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LICENSING REPORT

FOR

PRAIRIE ISLAND NUCLEAR GENERATING PLANT

UNITS 1 AND 2

SPENT FUEL CASK DROP EVALUATION

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REVISION NO.	DATE	RELEASED BY	CHARGE NUMBER
0	1-20-84		NOR-0199
1	9/5/84	H. E. Pearson	NOR-0199
2	10/4/84	H. E. Pearson	NOR-0199

8412270195 841221
PDR ADOCK 05000282
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QUADREX CORPORATION

DOCUMENT #: QUAD-1-83-017
REVISION NO. 1
SHEET 1 OF 1

REVISION SHEET

ITEM	DESCRIPTION	PAGE NO.
	The report was completely revised.	All

Revised by: A. J. Elliott

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Approved by: C. Hossain

FORM QAP-107-3

QUADREX

DOCUMENT #: QUAD-1-83-017
REVISION NO. 2
SHEET 1 OF 1

REVISION SHEET

ITEM	DESCRIPTION	PAGE NO.
	Minor corrections and clarifications were incorporated	2-1,2-2 2-4, 3-1 thru 3-6 4-2,4-5, 4-6,4-10, 4-12 5-2,5-4 5-5 6-1,6-2

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FORM
QUAP-107-3

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1.0 INTRODUCTION

This report, prepared for the Northern States Power Company (NSP), summarizes the evaluation of the consequences of postulated drops of selected spent fuel casks at the Prairie Island Nuclear Generating Plant, Units 1 and 2. The evaluation was performed to remove the temporary restriction on the use of 266 of the storage spaces at Prairie Island for storage of spent fuel. This restriction was imposed upon the facility on May 31, 1981 when approval was granted by the NRC for expansion of the capacity of the existing storage pools (reference 1-2).

The overhead handling system for the spent fuel cask at the Prairie Island plant was not designed as a single-failure-proof system. Therefore, in accordance with NUREG-0612 (reference 1-3) requirements, the consequences of postulated drops of selected spent fuel casks were evaluated. This evaluation included consideration of the four casks currently licensed for transporting PWR fuel assemblies. For each cask, a minimum thickness honeycomb crash pad was sized. In addition, the impact limiter for each cask was evaluated for its capacity to limit the impact force upon the spent fuel pool floor to acceptable levels.

The configurations of the spent fuel pool structures and the arrangement of the high density racks in the pool are described in section 2 of this report. This section also describes the cask load path, the safety related items in the vicinity of the load path, the postulated cask drop orientations and the characteristics of the spent fuel casks. Section 3.0 provides a summary of the results of structural, nuclear criticality, and radioactivity release evaluations. The evaluation procedures and methodologies are described in section 4.0 and 5.0. The conclusions are presented in section 6.0. Section 7.0 lists the references used.

2.0 DESCRIPTIONS

2.1 Spent Fuel Pool Structures

The fuel storage area at Prairie Island Nuclear Generating Plant is located between the two reactor buildings, and consists of a new fuel pit, two pools for storing spent fuel, and a canal for transfer of fuel elements. The two spent fuel storage pools are designated as pool no. 1 and pool no. 2 as shown in figure 2-1. Pool no. 1, the smaller of the two pools, has inside plan dimensions of 18'-11" x 18'-3". Pool no. 2 has inside plan dimensions of 18'-11" x 43'-5". Normal water depth for both pools is about 40 feet. Pool no. 1 is designated as the cask loading and unloading area. Pool no. 1 contains nine racks. The four 7 x 7 racks in the southeast corner would be removed to provide the cask loading and unloading area. Only the remaining five racks in pool no. 1 would be used for fuel storage during loading of the spent fuel cask. Pool no. 2 holds 21 racks. A typical spent fuel storage rack is shown in figure 2-2. 2 2

2.2 Cask Load Path, Safety Related Items, and Postulated Drops

The spent fuel storage facility is surrounded by a reinforced concrete enclosure. Access into the enclosure for the spent fuel cask is provided by a door and a narrow slot in the ceiling for attaching the spent fuel cask to the overhead load handling system. Because of these physical constraints, the dropping of the cask into pool no. 2 is not considered credible.

The proposed load path for the cask is illustrated in figures 2-3 through 2-5. The cask is lifted from the basemat, elevation 693 ft., through a large opening in the floor slab at the 755 ft. elevation, laterally transferred and aligned with the access door to the fuel pool area, moved directly north to above the loading and unloading area of pool no. 1 and lowered into the pool. The narrow slot in the ceiling of the enclosure prohibits any movement of the cask except in the north-south direction.

The only safety-related items that can be impacted during a postulated fuel cask drop into the pool is the floor of pool no. 1 and the fuel and racks in pool no. 1. During the cask alignment over pool No. 1, the overhead handling system could be moving south. If, while the cask is moving south, a postulated drop occurred, the cask would have a small horizontal velocity component that could result in the cask imparting a glancing blow to the fuel pool wall. The walls are 5 foot thick reinforced concrete and they would experience only minimum local damage resulting in no loss of water from the pool. Therefore, this blow was judged not to have any safety consequence. Along the remainder of the load path, the only safety-related item which could be affected by a postulated drop is one of the two inlet lines for the fuel pool cooling system. Failure of the corbel support could result in one of the inlet lines being struck by the falling support. The consequences of this event are evaluated in section 3.1. There is no safety-related equipment located below the impacted basemat. 2

Four drop orientations into the spent fuel pool are postulated. Drop position 1 is a straight vertical drop as shown in figure 2-6. Drop position 2, shown in figure 2-7, is similar to drop position 1 except the cask is tilted so its center of gravity is directly above a corner. For drop positions 3 and 4, it is postulated that as the cask is transferred, it is dropped so an edge of the cask catches the edge of the pool and the cask tips over and falls in a horizontal position until it first strikes the racks, then the floor. For drop position 3, it is assumed the racks are rigid and the cask attains the worst orientation for honeycomb penetration. For drop position 4, the rack crush and impact angle is determined and with these parameters the cask penetration into the honeycomb crash pad (if used) is obtained. These positions are illustrated in figures 2-8 and 2-9, respectively. 2

2.3 Description of Spent Fuel Casks

The four spent fuel casks considered in this study are currently licensed for off-site transportation of PWR spent fuel. They are the Nuclear Assurance Corporation NLI-1/2 and NLI-10 24 casks, the General Electric IF-300 cask, and the Transnuclear TN-8[illegible]. Each of these casks is provided with an impact limiter at each end to mitigate the consequence of an accident that might occur during transportation. All of the impact limiters are removable except for the IF-300 cask. The casks with removable impact limiters were evaluated in both configurations.

The characteristics of these casks are presented in table 2-1. The deceleration values presented in table 2-1 are for a 30-foot drop of the spent fuel cask in air with the lower impact limiter in place onto a hard unyielding surface. These deceleration values were determined by the cask vendors using analytical methods, and in some cases, from scale model testing. A separate value is provided for a bottom and an edge impact. The bottom impact value corresponds to the straight drop position 1 as illustrated in figure 2-6. The edge impact value corresponds to the inclined drop positions 2, 3, & 4 as illustrated in figures 2-7 through 2-9.

TABLE 2-1

SPENT FUEL CASK SUMMARY

Label	Supplier	Cask Weight[1] (Kips)	Heavy Load Weight[2] (Kips)	Decel.[3] (g)	References
NLI-1/2	Nuclear Assurance Corp	47.5	53.9	37.4 37.4	2-1, 2-2
NLI-10/24	Nuclear Assurance Corp	194.0	206.0	29.4 30.95	2-3, 2-4,
IF-300	GE	136.	145.0	234. 68.7	2-5, 2-6
TN-8L	Trans-nuclear	78.7	86.0	155. 80.	2-7, 2-8

Notes:

(1): The cask weight is for a fully loaded cask (with water, if applicable).

(2): The heavy load weight includes a fully loaded cask (with water, if applicable) and all equipment required for lifting and set down such as base plates, lifting yokes, wire ropes, and crane blocks.

(3): These are the deceleration values for the bottom and edge impacts respectively.

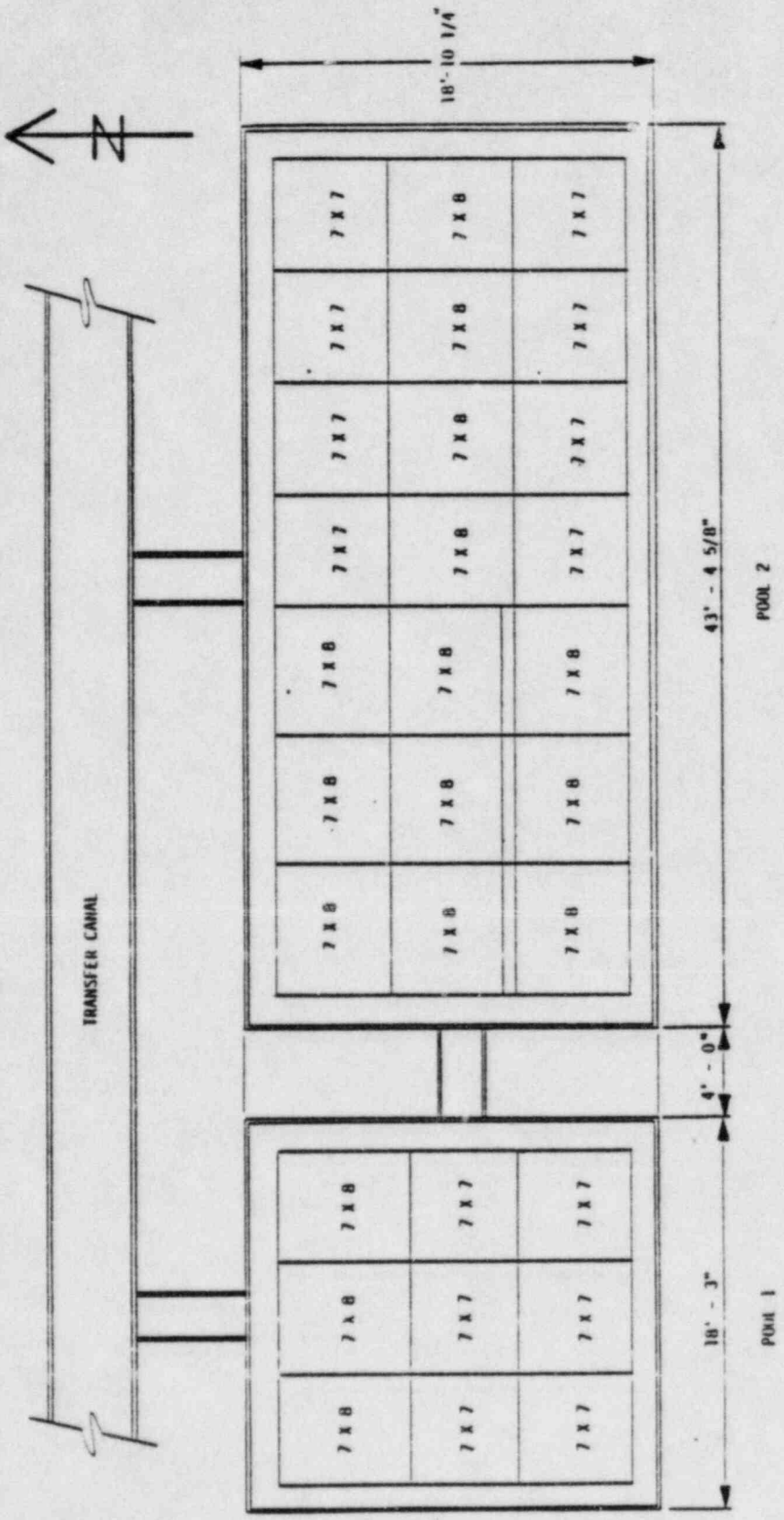

Figure 2-1. Arrangement of Spent Fuel Racks (Ref. 3-3))

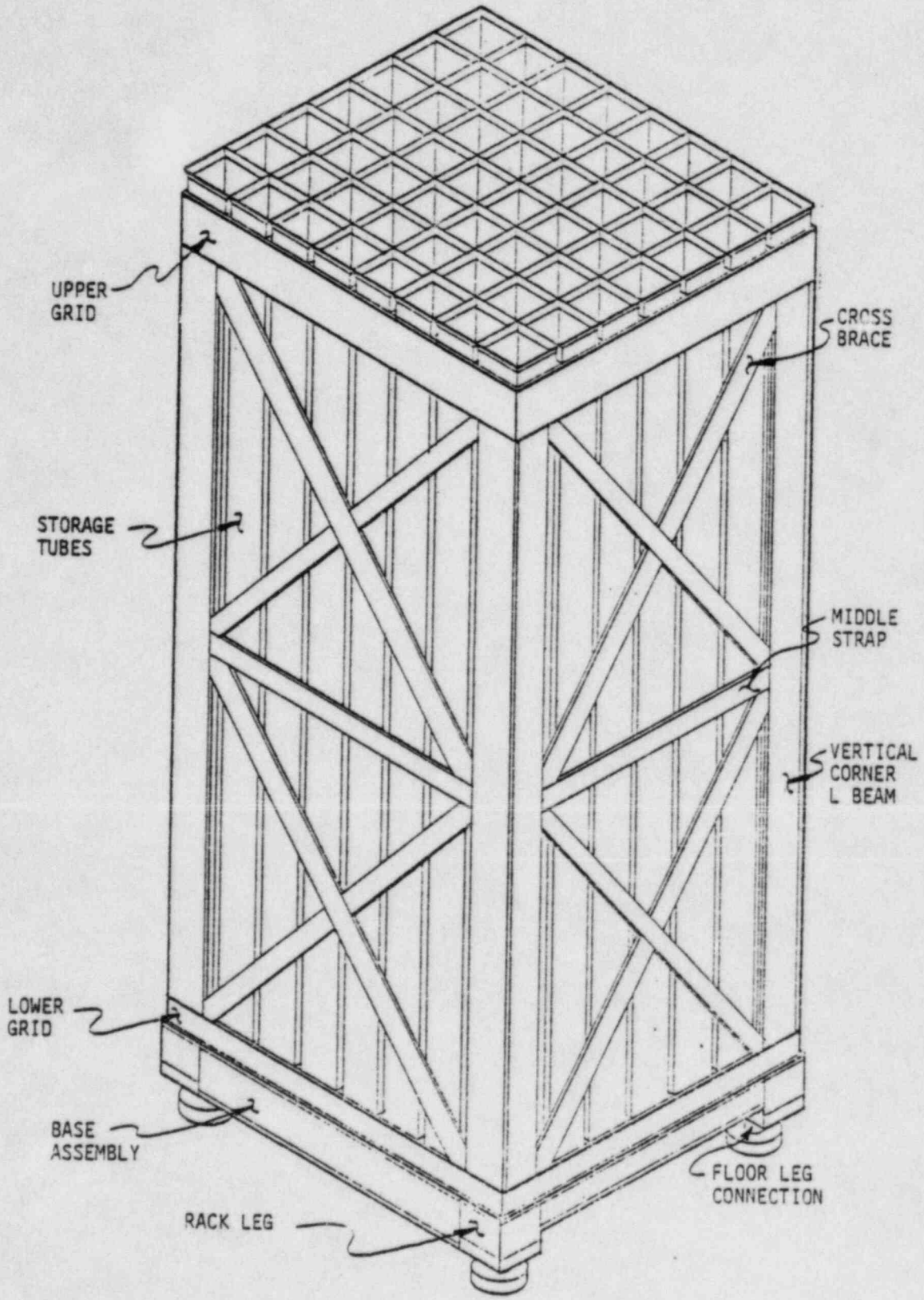

Figure 2-2. 7x8 Spent Fuel Storage Rack (Ref. 3-3)

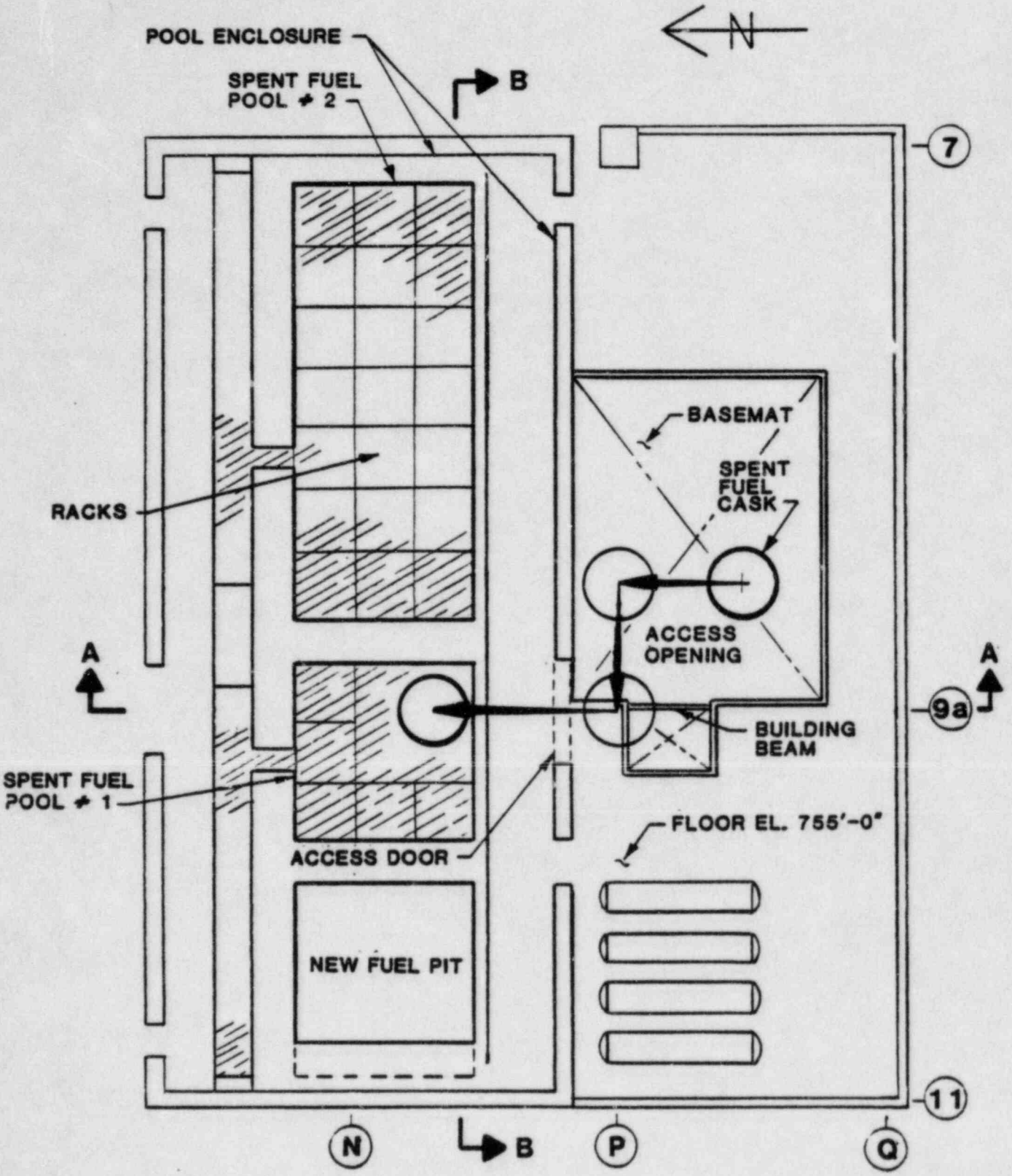

Figure 2-3. Load Path for Spent Fuel Cask - Plan View

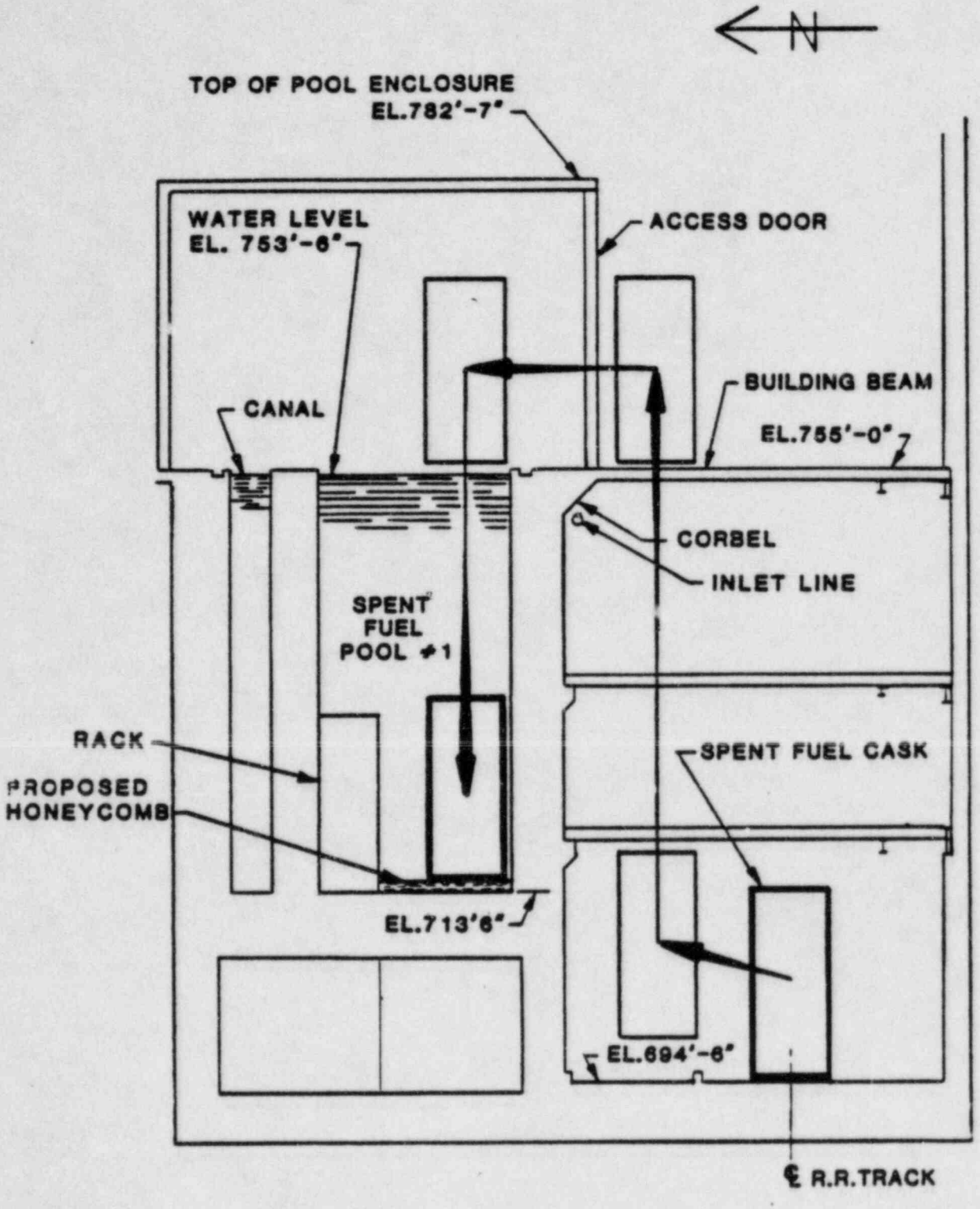

Figure 2-4. Load Path for Spent Fuel Cask-Elevation A
(Looking East)

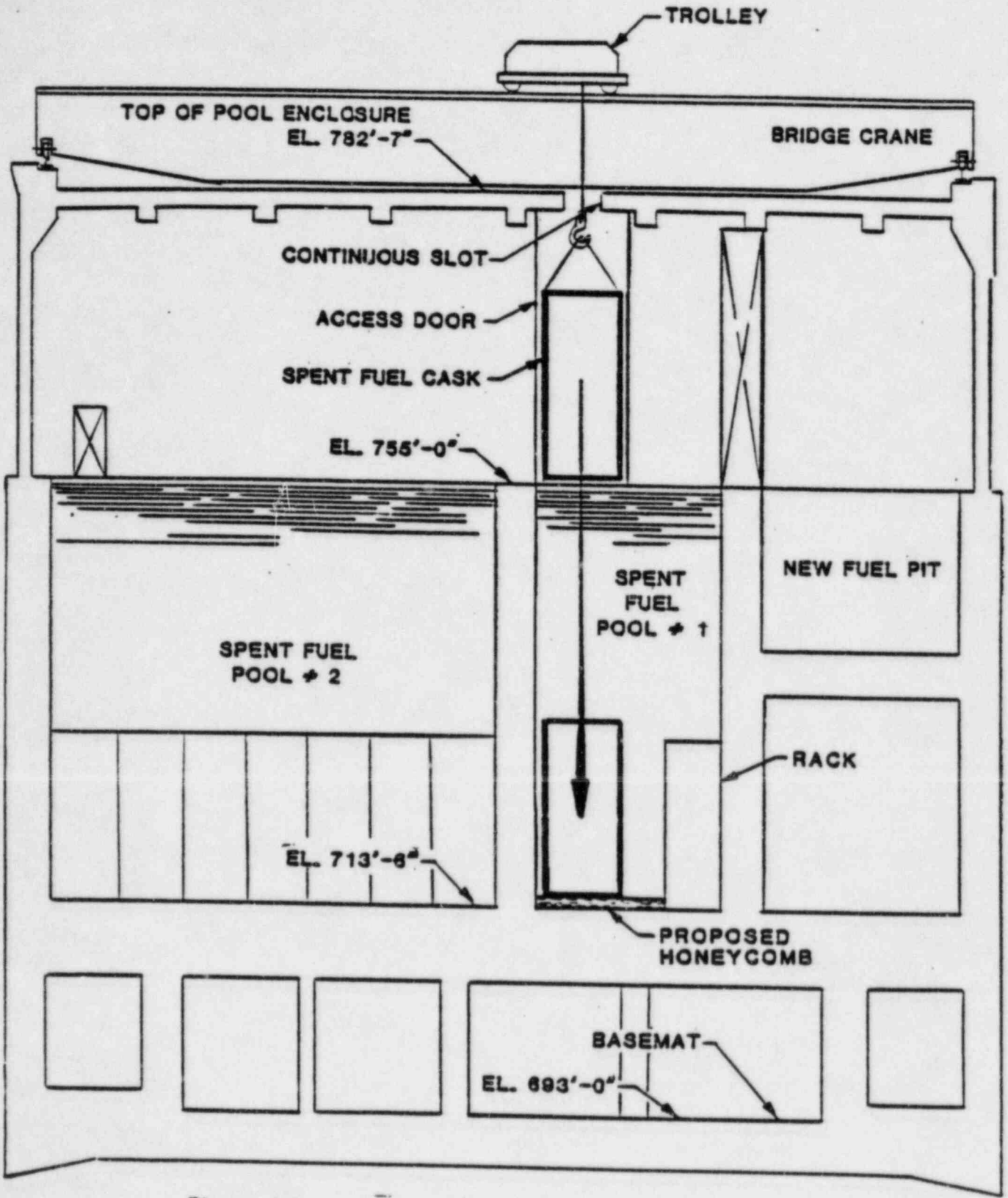

Figure 2-5. Load Path for Spent Fuel Cask - Elevation B (Looking South)

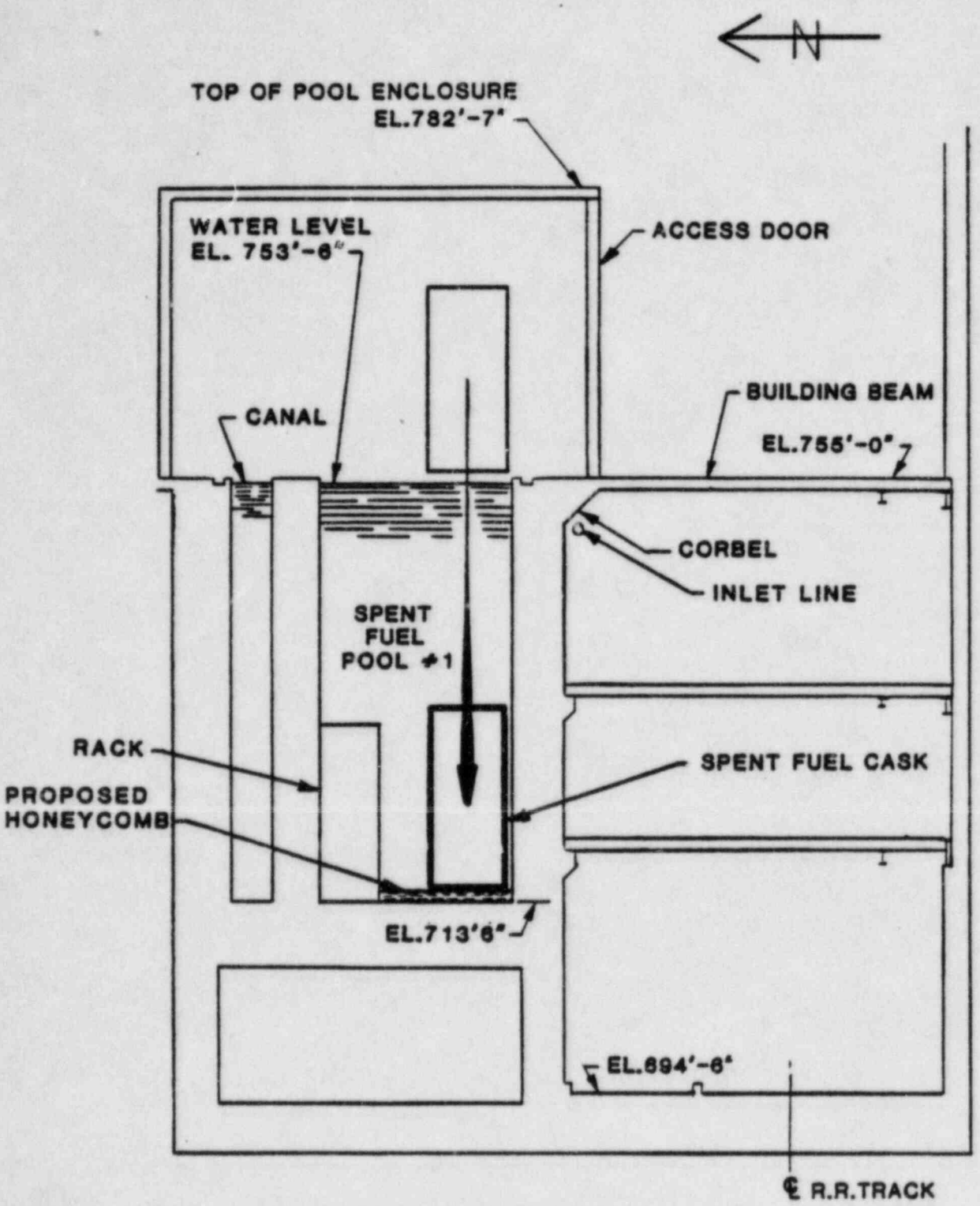

Figure 2-6. Postulated Drop Position 1

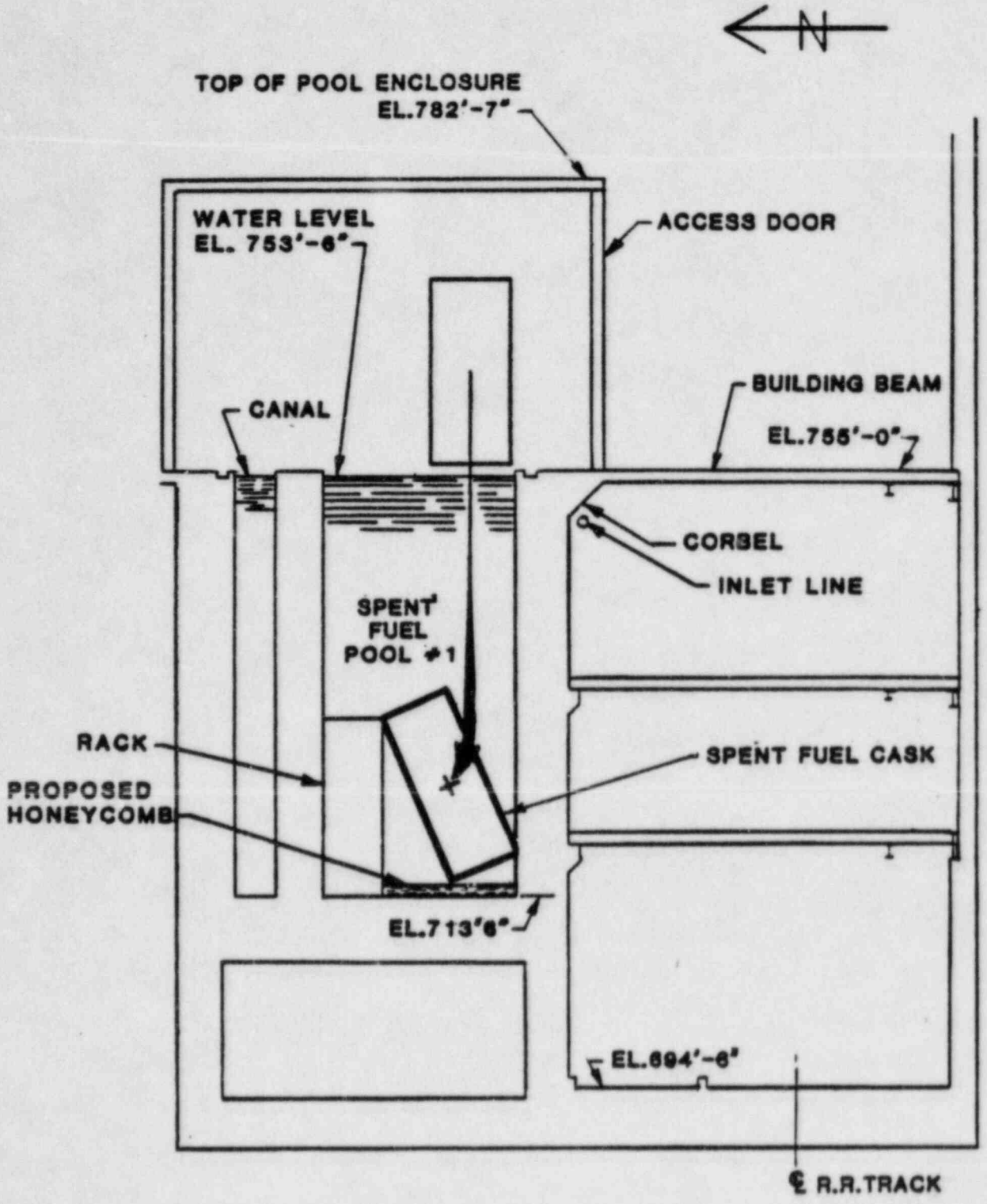

Figure 2-7. Postulated Drop Position 2

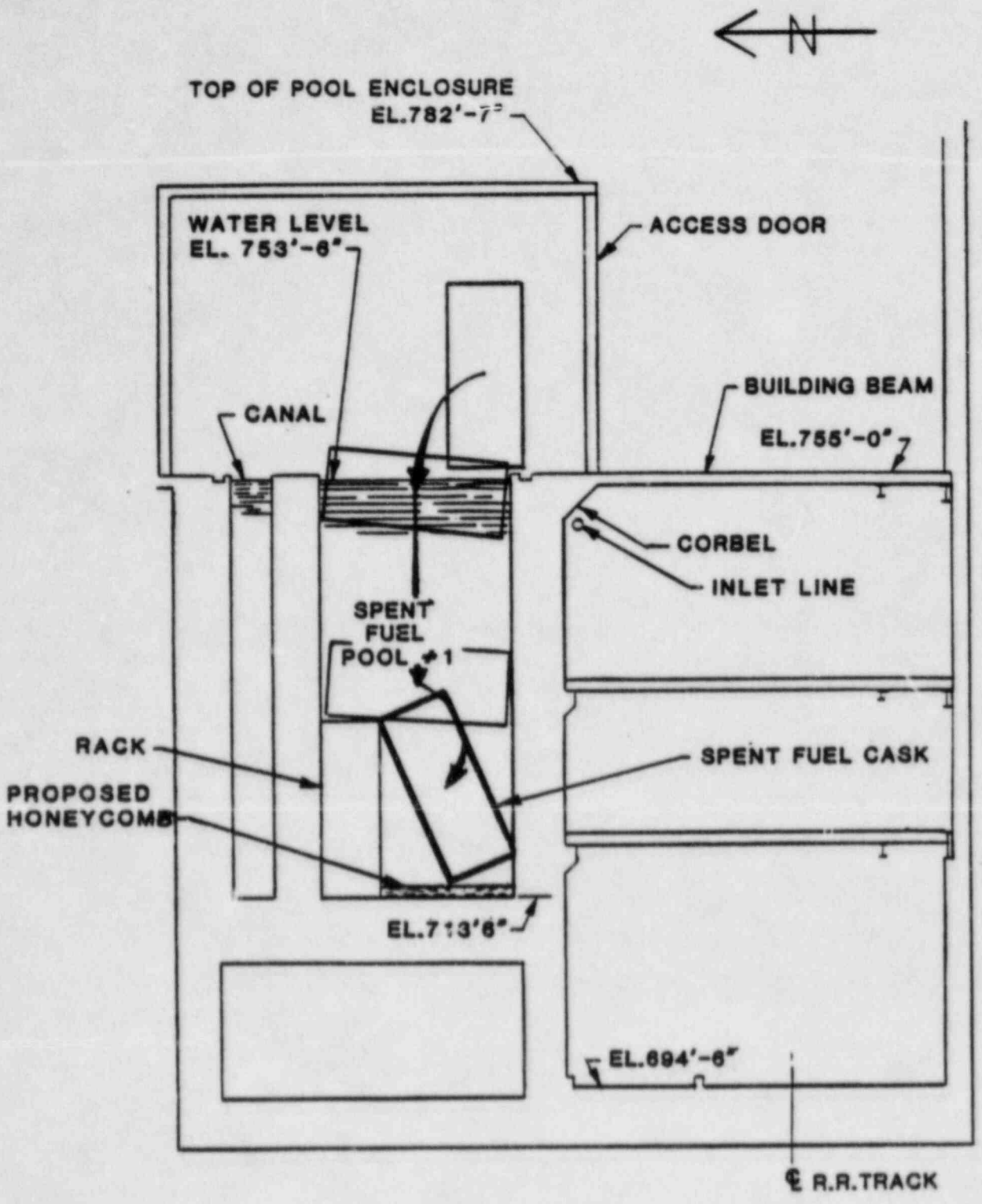

Figure 2-8. Postulated Drop Position 3

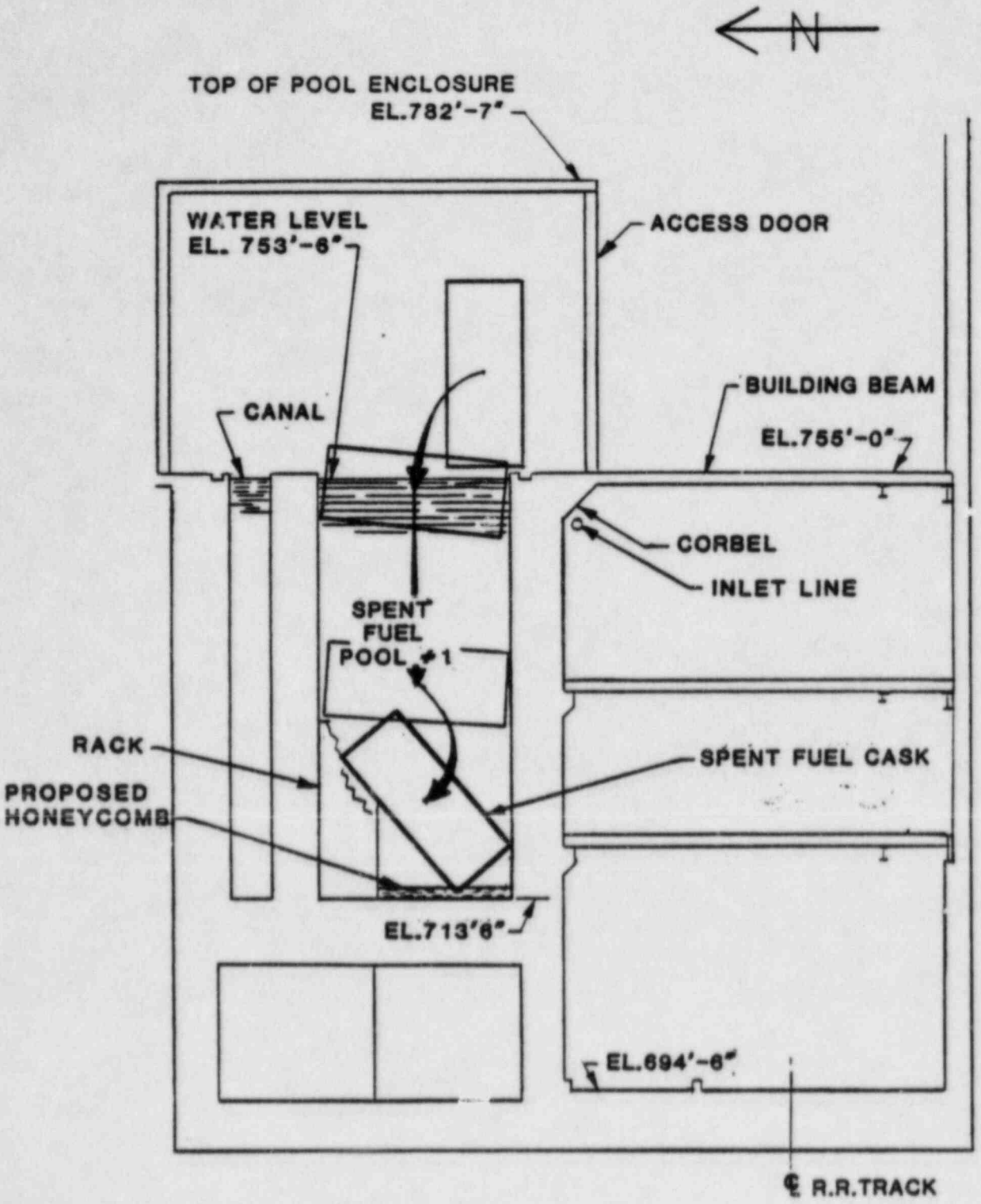

Figure 2-9. Postulated Drop Position 4

3.0 SUMMARY OF EVALUATION

The overhead handling system for the spent fuel cask was not designed as a single-failure-proof system. Therefore, in accordance with NUREG-0612 (reference 3-1) requirements, the consequence of postulated drops of selected spent fuel casks were evaluated. The results of the evaluation are provided in table 3-1. In addition to the drop evaluation results, the table also identifies the safety-related equipment or structure that may be affected if a spent fuel cask is accidentally dropped, the frequency of the lift, the hazard elimination category, and the thickness of the impacted spent fuel pool slab. 2

The evaluation results are summarized in this section. The details of the evaluation are provided in section 4.0 and 5.0.

3.1 Structural Evaluation Results

The evaluation considered the possible drops in the fuel handling area and in the fuel pool enclosure. Safety-related structures and components in the vicinity of the load path were assumed to be impacted by the postulated drops. These included the following:

- Basemat
- Building beam
- Corbel support
- Concrete floor slab at elev. 755'-0"
- Fuel pool floor
- Fuel racks
- Spent fuel

The postulated drops on the basemat, building beam, corbel support, and floors were evaluated to predict the extent of potential structural damage. The results show that the potential damage to the basemat, building beam, corbel supports and the concrete floor slab at elev. 755'-0" will not affect the safety-related function of the building. The nuclear criticality and

radioactivity release evaluations (see sections 3.2 and 3.3) show that even extreme damage to the impacted racks and spent fuel assemblies due to the postulated drops would not result in criticality condition nor unacceptable amount of release.

The structural evaluation of the corbel support indicated a postulated drop of the cask could result in the failure of the corbel support. The corbel support could fail by separating from the wall. The cask, since it would not be supported by the corbel support, would fall in the direction of the failing support, toward the hatchway, away from the pool. No damage to the pool would result. The falling corbel support could strike a fuel pool cooling inlet line, causing it to rupture. If this occurred, it would be necessary to isolate the fuel pool cooling system from the pool. The loss of water from the pool would be limited because there is no structural damage to the pool and the cooling system piping is so arranged that failure of any pipeline cannot drain the pool below the top of the stored fuel elements. If all cooling were lost and no other action was taken, the pool water temperature would rise until boiling at the surface began after several hours. For this condition, the maximum fuel clad temperatures would be 252°F. This is well below normal fuel operating temperature. The maximum evaporation rate after initiation of boiling is 44.7 gpm (reference 3-3). Two sources of makeup water that use inlets different than the cooling system inlets are available. They are the four demineralized water hose stations, each rated at 20 gpm and two fire hose stations, each rated at 95 gpm (reference 3-2). The total available capacity is 270 gpm which exceeds the maximum evaporation rate. As loss of the fuel pool cooling system was part of the design basis for the pool (see FSAR section 9.3.1 and USAR section 10.2.2). No further evaluation of this postulated event was performed. 2

The conservative structural evaluation of the fuel pool floor indicated that in order to limit the impact load to within the structural capacity of the floor, an energy-absorbing crash pad would be required unless adequate impact limiters are provided with the casks. For the four casks evaluated 2

in this study, the required thicknesses of honeycomb-type crash pads are shown in table 3-2 when the casks are used without the lower impact limiters attached. For two of the four casks (NLI-1/2 and NLI-10/24), it was determined that no additional crash pad would be necessary if the lower impact limiters are attached to the cask when the cask is lowered into the pool. Typical cask/crash pad combinations are shown in table 3-2 which limit the impact load to within the structural capacity of the pool floor slab.

3.2 Nuclear Criticality Evaluation Results

The nuclear criticality results are presented in figure 3-1 in the form of a curve of K_{eff} versus water/UO_2 volume ratio. It shows that if all the uncertainties defined in reference 3-1 are included, the maximum K_{eff} would be less than 0.95 if the boron concentration in the pool is maintained above 1800 ppm. Thus, the criticality criteria would be satisfied. 2

3.3 Radioactivity Release Evaluation Results

The results of the radioactivity analysis are presented in figure 3-2. Two curves are included in the figure: One curve is for the case that includes charcoal filters and the other curve is for the case that does not include charcoal filters. It was conservatively decided not to take credit for the charcoal fitlers. This is conservative because the fuel pool enclosure design includes a spent fuel pool special ventilation system which is initiated by high radiation in the spent fuel pool area. Credit was not taken for this system, due to the slot in the enclosure ceiling and the enclosure door both being open while the cask is being handled; however, some clean up would be realized by this system. Twerty-five percent of 10 CFR, part 100 values were used as the radiological dose acceptance criteria. The maximum number of fuel assemblies that can be damaged is 80. The plant technical specification limits the number of recently discharged fuel assemblies to 45 (reference 3-4). Therefore, all 45 recently 2

discharged fuel assemblies plus 35 other fuel assemblies were assumed to be damaged. The minimum number of days after reactor shutdown that a postulated fuel cask drop accident could occur without exceeding the dose acceptance criteria was determined to be fifty for the case that does not include charcoal filters . 2

TABLE 3-1. SUMMARY OF HAZARD ELIMINATION

Frequency[1] of Cask Lift (No./Yr.)	Safety-Related Equipment/Components/ Structures Involved in Postulated Drop	Hazard Elimination Category	Thickness of Impacted Structure (in.)
7 to 65	Basemat, Building Beam, Corbel Supports, and Floor at elevation 755 feet	3	N/A
	Fuel Pool Floor (Pool 1)	2,3	75
	Loaded Fuel Rack	3	N/A

Notes:

1. The frequency of lift is for normal plant operation. A cask has not yet been used.

2. Cask drop may result in minor local damage and yielding of the impacted slab, but the slab possesses sufficient capacity to absorb the impact energy without exceeding the maximum allowable ductility ratios. For the fuel pool floor, an energy-absorbing crash pad would be required. However, for NLI-1/2 and NLI-10/24 the lower impact limiter licensed for transportation use can be used in lieu of the crash pad on the pool floor.

3. The analysis demonstrates that a postulated crane failure and load drop will not damage safety-related equipment.

2

Table 3-2. REQUIRED THICKNESS OF HONEYCOMB CRASH PADS

Cask Label	Hexcel Honeycomb Label	Nominal Crush Strength (σ) nom (ksi)	Honeycomb Thickness (in)
NLI-1/2	1/8-2024-.003	1.12	35.7
NLI-10/24	1/8-5056-.002	.650	58.6
IF-300	1/8-2024-.003	1.12	46.7
TN-8L	1/8-2024-.003	1.12	40.0

2

Note: The crush strength and thickness are based on the honeycomb being in a stainless steel can.

2

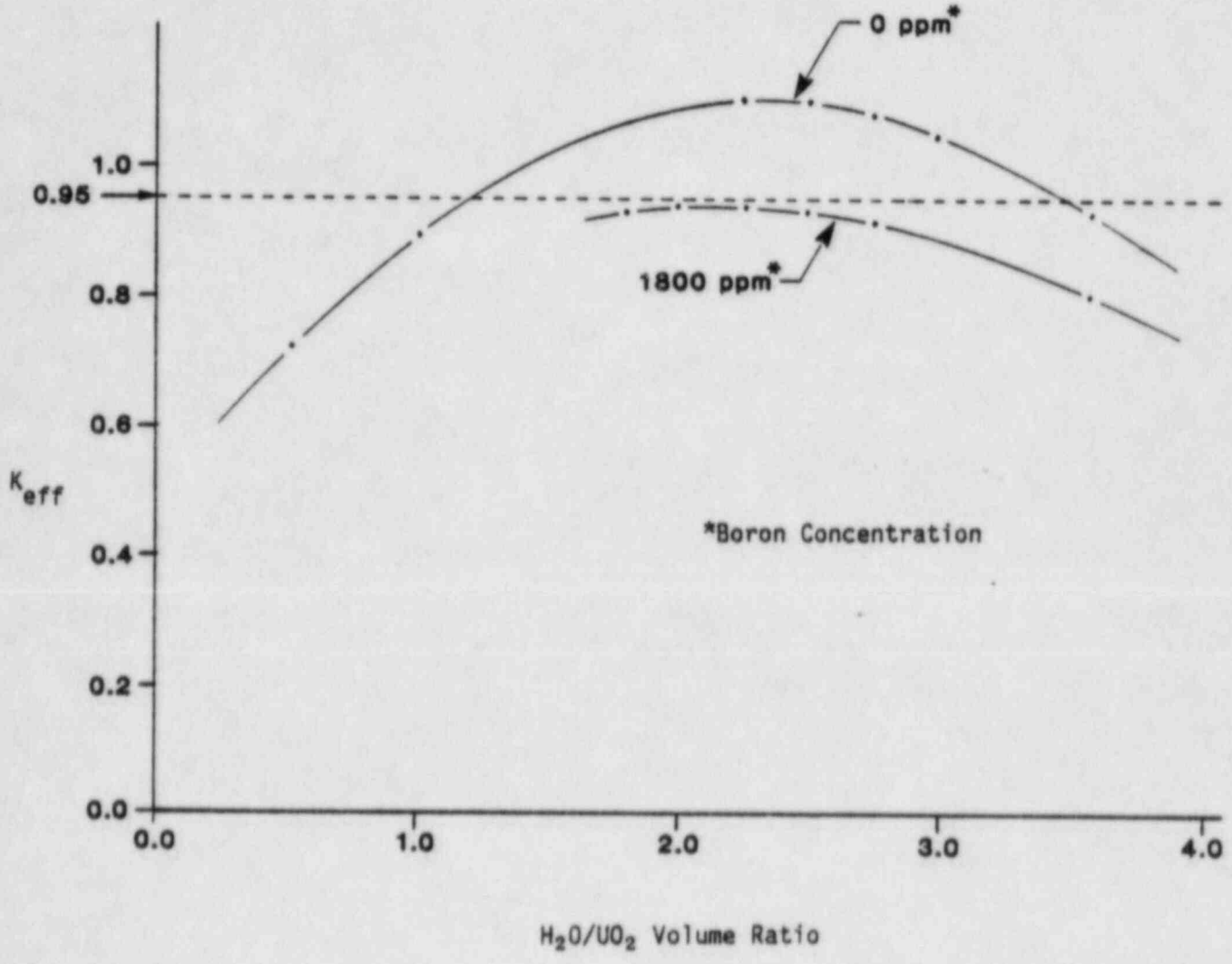

Figure 3-1. K_{eff} vs water/UO_2 volume ratio

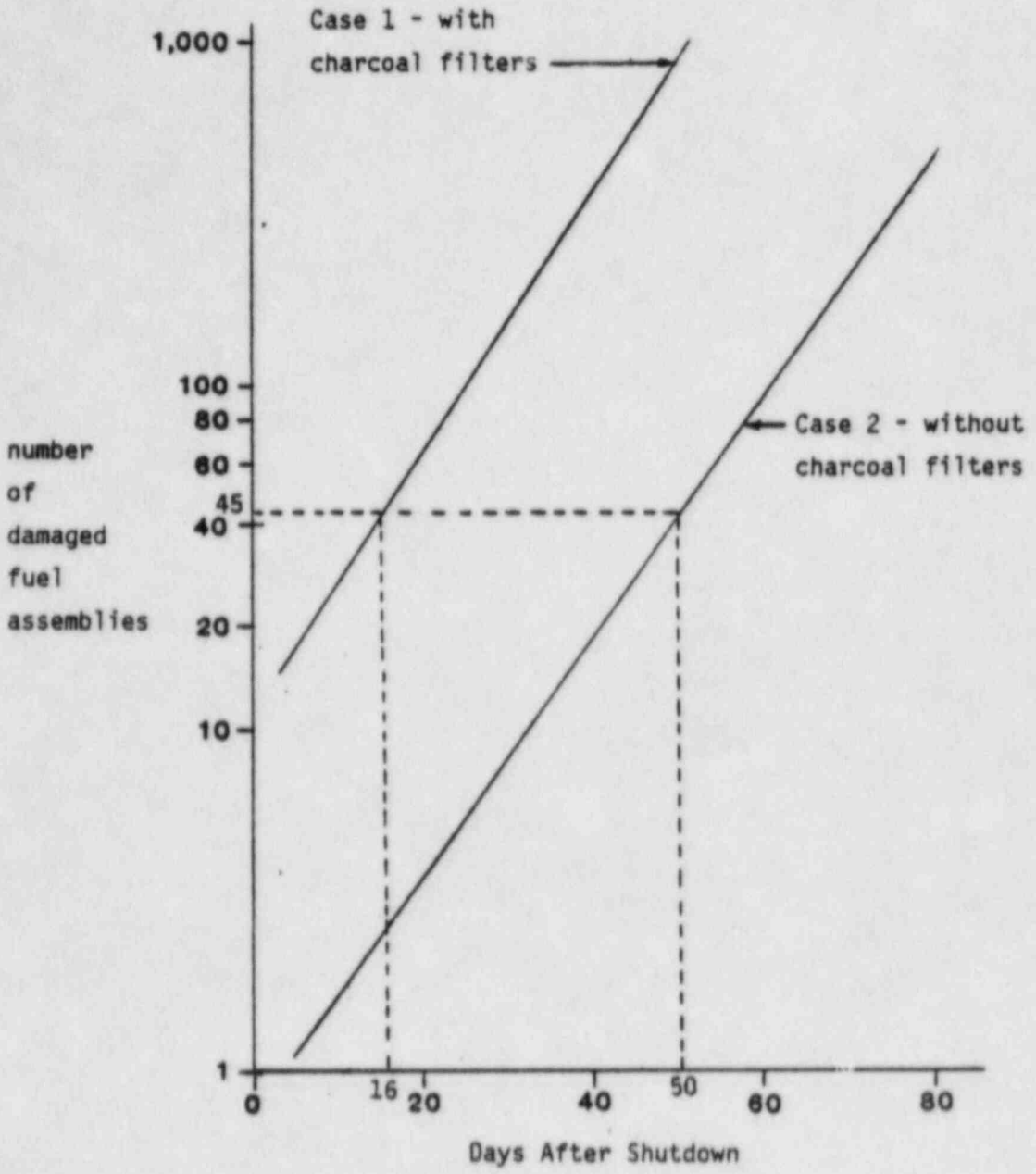

Figure 3-2. Number of Fuel Assemblies That Could be Damaged With the Dose Less than 1/4 of 10 CFR, Part 100 Values at the Exclusion Area Boundary Versus Time After Reactor Shutdown

4.0 STRUCTURAL EVALUATION METHODOLOGY

In this study two methods for mitigating the effect of dropping the cask onto the spent fuel pool floor were evaluated. The first method used a honeycomb crash pad on the pool floor to absorb the impact energy. The second method used the impact limiters currently used as protection against shipping accidents.

The sizing of the honeycomb crash pads consists of the following steps:

1. Determine the structural capacity of the spent fuel pool floor.
2. Determine impact velocities of the cask.
3. Evaluate the consequences of the cask impacting the fuel rack.
4. Size the honeycomb crash pad.

The evaluation of the impact limiters for mitigating the impact force upon the spent fuel pool floor to acceptable levels consists of the following steps:

1. Determine the structural capacity of the spent fuel pool floor.
2. Determine impact velocities of the cask.
3. Evaluate the consequences of the cask impacting the fuel rack.
4. Obtain the impact force of the cask with the impact limiter and compare the impact force to the structural capacity of the floor.

In steps 1 and 2, for the honeycomb crash pad sizing, the floor capacity and impact velocity were determined by assuming that the removable impact limiters would not be attached to the cask during the postulated drop. The capacities and impact velocities thus obtained were also used for evaluating the drop cases in which the removable impact limiters were assumed to be attached to the cask. For the IF-300, the impact limiter is not removable; therefore, the capacity and impact velocity are the same for both cases. For the other casks, the assumptions are conservative because the attached impact limiters would:

3-3 Licensing report for Prairie Island Nuclear Generating Plant units 1 and 2, Spent Fuel Storage Modification, QUAD-1-79-509, December 28, 1979.

3-4 Prairie Island Technical Specification.

4-1 "Licensing Report for Prairie Island Nuclear Generating Plant, units 1 and 2, Spent Fuel Storage Modification," Report no. QUAD-1-79-509.

4-2 "Other Category I Structures," USNRC Standard Review Plan, section 3.8.4

4-3 ACI 318-77, "Building Code Requirements for Reinforced Concrete," American Concrete Institute

4-4 ACI 349-76, "Code Requirements for Nuclear Safety Related Concrete Structures," American Concrete Institute, with 1978 supplement.

4-5 Structural Analysis and Design of Nuclear Plant Facilities, American Society of Civil Engineers, Manual no. 58, 1980

4-6 Topical Report BC-TOP-9-A, "Design of Structures for Missile Impact," rev. 2, Bechtel Power Corporation, September 1974.

4-7 Biggs, J. M., "Introduction to Structural Dynamics," McGraw-Hill Book Company, 1964

4-8 Wood, R. H., "Plastic and Elastic Design of Slabs and Plates," The Ronald Press Corporation, New York, 1961.

4-9 Ramakrishnan, V. and Arthur, P. D., "Ultimate Strength Design for Structural Concrete," Sir Isaac Pitman & Sons, Ltd., London.

1. Distribute the impact load over a larger area, thus mobilizing a larger resistance capacity of the floor.
2. Provide a larger frontal area, thus enhancing the drag force.
3. Provide a larger volume, thus increasing the buoyancy forces.

Increase in drag and buoyancy forces would decrease the impact velocity.

The analytical methods and assumptions used in the evaluation of plant structures subjected to the impact from a postulated spent fuel cask and rack drop are described in the following subsections. Subsection 4.1 provides the loads and allowables for which the pool structure was evaluated. The capacity of the spent fuel pool floor is presented in subsection 4.2. The impact velocity of the spent fuel cask upon the racks and floor is provided in subsection 4.3. The honeycomb crash pad design method is summarized in subsection 4.4. Evaluation of the postulated rack impact is given in subsection 4.5. Subsection 4.6 presents the evaluation of the cask impact with the impact limiters in place. 2

4.1 Loads and Allowables

The fuel pool structures were evaluated for the following loads:

a. Deadweight (D) - For the pool floor, the deadweight included the buoyant weight of the loaded spent fuel racks at 2.0 kips per square foot (reference 4-1) plus the weight of the water.

b. Impact Force (F_i) - This is the dynamic load applied to the building structure and produced by the deceleration of the impacting spent fuel cask. In all cases, this force was conservatively assumed to be a suddenly applied load (instantaneous rise time and infinite duration).

The structural adequacy was verified in accordance with the applicable portions of the USNRC standard review plan section 3.8.4 (reference 4-2). The allowable section strengths were taken from the American Concrete

Institute (ACI) Code 318-77 (reference 4-3). The allowable ductility ratios, presented in table 4-1, were taken from references 4-4, 4-5, and 4-6.

The impact force acting on the target structures during a postulated cask drop was determined and combined with the dead load to determine if it was within allowables. This was accomplished by satisfying the following inequality:

$$F_i \leq \frac{P_u - D}{DAF}$$

where,

P_u = ultimate capacity of the floor per the ACI Code 318 (reference 4-3).

DAF = Dynamic Amplification Factor. These were obtained from reference 4-7 for the applicable allowable ductility ratios presented in table 4-1.

D = Deadweight

4.2 Capacity of the Spent Fuel Pool Floor

The capacity of the spent fuel pool floor was determined for impact loads resulting from the drop of the spent fuel casks. The capacity was expressed as the magnitude of a suddenly applied load (instantaneous rise time and infinite duration) which when applied would result in exceeding the structural allowables. The following possible failure modes were considered:

- bending
- reaction shear
- punching shear

For each of these postulated failure modes, the resistance-displacement function of the structure was idealized as elasto-plastic. For determining the resistance capability, the ultimate capacities were calculated in accordance with the ACI Code 318-77 (reference 4-3). The allowable suddenly applied force was obtained by using the method presented in reference 4-7 and on the basis of the allowable ductility values given in table 4-1.

4.2.1 Bending Capacity

The ultimate moment capacity M_u was calculated according to the ACI Code 318-77 (reference 4-3). The ultimate capacity (P_u) for the reinforced concrete slab was determined by using the yield-line method of analysis (references 4-8 and 4-9). To ensure that the minimum collapse load or lower bound value was obtained, more than one yield-line pattern was considered.

The bending capacity was computed to be 6180 Kips by conservatively assuming a point load and by using the applicable allowable ductility ratio of 10.

4.2.2 Reaction Shear Capacity

The reaction shear capacity was determined in accordance with the ACI Code 318-77 (reference 4-3). Two cases were considered. For the first case, the cask was assumed to drop within a distance 'd' from the edge of the slab support (where 'd' is the effective thickness of the slab). For the second case, the cask was assumed to drop outside the distance 'd' from the slab support. The first case provided lower capacity, and conservatively was used in the evaluation.

Conservatively assuming that the impact limiter was not attached to the cask, and using the applicable allowable ductility ratio of 3.0, the

reactor shear capacities were computed for the four casks. The computed capacities are presented in table 4-2.

The reaction shear capacity of the floor slab for cask impact on the racks was conservatively determined by ignoring the support of the north-south wall under the slab. The applicable allowable ductility ratio was 1.6. Using this ductility limit, the reaction shear capacity was computed to be 6200 Kips. The capacity for all casks is identical because the load distribution was conservatively assumed to be equal to the load distribution for the cask with the minimum diameter.

4.2.3 Punching Shear Capacity

The punching shear capacity was determined in accordance with the ACI Code 318-77 (reference 4-3). For the direct impact of the cask upon the floor, the effective perimeter of the concrete section resisting the shear is within d/2 of the cask, where d is the effective thickness of the slab. For impact of the cask upon the rack, the effective perimeter of the resisting section is within d/2 of the minimum footprint of the cask upon the rack. Using these perimeters and the applicable allowable ductility ratio of 1.6 the punching shear capacities were computed. These are tablulated in table 4-2. | 2

4.3 Impact Velocities

The effects of the drag and buoyant forces on the impact velocity of the cask (the floor is impacted in drop positions 1 and 2, and the rack is impacted in drop positions 3 and 4) were included by following the procedure given in BC-TOP-9A (reference 4-10). The drop positions are defined in subsection 2.2.

For drop position 1, the axis of the cask is parallel to the direction of motion. For drop position 2, it is not; however, it is conservatively assumed to be parallel to the direction of motion. With the drag

coefficient of 0.85 from reference 4-10, the maximum impact velocity for the casks upon the floor for both positions was computed to be 42.1 fps. With the drag coefficient of 0.35 from reference 4-10, the maximum impact velocity upon the racks was determined to be 38.3 fps for drop positions 3 and 4. 2

4.4 Honeycomb Crash Pad

The honeycomb crash pad properties for each of the spent fuel casks are determined by the following steps:

- o obtain the maximum allowable crash force
- o select a common commercially available honeycomb
- o determine the maximum honeycomb penetration
- o calculate the honeycomb crash pad thickness.

These steps are described in the following paragraphs.

The impact force upon the floor is controlled by the honeycomb crush strength and the impact area of the cask. The postulated drop position 1 provides the maximum area; therefore, this postulated drop position would result in the maximum impact force on the floor. By using the floor capacities and the cask projected area, the maximum allowable honeycomb crush strength was determined that would result in an impact force below the floor capacity. For conservatism, it was assumed that the maximum allowable honeycomb crush strength included:

- A 30% increase in the static crush strength due to the dynamic loading condition (dynamic increase factor) (reference 4-11)
- A 15% increase in the nominal static crush strength (reference 4-12).

Based on these considerations, a nominal static crush strength was calculated. A common commercially available honeycomb was selected from reference 4-12 that had a nominal static crush strength less than the one calculated. The maximum allowable crush strength (σ_{max}) and the nominal crush strength of the honeycomb selected (σ_{nom}) are tabluated in table 4-3.

The maximum penetration of the cask into the honeycomb for the postulated drop positions was needed to select the honeycomb crash pad thickness. A total of four postulated drop positions were considered. For conservatism, it was assumed that the crush strength:

- did not include any dynamic increase factor
- was 15% less than the nominal static crush strength of honeycomb selected (reference 4-12).

The penetrations depths for all of the postulated drop positions are presented in table 4-3.

The required honeycomb pad thickness was obtained by multiplying the maximum penetration from all of the postulated drop positions by a factor (1.4) to ensure a constant crush force during deceleration of the cask (reference 4-11). These steps resulted in a crash pad thickness that may be reduced if the final design is based on properties derived from static and dynamic test results of the honeycomb types under consideration.

4.5 Spent Fuel Rack Impact

For drop positions 3 and 4, a sideways fall of the cask was postulated. Therefore, the racks are impacted prior to the cask impacting the honeycomb crash pad or floor. For this event, there are three considerations:

- o The maximum crush force from the rack upon the floor must be within the floor capacity..

- o The maximum rack crush must be within allowables to ensure that the rack does not "bottom out" during the crush process thus imparting a high impact force to the floor.

- o The impact angles and the energy absorbed by the rack are necessary for the drop position 4 honeycomb penetration calculations.

The ultimate compressive strength of the stainless steel storage tubes was determined by the conventional theory presented in reference 4-13. The column buckling load of the fuel rods was determined. The maximum crush force was obtained by adding the strength of the main tube with the fuel rod buckling strength. The impact area (foot print) was determined for the cask without the impact limiter by multiplying the cask diameter by the overlap dimension (O). The results with a comparison to the floor capacity are tabulated in tables 4-4 (for the bare cask) and 4-5 (for the cask with the impact limiter).

For the maximum crush calculations, the rack was conservatively assumed to be empty. A conservatively low crush force was obtained by using the experimental data provided in reference 4-12 and energy calculations.

For the postulated drop position 4, the rack was assumed to deform during the cask impact. For the honeycomb crash pad penetration, the impact angle and the energy absorbed by the racks was determined. In addition, the maximum rack deformation (S_c) is computed to ensure that the rack would not "bottom out" during the crush process thus imparting a higher impact force to the floor. The computed values are compared to the allowable values in table 4-6.

4.6 Cask Drop with Impact Limiters

The deceleration values presented in table 2-1 were used to evaluate the structural effect of the cask impact upon the spent fuel pool floor without the mitigating effect of the honeycomb crash pad. These values were determined by the cask supplier from a test in which the cask is dropped 30 foot in air onto a hard unyielding surface. This drop results in an impact velocity of 43.9 fps which is higher than the impact velocity of 42.1 fps with which the cask at the Prairie Island would impact the spent fuel pool floor. Therefore, the deceleration values obtained from the

30 foot air drop are valid for the postulated drop upon the spent fuel pool floor. The peak impact force was obtained from:

$$F_1 = W_T \{\frac{\ddot{x}}{g}\}$$

where,

W_T = Weight of the cask + the weight of the handling fixture + the block weight

$\ddot{x}$ = Deceleration in g's.

These values are tabulated in table 4-7 and compared to the floor allowables.

TABLE 4-1
ALLOWABLE DUCTILITY RATIOS

REINFORCED CONCRETE	Maximum Allowable Value of of The Ductility Ratio (μ)[1]	2
Flexure		
Beams and Slabs	$\frac{0.10}{\rho-\rho'} \leq 10$	
where		
ρ is the ratio of tensile reinforcement = $\frac{A_s}{bd}$		
ρ' is the ratio of compressive reinforcement = $\frac{A'_s}{b\ d}$		
Compression		
Walls and Columns	1.3	
Shear		
Load carried by concrete alone	1.3	
Load carried by concrete and reinforcing steel	1.6	
Load carried completely by reinforcing steel	3.0	
STEEL ELEMENTS		
Members proportioned to preclude lateral and local buckling	10	
Flexure, compression, and shear	10	
STEEL COLUMNS		
Proportioned to preclude elastic buckling	1.3	
STEEL TENSILE MEMBERS		
Stressed in tension only	$0.5\frac{e_u}{e_y}$	

where

e_u = ultimate uniform strain[2]

e_y = yield strain

[1] See references 4-4, 4-5, and 4-6.

[2] Ultimate uniform strain is the strain at ultimate stress. | 2

TABLE 4-2
FUEL POOL FLOOR CAPACITY FOR POSTULATED DROP
OF SPENT FUEL CASKS

Cask	CASK IMPACT ON THE FLOOR			CASK IMPACT ON THE RACK		
	Floor Capacity			Floor Capacity		
	Bending F_1(kips)	Reaction Shear F_1(kips)	Punching Shear F_1(kips)	Bending F_1(kips)	Reaction Shear F_1(kips)	Punching Shear F_1(kips)
NLI-1/2	6180.	4170.	4970.	6180.	6200.	6240.
NLI-10/24	6180.	7190.	6740.	6180.	6200.	7320.
IF-300	6180.	5390.	5690.	6180.	6200.	6610.
TN-8L	6180.	5630.	5830.	6180.	6200.	6460.

TABLE 4-3

HONEYCOMB CRUSH STRENGTHS AND CASK

PENETRATION DEPTHS FOR VARIOUS DROP POSITIONS

CASK Label	CRUSH STRENGTH (ksi)		MAXIMUM PENETRATION DEPTHS FOR DROP POSITIONS (in)			
	σ_{max}	σ_{nom}	1	2	3	4
NLI-1/2	2.37	1.12	10.7	15.8	25.0	22.7
NLI-10/24	1.01	.650	20.2	35.9	38.0	41.0
IF-300	1.69	1.12	15.8	24.7	32.7	32.0
TN-8L	1.59	1.12	8.49	20.9	27.5	28.0

Notes:

- σ_{nom} - nominal static crush strength of the honeycomb material selected
- σ_{max} - maximum allowable crush strength that includes a 30% dynamic increase factor and a +15% static strength increase
- Maximum Penetration Depth - These are conservatively based on a crush strength equal to 85% of the nominal static crush strength (σ_{nom}) 2
- The crush strength and penetration depths are for honeycomb in an air environment provided by a stainless steel sealing can.

TABLE 4-4
IMPACT FORCE ON THE FLOOR FROM THE RACKS WHEN
IMPACTED BY CASKS WITHOUT IMPACT LIMITERS

CASK	O (in)	N (no.)	F_{max} (kips)	Floor Capacity (kips)	Margin
NLI-1/2	53.62	27	1970.	6180.	3.1
NLI-10/24	55.12	52	3800.	6180.	1.6
TF-300	51.52	35	2560.	6180.	2.4
TN-8L	42.12	30	2190.	6180.	2.8

Notes:
- o N= number of tubes hit
- o O= cask overlap upon the rack from drop Positions 3 and 4
- o The margin equals the floor capacity divided by the impact force (F_{max})

TABLE 4-5
IMPACT FORCE ON THE FLOOR FROM THE RACKS WHEN
IMPACTED BY CASKS WITH IMPACT LIMITERS

CASK	O (in)	N (no.)	F_{max} (kips)	Floor Capacity (kips)	Margin
NLI-1/2	77.625	40	2920.	6180.	2.1
NLI-10/24	77.625	72	5260.	6180.	1.2
TF-300	77.625	53	3870.	6180.	1.6
TN-8L	77.625	55	4015.	6180.	1.5

Notes:
- o N= number of tubes hit
- o O= cask overlap upon the rack from drop Positions 3 and 4
- o The margin equals the floor capacity divided by the impact force (F_{max})

TABLE 4-6
SUMMARY OF RACK IMPACT ANALYSIS

CASK	Postulated Crush Depth S_c (in)	Allowable Crush Depth (in)	Margin
NLT-1/2	66.	127.	1.9
NLT-10/24	74.	127.	1.7
IF-300	74.	127.	1.7
TN-8L	52.	127.	2.4

TABLE 4-7

CASK DROP WITH IMPACT LIMITERS

AND NO CRASH PAD

Cask	Maximum Deceleration (g)	Total Weight W_T (kips)	Maximum Impact Force F_1 (kips)	Floor Capacity (kips)	Margin
NLI-1/2	37.4	53.9	2.02×10^3	4170.	2.06
NL-10/24	30.95	205.	6.34×10^3	6180.	.97
IF-300	234.	145.	3.39×10^4	5390.	.16
TN-8L	155.	86.1	1.33×10^4	5630.	.42

Notes:

- The margin equals the floor capacity divided by the impact force (F_1).
- The floor capacity for the NL-10/24 cask was very conservatively determined.

5.0 CRITICALITY AND RADIOACTIVITY RELEASE EVALUATION METHODOLOGY

5.1 Criticality Evaluation

A series of nuclear criticality analyses of spent fuel assemblies stored in the racks was performed to determine the effect on K_{eff} of a postulated cask drop onto the loaded racks which would change the spacing of the fuel assembly lattice. Since it was not practical to define the damaged rack configuration, maximum value of K_{eff} was determined by varying the spacing from normal storage position to the crushed condition.

5.1.1 Evaluation Criteria

The evaluation criteria used was as follows: Damage to fuel and fuel storage racks due to the accidental dropping of a heavy load would not result in a configuration of the fuel such that K_{eff} is greater than 0.95 (reference 5-1).

5.1.2 Analysis Method

The analysis method followed the assumptions and methodology described in reference 5-1. KENO (reference 5-2), a computer code which uses a three-dimensional Monte Carlo method was used to determine the K_{eff} values. The KENO code is used extensively in criticality safety calculations and is used as the standard for evaluating other criticality calculation models (reference 5-3). The code was validated against critical experiments performed with material compositions and geometric arrangements applicable to the present study (reference 5-4). The code was used to establish the basis for the ANSI Standard guide for Nuclear Criticality Safety in the Storage of Fissile Materials, ANSI N 16.5-1975 (reference 5-5).

The following assumptions were used in the calculations:

a. The fuel pool is filled with new fuel of high enrichment (see table 5#1).

b. The fuel is stored in an infinite array with only axial (vertical) neutron leakage allowed.

c. The pool temperature is 40°F.

d. The neutron absorption of the fuel assembly support structure is excluded.

e. The neutron absorption of the fuel assembly grids is excluded. 2

f. The neutron absorption of the U^{234} and U^{236} present in the fuel is excluded.

g. The neutron absorber boron-10 area density is 0.04 g/cm^2.

h. The water in the fuel storage pool contains 0 boron for one case and 1800 ppm for another case.

Of the above assumptions, 'a' through 'g' provided a conservative base for the calculations (reference 5-6). Assumption 'h' is a realistic condition that is allowed when considering postulated accidents (reference 5-5).

The effects of the heavy load drop on K_{eff} was evaluated by performing sensitivity studies using different water/UO_2 volume ratios and different soluble boron concentrations. The K_{eff} versus water/UO_2 volume ratio curves for two different boron concentrations was then generated. From these curves, the minimum soluble boron concentrations required to satisfy the evaluation criteria was determined. 2

5.1.3 Analysis Results

The results of the criticality analysis for different water/UO_2 volume ratios ranging from 0 ppm to 1800 ppm boron concentrations are presented in table 5-2. The worst case k_{eff} was determined to be 0.936 at a water/UO_2 ratio of 2.0. Table 5-3 presents the nominal value of K_{eff} for the worst case and the applicable uncertainties. (The uncertainties are defined in reference 3-3). The worst case K_{eff} with uncertainties is:

$$K_{eff} \leq k_o + \Delta k_1 + \Delta k_2 + (\Delta k_3^2 + \Delta k_4^2 + \Delta k_5^2 + \Delta k_6^2 + \Delta k_7^2)^{1/2}$$

where,

k_0 = nominal calculated k

Δk_1 = transport correction

Δk_2 = method bias

Δk_3 = fuel location effect

Δk_4 = storage tube pitch tolerance effect

Δk_5 = uncertainty in methods bias (95% confidence level)

Δk_6 = neutron absorber boron-10 tolerance effect

Δk_7 = storage tube dimensional effect

5.2 Radioactivity release

A radioactivity release analysis was performed to determine the effect on dose rate of a postulated cask drop which could impact the loaded spent fuel racks, damage the fuel cladding, and release radioactive material.

5.2.1 Evaluation Criteria

The evaluation criteria used was as follows:

The doses of radioactive material that may be released from the damaged spent fuel rods would be limited to less than one-quarter of the dose guidelines specified in 10 CFR, part 100 (reference 5-1). | 2

5.2.2 Analysis Method

The analysis method for the present study followed the assumptions and methodology described in references 5-1 and 5-7. The cladding on all of the fuel rods in the impacted fuel assembly was assumed to have ruptured, thus causing a release of fission product gases which were contained in the space or gap between the fuel pellets and the cladding. The percent of radioactive iodine inventory assumed to be released was based on guidelines in Regulatory Guide 1.25 (reference 5-7). For the noble gas (including kr-85), 100 percent of the inventory was assumed to be released. The fission product gases released were conservatively assumed to be 10 standard cubic feet per fuel assembly (0.3 cubic meters) per reference 5-1. The gas-bubbles would be released to the fuel pool water where they would rise to the surface of the pool. The water would scrub out approximately 99 percent of the iodine fission products (Iodine 131-135), but was not assumed effective in reducing the quantity of noble gases released to the fuel building atmosphere. The radioactive gases were then assumed to reach the spent fuel building atmosphere. For case 1, they were assumed to be exhausted to the environment through a charcoal filter system which would further reduce the quantity of airborne radioactive iodines. This filter was assumed not to be effective in removing noble

gases such as Krypton and Xenon. For case 2, the radioactive gases were assumed to be exhausted directly to the environment. The postulated dose consequences of a heavy load drop on fuel assemblies in the spent fuel pool were determined as a multiple of a single assembly fuel element. The assumptions used in the analysis are summarized in table 5-4.

5.2.3 Analysis Results

Figure 3-2 presents the results of the analysis in the form of two curves. The number of damaged assemblies that would produce doses equal to one-quarter of 10 CFR, part 100 guidelines are plotted against the days after shutdown. The curve for case 1 assumes the use of charcoal filters, but the other curve does not. The maximum number of fuel assemblies which can possibly be damaged due to a postulated heavy load drop is conservatively assumed to be 80. Based on the refueling data from reference 5-6, the maximum number of fresh assemblies that may be discharged into the pool is 45. Thus, of the 80 postulated damaged fuel assemblies, 45 are fresh and 35 would have been in the pool for at least 180 days. Radioactivity release contribution from these 35 assemblies would be negligible. The doses for both the exclusion area and low population zone were calculated. The radioactive material released from the reactor building was assumed to be within the two hour period per reference 5-7; therefore, the dosage calculated for the exclusion area bounds the dosage for the low population zone. The maximum number of days required to satisfy the criteria is 16 for case 1 (with charcoal filters), and 50 for case 2 (without charcoal filters). These are also shown in figure 3-2. 2

TABLE 5-1. FUEL DESIGN PARAMETERS FOR THE NUCLEAR CRITICALITY EVALUATION

Rod Array	14 x 14
No. of Fuel Rods	179
No. of Water Holes	17
Rod Pitch	0.556 in.
Pellet O.D.	0.3444 in.
Clad O.D.	0.400 in.
Clad Thickness	0.0243 in.
Clad Material	Zircaloy 4
Pellet Density, % T.D.	94.5
U_{235} Loading	39 g/cm
Nominal Active Fuel Length	144 in.

TABLE 5-2. K_{eff} vs. H_2O/UO_2 RATIO

H_2O/UO_2	K_{eff}	
Volume Ratio	0 ppm Boron	1800 ppm Boron
0.0[1]	[2]	---
0.5317	0.72614 ± 0.00387	---
1.0	0.89065 ± 0.00498	---
1.8	---	0.93238 ± 0.00419
2.0	---	0.93610 ± 0.00493
2.283	1.09829 ± 0.00501	0.93320 ± 0.00405
2.5	1.09747 ± 0.00502	0.92873 ± 0.00465
2.768	1.07806 ± 0.00506	0.91646 ± 0.00505
2.998	1.04346 ± 0.00586	---
3.566[3]	0.93414 ± 0.00804	0.80552 ± 0.00488

[1] Completely crushed

[2] Less than 0.73

[3] Normal undisturbed condition

TABLE 5-3. MAXIMUM K_{eff} WITH UNCERTAINTIES

Term	Value	Method
k_o	0.93610	KENO calculation
Δk_1	0	Included in Δk_2
Δk_2	0.003	Comparison with critical experiments
Δk_3	0	No room for fuel to move
Δk_4	0	No room for storage tube to move
Δk_5	0.00963	KENO and critical experiments 95% confidence level
Δk_6	0	Analysis is based on minimum boron content with 95% confidence
Δk_7	0	Analysis based on worst case tube dimensions
K_{eff}	0.94873	

where:

$$K_{eff} \leq k_o + \Delta k_1 + \Delta k_2 + (\Delta k_3^2 + \Delta k_4^2 + \Delta k_5^2 + \Delta k_6^2 + \Delta k_7^2)^{1/2}$$

4-10 Topical Report BC-TOP-9-A, "Design of Structures for Missile Impact," revision 2, Bechtel Power Corporation, September 1974.

4-11 Hexcell catalog no. TSB122, "Design Data for the Preliminary Selection of Honeycomb Energy Absorption System," 1982.

4-12 Hexcell catalog no. TSB120, "Mechanical Properties of Hexcel Honeycomb Materials," 1982.

4-13 Timoshenko, S. P. and J. M. Gere, "Theory of Elastic Stability" 2nd Ed., Engineering Societies Monograph Series.

5-1 U.S. Regulatory Commission, "Control of Heavy Loads at Nuclear Power Plants," U.S. NRC NUREG-0612, July 1980.

5-2 "KENO IV - An Improved Monte Carlo Criticality Program," ORNL-4938, Oak Ridge National Laboratory (November 1975), L. M. Petrie and N. F. Cross.

5-3 D. C. Hunt and D. Dickinson, "Comparative Calculational Evaluation of Array Criticality Models," Nuclear Technology, vol. 30, pp. 190-214, August 1976.

5-4 A. M. Hathout, R. M. Westfall, and H. L. Dodds, "Validation of Three Cross-Section Libraries Used with the SCALE System for Criticality Safety Analysis," NUREG/CR-1917, June 1981.

5-5 "Guide for Nuclear Criticality Safety in the Storage of Fissile Materials," ANSI N16.5-1975, April 1975.

5-6 Licensing report for Prairie Island Nuclear Generating Plant units 1 and 2, Spent Fuel Storage Modification, QUAD-1-79-509, December 28, 1979.

TABLE 5-4. ASSUMPTIONS FOR RADIOACTIVITY RELEASE ANALYSIS

	Reactor Type	PWR (Prairie Island units 1 and 2)
	Power Level (mwt)	1650
0-2 hour	X/Q (exclusion area boundary) sec/M^3	0.65×10^{-3} (reference 5-8)
	Peaking factor	1.2 (reference 5-1)
	Number of assemblies in Core	121
	Pool Water Decontamination Factor	100 (for radioactive iodines) (reference 5-6)
	Filter Efficiency Percent:	
	Elemental Iodine	95% (reference 5-7)
	Organic Iodine	95%
	Cooling Time (hours)	100 or greater

6.0 CONCLUSIONS

The following conclusions can be drawn from the results of the study:

a. The doses of radioactive material that may be released as a result of a postulated spent fuel cask drop would be less than one-quarter of 10 CFR 100 guidelines (300 rem thyroid, 25 rem whole body) at the exclusion area boundary if the impacted fuel has been out of the operating reactor for at least 50 days. This conservatively takes no credit for the charcoal filters. 2

b. The damage to the fuel and fuel storage racks that may result from a postulated drop of a spent fuel cask would not result in a fuel configuration such that K_{eff} is larger than 0.95 when 1800 ppm boron concentration is maintained in the water of the spent fuel pool.

c. The damage to the spent fuel pool that may result from a postulated drop of a spent fuel cask would not result in water leakage that could uncover the fuel. In order to ensure this, an energy-absorbing pad would be used on the pool floor or an impact limiter would be used on the cask. 2

d. Any postulated damage to the spent fuel inlet line would not have any safety-related consequences. Damage to any other safety-related equipment is not credible because of the separation between the spent fuel cask and the safety-related equipment.

e. The consequences of the postulated impact of the spent fuel pool floor by the NLI-1/2 and NLI-10/24 spent fuel casks with their lower impact limiters in place will be within the acceptable limits.

f. The consequences of the postulated impact of the spent fuel pool floor by the NLI-1/2, NLI-10/24, IF-300, and TN-8L spent fuel casks with the

honeycomb crash pads described in table 3-2 on the fuel pool floor will be within the acceptable limits.

7.0 REFERENCES

1-1 NSP Purchase Order D21644MQ, dated October 4, 1983.

1-2 Amendment #48 to DPR-42 (#42 to DPR 60 May 13, 1981).

1-3 U.S. Regulatory Commission, "Control of Heavy Loads at Nuclear Power Plants," U.S.NRC NUREG-0612, July 1980.

2-1 NLI Safety Analysis Report, Docket no. 701318, Rev. 1 (NLI-1/2)

2-2 Certificate no. 9010 - rev. no. 13, Docket no. 71-9010 (NLI-1/2)

2-3 Safety Analysis Report - NLI 10/24 Spent Fuel Shiping Cask - Doc. #71-9023, rev. 4, 2/76

2-4 Certificate no. 9023 - rev. no. 4, Docket no. 71-9023 (NLI-10/24)

2-5 Safety Analysis Report. GE document no. NEDO-10084-1; Design and Analysis Report IF 300 Shipping Cask, Feb. 1973 - Docket no. 70-1220

2-6 Certificate no. 9001 - rev. no. 18, Docket no. 71-9001 (IF-300)

2-7 TN-8 and TN-9 Safety Analysis Report Volume 1, rev. 8, April 8, 1980 by Transnuclear, Inc., Docket no. 71-9015

2-8 Certificate no. 9015, rev. no. 7, Docket no. 71-9015 (TN-8L)

3-1 U.S. Regulatory Commission, "Control of Heavy Loads at Nuclear Power Plants," U.S.NRC NUREG-0612, July 1980.

3-2 Updated Safety Analysis Report for Prairie Island Nuclear Generating Plant, Rev. 0, Dec. 1981 (section 10.2.2.3).

5-7 Regulatory Guide 1.25, "Assumptions Used for Evaluating the Potential Consequences of a Fuel Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors."

5-8 B. K. Grimes letter to all power reactor licensees, dated April 14, 1978.

5-9 Prairie Island Generating plant, USAR, Section 14.