PUBLIC SERVICE COMPANY OF COLORADO

FORT ST. VRAIN NUCLEAR GENERATING STATION

HEAVY LOAD ANALYSIS REPORT

PHASE II

PREPARED BY:

SARGENT & LUNDY

AND

TENERA CORPORATION

OCTOBER 15, 1985

HEAVY LOAD ANALYSIS REPORT

TABLE OF CONTENTS - PHASE II

TABLE OF CONTENTS - PHASE II (Cont'd)

HEAVY LOAD ANALYSIS REPORT

LIST OF TABLES - PHASE II

Fort St. Vrain

HEAVY LOAD ANALYSIS REPORT

LIST OF FIGURES - PHASE II

1.0 INTRODUCTION - PHASE II

This report is provided by Public Service Company of Colorado (PSC) as Phase II of the Heavy Load Analysis Report for Fort St. Vrain Nuclear Generating Station (FSV) in accordance with the guidelines of NUREG-0612 "Control of Heavy Loads at Nuclear Power Plants" and a letter from P. C. Wagner, NRC Region IV, to O. R. Lee, PSC, dated December 5, 1984 requesting reanalysis of heavy loads.

On June 14, 1985, PSC submitted Phase I of the Heavy Load Analysis Report. On August 8, 1985, PSC submitted an interim letter requesting a change in Phase II submittal date to October 15, 1985 and also listed excluded cranes and hoists. This Phase II report provides the remaining responses to the requested information on cranes and hoists with pagination continued from Phase I.

2.2 Specific Requirements for Overhead Handling System Operating in the Vicinity of Fuel Storage Pool.

2.2.1 Identify by name, type, capacity, and equipment designator, any cranes physically capable (i.e., ignoring interlocks, movable mechanical stops, or operating procedures) of carrying loads which could, if dropped, land or fall into the spent fuel pool.

RESPONSE

Fort St. Vrain does not have a fuel storage pool. Instead, up to one-third of the reactor core can be stored in the fuel storage facility, H-1401, below the refueling floor. Spent fuel elements are stored in nine steel lined vessels (wells) supported in three seismically designed concrete vaults with removable steel cover plugs in lead-filled steel cover slabs. The fuel storage facility is shown in Figures 2.1.3-1 through 2.1.3-5 in Phase I of this report and discussed in FSV updated FSAR Subsection 9.1.2.

FSV FSAR Subsection 9.1.2.3 states:

> "The design and operation of the fuel storage facility is such that spent fuel is adequately contained under all normal and abnormal conditions ... The design of the fuel storage wells will preclude criticality even when completely filled with fuel and flooded with water. The multiplication factor for the worst flooding situation is calculated to be less than 0.85."

The reactor building overhead crane is the only overhead handling system capable of carrying loads over the fuel storage facility. The appropriate information is provided in Table 2.1.1-1.

The reactor building overhead crane, G-7201, is a Whiting Corporation pendant controlled electric overhead traveling bridge crane with a 170-ton main hook sheave, a 50-ton main hook block, and a 17.5-ton auxiliary hook block. Crane hook coverage is shown in Figure 2.1.3-1.

The crane load combination consisting of the 170-ton main hook, snubber control system, lifting mushroom with the fuel handling machine (FHM) is the means by which fuel handling is performed over the PCRV, fuel storage facility, fuel shipping cask in the fuel handling port (pit), and on the reactor building refueling floor. All such operations are under procedural control. Moreover, procedural controls prevent the handling of any heavy loads over the PCRV during reactor operation.

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

No cranes are excluded in this area from the above category.

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 features). For each crane so evaluated, provide the load-handling system (i.e., crane-load-combination) information specified in Attachment 1.

<u>RESPONSE</u>

The reactor building crane has not been evaluated against the single-failure-proof criteria of NUREG-0612, Section 5.1.6. However, as described in PSC letter to NRC dated December 14, 1981, PSC has concluded that the reactor building crane, when utilizing the snubber lifting system, has sufficient design features so that the probability of a load drop is extremely small. The NRC has accepted this conclusion in letters dated March 6, 1984 and December 5, 1984.

The reactor building crane and snubber system are utilized for lifts of the fuel handling machine and auxiliary transfer cask. Accordingly, drops of these two loads have not been postulated for the purpose of evaluating FSV compliance with Criteria I, II, III or IV of NUREG-0612, Section 5.1 and are not addressed further in this Phase II response.

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

For those loads not addressed in the response 2.2.3, alternative 4 of NUREG-0612, Section 5.1.2, the analysis of heavy load drops in the spent fuel area, has been selected. The heavy loads, handled over the spent fuel area are listed in Table 2.1.3-1. The assumptions and method of analysis for heavy load drops are discussed in response 2.4.2d(3). Structural analysis of load drops on the lead-filled steel cover slabs of the fuel storage area indicate no damage to the cover slabs. Therefore, there is no damage to the fuel. Consequently, Criteria I, II, and III are satisfied since there is no fuel damage, no radiological release, no criticality problem, and no loss of spent fuel well cooling.

2.2.4 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

Neither alternative 2 nor 3 in NUREG-0612, Section 5.1.2 has been selected. No electrical interlocks or mechanical stops are used for limiting crane motion on the refueling floor.

2.2.4 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

No reliance is placed on crane operational limitations.

2.2.4 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

No reliance is placed on the physical locations of spent fuel.

2.2.4 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.

RESPONSE

No exception is taken to NUREG-0612, Appendix A. Information requested in Attachment 4 is provided in response 2.4.2d(3).

2.3 Specific Requirements of Overhead Handling Systems Operating in the Containment.

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 reactor building refueling floor overhead crane, G-7201, is a Whiting Corporation pendant controlled electric traveling bridge crane with a 170-ton main hook sheave, a 50-ton main hook block and an auxiliary hook block rated at 17.5 tons. This is the only crane capable of carrying heavy loads over the prestressed concrete reactor vessel (PCRV). During reactor operation, plant procedures do not permit heavy load movements over the PCRV.

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

No cranes are excluded in this area from the above category.

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

Refer to response 2.2.3.

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

No reliance has been placed on the installation and use of electrical interlocks or mechanical stops.

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 site specific consideration relied on is the prohibition of moving loads over the PCRV during reactor operation per Procedure MP-104-1.

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

A load drop analysis by GA Technologies for a reactor isolation valve dropping onto a PCRV refueling penetration indicated no fuel damage and that coolant loss from the depressurized reactor was within acceptable limits (Reference 4). Thus, Criteria I and II and the intent of Criterion III (i.e., sufficient coolant inventory is maintained to ensure adequate core cooling) are satisfied. This analysis bounds other potential load drops on the PCRV.

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.

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.

<u>RESPONSE</u>

Refer to response 2.2.3.

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:

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 of 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.

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 redundacy of safety-related equipment, discuss the basis for determining that load drops will not affect continued system operation (i.e., the ability of the system to perform its safety-related function).

(2) Where mechanical stops or electrical interlocks are to be provided, present details showing the areas where crane travel will be prohibited. Additionally, provide a discussion concerning the procedures that are to be used for authorizing the 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.

(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

For all overhead handling systems listed in Table 2.1.1-1, evaluations and analyses were performed pursuant to Section 2.4.2 above. To perform these analyses, load impact regions were defined for the crane hook/hoist coverage areas for each handling system. These load impact regions were defined based on a conservative definition of the extent of potential damage from postulated load drops.

For the two large overhead handling systems, the reactor building crane and the turbine building crane, multiple load impact regions were defined to cover the large hook coverage areas (see Figures 2.4-1 through 2.4-4). These regions were defined to include considerable overlapping to assure that postulated drops at the region boundaries were accounted for appropriately. Vertically, the load impact regions extended to the lowest elevation of the building, unless structural analyses demonstrated that this was not necessary. Nineteen load impact regions were defined for the reactor building overhead crane, and twelve load impact regions were defined for the turbine building overhead crane.

With regard to the remaining monorails and hoists, the load impact regions were conservatively defined to encompass the entire monorail coverage, including a minimum of 2 feet 6 inches on either side of the monorail (see Phase I Figures showing monorail coverage and Table 2.1.3-1). Vertically, the region boundaries were defined in the same manner as described above.

Table 2.4-1 provides an overview of the handling systems and load impact regions for which NUREG-0612 Criterion IV evaluations, relative to shutdown capability, were performed. The black dots in Table 2.4-1 indicate the types of evaluations performed for each handling system/load impact region.

Shutdown Models

The systems evaluations described below considered the effects of load drops on the ability to achieve and/or maintain the plant in a shutdown condition. The shutdown model utilized for most load impact regions was the forced circulation cooling model previously submitted to the NRC (Reference 2).

In addition, for certain load impact regions in the turbine building, a second shutdown model was utilized to assure acceptable consequences. This second model was utilized since it could not be readily demonstrated that forced circulation cooling could be accomplished using the Appendix R shutdown model for certain load drop scenarios from the turbine building overhead crane. The load drop scenarios of interest involve postulated drops of heavy turbine components following shutdown of the plant (core cooling is still required to remove decay heat). The shutdown model for these turbine building load impact regions relies on the equipment necessary to assure that potential consequences remain within those calculated for DBA-1, Permanent Loss of Forced Circulation (see FSV FSAR Section 14.10 and Appendix D). The basis for acceptability of the DBA-1 consequences for heavy load drop evaluations is that the dose consequences of DBA-1 are well within the dose acceptance criteria of NUREG-0612, Section 5.1, Criterion I. It is recognized that the DBA-1 consequences do involve some fuel damage. However, the NRC in NUREG-0612 recognized that fuel damage is a potential consequence of heavy load drops at nuclear power plants and specifically addressed this potential outcome by establishing the dose criteria in Section 5.1.

Assumptions Regarding Loss of Equipment

In general, equipment was assumed lost (i.e., inoperable or failed) based on a conservative judgment regarding the impacted area in a load region (described above). The area of postulated damage for a region was either assumed to be broader than the actual area of impact would likely be (e.g., the entire region at all elevations) or was constrained to a smaller area of influence by considering the characteristics of the load and/or load handling equipment, the results of structural analyses, load path restrictions within or over the region, and the physical and/or geometric structure of the region of the building.

Systems Evaluation Approach

The systems evaluations were performed on a load impact region basis. For each load impact region, the ability to accomplish the applicable shutdown model functions was evaluated. The evaluation involved selecting a load impact region and identifying which components in the shutdown model were within the region. This was accomplished by a detailed review and markup of drawings (e.g., P&IDs, piping plans, electrical cable plans) supplemented by plant walkdowns, when required.

The systems evaluations included a determination as to whether failure of system components within a region could result in the complete loss of a required shutdown model function.

Structural Analysis Methodology

Structural analyses were performed to evaluate the potential consequences of drops onto various floor slabs, e.g., a postulated drop of a reactor isolation valve onto the reactor building refueling floor. The basic purpose was to define the extent of postulated damage at elevations below the impacted floor slabs. Direct impact of equipment at lower elevations would have to be postulated if perforation or overall collapse of the floor system were predicted. Impacts of secondary missiles (i.e., scabbed concrete from the underside of the deck) were treated differently in that failures of large piping systems were not predicted from scabbing. The structural analysis methodology is described in response 2.4.2.d.

Results, Conclusions, and Future Actions

The summary of evaluation results in Table 2.4-1 indicates that there are potential problems in four load impact regions. For all other regions, it was successfully demonstrated that either (1) no shutdown model equipment could be impacted, (2) shutdown model equipment could be impacted, but system redundancy assured that shutdown capability was not compromised, or (3) for the turbine building overhead crane, that the minimum equipment required to limit the consequences to those calculated for DBA-1 was not damaged. These load impact regions where potential problems were identified are discussed below.

Reactor Building-Load Impact Region RB-3

The spent fuel shipping cask is placed in the loading port (see RB-3 on Figure 2.4-1) to facilitate loading of spent fuel with the fuel handling machine. In this location, the cask extends vertically below the refueling floor and is supported at its upper end by a ring support structure that is tied into the refueling floor structural steel beams. Structural analyses of a postulated drop of a spent shipping cask into the loading port were performed. The results of the analyses indicated that failure of the cask supports could not be precluded. Systems analyses indicated the potential areas of impact below the refueling floor where piping associated with both Trains A and B of the forced circulation cooling model could be impacted.

PSC is evaluating solutions for this load drop scenario and will implement action to either improve the reliability of this load movement or reduce the potential consequences.

Reactor Building - Load Impact Region RB-4

Load impact region RB-4 is defined, in plan, by the walls of the hot service facility and extends to the lowest elevation of the reactor building (elevation 4740').

Load drop scenarios into the hot service facility were analyzed structurally to determine if damage could be limited vertically in the building. There is shutdown system piping at the lowest elevation of the reactor building within this region that is common to both forced circulation cooling shutdown trains. Only two load drop scenarios, the drop of a removable hot service facility hatch cover from the reactor building crane 50-ton hook into the deep end of the hot service facility and the drop of the ATC from the reactor building crane 170-ton hook without the snubber control system into the shallow end of the hot service facility, could not be shown to have acceptable results. PSC is evaluating potential solutions for these load drop scenarios and will implement actions to either improve the reliability of these load movements or reduce the potential consequences.

Reactor Building - Load Impact Region RB-6

Load impact region RB-6 is defined in plan by the walls of the helium purification regeneration equipment pit and extends vertically to the lowest elevation of the reactor building. Scenarios of a load drop into this pit were analyzed structurally to determine if damage could be limited vertically within the pit space. There is shutdown system piping at the lowest elevation of the reactor building within this region that is common to both forced cooling shutdown trains. The load drop scenario of the removable slab from the reactor building crane 50-ton hook into this pit could not be shown to have acceptable results.

PSC is evaluating potential solutions for this load drop scenario and will implement actions to either improve the reliability of the load movement or reduce the potential consequences.

Reactor Building - Load Impact Region RB-27

Load impact region RB-27 is defined by the area under the PCRV Safety Valve Tank Head Monorail/Hoist (C30). Potential consequences of a heavy load drop of the tank head could not be limited vertically in the reactor building because the floor immediately below the hoist is grating, not concrete. Shutdown system components affecting the operation of both forced circulation cooling shutdown trains are located within the region at elevations below the hoist (elevation 4811 feet).

PSC has identified several options for addressing this issue and will implement actions to improve the reliability of the hoist system and/or reduce the consequences of the potential load drop.

TABLE 2.4-1

EVALUATION APPROACHES AND RESULTS
VS.
LOAD IMPACT REGIONS

EVALUATION APPROACH	REACTOR BUILDING																										
	REACTOR BUILDING OVERHEAD CRANE (C-1)																			HE PURIF. EQPT. HOIST			BEARING WATER PUMP HOIST (C-4)	BEARING WATER PUMP HOIST (C-6)	BEARING WATER MUP PUMP HOIST (C-12)	PCRV SAFETY VALVE TANK HEAD HOIST (C-30)	DEMIN FILTER HOIST (C-13)
																				C-16	C-17	C-18					
	LOAD IMPACT REGIONS																										
	RB-1	RB-2	RB-3	RB-4 *	RB-5	RB-6	RB-7	RB-8	RB-9	RB-10	RB-11	RB-12	RB-13	RB-14	RB-15	RB-16	RB-17	RB-18	RB-19	RB-21	RB-22	RB-23	RB-24	RB-25	RB-26	RB-27	RB-28
DOSE CALCULATIONS																											
STRUCTURAL ANALYSES			●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●				●	●
SYSTEMS EVALUATIONS	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●
EVALUATION RESULTS (NOTE 2)	e	e	P	P	e	P	e	e	e	e	e	e	e	e	e	e	e	e	e	e	e	e	b	b	b	P	e

* (NOTE 1)

TABLE 2.4-1 (continued)

EVALUATION APPROACHES AND RESULTS
VS.
LOAD IMPACT REGIONS

EVALUATION APPROACH	TURBINE BUILDING														CIRCULATING WATER PUMP PIT
	TURBINE BUILDING OVERHEAD CRANE (C-2)												CONDENSATE PUMP HOIST (C-23)	RX PLANT AIR HANDLING EQPT. MONORAIL (C-28)	CIRCULATING WATER PUMP HOIST (C-29)
	LOAD IMPACT REGIONS														
	TB-1	TB-2	TB-3	TB-4	TB-5	TB-6	TB-7	TB-8	TB-9	TB-10	TB-11	TB-12	TB-13	TB-14	CW-1
DOSE CALCULATIONS	●	●	●	●	●	●	●	●	●	●	●	●			
STRUCTURAL ANALYSES				●	●	●									
SYSTEMS EVALUATIONS	●	●	●	●	●	●	●	●	●	●	●	●	●	●	●
EVALUATION RESULTS (NOTE 2)	f	f	f	f	f	f	f	f	f	f	f	f	b	e (NOTE 3)	b

TABLE 2.4-1 NOTES

● - Indicates types of evaluations performed for each load impact region.

Note 1 - This load impact region, RB-4, also includes the Hot Service Facility Hoist (C20). Load drops from this hoist did not result in unacceptable consequences in load impact region RB-4.

Note 2 - Evaluation Results Key

a. Crane travel for this area/load combination is prohibited by electrical interlocks or mechanical stops.

b. System redundancy and separation precludes loss of capability of system to perform its safety-related function following this load drop in this area.

c. Site-specific considerations eliminate the need to consider load/equipment combination.

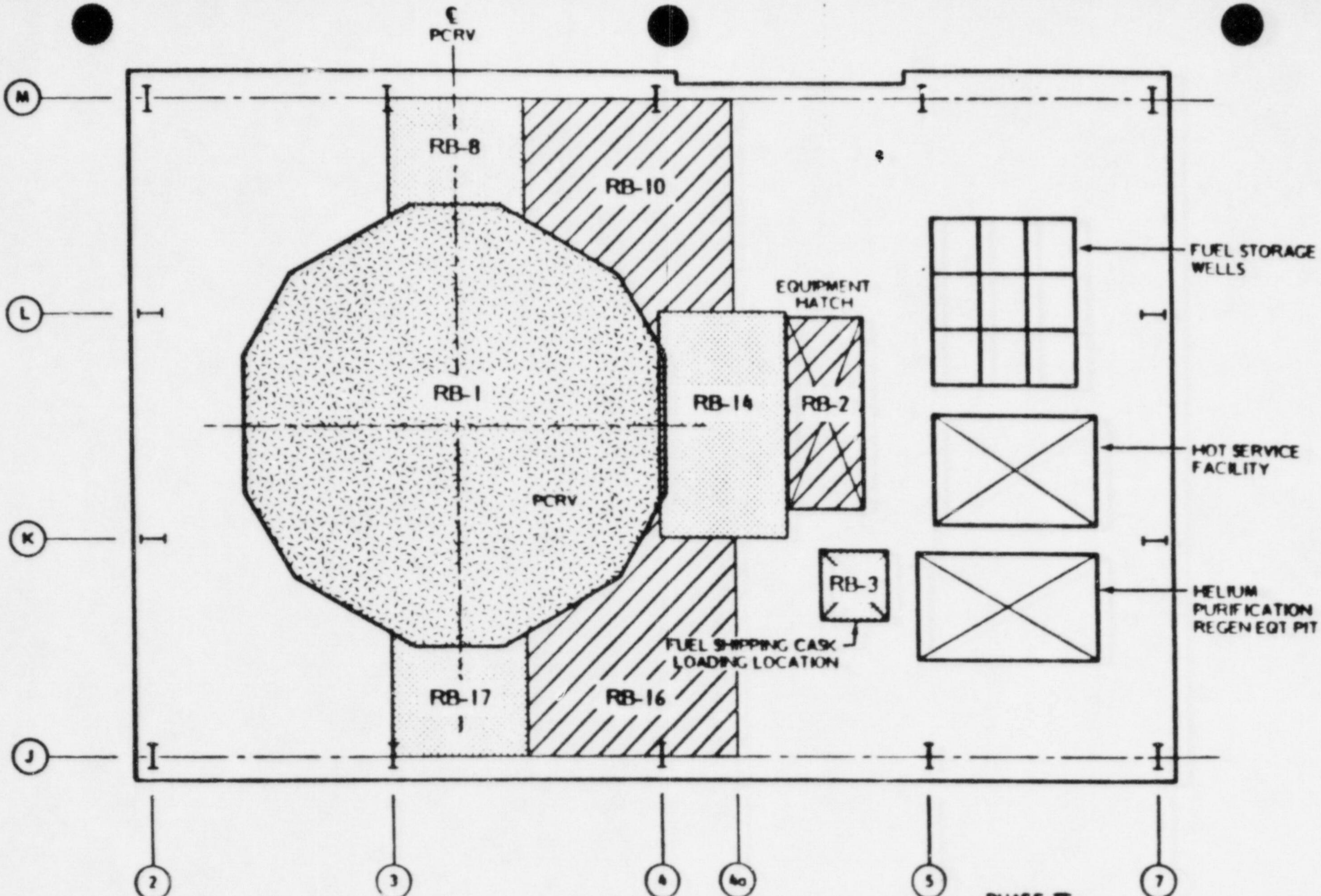
d. Likelihood of handling system failure for this load is extremely small (i.e., NUREG-0612, Section 5.1.6 satisfied).

e. Analysis demonstrates that crane failure and load drop will not damage shutdown equipment (forced circulation cooling trains A and B).

f. Analysis demonstrates that crane failure and load drop will not result in dose consequences in excess of NUREG-0612, Section 5.1, Criterion I.

P - Potential Problem Area

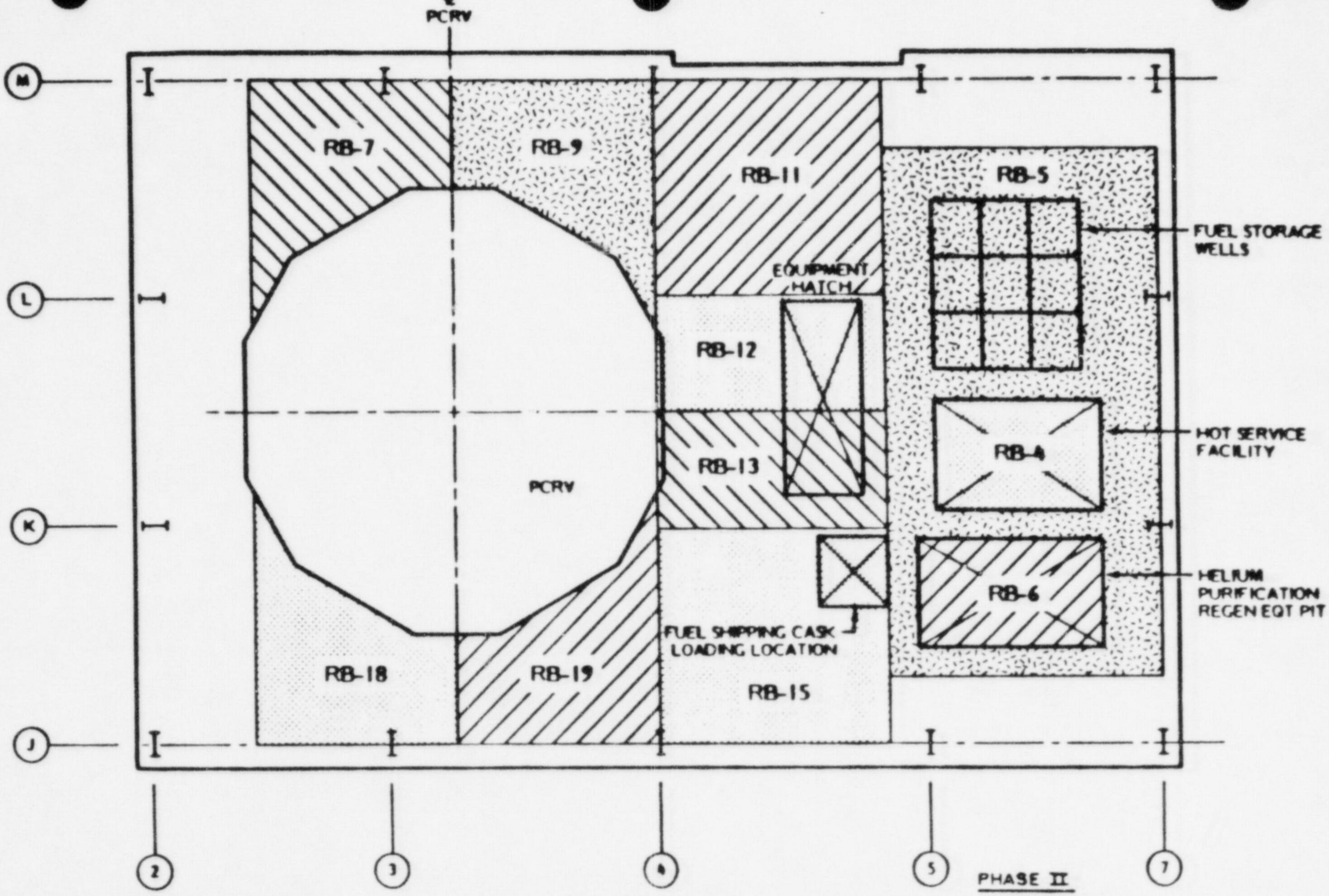
Note 3 - Alternative method relied on for reactor building cooling using chilled water system and air handling unit S-7320S. This method is independent of the turbine building areas underneath the hoist coverage of Reactor Plant Air Handling System Hoist (C28).



PHASE II
10-04-85

FIGURE 2.4-1

Reactor Building
Load Impact Regions Exposed
to Reactor Building Crane

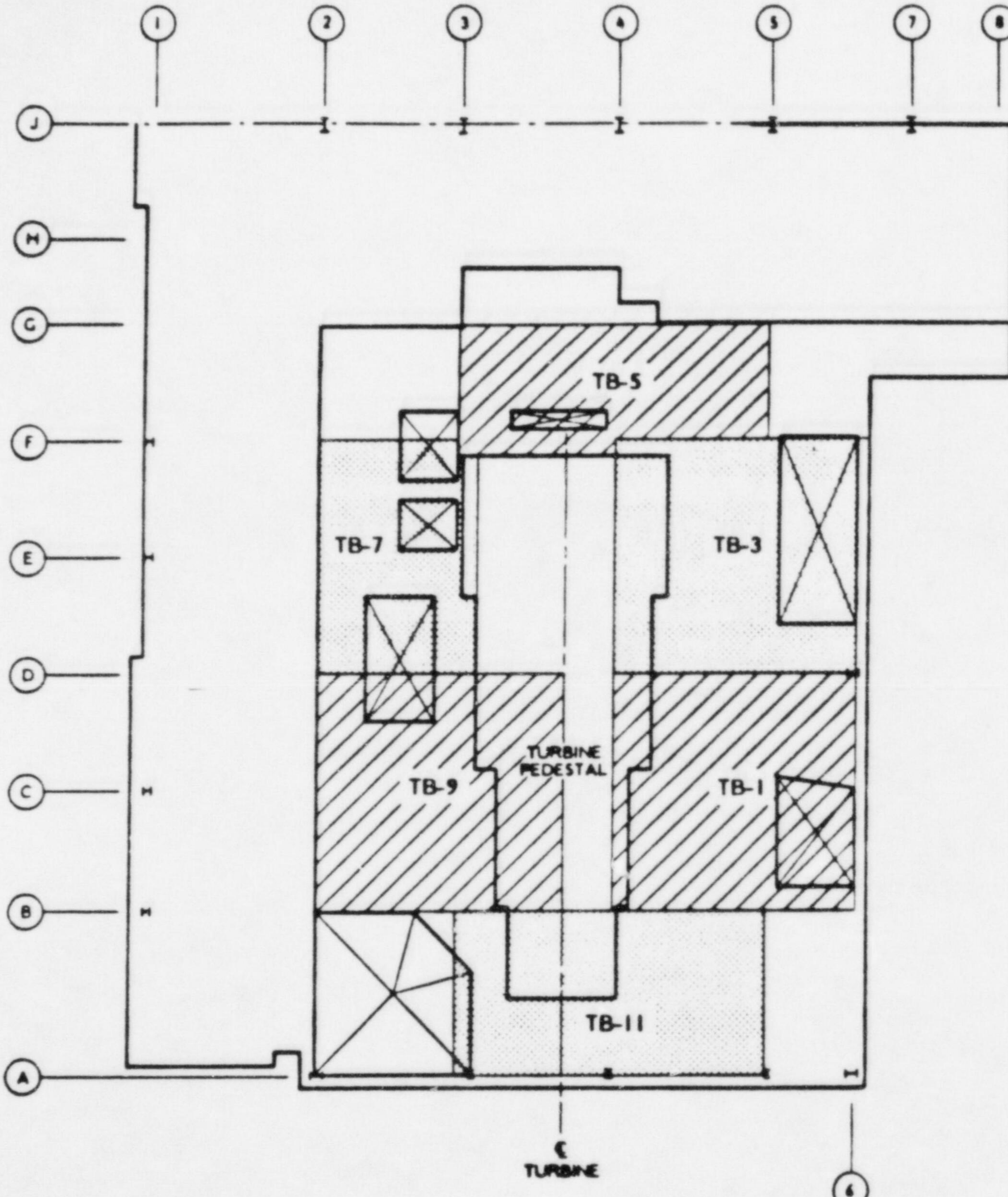


PHASE II

10-04-85

FIGURE 2.4-2

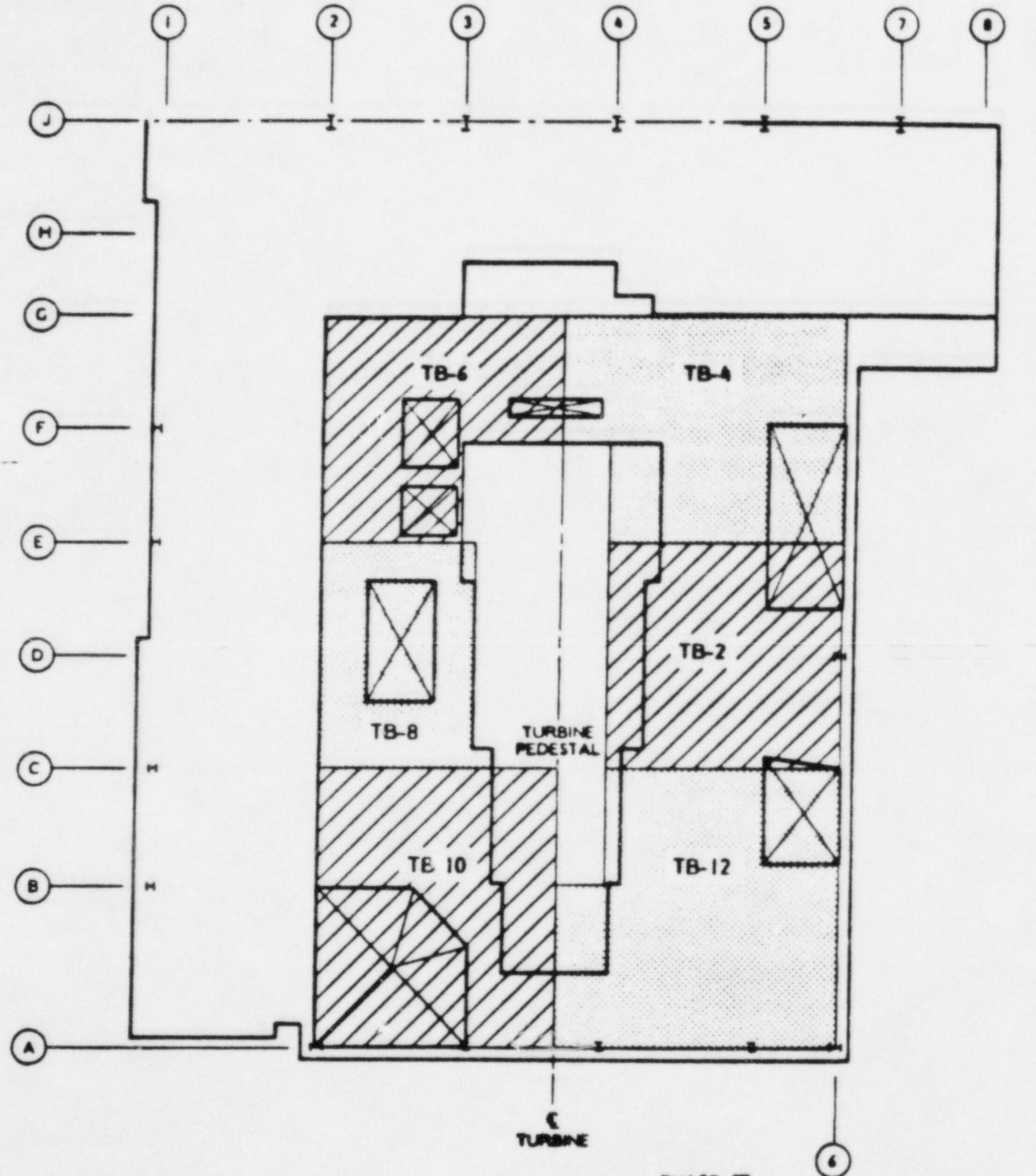
Reactor Building
Load Impact Regions Exposed
to Reactor Building Crane



PHASE II
10-04-85

FIGURE 2.4-3

TURBINE BUILDING
LOAD IMPACT REGIONS
EXPOSED TO TURBINE BUILDING CRANE



PHASE II
10-04-85

FIGURE 2.4-4

TURBINE BUILDING
LOAD IMPACT REGIONS
EXPOSED TO TURBINE BUILDING CRANE

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, <u>or partial compliance supplemented by suitable alternative or additional design features)</u>. For each crane so evaluated, provide the load-handling-system (i.e., crane-load-combination) information specified in Attachment 1.

<u>RESPONSE</u>

Refer to response 2.2.3.

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 constructed such that the hoisting system will retain its load in the event of seismic accelerations equivalent to those of a safe shutdown earthquake (SSE).

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

(3) The information requested in Attachment 4.

RESPONSE

2.4.2.d (1) Crane Handling System Seismic Criteria

DESCRIPTION

The turbine building crane is a Whiting Corporation overhead traveling bridge crane consisting of a single trolley, two girders and end trucks. This crane has been qualified per the requirements of CMAA Specification 70 (CMAA 70).

The turbine building crane rated load is 65 tons for the main hook. The top of the crane girder rail is at Elevation 4865'-10", which is approximately 76 feet above the grade level. The lifting height for the main hook is 80 feet below the crane rail. The span of the turbine building crane is 106'-7".

ANALYSIS

This crane was analyzed using a simplified approach for the safe shutdown earthquake (SSE) specified for the FSV plant. The seismic response spectra used in this analysis were obtained from Reference 8.

Overhead cranes exhibit three predominant modes of vibration: extension of the rope, strong axis bending of the girders and weak axis bending of the girders. The decoupled frequencies corresponding to these modes of vibration were determined and used to obtain spectral accelerations. These acceleration values were factored by 1.5 to account for higher modes and then applied to the crane and lifted load mass. However, if the decoupled frequencies in one direction (e.g., rope extension and girder strong axis bending) are close and may result in resonance, a 2-degree-of-freedom model is used. The effects of the two horizontal and vertical components of the earthquake were then combined by the square root of the sum of squares (SRSS).

The strong axis flexural stress in the girder was conservatively assumed to be at CMAA 70 allowables at the rated load. The SSE condition stress was determined by factoring this stress by the vertical spectral acceleration. In the weak axis direction, the seismic stresses were found based on the seismic loadings and the crane section properties.

The SSE allowable stresses were taken as 0.95 of the yield stress or 0.95 critical buckling. A damping value of 4% (percent of critical damping) was used in the analysis.

RESULTS

The major structural components of the turbine building crane were evaluated. The crane girder stresses were found to be within the SSE allowables for the combined effect of horizontal and vertical earthquake. The rope factor of safety indicates a substantial margin for the SSE condition.

Furthermore, the test load for these cranes is larger than the vertical seismic effect when the crane is holding the maximum critical load (MCL).

The crane will remain on its rails during an SSE event. The SSE seismic loading results in no uplift at the crane wheels. Safety stops which prevent lateral disengagement of the wheel and the rail have been provided.

CONCLUSION

The Fort St. Vrain turbine building crane major structural components possess adequate margin, as described above, to withstand the effects of the SSE specified for the Fort St. Vrain Plant. PSC's letter to the NRC dated December 14, 1981, indicates the design adequacy of the reactor building overhead crane to withstand the effects of the SSE specified for the Fort St. Vrain plant.

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

RESPONSE

No exceptions to NUREG-0612, Appendix A, are taken in the analysis.

2.4.2.d (3) The information requested in Attachment 4.

RESPONSE

The requested information for the analysis to demonstrate compliance with NUREG-0612, Section 5.1, is:

1. Initial Conditions/Assumptions:

   a. Weight of heavy loads are listed in Table 2.1.3-1.

   b. Impact areas of loads are listed in Table 2.4-1 and described in the response 2.4.2a and b.

   c. Safe load heights have been calculated based on the structural capability of the floor and will be implemented in revised station procedures.

   d. Drop location is the most critical drop area along the load path or within the designated load drop region.

   e. Impact area is assumed perfectly plastic when determining the post impact response of the system.

   f. The appropriate thickness of the slab was used in the analyses.

   g. Environmental drag forces are not considered.

h. The load combinations considered include:

- dead weight of the structural element,
- weight of the dropped load, and
- weight of the lifting device.

i. Material properties of steel and concrete:

- Structural Steel: ASTM A36 with a minimum yield strength of 36,000 pounds per square inch.
- Reinforcing Bars: ASTM A615 with a minimum yield strength of 60,000 pounds per square inch.
- Reinforced Concrete: Minimum compression strength is assumed to be 4,500 pounds per square inch. Construction specifications call for a compressive strength of 3,500 pounds per square inch. However, the increased strength is justified due to concrete aging (Reference 15) and the dynamic nature of impactive loads (Reference 1).

2. Method Of Analysis

A heavy load drop on structural elements causes plastic deformation of these elements. Immediately after impact, the dropped object and the target will move together with a common velocity. The motion stops when the kinetic energy is completely absorbed by the straining of the target. The final deflected shape of the target is reached when the velocity is zero. A safe load is one in which the ductility ratio for the target does not exceed acceptable limits (References 1, 11 and 12). The ductility ratio is the strain due to the impacting load divided by the elastic strain. The method of analysis follows the basic laws of motion and kinetic and strain energy.

When shutdown equipment is located immediately below the impact area under consideration and the dropped objects have a contact area of less than 4 square feet, the local damage to the impacted concrete structural element in the form of scabbing is considered. The dropped object is considered as a rigid missile and the maximum load which does not cause scabbing of the concrete is calculated according to the equations in Reference 11.

Safe load heights have been calculated based on the overall and local damage criteria, as applicable.

3. Conclusion

The conclusions are summarized in Table 2.4-1 and safe load heights and paths will be incorporated in revised plant operating procedures.

3.0 REFERENCES

1. "Appendix C - Special Provision for Impulsive and Impactive Effects (1977a)", ACI 349-76.

2. "Appendix R Evaluation: Fort St. Vrain Nuclear Generating Station, Report No. 1 - Shutdown Model (Rev. 4), and Report No. 2 - Electrical Reviews (Rev. 3)," Public Service Company of Colorado/TENERA Corporation, 1985.

3. Biggs, J. M., Introduction of Structural Dynamics, Copyright 1964 by McGraw-Hill, Inc.

4. "Report of Load Drop on PCRV Refueling Penetration" GA Technologies, Document Number 908280, dated October 1, 1985.

5. Fort St. Vrain Final Safety Analysis Report, Public Service Company of Colorado.

6. Fort St. Vrain Technical Specifications.

7. Fort St. Vrain updated Final Safety Analysis Report, Revision 3, Public Service Company of Colorado.

8. Horizontal Seismic Response Spectra for Fort St. Vrain Unit 1, S&L Report SAD 0377, January 15, 1981.

9. Manual of Steel Construction, American Institute of Steel Construction, Inc., New York, N.Y., Seventh Edition.

10. "Monorails and Underhung Cranes", American National Standard ANSI B30.11-1980.

11. Second ASCE Conference on Civil Engineering and Nuclear Power, Report of the ASCE Committee on Impactive and Impulsive Loads, Vol. V.

12. "Seismic Design for Nuclear Power Plants", Edited by R. J. Hansen, The MIT Press, Cambridge, Mass., and London, England, Copyright 1970, Second Printing 1972.

13. "Slings", American National Standard ANSI B30.9-1971.

14. "Standard for Special Lifting Devices for Shipping Containers Weighing 10,000 Pounds (4500 kg) or More for Nuclear Material", American National Standard ANSI N14.6-1978.

15. Winter, G., et.al., Design for Concrete Structures, Eighth Edition, McGraw-Hill Book Company, 1972.