calculations we have performed, we conclude that the dose rate is most likely to be about 40 R/hr. The reasons for the differences between this value and original values estimated by MPR and GPUN can be briefly summarized as follows:

° The original MPR calculations of dose rates from the 69 control rod guide assemblies which protrude above the plenum cover plate did not include any self-shielding effects - i.e., the shielding provided to the central assemblies by the assemblies near the edge of the plenum cover plate. When this self-shielding is taken into account we estimate the dose to be about 25 R/hr.

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- The GPUN calculation accounted for self-shielding by "homogenizing" the 69 control rod guide assemblies; i.e. by assuming that all the metal of the assemblies is spread uniformly over a cylinder which has the same height as the guide assemblies, and an outside radius equal to the distance from the plenum centerline to the outermost guide assembly. We consider that this over estimates the self-shielding effects, since it does not account for streaming which occurs through the spaces between adjacent control rod guide assemblies. This results in an underestimate of the dose rate.

- Neither the MPR or GPUN calculations accounted for the dose rate contributions from components located below the plenum cover plate but above the vessel water level. We understand the water level will be on the order of 12 inches below the vessel flange to ensure that no overflow occurs. Under this condition, the deep beams of the plenum assembly are exposed, and we estimate they could contribute about 10 to 15 R/hr to the dose rate. Thus, the total estimated dose rate could be about 40 R/hr, i.e. 25 R/hr from plenum components above the cover plate and 15 R/hr from components below the cover plate.

In summary, our current best estimate for the "planning basis" dose rate is 40 R/hr. The additional under head examinations which are to be carried out this month will provide additional information with regard to expected dose rates. Once agreement is reached on the expected dose rate, Bechtel can then proceed with a review of current head removal procedures to determine if any changes are needed to accommodate the higher dose rate.

Please call if you have any questions with regard to this letter.

Sincerely,

David G. Strawson

David G. Strawson

cc: H. Burton, DOE
V. Fricke, GPUN
W. Hamilton, TAAG

IX. Refueling Canal Clean-up Systems

Refueling canal cleanup was considered as a part of the broader question of maintaining water quality in the reactor vessel and both canals during defueling. TAAG notes that significant progress has been made in this regard during the period of this report. Major actions have been: (1) the development by GPU/Bechtel of source terms for soluble radioactivity and suspended solids for purposes of system design; (2) conceptual design of cleanup systems for the reactor vessel and the canal waters based upon these source terms; and (3) the formation of a Debris Defueling Working Group by EG&G/TIO.

The source terms for soluble radioactive species are dominated by ^{137}Cs; for this nuclide a continuous appearance rate of 2 Ci/day was selected, with occasional radioactivty spikes of 20 Ci contemplated. The 2 Ci/day continuous rate for ^{137}Cs is consistent with previous TAAG recommendations, as is the 20 Ci spike estimate. GPUN/Bechtel has estimated that some 32,000 lbs of debris is located in the reactor vessel which is of a size range sufficiently small to be suspended during defueling operations. However, the fraction of this inventory which will in fact be suspended during defueling will depend upon the extent to which the debris bed is agitated. GPUN/Bechtel expects that not more than 1% of the suspendible debris inventory would be dispersed by any defueling operation. Moreover, rapid settling of a large fraction of the suspended material can be expected; ignoring settling of any of the 320 lbs. of suspended material thus results in significant conservatism. GPUN/Bechtel has shown that reduction of suspended solids concentration corresponding to 320 lbs. of debris to an acceptable level of clarity (1 ppm) can be realized in reasonable periods of time for various contaminated volume configurations if a processing rate of 400 gpm is employed. Although the estimates assume perfect filtration of the solids, this factor is compensated by neglect of settling within the contaminated volume.

A sintered porous metal filter system had been described which seems to be particularly promising; TAAG endorses that development of this concept.

An approach which is similar (and likewise reasonable has been made to establish source terms for use in the design of the cleanup systems for the refueling canal and spent fuel pool. Based upon this approach, conceptual design of the cleanup systems has progressed.

TAAG recommends that the present conceptual design efforts continue.

In addition to the development of systems designed to maintain water quality in the reactor vessel and pools, efforts are also underway to remove the debris within the core region. Efforts in this regard have been initiated both by EG&G/TIO. TAAG studies are being conducted at Westinghouse Hanford and Burns & Roe. EG&G/TIO has formed a Debris Defueling Working Group for this purpose.

The TAAG study at Burns and Roe will consider the interrelationships between the new water cleanup and core debris handling systems and the other water treatment systems installed in the plant. This study will include consideration of the necessary independence of the various systems. Although interaction of the participants is encouraged, TAAG concurs with the continued independent development of the individual systems at this stage of development.

TAAG will continue to provide assistance and advice in the development of the systems.

X. Man-Rem Dose Re-Assessment

During the period of this report GPU/Bechtel presented to TAAG the results of their recent reassessment of the predicted Man-Rem exposure associated the TMI-2 clean up. This reassessment was performed using the assumption that the "Base Cost Program Reassessment" funding level would exist. In addition, the clean-up activities covered by this dose reassessment end with plant activity levels comparable to other operating nuclear plants and do not include decommissioning or reconstruction activities.

The results of this reassessment predict a Man-Rem exposure in the range of 16,000 to 28,000 Man-Rem (See table I). GPUN's previous (October 1981) estimates was 9,300 to 24,000 Man-Rem and the PEIS estimate in March 1981 was 2,000 to 8,000 Man-Rem. The increase in predicted exposure is a result of the additional information obtained during the calendar year 1982 clean-up activities.

Although the results of this reassessment represent a best estimate evaluation based upon the information available at the end of 1982, it is recognized that there still exists many unknown which could result in a significant change in the predicted exposure levels. In particular these unknowns include;

1. The effect of the recently discovered plate-out of Cesium in the pressure vessel head region has not been included in this study.

2. The exposure estimates are sensitive to the success and exposure cost to decontaminate the 282' elevation. A large uncertainty exists in predicting these values.

3. The reassessment assumes that the defueling will follow the predicted path. If difficulties occur, such as a warped plenum, additional exposure would be expected.

4. The lower regions of the core have not been explored and unanticipated damage may require further in-plant work.

Based upon the uncertainties inherent in this study, some of which are identified above, it is the judgement of the TAAG members that the actual Man-Rem estimate for the plant clean-up might be significantly greater than the value obtained from this study-as much as two to four times the amount in the GPU/Bechtel study.

TABLE I

TMI-2 PROGRAM REASSESSMENT
MAN-REM ESTIMATE
JANUARY 1983

MAN-REM SUMMARY

	SUBTOTAL	ASSESSED RANGE
Decontamination	9826	6800 - 12800
Reactor Disassembly and Defueling	4182	3000 - 5500
Radioactive Waste Management	1210	850 - 1600
Support Facilities and Systems	135	100 - 180
Plant Stability and Safety	1226	850 - 1600
Radiological Controls	3317	2300 - 4350
Expended Thru 12/31/82	1467	1467
Total	21363	16000 - 28000

XI. Addenda to the Third TAAG Report

The Third TAAG report covering the period of July - September 1982, discussed in section V.B.II.b. the testing of the Polar Crane Rigging. The followng Newport News Shipbuilding and Dry Dock Company inter-office memo dated 2/1/83 revises the TAAG recommendations relative to testing.

NN 26 (3321120)

INTER-OFFICE COMMUNICATION

NEWPORT NEWS SHIPBUILDING AND DRY DOCK COMPANY

A Tenneco Company

TO: W. H. Hamilton | **FILE NO.** TMI-2

FOR: Information | **DATE** 2/1/83

FROM: E. F. Sise, Jr.

SUBJECT: TAAG Research and Recommendations - Change of Recommendation for Load Testing Rigging for Reactor Vessel Closure Head Removal

Reference:
(a) TAAG Research and Recommendations Pertaining to TMI-2 - Reactor Vessel Closure Head Removal - Progress Made During July - September Meeting Period

Paragraph II.2 of reference (a) states, "This rigging should be tested using a test load which is one and one-half times the weight of the reactor vessel head, or is at least as great as the crane's static test load. If practical, this testing may be conducted in conjunction with the static load testing of the crane using the missile shield hold-down bolts." Further investigation has revealed that it may not be practical to adapt the closure head rigging to the missile shield hold-down bolts for load testing. Minor items, only, may be removed from the containment for load testing to one and one-half times the weight to which they will be subjected during the lifting operation.

Based on the above, Paragraph 11.2 of reference (a) is rewritten as follows:

"2. Rigging. TAAG recommends that the rigging be visually inspected for defects such as abnormal wear, distortion, cracks or other degradation of load bearing parts and trial fitted to the crane hook prior to use to ensure that the equipment is in good condition and that all points fit and operate as designed. This rigging should be tested, either statically or dynamically in conjunction with, and using the same test loads as those used for the crane load test(s). Following the load test(s) and preceeding the reactor vessel closure head lift, the rigging should be re-subjected to, at minimum, a thorough visual inspection. Any defects found during this inspection should be compared with the condition found during the initial inspection to determine if the defects were caused by load testing. If any defects which could adversely affect the load-carrying capability of the rigging are found, they should be repaired and the rigging should be re-tested and reinspected. Upon satisfactory final inspection, the rigging is considered acceptable for use."

EFSise Jr

E. F. Sise, Jr.

GWTucker:nmg