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TECHNICAL ADVISORY AND ASSISTANCE GROUP



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March 24, 1982
TAAG-0008

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1
First TAAG
Report

Dear Mr. Burton:

I am forwarding herewith a report entitled "Assessment of Defueling Plan For Three Mile Island, Unit 2", dated March 1, 1982. This report has been prepared by the members of the Technical Advisory and Assistance Group to summarize the work of that group regarding the defueling plan for TMI Unit 2.

It is requested that you distribute this report as indicated on the distribution list.

Very truly yours,

W. H. Hamilton /jeh
W. H. Hamilton
TAAG Chairman

WHH/jeh
Enclosure
cc: Mr. R. C. Arnold (w/o Enclosure)
Mr. H. Feinroth (w/o Enclosure)
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TAAG Members (w/o Enclosure)

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ASSESSMENT OF DEFUELING PLAN

FOR THREE MILE ISLAND, UNIT 2

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Abstract

In November, 1981, the Technical Assistance and Advisory Group (TAAG) was formed to assist and advise GPU Nuclear in the technical aspects of the clean-up of TMI-2. The formation of TAAG was done in cooperation with the Department of Energy and Nuclear Regulatory Commission. The first assignment for this group was to examine the defueling plan which had been developed for GPU Nuclear by their subcontractor, Bechtel National, Incorporated. Reported here are the results of the investigation. A significant part of the TAAG thinking is the importance of obtaining early physical evidence of the actual condition of the reactor internals as a basis for evaluating the existing defueling plan or developing alternate plans. A first step is a suggested program to determine the extent of core damage using control rod motion tests, electrical and mechanical surveys of the in-core instruments, and a visual examination inside the reactor vessel with a TV camera or other optical equipment.

A second part of the TAAG thinking is that emphasis should be placed on early reactor vessel head removal - again in the interest of learning the condition of the reactor internals as a basis for completing the defueling plan. Early head removal also will start the operations which, in themselves, will reveal further need for refinements to the plan. Importantly, early work on the head will enable the start of the defueling operations at a time sooner than now planned.

Other technical activities that are believed to warrant early attention in the future are also identified.

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ASSESSMENT OF DEFUELING PLAN FOR TMI-2

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ASSESSMENT OF DEFUELING PLAN FOR TMI-2

I. INTRODUCTION

A. Background

In mid-1981 GPU Nuclear determined that top priority should be given that portion of the recovery program which would lead to the reactor fuel being removed from the plant. Rather than completing a thorough decontamination of the containment building and clean-up of the reactor coolant system before proceeding with defueling, effort should be directed to lifting of the reactor vessel head and removal of the fuel with only a necessary minimum decontamination. The GPU Nuclear subcontractor - Bechtel National, Incorporated - was asked to develop a plan for fuel removal. Their initial efforts in developing the plan resulted in several issues of a comprehensive planning document, the most recent issue being on January 29, 1982. In parallel with the development of the comprehensive plan, initial work was started on the engineering of fuel handling and core disassembly tools. This work is being done by the TMI-2 Disassembly and Defueling Task Group under the direction of Bechtel Northern Corporation. The work has progressed to the stage of preparing functional requirements for the tooling and special equipment. Also being developed are the plans and specifications for the subcontracts which may be required.

Also in late 1981 the Department of Energy obtained funding authorization to expand their TMI research and development program to include in-situ examination and removal of the damaged reactor internals. The program develops technical data for the commercial nuclear power industry. The data include both experiences during the fuel removal and disposal and also the technology base for accident analysis to be derived from the state of the fuel and reactor structure.

This Department of Energy program supports engineering studies, tests, tooling and technique development in preparation for the defueling and participation in the defueling operations, core examination and fuel storage and shipment. The work is handled through the Technical Integration Office of EG&G Idaho at Three Mile Island.

Early in the development of the GPU Nuclear and Department of Energy programs and the Nuclear Regulatory Commission oversight program there was discussion of having a Technical Assistance and Advisory Group (TAAG). An experience with such a group in 1979 and 1980 with the design of the Submerged Demineralizer System had convinced GPU Nuclear, Department of Energy and the Nuclear Regulatory Commission that such a group could assist effectively in choosing technically sound approaches to the clean-up and defueling by bringing to bear expertise from over the United States.

Accordingly, in November, 1981, a group was formed; membership was agreed to by GPU Nuclear, Department of Energy and the Nuclear Regulatory Commission. The work of the group is funded through EG&G Idaho by the Department of Energy. Attending TAAG meetings as observers are representatives of EG&G Idaho, Department of Energy and Nuclear Regulatory Commission. The TAAG provides technical assistance and advice directly to the President, GPU Nuclear and documents its results to EG&G, Idaho to assist in execution of the DOE program.

The first task of the TAAG was to investigate the defueling plans already under development, as follows:

Perform an overview of the Bechtel National Defueling Plan, identifying those areas needing additional work and priorities for such work.

As part of the overview, give specific attention to alternative approaches.

Among specific issues to be considered is the question of fuel fines dispersed within the fuel handling canal, and its potential effect on habitability, tooling design, pool clean-up systems, etc.

Objective is to have a preliminary report by the beginning of March 1982.

An initial informational and educational meeting was held on December 21 and 22, 1981. Additional meetings were held in January and February. Also, some experimental work was done and some analysis and drawing work was performed by members of TAAG in their home organizations and through EG&G Idaho.

B. The TAAG Approach

In the first TAAG meetings the defueling plan was presented by representatives of the Bechtel National, Inc. and Bechtel Northern Corporation. The plan was documented in "Technical Plans for Reactor Disassembly and Defueling", initially issued February 26, 1981 and subsequently revised June 9, 1981 and January 29, 1982. The plan is a comprehensive and thorough document, laying out in detail the sequence of events required to remove the fuel from the reactor vessel and to store it in containers in the fuel storage canal. For each step in the sequence detail sheets are available, each describing the equipment requirements and technical problems to be resolved.

TAAG used this document as a basis for its deliberations. Detailed comments on the document are included in this report, section III.

The Bechtel plan is based on the core damage being within the range described in GEND 007, "TMI-2 Core Status Summary: A Basis for Tool Development for Reactor Disassembly and Defueling, February, 1981", prepared by EG&G, Idaho. But, as indicated in this report, there is a great deal of uncertainty in the conditions of the core and, hence, realistic plans can only be based on a broad spectrum of core conditions predicted to exist in the reactor vessel. There is also the possibility that a portion of the fuel has been swept into the reactor coolant and other plant systems. At the other end of the spectrum, some opinions are that the amount of damage is minimal and that most fuel assemblies can be removed using a normal defueling procedure.

From the outset the TAAG members were concerned about the basic assumption of the Bechtel plan, namely the core condition is within the bounds described in GEND 007.

Recognizing the uncertainty and leverage involved in the basic assumption for the plan, the TAAG members initiated a study of the possibility of accelerating the Bechtel plan for inserting a TV camera or other optical equipment into the reactor vessel to obtain visual verification of the condition of the reactor internals.

This visual inspection is referred to as a "quick look".

Initially, it was uncertain whether such an inspection would provide useful data or that the effort would be worth the time and occupational radiation exposure involved. Hence, some experimental and engineering work was initiated to help resolve the questions.

As that work progressed, other means of assessing the extent of core damage at an early time became apparent. These included moving control rods and testing the in-core instruments. Hence, in addition to providing some observations and engineering suggestions regarding the Bechtel defueling plan in this report, it is also proposed that careful consideration be given to determining the extent of core damage, first, by using the suggested control rod motion tests, the instrument tests and the "quick look" and, second, by emphasizing preparations for the lifting of the reactor vessel head at an early date and by surveying the systems for fuel deposits. In the thinking by TAAG members, both the first and second steps would provide information about the status of the reactor internals and permit a more focussed way to prepare for the fuel removal.

While the Bechtel defueling plan was comprehensive and well thought-out, the basic assumptions regarding core conditions might not be correct and hence the plan could lead to inadequate preparations, on the one hand, or be overly expensive, on the other.

If the core is more severely damaged than predicted by GEND-007, e.g., a "solid" mass in the bottom of the vessel, deformation of the core support structure or widely dispersed fuel fines in the coolant loops, a different emphasis in the preparations for defueling would be required. The development of milling or impact machine for vitrified material might be required. On the other hand, removal of large amounts of fuel from the loops would require special processes, not now being contemplated in the systems decontamination program.

If the core is less severely damaged than assumed, then the \$15 million planned for special tooling and the time required to build the tooling would not be required. Also time related expenditures during critical path tooling development would be several million dollars per month. Lesser damage would allow normal refueling tooling to be used, with some provision for individual fuel element failures - such as used by other nuclear plants. Not only would extensive special equipment not be required but complex procedures and special training of personnel also could be avoided.

TAAG also held a summary session with outside experts who could brief the members on the predicted condition of the core or on the degree of uncertainty in any prediction. This session confirmed the belief of the TAAG that the spectrum of possible core conditions is too broad - as of today - to permit the development of detailed plans and specifications. Further, exploration of core conditions using relatively simple and known techniques would be helpful in narrowing the spectrum.

In examining the defueling plan certain other auxiliary systems and equipments were examined. These are described in Section IV of the report. Since lifting equipment is important throughout the entire defueling and exclusive dependence is placed upon the installed reactor building polar crane, Section V of the report is devoted to that subject.

During the most recent TAAG deliberations, the TAAG was informed of a new development which will effect the pace of implementation of the defueling activities; namely delays in restart of Unit 1 will defer availability of needed funds to conduct the defueling by possibly a year or more. It appears that this development emphasizes the need for obtaining early knowledge of the actual conditions of the core to permit more efficient utilization of resources when they do become available.

II. APPRAISAL OF CORE DAMAGE

A. Summary of Prior Investigation

The reference document for core damage assessment, which was supplied to the TAAG, was GEND-007, "Three Mile Island Unit-2 Core Status Summary: A Basis for Tool Development for Reactor Disassembly and Defueling". This document dated May, 1981, summarized published data available at the time it was written. The specific mechanisms used for core damage estimates were:

- (1) Thermal-hydraulic computer calculations using water balance data derived from plant records.
- (2) Hydrogen generation estimates from containment atmosphere analysis and computer modeling of hydrogen production.
- (3) Fission product release measurements from analysis of water and gas volumes resulting from the accident, and interpreted in terms of fission product release experiments conducted in Reactor Safety Research programs.
- (4) To a lesser extent, other observations such as in-core instrument data and post-accident flow blockage.

The conclusions derived in GEND-007 were summarized in terms of maximum, minimum, and reference (intermediate) limits on core damage. The following paragraph from that report states the considerations of greatest impact on fuel removal.

"Factors of primary interest during reactor disassembly and removal of the core are the condition of the upper plenum, the amount of cladding oxidized, the presence of once molten materials such as liquified fuel and control rods, and the condition of the instrument and guide tubes. Some components of the upper plenum structure may have melted or fused together during the course of the accident necessitating the development of tooling and procedures for this contingency. It is evident that a bed of fragmented fuel and cladding has formed, perhaps extending to the core periphery. A few of the upper plenum components may rest on top of the debris. The amount of cladding oxidized, approximately 50%, is indicative of the fraction of the core which is brittle or fragmented. The presence of liquified fuel, or any once molten material, is enough to ensure that some areas of the debris will be fused together and that separation techniques and tools must be designed accordingly. The total weight of potential debris and embrittled cladding is 64,000 to 83,000 kg (140,000 to 184,000 lb)."

Some additional studies have been published subsequent to the reference report, notably NSAC-24. In general, these fall within the bounds of the above estimates, although they trend toward the higher part of the range. At the same time there has developed a "minority opinion" that fuel damage may be significantly less severe based on the following:

- (1) the basis for thermal-hydraulic calculations is uncertain;
- (2) substantially less hydrogen may have been generated and the production of part of the hydrogen was by mechanisms other than the zircaloy - steam reaction, and
- (3) some experimental data support a decrease in estimated fission product release as well as attribution of part of the released fission products to fuel leaching rather than release during the accident.

The TAAG spent considerable time evaluating this information and invited several scientists in the foregoing subject areas to present their evaluations. There was a general agreement within TAAG that evidence is indeed strong that a substantial fraction of the fuel (in the range of 50% with considerable uncertainty) did reach a very high temperature (approaching or possibly exceeding 2000°C). The contrary conclusion - that fuel damage was not severe - would contradict the entire basis (both experimental measurements

and computer modeling) of the present analytical methods used in many reactor safety analyses. Further consideration is described in Section II-C.

Direct Measurements on Systems. After the accident was controlled and the system stabilized, there was little done in the way of direct measurement on the system except for that required for continued control (such as analysis of all significant quantities of water and especially periodic analysis of the reactor coolant). For example, there has been no attempt to date to move control rods (see Section II-B-1.), to experiment with the instrument probes (see Section II-B-2.) or to identify the presence of fuel in the reactor coolant system piping outside the reactor vessel (see Section II-B-3.), although the presence of fuel in the system piping would be extremely important to the fuel removal strategy (estimates vary from very small amounts of fuel to tons). Such measurements have been made with other reactors in the past, using extreme collimation with γ -spectra analysis to monitor system components at a distance of several feet.

There is indication of transport of fuel and other materials out of the core vessel, via the analysis of several samples, especially of solids. These include solids in at least two samples of reactor coolant, two samples of water from the floor of the containment building, and, most definitive, a sample from a filter in the let-down/make-up system (MU-F-5B). The latter solids contained

approximately 6% uranium (fuel), 6-12% silver and 11% cadmium (control rods) and predominantly zirconium (cladding). There is no indication of the quantity of such solids on this filter or in any other location in the system. Further consideration of this approach is described in Section II-B-3.

In-core Instruments. For a few weeks following the accident a significant effort was expended on measurements with the in-core instruments (as well as many other endeavors). The individuals who did this work subsequently returned to their places of employment. To date, there is no ordered and coherent summary of this work, although there may be one in preparation. These instruments are described in Section II.B.(2).

There are general statements, some quoted frequently, to the effect that 51 of the 52 thermocouples survived the accident, along with some fraction of the self-powered neutron detectors (those in the lower and peripheral positions). "Survival" has been used by some analysis to impute that the instruments did not reach temperatures high enough to damage them. Upon examination, "survived" means only that the instrument continued to generate the signal it was supposed to give and its value seemed reasonable. Even in such cases, however, other connotations of the word, "survived", may be completely inapplicable. We found no adequate interpretation of this, although there is a common impression that such "survival" was in conflict with other core damage estimates; instead, it is concluded there is no evidence for physical survival of the thermocouples.

Many measurements of temperature indication have been made since the accident. Only limited measurements of the resistance of the instrument leads and attempts to measure the length of some probes by time domain reflectometry have been conducted. These measurements did not lend themselves to straightforward interpretation, and there appears to have been little attempt to draw conclusions about core damage from this.

Some time after the accident there was a proposal by Babcock & Wilcox to attempt to gather several types of information via the in-core instruments. Most, but not all, of the proposed measurements required access to the instrument terminations at the in-core service area, which is an area too radioactive for such access. None of the proposed tests were approved.

The TAAG concluded that significant information might be obtained through the in-core instruments by means of carefully designed measurements (not necessarily what the instruments were intended to measure), and that much of this could be obtained from outside containment. This information is described in Section II-B-2.

B. Prompt Steps to be Taken

1. Control Rod Motion Tests

a. Background

Based upon a review of the core design, upper plenum structure, control rod configuration and control rod mechanism design, it has been concluded that special tests exercising the TMI-2 control rod motors may provide additional information on the condition of the core.

The rod testing discussed in this section has been reviewed with GPU and DOE. GPU and DOE have agreed to perform these tests with funding to be provided by DOE through EG&G. The static testing has been successfully completed. Control rod mechanism proof/calibration testing at Diamond Power Specialty Corp is scheduled for performance during the first two weeks in March 1982. Criticality studies to support the safety evaluation for this testing are underway at Babcock and Wilcox. A specific date for the actual testing in TMI-2 will be dependent on the completion and approval of the safety analysis currently scheduled for spring 1982.

The TMI reactor has 69 movable control rods distributed throughout the reactor. Sixty one of the rods are "shim" rods which have scram capability and which were inserted fully into the core during the accident. The remaining eight rods are the Axial Power Shaping Rods (APSR). These rods do not scram and were at the 27% withdrawn position (about 35 inches out) at the time of the accident. Figure II-1 identifies the location of the

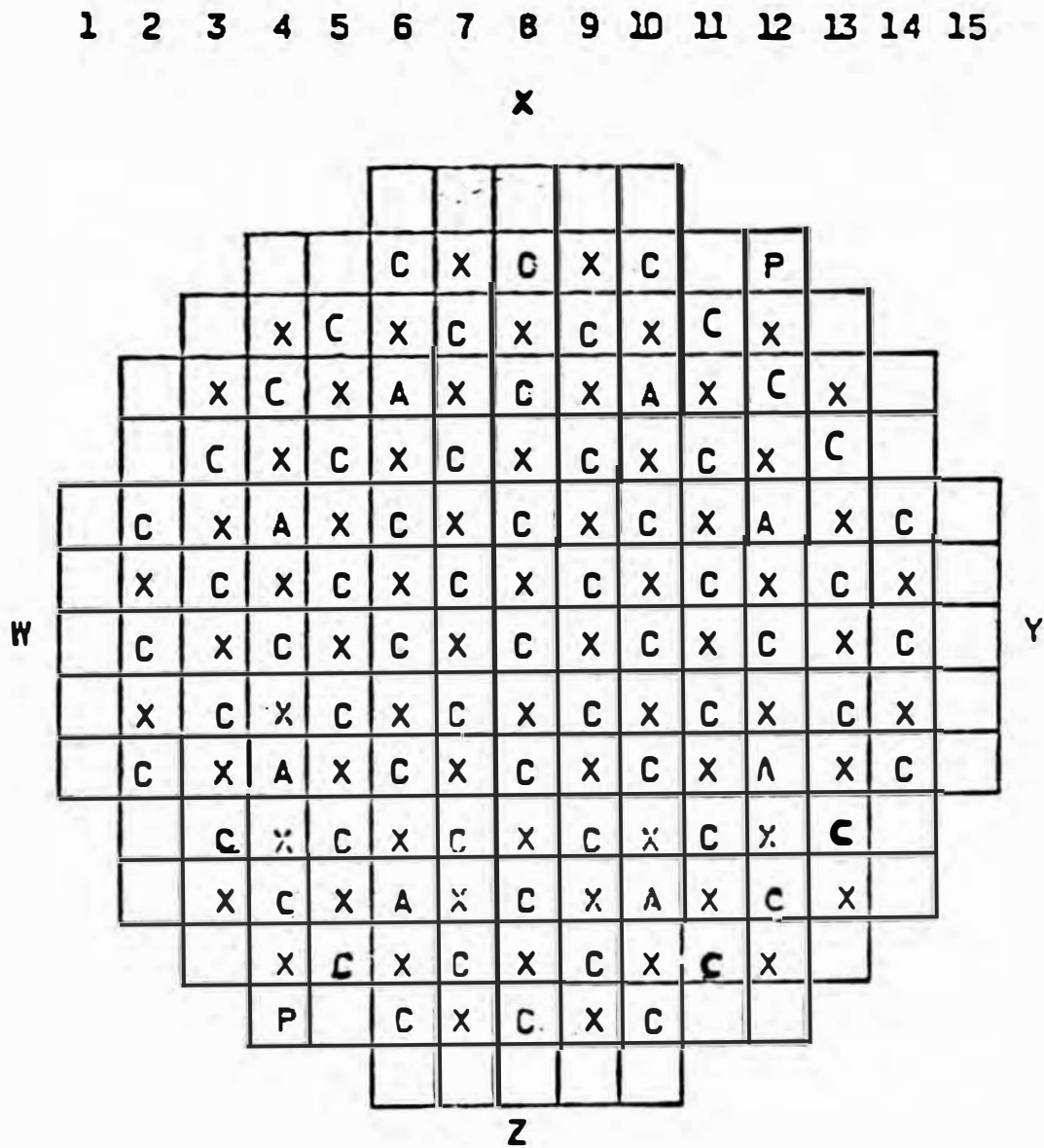
control assemblies in the TMI-2 core. Current predictions of damage resulting from the accident would suggest that a majority, or all of the control rods, both shim and APSR rods have melted and are not attached to the control rod lead screw. In addition predicted upper plenum accident temperatures suggest the possibility that the plenum structure may have been distorted or suffered melting of brazements and/or structure.

The purpose of the proposed tests is to determine through control rod motion;

- (1) If plenum distortion or melting has occurred to the extent rod motion is interfered with or prevented.
- (2) The extent of damage to the control rods, i.e. the weight of control rod remaining attached to the leadscrew.

In addition to providing the above information these tests will be used to position the APSR rods in the fully inserted position specified in Step C12 of the "Technical Plan for Reactor Disassembly and Defueling"

(TPO/TMI-005). Should these tests from outside the reactor building be successful, their accomplishment from outside the reactor building will reduce the radiation exposures and eliminate steps now planned to be done on the ASPR's from inside the reactor building.








-  CONTAINS NO CONTROL COMPONENT
-  CONTROL ROD ASSEMBLIES
-  AXIAL POWER SHAPING ROD ASSEMBLIES
-  BURNABLE POISON ROD ASSEMBLIES
-  PRIMARY NEUTRON SOURCES

FIGURE II-1 LOCATION OF CONTROL ASSEMBLIES

These rod motion tests could also be used to assist in selection of rods through which to make the "Quick Look."

b. Test Method

The proposed rod motion tests will use the installed control rod drive mechanisms. These mechanisms will be driven from the relay room outside containment. The power supply used will be the available single rod service power supply. Initial testing will be limited to the APSR rods. The results of this testing will be used to evaluate the desirability of testing selected shim rods. The test is performed in two phases, the static testing and dynamic testing.

(1) Static tests

The purpose of the static tests is to determine the electrical properties of the control rod drive mechanism stator and associated cabling. The results of these tests will be used to predict the electrical operability of the control rod mechanisms. The tests consist of the following measurements made at the relay room end of the electrical cables to the mechanisms.

1. Insulation resistance
2. Power circuit resistance
3. Power circuit inductance
4. Reflectometry and capacitance of power circuit

These tests, which had already been under preparation by EG&G, have been completed for the eight APSR mechanisms. The results indicate that the stators and cables associated with these mechanisms appear to be undamaged and are therefore suitable for operation in the dynamic tests.

(2) Dynamic tests

The purpose of the dynamic testing is to determine if the rods will move and what weight is lifted by the mechanism. The procedure will attempt to measure the force required to move the rod in order to determine 1) if sticking of the lead screw has occurred and 2) the mass of the control rod attached to the lead screw. The technique that will be used to measure this force will be to increase the electrical power to the mechanism until pole slippage stops indicating rod motion has occurred. Pole slippage occurs when the magnetic field generated by the stator moves but the rotor remains stationary. The TMI 2 control drive mechanism has

six separate windings in the stator. These windings are energized in pair-triple sequence to rotate the field in 15 degree increments.

Pole slippage will be detected by either listening to the mechanical noise generated by pole slippage through the use of a microphone or by observing the wave form of the power supplied to the mechanism. The use of pole slippage should provide a sensitive indicator of rod motion not subject to the inaccuracy of the rod position indicators ($\pm 1\frac{1}{2}$ inches). Determination of load to pole slippage will result in small rod motions.

Upon completion of the testing to determine whether the ASPR's will move, the determination can be made to position them in the fully inserted position.

Supplementary tests will be performed at the mechanism vendor's (Diamond Power Specialty Corp.) plant. These tests will be made to confirm the ability to detect rod motion through pole slippage measurements. In addition calibration curves will be developed for the shim and APSR mechanisms. These curves will plot mechanism power for mechanism motion, both in and out, as a function of load. These tests will be made for a dry mechanism

and a mechanism filled with water. The results of these tests will be used to develop the actual test for the TMI-2 APSR mechanisms.

c. Safety Evaluation

GPU has requested that TAAG assist in the preparation of the safety evaluation required to support the above rod motion tests. The primary safety consideration associated with these tests relate to assuring that the reactor remains shutdown during the tests. The approach under evaluation is to assure reactor shutdown through the use of dissolved boron in the primary coolant. It is proposed that sufficient boron will be provided to assure that criticality will not occur under any conceivable condition. Details of these criticality studies are presented in Section V.D of this report. Special procedures will be used to monitor the boron concentration during the test to assure that the required degree of subcriticality is maintained. Procedures will be prepared to terminate testing if the boron concentration should decrease and other procedures will be used to increase the boron concentration in this unlikely event.

2. Instrument Survey

a. Description

The TMI-2 reactor is equipped with in-core instrumentation in 52 fuel assemblies distributed throughout the core. The instruments are housed in sheathed detector assemblies that pass through pipes from the 347 foot elevation floor, through nozzles on the bottom of the reactor vessel and into the 52 instrumented fuel assemblies. Each of the detector assemblies contains 7 self powered neutron detectors (SPND's) one average flux detector, one thermocouple (TC), and a central calibration through into which other instruments can be inserted into the core during operation. One of the detector assemblies in fuel assembly N-8 is equipped with a movable detector which was parked about 9 feet below the core at the time of the accident. Figures II-2 and II-3 show the arrangement of the in-core detectors and their position in the core.

The existence of the in-core instruments provides potential for obtaining further direct evidence of the core condition, either through additional measurements of the existing instruments or through use of the access arrangement to introduce other measuring devices. The purpose of making these measurements is to obtain early evidence of specific characteristics of core condition or of the location of fuel. This information could be useful either separately or in conjunction with other efforts described in this section to appraise the core condition.

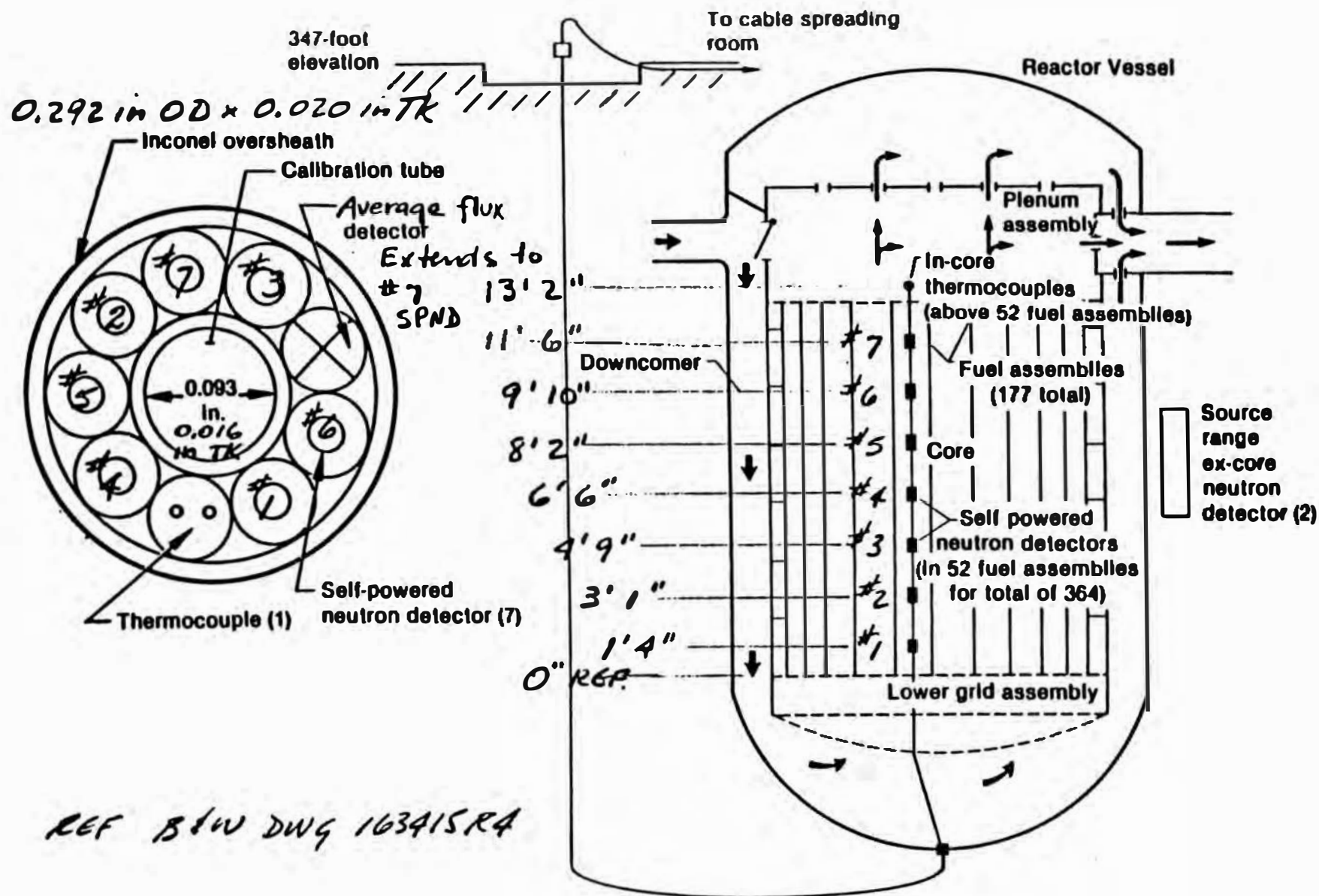


FIGURE II-2 INCORE DETECTION ARRANGEMENT

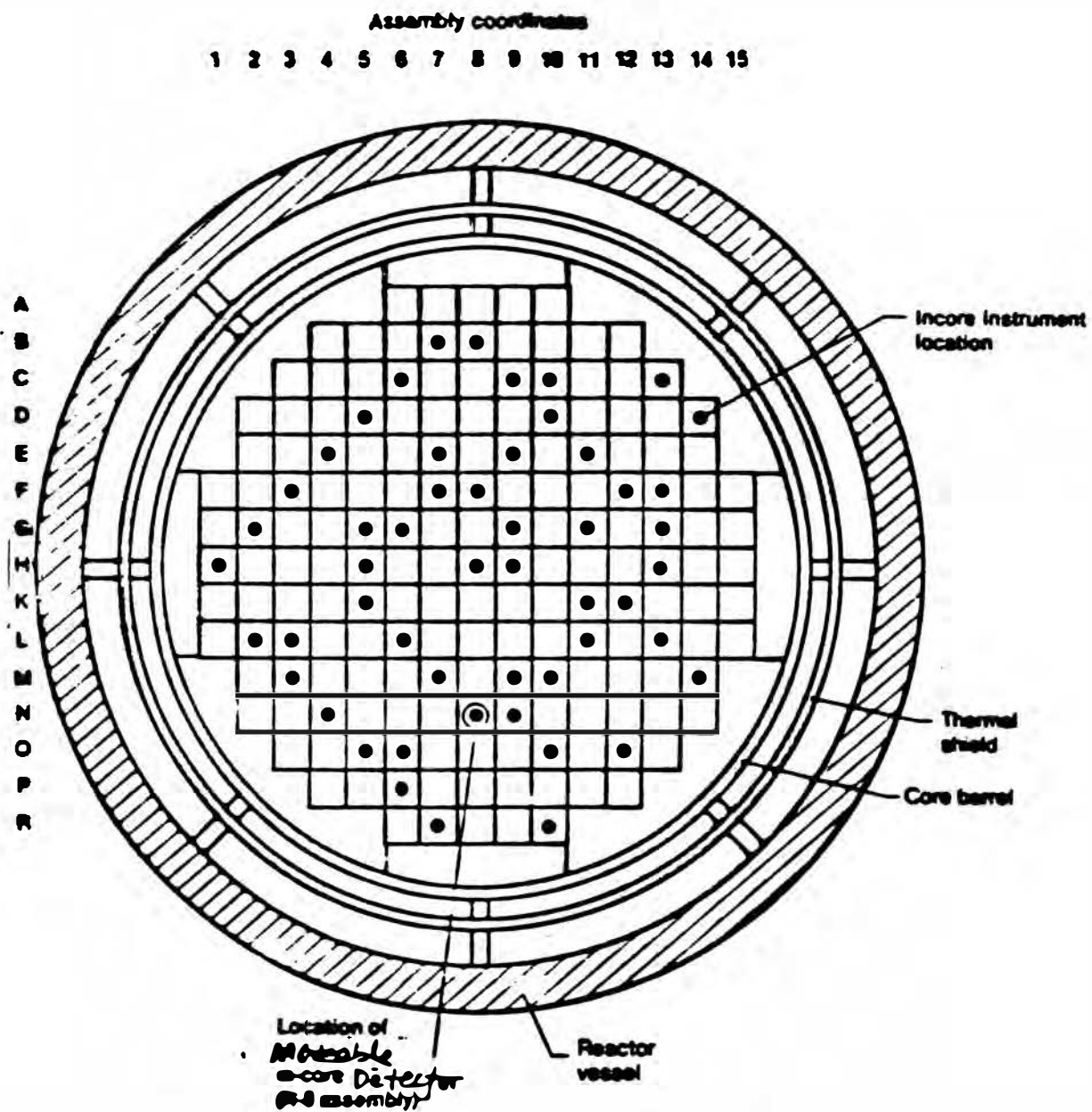


FIGURE II-3 IN-CORE INSTRUMENTATION POSITION

For example, GEND-007 indicates that the core support and lower core grid plate have not experienced damaging temperatures or loads. However, certain specific observations are unexplained. In particular, at about 227 minutes after the start of the accident all the SPND's in 8 detector assemblies (G9, F7, F8, E7, G6, G5, H5 and M9) went off scale within a minute. Even the lowest SPND's in these detectors, which are about 8 inches above the bottom of the active fuel, went off-scale and subsequently failed electrically. Also, the movable in-core SPND in the N8 position was found several days after the accident to have failed, even though it was parked about 9 feet below the core at the time of the accident.

Although these instrument readings could be explained by damage to cabling or connectors external to the detector assemblies, they could also indicate damage at the lower part of the core which is outside the range of reference damage for which the planning is being done.

Another condition that is not addressed in GEND-007 is the potential for high temperature damage to the core support. The upper plenum is believed to have experienced distortion or melting in the region just above the top of the fuel. However, the core support, which is loaded with the weight of the core, is not protected from similar high temperatures.

If the core support experienced high temperatures and lost strength, it could have achieved a final position on the stop lugs in the reactor vessel just below the core support. The shifting in position of all the core components could impose distortions not accounted for in planned tooling or defueling activities.

GEND-007 also concludes that 32,000 to 58,000 kg of loose core debris exists. It portrays the bulk of this loose debris to be laying on the lower, less damaged core remnants. The possibility that substantial amounts, i.e. tons, of the loose debris moved from the original location is not addressed. A reasonable probability exists in the judgement of TAAG that a substantial fraction of loose fuel debris could have been moved into the loops or into the bottom of the reactor vessel, under the core support.

In summary, additional use of the in-core instruments or their access arrangements could provide useful planning basis about specific possible damage conditions that are outside the present reference damage conditions as described by GEND-007. TAAG recommends that such a program be conducted.

b. Preliminary Tests of Existing Instruments

The incore instruments were in service and their signals were measured at various times during and after the accident. Following the accident Babcock and Wilcox, and Oak Ridge National Laboratory experimentally determined the characteristics of the SPND's and TC's under abnormal conditions to enable interpretation of the signals measured during the accident. These organizations, Electric Power Research Institute, and Idaho National Engineering Laboratory also analyzed the recorded signals and other data to determine conditions in the core during the accident.

The consensus of these analyses is generally consistent with the assessment of core damage in GEND-007. However, certain anomalies, such as the apparent survival of most of the TC's, appear in the various analyses and tend to bias the assessment toward the less severe range of damage. In the judgement of TAAG the apparent survival of the TC's is explainable and should not be interpreted to mean that the TC's are intact in their original locations. Their apparent survival is a characteristic failure mode of TC operating in a pressurized, conductive solution and has been experienced in test loop operation.

Similarly the measurements that showed the failure of most SPND's within several days after the accident should be interpreted as an indication of damage in core areas such as peripheral fuel assemblies and near the core bottom. Figure II-4 shows the locations of detectors that failed at the lowest level in the core.

These judgments are supported by the observation from photographs of the detector assemblies at the in-core service area. One of these photos is shown in Figure II-5. The white encrustation on the calibration tube of assembly D14 is positive indication of leakage through the detector assembly sheath and the calibration tube.

Further measurements of selected SPND's and TC's are being conducted by DOE and EG&G. These measurements include reflectometry which may be capable of determining the effective length of the instruments. Also a "crosstalk" measurement will be made by applying a signal to one instrument and measuring the response from other instruments in the same detector assembly. These measurements are planned for February and March 1982. They could provide direct evidence of the location of damage and be of particular significance in showing conditions at or below the bottom of the core. TAAG endorses these tests and urges their completion and use of the information in planning the disassembly and defueling.

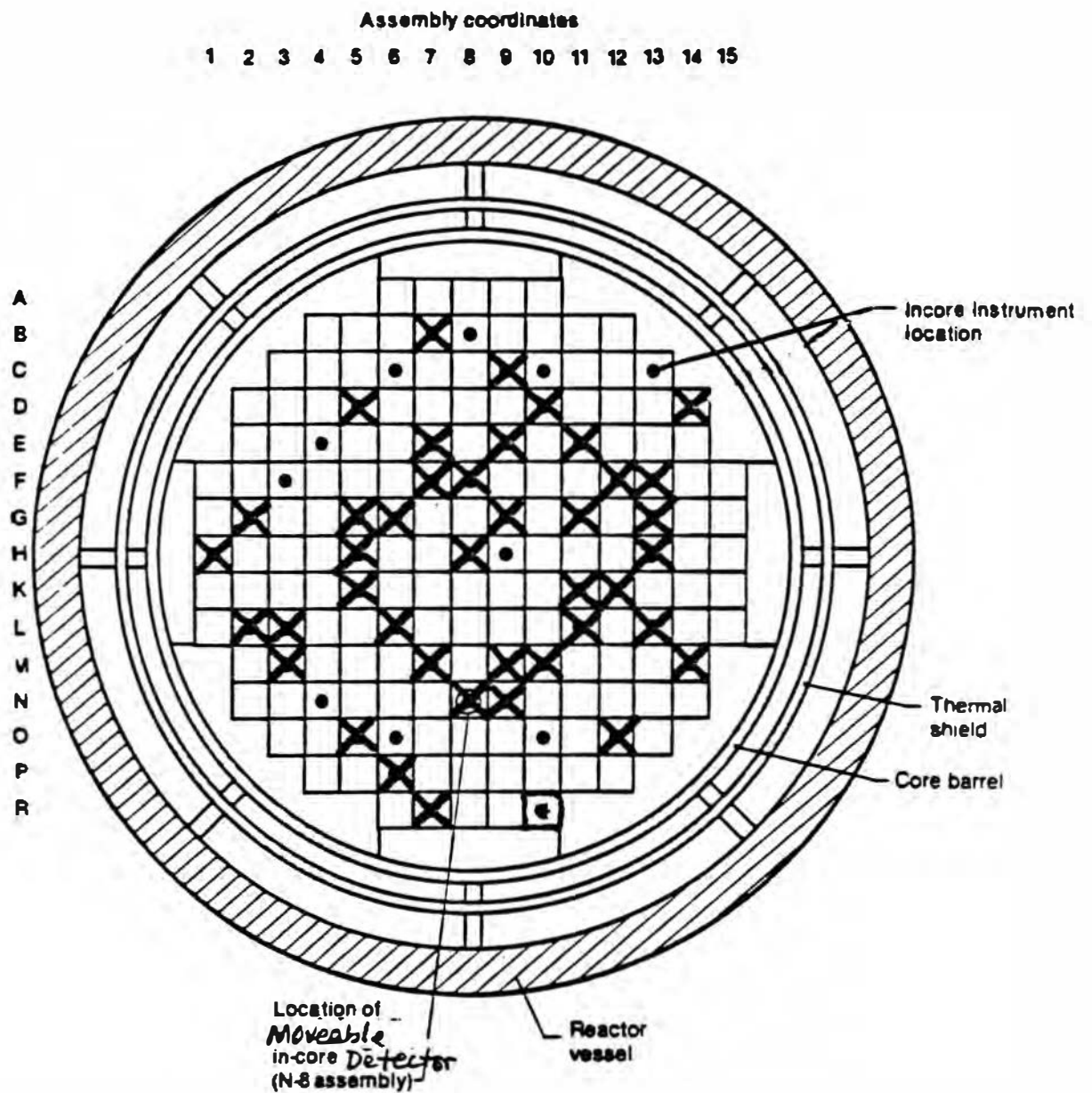


FIGURE II-4 LOCATION OF FAILED LEVEL 1 SPND'S

3

NOTE: X Indicates Failed SPND's by Electrical Measurements April 13, 1979.

CALIBRATION TUBE

WHITE ENCRUSTATION

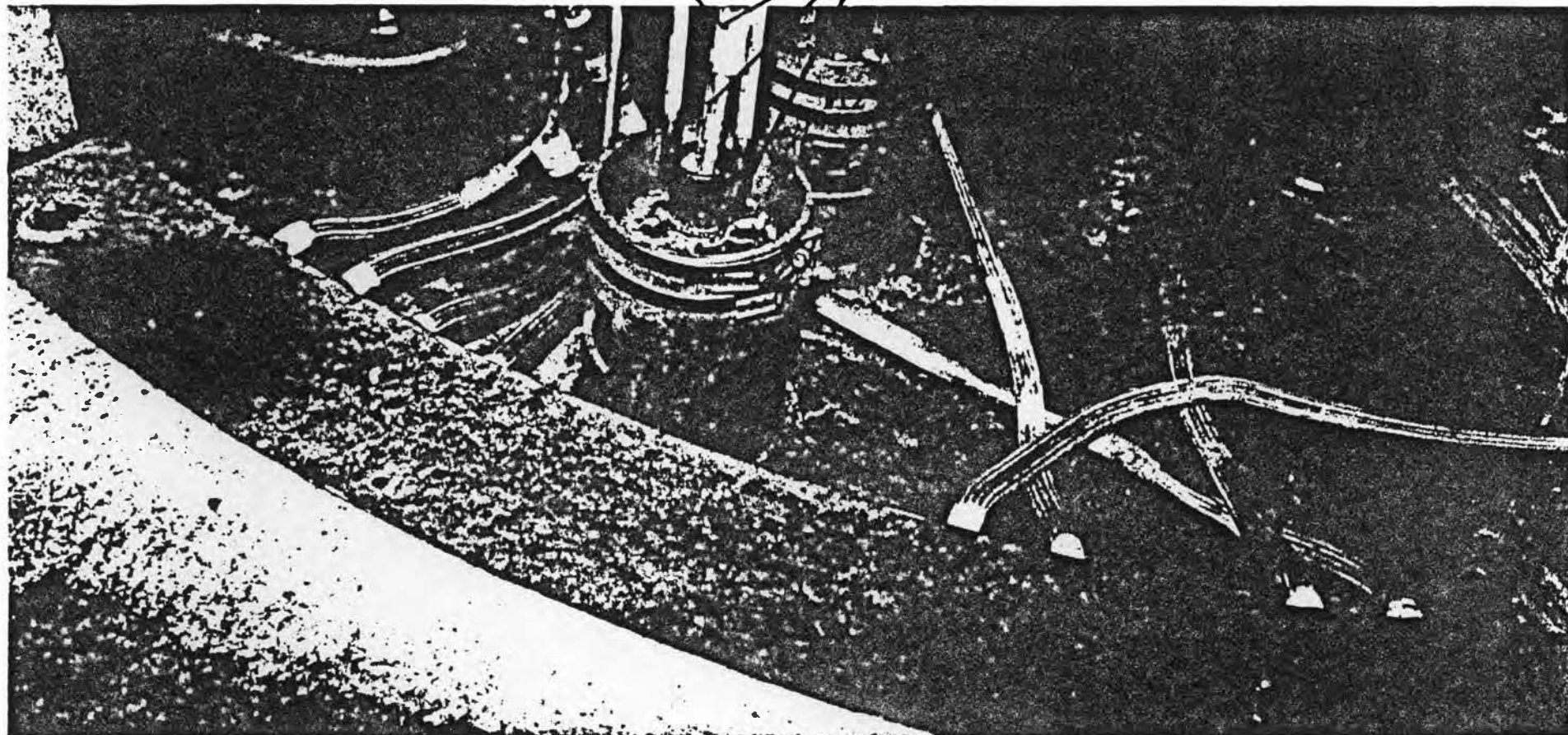


FIGURE II-5 Photograph of TMI-2 In-core Instrument Service Area (Post-accident)

c. Other Measurements Using In-core Access

TAAG further suggests that a program of additional measurements be considered in conjunction with other defueling activities. These suggestions are based in part upon the earlier Babcock and Wilcox document, "Preliminary Planning and Design Effort for TMI-2 Core Condition Mapping via Incore Detectors and Instrument Tubes", September 1980.

Possible measurements were described in this document that require removal of the incore detector assemblies from the guide tubes. Various detectors could then be inserted into the core through the guide tubes. TAAG recognizes several practical conditions that cause such measurements to be particularly burdensome including:

- (1) The TMI-2 plant is not equipped with the equipment to remove the detector assemblies.
- (2) The detector assemblies are highly contaminated and probably have radioactivity trapped in areas difficult to clean.
- (3) Removal of the detector assemblies could disturb the integrity of the external guide tubes which are fluid boundaries in the bottom of the reactor vessel.

TAAG suggests that measurements that require detector assembly removal be included only if they present the sole method of obtaining necessary data.

TAAG suggests that a measurement program consider:

(1) Additional evidence (photographs of the flange end of the detector assemblies, chemical samples of the white encrustations, direct leakage measurement) of leakage through the calibration tube. Leakage at this point shows definite rupture of the two fluid barriers of the detectors and confirms the radial damage pattern of the core.

(2) Dose rate measurement at the in-core service area of the detector assemblies to identify leakage through the assembly sheaths into the interspace between sheath seals at the flange end. Calculations indicate a dose rate of about 4 R/hr gamma from CS^{137} could be present at the top of the flange closure and this could be differentiated from flange to flange. These dose rates should be obtained before reactor coolant system drain-down and would also be useful in the radiation control engineering of the incore service area and for handling the detector assemblies during disassembly and defueling.

- (3) Mechanical probing through the calibration tube to detect obstructions that would indicate the level of damage throughout the core. This should be done after the water level is lowered in the reactor coolant system to preclude recontamination of the incore service area by leakage through calibration tubes.
- (4) Gamma probing through the calibration tube to help indicate quantity and location of fuel in the bottom of the reactor pressure vessel. Babcock and Wilcox suggested a thermal luminescent dosimeter for this measurement. Determination to conduct this measurement may depend upon the feasibility of other fuel locating surveys (see Section V-F).

TAAG expects that a measurement program including the above considerations can supply useful information that will lead to a focussing of the Defueling Plan on a realistic core damage estimate.

3. Quick Look into TMI-2

a. Introduction

If a leadscrew were removed through the top closure of a CRDM, a 1-1/2 inch diameter access hole 36 feet long is provided from the top of the mechanism to the upper end fitting on the fuel assembly. With one such access hole up to nine fuel positions and fittings can be viewed to assess core damage and damage to the lower portion of the upper plenum assembly.

b. Predicted Conditions at the Upper Plenum Assembly/Fuel Assembly Interface

There have been numerous studies by a number of different organizations on the predicted core damage resulting from the TMI-2 accident, e.g., NSAC, EG&G Idaho, NRC, ORNL, EPRI, etc. A document which summarizes many of their studies is GEND-007. Most of these studies indicate there may have been significant damage (e.g., melting and/or distortion) at the interface between upper plenum assembly and the fuel assembly. The range of predicted temperatures and the melting temperatures of the various components at upper plenum/fuel assembly are shown in Figure II-6. If these calculated temperatures are anywhere near correct there will be significant damage/distortion to the interface components.

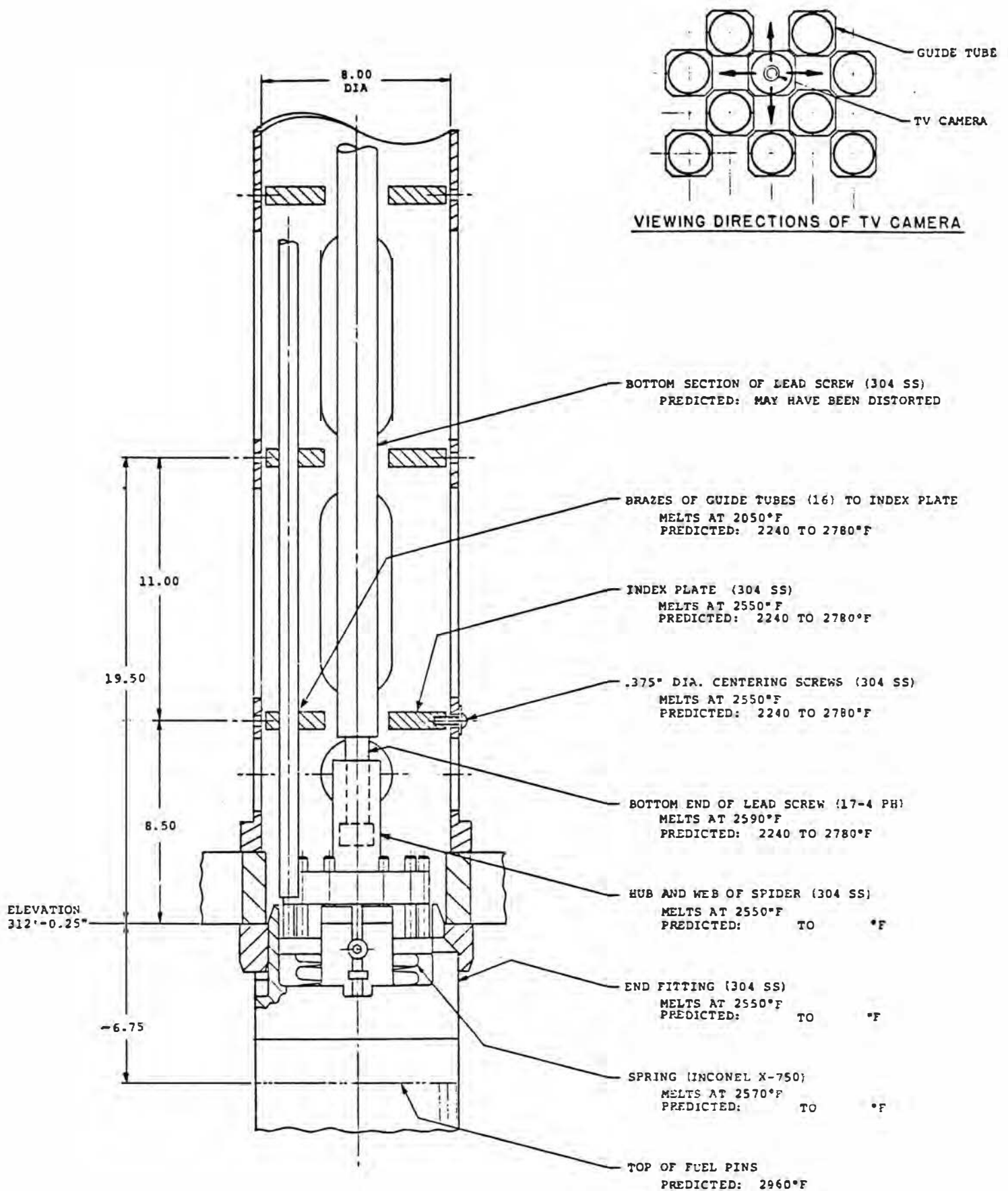


FIGURE II-6 PREDICTED AREAS OF DAMAGE AT TMI #2
(BASED ON GEND 007, PAGES 14 & 15)

Even if just the center of fuel assembly reached embrittling temperatures, some of the fuel assembly end-fittings may be out of the lower grid plate of the upper plenum assembly (e.g., the fuel hold-down spring and the Burnable Poison Rod Assembly retainer spring apply a down load on the fuel assembly end-fitting ranging from 583 to 951 pounds hot and cold, respectively).

- c. Areas of the Upper Plenum Assembly/Fuel Assembly Interface That Can Be Seen in the "Quick Look" to Assess As-Predicted Core Damage and Its Impact on Defueling

Drawing #SK1074-01-501* shows how a 1-1/4 inch TV camera can be manipulated down in a brazement assembly to inspect the interface between a fuel assembly end-fitting and the upper plenum assembly in up to nine different core positions through one leadscrew hole.

Some of the specific areas and information that can be obtained at the interface between fuel and upper plenum assembly by the leadscrew hole TV inspection are as follows:

- * This is a large drawing and will be provided separately by MPR Associates.

- (1) A check can be made of relative height between the top of the end-fitting centering tabs on the fuel assembly and the grid plate of the upper plenum assembly. Also, the condition of the centering tabs can be checked for fusing, melting, and distortion which would make it very difficult to remove the upper plenum assembly.

NOTE: Normally, a fuel assembly has less engagement with the grid plate when the plant is hot than cold (e.g., about 3/4 inch less), therefore the relative position of parts is an indication of core condition.

- (2) It would be possible to see debris laying on the flat surfaces of the index plates, arms of the spider, top of the grid plate in the end-fitting of the fuel assembly, etc. Such debris could then also be assumed to be elsewhere in the reactor coolant system.
- (3) When the "straight-on" TV camera is tilted to look out the 3 by 8 inches flow ports (see Figure 1 and Drawing SK-1074-01-501), one should be able to see the tops of four directly adjacent fuel assemblies which have burnable poison assemblies. The TV

inspection should also permit the checking for conditions at the point of engagement between the centering tabs on these four fuel assemblies and the upper plenum assembly.

- (4) When the TV camera is tilted further outward, one should also be able to see the upper end of the control rods, the lead screws, guide finger tubes, and index plates of four additional fuel assemblies. The centerline of these four assemblies will be about 16 to 17 inches away from the camera, which is a satisfactory distance for general viewing.
- (5) The TV camera should provide a good view of the guide tubes in the brazement to see if they have been distorted, their brazes melted, etc., and therefore, an indication of the temperature experienced by the core and the damage incurred.
- (6) The camera should be able to provide a view of the top ends of the fuel rods in each of the units inspected - providing some insight into the condition of the fuel rods.

Accordingly, from this TV inspection, it will be possible to obtain a good assessment and determination as to whether the predicted core/upper plenum assembly temperature predicted in the various studies covered by GEND-007 are anywhere near correct, or if they are grossly too high. This information should provide some factual hard information as to conditions of the core, and an assessment of the degree of difficulty of defueling the core. It should be possible to obtain a good assessment of how difficult removal of the upper plenum assembly will be. Further, the removed leadscrew will provide excellent material specimens to use in determining the degree of contamination and the temperatures reached in the region of the upper plenum and reactor internals.

It is not proposed at this time, but recent developments in radiation resistant fiber optics reveal that 1/4-1/2 inch borescopes attached to small, high performance colorvideo cameras also might be used if desired. State-of-the-art image enhancement of any videotapes is also feasible.

d. Preliminary Tests to Demonstrate Feasibility of
TV-Leadscrew Hole Inspection Technique

Tests were conducted at Lynchburg Research Center with a mockup brazement assembly to demonstrate the feasibility of manipulating a TV camera via a leadscrew hole so as to inspect up to nine core positions. This demonstration was very successful and a video tape was made to show the capability of the camera.

A test was also made at Diamond Power Division of B&W to confirm that the TV camera could pass through a CRDM unit. Physically, the camera passed through the mechanism satisfactorily. When the camera passes by the magnet in the torque-taker, the TV camera must be handled in a certain way to prevent the magnet in the CRDM torque-taker from affecting the TV camera. Using this handling technique, the TV camera made a successful inspection after it passed through the torque-taker.

Preparations are also being made to try the TV camera insertion in TMI-1 in the event that defueling of that reactor is undertaken.

e. General Procedure for Making a Leadscrew Hole TV Inspection

The following is a general procedure for making this type of TV inspection once a leadscrew is removed from a CRDM. (NOTE: For methods of removing a leadscrew from a CRDM, see Section IV.a)

The basic approach for this inspection will be to lower the TV camera from the top of the CRDM via the leadscrew hole down to the 10th index plate of the brazement in the upper plenum assembly without initially performing any inspections. The inspection will be started at that 10th index plate and then followed by an inspection of the fuel assembly that is directly below the removed leadscrew hole. In making this inspection, an effort should be made to try to go into all four of the lobes of the 10th index plate. After completing the inspection of the fuel assembly directly below the leadscrew hole, the TV camera will be raised and then tilted out each of four flow holes in the side of the tube enclosing the brazement to inspect the four adjacent fuel assemblies. Following the inspection of each adjacent fuel assembly, the camera will then be tilted further upward to inspect the next outward fuel assembly position. Using this approach, up to nine fuel assembly positions can be inspected via one leadscrew hole.

Following the inspection of the fuel end-fitting, the camera will be retracted into the leadscrew hole to inspect "C" guide tubes and "split" guide tubes that guide the individual control rod pins of a control rod assembly. Also, index plates #9 and #8, particularly the brazed areas, the surface of the index plates, and the 3/8-inch cap screws which center the index plates will be inspected.

f. Recommendations

TAAG recommends that GPU Nuclear proceed to demonstrate in TMI-1 the capability of performing a detailed TV inspection of the fuel/upper plenum assembly interface through the access hole provided by removal of a leadscrew. Upon the satisfactory completion of TMI-1 demonstration, TAAG recommends that GPUN proceed at all reasonable pace to perform such an inspection in TMI-2. Such an inspection would provide hard, first hand information on the condition of TMI-2 reactor internals.

C. Other Sources of Core Damage Information

Additional insight into core damage and, more generally, into problems related to fuel removal can be deduced from improvements or refinements of other information. Two of these are considered briefly here, namely improved analysis and characterization of selected samples and updated thermal-hydraulic calculations.

Radiochemistry and Material Characterization. Part of the uncertainty in estimates of core damage results from conflicting analytical data on fission products. Typical examples concern the release of Kr^{85} from the fuel, for which the estimate based on analysis of the Reactor Building atmosphere prior to containment venting was greater than the quantity found during the venting operation itself, or the release of Cs^{137} for which estimates of

the fraction released from the fuel vary from 37% to more than 60%, depending on the particular set of analyses used. Another example involves the numerous discrepancies between analyses of the same sample or very similar samples at different laboratories. This includes the very important first two samples of reactor coolant on March 29 and April 10, 1979.

In some cases there are good reasons for analytical data to be less precise than it could be. Many analyses are done for process control purposes or as proof of meeting technical specifications (water clean-up), and analytical accuracy suitable for these purposes is not adequate, for example, for accurate determination of fission product release from the fuel.

Similarly, routine analyses of the reactor coolant system water are marginally adequate, at best, for determining leak rates of fission products from the fuel; and recent data done on-site shows significantly more scatter than earlier analyses by Babcock and Wilcox. Problems arise when such analytical data are used for purposes other than the intended ones (such as fuel damage estimates or leak rates) because they can yield misleading or erroneous conclusions. It would be of great value in the longer term if important samples were identified and selected for more accurate analysis.

There also are instances when analytical techniques that could give very important information have not been used and the information has not been acquired. An example is the solids recovered from letdown filter MU-F-5B which contains a substantial amount of uranium; this is one certain location of fuel transported outside the reactor vessel. Microprobe and corresponding scanning electron microscopy would yield information about the nature of individual particles of fuel and other constituents (size, shape, possible source) that is important to estimating the extent of fuel transport out of the reactor core region. It is important that critical materials (such as this) be selected for characterization by the best methods currently available anywhere. TAAG urges that such characterization be made on the sample from MU-F-5B to assist in evaluating fuel transport into the reactor coolant system.

Thermal Hydraulic Calculations.

The TMI accident dramatized certain inadequacies in a number of models, computer codes, and even regulatory guides relating to reactor accidents. A number of thermal hydraulic studies of the TMI accident were made during 1979, and fuel damage estimates were based on these to a significant extent. In large measure because of this experience, there has been a substantial effort to upgrade this technology and to make the computer models more realistic. Since this work has now led to major improvements, it appears that reevaluation of the TMI accident using the improved computer codes would have significant value.

A comprehensive assessment of the state of this work is beyond the scope of this study, but the following developments were identified. A group at the Los Alamos National Laboratory under R. Smith is presently undertaking a reassessment of the TMI accident using the estimates from NSAC-24 for boundary conditions; the progress of this work should be followed. A group at Sandia has made substantial improvements in the MARCH computer code, but application to the TMI accident, specifically, is not planned; it is suggested that such a study would be of sufficient value that it should be undertaken. Further utilization of these methods should be pursued by the appropriate sponsoring agencies.

III. COMMENTS ON BECHTEL NATIONAL'S TECHNICAL PLAN FOR REACTOR DISASSEMBLY
AND DEFUELING REVISION 2, DATED JANUARY 1982

The TAAG considers that its most important statement about the technical plan is not about the plan itself but about the perceived intent of how the plan will be implemented. The intent appears to be that prior to commencing the first step of the plan all equipment, procedures and personnel must be ready to complete the plan through to the last step. Based on the experience of various members of TAAG, this is a commendable objective, but possibly not realistic based on the complexity of the total TMI-2 defueling job. TAAG suggests that the plan be reviewed for implementation in various increments, with the key thought of being able to stop due to technical or monetary constraints (i.e. one increment could be from start to head removal, with provision for a reactor vessel cover, and thus have a viable stop point for an indefinite period of time).

Additional benefits which would be derived from this approach include a) the ability to focus attention on the early phases of work, accomplish that work and if need be for any reason, stop until all preparations are ready for the next increment and then proceed and; b) due to the broad spectrum of possible reactor damage, data may be obtained which would allow efficient planning of tooling procedures, and qualification of manpower needs for follow-on increments of work.

A. General

1. As a result of presentations made to TAAG by the various groups involved in developing both tooling specifications and actual tooling, the following comments are made:

a. Mock-ups often do not duplicate "as-built" conditions. This can lead to improper assessment of tool functioning. Additionally, there seemed to be instances where tools are being designed with no apparent intent to test under postulated conditions - e.g. the device to shear the stuck control rod drive mechanism bolts.

b. ALARA considerations do not appear to be taken into consideration during tooling conceptual design and fabrication. Too much is being left to the field operations people to solve when in fact the problems could be more efficiently addressed early in design. Examples of this include:

(1) The through-head TV system which has significant radiological problems due to its ability to extract high levels of activity out of the reactor vessel because of many radioactivity collection points and uncontained cable which could highly contaminate the work area and

(2) The sample retrieval tool which has no design provisions for high activity samples.

2. It is difficult to assess the major concerns or risks (i.e. criticality, potential high radiation, exposure, etc.) for any particular operation. This seems important since it (the Plan) serves as the guide for all further work on a particular task and those individuals working that task should be aware of such risks. A paragraph titled "Precautions" is recommended so that significant issues such as criticality, unusual radiological issues, and other unusual risks can be identified in each step.
3. In the early part of the plan, several steps contain a number of activities that apparently could be accomplished from outside the reactor building (e.g., electrically checking the Axial Power Shaping Rods). Radiation doses would be reduced and any information developed would be available at an earlier date for planning of the defueling. Although this same activity may be conducted early from outside the reactor building as part of a "quick look", it also accomplishes a portion of the technical plan.
4. The technical plan as presented has all operations shown (planned and contingencies). As an aid to ease of understanding and concentration of effort, the TAAG suggests that a single, optimum sequence be shown such that management

and technical planning personnel can focus their attention on the main issues of starting and maintaining the sequence. A contingency can be shown in block form and referenced to a separate contingency sequence.

5. There are various steps in the plan which identify the potential for stuck or corroded bolts and other malfunctioning mechanical equipment. Often, the ultimate solution is to provide a tool to fracture the part in question. It is strongly recommended that, where possible, a penetrating lubricant be applied well ahead of the planned evaluation to minimize the amount of destruction removal necessary. This has proven to be a good technique in the area of ALARA.
6. It is suggested that the technical plan have certain identified key events, e.g., start of job, head removal, fuel removal, etc. These key events would be required to have formal prerequisite lists, which list equipment, procedures, deficiencies, qualifications, etc. for that event and future events. A list then becomes one of the management tools to determine if the specific event and future events can proceed as planned.
7. Included in the Plan should be a section which addresses the technical management (e.g. who can stop and start work, who makes final technical decisions, where the job will be supervised from, TV and audio communications into plant,

makeup of refueling crews, etc.) of the activities associated with the defueling operation, both for work inside and outside the containment as well as planning activities, and decision-making hierarchy. This entire defueling effort is an extremely complex management task and communications and decision making processes must be clearly defined and understood by all.

8. As a result of the safety analysis discussed in Section V.D, additional sections may have to be added to the plan (e.g., operation resequencing, new equipment specified and procedures developed) to assure that inadvertent criticality does not occur. Issues which need to be addressed include:
 - a. Secondary system level and pressure control sequenced into primary plant operations.
 - b. Because a high boron concentration is the primary shutdown mode, equipment and procedures for specifically monitoring for boron levels must be implemented.
 - c. Exclusion of nonborated water is an absolute necessity to assure that dilution does not occur. After review, this may affect secondary system work sequencing (e.g. drain down, refill, test, etc.)

B. Detailed Comments

We suggest that the following detailed comments be considered in make up of future and final technical plans and sequence of performing work.

<u>Section</u>	<u>Comment</u>
A.3	As discussed at length in the TAAG, a simpler "quick look" earlier than this step may yield information which would provide data necessary to establish core condition and tooling needs.
B.1 2.1.2	Additional discussion should be added relative to the containment recovery services building. As indicated, this facility is "if required" and it should be an absolute prerequisite. Provisions must be made in this facility to handle decontamination and repair of malfunctioning tooling. This equipment will be largely untried in a production mode, meeting many unknown conditions - and there will inevitably be problems with it. Without an adequate facility to repair it, much time could be lost.

8.1 2.1.9 Add an item which discusses the storage of radwaste both inside and outside the containment. There will be large quantities of radwaste which will have to be handled, stored in a manner which won't cause additional exposure to workers, and dispositioned.

8.1 2.1.4.b Specific goals for decontamination of the work area should be established. One goal should be based upon eliminating air breathing equipment to the maximum extent, which will improve productivity by a factor of 2 or 3. If this conflicts with the rest of the building decontamination, a technique to enclose the work area and provide it with breathable air should be considered. This is a cost consideration, but it could also significantly affect exposure, because of the increased productivity.

Shielded wait areas should be established so that people not directly involved in a work activity may wait in a lower radiation area. This could easily reduce exposure by 10%.

It is recognized that there will be instances where breathing apparatus will be required due to specific operations or due to the inability to adequately decontaminate. In these cases, communications equipment must be provided (Ref B.1, 2.1.1.d) to allow clear communication between workers.

B.1 2.1.1.i The HVAC system, or portable equipment, should be capable of local ventilation of work areas to control airborne radioactivity and, where required, be provided with spark arrestors to handle possible welding and burning operations.

B.3 The venting operation potentially involves a significant radiological hazard and installation of a radiation monitoring device at each vent point should be considered so that operations can be stopped if radioactivity levels increase above a predetermined point.

Consideration also might be given to sampling this gas prior to the actual venting to determine its chemical and radiological makeup. Such analysis could result in changes of venting rates, dilution of hydrogen, etc. Criteria should be established

for acceptable levels of gas after venting, purging (e.g., rates, allowable periods for securing purge based upon expected or measured gas generation rates) and sampling. The need for venting all control rod drive mechanisms is questioned,

The sequence of venting the mechanisms before lowering the level in the reactor vessel should be reconsidered. If the level is lowered first, this will minimize or eliminate possible contamination of the vent hoses when the mechanisms are vented.

B.4 Another means of drain down should be looked at based on the restricted flow of the letdown system (3 GPM approximate). Decay Heat Removal System could be an alternate to allow a quicker means of obtaining the desired level.

B.5 The method of water level control described in Section V.B of this report appears preferable to the techniques described in the technical plan, particularly from an exposure standpoint. Utilization of this suggested technique removes this step from the critical path.

B.7 See Section V.E of this report relative to the ionic iron problem. Use of a nitrogen blanket would appear preferable until such time that introduction of air to the system no longer causes a water clarity problem.

B.10 A consideration that does not appear anywhere in the references is leakage from the reactor vessel during the defueling. A likely location for leaks to occur would be through the 52 external guide tubes of the incore instrumentation. These tubes are 304L stainless steel, 1/2" Schedule 80, each are about 130 ft. long, about 60 or 70 ft. of which have been submerged since the start of the accident.

The end of the tube near the reactor vessel would have been hot or warm in its natural operating condition. The gradient from reactor vessel temperature down to ambient would have existed along the length of the external guide tubes. The guide tubes have numerous crevices at the supports and seal members. The tube passes through a seal where it goes from the RPV cavity into the incore pipe chase. The seal is part of the structural arrangement that may have precluded mixing of the water throughout the flooded reactor building.

Thus the chemistry of water at the tubes throughout the nearly 3 years since the accident may not be as shown by the samples. The steel also has welds which could be preferential sites for cracking or pitting due to the residual strains that would occur at those welds.

It would be prudent to analyze the effect of possible leakage from some number of these guide tubes and the capability to maintain boron concentration if leaks occurred.

Further to the possibility of leakage at the external guide tubes, Steps B10 and B11 cover pulling the incore detector assemblies out of the core. Consideration should be given to the possibility that a stuck in-core tube could impose significant forces on the external guide tubes and pull apart any tube that was in bad condition. As long as the in-core detectors remain in the external guide tubes, they would provide a partial plug and restrict any leakage that occurs through the external guide tubes. Since the location of the actual ends of the detector assemblies is unknown, it may be prudent not to attempt any withdrawal until the core is out.

Steps B11 and B10 deal with the partial withdrawal of in-core instruments. Included in these steps is the possibility that the old detector assembly will be cut and a cable attached to it. No comment or consideration is given to the likelihood that many of these detector assemblies are wetted up into the calibration tube and end plug at the 347' level. Such wetting could have occurred to any in-core detector that was melted-off in the core. Radcon procedures for this eventuality would be complex and should be identified in the plan to be pre-engineered if this step is retained.

B.12 Consideration should be given to installing the reactor vessel exhaust system on a control rod drive mechanism due to a much lower radiation level.

B.13 The auxiliary hoist of the polar crane should be given as much emphasis as the main hoist. It is anticipated that the auxiliary hoist will be used extensively throughout defueling.

B.17 It seems that the bridge travel mechanism should be refurbished also. An evaluation should be made to determine the cost and time to refurbish/modify the existing bridges versus supply new lightweight bridges exclusively for this defueling.

- C.14 2.3.1 Add ventilation equipment capable of handling welding and burning work as well as radioactivity in order to minimize airborne radioactivity.
- C.15 Clamping and removal of cut lead screws with subsequent removal from the plenum may be unnecessary. If the cut pieces were allowed to fall into the "rubble bed", exposure could be saved from two standpoints: (1) clamps would not have to be installed and (2) leadscrews would be removed from the reactor vessel when radiation levels should be lower (e.g., when the canal is filled).
- C.16 Plasma arc cutting is historically a "messy" technique, particularly in a radiological environment, for cutting. Precautions must be taken to limit the spread of radioactivity during this operation.
- C.18 Same comment as C.15.
- C.19 Same comment as C.15.
- C.22 Same comment as C.15.

- C.23 Same comment as C.15 and provision should be made for either a shielded container in which to store 1-6 leadscrews or for a shielded storage area.
- C.22 Leadscrew support tube - If you have to cut the leadscrew support tube, consider not clamping it and letting it drop after cutting.
- C.23 Removal of leadscrew - It may not be necessary to remove these leadscrews to lift the plenum. If exposure can be reduced, these leadscrews can be removed after the plenum is removed.
- C.23 2.3.1 Shielded container to store leadscrews should be included.
- C.25 Assuming that leadscrews can be uncoupled normally and they are fairly high level (5-10 R/hr), then support platform radiation levels may increase significantly. Shielding should be available to a) slip down over the mechanism or b) to go on the support platform (Note - no calculations seem to exist which establish acceptable loading).
- D.4 Consideration should be given to installing one long term seal only in lieu of this one and then the permanent one in step E.4.

D.6 Use of pure alcohol is strongly discouraged - history shows a high fire potential because of sparks (static electricity) and its rapid drying has a tendency to create airborne radioactivity problems. A mix of 50% water and 50% alcohol is a good decon agent and does not have the aforementioned problems.

O.16 It appears, based on the description, that the head will be lifted to the support stand in the "normal manner" e.g. - open on the bottom. Due to potentially higher than normal contamination levels, consideration should be given to covering the bottom of the head with a quickly installed boot.

E.3 This step briefly mentions "physical barriers" to minimize contamination spread to be discussed in G.15 (non existent). A concept is provided in Figure III-1 which utilizes a "refueling tank" type barrier coupled with water and air moving equipment designed to catch material at its source and minimize the amount of high activity water which must be processed. This unit would be installed prior to flooding the canal and must accomodate operations such as plenum jacking.

DEFUELING IN-CANAL CONTAINMENT CONCEPT TMI-II

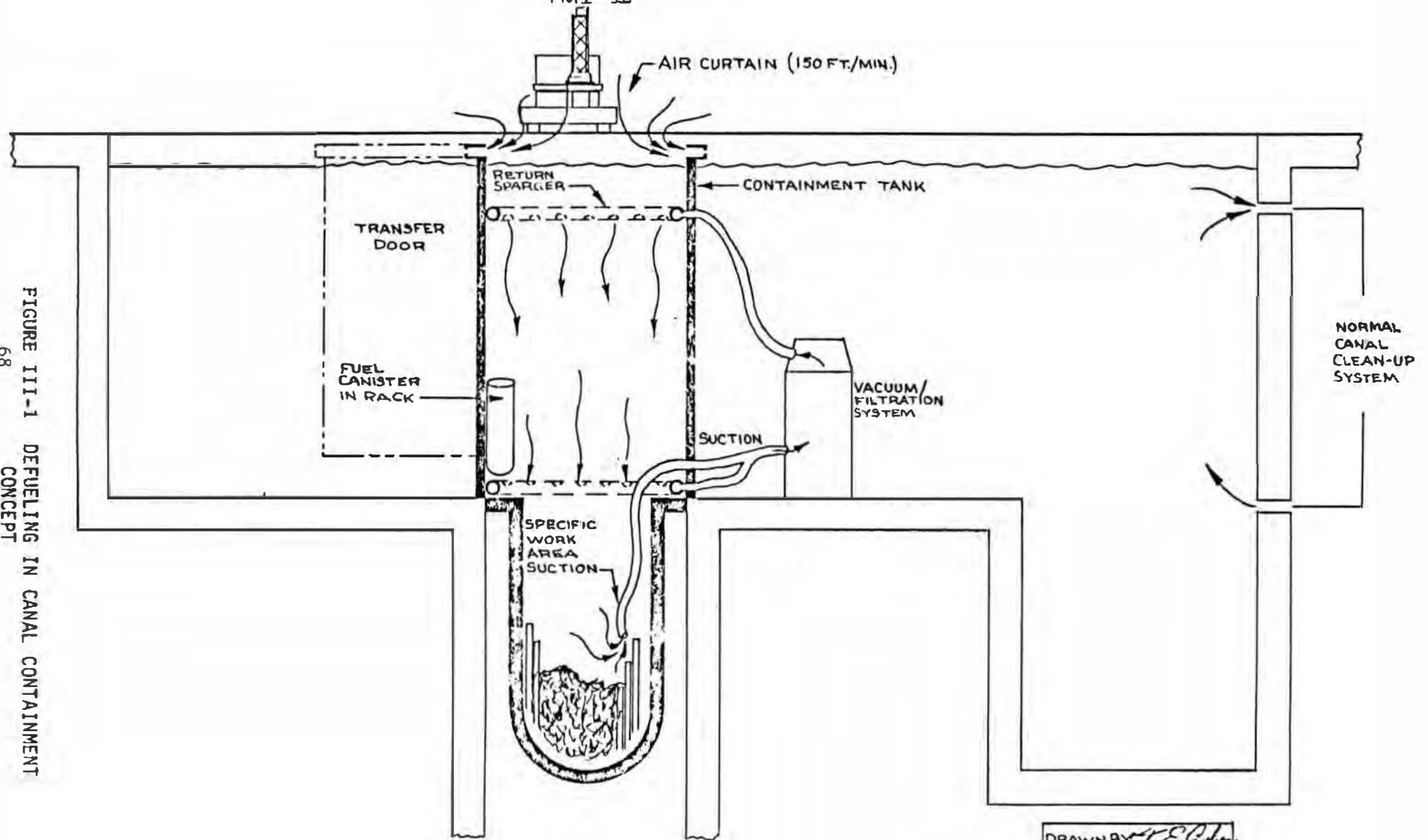


FIGURE III-1 DEFUELING IN CANAL CONTAINMENT
68

DRAWN BY: *CEC*
DATE: *2/1/75*
REVIEWED BY: *T.M.*
DATE: *1/21/76*

- E.4 See comment relative to D.4.
- E.8 See comment relative to physical barrier in E.3.
- F.5 It does not appear necessary to perform the "small initial lift" of the plenum - recommend a higher lift and commence with the video camera viewing.
- F.5 2.1 There should be no option regarding the test of the plenum lifting lugs if they are to be used.
- F.6 Plenum jacking - The criteria for the use of jacking devices and vertical impacting equipment has to be carefully established. The interaction of these devices on the polar crane are of particular concern. Also the criteria needs to be established as to when to go to plenum removal by dismantling the plenums.
- F.10 Consideration should be given to providing a physical barrier around the plenum (underwater) so that subsequent operations such as decon, inspection, etc. do not transfer large quantities of contamination into the canal.

F.11 2.1 Utilizing the approach suggested in E.3, the plenum would be drawn up into the "tank region", hydro vacuumed and the surrounding water cleaned up using the installed system. When the "tank" water and canal water were of comparable activity (e.g. within a factor of 5-10), then the tank door would be opened and the plenum transferred to its storage location.

G.1 Due to interaction between tool design and debris removal canisters, they must be designed concurrently to assure compatability.

G.2 2.2 The fuel would be loaded into cans which are placed within the "tank" and transferred as in F.11 2.1 above.

G.3 The plan includes removal of the water storage tanks from fuel pool A, but appears to leave the Submerged Demineralizer System (SDS) in fuel pool B. The SDS may be useful as the cleanup method for the fuel canal and fuel pool water in addition to the reactor coolant system and reactor building sump. The pumps that pump water from the temporary water storage tanks in the fuel pool A and served as feed to the SDS, could remain in fuel pool A for use in pumping fuel pool A water into the SDS for purification.

G.3 The cleanup system tanks have no provision for internal decontamination and, considering that sump water was processed through only a 40 micron filter, highly active sludge in the bottom of these tanks is probable and a technique for flushing it should be identified.

G.4 Prompt transfer of fuel from the site in order to minimize can storage time seems advisable (ref: resin storage problems). This would mean only minimal storage of fuel in the pool, would require a well sequenced shipping operation and may cause additional shipping costs.

IV LIFTING EQUIPMENT

A. Lead Screw Removal

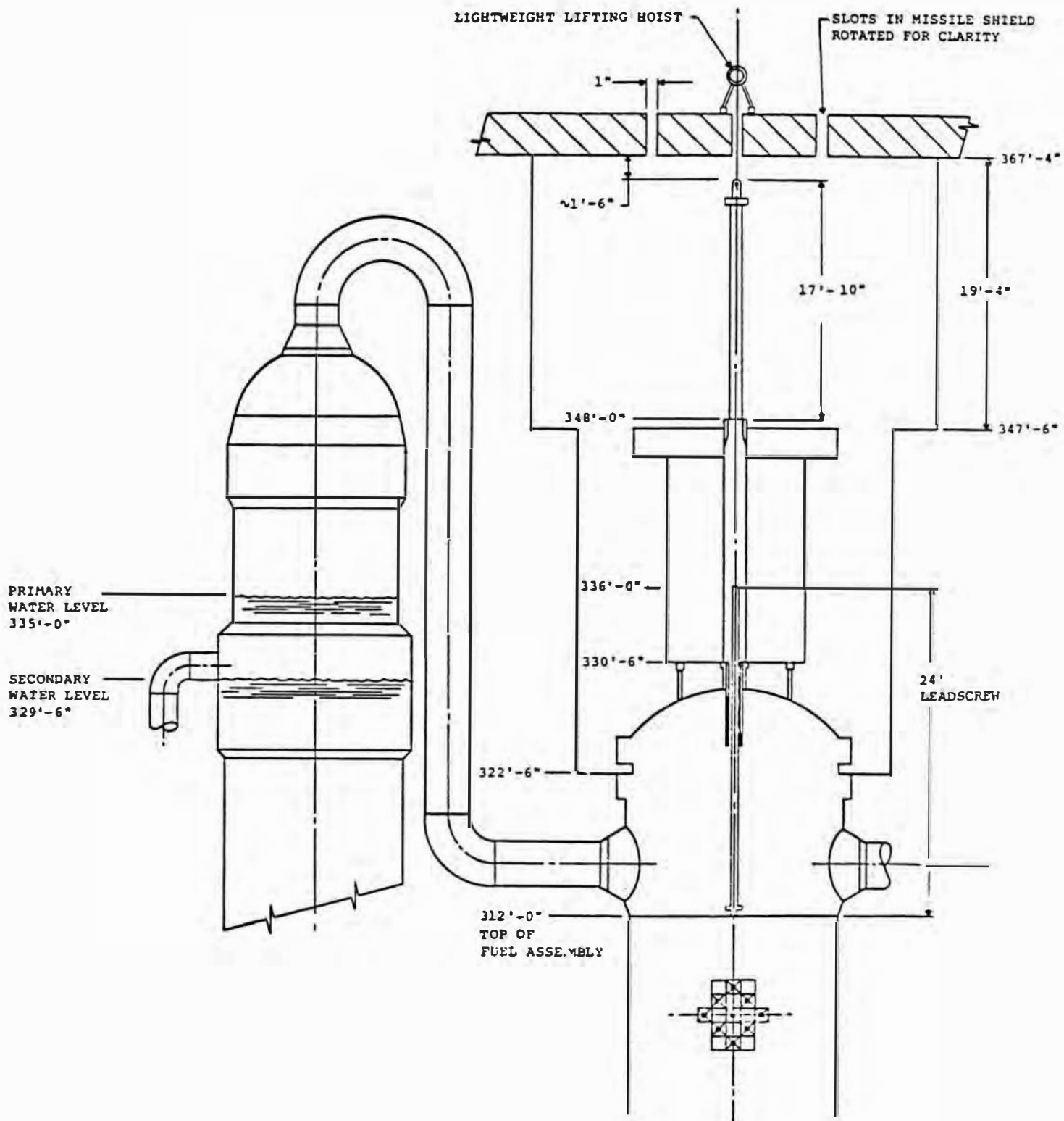
1. Purpose

The purpose of this section is to present the conceptual methods which could be utilized to remove a leadscrew on TMI Unit 2. The removal of the leadscrew would permit an internal inspection ("Quick Look") to be performed through the CRDM leadscrew hole using a 1-1/4" diameter underwater TV camera.

2. Methods for "Quick Look" Inspection

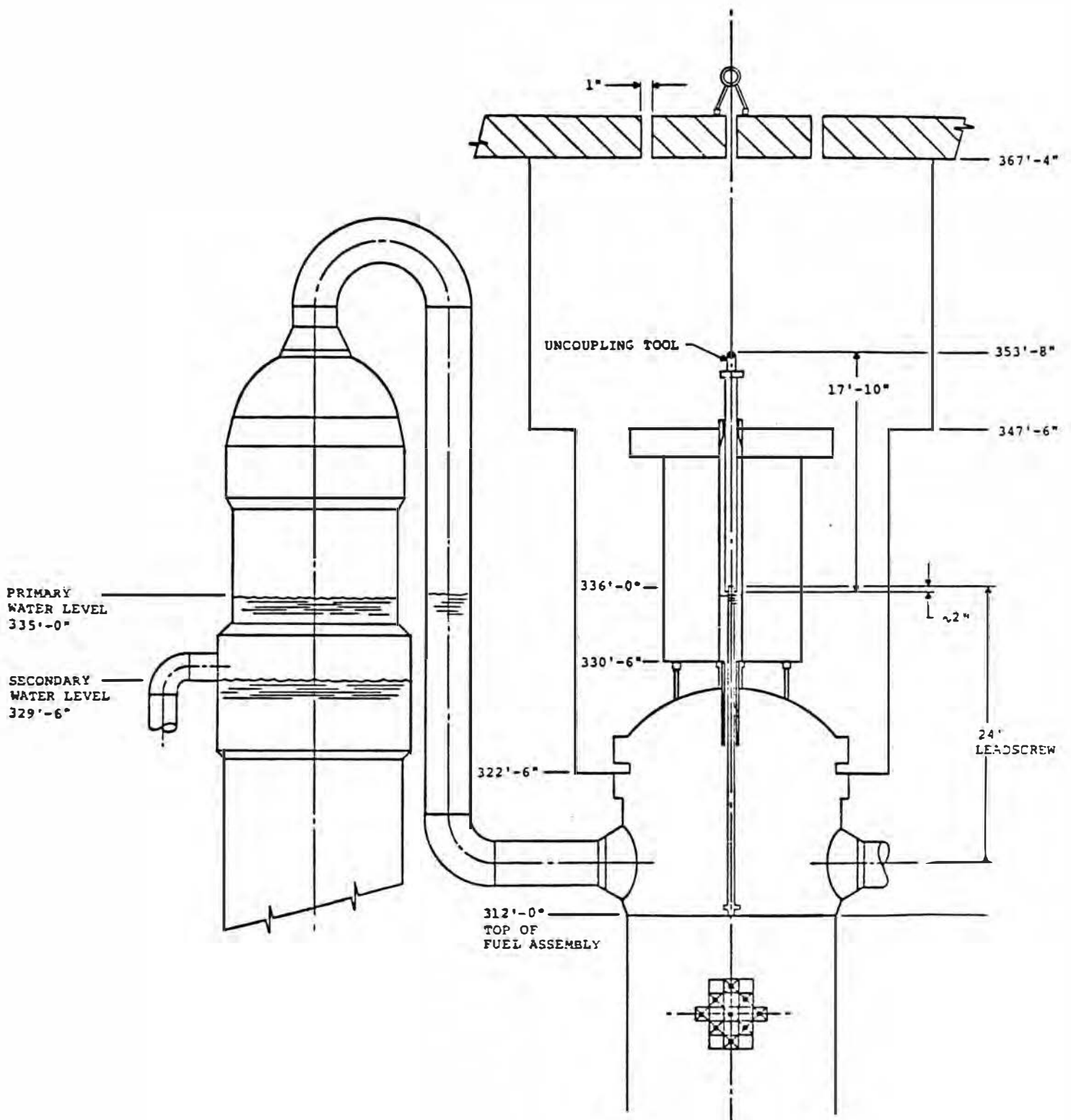
a. Leadscrew Removal with Missile Shield in Place (Concept 1)

Figure IV-1 shows the steps involved with removing a leadscrew with the missile shields in place. In this concept the ~ 1" separation space between the two center missile shields can be used to lower a lightweight hoist cable (~ 500 lb capacity) over the 7 CRDM in the center row of mechanisms (Row #8). The lightweight hoist can be mounted on the top of the missile shield and can be moved back and forth over the ~ 24' length of missile shield slot to permit tools and leadscrews to be handled to storage positions.

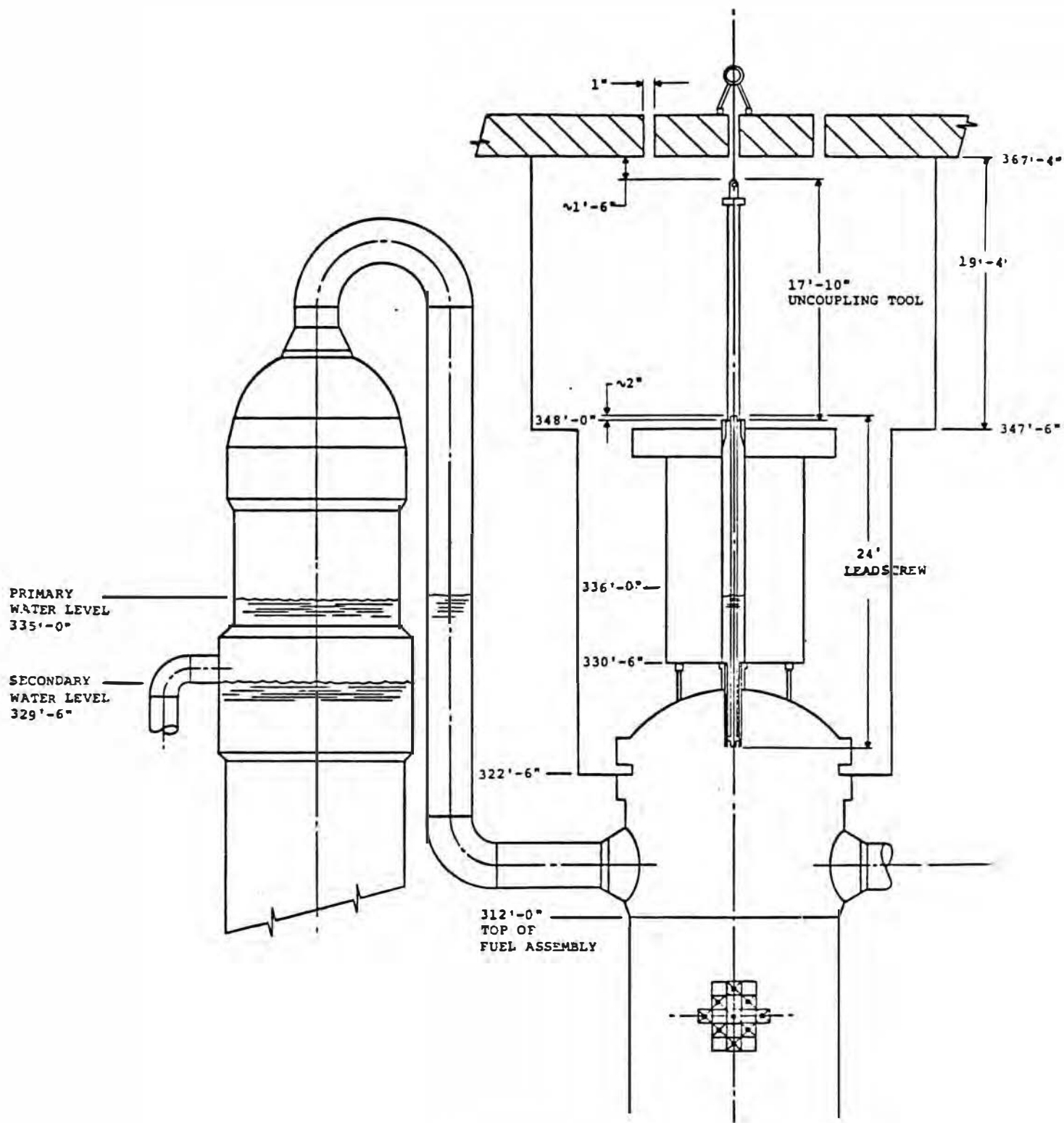


STEP 1

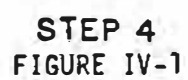
FIGURE IV-1 LEADSCREW REMOVAL WITH MISSILE SHIELD IN PLACE



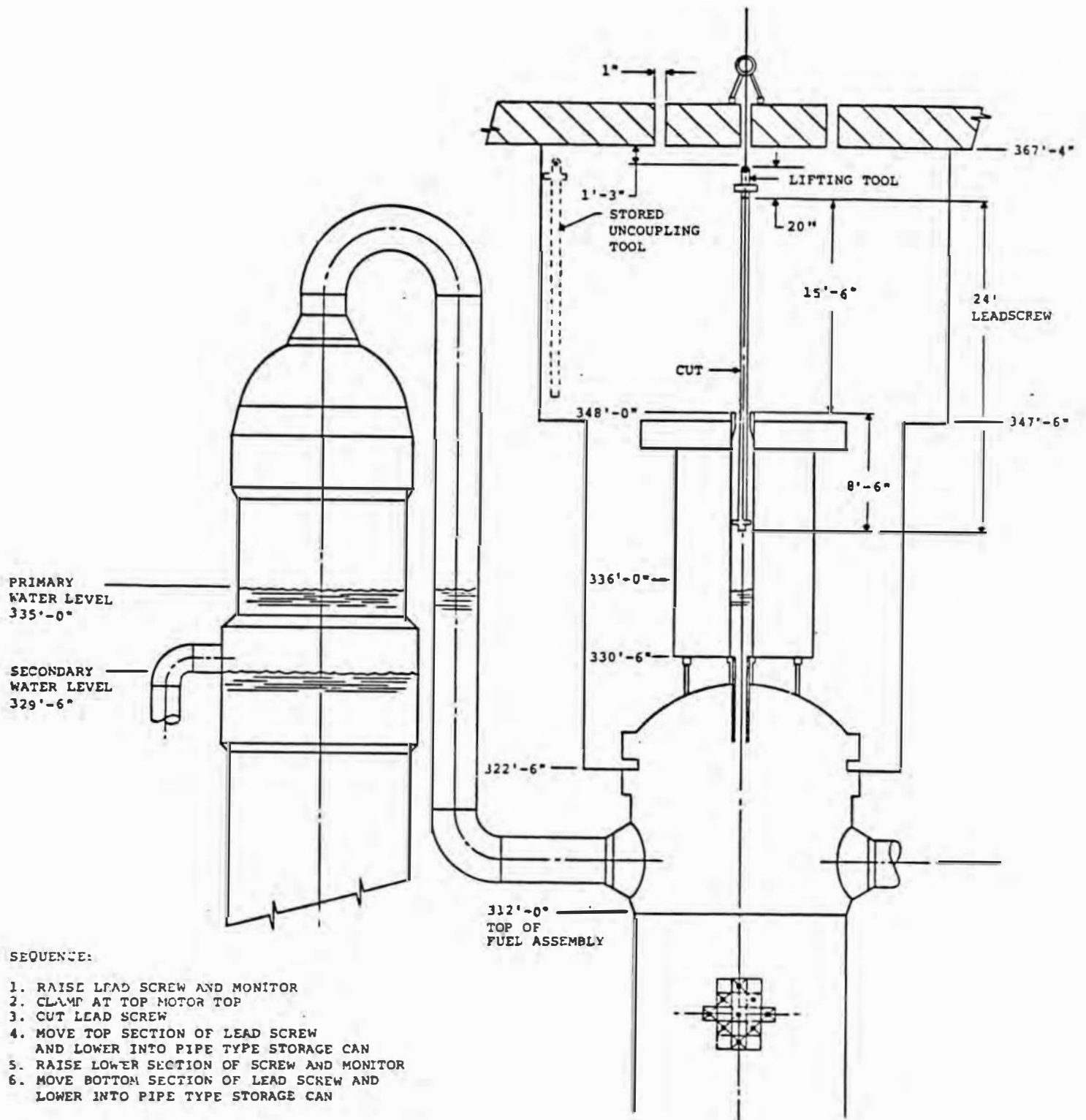
STEP 2
FIGURE IV-1
74



STEP 3
FIGURE IV-1



CONCEPT #1



STEP 5
FIGURE IV-1
77

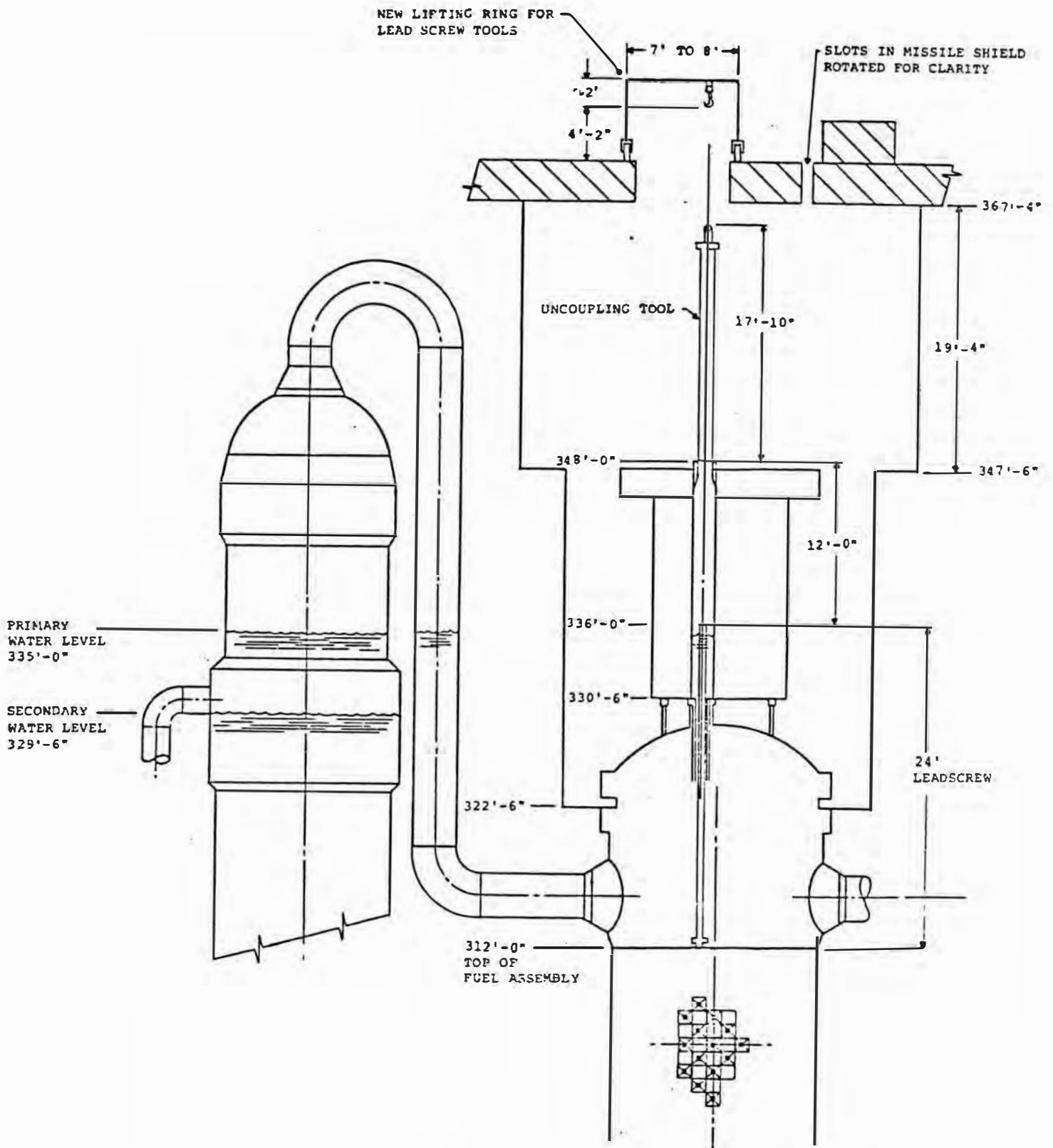
As a variation to this concept which would allow all CRDM to be serviced before the missile shields are removed, a lightweight handling system can be suspended from the underside of the missile shield (see Figure IV-3).

In concept 1 the clearance underneath the missile shields and the length of leadscrew requires that the leadscrew be cut once (see steps 5 & 6) to remove it. The cut sections of the leadscrew would be moved laterally by the hoist on the missile shield to PVC storage cans on the side of the canal wall. If the leadscrews are cut once, the two 12 foot sections will weigh about 75 lbs. If the leadscrew is cut in three places, the four 6 foot sections will weigh about 38 lbs. The number of cuts will depend on the length/weight which will permit easy manual removal of the leadscrew out of containment for detailed evaluation.

b. Leadscrew Removal with One Missile Shield Removed (Concept 2)

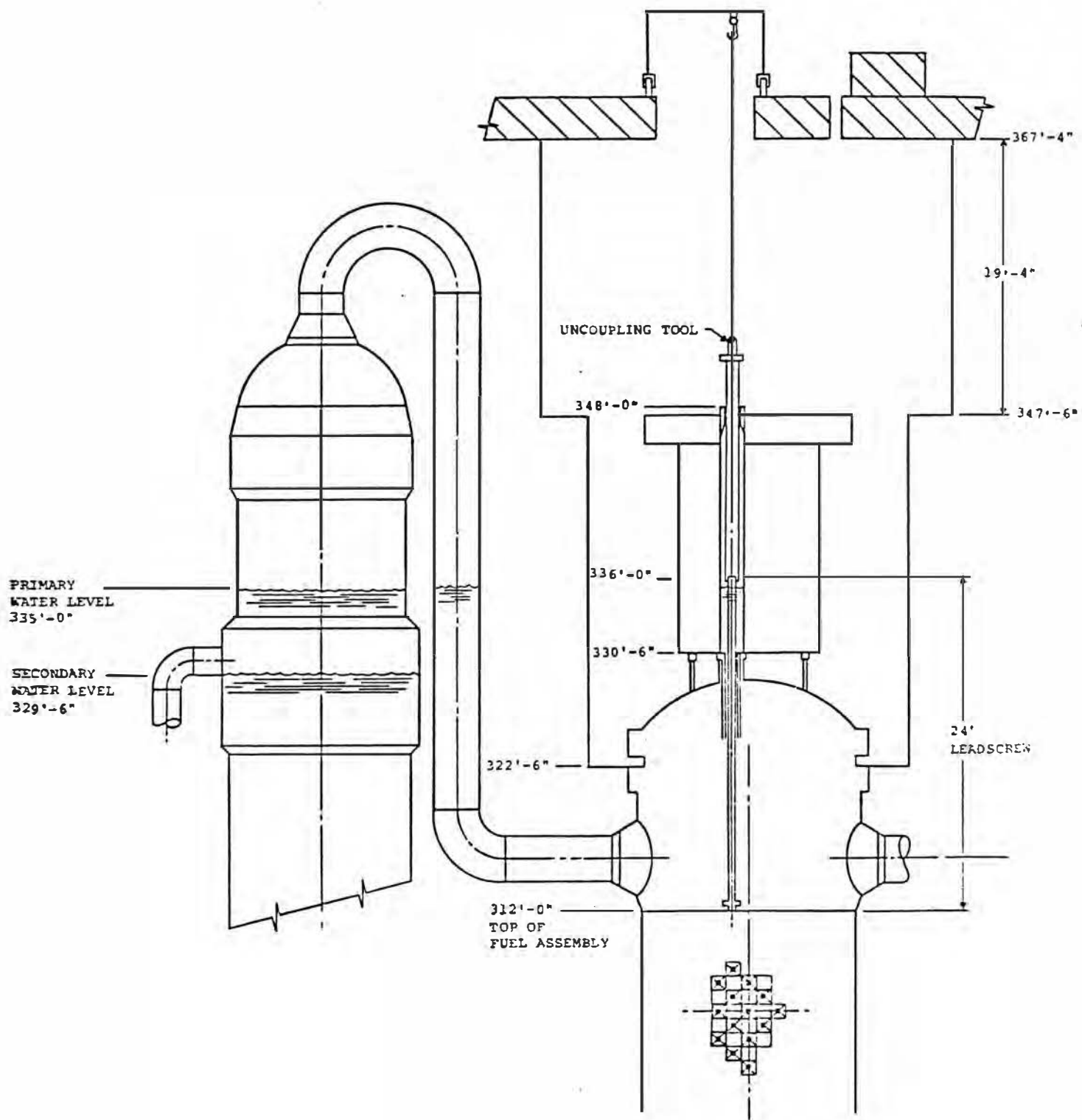
Figure IV-2 depicts the leadscrew removal method with one missile shield removed. This method requires that a gantry crane be installed at the 347'6" elevation over the hatch and a jib crane installed at the 367'4" level on primary shield (see Figure IV-4). These cranes would be utilized to transfer the missile shield lift rig, and hoisting cranes to the top of the missile shield.

CONCEPT #2

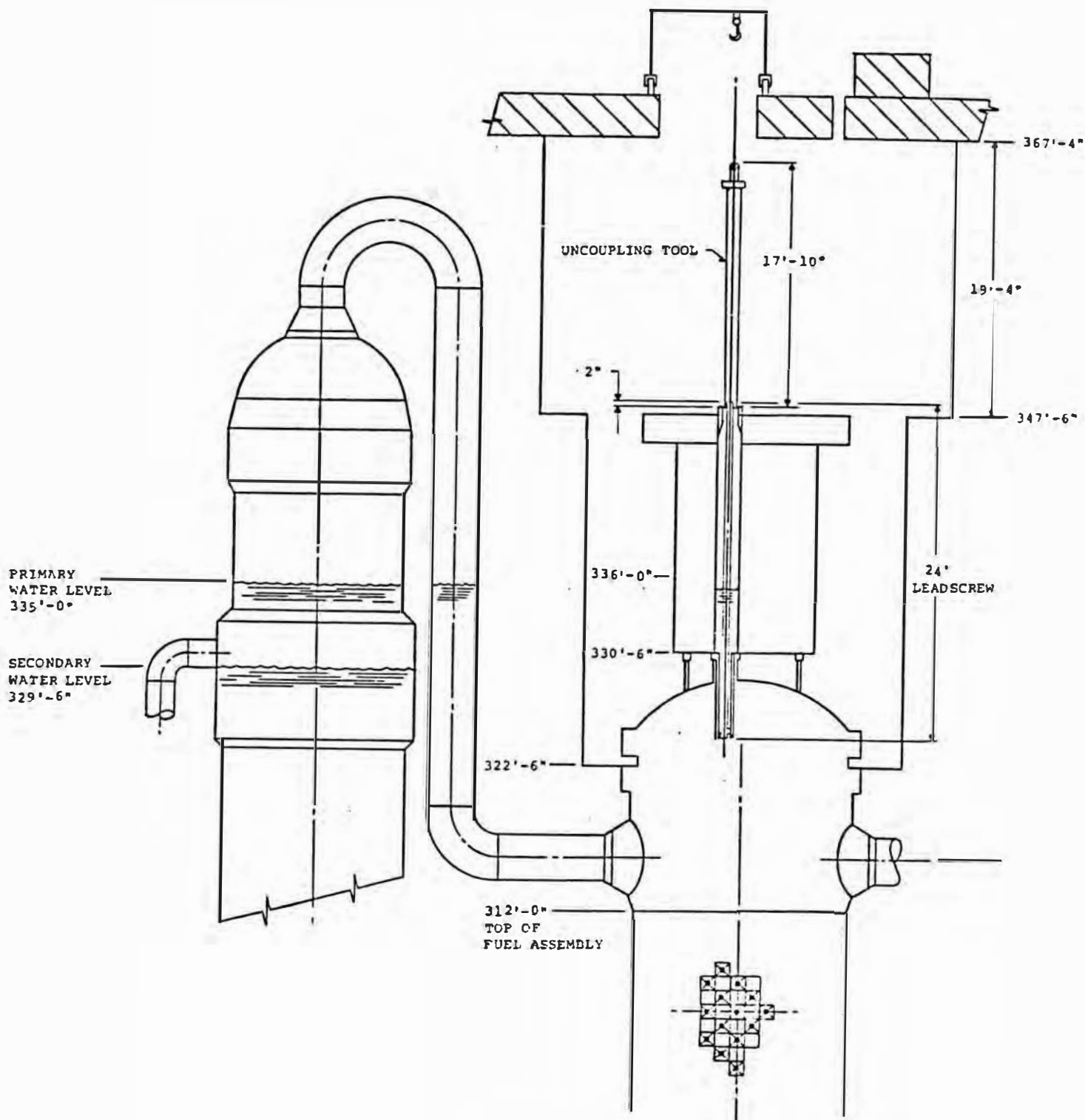


STEP A

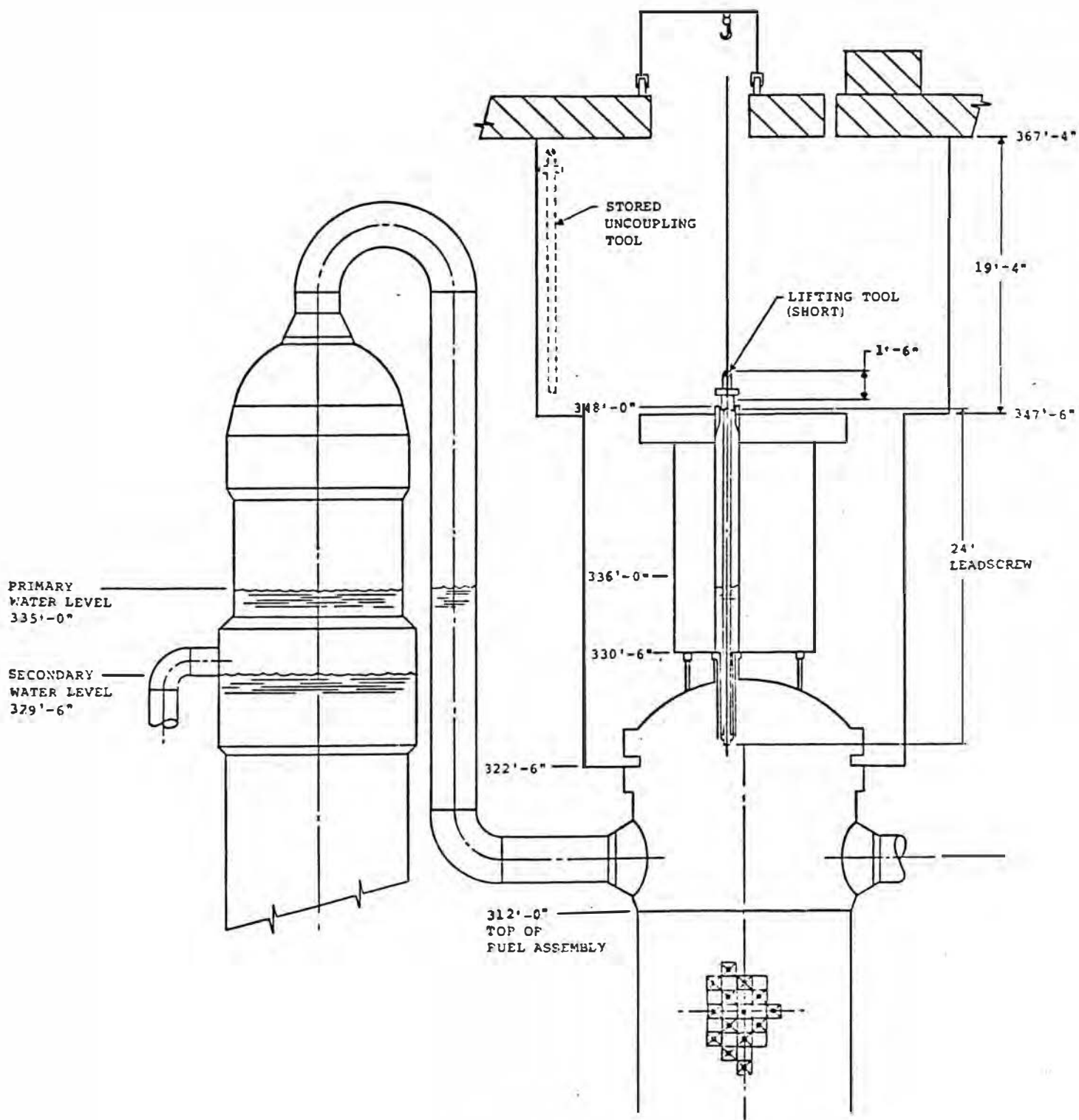
FIGURE IV-2 HEADSCREW REMOVAL WITH MISSILE SHIELD REMOVED



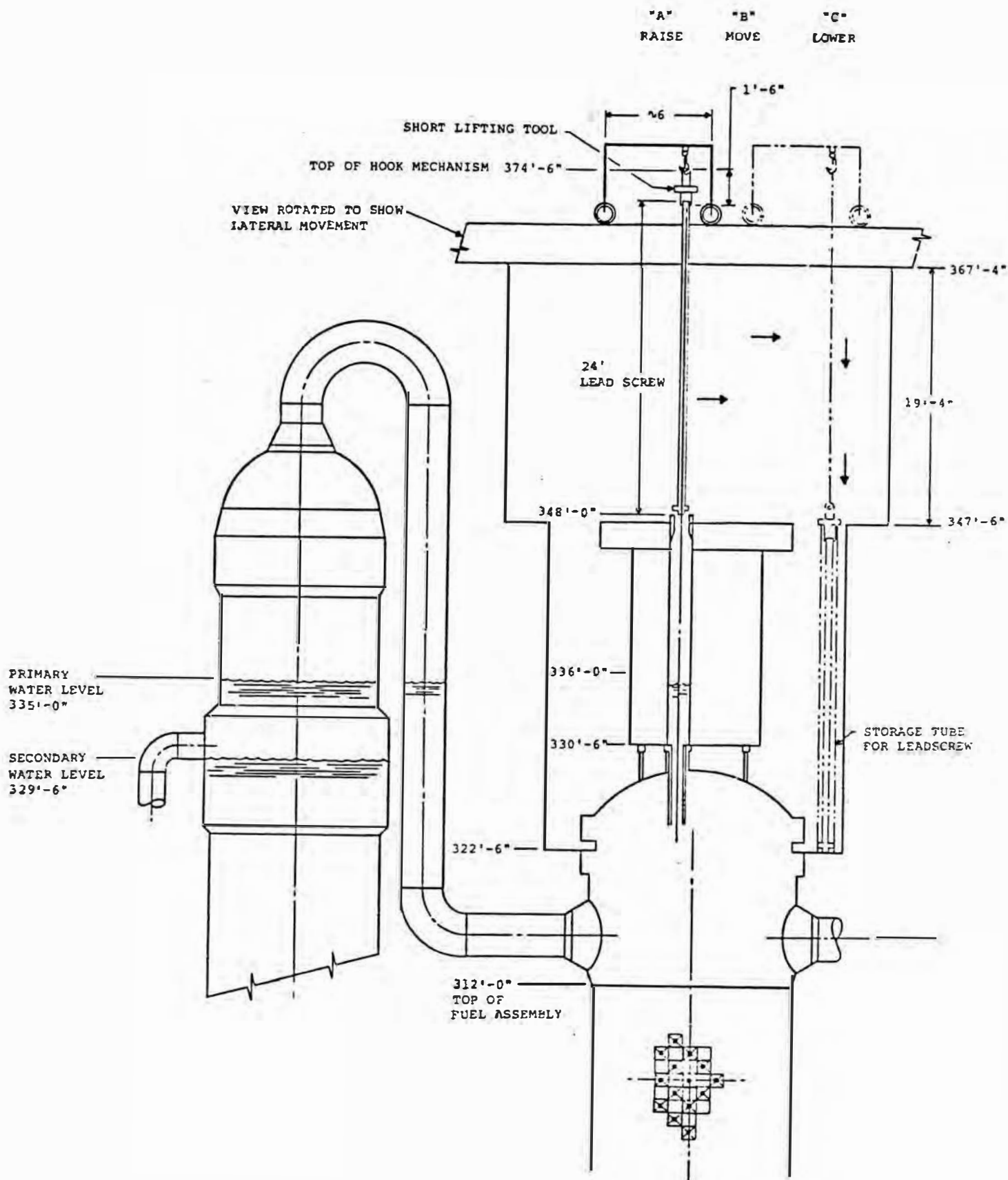
STEP B
FIGURE IV-2
80



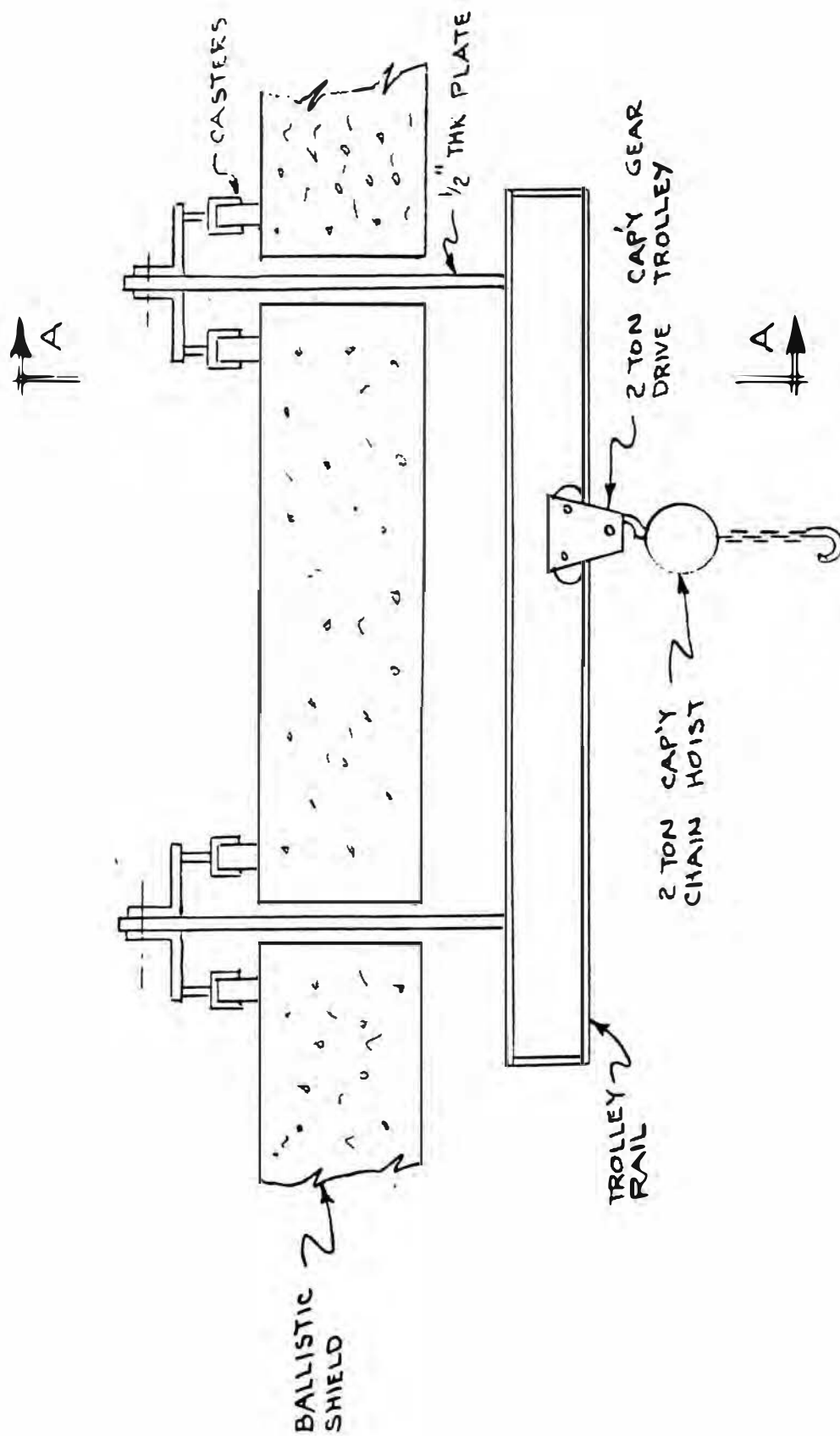
STEP C
FIGURE IV-2
81



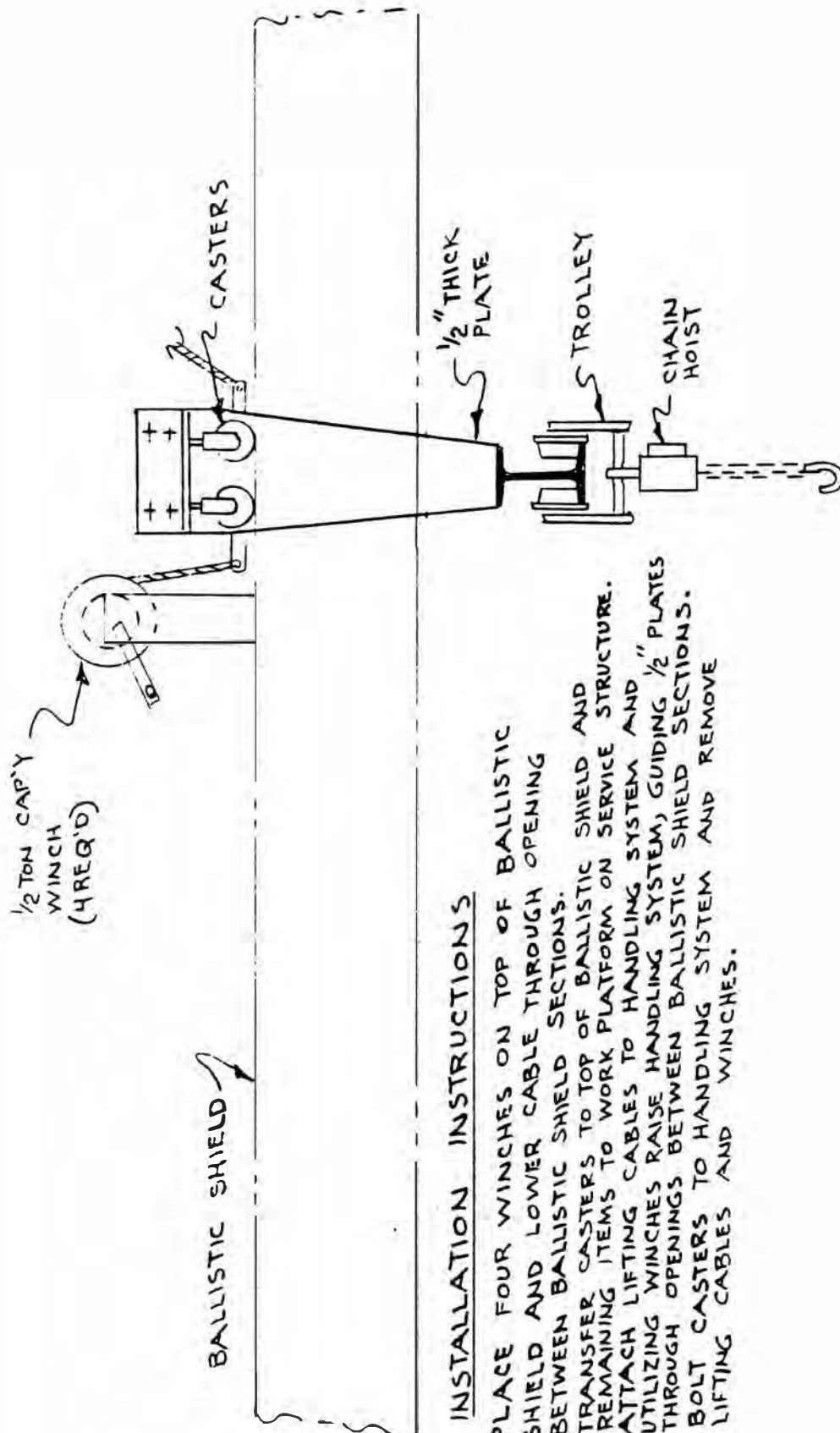
STEP D
 FIGURE IV-2
 82



STEP E
 FIGURE IV-2



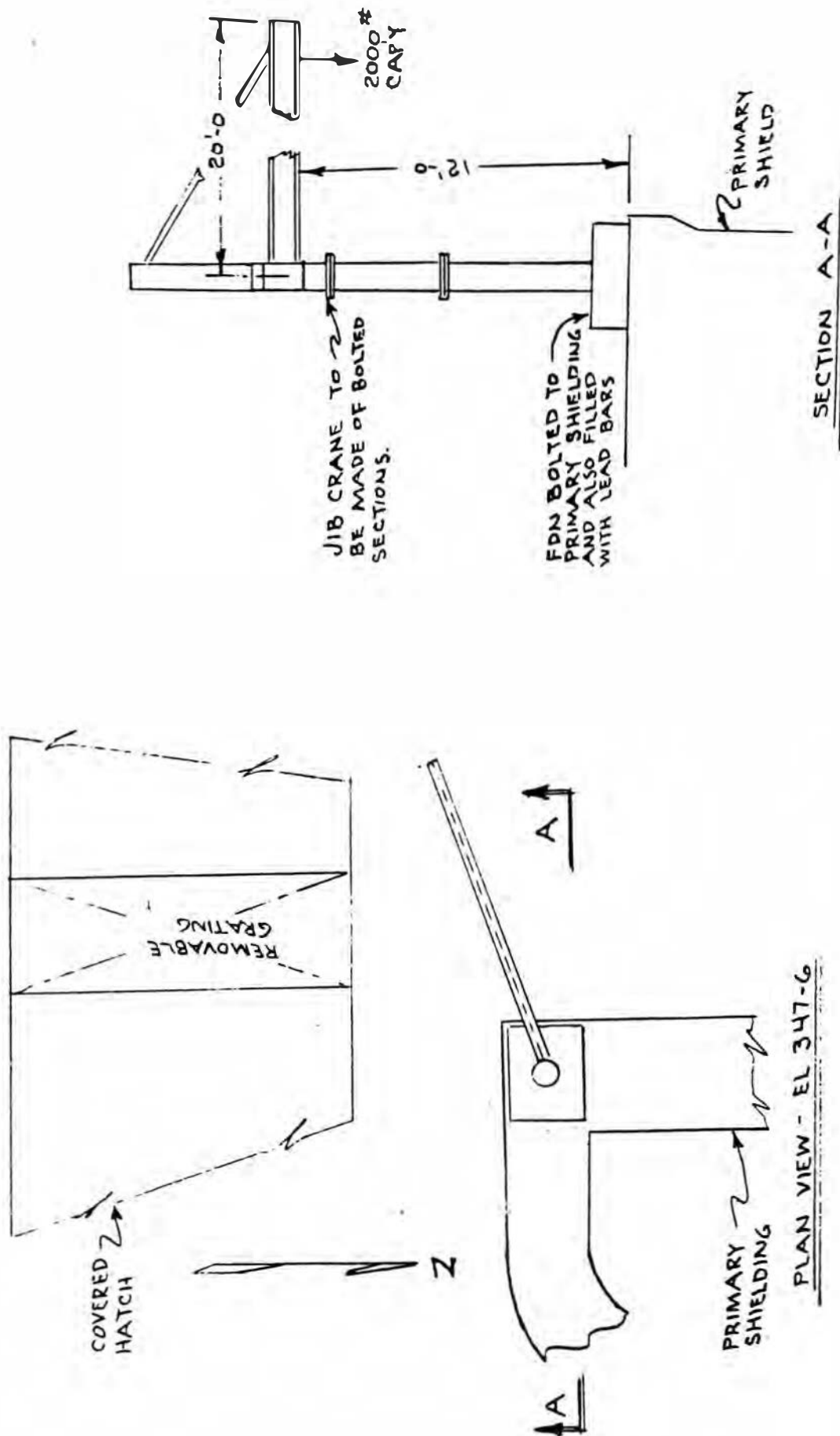
HANDLING SYSTEM TO BE INSTALLED
UNDER, AND SUPPORTED BY BALLISTIC SHIELD



INSTALLATION INSTRUCTIONS

1. PLACE FOUR WINCHES ON TOP OF BALLISTIC SHIELD AND LOWER CABLE THROUGH OPENING BETWEEN BALLISTIC SHIELD SECTIONS.
2. TRANSFER CASTERS TO TOP OF BALLISTIC SHIELD AND REMAINING ITEMS TO WORK PLATFORM ON SERVICE STRUCTURE.
3. ATTACH LIFTING CABLES TO HANDLING SYSTEM AND " PLATES UTILIZING WINCHES RAISE HANDLING SYSTEM, GUIDING 1/2 PLATES THROUGH OPENINGS BETWEEN BALLISTIC SHIELD SECTIONS.
4. BOLT CASTERS TO HANDLING SYSTEM AND REMOVE LIFTING CABLES AND WINCHES.

SECTION A-A



2000 LB CAPY JIB CRANE LOCATED ON
PRIMARY SHIELD

FIGURE IV-4

The missile shield lift rig would be transferred to the missile shield area in sections and then assembled. (See Figure IV-5 for description and operation of proposed missile shield lift rig.) After removal of the one missile shield, the lifting hoist for handling leadscrews and its tooling is brought in and assembled on top of missile shield (see step A) and the hoists guides are installed to contain the hoist trolley on the missile shield. The leadscrew would be removed (steps B thru E) in one section using the uncoupling tool and short lift tool and lowered into a PVC pipe type storage can located on the canal wall.

To permit removal of the leadscrew from containment building for evaluation, raise the leadscrew out of storage, cut leadscrew into sections and store in PVC pipe sections as discussed in paragraph IV.A.2.a above (Concept 1).

c. Leadscrew Removal with Two Missile Shields Removed (Concept 3)

Leadscrew removal for this concept* is similar to the prior concept except that both center missile shields are removed and stacked on adjacent shields to permit access to all the CRDM's. In addition, the leadscrew tools lift rig is extended to permit removal of the leadscrew in a

NOTE:

1. CONSTRUCTION TO BE 12" X 50" I (ASTM A440 STEEL)
2. WEIGHT OF EACH ASSEMBLY - APPROX 300*

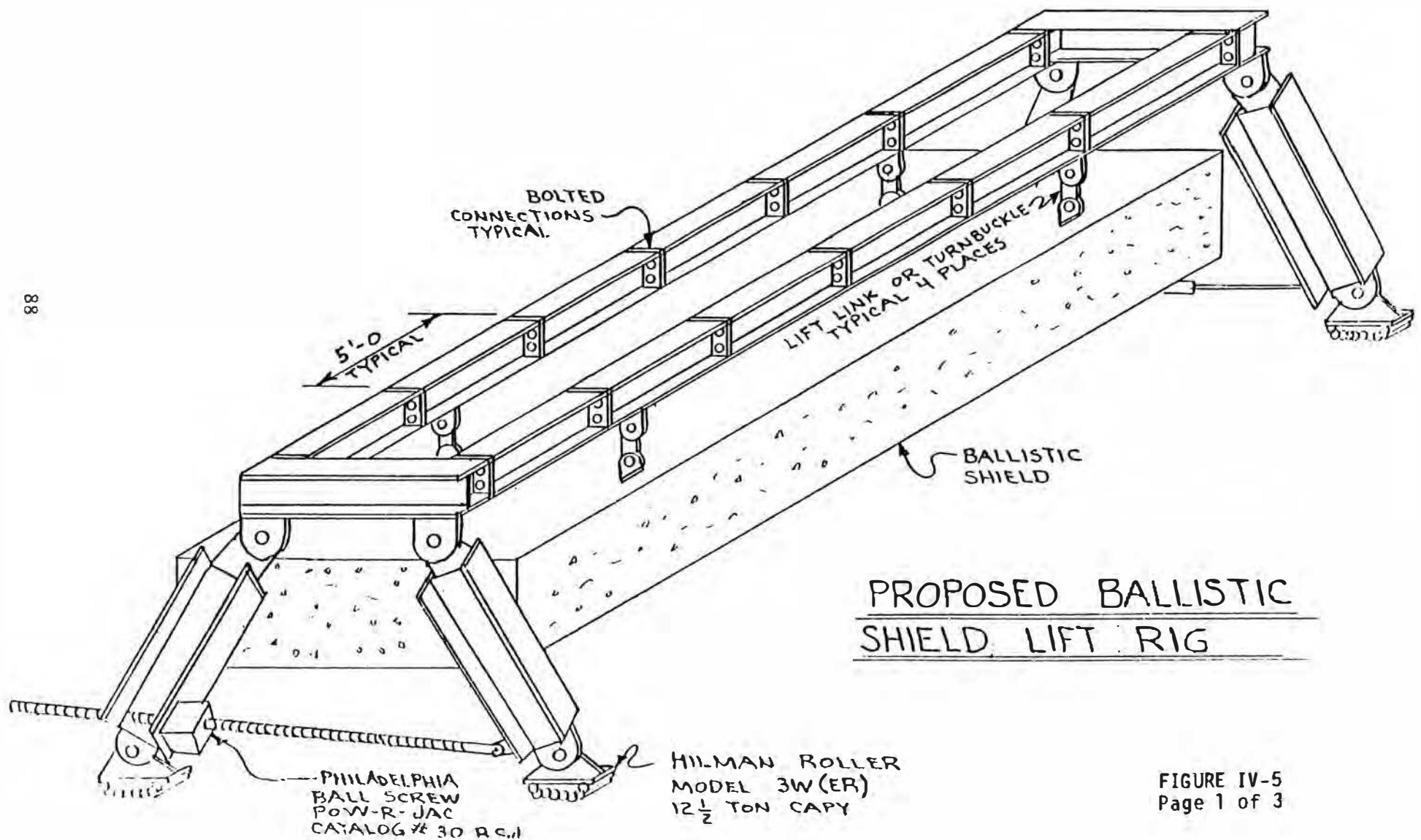
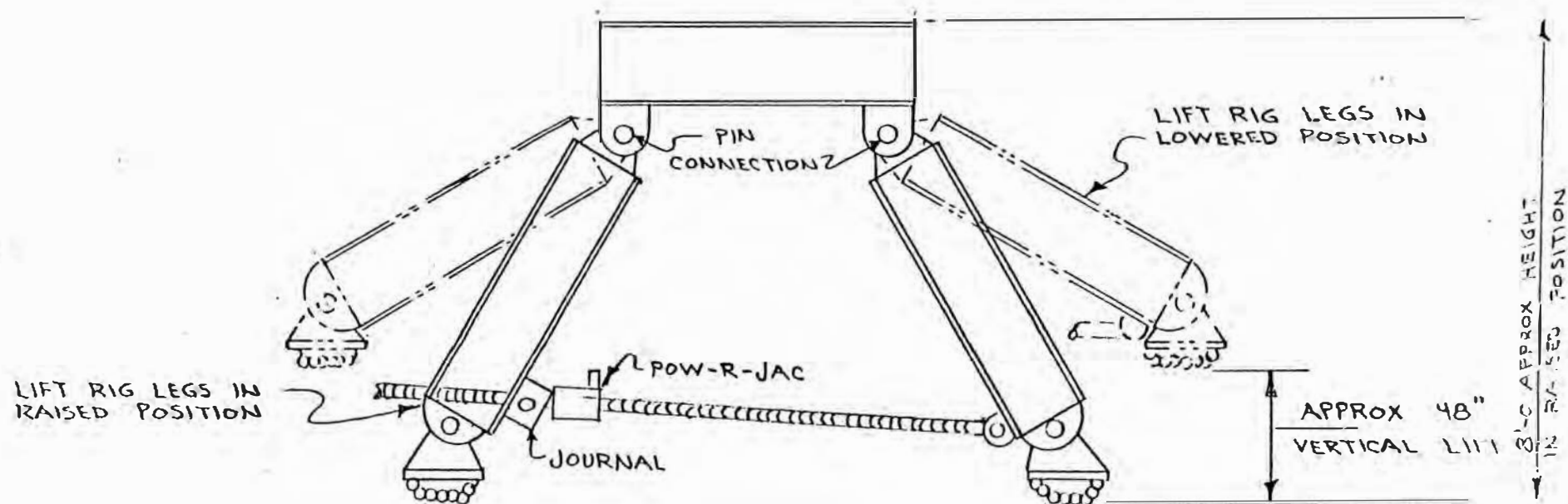


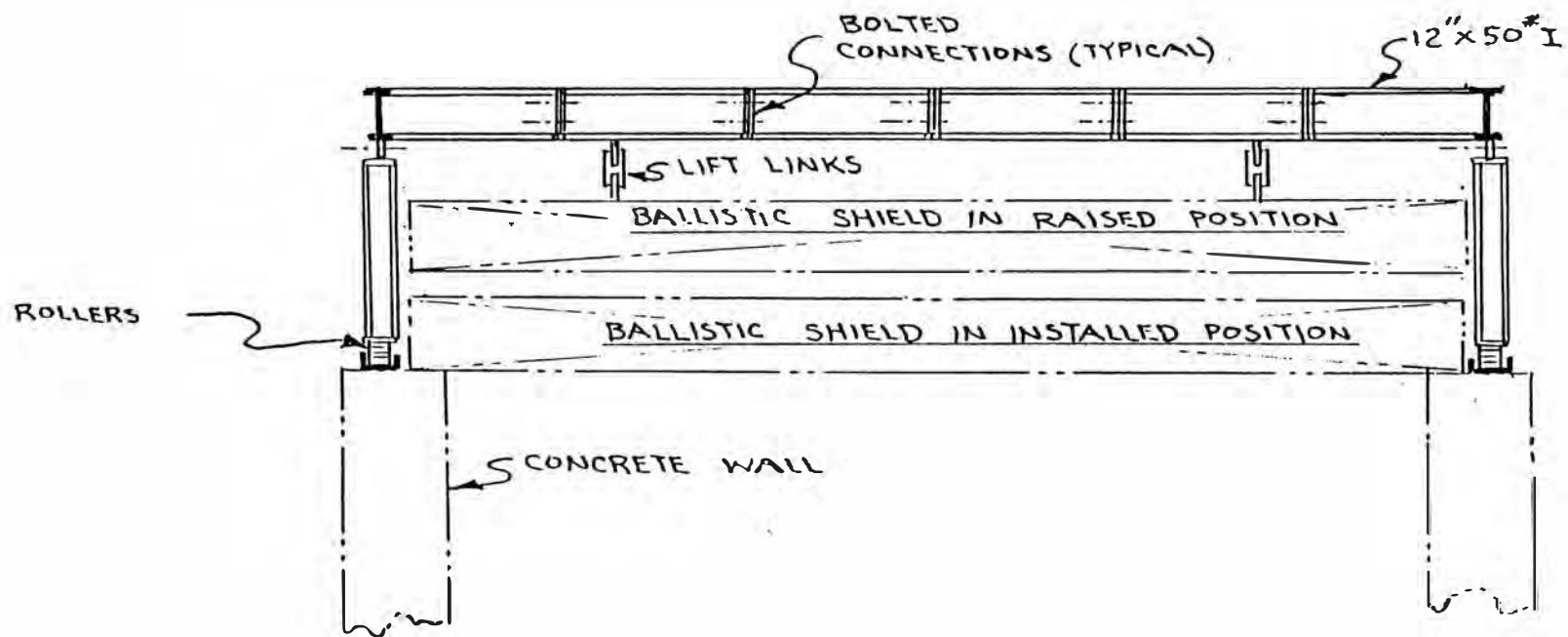
FIGURE IV-5
Page 1 of 3



NOTE:

LIFT RIG TO BE ASSEMBLED WITH LEGS EXTENDED IN LOWERED POSITION OVER BALLISTIC SHIELD, LIFT LUGS ATTACHED AND THEN LEGS RETRACTED USING SCREW JACK TO RAISE LIFT RIG WITH BALLISTIC SHIELD ATTACHED. BALLISTIC SHIELD IS THEN ROLLED TO STORAGE AREA.

OPERATION OF PROPOSED
BALLISTIC SHIELD LIFT RIG



VIEW SHOWING MOVEMENT OF A REMOVED BALLISTIC SHIELD
SECTION OVER AN INSTALLED SECTION

shielded container. After removal of the leadscrew into the shielded container, the container is moved laterally and lowered into storage position adjacent to the water pit wall.

To permit removal of the leadscrew from containment building for evaluation, raise the leadscrew out of its storage container, cut leadscrew into section and store in PVC pipe sections as in Concept 1.

* Newport News Drawing for Concept 3 will be forwarded by separate cover.

3. Man-Rem Study and Proposed Flow Charts

- a. Table 1 presents a man-rem and man-hour study for a "quick look" inspection via a leadscrew hole without removal of the missile shield (i.e., Concept 1). The total man-rem exposure for this type of "quick look" concept is 45.1. Of this total, 29.35 man-rem is for the "Quick Look" and 15.75 man-rem required for subsequent reactor head removal.
- b. Table 2 presents a man-rem and man-hour study for a "quick look" inspection with the missile shield removed (concept 3). The total man-rem exposure for their type

of "quick look" concept is 138.9. Of this total, 18.6 man-rem is for the "Quick Look", 101.8 man-rem for missile shield removal, and 18.4 man-rem required for subsequent reactor head removal.

- c. Table 3 provides a summary comparisons of man-rem and man-hours for Concept 1 and 3. The difference between Concept 2 and 3 is small.

4. Advantages/Disadvantages of Concepts

Table 4 provides a listing of the advantages and disadvantages for the three concepts presented.

TABLE 1

TMI UNIT 2 "QUICK LOOK" INSPECTION MAN REM STUDY
MISSILE SHIELDS INSTALLED - CONCEPT 1

Description	Manning	Time (hrs)	Man- Hours	Assumed Rad (R/Hr) Level	Man Rem
1. Install Temp. Lighting in Head Service Structure Area	2	4	8	.35	2.8
2. Prepare Tools & Equipment (Note: Qualification of 10 people 24 hrs. each not included)	2	16	32	Out of Rad Area	- -
3. Install Video Equipment for General Surveillance	2	8	16	.35	5.6
* 4. Transfer Lifting Hoist to Missile Shield Area	3	1	3	.3	.9
5. Install Draindown Hoses & Install & Calibrate Water Level Indic. for RC System	4	40	160	Out of R.B. Rad Area	- -
6. Lower Reactor Vessel Water Level No to EL335	2	24	48	Out of R.B. Rad Area	- -
7. Install CRDM Venting Tool/Hoses on Mechanisms (5CRDMs)	2	2	4	.35	1.4
8. Depressurize & Vent 5 Center CRDM's and RC System High Points	3 2	2 1	6 2	.3 .5	1.8 1.0
9. Activate Water Level Indication System	1	1	1	Out of R.B. Rad Area	- -
* 10. Remove CRDM Closure Insert Assy.	3	.5	1.5	.3	.45
11. Transfer leadscrew removal and cutting equipment to head service structure area.	3	1	3	.3	.9
12. Rig and Install Leadscrew Installation/Removal Tool (706564-1057) (1 Position)	3	.5	1.5	.3	.45
13. Apply Load and Unlatch Leadscrew (1 Positions)	3	.5	1.5	.3	.45
14. Untorque Leadscrew Nut	3	.5	1.5	.3	.45
15. Raise Leadscrew to Park Position and Install Restraining Device (1 Position)	3	.5	1.5	.3	.45

TABLE 1

TMI UNIT 2 "QUICK LOOK" INSPECTION MAN REM STUDY
MISSILE SHIELDS INSTALLED - CONCEPT 1

Description	Manning	Time (hrs)	Man- Hours	Assumed Rad (R/Hr) Level	Man Rem
16. Disconnect and Remove Leadscrew Installation/Removal Tool (1) Position	3	.5	1.5	.3	.45
* 17. Rig and Install Short Leadscrew Lift Tool (706568-1053) (1 Position)	3	.5	1.5	.3	.45
* 18. Raise Leadscrew to 12 feet and install clamp. (see step 5 if Concept 1)	4	1	4	.75	3
* 19. Install cutting equipment and cut lead screw	4	1	4	.75	3
* 20. Transfer first section of lead screw to storage	4	1	4	.75	3
* 21. Grapple, remove and place second section of lead screw in storage	4	1	4	.75	3
* 22. Transfer Leadscrew sections from Head Service Structure Area	4	1	4	.35	1.4
* 23. Transfer TV equipment to Head Service Structure Area	2	1	2	.3	.6
* 24. Install and Setup Core Inspection Equip. in Head Service Structure Area	4	1.0	4.0	.3	1.2
* 25. Perform Core Video Inspection (1 Position)	4	3	12.0	.3	3.6
* 26. Refill and Vent RCS (Note: <u>May not be required</u>)	2	.5	1	.35	.35
* 27. Bag & Remove TV Inspection Equip.	2	.5	1	.35	.35
TOTALS			356.6		45.1**
			Quick Look 69	Quick Look	29.35
			R.Head Rmvl. 287	R.Head Rmvl.	15.75

* Activities unique to quick look inspection and would not be required for the planned defueling.

** Cutting for leadscrew removal from R.C. Bldg. is included in this Man-rem estimate.

NOTES FOR TABLE 1

1. Man rem estimates are based on present radiological conditions. If large scale decontamination experiment is completed and RCS has been purified via SDS, lower radiation levels can be expected in general areas and in head service structure area.
2. It is estimated approximately 16 man rem, out of a total of 45 man-rem, will be expended to perform tasks which were originally planned for reactor disassembly except that now these tasks would be performed earlier at higher radiation levels.
3. Failure to uncouple lead screws could require attempts on others necessitating the expenditure of an additional 3 to 5 man rem for each lead screw. Since uncoupling is required for head removal anyway, it is not a penalty against a "quick look" inspection.

TABLE 2

TMI UNIT 2 "QUICK LOOK" INSPECTION MAN REM STUDY
TWO MISSILE SHIELDS REMOVED - CONCEPT 3

Description	Manning	Time (hrs)	Man-Hours	Assumed Rad (R/Hr) Level	Man Rem
* 1. Install Gantry Crane at Elevation 347'6" over hatch	6	16	96	.3	28.8
* 2. Install Jib Crane at Elevation 367'4"	6	8	48	.2	9.6
3. Install Temp. Lighting in Head Service Structure Area	2	4	8	.35	2.8
4. Prepare Tools & Equipment (Note: Qualification of 10 People 24 hrs each not included)	2	16	32	Out of Rad Area	- -
5. Install Video Equipment for General Surveillance	2	8	16	.35	5.6
* 6. Transfer Missile Shield Lift Rig to Head Service Structure Area	6	2	12	.3	3.6
* 7. Set-up Lift Rig for Missile Shield	5	8	40	.5	20.0
* 8. Raise & Transfer Missile Shield R2 on Top of Shield R1	5	2	10	.5	5.0
* 9. Raise & Transfer Missile Shield (R3) on Top of Shield (R4)	5	2	10	.5	5.0
* 10. Disconnect Missile Shield Lift Rig & roll to North Storage Area	3	1	3	.5	1.5
* 11. Assemble & Install Gantry Crane (4000 lb. CAP) on top of Stacked Missile Shields (Elev. 374')	5 1(QC)	16 1	80 1	.35	28.3
12. Install Draindown Hoses & Install & Calibrate Water Level Indic. System	4	40	160	Out of Rad Area	- -
13. Lower Reactor Vessel Water to EL 335	2	24	48	Out of Rad Area	- -
14. Install CRDM Venting Tool/Hoses on Mechanisms (5CRDM's)	2	2	4	.35	1.4
15. Depressurize & Vent 5 Center CRDM's and RC System High Points	3 2	2 1	6 2	.3 .5	1.8 1

TABLE 2

TMI UNIT 2 "QUICK LOOK" INSPECTION MAN REM STUDY
TWO MISSILE SHIELDS REMOVED - CONCEPT 3

Description	Manning	Time (hrs)	Man-Hours	Assumed Rad (R/Hr) Level	Man Rem
16. Activate Water Level Indication System	1	1	1	Out of Rad Area	- -
* 17. Remove CRDM Closure Insert Assy.	3	.5	1.5	.3	.45
18. Transfer leadscrew removal equip. to head service structure area	2	1	2	.3	.6
19. Rig & Install Leadscrew Installation/Removal Tool (706564-1057) (1 Position)	3	.5	1.5	.3	.45
20. Apply Load & Unlatch Leadscrew (1 Position)	3	.5	1.5	.3	.45
21. Untorque Leadscrew Nut	3	.5	1.5	.3	.45
22. Raise Leadscrew to Park Position and Install Restraining Device (1 Position)	3	.5	1.5	.3	.45
23. Disconnect & Remove Leadscrew Installation/Removal Tool (1 Position)	3	.5	1.5	.3	.45
* 24. Rig & Install Short Leadscrew Lift Tool (706568-1053)(1 Position)	3	.5	1.5	.3	.45
* 25. Remove Leadscrew & Close CRDM (1 Position)	4	1	4	.75	3
* 26. Transfer Leadscrew from Head Service Structure Area (1 Position)	4	1	4	.35	1.4
* 27. Transfer TV Equip. to Head Service Structure Area	2	1	2	.3	.6
* 28. Install & Setup Core Inspection Equip. in Head Service Structure Area	4	1.0	4.0	.3	1.2
* 29. Perform Core Video Inspection (1 Position)	4	3	12	.3	3.6

TABLE 2

TMI UNIT 2 "QUICK LOOK" INSPECTION MAN REM STUDY
TWO MISSILE SHIELDS REMOVED - CONCEPT 3

Description	Manning	Time (hrs)	Man- Hours	Assumed Rad (R/Hr) Level	Man Rem
* 30. Refill & Vent RCS (Note: <u>May</u> <u>not be required</u>)	2	12	24	.35	8.4
* 31. Bag & Remove TV Inspection Equip.	2	.5	1	.35	.35
TOTAL		M-hr total	640.5		138.9**
		Quick Look	354	Quick Look	120.35
		R.Head Rmvl.	286.5	R.Head Rmvl.	18.55

* Activities unique to quick look inspection and would not be required for the planned defueling.

** Cutting for leadscrew removal from R.C. Bldg. roughly estimated about another ~12 m-rem.

NOTES FOR TABLE 2

1. Man-rem estimates are based on present radiological condition. If large scale decontamination experiment is completed and RCS has been purified via SDS, lower radiation levels can be expected in general areas and in head service structure area.
2. It is estimated approximately 18 man rem, out of a total of 139 man-rem, will be expended to perform tasks which were originally planned for reactor disassembly except that now these tasks would be performed earlier at higher radiation levels.
3. Failure to uncouple lead screws could require attempts on others necessitating the expenditure of an additional 3 to 5 man rem for each lead screw. Since uncoupling is required for reactor head removal anyway, it is not a penalty against a "quick look" inspection.

TABLE 3

SUMMARY COMPARISON

Man-Hour Effort and Man-Rem StudyConcept 1 - Missile Shield Left in Place

<u>Man-Rem</u>	<u>Man-Hours</u>
<u>45.1</u> (total)	<u>356.5</u> (total)
29.35 (Quick Look)*	69 (Quick Look)
15.75 (Req'd for Reactor Head Removal)	287.5 (Req'd for Reactor Head Removal)

* The exposure to cut leadscrew and bring it out of containment is included in this estimate.

Concept 3 - Two Missile Shields Removed

<u>Man-Rem</u>	<u>Man-Hours</u>
<u>138.9</u> (total)	<u>640.5</u> (total)
18.6 (Quick Look)**	354 (Quick Look)
101.8 (Shield Removal)	286.5 (Req'd for Reactor Head Removal)
18.4 (Req'd for Reactor Head Removal)	

** About 12 additional M-Rem would be required to cut leadscrew into sections for removal out of Containment Building for evaluation.

TABLE 4

ADVANTAGES/DISADVANTAGES OF EACH "QUICK LOOK" METHOD

1. CONCEPT 1 - WITHOUT REMOVAL OF MISSILE SHIELDS

Advantages

1. Shortest possible time for "quick look" since missile shields do not have to be removed.
2. Lowest man-rem and man effort.
3. Does not require installation of a crane at 347'6" or 367'4" elevation.
4. Requires only a simple lightweight hoist on the missile shield to remove the leadscrew.
5. Requires approx. 94 man-rem less than Concept 3.
6. Cut leadscrew and its storage container are light and readily removable from the Containment Building for evaluation.

Disadvantages

1. Requires cutter to remove leadscrew from RV.
2. Requires one to three cuts for storage cans on canal wall for leadscrew sections.
3. Missile shield still installed and could limit future head area work until polar crane operational.

4. Can only reach 7 CRDMS through the center gap in the missile shields (Note: with a special hoist mounted under the missile shields all CRDMS can be disconnected.)

2. CONCEPT 2 - REMOVAL OF 1 MISSILE SHIELD

Advantages

1. Leadscrew removed from RV in one piece.
2. Permits future work on head area without missile shield.
3. Cranes available on 347'6" level and 367'4" level to support other work.
4. Better access to head area.

Disadvantages

1. Requires crane to be installed on 347'6" level and 367'4" level to handle equipment and tools.
2. Requires lift rig for lifting one missile shield.
3. Requires a gantry crane to be installed on top of missile shield to handle leadscrew.
4. Restricts access to head area with just one missile shield removed.
5. Requires leadscrew cutter & storage cans to remove leadscrews from Containment Building for evaluation.

3. CONCEPT 3 - REMOVAL OF 2 MISSILE SHIELDS

Advantages

1. Allows greatest access to all CRDM positions.
2. Leadscrew removed from RV in one piece.
3. Permits future work on entire head area without missile shield.
4. Unrestricted access to entire head area.
5. Cranes available on 347'6" level and 367'4" level to support other work.

Disadvantages

1. Requires crane to be installed on 347'6" level and 367'4" level to handle equipment and tools.
2. Requires lift rig for lifting one missile shield.
3. Requires a gantry crane to be installed on top of missile shield to handle leadscrew.
4. Requires Provision to restrain gantry crane on missile shield.
5. Highest man-rem. (138.9 man-rem)
6. Requires leadscrew cutter & storage cans to remove leadscrew from Containment Building for evaluation.

B. Polar Crane

The importance of the Polar Crane to the defueling and the obtaining of necessary information and the refurbishment of the crane as soon as practicable cannot be over stressed. It should also be mentioned that the auxiliary hoist of the Polar Crane, once missile shields have been removed, becomes a necessary piece of lifting equipment up to the point of head removal.

After review of reports, drawing, photographs and crane specifications, the following comments and recommendations are made.

1. After reviewing a technical report* on non destructive testing of wire rope, the report was discussed with Mr. Percy Malone, Naval Facilities Engineering Command, Code 453D, Alexandria, Virginia, who advised that, of the three methods tested, the Magnograph is preferred. He stated that this equipment will provide better inspection than can be performed by visual inspection alone and it will identify and measure internal defects such as broken wires and corrosion. Mr. Malone recommended that, prior to use of the Magnograph, it should be calibrated using a sample section of the same manufacturer's class, construction and size wire rope as that which is to be inspected. Based on the inspection results** that the wire rope inspected is in good condition, still well lubricated and with only a thin layer of surface rust on the rope, it

is considered that the Magnograph inspection of all the rope which can be made accessible by lowering the crane block to its lowest practical position and by removal of rust by wire brushing and visually inspecting the entire rope for pitting would provide adequate information to determine if the wire rope should be reused. The wire rope would not be acceptable if the cross-sectional metallic area is reduced by more than ten percent, if more than three randomly dispersed broken wires are found in one strand in one rope lay or six broken wires in one rope lay, or if significant pitting of individual wires is observed. It is also recommended that the wire rope be opened up in two or three representative locations using a marlinspike, screwdriver or similar tool to verify visually that the rope is not corroded internally.

* Technical Note No. N-1594, Nondestructive Test Equipment for Wire Rope, dated October 1980.

** Bechtel Northern Division Letter BLGE-0394, dated December 31, 1981.

2. Based on photographs and preliminary inspection results *** showing no evidence of heat damage to the crane structure, it is recommended that all rust be removed and all accessible areas, including the grooves of the hoist drums with the wire rope removed to the extent practical,

be visually inspected for cracks, deformation or other degradation. If found, deficiencies should be repaired or replaced. Nondestructive testing should be employed, if needed for resolving questionable defects or for inspection of critical load bearing welds. Hooks should be subjected to magnetic particle inspection prior to reuse (preferably after load testing). If sandblasting is used for rust removal, all wire rope should be protected against abrasion and/or adherence of blasting grit to the wire rope lubricant. After resolution of the above, the crane should be repainted as required.

*** Letter to G. W. Tucker from Thomas A. Russell, International Energy Associates, Ltd., dated January 20, 1982 and GPU Letter TMI-II-RE-0251, dated January 22, 1982.

3. The TAAG concurs with the performance of detailed inspections and replacement of any crane components such as bearings, flexible couplings, clutches, brakes, switches, etc., which are found to be in poor condition. The TAAG also concurs with a thorough inspection and checkout of electrical components and repair or replacement of all defective items. All of this work plus removal of old lubricants, where applicable, and replacement with new lubricants should be completed prior

to operational testing. It is recommended that the crane be then checked out with no load through all of its operational modes to ensure that it operates as designed. Special attention should be paid to proper operation of safety devices, such as limit switches, alarms, indicator lights, etc.

4. It is recommended that the crane be load tested to 312.5 short tons to establish a rated capacity of 250 short tons for performance of the work presently planned to be accomplished. This should provide approximately a 10:1 safety factor at the rated capacity, based on the original rope strength, thus allowing a reasonable margin for unknown contingencies. If any cracking sounds (pings) are heard emanating from the wire rope during load testing, it is recommended that the Magnograph and visual inspections be repeated to verify that breaking of wires does not occur during load testing. If more than one additional broken wire in one strand in one rope lay or more than three additional broken wires in one rope lay occur during load testing, replacement of the wire rope is recommended. Wire rope should be cleaned and lubricated prior to use.

5. It is expected that future inspection of the auxiliary hoist will reveal conditions similar to those encountered on the main hoist. If so, the preceeding recommendations except for capacity, load test weight, etc., would also apply to that hoist. Sufficient information to evaluate the load testing and rated capacity of the auxiliary hoist is not available but it is expected that downgrading of the rated capacity would be recommended if the existing wire rope is to be retained.

A. Water Level Measurement

1. Purpose

The purpose of this section of the report is to identify a potential way to monitor the TMI-2 reactor coolant level when the primary system is partially drained. In particular, it is to examine the possibility of monitoring the coolant level by a method which is outside of the containment building, and thereby avoid having to install such a system within the reactor containment building. Such a monitoring system will also avoid the need for personnel to enter the containment building to maintain, calibrate and check the level monitoring system.

2. Background

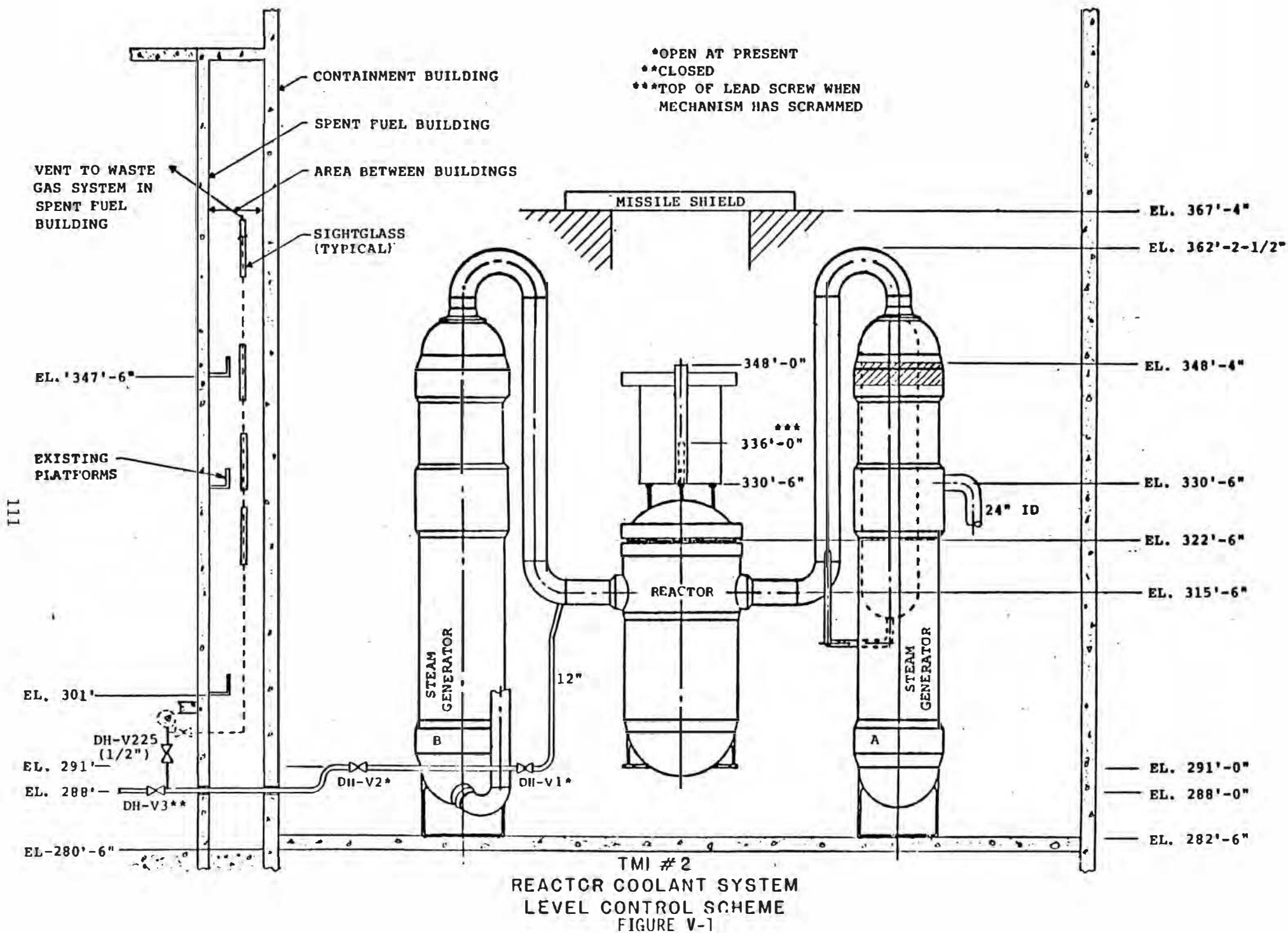
Presently, the TMI-2 reactor coolant system is basically filled and has the following approximate conditions: 80-100 psi and about 100°F.

There is an existing 12-inch decay heat removal line that comes off of the reactor coolant system hot leg just before the "B" steam generator. This 12-inch line comes off the reactor coolant system and proceeds out the containment

building and comes into the lowest level of the spent fuel building (see Figure V-1). There are three valves in this decay heat line, two of which are inside of containment (i.e., DH-V1 and DH-V2), and they are both presently open. The boundary of the reactor coolant system thus is DH-V3 which is located just inside of the lower level of the spent fuel building. Just after the decay heat line enters the spent fuel building and before DH-V3, there is a half-inch line which comes off the side of the 12-inch decay heat line. This one-half inch line is used to monitor low pressure in the reactor coolant system, and it has a valve (DH-V225). The elevations of the decay heat line relative to the reactor coolant system and the spent fuel building are shown in Figure V-2. The location of valves DH-V3 and DH-V225 in the spent fuel building can also be seen in Figure V-3. There is also space with access platforms between the containment building and the spent fuel building which can also be seen in these same figures.

3. Discussion

In view of the fact that the reactor coolant system boundary now comes outside of the containment building into the spent fuel building, it would appear there is an opportunity to tie into the half-inch line just before DH-V3 to make a water level monitoring system for the reactor coolant system when that system is partially drained for such operations as



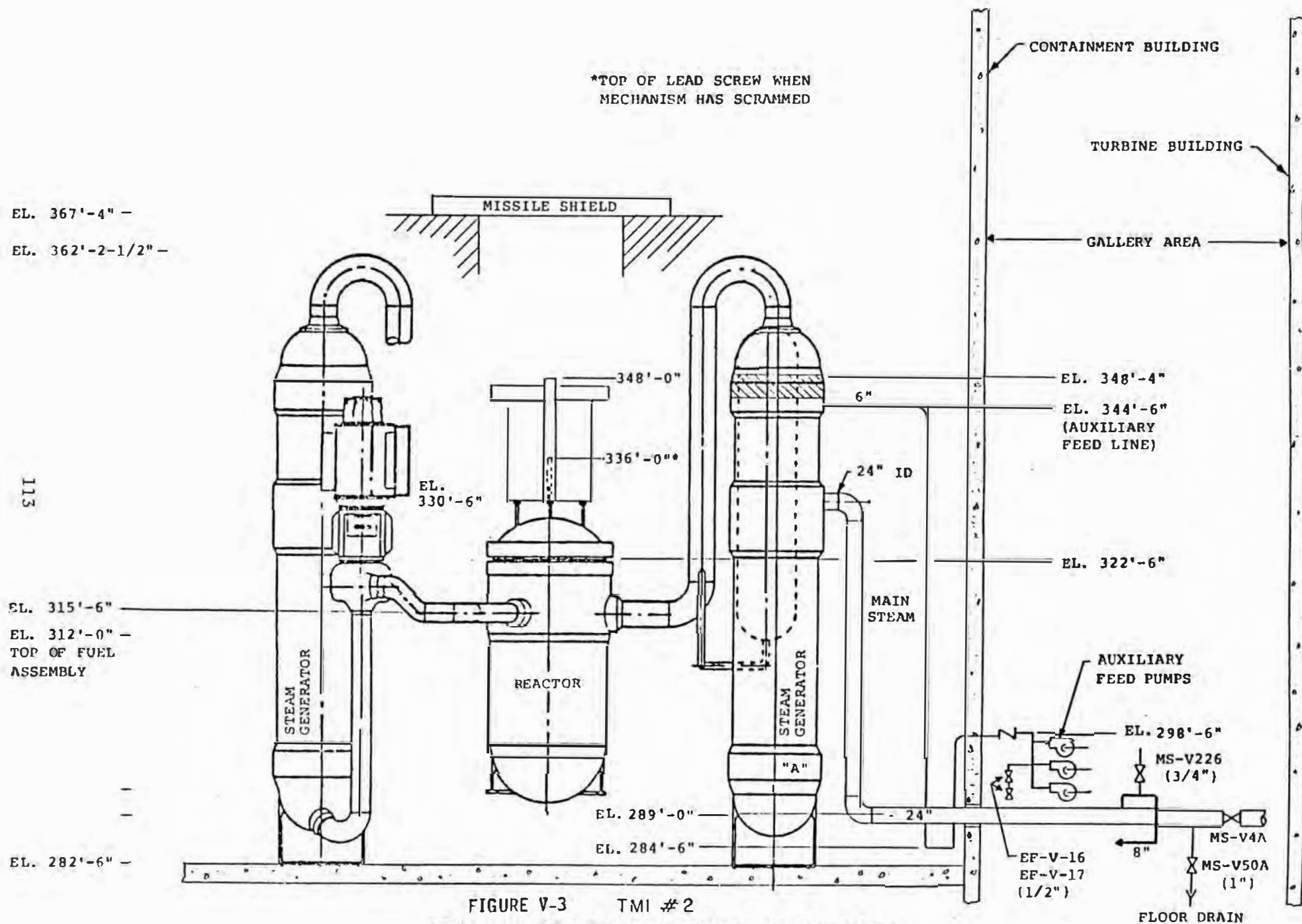


FIGURE V-3 TMI #2
 DRAINING OF SECONDARY SIDE OF "A" OTSG

control rod drive mechanism removal, head removal, etc.

Basically, a standpipe type level monitor can be run off the line associated with DH-V225 and proceed from the spent fuel building area into the area between the spent fuel building and the containment building (see Figures). The level monitor line can then be run vertically all the way to almost the top of the spent fuel building.

Conceptually, it is assumed that the level standpipe would have lexan or heavy duty tygon tubing sight glasses at three elevations (i.e., one near the top of the primary system "J" leg, one near the top of the control drive mechanisms, and another one down near the reactor vessel flange area). With regard to installing the level monitor in the area between the two buildings, one should also note on Figure 3 that access platforms already exist at various elevations within this area, and should help ease any installation problems. In addition, if one also wanted to monitor the reactor coolant water level by measuring the static head, a low pressure type gage could also be installed with DH-V225.

With regard to radiation levels in the monitoring standpipe, it was assumed that half-inch ID tygon tubing would be used to contain the coolant, and a coolant activity of 50 microcuries per milliliter was also assumed. With such assumptions, the radiation from the standpipe would be less than 20 mr/hr at

one foot, and 3 mr/hr at three feet. Accordingly, the radiation levels of the standpipe will be well within reason. It should be noted that the latest reading of reactor coolant activity is less than 20 microcuries per milliliter, so the radiation levels should be appreciably lower than what we calculated.

4. Recommendation

Have the appropriate TMI-2 group proceed to make a detailed examination of the installation of such a reactor coolant level monitoring system, since it can be installed outside of the containment building and should allow for easier access for monitoring and maintaining the level equipment. Based on preliminary discussions with various site personnel, we see no real major technical problem or reason why a level monitoring system (e.g., standpipe site glass, static head pressure instrument, etc.) cannot be installed relatively quickly at DH-V3 outside of the containment building.

B. Draining Steam Generators

1. Purpose

The purpose of this study is to identify a potential way to drain the secondary side of the steam generator down to the level of the main steam nozzles. It is understood that there is some concern that when the reactor coolant level is drained down in the steam generator, there is a possibility that:

- a. The steam generator could go critical due to fuel stuck in the tubes. The present condition is over-moderated and draining of the primary side could result in criticality.
- b. The reactor could go critical due to inleakage of unborated secondary coolant into the primary system as a result of steam generator tube leaks and the secondary water level being higher than the primary.

Without debating the technical merits or validity of such concerns, there is a possible method which avoids having to resolve this matter for a "quick look" inspection via a leadscrew hole in the TMI-2 core (i.e., removing a leadscrew through the top of a mechanism and using a TV camera to inspect down in the leadscrew hole).

If the secondary side water level can be drained to the bottom of the main steam nozzle (EL. 329'-6") and the water level in the reactor coolant system drained down (EL. 335') to release the leadscrew, then the primary coolant will still be approximately five feet above the secondary side water level (see Figure V-5). With such a water level configuration, those issues concerning the steam generator that could lead to the remote possibility of recriticality of the core are avoided.

2. Background and Discussion

To perform a "quick look" inspection (as defined above), the water level in the primary reactor coolant system needs to be lowered about 13 feet below the top closure of the control drive mechanism. The water level would be at EL. 335'. With such a water level, the top end of the leadscrew would be about one foot above the water so the uncoupling tool could activate the leadscrew release mechanism without getting the uncoupling tool wet, and thus minimizing any contamination problems associated with leadscrew uncoupling. Accordingly, the purpose of this study is to develop a potential way for draining the secondary side down below the level of where the primary side would be drained. The basic concept would be to drain the secondary water out of the steam generator by the main steam lines through the annuli around the two high level

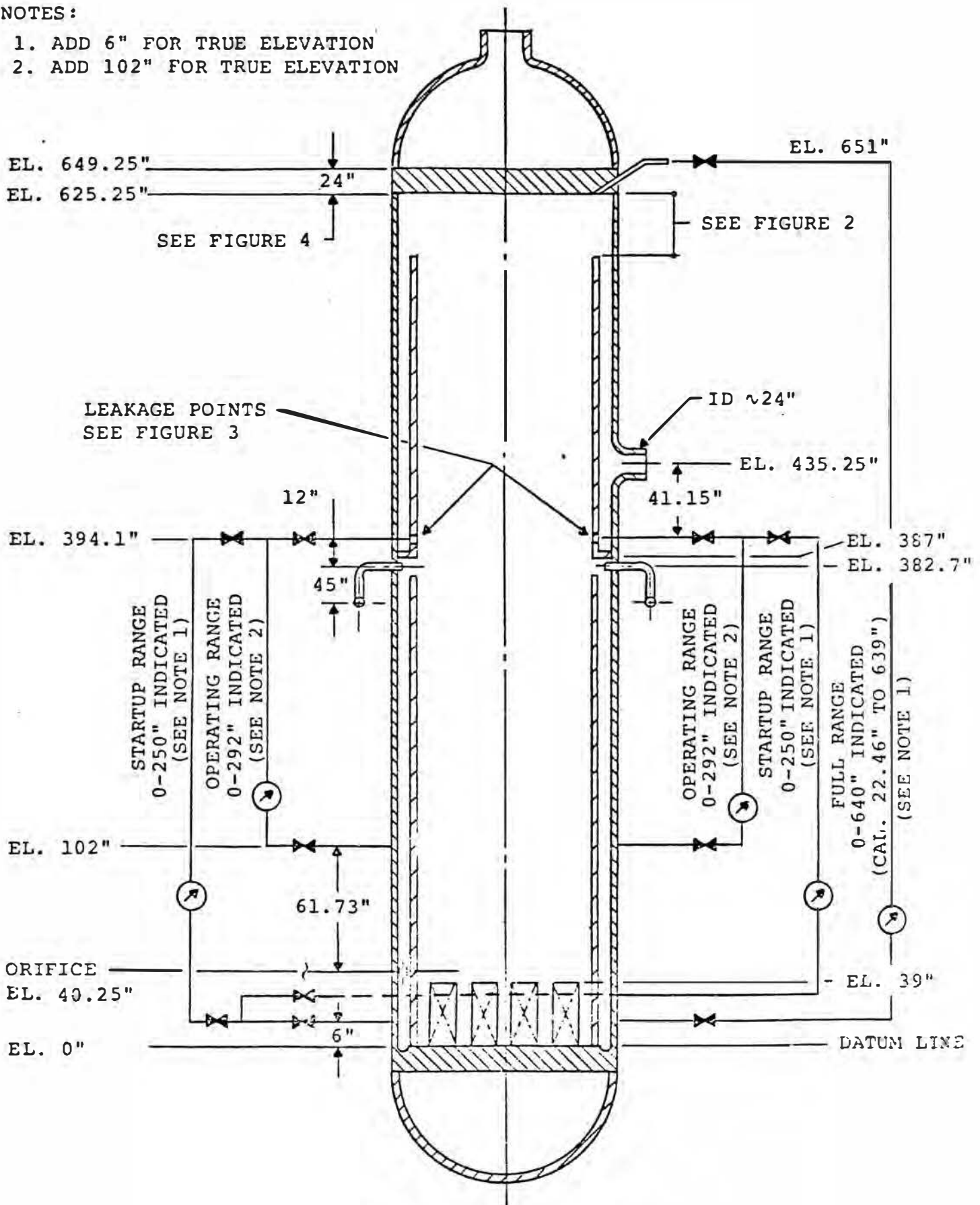
sensing lines (see Figures V-4, 5 and 6). There is approximately a 1/16-inch annulus around each of the two 1-inch OD high level sensing lines (see Figure V-6 for details).

For example, if the water level on the secondary side were eight feet above the main steam nozzles and atmospheric pressure on the top secondary water surface, approximately 33 gallons per minute could leak through these two annuli, and thus the secondary water inside the shroud area of the steam generator would drain out and down the main steam line. To vent the secondary side while draining, gas (nitrogen or helium) could be added through valves (EF-V-16 and EF-V-17) in the auxiliary feed system lines which feed to the top of the steam generator secondary side. With such a scheme, the secondary water should be able to be drained down to the bottom level of the main steam line which is elevation 329'-6". This secondary water can be drained out of the main steam pipes by one of several drain valves that are in the gallery area in between the turbine building and the containment building (see Figure V-5). The water drained from the secondary side can be measured to confirm the volume drained assuming the S.G. are completely full at present.

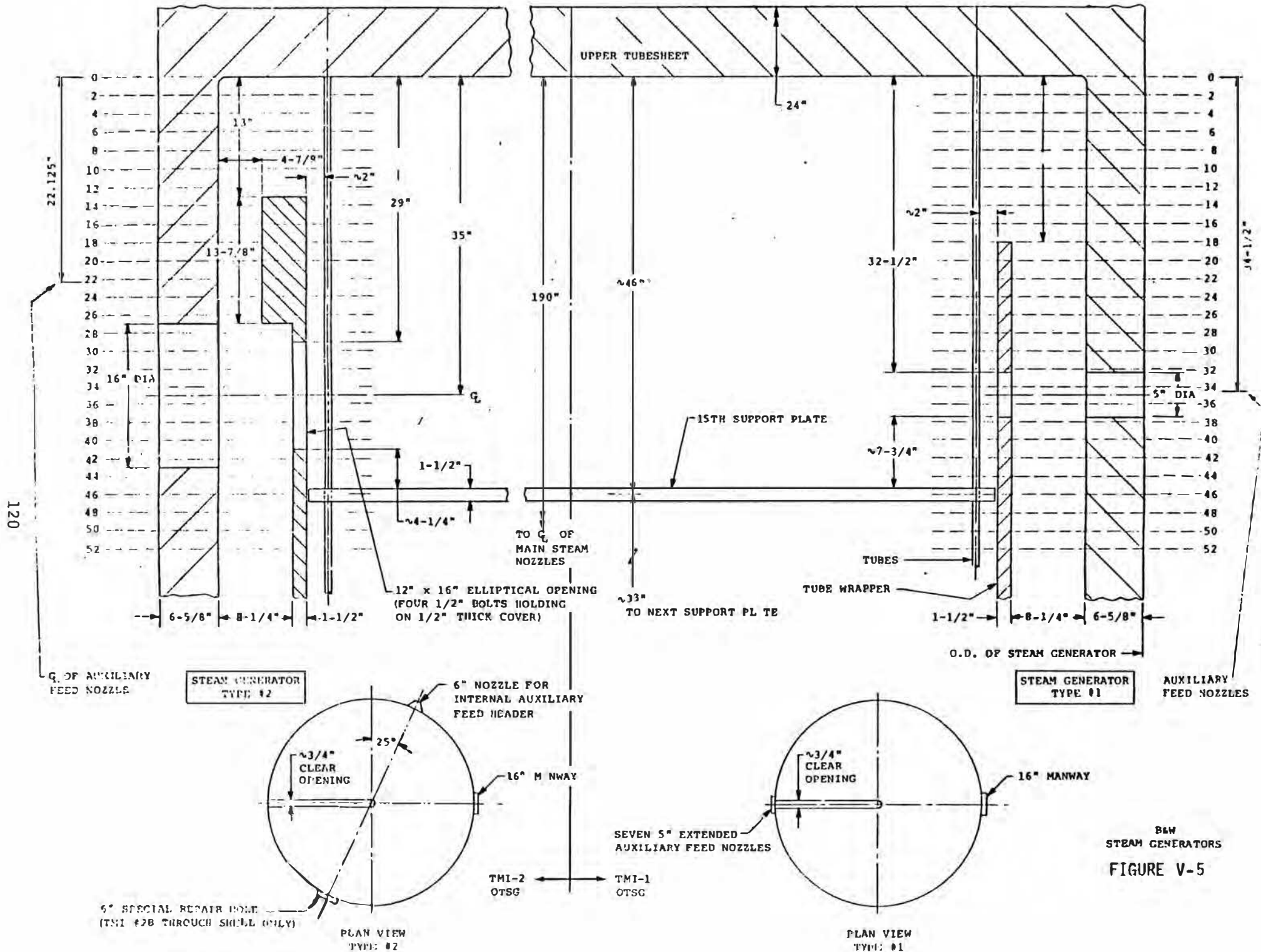
All of the above valves for draining and venting the secondary side are in low radiation areas (e.g., less than 5 mr/hr).

NOTES:

1. ADD 6" FOR TRUE ELEVATION
2. ADD 102" FOR TRUE ELEVATION



TMI UNITS 1 AND 2 OTSG
FIGURE V-4



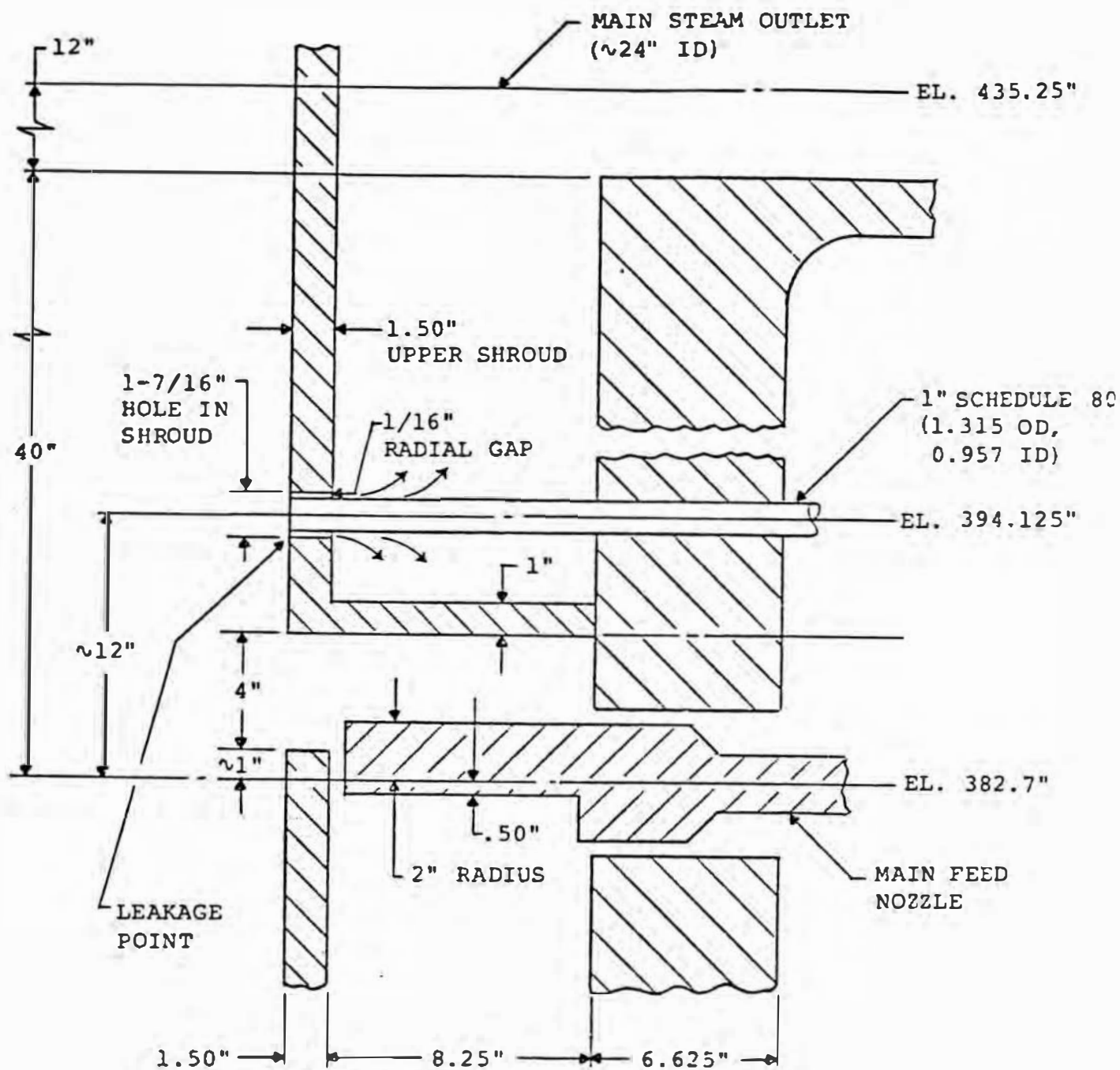


FIGURE V-6
 STEAM SYSTEM DETAILS

3. Recommendations

Have the appropriate TMI-2 group proceed to make a detailed examination of such a method for draining the secondary side. If the detailed review shows that this is a satisfactory draining method, it is recommended that implementation of the method proceed at an early date.

- C. FAILED FUEL HANDLING. The range of estimates of the condition and location of vessel internals is very broad, e.g. the fuel already outside the reactor is measurable in grams according to one expert. Because of the diversity of opinion regarding the core condition another estimates "more than three tons". This unique "failed fuel" complexity of TMI-2 prompted TAAG to obtain a first hand knowledge of existing industrial state-of-art failed fuel handling capabilities which might provide a base for TMI-2 defueling. A presentation was provided by Westinghouse on February 8 and by General Electric February 24. It is believed that the equipment and techniques observed or discussed are similar to those of other organizations, although other vendors and organizations also have useful inputs.

TAAG was impressed by the steady, tightly integrated evolution in tools, procedures, and training for handling failed fuel elements in defueling more than one-hundred reactors over the years. A large cadre of professional experts, technicians, and supporting capabilities has been established.

Vendors are accustomed to working in utility, licensed environments under tight schedules imposed by economics and radiation conditions. In some cases high radiation levels, fission gas evolution (Xe, etc.), and contamination by activation products (Co^{58} particularly) were encountered and overcome. Safe tooling and operating procedures have been developed and refined without loss of simplicity and reliability. Cores with 25% failed rods have been safely processed. Television cameras are used routinely for underwater viewing and manipulation of tools and core components.

Techniques have been developed for minimizing turbidity of water, for oxidation and controlled removal of Co^{58} , and for demineralization, degasification, and other water chemistry controls which are vitally important for handling even a relatively small number of failed fuel elements. One vendor representative said, "control of turbidity and water chemistry could be your most important problem". Colloidal iron was particularly troublesome.

Air operated tools have been developed for cutting and impact. Variations of these tools and the operating know-how appears relevant to breaking up and handling frequently postulated TMI-2 core materials such as fused "globbs" of Zr-U-Pu-Ag-In-Cd-Fe...

Special problems and know-how described by vendors included the handling & disposal of contaminated tools and laundry, communications equipment (throat microphones), xenon bubble evolution, and shut-down water chemistry. Security clearances and training can be a special problem if contract personnel are required - particularly on the TMI-2 scale.

On one hand, existing defueling experience provides an exceptional base to minimize TMI-2 clean-up costs and risks. On the other hand, the multifaceted defueling problem of TMI-2 will require better definition of the core condition to focus supporting efforts to avoid needless expenditure of time and money and, even more important, to minimize manpower training and qualification requirements.

TAAG reviewed failures and subsequent handling of Power Burst Facility (PBF) test assemblies. This input reinforced existing information and/or experience with breached or melted fuel encountered in many other reactors worldwide. (MTR, ETR, EBR-II, DFR, PRTR, BN-10 & BOR-60, Windscale, Hanford-K, etc.)

Experience with removal of damaged cores was reviewed in NUREG-0683, Volume 1, "Final Programmatic Environmental Impact Statement related to decontamination and disposal of radioactive waste resulting from March 28, 1979, accident, Three Mile Island Nuclear Station, Unit 2," March 1982.

The following summary provides a perspective on handling of failed fuel where significant core damage occurred.

Canadian NRX Reactor. All undamaged uranium fuel rods were withdrawn from the reactor. Many others were fused together. An attempt to pull out the defective fuel rods was unsuccessful because of breakage. The tops of reactor concrete shields and top thermal shield were removed to provide greater access to the fuel rods. Three fuel rods were pulled. The third rod being removed broke and allowed radioactive sludge to fall into the basement. The remainder of the fuel rods were grasped near the bottom and pulled out, with the work crew no longer trying to keep the lower portion of the calandria, or vessel, intact. All fuel rods were pulled into a standard fuel rod carrying flask or fuel can and transported to a water trench leading to an extraction plant.

Enrico Fermi I Reactor. On October 5, 1966, the Enrico Fermi I nuclear reactor, in Michigan, sustained partial meltdown of two fuel assemblies as the result of coolant-flow blockage. An estimated 10,000 Ci of fission products was released to the primary coolant and reactor cover gas. Special wedging tools of a chisel design were employed to break the bond between the two fused elements, and the elements were removed from the core for examination.

Canadian NRU Reactor. Following removal of the fuel rod fragments, a large cleanup effort was organized. An emergency headquarters was set up with CCTV and public address systems to the reactor hall, and an emergency change room was established. A vacuum system with a cyclone-type sand filter to prevent clogging was used to remove residual fuel debris.

Chemical decontamination experience to remove fuel failure debris, including fuel fragments, is very limited. Only one reactor, the Plutonium Recycle Test Reactor (PRTR), at Hanford, Washington, has undergone such a decontamination. The PRTR reactor system was decontaminated in 1962 after the failure of an experimental fuel element. Radiation levels were up to 250 R/hr at low spots. This experience demonstrated the quality jump in requirements to handle a melted fuel assembly compared to an assembly with a number of leaky rods.

There has been little experience with removing fuel debris from large reactors; however, tests have been conducted on a number of loops and pilot plants. Nitric acid was applied alternately with sodium hydroxide at the decontamination of the Homogeneous Reactor Experiment at Oak Ridge National Laboratory.

General Conclusion

The surveys cited above, of experience in handling failed fuel did not result in findings significant to the planning for removal of TMI-2 fuel. One observation is made that the level, types, and quality of photographs, audio and other documentation for TMI-2 failed fuel handling should receive priority.

D. Generalized Criticality Studies

1. Introduction

As stated in Section IIB1, TAAG was requested to assist in the preparation of a safety analysis to support the rod motion tests. The principal safety consideration involved with these tests is the possibility that mechanism/leadscrew/spider/control rod motion could result in core fuel rearrangement, which in turn could result in reactivity increases.

Shortly after the 1979 accident several independent studies were made to evaluate possible reactivity levels of the core based upon conservative estimates of core damage. These studies were made by Babcock and Wilcox (B&W), reported in the TMI-2 Criticality Evaluation Notebook, J. J. Woods and M. R. Gudorf. The Nuclear Regulatory Commission reported in A Further Evaluation of the Risk of Criticality at TMI-2, R. DiSalvo, et al., Oak Ridge National Laboratory reported in

TMI-2 Criticality Analysis - Parametric Studies and Overall

Results, R. M. Westfall, et al., and others. The results of these calculations indicated that primary coolant system boron concentrations in the range of 3000 to 3500 ppm would assure reactor shutdown. The conservatism inherent in these calculations was demonstrated by the fact that boron levels in the TMI-2 RCS ranged between 1150 ppm and 2250 ppm shortly after the accident without any noticable adverse effect.

2. Current Status of Predicted Shutdown Calculations

Recognizing that the possibility of core displacement exists as a result of the rod motion testing, TAAG members met on January 20, 1982 with GPU personnel to review the current status of core shutdown predictions. During this meeting, it was concluded that with the current undisturbed core conditions and boron concentrations, an adequate reactivity shutdown margin exists. However, a review of the assumptions used in the existing reactivity calculations identified some assumptions which could potentially effect calculated reactivity levels. Specific examples are: 1) the assumption that no core structural material was present. It is possible that the presence of zirconium would displace boron and result in a small decrease in shutdown margin, 2) the assumption that the core fuel was in the form of discrete pellets may not be limiting. A mixture of fuel pellets and fuel fines may result in higher reactivity levels. Subsequent discussions with B&W

and Bechtel Northern confirmed that recent calculations confirm the need to evaluate the effect of these additional second order effects on reactor shutdown and that some of these studies are underway. These B&W/Bechtel Northern studies are directed towards showing that shutdown is assured with existing boron concentrations even under extremely unlikely conditions of core rearrangement.

3. Limiting Worst Case Shutdown Study

During the above TAAG/GPU meeting it was agreed that the optimum approach to supporting the rod motion testing is to show, if possible, that core shutdown is assured regardless of the core geometry assumed and utilizing a combination of all worst case assumptions. If shutdown could be demonstrated under these conditions -- core rearrangement would not constitute a criticality consideration and, hence, minimal special safety precautions would be required during the test. In order to support the performance of such a limiting calculation, a table of parameters that should be included in the study was prepared. It included the following variables:

VARIABLES FOR WORST-CASE CRITICALITY

<u>Parameter</u>	<u>Potential Choices</u>
Fuel	UO ₂ ; U-235; PuO ₂ ; Pu-239/Pu-241; and/or UO ₂ /PuO ₂
Pellet Size	Range: Dissolved fuel, fractions of pellets, whole pellets, fused pellets (larger than fabricated).
Moderator	A. Water and soluble boron, with or without diluents. B. Steam (pure water), transition state to a dry core.
Diluent in Moderator	Inconel, Zircaloy, Zirc Oxide, Stainless Steel, etc., and/or fuel fines (from the Fuel Choices).
Burnup	Range: 0 MWD/MTU - to - maximum localized core burnup.
Fuel Enrichment	Dependent on fuel/burnup choice (example: highest enrichment fabricated, with statistical uncertainty applied).
Fuel Cell Conditions	Damaged fuel cell (fuel/moderator, with or without diluents) for various temperatures, boron concentrations, fuel packing fractions, and fuel sizes.
Moderator Temperatures	Range: 50°F (Tech Spec lower limit)-to-212°F for unvoided moderator). Note that this is the range to be reviewed where intermediate temperatures must also be considered, especially around the expected operating moderator temperature (approximately 90°F).
Moderator Chemistry	Review chemicals existing in coolant and other locations, and pH of moderator to determine most "reactive" choice.
Neutron Poisons	Only soluble boron; no control rods, no burnable poisons, no structural material which would cause reactivity reduction. As stated, diluents of structural material may cause positive reactivity effects, but larger volumes of structural material would cause a reactivity reduction.
Geometry	Range: Finite array (slab, sphere, cylinder) with worst credible neutron reflector (water, sand, concrete, etc.)-to-infinite array (heterogeneous/homogenous).

VARIABLES FOR WORST-CASE CRITICALITY

<u>Parameter</u>	<u>Potential Choices</u>
Neutron Leakage	Equivalent boron concentration reduction should be identified per neutron leakage chosen in cases where credit for neutron leakage has not been taken and could have been.
Computer Code(s)	Must be defensible for the conditions analyzed.

Following the TAAG/GPU meeting a telephone meeting was held between TAAG, GPU, Bechtel Northern, and B&W. As a result of both of these meetings a TAAG recommendation was forwarded to GPU. This recommendation was that B&W's current studies be expanded to include a limit analysis to support the preparation of the safety analysis for the rod motion tests. GPU has instructed Bechtel Northern to have these studies incorporated into the existing B&W studies. The present schedule for completing the rod motion safety analysis requires that these criticality studies be completed by April 1, 1982.

E. Canal Water

According to the present plan the reactor coolant system water will be processed to remove the bulk of the fission product activity remaining in that water, before work on the reactor vessel is initiated. It is anticipated that the concentrations of Cs^{137} and Sr^{90} can be reduced to $< 1 \text{ Ci/ml}$. During and following this processing, data should be obtained that will provide a much more accurate measure of the leach rate of fission products from the undisturbed fuel in its present state. This will provide a minimum source term for the cleanup system used during fuel removal. The leach rate will probably increase when fuel removal is initiated, and there is great disagreement about the extent of the increase. Source term estimates vary from around 1 Ci/day (the present rate) to perhaps 1000 Ci/day .

At some point the canal will be filled and in communication with the reactor coolant water. Continued processing of this water is required for two primary purposes: 1) to maintain acceptably low concentrations of radionuclides, whether dissolved or suspended, so that radiation from this source is not an important hindrance to carrying out the work, and 2) to maintain adequate water clarity to permit the required underwater manipulation of tools. The methods of water processing generally include two operations - filtration to remove particulates and ion exchange to remove dissolved ionic material along with some residual particulates.

Several different types of filters have been used successfully, including cartridge, precoat, powdex and deep-bed. There is one constraint not ordinarily encountered, namely criticality because of the possible presence of very large quantities of small fuel particles. This might limit the diameter of filters or require poisoning or dilution (as in deep bed filters). Experience with damaged fuel and otherwise suggests that the minimum flow rate should be around 250 gpm. Since much more particulate material is anticipated here, provision for greater capacity is recommended. Since there may be a large amount of fuel collected in the filters radiation levels may be far higher than normal; accordingly, the entire system must be designed with proper consideration for shielding, transfer, and transport. It may be noted that some sort of water vacuum cleaner is frequently considered for use, and that water will require similar filtration.

Ion exchange for removal of ionic radionuclides commonly uses organic resins for demineralization. In this case such a method is very inefficient because of the high concentration of boron that must be maintained for criticality control and the related quantity of sodium ion required for pH control (by partially neutralizing the boric acid). With such water organic ion exchangers are quite ineffective for cesium removal. The use of one or more zeolites, as used presently in the Submerged Demineralizer System for clean up of water in the Reactor Building, should be effective. The required throughput of the ion exchange system (F gpm) is

approximately determined by the leach rate (L Ci/day) of radionuclides from the fuel and the maximum acceptable concentration (C_p Ci/ml) in the fuel; $F = 0.18$ L/C.

The existing fuel pool cleanup system (cartridge filters and one 21-cubic foot demineralizer with 180 gpm flow rate capacity) is probably not suitable because of inadequate shielding and the likelihood that zeolites cannot be transferred hydraulically into and out of the "demineralizer". If capacity can be increased sufficiently the Submerged Demineralizer System might be used after appropriate modification; otherwise, a larger capacity system similar to it can be installed.

It is essential that adequate capacity be available, for both filtration and ion exchange. Since the capacity requirement is not well defined a modular system capable of expansion may be preferred. A second important point is that radiation levels may be very high for both filters and ion exchangers, so appropriate shielding, provision for connection and disconnection, and for transport must be available.

The TAAG suggestion is that the canal water system design be addressed early in the defueling schedule and that the installation and operation of that system be included in the Defueling Plan. Further detailed investigation of the canal water system by TAAG is intended.

F. Loop Survey

1. Description

With varying estimates of the amount of core (fuel) deterioration, concern has been expressed that fuel, in various forms, has migrated into the reactor coolant system and other support systems. Determination of the amount and location of this fuel ultimately is necessary to 1) assess possible criticality conditions in low points, dead legs and components such as the steam generator; 2) determine the extent of effort required to clean-up these systems; and 3) aid in radiation level predictions after the reactor coolant system has been cleaned up. The IAAG identified as an additional potential benefit that early assessment of fuel location and amount could aid in the overall appraisal of core damage.

2. Equipment

Discussions were held with EG&G, SAI, Bechtel and GPUJ personnel to determine the efforts underway. GEND 018 "Nondestructive Techniques for Assaying Fuel Debris in Piping at Three Mile Island Unit 2" was also reviewed. This document discussed several possible nondestructive techniques for fuel assessment which included a) acoustics, b) nuclear radiation,

c) thermal radiation, d) optical and e) pulsed eddy current. It was the conclusion of GEND 018 and a brief study made by one of the TAAG members that nuclear radiation (gamma ray spectrometry) was the best method for determining fuel location and amount outside the reactor vessel. A brief discussion of the equipment currently being used or under discussion follows:

a. Collimated gamma ray spectrometer.

Two pieces of equipment which fall into this category are the EG&G and SAI systems. The SAI system is currently being used for "in-containment" analysis work which is discussed below. This system consists of a Ge(Li) detector with its associated cryogenic supply and preamp, housed in a collimated shield and mounted on a wheeled cart. This unit is "portable" in so much that it can be wheeled around but it does weigh approximately 600 lbs. A future model is being discussed which weighs approximately 400 lbs. with the added feature of remote operation. Electronics, including the multichannel analyzer, data display, data recording devices and power supplies are located outside the containment and connected via cable to the in-containment equipment.

Although this equipment is "collimated", the angle of view is neither adjustable nor very narrow so that viewing specific components at a distance is not easily accomplished. It is noted that the addition of this capability would provide more versatility particularly in an environment such as the containment where there are physical and radiation restrictions in getting the detection equipment near the components of interest.

b. Portable, non or lightly collimated gamma ray spectrometry devices.

Although not currently being used or specifically planned for use, more portable devices have been discussed which offer the opportunity to 1) place gamma ray spectrometry devices in space restricted areas and 2) provide the ability to take more data points with less exposure.

The first of these units, discussed by SAI, is an enhanced cadmium telluride detector attached to essentially the same type of electronics discussed in paragraph a. above. Since cooling is not required and only a small amount of shielding is required for enhancement of sensitivity to the Cerium¹⁴⁴, the probe becomes much smaller. The other device discussed consists of a small, portable Ge(Li) detector (with cryostat), a small shield and a portable battery powered multichannel analyzer. (This is discussed in somewhat

more detail in a memorandum presented to the TAAG on February 5, 1982 from General Dynamics/Reactor Plant Services.)

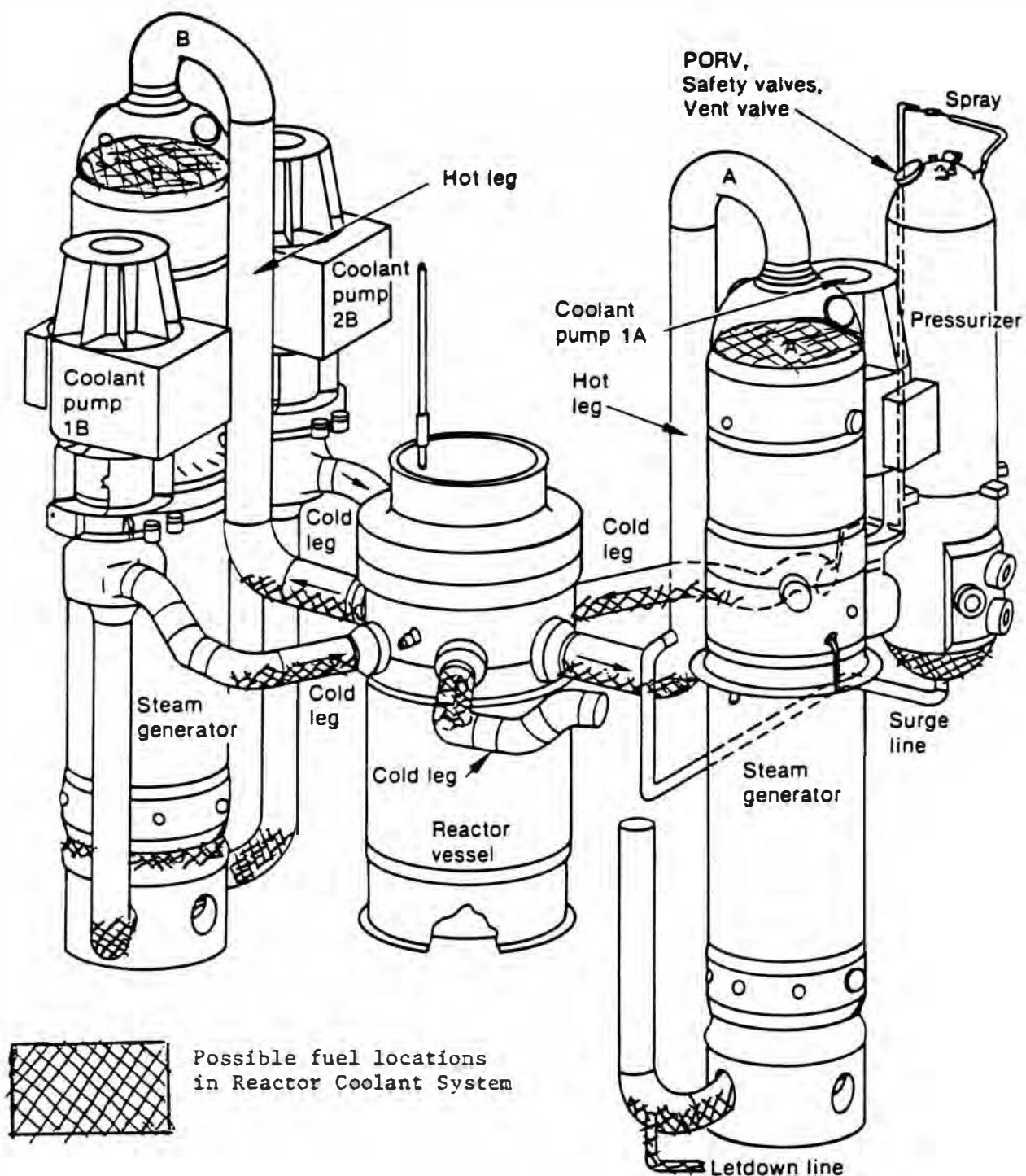
It is noted that during this investigation, several other pieces of equipment for performing analysis were found which had not been previously identified. Concern is expressed that perhaps more information is available which could lead to better equipment than is presently being contemplated or to a reduction in R&D costs associated with new equipment development.

3. Trials

Data obtained to date essentially has been focused on decontamination and not on finding fuel; although it is understood that an EPRI task group is addressing the fuel location issue. The equipment (the SAI collimated detector) has proven its capability to accurately analyze contamination on floor and wall surfaces as well as giving data as to radioactivity content of items such as containment coolers and selected tanks outside the containment. The equipment has also been able to quantify fuel (via Cerium¹⁴⁴) in the letdown filters as well as Reactor Coolant Bleed Tanks "A" & "C".

4. Procedure/Conclusion

It has been hypothesized (GEND 018) that fuel exists throughout the reactor coolant system and specifically in shaded areas shown in Figure V-7. At this time, due to high radiation levels within the "D" ring, it is unrealistic to attempt to directly analyze for the presence of fuel in any location within the "D" ring with possible exception of the top of the steam generators and a general view of the area within the "D" ring. Although the data obtained from the effort would be rather gross in nature and difficult to accurately interpret, it would give an idea as to whether or not much fuel exists within the "D" ring. Exposure estimate for this effort, including installation of a small portable crane is approximately 3 man-rem and the effort is therefore recommended.



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FIGURE V-7
POSSIBLE FUEL LOCATIONS

VI FUTURE EFFORTS

In the course of this investigation, technical problem areas came to light - areas of various types and magnitudes which might be the subject of future investigations. The group deliberated what areas warranted attention - viewed from their perspective. The areas for investigation were also discussed with GPU Nuclear management. At the conclusion of the discussions a suggested future program was developed and is reported here.

Items considered as efforts which need further attention included:

- 1) Procedures and equipment required for disposal of purification system demineralizer resin
- 2) Corrosion of purification system demineralizers and potential radiation hazards
- 3) Exposure estimates for defueling
- 4) Exposure estimates for decontamination
- 5) RCS water clean-up and quality
- 6) Systems and building decontamination chemical solutions

- 7) Systems and building decontamination processes and procedures
- 8) Long or short term planning of TMI-2 cleanup
- 9) Design of protective clothing
- 10) Debate about assumptions in SAI analysis of radioactive sources
- 11) Follow-up on "prompt damage appraisal" items
- 12) Technical management problems
- 13) Staffing of defueling operations
- 14) Facilities for defueling operations
- 15) Method and route of fuel removal from containment
- 16) Reactor vessel head removal procedure and equipment
- 17) Corrosion of instrument tubes on bottom of reactor vessel
- 18) Transfer of residual water from containment sump
- 19) General support to GPU in areas of criticality, chemistry and materials

Of the items on the preceeding list, three key activities appeared to warrant early attention. Others may warrant later attention.

A. Defueling Plan

Follow-up on the items developed during the first period of investigation.

Examples:

1. Rod motion test including completion of the safety report.
2. "Quick Look" into the reactor including procedure development.
3. Further investigation of the equipment and procedures for Reactor Vessel head removal.

B. Radiation Exposure Considerations

Examples:

1. Clean-up processing of RCS water.
2. Appraisal of SAI estimates of source levels.
3. Efforts being made to improve protective clothing and breathing apparatus.



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Deacidified using the Bookkeeper process.
Neutralizing agent: Magnesium Oxide
Treatment Date: Feb. 2007

Preservation Technologies
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