GPU 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

4410-85-L-0172
Document ID 0308A

September 11, 1985

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

Dear Dr. Snyder:

Three Mile Island Nuclear Station, Unit 2 (TMI-2)
Operating License No. DPR-73
Docket No. 50-320
Safety Evaluation Report for Heavy Load Handling
Inside Containment, Revision 2

Attached for your review and approval is Revision 2 to the Safety Evaluation Report (SER) for Heavy Load Handling Inside Containment. This revision increases the scope of the SER to include the handling of heavy loads over fuel canister storage racks in the Containment and Fuel Handling Buildings with fuel present.

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

Sincerely,

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

W/check $150.00 #00017620

F033
1/1

FRS/RBS/eml

Attachment

Enclosed: GPU Nuclear Check No. 00017620

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

GPU Nuclear

SER 15737-2-G07-105 REV. 2

ISSUE DATE October 29, 1984

☒ ITS

☐ NSR

☐ NITS

TMI-2

DIVISION

SAFETY EVALUATION REPORT

FOR

Heavy Load Handling

Inside Containment

COG ENG C. L. Rid DATE 7/12/85

RTR E. T. Smith DATE 7/11/85

COG ENG MGR. C. L. Rid for RRider DATE 7/11/85

NO	DATE	REVISIONS	BY	CHECKED	GROUP SUPERVISOR	MGR DESIGN ENGINEERING
2	8/1/85	Revised and Reissued for Use	RJR	BTE	ETS	CLR for RR
1	7/12/85	Revised and Reissued for Use	BTE	RJR	ETS	CLR for RR
				CHIEF ENGINEER		

8509160385 850911
PDR ADOCK 05000320
P PDR

EPD-33611 10/84

Title Safety Evaluation Report for Heavy Load Handling Inside Containment

Rev.	SUMMARY OF CHANGE
0	Initial Issue for Use
1	Revised to increase scope to include the handling of heavy loads over fuel canister storage racks with canisters present.
2	Revised to incorporate comments on Revision 1 and to correct minor typographical errors.

SHEET	LATEST REV	SHEET	LATEST REV	SHEET	LATEST REV	SHEET	LATEST REV	SHEET	LATEST REV	SHEET	LATEST REV	SHEET	LATEST REV
1	2												
2	2												
3	2												
4	2												
5	2												
6	2												
7	2												
8	2												
9	2												
10	2												
11	2												
12	1												
13	2												
14	1												
15	2												
16	2												
17	1												
18	1												
19	2												
20	2												
21	2												
22	2												
23	2												
24	2												
25	2												

REVISION STATUS SHEET

BECHTEL

DOCUMENT TITLE: Safety Evaluation Report for Heavy Load Handling Inside Containment

JOB 15737	REV
2-G07-105	2
PAGE 3 OF 25	

TABLE OF CONTENTS

1.0 INTRODUCTION

1.1 Background

During the TMI-2 recovery operations the lifting of heavy loads (2400 pounds or greater) is required. The hoists and cranes to be used for handling these loads include: the reactor building service crane, canister handling bridges, and other cranes and hoists.

1.2 Purpose

This Safety Evaluation Report (SER) provides a NUREG-0612 (Reference 1) evaluation of postulated heavy load drops, including a definition of load handling areas and demonstration that the effects of load drops in these areas will not reduce the margin of safety being maintained or create the potential for a criticality event within the containment or Fuel Pool 'A' (FPA) in the fuel handling building.

1.3 Scope

This SER addresses the handling of heavy loads within the containment and FPA during defueling and describes load handling areas and any necessary restrictions to be applied while handling these loads. As this SER does not address specific loads or specific load handling operations, offsite releases are only addressed generically in this SER. Additionally, rather than addressing specific load paths, this SER addresses an entire area (e.g. D-rings, hatch area, fuel transfer canal, or floor slab) as the area subject to the load drop. The results presented in this SER are based on evaluations of design drawings and calculations which determine the structural response and local damage of floor slabs and hatch covers. Load handling activities not included in this SER nor in other docketed SER's will be addressed on a case by case basis and be subject to NRC approval.

This SER will address activities associated with defueling but will not include fuel transfer from the spent fuel pool to the shipping cask or the handling of the fuel shipping casks.

For the purposes of this SER, the defueling canisters are treated as any other heavy load. Specific safety concerns associated with damage to dropped defueling canister and with the handling of defueling canisters filled with fuel are outside the scope of this SER and will be addressed in References 7 and 13.

Load handling areas included in the scope of this SER will be divided into three types of areas: unrestricted lift areas, restricted lift areas and exclusion areas described as follows:

1.3.1 Unrestricted Lift Areas

Unrestricted lift areas (ULA) are those areas where loads can be handled that are equal to or less than the rated load of the installed cranes or hoists.

1.3.2 Restricted Lift Areas

Restricted lift areas (RLA) are those areas where a restriction applies to the allowable lift height and/or weight of a load or load path to be used.

1.3.3 Exclusion Areas

Exclusion areas (EA) are those areas where heavy loads are not to be handled without further evaluation and specific prior approval of the NRC.

1.4 Organization

Section 2.0 consists of the description of the activities associated with the lifting of heavy loads.

Section 3.0 addresses the potential impact of load drops and the safety concerns associated with the movement of heavy loads in the containment and FPA in the FHB, summarizes the results of the analyses of the load drops postulated in this SER and includes any necessary load weight/lift height restrictions.

Section 4.0 presents the conclusions of this SER and Section 5.0 contains the list of references.

2.0 DESCRIPTIONS OF ACTIVITIES

As the goal of this SER is to provide generic direction for the handling of all heavy loads through defueling within the containment and in FPA, specific load handling activities are not identified. However, the following are prerequisites for performing any heavy load handling activity addressed in this SER:

(i) the performance of load handling activities will be by qualified personnel trained in the operation and safety of lifting and handling equipment.

(ii) appropriate procedures or Unit Work Instructions (UWI's) are available that clearly identify load paths which avoid the "exclusion areas" identified in Section 3.3.1 and 3.3.2 and identify the "restricted area" load handling limitations.

(iii) the crane lifting rigging and attachment points shall have been inspected and tested in accordance with approved procedures.

3.0 HEAVY LOAD DROP ANALYSIS

3.1 Introduction

The containment load drop analyses are based on the assumption that postulated load drops will result in the local failure of floors. An evaluation was made for heavy load drops in containment to ensure that the postulated failures cannot result in draining the reactor vessel below 314'-0", disabling all makeup paths to the reactor vessel (RV) or draining the fuel transfer canal (FTC). Reference 9 addresses heavy load drops over the reactor vessel which could potentially drain the reactor vessel below 314'-0".

Load drop analyses for load drops in FPA are based on the assumption that postulated load drops could result in local damage to the Fuel Canister Storage Racks (FCSR) and/or the fuel pool liner plate.

3.2 Identification of Loads

Loads handl ed inside the containment are anticipated to range up to a maximum of 25 tons, excluding the plenum; however, this SER addresses all loads up to the 170 ton rated capacity of the main hook of the polar crane.

This SER addresses all loads that may be handled inside FPA up to and including the design defueling canister weight of 3355 pounds.

3.3 Identification of Targets

The target for a postulated load drop is considered to be the floor and equipment in the region directly below the suspended load. Specific target areas will be identified in both the containment and FPA. These target areas will be differentiated based on their ability to withstand a specific load impact. The load handling areas are described as follows:

3.3.1 Containment Load Handling Areas

3.3.1.1 Reactor Vessel

The RV with PA removed prior to and following the installation of the defueling work platform (DWP) is considered an RLA.

3.3.1.2 Fuel Transfer Canal Deep End

The deep end of the fuel transfer canal (FTC) is that area of the FTC from 22'-6" to 40'-0" north of the RV centerline and 12'-0" east and west of the RV centerline.

3.3.1.2.1 The FTC deep end when no fuel canisters which contain fuel are present in the deep end is considered a ULA.

3.3.1.2.2 The FTC deep end with filled fuel canisters in the FCSR is considered an RLA.

3.3.1.3 Fuel Transfer Canal Shallow End

The FTC shallow end is that area south of the deep end, does not include the RV.

3.3.1.3.1 The FTC shallow end, north of the RV is considered an RLA.

3.3.1.3.2 The FTC shallow end, south of the RV is considered a ULA.

3.3.1.4 Northwest 'A' D-Ring and Seal Table

The northwest section of the 'A' D-ring and the seal table are considered EA's due to the presence of the incore instrument tubes in these areas. This EA encompasses the area inside containment, west of the FTC and north of the centerline of the 'A' once through steam generator (OTSG).

3.3.1.5 General Containment

The general containment excludes those areas described above and encompasses all other containment areas at all elevations. This area is considered a ULA if all unborated water sources in containment are isolated (See Section 3.5.2.2).

3.3.2 Fuel Handling Building Load Handling Areas

3.3.2.1 Fuel Pool 'A'

3.3.2.1.1 Fuel pool 'A' (FPA) prior to defueling canisters loaded with fuel being present in FPA is considered a ULA.

3.3.2.1.2 FPA with filled fuel canisters in FPA is considered an RLA.

3.4 Load/Target Interactions

The attached figures, 3.4-1 and 3.4-2 provide plans of the containment and FPA with allowed load handling areas identified. The classifications of various load handling areas are based on the evaluations developed in the following paragraphs; sections 3.4.1 through 3.4.2.1 have a one for one correspondence with sections 3.3.1 through 3.3.2.1.

3.4.1 Containment Load Handling Areas

3.4.1.1 Reactor Vessel

All loads to be handled over the reactor vessel are discussed and evaluated in detail in Reference 9.

3.4.1.2 Fuel Transfer Canal Deep End

3.4.1.2.1 The handling of loads over the deep end of the FTC without filled canisters present in the FTC presents no plant safety concerns. A drop in this area would not affect the stability of the core, drain or reduce the water level in the reactor coolant system or impact the availibility of makeup; in addition, containment access would not be prevented.

3.4.1.2.2 The handling of loads over the FCSR in the deep end of the FTC, when canisters are in the racks, will be restricted such that the potential energy will not be greater than that of a suspended fuel canister. The following equation will be used to determine the maximum plant elevation (H, maximum plant elevation in feet) to which a given weight (W, where W is in pounds and not greater than 3355 pounds per Reference 6) can be raised over the FCSR in the containment.

$$H = \frac{85,600}{W} + 308$$

3.4.1.3 Fuel Transfer Canal Shallow End

The analysis of load drops occurring in the FTC shallow end assumes that objects fall from their lift height unimpeded to the floor of the FTC and impact a point. This results in the transmission of the greatest potential impact energy directly to the FTC floor as no impact energy is assumed absorbed by the collapse of platforms or equipment.

3.4.1.3.1 The shallow end of the FTC north of the RV is classified as an RLA, partially based upon the review performed for Reference 10, and a review of loads that will be handled over this end of the FTC. This review examined the potential for failure of the floor at 322'-6" due to a drop of the plenum 12 feet and its impact on the availability of makeup to the RV and damage to the in-core tubes which could result in draining the RV. Based on this review, it was determined that loads can be handled in these areas without presenting the potential for draining the RV or impacting the availability of makeup to the RV, if they are limited in accordance with the following expression:

$$H = \frac{1,320,000}{W} + 322.5$$

Where H is the maximum plant elevation to which load W (in pounds) may be raised.

In the low probability event that excess dam leakage or a complete loss of the dam function occurred the water level in the deep end of the transfer canal and in fuel pool "A" would lower. Water shielding over both the plenum assembly and the canisters will be reduced, however flooding the canal could be completed to increase the water level and reduce the radiation exposure levels.

3.4.1.3.2 The shallow end of the FTC south of the RV is classified as a ULA, based upon the reviews performed for References 2 and 3, and a review of loads that will be handled over this end of the FTC. This review examined the potential for failure of the floor at 322'-6" and its impact on the availability of makeup to the RV and damage to the in-core tubes which could result in draining the RV. Based on this review, it was determined that loads can be handled in these areas without presenting the potential for draining the RV or impacting the availability of makeup to the RV.

3.4.1.4 Northwest 'A' D-Ring and Seal Table

This area is defined in section 3.3.1.4, illustrated on figure 3.4-1 and is an EA. This area has been identified as an area in containment where a load drop could impact the in-core tubes and potentially drain the RV.

3.4.1.5 General Containment

This area is classified as a ULA if all unborated water sources are isolated (See Section 3.5.2.2). This classification is based on the review performed for Reference 2 and 3 which demonstrated that load drops in these areas could not result in draining the reactor vessel, impacting the availability of makeup to the RV or an inadvertent criticality. Criticality is prevented by the isolation of non-borated water sources per paragraph 3.5.2.2.

3.4.2 Fuel Handling Building Load Handling Areas

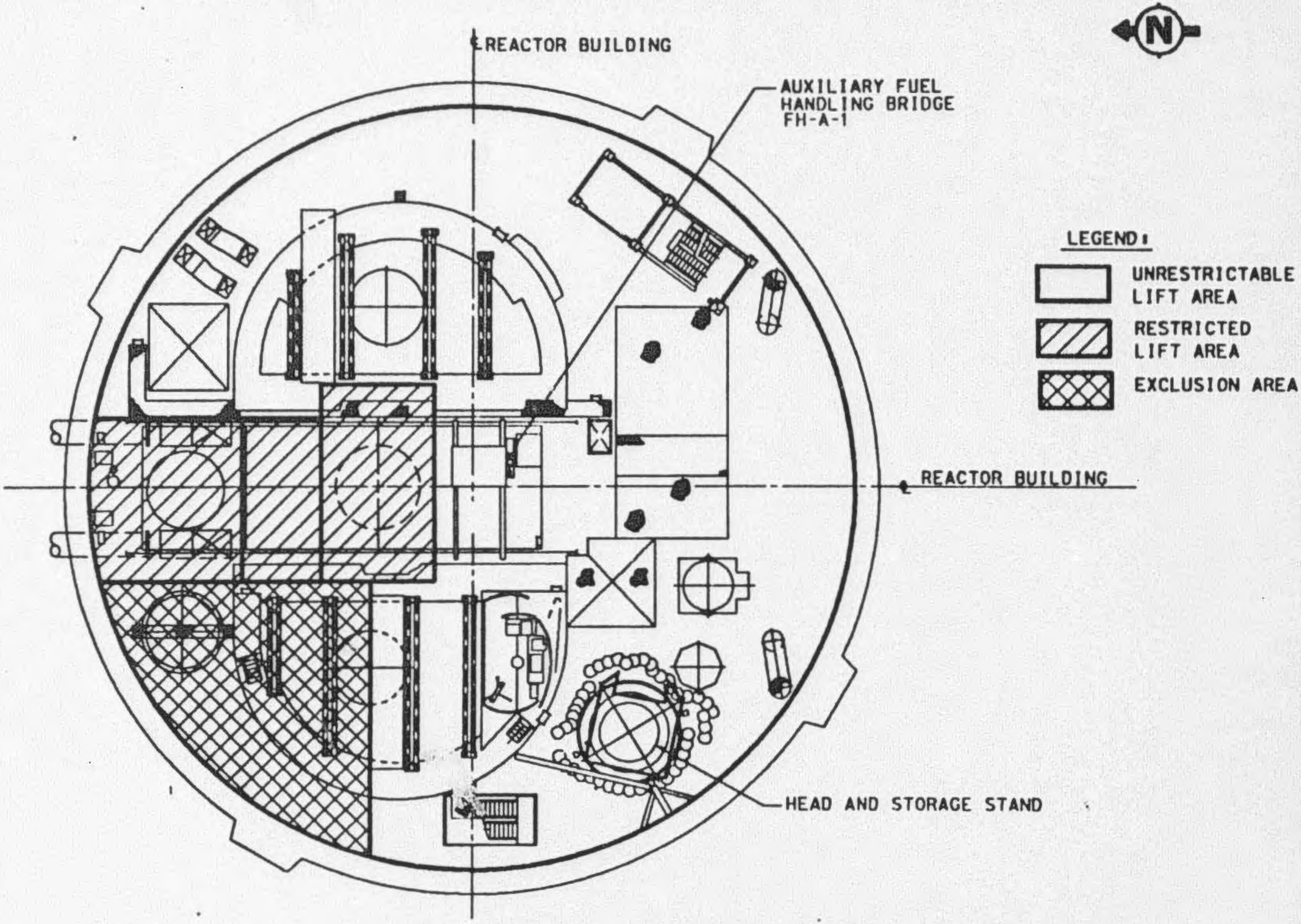
3.4.2.1 Fuel Pool 'A'

3.4.2.1.1 The handling of loads over FPA without filled defueling canisters present in the fuel pool presents no plaut safety concerns. Such a drop would not affect the stability of the core, drain or reduce the water level in the reactor coolant system, impact the availability of makeup or create the potential for an inadvertent criticality event. Therefore, the ULA classification for this area is appropriate.

3.4.2.1.2 The handling of loads over the FCSR in the fuel pool, when filled defueling canisters are in the racks, will be restricted such that the potential energy will not be greater than that of a suspended fuel canister. The following equation will be used to determine the maximum plant elevation (H, maximum plant elevation in feet) to which a given weight (W, where W is in pounds and not greater than 3355 pounds per Reference 6) can be raised over the FCSR in the FHB.

$$H = \frac{76,000}{W} + 308$$

Note: This expression is different than that provided in section 3.4.1.2.2 as the canister lift heights are different in the FHB than the containment.



PLAN EL. 347'-6"

FIGURE 3.4-1
POTENTIAL LOAD/IMPACT AREAS
REACTOR BUILDING

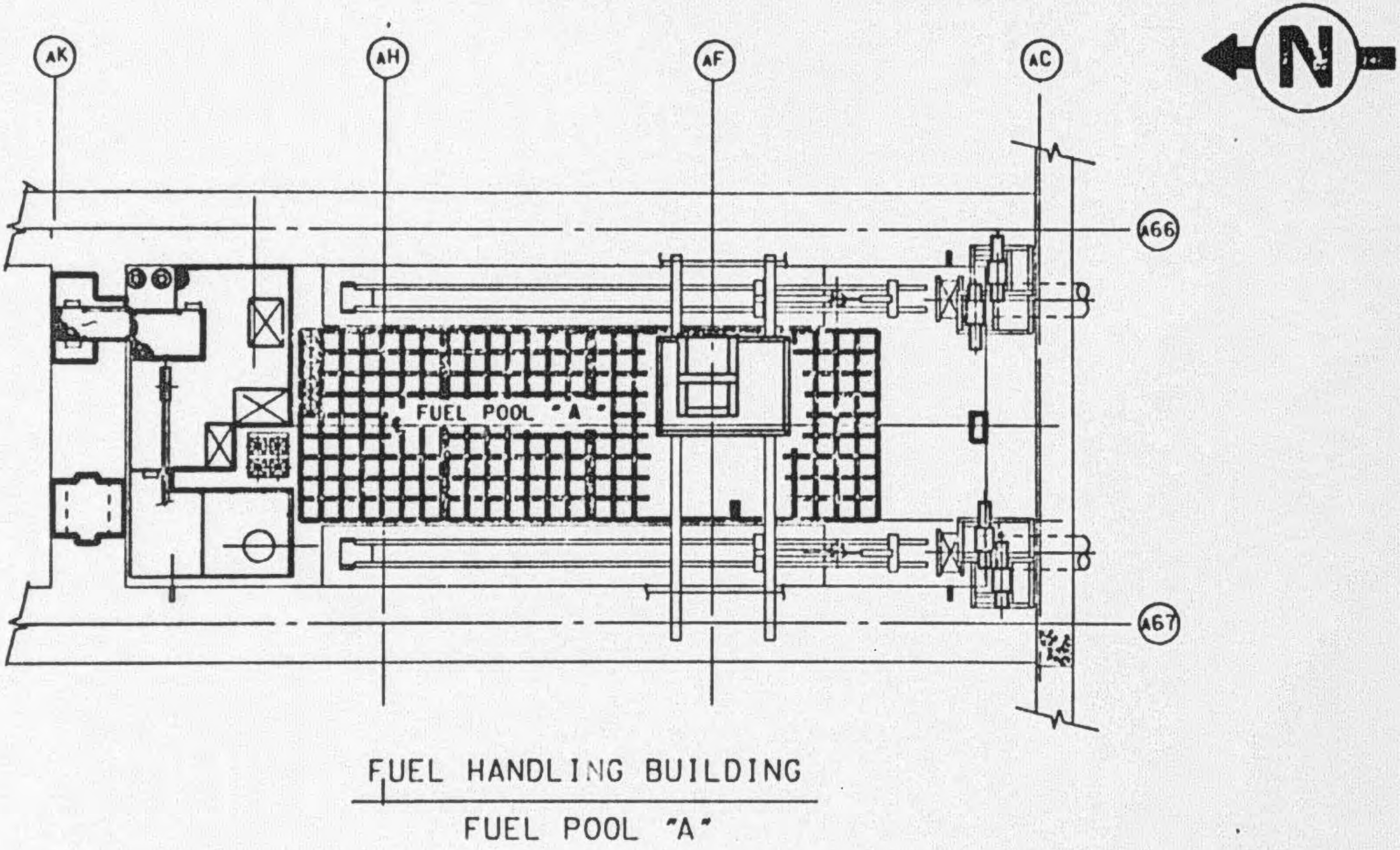


FIGURE 3.4-2
POTENTIAL LOAD/IMPACT AREAS
FUEL POOL "A"

3.5 Criteria Specific Evaluations (NUREG-0612)

This section evaluates the results of load drops postulated in this SER against the four criteria set forth in NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants."

3.5.1 NUREG-0612 Criteria

3.5.1.1 Criterion I:

Releases of radioactive material that may result from damage to spent fuel based on calculations involving accidental dropping of a postulated heavy load produce doses that are well within 10 CFR Part 100 limits of 300 rem thyroid, 25 rem whole body (analyses should show that doses are equal to or less than 1/4 of Part 100 limits).

3.5.1.2 Criterion II:

Damage to fuel and fuel storage racks based on calculations involving accidental dropping of a postulated heavy load does not result in a configuration of the fuel such that k_{eff} is larger than 0.95.

3.5.1.3 Criterion III:

Damage to the reactor vessel or the spent fuel pool based on calculations of damage following accidental dropping of a postulated heavy load is limited so as not to result in water leakage that could uncover the fuel (makeup water provided to overcome leakage should be from a borated source of adequate concentration if the water being lost is borated).

3.5.1.4 Criterion IV:

Damage to equipment in redundant or dual safe shutdown paths, based on calculations assuming the accidental dropping of a postulated heavy load, will be limited so as not to result in loss of required safe shutdown functions.

3.5.2 NUREG-0612 Evaluations

This section responds to each of the four NUREG-0612 criteria; sections 3.5.2.1 through 3.5.2.4 have a one for one correspondence with sections 3.5.1.1 through 3.5.1.4.

3.5.2.1 Any releases of radioactivity caused by the load drops addressed in this SER would be released within the containment or in the FHB. The containment or FHB would act as a physical barrier and prevent any liquid releases from escaping to the environment. Likewise, any additional particulates that may become airborne would be removed by the high efficiency particulate air (HEPA) filters so as not to exceed the limits established in Criterion I.

A bounding analysis was performed which assumes an instantaneous total release of the unaccounted for Kr-85 inventory from the reactor core. The amount released is assumed to be 31,300 curies of Kr-85 with the resulting dose estimated to be 9.7 millirem to the whole body for an individual located at the nearest site boundary and 1.8 mrem to the whole body for an individual located at the Low Population Zone (LPZ) Boundary. The meteorological dispersion parameters (X/Q) used were 6.1×10^{-4} sec/m^3 at the site boundary and 1.1×10^{-4} sec/m^3 at the LPZ boundary (as indicated in the FSAR).

An additional analysis was performed in Reference 7 in order to determine the maximum offsite dose due to any airborne particulates that may pass through the HEPA filters following the drop of a defueling canister. This analysis used conservative assumptions and calculated a critical organ (teenagers bone) dose of 2.96 Rem which is less than 4% of the 75 Rem acceptance criteria, 1/4 of the 10CFR Part 100 dose guidelines. The bone dose is presented since it was determined to be the critical organ based on comparisons of dose conversion factors for several organs, including the lung, kidney, liver and gastrointestinal tract, for the distribution of radionuclides available for release.

3.5.2.2 The dropping of heavy loads on the fuel canister storage racks (FCSR) without defueling canisters filled with fuel being present (in either the fuel pool or the FTC) poses no safety concern as there is no opportunity for a criticality event, radiation release or uncovering of fuel.

The handling of heavy loads over the FCSR with filled or partially filled canisters present will be maintained within the limits set forth in sections 3.4.1.2 and 3.4.2.1. This will ensure the FCSR are not damaged to such an extent as to cause a return to criticality.

Load handling over the reactor vessel and the associated safety issues are discussed in Reference 9.

As in previous load handling SER's the isolation of non-borated water sources during the handling of heavy loads to prevent the addition of non-borated water to the containment sump is necessary. Per Reference 4, the need to isolate the Reactor Building Chilled Water System in order to prevent a sump criticality event is no longer required. The systems identified below (per Reference 3) are potential sources of unborated water inside containment:

Reactor Building Fire Protection System
Reactor Building Demineralized Water System
Reactor Building Nuclear Services Closed Cooling Water System
Reactor Building Intermediate Closed Cooling Water System
Reactor Building Normal Cooling Water System
Reactor Building Nuclear Services River Water System
Main Steam and Feedwater Systems
Reactor Building Decontamination System
Decay Heat System (Auxiliary Spray)
Steam Generator Secondary Side Vents and Drains
Makeup and Purification System

As an alternative to isolating all unborated water sources for each heavy load handled inside containment, adherence to the load weight and height guidelines provided in Tables 3.5-1 through 3.5-9 will ensure that a dropped load will not fail the floor slab and consequently the unborated water systems located beneath the floor slab over which a load is being carried need not be isolated. Any unborated water systems which could be directly impacted by a load drop within the area of a particular load handling activity will be isolated until completion of that activity.

During any load handling activity with load weight/height in excess of the guidelines provided in the attached tables, all unborated water sources inside containment will be isolated unless it can be demonstrated that there is sufficient physical separation between the load handling area and specific systems to ensure no system failure in the event of a load drop.

3.5.2.3 As loads will not be handled over the in-core tubes, the load drops postulated in this SER could not drain the reactor vessel below the bottom of the reactor vessel hot leg, elevation 314'-0". Drainage to this level will not uncover the fuel. Makeup may be provided by the makeup system via redundant pathways to the reactor vessel.

The dropping of a heavy load in the deep end of the FTC or in FPA may result in local damage to the stainless steel liner plate. The extent of this damage will be determined by the shape and weight of the dropped load, and may range from denting, to perforation of the liner plate. The perforation of the liner plate may result in water being lost from FPA/FTC; this water would be collected by the liner leakage collection system and directed to the auxiliary building sump for FPA leakage or containment sump for FTC leakage. Necessary makeup would be provided from the borated water storage tank (BWST). The catastrophic failure of the slab in the deep end of the FTC is not considered credible due to the existence of a concrete support wall located at the center of the slab.

Reference 13 describes an analysis to determine the potential for criticality to occur in FPA/FTC due to a catastrophic failure of the liner causing FPA/FTC to be drained of water. This analysis determined a criticality event would not occur.

3.5.2.4 Criterion IV refers to "required safe shutdown functions" which are defined as those required to: maintain the reactor coolant pressure boundary, maintain subcriticality, remove decay heat, and maintain the integrity of components whose failures could result in excessive off-site releases.

The required safe shutdown functions that apply to the TMI-2 reactor in its current cooling mode and core configuration are:

1) The capability to maintain subcriticality.
2) Decay heat removal.
3) The capability to maintain the integrity of components whose failures could result in excessive off-site releases.

Reactor coolant will be maintained in the reactor system (RCS) above the reactor vessel nozzles for decay heat removal and reactivity control. Subcriticality will be maintained as described in section 3.5.2.2. Currently decay heat is removed by heat losses to ambient which has been demonstrated adequate to remove all decay heat (Reference 5) produced by the core material in the reactor vessel. As such, no additional equipment is necessary to remove decay heat.

Reactivity will continue to be controlled if the level of borated water in the RCS and FPA/FTC are maintained. Thus, dropping of a heavy load would only affect reactivity control if the load drop resulted in breaking in-core instrument tubes, since the breaking of the in-core instrument tubes would drain the reactor vessel below elevation 314'-0". However, for the load drops postulated in this SER, the breaking in-core instrument tubes will not occur because there are no in-core instrument tubes outside of the load handling exclusion areas.

The offsite releases are addressed in Section 3.5.2.1.

Consequently, safe shutdown will be maintained for load handling and load drop accidents postulated in this SER.

3.6 10 CFR 50.59 Evaluation

10CFR50, 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 of 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 specificaton, is reduced.

The planned load handling activities will not increase the probability of occurrence or the consequences of an accident or malfunction of equipment important to safety previously evaluated. The planned activities will not create the possibility of an accident or malfunction of a different type than any evaluated previously and have been shown not to be an unreviewed safety question. Since the operation of systems and equipment are in accordance with approved procedures to ensure compliance to technical specifications, the tasks included in this SER will not reduce the margin of safety as defined in the basis for any technical specification.

Therefore, it is concluded that the lifts described in this SER do not involve any unreviewed safety questions as defined in 10CFR Part 50, Paragraph 50.59.

4.0 CONCLUSIONS

The lifting of heavy loads and associated activities have been described and evaluated. 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 load drops have been shown not to compromise plant safety. The accidental releases of radioactivity have been evaluated and are bounded by the analyses presented in References 2 and 7. It is therefore concluded that the load lifts discussed in this SER can be performed without presenting undue risk to the health and safety of the public.

5.0 REFERENCES

1. NUREG-0612 "Control of Heavy Loads at Nuclear Power Plants" January 1980.

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

3. "Safety Evaluation Report for the Polar Crane Load Test," Rev. 0, February 1983.

4. NRC Letter from P. J. Grant to B. K. Kanga, NRC/TMI-84-052 dated July 17, 1984.

5. G. A. Hipp, et al., "Addendum to the TMI-2 Decay Heat Removal Report of April 1982," Revision 1, December 1982 (This report is Attachment 4 to GPUN Letter 4410-83-L-0052, March 15, 1983, B. K. Kanga to L. H. Barrett).

6. "Technical Evaluation Report for Fuel Canister Storage Racks," Rev. 0, January 1985.

7. "Safety Evaluation Report for Early Defueling," Rev. 1, April 1985.

8. Deleted

9. "Safety Evaluation Report for Load Handling Over the Reactor Vessel," 15737-2-G07-110, Revision 0, April 18, 1985.

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

11. Deleted

12. Deleted

13. "Technical Evaluation Report for Defueling Canisters," 15737-2-G03-114.

TABLE 3.5-1

REFUELING CANAL SLAB AREA, SHALLOW END OF CANAL, ELEVATION 322'-6"

MAXIMUM ALLOWABLE LOAD (LBS.)	MINIMUM EQUIVALENT DIAMETER OF LOAD DROP (INCHES)	MAXIMUM ALLOWABLE LIFT (FT.) ABOVE ELEVATION 322'-6"
10,000	1	11
10,000	3	35
10,000	6	38
10,000	9	40
10,000	12	42
10,000	18	48
10,000	24	54
10,000	36	68
5,000	1	24
5,000	3	108
5,000	6	110
5,000	9	110
5,000	12	110
5,000	18	110
5,000	24	110
5,000	36	110

TABLE 3.5-2

REFUELING CANAL SLAB AREA, DEEP END OF CANAL, ELEVATION 308'-0"

MAXIMUM ALLOWABLE LOAD (LBS.)	MINIMUM EQUIVALENT DIAMETER OF LOAD DROP (INCHES)	MAXIMUM ALLOWABLE LIFT (FT.) ABOVE ELEVATION 308'-0"
10,000	1	3
10,000	3	7
10,000	6	12
10,000	9	17
10,000	12	22
10,000	18	30
10,000	24	35
10,000	36	50
5,000	1	9
5,000	3	22
5,000	6	37
5,000	9	55
5,000	12	70
5,000	18	95
5,000	24	120
5,000	36	125

TABLE 3.5-3

NORTHWEST QUADRANT CONCRETE SLAB, ELEVATION 347'-6"
(EXCLUDING THE A D-RING AND THE REFUELING CANAL)

MAXIMUM ALLOWABLE LOAD (LBS.)	MINIMUM EQUIVALENT DIAMETER OF LOAD DROP (INCHES)	MAXIMUM ALLOWABLE LIFT (FT.) ABOVE ELEVATION 347'-6"
10,000	1	1
10,000	3	3
10,000	6	3
10,000	9	3
10,000	12	3
10,000	18	3
10,000	24	3
10,000	36	3
5,000	1	4
5,000	3	8
5,000	6	8
5,000	9	8
5,000	12	9
5,000	18	9
5,000	24	9
5,000	36	10

TABLE 3.5-4

NORTHEAST, SOUTHEAST, SOUTHWEST QUADRANTS CONCRETE SLAB, ELEVATION 347'-6"
(EXCLUDING THE D-RINGS, THE REFUELING CANAL AND HATCH AREAS)

MAXIMUM ALLOWABLE LOAD (LBS.)	MINIMUM EQUIVALENT DIAMETER OF LOAD DROP (INCHES)	MAXIMUM ALLOWABLE LIFT (FT.) ABOVE ELEVATION 347'-6"
10,000	1	1
10,000	3	7
10,000	6	7
10,000	9	8
10,000	12	8
10,000	18	8
10,000	24	8
10,000	36	9
5,000	1	4
5,000	3	15
5,000	6	15
5,000	9	16
5,000	12	18
5,000	18	22
5,000	24	23
5,000	36	25

TABLE 3.5-5

HATCH AREAS, ELEVATION 347'-6", WITH 3/8" S.S. CHECKERED PLATE

MAXIMUM ALLOWABLE LOAD (LBS.)	MINIMUM EQUIVALENT DIAMETER OF LOAD DROP (INCHES)	MAXIMUM ALLOWABLE LIFT (FT.) ABOVE ELEVATION 347'-6"
10,000	1	.25
10,000	3	1
10,000	6	2.5
10,000	9	2.5
10,000	12	2.5
10,000	18	2.5
10,000	24	2.5
10,000	36	2.5
5,000	1	.5
5,000	3	3
5,000	6	5
5,000	9	5
5,000	12	5
5,000	18	5
5,000	24	5
5,000	36	5

TABLE 3.5-6

HATCH AREAS, ELEVATION 347'6", with 4" S.S. GRATING

MAXIMUM ALLOWABLE LOAD (LBS.)	MINIMUM EQUIVALENT DIAMETER OF LOAD DROP (INCHES)	MAXIMUM ALLOWABLE LIFT (FT.) ABOVE ELEVATION 347'-6"
10,000	1	.25
10,000	3	1
10,000	6	2
10,000	9	2
10,000	12	2
10,000	18	2
10,000	24	2
10,000	36	2
5,000	1	.5
5,000	3	3
5,000	6	4
5,000	9	4
5,000	12	4
5,000	18	4
5,000	24	4
5,000	36	5

TABLE 3.5-7

ELEVATION 305' CONCRETE SLAB WEST OF HATCH

MAXIMUM ALLOWABLE LOAD (LBS.)	MINIMUM EQUIVALENT DIAMETER OF LOAD DROP (INCHES)	MAXIMUM ALLOWABLE LIFT (FT.) ABOVE ELEVATION 305'-0"
10,000	1	3
10,000	3	7
10,000	6	12
10,000	9	17
10,000	12	22
10,000	18	30
10,000	24	35
10,000	36	50
5,000	1	9
5,000	3	22
5,000	6	37
5,000	9	55
5,000	12	70
5,000	18	95
5,000	24	120
5,000	36	128

TABLE 3.5-8

ELEVATION 305' CONCRETE SLAB NORTH, SOUTH AND EAST OF HATCH

MAXIMUM ALLOWABLE LOAD (LBS.)	MINIMUM EQUIVALENT DIAMETER OF LOAD DROP (INCHES)	MAXIMUM ALLOWABLE LIFT (FT.) ABOVE ELEVATION 305'-0"
10,000	1	1
10,000	3	2
10,000	6	2
10,000	9	3
10,000	12	4
10,000	18	5
10,000	24	5
10,000	36	5
5,000	1	2
5,000	3	3
5,000	6	5
5,000	9	6
5,000	12	8
5,000	18	10
5,000	24	15
5,000	36	20

TABLE 3.5-9

HATCH AREA, ELEVATION 305'

MAXIMUM ALLOWABLE LOAD (LBS.)	MINIMUM EQUIVALENT DIAMETER OF LOAD DROP (INCHES)	MAXIMUM ALLOWABLE LIFT (FT.) ABOVE ELEVATION 305'-0"
10,000	1	0.25
10,000	3	0.83
10,000	6	0.83
10,000	9	0.83
10,000	12	0.83
10,000	18	0.83
10,000	24	0.83
10,000	36	0.83
5,000	1	0.5
5,000	3	1.67
5,000	6	1.67
5,000	9	1.67
5,000	12	1.67
5,000	18	1.67
5,000	24	1.67
5,000	36	1.67