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February 26, 1982
JPN-82-25

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

Attention: Mr. Domenic Vassallo, Chief
Operating Reactors Branch No. 2
Division of Licensing

Subject: James A. FitzPatrick Nuclear Power Plant
Docket No. 50-333
Control of Heavy Loads - NUREG-0612

References:
1. Letter, D. G. Eisenhut (NRC) to all Operating Reactors dated December 22, 1980
2. Letter, J. P. Bayne (PASNY) to T. A. Ippolito (NRC) dated October 15, 198. (JPN-81-82).

Dear Sir:

Reference 1 requested a review of heavy load-handling operations and a two-phase submittal of evaluations of their conformance to the guidelines of NUREG-0612. The Power Authority completed the first phase of this review and submitted the six month report via Reference 2. This report identified procedure changes necessary to meet the NUREG-0612 interim action guidelines. The Authority committed to implement these changes during the Reload 4/Cycle 5 refueling outage now in progress. In accordance with this commitment, these procedure changes have been completed.

The enclosed nine month report provides the results of the second phase of the review and the Power Authority's response to the items in Sections 2.2, 2.3 and 2.4 of Enclosure 3 to the December 22, 1980 letter. The analyses completed to date and summarized in this report indicate that the consequences of certain load drops do not, or might not, meet the guidelines of NUREG-0612. The Power Authority will prohibit the load lifts identified below until further evaluation demonstrates that the likelihood of the drop is sufficiently small or, constraints will be imposed on the lift so that the consequences of a drop are acceptable.

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1. The steam separator assembly will not be lifted into or out of the reactor vessel.

2. The refueling slot shield plugs will not be lifted unless both of the following conditions are met:

   a. no freshly discharged spent fuel is stored in the vicinity of the lift; and,

   b. no fuel is stored in non-boral racks in the vicinity of the lift.

3. No spent fuel or radioactive waste shipping casks will be lifted in the reactor building.

4. Recirculation pump motors will not be lifted in the northwest equipment hatch (Region 8) unless the plant is in the refueling condition (as defined in the enclosure).

5. No heavy loads will be lifted through the RHR heat exchanger hatches unless the plant is in the refueling condition (as defined in the enclosure).

The Power Authority will also evaluate reactor vessel head drop scenarios in addition to the one included in the enclosed report. These evaluations, and the others mentioned above, will be submitted as soon as they are completed. The load handling restrictions described above will assure the safety of load handling operations in the interim.

If you have any further questions, please do no hesitate to contact us.

Very truly yours,

Robert F. Burns for

J. P. Bayne
Senior Vice President
Nuclear Generation

cc: Mr. J. Linville
Resident Inspector
U. S. Nuclear Regulatory Commission
P. O. Box 136
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Mr. Ron Barton
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RESPONSES TO REQUESTS FOR INFORMATION
IN SECTIONS 2.2 AND 2.3 OF ENCLOSURE 3
TO NRC DECEMBER 22, 1980 LETTER

2.2 SPECIFIC REQUIREMENTS FOR OVERHEAD HANDLING SYSTEMS OPERATING IN REACTOR BUILDING

NUREG-0612, Section 5.1.4, provides guidelines concerning the design and operation of load-handling systems in the vicinity of spent fuel in the reactor vessel or in storage. 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 over spent fuel in the storage pool or in the reactor vessel.

RESPONSE: Three handling systems operating in the reactor building are capable of carrying loads over spent fuel in the storage pool or in the reactor vessel. These handling systems are described in Table I.

TABLE I

HANDLING SYSTEMS CAPABLE OF CARRYING LOADS
OVER SPEND FUEL IN THE STORAGE POOL
OR IN THE REACTOR VESSEL

CRANE	TYPE	CAPACITY	EQUIPMENT DESIGNATOR
Reactor Building Crane	Overhead Bridge	Main Hoist - 125 tons Aux. Hoist - 20 tons Aux. Hoist - ½ ton	CR-2
Refueling/Service Jib Cranes (3)	Pillar	750 lbs.	JC-25A,B,C
Refueling/Service Hoists (3)*	Base	750 lbs.	JC-27A,B,C

* The 3 refueling/service hoists are each mounted on the 3 refuel/service jib cranes.

ITEM 2.2-2 Justify the exclusion of any cranes in this area from the above rifying that they are incapable of carrying heavy loads or are permanently prevented from movement of heavy loads over stored fuel or into any location where, following any failure, such load may drop into the reactor vessel or spent fuel storage pool.

RESPONSE: The Refueling/Service Jib Cranes and the Refueling/Service Hoists may be excluded from the above category. Justification for exclusion of these handling systems was provided in our response to Section 2.1, Item 2, of our initial submittal responding to the NRC letter of December 22, 1980 (letter from George T. Berry to Darrell G. Eisenhut dated October 15, 1981). That response indicated that these hoists are being derated from 1,000 lbs. to 750 lbs. and will be clearly marked with the lower rating. Therefore, these hoists are excluded from the NUREG-0612 criteria since they will not be allowed to handle loads greater than 750 lbs.

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: The Reactor Building crane was evaluated to industry standards CMAA 70-1975 (reference 5) and ANSI B30.2-1976 (reference 6). It was found to meet these standards, with two exceptions which were justified in reference 3. Therefore, based on those evaluations, the reliability of the Reactor Building crane is demonstrated.

Notwithstanding the fact that the lifting system including the Reactor Building crane, lifting slings and strongbacks, complies with the intent of applicable industry standards and possesses demonstrated margins to failure, rather than relying on the reliability of the lifting system, an evaluation has been performed to assess the consequences of postulated drops of heavy loads. Therefore, although these heavy load drops need not be postulated, the load handling reliability of the Reactor Building crane was conservatively not relied on (except as indicated below), and the consequences of postulated load drops have been evaluated.

The only case where load handling reliability was considered was with respect to the main hoist load block and hook. NUREG-0612 (reference 3/4) 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, shield plugs, and the dryer and separator units. 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 the crane load block alone.

The only 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 Fitzpatrick Reactor Building crane is provided with two diverse upper limit switches to interrupt power to the hoist motor prior to "two blocking." When power is removed, holding brakes are automatically applied. One of the two limit switches is a geared limit switch driven off the drum shaft. The other is a counter weight switch that is released when the load block comes up against a trip bar; the trip bar will stop power to the hoist below the low point of the sheave assembly.

The holding brakes are solenoid released, and spring applied on loss of power to the solenoid. Two holding brakes are provided, either of which has sufficient capacity to hold the rated load (each brake is 150% of full motor torque). Additionally, inspection and maintenance procedures assure that the limit switches and holding brakes are functional and properly adjusted.

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

Nonetheless, an analysis of a load block and hook drop from the highest possible carry height onto the operating/refueling floor was performed to verify the capability of the floor to withstand the impact of such a drop. The results of the analysis indicate that while concrete scabbing on the underside of the floor is

predicted, no gross failure or penetration will result. The consequences of scabbing have been considered in the systems evaluations and were found to be acceptable.

Therefore, although drop of the load block and hook need not be postulated, even if they were to drop on the operating/refueling floor, the consequences are acceptable.

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 2.3 of this request. With respect to Criteria I through III, provide a discussion of your evaluation of crane operation in the Reactor Building and your determination of compliance.

ITEM 2.2-4a 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 bypass of such interlocks.

RESPONSE: An interlock system is provided for the Reactor Building Crane that prohibits movement of heavy loads over the Spent Fuel Pool during normal load handling operations. This interlock system is described in the response to Item 3.a in our initial submittal responding to the NRC's letter of December 22, 1980.

The keys which allow operation in modes other than the NORMAL mode must be obtained from the plant Shift Supervisor. Bypassing of any of the interlocks modes is controlled by plant procedures. Use of these procedures requires management approval and use of a Work Tracking Form (WTF). The Shift Supervisor must approve each WTF prior to commencing work. These procedures are contained in Maintenance Procedure MP 17.1. Deviations from procedures require approval in the manner described in the response to Item 3.b in our previous submittal.

Notwithstanding the fact that the interlock system prohibits movement of heavy loads over the Spent Fuel Pool, in the unlikely event that the interlock fails to protect against drops of certain items handled near the pool (e.g., the portable radiation shield and the refueling slot plugs), procedures will be implemented to assure that no fuel is stored in nonboral racks near the edge of the pool and that no newly spent fuel will be stored in that area. Therefore, the consequences of even unlikely drops into the spent fuel pool are determined to acceptably comply with the guidelines of NRC NUREG-0612.

One of the interlock modes (the Cask Handling Mode) described in that response does allow movement of heavy loads such as casks over a small 1' x 4' area in the southwest corner of the pool, i.e., load movement is prohibited directly over spent fuel. Concerns regarding cask tipping and cask impact on the pool floor have previously been raised by the NRC and addressed by the Authority. To address these concerns, we described, in our letter dated November 12, 1974, a Fuel Cask Drop Protection System proposed for installation at JAF. By letter dated April 22, 1977, we indicated that we would reevaluate the need for installation of that system prior to shipment of spent fuel, which was not anticipated before the late 1980s.

As a result of the current review of heavy load handling operations at JAF, we have determined that positive protection of spent fuel in the pool is provided through use of the Fuel Cask Drop Protection System. Therefore, installation of an approved licensed system will be complete prior to shipment of spent fuel at JAF.

ITEM 2.2-4b Where reliance is placed on the operation of the Stand-by Gas Treatment System, discuss present and/or proposed technical specifications and administrative or phyiscal controls provided to ensure that these assumptions remain valid.

RESPONSE: In no cases is reliance placed on operation of the Stand-by Gas Treatment System. However, JAF technical specifications require that secondary containment be maintained and that one train of the Stand-by Gas Treatment System be operable when handling irradiated fuel or fuel casks.

ITEM 2.2-4c 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 validity of such considerations.

RESPONSE: In no cases is reliance placed on other site-specific considerations. However, as discussed in response to Item 2.3 following, certain load lifts are scheduled to occur only during the refueling mode.

ITEM 2.2-4d 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.

The reactor pressure vessel (RPV) head shown in Figure 1 weighs 73 tons, including the weight of the RPV head strongback. Removal and reassembly of the head are accomplished according to Fitzpatrick Maintenance Procedures MP 4.1 and MP 4.2, respectively.

During normal refueling operations, the RPV head assembly is lifted out of the reactor cavity from about elevation 345', to the operating/refueling floor at elevation 369'-6". The vessel head is lifted at a time when no water is in the cavity. As a result, the evaluation of the head drop was performed assuming a 25 foot drop through air. Once at the desired height, the RPV head is moved south toward the head holding pedestal which rests on the operating/refueling floor. Reassembly is in the reverse order. Several head drop scenarios over the RPV can be postulated in the unlikely event of a failure of the reactor building crane. The potential for fuel damage, or a loss of safe shutdown capability affecting the ability to get water to the core for cooling purposes, was reviewed for the case of the RPV head drop from the normal carry height of the reactor building crane, 25 feet through air, impacting on the RPV flange. The general methods of analysis which are documented in references 9, 10, and 29 through 31 were used, along with parameters which are applicable to the Fitzpatrick plant. The RPV head drop was analyzed using two methodologies. The behavior of the RPV below the head flange, and at the support skirt, was analyzed for the load resulting from the 25 foot drop of the head, including consideration of load amplification due to the dynamic impact factor. Stresses and stability were

evaluated at significant regions. In addition to verifying the load carrying capabilities of the RPV, an assessment of the energy of the 25 foot drop of the head was performed. The objective of this evaluation was to verify that the RPV and skirt can survive the effects of the drop through energy absorption and dissipation. The resulting velocity of impact of this 25' drop through air, was calculated based on the equation of motion:

$$d = V_i t + \frac{1}{2} a t^2$$

Thus, the RPV head was found to impact the flange at a velocity of 40.1 feet per second. The energy of the RPV head drop can be calculated, considering the velocity at impact and recognizing that momentum is conserved when the RPV head impacts the flange. Therefore, assuming a reactor vessel head and assembly weight of 73 tons, the resultant energy was calculated to be 8.18×10^5 ft.-lbs.

The major portion of the impact load of the RPV head is transmitted directly to the RPV flange. The load path is then through the RPV shell to the supporting skirt which absorbs the entire impact. The dynamic model conservatively neglects energy absorption by the reactor internals. An assessment of the load path and supporting system revealed the response behavior of the system and the critical load to the component due to the load drop. The critical load was defined as that load which caused initial yielding of the weakest member. The system was then analyzed to determine its capability to absorb energy based on elastic response to this critical load. Thus, the energy absorbing capacity of the system was calculated to be 1.3×10^6 ft.-lbs. Since this is greater than the energy of the head drop, the RPV and support skirt are capable of absorbing the energy of the RPV head drop.

In addition to verifying that the energy of the head drop can be absorbed by the RPV and the skirt, stresses and stability at significant areas (such as the vessel wall, the lower vessel head area, the support skirt, and the RPV and skirt interface) were evaluated. In order to compute the stresses along the load path through the vessel and supporting skirt, it was necessary to calculate the dynamic impact factor, IF. It was assumed that the stresses are distributed in

the same manner as for the case of static loading; however, they are increased by this dynamic impact factor. According to reference 32, Roark and Young, this impact factor can be represented by the ratio:

$$IF = \delta_i/\delta = W_i/W = 1 + (1+2h/\delta)^{1/2}$$

where, δ_i = vertical deformation on impact

δ = static vertical deformation

W_i = force or effective weight upon impact

W = static force or weight of dropped load

h = height of drop

The above formula is based on the assumption that impact strains the elastic body the same way as static loading, i.e., that all of the kinetic energy of the moving body is expended in producing this strain. Actually, on impact some of the kinetic energy is dissipated, and this loss, which can be calculated by equating the momentum of the entire system before and after impact, is most conveniently taken into account by multiplying the available energy (measured by h or by v^2) by a factor K. The above equation can then be rewritten as:

$$IF = \delta_i/\delta = W_i/W = 1 + (1+2Kh/\delta)^{1/2}$$

A number of approximate models are available for estimating the energy loss factor K (reference 32). If a moving body of mass M strikes axially one end of a bar of mass M_1, the other end of which is fixed, then

$$K = \frac{1 + (1/3)(M_1/M)}{(1 + (1/2)(M_1/M))^2}$$

If there is a body of mass M_2 attached to the struck end of the bar, then

$$K = \frac{1 + (1/3)(M_1/M) + (M_2/M)}{(1 + (1/2)(M_1/M) + (M_2/M))^2}$$

Therefore, using the above expressions, the dynamic impact factor can be determined.

The stiffness of the RPV shell and the RPV support were modeled along with the associated masses of the actual system. The static deflection of the RPV flange due to the load was calculated, based on the spring stiffness of the load supporting system (the RPV shell, the bottom head region, and the support skirt region). Based on this deflection and the transfer of momentum between the head and the RPV upon impact, the dynamic impact factor of the head drop was calculated, and the resulting dynamic load was determined. The stress was then calculated for the reactor vessel, and was found to be significantly below the minimum yield stress of 50,000 psi, and the ultimate stress of 75,000 psi. In the bottom head area, and in the supporting skirt, the stresses were also significantly below code allowables.

In addition to evaluating the stress levels in the reactor vessel and support, the stability of the reactor vessel was also considered, to determine the potential for buckling. The reactor vessel was conservatively represented as a thin walled cylindrical tube under uniform longitudinal compression, and the theoretical buckling stress was calculated to be about 400,000 psi. Therefore, since the calculated axial load of the impact on the vessel wall is 10,800 psi, it is obvious that stability of the reactor vessel shell is not a problem. Similarly, a review of the buckling potential of the skirt shows that the potential for buckling in this area is not predicted, since the theoretical value is calculated to be about 165,000 psi.

Based on the evaluations above, reactor vessel integrity is maintained and no fuel damage is predicted as a result of the reactor vessel head drop. The limiting situation for fuel damage was judged to be the postulated drop of the upper internals package into the vessel. This includes an evaluation of both the steam dryer and the steam separator. A conservative structural evaluation was performed to determine if fuel integrity could be demonstrated for each of these postulated drops.

In order to determine the worst case between the dryer drop of 24' through air and the steam separator drop of 32.5' through water, the impact energies of each of these drops were determined. Based on the result of the impact energy evaluations (the separator drop energy was determined to be 3.4×10^5 ft-lbs and

Structural damage in Region 6 is predicted to be limited to scabbing of concrete under the refueling slab. Damage to equipment at the 344' elevation below this region was therefore evaluated on the basis of the potential for the scabbing. Loss of the only safe shutdown equipment that could be impacted in this region would result in the inability to utilize one of the two redundant Low Pressure Coolant Injection (LPCI) loops to provide makeup during reactor depressurization/cooling. The other loop of LPCI would be unaffected. In addition, other systems such as Low Pressure Core Spray could be utilized to accomplish this function. Therefore, the consequences of load drops in Region 6 were determined to acceptably comply with NRC evaluation criteria.

Region 7

Equipment, fuel containers, and shipping casks are moved in the southeast equipment hatch (Region 7). Structural evaluations were performed to determine the consequences of postulated drops in terms of overall structural failure and local structural response. For those heavy loads whose controlling mode of response was determined to be local in Table 2, the consequences of the load drops were found to be acceptable. That is, no penetration of the 272' elevation floor slab was predicted for those load drops. However, the bounding drops were determined to be those controlled by overall structural modes. Specifically, the shipping casks were determined to result in the worst case consequences.

Shipping casks have not been selected to date, however the 34 ton Chem-Nuclear Systems, Inc. (CNS) 4-45 cask is being considered as a possible candidate fortransporting radioactive material. This cylindrical cask is 173 1/8" long and 42 1/2" in diameter except for 31 5/8" at the end which is 40 1/2" in diameter. In addition, the consequences of handling accidents postulated for a larger, as yet unspecified fuel assembly cask weighing 110 tons, measuring 5' in diameter and 18' in height were also evaluated.

The 272' elevation concrete slab at the bottom of the equipment hatch is 2'3" thick and reinforced with #11 reinforcing bars each way - each face. The slab is supported by a 7' 2" thick, 12' wide beam that spans north - south, approximately 24' between the inside and outside crescent walls, and approximately 32' from

the inside crescent wall to a support pedestal which is located at the reactor side of the suppression pool. The beam is the principal load carrying member and has been provided for rail car loadings. The beam is located to the east side of the equipment hatch projection, leaving only the 2'3" slab for protection of the northwest quadrant under which the suppression pool is located.

For the cask drop evaluation, the potential for perforating the 2'3" slab in the northwest quadrant of the hatch was found to be high. Additionally, it was determined that the beam is subject to shear failure for postulated drops of either cask in the immediate vicinity north or south of the intersection with the supporting inside wall of the crescent area. Therefore, it was concluded that postulated drops in the equipment hatch area could cause abrupt failure of the concrete structure above the suppression pool leading to impact and the potential for loss of leaktight integrity of the suppresion pool.

Potential damage to safe shutdown equipment in this region was investigated in the vicinity of the hatch at the 272' elevation and below. Safe shutdown equipment in these areas whose loss could potentially result in the inability to achieve and maintain safe shutdown includes: 1) the suppression pool and 2) RHR service water piping affecting both of two redundant loops. The suppression pool is required to successfully accomplish reactor pressure relief/depressurization and initial cooldown if isolated from the main condenser. RHR service water is required to provide cooling water to the RHR heat exchangers, which are utilized to remove decay heat in the long term in all of several different possible system arrangements for accomplishing decay heat removal.

While the consequences of the bounding cask drop onto the track bay floor below the hatch were found to be unacceptable, several options exist and are currently being evaluated for developing a safe solution for moving casks in this region. Results of those evaluations will be provided to the NRC at a later date. No handling of casks in this region will be performed until a safe solution has been implemented.

Region 8

Region 8 includes the equipment hatch in the northwest quadrant of the reactor building. No heavy loads currently carried in this equipment hatch were determined to cause unacceptable structural response for a drop onto the floor slab at elevation 272' for this region. However, the potential for movement of a recirculation pump motor in this region may exist in the future. In such a case, the structural response of the floor at 272' would be bounded by the cask drop analyses previously described for Region 7, i.e., failure of the floor can not be precluded.

Potential damage to safe shutdown equipment in this region was investigated in the vicinity of the hatch at and below elevation 272'. The only safe shutdown equipment that could be impacted in these areas whose loss could eventually result in an inability to achieve safe shutdown is the suppression pool. The suppression pool is required to successfully accomplish reactor pressure relief/ depressurization and initial cooldown, if isolated from the main condenser. For this reason no loads that could result in damage to the suppression pool, if dropped, will be handled in this hatch unless the plant is shutdown and depressurized, and on an appropriate mode of long term cooling. The only such load currently anticipated is a recirculation pump motor, as mentioned above. With this restriction on load handling operations, the consequences of postulated load drops into Region 8 were determined to acceptably comply with NRC evaluation criteria.

Regions 9E and 9W

Regions 9E and 9W are the RHR heat exchanger hatches on the east and west ends of the reactor building. A systems approach to evaluating postulated load drops in these regions was performed. Potential damage to safe shutdown equipment in these regions was investigated separately in the vicinity of each of the hatch openings at the 326' elevation and the 300' elevation, and within and below the RHR heat exchanger cubicles at the 272' elevation. Evaluation was not required at the 344" elevation, because this area was included as part of the evaluation of Region 4. Investigations at the intermediate reactor building

elevations were undertaken because of the potential for a large RHR heat exchanger component (such as the shell) impacting the edge of the relatively small hatch opening at one of these intermediate levels and tipping over onto the floor at that elevation.

The only safe shutdown equipment whose loss could potentially result in an inability to achieve and maintain safe shutdown that could potentially be impacted at elevations 326' or 300' is piping associated with the Emergency Service Water System and Reactor Building Closed Cooling Water System at the 300' elevation. Either of these systems can provide cooling to the RHR pumps. Impact of the piping of interest could result in an inability to provide cooling to the pumps from either of these systems.

Safe shutdown equipment in the RHR heat exchanger cubicles at the 272' elevation whose failure could potentially result in an inability to cool the core include RHR service water piping. Loss of this piping could result in an inability to remove decay heat with the RHR heat exchangers. As indicated above in the discussion for Region 7, the RHR heat exchangers are required to remove decay heat in the long term in all of several different possible system arrangements for accomplishing decay heat removal. Because the ability to accomplish long term removal of decay heat is threatened by potential load drops in these regions, no heavy loads will be carried in these regions unless the plant is in the Refueling Condition, i.e. vessel head removed, reactor cavity filled, and spent fuel pool gate opened. As indicated in the general discussion above regarding safe shutdown functions and systems in various plant conditions, there are a number of ways to successfully cool the core when the plant is in this condition. With these restrictions on load handling operations, the consequences of postulated load drops in Regions 9E and 9W were determined to acceptably comply with NRC evaluation criteria.

FIGURE 1

REACTOR VESSEL ARRANGEMENT

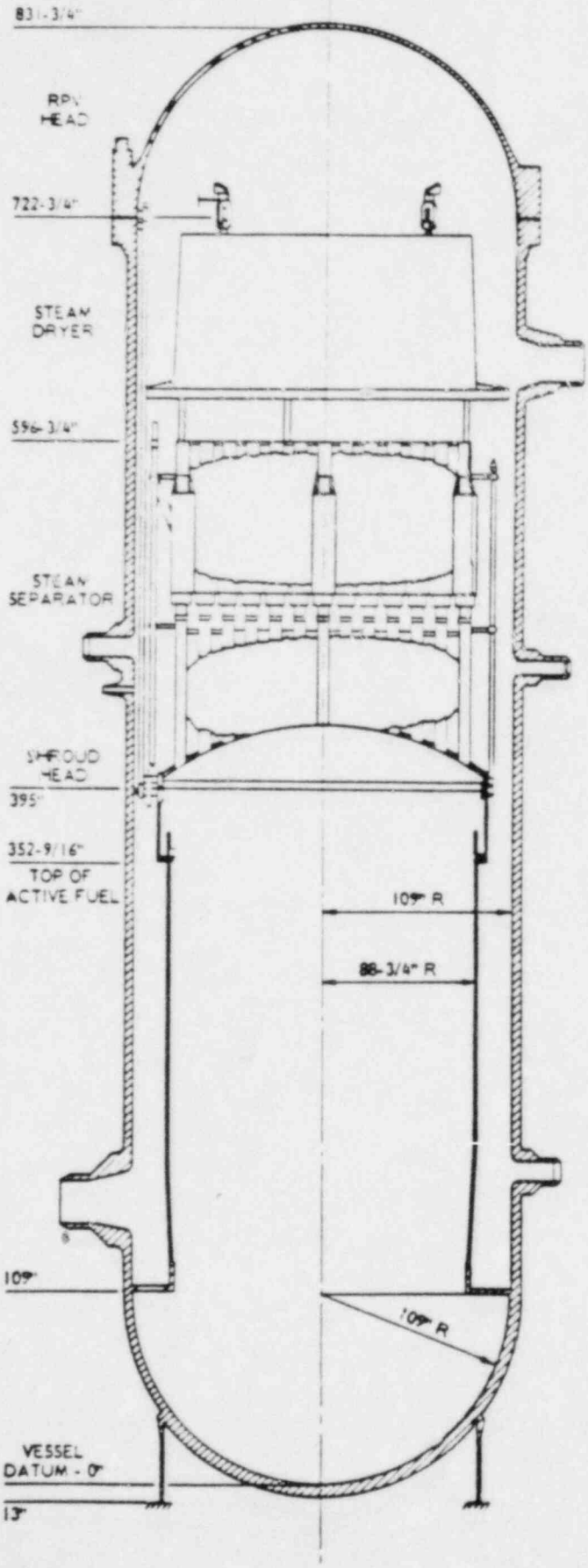


FIGURE 2

LOAD DROP EVALUATION

REGION I - REACTOR VESSEL

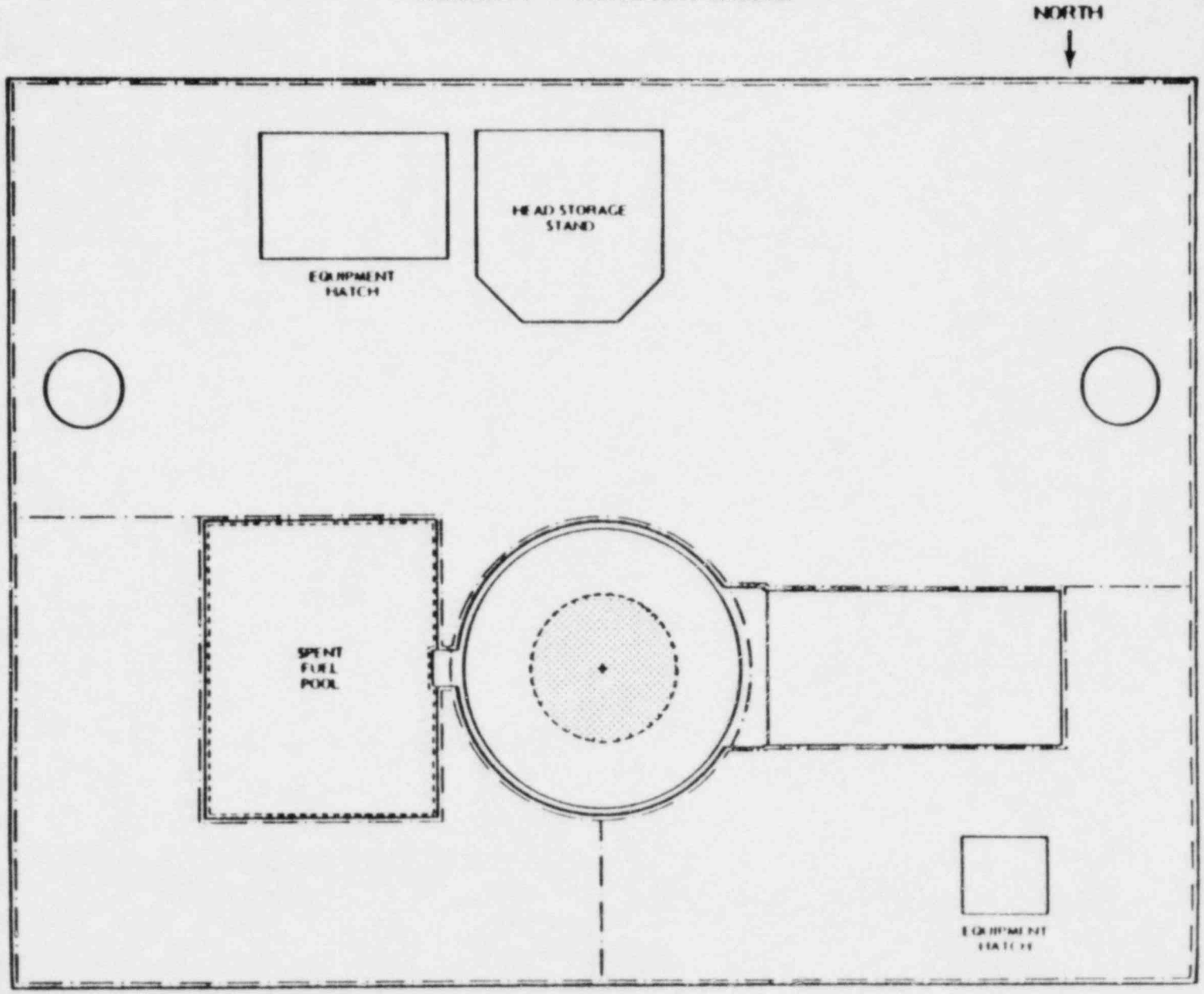


FIGURE 3

LOAD DROP EVALUATION

REGION 2 - SPENT FUEL POOL AREA

CASK HANDLING CONSIDERATIONS

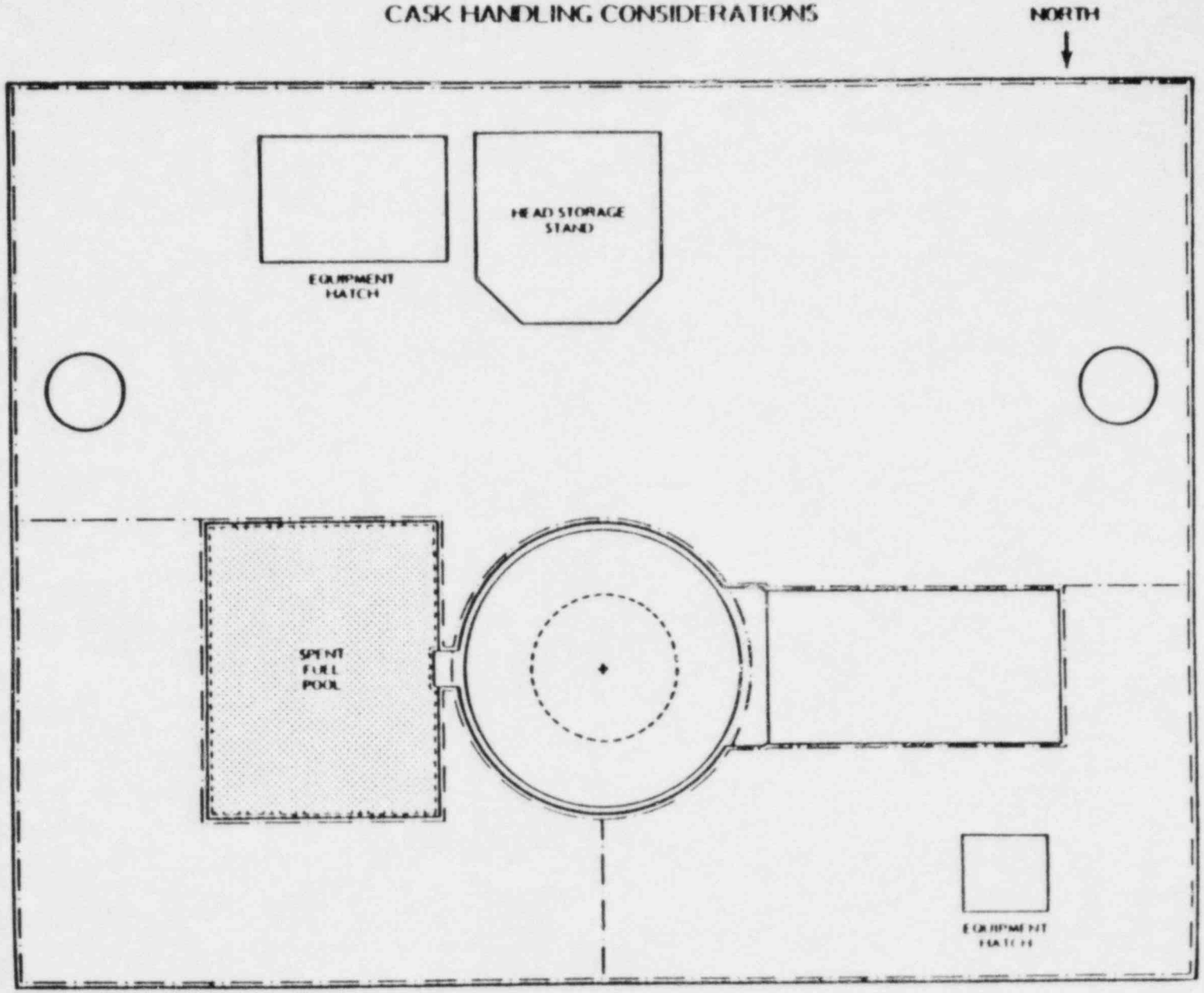


FIGURE 4

LOAD DROP EVALUATION

REGION 3 - INTERNALS STORAGE PIT

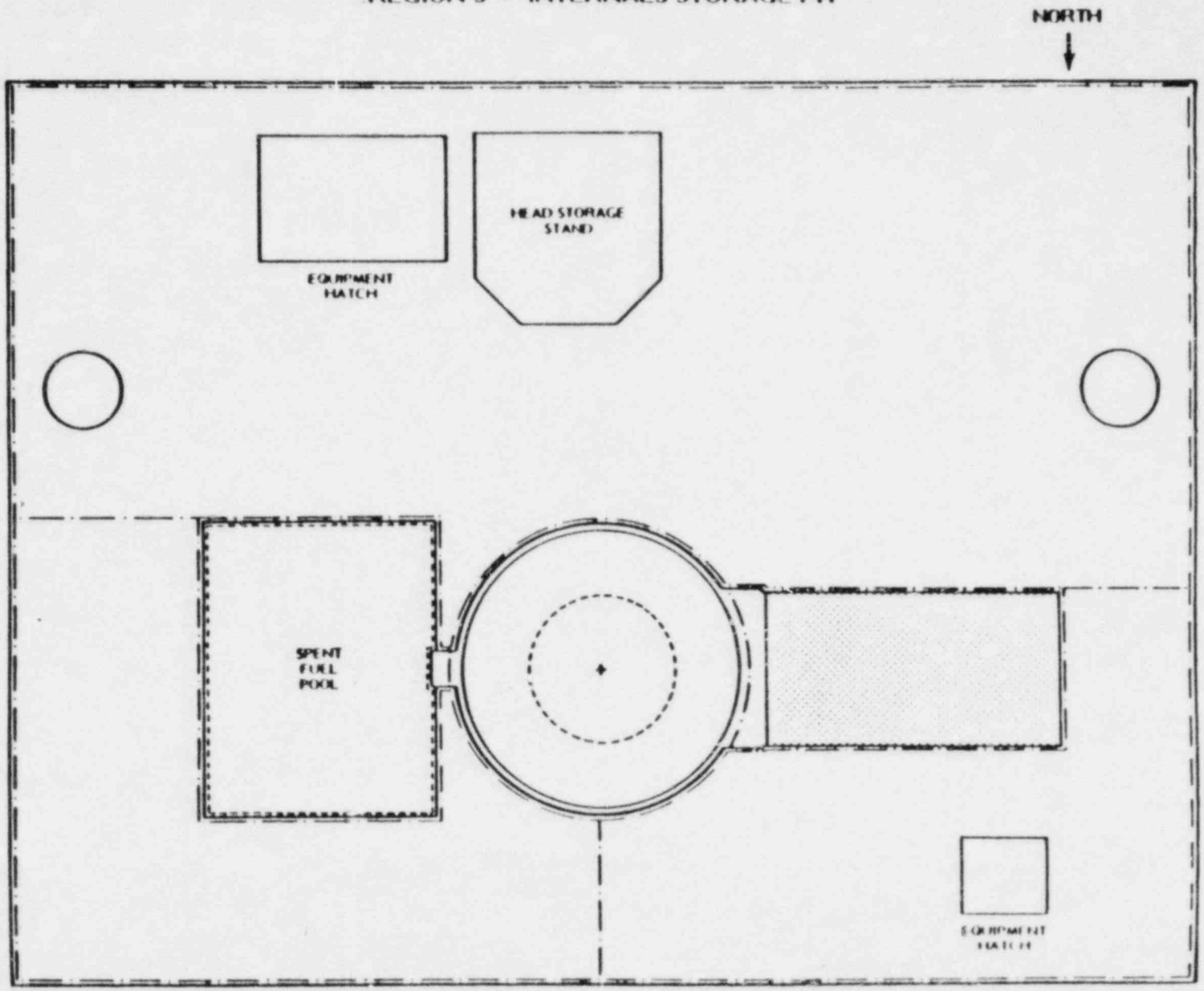


FIGURE 5

LOAD DROP EVALUATION

REGION 4 - SOUTH HALF-AREA

(Except RHR HX and Equipment Hatches)

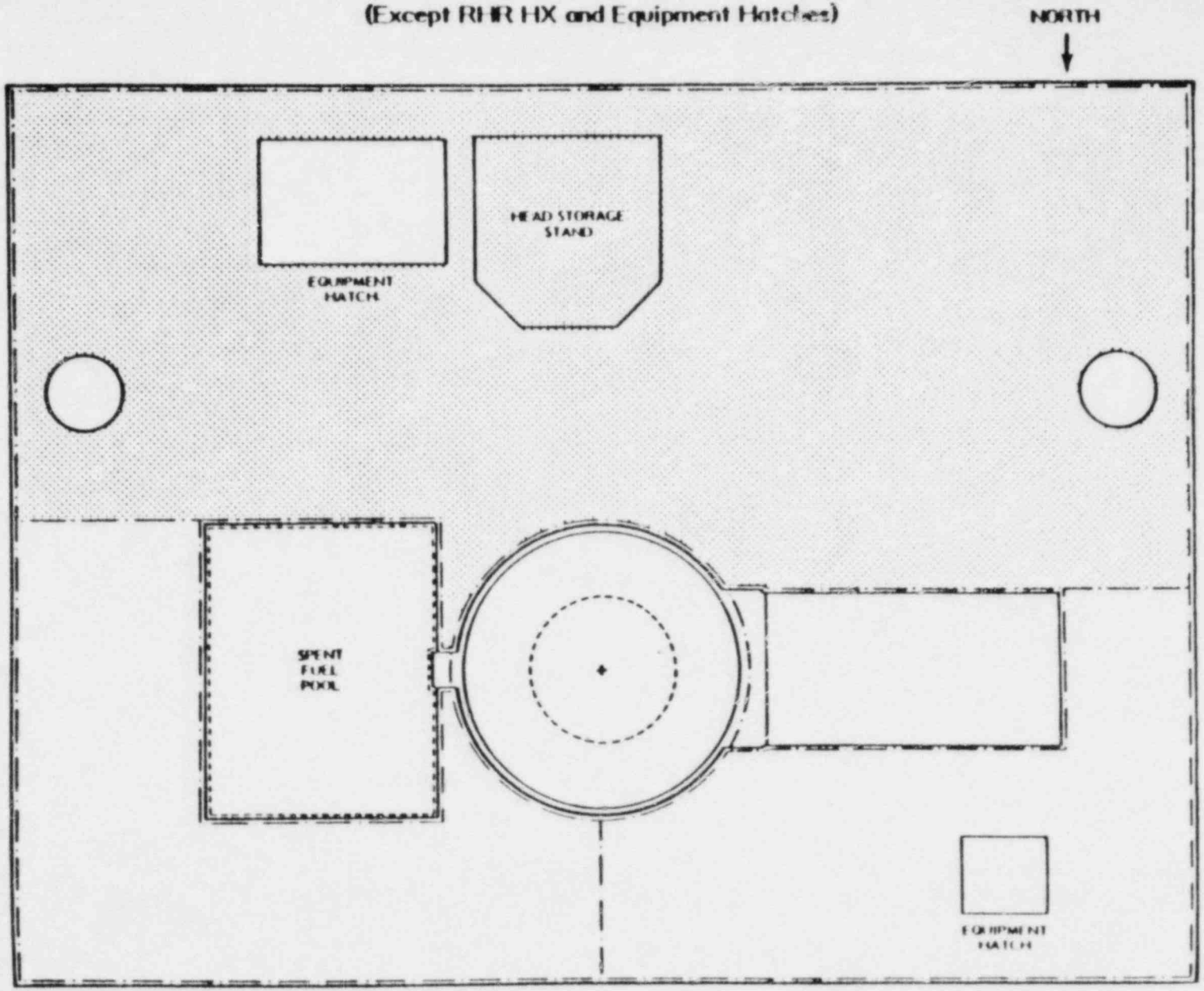


FIGURE 6

LOAD DROP EVALUATION

REGION 5 - NORTHEAST QUADRANT

(Except Spent Fuel Pool Area)

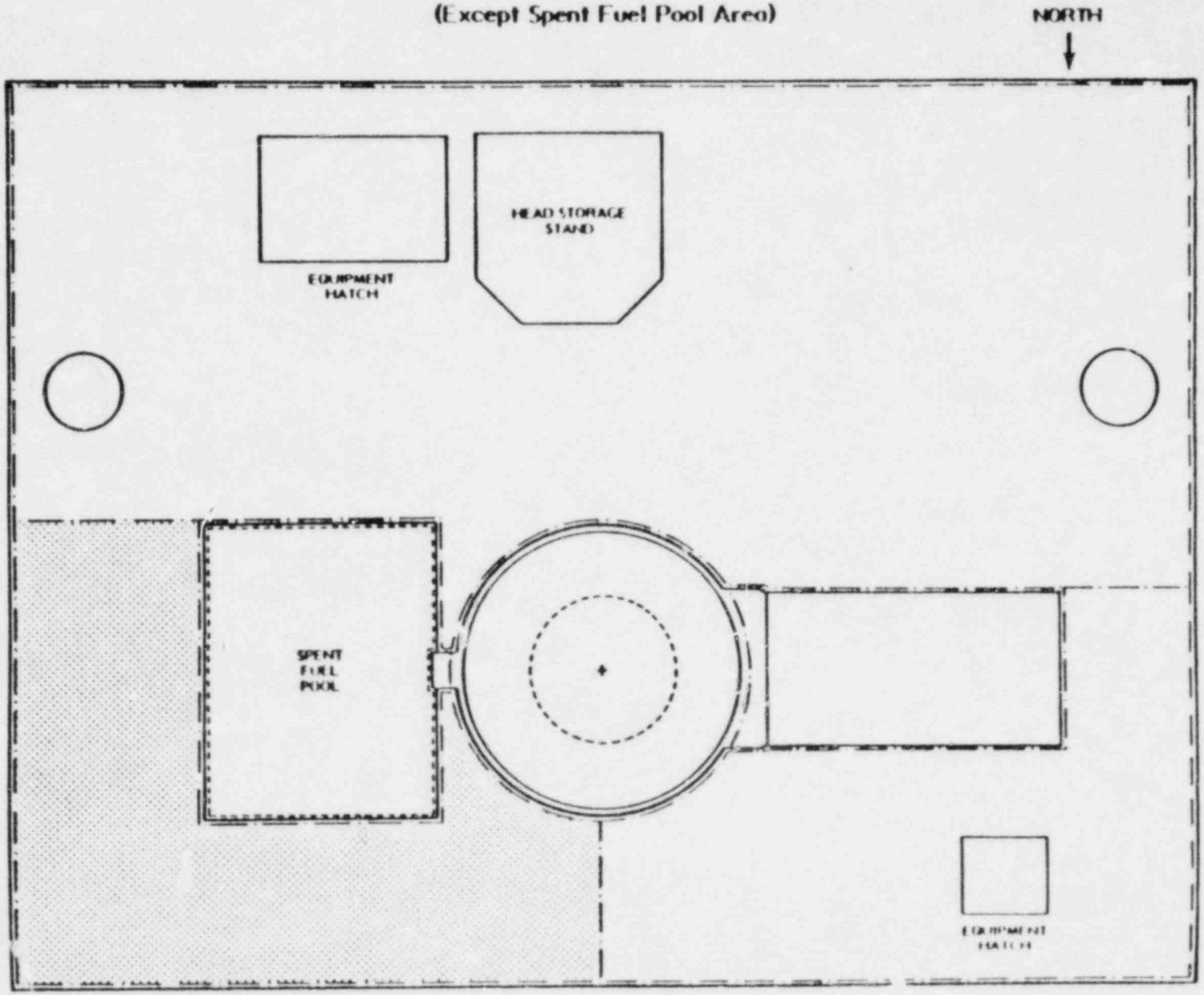


FIGURE 7

LOAD DROP EVALUATION

REGION 6 - NORTHWEST QUADRANT

(Except Equipment Hatch and Storage Pit Area)

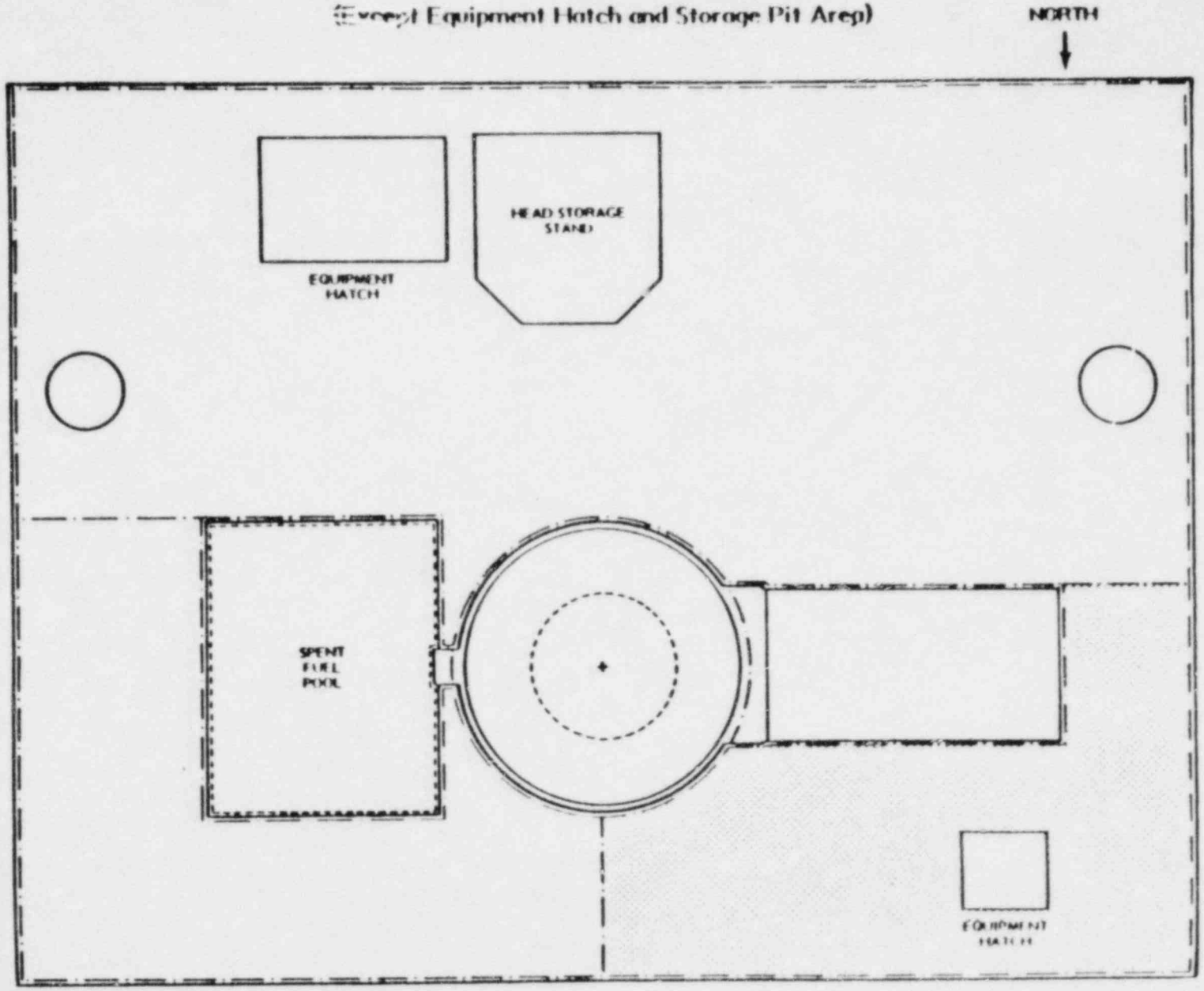


FIGURE 8

LOAD DROP EVALUATION

REGION 7 - SOUTHEAST EQUIPMENT HATCH

(Track Bay Floor at Elevation 272')

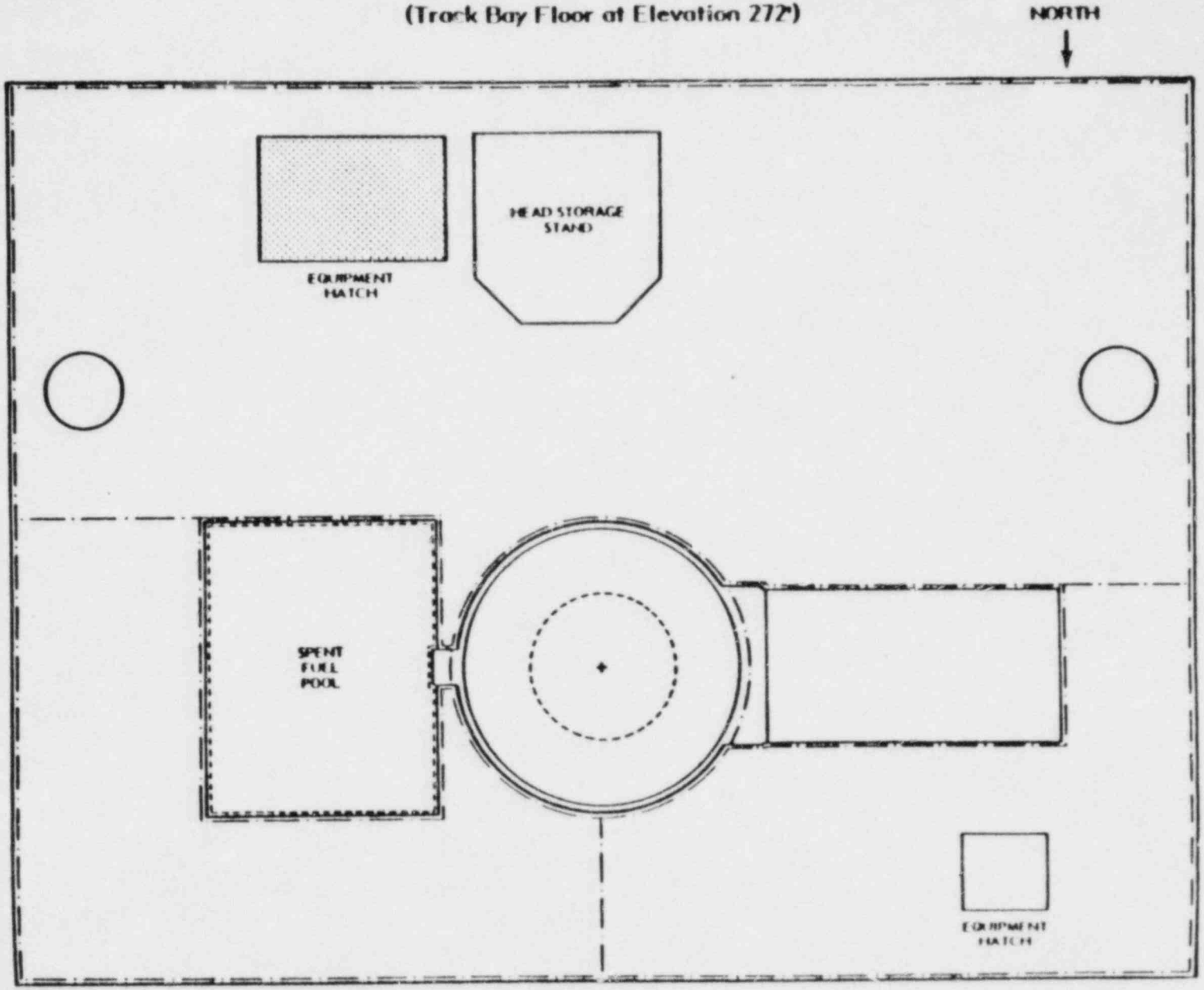


FIGURE 9

LOAD DROP EVALUATION

REGION 8 - NORTHWEST EQUIPMENT HATCH

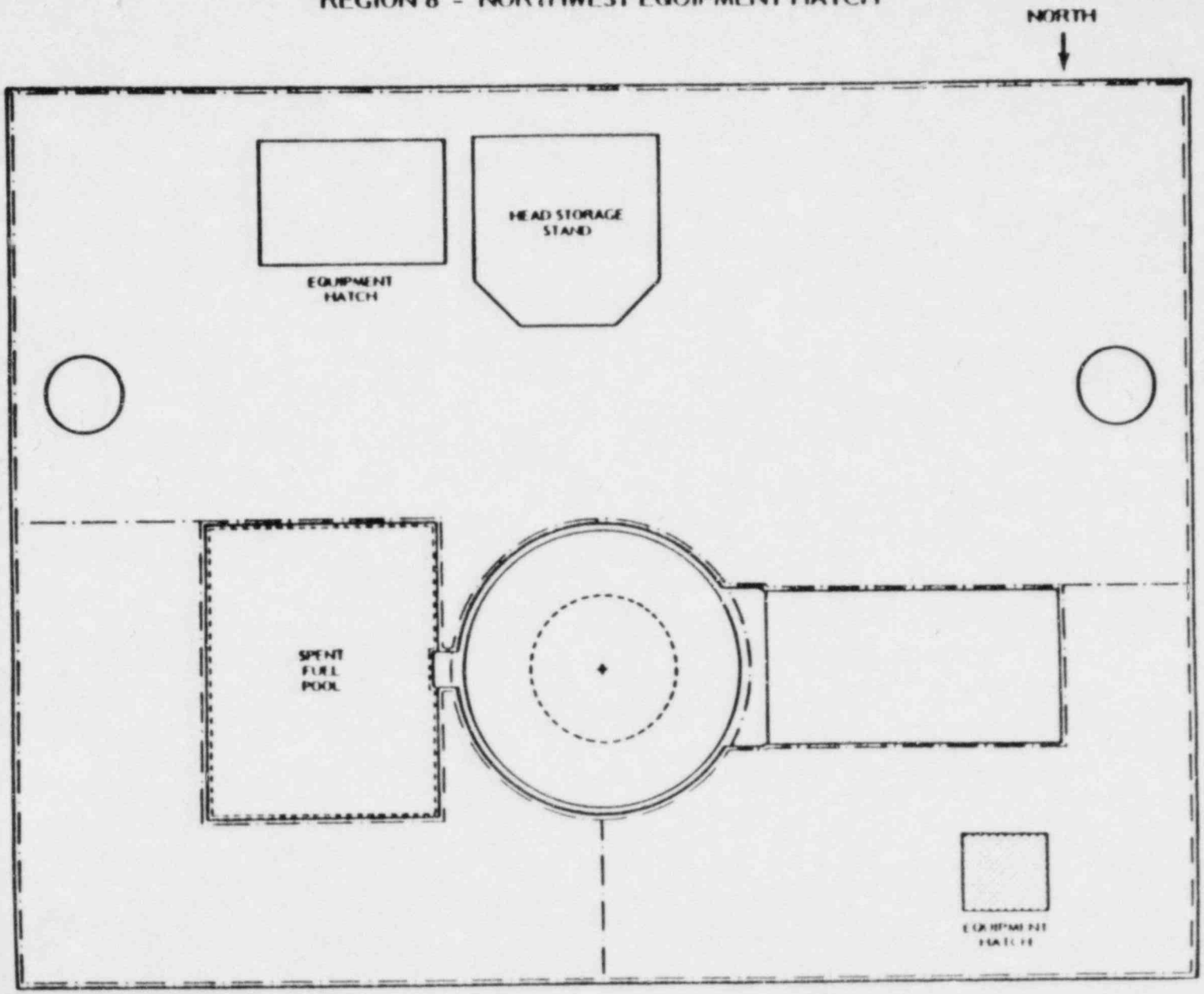
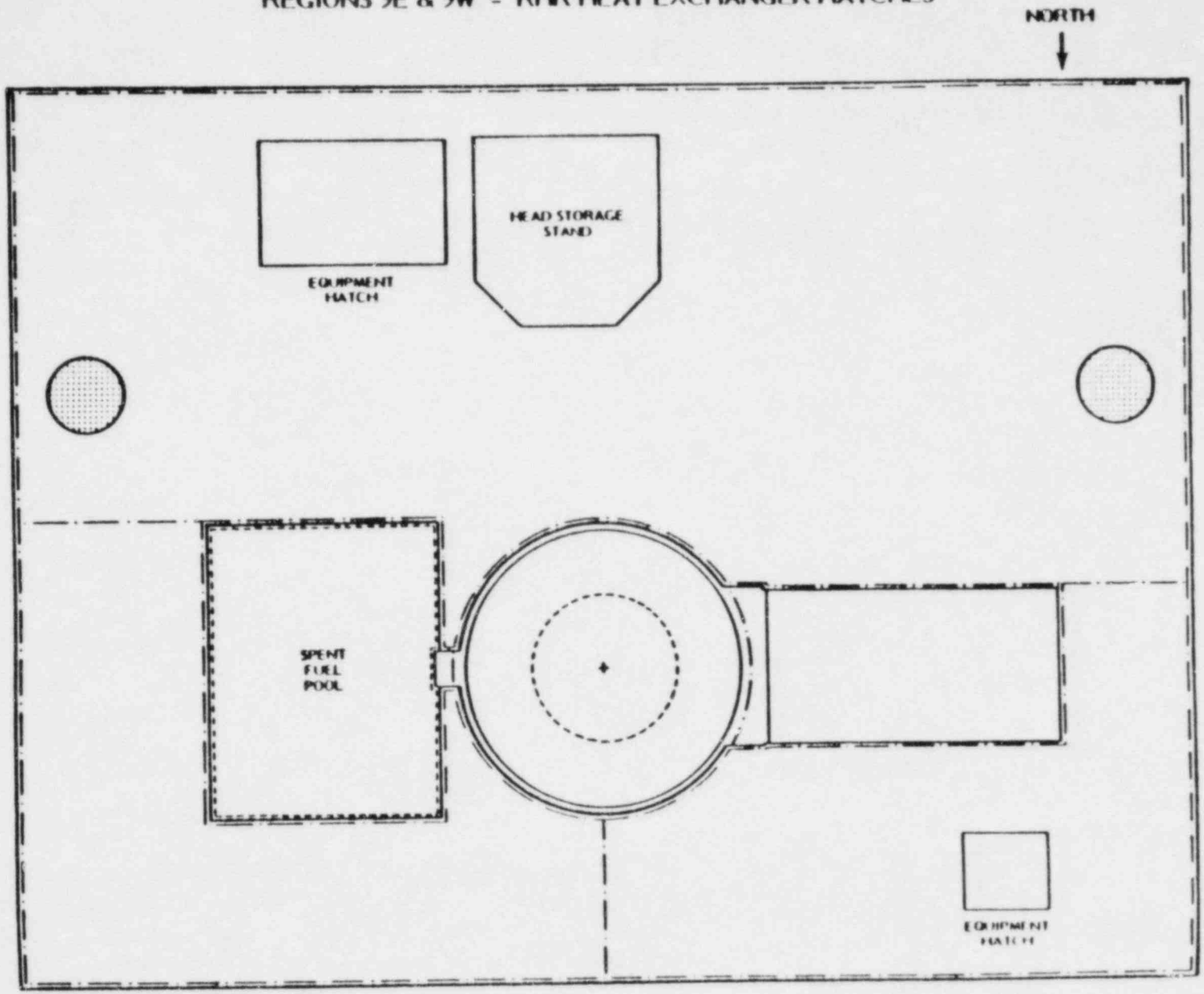


FIGURE 10

LOAD DROP EVALUATION

REGIONS 9E & 9W - RHR HEAT EXCHANGER HATCHES



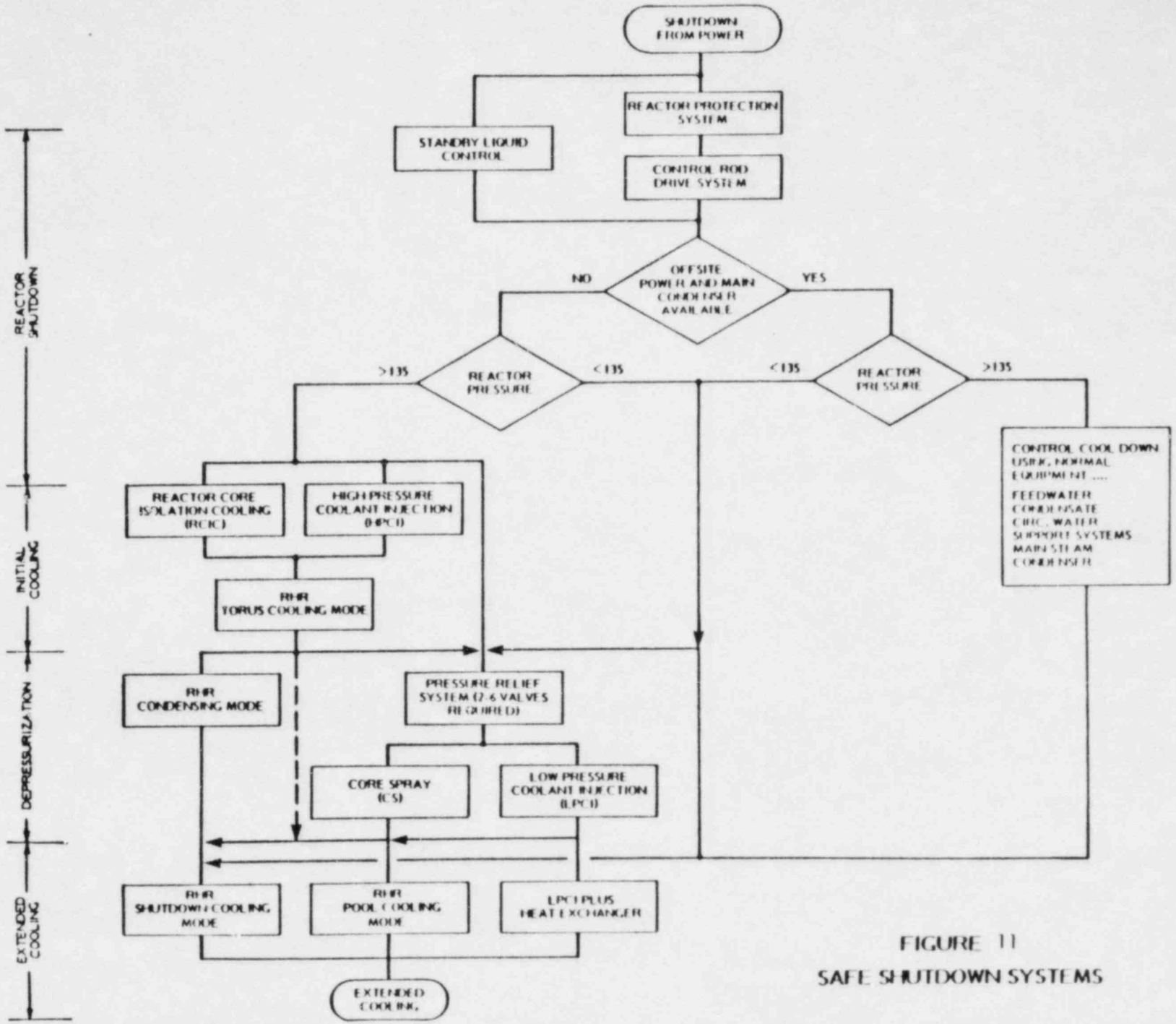
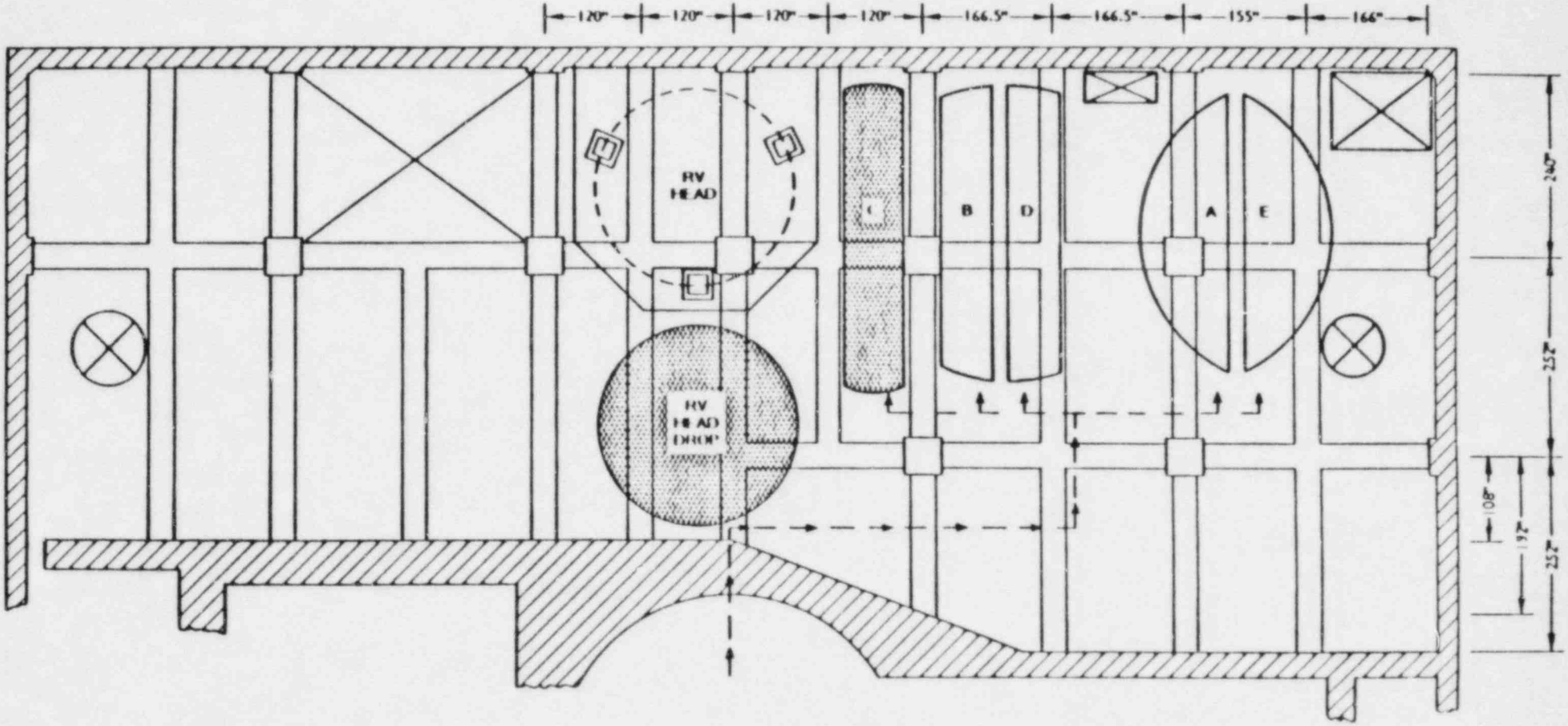


FIGURE 11

SAFE SHUTDOWN SYSTEMS

FIGURE 12

RV HEAD AND SHIELD PLUG
WORST CASE DROP ORIENTATIONS



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APPENDIX A

This appendix contains Table 3 from the Power Authority letter to NRC dated October 15, 1981 (JPN-81-82). Table 3 identifies Reactor Building Crane heavy loads.

In addition to those loads listed in Table 3, postulated drops of the Recirculation Pump motor were also evaluated.

TABLE 3

REACTOR BUILDING CRANE HEAVY LOADS[1]

	LOAD	SAFETY CLASS[2]	APPROX. WEIGHT (TONS)	APPLICABLE LIFT PROCEDURES	LIFTING EQUIPMENT	INTERLOCK MODE	HANDLING RESTRICTIONS
1.	Reactor Vessel Head & Strongback	1/3B	73	MP4.1&4.2 (7)	Head Strongback Turnbuckles & Shackles	Normal	Carry to minimum height necessary above vessel and refueling floor
2.	Drywell Head & Strongback	1/3B	48	MP4.1&4.2 (7)	Head Strongback Turnbuckles & Shackles	Normal	Carry to minimum height necessary above vessel and refueling floor
3.	Steam Dryer & Sling Assembly	1/3B	39	MP.4.1&4.2 (7)	Dryer/Separator Lifting Sling	Normal	Carry to minimum height necessary above vessel
4.	Shroud Head/Separator & Sling Assembly	1/3B	43.5	MP4.1&4.2 (7)	Dryer/Separator Lifting Sling	Normal	Carry to minimum height necessary above vessel
5.	Reactor Cavity Shield Plugs (5)& Shackles	3B	110 ea.	MP4.1&4.2 (7)	Slings, Turn-buckles & Shackles	Normal	Carry to minimum height necessary above refueling floor
6.	Internals Storage Area Shield Plugs (3)	3B	40-50 ea.	MP4.1&4.2 (7)	Slings & Shackles	Normal	Carry to minimum height necessary above refueling floor
7.	Refueling Slot Plugs (3)	3A	5.5 ea.	MP4.1&4.2 (7)	Slings & Shackles	Normal	Carry to minimum height necessary about refueling floor

TABLE 3

(continued)

	LOAD	SAFETY CLASS[2]	APPROX. WEIGHT (TONS)	APPLICABLE LIFT PROCEDURES	LIFTING EQUIPMENT	INTERLOCK MODE	HANDLING RESTRICTIONS
8.	Reactor Vessel Head Thermal Insulation	1/3A	10	MP4.1&4.2 (7)	Reactor Head Insulation Lifting Rig	Normal	Carry to minimum height necessary above vessel and refueling floor.
9.	Reactor Vessel Head Tensioners & Rig	1/3A	6	MP4.1&4.2 (7)	Reactor Head Stud Tensioner Rig	Normal	Carry to minimum height necessary above vessel and refueling floor.
10.	Spent Fuel Pool Gates (2)	2	1.3	MP4.1&4.2 (7)	Sling & Chain Fall	N/A	Lift with Chainfall.
11.	Portable Radiation Shield (Cattle Chute)	2/3A	14	MP4.1&4.2 (7)	Slings & Shackles	Normal	Do not carry over reactor vessel. Carry to minimum height necessary above refueling floor.
12.	Vessel Service Platform	1/3A	7	MP4.1&4.2 (7)	Service Platform Slings	Normal	Carry to minimum height necessary above vessel and refueling floor.
13.	Clean Up Filter Demineralizer Hatch Covers (2)	2/3A	4.35 ea	(7)	Slings & Shackles	Normal	Do not carry over reactor vessel. Carry to minimum height necessary above refueling floor.

TABLE 3

(continued)

	LOAD	SAFETY CLASS[2]	APPROX. WEIGHT (TONS)	APPLICABLE LIFT PROCEDURES	LIFTING EQUIPMENT	INTERLOCK MODE	HANDLING RESTRICTIONS
14.	Skimmer Surge Tank Tank Hatch Covers (2)	2/3A	3.7 ea	(7)	Slings & Shackles	Normal	Do not carry over reactor vessel. Carry to minimum height necessary above refueling floor.
15.	RHR Heat Exchanger Hatch Covers (2)	2/3A	4.15 ea	(7)	Slings & Shackles	Normal	Do not carry over reactor vessel. Carry to minimum height necessary above refueling floor.
16.	New Fuel Storage Vault Hatch Covers (3)	2/3A	3.75	(7)	Slings & Shackles	Normal	Do not carry over reactor vessel. Carry to minimum height necessary above refueling floor.
17.	Equipment Hatch NW quadrant) Hatch Covers (3)	2/3A	0.5 ea	(7)	Slings & Shackles	Normal	Do not carry over reactor vessel. Carry to minimum height necessary above refueling floor.
18.	Equipment Hatch SE quadrant) Hatch Covers(5)	2/3B[3]	1.3 ea	(7)	Slings & Shackles	Normal	Do not carry over reactor vessel. Carry to minimum height necessary above refueling floor.

TABLE 3

(continued)

	LOAD	SAFETY CLASS[2]	APPROX. WEIGHT (TONS)	APPLICABLE LIFT PROCEDURES	LIFTING EQUIPMENT	INTERLOCK MODE	HANDLING RESTRICTIONS
19.	Reactor Building Crane Load Block & Hook	2/3B[3]	3.1	(7)	N/A	Normal (when moving unloaded)	Do not carry over reactor vessel. Do not carry over equipment hatches except to make a lift through hatch. If over SE hatch, see footnote 3.
20.	Head Stud Rack	2/3A	1.5	MP4.1&4.2 (7)	Slings & Shackles	Normal	Do not carry over the reactor vessel. Carry to minimum height necessary above refueling floor.
21.	Shipping Cask CNS 4-45	2/3B[3]	34	(6) (7)	Lifting Yoke Supplied by Chem Chem Nuclear	Normal Cask Handling[4]	See Procedure for Lift[6]
22.	CNS 4-45 Cask Liner	2/3A/3B[3]	4	(7)	Sling provided with liner	Normal Cask Handling[4]	Carry to minimum height necessary above refueling floor. If over SE hatch, see footnote 3. Do not carry over reactor vessel.

TABLE 3

(continued)

	LOAD	SAFETY CLASS[2]	APPROX. WEIGHT (TONS)	OTHER APPLICABLE LIFT PROCEDURES	LIFTING EQUIPMENT	INTERLOCK MODE	HANDLING RESTRICTIONS
23.	Spent Fuel Shipping Cask (Non-selected as yet)	2/3B[3]	70-110	(6) (7)	Lifting Yoke	Normal Cask Handling[4]	See Procedure for for Lift[6]
24.	Fuel Channel Crate	2/3B[3]	1.2	(7)	Mesh Slings	Normal	Carry to minimum height necessary above refueling floor. If over SE hatch, see footnote 3. Do not carry over reactor vessel.
25.	New Fuel Container	2/3B[3]	1.9	(7)	Slings	Normal	Carry to minimum height necessary above refueling floor. If over SE hatch, see footnote 3. Do not carry over reactor vessel.
26.	RHR Heat Exchanger Shell	2/3A[5]	7.5	(7)	Slings & Shackles	Normal	Do not carry over reactor vessel. Carry to minimum height necessary above refueling floor.

TABL 1

(continued)

	LOAD	SAFETY CLASS[2]	APPROX. WEIGHT (TONS)	APPLICABLE LIFT PROCEDURES	LIFTING EQUIPMENT	INTERLOCK MODE	HANDLING RESTRICTIONS
27.	RHR Heat Exchanger	2/3A	20.5	(7)	Slings & Shackles	Normal	Do not carry over reactor vessel. Carry to minimum height necessary above refueling floor.
28.	Hydrolaser	2/3A	2	(7)	Slings & Shackles	Normal	Do not carry over reactor vessel. Carry to minimum height necessary above refueling floor.
29.	Recirculation Pump Motor	2/3A	20	(7)	Slings & Shackles	Normal	Do not carry over reactor vessel. Carry to minimum height necessary above refueling floor.

1 NUREG 0612 defines a heavy load as one that weighs more than the combined weight of a single spent fuel assembly and its associated handling tool. For reference, the weight of a fuel assembly, its associated handling tool, and channel at Fitzpatrick is approximately 750 lbs.

2 Safety Classes are defined in the response to Item 3.a.

3 These loads are classified as 3B because of their potential for damaging equipment below the track bay floor at the 272' elevation. These loads must be lifted over or up through the Reactor Building Equipment Hatch from the 272' elevation to the 369' elevation. The CNS 4-45 cask and Spent Fuel Shipping casks are also classified 3B because of their potential for damaging equipment below the refueling and spent fuel pool floors, if dropped.

4 Interlocks restrict movement of the cask to the cask loading area when the cask is over the spent fuel pool. See response to Item 3.a.

5 A RHR Heat Exchanger Shell or Tube Bundle, if pulled for maintenance or replacement, must be raised to the 369' elevation from the 272' elevation through the RHR HX Hatches.

6 Cask lifts will be governed by special lift procedures that will be prepared in advance of making the lifts.

7 All lifts will be addressed in a procedure governing load handling operations by the Reactor Building Crane.

the dryer drop energy was determined to be 1.2×10^6 ft-lbs) and the different impacted items, both the dryer drop and the steam separator drop were evaluated. In evaluating the consequences of the steam dryer drop, it was necessary to determine the sequence of "failure modes" for the drop. This was determined based on locations of first failure in the impact system. That is, the weakest links were assumed to fail first and the energy absorbed and dissipated in that failure was calculated and compared to the total energy of the drop. Similarly, the work done in other parts of the supporting structure in absorbing the impact of the dryer drop was calculated. In this manner, if the supporting structure could be shown to absorb and dissipate the energy of the dryer drop without damaging the fuel, the consequences of the drop were assumed to be acceptable.

The analysis for the dryer drop indicates that in actuality the weakest link in this drop is the crushing of the dryer assemblies themselves. This crushing of the dryer dissipates the total energy of the dryer drop and thus no other damage to the supporting structure is predicted. In addition, the loads of this drop onto the supporting structure were evaluated and found to be acceptable. Although crushing of the dryer is predicted, in order to be consistent with the guidelines of NUREG-0612, an evaluation was made to determine whether the supporting system could withstand the energies of the dryer drop assuming that no crushing of the dryer took place. Again in this case, the supporting system, including the shroud and shroud support, were found to acceptably withstand the energy of the dryer drop.

In addition to evaluation of drop energies, a dynamic impact factor was calculated for both the dryer drop and steam separator drop. The methodology for calculating this impact factor was similar to that described above for the reactor vessel head drop. A dynamic impact factor was calculated for the dryer drop, and the resulting total dynamic load on impact was determined. When the dryer drops, it lands on the four dryer support lugs. The calculated stress in these lugs exceeds the allowable shear stress for the lug material. Therefore, the four dryer support lugs are assumed to fail. Failure of these four support lugs results in the dryer dropping onto the steam separator. As was mentioned above, the energies of this drop were evaluated and found to be acceptable. In

addition, the stresses through the load path of the supporting system (i.e., steam separator, shroud, shroud support, reactor vessel, and skirt) were evaluated and found to be acceptable.

Based on the results of the reactor vessel head drop and steam dryer drop scenarios analyzed to date, both from the standpoint of stresses and energies, reactor vessel integrity is predicted and no fuel damage is expected. However, additional RPV head drop scenarios will be evaluated to provide additional assurance that vessel integrity can be maintained.

An analysis for a postulated drop of the steam separator assembly was also performed, in a manner similar to that for the steam dryer drop. However, whereas the consequences of the steam dryer drop were determined to comply with the criteria of NUREG-0612, the results of our analyses performed to date indicate that the consequences of the postulated steam separator drop could have a potential for causing fuel damage. Since the use of additional safety margins is desirable, we will undertake additional investigations to attempt to demonstrate that the likelihood of a steam separator drop, as analyzed, is sufficiently small or to suitably constrain the parameters of the drop to allow acceptable consequences to be demonstrated.

In addition, to assure that fuel integrity is maintained for other types of drops, investigations of the consequences of a postulated drop of the portable radiation shield or the refueling slot plugs into the reactor vessel were undertaken. In both cases, the consequences were determined to be conservatively bounded by the results of the steam dryer drop analysis. Therefore, reactor vessel integrity was predicted and no fuel damage is expected.

2.3 SPECIFIC REQUIREMENTS FOR OVERHEAD HANDLING SYSTEMS OPERATING IN PLANT AREAS CONTAINING EQUIPMENT REQUIRED FOR REACTOR SHUTDOWN, 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 drop which might prevent safe reactor shutdown or prohibit continue decay heat removal is extremely small, or that damage to such equipment from loads will be limited in order not to result in the loss of these safety-related functions. Cranes which must be evaluated in this section have been previously identified in your response to 2.1-1, and their loads in your response to 2.1-3-c.

ITEM 2.3-1 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 Reactor Building crane was evaluated to industry standards CMAA 70-1975 (reference 5) and ANSI B30.2-1976 (reference 6). It was found to meet those standards, with two exceptions which were justified in reference 3.

Notwithstanding the fact that the lifting system including the Reactor Building crane, lifting slings and strongbacks, complies with the intent of applicable industry standards and possesses demonstrated margins to failure, rather than relying on the reliability of the lifting system, an evaluation has been performed to assess the consequences of postulated drops of heavy loads. Therefore, although these heavy load drops need not be postulated, even if they were to occur, their consequences have been evaluated.

The only case where load handling reliability was considered was with respect to the main hoist load block and hook. NUREG-0612 (reference 34) 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, shield plugs, and the dryer and separator units. 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 the crane load block alone.

The only 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 Fitzpatrick Reactor Building crane is provided with two diverse upper limit switches to interrupt power to the hoist motor prior to "two blocking." When power is removed, holding brakes are automatically applied. One of the two limit switches is a geared limit switch driven off the drum shaft. The other is a counter weight switch that is released when the load block comes up against a trip bar; the trip bar will stop power to the hoist below the low point of the sheave assembly.

The holding brakes are solenoid released, and spring applied on loss of power to the solenoid. Two holding brakes are provided, either of which has sufficient capacity to hold the rated load (each brake is 150% of full motor torque). Additionally, inspection and maintenance procedures assure that the limit switches and holding brakes are functional and properly adjusted.

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

Nonetheless, an analysis of a load block and hook drop from the highest possible carry height onto the operating/refueling floor was performed to verify the capability of the floor to withstand the impact of such a drop. The results of the analysis indicate that while concrete scabbing on the underside of the floor is predicted, no gross failure or penetration will result. The consequences of scabbing have been considered in the systems evaluations and were found to be acceptable.

Therefore, although drop of the load block and hook need not be postulated, even if they were to drop on the operating/refueling floor, the consequences are acceptable.

ITEM 2.3-2 For any cranes identified in 2.1-1 not designated as single-failure-proof in 2.3-1, 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 methods 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 consideration 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 loads 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 validity of such considerations.

c. For interactions not eliminated by the analysis of 2.3-2-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 so evaluated, provide the load-handling-system (i.e., crane-load-combination) information specified in Attachment 1.

d. For interactions not eliminated in 2.3-2-b or 2.-3-2-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: The reactor building crane is normally used for maintenance operations which include moving of items above the operating/refueling floor at 369'-6" elevation, and movement of equipment from the track floor at 272' elevation up the SE equipment hatch to the operating/refueling floor. In addition, the reactor building crane can be used in a "cask handling mode" as discussed in our initial submittal responding to the NRC letter of December 22, 1980 (letter from George T. Berry to Darrell G. Eisenhut dated October 15, 1981).

Accordingly, evaluations of heavy load handling operations at Fitzpatrick require considerations of drops onto the operating/refueling floor at 369'-6" elevation, drops into the reactor vessel and spent fuel pool, drops into the internals storage pit, and drops onto the 272' elevation floor at the equipment hatches. Evaluations for drops into the reactor vessel and spent fuel pool were discussed previously in response to item 2.2.4.

The evaluation of heavy load handling operations at Fitzpatrick was performed by reviewing those heavy loads which could be carried over each region of the

reactor building. The reactor building was subdivided into nine regions of interest, covering the areas where heavy loads could be dropped (Figures 2-10). Accordingly, our responses to the above requests for information are provided below on a region by region basis.

A combination of systems evaluations and structural analyses were utilized to verify that damage following a postulated load drop would not preclude operation of sufficient equipment necessary to perform safe shutdown (NUREG-0612, Section 5.1, Criterion IV). A drop of each heavy load carried by the reactor building crane was postulated to occur onto the operating/refueling floor at elevation 369'-6" or equipment hatches, as appropriate. Worst case drop scenarios were evaluated, so that the consequences of a postulated drop of any of the heavy loads handled (see Appendix A) are bounded by the results presented herein.

Systems Evaluation Methodology - Safe Shutdown Evaluation

As part of the evaluation of heavy load handling operations at Fitzpatrick, a number of potential load drop regions in the reactor building were addressed by performing systems evaluations. The objective of the systems evaluations was to demonstrate that safe shutdown and long term cooling could be achieved and maintained assuming that certain combinations of equipment were lost due to a possible load drop. The results of the systems evaluations are summarized below in the discussions for each region.

In order to demonstrate the ability to safely shutdown and cool the core, it was necessary to (1) identify the safety functions required to achieve safe shutdown, (2) identify the plant systems required to accomplish these functions, (3) identify the equipment that could potentially be lost if a load drop were to occur in certain plant areas (designated as Regions), and (4) determine the resultant effects of the loss of this equipment on the safety functions required to achieve safe shutdown.

Plant Conditions

To determine the functions that must be accomplished to achieve and maintain safe shutdown, it was assumed for most regions investigated that the reactor was at 100% power at the time the load drop was postulated. For Regions 9E and 9W, however, it was assumed that movement of the load of interest in these regions would only be performed with the reactor in the shutdown and cold condition. Accordingly, the plant conditions associated with the cold shutdown and/or Refueling Condition[1/] were assumed as initial conditions for evaluating drop consequences in these regions. The functions required to be accomplished and systems included in the evaluation to accomplish these functions are desribed below.

Safe Shutdown Functions and Systems

In order to accomplish safe shutdown from 100% power, the following functions must be performed:

Reactor Scram or Shutdown
Monitoring of Critical Plant Parameters
Core Cooling (Initial)
Depressurization/Makeup
Extended Cooling

All of the above functions (except Scram and Extended Cooling) can be accomplished as part of normal plant cooldown with non-safety systems such as the feedwater, condensate, and circulating water systems. Nonetheless, no credit was taken for non-safety systems in performing the systems evaluations, i.e., the ability to accomplish safe shutdown and core cooling was evaluated assuming the use of safety systems only, except for those cases where the Refueling Condition was relied upon.

1/ The Refueling Condition referred to here is defined for purposes of the systems evaluations to be reactor head removed, reactor cavity (and possibly the Storage Pit) filled and Spent Fuel Pool Gate open.

The functions and specific systems that could be relied on to achieve and maintain safe shutdown are indicated in Figure II.

If the plant proceeds to the Refueling Condition,[1] then several cooling modes are possible. Examples are: (1) RHR Shutdown Cooling, (2) Spent Fuel Pool Cooling and Cleanup System Cooling and (3) RHR Fuel Pool Cooling. Certain of these cooling modes rely on non-safety equipment. However, they are considered since in the Refueling Condition significant time is available to establish alternate cooling. In addition, if any or all of the cooling systems referred to above were lost, the core and spent fuel in the storage pool would continue to be cooled by the body of water in the pool and the reactor cavity. All that is necessary is to provide a source of makeup water to replenish any loss of inventory. Makeup could be provided by hoses or by other arrangements from any of a number of different available water sources, if required. Accordingly, if a load is only handled during this plant condition, there is no single load drop scenario that could result in inability to cool the core.

Steps in the Systems Approach

The following summarizes the steps that were performed in the systems evaluations for each function/system required for safe shutdown:

1) Identify the system (including any support systems) components of interest.

2) For each potential load impact region evaluate potential for damage/loss of system components. If equipment could be impacted, assume it is lost.

3) Compare system equipment required (Item 1), with equipment lost (Item 2), and determine if the 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 function could be lost.

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 safe shutdown conclusion regarding a particular region is the composite for that region of the conclusions for all the systems required to accomplish the safe shutdown functions.

Structural Evaluation Methodology

Each of the heavy loads carried by the reactor building crane have been 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. The results of this evaluation are tabulated in Table 2. In each region where local response was evaluated the load drops were analyzed to verify that slab perforation (i.e. penetration entirely through the floor slab) did not occur. Scabbing of the concrete deck backface was evaluated for all loads. In cases where postulated drops were predicted to produce this effect, equipment and systems below that area which could be impacted by the scabbing were assumed to be lost for the purpose of performing the systems evaluations.

Where the controlling modes of the heavy load drop response were determined to be "overall structure" response modes, these load drops were evaluated to verify that gross and intolerable distortions of the primary structural members did not occur. By verifying that gross and possibly propagating failures would not occur for these load drops, the consequences of the load drops could be shown to be limited to scabbing.

TABLE 2

SUMMARY OF CONTROLLING STRUCTURAL BEHAVIOR RESULTING FROM POSTULATED REACTOR BUILDING CRANE HEAVY LOAD DROPS

	LOAD	APPROX. WEIGHT (TONS)	CONTROLLING MODE OF RESPONSE: OVERALL STRUCTURAL	CONTROLLING MODE OF RESPONSE: LOCAL
1.	Reactor Vessel Head & Strongback	73	X	
2.	Drywell Head & Strongback	48	X	
3.	Steam Dryer & Sling Assembly	39	X	
4.	Shroud Head/Separator & Sling Assembly	43.5	X	
5.	Reactor Cavity Shield Plugs (5) & Shackles	110 ea.	X	
6.	Internals Storage Area Shield Plugs (3)	40-50 ea.	X	
7.	Refueling Slot Plugs (3)	5.5 ea.		X
8.	Reactor Vessel Head Thermal Insulation	10	X	
9.	Reactor Vessel Head Tensioners & Rig	6		X
10.	Spent Fuel Pool Gates (2)	1.3		X
11.	Portable Radiation Shield (Cattle Chute)	14	X	X
12.	Vessel Service Platform	7		X

TABLE 2

(continued)

	LOAD	APPROX. WEIGHT (TONS)	CONTROLLING MODE OF RESPONSE: OVERALL STRUCTURAL	CONTROLLING MODE OF RESPONSE: LOCAL
13.	Clean Up Filter Demineralizer Hatch Covers (2)	6.35 ea		X
14.	Skimmer Surge Tank Tank Hatch Covers (2)	3.7 ea		X
15.	RHR Heat Exchanger Hatch Covers (2)	4.15 ea		X
16.	New Fuel Storage Vault Hatch Covers (3)	3.75		X
17.	Equipment Hatch NW quadrant) Hatch Covers (3)	0.5 ea		X
18.	Equipment Hatch SE quadrant) Hatch Covers (5)	1.3 ea		X
19.	Reactor Building Crane Load Block & Hook	3.1		X
20.	Head Stud Rack	1.5		X
21.	Shipping Cask CNS 4-45	34	X	X
22.	CNS 4-45 Cask Liner	4		X
23.	Spent Fuel Shipping Cask (Non-selected as yet)	70-110	X	X
24.	Fuel Channel Crate	1.2		X
25.	New Fuel Container	0.5		X

TABLE 2

(continued)

	LOAD	APPROX. WEIGHT (TONS)	CONTROLLING MODE OF RESPONSE: OVERALL STRUCTURAL	CONTROLLING MODE OF RESPONSE: LOCAL
26.	RHR Heat Exchanger Shell	7.5		X
27.	RHR Heat Exchanger	20.5	X	X
28.	Hydrolaser	2		X
29.	Recirculation Pump Motor	20	X	X

In the analyses of possible slab perforation, procedures recommended in references 9 and 10 were followed. The modified National Defense Research Committee (NDRC) formula (reference 35) was chosen because it has been shown to give the best fit with available experimental data (references 36 and 37). The NDRC formula predicts the possible depth of penetration of a solid missile. In order to determine the thickness of the reinforced concrete needed to resist impact without perforation or scabbing the Army Corps of Engineers formula was used (reference 38) in conjunction with the NDRC penetration formula. In addition, a 10% margin on concrete floor thickness was conservatively applied as recommended in reference 9.

Although limited penetration and scabbing were predicted for the set of bounding heavy load drops considered, in no case was the elevation 369'-6" slab predicted to be perforated for normal carry heights.

For those drops which were determined to be controlled by overall structural response, the methodology used to evaluate the consequences of the heavy load drops was one of characterizing 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 the load drop. It was assumed in those calculations, 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.

A four-step iterative step wise linear static analysis was performed using the STRUDL computer code (reference 7) to determine force-deflection for important points in the structural model. The computational 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. In this case, a plain grid

model was developed which allows loads and deflections normal to the plane of the grid, and rotations above the axes lying in the plane.

The basic steps in the STRUDL computation procedure were as follows:

1) The stiffness matrix is determined for each member and finite element. Members are considered as cantilivers.

2) If any member releases are specified, the local member stiffness matrix (step 1) is modified.

3) The applied member and element loads, if any, are processed.

4) When free joints or released support joints exist, the structural stiffness matrix is assembled in a global coordinate system.

5) The load vector is assembled and the global stiffness matrix in the load vector are modified to account for joint releases.

6) When there are free joints or release support joints, governing joint equilibrium equations are solved for the joint displacement.

7) The induced member distortions, member end forces, element strains and stresses are computed by back-substitution.

The model is successively loaded until the moment capacity of any section is exceeded. This moment capacity is defined by Chapter 10 of ACI 318-77 (reference 8). At this point the model is reconstructed, incorporating appropriate rotational member releases. Greater multiples of the drop load are then applied to the new model until moment capacities are reached at other sections. This procedure continues until the ultimate load of the slab/grid system is reached.

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 the analysis for Fitzpatrick heavy load drops. The hinge rotational capacity was used as a criterion to set a maximum allowable level of deflection for the slab/grid

system. The hinge rotational capacity for concrete structures was developed in references 11 and 12 based on test results given in references 13 and 14.

Rotations of the magnitude suggested in the above references and used for these analyses result in cracking which is confined to a region below (above) the tensile reinforcement. Generally speaking the section will remain intact with no crushing, spalling or scabbing, due to flexure; however, scabbing may occur as a result of shock wave motion associated with the reflection of tensile waves from the rear surface or shear plug formation. Therefore, it has been conservatively assumed that scabbing does occur due to these drops.

The load/deflection history up to the point of the maximum load coupled with the maximum allowable deflection, defines the maximum level of strain energy adsorption, provided that a shear failure has not occurred. At each load increment in the analysis (specified in terms of load amplification factors) the shear stress at limiting sections is checked and compared to allowables as specified in Chapter 11 of ACI 318-77 (reference 8). The load amplification factors are utilized as a convenient method of determining the slab/grid resistance and should be differentiated from dynamic load factors.

The moment diagram at the ultimate load and the slab/grid deflections as the ultimate load is reached, are each compared against allowables. In the case of the deflection, the allowable deflection is limited by the rotational capacity of the member. Integrating the slab/grid force-deflections under the loaded region, and performing an energy balance, the allowable drop heights for each heavy load considered are determined.

In addition to the conservatisms previously mentioned, the following conservatisms are also inherent in the methodologies used in the structural evaluations:

1) Static material strengths for concrete and steel were used, although test data shows that this property increases with the increased strain rates associated with dynamic loadings. For example, references 15 and 16 recommend dynamic increase factors of 1.25 for 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 17) of increased strength. Also, the average strength for structural steel is nearly a factor of 1.25 (reference 18) highered in 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) The criteria for hinge rotational capacity that was used corresponds to support rotations of the order of 2 degrees, with minimum cracking and no crushing or scabbing. To meet necessary performance requirements (i.e. halting propagating failure), larger rotations in the range of 5 to 12 degrees could be tolerated. Experimental observations (reference 19) suggest even further capability for well designed and well anchored slabs. Use of these larger rotational capabilities would have resulted in greater energy 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 20 through 28). It is expected that the shear capabilities of the beams at elevation 369' 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) 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 drop. In reality a more favorable load distribution would exist due to the load distribution capability of the slab.

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

Regions I and 2

Region I (Reactor Vessel) and Region 2 (Spent Fuel Pool) evaluations were discussed previously in response to item 2.2.4.

Region 3

Region 3 (Internals Storage Pit) was evaluated for the bounding load of the drop of the steam dryer. A structural evaluation was performed to determine whether overall structural failure, or local damage, could impact equipment necessary to accomplish and maintain safe shutdown.

The results of the analysis indicated that overall structural integrity of the storage pit would be maintained for a postulated drop of the steam dryer. Furthermore, it was concluded that a gross breach of the leak tight integrity of the pit would not occur, and that if in the unlikely event that leakage should occur, it would be extremely minor and would be limited to insignificant dripping through small cracks.

Scabbing of the concrete under the pit was assumed to occur. Therefore, damage to equipment in this region (at elevation 326' below the storage pit) was evaluated on the basis of potential scabbing of concrete from the underside of the storage pit floor. The evaluation determined that no safe shutdown equipment whose failure could result in an inability to accomplish and maintain safe shutdown could be impacted in this region.

Therefore, based on the structural and systems evaluations, the consequences of heavy load drops into Region 3 were determined to be acceptable.

The steam dryer assembly is mounted in the reactor vessel above the steam separator assembly. The dryer is cylindrically shaped, weighs 39 tons, and is 316 inches long and 214 inches in diameter. The separator is also cylindrically shaped, weighs 43.5 tons, is 201 inches long and 200 inches in diameter. Each unit must be lifted less than 6 feet above the storage pit floor when moving it into and out of the storage pit. The drop of the dryer was determined to be more controlling, because it free falls through air, versus the separator which falls through water.

Although each unit is highly crushable, no credit was given to energy absorbtion due to crushing. Instead, the units were conservatively assumed to be infinitely rigid.

The storage pit slab is 20 feet by 40 feet in dimension, and 5 feet thick. A yield line analysis assuming a uniformly loaded circular fan was conducted to determine the ultimate resistance of the slab. The deflection of the slab was calculated on the basis of slab rotational capacity. The absorbed energy was then obtained by integrating the deflection under the load. The maximum allowable carry height was therefore conservatively determined to be more than twice the normal carry height of the units.

Region 4

Region 4 includes the entire south half area of the 369'-6" elevation floor. Loads identified in Appendix A were evaluated for postulated drops over this area. On the basis of load weights, dimensions and drop heights; bounding loads were evaluated for both the overall structural mode and the local response mode of behavior previously discussed. For the overall structural mode, the bounding load drops were determined to be those for the reactor cavity shield plugs, and topple-over of the fuel cask. The five reactor cavity shield plugs (A, B, C, D, E) shown in figure 12 each weigh approximately 110 tons. The plugs are 6'1" thick and range between 33 to 38 feet in length, and 7 to 10 feet in width. The plugs are carried from the reactor cavity approximatley 6 inches above the refueling floor to the laydown area shown in figure 12.

The shield plug lifting slings were previously evaluated (reference 3) and found to meet industry standard ANSI B30.9-1971 (reference 4). In addition the reactor building crane was evaluated to industry standards CMAA 70-1975 (reference 5) and ANSI B30.2-1976 (reference 6), and found to meet those standards with two exceptions. Justification for those exceptions was provided in reference 3.

Notwithstanding the fact that the lifting system, including the crane and slings, complies with the intent of the applicable industry standards and possesses demonstrated margins to failure, an evaluation was performed for a postulated drop of any of the shield plugs onto the reactor building refueling floor at elevation 369'-6". The worst case drop was considered to be that of Plug C after the other four plugs had already been moved to their laydown area positions.

The concrete floor at elevation 369'-6" is generally 15 inches thick except at selected areas where it is 24 inches thick and at the reactor head laydown area where it is 30 inches thick. The floor slab is supported by a grid system of continuous concrete beams which vary in depth from 40 to 60 inches and in width from 26 to 30 inches, with reinforcement ratios of 0.12 to 1.6 percent. The slab/grid is supported around its periphery by shear walls and at intermediate points by concrete columns.

The overall structural response methodology discussed previously was used to verify that the concrete floor possessed sufficient internal strain energy capabilities to withstand the drop of the shield plug. It was assumed that momentum is conserved and 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.

The results of the bounding drop evaluation determined that the slab/grid system possessed sufficient energy absorbing capacities to withstand the drop of the cavity shield plug from heights significantly exceeding the normal carry heights.

In addition, the effects of local structural response from smaller and perhaps lighter drops were also evaluated in this region. Loads such as the portable radiation shield, refueling slot plugs, various hatch covers, and fuel channel crates were evaluated to assess the acceptability of postulated drops as limited by the concrete deck capability to resist perforation. The analysis methodology was as described previously for local response evaluations. For all such evaluations, in no case was the elevation 369'-6" slab predicted to be perforated for these loads.

Although spent fuel casks are not currently handled at Fitzpatrick, a drop of a cask was postulated to occur on the refueling floor at Region 4. Since fuel assembly and low-level radioactive material shipping casks have not been selected to date, the 34 ton (maximum loaded weight) Chem-Nuclear Systems, Inc. C.N.S. 4-45 cask and an as yet unspecified 110 ton fuel assembly cask were evaluated. The 110 ton cask was assumed to measure 5 feet in diameter and 18

feet in length. The C.N.S. 4-45 cask is 173 1/8 inches long and 42 1/2 inches in diameter, except for 31 5/8 inches at each end which is 40 1/2 inches in diameter. Each end of the cask has a cover and an impact limiter. However, for conservatism, the energy absorbing effects of the impact limiter was not accounted for.

The cask was assumed to be moved over the 369'-6" elevation deck to and from the equipment hatch, spent fuel, and cask washdown (head storage) area. The slab is 24 inches thick along the travel path between the spent fuel pool and the equipment hatch. Three concrete beams span the approximately 30 feet between the hatch and the spent fuel pool, and are the principal load carrying members.

Both overall and local structural response modes were evaluated for a postulated drop from the normal carry height of 6 inches followed by c topple-over onto the deck. The structural methodologies were as described above. Perforation of the slab is not predicted for either cask. Also, the postulated vertical drop does not cause unacceptable hinge rotation or shear failure, as defined by the criteria described earlier. However, for topple-over of the casks, a preliminary conservative analysis predicts the rotational capacity criteria to be exceeded. Therefore, overall structural failure can not be precluded.

While these initial evaluations have indicated that postulated cask drops onto this region of the refueling floor could result in overall structural failures, several possible options for preventing this result have been identified and are being evaluated. The results of these evaluations will be submitted to the NRC when they have been completed. Accordingly, no cask handling in this area will be performed until an acceptable solution has been implemented.

Since all of the structural evaluations, except those for cask topple-over, indicated that damage can be limited to concrete scabbing onto the elevation immediately below the refueling floor, systems evaluations were performed to determine the effects on safe shutdown of damage to equipment at elevation 344' in this region. Loss of the only safe shutdown equipment that could be impacted in this region could result in the inability to utilize one of the two redundant Low Pressure Coolant Injection (LPCI) loops to provide makeup during

reactor depressurization/cooling. The other loop of LPCI would be unaffected. In addition other systems, such as Low Pressure Core Spray could be utilized to accomplish this function. As indicated above, large casks will not be handled over this area of the refueling floor until an acceptable solution has been implemented.

Therefore, the consequences of postulated load drops in Region 4 were evaluated and, except for the case of cask topple-over, were found to acceptably comply with NRC evaluation criteria.

Region 5

Region 5 is the northeast quadrant of the refueling floor at elevation 369'-6". Structural evaluations of this region were determined to be bounded by those evaluations performed for Region 4, since the floor slab at this region is 24 inches thick versus 15 inches thick as evaluated for Region 4, and the heavy load carried in Region 5 are similar and bounded by those evaluated for Region 4.

Therefore, structural damage in Region 5 is limited to scabbing of concrete under the floor slab. The effects of damage to equipment at elevation 344' below this region was evaluated on the basis of potential for this scabbing. No safe shutdown equipment whose failure could result in an inability to accomplish and maintain safe shutdown could be impacted in this region. Therefore, the consequences of load drops in region 5 were determined to acceptably comply with NRC evaluation criteria.

Region 6

Region 6 includes the northwest quadrant of the refueling floor, except for the equipment hatch and the internal storage pit areas. The heavy loads carried in this area were determined to be similar to those carried in Region 4 and therefore the structural evaluations previously described for Region 4 apply to Region 6 also.