RESPONSE TO NRC REQUEST FOR INFORMATION

ON CONTROL OF HEAVY LOADS

NINE MONTH REPORT

FOR THE

POINT BEACH NUCLEAR PLANT

UNITS 1 & 2

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CONTROL OF HEAVY LOADS FOR POINT BEACH

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OVERHEAD HANDLING SYSTEM REVIEW

1. INTRODUCTION

This report is the second portion of the Point Beach Nuclear Plant evaluation of overhead handling systems as requested by Nuclear Regulatory Commission (NRC) letters of December 22, 1980 and February 3, 1981 clarification concerning control of heavy loads at nuclear power plants.

The six month report was submitted to the NRC in September 1981 and included the evaluation of the Point Beach overhead handling systems with regard to Section 2.1 of Enclosure 3 of the Nuclear Regulatory Commission's letter of December 22, 1980. This report addresses Sections 2.2, 2.3 and 2.4 of Enclosure 3 of the NRC letter of December 22, 1980 and documents the design review and evaluation of overhead handling systems at the Point Beach Nuclear Plant.

2. SPECIFIC REQUIREMENTS FOR OVERHEAD HANDLING SYSTEMS OPERATING IN THE VICINITY OF FUEL STORAGE POOLS

2.1 NRC QUESTION 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 which could, if dropped, land or fall into the spent fuel pool.

RESPONSE

The following table identifies those cranes which are physically capable (ignoring interlocks, moveable mechanical stops, or operating procedures) of carrying loads which could drop into the spent fuel pool.

Table 2-1

Crane	Type	Capacity (Tons)	Equipment Designator
Auxiliary Building Crane	Bridge	130/20	Z15
Spent Fuel Pool Crane	Bridge	1	Z17

These overhead handling devices and loads carried were addressed in the response to NRC questions 2.1-3 and Tables 4-12 and 4-31 of the Six Month Report.

2.2 NRC QUESTION 2.2-2

Justify the exclusion of any cranes in this area from the above category by verifying that they are incapable of carrying heavy loads or are permanently prevented from movement of the hook centerline closer than 15 feet to the pool boundary, or by providing a suitable analysis demonstrating that for any failure mode, no heavy load can fall into the fuel-storage pool.

RESPONSE

The spent fuel pool crane may be excluded from further consideration as the spent fuel elements weigh less than the defined heavy load of 1750 lbs. The consequences of a spent fuel element drop have been previously analyzed in Section 14.2.1 of the Point Beach Safety Analysis Report and found acceptable in the NRC Safety Evaluation Report.

2.3 NRC QUESTIONS 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., craneload-combination) information specified in Attachment 1.

RESPONSE

The auxiliary building crane will be modified to meet the guidelines of NUREG-0612, Section 5.1.6 or partial compliance supplemented by suitable alternatives or additional design features. Dependent upon equipment delivery, it is expected that the auxiliary building crane upgrade modifications can be completed within two years.

The information requested on Single-Failure-Proof Handling Systems in Attachment 1 to the NRC letter of December 22, 1980, is provided below.

Information on Single Failure Proof Handling System

2.3.1 NRC QUESTION ATTACHMENT 1-1

Provide the name of the manufacturer and the design-rated load (DRL). If the maximum critical load (MCL), as defined in NUREG 0554, is not the same as the DRL, provide this capacity.

RESPONSE

The supplier for the auxiliary building crane modifications has not been selected. This information will be submitted following selection of the supplier.

2.3.2 NRC QUESTION ATTACHMENT 1-2

Provide a detailed evaluation of the overhead handling system with respect to the features of design, fabrication, inspection, testing, and operation as delineated in NUREG 0554 and supplemented by the identified alternatives specified in NUREG 0612, Appendix C. This evaluation must include a point-by-point comparison for each section of NUREG 0554. If the alternatives of NUREG 0612, Appendix C, are used for certain applications in lieu of complying with the recommendation of NUREG-0554, this should be explicitly stated. If an alternative to any of those contained in NUREG-0554 or NUREG 0612, Appendix C, is proposed, details must be provided on the proposed alternative to demonstrate its equivalency.

RESPONSE

See the response to Question Attachment 1-1

2.3.3 NRC Question ATTACHMENT 1-3

With respect to the seismic analysis employed to demonstrate that the overhead handling system can retain the load during a seismic event equal to a safe shutdown earthquake, provide a description of the method of analysis, the assumptions used, and the mathematical model evaluated in the analysis. The decription of assumptions should include the basis for selection of trolley and load position.

RESPONSE

See the response to Question Attachment 1-1

2.3.4 NRC QUESTION ATTACHMENT 1-4

Provide an evaluation of the lifting devices for each single-failure-proof handling system with respect to the guidelines of NUREG-0612, Section 5.1.6.

RESPONSE

No special lifting devices are used with the auxiliary building crane. Lifting devices that are not specially designed will be replaced with slings meeting the requirements of ANSI B30.9-1971, "Slings". In the interim, as the slings are being replaced, the old slings have been derated by a factor of 2. This derating was accomplished by taking the lowest value for a particular diameter from the tables in B30.9-1971 for wire rope slings without regard to sling construction, splice, material and type of hitch and dividing the assumed value by 2. Table 2-2 shows the derated capacities of the slings.

TABLE 2-2
SLING CAPACITIES

CAPACITY (TONS)
FACTOR OF SAFETY = 10

Dia. (Inches)	Single Leg[2]	BRIDLE SLING[3] 2-LEG	BRIDLE SLING[3] 3-LEG	ENDLESS SLINGS
3/32	.12	.16	.24	
1/8	.21	.28	.42	
3/16	.47	.65	.95	
1/4	.18	.24	.37	.31
5/16	.28	.38	.55	.50
3/8	.40	.55	.80	.47
7/16	.55	.70	1.05	.95
1/2	.70	.90	1.40	1.0*
9/16	.85	1.15	1.70	1.05
5/8	1.05	1.40	2.10	1.4
3/4	1.40	1.90	2.85	1.9
7/8	1.95	2.50	3.75	2.9*
15/16	-	-	-	2.95
1	2.50	3.20	4.80	4.0*
1-1/8	3.15	3.85	5.50	4.2
1-1/4	3.70	4.60	7.0	5.4*
1-5/16	3.75	5.0	7.5	5.5
1-3/8	4.10	5.5	8.0	7.0
1-1/2	4.80	6.5	9.5	8.0
1-5/8	6.0	8.0	12.5	
1-11/16	-	-	-	9.0
1-3/4	7.0	9.5	14.0	
1-7/8	-	-	-	11.0
2	9.0	12.5	18.5	
2-1/4	15.5	-	-	15.5
2-5/8	21.0	-	-	21.0

* These capacities were derated to a factor of safety greater than 10 so they would not be of greater capacity than the following larger diameter sling.

NOTES

1. The attached table was developed from Tables 3 thru 14 of ANSI B30.9-1971 by taking the lowest capacity for a specific diameter ignoring sling construction, splice, material, and type of hitch, and derating by a factor of two.

2. For single leg slings using a vertical basket hitch D/d must be 20 or greater and the vertical angle should not exceed 30 degrees.

3. For Bridle slings do not exceed a vertical angle of 60 degrees or a horizontal angle of less than 30 degrees.

4. For endless slings using a vertical basket hitch D/d must be 5 or greater.

Slings used in the turbine building for carrying loads which do not pass over the control building will not be derated, and will not be replaced. All other sling requirements will apply to slings used for these non-safety related lifts. Slings which are used to carry miscellaneous loads over the control building will be derated as per the table. The slings used to carry the turbine rotor over its specified path will not be derated as the effects of the failure of this lifting system have been reviewed and determined acceptable.

This table will be used for old slings throughout the plant until the slings are replaced with the exception of the turbine building. It is expected that all slings used with the auxiliary building crane will be replaced prior to completion of modifications to the crane. Those old slings that are used with the auxiliary building crane have been derated by a factor of 2. When selecting a derated sling for use, the load used will be the sum of the static and maximum dynamic loads neglecting the loads imposed by the SSE. A dynamic load factor of 2 will be used to determine the load.

2.3.5 NRC Question Att. 1-5

Provide an evaluation of the interfacing lift points with respect to the guidelines of NUREG 0612, Section 5.1.6.

Response

Table 4-12 of the Six Month Report lists the loads handled by the auxiliary building crane. Only the following loads have interfacing lift points (lifting lugs or trunions).

New Fuel Shipping Cask
Spent Fuel Shipping Cask
Concrete Hatch Covers
Large Filter Cask
Small Filter Cask
Resin Cask
Watergate

Note: A dynamic load factor of 2 was used for all evaluations.

The new fuel shipping cask is owned by the contractor supplying the new fuel (Westinghouse). The spent fuel shipping casks are currently leased from various suppliers. The new fuel shipping container lifting lugs are designed such that any one of the four lifting lugs is capable of lifting the entire weight of a loaded container. Refueling Procedure 2A dated February 7, 1980 requires that all four lift points be used when handling the container. Based on the provisions above it is concluded that the lifting lugs are acceptable.

The spent fuel shipping cask lift points evaluation will be deferred until a shipping cask that is licensed is chosen for use at the Point Beach Nuclear Plant. No shipping cask movement over the spent fuel or safe shutdown equipment will be permitted until the evaluation is completed and compliance with NUREG-0612, Section 5.1.6(3) or its equivalent is confirmed or justified. Modifications, if required, will be completed prior to cask use.

An evaluation of the lugs for the concrete hatch covers, the large and small filter cask, the resin cask and watergate will be performed and submitted under a separate letter.

2.4 NRC Question 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 2.4 of this request. With respect to Criteria I through III, provide a discussion of your evaluation of crane operation in the spent fuel area and your determination of compliance.

Response

The spent fuel pool crane was identified in 2.2-1 above and was not categorized according to 2.2-3. As stated in the response to 2.2-2, this device carries spent fuel elements which weigh less than the defined heavy load of 1750 lbs. and therefore is excluded from further consideration.

3. SPECIFIC REQUIREMENTS OF OVERHEAD HANDLING SYSTEMS OPERATING IN THE CONTAINMENT

3.1 NRC Question 2.3-1

Identify by name, type, capacity, and equipment designator, any cranes physically capable (i.e., taking no credit for any interlocks or operating procedures) of carrying heavy loads over the reactor vessel.

Response

The following table identifies those cranes which are physically capable (ignoring interlocks, moveable mechanical stops, or operating procedures) of carrying heavy loads over the reactor vessel.

Table 3-1

Crane	Type	Capacity (Tons)	Equipment Designator
Containment Polar Crane	Polar	100/15	Unit 1 1-Z13 Unit 2 2-Z13
Reactor Pressure Vessel Head Circular Monorail	Monorail	2	Unit 1 None Unit 2 None

The above overhead handling devices and loads carried were addressed in the response to NRC Question 2.1-3 and in Tables 4-7, 4-9, 4-28 and 4-29 of the Six Month Report.

3.2 NRC Question 2.3-2

Justify the exclusion of any cranes in this area from the above category by verifying that they are incapable of carrying heavy loads, or are permanently prevented from the movement of any load either directly over the reactor vessel or to such a location where in the event of any load-handling-system failure, the load may land in or on the reactor vessel.

Response

None of the cranes identified in Question 2.3-1 above may be excluded from carrying heavy loads either directly over the reactor vessel or to such a location where in the event of load-handling-system failure, the load may land in or on the reactor vessel.

3.3 NRC Question 2.3-3

Identify any cranes listed in 2.3-1, 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

None of the cranes listed in Table 3-1 totally meet the single failure-proof criteria as outlined in NUREG-0612, Section 5.1.6.

The containment polar crane was designed in accordance with EOCI-61, "Specifications for Electric Overhead Traveling Cranes". In the response to Question 2.1-3-e of the Six Month Report, the design of the Containment Polar Crane was compared to CMAA-70 and Chapter 2-1 of ANSI B30.2-1976. This comparison showed that the polar crane essentially complies with the guidelines of the above standards and where deviations occur, justification is given or modifications will be made. Based on this comparison and the incorporation of additional design features, the polar crane is deemed to be highly reliable although not strictly single failure proof from a design standpoint.

3.4 NRC Question 2.3-4

For cranes identified in 2.3-1, above, not categorized according to 2.3-3, demonstrate that the evaluation criteria of NUREG-0612, Section 5.1, are satisfied. Compliance with Criterion IV will be demonstrated in your response to Section 2.4 of this request. With respect to Criteria I through III, provide a discussion of your evaluation of crane operation in the containment and your determination of compliance. This response should include the following information for each crane:

3.4.1 NRC Question 2.3-4-a

a. Where reliance is placed on the installation and use of electrical interlocks or mechanical stops, indicates 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 specification concerning the bypassing of such interlocks.

Response

No reliance is placed on the installation and use of electrical interlocks or mechanical stops for the cranes listed in Table 3-1 above.

3.4.2 NRC Question 2.3-4-b

b. 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 continued validity of such considerations.

Response

Reliance is placed on site-specific considerations for the Containment Polar Crane. Once the reactor vessel head is removed, the movement of any heavy loads over the open reactor vessel is prohibited procedurally and administratively unless specifically approved in advance by the Manager's Supervisory Staff. The exceptions to this are the removal and replacement of the upper internals, core support barrel and P.A.R. device. The core support barrel may only be lifted after all fuel has been removed from the vessel and therefore poses no threat to the continued removal of core decay heat or fuel damage.

A reactor vesssel head drop analysis will be performed to demonstrate compliance with the criteria of NUREG-0612, Section 5.1. The analysis will consider the guidelines of NUREG-0612, Appendix A for the analyses performed and where exceptions are taken, justification will be given. An evaluation of the upper internals drop will be reviewed in the head drop analysis. The results of the head drop analysis will be available in a report within one year.

The use of the P.A.R. device while fuel is in the vessel has been reviewed and found acceptable. During refueling, Technical Specification 15.3.8 (Appendix C of this report) requires that a minimmum boron concentration of 1800 ppm be maintained. The boron concentration is maintained at 2000 ppm and thus gives a K_{eef} of less than .90. NUREG-0612, Appendix A, Section 4.2.2(2) states that an acceptable method of demonstrating subcriticality is to demonstrate that K_{eff} for the uncrushed core is no greater than .90, then using the estimated 0.05 maximum reactivity insertion due to crushing show that K_{eff} is still less than .95. Based on a refueling K_{eff} of less than .90 and a 0.05 reactivity insertion the maximum K_{eff} is less than .95.

The present design provides radiation monitors with the capability of quickly detecting and isolating the containment including the purge and vent lines with the exception of the personnel access hatch. This system is presently being replaced with safety grade components that perform the same function. Technical Specification 15.3.8 provides for closure of the personnel access hatch after evaculation and also requires a third door having an automatic door closer which minimizes the exchange of inside air with outside air.

The above basis can also be applied to the movement of the vessel head and upper internals.

The plant procedures will be modified to ensure that requirements of Technical Specification 15.3.8 for refueling operations, are also met before movement of the vessel head, upper internals or P.A.R. device.

The reactor pressure vessel head circular monorail is an integral part of the reactor vessel head lifting structure. This monorail is used to position and move the reactor vessel studs, stud tensioners and the cavity seal ring and can only be used when the vessel head is in place and thus does not pose a threat to fuel assemblies in the core. The consequences of a drop of any of the the above loads on the vessel head are expected to be encompassed by the head drop analysis. This will be confirmed upon completion of the analysis.

3.4.3 NRC Question 2.3-4-c

c. Analyses performed to demonstrate compliance with Criteria I through III should conform with 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

As stated in the response to 2.3-4-b above, any exceptions to the guidelines of NUREG-0612 Appendix A, for the analyses performed, will be provided and justified in the future report of the reactor head drop analysis.

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

4.1 NRC Question 2.4-1

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

The Auxiliary Building Crane will be modified to meet the guidelines of NUREG-0612, Section 5.1.6. See the response to Question 2.2-3 above for additional information.

4.2 NRC Question 2.4-2

For any cranes identified in 2.1-1 not designated as single-failure-proof in 2.4-1, a comprehensive hazard evaluation should be provided which includes the following information:

4.2.1 NRC Question 2.4-2-a

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 method such that the impact area can be located on the plant general arrangement drawings. Figure 1 provides a typical matrix.

Response

Table 4-2 of the Six Month Report identifies those overhead handling systems which are in the vicinity of Safe Shutdown Equipment. This table is reproduced in this report as Table 4-1. The tables giving the information requested above for the handling devices listed in Table 4-1 are given in the updated Six Month Report.

Table 4-1

List of Overhead Heavy Load* Handling Devices in the Vicinity of Safe Shutdown Equipment

Item #	Description	
1	Circulating Water Pumphouse Monorail N - S	
2	Circulating Water Pumphouse Monorail E - W	
3	Reactor Pressure Vessel Head Circular Monorail	- Unit 1
5	Containment Polar Crane	- Unit 1
6	Containment Buttress Jib Cranes	- Unit 1
8	Auxiliary Building Main Crane	
10	Main Shop Crane	
12	Jib Crane Over Incore Instrumentation	- Unit 1
16	Turbine Building Main Crane	
18	Jib Crane Over Incore Instrumentation	- Unit 2
23	Containment Buttress Jib Cranes	- Unit 2
24	Reactor Pressure Vessel Head Circular Monorail	- Unit 2
25	Containment Polar Crane	- Unit 2
31	Facade Monorail at Column L - 8	- Unit 1
33	Facade Monorail at Column L - 15	- Unit 2
34	Facade Monorail at Column L - 16	- Unit 2

*Heavy Load defined as 1750 lbs. or greater - See Appendix A Definitions

4.2.2 NRC Question 2.4-2-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.

Response

All of the handling systems in Table 4-1 except those listed below, may be eliminated based on separation and redundancy of safe-shutdown equipment, mechanical stops and/or electrical interlocks, or other site specific considerations.

Item #	Description
8	Auxiliary Building Main Crane

4.2.2.1 NRC Question 2.4-2-b(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).

Response

Circulating Water Pumphouse Monorail N-S

This monorail may be eliminated based on separation and redundancy as there are six service water pumps available while only three pumps are required to safely shutdown the plant. There are no common cables, switchgear or piping under the load path of the monorail.

Circulating Water Pumphouse Monorail E-W

This monorail is eliminated based on separation and redundancy for the same reasons as described above for the N-S Monorail.

Reactor Pressure Vessel Head Circular Monorail Units 1 and 2

The drop of any single load from this monorail will not disable the removal of decay heat from the core due to redundancy and separation of the RHR supplies to the reactor vessel.

Containment Polar Crane Unit 1

The elimination of this crane is based on the capability of the plant to provide continued decay heat removal regardless of what load is dropped due to separation and redundancy or alternate decay heat removal paths such as safety injection.

The present design of the crane incorporates two limit switches in the reeving system, both in the same circuit, to prevent the two blocking accident. To provide separation and redundancy, the crane design will be modified to place one limit switch in the power circuit and one in the control circuit of the reeving system.

Containment Buttress Jib Crane Units 1 and 2

Further review of the containment buttress jib cranes has shown that they may be eliminated based on redundancy and separation. These cranes do not carry heavy loads over safe shutdown equipment except for the cables for a redundant diesel fuel oil transfer pump for diesel generator A and the residual heat removal suction line for Units 1 and 2. These suction lines are protected since they are embedded in the basemat concrete at the junction between the Containments and the Auxiliary Building.

Main Shop Crane

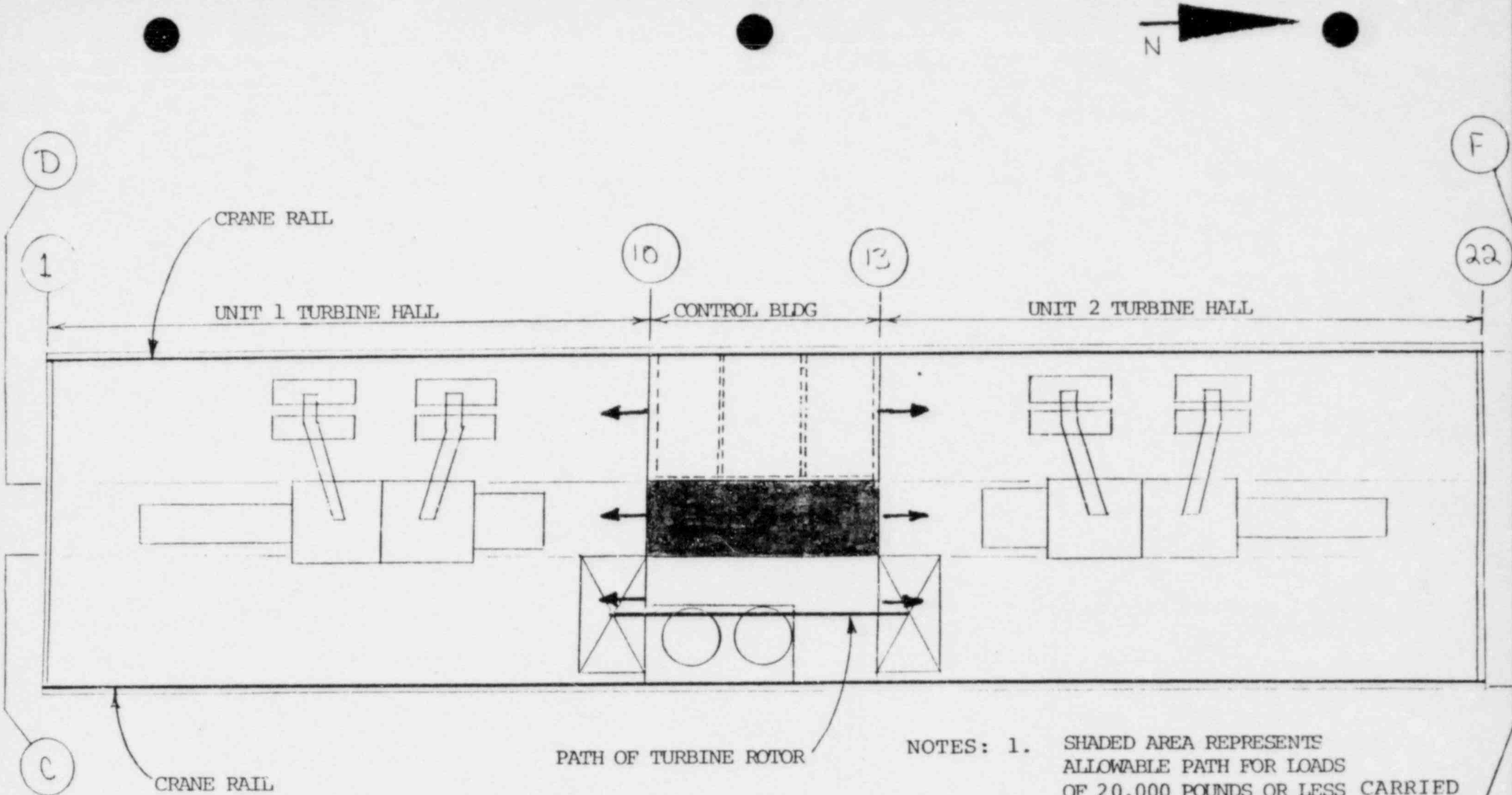
This crane may be eliminated as only the cables for one train of the auxiliary feedwater system may be impacted by a load drop leaving the redundant train available to supply the required feedwater.

Jib Cranes Over Incore Instrumentation Units 1

This jib crane may be eliminated due to separation and redundancy and the availability of safety injection as an alternate decay heat removal path.

Turbine Building Main Crane

Due to the possibility of loss of all safety and non-safety related 4.16 kv switchgear from a load drop over the area bounded by columns 10, 13, C and D on Figure 4-1 it is necessary that critical loads handled by the Turbine Building



NOTES: 1. SHADED AREA REPRESENTS ALLOWABLE PATH FOR LOADS OF 20,000 POUNDS OR LESS CARRIED AT 9 INCHES OFF THE FLOOR

2. MOVEMENT OF HEAVY LOADS IS UNRESTRICTED SOUTH OF COLUMN LINE 10 AND NORTH OF COLUMN LINE 13.

TURBINE BUILDING MAIN CRANE

FIGURE 4-1

Crane follow the load path indicated on the Figure. The only critical load is the spare LP turbine rotor. A load drop at any height on the slab bounded by the above columns would cause spalling and penetration of the floor above the switchgear.

All other loads weighing 20,000 pounds or less can be carried over the area above the switchgear at a maximum calculated height (9 inches) such that spalling of the concrete will not occur and damage the switchgear. See Appendix B for further information.

The load path for the critical load, which has been defined and shown in Figure 4-1 requires that the load be carried over the condensate storage tanks, diesel generators, service water piping, instrument air compressors and the service air compressor. The consequences of a drop on the above equipment were reviewed and determined to be acceptable. Loss of the condensate storage tanks will not affect ability to remove decay heat as the service water system provides a backup water supply for the auxiliary feedwater system. The service water lines are separated by about 70 feet, and run parallel to and very near column lines 10 and 13. Each line is fully capable of supplying all service water requirements to essential equipment.

The loss of both diesel generators was reviewed and determined to be less severe than the loss of all 4.16 kv switchgear which causes a prolonged loss of both onsite and offsite power (blackout). If the diesels are lost due to a load drop, offsite preferred power would still be available to supply the required loads, and the Technical Specifications concerning a loss of the diesel generators would be followed.

Loss of the instrument and service air compressor would not disable the diesels as the starting air receivers would still be available to start the diesels when required. Instrument and service air is not needed to safely shutdown the plant.

The present design of the crane incorporates two limit switches in the reeving system, both in the same circuit to prevent the two blocking accident. To provide separation and redunancy, the crane design will be modified to place on limit switch in the power circuit and one in the control circuit of the reeving system.

Jib Crane Over Incore Instrumentation - Unit 2

This crane may be eliminated based on separation and redundancy and the availability of both residual heat removal and safety injection for decay heat removal.

Containment Polar Crane - Unit 2

The elimination of this crane is based on the capabilty of the plant to provide continued decay heat removal due to separation and redundancy of safe shutdown equipment or alternate decay heat removal paths such as safety injection. The floor slab under the laydown area for the "B" reactor coolant pump flywheel was analyzed to determine the maximum height that the flywheel could be carried without structural failure. This analysis showed that the maximum height is 4 feet. The safe load path will indicate the maximum height allowed. A further discussion of the analysis is given in Appendix A.

The present design of the crane incorporates two limit switches in the reeving system, both in the same circuit to prevent the two-blocking accident. To provide separation and redundancy the crane design will be modified to place one limit switch in the power circuit and one in the control circuit of the reeving system.

Facade Monorails At Columns L-8 - Unit 1
L-15 - Unit 2
L-16 - Unit 2

These monorails may be eliminated from further considerations as they do not handle the defined heavy loads of 1750 lbs. or more. They are used strictly for handling the main steam relief valves which weigh 1250 lbs.

4.2.2.2 NRC Question 2.4-2-b(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 bypassing 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.

Response

No handling devices used at the Point Beach Nuclear Plant were eliminated from further consideration by use of mechanical stops or electrical interlocks.

4.2.2.3 NRC Question 2.4-2-b(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 continued validity of such considerations.

Response

No load/target combinations were eliminated on the basis of site-specific considerations at the Point Beach Nuclear Plant.

4.2.3 NRC Question 2.4-2-c

For interactions not eliminated by the analysis of 2.4-2b, 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 crane so evaluated, provide the load-handling-system (i.e., crane-load-combination) information specified in Attachment 1.

Response

The Auxiliary Building Crane will be modified to meet the guidelines of NUREG-0612, Section 5.1.6. See the response to Question 2.2-3 in Section 2.3 above for additional information.

4.2.4 NRC Question 2.4-2-d

For interactions not eliminated in 2.4-2-b or 2.4-2-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 constucted 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.

Response

The following interactions could not be eliminated by 2.4-2-b or 2.4-2-c above

Crane	Load	Weight
Unit 2 Containment Polar Crane	B Reactor Coolant Pumps Flywheel	14,000 lbs
Turbine Building Main Crane	All loads less than 17,000 lbs	17,000 lbs

The paragraphs below correspond to (1), (2) and (3) of the above question.

Containment Polar Crane

1) The Containment Polar Crane was not designed to retain the flywheel during a safe shutdown earthquake (SSE). The design basis for the crane required that it be in an unloaded condition during an SSE and that no part of the crane may become dislodged and fall on equipment or structures in the event of an earthquake.

2) No exceptions are taken to the analytical guidelines of NUREG-0612, Appendix A.

3) The information requested in Attachment (4) to the Commissions letter of December 22, 1980 is provided in Appendix A of this report.

TURBINE BUILDING MAIN CRANE

1) The Turbine Building Main Crane was not designed to retain the load in the event of an earthquake.

2) No exceptions are taken to the analytical guidelines of NUREG-0612, Appendix A.

3) The information requested in Attachment (4) to the Commissions letter of December 22, 1980 is provided in Appendix B of this report.

APPENDICES

Appendix A - Load Drop Analysis of Unit 2,
B Reactor Coolant Pump Flywheel

Appendix B - Load Drop Analysis of 17,000 lb. Main
Feed Pump Motor in the Control Building

Appendix C - Technical Specification 15.3.8
Refueling and Spent Fuel Assembly Storage

APPENDIX A

Load Drop Analysis of the 14,000 lb. Unit 2 Containment B Loop Reactor Coolant Pump (RCP) Flywheel.

A.1 Initial Conditions and Assumptions:

The RCP flywheel is lifted using three cables attached around the perimeter of the flywheel (see figure A-1). Two drop cases were considered. Case I assumes the flywheel drops straight down. The flywheel will impact over at east one of the steel beams under the slab. Case II assumes that one of the lift cables fails causing the flywheel to rotate before impacting the slab.

I: Straight Down Drop (See Figure A-1)	II: Rotational Drop (See Figure A-1)
a. Weight of RCP flywheel: 14,000 lbs.	a. 14,000 lbs.
b. Impact area of Load: 75 inch diameter	b. 2 inches by 26 inches
c. Drop height: 48 inches	c. 48 inches
d. Drop location: Midspan on smallest steel beam, (see figure A-2)	d. Center of largest slab panel, (see figure A-2)
e. Assumptions regarding credit taken in the analysis for the action of impact limiters: Impact limiters will not be used, so no credit was taken.	e. Impact limiters will not be used, so no credit was taken.
f. Thickness of floor slab: 4.5 inch slab with 1.5 inch metal decking	f. 4.5 inch slab with 1.5 inch metal decking
g. Assumptions regarding drag forces caused by the environment: No credit was taken for environmental drag forces.	g. No credit was taken for environmental drag forces.

I: Straight Down Drop

h. Load combination considered:

1.0D + 1.0L + 1.0I

D = dead load of slab

L = live loads on slab

I = impact loads on slab

i. Material properties of concrete and steel:

Steel:

Fy: dynamic value of 1.2 x yield strength = 1.2 x 36,000 = 43,200 psi

II: Rotational Drop

h. I.OD + 1,0L + 1.0I

D = dead load of slab

L = live loads on slab

I = impact loads on slab

i. Concrete: F'c dynamic value of 1.1 x ultimate strength = 1.1 x 4000 = 4400 psi

Fy: dynamic value of 1.2 x yield strength = 1.2 x 40,000 = 48,000 psi

V: poisson's ratio = 0.17

A.2 Method of Analysis

Load impact effects are assessed in terms of local damage and structural response.

Local damage, damage that occurs in the immediate viscinity of the impact area, is assessed in terms of perforation and spalling. The local damage evaluation ensures that the systems protected by the structural barrier would not be damaged by a load perforating the protective barrier or by creation of secondary missiles (spalling). In areas where slabs were poured on metal decking, the decking prevents spall particles from impacting protected systems.

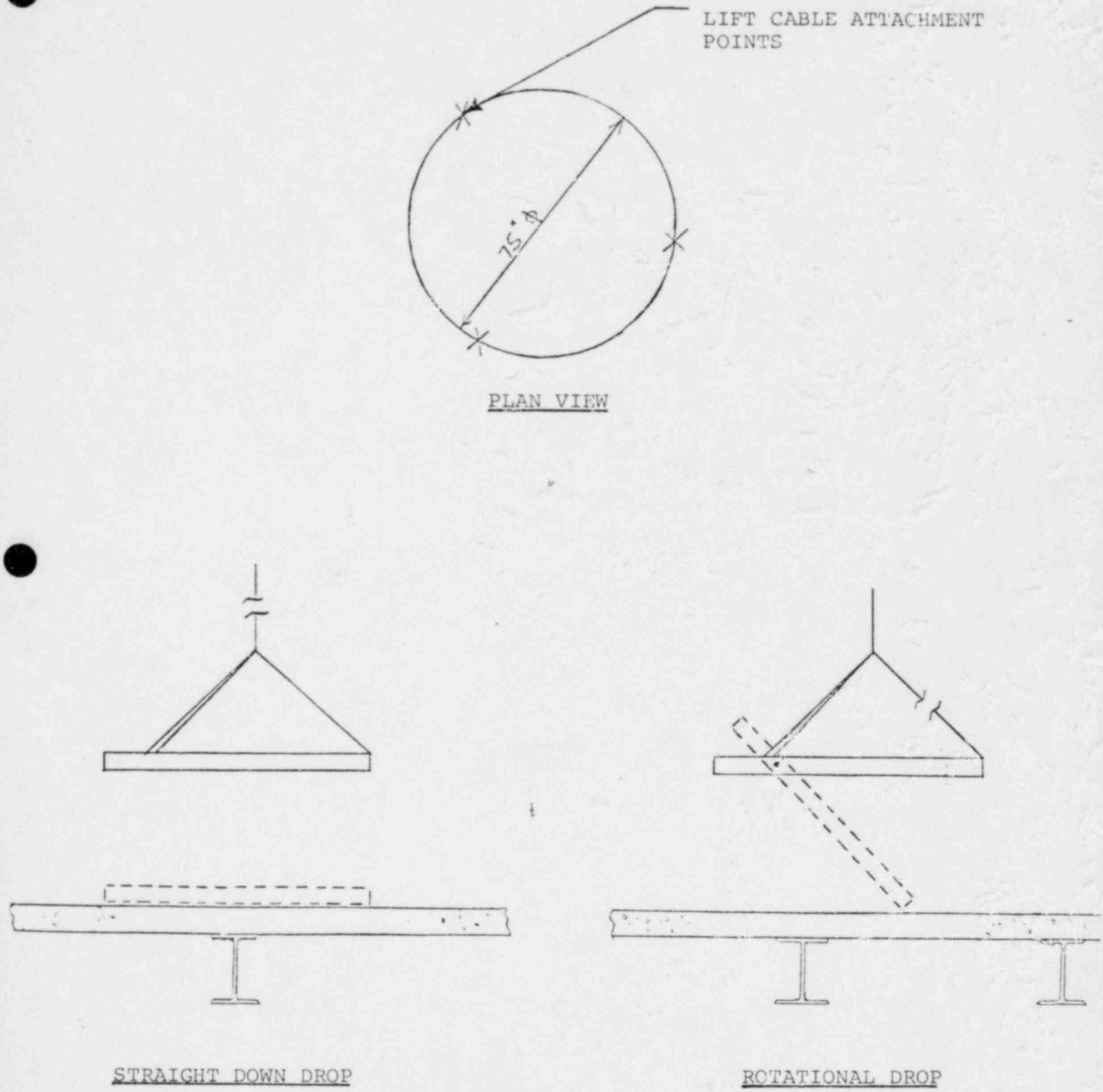
Structural reponse is assessed in terms of deformation limits and strain energy capacity. Structural response is determined by use of conservation of momentum and energy balance techniques.

A.3 Conclusion

The safety systems protected by the Containment El. 66' slab will not be impaired by the drop of the 14,000 lb. RCP flywheel from a height not exceeding 48 inches.

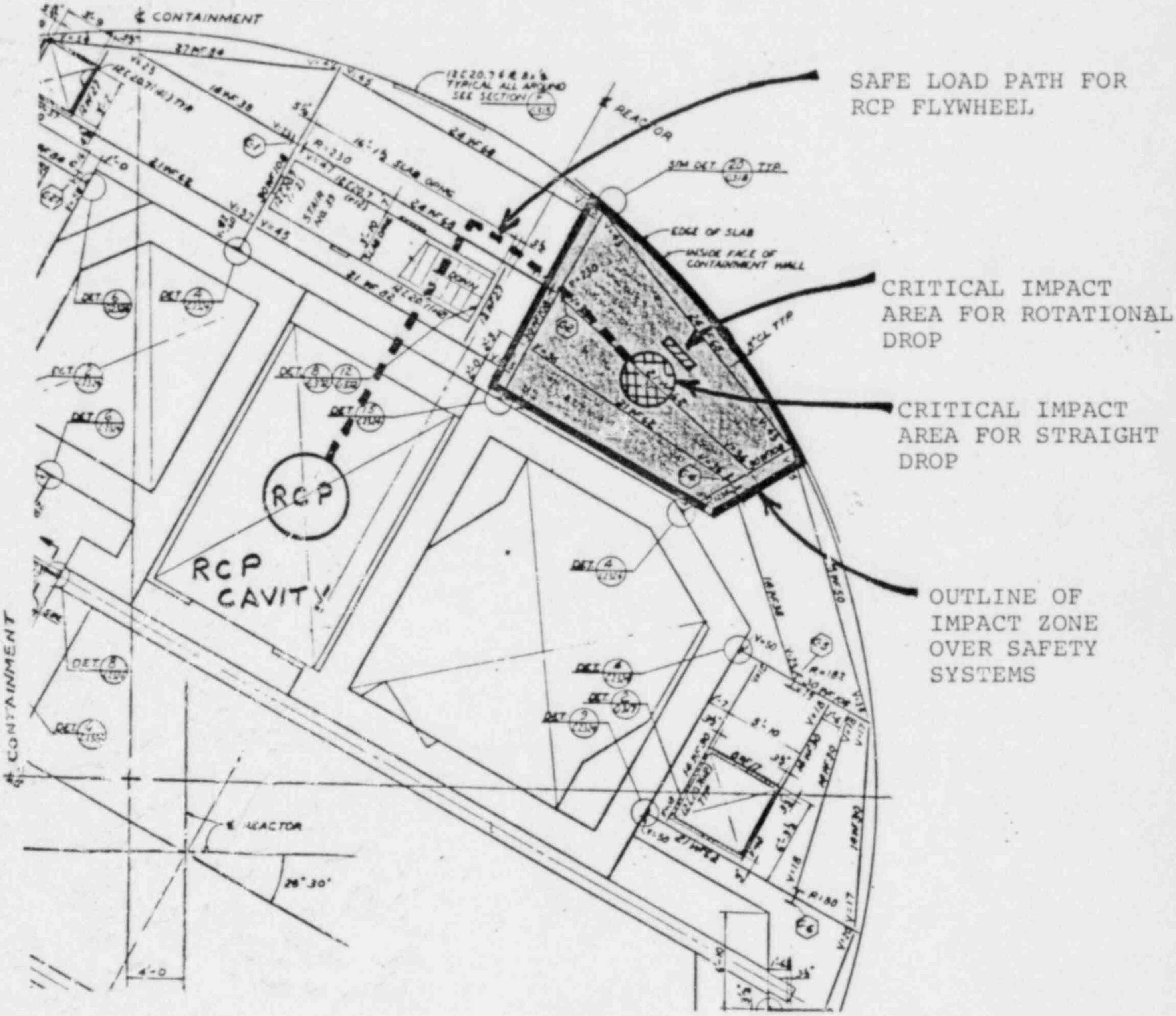
Safety systems under the floor are protected from local damage to the slab by the metal decking under the slab.

An energy balance analysis of the structural response indicates that the strain energy capacity of the slab and beams exceeds the strain energy required to prevent structural failure when subjected to a 14,000 lb. load dropped from 48 inches. While the slab panel will exhibit significant deflection, there are no safety systems near the bottom of this floor slab. Therefore the deflection of the slab is acceptable.



UNIT 2 CONTAINMENT
RCP FLYWHEEL DROP CONFIGURATION

FIGURE A-1



UNIT 2 CONTAINMENT EL. 66' SLAB
DROP LOCATIONS FOR RCP FLYWHEEL

FIGURE A-2

APPENDIX B

Load Drop Analysis of the 17,000 lb. Main Feed Pump (MFP) Motor in The Control Building.

B1. Initial Conditions and Assumptions

a. Weight of MFP Motor: 17,000 lbs.

b. Impact area of load: 64 inch equivalent diameter

c. Drop height: 9 inches

d. Drop location: center of largest slab panel, see figure B-1.

e. Assumptions regarding credit taken in the analysis for the action of impact limiters:

Impact limiters will not be used, so no credit was taken.

f. Thickness of floor slab: 8 inches

g. Assumptions regarding drag forces caused by the environment:

No credit was taken for environmental drag forces.

h. Load combination considered:

1.0D + 1.0L + 1.0I

D = dead load of slab

L = live loads on slab

I = impact loads on slab

i. Material properties of steel and concrete:

concrete: Fc': dynamic value of 1.1 x ultimate strength

Fy : dynamic value of 1.2 x yield strength = 1.2 x 40,000 = 48,000 psi

v : poisson's ratio = 0.17

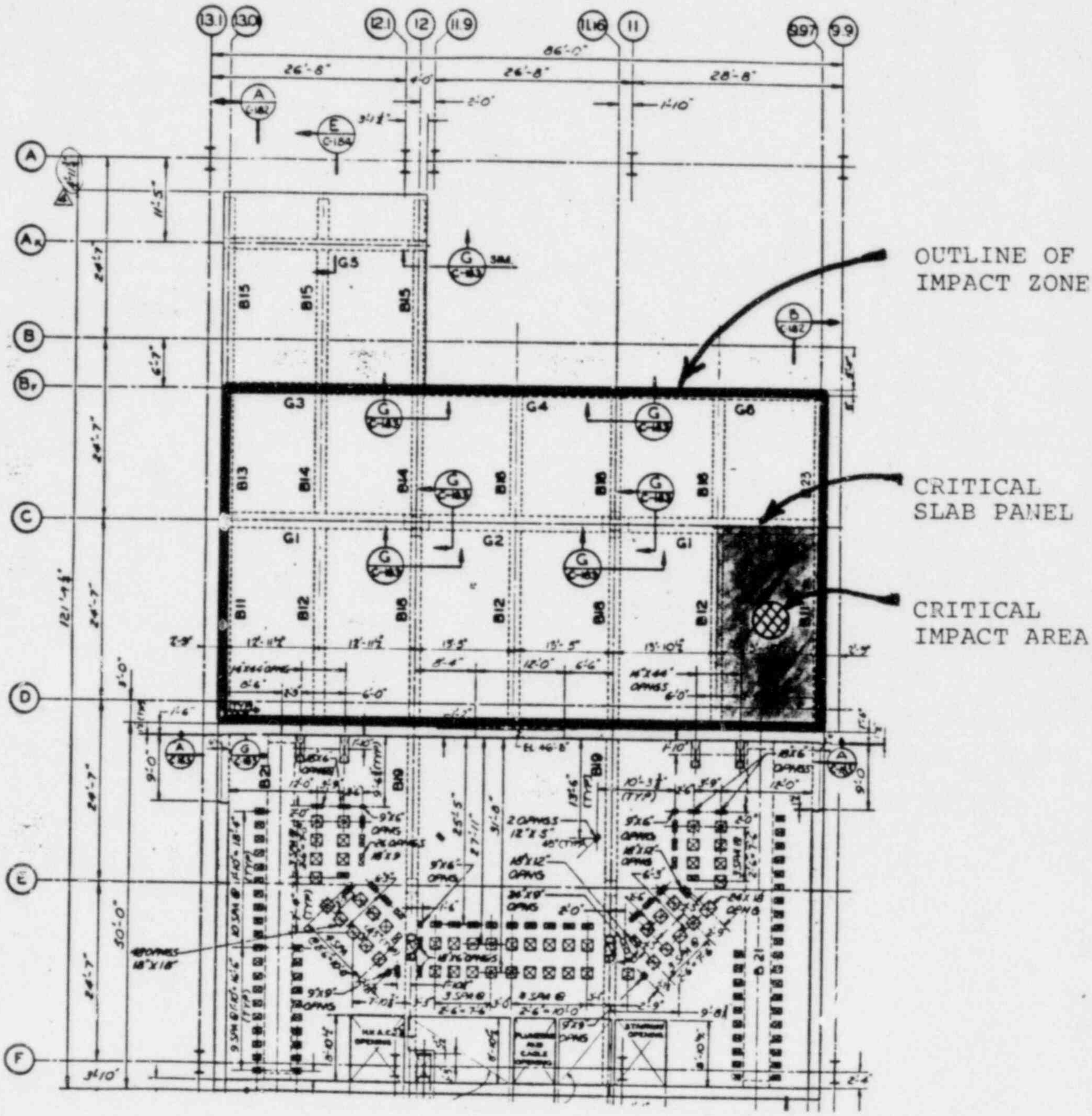
B2. Method of Analysis

(See Section A2)

B3. Conclusion

The safety systems protected by the Control Building El. 44' slab will not be impaired by the drop of the 17,000 lb. MFP motor from a height not exceeding 9 inches.

A local spalling and perforation assessment indicates that damage will not occur at drop heights of 9 inches or less. An energy balance analysis of the structural response shows that the strain energy capacity of the slab exceeds the strain energy required to prevent structural failure when subjected to a 17,000 lb. load dropped from 9 inches. While the slab panel will exhibit significant deflection, there are not safety systems directly under this floor slab. Therefore the deflection of the slab is acceptable.



CONTROL BUILDING EL. 44' SLAB
DROP LOCATION FOR MFP MOTOR

FIGURE B-1

15.3.8 REFUELING AND SPENT FUEL ASSEMBLY STORAGE

Applicability:

Applies to operating limitations during refueling operations and to operating limitations concerning the movement of heavy loads over or into the spent fuel storage pools.

Objective:

To ensure that no incident could occur during refueling operations, or during auxiliary building crane operations that would affect public health and safety.

Specifications:

A. During refueling operations:

1. The equipment hatch shall be closed and the personnel locks shall be capable of being closed. A temporary third door on the outside of the personnel lock shall be in place whenever both doors in a personnel lock are open (except for initial core loading).

2. Radiation levels in fuel handling areas, the containment and spent fuel storage pool shall be monitored continuously.

3. Core subcritical neutron flux shall be continuously monitored by at least two neutron monitors, each with continuous visual indication in the control room and one with audible indication in the containment available whenever core geometry is being changed. When core geometry is not being changed at least one neutron flux monitor shall be in service.

4. At least one residual heat removal loop shall be in operation.

5. During reactor vessel head removal and while loading and unloading fuel from the reactor, a minimum boron concentration of 1800 ppm shall be maintained in the primary coolant system.

6. Direct communication between the control room and the operating floor of the containment shall be available whenever changes in core geometry are taking place.

7. The containment vent and purge system, including the radiation monitors which initiate isolation shall be tested and verified to be operable immediately prior to refueling operations.

8. If any of the specified limiting conditions for refueling are not met, refueling of the reactor shall cease. Work shall be initiated to correct the violated conditions so that the specified limits are met, and no operations which may increase the reactivity of the core shall be made.

B. Limitations on Load Movements Over a Spent Fuel Pool*

1. One ton shall be the maximum load allowed over either the north half or south half of the spent fuel storage pool when spent fuel which has been subcritical for less than one year is stored in that half of the spent fuel pool.

2. Auxiliary building crane bridge and trolley positive acting limit switches shall be installed to prevent motion of the main crane hook over that half of the spent fuel pool which contains stored spent fuel which has been subcritical for less than one year.

3. When transporting loads exceeding one ton over a pool half which has fuel stored therein, the rigging between the transported load and the crane hook shall consist of either a single rigging device rated at six times the static and dynamic loads or dual rigging devices

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* These are interim requirements pending completion and implementation of NRC Generic Task A-36 "Control of Heavy Loads Near Spent Fuel."

each rated at three times the static and dynamic loads. The maximum permissible crane load shall be 39 tons for the main hook and six tons for the auxiliary hook.

4. Whenever possible, loads shall be carried over or placed in the half of the spent fuel pool that does not have any spent fuel assemblies stored therein.
5. Loads not exceeding 52,500 pounds may be carried over either pool half (or placed in the north half of the spent fuel pool) provided that that half of the pool contains no spent fuel assemblies.

Basis

The equipment and general procedures to be utilized during refueling are discussed in the Final Facility Description and Safety Analysis Report. Detailed instructions, the above specified precautions, and the design of the fuel handling equipment incorporating built-in interlocks and safety features, provide assurance that no incident could occur during the refueling operations that would result in a hazard to public health and safety.(1).

Whenever changes are not being made in core geometry one flux monitor is sufficient. This permits maintenance of the instrumentation. Continuous monitoring of radiation levels (A2 above) and neutron flux provides immediate indication of an unsafe condition. The residual heat pump is used to maintain a uniform boron concentration.

The shutdown margin indicated in Part A5 will keep the core subcritical, even if all control rods were withdrawn from the core. During refueling, the reactor refueling cavity is filled with approximately 275,000 gallons of borated water. The boron concentration of this water is sufficient to maintain the reactor subcritical approximately by 10% $\Delta k/k$ in the cold condition with all rods inserted, and will also maintain the core subcritical even if no control rods were inserted into the reactor.(2) Periodic checks of refueling water boron concentration insure that proper shutdown margin is maintained. Part A6 allows the control room operator to inform the manipulator operator of any impending unsafe condition detected from the main control board indicators during fuel movement.

During the refueling operation a substantial number of station personnel and perhaps some regulatory people will be in the containment. The requirements are to prevent an unsafe amount of radioactivity from escaping to the environment in the case of a refueling accident, and also to allow safe avenues of escape for the personnel inside the containment as required by the Wisconsin Department of Industry, Labor and Human Relations. To provide for these requirements, the personnel locks (both doors) are open for the normal refueling operations with a third temporary door which opens outward installed across the outside end of the personnel lock.(3) This hollow metal third door is equipped with weather stripping and an automatic door closer to minimize the exchange of inside air with the outside atmosphere under the very small differential pressures expected while in the refueling condition. Upon sounding of the containment evacuation alarm, all personnel will exit through the temporary door(s) and then all personnel lock doors shall be closed. As soon as possible, the fuel transfer gate value shall be closed to back up the 30 foot water seal to prevent escape of fission products.

The spent fuel storage pool at the Point Beach Nuclear Plant consists of a single pool with a four foot thick reinforced concrete divider wall which separates the pool into a north half and south half. The divider wall is notched to a point sixteen feet above the pool floor to allow transfer of assemblies from one half of the pool to the other.

In order to preclude the possibility of dropping a heavy load onto spent fuel assemblies stored in the spent fuel pool and causing a release or radioacivity which could affect the public health and safety, a number of precautionary measures have been incorporated into these limiting conditions for operation. No loads are permitted to be carried over freshly discharged spent fuel assemblies other than single spent fuel assemblies, handling tools and items weighing less than 2000 pounds. Limit switches are installed to prevent motion of the auxiliary building crane main hook over the half of the spent fuel pool which contains freshly discharged fuel.

When it is possible to keep all the discharged spent fuel assemblies in either the north and south half of the pool all heavy load transfers will be routed across the pool half which contains no stored fuel. When this is no longer possible, heavy loads will only be permitted to be carried over that half of the storage pool which contains spent fuel that has been subcritical for more than one year. The off site consequences of damaging such fuel assemblies are greatly reduced as the xenon and iodine fission product gases have decayed to essentially zero after one year.

In addition, the maximum load limits on the auxiliary building crane hooks have been selected such that a minimum safety factor of 10 exists between the permitted maximum load and the crane hook name plate rating times the minimum design safety factor. This results in a 39 ton limit on the 130 ton main hook and a six ton limit on the 20 ton auxiliary hook. The rigging between the auxiliary building crane hooks and the transported load must also be shown to have a safety factor of at least six over the static and dynamic loads if a single device is used and each rigging device must have a safety factor of three times the static and dynamic loads if dual straps, slings, or rigging devices are used. Dynamic loads include braking, accelerating, and slack loads.

Pending additional analysis which demonstrates that dropping a spent fuel shipping cask into the cask loading area of the north spent fuel pool will not cause an uncontrollable loss of spent fuel pool coolant or installation of the redundant crane hoisting mechanism described in Licensee's submittal of March 21, 1978, as amended; this specification (B3) precludes placing a spent fuel shipping cask into the cask loading area of the north pool when spent fuel is stored in the north half of the spent fuel pool unless the rigging devices described above are used and the weight is limited to 39 tons. Specification (B5) limits the size of the allowable load that can be placed in or carried across either the north or south half of the spent fuel pool without redundant rigging when fuel is not present in the respective half of the pool. The 52,500 pound limit is consistent with the analysis done for the potential effects upon spent fuel stored in the south spent fuel pool in the event of a postulated cask drop in the north spent fuel pool. (4)

References

(1) FSAR - Section 9.5.2

(2) FSAR - Table 3.2.1-1

(3) FSAR - Volume 5, Question 9.3

(4) FSAR - Appendix F