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Mr. Paul Leech
U. S. Nuclear Regulatory Commission
Environmental Projects Branch No. 2
Mail Stop P-522
Washington, D.C. 20555

Dear Mr. Leech:

Enclosed please find one copy of the Planning Study for Phase II of the TMI-II Recovery Operation. This study includes the activities associated with reactor internals disassembly, fuel removal, and system cleanup.

Sincerely,

L. J. Lehman, Jr.
Licensing Engineer, TMI-II

LJL:hah

Enclosure

cc: E. D. Fuller (w/o)
G. Mencinski (w/o)

8006130256

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TMI-2 PLANNING STUDY
FOR PHASE II

THREE MILE ISLAND - UNIT 2

PLANNING STUDY

FOR

PHASE II

REACTOR INTERNALS DISASSEMBLY,

FUEL REMOVAL,

AND

SYSTEM CLEANUP

BECHTEL POWER CORPORATION
MAY 28, 1980

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1.0 INTRODUCTION

1.1 Scope of Study

This Phase II Planning Study for TMI-2 Reactor Internals Disassembly, Fuel Removal, and System Cleanup was commissioned as a follow-on to the TMI-2 Initial Planning Study, Containment Entry and Decontamination, dated July 2, 1979. The Initial Planning Study (called Phase I) presented engineering concepts to develop plans for placing the containment in a configuration where reactor vessel head removal could begin. The purpose of this study is to summarize the planning, technical considerations, alternatives, procedural changes, and sequence of events for:

- examining the reactor core,
- opening the reactor pressure vessel,
- removing the reactor vessel internals,
- removing the fuel, and
- decontaminating the reactor coolant system.

The study is divided into major sections which describe:

- rigging, handling, and laydown considerations,
- control rod drive mechanism, vessel head, and upper internals removal,
- core inspection and fuel removal,
- reactor coolant system cleanup, and
- service systems and facilities required for this phase of the recovery.

This study was performed by Bechtel Power Corporation and Babcock & Wilcox (B&W). It will be followed by a Phase III report which will address the requalification and reconstruction necessary to return TMI-2 to commercial service.

1.2 Bases for Study

For the purposes of this study it is assumed that the containment decontamination (Phase I) has been completed to the point that Phase I activities will not interfere with Phase II activities. Decontamination activities should proceed from the upper levels of the containment downward and should be complete at least down to and including the 305' elevation. Table 1.2-1 presents the radiation levels assumed for various work areas in the containment during Phase II. Those levels should not be misconstrued to be sufficient objectives for defining completion of Phase I or limiting conditions prior to initiating Phase II.

Each section of the report contains recommendations and discussions which are based to varying degrees on factual data, inferred information, and supposition. Because the condition of the TMI-2 core is not known at this time and various ongoing study groups have produced a consensus, no attempt has been made to address all possible core conditions. For the purposes of this study an attempt is made to provide a reference or base case from which more detailed plans can be developed as knowledge of the core condition improves.

Some information is presented in terms of alternatives which should be considered depending on the actual conditions found during the recovery. Several issues or subject areas are identified where decisions are needed reasonably soon so as not to cause schedule impacts during the course of recovery. Where these issues are identified, alternative decisions and the impact of each are included, with a recommendation as to what course to pursue.

This study is based on available information, which is limited at this time. As more information is obtained or developed, many assumptions which are made can be refined to better reflect conditions expected during reactor disassembly. This study should therefore be updated during containment decontamination if the plans and methods presented here can be improved by incorporating the then available data.

1.3 Summary

This Phase II study addresses the prerequisites for proceeding with defueling and reactor coolant system cleanup and the proposed approach to those operations. Also discussed are the anticipated additions and/or modifications required to structures and equipment to support these activities.

Tables 1.3-1 through 1.3-5 summarizes several of the significant items and issues that will require consideration or resolution during the course of reactor coolant system (RCS) disassembly and fuel removal. Table 1.3-1 identifies the major uncertainties that may have a major impact on technical planning for certain activities and provides a suggested approach for resolution. Table 1.3-2 lists significant decisions which must be made to support RCS disassembly. Table 1.3-3 discusses research activities which could be performed to support the recovery as well as gain information which could be useful in generic research. Table 1.3-4 summarizes the extraordinary technical problems which are anticipated. Lastly, Table 1.3-5 provides a sampling of the recommendations found in this planning study which should be considered to accommodate RCS disassembly and fuel removal.

The rigging, handling, and laydown considerations addressed in Chapter 2.0 include a discussion of the polar crane refurbishment and points out the need for additional hoisting capabilities during this phase. The preparations which are expected to be necessary for the reactor vessel head and upper internals removal and laydown are presented. Similar discussions on lower internals handling are also presented. The possibility of fuel cask handling inside containment is discussed.

Chapter 3.0 on control rod drive mechanism (CRDM), reactor vessel head, and internals removal contains discussions of shielding provisions, general preparations and precautions, and plans for CRDM unlatching, head and upper and lower plenum removal. Studies will be conducted on a case-by-case basis to determine the viability of in-place decontamination of major components. One objective of component decontamination is to minimize shielding required for these components.

Special attention should be given to Chapter 4.0 on core inspection and fuel removal. This chapter addresses many of the issues which may be most sensitive and procedures which may present the most difficult technical problems. This chapter presents discussions of core inspection, vessel specimen holder removal, fuel removal, fuel canning and transfer operations, and special tooling and equipment which may be necessary for each of these tasks. There is also a discussion of fuel storage and ultimate disposition considerations and the impact of these issues on decisions regarding rigging and handling, spent fuel pool reclamation, and fuel canning.

Chapter 5.0 presents information concerning coolant purification, steam generator cleaning, mechanical and chemical cleaning of the reactor coolant system, and decontamination of the reactor vessel head and internals.

Chapter 6.0 discusses the service systems and facilities which will be required to support fuel removal and system cleanup, including remote viewing equipment, radiation monitoring instruments, communications, lighting, and decontamination systems.

The detailed procedures for removing the head, lifting the upper internals, defueling, lifting the lower internals, and RCS decontamination all will depend on the actual conditions encountered. These procedures should therefore have numerous hold points at which time all the available information may be evaluated prior to proceeding. They should be flexible enough to allow for the unexpected, and they must include consideration of a number of available options prior to proceeding beyond some point from which return to the original state could not occur.

TABLE 1.2-1

ASSUMED RADIATION LEVELS INSIDE
CONTAINMENT DURING PHASE II

Location	exposure Dose Rate
Top of Service Structure (after shielding)	500. mR/hr
Reactor Building General Background	50 mR/hr
Refueling Canal with Shielding on Service Structure	<1 R/hr

Note: Dose rates are not intended as objectives for defining the completion of decontamination or as limiting conditions prior to initiating Phase II activities.

MORE IMPORTANT:
WHAT ARE THEIR
PERSONNEL EXPOSURE
CRITERIA, WHAT DO
THEY INTEND TO ALLOW
PER WEEK, QUARTER, YEAR?

TABLE 1.3-1

MAJOR UNCERTAINTIES AND
APPROACH TO RESOLUTION

Uncertainty: Reactor vessel head radiation levels. These levels are difficult to predict and may require removal or redesign of the existing head storage stand to allow for more efficient head decontamination, inspection, and shielding.

Approach to Resolve

1. Complete a detailed study of expected contamination levels on the inside surface of the reactor vessel head.

2. After completion of containment decontamination and decontamination of the outer surfaces of the reactor vessel head and closure, measurements should be taken of the radiation levels through the head. Adjust these measured levels with the predicted contribution from other sources (upper internals and the reactor coolant).

Uncertainty: Radiation levels from the steam generator upper tubesheet. These levels could pose a severe radiation hazard if significant amounts of fuel debris are trapped on the upper tubesheet, especially if the water in the steam generator (which is providing some degree of shielding) is removed. For this reason it may be impractical to drain the reactor coolant out of the steam generators and lower the RCS water level prior to head removal.

Approach to Resolve

1. After completion of containment decontamination and initial RCS flushing, perform a radiation survey around the exterior of the upper steam generator to determine the presence of fuel debris.

2. If it is determined that there are significant amounts of fuel debris on the tubesheet, methods should be developed to remove the debris prior to RCS draining.

TABLE 1.3-1 (Cont.)

Uncertainty: Core conditions and impact of conditions on reactor disassembly and defueling operations.

Approach To Resolve

1. Monitor results of on-going study groups assessing core condition.

2. Develop plans for core inspection prior to head removal.

3. Removal of surveillance specimens as soon as possible for detailed metallurgical analysis.

4. Provide a wide range of tool designer experience to accommodate special tooling needs during reactor disassembly and defueling.

Uncertainty: The availability of permanent offsite storage space for damaged TMI-2 fuel.

Approach to Resolve

1. Provide for interim storage of damaged fuel in the onsite spent fuel pool using failed fuel canisters and special storage racks.

2. Monitor the availability of storage space at existing independent spent fuel pools (GE-Morris, Barnwell, West Valley) and the licensing activities for new independent storage facilities.

TABLE 1.3-2

DECISIONS

DECISION 1: Whether to decontaminate the containment down to elevation 282' prior to reactor disassembly. The radiation contribution from the sources around elevation 282' if not decontaminated should be evaluated. ALARA considerations may dictate that decontamination of elevation 282' be deferred until required for reconstruction. If the unit is not to be refurbished for continued operations, decontamination of elevation 282' may be waived or could be done to less stringent criteria pending establishing criteria for decommissioning.

DECISION 2: Whether to decontaminate reactor internals in the reactor pressure vessel when RCS decontamination is performed. The alternatives are to decontaminate the internals in the fuel transfer canal or not decontaminate at all.

TABLE 1.3-3

RESEARCH ACTIVITIES IN
SUPPORT OF RECOVERY

1. Movement of selected control element assemblies (CEAs) could provide some information related to potential fuel damage. This exercise could give an indication of damage not obtainable by visual inspection. For example, if the top and bottom end fittings look good, but CEA won't move, it is a possible indication of significant deformation within and along the length of the fuel assembly. A study should be performed to determine how such an exercise could be performed; what information could be obtained, interpreted, and utilized; and what the risks are associated with CEA travel of 1 or 2 percent of range.

2. The fuel fabricator should be requested to review the in-core instrument data (both power and temperature information is available) and provide some assessment, fuel element by element, of the possible core condition. This assessment, confirmed by a visual assessment after the head and upper plenum have been removed, could be used to aid in developing an optimum defueling plan. Recommendations on maximum allowable forces exerted on individual elements and/or specific locations on an element could also be used as input in designing special damaged-fuel handling tools and canisters.

3. The feasibility of remote visual examination of core components through the use of optical equipment (such as a borescope or fiberoptics) during reactor disassembly and defueling operations.

4. The design of specialized, damaged-fuel and CEA handling tools and damaged-fuel cans should proceed as soon as possible. An effort should be initiated to conceptualize alternative methods and tools for damaged-fuel handling, and then select those alternatives which appear to have the most utility for the situation.

5. The split seal ring designed for Unit 2 may represent an unacceptable potential leakage path from the refueling canal. However, any modifications to the seal must not result in an unacceptably long installation time, since the radiation levels near the reactor vessel head are postulated to be very high. A study to determine the optimum resolution of this potential problem should be completed soon enough that modifications can be made to the seal ring well before its use.

6. B&W should be requested to provide recommendations related to core reactivity control during all RCS disassembly activities such as control rod and axial power shaping rod (APSR) uncoupling, head removal, etc.

Table 1.3-3 (Cont.)

7. B&W should be requested to evaluate the consequences of free fall of an APSR from its existing position to the fully inserted position after uncoupling.

8. The feasibility of underwater reactor vessel head stud detensioning should be examined.

TABLE 1.3-4

SUMMARY OF EXTRAORDINARY
TECHNICAL PROBLEMS

1. A modified reactor vessel head storage stand with lead shielding may exceed the design floor loading under the storage stand.

2. Usable space on elevation 347' is very restricted and may not be sufficient to accommodate laydown and working space requirements.

3. A seal ring which will not leak under any circumstances is required during defueling operations.

4. Possible deformation of the reactor vessel head and plenum could retard RCS disassembly.

5. Fuel accountability for the severely damaged core will require a novel approach or a relaxation of accountability requirements.

6. Methods of criticality control and monitoring must be developed for all RCS disassembly and defueling activities.

TABLE 1.3-5

RECOMMENDATIONS

1. The reactor cavity shield blocks should be removed from the containment.

2. Two 15-ton cherry pickers should be provided inside the containment to assist hoisting and movement of materials. These cherry pickers could be placed across the reactor coolant pump (RCP) supports at elevation 369', one for each D ring.

3. A bridge-type service hoist should be provided on tracks along the D-ring rails to assist in defueling activities.

4. A skid-mounted underwater filter system should be used for fuel transfer canal cleanup.

5. The neutron shield tanks should be removed from the containment.

6. An air control boundary should be provided in lieu of the containment equipment hatch to expedite defueling activities.

7. Fuel canisters should be designed to allow loading from the top or the side.

8. The fuel transfer tube system should be modified to use a cable drive instead of the existing underwater air-motor-powered chain and sprocket drive.

2.0 RIGGING, HANDLING, AND LAYOUT CONSIDERATIONS

2.1 General Considerations

A basic problem that will be encountered while conducting any activities on elevation 347' will be congestion of personnel and equipment. The very limited amount of working and storage space available at this elevation must be carefully factored into the technical planning for the various activities relating to RCS disassembly and fuel removal. Optimizing the flow of personnel and equipment to and from this area will help minimize radiation exposures and schedule delays.

Usable storage and work space will be especially limited at elevation 347' while the reactor vessel head is in its storage stand. Furthermore, Stairway No. 1 may not be available for use due to the proximity to the head. To allow access, the area between the south end of the pool and the floor hatchway should be left clear (see Figure 2.1-1). Enough of the hatchway will have to be left open to allow passage of the largest piece of equipment to be moved between elevations 305' and 347' during defueling. Prior to removal of the head, the need for temporary equipment and materials at elevation 347' should be evaluated. Only items with a specific purpose should be left at elevation 347'. If possible, large bulky items such as the neutron shield tanks and shield blocks should be removed and placed elsewhere within the containment. Any excess scaffolding, cleanup equipment, tools, etc., should be removed to allow room for special tooling equipment and materials needed to support the defueling effort.

2.2 Polar Crane

In preparation for defueling, the polar crane will be used to perform the work items listed below. With the exception of removal of the neutron shield tanks and the head insulation, the polar crane is the only piece of hoisting equipment capable of performing these tasks and will have to be refurbished in order to make the lifts. This listing is not intended to represent the sequential order of events, nor should it be construed to be all inclusive.

2.2.1 Auxiliary Cranes

The auxiliary cranes, as shown on Figure 2.1-1, will be lifted from elevation 305' and placed in position.

2.2.2 Refueling Crane

The refueling crane trolley should be removed from above the pool to an area where it can be decontaminated and refurbished to support normal refueling if the plant is to be put back in operation.

2.2.3 Reactor Coolant Pump (RCP) Motors

The RCP motors will be removed from their supports and lowered to elevation 305' for removal from containment. The RCP motors will be used as the design basis items for handling heavy loads inside the containment. Figures 2.2-1, 2.2-2, and 2.2-3 illustrate that the RCP motors will pass through the containment equipment hatch in an upright position, a requirement due to the design of the thrust bearings. Figure 2.2-4 illustrates the relative size of other large items such as the shield blocks and selected casks, which will pass through the equipment hatch.

2.2.4 Shield Blocks

The shield blocks will be removed from above the refueling pool and placed on top of D rings or lowered to elevation 305' for removal from containment. After the reactor coolant pump motors are removed, it may be desirable to place one or more of the shield blocks across the D rings and use for laydown. However, prior to this removal from the containment is recommended to reduce congestion and provide a less hampered working space.

2.2.5 Refueling Pool Indexing Fixture

The indexing fixture can be removed from the refueling pool and lowered to elevation 305' for removal from containment.

2.2.6 Shielding

Shielding material can be raised from elevation 305' and placed around the service structure and the reactor vessel head storage stand.

2.2.7 Head Insulation

Insulation will be removed from the reactor vessel head. Figure 2.1-1 illustrates two auxiliary cranes in position to assist in this operation. This insulation should be removed and set on elevation 347' in the quickest manner possible to minimize exposure. A method of lifting several panels at one time should be developed.

2.2.8 Neutron Shield Tanks

The neutron shield tanks will be removed from the floor of the refueling canal and lowered to elevation 305' for removal from the containment.

2.2.9 Hatchway

Hatch covers and supports can be installed in the floor hatch at elevation 347' to provide for maximum floor space during defueling (Figure 2.1-1). Part of the floor hatch should remain open to allow the passage of equipment.

2.2.10 Reactor Vessel Head

The service structure and head will be raised and set on the support stand.

2.2.11 Plenum

The plenum will be removed from the reactor vessel and set in the deep end of the refueling pool.

2.2.12 Casks

If it becomes necessary to handle fuel casks in the containment, the casks can be raised from elevation 305' and set in the refueling pool. The fuel casks can be loaded underwater, then removed from the refueling pool and lowered to elevation 305'.

2.2.13 Core Support Assembly

The plenum, if not removed from the containment, will have to be moved from the deep end of the refueling pool to the shallow end. Then the core support assembly can be removed from the reactor vessel and placed in the deep end of the pool.

2.3 Other Hoisting Considerations

Two cherry picker cranes (approximately 15-ton) are proposed to be installed on the shield walls straddling the RCP supports at elevation 369'-6". These cranes can be used during the preparation for fuel removal and cleanup (see Figure 2.1-1). These cranes would serve to raise materials to the 347' level, position materials on this level, assist in rework of the refueling cranes, and in general help with material handling. Cranes in this capacity range are also small enough to be repositioned by the polar crane without much difficulty. After completion of fuel removal and prior to cleanup of the RCS, these cranes can be used to move tools, cleanup materials, insulation, scaffolding, grating, etc., in and out of the steam generator compartments, called the D rings.

The installation and use of an auxiliary service hoist described in Sections 3.2.3 and 4.5.1 should be considered in supporting the defueling effort. This service hoist could be a bridge type, as shown in Figure 3.2-2, or a gantry type which could also be supported by rails installed on the D rings.

2.4 Laydown Considerations

The existing head laydown stand does not provide good access to the underside of the head. Once the head is on the stand it will be difficult to get monitoring or cleanup equipment in place. The head stand will have to support approximately 5,000 pounds of shielding, in addition to the weight of the head and the service structure. Visual inspections, radiation monitoring, decontamination, and nondestructive examination would be planned once the head is in storage. The existing head stand should, therefore, be modified or replaced with a stand designed to provide good access and to support the additional weight.

It may also be desirable to have lead shielding around the stand. A structural evaluation will be required to assure the floor can support the added loads. If necessary, the elevation 347' floor could be braced from below or the head storage stand could be modified to distribute the load across the floor. Bracing the floor under the storage stand could affect drain lines, HVAC ducting, and electrical trays under the elevation 347' slab. The head stand and the modifications will be directly above the containment equipment hatch, which will be the main entrance and exit path for personnel and equipment. For these reasons, distributing the load on the elevation 347' floor would be the preferred approach from a schedule impact standpoint.

The reactor vessel upper internals (plenum) will be removed with the refueling pool flooded. Upon removal from the reactor the plenum will be placed in the deep end of the refueling pool. This will provide maximum shielding and will minimize worker dose. After completing the defueling, the plenum can be moved to the shallow end of the pool and the much longer lower internals (core support assembly) will be placed in the deep end of the pool. The plans for movement and storage of reactor internals will be reassessed as storage space requirements become better defined.

The top of the core support assembly will come out of the water in the refueling pool as it is raised from the reactor and shifted to the deep end of the pool. This may require shielding to be draped over the assembly before it is raised, remote control of the lifting crane, or other means of reducing radiation levels as the lower internals are moved.

2.5 Reclamation of the Spent Fuel Pool

The fuel pool waste storage system, which consists of two 25,000-gallon tanks and four 15,000-gallon tanks, is presently located in spent fuel pool "A". The submerged demineralizer system is currently planned for spent fuel pool "B". Reclamation of both pools will be required as part of the recovery effort.

Reclamation of at least the "A" spent fuel pool is of special importance because of the need to provide storage racks for the damaged core.

2.5.1 Fuel Handling

There are only two means of egress from the containment for spent fuel: through the fuel transfer tubes or through the equipment hatch. The normal means is through the fuel transfer tubes from the refueling canal inside containment to the "A" spent fuel pool for storage prior to shipment offsite. All of the procedures, tools, and the basic containment design are based upon this normal method of handling spent fuel.

The normal method of fuel handling requires storage racks in the "A" spent fuel pool. This is the location of the fuel building end of the fuel transfer tubes and spent fuel storage racks. The "A" spent fuel pool is sufficient for unloading the core, but the "B" spent fuel pool is also required to provide access to the cask loading pit to transfer spent fuel into a shipping cask for shipment offsite. Figure 2.5-1 provides an elevation view of the fuel transfer canal in the containment building and the spent fuel pool in the fuel building.

A second method is available by loading fuel into a spent fuel shipping or transfer cask in the refueling canal inside the containment, with entry and egress of the spent fuel cask through the containment equipment hatch. The normal method is preferable for removing fuel from the containment, since it requires minimum modifications to either the plant or operating procedures.

Since removal and reinstallation of the equipment hatch is a long, involved process, this method would require a significant schedule extension. If it becomes apparent that fuel cask handling through the equipment hatch is likely, it is recommended that an air control door be installed onto the equipment hatch bolt circle to serve as the containment boundary during fuel handling, and to allow quick [illegible] during movement of the cask or other materials. This door would only have to be designed for air control with a slight pressure differential, and not for the containment design pressure which applies only during reactor operation.

2.5.2 In-Containment Spent Fuel Handling

The rigging and handling of a spent fuel shipping or transfer cask inside containment is feasible, but will add significantly to the schedule for defueling the reactor, due to the difficulties and safety considerations associated with rigging and handling of large, massive components inside containment.

Another disadvantage of this method relates to the issue of ultimate disposition and/or reasonably long-term intermediate storage of the damaged spent fuel. Until such time as some decision is made regarding where the fuel is to be shipped, it must remain at the TMI site. The only location presently available for temporary storage of the fuel is either the "A" or "B" spent fuel pools. It is possible that space will be available in the transfer canal for limited storage of fuel (up to 1/6 core) and consideration will be given to the installation of racks for this purpose. Damaged fuel will probably have to be canned before it is moved (see Section 4.5.7). If possible, these cans will be designed to allow movement of the fuel canisters through the transfer tubes. However, if specially configured fuel canisters are required which do not fit the transfer tubes, then removal through the containment hatch using a cask must be considered.

In-containment spent fuel cask handling will require a lift which takes a cask over some part of the Reactor Coolant System. It should be anticipated that a cask drop analysis will be required to show that the consequences of such an accident are acceptable.

2.5.3 Requirements for Spent Fuel Pool Reclamation

It is recommended that a program be initiated to reclaim the "A" pool on a schedule such that it is available for use prior to its scheduled need, that is, when fuel is to be transferred from the containment. The "B" spent fuel pool should also be reclaimed, but it may be deferred if the fuel is to be temporarily stored onsite until fuel shipments are to be made. Such a program should include at least the following items:

- Removal of the existing tank farm in the "A" spent fuel pool

- Inspection and repair of any leaks or other damage which may have occurred to the spent fuel pool liner

- Installation of storage racks suitable for storage of the core pieces in failed fuel cans

- Modification of the fuel transfer tube mechanism to a cable system allowing positive control of fuel transfer through the tube.

- Replacement of the fuel transfer basket to accept a failed fuel canister slightly larger than a fuel assembly.

Reclamation of the "B" spent fuel pool should involve only removal of the Submerged Demineralizer System. The Submerged Demineralizer System will be completely installed and supported in a cage assembly which will rest on the curb and/or floor around the "B" pool and no welding will be done to the pool liner. Since no cutting of welds to the liner plate should be necessary, liner damage is not expected; however an inspection of the liner should be performed.

Section 4.8, Fuel Storage and Shipment, also contains discussions which have a bearing on the issue of spent fuel pool reclamation related to storage considerations.

2.5.4 Effects of Reclamation of the Spent Fuel Pools

Reclamation of the spent fuel pools is needed for fuel handling operations, but it has an impact on radwaste management activities. Reclamation of the pools cannot be initiated until after the tanks in the "A" pool are emptied. This will require the 110,000 gallons of water to be processed and stored. Following reclamation, the pool can be refilled with borated water which meets the water quality specifications for the pool.

Since the objective of reclaiming the "A" spent fuel pool is to allow fuel handling and temporary storage, any structures, modifications, or additions which interfere with the fuel-handling bridge or fuel handling activities will have to be returned to an acceptable configuration.

Reclamation of the spent fuel pools should be accomplished to support fuel handling operations, but the plan should be carefully reviewed to assure that the impact on radwaste management activities is minimized.

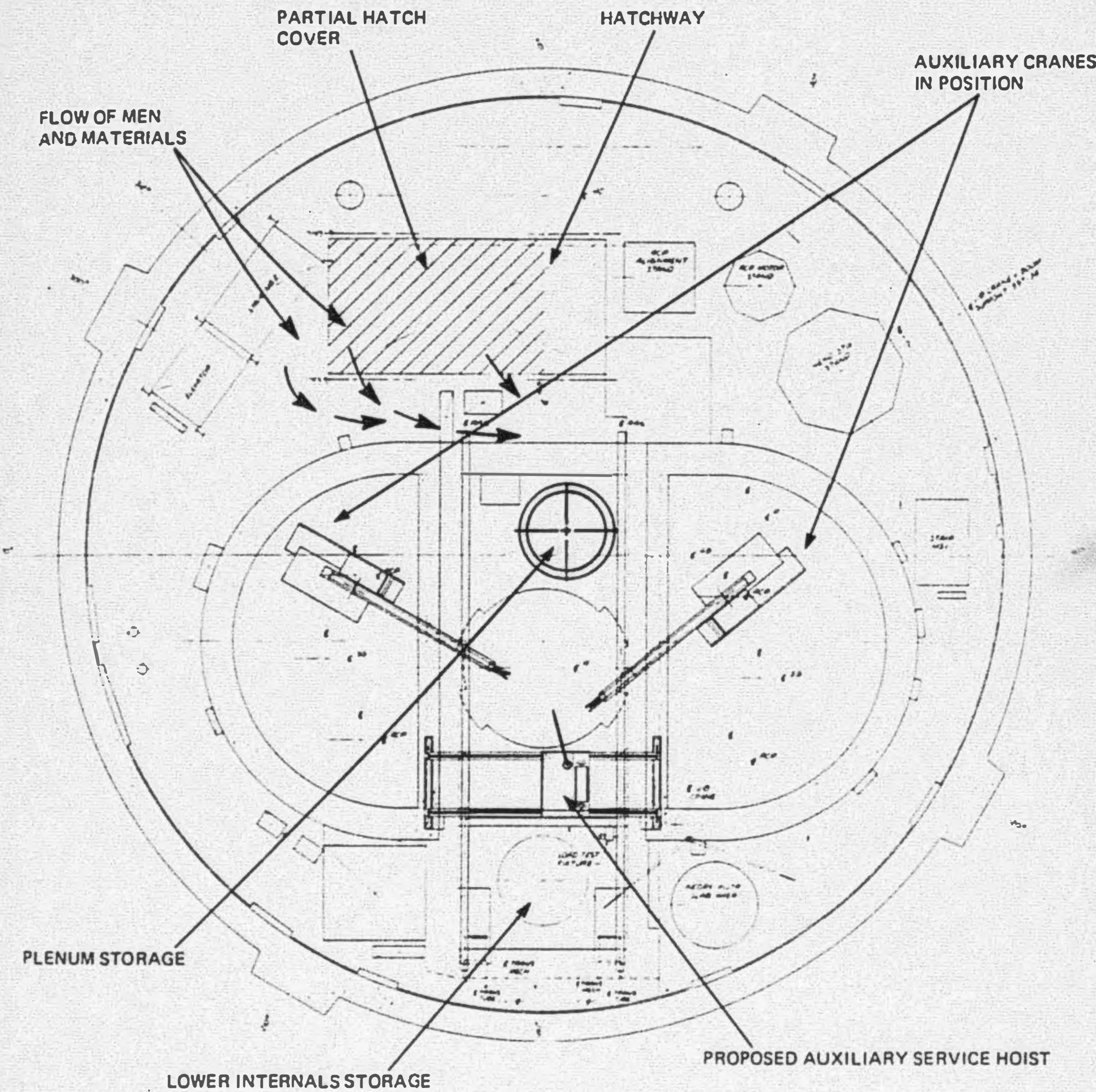


FIGURE 2.1 — 1
TMI-2 CONTAINMENT BUILDING
PLAN VIEW AT EL. 347'-6"

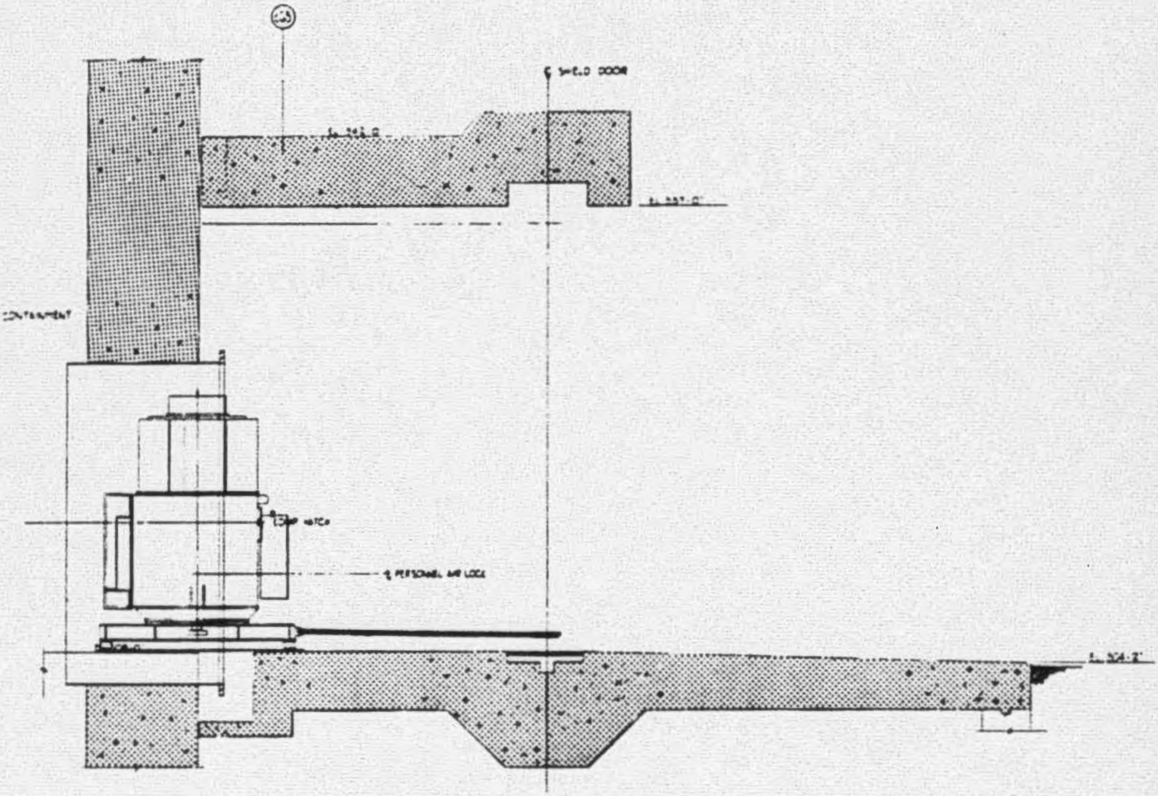


FIGURE 2.2–1
REMOVAL OF RC PUMP MOTOR THROUGH CONTAINMENT EQUIPMENT HATCH

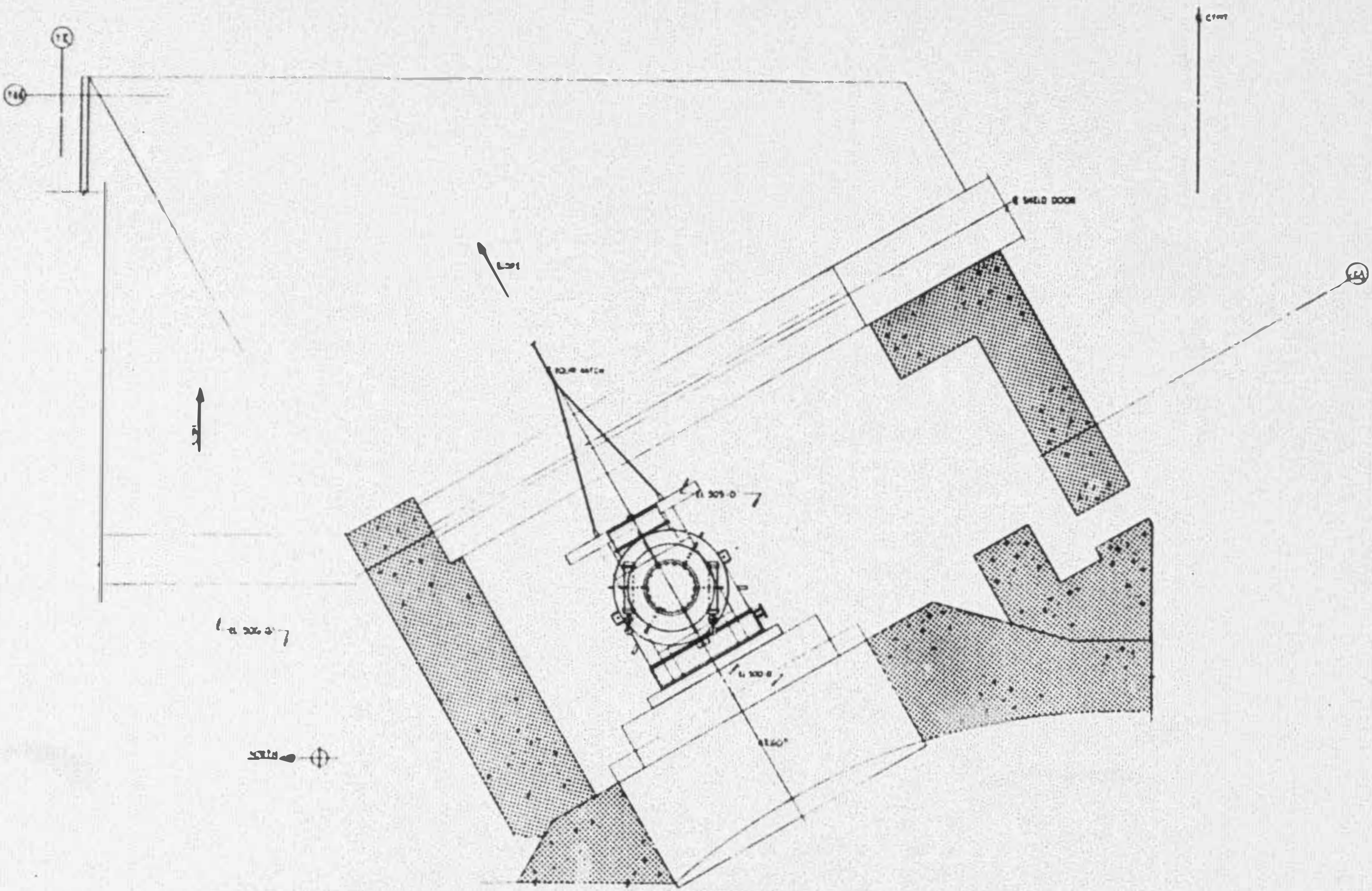


FIGURE 2.2–2
REMOVAL OF RC PUMP MOTOR THROUGH CONTAINMENT EQUIPMENT
HATCH – PLAN VIEW

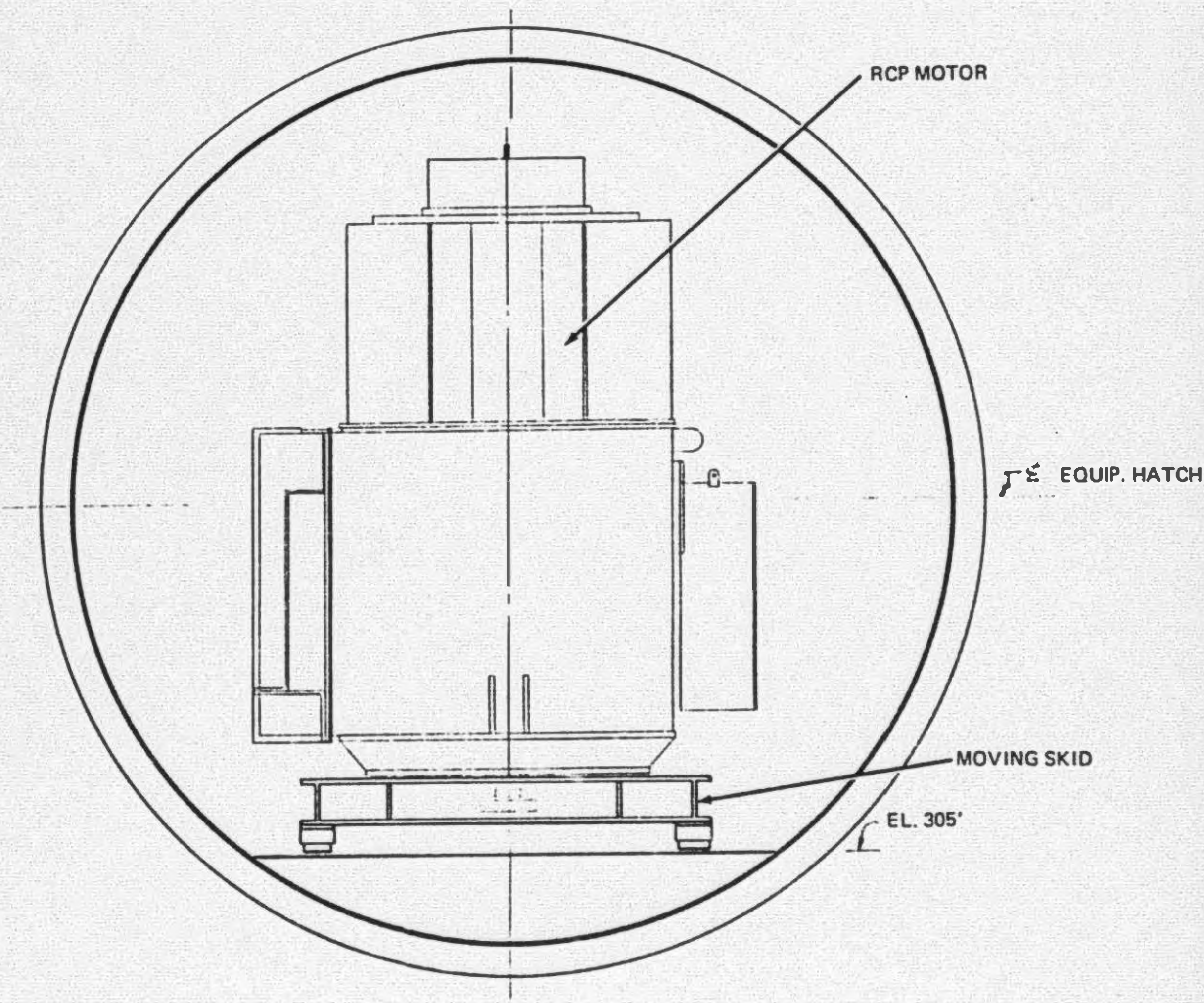


FIGURE 2.2–3
ELEVATION VIEW OF CONTAINMENT EQUIPMENT HATCH WITH RCP MOTOR POSITIONED IN HATCH OPENING

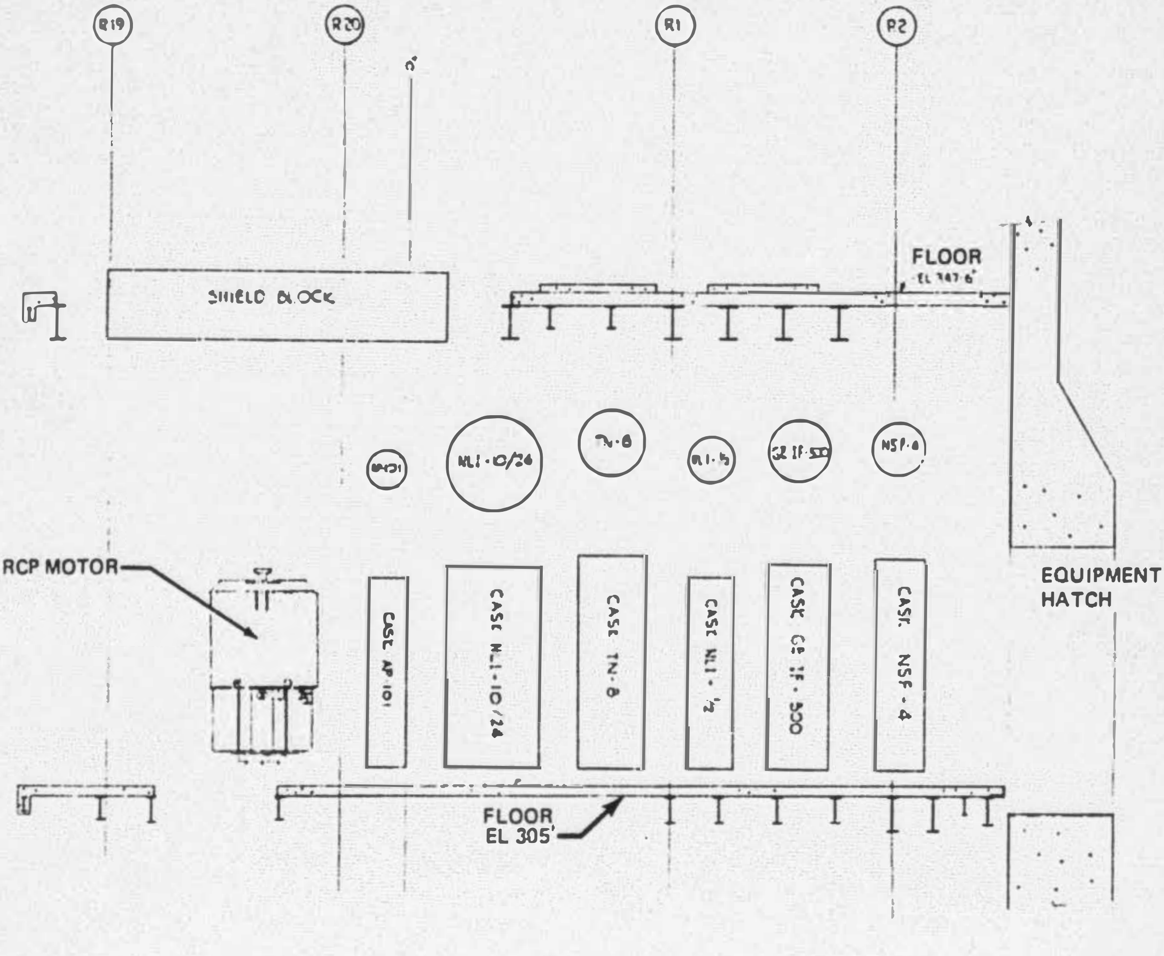


FIGURE 2.2–4
ILLUSTRATION OF RELATIVE SIZES OF LARGE BULKY ITEMS WHICH POTENTIALLY WILL PASS THROUGH EQUIPMENT HATCH

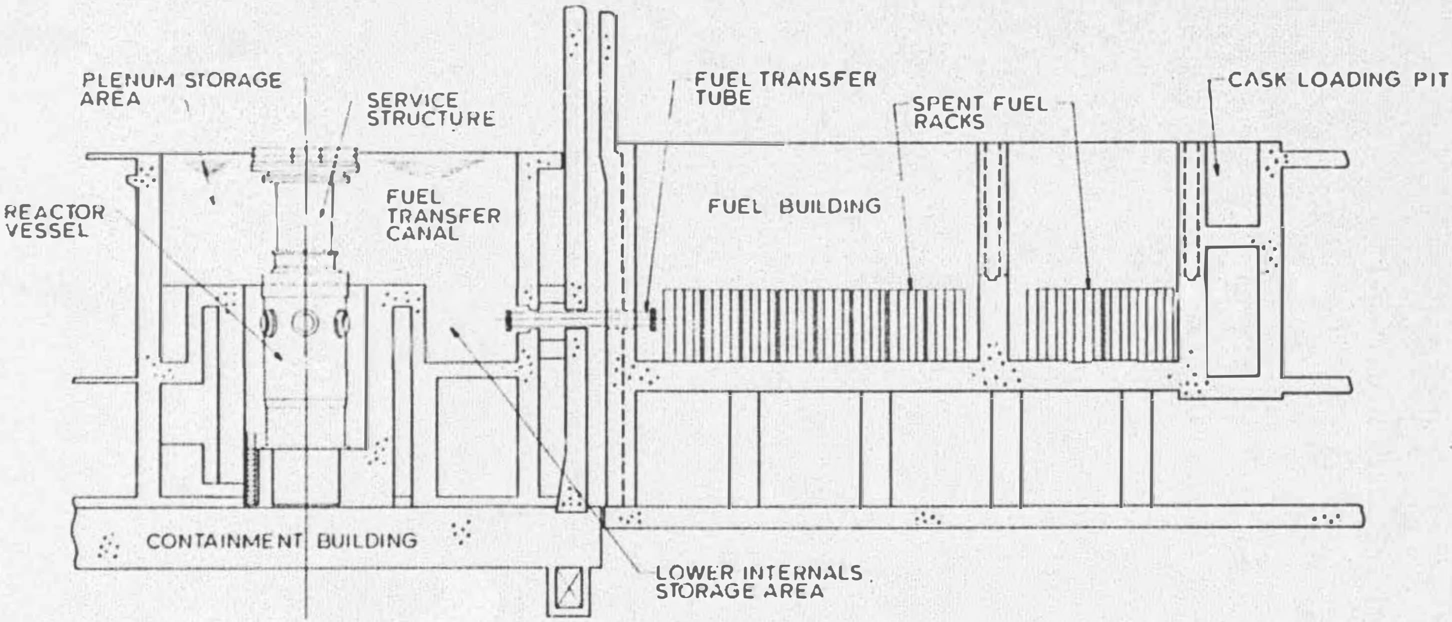


FIGURE 2.5-1
ELEVATION VIEW OF CONTAINMENT AND FUEL HANDLING BUILDING

3.0 REACTOR DISASSEMBLY

3.1 Introduction

This section examines the tasks and proposed methods of removing the reactor vessel head and plenum assembly to expose the fuel and, subsequent to fuel removal, removing the core support assembly (see Figure 3.1-1).

Although radiation levels and other conditions are not specifically known, it can be expected that throughout the reactor coolant system decontamination and defueling efforts the radiological control problems from both the potential spread of contamination and the direct radiation sources will be considerably higher than could be found during normal refueling. Alternate approaches will be discussed where the individual task in the sequence may have to depart from the normal or base approach.

Each subsection that deals with an individual task includes a discussion of the base method and alternates and of expected shielding, training, tooling, and special equipment needs.

3.1.1 Procedures

A complete package of procedures will have to be developed for this work. Many standard refueling procedures can be modified for use, particularly if the base method is feasible. These alternate procedures should be prepared to the extent possible to enable work to proceed promptly should conditions require the alternate methods. It should be recognized that it may not be practicable for the alternate procedures to be fully prepared prior to complete knowledge of conditions.

Although the procedures used will have the usual set of check points and prerequisites, the following three key hold points are thought to be of enough significance to warrant discussion:

- Prior to any axial motion of drives which may be coupled to control components, a hold should be established to allow thorough evaluation of potential core reactivity consequences.

- Prior to RV head lift, a hold should be established to allow thorough evaluation of potential core reactivity consequences as a result of inability to uncouple or inability to verify uncoupling of control assemblies.

- Prior to plenum lift, a hold should be established to allow thorough evaluation of inspections performed through and around the plenum cylinder. During the plenum lift sequence (e.g. 1 foot and 2 feet above flange), additional holds should be established for inspection and evaluation.

3.1.2 Assumptions

The following is a list of assumptions made in developing this plan. As more specific conditions are known, the assumptions may be changed and the methods can be revised accordingly.

- Control rod drives may be inoperable due to stator damage.
- Radiation levels on top of the service structure will range between 5 and 20 R/hr following building cleanup and without temporary shielding.
- Radiation levels on the work platform at the top of the service structure will be about 500 mr/hr with temporary shielding.
- Reactor building general background radiation will be about 50 mr/hr after cleanup.
- Radiation levels in the canal with shielding on the service structure will be about 500 mr/hr.
- Performance of all work will require respiratory protection.
- Components will be treated as reusable until proven otherwise.
- General containment cleanup is completed prior to the start of actual disassembly work so that access to work areas is as unencumbered as is possible.
- The polar crane will be available prior to the start of work. Decontamination and refurbishment of the polar crane is discussed in Section 9.7 of the Initial (Phase I) Planning Study.
- Capability is available for remote television supervision of work in the fuel transfer canal and on the elevation 347' operating floor.

3.1.3 Base Methods and Alternatives

In Sections 3.3 through 3.7, the base methods discussed closely parallel normal refueling operations except that consideration is given to:

- Radiation exposure reduction through time reduction, shielding, and distance from the source.
- Equipment deficiencies which have a high probability of occurence due to exposure to abnormal operating and/or environmental conditions.

For the base method, work on CRDM uncoupling will be done at the top of the service structure from a shielded platform resting on the top of the motor tubes. Local shielding should be provided on the seismic plates. Most work will be done with normal tooling for this operation except for the APSRs which are partially withdrawn (20 percent) and will require special treatment.

Work on the RV head closure area is expected to be done locally in the fuel transfer canal for the base plan. Shielding and tooling changes are suggested for radiation exposure reduction.

Rigging for all major lifts is by the normal methods except for special leveling devices used for a portion of the plenum assembly lift. Execution of the lifts should be modified from the normal means to allow remote observation and inspections and to provide water shielding. Detailed inspections are suggested during these lifts.

The alternate methods discussed are intended to accommodate high radiation levels or significant equipment deficiencies. These methods generally require substantial procedure and tooling changes. In some cases, concept and tooling development work is needed to validate the alternate procedure.

Also included in these sections are:

- Special considerations for local shielding to minimize the effect of anticipated high radiation level areas.

- Specific training recommendations for the activity considered in each subsection.

- Special tooling and equipment needs for both the base methods and the alternate methods.

Some facilities modifications are suggested to provide for shielding and to enhance operations. Training facilities are suggested for mockup of non-standard operations such as uncoupling axial power shaping rods and all operations for personnel and supervision training.

3.1.4 Plant Conditions

This section discusses expected conditions and operational control of the reactor coolant system prior to and during the reactor disassembly work. Also included is chemistry control.

3.1.4.1 RCS Operating Conditions and Control

At the time reactor disassembly is anticipated, the RCS is expected to be at near normal refueling temperatures, 100-125 F, and at or slightly above atmospheric pressure. RCS inventory processing should be completed, thus reducing RCS coolant activity from current conditions to near normal refueling conditions even though some continuous leaching of fission products from the damaged fuel and from plateout in the RCS is expected. Reactor coolant bleed holdup tank (BHUT) (or other alternate tankage) capacity should be available to drain the RCS to a level below the reactor vessel flange. The borated water storage tank (BWST) should be filled to provide inventory to fill the fuel transfer canal following RV bead removal.

To prepare for reactor disassembly, the RCS must be depressurized and the excess inventory drained via the makeup system letdown line through MU-V8 to the bleed holdup tanks. If, due to blockage, this path is not available, an alternate path from the RCS through the decay heat and spent fuel pool cooling systems to the bleed holdup tanks is available. Following (or during) RV head removal, the fuel transfer canal should be filled from the BWST via the spent fuel pool cooling system. When the fuel transfer canal is flooded, the canal water must have the same boron (boron acid) concentration as the water in the reactor vessel to ensure that the core is maintained in a subcritical condition.

Continuous circulation of reactor coolant through the core can be maintained by the mini decay heat removal system or, if processing of RCS inventory is complete, by the normal decay heat removal system.

3.1.4.2 Chemistry Control

Steam Generators (Secondary Side)

When preparations are made for reactor disassembly, the steam generators are expected to be isolated since the mini decay heat removal system should be operating to remove decay heat. The chemistry conditions on the isolated secondary side of the steam generators should be in accordance with those listed in the B&W TMI-2 Recovery Project Water Chemistry Manual, which are listed in Table 3.1-1.

Conditions should be maintained with ammonia and hydrazine under a nitrogen blanket. These layup conditions should be maintained at all times except when it is necessary for a steam generator to be empty for maintenance and inspection operations. After maintenance and inspection the wet layup condition should be restored with water containing ammonia and hydrazine and a nitrogen blanket.

Wet layup chemistry conditions should be provided for the feedwater system as listed in the B&W TMI-2 Recovery Project Chemistry Manual, which are listed in Table 3.1-2.

Reactor Coolant System

Prior to removal of the reactor vessel head, the RCS chemistry conditions should be maintained as listed in the B&W TMI-2 Project Water Chemistry Manual, which are listed in Table 3.1-3.

3.1.4.3 Reactor Component Conditions

The major elements of the accident which may have caused abnormal physical conditions in the reactor coolant system are the prolonged heating period during core uncovering followed by several rapid cooling transients caused by refill of the reactor vessel (by the turning on of the makeup pumps) and the operation of the RC pumps, and prolonged exposure of the components to the humid reactor building atmosphere for the months which followed the incident.

The former of these is anticipated to have caused mechanical deformation in certain components while the latter is expected to have caused some physical deterioration of the exposed exterior of the components.

A possibility exists that the reactor vessel head, and its closure studs are mechanically deformed due to the temperature transients and the corresponding thermal loadings which occurred. The amount of deformation should be quantified prior to the time of disassembly. Base and alternate methods in Section 3.5 address potential problems caused by deformation.

The same temperature transients may have caused distortion of the plenum assembly which may present removal difficulty. The possibility for plenum distortion and alternate methods for plenum removal in case of distortion and binding are currently being evaluated. Alternate removal methods in case of keyway binding are discussed in Section 3.6.

3.1.5 Schedule

Figure 3.1-2 is a logic sequence of the base method reactor disassembly subtasks. Included are the prerequisites which are noted throughout this report. Also noted are the points at which alternate methods would have to be implemented should they be required due to physical or environmental conditions.

3.2 Preparations

This section deals with facilities, support systems, and activities prior to beginning the CRDM uncoupling operation.

3.2.1 Missile Shield Removal

The sequence and method for removing the missile shields should remain unchanged. These lifts are made with the polar crane and pendant cables. The lifting cables should be inspected, replaced (if necessary) and then load tested prior to the lift due to expected loss of cable lubrication as a result of the harsh containment environment. Final location for the shield blocks should be carefully considered to preclude interference with subsequent operations and/or to maximize laydown space capacity. (Refer to Section 2.)

3.2.2 Head Storage Stand Preparation

A lead liner (Figure 3.2-1) should be built into the closure head storage stand for shielding during work on the closure head and adjacent areas on the operating floor and the 305' elevation. Shielded enclosures should also be provided for the closure head key blocks. In addition, provision for shield walls around the storage stand should be considered to provide additional shielding and for potential use in later RV head decontamination. Detailed study may indicate complete removal or redesign of the existing storage stand to provide for more efficient RV head decontamination, inspection, and rework.

3.2.3 Auxiliary Service Hoist

An auxiliary service hoist, shown in Figure 3.2-2, which spans the fuel transfer canal, should be provided. This hoist will be used for light lifts in the canal, such as lifts above the service structure for control rod drive uncoupling. The service hoist will relieve the polar crane for other lifts in the containment and is better suited for controlling the light critical lifts associated with reactor disassembly and, later, for fuel removal.

The hoist should be installed on rails on top of the D-ring shield walls adjacent to the fuel transfer canal. However, since work on the service structure may be done from an elevated platform (an alternate CRDM uncoupling method) and the leadscrews may be removed while the RV head is on the reactor vessel in order to flush the CRDMs, provisions should be made at time of installation to provide for a temporary lift height of 15 feet above the 367' elevation (top of D-ring shield walls). Additionally, this hoist should be installed to allow it to travel as close as possible to the containment wall at the deep end of the canal and at least 6 feet beyond the shield walls at the shallow end of the refueling canal.

Rail extensions, used for this purpose, must be removable. Extending the length of travel along the north-south axis will allow this hoist to be used for handling equipment directly between the elevation 347' operation floor and the bottom of the fuel transfer canal. Sizing the hoist equipment to lift at least 10 tons would allow moving equipment such as stud tensioners in and out the canal without the use of the polar crane.

The control pendant should be heavy duty and long enough to allow an operator to control the hoist from the 347' elevation at either end of the canal and slightly behind the D-ring shield walls. Radio control should be considered to avoid the pendant length but a redundant pendant control should be provided in any case.

This hoist should be designed as a permanent installation, since it will be useful for future outages.

3.2.4 Hot Machine Shop

Due to space limitations and radiation levels in the containment, bot machine shop capabilities should be provided outside containment. Controlled and direct access from the containment to the shop should be available.

3.2.4.1 Tooling Modifications

There is big' probability that existing tooling will need modification and special tooling will need to be fabricated as the reactor work proceeds. The hot machine shop should contain the following equipment:

- Power hacksaw
- Lathe
- Milling machine
- Drill press
- Welding equipment
- Overhead crane with 30' lift capacity
- Assembly area

3.2.4.2 Storage

Storage space for special equipment and tools for use in reactor disassembly is needed. This facility should include storage for mechanical and electronic equipment. Shop maintenance facilities for the electronic equipment should also be provided.

Storage is also needed for contaminated equipment removed from the containment. Following decontamination or when discarded as waste the controlled storage space will no longer be needed.

3.2.4.3 Training

A separate training facility should be provided for mockup testing and mockup and classroom training.

Training aids should include video tapes, equipment drawings, and equipment mockups where possible. Training instructors who are specialists in this work should be used extensively (see Section 3.11).

3.2.5 Support Systems

Due to the anticipated radiation levels and limited personnel resources, much of the reactor work will be controlled from the command center. Therefore, video and communications systems linking the command center with the work crews in the containment should be provided. The video and communications systems will allow remote management, supervision, and technical evaluation from the command center thus avoiding exposure of the key personnel involved. These systems will also be employed for personnel training purposes.

3.2.5.1 Communications

An audio communications system between the work crews and the command center will be required for all the reactor disassembly work. Two-way radio and microphone/headset equipment, which can be used in conjunction with breathing apparatus, should be provided. A detailed discussion of the communication system is presented in Section 6.3.

3.2.5.2 Video Support

Extensive video systems will be needed to support all reactor disassembly tasks. Area coverage of the fuel transfer canal and 347' elevation should be provided. Special cameras and systems should also be provided for key tasks and inspections.

A typical application of video systems is the closure head lift (Figure 3.2-3). During this mode, the video system will permit checking of the closure head for level, entangled debris, and damage. Taping of these events will permit later evaluation as well as provide for permanent records.

Other video applications would be plenum assembly inspection prior to and during removal as well as detailed video examination to determine whether or not the plenum can be used again. Detailed examinations of the core to record as-found condition will also aid in planning for fuel removal.

A detailed discussion of remote viewing is presented in Section 6.1.

3.2.5.3 Fuel Transfer Canal Cleanup System

Cleanliness of the fuel transfer canal water is important to minimize radiation fields due to potential suspended crud in the system and to improve water clarity for underwater inspection. A cleanup system should be provided for this purpose.

A skid-mounted assembly consisting of filters, pumps, valves, and interconnecting piping is suggested for this service. See Figure 3.2-4. The filter cartridges can be isolated and removed for disposal. Shielded containers should be provided for the spent filter cartridges. The changeout operation should be done remotely with long-handled tools and the auxiliary service crane (Section 3.2.3).

Placement of the system is important due to physical space limitations, potential radiation fields produced by contamination trapped in the filters, and the length of suction and discharge lines. A potential location for the system is at the north end of the fuel transfer canal near the containment wall on the elevation 347' operating floor. This location is away from the reactor disassembly work area. A system with submersible filters located in the fuel transfer canal should be investigated in order to provide additional shielding.

3.2.5.4 RV Head Vacuum System

In order to prevent gaseous releases through the open CRDMs during uncoupling, a slight vacuum could be applied to the reactor. There are eight vent valve thermocouple nozzles which terminate in blind flanges outside the service structure and above the insulation panels which could be utilized for access to the gas space under the RV head. A slight vacuum could be applied to the gas space discharging to the normal or a special vent header.

3.2.6 Fuel Handling Bridges

Fuel distortion and structural weakness are expected in the TMI-2 core. The existing fuel handling equipment is designed so that the fuel is lifted from the end fitting, thus applying some tensile load to the fuel assembly. There is also a close tolerance between the fuel handling mast and the fuel assembly which will not accommodate significant distortion. These conditions would prohibit the use of existing fuel handling equipment.

Since the main and auxiliary fuel handling bridges (see Figure 3.2-5) are not expected to be needed for handling fuel by the normal method, it is suggested that the bridges be modified for use as work platforms.

The trolley assembly with the control panel and fuel/control component mast should be removed and stored outside containment

(Figure 3.2-6). The bridges (spanning the canal) should then be modified to function as work platforms. These modifications should provide:

- working platform with safety railings
- shielding under platform as required
- storage of long tools (hanging)
- storage and handling of underwater video cameras and lights
- power, water, air, vacuum, and communications.

The bridges are expected to be used for handling tools for remote operations, for inspections, and for cleaning loose debris.

3.2.7 Polar Crane

The polar crane must be operable. A power rotation feature should be provided for the main hook to rotate major components remotely. Pendant type controls for the crane should be provided which can be operated from the 347' elevation or the 371' elevation. A high accuracy load cell should also be provided. Several commercial loadcell units are available and should be evaluated.

3.2.8 Internals Storage Stand

The internals storage stand should be in (or placed in) the deep end of the fuel transfer canal. See Figures 3.2-7 and 3.2-8. In order to avoid interference when placing the plenum assembly in the storage stand, the storage stand must be level within .010 inch per foot, not to exceed 1/8 inch side to side. The stand should be centered between the north and south walls of the deep end of the fuel transfer canal and centered between the east and west upenders.

3.2.9 Plenum Inspection Station

Since inspection of the lower portion of the plenum cylinder is difficult while the plenum is in the internals storage stand, an inspection station should be provided consisting of underwater television cameras and underwater lights and supporting stands. The inspection station should be on the fuel transfer canal floor approximately half way between the reactor vessel and the deep end of the canal. See Figure 3.2-9.

This station would be used to inspect the entire upper grid area of the plenum for damage or debris by moving the plenum over the camera.

The cylindrical portion of the plenum can also be inspected in this location by the use of additional cameras operated from the fuel handling bridge work platforms.

3.2.10 Laydown Space Requirements

It is anticipated that a number of in-containment operations will be conducted in parallel with the reactor disassembly work, and, therefore, laydown space will require careful planning.

For the reactor disassembly work, laydown space for the components involved is as follows:

- Missile shield blocks
- RV head stand
- Internals storage stand
- Lifting tripod, pendants, and handling adapters
- Tooling staging area

Any smaller components (i.e., insulation, CRDMs, etc.) which are to be decontaminated, inspected, or discarded, should be removed from containment.

3.2.11 Containment Elevator

The existing elevator should be evaluated for possible use prior to the beginning of the reactor disassembly work.

3.2.12 Local Shielding

Local shielding should be available in a number of locations for reactor disassembly. Movable panels should be provided in the canal for use during detensioning, seal plate installation, and other operations. Other locations which may need local shielding are expected to be identified by radiation surveys.

3.3 Control Rod Drive Mechanisms

This section describes methods of uncoupling both axial power shaping rod and control rod drive leadscrews from their control components.

3.3.1 Axial Power Shaping Rod (APSR) Uncoupling Methods

APSRs are not used to achieve or maintain shutdown conditions. The drive mechanisms used to drive APSRs are designed so that in the event of a reactor trip there is no automatic insertion of the APSRs. They are, therefore, in the same position, approximately 20 percent withdrawn, as they were at the time of the reactor trip on March 28, 1979. The fact that the APSRs remained partially withdrawn during the transient should have precluded damage to the coupling and should permit uncoupling by near standard methods. However, it is possible that some APSR assemblies may be stuck and it may not be possible to verify uncoupling. It is assumed that stators may be inoperable and normal leadscrew backing cannot be accomplished.

Base Method

Prior to uncoupling, the vacuum system described in Section 3.2.5.4 should be installed. This will provide an air flow path down through the CRDMs as the closure assemblies are removed, thus prohibiting gases from escaping, uncontrolled, through the CRDMs to the containment.

During the uncoupling operation, radiation exposure may be reduced by uncoupling all mechanisms prior to parking or removing any leadscrews.

If the APSRs were free to move axially and were uncoupled in the present position, they would free fall to the inserted position; therefore, the APSPs should be driven down to the normal uncoupling position. After verify the operability of the existing stator, the APSRs should be driven down to the normal uncoupling position with a portable service power supply and the existing or a replacement stator. Uncoupling by the normal method and verifying uncoupling by free rotation of the leadscrew would then follow. The leadscrew can then be moved to the parked position using the service power supply and stator.

As in normal refueling operations, an elevation in radiation levels will probably be noted when the leadscrews have been withdrawn to the parked position. This is due to a significant amount of crud which is normally deposited on the leadscrew surfaces. At this point the decision could be made to remove the leadscrews from the control rod drive mechanism. This would entail using the auxiliary device hoist, noted in Section 3.2.3, to raise the leadscrew into a shielded container to contain contamination and moving to a storage inspection station for decontamination and inspection. Removal of the leadscrews would accomplish a reduction in exposure levels around the service structures as well as providing a flow path for flushing the

CRDM into the reactor coolant system, which would be a significant aid in the CRDM disassembly and inspection to follow. The decision to proceed with leadscrew removal while the RV head is on the reactor vessel should be based upon radiation surveys.

Alternates

If a replacement stator and power supply cannot be used or the rod is stuck and cannot be driven down, the leadscrew may be uncoupled from the APSR in its present position. If the rod is free, it should move down to the inserted position and uncoupling can be verified by free rotation of the leadscrew. An evaluation of the consequences of an APSR free fall from its existing position to the fully inserted position should be done. If the rod is stuck and will not move down, uncoupling should be verified by differential weight measurement. The leadscrew should be moved to the parked position by rotating the torque tube.

If radiation levels on top of the service structure are unacceptably high, an alternate shielded work platform should be provided approximately 10 feet above the service structure. This alternate would require lengthening the existing tools.

If the above methods are unsuccessful in driving down the rod or uncoupling, cutting through the entire drive assembly just below the roller nut drive (Figure 3.3-1) should be considered. This method may require additional access by removing adjacent drives, providing access through the service structure, or removal of the service structure. Procedure development, tooling development, and mockup testing for this alternate would be required.

3.3.2 Control Rod Uncoupling Methods

The control rods tripped early in the incident and together with the coupled leadscrews should be in the fully inserted position.

Base Method

The control rods should be uncoupled by the normal method using existing tooling. Once uncoupled, the auxiliary service hoist, noted in Section 3.2.3, would be used to raise the leadscrews to the parked position.

An elevation in radiation levels due to crud will probably be noted when the leadscrews have been withdrawn to the parked position. As with the APSR leadscrews described in 3.3.1 above, a decision concerning removal of the leadscrews should be made at this point. The removal of leadscrews will lower radiation levels around the RV head service structure and also will provide a flow path for flushing the CRDM.

Alternates

If radiation levels prohibit personnel access to the top of the service structure for the uncoupling operation, an elevated platform could be fabricated in such a way that personnel would have an additional 10 feet of distance, which would lower exposure levels. The platform would be shielded with movable plate so that access through the platform would be provided over CRDM locations to allow the uncoupling operation to be performed. The use of the elevated platform would necessitate that all tooling be lengthened by 10 feet.

As in the case with APSRs, if a control rod assembly cannot be uncoupled successfully, cutting through the entire drive assembly just below the roller nut drive should be considered (in Figure 3.3-1). This method may require additional access by removing adjacent drives, providing access through the service structure, or removal of the service structure. Procedure development, tooling development, and mockup testing for this alternate is required.

3.3.3 Shielding

Shielding for the uncoupling operation could be provided by sheets of lead placed around and between the tops of the CRDMs. Additionally, lead sheet could be sandwiched between sheets of plywood, forming sections that can be placed directly on top of the CRDMs and arranged to provide access to one CRDM at a time.

In the event that the elevated platform is used, the floor of the platform would be covered with the same movable section described in the preceding paragraph.

3.3.4 Personnel and Training

A CRDM mockup should be provided for training purposes and for testing of alternate tooling when required.

Video tapes of actual uncoupling operations are available and, with an instructor, should be used for training of craft labor and supervision. Special operations can be mocked up and put on video tape for training purposes if required.

This operation should be directed by experienced service personnel. Drive mechanism uncoupling is a difficult task in reactor disassembly to cover adequately using only audio and video. Key service personnel may be required locally if problems should arise.

3.4 Preparations for Reactor Vessel Head Removal

This section includes the miscellaneous tasks needed prior to RV closure stud detensioning.

3.4.1 Neutron Shield Tank

Twelve neutron shield tanks are hinged to the floor of the canal, using the canal seal plate bolting ring and filled with water. The hinge pins must be pulled to permit the tanks to be lifted. Water, if any remains in the tank, need not be drained before the tanks are lifted. An existing tripod rig should be used to lift each tank. The neutron shield tanks should be removed from the containment. If needed, remote handling of these tanks is possible following fabrication and checkout of new tooling.

3.4.2 Head Insulation

The RV head insulation consists of 16 pieces of insulation approximately 83-9/16 inches high, 40-1/2 inches wide, and 4 inches thick. The insulation is the reflective type and is constructed with layers of sheet metal with air space between them. The air spaces, however, are not sealed and the panels probably have high levels of both internal and external contamination. The panels are removed individually by installing a handling lug on the top and lifting with a crane. Each panel shall be bagged as it is removed to contain contamination. Once bagged the panels can be stored or disposed of outside of containment.

High radiation levels can be expected in the work area (top of panels) due to "shine" from motor section of CRDMs. To reduce levels for insulation removal and subsequent stud work, the outside diameter of the service structure could be banded by lead sheet (Figure 3.4-1) for a distance of approximately 4 feet above the top of the insulation panels. To further reduce exposure during removal, service structure cooling fan covers could be fabricated and installed. A portable work platform and the short lifting posts provided for Unit 1 could be used for this work. A system for compacting insulation inside containment should be considered, to facilitate removal.

3.4.3 Canal Seal Plate

Due to the extended period of time that the seal plate will be in place, proper installation is important. Figure 3.4-2 illustrates the recommended seal plate design. The canal can be partially flooded for leak testing the canal seal plate prior to the RV head lift.

Alternates to the normal seal plate arrangement are as follows:

- More durable gasket materials should be considered. These gaskets should be cemented in place and all potential leak paths should be eliminated. Gasket seating surfaces should be thoroughly cleaned.

Coating the sealing surfaces with zinc chromate, as described in the existing plant procedure, is no longer recommended.

- Welding the seal plate in place should be considered.

3.4.4 Control Rod Drive Cooling Water Spool Pieces

These spool pieces are used to connect the CRDMs to the cooling water supply. They are located between the top of the RV service structure and cooling water line on the side of the Fuel Transfer Canal at the 347' elevation. Depending on radiation levels, the spool pieces will be disconnected by one of two methods. If levels are reasonably low, connections will be unbolted using standard procedures and practice. This will require the use of hand wrenches at the top of the service structure. If, however, levels are high, a remote controlled power saw could be used to sever the cooling lines at a point to allow clearance for head removal from the canal.

3.4.5 Fuel Transfer Tube Covers and Canal Drain Cover

The drain cover in the bottom of the deep end of the fuel transfer canal will have to be installed prior to flooding the canal. Due to the long period the drain cover will be in place, consideration should be given to redundant sealing method such as welding.

The fuel transfer tube covers are 30-inch blind flanges. (See Figure 3.4-3.) Each is approximately 700 pounds and is held in place by 28 studs with hex nuts on both ends of the studs. The blind flanges are equipped with a lifting eye for use in removing to the laydown area out of the fuel transfer canal. Air- or electric-driven equipment should be used to remove nuts after the nuts are broken loose. Any foreign deposits should be removed from the threaded studs and a small amount of lubricant applied before running the nuts off. The normal procedure is to use the auxiliary hook on the polar crane for removing these covers. If the auxiliary hoist discussed in Section 3.2.3 is installed with extended rails as recommended, it could be used for this work.

Radiation levels in the fuel transfer canal may require the use of portable shielding for work on the drain cover and fuel transfer tube flanges. A portable shielded work enclosure could be provided for this purpose.

3.4.6 Electrical and Instrumentation Disconnects

Electrical disconnection for the CRDs is expected to be difficult due to potential corrosion of normally tight fitting terminals. Since the containment environment has probably ruined the cable, these connections could be cut to facilitate head removal. Radiation levels at the top of the canal, where the cables are located, may require the cutting operation to be done remotely. However, tooling will remain basically the same despite different length controls and handling methods. Actual cutting can be performed with commercially available cable shears. Once the cables have been removed from the service structure, the cables and cable trays running along the side of the transfer canal can be removed, allowing more room and also eliminating an area of high contamination that cannot be effectively cleaned.

3.4.7 Remote In-Vessel Inspection

Consideration should be given to the use of remote means for inspection of the reactor vessel interior prior to RCS disassembly. It may be possible to insert fiber optics, miniature video equipment, or other diagnostic instruments through reactor vessel penetrations (control rod guide tubes, incore detector guide tubes) to inspect the core and the reactor vessel internals. Such inspections would aid technical planning for RCS disassembly and defueling and could be useful in detecting physical interferences and regions of severe damage.

3.5 Reactor Vessel Closure Head Removal

The work described below is shown in sequence in Figure 3.1-2. The sequence includes closure stud detensioning, closure stud removal and parking, and RV head lift.

3.5.1 Detensioning

Detensioning requires that each of the 60 studs be stretched and the castellated nut backed off, in order to remove the clamping load between the head and the RV. This is normally done in two detensioning passes with work crews in the fuel transfer-canal using locally operated equipment (See Fig. 3.5-1).

Difficulty is anticipated due to oxidation of closure studs and nuts, further complicated by anticipated high radiation levels.

Regardless of which method is used, an early thorough soaking of the studs and nuts with an approved lubricant (methyl salicylate has been previously useful) is suggested. Wire brushing of the exposed thread stud area is also recommended.

3.5.1.1 Base Plan

It is recommended that the base plan include installation of improved motorized engaging nut drives (MEND) and a modification to include a power-operated nut turning mechanism on the stud tensioners. Power-operated tensioner hoists should also be added to the service structure monorail to reduce time in transferring stud tensioners.

Other than the above modifications, the base plan should follow normal procedures for detensioning. Detensioning is accomplished with two tensioners in two complete detensioning passes. In order to avoid radiation exposure contribution from the lower CRDM area, tensioner positioning should be done from the canal floor rather than from the RV head flange.

3.5.1.2 Alternates

- This alternate should be implemented if radiation exposure due to proximity to the RV head flange is unacceptably high.

  The stud tensioning can be adapted for semiremote operation. With this modification, the tensioners are positioned locally at the RV head flange but are engaged, pressurized, and disengaged remotely. The stud nut would be turned locally after pressurizing the tensioner. This method should include provisions for a remote control panel and close video coverage of the tensioner.

- If radiation exposure rates are not substantially reduced by distance in the alternate above, a totally remote, underwater detensioning method should be considered (See Figure 3.5-2).

  Modification to existing tensioners for underwater operation is possible, but positioning methods need to be developed and implemented.

  Consideration should also be given to the use of an entirely different tensioning system. Units are commercially available for underwater operation but would require custom fabrication or modification for this application.

- If problems exist with stud detensioning stud heaters can be used.

  The closure studs are fabricated with a vertical hole through the center for the entire length. After removal of the elongation measuring rods, a heater could be inserted into the studs. Preliminary calculations show that a temperature of 300 F should be sufficient to relax the stud load and could be achieved in approximately 1 hour. The closure nuts would then have to be backed off to complete the unloading sequence. This method should also be considered for use with problem studs, since this method of elongation allows direct access to the castellated nut.

  Additional development and analysis is needed for this alternative.

3.5.1.3 Shielding

Shielding around the lower portion of the service structure should remain in place for the detensioning operation (Figure 3.4-1).

If a semiremote tensioner operation is performed, additional temporary shielding can be provided around the remote control panel, if needed.

For the underwater alternative, shielding is provided by the flooded fuel transfer canal.

3.5.1.4 Personnel and Training

The detensioning operation may be performed by inexperienced personnel. The actual operation should be directed by service personnel familiar with tensioner operation and detensioning techniques. Training should be provided by video tapes and actual mockups. Training tapes on stud tensioner operation, rigging and setup are available and should be extensively utilized. A closure stud mockup (Figure 3.5-3) should be provided for training and for checkout of new or modified equipment to facilitate actual performance and to minimize worker exposure.

3.5.2 RV Stud Removal

This section covers removal of the studs from the RV flange and parking the studs on the RV head flange.

It should be recognized that some number of studs could be stuck and will not rotate by any method below. In this event, the stuck studs can be left in place in the reactor vessel flange and RV head, plenum, and fuel removal should proceed. During this period, tooling and procedures could be prepared to remove the stuck studs. Experience indicates that vertical impact while applying torque should be successful.

3.5.2.1 Base Method

The base method should be the normal removal sequence using existing tooling (See Figure 3.5-4). The tooling rotates the studs out of the stud holes and, by using a spacer under the nuts, the studs are parked on the RV head flange. In this position, the studs do not extend below the RV flange mating surfaces. Work should be done only from the canal floor to limit exposure from lower portion of the CRDMs.

3.5.2.2 Alternates

- If radiation levels indicate the need, the removal tool can be adapted for semiremote operation. The removal tool would be operated from a distance while the studs would be lifted and parked locally. This method would also require provision to accommodate the reaction force of the tool torque.

- Underwater operations should also be considered. This approach would require new tooling design, development, and testing.

3.5.2.3 Shielding

Shielding around the lower portion of the service structure should remain in place for the stud removal operation (Figure 3.4-1).

If semiremote operations are used, additional temporary shielding around the control panel can be provided, if needed.

For the underwater alternative, shielding is provided by the flooded fuel transfer canal.

3.5.2.4 Personnel and Training

The closure stud mockup (Figure 3.5-4) is suitable for training inexperienced personnel and supervision for stud removal and should be provided. This operation should be directed remotely by experienced personnel. Extensive use of video tape and instructor training is suggested.

3.5.3 RV Head Rigging and Lifting

This section describes the RV head rigging and the lift. Included are those special measures needed as a result of unknown damage. It should be recognized that, at the time of this operation, considerably more information relative to the CRDM uncoupling will be known. A hold point for evaluation of that information should be inserted here.

3.5.3.1 Base Method

Rigging for the RV head lift will be by the normal method. The lifting tripod is used with three pendant lifting cables, two with adjustable turnbuckles. Due to the exposure to the containment environment, it is possible that lubricant was forced out of the existing cables which have remained in the containment and therefore, load testing is required and inspection of cables for possible replacement should be considered. If replacement is selected, a solid rod lifting pendant could be used. Rigging is accomplished by pinning the lift cables to the RV head and the lift is made using the tripod and polar crane. Refurbishment of the polar crane is assumed to have been completed prior to this operation.

The base method should include provisions for raising water level in the fuel transfer canal to the top of the RV head flange for shielding prior to the lift. This operation also allows an additional leak check of the seal plate.

Underwater cameras at the three lift points should be used to monitor the RV head lift (Figure 3.2-4). Initially, their function will be to assure a level lift. The cameras should be equipped with zoom, pan tilt, and lighting in order to perform inspections under the RV head early in the lift. This point is expected to be about 2 feet above the RV flange.

3.5.3.2 Alternates

Although no alternates, as such, have been considered for RV head rigging and removal, conditions may necessitate the following:

If one or more control components could not be uncoupled and it was necessary to cut through the CRDM assembly as described in Sections 3.3.1 and 3.3.2, a section of leadscrew remains coupled to the control component and extends through the penetration in the RV Head. It will be necessary to ensure that part of leadscrew is not moved when the RV head is lifted. This can be done either by video observation during the lift or by the use of a dipstick tool, which is a simple rod arrangement in which a long rod is lowered through the service structure and is supported by the top of the segment of leadscrew. When the head is lifted relative motion of the RV head to the leadscrew can be observed.

In the event that one or more RV head studs cannot be removed from the RV flange and it is decided to leave them in place, it will be necessary to remove the stud nut and spherical washer set so that the RV head may be removed. Every effort should be made to remove the stud prior to removing the RV head because the stud may be an obstacle to core examination and fuel removal and efforts to remove that stud may result in tooling or parts of tooling falling into the core.

3.5.3.3 Shielding

Shielding for rigging to the RV head should continue to be provided around the lower portion of the service structure, per Figure 3.4-1. This may have to be removed for the lift due to lifting pendant limitations.

Fuel transfer canal water level should be raised to some level to provide shielding over the plenum assembly after the head has been removed. It may be prudent to fill the canal to refueling level prior to head removal but this option will have to be carefully examined to determine any detrimental effects to the RV head or CRDMs.

It should be noted that when moving the RV head to the storage stand high radiation levels could exist due to plateout of fission products and the potential for core debris on the underside of the RV head.

After removal and placement on the storage stand, the contribution to exposure by the RV head on the 347' elevation may continue to be significant even though the stand has been shielded. A concrete (or other suitable material) shield wall around the storage stand should be considered to reduce exposure. If the band of shielding per Figure 3.4-1 was removed for the lift, it could be reinstalled after RV head placement on the stand.

3.5.3.4 Personnel and Training

Video tape and instructor training is suggested for this operation.

Control of the lift, monitoring of radiation level, monitoring video display, and control of the work crew should be exercised from the command center by experienced service personnel. Training should include emphasis on coordination of these activities.

3.6 Plenum Removal

This section covers the base method and alternatives for plenum removal.

Prior to the lift, a video inspection should be made of the plenum with attention given to the following areas:

- The annulus between the core support assembly ID and the plenum cylinder OD, with the main interest around the area of the LOCA restraints and the interface between the upper grid and the bolt ring area of the core support assembly.

- Inside the plenum cylinder, where accessible, at the interface between the column weldments and the upper grid. This will allow inspection of the top of a number of fuel assemblies.

- Plenum keyways to the reactor vessel key interface.

- Plenum cover elevation relative to the reactor vessel flange. Normally the fuel assembly springs will raise the plenum about 1/2 inch above the RV flange.

- The inside of the control rod guide tubes for inspection of the top of the control components and fuel assemblies.

Also prior to execution of this lift, removal and inventory of all loose material which can be reached should be completed. The plenum is flat and horizontal and can be readily vacuumed.

3.6.1 Base Method

Plenum removal should be performed underwater to limit exposure. The indexing fixture which serves to extend the guide keys of the reactor vessel and is normally used for removal and installation should not be used, in order to allow more flexibility during the move and to facilitate close video inspection during the move.

Any CRD leadscrews previously cut should again be cut level with the top of the CR guide tubes to allow rigging the plenum for the lift.

The lift rigging includes the tripod with attached internals handling adapters with adjusting turnbuckles and latchboxes to latch the plenum lifting lugs. Since the turnbuckles cannot be adjusted under load and the plenum cannot be blocked up due to access for the initial part of the lift, remotely operable leveling devices should be used in place of the turnbuckles for leveling the plenum assembly.

As in the case of the RV head lift, to ensure that no control components are being withdrawn with the plenum, dipstick gauging should be used to monitor relative motion of the previously cut lead-screws and the plenum cylinder. Several other control component locations should also be dipstick gauged to provide assurance that there is no general condition existing that would cause the fuel to lift with the plenum (Refer to Section 3.5.3.2.).

Video coverage of the plenum cover flange should be provided during the initial part of the lift. The lift should proceed slowly while monitoring the load cell for about 1 foot. At this point, a video scan of control rod guide tube locations should be repeated. Some viewing should now be possible of the tops of fuel assemblies adjacent to the CR guide tubes. A hold for evaluation of this inspection should be implemented.

The lift should then proceed to about 2 feet. At this point, larger underwater cameras with better lighting can be inserted into the annulus between the plenum and the core support assembly. Video inspection of the entire top of the core should now be possible. A second evaluation hold point should be implemented. The plenum should be blocked up for this inspection to eliminate interfaces between lift rigging and inspection equipment.

If debris is moving up with the plenum, it may be possible to show that this condition is allowable during the lift or long-handled tools may be required to release the debris. This can only be evaluated when the condition occurs. This operation would require close video coverage and long-handled tools used from above. After the lift has begun, the plenum should not be reseated due to the potential of causing additional damage if fuel movement bas taken place after the plenum was lifted.

Interference between the plenum bumpers and the core support assembly vent valves is a possibility. The plenum should be moved carefully through this zone to minimize potential hangup at this interface.

The lift should continue until the plenum clears the reactor vessel. The plenum should then be moved to an area adjacent to the RV where video cameras have previously been positioned to view upward at the plenum assembly grid. The plenum should then be translated so that the entire lower portion of the plenum has been inspected for debris and damage. Any accessible debris should be removed prior to placing the plenum on the storage stand. The exterior portion of the plenum should also be video inspected, particularly through the large flow holes in the cylindrical shell.

The plenum should then be placed in the storage stand in the deep end of the fuel transfer canal. If necessary later, the plenum can be placed on the canal floor at either end of the shallow portion of the fuel transfer canal. It should be placed on stainless steel bars or beams for better weight distribution.

3.6.2 Alternatives

The following alternatives address potential difficulty in removal of the plenum from the vessel, since deformation or debris may cause plenum keyway-to-RV key interference.

- If key to keyway binding occurs (Figure 3.6-1) additional lift force should be applied and an impact device used as shown in Figure 3.6-2. This type of device is commercially available and can be adapted for this service.

- If impact does not free the interference, the keyways in the plenum flange should be cut out using an underwater air arc remote machining, or other suitable means. Care should be taken in order to avoid upset of the key material since the core support assembly will also have to clear these keys.

3.6.3 Shielding

Shielding will be provided by water in the fuel transfer canal.

3.6.4 Personnel and Training

Video tape and instructor training for the base method lift should be provided.

Close coordination of the extensive video coverage with the lift is needed and should be included in the training program.

3.7 Core Support Assembly Removal

This section addresses removal of the core support assembly (CSA) following fuel removal.

Prior to the lift, removal and inventory of all accessible loose debris should be completed. Video inspections of the annulus between the reactor vessel and core support assembly, as well as the lower spherical dome under the CSA, should also be completed prior to the lift. The internals storage stand must be available in the deep end of the canal. The plenum should be placed at the south end of the fuel transfer canal on the canal floor.

3.7.1 Base Plan

The rigging for the CSA lift is identical to that for the plenum lift (Section 3.6.1). The base method is a normal lift with the fuel transfer canal full and no indexing fixture. No unusual inspections are necessary during the lift. The bottom CSA should be inspected in the same manner as the plenum (see Section 3.6.1). Following inspection and debris removal if required, the CSA should be set in the storage stand in the deep end of the fuel transfer canal.

3.7.2 Alternate

The two plenum assembly alternates apply to the core support assembly as well. (See Section 3.6.2.)

- Impact
- Cut clearance for the keyways

3.7.3 Shielding

Shielding is provided by water in the fuel transfer canal.

3.7.4 Personnel and Training

Training for the normal removal method should be by video tape and instructor.

3.8 Surveillance Specimen Capsule Removal

Surveillance specimen capsule holders are located in the annulus between the core support assembly thermal shield and the reactor vessel. The three capsule holders, located around the core, each contain two specimen capsules and are accessible through access holes in the core support assembly flange, once the upper plenum assembly has been removed.

Each of the six installed specimen capsules contains specimens of reactor vessel base metal and weld metal for tensile, charpy, and fracture toughness testing. In addition dosimetry and alloy temperature monitor specimen are included.

TMI-2 served as a host reactor for the irradiation of capsules from TMI-1 and B&W users' group capsules. Three of the six installed capsules are for TMI-2; one from TMI-1, and two for B&W users.

All six specimen capsules should be removed and stored for future reinstallation and one of the TMI-2 capsules should be opened for evaluation.

Evaluation of the specimen can contribute to the assessment of the condition of components by yielding information on the maximum temperature seen by the specimen, as well as effects of the transient on reactor vessel materials and any changes in mechanical properties of the materials.

3.8.1 Base Method

Unless there is other justification, specimen removal should be accomplished prior to fuel removal and core support assembly removal. This will allow early examination of specimen which could have bearing on determining the condition of components.

Procedures and tooling for specimen capsule installation, removal, and handling presently exist and should not have to be modified for the removal operation.

3.8.2 Alternate

The capsules may be removed with the CSA in the deep end of the fuel transfer canal in the internals storage stand. Access to the specimen holders on the CSA would be easier if this alternate is used.

3.8.3 Shielding

Shielding is provided by the canal water level.

A shielded container will be required for shipping the capsules.

3.8.4 Personnel and Training

A training mockup should be provided with a set of tooling for the selected removal method.

3.9 Tooling and Fixtures

Depending on the method chosen to perform operations, existing tooling may be used as is on modified so as to be compatible with required working conditions.

3.9.1 Control Rod Drive Mechanism Uncoupling

- Dipstick

  To be inserted into the CRDM to determine the elevation of axial power shaping rod leadscrews. Can also be used during RV head lift to gauge relative motion between the RV head assembly and control components or their leadscrews.

- Uncoupling Tool

  Should be lengthened to allow operation from a platform over the service structure if radiation level requires the use of that alternate.

- Leadscrew Installation/Removal Tool

  Used for installation and removal of leadscrews and should be lengthened if the raised platform is used for removal operation.

- Load Cell

  Used to verify that control rods have been uncoupled. Range should be 0-500 pounds with accuracy of ±1 percent at 150 pounds.

- Motor Tube Extension

  Necessary only if elevated platform is used for CRDM uncoupling - it would allow the use of a pneumatic uncoupling tool ("jumping jack") for uncoupling.

- Special Stator Assembly

  The existing stators on the CRDMs are assumed to be inoperable. This assumption can be confirmed by electrical tests. The removal of existing stators from APSRs would allow the special stator to be installed to drive APSRs to their fullest position and then to park the leadscrew.

3.9.2 Reactor Vessel Head Removal

- Man Lift

  To facilitate personnel access to the fuel transfer canal. Existing access is by circular stairway or ladder, neither of which is compatible with use of respiratory protection equipment.

- Motorized Engaging Nut Drive (MEND) Units

  An air drive system, for installation on stud tensioners, which drives the stud tensioner engaging nut. Provides a more efficient and reliable method of stud tensioner engagement to the RV studs.

- Remote Operation Package for Tensioners

  Stud tensioner operations can be adapted to remote control by changing manually operated valves to solenoid operated valves with pendant controls. This modification can be designed and tested if radiation levels determine this alternate to be necessary.

- Underwater Detensioning Units

  These units have been used in other applications. A study should be performed to determine if the system could be used on the existing closure.

- Nut Turning Mechanism Modification

  With the RV stud under tension the present method of unseating the stud nut is a hand-operated mechanism (an integral part of the stud tensioner) which provides some slight mechanical advantage. A power-operated turning mechanism should be developed and adapted to the stud tensioner to alleviate potential problems with stud nuts.

- Nut Turning Mechanism Modification

  Stuck nuts are anticipated. A power-operated turning mechanism should be developed and adapted to the stud tensioners.

- Wire Brush for Closure Studs

  This tool would allow rust to be removed from the closure stud threads to allow ease of engagement of the tensioner and removal of the closure nut.

- Measuring Rod Puller

  To remove measuring rods if stud heaters are used.

- Tensioner Hoists

  New tensioner hoists should be considered to improve time span for positioning.

- Guide Cones (Figure 3.5-2)

  Guide cones to protect adjacent studs from being damaged by remotely located tensioners.

3.9.3 Reactor Vessel Stud Removal

- Stud Removal Tool

  The base plan uses this tool as is. If semiremote operations are planned, this tool could be modified by extending the controls for operation at a remote location. Close video support should be used with the remotely operated tool.

- Underwater Stud Removal Tooling

  If this alternate is selected, the underwater tooling will have to be developed.

3.9.4 RV Head Rigging and Lifting

- Pendants

  If replaced, the solid rod lifting pendants should be used.

3.9.5 Plenum Removal

- Dipstick

  For monitoring relative motion between leadscrew and plenum. (See Section 3.5.3.2.)

- Impact Device

  This device should be considered for specific application to plenum and core support assembly removal. Limits on its use should be established and testing of the device should be done. The basic unit is commercially available.

- Keyway Cutting Tools

  Methods and tooling for cutting and enlarging keyways in both the plenum cylinder and the core support assembly should be investigated. Underwater air arc is possible but hard to control. Other alternatives should be pursued.

- Underwater Vacuum System and Filters

  This system should be available for use in removing debris from the plenum, the core support assembly, and the reactor vessel.

- Support Blocks

  For plenum inspections at approximately 2 feet above the RV flange (see Section 3.6.1.).

3.9.6 Core Support Assembly Removal

The impact device and keyways cutting tools in Section 3.9.5 apply here as well.

3.9.7 Surveillance Specimen Capsule Removal

- Handling Tools

  Existing handling tools can be used. Length of tools may have to be modified to make them compatible with removal method to be used.

- Transfer Containers

  A transfer container which will accommodate the capsule(s) and be compatible with the final configuration of the fuel transfer mechanism should be provided for capsule transfers.

3.10 Temporary Shielding

An important step in the reactor disassembly plan will be to provide temporary shielding over significant radioactive sources in the area where work is to be performed. Anticipated radioactive sources which may require temporary shielding are identified throughout this study, and the various protective measures which can be taken are described. The objective of temporary shielding is to reduce the dose rates in the area of work to levels which allow adequate time in the area without exceeding allowable dose limits. Temporary shielding will be provided to reduce dose rates to ALARA.

Table 1.1-1 provides approximate area radiation levels which are assumed in the containment prior to RCS disassembly and fuel removal. As stated in Section 1.2 these levels should not be considered as either necessary or sufficient objectives for commencing Phase II. Nevertheless, these levels are helpful in planning worker stay times for activities in these general areas. The need for temporary shielding at various locations will be better defined as the technical plans for RCS disassembly, component laydown and storage, and personnel access are finalized.

The level of contamination on RCS components is also important in determining the amount of shielding required. As preparation for RCS disassembly progresses, expected contamination levels should be predicted, using all available data on core damage levels and reactor coolant isotopic concentrations. Of particular importance will be an accurate assessment of the contamination levles on the inside surfaces of the reactor vessel head. This has a large impact on personnel access and shielding requirements on elevation 347' in the vicinity of the reactor vessel head storage stand.

Temporary shielding materials that should be considered for use are as follows: lead sheet, lead brick, steel brick, steel sheet, water cans, concrete block, and poured concrete.

3.11 Training

A training program for the reactor disassembly work should be prepared well in advance of actual work and executed consistent with the anticipated work schedule. Consideration should be given to:

- Unknown conditions inside the reactor
- Heavy use of inexperienced personnel
- High turnover rate due to radiation exposure limits
- Remote supervision of most of the reactor disassembly sub-tasks
- Coordination of several disciplines involved in the work
- Specific containment environmental conditions.

3.11.1 Training Facility

The training facility should consist of classrooms and an equipment mockup area. Classrooms should be equipped with audio-visual equipment, equipment models, and drawings. One of the classrooms should be located adjacent to the mockup area. This classroom should have the capability to set up a working model of the command center to simulate remote communication and supervision conditions.

The equipment mockup area would allow training on the mockups and could be used for testing of new or modified tooling and equipment.

3.11.2 Training Program

The program should be developed in matrix form addressing the training subjects relative to the needs of each discipline involved in the work and provide the opportunity for the disciplines to train as a team. Figure 3.11-1 is an example of such a matrix.

The general training subjects which should be included in the training program are:

- Site Access HP and Security

  This is the normal access training now being provided at TMI for all incoming personnel.

- HP and Radiation Control for Containment Conditions

  This subject should be included to provide more specific training relative to the known containment conditions. Radiation Work Permits should be available and used for this subject.

- Reactor Disassembly Plan

  Personnel responsible for the execution of the work and for evaluation of the information available during the work should be thoroughly familiar with the overall plan. Consideration should be given to beginning this training well in advance of the execution phase to allow visits to other reactor sites if needed.

- Individual Task

  Classroom training for the individual sub-tasks should be provided for the team of workers who will execute the sub-task. The instructors should be the supervisors and engineers who are most familiar with the sub-tasks to be performed.

- Mockup Training

  Where practicable and needed, mockups should be used for training. To the extent possible, the physical conditions should be simulated. Workers should be dressed in accordance with the radiation work permit and remote communications and video should be used for supervision.

Documentation of completion of the training program requirements should be provided. Access badges should be color coded or otherwise marked to easily identify workers who have completed the required training.

3.12 Health Physics Considerations

A number of factors should be considered when establishing health physics procedures and training programs for the reactor disassembly work. Some of these are:

- Workmen may be inexperienced. During a normal outage there are experienced maintenance personnel locally supervising disassembly operations who control the inexperienced personnel under their direction to assure that station HP procedures and good radiation work practices are followed. This close attention will not be possible using the command center concept for supervision of the work.

- HP technicians will probably be temporary help with less specific TMI experience than the permanent station personnel normally assigned.

- Significant beta and alpha activity is expected in the area and potential rupturing of previously intact fuel pins may cause release of gas containing fission products.

- Personnel staging areas should be established in the containment. When not participating in a specific work task, workers should return to these staging areas.

- Due to surface contamination throughout the containment it will be necessary to control contamination entering as well as leaving the work area. As a minimum, gloves and shoe covers should be changed on entering as well as leaving the work areas.

- All equipment used in the containment should be considered as contaminated on external surfaces. All components to be removed from work areas should be bagged or sprayed with a strippable coating to avoid spread of contamination.

- Handling of tools and inspection equipment in the fuel transfer canal reactor vessel is expected to be extensive and will require close control to avoid worker contamination.

- Training programs should place significant emphasis on known containment environmental conditions applicable in the specific work areas.

Instrumentation for remote monitoring of radiation levels in each work area should be provided. Monitoring and alarm functions for these instruments should be located in the command center. Movable remote monitors should be placed at pre-selected locations in containment prior to any disassembly operation which has potential of changing work area radiation levels. The use of remote readout dosimeters for work crews should be investigated. A detailed discussion of radiation monitoring instrumentation is provided in Section 6.2.

TABLE 3.1-1

SECONDARY SIDE CHEMISTRY CONDITIONS

pH at 77 F	9.5-10.5
Ammonia, ppm	10 initially, 2-20 range
Hydrazine, ppm	100 initially, 50-200 range
Cation conductivity, μmho/cm	10 max
Sodium, ppm	2 max
Chlorides, ppm	1 max

TABLE 3.1-2

FEEDWATER SYSTEM CHEMISTRY CONDITIONS

pH at 77 F	9.5-10.5
Ammonia, ppm	10 initially, 2-20 range
Hydrazine, ppm	100 initially, 50-200 range
Chloride, ppm max	1 max

TABLE 3.1-3

RCS CHEMISTRY CONDITIONS

pH at 77 F	7.5
Boron concentration, ppm	3000-4000
Hydrogen content, std cc/kg	5-15

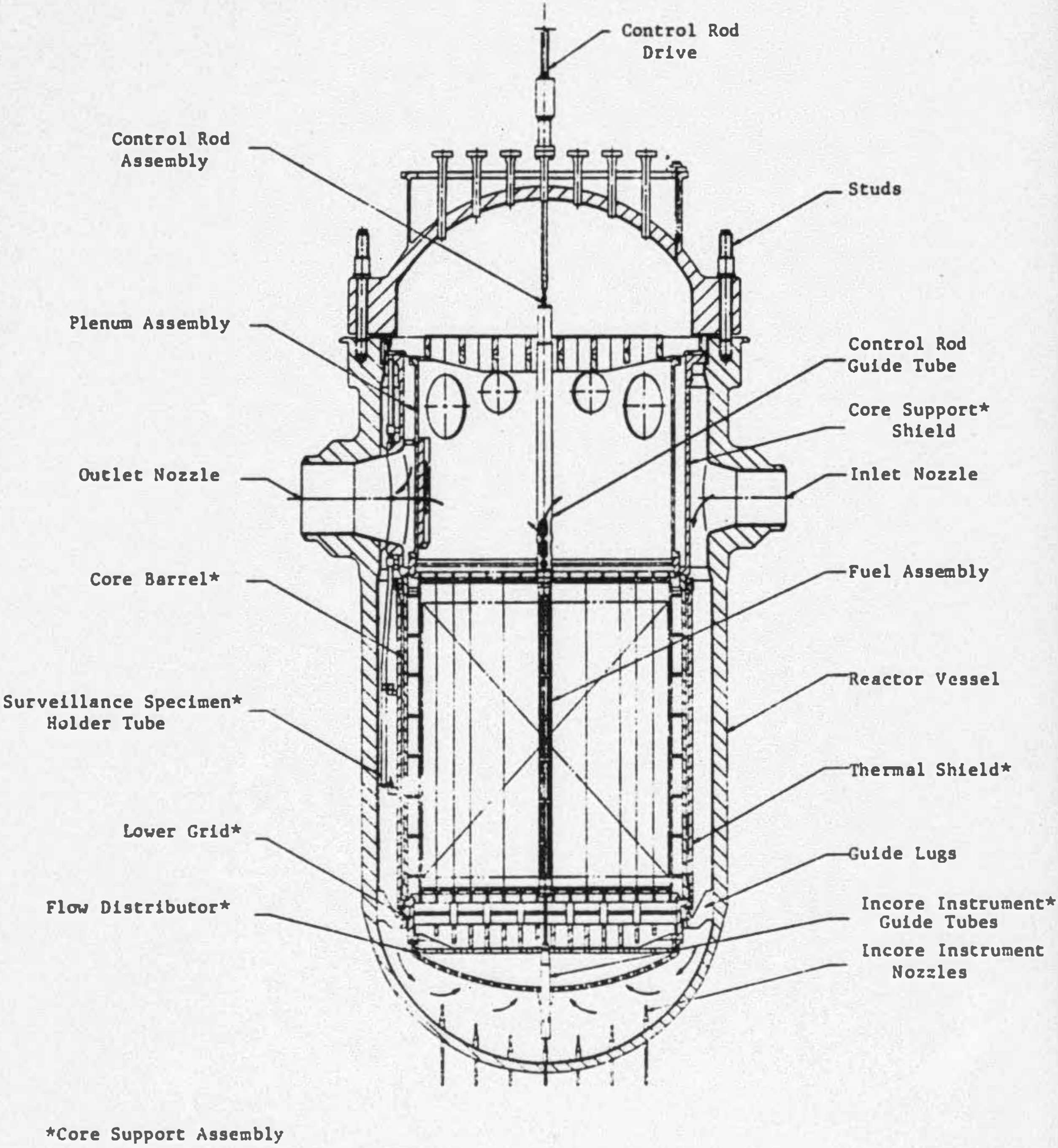


FIGURE 3.1–1
REACTOR VESSEL ASSEMBLY

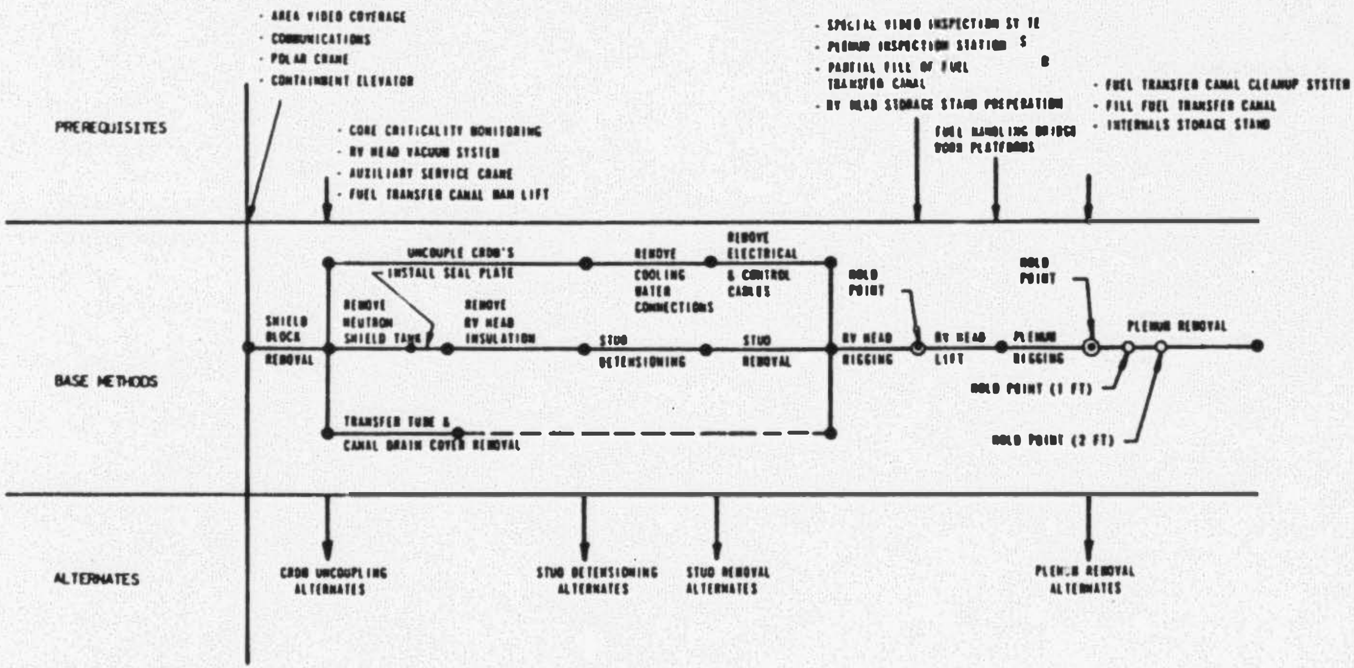


FIGURE 3.1–2
REACTOR DISASSEMBLY EVENTS LOGIC SEQUENCE

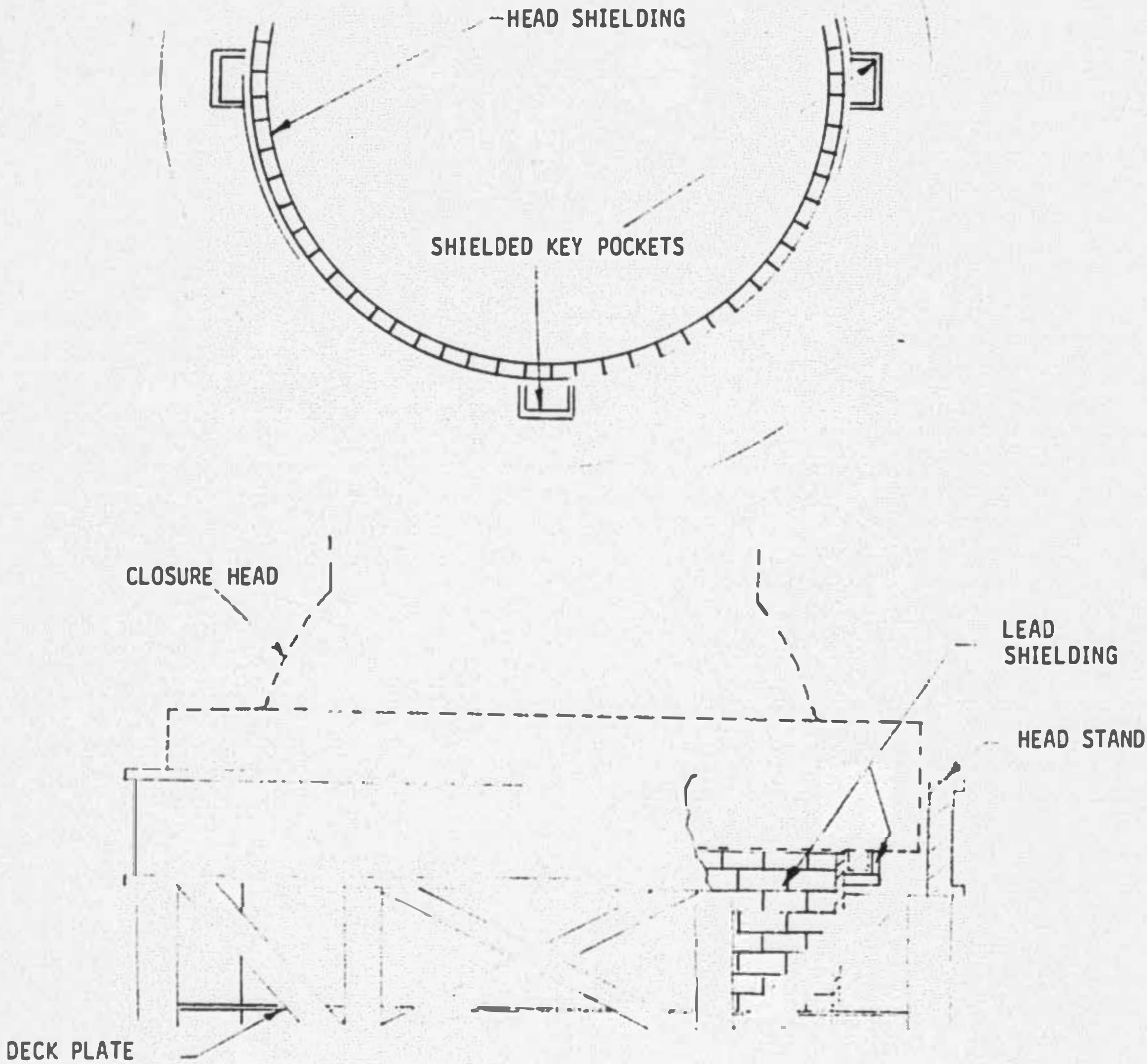


FIGURE 3.2–1
REACTOR VESSEL HEAD STORAGE STAND SHIELDING

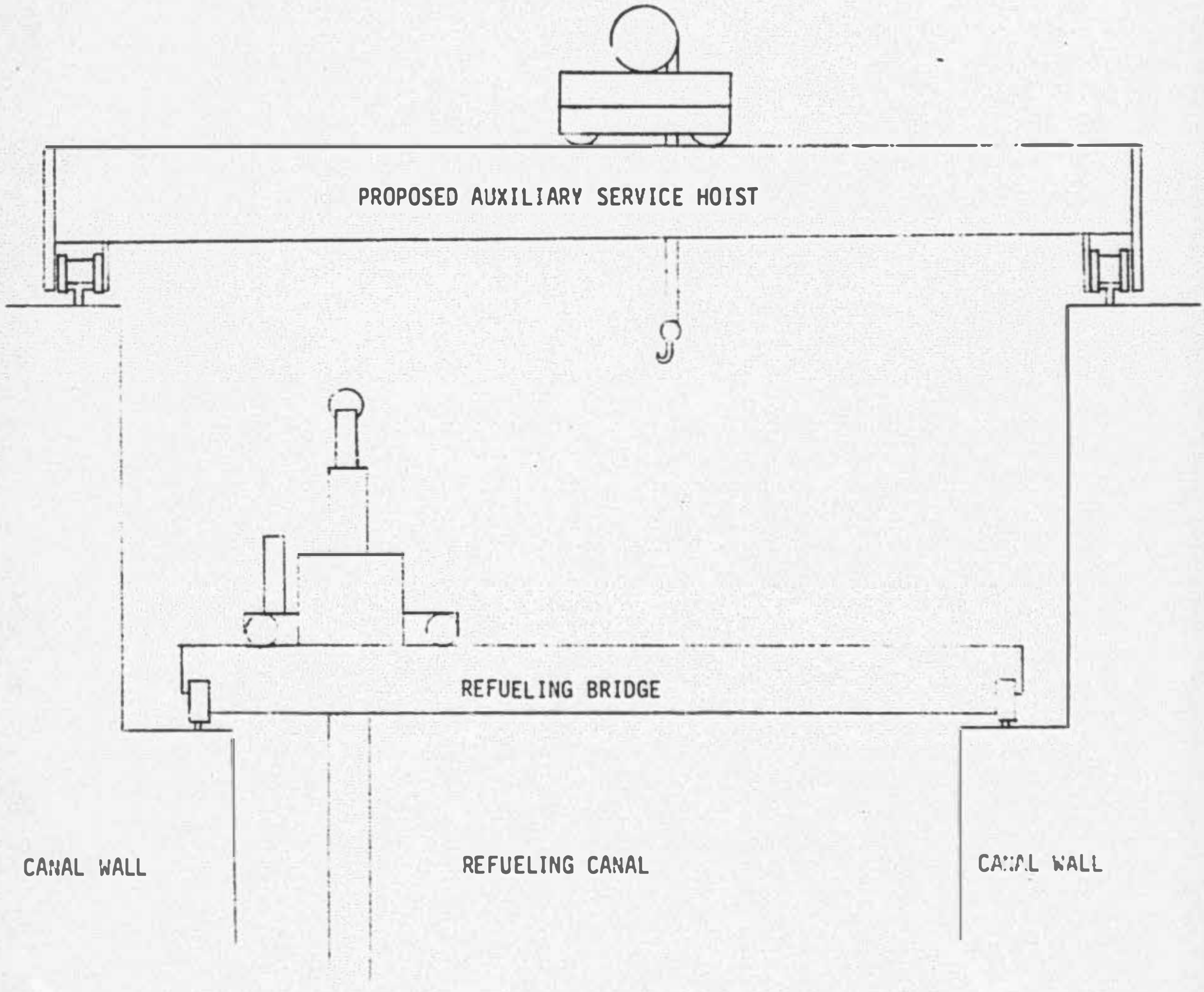


FIGURE 3.2–2
PROPOSED AUXILIARY CRANE

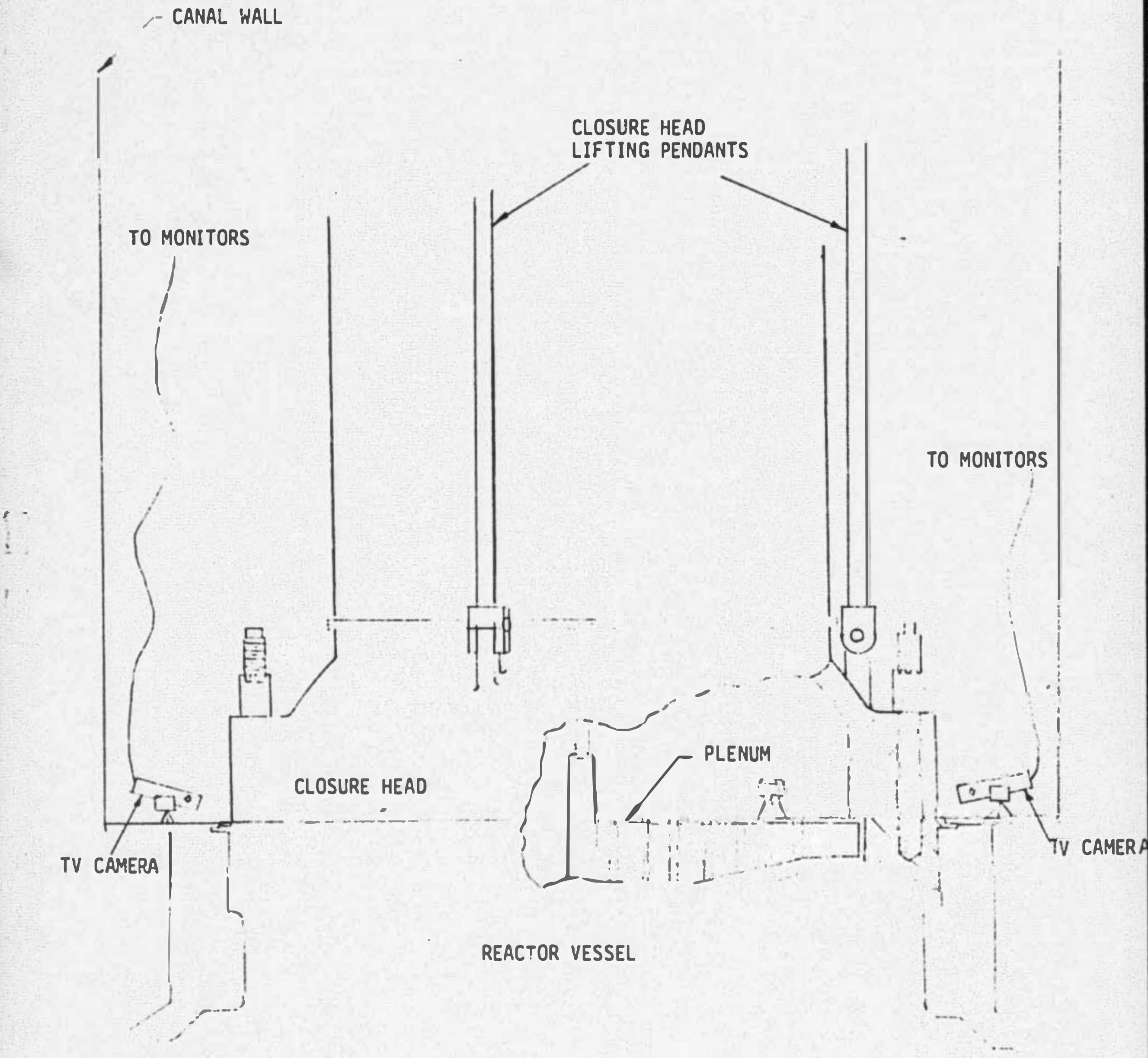


FIGURE 3.2–3
SUPPORT VIDEO SYSTEM

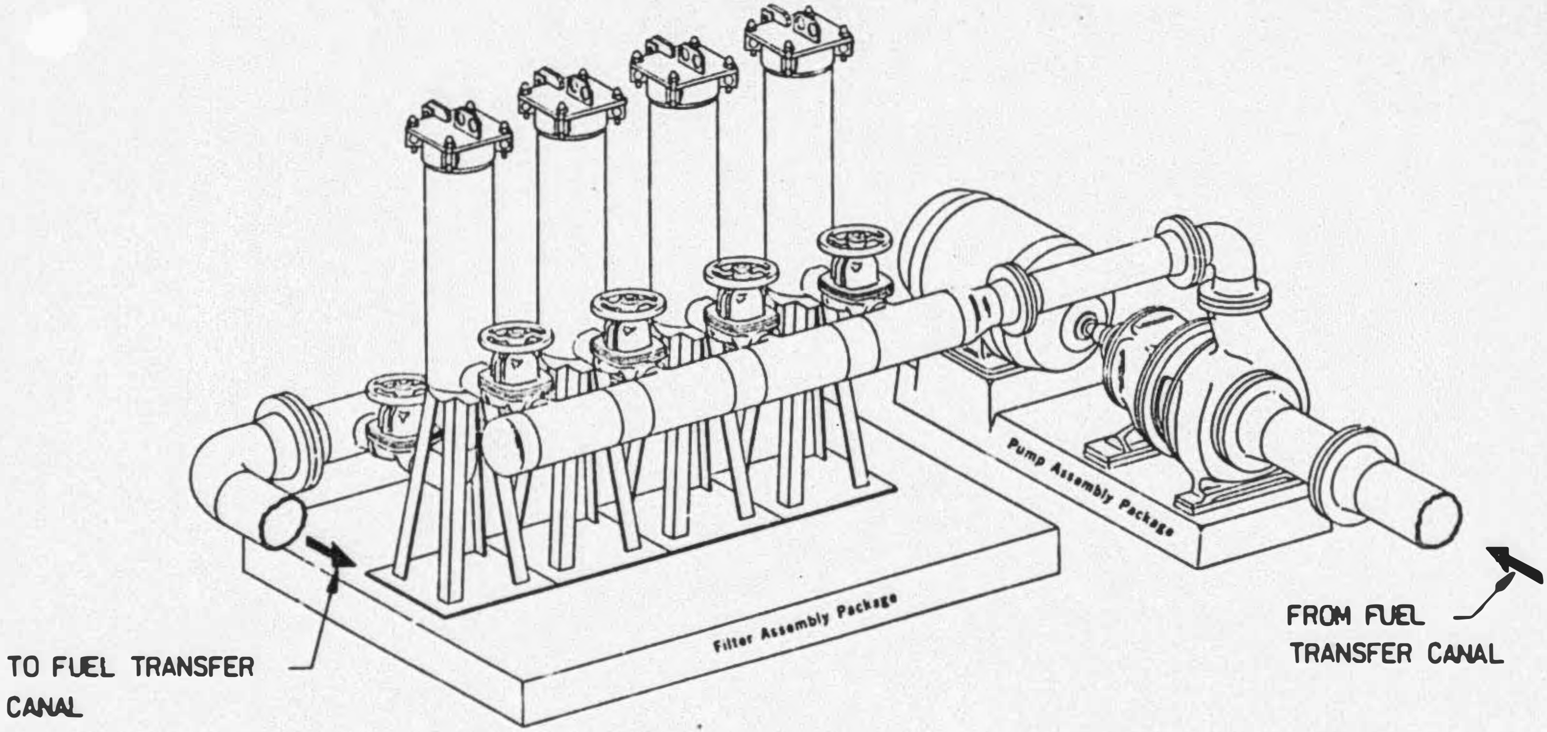


FIGURE 3.2–4
FUEL TRANSFER CANAL CLEANUP SYSTEM

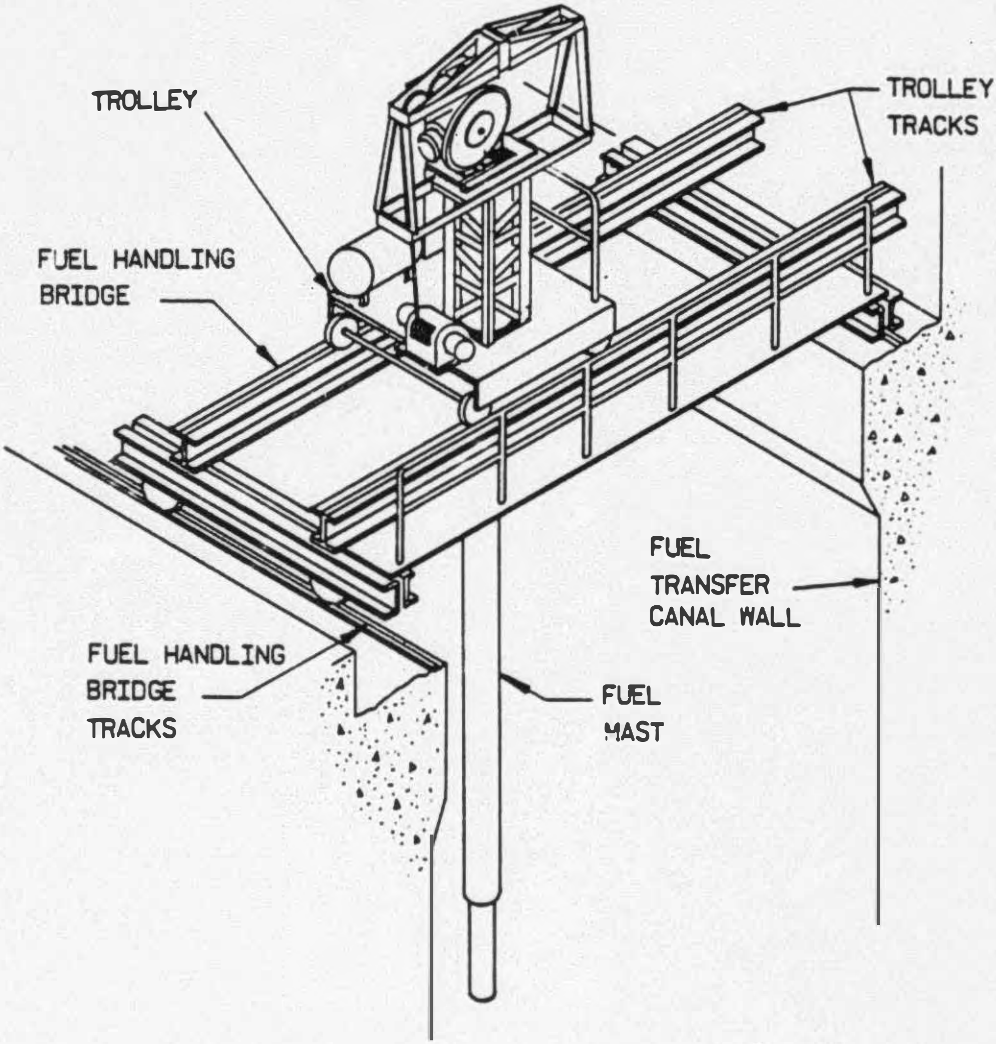


FIGURE 3.2–5
AUXILIARY FUEL HANDLING BRIDGE

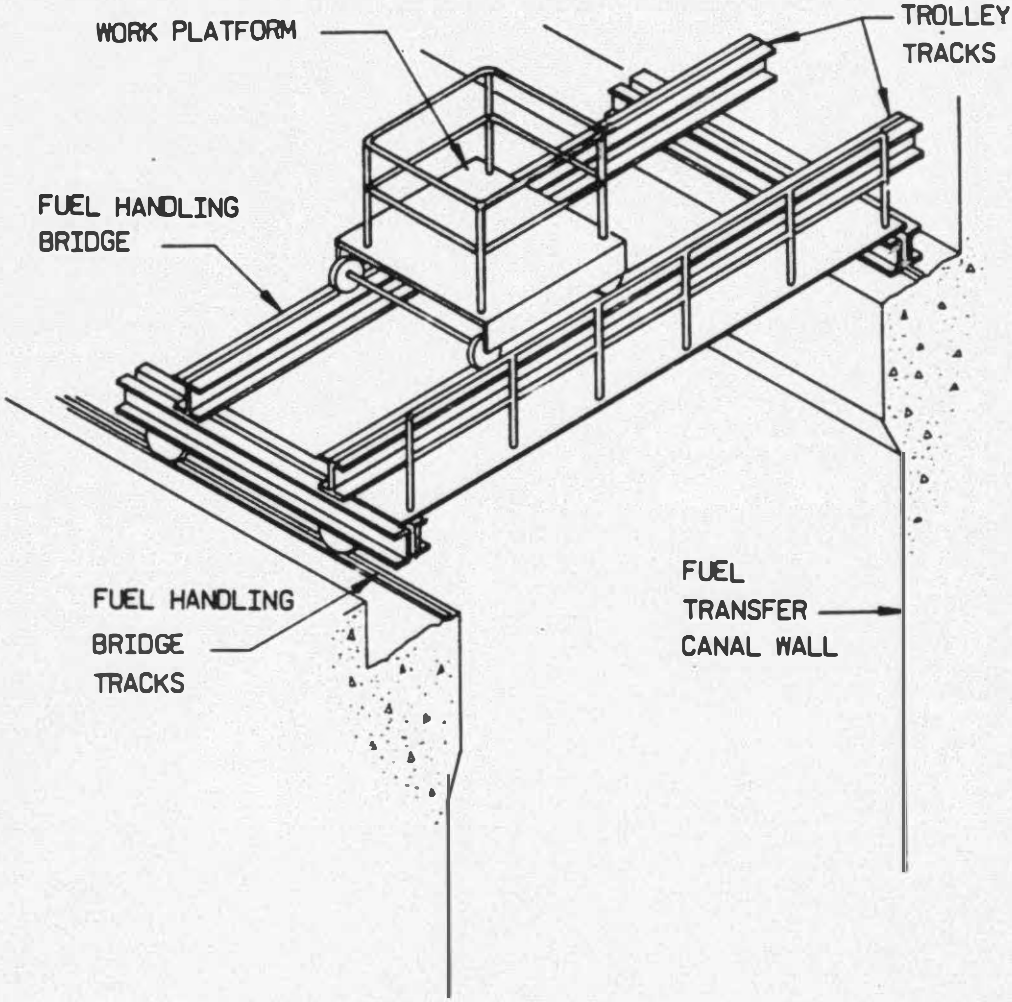


FIGURE 3.2–6
MODIFIED AUXILIARY FUEL HANDLING BRIDGE

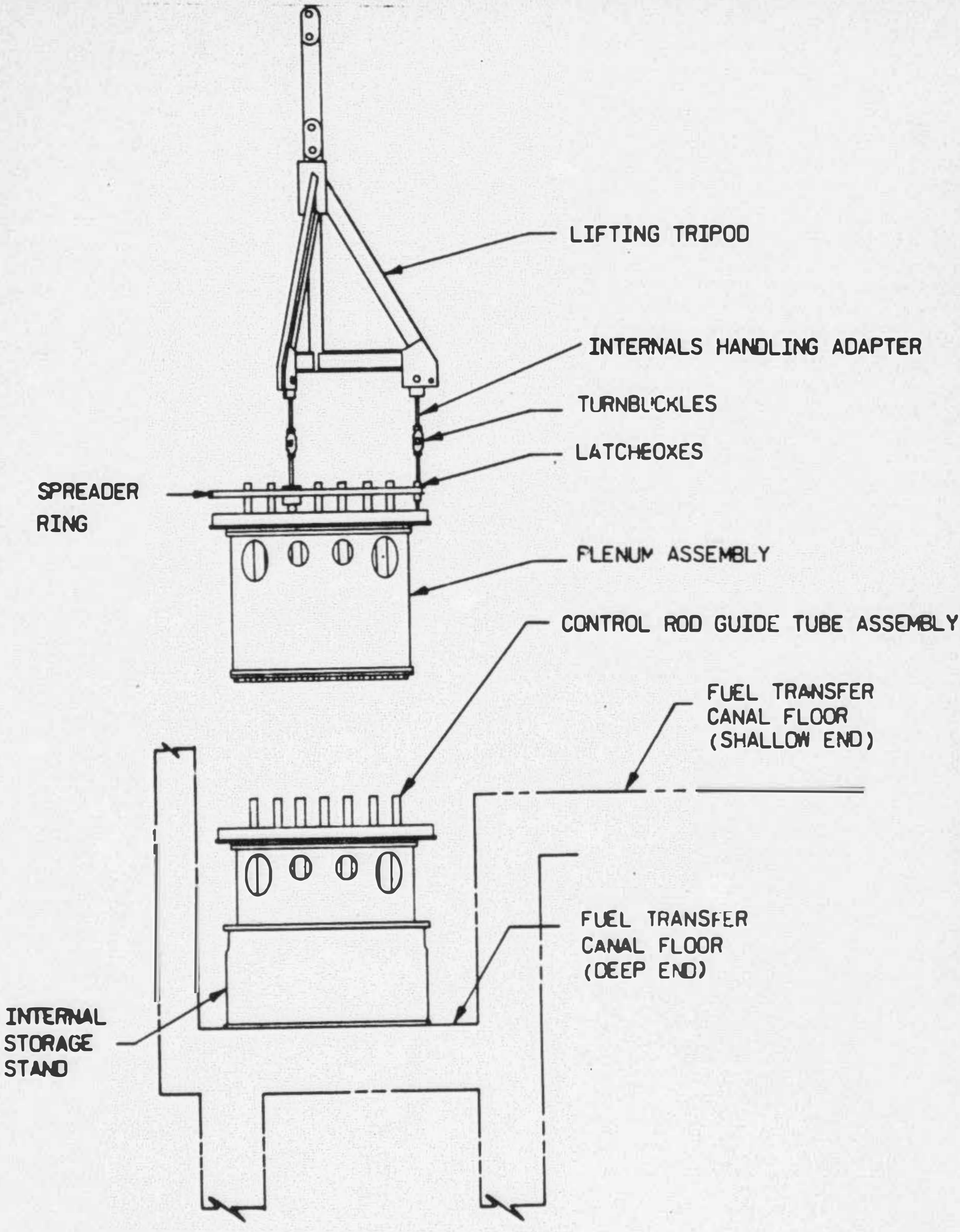


FIGURE 3.2–7
PLENUM ASSEMBLY STORAGE AREA

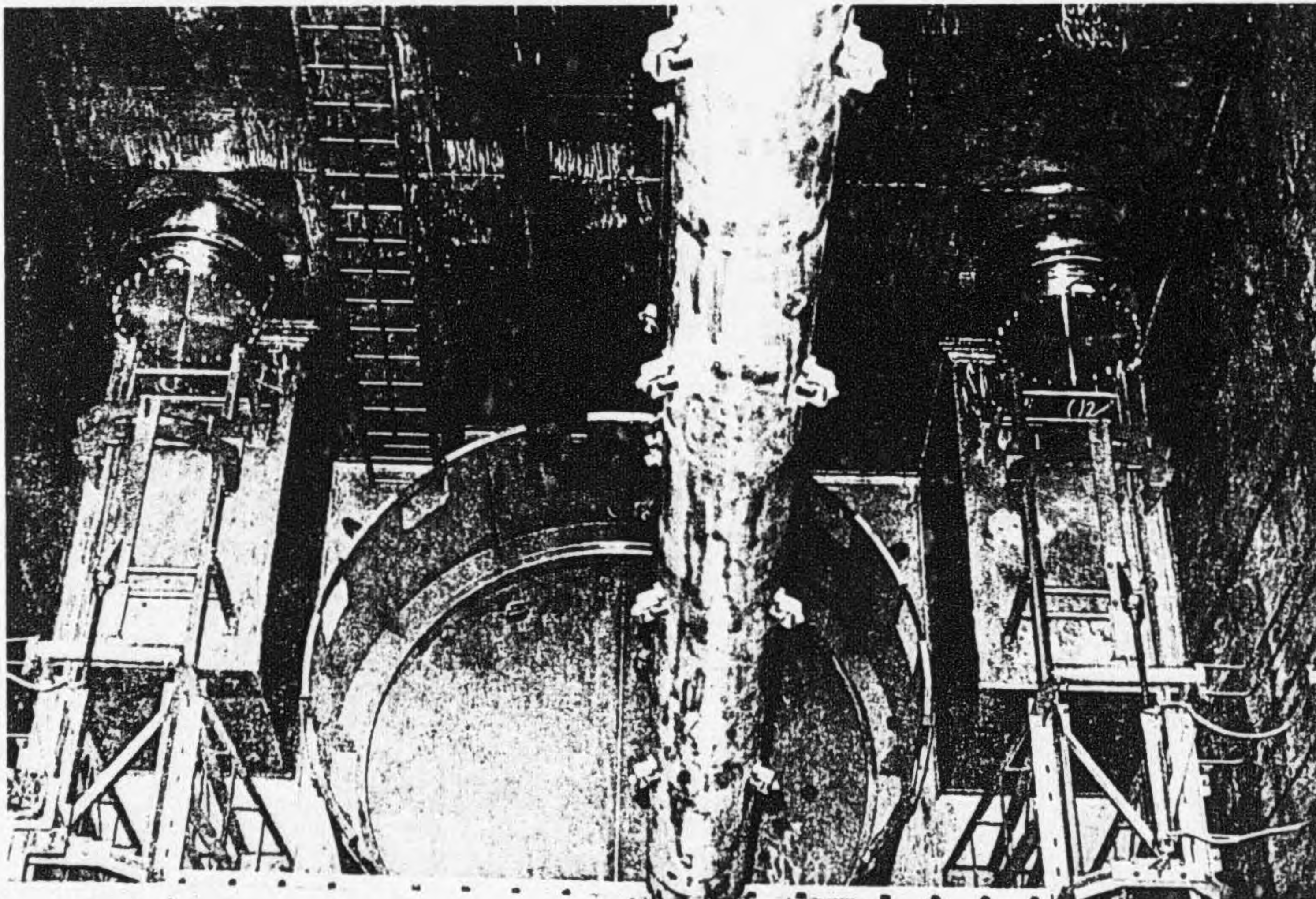


FIGURE 3.2–8
INTERNALS STORAGE STAND IN PLACE IN DEEP END
OF FUEL TRANSFER CANAL

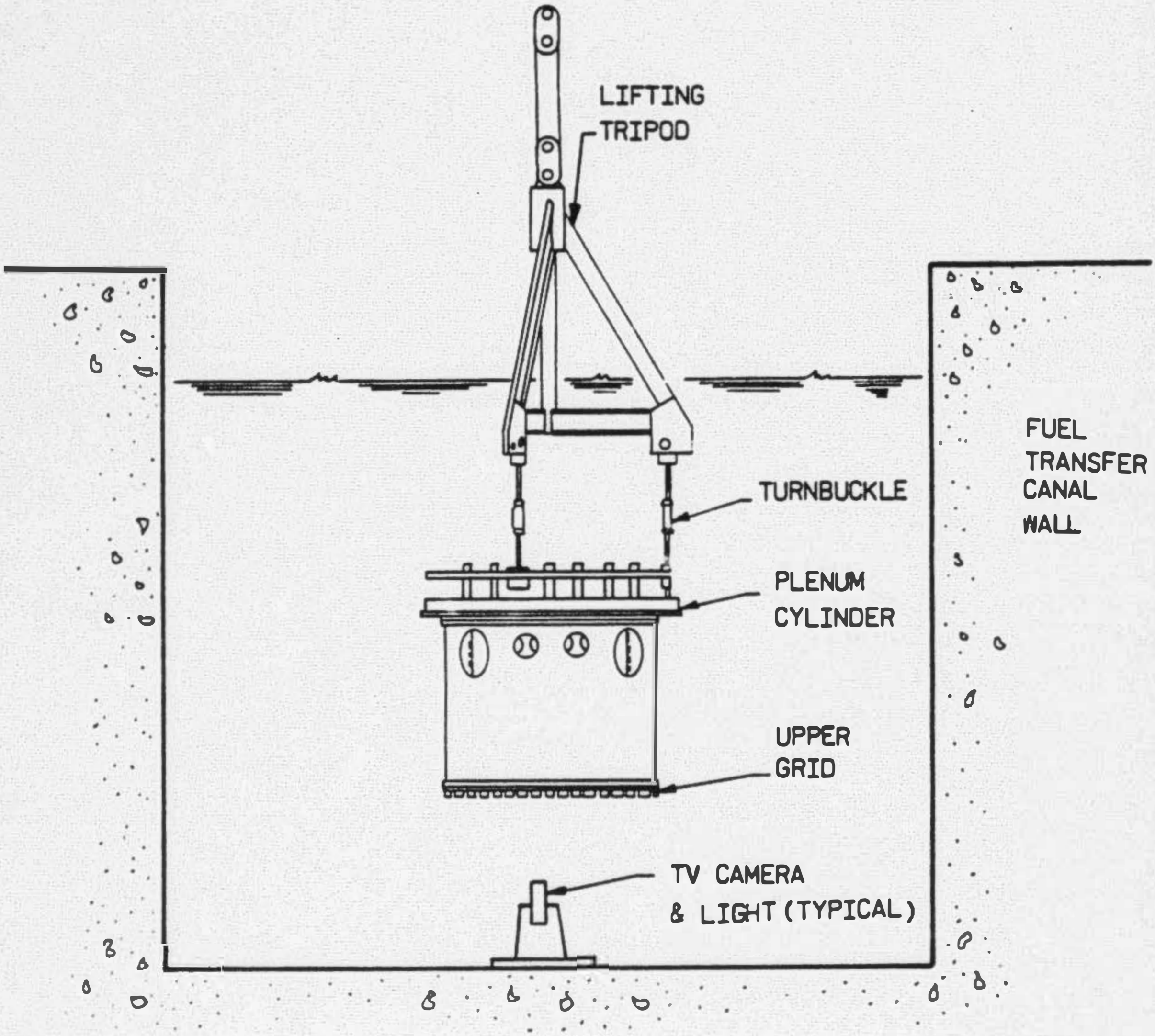


FIGURE 3.2–9
PLENUM INSPECTION STATION

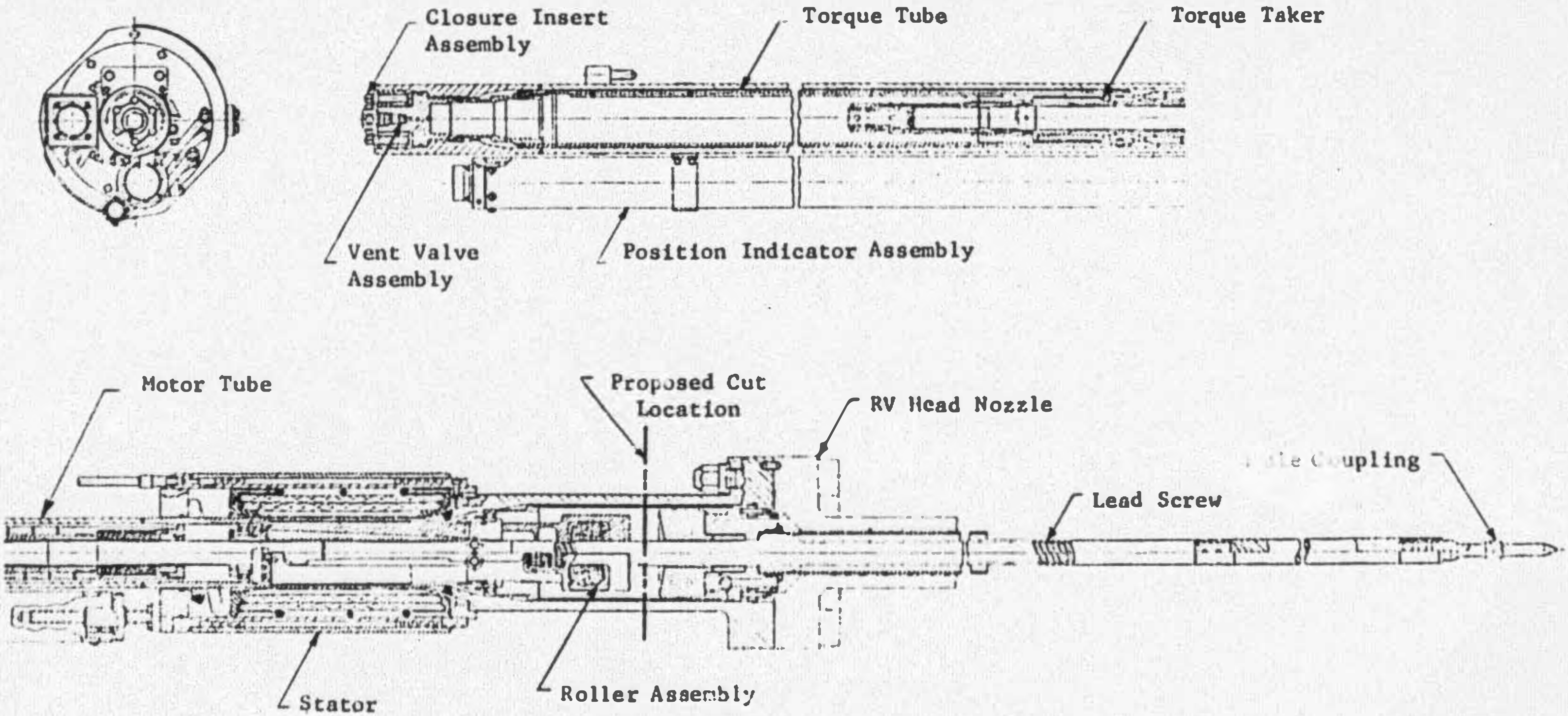


FIGURE 3.3–1
CONTROL ROD DRIVE MECHANISM

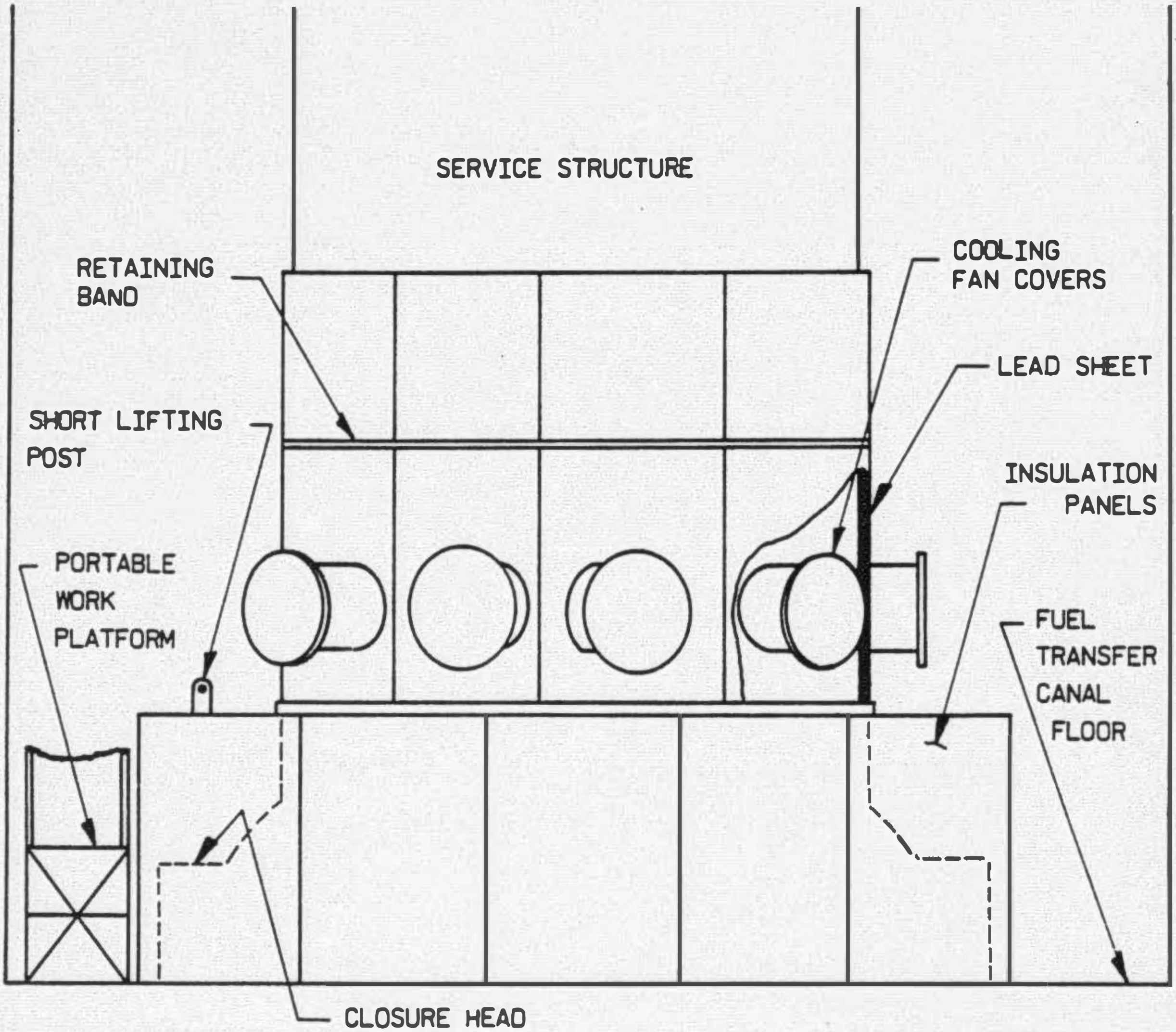


FIGURE 3.4—1
CLOSURE HEAD INSULATION REMOVAL

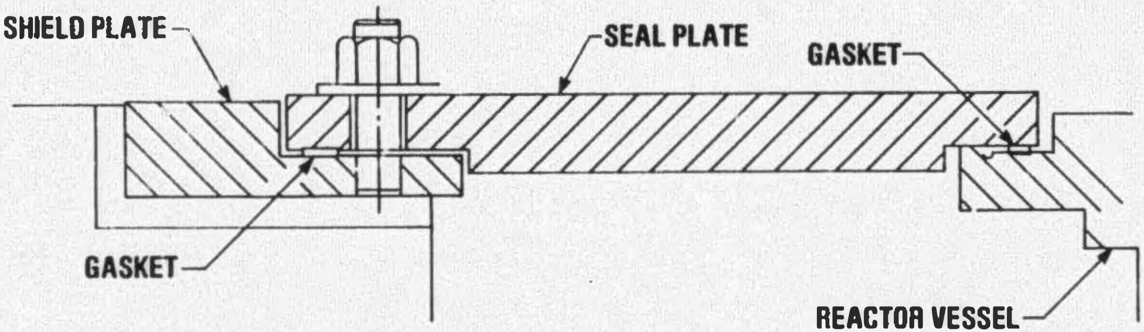


FIGURE 3.4–2
CANAL SEAL PLATE

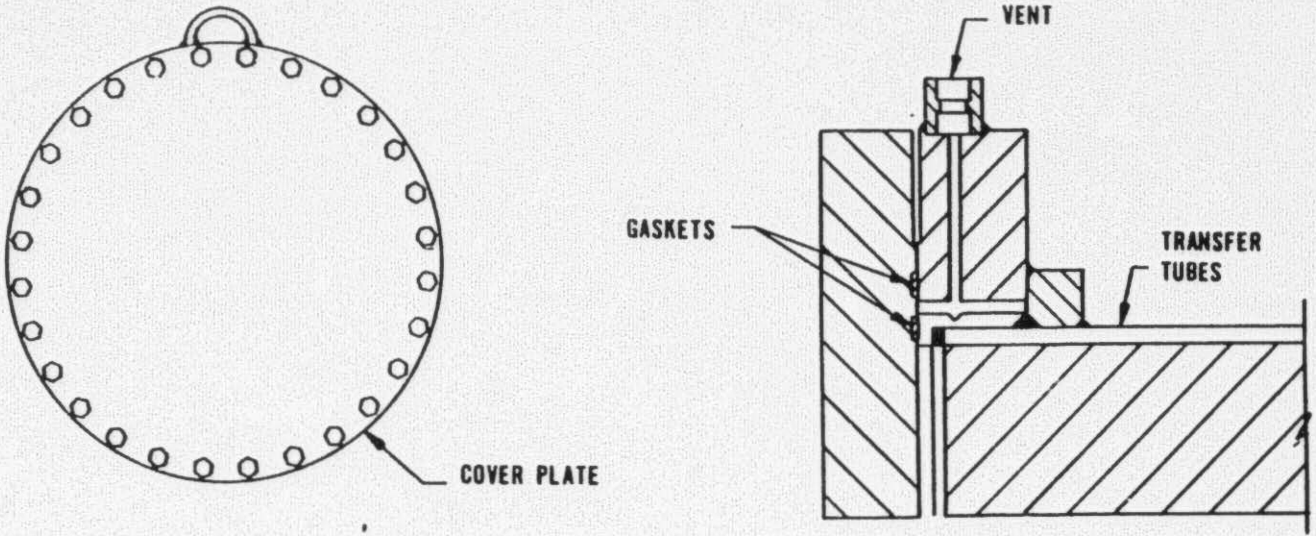


FIGURE 3.4–3
TRANSFER TUBE COVER

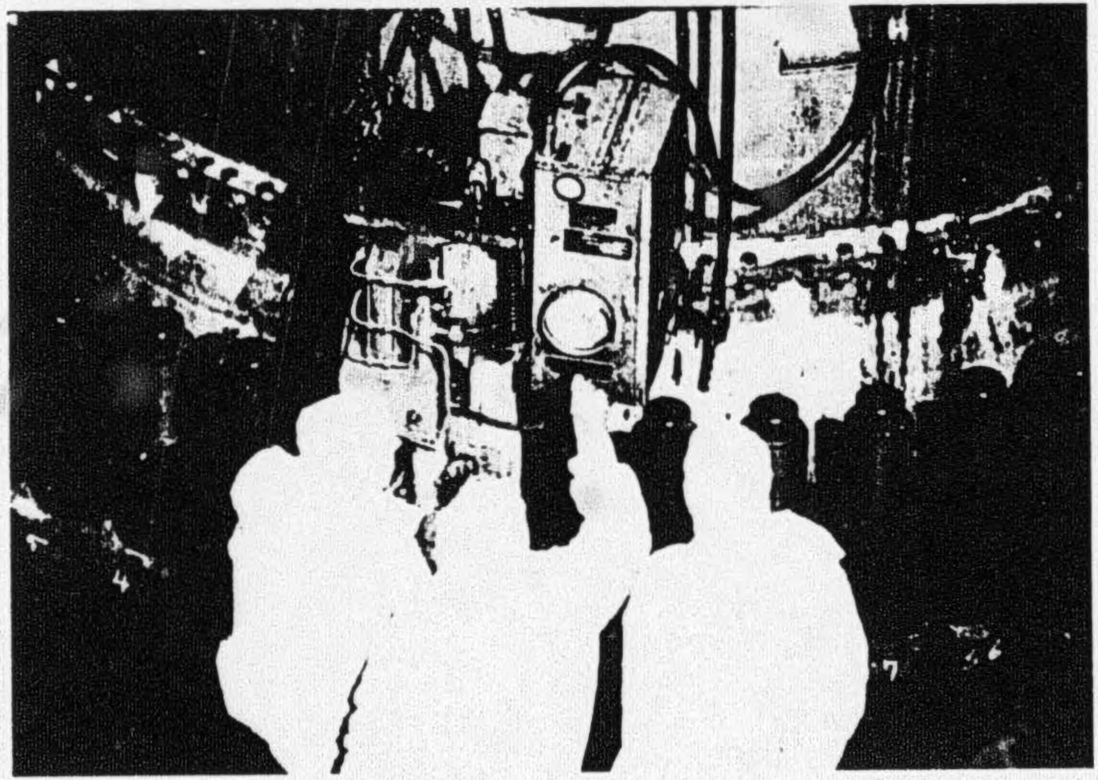


FIGURE 3.5–1
STUD TENSIONER OPERATIONS (TYPICAL)

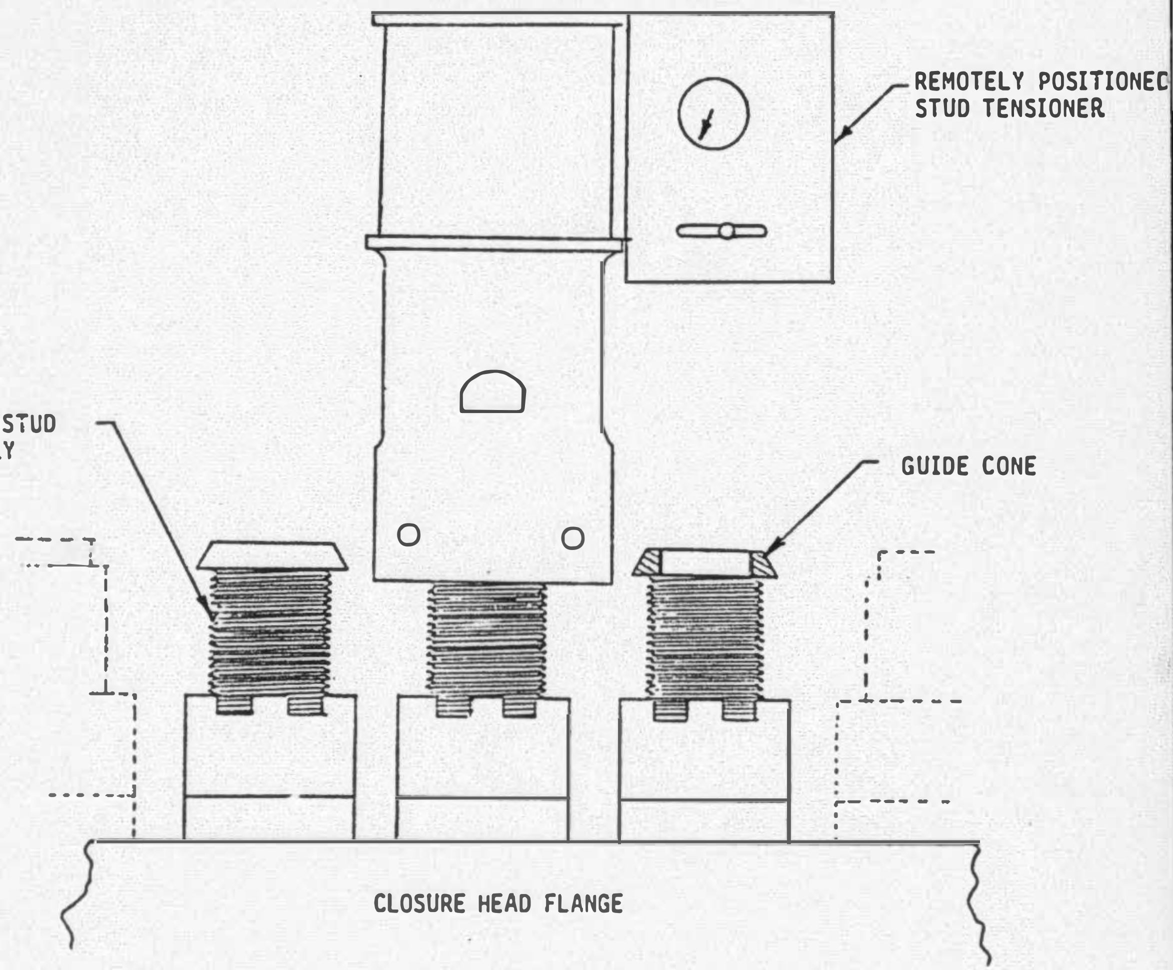


FIGURE 3.5–2
REMOTELY POSITIONED STUD TENSIONER

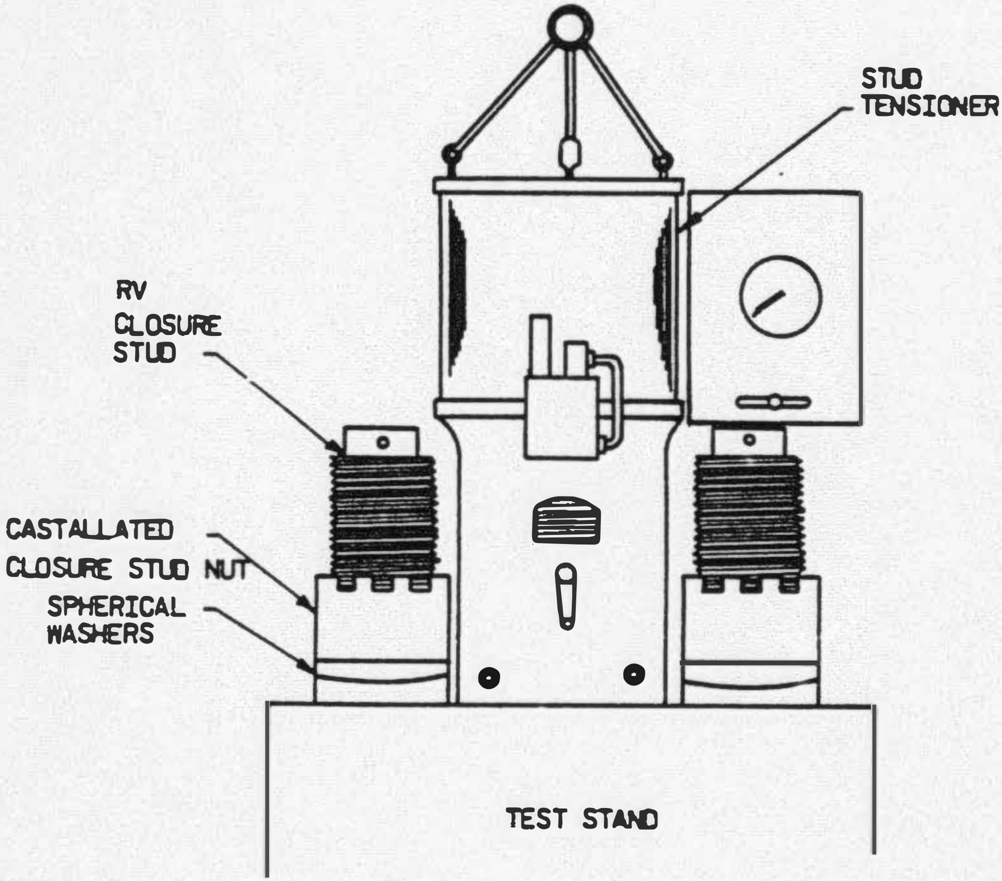


FIGURE 3.5–3
STUD TENSIONER TRAINING AND TEST STAND

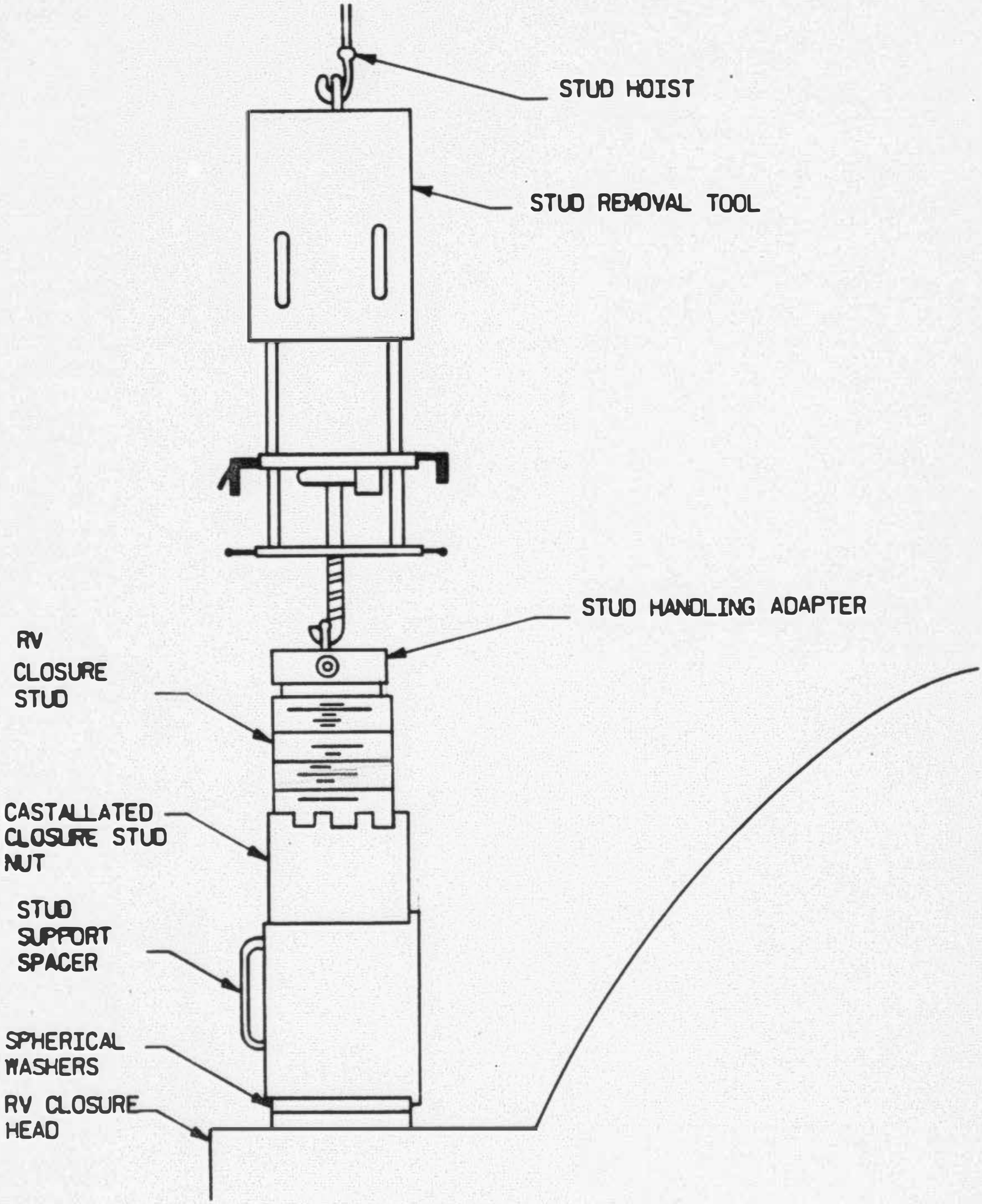


FIGURE 3.5–4
STUD REMOVAL TOOL

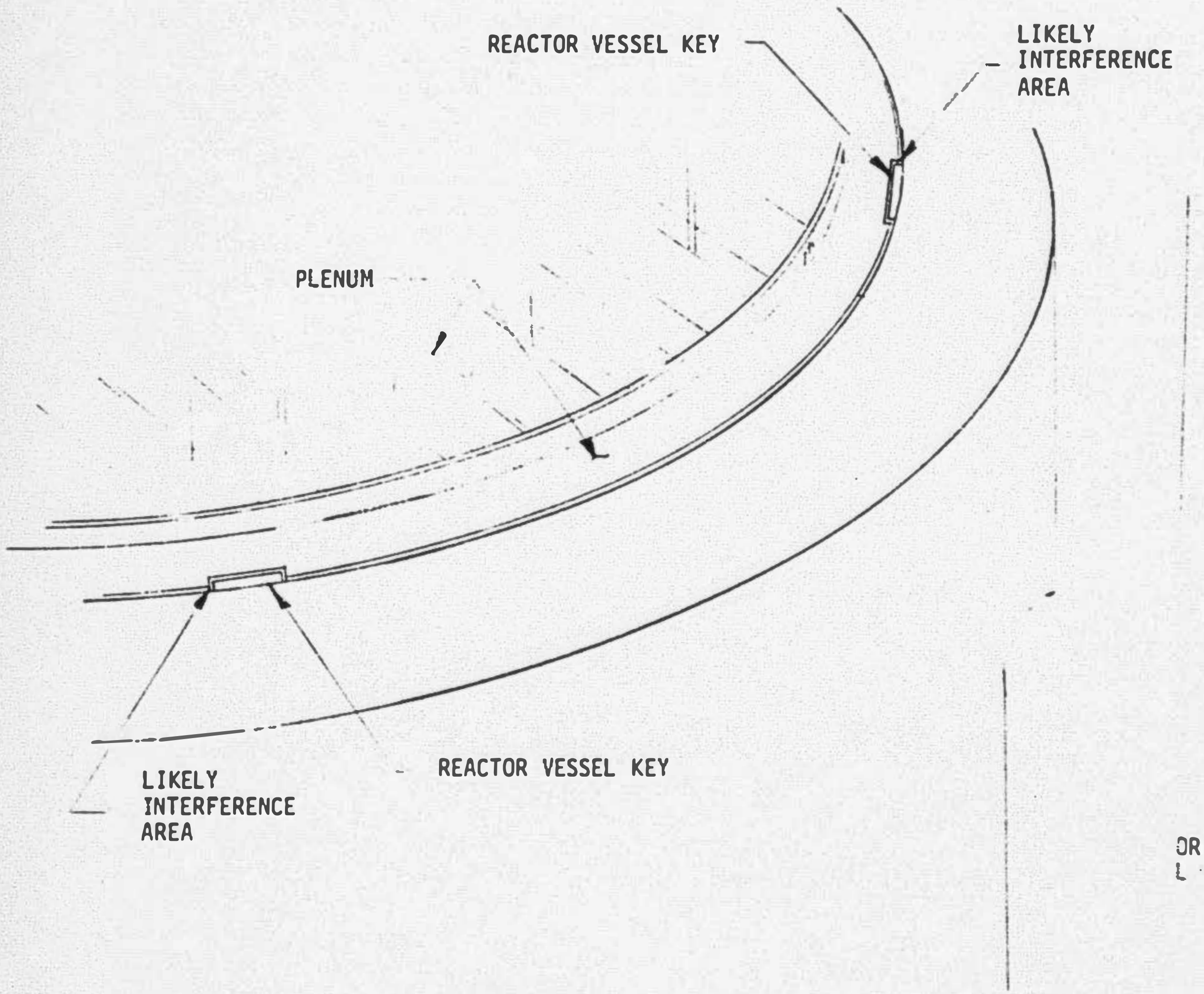


FIGURE 3.6–1
PLENUM INTERFERENCES

DISCIPLINE/TRAINING SUBJECTS	SITE ACCESS HP & SECURITY	HP & RAD. CONTROL FOR CONTAINMENT CONDITIONS	REACTOR DISASSEMBLY PLAN	INDIVIDUAL TASK	MOCKUP
Managers & Supervisors	X	X	X		X
Technical Support Engineers	X		X		X
Video Engineers	X	X		X	X
HP Technicians	X	X		X	X
Craft Labor	X	X		X	X

FIGURE 3.11–1
TRAINING PLAN MATRIX

4.0 CORE INSPECTION AND FUEL REMOVAL

4.1 Introduction

This section of the Planning Study addresses the activities, equipment, and concepts necessary to support the defueling operation at TMI-2. As a planning document, it provides only general guidelines. Detailed work procedures and equipment specifications will be developed in the technical planning of each work event required to achieve reactor defueling. The areas examined and discussed in the following text include:

- A General Description of the Core Damage - The core condition has been assessed and classified into 3 categories representing minor to severe damage. The definitions of the 3 levels and removal methods are discussed for each. Methods range from near normal removal for lightly damaged assemblies to dissection and removal of fragments for severely damaged fuel.

- Special Considerations for Criticality and Nuclear Material Accountability - Parametric and geometric control of fuel reactivity are discussed and recommendations for supportive studies are presented. Detectors and sources required for criticality monitoring are discussed. Requirements and problems to be addressed in the inventory of the special nuclear material are outlined. Ways to locate and retrieve nuclear material include the use of optical inspection equipment and mechanical techniques.

- Special Tooling Needs - Depending on the actual condition of the core, various kinds of tools will be required for removing components and debris. Recommendations are made for devices that will aid fuel removal and minimize personnel exposure. Among the equipment recommendations are suction pumps, filters, hoists, pole-mounted gripping and measuring devices, saws, torches, assembly removal devices, TV cameras, and boroscopes. Personnel training in special tool manipulation is suggested.

- In Place Inspection Plan - The desirability of an early determination of actual core damage to prevent impact on tool development is emphasized. Prior to closure head removal it is proposed that a partial assessment be made by lowering visual monitoring equipment through the CRDM nozzles or other access points. After closure head removal prior to the plenum lift, additional thorough visual inspections are envisioned and a detailed inspection is needed after plenum removal. Two viewing centers (one local and one remote) are proposed with the remote center having recording capability.

- Facility Modifications for Fuel Movement and Inspection -Recommendations are made for changes in the fuel transfer system drive to a cable drive and above water winches to be located in the fuel storage building to improve reliability and maintainability. A discussion of assembly inspection during removal by video means is discussed. A method is also proposed for scanning fuel prior to removal from the core. An inspection station is recommended.

- Consideration for Fuel Canning and Storage - Suggestions for the design of fuel containers which will be compatible with the damage level of the fuel and with special fuel removal tools is given. Side loading cans and top loading cans are required to accommodate badly damaged assemblies, yet they must be able to fit existing equipment such as upenders, transfer tubes, storage racks, and casks.

  Considerations for removing and placing assemblies into cans are given. Concerns are with weakened assemblies and gas releases. A discussion of patterns of fuel movement from the reactor vessel to either storage in the fuel pool or shipment in casks is presented. Problems with crane access and floor loadings with cask usage may arise.

- Contamination Control - Methods to reduce radiation and personnel exposure are discussed. Among the suggested methods are water surface skimmers, underwater washdown, and remote filter cartridge changeout. A radiation control area in which to repair tooling is suggested.

- Organizational Considerations - A discussion of the organization required for defueling is presented. Functions and responsibilities of key personnel are given.

- Study Recommendations - Studies recommended to provide insight into the equipment needs and safety aspects of defueling are listed in Table 4.1-1.

The defueling operation of TMI-2 will require development of new equipment, concepts, and procedures because of the extensive core damage which may have occurred. In order that these design and planning activities have the most reasonable estimates of the core mechanical condition, upon which to base design calculations and decisions, assessments of the mechanical integrity of the fuel assembly hardware should be continued and updated. The assessments should evaluate fuel assembly component condition as a function of radial and axial position in the core, using transient time and temperature data which includes and brackets the transient believed to have occurred at TMI-2. This would include updated assessments of the damage in centrally located high-power fuel assemblies, intermediate-power fuel assemblies, and lower power fuel assemblies on the core periphery. The latter are believed to be less extensively damaged. The specific areas to be evaluated would include:

- guide tube oxidation and hydriding; mechanical properties in the oxidized condition.

- fuel rod oxidation and hydriding; mechanical properties in the oxidized condition.

- assessment of eutectic formations between the Zircaloy-4 fuel rods and Inconel 718 grid, Zircaloy-4 guide tube and type 304 stainless steel guide tube nuts, and Ag-In-Cd (Control Component Poison Material) and type 304 stainless cladding.

- analysis of current incore thermocouple data which may provide supporting data for the damage prediction.

- CRD-CR coupling damage.

These areas of evaluation are considered critical in supporting design efforts for fuel assembly removal and disposal. Due to the expected varying levels of structural degradation to the fuel assemblies, the knowledge of the fuel assembly condition and load-bearing capabilities of the fuel assembly hardware are envisioned as being necessary for design of removal equipment, handling equipment, and supporting devices in storage or shipping containers. These assessments would be made using available industry-wide data on oxidation and hydriding behavior of Zircaloy, mechanical properties of oxidized Zircaloy-4 tubing, eutectic reactions, and thermal relaxation. However, it is anticipated that some of this information may not be readily available or applicable for the time and temperature conditions under consideration.

In some cases, small-scale bench experiments may be useful in providing needed data. Where larger scale experiments are required to provide the missing information, the schedule for removal operations may preclude obtaining this data in a timely manner. Where data cannot be obtained, worst case conditions should be assumed in equipment design and procedure development.

For the purpose of the planning study the core condition assumed was:

- fuel assemblies are oxidized to the degree that conventional handling methods are unacceptable.

- Peripheral assemblies may be physically intact. However, the peripheral assemblies may also be damaged to the extent that they cannot be removed by lifting either from the top or bottom.

- central core assemblies have sustained significant physical damage, upper end fittings missing, disintegration of fuel rods and guide tubes down to some intermediate grids, shards of cladding, fuel pellets, etc., in a random layer over the intact portion of the core.

- CRAs, LBPRAs, APSRs, etc., are inserted and may be either mostly intact or quite severely damaged.

Other core damage assessments have been produced which differ from the foregoing. Continuing studies may result in modified core damage assessments. Some recent NRC studies (Reference 1) suggest extensive damage over the top portion of the total core with a crust of molten Zr-ZrO eutectic physically bonding parts of the core together. Preliminary NSAC studies (Reference 2) suggest somewhat less damage than the NRC studies do.

Current incore thermocouple data indicates that the degree of core damage may be less than that indicated by the NRC studies.

4.2 Criticality Monitoring and Control

4.2.1 Criticality Control

The prevention of an accidental criticality during handling of the damaged TMI-2 fuel requires complete understanding of the critical parameters of the fuel and careful assimilation of these critical data into the design of all processes and equipment involved.

The TMI-2 core is a three-batch fuel loading containing three specific uranium enrichments. The distribution in the original core was as follows:

- 56 fuel assemblies with 1.98 w/o U-235
- 61 fuel assemblied with 2.64 w/o U-235
- 60 fuel assemblies with 2.96 w/o U-235

The core exposure at the time of the incident was 94.6 effective-full-power days. Calculations should be performed for the actual core conditions to establish an upper bound on criticality limits.

The parameters by which criticality may be controlled include limits on the size and the shapes of containers of fissile material (called geometry controls), limits on the amount of moderation permitted, and the addition of neutron absorbers (called poisons). For most applications, it is necessary to control a combination of these parameters and this will, no doubt, be the case in the TMI-2 core. The interaction between the various parameters is sufficiently complex that careful studies must be made of the effect of each of these parameters on the others. Studies should be performed for different conditions of reflections, isotopic composition, geometry, and poison material. It may be desirable to include other poison materials since some accident conditions are difficult to control with boron alone as a poison. However, initial analyses (Reference 3) indicate that criticality events are improbable with boron concentrations of 3500 ppm or greater. Previous analyses used to establish current technical specifications should be reviewed prior to defueling.

The generic data described above can be used in the design of processes and equipment for the handling of fuel material which is no longer restrained by fuel pins or fuel assemblies. It is most important that liaison be maintained between the various groups involved to see that nuclear criticality safety requirements are included in all designs.

It is possible for containers which are individually subcritical to constitute a critical system through the exchange of neutrons. The handling and storage of containers of fuel must take this face into account. Additional calculations of this interaction effect will be required and appropriate liaison must be maintained to see that storage rack designs consider the appropriate data.

Fuel assemblies which still contain fuel but which have been extensively damaged may constitute still another criticality safety problem. For example, bowed fuel pins can change the moderation present within an assembly and thereby change the reactivity of a single assembly. Such possible situations must be considered and prepared for. This preparation could take the form of calculations which determine the reactivity of a fuel assembly under conditions of variable pin pitch.

Possible dispersion of fuel by the fuel unloading processes should be considered. If the unloading processes can cause fuel to move from the core region, other possible locations within the reactor system should be considered and appropriate precautions taken. The generic data described earlier will be appropriate for this consideration.

The fuel material from damaged or destroyed fuel pins may be retained in locations (or in grids) where accidental movement could cause a more reactive configuration. Therefore, data should be generated to determine the reactivity of the system as a function of the amount of damage and the size of these accumulations. Using these data, decisions can be made regarding the increase in reactivity which might result from unloading activities. Provisions can then be made to control the possible increase in reactivity. Methods for mechanically inserting poison material into the fuel assemblies or regions of debris should be analyzed.

It is important to a safe defueling of TMI-2 that the above information be generated before any operations are undertaken.

The study would collect and categorize all previous studies performed on criticality potential for TMI-2. The analyses should address all potential critical geometries (i.e., slabs, spheres, and cylinders) for various fuel arrays. This would include loose debris, intact fuel assemblies, fused material, etc., based on the core damage assessment.

4.2.2 Criticality Monitoring

Difficult problems will exist in attempting to monitor the reactivity of the core during disassembly. Prior to head removal, plans should be considered to insert a fission chamber into fixed incore detector locations to monitor the reactivity of the system. This would necessitate removal and disposal of one or more existing incores. Another possibility is to lower the fission chamber through a CRDM nozzle or CRD guide tube to monitor the upper fuel region.

In monitoring the reactivity of any system, the location of the sources and detectors is of primary interest. Poor location of sources and detectors relative to the fissile material can mask real changes in reactivity or cause false indications of a reactivity change. A great deal of thought and analysis must go into location of the sources and detectors if the resulting measurements are to be indicative of the reactivity of the system.

The instrumentation for this monitoring will be fission chambers and associated electronics. Neutron sources such as polonium-berylium or americium-berylium may also be required.

Information should be obtained on the condition of the startup instrumentation channels of the reactor. These may be very useful in conjunction with the proposed new hardware described above.

Before disassembly of the core is started and at frequent stops as disassembly proceeds, the effective multiplication of the core should be determined. In this way, operations which may inadvertently increase the reactivity can be detected and stopped or modified. A technique for measuring the effective multiplication of the system should be determined.

The unloading technique which is least likely to result in reactivity increases and which is most compatible with the required measurements should be developed. The precise order in which steps are to be carried out should be carefully thought out and documented. Such preplanning will allow the maximum information to be developed at each stage in the process and will go far toward assuring a smooth operation.

4.3 Special Material Accountability

Since nuclear power reactor operators do not normally measure special nuclear material at a reactor site except by individual fuel assemblies, accounting requirements for a nuclear power reactor station are relatively minimal when compared to a fuel manufacturing plant, as reflected in Federal Regulations 10 CFR 70.51(b), (c), and (d). The requirements at a reactor station are based upon security and item control (each fuel bundle and its fuel rods). The manufacturers' values for special nuclear materials are accepted for the contained fuel in each received bundle; the amount of fuel in a discharged bundle is based on calculations reflecting the estimated burnup to which the bundle has been subjected. Final accounting reconciliation can only come after fuel recovery which is not now permitted. Reactor operators are required to inventory (by piece unit) annually and maintain procedures "sufficient to enable the licensee to account for the special nuclear material in his possession under license."

Special material accounting for the TMI-2 recovery must be reflected in plant procedures which should be "sufficient" to enable accounting for the material possessed. Since the plant's Material Control and Accounting procedures are approved by the NRC, the test of what is sufficient is, in general, a negotiated item. The key question to be answered is, can the removed fuel be measured? It is assumed that the fuel can no longer be adequately accounted fo by item count of discrete components and supplemental procedures will have to be employed. It should be noted, however, that the highly radioactive nature of the TMI-2 core will on one hand limit the measuring possibility but will also restrict the possibility of diversion of the fuel.

The best available means of accounting for the fuel should be studied carefully. It should be recognized that if component mechanical failure is extensive, mechanical measurement techniques may not be sufficient to make a reliable assessment. Also, mechanical measurement during fuel assembly withdrawal may greatly hamper fuel movement and cause unnecessary exposure to personnel performing the work.

By knowing the weight of the core and recording the lift weight, a certain amount of accountability following removal can be obtained. Collection on filters and in other waste areas also will limit the effectiveness of any attempt to account for all the fuel which went into the assemblies. It may be necessary, therefore, to perform a detailed inspection of the reactor coolant system using video equipment or, if the system is "clean enough," a direct inspection may be necessary. The use of pole-mounted cameras, borescopes, fiber optics, and radiation detection equipment may be necessary. The site work preparation should ensure the capability to inspect the entire system, including dead legs such as drain lines.

4.4 Radiation and Contamination Control

Radiological safety may be reasonably assured through normal personnel access and stay time controls except for the accidental release of gaseous products from any remaining intact fuel rods that fail during the handling operation. It is expected that there are few intact rods remaining, but assurance of this condition, even when examination of the core is possible, is highly unlikely. Accordingly, all plans for movement of assemblies that are not failed should reflect an absolute minimum of handling operations from core to isolation canister. This approach will minimize the risk of a gas release to the containment.

Techniques for contamination control should provide for underwater cleanup of the plenum and other reactor internals, underwater cleanup of tooling used to move fuel, surface skimmer to remove radioactive materials which float, washdown of fuel assemblies and transfer cans underwater, and cleanup of reactor vessel using an underwater washing technique. A separate radiation control area should be established large enough to allow repair of tooling used for fuel movement. The system (Figure 4.4-1) used to provide for the underwater washdown and skimming must consider the following:

- Extremely high radiation fields around the filter will dictate remote (underwater) cartridge changeout,
- Design must consider critical geometry,
- Loading of used filter cartridges in fuel handling can which can be transferred to spent fuel pool,
- Size of equipment should be limited to ensure ease of disposal, since high contamination levels could prevent reuse.

4.5 Core Removal

4.5.1 Preparation for Core Removal

4.5.1.1 Tool Checkout and Training

Due to the possibility of higher than normal radiation levels, the defueling team should undergo a vigorous training program. This training program should include: anticipated fuel status, tooling familiarization and repair, a short course on fuel assembly construction in addition to practice in handling the tools during underwater operations.

In order to ascertain the effectiveness of various tools in handling fuel assembly parts and partial fuel assemblies, a checkout and training program should be performed. The tool checkout program would consist of operating the tools underwater to retrieve partial assemblies and assembly parts. Tool checkout could be performed at some convenient location such as the "B" spent fuel pool to train personnel and to checkout key disassembly procedures.

The training program would be attended by all personnel including video, health physics, and radiation monitoring. This large group of trained personnel would be required so that there will be the capability for most team members to participate in the hands-on defueling should additional personnel be required because of unforeseen circumstances, i.e., radiation monitoring and tracking personnel would be familiar with the tools, enabling them to operate them. This provides each shift with fuel handlers and backups.

4.5.1.2 Special Tool Requirements

Assuming the condition of the core to be as described in Section 4.1, a number of special tools not used in normal core removal will be needed.

To the extent that the assumed core conditions are substantiated by visual assessment, the following classes of tooling will be required. Tools should be designed for easy decontamination in the event of repair, modifications, or other required closed handling. Consideration should be given to producing these tools from materials which are good neutron poisons so that extra criticality control is provided by the presence of the tools.

- A means of debris removal is needed such as a suction pump equipped with storage canisters containing suitable strainers and filters to permit capture of the fragmented core material. The canisters must be designed to meet regulations on shipping, to be compatible with existing shipping casks, and to ensure nuclear criticality safety. They must also be compatible with handling equipment and storage/reprocessing

- Serve as a backup to the fuel bridges,

- Help in the handling of special fuel movement tools, and

- Capability for remote operation, as well as operation from the fuel handling bridge.

This crane would also be available to perform specific duties associated with the process of loading damaged fuel into cans, which is complicated by video and washdown requirements. This crane should have a 10-ton capacity (minimum). The overhand portion would require counter-balancing. The controls should consider the need for precision (slow) movement as well as for rapid movement to help minimize worker exposure time. The conflicting requirements for precision and speed may require a multispeed system.

The present projections of core damage preclude the use of the fuel-handling bridge with its mast. A simplified bridge using the rail system should be incorporated in place of the fuel-handling bridge. This type of system will provide a platform from which the inspection equipment and special handling tools can be used for maximum efficiency and safety.

4.5.2 In Place Core Inspection

An early determination of actual core damage is highly desirable to provide lead time necessary for the development of tools and techniques which will be required for core removal. Prior to reactor vessel closure head removal, it may be possible to make a partial assessment of the upper core condition by removing selected CRDMs and lowering visual monitoring equipment through the CRDM nozzles.

After removal of the closure head, cameras can be lowered through the peripheral area of the plenum, part of the top of the core can be examined, and an initial inspection of the periphery of the core can be accomplished. Additional inspection of the core through the 69 control rod guide tubes can also be performed. If the remaining fuel assemblies have to be inspected prior to the initial plenum lift, holes may have to be drilled in the top surface of the plenum assembly to provide access. This should be done only if absolutely necessary. Locations of holes can be determined best following initial video examinations.

These inspections would also provide information on debris or fuel which might have migrated into the plenum assembly, as well as possible mechanical distortions, all of which may impact procedures used for plenum removal.

A detailed examination by visual methods should be performed when the upper plenum structure is removed and the top of the core is exposed. This examination will provide input to the types and degrees of fuel assembly damage incurred during the transient. The information will provide a firm basis for selection of the fuel removal technique most likely to be successful and a sequence for removal of the fuel assemblies.

The inspection of the core should be conducted from the fuel-handling equipment bridge. Two viewing centers could be set up, one in the command center outside the containment building and one on the bridge itself for the crane operator. The command center should have videotape and audio recording capabilities, as well as capability to record character information on the videotape. Provisions to record "still" video shots and to transmit these pictures offsite should be provided. Capability for photographic records including stereographic photos should be provided.

Good water clarity would be an important contribution to the successful inspections of the core. The visual activity afforded by clean water would also be of benefit during fuel removal operations.

4.5.3 IDA Withdrawal/Removal Methods

Prior to defueling the reactor core, the Incore Detector Assemblies (IDA) should be withdrawn from the fuel assemblies. The fuel assembly can be removed without removing the incore detector; however, the incore detectors could then complicate the removal of fuel assemblies. The IDA locations are shown in Figure 4.5-1.

Removal of the IDA may provide an indication of reactor core damage. A detector that is difficult to remove the first 20 feet may be indicative of a damaged fuel assembly.

The discussions presented in this section are based on the assumption that the incores are essentially intact but may be "stuck." Correlation of incore data with core transient data indicates that the incore detectors may be intact. However, fuel assembly damage and/or debris in the guide tubes may prevent their withdrawal. Plans for fuel removal assume that a large number of these incores will not be withdrawn in the normal fashion. Considerations which affect the means used to withdraw stuck incores include:

- Pulling on the incore puts a load on the already damaged (weakened) fuel assembly.
- Debris trapped in the guide tube is likely to have a high radiation level and could contain fuel and fission products.
- Pulling a fuel assembly out with a stuck incore will be restricted by the load which can be applied by the special fuel movement tools to the weakened assemblies.

As a result of present experience with stuck incores, a procedure and equipment have been developed which could be employed within the limits already discussed. This method is briefly discussed herein.

If the IDA does not comes free, force is maintained on the incore detector, and two cable stoppers are installed on the cable at the closure assembly. The force on the IDA is released, and the cable is sheared, leaving at least 1 foot above the cable stoppers. An 8-foot extension is welded to the sheared IDA, and force is reapplied to allow removal of the cable stoppers. The pulling force is then released.

When the respective fuel assembly is lifted, the incore detector can be cut in the core region. The incore can then be removed either from the core side or the incore service area.

A system (Figure 4.5-2) should be planned to flush the incore guide tubes toward the reactor vessel. This system would be applied to each detector/guide tube prior to attempting withdrawal. It would be desirable also to continue flushing during the withdrawal to prevent debris from reentering the guide tube.

4.5.4 Control Component Removal Methods

The methods employed for control component removal will depend largely on the condition of the host fuel assembly. The general damage categories into which the fuel can be classified are listed in Section 4.5.5.

Although it may be mechanically possible to remove the control components from Damage Level 1, and in some cases Damage Level 2 fuel, it may not be feasible due to criticality or safety concerns. If it becomes permissible and necessary to remove control components from fuel assemblies, they will be removed with the special tools described in 4.5.1-2, inspected, and placed in storage containers. If it is not necessary to separate the control assembly from the fuel, it will be removed with the fuel assembly.

Those Damage Level 2 and 3 components which cannot be separated from the fuel or which have no requirement for separation will be disposed of in the same manner as the host assembly.

A core map showing control component locations is shown in Figure 4.5-3.

4.5.5 Fuel Removal Methods

After the core inspections described in 4.5.2, it is anticipated that the condition of the fuel assemblies can generally be categorized as follows:

Damage Level 1 - Weakened structurally with pin failures and warpage. Major parts remain intact.

Damage Level 2 - Significant structural damage and oxidation with fragmented components forming a layer of debris over the intact core portion. No eutectic bonding exhibited.

Damage Level 3 - Crust of metal oxide eutectic formed which bonds parts of the core together.

The method employed for removing a fuel assembly will depend on the damage level. Some methods to be considered for removal of each level are given below:

Damage Level 1 - Removal of assemblies with this damage level will be attempted first. A TV camera will be positioned to provide a good view of the upper end fitting. A probe will be inserted into each guide tube in turn and the location of any obstructions noted. Rods with expansion mechanisms on their lower ends will be arranged on a spacer to fit the guide tube pattern and their upper ends will connect to a hoist. These rods will be inserted into the guide tubes to the lower end fitting and the expansion mechanisms will be activated. The lifting forces generated are transmitted to the fuel assembly only at the lower ends of the guide tubes, where the probability of structural integrity is best. The assembly will be lifted out and placed in its storage container after being given a visual examination.

The process will be repeated until removal of all Level 1 assemblies has been accomplished.

Damage Level 2 - Visual examination leads to the conclusion that the assembly has significantly more deterioration than a Level 1 category assembly. Since portions of the assembly may be missing and since the assembly may be structurally unsound down to perhaps, 3 feet from the lower end fitting, several methods may be necessary to remove the assembly. Rubble and debris present in the upper part of the core, but which is not metallurgically bonded, will be removed with the special tools described in 4.5.1-2 to provide clear access to the remaining portions of the fuel assembly.

Removal of the remainder of the assembly may require lifting by a guide tube expansion mechanism and hoist (as described in Damage Level 1 above) or the use of the side-access device, or both, depending on the core location and unique condition of the particular assembly. Any loose pieces falling to the lower vessel cavity upon assembly movement will be retrieved when feasible. Assemblies will be visually examined when removed from the core and placed in storage containers. Side loading cans and side loading tools may be required for assemblies with this degree of damage.

Damage Level 3 - Fuel assemblies with this damage level may be mechanically or metallurically bonded together so that removal intact may be impossible. These fuel assemblies will be separated by cutting them apart. If no fuel assembly can be removed intact, the entire core may have to be cut up with the fragments being removed in storage containers. High speed saws, lasers, or torches may be used to cut or separate these highly damaged assemblies. Special suction equipment and mechanical grappling devices may be used to retrieve debris resulting from the cutting operation.

4.5.6 Inspection During Fuel Removal

A complete video scan of the top of the core can be performed to ascertain and record the initial core conditions. A camera system that views the sides of assemblies as they are withdrawn from the core should be investigated. With such a system, the condition of each fuel assembly can be recorded as it is withdrawn and categorized as to desirability for future inspection. This system would minimize the handling of fuel for inspection and allow direct movement to the fuel canning station.

Once the first assembly has been removed a camera can be lowered into the vacated location to determine the condition of adjacent fuel assemblies.

An inspection station for more detailed inspection of selected assemblies prior to canning should be considered; however, every attempt should be made to minimize fuel handling and still remain consistent with required inspection.

4.5.7 Fuel Canning

The actual procedure for movement of the fuel into the cans would depend on many factors which will not be known until the condition of the containment following decontamination and the integrity of the fuel are assessed. Inspections conducted prior to lifting the plenum, radiation mapping prior to head removal, and incore withdrawal operations will all help to determine fuel assembly status. However, the actual status cannot be verified until the initial video inspection. The radiation levels encountered following containment decontamination will also influence the fuel movement pattern selected. The location of the can loading operation (deep end vs reactor vessel) could impact the final procedure. Criticality control (to be addressed by an ongoing study) is not anticipated to be a problem but should be addressed by the procedure. The actual procedure to be followed during the can loading operation should address the following areas:

- As an assembly is lifted from the vessel (to be loaded in a can at the vessel or moved) video coverage can provide a record of fuel assembly status.
- Following removal of an assembly a video inspection of the remaining assemblies can be used to evaluate which assembly will be moved next, to determine what method will be used (side, top grapple, etc.) to move it, and to provide an additional record of core status.
- Procedures should provide guidance on which lifting tool is to be used.
- Concerns with rupture of weakened assemblies and release of gases should be addressed.
- Vacuuming (if deemed desirable) would also be covered generically in the procedure.
- Operation of the canal cleanup system should be addressed.
- Lifting of tools out of water should be addressed, i.e., do not lift them out of water without radiation monitoring and wash them down if necessary.
- The procedures used should identify which assemblies (if any) are to receive detailed inspection.
- Key hold points should be identified in the procedures such as (a) following initial core scan, and (b) if any assembly comes apart or is severely damaged (other than intentional cutting, etc.) as a result of the movement operation.

4.5.8 Fuel Movement Patterns

As a result of the overriding considerations of personnel exposure, contamination control, and criticality, the amount of activity which is performed in the reactor building should be evaluated. Figure 4.5-4 shows three possible fuel movement patterns. The patterns shown are for illustration of what impact the various considerations could have. The radiation levels shown depict three different background conditions.

Figure 4.5-4A depicts fuel movement should low radiation levels be encountered. For better contamination control, a washdown/vacuum step (2) is performed on the majority of the assemblies. This step would help to remove loose contamination and/or fuel, thereby reducing future problems n handling the assemblies. The assemblies are then canned (4) or inspected prior to canning (3) and (5), then sent to the spent fuel pool (7). Path (6) is shown for those assemblies or portions of assemblies deemed too weak to risk movement without a can.

Figure 4.5-4B depicts fuel movement should intermediate radiation levels be encountered. In this case fuel movement will be much the same as in 4.5-4A except that the majority of the fuel will be scanned by video as it is removed from the core, then loaded directly into cans for movement into the spent fuel pool.

Figure 4.5-4C represents a fuel movement pattern which could be employed for high radiation levels. The majority of the fuel would be (1) scanned by video as it is removed from the core, then loaded directly into cans for movement (4) (5) to the spent fuel pool. A limited number of specifically designated assemblies would be moved (2) from the core to the inspection area, inspected, loaded into cans, and moved to the spent fuel pool (3) and (5). The assemblies to be inspected would include those designated for further research so that any changes following removal would be accounted for.

The wash/vacuum station is shown removed from the vessel area so that it will not interfere with the removal of assemblies from the vessel. It is anticipated that the critical (time) part of the operation will be the removal from the vessel. Additionally, an operation which could present visibility problems should be closely evaluated to ensure that the maximum possible visibility is maintained in the reactor vessel area.

A backup fuel movement path would be to load the fuel over the vessel into cans, then place the cans directly into a shipping cask in the deep end of the canal. The cask would then be decontaminated and placed on its transporter for offsite shipment.

This procedure should be considered only in the event that the fuel cannot be transferred to the spent fuel pool through the transfer tube. The time required for this type operation would significantly impact the total time and personnel exposure required to remove the core.

4.5.9 Fuel Outside the Core Region

As a result of reactor coolant pump operation following the core damage caused by the accident it is possible to have fuel distributed throughout the reactor coolant system. The most likely areas for deposition of this material would be horizontal surfaces in regions of low flow velocities, such as the upper steam generator tubesheet and the bottom of the reactor vessel. At various phases of the operation, these two areas will be accessible for cleanup. The special considerations of high radiation levels and criticality control must be addressed for both of these potential locations. The reactor coolant system decontamination flush should consider the unlikely occurrence of having fuel trapped in dead legs such as instrument lines, letdown lines, and drain lines. Careful radiation surveys should detect such deposition. The initial fuel scan and subsequent fuel assembly inspections will help to ascertain the potential for fuel outside the core.

Several areas where fuel debris is likely to have been deposited are:

a. OTSG Upper Tubesheet

The long lead time required to prepare equipment for fuel removal from the upper tubesheet and the complication of being above the water level for head removal and defueling dictates that this problem be identified as early as possible. As early as feasible, from a building entry standpoint, radiation surveying of the upper head region should be initiated. If this survey indicates the need to remove fuel from the upper head, then construction of any special equipment should begin. (Equipment design and operating techniques should be identified prior to this.) This operation should be completed early enough not to interfere with the head removal schedule, which will require steam generator draining below the upper tubesheet. Depending on the radiation level the use of a wash/vacuum system to remove any fuel from the tubesheet may be possible. This system should be portable and have sufficient shielding of any filters used. The filter design should allow transfer underwater of the cartridge to a fuel transfer can for movement to the spent fuel pool.

The possible use of mechanical manipulators to pick up debris and place it in a shielded container should also be considered. Video coverage could be provided through the inspection hand hole.

b. Reactor Vessel

There is a concern that the bottom of the reactor vessel contains fuel or debris which went through the reactor coolant system, as well as some which found its way there directly from the core. This fuel will remain covered with borated water, for heat removal and criticality control, throughout the defueling operation. Following the defueling, the reactor vessel internals should be removed to provide access to the bottom of the vessel. In the event that the internals are not removed access to the bottom can be made through the lower grid plate. This debris can be removed using the same wash/vacuum system used on the core to sluice it into a container. These containers should be suitable for loading into a fuel can for movement to the spent fuel pool.

c. Other Locations

Samll quantities of fuel or debris may become lodged in dead legs such as drains and letdown lines, or areas such as the reactor coolant pump, RCS piping, and steam generator lower head. Methods used to remove the debris from these locations need to employ means to assure criticality safety while minimizing radiation exposure. For example, when the RCS is flushed it may be necessary for the initial flushing to employ boric acid for reactivity control. Disassembly of components to remove these materials could result in large exposures to personnel, as well as contaminating the surrounding areas.

Considerations of schedule and the sequence of events during decontamination flushing dictate the approach selected for ensuring cleanup of these areas. Radiation surveys, early in the program, can be used as a basis for selecting the methods to be used. Additional areas of specific concern include: pressurizer spray valve, pressurizer vent valve, pressurizer electromatic relief; reactor coolant drain tank; makeup system letdown line, letdown coolers, letdown orifice and seal return line; and core support locations such as in the areas of the former plates. These areas should all be examined from a radiation standpoint. The former plate area bas access holes for flow, but they are too small to reach with a vacuum. An extensive radiation survey should be performed to help assess the amount of radioactive material that may be trapped. A system flush or the decontamination flush discussed in Section 5.0 could be used to remove this material. The effectiveness of the removal can be determined by taking before and after radiation surveys.

4.6 Fuel Containers

Considerations for fuel container development should include, but not be limited to, the following criteria:

- The need for a material suitable for long-term storage.
- Criticality safe configuration.
- Compatibility with existing fuel shipping casks.
- Ability to fit in upenders, pass through the transfer tube, and be stored in spent fuel pool storage racks.
- Handling should be compatible with fuel-handling capabilities including lifting devices in fuel transfer canal and spent fuel pool.
- The ease of loading the damaged assemblies is important to minimize the radiation exposure of personnel performing the operation.
- The cans should be sealed to prevent the spread of contamination and for shipping considerations. Unsealed cans for cooling and chemistry control should also be evaluated.
- The minimum opening for loading the fuel should be as large as possible to allow the assemblies to fit inside the cans with a minimum effort.
- Side-loading cans may be necessary for those assemblies too severely damaged to be handled from the top or a sufficient number of side-handling tools should be provided to allow the tool to stay with the assembly in a top-loaded can.
- Can design should consider working conditions underwater with long manual tooling, such as restricted visibility due to crud in the water, and the need for respirators and gloves.
- The containers will require approval by government agencies.
- Cans should have identification markings that can be easily read from above.
- A method to vent or verify that the containers are full of water and will not be buoyant when restraints are released after shipping in a cask.

The preceding list of considerations for fuel can design was based on the assumptions that all of the fuel will be loaded into cans for movement. The bases for this assumption are that the structural integrity of the assemblies will be questionable; that the handling tools will be capable of separating even the severely damaged assemblies; that the handling tools will have the capability to alter the configuration of severely damaged assemblies to fit the can; and that even good assemblies could have portions of other assemblies trapped which could become dislodged during fuel movement. Due to the public concern over shipment of radioactive materials in general and the special nature of this fuel, plans should include the capability for interim storage onsite in the fuel movement cans. Present core damage estimates indicate that handling of the assemblies from the top may not be possible in some cases. Therefore, means to separate, lift, and load the assemblies into cans using tooling which handle the assemblies from the side may be required. This dictates the need for side-loading cans or tooling which is designed to remain with the assembly in the can. Any can design which does not fit in existing shipping casks should be avoided due to the large expense and long lead time required to acquire and license new casks. The possibility of higher than normal radiation fields, worse than normal pool visibility, plus the need to work in more restrictive anti-Cs than normal, dictate large and simple grappling devices for items such as can covers, etc.

In addition to the one-assembly containers discussed above there should be some containers for handling of debris. The maximum size of these containers would be determined by criticality studies as well as the transfer tube and cask cavity dimensions. Containers should be provided for any filter cartridges used in the cleanup system. Stacking of smaller containers inside of a fuel assembly container should be considered but must be examined from a criticality standpoint.

4.7 Fuel Transfer System Modification

Difficulties have been experienced with the existing fuel transfer system. A recommended modification to the fuel transfer system replaces the existing underwater air-motor-powered chain and sprocket drive with a cable drive. (Figure 4.7-1) The features of the system are:

- o Positive and reliable operation
- o Increased allowable track misalignment tolerances
- o Manual operation capability
- o Overload protection
- o Improved dry operation capability
- o Less maintenance (most mechanical parts located above water level)

Winch(es) for each transfer system would be mounted above the water level in the fuel storage building. The winch(es) and cables control the movement of the fuel transfer carriage between the reactor building and the fuel storage building. Each winch is provided with a load limiting device to deenergize the winch when cable tension builds up to a preset limit. The winches are interlocked to the gate valves to prevent carriage operation unless the gate valves are fully open. All cable drive components are contained within the fuel storage building, eliminating the necessity to remove the reinstall cables when the system is not in use.

The existing carriage is used as is except that all chain and sprocket drive components and anti-overturn lugs are removed. Lubrite bushings are installed in the carriage wheels. Lateral guide rollers and cable attaching lugs are provided for attachment to the carriage.

The existing track sections are used as is, provided that they meet the alignment criteria required for the cable drive system. Additional removable sections are provided to span the gap on the reactor building side of the transfer tube.

4.8 Fuel Storage and Shipment

4.8.1 Direct Shipment Considerations

Section 4.5.7 discusses some of the implications of loading fuel cans directly into spent fuel shipping casks inside containment. Another consideration which must enter into the evaluation of this option is the issue of interim storage and ultimate fuel disposition. This issue bas plagued the nuclear industry for some time, but it has special impact on Three Mile Island, since an entire core could be extensively damaged.

If the direct shipment option is to be used, there must be someplace to which to ship the fuel. As present, unless the federal government makes a facility available, there is not such place. Even if the federal government makes some facility available, it is not clear that such a facility will have the capability for fuel handling immediately available. If additional fuel handling and storage capabilities must be constructed, it is likely that such an effort would require a schedule which is not compatible with the TMI recovery schedule.

It may be concluded that a direct shipment option for the entire core does not exist. Temporary fuel storage should, therefore, be a design basis.

4.8.2 Temporary Storage Requirements

Once fuel removal operations have begun it will be desired to complete them as swiftly and safely as possible. First, removing the damaged fuel from the reactor vessel, canning it, and locating the cans in a specially designed damaged fuel can rack stabilizes the damaged fuel, assuring a more controlled and safer environment. Second, canning the damaged fuel limits the primary source of radioactivity release, thereby reducing the production of additional volumes of radioactive liquid wastes. Finally, removing the fuel from the reactor vessel allows cleanup of the reactor coolant system unimpeded by fuel in the core.

It is anticipated that spent fuel will be removed, canned, and prepared for transfer from the refueling canal much faster than spent fuel shipping casks could be handled, loaded, and make the round trip to and from a depository. Therefore, a backlog of fuel cans would have to be stored onsite temporarily even if the ultimate fate of the fuel were known prior to defueling.

The entire core will be placed in failed fuel cans. It is expected that more than 177 failed fuel cans will be needed, since some elements may have to be packaged in two or more cans and debris from failed fuel assemblies will also be packaged in failed fuel cans. It is more appropriate to assume that at least 250 failed fuel cans will be used to remove the fuel. This number will allow two cans for each element in the central region of the core and one can for the peripheral assemblies which are expected to be structurally intact.

The "B" spent fuel pool appears to be large enough to hold in excess of 460 cans in non-poisoned high-density racks with 14-inch-on-centers spacing. This would allow storage of the entire damaged core and full core load. A sketch of the A and B spent fuel pools with high-density spent fuel racks is shown on Figure 4.8-1.

4.9 Organization

The organizational arrangement used to implement the core inspection and fuel removal work plan should recognize several important factors:

- Single point responsibility is important to overall performance.
- Several disciplines are involved in each of the subtasks.
- Flexibility of total force is needed.
- Timely engineering support will be required.
- To the extent possihle, the work will be directed from the Command Center.
- Methods, procedures and tooling are likely to change during the course of the work.
- A significant amount of preparation work prior to actual implementation is needed both off-site and on-site.

4.9.1 Functions

The following functions should be incorporated into the fuel removal organization:

- Management and Supervision
- Health Physics
- Planning and Scheduling
- Technical Support
- Procedure Engineering
- Tool Engineering
- Facility Engineering
- Training
- Labor

Several of these functions should be activated for preparation off-site well in advance of site buildup.

Onsite, these functions should be organizationally arranged to accommodate shift and schedule requirements.

4.9.2 Responsibilities of Key Personnel

Descriptions of each job function should be prepared including interfaces with other site organizations. Some responsibilities for key personnel area:

Task Manager - responsible for overall execution of fuel removal work.

Shift Supervisors - responsible for activity in process relative to fuel removal.

Technical Support Engineers - responsible for review and evalution of inspection as it is available. Provide engineering recommendations as needed, monitor operations, and maintain records of fuel location and waste generation.

Health Physicists - responsible for radiation control and personnel exposure.

Video Specialists - responsible for performing special video inspections and maintaining video equipment.

Equipment Operators - responsible for performing equipment manipulations to remove and store fuel.

In order to maintain steady work progress, more than one shift is recommended during fuel removal. A determination of the total number of shifts needs to consider the working conditions (working in breathing apparatus), the amount of physical exertion, the need for extras on each shift, and the need for supervisory personnel on each shift. Use of three shifts, for example, will require three teams of supervisory personnel.

Another major concern is the amount of radiation to which personnel will be exposed. The level of radiation, the working conditions, and the amount of manual tool work could dictate that teams working inside the containment be relieved every one to two hours. The video team, however, could consist of as few as two people since their time in the containment should be minimal.

Related, but separate, tasks such as can loading versus inspection can be performed by separate teams. These teams should be familiar with the overall operation to enable maximum flexibility should one phase start to lag the other. By having personnel familiar with the operation of other teams, they could, if required, be trained to perform these other functions.

4.9.3 Site Buildup

Staff buildup on site should begin several months in advance of the scheduled execution of the work in containment.

Some of the functions which will be needed early are:

- Facility engineering for early input and coordination of facility changes and deletions in and outside containment which directly support the reactor defueling work.
- Procedure engineering to assure timely review and approval of the work procedures.
- Planning and scheduling for integration of the reactor defueling work into onsite activities and to establish the final schedule for assignment of other resources.

Lead time for buildup of resources should also be consistent with the pre-established training requirements.

TABLE 4.1-1

SUMMARY OF STUDIES AND TESTS RECOMMENDED

- Fuel Assembly Mechanical Integrity Assessments.
- In-Vessel Criticality Studies to Determine the Safety Aspects of Moving Damaged Components.
- Ex-Vessel Criticality Studies to Determine Safe Methods of Handling Fuel Bearing Components or Debris.
- Develop Techniques for Core Inspection.
- Study Core Disassembly Methods and Procedures.
- Develop Method for Measuring Sub-Critical Multiplication.
- Study and Evaluate Acceptable Ways of Fuel Canning.
- Study Fuel Transfer and Storage Problems.
- Develop Techniques for Obtaining a Reliable Inventory of Fissionable Material.
- Study and Develop Techniques for Radiation and Contamination Control.
- Prepare Tests to Ascertain the Mechanical Integrity of the Fuel and Operability of the CRDMs.
- Prepare Functional Tests for Special Tools and Cleaning Equipment.

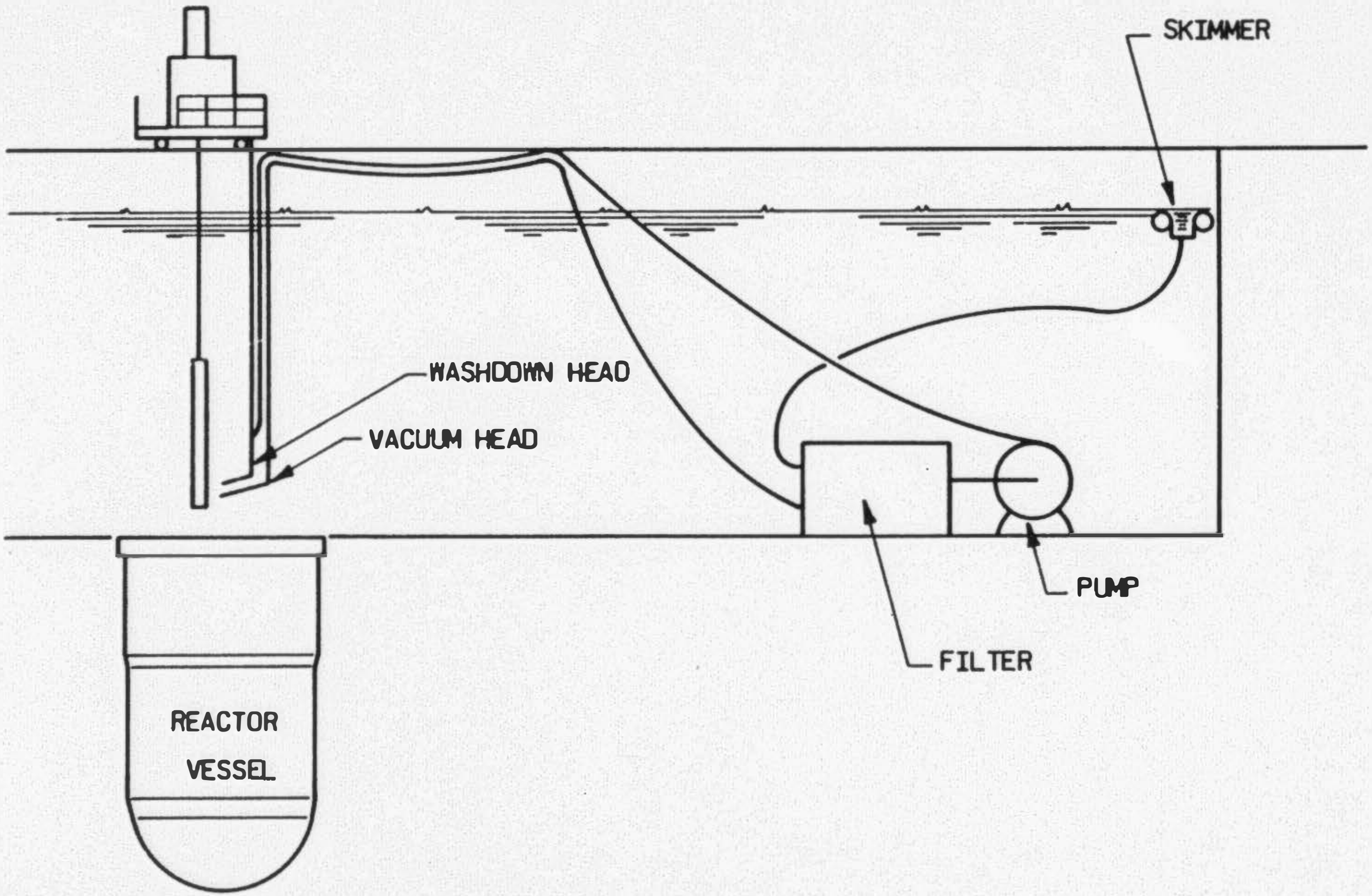


FIGURE 4.4–1
UNDERWATER WASHDOWN AND SKIMMING SYSTEM

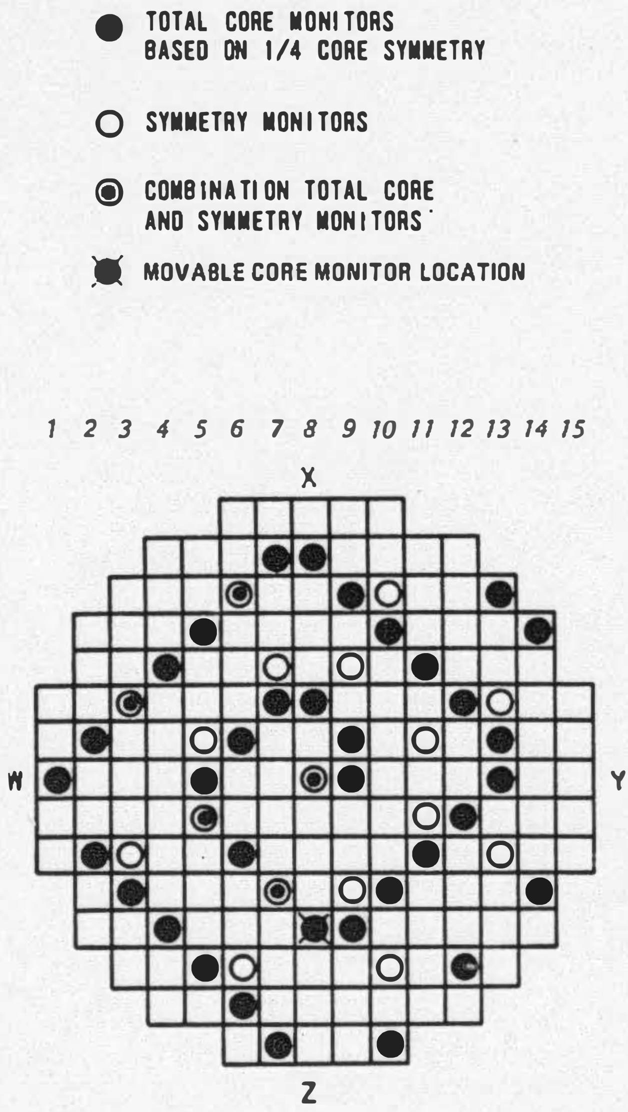


FIGURE 4.5–1
INCORE DETECTOR LOCATIONS

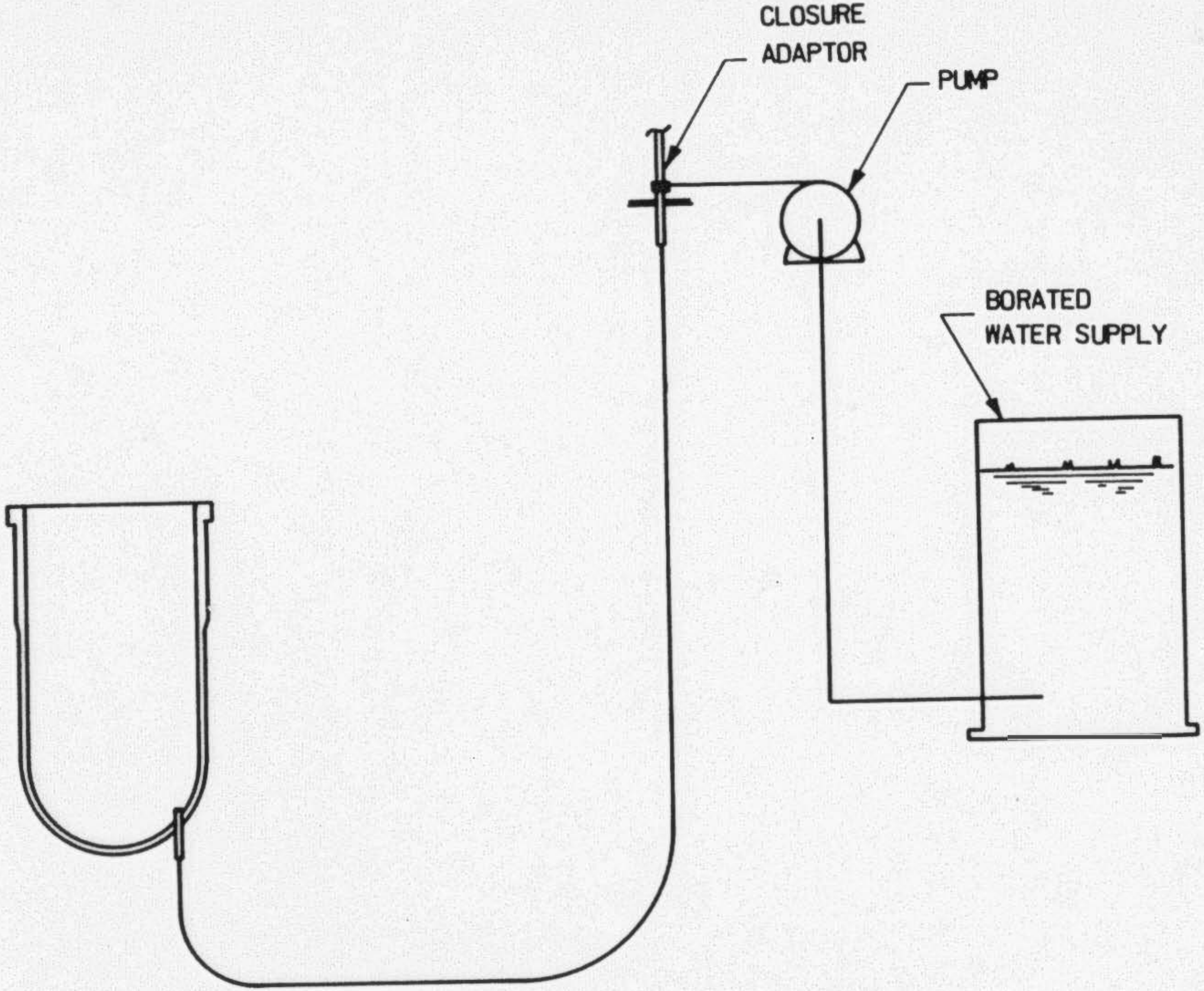
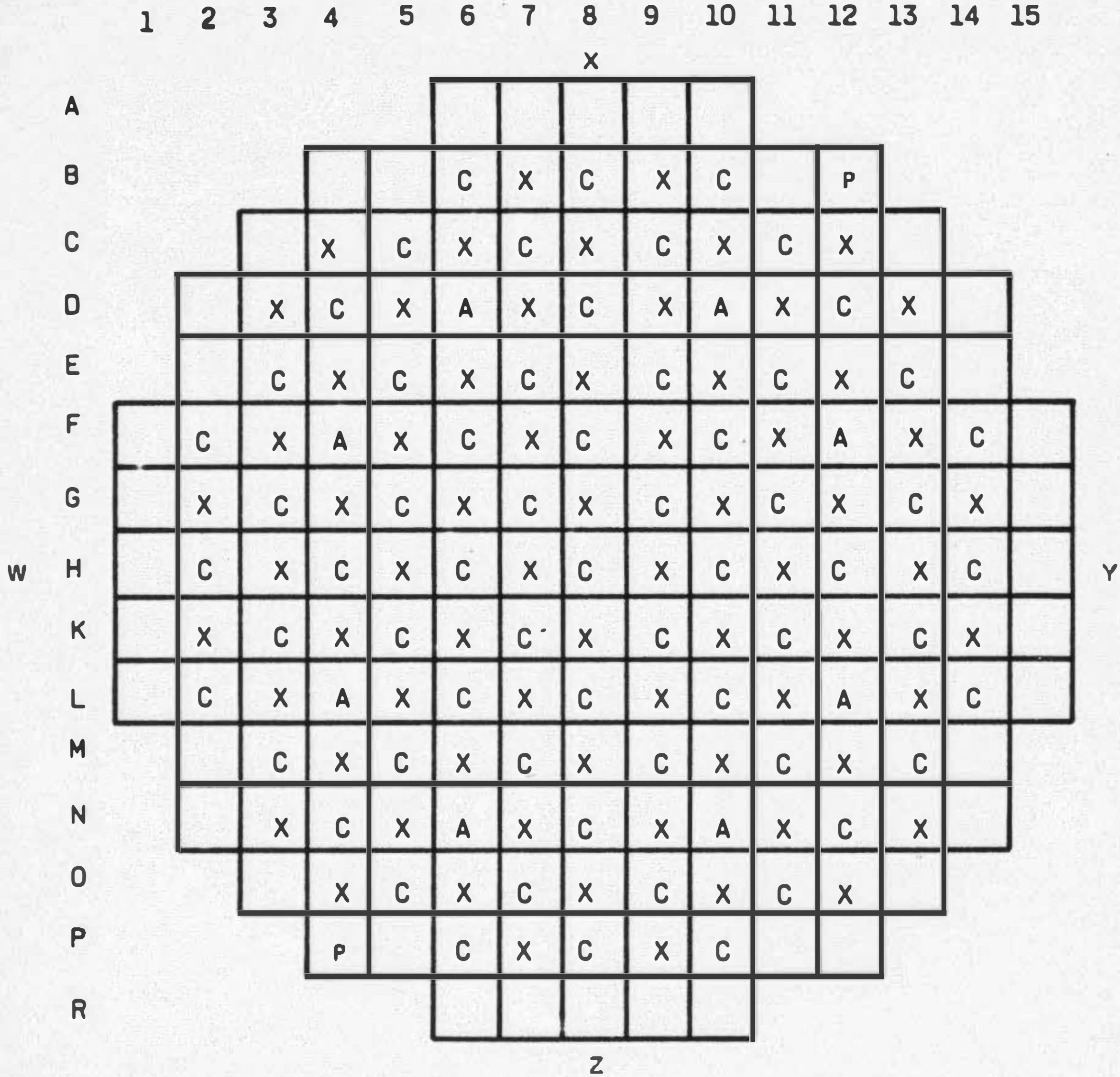


FIGURE 4.5–2
INCORE GUIDE TUBE FLUSHING SYSTEM



CONTAINS NO CONTROL COMPONENT

C CONTROL ROD ASSEMBLIES

A AXIAL POWER SHAPING ROD ASSEMBLIES

X BURNABLE POISON ROD ASSEMBLIES

P PRIMARY NEUTRON SOURCES

FIGURE 4.5–3
LOCATION OF CONTROL ASSEMBLIES

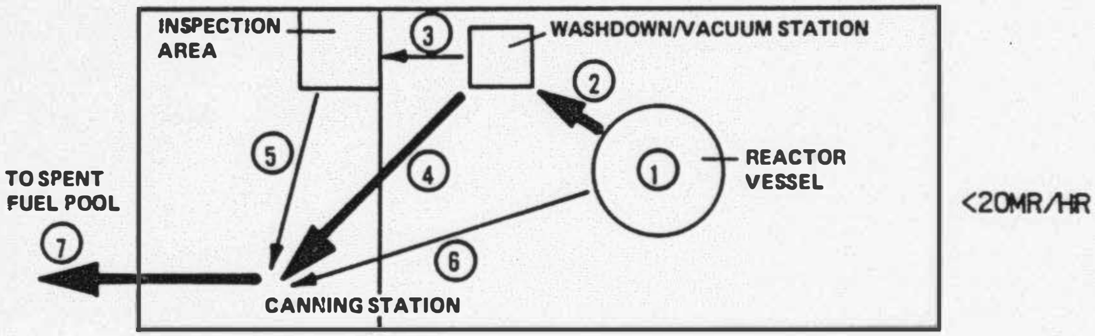


FIGURE 4.5–4A

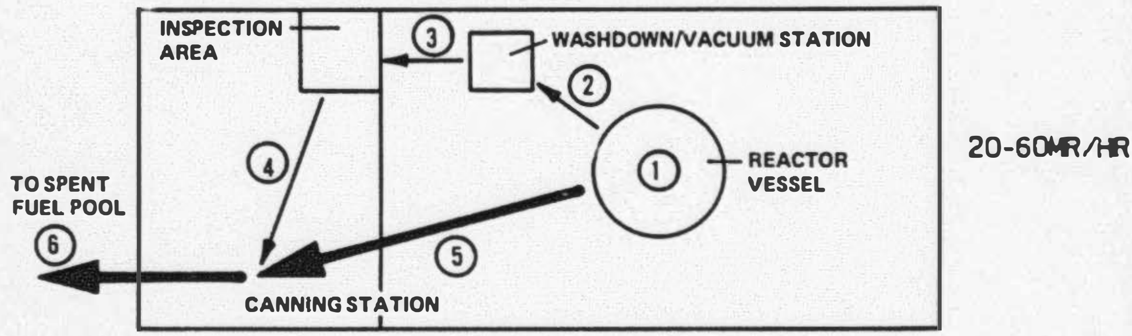


FIGURE 4.5–4B

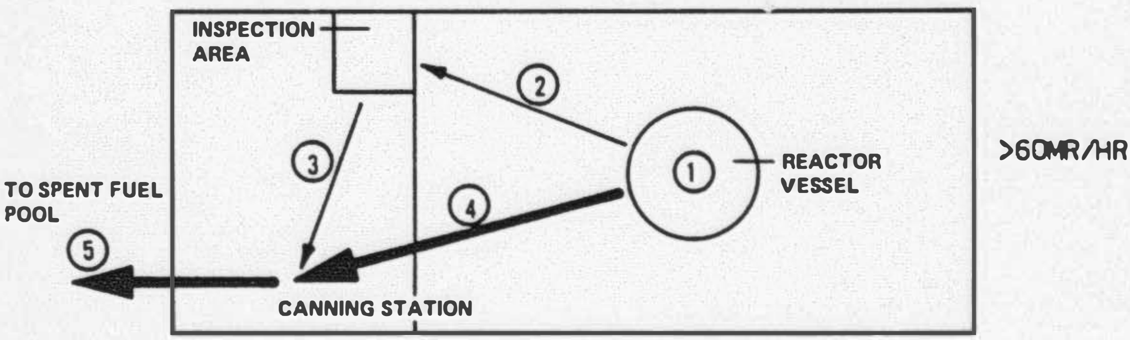
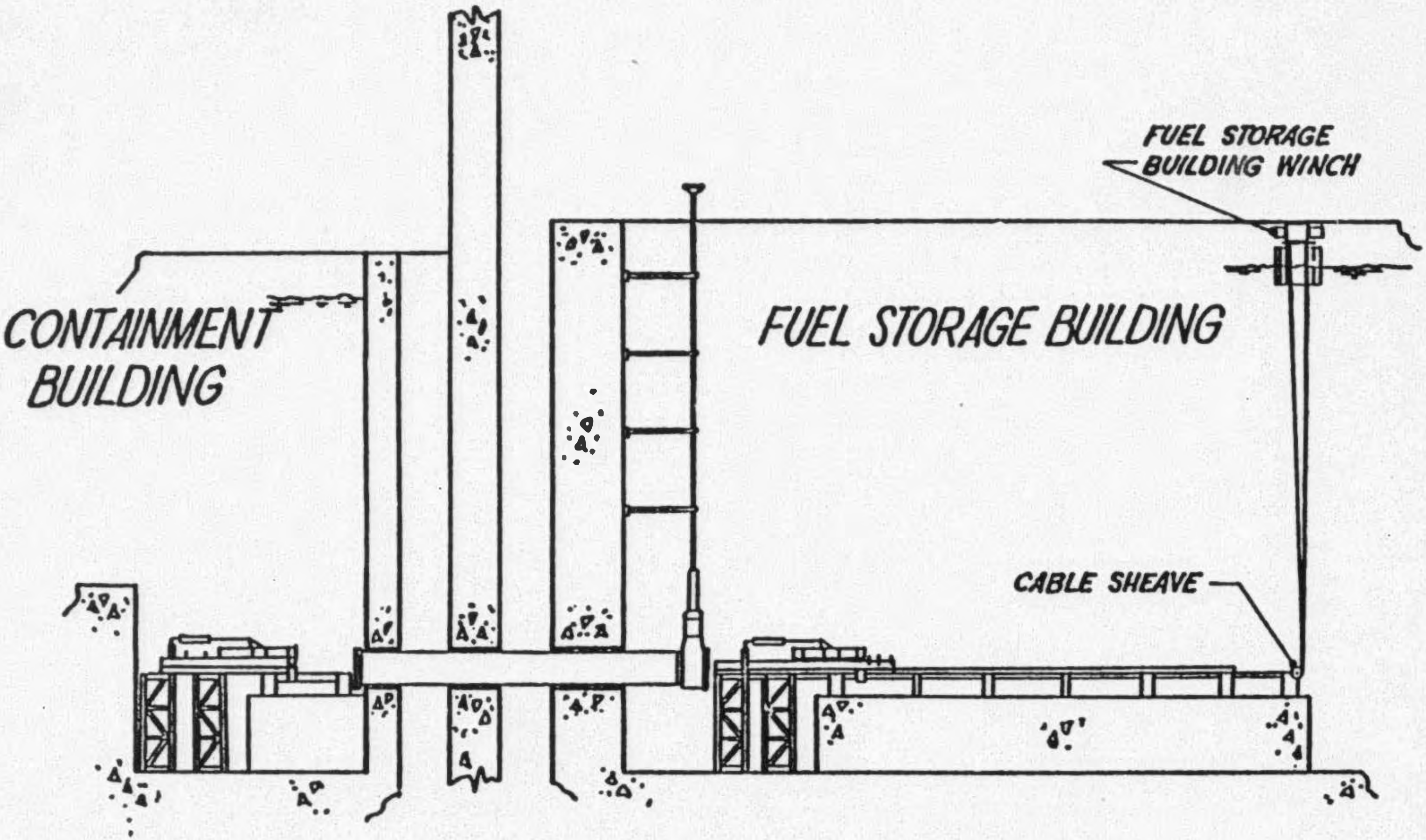


FIGURE 4.5–4C

FIGURE 4.5–4
EXAMPLES OF FUEL MOVEMENT PATTERNS



SINGLE WINCH CABLE DRIVE
TRANSFER SYSTEM

FIGURE 4.7–1

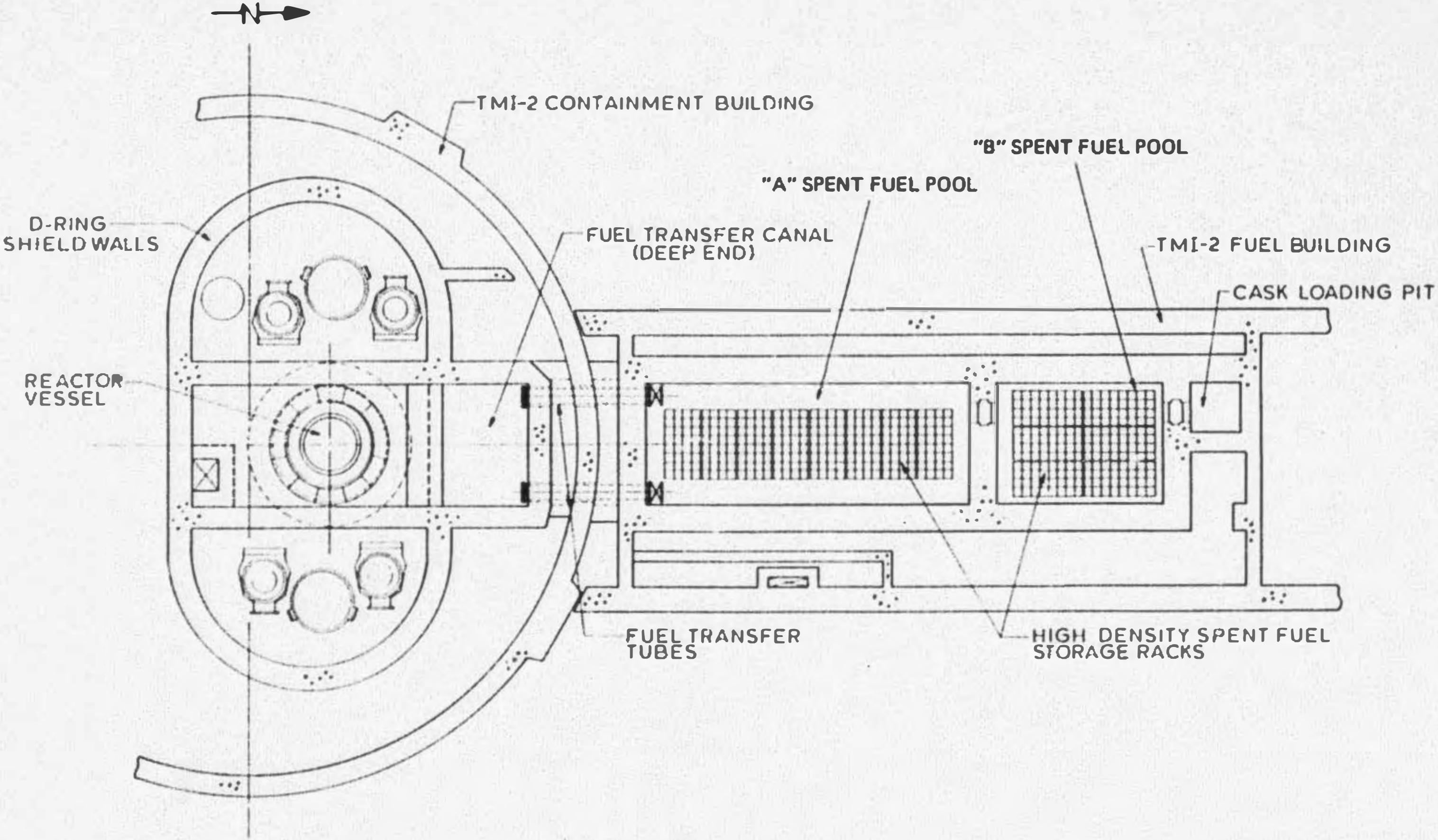


FIGURE 4.8-1
FUEL HANDLING AREAS IN CONTAINMENT AND FUEL HANDLING BUILDING

5.0 REACTOR COOLANT SYSTEM DECONTAMINATION

5.1 Introduction

5.1.1 Background

The contamination of the reactor coolant system (RCS), as a result of the release of fission products and fuel from the reactor core area, is a significant problem that must be addressed in the overall recovery program. The potential distribution of fuel or core debris throughout the RCS must be considered; any fragments must be removed in order to permit proper future reactor operations and to reduce a source for future fission product release to the RCS. In addition, the fission product activity in the RCS contributes to the containment radiation levels, thereby complicating the complete recovery and ultimate requalification of the RCS. In response to these concerns, a comprehensive RCS decontamination program should be established and integrated into the overall recovery planning.

5.1.2 Objectives

The principal objectives of the decontamination of the Reactor Coolant System should include:

- Reduction of the containment radiation levels attributable to the activity inside the RCS while maintaining component and pressure boundary integrity.

- Removal of fuel fragments and related debris from the RCS.

A program designed to meet these objectives should consider the cost and schedule limitations of the recovery effort while attempting to reduce the containment radiation levels as early as possible.

5.1.3 General Decontamination Program

The RCS decontamination effort should be directed to each of the potential problem areas and should include primary and alternate plans for the removal of the radioactivity and fuel debris. The activity associated with the RCS comes from three basic sources:

- Soluble activity consisting primarily of soluble fission products,

- Insoluble activity composed of fuel pellets and cladding fragments containing fission products, and

- Activity deposited on or absorbed into the corrosion film, including activated corrosion products as well as concentrations of soluble and insoluble fission products.

Each of these sources may require a different removal technique to effectively reduce the activity. In addition, special considerations may be required when working to collect and account for the distributed fuel fragments in the RCS.

5.1.3.1 Soluble Activity

The soluble activity can be removed from the RCS either by draining the contaminated coolant or recirculating the coolant through ion exchange beds and back into the system. A third possible approach would be to dilute and remove the soluble products by a feed and bleed method or partial drain and refill. This would, however, add to the radwaste processing volumes while having a favorable feature of being usable before and during defueling activities without a major equipment impact. Since the vessel water cannot be drained before defueling, some method must be provided to recirculate the reactor vessel water to facilitate activity removal from the entire system.

The optimum approach will probably be to drain the water from the steam generators and loops and process the water as it is being removed. This approach is advantageous from the standpoint of minimizing radwaste volumes; however it may not be advantageous from other standpoints such as radiation exposures. The steam generators may be drained without draining the reactor vessel. Another approach would be to store the drained water for subsequent processing. The water in the vessel would then be purified by recirculation through ion exchange. This should be initiated as soon as practical and cleanup maintained at maximum flow throughout the defueling period. The decay heat removal arrangement being used at the time (e.g., mini-decay heat removal system) can be used to circulate the reactor vessel water to the purification equipment.

The vessel contains about 26,000 gallons of water and the remainder of the system about 59,000 gallons of water. Draining the loops thus lowers the volume to be reprocessed by inline purification by a factor of three.

Soluble activity should be removed before defueling. This will lower radiation levels at the refueling canal water surface during fuel handling, and lower radiation levels on the canal surfaces after draining. The refueling water will also require less processing after use. Boron control must be maintained during the processing of this water. Also, core flow should be kept at low velocities, to minimize potential for further fuel damage. Resin beds must be heavily shielded and designed for remote maintenance.

5.1.3.2 Fuel Fragments

There is a probability that some fuel fragments are outside the core. The nature of these fragments (pellets, powder, particulate) is not known, nor is the location or amount. Some correlations to other events which produced loose debris in the RCS at other plants can give an insight to most probable collection locations. The likely areas for collection of loose core material include:

- bottom of the reactor vessel
- upper tubesheet of the steam generators
- instrumentation lines and drain connections in low areas of the RCS piping
- lower dome area of the steam generators
- horizontal surfaces of the plenum assembly
- horizontal surfaces of the core support assembly
- pressurizer and surge line and spray line
- letdown system.

In order to retain a safe margin to any criticality possibility while collecting fuel debris, adequate precautions in terms of neutron poisons or ever-safe geometry should be taken. This could be done by providing for the collection vessel for any fuel fragments to contain enough borated water or other neutron poison to control the neutron population or by designing the collection system to maintain ever-safe geometry. Underwater vacuuming can be used to pick up the loose fuel and cladding material (and a significant portion of the loose insoluble activity) depending on just how finely broken and dispersed the fragments are. Because of the high radiation levels which could be encountered, these vacuuming operations may have to be conducted using automatic or remote control equipment. Detailed isotopic radiation surveys of the RCS should be conducted in order to identify those areas that contain the insoluble fission products. These areas then should be concentrated upon as they are likely to contain the fuel fragments as well.

Areas in the bottom of the reactor vessel and the top and bottom of the steam generators, the pressurizer, and the pressurizer surge line should receive special attention as soon as access to these areas is possible, due to the tendency for debris to collect there.

5.1.3.3 Insoluble Activity

In addition to the material accessible for removal by vacuuming, a considerable amount of debris could be present in other portions of the RCS. Mechanical decontamination techniques such as flushing, hydrolasing, etc., could be tried on these areas to remove loosely adhered particles. In conjunction with these operations, if radiation surveys indicate the presence of fission products in such areas as instrument lines and vent and drain lines, these areas could be flushed to the RCS and the activity removed by the mechanical technique being used.

5.1.3.4 Deposited and Absorbed Activity

The corrosion product and fission product activity contained as part of the corrosion film may require different techniques for removal. While preceeding decontamination efforts for loose fragments and soluble activity may eliminate the need for further activity removal, there could also be a significant portion of the activity which has become absorbed into the corrosion film. This depends to a large extent on the particle size of the activated products. Since this information is not presently known for the chemistry conditions present at TMI-2, additional information is required to help assess the possibility of removing this absorbed activity. Information that would be particularly valuable would be the analysis of specimens from the RCS, such as steam generator manway covers or control rod drive mechanisms. These items could provide valuable data on the composition of the corrosion film in the RCS. In general, chemical decontamination techniques should be considered for removal of part or all of the activated corrosion layer. This approach will be discussed in more detail in the following section. It should be noted that the short operating time of the plant should have minimized the effects due to deposition of activated corrosion products such as cobalt. However, to effectively remove absorbed fission product activity, the corrosion layer should be also removed.

5.2 Decontamination Alternatives

Reactor coolant system decontamination should reduce the soluble and insoluble activity by appropriate means while minimizing the man-rem exposure expended and protecting the integrity of the pressure boundary. The ability to apply available mechanical and chemical decontamination techniques should be studied to identify the need for additional development work to produce the desired results. In addition to decontaminating the entire RCS, attention should be given to the individual component decontamination required to allow specific inspections and repair requirements associated with the recertification of the equipment. The decontamination requirements of each component can best be determined after detailed radiation surveys identify the activity present. This information together with the inspection and repair needs will provide a significant input to the decontamination planning.

5.2.1 Mechanical Decontamination

Several mechanical decontamination techniques may be considered as discussed below.

As part of the program to reduce the insoluble activity levels, a purification filter may be useful. This was discussed earlier as part of the activity removal before and during the defueling operation. After the fuel has been removed further filtration could be employed by flushing on the entire RCS. This could be done by running reactor coolant pumps and circulating water through filters. One location for filters is in the reactor vessel internals (or other suitable area for high flow filters, such as a flow screen similar to that used during hot functional testing). In addition, these filters could contain resin to further reduce soluble activity while collecting loose debris and fragments which had not already been removed from the system. Consideration should be given to the amount of fuel that could collect on such filters for reactivity control purposes. Further, depending on the quantity of filterable material actually present, change-out of the filters could be required. However, this approach does allow the entire RCS to be treated at one time and makes use of equipment already available (reactor coolant pumps). With this approach, backflushing lines containing possible fuel fragments and insoluble fission products from areas which normally are under low- or no-flow conditions could increase the total amount of activity removed.

Additional mechanical means such as hydro laser and scrubbing could be applied with a vacuuming system to large accessible surfaces such as the reactor vessel, head, and internals.

5.2.1.1 Reactor Vessel Head

Some of the early recovery plans include assessments of the condition of large components such as reactor vessel head.

A partial decontamination of the head could be required to permit reasonable access for inspection. A high velocity washing such as with a hydro laser could be applied. If this were scheduled before the head is transported to the head stand, a means of collecting the rinse water such as in the refueling canal area would be needed. In order to perform this with the head on the stand, modifications to the stand could be made to collect the waste. Since the majority of the inspection concerns will probably address the seating surfaces of the head, this area could be shielded to reduce the shine from other areas of the head. The head should not be dipped into a cleaning solution because of the carbon steel exterior. A chemical cleaning while on the head stand may be studied, although this is likely to require major modifications to the stand while shielding and remote inspection techniques could achieve the desired results. Similarly, cleaning of the control rod drive mechanisms could be done while the head is still installed or while it is on the head stand. Disassembly of the CRDMs for inspection requires removal of the leadscrews. If these can be removed while the head is still on the vessel, a backflushing path becomes available for the mechanisms. The waste would be flushed into the RCS and the dose rate of the head thereby reduced for subsequent operations due both to the leadscrew and the flushing. An alternate approach would be to perform this cleaning on the head stand. This operation could then be coordinated with any stand modifications required for head decontamination. Reducing the activity associated with the drives early is important to allow early removal from the reactor vessel head. Flushing the drives must be considered carefully, however, since it could result in raising the activity level near the top of the drives if leadscrews are not removed and therefore flushing is done from bottom to top.

5.2.1.2 Once Through Steam Generators

Since access to the steam generators for component cleaning could be severely restricted by the radiation levels present, these components present a particular problem. The upper and lower portions, plenum or dome areas, allow some access through manways but close access is required if much cleaning is necessary for the upper tubesheets or tubes. As a result, remote decontamination techniques may have to be employed. A wet vacuum method could provide for debris removal and could be controlled remotely. Since core fragments may be lodged inside OTSG tubes, a means of identifying the affected tubes and either clearing or plugging the tubes should be considered. This could be done by remote free-path inspection from either the upper or lower dome areas. Additional decontamination of the tubes could be done by flushing each tube with a jet stream from the top while collecting the rinse water at the bottom and either recirculating it through a filter or collecting it for treatment and disposal. A device similar to the type used for the free path inspection could be modified for positioning the hydro laser. Setting up this equipment generally requires close access to the generator, and special dome shielding could be employed to help control total exposures.

5.2.1.3 Reactor Vessel Internals

The lower internals are activated by core neutrons. Radiation level measurements made during the surveillance specimen holder tube replacement work at other plants in 1976 showed 3000-6000 R/hr radiation fields adjacent to the lower internals. TMI-2 has operated only 93 EFPD, and there will be significant decay time before internals inspection. However, radiation levels from activation of base metal are expected to be at least 100 R/hr when inspection is required. Decontamination will not remove this source. Remote inspection will be required because of the activation. Therefore, no special decontamination of the lower internals is recommended other than debris removal.

The upper internals (plenum) are further from the core, and have less activation. Decontamination would probably make the upper part of the plenum accessible to inspection. The activation contribution to dose rate from the lower part of the plenum is unknown. If detailed inspection of the plenum is needed, detailed calculations should be performed to identify the amount of activation present. The effects of decontamination could then be better identified.

If studies show plenum decontamination to be useful, they could best be decontaminated with the total system after fuel is removed. No special setup would be required, and no additional radwaste volume generated. If plenum decontamination is needed but RCS decontamination is not planned, a special plenum decontamination system must be developed.

Other areas of the RCS requiring special attention include the letdown and makeup system lines as well as vents, drains, instrumentation connection, and the sample lines. An evaluation on a case-by-case basis could be made of these areas considering such items as:

- cost and ease of replacement
- cost to shield
- extent of contamination

Finally, one major component, the pressurizer, deserves particular attention. Due to low flow velocities and limited access, the pressurizer may not be effectively decontaminated by application of the techniques discussed thus far, with the possible exception of hydrolasing. Removal and decontamination or replacement of the pressurizer heaters could resolve part of the concern. The same considerations should be given to the spray line, spray valves, and vent connections. The replacement costs and exposure costs factor heavily into those considerations. Contamination in the pressurizer vessel should be swept into the reactor coolant system before RCS cleaning, if practicable. If necessary, the pressurizer could be isolated and cleaned separately. This separate decontamination could be mechanical or chemical, with connections being made to the spray and surge lines for a recirculation path.

5.2.2 Chemical Decontamination

A comprehensive chemical decontamination effort may be required to meet all of the objectives of the decontamination program. A material compatibility testing program along with a solvent research and development program should preceed the actual implementation. A long lead time is required to identify the particular type of film present in this RCS and to develop and qualify the solvent for use. Part of the qualification of any solution would be to identify the long term effects on the integrity of the reactor coolant system pressure boundary. Chemical techniques such as electropolishing could be used for local decontamination with full or partial system decontamination also done by chemical means. The long contact times that will be encountered as a result of the filling and draining of the RCS with a chemical solution should be considered in evaluating the effects of any solvents. Furthermore, in order to completely drain the system, the reactor vessel head has to be removed, adding significantly to the contact time of the reactor vessel. A means to pump out the reactor vessel with the head on could be devised to reduce this time. Other factors relating to the chemical approach include a means to agitate the solution and provide a heat input. The reactor coolant pumps could perform both of these functions.

Both weak and strong chemical decontaminations are alternatives which require extensive preparations. However, there could be considerable activity absorbed in the corrosion layer which will not be removed by other methods. This possibility should be considered in order to allow some work to be performed in support of a chemical program until enough evidence is gathered to determine the condition of the corrosion layer in the RCS.

5.3 Recommended Decontamination Program Approach

Since both the mechanical cleaning and chemical decontamination alternatives have positive aspects as well as limitations, a balanced combination is recommended to provide the assurance of meeting the stated objectives. This will permit the development work to proceed for the chemical approach in time to be implemented should it be required. The mechanical techniques may be applied both before and after a chemical cleaning to provide for optimum activity removal. As more information about the condition of the RCS becomes available, it should be factored into the decontamination program.

Information such as detailed component radiation surveys and swipe samples from interior RCS surfaces should be obtained as access to these areas becomes possible. This type of data could define the nature and composition of the contamination and therefore provide means to evaluate the potential effectiveness of the various decontamination techniques being studied. It may be possible to collect enough evidence to concentrate the overall decontamination efforts on a single approach. Planning and testing should begin soon in order to be prepared with a complete decontamination program regardless of any new data received.

Another source of valuable information concerning the activity and tenacity of the film coating the RCS could be available for artifacts such as control rod drive mechanisms and steam generator inspection port or manway covers. Laboratory evaluation of the various decontamination techniques to be used in the field could be performed on such items to help eliminate or modify the methods to achieve acceptable results.

The need for specific component decontamination to support other recovery requirements should be identified in the integration of recovery efforts. The schedule of recovery efforts should also consider the impact of decontamination work, including contingency planning should initial decontamination efforts not conclude with the expected results.

5.4 Support Systems and Equipment Needs

In order to effectively carry out a reactor coolant system decontamination, equipment used for the containment decontamination program could be used for RCS cleanup, such as radwaste storage, scrubbing, and hydro lasing apparatus and temporary shielding. Remote video coverage for inspection, monitoring, and supervision of the decontamination tasks should be provided. The reactor coolant pumps may be used to provide circulation within the RCS. Since the pumps are required for future plant operations, they should be fully refurbished at the earliest opportunity to facilitate the cleanup. This implies that pump supporting systems must be available also. Special equipment may be required for component cleaning activities such as reactor vessel cleaning, steam generator and pressurizer cleanup, and equipment to remove and shield the pressurizer heaters. This equipment should be designed and tested after evaluation of radiation survey information to provide some indication of the environment expected during the actual decontamination operation.

5.5 General Planning Requirements

The reactor coolant system decontamination program has many aspects which have to be integrated into the overall recovery effort. Present PWR decontamination technology is oriented toward removing well-formed, tenacious corrosion films. Because of the conditions in the TMI-2 system, separate research efforts may be needed for developing decontamination techniques. Plans for handling and disposing of the radwaste volumes generated should be incorporated into the containment recovery operation. An estimate of the man-rem costs of RCS cleanup should be made both for the overall system and for individual components. The decision to pursue further decontamination attempts has to consider this cost against the consequences of leaving the system as is. Since a major effort will be made to inspect and recertify the RCS, much of this effort will not have to be duplicated as a post-decontamination evaluation. This represents a considerable savings since this cost (inspection/ certification) could otherwise by a large part of any decontamination program.

5.6 Data Requirements

In order to assess the condition of the RCS and true magnitude of the contamination problem, information from the in-containment radiation survey is essential. Detailed radiation surveys around the RCS piping and accessible major components is required to evaluate the contribution to the general area and local radiation levels from the activity inside the system. The exterior surface contamination is distributed throughout containment and must be considered when evaluating the RCS survey information. As interior surfaces of the reactor coolant system can be made accessible, attempts should be made to determine what activity is present and how it can be removed. This can be accomplished in the upper dome of the steam generators and the reactor vessel head during the defueling evolutions. Evaluation of radiation survey information should distinguish between the radiation levels attributable to both soluble and insoluble activity. Reactor artifacts should also be obtained as early as possible in order to assess the distribution of the activity within the RCS. With information on the amount and type of activity deposited on surfaces and the ease of its removal, decontamination efforts can be focused on a reference method.

6.0 SERVICES AND FACILITIES

6.1 Remote Viewing

6.1.1 Introduction

The objective of this section is to define the scope and the criteria for a closed circuit television (CCTV) system for a number of different operations, provide a remote vantage point for observations, and make visual and audio records for examination later, or for training purposes.

Since radiation levels and other conditions are not specifically known, contingency options are included to address as many variables as possible. In the following conceptual discussion, the various components and devices are identified, described, and related to each other as an operating system.

It is expected that high-resolution still photography will also be employed to maintain a pictorial record of conditions.

6.1.2 System Functions

The CCTV system should provide a means of continuously surveying equipment or work activities inside the containment from monitoring and control stations located in the command center. The system should provide the means to remotely control from the monitor/control stations all cameras and their functions, such as pan, tilt, focus, zoom, and on/off.

Additionally, the system should provide the means to record the output of any camera on video cassette recorders (VCR).

Work crews waiting to enter the containment should be provided with the capability of monitoring (but not controlling) any camera in the system. A space and monitor(s) should be provided in the personnel access facility for this purpose, so that each team may familiarize itself with the in-containment activities prior to entry.

Training facilities should also be provided comprising a color VCR, a color monitor, and a portable (battery pack) color camera and color VCR. The color monitor and VCR should be in a cabinet on casters so that it could be used in the command center or other locations for training sessions.

6.1.3 System Design Requirements

The design of this system should be based on visually monitoring and/or recording all aspects of the defueling and RCS cleanup operations, including monitoring of worker exposure time intervals. Extensive video systems will be required to control and support the various cleanup, decontamination, and reactor disassembly efforts. The following are some of the specific areas and functions to be covered:

- Area coverage of the 347' elevation
- Area coverage of the fuel transfer canal
- RV closure stud removal
- RV head lift (underwater cameras)
- Plenum lift (underwater cameras)
- Core top examination (underwater cameras)
- Fuel assembly removal (underwater cameras)
- Internal core examination (underwater cameras)
- Special operations dictated by fuel condition (underwater cameras)

6.1.3.1 Cameras

The cameras will fall into two broad categories: conventional and underwater.

The conventional (nonsubmersible) cameras should be enclosed in environmental housings, capable of operation in high humidity and high ambient temperatures. The cameras should be radiation tolerant.

The cameras should be fitted with remotely controlled zoom lenses to provide for both closeup and distant viewing.

Extreme variations in lighting can be expected, ranging from perhaps 0.1 footcandles or less to as much as 250 footcandles or more. The camera should therefore be selected to provide adequate sensitivity to the lower level, with an automatic attenuation for the higher levels. All cameras should have digitized camera number identification as part of the composite video.

The underwater cameras should be capable of submergence in water and be radiation tolerant. Four cameras should be equipped with right-angle viewing adapters to permit viewing of the fuel assemblies as they are lifted from the core, and for inspecting vacated fuel assembly channels. All underwater cameras should be equipped with self-contained lights.

The conventional cameras should be installed on remotely controlled pan-and-tilt mounts which may be installed on heavy duty tripods, camera dollies, or wall mounts, depending on the area to be covered. Three of the underwater cameras should be installed on underwater pan-tilt mechanisms.

6.1.3.2 Video Monitors

There should be two monitor/control stations in the command center. Each monitor/control station should be equipped with four 9-inch monochromatic (black and white) monitors, nested together in a two-over-two configuration. One station should be dedicated to non-underwater operations, while the other station should serve all underwater monitoring and examinations.

A separately switched 23-inch monitor should be provided for both monitor/control stations, and should be capable of monitoring all cameras in both viewing groups. This monitor would be used by interested casual observers, such as visitors and management. In a similar arrangement, a separate 23-inch monitor should be placed in the personnel access facility.

Four 9-inch monitors, in a two-over-two configuration, should be located on the fuel handling bridge. These monitors should be capable of monitoring all of the underwater activities by selective switching. They will be used by the crane operator for examination of the core and fuel assemblies.

6.1.3.3 Remote Control

All cameras should have remote control capabilities for all functions including off-on.

In order to minimize the number of circuits entering and leaving the containment through penetrations, a digital or multiplexed control system should be employed.

6.1.3.4 Recording Devices

In order to provide a permanent record of selected events or operations for archives or training, video recorders should be utilized.

A video cassette recorder should be located in the viewing center with the capability of selecting and recording the video from any one of the seventeen cameras. A video cassette recorder should be provided for training purposes, with the capability to switch video from any camera in the system to the recorder. The output of the recorder should be capable of being displayed on the training monitor.

6.1.3.5 Color Video Equipment

In order to provide additional training facilities, color video equipment should be provided for playback of offsite produced training tapes, as well as recording special onsite events. The color equipment should not be connected into the monochrome system.

Color equipment should consist of a battery-powered portable color VCR with hand-held camera, plus a stationary VCR and monitor.

The recording format selected should be compatible with existing offsite equipment.

6.1.3.6 Power Requirements

Except for the battery portable equipment discussed in Section 6.1.3.5, all equipment should be operated on 120 volt AC, 60 Hertz grounded neutral system. Connections should be made through standard grounded plugs.

6.1.3.7 Containment Penetration Requirements

Although there should be enough spare conductors on existing penetrations for the control cables, there probably will not be enough triaxial penetrations for the video cables.

For this system, the best approach is to route the cables through temporary penetrations provided at the access control point or through nearby spare containment penetrations and sealed with epoxy or impermeable mastic material.

6.1.4 System Description

The conceptual design of this system permits remote visual surveillance of all aspects of reactor coolant system cleanup as well as defueling operations. The RCS cleanup operation involves more conventional types of CCTV equipment, whereas the defueling operation requires the use of underwater, radiation-resistant cameras in addition to the conventional monitoring and control equipment. For this reason, the system description will be presented as two subsystems.

6.1.4.1 General Surveillance and RCS Cleanup

The general surveillance/RCS cleanup system should be composed of ten remotely controlled monochrome cameras, with pan-tilt and zoom capabilities. Cameras should be mounted on rugged tripods, dollies, or wall mounts, depending on the nature of work, the work location, and the permanency of the location.

The interconnection arrangement of this group of cameras is shown on Figure 6.1-1.

The interconnection arrangement of the training and VIP monitors is shown on Figure 6.1-2.

6.1.4.2 Reactor Vessel Internals and Defueling

The underwater system should be basically the same as the conventional system except for the underwater cameras and underwater pan/tilt mechanisms. Each of the cameras has a separate control unit, located on the fuel handling bridge. The interconnection arrangement is shown on Fig. 6.1-3.

Three of the underwater cameras, equipped with zoom lenses and self-contained or attached underwater lights, should be installed on underwater pan/tilt heads. They should be located at the 322'-6" elevation near the reactor vessel flange at the three lift points to monitor the RV head lift. In addition, they should be used to perform inspections under the RV head early in the lift. Means should be provided to remove these cameras while still under water to permit their use elsewhere.

Four cameras should be equipped with right-angle viewers and self-contained lights and arranged on a movable mounting ring so that all four sides of a fuel assembly can be examined as it is being withdrawn from the core.

6.1.4.3 Monitoring Stations

A monitor/control center should be set up within the command center with communications to remote locations, as described in Section 6.3.

The monitors should be arranged in two groups of four, in a two-over-two arrangement. One group should be used for conventional cameras, while the second group should be used for all underwater work.

A 23-inch monitor capable of independently monitoring all cameras in the system should be located nearby the monitor/control center, preferably in a separate room. This monitor should be used for visitor and management briefing. Another 23-inch monitor, capable of monitoring all cameras in the system, should be set up in the personnel access facility.

In-containment monitors should be set up on the fuel handling bridge to monitor defueling operations and assist crane operators. Four 9-inch monitors in a two-over-two arrangement should be utilized.

6.1.5 System Operation

Defueling and RCS cleanup, as well as other in-containment operations, should be controlled from the command center. The system should be utilized for management and supervision, rather than strictly monitoring purposes.

For the most part, all conventional cameras will be used to provide coverage of the fuel transfer canal and the 347'-6" elevation. The cameras should be positioned as required to cover a given operation of the cleanup, and, once in position, should be controlled by the command center.

The removal of the studs from the RV flange and parking the studs on the RV head flange should be covered by three or more tripod-mounted cameras. If the radiation levels are too high, and the removal tool is adapted to semiremote operation, close video support via these cameras will be required.

Prior to RV head removal, and before the refueling canal is flooded, three underwater cameras mounted on underwater pan/tilt mechanisms should be installed near the RV flange at the three RV head lift points. The method of installation of the cameras and pan/tilt mechanisms should permit their removal by remote means while under water. This will permit their use for other purposes after the head and plenum are removed. These cameras will be used to monitor the RV head initially to assure a level lift. During this time, each camera output will be displayed on a separate monitor, both in the command center and at the fuel handling bridge. The cameras will also be used to perform inspections under the RV head early in the lift, approximately 2 feet above the RV flange.

Video coverage of the plenum cover flange should be provided by one or more cameras during the initial part of the lift. Video inspections of the plenum should continue throughout the lift, including the exterior.

Much of the video coverage should be recorded on tape, both for record purposes and for additional review later.

During removal of the first fuel assembly, four cameras should be positioned in such a manner that all four sides can be inspected as the assembly is removed. The entire removal sequence should be recorded on video tape. For this purpose, four cameras with right-angle viewers and zoom lenses should be vertically mounted on a suitably sized ring spaced 90 degrees apart. Each camera should be oriented in such a way that it will "look" toward the center of the ring. As the fuel assembly passes through the field of view of the camera, the zoom lens should be adjusted to provide the desired coverage.

Fuel channel inspection should be accomplished after the first fuel assembly is removed. A camera with a right-angle viewer and zoom lens is lowered into the channel and all sides are inspected by rotating the vertically hung camera.

Most of the conventional cameras and accessories can be used during the reconstruction phase for traffic control, HP purposes, and security functions. At that time the monitor/control stations could be relocated to other, convenient locations.

6.2 Radiation Monitoring System

6.2.1 Introduction

The radiation monitoring system for the initial containment entry and decontamination phases was originally discussed in the Planning Study for Containment Entry and Decontamination, Section 4.2.13, Containment Radiation and Radioactivity Monitoring Program.

The assumption was made that all previously installed radiation monitors and reactor instrumentation inside the containment could not provide the required reliability and accurate indications, and these monitors should be replaced, or at least supplemented until they can be replaced.

The recommendation for the radiation monitoring system described in this section is to provide protection during initial containment entry and decontamination (Phase I), reactor disassembly, fuel removal, and reactor coolant system cleanup operations (Phase II).

The fundamental objective of the radiation protection program is to limit radiation doses. The radiation monitoring system (RMS), in part, indicates the effectiveness of this program. The basic function of the RMS for the initial containment entry, reactor vessel internals disassembly, fuel removal, and reactor coolant system decontamination is to provide:

(1) Rapid and reliable information on the magnitude and location of the possible radiation hazards, associated with the operations, for the purpose of defining the type and extent of any necessary countermeasures, implementing emergency procedures. or reevaluating/redirecting the operation activities.

(2) An assessment of any hazards from airborne radioactive materials to compare the concentration of radioisotopes in the atmosphere to the maximum permissible concentration (MPC) levels, and to serve as a basis for the planning of immediate corrective action.

(3) An assessment of the adequacy of controls on the releases of the radioactive materials to the environment.

(4) An estimation of the probable upper limits of the potential exposure to the personnel, necessary for proper planning of all activities, time frame and input for the selection of the equipment to be used.

(5) Rapid determination of the possible contamination from processes and operations of Phases I and II.

(6) An input of prediction for any long-term changes or trends in the radiation levels and/or radioactivity concentrations resulting from the operation of equipment, and other related activities.

(7) A demonstration of compliance, if required, with the applicable regulations, environmental standards, technical specification, or other operational limits.

To achieve the foregoing functions the radiation monitoring system should be designed to:

(a) Continuously detect and record the level of radioactivity in certain process and all effluent streams to ensure compliance with the requirements of the Code of Federal Regulations, Title 10, Parts 20 and 50, Regulatory Guides 1.21, and technical specifications.

(b) Provide personnel with a continuous indication of the ambient beta, gamma, and neutron radiation and radioactivity level in selected areas.

(c) Provide alarm and indication in the command center and the control room for monitored points as well as the initiation of protective functions, as required, on abnormal increases in radiation level above the existing conditions in an area, or systems.

(d) Provide redundant systems through the measurement of effluents, both at their point of releases, and at their potential source points.

Separate, independent and redundant monitors are to be installed for selected measurements to increase level of reliability.

(e) The final selection of the type of monitors, measurement ranges, and their sensitivities will be based on the predominant isotopes and their physical forms in the system or area to be monitored.

Functionally, the RMS is divided into the following pattern:

- Area radiation monitoring (ARM) system
- Continuous air radioactivity monitoring (CAM) and sampling system

- o Process and effluent monitoring system

- o Criticality monitoring system

6.2.2 Area Radiation Monitoring System (ARM)

Area radiation monitors will utilize detectors to detect ambient beta, gamma, and neutron radiation.

ARM readout will be mR/hr for beta and gamma, and mrem/hr for neutron radiation.

The radiation detectors may be Geiger-Mueller (GM) tubes, scintillation detectors, proportional counters, ionzation chambers, or other commonly acceptable detectors, as required, to meet specific requirement for the particular measurements.

The ARM includes fixed mobile (to allow flexibility in locating monitors near the work teams) and portable monitors. Fixed and mobile ARM provide local and remote (command center) continuous display, local and remote (command center, control room) alarms, and recording (command center) of radiation levels during all operations. Alarm indication, audible and visual, includes both high radiation levels and monitor channel* failure. A fixed ARM will be continuously recorded on a multipoint recorder.

A mobile ARM will use a singlepoint recorder to obtain a trend chart for different areas.

6.2.2.1 Containment Building

The beta-gamma fixed, mobile and portable ARM will be used to continuously monitor the radiation level in all areas where work is expected to be performed. These include the area near the RPV head closure assembly adjoining the operating floor, areas adjacent to the fuel transfer canal, and areas adjacent to the fuel storage pool.

Some of the ARM will be mobile type to provide flexibility in locating them in close proximity to working teams.

In addition to the mobile monitors, permanent ionization chamber monitors will be installed to replace the existing G-M detectors.

These ARMs will be capable of detecting wide range radiation levels without saturation by high flux, and responding rapidly to a step change in radiation at any level.

The new monitors will be installed at the same locations, or at areas where background radiation levels are expected to be the highest.

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*ARM Channel: A radiation sensitive detector, check source, electronic processing and alarm circuitry, and interconnecting cables used to measure and display the exposure rate present at the detector location.

The high radiation areas can be determined from the periodic surveys taken throughout the containment.

High radiation or loss of a channel will be alarmed locally, in the control room and the command center.

Power for the fixed monitors will be provided from the same source as the existing monitors being replaced. Power supply for the mobile monitors will be provided from the nearest available power source.

6.2.2.2 Containment Recovery Service Building (CRSB)

The monitors will be located in areas where access is restricted, and where radiation levels could possibly increase due to postulated occurrences. The local display unit, located in or adjacent to the monitored area, will be provided.

Alarm, audible and visual, will be provided in the local display units and in the command center. A shared alarm for the CRSB will be provided in the control room. Alarm indications will include both high radiation levels and channel failure.

6.2.2.3 Command Center

The shielding of the command center will be designed to ensure that the zone I* designation will apply at all times. Area radiation monitors will be located at selected areas with display, recording, and alarm located in the command center and shared alarm in the control room.

6.2.2.4 Radwaste Staging Area (RSA)

A fixed ARM will be provided in the radwaste staging area. These monitors will have local display units which include both audible and visual alarms, indication and alarm in the command center and a common alarm in the control room. Selected monitors should serve as area monitors for personnel protection as well as the monitors of the radiation levels of the waste containers during movement, storage, and off-site disposal.

6.2.2.5 Personnel Access Facility (PAF)

Area radiation monitors will be included in selected areas of the personnel access facility, with local display units providing audible and visual alarms, indication and alarms in the command center and a common alarm in the control room.

* Zone I: Provides unlimited access where maximum design radiation dose rate is ≤0.5 mRem/hr.

6.2.2.6 Auxiliary Building

New area radiation monitors in the auxiliary building will be added. A local display unit (which includes both audible and visual alarms indication), alarms in the command center, and common alarms in the control room will be provided.

6.2.3 Continuous Air Radioactivity Monitoring (CAM) and Sampling System

The air monitors will provide capabilities for continuous sampling and continuous measurement and will utilize beta and beta-gamma sensitive detectors.

The monitors consist of beta detectors for detecting noble gases, and beta or gamma detectors with fixed or movable filters which sample air for detecting radioactive airborne particles. The type of filters (fixed versus movable) will be established based on existing and predicted radioactivity concentrations.

The monitor ranges of 10^{-7} to 10^{-1} μCi/cc for noble gases, and 10^{-11} to 10^{-6} μCi/cc for particles will be used.

The CAM will include fixed and mobile air monitors. Each CAM will provide local and remote (command center) continuous display, local and remote (command center) alarms, combined alarm in the control room, and recording (command center) of radioactivity concentrations during all operations.

A recording system will be used to provide a continuous record of the particulate and noble gas concentrations for permanent record purposes.

Alarm indication, audible and visual, includes both high concentration level and air monitor channel failure. In addition, mobile high volume continuous air samplers with charcoal filters to obtain samples for onsite halogens analysis will be provided.

The capability of detecting halogens should be based on the detection of iodine-131 (range of 10^{-10} to 10^{-5} μCi/cc) providing two single-channel analyzers.

6.2.3.1 Containment Building

The fixed and mobile CAM will be used to sample and measure the airborne particles and noble gases in all areas where work must be performed.

During the decontamination and cleanup operation the CAM will be a mobile type. In addition, the high volume air samplers with charcoal or particulate filters will be used.

6.2.3.2 Containment Recovery Service Building (CRSB), and Radwaste Staging Area (RSA)

Fixed air monitors and mobile air samplers will be provided during decontamination operation, movement, and storage of waste containers.

6.2.4 Process and Effluent Monitoring System

The radiation monitors will be installed in the effluent streams of the various ventilation and selected process systems which could discharge radioactivity to the evironment.

The RMS may interface with the following systems:

(a) Reactor Building Equipment Drains

(b) Fuel Handling Building Ventilation

(c) Containment Purge

(d) Containment Recovery Service Building Ventilation

(e) Reactor Building Ventilation

Each channel of process radiation monitoring system will consist of a remotely-located control cabinet. The detectors will be located adjacent to the monitored process. The control, readout, recording, and power supply instrumentation will be installed in the radiation monitoring control cabinet with their respective alarm annunciators located in the control room and the command center.

Each monitor channel will provide indication in the command center and in the control room and will have the capability of initiating protective or control functions on high radioactivity levels. The system will be redundant through the measurement of the effluents, both at their point of release, and at their potential source points. Wide range monitors will be selected to cover the broad range of activities expected during all conditions.

Where particulate activity levels will be measured, the system will incorporate isokinetic sampling probes.

Each particulate and gaseous monitoring channel will use a beta detector to monitor the particulate and gaseous activity. A gamma detector with spectrometer analysis will be used in the halogen monitor. This monitor will use a charcoal cartridge assembly to absorb halogen. The liquid monitors utilize assembly with a gamma detector to monitor gross gamma activity.

The effluent monitors are considered permanent.

6.2.4.1 Reactor Building Sump Discharge Line

The radiation levels of the liquid being pumped from the reactor building sump to areas outside containment will be monitored by detectors attached externally to the sump discharge line.

One detector will monitor the lower range (0~100 mR/hr), a second detector will be used to monitor the higher range (≧100 mR/hr).

The radiation level can be related to activity concentrations by means of grab samples analyzed in the laboratory.

A grab sample connection should be provided in the sump pump discharge line. Sump pump discharge samples will be used to correlate the detectors.

The reactor building sump pump discharge radiation detectors will be continuously recorded on a 2-pen recorder in the command center. A 2-pen recorder would provide a continuous record from each of the two detectors, one for each range. A high radiation signal will be alarmed locally, and in the command center.

The high radiation signal, which will be set to actuate at a predetermined level so as to avoid discharging highly contaminated liquids from the containment, will automatically terminate pumping and close the sump pump discharge line containment isolation valves.

6.2.4.2 Reactor Coolant Letdown Line

This monitoring channel will have a single, external detector continuously monitoring the reactor coolant letdown stream.

The monitor will measure the gross gamma radioactivity in this line.

The measurements will be indicated and recorded in the control room.

Alarms will be provided in the command center, and in the control room.

Provisions will be added for periodic grab sample analysis in the laboratory.

6.2.4.3 Containment Ventilation Monitoring System

This system will have two redundant detection systems complete with individual isokinetic sampling to continuously monitor the containment ventilation for particulate activity, noble gases, and halogens. The measurements will be indicated and recorded on the radiation monitoring control cabinet with alarm in the control room and the command center.

6.2.4.4 Containment Atmosphere Monitoring System

Two detection systems will monitor the containment atmosphere for particulate activity and halogens. Each detector will continuously draw a sample from the sample lines at different points within the containment. Each monitor consists of particulate and halogen filters.

Measurements should be indicated in the control cabinet with alarm in the control room and the command center.

6.2.4.5 Water Station Effluent

A single external detector will monitor continuously the station water's final discharge path to the collecton box. The detector will measure gross gamma activity. The measurements will be indicated and recorded in the radiation monitoring control cabinet. An alarm will be provided in the control room and the command center.

6.2.4.6 Waste Gas Monitor

The waste gas discharge line will be continuously monitored by two off-line noble gas detection systems. The measurements will be indicated and recorded on the radiation monitoring control cabinet. An alarm will be provided in the control room and the command center.

In addition, an interlock for closing the waste discharge header valve will be provided on high radiation signal.

6.2.4.7 Containment Recovery Service Building Exhaust Monitor

This system will monitor the air particulate activity levels from the containment recovery service building HVAC exhaust duct, and continuously collect halogens samples for laboratory analysis.

Continuous indication, recording, and alarm will be provided in the command center. A common alarm* will be provided in the control room.

6.2.4.8 Personnel Access Facility Exhaust Monitor

This system will monitor air particulate activity levels and continuously collect halogens samples for laboratory analysis.

Continuous indication, recording, and alarm will be provided in the command center. A common alarm will be provided in the control room.

*Common Alarm: Combined alarm for CRSB, PAF, and radwaste staging area exhaust system.

6.2.4.9 Radwaste Staging Area Exhaust Monitor

The radwaste staging area exhaust HVAC duct will be equipped with monitors for air particulate activity levels, and halogens samples will be continuously collected for laboratory analysis.

Continuous indication, recording, and alarm will be provided in the command center and there will be a common alarm in the control room.

6.2.4.10 Fuel Building Exhaust Monitor

The fuel building exhaust will have redundant monitors providing sample of air from the HVAC ductwork upstream of the first isolation valve. Continuous monitoring of noble gases and sample collection for laboratory analysis of particulates and halogens will be provided. Alarm for both channels, recording, and indication will be provided in the control room, with alarm in the command center.

6.2.5 Criticality Monitoring System

The criticality monitoring approach described in this section will be implemented during movement of the removed fuel inside and outside the containment, and decontamination of the reactor coolant system.

Criticality monitoring program for the vessel internals disassembly and fuel removal is discussed in Section 4.2.2, Criticality Monitoring.

Criticality monitors will be provided to detect the changes of neutron and gamma-flux levels which will be an indication of changing conditions, e.g., changing moderator configurations, mass geometry, disappearing of neutron absorbers, etc.

A monitoring system, using neutron and gamma radiation detectors in coincidence to minimize any spurious signals, will provide audible and visual alarm which will be clearly distinctive from any other radiation monitoring alarms provided locally, in the command center and in the control room.

For the criticality events there will be an automatic evacuation signal over the PA system. Alarm will be keyed to the building or area involved.

The sensitivity of the criticality monitoring system should be adequate to detect a criticality event that produces an absorbed dose in soft tissue of 20 rads of combined neutron and gamma radiation at an unshielded distance of 2 meters from the fissionable materials within 1 minute.

In addition, the criticality monitors will provide continuous indication of increasing neutron and gamma flux levels in the control room and the command center.

6.3 Communications System

6.3.1 Introduction

The communications system suggested for use during the initial entry and decontamination phases is described in Reference 1, Section 4.2.12. It was assumed that the existing communications system inside containment would not be operable until such time that it could be checked out for serviceability. Should any of the existing system be found operable, it may then be used to supplement the communications system suggested in the following paragraphs, for the cleanup and defueling phases.

The communications system described below does not cover radio control of cranes or robotic devices, which if used would come with their own dedicated command, feedback, and signal links determined and integrated by the respective manufacturer.

6.3.2 System Functions

The communications system should provide audio communications between a central monitoring unit located in the command center and members of the work crews in the containment, personnel in the fuel handling building, and members of appropriate support facilities located on-site. This system could be augmented by, the visual monitoring capability of the closed circuit television (CCTV) system discussed in Section 6.1.

6.3.3 System Objectives

The communications system should satisfy the following objectives:

a. The system should be a two-way voice communications system. The system should be capable of allowing any of the work crew members, fuel handling building personnel, or appropriate support facility personnel to verbally communicate with the central control unit in the command center or with each other at will. The system could also be capable of allowing work crew members to verbally communicate with one another, independent of the central control unit, by switching to a separate channel.

b. The central control unit in the command center should be equipped with earspeakers and microphone headsets in addition to separate speakers and a swivel microphone so as to be able to monitor and control all frequencies being used.

c. The audio receiver/transmitter unit worn by personnel in the containment, fuel handling building, and/or appropriate support facilities should be self-contained and compact to permit free, unrestricted movement. The headsets worn by the work crew personnel should allow them to send and receive without any deliberate use of their hands and should be capable of being worn comfortably beneath anti-contamination headdress and air masks.

6.3.4 System Description

The communications system to be used should be as simple and uncomplicated as possible, utilizing conventional proven components to ensure system availability and to minimize system downtime. As discussed in Reference 1, Section 4.2.12.3, a completely untethered battery operated radio system has been proven to afford the greatest flexibility and freedom of movement. For this application, a simplex radio communication system which both receives and transmits on a single frequency should provide an adequate communication system. Since they are all tuned to the same frequency, any one individual may talk and all others will hear his conversation. The advantages and disadvantages of a simplex radio communication system are provided in Table 6.3-1. An alternative to the simplex system would be a duplex system. In a duplex system, the central control unit (base station) would transmit on, and the portable "Walkie Talkie" type radios would receive on, one frequency, while the portable radios would transmit on, and the base station would receive on, a second frequency. This would allow the central control unit operator and/or the work crew members to simultaneously receive and transmit. The advantages and disadvantages of a duplex radio communication system are provided in Table 6.3-1. Figure 6.3-1 illustrates a typical two-way communication system.

Coaxial cable would need to be routed through a spare electrical penetration in the containment to carry the signal from the central control unit to the antenna located as high in the containment as possible, preferably on the polar crane rail. Additional antennae should be provided in the fuel handling building and in the containment recovery service building. These antennae may be a LOSSI line type which radiate along the entire length of cable. This type antennae are limited to applications where there are few physical obstructions.

Members of the work crews, as well as personnel in the fuel handling building and in appropriate support facilities, should be equipped with a portable "Walkie Talkie" type radio. These portable radios should be lightweight and capable of being attached to their body, such as to the belt, and should be cable-connected to an earphone/microphone headset worn by the work crew member. The portable radio should be battery operated complete with a rechargeable battery good for at least eight hours of continuous service. At least one recharging unit capable of charging multiple batteries simultaneously should be provided with the system. The battery in the portable radios should be capable of being recharged without the radio case being attached so that a spare battery can be installed in the radio, thus not rendering it inoperable while the battery is being recharged.

In similar applications, voice-actuated microphones have been proven to be too sensitive for this type of service, and therefore either a press-to-talk pad-type actuator or a balanced-pressure-type microphone should be used in conjunction with the headset microphones. The press-to-talk pad-type actuators can be located at a convenient location on the team member's body to allow him to be able to press the switch with his arm, elbow, or by pressing a part of his body against an obstruction to talk and would not require him to specifically use his hands. The balanced-pressure type microphone actuates on voice levels above background noise. The advantages and disadvantages of these two types of actuators are presented in Table 6.3-2 .

The headset should be an integral earphone and microphone unit capable of being worn comfortably with anti-contamination headdress and air masks. The microphone could be located at the throat, mouth, or cranium. A throat microphone could cause muffled transmissions as well as possibly becoming uncomfortable, especially if air masks are being worn. A mouth microphone could interfere with wearing the air mask. A cranium microphone is recommended since it would provide clear communications and should not interfere with either the work crew member's activities or with his wearing an air mask.

The central control unit should be a separate communications console located in the command center close to the CCTV viewing panel. The central control unit operator should be able to transmit to, and receive communications from, any individual in the containment, fuel handling building, or appropriate on-site support facility at will.

6.3.5 System Operation

Voice communications with the work crew members inside the containment, personnel inside the fuel handling building, or personnel in the appropriate on-site support facilities should originate from and be controlled by the central control unit operator located in the command center. The central control unit operator should have a continuous communications link with the personnel having portable radios inside the containment and in the fuel handling building (if manned). If the existing containment page system is determined to be operable, it could be disconnected from the plant page system and dedicated to the cleanup and defueling operations under the control of the central control unit operator as a backup emergency notification communication system. Any other existing in-containment communications systems found to be serviceable could be used as backups to the above described systems if required.

During the reconstruction phase, the sound-powered telephone system should be made serviceable, which could function to augment the portable radios. The system could be suitably enhanced by providing additional handset stations, paging speakers, and portable handset/speaker stations.

A recorder should be located in the command center to record all voice communications for playback at a later time if necessary. The recorder could be a voice-actuated or a continuous running, multichannel, low-speed recorder that records on one reel of tape. One channel of the recorder should be dedicated to recording time codes, to enable identification of the recording at a later date, while the remaining channels can be used to record audio communications on each frequency, from each work crew or from each location. For this application, a multichannel Dictaphone Series 4000 Communications/Logging Retreival System, or equal, should give satisfactory service.

The radio communications system could contain several optional features. These features include a talk-around, time-out timer, emergency signal, selective signalling, and private line or protected voice capabilities.

The talk-around feature would apply mainly to a duplex system. This feature would allow the work crew members to communicate among themselves without tying up the central control unit circuit. The "Walkie Talkie" radios would be equipped with a switch which, when operated, would allow the radio to transmit at the same frequency as the base unit, but at a much more reduced power level. However, the central control unit operator would still have the capability of communicating with the work crew members without interference, since the output from the base unit is much stronger than that of the portable radios. For a simplex system, this feature would not be required, since only one frequency is used to both transmit and receive.

The time-out timer feature automatically shuts off a keyed transmitter after a selected time interval. This feature would prevent the disabling of the communications system by a transmitter being inadvertently left keyed-in, especially if a simplex system is used. The emergency signal feature could be initiated by any work crew member to initiate a visual alarm on the central control unit if he is in danger and is unable to communicate. The central control unit would provide a visual readout identifying the specific radio that is transmitting the signal. A list would have to be kept to correlate alarm signals to the assigned radio in order to determine which individual was in trouble. The selective signalling feature would provide the central control unit operator with the capability of selectively paging any specific individual without tying up the communications channel. When a page request is initiated, the number would be passed through the local telephone exchange to the paging terminal via a single telephone trunk connection. After a validity verification, the terminal would activate the radio transmitter through a two-wire telephone line. The transmitter would broadcast a radio signal, which is then detected by the paging receiver, which is worn by the person being paged.

The private line and protected voice features use a coding and decoding system which would provide the capability of preventing any receiver outside the plant that does not have this feature from monitoring the communications taking place. The private line feature would superimpose an audio carrier on the selected transmitting frequency, and the receiver would then be designed to key only if this audio carrier is present. The protected voice feature would automatically scramble the transmission so that anyone listening on the same transmitting frequency would hear only electrical noise unless the receiver was equipped with an identical de-scrambler. The protected voice feature would require a special industrial license or a waiver on the existing license, plus it cannot be used for a permanent installation.

To provide greater reliability of the communications system, the central control unit should be powered from a non-interruptible power source with the capability of being automatically transferred to a nickel-cadmium battery power supply. Consideration could be given to supplying power to the central control unit from an on-site vital power source. However, if this is done, an evaluation would also have to be made on the possibility of the radio communication system jeopardizing the reliability of this vital power source. Provision of redundant transmitters should be considered to enhance system reliability.

6.4 Temporary Lighting and Power System

6.4.1 Introduction

The temporary lighting and power systems suggested for use during the initial entry and decontamination phases were described in Reference 1, Section 4.2.4. It is assumed that the existing lighting and power systems in the containment are inoperable until such time that they can be checked for serviceability. Should any of the existing systems be found operable they should be utilized in lieu of, or to supplement, the temporary systems described herein.

6.4.2 System Functions

The temporary lighting and power system provides adequate lighting and power for the initial entry and decontamination phases of the recovery effort. The lighting should include overall area illumination from portable lighting fixtures which are set in fixed locations throughout the containment. Portable fixtures should be utilized to provide higher illumination levels, as required by the visual task at specific work locations. These fixtures can be moved from one location to another as the work is completed in a given area. Battery-powered lights should be provided for backup egress lighting in the event that normal power fails. These fixtures should be designed to detect loss of normal power and turn on automatically similar to the normal plant emergency lighting design.

Lighting levels should be consistent with the requirements of the closed circuit television system described in Section 6.1.

Temporary power outlets should be provided for the temporary lighting, closed circuit TV, electric heaters, electric steam jennys, hand-powered tools, radiation monitors, and welding outlets.

6.4.3 System Design Requirements

The temporary lighting or power can be served from the existing electrical system in Unit 2 via the containment recovery service building electrical system. Service at 480 volt, three phase, and 120/208 volt, three phase, four wire will be sufficient for all temporary electrical requirements.

The lighting system should be designed to provide adequate lighting levels for safe egress throughout the containment and with equipment suitable for the environment inside the containment.

General area lighting should be provided by permanently located portable fixtures served by portable waterproof cables and locking type connectors. The fixtures should be height adjustable and capable of being aimed in any direction and should be watertight with protective guards to prevent damage to glass lenses or lamps. Lamps should be 500 watt or larger tungsten halogen (quartz iodine) type as required. Mercury vapor lamps should not be utilized in order to preclude introduction of mercury into the NSSS due to possible lamp breakage. Ease of maintenance, setup, and lamp replacement and portability should be considered in fixture selection. The fixtures should be compact or capable of being disassembled so that the disposal of contaminated parts may be easily accomplished.

Lights should be installed in key locations such as equipment rooms or areas, passages, or stairways. Additional lighting as required should be added to work areas to supplement general area lighting for TV coverage purposes.

Portable hand-held incandescent-type work lights of a watertight design should be used to provide illumination in confined spaces where normal lighting cannot reach. Fixture design should permit hanging of the fixture from supports and the fixture should be provided with globe and guard for lamp protection. Cable for portable hand-held fixtures should be provided with watertight locking plugs.

Since personnel safety is of prime importance, shock hazard should be eliminated by suitably grounded equipment which normally consists of a separate ground wire in all portable cables and ground connections in portable equipment which provide a continuous cable connection to the station ground grid. Ground fault circuit interrupters should be provided for circuit ground fault protection.

Portable emergency lights, battery operated, should be provided in the event that normal power is lost, to permit safe egress throughout the containment work areas. These fixtures should monitor the temporary power system and switch on when the power is lost. Lighting levels for emergency lighting should approximate the original plant design with the exception that during critical operations additional battery-powered fixtures may be required to provide adequate illumination of the work area until the critical task is completed or put in a safe state.

The temporary power to the containment from the containment service building electrical distribution system will be supplied via bulkhead-type receptacles and connectors at the temporary door and closure which will replace the present equipment hatch. The bulkhead receptacles will permit uninterrupted temporary power supply to the containment when it is required to close the temporary door.

Temporary power distribution can be accomplished by portable spot-power distribution units supplied by portable 480-volt cables. These units can provide watertight locking receptacles for low voltage outlets to supply all portable lighting and power consuming equipment required for decontamination. These units should be served by 600-volt heavy duty power cable with a ground conductor and watertight locking connectors similar to the cable used in the mining industry. The rugged cable construction will prevent damage to the cables and provide additional protection against short circuits and possible fires.

Sufficient quantities of spare fixtures, lamps, and extension cords should be stocked in the containment to reduce maintenance/ replacement time. Extension cord should be kept to the shortest practicable length to reduce quantity of contaminated material and to reduce the possible spread of contamination.

6.5 Decontamination Systems

6.5.1 Introduction

6.5.1.1 Background

The overall recovery planning should include consideration of supporting systems and equipment to support the Reactor Coolant System (RCS) decontamination program. As indicated in Section 5.0, "Reactor Coolant System Decontamination," a number of decontamination techniques are available. Formulation of a detailed program should ensue as more is learned about the containment and RCS environments; however, some support requirements can be identified now and contingency planning initiated thereon. A considerable amount of radioactive waste processing can be foreseen in terms of initial RCS cleanup (prior to defueling) and potential chemical decontamination (after defueling). To enable contingency planning, decisions must be made soon regarding the availability of existing systems for this effort.

6.5.1.2 Mechanical Techniques

Since the containment decontamination program precedes the RCS decontamination, a considerable amount of mechanical decontamination equipment should be readily available. As there are a number of experienced companies dealing in mechanical decontamination it is expected that relatively short lead times will be required to acquire all but very specialized equipment, which may have to be designed specifically for the task. Therefore, to some extent, RCS decontamination planning will be directed by the containment decontamination effort. Generally speaking, it is expected that mechanical decontamination techniques can be readily supported by the then existing plant facilities.

6.5.1.3 Chemical Techniques

Available information relative to the reactor coolant chemistry and plant operating history indicates that some type of chemical decontamination may be necessary after defueling. Not a great deal of experience can be directly related to the TMI situation because (1) most RCS decontaminations, to date, have dealt with plants involving different materials, different coolant chemistry, and different chemical composition of corrosion layers; and (2) the RCS at TMI has cesium- and strontium-related compounds versus the typical corrosion products. Uncertainty about the nature and extent of the corrosion layers and deposition within the RCS precludes selection of a specific process at this time. The spectrum of chemical decontamination techniques ranges from single-step dilute, or "soft," chemical solutions producing relatively small volumes of filter and demineralizer resin waste products to multi-step concentrated, or "hard," chemical solutions producing as much as approximately 1.5 million gallons of chemical solutions and rinse water which must be disposed of. There is also a question as to whether a full or partial system chemical decontamination will be required, or desired.

It is expected that specific process development will be dependent on analysis of representative samples of the reactor coolant system piping and probably subsequent research. In view of the potential 2-1/2-year lead time required to develop an acceptable process, engineer an accompanying system, and provide equipment and personnel onsite for the decontamination, preliminary study contracts should be let as soon as possible to some of the more experienced companies dealing in chemical decontamination.

6.5.2 Chemical Decontamination Efforts

The following discussion is intended only to provide some background to show the potential scope of a chemical decontamination effort. It must be realized that differences in plant materials, operating time at high temperatures, coolant chemistry, and the resultant nature of the corrosion or deposition films preclude wholesale direct application of previous decontamination efforts to the TMI situation.

6.5.2.1 SENA Decontamination

At the SENA plant, an 825-MWt PWR located near Chooz, France, a chemical decontamination was accomplished on the steam generators and a portion of the coolant loops using the two-step alkaline permanganate/oxalic acid-citrate (AP-Citrox) process, with TURCO solvents, after approximately 2400 EFPH of operation. Virtually all of the decontamination equipment requirements were met using existing standard equipment, although a number of design modifications had to be made to accommodate plant-specific process requirements such as providing a demineralized water purge flow through the main coolant pump motor to prevent any decontamination solution contact with the pump's graphitar bearings. Extensive review of material compatibility, testing of components and modified systems, training of operators, and preparation of operating and emergency procedures, and trial operations, were necessary prior to commencing the decon. The required solution temperatures and agitation were achieved by operating the main coolant pumps. In addition to the waste water generated during testing and trial operations, three system volumes of rinse water followed each chemical application, all of which had to be processed. The decontamination effort was successful in that DFs achieved ranged from 40 to 200.

6.5.2.2 Douglas Point Decontamination

The August-1975 decontamination of the Douglas Point plant in Ontario, Canada, was the first large-scale demonstration of the fully developed CAN-DECON process (a process using a regenerative dilute reagent) and was directed primarily at the carbon steel piping and valves. The Douglas Point reactor is a 200 MWe CANDU PHWR (CANada Deuterium Uranium Pressurized Heavy Water Reactor) which has been operational since mid-1967.

The AP-Citrox process was considered when a study was started to develop a decontamination procedure which would be effective on carbon steel. For a number of reasons, the most predominant being closely linked to the heavy water reactor design, AP-Citrox was not considered feasible. Subsequent development led to the CAN-DECON concept. This process involves the addition of dilute chemical reagents directly to the coolant of a shutdown reactor. The reagent attacks the surface oxide layer and releases both particulate and dissolved material to the coolant. A continuous high flow of coolant is passed through sub-micron filters and cation-exchange resin in the reactor purification circuit. The filters remove the insoluble matter while the resin removes dissolved metal ions and at the same time regenerates the reagents, which are recirculated back to the primary circuit where they again attack the contaminated surfaces. The process is continued for as long as contaminants are being removed, until the ion-exchange resin is spent, or until the allotted time has expired. The reagents and remaining dissolved corrosion products are then removed by mixed-bed resin.

Using this process, with the reactor core in place, the Douglas Point system was decontaminated with a commercial reagent, Nutek L-106. Since the existing purification system capacity was inadequate, an additional purification system was added. Prior to injecting the reagent, the coolant system alkalinity was reduced by passing the coolant through a mixed-bed ion exchanger. The operating temperature maintained during the process was 90 C (194 F). Upon completion of the decontamination, the coolant chemistry was restored and the reactor was ready for startup.

Waste generation consisted of approximately 1000 cubic feet of spent filter cartridges and ion-exchange resins. The entire process, including time required to interchange the temporary and permanent purification systems (before and after the decontamination), took a total of 72 hours, resulted in DFs ranging from 3 to 6 for carbon steel surfaces, and removed approximately 250 Ci or Co-60 from the system. In terms of the initial objectives the treatment was considered successful and is intended to be repeated on a periodic basis to keep primary plant radiation levels down. Development work is in progress to apply this technique to light water reactors.

6.5.2.3 Dresden Decontamination

Planning for the Dresden 1 decontamination, now in the preparatory stages was initiated in 1972 when solvent selection studies were undertaken by Dow Chemical Company. The plant is a 700 MWt (207 MWe) BWR, located near Morris, Illinois, which has been operational since August of 1960 and has accumulated approximately 80,000 EFPH. After extensive study of commercially available solvents, a decision was made to develop a solvent specifically for this project with the hope of achieving higher decontamination factors and reducing waste volumes. The resultant Dow NS-1 solvent was then demonstrated compatible with exposed materials during an intensive materials test program. The subsequent engineering effort necessitated addition of some process-specific pumps and a heat exchanger, and approximately 500,000 gallons of liquid storage capacity. Since the existing radwaste treatment capabilities were limited, a new radwaste facility was also added for concentrating and solidifying the expected liquid wastes.

The process is to be applied after the reactor core is defueled. The NS-1 solvent will be injected, and circulated for approximately 100 hours at an operating temperature of 225 F (107.2 C). The first of 2-3 rinses will contain chemicals to remove any potential copper deposits. The combined solvent and rinses are expected to generate approximately 440,000 gallons of liquid waste for subsequent processing. This cleaning is expected to remove approximately 3,000 Curies of activity and 1,100 pounds of particulate from the system, achieving decontamination factors of 100 or more.

The Dresden decontamination is a first, in terms of a full-scale large commercial reactor decontamination. Extensive research, testing, and engineering design were involved, and it is expected that future decontaminations can take advantage of the established criteria and many answered questions evolving from this effort.

6.5.3 Planning Considerations

Regardless of the type of chemical decontamination process selected, certain common requirements should be met. A number of these considerations will require long-range planning and should be integrated into the overall recovery planning effort. Use of available existing equipment and systems should be a primary consideration as plant recovery will eventually necessitate their use in any event. Assessments of these should be made now to enable timely availability of their functions to assist in this decontamination effort.

6.5.3.1 Heat Source/Circulation

Virtually all of the chemical methods available employ solution temperatures in the 170-190 F (76.6-87.7 C) range and require agitation of the solution. The most desirable means of achieving this at TMI would be to use the reactor coolant pumps. The energy developed by the operating pumps themselves is sufficient to heat the solutions to the required temperatures. In fact, excessive heat would probably need to be removed to maintain an upper temperature limit. Use of the steam generators and accompanying secondary system in this regard appears to be the most practical method of temperature control.

The alternative is to provide additional (and probably very large) pumps and possibly heat exchangers to accomplish the desired objectives. In addition to the costs inherent with this alternative are the space and interface considerations.

Presently, the status of the reactor coolant pumps and their motors is uncertain. To make use of them in this effort, their status must be evaluated as soon as reasonably possible to allow time for any necessary refurbishment. Use of the reactor coolant pumps further implies that their support systems must also be available. An additional consideration with this option is any limitations imposed on pump operation without the reactor core in place, as the chemical decontamination is expected to commence after defueling. Furthermore, pressurizer heaters and spray or vent valves should be operable, to enable pressurizing the RCS sufficiently to operate the reactor coolant pumps.

6.5.3.2 Injection/Drain Paths

Means should be provided to inject and drain the chemical solutions, and rinse water if necessary. The existing makeup and purification system would be ideally suited for this purpose. If necessary, the decay heat removal system could also be adapted for use. Both of these systems already are connected to the RCS and have the needed pumps, heat exchangers (for cooling the removed solutions), and interconnections with tankage, chemical addition systems, demineralized water, and radwaste systems. Some system modifications may still be required, dependent on the chemical method used. The makeup and purification system is also a support system for the reactor coolant pumps, since it supplies seal injection water to them.

The alternative would be to provide auxiliary equipment which would in effect duplicate existing equipment. Equipment costs, space, and interfaces with this auxiliary equipment must be considered, and compared to restoring availability of the existing systems, in particular the makeup and purification system. It should also be noted that the makeup and purification system may also be very useful much earlier in the RCS decontamination effort, in terms of reactor coolant cleanup prior to defueling.

6.5.3.3 Waste Considerations

As indicated earlier, chemical decontamination waste generation may vary from small volumes of filters and demineralizer resins to large volumes of liquid waste which would require subsequent processing and disposal. The existing reactor coolant and miscellaneous liquids radwaste disposal systems can provide a considerable amount of storage capability and some processing capability. These facilities can be integrated with the newly installed EPICOR II system, and the presently planned submerged demineralizer system and/or the planned evaporator/solidification facility to process and dispose of these wastes. This of course presumes that these systems are available at the time they are needed, and calls for some long-term logistics planning now to ensure that this is the case. This may reveal a need for additional waste storage or processing capability.

6.5.4 Planning Approach

As chemical decontamination appears to be the longest lead time phase of the RCS decontamination effort, planning for this should be considered the point of reference. The following summarizes the major considerations involved:

a. Selection of potential vendors and commencement of preliminary studies with selected processes, in addition to the selection of a laboratory to study the surface to be decontaminated and make recommendations accordingly.

b. Assessment of reactor coolant pump operability, restoration. and useability.

c. Restoration and usability of the makeup and purification system.

d. If possible, select techniques and chemicals compatible with waste systems.

e. Pressurizer operability should be established since it will be required to assure adequate net positive suction head for the RCPs.

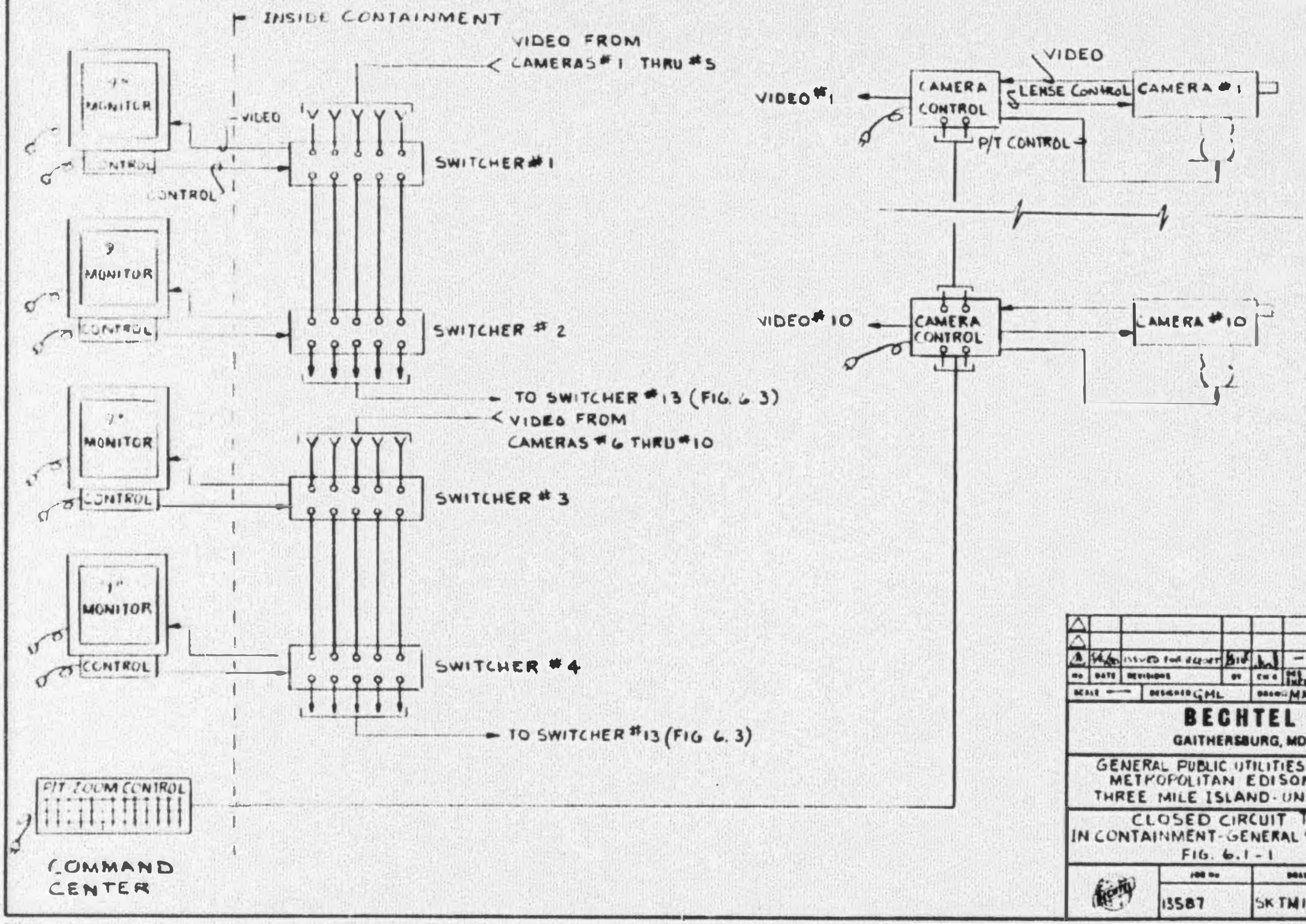
Since item a. above is especially dependent on obtaining representative samples of the RCS piping, a special effort must be made to obtain these as soon as possible after containment reentry. Logistics planning for accommodating generated wastes should also factor in wastes generated during containment decontamination as well as initial cleanup of reactor coolant activity prior to defueling. As the initial stages of the RCS decontamination procedure take shape and mechanical decontamination methods are more firmly defined, this input will also have to be factored into the logistics planning.

TABLE 6.3-1

SIMPLEX SYSTEM		DUPLEX SYSTEM	
Advantages	Disadvantages	Advantages	Disadvantages
1. Transmits and receives on one frequency	1. Cannot transmit/receive while receiving/transmitting	1. Can transmit and receive simultaneously	1. Requires more cable and antennae
2. Requires minimum cables and antennae	2. If any transmitter is accidentally keyed, whole system becomes inoperable	2. Allows private intra/inter team communications without tying up system	2. Requires two or more frequencies
3. Less sophisticated than duplex		3. If any transmitter is accidentally keyed, only one frequency becomes inoperable	3. More sophisticated than simplex and portables more complicated
4. Less costly			4. More costly

TABLE 6.3-2

PRESS-TO-TALK PAD		BALANCED-PRESSURE	
Advantages	Disadvantages	Advantages	Disadvantages
1. Easy to operate	1. Requires physical action by operator	1. No physical action required by operator	1. Miscellaneous loud noises (i.e. coughs, equipment noise, etc.) may key transmitter inadvertently
2. Will not key transmitter until operated	2. May be inadvertently bumped and key transmitter	2. No extra components required besides microphone	2. May not be able to locate on person in comfortable location and still operate properly
3. Does not need to be operated by hands		3. Keys transmitter on noise levels above background only	



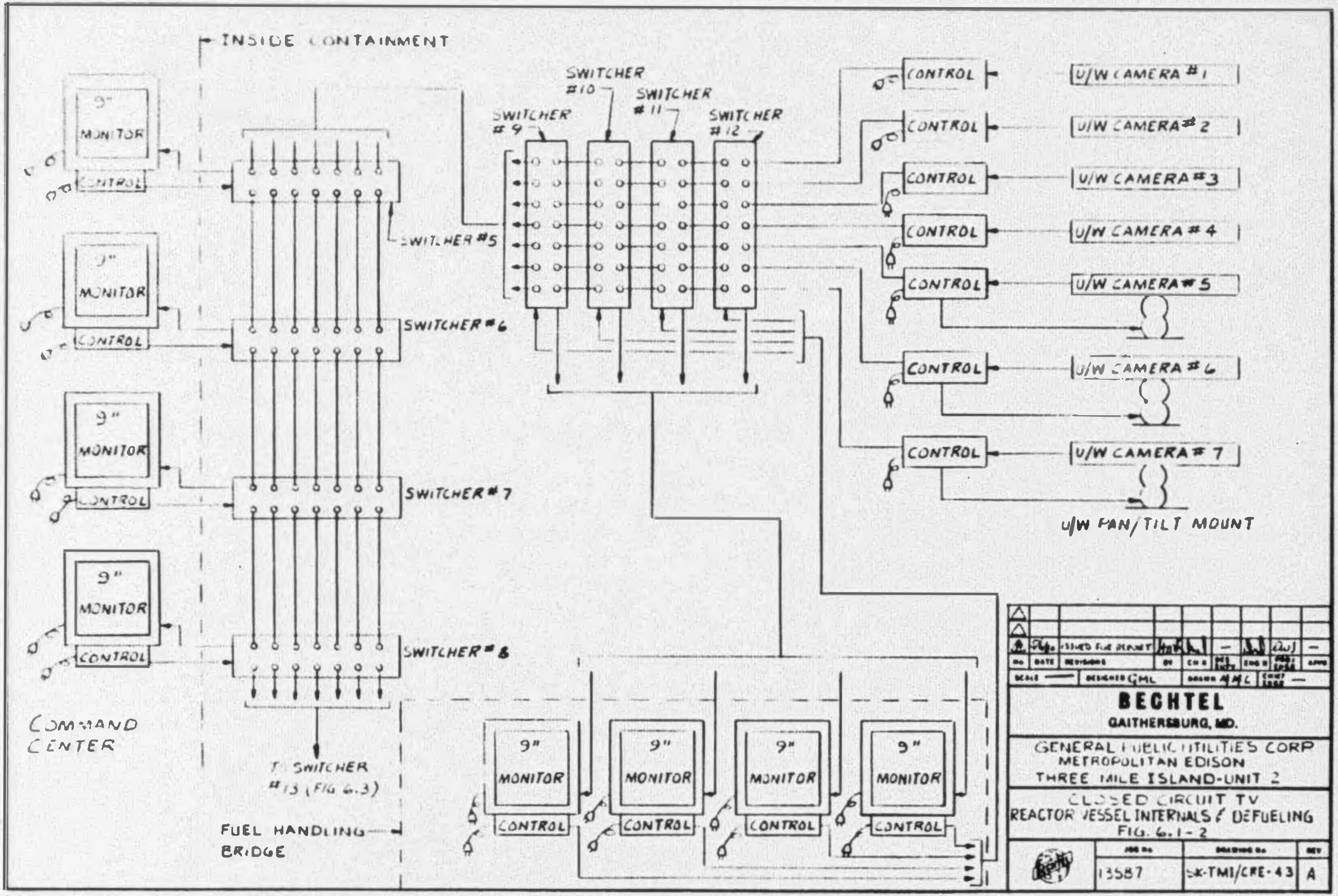
A		ISSUED FOR REPORT			—			—
NO	DATE	REVISIONS	BY	CHK				APPD
SCALE —	DESIGNED GML		DRAWN MM C.					—

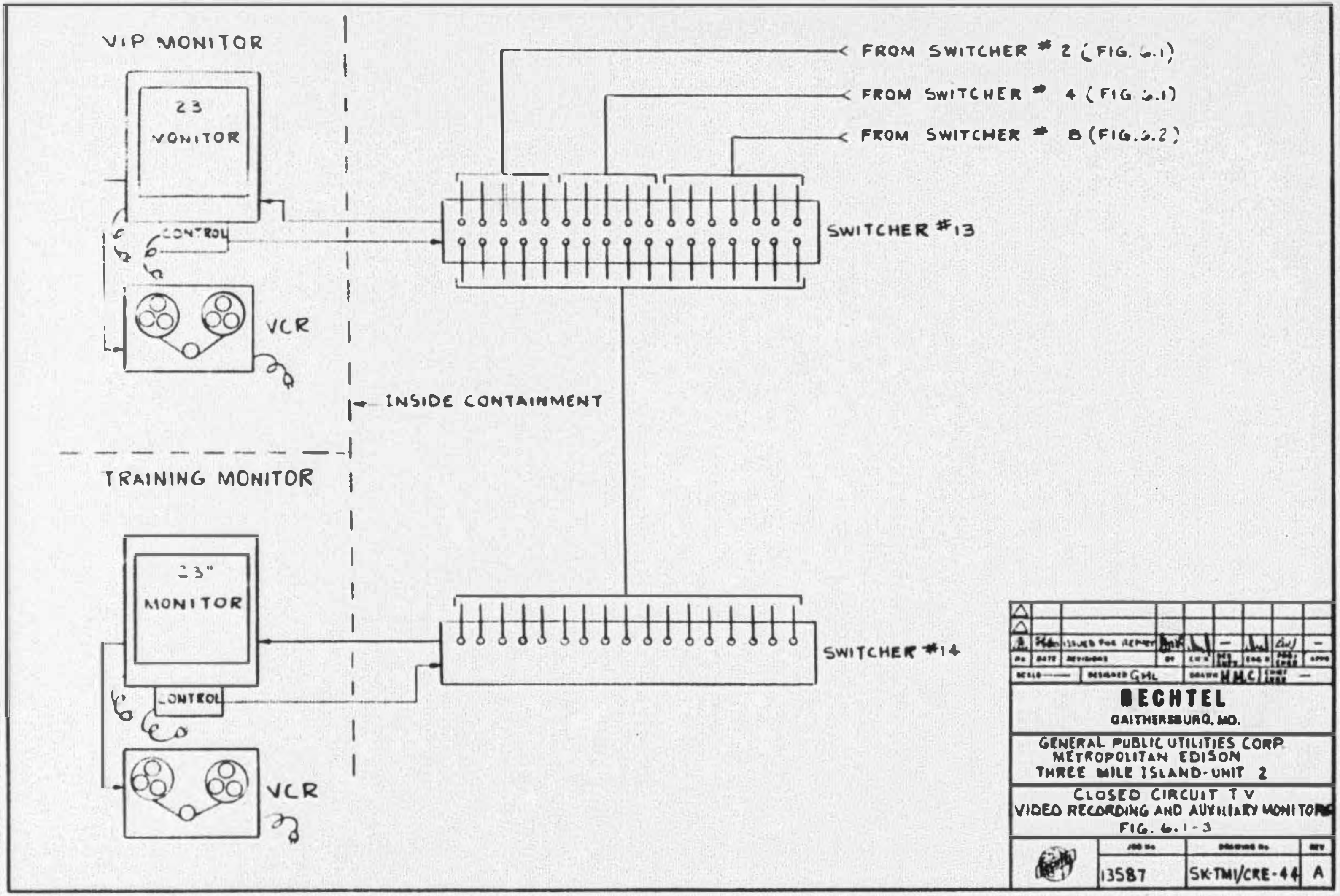
BECHTEL
GAITHERSBURG, MD.

GENERAL PUBLIC UTILITIES CORP.
METROPOLITAN EDISON
THREE MILE ISLAND-UNIT 2

CLOSED CIRCUIT TV
IN CONTAINMENT-GENERAL SURVEILLANCE
FIG. 6.1-1

JOB No	DRAWING No	REV
13587	SK TMI/CRE-42	A


INSIDE CONTAINMENT
9" MONITOR
CONTROL
SWITCHER #5
SWITCHER #6
SWITCHER #7
SWITCHER #8
SWITCHER #9
SWITCHER #10
SWITCHER #11
SWITCHER #12
CONTROL
U/W CAMERA #1
U/W CAMERA #2
U/W CAMERA #3
U/W CAMERA #4
U/W CAMERA #5
U/W CAMERA #6
U/W CAMERA #7
U/W PAN/TILT MOUNT
COMMAND CENTER
TO SWITCHER #13 (FIG 6.3)
FUEL HANDLING BRIDGE
BECHTEL
GAITHERSBURG, MD.
GENERAL PUBLIC UTILITIES CORP
METROPOLITAN EDISON
THREE MILE ISLAND-UNIT 2
CLOSED CIRCUIT TV
REACTOR VESSEL INTERNALS & DEFUELING
FIG. 6.1-2
13587
SK-TMI/CRE-43
A



ISSUED FOR REPORT

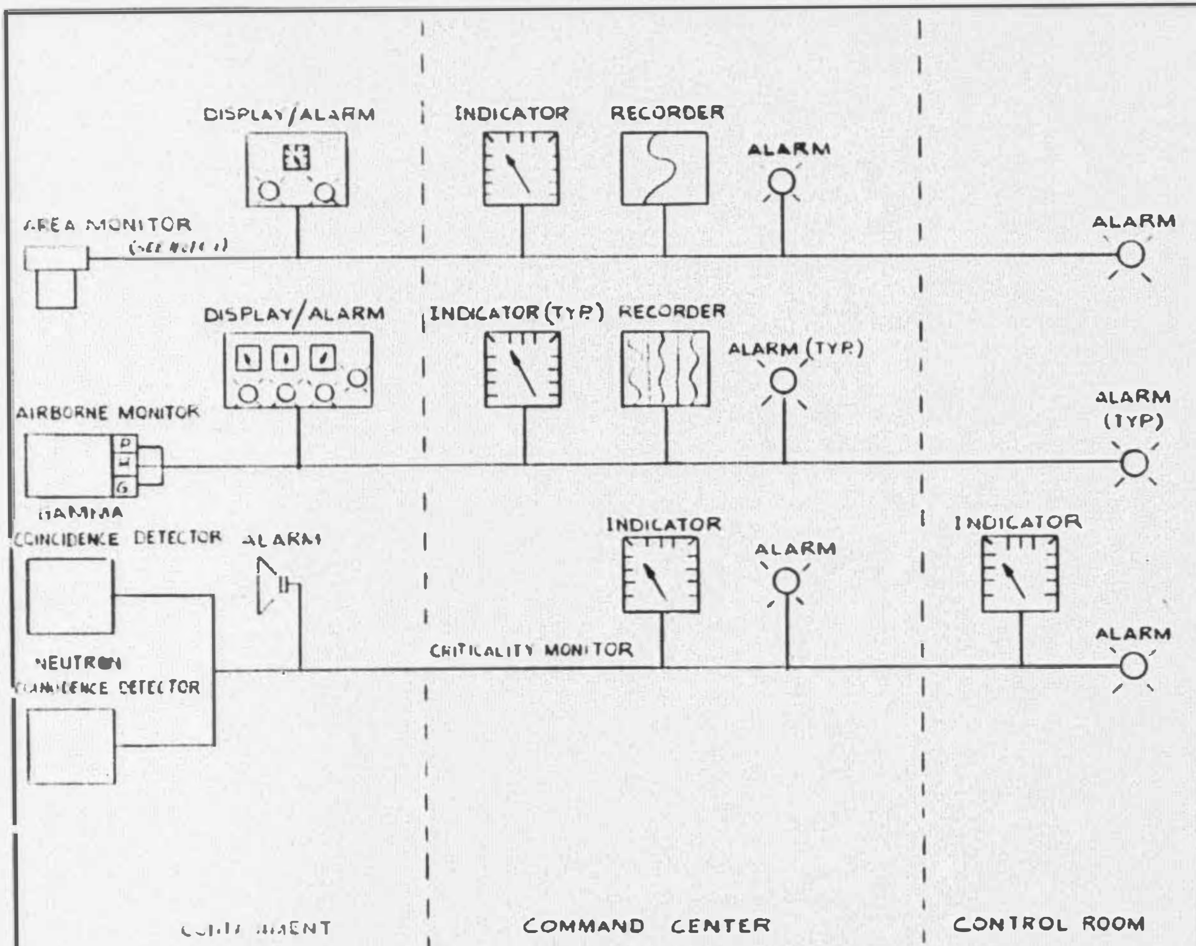
DESIGNED GML DRAWN MLC

BECHTEL
GAITHERSBURG, MD.

GENERAL PUBLIC UTILITIES CORP.
METROPOLITAN EDISON
THREE MILE ISLAND - UNIT 2

CLOSED CIRCUIT TV
VIDEO RECORDING AND AUXILIARY MONITORS
FIG. 6.1-3

JOB No 13587 | DRAWING No SK-TMI/CRE-44 | REV A



LEGEND

P - PARTICULATE
H - HALOGEN
G - NOBLE GAS

NOTES:

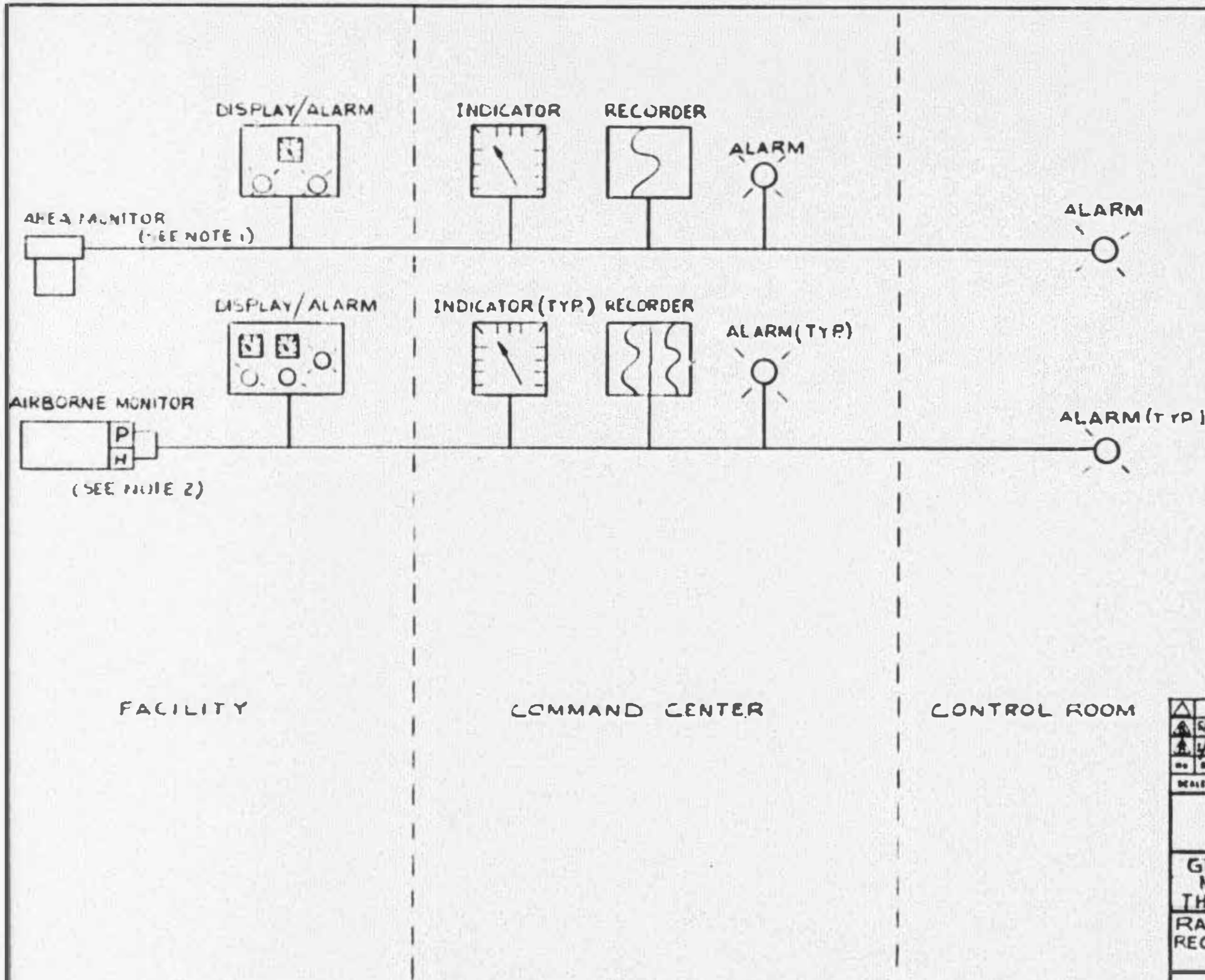
1. FOR AREA MONITORS WHICH ARE REPLACEMENTS FOR EXISTING MONITORS, THE RECORDER AND AN INDICATOR ARE PROVIDED IN THE CONTROL ROOM.

BECHTEL
GAITHERSBURG, MD.

GENERAL PUBLIC UTILITIES CORP
METROPOLITAN EDISON
THREE MILE ISLAND — UNIT 2

RADIATION MONITORING SYSTEM
CONTAINMENT MONITORING
FIGURE 6.2-1

13587 | SK-TMI/CRE-4 | B



LEGEND

P – PARTICULATE
H – HALOGEN

NOTES:

1. PROVIDED IN THE CONTAINMENT RECOVERY SERVICE BUILDING, COMMAND CENTER (NO LOCAL DISPLAY/ALARM), RADWASTE STAGING AREA AND PERSONNEL ACCESS FACILITY.
2. PROVIDED IN THE RADWASTE STAGING AREA

NO	DATE	REVISIONS	BY	CHK'D		ENG'G		APPR
B		ISSUED FOR REPORT		NJ	—			—
A		ISSUED/REVIEW		—		—		—

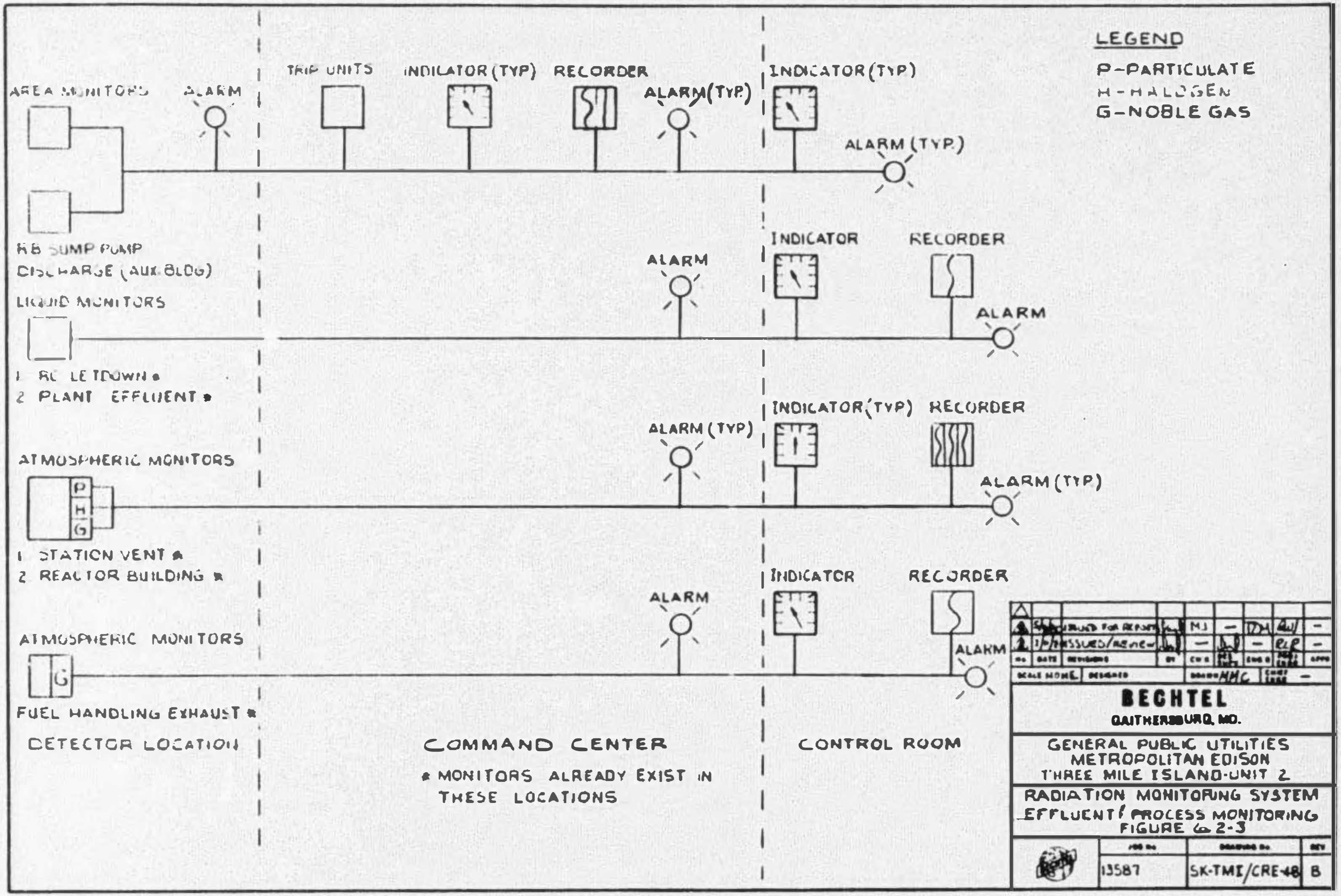
SCALE NONE | DESIGNED | DRAWN MMC | CHKD —

BECHTEL
GAITHERSBURG, MD.

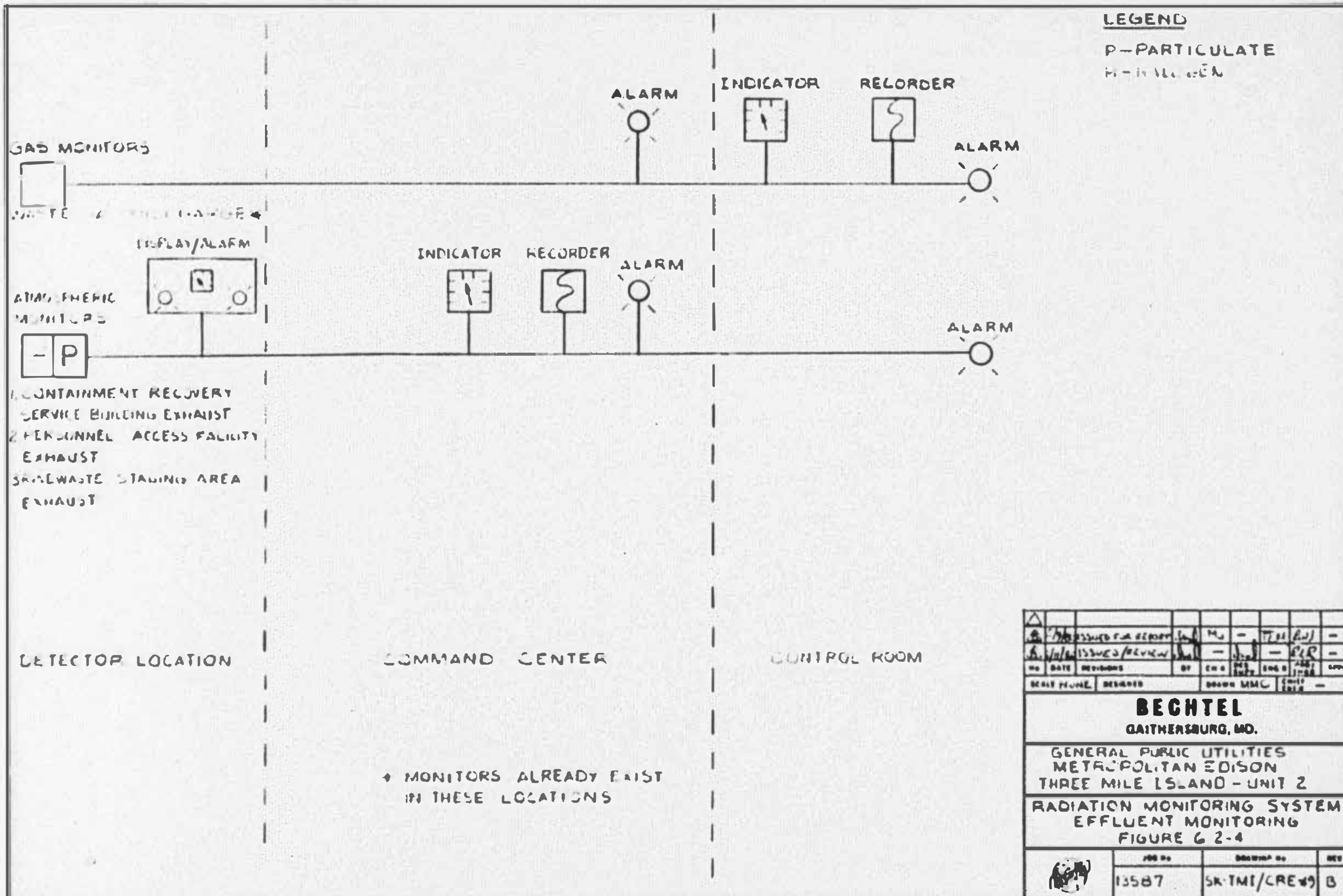
GENERAL PUBLIC UTILITIES CORP.
METROPOLITAN EDISON
THREE MILE ISLAND-UNIT 2

RADIATION MONITORING SYSTEM
RECOVERY FACILITY AREA MONITORING
FIGURE 6.2-2

JOB No.	DRAWING No.	REV
13587	SK-TMI/CRE-47	B



LEGEND
P-PARTICULATE
H-HALOGEN
G-NOBLE GAS
AREA MONITORS
ALARM
TRIP UNITS
INDICATOR (TYP)
RECORDER
ALARM (TYP.)
INDICATOR (TYP)
ALARM (TYP.)
RB SUMP PUMP
DISCHARGE (AUX BLDG)
INDICATOR
RECORDER
ALARM
ALARM
LIQUID MONITORS
1. RC LETDOWN *
2. PLANT EFFLUENT *
INDICATOR (TYP)
RECORDER
ALARM (TYP)
ATMOSPHERIC MONITORS
P
H
G
ALARM (TYP)
1. STATION VENT *
2. REACTOR BUILDING *
INDICATOR
RECORDER
ALARM
ATMOSPHERIC MONITORS
G
ALARM
FUEL HANDLING EXHAUST *
DETECTOR LOCATION
COMMAND CENTER
* MONITORS ALREADY EXIST IN THESE LOCATIONS
CONTROL ROOM
ISSUED FOR REPORT
ISSUED/REVIEW
SCALE NONE
BECHTEL
GAITHERSBURG, MD.
GENERAL PUBLIC UTILITIES
METROPOLITAN EDISON
THREE MILE ISLAND-UNIT 2
RADIATION MONITORING SYSTEM
EFFLUENT / PROCESS MONITORING
FIGURE 6.2-3
13587
SK-TMI/CRE-48
B



LEGEND
P - PARTICULATE
ALARM
INDICATOR
RECORDER
ALARM
GAS MONITORS
INDICATOR
RECORDER
ALARM
ALARM
P
CONTAINMENT RECOVERY
SERVICE BUILDING EXHAUST
PERSONNEL ACCESS FACILITY
EXHAUST
EXHAUST
DETECTOR LOCATION
COMMAND CENTER
CONTROL ROOM
* MONITORS ALREADY EXIST
IN THESE LOCATIONS
ISSUED FOR REPORT
ISSUED/REVIEW
DRAWN MMC
BECHTEL
GAITHERSBURG, MD.
GENERAL PUBLIC UTILITIES
METROPOLITAN EDISON
THREE MILE ISLAND - UNIT 2
RADIATION MONITORING SYSTEM
EFFLUENT MONITORING
FIGURE 6 2-4
13587
SK-TMI/CRE-9
B

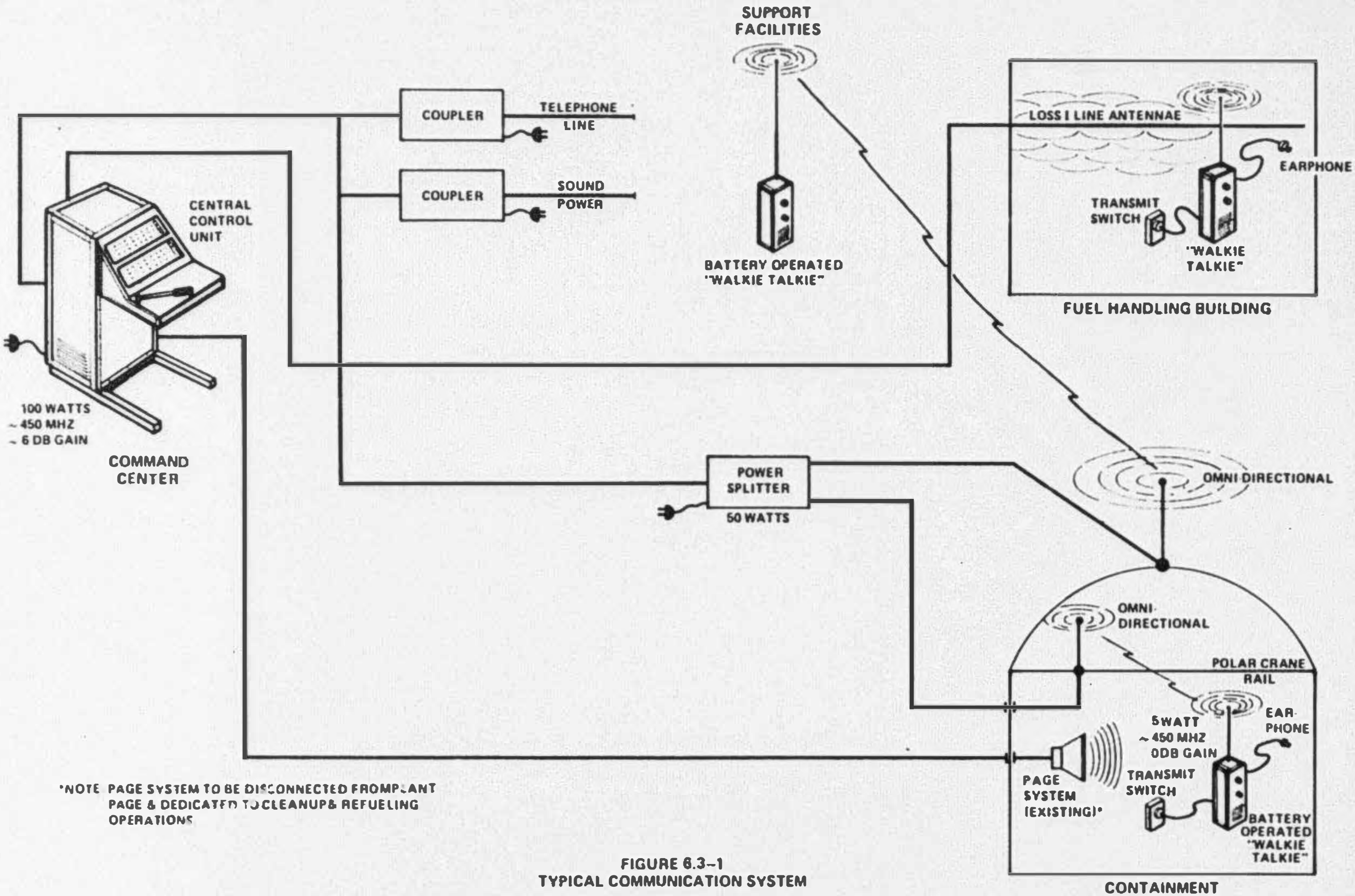


FIGURE 6.3-1
TYPICAL COMMUNICATION SYSTEM

LIST OF REFERENCES

1. NRC TMI-2 Special Inquiry Group Report, Vol. II, Part 2; January 1980.

2. NSAC-1, Analysis of the Three Mile Island - Unit 2 Accident, NSAC/EPRI, July 1979

3. NPGD-TM 534, TMI-2 Criticality Evaluation, Babcock and Wilcox, December 1979.

APPENDIX A

General Plant Description

Three Mile Island Unit 2 utilized a nuclear steam supply system incorporating a pressurized water reactor and a two-loop reactor coolant system.

The reactor core consists of 177 fuel assemblies of 208 fuel rods each. The rods are made of slightly enriched uranium dioxide pellets enclosed in Zircaloy tubes pressurized with helium.

Water serves as both moderator and coolant, and is circulated through the reactor vessel and core by four vertical reactor coolant pumps, two located in each loop. The coolant water heated by the reactor is circulated through two vertical once-through steam generators, one in each loop, in which heat is transferred to the secondary system to produce slightly superheated steam, and then returned to the pumps to repeat the cycle.

The steam produced in the steam generators is supplied via two main steam lines from each steam generator to a tandem compound two-stage reheat four-flow turbine. The exhaust steam is condensed in a dual-pressure surface condenser to which cooling water is supplied in a closed circulating water system cooled by two natural-draft cooling towers.

The reactor is controlled by a combination of the regulation of the concentration of soluble neutron absorber (boric acid) and the movement of mechanical control rods made of an alloy of silver-indium-cadmium enclosed in stainless steel tubing. A reactor protection system is provided which automatically initiates appropriate corrective action whenever a safety-related condition monitored by the system approaches pre-established limits. This system and an engineered safety features actuation system will act to shut down the reactor, close isolation valves, and initiate operation of the engineered safety features should any or all of these actions be required.

Engineered safety features for this plant include an emergency core cooling system which consists of a core flooding system and both high and low pressure injection systems with provisions for recirculating the borated water after the injection phase. Combinations of these systems will assure core cooling for the complete range of postulated reactor coolant pipe breaks. Other engineered safety features include the containment, containment isolation valves, reactor building spray system, reactor building air cooling system, and combustible gas control systems.

The nuclear steam supply system is housed in a carbon-steel-lined prestressed reinforced concrete containment identified as the reactor building. The auxiliary and fuel-handling building contains most of the engineered safety features components, radioactive waste management systems, new and spent fuel storage facilities, and various related auxiliary systems.

Reactor Coolant System

The components that comprise the reactor coolant system are the reactor vessel, two vertical once-through steam generators, four shaft-sealed reactor coolant pumps, an electrically heated pressurizer and interconnecting piping. The system is arranged in two heat transport loops, each of which has two reactor coolant pumps and one steam generator. The reactor coolant is transported through piping connecting the reactor vessel to the steam generators and flows downward through the steam generator tubes, transferring heat to the steam and water on the shell side of the steam generator. In each loop, the coolant is returned to the reactor vessel through two lines, each containing a coolant pump.

Figures A-1 and A-2 are elevation and plan views of the system arrangement. All inlet and outlet nozzles are located at the same elevation and are placed 60 degrees apart on reactor vessel. The outlet piping leaving the reactor vessel is 36 inches ID and runs in a horizontal plane for a few feet, undergoes a 90-degree bend, runs upward for approximately 25 feet, enters a 180-degree bend, and enters the top plenum of the steam generator. At the steam generator outlet the loop branches into two 28-inch pipes. Each 28-inch pipe leaves the steam generator through a bend, runs upward for approximately 20 feet, and enters the suction nozzle of a reactor coolant pump. The reactor coolant pumps are located at an elevation approximately 2 feet above the horizontal plane of the reactor coolant inlet nozzle. Upon leaving the pump discharge nozzle the pipe drops down through two 45-degree bends into a horizontal run and enters the reactor vessel. The pressurizer is connected to the reactor coolant piping by a 10-inch surge line and a 2-1/2-inch spray line. The surge line is connected to the 36-inch piping as shown in Figure A-2, and the spray line is connected to the 28-inch piping on the discharge side of the reactor coolant pump as shown in Figure A-1.

The general design considerations for the entire reactor coolant system are as shown in Table A-1. All materials in contact with reactor coolant fluid are stainless steel or Inconel. All pressure-containing components of the system are designed, fabricated, inspected, and tested to applicable codes as listed in Table A-2.

The major piping in the coolant system are the 28-inch-ID cold leg piping and the 36-inch-ID hot leg piping from the reactor vessel to the steam generators. Also included is the 10-inch pressurizer surge line and the 2-1/2-inch pressurizer spray line. The system piping also includes those auxiliary system connections that are necessary for operation. In addition to drains, vents, pressure taps, temperature elements, and injection connections, there is a flow meter in each 36-inch line to the steam generators to provide a means for determining the flow through each loop.

The 28-inch and 36-inch piping are carbon steel clad with austenitic stainless steel. Short sections of 28-inch stainless steel transition piping are provided between the pump casings and the

28-inch carbon steel lines. Stainless steel or Inconel safe-ends are provided for field welding the nozzle connections to smaller piping. The piping safe-ends are designed so that there is no furnace-sensitized stainless steel in the pressure boundary material. This is accomplished by either installing stainless steel safe-ends after stress relief or using Inconel. Smaller piping, including the pressurizer surge and spray lines, is austenitic stainless steel. All piping connections in the reactor coolant system are butt-welded except for the flanged connections on the pressurizer for relief valves.

Thermal sleeves are installed where required to limit the thermal stresses developed because of rapid changes in fluid temperatures. They are provided in the four high-pressure injection nozzles on the reactor inlet pipes, the two core flooding nozzles on the reactor vessel, and the surge line and spray line nozzles on the pressurizer. The reactor coolant piping is designed according to American National Standards Institute B31.7 and to associated nuclear code cases.

Reactor Vessel Description

The reactor vessel (RV) is 14'-1 5/8" in diameter by 31'-7" high by 8" thick and consists of a cylindrical shell, a spherically dished bottom head, and a ring flange to which a removable reactor closure head is bolted. A cylinder attached to the vessel's shell supports the vessel and extends downward to a flanged base ring which is bolted to the foundation of the building. The reactor vessel head is 14'-2 5/8" in diameter by 5'-8 13/16" high by 6 5/8" thick and is spherically dished and welded to a ring flange which mates with and is bolted to the vessel with sixty (60) 6 1/2" diameter studs. The studs are threaded into tapped holes in the reactor vessel flange and penetrate through holes in the RV head flange. The clamping load between the RV and the RV head is maintained with spherical washers under castellated nuts threaded on the studs. All internal surfaces of the vessel are clad with stainless steel weld deposit.

Two concentric metallic O-rings provide the pressure integrity seal between the closure head and the vessel's flanges. The reactor vessel's ring flange includes an internal ledge to support the core and the internal structural components.

The vessel has two outlet nozzles through which the reactor coolant is transported to the steam generators, and four inlet nozzles through which reactor coolant re-enters the reactor vessel. Two smaller nozzles between the reactor coolant nozzles serve as inlets for decay heat cooling and emergency cooling water injection (core flooding and low pressure injection engineered safety function). The reactor coolant and the control rod drive penetrations are located above the top of the core to maintain a flooded core in the event of a rupture in a reactor coolant pipe or a control rod drive pressure housing.

The bottom head of the vessel is penetrated by fifty-two (52) small incore instrumentation nozzles. The closure head is penetrated by sixty-nine (69) flanged nozzles which provide for attaching the control rod drive mechanisms and for control rod extension shaft movement, and eigbt small flanged nozzles for vent valve thermocouple installation. The thermocouple nozzles are not used and therefore are blind flanged.

Core guide lugs welded to the reactor vessel inside wall limit a vertical drop of the reactor internals and core to 1/2 inch or less and prevent rotation about the vertical axis in the unlikely event that a major internals component fails. These lugs provide shock support for the internals and control the motion of the lower end of the core support assembly while under the influence of horizontal seismic loads.

Surveillance specimens made from reactor vessel steel are placed between the reactor vessel's wall and the thermal shield. These specimens are examined periodically to evaluate nil-ductility transition temperature changes in the material.

Metal reflective insulation is used on the exterior of the vessel from the closure flange down to the support cylinder, on the exterior of the bottom head dome, and on all insulated surfaces of the support skirt. Removable metal reflective insulation panels enclose the top bead closure flange and studs. These panels must be removed for stud detensioning. Metal reflective insulation is used on the closure head dome.

The incore instrument penetrations are field-weld-joined to pipes that contain two 90-degree bends and terminate in bolted sealing flanges in a shielded area at a higher elevation in the reactor building. These tubes contain incore detector assemblies for measuring the neutron flux and temperatures in the reactor core. The incore detector assemblies are subject to the reactor coolant pressure.

The reactor coolant system is protected against over-pressure by the high-pressure trip (set at 2350 psig) and ASA code relief valves (set at 2500 psig) on the pressurizer's top head.

Reactor Vessel Material

The reactor vessel is fabricated from low-alloy manganese-molybdenum steel (plate specification SA-533B Class I and forging specification A-508), which is readily fabricated in both plate and forging forms. This material is heat-treated specifically to obtain good notch ductility to ensure a low ductility transition temperature (NDTT) and thereby give assurance that the finished vessel can be initally hydrostatically tested and operate at near room temperature without restrictions. The studs, nuts, and washers are fabricated from A540 Grade B23, which has a minimum yield point of 130,000 psi. The studs, when tightened for operating conditions, will have a tensile stress of approximately 30,000 psi.

Reactor Internals

The reactor internals support the core, maintain fuel assembly alignment, limit fuel assembly movement, and maintain CRA guide tube alignment between fuel assemblies and control rod drives. They also direct the flow of reactor coolant, provide gamma and neutron shielding, provide guides for incore instrumentation between the reactor vessel's lower head and the fuel assemblies, support the surveillance specimen assemblies in the annulus between the thermal shield and the reactor vessel wall, and support the internals vent valves.

The internal components include the plenum assembly and the core support assembly. Figure 3.1-1 shows the reactor vessel, reactor vessel internals arrangement, and the reactor coolant flow path. Figure A-3 is a cross section through the reactor vessel. All the internal components can be removed from the reactor vessel to allow inspection of the reactor internals and the reactor vessel internal surface.

The reactor internals are fabricated from SA-240 (type 304) material and designed to within the allowable stress levels permitted by the ASME Code, Section III, for normal reactor operation and transients. The structural integrity of all core support assembly circumferential welds is ensured by compliance with the ASME Code, Sections III and IX, radiographic inspection acceptance standards, and welding qualification.

Material for the reactor internals bolting is subjected to rigid quality control requirements to ensure the structural integrity. All core support circumferential weld joints in the internals shells are inspected according to the requirements of the ASME Code, Section III.

Plenum Assembly

The plenum assembly is located directly above the reactor core and is removed as a single component before refueling. It consists of a plenum cover, upper grid, CRA guide tube assemblies, and a flanged plenum cylinder with openings for reactor coolant outlet flow. The plenum cover is a series of parallel flat plates intersecting to form square lattices with a perforated top plate and flange; it is attached to the plenum cylinder's top flange. The perforated top plate is 1/2-inch thick and has sixty-nine (69) 8.520-inch-diameter holes. The plenum cover is attached to the top flange of the plenum cylinder by a flange. The three T-shaped lifting lugs are provided for handling the plenum assembly. These lugs, 17-1/4 inches high, 3 inches thick, and 3 inches (min) wide, are bolted to the lattice systems of the cover. The CRA guide tubes are welded to the plenum cover top plate and bolted to the upper grid. CRA guide assemblies provide guidance and protect the CRA from the effects of coolant crossflow and provide structural attachment of the grid assembly to the plenum cover.

The plenum cylinder is a 130-inch-OD, 1-1/2-inch-thick cylinder with flanges on both ends to connect the cylinder to the plenum cover and upper grid. Six 34-inch-diameter holes and four 22-inch-diameter holes in the plenum cylinder provide a flow path for the coolant water.

Locating keyways in the plenum assembly cover flange engage the reactor vessel's top flange locating keys to align the plenum assembly with the reactor vessel, reactor closure head control rod drive penetrations, and the core support assembly. The bottom of the plenum assembly is guided by the inside surface of the lower flange of the core support shield.

The plenum assembly upper grid, attached to the bottom of the plenum assembly, is a rolled plate with machined holes that locate the lower end of the individual CRA guide tube assembly relative to the upper end of a corresponding fuel assembly.

Alignment between fuel assemblies is provided by pads bolted to the lower surface of the upper grid.

Core Support Assembly

The core support assembly consists of the core support shield, core barrel, lower grid assembly, flow distributor, thermal shield, incore instrument guide tubes, surveillance specimen holder tubes, and internals vent valves. Static loads from the assembled components and fuel assemblies and dynamic loads from CRA trip, hydraulic flow, thermal expansion, seismic disturbances, and loss-of-coolant accident considerations are all carried by the core support assembly. The assembly's components are described below.

The core support shield is a large, flanged cylinder which mates with the reactor vessel opening. The forged top flange (146 inches ID) rests on a circumferential ledge in the reactor vessel top closure flange. The core support shield lower flange (136 inches ID) is bolted to the core barrel. The inside surface of the lower flange guides and aligns the plenum assembly relative to the core support shield. The cylinder wall has two nozzle openings for coolant flow. These openings are formed by two forged rings, which seal to the reactor vessel outlet nozzles by the differential thermal expansion between the stainless steel core support shield and the carbon steel reactor vessel. The nozzle seal surfaces are finished and fitted to a predetermined cold gap providing clearance during core support assembly installation and removal. At reactor operating temperature, the mating metal surfaces are in contact to make a seal without exceeding allowable stresses in either the reactor vessel or the internals. The cylinder wall also has eight holes in which vent valve mounting rings are welded.

The core barrel supports the fuel assemblies, lower grid, flow distributor, and incore instrument guide tubes. The barrel consists of a flanged cylinder, a series of internal horizontal spacers bolted to the cylinder, and a series of vertical plates bolted to the inner surfaces of the horizontal spacers to form an inner wall enclosing the fuel assemblies. The upper flange of the cylinder is bolted to the mating lower flange of the core support shield assembly, and the lower flange is bolted to the mating flange of the lower grid assembly. All bolts are lock-welded after final assembly.

The lower grid assembly provides alignment and support for the fuel assemblies, supports the thermal shield and flow distributor, and aligns the incore instrument guide tubes with the fuel assembly instrument tubes. The lower grid consists of two lattice structures separated by short tubular columns and surrounded by a forged flanged cylinder. The top flange of the forged cylinder is bolted to the lower flange of the core barrel. A perforated flat plate, midway between the two lattice structures, aids in distributing coolant flow through the core. Alignment between fuel assemblies is provided by pads bolted to the top surface of the upper lattice structure.

The flow distributor is a perforated, dished head with an external flange which is bolted to the bottom flange of the lower grid. The distributor supports the incore instrument guide tubes and distributes the reactor coolant entering the bottom of the core. The thermal shield is a stainless steel cylinder 152 inches in OD, installed in the annulus between the core barrel cylinder and the reactor vessel's inner wall. The shield reduces the neutron and gamma internal heat generation in the reactor vessel wall and thereby reduces the resultant thermal stresses.

Holder tubes are installed on the core support assembly's outer wall to contain the surveillance specimen assemblies. The tubes extend from the top flange of the core support shield to the lower end of the thermal shield.

<u>RV Head and Internals Lifting Equipment</u>

The equipment used to lift the reactor vessel head and reactor internals is normally stored in containment.

A lifting tripod is attached to the polar crane hook and positions each of three pendants relative to each other for the three point lifts of the RV head, plenum assembly, and core support structure. See Fig. 3.6-2.

Pendant lifting cables are pinned to the tripod and lifting lugs on the RV head for the RV head lift. Two turnbuckle assemblies are installed between two pendants and the tripod for leveling adjustment.

Three internals handling adaptors are pinned to the tripod for plenum assembly and core support assembly lifts. These adaptors include a turnbuckle in each for leveling adjustment. The lower ends of the adaptors have a latch box attached which can be locally or remotely operated to latch and unlatch the adaptors to the T-shaped lifting lugs that are a part of the plenum assembly and the core support assembly. A spreader ring is used to position the lower ends of the handling adaptors relative to each other.

Fuel Assemblies

The fuel for the reactor is in the form of sintered pellets of low-enriched uranium dioxide clad in Zircaloy-4 tubing. The ends of the tubing are sealed by Zircaloy-4 end caps attached to the tubing by welding. The cladding, fuel pellets, end caps, and fuel support components form a "fuel rod." The basic fuel assembly is made up of 208 fuel rods, 16 control rod guide tubes, 1 instrumentation tube assembly, 7 spacer sleeves, 8 spacer grids, and 2 end fittings. The guide tubes, spacer grids, and end fittings form a structural cage which contains the rods and tubes in a 15 X 15 array. The center position in the assembly is reserved for instrumentation, and the remaining 16 positions are provided for the guide tubes which guide the control rods. Figure A-4 shows the construction features of a fuel assembly. Fuel assembly components, materials, and dimensions are listed below.

Item	Material	Dimensions, in.
Fuel Rod		
Fuel	UO_2 sintered pellets	0.370 dia.
Fuel cladding	Zircaloy-4	0.430 OD X 0.377 ID X 153-1/8 long
Fuel rod pitch		0.568
Active fuel length		144
Ceramic spacer	ZrO_2	0.368 dia.
Corrugated tubing	SS, type 321	10 and 14
Fuel Assembly		
Fuel assy pitch		8.587
Overall length		165-3/4
Control rod guide tube	Zircaloy-4	0.530 OD X 0.016 ID
Instrumentation tube	Zircaloy-4	0.493 OD X 0.441 ID
End fittings	SS, grade CF3M	
Spacer grid	Inconel-718 strips	0.020 thick exteriors 0.016 thick interiors
Spacer sleeve	Zircaloy-4	0.554 OD X 0.502 ID

The fuel assemblies are designed to operate satisfactorily to design burnup and to retain adequate integrity at the end-of-life to permit their safe removal from the core. The cold-worked Zircaloy-4 cladding is designed to be free-standing, to contain fission products, and to maintain dimensional integrity. Fuel rods are held in place by spacer grids that maintain dimensional control of the fuel rod spacing throughout the design lifetime, without impairing cladding integrity.

The fuel assembly upper end fitting and the control rod guide assembly in the internals structure are indexed to the grid plate above the fuel assemblies to ensure continuous alignment of the guide channels for the CRA and APSRA. The rod travel is designed to keep the rods engaged in the fuel assembly guide tubes so that the CRAs and APSRAs can always be inserted into the core.

Control Rod Assembly

Each control rod assembly has 16 control rods, a stainless steel spider, and a female coupling. Figure A-5 shows the control rod assembly configurations. The 16 control rods are attached to one side of the spider, and the female coupling is attached to the other. The control rod drive connects to the CRA at the female coupling by a male-female connection. On the side of the spider opposite the control rod drive connection, the 16 control rods are tightly connected. When the CRA is inserted into the fuel assembly's upper end fittings, the rods are guided by the guide tubes of the fuel assembly. Full-length guidance for the CRA is also provided in the upper plenum assembly by the control rod guide assembly. When the reactor closure head is installed and the control rod drives are attached, the CRA cannot be withdrawn far enough to cause disengagement of the control rods from the fuel assembly guide tubes. The CRAs are designed to permit maximum conformity with the fuel assembly guide tube throughout the stroke.

Each control rod contains a section of neutron absorber material - an alloy of silver-indium-cadmium clad in cold-worked, type 304 stainless steel tubing and type 304 stainless steel (upper and lower end pieces). The end pieces are welded to the tubing to form a water- and pressure-tight container for the absorber material. The stainless steel tubing provides the structural strength of the control rods, prevents corrosion of the absorber material, and eliminates possible silver contamination of the reactor coolant. Above the absorber material is a spacer of corrugated tubing which keeps the absorber material in place during shipping and handling. Pertinent data on the CRAs are as follows:

Item	Data
Number of CRAs	61
Number of control rods per assy	16
OD of control rod, in.	0.440
Cladding thickness, in.	0.021
Cladding material	type 304 SS, cold-worked
End plug material	type 304 SS, annealed
Spider material	SS, grade CF3M
Poison material	80% Ag, 15% In, 5% Cd
Female coupling material	type 304 SS, annealed
Corrugated tubing	type 321 SS
Length of poison section, in.	134
Stroke of control rod, in.	139

Control Rod Drive Mechanism

The control rod drive mechanism is used to raise, lower, and maintain control rod position and axial power shaping rod position in response to the control rod drive motor control systems. The control system provides a sequentially programmed dc input to the drive motor. The motor has seven input leads, one for each phase of the six-phase star-connected winding and one for the neutral return line. The stator coils are sequentially energized in a repetitive 2-3-2-3 manner to produce a rotating magnetic field around the rotor assembly. As the stator coils are progressively energized, the rotor rotates (steps) to orient itself to a new position. This rotary motion is translated into linear motion of the leadscrew and the attached control rod.

Each control rod drive mechanism is associated with a control rod (CR) or axial power shaping rod (APSR) assembly. The CR or APSR assembly spider is attached to the leadscrew extension shaft by means of a breachlock-type connector. This breechlock connector must be uncoupled to allow removal of the reactor vessel head.

The leadscrew is the connecting link between the rotor assembly and the control rod. When the rotor assembly rotates, the leadscrew is kept from rotating by keying it to the torque tube through the torque taker. The leadscrew travels along the vertical centerline of the drive. In its travel the leadscrew passes through the torque tube, the rotor assembly, and the thermal barrier. The leadscrew is uncoupled from the control rod assembly by rotating it (the leadscrew) within the motor tube. This requires removal of the CRDM closure insert assembly and insertion of the uncoupling tool.

The motor tube forms the primary coolant pressure boundary when bolted to the reactor head. It is a four-section weldment housing the rotor assembly, the thermal barrier, the leadscrew, and the torque tube assembly. The center (motor) section is machined from Inconel-clad, low-alloy steel with the magnetic properties required for stator/rotor coupling. The upper end of the motor tube is threaded to accommodate the top closure components. The motor tube is bolted to its mating flange on the RV closure head with eight (8) bolts 1-1/8" diameter and a split bolting ring.

The rotor assembly, when actuated through a magnetic coupling by the stator, engages, holds, and positions the leadscrew. The major components of the rotor assembly are the rotor tube, the segment arms, the roller nuts, and the bearings.

The thermal barrier restricts the circulation of hot primary fluid and acts as an insulator between the reactor vessel head and the drive. A clearance hole is drilled through the center of the housing to allow unrestricted leadscrew travel. Four ball-check valves are installed at the base of the thermal barrier to permit in-flow to the mechanism to replace the volume left by the exiting leadscrew during a reactor trip.

A stator assembly is provided external to the CRDM pressure boundary. The stator coils are sequentially energized to provide a rotating magnetic field thus rotating the CRDM rotor. The rotary motion is translated into axial leadscrew motion.

Cooling water inlet and outlet connections are provided to each stator assembly near the top of the CRDM. These are connected via small bore tubing, to a cooling water header arrangement which is an integral part of the service structure. Cooling water supply to and return from the service structure is provided by lines adjacent to the service structure. Flanged bolted connections between these lines and the service structure must be unbolted for RV head removal.

Electrical power and control cables are provided from bulkhead connections just above the fuel transfer canal adjacent to the service structure to the individual CRDM stators position indicators and thermocouples. These cables must be disconnected to allow RV head removal.

A cylindrical service structure 126" diameter and 17' high is bolted at the lower end to its support skirt on the RV closure head. All the CRDMs are inside the service structure. The upper portion of the service structure supports the cooling water heaters and electrical cabling, provides vertical alignment of the CRDMs, and service as a work platform for work in the CRDMs.

Reactor Coolant Pumps and Motors

The reactor coolant pumps supply a medium for removing heat from the reactor core. They circulate reactor coolant through the reactor vessel to the steam generators for heat removal, and return it to the reactor vessel. There are four reactor coolant pumps for this purpose - two in parallel between each steam generator and the reactor vessel. The pumps are vertical suction, horizontal discharge, single-stage centrifugal units sealed with a combination of three face-type mechanical seals and driven by constant-speed, vertical, squirrel-cage induction motors. The pumps are rated at a nominal 92,200 gpm with a discharge head of 362 feet of water at reactor coolant operating temperature and pressure, and the motors have a rating of 9,000 hp. The combined pump-motor moment of inertia of over 70,000 lb-ft^2 is designed to ensure an adequate flow coastdown time for core protection against departure from nucleate boiling (DNB) following a pump trip.

Figure A-6 shows the reactor coolant pump and motor. The pump is approximately 15 feet high and the motor is over 17 feet high, giving the installed pump and motor an overall height of approximately 32 feet.

The Allis-Chalmers pump motors are designed to develop 9,000 hp at 1,200 rpm from a 6,600-volt, 3-phase power supply. When pumping cold reactor coolant, the pumps have a full load current of approximately 715 amperes and a brake horsepower of over 9,500. Locked-rotor starting current is approximately 3,600 amperes. At normal hot operating conditions, the motor develops about 7,400 hp with a current of about 550 amperes. The motors are totally enclosed and air cooled. They are capable of withstanding reactor building design pressure at normal room temperature for two days. This allows for building pressure and leak testing. Hot air is discharged from the air housing into the air cooler housing, where it is cooled by two air-to-water heat exchangers and directed back into the air housing surrounding the motor. Cooling water requirements for the heat exchangers are 176 gpm at 105 F.

The reactor coolant pumps are Bingham units, type RQV 28 x 28 x 41 (radial quadrant volute, 28-inch-diameter suction nozzle; 28-inch-diameter discharge nozzle; 41-inch-diameter impeller) designed for a system pressure of 2,500 psig at 650 F.

As shown in Figure A-6, the pump requires both cooling water and seal injection water. The seal injection water establishes the reactor coolant system boundary at the pump restriction bushing (at the top of the suction bowl), preventing the release of radioactive contaminants from the reactor coolant system through the pumps. Seal injection water also provides the primary means for heat removal from the pumps. Cooling water is supplied to the pump seal water heat exchangers and cooling jacket to prevent the pump as a whole from attaining the reactor coolant system operating temperature and damaging portions of the pump. The mechanical seals at the top of the pumps may be damaged if the differential temperature across them exceeds 175 F; the O-rings in the stuffing box area of the pump may be damaged if the pump temperature exceeds 400 F; and critical tolerances in the pumps may be closed up if the pump shaft is allowed to heat up to more than 200 F above its normal operating temperature.

The pump is capable of operating indefinitely after losing either the seal injection water supply or the cooling water supply, but it must not be operated or allowed to stand idle in a hot loop without at least one of the coolant supplies in service.

Steam Generators

As shown in Figure A-7, the OTSG is a straight-tube, straight-shell heat exchanger. The primary fluid or reactor coolant is on the tube side, and the secondary fluid is on the shell side. The primary fluid from the reactor outlet enters the 36-inch inlet nozzle at a temperature of about 603 F. The primary fluid gives up heat to the secondary fluid as it flows through the tubes and leaves through the two 28-inch-ID outlet nozzles at a temperature of 555 F.

The tubes are supported by tube support plates which have broached openings to permit flow between the plate and the tube. The support plates are fixed longitudinally by a system of support rods, which are welded to the lower tubesheet. This system permits positive placement of each support without restricting the thermal movement of the supports within the cylindrical baffle. The cylindrical baffle comprises two pieces; the lower section is bolted to the bottom tubesheet, and the upper section is welded to the shell just below the steam outlet nozzles. Alignment pins hold both sections radially in the shell.

As shown in Figure A-8, feedwater enters the boiler through 32 spray nozzles connected to the 14-inch-OD main feedwater header. The condensing action of the cold feedwater (455 F at full load) draws steam through the circumferential space between the upper and lower cylindrical baffles. This steam heats the feedwater rapidly to the saturation temperature of about 535 F, which prevents thermal shock of the shell. The flow of bleed steam is inherently self-regulating; any change in feedwater flow changes the rate of condensation, thus changing the rate of bleed steam flow.

A mixture of saturated steam and water forms in the downcomer. The level and density of the downcomer fluid are set by the static head and pressure drop between the bottom of the tube nest and the bleed point. An adjustable orifice in the lower section of the downcomer is used to adjust the dynamic characteristics of the steam generator. The fluid enters the tube nest through the ports in the lower portion of the cylindrical baffle, boils in the tube bundle, and the superheated steam exits through the two 24-inch-ID steam outlet nozzles.

An internal auxiliary feedwater header at the top of the steam generator distributes small quantities of feedwater within the tube bundle. The auxiliary feed header can be used to fill the steam generator in the unlikely event that the steam generator is hot with the secondary side empty. This method of filling prevents thermal shock of the vessel wall and tubesheet. The auxiliary feedwater header is also used to inject feedwater into the steam generator upon loss of all four reactor coolant pumps. This ensures natural circulation of the reactor coolant after coastdown of the reactor circulating pumps.

The materials selected for use in the OTSG vary in properties and methods of fabrication. The types and locations of the materials in the primary components of the OTSG are shown in Figure A-9. The heads, the tubesheets, and the support skirt are low-alloy steel (1% Mn, 0.5% Mo), while the remainder of the shell and nozzle material is carbon steel.

Pressurizer

The pressurizer (and associated components, controls and instrumentation) establishes and maintains the reactor coolant system pressure within prescribed limits and provides a surge chamber and a water reserve to accommodate reactor coolant density (volume) changes during operation. Relief valves connected to the pressurizer protect all reactor coolant system components from exceeding the design pressure.

The pressurizer and pressure control equipment includes the pressurizer vessel, pressure relief valves, electric immersion heaters, and pressure sensing and control devices. The general arrangement of the vessel is shown in Figure A-10, and the design and operating data are tabulated in Table A-3. The vessel is a vertically mounted cylinder with hemispherical ends; it is connected to the reactor outlet piping by a surge line which terminates in the lower end of the vessel. Penetrations (nozzles) are provided for electric heaters, level and temperature measurements, spray line, vents, sample line, and relief valves. A manway is installed for internal inspection and spray nozzle maintenance.

The pressurizer heaters are replaceable immersion heating elements installed in the lower portion of the pressurizer. The elements are divided into three bundles for installation purposes and are further divided into five banks of varying numbers of elements for control purposes. Heater data are presented in Table A-4. Heater banks 1 and 2 are sized to provide adequate heating capacity to compensate for normal heat losses and spray valve bypass flow during steady-state operation. Heater banks 3, 4, and 5 provide additional heating capacity during reactor startup and during load changes.

The pressurizer spray line originates at the reactor inlet piping and terminates in the top of the pressurizer with a spray nozzle. The remote auto/manual motor-operated spray valve is actuated (opened) when the reactor coolant system pressure exceeds normal limits. Since the spray valve is normally closed during operation, a bypass valve is provided to maintain the spray line and components at the same temperature as the reactor inlet and to minimize boric acid concentration difference between the pressurizer water and the reactor coolant.

Fuel Storage and Handling

New Fuel Storage - The new fuel storage pit, which is shared with Unit 1, provides dry storage for approximately one full core (180 fuel assemblies). The storage pit and racks are designed to seismic Category I requirements. The racks have a spacing which is sufficient to maintain a $K_{effective}$ of 0.90 or less even in the event the storage area were flooded with unborated water.

Spent Fuel Storage - Spent fuel will be stored under water in the spent fuel storage pool. The spent fuel storage racks are designed to prevent fuel assemblies from being placed in other than their prescribed locations. The spent fuel storage racks and the spent fuel pool can accommodate a total of approximately two and one-half cores (448 fuel assemblies), representing normal maximum storage of one and one-third cores, plus postulated unloading of a full core from the reactor.

The fuel pool is of reinforced concrete construction with a stainless steel liner, and is designed to seismic Category I requirements. The spent fuel storage racks are designed to withstand the maximum uplift forces of the spent fuel pool bridge hoist. The racks have a center-to-center spacing which is sufficient to maintain a $K_{effective}$ of 0.90 or less, assuming the pool water is unborated. The facility is designed to prevent the cask handling crane from traveling over, or in the vicinity of, the spent fuel storage areas, thereby precluding damage to the stored fuel in the event of a dropped cask.

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Fuel Storage and Handling

New Fuel Storage - The new fuel storage pit, which is shared with Unit 1, provides dry storage for approximately one full core (180 fuel assemblies). The storage pit and racks are designed to seismic Category I requirements. The racks have a spacing which is sufficient to maintain a $K_{effective}$ of 0.90 or less even in the event the storage area were flooded with unborated water.

Spent Fuel Storage - Spent fuel will be stored under water in the spent fuel storage pool. The spent fuel storage racks are designed to prevent fuel assemblies from being placed in other than their prescribed locations. The spent fuel storage racks and the spent fuel pool can accommodate a total of approximately two and one-half cores (448 fuel assemblies), representing normal maximum storage of one and one-third cores, plus postulated unloading of a full core from the reactor.

The fuel pool is of reinforced concrete construction with a stainless steel liner, and is designed to seismic Category I requirements. The spent fuel storage racks are designed to withstand the maximum uplift forces of the spent fuel pool bridge hoist. The racks have a center-to-center spacing which is sufficient to maintain a $K_{effective}$ of 0.90 or less, assuming the pool water is unborated. The facility is designed to prevent the cask handling crane from traveling over, or in the vicinity of, the spent fuel storage areas, thereby precluding damage to the stored fuel in the event of a dropped cask.

Spent Fuel Pool Cooling and Cleanup Systems - The spent fuel pool cooling and cleanup is designed to maintain the water quality and clarity of the pool water and to remove the decay heat generated by the stored spent fuel assemblies. The cooling system consists of two 50-percent-capacity spent fuel pool cooling pumps and heat exchangers. The capability to supply makeup to the pool is by permanently installed connections to the borated water storage tank. In addition, the fuel pool piping is arranged so that the pool cannot be inadvertently drained to uncover the fuel. All inlet and outlet connections to the pool are made at the approximate normal water level in the pool.

The spent fuel pool cooling system, with both pumps and heat exchangers operating, has a capacity to maintain the spent fuel pool water temperature below 116 degrees Fahrenheit with one-third of a core stored, and below 145 degrees Fahrenheit when one and one-third cores are stored in the spent fuel pool. If up to one and one-third cores are stored in the spent fuel pool and it becomes necessary to unload a full core from the reactor, the decay heat removal system would be utilized if necessary to supplement the spent fuel cooling system.

Fuel Handling System - The fuel handling system provides the means of transporting and handling fuel from the time it reaches the plant in an unirradiated condition until it leaves after it has been removed from the reactor. The system consists of the fuel transfer canal, fuel storage handling bridge, the fuel handling building crane (shared with Unit 1), main and auxiliary fuel handling bridge, handling equipment, and the fuel transfer system.

The design of the fuel-handling facility is such that no cranes capable of carrying heavy loads can operate over the spent fuel pool. Travel of the crane handling the spent fuel cask is limited by the physical length of rails, with administratively imposed automatic interlocks to preclude an approach to the spent fuel pool area. The beam of the hoisting yoke for the cask will be oriented in such a way that any trunnion or yoke failure will impart rotational and lateral motion in a direction perpendicular to the direction to the spent fuel pool and not toward the pool. The cask handling crane is also designed so that the cask vertical lift will be less than 30 feet above a restraining surface during any moving sequence.

TABLE A-1

GENERAL DESIGN CONSIDERATIONS

System heat output, MWt	2,584[a]
Operating pressure, psig	2,155[b]
Full-power reactor inlet temp, F	555
Full-power reactor outlet temp, F	603
Design pressure, psig	2,500
Design temperature, F	650[c]
Coolant volume, incl pressurizer, ft^3	11,800
Total reactor flow (4 pumps), gpm	352,000

[a] Includes 16 MWt contribution from the energy imparted to the primary coolant by the coolant pumps.

[b] At sensing nozzle on the reactor outlet pipe.

[c] Except pressurizer and surge line, designed for a temperature of 670 F.

TABLE A-2

REACTOR COOLANT SYSTEM CODES AND CLASSIFICATIONS

Component	Code	Classification
Reactor vessel	ASME III[a]	Class A
Steam generator	ASME III[a]	Class A
Pressurizer	ASME III[a]	Class A
Reactor Coolant pump		
Casing	ASME III[a]	Class A
Motor	IEEE,[b] NEMA,[c] ASA[d]	Class A
Piping and valves	ANSI[e] B31.7-1968 and associated nuclear code cases	

[a] American Society of Mechanical Engineers, Boiler and Pressure Vessel Code, Section III, Nuclear Vessels
[b] Institute of Electrical and Electronics Engineers
[c] National Electrical Manufacturers Association
[d] American Standard Association No. C50.2-1155, C50.20-1954
[e] American National Standards Institute No. B31.7-1968

TABLE A-3

PRESSURIZER DESIGN DATA

Design pressure, psig	2,500
Operating pressure, psig	2,155
Design temperature, F	670
Operating temperature, F	649.5
Steam volume, ft^3	700
Water volume, ft^3	800
Hydrotest pressure, psig	3,125
Electric heater capacity, kW	1,638
Dimensions	
Overall height, ft/in	44/11.75
Shell OD, in.	96.375
Shell minimum thickness, in.	6.188
Dry weight. lb	291,000

Nozzles Function	No.	ID, in.	Material
Surge line	1	10, Sch 140	CS, SS clad[a]
Spray line	1	4, Sch 120	CS, SS clad[b]
Relief valve	3	2.5	CS, SS clad [a]
Vent	1	1, Sch 160	Inconel
Sample	1	1, Sch 160	CS, SS clad[b]
Temp well	1	0.375	Inconel
Level sensing	6	1, Sch 160	CS, SS clad[b]
Heater bundle	3	19.125	CS, SS clad
Manway	1	16	CS, SS clad

[a] With stainless steel (SS) safe-end added after stress relief.

[b] With Inconel safe-end.

TABLE A-4

PRESSURIZER HEATER DESIGN AND OPERATING DATA

Element/Assembly Data

Total number of elements	117
Element rating, kW	14
Element length (overall), ft-in.	9-2.75
Element OD, in.	0.66
Number of assemblies[a]	3
Number of elements per assembly	39
Assembly rating, kW	546
Overall rating (all heaters), kW	1,638

Bank Data[b]

Bank No.	No. of elements/bank	Bank rating, kW
1	3	42
2	9	126
3	27	378
4	42	588
5	36	504

[a] Heaters are equally divided into these assemblies, which are installed in individual pressurizer vessel nozzles.

[b] Heaters are divided into five banks for control.

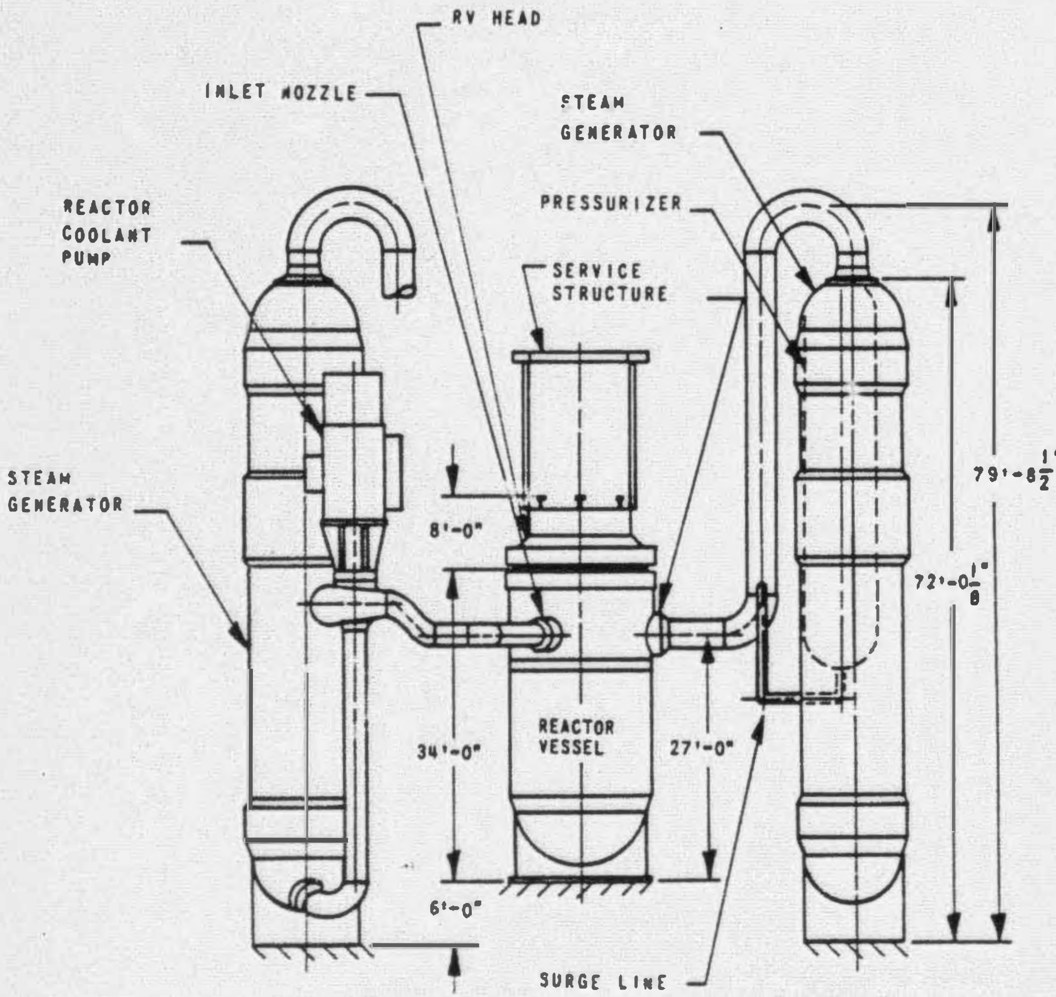

FIGURE A–1
REACTOR COOLANT SYSTEM ARRANGEMENT – ELEVATION

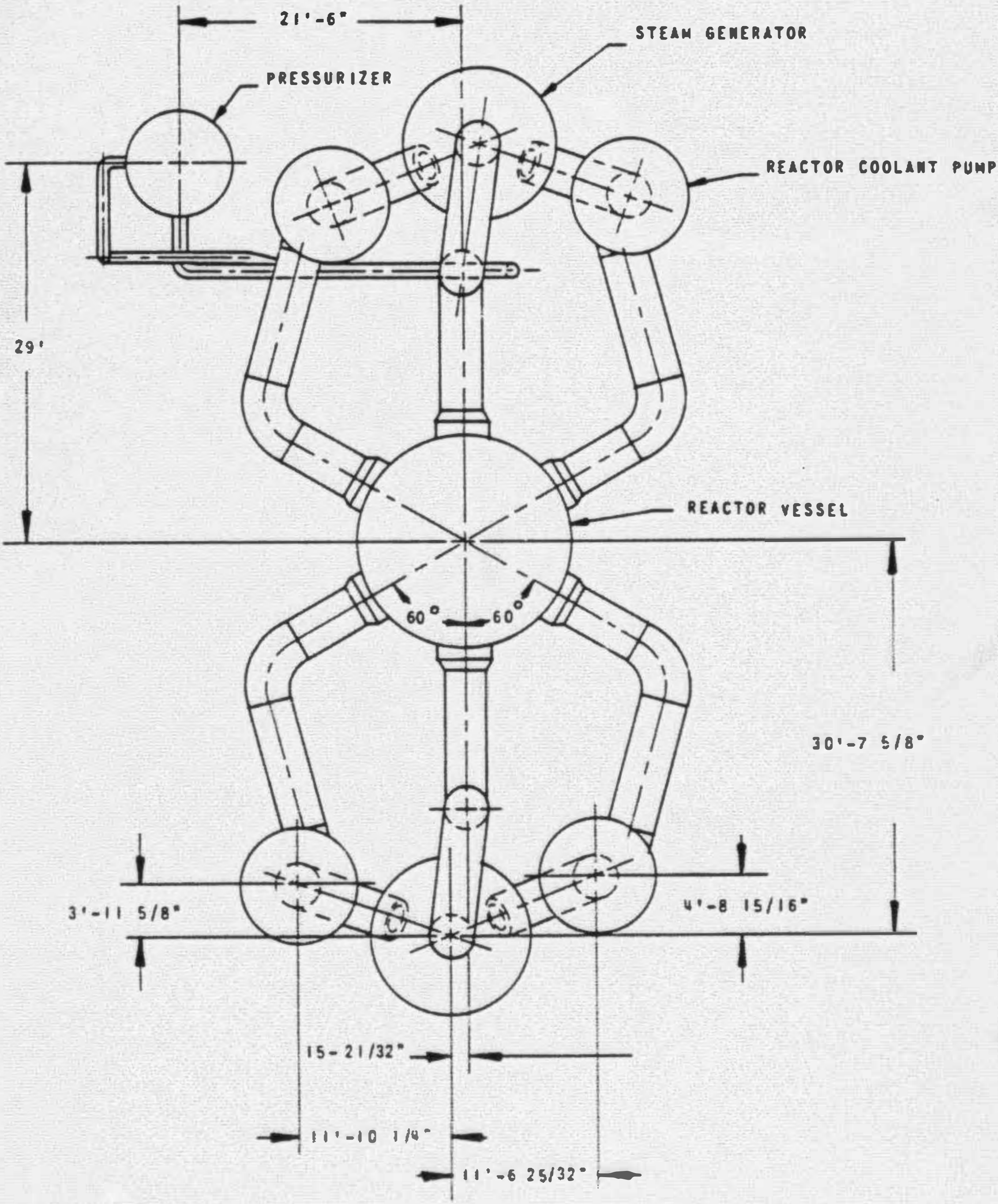

FIGURE A–2
REACTOR COOLANT SYSTEM ARRANGEMENT – PLAN VIEW

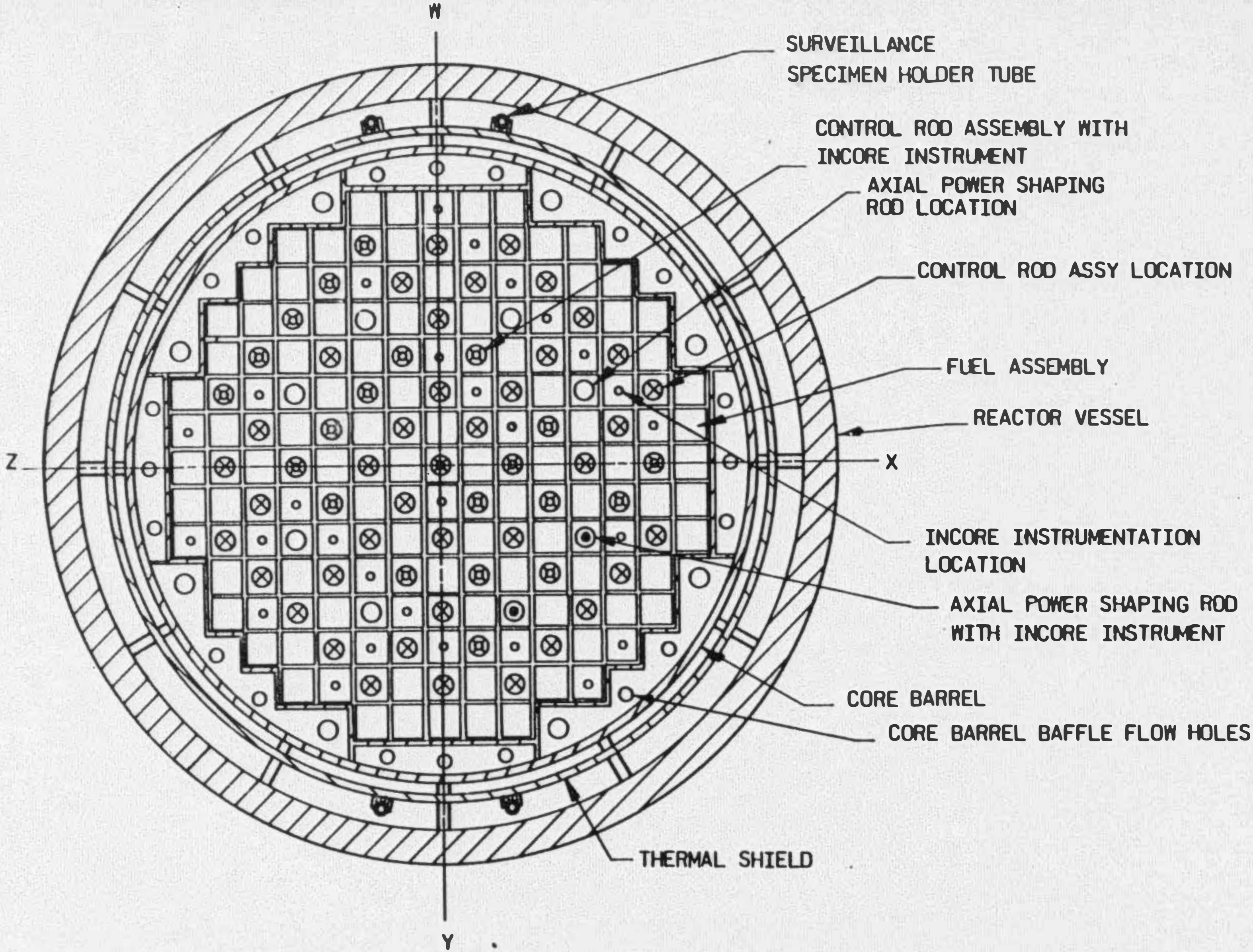

FIGURE A–3
REACTOR VESSEL INTERNALS – CROSS SECTION

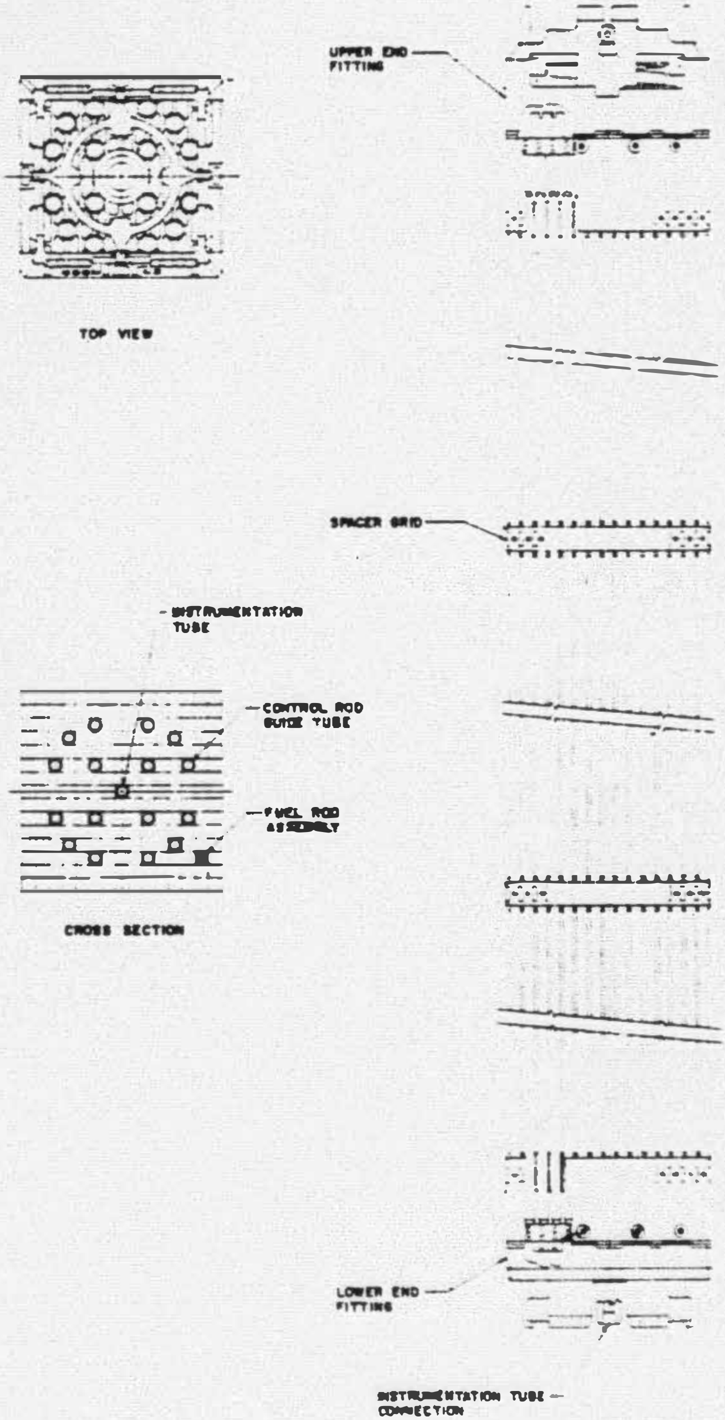

FIGURE A–4
FUEL ASSEMBLY

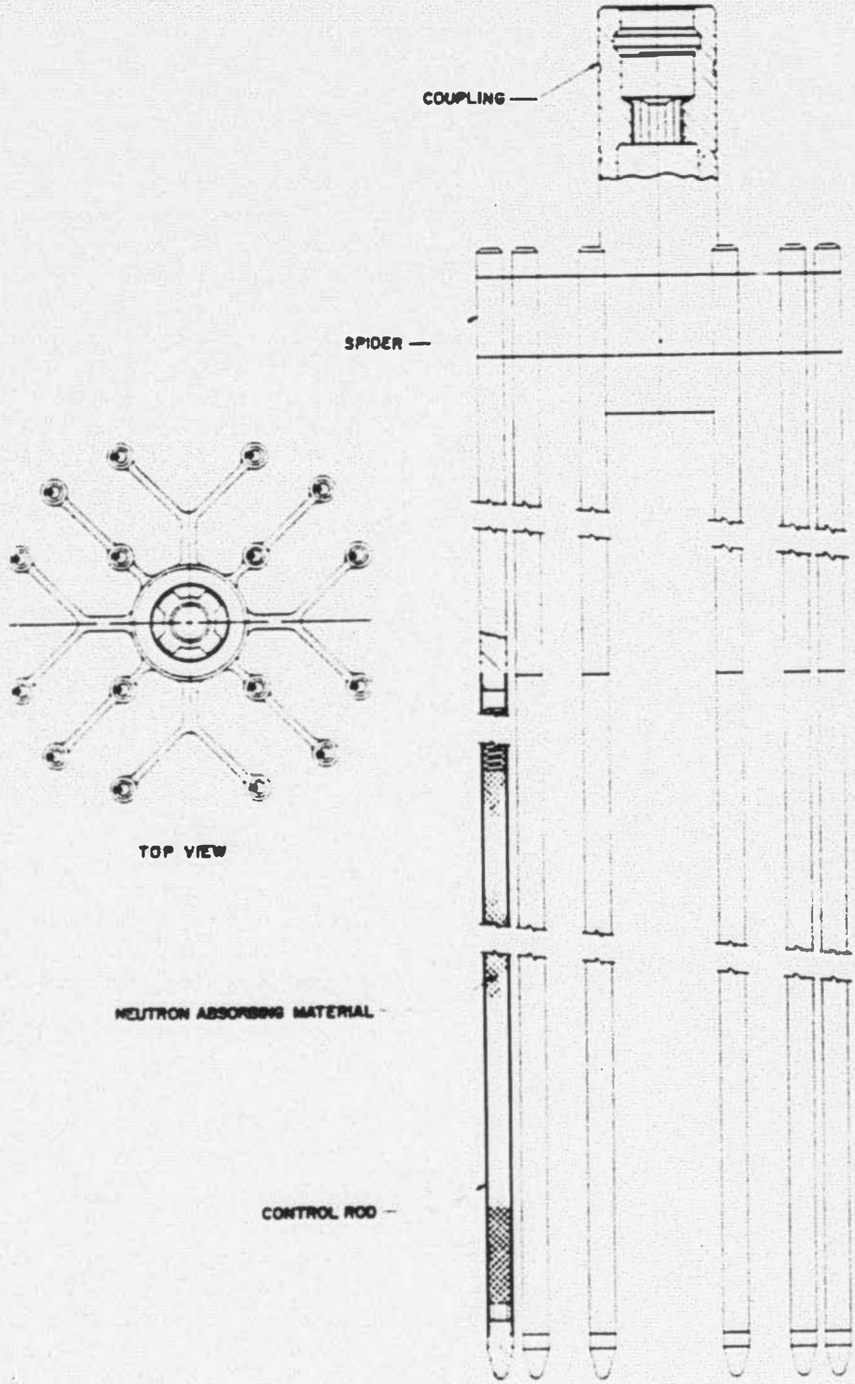

FIGURE A–5
CONTROL ROD ASSEMBLY

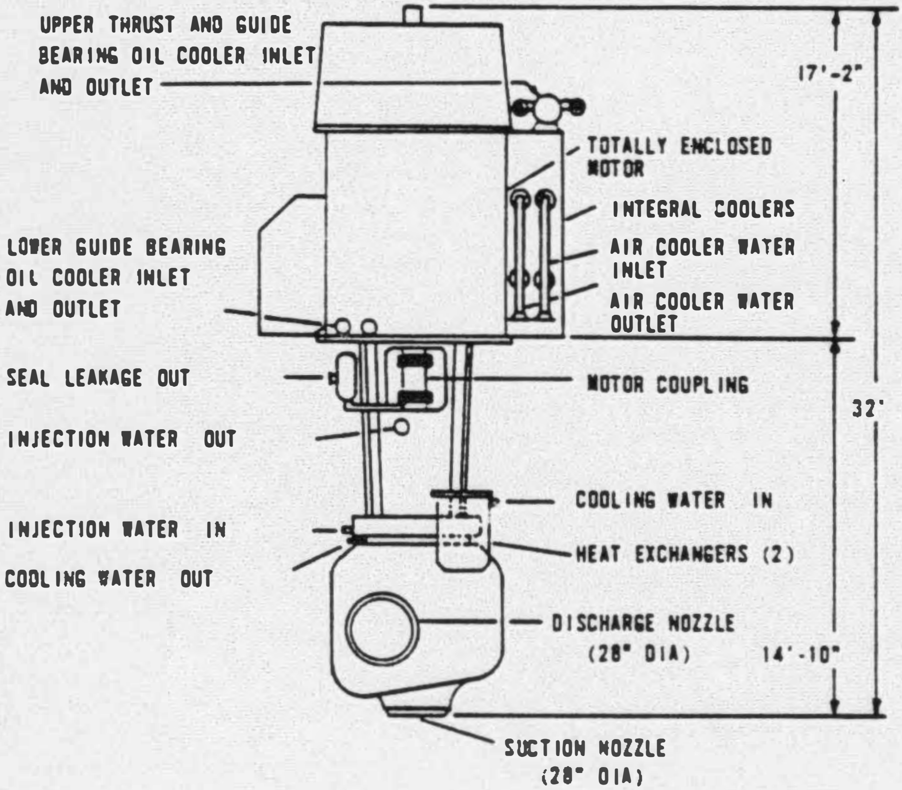

FIGURE A–6
REACTOR COOLANT PUMP AND MOTOR

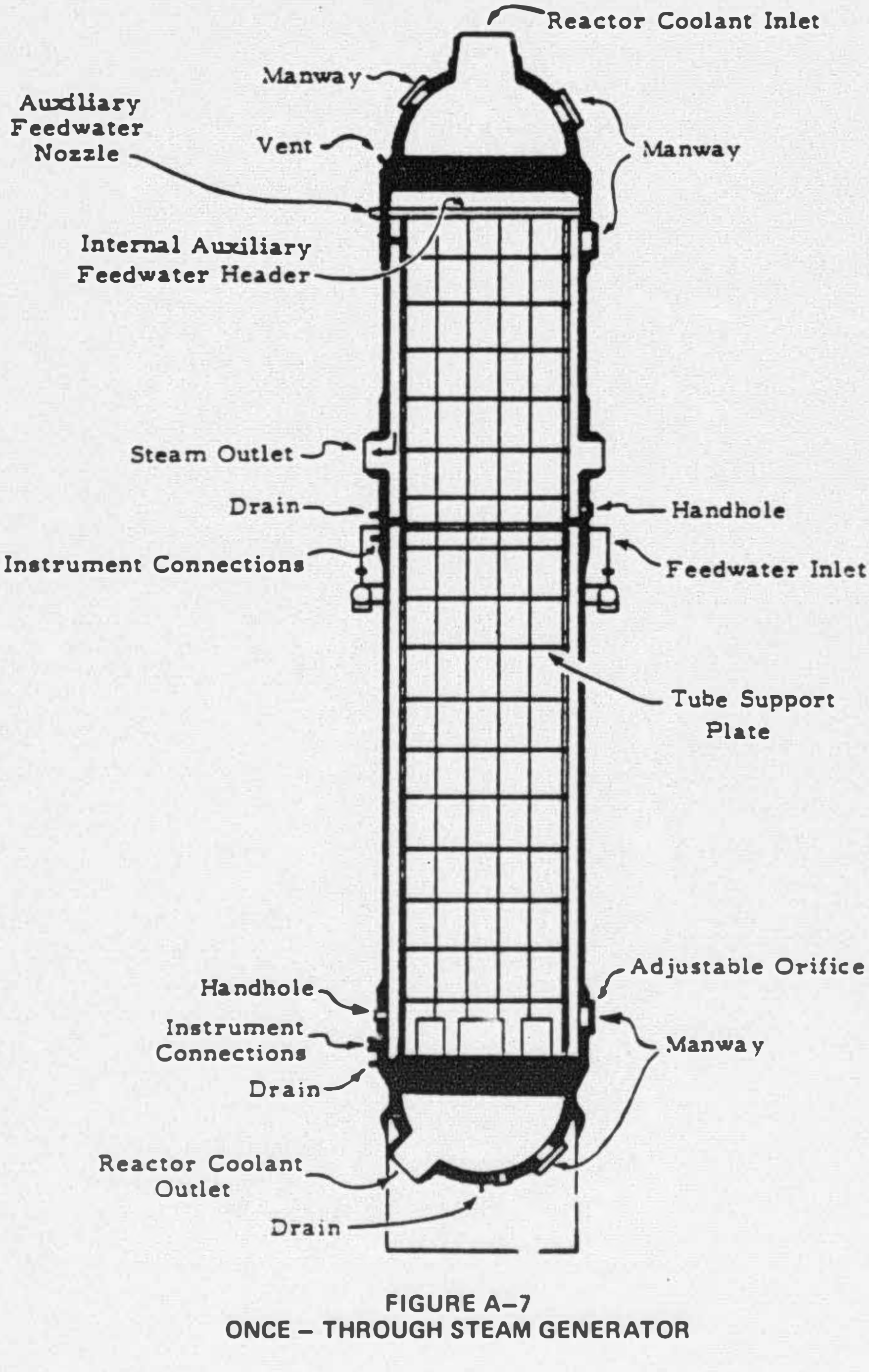

FIGURE A–7
ONCE – THROUGH STEAM GENERATOR

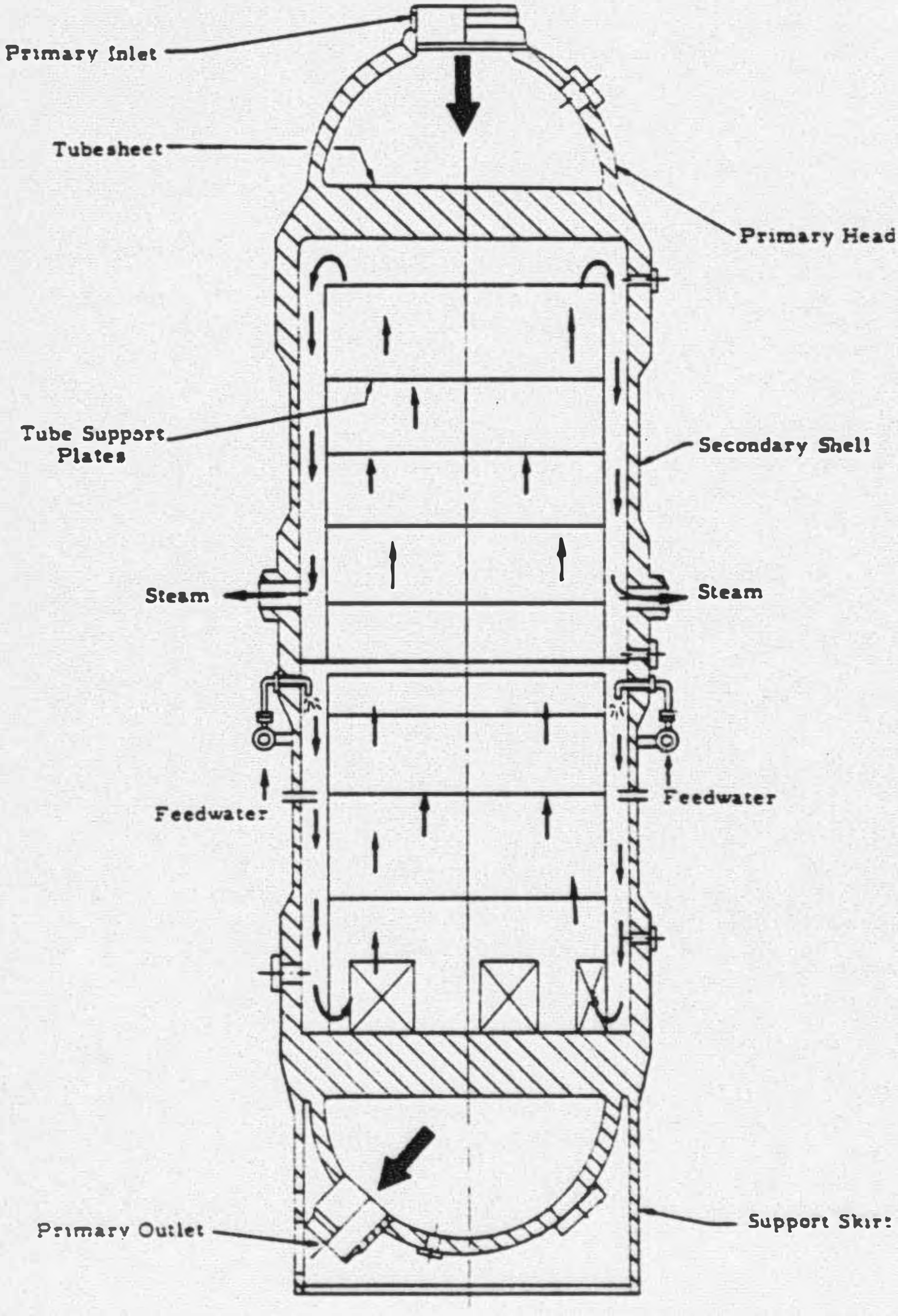

FIGURE A–8
LONGITUDINAL SECTION OF THE OTSG

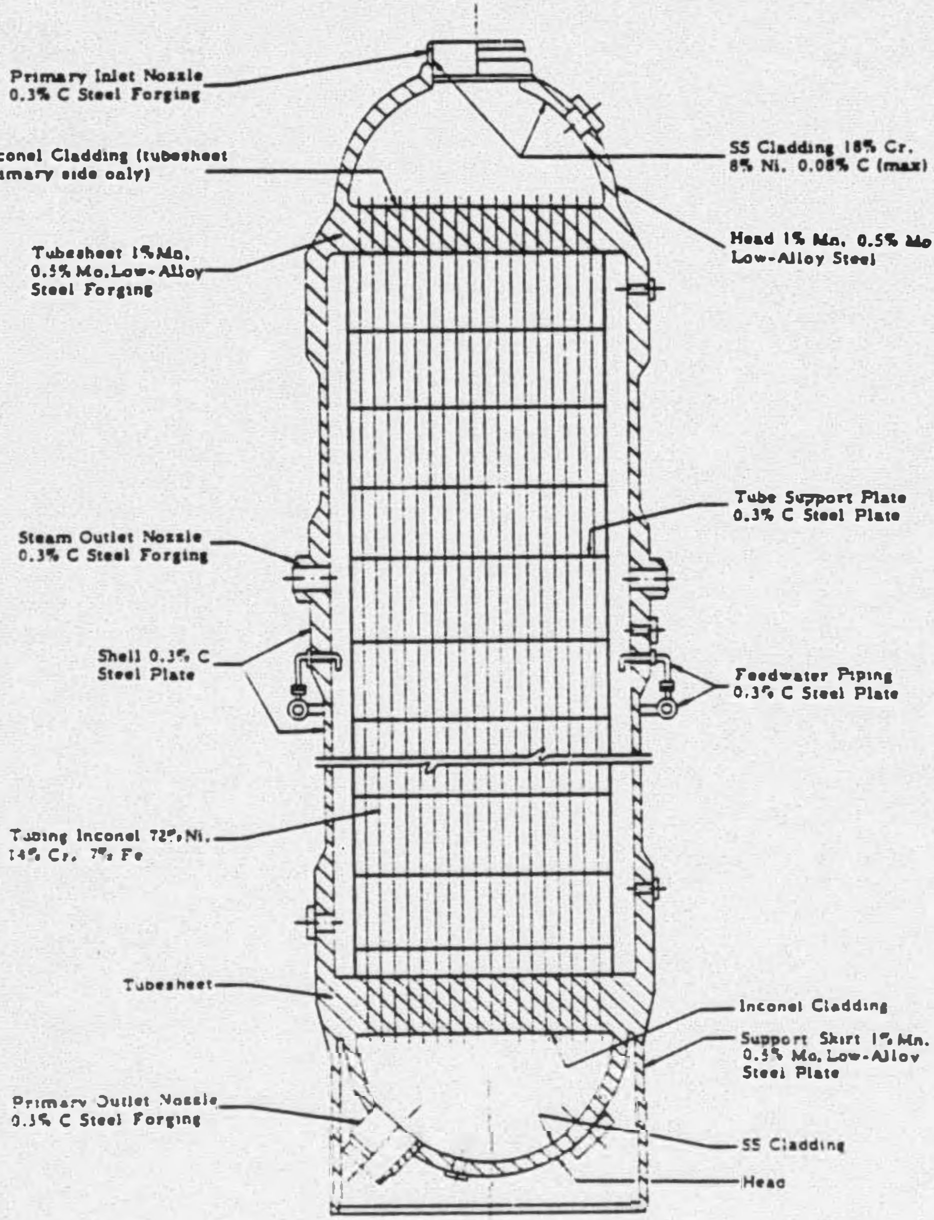

FIGURE A–9
MATERIAL OF CONSTRUCTION OF THE OTSG