GPU Nuclear

SAFETY ANALYSIS

SA # 4350-3153-85-1

Rev. # 4

TITLE

LOAD HANDLING INSIDE CONTAINMENT

AND

OVER FUEL POOL "A"

SAFETY EVALUATION REPORT

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No. 4350-3153-85-1

Title: Load Handling Inside Containment and Over Fuel Pool "A" Safety Evaluation Report

Rev.	SUMMARY OF CHANGE	Approval	Date
0	Initial issue for use.		10/84
1	Revised to increase scope to include the handling of heavy loads over fuel canister storage racks with canisters present.		7/85
2	Revised to incorporate comments on Revision 1 and to correct minor typographical errors.		9/85
3	Revised to correct equations for handling loads over the fuel canister storage racks, delete equations for load handling in the shallow end of the fuel transfer canal, and to correct revisions of references.		6/86
4	Revised to demonstrate that a load drop in the area west of the Fuel Transfer Canal and North of the "A" Once Through Steam Generator "OTSG" centerline is highly unlikely based on the use of rigging equipment designed as single-failure-proof or with double the required load capacity. In addition, this revision deletes the lift height restrictions for the general containment as there is currently no need to protect unborated water sources in the Reactor Building from potential load drops. The SER title has also been changed to more accurately reflect the scope of the SER.		5/87

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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 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. This SER also defines certain lift areas inside containment requiring additional prerequisites for load handling regardless of the weight of the load. As this SER does not address specific loads or specific load handling operations, off-site 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 SERs 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 additional prerequisite lift 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 Additional Prerequisite Lift Areas

Additional Prerequisite Lift Areas (APLA) are those areas where an additional prerequisite is necessary to provide greater assurance that a drop of a load handled within these areas would be extremely unlikely, if not incredible.

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 (UWIs) are available that clearly identify load paths, applicable "restricted area" load handling limitations, and applicable additional prerequisites given in this SER.

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, within ULAs or RLAs, to ensure that the postulated failures cannot result in draining the reactor vessel (RV) below 314'-0", disabling all makeup paths to the

RV or draining the fuel transfer canal (FTC). Because of the additional prerequisite on load handling within the APLA, a load drop in this area is not considered credible and, thus, the consequences of a postulated load drop in this area are not further evaluated. The bounding consequences of a postulated load drop in the APLA is the potential draining of the RV which has been evaluated in Reference 8. Reference 9 addresses heavy load drops over the RV which could potentially drain the RV 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

A load is the total weight suspended below the hook of a crane or hoist which is subject to a drop should the rigging or an attachment point fail. Polar crane failure that causes the load blocks to fall can occur due to overstressing of the hoist cables. Overstressing the hoist cable can be caused by overloading the crane. Overloading of the crane is precluded by administrative controls which limit the load on the crane. For load handling activities outside of the APLA, the load on the crane is limited to the rated capacity of the crane load block being used. For the load handling activities inside the APLA the load on the crane is limited to one-half of the rated capacity of the crane load block being used. Additionally, good rigging practice will ensure that the load to be handled has a free lift path to prevent load hang-up. Hoist cable overstressing can also occur due to "two-blocking" (i.e., the load block is raised to the extent that it contacts the crane structure). "Two-blocking" is precluded by at least one upper limit switch and administrative controls which restrict lift heights to provide minimal clearances over obstructions in the load path. Thus, the main or auxiliary load blocks on the polar crane are not considered loads. Loads handled inside the containment are anticipated to range up to the weight of a RV missile shield (approximately 40 tons), excluding the plenum; however, this SER addresses all loads (including light loads in the APLA) 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 APLAs due to the presence of the incore instrument tubes in these areas. This APLA 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.

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 (pages 11 and 12, respectively) 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 RV 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 availability 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.

$$H = \frac{37,000}{W} + 322$$

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, as a load drop in this area could result in damage to the floor at 322'-6" and possibly impact the availability of normal makeup to the RV or damage the incore tubes which could result in draining the RV.

Load handling in this area without lift height restrictions may create a potential for local damage such as spalling of concrete from the bottom of the floor slab which could in turn impact the incore instrument cable chase. In order to preclude any spalling that might occur load/lift height limits have been established. These limits are presented in Table 3.5 and will be used for load handling in the north half of the shallow end of the FTC.

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 incore 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 APLA. A postulated load drop in this area could impact the incore tubes and potentially drain the RV. Thus, additional prerequisites will be placed on load handling activities in this area to assure that the potential for a load drop is extremely small. To minimize the probability of a load drop, load handling will not be performed in this area if it can be reasonably avoided. Loads handled by a crane or hoist will be rigged such that the rigging design is either singe-failure-proof, or the rigging components have double the load capacity required by applicable TMI-2 procedures. Rigging components are all components within the handling system that transmit the load from the load attachment point to the hook of the crane or hoist. The single-failure-proof rigging design will be such that a single failure of any rigging component will not cause the loss of control of the load and the remaining load bearing rigging components satisfy the design requirements of applicable TMI-2 procedures. In addition, the total load carried by the crane or hoist (which includes the weight of the load, rigging, and the crane or hoist load block) will be limited to no greater than one-half of the load rated capacity of the crane or hoist.

Loads handled by the polar crane may necessarily cause both load blocks to enter the APLA. Polar crane failure which causes either, or both, load block(s) to fall is very unlikely since overloading the crane is precluded. Overloading the crane is precluded by the load weight restriction given above for the load block in service and by the assurance that a free lift path exists prior to raising the load. "Two-blocking" either load block, which could overstress the hoist cables and cause the load block to fall, is precluded by the availability of at least one upper limit switch for each load block and administrative controls which limits the lift height of the load (and, hence, the height of the load block in service) and prohibits the operation of both hoists at the same time.

Small loads that can be man-handled do not require the additional prerequisites given above. However, if a small load is handled in an area where a dropped load would have a clear path to exposed incore tubes (e.g., inside the incore tube chase), then the load is to be tied-off to prevent the drop. The tie-off(s) will be attached to an existing plant structure that has been analyzed to ensure that the loading, imparted on the structure from the dropped load, does not exceed the yield stress of the structure. The tie-off(s) will also satisfy the design requirements of applicable TMI-2 procedures.

3.4.1.5 General Containment

This area is classified as a ULA. 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 RV or impacting the availability of makeup to the RV.

3.4.2 Fuel Handling Building Load Handling Areas

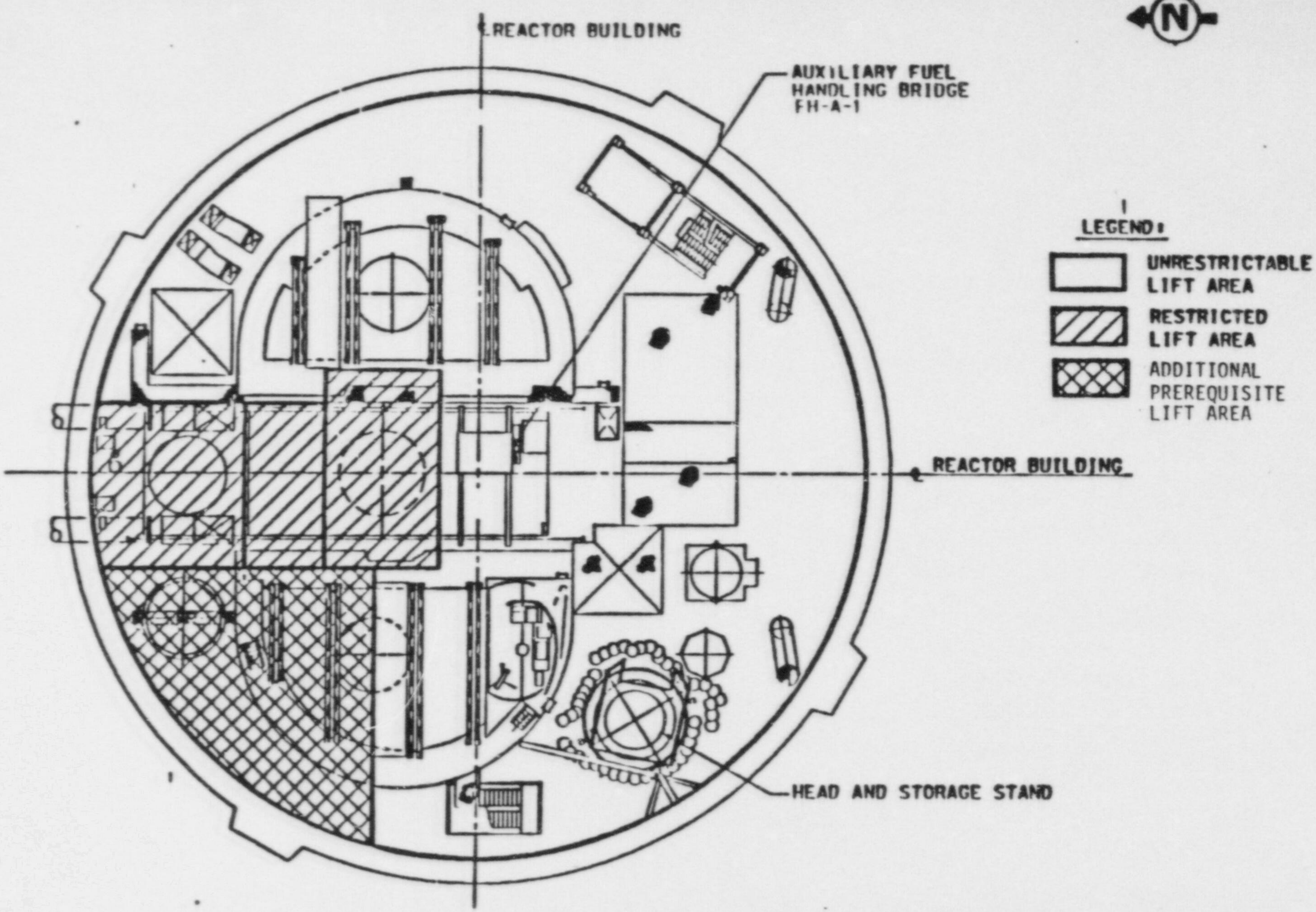
3.4.2.1 Fuel Pool 'A'

3.4.2.1.1 The handling of loads over FPA without filled defueling canisters present in the fuel pool presents no plant 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.

$$H = \frac{37,000}{W} + 321$$

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



PLAN EL. 347'-6"

FIGURE 3.4-1
POTENTIAL LOAD/IMPACT AREAS
REACTOR BUILDING

FIGURE 3.4-1

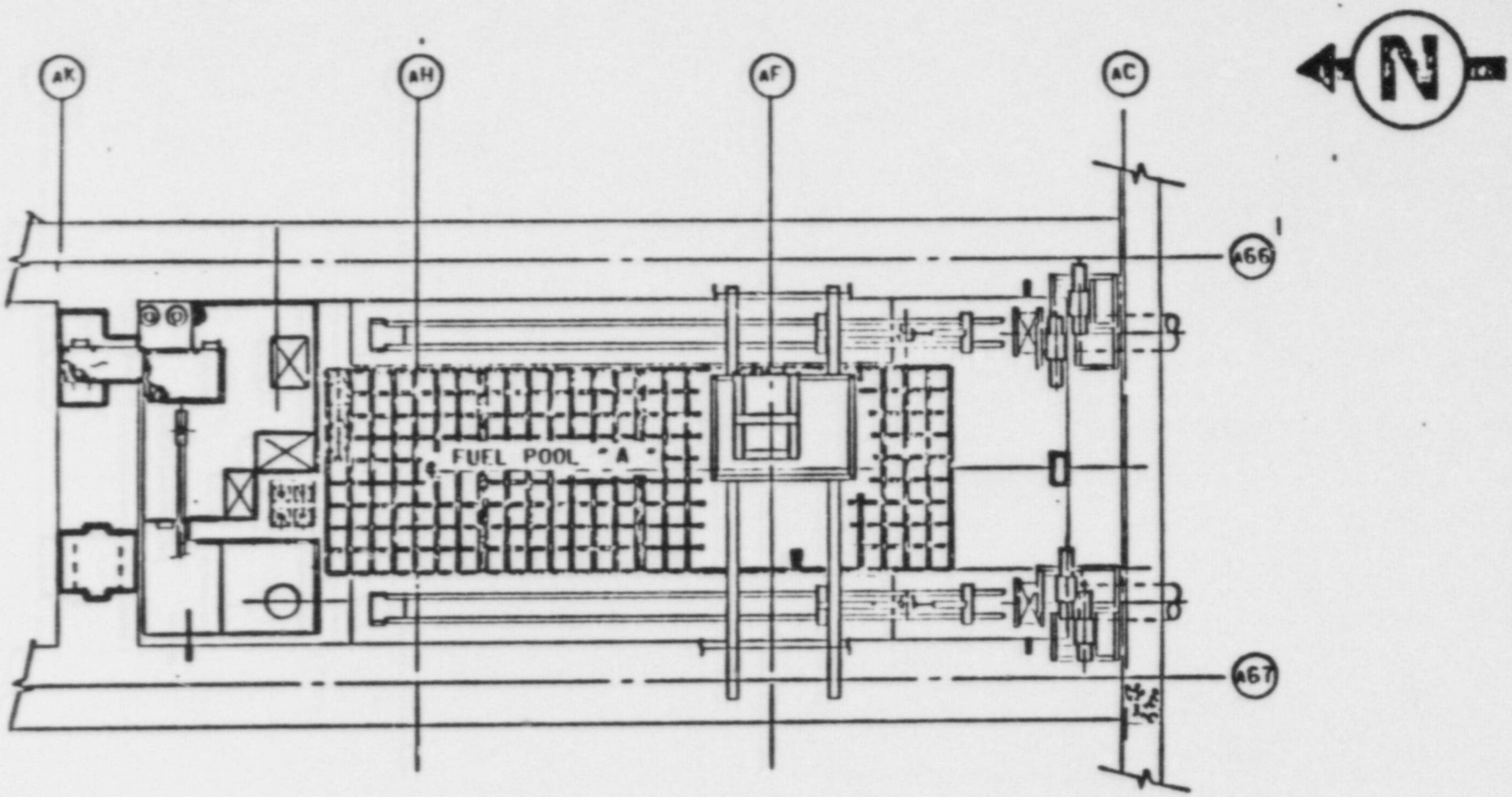


FIGURE 3.4-2

FUEL HANDLING BUILDING

FUEL POOL "A"

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

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3.5 NUREG-0612 Evaluation

NUREG-0612, "Control of Heavy Loads at Nuclear Power Plants," provides guidelines that Licensees can use to assure the safe handling of heavy loads. Section 5.1 of NUREG-0612 states that the objectives of the guidelines are to "assure that either (1) the potential for a load drop is extremely small, or (2) for each area addressed, the following evaluation criteria are satisfied."

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).

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.

Criterion III:

Damage to the RV 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).

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.

It is important to note that NUREG-0612 Section 5.1 provides an either/or recommendation. Licensee may either assure that the potential for a load drop is extremely small or demonstrate compliance with the four (4) criteria for postulated load drops.

Load handling activities within the scope of this SER (with the exception of those in the APLA) are evaluated by postulating a load drop and demonstrating that the results satisfy the criteria of NUREG-0612 Section 5.1. Section 3.5.1 of this SER presents the results of the evaluations performed for postulated load drops. Load handling activities within the APLA are evaluated by showing that the potential for a load drop is extremely small. Section 3.5.2 of this SER demonstrates that load handling activities inside the APLA meet the NUREG-0612 objective of assuring that acceptable measures are provided to control loads.

3.5.1 NUREG-0612 Criteria Evaluations

This section provides an evaluation of the four (4) NUREG-0612 criteria for load handling activities within the scope of this SER (with the exception of activities in APLA).

3.5.1.1 Criterion I

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 off-site 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.

3.5.1.2 Criterion II

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 RV and the associated safety issues are discussed in Reference 9.

Unlike previous load handling SERs 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 no longer required. Reference 11 has shown that the quantity of fuel present in the Reactor Building basement, based on conservative assumptions, is not sufficient to achieve criticality; thus, non-borated water may be introduced into the Reactor Building. The volume of non-borated water in the Reactor Building basement is limited to 70,000 gallons, based in the evaluation given in Reference 12. However, mitigating methods are available (e.g., closing valves in the affected systems to terminate the inflow of non-borated water, to pumping out the non-borated water) to either limit the volume of water below 70,000 gallons or to reduce the volume of water to an acceptable value in a timely manner.

3.5.1.3 Criterion III

Load drops postulated outside of the APLA could not drain the RV below the bottom of the RV 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 RV. Adherence to the load weight and height guidelines provided in Table 3.5 will ensure that a dropped load, in the northern half of the shallow end of the FTC, does not affect the incore instrument cable chase as described in Section 3.4.1.3.1.

The dropping of a heavy load, handled in accordance with the guidelines contained in this SER, 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.

3.5.1.4 Criterion IV

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 Coolant System (RCS) above the RV nozzles for decay heat removal and reactivity control. Subcriticality will be maintained as described in Section 3.5.1.2. Currently decay heat is removed by heat losses to ambient which has been demonstrated adequate to removal all decay heat (Reference 5) produced by the core material in the RV. As such, no additional equipment is necessary to removal decay heat.

Reactivity will continue to be controlled if the level of borated water in the RCS is maintained. Thus, dropping of a heavy load would only affect reactivity control if the load drop resulted in breaking incore instrument tubes, since the breaking of the incore instrument tubes would drain the RV below elevation 314'-0". However, for the load drops postulated outside of the ALPA, the breaking of incore instrument tubes will not occur as discussed in Section 3.5.1.3. Load handling inside the APLA is addressed in the following section.

The off-site releases are addressed in Section 3.5.1.1.

Consequently, safe shutdown will be maintained for load handling and load drop accidents postulated outside the APLA.

3.5.2 Load Handling in APLA

Guidelines for ensuring that the potential of a load drop is extremely small are provided in Section 5.1.1 of NUREG-0612. These guidelines have been identified in NRC Generic Letter 85-11, "Completion of Phase II of "Control of Heavy Loads at Nuclear Power Plants' NUREG-0612," to be sufficient to reduce the risks of heavy load handling such that the potential for a load drop is extremely small. Specifically, NRC Generic Letter 85-11 states:

> "Our review <of utilities' submittals for Phase I> has indicated that satisfaction of the Phase I guidelines assures that the potential for a load drop is extremely small. We have noted improvements in heavy load handling procedures and training and crane and handling tool inspection and testing. These changes have been geared to limiting the handling of heavy loads over safety-related equipment and spent fuel to the extent practical, but where this can not be avoided, to accomplishing it with the operational and other features of

the program implemented in Phase I. We, therefore, conclude that the guidelines of Phase I are adequately providing the intended level of protection against load drop accidents."

TMI-2 Procedure 4000-ADM-3890.02, "Control of Lifting and Handling Program," incorporates all the guidelines given in Section 5.1.1 of NUREG-0612, for the control of heavy loads, with the exception that the selection of slings does not include dynamic loads. The inherent safety factor in the design of slings more than compensates for the additional loads that could be imparted in the lifting equipment from dynamic loads. Therefore, based on Generic Letter 85-11, it may be concluded that the guidelines of NUREG-0612 are met through the implementation of Procedure 4000-ADM-3890.02. However, because the handling of loads within the APLA may necessitate load movements which, should failure occur, cause damage to the incore instrument tubes and result in an unisolable leak from the RV (i.e., a safe load path is not available), additional precautions above existing load handling requirements will be required to provide greater assurance against a load drop. These additional precautions are given in Section 3.4.1.4.

Based on this approach of taking additional precautions in addition to the existing procedural requirements, it is concluded that load handling within the APLA can be performed safely and in accordance with the guidelines of NUREG-0612.

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 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 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. Considering the additional prerequisite for load handling in the APLA, the probability of a load drop is greatly reduced to the extent of being extremely small. 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 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. Load handling activities can be safely performed inside the APLA because of the extreme unlikelihood of a load drop. It is therefore concluded that the load handling activities 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," Revision 5, February 1984.

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

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

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 GPU Nuclear letter 4410-83-L-0052, dated March 15, 1983, B. K. Kanga to L. H. Barrett).

6. "Technical Evaluation Report for Fuel Canister Storage Racks," Revision 1, May, 1986.

7. "Safety Evaluation Report for Defueling of the TMI-2 Reactor Vessel," Revision 10, May, 1986.

8. "Safety Evaluation Justifying the Non-Seismic Design of TMI-2 'Post-Accident' Systems," GPU Nuclear letter 4410-85-L-0077, dated April 16, 1985, F. R. Standerfer to B. J. Snyder.

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. "Safety Evaluation Report for Reactor Building Sump Criticality Evaluation," GPU Nuclear letter 4410-86-L-0009, dated January 23, 1986, F. R. Standerfer to B.. J. Snyder.

12. "Technical Specification Change Request No. 46," GPU Nuclear letter 4410-84-L-0154, dated November 6, 1984, F. R. Standerfer to B. J. Snyder.

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

TABLE 3.5

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