Nuclear

GPU Nuclear Corporation
Post Office Box 480
Route 441 South
Middletown, Pennsylvania 17057-0191
717 944-7621
TELEX 84-2386
Writer's Direct Dial Number:

(717) 948-8461

4410-86-L-0049
Document ID 0317A

May 15, 1986

TMI-2 Cleanup Project Directorate
Attn: Dr. W. D. Travers
Director
US Nuclear Regulatory Commission
c/o Three Mile Island Nuclear Station
Middletown, PA 17057

Dear Dr. Travers:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
Defueling Safety Evaluation Report - Revision 10

Attached for your approval is Revision 10 to the Defueling Safety Evaluation Report (SER). The title of this SER has been changed to facilitate incorporation of future defueling operations by appropriate revision to this baseline document. Change bars indicate all changes incorporated since Revision 4. Revision 5 through 9 were internal revisions which were not issued for use. This SER has been revised to include core region defueling tools and activities, lower head vacuuming, and the use of core bore equipment as a defueling tool. The SER also has been revised to incorporate tooling and activity changes addressed in GPU Nuclear letters 4410-86-L-0001, dated January 17, 1986 (Use of Debris Canisters); 4410-86-L-0005, dated January 20, 1986 (Fines/Debris Vacuum System Modifications); 4410-86-L-0021, dated January 22, 1986 (Hydraulic Impact Chisel); 4410-86-L-0027, February 3, 1986 (Hydraulic Cutoff Saw); 4410-86-L-0051, dated March 26, 1986 (Hydraulic Shredder); and 4410-86-L-0062, dated April 11, 1986 (Canister Head Gasket Replacement).

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

w/ check $150.00
#00023373
D009
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Sincerely,

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

FRS/RBS/eml

Attachment

Enclosed: GPU Nuclear Corp. Check No. 00023373

SA-4350-3261-85-1

GPU Nuclear

SER 15737-2-G07-108 REV. 10

ISSUE DATE March 5, 1985

☒ ITS

☐ NSR

☐ NITS

TMI-2

DIVISION

SAFETY EVALUATION REPORT

FOR

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.Ride DATE 3/5/85

NO	DATE	REVISIONS	BY	CHECKED	GROUP SUPERVISOR	MGR DESIGN ENGINEERING
6	12/13/85	Issued for Comment	JCR	RJL	ETS	CR for RLR
5	12/6/85	Revised and Reissued for Use	JCR	RJL	ETS	CR for RLR
10	5/5/86	Revised and Reissued for Use	JCR	D	ETS	CR for RLR
9	3/5/86	Revised and Reissued for Use	JCR	D	ETS	CR for RLR
8	1/31/86	Revised and Reissued for Use	JCR	D	ETS	CR for RLR
7	1/2/86	Revised and Reissued for Use	JCR	RJR	ETS	CR for RLR
0	3/5/85	Issued for Use	JCR	RJR	ETS	CR for RLR
				CHIEF ENGINEER	N/A	

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EPD-33611 10/84

GPU Nuclear	NO. 15737-2-G07-108
Title Safety Evaluation Report for Defueling of the TMI-2 Reactor Vessel	PAGE 2 OF 89

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
3	Update to incorporate responses to NRC comments, revised shielding section, 550 lb limit on fuel canister loading, removal of 4 ft. exclusion zone requirement, and a minor change to words on pyrophoricity.
4	Update to incorporate boration of defueling system hydraulic fluid, and include reference to Foreign Material Controls Report (Reference 22).
5	Update to incorporate title change and inclusion of appendix to address bulk defueling activities.
6	Update to incorporate a detailed description of the impact chisel, use of the vacuum system for lower vessel head vacuuming, the use of the core bore equipment as a defueling tool, and to make editorial change to be consistent with the plant technical specifications.
7	Update to incorporate resolutions to site comments.
8	Update to incorporate knockout canister's capability of accepting particles greater in size than standard fuel pellets, the use of the vacuum system without the filter canisters, and to address the use of debris containers.
9	Update to incorporate a detailed description of hydraulic cutoff saw and to make minor editorial changes.
10	Update to incorporate appendices into main body of text and to include changes/updates to account for defueling experience to date.

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 Defueling of the TMI-2 Reactor Vessel

JOB 15737

SPEC. NO. 2-G07-108

REV. 10

EPD-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 the activities associated with 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

The activities associated with the defueling of the TMI-2 reactor vessel are the removal of core material from the reactor vessel, encapsulation of these materials within specially designed canisters, dewatering of the filled canisters, and placement of the canisters into the storage racks located in spent fuel pool 'A'. Defueling can be subdivided into four activities. These are:

- o Initial defueling - removal of fuel element end fittings and other loose debris, including vacuumable "fines", from the rubble bed. Specific activities include pick-and-place and vacuuming of loose debris.
- o Core region defueling - removal of debris from the core region of the reactor vessel which remains after the completion of initial defueling. This phase is differentiated from initial defueling in that significant sizing operations will be performed (e.g., separating and cutting of fused fuel assemblies and other large pieces of core debris). It also is intended that removal of the "hard crust" will be accomplished during this phase. It is noted that some activities performed during core region defueling will be similar to those performed during initial defueling.
- o Lower head defueling - removal of debris from the lower reactor vessel head. Vacuuming and other debris removal techniques will be used during this phase.
- o Core support assembly defueling - removal of debris from the core support assembly.

To the extent that specific defueling activities have been defined and tools designed, these activities and tools are addressed in this SER. Specifically, initial defueling and core region defueling activities are addressed in this SER. Additionally, vacuuming of debris from the lower head of the vessel, a primary activity associated with lower head defueling, is addressed herein. To perform the defueling activities, special equipment has been or is being developed. The following equipment has been defined to date, and its use is described in other sections of this SER. This equipment includes:

- 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 may be used to position and to 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 the canisters are resident either within the reactor vessel or within 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 canisters located within the deep end of the fuel transfer canal and within spent fuel pool "A";
- o A vacuum system that will remove fuel fines and other debris from the debris bed;
- o A cutting station to allow sectioning of debris that is too large to be placed in a canister directly and is too small to be cut in-place;
- o A remote manipulator that will provide the capability to access areas of the vessel which are not directly below the work platform working slots;
- o A manual tool positioner to provide a rigid mount for the remote manipulator and other core cutting and core debris removal equipment;
- o A partial fuel assembly removal tool to be used for removing partial fuel assemblies;
- o An incore instrument cutting tool which is designed to cut the incore instrument guide tubes and instrument strings;
- o Abrasive and water jet cutting equipment, including an abrasive saw and an abrasive/water jet cutting system, to perform remote, underwater cutting of hard materials with and without the use of abrasives;
- o Attachments and modifications to the basic vacuum system to remove debris in areas not accessible to the originally designed system;
- o An impact chisel to break apart hard materials;
- o A hydraulic shredder to reduce core debris, thus facilitating the placement of debris into fuel canisters or debris baskets;
- o The equipment used for acquisition of core stratification samples may be used to provide access to the lower regions of the vessel;
- o Debris containers to be used for removing fuel assembly upper end fittings, control component spiders, or other structural material from the reactor vessel.

As defueling operations proceed, the possibility exists that activites or equipment described in this report will need to be modified or new activites and/or tooling developed. For initial defueling, core region defueling and lower vessel head vacuuming, modifications to existing activities or equipment or the introduction of new activities or tooling will be reviewed in accordance with TMI-2 administrative procedures to ensure that no potential hazards or safety concerns, not bounded by this SER, are created. If no such hazards or safety concerns are created, defueling may proceed based on the new or modified activities or tooling without a requirement to revise this SER.

As the defueling activities associated with lower head and core support assembly become better defined, these activities will be reviewed in accordance with TMI-2 administrative procedures to ensure that no potential hazards or safety concerns, not bounded by this SER, are created. If no additional hazards or safety concerns are created, these activities may proceed based on this SER. If these activities are determined to involve issues or concerns not adequately addressed herein, they will be the subject of separate SER's or will be incorporated into subsequent revisions of this SER.

In addition to the equipment listed above, specially designed defueling canisters will be used to contain the core material for shipment and storage offsite. The design features of the canisters 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 canisters; and
- o Placement of the canisters into storage racks located in spent fuel pool "A".

The scope of this SER does not cover activities related to the transfer of canisters from fuel pool "A" for shipment offsite from TMI-2. Additionally, this SER does not cover activities associated with sample collection or core stratification sample acquisition.

1.3 Organization

Section 2 of this SER describes the major activities and equipment needed for defueling of the reactor vessel. Section 3 identifies the components and systems affected by or involved with defueling. Section 4 identifies and discusses the safety concerns associated with defueling. Section 5 presents an evaluation of the occupational radiation exposure considerations associated with defueling. Section 6 presents an assessment of the impact of defueling on plant activities including any impact on the operation of Unit 1. Section 7 presents a 10 CFR 50.59 evaluation for defueling and Section 8 assesses the environmental impact resulting from occupational exposure and radioactive releases to the public. Section 9 presents the conclusions of the safety evaluation and Section 10 contains a list of references.

The figures included in this safety evaluation provide a general concept of the defueling systems. These figures are not intended to provide exact design details, which are subject to change.

1.4 Licensing Documents Associated With Defueling

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

- Safety Evaluation Report for Defueling (this document)
- Safety Evaluation Reports for Heavy Load Handling
- 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
- Safety Evaluation Report for Core Stratification Sample Acquisition

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

1.4.1 Safety Evaluation Report for Defueling

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

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

This report establishes the boron concentration needed in the reactor coolant system to ensure subcriticality during defueling activities. The report describes the various analyses performed to establish 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 are used during defueling activities 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 spent fuel pool "A". It provides the information needed to demonstrate that the canister storage racks 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 included in the appropriate operating procedures.

1.4.8 Safety Evaluation Report for Core Stratification Sample Acquisition (Reference 23)

This report describes the activities to be performed during the acquisition of core stratification samples. It describes the equipment to be used and provides the information necessary to demonstrate that these activities can be performed safely.

2.0 Major Activities and Equipment

Defueling activities are performed in accordance with detailed approved procedures. The type of operations performed during the 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 canisters
- sectioning of oversized debris and end fittings for loading into fuel canisters
- sectioning of fused debris in the core debris bed
- removal of partial length fuel assemblies from the core
- cutting of incore instrument guide tubes and instrument strings
- boring through the debris bed to gain access to the lower vessel regions
- removal of debris from lower vessel head region by vacuuming
- use of a hydraulically operated shredder to reduce the size of core debris
- replacement of loaded fuel canisters head gaskets in the fuel handling building
- transferring loaded defueling canisters out of the reactor vessel
- dewatering of the loaded defueling canisters in the reactor vessel and/or spent fuel pool "A"
- storage of defueling canisters in spent fuel pool "A" awaiting shipment

2.1 Equipment Installation

Defueling equipment is fabricated in subassemblies to enable the equipment to be transported into the reactor building through the existing personnel airlock. Equipment is brought into the reactor building at the 305'-6" elevation and transported to the operating deck area at the 347'-6" elevation. The main structure of the work platform is in position over the reactor vessel. Additionally, the canister positioning system is installed within the vessel. Other defueling equipment is, in general, of modular design for minimum assembly and installation time and ease of maintenance.

Prior to being placed in service, equipment is 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 undergoes an acceptance checkout and turnover program on site.

2.2 Defueling Operations and Equipment

Except for water processing activities, debris and fuel handling operations and canister loading takes 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 are supervised by an appropriately licensed operator.

The water level within the reactor vessel/IIF is determined and controlled to maintain adequate shielding and minimize dose rates to personnel on the work platform and in surrounding areas. It is expected that the normal water level will be approximately 5 feet above the vessel flange.

Prevention of inadvertent lifting of debris high enough to cause excessive exposure to personnel on the shielded work platform during defueling activities is generally implemented mechanically through the use of rigging bars. It is planned to have three standard rigging bar lengths. The different length bars may be combined in several different ways so as to accommodate the various defueling tools. Using the rigging bars per administrative controls prevents debris from being lifted above elevation 323'-6".

The above approach only applies to those tools handled by the reactor building service crane. For tools hanging from the work platform jib cranes, no rigging bars are required, as the vertical movement limitations of the jib cranes prevent the lifting of debris above elevation 323'-6".

It is not considered necessary to mechanically restrict the lifting of all long-handled tools such that all debris remains below elevation 323'-6" during defueling operations. To do so could potentially result in unnecessary cutting and handling activities that would result in unnecessary occupational exposure. Consequently, administrative controls have been established to allow for lifting debris above elevation 323'-6". The governing criteria is the maintenance of acceptable radiation levels on the defueling work platform. In addition to the design features and administrative controls, radiation monitoring capabilities are provided on the defueling work platform to alert the operators to dose rates that may be unacceptable. Acceptable dose rates are established by Radiological Controls for the planned activities.

Procedures have been developed that require loads greater than 550 lbs. (weight measured in air or 490 lbs. measured in water) be in a resting (i.e., no loads on handling equipment) position prior to being released in the canister. This limit is based on results of dynamic impact evaluations which show that the lower support plate of the fuel canisters can withstand loads of at least 350 lbs. (weight measured in air) dropped the full canister length without a failure of the lower plate to shell weld. When credit is taken for the drag forces of the water in the canister this weight limit increases to 550 lbs.

When the loading of a fuel canister is completed, the canister closure head is installed. The bulkhead seal surface and bolt holes are cleaned and inspected to ensure they are free of any debris. The seal surface of the closure head is inspected to ensure it is clean, seals are acceptable and the required number of bolts are in place. The head is then installed on the canister and the required number of bolts torqued under water. Prior to removal of the fuel canister from the reactor vessel, two relief valves are installed or the canister is vented. When the loading of a knockout or filter canister is completed, the canister is vented or the flow fittings are plugged and two pressure relief valves 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 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 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 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 minimizes the air intake flow into the support structure and work platform region. This ensures 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 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 slots. 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 serves 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 is 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 supports the control consoles for operation and monitoring of the defueling equipment. The platforms are comprised of beam and plate subassemblies that were 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 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 defueling. 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 is incorporated into the SCSB design to prevent lateral movement during 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.

Long-Handled Tools

During defueling long-handled tools are used. Some of them are sectioned for ease of handling and storage. The operator operates the tools through the slots in the rotatable shielded work platform. Most tools are 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 are provided; both types of tools have been used previously in remote handling operations. The light weight tools are made with 1-1/2 inch aluminum or stainless steel 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. All end effectors are 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.

Fines/Debris Vacuum System

The fines/debris vacuum system as originally designed for use during defueling activities is illustrated in Figure 2-7. That original design is composed of a pump, piping, valving and knockout and filter canisters. The system is designed to remove fuel debris from the debris bed. Adaptations to the original design are such that the system may be used to collect vacuumed debris in both knockout and filter canisters, connected in series, or, based on operational considerations, the system may be operated using other combinations of canisters (e.g., only single knockout canister, two knockout canisters in series). Each such combination will be reviewed in accordance with existing TMI-2 administrative procedures to ensure no additional safety concerns are created.

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 a canister, the system is equipped with a means for determining the canisters' weight during loading. Knockout canister weighing is designed to be performed using the knockout canister connect assembly module, which is located under the rotatable work platform. After a mechanical interlock between the Canister Positioning System (CPS), which is carrying the knockout canister, and the knockout canister connect assembly module has been established, the knockout canister is lifted from the CPS a predetermined distance to allow weighing of the canister. The filter canisters are weighed using the Filter Canister Weighing System (Figure 2-8). The Filter Canister Weighing System is attached to the filter canister and suspended from the shielded work platform deck shielding plate. The existing load cells used in weighing both the knockout and filter canisters have a 0-7500 lb. range with 0.5% accuracy. This results in a ± 17 lb. accuracy for the design weight of a fully loaded canister (i.e., 3355 lbs). In each case the load cell readout is provided on the control console. If the above weighing systems are not available, alternate means of weighing the canisters will be provided and approved in accordance with TMI-2 administrative procedures.

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. The nozzle may be equipped with borated (i.e., ≥4950 ppm) water jets or other means to increase the vacuuming efficiency. Over-pressure protection of canisters is ensured since the maximum pump discharge pressure is 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 since the shielding on the shielded work platform will protect the operators from high dose rates.

In addition, the vacuum system will be adapted to permit vacuuming of debris in the lower vessel head region. Adaptations will be upstream of the existing vacuum system inlet connections. The vacuum system itself will be operated within design parameters. The adaptations to be made are summarized as follows:

- A fuel canister, installed in the single canister support bracket, with a temporary closure head, providing system inlet and outlet connections, may be used. Prior to removing the fuel canister from the vessel, the temporary head will be removed and a standard fuel canister head installed.

- A simplified vacuum nozzle will be used. The nozzle will not have mechanical unclogging capability. The system will be backflushed to clear the nozzle, as required.

- If the fuel canister is used, vacuum hose, with appropriate connections, will be used to connect the nozzle to the temporary fuel canister closure head. Similar hose will then be used to connect the closure head to the existing vacuum system connection. If the fuel canister is not used, a longer length hose will be used to connect the nozzle to the existing vacuum system knockout canister.

The inlet connection on the temporary fuel canister closure head will be equipped with a nozzle which will cause all debris entering the fuel canister to be directed vertically downward. This approach minimizes the potential for damage to the boral shroud. As previously described, a limit of 550 lbs. (weight measured in air) has been established as the maximum weight of debris that can be released into a fuel canister without being in a resting (i.e., no load) position. This limit was established to prevent damage to the lower support plate. As all of the vacuumed debris will weigh considerably less than 550 lbs., lower head vacuuming operations are not expected to result in any damage to the canister lower support plate.

To gain adequate access to the lower head area, the reactor vessel vent valves will be removed using tools and procedures similar to a normal removal (i.e., mechanical disassembly). The vent valves will be removed through the rotatable defueling work platform access slots and may be transferred unshielded in air to an appropriate storage location. Relocation or other measures will be taken to ensure acceptable doses to personnel during valve removal and storage.

Canister Positioning System

The canister positioning system (CPS) is shown in Figure 2-9. 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". Lowering or raising of the canister sleeve can be accomplished only when the canister sleeve is under the open slot. Canisters are moved into the loading position by manually rotating the CPS from above the platform.

Cutting Station

The function of the cutting station is to clamp and section debris which is too large to be loaded directly into fuel canisters and is too small to be cut in-place. The cutting station is intended to be used to section debris composed primarily of ductile material (e.g., 304 stainless steel, zircaloy and inconel). The more brittle material will be sectioned by other tools while still on the core surface.

The use of the cutting station requires that the debris be removed from the core region and positioned in the station. This will be accomplished via the long-handled tools. Clamping and cutting of the debris will occur at an elevation approximately 9-1/2 feet below the normal water level. Cutting of the debris will be accomplished by a hydraulically operated saw.

The cutting station is suspended from rails on the defueling work platform's working slot. Installation and maintenance of the cutting station will be performed via the working slots.

Remote Manipulator

The remote manipulator will be available to support long-handled tool operations by handling defueling tools, components of the viewing system and fuel debris. The manipulator, mounted to the manual tool positioner, will have approximately a 4-foot reach and have six degrees of freedom of motion. It is intended that the manipulator be controlled from inside the reactor building; however, the option of controlling it from outside the reactor building and thus, reducing personnel exposure, may be available. The manipulator will be capable of performing defueling operations in areas of the vessel which are not directly below the work platform working slots, in parallel with other defueling operations.

Manual Tool Positioner (MTP)

The MTP will provide a rigid mount, capable of positioning defueling tools and the remote manipulator in all areas of the core support assembly. Mounting and removal of the defueling tools and the remote manipulator will be accomplished via the working slots using long-handled tools.

The MTP consists of a post supported by a carriage which, in turn, is supported by the rails on the work platform. The MTP can travel along the working slots via these rails. A stabilizer, clamped to the post and supported by the core support assembly baffle plates, may be used to increase the load capability of the MTP. The stabilizer may also be used to stabilize debris in the core during cutting operations. More than one MTP may be used if required.

Abrasive/Water Jet Cutting System

The abrasive/water jet cutting system will be provided to perform remote, underwater cutting of hard materials with and without the use of abrasives. The equipment is designed to be handled by the remote manipulator. The remote manipulator, in turn will be used to position the equipment at the cutting location. Nozzles with orifice diameters compatible with abrasive/water jet cutting, water jet cutting, and abrasive and/or water jet surface cleaning will be used. An abrasive feed system will be used to supply abrasive grit to the nozzle. The feed system will be capable of isolation from the water jet to allow for water jet cutting and cleaning without the use of abrasives. The abrasive/water jet cutting system can be turned off using an emergency shutoff feature provided at the shielded work platform.

Water used for the abrasive/water jet cutting operations will be borated to a concentration of at least 4350 ppm. The supply water for cutting will be extracted from the filter train of the DWCS or from another borated water supply and will be returned to the reactor vessel via the waterjet. If hoses, rather than hard piping, are used for the water supply line, appropriate controls will be established to avoid the possibility of connecting to the wrong water supply.

Incore Instrument Cutting Tool

The incore instrument cutting tool is designed to cut the incore instrument guide tubes and instrument strings below the fuel assembly lower end fittings, as required. Due to the geometry of the region, the cutting of the guide tube below a fuel assembly must be accomplished from a neighboring fuel assembly location. The actual area of cutting is in the 5-1/4" high space between the bottom of the lower grid rib section and the top of the lower grid distributor plate. The tool is sized to permit installation into the space vacated by one previously removed fuel assembly. Operators on the shielded work platform lower the tool into place using a long-handled pole. Once the tool is in place, a hydraulic motor is used to power a reciprocating saw mechanism located at the end of the tool. The handling pole is turned to bring the saw blade into contact with the guide tube. When the cut is completed, the saw blade can be retracted and the tool can then be removed.

Core Bore Equipment

The equipment used for acquisition of core stratification samples may also be used during defueling to provide access to the lower region of the vessel. Core stratification sample acquisition activities will use a hollow drill bit to withdraw core samples. When the core bore machine is used for defueling, a solid face drill bit may be used. The majority of core bores during defueling will be in the core region above the lower core support plate. Any core bores which pass below the lower support plate will be aligned to pass through existing openings in the lower core support assembly and the length of drill string will be controlled to prevent the drill from exceeding the maximum allowable drill depth. A detailed description of the core bore equipment and its operation is provided in the Safety Evaluation Report for Core Stratification Sample Acquisition (Reference 23).

Hydraulic Cutoff Saw

A hydraulic cutoff saw, which is essentially a hydraulically driven hack saw, may be utilized for sizing structural material or other debris. This will facilitate the loading of debris into defueling canisters, debris baskets or debris containers. Debris will be secured for cutting using a hydraulically actuated clamping station. Both the saw and the clamping station will be located within the reactor vessel/IIF and supported from the defueling work platform.

Impact Chisel

The impact chisel is a hydraulically actuated, miniature jackhammer-type tool for use in breaking apart hard materials. The tool is designed for effectiveness on brittle, friable materials. The maximum output of the chisel is 2000 impacts/minute and the chisel can impart an impact energy of approximately 20 foot-pounds. The impact chisel will be supported by either a long-handled pole or the MTP. The chisel's angle of attack can be varied remotely to achieve any angle from vertical to horizontal. Different bit types are provided for the various anticipated chiseling operations.

Viewing System

The viewing system used within the IIF and reactor vessel to support the defueling 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. The operation of the manual tool positioner, the remote manipulator and other cutting equipment allows defueling activities to be performed in areas of the vessel not directly below the work platform working slots. These activities require viewing capabilities under the work platform. Thus, camera mounts may be hung from the work platform support beams or the core support assembly baffle plates.

Abrasive Saw

An abrasive saw may be used to perform underwater cutting operations. This tool is designed to be attached to the MTP. The abrasive saw is powered by a hydraulic motor. Different blade materials will be provided for the cutting of the different types of debris materials. The abrasive saw is designed to be installed through the working platform slots.

Top Access Partial Fuel Assembly (PFA) Removal Tool

The top access PFA removal tool is available to provide the capability to remove partial fuel assemblies from the core region of the vessel. The intended use of this tool is to install a stainless steel anchor bolt into the top portion of a fuel assembly stub via a series of impacts. These impacts would drive the approximately 27-inch long anchor bolt up to 23 inches into the center of the upper portion of an intact fuel assembly stub. The maximum possible impact energy to the anchor bolt is 6,600 foot-pounds. The tool will then be used in conjunction with the reactor building service crane to lift and translate the fuel assembly stub to a fuel canister for loading into the canister.

Hydraulic Shredder

The hydraulic shredder will be used to reduce fuel pins (with and without fuel) and inconel spacer grids to sizes which will facilitate placement into fuel canisters or debris baskets. The shredder will be suspended below the defueling work platform using a support structure attached to the platform. The shredder is hydraulically powered using a working fluid borated to at least 4350 ppm. The shredder hydraulic system has a capacity of approximately 130 gallons and is independent of the existing hydraulic system used to power the other defueling tools. Additionally, the gear box has been sized to contain less than 2 gallons of lubricating oil. The control console will be located on the work platform and the shredder will be located in the reactor vessel close to the rubble bed. As planned, the fuel/debris will be loaded into an inlet hopper and the shredder output will be discharged to a transfer container which is then emptied into a fuel canister or debris basket. Discharge from the shredder may also be directed to the debris bed. Non-shreddable material lodged in the shredder will be retrieved through manual manipulations using long-handled tools.

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

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.

The hydraulic system is used to supply hydraulic fluid at the required operating pressures and flow rates for the operation of most of the defueling equipment. The hydraulic system working fluid is borated to a concentration of at least 4350 ppm. For tools/equipment where hydraulic fluid other than that supplied by the hydraulic system is used (e.g., core bore, hydraulic shredder), the fluid used will be borated to at least 4350 ppm or the quantity/location of the fluid will be evaluated to ensure no safety concerns exist.

Debris Containers

To expedite access to the vacuumable fuel debris in the core region, debris containers may be used for the removal and storage of fuel assembly upper-end fittings, control component spiders, or other structural material from the reactor vessel. The debris containers are similar in outside design to the fuel canisters described in Reference 1, except for the upper closure head and the number of bolt holes. Additionally, the debris containers are not required to have the internal boral plates, concrete filler, recombiner catalyst, or dewatering capability.

The bulkhead opening will be enlarged to facilitate loading of end fittings or other structural material. The enlargement is limited to allow usage of existing closure heads. As a method of differentiating these containers from fuel canisters, four of the eight bolt holes on the container and closure head have been eliminated. The debris containers will be partially assembled on site using spare or rejected parts from the canister vendors. The assembly will use site-approved welding procedures.

During loading, the use of the seal cover will be eliminated. This will aid in identification of the type of container as well as facilitating end fitting loading. After the debris containers are loaded, they will be closed and stored in the spent fuel pool "A" storage racks. Use of these debris containers for shipment will require a separate licensing submittal.

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 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, through the personnel airlock. The empty canisters are then transported to elev. 347'-6" and lowered into the Fuel Transfer Canal (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 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, the loaded canisters are prepared for transfer as discussed in Section 2.2. In addition, the canisters may 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 and storage will be minimized. However, this partial dewatering is not required to safely transfer the canisters to the Fuel Handling Building (FHB) for dewatering and storage, as overpressure protection is provided.

Once canister preparation 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. Routine access to the north end auxiliary work platform will not be permitted 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 the canal storage racks or one of the two fuel transfer system upenders. Some vacuum system filter canisters may be transferred to the DWCS filter canister storage locations. These filter canisters may then be reused as cleanup filters in the DWCS. Shielding for all canisters in the canal 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 spent fuel pool "A", where dewatering is to be performed. Additionally, a loaded canister may be returned to the reactor vessel by just reversing the process taken to place the canister in the FHB.

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 may be used as an indication that the canister is dewatered.

The CHB is equipped with a load cell which will permit the CHB to 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 restriction. The shipping weight restriction is 2800 lbs.; however 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.

Due to the excessive leakage past the fuel canister head gasket seals that has been experienced to date, the original gaskets are to be replaced. For those canisters that have already been loaded with debris, this gasket replacement will be performed in the FHB. To perform this activity, the loaded canister must first be positioned in the dewatering station where the canister head will be removed. The head will then be transported, via crane, to a specially designed work table, which is attached to the dewatering station platform shield wall. The work table is rotatable, thus permitting the canister head to be turned over, which allows access to the gaskets on the underside of the canister head. After rotating the head, the original gaskets will be removed and new gaskets inserted. To minimize the potential for the spread of contamination to the spent fuel pool water, a temporary cover will be placed on the canister once the head is removed. Prior to returning the canister head to the canister for reinstallation, the head will be visually inspected for damage. If any damage is found, the head will be repaired or replaced.

2.4 Off-Normal Events

Potential off-normal events, such as electric power failure, loss of compressed air, failure of viewing systems, jamming of defueling equipment, tool failures, and operator error have been considered during the design of the 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 defueling activities until repairs can be made, but will not cause unacceptable risk to the health and safety of the public.

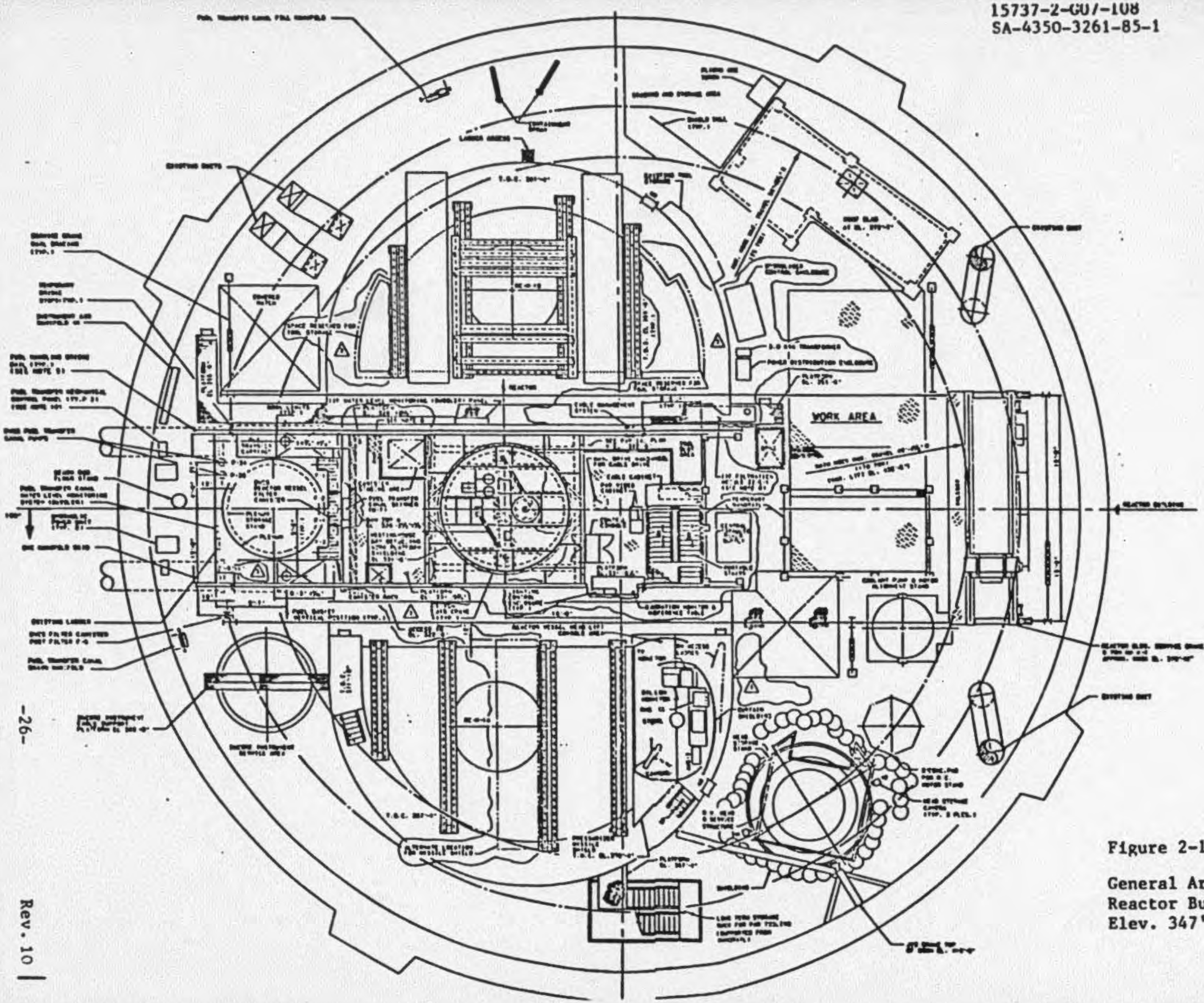

Figure 2-1

General Arrangement
Reactor Building
Elev. 347'

Pick-and-Place

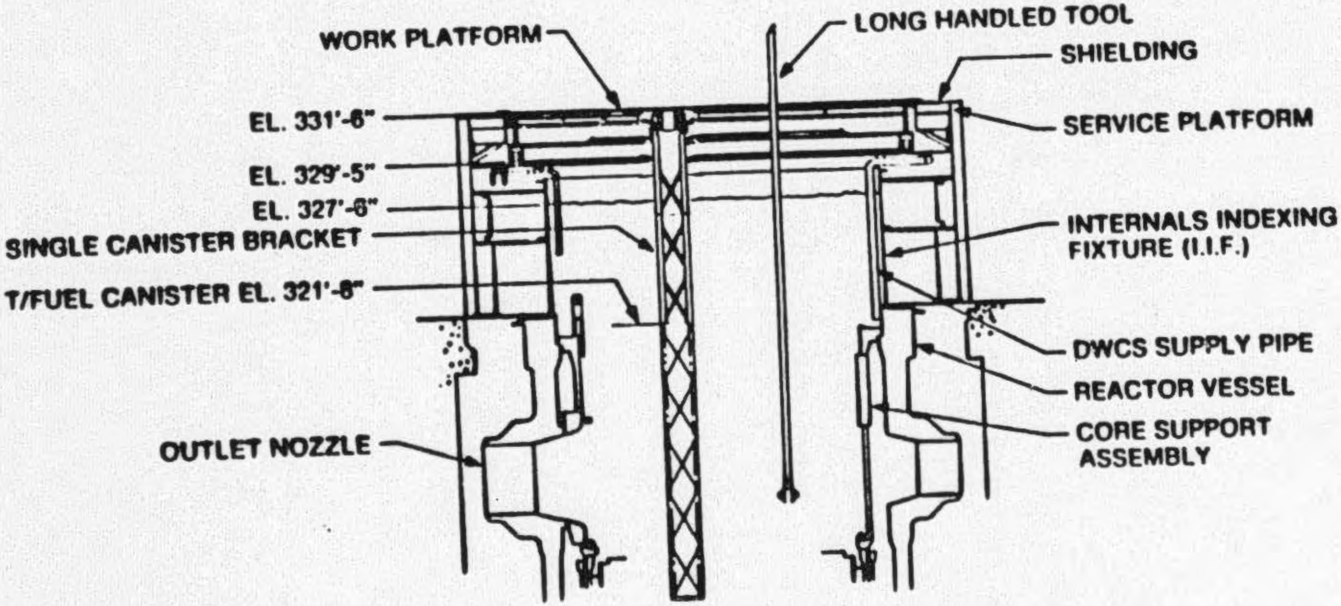

Figure 2-2

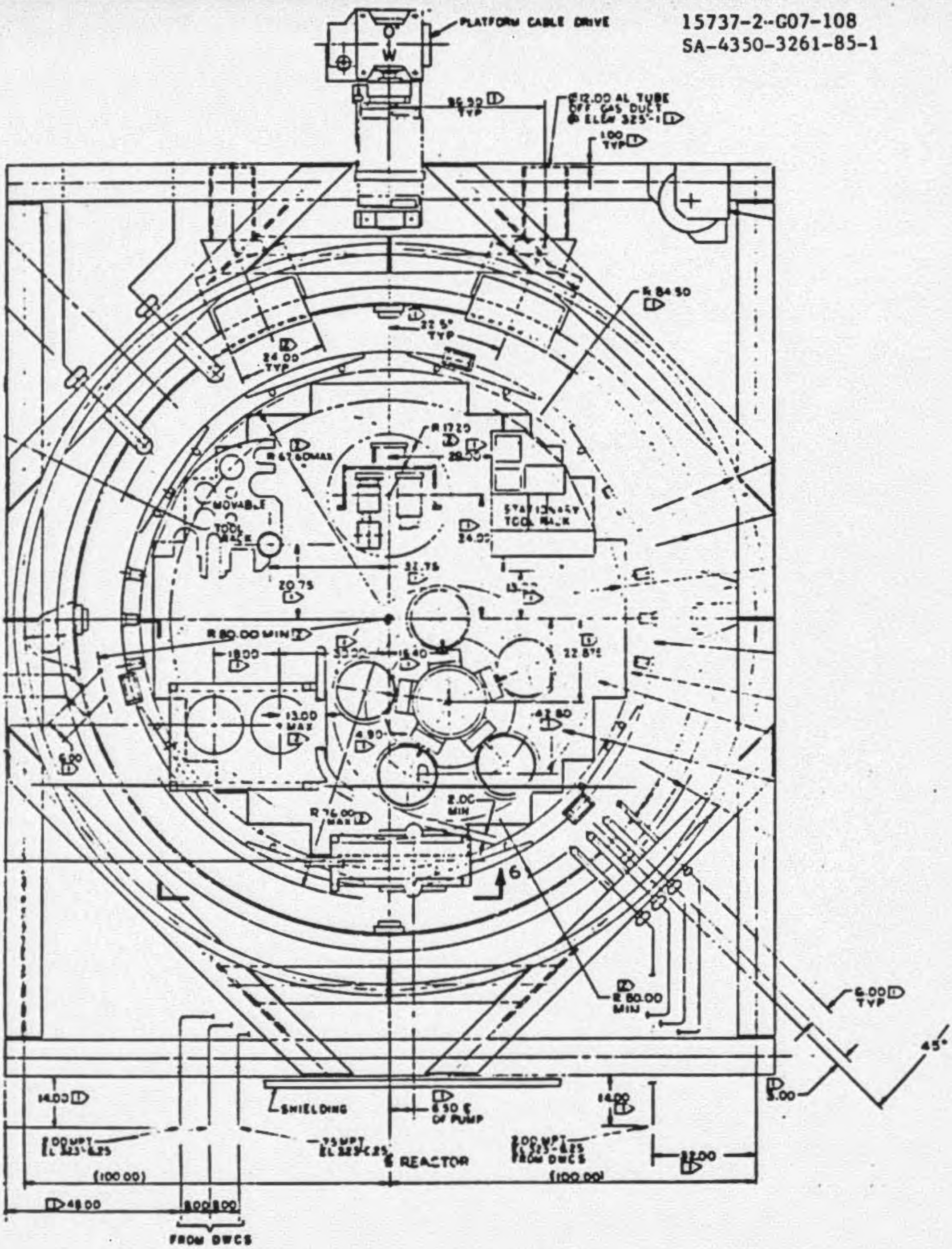

Figure 2-3 Piping Routed Through Support Structure

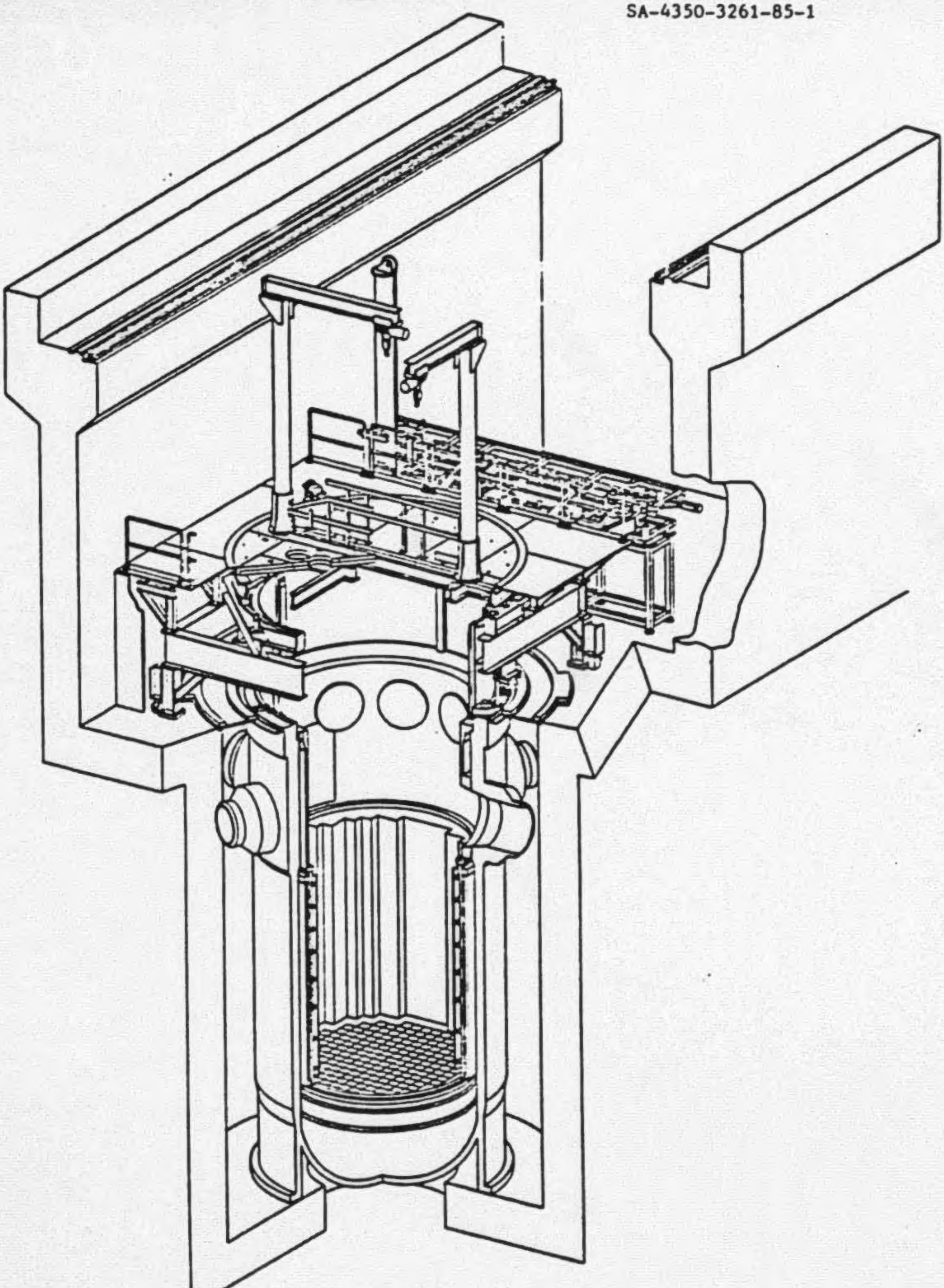

Figure 2-4 Early Defueling System

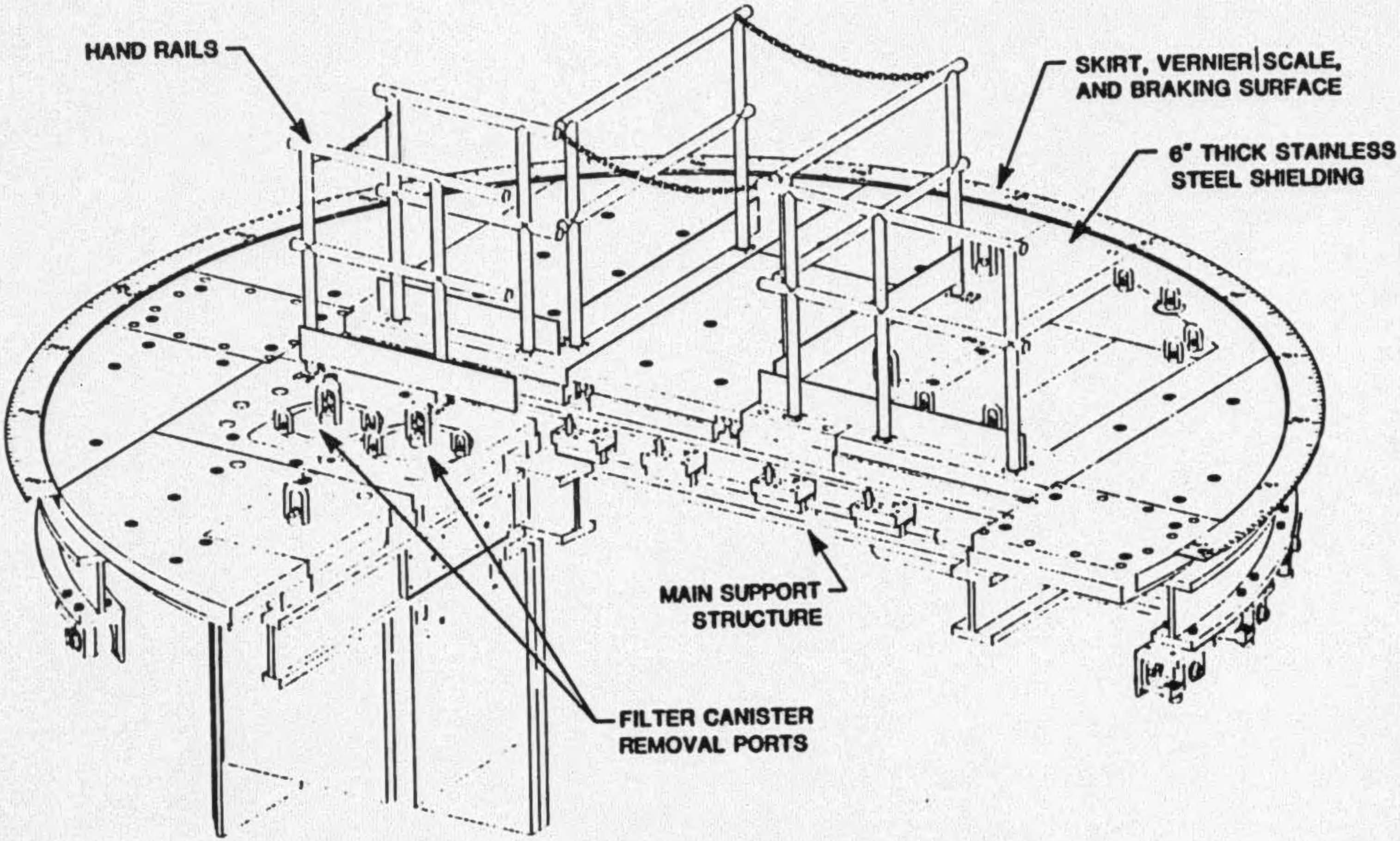

Figure 2-5 Shielded Work Platform Assembly

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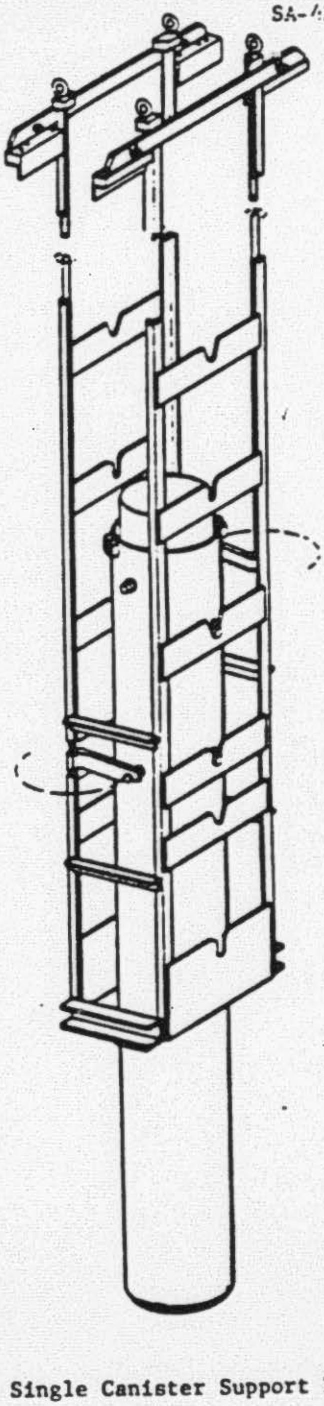


Figure 2-6 Single Canister Support Bracket

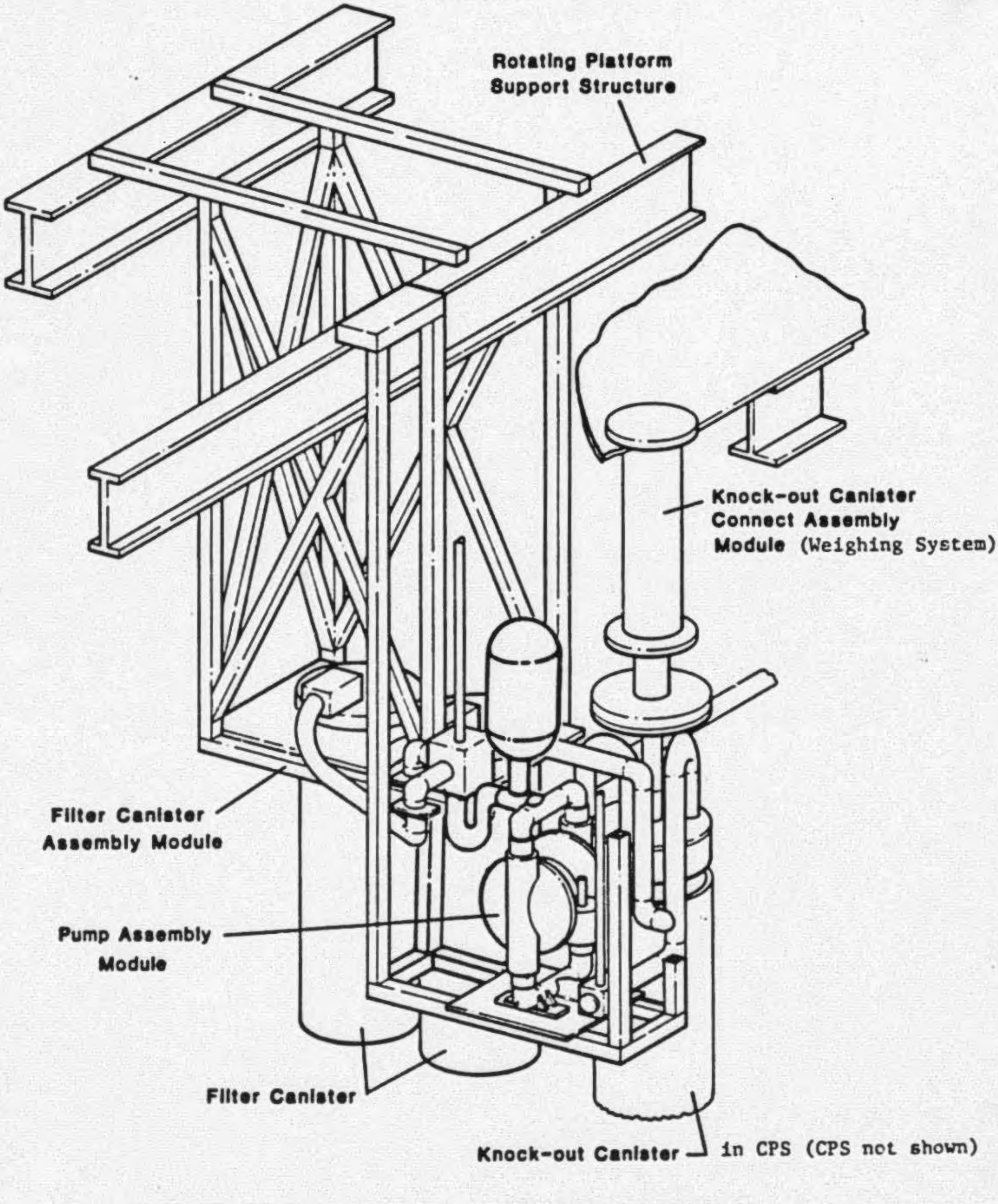


FIGURE 2-7 FINES/DEBRIS VACUUM SYSTEM

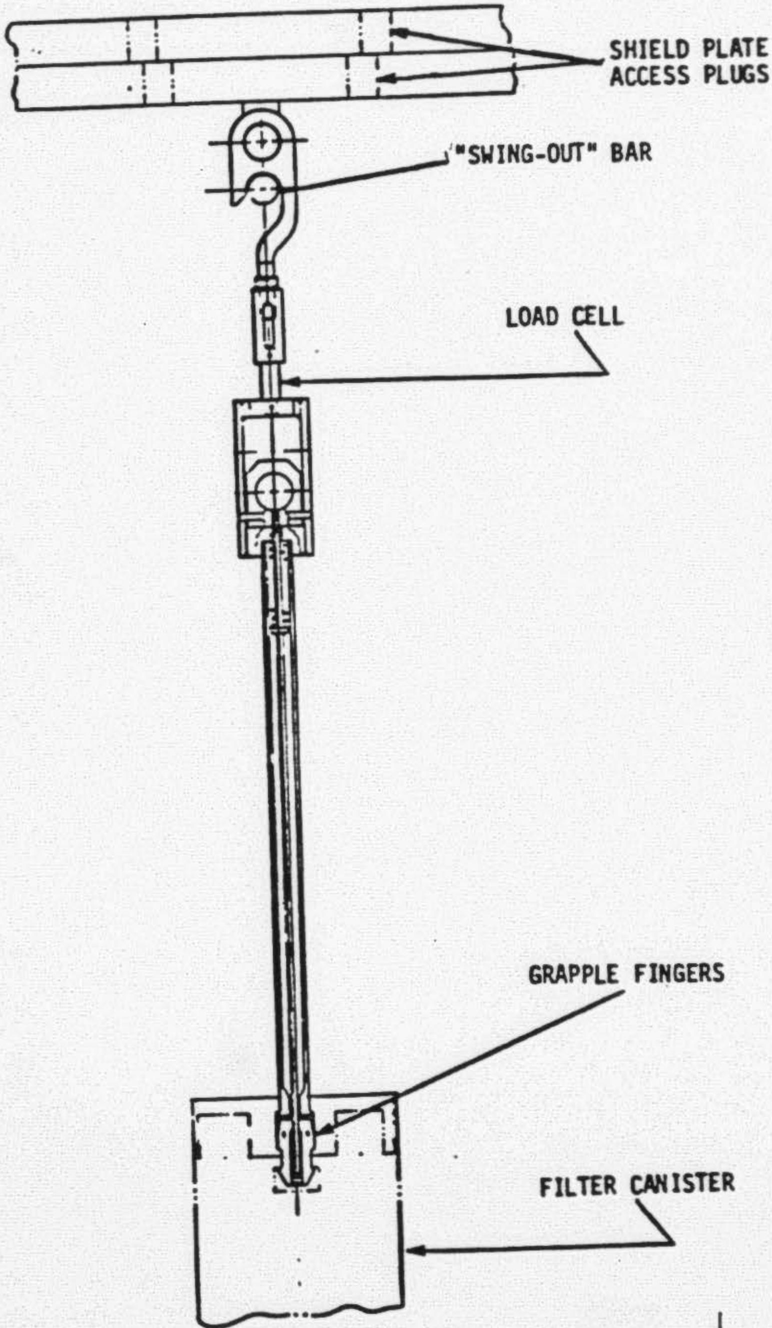
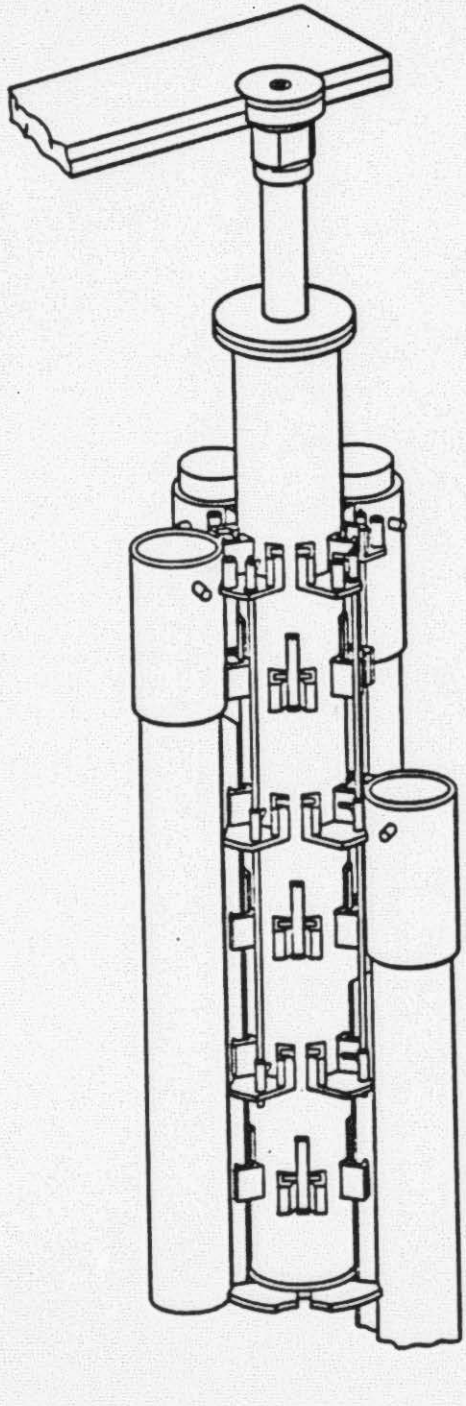


FIGURE 2-8 FILTER CANISTER WEIGHING SYSTEM



CANISTER POSITIONING SYSTEM

Figure 2- 9

3.0 Components and Systems Affected

3.1 Containment

During defueling operations, the uncontrolled release of radioactive material to the environment is prevented by maintaining the integrity of the reactor building containment. All defueling activities and filled canister transfer operations will be stopped, as required by Technical Specifications, any time 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 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 as discussed in Section 2.2.

The RCS water will be sampled at a frequency sufficient (i.e., 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 provides another means of RCS sampling.

3.2.2 Reactor Coolant System Level Monitoring

Level monitoring is accomplished through at least two of 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 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.

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 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 defueling, decay heat removal will be by loss to ambient cooling. The MDHRS provides a flow path 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 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 defueling to provide water shielding over the defueling equipment 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 defueling. It is designed to be operated on an as needed basis 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. In the event of the unavailability of the DWCS, the Submerged Demineralizer System (SDS) and a temporary reactor vessel filtration system can be used to remove radioactive ions and to maintain radioactivity at acceptable levels in both the RCS and FTC.

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 entire canal could be flooded to re-establish the normal defueling water level in the deep end of the FTC, if necessary. Another postulated condition requiring flooding of the canal is excess IIF leakage.

The water level in the shallow end of the FTC following flooding 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 underwent 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.

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 has been installed in the northeast end of spent fuel pool "A".

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 overpressurization 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 provides most of the necessary shielding. The FHB DS work platform has been 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 (i.e., $\geq$4950 ppm). A hose connecting the FTC fill manifold to a manifold on the shielded work platform supplies 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 prevents 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 based on approximately 26 ft^2 in openings in the platform. The system is operated on an as needed basis. 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 defueling operations, one in the reactor building and one in the FHB. The existing storage fuel handling bridge, with a new canister handling trolley, transfers loaded canisters in the FHB spent fuel pool "A" area. The existing auxiliary fuel handling bridge, with a new canister handling trolley, transfers 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 are used to provide north-south movement of the canisters. The new trolleys provide east-west movement of the canisters. The bridges, trolleys and hoists are motor-driven and controlled from the trolleys. Each new trolley includes 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) and is 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 of the collar (i.e., approximately 30 inches) in the reactor building 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 hoists 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 defueling, the CHB's were 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, and have a factor of safety of 5 based on ultimate strength and 3 based on yield strength.

3.3.2 Polar Crane

The reactor building polar crane was used to lift heavy loads during installation of the early defueling equipment and may be used during subsequent defueling 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 spans the FTC. This crane is used to handle tools, equipment, shielding and empty canisters. The crane is also used to handle long-handled tools during fuel canister loading operations. To prevent fuel canister overloading during loading operations, the RBSC is used to weigh the fuel canisters on an as-needed basis. The RBSC, in conjunction with an appropriate lifting tool, and an ordinary weight scale are used to weigh the open fuel canisters. The RBSC may also be used to weigh canisters as part of in-vessel dewatering. The crane rigging is such that the canister cannot be lifted to a height that would cause excessive exposure to personnel on the shielded work platform.

The crane rails run the length of the canal on top of the D-rings. This allows the crane access to the entire length of the canal and to the hatch between elev. 347'-6" and elev. 305'. This crane provided load handling capability for the numerous small loads required to be handled during defueling preparations and operations without requiring use of the polar crane. The crane was load-tested prior to use. The crane complies 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 comply 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 defueling identified the following safety issues:

- criticality control
- boron dilution
- release of radioactivity
- hydrogen evolution
- pyrophoricity
- heavy load drop
- fire protection
- decay heat removal
- use of the core bore equipment
- reactor vessel integrity

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 ensure that this situation would not occur, a review of materials that may be located on the defueling work platform or handled within the reactor vessel has been made. The effect of these materials on the shutdown margin, assuming they were brought into contact with the fuel, was determined considering the materials as moderators and reflectors. For the purpose of this evaluation, the RCS boron concentration was assumed to be 4950 ppm, the lower procedural operating limit. A quantity of the various materials was determined such that the resultant $k_{eff} \leq 0.99$ for all credible situations (Reference 22). Additionally, the abrasive grit to be used in the abrasive/water jet system has been evaluated to ensure that the addition of this material to the reactor vessel will not result in a criticality safety concern. Controls have been implemented to ensure that limitations on material type and quantity are not violated (e.g., only abrasive grit materials that have been evaluated for their effect on k_{eff} will be used, hydraulic system working fluid is borated to at least 4350 ppm).

4.2.2 Canister Handling Operations

Canister handling operations raise three areas of concern regarding criticality. The first deals with the transport of the canisters in the canister transfer shield, the second deals with dewatering of the canisters, and the last is related to the removal of the canister head for gasket replacement.

Relating to the first concern, lead and steel 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 (i.e., $k_{eff} \leq 0.95$) 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). 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.

With regards to the gasket replacement activities, Reference 1 analyses demonstrate that the maximum k_{eff} for a single, loaded fuel canister moderated with unborated water is 0.857. The removal of the canister head would not be expected to appreciably affect this value. Additionally, the canister will remain in spent fuel pool "A" during the gasket replacement. As the Technical Specifications require the water in spent fuel pool "A" to be borated ($\geq$4350 ppm), the water within any open fuel canister will also be borated. Taking credit for the borated water within the canister would reduce k_{eff} to a value well below 0.857. Consequently, it is concluded that the planned activities associated with the head gasket replacement do not result in a canister k_{eff} exceeding 0.95. Additionally, if any of the open canister's contents are spilled into spent fuel pool "A", subcriticality is ensured by the 4350 ppm boron concentration in the pool.

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.

4.2.4 Debris Containers

The following evaluation was performed independent of the debris containers' fuel inventory; thus, no restrictions are required regarding the amount of fuel loaded into the containers. However, efforts are made to limit the amount of fuel entering the containers. These efforts include limiting fuel rod end stubs to approximately two (2) inches and limiting debris to structural materials with no significant quantities (i.e., chunks or agglomerations) of unidentifiable material attached. Thus, the containers are not expected to contain significant quantities of fuel.

When debris containers are being loaded in the reactor vessel or temporarily stored in either the fuel transfer canal or spent fuel pool "A", they will be submerged in water having a boron concentration of at least 4350 ppm. Since the containers are vented, the boron concentration of the water within the containers was assumed equal to that of the surrounding water. Previous analyses (Reference 5) have demonstrated that the core will remain shutdown, with a $k_{eff} \leq 0.99$, when the RCS water is borated to a concentration of at least 4350 ppm. Although differences exist between the assumptions used in the Reference 5 analyses and those that would actually be used for an explicit analysis of submerged containers, direct application of the Reference 5 results to this evaluation are conservative. The differences are noted here to demonstrate this conservatism:

- The Reference 5 model included the entire core. With the containers being significantly smaller than the core, the containers experience significantly more neutron leakage, thus, k_{eff} is reduced.

- The Reference 5 model did not consider structural material; whereas the majority of the containers' inventory will be comprised of structural material. As previous analyses have shown that structural material, such as the stainless steel in end fittings, tends to act as a neutron poison, k_{eff} would be reduced if the structural material was considered.

Therefore, based on the results of the Reference 5 analyses and the conservatisms mentioned above, it can be concluded that the containers will be critically safe (i.e., $k_{eff} \leq 0.99$) when one or more containers are submerged in water and the containers contain water that has a boron concentration of at least 4350 ppm.

When the containers are within the canister transfer shield (CTS), the lead and steel walls of the CTS will act as an additional neutron reflector, tending to increase k_{eff}. To demonstrate that the containers will be critically safe when within the CTS, the analyses of References 5 and 22 are used. In Reference 22, a 65-cm (25.6-inch) thick lead shell is applied to the outside of the core region. The resultant increase in k_{eff} was 0.03% Δk. Though this analysis was completed for a boron concentration of 4950 ppm, the increase in k_{eff} is expected to be similar for a 4350 ppm boron concentration. When this increase is added to the k_{eff} of 0.9896 calculated for the core at 4350 ppm (Reference 5), the resultant k_{eff} is still below 0.99. Consequently, it is demonstrated that the entire core will remain critically safe after the addition of a 65-cm thick lead reflector.

Additional conservatisms to be considered when this result is applied to containers within the CTS include:

- Both the Reference 5 and 22 models included the entire core. Additionally, the CTS walls are significantly thinner (approximately 6.5 inches) than the 25.6-inch shell. With the containers being significantly smaller than the core and the CTS walls thinner than the modelled shell, the containers within the CTS will experience significantly more neutron leakage, thus reducing k_{eff}.

- The Reference 5 and 22 models did not consider structural material; whereas the majority of the containers inventory will be comprised of structural material. As previous analyses have shown that structural material, such as the stainless steel in end fittings, tends to act as a neutron poison, k_{eff} would be reduced if the structural material was considered.

- The actual boron concentration of the RCS will be administratively maintained ≥4950 ppm. Technical Specifications require the water in spent fuel pool "A" to have a minimum concentration of 4350 ppm boron; however, it is expected that the fuel transfer canal and spent fuel pool "A" will be operated at approximately 4500 ppm boron. Any concentrations greater than 4350 ppm will cause a reduction in k_{eff} below the value calculated in Reference 5.

- The Reference 5 value of k_{eff} was determined assuming the presence of an eight-inch stainless steel reflector on the outside of the core; thus, the k_{eff} value of 0.9896 already takes credit for some reduced neutron leakage.

Since it has been demonstrated that the entire core will remain critically safe after the addition of a 65-cm thick lead reflector and that additional conservatisms tend to reduce k_{eff} even further, it is concluded that the containers will remain critically safe when within the CTS.

Finally, the effect of the storage of these containers on the neutron multiplication of defueling canisters located in the storage racks was evaluated. The positioning of these containers near defueling canisters must not cause the k_{eff} of a defueling canister to exceed the canister licensing criteria. This is accomplished by separating the containers from any defueling canister by at least one empty storage cell. An individual debris container need not be separated from another debris container by the one empty cell as the 4350 ppm boron concentration ensures that the containers' k_{eff} will remain below 0.99.

4.3 Boron Dilution

The RCS temperature and chemistry will not be affected significantly by defueling; hence, boron solubility will remain essentially unchanged. The only way the RCS boron concentration can be changed in an uncontrolled manner during defueling is by dilution of the RCS coolant with water that is either unborated or borated below 4952 ppm.

Potential sources of this water are the various systems connected to the RCS, including the secondary system. Systems which potentially contain coolant with boron concentrations less than 4950 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 ≥4350 ppm. Reference 6 provides an evaluation of 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 defueling to ensure that the RCS boron concentration will remain ≥4350 ppm.

The use of the ultra high pressure pump of the abrasive/water jet cutting system for reactor building decontamination activities as well as in-vessel cutting operations presents a boron dilution safety concern not specifically addressed in Reference 6. The supply water to the pump during cutting operations is borated to ≥4950 ppm. During reactor building decontamination operations, the pump is supplied with water from a low boron (or unborated) water source. Consequently, when the decontamination water source is connected to the pump, the pump must not be used for in-vessel operations to avoid introducing low boron water into the fuel regions of the reactor vessel. In addition, when changing from the decontamination mode of operation to the in-vessel cutting mode, the residual low boron water in the pump must be purged prior to inserting the cutting tool into the fuel region.

In order to eliminate the above mentioned safety concerns, the low borated water supply hose is coupled permanently to the decontamination nozzle hose and the ≥4350 ppm borated water supply hose is permanently coupled to the abrasive/water jet cutting tool hose. A common, mating fixture is attached to the pump supply and discharge lines. To further assure that the low borated water source will not be used for in-vessel cutting, the set of hoses for decontamination activities will have a permanent tag attached to the connecting fixture which warns plant personnel not to use that set of hoses for in-vessel operations. Operation of the ultra high pressure pump for either in-vessel or decontamination activities will be performed only after verification by procedure that the proper hose pair have been connected to the pump.

The ultra high pressure pump will contain approximately five gallons (according to the pump manufacturer) of low borated water at the cessation of the decontamination mode of operation. Upon starting the pump for in-vessel cutting operations (after disconnecting the decontamination hoses and connecting the abrasive/water jet cutting hoses) this residual water may be injected into the reactor vessel. Reference 6 summarizes evaluations which have been performed to address the issue of mixing within the RCS volume. It is stated in the reference that unborated water entering the reactor vessel at the elevation of the hot and cold leg piping is likely to rise directly to the internal indexing fixture (IIF) rather than flow down to the core region.

Administrative controls will require that the abrasive water jet cutting nozzle be discharged in a radial direction above the reactor vessel flange and away from suction lines to other systems within the reactor vessel for a minimum of five minutes (pump flow rate is approximately 2.8 gpm and residual water in pump is approximately 5 gallons) after the pump has been used for decontamination activities. This will minimize the potential for the low borated water to enter the debris bed or RCS processing systems prior to mixing with the RCS water.

The water in the RCS is maintained at a boron concentration of $\geq$ 4950 ppm and must be verified to be at this level prior to the use of the ultra high pressure pump for in-vessel cutting following the use of the pump for decontamination activities. The introduction of five gallons of unborated water to the upper region of the reactor vessel during purging of the residual pump water will have a negligible affect on overall RCS boron concentration.

4.4 Release of Radioactivity

The radiological impact of radionuclide releases to both the FHB and reactor building atmospheres, as well as to the environment, from defueling activities has been evaluated. During 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 defueling systems and activities do not introduce potential liquid effluent release pathways.

The tasks associated with the preparations for 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 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 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 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 defueling of the reactor vessel or be decontaminated (e.g., flushed), as required for radiological control, as they are removed from the water. The canisters and defueling tools are designed, where practicable, to be easily decontaminated. This minimizes the potential for fuel 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 defueling. Prior to transfer, the canisters may be partially dewatered in the reactor vessel to ensure that the catalytic recombiner is uncovered. This action will minimize pressure buildup which could lead to the opening of the canister relief valves. If the relief valves lift, particulates may be released. The most likely release point is the FTC or spent fuel pool "A" water. Other defueling activities that may result in releases of particulates to the FTC or spent fuel pool "A" include the removal of fuel canister heads to perform gasket replacement and the use of vented canisters or of debris containers. If particulate release occurs, the 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 alternate water cleanup system, will be used, as necessary, to keep contamination to acceptable levels. In the unlikely event that a relief valve lifts during the time in which the canister is within the CTS, the resulting release is expected to be minimal. Additionally, these 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 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 or other equipment 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 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 defueling activities (e.g., 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 defueling activities. The maximum annual average meteorological dispersion coefficient (X/Q) is $2.27x10^{-6}$ 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 air dose 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 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 a drop from heights exceeding the drop heights for canisters in the reactor building, with only minor deformation and no observed cracking (Reference 12). Therefore, the bottom head of the canister would not be expected to crack or rupture.

- o 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.

- o The top portion of each canister contains the most likely leakage path. 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.

- o 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. Therefore, 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 particles larger than 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.

It is noted that this accident scenario (i.e., entire canister contents spilled out on dry surface) represents a worst case accident. For other canister accidents (e.g., stuck open relief valve) the amount of the canister's content released would be smaller, and thus, the corresponding offsite doses would be smaller.

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 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 system dilutes gases that are released or evolved during defueling activities before they are released into the reactor building. Any hydrogen evolved during defueling will be diluted by the off-gas treatment system as required and thus, will not reach a combustible concentration in the reactor building.

While the canisters or debris containers 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. Subsequently, the hydrogen could be released via vent openings or the relief valves. Any hydrogen released will be to either the FHB or reactor building depending on the transfer or 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 or debris containers.

The combustion of hydrogen within the canister transfer shield is not expected to occur. The canisters or debris containers 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 or debris container 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. Therefore, 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 defueling would be conducted with water coverage. It was concluded that zirconium hydride would not ignite under water. Since defueling operations will be conducted under water, GPU concurs that there is no potential for a pyrophoric zirconium or uranium/zirconium alloy reaction in the reactor vessel during defueling.

The concern over pyrophoric materials is presently focused on the potential for metallic zircalloy, U/Zr alloy (from chemical interaction of UO_2/Zirc 4 clad), 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. However, the possibility exists that some form of the metallics may be encountered during the defueling operations involving the lower regions of the core (e.g., shredding, sawing, water jet abrasion). Zircalloy, being a ductile metal even after irradiation, did not break up into small particles under the high temperature steam environment of the TMI-2 accident. Rather, the material oxidized, and it is the oxide which will break up as a consequence of thermal shock or abrasion. However, during the defueling process, it is possible, as a result of cutting operations, that fresh (i.e., unoxidized) metal or alloy 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 the upper core debris indicates only about 1.5 % of the particulate matter and an even lower percentage of lower core debris is less than 50 microns. The abrasive water jet and sawing operations may generate a distribution of particle sizes which could create some increase in the amount of particles in the 50 micron range. The amount of these particles will be dependent on the extent of these operations. However, all of these mechanical operations occur under water where any new surfaces rapidly oxidize and the corresponding reaction energy is readily dissipated. It should further be noted that use of the shredder will not generate a significant amount of particles in this size range.

(2) The general criteria for metal and alloys to exhibit pyrophoric reactions involve: 1) newly created surfaces with high surface-to-volume ratios in an oxygen free environment being rapidly exposed to an oxygen environment, or 2) storage of high surface-to-volume metallics under conditions of favorable heat transfer, presence of organics, and a pilot ignition source; neither of these criteria are anticipated during defueling. Further exposure of pyrophoric material with high surface-to-volume ratios due to the defueling activities would initially be under water, where oxidation would again occur. The rate at which this oxidation will occur has been determined to be extremely high, thus the resultant time for all newly created surfaces of a reactive metal to be oxidized will be small when compared to the time required for canister loading and transfer operations.

(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, pilot ignition tests have been conducted on a sample of material removed from the plenum upper core debris and lower reactor vessel debris to determine their potential for pilot ignition. Attempts were made to "pilot ignite" these subject materials 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 planned defueling activities or postulated accidents, are such that it is not possible 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

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. Handling of the hydraulic shredder is specifically addressed in Reference 24.

4.8 Fire Protection

Fire Protection during the defueling activities is 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 is available for defueling. This ensures that the potential for and consequences of a fire are minimized.

4.9 Decay Heat Removal

During 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 314'-0" has been performed (Reference 4). The results of this best estimate analysis show that loss to ambient cooling will maintain the RCS bulk temperature at less than 170°F during defueling activities. The video system lighting and other defueling equipment will add additional heat to the reactor vessel water; however, it is not considered a safety problem, as the lights or other equipment can be turned off to eliminate heat input should the water temperature increase to unacceptable levels. The RCS water temperature is monitored in accordance with TMI-2 Technical Specifications and the Recovery Operations Plan.

4.10 Use of the Core Bore Equipment

A safety evaluation of the use of the core bore equipment is provided in Reference 23.

4.11 Reactor Vessel Integrity

The only postulated failure mechanism which could result in the draining of the reactor vessel water is damage to the incore instrument tubes or nozzle welds. Damage to an incore instrument tube outside the vessel would occur by pulling on an incore instrument string if the instrument string could impart excessive loading to the instrument tube. An incore instrument tube nozzle could be damaged by directly or indirectly imparting a load to the nozzle. Failure of either a nozzle weld or instrument tube could result in unisolable RCS leakage.

During defueling activities, structurally intact incore instrument strings are not expected to be found in the loose rubble; however, it is recognized that they may be found in other regions of the vessel. An evaluation was performed to assess the effects of applying limited loads to the incore instrument strings within the reactor vessel. This evaluation is described below.

Reference 25 reports the results of a thermal-hydraulic and structural analysis of the reactor vessel lower head during the TMI-2 accident. The results of the thermal analysis are not factored into the structural analysis and consequently they are relatively independent of each other. However, the thermal analysis shows that "the upper part of the incore instrument nozzle does reach temperatures well above its melting point because it is surrounded by hot material and has no cooling except for conduction down the nozzle and into the lower head. The half inch or so of the nozzle above the surface of the lower head provides sufficient cooling to keep this welded portion of the nozzle from melting." The report further shows that "although a significant portion of the lower head could be 1600°F or hotter" temperatures significantly higher are unlikely because the ultimate strength of the base material at these elevated temperatures is insufficient to prevent head failure. Obviously gross head failure did not occur since the vessel has demonstrated significant pressure retaining capability. Therefore, it is concluded that melting of any significance in the lower head and distortion of same are highly unlikely.

If, however, the material in the lower head were sufficient to have melted the upper portion of the incore nozzle, logic would indicate that the incore instrument string had also melted. This is confirmed by electrical resistance data which indicates that many of the incore thermocouples in the center of the core presently terminate at or near the reactor vessel wall. It would, therefore, appear highly unlikely that incore instrument strings remain above the CSA lower grid in the center 50% of the core cross-section.

Reference 25 also developed a structural loading criteria based on a calculation of the minimum wall thickness in various segments of the lower head needed to support the pressures experienced in the RCS during the TMI-2 accident. At a metal temperature of 1600°F, a 0.030-inch thick incore nozzle to vessel weld would support the 2000 psi pressure experienced during the accident. Note that this weld, to support pressure, is only required to bridge an axial gap between the nozzle O.D. and the vessel wall hole I.D. of 0.005-inch. Obviously little metal is required in such a small annulus to support the pressure.

A significant conservatism in the analysis is the choice of 1600°F as the maximum weld temperature considered in the structural portion of the report. This temperature was chosen because it represented the highest temperature for which approved material ultimate strength data existed. It is, however, approximately 1000°F below the temperature required to melt inconel 600 (i.e., 2540°F). As an example of the conservatism noted above, inconel at 1900°F is expected to exhibit an ultimate strength of approximately 7000 psi. Therefore, if the weld reached 1900°F, the thickness would have to be approximately 0.058 inches thick.

Assuming a conservative temperature of 1600°F, the following loads represent what the 0.030-inch thick weld could now sustain at room temperature as reported in Reference 25:

a. Axial Force - 5400 pounds
b. Bending Movement - 1400 inch-lbs.
c. Twisting Torque - 5800 inch-lbs.

Higher loads could be calculated for greater thickness (i.e., 0.058 inches at 1900°F).

A review of the potential loading methods which could transmit loads from the incore instrument string to the incore nozzle has been performed and it was concluded that only axial tension loads could be transmitted from the incore instrument strings above the CSA to the incore nozzles below the flow distributor. Based on this limited potential, the GPUN material laboratory in Reading, Pennsylvania was directed to experimentally determine the tensile load needed to break a new incore instrument string. Three tests were performed, all of which resulted in incore string breakage at loads of 3800 to 3950 pounds. Thus, the strings should break prior to resulting in damage to the incore instrument nozzle weld.

The application of an upward axial load to an incore instrument string within the reactor vessel could result in a compressive load to the inside radius of the incore piping and a bending movement to the pipe. The potential loads were determined to be low when compared to loads in the original design.

To minimize the potential for damage to the nozzle welds, the vacuum equipment to be used in the lower head has been designed to minimize the force that can be imparted to the guides tubes or nozzles.

Based on the above, it is not expected that the incore instrument tubes or nozzle welds will be damaged during defueling such that unisolable RCS leakage occurs. However assuming damage to occur, an analysis has been performed (Reference 19) which demonstrates that sufficient makeup water would be available, following the simultaneous shearing off of all 52 incore nozzles at the inside vessel wall, such that the core debris would remain covered with borated water. As it is considered extremely unlikely that the defueling operations could damage all 52 nozzles and/or tubes, the analysis is considered to provide a conservative limit on any RCS leakage, via the incore instrument tubes, that may be experienced during defueling.

5.0 Radiological Considerations

5.1 External Exposures

All individuals entering the reactor building are monitored for external radiation exposures according to established radiological controls procedures. All external radiation exposures are maintained within the dose equivalent limits established in 10 CFR 20. All personnel exposures are maintained as low as is reasonably achievable (ALARA). Administrative dose limits are applied according to established procedures to ensure that the 10 CFR 20 limits are not exceeded. Extremity monitoring is performed, as needed, according to existing radiological controls procedures.

Radiation exposure rates inside the reactor building are monitored during defueling operations. The Radiological Controls Department determines the requirements for radiation monitoring for personnel protection during defueling.

5.2 Internal Exposures

All individuals entering the reactor building are 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 are maintained as low as is reasonably achievable and within the limits established by 10 CFR 20. Airborne radioactivity in work areas is monitored according to established procedures. Air sampling for particulates is performed using such devices as lapel samples and grab samples. Tritium grab samples are 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 determines if the use of respiratory devices is ALARA for a particular task. This determination is based on 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.

During defueling activities the ranges of airborne radioactivity concentrations encountered through March 31, 1986 have been:

Isotope	Reactor Building Concentration (µCi/ml)
Cs-137	8E-11 - 5E-10
Cs-134	6E-11 - 3E-10
Sr-90	5E-11 - 3E-10
H-3	2E-8 - 3E-7
Pu-239 & Pu-240	5.2E-16 - 4.9 E-15

It is expected that the concentrations will remain in or about these ranges for the remainder of the defueling operations.

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 wiped/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" or the direct venting of canisters and/or debris containers to 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

Table 5.3-1 provides the jobhours and corresponding person-rem expended during defueling through March 31, 1986. Table 5.3-2 shows the estimated jobhours and person-rem for the remainder of the defueling activities. Included in the estimated jobhours are installation, operation, maintenance and decontamination and removal of the defueling equipment. The estimated person-rem at the different locations were evaluated using the average dose rates during defueling activities through March 31, 1986. The dose rates are:

Location	Dose Rate (mrem/hour)
305' elevation	75
347' elevation	50
Canister handling bridge (in containment)	20
Canister handling bridge (in FHB)	2
Defueling platform	10

Taking the totals from Tables 5.3-1 and 5.3-2 shows that the total estimated occupational exposure to complete the defueling operations is approximately 1400 person-rem. As more training and experience with actual defueling activities is gained, these estimates will be reviewed and updated when necessary.

5.4 Measures Taken to Reduce Occupational Exposure

5.4.1 ALARA

During the design of the defueling tools and the planning of defueling activities, the principles of ALARA were considered. In studying the alternatives for 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 are 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 reduces the time needed to complete a task. The higher radiation areas are identified to personnel and the work is structured to avoid these areas to the extent practical. Practice sessions are 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 is 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 is utilized. The long-handled tools are representative of the actual tool lengths to be used. In general, extensive training of workers on the DTA and other mock-ups is used to familiarize the workers with the tasks to be performed. This training results in increased worker efficiency thus, less in-containment time and less personnel exposure is 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 defueling tools are assembled and tested outside of the reactor building, then disassembled during the mock-up training. Where practicable, shielding is provided, where required, and all tools are fabricated of approved materials 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 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 defueling operations, the control station is located on the auxiliary work platform in the south end of the canal.

5.4.2 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 defueling and considering various shielding configurations to minimize the radiological impact of these sources. The analyses were performed to ensure that 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 defueling. Operations during defueling are governed by actual measured radiation dose rates. Adequate precautions, such as shielding or personnel relocation, are 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.

- 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, activation products, and actinides was distributed throughout the 93.1 metric tonnes of UO_2 which comprised the total initial fuel load. The core inventory at shutdown was predicted by the computer code ORIGEN-2 (Reference 21). The predicted core inventory of cobalt-60 was adjusted to reflect B&W specific cobalt-59 impurity levels in structural materials. The predicted core inventory of cesium-137 was reduced by the number of curies removed by water processing, and the inventory of cesium-134 was adjusted to reflect actual cesium-134 to cesium-137 ratios. No retention of noble gases was assumed. The core inventory was decayed to October 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. The free volume in the canister was assumed to be water for canisters in the reactor vessel and was assumed to be air (i.e., canister dewatered) for canisters being transferred.

- Reactor Coolant Source Term. The reactor coolant in the IIF and reactor vessel contains dissolved radioactive materials. The water level was assumed to be at 327'-6". Various concentrations of radioactivity were evaluated to estimate dose rates for the start of defueling and after water processing has reduced cesium concentrations to an equilibrium level. Radioactive particulates suspended in the water were not considered.

- Fines/Debris Vacuum System Source Terms. The fines/debris vacuum system is designed to remove fines/debris 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 defueling systems, a structured program of independent verification was instituted during the early design stage. Shielding requirements for major 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 defueling systems shielding components. Dose rates were calculated for normal defueling operations. Note that all dose rates are from the sources described and do not include background radiation.

For purposes of these analyses, normal defueling operations were assumed to entail manual defueling with the entire tool slot in the rotatable platform unshielded. During normal operations it was assumed that all areas of the work platforms were occupied. 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 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 has vertical shielding attached which will extend from the canal floor to the platform elevation. This vertical shield essentially spans the width of the refueling canal and is 2-inch thick steel.

- 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 defueling operations. The platform is 3-inch thick steel plate.

- Auxiliary Work Platforms. The auxiliary work platform on the south end of the canal is constructed of 1-inch thick steel shielding or equivalent. The dose rates to operators is limited to approximately 2 millirem/hour from sources in the vessel during normal 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 on the north end auxiliary work platform from sources in the reactor vessel to less than 1 millirem/hour during normal operations.

- Unshielded Sources. During normal operations, workers are positioned along the open tool slot in the rotatable platform. Dose rates to operators from the reactor coolant were calculated assuming predicted concentrations of key radionuclides which were based on anticipated water processing activities. A dose rate of 11 millirem/hr at 3-feet above the work platform at the open tool slot was calculated, assuming the following reactor coolant concentrations: 0.03 μCi/ml cesium-137, 0.1 μCi/ml antimony-125, and 0.007 μCi/ml cobalt-60. These were the expected concentrations after approximately one month of processing reactor coolant through SDS on a 5 day cycle. 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.

- Canister Shielding. If needed to minimize the dose rates to operators from canisters in the CPS, a 4-inch thick steel shield plug will be placed on fuel canisters after they are loaded and placed in the topmost canister position. In addition, a 1-inch thick steel collar was incorporated into the canister sleeve design in the CPS. The filter canister support structure design incorporates vertical shielding as well. This steel shield collar extends approximately 2 feet down from the top of the canister source region. These precautions limit the maximum contribution from all canisters in the vessel to approximately 3 millirem/hour at the open tool slot.

- 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 postulated deep end water concentrations equivalent in dose rate to 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 estimated dose rate 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 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 consists of a fixed mast cylindrical shield of 2-1/2 inch thick lead, and a 9 foot long sliding collar of 1-1/2 inch thick lead. The collar will be used to ensure that the entire canister is shielded during all transfer operations.

  When the CTS is positioned over the work platform, a total of 4-inch thick lead shielding extends approximately 9-feet above the work platform. Dose rates to operators on the work platform were calculated as approximately 47 millirem/hour at a location 3-feet from the CTS from direct radiation from the canister being transferred. The dose rate drops to 42 millirem/hr at a distance of 7-feet from the CTS and to 23 millirem/hr at a distance of 14-feet from the CTS. Because the CTS does not have a shielded bottom, there will be an unshielded beam of radiation directed downward from the bottom of the CTS during canister transfers. Because of this, there is a potential for increased dose rates on the defueling work platforms due to radiation scattered from surfaces in the path of this beam. Of the potential scattering surfaces, the worst case dose rate consequences were from scatter from the 6-inch thick steel rotatable work platform. The 4-1/2 inch gap between the bottom of the fixed mast and the work platforms provides the pathway for scattered radiation to reach personnel locations. The maximum dose rate due to scatter at 3-feet above the platform was 13 millirem/hour at a distance of approximately 2-feet from the CTS. At 7-feet from the CTS the scattered dose rate was 9 millirem/hour at 3-feet above the work platform.

  For a short time during the lowering of a canister into the deep end of the canal, a part of the canister is shielded only by the 1-1/2 inch thick lead CTS collar. Dose rates on the north end auxiliary work platform are calculated to be as high as 560 millirem/hour close to the CTS during this short time period. Routine access to the north end auxiliary work platform during the lowering of canisters into the deep end of the canal is not permitted. The maximum dose rates on the service platform during any transfer operation were calculated as 120 millirem/hour at the north edge and 39 millirem/hour at the reactor vessel center line. The maximum dose rate on the south end auxiliary work platform during transfer to the deep end was 15 millirem/hour.

The CTS also incorporates a 3-inch thick lead shield plug above the top of the canister. Calculated dose rates to the operators on the canister handling bridge are at most 5 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 is at most 24 millirem/hour during any transfer operation. 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 canal floor areas under the platforms is permitted during canister transfers.

- Shield Boots. A vertical shield extending down from the work 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 are of 5-inch thick steel and extend 2-feet into the reactor coolant. This shielding limits whole body dose rates along the open slot to approximately 25 to 50 millirem/hour. Due to interferences with CPS or vacuum system piping, some sections of the boot are less than 5-inch thick steel and some sections do not extend the full 2-feet into the water. These cases have been evaluated individually to ensure that dose rates are maintained in the ranges calculated for the complete boot. These dose rates are calculated for canister transfer from the CPS. Transfers of filter canisters would result in similar dose rates.

- Single Canister Support Bracket. 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 is 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 are 6-inch thick steel and are 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 approximately 12 to 25 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 100 millirem/hour or less. 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, 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 defueling. The canister source terms were based on 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 reactor coolant source terms reflected the calculated concentrations of soluble radioactive materials based on expected water processing scenarios before and after the start of defueling. Defueling with operators working directly over the open slot is constrained by the dose rates present and will be temporarily interrupted, if necessary, if the reactor coolant concentrations (e.g., resulting from a crud burst) cause dose rates to increase above acceptable levels. A range of water concentrations at the start of defueling was evaluated to provide predictions of dose rate variability due to different water processing scenarios. The scenarios are:

Scenario 1. No processing prior to the start of defueling
Scenario 2. One processing for cesium, antimony and cobalt removal
Scenario 3. Same as Scenario 2, with one additional processing for cesium removal.

In addition, an equilibrium water concentration was calculated assuming Scenario 2 prior to the start of defueling. This equilibrium concentration was calculated for one month after the start of defueling, assuming that the reactor coolant was processed through SDS on a 5 day cycle. The dose rates at 3-feet above the work platform along the open tool slot for these cases are given below.

	Concentrations (μCi/ml)			Dose Rate
	Cs-137	Sb-125	Co-60	(mrem/hr)
Scenario 1	0.39	0.19	0.013	46
Scenario 2	0.20	0.10	0.007	24
Scenario 3	0.09	0.10	0.007	15
Equilibrium	0.03	0.10	0.007	11

Opening the T-slot increases the dose rates to operators working at that location. At 3-feet above the platform, and assuming the equilibrium water concentrations, the dose rate increase due to opening the T-slot is 4 millirem/hour, for a total of 15 millirem/hour from the reactor coolant.

Early in the defueling planning, the Radiological Controls Department developed dose rate goals for the defueling effort. These dose rate goals were given as average millirem per hour for specific work locations. The dose rate goals included background radiation and transit doses. The dose rate goals were:

- o 30 millirem/hour on the reactor building canister handling bridge
- o 15 millirem/hour on the defueling work platform

To ensure that the defueling systems were adequately designed to permit achievement of the dose rate goals, it was a design parameter that the dose rate contribution from the defueling systems be limited to the following values:

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

The calculated dose rate at the open slot due to sources in the vessel are dominated by the contribution from radioactivity in the reactor coolant. Only a small fraction of the total dose rate was calculated to come from canister or vacuum system sources. The maximum dose rate at the open tool slot, assuming the equilibrium reactor coolant concentrations, was 13-15 millirem/hour at 3 feet above the platform. This only slightly exceeds the dose rate design parameter of 12 millirem/hour.

The dose rates on the work platforms were calculated to be approximately 2 millirem/hour or less during normal operations, which meets the dose rate design parameter. The maximum dose rate to operators on the canister handling bridge trolley during a canister transfer was 5 millirem/hour from the canister in the CTS. This meets the dose rate design parameter of 10 millirem/hour.

The maximum dose rate at 7-feet from the CTS during canister transfer was calculated as 51 millirem/hour, including direct and scattered radiation. This exceeds the dose rate design basis of 15 millirem/hour. However, to meet the average personnel dose rate goals, operators may be temporarily relocated on the work platform area during certain transfer operations.

A significant effort was expended throughout the design process to ensure that defueling systems will provide adequate radiation protection for operators and will result in the lowest reasonably achievable collective dose for defueling operations. Finally, it is noted that to date the overall exposure to workers has been less than what the above calculations predicted.

6.0 Impact of Defueling on Plant Activities

The major potential impact of defueling on plant activities is 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 defueling accident in the FHB, within the scope of this safety evaluation, is a canister drop. 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 are 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 would be no airborne particulate 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 would be within acceptable limits as demonstrated in section 4.4.2.1.

It can therefore be concluded that 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 defueling activities for TMI-2 and routine refueling activities at a typical commercial 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 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 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 or debris containers underwater in the reactor vessel. The canisters/debris containers are designed to ensure subcriticality during all phases of defueling. Once fuel has been loaded into the canisters or debris containers, the defueling process is very similar to the handling of a typical fuel bundle. Therefore, defueling activities at TMI-2 are comparable to typical refueling activities described in the TMI-2 FSAR and the consequences of postulated 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 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, this event 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 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 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 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 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 defueling SER are comparable to and bounded by similar events postulated and analyzed in the TMI-2 FSAR. Generally, the consequences of the postulated 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 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 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 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.

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

The variety of postulated events analyzed in this SER consider the spectrum of event types which potentially could occur during the defueling process or as a result of the defueling process. A comparison of these events with those comparable events in the FSAR demonstrates that the type events postulated for the 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 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.

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

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 defueling process. Subcriticality is ensured by establishing the boron concentration at greater than 4350 ppm during the 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 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 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 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 Environmental Assessment

The defueling activities have been assessed and it is concluded that these activities will be performed with no unacceptable consequences to the health and safety of the public or workers.

Releases to the public resulting from planned defueling activities are not expected to be significantly different than releases during previous work conducted in the reactor building (see Section 4.4). Past releases of radioactivity to the environment have been well within the limits of the TMI-2 Environmental Technical Specifications. Specifically regarding the potential tritium release, Section 4.4.1.2 explains that the releases to the environment will not increase over past releases. Section 4.4.1.3 provides a rationale for assessing the potential release of krypton-85 and presents the offsite dose consequences of a sudden release of krypton-85 to the environment. These dose consequences are less than 1% of 10CFR50, Appendix I dose limits, which is considered acceptable. Therefore, the releases to the public resulting from planned defueling activities will not result in a significant environmental impact.

Two accidents with potential offsite dose consequences have been evaluated. These are an instantaneous release of all unaccounted for krypton-85 (Section 4.4.2.1) and the dropping of a defueling canister (Section 4.4.2.2). In both cases the analyses were performed using extremely conservative assumptions in order to provide bounding results. Using the conservative assumptions, the results were found to be within past analyses that have been found to have acceptable consequences. The krypton-85 release was found to be less than 1% of the 10CFR100 dose guidelines for accidents. The environmental impact for this postulated release of 31,300 curies of krypton-85 would be less than that evaluated for the decontamination of the reactor building atmosphere presented in NUREG-0662, "Final Environmental Assessment for Decontamination of the Three Mile Island Unit 2 Reactor Building Atmosphere," (Appendix C to Reference 20). The conclusion of NUREG-0662 was that no significant environmental impact would result from the release of 57,000 curies of krypton-85 to the environment. The canister drop resulted in doses that were less than a fuel handling accident as described in the TMI-2 FSAR.

The environmental impact from planned defueling activities resulting from occupational exposure has been reevaluated by the NRC in Supplement No. 1 to Reference 20. This evaluation estimates an occupational exposure in the range of 2600 to 15000 person-rem for reactor disassembly and defueling. The projected occupational exposure for the planned defueling activities is expected to be within the range given in Supplement 1, if not lower (see Section 5.3). Thus, the environmental impact resulting from occupational exposure during the planned defueling activities is bounded by Reference 20.

Therefore, the planned defueling activities will be performed with no significant environmental impact.

9.0 Conclusions

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 defueling activities can be performed without presenting undue risk to the health and safety of the public.

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