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November 1, 1984

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
Reactor Building Heavy Loads

Attached for your approval is the safety evaluation for handling heavy loads inside containment as committed to in GPU Nuclear Letter 4410-84-L-0131 dated August 10, 1984. This safety evaluation expands the analyzed load path area for heavy loads inside the TMI-2 containment. The safety analysis does not address the handling of heavy loads over the reactor vessel, incore instrument seal table and guide tubes, or the northwest corner of the A "D"-ring. Additionally, the safety evaluation does not address movement of heavy loads over the deep end of the fuel pool with fuel canisters present. Movements of heavy loads over these areas will be separately evaluated and approved pursuant to Technical Specification 3.10.1.

With the approval of this safety evaluation, GPU Nuclear will no longer submit Unit Work Instructions (UWI's) that implement load handling over the analyzed areas for approval. UWI's that implement load handling over the restricted areas, along with the associated safety evaluation, will continue to be submitted for approval.

D009 w/check $150.00
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Add:	Ltr	Encl
NRR Singh	4	4
NRR Neighbors	1	1
NRR/DL/TAPMG	1	1
NRR/DSI/AEB	1	1
NRR/DSI/ASI	1	1

In accordance with the requirements of 10 CFR 170, "Licensing Fees", an application fee of $150.00 is enclosed.

If you have any questions concerning this information, please call Mr. J. J. Byrne of my staff.

Sincerely,

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

FRS/RBS/jep

Attachment

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

15737-2-G07-105

SAFETY EVALUATION REPORT
FOR
HEAVY LOAD HANDLING
INSIDE CONTAINMENT
THREE MILE ISLAND UNIT 2

Rev. 0
October, 1984

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1.0 INTRODUCTION

1.1 Background

During the TMI-2 recovery operations the need for the lifting of loads greater than that which can be hand carried will necessitate the use of various hoists and cranes. The hoists and cranes to be used for handling these loads include the main hoist of the polar crane, the auxiliary hoist of the polar crane (to be requalified to 25 ton capacity), the reactor building service crane, and other smaller cranes and hoists within the containment.

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.

1.3 Scope

This SER addresses the handling of heavy loads within the containment, load handling areas and any necessary restrictions to be applied while handling these loads. The areas above the in-core instrument seal plate, the reactor vessel, and the northwest corner of the A D-ring are identified as "exclusion areas" where heavy loads are not to be handled without specific approval of th NRC, in accordance with Technical Specification 3.10.1 "Crane Travel - Containment Building."

In addition to these three exclusion areas, the deep end of the fuel transfer canal (FTC) is an exclusion area when either of the following two conditions exist:

a. Fuel filled canisters are present in the deep end of the FTC.
b. Fuel filled canisters are present in the fuel handling building fuel pool and one or both of the fuel transfer tubes are open.

Loads to be handled over the three exclusion areas or over the deep end of the FTC when either of the above two conditions exist will be evaluated and approved by the NRC on a case by case basis prior to the load handling operation. Additionally, the handling of canisters filled with fuel are outside the scope of this SER and will be treated in separate SER's.

Since this SER does not address specific loads and load handling operations, offsite releases are only addressed generically in this SER. 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.

1.4 Organization

Section 2.0 consists of the description of the activities associated with the lift 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 summarizes the results of the analyses of the load drops postulated in this SER.

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

2.0 DESCRIPTIONS OF ACTIVITIES

As the goal of this SER is to provide generic direction for the handling of all heavy loads within the containment, specific load handling activities are not identified. However, the following activities should precede all load handling activities:

(i) qualified personnel are available for load handling
(ii) appropriate procedures or Unit Work Instructions (UWI's) are available that clearly identify load paths which avoid the "exclusion areas" identified in Section 1.3
(iii) personnel involved have been instructed in crane safety and operations
(iv) an inspection of rigging and its attachment points should be conducted prior to the movement of heavy loads
(v) the crane shall have been inspected and tested in accordance with the applicable ANSI Standards.

3.0 HEAVY LOAD DROP ANALYSIS

3.1 Introduction

The reactor building heavy load drop analysis is based on the assumption that postulated load drops will result in the local failure of floors. An evaluation was made to ensure that the postulated failure cannot result in draining the reactor vessel below 314'-0" or disabling all makeup paths to the RPV.

3.2 Identification of Loads

This SER addresses the handling of any load in excess of 2400 lbs. inside the containment. It is anticipated that these loads will range up to a maximum of 25 tons; however, this SER addresses all "heavy loads" (as defined by NUREG-0612) from 2400 lbs. up to the 170 ton rated capacity of the main hook of the polar crane.

3.3 Identification of Targets

The target for a postulated load drop is considered to be all the floors and equipment in the region directly below the suspended load.

3.4 Load/Target Interactions

The attached figures, 3.3-1 through 3.3-3, provide plans of the containment with the allowed load handling areas identified. These figures are partially based upon the reviews performed for References 2 and 3, and a review of the failure of the floor at Elevation 347'-6" and its impact on the availability of makeup to the reactor vessel. Failure of the floor resulting in damage to the in-core tubes, which could result in draining the reactor vessel, is precluded by restriction of load handling to areas outside the identified exclusion zones.

3.5 Criteria Specific Evaluation (NUREG-0612)

This section evaluates the results of heavy 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 This SER specifically excludes from its scope the handling of heavy loads over areas containing spent fuel, thus eliminating the potential for a load drop resulting in damage to spent fuel. Any activity releases caused by the load drops addressed in this SER would be released within the containment. The containment 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 of the containment purge exhaust system so as not to exceed the limits established in Criterion I.

The analyses in Reference 2 (for Krypton 85) showed that even when utilizing "worst case" assumptions (i.e. dropping the reactor vessel head on the reactor with no containment), the maximum whole body dose is 12 millirem compared to a limit of 6250 millirem. In addition, specific evaluations will be performed and results reported if conditions are not bounded by the referenced analysis. Thus Criterion I is met.

3.5.2.2 This SER does not allow the handling of heavy loads over the three exclusion areas or the fuel canister storage racks when filled canisters are in place. Therefore, as heavy loads will not be handled over these areas, which contain significant amounts of fuel, fuel reconfigurations are not postulated as a result of a heavy load drop.

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. However, the systems identified below will require isolation (per Reference 3) prior to handling heavy loads within the 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

The isolation of these non-borated water sources during the handling of heavy loads will ensure that a sump criticality event cannot occur; thus Criterion II is met.

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 and any necessary makeup may be provided by the makeup system via redundant pathways to the reactor vessel; thus Criterion III is met.

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, reach and 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.

The reactor coolant pressure boundary needs to be maintained insofar as reactor coolant must be maintained in the reactor coolant system (RCS) for decay heat removal and reactivity control. 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 is 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 of 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.

Based on the above, Criterion IV is met.

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 possibility 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. 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 question 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 Reference 2. 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).

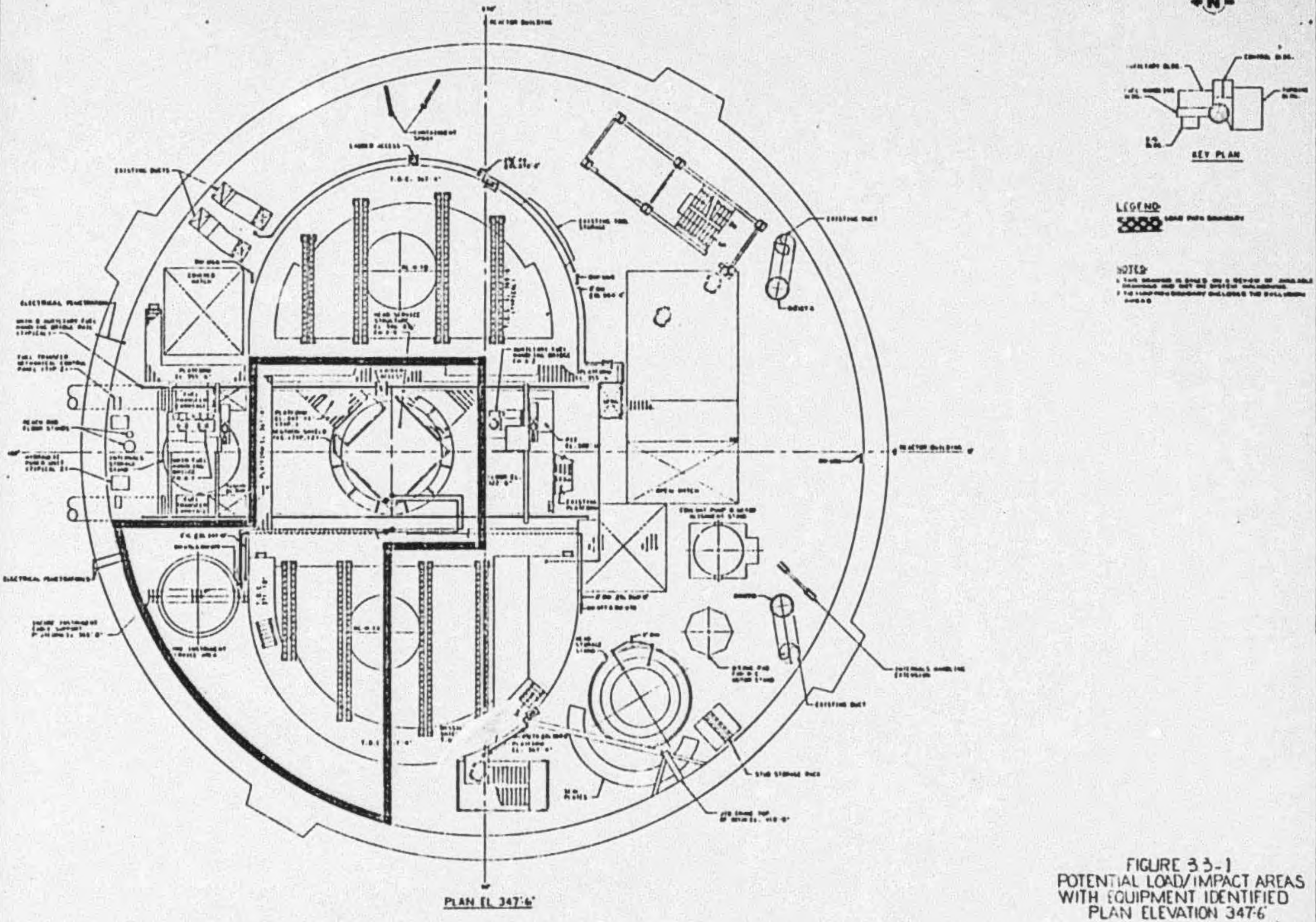

FIGURE 3.3-1
POTENTIAL LOAD/IMPACT AREAS
WITH EQUIPMENT IDENTIFIED
PLAN ELEVATION 347'-6"

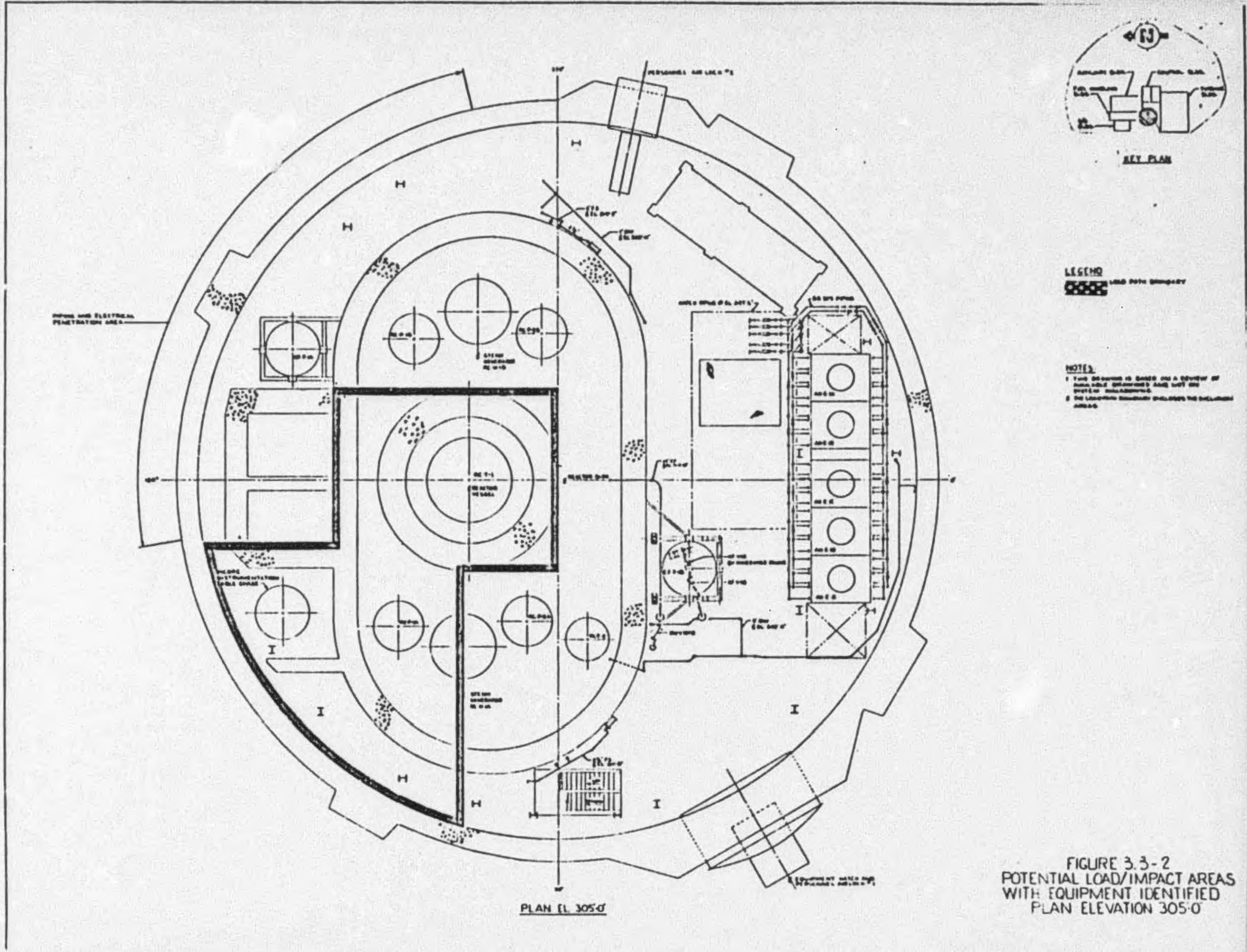

FIGURE 3.3-2
POTENTIAL LOAD/IMPACT AREAS
WITH EQUIPMENT IDENTIFIED
PLAN ELEVATION 305'0"

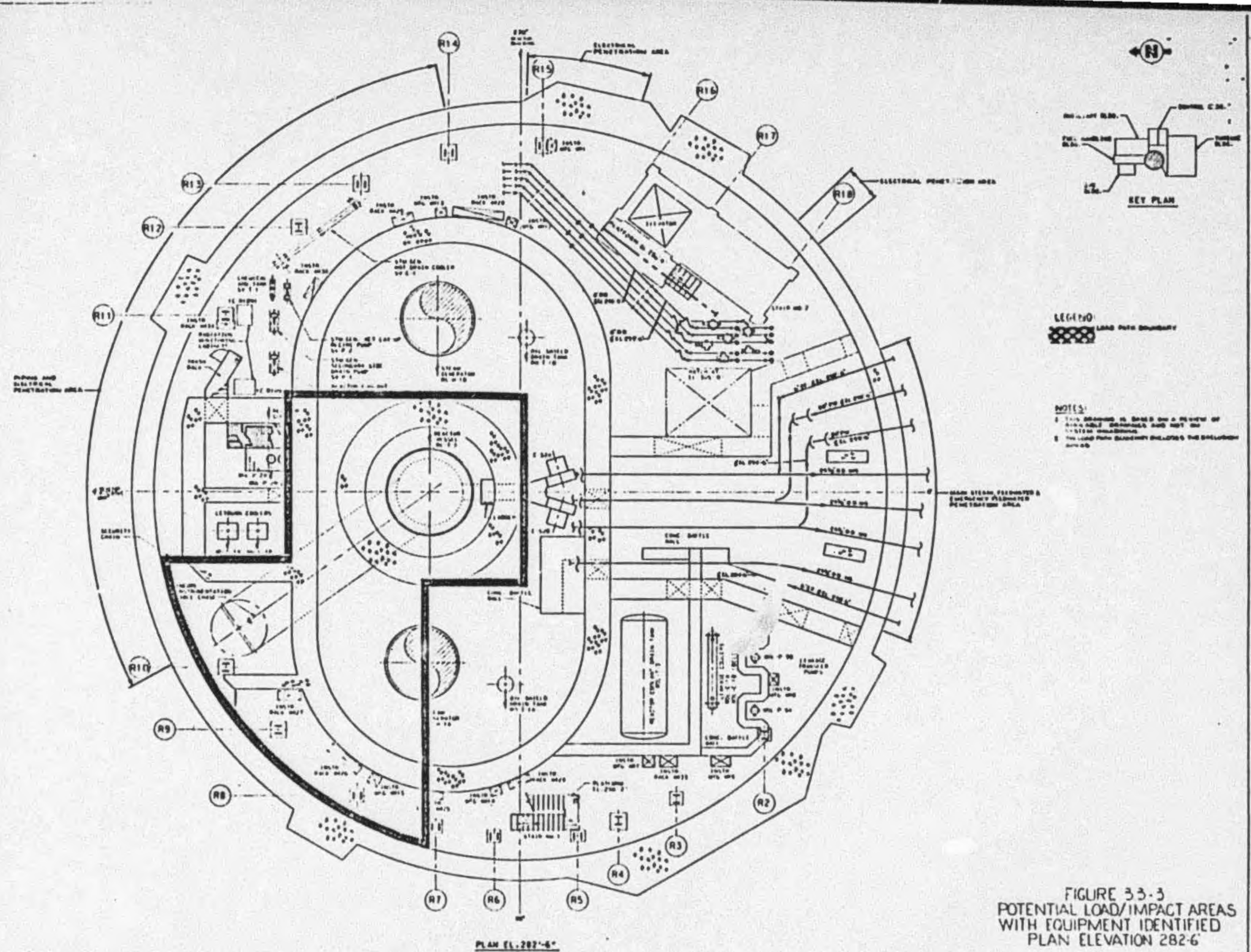

FIGURE 3.3-3
POTENTIAL LOAD/IMPACT AREAS
WITH EQUIPMENT IDENTIFIED
PLAN ELEVATION 282'-6'