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WILLIAM D. HARRINGTON
SENIOR VICE PRESIDENT
NUCLEAR

July 13, 1983

BECo Letter No. 83-181

Mr. Darrell G. Eisenhut, Director
Division of Licensing
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

License No. DPR-35
Docket No. 50-293

Subject: NUREG 0612: Control of Heavy Loads

References: (A) NRC Letter, "Control of Heavy Loads," dated December 22, 1980.

(B) Boston Edison Letter from Mr. W.D. Harrington to Mr. Darrell G. Eisenhut dated February 28, 1983.

Dear Sir:

In the letter of February 28, 1983, Boston Edison committed to provide a report which would satisfy the six and nine month reports as described in your letter of December 22, 1980.

Accompanying this letter is a report consisting of two enclosures. Enclosure 1 provides responses to Franklin Research Center's draft TER-C5257-109, dated February 12, 1982. Enclosure 2 responds to requests for information contained in Enclosure 3, Sections 2.2 and 2.3 of your December 22, 1980 letter.

We believe this report demonstrates that most of Pilgrim's lifting devices and procedures satisfactorily meet the guidelines of NUREG-0612. However, a number of procedural changes and modifications have been indicated necessary to ensure complete compliance with the goals of the NUREG. We shall complete all such modifications and procedure emplacements by December 31, 1984.

We believe this report satisfies the six and nine month report requirements. Should you require any information regarding this submittal, please contact us.

Very truly yours,

W D Harrington

PMK/mat

Attachments: Heavy Loads Report

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ENCLOSURE I
RESPONSES TO FRANKLIN RESEARCH CENTER
DRAFT TER - C5257-109 DATED FEBRUARY 12, 1982

NRC Request: (from draft Franklin TER Section 2.1.1.c)

Although FRC agrees with the Licensee about the applicability of NUREG-0612 to the reactor building bridge crane, insufficient information has been provided to allow an evaluation of compliance of other load handling systems with the guidelines. The Licensee should provide documentation of compliance or justification for exclusion.

RESPONSE:

As a result of this FRC item and a better understanding of the criteria for excluding handling systems, BECo has reevaluated its initial response. Plant arrangement drawings, vendor equipment lists and area surveys were utilized to identify all handling systems that could carry heavy loads and to develop justification for the exclusion of particular handling systems from the scope of NUREG-0612. The handling systems identified as being within the scope of NUREG-0612 are:

	Handling System	Location	Capacity
(1)	Reactor Building Bridge Crane	RB - 117' el.	100 ton (M) 5 ton (A)
(2)	Turbine Building Bridge Crane	Turbine Building - 51' el.	165 ton (M) 25 ton (A)
(3)	RHR Pump and Motor Hoists/Monorails (2)	SE & NW Quadrants RB - 23' el.	5 ton/ea
(4)	Recirculation Pump and Motor Hoist/ Monorail	Drywell	20 ton
(5)	Fuel Pool & Reactor Water Cleanup Filter Equipment Hatch Hoists/Monorails (2)	SE Quadrant RB - 91' el.	5 ton
(6)	Reactor Auxiliary Bay Equipment Hatch Hoist/ Monorail	Reactor Auxiliary Bay - 23' el.	5 ton
(7)	Recirculation Pump MG Set Hoist/Monorail	RB - 51' el.	8 ton

A listing of the handling systems excluded and the basis for exclusion are provided below:

a) Channel Handling Boom

This is a 200 lb. crane located on the refueling floor. The 200 lb. rating is less than that of a heavy load, where a heavy load is greater than the combined weight of a single spent fuel assembly and its handling tool (1500 lbs. for Pilgrim Station). No load drop could result in damage to safe shutdown or decay heat removal equipment, and this hoist cannot carry loads over the spent fuel pool or reactor vessel. Therefore, NUREG-0612 does not apply to this equipment.

b) A & B Refueling Jib Cranes

These are ½ ton cranes over the spent fuel pool on the refueling floor. One also may be used over the reactor vessel to aid in refueling operations. The ½ ton rating is less than that of a heavy load. Also, these cranes do not travel over safe shutdown equipment. Therefore, NUREG-0612 does not apply to this equipment.

c) CRD Maintenance Hoist/Monorail

The 2 ton hoist for the CRD maintenance hoist is located in the CRD Maintenance Shop at elevation 23' in the southeast quadrant of the Reactor Building. This hoist/monorail system is no longer in use, therefore NUREG-0612 requirements do not apply to this equipment.

d) Decontamination Room Hoist

This hoist is located near the North wall on elevation 23' of the reactor building. It is no longer used. Therefore, the requirements of NUREG-0612 do not apply to this equipment.

e) Fuel Rod Storage Hoist

The 2 ton fuel rod storage hoist is used to move fuel in the spent fuel pool and reactor vessel. This hoist is used only to move fuel and smaller tools i.e., no heavy loads are handled. In addition, no safe shutdown or decay heat removal equipment could be damaged by this crane. Therefore, NUREG-0612 requirements do not apply to this crane.

f) Feedwater Heater A-Frame
Reactor Feed Pump Hoist

These handling systems are located on elevation 51' of the Turbine Building. Although there is cabling associated with safe shutdown systems located at elevations below the 51' el. in this area, loss of these cables would not result in an inability to accomplish safe shutdown. Therefore, NUREG-0612 requirements do not apply to this equipment.

g) Turbine Basement Hoist
Radwaste Area Bridge Crane
Reactor Clean-up Sludge Disposal Bridge Crane
Off-gas Filter/Shipping Cask Monorail Hoist
Retention Building Prefilter Monorail Hoist

These cranes are located in the radwaste area of the plant, the main stack, or the Retention Building. No spent fuel or equipment necessary for safe shutdown or decay heat removal is located in these areas. Therefore, the requirements of NUREG-0612 do not apply to this equipment.

h) Electric Wire Rope Hoist
Machine Shop Decontamination Trough Davit

These cranes are located in the machine shop area of the plant. There is no spent fuel, safe shutdown or decay heat removal equipment that could be damaged by a load drop from either of these cranes. Therefore, the requirements of NUREG-0612 do not apply to this equipment.

i) General Area Monorail

This 1500 lb monorail is located at the 91'3" el. of the Reactor Building and traverses over the General Tool Cage, Receiving Cage, Relief Valve Cage and the Rigging Tool Cage. The 1500 lb rating for this monorail system is less than that of a heavy load for PNPS. Therefore, the requirements of NUREG-0612 do not apply.

j) Turbine Building Mezzanine Monorail

This monorail is located at the 64' el. of the Turbine Building. It is used for maintenance of the Reactor and Turbine Building exhaust fans. A load drop from this

monorail could not impact any safe shutdown equipment. Therefore, the requirements of NUREG-0612 do not apply to this monorail.

k) Diesel Generator Monorails (4)

Each diesel generator has a monorail along each side running the entire length of the diesel generator. The monorails are only used for maintenance of their respective diesel generators. A drop from a monorail could only affect one diesel generator which would already be out of service for maintenance. Therefore, the requirements of NUREG-0612 do not apply to this monorail.

Tables 1, 2, and 3 provide listings of loads, load weights and load handling procedures for the Reactor Building Crane, Turbine Building Crane, and Monorail/hoist systems.

Evaluation of Turbine Building Crane Against NUREG-0612, Section 5.1.1 - General Guidelines

The Turbine Building Crane was not previously addressed in Boston Edison's June 25, 1981 submittal. Information is provided below to address the General Guidelines in NUREG-0612, Section 5.1.1.

NUREG Section 5.1.1(1) - Safe Load Paths

As indicated in Table 2, there are two distinct load handling areas in terms of potential load drop consequences associated with the Turbine Building Crane. These are Load Impact Regions 16 and 17 (see Figures 10 and 11 in Enclosure 2 to this submittal). Load drops in Region 16 will not affect the ability of safe shutdown systems to perform their safety functions.

On this basis, and conservative structural analyses of the Region 17 floor area, load handling procedures will impose two levels of administrative control. As indicated in Enclosure 2, more rigorous structural analyses are being performed that may result in refinement of the specifics of these administrative restrictions. The first level of administrative control is illustrated in Figure 1 and involves defining an exclusion area over which only certain heavy loads can be carried. All other heavy loads will be handled by the Turbine Building Crane in Region 16 outside of this exclusion area.

The second level of administrative control involves defining specific safe load paths for certain limiting loads within Region 17. The loads involved are listed in Table 2. The safe load paths are illustrated in Figure 2. In addition to safe load paths, procedures will specify limits on carry heights within this region.

NUREG Section 5.1.1(2) - Procedures

Load handling procedures containing the information described in NUREG-0612, Section 5.1.1(2) will be developed and implemented for the Turbine Building Crane prior to movement of any heavy loads within the exclusion area.

Deviations from safe load paths will be controlled in the manner described in the response to draft TER Open Item 2.1.2.c (page 10 of this enclosure).

NUREG Section 5.1.1(3) - Crane Operators

Crane operators will be qualified in accordance with ANSI B30.2.

NUREG Section 5.1.1(4) - Special Lifting Devices

Special lifting devices utilized with the Turbine Building Crane are used for lifts over Region 16 only. Because load drop consequences in Region 16 will not affect the ability of safe shutdown systems to perform their safety functions, comparison of these lifting devices to the requirements of ANSI N14.6-1978 was judged not to be warranted.

NUREG-0612, Section 5.1.1(5) - Non Special Lifting Devices

Sling selection, use and maintenance will be controlled as indicated in the response to draft TER Open Item 2.1.6.b (page 19 of this enclosure).

NUREG Section 5.1.1(6) - Handling System Design

The Turbine Building Crane was designed to crane industry standard EOCI-61. As a result, the Turbine Building Crane design has been evaluated by verifying that the 10 CMAA-70 requirements identified in draft TER Open Item 2.1.8.c have been satisfied or compliance justified by equivalent means. The results and conclusions of this comparison are provided in the response to draft TER Open Item 2.1.8.c (page 20 of this enclosure).

NUREG Section 5.1.1(7) - Handling System Inspection, Testing and Maintenance

Maintenance procedures will be implemented for the Turbine Building Crane that are consistent with the inspection, testing and maintenance guidelines of ANSI B30.2-1976.

Evaluation of Monorails/Hoists Against NUREG-0612, Section 5.1.1 - General Guidelines

The handling systems (listed as 3 through 7 above) were not previously addressed in Boston Edison's June 25, 1981 submittal. Information is provided below to address the General Guidelines in NUREG-0612, Section 5.1.1 for each of these monorail/hoist systems.

NUREG Section 5.1.1(1) - Safe Load Paths

Load paths for monorail systems are defined by the limits of the monorail track. In one case, however, limitations on movement along the monorail track were judged to be necessary.

Heavy load movements with the Recirculation Pump MG Set Monorail/Hoist are not generally anticipated during power operation. This is because Technical Specifications limit the time that a Recirculation System loop can be out of service to only 24 hours prior to proceeding to the cold shutdown condition.

If heavy loads are ever moved in this area at power, then the concern becomes the potential for damage to CRD hydraulic control units below the 51' el. deck in this area. On the basis of structural analysis of load impacts on this deck, it was determined that certain administrative controls were prudent for this area if heavy loads are to be moved at power. These administrative controls are illustrated and described in Figure 3. They will be incorporated into load handling procedures applicable to this handling system.

NUREG Section 5.1.1(2) - Procedures

Load handling procedures containing the information described in NUREG-0612, Section 5.1.1(2) will be developed and implemented for each of the identified monorail/hoist systems prior to lifting any heavy loads with these handling systems.

NUREG Section 5.1.1(3) - Crane Operators

Hoist operators for the identified monorail/hoist systems will be qualified in accordance with ANSI B30.2-1976.

NUREG Section 5.1.1(4) - Special Lifting Devices

No special lifting devices have been identified that are used with the identified monorail/hoist systems.

NUREG Section 5.1.1(5) - Non Special Lifting Devices

Sling selection, use and maintenance will be controlled as indicated in the response to draft TER Open Item 2.1.6.b (page 19 of this enclosure).

NUREG Section 5.1.1(6) - Handling System Design

The design of the following handling systems has been evaluated as described below.

- o RHR Hoist/Monorails (2) - 5 ton capacity
- o Recirculation Pump Motor Hoist/Monorail - 20 ton
- o Fuel Pool and Reactor Water Cleanup Filter Equipment Hatch Hoist/Monorails (2) - 5 ton
- o Reactor Auxiliary Bay Equipment Hatch Hoist/Monorail - 5 ton
- o Recirculation Pump MG Set Hoist/Monorail - 8 ton

These handling systems are single monorail tracks suspended from Reactor Building, Reactor Auxiliary Bay, or Drywell structural I-beams. The criteria of ANSI B30.2 and CMAA-70 are not applicable to the design of handling systems such as these monorails and hoists. Accordingly the design of the monorail/hoist systems was compared to the criteria in applicable standards, i.e. ANSI B30.11, "Monorail Systems and Underhung Cranes - 1980," and ANSI B30.16, "Overhead Hoists -1973."

Based on a point-by-point comparison to these standards, it was found that these monorail systems conform to the criteria in these current standards, including requirements for maximum monorail deflection and monorail stress design safety factors with the exception of demonstrating a design safety factor of 4 for hoist components, and providing a warning label on hoists.

ANSI B30.16 requires that for hoist components the stress due to the rated load shall not exceed 25% of the average ultimate material strength. Stress design safety factors are not available for the chain type hoists, and the complexity of these devices does not lend them to performance of stress analyses. However, safety margins were adequately addressed in the purchase specifications by requiring rigorous load testing. The following summarizes the load tests required in the purchase specifications for these hoists:

(1) Proof testing of the hook to 200% of rated capacity, followed by NDE to demonstrate no permanent deformation;

(2) Proof test of the load chain to 300% rated capacity;

(3) Load test by the supplier at 150% rated capacity for the completed hoist and trolley, prior to shipment; and

(4) Performance of a 150% load test of the hoists, trolley and monorail through raising, lowering and travelling operations. This test was performed after final installation.

Additionally, inspection and maintenance procedures will be implemented that assure monorail and hoist components remain in good working condition. The load tests performed on these hoists and the continuing inspection and maintenance demonstrate an adequate achieved safety margin.

ANSI B30.16 requires a WARNING label on the hoist or load block to caution personnel against overloading; the danger of using twisted; kinked or damaged hoist cable, slings or hoist chains; using a damaged or malfunctioning hoist; lifting people; or operating the hoist with other than manual power. The Bechtel specification for these hoists did not require application of such a label, and an inspection of these hoists has verified that such warning labels are not affixed to these hoists. Boston Edison will add warning labels to these hoists that meet ANSI B30.16.

NUREG Section 5.1.1(7) - Handling System Inspection, Testing and Maintenance

Maintenance procedures will be implemented for the identified monorail/hoist systems that are consistent with the inspection, testing and maintenance requirements of ANSI B30.11-1980 and ANSI B30.16-1973.

NRC Request: (from draft TER Section 2.1.2.c)

The Pilgrim Nuclear Power Station does not comply with Guideline 1 of NUREG-0612. To comply, the Licensee should perform the following:

1. Verify that safe load paths are clearly marked.

2. Verify that deviations from established load pathways require written alternatives which must be specifically approved by the plant safety review committee.

RESPONSE:

1. Safe load path markings - As indicated in Boston Edison's submittal of June 25, 1981, marking of safe load paths on the refueling floor or the Turbine Building operating deck is not practical because of the extensive use of temporary coverings on the floor during outages. Further, current procedures and practices provide alternative visual aids to the crane operator during heavy load movements that meet the intent of this NUREG-0612 guideline. These practices are summarized below:

   (a) The crane operator/signalman will verify the load path prior to load movement to assure that it is clear of obstructions.

   (b) The signalman will have load movement procedures including figures indicating the safe load path in his possession or will have reviewed the specific load paths involved prior to movement and will direct the crane operator along the designated load path in accordance with the procedures.

2. Deviations from Safe Load Paths - Heavy load handling procedures which include safe load paths are safety-related procedures and accordingly changes to these procedures (including deviations from safe load paths) are controlled. For PNPS this involves preparation of a safety evaluation and review and approval of the Operations Review Committee (ORC).

NRC Request: (from draft TER Section 2.1.5.b)

The Licensee has provided insufficient information to allow evaluation of the compliance of special lifting devices at the Pilgrim plant with Guideline 4 of NUREG-0612.

RESPONSE:

Special lifting devices utilized with the Reactor Building Crane that have been evaluated against the requirements of ANSI N14.6-1978 for the purpose of developing a response to this item are listed below:

1. Head Strongback
2. Dryer/Separator Lifting Sling Assembly

The dryer/separator sling assembly and the head strongback were evaluated against ANSI N14.6 as described below.

Description of Dryer and Separator Sling

The dryer and separator sling is used to remove and install the dryer and the steam separator assembly. The device is a cruciform steel frame attached to a hook box by four wire ropes with turnbuckles. The four ends of the cruciform frame are each fitted into a bell-shaped housing which is open and flared at the bottom. A hole passes through two sides of the housing for the lifting pin travel. Each lifting pin is actuated by a double-acting air piston. The lifting pin, in turn, actuates an air valve at the end of the pin's travel. This air valve gives positive indication by way of a pressure gauge, that the lifting pin is fully inserted into the dryer and separator lifting lug. A lifting eye, located on top of each I-beam, is connected to a turnbuckle and a wire rope. The wire ropes are attached to the hook box by spelter sockets and pins. The hook box contains a slot at the top which is sized to accommodate the double hook

of the crane. Two hook pins pass through the hook box to engage the crane hook.

Description of the Head Strongback

The head strongback is used to hoist the drywell head and the reactor vessel head. The device consists of four lifting arms mounted at right angles between top and bottom four-point star plates. The top plate has a slot through which the double hook of the crane passes to engage the two hook pins. The strongback is attached to lifting lugs on the drywell head and reactor vessel head, and to lifting lugs at the end of each arm of the strongback, by turnbuckles and anchor shackles.

For the reasons listed below, the detailed comparison of the Dryer/Separator Sling Assembly and the Head Strongback to ANSI N14.6-1978 was limited to Sections 3.2 and certain parts of Section 5 of the standard.

1) These devices were designed by General Electric Company prior to the existence of ANSI N14.6-1978. In this regard, there are a number of sections in the standard that it is not reasonable to apply in retrospect. These are the sections entitled, Designer's Responsibilities (Section 3.1); Design Considerations (Section 3.3); Fabricator's Responsibilities (Section 4.1); Inspector's Responsibilities (Section 4.2); and Fabrication Considerations (Section 4.3). Because documentation is not available to assure that all of the subparts of these sections were met, they have not been addressed item by item for the purpose of identifying and justifying exceptions. However, information on the design drawings indicate that sound engineering practices were placed on the fabricator and inspector by the designer for the

purpose of assuring that the designer's intent was accomplished. On this basis, there is reason-able assurance that the intent of the sections of the standard listed above was, in fact, accomplished in the design, fabrication, inspection, and testing of these devices.

2) Section 1.0, Scope; Section 2.0, Definitions; Section 3.4, Design Considerations to Minimize Decontamination Effects in Special Lifting Device Use; Section 3.5, Coatings; Section 3.6, Lubricants; and Sections 5.2.3, 5.3.4 and 4.3.5 related to functional testing of non-load bearing parts are not pertinent to load handling reliability of the devices and, therefore, have not been addressed for the purpose of identifying and justifying exceptions.

3) Section 6, Special Lifting Devices for Critical Loads, is applicable to critical loads. A critical load is defined in the standard as:

> "Any lifted load whose uncontrolled movement or release could adversely affect any safety related system when such system is required for unit safety or could result in potential off-site exposures comparable to the guideline exposures outlined in <u>Code of Federal Regulations</u>, Title 10, Part 100."

The applicability of Section 6.0 of ANSI N14.6-1978 is discussed in NUREG-0612 Section 5.1.6. This Section of the NUREG indicates that special lifting devices utilized for heavy load handling with cranes that rely on upgrading to single-failure-proof to demonstrate compliance, should comply with ANSI N14.6-1978, Section 6.0. The Reactor Building Crane with which the two special lifting devices referred to above are utilized has not been upgraded to single-failure-proof. Therefore, Section 6.0 does not apply.

ANSI N14.6 - Section 3.2

Section 3.2 of ANSI N14.6-1978 establishes design criteria for special lifting devices. Specifically, it establishes (1) stress design factors for load bearing members and (2) brittle fracture criteria for materials used in load-bearing members.

Design documents necessary to verify compliance or identify exceptions to these criteria are not available. Nonetheless, it is believed that adequate verification of the design safety margins have been demonstrated based on the following:

(1) Proof Load Tests - The Dryer and Separator Sling Assembly and Head Strongback were required by drawing specification to be proof-tested at 125% of their rated capacity. Thorough visual, dimensional and NDE examinations were required following the proof test.

(2) In Service Examinations - Both devices have been utilized on many occasions to perform the lifts for which they were designed with no evidence of overstress or permanent deformation.

(3) The devices are utilized only for the lifts for which each was specifically designed. They have not been and will not be used for any other purpose. Therefore, the possibility of an overload situation is extremely remote.

(4) As indicated in the discussion below regarding inspection and maintenance, the devices will periodically be subjected to necessary visual, dimensional, and nondestructive examination. This should assure that any indication of overstress will be detected and action taken to repair or replace the damaged components.

ANSI N14.6 - Section 5

The comparison of current practices for inspection, testing, and maintenance of the dryer/separator sling and head strongback to Section 5 of ANSI N14.6-1978, as supplemented by NUREG 0612, Section 5.1.1(4), found that certain changes to PNPS procedures were required in order to satisfy the inspection and test requirements in ANSI N14.6. These changes will be made so that the PNPS inspection program complies fully, with the following exceptions:

Exception I: Section 5.3.1 of the Standard requires that either load testing or a comprehensive inspection program be undertaken annually for each special lifting device to assure continuing compliance with the standard. The option of periodic inspection has been chosen for the PNPS lifting devices.

ANSI N14.6 was developed to be applicable "for special lifting devices for shipping containers weighing 10,000 pounds or more for nuclear materials," most notably lifting devices for casks. The service environment for lifting devices such as casks is different and generally more severe than the service environment for the PNPS lifting devices. Accordingly, it is our position that a less restrictive inspection program is warranted to assure continued serviceability for the PNPS lifting devices than that which is specified in ANSI N14.6. We propose that the full set of inspections prescribed in ANSI N14.6-1978, Section 5.3.1(2) be completed on a five-year interval. Additionally, thorough visual examinations prior to each period of use of the lifting devices will be undertaken. This inspection program is judged to be equivalent to the intent of ANSI N14.6 and to provide sufficient periodic inspection and examination to identify wear or degradation that could potentially reduce design safety margins.

The bases for the extended frequencies for certain inspections are as follows:

o Frequency of Usage

Since the lifting devices identified for PNPS are typically used on an annual basis to support refueling operations, the frequency of use is considerably less than that of the special lifting devices for which ANSI N14.6 was developed. Special lifting devices for items such as casks

are potentially used between 50 to 100 times annually. The reduced frequency of use limits the number of stress cycles to which the PNPS devices are subjected and, in turn, the cumulative usage factor and the potential for abuse and damage.

o Controlled Environment

The PNPS lifting devices are stored inside the Reactor Building in a dry, chemical free environment. All lifting devices are inspected for cleanliness and cleaned prior to each use. On the contrary, the lifting devices for items such as casks for which ANSI N14.6 was developed are subjected to harsh environments that may include rain, road dust, road salt, and other potentially deliterious materials, as well as greater abuse since they are transported on open truck flatbeds. Furthermore, as part of normal service, casks and their lifting devices must be decontaminated, which requires the use of various acidic and caustic solutions. The absence of potentially corrosive compounds and solutions lessens the likelihood of environmental service related damage to the PNPS lifting devices.

In conclusion, the service conditions are relatively mild and operating procedures provide substantial assurance that heavy loads will be handled in a safe manner, minimizing the potential for damage of the lifting devices. The comprehensive PNPS visual and 5-year dimensional and nondestructive examination program will adequately confirm that design margins of safety have not been compromised due to potential service related mechanisms of degradation.

Exception 2: Plant procedures do not specify a visual inspection by maintenance or other nonoperating personnel at intervals of three months or less as required by Section 5.3.7 of ANSI N14.6-1978. Between periods of usage, these devices are stored in a specific location under a controlled environment and are not subjected to any other usage except the dedicated and specific usage mentioned in the description of the devices. Procedures require that the devices be inspected and examined by qualified personnel prior to usage as described above. Based on the controlled storage, dedicated usage, and the complete inspection schedule, the equivalency of Section 5.3.7 is demonstrated.

Exception 3: Section 5.3.3 of ANSI N14.6-1978 requires that special lifting devices be load tested according to Section 5.2.1 to 150% of maximum load following any incident in which any load-bearing component may have been subjected to stresses substantially in excess of those for which it was qualified by previous testing, or following an incident that may have caused permanent distortion of load-bearing parts. Since distortion may already have occurred or defects may have already developed due to the overstressed condition, it seems more prudent and practical to perform the dimensional examinations for deformation and the nondestructive examinations for defects to determine whether the device is still acceptable for use rather than to subject the device to 150% load testing. If major repairs are required, the device shall be repaired or modified and then tested to 150% load followed by examination for defects or deformation. Major repairs are defined in Section 5.3.2 of ANSI N14.6-1978. This alternative achieves the same objective as Section 5.3.3 of the standard.

Exception 4: Section 5.2.1 of the standard requires an initial load test of 150% of rated load. The PNPS special lifting

devices were subjected to proof load tests of 125% of rated load. This is consistent with industry standards for other heavy lifting equipment such as cranes and therefore, is judged to be adequate to demonstrate load carrying capacity substantially in excess of rated load.

NRC Request: (from draft TER Section 2.1.6.b)

The Licensee has provided insufficient information to allow evaluation of compliance of the Pilgrim plant's not-specifically-designed lifting devices with Guideline 5 of NUREG-0612.

RESPONSE:

Heavy load handling procedures for all handling systems included within the scope of NUREG-0612 require that sling selection be based on the applicable sections of ANSI B30.9-1971 for various sling types. This involves:

(1) An accurate designation of the load weight,

(2) the addition of a dynamic load factor to the load weight of 1/2% of the load weight for each ft/min of hoist-speed (this factor is designated in the procedures for specific hoists), and

(3) selecting a sling of appropriate capacity for the combined load weight and dynamic loading factor.

All slings will be inspected, tested, repaired and replaced in accordance with the applicable sections of ANSI B30.9-1971.

NRC Request: (from draft Franklin TER, Section 2.1.8.c)

The Pilgrim Nuclear Power Station complies with NUREG-0612, Section 5.1.1, Guideline 7, to a substantial degree, on the basis of compliance with EOCI-61 criteria. However, the Licensee should provide information to verify that (1) the following CMAA-70 requirements have been satisfied for cranes subject to this review or (2) the requirements of CMAA-70 have been satisfied by equivalent means:

1. nonsymmetrical girder sections were not used in construction of the cranes

2. any longitudinal stiffeners in use conform to the requirements of CMAA-70, and allowable h/t ratios in box girders using these stiffeners do not exceed ratios specified in CMAA-70

3. girders with b/c ratios in excess of 38 were not used

4. fatigue failure was considered in crane design and the number of design loading cycles at or near rated load was less than 20,000

5. the sum of maximum crane load weight and the weight of the bottom block, divided by the number of parts of rope, does not exceed 20% of the manufacturer's published rope breaking strength

6. drum design calculations were based on the combination of crushing and bending loads

7. drum groove depth and pitch conform to the recommendations of CMAA-70

8. mechanical load brakes or hoist holding brakes with torque ratings of approximately 125% of the hoist motor torque were used

9. any static control systems in use conform to the requirements of CMAA-70

10. controllers used were of the spring-return or momentary-contact pushbutton type.

Response:

The requested verification is provided below for the Reactor Building and Turbine Building Overhead Bridge Cranes.

Reactor Building Crane

The PNPS Reactor Building Crane was built prior to the issuance of ANSI B30.2-1976 and CMAA 70-1975. This crane was designed and fabricated by Crane

Manufacturing and Service Corporation in accordance with EOCI-61, "Specifications for Electric Overhead Traveling Cranes-1961," and additional criteria contained in Bechtel Specification No. 6498-M-23, Rev. 1, October 24, 1968. These specifications addressed certain, but not all, of the criteria in ANSI B30.2-1976 and CMAA 70-1975. To address the 10 points identified in the Franklin Research Institute's TER where CMAA-70 and ANSI B30.2 are more restrictive than EOCI-61, a design evaluation of the PNPS Reactor Building Crane was performed. The following summarizes our findings for these 10 points.

(1) Torsional Forces - CMAA 70 specifies that twisting moments be determined based on the horizontal distance between the center of gravity and the shear center of the girder section. EOCI-61 requires twisting moments to be based on the distance between the load center of gravity and the beam center of gravity. Since the PNPS Reactor Building Crane girders are symmetrical box sections, these two requirements are the same. Since the trolley rails are located over the centerline of the girders, there are no appreciable torsional forces on the girders. Thus PNPS Reactor Building Crane satisfies CMAA 70 criteria relative to torsional forces.

(2) Longitudinal Stiffeners - CMAA 70 specifies a minimum moment of inertia for longitudinal stiffeners, maximum width to thickness ratio, and stiffener location along the web plate. EOCI does not provide similar guidance. For the PNPS Reactor Building Crane, application of the CMAA criteria requires that the moment of inertia be greater than I_o = 52.15-in.^{4}, the width to thickness ratio should be less than 12, and the stiffener should be located 0.4 of the distance from the compression plate to the web neutral axis. The actual moment of inertia is 159-in.^{4}, the stiffener width to thickness ratio is 9.6, and the stiffener centerline is located 0.43 of the distance from the

compression plate to the web neutral axis. Thus the CMAA criteria relative to longitudinal stiffeners are sat'sfied for this crane.

CMAA 70 specifies that l/h (l = girder span; h = web height) should be less than 25; EOCI-61 has no limit on l/h. For the PNPS Reactor Building Crane, l/h = 1228 in./84 in. = 14.6. Therefore, CMAA 70 is satisfied.

In addition, CMAA 70 specifies that h/t be less than

$C(K+1) \sqrt{\frac{17.6}{f_c}}$ and less than M, where:

t	=	web thickness = 3/8 in.
C	=	162 (the PNPS Reactor Building Crane has one longitudinal stiffener)
K	=	$f_t/f_c = 1.0$
f_t	=	max. tensile stress = 16.0 ksi
f_c	=	max. compressive stress = 16.0 ksi
M	=	376

Therefore according to CMAA 70, h/t should be less than 340.2 and less than 376. h/t = 84/(3/8) = 224. Therefore, CMAA 70 is satisfied.

(3) Basic Allowable Stresses - EOCI-61 is more conservative than CMAA 70 for allowable tension, compression, and shear stresses, if b/c is less than 38 (b is distance between web plates and c is the thickness of the cover plate). For the PNPS Reactor Building Crane, b/c is 24 in./1.5 in. = 16. Therefore, CMAA 70 is satisfied.

(4) Fatigue Failure and Cyclic Loading - CMAA 70 specifies that fatigue failure be considered in the crane design, and also specifies an allowable stress range for crane structural members that are subject to cyclic loading of greater than 20,000 over the life of the crane. The number of cycles for any of the crane members will be less than 2,000 over the life of the PNPS Reactor Building Crane. Based on this, failure due to cyclic fatigue should not be of concern for this crane, and the CMAA 70 criteria for cyclic loading are satisfied.

(5) Hoisting Rope - CMAA 70 specifies a 5:1 hoisting rope safety factor for the rated load plus bottom block divided by the number of parts of rope. For the PNPS Reactor Building Crane, the resulting safety factor for the main hoist is:

bottom block = 6,100 lbs.
rated load = 200,000 lbs.
parts of rope = 12 (1-1/8" each)
rope published breaking strength = 113,000 lbs.
resulting safety factor = 113,000/((200,000+6,100)/12)=6.57:1

For the aux. hoist:

block = 20 lbs. (8 1/2 ton capacity)
rated load = 10,000 lbs.
parts of rope = 1 (7/8" dia. - Type 304)
rope published breaking strength = (g.t.) 56,000 lbs.
resulting safety factor = 56,000/10,020 = 5.6:1

Therefore the ropes satisfy the criteria in CMAA 70.

(6) Hoist Drum Loads - CMAA 70 specifies that drum design should consider combined crushing and bending loads; however, EOCI 61 is not as specific. The Bechtel design specification for this crane required the design to consider combined crushing and bending loads. Therefore CMAA 70 is satisfied.

(7) Hoist Drum Groove - CMAA 70 specifies minimum drum groove depth and drum groove pitch; EOCI 61 does not provide such specific guidance. For the Pilgrim Reactor Building Crane, this guidance would require minimum drum groove depth and pitch of 0.42 in. and 1.25 in. respectively for the main hoist, and 0.33 in. and 1.0 in. for the aux. hoist. The actual dimensions are 0.4375 in. and 1.25 in. for the main hoist and 0.344 in. and 1.0 in. for the aux. hoist. Thus, the CMAA criteria are satisfied.

(8) Hoist Holding Brakes - CMAA 70 and ANSI B30.2 require that holding brakes have minimum torque ratings (relative to motor torque) of 125% if used with control braking other than mechanical; 100% if used with mechanical control braking.

For the PNPS Reactor Building Crane, the design specification called for brakes that are 150% of the full rated torque of the hoist motor, for both the main and auxiliary hoists. The main hoist brakes actually installed have a total torque rating of 370% of the full rated motor torque. The hoist holding brakes are, however, provided with a time delay so that both brakes are not applied at the same time. Each brake thus has a rating of 185% full motor torque.

The auxiliary hoist also has two holding brakes, applied with a time delay. Each brake has a torque rating of 200% full motor torque.

Therefore the holding brakes satisfy CMAA 70.

(9) Static Controls - CMAA 70 includes various criteria for crane static controls; EOCI only addresses crane magnetic controls. Since the Pilgrim Reactor Building Crane uses a D.C. magnetic control system, the CMAA 70 criteria on static controls are not applicable.

(10) Restart Protection - CMAA 70 establishes criteria for restart protection for cranes not provided with spring-return controllers or momentary contact pushbuttons; this is not addressed in EOCI 61. These CMAA 70 criteria are not applicable to the PNPSReactor Building Crane since this crane has spring-return pushbutton controls.

Turbine Building Bridge Crane

The PNPS Turbine Building Bridge Crane was also built prior to the issuance of ANSI B30.2-1976 and CMAA 70-1975. This crane was designed and fabricated by Whiting Corporation in accordance with EOCI-61, "Specifications for Electric Overhead Traveling Cranes-1961," and additional criteria contained in Bechtel Specification No. 6498-M-12, Rev. 1, February 19, 1968. These specifications addressed certain, but not all, of the criteria in ANSI B30.2-1976 and CMAA 70-1975. To address the 10 points identified in the Franklin Research Institute's TER where CMAA-70 and ANSI B30.2 are more restrictive than EOCI-61, a design evaluation of the PNPS Turbine Building Crane was performed. The following summarizes our findings for these 10 points.

(1) Torsional Forces - CMAA 70 specifies that twisting moments be determined based on the horizontal distance

between the center of gravity and the shear center of the girder section. EOCI-61 requires twisting moments to be based on the distance between the load center of gravity and the beam center of gravity. Since the PNPS Turbine Building Crane girders are symmetrical box sections, these two requirements are the same. Since the trolley rails are located over the centerline of the girders, there are no appreciable torsional forces on the girders. Thus the PNPS Turbine Building Crane satisfies the CMAA 70 criteria relative to torsional forces.

(2) Longitudinal Stiffeners - CMAA 70 specifies a minimum moment of inertia for longitudinal stiffeners, maximum width to thickness ratio, and stiffener location along the web plate. EOCI does not provide similar guidance. For the PNPS Turbine Building Crane, application of the CMAA criteria requires that the moment of inertia be greater than I_o = 169-in.^{4}, the width to thickness ratio should be less than 12, and the stiffener should be located 0.4 of the distance from the compression plate to the web neutral axis. The actual moment of inertia is 513-in.^{4}, the stiffener width to thickness ratio is 11, and the stiffener centerline is located 0.56 of the distance from the compression plate to the web neutral axis. The CMAA criteria on stiffener moment of inertia and width to thickness ratio are satisfied. Location of the longitudinal stiffener closer to the neutral axis means that less resistance to buckling of the web plate is provided, for the same stiffness afforded by the stiffener. Since for this crane, the actual stiffener moment of inertia is significantly above the CMAA 70 minimum (513/169 = greater than 300%), the design of the stiffener compensates for the deviation from CMAA 70 in the location of the longitudinal stiffener, and provides an adequate alternative to the CMAA criteria.

CMAA 70 specifies that l/h (l = girder span; h = web height) should be less than 25; EOCI-61 has no limit on l/h. For the PNPS Turbine Building Crane, l/h = 1228 in./84 in. = 14.6. Therefore, CMAA 70 is satisfied.

In addition, CMAA 70 specifies that h/t be less than

$C(K+1) \sqrt{\frac{17.6}{f_c}}$ and less than M, where:

t	=	web thickness = 3/8 in.
C	=	162 (the PNPS Turbine Building Crane has one longitudinal stiffener)
K	=	$f_t/f_c = 1.0$
f_t	=	max. tensile stress = 16.0 ksi
f_c	=	max. compressive stress = 16.0 ksi
M	=	376

Therefore according to CMAA 70, h/t should be less than 340.2 and less than 376. h/t = 84/(3/8) = 224. Therefore, CMAA 70 is satisfied.

(3) Basic Allowable Stresses - EOCI-61 is more conservative than CMAA 70 for allowable tension, compression, and shear stresses, if b/c is less than 38 (b is distance between web plates and c is the thickness of the cover plate). For the PNPS Turbine Building Crane, b/c is 19.75 in./1.875 in. = 10.5. Therefore, CMAA 70 is satisfied.

(4) Fatigue Failure and Cyclic Loading - CMAA 70 specifies that fatigue failure be considered in the crane design, and also specifies an allowable stress range for crane structural members that are subject to cyclic loading of greater than 20,000 over the life of the crane. The

number of cycles for any of the crane members will be less than 13,200 over the life of the PNPS Turbine Building Crane. Based on this, failure due to cyclic fatigue should not be of concern for this crane, and the CMAA 70 criteria for cyclic loading are satisfied.

(5) Hoisting Rope - CMAA 70 specifies a 5:1 hoisting rope safety factor for the rated load plus bottom block divided by the number of parts of rope. For the PNPS Turbine Building Crane, the resulting safety factor for the main hoist is:

bottom block = 18,000 lbs.
rated load = 330,000 lbs.
parts of rope = 16 (1¼" each)
rope published breaking strength = 123,000 lbs.
resulting safety factor = 123,000/((330,000 + 18,000)/16)=5.66:1

For the aux. hoist:

block =
rated load = 50,000 lbs.
parts of rope = 12 (9/16" each)
rope published breaking strength = 26,000 lbs.
resulting safety factor = 26,000/((50,000+1800)/12)=6:1

Therefore the ropes satisfy the criteria in CMAA 70.

(6) Hoist Drum Loads - CMAA 70 specifies that drum design should consider combined crushing and bending loads; however, EOCI 61 is not as specific. A Whiting Corporation representative stated that the drum design for Whiting cranes considers both crushing and bending loads. Therefore, CMAA 70 is satisfied.

(7) Hoist Drum Groove - CMAA 70 specifies minimum drum groove depth and drum groove pitch; EOCI 61 does not provide such specific guidance. For the PNPS Turbine Building Crane, this guidance would require minimum drum groove depth and pitch of 0.47 in. and 1.375 in. respectively for the main hoist, and 0.21 in. and 0.69 in. for the aux. hoist. The actual dimensions are 0.47 in. and 1.375 in. for the main hoist and 0.21 in. and 0.69 in. for the aux. hoist. Thus, the CMAA criteria are satisfied.

(8) Hoist Holding Brakes - CMAA 70 and ANSI B30.2 require that holding brakes have minimum torque ratings (relative to motor torque) of 125% if used with control braking other than mechanical; 100% if used with mechanical control braking.

For the PNPS Turbine Building Crane, the design specification called for brakes that are 150% of the full rated torque of the hoist motor, for both the main and auxiliary hoists. The main hoist brake actually installed has a total torque rating of 168% of the full rated motor torque.

The auxiliary hoist holding brake has a torque rating of 230% full motor torque.

Therefore the holding brakes satisfy CMAA 70.

(9) Static Controls - CMAA 70 includes various criteria for crane static controls; EOCI only addresses crane magnetic controls. Since the PNPS Turbine Building Crane uses a magnetic control system, the CMAA 70 criteria on static controls are not applicable.

(10) Restart Protection - CMAA 70 establishes criteria for restart protection for cranes not provided with spring-return controllers or momentary contact pushbuttons; this is not addressed in EOCI 61. The PNPS Turbine Building Crane motion control is provided in the cab by GE General-Duty Surface-Mounted Master Switches which do not have spring return. The control circuitry design includes an undervoltage release which prevents motor restart unless these controllers are in the neutral position. Crane motion control from the floor is provided by a pendant-mounted pushbutton station. These push buttons are spring-return. The CMAA 70 criteria are satisfied.

NRC Request (from draft TER Section 2.2.1)

Technical Specifications (Interim Protection Measure 1, NUREG-0612, Section 5.3(1))

"Licenses for all operating reactors not having a single-failure-proof overhead crane in the fuel storage pool area should be revised to include a specification comparable to Standard Technical Specification 3.9.7, 'Crane Travel - Spent Fuel Storage Pool Building,' for PWR's and Standard Technical Specification 3.9.6.2, 'Crane Travel,' for BWR's, to prohibit handling of heavy loads over fuel in the storage pool until implementation of measures which satisfy the guidelines of Section 5.1."

a. Summary of Licensee Statements and Conclusions

The Licensee made no statement and expressed no conclusions regarding this interim protection measure.

RESPONSE:

PNPS procedures define specific safe load paths for heavy loads handled in the vicinity of the spent fuel pool. These safe load paths assure that no heavy loads pass over the spent fuel pool with the exception of 2 loads which must pass over certain peripheral areas of the pool. The 2 loads are casks which are carried over the cask laydown area and the spent fuel gate, which is moved within the pool along the wall adjacent to the reactor cavity and periodically lifted out of the pool for maintenance. None of these safe load paths pass directly over spent fuel in the spent fuel pool. In addition, at certain times the Reactor Building Crane load block may be moved over the spent fuel pool. The load block has not been postulated to drop into the spent fuel pool on the basis that redundant upper limit switches will be provided and redundant holding brakes exist for the crane hoists. These redundant devices will assure that the probability of a two-blocking or uncontrolled lowering event will be sufficiently small to eliminate consideration of a postulated load block drop.

The procedures that include the safe load paths are safety related procedures and therefore, deviations are tightly controlled as described previously in the response to the open item related to draft TER Section 2.1.2.c. No additional procedural controls or technical specifications are judged to be necessary at this time.

NRC Request: (from draft TER Section 2.2.3)

Special Reviews for Heavy Loads Over the Core (Interim Protection Measure 6, NUREG-0612, Section 5.3(1)

"Special attention should be given to procedures, equipment, and personnel for the handling of heavy loads over the core, such as vessel internals or vessel inspection tools. This special review should include the following for these loads: (1) review of procedures for installation of rigging or lifting devices and movement of the load to assure that sufficient detail is provided and that instructions are clear and concise; (2) visual inspections of load bearing components of cranes, slings, and special lifting devices to identify flaws or deficiencies that could lead to failure of the component; (3) appropriate repair and replacement of defective components; and (4) verify that the crane operators have been properly trained and are familiar with specific procedures used in handling these loads, e.g., hand signals, conduct of operations, and content of procedures."

a. Summary of Licensee Statements and Conclusions

The Licensee made no statements and expressed no conclusions regarding this interim protection measure.

RESPONSE:

Items (2), (3) and (4) are adequately addressed in previous responses in Boston Edison's submittal of June 25, 1981 as supplemented by responses in this submittal. The review of PNPS procedures requested in Item (1) has been performed and it has been confirmed that sufficient detail and clarity are provided for the subjects of interest.

TABLE I

HEAVY LOADS - REACTOR BUILDING CRANE

Heavy Loads	Approx. Weight (Tons)	Procedures
Waste Debris Shipping Casks	25	6.9-167
Vessel Head Insulation	10	3.M.4-48
Cattle Chute	5	3.M.4-48
Head Strongback	2	3.M.4-48
Reactor Shield Plug (9)	(3) 68 (6) 72	3.M.4-48
Drywell Head	43	3.M.4-48
Reactor Vessel Head	81	3.M.4-48
Steam Dryer Assembly	27	3.M.4-48
Moisture Separator Assemly	42.5	3.M.4-48
Spent Fuel Pool Gates (2)	0.6 (Max)	3.M.4-48
Dryer/Separator Storage Pit Shield Plugs	43	3.M.4-48
Refueling Slot Plugs (4)	6.6 (Max)	3.M.4-48
Spent Fuel Shipping Casks	26	6.9-164
Vessel Service Platform	4	3.M.4-48
Miscellaneous Tool Boxes	3	*
New Fuel Crates	1	*

* General heavy loads handling procedure to be implemented

TABLE 2

HEAVY LOADS - TURBINE BUILDING CRANE

Loads Lifted in Region 16 - Load Drops in Region 16 Will Not Result in Loss of Safe Shutdown Capability - Lifts of These Loads Will Be Excluded From Region 17

Heavy Loads	Approx. Weight (Tons)	Procedures
H.P. Turbine Outer Shell	72.7	*
H.P. Turbine Rotor	61.7	*
L.P. Turbine Hoods (2)	57.8	*
L.P. Turbine Inner, Upper Casing (2)	60.3	*
L.P. Turbine Rotor (2)	147.2	*
Generator Outer Shield (Upper)	6.6	*
Generator Rotor	115	*
L.P. Rotor Lifting Beam	6T	*
Condensate Pumps		*
Motor	11.6	
Pump	9.8	
Alternator		
Rotor	7.5	
Startor	1.5	
Miscellaneous Turbine Parts, Small HX's, Steam Valve Components, Tool Boxes, etc.	Varies	*

Loads Lifted in Region 17 - Load Drops in Region 17 Potentially Have Safety Consequences - see Enclosure 2 for Evaluation

Heavy Loads	Approx. Weight (Tons)	Procedures
Feed Water Heaters/Drain Coolers		*
E103A&B	34.25	
E104A&B	26.6	
E105A&B	22.2	
E106A&B	20.5	

TABLE 2

HEAVY LOADS - TURBINE BUILDING CRANE

(Continued)

Heavy Loads	Approx. Weight (Tons)	Procedures
Reactor Feed Pump & Motor		*
Rotor	8 (Max)	
Startor	8 (Max)	
Pump	4 (Max)	
L.P. Turbine Diaphragms (44 pieces)	2.5-10	*

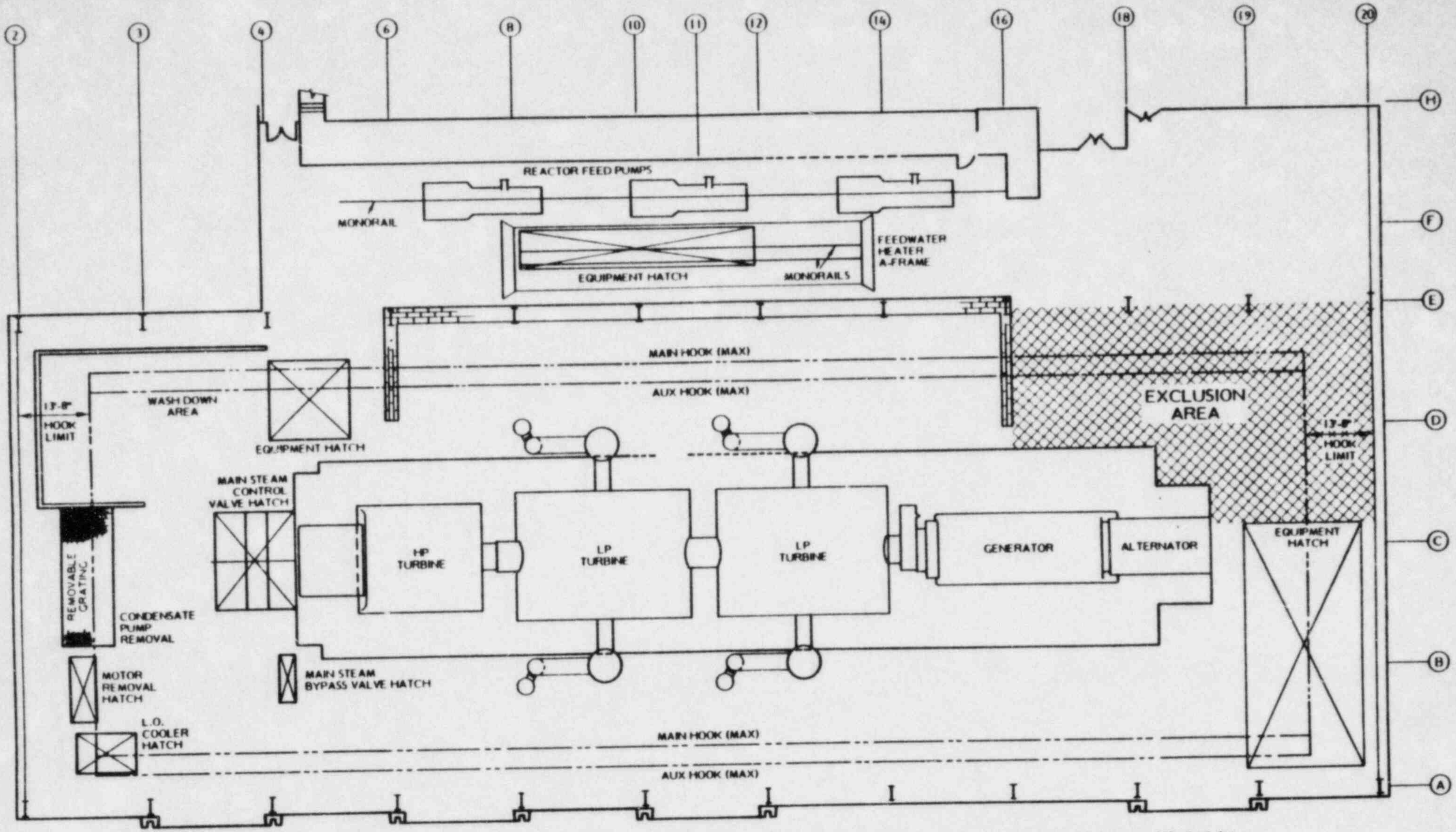
* General heavy loads handling procedure to be implemented

TABLE 3

HEAVY LOADS - HOIST/MONORAIL SYSTEMS

Monorail/Hoist	Heavy Loads	Approx. Wgt (Tons)	Procedures
RHR Pump and Motor Hoists (2)	RHR Motor	2.35	*
Reactor Recirc Pump and Motor Hoist	Recirculation Pump Motor	14	*
Fuel Pool and Reactor Water Clean Up Filter Hoists (2)	Hatch Cover	3/4	*
Reactor Auxiliary Bay Equipment Hatch Hoist	Hatch Cover	3/4	*
Recirculation Pump MG Set Monorail	Fluid Drive Pumps and Motors	1/2 - 1	*
	Fluid Drive Cooler	1.8	*
	Motor Generator Armature	4.5	

* Load handling procedures will be developed by Boston Edison prior to any future heavy load lifts with these handling systems.



Note 1: The following heavy loads may be lifted by the Turbine Building Crane within the exclusion area. However, lift height and procedural restrictions must be followed when such lifts are made.

(1) Turbine Diaphragms
(2) Feedwater Heaters: E103A&B, E105A&B and E104A&B
(3) Feedwater Drain Coolers: E106A&B
(4) Reactor Feed Pump Motor and Pump Components

FIGURE 1

EXCLUSION AREA FOR ALL HEAVY LIFTS BY THE TURBINE BUILDING CRANE EXCEPT THOSE LISTED IN NOTE 1
TURBINE BUILDING PLAN EL. 51'-0"

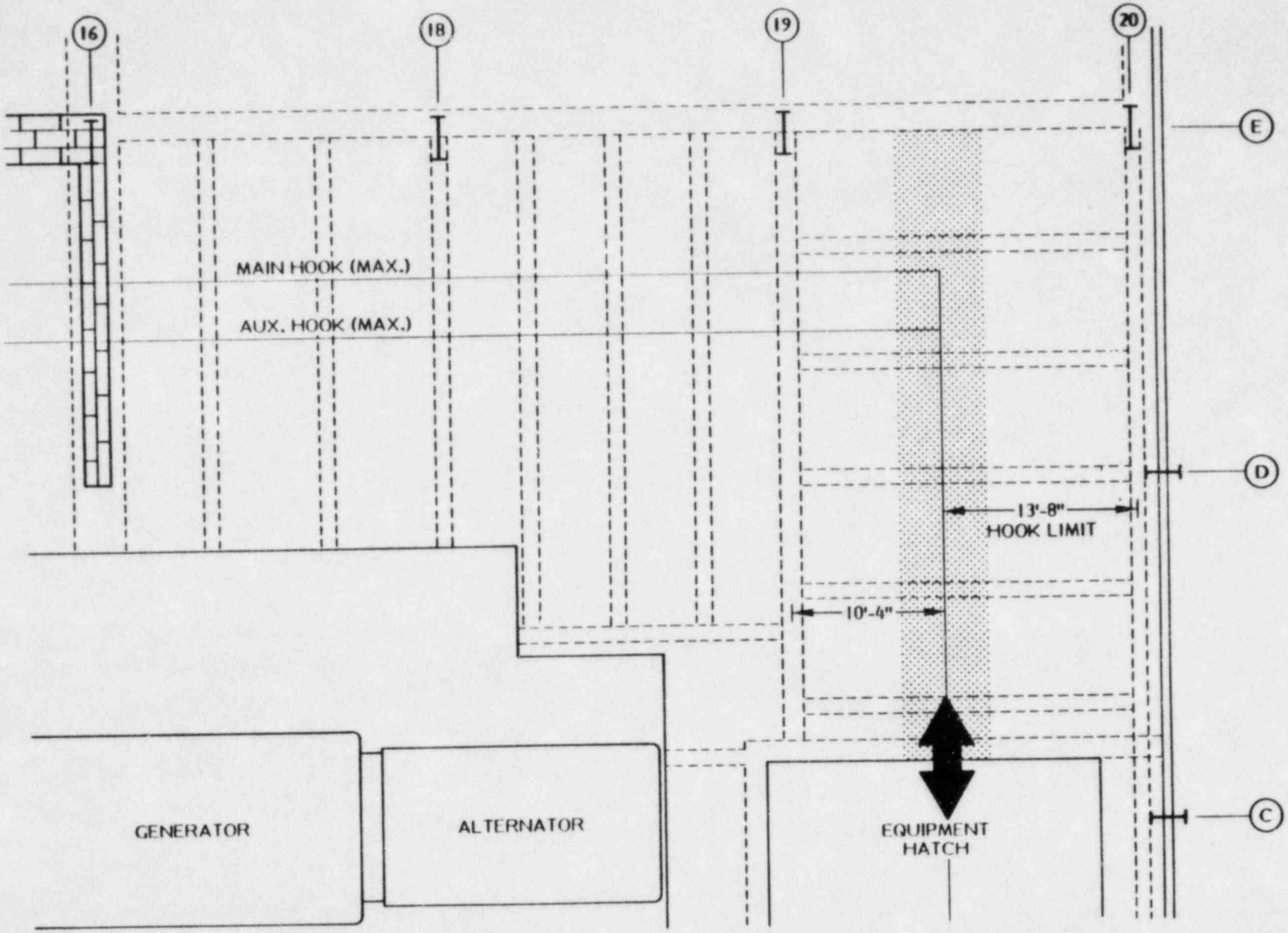
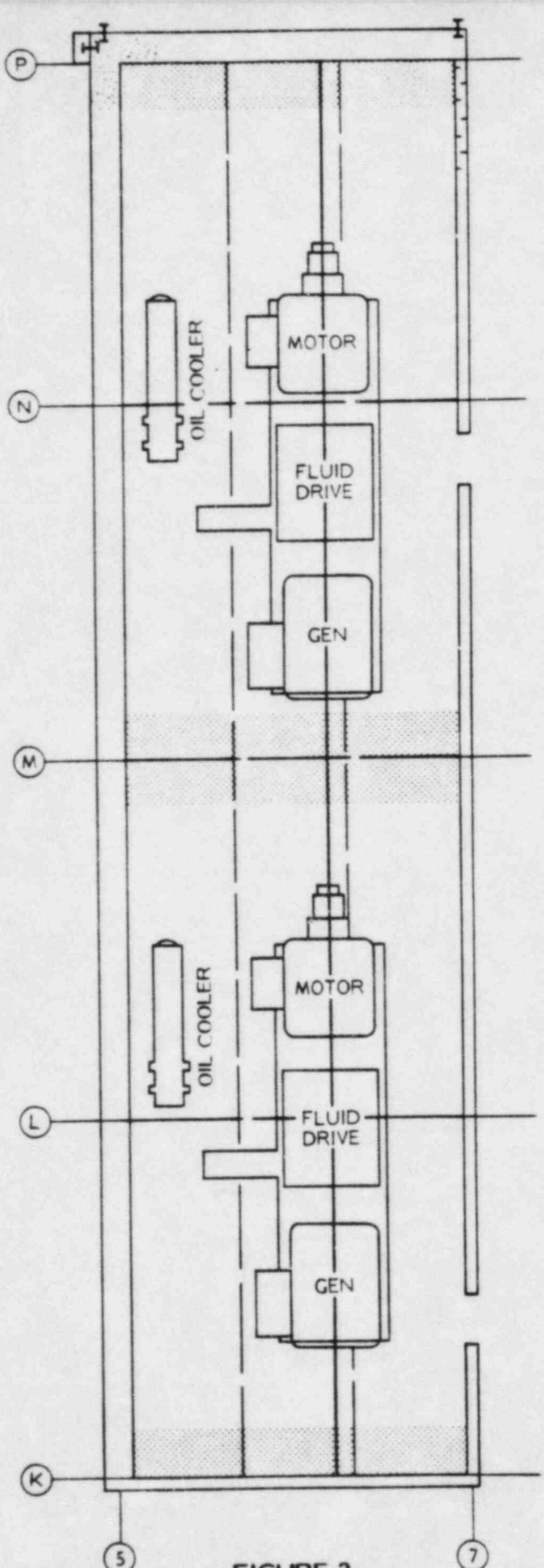


FIGURE 2

INTERIM SAFE LOAD PATH
FEEDWATER HEATER AND REACTOR FEED PUMP COMPONENTS
TURBINE BUILDING 51' EL.



NOTE:

If heavy lifts are to be made during power operation, the centerline of the hoist hook is not to be moved over the shaded areas until the load has been lowered to a height 1 foot or less above the 51'el deck.

FIGURE 3

EXCLUSION AREAS
RECIRC. PUMP
MG SET MONORAIL
EL. 51'-0"

ENCLOSURE 2

RESPONSES TO REQUESTS FOR INFORMATION IN SECTIONS 2.2 AND 2.3 OF NRC LETTER DATED 12/22/80

OVERVIEW OF NUREG-0612 EVALUATIONS

Boston Edison has identified the following fixed handling systems at Pilgrim Nuclear Power Station (PNPS) to which NUREG-0612 is applicable. They are:

	Handing System	Location	Capacity
(1)	Reactor Building Bridge Crane	RB - 117' el.	100 ton (M) 5 ton (A)
(2)	Turbine Building Bridge Crane	Turbine Building 51' el.	165 ton (M) 25 ton (A)
(3)	RHR Pump and Motor Hoists/Monorails (2)	SE & NW Quadrants RB - 23' el.	5 ton/ea
(4)	Recirculation Pump and Motor Hoist/ Monorail	Drywell	20 ton
(5)	Fuel Pool & Reactor Water Cleanup Filter Equipment Hatch Hoists/Monorails (2)	SE Quadrant RB - 91' el.	5 ton
(6)	Reactor Auxiliary Bay Equipment Hatch Hoist/ Monorail	Reactor Auxiliary Bay - 23' el.	5 ton
(7)	Recirculation Pump MG Set Hoist/Monorail	RB - 51' el.	8 ton

There are two basic approaches available to demonstrate compliance with the NUREG-0612 guidelines. They are: (1) demonstrate adequate load handling reliability, or (2) demonstrate that load drop consequences are within the limits of Criteria I-IV listed in Section 5.1 of the NUREG. In all cases for PNPS, the approach has been a demonstration of acceptable consequences.

A combination of systems, structural, criticality and dose evaluations has been utilized to address the NUREG-0612 guidelines for PNPS. To assist in performing these evaluations the relevant plant areas were subdivided into potential load impact regions. The Reactor Building was subdivided into fourteen (15) load impact regions; eight of the regions on the 117' el. refueling floor and the others corresponding to the areas over which the various monorails could travel. The Turbine Building was subdivided into two (2) load impact regions. The subdivisions were based, in part, on the configuration of the buildings and a knowledge of specific locations where heavy loads are typically handled. The load impact regions are described in Table I of this enclosure.

Major component laydown areas for the refueling floor are illustrated in Figure I. The eight load impact regions for the Reactor Building refueling floor are illustrated in Figures 2 through 9. The two load impact regions for the Turbine Building are illustrated in Figures 10 and 11.

Tables 1, 2 and 3 in Enclosure 1 to this submittal identify the loads and applicable load handling procedures for the identified handling systems.

Table 2 of this enclosure relates the defined load impact regions with the four criteria listed in Section 5.1 of NUREG-0612. As indicated in the table, the majority of the regions were defined to assist with and focus safe shutdown evaluations to address NUREG-0612 Criterion IV.

Table 3 of this enclosure indicates the types and combinations of evaluations employed by Boston Edison for each region.

Evaluation Methodology

Systems Evaluations

As noted above, systems evaluations were utilized for many of the regions to evaluate potential load drop consequences. The objective of these systems evaluations was to determine whether defined safety functions could be accomplished assuming that certain equipment was made inoperable as a result of a postulated load drop.

The steps used to perform systems evaluations of the potential effects of load drops inside the reactor building are outlined below:

1) define the safety functions which must be accomplished;

2) identify the systems and support systems relied on to accomplish each safety function;

3) for each load impact region, identify the applicable safety functions and systems and determine which components of those systems could be affected by a heavy load drop within the region;

4) perform an analysis of the effects of failure of those components on the ability to accomplish the applicable safety functions.

The safety functions and systems defined for the purpose of performing the PNPS systems evaluations are illustrated in Figures 10 and 11. The results and conclusions of the systems evaluations are described in the responses to Request for Information Item 2.3.

Structural Analyses

As indicated in Table 5, a number of load impact regions were addressed utilizing structural analyses/evaluations. The load drop scenarios addressed or supported with structural analyses included:

1) drops onto and into the reactor vessel;

2) drops onto the spent fuel pool floor concrete slab, and;

3) drops onto concrete slabs over safe shutdown equipment.

The results and conclusions of the structural analyses are described as appropriate in the responses to Request for Information Items 2.2 and 2.3. The steps in the structural evaluation approach and general methodology are discussed below:

The general steps used to perform the structural evaluations are outlined below:

1. Identification of heavy load, handling systems, and handling locations including a full characterization of the load weight, dimensions, material properties, and structural characteristics.

2. Development of postulated drop scenarios based upon realistic consideration of plant procedures.

3. Review of important structural engineering aspects of impacted structural elements to fully characterize behavior. For reinforced concrete and steel elements, identify drops which control "local" response (e.g., penetration, scabbing, spalling, perforation, etc.); loads that control "overall" structural response (e.g., large inelastic deformations or abrupt failure of principal structural members, etc.); and/or loads that may induce behavior that exhibits combined response such that either overall or local failure modes would control.

4. Incorporating 1 through 3 above, provide early input to the systems evaluations to factor structural information into systems evaluations assumptions.

5. Conduct detailed structural evaluations that include:

    a. Specification of impact energy considering, as appropriate, the energy dissipated due to the transfer of momentum, fluid drag, bouyancy, etc.;

    b. Model develoment for assessing dynamic response utilizing empirical data as necessary;

    c. Development of failure criteria based upon stability or leak tightness considerations;

    d. Computation of the strain energy absorbed prior to reaching the prescribed performance limits;

    e. Assessment of structural response and structural consequences of drop.

The structural evaluation methodology and criteria generally follow the recommendations made by the American Society of Civil Engineers Technical Committee on Impulse and Impact Loads (Reference 1). These recommendations are supplemented by a large body of experimental and analytical information which is documented in reports which have been published by government, university, and industry organizations.

The evaluation methodology and criteria which are addressed below consider the two potential modes of structural behavior, local effects and overall structural response, respectively.

Local Impact Response Evaluation

Local impact response may lead to severe damage such as crushing, perforation, and concrete ejection in the vicinity of the impactive load; however, overall dynamic response of the structure in the form of reactions away from the load are insignificant. The complex nature of local impact response of reinforced concrete requires evaluation using empirical formulae that are experimentally derived. The modified National Defense Research Committee (NDRC) formula (Reference 2) was chosen because it has been shown to give the best fit with available experimental data (References 3 and 4). The NDRC formulae for the depth of penetration, x (inches), of a solid cylindrical missile are given by:

$$x = \left[4\,KNWd\left(\frac{V}{1000d}\right)^{1.8}\right]^{1/2} \qquad \text{for } \frac{x}{d} \leq 2.0 \qquad (1)$$

or

$$x = KNW\left[\frac{V}{1000d}\right]^{1.8} + d \qquad \text{for } \frac{x}{d} \geq 2.0 \qquad (2)$$

where W = weight of the missile (pounds)
d = diameter of missile (inches)
V = impact velocity of missile (feet/second)
N = missile shape factor

= 0.72 flat-nosed missiles

= 0.84 blunt-nosed missiles

= 1.00 spherical-nosed missiles

= 1.14 sharp-nosed missiles

K = concrete penetrability factor

= $180/\sqrt{f'c}$ ($f'c$ = concrete compressive strength in pounds/square inch)

The thickness of reinforced concrete needed to resist impact without perforation and scabbing are given by the following Army Corps of Engineers formulae which can be used in conjunction with equations 1 and 2 (Reference 5).

$$\frac{t_s}{d} = 2.12 + 1.36 \left(\frac{x}{d}\right) \quad \text{for } 0.65 \leq \frac{x}{d} \leq 11.75 \tag{3}$$

$$\frac{t_p}{d} = 1.32 + 1.24 \left(\frac{x}{d}\right) \quad \text{for } 1.35 \leq \frac{x}{d} \leq 13.5 \tag{4}$$

where t_s = concrete thickness required to prevent scabbing

t_p = concrete thickness required to prevent perforation

Equations 3 and 4 were later extrapolated for small values of x/d (Reference 6) giving,

$$\frac{t_s}{d} = 7.91 \left(\frac{x}{d}\right) - 5.06 \left(\frac{x}{d}\right)^2 \quad \text{for } \frac{x}{d} \leq 0.65 \tag{5}$$

$$\frac{t_p}{d} = 3.19 \left(\frac{x}{d}\right) - 0.718 \left(\frac{x}{d}\right)^2 \quad \text{for } \frac{x}{d} \leq 1.35 \tag{6}$$

A 10 percent margin on thickness has been applied in the use of equations 3 through 6 as recommended in Reference 1, except for concrete sections backed by steel decking where the equations were used directly.

The effects of shape and deformability have been conservatively accounted for in the case of the NDRC formula by adjusting the missile shape factor, N, and/or using "equivalent" diameters.

Overall Structural Response Evaluation

Overall structural response results from the dynamic interaction of the impactive load and the structure which it impacts.

The resultant complex forcing function produces in-structure dynamic reactions in the forms of forces, moments, and shears at points away from the impactive load. As a rule, this forcing function is unknown; however, occasionally it can be estimated by incorporating knowledge of the characteristics of the dropped load (weight, size, shape, deformability), characteristics of the impacted structure (material properties, structural configuration), and the impact conditions (velocity, orientation).

The following discussion addresses the use of energy balance methods for the evaluation of reinforced concrete and structural steel structures. These techniques do not require explicit knowledge of the forcing function.

The load drop methodology incorporates the conservation of energy and momentum to calculate the transmitted kinetic energy and maximum displacement to investigate the important modes of overall reinforced concrete structural behavior. The objective of this methodology is to characterize structural behavior in terms of the available strain energy up to prescribed performance limits. These limits are dictated by either ductile or brittle modes of failure. The ductile mode is characterized by large inelastic deflections without complete collapse, while the brittle mode may result in partial failure or total collapse. The available internal strain energy that can be absorbed by the concrete floor system without reaching those limits of unacceptable behavior is balanced against the externally applied energy resulting from a heavy load drop. It has been assumed that momentum is conserved, and the kinetic energy of the drop drives the mass of the floor and induces strain. As an additional

conservatism, no credit has been taken for potential sources of energy dissipation through local deformation in the forms of concrete crushing and penetration.

The following sections discuss specific details applicable to the evaluation of reinforced concrete and steel structures, respectively.

Reinforced Concrete Structures

Generally, the ultimate load of a concrete slab or beam system is reached prior to exceeding the hinge rotational capacity of particular sections provided that an unstable mechanism has not formed. The hinge rotational capacity was used as a criteria to set the maximum allowable level of deflection for the concrete slab or beam system. The hinge rotational capacity for concrete structures was developed in References 7 and 8 based on test results given in References 9 and 10 and is given as:

$$r_u = 0.0065\ (d/c) \leq 0.07 \qquad (7)$$

where

r_u = rotational capacity of plastic hinge (radians)

d = distance from the compression face to the tensile reinforcement

c = distance from the compression face to the neutral axis at ultimate strength

The maximum deflection for a concrete slab or beam with a plastic hinge at its center is then given by:

$$Xm = (r_u L/4) \qquad (8)$$

where,

Xm = maximum deflection

L = span of beam

Rotations of the magnitude governed by equation 7 result in cracking which is confined to a region below (above) the tensile reinforcement. Generally speaking, the section will remain intact with no crushing, spalling, or scabbing due to flexure; however, scabbing may occur as a result of shock wave motion associated with the reflection of tensile waves from the rear surface or shear plug formation. It has been conservatively assumed that scabbing does occur.

The load/deflection history up to the point of the ultimate loading, coupled with the maximum allowable deflection, defines the maximum level of strain energy absorption provided that a shear failure has not occurred. The shear stress at limiting sections was checked and compared to allowables as specified in Chapter II of ACI 381-77 (Reference 11).

<u>Structural Steel Structures</u>

The maximum response of structural steel elements is determined using the commonly applied energy balance method (References 1, 12, and 13) by equating the externally applied kinetic energy to the available internal strain energy. The maximum permissible deflection of each structural element is given in terms of an allowable ductility ratio which is defined as:

$$\mu = \frac{Um}{Uy} \qquad (9)$$

where Um = maximum permissible deflection
Uy = deflection at the effective limit

The allowable structural steel ductility ratios for impact loads have been taken from Reference 1 and are as follows:

MODE OF RESPONSE	ALLOWABLE DUCTILITY RATIO
1. Flexure	
- open sections	12.5
- closed sections	20
2. Shear	5
3. Compression	$\frac{14 \times 10^4}{F_y(\frac{KL}{r})^2} \leq 10$
4. Torsion	$0.5 \cdot \frac{\epsilon u}{\epsilon y}$

where Fy = minimum yield stress of the steel

K = theoretical effective length factor for compression member

L = length of compression member

ϵu = ultimate strain

ϵy = yield strain

The effective yield limit corresponds to the inflection point of an equivalent elasto-plastic resistance displacement curve as defined by Newmark (Reference 14). For simplicity, an equivalent elasto-plastic resistance displacement curve was developed by setting the maximum resistance equal to the actual minimum yield resistance. This procedure is conservative because it neglects the strain energy associated with the strain hardening mechanism.

Discussion of Structural Margins

In addition to the conservatisms previously mentioned, the following conservatisms are also inherent in the methodology used in the evaluation:

1. Static material strengths for concrete and steel are used. Test data shows that this property increases with the increased strain rates associated with dynamic loadings. For example, References 13 and 15 recommend dynamic increase factors of 1.25 for the compressive strength of concrete and 1.20 for the flexural, tensile, and compressive strength of structural steel.

2. Design (minimum) material properties for steel are used. The average strength for structural steel is nearly a factor of 1.25 (Reference 11) higher than the minimum yield requirement specified by ASTM. While these factors above minimum code strength exist and contribute to structural margins, they are not used in the evaluation.

3. Equation 7 for hinge rotational capacity is used. This corresponds to rotations of the order of 2 degrees with minimum cracking and no crushing or scabbing. To meet necessary performance requirements (i.e., halting propagating failures), larger rotations in the range of 5 to 12 degrees could be tolerated. Such rotations would lead to crushing, spalling, and scabbing of the section (Reference 15); however, overall load carrying capability is expected to remain intact. Experimental observations (Reference 18) suggest even further capability for well-designed and well-anchored slabs. Failure modes at such levels initially appear to be controlled by yielding in shear and flexure followed by membrane stretching until failure occurs, normally at the support edge of the slab. Use of these larger rotational capabilities would have resulted in greater energy absorbing capabilities of the floor system.

4. The analysis uses ACI 318-77 allowable shear stresses. A significant body of data suggests the existence of higher shear capabilities. (References 18 through 26).

5. The structural loads are distributed directly under the dropped heavy load. In reality, a more favorable load distribution would exist due to the load distribution capability of the slab.

6. No credit is taken for local energy dissipation associated with any crushing of the load itself or the immediate surface of the floor.

RESPONSES TO REQUESTS FOR INFORMATION IN SECTIONS 2.2 AND 2.3 OF ENCLOSURE 3 TO NRC DECEMBER 22, 1980 LETTER

2.2 SPECIFIC REQUIREMENTS FOR OVERHEAD HANDLING SYSTEMS OPERATING IN REACTOR BUILDING

NUREG-0612, Section 5.1.4, provides guidelines concerning the design and operation of load-handling systems in the vicinity of spent fuel in the reactor vessel or in storage. Information provided in response to this section should demonstrate that adequate measures have been taken to ensure that, in this area, either the likelihood of a load drop which might damage spent fuel is extremely small, or that the estimated consequences of such a drop will not exceed the limits set by the evaluation criteria of NUREG-0612, Section 5.1, Criteria I through III.

ITEM 2.2-1 Identify by name, type, capacity, and equipment designator, any cranes physically capable (i.e., ignoring interlocks, moveable mechanical stops, or operating procedures) of carrying loads over spent fuel in the storage pool or in the reactor vessel.

RESPONSE: The only heavy load handling system operating in the Reactor Building that is capable of carrying loads over spent fuel in the storage pool or in the reactor vessel is the Reactor Building Crane. The Reactor Building Crane is an overhead bridge crane with a main hoist of 100 tons and an auxiliary hoist of 5 tons.

ITEM 2.2-2 Justify the exclusion of any cranes in this area from the above verifying that they are incapable of carrying heavy loads or are permanently prevented from movement of heavy loads over stored fuel or into any location where, following any failure, such load may drop into the reactor vessel or spent fuel storage pool.

RESPONSE: Justification for exclusion of handling systems operating in the vicinity of the spent fuel pool is provided in the response to draft Franklin TER Section 2.1.1.c in Enclosure 1 to this submittal.

ITEM 2.2-3 Identify any cranes listed in 2.2.-1, above, which you have evaluated as having sufficient design features to make the likelihood of a load drop extremely small for all loads to be carried and the basis for this evaluation (i.e., complete compliance with NUREG-0612, Section 5.1.6, or partial compliance supplemented by suitable alternative or additional design features). For each crane so evaluated, provide the load-handling-system (i.e., crane-load-combination) information specified in Attachment 1.

RESPONSE: The Reactor Building crane was evaluated to industry standards CMAA 70-1975 and ANSI B30.2-1976. It was found to meet these standards as indicated in the response to draft Franklin TER Item 2.1.8.c in Enclosure 1 to this submittal.

The only case where load handling reliability was considered was with respect to the main hoist load block and hook. NUREG-0612 requires that the load block and hook be considered as a heavy load. The load block is used for handling numerous loads, including the reactor vessel head, drywell head, shield plugs, and the dryer and separator units. In moving these loads, the hook, load block, rope, drum, sheave assembly, motor shafts, gears, and other load bearing members are subjected to significant stresses approaching the load rating of the crane. By comparison, these components are subjected to a considerably smaller load when only the hook and load block are being moved. Based on this, it is not considered feasible to postulate a random mechanical failure of the crane load bearing components when moving the crane load block alone.

The only feasible failure modes for dropping of the main hook and load block would be:

1) A control system or operator error resulting in hoisting of the block to a "two blocking" position with continued hoisting by the motor and subsequent parting of the rope (this situation can be prevented by operator action prior to "two blocking" or by an upper limit switch to terminate hoisting prior to "two blocking"); and

2) Uncontrolled lowering of the load block due to failure of the holding brake to function (the likelihood of this can be made small by use of redundant holding brakes).

The PNPS Reactor Building crane is currently provided with one upper limit switches to interrupt power to the hoist motor prior to "two blocking." A second upper limit switch will be added. When power is removed, holding brakes are automatically applied.

The holding brakes are solenoid released, and spring applied on loss of power to the solenoid. Two holding brakes are provided for each of the main and auxiliary hoists. Each brake has sufficient capacity to hold the rated load (each brake is greater than 150% of full motor torque). Additionally, inspection and maintenance procedures assure that the limit switches and holding brakes are functional and properly adjusted.

With the provisions described above, the two limit switches will reduce the likelihood for "two blocking" and the two holding brakes will reduce the likelihood of uncontrolled lowering of the load block. Based on these features, it is concluded that a drop of the load block and hook is of sufficiently low likelihood that it does not require load drop analyses.

ITEM 2.2-4 For cranes identified in 2.2.-1, above, not categorized according to 2.2-3, demonstrate that the criteria of NUREG-0612, Section 5.1, are satisfied. Compliance with Criterion IV will be demonstrated in response to Section 2.3 of this request. With respect to Criteria I through III, provide a discussion of your evaluation of crane operation in the Reactor Building and your determination of compliance.

ITEM 2.2-4a Where reliance is placed on the installation and use of electrical interlocks or mechanical stops, indicate the circumstances under which these protective devices can be removed or bypassed and the administrative procedures invoked to ensure proper authorization of such action. Discuss any related or proposed technical specifications concerning the bypass of such interlocks.

RESPONSE: There are no interlock systems or mechanical stops provided for the Reactor Building Crane that prohibit movement of heavy loads over the spent fuel pool or reactor vessel during normal load handling operations. Safe load paths implemented in plant procedures are relied on to prevent inadvertent movement of heavy loads over the spent fuel pool. Loads that must be moved over the spent fuel pool or reactor vessel were appropriately evaluated as described in the following responses.

ITEM 2.2-4b Where reliance is placed on the operation of the Stand-by Gas Treatment System, discuss present and/or proposed technical specifications and administrative or phyiscal controls provided to ensure that these assumptions remain valid.

RESPONSE: Reliance is placed on secondary containment integrity and operation of the Stand-by Gas Treatment System for several postulated drops of heavy loads into the spent fuel pool. These postulated drops involve (I) drop of a spent fuel pool gate from above the pool and (2) drop and topple over of a refuel slot plug as it is being moved along the eastern edge of the pool. The drops of interest would be following a refueling operation when freshly spent fuel is in the pool. PNPS procedures will require that secondary containment be established and that one train of the Stand-by Gas Treatment System be operable when these load movements are made.

ITEM 2.2-4c Where reliance is placed on other site-specific considerations (e.g., refueling sequencing), provide present or proposed technical specifications, and discuss administrative or physical controls provided to ensure the validity of such considerations.

RESPONSE: As indicated in the following response to Item 2.2-4d, reliance is placed on the sequence and timing of certain load handling operations near the spent fuel pool.

In addition, as discussed in response to Item 2.3 in this enclosure, certain load lifts are scheduled to occur only during cold shutdown conditions which limits the safety functions that must be accomplished following postulated load drops.

ITEM 2.2-4d Analyses performed to demonstrate compliance with Criteria I through III should conform to the guidelines of NUREG-0612, Appendix A. Justify any exception taken to these guidelines, and provide the specific information requested in Attachment 2, 3, or 4, as appropriate, for each analysis performed.

RESPONSE:

There are three potential consequences of interest when considering load drops onto the open reactor vessel. They are: 1) loss of reactor vessel integrity, 2) fuel cladding damage and the resultant radiological dose, and 3) fuel crushing and the possibility of a resulting criticality condition. Criteria I through III in Section 5.1 of NUREG-0612 address each of these potential consequences. As indicated in Table 2, Criteria I through III were considerations for Regions 1 (reactor vessel) and 3 (spent fuel pool area). The evaluations below have been performed to address these issues.

Reactor Vessel

The postulated load drops of interest onto or into the reactor vessel are (1) the reactor vessel head, (2) moisture separator assembly and (3) steam dryer assembly. A detailed finite element elastic-plastic analysis was performed to evaluate the effects of these drops onto the PNPS reactor vessel. These analyses and their results are described in General Electric Report No. NSEO-84-0982 entitled "Structural Analysis of Pilgrim/Boston Edison Company Vessel Head Drop, Shroud Head Assembly Drop, and Steam Dryer Assembly Drop Conditions -October 1982." This report was previously transmitted to NRC for review by letter dated February 28, 1983 from W. D. Harrington to Darrel G. Eisenhut (ltr #83-59).

The analysis results indicate no damage to fuel in the core or loss of vessel integrity. Accordingly, Criteria I through III are met for drops onto the reactor vessel.

Spent Fuel Pool - Criteria I and II - Dose and Criticality

Based on the safe load paths defined by Boston Edison, the heavy loads carried over or near the spent fuel pool that were considered for potential impact of spent fuel in the pool are (1) a shipping cask (cask topple over after impacting the north edge of the spent fuel pool), (2) a spent fuel pool gate, (3) refueling slot plugs and (4) the "cattle chute" (refueling shield).

With regard to the potential for criticality as a result of fuel/rack crushing in the spent fuel pool, the generic evaluation in NUREG-0612 was used as the basis for concluding that a criticality event is not possible. The PNPS spent fuel pool rack design is of the high density type. It relies on geometry and boral plates in the racks to assure that fuel in the pool remains subcritical with substantial margins, during normal and abnormal conditions.

The NRC states in Section 2.2.4.2 of NUREG-0612 that, in the case of BWR spent fuel racks with boral poison cans:

> "crushing the fuel would not significantly increase the k_{eff} of the fuel. For racks with boral poisons, it seems inconceivable that any load which might fall on the spent fuel pool would separate the fuel from the poison cans and subsequently push the assemblies together to form a critical mass. Therefore it appears that postulated load drop events would not cause a criticality in a BWR spent fuel pool that uses boron plate type racks."

On this basis, it is concluded that NUREG-0612 Criterion II is met for postulated heavy load drops into the spent fuel pool.

With regard to potential offsite doses, Figure 2.1-2 of NUREG-0612 was utilizied to evaluate dose consequences. The maximum number of fuel assemblies that could be damaged from a fuel pool gate drop was determined to be 9. The postulated load drop scenario of interest is an end-on drop of the gate from approximately 1 foot above the 117' el. refueling floor. This corresponds to a drop of the gate when it is being returned to the pool following maintenance during a refueling outage. A flat or lengthwise side-on drop would not result in sufficient energy being transferred to individual rods to cause loss of cladding integrity.

The maximum number of fuel assemblies that could be damaged from a refuel slot plug drop was determined to be 50. Figure 2.1-2 of NUREG-0612 indicates that for 50 assemblies damaged with credit for charcoal filters, on the order of 11 day decay times would be necessary to demonstrate acceptable consequences. The postulated load drop scenario of interest is a drop of slot plug to the 117' el. floor adjacent to the east wall of the spent fuel pool and subsequent topple over into the pool. Both for the gate and the slot plugs, the drops of interest would be drops following completion of refueling operations. The reason for this is that when the gates and plugs are removed prior to refueling operation (potentially less than 24 hours after shutdown), there is no newly offloaded spent fuel in the spent fuel pool. A conservative time of 14 days was used for spent fuel decay time, corresponding to the minimum time to complete refueling operations and reach the point in the outage when the gates and plugs will be returned to their normal location. This 14 day decay time clearly exceeds the decay time necessary to demonstrate acceptable consequences for the gate drop and exceeds the 11 day decay time for the plug drop. Accordingly, the NUREG-0612 dose criteria are met for these two load drop scenarios.

A cask topple into the spent fuel pool from the north end could result in impact of spent fuel in the storage racks. Such a drop could damage a substantial number of fuel assemblies (potentially on the order of 200 assemblies). The potential offsite doses associated with damage of this extent could potentially exceed the allowable offsite doses stated in Criterion I of NUREG-0612 depending on the decay time of the fuel impacted. For this reason, adequate decay time will be allowed for fuel assemblies in the region of the spent fuel pool near the cask laydown area prior to cask movement near the spent fuel pool. The following "prerequisite" will be incorporated into load handling procedures for casks.

"Prior to movement of the cask over or into the spent fuel pool, all spent fuel within a distance L (where L is the longest dimension of the cask) from the north edge of the pool shall have decayed a minimum of 60 days."

With this restriction, Criterion I of NUREG-0612, Section 5.1 will be met for a postulated cask drop.

With regard to Criterion I for potential drops of the "cattle chute" into the spent fuel pool, specific load handling restrictions will be placed in the appropriate sections of plant procedures. These restrictions will sufficiently reduce the likelihood of the "cattle chute" actually dropping into the spent fuel pool to eliminate having to consider this drop scenario. The restriction will be to limit the carry height of the cattle chute to 6" above the 117' el. floor or obstructions such as curbs, electrical junction boxes, etc., when moving the load along the floor adjacent to the east wall of the spent fuel pool, and to specify that in no case shall the "cattle chute" be carried over any portion of the pool. These restrictions will assure that, if the load is dropped while traversing along the east side of the pool, it will land on the refueling floor and remain upright, i.e., will not topple into the spent fuel pool. This is further assured by the fact that, (1) the "cattle chute" center of gravity is low because most of the mass is concentrated in the lead shielding at the bottom of the load (topple over, if dropped from low heights would not be expected), and (2) if dropped near the edge of the pool, impact of the pool curb would tend to rotate the "cattle chute" away from the pool. On this basis, Criterion I is not a consideration for a postulated "cattle chute" drop near the spent fuel pool.

<u>Spent Fuel Pool - Criterion III (Pool Integrity)</u>

As indicated above, the loads postulated to accidentally fall into the spent fuel pool are (1) a cask, (2) a refuel slot plug, and (3) a spent fuel pool gate. With regard to the cask, the potential for loss of fuel pool integrity has previously been evaluated. The results are presented in Section 10.3.6 of the PNPS FSAR. It indicates there that analysis has demonstrated that "damage to the floor will not result in a leakage rate greater than the pool makeup capability." Accordingly, Criterion III is met for a cask drop.

With regard to a postulated drop of a refuel slot plug, it is inconceivable that such a drop would result in impact of the pool floor; the plug instead would impact the spent fuel storage racks. A drop of a spent fuel pool gate to the pool floor has been analyzed. The results indicate that damage to the floor will not result in leakage greater than the pool makeup capability. Therefore, Criterion III is met for each of these two postulated drops.

2.3 SPECIFIC REQUIREMENTS FOR OVERHEAD HANDLING SYSTEMS OPERATING IN PLANT AREAS CONTAINING EQUIPMENT REQUIRED FOR REACTOR SHUTDOWN, DECAY HEAT REMOVAL, OR SPENT FUEL POOL COOLING

NUREG-0612, Section 5.1.5, provides guidelines concerning the design and operation of load-handling systems in the vicinity of equipment or components required for safe reactor shutdown and decay heat removal. Information provided in response to this section should be sufficient to demonstrate that adequate measures have been taken to ensure that, in these areas, either the likelihood of a drop which might prevent safe reactor shutdown or prohibit continued decay heat removal is extremely small, or that damage to such equipment from loads will be limited in order not to result in the loss of these safety-related functions. Cranes which must be evaluated in this section have been previously identified in your response to 2.1-1, and their loads in your response to 2.1-3-c.

ITEM 2.3-1 Identify any cranes listed in 2.1-1, above, which you have evaluated as having sufficient design features to make the likelihood of a load drop extremely small for all loads to be carried and the basis for this evaluation (i.e., complete compliance with NUREG 0612, Section 5.1.6, or partial compliance supplemented by suitable alternative or additional design features). For each crane so evaluated, provide the load-handling-system (i.e., crane-load-combination) information specified in Attachment 1.

RESPONSE:

A discussion regarding the Reactor Building crane is provided in the response to Item 2.2 in this enclosure. The designs of the remaining handling systems addressed in this submittal, including the Turbine Building Bridge Crane, have also been evaluated against appropriate industry standards and have been found to comply with certain justified exceptions. These evaluations are included in the response to draft Franklin TER Item 2.1.1.c and 2.1.8.c in Enclosure 1 to this submittal.

As in the case of the Reactor Building Crane, the Turbine Building Crane hoist hook and load block have been eliminated as potential heavy load drops on the basis of load handling reliability. The evaluation for the Reactor Building Crane in the response to Item 2.2-3 is equally applicable to the Turbine Building Crane and accordingly, is not repeated here. In this regard, a second upper limit switch will be added to the Turbine Building Crane to assure adequate protection against a potential two blocking event.

ITEM 2.3-2 For any cranes identified in 2.1-1 not designated as single-failure-proof in 2.3-1, a comprehensive hazard evaluation should be provided which includes the following information:

a. The presentation in a matrix format of all heavy loads and potential impact areas where damage might occur to safety-related equipment. Heavy loads identification should include designation and weight or cross-reference to information provided in 2.1-3-c. Impact areas should be identified by construction zones and elevations or by some other methods such that the impact area can be located on the plant general arrangement drawings. Figure 1 provides a typical matrix.

b. For each interaction identified, indicate which of the load and impact area combinations can be eliminated because of separation and redundancy of safety-related equipment, mechanical stops and/or electrical interlocks, or other site-specific considerations. Elimination on the basis of the aforementioned consideration should be supplemented by the following specific information:

(1) For load/target combinations eliminated because of separation and redundancy of safety-related equipment, discuss the basis for determining that load drops will not affect continued system operation (i.e., the ability of the system to perform its safety-related function).

(2) Where mechanical stops or electrical interlocks are to be provided, present details showing the areas where crane travel will be prohibited. Additionally, provide a discussion concerning the procedures that are to be used for authorizing the byassing of interlocks or removable stops, for verifying that interlocks are functional prior to crane use, and for verifying that interlocks are restored to operability after operations which require bypassing have been completed.

(3) Where load/target combinations are eliminated on the basis of other, site-specific considerations (e.g., maintenance sequencing), provide present and/or proposed technical specifications and discuss administrative procedures or physical constraints invoked to ensure the validity of such considerations.

c. For interactions not eliminated by the analysis of 2.3-2-b above, identify any handling systems for specific loads which you have evaluated as having sufficient design features to make the likelihood of a load drop extremely small and the basis for this evaluation (i.e., complete compliance with NUREG 0612, Section 5.1.6, or partial compliance supplemented by suitable alternative or additional design features). For each so evaluated, provide the load-handling-system (i.e., crane-load-combination) information specified in Attachment 1.

d. For interactions not eliminated in 2.3-2-b or 2.3-2-c, above, demonstrate using appropriate analysis that damage would not preclude operation of sufficient equipment to allow the system to perform its safety function following a load drop (NUREG 0612, Section 5.1, Criterion IV). For each analysis so conducted, the following information should be provided:

(1) An indication of whether or not, for the specific load being investigated, the overhead crane-handling system is designed and constructed such that the hoisting system will retain its load in the event of seismic accelerations equivalent to those of a safe shutdown earthquake (SSE).

(2) The basis for any exceptions taken to the analytical guidelines of NUREG 0612, Appendix A.

(3) The information requested in Attachment 4.

<u>RESPONSE</u>

The types of evaluations performed for each defined load impact region are illustrated in Table 3. The evaluation approaches and methodologies are described in the introductory sections to this enclosure.

The results of the evaluations indicated that for the large majority of the load drop scenarios safe shutdown and/or decay heat removal could be accomplished

relying on plant safety systems to perform their normal design functions. There were, as well, several potential problems identified in some of the load impact regions. These potential problems are described below along with a description of Boston Edison's plans for resolving each.

<u>Region 4 - North Equipment Hatch</u>

On the basis of preliminary structural evaluation of a drop of a shipping or waste debris cask through the north equipment hatch from the 117' refueling deck elevation to the 23' el. floor, it was concluded that the floor could potentially fail. This failure could result in damage to the torus, below elevation 23', potentially resulting in loss of torus water inventory and subjecting safety related equipment to a radioactive steam environment should ADS operation or primary pressure relief become necessary. To address this situation, Boston Edison will as an interim measure, revise cask load handling procedures to require that the cask be lifted vertically in the northwest quadrant of the hatchway. The torus is located below the 23' el. floor in the southeast quadrant of the hatch opening. In addition, the load orientation will be such that the load impact will to the maximum extent possible pick up the load resistance capability of a major structural wall (torus compartment wall) that runs diagonally from the southwest to the northeast approximately below the center line of the 23' el. floor exposed to the hatch opening.

In the long term, Boston Edison will either (1) undertake more sophisticated structural analyses that could include:

(a) an investigation of membrane action capability and higher hinge rotational capability,

(b) consideration of specific load impact limiters integral to the cask, and

(c) explicit consideration of the energy dissipation capability of the cask transport vehicle, OR

(2) Will design and fabricate energy absorbing pads for temporary installation on the 23' el. floor during heavy load movements in the equipment hatch.

Region 2 - Dryer/Separator Pool

Reactor Building Closed Cooling Water System (RBCCW) piping is located below the dryer/separator pool floor. The piping involved if breached could potentially result in loss of both trains of RBCCW which provides cooling support to the RHR System, Core Spray System, HPCI and RCIC. Structural evaluations indicate that for drops of the dryer or separator from normal carry heights the floor will remain intact. However, for two other postulated load drops, structural analyses could not adequately demonstrate that the pool floor would protect the safety equipment below. These postulated load drops are drops of a dryer/separator pool slot plug from the 117' el. and drop of a cask from the 117' el. to the pool floor.

Dryer/Separator Pool Plug Drop - Upon further evaluation, it was concluded that based on handling procedures, plant configuration and load movement sequencing, the probability of a pool plug drop to the pool floor was sufficiently small to preclude postulating the drop.

The Dryer/Separator Pool plugs are stacked vertically. Slots in the pool walls are engaged by the plugs such that any drop of a plug as it is being raised or lowered to and from the refueling floor would result in falling back down into its normal location, i.e., no impact of the pool floor. Safe load paths are stringently adhered to when moving to and from their normal position. Further, dryer/separator pool plugs are typically removed and installed twice during a refueling outage, i.e., 4 movements total. For two of these movements the separator would be in its storage location at the west end of the dryer/separator pool and would protect the pool floor from direct impact from an accidental plug drop.

Accordingly, the only conceivable way a plug could drop to the pool floor would be for a drop to occur during one of the two lifts that the separator is not present in the pool and precisely during the lift when the plug is not engaged in its slots, but is still over the edge of the pool. The likelihood of a drop occurring at this point is low because of (1) the small time the plugs will be in the undesired position, and (2) the structural integrity of the crane and lifting equipment will be verified upon initial lift off and hold of the plug, i.e., prior to the plug being in the undesired position.

Cask drop - The designated cask washdown area for PNPS is in the dryer/separator pool. As a result of the potential problems associated with a cask drop to the pool floor, the cask washdown area will be moved to the reactor head storage stand. This will eliminate the possibility of a cask drop to the pool floor.

Regions 6, 12 and 13

Regions 6, 12 and 13 are defined by hatch openings at the 91' el. of the Reactor Building (see Table 1). The component of interest is RHR piping in the compartments below these hatch openings. The only heavy loads carried in these regions are the hatch covers themselves which weigh approximately 1500 lbs. The possibility of hatch cover drop through a hatch opening will essentially be eliminated by appropriate procedural controls on hatch movement that will include limitations on lift height and load orientation and a prohibition against movement of a hatch cover over an open hatch. No additional measures are judged to be necessary.

Region 7 - 117' el. Refueling Floor

Systems evaluations have concluded that safe shutdown and decay heat removal can be accomplished if damage in Region 7 can be limited to the elevation immediately below the 117' el. (i.e. the 91' el.). Accordingly, structural analyses were undertaken to determine if the floor could adequately resist large load impacts such that overall floor collapse or perforation would not occur.

These structural analyses revealed that the floor system potentially has substantial load impact resistance. However, premature failure of certain clip angle connections at the ends of the steel beams supporting the floor could result in the floor system not realizing its full potential to resist the load impact. The load drop scenarios of interest are the reactor vessel head and reactor cavity shield plugs.

As a result of these conclusions, Boston Edison plans to design and install modifications to the steel beam connections of interest that will allow the floor to resist load impacts of these large heavy loads from drop heights up to their normal carry height.

Region 17 - Turbine Building 51' el.

The elevations below the 51' el. turbine deck in Region 17 (Figure 11) contain electrical equipment associated with the operation of both trains of safe shutdown system components. This equipment is separated vertically (Train A at the 37' el. and Train B at the 23' el.). Conservative structural analyses were performed of this area to determine if damage from postulated heavy load drops could be limited to the 37' el. immediately below the 51' el. slab. This involves demonstrating no perforation of the slab or overall collapse of the floor system.

Based on these structural analyses, it was determined that the Turbine Building 51' el. floor system had the potential for similar problems with the beam connections as described above for the Reactor Building 117' el. However, because the weights of the loads considered for this area (see Table 2 of Enclosure 1 to this submittal) are considerably less than those in Reactor Building situation, it was determined that the load drop parameters (e.g. carry height and load path) could be restricted with the implementation of procedural controls such that floor system failure would not occur. These procedural controls are described in Enclosure 1 to this submittal in the response to draft Franklin TER Open Item 2.1.1.c. More rigorous structural analyses of this floor system currently being performed may result in refinements to these procedural controls. If so, they will be incorporated into plant procedures.

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TABLE I
LOAD IMPACT REGIONS
DEFINED FOR EVALUATION PURPOSES

Region	Description	Handling System
1	Reactor Vessel	Reactor Bldg. Crane
2	Dryer/Separator Storage Pool including elevation 74'3" below pool floor	Reactor Bldg. Crane
3	Spent Fuel Pool including elevation 51' below pool floor	Reactor Bldg. Crane
4	North equipment hatch at elevation 117' to hatch access area at elevation 23' including elevation (-)17'6" below hatch access area	Reactor Bldg. Crane
5	Mid equipment hatch at elevation 117' to hatch access area at elevation 91'3" including elevation 74'3" below hatch access area	Reactor Bldg. Crane
6	South equipment hatch at elevation 117' to hatch access area at elevation 91'3" including elevation 74'3" below hatch access area and elevation 51' below SFP demineralizer space	Reactor Bldg. Crane
7	North half of refueling deck at elevation 117' including north half of elevation 91'3"	Reactor Bldg. Crane
8	South half of refueling deck at elevation 117' including south half of elevation 91'3"	Reactor Bldg. Crane
9	RHR loop A monorail including area at elevation 23' and RHR A equipment space, elevation (-)17'6"	RHR A Pump and Motor Monorail/Hoist
10	RHR loop B monorail including area at elevation 23' and RHR B equipment space, elevation (-)17'6"	RHR B Pump and Motor Monorail/Hoist
11	RECIRC Pump Motor Monorail including area inside drywell below elevation 37'	Recirc Pump Motor Monorail/Hoist
12	Monorail at Reactor Building elevation 113'-3½", above SFP filter space, including area at elevation 91'3", the SFP filter space at el. 74'3", and elevation 51' below SFP filter space.	Spent Fuel Pool Filter Equipment Hatch Monorail/Hoist

TABLE I
LOAD IMPACT REGIONS
DEFINED FOR EVALUATION PURPOSES

(Continued)

Region	Description	Handling System
13	Monorail at Reactor Building elevation 113'-3½", above clean-up filter demineralizer spaces, including area of elevation 91'3", elevation 74'3", below load path and elevation 51" below clean-up filter deminerlizer spaces.	Reactor Water Cleanup Filter Equipment Hatch Monorail/Hoist
14	Reactor Auxiliary Bay Monorail, including area and hatch at elevation 23', area and hatch at elevation 3' and area at elevation (-)17'6"	Reactor Auxiliary Bay Equipment Hatch Monorail/Hoist
15	Recirc pump MG Set space at elevation 51' including corresponding area at elevation 23'.	Recirculation Pump MG Set Monorail/Hoist
16	Majority of turbine deck at elevation 51' including southeast equipment hatch and corresponding areas at elevations 37', 23', and -1' (also 6').	Turbine Building Bridge Crane
17	Load pickup area on turbine deck at elevation 51', northeast of turbine, including corresponding area at elevation 37'.	Turbine Building Bridge Crane

TABLE 2

LOAD IMPACT REGIONS
VS.
NUREG CRITERIA

LOAD IMPACT REGIONS / NUREG-0612 CRITERIA (SECTION 5.1)	REACTOR BUILDING AND AUXILIARY BAY															TURBINE BUILDING CRANE	
	REACTOR BUILDING CRANE								MONORAIL/HOIST SYSTEMS								
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
I (DOSE LIMITS)	X		X														
II (CRITICALITY)	X		X														
III (RV OR SFP INTEGRITY)	X		X														
IV (SAFE SHUTDOWN)		X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X

TABLE 3

EVALUATION APPROACHES UTILIZIED
VS.
LOAD IMPACT REGIONS

LOAD IMPACT REGIONS / ANALYSIS/ EVALUATION APPROACH	REACTOR BUILDING AND AUXILIARY BAY															TURBINE BUILDING CRANE	
	REACTOR BUILDING CRANE								MONORAIL/HOIST SYSTEMS								
	1	2	3	4	5	6	7	8	9	10	11	12	13	14	15	16	17
DOSE			X														
CRITICALITY			X														
STRUCTURAL	X	X	X	X	X	X	X	X	X						X		X
SYSTEMS		X	X	X	X	X	X	X	X	X	X	X	X	X	X	X	X

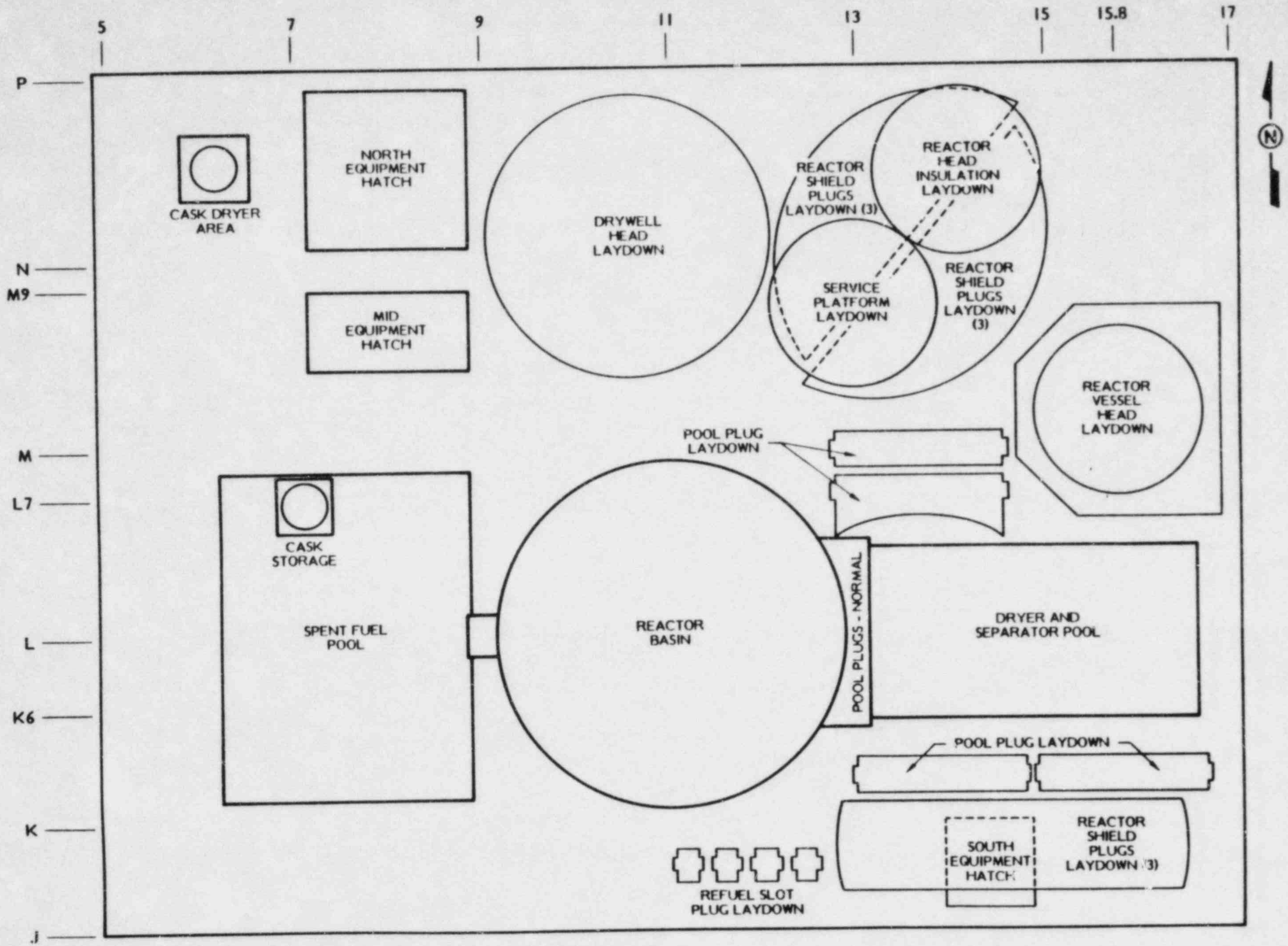

FIGURE 1

MAJOR COMPONENT LAYDOWN
REACTOR BUILDING - 117' EL

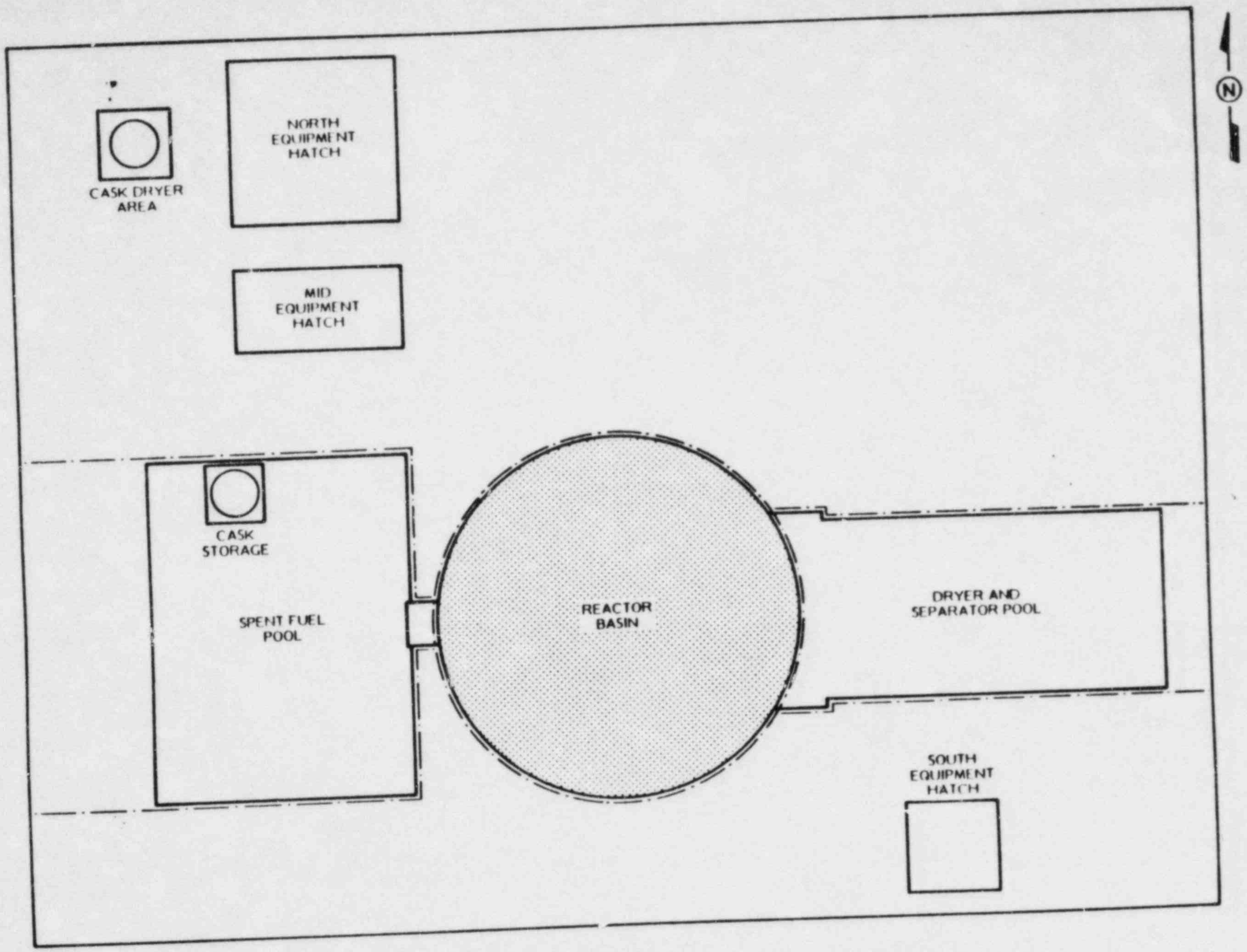

FIGURE 2

LOAD IMPACT REGION I - REACTOR BASIN
REACTOR BUILDING - 117' EL

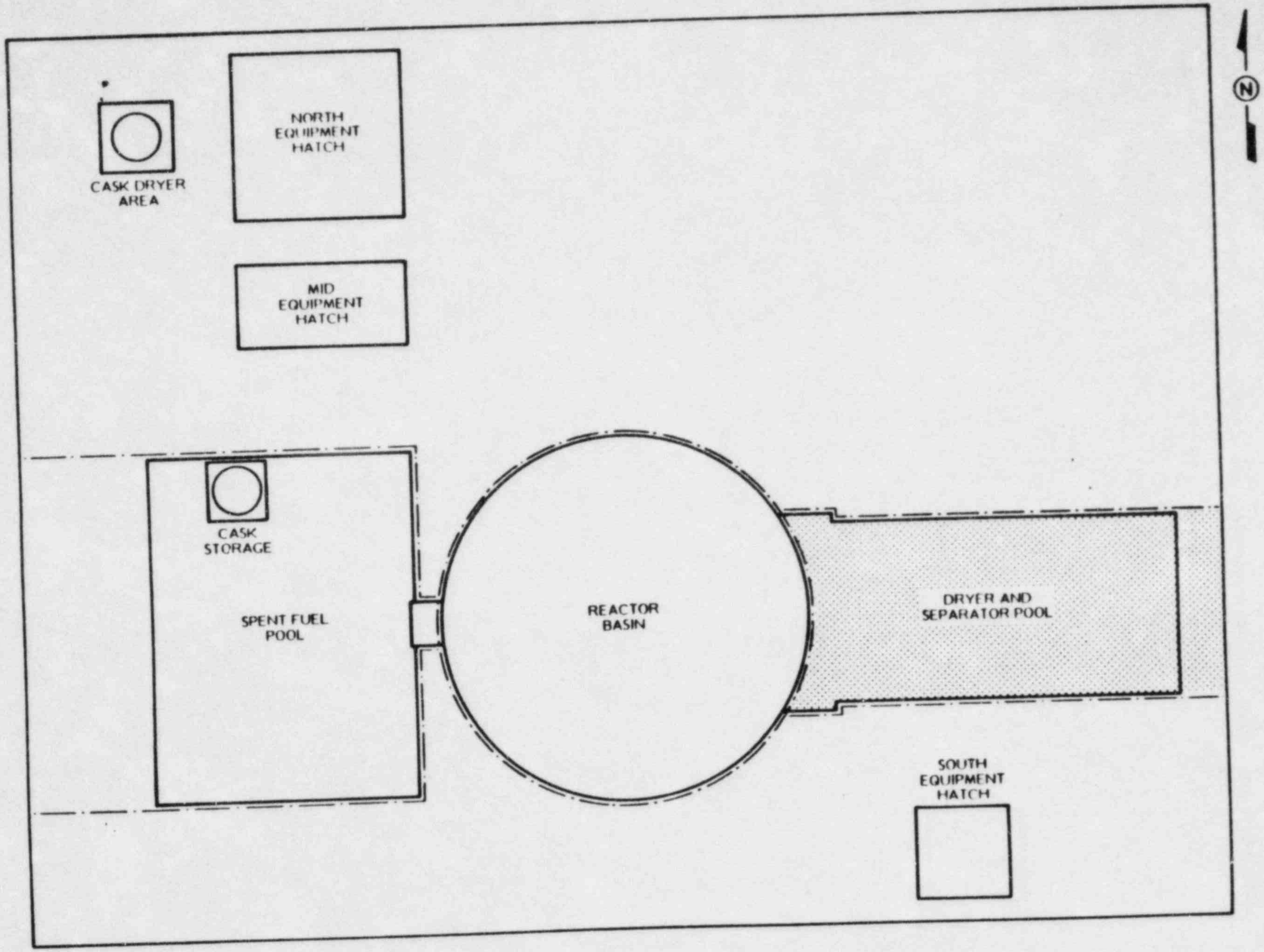

FIGURE 3

LOAD IMPACT REGION 2 - DRYER AND SEPARATOR POOL
REACTOR BUILDING - 117' EL

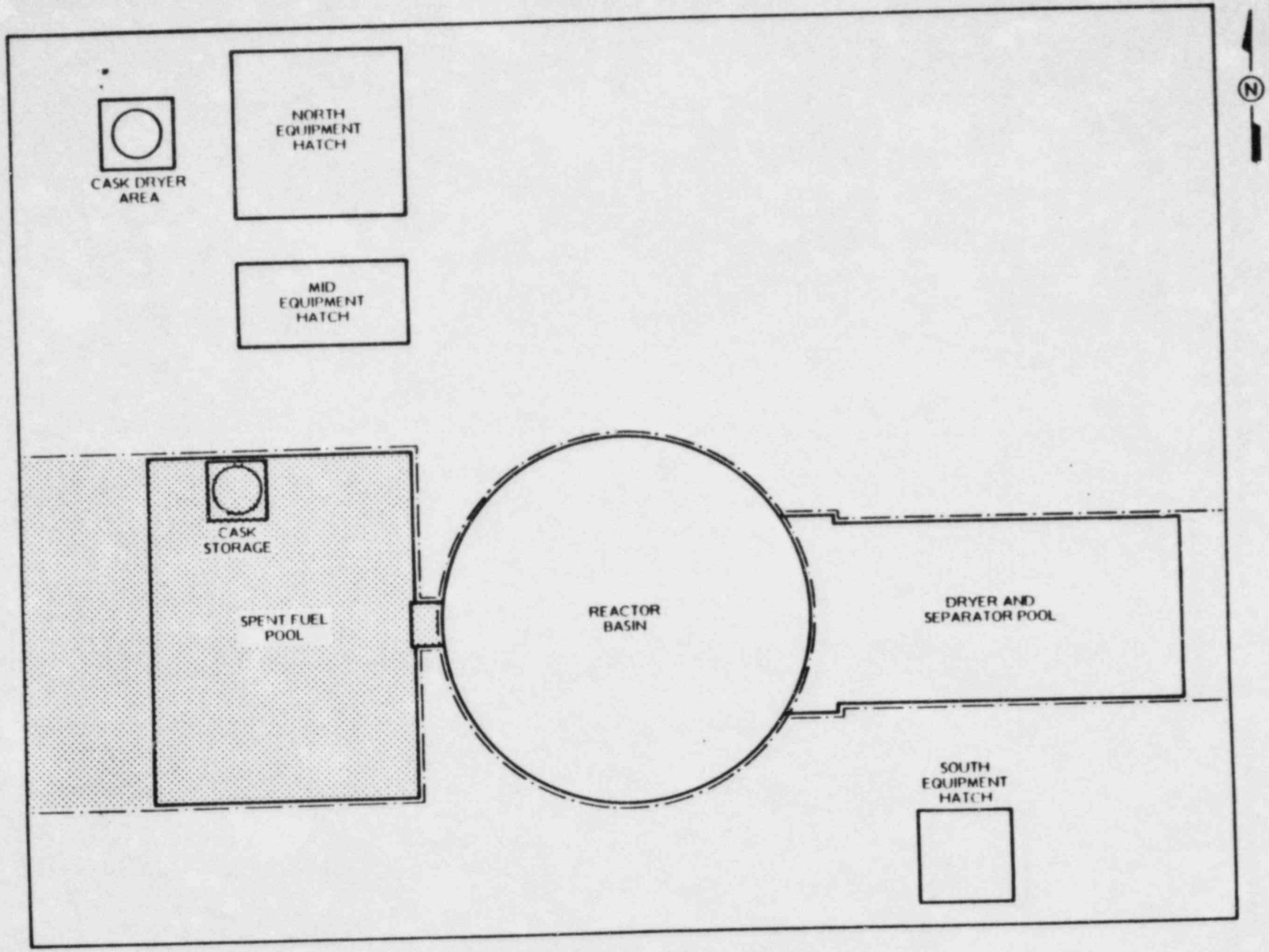

FIGURE 4

LOAD IMPACT REGION 3 - SPENT FUEL POOL
REACTOR BUILDING - 117' EL

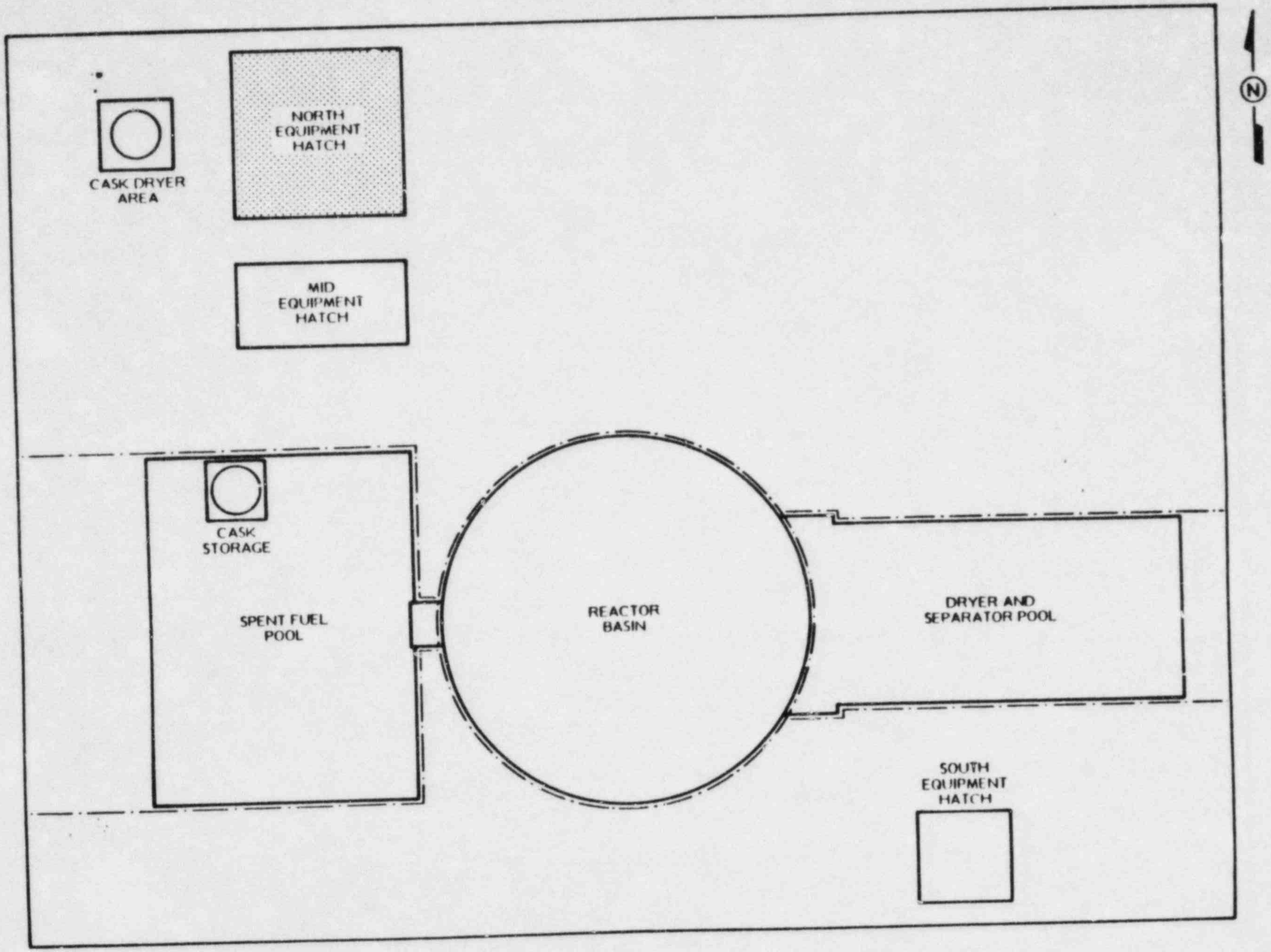

FIGURE 5

LOAD IMPACT REGION 4 - NORTH EQUIPMENT HATCH
REACTOR BUILDING - 117' EL

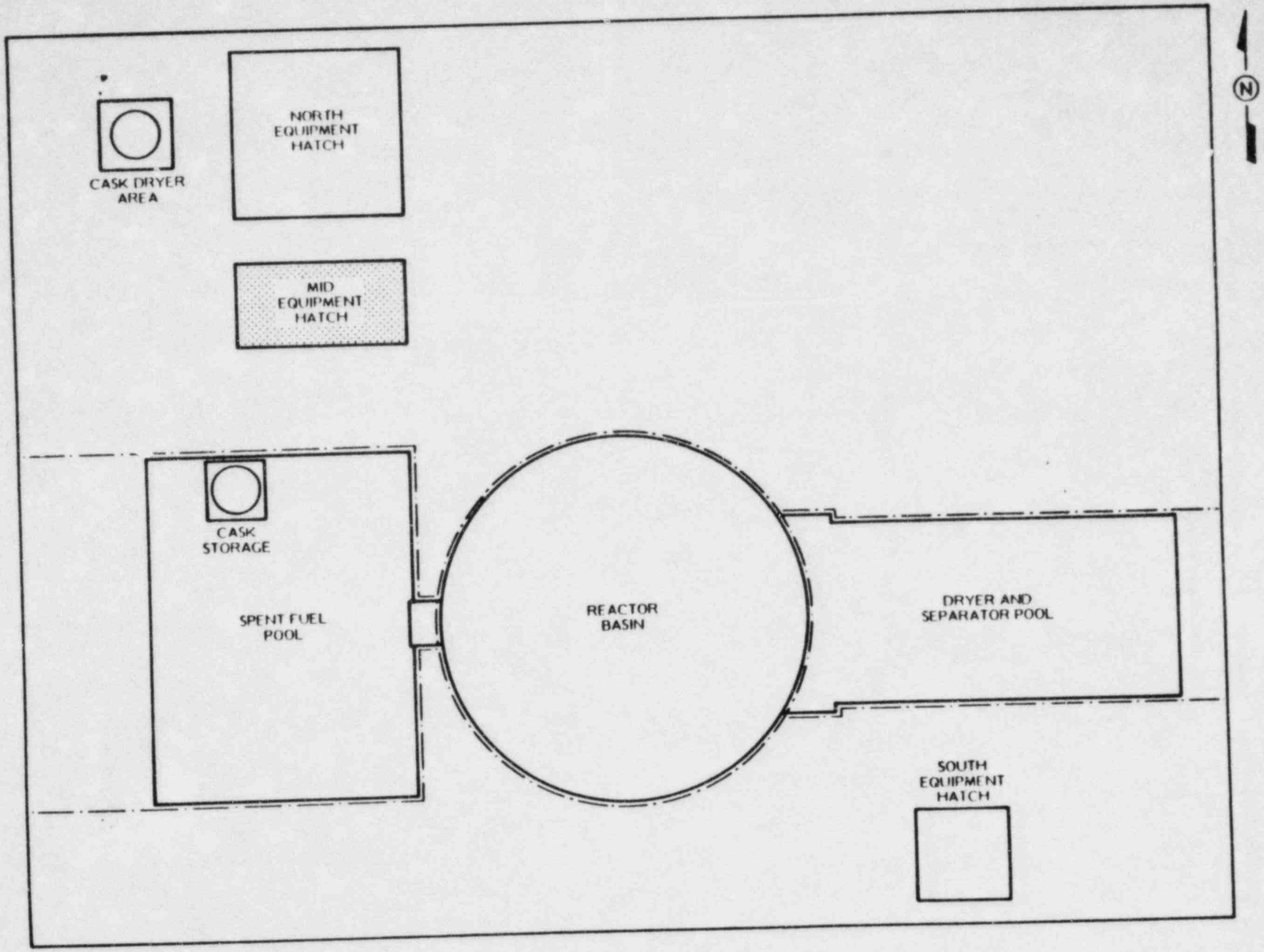

FIGURE 6

LOAD IMPACT REGION 5 - MID EQUIPMENT HATCH
REACTOR BUILDING - 117' EL

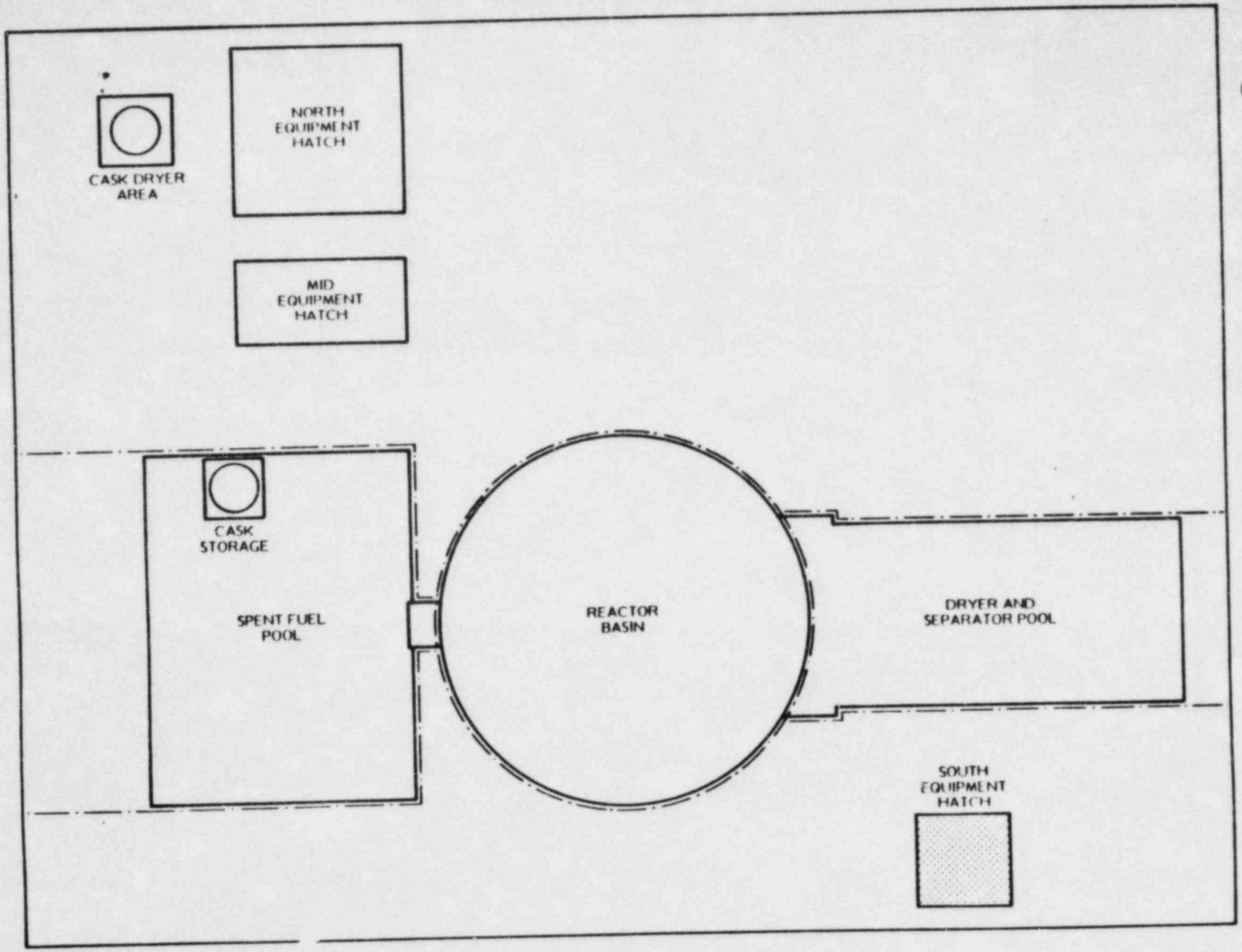

FIGURE 7

LOAD IMPACT REGION 6 - SOUTH EQUIPMENT HATCH
REACTOR BUILDING - 117' EL

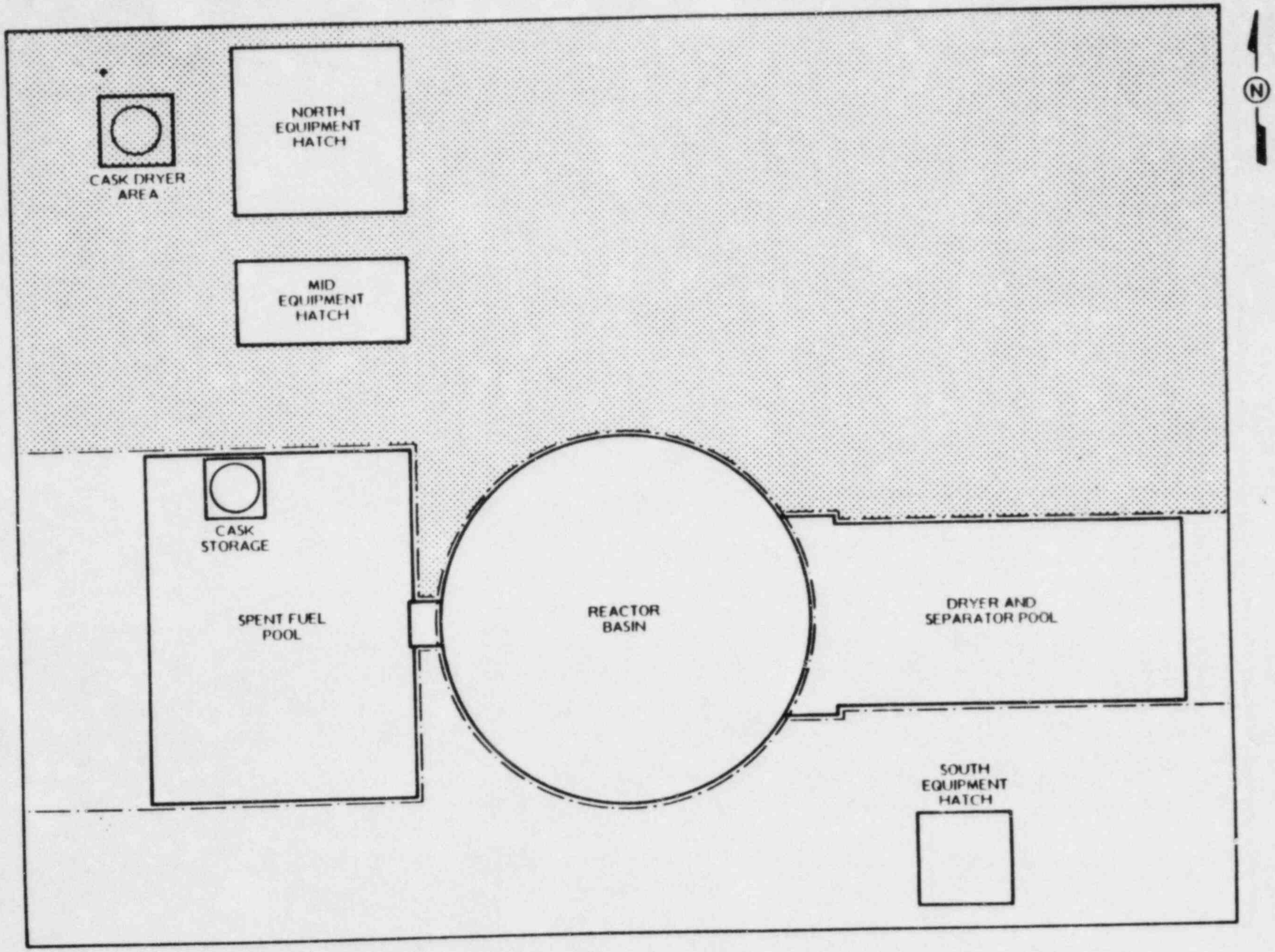

FIGURE 8

LOAD IMPACT REGION 7 - NORTH REACTOR BUILDING
REACTOR BUILDING - 117' EL

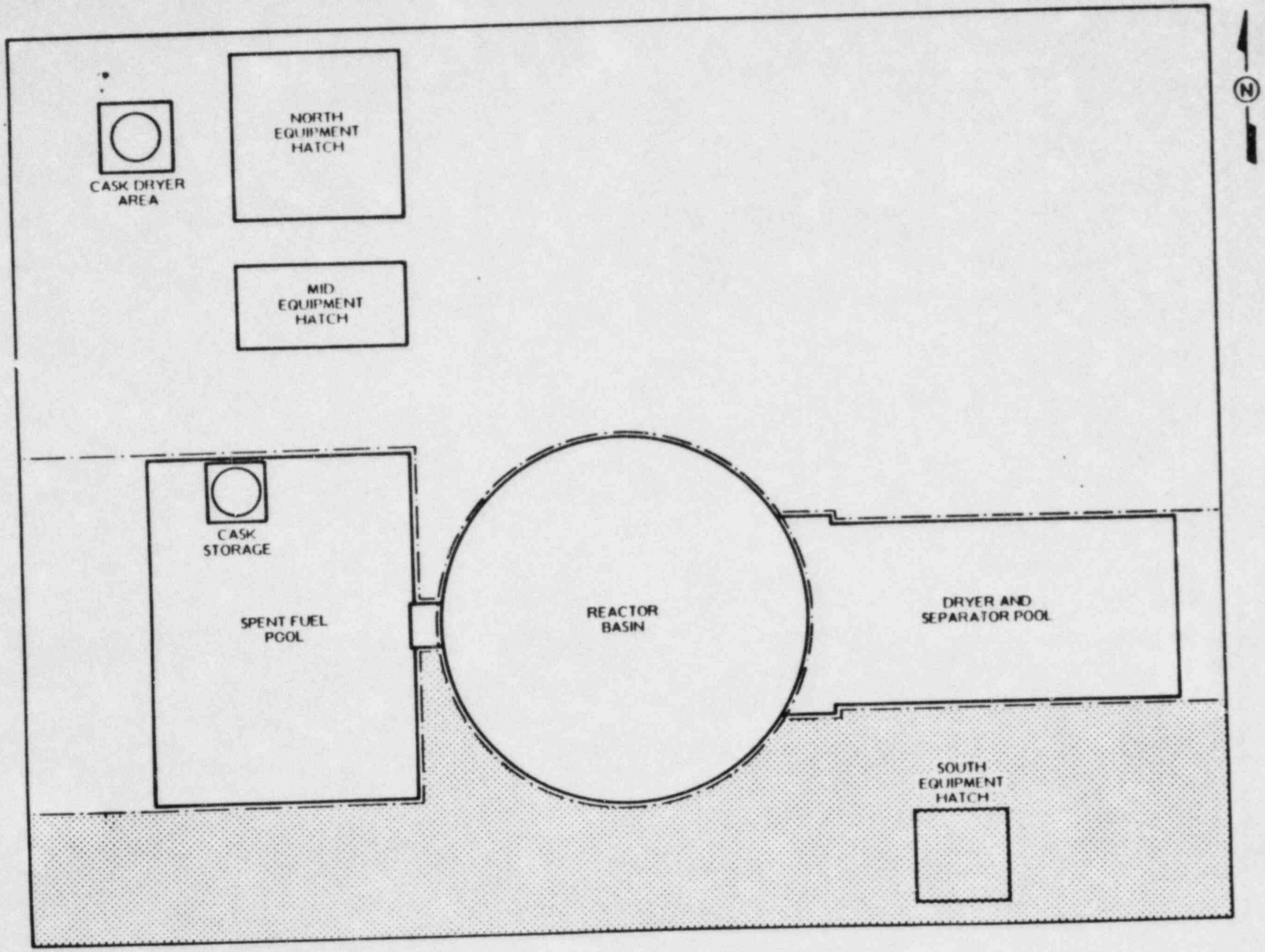

FIGURE 9

LOAD IMPACT REGION 8 - SOUTH REACTOR BUILDING
REACTOR BUILDING - 117' EL

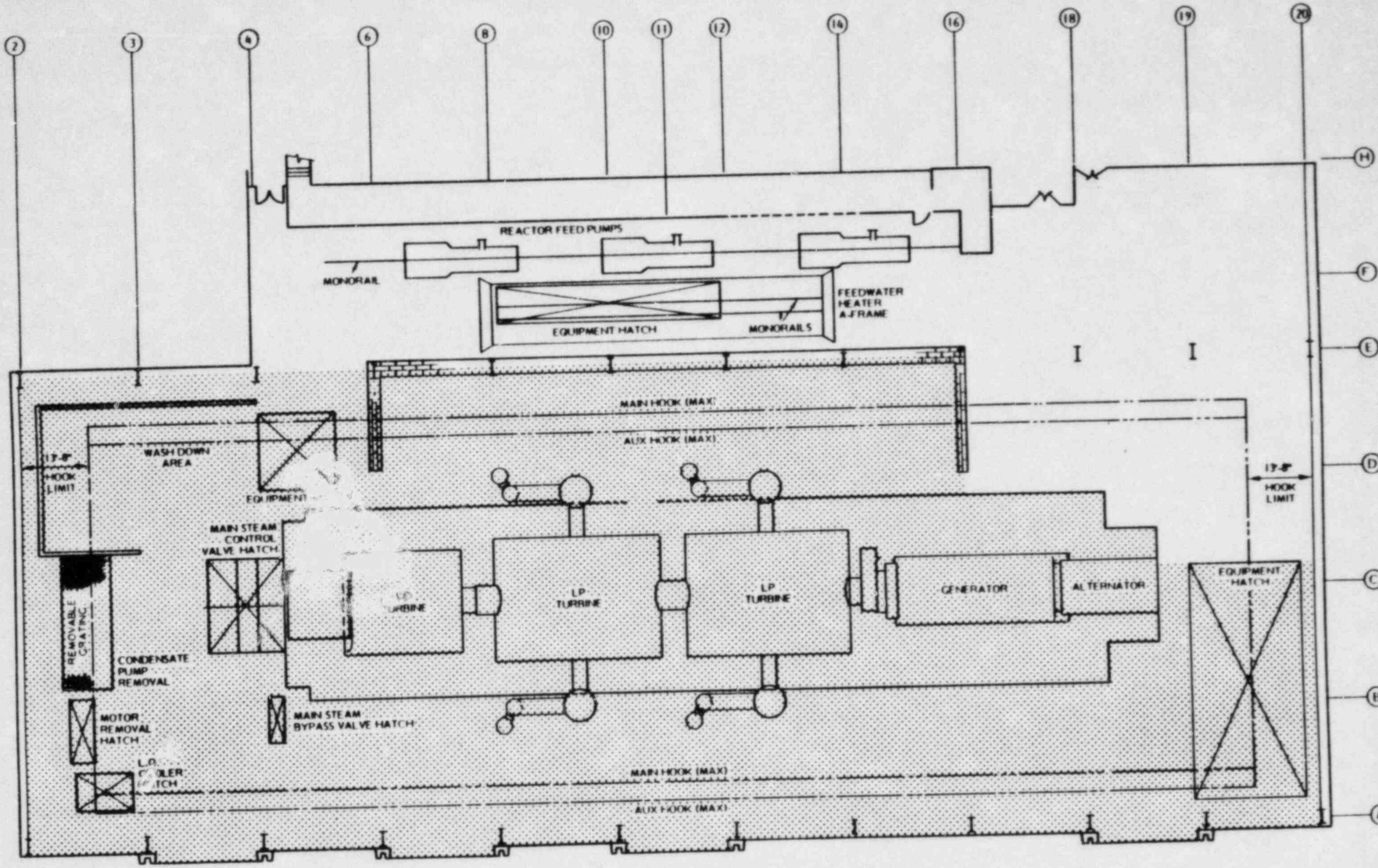

FIGURE 10

LOAD IMPACT REGION 16
TURBINE BUILDING PLAN EL. 51'-0"

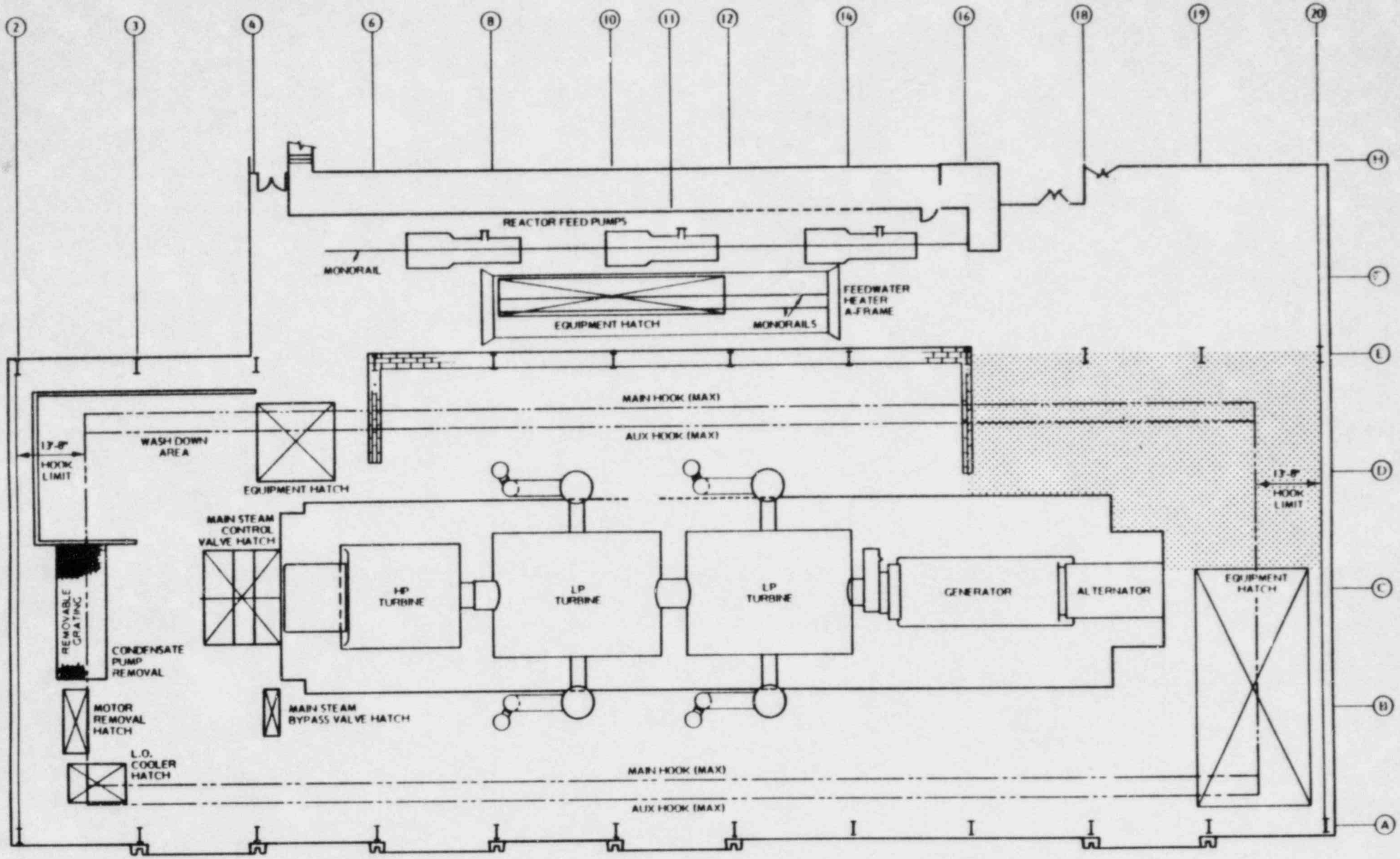

FIGURE 11

LOAD IMPACT REGION 17
TURBINE BUILDING PLAN EL. 51'-0"

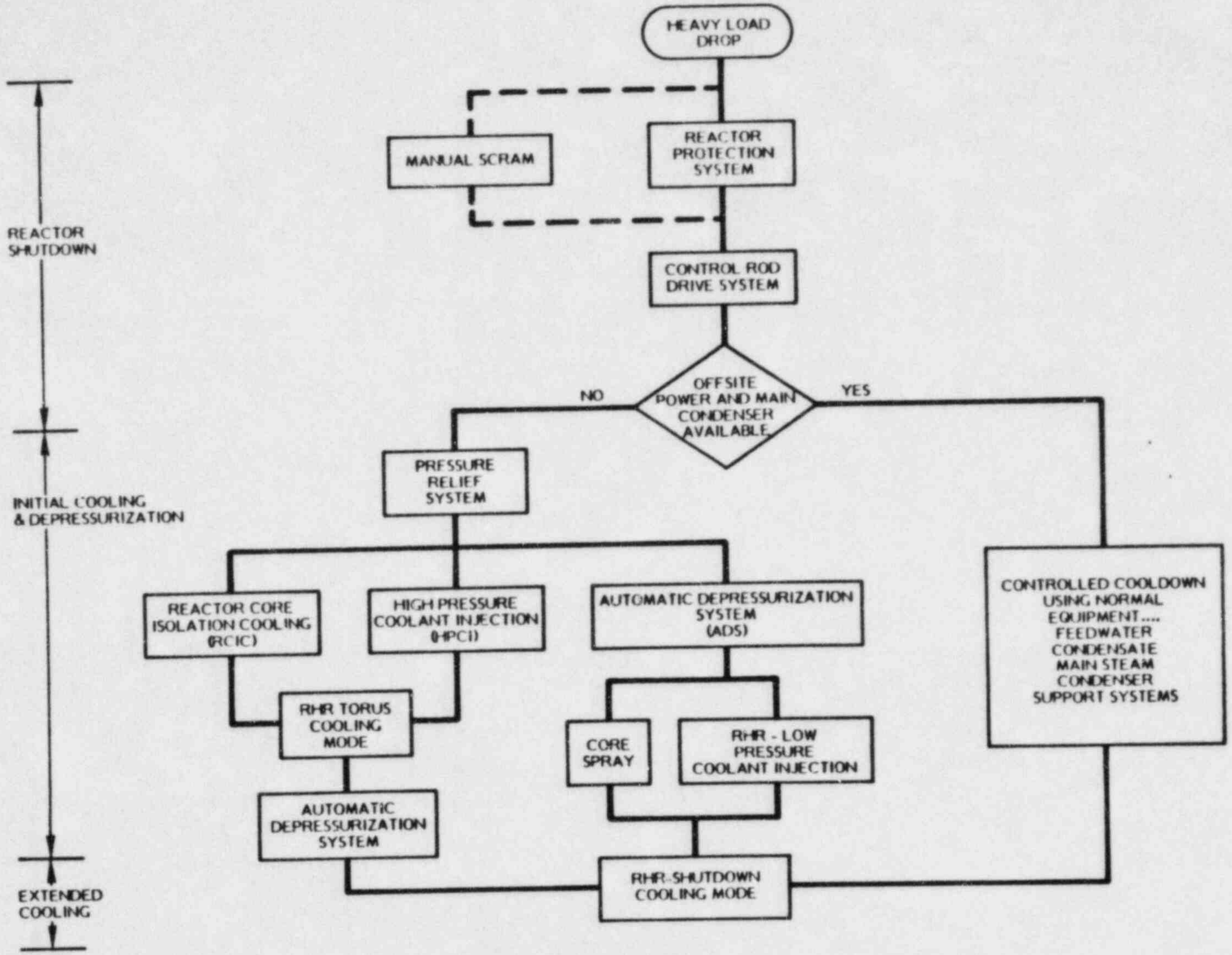

FIGURE 12

SAFE SHUTDOWN CONCEPT FOR HEAVY LOAD EVALUATIONS

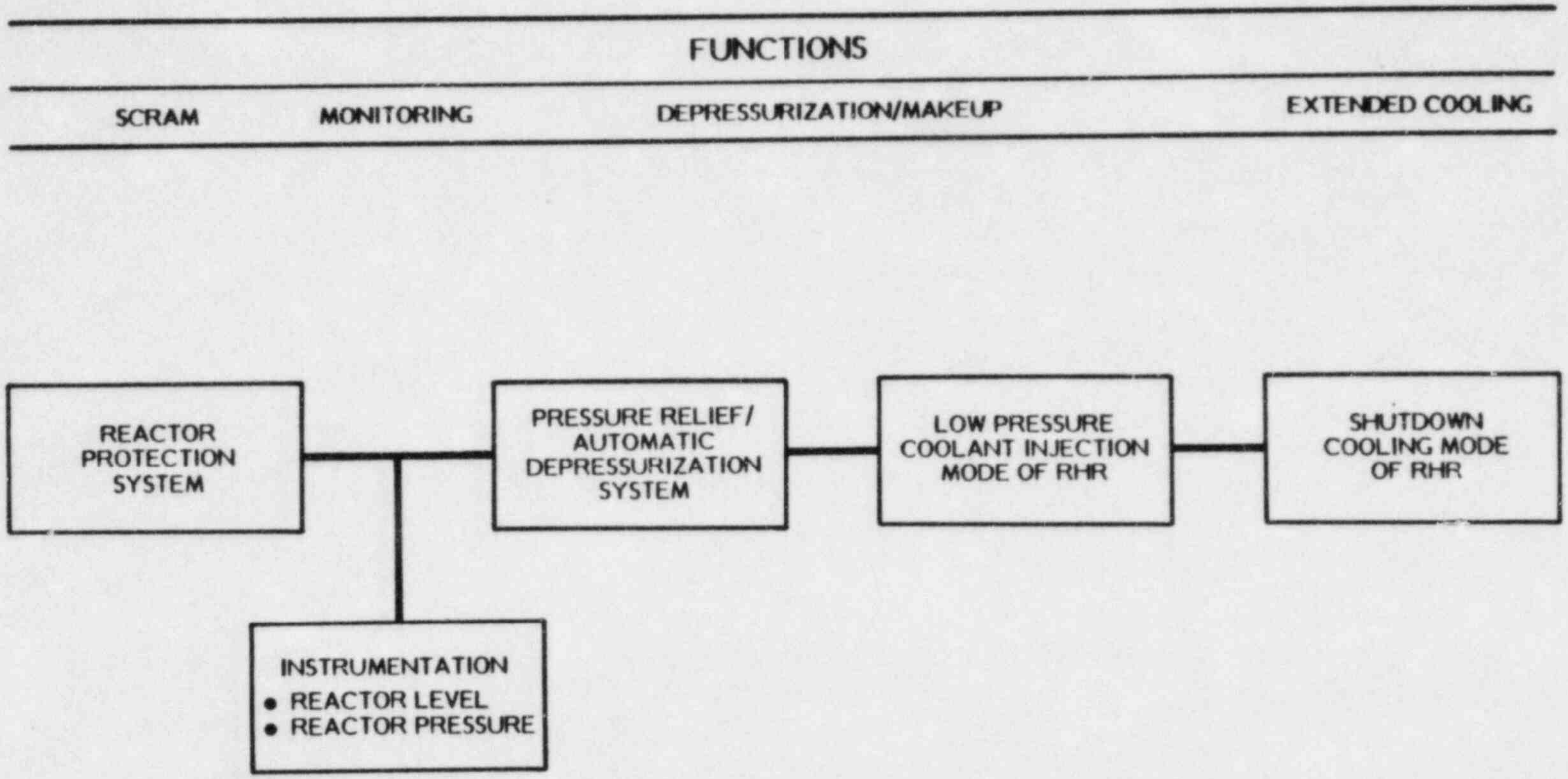

FIGURE 13

SAFE SHUTDOWN SYSTEMS
SELECTED FOR LOAD DROP EVALUATIONS