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4410-85-L-0248
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December 31, 1985

TMI-2 Cleanup Project Directorate
Attn: Dr. W. O. 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
Safety Evaluation Report for
Core Stratification Sample Acquisition, Revision 3,
Response to NRC Comments on Safety Evaluation Report for
Core Stratification Sample Acquisition, Revision 1

Attached for your review and approval is Revision 3 to the Safety Evaluation (SER) for Core Stratification Sample Acquisition activities. This revision includes responses to portions of the NRC comments on Revision 1 of the SER and proposes increasing the depth of the core bore activity down to and through the flow distribution plate. A discussion of reactor vessel integrity and revised man-rem estimates has been included.

Also attached are GPU Nuclear's responses to the NRC comments on Revision 1 of the SER. NRC comments were previously provided by NRC Letter NRC/TMI-85-095, W. O. Travers to F. R. Standerfer, dated November 22, 1985.

Sincerely,

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

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FRS/RBS/eml

Attachments

ATTACHMENT
(4410-85-L-0248)

Response to NRC Comments on Core Stratification Sample Acquisition SER, Revision 1

1. COMMENT:

Describe your control program that assures that a length of drill string long enough to reach the lower vessel head is not available. This should include the case where a drill hole is abandoned due to a broken bit.

RESPONSE:

The length of drill string inside containment at any time is administratively limited to no more than required for a single core bore of a depth not exceeding the maximum depth limit (i.e., the fully extended drill string will be no less than 6 inches from the reactor vessel inner wall). Additional sections of drill pipe will be allowed into the reactor building only after previously used sections have been disposed of properly. This disposal may include placement in a defueling canister or direct removal from the reactor building. In addition, the drill rod and casing pipe storage racks have been designed to limit the amount of pipe that may be stored at the drilling platform, so that extra pipe is not available.

In the event that a drill bit breaks, the hole will not be "abandoned" until all sections of the drill string, including broken pieces, are removed from the hole. Consequently, a broken drill string will not impact the drill string length control program. If for some unforeseen reason, a portion of the drill string cannot be removed, the next core bore will not begin until the drill string storage rack at the drilling platform has been filled and all other lengths of drill string have been disposed of and/or accounted for.

2. COMMENT:

What load restrictions (i.e., torsional, horizontal and vertical force limits) will be placed on the core bore equipment to ensure that incore instrument nozzles will not be degraded. What is their bases?

RESPONSE:

No load restrictions will be placed on the core bore equipment specifically for protection of incore instrument nozzles. The basis for this is provided below.

The core bore operation will exert a downward force on the core region debris bed and on the core support assembly. This downward force is automatically controlled and will not exceed 10,000 pounds based on the operational limitations of the core bore equipment. This downward force cannot be imparted on the lower reactor vessel head incore instrument nozzles unless there is a direct, solid link between the drill bit and nozzle. Since none of the drill locations will be directly over an incore nozzle, this link can only be created by debris. To ensure that incore nozzle loading will be precluded, the depth of core bore will be limited such that the bit will not pass into the rubble bed in the lower vessel head region. The determination of maximum drill depth will be based on TV camera viewing of the lower head region immediately below the drill path prior to the start of drilling at a particular location. Consequently, the drill bit downward force can only be exerted in the rubble above the flow distribution plate where the force will be distributed to the core support assembly and is unlikely to impart a load on the nozzles.

If the drill string/bit were capable of "catching" an incore instrument string and wrapping the string around the drill bit as it rotates, a stress could be imparted to an instrument nozzle or to the instrument tube below the vessel lower head. This type of event is not considered credible for the following reasons:

a. The drill bit/string configuration is such that there is no feature which could grab and hold an instrument string.

b. Each core bore will be centered over a fuel assembly which has no instrument string. If an adjacent instrument fuel assembly collapsed into the path of a core bore, the bit would drill through the assembly and sever the string. The only other drill bit contact with an instrument string would have to be with a "loose" string from an adjacent fuel assembly location. The instrument strings in an intact core are contained within an instrument tube in the center of a fuel assembly. It is not considered feasible that the surrounding fuel assembly and instrument tube could disintegrate or melt, thus exposing the instrument string, without the destruction of the instrument string.

3. COMMENT:

Please provide details of drilling platform and actions including:

a. Load and load distribution of platform

b. Location of supports and contact points

c. Dynamic effect of drilling action

d. Total weight of drill rig

RESPONSE:

The Drill Indexing Platform Structure Assembly is comprised of three major subassemblies identified as the Wing Assembly (consisting of 4"x4"x1/4" structural tubing), the Upper Level Assembly (consisting of

6"x6"x1/4" structural tubing), and the Lower Level Assembly (consisting of 4"x4"x1/4" and 6"x6"x1/4" structural tubing). Drawing 419932 in Volume II of EGG-TMI-6824, "TMI-2 Core Stratification Sample Project System Design Description Drawings" shows these assemblies. The structure assembly was evaluated for structural integrity using four different load cases with each load case assuming three different indexing positions for a total of twelve load combinations. Table 1 shows the loads in common to the twelve load combinations. This total load is 22, 400 pounds. Table 2 shows the additional loads caused by drilling actions as described in the table. Table 3 gives a summary of the twelve load combinations. Figure 1 graphically presents the locations of the applied loads. The maximum primary stresses to the main structural members are shown in Table 4 and are shown to be much less than the AISC allowables.

TABLE 1

WEIGHT LOADS FOR ALL LOAD COMBINATIONS

Label	Description	Location of Application*	Magnitude
-	Drill Indexing Platform Structure (3400 lb total)	(Distributed)	6" tube: 40 lb/ft 4" tube: 15 lb/ft
W1	Drill Indexing Platform Left Wing (1250) plus 2 operators (200 each)	El. 337.833	1650 lb
W2	Drill Indexing Platform Right Wing (1250)	El. 337.833	1250 lb
W3	Drilling equipment center of gravity Drill Unit (3535) Tilt platform (2200) Underwater structure (2650) Drill roller platform (760) Casing (300) Drill tube, core barrel, bit (550) Middle and top clamp (350 & 275) Bottom clamp (130) Hydraulic control panels (300)	El. 337.833 24 inches from 'P'	11,050 lb
W4	Cask (4700) and cask roller platform (350)	El. 333.886 below 'P'	5050 lb

*NOTE: Weight W3 and W4 have three alternate locations relative to point 'P' as specified by dimension "X." (See Table 3 and Figure 1).

TABLE 2

ADDITIONAL LOADS FROM CORE

DRILLING (REACTION LOADS ON BIT)

Case	Label	Description	Location of Application*	Magnitude
1.		Initiation of drilling #1	El. 305	
	F1	Anticipated force on bit, x-dir.& vert.	below 'P'	1118 lb
	M1	Anticipated moment on bit		100 ft-lb
2.		Initiation of drilling #2		
	F2	Anticipated force on bit, z-dir.& vert.		1118 lb
	M1	Anticipated moment on bit		100 ft-lb
3.		Bit jams while pushing		
	F3	Maximum force on bit		17,500 lb
	M2	Maximum moment on bit		5200 ft-lb
4.		Bit jams while pulling		
	F4	Maximum force on bit		10,000 lb
	M2	Maximum moment on bit		5200 ft-lb

*NOTE: All reactions have three alternate locations below point 'P' as specified by dimension "X." (See Table 3 and Figure 1)

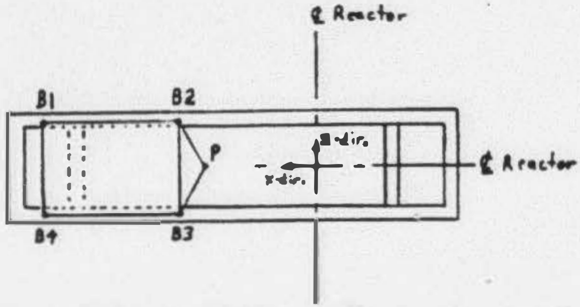
TABLE 3

SUMMARY OF LOAD COMBINATIONS

Load case (all cases include weight loads)	Indexing Position of Point 'P' (Location specified by dimension "x")		
	x = 1.01 ft	x = 2.02 ft	x = 4.05 ft
1.	1A	1B	1C
2.	2A	2B	2C
3.	3A	3B	3C
4.	4A	4B	4C

FIGURE 1

LOCATIONS OF LOADS



Note: Bolts B1, B2, B3, and B4 are fixed relative to point 'P'. The location of 'P' is specified by dimension 'x'.

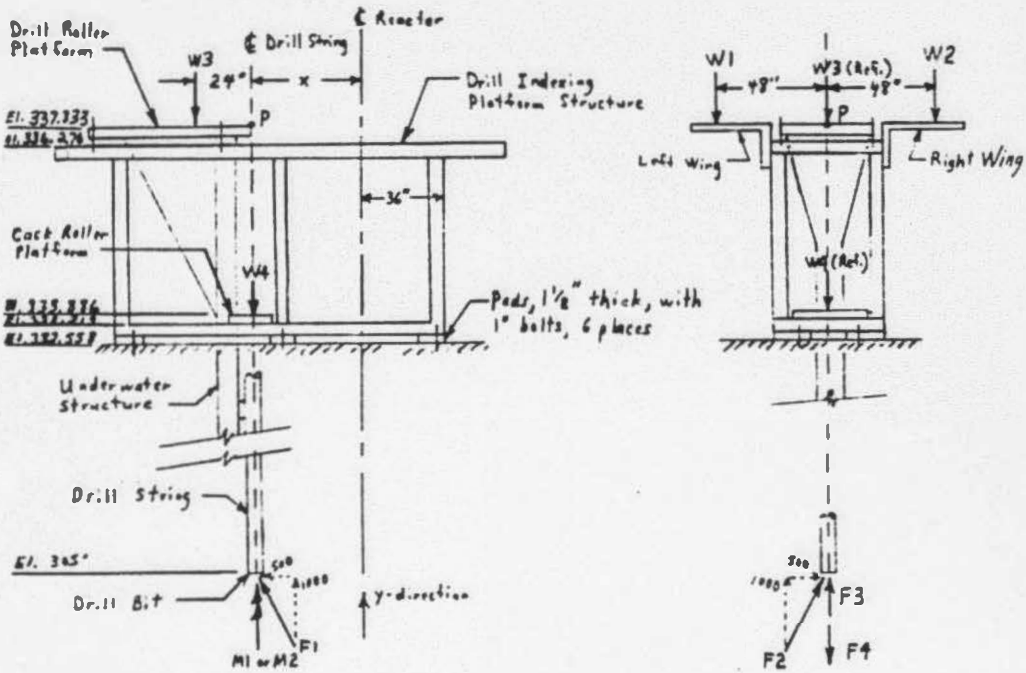


TABLE 4

MAXIMUM PRIMARY STRESSES AND THE ALLOWABLE
STRESSES FOR THE MAIN STRUCTURAL MEMBERS

	6"x6"x1/4" TUBING		4"x4"x1/4" TUBING	
	MAXIMUM	ALLOWABLE	MAXIMUM	ALLOWABLE
SHEAR (psi)	1548	14,400[a]	277	14,400[a]
BENDING (psi)	5122[b] (z)	23,760[c]	2409[d] (z)	23,760[c]
	656[b] (y)	23,760[c]	1967[d] (x)	23,760[c]
COMPRESSION (psi)	130	21,255[e]	2490	19,723[e]
STRESS LEVEL	0.249	< 1[f]	0.310	< 1[f]

NOTES:

a - 0.4 x Fy, Fy = 36,000 psi for A36 Steel, AISC, "Manual of Steel Construction", Eighth Edition 1980, Appendix, Table 2, Pg. 5-73

b - Bending Moment About Z-Axis and Y-Axis

c - 0.66 x Fy

d - Bending Moment About Z-Axis and X-Axis

e - AISC, "Manual of Steel Construction", Eighth Edition, 1980, Table C1.8.1; Eq. (2.a.1-1) & Eq. (1.5-1)

f - Ibid, eq. (1.6-2)

4. COMMENT:

Please provide analyses and calculations to substantiate that:

a. Existing structure can take the platform load and operating forces;

b. Reactor vessel can take the operating forces; and

c. Accidents in operation will not damage the system.

RESPONSE:

a. The Defueling Work Platform (DWP) is positioned over the Reactor Vessel and is supported from the refueling canal floor by the Shielded Support Structure (SSS). The DWP consists of a circular beam approximately 17 feet in diameter with interior cross beams as shown in Figure 2. The platform circular and cross beams are welded girders made of 304 stainless steel. The platform is supported by 22 wheel assemblies that ride on a rail mounted on the SSS shown in Figure 3. The SSS consists of frame work made of ASTM A36 carbon steel resting on four columns as shown in Figure 4.

Structural analyses have been performed to evaluate the structural integrity of the DWP and the SSS. The analyses found that the load case shown in Figure 5 with the shield collar at Location 'A' (Denoted as "Case 2") to be the most limiting for the defueling equipment cases. All cases considered the live load of 100 p.s.f. (total of 24,300 lbs.) and the platform shielding weight (total of 81,101 lbs.). Two cases were considered for the core drilling equipment. The core drilling cases considered the total live load and the total platform shielding weight along with the loads denoted as "A", "D", and "E" shown in Figure 5. Each core drilling case also includes the loads shown in Table 1 of the response to Comment 3 applied at the Reactor Vessel centerline. The two core drilling cases differ by the loads imposed on the platform during core drilling operations. Core drilling Case 1 assumes a 14,000 ft-lb moment across the working slot at the Reactor Vessel centerline to evaluate the reaction loads on the bit described as Cases 1 and 2 in Table 2 of the response to Comment 3. Core drilling Case 2 assumes a 10,000 pounds downward force applied at the Reactor Vessel centerline to evaluate the reaction load on the bit described as Case 4 in Table 2 of the response to Comment 3.

Table 5 shows the results of the three loading cases described above. All results are within the design limits for 304 stainless steel as taken from the ASME Boiler and Pressure Vessel Code Section III, Division 1 - Appendix XVII. The nodes identified in Table 5 are depicted in Figure 6. Table 6 shows the structural design criteria used for the DWP and the SSS. The allowable stress for all the components given in Table 5 (with the exception of the filter canister support beam) is 20,000 psi. The allowable stress of the filter canister support beam is 18,000 psi.

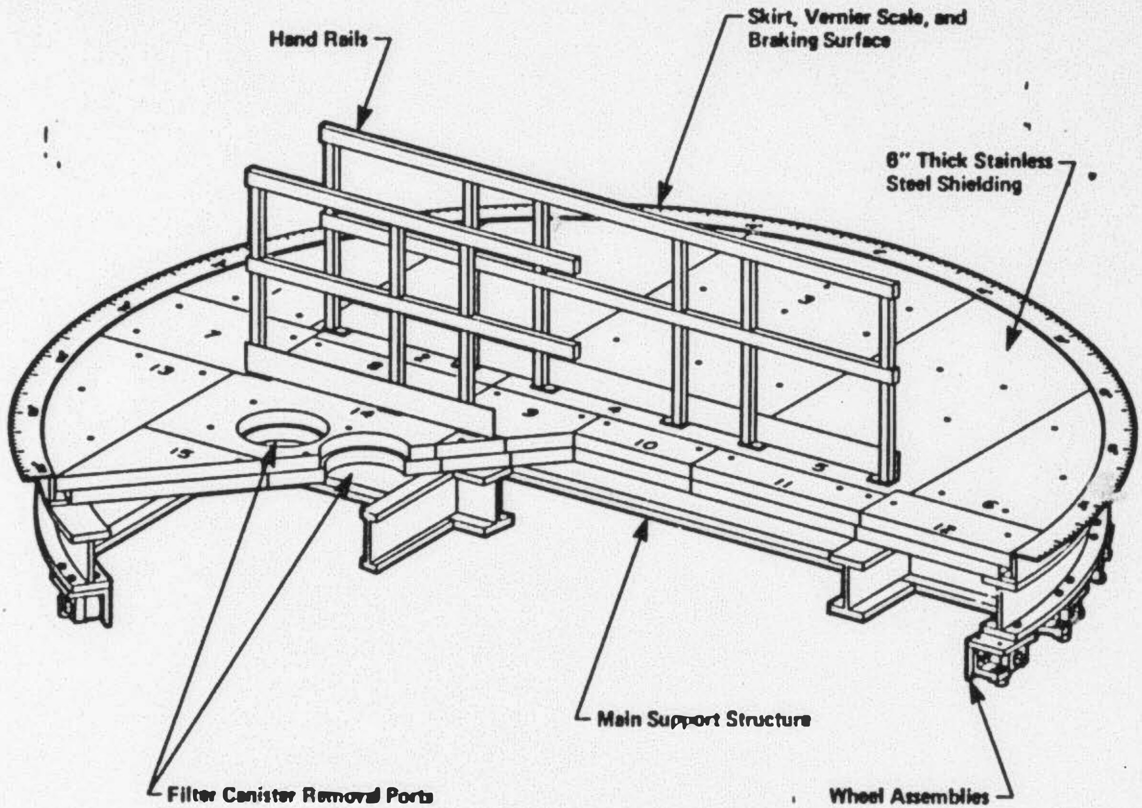


FIGURE 2 SHIELDED WORK PLATFORM

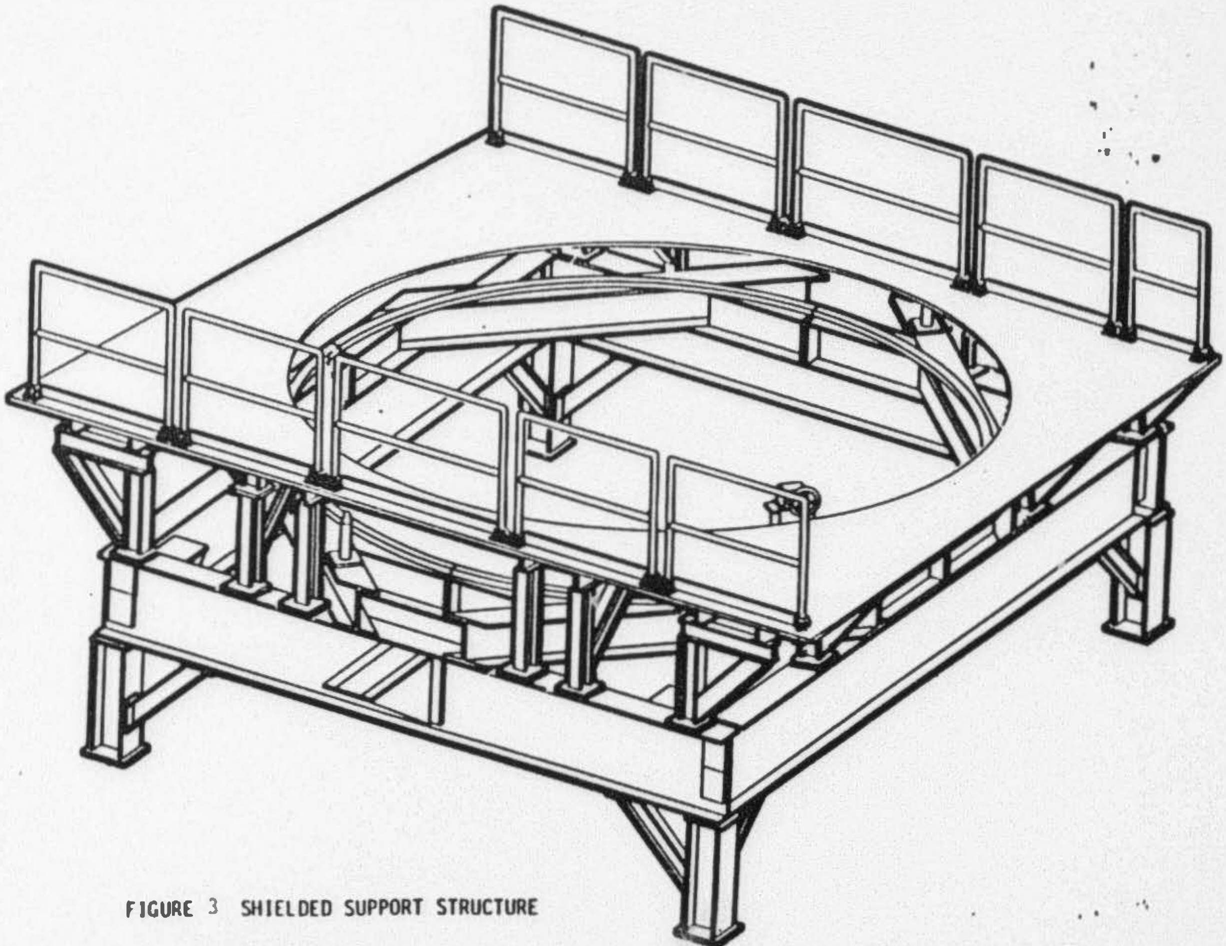


FIGURE 3 SHIELDED SUPPORT STRUCTURE

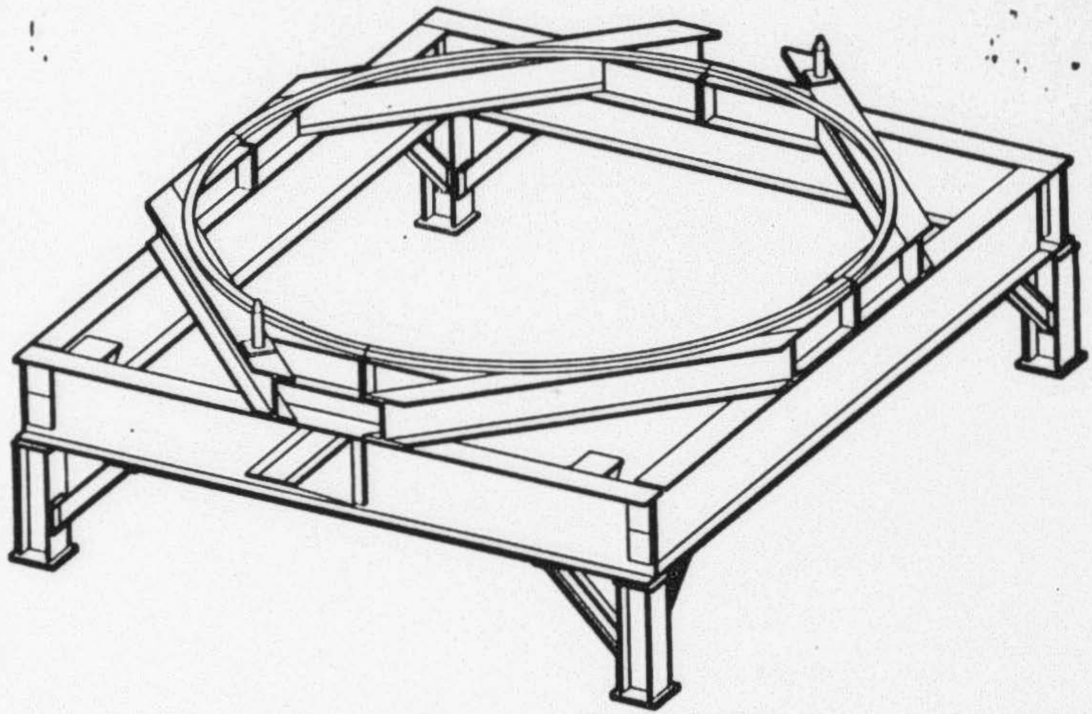
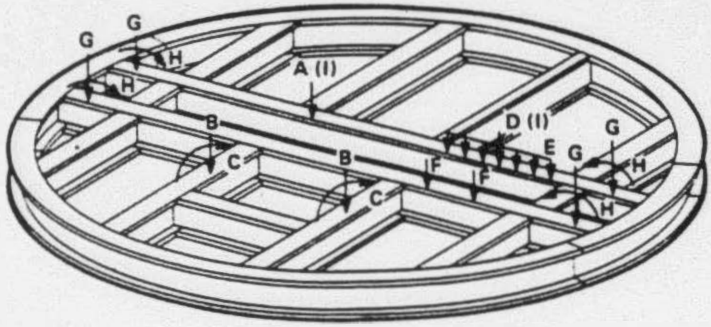


FIGURE 4 SUPPORT STRUCTURE



A. Canister Positioner (24000 lb) + Vertical Shielding (9086 lb)
B. Tool Positioner (7500 lb)
C. Tool Positioner Reaction (5000 ft lb)
D. Filter Canisters (7000 lb)
E. Vertical Shielding (10055 lb total)
F. Tool Rack (2500 lb)
G. Jib Crane Weight (750 lb)
H. Jib Crane Moment (8000 ft lb)
I. Shield Collar (10000 lb) - (Can be at Location A or D)

FIGURE 5 WORK PLATFORM LOADING CONDITIONS UNDER NORMAL OPERATION (LOAD CASES 1 and 2)

FIGURE 6 WORK PLATFORM MODEL WITH NODES LABELED

ATTACHMENT
(4410-85-L-0248)

TABLE 5

MAXIMUM STRESS RESULTS FOR THE WORK PLATFORM

	Component	Nodes	Max. Stress (σ_{bv}) Defueling Equipment (Case 2)	Core Drilling Case 1	Core Drilling Case 2
A.	Main Cross Beam #1 (Canister Positioner)	92-93	14,886	14,593	14,220
		91-92	14,201	14,989	14,520
		90-91	13,353	15,456	14,869
		89-90	12,435	15,851	15,147
		88-89	11,014	15,857	15,153
B.	Main Cross Beam #2 (Tool Positioner	106-107	8,665	6,579	6,669
		107-108	8,138	6,740	6,801
		108-109	9,090	6,866	6,895
		109-110	9,063	6,916	6,914
		110-111	8,960	6,916	6,914
C.	Tool Positioner	131-132	1,081	605	609
		132-133	1,139	716	722
		133-134	1,123	754	762
D.	Filter Canister Support Beam	726-727	1,127	1,139	1,134
		727-728	1,122	1,139	1,133
		728-88	920	939	934
E.	Jib Crane Supt. Beams	820-821	10,936	12,242	11,725
		718-83	9,935	11,040	10,571
		717-718	9,746	10,821	10,361
		822-823	11,937	12,223	11,987
F.	Circular Beam	2-3	2,146	2,076	2,042
		17-810	2,608	2,582	2,521

TABLE 6

STRUCTURAL DESIGN CRITERIA

	WORK PLATFORM 304 SST	SUPPORT STRUCTURE A36 STEEL
TENSION	$F_t = 0.6\ F_y$	$F_t = 0.6\ F_y$
SHEAR	$F_v = 0.4\ F_y$	$F_v = 0.4\ F_y$
BENDING	$F_b = 0.66\ F_y$ (compact sect.)	$F_b = 0.66\ F_y$
COMPRESSION	$F_a = 0.6\ F_y$ (consider buckling)	$F_a = 0.6\ F_y$ (consider buckling)
BEARING	$F_p = 1.5\ F_{ult}$ (for bolts)	$F_p = 1.5\ F_{ult}$ (for bolts)
BOLTS	ASTM A325	ASTM A325
-tension	$F_{bt} = F_{ult}/2$	$F_{bt} = 44$ ksi
-shear	$F_{bv} = 0.62\ F_{ult}/3$	$F_{bv} = 21$ ksi

304 SST
$F_y = 30$ ksi
$F_{ult} = 70$ ksi

A36 STL
$F_y = 36$ ksi
$F_{ult} = 58$ ksi

A325 BOLTS
$F_y = 81$ ksi
$F_{ult} = 105$ ksi

b. Since the core bore equipment is supported by the DWP, which is in turn supported from the floor of the fuel transfer canal, the static equipment loads associated with the core bore operation are not imparted to the reactor vessel. The only significant operating force imparted to the vessel is the downward force exerted by the drill bit face. The core bore equipment is designed such that even with a single control system failure, the maximum force to the vessel will not exceed 10,000 pounds. Since the vessel has with stood pressures of up to 2300 psig following the TMI-2 accident, a force of 10,000 pounds transmitted to the vessel will not cause failure of the vessel. Damage to the reactor incore instrument nozzles is unlikely as discussed in the response to NRC comment Number 2.

c. Accidents which could cause damage to the reactor coolant system resulting from the operation of the core bore equipment are addressed in response to NRC comments 1, 2, 4b, and 5.

5. COMMENT:

Assuming failure of an incore instrument nozzle, what is the maximum leak rate? (Include comparison to makeup rate and recirculation rate.)

RESPONSE:

As discussed in the response to NRC comment Number 2, a failure of an incore instrument nozzle as a consequence of the core bore activities is considered unlikely. If, however, a nozzle fails, the resulting leak rate from the vessel will be approximately 0.4 gpm as discussed in the SER for Heavy Load Handling Over the TMI-2 Reactor Vessel (the nozzle failure mechanism as a result of loads imparted by the core bore operation will be similar to the failure mechanism which results from loads imparted by a load drop accident). This leak rate is negligible compared to the makeup and recirculation capability as described in Technical Specification Change Request No. 46, Recovery Operations Plan Change Request No. 46 approved by NRC letter dated August 8, 1985.

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SER 15737-2-G07-109 REV. 3

ISSUE DATE July 3, 1985

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TMI-2

DIVISION

SAFETY EVALUATION REPORT

FOR

Core Stratification Sample

Acquisition

COG ENG [illegible] DATE [illegible]

RTR [illegible] DATE 7/2/85

COG ENG MGR. [illegible] for R.L.R. DATE 7/3/85

NO	DATE	REVISIONS	BY	CHECKED	GROUP SUPERVISOR	MGR DESIGN ENGINEERING
3	12/13/85	Revised and Reissued for Use	[illegible]	RJR	ETS	[illegible] for RLR
2	11/15/85	Revised to Extend Scope	[illegible]	RJR	ETS	[illegible] for RLR
1	8/16/85	Revised and Reissued for Use	RJR	[illegible]	ETS	[illegible] for RLR
0	7/3/85	Issued for Use	[illegible]	RJR	[illegible]	[illegible]
				CHIEF ENGINEER		

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NO. 15737-2-G07-109

Title: Safety Evaluation Report for Core Stratification Sample Acquisition

Rev.	SUMMARY OF CHANGE
0	Issued for Use
1	Revised to incorporate comments on Revision 0 and correct minor typographical errors
2	Revised to increase depth of core bores, address Kr and Cs releases, revise man-rem estimate, clarify discussion of reactor coolant system integrity, and revise load handling requirements.
3	Revised to incorporate responses to NRC comments.

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

BECHTEL

DOCUMENT TITLE Safety Evaluation Report for Core Stratification Sample Acquisition

JOB 15737

2-G07-109

REV 3

Table of Contents

1.0 Introduction

1.1 Purpose

The purpose of this Safety Evaluation Report (SER) is to demonstrate that the activities associated with the acquisition of the core stratification samples can be accomplished without presenting any undue risk to the health and safety of the public.

1.2 Scope

The scope of this SER includes the activities associated with the installation, operation and removal of the core boring machine, acquisition of the core samples, the transfer of the samples from the machine to the defueling canisters (provided for other debris removal operations, Reference 2) and viewing the lower vessel region through the bored holes. These activities will be performed after the defueling work platform and canister positioning system have been installed. Early defueling activities are planned prior to core stratification sampling (core bore) activities but this is not a prerequisite.

1.3 Background

In order to determine the extent and distribution of the various damage phenomena of the TMI-2 reactor core, removal and examination of core samples are desirable. Evidence collected and analyses performed since the accident indicate that the reactor core damage includes cladding oxidation, fuel liquefication as a result of decay heat from fission products, control rod failure and relocation of control material, interaction of AgInCd and stainless steel materials with fuel and cladding, and fragmentation upon quenching (Ref. 8). The core samples will provide a basis for establishing a correlation between these phenomena and fission product release from the fuel and fission product retention in the core. The removal of the core samples will also provide information to determine the maximum temperature reached and material interactions.

Finally, the core drilling will be used to gain access to the space below the reactor core support plate for remote visual inspection.

1.4 Organization

Section 2.0 of this SER describes the major activities and equipment needed for the Core Stratification Sample Task. In addition this section addresses the core samples, drilling locations, core boring operation, and the failure modes and effects analysis (FMEA).

Section 3.0 describes the radiological considerations of the planned activities including an assessment of the expected external and internal occupational radiation exposure. It also describes shielding considerations and other steps taken to maintain the occupational exposure ALARA.

Section 4.0 addresses the safety concerns associated with the planned activities including a 10 CFR 50.59 evaluation.

Sections 5.0, 6.0 and 7.0 contain the environmental assessment, conclusion and references for this SER, respectively.

2.0 Description of Activities

The activities that must be conducted prior and subsequent to the core stratification sampling, as well as the actual core sample drilling and removal, are described in this section.

2.1 Support Activities

Extensive operational testing prior to actual operation will be conducted under simulated conditions to assure successful core sample acquisition. A "dummy" fuel assembly has been constructed for these tests to verify drilling capabilities in the fuel assembly structure.

Concurrent with this testing, TMI personnel will be trained in the operation of this equipment. This training may include conducting planned operational steps in respirators and anti-C clothing where needed to obtain accurate time estimates and operational constraint information in order to project a reasonable ALARA review. These training sessions will also be used to verify that the operating procedures will result in safe equipment operation.

All equipment necessary to obtain the core sample will be transported into the reactor building through personnel airlocks. The core drilling equipment, support structure and interface platform will be staged and transported into the reactor building on the 305' level. Final assembly and staging of equipment will take place on the 347' level prior to hoisting the assemblies into place on the Defueling Work Platform (DWP). The Defueling Water Cleanup System will be operational prior to and during the core stratification sampling.

2.2 Core Samples

Core samples will be obtained from three or more locations and each sample will have a maximum design length of 8 feet. The first stage of each core bore will include boring down to the lower grid assembly with the drill bit centered on a fuel assembly which has a vacant instrument tube (i.e. not containing an incore instrument). This initial stage of drilling will provide the desired core sample but will also attempt to provide a pathway for TV camera access (after removal of the sample) to the vessel region below the core. If after boring to the lower grid assembly, the TV camera cannot be lowered beyond the lower grid assembly due to obstructions, a second deeper core bore of a reduced diameter will be made down through the flow distributor plate. It will be stopped prior to reaching the rubble in the lower head and/or will drill no closer than 6 inches to the vessel lower head. This second stage of core bore may produce a second sample as a byproduct of the drilling operation.

2.3 Drilling Locations

Figure 2.3-1 shows the candidate reactor fuel assembly locations within the TMI-2 core. Figure 2.3-1 shows the drilling location within a fuel assembly. It is planned that drilling locations will be selected from reactor fuel positions K-9, D-4, N-12 and other alternative locations shown in Figure 2.3-1 in fuel bundles not containing instrumentation strings (Note: later core data or evaluations could change the locations from those given here). For the second stage of core bore for camera access, the hole will align with the corresponding existing inspection access path through the lower grid assembly and flow distributor and will provide access for a 2.0 inch diameter cylindrical instrument (TV) viewing package.

2.4 Core Drilling Equipment Description

The following equipment will be used for the core drilling operation: the core drilling machine, core barrel and drill bit, drill indexing platform, core sample transfer cask, underwater support structure, underwater clamp assembly, flush water supply tank, inert gas purge equipment, sample storage basket assembly, drill unit instrumentation, and video inspection system. The configuration of most of this equipment is illustrated in Figures 2.4-1 and 2.4-2. These are briefly described below. Additional equipment details are provided in Reference 10.

a. Core Drilling Machine

The drill unit is a self-contained machine consisting of a hydrostatically driven spindle and hydraulically actuated feed cylinders. The spindle is equipped with a hydraulically actuated drill chuck in order to apply controlled rotational torque ranges and speed to the drill string. The cylinders are used to apply the downward force on the drill bit and to move the drill and casing strings into and out of the hole by moving the spindle vertically. The working fluid for this unit consists of approximately 27 gallons of HOUGHTO-SAFE 620.

The hydrostatic drive system is powered by a three phase, 480 volt electric motor which will be fed by a portable power center. The spindle is driven through a multi-speed gear system in order to provide the capability to change the speed and torque ranges to the spindle. The same electric motor will provide power to a hydraulic pump which in turn will provide power to the cylinders, the spindle chuck and to two separately mounted clamps used in handling the drill rod and casing strings. The drill unit is instrumented with a control system which will be capable of monitoring and controlling the drilling process.

The drill unit will be operated automatically but will include an override and controls for manual drilling and rod handling operating modes. Details of the operation modes are provided in Reference 10. Instrumentation will provide visual indication of rotational speed and torque on the drill string and the weight (force) applied to the drill bit. The drill unit will be equipped with a data acquisition system to record information on the material being drilled, such as rubble, solidified mass, standing fuel arrays and voids. Reference 10 describes the data acquisition system in more detail. The data interpretation will provide elevation and thickness information on the material penetrated with a goal of locating boundaries within $\pm$ 1.00 inch vertically.

The drill unit will be used as the support for a TV camera manipulator to view through the core bore hole.

The drill unit instrumentation will provide safety monitoring and control safe equipment shutdowns. The instrumentation also provides information on the drill unit stabilizing structure to alert the drill unit operator when the structure has moved excessively in the horizontal plane (side loading) during the initial drilling operation.

The drill unit instrumentation will provide input to a computer controller which will monitor the following parameters: revolution per minute (RPM), weight on drill bit, torque, depth (rate of penetration) and the water flow rate (drill bit coolant). Any deviation from pre-set value ranges for weight on bit, torque, and water flow will cause an automatic shutdown of the unit.

If the surface that the drill bit encounters is very hard and on a slope from horizontal, the drill bit may move sideways causing a side loading on the stabilizing platform. The drill unit Out-of-Tolerance Indicator will monitor the horizontal deflection and provide visual alarm indication to the drill unit operator.

There is no potential of drilling through the reactor lower head due to the physical limitation of the casing and drill bit lengths available to the operator i.e., limited rack space in the drill rod and casing rack on the defueling platform.

The core drill drive and feed equipment will be designed to mount on the Defueling Work Platform (DWP) via an interface platform. A mounting platform will be configured to permit the drill to access all of the primary and alternate target positions and will be equipped with anchoring devices to prevent dislocation of the drill once it has been positioned over a given target. The equipment will be designed to meet the load limits of the DWP and no loads other than those from the drilling operation will be imparted to the reactor vessel.

Triangulation will be used to position the drill unit within $\pm$ 0.125" horizontally prior to drilling, and to orient the drill string parallel within $\pm$ 0.05 degrees with the reactor vessel centerline as determined from measurements to the plane of the reactor vessel flange. During drilling the drill bit drift will be limited to $\pm$ 0.5" at the lower core support grid elevation.

This is a project limit only; exceeding this limit may prevent the video package from being lowered through the core support assembly, due to the location of the inspection holes, but will present no safety concern.

b. Core Barrel & Drill Bit

The drill core barrel function is to contain and protect the core sample, support the drill bit, transmit drilling forces, and channel flush water to the bit. A double tube core barrel will be used to permit the inner tube to remain stationary around the core sample while the outer tube rotates. A swivel mechanism built into the top of the core barrel allows flush water to be channeled to the drill bit which keeps it clean and cool. A series of flutes (channels) in the drill bit allows water to pass back to the reactor vessel as it is displaced by the core sample.

The overall length of the bottom section of drill pipe containing the core barrel will be limited in order for the core barrel to completely fit into the fuel canister.

The drill bit has a cast matrix crown with cutter inserts made of tungsten carbide with synthetic diamond bonded to it.

c. Drill Indexing Platform

The function of the drill indexing platform is to support the drill unit, transfer cask, and underwater structure, and to accommodate positioning of these components at the proper locations for drilling. The platform mounts to an interface platform which bolts to the rotatable Defueling Work Platform above the reactor vessel.

The Defueling Work Platform rotates about the reactor centerline and will be used to position the drill indexing platform in the approximate circumferential location for drilling.

The drill indexing platform is designed to translate horizontally (index) in both circumferential and radial directions as required for accurate positioning. The platform has two roller platforms, one platform for the drill unit and underwater structure and one platform for the transfer cask. These two platforms can be moved radially independently of each other by means of hand-cranked rack and pinion gear mechanisms. The cask can also be moved circumferentially independently of the other components by translation on slide bearings by means of a hand-operated lead screw. This will be used when positioning the cask off-center above a fuel canister when lowering a core sample or used casing section into the fuel canister mounted on the canister positioning system.

Four hydraulic jacks will be used to lift the drill roller platform off its tracks and clamp it in position. These jacks are part of the hydraulic system which is electrically powered and contains approximately 2 gallons of HOUGHTO-SAFE 620 as the working fluid. Mechanical devices are provided to back up these hydraulic devices.

The drill indexing platform structure has removable grating to allow for multiple positioning of the drill unit and has two operator access (wing) platforms with handrails.

d. Core Sample Transfer Cask

The transfer cask allows the core barrel to be moved over the fuel canister and then lowered into the canisters. The cask incorporates a water manifold to supply borated rinse water around the circumference of the casing or drill pipe for decontamination as they are withdrawn from the reactor.

e. Underwater Support Structure

Tests determined that stabilization of the drill casing is required at about the 314' elevation. Therefore, a 25 ft. 3 in. long tubular girder underwater stabilizing structure will be employed. The upper end terminates in a plate which bolts to the tilting platform. A plate at the lower end contains a tapered funnel guide and a 4.69 in. hole for guiding and stabilizing the drill casing. A casing clamp is also required approximately 4 ft. 8 in. under the reactor water level to separate the casing and allow the cask with sample specimen to be moved to a position above the fuel canister. The underwater stabilizing structure satisfies both of these requirements.

The underwater structure and tilting platform assembly will be lowered in position through, and bolted to the top of the drill indexing roller platform by the reactor building service crane or the polar crane. Alignment of the drill unit and underwater structure will be accomplished by four jacks that can be adjusted to correct the drilling axis by use of a hydraulic system. The stabilizing structure will be aided in providing the rigidity and alignment necessary for accurate drilling by use of strain gages that will alarm when the drilling axis has deflected out of tolerance.

f. Underwater Clamp Assembly

The lower casing clamp consists of a pair of diametrically opposed serrated jaws and funnel guide located approximately 4 ft. 8 in. under the reactor water level and welded to the underwater structure. The clamp is hydraulically operated from a remote position located on the drill platform. Using a demineralized water medium, a hand pump actuates a double acting cylinder which forces the jaws either closed to hold the casing section while engaging (disengaging) the adjacent threaded joint or open to allow free operations of the drill string. A relief valve is used to prevent overpressurization of the system. A manually operated tool has been designed and will be available to insert into the reactor vessel and operate the clamp should the primary clamp fail.

g. Flush Water Supply

During drilling operations, borated water will be required for flushing drill fines and cooling the bit. Borated water will also be used to rinse the drill tube and casing during removal from the reactor. This water will be provided by a standard 1/2 to 6 gpm positive displacement pump. A diversion valve will be provided so that pump suction can draw borated water from the IIF, beneath the DWP. The pump suction will be taken from an area remote from potential sources of unborated moderators. This is the primary source of flush water. A secondary source is provided by two 55-gallon flush water supply tanks, which are used to rinse the swivel supply line prior to breaking its connection to the drill string. These tanks will be filled from the BWST via the canal fill system with a verification sample taken after filling the tank.

h. Inert Gas Purge Equipment

In order to protect the core bore samples from oxygen, the transfer cask will be purged with inert gas to protect the core barrel and sample as they are removed from the reactor. This will be done by inserting a wand into the lowest section of drill tube containing the sample. As the water drains out of the drill pipe and core barrel, it will be replaced by the inert gas. After the last section of drill tube is removed from the top of the core barrel, a plug will be inserted into the top of the core barrel. The total amount of available inert gas will be limited to an amount less than could accumulate in the reactor area and cause a personnel hazard.

i. Sample Storage Basket Assembly

After the core sample is raised into the transfer cask, the sample will be transferred horizontally and lowered into a basket assembly which would be installed in a fuel canister. These canisters are on the underwater rotating carousel that will be used to position the basket below the transfer cask's path. The basket assembly is self draining and is presently designed to receive two large and two small core barrels, with samples, and two contaminated casing sections per fuel canister. The core barrels and casing sections are lowered into the fuel canister baskets by means of lifting plugs threaded into their top ends and attached by a stainless steel lanyard to an overhead bridge crane.

j. Drill Unit Instrumentation

The drill unit will be instrumented with a control system which will be capable of controlling the drilling process from control parameters experimentally determined by EG&G testing. The principle method of drilling control will be to program in a specific drill bit rpm and utilize the weight on the bit as a variable to control torque within a predetermined control band. The automatic operation can be overridden to control the drill unit manually, if required. The equipment safe shutdown feature described in section 2.4a will override both the automatic and manual operation modes.

k. Video Inspection System

In order to inspect the lower reactor head region, two cameras with cabling, a camera manipulator, and video recording system will be used. The camera head is equipped with a 16 m lens.

The camera manipulator assembly consists of a shaft to provide camera support, indexing, and rotation, a set of clamps to lock the shaft to the drill unit. a rotating head to provide radial scanning in 30 degree increments, a water connection to provide borated flushing water to clear the viewing area of floating particulate, and a camera indexing fixture to assure camera indexing to a specific reactor compass heading.

The video recording system consists of a 6 Mhz bandwidth video recorder, a video distribution system, a pair of video printers, and a video quantitizer.

2.5 Core Boring Operation

The core drilling equipment will be staged and assembled onto the DWP and prepared for operation. The locations of the core samples will be predetermined and the drill unit and rotating work platform will be positioned at the desired coordinates over the reactor core.

To complete the drill unit alignment over the sample location an electronic leveling system will be used in conjunction with the theodolite sighting system (i.e., surveying instrument). Any required adjustments, as indicated by the theodolite sighting system, will be made to the drill unit roller and tilting platforms. These adjustments will properly align the drill rig in the horizontal plane and ensure that the drill axis is perpendicular to the horizontal plane and directly over the target fuel assembly location. The platforms once properly aligned will be held rigidly in place by clamps during the operation of the drill unit.

The drill piping and casing piping sections are assembled in the drill chuck while being supported in place by the lower clamp assemblies. Core drilling starts at the top of the debris bed and continues until the lower fuel assembly end fitting is completely penetrated. If needed, the casing is then drilled down around the drill pipe until the casing shoe rests on or near the top of the lower end fitting grid. In order to prevent drilling into the reactor vessel wall (lower head), the drill pipe (string) length available on the DWP at any given time will be limited so that the drill cannot physically reach the vessel lower head.

The core drilling will be controlled from the defueling work platform. The drill unit is instrumented to provide continuous information on drilling parameters.

After the core drill penetrates the fuel assembly lower end fitting, the upper section of the casing, a minimum of one foot above the core debris, will be removed by remotely disconnecting a joint in the casing. The drill piping containing the core sample in the lower 11 foot pipe section is raised while rotating the drill string. While withdrawing the core sample the pipe and casing sections are removed until the section containing the sample has been withdrawn up into the lead transfer cask. The drill pipe sections being removed are sprayed down with borated water as they are retracted through the transfer cask (to remove loose contamination) using a spray system piped into the cask.

The drill pipe section containing the core sample is purged by an inert gas blanket while in the transfer cask and then capped prior to transfer to the fuel canister.

The transfer cask is translated horizontally to a position directly over the fuel canister which has previously been located at the loading station by rotation of the canister positioning system. The core sample pipe is then grappled by the overhead bridge crane and lowered into the fuel canister. When the canister is full, it is capped and the canister positioning system will be rotated to bring an empty canister into loading position.

After completion of the first stage of the core bore the video inspection manipulator with camera and cable will be lowered through the casing pipe to a position in the fuel assembly just above the lower end fitting. If blockage is observed, then a deeper, slightly reduced diameter (2.2") core drill will be performed to clear a pathway through the flow distributor to provide video manipulator access to the reactor vessel bottom head region. The lengths and number of drill pipe sections are controlled so that the drill string cannot reach the lower reactor vessel head. Additional sections of drill pipe will be allowed into the reactor building only after previously used sections have been disposed of properly. This disposal may include placement in a defueling canister or removal from the reactor building. The drill rod and casing pipe storage racks have been designed to limit the amount of pipe that may be stored at the drilling platform so that extra pipe is not available.

When clear of the blockage, the video inspection manipulator with camera and cable will be lowered through the casing and the drill hole to the lower head region. Then, video inspection is performed of the lower vessel head area at that location.

If inspection of the lower vessel area indicates that a grab sample is justified, existing grab sample tooling will be utilized to obtain a sample. The sample will be placed in a special sample cask for shipment.

After the core samples have been removed from the reactor and the lower head inspected, the hole in the fuel assembly end fitting will be sealed, if required, to prevent large particles from falling into the lower vessel. The casing, if used during the drilling operation, will stay in place, until the end fitting plug has been installed. The end fitting plug has a lead-in chamfer to facilitate entry into the drilled end fitting hole. The plug also has a lifting eye and a wire rope to lower it into the hole. The wire rope will be cut and retrieved after the plug is installed, prior to casing removal. The casing pipe section is then withdrawn into the transfer cask and transferred to and lowered into a fuel canister. The lowering of the casing pipe sections into the defueling canister requires the use of a crane. Other raising and lowering operations may be completed by the use of the core drilling machine. Loaded fuel canisters will be transferred to the appropriate storage locations. The fuel canister transfer will be performed in accordance with the "Safety Evaluation Report for Early Defueling of the TMI-2 Reactor Vessel," (Ref. 2).

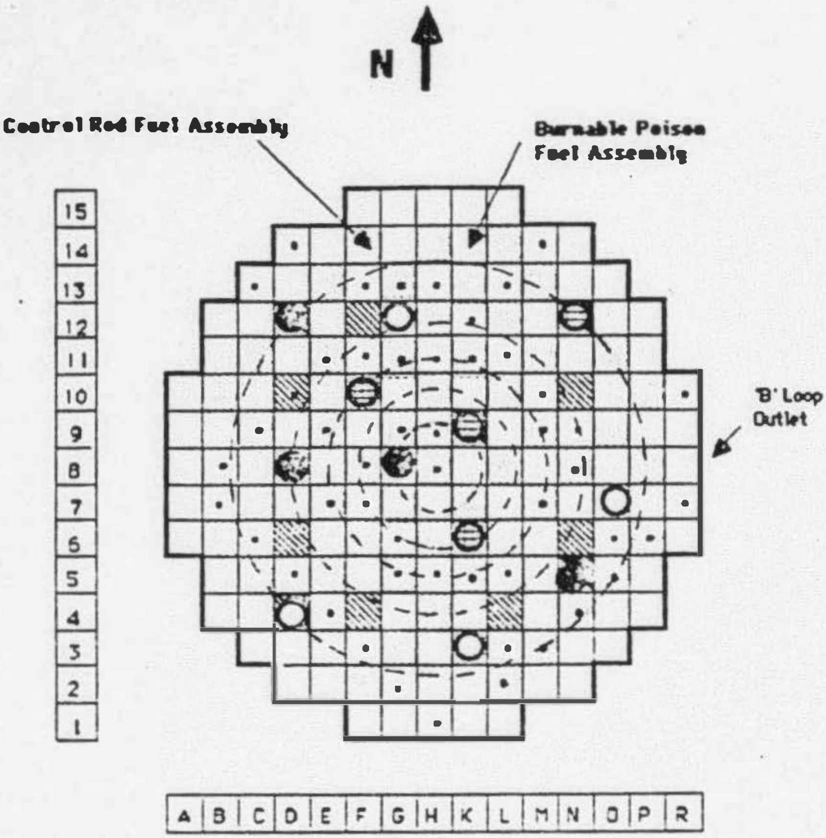
Upon completion of the core sample acquisition tasks, the core drilling equipment will be removed, disassembled and disposed of in accordance with GPU requirements for handling of contaminated and potentially contaminated materials.

2.6 Failure Modes and Effects Analyses (FMEAs)

Failure modes and effects analyses were performed for the following systems: drill unit, tilt platforms, underwater structure, drilling platform, drill unit indexing platform, transfer cask, cask indexing platform, drilling flush water, inert gas purge, appurtences, drill string casing, video examination system, drilling machine computer control, data acquisition system and theodolite system. The FMEAs are described in detail in Reference 4.

The only failure mode with safety significance is a failure which could cause the core boring unit to drill through the lower vessel head and cause loss of reactor water. The means to prevent this type of failure is to limit the length of drill bit and casing available to the operators. The drilling system design will insure that neither the drill bit nor the casing shoe can reach the reactor vessel pressure boundary. This will be accomplished by restricting core bore activities to within a cylindrical envelope. The envelope will be approximately 8'-1" in diameter and will be centered on the reactor vessel centerline. The bottom of the cylindrical envelope (i.e., at the maximum depth of core bore) will be no closer than 6" from the inner wall of the reactor vessel lower head at the outer edge of the cylindrical envelope. However, as discussed in section 4.4, the drilling operation will be administratively limited to prevent drilling into the rubble bed, if the rubble bed extends into the envelope.

Quality Control will verify the total length of the drill rod and casing pipe taken inside containment and staged on the DWP. This will ensure the total drill pipe length available to the equipment operators will be insufficient to reach within 6" of the reactor vessel wall.



High Priority
K6
K9
F10
N12

Medium Priority
N5
D12
G8
D8

Lowest Priority
D4
G12
K3
O7

Figure 2.3-1 **Core Bore Locations**

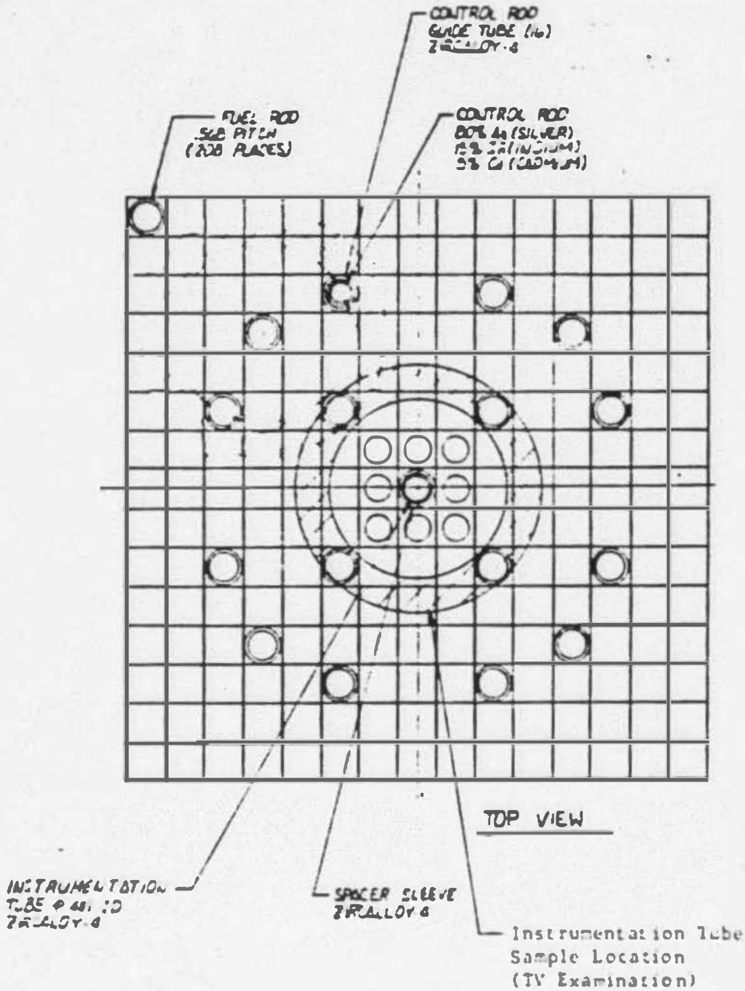


Figure 2.3-2 Fuel Assembly Drilling Locations

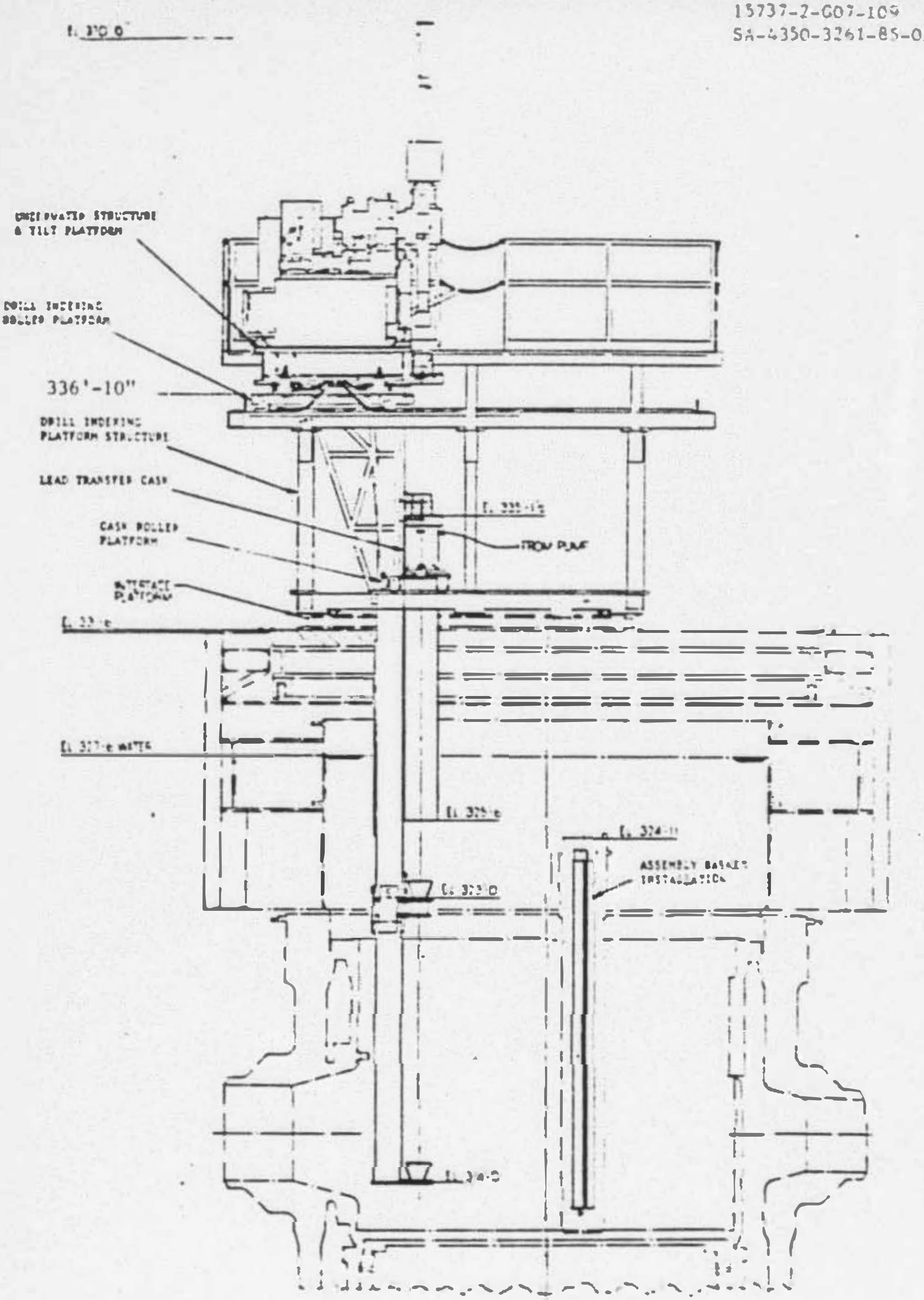


Figure 2.4-1 Core Stratification Sampling Equipment Assembly

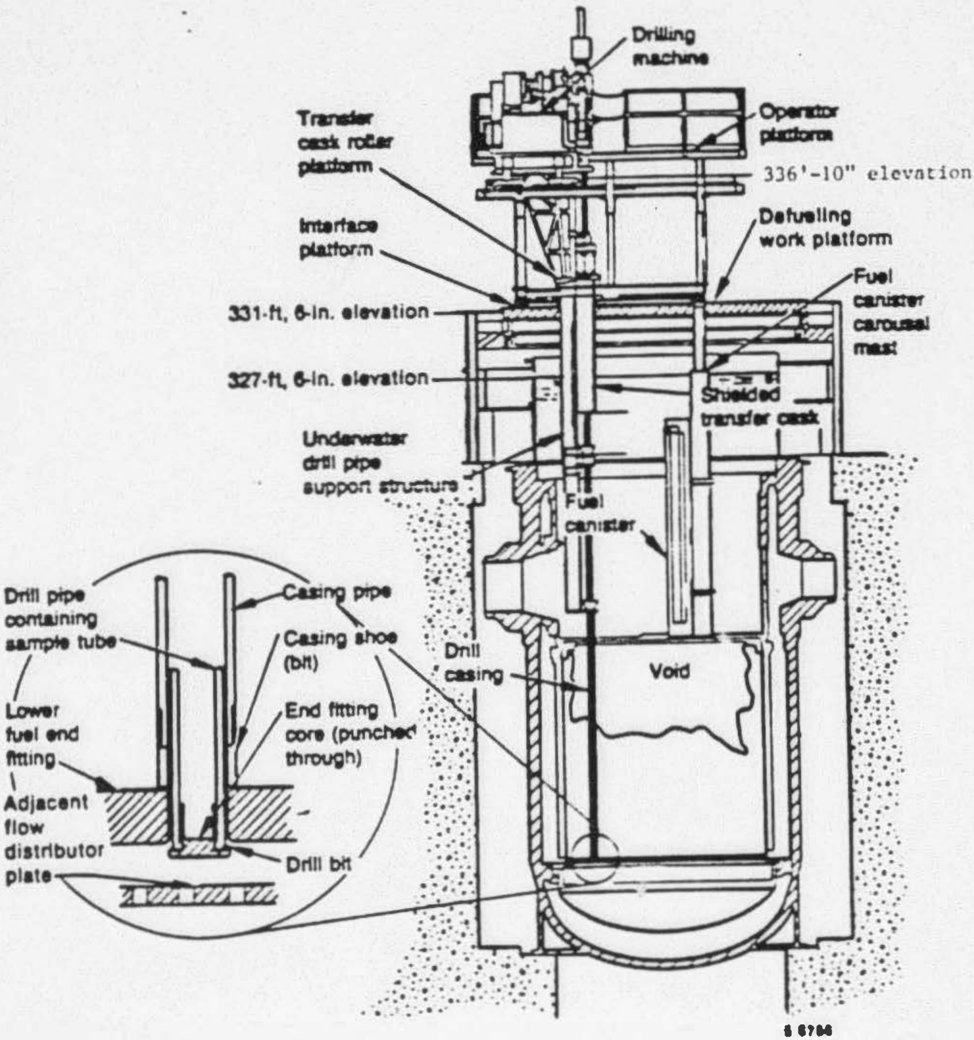


Figure 2.4-2 Core Stratification Sampling

3.0 Radiological Considerations

3.1 External Exposures

All individuals entering the reactor building will be monitored for external exposures in accordance with radiological control procedures to ensure personnel exposures are maintained ALARA and within 10CFR20 dose equivalent limits. Administrative control points in accordance with radiological controls procedures will be used in order to assure specified dose limits are not exceeded. Extremity monitoring will be performed as needed in accordance with existing procedures. Radiological Controls Department personnel will continuously monitor dose rates in the reactor building during the sample acquisition and supportive activities.

The collective personnel radiation exposure to workers during the core boring and sample transfer operations and during the supportive activities in the reactor building has been estimated. This estimate was developed based on projected person-hour requirements and reactor building exposure rates associated with these activities. The average dose rate during staging, assembly, disassembly and removal was taken as 50 mrem/hr. The average dose rate during core drilling operations and video inspection at 5 locations was estimated to be 25 mrem/hr. The collective dose is estimated below:

Activity	Dose Rate (mR/hr)	Total Hours	Man-Rem
Assembly & Disassembly	50	346	17.3
Drill Handling	25	416	10.4
Core Drilling	25	624	15.6
Totals		1,386	43.3

Person-rem for radiological controls support is not included in the above estimate. From a review of historical data it is assumed that person-rem for the radiological controls group will be 20% of that accummulated by other groups in containment. Based on this, the estimate for radiological controls support is 8.7 person-rem, and the total for all groups is estimated at 52 person-rem.

Due to the uncertainty in the person-hour estimate and the radiological conditions which will exist during the operation, it is estimated that the total exposure could vary by up to $\pm$ 30 percent. Considering these uncertainties, 35 to 70 person-rem has been selected to be used as the estimate for the performance of the activities scoped in this SER, including radiological controls support.

3.2 Internal Exposures

All individuals entering the reactor building will be monitored for internal radiation exposures according to established procedures. This monitoring will be accomplished by periodic whole body counting or bioassay, or both. All exposures to airborne radioactivity will be maintained ALARA and within the limits established in 10CFR20. Airborne radioactivity in work areas will be monitored according to established procedures. Air sampling for particulates will be performed using devices such as breathing zone air samplers and grab samples. Tritium grab samples will be taken as required according to established procedures.

Respiratory protection has been used to minimize the uptake and deposition of airborne radioactivity in the body. The use of respiratory protection devices can, by reducing uptakes of radioactive materials, result in overall dose savings (internal and external); however, if they impede work, total dose can increase by causing an elevated external dose. Current radiation protection guidance as expressed in International Commission on Radiation Protection Publication-26 (ICRP-26) considers both external and internal sources of exposure and recommends minimizing the sum of them.

For soluble cesium-137, the internal dose is 2.5 mrem (received over several years, T_{eff} 70 days) for each hour of exposure at MPC. For soluble strontium/yttrium-90, the bone dose is approximately 15 mrem (received over 50 years, T_{eff} 6400 days) for each hour of exposure at MPC. Even if there is no overall savings in the total dose due to elimination of a respirator for a given task (that is, the increased internal dose exactly offsets decreased external dose), the fact that the internal dose is calculated on a fifty year dose commitment whereas external dose is deposited instantly means that the rate of dose deposition is reduced on an overall basis.

The Radiological Controls Department, via the prework radiological review process, shall determine if the use of respiratory devices for a task is ALARA. This review will examine the current radiological conditions in the work area, the potential of the task or other concurrent tasks to perturb the radiological conditions and when available, review the results of previous airborne activity measurements in the work area for similar tasks.

3.3 ALARA Considerations

The objective of minimizing occupational exposure has been a major goal in the planning and preparation for all activities in the containment. The actions that have been taken or are being planned toward meeting this objective are summarized in this section. These actions will minimize the time personnel must work in radiation fields, maximize the distance between personnel and radiation sources to the extent practicable, and utilize shielding where appropriate to meet the ALARA objective. Protective clothing and respirators will be used as necessary to reduce the potential for external contamination and internal exposure of personnel.

Execution of individual tasks are maintained ALARA by a detailed radiological review by Radiological Engineering and mock-up training where appropriate. Training of workers on a mock-up will familiarize the workers with tasks to be performed. This training will result in less time and personnel exposure in the reactor building.

Equipment has been designed with the intent of keeping radiation exposures ALARA by minimizing in-containment assembly and simplifying operation.

3.4 Shielding

When the 3/8 inch thick stainless steel core barrel containing the sample is removed from the water, lead shielding with an approximate thickness of 3" will be used in the core sample transfer cask to minimize direct radiation levels. A radiological analysis performed for different source terms predicts exposure rates on the outside surface of the cask of 15 mrem/hr for cesium-leached fuel (Ref. 4). To prevent streaming, two temporary shielding inserts will be placed at the top of the transfer cask around the core barrel during sample transfer.

The cask incorporates a water manifold to supply borated rinse water around the circumference of casing or drill tubes for decontamination as they are withdrawn from the reactor.

4.0 Safety Concerns

To ensure that the sample acquisition operation is conducted in a safe manner, the following potential safety concerns have been evaluated.

4.1 Release of Radioactivity

The planned activities associated with the sampling operation are not expected to release any appreciable amounts of gaseous or particulate activity. Any potential releases of radioactivity are enveloped by the dose assessment performed for early defueling activities in the Safety Evaluation Report for Early Defueling of TMI-2 Reactor Vessel (Ref. 2). These analyses demonstrate that any potential release will be within allowable limits.

The planned core bore activity has the potential to release Kr-85 from the core debris and from the reactor building. The offsite radiological consequences and the measures employed to minimize exposure to in-containment personnel, from this potential release of Kr-85, are addressed below.

Kr-85 is a fission product with a half-life of 10.7 years which decays primarily (99.6%) via beta emission at a maximum energy of 0.672 MeV. The amount of Kr-85 available for release from the core cannot be quantified because the complete characterization of the core (the existence and extent of voids and intact fuel pins) is not available. The amount of Kr-85 in the core has been postulated to be as low as zero and as high as 31,300 Curies. The amount of Kr-85 entrapped within voids in the core is also speculative. Therefore, the amount of Kr-85 available for release during the core bore activities cannot be quantified.

To assess offsite radiological consequences a "worst-case" bounding analysis has been performed. This analysis is presented in the SER for Heavy Load Handling Over the TMI-2 Reactor Vessel (Ref. 6). This analysis assumes the release of 31,300 Curies of Kr-85 to the environment and shows that the radiological consequences are acceptable from an accident release perspective. The resulting dose given in the SER is 9.7 millirem to the whole body. The corresponding dose to the skin is 810 millirem using the dose conversion factors given in Regulatory Guide 1.109. The atmospheric dispersion factor used to obtain the dose of 810 millirem is 6.1×10^{-4} sec/m^3 which is given in Appendix 2D of the TMI-2 FSAR.Accident release calculations take no credit for mitigating actions by control room operators. However, if a large release of Kr-85 from the core were to occur, operator action could be taken to minimize the offsite doses. Upon indication from a portable detector or from the plant vent radiation monitor, the containment purge could be secured. Purging the containment could then be limited to only during favorable meteorology (i.e., weather which ensures a rapid dispersion) which would minimize the offsite dose due to a Kr-85 release. Based on a controlled release scenario, the expected exposure to the skin of an individual offsite would be much less than 810 millirem since the atmospheric dispersion factor would be much less than the atmospheric dispersion factor used for accident assessments. The largest atmospheric dispersion factor on an annual average basis is 2.27×10^{-6} sec/m^3 in the SE sector at the site boundary (TMI-1 Offsite Dose Calculation Manual). Thus, the highest expected dose to the skin for an individual offsite is 3 millirem from the release of 31,300 curies of Kr-85. This dose is well within the 15 millirem limit given in 10CFR50, Appendix I.

Since the amount of Kr-85 available for release during the core bore cannot be quantified, protective measures will be employed to minimize exposures to in-containment personnel. Kr-85 in the IIF air space will be shielded by the rotatable work platform. To provide indication of large releases of Kr-85, a portable detector will be located on the core bore platform or the rotatable work platform, and an alarming radiation monitor will be in place to monitor below the rotatable work platform. If significant increases of beta radiation are detected, workers will secure the core bore apparatus in a safe configuration and will vacate the reactor building at the direction of Radiological Control personnel. The need for air samples to be analyzed for Kr-85 will be determined by the Radiological Controls Department.

The core boring operation also has the potential for releasing significant quantities of fuel fines and soluble radioactive materials to the reactor coolant due to the pulverization of fuel and fuel debris as the drill penetrates these materials. These materials may then become distributed in the reactor vessel water volume, which may cause an increase in dose rates above the water surface. Because of the high density of UO_2, fuel particles will tend to settle rapidly in the water; however, soluble materials may become dispersed throughout the water volume. The defueling water cleanup system (DWCS) has been designed to provide sufficient water cleanup capability so that soluble activity and particulate levels are returned to normal within a reasonable amount of time after such release. Assuming a 20 Curie release of Cs-137, and assuming a uniform mixing in the reactor vessel, the water activity would increase by 0.18 μCi/ml. This would cause a dose rate increase of approximately 14 millirem along the open slot in the rotatable platform, assuming the entire tool slot is open. The dose rate increase would be less on the core bore platform due to the increased distance from the source.

The cesium and other soluble radioactive materials will tend to stay dissolved in the reactor coolant and will not significantly increase reactor building airborne radioactivity or airborne releases to the environment.

To minimize the potential dose rate increase to workers the following precautions will be considered:

- o Radiation dose rates on the core bore platform and/or on the defueling work platform will be continuously monitored while personnel are present during the core bore operation.

- o All shield panels in the rotatable platform will be in place, excepting those panels required to be removed for placement of the core bore apparatus.

- o The DWCS or other water cleanup capability will be operable during core bore operation.

Considering the expected releases and the resulting dose rates, these precautions are considered adequate to protect personnel in the event of soluble or particulate releases to the reactor coolant.

4.2 Criticality

The only credible means of attaining criticality of the fuel contained in the vessel is through deboration of the RCS water or introduction of "foreign" materials to the reactor vessel. The potential for boron dilution during defueling will be addressed in the "Hazards Analysis: Potential for Boron Dilution of Reactor Coolant System," (Ref. 3). The analysis for potential boron dilution during defueling will envelope the core sample acquisition activities.

Components of the core boring system which contain hydraulic fluid which could potentially cause local deboration in the core are classified important to safety to minimize the potential for failure of these components.

The main concern is leakage of hydraulic fluid from a hose break or from the reservoir attached to the core drilling machine. The two closed hydraulic systems on the drill unit contain a total of about 27 gallons of HOUGHTO-SAFE-620 hydraulic fluid. HOUGHTO-SAFE-620 is a mixture of 40% water, 40% glycol, 15% polyglycol and 5% additives which although slightly heavier than water, also is completely miscible in water. Therefore, it is unlikely that hydraulic fluid entering the top of the reactor pressure vessel could move down to the core region without being significantly diluted in the water above the core. The only pathway for the hydraulic fluid is via leakage onto the surface of the vessel water from the top of the vessel.

There is a low level sensor on the hydraulic fluid reservoir which would initiate shutdown of the drill unit within 1 second of detecting a loss of 1/2 gallon of hydraulic fluid. Therefore, leakage from a hose break would be limited to 1 1/2 gallons. A drip pan with a capacity of 8 gallons will be provided to collect minor leakage. Due to a limit in reservoir (tank) size no more than 27 gallons of HOUGHTO-SAFE-620 is available to leak into the top of the reactor vessel, even if the controls or drip protection failed to limit the leakage. A drip pan with a capacity of 8 gallons will be provided to collect minor leakage. Based on information and analyses presented in Reference 3, any leakage into the reactor vessel would be diluted by the reactor coolant.

The water used for drill bit flushing/cooling will be supplied from the reactor vessel; the flush water supply tank will be used as a secondary source. The water in the flush water supply tank will be borated to a concentration within the limits required by the Technical Specifications. To ensure that the flush water is adequately borated at the start of the core boring operation, a sample will be taken and analyzed for boron concentration. During the core boring operation the flush water supply tank will be refilled from the Borated Water Storage Tank.

A leak of hydraulic fluid from the drill unit could enter the RCS possibly causing deborated moderator to be injected as flush water into the drill bit. In order to prevent this, the relative location of flush water suction will be from an area remote from potential sources of unborated moderators and the drill's hydraulic fluid introduction will be such that insignificant deboration would take place in the event of a leak. The flush water suction line arrangement will also minimize possible deboration of the flush water in the unlikely event of deboration resulting from the wrong resins being added to the DWCS ion exchangers.

The addition of the steel drill casing and core barrel into the core region does not represent a configuration more reactive than that analyzed in the "Criticality Report for the Reactor Coolant System," (Reference 7). Additionally, the use of other "foreign" materials (e.g. cable, camera, etc.) during the sample acquisition activities will not increase the k_{eff} of the reactor coolant system above 0.99. The process used to review and control these materials is discussed in Section 4.2.1 of Reference 2.

The manually operated underwater casing clamp hydraulic system contains 1.4 gallons of demineralized water. Leakage of this small amount of water has been shown, in the foreign materials report (Reference 11), to not be a criticality hazard, even if introduced directly into the rubble bed.

4.3 Pyrophoricity

In order to avoid the possibility of a hypothetical pyrophoric reaction, core samples shall be maintained in an essentially oxygen-free environment following withdrawal from the RCS water. This will be accomplished by purging the core barrel and core sample with an inert gas as they are removed from the reactor. This will be done by inserting a wand into the lowest section of drill tube containing the sample prior to it being raised from the water. As the water drains out of the drill tube and core barrel, it will be replaced by the inert gas. After the last section of drill tube is removed from the top of the core barrel, a plug will be inserted into the top of the core barrel prior to transferring it to the defueling canister. For a more detailed discussion of pyrophoric events in the RCS, see Reference 9.

The heat generated by drill bit friction will not increase the potential for a pyrophoric event since this heat will be readily dissipated by the boring tool flush water. The drill unit will be automatically shut down upon loss of flush water.

4.4 Reactor Coolant System Integrity

Load handling activities during the equipment installation, operation and removal will be performed in accordance with the "Safety Evaluation Report for Heavy Load Handling", (Ref. 5) and with the "Safety Evaluation Report for Heavy Load Handling over the Reactor Vessel", (Ref. 6). The analyses presented in these two references demonstrate that any potential drop accidents associated with the core sample acquisition activities will not impact the health and safety of the public.

During the installation and removal of the core boring equipment, some loads to be handled have been identified which will exceed the height/weight limitations presented in Reference 6. An analysis has been performed to bound all heavy load handling during the installation and removal activities associated with the core drilling equipment. The maximum load and lift height in this analysis are a 5000 pound load being raised to a plant elevation of 339'-0". The analysis determined that this lift height is allowable if the following conditions are met:

i. The load is maintained within a 3'-6" limit from either side of the DWP support structure center line (N-S or E-W centerline).

ii. The T-slot tool positioner, tool rack, jib cranes and transfer shield-shield collar will not be on the DWP during this load handling activity.

iii. The core bore drill rig, flush water tank and hydraulic control assembly will not be installed prior to this load handling activity.

Loads less than 2000 pounds that will be handled above the DWP will be handled in accordance with the following:

$$H = \frac{17500}{W} + 331.5$$

Where H is the maximum plant elevation to which the load may be raised in feet, and W is the weight of the lifted load including the weight of the rigging which is rigidly attached to the load, in pounds. H is the maximum elevation of the lowest rigid point of the suspended load.

Any additional load handling activities that are identified that exceed the limitations of References 5 and 6, will be evaluated on a case by case basis.

The core bore operation will exert a downward force on the core region debris bed and on the core support assembly. This downward force is automatically controlled and cannot exceed 10,000 pounds based on the operational limitations of the core bore equipment (Reference 10). This downward force cannot be imparted on the lower reactor vessel head incore instrument nozzles unless there is a direct, solid link between the drill bit and a nozzle. Since none of the drill locations will be directly over an incore nozzle, this link can only be created by debris. To ensure that incore nozzle loading will be precluded, the depth of core bore will be limited such that the bit will not pass into the rubble bed in the lower vessel head region. The determination of maximum drill depth will be based on TV camera viewing of the lower head region immediately below the drill path prior to the start of drilling at a particular location. Consequently, the drill bit downward force can only be exerted in the rubble above the flow distribution plate where the force will be distributed to the core support assembly and is unlikely to impart a load on the nozzles.

The potential for core boring activities to damage incore instrument tubes outside of the reactor vessel has also been evaluated. If the drill string/bit were capable of "catching" an incore instrument string and wrapping the string around the drill bit as it rotates, a stress could be imparted to an instrument tube below the vessel lower head. This type of event is not considered credible for the following reasons:

a. The drill bit/string configuration is such that there is no feature which could grab and hold an instrument string.

b. Each core bore will be centered over a fuel assembly which has no instrument string. If an adjacent instrumented fuel assembly collapsed into the path of a core bore, the bit would drill through the assembly and sever the string. The only other drill bit contact with an instrument string would have to be with a "loose" string from an adjacent fuel assembly location. The instrument strings in an intact core are contained within an instrument tube in the center of a fuel assembly. It is not considered feasible that the surrounding fuel assembly and instrument tube could disintegrate or melt, thus exposing the instrument string, without the destruction of the instrument string.

4.5 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.

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.

The FSAR for TMI-2 evaluated a variety of Design Basis Accidents. In general each of those events represented a substantially more consequential accident than any that can be credibly postulated for boring into the TMI-2 core in its present condition.

Each of the questions for determining if core boring activities involve an unreviewed safety question are evaluated below.

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?

The uncontrolled addition of unborated water to the reactor coolant system during core boring activities is addressed in Reference 3. It describes how the potential sources of unborated water are isolated from the reactor coolant system and the measures to be taken to detect a boron dilution event should one occur.

The introduction of foreign materials into the reactor core region is discussed in section 4.2

Any release of radioactivity during core boring activities will be to the containment. During core boring activities containment isolation capability will exist and the ventilation system for the containment will be operated in accordance with the technical specifications and will, therefore, not increase the probability of an uncontrolled release of radioactivity. As for the consequences of a release of radioactivity, as stated in Section 4.1 of this report the consequences of releases of radioactivity are not increased over previous analyses present in Reference 2.

A fuel handling accident related to core boring activities that could result in a release of radioactivity to the environment is the dropping of a fuel canister onto the dry portion of the refueling canal. An evaluation of this event was performed in Reference 2. The resulting offsite whole body dose is less than the limiting fuel handling accident whole body dose presented in the FSAR.

As described in this safety evaluation, core boring activities will not adversely affect equipment classified important to safety (ITS), thus the probability of a malfunction of ITS equipment is not increased. It is also concluded that the consequences of a malfunction of ITS equipment is not increased for the following reasons:

- the results of accident analyses described in Section 4 of this report are within the bounds of previous analyses presented in the FSAR,
- removal of decay heat is passive in nature, thus no impact should DHR system fail, and
- the existing RCS boron concentration will ensure subcriticality for all credible fuel configurations, not requiring any active system.

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

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

The various types of accidents which could occur during core stratification sampling are discussed in Sections 4.1 through 4.4. None of these accidents are of a different type than those previously analyzed.

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

Technical Specification safety margins at TMI-2 are concerned with criticality control and prevention of further core damage due to overheating. As demonstrated by this Safety Evaluation Report, Technical Specification safety margins will be maintained throughout the core boring process. Criticality is controlled by establishing a boron concentration at greater than 4350 ppm during the core boring process and ensuring that this concentration is maintained by monitoring the concentration and isolating potential deboration pathways. Also, the quantity of "foreign" materials that could be introduced to the vessel will be strictly controlled. The ability to prevent further core damage due to overheating is not affected by core boring as systems will continue to be in place to add borated cooling water to the core in the event of an unisolable leak from the reactor vessel.

In conclusion, the core stratification sampling activities do not

- o increase the probability of occurence 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 core stratification sampling activities do not constitute an unreviewed safety question.

No Technical Specification changes are required to conduct the activities bounded by this SER.

5.0 Environmental Assessment

The activities associated with core drilling have been assessed and it has been 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 the planned core drilling activities are not expected to be significant. Specifically, the potential impact of an environmental release of krypton is assessed in Section 4.1. This accident is the core drill penetrating a pocket in the core filled with 31,300 Curies of Kr-85. The analysis of this accident was performed using conservative assumptions in order to provide bounding results. Using these conservative assumptions, the results of the analysis were found to be within the bounds of previous analyses that have been found to have acceptable consequences. Therefore, the planned activities will be performed with no significant environmental impact.

6.0 Conclusions

The sample acquisition, transfer, and associated activities, have been described and evaluated. The evaluations have shown that the task and equipment employed follow the continued commitment to maintain radiation exposure levels ALARA. The evaluations have also shown that no detectable increase of radioactivity releases to the environment will result from the planned activities. The consequences of postulated accidents with respect to potential core disturbances and loadings on the reactor vessel have been shown not to compromise safety. The normal and accidental releases of radioactivity have been evaluated and are bounded by the analyses presented in Reference 2. It is therefore concluded that the core sampling acquisition and the associated activities, can be performed without presenting undue risk to the health and safety of the public.

7.0 References

1. "Safety Evaluation Report for Removal of the TMI-2 Reactor Vessel Head", Rev. 5, February, 1984.

2. "Safety Evaluation Report for Early Defueling of the TMI-2 Reactor Vessel", Rev. 4.

3. "Hazards Analysis: Potential for Boron Dilution of Reactor Coolant System," document 4430-84-007R.

4. "TMI-2 Core Stratification Sampling Project, Failure Modes and Effects Analysis (FMEA)", EG&G Idaho, Inc., CSS-133, December 26, 1984.

5. "Safety Evaluation Report for Heavy Load Handling Inside Containment," Rev. 2.

6. "Safety Evaluation Report for Heavy Load Handling Over the TMI-2 Reactor Vessel", Rev. 0.

7. "Criticality Report for the Reactor Coolant System", Rev. 0, October 1984.

8. "Requirements Document for TMI Core Stratification Sampling Project", EG&G Idaho, Inc. Report No. PF-NME-84-004, Rev. 1, November 1984.

9. "Safety Evaluation Justifying the Non-Seismic Design of TMI-2 'Post-Accident' Systems", Rev. 0.

10. "TMI-2 Core Stratification Sampling Project System Design Description," EGG-TMI-6824, Vol. I.

11. "Report on Limits of Foreign Materials Allowed in the TMI-2 Reactor Coolant System During Defueling Activities", Rev. 1, 15737-2-N09-002, September 1985.