KEWAUNEE NUCLEAR POWER PLANT

NINE-MONTH RESPONSE TO

GENERIC LETTER 81-07

NUREG 0612

CONTROL OF HEAVY LOADS

Submitted by

Wisconsin Public Service Corporation
Green Bay, Wisconsin 54305

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1.0 Introduction

This comprises WPSC's nine-month response to Mr. D. G. Eisenhut's letter of December 23, 1980, concerning the control of heavy loads. Initially, Wisconsin Public Service Corporation (WPSC) responded to that request by letters dated June 22, 1981; August 17, 1981; and October 9, 1981. By letter dated May 17, 1982, Mr. S. A. Varga informed WPSC that these submittals provided insufficient evidence for the NRC to conclude that an evaluation with respect to the guidelines of NUREG 0612 was not required. He therefore requested WPSC to supply the information requested in Mr. Eisenhut's letter of December 22, 1980. Our six-month response was submitted on December 23, 1982. Two items were not resolved in that submittal. The additional investigation concerning these items has now been completed and is included in this response.

Section 2.0 discusses cranes with safety related equipment in their immediate vicinity in response to the information requested from Section 2.1, 2.2, 2.3, and 2.4 of Enclosure 3 of Mr. D. G. Eisenhut's letter dated December 22, 1980.

Section 3.0 provides a brief summary of this submittal.

2.0 RESPONSE TO INFORMATION REQUESTED

NRC QUESTION 2.1.3

With respect to the design and operation of heavy-load handling systems in the containment and the spent fuel pool area and those load handling systems identified in 2.1-1, above, provide your evaluation concerning compliance with the guidelines of NUREG 0612, section 5.1.1. The following specific information should be included in your reply:

e. Verification that ANSI B30.2-1976, chapter 2-2, has been invoked with respect to crane inspection, testing and maintenance. Where any exception is taken, sufficient information should be provided to demonstrate the equivalency of proposed alternatives.

RESPONSE

The turbine building crane, auxiliary building crane, and containment polar crane are tested, maintained, and inspected in a manner that satisfies Chapter 2-2 of ANSI B30.2.0-1976.

The preoperational tests conducted on the above mentioned cranes were performed in a manner that meets the intent of Chapter 2-2 (ANSI B30.2.0-1976) but not the exact letter of the guide. In Section 2-2.2.2, Rated Load Test, it is stated, prior to initial use:

"Transport the test load by means of the bridge for the full length of the runway in one direction with the trolly as close to the extreme right hand end of the crane as practical and in the other direction with the trolly as close to the extreme lefthand end of the crane as practical."

Although the bridge and trolly movement was not tested exactly as stated in the ANSI Standard during pre-operational testing, sufficient testing was performed to ensure crane operability. Pre-operational tests for the three above mentioned cranes included a lift of rated load with bridge movement, trolly movement, raising, and lowering the load, and holding with the brake. The 125% rated load test was also performed as were other various pre-operational tests. The results of the preoperational tests were acceptable to the crane manufacturer and the WPSC start-up crew.

Item g

Exceptions, if any, taken to ANSI B30.2.0-1976 with respect to operator training, qualification, and conduct.

RESPONSE

There are two sections in ANSI B30.2.0-1976, Chapter 2-3 Qualification for Operators, with which we have previously not endorsed. These are written examinations for the crane operators and the standard hand signals for controlling overhead and gantry cranes.

Starting with the 1984 crane training refresher course, examinations will be given to the participants following completion of the course.

The hand signals presently used at the Kewaunee Nuclear Power Plant for controlling overhead and gantry cranes are those signals included in the

WPSC Safety Rule Book. These hand signals are similar to the standard hand signals presented in ANSI B30.2.0-1976. Historically the WPSC hand signals have not presented ambiguities or confusion. In the future we plan to incorporate the ANSI hand signals into the WPSC Safety Rule Book.

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 cranes are the only cranes which are physically capable (i.e., ignoring interlocks, moveable mechanical stops, or operating procedures) of carrying loads into the spent fuel pool area.

Crane/Type	Capacity Main Hook	Capacity Auxiliary Hook	Location
Auxiliary Building Fuel Handling Crane	125 ton	10 ton	Auxiliary Building
Spent Fuel Pool Bridge and Hoist	3 ton	NA	Spent Fuel Pool

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

See response to NRC Question 2.2-4.

NRC QUESTION 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 cranes identified in response to NRC Questions 2.2-1 are not in strict agreement with NUREG 0612 Section 5.1.6 "Single-Failure-Proof Handling Systems."

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. This response should include the following information for each crane:

a. Which alternatives (e.g., 2, 3, or 4) from those identified in NUREG 0612, Section 5.1.2, have been selected.

RESPONSE

Spent Fuel Pool Bridge and Hoist

See six-month response to NRC Question 2.1.2 item 16, page 4.

The spent fuel pool bridge and hoist is used to lift new fuel assemblies or spent fuel assemblies, neither of which qualify as heavy loads as defined in NUREG 0612. Based upon our six-month response and the above discussion, this crane is excluded from any further discussion.

Auxiliary Building Fuel Handling Crane

See six-month response to NRC Question 2.1.2 item 20, page 10.

This crane meets Alternative 2 of Section 5.1.2 of NUREG 0612 except for item (a) where the electrical interlocks prevent movement of the overhead crane load block within five feet horizontal of the spent fuel pool. In reviewing the safe load paths shown for the heavy loads handled by this crane (see Attachment 4 of six-month response), we find that only the Irradiated Reactor Vessel Surveillance Capsule Shipping Cask is required to be moved over the spent fuel pool. The spent fuel shipping cask is not shown because the

Kewaunee Nuclear Plant has not acquired a cask and, in addition, when the final four racks are installed the spent fuel pool will have the capability to provide storage space for all spent fuel until the year approximately 2001. (With full core unload reserve.) Both of these loads are excluded from further consideration based upon the justification presented in the six-month response to NRC Question 2.1.2 item 20, page 11.

Item b

> If Alternative 2 or 3 is selected, discuss the crane motion limitation imposed by electrical interlocks or mechanical stops and indicate the circumstances, if any, under which these protective devices may be bypassed or removed. Discuss any administrative procedures invoked to ensure proper authorization of bypass or removal, and provide any related or proposed technical specification (operational and surveillance) provided to ensure the operability of such electrical interlocks or mechanical stops.

RESPONSE

See our six-month response to NRC Question 2.1.2, item 20, page 11, which indicates which heavy loads require the electrical interlocks to be placed in bypass.

Administrative Control Directive (ACD) 4.1, Operations Group Organization, Section 2, Shift Supervisor's responsibilities, states:

q. Maintains plant security and key control

This includes the key to the auxiliary building fuel handling crane bypass control switch. The crane can't be placed into bypass without unlocking the crane bypass control switch. The Shift Supervisor authorizes the removal of the crane bypasses by his issuing the crane bypass control switch key.

See Kewaunee Nuclear Plant Technical Specifications Section 3.8, Refueling Specification A.7 concerning the movement of heavy loads in the vicinity of the Spent Fuel Pool.

Item c

Where reliance is placed on crane operational limitations with respect to the time of the storage of certain quantities of spent fuel at specific post-irradiation decay times, provide present and/or proposed technical specifications and discuss administrative or physical controls provided to ensure that these assumptions remain valid.

RESPONSE

Not applicable

Item d

Where reliance is placed on the physical location of specific fuel modules at certain post-irradiation decay times, provide present and/or proposed technical specifications and discuss administrative or physical controls provided to ensure that these assumptions remain valid.

RESPONSE

Not applicable

Item e

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.

Criteria I

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

RESPONSE

Attachment 3 of the six-month response lists those heavy loads which have the potential for impacting spent fuel (only loads requiring the spent fuel pool electrical interlocks to be placed in bypass need be considered). These include:

- Spent Fuel Shipping Cask

The Kewaunee Nuclear Power Plant does not have a spent fuel shipping cask. The spent fuel storage capacity of the spent fuel pool has been increased. When the final four racks are installed the increased capacity will provide storage space for all spent fuel until approximately year 2001 - (with full core unload reserve). Accordingly, no spent fuel cask handling operations are currently planned. A detailed analysis of the consequences of an accidental drop of a spent fuel shipping cask will be made and procedures will be written prior to first use of a spent fuel cask.

- Pool Divider Gates

For relocation of the divider gates the crane is allowed to operate over the entire spent fuel pool area. Although this operation is under strict procedural control and safe load paths are defined, an accidental drop of the bottom block of the crane could cause damage to certain spent fuel elements.

The extent of the damage to spent fuel elements from an accidental drop of a pool divider gate is evaluated to be less than the damage due to a postulated turbine missile accident described in Section 14.2 of the Updated FSAR.

- Irradiated Reactor Vessel Surveillance Capsule Shipping Cask

The north spent fuel pool (1A) is reserved for loading this shipping cask; interlocks on the Auxiliary Building Crane prevent the transport of heavy loads, such as the shipping cask, over the large spent fuel pool (1B).

The auxiliary building crane has been provided with an interlock system which precludes the trolley from passing over the area of the large spent fuel pool (1B). This interlock system can include the north spent fuel pool if required. When the cask is required to be moved over the north spent fuel pool (1A), the interlocks preventing the crane from moving over the south spent fuel pool (1B) will be maintained, thereby precluding the possibility of damaging spent fuel due to a cask drop. Redundant limit switches are furnished to assure that the exclusion area is not inadvertently traversed by the malfunction of a limit switch.

An override feature is provided to administratively allow free movement of the trolley when spent fuel is not stored in the pool. The override is achieved by the use of a key lock switch. The key will be under the control of the shift supervisor.

The protection provided to minimize the effects of a dropped cask accident is presented in Table 9.5-2 of the Updated FSAR.

- Spent Fuel Storage Racks

The Kewaunee Nuclear Plant has the capability to provide storage space for all spent fuel until the year approximately 2001 (with full core unload reserve) provided the spent fuel rack modification is completed. Amendment No. 26 to Facility Operating License No. DPR-43 changed the Technical Specifications to authorize an increase in the storage capacity of the Spent Fuel Pool at the Kewaunee Nuclear Plant.

When the final racks are installed, considerations will be given, including the guidance of NUREG 0612, to ensure that safe load handling practices are followed.

Criteria II

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

RESPONSE

The following justification has been used to exclude from further consideration the criticality concerns for the spent fuel pool under dropped heavy load conditions:

1) The Potential For Criticality of PWR Fuel discussion and conclusions detailed in Section 2.2 of NUREG 0612. In addition, these conclusions can be considered conservative in that, Kewaunee uses 14 x 14 fuel, and the conclusions presented in Section 2.2 of NUREG 0612 considered 15 x 15 fuel.

2) Our spent fuel rack design. (Reference: NUS Criticality Analysis; See letter from E. W. James (WPSC) to V. Stello (NRC) dated November 14, 1977)

3) Our cold shutdown Technical Specification shutdown margin requirement (10% or $K_{eff} < .90$).

4) Spent Fuel Pool boron Technical Specification of 2100 ppm during refueling operations.

Criteria III

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

RESPONSE

A load drop analysis for an estimated 30 ton cask was performed and is reported in the Updated FSAR Section 9.5. It was concluded from this analysis that, if the cask is dropped in the small pool (north pool), the large pool (south pool) will not lose water.

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 cranes are the only cranes physically capable (i.e., taking no credit for any interlocks or operating procedures) of carrying heavy loads over the reactor vessel.

Crane/Type	Capacity Main Hook	Capacity Auxiliary Hook	Location
Manipulator Crane	3 ton	NA	Containment Building
Containment Polar Crane	230 ton	20 ton	Containment Building
Galion Crane	12.5 ton	NA	Containment Building

Item 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

Manipulator Crane

See six-month response to NRC Question 2.1.2 item 17, page 5.

The manipulator crane is used to lift new assemblies or spent fuel assemblies, neither of which qualify as heavy loads as defined in NUREG 0612. The multitude of safety features and interlocks to prevent an accidental load drop for this crane are described in Section 9.0 of the Updated FSAR. Based upon our six-month response and the above discussion, this crane is excluded from any further discussion.

Galion Crane

The galion crane use is limited to moving miscellaneous items, usually weighing less than a heavy load, within the containment building. This crane has a telescopic four section boom with an attached 15 foot self-storing jib. Normal operation of this crane is with the boom fully extended. Operating in this manner limits the loads which can be handled by this crane to those loads that weigh less than a heavy load. Therefore, based upon this operation, this crane is excluded from any further consideration.

Item 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

Containment Polar Crane

The containment polar crane identified in response to NRC Question 2.3-1 is not in strict agreement with NUREG 0612 Section 5.1-6 "Single-Failure-Proof Handling System."

Item 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:

a. 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 specification concerning the bypassing of such interlocks.

RESPONSE

Containment Polar Crane

No electrical interlocks or mechanical stops are installed on the containment polar crane.

Item 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

Reactor Building Polar Crane

See Drawing No. CHL-1 in six-month response for exact location.

The reactor building polar crane is used to handle heavy loads in the containment building. Heavy loads handled by this crane are: reactor vessel and pressurizer missile shield, reactor vessel head, upper and lower internals, reactor coolant pumps (including motor and flywheel), inservice inspection tool, and reactor vessel studs (in handling box only); see Attachment 3 in six-month response for additional information concerning these loads. Except for the pressurizer missile shield, the movement of these heavy loads with the reactor building polar crane is done only when the reactor coolant system is in the cold shutdown or refueling shutdown condition. In addition, during refueling operations, containment integrity other than the fuel transfer tube must be maintained for activities affecting core reactivity.

See Attachment 4 in the six-month response for the sketches which identify the specific pathway for the movement of the heavy loads. These specific pathways (safe load paths) were developed with the following considerations:

a. Minimize the potential for a heavy load drop to impact irradiated fuel or to impact safe shutdown equipment.

b. Shortest distance between the component and its designated lay down area.

c. Limits imposed upon crane travel due to the design of the crane and maximum travel of the crane.

d. Reactor coolant system conditions required prior to the movement of specific components.

e. Personnel safety.

Written procedures will be generated identifying the applicable requirements from NUREG 0612, Section 5.1.1(2) for the loads identified in these sketches. Written procedures will be generated if deviations from approved specific pathways (see sketches in six-month response, Attachment 4) are necessary.

The steam generators or the residual heat removal system is the safe shutdown equipment that is required for continued decay heat removal. Due to the arrangement of both steam generators and their associated feedwater (auxiliary and main feedwater) and steam piping inside containment, no single load, if dropped, can remove both trains from service at the same time.

A redundant residual heat removal supply piping is provided in the reactor building to pump water into the reactor vessel and prevent a boil off. The physical separation between the redundant supply piping in our judgment is adequate to preclude damage to both trains due to a single heavy load drop accident. One section of the residual heat removal return piping is common to both trains; however, three floor elevations would protect it from a load drop, and there are isolation valves which could be closed to prevent any loss of reactor coolant.

In the event of a failure resulting in the unavailability of the residual heat removal system, the volume of the reactor coolant system, even at its minimum, provides a heat sink for the relatively low heat that is generated by the core. This allows a sufficient amount of time to provide an alternate means of heat removal. Because of the amount of time available for action, it is more appropriate to address the potential for a loss of decay heat removal through a procedure. A procedure addressing this is in existence at the Kewaunee Plant, as indicated in our response to IE Bulletin 80-12, dated June 20, 1980. We feel that our procedures adequately address the concerns of assuring decay heat removal and no further action is necessary in this regard.

During refueling operations current technical specifications require that the following conditions be met:

T.S.3.8 Refueling

a. During refueling operations

1. The equipment hatch and at least one door in each personnel air lock shall be closed. In addition, at least one isolation valve shall be operable or locked closed in each line, other than the fuel transfer tube, which penetrates the containment and which provides a direct path from containment atmosphere to the outside.

2. Radiation levels in fuel handling areas, the containment and the spent fuel storage pool shall be monitored continuously. High activity levels shall be cause for closing the normal vent path.

5. During reactor vessel head removal and while loading and unloading fuel from the reactor, the minimum boron concentration of 2100 ppm shall be maintained in the Reactor Coolant System, and verified by sampling daily.

8. The containment ventilation and purge system, including the radiation monitors which initiate containment ventilation isolation, shall be tested and verified to be operable immediately prior to a refueling operation.

In addition, when the reactor vessel head is being removed or replaced, the equipment hatch and at least one door in each personnel airlock shall be closed.

Item 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.

Criteria I

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

RESPONSE

During refueling operations the containment air quality is maintained by ventilation. The containment vessel is serviced by the containment purge and vent system, which provides fresh tempered air at a rate that will result in 1 1/2 air changes per hour (33000 CFM). Supply air is provided by a fan through a fresh air filter and heating coils. Exhaust air is drawn through filter assemblies which will remove 99.97% of all particulate matter of 0.3 microns and larger. If there are high radiation levels in the containment building, the containment purge and ventilation system can be aligned to a charcoal absorber filter which will remove 99.9% of elemental iodine and 95.0% of methyl iodide at 70% relative humidity.

Before the reactor vessel head and upper internals are lifted above the reactor vessel, the equipment hatch and at least one door in each personnel airlock shall be closed. In addition, when moving the upper internals with fuel in the reactor, we conservatively apply the same operating limitations as we do for the refueling operation itself. Therefore, should there be high radiation levels resulting from fuel damaged from dropping the reactor vessel head, the radiation would be contained within containment. The containment purge and ventilation system automatically isolates from the containment vessel on a high-radiation signal by closing the two valves in both the supply and exhaust ducts. The valves in each duct are located adjacent to the containment vessel, one inside and one outside.

Criteria II

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

RESPONSE

The neutronics analysis for the PWR Core is evaluated on the basis of the NUREG 0612 criteria (page A-5) and the value given in Table 3.2.1 of the Updated FSAR.

K_{eff} for the uncrushed core in 2100 ppm boron concentration is less than 0.90. Then using the estimated .05 maximum reactivity insertion due to crushing from NUREG 0612, the maximum achievable K_{eff} is still less than 0.95, which meets the requirement of NUREG 0612.

Criteria III

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

RESPONSE

An accidental drop of the reactor vessel head and the lifting gear onto the vessel during a heavy load handling operation was postulated with the following assumptions.

The scope of work for the polar crane was limited to investigation of the reactor vessel safety injection system's ability to supply cooling water to the reactor after a reactor vessel head removal/replacement related accident. The reactor vessel head is assumed to fall from the high hook position of 716'-6" straight down to the reactor vessel mating surface at elevation 623'-7".

The total weight lifted by the crane includes weights of vessel head, crane block, lifting gear, and control rod drive mechanism (CRDM). It is estimated to be 175 kips. During the plastic impact, the potential energy of the load will be absorbed by the steel columns supporting the reactor vessel by elasto-plastic deformations. The main coolant pipes are assumed to crack due to the drop. The coolant will be collected by the drainage system. However, the safety injection lines will remain intact and within the elastic range. The maximum calculated vertical displacement of the line is 2.79 inches. The maximum calculated bending stress of the safety injection lines is 21.3 ksi. The safety injection system will continue to pump water into the reactor vessel and prevent a boil off.

The calculations take no credit for dissipation of energy in distorting and/or destroying CRDM system and main coolant pipes or the concrete encasement around the steel columns supporting the vessel.

NRC QUESTION 2.4

Item 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

See response to NRC Question 2.4 item 2.a below.

Item 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:

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.

RESPONSE

See Attachment 2 in the six-month response which lists the overhead heavy load handling systems in the vicinity of safe shutdown equipment.

See six-month response to NRC Question 2.1.2 which provides our justification for excluding overhead-load-handling systems from further consideration. Our justification for exclusion is based upon there being sufficient physical separation from any load impact point and any safety-related component such that no heavy load drop can result in damage which would render inoperable any system or component required for plant shutdown or decay heat removal.

See Attachment 3 in our six-month response for those loads that meet the heavy loads criteria.

Attachment 4 in the six-month response identifies the physical location for the load handling system in the plant. (Drawing CHL-1, CHL-2 and CHL-3)

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 considerations 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).

RESPONSE

Our six-month response to NRC Question 2.1.2 provides a detailed discussion of each load handling system in the vicinity of spent fuel or safe shutdown equipment and justification for its exclusion from further consideration.

(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

See our six-month response to NRC Question 2.1.2 item 8, page 3 and item 20, page 10 which:

- identifies what load handling systems will be modified
- provides a discussion concerning the bypassing of electrical interlocks or removal of mechanical stops,
- describes the areas where crane travel will be prohibited.

(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

See our six-month response to NRC Question 2.1.2 which discusses load-handling-systems and their justification for exclusion from further consideration.

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

RESPONSE

None

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).

RESPONSE

None

3.0 Summary

A response to specific items of D. G. Eisenhut's December 23, 1980, letter and NUREG 0612 "Control of Heavy Loads" has been provided. This response addresses those items not previously answered in our December 23, 1982, six-month response. This report concerns itself with overhead handling systems operating in the vicinity of fuel storage pools, operating in the containment, or operating in plant areas containing equipment required for reactor shutdown, core decay heat removal, or spent fuel pool cooling.