GULF STATES UTILITIES COMPANY

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File Nos. G9.5, G9.19.2

Mr. Harold R. Denton, Director
Office of Nuclear Reactor Regulation
U.S. Nuclear Regulatory Commission
Washington, D.C. 20555

Dear Mr. Denton:

River Bend Station - Unit 1
Docket No. 50-458

Gulf States Utilities Company (GSU) provides in Attachment 2 the information requested in Sections 2.2, 2.3 and 2.4 of Enclosure 3 to the Nuclear Regulatory Commissions' (NRC) December 22, 1980 letter. The letter requested a review of the planned controls for handling heavy loads at River Bend Station (RBS). GSU has performed a comparison of these controls to the guidelines of NUREG-0612 to demonstrate that the guidelines have been or will be met and to satisfy the NRC RBS Safety Evaluation Report outstanding item No. (12).

On March 1, 1984 GSU submitted a report addressing the first phase of the requested review (Section 2.1 of Enclosure 3 to the NRC's December 22, 1980 letter). A technical evaluation report was performed by EG&G Idaho, Inc. on GSU's phase one submittal. As a result of their review, additional information was requested in a NRC letter dated September 14, 1984. GSU's response to the NRC's request for additional information is addressed in Attachment 1.

Sincerely,

J. E. Booker

J. E. Booker
Manager-Engineering,
Nuclear Fuels & Licensing
River Bend Nuclear Group

erg
JEB/RJK/kt

Attachment

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ATTACHMENT 1

Guideline 1, NUREG-0612, Article 5.1.1(1):

NRC Request:

Information is needed to confirm that for other heavy loads handled by the Polar crane and the other nonexempt cranes or hoists (35 identified loads), safe load paths are followed. Additional evaluation and information, as discussed, is needed for RBS to show consistency with NUREG-0612 Guideline 1, for 35 loads.

GSU Response:

Of the 35 identified loads, Safe Load Path designations/restrictions, were necessary for approximately 17 loads handled by three cranes and two monorails, and are presently being incorporated into RBS load handling procedures. The approach for this selection is based on the conclusions found in the phase II report (note pgs. 1-5, 29-31).

Guideline 2, NUREG-0612, Article 5.1.1(2):

NRC Request:

The information submitted provides a commitment to develop procedures. However, the five specific requirements that these procedures should include (see 2.3.2 above) have not been addressed. In the preparation RBS should include all of the requirements or justify exceptions. Suitable resolution of the Safe Load Path guidelines must be established before procedures consistent with Guidelines can be written. At a minimum, procedures should cover handling of those loads listed in Table 3-1 of NUREG-0612. These procedures should include: identification of required equipment; inspections and acceptance criteria required before movement of load; the steps and proper sequence to be followed in handling the load; defining the safe path; and other special precautions.

RBS should supplement the commitment to develop procedures with information or statements to confirm that the procedures incorporate all of the requirements specified in NUREG-0612 Section 5.1.1(2).

GSU Response:

GSU is developing Load Handling Procedures for systems of concern to contain the requirements of 5.1.1(2).

Guideline 4, NUREG-0612, Article 5.1.1(4)

NRC Request:

Confirm whether there are three or only two "Special Lifting Devices." If in fact there are three identify the third one and provide information to verify its consistency with NUREG-0612 Guideline 4 requirements.

GSU Response:

Only two special lifting devices exist as described in phase I. The third lifting device was removed from the RBS design during phase I evaluation. Based on phase II evaluation results, the five loads handled by the special lifting devices are not "critical loads" as defined by ANSI N14.6.

Guideline 7, NUREG-0612, Article 5.1.1(7)

NRC Request:

A suitable resolution of the inconsistent exception concerning initial testing will bring RBS into acceptable consistency with the NUREG-0612 Guideline 7.

GSU Response:

There is no longer any exception to initial crane testing identified since test loaded movement of bridge and trolley are included in RBS start-up procedures for the three cranes of concern.

RESPONSES TO REQUESTS FOR INFORMATION IN SECTIONS 2.2, 2.3, AND 2.4 OF ENCLOSURE 3 TO NRC DECEMBER 22, 1980 LETTER

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-handing 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 that the estimated consequences of such a drop will not exceed the limits set by the evaluation criteria of NUREG-0612, Section 5.1, Criteria I through III.

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

RESPONSE: At River Bend, there are two spent fuel storage pools. One in the Fuel Building at the 113' el. capable of storing up to 525% of a full core and another in the Reactor Building capable of storing up to 32% of a full core. (See Figures 7 and 6, Regions 19 and 2 respectively). The spent fuel storage pools and racks are described in FSAR Section 9.1.

With regard to the Fuel Building, there are no cranes capable of carrying heavy loads over the spent fuel pool. In the Reactor Building, the Polar Crane is capable of carrying loads over the spent fuel storage area. It is an overhead bridge crane mounted on a circular rail with a main and auxiliary hoist with capacities of 100 and 5 tons, respectively.

ITEM 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: With regard to the Fuel Building spent fuel storage area, the Spent Fuel Cask Trolley is located at el 151'. However, the spent fuel pool is protected from drops from this crane by restriction (i.e., location of racks) of the limits of cask crane travel as discussed in subsections 9.1.4.2.2.1 of the River Bend FSAR. The Fuel Building Bridge Crane will be provided with fixed mechanical stops prior to storage of any spent fuel in the spent fuel pool. These stops will restrict crane movement such that it is incapable of carrying heavy loads over spent fuel.

With regard to both the Fuel Building pool and the Reactor Building spent fuel storage area, there are 1/2 ton capacity jib cranes that can lift loads over the spent fuel storage racks. As indicated in our initial response, however, these cranes are only used to carry lighter loads, such as channels, control rods or fuel assemblies that do not qualify as heavy loads.

ITEM 2.2.3 Identify any cranes listed in 2.2.1, above, which you have evaluated as having sufficient design features to make the likelihood of a load drop extremely small for all loads to be carried and the basis for this evaluation (i.e., complete compliance with NUREG-0612, Section 5.1.6 or partial compliance supplemented by suitable alternative or additional design features). For each crane so evaluated, provide the load-handling-system (i.e., crane-load-combination) information specified in Attachment 1.

RESPONSE: It has not been necessary to evaluate the Polar Crane against the criteria of NUREG-0612, Section 5.1.6.

ITEM 2.2.4 For cranes identified in 2.2.1, above, not categorized according to 2.2-3, demonstrate that the criteria of NUREG-0612, Section 5.1, are satisfied. Compliance with Criterion IV will be demonstrated in response to Section 2.4 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.

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.

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 ad discuss administrative or physical controls provided to ensure that these assumptions remain valid.

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.

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: The heavy loads that could be handled by the Polar Crane were identified in Table 4 in GSU's initial submittal to NRC regarding the heavy loads issue. As indicated in the response to Item 3.a of that submittal, both procedural restrictions and Technical Specifications have been developed to prevent carrying heavy loads over spent fuel in the racks of the Containment fuel pool. Therefore the requirements of Criteria I through III of NUREG-0612 are satisfied.

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 that the estimated consequences of such a drop will not exceed the limits set by the evaluation criteria of NUREG-0612, Section 5.1, Criteria I through III.

ITEM 2.3.1 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 heavy loads over the reactor vessel is the Containment Polar Crane.

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: The only other handling system inside the containment capable of moving loads over the vessel is the Refueling Platform used for refueling operations. Its' function is to handle single fuel assemblies and perform other vessel servicing functions, i.e., no heavy loads as defined in NUREG-0612 are handled by this handling system.

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 I.

RESPONSE: As indicated in the response to Item 2.2.3 above, it has not been found to be necessary to evaluate the Polar Crane against the criteria of NUREG-0612, Section 5.1.6.

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:

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 specifications concerning the bypassing of such interlocks.

RESPONSE: The crane load blocks have not been included in any of the heavy load drop evaluations described in subsequent responses for the reasons given below:

NUREG-0612 requires that the load block and hook be considered as a heavy load. The load block is used for handling numerous loads, including the reactor vessel head, drywell head, steam dryer, and moisture separator. In moving these loads, the hook, load block, rope, drum, sheave assembly, motor shafts, gears, and other load bearing members are subjected to significant stresses approaching the load rating of the crane. By comparison, these components are subjected to a considerably smaller load when only the hook and load block are being moved. Based on this, it is not considered feasible to postulate a random mechanical failure of the crane load bearing components when moving either the main hoist or auxiliary hoist load block without a load.

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

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

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

The River Bend polar crane main hoist and auxiliary hoist are each provided with redundant and diverse upper limit switches to interrupt power to the hoist motor prior to "two blocking". When power is removed, holding brakes are automatically applied.

The holding brakes are solenoid released, and spring applied on loss of power to the solenoid. Two holding brakes are provided for each hoist on the polar crane; each holding brake has sufficient capacity to hold the rated load. Additionally, inspection procedures assure that the limit switches and holding brakes are functional and properly adjusted.

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

ITEM 2.3.4.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 validitiy of such considerations.

RESPONSE: Loads only lifted over the vessel when the reactor vessel head or moisture separator is in place were not considered as loads that could potentially drop into the core. These are: the drywell head and the steam dryer. No administrative controls are required to enforce this situation, because it is physically impossible to disassemble or reassemble the reactor such that these loads would be carried over an exposed core.

In addition, the portable refueling shield is installed in the reactor well after the head has been removed, but before the dryer or separator has been removed. It is removed from the reactor well after the dryer and separator have been installed. This sequencing is enforced by written procedures governing the installation and removal of the portable refueling shield and will be strictly enforced by individuals in charge of lifts by the Polar Crane.

ITEM 2.3.4.c Analyses performed to demonstrate compliance with Criteria I through III should conform to the guidelines of NUREG-0612, Appendix A. Justify any exception taken to these guidelines, and provide the specific information requested in Attachment 2, 3, or 4, as appropriate, for each analysis performed.

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

Analyses were performed to determine the structural consequences of dropping the vessel head or the shroud head assembly during maintenance operations. The consequences of dropping the steam dryer assembly can be extrapolated from the analysis of the shroud head assembly drop, since a steam dryer drop would generate less kinetic energy than the shroud head assembly drop and the impacted structure would be the same for both cases. The shroud head and dryer were assumed to be dropped from a height sufficient to generate the steady state velocity of the two assemblies as they move through water inside the vessel and are under the action of the fluid drag forces. An axisymmetric impact of the shroud head and dryer assembly on the main body of the shroud is assumed. In addition, nonaxisymmetric impact of the shroud head on the shroud was also considered. It was postulated that the vessel head would be dropped from a height of approximately 39 ft. above the vessel-head flange, and that at impact the head would be rotated 90° from the inplace orientation causing a point impact on the vessel. This height conservatively bounds the maximum possible carry height as limited by physical restrictions.

The vessel loads due to the postulated impacts were determined by dynamic, elastic-perfectly plastic finite element analyses. In the vessel head impact analysis the vessel was characterized by isoparametric quadrilateral elements using the ANSYS computer program.

The model reflects the longitudinal, lateral, extensional and inextensional effects. In the shroud head impact analysis the impacted shroud body is characterized by axisymmetric finite elements using the ANSYS computer program.

The results of the finite element analyses can be summarized as follows:

Drop of:	Consequences
Vessel Head	Local yielding of the vessel top flange Reactor skirt does not yield, does not buckle, and will remain stable. Vessel maintains its normal position. No damage to the fuel rods and hence no release of radioactive materials.
Shroud Head/Steam Dryer	No yielding of the upper shroud and shroud support struts. No instability of the shroud support structure. Damage to the internal components is minor and does not impact structural stability. No damage to the fuel rods and hence no release of radioactive materials.

In addition drops of the drywell head onto the vessel head and the portable refueling shield onto the separator (dryer conservatively not relied on to mitigate the drop) were evaluated by comparing the available drop energies to those in the analyses described above. Based on this comparison, these drops were found to be bounded by the head drop and dryer drop analyses.

On the basis of the analyses described above, it is concluded that NUREG-0612 Criteria I-III are met for all postulated drops into the reactor well.

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 that 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 2.1.3.3.

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., complete compliance with NUREG-0612, Section 5.1.6, or partial compliance supplemented by suitable alterantive 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 handling systems of interest specified in response to Item 2.1.1 are also listed below. It has not been necessary to evaluate any of these handling systems aginst the criteria of NUREG-0612, Section 5.1.6.

Handling System	Capacity (Tons)	Location
Reactor Building Polar Crane/Aux Hoist	100/5	Reactor Building
Drywell MSIV and Relief Valve Monorail	3	Reactor Building
Fuel Building Bridge Crane	15	Fuel Building
Spent Fuel Cask Trolley/Aux Hoist	125/15	Fuel Building
MSIV Monorails	8/5	Auxiliary Building
MSIV and Feedwater Isolation Valve Monorails	3	Auxiliary Building
Feedwater Valve Hoists	3	Auxiliary Building

Handling System	Capacity (Tons)	Location
RHR A Pump Monorail	8	Auxiliary Building
RHR B & C Pump Monorail	8	Auxiliary Building
Auxiliary Building Tunnel Plug Monorail	6	Auxiliary Building
Hoist Area Monorail	5	Control Building
Floor Plug Monorail	5	Control Building
Control Building Equipment Handling Area Monorail	5	Control Building

ITEM 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 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.

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).

(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.

c. For interactions not eliminated by the analysis 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.

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: Table 1 identifies the potential impact regions and the associated load handling systems. The loads corresponding to these handling systems were previously identified in response to item 2.1-3c. Figures 1 through 11 further describe the potential impact regions. For the handling systems identified in Table 1 (also listed in 2.4.1 above) a combination of systems and structural evaluations was utilized to determine if Criteria IV of NUREG-0612 is met for all postulated load drop scenarios. To assist these evaluations, a set of safety functions were identified corresponding to these criteria. The goal of these evaluations then became to demonstrate that the applicable safety functions could be accomplished for all load drop scenarios.

SYSTEMS EVALUATION METHODOLOGY

As part of the evaluation of heavy load handling operations at River Bend Station, a number of potential load drop regions in the Reactor, Auxiliary, Fuel and Control Buildings, were addressed by performing a "systems evaluation".

The objectives of the "systems evaluations" were to demonstrate that safe shutdown, long-term cooling, and fuel pool cooling could be achieved and/or maintained assuming that certain equipment was lost as a result of postulated load drops.

In order to demonstrate the ability to accomplish these objectives, it was necessary to:

1. Identify the load drop regions.
2. Identify the safety functions required to be accomplished for each region.
3. Identify the systems needed to accomplish the identified safety functions.
4. Identify the equipment associated with these systems, including support systems, that could potentially be lost if a load drop were to occur in the region.
5. Determine the effects of loss of this equipment on the ability to accomplish the identified safety functions.

SAFETY FUNCTIONS

The systems evaluations address the capability to perform the following safety functions:

1. Spent Fuel Cooling - upper Reactor Building fuel pool and Fuel Building spent fuel pool.
2. Extended Core Cooling.
3. Reactor Shutdown and Cooldown to extended core cooling.

To determine which safety functions are applicable to each load drop region, initial plant conditions were assumed during load handling operations as discussed below.

Spent Fuel Cooling

Load handling operations in any region could take place when spent fuel is stored in the Fuel Building spent fuel pool and the Reactor Building fuel pool. Therefore the capability to perform spent fuel cooling in these storage locations must be verified for postulated load drops in all regions subject to systems evaluations. The reactor building fuel pool contains spent fuel racks capable of storing up to 32% of a full core. The Fuel Building spent fuel pool has a capacity of 525% of full core. For the purpose of evaluating whether or not this safety function could be accomplished, heavy load drops were postulated to occur when there was recently discharged irradiated fuel in the containment spent fuel storage racks and in the Fuel Building spent fuel storage racks.

Extended Core Cooling

Load handling operations in any region could occur with fuel in the reactor vessel. Therefore, for all postulated load drops, the capability for extended core cooling must be verified.

Reactor Shutdown and Cooldown to Extended Core Cooling

It is not anticipated that any of the heavy loads handled by the Containment Polar Crane or the Drywell MSIV and Relief Valve Monorail would be lifted until the plant has been shut down for some time. Accordingly, all heavy load drops inside the Reactor Building were postulated to occur when the reactor was shut down and cooled down. This condition corresponds to either the Cold Shutdown or Refueling Operational Conditions. Load handling operations in the Auxiliary, Fuel, and Control Buildings could take place during any plant operational condition. Accordingly, the capability to perform this safety function must be verified for postulated load drops in regions 10 through 18, 20 and 21.

SYSTEM REQUIREMENTS

For the purpose of performing the systems evaluations described in this appendix, only "safety systems" were selected to accomplish each of the safety functions of interest. This means that many of the systems used to accomplish normal plant shutdown, such as offsite power and condensate, feedwater and circulating water systems, were conservatively not relied on to perform the systems evaluations.

Further, although there are several combinations of safety systems that may be used for safe shutdown, initially only certain systems were assumed to be available. Additional systems, or portions of systems, were included in the review only if it was determined to be necessary or prudent to do so.

Cooling of spent fuel in the Fuel Building and/or the upper containment fuel pool during maximum postulated heat loads can be accomplished by the Fuel Pool Cooling and Cleanup (FPCCU) system. The fuel pool cooling subsystem of the FPCCU consists of two 100-percent capacity pumps and coolers, and associated piping, valves, and instrumentation. Normal cooling of the FPCCU system coolers is provided by Reactor Plant Component Cooling Water (RPCCW). Upon loss of RPCCW pressure, and as assumed in these systems evaluations, the Standby Service Water (SSW) system provides cooling to the FPCCU heat exchangers. SSW uses the RPCCW system piping which is isolated from the balance of the RPCCW system by redundant (Division 1 and 2) isolation valves. Although one loop of the FPCCU system can provide adequate spent fuel cooling during maximum design basis heat loads, both FPCCU loops are normally used during reloads. When fuel is in the containment and Fuel Building pools (e.g. reload) one loop of the FPCCU systems is aligned to the Fuel Building and the other loop to containment. If either FPCCU loop fails the operable FPCCU loop is aligned to the Fuel Building and the standby RHR system loop is aligned to containment.

Figure 12 displays the systems, selected for evaluation purposes, to perform the Reactor Shutdown, Cooldown, and Extended Core Cooling functions. Evaluation of the systems included all necessary support systems. Performance of the safety functions by other safety related systems was considered when required by the potential effects of the postulated load drops.

LOAD DROP MATRICES

For each safety function, a matrix was developed that outlined each of the regions to be evaluated and the critical system components relied on to perform the safety function. The critical components include required support functions such as cooling water, power supplies, and electrical cabling.

Each region of concern was evaluated to determine which of the system components of interest could be lost as a result of a load drop. The evaluation was performed by conducting plant walkdowns and reviewing plant drawings, system descriptions and fire hazards evaluations. This information was then entered into the matrices.

STEPS IN THE SYSTEMS APPROACH

The following summarizes the steps that were performed in the systems evaluations for each safety function evaluated:

Completion of Matrix

(i) Identify the system (including any support systems) components selected for accomplishing the safety function of interest, organize into component groups for purposes of evaluation, and enter at top of matrix columns.

(2) For each region of concern, evaluate the potential for damage/loss of system components based on a detailed-review of the equipment and piping layouts, electric cable locations and results of fire hazards analysis electrical review of the region. Enter the components assumed lost in the appropriate box on the matrix.

(3) Compare system equipment required (Item 1), with equipment lost (Item 2), and determine if the safety function for which the system is relied on could be lost.

(4) Review for other potential system interactions based on equipment damaged/lost and determine if safety function could be lost.

Conclusions

(5) If the system evaluation reveals that the system could accomplish its safety function following a load drop into the region of interest, then no further evaluation is necessary.

(6) If the system evaluation reveals that the system function could potentially be lost, then evaluate the possibility of relying on alternative safety systems to accomplish the same function following a postulated load drop into the region.

(7) The overall safety function conclusion regarding a particular region is the composite for that region of the conclusions for all the systems required to accomplish the safety function.

STRUCTURAL EVALUATION METHODOLOGY

Structural load drop analyses performed to support evaluations related to safety functions described above, typically involved determination of structural response of concrete floor slabs to dynamic impact loadings. The heavy loads which could potentially be dropped onto various floor slabs were evaluated to identify loads which control local response (e.g. penetration, scabbing, spalling, perforation, etc.); loads that control overall 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.

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 perforation. 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, it was decided that the consequences of scabbing would be considered in the systems evaluations, i.e., its potential for damaging equipment below the floor was evaluated.

Where the controlling mode of response is listed as "overall structural", these load drops were evaluated to determine the potential for producing gross and intolerable distortions of primary structural members and possibly propagating failures.

Overall Structural Response Evaluations

A model of each fl[illegible]ation was developed with the objective of evaluating structural behavio[illegible]stulated 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 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 stiffness properties of the supporting beam grid, where applicable, were represented assuming an effective reinforcement for the beam/slab composite section consistent with ACI 318-77 (Reference 1).

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 (Reference 1).

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 criteria to set the maximum allowable level of deflection for the slab/grid system. The hinge rotational capacity for concrete structures was developed in References 2 and 3 based on test results given in References 4 and 5 and is given as:

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

where,

r_u = rotational capacity of plustic 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 (1) 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 reflect'on 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 allowcble 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 11 of ACI 318-77 (Reference 1).

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, References 6 and 7 recommend dynamic increase factors of 1.25 for the compressive strength of concrete and 1.20 for the flexural, tensile and compressive strength of structural steel.

2) Design (minimum) material properties for 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 (Reference 8) of increased strength. Also, the average strength for structural steel is nearly a factor of 1.25 (Reference 9) 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 propgating failures), larger rotations in the range of 5 to 12 degrees could be tolerated. Such rotations would lead to crushing, spalling and scabbing of the section (Reference 7); however overall load carying 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 Vf'c to 20 Vf'c (References 11-19). It is expected that the shear capabilities for these beams would tend to be in the higher end of the range since the majority of the beams are "deep". Deep beams behave as tied-arches with significant reserve capacity.

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 shield plug. 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 shield plug or the immediate surface of the floor.

Local Structural Response Evaluations

Selected loads such as the portable radiation shield (cattle chute), various hatch covers and 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.

Procedures recommended in References 20 and 21 were followed. The modified National Defense Research Committee (NDRC) formula (Reference 22) was chosen because it has been shown to give the best fit with available experimental data (References 23 and 24). The NDRC formula for the depth of penetration, x (inches), of a solid cylindrical missile is given by:

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

or

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

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 are given by the following Army Corps of Engineers formulae which can be used in conjunction with equations (2) and (3) (Reference 25).

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

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

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 (Reference 20) giving,

$$ts/d = 7.91\,(x/d) - 5.06\,(x/d)^2 \quad \text{for} \quad x/d \leq 0.65 \tag{6}$$
$$tp/d = 3.19\,(x/d) - 0.718\,(x/d)^2 \quad \text{for} \quad x/d \leq 1.35 \tag{7}$$

A 10 percent margin on thickness has been applied in the use of equations (6) thru (7) as recommended in Reference 20.

Limited penetration and scabbing was predicted for the set of bounding heavy load drops considered; however, in no case were the concrete slabs predicted to be perforated.

OVERALL RESULTS AND CONCLUSIONS

Table 2 provides the overall results of the systems evaluations. The term "OK" is used where the systems evaluations demonstrated sufficient redundancy and separation of equipment to assure the capability to perform the required safety function. As indicated, the capability to perform spent fuel cooling, extended core cooling and safe shutdown, is maintained for postulated load drops in most regions.

The term "Potential Problem" is used to indicate those situations for which additional evaluations and/or restrictions were necessary to assure the capability to perform the required safety functions. These situations are discussed further in the sections below.

1. Regions 2, 7, and 20

The potential problems in the above regions all involve the loss of spent fuel cooling. In Regions 2 and 7 the loss of cooling capability can be limited to the containment fuel pool by operable isolation valves thereby leaving the Fuel Building spent fuel cooling intact. For load drops in Region 20 there is the potential for loss of spent fuel cooling to both the Fuel and Reactor Building pools.

The overall result of these system losses was judged to be acceptable, however, based on the following rationale. If both FPCCU and RHR cooling to the Reactor Building fuel pool and/or FPCCU cooling to the Fuel Building spent fuel pool were lost, hoses could be routed to the pools to provide makeup from any available water source. Therefore, spent fuel in the pools would always remain covered with water. Makeup would only be necessary if boiloff from the pool were to occur. Whether or not boiloff would occur, and when, would be highly dependent on the amount, power history, and decay history of spent fuel in the storage racks; however, even under the most limiting heat load and water level conditions several hours would be available to provide an alternative water source. As long as makeup can be provided to the pool, there will be no spent fuel damage and, therefore, no off-site dose consequences of significance. For this reason, it is judged that there is reasonable assurance that Safety Function I, spent fuel cooling, can be accomplished following all postulated load drops.

2. Regions 13 and 14

In a limited portion of the load handling areas of these regions, there is the potential for a dropped load to damage RHR piping that is nonisolable from the suppression pool. Since the subject piping is in proximity to the valves whose maintenance may involve use of the monorail it is not realistic to restict the load path to satisfy this concern. The potential for opening a nonisolable vent path from the suppression pool to the Auxiliary Building is a concern when reactor coolant discharge (depressurization) to the suppression pool is necessary. Since such discharge is not necessary when the plant is in Cold Shutdown or Refueling, load handling operations should be restricted to those Operational Conditions. The potential effects of a load drop in these regions are acceptable with respect to the Extended Core Cooling and Spent Fuel Cooling safety functions. Based on the above, heavy load handling in Regions 13 and 14 will be restricted to when the plant is in Cold Shutdown or Refueling to resolve this potential problem.

3. Region 21

Based on the systems evaluation in Region 21 the consequences of a cask drop would be unacceptable due to the potential for damaging both Divisions of Standby Service Water. However, we have performed a cask drop analysis for this Region. The results of that analysis indicate that the effects of a cask drop in the washdown area (area above the tunnel with safe shutdown equipment) are limited to localized spalling of the concrete which will be contained by metal decking installed below the washdown area floor. The structural analysis requires that the cask be carried at a height (bottom of cask) no greater than 6 inches above Fuel Building Elevation 113'-0". With this carry height limit imposed the results of a heavy load drop in this area are acceptable.

4. Additional Administrative Requirements

In addition to the heavy load handling requirements discussed in Sections 2 and 3 above, lift height restrictions for loads handled by the Auxiliary Building Tunnel Plug Monorail were required as part of the Region 16 systems evaluation. A structural evaluation was performed to determine limits necessary to assure that a postulated load drop would not cause structural damage to the 95' elevation floor. Based on that evaluation loads up to 2 tons may be carried no greater than 12 feet and loads up to 6 tons no greater than 5 feet above the 95' elevation floor. All limits necessary as a result of the heavy loads handling evaluation will be imposed administratively by procedures.

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TABLE I
LOAD DROP REGIONS

REGION	DESCRIPTION	HANDLING SYSTEMS
1	Reactor vessel and area above reactor vessel. RB Elevations 70' through 186'3".	Polar Crane
2	Upper containment fuel storage pool and dryer storage area. RB Elevations 146' through 186'3".	Polar Crane
3	Reactor building hoist area including level below hoist area floor. RB Elevations 70' through 186'3".	Polar Crane
4	Reactor building - east side. RB Elevations 162'3" through 186'3".	Polar Crane
5	Separator storage area (north). RB Elevations 162'3" through 186'3".	Polar Crane
6	Separator storage area (south). RB Elevations 137'3" through 186'3".	Polar Crane
7	Reactor building - west side. RB Elevations 162'3" through 186'3".	Polar Crane
8	Containment drywell - east side. RB Elevations 70' to 154'7".	Drywell MSIV and Relief Valve Monorail
9	Containment drywell - west side. RB Elevations 70' to 154'7".	Drywell MSIV and Relief Valve Monorail
10	Main Steam line space. AB Elevations 114' to 161'3".	MSIV Monorails
11	RHR B transfer spaces. AB Elevations 70' to 134'6"	RHR B&C Pump Monorail

TABLE I
(Continued)

REGION	DESCRIPTION	HANDLING SYSTEMS
12	RHR A transfer spaces. AB elevations 70' to 134'6".	RHR A Pump Monorail
13	Mainsteam/Feedwater piping area - east side. AB Elevations 95'9" to 133'.	MSIV and FWIV Monorail FW Valve Hoist
14	Mainsteam/Feedwater piping area - west side. AB Elevations 95'9" to 133'.	MSIV and FWIV Monorail FW Valve Hoist
15	RHR C transfer space. AB Elevations 70' to 105'6".	RHR B&C Pump Monorail
16	Auxiliary Building Tunnel access region. AB Elevations 70' to 108'.	Aux. Bldg. Tunnel Plug Monorail
17	Control Building equipment handling region. CB Elevations 70' to 156'.	Hoist Area Monorails
18	Control Building Tunnel access region. CB Elevations 70' to 115'.	Floor plug Monorail
19	Spent fuel storage pool - west end. FB Elevations 70' to 138'.	Fuel Building Bridge Crane
20	Fuel Building region west of SFP. FB Elevations 70' to 138'.	Fuel Building Bridge Crane
21	Cask handling area. FB Elevations 70' to 151'.	Spent Fuel Cask Trolley

TABLE 2

OVERALL RESULTS

Region	Safety Function	1 Spent Fuel Cooling	2 Extended Core Cooling	3 Safe Shutdown	Resolution (Section #)
Reactor Building	2	Potential Problem	OK	N/A	1
	3	OK	OK	N/A	
	4	OK	OK	N/A	
	5	OK	OK	N/A	
	6	OK	OK	N/A	
	7	Potential Problem	OK	N/A	1
	8	OK	OK	N/A	
	9	OK	OK	N/A	
Auxiliary Building	10	OK	OK	OK	
	11	OK	OK	OK	
	12	OK	OK	OK	
	13	OK	OK	Potential Problem	2
	14	OK	OK	Potential Problem	2
	15	OK	OK	OK	
	16	OK	OK	OK	4 *
Control Building	17	OK	OK	OK	
	18	OK	OK	OK	
Fuel Building	20	Potential Problem	OK	OK	1
	21	Potential Problem	Potential Problem	Potential Problem	3

* Discusses necessary lift height restrictions

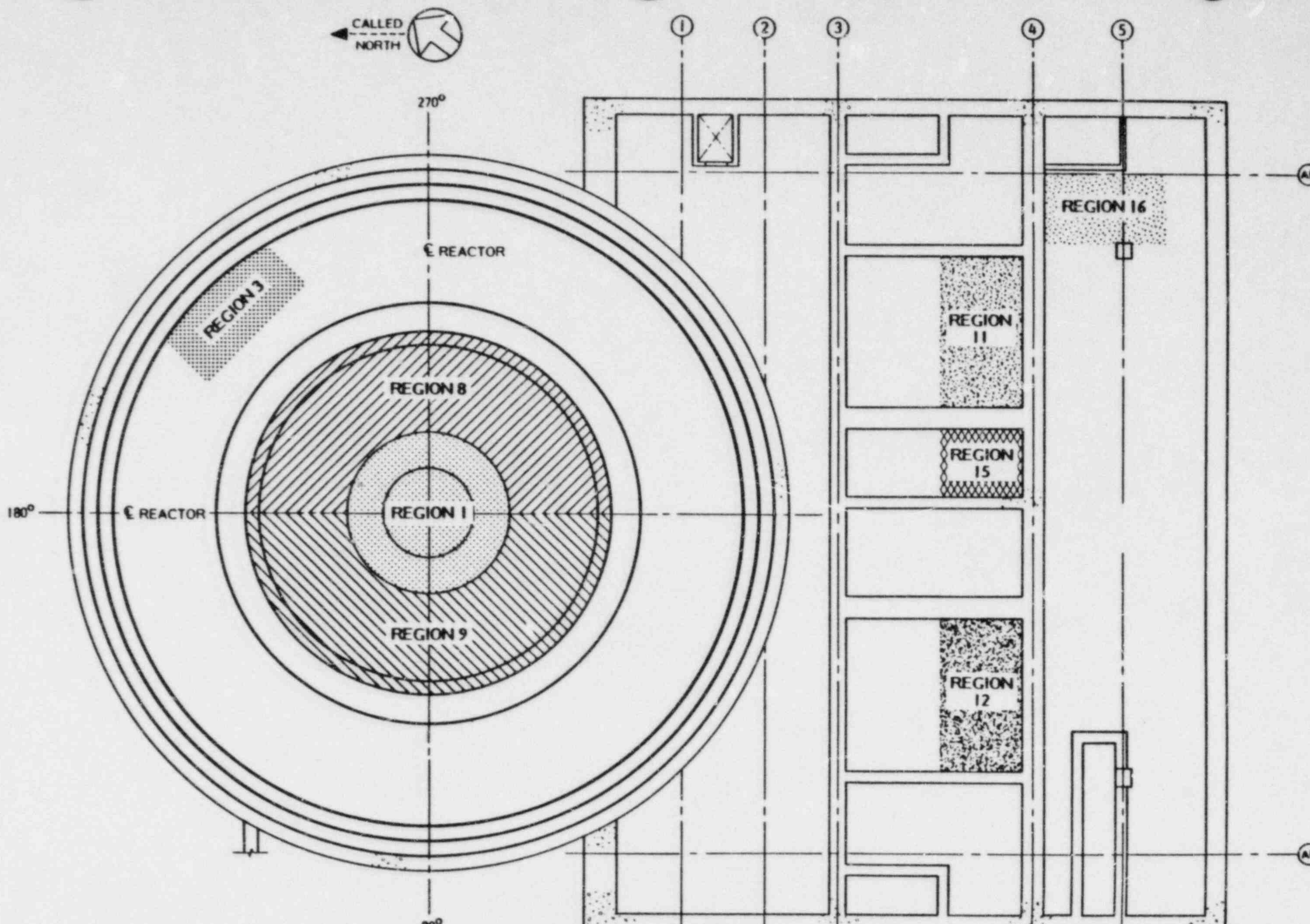

FIGURE 1

RIVER BEND STATION
HEAVY LOAD DROP IMPACT REGIONS
REACTOR BUILDING/AUXILIARY BUILDING EL 70'-0"

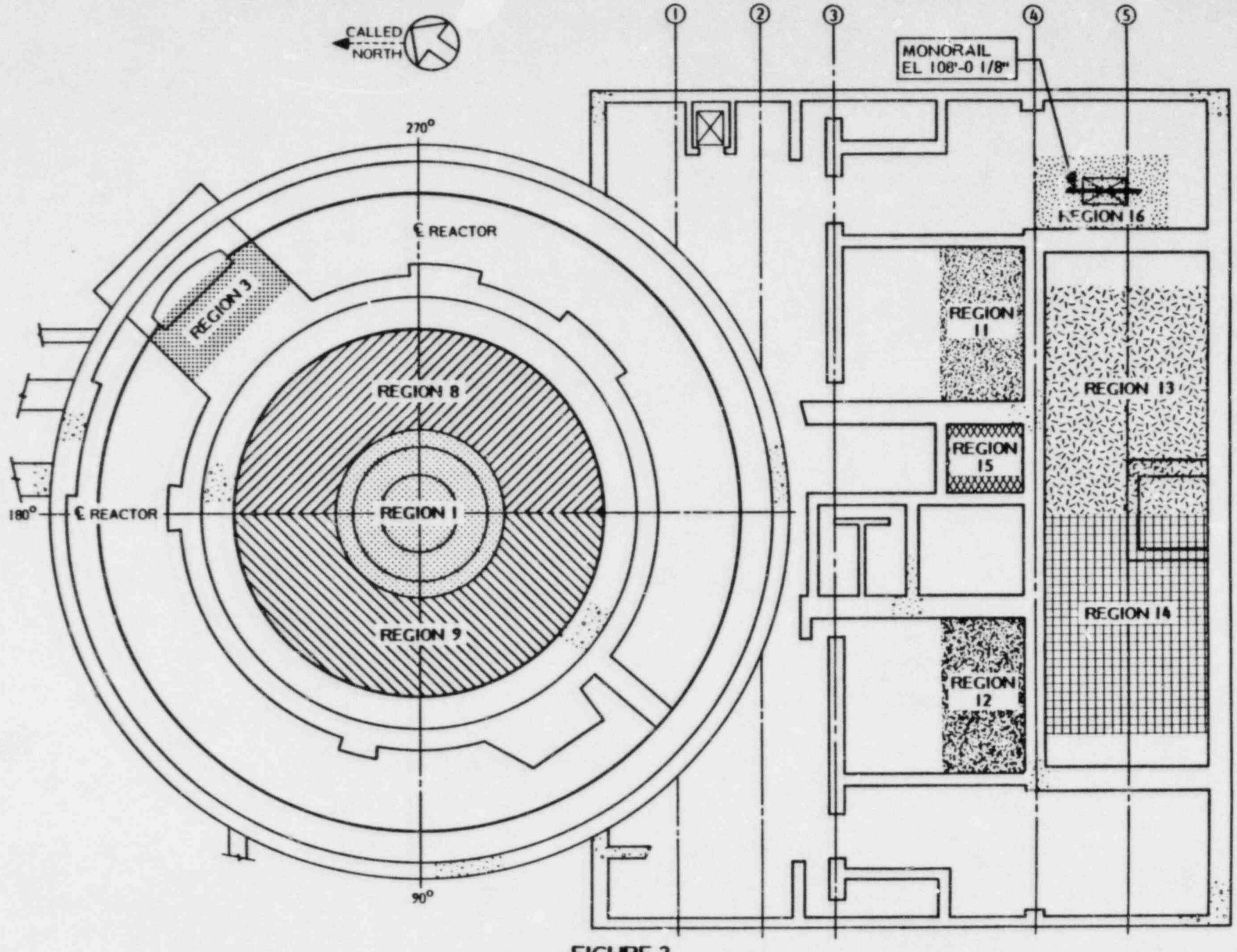

FIGURE 2

RIVER BEND STATION
HEAVY LOAD DROP IMPACT REGIONS
REACTOR BUILDING/AUXILIARY BUILDING, EL 95'-9"

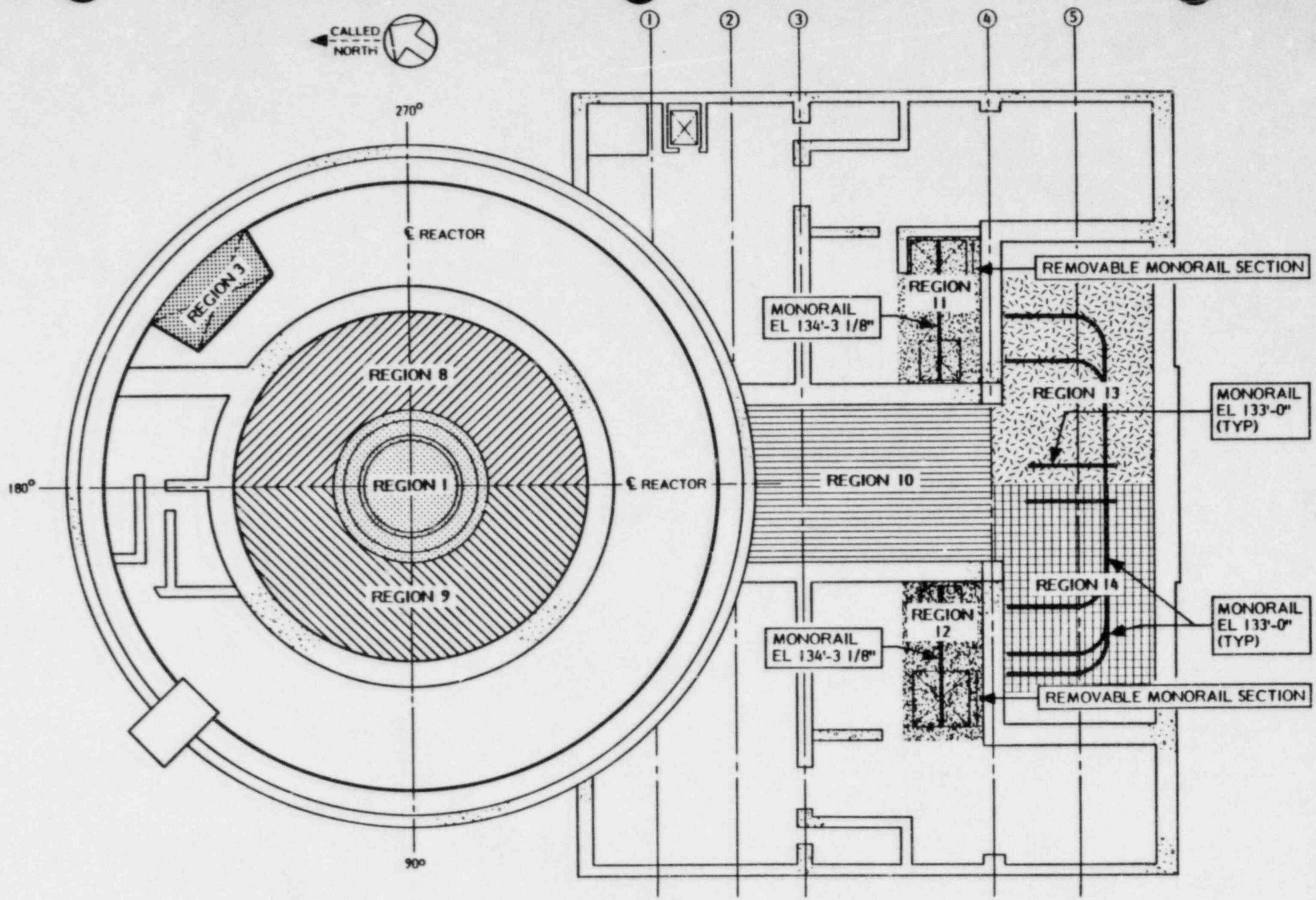

FIGURE 3

RIVER BEND STATION
HEAVY LOAD DROP IMPACT REGIONS
REACTOR BUILDING/AUXILIARY BUILDING, EL 114'-0"

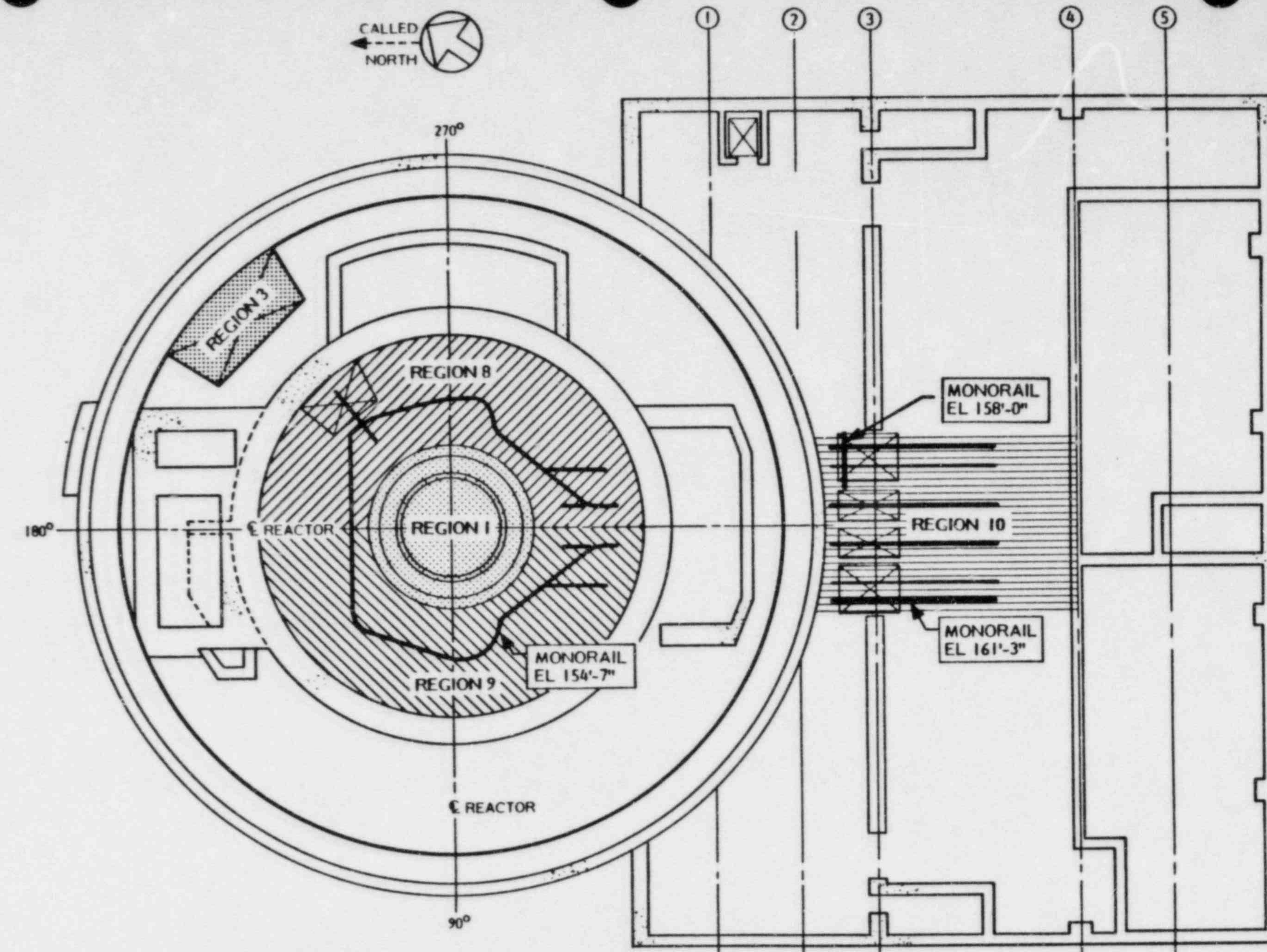

FIGURE 4

RIVER BEND STATION
HEAVY LOAD DROP IMPACT REGIONS
REACTOR BUILDING/AUXILIARY BUILDING, EL 141'-0"

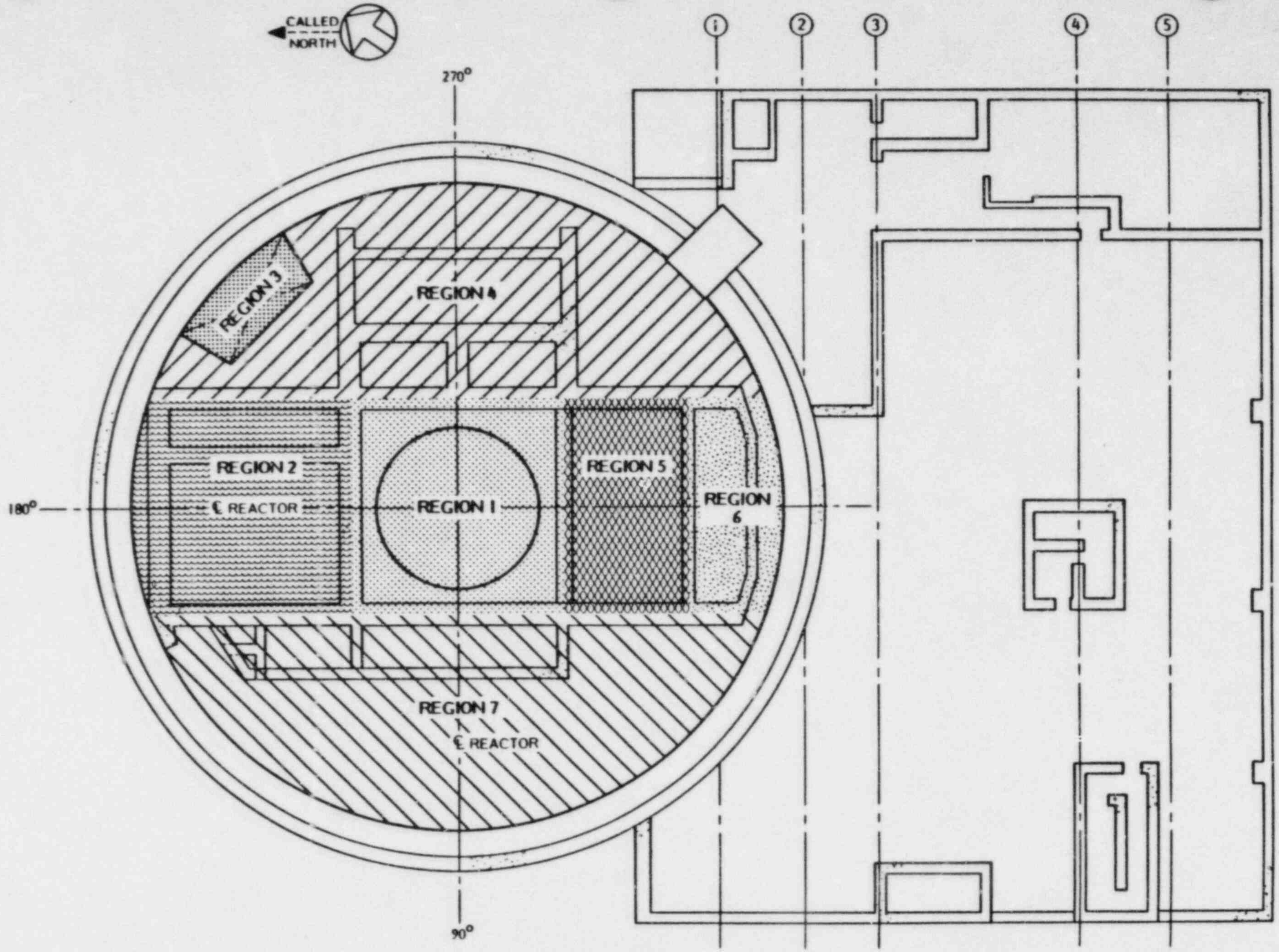

FIGURE 5

RIVER BEND STATION
HEAVY LOAD DROP IMPACT REGIONS
REACTOR BUILDING/AUXILIARY BUILDING, EL 162'-3"

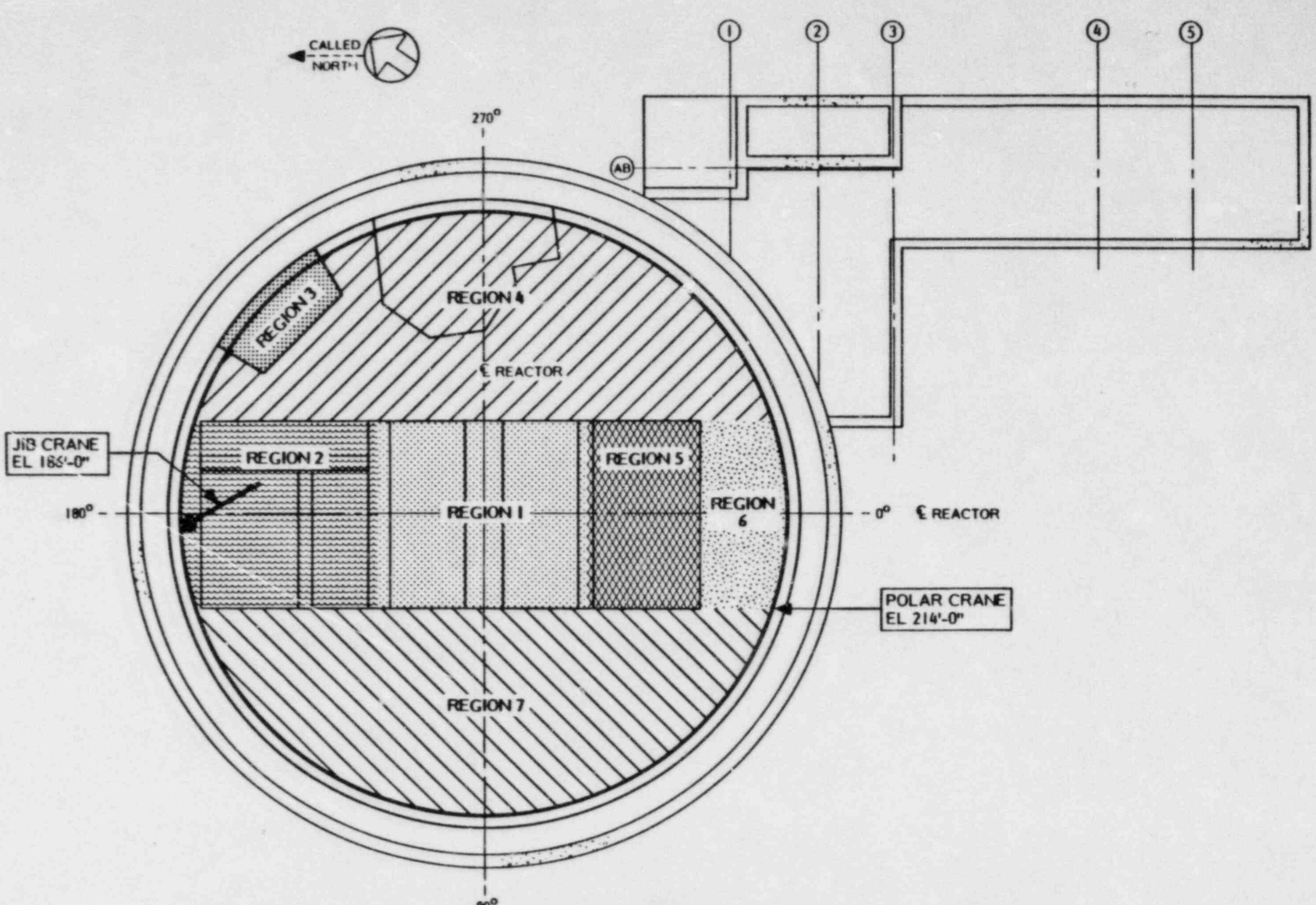

FIGURE 6

RIVER BEND STATION
HEAVY LOAD DROP IMPACT REGIONS
REACTOR BUILDING/AUXILIARY BUILDING, EL 186'-3"/EL 185'-0"

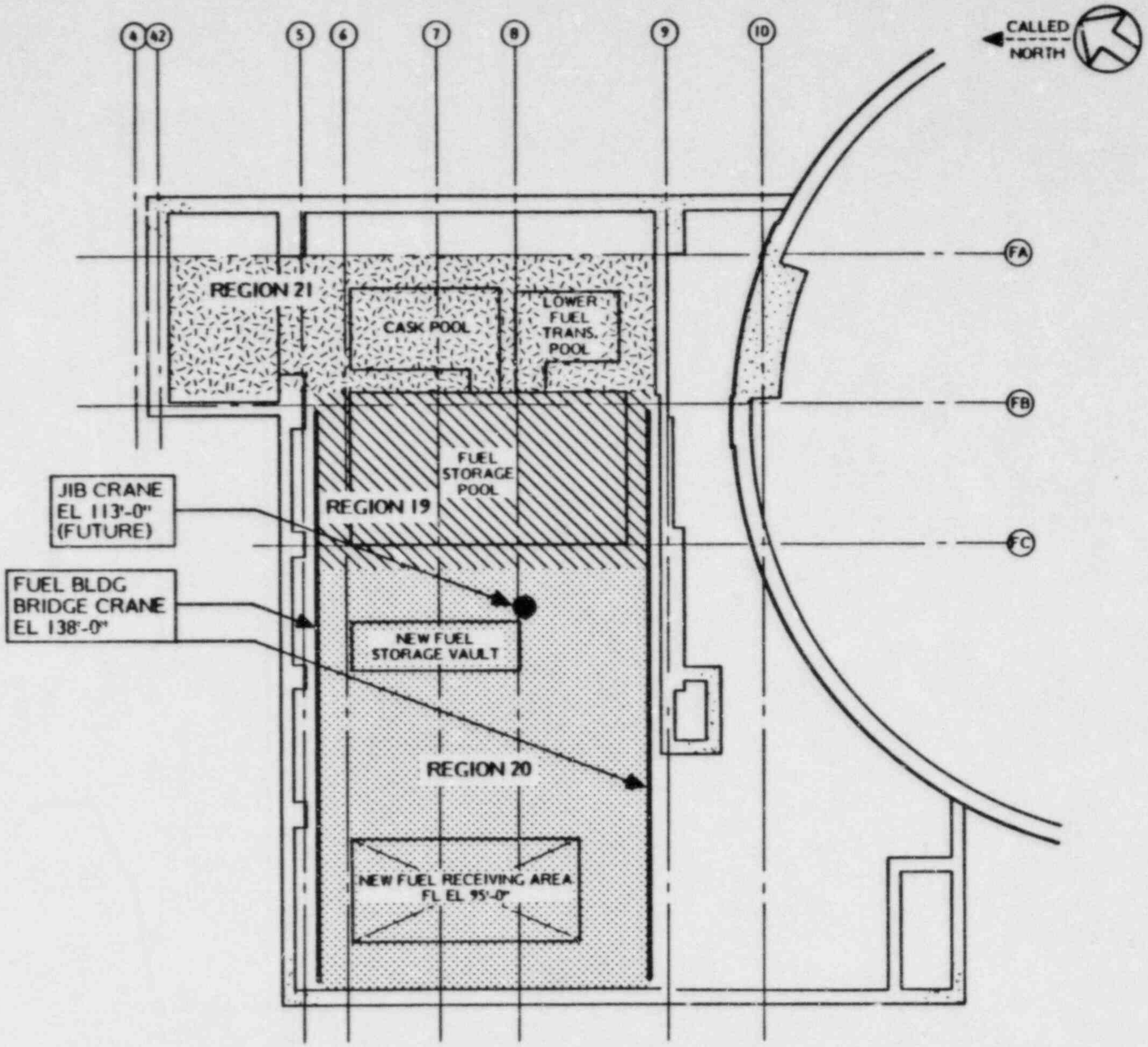

FIGURE 7

RIVER BEND STATION
HEAVY LOAD DROP IMPACT REGIONS
FUEL BUILDING, EL 113'-0" AND BELOW

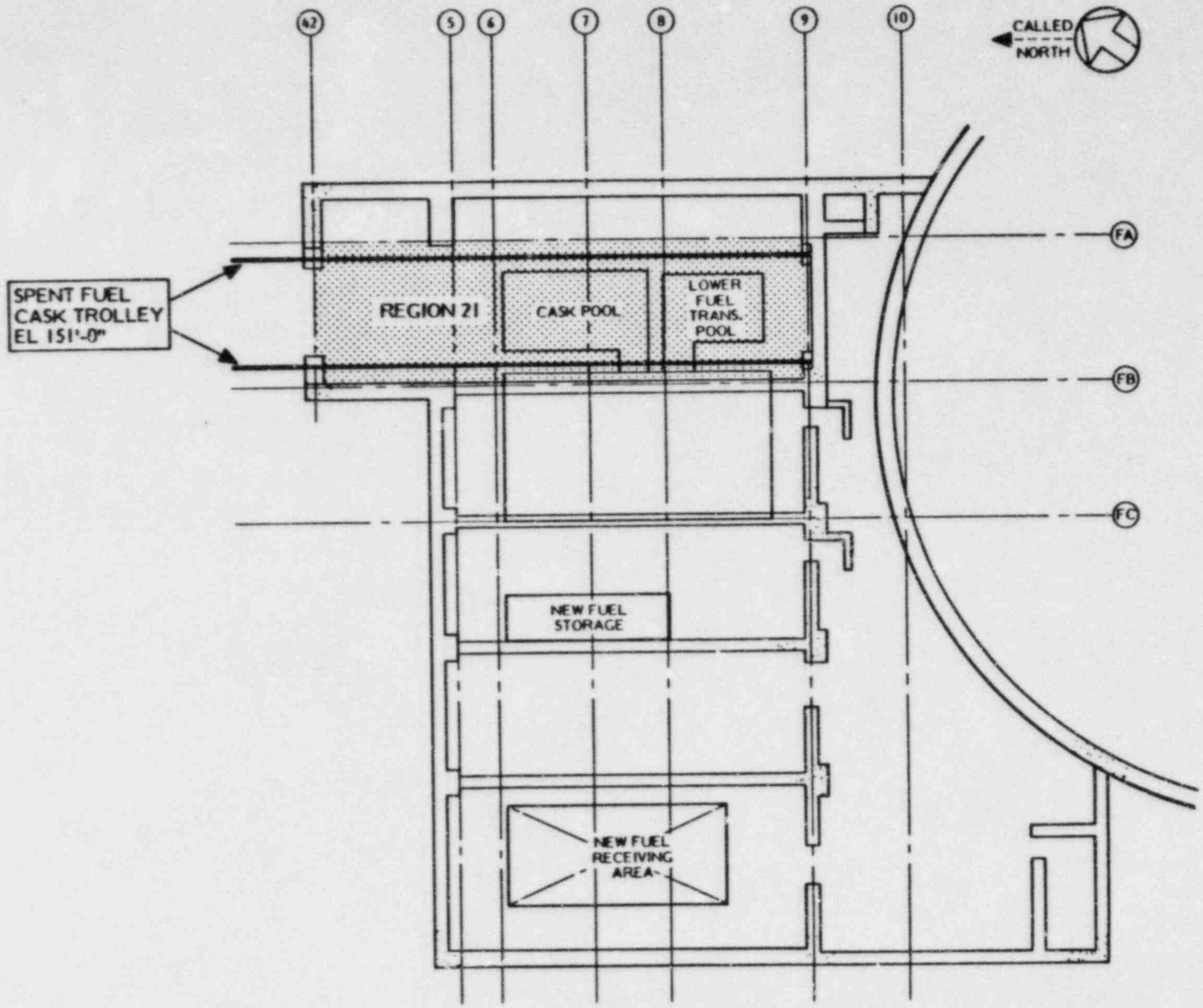

FIGURE 8

RIVER BEND STATION
HEAVY LOAD DROP IMPACT REGIONS
FUEL BUILDING, EL 148'-0"

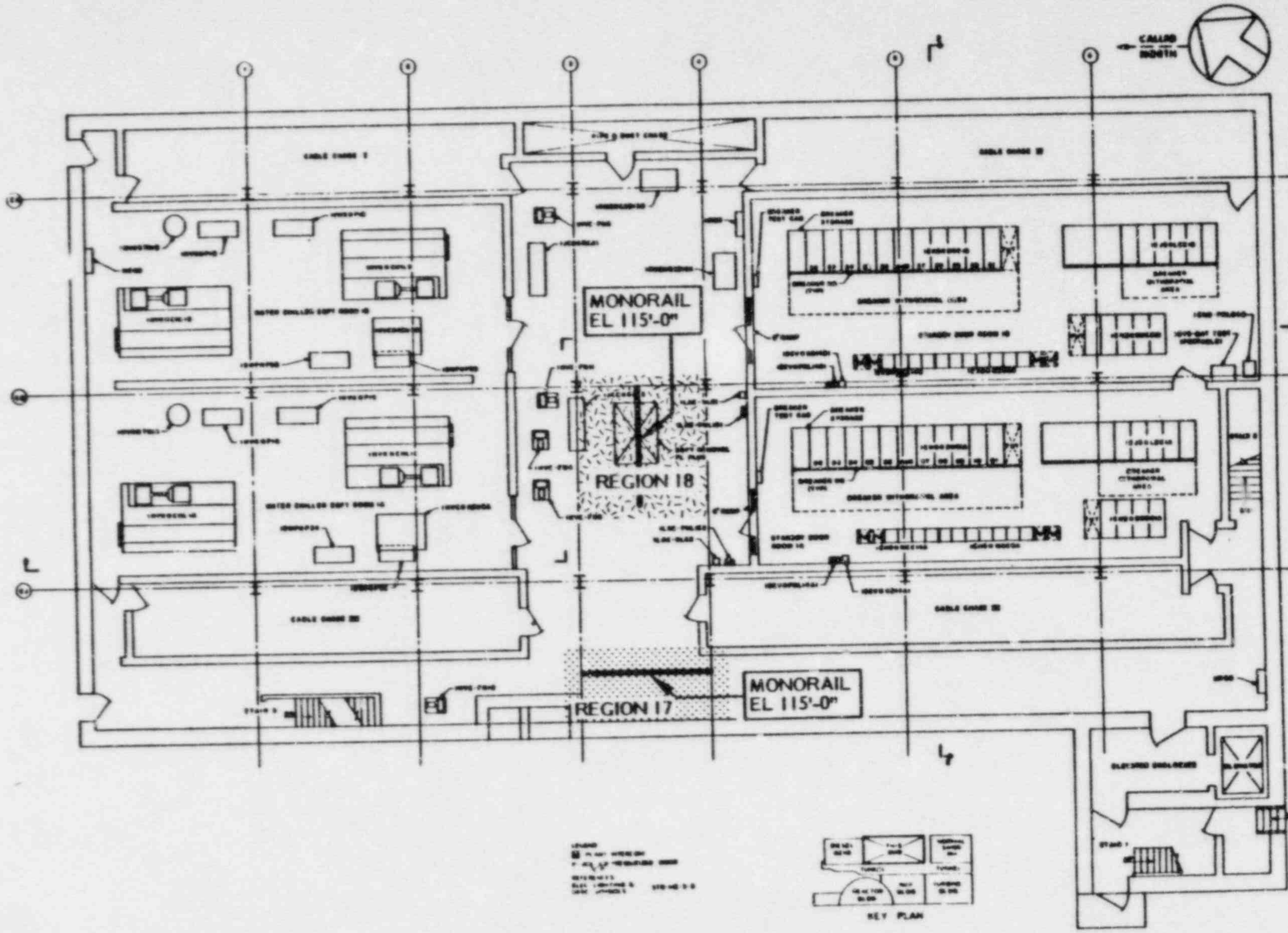

FIGURE 9

RIVER BEND STATION
HEAVY LOAD DROP IMPACT REGIONS
CONTROL BUILDING, EL 98'-0" AND EL 70'-0"

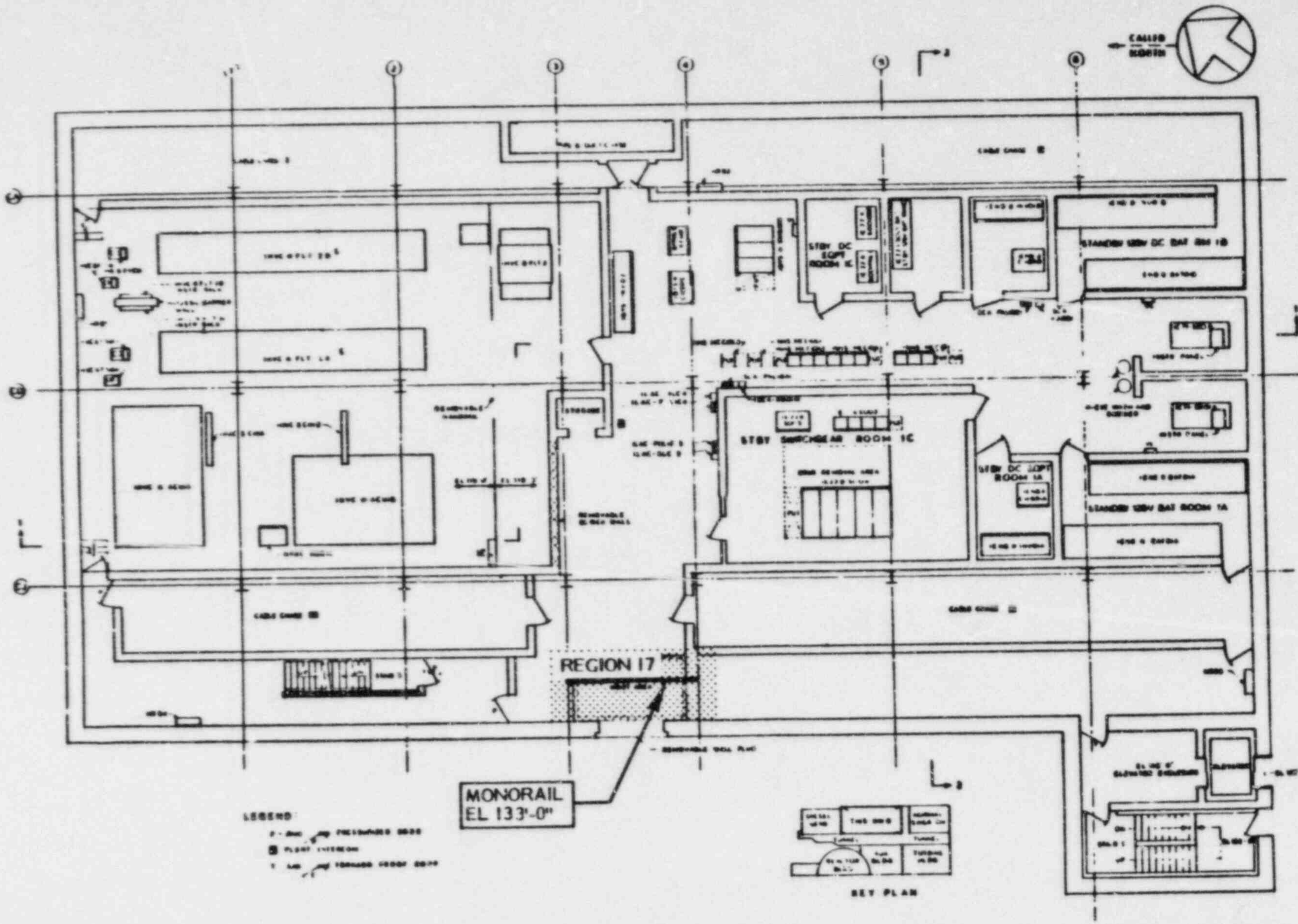

FIGURE 10

RIVER BEND STATION
HEAVY LOAD DROP IMPACT REGIONS
CONTROL BUILDING, EL 115'-0"

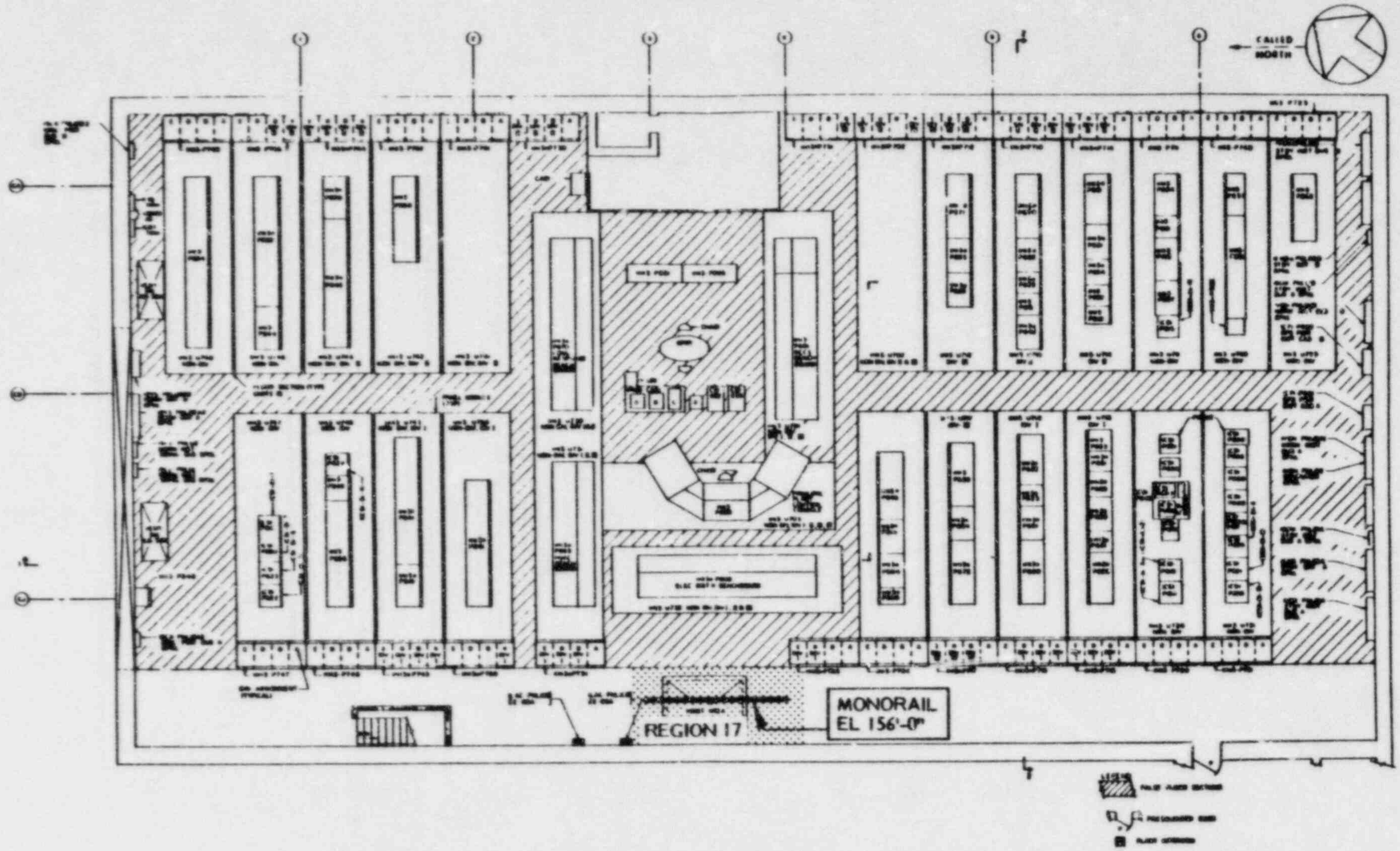

FIGURE 11

RIVER BEND STATION
HEAVY LOAD DROP IMPACT REGIONS
CONTROL BUILDING, EL 136'-0"

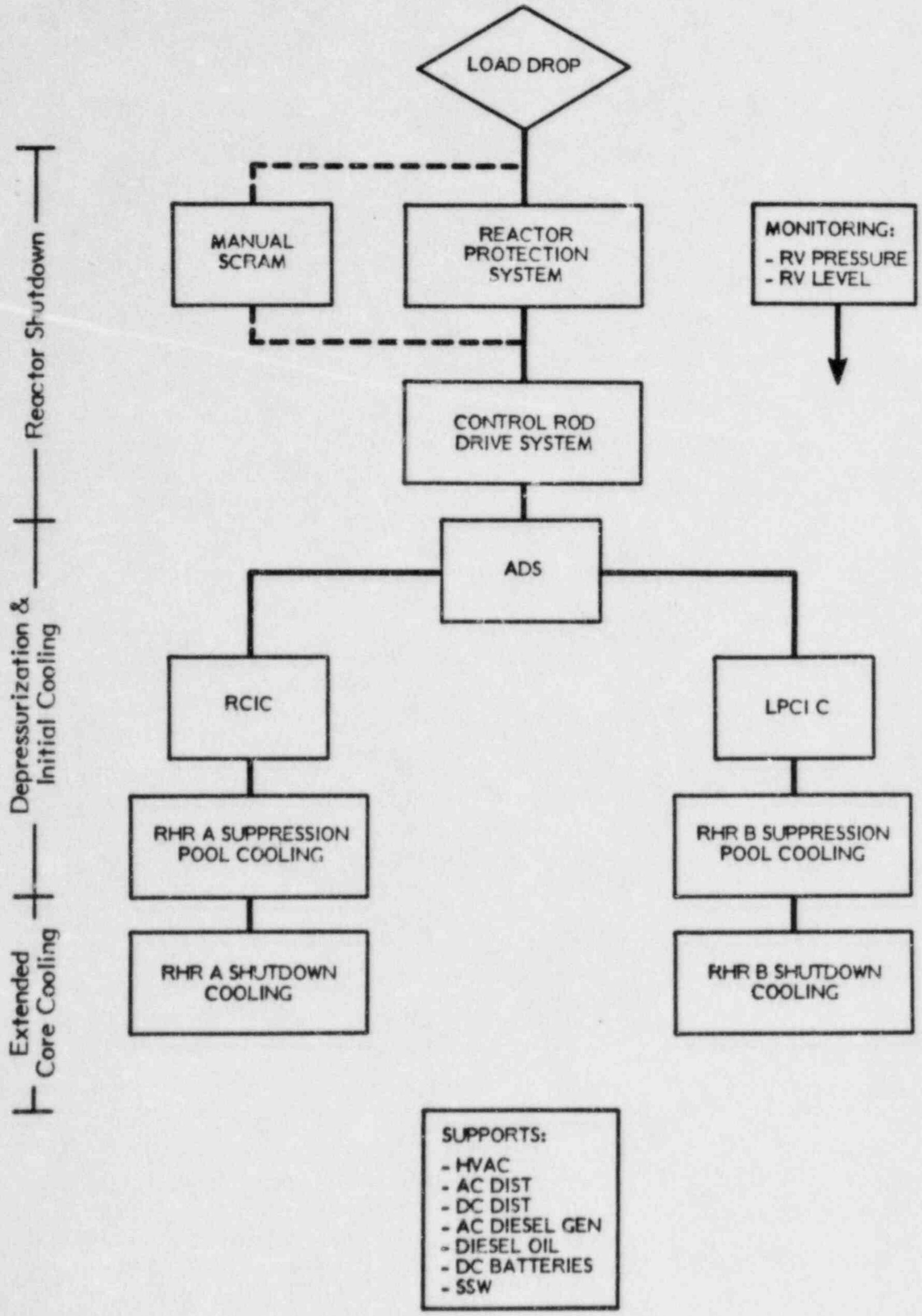

FIGURE 12

SAFE SHUTDOWN SYSTEMS