MAINE YANKEE ATOMIC POWER COMPANY •

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June 3, 1981
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U.S. NUCLEAR REGULATORY COMMISSION

United States Nuclear Regulatory Commission
Washington, D. C. 20555

Attention: Office of Nuclear Reactor Regulation
Division of Licensing
Mr. Robert A. Clark, Chief
Operating Reactors, Branch No. 3

Reference: (a) License No. DPR-36 (Docket No. 50-309)
(b) USNRC Letter to MYAPCo dated March 12, 1981
(c) USNRC Letter to MYAPCo dated April 29, 1981

Subject: Maine Yankee Fuel Storage Modification

Dear Sir:

Your letters, Reference (b) and (c) posed thirty-two questions relating to Maine Yankee's proposed spent fuel storage modifications.

Enclosed are responses to these questions. The responses are as complete as they can be at this time. In some cases, as indicated in certain responses, additional information will have to be provided at a later time.

At this point, a vendor has been selected to supply the spent fuel racks we propose to use. Design information specific to this vendor's rack design is being used to develop responses to the questions which remain open at this time.

When these responses are available, they will be submitted promptly. Also at that time, the vendor's design report will be provided.

Following submittal of the outstanding responses and vendor design report, all relevant information will be assembled and submitted in the form of a complete report on all aspects of the proposed spent fuel storage modifications. This complete report, which the staff requested during our meeting of March 3, 1981, will take some time to prepare because it requires a complete rewrite of all information relating to the proposed change, but we will submit it as soon as possible.

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U.S. Nuclear Regulatory Commission
Attn: Mr. Robert Clark

It is our understanding that Staff review of Maine Yankee's proposed spent fuel modification has been assigned a high priority for completion. We appreciate the attention the Staff is giving this matter. If we can be of assistance beyond the steps outlined above, please do not hesitate to call upon us.

Sincerely yours,

MAINE YANKEE ATOMIC POWER COMPANY

John H. Garrity

John H. Garrity, Director
Nuclear Engineering & Licensing

JHG/plb

Enclosure

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1. Question

In regard to the proposed reracking provide the following additional information:

a. With the aid of a drawing, describe the travel paths of the old and the new storage racks. Quantify the number of stored spent fuel assemblies that will be stored in the pool at the time reracking occurs.

   Identify all equipment located below or within the area of influence of a dropped load along the travel paths which is essential in attaining a safe shutdown or in mitigating the consequences of an accidental load drop.

b. With the aid of drawings describe the old and the new storage racks and their weights. Further, provide a description of the rigging that is interposed between the storage racks and the crane hook. Identify all cases where the proposed rigging does not meet the requirements set forth in Section 5.1.6(1) of NUREG-0612 "Control of Heavy Loads at Nuclear Power Plants," and explain why they are acceptable.

Response

The phasing scheme is depicted on figures 1, 2, 3, 4 and 5 (attached).

The following is a key to the information on each of these figures:

- Existing racks are designated with small letters.
- The new racks are designated with a large letter.
- The roman numeral superscript to the large letters indicate the phase in which the rack is installed.
- The number at the lower right corner of each rack indicates the number of stored fuel assemblies in that rack following completion of the indicated phase.
- The number in parenthesis in the lower right corner of each rack indicates the available storage spaces in the rack.

Figure 1 shows the pool layout as is with the existing racks. Please note that Racks X, Y and Z are not installed yet; however, rack X is on site and will be placed into service should full core discharge be necessary. (Racks X, Y, and Z would constitute the final phase of the reracking approved via amendment 11 issued in 1975).

The racks are replaced as follows: The rack to be removed is disconnected from adjacent racks after the fuel assemblies stored within it are moved to other racks. The rack is then lifted six (6) inches off the liner with the temporary crane, moved away from its location and lowered into an area which can be reached by the yard crane. The rack is then picked up and removed with the yard crane. The new racks will be installed in a similar manner by reversing the sequence.

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The phasing scheme is as follows:

Phase I (Figure 2) to be completed by October 1983:

a. Install Rack Z
b. Remove fuel from racks C, E and O.
c. Remove racks C, E and O.
d. Install new racks Z^{I}, T^{I}, M^{I}, G^{I} and C^{I}.
e. Dispose of racks C, E, O and Y.

Inventory: 650 fuel assemblies, 259 empty storage spaces and 63 emergency storage spaces.

Phase II (Figure 3) to be completed by December 1984

a. Remove fuel from racks A and N.
b. Remove racks A and N.
c. Install new racks Y^{II}, S^{II}, L^{II}, F^{II} and B^{II}.
d. Dispose of racks A and N.

Inventory: 722 fuel assemblies, 335 empty storage spaces and 63 emergency storage spaces.

Phase III (Figure (4) to be completed by January 1986

a. Remove fuel from racks F, M, G and L.
b. Remove racks F, M, G and L.
c. Install new racks X^{III}, R^{III} and K^{III}.
d. Remove fuel from racks Z, K and J.
e. Remove racks Z, K and J.
f. Install new racks W^{III}, P^{III}, J^{III}, V^{III} and U^{III}.
g. Dispose of racks F, G, J, K, L, M and Z.

Inventory: 795 fuel assemblies, 283 empty storage spaces and 63 emergency storage spaces.

Phase IV (Figure 5) to be completed by February 1987

a. Remove fuel from racks P and D.
b. Remove racks P and D.
c. Install new racks N^{IV}, O^{IV}, I^{IV}, H^{IV}, D^{IV}, E^{IV} and A^{IV}.
d. Dispose of racks D, P and X.

Inventory: 867 fuel assemblies and 509 empty storage spaces.

No equipment which is essential in attaining a safe shutdown or in mitigating the consequences of a load drop is located below the travel paths of the racks.

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The weights (dry, in air) of the older design spent fuel racks shown in Figures 1-5 are:

Rack	Size	Weight (lbs)	PAR Dwg. #
P	8 x 5	10,000	A-23640-D
X	9 x 7	15,700	A-23140-D
Y	9 x 6	13,400	A-23150-D
C	9 x 7	15,700	A-23140-D
D	10 x 7	12,500	A-17925-D
Z	10 x 7	17,200	A-19698-D
G	10 x 5	12,500	A-23130-D
F	10 x 5	12,500	A-23130-D
A	10 x 7	17,200	A-19698-D
E	10 x 7	17,200	A-23670-D
J	9 x 7	15,700	A-19697-D
K	10 x 7	17,200	A-19698-D
L	10 x 5	9,000	A-17926-D
M	10 x 5	9,000	A-17926-D
N	10 x 7	12,500	A-17925-D
O	10 x 5	9,000	A-17926-D

The column labelled "drawing number" indicates which of the attached drawings depicts the design of each of the old spent fuel racks.

The weights of the new storage racks shown on Figure 5 are:

Rack	Size	Weight
L,M,H,I,S,O,N,Y,V	8x8	19,200
D,J,P,W	7x8	16,800
A,B,C,E,F,G K,R,T,U,X,Z	6x8	14,400

Rigging that is interposed between racks and the yard crane hook is a standard 4 pick sling. The rigging meets the guidance of Section 5.1.6(1) of NUREG-0612.

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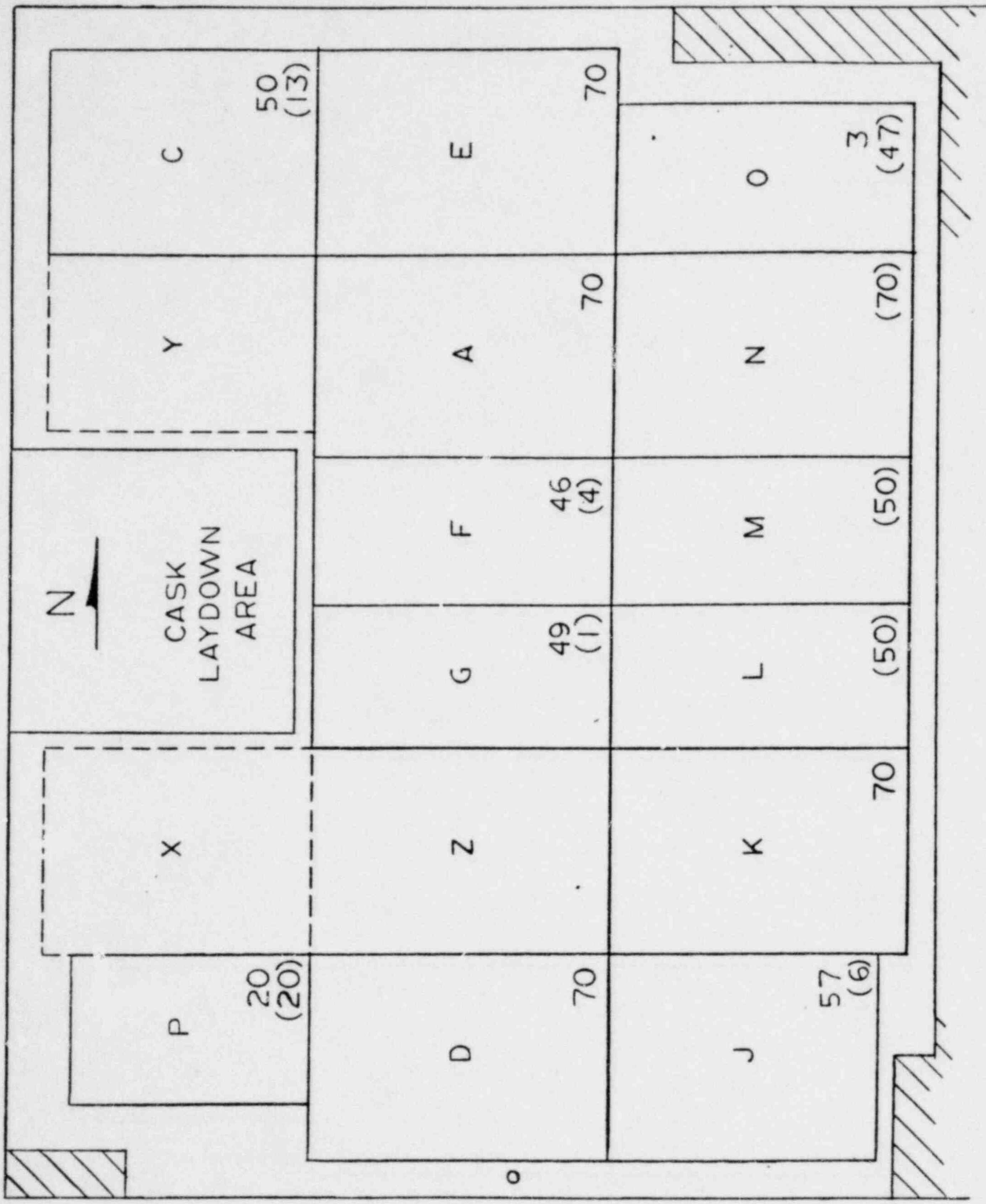
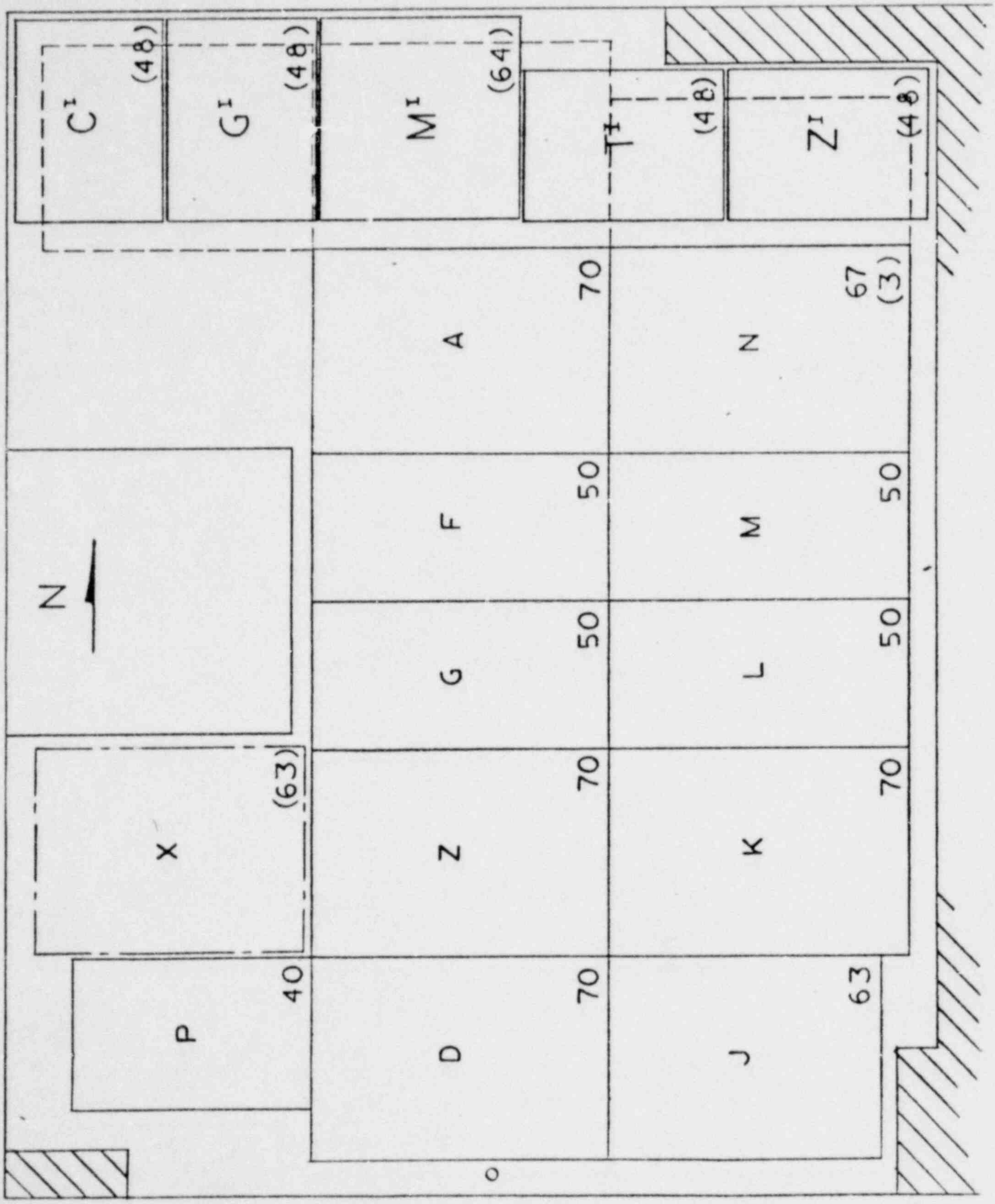
FIGURE 1



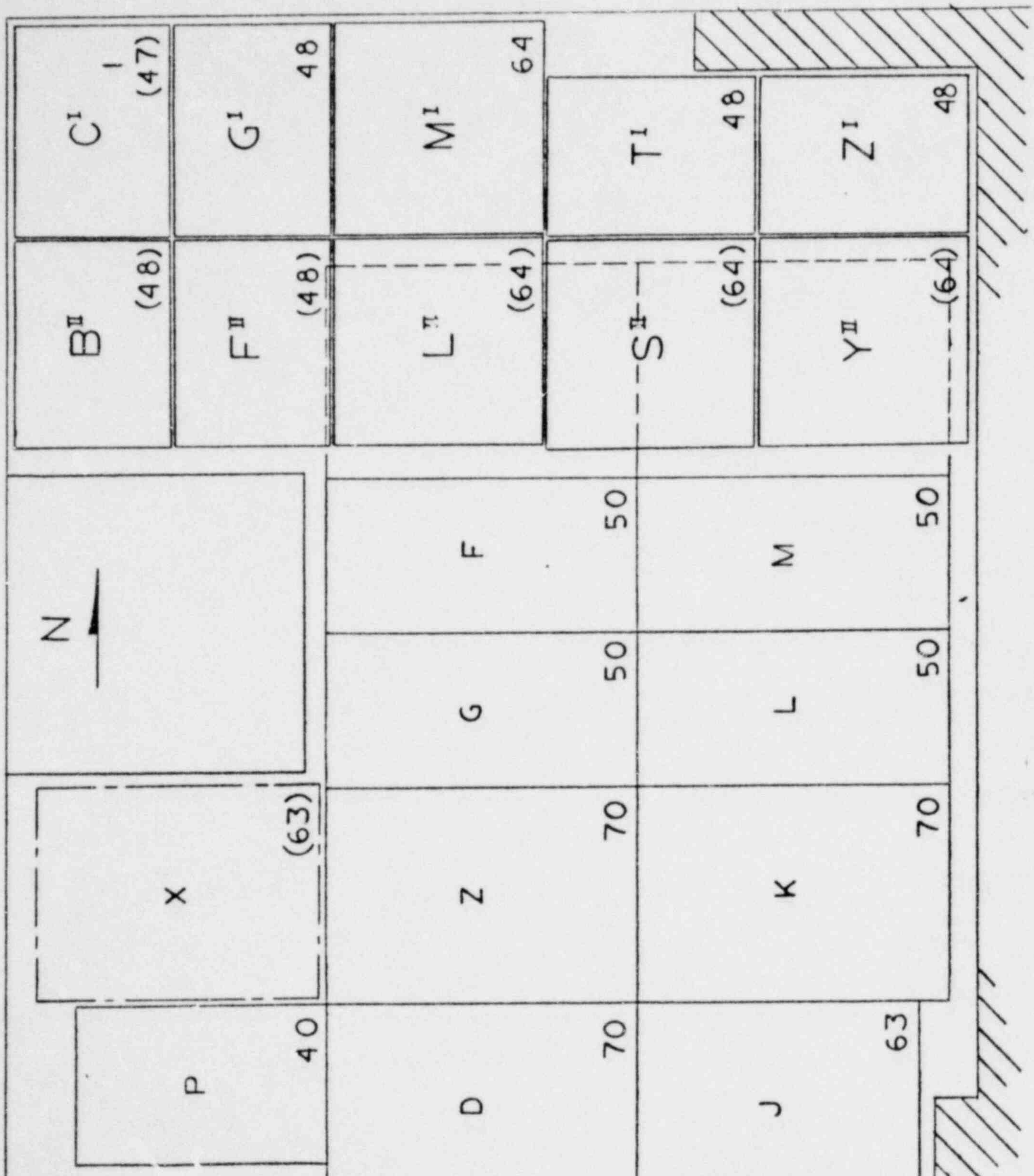
PHASE I

FIGURE 2



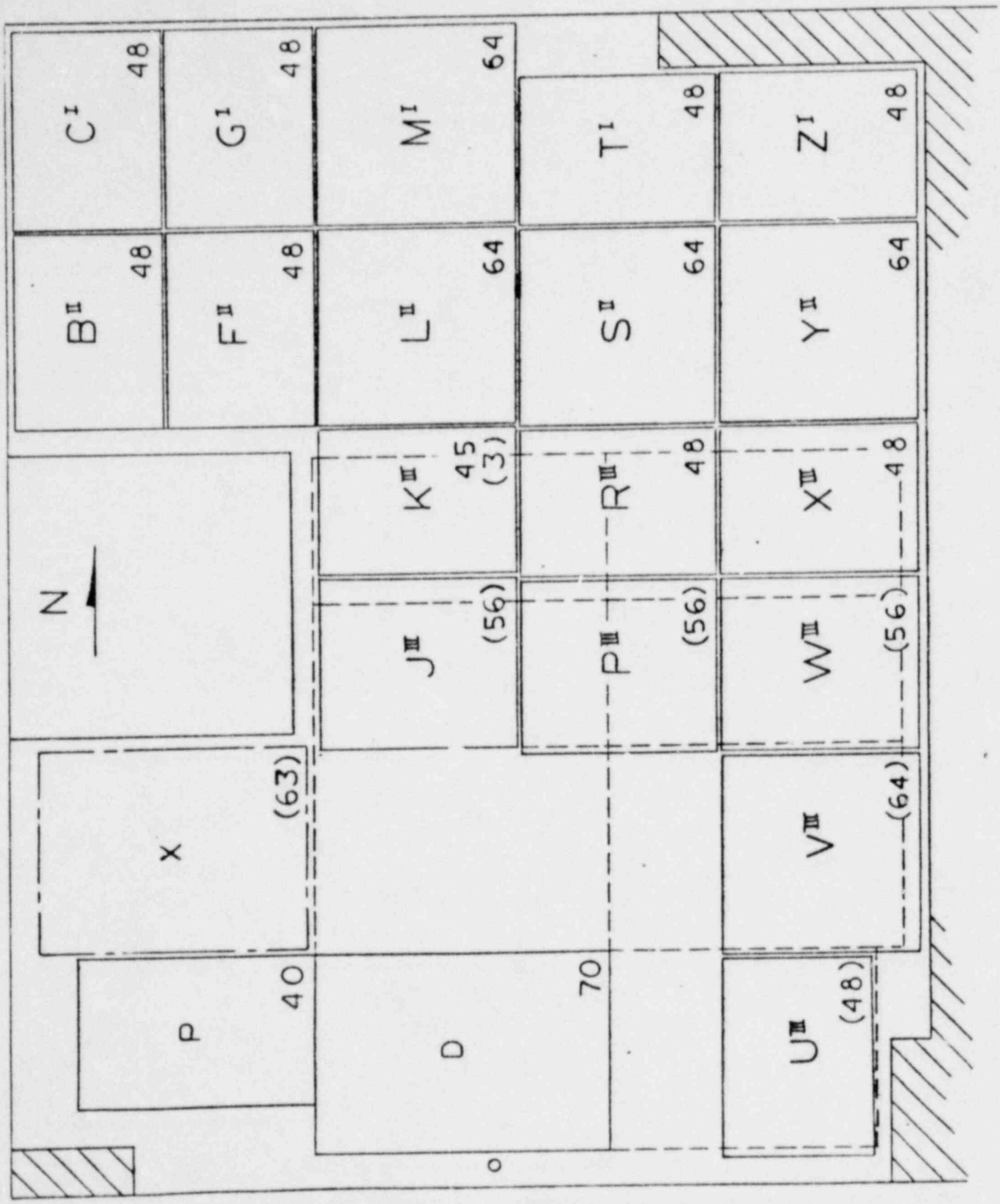
PHASE II

FIGURE 3



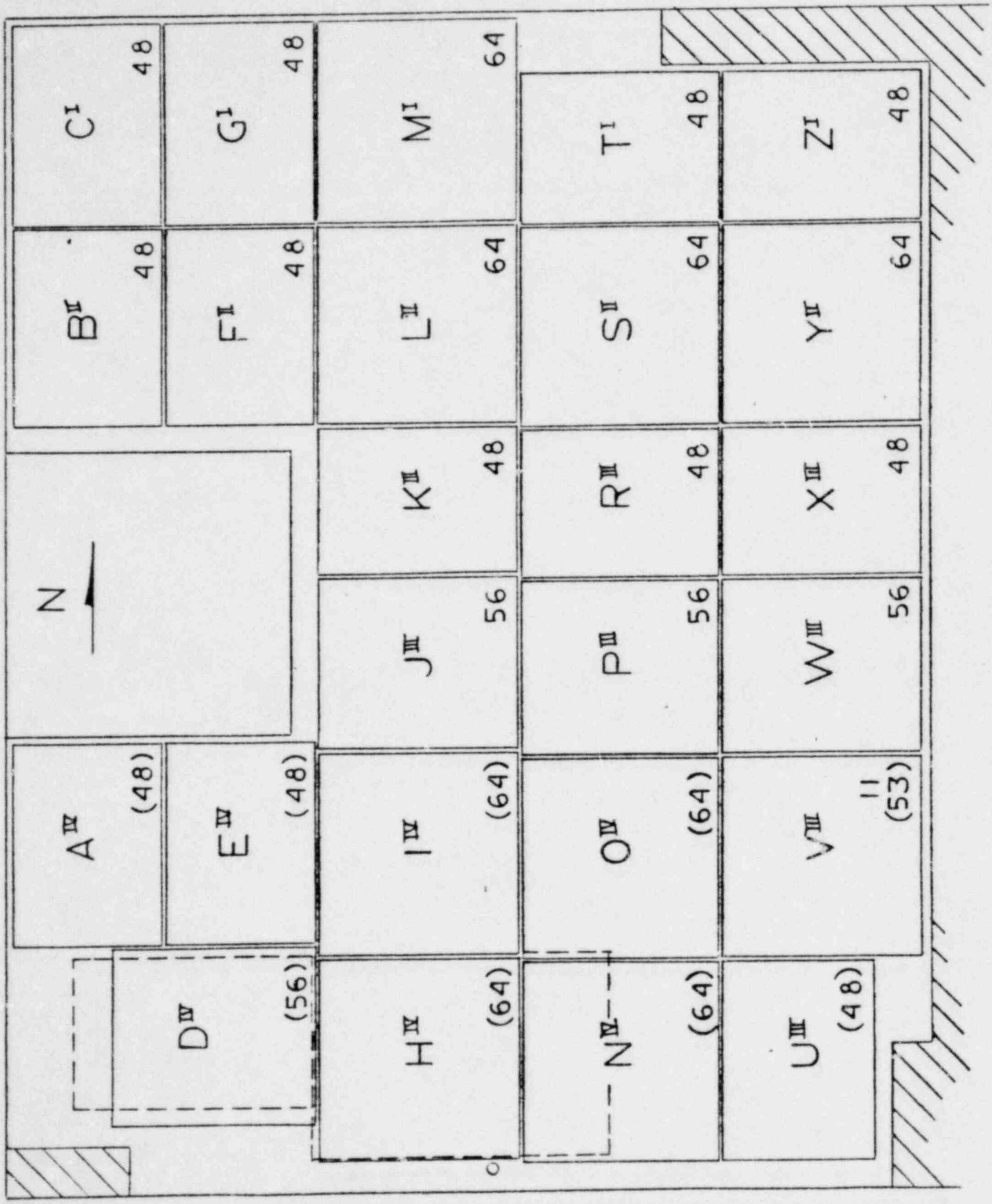
PHASE II

FIGURE 4



PHASE IV

FIGURE 5



6/3/81

2. Question

In the event the overhead crane has not been reviewed by the staff and found to be in compliance with NUREG-0554, describe and discuss what additional measures will be implemented in order to augment its reliability during this modification.

Response

Modifications to the yard crane which have already been implemented since initial installation to augment its reliability include the following items:

- A second upper limit switch of different design from the existing limit switch.
- A spring type overload switch which will shutoff power to the main hoist motor and lock both holding brakes.
- A centrifugal overspeed switch for the main hoist which sets the brakes regardless of control switch position.
- Electrical interlocks to prevent travel over stored spent fuel.

Periodically, at refueling intervals, a magnetic particle and dye penetrant test are performed on the yard crane hook along with a complete visual inspection performed by a factory representative. Maine Yankee does not have any further plans for additional measures to augment its reliability during this modification.

Description of Fuel Building Overhead Crane
Mfg., Whiting Corp.

Capacity	
Main Hoist	125 tons
Auxiliary Hoist	20 tons
Span	32 feet
Maximum Lift	
Main Hoist	65'-0"
Speed	
Main Hoist	3 FPM
Bridge Travel	50 FPM
Trolley	40 FPM

Motors	Hoist	Travel	Traverse	Aux. Hoist
Horsepower	30	10	5	30
Volts	460	460	460	460
RPM	900	1,200	1,200	900
Amperes F.L.	51	71	8.5	51

6/3/81

3. Question

In reference to the "Schedule of Change" presented in your September 29, 1980 letter, it appears that reracking operations may extend over a considerable length of time, i.e., "reracking will be implemented on a phase basis to provide an increasing capacity consistent with the normal refueling cycle." Describe and discuss the considerations that would lead you to schedule the reracking operations over more than one normal refueling cycle. The discussion should include the relative merits, in terms of reliable load handling operations, of completing the modification in one or more closely spaced refueling cycles as opposed to numerous refueling cycles.

Response

The reracking scheme is based on the estimated latest possible dates by which racks have to be replaced in order to accomodate refueling discharges and maintain full core discharge capability until all new racks are installed assuming deferral of pin storage. The phasing scheme meets the dual objectives of providing adequate spent fuel storage capability and reducing cash flow in early years to the minimum consistent with the first objective. In other words, Maine yankee does not wish to procure spent fuel racks until they are likely to be needed.

During the period when racks are being replaced, a modest amount of fuel will be consolidated to allow refinenent of techniques, tooling, and procedures. The temporary spent fuel cask rack would only be used to provide additional storage space for full core rejection should the need ari

It should also be noted that other spent fuel storage options could conceivably become available in the future. Delaying rack expenditures until they are necessary by phased reracking permits evaluation of these potential options without bias introduced by consideration of early sunk costs.

Early licensing of reracking, pin storage, and the temporary spent fuel cask area rack is prudent because it provides the capability to accomodate spent fuel in an approved manner.

Maine Yankee does not believe there is any effect, in terms of reliable load handling, attributable to phased reracking as opposed to reracking in one or more closely spaced refueling cycles.

6/3/81

4. <u>Question</u>

Responses numbered 11 and 12, dated February 15, 1980, state that an empty rack will not be moved over fuel storage racks. In this regard, using Figure 2 of the February 15, 1980 submittal, provide the following additional information: Sequentially describe the reracking operations, and the travel paths in sufficient detail so as to enable the staff to conclude all reasonably practical measures will be taken to place the least amount of stored fuel at risk in the event of a load drop.

<u>Response</u>

The reracking operation is described in the response to question 2. That response shows that reracking can be accomplished without placing any stored fuel at risk provided reracking is allowed to commence while there remains a sufficient number of empty storage cells to accomodate removal of all spent fuel from a set of racks to be moved in a single phase, selected so as to clear a free path to the spent fuel cask laydown area.

6/3/81

5. Question

Describe all of the handling tools and loads that are normally carried above the stored spent fuel.

Verify, in each case, that the maximum potential kinetic energy capable of being developed when dropped from their maximum elevation will be less than the kinetic energy of one fuel assembly and its associated handling tool when it is dropped from the height at which it is normally handled above the spent fuel storage racks.

Response

There are no handling tools normally carried above the stored spent fuel during reracking that exceed a weight of 100 lbs., thus the kinetic energy that could be developed in a load drop must be less than that associated with a fuel assembly drop.

The only heavy load that may be carried over stored spent fuel is a temporary crane used to reposition storage racks within the fuel pool. This crane weighs about 5000 lbs. and is handled by the overhead crane during installation. There is a factor of 25 on allowable crane load to temporary crane weight. Also, during installation of the temporary crane, double slings are used between the overhead crane hook and temporary crane I-Beam to provide redundancy on the lifting fixture.

The temporary crane described above has been used throughout the reracking approved via Amendment 11 to the Maine Yankee license.

6/3/81

6. Question

In regard to the short term spent fuel pool storage rack which you propose to insert and remove from the spent fuel cask laydown area in the spent fuel pool provide the following additional information:

Describe and discuss how you meet the criteria in Section 5.1 of NUREG-0612 "Control Of Heavy Loads At Nuclear Power Plants", assuming the rack is dropped.

Response

Due to the physical location of the cask laydown area in the Maine Yankee spent fuel pool installation/removal of short-term spent fuel pool storage rack can be accomplished without travel over stored spent fuel or existing spent fuel storage racks. In addition, the yard crane is provided with electrical interlocks to prevent travel over stored spent fuel.

Any rigging that is interposed between storage rack and the crane hook will meet the requirements set forth in Section 5.1.6(1) of NUREG-0612.

6/3/81

7. Question

In establishing the adequacy of local cooling in a typical row of 35 storage cells (Appendix B of the September 29, 1980 submittal) it is stated it was conservatively assumed that the storage cells contained consolidated fuel rods that have decayed for a minimum of 120 days. Since, we believe, the decay heat load of freshly off loaded fuel assemblies (3 days decay) may be as much as four times the above assumed heat load, describe and discuss the measures that will be implemented in order to provide assurance that this heat load will be so distributed throughout the pool in such a fashion that the distributed heat load in any row will not exceed that assumed in your analysis.

Response

The Thermal-hydraulic analysis of the adequacy of local cooling for the modified spent fuel racks was discussed in Attachment B of the September 29, 1980 submittal. Two cases were addressed, one dealing with stored fuel pins, the other dealing with as discharged fuel assemblies.

1. A row of 35 consolidated fuel storage bundles compacted 120 days after discharge from the core.
2. A row of 35 fuel assemblies (not consolidated) discharged after 3 days in-reactor cooling, with assembly average exposures of 44,500 MWD/MTU, a conservatively high burnup.

Results of these analyses were reported in Table 2.1 of Attachment B of the September 29, 1980 submittal. Maximum outlet temperatures for either consolidated pin bundles or freshly discharged fuel assemblies were similar. This is attributable to the higher fluid flow resistance of the consolidated pin bundles. However, no bulk boiling occurs in either case.

The heat load of each consolidated pin storage bundle was assumed to be 33.54 BTU/sec. This corresponds to 120 days cooling after infinite burnup with a power peaking factor of 1.5.

The heat load of each fuel assembly was assumed to be 64.6 BTU/sec. Using Branch Technical Position APCSB 9-2, for 3 days cooling and 3 year operating times this corresponds to equivalent burnup of 44.5 KMWD/MTU.

Administrative controls will be utilized to ensure that a minimum of 120 days cooling time is allowed before any as discharged spent fuel assembly is disassembled and its spent fuel pins stored in a consolidated array.

8. Question

a. Appendix B of the September 29, 1980 submittal states that during a full core discharge the decay time in the reactor vessel will be adjusted such that the bulk pool temperature will not exceed 154°F or the cooling capacity of the spent fuel pool cooling system. Since the decay heat load of a recently discharged full core significantly exceeds the capacity of the pool cooling system (assuming a decay time of 4 days), quantify and discuss the required adjusted full core decay times following each incremental increase in pool heat load due to previous normal discharges up to the pool's full storage capacity. The discussion should be in sufficient detail as to enable the staff to perform an independent review and evaluation.

b. Verify that the decay heat loads have been calculated in accordance with Branch Technical Position APCSB 9-2.

Response

The limitations described concerning bulk fuel pool temperature and cooling capacity of the spent fuel pool cooling system were taken from FSAR Section 9.8.1. These limitations have been adhered to because they have been previously reviewed and approved, not because they conform to any safety or regulatory requirement.

The design heat load of the spent fuel pool cooling system is 22×10^6 BTU/hr. or 6.44 MWT. This heat load can be accomodated by the spent fuel pool cooling system without exceeding a steady state bulk spent fuel pool temperature of 154°F under FSAR design conditions.

The required decay times for full core discharge have been calculated, assuming steady state conditions, design heat removal capability, and without credit for heat transport through the spent fuel pool walls and floor, by determining the decay time necessary to reduce the decay heat of the full core to be rejected to less than the difference between the design heat load and the heat load supplied by stored spent fuel already in the spent fuel pool prior to full core rejection.

The decay heat load of a full core discharge after 4 days of in-reactor cooling does exceed the design heat load of the pool. Typically, however, it requires 7 days after shutdown to remove the reactor head and prepare to move fuel and 30 minutes to move a fuel assembly from the reactor to the spent fuel pool. Hence, it would take approximately 10 days after shutdown to completely discharge a full core to the pool under ideal conditions.

Table 1 provides the decay times required before all 217 fuel assemblies of a full core discharge could be placed in the spent fuel pool without exceeding the design heat load. Two conditions are provided, present pool loading and projected loading at end of the plant life. The end-of-plant life loading assumes all storage locations to be filled with consolidated pin bundles except the 217 required for the freshly discharged fuel. Average core operating cycle length was assumed to be 13 months inclusive of outage time. Core power was assumed to be 2630 MWt. All fuel already in the pool was assumed to have operating histories of three years at 2630 MWt. The discharged full core was assumed to consist of 1/3 with 3 cycles operation, 1/3 with 2 cycles and 1/3 with 1 cycle.

Administrative controls will be utilized to ensure the design heat load is not exceeded. The technique for determining required decay times for contemplated discharge of spent fuel illustrated in Table 8.1 will be utilized.

Table 8.1
Required Decay Times for Full Core Discharge
Without Exceeding Design Heat Load

Cycle 6 Reload (Includes spent fuel discharged through end of Cycle 5)

A. Design Heat Load =	6.44 MWT
B. Pool Heat Load Prior to Discharge =	0.48 MWT
C. Available Heat Load for Discharge =	5.96 MWT
D. Required Cooling Period for Full Core Heat to Decrease Below (C) =	13 days

End-of-Pool Life (Cycle 30 Discharge)

A. Design Heat Load =	6.44 MWT
B. Pool Heat Load Prior to Discharge =	1.01 MWT
C. Available Heat Load for Discharge =	5.43 MWT
D. Required Cooling Period for Full Core Decay Heat to Decrease Below (C) =	17 days

All heat loads used in pool cooling analyses have been calculated in accordance with Branch Technical Position APCSB 9-2.

6/3/81

9. Question

Assuming the spent fuel pool contains the maximum possible decay heat load and the pool's bulk water temperature is at its maximum operational value, provide the time interval before bulk boiling occurs under the following conditions:

a. one of the two spent fuel pool cooling water pumps fails, and

b. all external pool cooling is lost.

Response

The maximum possible spent fuel pool heat load will be administratively limited to the FSAR design heat load as described in the response to question 8.

Loss of one Spent Fuel Pool cooling pump will not result in bulk boiling at the design heat load. Pool bulk temperatures will stabilize below 190°F.

Since the fuel pool heat load has not been increased beyond the FSAR design spent fuel pool heat load, the 7.8 hours time to pool bulk boiling reported in Section 2.3 of the attachment to the September 18, 1979 modified spent fuel pin storage submittal (WMY 79-97) remains applicable.

6/3/81

10. Question

For both the component cooling water system and the spent fuel pool cooling systems, identify all deviations from the positions set forth in Regulatory Guides 1.13, 1.26 and 1.29 and demonstrate the acceptability of the deviations.

Response

The proposed modification will require no design changes to the component cooling or spent fuel pool cooling systems. The design heat load and spent fuel pool cooling and heat removal system design and performance remain as described in the FSAR. All have been previously reviewed and approved.

The proposed design of the new spent fuel storage racks will meet all requirements of Regulatory Guides 1.13, 1.26, and 1.29.

6/3/81

11. Question

Describe and discuss all modifications that have been made to the spent fuel pool cooling system that could possibly alter its capacity or reliability since the operating license was granted.

Response

The only modification to the spent fuel pool cooling system that has been performed since an operating license was granted was a minor piping relocation that provided improved surface skimming operations for pool purification. This change did not alter the pool's cooling capacity or reliability. The previous review and approval of this system remains applicable.

6/3/81

12. Question

Assuming that pool boiling were to occur what would be the maximum possible boil off rate? Identify and quantify the makeup rate capability of all sources of makeup water which complies with Regulatory Guide 1.13, "Spent Fuel Storage Facility Design Basis" and Regulatory Guide 1.26 "Quality Group Classifications and Standards for Water-Steam-and Radio-Waste-Containing Components of Nuclear Power Plants". Also, indicate the required length of time before makeup water could be made available from each of these sources.

Response

Assuming no credit for conduction losses from the pool water through the pool walls and floor, the maximum boil off rate would be 50 GPM at the design heat load.

Normal or routine makeup to the spent fuel pool is accomplished via the Chemical and Volume Control System (CVCS). The operator has the option of utilizing any of the following methods:

a) blended makeup of demineralized water and concentrated boric acid

b) batch makeup of demineralized water and/or concentrated boric acid

c) batch makeup from the Refueling Water Storage Tank

The valve lineups required to perform any of the above tasks take less than 15 minutes. All of the above makeup options can provide greater than 150 gpm flow into the spent fuel pool.

Additionally, there are at least three (3) primary grade water hose connections in the vicinity of the spent fuel pool. Each of these connections can provide approximately 20 gpm makeup flow to the spent fuel pool via hoses. The combined makeup from this source would be in excess of 60 gpm. Again, this option could be implemented in less than 15 minutes.

In an emergency situation, where normal makeup means were not available, makeup from the fire main system is also available. Using one or more fire hoses, makeup rates in excess of 150 gpm could be accomplished. This makeup flow could be established in less than 20 minutes.

This response applies equally to the spent fuel pool with or without increased spent fuel storage capability because the design heat load remains as described in the FSAR as was discussed in the response to question 8.

6/3/81

13. Question

Indicate if these proposed modifications conform with NRC position on spent fuel pool modification entitled "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", issued on april 14, 1978 and later amended on January 18, 1979. If not, identify and justify the deviations.

Response

The proposed modification for the spent fuel storage racks will meet the NRC staff positions described in "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications."

14. Question

Provide sufficient details (discussion, sketches and schematics) of the racks, rack base supporting structures, racks arrangement in the pool, the spent fuel pool, and all gaps (clearance and expansion) of the rack structure and fuel bundles.

Response

This information is vendor design specific and will be provided later.

6/3/81

15. Question

Provide a description of all items (weights, heights, and kinetic energy) which may be moved over the spent fuel assemblies and the pool floor liner. State which of these items is the critical one for the fuel pool and for the fuel pool liner.

Response

A fuel bundle-handling fixture drop on the fuel pool and liner results in a kinetic energy of 31 ft-kips. In order to exceed this, an item dropped from the fuel handling crane must exceed a weight of 650 lbs. There are no items which exceed 100 lbs that normally pass over the liner from crane height.

6/3/81

16. Question

Describe the modifications which are required to the present racks. Explain in detail the load path along which all postulated forces are transmitted to the spent fuel pool structures.

Response

This question is no longer applicable. The currently used racks will be replaced with racks of a new design. The new design is described in detail in the response to Question 14.

6/3/81

17. Question

Verify that the increased compressive loads on the plastic insulator have been considered and that no degradation occurs which could decrease its insulating integrity.

Response

This question is no longer applicable. The currently used racks will be replaced with racks of a new design. The material used in the construction of the new racks is stainless steel which will not require plastic insulators because it is compatible with the spent fuel pool liner material.

6/3/81

18. Question

Indicate how the increased loads on the slab have been considered. The increased loads should include all loads resulting from the racks and loads generated from heavy drop accident.

Response

The increased loads transmitted by the rack legs to the pool floor are computed for each time step of the analysis. These loads are used to determine the local bearing stress, punching shear stress and the overall floor load on the reinforced concrete floor. Since the pool floor is a 6-foot thick concrete slab resting on bedrock, the increased loads produced by the rack weight and consolidated bundles are well within the load carrying capacity of the slab.

The allowable stresses for bearing and shear are defined by Sections 10.14 and 11.10 of American Concrete Institute (ACI 318-63) which is the code originally used in the pool design per the FSAR. The bearing and punching shear stresses are determined using the maximum impact foot force from the time history analysis.

The overall floor load is obtained by determining the forces transmitted by the rack feet for one rack and calculating a total for all racks by SRSS due to the random occurrence of the impact forces during a seismic event.

6/3/81

19. Question

Provide the load combinations, the acceptable criteria and the reference standards or papers used in the design of the spent fuel racks. Also, provide a discussion on the fabrication techniques (include welding) that will be used during the construction of the racks.

Response

The response to this question is vendor design specific and will be provided later.

6/3/81

20. Question

For the accident drop of any heavy object (including the fuel assembly and shipping cask) over the fuel pool provide the following:

a. The assumptions, method of analysis, ductility ratios and allowable stresses or strains used in the analysis to insure that the leak-tightness of the fuel pool liner is maintained.

b. The code and load combination used in the design of fuel pool liner and the slabs.

c. The assumptions, method of analysis, the calculated and allowable stresses for the concrete slab which is affected by this accident.

Also, discuss the effects and consequences of a direct drop over the liner versus a direct drop over the fuel pool racks (straight, inclined and through). State which is more critical in the design of the liner plate and the concrete slab.

Response

There are no current plans to utilize a spent fuel shipping cask in the Maine Yankee spent fuel pool for the foreseeable future. When the option to ship spent fuel off-site again becomes available, an evaluation will be made for the cask drop accident on the liner and the concrete slab. Until this evaluation is completed, spent fuel shipping casks will not be lifted over the fuel pool.

The limiting drop accident is the drop of a consolidated fuel bundle from 18" above the racks directly onto the stainless steel liner. (This is the maximum height the crane can raise a fuel bundle with the spent fuel pool handling tool due to physical limitations.)

An evaluation of the consequences of a bundle drop on the liner due to the increased weight of a pin storage bundle was conservatively performed utilizing the modified NRC Formula for penetration of concrete by a missile.

Assuming the bundle to drop at an angle such that an edge of the lower end fitting contacts the liner, both an old or new weight bundle penetrates the liner. The consolidated bundle penetrates the concrete by only about four inches more than the existing bundle for which the spent fuel pool was previously licensed.

In addition, this calculation is very conservative since:

1. The bundle is considered infinitely stiff (no absorbtion of energy by bundle).

2. Line of impact coincides with the line at the c.g. of the bundle and edge which is penetrating into the concrete at all times during impact.

3. Calculated kinetic energy is based on a vertical drop.

6/3/81

4. The contact surface is assumed constant thru the penetration when in fact it increases significantly once initial penetration is made. This local penetration has no overall effect on the 6-foot thick slab (Ductility Ratio). This slight increase in penetration depth caused by the increased bundle weight has no effect on leakage from the pool since any seepage would be dependent primarily on the porosity of the concrete. Any seepage out of the pool is well within the makeup capability of the spent fuel cooling system.

21. Question

Provide step by step detailed discussion on how the seismic effects on the racks have been considered. Provide, also, a discussion on the sliding and stability of the racks, the friction forces due to the sliding, the floor response spectra or time history, the damping values and applicability of Regulatory Guide 1.92.

Response

The response to this question is vendor design specific and will be provided later.

6/3/81

22. Question

Due to the gaps between assemblies and the wall of the guide tubes, additional loads will be generated by the impact of the fuel assemblies during a postulated seismic excitation and sliding. Provide the justifications and the numerical values of these dynamic magnification factors due to the impact. Provide, also, sufficient details describing the gaps, the guide tubes and the boundary conditions of the fuel bundle inside the guide tubes.

Response

See response to question No. 21.

6/3/81

23. Question

With regard to the issue of heavy drop accident (straight, through, and inclined) over the spent fuel racks, provide the following:

a. Sketches, schematics and discussions regarding the shape of the impact area.

b. Detail justification on why there will be no geometric distortion of the racks and how the structural criteria established for this case can be met.

Response

The response to this question is vendor design specific and will be provided later.

6/7/81

24. Question

Discuss the method used to account for the sloshing water on the fuel pool and fuel racks.

Response

1. Loads on Fuel Pool walls are not affected by this proposed change.

2. The racks are located below any free surface wave activity. It is concluded that the rack elevation compared to the pool water elevation is such that rigid body motion rather than sloshing loads is applicable to the rack design.

6/3/81

25. Question

Provide a detailed discussion of the analysis used to calculate the stresses due to the fuel handling uplift accident, thermal loads, dead loads and friction loads. The model used and the assumptions made should also be provided.

Response

The response to this question is vendor design specific and will be provided later.

6/3/81

26. Question

Due to thermal or seismic movement, friction forces will be present between the racks and the fuel pool liner. Discuss how these friction forces have been incorporated in the analysis. Provide also the numerical values with justifications of the coefficient of friction used in the analysis.

Response

The response to this question is vendor design specific and will be provided later.

6/3/81

27. Question

Provide design details to allow us to evaluate the compatibility of the materials of construction and the poison material in the redesigned racks with respect to galvanic and other corrosion processes.

Response

The response to this question is vendor design specific and will be provided later.

6/3/81

28. Question

If venting of the "containment pocket" for the poison material is not provided, explain the method used to mitigate the effects of gas buildup.

Response

The response to this question is vendor design specific and will be provided later.

6/3/81

1. Question

The text (page 3 of Attachment B of the September, 1980 submittal) asserts that all mechanical uncertainties were treated by assuming "worst case" conditions. But Figure 3.1 indicates that the effective multiplication factor is approximately 0.929 at the 10.5 inch nominal spacing. This seems inconsistent. Please provide a listing of mechanical parameters with the nominal and "worst case" value of each.

Response

Pessimistic values of important mechanical parameters have been used to calculate multiplication factors which conservatively represent the 10.5 inch center-to-center "nominal" configuration.

The nominal 10.5-inch center-to-center spacing was represented in the calculational model by a 10.45" center-center spacing, resulting in an infinite multiplication factor of 0.929.

Other mechanical parameters of importance in the fuel storage rack calculation are the thickness and length of the Poison plate and the thickness of the water gap in the flux trap.

The values used in the calculational model are compared to the nominal mechanical parameter values in Table 1.

Table 1
Mechanical Parameters (inches)

	Nominal	Utilized
Poison Thickness (Perpendicular to fuel assembly face in horizontal plane)	0.085	0.084*
Poison Length (Parallel to fuel assembly face in horizontal plane)	8.0±0.03	7.97
Water Gap in Flux Trap (Perpendicular to fuel assembly face in horizontal plane)	1.12±0.08	0.99

*Minimum in Design Specification

This response will be revised to take into account vendor design specifics at a later time.

6/3/81

2. Question

The conclusion (page 3 of Attachment B of the September, 1980 submittal) that compacted assemblies are less reactive than the design lattice was based on calculations at a 12.0 inch pitch. Provide arguments (or calculational results) to show that this conclusion is also valid at 10.5 inch pitch. Also explain the fact that optimally moderated compact assemblies have a smaller effective multiplication factor that the design assembly (0.760 vs 0.773 in the September, 1979 submittal).

Response

The effect of compaction, a decrease in the assembly water to metal ratio, results in a decrease in reactivity because the stored pin array is more undermoderated than the design assembly. Assembly calculations have shown a reactivity differential of 16% $\Delta k/k$ between a compacted 4.1 w/o U-235 stored pin assembly and a 4.1 w/o U-235 fuel assembly. This reactivity difference assures the fuel assembly is bounding with respect to reactivity for fuel storage rack calculations.

The differential in multiplication factor between the optimally moderated stored pin configuration and the design assembly (0.760 vs. 0.773) is a result of two factors.

First, stainless steel is the material used for structural elements of the pin storage cages. This stainless steel is a stronger neutron absorber than the zircaloy & structural elements utilized in the fuel assembly design. Also, the structural configuration of fuel assemblies creates water gaps not present in stored pin arrays.

Second, the multiplication factor of 0.773 representing currently used spent fuel racks and design fuel assemblies was taken from design calculations for those racks. The multiplication factor of 0.760 representing stored pin assemblies in current racks utilized as built data for the current racks.

3. Question

The implication in the September, 1980 submittal is that the reracking will take place in stages so that some portions of the racks will have 10.5 inch spacing while others have 12.0 inch spacing. Comment on the criticality impact of interfaces between two such portions.

Response

The mechanical designs of the two fuel storage racks, 10.5-inch and 12.0-inch spacings, are equivalent. The 10.5-inch spacing is a more reactive configuration than the 12.0-inch spacing and constitutes the limiting condition. From a criticality standpoint, the interface between 10.5-inch and 12.0-inch racks will be less reactive than the nominal 10.5-inch racks.

The reracking design is based on consideration of an infinite array of 10.5 inch center-to-center racks. Insertion of the considerably less reactive 12.0 inch center-to-center racks into this array results in a reactivity decrease, thus there is no impact on criticality resulting from juxtaposition of racks of differing center-to-center spacing.

6/3/81

4. Question

Describe the samples and instrument readings and the frequency of measurement that will be performed to monitor the water purity and need for spent fuel pit cleanup system demineralizer resin and filter replacement. State the chemical and radiochemical limits to be used in monitoring the spent fuel pool water and initiating corrective action. Provide the basis for establishing these limits. Your response should consider variables such as: boron, gross gamma and iodine activity, demineralizer and/or filter differential pressure, demineralizer contaimination factor, pH, and crud level.

Response

The spent fuel pool will be analyzed at least monthly for the following parameters whenever reracking or pin storage operations are taking place:

PH
Conductivity
Boron Concentration
Ammonia
Sodium ions
Chloride ions
Gross gamma activity
Specific Radionuclides (upstream and downstream of the demineralizer)

Boron concentration is maintained above the refueling concentration.

Chloride ion concentration is maintained less than 0.1 ppm.

Gross gamma activity is maintained less than 1.0 E-2 uci/cc.

The decontamination for (D.F.) across the demineralizer is maintained greater than 1.0. If any of the latter three limits are exceeded the spent fuel pool demineralizer is normally considered exhausted and a fresh change of resin is installed and demineralizer returned to service.

The pre and post filters normally operate with a differential pressure range of 3 to 15 psi. Filters are replaced when the differential pressure exceeds 20 psi.

These limits are based on previous operating experience with primary consideration for the plant ALARA program.

6/3/81