SAFETY EVALUATION BY THE OFFICE OF

NUCLEAR REACTOR REGULATION

SUPPORTING FACILITY MODIFICATIONS

TO INCREASE THE CAPACITY OF THE

SPENT FUEL STORAGE POOL

PROVISIONAL OPERATING LICENSE NO. DPR-45

DAIRYLAND POWER COOPERATIVE

LACROSSE BOILING WATER REACTOR

DOCKET NO. 50-409

Date: July 13, 1979

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1.0 Introduction

By letters dated April 20, June 7, July 11, August 7, September 25, October 4, and November 29, 1978, Dairyland Power Cooperative (DPC) (the licensee) proposed to increase the storage capacity of the Fuel Element Storage Well (FESW), at LaCrosse Boiling Water Reactor (LACBWR) from 134 to 440 fuel assemblies. DPC provided additional information to support their proposal by letters dated October 26, November 20, 1978, and January 4 and 31, February 14, March 1, May 17 and June 26, 1979. The increased capacity would be achieved by replacing the originally installed spent fuel storage racks with new racks of a two-tier design. These new racks will utilize a B_4C/polymer composite material between storage locations as a neutron absorber.

2.0 Discussion

A general description of the new two-tier rack design is contained in DPC's submittal of April 20, 1978. Pertinent excerpts follow.

2.1 Storage Rack Arrangement

The arrangement of the storage racks in the LACBWR fuel storage well is shown in Figure 1. From this figure it can be seen that the fuel storage well has two (2) storage racks with a 9x8 array of fuel storage locations and two (2) storage racks with a 4x10 array of fuel storage locations. Each storage location is capable of storing two (2) fuel assemblies in a two-tier configuation (i.e., one assembly positioned above the other). Fuel assemblies stored in the lower tier are always accessible (i.e., for periodic surveillance) as long as the upper tier location is vacant.

The top of the fuel storage racks would be about 30 inches below the fuel transfer canal. Therefore, in the event of a postulated failure of the canal gate or pressure vessel to cavity seal, the fuel stored in the upper tier locations will remain covered with water.

Floor area is provided at the south end of the fuel storage well for the spent fuel shipping cask. This area is also used to store the core spray bundle during refueling operations. The existing LACBWR crash pad which was provided to ensure the structural integrity of the fuel storage well in the event of a cask drop accident will be modified to fit into the area.

2.2 Storage Rack Description

Each storage rack consists of a welded assembly of fuel storage cells spaced 7 inches on center. Each rack, however, is fabricated in two sections designated upper tier rack and lower tier rack sections.

The upper-tier rack section consists of two "egg-crate" grid structures which position and secure the fuel storage cells. A typical cross-section

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for the fuel storage cell is shown in Figure 2. The cell consists primarily of four angles which are welded to the upper and lower egg-crate grids and which provide full length support for the stored fuel assembly. The four angles are stiffened with side plates welded to the angles. Outer side plates are added to the opposite sides of the cell to form the poison compartments as can be seen in the cell cross-section. The fuel storage cells are positioned such that every other cell is rotated 90°. This cell configuration results in a poison plate being placed between each adjacent fuel assembly in each orthogonal direction. The bottom grid of the upper-tier rack section is designed to sit on and interlock with the lower-tier rack section.

The lower-tier rack section also consists of two "egg-crate" grids which position and secure fuel storage cells essentially identical to those described above. Support feet attached to the lower grid raise the rack above the floor to the height required to provide a cooling water supply plenum (for natural circulation) (Figure 3). Each support foot contains a remotely adjustable jackscrew to permit the rack to be leveled following installation. The jackscrews will be accessible through the storage cells. The lower tier fuel assembly support plates are welded to the lower egg-crate grid. The upper egg-crate grid (of the lower tier rack) has four seating surfaces in each storage location to support the remotely installable fuel assembly support plate provided for the upper tier locations. The upper tier fuel assembly support plate which can be remotely locked into the grid, is provided with tow pins for securing the fuel assembly shroud with the shroud locking ring. Each support plate contains a cooling flow orifice.

The horizontal seismic loads are transmitted from the rack structures to the fuel storage well walls at three elevations (the top grid of the upper tier rack section, the top grid of the lower-tier rack section and the bottom grid of the lower-tier rack section) through adjustable pads attached to the rack structures. The thickness of these pads are adjusted as required to accommodate variations in the storage well walls and to provide the small gaps needed for thermal expansion. Lateral bracing will be provided around the periphery of the cask setdown/core spray bundle storage area to ensure proper transfer of the seismic loads across and/or around this area at the three rack elevations. The vertical dead-weight and seismic loads are transmitted to the storage well floor by the rack support feet.

The fuel storage racks and associated seismic bracing are to be fabricated from Type 304 stainless steel.

3.0 Evaluation

3.1 Criticality Analysis

The LaCrosse fuel pool criticality calculations are based on unirradiated fuel assemblies with no burnable poison. Calculations were made for both stainless steel and Zircaloy clad fuel elements. The fuel loadings which were used in the analyses were nominal values of 22 4 grams of uranium-235 per axial centimeter of fuel assembly for the stainless steel clad fuel elements and 16.6 grams of uranium-235 per axial centimeter of fuel assembly for the Zircaloy clad fuel elements.

Nuclear Energy Services, Incorporated (NES) performed the criticality analyses for DPC. NES made parametric calculations by using the HAMMER computer program to obtain four-group cross sections for EXTERMINATOR diffusion theory calculations. The blackness theory program, BRM, was used to calculate the thermal and epithermal group cross section for the boron region. This calculational method was used to determine the nominal k_∞ and then the effects of design and fabrication tolerances, changes in temperature, voids in the pool water, and abnormal dislocations of fuel assemblies in the racks. To obtain its January 31, 1979 response to our request for additional criticality information, NES used the KENO IV Monte Carlo program to calculate the increase in k_∞ that would be obtained if it was assumed that all of the boron carbide particles were the maximum size of .020 inches. NES found that this increase would be 0.9 percent. Adding this increment to NES's previously calculated value for the nominal reference configuration gives a k_∞ of 0.922. With all of the other effects listed above included NES's maximum value of k_∞ is 0.925. NES found that if a fuel assembly in the shipping cask area of the pool was brought up to the outside of a fully loaded rack, the k_∞ could increase by 0.004. Thus NES's "worst abnormal" k_∞ is 0.93.

Moreover, NES calculated the reference configuration with the more rigorous KENO IV Monte Carlo program. This resulted in a lower k_∞ than was obtained with the EXTERMINATOR program; so NES assumed that no calculational bias was needed on the worst case k_∞ of 0.93.

By letter dated January 22, 1979 DPC stated that two inspections will be performed, on site, to verify the presence of the boron plates in the racks. First, every location will be checked by visually observing the boron plates through a test hole which will be in every one of the stainless steel jackets. Second, a blackness test shall be performed on ten percent of the neutron absorber plate locations, which will be randomly selected in each storage rack. DPC also stated that if the blackness test reveals any missing boron plates, all of the plate locations will be checked.

DPC stated in its letter of January 4, 1979 that it will perform a surveillance test on coupons of the B_4C/Polymer Composite plates to verify the continued presence of the boron in the plates in the pool over the complete life of the storage racks.

Considering the fact that the neutron absorber plates in these racks only partially cover the fuel assemblies (i.e., 3.94 inches out of 5.65 inches) the preceding calculated results compare favorably with the results of calculations made with other methods for fuel pool storage lattices with

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boron plates. By assuming new, unirradiated fuel with no burnable poison or control rods, these calculations yield the maximum neutron multiplication factor that could be obtained throughout the life of the fuel assemblies. This includes the effect of the plutonium which is generated during the fuel cycle.

We find that DPC's onsite tests for the presence of the boron is acceptable provided that if any boron plates are found to be missing in the blackness test, the NRC is promptly notified. Moreover, we have determined that DPC's proposed surveillance test is an acceptable way to verify the continued presence of the boron over the life of the racks.

The results of our review indicate that all factors which could affect the neutron multiplication factor in this pool have been conservatively accounted for. We have also found that if the fuel loading is limited to no more than 22.6 grams of uranium-235 per axial centimeter of fuel assembly for the stainless steel clad fuel elements and to no more than 16.6 grams of uranium-235 per axial centimeter of fuel assembly for the Zircaloy clad fuel elements, the maximum neutron multiplication factor in this fuel well with the proposed racks will not exceed 0.95. This is NRC's acceptance criterion for the maximum (worst case) calculated neutron multiplication factor in a spent fuel pool. This 0.95 acceptance criterion is based on the uncertainties associated with the calculational methods and provides sufficient margins to preclude criticality in the fuel pool.

In summary, we find that when any number of the fuel assemblies, with no more than 22.6 grams of uranium 235 per axial centimeter of fuel assembly for the stainless steel clad fuel elements and no more than 16.6 grams of uranium 235 per axial centimeter of fuel assembly for the Zircaloy clad fuel elements, are loaded into the proposed racks, the neutron multiplication factor will be less than the 0.95 limit. To preclude the possibility of the neutron multiplication factor k_{eff} in the fuel pool exceeding the 0.95 limit without being detected, the use of these high density storage racks for fuel assemblies which have stainless steel clad fuel elements with more than 22.6 grams (allowing for local deviations) of uranium 235 per axial centimeter of fuel assembly and those which have Zircaloy clad fuel elements with more than 16.6 grams of uranium 235 per axial centimeter of fuel assembly should be prohibited pending our review and approval. Based on the above we have concluded that the proposed rack design is acceptable from the standpoint of criticality.

3.2 Spent Fuel Cooling

The licensed thermal power for the LACBWR is 165 MW_t. DPC plans to refuel this plant annually. This will require the replacement of about twenty four of the seventy two fuel assemblies in the core every year. In its June 7, 1978 submittal, DPC assumed a 3-day (72 hour) decay time for calculating the maximum heat generation rates in the fuel pool for the annual refueling and a 7-day (168 hour) decay time for a full core offload. With these decay times, DPC used the method given in the NRC Standard Review Plan, NUREG-75/087, to calculate 0.93×10^6 BTU/hr as the maximum possible heat load for any annual refueling.

The spent fuel pool cooling system consists of two pumps and one heat exchanger. Each pump is designed to pump 260 gpm (1.3×10^5 pounds per hour). Only one pump is operated at a time; so there is a backup pump available. At the design flow rate, the heat exchanger is designed to transfer 2.6×10^6 BTU/hr from 120° fuel pool water to 90° Component Cooling water, which is flowing through the heat exchanger at a rate of 1.3×10^5 pounds per hour.

In its August 7, 1978 submittal, DPC included an analysis of the natural convection cooling of the spent fuel in the proposed two-tier racks. This showed that the maximum increase in water temperature in the storage container with minimum natural circulation flow will be less than 24°F and that the temperature of all of the fuel pool water will always be far below the saturation temperature.

Using the method given on pages 9.2.5-8 through 14 of the NRC Standard Review Plan, with the uncertainty factor, K, equal to 0.1 for decay times longer than 10^7 seconds, we calculate that the maximum peak heat load during the 1990 refueling could be 1.1×10^6 BTU/hr and that the maximum peak heat load for a full core offload that essentially fills the pool could be 2.0×10^6 BTU/hr. This full core offload was assumed to take poace one year after the 1987 refueling. We also find that the maximum incremental heat load that could be added by increasing the number of spent fuel assemblies in the pool from 133 to 440 will be 0.2×10^6 BTU/hr. This is the difference in peak heat loads for full core offloads that essentially fill the present and the modified pools. Since the spent fuel pool cooling system is designed to remove 2.6×10^6 BTU/hr from 120°F fuel pool water with one pump operating, the maximum fuel pool outlet water temperature will be less than 120°F for the maximum heat load, 2.0×10^6 BTU/hr, which we calculate for any full core offload.

In the unlikely event that both spent fuel pool cooling pumps fail just after a full core offload which generates 2.0×10^6 BTU/hr the fuel pool water would heat up at a rate of about 9°F/hr. Thus assuming an initial outlet water temperature of 120°F it would be ten hours before boiling would commence. During boiling a makeup rate of four gallons of water per minute (gpm) would be required to keep the pool full. In its January 31, 1979 letter, DPC stated that either the overhead storage tank or the Demineralized Water Hose Station could provide this make up. We have concluded that ten hours would be sufficient time to provide a 4 gpm source of makeup water for the spent fuel pool.

Based on the above we have concluded that the present cooling capacity for the spent fuel pool at the LaCrosse Boiling Water Reactor will be sufficient to handle the incremental heat load that will be added by the proposed modifications. We also conclude that this incremental heat load will not alter the safety considerations of the spent fuel cooling from those previously reviewed and found to be acceptable.

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3.3 Structural Analysis and Acceptance Criteria

The loads, loading combinations, and acceptance criteria are in accordance with Section 3.8.4 of the NRC Standard Review Plan. The allowable stresses for both type 304 and 17-4 PH stainless steels are in accordance with Section III of the ASME B&PV Code. The allowable stresses for the stainless steel welds are as specified in Table 1.5.3 of the AISC Code.

The seismic analysis performed on the racks was a modal response spectrum analysis using 1.0% equipment damping for 1/2 safe shutdown earthquake (SSE) and 2.0% for SSE. Loads, stresses, and deflections were determined for an 8X9 - 4X10 rack coupled model. This coupled model of the structural grid arrays, including the region for control rod storage, represents the structural case having the lowest frequencies of vibration. The seismic analyses included the weight of the rack structure, fuel assemblies, and contained and hydrodynamic water mass. The "rattling" effects of the fuel assemblies/cells impacting have been accounted for. The sloshing effects of water on the racks and pool structure have been considered. The combination of modes and spatial components are in accordance with Regulatory Guide 1.92.

The racks have been designed to withstand the local as well as gross effects of a dropped fuel assembly. Drops on top of a double tier rack, subsequent tipping, and straight drops directly through cells in both a flexible location and over one of the support feet were considered. Because of the nature of the loading, the static yield strength was increased based on GE data and substantiated by tensile tests performed by NES.

The effects from a postulated stuck fuel assembly have been bounded by examining the case of the grapple being hooked onto the storage cell with an upward force of 4,000 lbs.

Because of the increased loading imparted to the pool resulting from this increase in storage capacity, a structural analysis was performed of the pool walls and floor. The load combinations considered were per Section 3.8.4 of the Standard Review Plan and the allowable stress/ load limits were taken from the ACE-318-71 Code.

Cask drop analyses were performed to evaluate the consequences of a postulated cask drop on top of the crash pad and the new spent fuel storage racks. Linear and non-linear analyses, using energy balance techniques, were done to determine the maximum deformations and reaction loads developed in the crash pad and racks as a result of the kinetic energy of the cask drop. The adequacy of the pool floor to withstand these reaction loads was evaluated. Reanalysis of the crash pad was necessary due to modifications to the pad required to accommodate the layout of the new high density spent fuel storage racks.

The spent fuel pool drain line is routed off the bottom of the pool. Since a break in this line could lead to a loss of cooling water and

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compromise the integrity of the stored spent fuel, a seismic analysis of this piping was performed and a second check valve added between the pool connection and pipe anchor, i.e., the forced circulation pump cubicle wall. A modal response spectrum analysis, using 1% and 2% damping for the OBE and SSE, respectively, was performed on this approximately 20 feet of piping. In addition, thermal and weight analyses were done. The seismic responses were combined in accordance with the requirements of Regulatory Guide 1.92. Load combinations and allowable stresses are in accordance with Subsection ND of Section III of the ASME Code and Regulatory Guide 1.48.

Seismic response spectra utilized were taken from Gulf United Services Report No. SS-1162, entitled "Seismic Evaluation of the LaCrosse Boiling Water Reactor", dated January 11, 1974. These spectra were developed using an SSE ground acceleration value of 0.12 g. This document currently under staff review, was submitted in support of the full term operating License application.

The new racks will be installed in accordance with a detailed installation plan developed by DPC. Installation of the new racks will not commence until all rack sections and associated components are on-site and ready for installation. New racks positioned for the purpose of shuffling previously stored fuel will be secured by safety lines to prevent sipping.

The design, fabrication, and installation procedures; the structural design and analysis procedures for all loadings, including seismic and impact loading; the load combinations and structural acceptance criteria; and the applicable industry codes were all reviewed in accordance with the Branch Technical Position (BTP) entitled, "Review and Acceptance of Spent Fuel Storage and Handling Applications."

Results of the seismic and structural analyses show that the racks are capable of withstanding the loads associated with all design loading conditions. Also, impact due to fuel assembly/cell interaction will result in no damage to the racks or fuel assemblies themselves.

Results of the dropped fuel assembly analyses show that local rack deformation will occur, but indicate that gross stresses meet the applicable allowables and that no damage to the pool floor or liner will occur. The analyses indicate that no more than two equivalent fuel assemblies will be damaged as a result of a dropped fuel assembly.

Results of the stuck fuel assembly analysis, bounded by evaluating the effects of the grapple being hooked onto the storage cell, show that the stresses are below those allowed for the applicable loading combination.

Results of the structural analysis of the pool show that the present load carrying capacity of the pool is adequate.

Cask drop analyses results indicate that although incurring permanent damage, the modified crash pad is capable of performing its intended function and will protect the pool floor from any damage. Results of a postulated cask drop analysis on the racks indicate that extensive local rack deformation and fuel damage will occur. The reaction loads generated during the cask drop will lead to failure of the rack base structure and support legs. Using empirical missile equations developed by the Ballistic Research Laboratory and Stanford Research Institute, the impacting grid structure could penetrate the 3/8" pool floor liner plate up to a depth of 0.28". Therefore, additional 3/8" stainless steel barrier plate will be provided on top of the pool floor liner under the rack structures to ensure that the existing liner will not be structural damaged. With the additional barrier plate in place, a dropped cask will not damage the pool liner or floor sufficiently to adversely affect the leak-tight integrity of the storage well (i.e., would not cause excessive water leakage from the FESW).

Results of the spent fuel pool drain line analyses show that all stresses are within ASME Code allowables. Further, stresses associated with the faulted plant condition loads are below the upset plant condition allowables.

All rack and support frame components are fabricated of type 304 stainless steel and 17-4 PH stainless steel. The 17-4 PH stainless steel being utilized as support leg components will be heat treated at 1100°F, the surface film removed by pickling and correct heat treatment verified by hardness testing of test samples heat treated along with each lot of material.

The type 304 and 17-4 PH stainless steels and the type 348H stainless steel cladding used in the new storage racks are compatible with the storage pool environment, which is oxygen-saturated, high purity demineralized water and controlled to a pH 4.5-8 and a maximum temperature of 150°F. In this pool water environment, the corrosive deterioration of the 304 alloy should not exceed 5.90×10^{-5} inches in 100 years, which is minute relative to initial thickness. Dissimilar alloy interaction (electrolytic or galvanic corrosion) between the 304 and 17-4 PH stainless steels of the storage racks, the 348H stainless steel cladding of the spent fuel assemblies (or Inconel and Zircaloy in spent fuel assemblies considered for possible future use), and the 304 stainless steel pool liner will be of no significance because of the minute electrical potential differential.

The composite plate of B_4C/polyethylene resin neutron absorber material is not affected by chemical degradation because it is inert to the pool water environment. Irradiation will cause off-gassing of the polyethylene resin. However, venting of the storage cell will allow generated gas to escape and prevent bulging of the stainless steel shroud encapsulating the neutron absorber plates.

Based on the discussion and evaluation above, we find that the new proposed LaCrosse spent fuel storage racks and the design and analyses performed for the racks and pool are in conformance with established criteria, codes and standards specified in the staff position for acceptance of spent fuel storage and handling applications. Further, we find the design and analyses performed for the crash pad and pool drain line acceptable. Therefore, we conclude that the subject modification proposed by the licensee is acceptable, and satisfies the applicable requirements of the General Design Criteria 2, 4, 61, and 62 of 10 CFR, Part 50, Appendix A.

3.4 Occupational Exposure

We have estimated the increment in onsite occupational dose resulting from the proposed increase in stored fuel assemblies on the basis of information supplied by the licensee for dose rates in the spent fuel area from radionuclide concentrations in the FESW water and the spent fuel assemblies. The spent fuel assemblies themselves in the double tier will contribute a small fraction of the dose rates in the pool area because of the depth of water shielding the fuel. A Technical Specification change will require the licensee to provide a minimum of 16 feet of water above the spent fuel assemblies. This depth of water will reduce dose rate levels from the spent fuel elements to small fractions of that provided by the radionuclide concentration in the water. Consequently, the occupational radiation exposure resulting from the additional spent fuel in the pool is negligible. Based on present and projected operations in the spent fuel pool area, we estimate that the proposed modification would add less than one percent to the total annual occupational radiation exposure at this facility. The small increase in radiation exposure will not affect the licensee's ability to maintain individual occupational doses to as low as is reasonably achievable and within the limits of 10 CFR Part 20. Thus, we conclude that storing additional fuel in a double tier in the FESW will not result in any significant increase in doses received by occupational workers.

3.5 Radioactive Waste Treatment

The plant contains waste treatment systems designed to collect and process the gaseous, liquid and solid wastes that might contain radioactive material. There will be no change in the waste treatment systems because of the proposed modification.

The only waste treatment system affected by the modification of the FESW is the liquid radwaste system. The proposed modification to the FESW is expected to increase the amount of leakage from the pool negligibly. The present annual average leakage rate which depends on pool water level is about 6 gallons per hour. This leakage rate is higher than normal for spent fuel pools but small (10%) compared with other normal waste stream flows into the LaCrosse radwaste treatment system. The leakage is collected and processed through the liquid radwaste system prior to discharge. The licensee has operated the plant within the plant radiological effluent Technical Specifications with this leakage.

The licensee has been unable to reduce pool leakage rate below the present value and believes that the leakage is from the bottom or near the bottom of the pool based on tests which varied the pool water level. Because of spent fuel in the pool, it is difficult for the licensee to determine exactly where the leakage is and make repairs.

As part of the proposed modification, the licensee has committed to attempt to detect and eliminate the pool leakage. The licensee has a program which includes techniques such as acoustic detection and eddy current testing to try to find the leak. This will be done after the pool liner is cleaned and portions of the pool floor are exposed while the old fuel racks are being replaced. We will require the licensee to submit a report concerning the pool leakage within 120 days after the modification of the pool. This report would describe the program and its results in reducing pool leakage. It would also describe additional actions planned regarding any remaining pool leakage.

No increase in the present pool leakage is expected until spent fuel is stored in the upper tier and the pool water is raised to provide more shielding for this spent fuel. The expected leakage then has been determined by the licensee to be about 7 to 10 gallons per hour. This flow is still less than 10% of the monthly average volume of liquids processed by the plant liquid radwaste treatment system. This increase should not reduce the capability of the plant liquid radwaste treatment system to process water and to keep releases of radioactive liquids within the requirements of the plant Technical Specifications. The radioactive effluent specifications would not be changed.

We have reviewed the liquid radwaste treatment system based on the expected leak rates from the FESW. We conclude that the additional liquid releases from the plant that might result from the proposed modification are a small fraction (approximately 1%) of the average annual liquid releases from the plant. This does not change the evaluation of the liquid treatment system contained in our Safety Evaluation dated January 1963. Based on our evaluation of the liquid radwaste treatment system, we conclude that the system would be capable of limiting radioactive releases to a small fraction of the limits of 10 CFR Part 20 and of the plant radiological liquid effluent Technical Specifications.

3.6 Postulated Accidents

3.6.1 Fuel Handling Accidents

The NRC staff is generically considering load handling operations in the vicinity of spent fuel pools to determine the liklihood of a heavy load impacting fuel in the pool and, possible radiological consequences of such an event. Because LaCrosse will be required to prohibit loads, other than a spent fuel shipping cask and reactor vessel internals which are stored in the pool during refueling, greater than the nominal weight of a fuel assembly to be transported over spent fuel in the FESW, we have concluded that the likelihood of any other heavy load handling accident is sufficiently small that the proposed modification is acceptable and no additional restrictions on load handling operations in the vicinity of the FESW are necessary while our generic review is under way.

We have evaluated the potential consequences of postulated fuel handling accidents for the proposed modification, with fuel stored in double tiers. Our previous evaluation dated October 22, 1975, for storing spent fuel in a single tier is no longer appropriate for the proposed double tiering. For this present evaluation, we have assumed the equivalent of all the fuel pins in two freshly discharged fuel assemblies are damaged and the fuel is discharged from the reactor 72 hours after shutdown. The other assumptions, which are the same as in the previous evaluation, are given in Table 1. The estimated potential consequences are 162 Rem to the thyroid and 2 Rem to the total body at the Exclusion Area Boundary (EAB). The potential consequences at the Low Population Zone are smaller than those at the EAB. These are within the guidelines of 10 CFR Part 100 and are, therefore, acceptable.

We believe this postulated accident analysis is appropriately conservative for the following reasons: The probability of the postulated fuel handling accident involving extensive release of radioactivity is small. There have been several hundred reactor-years of plant operating experience with only a few incidents involving the dropping of spent fuel, none of which resulted in measurable releases of radioactivity. The likelihood of a dropped fuel assembly directly striking another assembly stored in the racks, an impact which results in the greatest energy available for crushing fuel pins in both assemblies, is small due to moments of drag forces exerted by the water causing it to fall in a tipped orientation. The licensee does not plan to store freshly discharged assemblies in both the upper and lower tiers; thus, an assembly dropped on the one stored in the upper rack position should not initiate an impact involving three freshly discharged assemblies. Furthermore, there are steel plates positioned at the bottom of each of the upper tier fuel storage cells which should render three assembly impacts unlikely. However, even if it is assumed that a freshly discharged assembly is stored on top of another one, the storage cell lower protective plate is not securely in place and a freshly discharged fuel assembly is dropped directly on an assembly in the upper rack such that all the fuel pins in three equivalent assemblies are damaged, offsite accident consequences would be within the exposure guidelines of 10 CFR Part 100.

3.6.2 Cask Drop Accidents

The evaluation of the potential consequences of a drop of the spent fuel shipping cask into the FESW is considered in our Safety Evaluation dated October 22, 1975, for the first modification of the FESW. The potential consequences in that evaluation would not be changed by this proposed action because the previous evaluation assumed that the pool was filled with spent fuel, including 24 freshly discharged assemblies (a normal refueling), and all this fuel was ruptured in the accident. The proposed action, taking into account the additional old fuel that may be in the pool, would not change the conclusion in the Safety Evaluation dated October 22, 1975, that the potential consequences of this postulated accident were within the 10 CFR Part 100 exposure guidelines are therefore, acceptable. The previous evaluation, however, did not consider a freshly discharged full core offload in the pool. For the postulated consequences of a cask drop into the pool with a freshly discharged core in the pool to be well within the 10 CFR Part 100 exposure guidelines, the containment must be isolated if the spent fuel in the pool has decayed less than 43 days. We have determined that the Technical Specifications should be changed to require containment isolation if the spent fuel shipping cask is near the pool and if spent fuel in the pool has decayed less than 43 days. On the rare occasion when the entire core may be off loaded from the reactor vessel into the FESW the required decay time will be extended to 51 days to compensate for the reduced depth of water (16 feet) above the spent fuel pool. The licensee agrees and the proposed technical specifications revisions include this requirement. On this basis, we have concluded that the potential consequences of a spent fuel cask drop into the pool will be well within the exposure guidelines of 10 CFR Part 100 and therefore, acceptable.

The potential radiological consequences of dropping the reactor vessel internals are bounded by the evaluation for dropping the spent fuel shipping cask into the pool.

3.7 Installation of New Racks

3.7.1 Criticality Consideration

DPC plans to temporarily install one of the proposed double tier racks in the cask loading area near the southside of the pool. This rack will be over 18 feet high. Since the pool is only 11 feet square, it will not be possible for it to tip over even when all of the other racks are removed from the pool. However, in their October 16, 1978 submittal DPC stated that all temporarily installed racks will be secured by safety lines when they contain fuel assemblies. DPC plans to transfer the fuel assemblies that are in the racks along the north wall of the pool to the temporarily installed double tier rack on the southside. This will allow removal of two rows of racks along the north wall and temporary installation of a small double tier rack in the northwest corner of the pool. This will provide enough temporary storage space so that the old racks in the northeast corner of the pool can be emptied, removed, and permanently replaced by a large, double-tiered rack. There will then be sufficient storage space for the fuel assemblies in the pool so that the complete rack change can be made without moving racks containing fuel assemblies.

Since the horizontal dimensions of the fuel pool are only eleven feet by eleven feet there is not enough room for the racks to swing very much as they are lowered to the bottom of the pool. Thus if one of the racks, which was being moved into or out of the pool, were to accidentally drop, it is highly unlikely that the racks in the bottom of the pool could be significantly compressed in a horizontal direction. But this is the only way the k_{eff} could be increased. Compression of the filled racks from the top would squeeze water out of the fuel assembly itself and thereby reduce the k_{eff}. Thus it is highly unlikely that the k_{eff} would be increased by dropping a rack.

We conclude that there is reasonable assurance that the health and safety of the public will not be endangered during the installation of the proposed racks.

3.7.2 Occupational Radiation Exposure

We have reviewed the licensee's plans for the removal and disposal of the low density racks and the installation of the high density racks in a double tier with respect to occupational radiation exposure. The licensee has discussed two plans for doing this work. The first plan assumes the old racks can be disassembled remotely from above water with specially designed tools. The occupational radiation exposure for this operation is estimated by the licensee to be about 16 man-rem. If the work can not be accomplished remotely, the alternate plan will use divers to disassemble the old racks. Because of the exposure to the divers and increased exposure to other personnel, the occupational exposure commitment would be increased to about 23 man-rem.

We consider these to be reasonable estimates that represent a small fraction of the total man-rem burden from occupational exposure at the facility. The licensee will make every effort to mimimize exposure and, at the same time, provide the necessary fuel storage capacity. Consequently, we conclude that exposures will be as low as is reasonably achievable during spent fuel storage rack modifications.

3.8 Proposed Technical Specification Changes

The licensee has proposed changes to the Technical Specifications to provide additional assurance that the FESW sub-criticality requirements are not violated and that radiation exposure that could effect workers (occupational exposures) or the health and safety of the public is as low as reasonably achievable.

T.S. 2.12.3 would establish a limit of 16.6 grams/axial centimeter for Zircaloy clad fuel and 22.6 gms/axial centimeter for stainless steel clad fuel assemblies.

As discussed in Section 3.1 of this Safety Evaluation, the above limits will assure that the maximum neutron multiplication factor in the LACBWR (FESW) will not exceed k_{eff} of 0.95 which is acceptable.

T.S. 2.12.5 would require that fuel assemblies stored in the spent fuel storage racks be covered by at least 16 feet of water.

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As discussed in Section 3.4 above, 16 feet of water above the irradiated fuel assemblies provides adequate shielding for storing additional fuel in the FESW. With this restriction no significant increase in doses would be received by occupational workers. We also conclude in Section 3.6 above that offsite consequences due to an accident during fuel handling or spent fuel shipping cask movement are well within the radiation exposure guidelines of 10 CFR Part 100 and are therefore, acceptable.

T.S. Section 4.2.1.9 would require that the containment building be isolated if a spent fuel shipping cask is being moved near or within the FESW that contains spent fuel elements with less than 43 days of decay time. As discussed in Section 3.6.2 above, the calculated potential consequences of a spent fuel cask drop into the FESW are well within the radiation exposure guidelines of 10 CFR Part 100 relating to health and safety of the public and are, therefore, acceptable.

T.S. 4.2.8.4 would be a new provision requiring that at least 72 hours must elapse before fuel assemblies are removed from the core for storage in the FESW.

As discussed in Section 3.6.1 of this Safety Evaluation this restriction assures that the potential radioactive releases resulting from a fuel handling accident are within the guidelines of 10 CFR 100, and are, therefore, acceptable.

T.S. 4.2.8.5 would impose a new limit on objects that could be handled over the spent storage well with certain exceptions.

Section 3.6 of this evaluation considered the accidental dropping of a fuel element or a shipping cask in the FESW. It was concluded that neither of these events would cause radioactive releases and offsite consequences greater than the guidelines of 10 CFR Part 100. To assure that no other heavy objects are moved over the FESW that have not been evaluated, the new restriction will be imposed.

4.0 Summary

We have concluded that the proposed modification to the LACBWR FESW and the proposed changes to the Technical Specifications are acceptable because:

1) The calculated effective neutron multiplication factor (k_{eff}) of the fuel stored in the new storage racks meets our acceptance criterion.

2) Adequate cooling of the stored spent fuel will be provided by the existing FESW cooling system.

3) The mechanical and structural design of the new spent fuel storage racks, and FESW drain line meet appropriate criteria.

4) The expected increase in occupational radiation exposure to individuals due to the storage of additional fuel in the FESW is negligible.

5) The increase in plant liquid releases of radioactivity from pool leakage would be negligible compared with the limits of 10 CFR Part 20 and of the plant effluent Technical Specifications.

6) The potential consequences of the postulated design basis accident for the FESW, i.e., the rupture of the fuel pins in two fuel assemblies and the subsequent release of the radioactive inventory within the gap, are acceptable.

7) The likelihood of an accident involving heavy loads in the vicinity of the spent fuel pool is sufficiently small that no additional restrictions on load movement are necessary while our generic review of the issues is under way.

8) The removal of the existing racks and the installation of the new racks will be conducted in a manner that will not endanger the health and safety of the public and will result in occupational radiation exposure that is as low as reasonably achievable.

5.0 Conclusion

We have concluded, based on the considerations discussed above, that: (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, and (2) such activities will be conducted in compliance with the Commission's regulations and the issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public.

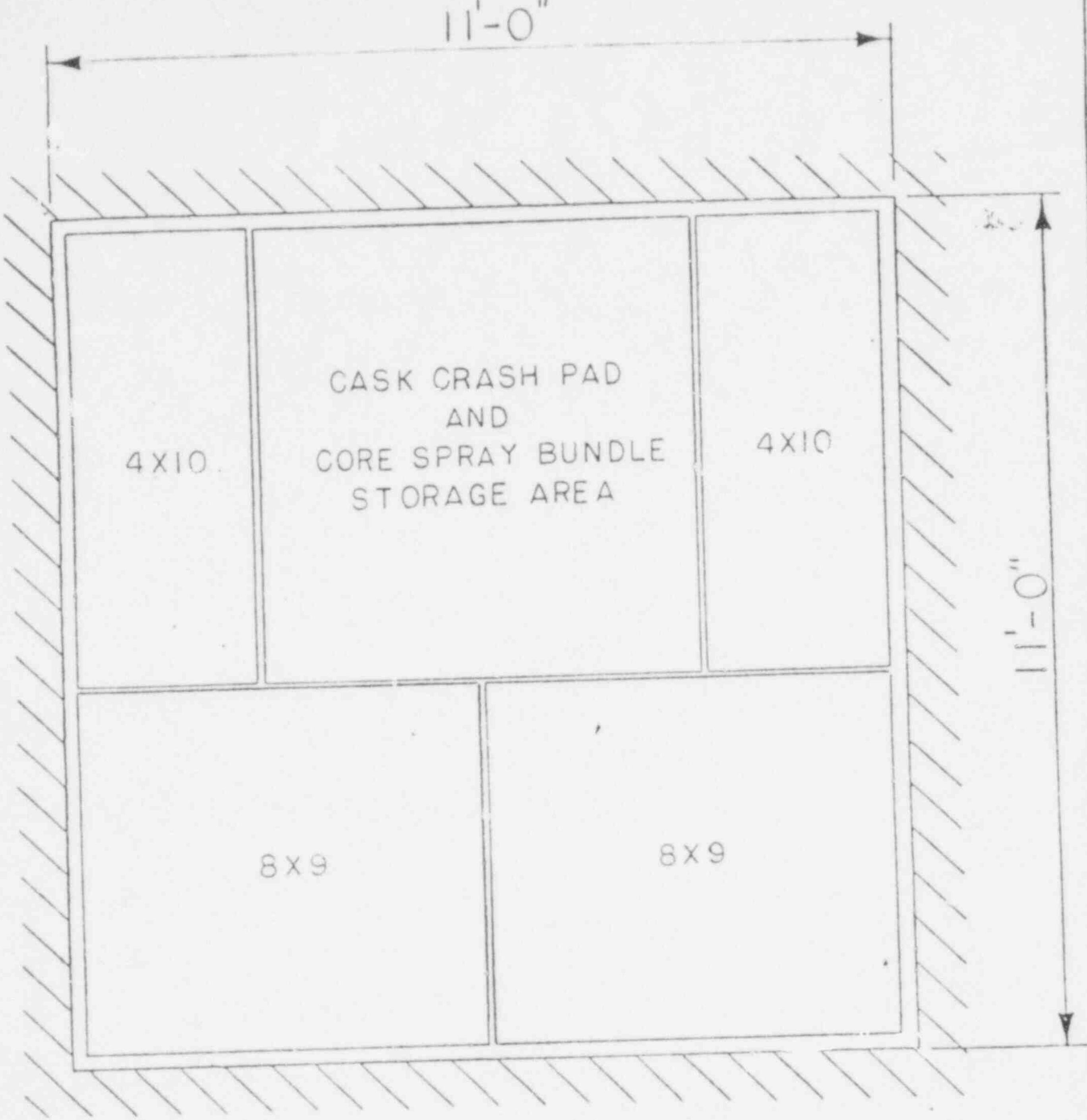
TABLE 1

ASSUMPTIONS USED IN EVALUATING

LACBWR FUEL HANDLING ACCIDENTS

Power Level	165 Mwt
Total Number of Fuel Rods in Core	7200
Number of Fuel Rods Damaged	200
Shutdown Time	72 hours
Radial Peaking Factor	1.5
Inventory Released from Damaged Rods	
Iodines and Noble Gases (not Kr 85)	10%
Kr 85	30%
Pool Decontamination Factors	
Iodines	*100
Noble Gases	1
X/Q Values, sec/m^3	
0 - 2 hours @ 1,109 ft	2.2×10^{-3}
0 - 8 hours @ 3 miles	3.8×10^{-5}
8 - 24 hours @ 3 miles	2.5×10^{-5}
24 - 96 hours @ 3 miles	9.1×10^{-6}
96 - 720 hours @ 3 miles	2.6×10^{-6}

*In accordance with Reg. Guide 1.25 for iodine that passes through 23 feet of water.



PLAN

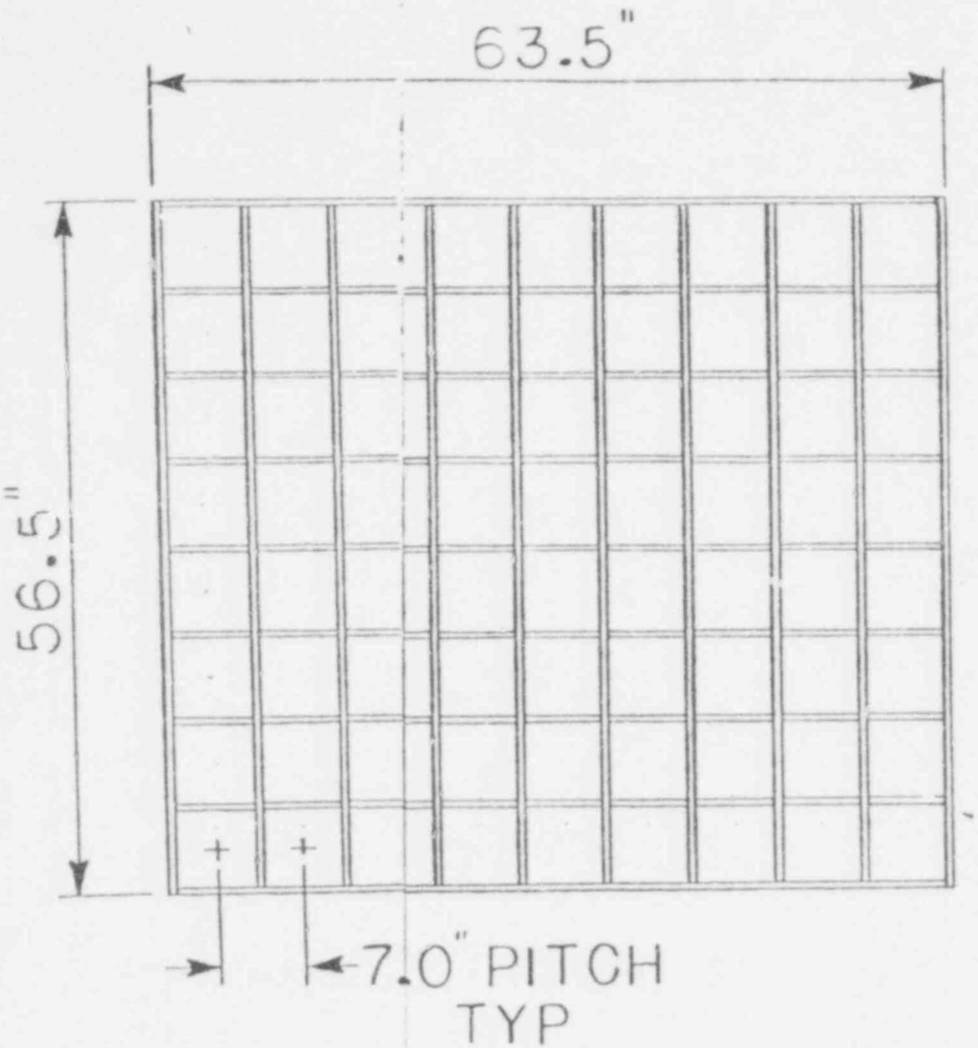
TWO TIER FUEL STORAGE RACK ARRANGEMENT

N

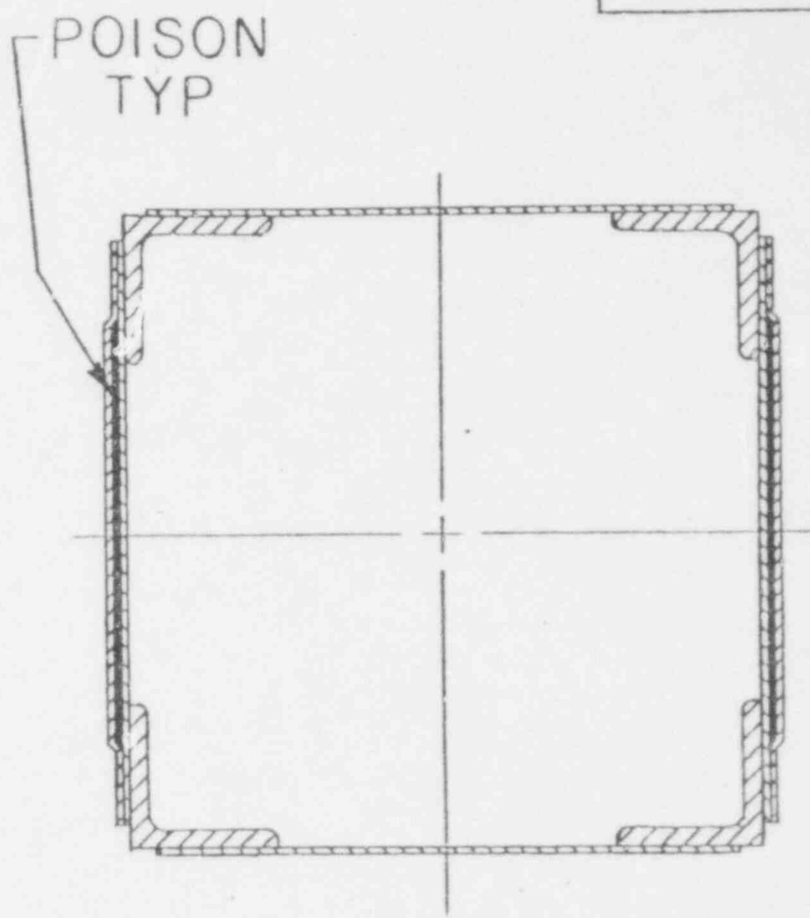
(LACBWR)

SK-5101-237-2

FIGURE 1



TOP VIEW



TYPICAL SECTION THRU FUEL STORAGE CELL

FIGURE 2

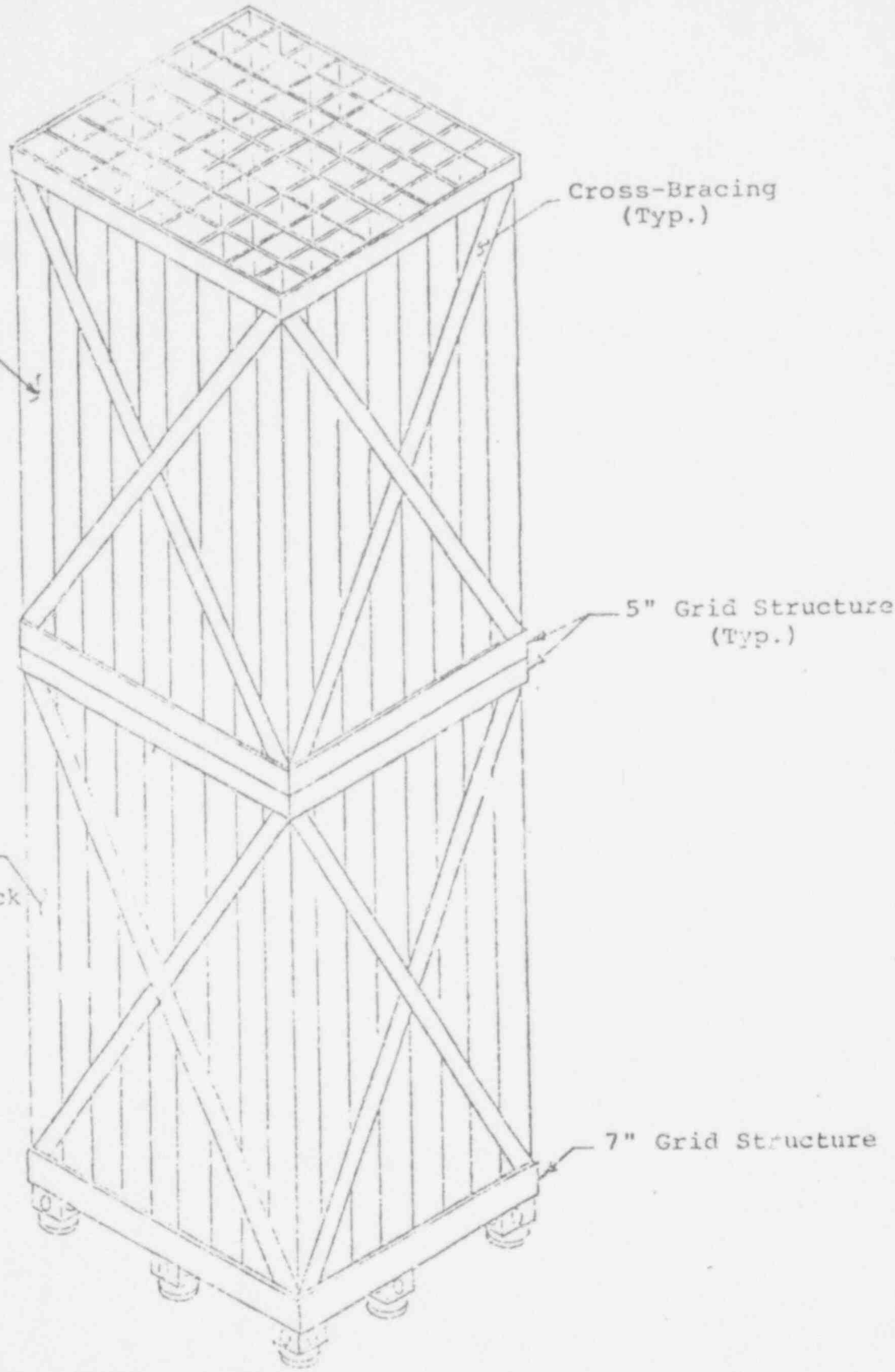


FIGURE 3

8x9 SPENT FUEL STORAGE RACK