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ENCLOSURE

Enclosure 1

Supplement 1, December 1977, to "Design Report & Safety Evaluation for Replacement of Spent Fuel Pool Storage Racks, August 1977".

Enclosure 2

Response to 10/25/77 NRC Request for Additional Info. on Monticello Replacement of Spent Fuel Pool Storage Racks.

PLANT NAME: Monticello
RJL 12/13/77

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NORTHERN STATES POWER COMPANY

MINNEAPOLIS, MINNESOTA 55401

December 8, 1977

Mr Victor Stello, Director
Division of Operating Reactors
c/o Distribution Services Branch, DDC, ADM
U S Nuclear Regulatory Commission
Washington, DC 20555

Dear Mr Stello:

MONTICELLO NUCLEAR GENERATING PLANT
Docket No. 50-263 License No. DPR-22

Replacement of Spent Fuel Pool Storage
Racks - Supplement 1

On August 17, 1977, we submitted a document entitled, "Design Report and Safety Evaluation for Replacement of Spent Fuel Pool Storage Racks, August 1977". Attached is Supplement 1 to that report which includes replacements for the cover sheet and Pages 7, 22, 23 and 25 and new Pages 25a and 25b. This supplement provides additional seismic information.

Also attached are the written responses to the related additional information requested by our NRC Project Manager during an October 25, 1977 telephone conversation.

Yours very truly,

L.O. Mayer

L O Mayer, PE
Manager of Nuclear Support Services

LOM/MHV/deh

cc: J G Keppler
G Charnoff
MPCA
Attn: J W Ferman

Attachment

773470050

Enclosure 1

Supplement 1, December 1977, to "Design Report and Safety Evaluation for Replacement of Spent Fuel Pool Storage Racks, August 1977".

Instructions for Filing Supplement 1

1. Remove the following pages from the August 1977 report:

   cover sheet
   7
   22
   23
   25

2. Insert the following attached pages into the August 1977 report (pages are identified as Supplement 1, December 1977):

   cover sheet
   7
   22
   23
   25
   25a
   25b

3. The pages removed in step 1, above, may either be discarded or attached at the end of the August 1977 report for future reference. If the latter option is used, mark each of the old pages conspicuously with the word "SUPERCEDED".

MONTICELLO NUCLEAR GENERATING PLANT

Docket No. 50-263 License No. DPR-22

August 1977

DESIGN REPORT AND SAFETY EVALUATION

FOR

REPLACEMENT OF SPENT FUEL POOL STORAGE RACKS

Incorporating:

Supplement 1
December, 1977

3.2 Fuel Storage System Construction

The fuel storage module is a fabricated stainless steel structure composed of fuel storage tubes, made by forming an outer tube and an inner tube of 304 stainless steel with an inner core of Boral* into a single fabricated tube. The outer and inner tubes are welded together after being sized to the required dimensional tolerances by a patented process. The completed storage tubes are fastened together to form a 13x13 storage module. Each 13x13 module is approximately 7 feet square and 14 feet high and provides storage space for 169 BWR fuel assemblies.

Cylindrical columns, 8-inches long are welded to the underside of the module base assembly. The columns transfer the module forces to the fuel storage pool slab and raise the module above the floor of the fuel pool sufficiently to permit natural circulation of cooling water to flow to the modules.

4.0 SAFETY EVALUATION

4.1 Criticality Analysis

4.1.1 The Principal Analytical Model

The criticality analysis calculations were performed with the MERIT computer program. The MERIT program is a Monte Carlo program which solves the neutron transport equation as an eigenvalue or a fixed source problem including the neutron shielding problem. This program is especially written for the analysis of fuel lattices in thermal nuclear reactors. Geometries with up to three space dimensions and neutron energies between 0 and 10 MeV can be handled. The MERIT program uses cross sections processed from the ENDF/B-IV library tapes.

4.1.2 The Model for Verification

The qualification of the MERIT program rests upon extensive qualification studies including Cross Section Evaluation Work Group (CSEWG) thermal reactor benchmarks (TRX-1, -2, -3, -4) the B&W UO_2 and PuO_2 criticals, Jersey Central experiments, CSEWG fast reactor benchmarks (GODIVA, JEZEBEL), the KRITZ experiments, and in addition, comparison with alternate calculational methods. Boron was used as solute in the moderator in the B&W UO_2 and B&W PuO_2 criticals, and as a solid control curtain in the Jersey Central experiments. The MERIT qualification program has established a bias of .005 $\pm$ .002 (1σ) Δk with respect to the above critical experiments. Therefore, MERIT underpredicts k_{eff} by 0.5 percent Δk.

* Product of Brooks & Perkins, Inc. Consisting of a layer of B_4C-Al matrix bonded between two layers of aluminum.

4.3.1.9 The base plate of each storage module is raised above the floor of the pool sufficiently to permit natural circulation of cooling water flow to the modules. Analysis has confirmed that frictional forces between module support and the floor and the low seismic overturning moment of the racks make them stable **under all conditions** of storage.

4.3.2 Analytical Methods

Appropriate modeling of the fuel storage module was developed for each structural component and mass values assigned over the height in eleven mass nodes. The modules were combined into an idealized 8-module array and the pool wall was included to determine hydrodynamic mass effects. The modules were analyzed as a cantilever beam attached to a rigid base, using DYSEA, a GE-developed version (qualified level 2) of SAP-IV modified to derive loads in a water filled rectangular pool. These loads were derived for the horizontal and vertical accelerations specified in the General Electric BWR Systems Department seismic criteria document and were compared to the allowable stresses in the reference documents. The analysis indicates that the derived loads do not overstress the module; thus, it can be concluded that the modules are not overstressed for the Monticello application since the Monticello accelerations at the fuel pool elevation are 0.2g (SSE) and the analysis was done for 3g (SSE).

Monticello accelerations are those in Ref. 4.3.1.4, Bechtel, Inc. for the Monticello building fuel pool elevations and those shown herein in Figure 4.3-2 for the pool floor elevation.

Generic accelerations are those in GE Document 384HA137, Rev. 1, "BWR-6 Seismic Design Specification".

Table 4.3-1 compares the forces for the GE specified seismic accelerations at the fixed-base module frequency of 12.17 hz with those for Monticello.

4.3.3 Discussions of Results

4.3.3.1 The natural frequencies of the 13x13 module were calculated by accounting for the stiffness of the modules and support columns and the hydrodynamic effect of the surrounding fluid. These natural frequencies were found to be greater than 8.0 hz in the horizontal direction and 35.0 hz in the vertical direction. Frequencies of 5.5 hz and 17.0 hz were used to obtain the spectral accelerations used in the force analysis which produce conservative values relative to the higher natural frequencies.

4.3.3.2 Maximum displacement at the top of the modules for the X direction or the Y direction (the modules are symmetrical) is 0.07 inch.

4.3.3.2 (Cont'd)

Nominal spacing between modules is 2-in. so no interaction between modules as a consequence of SSE is considered.

4.3.3.3 The only applied loads to the module are the seismic loads. These were calculated to be at the top of the fuel support members (bottom of tube). The loads in the X, Y and Z direction occur simultaneously. Since the OBE loads are ∿90% of SSE loads and the OBE stress allowables (with the exception of the buckling allowables) are 50% of SSE allowables, OBE is limiting.

4.3.3.4 Thermal stresses were calculated and found to be insignificant.

4.3.3.5 The fuel pool floor loading was re-analyzed by the plant architect-engineer and found to be acceptable per 4.3.1.4.

4.3.3.6 The eleven-node module with fixed base was modified and analysis was performed of the module plus its support structure. Since the response of the module and column support system is primarily rigid body motion, adequate representation of the system can be made by a 2-node lumped-mass model. The lumped mass at the top was chosen to preserve the base shear force of the first node and the height of the model was selected to preserve the over-turning moment at the base for both first node and rigid body motion. The stiffness was selected to preserve the fundamental frequency of the module and the support columns.

The critical locations for maximum compression and shear stresses are at the base of the module in the areas near the support columns. Figure 4.3.1 shows the path of the shear forces from the fuel elements to the support columns. Based upon this structural behavior, shear stresses in the fittings, fuel support plates and bottom tube elements were developed and compared to the allowable stresses (Table 4.3-3). None of the allowables are exceeded for either OBE or SSE conditions.

The mechanism for transferring shear forces to the pool slab is through friction resistance provided by the normal force due to the submerged weight of the module through its support columns resting on the pool floor liner. A minimum value of 0.31 for the coefficient of sliding friction for stainless steel to stainless steel was assumed in the analysis. This value has been verified by recent tests of stainless steel materials. (Ref: Rabinowicz,Ernest, "Friction Coefficient Value For a High Density Fuel Storage System," report to General Electric Co., Nuclear Energy Programs Operations, 20 October, 1977). A value of 0.31 is sufficient to ensure that sliding does not occur for earthquake motions corresponding to the OBE and SSE and provides a factor of safety for sliding and over-turning greater than 1.5 and 1.1 for the OBE and SSE, respectively. An additional non-linear analysis for sliding was performed to determine relative displacements if the coefficient of friction were less than 0.31. These results are listed in Table 4.3-4.

The seismic horizontal floor time history and response spectra are those developed in 4.3.1.4 and are shown in Figure 4.3-2.

Table 4.3-2
(Deleted)

TABLE 4.3-3

Comparison of Maximum Shear (psi), Calculated

To Allowable

	OBE		SSE	
	Calculated	Allowable	Calculated	Allowable
Combined Fitting -				
- Normal Stress	1,960	16,500	2,160	33,000
- Shear Stress	1,125	11,000	1,480	22,000
Tube Local Shear at Bottom	496	11,000	650	22,000
Support Plate Weld Shear	100	11,000	130	22,000

TABLE 4.3-4

Sliding Analysis Displacements

Coefficient of Friction	Maximum Non-linear Sliding Displacements (in.)
0.10	0.49
0.15	0.23
0.20	0.13
0.25	0
0.30	0

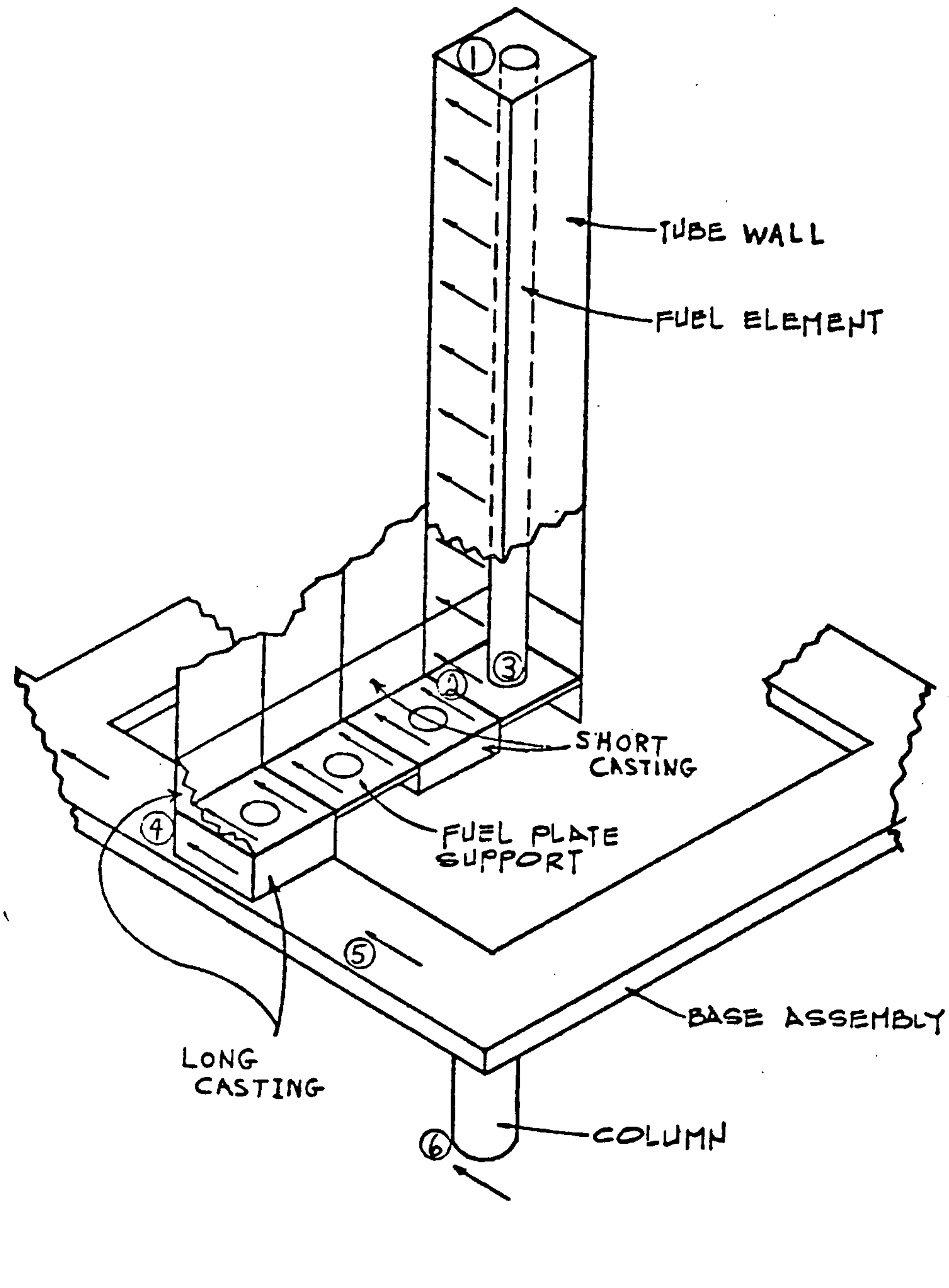
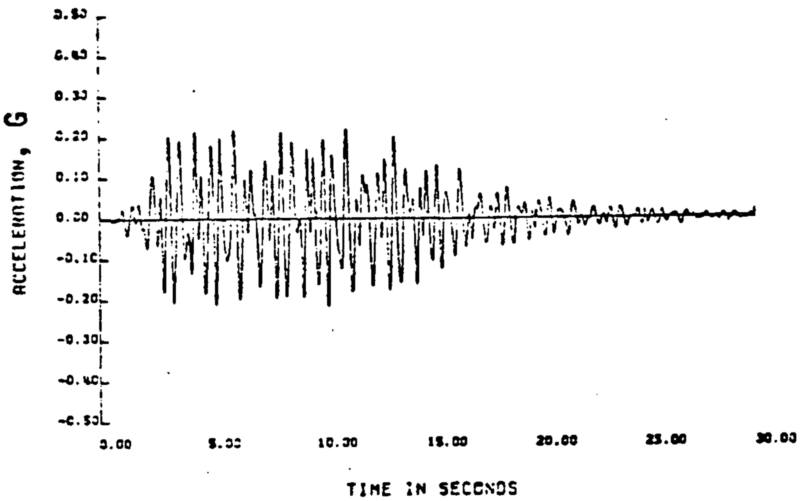
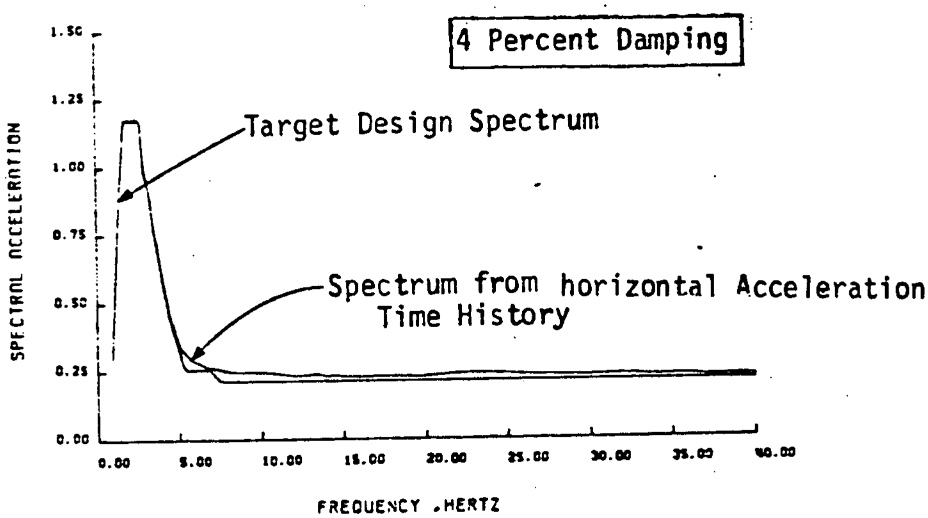


FIGURE 4.3-1 PATH OF SEISMIC HORIZONTAL FORCES IN MODULE



Horizantal Acceleration Time History



Floor Response Spectra

FIGURE 4.3-2 SEISMIC HORIZONTAL FLOOR TIME HISTORY AND RESPONSE SPECTRA COMPARISON

Enclosure 2

Response to 10/25/77 NRC
Request for Additional Information
on
Monticello Replacement of
Spent Fuel Pool Storage Racks

Q.1 Provide the number of grams of uranium-235 per axial centimeter of fuel assembly that was used in your criticality calculations. We intend to incorporate this information as a Technical Specification limit on fuel assemblies that are to be placed in these high density storage racks.

A.1 The criticality calculations did not assume a limit on grams of uranium-235. The axial distribution of U-235 in a fuel assembly is not a fully meaningful specification for Monticello fuel. This fuel is manufactured with variable rod enrichments and burnable poison, gadolinia, integral to the fuel. The reactivity of such fuel is a function of both enrichment (or axial U-235 distribution) and gadolinia content. For this reason it is deemed prudent to use a k-infinite limit for purposes of nuclear criticality safety in storage of the fuel.

If a Technical Specification limit is desired for the High Density Fuel Storage System, we would recommend k-infinite limit ($k_\infty \leq 1.35$ based on a BWR lattice pitch at 20°C.).

Q.2 In Section 4.1.2 of your submittal, on the verification of the criticality calculations, you state there were solid boron control curtains in the Jersey Central experiments. How does the areal density of boron ten atoms between the fuel assemblies and the thicknesses of the water channels next to the boron plates in the Jersey Central experiments compare with those in the proposed storage racks? Please provide any experimental confirmation that you may have for the calculated neutron multiplication factors in a BWR fuel assembly lattice with a 6.5 inch pitch, with approximately 0.25 inch water gap between the fuel assembly and the boral plate, and with about 1.6×10^{21} boron ten atoms per square centimeter between the fuel assemblies.

A.2 The areal density of boron-ten in the Jersey Central control curtains is 0.00597 grams $^{10}B/cm^2$ compared to 0.013 grams $^{10}B/cm^2$ for the High Density Fuel Storage System. The nominal thickness of the water channel between the surface of the Jersey Central control curtain and the outside of the fuel channel is 0.378 cm. compared to a 1.15 cm. water gap from the inside of the HDFSS storage cell to the outside of the fuel channel (when present).

There has been no direct experimental confirmation of this system. Such experiments for a system as substantially subcritical as this one are not necessary.

Q.3 In regard to Figure 4.1-1 of your submittal, what is the origin of the water gaps shown in the four corners?

A.3 The origin of the water gaps in the simplified cell is the combination of the nominal gap between storage tubes and the maximum gap between perpendicular boral plates. For conservatism and simplicity, the stainless steel in the corners was neglected and replaced with moderator.

Q.4 Provide the nominal and minimum thicknesses of the Type 304 stainless steel in the inner and outer storage tubes.

A.4 The 304 stainless steel tube walls have dimensions as listed in the table below.

304 SS Tube Walls (Ref ASTM A-240, A-480)

Dimensions in Inches

Wall	Nom	Tolerance	Minimum
Outer	.090	$\pm$ .008	.082
Inner	.0355	$\pm$ .004	.031

Q.5 Provide the nominal and minimum dimensions of the inner storage tubes.

A.5 The minimum dimension of a fuel storage space is 6.05 inches square, projected for the full length of the storage space. The nominal fuel storage space dimension is 6.250 inches square for a tube storage location and 6.261 inches square for a non-tube location.

Q.6 Provide the nominal density of boron carbide in the boral and the nominal thickness of the unclad boral sheets.

A.6 The Boral product used in the High Density Fuel Storage System consists of a 0.056 inch layer of B_4C-Al matrix sandwiched between two 0.010 inch aluminum sheets. The density of boron-10 is 0.013 grams/cm^2 minimum, corresponding to a boron carbide density of approximately 0.1 gm/cm^2.

Q.7 Provide the change in k_∞ for this high density storage lattice with a small change in uranium-235 enrichment.

A.7 A variation in enrichment has no significance in BWR fuel design as stated in reply to Question 1. In addition to average bundle enrichment, the k-infinite of a BWR bundle is dependent on geometry, enrichment distribution, gadolinia distribution, etc.

Q.8 Provide the change in k_∞ for this high density storage lattice with a small change in the areal density of the boron ten atoms between fuel assemblies.

A.8 The sensitivity to boron concentration was determined from previous analyses of a similar rack design. The change in cell k_∞ from .010 to .015 grams B^{10}/cm^2 was approximately 2% Δk.

Q.9 Provide the k_∞ for a filled lattice of these storage tubes held in a close packed condition, i.e., the minimum possible pitch.

A.9 The k_∞ of the HDFSS with fuel stored in the minimum possible pitch has not been determined. It has been shown, however, that the k_∞ of the storage cells decreases with both decreasing moderator density and increasing pitch. Therefore, the nominal spacing of 6.563 inches is the pitch that gives the maximum k-infinite.

The dimensions of the materials constituting the model cell were maintained constant for the cases evaluated. Therefore, if a lesser pitch is considered, the result of moving the fuel closer together is to exclude moderator, or effectively decrease the moderator density. Table 6 of Table 4.1-1 of the Design Report shows that the result of decreasing moderator density is a decrease of reactivity (k_∞). Thus, the k_∞ of storage tubes with minimum possible pitch will be less than that of the nominal 6.563 in. pitch case.

Q.10 Provide the amount of the increased pitch in case 4 of Table 4.1-1.

A.10 The pitch in Case 4 of Table 4.1-1 is 6.832".

Q.11 In Section 4.4 of your submittal, you stated that a dimensional inspection of the neutron absorber plate locations will be performed at the pool site. Describe in detail how you propose to use this and other tests to show that there will be a sufficient number of boral plates in the racks with a sufficient amount of boron ten isotope in the plates to maintain the $k_{eff} \leq 0.95$.

A.11 To verify that there will be a sufficient number of boral plates in the fuel storage modules with a sufficient amount of B^{10} isotope in the plates to maintain the $k_\infty \leq 0.95$, the following program has been developed.

The boron carbide used in the Boral sheets is certified as to its B^{10} isotopic content. Samples of each Boral sheet are chemically analyzed to determine the boron content. These data are statistically evaluated such that the samples are representative of the entire area of the Boral plate. It is verified that the minimum B^{10} content, at a 95% confidence level, meets or exceeds specification requirements. Analyses are performed to establish the correlation between the B^{10} content and the thickness of the Boral sample. The Boral sheets are dimensionally inspected and the thickness data are statistically analyzed to verify the sheet meets the minimum thickness requirement over its entire area at a 95% confidence level. These thickness data are compared with the correlation data to provide additional assurance that the B^{10} content meets or exceeds specification requirements. The Boral is inserted into a tube assembly only after it has been verified that each of the above inspections and evaluations has been successfully performed. The Boral plates are placed between the inner and outer walls of the tube assembly. The assembled storage tube is hydro-formed, "locking" the Boral plates in place. The inner and outer walls are welded together at each end of the tube encapsulating the Boral. It is then not possible to remove the Boral without destroying the tube. The thickness of the storage tube wall is measured after tube assembly. These data are statistically analyzed and the entire tube assembly population, known to contain Boral, is uniquely identified by the average thickness (mean) and the standard deviation.

Presence of the neutron absorber material in the fabricated fuel storage module is verified by visual examination and dimensional inspection. The thickness of the Boral plate is different than commercially available alumunim or SSt sheets. Materials of standard thickness used in place of Boral would be detected by the significant difference in wall thickness measurements for tube walls which contain Boral as opposed to tube walls which contain non-Boral materials.

The thickness of the walls of the module assembly will be measured at the reactor site before installation. These data will be statistically analyzed such that the individual fuel storage module is uniquely identified by the mean and the standard deviation. The two sets of wall thickness data will be statistically compared to determine, at a 95% confidence level, if there is a significant difference between the individual tube wall thickness data and the module assembly data. If a significant difference does not exist it indicates the module was made from tubes known to contain neutron absorbing material and the module will be accepted as containing the required amount of Boral plates.

Q.12 Describe the procedures that will be used to remove the present racks and install the new ones. Specifically, discuss how you will preclude the possibility of dropping or tipping a rack onto the spent fuel in the pool.

A.12 The spent fuel storage modification at Monticello will be carried out in two or more phases. The first phase entails placement of new spent fuel racks (hereafter called modules) 1 through 4 in locations shown on the attached Figure 1. Placement of modules 5 through 13 will occur as a later phase or phases as modules become available from the manufacturer.

Presently the spent fuel pool at Monticello contains 37 standard General Electric spent fuel racks (hereafter referred to as racks) with a capacity of 20 spent fuel assemblies each. A total of 616 spent fuel assemblies are stored in 31 of these racks. Also the pool contains two racks for defective fuel canisters or control rod blades, each rack capable of holding up to ten canisters or control blades, as needed. One rack is designated as a control blade rack the other as a defective-fuel rack, although fuel has never been damaged to the extent that placing it in a sealed canister has been necessary. The existing pool layout is shown on Figure 2, attached.

During pool preparations and module installation the following general guidelines will be adhered to:

A. All racks will be emptied of their contents prior to unbolting and moving.

B. Racks and modules will be kept low in the pool during horizontal movement. They will not be moved over spent fuel.

C. Vertical movement of racks and modules in and out of the pool will be done in designated areas. All racks removed from the pool in preparation for the first phase will be lifted from the southeast corner of the pool. Modules 1 through 4 will be lowered into the southeast corner of the pool. All other racks removed from the pool will be lifted from the northwest corner of the pool. Modules 5 through 13 will be lowered into the northwest corner of the pool.

D. Racks will be washed down to remove contamination as they are lifted out of the pool water.

E. Racks or modules which are out of the fuel storage pool shall not be moved within 12 feet of the fuel pool wall except when in transit in or out of the pool in the manner specified in item C above.

F. New modules will be handled with the modified reactor building crane main hoist with a special lifting fixture, both of which are designed with a high safety factor.

The specific procedure currently being considered for pool preparations and module installation, which meets the above criteria, is as follows:

1. Move rack 24 (See Figure 2) to the south end of the pool.

2. Remove the defective-fuel rack from the pool.

3. Move rack 24 into the location vacated by the defective-fuel rack.

4. Move rack 31 into the location vacated by rack 24.

5. Remove the work table from the southeast corner of the pool.

6. Remove racks 29, 30, 32, 33, 34 and 35. After removal from the water, move racks directly to the south away from the pool.

7. Install modules 2, 3, 4 and 1 respectively, approaching the fuel pool from the south and lowering the modules into the southeast corner of the pool.

8. Move fuel from racks 1 through 25 and 28 into the new storage modules. (Approximately 737 assemblies will be stored in modules 1 through 4 and racks 36, 37, 26 and 27 after the Fall 1978 refueling outage.)

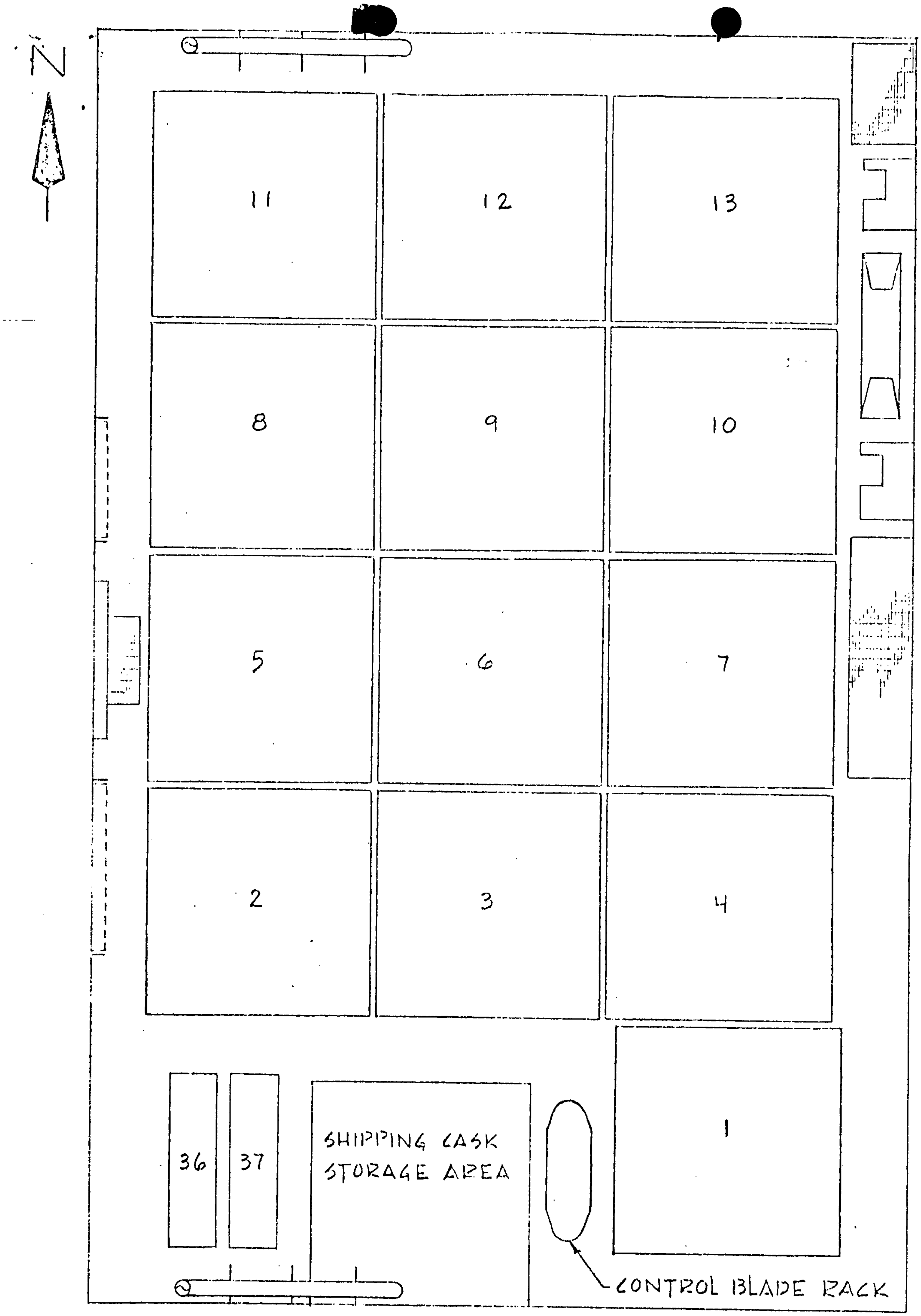
9. Remove racks 1 through 25 and rack 28 via the northwest corner of the pool. After removal from the water, move racks directly to the west away from the pool.

10. Install module 7.

11. Move fuel from racks 26 and 27 to southeast locations of module 7.

12. Remove racks 26 and 27.

13. Install modules 6, 5, 8, 9, 10, 13, 12 and 11, respectively, approaching the fuel pool from the west and lowering the modules into the northwest corner of the pool.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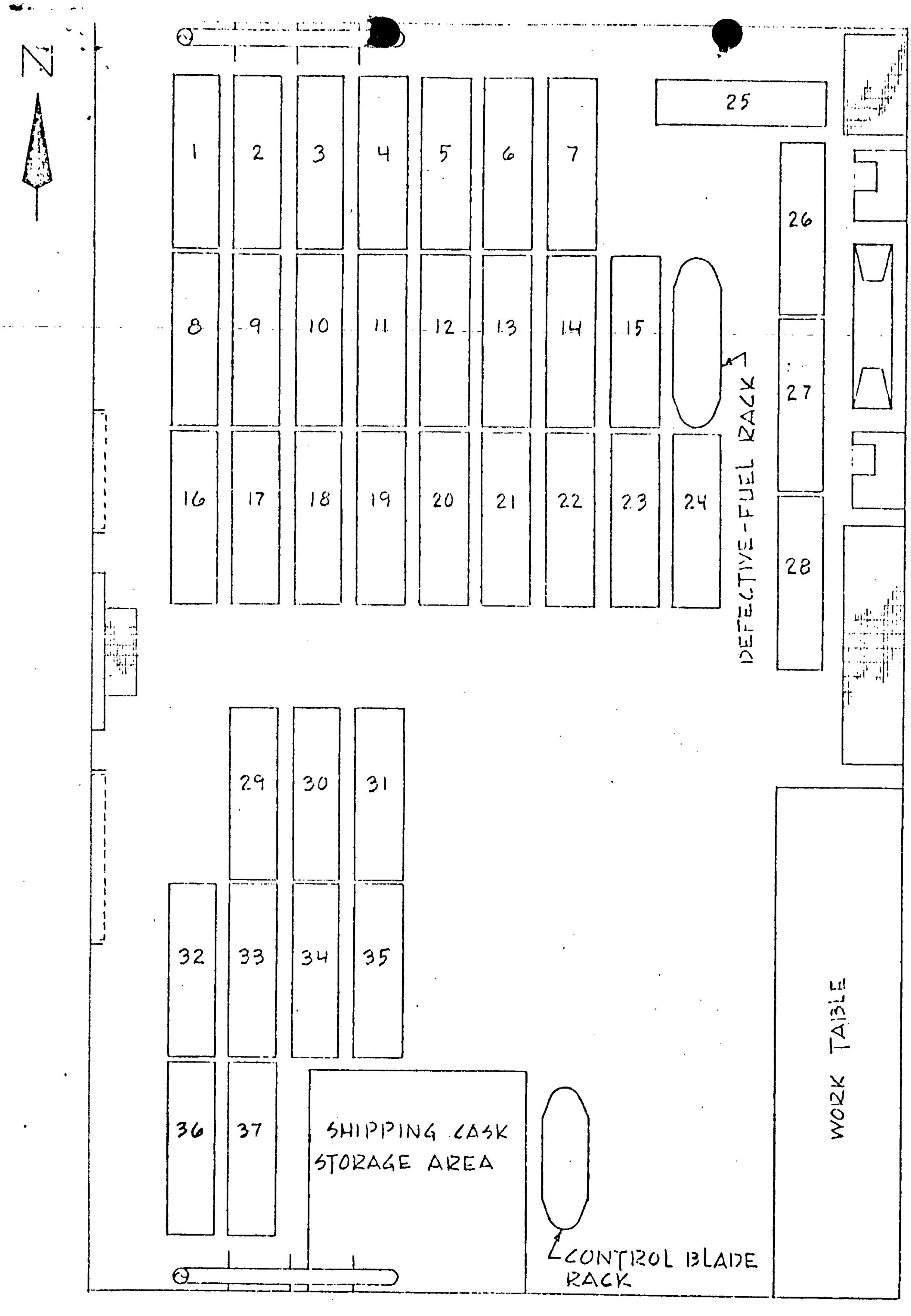
SHIPPING CASK STORAGE AREA
CONTROL BLADE RACK

FIGURE 1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DEFECTIVE-FUEL RACK
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SHIPPING CASK STORAGE AREA
CONTROL BLADE RACK
WORK TABLE

FIGURE 2