R0260

EG&G Idaho, Inc.
PO Box 88, Middletown, PA 17057

File 74-01
make New File "8th TAAG Report"

August 8, 1984

Distribution

TRANSMITTAL OF EIGHTH TAAG REPORT - Hmb-173-84

Dear Sirs:

Enclosed herewith is the eighth report of the Technical Assistance and Advisory Group (TAAG). This report covers the TAAG activities during the period from December 1, 1983 to April 1, 1984.

Very truly yours,

H. M. Burton, Manager
Technical Information & Examination Program

mrr

Enclosure:
As Stated

cc: J. O. Zane, w/o enclosure

Distribution

TAAG
R. S. Brodsky
N. M. Cole
E. A. Evans/F. J. Leitz, Jr.
W. H. Hamilton
T. A. Hendrickson/C. W. Hess
A. P. Malinauskas/E. D. Collins
E. F. Sise/R. L. Earl

U.S. DOE
W. W. Bixby
F. E. Coffman
J. P. Hamric
D. J. McGoff
F. A. Ross
F. L. Sims

EG&G Idaho, Inc.
R. H. Beers
W. A. Franz
K. C. Sumpter

GPU Nuclear & Bechtel
J. J. Barton
P. R. Clark
J. C. DeVine, Jr.
R. H. Fillnow
R. L. Freemerman
B. K. Kanga
C. A. Negin
R. L. Rider
K. DeConte (6)

Distribution (Continued)

U.S. NRC
P. Grant (4)
B. Snyder
R. Weller

EPRI
J. Taylor
A. Roberts
F. L. Schwartz

Other
Dr. J. C. Fletcher, Univ. of Pittsburgh

DRAFT

EIGHTH REPORT

OF THE

Technical Assistance and Advisory Group (TAAG)

Three Mile Island Unit 2 (TMI-2)

For the Period December 1, 1983, to April 1, 1984

TAAG MEMBERS

W. H. Hamilton, Chairman
R. Brodsky
N. M. Cole
E. A. Evans/F. J. Leitz, Jr.
E. F. Sise/R. L. Earl
T. A. Hendrickson/C. W. Hess
A. P. Malinauskas/E. D. Collins

Participating

F. A. Ross, DOE
H. M. Burton, EG&G
R. Weller, WRC
A. Roberts, EPRI

CONTENTS

		PAGE
1.0	INTRODUCTION	1
1.1	SCOPE AND APPROACH	2
1.2	SUMMARY OF RECOMMENDATIONS FROM THE EIGHTH TAAG REPORT	2
2.0	SYSTEM DECONTAMINATION	4
2.1	ASSESSMENT OF TECHNICAL PLANS TO DECONTAMINATE PIPING SYSTEMS	4
2.2	DECONTAMINATION SKID	4
2.3	CADD	5
2.4	PRIORITY	6
2.5	CHARACTERIZATION	6
3.0	USE OF THE ROBOT	8
4.0	DEFUELING PLANS	11
4.1	CURRENT CONSIDERATIONS AND RELATED BACKGROUND ON SHREDDING MACHINES	11
4.2	TMI-2 DEFUELING CONCEPT WHICH ALLOWS THE USE OF LONG HANDLED DEFUELING TOOLS	16
5.0	DATA ACQUISITION BEFORE AND AFTER PLENUM REMOVAL	18
5.1	DATA TO SUPPORT MECHANICAL DISASSEMBLY	19
5.2	DATA TO SUPPORT SELECTION OF DEFUELING METHOD	21
5.3	DATA TO SUPPORT PROCEDURE DEVELOPMENT	22
5.4	DATA TO SUPPORT TOOL DESIGN	23
5.5	DATA TO SUPPORT POST-DEFUELING OPERATIONS	25
6.0	REVIEW OF PEIS SUPPLEMENT	27
7.0	CLEANUP SYSTEM FILTERS	31
8.0	DISPOSITION OF RECOMMENDATIONS IN SIXTH AND SEVENTH TAAG REPORTS	32

ATTACHMENT

TPO/TMI-135, TMI-2 Defueling Concept Which Allows the Use of Long Handled Defueling Tools

SECTION 1.0

INTRODUCTION

In a letter dated February 2, 1984, Mr. B. K. Kanga, Director, Three Mile Island Unit 2 (TMI-2), requested that the Technical Assistance and Advisory Group (TAAG) address the following matters for the period ending April 1, 1984:

1. Assess the technical plans being developed by GPUN, underwritten by EPRI, to decontaminate piping systems.

2. Assess the plans for use of the robot to be delivered in 1984 for use on the 282' level.

3. Keep current on Westinghouse defueling plans and provide technical comments as specific proposals are made. Research what experience there is in reprocessing and other facilities with cutting up of fuel assemblies.

4. As a carryover from the last period of TAAG activities, evaluate the data required before and after plenum lift. Include in the evaluation identified uses of the data, concepts for acquiring the data and methods for evaluation.

5. Perform a review of the NRC supplement to the PEIS and provide comments.

6. Review the design and testing of filters for the canal water clean-up system.

1.1 SCOPE AND APPROACH

This report responds to these work items. One section of the report addresses each of the six work items. The recommendations are summarized below.

In the last section of this report the GPU Nuclear responses to the Sixth and Seventh TAAG Report recommendations are tabulated.

1.2 SUMMARY OF RECOMMENDATIONS FROM THE EIGHTH TAAG REPORT

1. System Decontamination

   TAAG recommends: 1) that work on a decontamination skid be terminated and an experienced subcontractor be used, 2) that CADD decontamination-related work be terminated, 3) that priority for AFHB work be re-determined, and 4) that procedural documents be prepared with users in mind.

2. Use of the Robot

   The use of the EPRI-sponsored robot in containment was investigated. It is concluded that there are several tasks that can be performed on Els. 282' and 305'. These are presented in Section 3.0, along with suggested enhancements to the robot.

3. Defueling Plans

A defueling concept that incorporates a dry transfer from the reactor vessel to the deep end of the transfer canal is recommended. This concept allows the use of long handled tools and requires a minimum of development. Refer to Section 4.0.

4. Data Acquisition Before and After Plenum Removal

The data needs have been grouped into five categories. Seventeen individual data-collection tasks have been placed into these categories. Details are presented in Section 5.0.

5. Programmatic Environmental Impact Statement (PEIS) Supplement

TAAG recommends that the PEIS supplement include comparisons of the exposure in the TMI-2 cleanup to other cases of radiation exposure to put the TMI-2 recovery program in a more realistic perspective.

6. Cleanup System Filters

TAAG makes no recommendations, since the filter test has not yet been completed.

SECTION 2.0

SYSTEM DECONTAMINATION

2.1 ASSESSMENT OF TECHNICAL PLANS TO DECONTAMINATE PIPING SYSTEMS

TAAG was requested to review and to assess the technical plans being developed by GPU Nuclear, underwritten by EPRI, to decontaminate piping systems. The specifics of the technical plans have been made available in the form of presentations to TAAG and planning studies prepared on the topic.

In general, TAAG finds that the program to date is technically sound. It is well thought out, thorough, and comprehensive. TAAG finds the efforts to identify acceptable decontamination solutions for the TMI-2 types of contamination especially well done and informative. However, there are some aspects of the approach that TAAG believes are in need of modification.

2.2 DECONTAMINATION SKID

An integral part of the current program is the engineering, design, construction, and development of a decontamination skid to be used in the plant. This effort is not necessary. It is possible to contract with an outside firm for decontamination services, as a number of suppliers already have mobile decontamination systems. Not only will this eliminate another research, design, and development (RD&D) program, it will yield a more usable product for EPRI. This would give EPRI controlled data about the operation and performance of a commercially available system. GPU Nuclear would save the RD&D, construction, testing, and training costs associated with developing a decontamination system for this project.

TAAG recommends that all work on the decontamination skid be terminated and that the actual decontamination of the Auxiliary and Fuel Handling Buildings (AFHB) be performed by one or more contractors who specialize in providing this service.

2.3 CADD

Several presentations to TAAG mentioned the use of a computer-aided design and drafting (CADD) system as a tool to support the characterization and decontamination efforts.

The analytical functions of a CADD model are not necessary. First, the complexity and precision of the proposed calculations (pipe volume/unit length, flow rate required to achieve a target fluid velocity, and the time required to flush a target volume of water through the line) do not demand computer assistance. To achieve the accuracy required for the proposed calculations, tables showing the relationships between line size and flow rate, and between line size and water volume per unit length of pipe would enable an engineer equipped with piping drawings to generate the required data quickly. Second, developing, debugging, and utilizing the software required to perform these analyses are time-consuming and costly tasks. The software associated with 3-D CADD modeling is not readily amenable to add-on analytical programs.

TAAG recommends that CADD work planned in association with decontamination of piping systems be terminated.

2.4 PRIORITY

Based on the recent effort to develop a coherent plan for the TMI-2 Recovery Project, it is not clear that the decontamination of AFHB systems should be given high priority at this time. Technical Plan TPO/TMI-115, TMI-2 Program Strategy, clearly identifies three phases of the project. It is not clear that very much effort should be expended in the AFHB until Phase III "Cleanup", or possibly in "Future Work". The priority for the AFHB decontamination effort should be re-determined based on this overall strategy for the project. EPRI's interest in the decontamination project was in the data acquisition potential of a carefully designed decontamination effort. It is possible that this can be achieved without complete decontamination of the AFHB.

2.5 CHARACTERIZATION

The approach to characterization of the AFHB is needlessly complicated by the use of arbitrary designations for cubicles. For instance, instead of referring to the "A" Makeup Pump cubicle, the cubicle is referred to as AX007. While this approach may have advantages for data processing, it has practical limitations as far as implementation is concerned. The operations personnel at the site are used to referring to cubicles by their function or by the equipment housed within them. Redesignating cubicles with arbitrary alphanumeric labels is certain to add a layer of confusion to an already elaborate task. When this approach was first presented to TAAG (Planning Study TPO/TMI-032, AFHB Cubicle/System Characterization), it seemed reasonable in the context of that study. However, that methodology is now being carried forward into the decontamination phase of the effort.

In addition to redesignating the cubicles, at least one drawing has been prepared with renumbered column lines. This is certain to cause confusion, and all drawings prepared for this effort should use existing column line designations.

TAAG believes that all documents prepared for this effort should be "user friendly" by utilizing existing terminology for plant features.

SECTION 3.0

USE OF THE ROBOT*

TAAG was requested to assess the plans for the use of the robot(s) on El. 282'6" of the reactor building. Three robots have been purchased. The first of the robots is currently undergoing proof of principle testing on site. The TAAG review of the use of the robot was assisted by presentations to TAAG by site personnel, who described the physical characteristics and capabilities of the robots. Actual plans for the use of the robot have not been formalized by GPU Nuclear. So, TAAG developed an independent review of the uses of the robot.

The TAAG review began with an evaluation of the areas of the reactor building basement that are accessible to the robot. The robot has 120 feet of power and control cable. If the robot is lowered into the basement via the access hatch, as currently planned, this cable will permit the robot to access most of the basement. However, the length of the cable is not the limiting factor as far as access is concerned. Three areas were identified that will prohibit the movement of the robot.

The first is in the southwest quadrant of the building near the leakage transfer pumps (WDL-P-9A&9B). The shield wall separating these pumps from the leakage coolers (WDL-C-1A&1B) comes within 31" of the reactor building liner.

*Note: This section is excerpted from TPO/TMI-134. Readers are urged to review that TAAG Planning Study in detail for further information on this topic.

Due to the shape of the reactor building at that elevation, the actual floor space is only 19". Since the robot is 30" wide, it is impossible for the robot to get past this area in the building.

The second area that will restrict access of the robot is located in the northeast quadrant of the reactor building near the steam generator hot drain cooler (SV-C-1). This area houses three pumps and their associated piping and conduit, which present an impenetrable barrier to the robot. The overhead in this area is also congested by a number of large bore lines that penetrate the reactor building wall.

The third area that will limit the access of the robot is located to the southwest of the building center. The doorways to the reactor coolant drain tank (WDC-T-3) cubicle and through the secondary shield wall both have locked security doors. The first stage of the robot development program does not include robotic arms or manipulators to open doors. Hence, the robot will not be able to access these areas until a manipulator can be developed to open these doors.

If a manipulator to open the security doors is developed for the robot, and if the doors still function after 3 years of immersion in water, the robot can enter the reactor coolant drain tank cubicle and the secondary shield wall. However, secondary obstructions restrict access in these areas.

If the robot's tires are changed to permit it to climb and to descend stairs, the robot can utilize the installed platform inside the secondary shield walls to gain access to the "B" steam generator. If the robot could climb and descend stairs, it would offer another way for the robot to gain access to the basement. It would then be possible to drive the robot down the open stairway into the west quadrant of the building. Although this approach would use approximately 60' of cable instead of the 20' of cable necessary to lower the robot through the hatch, the cable length is still not limiting.

Independent of the considerations of movement restrictions, a list of tasks to be performed by the robot was developed. In the basement, four general types of efforts were identified: visual data acquisition, radiological examinations, decontamination data acquisition, and special tasks.

TAAG has suggested that significant amounts of decontamination could be achieved by leaching cesium from the concrete by keeping the surface wet with clean water. One of the ways to achieve this is to install a set of spray nozzles directed at the surface. The robot enables this experiment to be performed inside the impingement shield walls, which would isolate the test area from other radiation sources.

SECTION 4.0

DEFUELING PLANS

TAAG reviewed two issues related to TMI-2 defueling. The first is an assessment of shredding machines. The second is an evaluation of an alternate approach to defueling. Each issue is discussed below.

4.1 CURRENT CONSIDERATIONS AND RELATED BACKGROUND ON SHREDDING MACHINES

In January, the Westinghouse candidate plans for defueling were presented to TAAG. The review of these plans by GPUN/Bechtel during the next two months was followed as well.

TAAG members were asked to investigate any previous experience in spent fuel reprocessing with the use of shredding machines and/or other devices for disintegration of fuel rods. Concerns of interest were: (1) feasibility for use at TMI-2, (2) potentials for fuel/fission product release and pyrophoric reactions, and (3) potential particle size generation.

TAAG obtained the following information:

(1) The predominant type of machine used to disintegrate the fuel rods/assemblies in preparation for leaching of the oxide fuel from the cladding hulls into nitric acid solutions is a shearing device which is operated in an air atmosphere. These machines are generally too large and too complicated for adaptation to use at TMI-2.

(2) Comparatively little experience has been obtained with a shredding machine, and only then with mock-up fuel rods rather than real irradiated fuel. (Shredders have been used more extensively for volume reduction of contaminated wastes.) These operations have also been done in an air atmosphere, and varying degrees of success have been obtained toward preparing debris that is satisfactory for leaching. However, from all indications, disintegration into particle sizes that are adequate for slurry transfer can be achieved. In the trials, some evidence of difficulty was encountered in feeding whole assemblies or other large pieces into the shredding machine. Also, indications of pyrophoric reactions (sparks, etc.) were observed.

(3) Particle size distribution from shearing light water reactor (LWR) fuel rod assemblies was reported in NUREG/CR-0866, prepared by Davis, West, and Stacy at Oak Ridge National Laboratory. Of particular interest in that report is the table (shown herein as Table 1) that indicates that the portion of particles which are <10 μm in size (those that would settle slowly if suspended in water) ranged from 0.056 to 3.71% (wt.). Thus, if all of the TMI-2 core (~120,000 kg) is disintegrated, one might expect from ~70 to ~4500 kg of slowly settling solids to be produced.

(4) With regard to fission gases, releases of 0.05% of the tritium, 1.0% of the krypton-85, and 32% of the carbon-14 were measured while shearing Saxton Reactor fuel rods at the Savannah River Laboratory (SRL Report No. DPST-AFCT-77-1-2).

(5) Potential increases in dissolved activity were measured in a series of experiments in which fully irradiated fuel fragments (two different size fractions - one in which the particles were 1000 μm or larger and one in which the particles were 300 μm or larger) were leached with water containing 3270 ppm of boron and 1050 ppm sodium (pH = 8.0). This work was done by Mitchell, Goode, and Vaughen at Oak Ridge National Laboratory and was reported in ORNL/TM-7546. Estimates of the percentage of fuel and fission product inventory were tabulated (these data are shown herein as Table 2). Of particular interest was that <0.5% of the radiocesium and <0.022% of the transuranics were dissolved.

In their defueling plans, Westinghouse proposed that remotely operated tools with a wide variety of "end-effectors" (shovels, scoops, saws, etc.) would potentially be needed, and that they favored a clamshell device to pickup and place the core material into the shredder. However, appropriate tooling cannot be accurately specified and designed until the character and configuration of the material underneath the rubble bed is determined. A number of data acquisition methods (probing, sampling, instrumental mapping, etc.) have been considered.

Table 1. Ranges of mass distribution by particle size for irradiated fuels

Particle sieve size (μm)	Weight in sizes less than the indicated value (%) Minimum	Maximum	Particle sieve size (μm)	Weight in sizes less than the indicated value (%) Minimum	Maximum
1	0.000038	0.12	140	7.56	36.2
2	0.00045	0.39	160	8.30	39.3
4	0.0042	1.12	180	8.99	42.1
6	0.014	1.96	200	9.65	44.7
8	0.031	2.83	250	11.2	50.2
10	0.056	3.71	300	12.5	54.7
15	0.15	5.88	350	13.7	58.5
20	0.30	7.95	400	14.9	61.7
30	0.71	11.7	450	15.9	64.5
40	1.26	15.1	500	16.9	66.9
50	1.90	18.1	600	18.7	70.9
60	2.61	20.8	700	20.3	74.1
80	4.18	25.6	800	21.7	76.7
100	5.87	29.6	900	23.0	78.9
120	6.77	33.2	1000	24.2	80.7

Source: Table II, NUREG/CR-0866

Table 2. Estimated percentage of fuel inventory leached from the fuel over time

Substance	Time (y) 1	2	3	4
Uranium	0.009-0.087	0.10-0.11	0.011-0.13	0.011-0.14
Gross alpha[a]	0.004-0.022	0.005-0.026	0.005-0.030	0.005-0.036
^{3}H	0.13-0.64	0.18-0.70	0.21-0.74	0.24-0.76
^{90}Sr	0.066-0.12	0.072-0.15	0.076-0.16	0.079-0.18
^{106}Ru	0.016-0.093	0.017-0.10	0.018-0.011	0.018-0.11
^{125}Sb	0.30-1.2	0.32-1.5	0.33-1.6	0.34-1.8
^{129}I	5.6-7.4	5.7-7.5	5.7-7.6	5.8-7.6
^{134}Cs	0.29-0.41	0.30-0.46	0.31-0.49	0.32-0.52
^{137}Cs	0.28-0.46	0.30-0.51	0.30-0.55	0.31-0.57
^{144}Ce	0.012-0.066	0.013-0.074	0.013-0.079	0.014-0.083
^{154}Eu	0.012-0.063	0.014-0.073	0.015-0.080	0.016-0.086

[a]Plutonium.

Source: Table 4, ORNL/TM-7546

4.2 TMI-2 DEFUELING CONCEPT WHICH ALLOWS THE USE OF LONG HANDLED DEFUELING TOOLS

TAAG reviewed the various plans being considered for defueling the reactor once the head and plenum were removed. These plans included a concept for using robot-like equipment with various end-effectors to break up and remove fuel material. Also included were schemes for shredding the fuel material and then slurrying the material into canisters. The concepts were used in conjunction with the canal full of water.

While these plans were all being considered, a "dry" defueling concept was conceived wherein the fuel transfer from the vessel would be done in a shielded cask, with the canal dry. The concept features long handled tools, a transfer cask for 130" fuel canisters, a shield platform above the reactor vessel, an essentially dry transfer canal, and a lowered level in spent fuel pool A.

Advantages of this concept include avoidance of major development projects, depth of water amenable to manual operation of tools, reduced volume of water to be processed, and lower ambient radiation fields where personnel will be working.

This concept has been published as a separate TAAG Planning Study* and is an Attachment to this report.

---

*TAAG. June 1984. TMI-2 defueling concept which allows the use of long handled tools. TAAG Planning Study TPO/TMI-135. Middletown, PA: GPU Nuclear.

After weighing the various concepts, TAAG has concluded that the "dry" defueling has many advantages, as noted in TPO/TMI-135, and recommends that the "dry" defueling concept be adopted for planning the detailed defueling procedures and equipment.

SECTION 5.0

DATA ACQUISITION BEFORE AND AFTER PLENUM REMOVAL

In a letter dated September 20, 1983, from B. Kanga, Director, TMI-2, TAAG was requested to provide an assessment of data acquisition needs before and after plenum removal. The assessment was to identify the uses for the data, concepts for acquiring data, and methods for evaluation. In addition, to the extent practical, TAAG was to provide estimates of what the data might show and of the impact it would have on fuel removal equipment and fuel removal methods.

The following discussion summarizes the results of TAAG's assessment. The information used for this assessment includes GPU Nuclear's studies and meetings conducted prior to March 1984. The format of this report is to individually list and discuss each item of data. The order of these items as they appear on the list is not significant and does not reflect on the relative importance of the individual item. The testing outlined in this report is that considered necessary to support defueling of the reactor and does not include taking data in order to better understand the engineering or scientific aspects of the accident.

The data needs have been grouped into five categories:

1. Data to support mechanical disassembly before defueling operations begin.

2. Data to be used for selecting the reactor defueling method.

3. Data to support development of the specific procedures for the defueling operation.

4. Data to support development of the tools and fixtures needed for defueling.

5. Data to support post-defueling operations, such as shipping.

Recognizing the large number of variables that affect the test schedule at TMI-2, no attempt is made in this report to identify a specific schedule for the tests discussed. Experience has shown that almost all data obtained to date have had a significant impact on the defueling plans. This past experience emphasizes the advantage of obtaining this data as early as possible.

5.1 DATA TO SUPPORT MECHANICAL DISASSEMBLY

1. DATA: Determine if the head bolts are free.
   METHOD: Partially de-torque the head bolts.
   IMPACT: If bolts are frozen, special procedures and tools will have to be developed to permit their removal. (This effort is complete.)

2. DATA: Measure the clearance between the plenum and the pressure vessel wall.

METHOD: Manual remote probing under video observation.

IMPACT: If it is determined that adequate clearance does not exist, it may be necessary to develop special tools and procedures for the removal of the plenum.

3. DATA: Determine the positions of the vent valves.

METHOD: Remote visual observation.

IMPACT: If the valves are open, they could interfere with the withdrawal of the plenum. If this is the case, it will be necessary to develop tools and procecures to correct the interference.

4. DATA: Determine if debris has accumulated between the plenum and the core support assembly.

METHOD: Remote visual observation.

IMPACT: If debris has accumulated between the assembly and the plenum, the debris could interfere with the lift of the plenum. If there is a significant accumulation, it may be necessary to remove the debris prior to plenum removal.

5.2 DATA TO SUPPORT SELECTION OF DEFUELING METHOD

1. DATA: Determine the visibility that will exist during the defueling operations.

   METHOD: A. Stir the debris bed with a reach rod while observing turbidity and settling rates with a TV monitor.

   B. Measure the particle size distribution of debris samples from the core.

   C. Make observations during the proof testing of the "shredder".

   D. Measure turbidity and settling rates of debris generated during the proof testing of tools to be used to disassemble the core.

   IMPACT: The degree of visibility will influence selection of the defueling procedures and the defueling sequence. It is likely that the data from these tests will emphasize the need to initially vacuum the core region. In addition, the results may indicate that defueling must be performed in small increments, followed by cleanups.

2. DATA: Evaluate the cesium release that will accompany defueling operations.

   METHOD: Perform release rate tests of shredded core samples taken from various regions of the core.

   IMPACT: The results of these tests will provide information required to confirm design and operational planning.

3. DATA: Determine the distribution of 2.96% enriched uranium within the rubble bed.

METHOD: Measure the enrichment of fuel particles contained in the core debris samples that have been obtained for other purposes.

IMPACT: These measurements can be used to support the conservatism in the boron concentration specified for reactivity control.

5.3 DATA TO SUPPORT PROCEDURE DEVELOPMENT

1. DATA: Determine the extent of damage to the lower core support structure and the amount of material located in the lower inlet plenum.

METHOD: A. Insert a remote TV camera between the core barrel and the pressure vessel wall, and into the lower plenum volume.

B. Monitor for the presence of fuel material in the lower pressure vessel volume by taking radiation measurements outside the pressure vessel (gamma monitoring or neutron track counting).

C. Evaluate existing neutron track data.

IMPACT: The results of these measurements/observations will provide advance indication of the extent of damage in the lower core/core structure. This, in turn, will provide a basis to develop tools and establish defueling sequences.

2. DATA: Generate a detailed core map to determine fuel removal sequences and to determine the extent of core change after specific defueling operations, such as vacuuming.
METHOD: Remote TV should be used to map all accessible areas of the core. This effort is underway.
IMPACT: This core map will be necessary to evaluate the effectiveness of the defueling operations. It will be necessary to update the map periodically during the defueling operations.

3. DATA: Determine the amount of debris on the upper surface of the plenum's lower plate.
METHOD: Remote TV inspections at one location.
IMPACT: These data can be used to confirm predictions of the radiation exposure associated with removal and storage of the plenum. Although unlikely, the data would indicate if the plenum has to be flushed prior to removal.

5.4 DATA TO SUPPORT TOOL DESIGN

1. DATA: Confirm, with realistic proof tests using unirradiated fuel, the predicted performance of essential elements of the defueling hardware. Evaluate the performance of proposed tooling in a prototypical environment under expected conditions. These tests should include measurements to evaluate the effect of wear on the reliability and maintainability of tools and transfer systems.

METHOD: A. Evaluate the performances of the defueling tools in a realistic defueling environment.

B. Confirm performance predictions for the defueling water cleanup system (DWCS) canister filters using effluent from the above tests.

IMPACT: These tests are expected to uncover deficiencies that will have to be corrected by equipment modifications or procedural changes.

2. DATA: Determine the material properties of the fuel to be removed.

METHOD: A. Probe the rubble bed with probes, long drill bits, and samplers, at various depths and radial locations, to obtain the properties of the rubble bed (also map its perimeter).

B. Using a remote grapple, remove a section of fuel rod currently on the top of the debris bed.

IMPACT: These tests provide a basis for tool development. Based upon experience to date, it is likely the results of these tests will have a significant impact on tool design and procedure development.

3. DATA: Determine the effort required to separate the upper end fittings from the plenum.

METHOD: A reach rod should be inserted into a leadscrew guide tube, possibly before the head is removed. This tool should have a manual impact hammer at its upper end. Tests should be made to determine if this tool can dislodge upper end fittings.

IMPACT: These tests are expected to show that the upper end fittings can be removed remotely with a relatively simple tool. If this is not the case, it may be necessary to remove the plenum with the upper end fittings attached. In this case, special shielding may be required for the transfer operations and around the storage stand.

5.5 DATA TO SUPPORT POST-DEFUELING OPERATIONS

1. DATA: Evaluate the likelihood that pockets of hydrogen gas have, or could, accumulated in the core volume.

METHOD: Observe (by TV) the surface of the debris bed for the release of gas bubbles while tests that disturb the bed are performed.

IMPACT: It is anticipated that gas pockets will not be found and that it will not be necessary to take any special precautions to prevent hydrogen burning during defueling. (Pyrophoric reactions of metal fines under water are believed to be a result of the accumulation of hydrogen gas within the material bed.)

2. DATA: Determine the anticipated hydrogen generation rate inside loaded fuel canisters.

METHOD: Measure gas generation rates in fuel samples.

IMPACT: These data are necessary to determine if there is a need to provide special features in the container to protect against a hydrogen burn.

3. DATA: Determine the distribution of fuel in the primary system outside the pressure vessel.

METHOD: Radiation monitoring using neutron track devices or collimated gamma detectors.

IMPACT: The data will provide an early indication of the extent of fuel redistribution and the cleanup effort that will be required. It is anticipated it will be shown that fuel is distributed throughout the primary system.

5. DATA: Determine the magnitude and decay of the radiation field associated with core debris samples.

METHOD: Collect laboratory measurements.

IMPACT: These data will predict shielding requirements and waste management impacts.

SECTION 6.0

REVIEW OF PEIS SUPPLEMENT

The attached letter was sent to NRC to comment on the supplement to the PEIS. The intent of this comment was to put in perspective that the TMI-2 clean-up exposure compares favorably with other personnel exposures in typical living conditions.

WILLIAM H. HAMILTON

4601 BAYARD STREET, APT. 307
PITTSBURGH, PENNSYLVANIA 15213
TELEPHONE: 412-683-8826

February 27, 1984

Dr. Bernard J. Snyder, Program Director
Three Mile Island Program Office
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D. C. 20555

Dear Dr. Snyder:

The Technical Advisory Group has reviewed the Nuclear Regulatory Commission's draft Supplement 1 to NUREG-0683, Programmatic Environmental Impact Statement (PEIS) on the Three Mile Island cleanup activities.

Our concerns are centered on the Section 3/3 discussion of health effects. We consider the draft section somewhat misleading and recommend the specific changes discussed below to provide a more realistic and comprehendible focus:

1. NRC's estimate for cumulative occupational radiation doses associated with the Current Cleanup Plan is 13,000 to 46,000 person-rem (Table 3.1).

   To provide some perspective on this cleanup dose, which is expected to occur over a five to ten year period, we recommend that the text include a comparison with several examples of common radiation releases in the area. For instance, (1) persons living in the vicinity of TMI receive approximately 24,000 person-rem each year of additional exposure through the use of natural gas in their homes (Reference 1); and (2) the total exposure to area residents due to potassium-40 in the blood and tissues of their bodies is approximately 43,000 person-rem/year (References 2 through 6). These doses assume a population of approximately 2.16 million within a 50 mile radius of the site. Hence, the annual exposure due to the cleanup will be about an order of magnitude less than the local radiation exposure due to these common sources.

2. In Section 3.3, the fourth paragraph states:

   "....For the minimum-collective-dose case (13,000 person-rem) it is expected that 2 additional fatal cancers would be caused. For the maximum-dose case (46,000 person-rem), 6 additional cancer fatalities would result. Although it is possible to compute a range of probabilities for cancer induction among average individual workers based on the above figures, the results of such a calculation may not bear a close relationship to actual risks since the work force size and collective dose associated with the various tasks can differ by large factors, rendering inapplicable the concept of an average individual worker."

These mortality figures were derived based on a factor of 131 fatal cancers in the exposed workers per one million person-rem.

We recommend the deletion of this discussion. Without considering such factors as the work force size and the collective dose associated with individual tasks, statements such as "6 additional cancer deaths would result" are meaningless. Discussions of licensee administrative controls and the risks to individuals associated with the maximum allowable doses during given time periods (as provided in subsequent paragraphs) present a much clearer picture.

3. In the last paragraph of Section 3.3, the probability of genetic effects among the offspring of the work force should be expressed in terms of increased risk to the individual worker, rather than as a flat number based on 260 genetic effects per one million person-rem.

We appreciate this opportunity to review on the draft Supplement to the PEIS and trust that our comments will be properly considered in the final document.

Sincerely,

William H. Hamilton

William H. Hamilton
Chairman
Technical Assistance and Advisory Grou

WHH/ep

cc: Mr. Bahman K. Kanga, Director
General Public Utilities Nuclear/Bechtel TMI-2

Mr. Harold M. Burton, Manager
EG&G Idaho, Inc.

Mr. Adrian Roberts
Electric Power Research Institute

References: (1) EPA Report EPA-520/1/73-004, pages 29 through 31.
(2) Page 57 of "The Fight over Nuclear Power" by Drs. Bodansky and Schmidt.
(3) EPA 520/1-77-009, Pages 29 and 34.
(4) Dr. Edward Teller, Wall Street Journal, July 31, 1979.
(5) J. M. Smith, Jr., GE, "Natural Background Radiation and the Significance of Radiation Exposure".
(6) National Council on Radiation Protection and Measurement Report Nos. 45 and 56.

SECTION 7.0

CLEANUP SYSTEM FILTERS

The filters in the canal water cleanup system are crucial to the ability of the system to meet the water clarity requirements. Currently, a sintered metal filter has been selected for use in this application. TAAG has been concerned about the ability of these filters to meet the requirements and has been anxious to review the test results. The test data have not yet been obtained due to equipment trouble with the test loop. TAAG will evaluate the test data during the next period.

SECTION 8.0

DISPOSITION OF RECOMMENDATIONS IN SIXTH AND SEVENTH TAAG REPORTS

The dispositions of recommendations in the Sixth and Seventh TAAG Reports are summarized in the tables that follow.

DISPOSITION OF TAAG RECOMMENDATIONS: REPORT # 6

RECOMMENDATION I.D.	STATEMENT	DISPOSITION BY GPU
III.A	The section of the leadscrew support tube should be carefully examined.	Examination is being implemented.
III.C	It is recommended that torque be limited to use of 400-500 ft-lbs to remove CRDM to closure head bolts.	The basis for the recommendation is considered incorrect. If threads are stripped, they can be replaced.
IV.	1) Sample and inspect the plenum debris to determine if it will become suspended during plenum removal.	Debris will be suspended to some extent.
	2) If the debris can be suspended, take steps to clean the plenum or to fix the debris to the plenum.	Debris will be contained.
	3) A small, disposable filtration system may be acceptable to monitor water clarity around the plenum for this evolution.	Current plans require that the DWCS be available prior to defueling. A separate filtration system is not envisioned.
	4) Steps should be taken to pre-assemble the cleanup system on skids, eliminating the need for construction in the canal after plenum removal.	System will be pre-assembled to the extent practicable.
V.A	Defer El. 282'6" decontamination until after defueling.	This is now included in the TMI-2 Strategy Plan.
V.A	Consider reflood of the basement.	Current position is to not reflood.
V.A	Shield floor penetrations.	Has been implemented.
V.C	Isolate El. 282'6" to reduce airborne contamination.	Air flow has been changed for this purpose
V.D	Pathways approach.	This has been partially implemented.
VI.A 1	Use actual weight of plenum.	Will use design weight.
VI.A 2	Include video and load cell monitoring during plenum lift.	Included.

DISPOSITION OF TAAG RECOMMENDATIONS: REPORT # 7

RECOMMENDATION I.D.	STATEMENT	DISPOSITION BY GPU
II.	Permanently installed gamma monitoring equipment is not required in the TMI-2 reactor building.	Concur.
II.	The program for the maintenance, calibration, and use of personnel dosimeters should be reviewed to assure that there are no deficiencies that would reduce the effectiveness of this protection.	Existing program is effective and is monitored to ensure level of performance.
II.	Safety evaluations for operations such as head and plenum lift, defueling, etc., should include an evaluation of what, if any, special radiation monitoring equipment is required.	Concur.
III.	Further analysis of the SSTR data should be pursued to ascertain if the limitations of the experiment can be overcome analytically to yield more conclusive information about the location of the neutron source.	HEDL has action.
III.	The collimated spectrometer used to survey the "A" purification demineralizer should be used to survey the same locations as the SSTR measurements in order to complement the SSTR data. This examination must be performed before March 1984.	Was not accomplished prior to installation of the canal seal plate due to scheduling difficulties.
III.	The use of miniature ionization chambers inserted into incore instrument calibration tubes should be pursued provided that a complete safety evaluation of the consequences of a loss of coolant can be performed.	Concur. Safety evaluation may rely on the low risk posed by this operation.
III.	The video examination of the lower regions of the reactor vessel should be performed as soon as practical after head removal.	This is not in current plans. TAAG consid this to be an open item remaining to be resolved.

DISPOSITION OF TAAG RECOMMENDATIONS: REPORT # 7

RECOMMENDATION I.D.	STATEMENT	DISPOSITION BY GPU
III.	The metal disintegration machine (MDM) should be purchased and a mockup for training should be fabricated.	This is not in current plans.
III.	The inspection crew to be used for this examination should be the same inspection crew that was used in the original "Quick-Look" inspection in July 1982.	This is not in current plans. TAAG considers this to be an open item remaining to be resolved.
V.	The measurements confirm that current plans for dry head removal followed by the installation of the internals indexing fixture to shield the plenum are appropriate and should not be changed due to underhead dose rates.	Concur.
VII.	A flat, non-ribbed gasket should be used to seal the internals indexing fixture to the reactor vessel flange.	Accepted.
VII.	Adhesive should not be used across the entire face of the gasket.	Under evaluation. TAAG considers this to be an open item remaining to be resolved.
VII.	The lower support ring of the internals indexing fixture and the reactor vessel flange should be examined to determine their condition. The reactor vessel flange should be cleaned prior to placement of the IIF.	Concur.
VII.	The internals indexing fixture should be structurally evaluated to determine if the loads, such as the work platform, will affect the seal.	Evaluation has concluded that the seal will not be affected.
VII.	A small pump should be considered as a backup should the internals indexing fixture seal leak.	Not in current planning.

RECOMMENDATION I.D.	STATEMENT	DISPOSITION BY GPU
VIII.	Present plans do not preclude replacing the reactor vessel head on the reactor vessel. However, TAAG could not identify any compelling technical reason why the head should be replaced.	Concur.
VIII.	If the head is to be put back on the reactor vessel, elastomer gaskets would not be advisable. TAAG recommends that soft metal "O" rings be used. The most straightforward approach might be to use new "O" rings of the original design and to retension the reactor head studs to their design value.	If a need to replace the head arises, O-rings will be considered.
IX.	Licensing and all other paperwork processing activities should proceed because these tasks control the schedules for head and plenum removal. These paperwork tasks should continue even in a period of reduced funding.	Evaluated on a case-by-case basis.
IX.	The development program for the fuel transfer canal water filtration system should continue as a priority task since this task is the critical path for plenum removal.	Concur.
IX.	A number of improvements in the presentation of the information in the schedules should be made to assist management understanding of the project schedule.	Concur.
IX.	TAAG recommends that this effort be continued over the next period in order to evaluate finalized schedules.	Was not included in next period activities.

DISPOSITION OF TAAG RECOMMENDATIONS: REPORT # 7

RECOMMENDATION I.D.	STATEMENT	DISPOSITION BY GPU
X.	A number of improvements in the administration of technical planning documents should be made to reduce document proliferation and to specifically identify revisions. The "primer" type of document should be eliminated.	TAAG review is acknowledged. In some cases primer type documents, i.e., technology assessments, are useful.
X.	A water management group should be formed to aid in the management of water to assure separation of various borated and non-borated water streams.	Water management is jointly conducted by technical planning and plant operations. A reorganization is not required.
X.	TAAG had specific comments on 21 technical planning documents. Two of these documents, TPO/TMI-058, Solid Waste Handling and Packaging Facility, and TPO/TMI-066, Interim On-Site Storage Facility, contained serious technical deficiencies that need to be addressed prior to further consideration.	Comments are acknowledged. A long-range solid waste management plan is currently scheduled that will reconsider the two referenced studies. Reexamination indicates that the basic need still exists for a packaging and handling facility and for an interim storage facility to avoid interruption of recovery activities. Therefore, the basic conclusions of these two studies prevail.
X.	TAAG is concerned with the possibility of boron dilution using the sump suction pump method described in TPO/TMI-063, for accommodating leaking in-core instrument pipes. TAAG considers this to be an open item.	The sump is not pumped to the reactor until a sufficient inventory of water is in the building to ensure proper boron concentration. Therefore, GPU believes that the TAAG concern is resolved.
XI.	TAAG recommends that the examinations recommended in Section 3 of this report (#7) be prerequisites for any CSA removal technical planning. If those examinations are expedited, technical planning for CSA removal can begin in the fourth quarter of 1984.	Some of the examinations will be conducted. Refer to comments above. The start of planning activities cannot wait for the results of these examinations. When examinations are conducted, the results will be used to refine the plans.

Attachment A

MPR ASSOCIATES, INC.

May 30, 1984

TMI-2 DEFUELING CONCEPT WHICH ALLOWS THE USE OF LONG HANDLED DEFUELING TOOLS

INDEX OF CHAPTER

	Page
I. Introduction and Background	1
II. Description of Defueling Concept Using Long Handled Tools	3
A. Features to Achieve 15-25 Foot Working Water Depth For Long Handled Tools	3
1. General Description of the Concept	3
2. Water Level in the Reactor Vessel	5
a. Working Water Depth	5
b. Shielded Work Platform	6
c. Water Processing and Ventilation Features	7
d. Water Depths for Handling and Loading Canisters	8
3. Features and Operations in the Deep End of the Reactor Cavity Pit and Spent Fuel Pools	8
a. Gate to Increase Water Depth in the Deep End of the Reactor Cavity and Spent Fuel Pool While Isolating and Maintaining a Lower Water Level in the Reactor Vessel	8

Page

b. Handling Canisters in the Deep End of the Reactor Cavity 9

c. Handling Canisters in the Spent Fuel Pool 10

B. Separation of High Activity Water from Low Activity Water and the Beneficial Impact on Water Cleanup and Processing Facilities 11

1. Shielding the High Activity Reactor Vessel Water and Isolation of this High Activity Water from the other Water Pits with Relatively Clean Water so they are not a Significant Radiation Source to Workers 11

2. Separation of High Activity Water may permit use of Existing Water Cleanup and Processing System 11

C. Radiation Levels During Long Handled Tool Operations 12

1. Radiation from the Working Slot Area Due to Activity in the Reactor Water 12

2. Canister Handling 13

3. Fuel Pin Handling 13

D. Concept of a Vacuum System to Remove Loose Core Debris in the Long Handled Tool Defueling Arrangement 14

E. Use of Automated/Remote Tool in the Long handled Tool Defueling Concept 15

III. Use of the Reactor Cavity to Reduce Radiation Levels 16

IV. Dry Handling of the Upper Plenum Assembly 17

		Page
V.	Summary of Features For Defueling Concept	19
	A. Separation of High Activity Water and Low Activity Water and Minimizing Volume of High Activity Water	19
	B. Potential for Using Existing Water Processing and Cleanup Systems with Minimum Modifications	19
	C. Working Water Depths and Radiation Levels are Satisfactory for Use of Practical Long Handled Tool Defueling Concept - This is Based on Extensive Past Experience [Note: Concept does not preclude use of Automated/Remote Defueling Tools, but is not dependent on them]	20
	D. Tooling and Equipment for Concept Are Straightforward and Quite Simple (e.g. No Development Work is Required)	20
VI.	General Assessment of the Concept	21

FIGURES

1. Long Handled Tool Defueling Concept - Overall Layout

2. Long Handled Tool Defueling Concept - Reactor Cross Section

3. Long Handled Tool Defueling Concept - Plan Section

4. Concepts for Configuration of Working Slot and Slot Cover

5. Canister Loading Elevations in Reactor Vessel

6. Radiation Levels While Using Hand Tools from Top Shielded Work Platform

7. Dose from a Fuel Canister During Long Handled Tool Defueling Operations

8. Dose from a Single Fuel Rod Long Handled Tool Defueling Operation

9. Conceptual Arrangement for the Initial Loose Debris Vacuuming System in the Long Handled Tool Defueling Concept

10A. Layout Showing Concept of the Automated/Remote Arrangement in the Present Reference TMI-2 Defueling Concept (see Figure 1 of GPUN Letter of Feb. 28, 1984 to EG&G)

10B. Conceptual Layout Showing How the Automated/Remote Tooling is also Compatible with the Long Handled Tool Defueling Concept

11. Dry Upper Plenum Removal

12. Conceptual Layout of Dutchman on Which Shielded Work Platform is Positioned

13. Radiation Survey of Reactor Cavity

TMI-2 DEFUELING CONCEPT WHICH ALLOWS THE USE OF LONG HANDLED DEFUELING TOOLS

I. Introduction and Background

The present defueling concept being considered for TMI-2 basically involves filling the entire reactor cavity with water (very similar to a normal defueling operation) and using automated/remote tools to grapple and grind the remains of the core. The ground up fuel debris is then to be slurried into canisters. With this concept, the working water depth to the top of the loose core debris is approximately 39 feet. The working water depth to the lower grid plate is approximately 48 feet. With these working water depths, use of long handled tools is impractical and inefficient except for very isolated and specific types of functions.

A concern has been raised about the present defueling concept and the fact that the water depth is beyond the practical limits for using long handled tools. Further, there is concern that the present contemplated system is almost totally dependent on the use of automated/remote tools and the fuel grinding/slurrying approach with which there is no significant experience. Long handled tools have been proven to be very effective in dealing with unplanned and unforeseen evolutions which invariably occur in this type of defueling operation. Accordingly, the question has been raised <u>whether there is a practical concept for defueling TMI-2 which would allow the use of long</u>

handled tools and, thus, not be totally dependent on automated/remote defueling tools.

Experience has shown that there are two practical requirements that have to be achieved to make the use of long handled tools viable. Specifically:

- The first requirement is that the working water depth for the long handled tools needs to be in the range of 15-25 feet. If the depth of water goes beyond 25 feet, the degree of difficulty in effectively using long handled tools increases very rapidly with small increases in depth. Working water depths of 40 feet and greater are not practical except for isolated and specific types of functions. Accordingly, it is desirable to determine if there is a defueling concept for TMI-2 which would allow the water to be in the 15-25 feet depth range for the areas where the bulk of the work has to take place.

- The second requirement is that the radiation level for people using the long handled tools must be kept down in the 10-50 mr/hr range while working over the pit. Such exposures would be in the range of that experienced in normal refueling operations.

Accordingly, TAAG undertook a brief investigation to determine if there is such a defueling concept that would meet the above requirements so that long handled tools could effectively be used. Obviously, if there are automated tools that are well-proven and effective, the concept should not preclude the use of such tools; however, the concept must not be dependent on their satisfactory operation.

II. Description of Defueling Concept Using Long Handled Tools

A. Features to Achieve a 15-25 Foot Working Water Depth For Long Handled Tools

1. General Description of the Concept

The basic approach in this concept is to retain the water within the reactor vessel and to use a rotating shielded deck on top of the reactor vessel from which people using the long handled tools could operate. This deck would provide the shielding so that the contaminated water in the reactor vessel would not be a major exposure source for personnel working the tools. This shielded platform is installed 1 to 2 feet above the reactor vessel flange by having a cylindrical dutchman between it and the vessel (see Figures 1 and 2). The various lines for water treatment and for air ventilation to control any off-gassing would be run into the reactor vessel through the dutchman and would not impact operation of the rotating shielded platform.

The shielded platform has about a 2-foot wide slot in it through which the long handled tools would be operated. This tool working slot width and the platform shielding should limit the radiation fields to the order of 10-50 mr/hr while operating the long handled tools. The working slot on the shielded platform is covered by hinged doors or plugs of about 2-inch thick steel or possibly lead glass (with a protective cover) to increase visibility. These doors can be opened to gain access to the reactor vessel and permit lateral movement of the long handled tools to deposit a canister in the transfer boot. Various concepts for the slot configuration and slot covers can be seen on Figure 4.

By taking advantage of the shielding provided by the reactor pit cavity walls and the clean stainless steel liner of the pit cavity, exposure due to other radiation sources such as the painted surfaces elsewhere in the building are avoided. Thus, general background radiation levels can be very low for the defuelers down in the reactor cavity (e.g. even in its present undecontaminated state, parts of reactor cavity currently read as low as 20 mr/hr).

To transfer canisters loaded with fuel debris from the reactor vessel, a shielded transfer boot and a very simple transfer cask would be involved (see Figure 2). The shielded transfer boot allows passage of the radioactive canister out of the water and into the cask. This simple cask would weigh about 15 tons. A similar shielded transfer boot is located over the fuel transfer device in the deep end of the pit.

To provide adequate water shielding for the operations that take place in the deep end of the reactor cavity pit and in the spent fuel pool, it would appear that a 4 to 6 foot gate needs to be installed to allow the water level in the deep end of the reactor cavity and the spent fuel pit to be raised above the elevation of the reactor vessel flange. Once the fuel canister is transferred out of the reactor building and into the spent fuel building, it is removed from the plant's existing horizontal fuel transfer system by means of a transfer boot and simple cask to deposit the canister in an appropriate storage rack location. The overall layout of this concept and key features are shown in Figures 1 and 2.

2. Water Level in the Reactor Vessel

a. Working Water Depth

By using a shielded work platform on top of the reactor vessel, the water level can basically remain at the reactor vessel flange or at least no more than a foot or so above the flange. As can be seen in Figure 2, with the shielded work platform approach, the working water depth to the top of the existing loose core debris pile is approximately 16 feet and the working depth all the way to the lower grid plate is only about 26 feet. Accordingly, this approach results in working water depths that are very practical for long handled tool defueling operations.

b. Shielded Work Platform

The shielded work platform is approximately 12" thick and is assumed to have no more than 3" of lead shielding (see Figures 2 and 3). The platform also includes a transfer boot as shown in Figure 2. The transfer boot has provisions for a lateral side gate which would allow long handled tools to move canisters or other objects between the slotted work area of the shielded platform and the center of the transfer boot without requiring lifting the canister out of the water. Likewise, the transfer boot at the "up-ender" would be similarly designed with a side opening gate. This concept of using transfer boots to move fuel between water and a transfer cask has been used extensively at ETR and ATR and in other non-commercial applications for over 20 years.

c. Water Processing and Ventilation Features

The shielded working platform is positioned on an approximate 2 foot high dutchman that fastens directly to the reactor vessel flange (see Figure 2). This dutchman keeps the working water depth low, yet allows the necessary water purification system to draw water out and return the water to the reactor vessel. Likewise, this dutchman also allows an air sweep system to collect any gases that may evolve off during the defueling operations to be drawn off from the underside of the shield platform so as to minimize any airborne activity problems that may occur. This approach of air and water processing through the dutchman would not require activation of any of the systems currently attached to the reactor coolant system (e.g. decay heat removal, mini-decay heat removal, etc.), and therefore, may possibly help eliminate such systems from the plant's Tech. Specs. A conceptual layout of the dutchman is shown in Figure 12.

d. Water Depths for Handling and Loading Canisters

The water depths available in the reactor vessel when manually picking up and loading core debris objects (e.g. fuel rods, end-fittings) into canisters are shown in Figure 5. This figure is based on the assumption that the loose core debris has already been vacuumed into canisters.

As can be seen in Figure 5, a 6-foot long object can be loaded into a canister and there will still be 3'-8" of water available for shielding. This depth of water is more than adequate to make the dose to workers on the shielded platform very low. The radiation levels resulting from raising various objects near the water surface are discussed in Section II.C below.

3. Features and Operations in the Deep End of the Reactor Cavity Pit and Spent Fuel Pools

a. As can be seen in Figure 2, there is a gate approximately 6 feet high installed between the reactor vessel and the deep end of the reactor cavity pit. The original plant construction included the necessary provisions in the cavity walls and floor for a gate slot at this location. Accordingly, only the gate itself needs to be provided. The water level behind this gate is basically set by the water level over the spent fuel racks in the auxiliary fuel handling building (see Figure 1). Since the deep end of the reactor cavity and the spent fuel pool are connected by the transfer tubes, the water levels in these two pools will be at the same level. The water level shown in Figures 1 and 2 also provides adequate shielding for the storage of the upper plenum assembly and provides sufficient water over the "up-enders" to allow the transfer and handling of canisters. A detailed analysis of the concept may allow further reduction of this water level by a foot or so, but at this point, it is not desirable to reduce the margin in this area. In any case, this gate size and water depth appear to be reasonable.

b. The handling of canisters in the deep end of the reactor cavity also involves the use of a transfer boot (e.g. mounted on a stand) over the "up-ender" of the fuel transfer mechanism. With this arrangement, the transfer cask moves back and forth between the reactor vessel transfer boot and the "up-ender" transfer boot (see Figure 2). Conceptually, this is not too different from the fuel transfer operations that normally occur with the fuel handling manipulator that moves fuel between the reactor vessel and the "up-ender" during refueling operations in a nuclear power plant. To facilitate this transfer operation at TMI-2, the orientation of the rotating shielded platform's transfer boot and the "up-ender" transfer boot could be set so that the polar crane bridge is in a fixed position and only the polar crane trolley has to move back and forth between these two transfer points. Thus, canister transfers would not require the movement of the main frame of the polar crane. This should greatly facilitate the speed at which the transfer cask can be handled.

c. A transfer boot is also located in the spent fuel pool. This transfer boot could either be fastened to a cask which is handled by the spent fuel building crane or the cask and a transfer boot could be suspended from the existing spent fuel handling bridge. In either of these arrangements, the boot and cask could move back and forth from the spent fuel pool "up-ender" and to the various spent fuel storage racks where the canisters are to be stored.

B. <u>Separation of High Activity Water from Low Activity Water and the Beneficial Impact on Water Cleanup and Processed Facilities</u>

1. To provide shielding over the water through which the long handled tools are being used, a rotating shielded platform is provided as discussed previously. Secondly, by separating the more highly contaminated water in the reactor vessel from the relatively clean water in the deep end of the reactor cavity pit and in the spent fuel pit areas, the water in these later two pits does not become a major radiation source. This would not be the case if the entire water pit was connected directly to the reactor vessel water.

2. Another advantage of separating the reactor vessel water from the pit water is that the volume of water remaining in the reactor vessel is relatively small and, therefore, it should greatly reduce the total volume of water that has to be processed through the <u>Reactor Vessel Portion</u> of the defueling water cleanup (DWC) system. This reduced processing volume should result in a reduction in the number of filters and ion exchangers that will have to be used, handled, and shipped off site for waste disposal. In addition, this long handled tool defueling concept will also reduce the volume of water in the deep end of the reactor cavity and in the spent fuel pit. This in turn will reduce

the total volume of water that has to be processed through the <u>Fuel Pool Portion</u> of the defueling water cleanup (DWC) system. Thus, by separating reactor cavity deep end and spent fuel pools from the reactor vessel, the water activity should be lower in the pools and the reduced processing volume will also result in fewer filters and ion exchangers that will have to be used, handled and disposed of in cleaning up the water in these pools.

C. <u>Radiation Levels During Long Handled Tool Operations</u>

1. As indicated in the general description section above, the long handled tools are operated through a slotted section in the rotating shielded platform. Figures 2 and 4 show the workup slot to be about 2 feet wide and covered with hinged shield doors or block sections with access holes for the long handled tools. An assessment was made of the worker radiation exposure levels as a function of various working hole sizes assuming that the water has an activity of 1 μc_i/cc. These are tabulated in Figure 5. A 2-foot diameter working hole results in exposure of about 35 mr/hr on the shielded platform. A 1-foot diameter working hole results in about 10 mr/hr.

2. An assessment was made of the radiation level generated by a canister that was fully loaded with maximum activity debris (see Figure 6). As can be seen in this figure, even if such a very highly and fully loaded canister was brought to within 3 feet of the water surface, the radiation level at 2 feet above the shielded platform, assuming a 2-foot opening, would be about 380 mr/hr and about 40 mr/hr if the water depth is maintained at 4 feet. Accordingly, about 4 feet of water shielding should be quite adequate to permit handling of a canister.

3. An assessment was also made of the radiation level resulting from handling a maximum burn-up fuel rod assuming no leaching or loss of fission products (see Figure 7). As can be seen in this figure, such a rod, if raised to 1 foot of the water surface, would result in a radiation level of about 370 mr/hr at work elevation and about 50 mr/hr if 2 feet of water is maintained. Based on the depth of water available in this concept, sufficient water depth should be available to permit handling of fuel pins, debris and canisters to still maintain low radiation levels in the work areas.

D. Concept of a Vacuum System to Remove Loose Core Debris in the Long Handled Tool Defueling Arrangement

Presently there is about 3 feet of relatively loose and small size debris resting on the TMI-2 core. It is proposed that one of the first actions after the plenum has been removed is to vacuum up this loose debris. Figure 9 shows conceptually how such a vacuuming system can be installed and used with a long handled tool defueling concept. In this concept a long handled tool (with vacuum suction hose, TV, and lights mounted on the end of the tool) can be manually manipulated through a working slot to vacuum the debris. Sufficient direct visibility will most likely be available for this type of operation since water is relatively shallow and fine tool positioning is not required. If direct visibility should be a problem, the TV camera should provide adequate viewing. The vacuum system pump unit and bracket assembly to support the canisters are positioned in the reactor vessel by suspending them in an access port of the shielded work platform. The canisters are positioned in the access port adjacent to the working slot. Thus, by using an off-set hand tool in the working slot, canisters can be installed and moved to the transfer boot for removal from the reactor vessel.

E. <u>Uses of Automated/Remote Tools in the Long Handled Tool Defueling Concept</u>

The present reference TMI-2 defueling concept uses automated/remote tooling. The tooling is positioned in the reactor vessel flange under about 23/24 feet of water. The reference tooling concept is shown on Figure 10A.

As indicated previously, it is desired that the long handled tool defueling concept should not preclude the use of automated/remote tooling if it is reliable; however, this defueling concept should not be dependent on it. Figure 10B shows conceptually how automated/remote tools can be used in conjunction with the shielded work platform of the long handled tool defueling concept.

In this regard it should be noted that <u>drive mechanism portions of the automated/remote tools are above water, directly accessible for manual maintenance and adjustment instead of being 23/24 feet under water as in the present reference concept</u>. Also, with the plenum stored in its normal position in the deep end of the reactor cavity, the south end of the cavity with its low radiation levels is available for the control units associated with such automated/remote equipment, as well as for a supervisor and worker staging area.

III. Use of the Reactor Cavity to Reduce Radiation Levels

The existing floors and walls of the normal refueling work area (i.e., 347' elevation) are basically painted. These painted floors and the walls are a significant source of radiation to workers. Therefore, if defueling operations take place at that present elevation (elev. 347'), the nominal exposure just from activity entrapped in the paint and concrete of the floors and walls is understood to be in the order of about 100 mr/hr, even after numerous decontamination efforts have been made. However, the reactor cavity pit which is approximately 25 feet deep is stainless steel lined rather than painted. The bottom of the reactor cavity has some of the lowest readings in the TMI-2 containment building, even without any significant decontamination (e.g. some areas of the cavity are as low as 20 mr/hr at present - see Figure 13). Accordingly, it is considered that the long handled tool defueling concept can take advantage of the fact that the reactor cavity stainless steel walls and floor are not a source of trapped contamination. With further simple cleaning of the stainless steel liner, surface contamination as a radiation source in this area should, for all practical purposes, be eliminated. Since the reactor cavity walls are in essence shield walls, they are used in this concept to protect personnel working on top of the shielded platform on the reactor vessel from other building radiation sources. Assuming that further cleaning can get the radiation down in the cavity to near normal background, the only radiation that would have to be contended with is that coming from the working slot of the shielded platform on the reactor vessel. If there is radiation shine from the containment dome area back down into the reactor cavity, as some have speculated, it should be relatively easy to install additional beams across the pit and provide a temporary shield and herculite cover to prevent any of the dust from the other parts of the building from affecting the working level down at the bottom of the reactor cavity.

In essence, it appears very practical in this concept to separate the reactor cavity work area from the remainder of the building and therefore drastically reduce the general background radiation in the cavity to a level experienced in current nuclear plants during refueling. Further, it also offers the potential of isolating the cavity working area from the rest of the containment which is a source of airborne activity problems. It thus offers the potential of eliminating the need for respirators for work in the reactor cavity.

IV. Dry Handling of the Upper Plenum Assembly

Presently many of the OTSG nuclear plants move the upper plenum assembly from the reactor vessel to the deep end of the reactor cavity with just the deep end of the cavity flooded with water (e.g. this is commonly referred to as a dry transfer). The plants have found this approach very effective from ALARA point of view and this allows various work tasks to proceed at the reactor flange area without having to go through a major decontamination effort of the stainless steel cavity liner since the cavity above the reactor vessel did not have to be flooded (i.e. decontamination of the cavity after flooding is normally a major and time-consuming effort). This dry transfer has been found to be ALARA effective even though the upper internals assembly in these plants have seen many years of operation and thus have significant activity in the lower portion of the assembly.

An assessment was made using the dry transfer technique for handling the upper plenum assembly at the TMI-2. While TMI-2's upper plenum is not as highly activated as those in other power plants, it has very significant surface contamination (e.g. 1300 $\mu c_i/cm^2$). Figure 11 shows the estimated radiation levels at the south end of the TMI-2 reactor cavity during dry plenum transfer, assuming the plenum has the maximum surface activity which has been measured (1300 $\mu c_i/cm^2$) and assuming that there are no shield walls at the end of the cavity. The radiation levels, even with these conservative assumptions, are 2 r/hr at the very edge of the cavity. Normally, crane operators stand behind the "D" ring wall or back against containment walls for transfer operation and as a result the exposure dose is very small. If problems develop during such transfer at TMI-2, the reactor cavity could then be flooded. Accordingly, a dry transfer of the upper plenum at TMI-2 appears very practical and quite attractive, regardless of the defueling concept used.

V. Summary of Features For Defueling Concept

A. There is separation between high activity water and low activity water (i.e. high activity water is retained in the reactor vessel). The impact of this most important feature is summarized below:

1. It limits the volume and surface area of high activity water to just that in the reactor vessel (some 30,000 gallons). In contrast, the present refueling concept requires flooding the entire reactor cavity which contains some 300,000-400,000 gallons of water.

2. By reducing the area of highly contaminated water and shielding that area (i.e. the shielded work platform), it should greatly reduce the radiation source caused by the high activity water and thus reduce exposure to defueling workers.

B. The reduction of the high activity water volume and the separation feature discussed in II.B. above should reduce the total volume of water processed by both the reactor vessel portion and the spent fuel pool portion of the defueling water cleanup (DWC) system. This reduced volume of water that has to be processed should in turn significantly reduce the number of filters and ion exchangers that will have to be used, changed and disposed of as waste.

C. The water level is near the reactor vessel flange and a shielded work platform over the reactor vessel serves as a defueling work area. The advantages of these features are as follows:

1. The defueling working water depth is in the range of 16-27 feet, thus allowing the use of manual long handled defueling tools. This arrangement does not preclude the use of automated/remote tools, but the concept is not dependent on them.

2. The reactor vessel cavity walls and floors, since they are not painted surfaces like the 347' levels, should not be a radiation source. In essence, the reactor vessel cavity acts as a shielded cubicle to minimize exposure to defueling personnel working on the shielded platform. The south area of the reactor vessel cavity is a good staging area for defueling equipment and provides a low radiation area for other potential uses since the upper plenum assembly is stored in the deep end of the reactor vessel cavity (e.g. its normal storage location).

D. The tooling and equipment requirements are straightforward and mechanically simple. They are non-developmental and no research effort is required. This basic defueling concept has been successfully used for many years in test reactors and certain other non-commercial power applications.

VI. General Assessment of the Concept

There have been numerous questions and issues raised concerning the present defueling concept and how this new concept addresses those issues. Comments as to how the long handled tool defueling concept answers these various questions are summarized in Table 1.

Table 1
Long Handled Tool Defueling Scheme

Concern	How Addressed in Long Handled Tool Defueling Scheme
1. Some of the schemes currently being discussed involved developmental equipment such as remotely operated arms, shredders, etc.	1. Defueling can be done entirely with long handled tools, although the use of more developmental equipment is not precluded.
2. Long handled tools would not be practical because the water working depth would exceed the maximum practical length of about 25 feet.	2. All operations can be done with long handled tools. The water working depth does not exceed 26 feet in length if shielded work platform is mounted on the reactor vessel and the reactor cavity is dry except for the deep-end which is flooded.
3. Radiation exposure to workers would be excessive, thus requiring the use of the automated/remote defueling concept.	3. All defueling work would be done in the reactor cavity area, which is one of the lowest radiation areas in containment - now 20 to 30 mRem/hour. This dose can probably be reduced by decontamination since the S.S. liner should be relatively easy to clean. A shielded work platform will control the dose rate from the defueling operations in the reactor vessel.
4. Defueling operations will require visibility, hence, the development of a high capacity Defueling Water Cleanup System (DWCS) is a prerequisite for defueling.	4. It may not be true that all defueling activities require visibility. The vacuum defueling evolutions can also, in all likelihood, be performed blind, or with very limited visibility. Later grappling operations will require a degree of visibility, but complete water clarity may not be a requirement even then. Specifically, since the Long Handled Tool Defueling approach reduced the water working depth, the water clarity requirements to obtain sufficient visibility are likewise reduced. Accordingly, it does not appear that the DWCS is a prerequisite for the initial vacuuming of the loose core debris. Also, the level of water clarity/filtration required (e.g. the ppm required) in the DWCS effluent for later defueling operations will be somewhat reduced due to the shallower water depths with this defueling approach.

5. Developmental programs to utilize automated/remotely operated defueling equipment and Mott Sintered Metal DWCS filters control the start of defueling operations.

5. Remotely operated defueling equipment and the DWCS filters are not prerequisites for initial defueling (e.g. removal of the loose core debris). Initial defueling can be accomplished "blind" with simple hand tools and a rudimentary dredging/vacuuming system.

If the development programs for automated/remote equipment experience major delays, manual tools can be used to defuel the reactor. Likewise, if problems develop with the filters for the DWCS, this defueling concept may allow relaxation of the 1 ppm suspended solid requirement (e.g. for visibility) for the effluent from the filters due to the shallow water depths.

The Long Handled Tool Defueling scheme has large body of experience and is similar to one used for more than 25 years at the ETR & ATR test reactors and other non-commercial applications.

The Long Handled Defueling Concept does not preclude the use of automated/remote tooling if they are proven and reliable.

6. Cesium released during defueling will contaminate water in the refueling canal, and increase radioactive exposure to workers.

6. All of the cesium released during defueling would be confined in the reactor vessel. Workers would be protected from the resultant radiation by the shielded work platform. Hence, the radiation hazard from any cesium spike would not become a general area problem, and would not affect defueling. In addition, with the reduced volume of the reactor vessel water and the complete separation of the water in the fuel transfer and spent fuel pool areas from the reactor vessel, the reactor vessel portion of the DWCS should be able to more effectively and quickly handle any cesium release if it should occur.

. Radiolytic gasses (hydrogen and oxygen), krypton, and tritium may be released during defueling and may be a hazard to workers.	7. The area beneath the shielded work platform is vented to the reactor building purge system and would not be a hazard to the workers.
. Water clarity will be a problem during defueling even if television cameras are placed in the reactor vessel.	8. The problem would be minimized by attaching a TV camera and a light source near the operating end of the long handled tools. Further, the reduced water working depth in the long handled tools concept (17 to 26 feet) greatly reduces clarity requirements for visual observations.
. The Long Handled Tool Defueling scheme involves too many handling steps when the canisters are transferred from the shielded deck to the upenders.	9. The number of steps to transfer the fuel are similar to the number of steps required in a normal defueling operation. The steps are simple and straight forward.
0. The Long Handled Tool Defueling scheme could not be developed and put in place by April 1, 1985.	10. The scheme is the least developmental of any being considered and therefore could be ready before any of the others. Its equipment is mechanically straight forward and simple.
1. The Long Handled Tool Defueling scheme would require the opening of the R.B. Equipment Hatch.	11. The scheme does not require the opening of the Equipment Hatch; although, opening the hatch would assist the installation and checkout of the defueling equipment regardless of the scheme used!
2. The reactor cavity will need to be filled with water to permit the transfer of the plenum.	12. The dose rates associated with a dry plenum lift are not so high as to require the entire reactory cavity to be flooded. A dry plenum lift is feasible providing the capability to flood the reactor cavity is provided in case of emergency, (e.g., a failure of the polar crane during the lift).
3. The reactor cavity will need to be filled with water or a special shield box will need to be constructed, in order to shield the workers from the plenum stored in the south end of reactor cavity.	13. The plenum can be stored in the deep end of the reactor cavity in its normal location. The shield water provided in the deep end for fuel canister upending and transfer is also adequate to shield the plenum and to protect the workers.

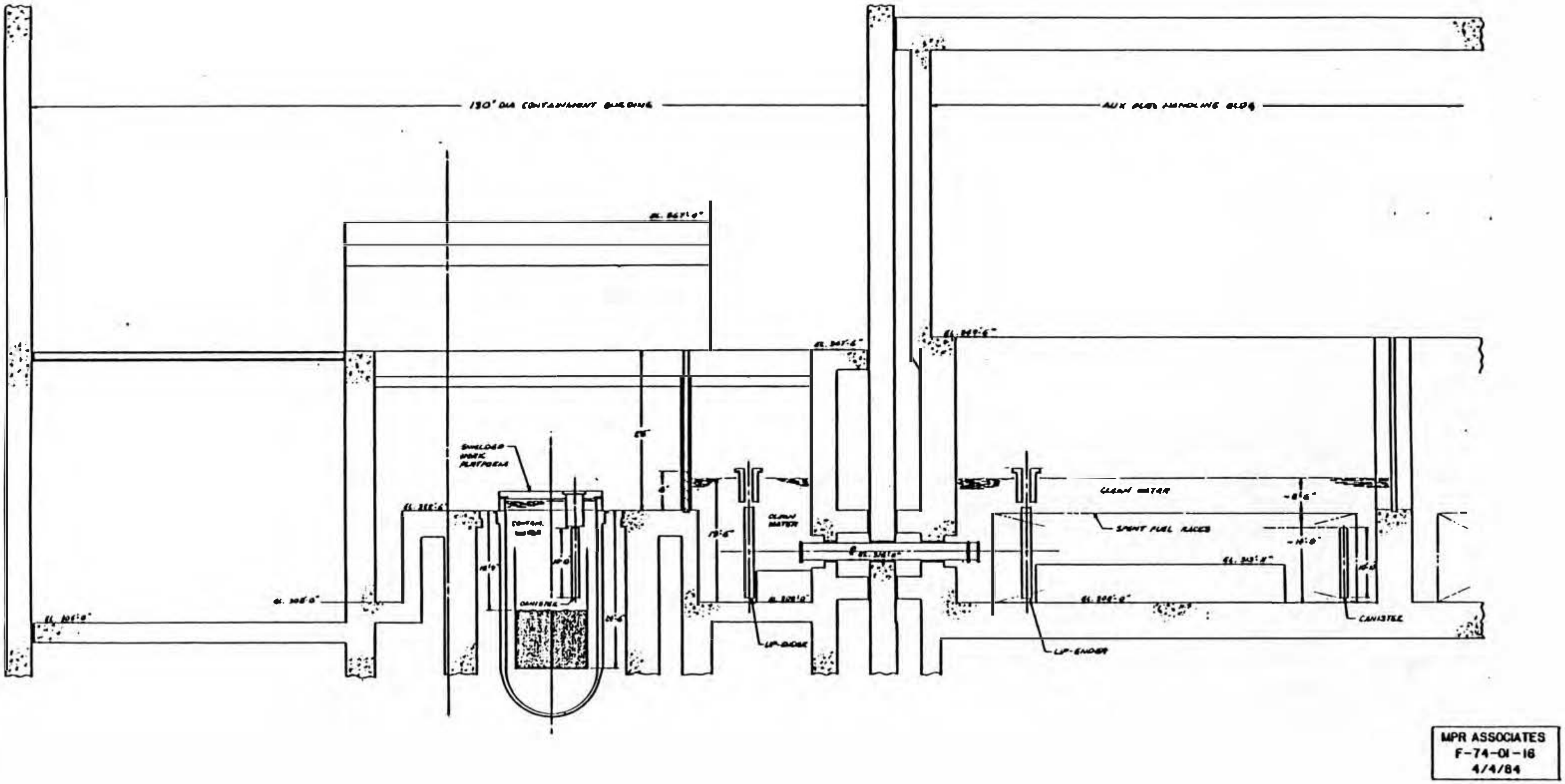
14. The plenum must be stored in the shallow end (ie the south end) of the reactor cavity to permit the use of the deep end for storing a large number of fuel canisters.

15. The vacuum system will require a large number of canisters to be staged in the deep end of the reactor cavity due to the need to operate various types of solids removal devices in series to effect defueling.

14. The fuel canisters should be transferred to the "A" Fuel Pool as soon as they are loaded into one of two the transfer mechanisms. The plenum can be stored in its normal location in the deep end of the pool. A few extra storage spaces for cannisters (e.g. 8) can be provided in the deep end even with the plenum stored there.

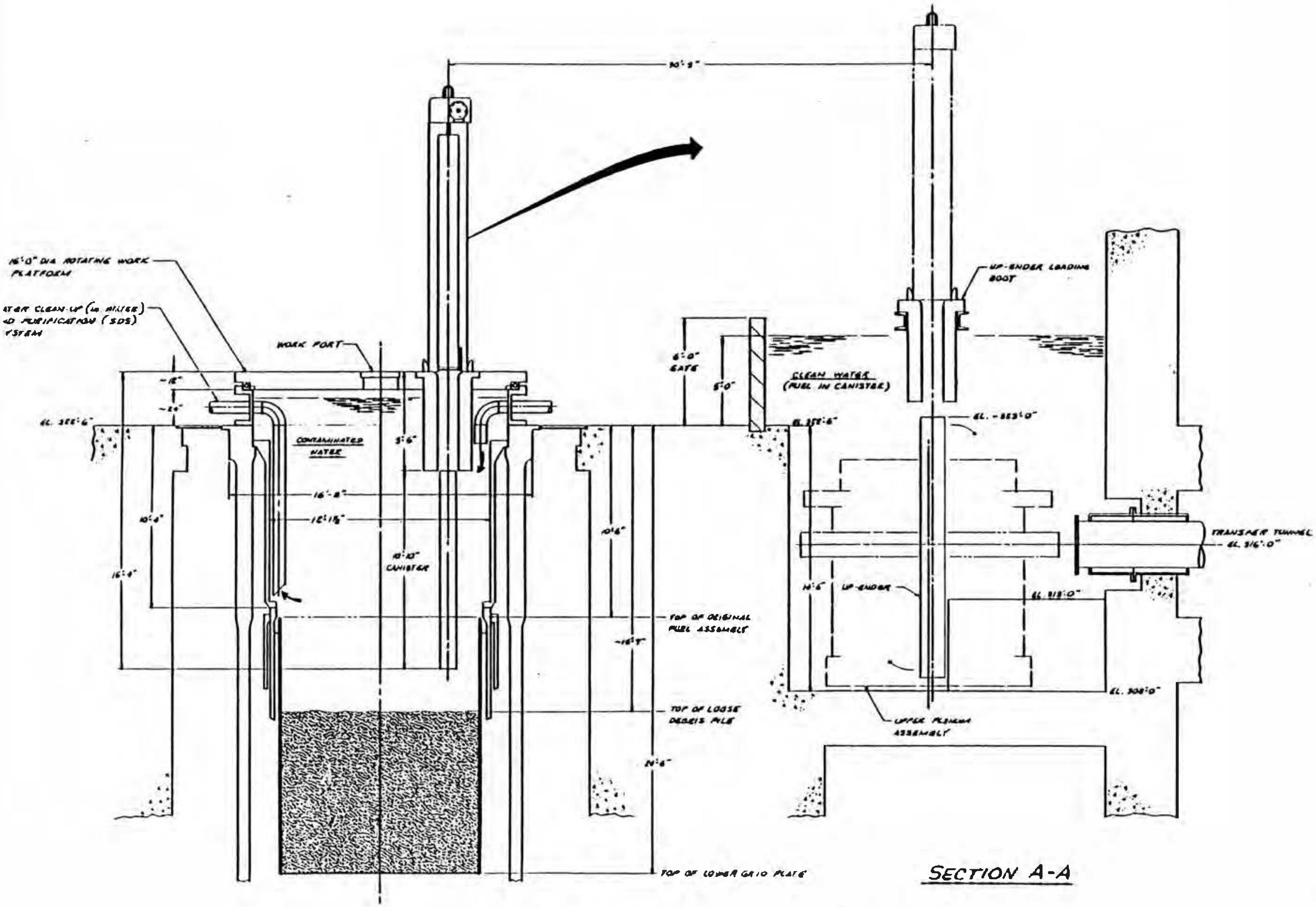
15. The vacuum system developed for the initial rubble removal should be designed to use one sort of solids removal device at a time. Knock-out drums should be able to remove the majority of all fuel debris. The fines can be returned to reactor vessel to be removed at a later time. Such a vacuum system can be contained within reactor vessel.

Shredder/vacuum system developed should be sized to be located in the reactor vessel. This will eliminate the need for complex encasement systems and for breaking connections of the debris transfer system in the clean water in the deep end of the reactor cavity.



LONG HANDLED TOOL DEFUELING CONCEPT
OVERALL LAYOUT

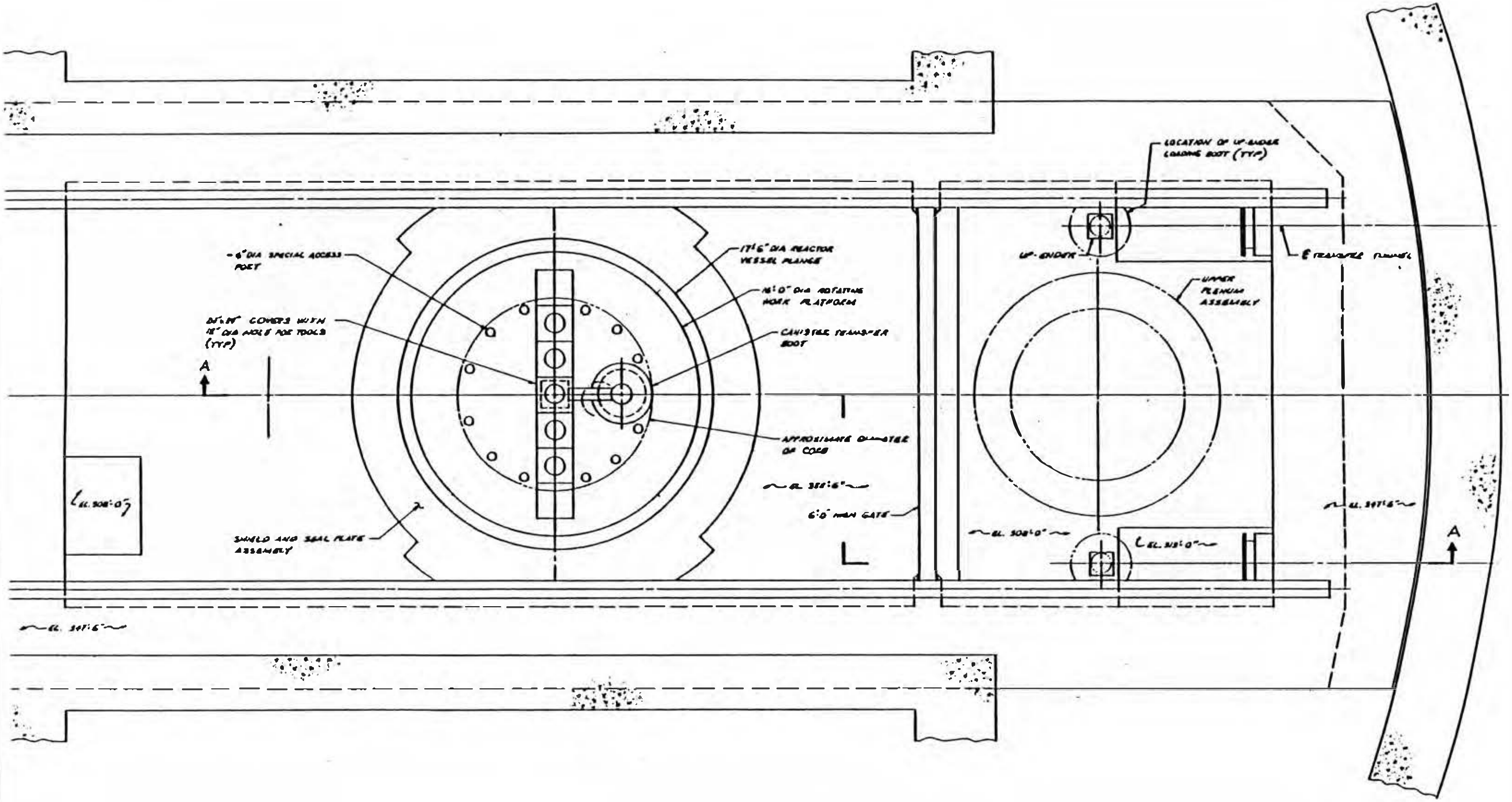
FIGURE I



MPR ASSOCIATES
F-74-01-18
4/4/84

LONG HANDLED TOOL DEFUELING CONCEPT
REACTOR CROSS SECTION

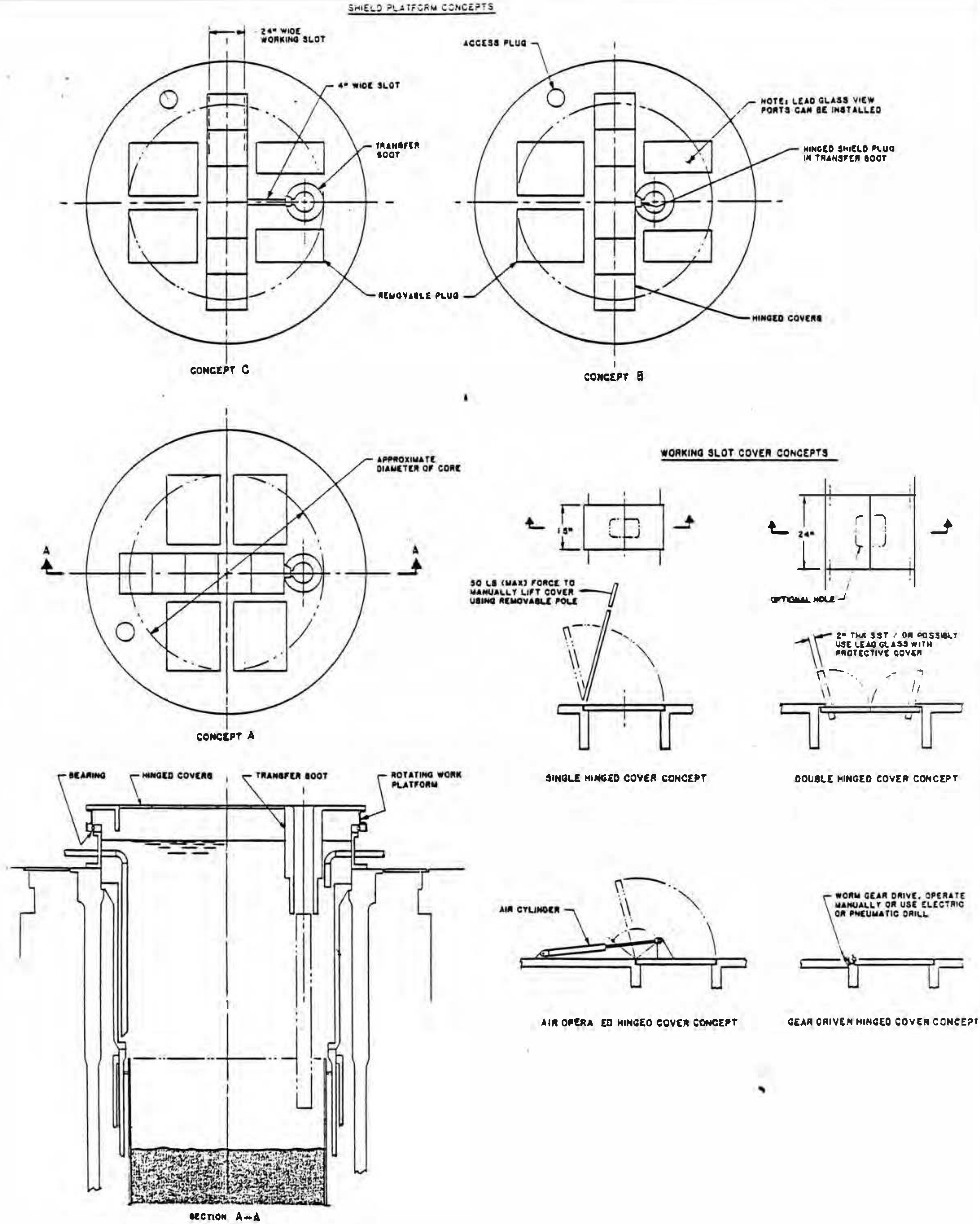
FIGURE 2



LONG HANDLED TOOL DEFUELING CONCEPT
PLAN VIEW

FIGURE 3

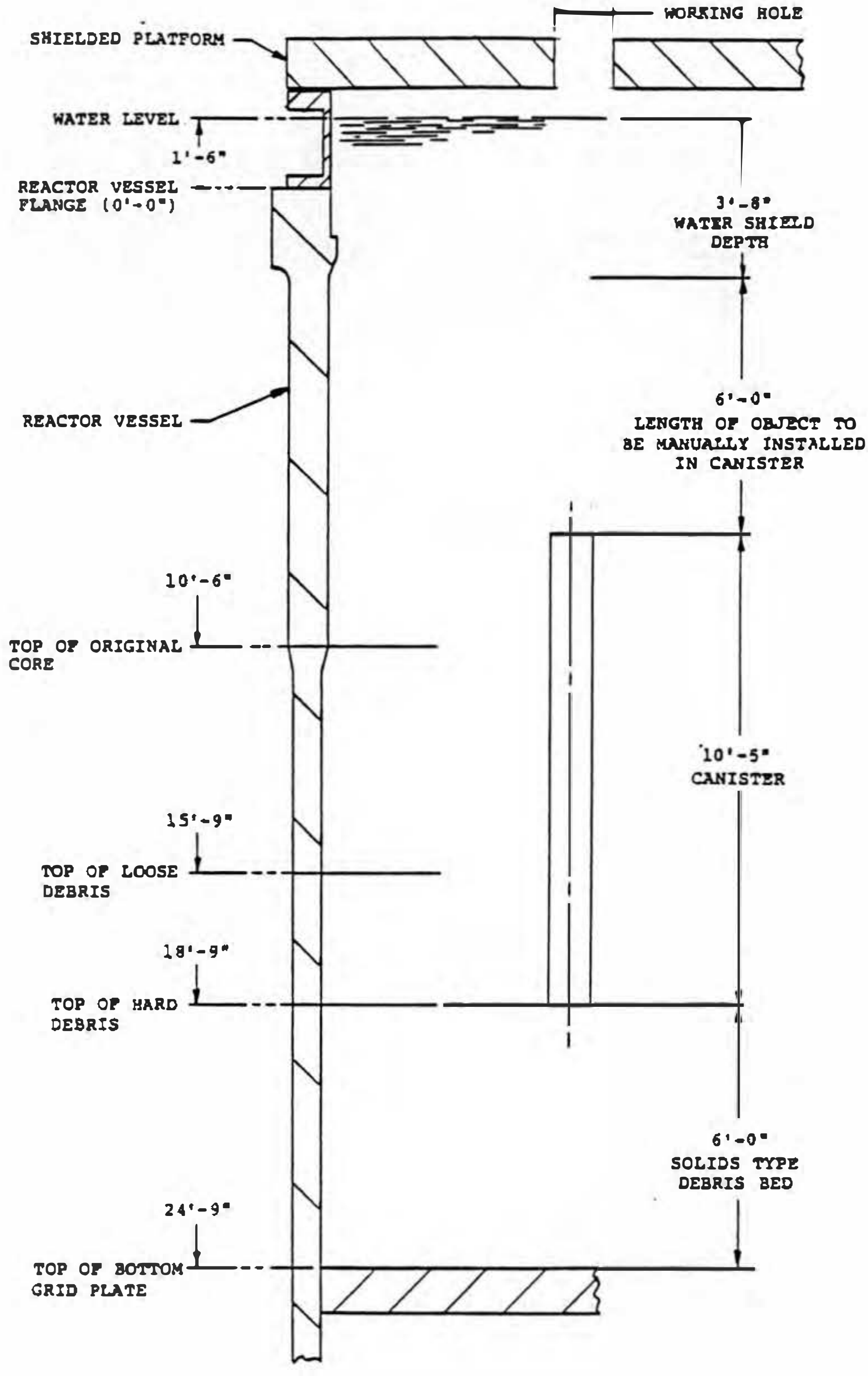
MPR ASSOCIATES
F-74-01-17
4/4/84



WORKING SLOT & COVER CONCEPTS
FOR ROTATING WORK PLATFORM

FIGURE 4

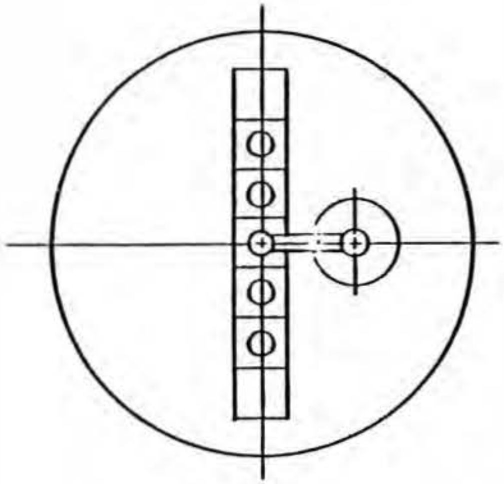
MPR ASSOCIATES
F-74-01-24
4/30/84

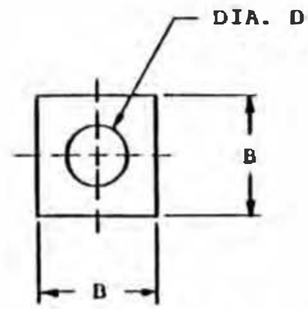


CANISTER LOADING ELEVATIONS IN REACTOR VESSEL

FIGURE 5

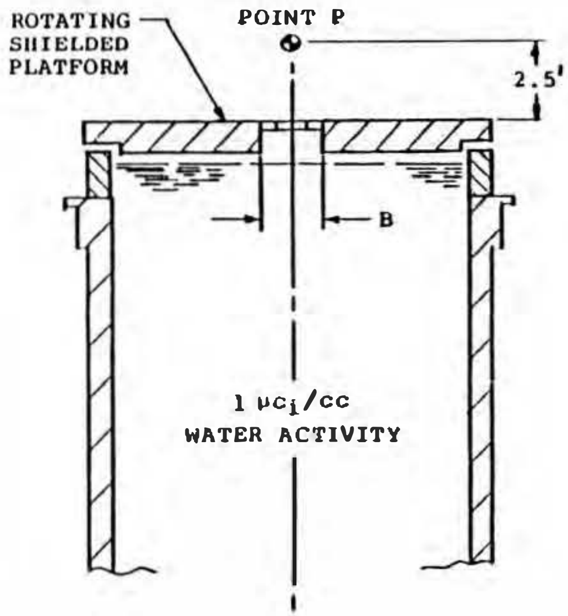
MPR ASSOCIATES
F-74-01-21
4/20/84

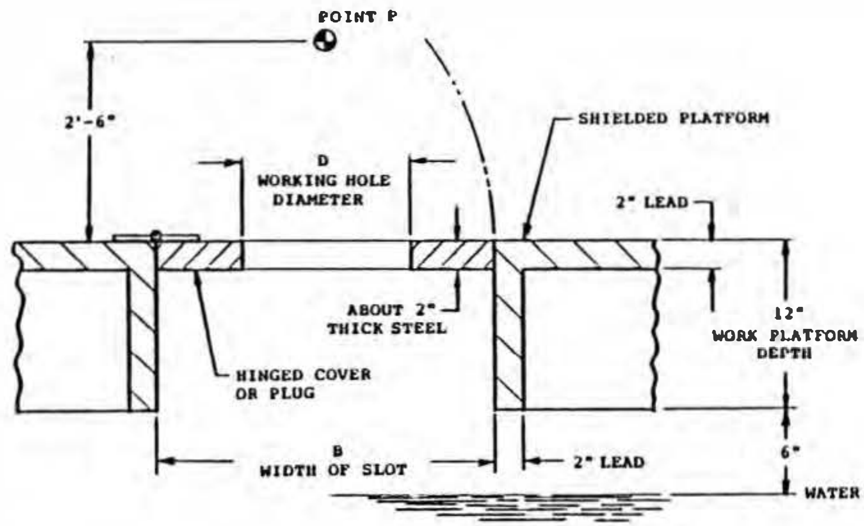




THE "B" DIMENSION VARIES WITH SIZE OF WORKING HOLE ("D"). THE "B" DIMENSION IS EQUAL TO APPROXIMATELY D + 1 FOOT.

DIA. D (FEET)	DOSE AT POINT P (MR/HR)
4	116
3	71
2	34
1	9

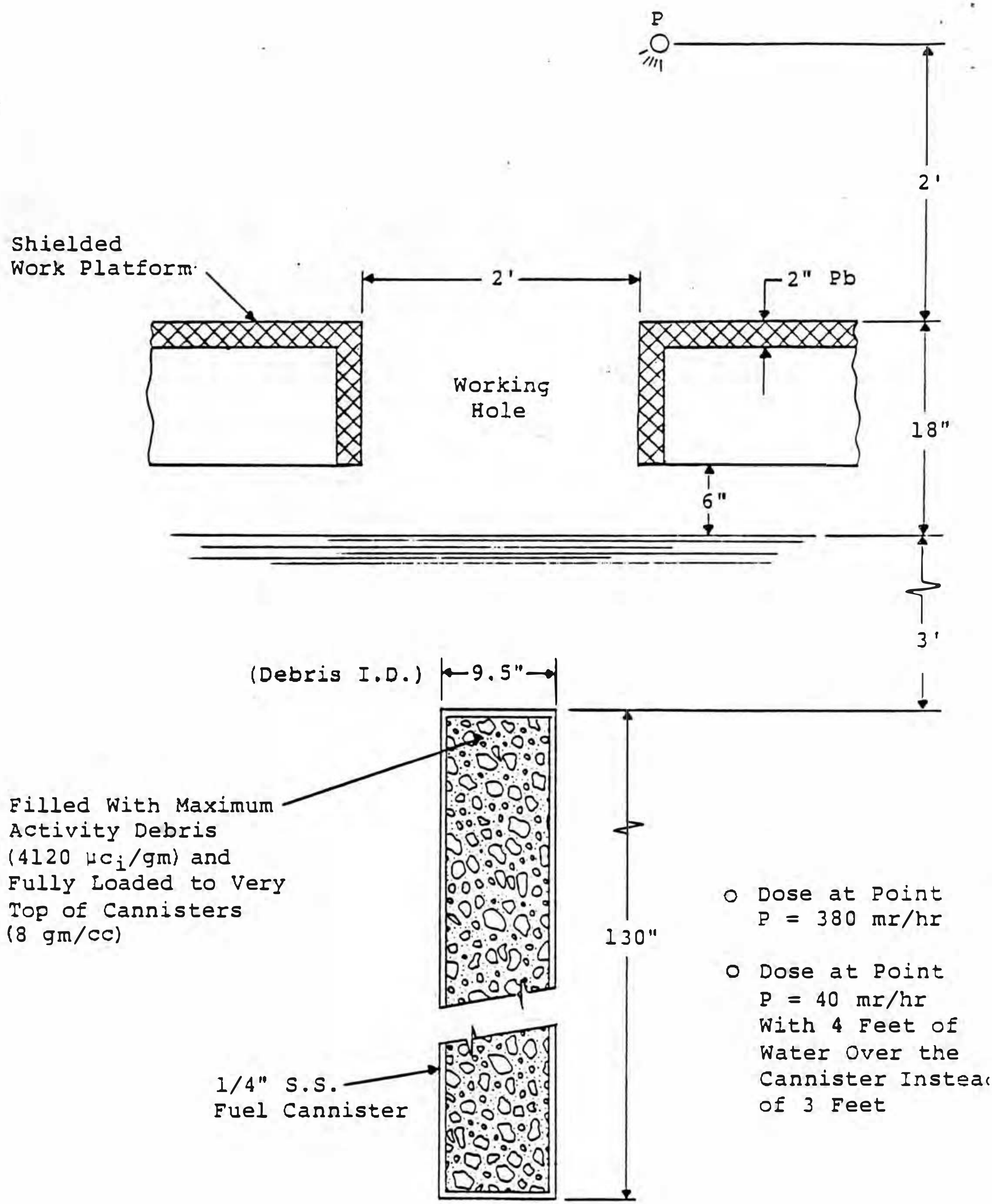




RADIATION LEVELS WHILE USING HAND-TOOLS FROM TOP OF SHIELDED PLATFORM THAT IS POSITIONED ON TOP OF REACTOR VESSEL

MPR ASSOCIATES
F-74-01-19
4/19/84

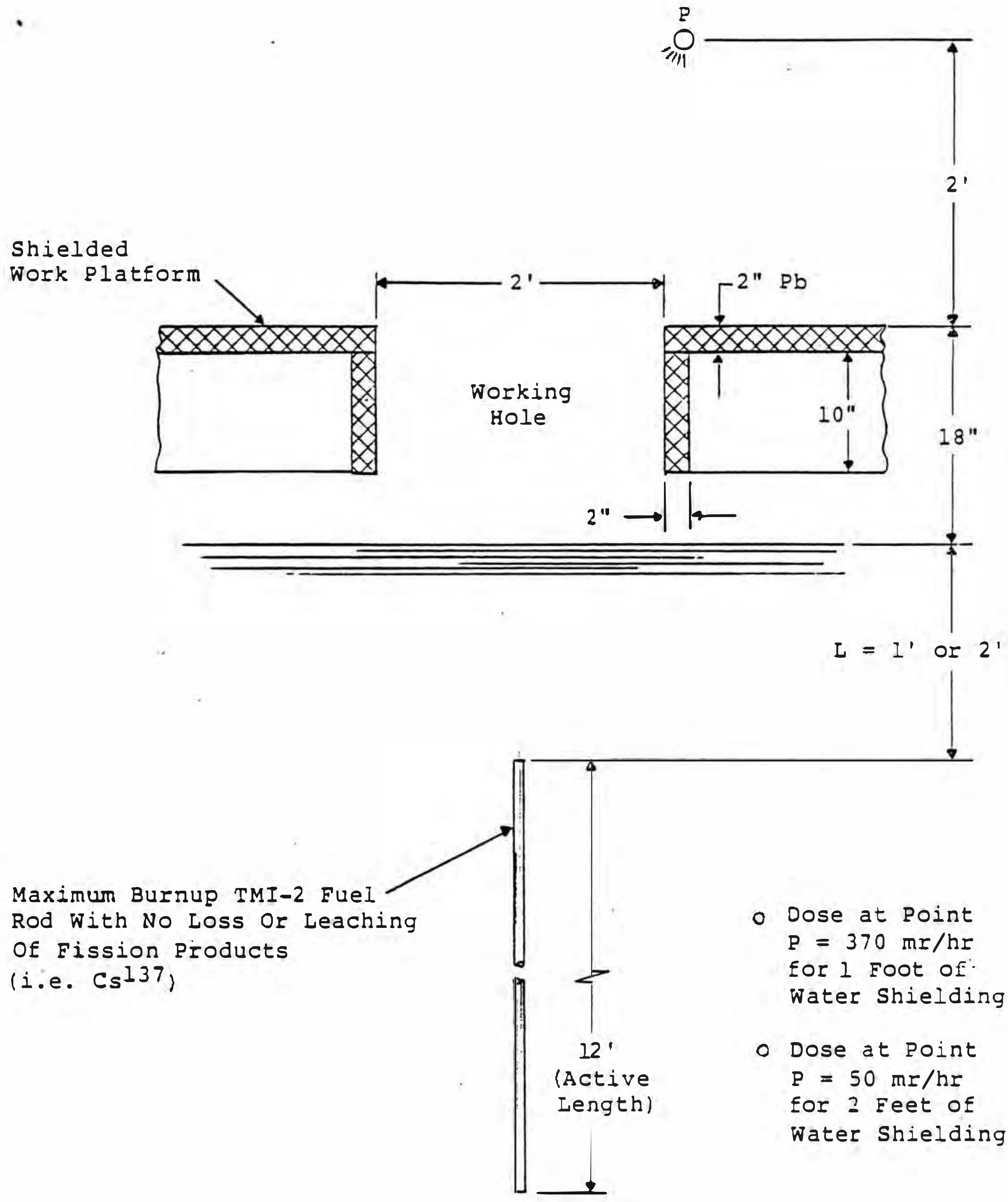
FIGURE 6



DOSE FROM A FUEL CANNISTER
DURING DEFUELING OPERATIONS

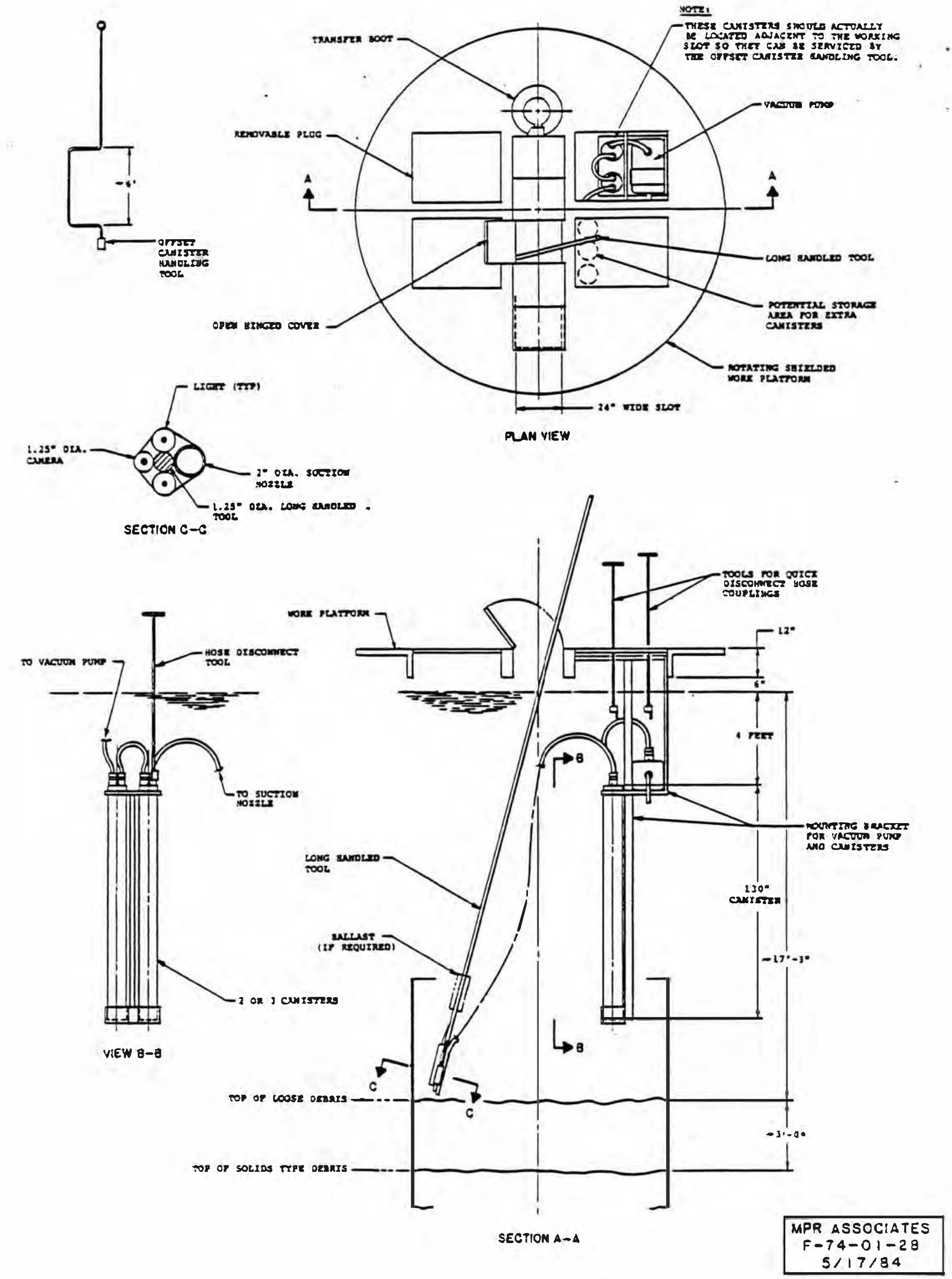
FIGURE 7

MPR ASSOCIATES
F-74-01-22
4/26/84



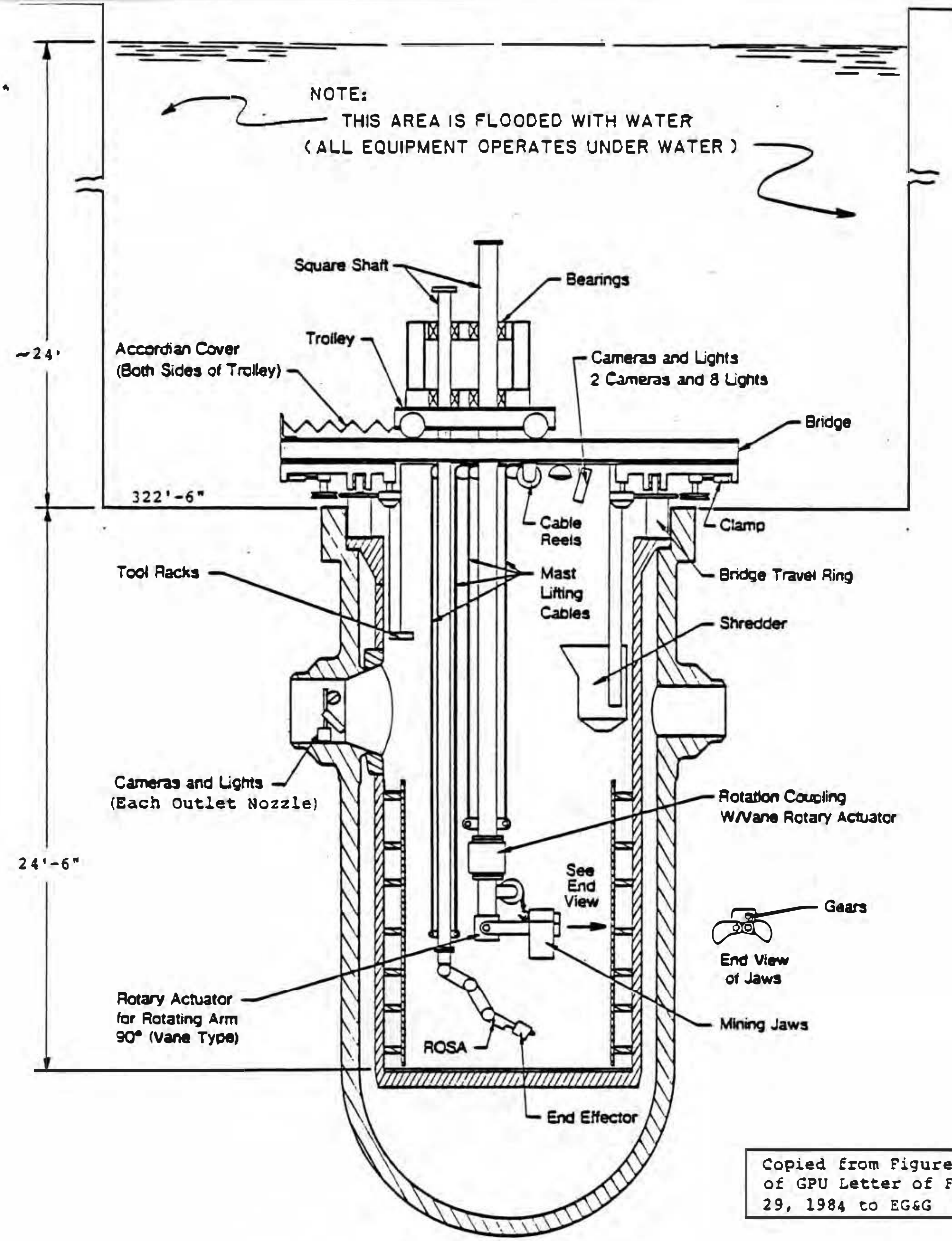
DOSE FROM A SINGLE ROD
DURING DEFUELING OPERATIONS

FIGURE 8



CONCEPTUAL LAYOUT OF LOOSE DEBRIS VACUUMING SYSTEM

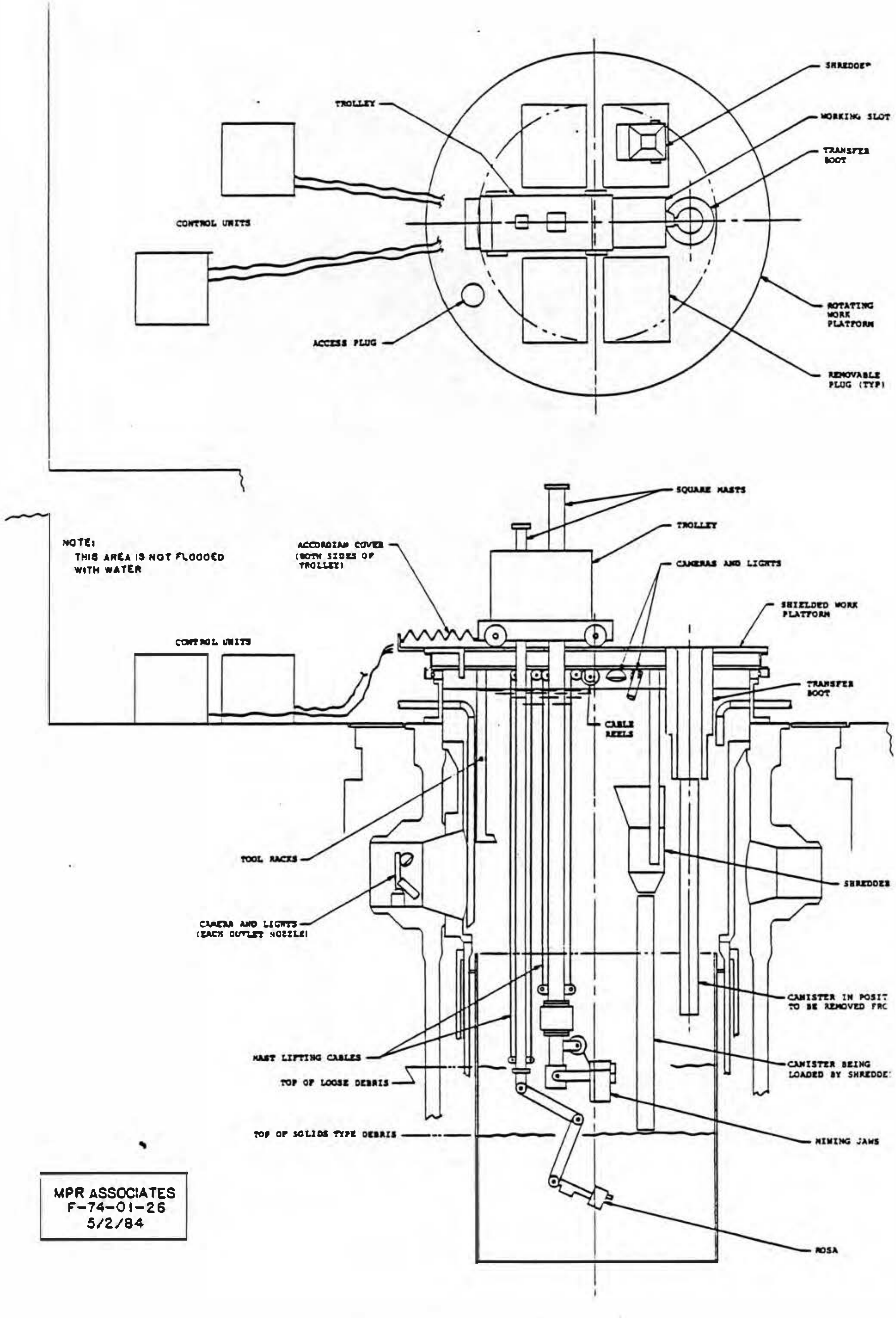
FIGURE 9



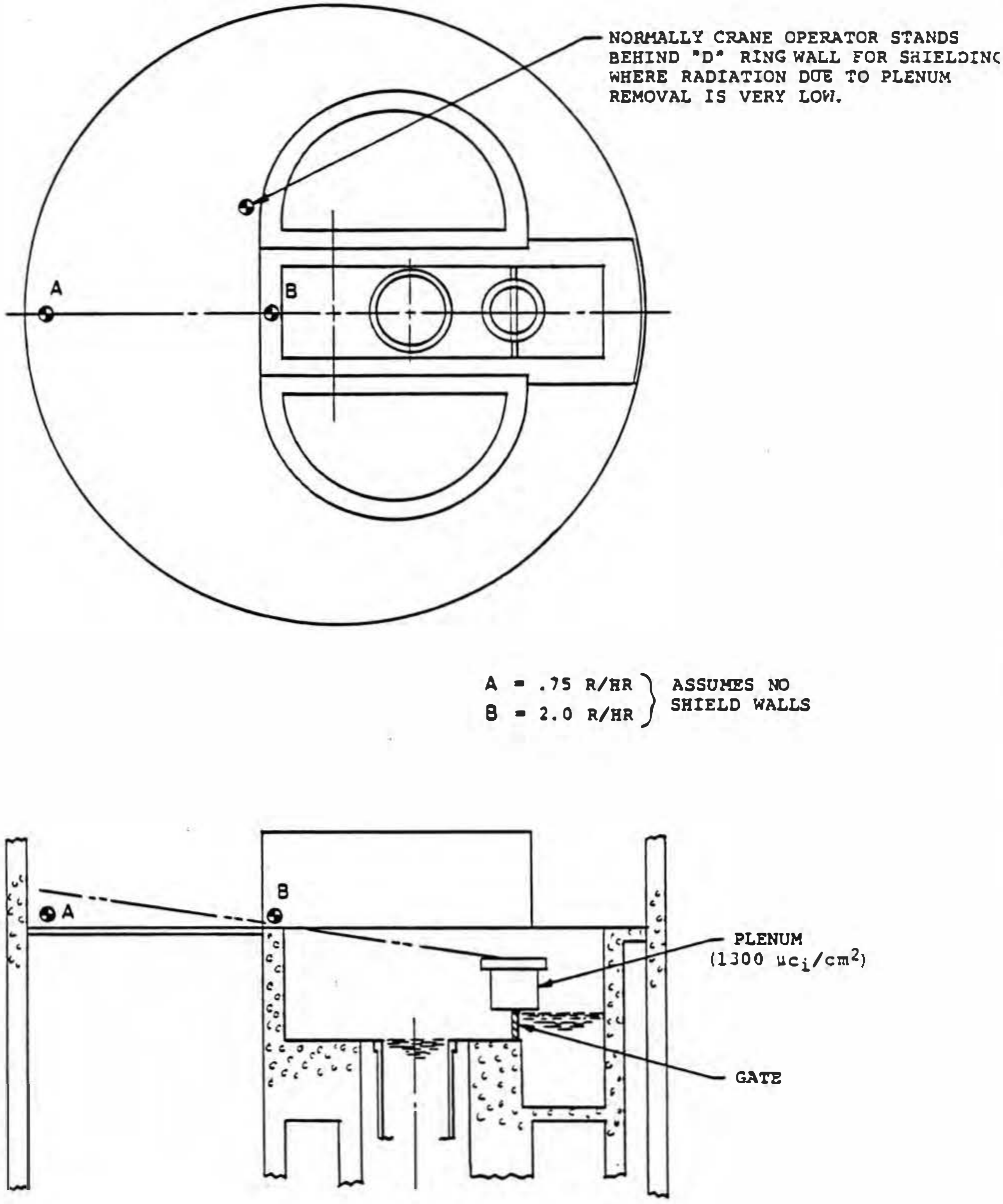
Copied from Figure 1 of GPU Letter of Feb. 29, 1984 to EG&G

PRESENT REFERENCE
TMI-2 AUTOMATIC/REMOTE DEFUELING CONCEPT
(REACTOR CAVITY FLOODED)

FIGURE 10A

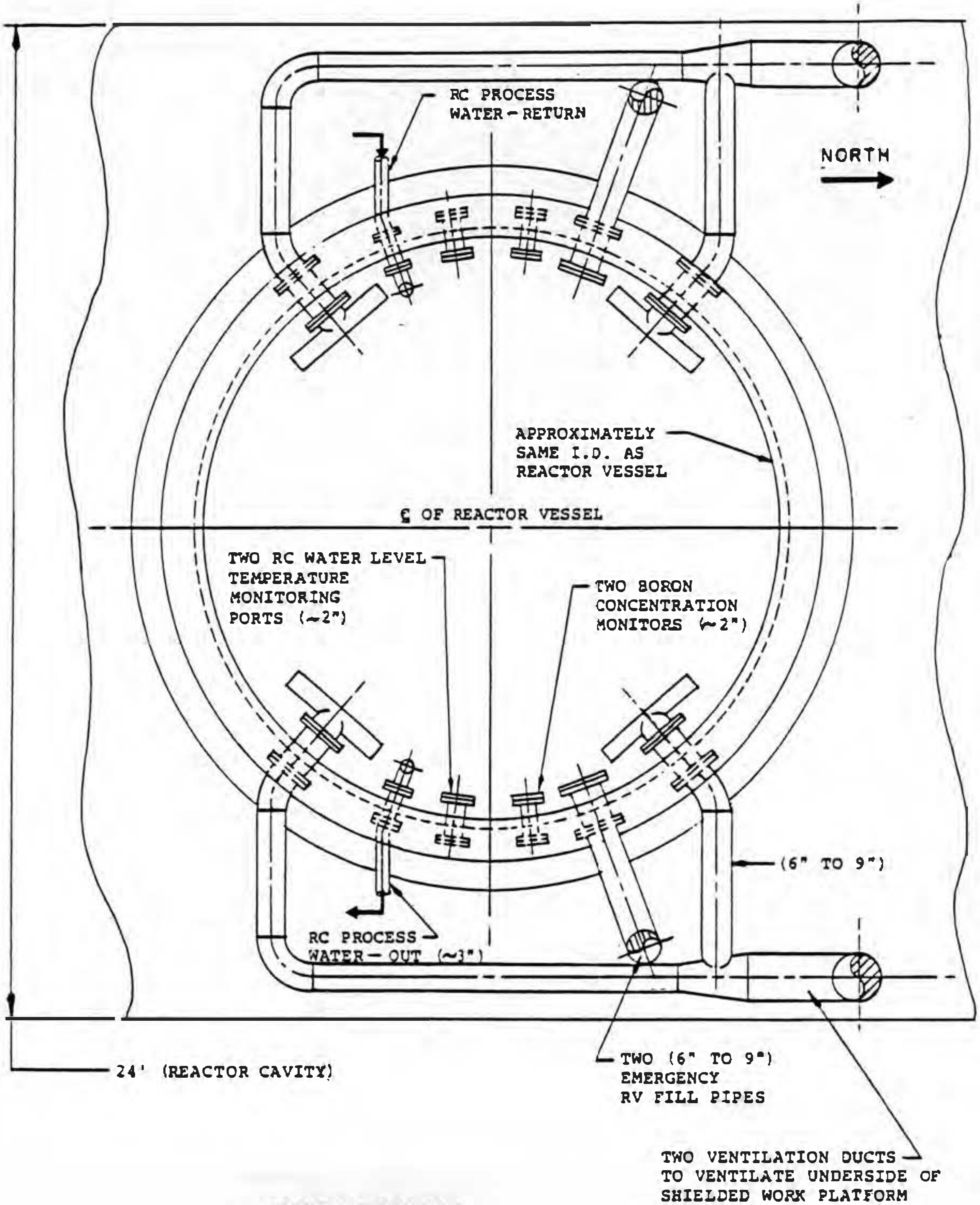


USING AUTOMATED / REMOTE DEFUELING EQUIPMENT
WITH THE LONG HANDLED TOOL DEFUELING CONCEPT



DRY UPPER PLENUM REMOVAL

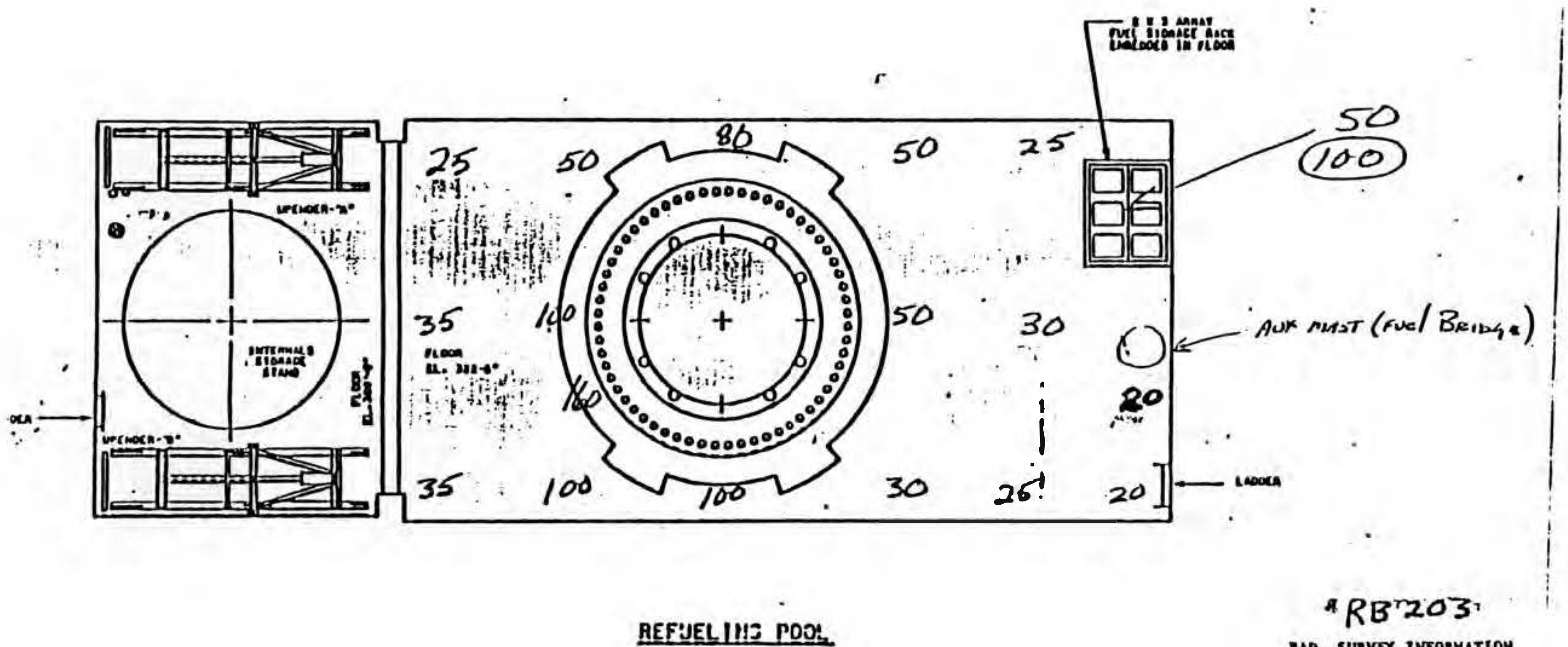
FIGURE 11



CONCEPTUAL LAYOUT OF
DUTCHMAN ON WHICH
SHIELDED WORK PLATFORM
IS POSITIONED

MPR ASSOCIATES
F-74-01-27
5/11/84

FIGURE 12



#RB203

RAD. SURVEY INFORMATION

LOCATION 322 Fuel Pool

REASON Rad Eng Req

DATE 12-22-83 TIME 100

SURVEY REQ. NO. N/A

INST. RO2A CAL DUE 1-19

S.N. 495 B.T. 2

TECH

REVIEWED

REACTOR CAVITY RADIATION SURVEY

Figure 13