John D. O'Toole
Vice President

Consolidated Edison Company of New York, Inc.
4 Irving Place, New York, NY 10003
Telephone (212) 460-2533

December 3, 1981

Re: Indian Point Unit No. 2
Docket No. 50-247

Director of Nuclear Reactor Regulation
U. S. Nuclear Regulatory Commission
Washington, D. C. 20555

ATTN: Mr. Darrell G. Eisenhut, Director
Division of Licensing

Dear Mr. Eisenhut:

Your letter dated July 31, 1980, requested a review of the controls for handling heavy loads at Indian Point 2, the implementation of certain recommendations regarding these controls, and the submittal of information to demonstrate that the recommendations have been implemented. This first submittal was transmitted to the NRC by letter dated June 22, 1981. This letter transmits the second submittal.

We have conducted the second phase of the requested review and the additional information requested to address the applicable items in Sections 2.2, 2.3, and 2.4 of Enclosure 3 to your July 31, 1980 letter is provided in the Enclosure to this letter. Any changes or modifications that have been or will be completed to satisfy the guidelines of NUREG-0612 have been identified in the Enclosure.

In your June 22, 1981 report, we had indicated that it was necessary to confirm that certain criteria in ANSI N14.6-1978 were met for the Reactor Vessel Head and Internals Lifting Rigs. We have reevaluated our previous response (Item 3d) on these devices and have determined that the information provided is sufficient to demonstrate adequate load handling reliability for these devices.

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This submittal completes our responses to all requests for information in your July 31, 1980 letter.

Should you or your staff have any questions, please contact us.

Very truly yours,

John D. O'Toole

John D. O'Toole
Vice President

Encl.

50-247

CONTROL OF HEAVY LOADS
Second Submittal

Received wth ltr dtd 12/03/81

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Enclosure

Control of Heavy Loads
Second Submittal

Consolidated Edison Company of New York, Inc.
Indian Point Unit No. 2
Docket No. 50-247
December, 1981

RESPONSES TO REQUESTS FOR INFORMATION IN SECTIONS 2.2, 2.3, AND 2.4 OF ENCLOSURE 3 TO NRC JULY 31, 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-handling systems in the vicinity of stored, spent fuel. Information provided in response to this section should demonstrate that adequate measures have been taken to ensure that in this area, either the likelihood of a load drop which might damage spent fuel is extremely small or 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.

RESPONSE: For the reasons given in the response to Item 3 of our June 22, 1981 submittal, the Spent Fuel Storage Building crane has been excluded from consideration until such time as a decision is made regarding a spent fuel shipping cask. Currently, no heavy loads are routinely carried within the vicinity of the spent fuel pool.

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 Gantry Crane. The crane was designed by Whiting Corporation and possesses a 175-ton main hoist and a 35-ton auxiliary hoist.

ITEM 2.3.2. Justify the exclusion of any cranes in this area from the above category by verifying that they are incapable of carrying heavy loads, or are permanently prevented from the movement of any load either directly over the reactor vessel or to such a location where in the event of any load-handling-system failure, the load may land in or on the reactor vessel.

RESPONSE: The Manipulator Crane used for refueling operations. It is sized to handle single fuel assemblies, i.e., no heavy loads as defined in NUREG 0612 are handled by this handling system. In addition, four jib cranes have recently been installed to service each of the Reactor Coolant Pumps. These jib cranes have a 2-ton capacity and are used to lower and raise tools and small parts for performing maintenance on the pumps. Any load drops by these jib cranes would be bounded by postulated drops of a Reactor Coolant Pump motor (see Region 6 evaluation in response to Item 2.4.2.b(1).

ITEM 2.3.3. Identify any cranes listed in 2.3.1 above which you have evaluated as having sufficient design features to make the likelihood of a load drop extremely small for all loads to be carried and the basis for this evaluation (i.e., complete compliance with NUREG 0612, Section 5.1.6, or partial compliance supplemented by suitable alternative or additional design features). For each crane so evaluated, provide the load-handling-system (i.e., crane-load-combination) information specified in Attachment 1.

RESPONSE: A probabilistic failure analysis of the Polar Crane has been performed applicable to removal and installation of the Reactor Head and the Upper Internals. Drops of these two components are controlling with respect to evaluating the potential for damaging the vessel nozzles or fuel in the core. The failure analysis utilized fault tree methodology and addressed all ways the polar crane system could fail, including failure of control circuitry, protective devices, brakes, structural failures of the crane or lifting rigs, and operator errors. The results of this analysis indicated that the probability of dropping the head or internals after initial lift off and leveling of the load is extremely small. Initial lift off and leveling of the load involves raising the load a height of approximately 1.5 feet. The duration of this operation is approximately 15 minutes. Although still small, the probability of a drop during initial lift is somewhat larger than a drop from a greater height. Therefore, structural analyses have been performed to determine if the vessel nozzles or fuel in the core could be damaged if such a drop during initial lifting should occur. These are described in the response to Item 2.3.4.c.

One other load is carried over the open reactor vessel that could potentially damage fuel in the vessel. This is the Reactor Vessel Weld ISI tool. Its weight is approximately 5 tons. For this particular lift, which is performed by the Auxiliary Hoist, adequate load handling reliability will be assured on the same basis as for loads lifted by the Auxiliary Hoist in the Annulus Region. This basis is described in the response to Item 2.4.1.

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: In no cases is reliance placed on mechanical stops or electrical interlocks.

ITEM 2.3.4.b. Where reliance is placed on other, site-specific considerations (e.g., refueling sequencing), provide present or proposed technical specifications and discuss administrative or physical controls provided to ensure the continued validity of such considerations.

RESPONSE: Loads always lifted when the reactor vessel head is in place were not considered as loads that could potentially drop into the core. These are: the CRDM missile shields, the CRDM missile shield support beams, and the reactor vessel head stud tensioners. No administrative controls are required with these lifts since it is physically impossible to disassemble or reassemble the reactor such that these loads would be carried over an open vessel. In addition, the lower internals package can only be removed when the reactor is defueled and, therefore, is not a threat to either spent fuel or core cooling.

There are a number of other loads that could be moved within the containment when the reactor vessel head is removed. Procedures prohibit movement of any of these loads, including the crane load block, over the refueling cavity when the reactor vessel head is removed and there is irradiated fuel in the vessel. These procedures will be reviewed with operators as part of the qualification and training program and will be strictly enforced by individuals in charge of lifts by the Polar Crane. These administrative controls are judged to be adequate to preclude postulating that any of these loads drop into or onto an open reactor vessel.

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

RESPONSE: There are three potential consequences of interest when considering load drops onto the open reactor vessel. These 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-III in Section 5.1 of NUREG 0612 addresses each of these potential consequences. The evaluations below have been performed to address these issues:

Reactor Vessel Integrity - Structural Evaluation

During normal refueling operations, the reactor pressure vessel (RPV) head assembly is initially lifted a small distance above the flange and checked for levelness. It is then raised a height of 29.0 feet above the flange. Once at the desired height, the RPV head is moved west towards its storage stand which rests on the operating deck. Reassembly is in reverse. The potential for fuel damage or a loss of safe shutdown capability affecting the ability to get water to the core resulting from a loss of integrity of RPV nozzles was evaluated.

Based on the failure analysis described in the response to Item 2.3.3, the RPV head was assumed to drop 1.5 feet through air impacting on the RPV flange. The general methods of analysis which are documented in WCAP-9198[1/] were incorporated using parameters which are applicable to the Indian Point plant.

1/ Alexander, D. W., Shakeley, R., and Dudek, D. F., Reactor Vessel Head Drop Analyses," WCAP-9198, Westinghouse Electric Corporation, January, 1978.

The weight of the RPV head was taken to be 347,000 lbs., including the weight of the load block and the Head Lifting Rig. The RPV head was found to impact the RPV flange at a velocity of 9.83 ft./sec. During the postulated head drop, the RPV head loads the shell, but does not load the fuel. Since the head is postulated to be lifted to only 1.5 feet at the time of the drop, the head is still engaged on its guide studs and the control rod drive shafts are still within their respective head penetrations. For this reason, the drop is not expected to impart a significant impact load to the control rod drive shafts. Loading the control rod drive shafts is the only feasible mechanism for loading the fuel as a result of this drop. On this basis, damage to the fuel is not predicted.

The major portion of impact load of the RPV head is transmitted directly to the RPV flange. The load path is through the RPV shell to the two inlet nozzles and two outlet nozzles from which the RPV is supported. The dynamic model conservatively neglects energy absorption by the guide studs or the reactor internals. The stiffnesses of the RPV shell, the supported inlet and outlet nozzles, and the RPV support are modeled along with the associated masses of the actual system. The total impact load was calculated to be 47.2 million pounds. The load was assumed to be distributed to each nozzle in proportion to their stiffness resulting in a maximum principal stress in the outlet nozzle of 26,750 psi. This compares to an allowable stress of 84,000 psi. Based upon this calculation, a loss of nozzle integrity is not predicted, and the reactor coolant pressure boundary remains intact. Therefore, sources of cooling water which are provided through attached piping such as RHR or safety injection remain available.

In performing the RV head drop analysis, the following exceptions were taken to Appendix A of NUREG 0612:

(1) NUREG 0612 requires that the RPV head drop be evaluated for a fall from its maximum height. This evaluation was limited to a nominal height of 1.5 feet corresponding to a drop during initial liftoff. The basis for this exception is provided in the response to Item 2.3.3.

(2) The evaluated head drop is bounding in producing a maximum load to the RPV nozzles and the fuel. Off center drops over the RPV are not evaluated because the head is assumed to drop when still engaged on the guide studs. The orientation for drops is essentially flat and flange to flange based on the small drop height assumed and the fact that the head is still engaged on the guide studs.

Fuel Damage - Structural Evaluation

As indicated above, no fuel damage was predicted as a result of a reactor vessel head drop. However, the limiting situation for fuel damage was judged to be the postulated drop of the upper internals package into the vessel. A conservative structural evaluation was performed of a drop of the upper internals during initial lifting as described below.

During normal refueling operations, the reactor vessel upper internals package is initially lifted a small distance and checked for levelness. It is then lifted approximately 25 feet above the top of the core. Once at the desired height, the upper internals is moved west towards its storage stand which rests on the refueling cavity floor. For the reasons described in Section 2.3.3, it was postulated that the Polar Crane or the Upper Internals Lifting Rig fails during initial liftoff of the upper internals. The height of this drop was assumed to be 1.5 feet.

The impact velocity was calculated to be 9.83 feet per second. The effects of drag, bouyancy, and a "dashpot" effect due to the tight tolerance with the core barrel were conservatively neglected.

The total kinetic energy for the drop of the 143,000 pound (including load block and lifting rig) upper internals structure was calculated to be 2,145 thousand foot-pounds. This energy is assumed to be dissipated evenly by each of the 193 fuel assemblies in the core. The fuel cladding was considered to fail at a plastic strain of 1 percent. This criteria is based upon the irradiated properties of Zircalloy-4, the cladding material.

The impact load is transmitted from the upper core plate to the upper nozzle of the fuel assembly, through the 20 guide tubes, and to the lower nozzle of the fuel assembly. The fuel rods are not significantly loaded unless either the upper nozzle contacts the fuel due to elastic shortening and/or buckling of the guide tubes or the fuel assembly deflects laterally as a composite element.

It was found that the guide tubes reach their elastic limit prior to buckling elastically. The energy absorbed by axial deformation up to the elastic limit is 25,900 foot-pounds for the entire core. It is expected that the guide tubes would then buckle inelastically. The additional energy absorbed in this failure mode until the fuel assembly upper nozzle impacts the fuel rods is neglected.

Individual fuel rods are predicted to buckle elastically between spacer guides at a load of 120 pounds. This corresponds to 8,730 foot-pounds of energy due to axial deformation for the entire core. The additional energy of 180 thousand foot-pounds can be absorbed beyond the point of critical buckling through bending until the cladding strain reaches a value of 1 percent plastic. The fuel rod is assumed to take a sinusoidal shape based upon a pinned-pinned boundary condition. Accordingly, the deflection along the fuel rod is given by,

$$Y = A \sin \frac{\pi}{L} X \tag{1}$$

where
L = length of fuel rod between spacer grids
A = lateral deflection of fuel rod at mid span
X = distance along span
Y = lateral deflection of fuel rod at a distance X along the span

From beam theory,

$$1/R = \frac{d^2y}{dx^2} = -\frac{M(x)}{EI} \tag{2}$$

where
R = radius of curvature
M(x) = moment at a point x
E = youngs modulus
I = moment of inertia

The strain energy in bending is given by,

$$U_b = \frac{1}{2}\int_0^L \frac{M(x)^2}{EI}\,dx \tag{3}$$

From (2) and (3), it follows that

$$U_b = \frac{1}{2}\int_0^L EI\left(\frac{d^2y}{dx^2}\right)^2 dx \tag{4}$$

Differentiating the approximated deflection curve (1),

$$\frac{d^2y}{dx^2} = -\left(\frac{\pi}{L}\right)^2 A \sin\frac{\pi}{L}x \tag{5}$$

and substituting (5) into (4),

$$U_b = \frac{\pi^4 A^2 EI}{2L^4}\int_0^L \sin^2\left(\frac{\pi}{L}\right)x\,dx$$

$$U_b = \frac{\pi^4 A^2 EI}{4L^3} \tag{6}$$

From (6) it follows that

$$A = \sqrt{\frac{4U_b L^3}{\pi^4 EI}} \tag{7}$$

From (2) and evaluating (5) at x = L/2,

$$y_R = A\left(\frac{\pi}{L}\right)^2 \tag{8}$$

The bending strain at any fiber at a distance y from the neutral axis is given by,

$$\epsilon_b = y/R \tag{9}$$

Substituting (8) into (9)

$$\epsilon_b = Ay\left(\frac{\pi}{L}\right)^2 \tag{10}$$

Combining the bending strain from (10) with the axial strain, a total strain of 0.22 percent was calculated. This compares to a yield strain of 0.29 percent and the allowable plastic strain of 1 percent.

Based upon this analysis where the total kinetic energy is conservatively assumed to be taken by the fuel, a fission product release is not predicted from the fuel.

Criticality Considerations

The potential for a criticality condition as a result of a load drop into the core has been evaluated independent of the specific load being considered. Criterion II, Section 5.1 of NUREG 0612 requires that the resultant keff not be greater than 0.95. Additionally, Section 4.2 of Appendix A to NUREG 0612 provides guidelines for neutronics analyses of PWR cores. Since the Indian Point reactor utilized the same fuel geometry analyzed in Section 2.2 of NUREG 0612, we believe the analyses are applicable. In this case, the maximum increase in keff due to fuel crushing would be about 0.02. Since the Indian Point Technical Specifications require at least 10% $\Delta k/k$ during reactor vessel head removal and while loading and unloading fuel from the reactor, Criterion II of Section 5.1 is satisfied as the maximum achievable keff is less than 0.92.

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

RESPONSE: The load handling reliability of three handling systems has been evaluated because of the potential impact of loads on equipment required to achieve and maintain safe shutdown. The evaluation of each is described below:

<u>Auxiliary Hoist of the Polar Crane</u>

The Polar Crane Auxiliary Hoist has a capacity of 35 tons and has a hook travel that can service the Annulus Region between the containment wall and the crane wall. For the purpose of addressing the NUREG 0612 guidelines for this region of the containment, the load handling reliability of the Auxiliary Hoist has been evaluated against the criteria of Section 5.1.6. Based on the discussion below, adequate load handling reliability of the Auxiliary Hoist in the Annulus Region is demonstrated and, therefore, with one exception, load drops into this region have not been postulated.

The auxiliary hoist is mounted on the trolley frame and fully satisfies the criteria in CMAA-70-1975 and ANSI B30.2-1976. For most load handling operations, the auxiliary hoist satisfies the intent of Section 5.1.6 of NUREG 0612 (i.e., dual load path or increased safety factors of 10:1 in lieu of normal 5:1).

The auxiliary hoist components are designed with a 5:1 design safety factor on ultimate strength. For loads of less than 17.5 tons, the design safety factor for the hoist will be better than 10:1. With the exception of the containment equipment hatch door/airlock, all loads typically carried in the Annulus Region are less than 17.5 tons. The equipment hatch door weighs approximately 25 tons. For this reason an evaluation of the consequences of a postulated drop of this door into the Annulus Region has been performed. This evaluation is included in the response to 2.4.2.b(1).

In addition, the auxiliary hoist has eight parts of 7/8" wire rope. Based on published breaking strengths, the rope has a breaking strength of 245 tons. This gives a factor of safety for the wire rope of better than 14:1 for loads less than 17.5 tons and approximately 10:1 for the 25 ton equipment hatch door. Furthermore, redundant holding brakes are provided of greater than 150% capacity that are engaged when power to the hoist is lost or removed. To satisfy the intent of Section 5.1.6 of NUREG 0612, the following actions will be taken:

(1) Certified slings (ANSI B30.9) will be utilized with the auxiliary hoist for loads lifted in the Annulus Region.

(2) An extensive inspection program will be provided for ropes, brakes, and limit switches. This will include a thorough visual inspection prior to each refueling outage and checking for proper functioning of brakes and limit switches.

(3) More stringent criteria on rope replacement will be utilized (replace when five or more randomly distributed wires in one rope lay are found damaged, in lieu of the ANSI B30.2 criteria of 12 or more).

(4) A second upper limit switch will be installed on the auxiliary hoist.

(5) As indicated in Consolidated Edison's June 22, 1981 submittal, load handling and operator qualification procedures have been upgraded to meet the guidelines of NUREG 0612 and ANSI B30.2-1976.

Auxiliary Feedwater Pump Building Monorail

To assure that the likelihood of a load drop is sufficiently small that a load drop need not be postulated from the Auxiliary Feedwater Pump Building Monorail, the design of this handling system was compared to the criteria of Section 5.1.6 of NUREG 0612. Since NUREG 0612 pertains to overhead bridge cranes, it is not directly applicable to handling systems such as this monorail hoist. Accordingly, this comparison was performed to assure that the intent of the Section 5.1.6 criteria is satisfied. The following provides the results of this comparison:

(1) The monorail and its attaching hardware were designed to AISC specifications for a rated load of 5 tons. The AISC specifications call for a design safety factor of 5:1 against ultimate strength for the maximum stress. This gives an ultimate capacity of 25 tons or a safety factor of 13:1 for the maximum loads anticipated for these monorails.

(2) This monorail does not have a hoist permanently attached. To provide an increased margin to meet the intent of Section 5.1.6 of NUREG 0612, procedures will require use of a hoist with ratings that are at least 2 times greater than the weight of the load to be handled. Hoists that meet ANSI B30.16 or some other equivalent industry standard will be used. These hoists are designed to manufacturer's specifications that require all components to meet a design safety factor of better than 5:1 on ultimate strength. This gives a design safety factor of better than 10:1 for the loads that would be handled over the auxiliary feedwater pumps.

(3) Certified slings (ANSI B30.9) will be utilized when handling loads with the auxiliary feedwater pump monorails.

(4) Dynamic loads need not be considered for these hoists. The hoists are hand-driven type with a pawl-racket holding device that is secured by a friction type disc brake. Lowering is accomplished by driving against the holding brake. The dynamic load would only occur on hoisting due to the pawl action; however, this load would be small. For these hoists, a load drop during hoisting is not of safety concern. The concern is only if a drop were to occur in transporting the load along the monorail over an auxiliary feedwater pump, motor, or piping.

(5) Hoists and the monorail system are inspected and maintained in accordance with ANSI B30.11, ANSI B30.16, and manufacturer's criteria.

PAB Component Cooling Water Pump Monorail Spur Track

To assure that the load handling reliability for this system is sufficiently high so that a load drop need not be postulated, a comparison of this handling system to the criteria in NUREG 0612, Section 5.1.6, was performed in the same manner as for the Auxiliary Feed Pump Monorail. The following provides the results of this review which demonstrate that the intent of NUREG 0612, Section 5.1.6, is satisfied:

(1) The monorail and its attaching hardware were designed to AISC specifications for a rated load of 7 tons. The AISC specifications call for a design safety factor of 5:1 against ultimate for the maximum stress. This gives an ultimate capacity of 35 tons or better than 45:1 for the maximum loads anticipated to be handled by this monorail over the component cooling water pump components.

(2) The hoist was designed to manufacturer's specifications that require all components to meet a design safety factor of better than 5:1 on ultimate strength. The hoist has a rated capacity of 7 tons. As with the monorail, this gives a design capacity of 35 tons or better than 45:1 for the maximum loads that would be handled by this hoist over the component cooling water pump components.

(3) Certified slings (ANSI B30.9) will be utilized when handling loads with the auxiliary feedwater pump monorails.

(4) Dynamic loads need not be considered for this hoist. The hoist is a hand-driven type with a pawl-rachet holding device that is secured by a friction type disc brake. Lowering is accomplished by driving against the holding brake. The dynamic load would only occur on hoisting due to the pawl action; however, this load would be small. The extremely large design safety factors for these loads provide more than enough margin for any such dynamic loads.

(5) Hoists and the monorail system are inspected and maintained in accordance with ANSI B30.11, ANSI B30.16, and manufacturer's criteria.

ITEM 2.4.2. For any cranes identified in 2.1.1 not designed as single-failure-proof in 2.4.1, a comprehensive hazard evaluation should be provided which includes the following information:

ITEM 2.4.2.a. The presentation in a matrix format of all heavy loads and potential impact areas where damage might occur to safety-related equipment. Heavy loads identification should include designation and weight or cross-reference to information provided in 2.1.3.c. Impact areas should be identified by construction zones and elevations or by some other method such that the impact area can be located on the plant general arrangement drawings. Figure 1 provides a typical matrix.

RESPONSE: The requested information is provided in Attachment 1, Tables 2 through 10 and Figures 2 through 10. Layout drawings showing the location and surrounding equipment for the monorail systems of interest were included in our June 22, 1981 response.

ITEM 2.4.2.b. For each interaction identified, indicate which of the load and impact area combinations can be eliminated because of separation and redundancy of safety-related equipment, mechanical stops and/or electrical interlocks, or other site-specific considerations. Elimination on the basis of the aforementioned considerations should be supplemented by the following specific information:

RESPONSE: This information is provided on Tables 2 through 10 in Attachment 1: see those items relying on hazard elimination Category C (right-hand column).

ITEM 2.4.2.b(1): For load/target combinations eliminated because of separation and redundancy of safety-related equipment, discuss the basis for determining that load drops will not affect continued system operation (i.e., the ability of the system to perform its safety-related function).

RESPONSE:

DIESEL GENERATOR BUILDING OVERHEAD HOIST

Load drops from the Diesel Generator Building Hoist will have no affect on the capability to accomplish and maintain safe shutdown. This is because power to accomplish safe shutdown can be provided without reliance on the emergency diesels. Safe shutdown can be accomplished by either using offsite power or a backup emergency power source provided by one of three gas turbines. These gas turbines are independent of the diesel generator building and are unaffected by loss of the diesel generator units. This backup emergency power capability was developed as a result of the fire protection review and has been approved by the staff in the NRC's fire protection SER for Indian Point 2. The gas turbines are able to supply loads required for safe shutdown. Technical specifications for the gas turbines have been incorporated into the Indian Point 2 operating license.

CONTAINMENT POLAR CRANE

Systems evaluations were performed for a number of the regions inside containment. The approach and assumptions used to perform these evaluations are described below. The evaluation of each region for which systems evaluations were utilized is also provided.

Evaluations

Postulated load drops were evaluated using systems evaluations in Regions 2A, 3, 4, 5, 6, 7, 8, 9, and 10 (shown in Figures 2 through 10). These systems

evaluations typically involved determining whether a load drop could cause loss of the primary core cooling mode at the time of the drop or, if the primary cooling mode could be lost, determining if backup cooling modes could be lost from the same drop.

Plant Conditions and Cooling Modes

The initial plant conditions for all systems evaluations was taken to be the "Cold Shutdown" or "Refueling Operation" condition as defined in the facility Technical Specifications. Heavy load handling operations typically don't begin until at least four days after shutdown. Cooling for both of these conditions is normally provided by the RHR loop of the Auxiliary Coolant System. Cases were considered for the situations of both RV head removed and RV head in place. Backup cooling modes, in the event of loss of RHR cooling, were identified from plant emergency procedures for loss of all RHR cooling. Several backup modes of cooling are possible. Not all backup modes were included in the evaluations, i.e., sufficient core cooling capability could be demonstrated without the need to include all possible modes identified in the procedures.

Event Trees

In order to identify which combinations of equipment failures could potentially result in a loss of core cooling capability, a set of event trees was developed. These event trees cover five cases that could be encountered for load drops inside containment.

They are:

Case 1 - Reactor Vessel Head Removed - Load Drop Does Not Result in an Unisolable Reactor Coolant System (RCS) Pipe Break

Case 2 - Reactor Vessel Head in Place - Load Drop Does Not Result in an Unisolable RCS Pipe Break

Case 3A - Reactor Vessel Head Removed - Load Drop Results in an Unisolable RCS Pipe Break

Case 3B - Reactor Vessel Head In Place - Load Drop Results in a Small Unisolable RCS Pipe Break

Case 3C - Reactor Vessel Head In Place - Load Drop Results in a Large Unisolable RCS Pipe Break

The event trees for these cases are displayed in Figures 11 through 15.

The event trees for the most part identify success and failure paths at the system level. For any particular load drop, the success or failure of a particular system was evaluated by determining whether any of the components required for operation of that system located inside containment could potentially be damaged by the load drop. If components could be damaged, then a determination was made as to whether loss of that system component could result in loss of the system function. Once the success or failure of the system of interest for each case was determined, the path on the event tree corresponding to the particular load drop event being postulated could be identified.

If the path for a particular drop scenario corresponded to successful maintenance of core cooling (indicated by the term "OK"), then no further evaluation of that drop scenario was required. If the path was one that culminated with an asterisk, then alternative core cooling modes were considered, i.e., cooling modes other than those initially included in the event trees. (As indicated above, all possible cooling modes were not necessarily included on the event trees).

Assumptions Regarding Loss of Equipment

The loss of equipment was evaluated in a conservative manner using the following assumptions:

(1) Except in cases where more localized damage could be justified, all equipment in a given region (at all elevations) was assumed to be lost. In the cases of Regions 6 and 7, the regions were subdivided for evaluation purposes into four subregions corresponding to each quadrant of the containment. This is justified for Region 7 because the effects of load drops below the operating deck, if there

should be any, are expected to be localized, i.e., gross failure of large sections of the operating deck is not predicted. The deck was subdivided into four quadrants roughly corresponding to the NE, SE, SW, and NW regions of the floor. This was chosen because load handling and laydown areas are, for the most part, restricted to the four corner areas on either side of the two steam generator compartments.

Each of Regions 8 and 10 was subdivided into two subregions (North and South) for evaluation purposes.

(2) If RCS piping or connecting piping was in the region, an RCS pipe break was assumed to occur and its effect on core cooling evaluated assuming the simultaneous loss of other equipment in the region that could be impacted.

(3) In the case of Region 6, Reactor Coolant Pump Motor drops down through the corresponding openings in the operating deck were assumed to affect a significantly larger area below the deck than defined for the region at the 95' el.

(4) If instrumentation required to operate a component was within an impacted region, the component was assumed to be lost, e.g., if a steam generator level instrument was predicted to be lost, then the affected steam generator was assumed to be lost.

Steps in the Systems Approach

The steps used to perform systems evaluations of the potential effects of load drops inside the crane wall are outlined below:

(1) Select a region for consideration.

(2) Identify the equipment within the region that could be important to core cooling considerations.

(3) Identify the cases (event trees) that apply to that region.

(4) Assuming the equipment within the region is lost, determine whether the system function is lost.

(5) Using the results of (4), i.e., success or failure of the system, determine for each case which path on the event tree represents the load drop event being considered.

(6) If the path represents successful maintenance of core cooling for all cases, then no further analyses are required for the region.

(7) If the path represents a failure to demonstrate adequate core cooling with the core cooling modes included in the event tree, then either consider alternative cooling modes or perform a more detailed evaluation of possible equipment failures and reenter event trees.

Systems Evaluation Results

Evaluation of Regions 3 and 4 - Areas Over RHR Heat Exchanger Compartment

There are two potential drop areas of interest that make up Regions 3 and 4. The first is the grating in the NW quadrant of containment. Although plant procedures prohibit movement of heavy loads over this region, there are no physical limitations that would prohibit Polar Crane travel over the region. In addition, the capacity of the grating is such that it can not be shown to withstand load drops of any significant weight or height of carry.

The second drop area is the head storage stand area. Structural analysis of a drop of the head on its storage stand predicts that scabbing from the underside of the 95' elevation slab into the RHR Heat Exchanger compartment could occur.

In lieu of demonstrating that the intervening structures, i.e., the operating deck or the grating, can protect the equipment below from the effects of a load drop, a system evaluation was performed. The equipment identified in Table I was assumed lost as a result of a load drop on Regions 3 or 4. This equipment is located in the RHR Heat Exchanger Compartment. Also indicated in Table I is whether or not the equipment failures are predicted to result in loss of the system function. In some cases, remarks are included to explain the system failure conclusions.

TABLE I - SYSTEMS EVALUATION OF REGIONS 3 AND 4

SYSTEMS OF INTEREST	EQUIPMENT IN REGION ASSUMED LOST	IDENTIFICATION	IS SYSTEM ASSUMED LOST	REMARKS
RCS and Connecting Piping	None		No RCS Pipe Break	
RHR	Heat Exchangers (2) Inlet & Outlet Piping Inlet & Outlet Valves	Inlet-line 9, Outlet 361	Yes	
CCW	Inlet & Outlet Piping to RHR Heat Exchangers and to Recirculation Pumps		Yes	
Recirculation portion of SI	Pumps (2) Discharge Piping to RHR Heat Exchanger Sump	Line 293	Yes	
HPI Portion of SI	Piping Valves	Line 351 from Acc. #1 to Loop 1 Cold Leg MOV 894A Chk 895A Line 355 (to 351) Chk 838A Line 56 (to 351) Chk 857J	No	Line 56 can be isolated from remainder of SI system by MOV-856A outside Crane Wall
Fan Coolers	None		No	Located in the Annulus
Steam Generators	None		No	
Feedwater	None		No	
Atmosphere Steam Dump	None		No	

Note: Other systems unaffected by a drop in this region are Pressurizer/PZR pressure control, RC pumps, and CVCS.

The conclusions regarding the system failures were then used to enter the event trees applicable to the postulated load drop. The applicable event trees are those for Cases 1 and 2 (see Figures 11 and 12 in Attachment 1). The Case 3 event trees are not applicable, because no unisolable RCS pipe breaks are predicted as a result of drops into this compartment.

For Case 1, since RHR is predicted to fail, the primary cooling mode is assumed lost. However, the backup cooling mode, HPI and the Fan Cooler units, are not predicted to be lost as a result of the drop. Therefore, successful core cooling is maintained. The path on the event tree representing this success is Path 5.

For Case 2, again RHR is predicted to fail. However, none of the equipment in either of the two backup cooling modes displayed in the event tree is predicted to fail. Therefore, core cooling is maintained, as represented by Path 23.

Evaluation of Region 2A - Area In Annulus Region Between Columns 3 and 17

As indicated in the response to Item 2.4.1, there is one portion of the Annulus Region between the crane wall and the containment liner that cannot be eliminated on the basis of a demonstration of load handling reliability. This is the area over which the containment equipment hatch door/airlock is carried (see Figure 2). The hatch door weighs approximately 25 tons. Accordingly, a systems evaluation of the potential consequences of a drop of the equipment hatch door into Region 2A was performed.

The equipment and systems assumed lost are indicated in Table 2. The very conservative assumption that all equipment within the region is lost was made. This information was used to enter the event trees for Cases 1 and 2. Cases 3A, 3B and 3C were not considered because no unisolable RCS pipe breaks are predicted.

For Case 1, successful core cooling is predicted. Path 5 represents the success path. For Case 2, successful core cooling is also predicted as represented by Path 36.

TABLE 2 - SYSTEMS EVALUATION OF REGION 2A

SYSTEMS OF INTEREST	EQUIPMENT IN REGION ASSUMED LOST	IDENTIFICATION	IS SYSTEM ASSUMED LOST	REMARKS
RCS and Connecting Piping	None		No RCS Pipe Break	Check valves located inside Crane Wall
RHR	Piping	Line 361 to Loop 4 cold leg injection line or Line 356 to Loop 2 cold leg	Yes	
CCW	None		No	
HPI portion of SI	Piping, Valves	Line 16 to Loops 1, 2 & 4	No	Isolate line 16 outside containment at SI pumps
Recirculation portion of SI	None		No	
Coolers	None		No	Fan coolers would not be damaged. Potentially cooling lines could be, but each cooler is on separate line penetrating containment and therefore the break could be isolated.
Instrumentation Circuits	Steam Generator Level Indication		Yes	All level indication could be lost, however, steam generators would be available

TABLE 2 - SYSTEMS EVALUATION OF REGION 2A
(continued)

SYSTEMS OF INTEREST	EQUIPMENT IN REGION ASSUMED LOST	IDENTIFICATION	IS SYSTEM ASSUMED LOST	REMARKS
	RC Pump Seal & Cool. Water Flow		No	Cooling flow information would be lost, but RC pumps still available
Instrument Rack	Pressurizer Press and Level Indicators		Yes	
Chemical and Volume Control System - Charging and Letdown	None		No	
PORV	None		No	

Evaluation of Region 5 - Reactor Cavity

The load drops of interest for this region include the reactor head and the upper internals package. Load drops of either of these two components could potentially damage the reactor cavity liner and floor to the extent that equipment below could be impacted or refueling water discharged to the containment floor. The issue is whether or not equipment below that is required to maintain core cooling could be impacted or damaged from flooding.

There is no RHR or CCW equipment below the cavity floor. Therefore, the primary cooling mode is predicted to be unaffected by the postulated load impact. Further, as part of a previously performed ECCS performance analysis, a water level inside containment has been calculated based on a larger volume of water than could be discharged from the reactor cavity.

The water volume used for the ECCS analysis was over 420,000 gallons. The approximate volume of the reactor cavity is 300,000 gallons. The water level calculated for the ECCS analysis resulted in a water level up to about the 50' elevation or approximately 4' above the floor level of 46'. We have used the 50' elevation water level as a bounding value for evaluating the possible flooding effects of a loss of reactor cavity integrity. Our review indicated that there are no RHR or CCW valves affected by a 50' elevation water level. Therefore, the primary mode of cooling in the cold condition is not predicted to be lost as a result of a postulated heavy load drop onto the reactor cavity floor.

Evaluation of Region 6 - Reactor Coolant Pumps

Region 6 was subdivided into four subregions: 6NE, 6SE, 6SW, and 6NW for evaluation purposes. The load drop of interest for these regions is a drop of a Reactor Coolant Pump Motor onto the pump. This could potentially occur from a height above the 95' elevation when raising or lowering a pump motor through the grating covered hatch at the operating deck. The Reactor Coolant

Pump/Motor Mating surface is located at about the 70' elevation. Accordingly, a drop of a 32-ton pump motor onto the pump of over 25' could be postulated.

To evaluate the consequences of such pump motor drops, a systems evaluation was undertaken for each of the four regions. The equipment and associated systems identified in Table 3 were assumed lost as a result of a pump motor drop. The system failures identified were then used to enter the event trees. For the pump motor drop, all cases were considered. The results are presented in Table 4. As Table 4 indicates, core cooling can be maintained for all postulated drop scenarios.

Evaluation of Region 7 - Operating Deck - 95' Elevation

Region 7 was subdivided into four subregions: 7NE, 7SE, 7SW, and 7NW for evaluation purposes. The postulated load drop of principal interest is the RC Pump motor on the operating deck.

Structural evaluations were performed to verify that drops onto the operating deck could not result in gross failure of large sections of the deck, i.e., localized failures only such as scabbing from the underside of the deck are anticipated.

A systems evaluation very similar to that performed for Region 6 was performed for Region 7. Initially, the very conservative assumption was made that the equipment and associated systems identified in Table 5 for each region were lost. The system failures identified were then used to enter the event trees. All cases were considered. The results are presented in Table 6.

Table 6 indicates that if all equipment of interest in subregion 7NE is assumed lost, core cooling can not be accomplished by the cooling modes included in the event trees for Cases 2 and 3B. The principal difference in the analysis of Regions 6NE and 7NE is that the PORV piping is assumed lost for Region 7NE. The PORV Piping runs from the top of the Pressurizer out of the NW corner of

TABLE 3 - SYSTEMS EVALUATION OF REGIONS 6NE, 6NW, 6SE, AND 6SW

SYSTEMS OF INTEREST	EQUIPMENT IN REGION ASSUMED LOST	IDENTIFICATION	IS SYSTEM ASSUMED LOST	REMARKS
RCS and Connecting Piping	RCS piping	Loop Cold Legs - one per region	RCS Pipe Break Possible	
RHR	Piping	Cold Leg Injection Lines - one per region. RHR Return Line in 6SE	Yes	
CCW	Piping	RCP Cooling Lines	No	The CCW to RCPs can be isolated from CCW loop to RHR Hx
Recirculation portion of SI	None		No	
[illegible] portion [illegible]SI	Piping	Injection Lines to Cold Legs - one per region. Injection Lines to Hot Legs in Regions 6NW, 6SW, 6SE	No	Broken injection lines can be isolated from remainder of SI system by MOVs located outside of Crane Wall
Reactor Coolant Pump (RCP)	RCP and Associated Auxiliaries	One pump per region	Yes	Can affect one pump only
Pressurizer	Pressurizer Instrumentation, and Pressure Control	• PZR Heaters & Spray Lines • Level and Pressure Instruments	For 6NE drop only	Located in Loop 4. Therefore, assumed lost for 6NE

TABLE 3 - SYSTEMS EVALUATION OF REGIONS 6NE, 6NW, 6SE, AND 6SW

(continued)

SYSTEMS OF INTEREST	EQUIPMENT IN REGION ASSUMED LOST	IDENTIFICATION	IS SYSTEM ASSUMED LOST	REMARKS
Fan Coolers	None		No	Annulus
Steam Generators	None		No	
Feedwater	None		No	
Atmospheric Dump	None		No	
Chemical and Volume Control System - Charging and Letdown	Piping	Charging and Letdown Lines	For 6SW and 6SE only	Charging and letdown connections are in Loop 1. Piping travels in vicinity of 6SE RCP enroute to and from regenerative heat exchanger in SE quadrant.

TABLE 4

REGION 6 - EVENT TREE ASSESSMENT

SUBREGION	CASE	PATH	CONCLUSION
6NW	1	5	OK
	2	23	OK
	3A	1	OK
	3B	1	OK
	3C	1	OK
6SW	1	5	OK
	2	19	OK
	3A	1	OK
	3B	18	OK
	3C	1	OK
6SE	1	5	OK
	2	23	OK
	3A	1	OK
	3B	18	OK
	3C	1	OK
6NE	1	5	OK
	2	15	OK
	3A	1	OK
	3B	14	OK
	3C	1	OK

TABLE 5 - SYSTEMS EVALUATION OF REGIONS 7NE, 7NW, 7SE, AND 7SW

SYSTEMS OF INTEREST	EQUIPMENT IN REGION ASSUMED LOST	IDENTIFICATION	IS SYSTEM ASSUMED LOST	REMARKS
RCS and Connecting Piping	RCS Piping	RCS Cold and Hot Leg Piping	RCS Pipe Break Possible	
RHR	Piping	Cold leg injection - one per region. RHR return line in 7SE	Yes	
CCW	Piping	RCP cooling lines	No	The CCW to RCPs can be isolated from the CCW to the RHR Hx
HPI portion of SI	Piping	Injection lines to to Cold Legs - one per region - Injection lines to Hot Legs in 7NW, 7SW, 7SE	No	Injection lines can be isolated from remainder of SI system by MOV located outside Crane Wall
Steam Generator	Piping - steam/ blowdown		No	Affects one loop only
Recirculation portion of SI	None		No	
Reactor Coolant Pump	Auxiliaries	One pump per region	No	Can affect one pump only
Pressurizer	Instrumentation, Pressure Control and Pressure Relief	• PZR Heaters & Spray Lines • Level & Pressure Instruments • PORV Piping	For 7NE drops only	

TABLE 5 - SYSTEMS EVALUATION OF REGIONS 7NE, 7NW, 7SE, AND 7SW

(continued)

SYSTEMS OF INTEREST	EQUIPMENT IN REGION ASSUMED LOST	IDENTIFICATION	IS SYSTEM ASSUMED LOST	REMARKS
Fan Coolers	None		No	
Feedwater	Piping		No	Piping separated to steam generator
Atmosphere Dump	None		No	
Chemical & Volume Control System - Chg and Letdown	Piping		For 7SW and 7SE drops only	Charging and letdown are from Loop 1 in SW quadrant. Piping travels into SE quadrant to and from regenerative heat exchanger.

TABLE 6

REGION 7 - EVENT TREE ASSESSMENT

SUBREGION	CASE	PATH	CONCLUSION
7NW	1	5	OK
	2	23	OK
	3A	1	OK
	3B	1	OK
	3C	1	OK
7SW	1	5	OK
	2	19	OK
	3A	1	OK
	3B	18	OK
	3C	1	OK
7SE	1	5	OK
	2	19	OK
	3A	1	OK
	3B	18	OK
	3C	1	OK
7NE	1	5	OK
	2	17	Consider Alternative Cooling Modes
	3A	1	OK
	3B	16	Consider Alternative Cooling Modes
	3C	1	OK

the Pressurizer Compartment at about the 127' elevation; down the Pressurizer Compartment wall to the 103' elevation. It then runs northwestward across the operating deck to a penetration in the floor just inside the crane wall. It proceeds downward at an angle through the crane wall to the Annulus region where it ultimately ties into the Pressurizer Relief Tank.

Prior to considering alternative cooling modes as the event trees suggest, we attempted to determine if the assumption that all equipment in the Region 7NE is lost from a single load drop is a reasonable one. Our conclusion was that it was not reasonable to assume the loss of PORV piping in conjunction with loss of RHR. The RHR injection line that could be lost from a load drop is below the operating deck east and south of the RC Pump in Loop 4. The PORV discharge line is above the operating deck, north and west of this pump and the Pressurizer Compartment. It is extremely unlikely that a load drop that could damage one system could also damage the other. For this reason, it is concluded that core cooling can be maintained for Cases 2 and 3B.

Evaluation of Region 8 - Steam Generators

A load drop onto Region 8 could impact one or two steam generators. However, a breach of the steam generator shell at cold conditions would have no effect on the primary core cooling mode (RHR).

Evaluation of Region 9 - Instrument Racks - NE Quadrant

A heavy load drop could potentially penetrate the grating over the instrument rack and valve access area in the NE quadrant of the operating deck. The equipment and systems assumed to be lost are indicated in Table 7. This information was used to enter the event trees for Cases 1 and 2. Cases 3A, 3B, and 3C were not considered because no unisolable RCS pipe break is predicted.

For Case 1, successful core cooling is predicted. Path 5 represents the success path. For Case 2, successful core cooling is represented by Path 23.

TABLE 7 - SYSTEMS EVALUATION OF REGION 9

SYSTEMS OF INTEREST	EQUIPMENT IN REGION ASSUMED LOST	IDENTIFICATION	IS SYSTEM ASSUMED LOST	REMARKS
RCS and Connecting Piping	None		No RCS Pipe Break	
RHR	Piping	Line 361 to Loop 4 cold leg injection line	Yes	
CCW	None		No	
HPI portion of SI	None		No	
Recirculation portion of SI	None		No	
Fan Coolers	None		No	
Instrumentation Racks	Steam Generator Level Indication		Yes	All level indication could be lost, however, steam generators would be available
	RC Pump Seal & Cool. Water Flow		No	Cooling flow information could be lost, but RC pumps still available
	RC Flow		No	Not useful
Instrument Sensing Lines	Pressurizer Press and Level Indicators		No	One channel of Pressure and Level is routed outside the Crane Wall.
Chemical and Volume Control System - Charging and Letdown	None		No	

Note: Other systems unaffected by this load drop include the pressurizer, RC pumps, Steam Generators and Feedwater, and Atmospheric Dump.

Evaluation of Region 10 - Slabs Between Steam Generators

The load drop of interest for this region is the drop of a CRDM Missile Shield. The area between the steam generators is the laydown area for the four 23-ton missile shields (two on each side). Since the missile shields are only lifted when the reactor vessel head is in place, Cases 2 and 3A do not apply. Path I is applicable for Case 2, because no damage to RHR or CCW piping is predicted if the missile shield drop were to result in damage to equipment below the slabs. Further, the potential consequences of a postulated RCS pipe break from a drop are bounded by the RCS pipe break cases considered for Regions 6 and 7.

ITEM 2.4.2.b.(2) Where mechanical stops or electrical interlocks are to be provided, present details showing the areas where crane travel will be prohibited. Additionally, provide a discussion concerning the procedures that are to be used for authorizing the bypassing of interlocks or removable stops, for verifying that interlocks are functional prior to crane use, and for verifying that interlocks are restored to operability after operations which require bypassing have been completed.

RESPONSE: Neither mechanical stops or electrical interlocks have been relied on.

ITEM 2.4.2.b.(3) Where load/target combinations are eliminated on the basis of other, site-specific considerations (e.g., maintenance sequencing), provide present and/or proposed technical specifications and discuss administrative procedures or physical constraints invoked to ensure the continued validity of such considerations.

RESPONSE: No load/target combinations were eliminated on the basis of site specific considerations.

ITEM 2.4.2.c. For interactions not eliminated by the analysis of 2.4-2-b, above, identify any handling systems for specific loads which you have evaluated as having sufficient design features to make the likelihood of a load drop extremely small and the basis for this evaluation (i.e., complete compliance with NUREG 0612, Section 5.1.6, or partial compliance supplemented by suitable alternative or additional design features). For each crane so evaluated, provide the load-handling-system (i.e., crane-load-combination) information specified in Attachment 1.

RESPONSE: See response to 2.4.1.

ITEM 2.4.2.d For interactions not eliminated in 2.4-2-b or 2.4-2-c, above, demonstrate using appropriate analysis that damage would not preclude operation of sufficient equipment to allow the system to perform its safety function following a load drop (NUREG 0612, Section 5.1, Criterion IV). For each analysis so conducted, the following information should be provided:

RESPONSE: All handling systems and load impact regions have been evaluated.

ATTACHMENT I

CRANE: CONTAINMENT POLAR CRANE

E 1

LOCATION	CONTAINMENT BUILDING		
IMPACT AREA / LOADS	REGION 1 - REACTOR VESSEL (SEE FIGURE 1) 1A - REACTOR VESSEL HEAD REMOVED		
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
RV HEAD (169 TONS)	Vessel flange is at 69' el. Fuel in core is at 56' el. Head and internals assumed dropped 1.5' after lift off.	Reactor Vessel - Vessel Integrity Considerations	e. Head drop analysis indicates stress is within code allowables
		Irradiated Fuel Assemblies in the core	e. Breach of fuel cladding not predicted for head drop.
REACTOR INTERNALS (67 TONS)		Irradiated Fuel Assemblies in the Core	e. Breach of fuel cladding not predicted for internals drop. Criticality not predicted assuming optimum uranium-water ratio from fuel crushing.
ISI TOOL (5 TONS)		Irradiated Fuel Assemblies in the Core	d. Likelihood of handling system failure for this load is extremely small.
CRANE LOAD BLOCK (4.5 TONS) REACTOR COOLANT PUMP MOTORS (32 TONS) CONCRETE HATCH COVER (7.5 TONS) PZR MISSILE SHIELD (7.6 TONS)		Reactor Vessel and Irradiated Fuel Assemblies in the Core	Procedures prohibit carrying any of these loads over the reactor cavity when the head is removed and irradiated fuel is in the vessel.

CRANE: CONTAINMENT POLAR CRANE

E 1 (CONTINUED)

LOCATION	CONTAINMENT BUILDING		
IMPACT AREA / LOADS	REGION 1B - REACTOR VESSEL REACTOR VESSEL HEAD IN PLACE (SEE FIGURE 1)		
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
CRDM MISSILE SHIELD BLOCKS (23 TONS) CRDM MISSILE SHIELD SUPPORT BEAMS RV HEAD STUD TENSIONERS CRANE LOAD BLOCK (4.5 TONS)	Impact area would be shield support beams at approximately the 95' el or the vessel head lifting rig, slightly below the 95' el.	Reactor Vessel - Vessel Integrity Considerations	e. Bounded by RV Head Drop Analysis in terms of load on vessel nozzles. The shield blocks (heaviest load) would impact the shield support beams and possibly head rig, if dropped. Expected that only a small amount of energy would be transferred to the nozzles, if any. Loss of RCS pressure boundary integrity at cold conditions from possible damage to CRDM housings as a result of a drop will have no effect on the capability to cool the core. Damage to housings would be expected to be very limited because of protection afforded by RV head lifting rig which is permamently in place.

TABLE 2

CRANE: CONTAINMENT POLAR CRANE

LOCATION	CONTAINMENT BUILDING		
IMPACT AREA / LOADS	REGION 2 - ANNULUS REGION BETWEEN THE CRANE WALL AND THE CONTAINMENT LINER (SEE FIGURE 2) 2A - ANNULUS REGION - BETWEEN COLUMNS 3 and 17		
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
PZR MISSILE SHIELD (7.6 TONS) RV HEAD STUD TENSIONERS CONTAINMENT EQUIPMENT HATCH PLUG (25 tons)	Impact area would be grating or checkered plate at the 95' el. Equipment is at lower elevations.	Equipment required to maintain Long Term Cooling.	d. Likelihood of handling system failure for heavy loads handled by the Auxiliary Hoist in the Annulus Region is extremely small. e. There is adequate separation of equipment to assure core cooling in the event of a load drop into the Annulus Region between columns 3 and 17. This is the region over which the 25-ton equipment hatch door travels to and from its laydown area.

TA[illegible]3

CRANE: CONTAINM[illegible]T POLAR CRANE

LOCATION	CONTAINMENT BUILDING		
IMPACT AREA / LOADS	REGION 3 - AREA OVER GRATING WITHIN CRANE WALL IN NORTHWEST QUADRANT OF CONTAINMENT (SEE FIGURE 3)		
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
Systems evaluation is independent of load considered.	Impact area is 95' el. Equipment is at lower elevations.	• RHR Heat Exchangers and associated piping • CCW piping to RHR Heat Exchangers and Recirculation Pumps • Recirculation pumps and sump • 1 of 4 HPI cold leg injection lines • 1 of 4 RHR cold leg injection lines	c. Evaluated loss of RHR cooling at cold conditions, both with head off and head in place. Core cooling maintained with equipment that is unaffected by a postulated load drop into the region. In addition, loads are prohibited from movement over this region when there is irradiated fuel in the reactor vessel.

CRANE: CONTAINMENT POLAR CRANE

E 4

LOCATION	CONTAINMENT BUILDING		
IMPACT AREA / LOADS	REGION 4 - RV HEAD STORAGE AREA (SEE FIGURE 4)		
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
Systems evaluation is independent of load considered.	Impact area is 95' el. Equipment is at lower elevations.	• RHR Heat Exchangers and associated piping • CCW Piping to RHR Heat Exchangers and recirculation pumps • 1 of 4 HPI cold leg injection lines • 1 of 4 RHR cold leg injection lines	See Discussion for Region 3

CRANE: CONTAINMENT POLAR CRANE

BLE 5

LOCATION	CONTAINMENT BUILDING		
IMPACT AREA / LOADS	REGION 5 - REFUELING CANAL (SEE FIGURE 5)		
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
Systems evaluation is independent of load considered.	Impact area is bottom of refueling canal at approximately 69' el. in west end and 60' el. in east end.	Possible Equipment Required to Maintain Long Term Cooling - Concern is leakage from pool to levels below and possible resultant flooding damage to equipment.	e. Volume of water in refueling canal is less than volume of water that could be dumped to the containment floor during a large LOCA. Flooding of safety-related components was previously evaluated for LOCA - modifications made to assure operability of all components. Checked for additional components associated with normal long term cooling mode, i.e. RHR. All are above LOCA water level. In addition, most of the loads listed are prohibited from movement over the refueling canal by procedure.

CRANE: CONTAINMENT POLAR CRANE

LOCATION	CONTAINMENT BUILDING		
IMPACT AREA / LOADS	REGION 6 - AREA AROUND REACTOR COOLANT PUMPS - FOUR SEPARATE REGIONS; ONE IN EACH QUADRANT OF CONTAINMENT (FIGURE 6)		
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
SYSTEMS-EVALUATION IS INDEPENDENT OF LOAD CONSIDERED	Impact area is RCP motor/pump connection at 69' el. Opening for motor removal is at 95' el.	For Each of Four Regions • 1 of 4 RHR Cold Leg Injection Lines • RHR Return Line From Loop 2 Hot Leg (SE Quardrant Only) • CCW Piping to RC Pump • Reactor Coolant Pump • Pressurizer Spray, Heaters and Instruments (NE Quadrant Only)	c. There is adequate separation of equipment to assure that core cooling can be maintained in the event of loss of the primary cooling mode (RHR) and/or a RCS pipe break at cold conditions.

TABL

CRANE: CONTAIN T POLAR CRANE

LOCATION	CONTAINMENT BUILDING		
IMPACT AREA / LOADS	REGION 7 - OPERATING DECK INSIDE CRANE WALL - FOUR SEPARATE REGIONS: ONE IN EACH QUADRANT OF CONTAINMENT (SEE FIGURE 7)		
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
SYSTEMS EVALUATION IS INDEPENDENT OF THE LOAD CONSIDERED	Impact area is at 95' el. Equipment is at lower elevations.	Same equipment as identified for Region 6 except for following additions: • Damage to charging 2nd letdown piping can occur in both SW and SE Quadrants • Damage to a Steam Generator and its associated piping could occur • Damage to PORV piping to the Pressurizer Relief Tank could occur in the NE Quadrant (piping exposed above 95' el. only).	c. There is adequate separation of equipment to assure that core cooling can be maintained in the event of loss of the primary cooling mode (RHR) or a RCS pipe break at cold conditions.

TABLE 8

CRANE: CONTAINMENT POLAR CRANE

LOCATION	CONTAINMENT BUILDING		
IMPACT AREA / LOADS	REGION 8 - STEAM GENERATORS (SEE FIGURE 8)		
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
SYSTEMS EVALUATION IS INDEPENDENT OF THE LOAD CONSIDERED	Impact area is top of steam generator shell at approximately 125' el.	STEAM GENERATORS	e. Consequences of Steam Generator shell rupture from a load drop have no effect on fuel in the core or core cooling capability at cold conditions.

LOCATION	CONTAINMENT BUILDING		
IMPACT AREA / LOADS	REGION 9 - GRATING OVER INSTRUMENT RACKS AND VALVE ACCESS AREA - NE QUADRANT OF OPERATING DECK (SEE FIGURE 9)		
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
SYSTEMS EVALUATION IS INDEPENDENT OF THE LOAD CONSIDERED	Impact area is grating at the 95' el. Instrument Racks and valve access located at 68' el.	• 1 of 4 RHR Cold Leg Injection Lines • Steam Generator Level Indication • Pressurizer Pressure and Level Indication	c. There is adequate equipment separation to assure that core cooling capability is maintained in the event of loss of the primary cooling mode (RHR).

CRANE: CONTAINMENT POLAR CRANE

LOCATION	CONTAINMENT BUILDING		
IMPACT AREA	REGION 10 - AREA BETWEEN STEAM GENERATORS - SLABS - BOTH NORTH AND SOUTH OF REFUELING CANAL WITHIN STEAM GENERATOR ENCLOSURES (SEE FIGURE 10)		
LOADS	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
SYSTEMS EVALUATION IS INDEPENDENT OF THE LOAD CONSIDERED	Impact area is 95' el. Equipment located at lower elevations.	• RCS piping	c. & e. No damage to equipment for primary core cooling mode (RHR) predicted. RCS pipe-break consequences bounded by Region 6 and 7 evaluations.

TABLE 11

CRANE: AUXILIARY FEEDWATER PUMP BUILDING MONORAIL (PUMPS 21 AND 23)

LOCATION	AUXILIARY FEEDWATER PUMP BUILDING		
IMPACT AREA / LOADS	• MOTOR DRIVEN FEEDWATER PUMP HOUSING		
	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
FEEDWATER PUMP MOTOR (2860 lbs)		Motor driven pump that is not out for service and the steam driven pump. Potential effects are: (1) a drop of the motor on the pump housing for the pump that is out of service could potentially lead to flooding damage of the motor driven pump that is operable or the steam driven pump, or (2) a drop of the motor on the operable motor driven pump could take out the operable motor driven pump and lead to flooding damage of the steam driven feedwater pump.	d. Likelihood of load drop is extremely small.

TABLE 2

CRANE: AUXILIARY FEEDWATER PUMP BUILDING MONORAIL (PUMP 22)

LOCATION	AUXILIARY FEEDWATER PUMP BUILDING		
IMPACT AREA	• STEAM DRIVEN AUXILIARY FEEDWATER PUMP • STEAM DRIVEN PUMP SUCTION OR DISCHARGE PIPING		
LOADS	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
FEEDWATER PUMP TURBINE (1.5 TONS) TURBINE MISSILE SHIELD (1.9 TONS)		Motor driven feedwater pumps. The potential effect is that damage to the pump housing or suction piping could lead to flooding damage to motor driven feedwater pumps or electrical equipment if the suction line is not isolated. Plant procedures require isolation of the suction line prior to making heavy lifts over the pump.	d. Likelihood of handling system failure for these loads is extremely small.

TABLE 13

CRANE: PAB COMPONENT COOLING WATER PUMP MONORAIL SPUR TRACK

LOCATION	AUXILIARY BUILDING		
IMPACT AREA	• COMPONENT COOLING WATER PUMPS OR MOTORS • COMPONENT COOLING WATER PIPING • COMPONENT COOLING WATER PUMP MOTOR POWER CABLES		
LOADS	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
STEEL GRATING (APPROX. 800 lb)	Grating is at el. 80' (Pumps are at el. 68')	Component Cooling Water Pumps. The potential effect is that a load drop could cause loss of all three component cooling water pumps.	d. Likelihood of handling system failure for this load is extremely small.
PIPING SUPPORT "I" BEAM (500-600 lbs)	"I" Located at el. 74'	Same as above.	
CCW PUMP MOTORS (1475 lbs)	Motors located at el. 68'	Same as above.	

TAB[illegible]4

CRANE: DIESEL GENERATOR BUILDING CRANE

LOCATION	DIESEL GENERATOR BUILDING		
IMPACT AREA	• DIESEL GENERATOR UNITS • DIESEL GENERATOR LOCAL CONTROL PANEL		
LOADS	ELEVATION	SAFETY-RELATED EQUIPMENT	HAZARD ELIMINATION CATEGORY
DIESEL GENERATOR COMPONENTS (LESS THAN 2 TONS; MOST ARE ON THE ORDER OF 200-400 LBS)	Floor el.72'	• Diesel Generator Units. The potential effect is that drop of a component such as a lube oil cooler could damage small piping, such as fuel or lube oil, on adjacent operable D.G. units, or could damage the local control panels causing loss of D.G. control from the control room.	b. Alternate shutdown capability deriving emergency power from one of three gas turbines would be available.

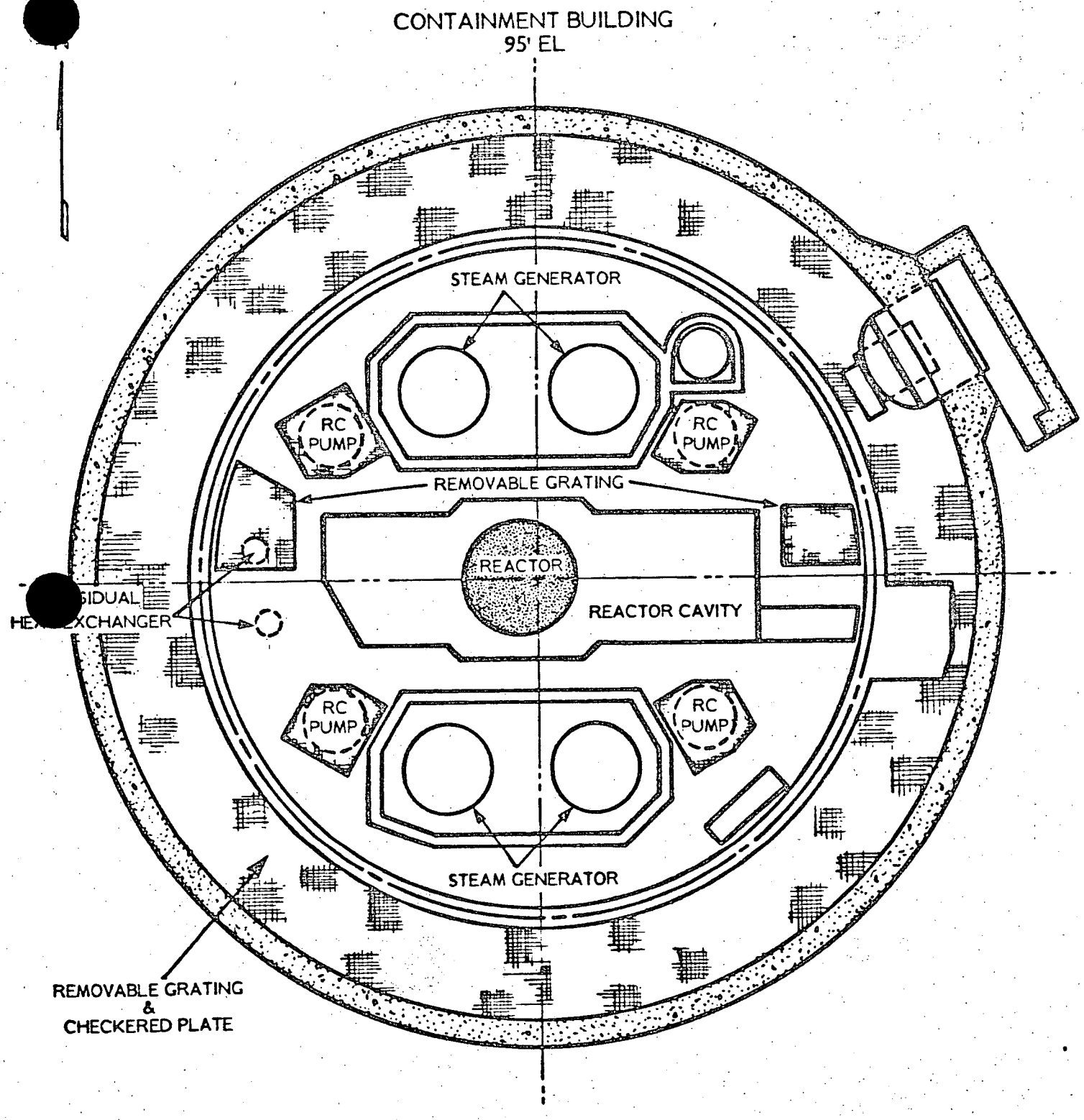

FIGURE I

REGION I - REACTOR VESSEL

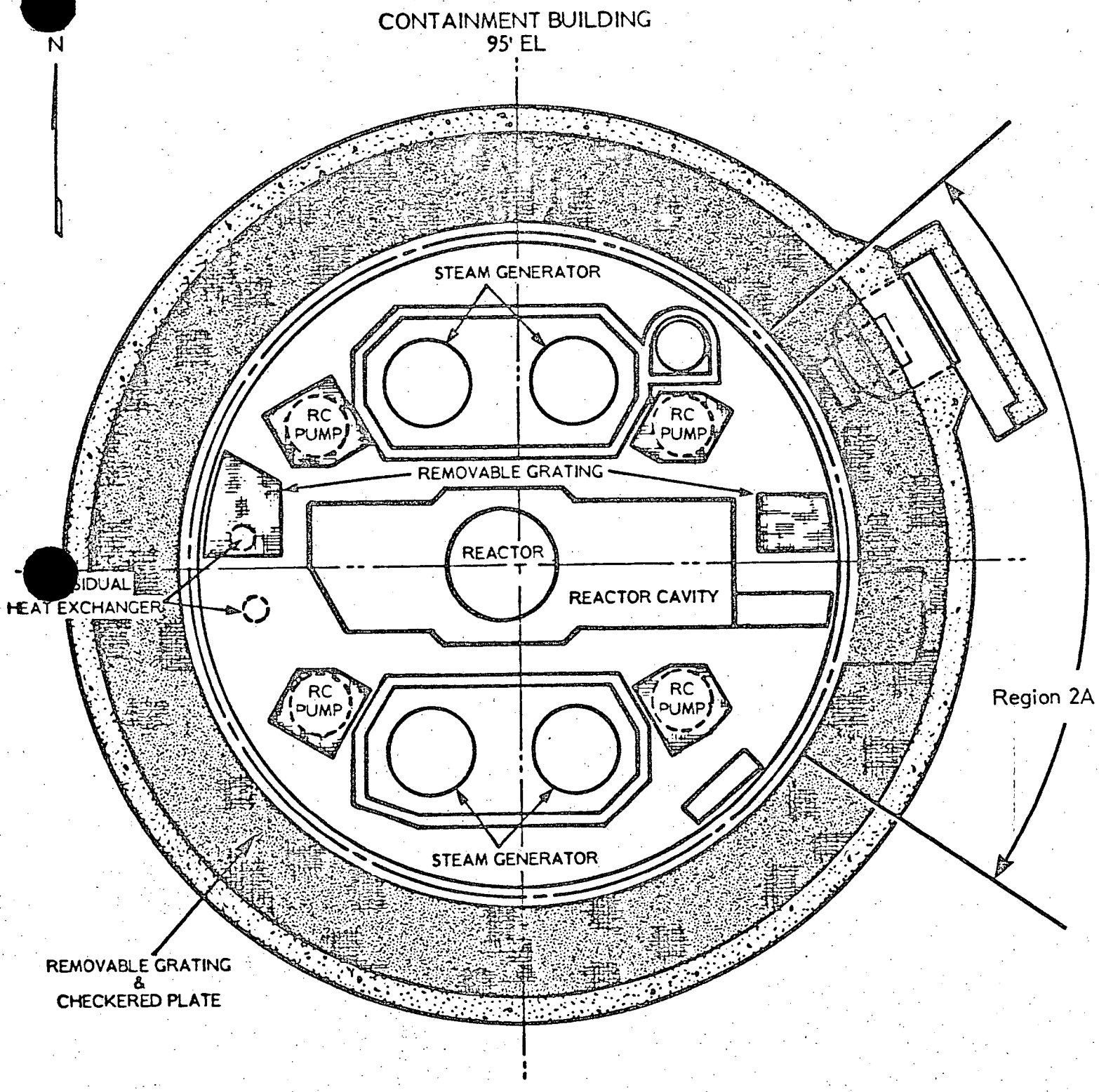

FIGURE 2
REGION 2 - ANNULUS

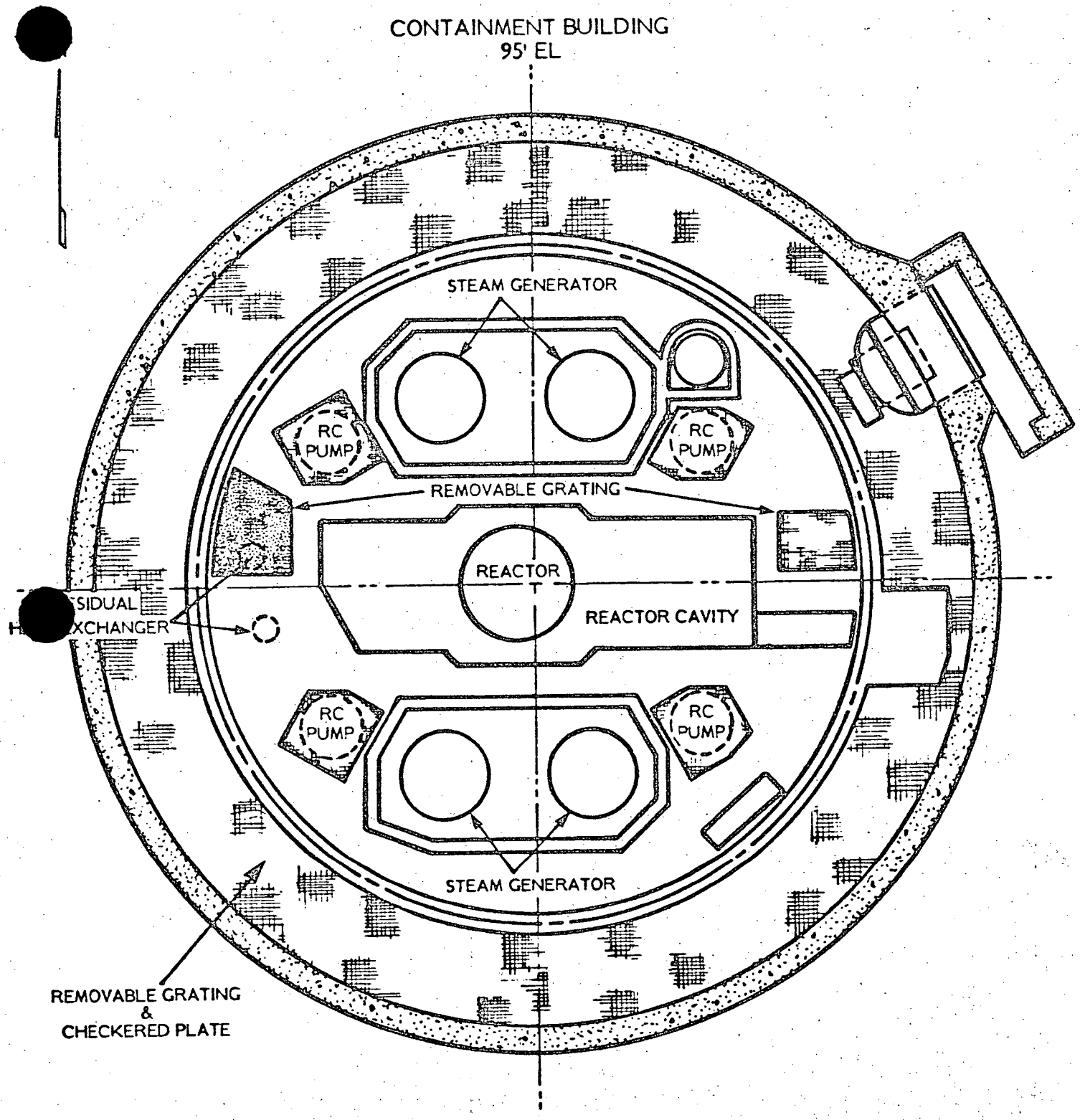

FIGURE 3

REGION 3 - GRATING OVER RHR HEAT EXCHANGERS

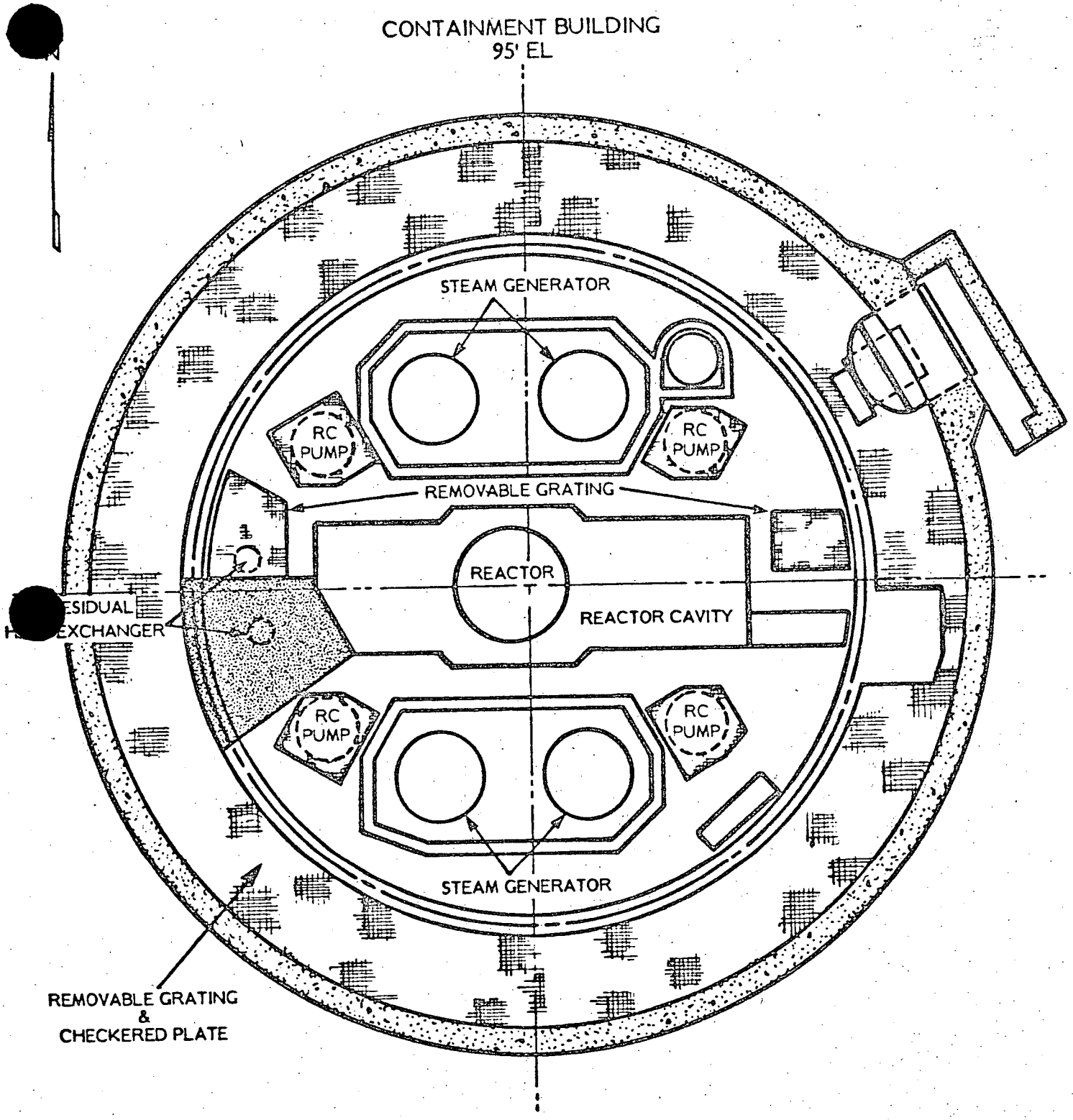

FIGURE 4

REGION 4 - OPERATING DECK HEAD STORAGE STAND
OVER RHR HEAT EXCHANGERS

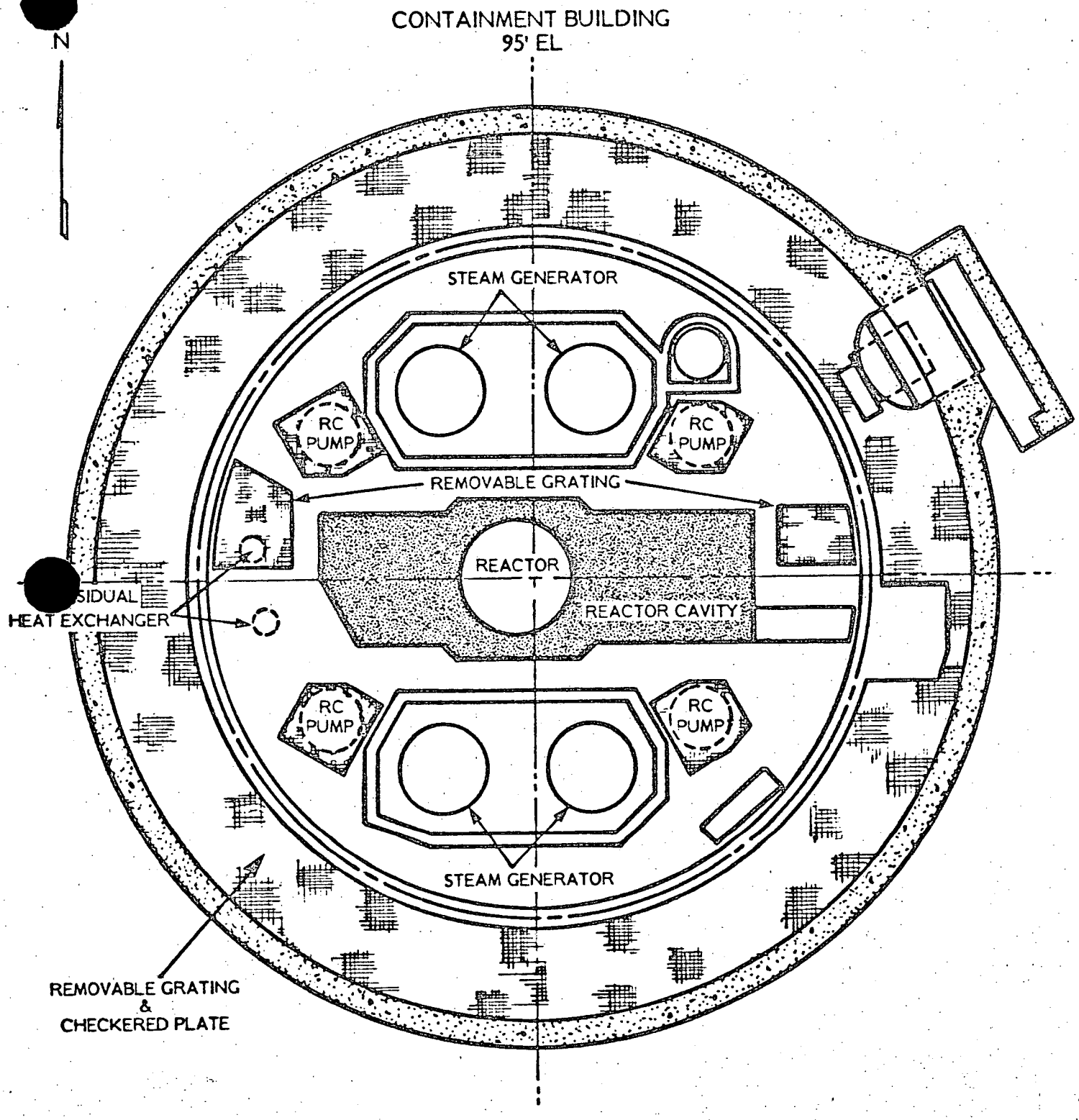

FIGURE 5

REGION 5 - REACTOR CAVITY

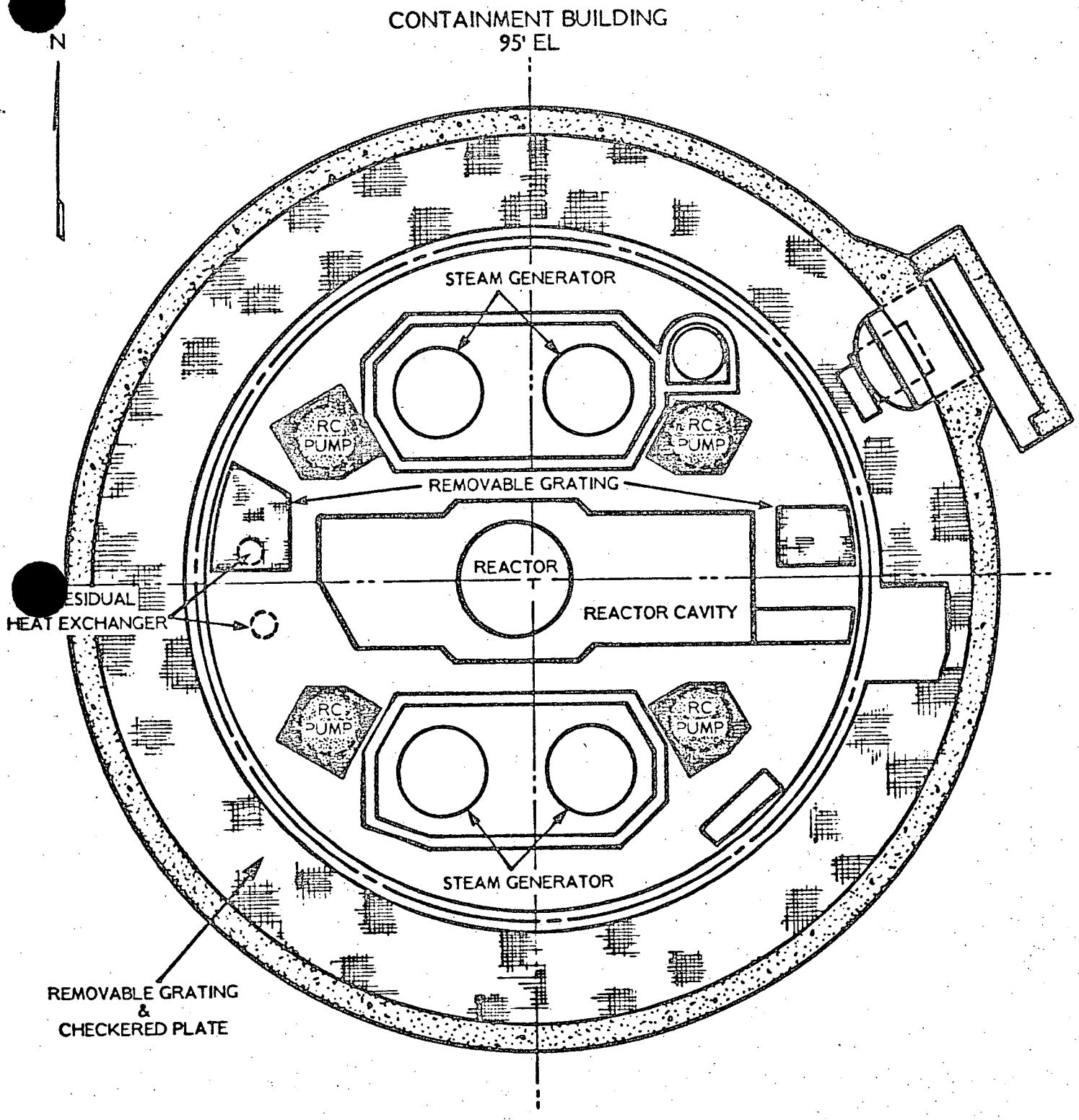

FIGURE 6

REGION 6 - REACTOR COOLANT PUMP AREA

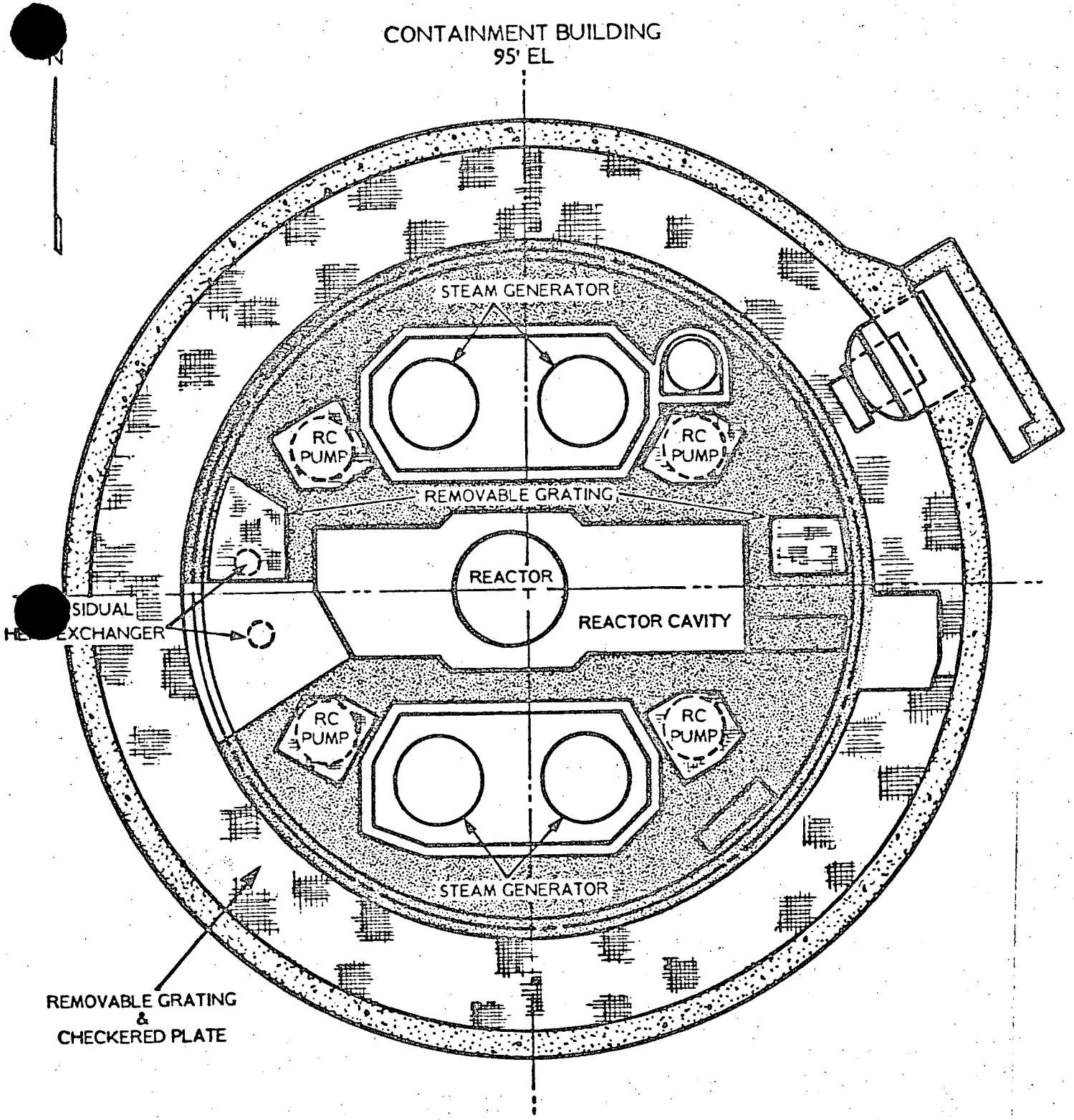

FIGURE 7

REGION 7 - OPERATING DECK INSIDE CRANE WALL

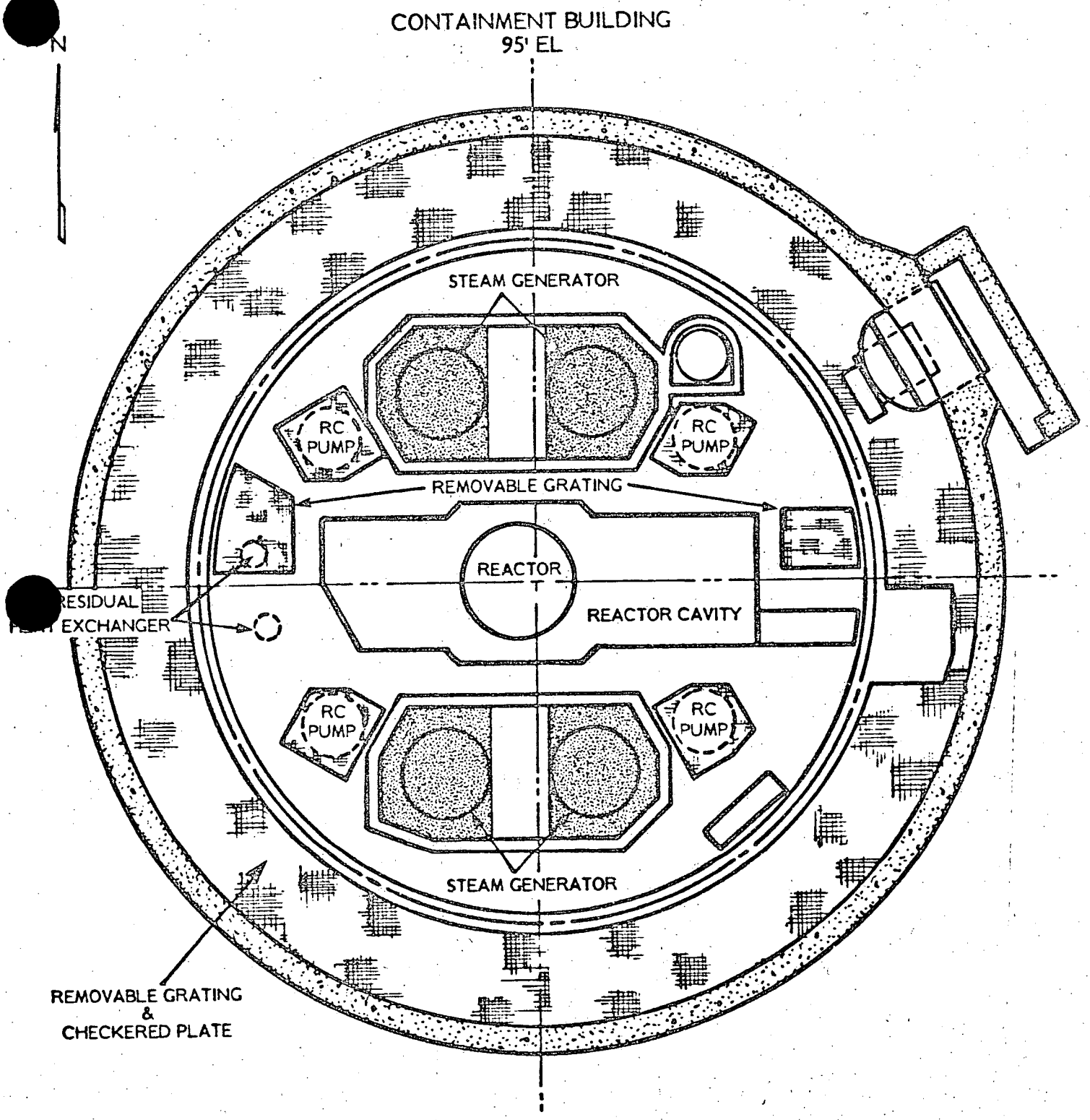

FIGURE 8

REGION 8 - STEAM GENERATORS

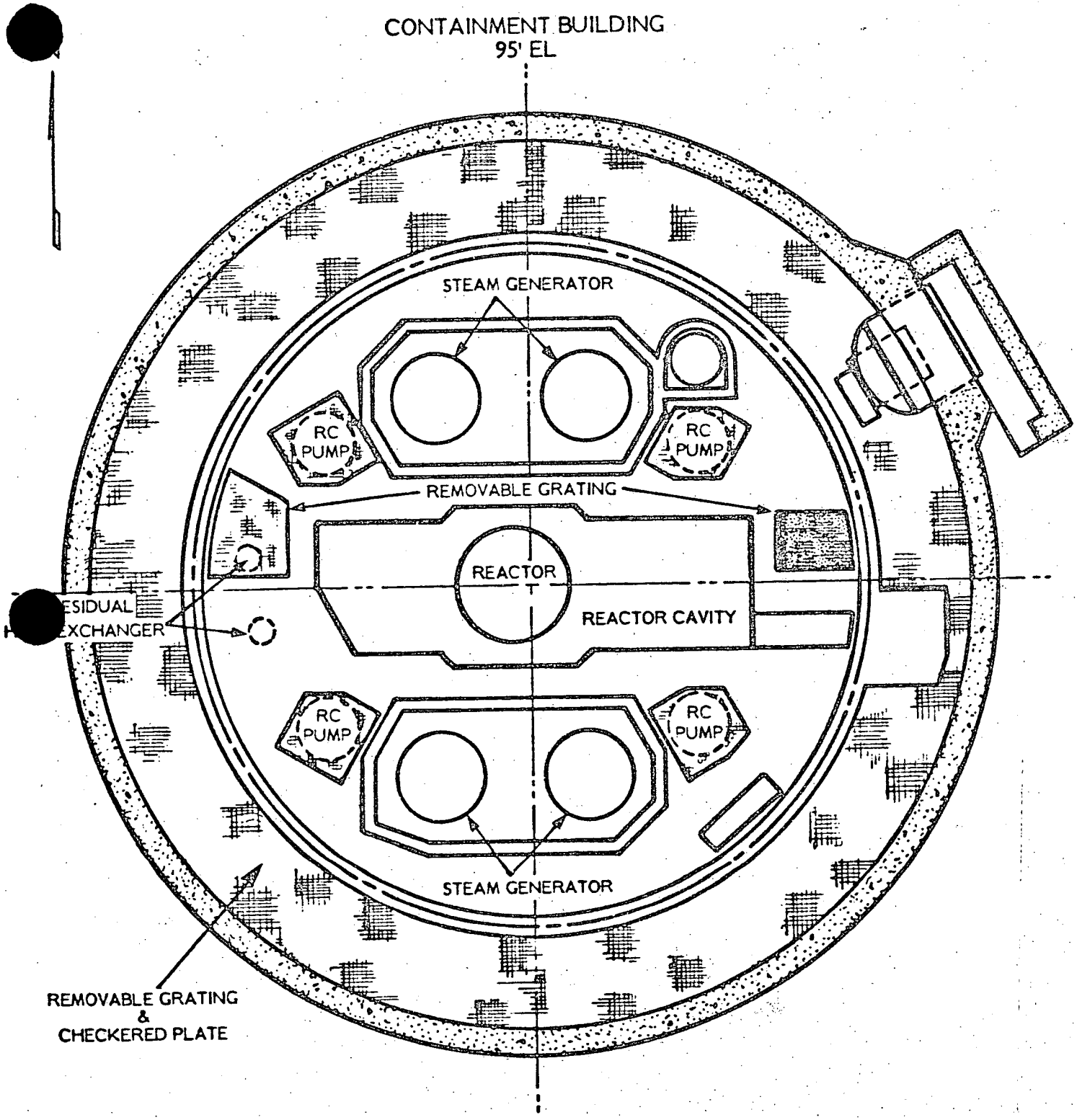

FIGURE 9

REGION 9 - GRATING OVER INSTRUMENT RACKS AND VALVE ACCESS AREA

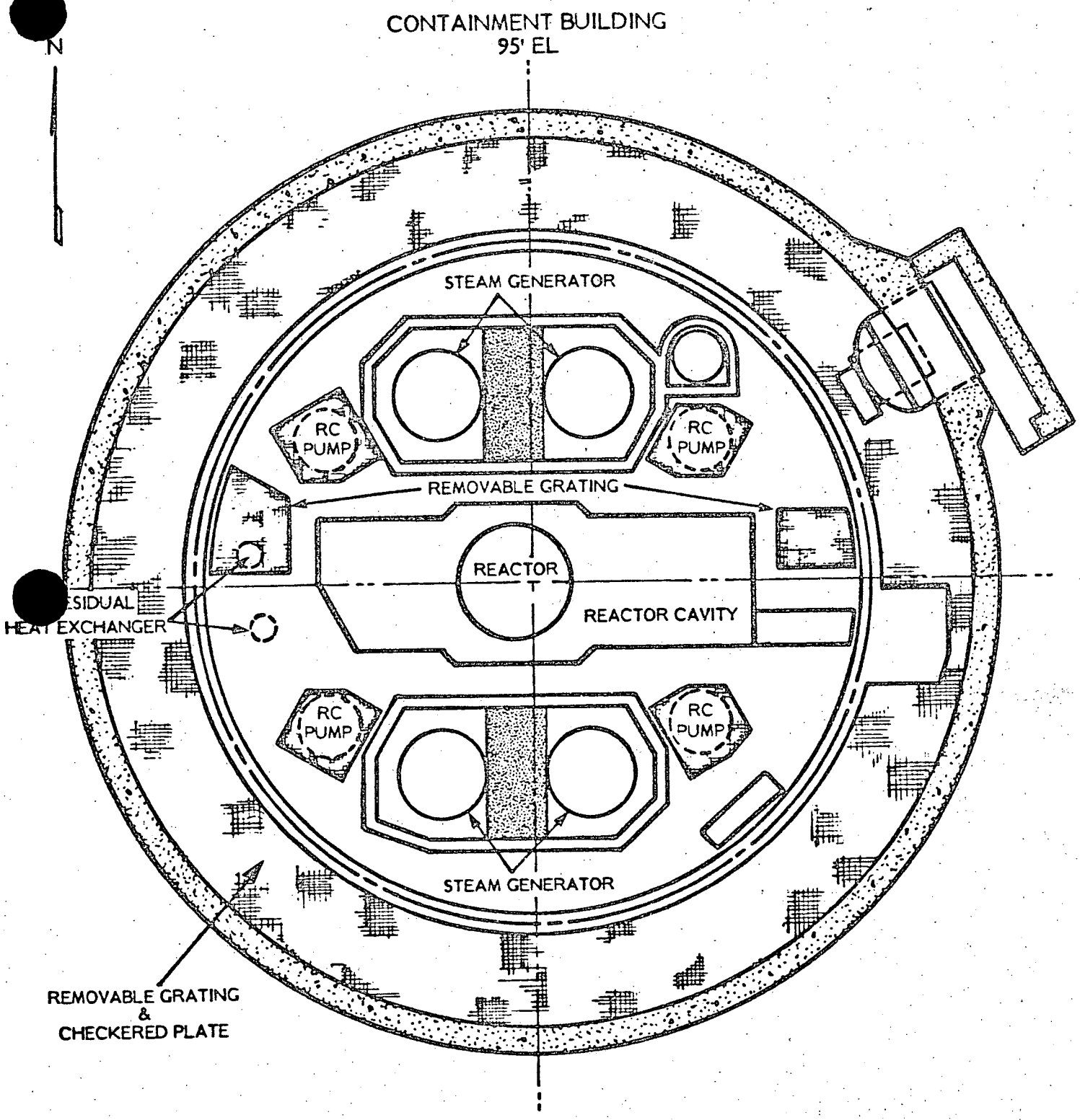

FIGURE 10

REGION 10 - SLABS BETWEEN STEAM GENERATORS (95' EL)

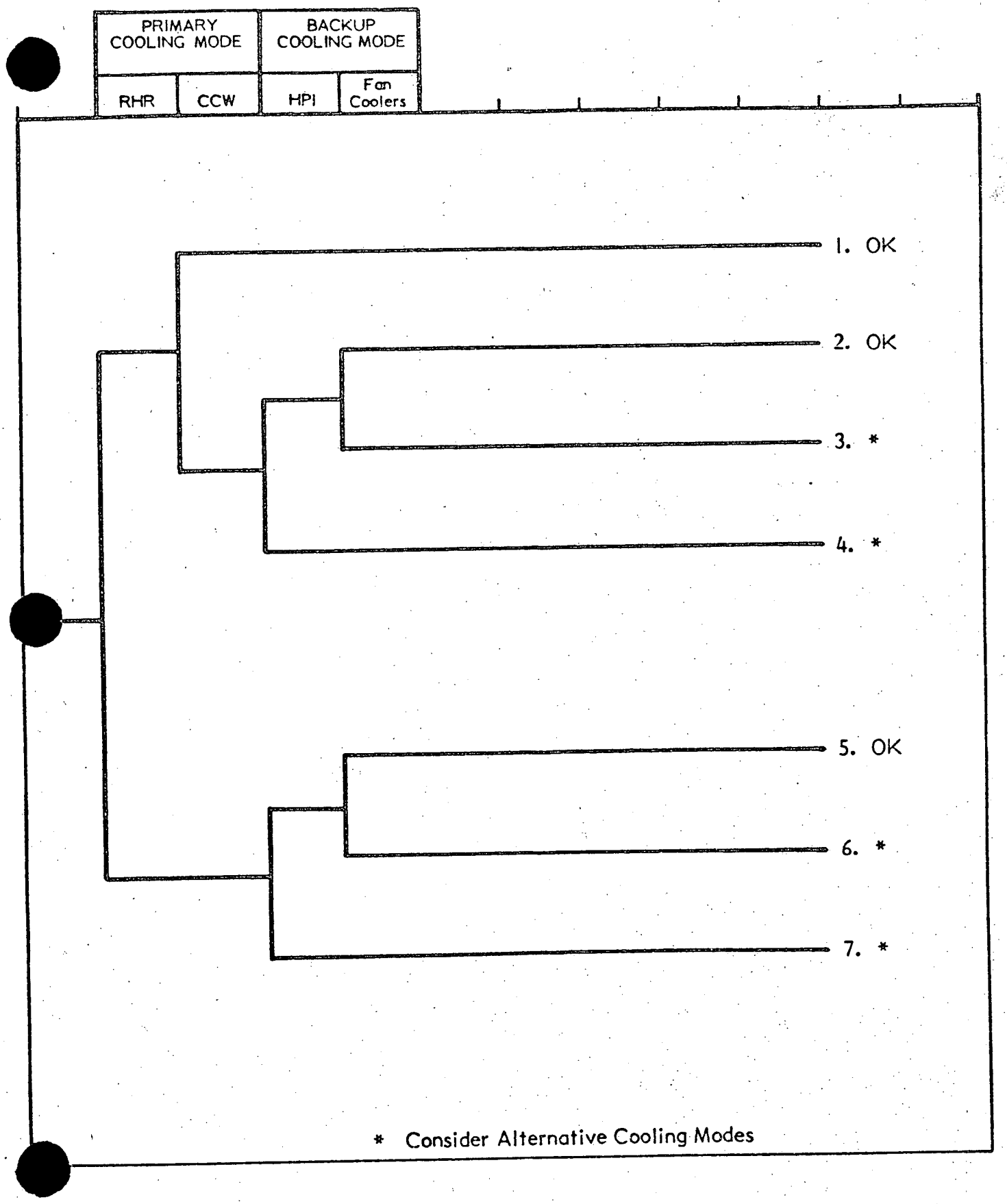

FIGURE 11

CASE 1 - RV HEAD REMOVED - NO RCS BREAK

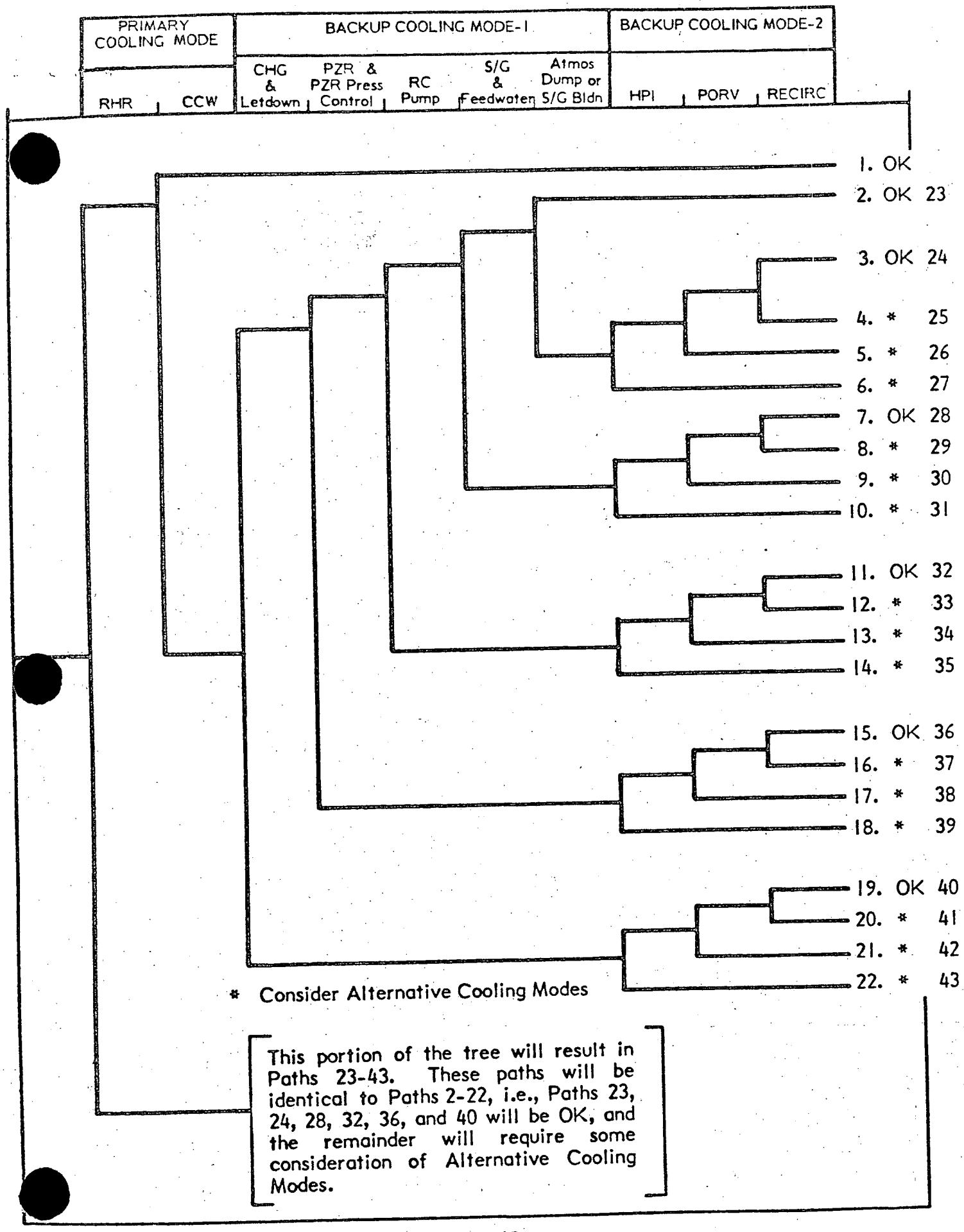

FIGURE 12

CASE 2 - RV HEAD IN PLACE - NO RCS BREAK

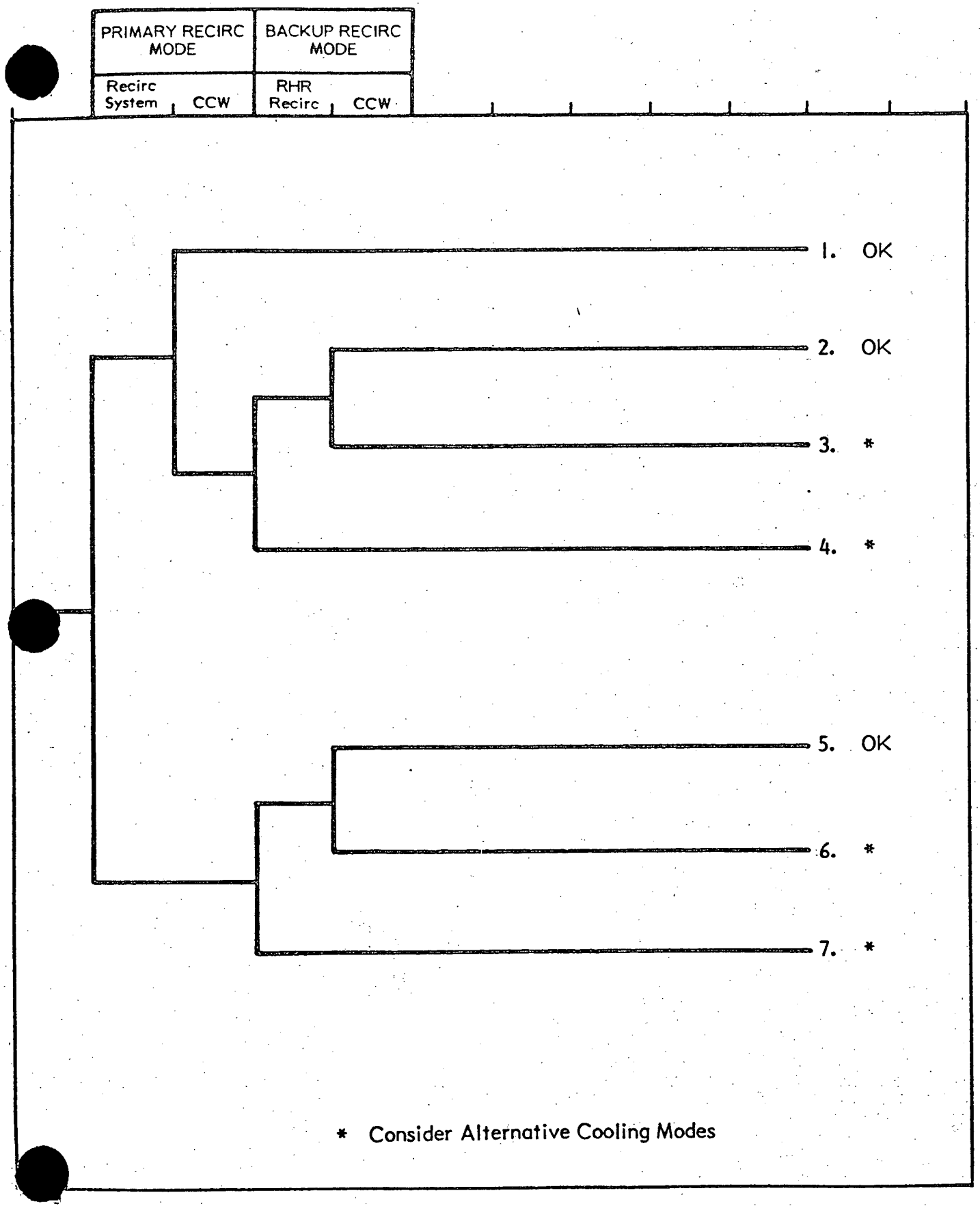

FIGURE 13

CASE 3A - RV HEAD REMOVED - RCS BREAK

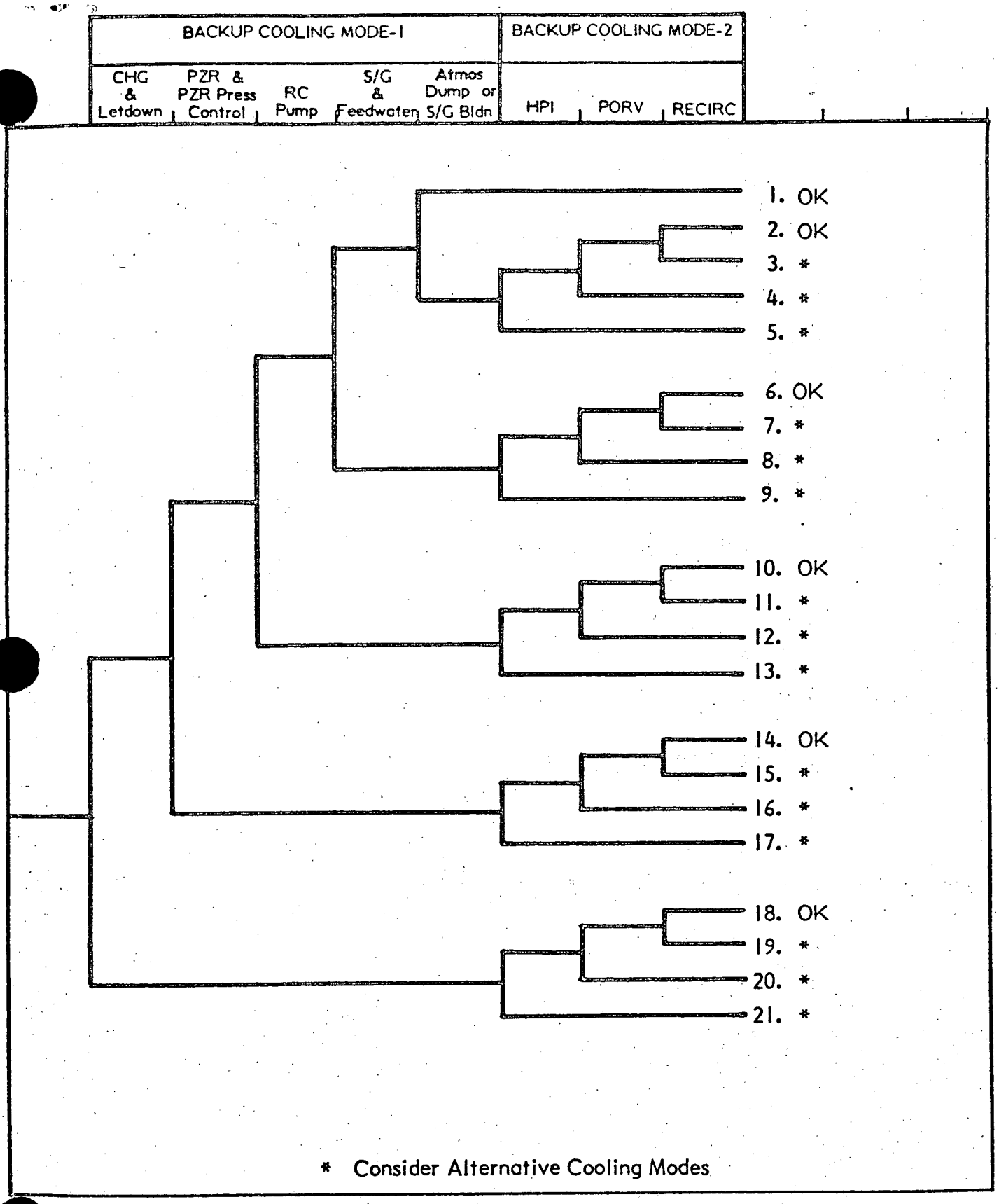

FIGURE 14
CASE 3B - RV HEAD IN PLACE - SMALL RCS BREAK

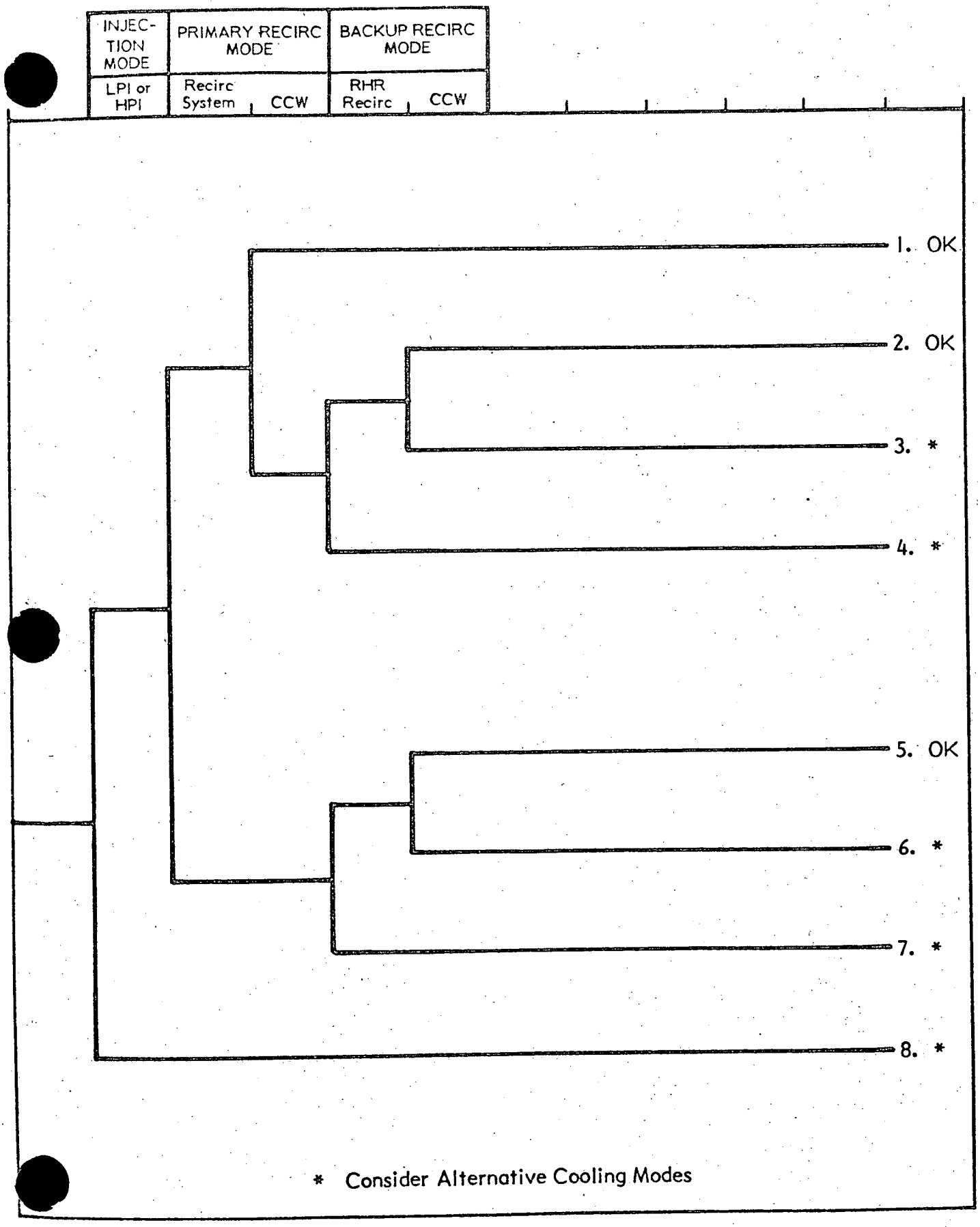

FIGURE 15
CASE 3C - RV HEAD IN PLACE - LARGE RCS BREAK