CONTROL OF HEAVY LOADS FOR SAN ONOFRE NUCLEAR GENERATING STATION UNIT I

FINAL REPORT

Submitted to:
Southern California Edison Company
P.O. Box 800
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1.0 INTRODUCTION

The NRC's letter of December 22, 1980, requested a review of the controls for handling heavy loads at San Onofre Nuclear Generating Station, Unit 1 (SONGS 1), the implementation of certain recommendations regarding these controls, and the submittal of information to demonstrate that the recommendations have been implemented.

A report was submitted in April 1982 which addressed the information required in Section 2.1 of Enclosure 3 of the December 22, 1980 letter. This report is responsive to the information required in Sections 2.2, 2.3 and 2.4 of Enclosure 3 of the December 22, 1980 letter. In addition, in Section 4.0 of this report, an evaluation of lift rig designs is provided. This had been identified in the April 1982 report as requiring further evaluation.

2.0 IDENTIFIED HANDLING SYSTEMS

The heavy load handling systems and loads required to be addressed in this report were identified in our submittal of April 1982. The handling systems are:

Inside Containment

Reactor Service Crane

Outside Containment

Turbine Gantry Crane

Spent Fuel Bridge Crane

3.0 RESPONSES TO REQUESTS FOR INFORMATION IN SECTIONS 2.2, 2.3, AND 2.4 OF ENCLOSURE 3 OF NRC DECEMBER 22, 1981 LETTER

ITEM 2.2 SPECIFIC REQUIREMENTS FOR OVERHEAD HANDLING SYSTEMS OPERATING IN THE VICINITY OF FUEL STORAGE POOL

NUREG 0612, Section 5.1.2, provides guidelines concerning the design and operation of load-handling systems in the vicinity of stored, spent fuel. 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 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.

RESPONSE: As described in our April 1982 report, no heavy loads are lifted over spent fuel in the spent fuel pool. Lifts of the spent fuel cask by the turbine gantry crane are restricted to an area away from the spent fuel, which precludes direct or indirect interactions in the event of a load drop. As shown in Figure I, the spent fuel pit bridge crane is used to withdraw and replace the gate which isolates the fuel transfer system from the remainder of the spent fuel pit. In this case Technical Specification 3.8.B.I precludes lifting of the gate over the spent fuel pool. This restriction on operation makes most of the criteria of NUREG 0612, Section 5.1.2 not applicable to SONGS I. The only potential mechanism for interaction with spent fuel would be if a failure of the spent fuel crane/lifting sling occurred when the gate had been lifted out of its slot. In the unlikely event that the gate fell into the pool, analyses have been performed for an impact of the gate with the pool bottom slab. A structural evaluation was performed to determine whether a drop of the gate could result in a breach of the slab. The results of the analysis indicate that overall structural integrity of the pool bottom slab would be maintained and the leak-tight integrity of the pool would not be breached. Additionally, we examined the potential for the gate to be dropped such that it could enter the pool and strike spent fuel assemblies. The spent fuel bridge crane which lifts the gate is constructed of a steel I-beam network which extends to the maximum height of the hoist. The crane is designed such that, when in position to lift the gate, the hoist and its load are

separated from the pool area containing storage racks by this network of structural steel I-beams. Therefore, the bridge crane, itself, would prevent the gate from entering the pool in the unlikely event the gate is dropped after it clears the pool. Therefore, we have concluded that NUREG 0612 criteria are satisfied.

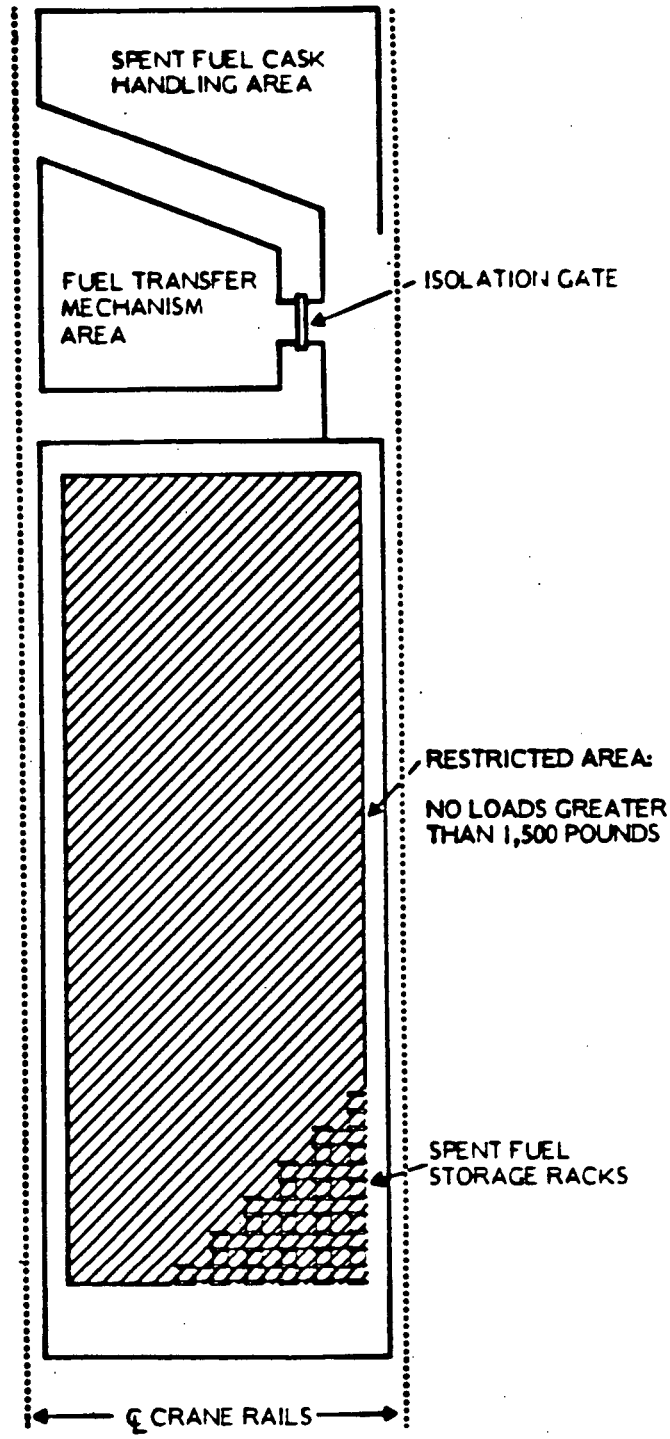

FIGURE I

SPENT FUEL PIT BRIDGE CRANE
SERVICE AREA

ITEM 2.3 SPECIFIC REQUIREMENTS OF OVERHEAD HANDLING SYSTEMS OPERATING IN THE CONTAINMENT

NUREG 0612, Section 5.1.3, provides guidelines concerning the design and operation of load-handling systems in the vicinity of the reactor core. Information provided in response to this section should be sufficient to 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 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 2.3-1. 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 only handling system within containment physically capable of carrying loads over the reactor vessel is the reactor service crane. The crane was designed by Harnischfeger Corporation and possesses a 110-ton hoist and a 20-ton auxiliary hoist.

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: There are no other cranes inside the containment capable of lifting heavy loads as defined in NUREG 0612.

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: Two aspects of load handling operations involving the reactor service crane were evaluated on the basis of design features that resulted in the likelihood of a load drop being extremely small. These were the potential drop of the main hoist load block and hook in an unloaded condition and potential

drops of the reactor vessel head and upper internals. Each of these cases is discussed below.

Load Block: The reactor service crane was evaluated to industry standards CMAA 70-1975 and ANSI B30.2-1976 as required by NUREG 0612. This evaluation, provided to you in our 6-month report, indicates that the crane complies with the intent of these standards and possesses demonstrated margins to failure. 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, upper internals, missile shield and auxiliary shield. 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 negligible loads 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 crane failure modes that could result in dropping of the main hook and load block are:

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 main hoist of the reactor service crane has a gear-driven limit switch that is connected to and driven by the hoisting drum. These switches are provided to prevent the cables from being completely unwound from the drums and to prevent overtravel of the hooks in the lifting and lowering direction. In addition, a block-actuated limit switch is provided for each hook to prevent overtravel of the hook in the upward direction and to prevent a two-blocking event should the gear-operated limit switch be out of adjustment or fail to operate.

The main hoist is also equipped with redundant electric motor brakes. The brakes are shoe-type electric brakes which serve as parking and service brakes. In addition, a magnetorque motor is supplied which is an electric load brake used for speed control. This load brake will prevent a free-field fall of the load in the event that the hoisting motor fails and the brakes fail to engage when the operator pushes the stop hoist control.

With the provisions described above, the limit switches reduce the likelihood for "two blocking" and the holding brakes 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.

Head and Internals Lifts: Lifts of the reactor vessel head and reactor internals by the reactor service crane have been analyzed on a probabilistic basis. The study identified and quantitatively analyzed, using fault tree methods, the potential mechanisms for drops of the reactor vessel head. The study was performed in accordance with the following steps:

- Description of the reactor service crane system and associated testing, maintenance, inspection, training and lift procedures regarding removal and installation of the reactor head and upper internals during refueling
- Event identification and fault tree construction—determination of all the ways the reactor service crane system could fail
  - (1) Structural failure while subjected to normal load conditions
  - (2) Structural failure due to excessive load
    - i) Two-blocking event
    - ii) Load hangup event
  - (3) Overspeed event—loss of hoisting or lowering capability coupled with loss of brakes

- Qualitative analysis--find minimal cut sets and establish all single failure events leading to system failure
- Probabilistic analysis
  - (1) Find sources of data and determine applicability to SONGS I operations
  - (2) Compute probability of the Top Event
  - (3) Probabilistically rank basic events and min cut sets (i.e., conduct a sensitivity analysis)
- Conclusions, recommendations and results.

The Top Event for the analysis was defined in terms of two individual events:

- Drop during removal
- Drop during installation

These two events will generate the same load drop scenarios with two exceptions:

- During installation a two-blocking event would most likely occur above the reactor head laydown area. Hence, this scenario is not considered during installation.
- A reactor head or upper internals load hangup event could only occur during removal. Again, this scenario is not considered during installation.

During removal operations, the head and upper internals are initially lifted several inches and carefully inspected. The load remains suspended in that position for approximately 15 minutes before further lifting. To account for these operations, the analysis was segregated into two types of potential load drops:

- Drop during initial lift
- Drop after initial lift.

A drop during initial lift could result from a load hangup event or lifting system structural failure. This type of drop would occur at a height of no more than a few inches above the flange and is of no safety significance. The probability of a structural failure or load hangup following this initial lift is significantly reduced.

Tables 3-1 and 3-2 summarize the probabilities of various load drop scenarios for the lifts of the vessel head and upper internals. The results indicate that the dominant failure mechanisms for both lifts are those related to the occurrence of a load drop during the initial lift and hold. Because the initial lift height is limited to less than two feet, the consequences of a load drop at this stage of a lift are considered minimal. The mean probabilities of failures leading to dropping of the loads subsequent to the initial lift phases are on the order of 10^{-6} per lift or less. Accidents of this low of probability are acceptable without further analyses. Overspeed events were found to be of negligible importance. The specific results for each load are discussed below.

REACTOR VESSEL HEAD

The drop of the head during the initial lift and hold dominates the failure mode for this load. A load hangup event is the major contributor to this scenario.

Failures subsequent to the initial lift are primarily due to structural failures. The mean probability of failure is on the order of 10^{-6} per lift and is sufficiently small that specific analyses of the consequences of a load drop are not necessary.

TABLE 3-1

MEAN PROBABILITY OF VARIOUS REACTOR HEAD DROP SCENARIOS PER LIFT (BEST ESTIMATE CASE)

Load Drop Scenario	Mean Probability
o Drop During Initial Lift	1.36×10^{-4}
- Load Hangup Event	1.0×10^{-4}
- Structural Failure During Initial Lift	3.7×10^{-5}
o Drop After Initial Lift	3.2×10^{-6}
- Structural Failure After Initial Lift	2.7×10^{-6}
- Two-Blocking Event	6.1×10^{-7}
- Overspeed Event	2.2×10^{-9}

TABLE 3-2

MEAN PROBABILITY OF VARIOUS UPPER INTERNALS DROP SCENARIOS PER LIFT (BEST ESTIMATE CASE)

Load Drop Scenario	Mean Probability
o Drop During Initial Lift	4.7×10^{-5}
- Structural Failure During Initial Lift	3.7×10^{-5}
- Load Hangup Event	1.0×10^{-5}
o Drop After Initial Lift	6.1×10^{-7}
- Two-Blocking Event	6.1×10^{-7}
- Overspeed Event	2.2×10^{-9}
- Structural Failure After Initial Lift	NIL

UPPER INTERNALS

The drop of the upper internals during the initial lift and hold is also the dominant failure mode for this lift. Because a load cell is used during this lift, the failure probability is on the order of 10^{-5} per lift.

Failures subsequent to the initial lift are primarily due to two-blocking events. The mean probability of failure is on the order of 10^{-6} to 10^{-7} per lift and is sufficiently small that specific analyses of the consequences of a load drop are not necessary.

MISCELLANEOUS MAINTENANCE OPERATIONS

Certain lifts might be required in the future, which were not specifically identified and evaluated in this report. Such heavy load handling operations could be acceptable on a low probability basis if either the frequency were low (e.g. less than once per year) or if there are sufficient design safety factors to meet the criteria of NUREG-0554 and Appendix C to NUREG-0612. The probability evaluations discussed above could be applicable to these miscellaneous operations if the lifts were only infrequently performed, and if similar lifting procedures and reliable lifting devices were used. For loads which result in crane and lifting device safety factors double that required by NUREG-0612 (e.g. less than one half the crane and lifting device capacity), the conservative design standards would be sufficient, when coupled with the reliability analysis of the Reactor Service Crane, to meet the criteria of NUREG-0554 and Appendix C to NUREG-0612.

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

ITEM 2.3-4-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: No reliance is placed upon electrical interlocks or mechanical stops.

ITEM 2.3-4-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: The only such considerations are the physical protection provided by the reactor vessel head when in place or the absence of fuel in the reactor vessel. Loads lifted only when the reactor vessel head is in place and the unit is in a shutdown mode or the reactor is defueled were not considered as loads that could potentially interact with the core. These are: the CEDM ventilation duct, the core support barrel, the auxiliary and missile shields, the ISI tool, the sand tanks, and the head stud tensioners and rack.

ITEM 2.3-4-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: No analyses were necessary.

ITEM 2.4 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

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 load drop which might prevent safe reactor shutdown or prohibit continued decay heat removal is extremely small, or damage to such equipment from load drops 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 2.4-1. 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., 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: For the same reasons discussed in Section 2.3-3, a drop of the turbine gantry crane load block and hook when unloaded was not considered credible.

ITEM 2.4-2. For any cranes identified in 2.1-1 which are not designated as single-failure-proof in 2.4-1, a comprehensive hazard evaluation should be provided which includes the following information:

ITEM 2.4-2-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: Two heavy load handling systems, the Reactor Service Crane and the Turbine Gantry Crane, have the potential of carrying heavy loads over safe shutdown equipment. The reactor service crane is used during cold shutdown or refueling mode when the RHR system (inside of containment) is in operation. Nonroutine in-containment lifts for purposes of maintenance in operating modes other than 5 or 6 will be considered on a case by case basis, and were not reviewed in this evaluation. The requested information for potential interactions of the Reactor Service Crane and Turbine Gantry Crane is presented in Tables 3-3 through 3-6. Layout drawings showing the location of equipment are provided in Figure 2 for the Reactor Service Crane, Figure 3 for the Turbine Gantry Crane and in the April 1982 report.

As discussed in the April 1982 report, the Turbine Gantry Crane is used for lifts related to major turbine generator maintenance, new fuel shipping container,

spent fuel cask and other miscellaneous loads. Special procedures and load paths are designated for handling activities involving the spent fuel cask which preclude interaction with safe shutdown equipment. When heavy loads are lifted during turbine generator maintenance, the reactor is in cold shutdown of necessity, and the residual heat removal system, located inside of containment, is not susceptible to interaction with Turbine Gantry Crane load drops. Therefore, our evaluation was restricted to the North Turbine Deck extension as described in the April 1982 report and shown in Figure 4. Instead of evaluating specific interactions of potential heavy loads with safe shutdown equipment, an envelope of loads was defined which could be dropped without penetrating the north turbine deck extension or causing scabbing of concrete which could interact with safety-related equipment underneath the deck. These results are presented in response to Items 2.4-2-b(3) and 2.4-2-d.

CRANE: REACTOR SERVICE CRANE

TABLE 3-3

LOCATION	CONTAINMENT SPHERE		
IMPACT AREA / LOADS	AREA AROUND REACTOR COOLANT PUMP A		
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
Reactor Coolant Pump Motor	14'	RHR Heat Exchanger	See discussion under Item 2.4-2-b(3)
Reactor Coolant Pump Motor Removal Hatch	42'	RCS, RHR Heat Exchanger	See discussion under Item 2.4-2-d

CRANE: REACTOR SERVICE CRANE

TABLE 3-4

LOCATION	CONTAINMENT SPHERE		
IMPACT AREA / LOADS	AREA AROUND REACTOR COOLANT PUMP C		
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
Reactor Coolant Pump Motor	14'	RHR Piping and Equipment	See discussion under Item 2.4-2-b(3)
Reactor Coolant Pump Motor Removal Hatch	42'	RCS, RHR equipment	See discussion under Item 2.4-2-d

CRANE: REACTOR SERVICE CRANE

TABLE 3-5

LOCATION	CONTAINMENT SPHERE		
IMPACT AREA / LOADS	REFUELING DECK		
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
Reactor Coolant Pump Motor	42'	RHR equipment	See discussion under Item 2.4-2-d
Reactor Vessel Head	42'	RHR equipment	See discussion under Item 2.4-2-d
Reactor Internals (Impact area is bottom slab of refueling cavity)	42'	RHR equipment	See discussion under Item 2.4-2-d
Auxiliary Shield	42'	RCS	See discussion under Item 2.4-2-b(3)
Missile Shield	42'	RCS	See discussion under Item 2.4-2-b(3)

CRANE: TURBINE GANTRY CRANE

TABLE 3-6

LOCATION	TURBINE DECK		
IMPACT AREA / LOADS	NORTH TURBINE DECK		
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
Hot Tool Box	42'	Safety related piping and cables	See discussion under Item 2.4-2-b(3) and Item 2.4-2d
New Fuel Container	42'	Safety related piping and cables	See discussion under Item 2.4-2-b(3) and Item 2.4-2d
Misc. High Density Equipment	42'	Safety related piping and cables	See discussion under Item 2.4-2-b(3) and Item 2.4-2d
Misc. Low Density Equipment	42'	Safety related piping and cables	See discussion under Item 2.4-2-b(3) and Item 2.4-2d

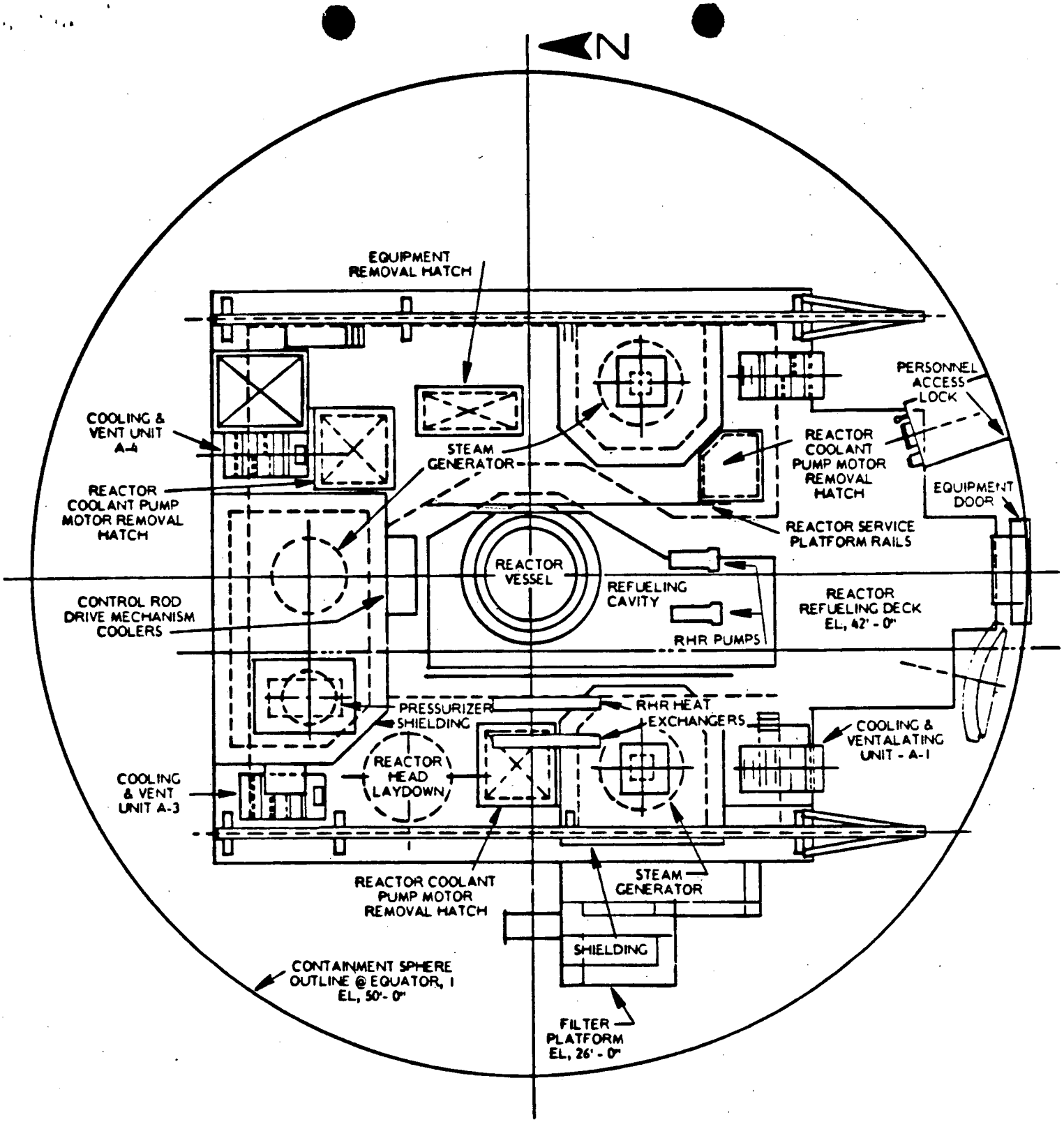

FIGURE 2

CONTAINMENT GENERAL ARRANGEMENT AT REFUELING DECK
(Elevation 42 feet)

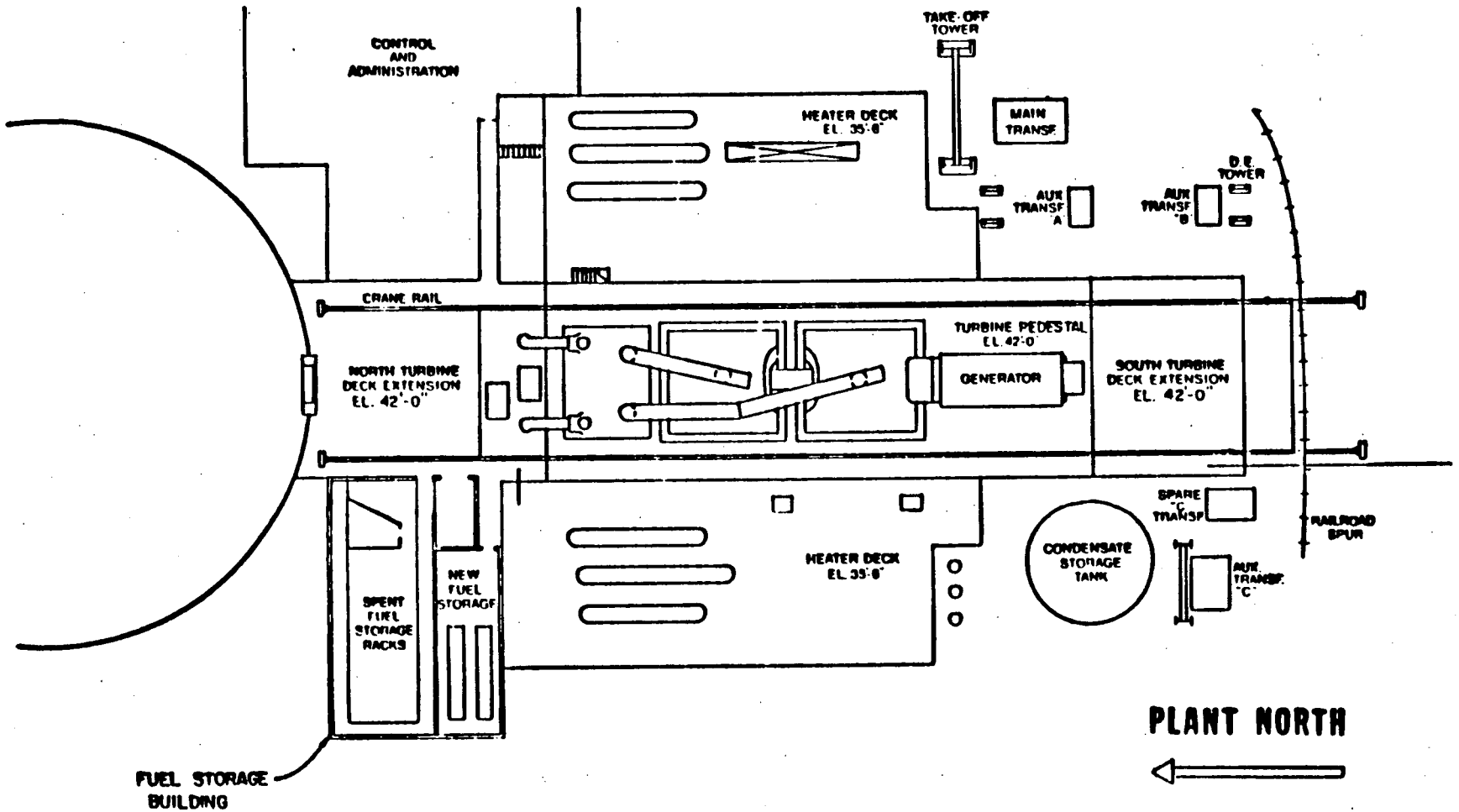

FIGURE 3

TURBINE GANTRY CRANE SERVICE AREA

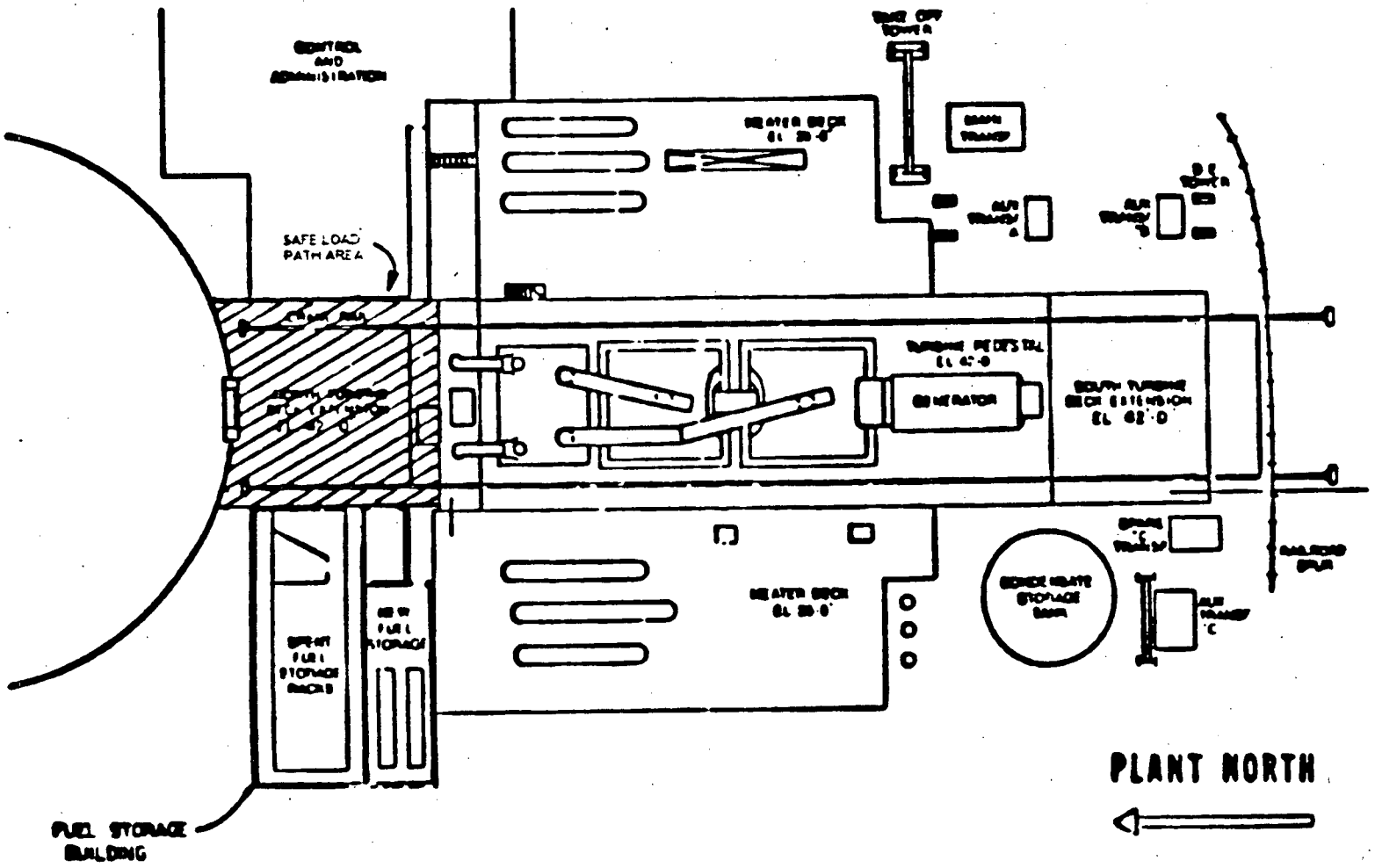

FIGURE 4

TURBINE GANTRY CRANE
SAFELOAD AREA

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ITEM 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. Elimination on the basis of the aforementioned considerations should be supplemented per items 2.4-2-b(1), (2), and (3) as follows.

RESPONSE: Not applicable to SONGS I.

ITEM 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: No load/target combinations were eliminated because of separation and redundancy of safety-related equipment.

ITEM 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: Neither mechanical stops or electrical interlocks are to be provided.

ITEM 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: Administrative controls will be applied to restrict handling of heavy loads in the following cases:

Reactor Service Crane

As discussed in response to Item 2.4-2-d, load lift heights for heavy loads carried over the refueling deck should be limited to assure that in the event of a load drop no adverse overall structural effects would occur and no scabbing of concrete would occur which could impact safety-related equipment under the refueling deck. Generally, the minimum required lift height is that necessary to

clear obstructions in the load path, but the lift heights will not exceed those maximum heights provided in Table 3-7 except as discussed below.

Certain lifts of the reactor coolant pump motors or other major loads have the potential to damage reactor coolant system piping or RHR components for drops down the removal hatches or lifts above the refueling deck for purposes of clearing the missile shield and/or the fuel manipulator machine. NUREG-0612 (Section 5.1, Criterion IV) allows such lifts provided the postulated drop would not result in loss of required safe shutdown functions. Therefore, prior to these handling operations, procedures will be in place which require an operable safe shutdown path that is independent of potential equipment damage that could result from the load being dropped.

In the case of the reactor vessel head, the lift should be limited to the minimum height necesssary to set the head down on its stand. A drop in this case, could result in spalling of concrete from the refueling deck that could interact with the RHR equipment. The drop of the reactor internals into the refueling cavity was selected as the bounding load since other loads (e.g. miscellaneous tools, the ISI tool and the stud rack) normally carried across and/or into the refueling cavity are smaller. Therefore, the consequences of postulated drops of these smaller miscellaneous loads are within the limits of NUREG-0612 criteria, since the drop of the internals was found to comply with those evaluation criteria. The drop of the reactor internals into the reactor cavity could cause spalling that might interact with RHR equipment. However, since the reactor would be in refueling mode with the cavity flooded during either of these lifts, residual heat could be removed by alternative means if the RHR system was not available (i.e. the refueling water pumps through the recirculation heat exchanger to the cold leg injection flow path).

The two heaviest loads lifted by the Reactor Service Crane involve the auxiliary shield and missile shield.

The auxiliary shield is a large reinforced concrete slab about 21 feet wide, 27 feet long and 18 inches thick. The shield slides down a slot to separate the reactor vessel cavity from the refueling canal during normal operation and is removed prior to refueling to allow removal and storage of the upper internals and fuel assemblies under water. The auxiliary shield is removed by the reactor

service crane and stored on top of steam generator "B" enclosure (about 25 feet above the Refueling Deck). The lift is conducted in two steps. First, the auxiliary shield is lifted in a vertical position by a two part sling attached to lifting lugs in the top edge of the slab and carried over the missile shield toward Steam Generator "B". The bottom edge of the auxiliary shield is then positioned onto special missile shield lugs which facilitate rotating the auxiliary shield from a vertical to horizontal position on top of the missile shield. The auxiliary shield is then lifted in a horizontal position by a four part sling to the top of the steam generator enclosure shield. This two step lift is required since the crane cannot lift high enough to reach the top of the steam generator enclosure shield with the auxiliary shield in the vertical position. The auxiliary shield is reinstalled in the reverse sequence of operations.

The missile shield is also a large reinforced concrete slab, about 23 feet wide, 27 feet long and 2 feet thick which spans the top of the reactor cavity during normal operations. The shield is removed prior to refueling or other maintenance activities requiring access to the reactor vessel, control rod drive mechanisms or other equipment within the reactor cavity. The missile shield is stored on top of Steam Generator "B" enclosure shield (on top of the auxiliary shield if it too has been removed). It is lifted in its normal horizontal position by a four part sling attached to lifting lugs in the missile shield. In addition to the missile shield being the heaviest load lifted by the Reactor Service Crane, this shield is also the highest load lifted. As such these loads are carefully controlled and a minimum of 5 individuals are required to monitor, control and conduct the lifts.

While the control of these heavy load lifts has been sufficient to prevent their accidental drop during over 14 years of operation at SONGS 1, additional procedural requirements and design reviews identified in our February 5, 1982 letter and six month report (transmitted by our letter dated April 1, 1982) have increased our confidence that these loads will be safely handled. Therefore, the new administrative controls and the existing safety record at SONGS 1 provide sufficient assurance of safe handling when the limited frequency of these handling operations is considered. For example, the probabilistic evaluation of the reactor vessel head lift (see response to Item 2.3-3) demonstrated how unlikely a load drop is, given the design of the Reactor Service Crane and the administrative controls on lifting heavy loads.

Turbine Gantry Crane

Heavy loads carried over the North Turbine deck extension should be limited to the safe load paths, weight limits and height restrictions discussed in response to Item 2.4-2-d. In general, load lift heights during any mode of operation should be limited to assure that in the event of a load drop no scabbing of concrete would occur which could impact safety related equipment under the deck. Generally, the minimum required lift height is that necessary to clear obstructions in the load path, but the lift heights will not exceed those maximum heights provided in Table 3-7. The safe load paths for large loads are shown in Figure 5. These paths should be used to carry any load over 7,000 lbs. up to the maximum load in this area, which is the 20,000 lbs hot tool box. High density heavy loads are defined as loads which are 2½ feet in diameter or less and weigh 2,000 to 5,000 lbs. Lower density heavy loads are defined as loads which are 5 feet in diameter or greater and weigh 2,000 to 10,000 lbs. Based on the evaluated conditions above, both high and low density loads can be safely carried anywhere without producing scabbing underneath the deck.

ITEM 2.4-2-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: Not applicable.

ITEM 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).

RESPONSE: Certain crane/load combinations have been evaluated to assure operability of minimum safe shutdown equipment by structural analysis.

Each of the heavy loads carried by the reactor service crane and turbine gantry crane have been evaluated to identify loads which control local response (e.g., penetration, scabbing, spalling, performation, etc.); loads that control overall

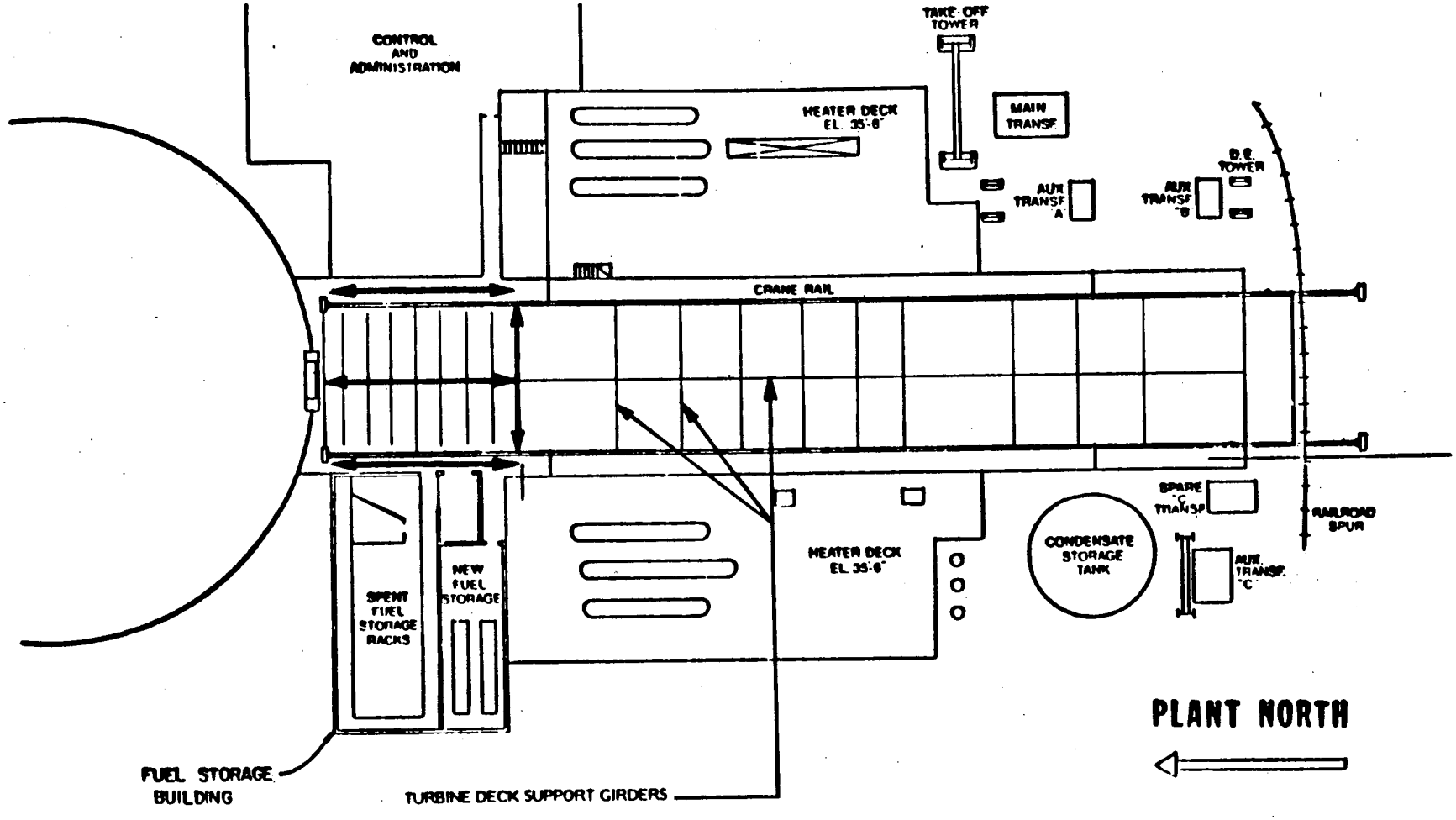

FIGURE 5

TURBINE GANTRY CRANE SAFELOAD PATHS

structural response (e.g., large inelastic deformations or abrupt failures 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. The results of this evaluation are tabulated in Table 3-7.

Where the controlling mode of response to postulated load drops is listed as "local," these loads were evaluated to determine the potential for slab penetration or performation. Scabbing of the concrete deck backface was evaluated for all loads. Where it was found that postulated drops are capable of producing this scabbing effect, the potential for impact of scabbing to safety-related systems was evaluated.

Postulated drops of the reactor coolant pump motor, new fuel shipping container, and other miscellaneous equipment fall in this category of loads which control local response, and bound other load drops that potentially lead to local effects. A discussion of the local effects evaluation methodology is provided below.

Where the controlling mode of response is listed as "overall structural," these load drops were evaluated to determine the potential for producing gross distortions of primary structural members and possibly propagating failures. Postulated drops of the reactor vessel head, reactor coolant pump motor and hot tool box, various hatch covers, and the internals, fall in this category and bound other load drops that potentially lead to "overall structural" effects.

The results of these analyses indicate that restricted load lift heights would assure no local scabbing of concrete and no adverse overall structural effects for the most limiting loads lifted over the refueling deck and north turbine deck. For less limiting loads, increased lift heights could be allowed with equivalent structural response.

Overall Structural Response Evaluations

A model of each floor location was developed with the objective of evaluating structural behavior for postulated flat and oblique drops of these loads.

A load drop methodology was developed to investigate the important modes of structural behavior. The objective of this methodology is to characterize

TABLE 3-7

STRUCTURAL LOAD EVALUATIONS

Load	Approximate Weight (Tons)	Maximum Height (Inches)	Controlling Mode of Response	
			Overall Structural	Local
TURBINE DECK				
Hot Tool Box	10	6	X	
New Fuel Container	2.5	18		X
Miscellaneous High Density Equipment[1]	2.5	18		X
Miscellaneous Low Density Equipment[2]	5	12		X
REFUELING DECK				
Reactor Cavity Seal Ring	10	36	X	
Stud Tensioners	2	120		X
Stud Rack	20	12	X	
Reactor Coolant Pump Motor	31	6	X	X
Westinghouse ISI Tool	5	48		X
RC Pump Hatch	19	12	X	X
Miscellaneous High Density Equipment[1]	2.5	120		X
Miscellaneous Low Density Equipment[2]	5	72		

[1] 2½ feet or less in diameter

[2] 5 feet or greater in diameter

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 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 was taken for potential sources of energy dissipation such as concrete crushing and penetration.

An iterative step-wise linear static analysis was performed for the postulated load drops whose controlling mode of response was determined to be "overall structural." The objective was to determine force-deflection for important points in the structural model. The computation procedure of the analysis is based on a network interpretation of the governing equations, the principal feature of which is the segmentation in processing of the geometrical, mechanical and topological relationships of the structure. This allows a concise and systematic computation algorithm that is applicable for different structural types.

For each impacted structural system (floor slab or slab-beam composite), a model was developed and the response of the system to the dynamic impact loading was determined. The model was loaded in the direct vicinity of the drop location. This is considered to be conservative in view of the fact that the slab will help transfer load away from the drop vicinity and result in a more favorable redistribution of the load.

The model was loaded until the moment capacity of any section or the allowable deflection was reached. This moment capacity is defined by Chapter 10 of ACI 318-77.

Generally, the ultimate load of a slab/grid system is reached prior to exceeding the hinge rotational capacity of particular sections provided that an unstable mechanism has not formed. This was found to be the case in this analysis. The hinge rotational capacity was used as a criterion to set the maximum allowable

level of deflection for the slab/grid system. The hinge rotational capacity for concrete structures was developed based on test results and is given as:

$$r_U = 0.0065\,(d/c) \leq 0.07 \qquad (1)$$

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 beam with a plastic hinge at its center is then given by:

$$Xm = (r_U L)/4 \qquad (2)$$

where

Xm = maximum deflection,

L = span of beam

Rotations of the magnitude governed by Equation I 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. The potential for scabbing was evaluated for all load drops.

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 318-77.

For each area where the potential for overall structural response modes was considered possible, an assessment of the bounding drop was made. The criteria

for selection was impact energy of the postulated drop.

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 were used. Test data shows that this property increases with the increased strain rates associated with dynamic loadings. For example, 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 are recommended.

2) Design (minimum) material properties for concrete and steel were used. No increase was taken for the aging of concrete which can amount to a factor of up to 1.35 of increased strength. Also, the average strength for structural steel is nearly a factor of 1.25 higher than the minimum yield requirement specified by ASTM. While these factors above minimum code strength exist and contribute to structural margins, they were not used in the evaluation.

3) Equation (1) for hinge rotational capacity was used. This corresponds to support 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; however, overall load carrying capability is expected to remain intact. Experimental observations 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 flexture 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 grid system.

4) The analysis used ACI 318-77 allowable shear stresses. A significant body of data suggests the existence of higher shear capabilities on the order of 10 f'c to 20 f'c. It is expected that the shear capabilities for the thick concrete slabs would tend to be in the higher end of the range.

5) In many cases, the analysis neglected the two-way resistance capability of the slab. It is expected that the slab would contribute increased strength particularly at larger deformations.

6) The load was distributed directly under the dropped load. In reality a more favorable load distribution would exist due to the load distribution capability of the slab.

7) No credit was taken for local energy dissipation associated with any crushing of the loads or the immediate surface of the floor.

Local Structural Response Evaluations

Selected loads such as the new fuel container, reactor coolant pump motor and various equipment were evaluated to assess the acceptability and potential consequences of postulated drops. The acceptance criteria were based on the capability of the concrete slabs to resist perforation, penetration, and underside scabbing.

The modified National Defense Research Committee (NDRC) formula was chosen because it has been shown to give the best fit with available experimental data. The NDRC formula for the depth of penetration, x (inches), of a solid cylindrical missile is given by:

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

$$x = KNW\,(V/1000\,d)^{1.8} + d \quad \text{for } x/d \geq 2.0 \qquad (3)$$

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 pounds/square inch)

The thickness of reinforced concrete needed to resist impact without perforation and scabbing is given by the following Army Corps of Engineers formulae which can be used in conjunction with equations (2) and (3).

$$ts/d = 2.12 + 1.36\,(x/d) \text{ for } 0.65 \leq x/d \leq 11.75 \quad (4)$$

$$tp/d = 1.32 + 1.24\,(x/d) \text{ for } 1.35 \leq x/d \leq 13.5 \quad (5)$$

where

ts = concrete thickness required to prevent scabbing

tp = concrete thickness required to prevent perforation

Equations (4) and (5) were later extrapolated for small values of x/d, giving

$$ts/d = 7.91\,(x/d) - 5.06\,(x/d)^2 \text{ for } x/d \leq 0.65 \quad (6)$$

$$tp/d = 3.19\,(x/d) - 0.718\,(x/d)^2 \text{ for } x/d \leq 1.35 \quad (7)$$

A 10 percent margin on thickness has been applied in the use of equations (6) and (7).

Limited penetration and scabbing was predicted for certain of the bounding heavy load drops considered; however, in no case was the refueling deck (el. 42) concrete slab predicted to be perforated. In addition, no scabbing of the north turbine deck was expected.

4.0 LIFT RIG EVALUATION

In the six-month report certain information requested in Item 3(d) was not supplied for the reactor vessel head, core barrel and internals lifting rigs. At that time our evaluation had not been completed because of a need to obtain information from the fabricator. Specifically, comparisons to the design requirements in ANSI Standard N14.6-1978, Sections 3, 5 and 6, were not complete and all exceptions were not identified. The core barrel lifting rig is only used when all fuel has been removed from the reactor vessel. Therefore, this lifting rig is not required to be evaluated to NUREG 0612 criteria since drops could not impact either spent fuel or safe shutdown equipment under those conditions.

Both the reactor vessel head and internals lifting rigs were designed and fabricated before ANSI Standard N14.6 was issued. Additionally, this standard is not applicable to these lifting rigs, since it was issued for lifting rigs associated with irradiated fuel shipping casks. Lifting rigs for these loads might be expected to have far greater usage, in less controlled environments and without the tight procedural controls imposed upon the lifting rigs for the reactor vessel head and internals. Therefore, we believe a detailed comparison to Section 6.0 of the ANSI standard is not useful or warranted. We have, however, reviewed the design requirements for the reactor vessel head and internals lifting rigs and believe that appropriate and conservative requirements were imposed. These design requirements are discussed below. Each lifting rig is subjected to inspection prior to use.

Reactor Vessel Head Lift Rig

The reactor vessel head lift rig was analyzed to determine compliance with applicable criteria in Section 3 of ANSI N14.6. These stress analyses demonstrated that the criteria of ANSI N14.6 were satisfied. The minimum safety factor for any load-bearing component in the reactor vessel lift rig is greater than that required by ANSI N14.6; that is, greater than 3 on yield strength and greater than 5 on ultimate strength. Based on the evaluations performed, it is concluded that the reactor vessel head lift rig is adequate.

Internals Lift Rig

The design of the internals lift rig used the ASME Boiler and Pressure Vessel Code, Section III, Article NA. In addition to the conservatisms implicit in this code, a design load of twice the operating load was assumed. Therefore, the maximum stresses in the lift rig are essentially a factor of 3 less than the conservative yield stresses set by code for the materials. In addition, the factors of safety on ultimate strength exceed the factor of 5 established by ANSI N14.6-1978 design criteria. Based on use of the ASME code requirements and on doubling the load for design, it is concluded that the internals lift rig is adequate.

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