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4410-85-L-0091
Document ID 0235A

May 20, 1985

TMI Program Office
Attn: Dr. B. J. Snyder
Program Director
US Nuclear Regulatory Commission
Washington, DC 20555

Dear Dr. Snyder:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
Safety Evaluation Report for Early Defueling
of the TMI-2 Reactor Vessel

Attached for your review and approval is Revision 1 of the Safety Evaluation Report (SER) for Early Defueling of the TMI-2 Reactor Vessel. This SER addresses those activities associated with the initial phase of defueling of the reactor vessel. Section 4 of this SER notes that a revised total dose estimate is being developed and will be available prior to the start of defueling. The revised estimate will reflect the most recent experience gained during mock-up and associated training and will include projected dose rates that may differ significantly from those included in this SER.

The SER concludes that early defueling activities do not constitute an unreviewed safety question and can be performed without undue risk to the health and safety of the public.

D009 w/check $150.00
1/1 # 00015952

Per the requirements of 10 CFR 170, an application fee of $150.00 is enclosed for review of this document.

Sincerely,

Thomas F. Demmitt

for F. R. Standerfer
Vice President/Director, TMI-2

FRS/RBS/

Attachment

Enclosed: GPU Nuclear Check No. 00015952

cc: Deputy Program Director - TMI Program Office, Dr. W. D. Travers

GPU Nuclear

SER 15737-2-G07-108 REV. 2

ISSUE DATE May 9, 1985

☒ ITS

☐ NSR

☐ NITS

TMI-2

DIVISION

SAFETY EVALUATION REPORT

FOR

Early Defueling of the

TMI-2 Reactor Vessel

COG ENG G K Boldt DATE 3/5/85

RTR E T. Smith DATE 3/5/85

COG ENG MGR. C. L. Rid for R. L. Rider DATE 3/5/85

NO	DATE	REVISIONS	BY	CHECKED	GROUP SUPERVISOR	MGR DESIGN ENGINEERING
LW 2	5/9/85	Revised and Reissued for Use	JCR	RJR	ETS	CLR for RLR
LW 1	4/17/85	Revised and Reissued for Use	KR	RJR	ETS	CLR for RLR
LW 0	3/5/85	Issued for Use	JCR	RJR	ETS	CLR for RLR
				CHIEF ENGINEER	N/A	

EPD-33611 10/84

Rev.	SUMMARY OF CHANGE
0	Initial issue for use
1	General update to incorporate comments on Rev. 0
2	Update to incorporate in-vessel dewatering, change in scope definition and editorial comments

SHEET	LATEST REV.	SHEET	LATEST REV.	SHEET	LATEST REV.	SHEET	LATEST REV.	SHEET	LATEST REV.	SHEET	LATEST REV.	SHEET	LATEST REV.
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REVISION STATUS SHEET

BECHTEL

DOCUMENT TITLE: Safety Evaluation Report for Early Defueling of the TMI-2 Reactor Vessel

JOB 15737

SPEC. NO. 2-G07-108

REV. 2

GPD-33000-T 10/83

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Table of Contents

1.0 Purpose, Scope and Organization

1.1 Purpose

The purpose of this Safety Evaluation Report (SER) is to demonstrate that all activities associated with early defueling of the TMI-2 reactor vessel can be accomplished without causing unacceptable risk to the health and safety of the public.

1.2 Scope

For the purposes of this SER, the phrase "early defueling" is defined as:

> All activities associated with removal of fuel element end fittings and other loose debris — including vacuumable "fines" — from the damaged core, encapsulation of those materials within specially designed canisters, dewatering of the filled and sealed canisters, and placement of the sealed canisters into storage racks located in spent fuel pool "A".

It is important to recognize, however, that per this definition, early defueling actually consists of two types of activities. These are:

1. Removal of end fittings, structural materials, and related loose debris which will <u>not</u> involve removal of significant amounts of fuel material; and

2. Removal of intact segments of fuel rods, and other pieces of core debris, and removal of loose "fines" by vacuuming operations.

Thus the first is defined as "core alteration", and the second is defined as "defueling"; i.e., removal of significant quantities of fuel.

The design features of the canisters that will be used during early defueling are described in Reference 1. Handling and use of these canisters is described in this SER. This includes:

- o Movement of the canisters into and out of the reactor vessel;
- o Filling of the canisters;
- o Closure (sealing) of the filled canisters and installation of pressure relief devices;
- o Dewatering of the sealed canisters; and
- o Placement of the sealed canisters into storage racks located in spent fuel pool "A".

Various equipment, in addition to the canisters, is required to facilitate performance of early defueling. The most notable examples are:

- o A shielded, rotatable work platform installed over the reactor vessel and its support structure, rotation drive unit, cable management system, and two jib cranes;
- o Long-handled tools that will be manipulated manually by operators from the shielded work platform through working slots in the platform;
- o A single canister support bracket that will position and support one canister within the reactor vessel during loading operations;
- o A canister positioning system that will support five canisters within the reactor vessel and interface with the canister handling bridge and vacuum system.
- o Inert gas injection systems that can displace water from the canisters while they are resident either within the reactor vessel or in spent fuel pool "A";
- o Bridge cranes for (1) movement of the canisters from one location to another, and/or (2) mechanical assistance to the operators in manipulating the long-handled tools;
- o A fuel transfer system to move the canisters between the reactor building and the fuel handling building (FHB);
- o Storage racks for sealed canisters located within the deep end of the refueling canal and within spent fuel pool "A";
- o A vacuum system that will remove fuel fines and particles up to and including the size of a fuel pellet from the debris bed.

This SER does not cover activities related to the transfer of canisters from fuel pool "A" for shipment off-site from TMI-2.

1.3 Organization

Section 2 of this SER describes the major activities and equipment needed for early defueling of the reactor vessel. Section 3 identifies the components and systems affected by or involved with early defueling. Section 4 identifies and discusses the safety concerns associated with early defueling. Section 5 presents an evaluation of the occupational radiation exposure considerations associated with early defueling. Section 6 presents an assessment of the impact of early defueling on plant activities including any impact on the operation of Unit 1. Section 7 presents a 10CFR 50.59 evaluation for early defueling and Section 8 presents the conclusions of the safety evaluation. Section 9 contains a list of references. The figures included in this safety evaluation provide a general concept of the early defueling systems. These figures are not intended to provide exact design details which are subject to change.

1.4 Licensing Documents Associated With Early Defueling

There are several licensing documents associated with early defueling activities. Included are:

- o Safety Evaluation Report for Early Defueling (this document)
- o Safety Evaluation Reports for Heavy Load Handling
- o Technical Evaluation Report for Defueling Water Cleanup System

- Criticality Report for the Reactor Coolant System
- Technical Evaluation Report for Defueling Canisters
- Technical Evaluation Report for Defueling Canister Storage Racks
- Hazards Analysis: Potential for Boron Dilution of Reactor Coolant System

A description of the scope of each of these documents is provided below.

1.4.1 Safety Evaluation Report for Early Defueling

This SER is the basic document describing the early defueling activities and providing the justification for concluding that early defueling activities will be performed in a safe manner. It provides a summary description of the early defueling activities and the equipment and systems to be used. The document assesses the safety aspects of early defueling activities and various accidents that could potentially occur while performing the early defueling activities. It also assesses the environmental impact of the early defueling activities. Where appropriate, this SER references the other licensing documents associated with the early defueling activities.

1.4.2 Safety Evaluation Reports for Heavy Load Handling (References 8 and 19)

These documents provide the NUREG-0612 evaluation for handling heavy loads during early defueling activities. The criteria to be satisfied are provided along with guidelines (e.g. lift height vs. weight) to ensure compliance. The effects of dropped loads on plant structures and systems are addressed.

1.4.3 Technical Evaluation Report for Defueling Water Cleanup System (DWCS) (Reference 2)

This TER presents the design bases for the completed DWCS and provides the information needed to demonstrate that the system will satisfy its safety functions. It addresses interfacing requirements with other plant systems and equipment; in particular, interface requirements with the canisters are discussed. The system described in the DWCS TER is more extensive than the portion to be used for early defueling activities described in this SER (see Section 3.2.7 for a description of the cleanup system that will be used during early defueling).

1.4.4 Criticality Report for the Reactor Coolant System (Reference 5)

This report establishes the boron concentration needed in the reactor vessel coolant to ensure subcriticality during early defueling activities. The report describes the various analyses that have been performed in establishing the boron concentration and the conservatisms inherent in those analyses.

1.4.5 Technical Evaluation Report for Defueling Canisters (Reference 1)

This TER presents the design bases for the three types of canisters (fuel, filter, and knockout) that will be used during early defueling and it provides the information needed to demonstrate that the canisters can safely perform their functions.

1.4.6 Technical Evaluation Report for Defueling Canister Storage Racks (Reference 9)

This TER presents the design bases for the canister storage racks in both the Fuel Transfer Canal (FTC) and the spent fuel pool. It provides the information needed to demonstrate that the canister storage racks will perform their safety functions. Summaries of the relevant analyses are included.

1.4.7 Hazards Analysis: Potential for Boron Dilution of Reactor Coolant System (Reference 6)

This report addresses the potential for a boron dilution event within the reactor coolant system, describes measures taken to lessen the potential for a boron dilution event, and provides the bases for establishing the sampling frequency requirements to be included in the appropriate operating procedures.

2.0 Major Activities and Equipment

Early defueling consists of two type activities; "core alteration" and "defueling", as defined earlier. These activities will be performed in accordance with detailed approved procedures. The synopsis provided below is intended to give a general overview of these activities.

The majority of the early defueling tooling will be available prior to the start of early defueling; however, the actual installation and sequence of use of early defueling tools will be phased and optimized as detailed procedures are developed. The activities comprising early defueling are described below, for information only, and are not intended to define procedural requirements for the sequencing of early defueling operations. A general arrangement of the 347' elevation of the reactor building (Figure 2-1) provides the layout of the early defueling systems.

Core Alteration

The goal of core alteration activities is to remove the material on the surface of the debris bed necessary to provide the clearance needed to install the vacuuming equipment. There is a pick-and-place capability using long-handled tools and debris loading baskets during this phase. Figure 2-2 illustrates the general arrangement of in-vessel equipment for core alteration activities.

The key early defueling components for core alteration, in addition to the work platform and support structure, are the single canister support bracket and miscellaneous long-handled tools. If conditions permit installation, the canister positioning system may be installed at this time. Debris baskets may be used to increase the fuel canister loading efficiency and packing density. Types of operations to be performed during this initial phase of early defueling include:

- loading fuel canisters and both top and side loading debris baskets with debris small enough to be loaded without significant size reduction operations
- loading filled debris baskets into fuel canisters
- dewatering and transferring loaded canisters out of the vessel
- moving large debris fragments to make way for vacuuming

Defueling

The goal of defueling activities is to vacuum as much of the fuel debris fines as practical from the core region following initial pick-and-place of end fittings and other structural material from the surface of the debris bed. However, additional pick-and-place of intact fuel rod segments and other pieces of core debris may be required during vacuuming. Equipment used in the core alteration activities will be augmented by the vacuum system and the canister positioning system. If the single canister support bracket continues to serve productively, it need not be removed. Types of operations to be performed during these defueling activities include:

- loading fuel canisters and both top and side loading debris baskets with debris small enough to be loaded without significant size reduction operations
- loading filled debris baskets into fuel canisters
- moving large debris fragments to make way for vacuuming
- vacuuming fines/debris into knockout and filter canisters
- dewatering and transferring loaded defueling canisters out of the reactor vessel

2.1 Equipment Installation

Early defueling equipment is fabricated in subassemblies to enable the equipment to be transported into the reactor building through the existing personnel airlock. Components will be brought into containment through the personnel airlock or the equipment hatch airlock barrel (i.e., contingent upon receipt of approval to remove the equipment hatch airlock barrel) at the 305'-6" elevation and transported to the operating deck area at the 347'-6" elevation. The main structure of the work platform can be assembled on the 347'-6" elevation and lowered into position over the reactor vessel, or it can be assembled over the reactor vessel. Similarly, the main spindle of the canister positioning system can be assembled inside the reactor building before being lowered into the vessel. Other early defueling systems are of modular design for minimum assembly and installation time and ease of maintenance. If the equipment hatch airlock barrel is removed, more extensive pre-assembly of the early defueling equipment may be done outside the reactor building.

Prior to being placed in service, the equipment will be functionally tested to assure that it will interface as designed and perform as intended. Functional testing may be performed at the manufacturer's facility, on site at the Defueling Test Assembly (DTA) and/or other areas at the site. All equipment will undergo an acceptance checkout and turnover program on site.

2.2 Early Defueling Operations and Equipment

All debris and fuel handling operations and canister loading will take place inside a water-filled structure comprised of the reactor vessel and the IIF, as modified for head removal and plenum removal operations. These operations will be supervised by an appropriately licensed operator.

The water level will be maintained at an elevation approximately 5 feet above the vessel flange during early defueling. The actual level will be determined and controlled to maintain adequate shielding and minimize dose rates to personnel on the work platform and in surrounding areas.

Vertical movement of debris and fuel will be limited so that there is at least an equivalent of 4 feet of water shielding above fuel and/or debris being handled inside the reactor vessel. This will prevent the inadvertent lifting of fuel or activated debris high enough to cause excessive exposure to personnel on the work platform. The lift restriction will be implemented mechanically through the use of rigging bars and limit switches..

Pulling on in-core instrument strings is not expected to occur during the early defueling activities. However, if the early defueling activities progress to a stage where an in-core instrument string is likely to be pulled, necessary precautions will be implemented to protect the integrity of the reactor vessel boundary.

When the loading of a fuel canister is completed, the canister closure head will be installed. The bulkhead seal surface and bolt holes will be cleaned and inspected to ensure they are free of any debris. The seal surface of the closure head will be inspected to ensure it is clean, seals are acceptable and all bolts are in place. A pressure relief valve will be installed on the closure head before the head is staged into the work area. The head will then be installed on the canister under water. After the installation of the closure head is complete, a second relief device will be installed. When the loading of a knockout or filter canister is completed, the flow fittings will be plugged and the two pressure relief devices installed. Canisters may be dewatered prior to transfer to the FHB.

The work platform is supported by the refueling canal floor via a support structure. The operators will stand on the rotatable shielded work platform nine feet above the reactor vessel flange. The work platform and support structure provide shielding to minimize the contribution of in-vessel sources to personnel exposure. Between the work platform and support structure, various lines are routed into and out of the IIF (Figure 2-3)for water treatment and air ventilation to control off-gassing. This water treatment and off-gas control piping is stationary and does not impact operation of the rotatable shielded work platform.

The work platform has a full diameter, 18-inch wide slot, with a 24 inch wide T-shaped extension, through which the long-handled tools are operated. The width of this tool working slot and the work platform shielding are designed to limit the radiation fields on the platform while the long-handled tools are being operated. When not in use, the working slot can be covered by plugs of 6-inch thick steel to further reduce the dose rates on the platform.

The equipment used in early defueling is described briefly below. A cutaway view of the canal and reactor vessel (Figure 2-4) shows the placement of some of the major early defueling tools and support structures.

Support Structure

The shielded support structure supports the shielded rotatable work platform and a stationary (service) platform area. The support structure also provides piping penetrations, an off-gas seal and shielding. The piping includes that necessary to support the DWCS, off-gas control system, RCS sample system and the level monitoring (i.e., bubbler) system. The off-gas seal will minimize the air intake flow into the support structure and work platform region. This will ensure that the off-gas system is effective in controlling fission gas. Shielding plates are located on the service platform and along the north side of the shielded support structure to reduce radiation levels in operator work areas.

Shielded Rotatable Work Platform

The shielded rotatable work platform is approximately 17-feet in diameter. The perimeter of the platform is a fabricated wide flange beam with roller assemblies mounted on the lower flange. The roller assemblies mate with the support rail mounted on the support structure. A cable drive system provides the rotational drive for the platform. This platform supports 6-inches of stainless steel shielding as well as the early defueling tools and their reaction loads and the operators. Three transfer ports are provided to allow canisters to be installed and removed through the work platform. Two removable jib cranes are mounted on the shielded work platform to aid the operators in manipulating the long-handled tools in the tool working slot. To avoid inadvertent movement of the shielded work platform, a manual disc-type brake is attached to the service platform of the shielded support structure. The skirt on the shielded work platform will serve as the disc. Additionally the skirt design includes a 1-1/2 inch hole which, when aligned with a 1-inch hole in the service platform, permits installation of a pin to verify proper shielded work platform orientation for canister removal. With the brake or pin in place, a torque limiter will be used to prevent the cable drive system from rotating the work platform. A cutaway view of the shielded work platform is shown in Figure 2-5.

Auxiliary Work Platforms

Auxiliary work platforms are provided to the north and south of the shielded work platform. The north side platform extends over the canal dam and the south side platform extends to the south end of the canal. The function of the auxiliary work platforms is to provide a staging area for personnel and equipment at the work platform elevation. The south platform will support the control consoles for operation and monitoring of the early defueling equipment. The platforms are comprised of beam and plate subassemblies that will be field-connected to the main support structure. The platforms are supported by the canal floor. The south platform is designed to be continuously manned during early defueling activities; thus it is shielded to reduce radiation levels to personnel. The north platform is not a shielded structure.

Single Canister Support Bracket (SCSB)

The SCSB (Figure 2-6) may be used to support and position a single fuel canister during early defueling prior to the installation of the canister positioning system. The SCSB also has provisions to enable a knockout canister to be installed and used in conjunction with the vacuum system if this should prove to be necessary. The SCSB is located in the long-handled tool slot on the work platform and moves on rails mounted to the shielded work platform. The SCSB is moved manually along the length of the slot within the area bounded by the core former plates. A brake that prevents SCSB lateral movement is incorporated into the design to prevent movement during early defueling. The sleeve handling tool, supported from above by a crane, lifts the canister sleeve, rotates and repositions the sleeve at other elevations, as required, to facilitate canister loading. The canister can be accessed from either end of the full length slot. The SCSB is also designed to interface with the canister handling bridge. The present condition of the debris bed should allow enough clearance for initial installation of the SCSB. However, if this is not the case, limited debris rearrangement operations can be completed to provide clearance. To lower the canister to elevations that allow the loading of debris, further debris rearrangement may be required.

Long-Handled Tools

During early defueling, long-handled tools will be used for limited pick-and-place operations and cutting of debris. The tool handles will be approximately 30-feet long. Some of them will be sectioned for ease of handling and storage. The operator will operate the tools through the slots in the rotatable shielded work platform. Most tools will be supported by an overhead crane that provides vertical and lateral motion. Several cranes are available for use, including the two jib cranes on the platform, the reactor building service crane and the polar crane. However, some of the tools can be supported by the hand rail or be hand-held. The working ends of the tools, in most cases, have hydraulic actuation.

Two types of long-handled tools will be provided; both type tools have been used previously in remote handling operations. The light weight tools are made with 1-1/2 inch aluminum pipe that is joined with pipe couplings and has integral working ends. The heavy duty tools are made with 3-inch, schedule 10, stainless steel pipe. When staged in the reactor vessel, in preparation for use, the working portion (i.e., end effector) of the tools will be installed in a tool rack located under the shielded work platform. A common handling section will couple remotely to each end effector to form the completed tool; however, some end effector changeout on the work platform may be necessary. All end effectors will be attached to the long pole handles using bolts and couplings. Two locator pins are used to assure that any hydraulic connections are aligned prior to connection.

Viewing System

The viewing system that will be used within the IIF and reactor vessel to support the early defueling vacuuming and pick-and-place activities consists primarily of the same video equipment used for the plenum inspections. The cameras and lighting systems are a combination of stationary-mounted and mobile units with various articulating features. The operators can manipulate long-handled tools to position the cameras and lights for optimum viewing.

Fines/Debris Vacuum System

The fines/debris vacuum system needed for defueling activities is illustrated in Figure 2-7. The vacuum system is composed of a pump, piping, valving and knockout and filter canisters. The system is designed to remove particles up to and including the size of a fuel pellet from the debris bed. The vacuum system is located under and supported from the rotatable work platform. It has a control console mounted on the south end auxiliary platform. To prevent overloading of either a knockout or filter canister, the system is equipped with a means for continuous monitoring of the canister weight during loading. The vacuum pickup nozzle is connected to a defueling canister by a flexible hose and is manually manipulated by a long-handled tool supported from the hand rail or from one of the jib cranes on the shielded work platform. Over-pressure protection of canisters is ensured by limiting the pump discharge pressure to a value below the canister design pressure. The system is modular to permit remote installation and removal of the pump, load cell, valving and piping sections. A leak in the submerged piping on the discharge side of the vacuum pump is unlikely to result in a radiation hazard at the surface of the water. However, if such occurs, the shielding on the shielded work platform will protect the operators from high dose rates.

Canister Positioning System

The canister positioning system (CPS) is shown in Figure 2-8. Its purpose is to support fuel and knockout canisters (i.e., a total of five) and interface with the canister handling bridge and the vacuum system. The elevation of each canister in the CPS may be changed to minimize the canister height above the core debris bed as the bed level is lowered. Canister elevation is adjusted by changing the height of the support sleeves. The CPS can position the top of the canisters at elevations 324'-11", 321'-0", and 317'-0". Only the canister under the open slot can be lowered. Canisters are moved into the loading position by manually rotating the CPS from above the platform.

Control System

The control system provides central hydraulic power, electrical power distribution, controls and instrumentation including the viewing system monitors and the overall cable management system for the following systems:

- o Viewing System
- o Shielded Work Platform Drive System
- o Vacuum System
- o Hydraulic System for Long-Handled Tools

The control systems, control console, electrical power distribution equipment, hydraulic power equipment, and instrumentation are located on the auxiliary work platform south of the reactor vessel. A local service panel for equipment plug-in will be located on the shielded work platform for operator use.

Cable Management System

The cable management system prevents entanglement and binding of cable and hose assemblies rotated onto the shielded work platform. The cable management system routes stationary and moving cables and hoses (electrical, hydraulic and pneumatic) from the electrical power distribution center to the required positions. The cables are routed through a powertrack from the canal floor onto the rotatable shielded work platform. As the shielded work platform rotates, the powertrack wraps around the support structure of the shielded work platform. The total takeup travel required for $\pm$ 190° rotation of the shielded work platform is 30 feet. To accommodate this a movement, the powertrack is threaded through a drum housing that has a 15-foot travel capacity and is designed to take-up and let-out the powertrack in a fashion similar to a block and tackle arrangement. The drum housing is attached, by a cable, to a counterweight that keeps the powertrack taut throughout its travel.

2.3 Canister Handling

Empty canisters are brought into the reactor building horizontally, either through the personnel airlock or through the equipment hatch airlock barrel (i.e., contingent upon receipt of approval to remove the equipment hatch airlock barrel). The empty canisters are transported to elev. 347'-6" and then lowered into the FTC where they are stored vertically in racks in the shallow portion of the north end of the canal. Empty canisters can also be brought into the reactor building using the fuel transfer system, if necessary. The empty canisters can be placed in the reactor vessel early defueling system using either the Canister Handling Bridge (CHB), the reactor building service crane, or the polar crane.

To transfer loaded canisters from the reactor vessel, the CHB is required (see Section 3.3.1). Prior to transfer, it is planned that the loaded canisters will be partially dewatered within the reactor vessel. This dewatering is intended to expose at least 50% of the recombiner catalyst within an inert cover gas; thus, the potential for significant hydrogen pressure buildup during canister transfer will be minimized. However, this partial dewatering is not required to safely transfer the canisters to the FHB for dewatering and storage.

Once the partial dewatering is complete, the CHB is positioned over the canister removal port, the tool slot, or the filter canister removal ports, depending on where the canister to be transferred is located. The canister removal port and the filter canister removal ports have steel shielding in the air gap under the work platform. This shielding is designed to extend down into the water. The canister transfer shield bottom is at elev. 331'-10 1/2", providing a nominal 4-1/2 inch clearance between the work platform and the shield. During canister transfers, a shield collar will be lowered to touch the work platform. The collar will automatically stop when it touches the work platform and the work platform has been designed to withstand the force imposed by the collar. Vertical shielding between the platform and the water is not present at the alternate transfer location in the tool slot or on the single canister support bracket. Shielded panels for the tool slot will be available to close off the portion of the slot not being used for canister transfer. Use of these shield plugs or other shadow shields, or relocation of personnel, may be used to minimize worker exposures during transfers from these locations. Access to the north end auxiliary work platform may be restricted when lowering filled canisters into the deep end of the FTC.

The CHB will remove canisters from the vessel, transfer them to the north (deep) end of the canal, and then lower the canisters into either the canal storage racks or one of the two fuel transfer system upenders. Shielding in either of these locations is provided by the canal water. Canisters will be transported from the canal storage racks to the fuel transfer system upender using the CHB.

The fuel transfer system upender in the reactor building will lower the canister into a horizontal position to enable its movement into the FHB. The canister will be brought back into the vertical position by an upender located within spent fuel pool "A" in the FHB. The canister is removed from the fuel transfer system by the CHB in the FHB (see Section 3.3.1) and is transferred to either a submerged storage rack location or the dewatering station in spent fuel pool "A".

If the canisters are initially placed in the storage racks, the CHB will be used to transfer the canisters from the storage racks to the dewatering station located in fuel pool "A" where dewatering is to be performed.

Once the canister is lowered into the dewatering station, inert gas is used to purge the standing water from the canister. The water and inert gas flow through a sight glass and filter and into a holding tank. The sight glass permits the detection of gas bubbles which are an indication that the canister is dewatered.

The CHB will be used to weigh the canister before and after dewatering in the FHB to determine the amount of water removed during dewatering and to demonstrate that the dewatered canister complies with the maximum shipping weight restrictions. The shipping weight restriction is 2800 lbs.; 5% of the canisters are allowed to be up to 5% overloaded (i.e., 2940 lbs). After the canister is dewatered, it is filled with inert gas to a positive pressure. The CHB is then used to remove the canister from the dewatering station and move it to a storage rack location.

The CHB's can also replace the filter canisters in the DWC system racks located in spent fuel pool "A" and in the deep end of the FTC. These canisters are dewatered and stored in the FHB after use via the same procedures used for canisters transferred from the reactor vessel.

2.4 Off-Normal Events

Potential off-normal events, such as electric power failure, loss of compressed air, failure of viewing systems, jamming of early defueling equipment, tool failures, and operator error have been considered during the design of the early defueling systems. It was the design objective to preclude safety concerns during such potential events. This safety evaluation addresses those off-normal events which have a potential impact on the health and safety of the public. Other off-normal events, such as those noted above, may cause a delay in the early defueling activities until repairs can be made, but will not cause unacceptable risk to the health and safety of the public.

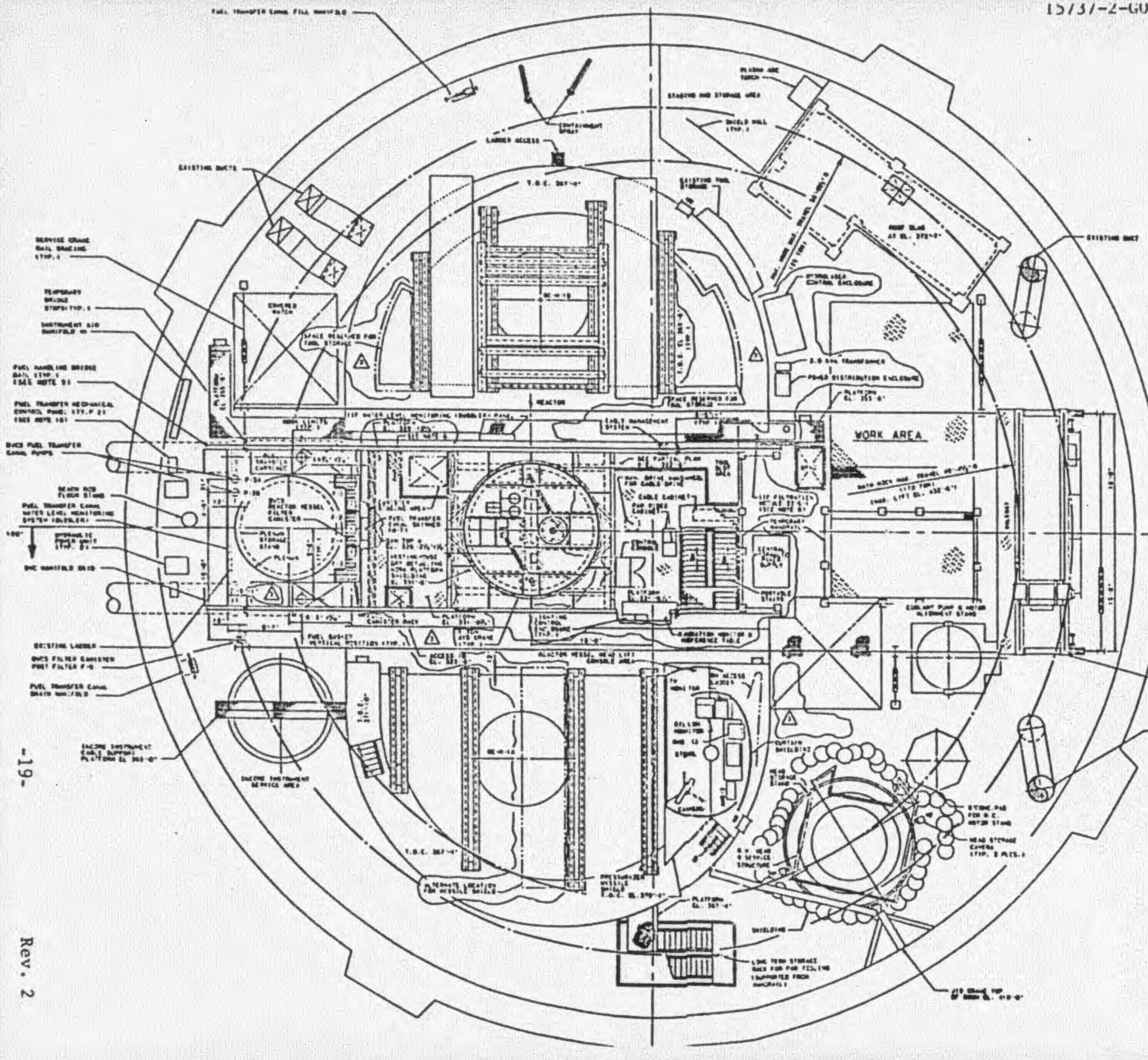


Figure 2-1

General Arrangement
Reactor Building
Elev. 347'

Core Alterations

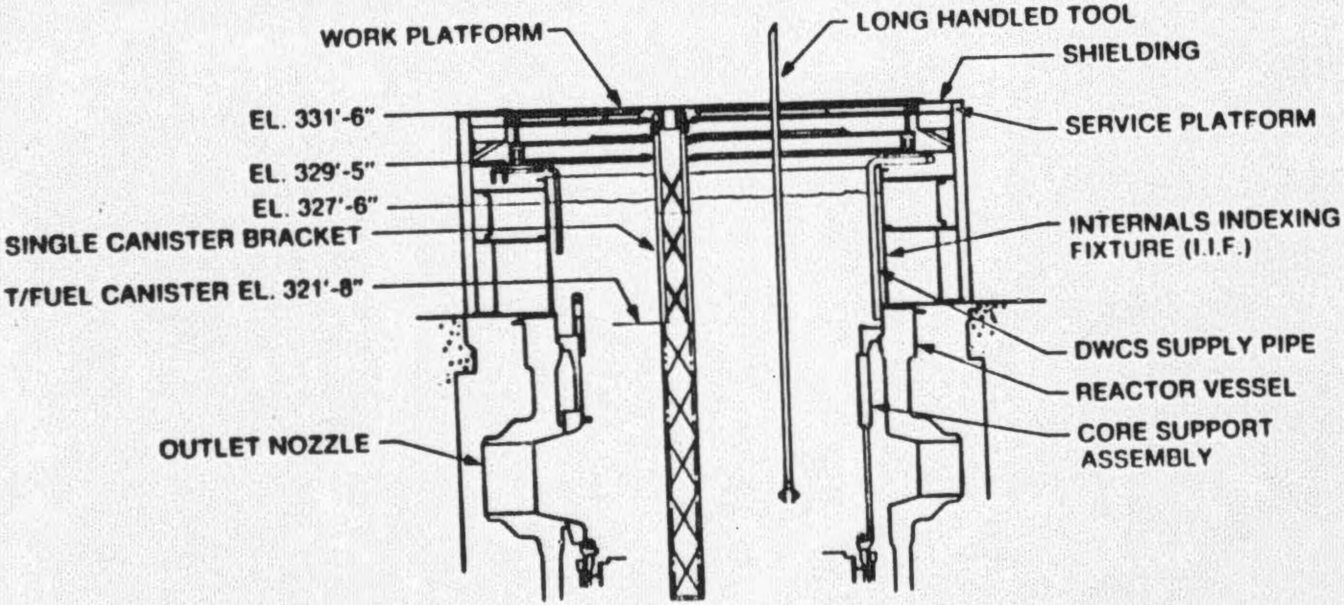


Figure 2-2

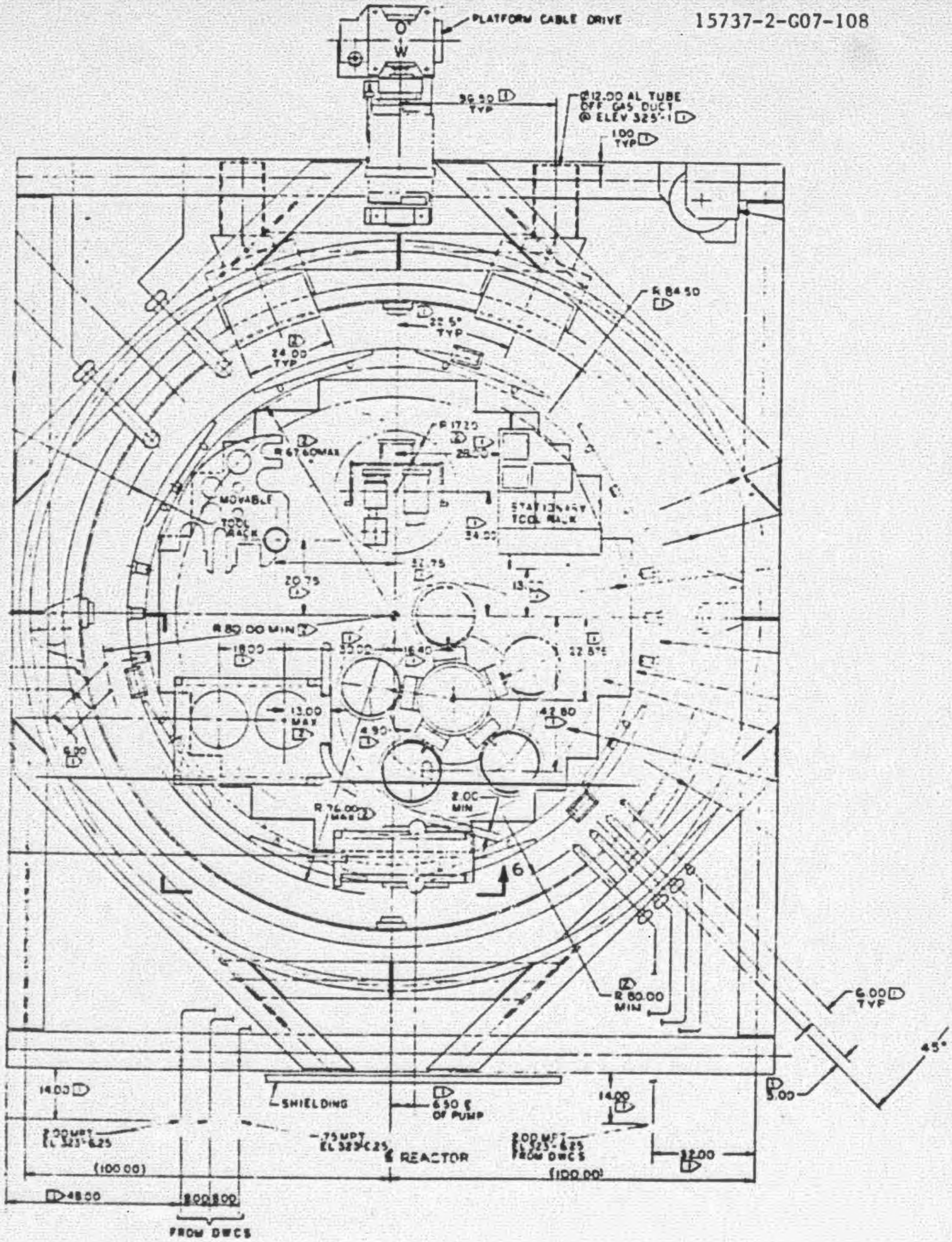


Figure 2-3 Piping Routed Through Support Structure

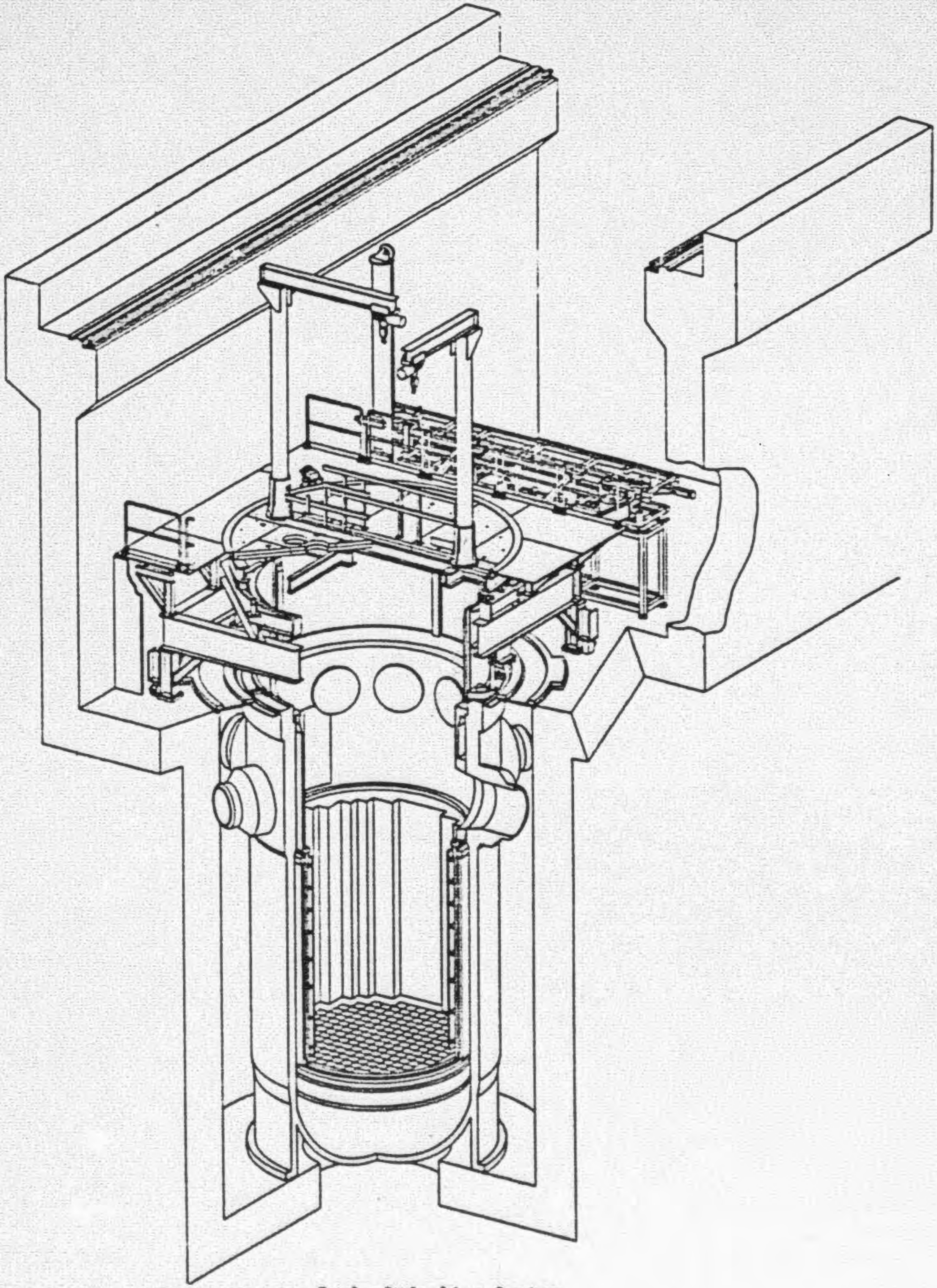


Figure 2-4 Early Defueling System

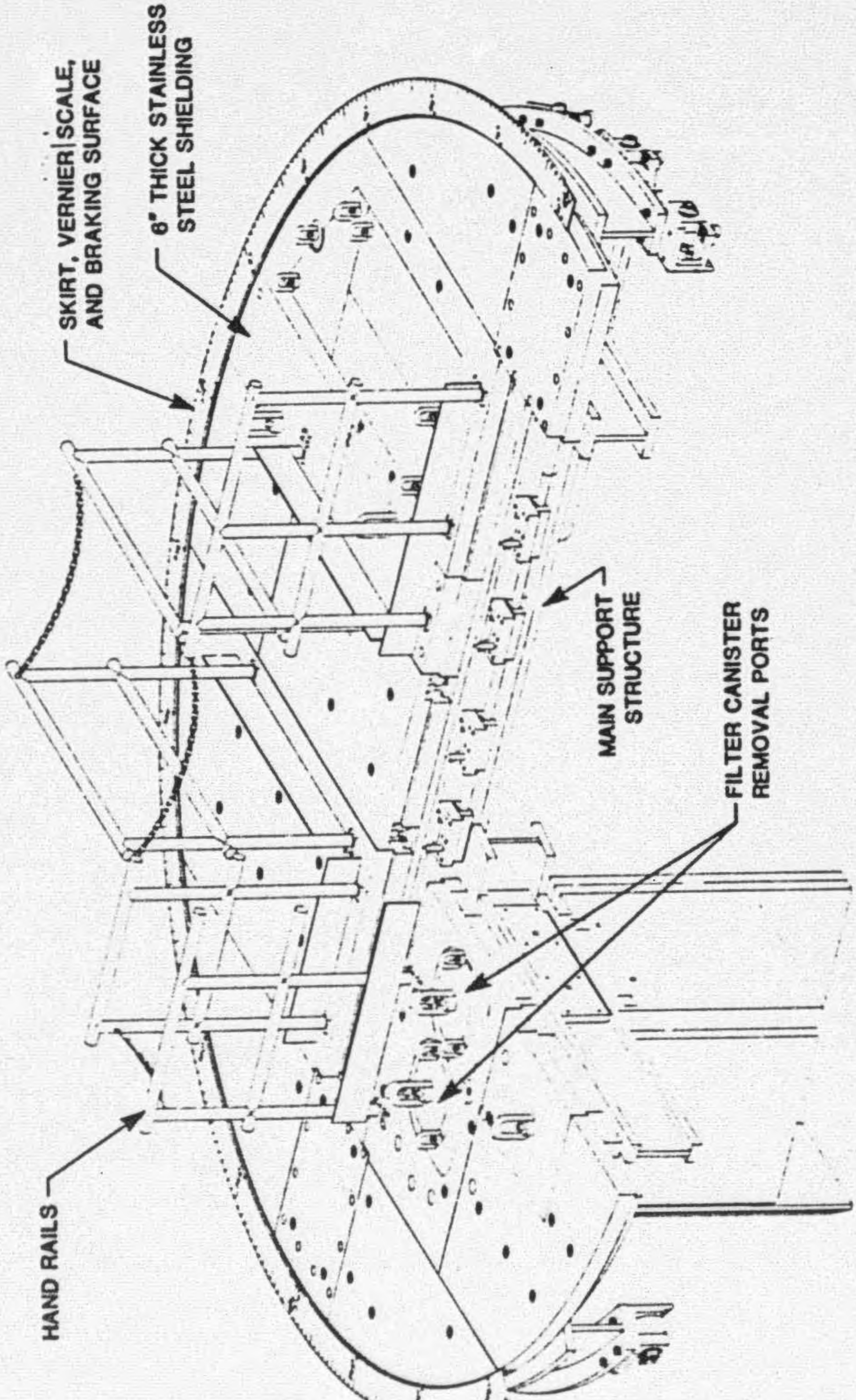


Figure 2-5 Shielded Work Platform Assembly

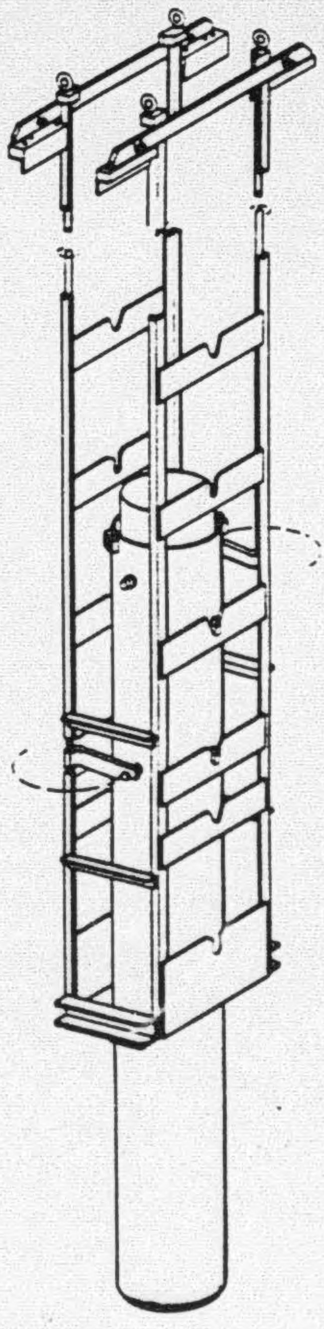


Figure 2-6 Single Canister Support Bracket

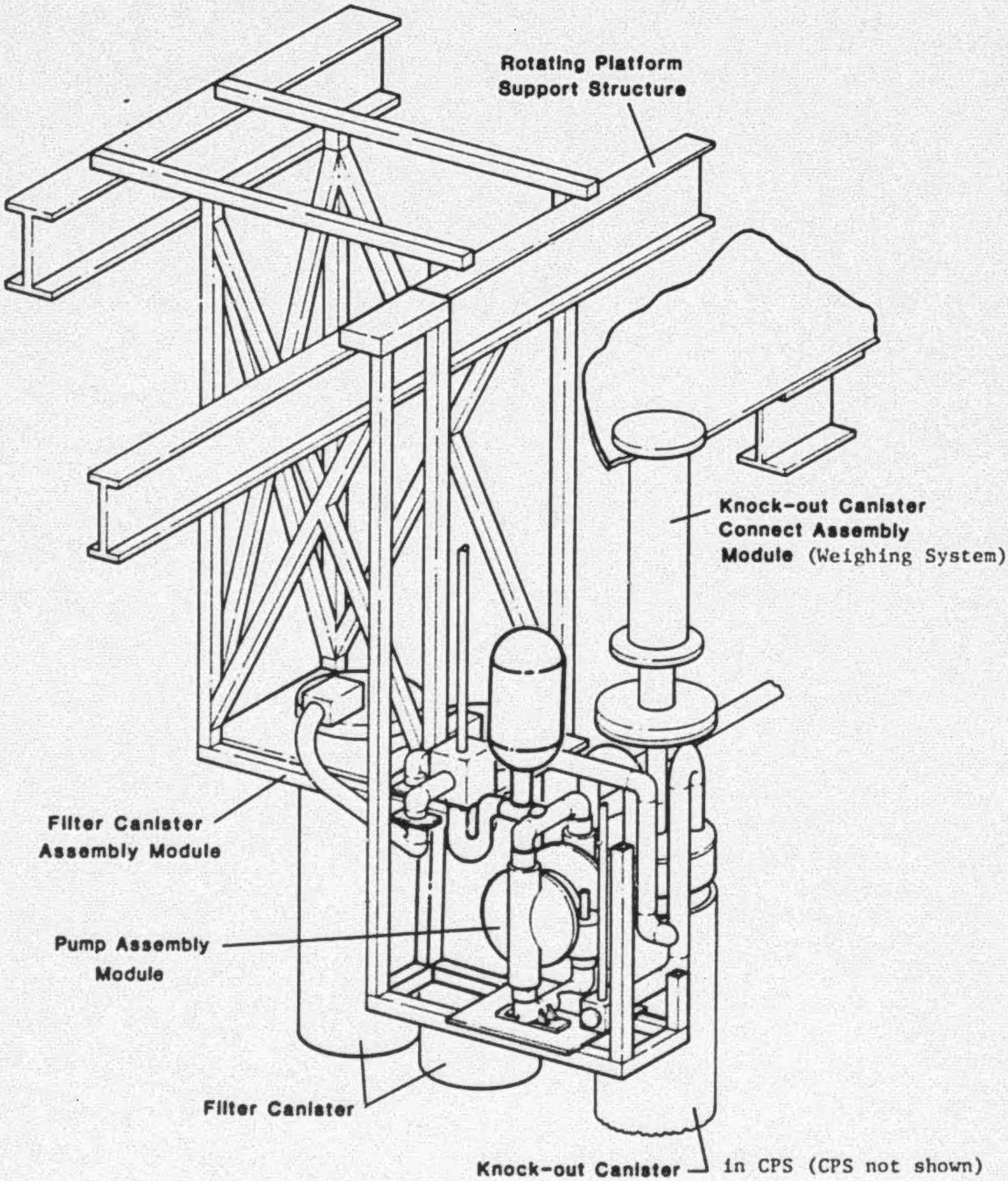
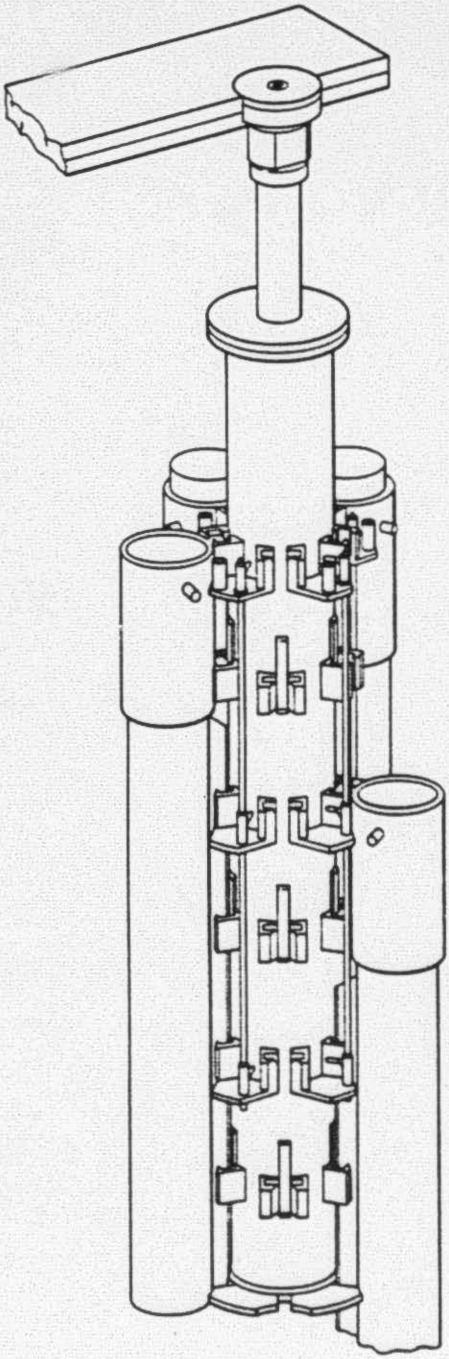


FIGURE 2-7 FINES/DEBRIS VACUUM SYSTEM



CANISTER POSITIONING SYSTEM

Figure 2- 8

3.0 Components and Systems Affected

3.1 Containment

During early defueling operations, the uncontrolled release of radioactive material to the environment is prevented by maintaining the integrity of the reactor building containment. Core alterations, defueling and filled canister transfer operations will be stopped, as required by Technical Specifications, any time the equipment hatch airlock barrel is removed or both air lock doors are opened. Containment integrity prevents an unacceptable release of radioactive materials to the environment and will be maintained in accordance with Technical Specifications. The containment purge systems will be operated in accordance with procedures approved by the NRC pursuant to Technical Specification 6.8.2.

3.2 Systems

3.2.1 Reactor Coolant System (RCS)

During early defueling activities, the safety function of the RCS is to provide a sufficient volume of adequately borated water to maintain subcriticality of the core and to provide decay heat removal. In addition, the RCS serves as a water shield for radiation sources inside the vessel. The RCS will be in a depressurized condition with the water level maintained approximately 5 feet above the vessel flange; i.e., within the IIF.

The RCS water will be sampled at a frequency sufficient (i.e., to be determined based on Reference 6) to ensure adequate protection against the effects of a boron dilution event. The sampling will be accomplished using an RCS sampling system which provides remote sampling capability. The system takes suction from the 315' elevation inside the vessel. Inadvertent rapid drain down via suction is precluded by the use of small bore piping. Representative sampling is assured by normal thermal mixing in the vessel and adequate line purge prior to sampling. In addition, sampling will be performed in accordance with Recovery Operations Plan 4.4.9. When the DWCS is operating, it will provide another means of RCS sampling.

3.2.2 Reactor Coolant System Level Monitoring

Level monitoring is accomplished through three independent systems in the RCS. There is a pressure transmitter connected to the RCS hot leg which monitors the water level using the hydrostatic head of the water over the transmitter. The transmitter has a digital readout located in the control room. A local level indicator, located at the 282' elevation of the FHB, is also provided and connected in parallel with this transmitter. During a loss of power this local indicator could be used to provide level information.

A new bubbler system, similar to the one used since the installation of the IIF, functions as a second level monitoring system. The bubbler is an open ended system into which air is introduced at a constant flowrate. The air pressure equals the hydrostatic head of water above the open end of the tube and is sensed and displayed as the water level. The readout is located in the control room.

A third means of level indication is a tygon tube attached to the cold leg. During a loss of power, this tube would still be available to monitor water level.

The water level in the IIF will be maintained at an elevation approximately 5-feet above the vessel flange. Low and high level alarms will be set to signal any unacceptable trends (i.e., increase or decrease) in the water level.

3.2.3 Standby Reactor Coolant Pressure Control System (SPCS)

The normal function of the SPCS is to maintain the RCS volume at desired levels. During early defueling, when the RCS is in a depressurized condition, the SPCS, if available, could provide additional capability for makeup of borated water. The SPCS is capable of injecting borated water from the charging water storage tank into the RCS to ensure that the desired level is maintained and to ensure that the reactor coolant can be maintained within the prescribed boron concentration limits.

3.2.4 Mini-Decay Heat Removal System (MDHRS)

The MDHRS is designed to provide for heat removal from the RCS by forced circulation through the core. During early defueling, decay heat removal will be by loss to ambient cooling. The MDHRS, if available, could provide a backup to loss to ambient cooling. The MDHRS may also provide a pumping capability for the transfer of water from the borated water storage tank to the RCS.

3.2.5 Nitrogen System

The Nitrogen for Nuclear and Radwaste System stores and supplies nuclear grade nitrogen at various pressures to several different systems. The nitrogen is used as a diluter of hydrogen and radwaste gas, as a purge gas, to maintain system pressures and as a non-corrosive cover gas.

3.2.6 Internals Indexing Fixture (IIF)

The IIF was installed on the reactor vessel after the head was removed. It has been modified and bolted to the reactor vessel flange and partially filled with RCS water. The IIF will remain in place during early defueling of the reactor vessel to provide water shielding over the early defueling equipment and canisters located within the reactor vessel. More details of the IIF can be found in reference 4.

3.2.7 Defueling Water Cleanup System (DWCS)

The DWCS is intended to be the primary water processing system during early defueling and is designed to remove radioactive ions and particulate matter from the water in the deep end of the FTC, spent fuel pool "A" and the reactor vessel. The system is composed of two major subsystems which allow greater processing flexibility. These two subsystems are the Reactor Vessel Cleanup System and the FTC/Spent Fuel Pool Cleanup System. Further DWCS detail and the system safety evaluation are contained in Reference 2.

The complete DWCS is not required to begin early defueling. It is planned that the filtration portion of the Reactor Vessel Cleanup System will be operating at the start of early defueling to remove particulate matter. However, initial processing of the RCS to remove radioactive ions and maintain RCS radioactivity at acceptable levels will be accomplished using the Submerged Demineralizer System (SDS) on an interim basis. Processing would be accomplished, if required, by taking a slip stream from the DWCS through SDS. This method would provide an adequate process capability for the RCS until DWCS becomes fully operationaly.

During this interim period, processing of water in the FTC will be accomplished, if required, by routing the FTC water through the SDS via the FTC drainage system. A direct return path to the FTC from SDS may be provided by using portions of the DWCS Fuel Canal/Spent Fuel Pool Cleanup System.

3.2.8 Canal Dam

The storage of the plenum assembly and defueling canisters and canister transfer operations require that the deep end of the refueling canal be filled with water. A dam weighing approximately two tons, fabricated from stainless steel, is placed in the existing keys in the floor and the east and west walls of the refueling canal. This dam allows the deep end of the canal to be filled to a level above the floor of the shallow end of the canal (el. 322'-6"). The main structure of the dam is a 3/8 inch stainless steel plate with stiffeners behind and along the perimeter of the plate. The dam has two redundant parallel inflatable gaskets along the dam edges which provide a leaktight seal. The canal dam is addressed in more detail in Reference 3.

In the low probability event that excess dam leakage or a complete loss of the dam function occurs, the water level in the deep end of the FTC and in spent fuel pool "A" would be lowered. The tops of the canisters positioned in these areas will remain below the resulting water level; thus, the canisters will remain shielded. The level of water shielding over both the plenum assembly and the canisters will be reduced; however, the canal could be flooded to increase the water level and reduce radiation exposure levels, if necessary.

The water level in the shallow end of the FTC following dam failure would be well below the water sensitive electrical equipment associated with the defueling work platform, including the cable management system and the platform drive motor. Consequently, dam failure will not result in failure of the defueling platform systems.

3.2.9 Fuel Transfer System

The Fuel Transfer System (FTS) is used to move canisters between the reactor building and spent fuel pool "A". The FTS has been modified to handle asymmetrically loaded fuel canisters weighing up to 3355 lbs. Canisters are handled by the FTS in the same manner as fuel assemblies. The basket on the FTS has been modified to accept canisters and the drive has been modified to be more easily maintained and less sensitive to track misalignment. The fuel transfer system will undergo full operational testing and turnover prior to use.

3.2.10 Dewatering Systems for Defueling Canisters

The Dewatering Systems (DS) are designed to purge water from submerged defueling canisters using inert gas. There are two locations at which dewatering may be performed. The first is inside the reactor vessel where canisters may be dewatered prior to removal from the reactor vessel. The second is in the dewatering station in the FHB. The DS also provides an inert cover gas.

Normally, canisters will be dewatered initially inside the reactor vessel to expose the hydrogen/oxygen recombiner catalyst. Exposure of the catalyst will control the composition of gas and potential pressurization of the canisters due to radiolytic decomposition of water.

In-Vessel DS

The in-vessel DS can be utilized to partially dewater canisters prior to transfer to the deep end of the FTC. Canisters will be dewatered using bottled inert gas located on or near the work platform. The connection to the canisters from the inert gas supply will be via a 1/4-inch supply hose. The inert gas supply pressure will be below the design pressure of the canisters to prevent overpressurization of the canisters.

The water displaced from the canisters, as well as any excess inert gas, will be vented directly to the reactor vessel. Any airborne particulate activity resulting from the in-vessel dewatering will be removed by the off-gas system provided by the work platform.

FHB DS

The FHB DS will be installed in and around the northeast end of spent fuel pool "A". The system will be designed to be installed with the fuel pool either flooded or dry. Installation will be in accordance with approved procedures.

There is no source of unborated water to the FHB DS which would dilute the boron concentration of the water removed from the canisters. The FHB DS pumps have been sized to prevent accumulation of significant quantities of fuel in them and the piping is designed to prevent the same. Protection for the defueling canisters from over-pressurization during dewatering is ensured by limiting the FHB DS operating pressure to below the canister design pressures.

The majority of the potentially radioactive components of the FHB DS are submerged. Consequently, the fuel pool water will provide most of the necessary shielding. The FHB DS work platform will be provided with additional shielding to reduce the dose rates to personnel on the platform from those components that are not submerged.

The FHB DS vents excess inert gas through automatic vent valves to a holdup tank. This tank is then vented to the SDS off-gas filter via a single tie-in to the 6-inch, off-gas line. The most significant expected consequence of this tie-in is that the SDS off-gas filters may have to be changed out more frequently.

The planned activities associated with the operation of the FHB DS are not expected to release any appreciable amounts of gaseous or particulate activity. However, any particulate activity will be processed by the HEPA filters in the SDS off-gas system, ensuring a negligible release to the environment.

3.2.11 Decontamination Spray System (DSS)

The DSS provides the ability to flush radioactive debris from the surface of the canisters, end effectors, long-handled tools, and other equipment as each item is removed from the reactor vessel. The system tubing, fittings and nozzles are mounted to the underside of the shielded work platform, ensuring all flush water and radioactive debris will be confined to the reactor vessel. The flush water will be borated ($\geq$ 4350 ppm). A hose will be installed from the FTC fill manifold to a manifold on the shielded work platform to supply the borated water.

3.2.12 Off-Gas System

The off-gas system creates an airflow through the work platform, into the IIF enclosure, and out to the reactor building atmosphere through a filtration unit. This airflow through the platform will prevent radioactive gases produced under the platform from reaching personnel working on the platform. The 4000 scfm filtration unit, equipped with HEPA filters and a moisture-separating, multi-density prefilter, can maintain an average flow velocity of 150 fpm through the work platform openings. The filtration unit is connected to the IIF enclosure via two, twelve-inch flexible ducts and discharges directly to the reactor building atmosphere in an area located away from the platform.

3.3 Material Handling Equipment

3.3.1 Canister Handling Bridges (CHB)

Two CHB's are required for canister transport during early defueling operations, one in the reactor building and one in the FHB. The existing storage fuel handling bridge, with a new canister handling trolley, will transfer loaded canisters in the FHB spent fuel pool "A" area. The existing auxiliary fuel handling bridge, with a new canister handling trolley, will transfer canisters in the FTC area of the reactor building. Shielding for the canisters is provided by a fixed, shielded mast (i.e., canister transfer shield) attached to the trolley.

The existing auxiliary and storage fuel handling bridges will be used to provide north-south movement of the canisters. The new trolleys will provide east-west movement of the canisters. The bridges, trolleys and hoists are motor-driven and controlled from the trolleys. Each new trolley will include the canister transfer shield, grapple, grapple guiding tool, hoists, cable and hose reels, and a load cell with a digital readout. The trolleys are capable of centering the canister handling grapple over all canister locations, using a digital location system. Visual matchmarks on the CHB trolley will provide backup to the digital system.

The canister transfer shield (CTS) is made of stainless steel encased lead (2-1/2 inches Pb) fixed to the floor of the trolley. The CTS is provided to reduce radiation fields to an acceptable level during canister transfer. There is a 3-inch thick lead shield on top of the grapple tool to minimize the radiation exposure to the operator on the trolley. A 9-foot high shield collar is located on the lower end of the CTS in both the reactor building and FHB. The collar on the CTS in the FHB consists of 1-1/2 inches of lead. The upper portion of the collar in the reactor building also consists of 1-1/2 inches of lead. The lower portion (i.e., approximately 30 inches) consists of 4-inches of lead. These collars are retractable to provide any necessary clearances during movement. The collars can extend down into the water when over the transfer mechanism or racks to eliminate any gaps between the water level and the lower end of the CTS.

The CHB grapple will hoist canisters out of the vessel, through the work platform and into the transfer shield. The CHB grapple is an air-operated, single-point grapple which fails in the engaged position. The grapple actuating system consists of an air cylinder coupled to a spring-loaded cam rod. The camming action causes the bottom end of three pawls to rotate outward and engage the special handling ledge integral to the defueling canister lids. A grapple engaged/disengaged position indicator light is provided on the canister handling trolley to give indication that the grapple has engaged.

The grapple is the primary means of preventing a canister drop. While a canister is being raised or lowered from the CTS, it is the only retaining device. However, during canister transport, two redundant canister retention mechanisms, located at the bottom of the transfer shield, provide a diverse secondary means of preventing a canister from falling out of the shield in the unlikely event of grapple failure or accidental release. An engaged/disengaged indicator light is provided on the canister handling trolley to give positive indication that the retention mechanisms are functioning properly.

The components and structure of the CHB's are designed to conform to safety standards of ANSI B30.2-1983 and ANSI B30.16-1978. Prior to their use in early defueling, the CHB's will be tested to meet the requirements of ANSI B30.2 and the TMI-2 Lifting and Handling Program. The critical load bearing components of the CHB trolley and transfer shield, including the canister retention mechanisms, were designed to conform to NUREG-0612, Section 5.1; i.e., the factor of safety of materials used is 5 based on ultimate strength and 3 based on yield strength.

3.3.2 Polar Crane

The reactor building polar crane will be used to lift heavy loads during installation of the early defueling equipment and may be used during early defuelng operations. The capability of the polar crane to lift these loads has been demonstrated by the polar crane load test and the reactor vessel head lift which is addressed in Reference 4. The polar crane will not be used to handle fuel-filled canisters.

3.3.3 Reactor Building Service Crane

The Reactor Building Service Crane (RBSC) is a 5-ton capacity crane which will span the FTC. This crane will be used to handle tools, equipment, shielding and empty canisters. The crane will also be used to handle long-handled tools during fuel canister loading operations. To prevent fuel canister overloading during loading operations, the crane will also be used to weigh open fuel canisters. The crane lifting rigging will include a rigging element to ensure that the equivalent of 4-feet of water will be maintained above any fuel debris during canister weighing or loading operations. The crane rails run the length of the canal on top of the D-rings. This will allow the crane access to the entire length of the canal and to the hatch between elev. 347'-6" and elev. 305'. This crane will provide load handling capability for the numerous small loads which are expected to be handled during early defueling preparations and operations without requiring use of the polar crane. The crane will be load-tested prior to use. The crane will be in compliance with the TMI-2 Lifting and Handling Program and ANSI B30.2.0-1983, Overhead and Gantry Cranes.

3.3.4 Jib Cranes

Two 1-ton capacity jib cranes are located on the shielded work platform to assist operators in manipulating long-handled tools within the slot. The reach of the jib cranes is such that a direct vertical lift can be made at any location along the slot. The jib crane boom can be aligned along the axis of the long-handled tool slot at any angular orientation of the shielded work platform. This configuration allows direct translation of debris to the fuel canister and/or intermediate loading baskets. This jib crane arrangement also provides easy rotation away from the path of the canister transfer mechanism. The maximum lift above the platform with the cranes is 13'-6". With the 1-ton capacity, the jib cranes cannot handle full canisters. The jib cranes will be in compliance with the TMI-2 Lifting and Handling Program, specifically ANSI B30.11-1980.

4.0 Safety Concerns

4.1 General

An evaluation of the activities associated with early defueling identified the following safety issues:

- o criticality control
- o boron dilution
- o release of radioactivity
- o RCS hydrogen evolution
- o pyrophoricity
- o heavy load drop
- o fire protection
- o decay heat removal

Each of these issues is discussed below.

4.2 Criticality Control

4.2.1 Reactor Coolant System

Criticality calculations have been performed to determine the minimum boron concentration required in the RCS to maintain a 1% Δk shutdown margin. A conservative and bounding fuel model was used to determine this boron concentration. The results of this design basis model also account for computer code uncertainty. A boron concentration of 4350 ppm will assure that the reactor core is maintained subcritical with $k_{eff} \leq 0.99$ during all reactor disassembly and defueling operations. This includes the movement of any reactor component, including fuel, within the vessel, whether planned or due to an accident such as a heavy load drop. Reference 5 provides the basis and models used in the selection of a subcritical boron concentration for defueling.

With the introduction of additional materials to the RCS, the potential exists for the RCS reactivity to increase. This could occur if the introduced materials were to act as neutron moderators and/or reflectors or were able to dilute the boron concentration to below 4350 ppm. To ensure that this situation does not occur, a review of materials that may be located on the defueling work platform or handled within the reactor vessel will be made. The effect of these materials on the shutdown margin, assuming they were brought into contact with the fuel, will be determined considering the materials as moderators and reflectors. For the purpose of this evaluation, the RCS boron concentration will be assumed to be 4950 ppm, the lower procedural operating limit. A quantity of the various materials will be determined such that the resultant k_{eff} is less than 0.99 for all credible accidents. For example, the design of the early defueling tooling hydraulic system will have specific features to limit the amount of hydraulic fluids which could potentially be introduced into the RCS such that the k_{eff} criterion is met. Procedural controls will be implemented to ensure that limitations on material type and quantity are not violated.

4.2.2 Canister Handling Operations

Canister handling operations raise two areas of concern regarding criticality. The first deals with the transport of the canisters in the canister transfer shield and the other deals with dewatering of the canisters.

Relating to the first concern, lead in the canister transfer shield and shield collar will act as a reflector for neutrons when a canister filled with core debris is placed inside the shield. Criticality calculations were performed to verify that an adequate shutdown margin ($k_{eff} \leq 0.95$) will be maintained during operations involving the canister transfer shield.

The criticality analyses for the various configurations using the canister transfer shield have been analyzed using KENOIV (Reference 11).

The calculational models for the canister in the transfer shield assume the following conservative conditions:

1. Batch 3 unirradiated fresh fuel only.
2. Enrichment: 2.98 w/o U235 (2.96 w/o + 2σ).
3. No cladding or core structural material.
4. No soluble poison or control materials from the reactor core.
5. Optimal fuel lump size and volume fraction and optimal water moderator density (except in parametric cases for the optimization study).

Insertion studies were performed which concluded that the 100% canister insertion level was the most reactive. The results of the calculations indicated that no poison material is required in the design of the transfer shield as k_{eff} will remain below 0.95. These results are valid for standard, unruptured canisters and for canisters with ruptured internals. Further details are presented in Reference 1.

During dewatering, the criticality concerns can be considered bounded by the results provided in Reference 1; the analyses performed for Reference 1 were completed with optimal fuel/moderator ratios.

4.2.3 Canisters In Storage Racks

Criticality calculations have been performed to demonstrate that the defueling canisters array in the storage racks will maintain a $k_{eff} \leq 0.95$. Further details are presented in Reference 1.

4.3 Boron Dilution

The RCS temperature and chemistry will not be affected significantly during early defueling; hence, boron solubility will remain essentially unchanged. The only way the RCS boron concentration can be changed in an uncontrolled manner during early defueling is by dilution of the RCS coolant with water that is either unborated or borated below 4350 ppm.

Potential sources of this water are the various systems connected to the RCS, incuding the secondary system. Systems which potentially contain coolant with boron concentrations less than 4350 ppm have been identified and isolated to assure that they will not be credible sources of boron dilution. The water in the deep end of the FTC and in fuel pool "A" will be maintained at boron concentrations $\geq$ 4350 ppm. Reference 6 provides an evaluation of all potential dilution paths and the isolation boundaries for these paths. This reference provides dilution detection criteria (e.g., level monitoring, sampling frequency) for static conditions and all modes of water processing during early defueling to ensure that the RCS boron concentration will remain $\geq$ 4350 ppm.

4.4 Release of Radioactivity

The radiological impact of radionuclide release to both the FHB and reactor building atmospheres, as well as to the environment, from early defueling activities have been evaluated. During early defueling, containment integrity will be maintained in accordance with Technical Specifications. All gaseous release pathways to the environment from both the reactor building and FHB will be filtered and monitored, preventing an uncontrolled release of radioactivity to the environment. Radionuclides released to the environment would be in the form of gaseous effluents because early defueling systems and activities do not introduce potential liquid effluent release pathways.

The tasks associated with the preparations for early defueling are not significantly different in their potential for increasing airborne radioactivity than previous work in the reactor building. Consequently, these tasks are not expected to increase the normal background airborne levels currently experienced in the reactor building.

During early defueling activities, radiological releases to the environment will be limited and controlled by filtering and monitoring release pathways. Suspended particulate activity will be removed during filtration and will not be available for release to the environment as a result of early defueling operations. Postulated releases to the environment from either the reactor building or the FHB will not produce offsite doses which exceed allowable limits.

Releases of Krypton-85 will be monitored and an alarm indicating high Krypton-85 levels at the release point will be located in the control room. The need for local monitoring of Krypton-85 in the reactor building during early defueling will be determined by the Radiological Controls Department.

Monitoring at potential release points for alpha-emitting particulates will be conducted to meet the requirements in the TMI-2 Environmental Technical Specifications. Beyond this, the practice will be to analyze the weekly vent samples for gross alpha emitters. Additionally, portable air samples and breathing zone air samples will be routinely analyzed for alpha activity. In the event any of these samples begin to show a significant increase in the frequency of alpha detection (i.e., in excess of levels expected for background) or if the presence of plutonium or other alpha emitters is suspected, the level of analytical scrutiny for the alpha emitters will be increased appropriately to address the situation.

4.4.1 Normal Operations

4.4.1.1 Particulate Releases

All surfaces and equipment, including defueling canisters, which may come in contact with fuel fines will either remain under water during the early defueling of the reactor vessel or be decontaminated (i.e., flushed), as required for radiological control, as they are removed from the water. The canisters and early defueling tools are designed, where practicable, to be easily decontaminated. This will prevent fuel from being removed from the vessel in an uncontrolled manner. An off-gas system is located under the shielded work platform to remove particulates which may become airborne from the RCS during early defueling. Prior to transfer, it is planned that the canisters will be partially dewatered in the reactor vessel to ensure that the catalytic recombiner is uncovered. This action will minimize hydrogen pressure buildup leading to the opening of the canister relief valves. Potentially, some particulates could be released to the FTC or spent fuel pool "A" water. If this occurs, these particulates will either settle out on the bottom of the pools, or they will be entrained in the water which will be monitored for this type of contamination. The DWCS, or an interim water cleanup system, will be used, as necessary, to keep contamination to acceptable levels. Radioactive particulates will be filtered out of the gaseous effluents and will not be available for release in the environment under normal operational conditions.

4.4.1.2 Tritium Releases

Tritium exists primarily as tritiated water. Due to evaporation, some of the tritium in the reactor coolant will become airborne. Although early defueling will not create new sources of tritium in the water, the operation of the off-gas system and the additional heat added by the underwater lighting may increase the evaporation rate of the RCS water. Thus, a slight increase in the rate at which tritium is released to the reactor building atmosphere is possible. However, the off-gas system will dilute the tritium as it is released to the reactor building; therefore, tritium concentrations will not reach unacceptable levels in the reactor building, nor will tritium releases from the reactor building have any unacceptable affect on the health and safety of the public.

4.4.1.3 Krypton-85 Releases

Krypton-85 is an inert gas and will not be removed by the reactor building or FHB filter systems. There is a possibility that the Krypton-85, which is assumed to be in the reactor core, may be released as a result of early defueling activities. An analysis of the offsite doses from postulated Krypton-85 releases to the environment has been made for normal operations. This analysis included all early defueling activities (i.e., canister filling, canister transfer, canister relief valve opening in spent fuel pool "A" or FTC, and dewatering). The range of values for Krypton-85 readily available for release has been estimated to be 0-100 Ci, with the most likely value being approximately 30 Ci (ORNL/TM-8730). For the purposes of this safety evaluation, it is assumed that 100 Ci of Krypton-85 is available for release during early defueling activities. The maximum annual average meteorological dispersion coefficient (X/Q) is 2.27×10^{-6} sec/m^3 and occurs in the SE sector at the site boundary (See Offsite Dose Calculation Manual). Using Regulatory Guide 1.109 methodology and dose conversion factors, the maximum offsite total body dose is 0.0001 mrem/yr, the skin dose is 0.01 mrem/yr, the gamma airdose is 0.0001 mRad/yr, and the Beta air dose is 0.014 mRad/yr. These doses are less than 1% of the 10CFR50, Appendix I dose limits.

4.4.2 Accident Conditions

The possibility of an accident in the course of the early defueling activities is remote. However, environmental releases, even under accident conditions, will be controlled and filtered. The offsite dose consequences from two postulated scenarios were evaluated. These scenarios represent the worst credible accidents. Therefore their offsite dose consequences will be the most severe and all other postulated accidents are expected to result in offsite doses which are lower than those presented. The two postulated accidents are:

a. instantaneous release of all unaccounted for Krypton-85
b. canister drop accident onto a dry canal floor

4.4.2.1 Krypton-85 Release

This accident assumes an instantaneous release of 31,300 Ci of Krypton-85, which represents the unaccounted for Krypton-85 inventory remaining after the controlled reactor building purge of June-July 1980, as decayed to July 1, 1985. The accident meteorological dispersion parameters (X/Q) are 6.1×10^{-4} sec/m^3 for the Site Boundary (See FSAR, Appendix 2D) and 1.1×10^{-4} sec/m^3 for the Low Population Zone (LPZ) (i.e., 3218 m) (See FSAR chapter 15.1.21). Using Regulatory Guide 1.109 methodology and dose conversion factors, the maximum offsite whole body dose is 0.0097 rem (9.7 mrem) and occurs at the Site Boundary. The LPZ whole body dose is 0.0018 rem (1.8 mrem). These doses are less than 1% of the 10CFR100 dose guidelines for accidents.

4.4.2.2 Canister Drop Accident

When a canister is being raised into or lowered from the CTS, a failure of the grapple would result in dropping the canister. This would occur over the reactor vessel, the deep end of the FTC, or fuel pool "A". In all cases the canister would be dropped into water having a boron concentration of 4350 ppm or greater. Therefore, subcriticality would be ensured under any leakage condition. Should the canister leak, any particulate activity would remain in the water and would not be released to the environment. Any Krypton-85 that may be released from the canister would result in offsite doses less than the doses described in Section 4.4.2.1.

As noted in Section 3.3.1, the CTS is designed with diverse means for preventing a canister drop accident while the canister is being transported from the reactor vessel to the deep end of the FTC. Since multiple failures are required for a canister drop accident to occur over the dry portion of the FTC, such an event is considered extremely unlikely. However, should multiple failures occur and a canister is dropped onto the dry portion of the refueling canal, there is the potential for canister leakage.

Potential canister leakage is limited by the following features:

- o Limited space is available for leakage of canister contents due to the small inner diameter of the canister transfer shield. The maximum annular space width is estimated at 1/2 inch. The small clearance between the canister and the shield will provide structural support along the length of the canister and prevent a total circumferential rupture of a canister; therefore, leakage would be expected to occur only at the extreme ends of the canister.

- o Vertical drop tests have shown that the bottom head of the defueling canisters can withstand an impact energy of 51,300 ft-lbs with minor deformation and no observed cracking. This corresponds to an impact velocity of 34 ft/sec on an unyielding surface (Reference 12). This impact load exceeds the calculated impact load for a canister drop in the reactor building. Therefore, the bottom head of the canister would not be expected to crack or rupture.

- If a canister drop were to occur over the dry portion of the canal, by design the lift height of the load is such that the canister will not fall completely out of the transfer shield. This ensures any impact will occur on the canister bottom head.

- The top portion of each canister contains the most likely leakage path. Under normal conditions, the canister vent and drain connections on the upper head may offer a leakage path from the canister during connect/disconnect operations. These connections, Hansen quick disconnects, have integral shutoff valves and will be capped before shipping. If leakage should occur, it is expected to consist of fuel fines, gases, and water vapor. The clearances in the fittings and the connecting tubes will not pass large fuel particles such as fuel pellets.

- The upper closure head nozzles on the canisters are protected by a steel skirt. Under postulated drop accidents, direct impact loads on the canister can be minimized. There is no defined mechanism for dropping something inside the skirt which would directly impact the nozzles. Therefore, leakage from the canister due to a direct impact on the nozzles is not credible.

These design features of the canister and the handling equipment make the potential for a leak very small. It is expected, under design drop conditions, that no leakage will occur. However, for the purposes of this safety evaluation, leakage from the canister is assumed to occur. Since the amount of leakage cannot be quantified, it was conservatively assumed that the entire canister contents would leak onto the dry canal floor.

To assess the offsite exposure consequences of the postulated canister drop, an estimate of the fraction of the canister contents becoming airborne into the reactor building atmosphere and released to the environment was made. To evaluate this fraction, a literature review of experimental and calculated suspension factors was performed. Only suspended particles were assumed to be available for offsite release. The results of the literature review are presented below.

Experiments have been performed (reference 14) to determine what weight percent of a dry powder will become airborne after a spill. The powders used in the experiments were dry titanium dioxide and depleted uranium dioxide. These powders were released in a free fall spill through static air. Particle sizes ranged up to 75 microns with approximately 98% of the powders having particle sizes 20 microns or less. The results of these experiments have suggested that 0.12 w/o of the particles will become airborne during a spill. The study also showed that particles less than 10 microns in diameter accounted for approximately 40% of the airborne mass. This tends to suggest that the larger particles have less of a tendency to become airborne than the smaller ones. Therefore, although these tests did not cover the entire range of particle sizes of interest, they do cover the lower end of the range where particles have the greatest tendency to become airborne.

Additional data (Reference 15) tends to confirm the 0.12 w/o estimate referenced above. One micron particles on a stainless steel surface, exposed to a 20 mph wind, had 0.29 w/o of the total mass become airborne. The airflow inside the reactor building and FHB is much less than 20 mph; therefore, a more applicable experiment may be one with one micron particles on a stainless steel surface in a 2.5 mph wind. For this case, 0.075 w/o of the total mass became airborne. Following a spill of dry powder (i.e., particles $\leq$ 75 microns), a reasonable estimate of the percentage of the powder becoming airborne is 0.12 w/o.

Of the three types of canisters, only the filter canister would be expected to contain mostly fuel fines of the size that would exhibit a tendency to become airborne and result in a 0.12 w/o airborne fraction. These canisters are expected to contain fines in the range of 140 microns down to 0.5 microns. The fuel canister is expected to contain large pieces of core debris and the knockout canister is expected to contain debris ranging in size from 140 microns up to whole fuel pellets. The fines in the filter canisters are not free but are contained within the filter media and would not become as readily airborne as dry powder. Additionally, at the time the canister is over the dry portion of the refueling canal, the canister contents are wet. Consequently, they would not be expected to exhibit as great a tendency to become airborne.

The 0.12 w/o airborne release fraction for dry powders (i.e., particles $\leq$ 75 microns) discussed above was used to assess the offsite exposure consequences resulting from the postulated canister drop. This conservative assumption bounds any possible dose consequences from a canister drop accident.

The offsite doses resulting from the postulated canister drop in the dry portion of the refueling canal were evaluated using the assumptions listed in Table 4.4-1, the dose conversion factors listed in NUREG-0172 (Reference 16) and the organ dose calculation methodology consistent with Regulatory Guide 1.109 (Reference 17). The whole body dose calculation methodology is consistent with Regulatory Guide 1.4. The radionuclide inventory of the canister was based on 1% of the core inventory as given in GEND INF-019 (Reference 18), decayed to July 1, 1985, applying a peaking factor of 1.9. Containment integrity was assumed to be maintained and a HEPA filter efficiency of 99% was used. Table 4.4-3 presents the offsite doses for the whole body, thyroid and bone.

The bone dose is presented since it was determined to be the critical organ. The critical organ determination was made based on comparison of dose conversion factors for several organs, including the lung, kidney, liver and gastrointestinal tract, for the distribution of radionuclides available for release.

Table 4.4-1

Assumptions Used to Assess Offsite Doses from Canister Drop Accident

Canister Inventory	Table 4.4-2
Canister Contents Released	100%
Canister Contents Airborne - Particulates	0.12 w/o
Canister Contents Airborne - Gases	100%
HEPA Filtration Efficiency	99%
Radionuclides Released to Environment	Table 4.4-2
Accident Condition X/Q	
0-1 Hour Exclusion Boundary	6.1 E-4 sec/m^3
0-8 Hour Low Population Zone	1.1 E-4 sec/m^3

Table 4.4-2

Radionuclide Inventories Used in Canister Drop Accident Analysis

Radionuclide	Activity (Ci) In Canister	Activity (Ci) Released to Environment
H-3	5.76 E+1	5.76 E+1
Kr-85	1.31 E+3	1.31 E+3
Sr-90	1.23 E+4	1.48 E-1
Y-90	1.23 E+4	1.48 E-1
Ru-106	8.49 E+2	1.02 E-2
Sb-125	4.60 E+2	5.52 E-3
Te-125m	1.12 E+2	1.34 E-3
Cs-134	4.62 E+2	5.54 E-3
Cs-137	1.38 E+4	1.66 E-1
Ba-137m	1.31 E+4	1.57 E-1
Ce-144	1.85 E+3	2.22 E-2
Pr-144	1.85 E+3	2.22 E-2
Pm-147	9.50 E+3	1.14 E-1
Sm-151	2.01 E+2	2.41 E-3
Eu-154	9.77 E+1	1.17 E-3
Eu-155	2.51 E+2	3.01 E-3
U-238	5.13 E-1	6.16 E-6
Pu-238	1.44 E+1	1.73 E-4
Pu-239	1.71 E+2	2.05 E-3
Pu-240	4.56 E+1	5.47 E-4
Pu-241	2.95 E+3	3.54 E-2
Am-241	3.61 E+1	4.33 E-4

Table 4.4-3

Offsite Dose Resulting from Postulated Canister
Drop on the Dry Portion of Refueling Canal

Organ	Dose (Rem) Exclusion Boundary	Dose (Rem) Low Population Zone
Whole Body	4.3 E-4	7.7 E-5
Thyroid	1.9 E-3	3.5 E-4
Bone	2.96	0.53

4.5 Hydrogen Evolution

During early defueling activities, the reactor vessel will be covered by the shielded work platform. An off-gas system has been designed to provide an air in-flow through the top of the work platform. This will dilute gases that are released or evolved during early defueling before they are released into the reactor building. Off-gases, including hydrogen, are controlled by the off-gas treatment system. Any hydrogen evolved during early defueling is diluted by the off-gas treatment system and will not reach a combustible concentration in the reactor building.

While the canisters are being transported in the canister transfer shield or are in storage in either the FTC or spent fuel pool "A", radiolytic generation of hydrogen may occur within the canister. Thus, hydrogen could be released through opening of a canister relief device. However, partial dewatering in the reactor vessel and dewatering in spent fuel pool "A" will minimize the potential for hydrogen pressurization by exposing the hydrogen to the catalytic recombiner.

In the case of the stored canisters, this hydrogen will be released to either the FHB or reactor building depending on the storage location. The hydrogen will be diluted by the large surrounding atmospheres of these buildings. Consequently a combustible concentration of hydrogen will not be reached in either building due to the release of hydrogen from the canisters.

The combustion of hydrogen within the canister transfer shield is not expected to occur. The canisters are to be in the transfer shield for short periods of time (i.e., during transfer from the reactor vessel to the deep end of the refueling canal or during canister handling in the FHB). However, it is recognized that a canister could be in a transfer shield for extended periods of time. Even if a hydrogen generation rate within a single canister was postulated such that it was sufficient to cause a canister relief device to discharge to the transfer shield, there are no ignition sources inherent in the design or operation of the transfer shield and the top of the transfer shield is vented (i.e., approximate vent area of 15 square inches) to the surrounding building volume. Consequently, since it is planned that the canisters will be partially dewatered prior to lifting into the transfer shield and in view of the provisions of the shield design, it is considered very unlikely that a hydrogen combustion incident can occur in the canister transfer shield. However, should hydrogen combustion occur, the resultant loadings on the canister and the transfer shield are expected to be within the structural capabilities of the canister and the transfer shield.

4.6 Pyrophoricity

The NRC has evaluated the potential for zirconium hydride fires (Reference 20; section 13.10.1.3). This evaluation assumed that operations relating to early defueling would be conducted with water coverage. It was concluded that zirconium hydride would not ignite under water. Since early defueling operations will be conducted under water, GPU concurs that there is no potential for a pyrophoric zirconium reaction in the reactor vessel during early defueling.

After the canisters are loaded with debris, the dewatering operation in the FHB will remove enough water from the canister to assure that more than one-half of the catalytic recombiner is not submerged. When the catalyst is no longer submerged, it will function to control the hydrogen and oxygen concentrations in the canister. An inert cover gas (e.g., argon) will blanket the core debris in the canister after the dewatering process.

The concern over pyrophoric materials is presently focused on the potential for metallic zircalloy and zirconium hydride fines existing in the dewatered canisters. The manner in which the fuel deteriorated during the accident makes the presence of these species, in a pyrophoric form, highly unlikely in the present configuration of the core rubble bed. Zircalloy, being a ductile metal even after irradiation, would not break up into small particles under the high temperature steam environment of the TMI-2 accident. Rather, the material oxidizes, and it is the oxide which breaks up as a consequence of thermal shock or abrasion. However, during the early defueling process, it is possible, as a result of cutting operations, that fresh (i.e., unoxidized) metal surfaces, including small chips and fines, could be created.

Considerable analyses have been conducted since the pyrophoric concern was initially raised and are summarized in Reference 10. The analyses indicate that three conditions must exist to initiate and maintain a pyrophoric reaction:

(1) The pyrophoric material must have a high surface to volume ratio of the nature of powder. Experience indicates that moist zirconium fines of less than 10 microns will burn. However, existing analysis of core debris indicates only about 1.5 % of the particulate matter is less than 45 microns. The early defueling activities are not likely to generate significant additional quantities of fines in the size range of concern.

(2) The pyrophoric material must exist in an oxygen depleted environment and then be suddenly exposed to oxygen. The surface of the core pyrophoric material has been exposed to oxygen in the water since the accident. Thus, oxidation that has already occurred would limit a pyrophoric reaction to material that is

freshly exposed. The early defueling process is not likely to expose significant quantities of debris in the size range specified in item (1) above. Further, any additional exposure of pyrophoric material due to the early defueling activities would initially be underwater, where oxidation would again occur at some rate.

(3) The oxidation rate must exceed the heat transfer rate to the surrounding environment. The oxidized debris that will be mixed with any pyrophoric material acts as a diluent and minimizes the potential for ignition and propagation.

In addition to the above considerations, tests have been conducted on a sample of mater[illegible] removed from the plenum to determine its pyrophoricity. Attempts were made to "pilot ignite" the subject material by conducting a spark test and a flame test. The results found "no pyrophoric characteristic" for the material tested.

In summary, theoretical analysis and experimental data indicate that the characteristics of the material currently in the reactor vessel, or as it may be modified during early defueling, are such that it is highly unlikely to sustain a pyrophoric reaction. This conclusion is not dependent on continued submergence of the material in water. Thus, it is not considered reasonable to postulate a pyrophoric reaction of exposed fuel debris as a significant driving force for radionuclide transport.

4.7 Heavy Load Drop

The NRC Generic Letter on the control of heavy loads requires licensees to address the guidelines of NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

Section 2.3 of the letter requires information sufficient to demonstrate adequate measures have been taken to ensure that, in the vicinity of the reactor core, either the likelihood of a load drop which might damage the fuel is extremely small or the estimated consequences of such a drop will not exceed the limits set by the evaluation criteria I-III of NUREG-0612, Section 5.1. Section 2.4 of the letter requires information sufficient to demonstrate that, in the vicinity of equipment or components required for safe reactor shutdown and decay heat removal, either the likelihood of a load drop which might prevent safe reactor shutdown or prohibit continued decay heat removal is extremely small or that damage to such equipment will be limited so as not to result in the loss of these safety-related functions (Criterion IV).

Heavy load handling in both the reactor building and FHB, excluding heavy loads handled over the reactor vessel, is addressed in Reference 8. Heavy loads handled over the reactor vessel are addressed in Reference 19.

4.8 Fire Protection

Fire Protection during the early defueling activities will be provided in accordance with the requirements of the Fire Protection Program Evaluation, Revision 1, and TMI-2 Administrative Procedure 4000-ADM-3680.02, Control of Combustible Materials. Existing fire detection or fire extinguishing equipment in the reactor building will be available for early defueling. This will ensure that the potential for and consequences of a fire are minimized.

4.9 Decay Heat Removal

During early defueling activities the water level in the RCS will be at an elevation approximately 5 feet above the vessel flange (i.e., approximately 327'-6"). An analysis of decay heat removal ability with the RCS water level at elevation 321'-6" has been performed (Reference 4). The results of this conservative analysis show that loss to ambient cooling will maintain the RCS bulk temperature at less than 170°F during early defueling activities. The video system lighting will add additional heat to the reactor vessel water; however, it is not considered a safety problem, as the lights can be turned off to eliminate heat input should the water temperature increase to unacceptable levels. The RCS water temperature will be monitored during early defueling in accordance with TMI-2 Technical Specifications and the Recovery Operations Plan.

5.0 Radiological Considerations

5.1 External Exposures

All individuals entering the reactor building will be monitored for external radiation exposures according to established radiological controls procedures. All external radiation exposures will be maintained within the dose equivalent limits established in 10CFR20. All personnel exposures will be maintained as low as is reasonably achievable (ALARA). Administrative dose limits are applied according to established procedures to ensure that the 10CFR20 limits are not exceeded. Extremity monitoring will be performed, as needed, according to existing radiological controls procedures.

Radiation exposure rates inside the reactor building will be monitored during early defueling operations. The Radiological Controls Department will determine the requirements for radiation monitoring for personnel protection during early defueling. A multi-channel radiation monitoring system has been installed. This system consists of multiple area radiation monitors which will be positioned at strategic locations throughout the reactor building. Radiological Controls Department personnel will use this system and/or other instrumentation to continuously monitor dose rates in the reactor building during early defueling activities.

5.2 Internal Exposures

All individuals entering the reactor building will be monitored for internal radiation exposures according to established procedures. This monitoring may be accomplished by routine breathing zone air sampling and periodic whole body counting.

All exposures to airborne radioactivity will be maintained as low as is reasonably achievable and within the limits established by 10CFR20. Airborne radioactivity in work areas will be monitored according to established procedures. Air sampling for particulates will be performed using such devices as lapel samples and grab samples. Tritium grab samples will be taken as required according to established procedures.

Respiratory protection has been used to minimize the uptake and deposition of airborne radioactivity in the body. The use of respiratory protection devices can, by reducing uptakes of radioactive materials, result in overall dose savings (internal and external); however, if they impede work, total dose can increase by causing an elevated external dose. The Radiological Controls Department shall determine, via the prework radiological review process, if the use of respiratory devices is ALARA for a particular task.

This review will include an examination of the current radiological conditions in the work area, an assessment of the potential for the task or other concurrent tasks to perturb the radiological conditions and a review of the results of previous airborne activity measurements in the work area for similar tasks.

An estimate of airborne radioactivity to be encountered by individuals performing early defueling activities was derived from the continuous air monitors positioned on the IIF during plenum inspection activities to monitor radioactive particulates and tritium grab samples taken in the reactor building prior to, during and following reactor vessel head removal. These levels are given below:

Isotope	Reactor Building Concentration (μCi/cc)
Cs-137	3.5 E-10
Cs-134	1.3 E-11
Sr-90	3.7 E-11
H-3	1.1 E-7
Gross alpha	6.2 E-14

Breathing zone samples from workers on the IIF platform during plenum inspection and end fitting separation activities showed a gross beta-gamma activity equivalent to an airborne concentration of 8.4 E-10 μCi/ml for all beta and gamma emitting nuclides. Alpha activity was below the minimum detectable for the breathing zone samples. The planned activities are not expected to increase the tritium or particulate levels inside the reactor building (see Section 4.4). The additional release of tritium to the reactor building atmosphere, due to evaporation of the reactor coolant, is not expected to significantly increase the tritium level in the reactor building atmosphere because of the low tritium concentration in the reactor coolant (i.e., approximately 0.03 μCi/ml). Equipment and defueling canisters being removed from the reactor vessel will be sprayed down, as necessary, to remove particulates and enhance radiological control. Therefore, the amount of particulates resulting from these activities with the potential for becoming airborne is not expected to significantly increase the particulate concentration in the reactor building atmosphere. The opening of the canister relief valves while the canisters are located in the deep end of the FTC or spent fuel pool "A" may cause particulates to be released to the surrounding water. However, these particulates will be entrained in the water and their potential for becoming airborne is minimal.

5.3 Exposure Estimates

Due to the nature and duration of the early defueling activities, the development of detailed man-rem estimates for early defueling is impractical at this time. As the early defueling plans become better established and the man-hour estimates are well defined, a refined man-rem estimate will be developed which will be available prior to the start of early defueling. However, the comparative collective personnel radiation exposure to workers during different defueling options, using non-specific manhour assumptions, has been evaluated. Doses were evaluated on the basis of target dose rates for work locations in the TMI-2 reactor building and estimated work hours. Based on recent dose rate reductions in the reactor building, proposed dose reduction options and recommended dose rate targets, the work locations were assigned dose rates as listed in the table below:

Location	Mean Millirem/Manhour (Dose Rate Target)
305' elevation	110
347' elevation	55
Canister handling bridge (in containment)	30
Canister handling bridge (in FHB)	2
Defueling platform	15

The dose rate targets are considered to be reasonably achievable. By the start of early defueling, the expected mean millirem/manhour at specific areas of the 305' and 347'-6" elevation are 85 and 45 millirem/hour respectively. These dose rate decreases will be achieved by floor scabbling, decontamination and source shielding. However, man-rem estimates were based on the dose rate targets.

The dose rate target for the canister handling bridge inside containment is based on decontamination of the bridge, the effect of dose reduction on the 347' elevation, decontamination of the FTC liner, shielding of the operator area on the bridge and use of the canister transfer shield.

The work platform target can be achieved by shielding the platform as described, maintaining water activity at the defueling water cleanup system design specification and decontaminating the FTC liner.

The FHB dose rate is based on current area dose rates which are not expected to change significantly. An evaluation has shown that the canisters will be well shielded by the pool water and will have minimal impact on dose rates other than at locations just above the surface of the spent fuel pool.

The comparative evaluation indicated that the defueling option selected would take approximately 700 man-rem. These estimates should not be construed as actual man-rem for this defueling option; the actual manhours required are not available at this time. However, this comparison did indicate that the defueling option selected was the most man-rem efficient option of the manual techniques reviewed.

Table 5.3-1 shows the estimated manhours and associated man-rem for this defueling option. These estimates include both early and bulk defueling activities. Included in the manhours are installation, operation, maintenance, decontamination and removal of the defueling equipment. The manhours are broken into locations corresponding to the target dose rate locations. Detailed exposure estimates will be developed on a task-by-task basis as part of the ALARA review of in-containment work and to insure that each activity is performed in a man-rem effective manner.

Table 5.3-1

Man Rem Estimates for Defueling Option Selected

	Manhours	Manrem
Preparations		
305'	255	28
347'	2300	127
Canister handling bridge (containment)	440	13
Defueling platform	1245	19
Canister handling bridge (FHB)	-	-
Operations		
305'	-	-
347'	-	-
Canister handling bridge (containment)	2330	70
Defueling platform	14475	217
Canister handling bridge (FHB)	3265	7
Maintenance		
305'	-	-
347'	-	-
Canister handling bridge (containment)	155	5
Defueling platform	1370	21
Canister handling bridge (FHB)	435	1
Decontamination and Removal		
305'	252	28
347'	2264	125
Canister handling bridge (containment)	600	18
Defueling platform	1950	29
Canister handling bridge (FHB)	-	-
Total	31336	708

5.4 Measures Taken to Reduce Occupational Exposure

5.4.1 ALARA

During the design of the early defueling tools and the planning of early defueling activities, the principles of ALARA were considered. In studying the alternatives for early defueling operations and equipment, ALARA was considered on a judgmental basis. Specific design changes were made on the tools and equipment to enhance performance of certain operations. Operational sequences were reviewed and changed to allow performance of work in lower radiation areas, where possible.

The objective of minimizing occupational exposure has been a major goal in the planning and preparation of all activities in the containment. Protective clothing and respirators will be used as required to reduce the potential for external contamination and internal exposure of personnel.

Extensive planning of tasks to be conducted in a radiation field and training of personnel will reduce the time needed to complete a task. The higher radiation areas will be identified to personnel and the work will be structured to avoid these areas to the extent practical. Practice sessions will be utilized, as necessary, to ensure that personnel understand their assignments prior to entering the reactor building. Planning and training are proven methods of ensuring that personnel are properly prepared to conduct the assigned tasks expeditiously.

Execution of individual tasks are maintained ALARA by a detailed pre-task radiological review by Radiological Engineering and mock-up training. The need for mock-up training will be determined on a case-by-case basis. A detailed mock-up, the Defueling Test Assembly (DTA), simulating the configuration and orientation of the rotatable work platform, vacuum system, T-slot, working slot, handrails, single canister support bracket, canister positioning system and the debris bed will be utilized. The long-handled tools will be representative of the actual tool lengths to be used. Extensive training of workers on the DTA and other mock-ups will familiarize the workers with the tasks to be performed. This training will result in increased worker efficiency; less in-containment time and personnel exposure will be required.

Tooling has been designed with the intent of keeping radiation exposures ALARA. Because of the large component sizes, the equipment must be assembled inside the reactor building. The components were designed for rapid assembly. The early defueling tools will be assembled and tested outside of the reactor building, then disassembled during the mock-up training. Shielding is provided where required and all tools are fabricated of stainless steel and/or aluminum with smooth inside and outside surfaces and no blind holes for ease of decontamination. Flushing and draining holes are provided, where required, with flushing capability from the top of the tooling. Spray rings are located under the rotatable work platform to provide a washdown capability for canister removal areas and the long-handled tool slot.

Savings of in-containment time is realized by simplifying the communication and control required during early defueling by having a centralized operating and control station. This centralized station allows all operations to be conducted with constant monitoring and coordination. To minimize radiation exposure to personnel and provide control functions near the early defueling operations, the control station is located on the auxiliary work platform in the south end of the canal.

5.4.2 Early Defueling Systems Design Radiation Shielding

A series of calculations was performed to evaluate the dose rates to personnel considering the various components which could constitute radiation sources during early defueling and considering various shielding configurations to minimize the radiological impact of these sources. The analyses were performed to ensure that early defueling systems were adequately designed to minimize personnel occupational exposures based on theoretical or design basis source terms. The shielding configurations and dose rates in this section are provided as an estimate of conditions which may exist during early defueling. Operations during early defueling will be governed by actual measured radiation dose rates. Adequate precautions, such as shielding or personnel relocation, will be used as necessary to ensure worker safety and to minimize collective personnel exposures.

The primary sources were identified as: loaded defueling canisters in the reactor vessel, the radioactivity in the reactor coolant, and the particulates in the vacuum system components. Source terms were developed for these items to use throughout the shielding review program.

- o Canister source term. Source terms were developed for the three different types of canisters: fuel, knockout and filter canisters. The source term in general was calculated by assuming that the total core inventory of fission products was distributed throughout the 93.1 metric tonnes of UO_2 which comprised the total initial fuel load. No retention of noble gases was assumed. The core inventory was decayed to July 1, 1985. The core debris was assumed to be composed of UO_2, zircaloy and stainless steel in the same proportion as originally present in the core region. Each canister was assumed to contain its maximum permitted weight of fuel debris. The weight of fuel debris is limited by the maximum allowable weight of a loaded, fully dewatered canister. The fuel debris was assumed to contain the average core fission product specific activity, and the debris was assumed to be distributed homogeneously throughout the usable canister volume.

- o Reactor Coolant Source Term. During early defueling, the reactor coolant in the IIF and reactor vessel will contain dissolved radioactive materials. The water level was assumed to be at 327'-6". The dose rate contribution from the reactor coolant is based on a water concentration of 0.02 μCi/ml cesium-137. The sensitivity of the dose rate to early defueling operators from increased water concentrations also was investigated.

- o Fines/Debris Vacuum System Source Terms. The fines/debris vacuum system is to be used during early defueling to remove fines/debris from the core region by vacuuming. The source terms for these components were derived from the system design bases. The water upstream of the knockout canister was assumed to contain 8000 ppm fuel debris, which is based on the expected loading rate of the knockout canister. Components downstream of the knockout canister were assumed to contain water with 1400 ppm solid fuel debris suspended in it.

To provide an additional confidence level to the shielding analyses performed for the early defueling systems, a structured program of independent verification was instituted. Shielding requirements for major early defueling components (e.g., rotatable platform, stationary platform, canister shield collars and canister shield plugs) were calculated independently by GPU Nuclear and the hardware vendor.

Calculated dose rates were compared during shielding design and any significant differences were investigated. A simplified standard calculation test case was performed to compare computer code results. Results of the independent verification program indicated good agreement between the two analytical programs.

Many separate calculations were performed for various sources, shielding components, and operational scenarios. From these calculations came the final design recommendations for early defueling systems shielding components. Dose rates were calculated for normal early defueling operations. Note that all dose rates are from the sources described and do not include background radiation.

For purposes of these analyses, normal early defueling operations were assumed to entail manual defueling with the entire working slot in the rotatable platform unshielded. During normal operations it was assumed that all areas of the work platforms were occupied and that the north end canal floor area (i.e., between the support structure and the refueling canal dam) was accessible. However, access under the auxiliary work platforms on the north and south end of the FTC was not considered part of normal operations. The individual shielding components described below were evaluated for their impact on dose rates in work areas.

- o Shielded Work Platform. The shielded work platform was designed to limit dose rates to operators to less than 1 millirem/hour from sources in the reactor vessel during normal early defueling operations. These sources included 5 fully loaded fuel canisters in the canister positioning system (CPS), 2 fully loaded filter canisters, vacuum system components, and the reactor coolant. The shielding requirement for the platform was determined to be 6-inch thick steel.

- o Vertical Support Structure Shielding. The north end of the support structure will have vertical shielding attached which will extend from the canal floor to the platform elevation. This vertical shield will essentially span the width of the refueling canal and will be 2-inch thick steel.

- o Service Platform. The service platform was designed to limit dose rates to operators to less than 1 millirem/hour from sources in the vessel during normal early defueling operations. The platform is to be constructed of 3-inch thick steel plate.

- o Auxiliary Work Platforms. The auxiliary work platform on the south end of the canal will be field erected. It is constructed of 1 to 2-inch thick steel shielding or equivalent. The dose rates to operators will be limited to approximately 1 millirem/hour from sources in the vessel during normal early defueling operations. The auxiliary work platform on the north end of the canal is not a shielded structure. However, the 2-inch thick steel vertical support structure shield effectively reduces dose rates from sources in the reactor vessel to approximately 1 millirem/hour during normal operations.

- o Unshielded Sources. During normal operations, workers will be positioned along the open slot in the rotatable platform. The maximum dose rate from the reactor coolant to workers at the slot is 2.6 millirem/hour, assuming a water concentration of 0.02 μCi/ml cesium-137. Dose rates from the water would be somewhat higher if the entire working slot and T-slot were unshielded. The maximum dose rate from the vacuum system components along the open slot was calculated as approximately 1.3 millirem/hour. Vacuum system components considered were the pump and piping; excluded from considerations were the filter canisters. In the event that the shield plate directly above the vacuum pump must be removed for maintenance, the dose rate from the vacuum system components will be less than 50 millirem/hour below the platform elevation.

- o Canister Shielding. It was considered an ALARA goal to limit the dose rate from the loaded canisters in the vessel (i.e., including filter, knockout and fuel canisters) to a contribution approximately equivalent to that from the reactor coolant. To accomplish this goal, a 4-inch thick steel shield plug will be placed on all fuel canisters after they are loaded and placed in the topmost canister position. In addition, a 1-inch thick steel collar will be incorporated into the canister sleeve design in the CPS. The filter canister support structure design incorporates vertical shielding as well. This steel shield collar will extend approximately 2 feet down from the top of the canister source region. These precautions will limit the maximum contribution from the canisters to less than 5.5 millirem/hour at the open slot.

- o Sources in the Canal Deep End. Dose rates at the work platform due to sources in the deep end of the FTC were evaluated. These sources include the plenum in its storage location, loaded fuel and filter canisters in the storage racks and a postulated deep end water concentration of 0.02 μCi/ml cesium-137. The total dose rate from these sources to operators on the work platform was less than 2 millirem/hour. This was a very conservative estimate since the primary dose contributor (1.3 millirem/hour) was the water in the deep end which was assumed to have an activity of 0.02 μCi/ml cesium-137. The activity in this water should be maintained at a much lower concentration since there is no large source of contamination in the deep end during normal early defueling operations. This estimated dose rate also does not take credit for the shielding effect of the dam which is 3/8 inch thick steel.

It was considered desirable to design the early defueling systems such that workers could remain on the work platform during canister transfers from the CPS. Therefore, special shielding components were designed to maintain reasonably low dose rates on the work platform during transfer operations.

- Canister Transfer Shield and Shield Collar. The canister transfer shield (CTS) was designed to be supported from the canister handling bridge. The CTS includes a 9 foot long sliding shield collar which will be used to ensure the entire canister is shielded during all transfer operations. The CTS is designed to provide a 2-1/2 inch thick lead shield along the entire canister length and a collar to provide an additional minimum of 1-1/2 inch thick lead shielding.

  When the CTS is positioned over the work platform, the 4-inch thick lead shielding extends approximately 9-feet above the work platform. Dose rates to operators on the work platform will be approximately 16 millirem/hour at a location 2-feet from the CTS. The CTS also incorporates a 3-inch thick lead shield plug above the top of the canister. Dose rates to the operators on the canister handling bridge will be less than 9 millirem/hour from the canister being transferred. This dose rate does not take credit for any structural material in the bridge. The contribution to the dose rate to workers on the 347'-6" elevation around the canal will be less than 10 millirem/hour. This dose rate does not take credit for the extra lead shielding provided by the collar.

  The underside of the CTS is not shielded and no access to the north end canal floor will be permitted during canister transfers. Dose rates to operators working on the north end auxiliary work platform will be comparable to those present when the CTS is positioned over the early defueling work platform (i.e., approximately 16 millirem/hour at a location 2 feet from the CTS).

- Shield Boots. A vertical shield extending down from the platform into the reactor coolant is used during transfers to shield the canister between the water and the CTS. This "boot" structure is positioned at the primary canister transfer location for the CPS and also surrounds the two filter canister locations. These structures will be constructed of 5-inch thick steel and will extend 2-feet into the reactor coolant. This will limit whole body dose rates along the open slot to 27 millirem/ hour. Extremity dose rates may approximate 150 millirem/hour on contact with the platform surface. Due to interferences with CPS or vacuum system piping, some sections of the boot may be less than 5-inch thick steel or may not extend the full 2-feet into the water. These cases will be evaluated individually to ensure that dose rates will be maintained in the ranges calculated for the complete boot. These dose rates are calculated for canister transfer from the CTS. Transfers of filter canisters would result in similar dose rates.

- Single Canister Support Bracket. During early defueling, before the installation of the CPS, a single canister support bracket (SCSB) may be used. The SCSB consists of a single canister suspended from the rotatable platform in the center of the slot. The SCSB can be located anywhere along the slot within the diameter of the core former. During normal operations there will be a dose rate increase along the open slot of less than 1 millirem/hour since there is greater than 6 feet of water shielding above the canister top. Shielding calculations were performed assuming that during transfers from the SCSB the working slot will be closed with shielded panels during each transfer from SCSB. These shielded panels will be 6 inch thick steel and will be designed so that there is no radiation streaming between the panels and the CTS. Assuming that the top of the canister is at the platform elevation, dose rates on the rotatable platform would be less than 12 millirem/hour. Assuming that the canister transfer is from the alternate canister transfer position for the CPS, dose rates at the south end auxiliary work platform would be less than 100 millirem/hour. Dose rates on the auxiliary work platform at the south end of the FTC would be lower if the SCSB were positioned in the north half of the reactor vessel during transfer. Since the SCSB is a limited use item to be used primarily until the CPS is installed, this operational constraint was considered to provide adequate protection of personnel.

The analytical approach taken in the shielding program was to provide a reasonably accurate assessment of the worst case radiological impact expected during early defueling. The canister source terms were based on 100% fission product retention (i.e., excluding noble gases) and the maximum weight of fuel debris permitted to be loaded into a canister. For normal operations, the maximum number of canisters was assumed to be present. No shielding credit was taken for the massive structural components of the support structure or the work platforms. The assumption of 100% fission product retention is very conservative. Cesium-137 is the predominant gamma dose rate contributor at TMI-2. Greater than 40% of the core inventory of cesium-137 has been removed by water processing. It is clear that much of the debris has a significantly reduced cesium source and that many canisters will produce dose rates lower than those predicted.

The reactor coolant source term reflected the assumed design basis capability of the DWCS to remove soluble radioactive materials. Early defueling with operators working directly over the open slot will be constrained by the dose rates present and will be temporarily interrupted, if necessary, if the reactor coolant concentrations cause dose rates to increase above acceptable levels. In addressing the platform and support structure shield thicknesses, however, it was considered that much higher concentrations could exist and personnel access would still be desirable. Dose rates were calculated for much higher concentrations. These results (see below) show that even with upset water conditions, personnel access to the platform and to the north end canal area will not be overly restricted.

- o Dose rates 18 inches above the work platform assuming the working slot and T-slot are closed (i.e., shielded with 6 inch thick steel). The canister, vacuum system and water source are considered.

	Water Concentration (Cesium-137)		
	0.02 μCi/ml	0.2 μCi/ml	1.0 μCi/ml
Dose Rates (mrem/hr)	0.01	0.03	0.13

- o Maximum dose rate 18 inches above the work platform assuming the operator is at the open slot. Only the water source is considered.

	Water Concentration (Cesium-137)		
	0.02 μCi/ml	0.2 μCi/ml	1.0 μCi/ml
Dose Rates (mrem/hr)	6	26	130

- Dose rates 18 inches above the work platform assuming the slot is open and the operator is at various locations on the platform. The water source, vacuum system and canisters are considered.

	Water Concentration (Cesium-137)		
	0.02 μCi/ml	0.2 μCi/ml	1.0 μCi/ml
Dose Rates (mrem/hr)			
Above center of CPS	0.03	0.16	0.73
Above filter canister*	0.59	5.8	28.9
Above vacuum pump	0.01	0.03	0.13
At canal floor, north of support structure	12.9	14.9	23.9

*Dose rate increases at higher concentrations are primarily due to radiation streaming from the water through the open working slot.

To ensure that the early defueling systems are adequately designed to permit achievement of the dose rate goals set for these activities, it was a design basis that the dose rate contribution from the early defueling systems be limited to the following values:

- 12 millirem/hour at the open slot
- 2 millirem/hour on the work platform
- 10 millirem/hour on the canister handling bridge during canister transfer
- 15 millirem/hour at a location 7 feet from the CTS during canister transfer

The evaluations performed show that these dose rate increases are achievable based on current designs. In fact, designs were optimized to provide for even lower dose rate increases. Based on the evaluations performed, the maximum increases expected during normal operations from sources in the vessel are:

- 9.4 millirem/hour at the open slot
- <1 millirem/hour on the work platform
- 8.8 millirem/hour on the canister handling bridge during canister transfer
- 6 millirem/hour at a location 7 feet from the CTS during canister transfer

A significant effort has been expended throughout the design process to ensure that early defueling systems will provide adequate radiation protection for operators and will result in the lowest reasonably achievable collective dose for early defueling operations.

6.0 Impact of Early Defueling on Plant Activities

The major impact of early defueling on plant activities will be the effect of fuel movement in Unit 2 on operations in Unit 1. A condition has been imposed by the Atomic Safety and Licensing Board on TMI Unit 1 restart that either (1) the effects of TMI-2 fuel movement on TMI-1 personnel in the FHB have been addressed and the NRC has approved the applicable procedures, or (2) work in the Unit 1 area of the FHB will be suspended during Unit 2 fuel movement.

Due to the environmental barrier which isolates the Unit 1 auxiliary building from the FHB areas of Unit 1 and Unit 2, the only Unit 1 area that potentially would be affected by Unit 2 fuel movement is the Unit 1 FHB area. The worst case early defueling accident in the FHB, within the scope of this safety evaluation, is a canister drop. An evaluation has been performed to determine the radiological consequences due to a canister drop in the FHB. The scope of this safety evaluation ends with the storage of the canisters in the racks; therefore, all canister movements in the FHB covered by this document will be made over spent fuel pool "A". Consequently, any postulated canister drops would be into the pool. Even though the design specifications of the canister allow for canister leakage it is not expected that leakage would result from such a drop. If any leakage were to occur, it would occur underwater; therefore, there will be no airborne releases from a canister drop and any debris that is released into the water would be shielded by the pool water so that the contribution to the area dose rate would be negligible. Ultimately the debris would be cleaned up by the DWCS or an alternate cleanup system. Releases of Krypton-85 will be within acceptable limits as demonstrated in section 4.4.2.1.

It can therefore be concluded that early defueling operations in Unit 2 will not affect personnel in Unit 1.

7.0 10 CFR 50.59 Evaluation

10 CFR 50, Paragraph 50.59, permits the holder of an operating license to make changes to the facility or perform a test or experiment, provided the change, test, or experiment is determined not to be an unreviewed safety question and does not involve a modification of the plant technical specifications.

10 CFR 50, Paragraph 50.59 states a proposed change involves an unreviewed safety question if:

a) The probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report may be increased; or

b) The possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report may be created; or

c) The margin of safety, as defined in the basis for any technical specification, is reduced.

Although there are notable differences between the proposed early defueling activities for TMI-2 and routine refueling activities at a typical commerical nuclear power plant, the activities are not fundamentally different and are, in fact, sufficiently similar to be justly compared. This evaluation proposes to compare similar events to demonstrate that TMI-2 activities are bounded by the TMI-2 FSAR.

A significant consideration in planning and conducting early defueling activities at TMI-2 and a major variation from routine fuel handling operations is the absence of fuel cladding which was lost during the accident. To offset this condition, i.e., the apparent consequences of the loss of this fuel containment barrier, specific protective measures have been provided. Containment integrity will be maintained to mitigate the possible consequences of postulated events involving uncontained fuel in the vessel. The boron concentration in the RCS has been increased to a level adequate to ensure subcriticality for all core configurations. Finally, the proposed early defueling process calls for restoration of a fuel containment barrier prior to removal of the fuel from the reactor vessel. The latter will be accomplished by loading the fuel in canisters which will be sealed while underwater in the reactor vessel. The canisters are designed to ensure subcriticality during all phases of early defueling. Once fuel has been loaded into the canisters, the early defueling process is very similar to the handling of a typical fuel bundle. Therefore, early defueling activities at TMI-2 are comparable to typical refueling activities described in the TMI-2 FSAR and the consequences of postulated early defueling events may be evaluated accordingly.

The FSAR for TMI-2 evaluated a variety of postulated events to bound the range of possible events and their offsite dose consequences. Section 4 of this SER similarly analyzes a variety of events to bound the range of possible early defueling events and their offsite dose consequences. To demonstrate that the range of events postulated in this SER are bounded by those analyzed in the TMI-2 FSAR, the two events postulated in this SER are those with the greatest potential consequences and were compared to similar events analyzed in the TMI-2 FSAR.

SER	FSAR
1) Canister Drop Accident	Fuel Handling Accident
2) Krypton - 85 Release	Waste Gas Decay Tank Rupture

Canister Drop Accident/Fuel Handling Accident

The drop of a fuel bundle was analyzed in Section 15.1.21 of the TMI-2 FSAR. The analysis assumed the dropping of the "hottest" fuel assembly in the FHB or the reactor building. As a result of the drop, the cladding suffered mechanical damage and the gap activity was released. In both the FHB and reactor building, the event was assumed to occur underwater. The activity was released to the building atmosphere and subsequently to the environment through the unit vents. Additional assumptions were made regarding the "scrubbing" effect of the water, the operation of the ventilation systems and other parameters which would affect the consequences of the event. As reported in Table 15.1.21-3, of the TMI-2 FSAR, the vent resulted in accumulated doses of 46 rem thyroid and 3 rem whole body for the two hour exclusion boundary and 4.8 rem thryoid and 1 rem whole body for the 30-day Low Population Zone (LPZ).

The drop of a filled defueling canister has been analyzed and is reported in section 4.4.2.2 of this SER. This analysis assumes the drop of the defueling canister into the dry portion of the FTC. The radionuclides listed in Table 4.4-2 of this SER were assumed to be released to the reactor building ventilation system and subsequently to the environment through the unit vent. The accumulated doses resulting from this postulated event are reported to be 0.43 mrem whole body for the two hour exclusion boundary dose and 0.077 mrem whole body for the 30-day LPZ. The thyroid dose was calculated to be 1.9 mrem for the two hour exclusion boundary dose and 0.35 mrem for the 30-day LPZ dose. In addition, the critical organ dose for the assumed distribution of radionuclides was calculated to be 2.96 rem to the bone for the two hour exclusion boundary dose and 0.53 rem to the bone for the 30-day LPZ dose.

As can be seen, the dose consequences of the canister drop accident are far less than those resulting from the fuel bundle drop accident. The thyroid and whole body doses are virtually non-existent for the canister drop accident. However, a comparison of the calculated critical organ doses (i.e., thyroid for the fuel bundle drop and bone for the canister drop) reveals that the consequences of the canister drop accident are at least a factor of ten less (2.96 rem vs. 46 rem). Despite notable differences in the two events, they are judged to be similar. Thus, the accident analyses contained in the TMI-2 FSAR clearly bound the consequences of canister drop events postulated for early defueling.

KRYPTON - 85 RELEASE/WASTE GAS DECAY TANK FAILURE

The rupture of the waste gas decay tank has been analyzed in Section 15.1.17 of the TMI-2 FSAR. This accident postulates the rupture of the waste gas decay tank and the consequent release of large quantities of the gaseous fission products to the auxiliary building ventilation system and to the environment through the unit vent. The NRC has evaluated this postulated event and reported the consequent doses to be 6 rem whole body for the two hour exclusion boundary, <1 rem whole body for the 30 day LPZ dose, and "negligible" thyroid for both cases.

The comparable event during TMI-2 early defueling operations would be the release of all unaccounted-for krypton-85 from the TMI-2 core. The analysis postulates the release of the krypton-85 to the reactor building ventilation system and subsequently to the environment through the unit vent. This scenario represents the maximum gaseous fission product release which can be postulated for TMI-2 in its current condition. The analysis is presented in Section 4.4.2.1 of this SER. The calculated doses for this event are 9.7 mrem whole body for the two hour exclusion boundary, 1.8 mrem whole body for the 30 day LPZ and zero rem thyroid for both cases.

A comparison of the dose consequences of the postulated waste gas decay tank rupture and the postulated release of krypton-85 clearly results in the conclusion that the waste gas decay tank rupture is bounding. Although these events are not identical, both events result in the release of large quantities of gaseous fission products. Thus, they can be considered comparable.

OTHER COMPARABLE EVENTS

A variety of other type postulated events have been analyzed in this SER. They include: (1) criticality, (2) boron dilution, (3) pyrophoricity, and (4) heavy load drops. The analysis of these events resulted in the conclusion that recovery with in-plant systems is possible prior to a postulated event resulting in an accident and consequent offsite dose. Analysis of comparable events in the FSAR led to a similar conclusion concerning offsite dose consequences.

For example, the chemical and volume control system malfunction analyzed in the FSAR is comparable to the Reactor Coolant System deboration analyzed for early defueling; i.e., both are boron dilution events. The malfunction of the chemical and volume control system has been analyzed in Section 15.1.4 of the TMI-2 FSAR. The event assumes a combination of multiple valve maloperations or failures and operation of more than one makeup pump while the Reactor Coolant System is operating at reduced pressure. This scenario describes the maximum credible boron dilution accident, resulting in a positive change in reactivity which is managed by plant automatic control systems.

Section 4.3 of this SER discusses postulated boron dilution events during early defueling. Potential boron dilution pathways have been identified. Each postulated event has been analyzed to assure that early recognition and timely response are possible to ensure the boron concentration is maintained at an acceptable level and subcriticality is assured.

A comparison of the postulated events indicates that the consequences of the deboration events analyzed in this SER do not exceed the consequences of the Chemical and Volume Control System malfunction analyzed in the TMI-2 FSAR. In fact, none of the events analyzed in this SER resulted in a significant offsite dose or other consequences and all were managed without approaching criticality.

SUMMARY

As a result of the above review, it is concluded that the type of events postulated and analyzed in this early defueling SER are comparable to and bounded by similar events postulated and analyzed in the TMI-2 FSAR. Generally, the consequences of the postulated early defueling events are much less significant because:

1. Fission product inventory has significantly decayed; therefore, source terms have been substantially reduced.

2. Potential for criticality has been eliminated by increasing the boron concentration in the RCS, isolation of all dilution sources, and removing dependence on control rods for assurance of shutdown.

3. Low decay heat load (12 Kw) eliminates dependence on all decay heat removal mechanisms except passive heat loss to the ambient environment.

4. Low decay heat load coupled with the open RPV eliminates any potential for pressurization type accidents.

10 CFR 50.59 REVIEW

To determine if early defueling activities involve an unreviewed safety question, the three key questions must be evaluated.

Has the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the Safety Analysis Report been increased?

A variety of events have been analyzed in this SER. It has been demonstrated that these events are bounded by comparable events analyzed in the FSAR. Two postulated events have potential offsite dose consequences: (1) a defueling canister drop, and (2) a Krypton - 85 release. It was shown that the potential consequences from these two events are substantially less than the potential consequences of comparable events analyzed in the FSAR. Also, by analysis of other postulated events, it has been demonstrated there are no events whose potential consequences exceed those analyzed in the TMI-2 FSAR.

By analyzing postulated events and reviewing various safety mechanisms, i.e., fire protection and decay heat removal, it has been demonstrated that early defueling activities will not adversely affect equipment classified as important-to-safety (ITS). Consequently, it is concluded that the probability of a malfunction of ITS equipment or the consequences of a malfunction of ITS equipment has not been increased.

Therefore, it is concluded that the early defueling activities do not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report.

Has the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report been created?

The variety of postulated events analyzed in this SER consider the spectrum of event types which potentially could occur during the early defueling process or as a result of the early defueling process. A comparison of these events with those comparable events in the FSAR demonstrates that the type events postulated for the early defueling process are similar to and bounded by the FSAR. In addition, no new event type has been identified which is different than those previously analyzed in the FSAR. Therefore, the early defueling process has not created the possibility of occurrence of an accident or malfunction of a different type than evaluated previously in the Safety Analysis Report.

Has the margin of safety, as defined in the basis for any technical specification been reduced?

Technical Specification safety margins at TMI-2 are concerned with criticality control and prevention of further core damage due to overheating. As demonstrated by this Safety Evaluation Report, Technical Specification safety margins will be maintained throughout the early defueling process. Subcriticality is ensured by establishing the boron concentration at greater than 4350 ppm during the early defueling process and ensuring that this concentration is maintained by monitoring the boron concentration and inventory levels and by isolating potential deboration pathways. The ability to prevent further core damage due to overheating is not affected by the early defueling process. Systems will remain in place to add borated cooling water to the core in the event of an unisolable leak from the reactor vessel.

CONCLUSION

In conclusion, the early defueling activities do not:

- o increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated in the safety analysis report, or
- o create the possibility for an accident or malfunction of a different type than any evaluated previously in the safety analysis report, or
- o reduce the margin of safety as defined in the basis for any technical specification.

Therefore, the early defueling activities do not constitute an unreviewed safety question.

No Technical Specification changes are required in addition to those requested in Technical Specification Change Request No. 47, as approved by Amendment to Order issued April 23, 1985 to be effective May 31, 1985, to conduct the activities bounded by this SER.

8.0 Conclusions

Early defueling and associated activities have been described and evaluated. The evaluations have shown that the tasks and tooling employed follow the continued commitment to maintain radiation exposure levels ALARA. The evaluations have also shown that the radioactivity releases to the environment that will result from the planned activities will not exceed allowable limits. Finally, it has been demonstrated that the consequences of postulated accidents with respect to potential core disturbances will not compromise plant safety. Therefore, it is concluded that early defueling activities can be performed without presenting undue risk to the health and safety of the public.

9.0 References

1. Technical Evaluation Report for Defueling Canisters, Rev. 0 15737-2-G03-114, March 1985.

2. Technical Evaluation Report for Defueling Water Cleanup System, Rev. 3, 15737-2-G03-106, January 1985.

3. Safety Evaluation Report for Plenum Lift and Transfer, Rev. 3, 15737-2-G07-106, April 1985.

4. Safety Evaluation Report for Removal of the TMI-2 Reactor Vessel Head, Rev. 5, 15737-2-G07-102, February 1984.

5. Criticality Report for the Reactor Coolant System, Rev. 0, 15737-2-N09-001, October 1984.

6. Hazards Analysis: Potential for Boron Dilution of Reactor Coolant System, Rev. 2 (to be issued).

7. TMI-2 Reactor Building Purge - Kr-85 Venting, GEND-013, page 93.

8. Safety Evaluation Report for Heavy Load Handling, 15737-2-G07-105 (to be issued).

9. Technical Evaluation Report for the Fuel Canister Storage Racks, Rev. 0, 15737-2-G03-113, January 1985

10. GPU TPO/TMI-127, Rev. 0, "Technical Plan for Pyrophoricity", December 1984.

11. "KENO4, An Improved Monte Carlo Criticality Program," (B&W Version of ORNL Code, KENOIV), NPGD-TM-503, Rev. B

12. TMI-2 Defueling Canisters Final Design Technical Report, B&W Document 51-1152641-03, April 1985.

13. TMI-2 Auxiliary and Fuel Handling Building Decontamination Effort, Task 9.1, Generic Criticality Safety Parameters, March 1983.

14. NUREG/CR-2139, Aerosols Generated by Free Fall Spills of Powders and Solutions in Static Air, Battelle Pacific Northwest Laboratory, December 1981.

15. Mishima, J., Potential Aerosol Generation Mechanisms from Damaged Shipping Packages, BNWL-SA-5742, July 1976.

16. NUREG-0172, "Age Specific Radiation Dose Commitment Factors for a One-Year Chronic Intake," November 1977.

17. Regulatory Guide 1.109, "Calculation of Annual Doses to Man from Routine Releases of Reactor Effluents for the Purpose of Evaluating Compliance with 10CFR50, Appendix I", Rev. 1, October 1977.

18. GEND-INF 019 Vol. II, May 1983, "Estimated Source Terms for Radionuclide and Suspended Particulates During TMI-2 Defueling Operations Phase II."

19. Safety Evaluation Report for Heavy Load Handling over the TMI-2 Reactor Vessel, Rev. 0, 15737-2-G07-110, April 1985.

20. NUREG-0683, Final Programmatic Environmental Impact Statement, March 1981.