Docket No. 50-255

CONSUMERS POWER COMPANY
PALISADES NUCLEAR GENERATING STATION

SPENT FUEL POOL MODIFICATION
DESCRIPTION AND SAFETY ANALYSIS

November 1976

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1.0 INTRODUCTION

Because of uncertainties in the availability of future fuel reprocessing facilities, Consumers Power Company (CPCo) plans to increase the storage capacity of the spent fuel pool at the Palisades Nuclear Generating Plant to prevent a shortage of spent fuel storage capacity. The proposed method of accomplishing this increase is to install high density spent fuel storage racks of smaller center to-center spacing between assemblies, utilizing neutron absorbing material to maintain the required margin of subcriticality.

The original plant design assumed a viable fuel reprocessing industry in the United States by the time the plant commenced operations. Therefore, the original spent fuel pool was sized to accommodate 1 2/3 cores with racks provided for 1 1/3 cores, the assumption being that the 1/3 core discharged each year would be transferred to a reprocessing facility prior to the next year's refueling. Therefore, the pool would always have the capability to accept a full core offload. However, there is not now and will not for the near term be a capability for reprocessing in the U.S. Therefore, additional spent fuel generated as a result of reactor operation cannot be disposed of and must be stored.

At the present time, one full core is being stored in the spent fuel pool. The plant is scheduled to be shut down for refueling in August 1977. This will necessitate the storage of an additional 1/3 core, completely filling the present storage racks. CPCo finds this condition to be unacceptable since it would not permit any core unloading should it be necessary. CPCo deems it necessary to increase the capacity of its spent fuel pool. Therefore, CPCo requests the approval of the NRC to increase the capacity of its spent fuel pool to 798 elements. This increase allows the storage of normal spent fuel until 1985 and retain a capability to offload a full core up to that time. This report discusses in detail the various design features incorporated in this modification and demonstrates they will have no detrimental effect on the health and safety of the public.

2.0 GENERAL DESCRIPTION

2.1 Present Design

As described in the FSAR, the spent fuel storage pool is located in the auxiliary building adjacent to the containment. The pool contains four spent fuel storage racks with a capacity of 276 spent fuel elements and one rack for storage of control rods. It is lined with stainless steel and has reinforced concrete walls and floor 4 1/2 to 6 feet thick.

The present fuel racks are stainless steel with a center-to-center spacing of 11 1/4 inches. There are two 1/4 inch stainless steel plates between each pair of fuel assemblies. At design temperature with no credit taken for soluble boron in the pool water, the maximum K_{eff} is less than 0.95. A recessed area is provided in the pool for a spent fuel shipping cask.

The fuel pool cooling system is a closed loop system consisting of two half-capacity pumps, a full-capacity heat exchange unit consisting of two heat exchangers in series, a bypass filter, a bypass demineralizer, a booster pump, piping, valves, and instrumentation.

The spent fuel pool cooling system has a heat removal capability of 23×10^6 BTU/hr. The spent fuel cooling system is conservatively designed to maintain pool average temperature at less than 125^oF with 1/3 core of fully burned up fuel in the pool, 36 hours after reactor shutdown. A single failure of the cooling system would increase pool temperature by only 3^oF. The water in the spent fuel pool is normally borated to 2000 ppm. The entire fuel pool cooling system is tornado-protected and is located in a Seismic Class 1 structure.

Fuel pool makeup water is supplied from the Safety Injection and Refueling Water (SIRW) tank. A secondary backup supply of water is available from the fire system. This would be utilized to replenish the fuel pool water inventory in the event of considerable loss of pool water.

The clarity and purity of the water in the spent fuel pool are maintained by passing a portion of the flow through the bypass filter and/or demineralizer. Skimmers are provided in the spent fuel pool to remove accumulated dust from the pool.

Connections are provided for a temporary tie-in to the shutdown cooling system to provide for additional heat removal in the event that a full core has to be unloaded into the pool. These connections also provide a backup capability for the fuel pool heat exchangers.

The fuel pool cooling system is connected by valved piping to the reactor refueling cavity for additional cooling of the reactor cavity water during spent fuel transfer.

Two fuel tilt pits are located in the fuel building adjacent to the spent fuel pool and connected to it by canals which are closed off by dam blocks. One tilt pit is used for normal fuel transfer activities. The second tilt pit was provided to accommodate an additional unit then being considered. This second pit was not lined at the time of construction. Presently, it is being lined with stainless steel to permit its use for additional spent fuel storage.

2.2 Proposed Modification

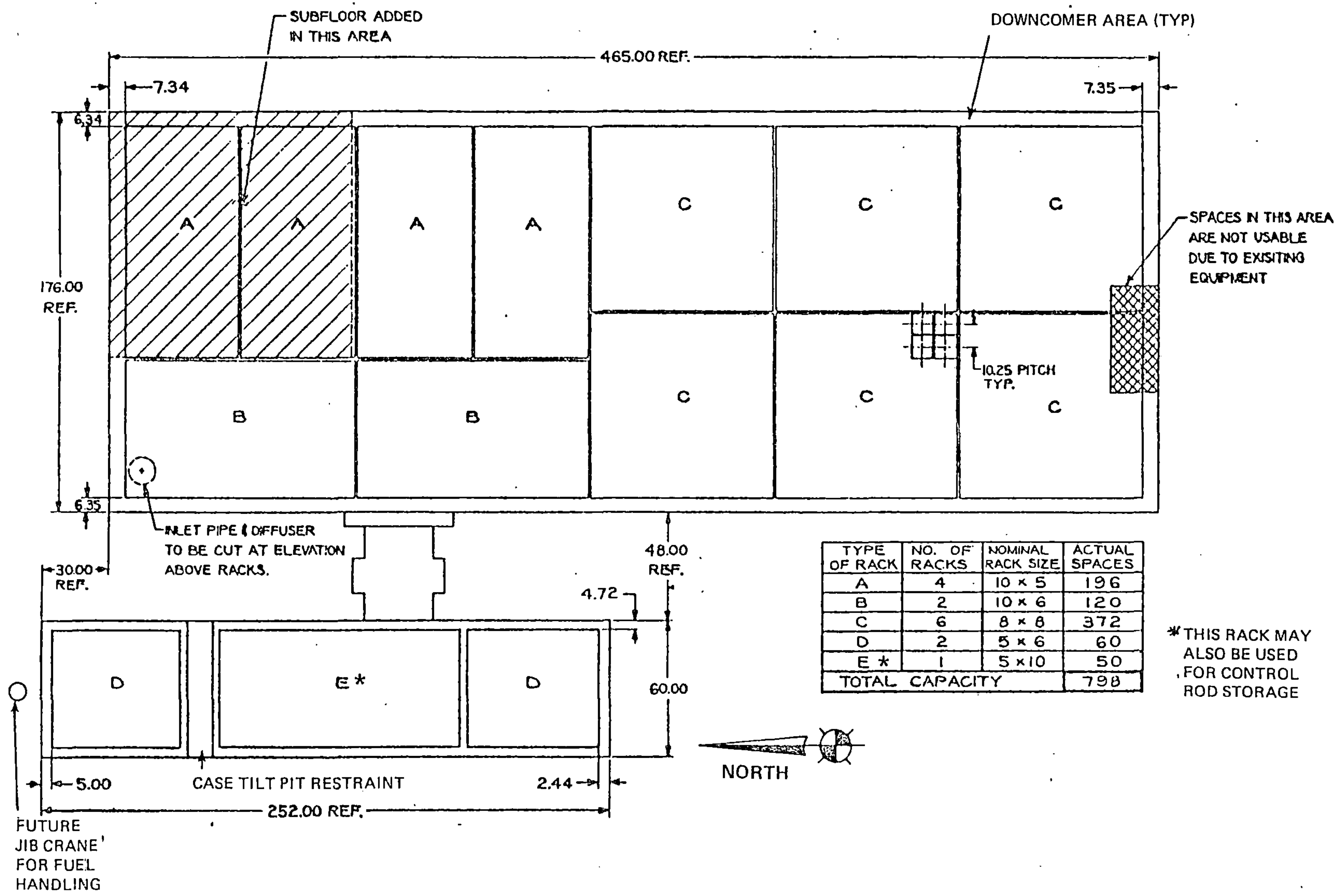
The proposed fuel rack modification, which conforms in all respects to Safety Guide 13 (USNRC RG 1.13), will involve removing the existing fuel and control rod racks and replacing them with new racks with smaller center-to-center spacing. Each individual storage location consists of two concentric 1/8" austenitic type 304 stainless steel square cans with the annular space occupied by B_4C neutron absorber plates to ensure subcriticality.

A rack assembly consists of a rectangular array of storage cans with a minimum 10.25 inches center-to-center spacing of the fuel assemblies. The array size of each rack was chosen to optimize use of pool space as shown in Figure 2-1. The expanded spent fuel storage capacity is 798 assemblies.

The new racks are seismic Category I and are restrained to the pool wall at the top and bottom of each rack to prevent excessive movement of the racks under postulated seismic accelerations. Provisions are made in the design to accommodate thermal expansion.

The present cask laydown area will contain two 50-element racks which will normally be used to store fuel during full core offloads. These two racks may be removed to allow placement of the spent fuel shipping cask or to allow the use of fuel inspection and repair equipment, etc., providing that the anti-tipping device is installed to provide seismic restraint. At the time that fuel shipment resumes, the two racks in the cask area may be removed and the cask anti-tipping device (described in Appendix J to the FSAR) will be installed in the pool. This device will provide anti-tipping protection and will act as a seismic restraint for the remiaining racks.

The presently unused tilt pit will be used for spent fuel and control rod storage and as an alternate cask laydown area. Control rods and fuel stored in the rack in this tilt pit use a rack design with slightly larger cans than those used in the other racks. A jib crane will be added to facilitate fuel handling in the tilt pit area. This is fully discussed in Sections 3.0 and 4.0. To minimize heat generation in the tilt pit, normally only fuel decayed for at least one year will be stored there. When fuel with a shorter decay time is stored in the tilt pit, thermal conditions will



TYPE OF RACK	NO. OF RACKS	NOMINAL RACK SIZE	ACTUAL SPACES
A	4	10 × 5	196
B	2	10 × 6	120
C	6	8 × 8	372
D	2	5 × 6	60
E *	1	5 × 10	50
TOTAL CAPACITY			798

Figure 2-1. Palisades Plant Spent Fuel Storage Rack Arrangement

will be monitored to insure that the criteria delineated in Section 6.3 are met.

The additional heat load due to the increased number of spent fuel assemblies can be accommodated by the spent fuel pool cooling system without modification except for addition of a direct cooling water supply to the tilt pit. This is discussed in detail in Section 6.0.

3.0 MECHANICAL DESIGN

3.1 Spent Fuel Storage Rack

Each fuel assembly will be stored in a concentric can roughly 12' long and with an inside square cross sectional length of 8.56". Each storage cell will consist of two concentric 1/8" cans with neutron absorber plates installed in the annular gap between the cans. The top and bottom of the two concentric cans will be closed with spacers and seal welded to provide a water tight annulus within which the neutron absorber will be held. A 1/4" diameter rod will be run the length of each corner of the annulus and welded in place to maintain the spacing between cans and to provide lateral support for the absorber plate. A 3/8" thick fuel support plate will be welded at the bottom of the can to provide support for the fuel. The plate will contain a 5" diameter hole to allow cooling water to flow upward through the fuel assembly to provide for removal of the decay heat from the fuel element. The plate will also contain four 3/4" holes to accept the two fuel assembly alignment pins. The top of each can will be flared slightly to facilitate fuel assembly insertion. The rack will be constructed of stainless steel with the exception of the B_4C absorber plates (see Figure 3-1).

3.2 Neutron Absorber

The neutron absorber plate is B_4C powder bonded together in a carbon matrix. The absorber is 50% B_4C by volume with the remainder being carbon and voids. Specifications for the B_4C powder used for the absorber plates will require that the median particle size be 125 microns by volume consistent with maintaining the criticality allowance for heterogeneity.

The absorber is fabricated in 0.21" (minimum) thick plates. These plates are inserted in the annular spaces formed by the concentric square cans. The B_4C plate is chemically inert in borated water

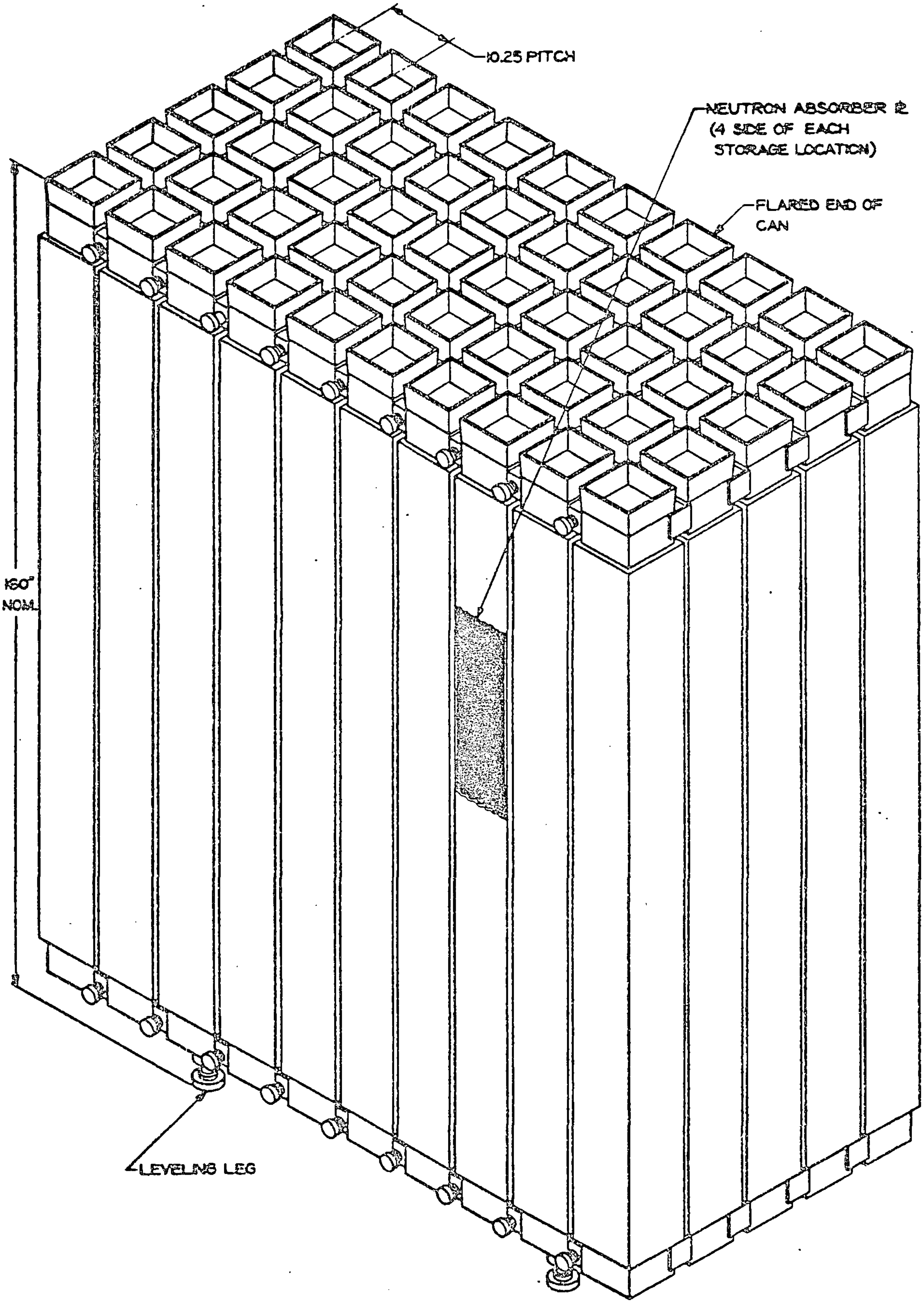


Figure 3-1. Palisades Plant Typical Fuel Rack Isometric

and is thermally stable under all temperatures expected in the pool. In the unlikely event of a postulated seismic event, or significant vibrations, the neutron absorber plates are restrained from shifting by the spacers and the concentric tubes. In the extreme case of the postulated seismic event and assuming the worst mechanical tolerances, there will be no settling of the neutron absorber material below the top of the active fuel. The B_4C plates are inserted and the spacers seal welded to form an envelope around the neutron absorber material. All these seal welds will be dye penetrant inspected. The neutron absorber material used in these racks is of the same type approved for the Connecticut Yankee Rack Modification (Docket 50-213).

3.3 Fuel Rack Assemblies

The assembled cans are formed into rack assemblies by attachment to rectangular stainless steel bars. The bars run horizontally near the top and bottom of the rack assembly forming a unitized lattice arrangement. Each can is continuously welded on all four sides to the lattice forming a single rigid structure, the rack assembly. There are five different sizes of rack assemblies, the size of each having been chosen to maximize the storage capacity of the pool (see Figure 2-1). The racks are:

Type	Spaces
A	10 x 5
B	10 x 6
C	8 x 8
D	6 x 5
E	10 x 5

In the main pool all the racks are similar, having 8.56" square inner cans and a 10 1/4" center-to-center spacing. In the tilt pit pool the E type rack designed for storage of control as well as fuel, has a 9" square inner can, and is arranged on a 10.25" by 11.25" center-to-center spacing. The two D racks in the tilt pit pool are similar to racks in the main pool.

Each of the racks is supported by four legs. Each of the legs has the capability of being adjusted to compensate for any tilt in the floor. The leg supports are located between fuel cells and are reinforced with gussets.

The racks in the main pool are restrained during a seismic event by compression-type restraints on the periphery of the array. These restraints, in two rows, one near the top and the other near the bottom of the rack assembly, will be set during installation to have clearances to accommodate expansion due to temperature changes of the pool water. The maximum gap between the restraints and the pool wall will be approximately 0.3" and will accommodate a temperature increase from 70° to 220°F.

The racks in the tilt pit are provided with lugs that will mate with keyways embedded in the pool side walls. These keyways will have a clearance to allow for thermal expansion. The keyway restraints transfer the seismic load to the east and west walls.

A jib crane will be installed adjacent to the tilt pit to facilitate fuel handling. The north side of the tilt pit can serve as an alternate cask laydown area if the 30 element rack is removed. This capability will also extend plant operation for one more core cycle if the need should arise.

Each rack is provided with four threaded holes at the top attached to the lattice that will allow the attachment of lifting eyes both for initial installation and, in the case of the A and D type racks so they may be removed to permit installation of the shipping cask, etc. The racks are designed only to be lifted while empty.

3.4 Codes, Standards, and Practices for Fuel Assembly Rack Design, Construction, and Assembly

The following are the codes, standards, and practices to which the fuel assembly racks will be designed, constructed, and assembled. (Revisions utilized are those in effect as of November 1, 1976)

1. Design Codes

   a. AISC specification for the design, fabrication and erection of structural steel for buildings, 1969, including supplements 1, 2 and 3.

   b. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plant Components (Tables I-7.0 and I-8.0 are used for allowable stress values for materials of construction).

2. Material Codes

   a. ASME Specification SA-240, Specification for Stainless and Heat-Resisting Chromium and Chromium-Nickel Steel Plate Sheet and Strip for Fusion-Welded Unfired Pressure Vessels.

   b. ASME Specification SA-320, Specification for Alloy Steel Bolting Materials for Low Temperature Service.

   c. ASME Specification SFA-5.9, Corrosion Resisting Chromium and Chromium Nickel Steel Welding Rods and Bare Electrodes.

3. Welding Codes

   a. ASME Boiler and Pressure Vessel Code, Section IX-1974 Welding and Brazing Qualifications.

4. Quality Assurance, Cleanliness, and Package Requirements

   a. RG 1.37 Quality Assurance Requirements for Cleaning of Fluid Systems and Associated Components of Water-Cooled Nuclear Power Plants, 3/16/73

   b. RG 1.38 Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants, 3/16/73

   c. RG 1.28 Quality Assurance Program Requirements - Design and Construction (Safety Guide 28), 6/7/72

   d. RG 1.88 Collection, Storage, and Maintenance of Nuclear Power Plant Quality Assurance Records, Rev 2, 10/76

   e. RG 1.64 Quality Assurance Requirements for the Design of Nuclear Power Plants, Rev 2, 6/76

   f. ANSI N45.2 Quality Assurance Program Requirements for Nuclear Power Plants, 1971

   g. ANSI N45.2.9 Requirements for the Collection, Storage, and Maintenance of Quality Assurance Records for Nuclear Power Plants, 1974

   h. ANSI N45.2.11 Quality Assurance Requirements for the Design of Nuclear Power Plants, 1974

i. ANSI N45.2.12 Requirements for Auditing of Quality Assurance Programs of Nuclear Power Plants (draft)

j. ANSI N45.2.13 Quality Assurance Requirements for Control of Procurements of Equipment, Materials and Services for Nuclear Power Plants, 1976

3.5 Absorber Loading Verification

In order to assure that the neutron absorbing material is loaded as assumed in the criticality analysis, a verification procedure for absorber loading will be conducted. Each partically fabricated concentric can will be weighed prior to insertion of the absorber material. After the B_4C plates are inserted and just prior to welding the assembly to the base, the completed assembly will again be weighed. The incremental increase in weight will be compared to a predetermined reference to provide a confirmation of the absorber material loading.

In order to assure that no neutron absorber plates have been omitted during rack construction, a site quality control program will be implemented for each rack. Randomly selected cans based on statistical sampling methods will be tested using a neutron source. Differences in count rates will identify the absence or presence of B_4C plates.

3.6 Fuel Rack Installation Procedure

New fuel racks will be installed in the present cask laydown area and in the tilt pool. The existing fuel will be transferred to these new racks. The old racks will then be removed and the new ones installed using a detailed written procedure designed to preclude any possibility of dropping a rack on the stored fuel elements. This procedure will be approved in advance by the Plant Review Committee (PRC).

Fuel rack handling will be made with the existing crane facilities. Crane movement will be controlled by written administrative procedures which will prohibit the movement of spent fuel racks or control rod racks directly over locations in the pool where fuel assemblies are being stored.

3.7 Material Compatibility

Because the replacement racks, their associated hardware and the seismic restraints are of all stainless steel construction, as is the spent fuel pool liner, there is no potential for galvanic corrosion. Material compatibility between the spent fuel pool and the new storage racks and between the fuel assemblies and the new storage racks is also not a problem as stainless steel has been shown to be compatible with both fuel assemblies and spent fuel pool water.

4.0 CRITICALITY CONSIDERATION

The racks in the main pool are designed for a 10.25 inch center-to-center spacing with B_4C plates around each assembly. The results of the criticality analyses are as follows:

1. The center-to-center spacing of 10.25 inches between fuel assemblies with neutron absorber surrounding each fuel assembly results in a k_∞ of 0.872 under nominal conditions.

2. The worst case situations, considering maximum variations in the position of fuel assemblies within the storage rack, neutron absorber positioning, variations in can dimensions, the most reactive temperature, calculational uncertainties and worst case accidents result in a k_∞ of 0.924 with a confidence level of 95%.

4.1 Assumptions and Method of Analysis

The referenced set of calculations were based upon the following assumptions:

1. Fresh fuel of 3.05 weight % U-235 nominal average enrichment

2. Water temperature of 68°F

3. No credit taken for soluble poison

4. Fuel racks are infinite in three dimensions

5. Control rods and other fixed poisons are not present in the fuel assembly.

The majority of the calculations were performed with methods commonly employed in light water reactor design, i.e., four-group diffusion theory cell calculations using PDQ-07. The cross sections for these calculations are generated with NUMICE, the NUS version of LEOPARD. This code uses the same cross section library tape and calculational techniques as LEOPARD. The cross sections for the poison are generated using "blackness theory"[(1)] routines available in NUMICE. This is a well-established technique for treating slab absorbers in diffusion calculations and has been used in previous poisoned rack analyses approved by the NRC, specifically Connecticut Yankee.

Selected cases were checked and the final design multiplication factors were verified with Monte Carlo criticality calculations using KENO with 123-group cross sections. The 123-group cross section library is generated from the basic GAM-THERMOS library using XSDRN (P_3, S_8).

4.2 Results of Analysis

Figure 4-1 shows the geometry of the fuel rack used in the referenced design calculations. Section 4.3 summarizes the results of both the diffusion theory and Monte Carlo calculations. In general, the four-group PDQ diffusion calculations produce k_∞ values about 0.015 lower than the Monte Carlo calculations.

Calculational uncertainties in the use of PDQ with cross sections based on the LEOPARD library have been obtained by comparing the results of a series of benchmark calculations with critical experiments. These comparisons[(2)] have shown that the average

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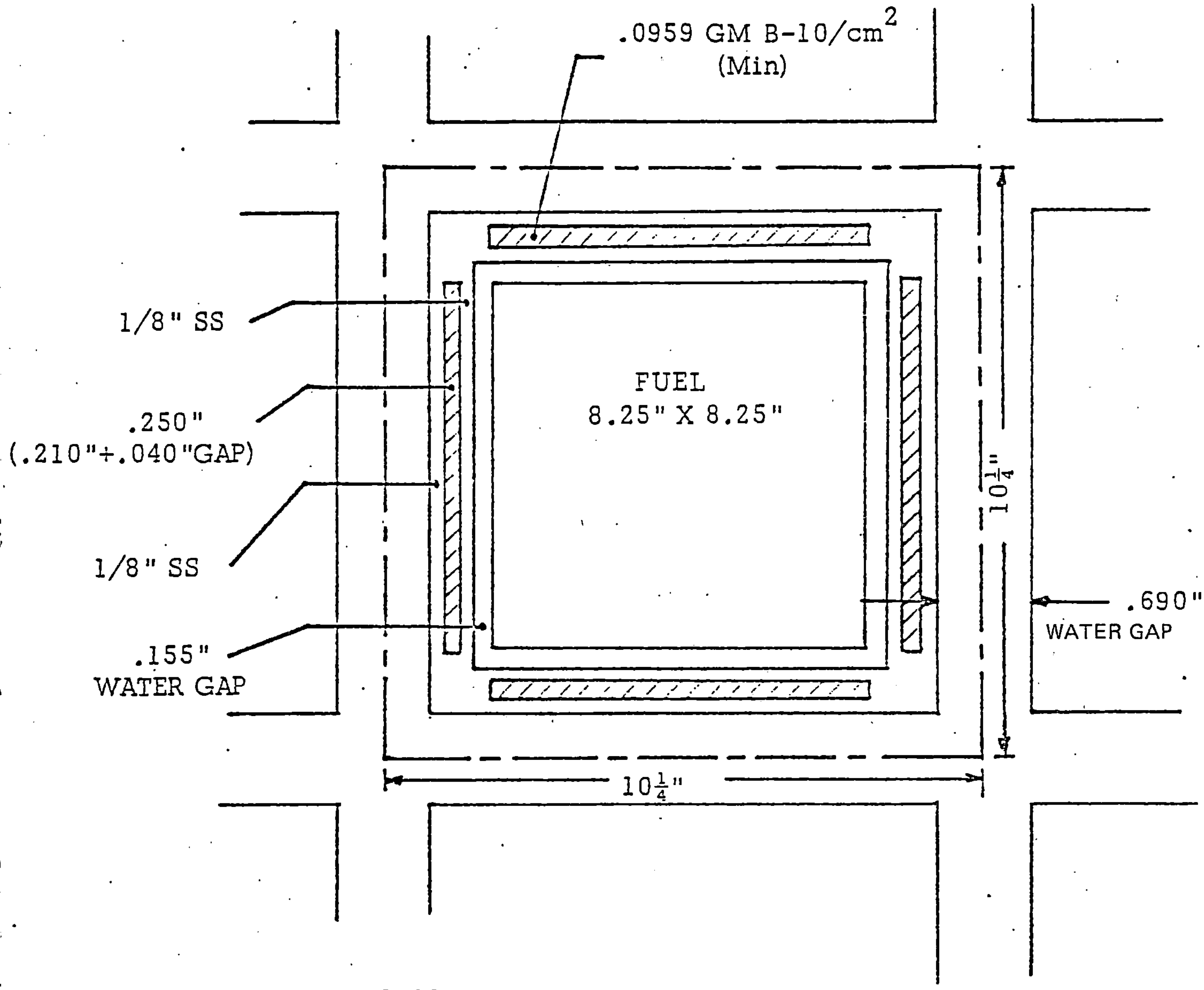
(1) WAPD-218 "A Theoretical Method for Determining the Worth of Control Rods" by A. F. Henry, Bettis Atomic Power Division, August, 1959.

(2) WCAP-3269-25 "Calculation of Lattice Parameters and Criticality for Uranium Water Moderated Lattices" by L. E. Strawbridge, Westinghouse Electric Corporation, September 1963.



B$_4$C WIDTH: 8.26"

FUEL LOADING: 37.55 GM U-235/cm HEIGHT (nominal)

38.30 GM U-235/cm HEIGHT (Tolerance)

Figure 4-1. Geometry of the Reference Storage Cell difference between the calculations and experimental results was 0.009 Δk. The KENO code, using the 123-group GAM-THERMOS cross section library, has been extensively benchmarked also.

For a series of ten critical experiments reported[3] the average k_{eff} as calculated using KENO and 123-group cross sections was 0.9914 $\pm$ .0020. Using the same method, NUS has performed another benchmark on one of the Yankee critical experiments[4] with Ag-In-Cd cruciform control rods banked at 26.37 cm from the bottom of the fuel. The calculated k_{eff} was 1.008 $\pm$ 0.006. On the basis of the above comparisons with criticals, a calculational uncertainty of 0.008 Δk was assigned to the KENO calculations.

Also, statistical analysis of the Monte Carlo results shows a standard deviation of $\pm$ 0.004, giving a 2σ uncertainty of 0.008 Δk_∞. Thus, an additional 0.008 Δk uncertainty is assigned to the KENO calculations.

The worst case criticality condition was obtained by using the maximum tolerances for the positioning of the fuel assemblies within the storage can as well as the relative can-to-can positioning. The rack tolerances are calculated on an overall rack width basis, such that cumulative tolerances between cans are accounted for. The calculation was performed at a water temperature of 68°F. The minimum boron carbide content of the neutron absorber plates is specified to be 0.0959 gm B-10/cm^2 plate based on a 0.21 inch thickness. Production variations are expected to provide up to 10% increase in B-10 content. This would result in a decrease in k_∞ ; no credit for this effect has been taken.

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(3) "Validation of Monte Carlo Calculations of Shipping Cask Systems" by L. M. Petrie and P. G. McCarty, ORNL, CONF 731101-14, 1973.

(4) "Yankee Critical Experiments - Measurements on Lattics of Stainless Steel Clad Slightly Enriched Uranium Dioxide Fuel Rods in Light Water" by P. W. Davison, et al., YAEC-94 April, 1959, page 82.

The unlikely case of one absorber plate missing from one side of one can in a group of 25 storage cells was calculated. The results show that the increase in reactivity is 0.002 Δk_∞ (PDQ analysis).

4.3 Worst Case Analysis of Tolerances and Calculational Uncertainties

The following are the results of the KENO analysis of the worst case of tolerances and calculational uncertainties:

Nominal Conditions, k_∞	0.872
Enrichment, 3.05%	
Mechanical Spacing, 10.25"	
Pool Temperature, 68°F	
B_4C Particle Self-Shielding, Δk_∞	0.004

Worst Tolerances, Δk_∞	
Enrichment, 102% of nominal	0.004
Loss of Poison	0.002
Mechanical Tolerances	0.023
Pool Temperature (40°F)	0.003
Worst Tolerances	0.032*

*An algebraic sum overestimates the effect of combining such tolerances. The root mean square of the first three tolerances, plus the pool tolerance, may be more appropriate, which yields a total tolerance effect of approximately 0.026.

Calculational Uncertainties, Δk_∞

KENO Benchmark	0.008
Statistics (2σ)	0.008
Total Calculational Uncertainties	0.016

Maximum, k_∞

Nominal, k_∞		0.872
B_4C Particle Self-Shielding, k_∞	=0.004	
Worst Tolerances	0.032	
Calculational Uncertainties	0.016	
	0.052	
MAXIMUM, k_∞		0.924

4.4 Parametric Studies

The base case, as established in the preceding sections, refers to the rack design with 10.25 inch spacing, 3.05 w/o nominal enrichment, 0.0959 gm B-10/cm^2 plate B_4C loading and 68°F pool water temperature. The k_∞ of the base case is 0.872 based on the 123-group KENO calculation. Parametric studies were performed to determine the effect on k_∞ of varying the base case conditions one at a time. The results are presented below:

1. k_∞ vs. Center-to-Center Spacing (KENO)

	10.125"	+0.0125 Δk_∞
(Nominal)	10.250"	(Base)
	10.375"	-0.0125 Δk_∞

2. k_∞ vs. Enrichment (PDQ)

(Nominal)	3.05 w/o U235	(Base)
(102%)	3.11 w/o U235	+0.0036 Δk_∞
(110%)	3.36 w/o U235	+0.0173 Δk_∞

3. k_∞ vs. B-10 Loading (PDQ)

	0.0850 gm B-10/cm^2	+0.0039 Δk_∞
(Nominal)	0.0959 gm B-10/cm^2	(Base)
	0.1050 gm B-10/cm^2	-0.0039 Δk_∞

4. k_∞ vs. Water Temperature (PDQ)

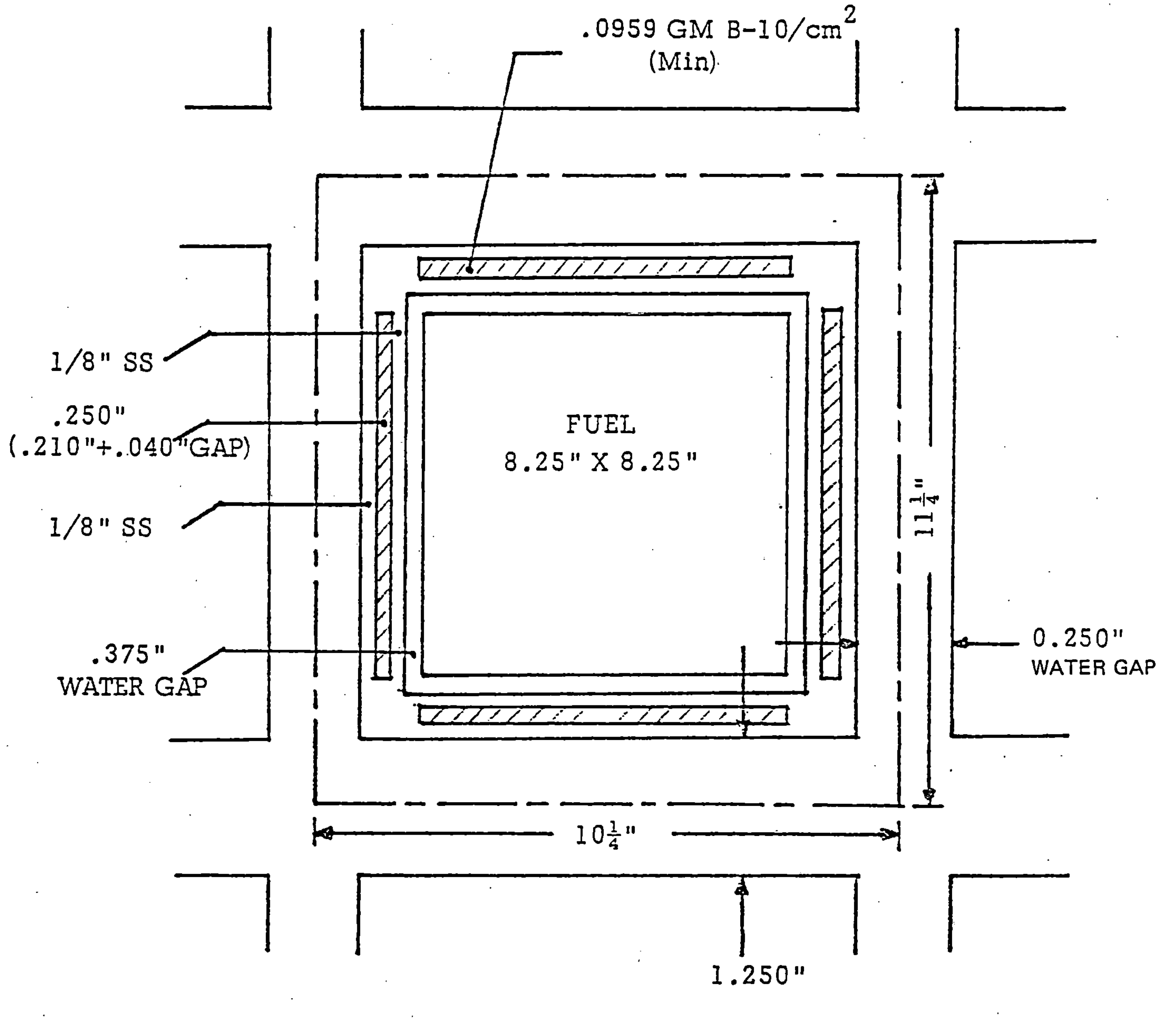
	40°F	+0.0031 Δk_∞
(Nominal)	68°F	(Base)
	100°F	-0.0023 Δk_∞
	212°F	-0.0243 Δk_∞

4.5 Accident Reactivity Analysis

Two fuel handling incidents were analyzed: (a) a fuel assembly drop during spent fuel handling landing horizontally on top of storage racks; (b) a fuel assembly inadvertently brought alongside the outer periphery of the storage racks in the vertical position if racks are removed (see Section 3.3). For the case (b) accident, which is the worst, the k_∞ of the pool was determined (PDQ) to be about 0.001 Δk_∞ higher than the nominal k_∞. Thus, this situation affects reactivity only slightly.

4.6 Tilt Pit Reactivity

The tilt pool contains a 50-storage cell special rack to accommodate storage of control rods and fuel assemblies. The geometry of each cell in this special rack is presented schematically in Figure 4-2. On each side of the special rack there is a 30-storage cell rack incorporating the main rack design shown in Figure 4-1. Criticality calculations under the assumptions and with the KENO methods described in Section 4.1, showed that the E-type rack nominal infinite multiplication factor is 0.883.



B_4C WIDTH: 8.26"

FUEL LOADING: 37.55 GM U-235/cm HEIGHT (Nominal)

38.30 GM U-235/cm HEIGHT (Tolerance)

Figure 4-2. Geometry of the Tilt Pool Rack Design

5.0 STRUCTURAL ANALYSIS

5.1 Loads & Loading Criteria

In accordance with Regulatory Guide 1.13, the spent fuel storage racks were designated Seismic Category I. Structural integrity of the fuel racks when subjected to normal and abnormal loads, as as well as the DBE, is demonstrated with respect to the NRC Standard Review Plan Section 3.8.4. In accordance with this Review Plan, the following loads, load combinations, and structural acceptance criteria were considered:

5.1.1 Loads

a. Normal Loads

i. Dead Loads - dead weight of rack and fuel assemblies and hydrostatic loads

ii. Live Loads - effect of lifting empty rack during installation

iii. Thermal Loads - uniform thermal expansion of racks due to change in average pool temperature from 70-220^{o}F and a thermal gradient between adjacent storage boxes of 35^{o}F.

b. Severe Environmental Load - Operating Basis Earthquake (OBE)

c. Extreme Environmental Load - Design Basis Earthquake (DBE)

d. Accidental drop of a spent fuel assembly from a height consistent with fuel handling operations which is 15 inches above the top of the racks.

e. Postulated stuck fuel assembly which causes an upward force of 1750 lbs, which is the limit switch setting on the crane.

5.1.2 Load Combinations

The spent fuel storage racks were analyzed using elastic working stress design methods for the following applied loads:

a. Dead Loads Plus Live Loads

b. Dead Loads Plus OBE

c. Dead Loads Plus Thermal Loads Plus OBE

d. Dead Loads Plus Thermal Loads Plus DBE (SSE)

e. Dead Loads Plus Fuel Assembly Drop

f. Dead Loads Plus Stuck Fuel Assembly.

Live loads were not included in load combinations b. through f., since the only live load on the rack was that due to lifting, and lifting of the racks was performed with the racks empty.

5.1.3 Structural Acceptance Criteria

The following were the strength limits for each of the above load combinations:

Load Combination	Strength Limit
a.	1.0 S
b.	1.0 S
c.	1.5 S
d.	1.6 S

Load Combination	Strength Limit
e.	1.6 S (except as noted below)
f.	1.6 S (except as noted below)

where S was the required section strength based on the elastic design methods and the allowable stresses defined in Part 1 of the AISC "Specification for the Design, Fabrication and Erection of Structural Steel for Buildings," February 12, 1969, included Supplement Numbers 1, 2 and 3. (Supplement 3 was effective June 12, 1974.) For load combinations e. and f., local stresses might exceed the limits, provided there was no loss of function of the fuel rack.

5.2 Seismic Analysis

The individual fuel racks described in Section 3.0 will be of all-welded construction. The main pool racks will rest on the floor and butt against one another at the top and the bottom. At the perimeter of the pool there will be clearance between the pool wall and the upper and lower supports sufficient to allow for thermal expansion of the racks. The tilt pit racks will rest on the floor and will be keyed to the pool walls.

The seismic loading of a typical fuel rack (See Figure 5-1) was determined from a response spectrum modal dynamic analysis, in which the stiffness of the fuel assembly was neglected. However, the mass of the fuel assemblies and an effective mass of water were considered to be uniformly distributed along the storage tubes. The appropriate floor response spectra at 4% damping for the OBE and DBE were employed. The FSAR indicated the use of 2% damping for steel framed structures, and this has been increased 2% to account for the additional damping due to the surrounding water, as recommended by Newmark and Rosenbleuth in Fundamentals of Earthquake Engineering.

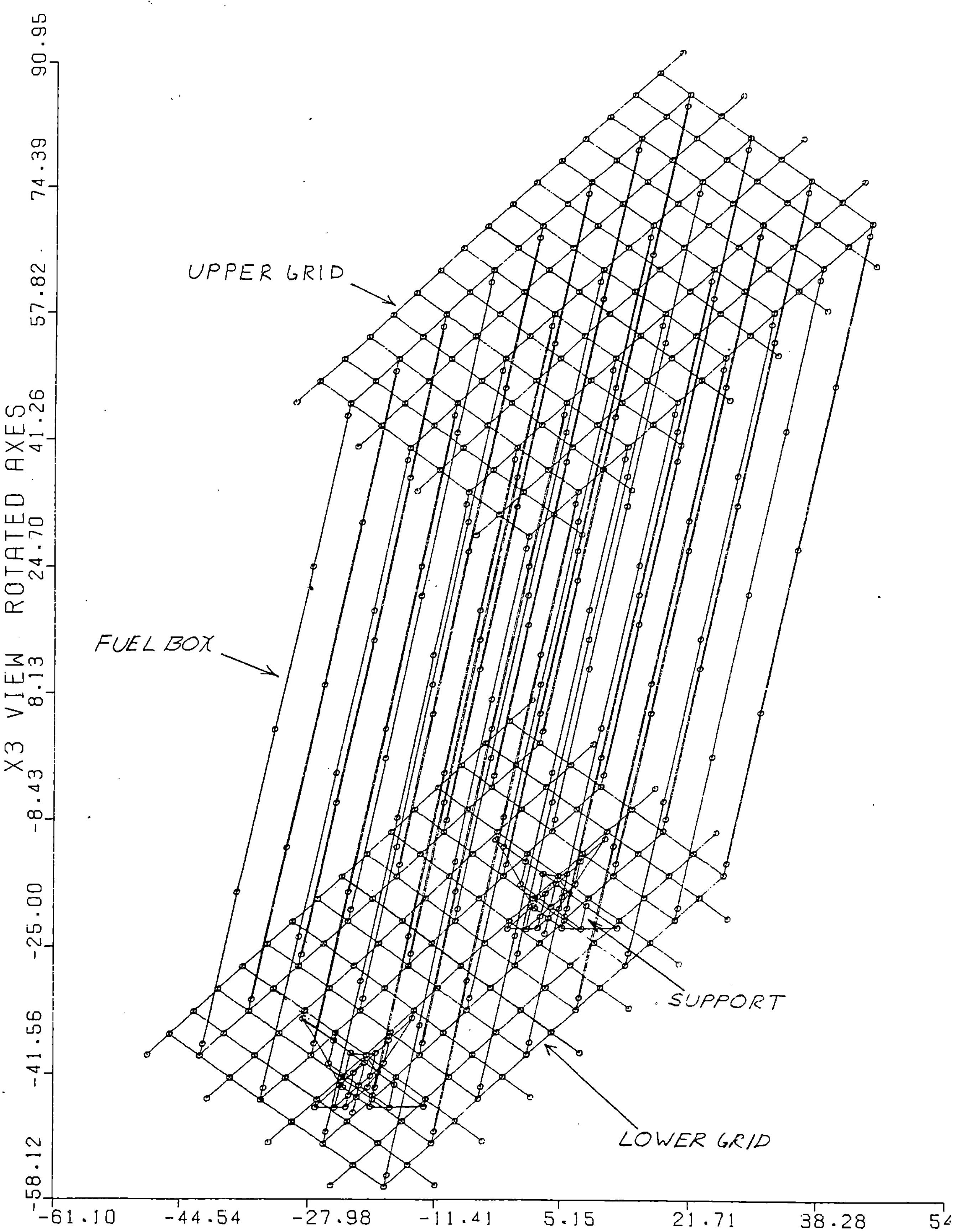


Figure 5-1. Palisades 8 x 8 Fuel Rack Symmetry Model
(One-half Rack)

The ground motion response spectra used for design were based on the work of Housner, commonly called the Housner Spectra. For developing the floor response spectra used in the design of spent fuel racks, a ground motion time history was used as input to a lumped mass model. Specifically, the time history record of the 1952 Taft event was used for this purpose.

As a check on the adequacy and conservativeness of the Taft time history as a seismic design input, a comparison was made of the spectra associated with the Taft time history with the Housner spectra. This was done for a 7 1/2 percent damping, a value consistent with the anticipated response of the concrete structure, as well as 4 percent damping, the value to be used in the design of the fuel racks which are primarily steel frames.

The Taft response spectra, given in Figures 5-2 and 5-3, was obtained by exciting a single-degree-of-freedom oscillator by the Taft time history for a duration of 30 seconds and determining the peak oscillator response in the forced vibration range at a total of 78 frequency points. The first 74 points, representing the frequency range from 0.03 to 33 hertz, employed the frequency increment criteria normally used in modern nuclear plant design. In addition, four more frequency points were added beyond 33 hertz to cover the frequency range up to 49 hertz.

Figures 5-2 and 5-3 also show the corresponding Housner Spectra for 7 1/2 percent and 4 percent damping, respectively, as taken from the FSAR.[5] From Figure 5-2, it is clear that the response spectrum from the Taft time history envelopes the Housner response spectrum used for design at all frequencies of interest to the structure. Also, from Figure 5-3, it is seen that the response spectrum associated with the Taft time history adequately envelopes the design response spectrum at 4 percent damping. More

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(5) "Final Safety Analysis Report, Consumers Power Company, Palisades Nuclear Power Plant."

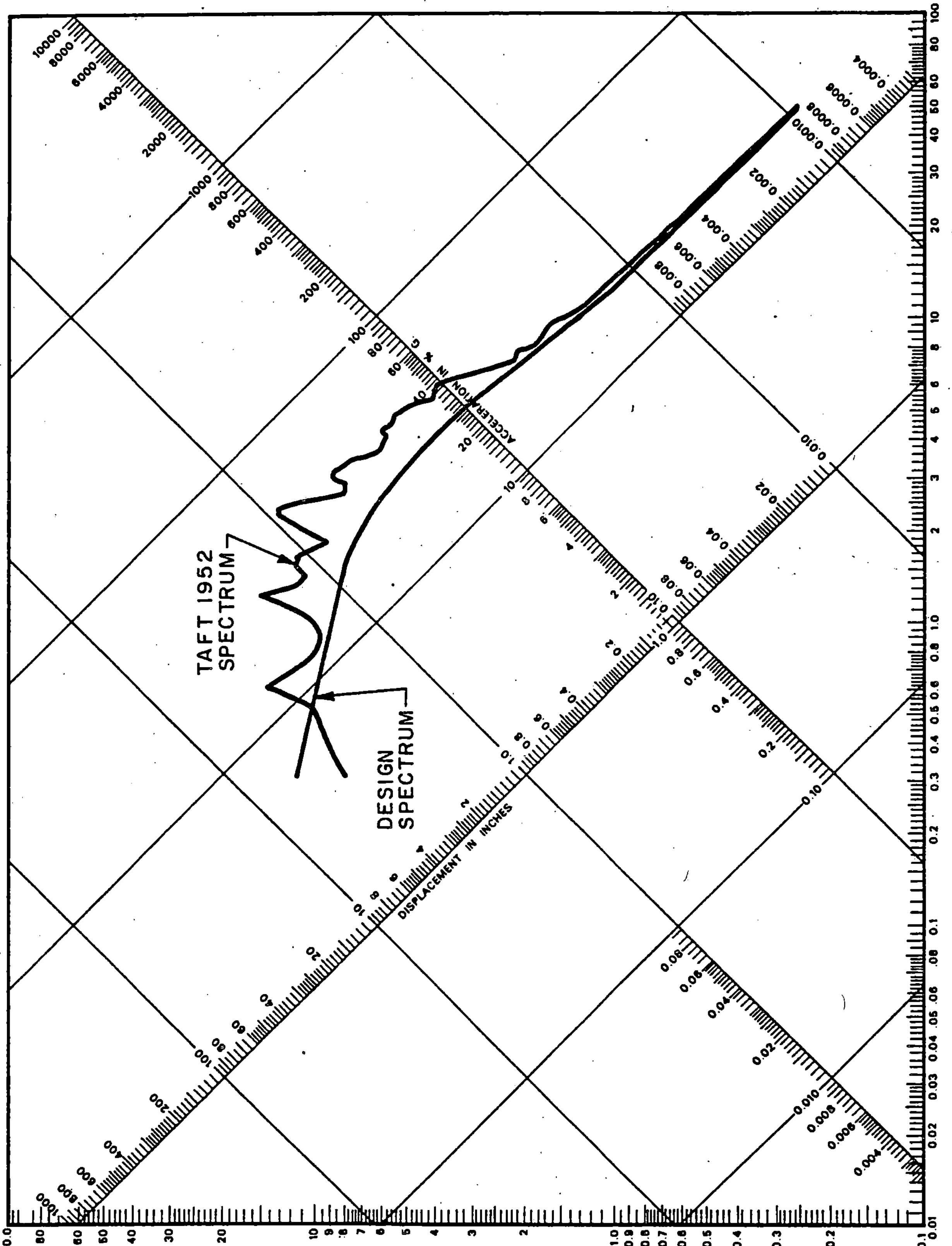

Figure 5-2. Ground Response, Housner and Taft, 1952
(7 1/2% Damping)

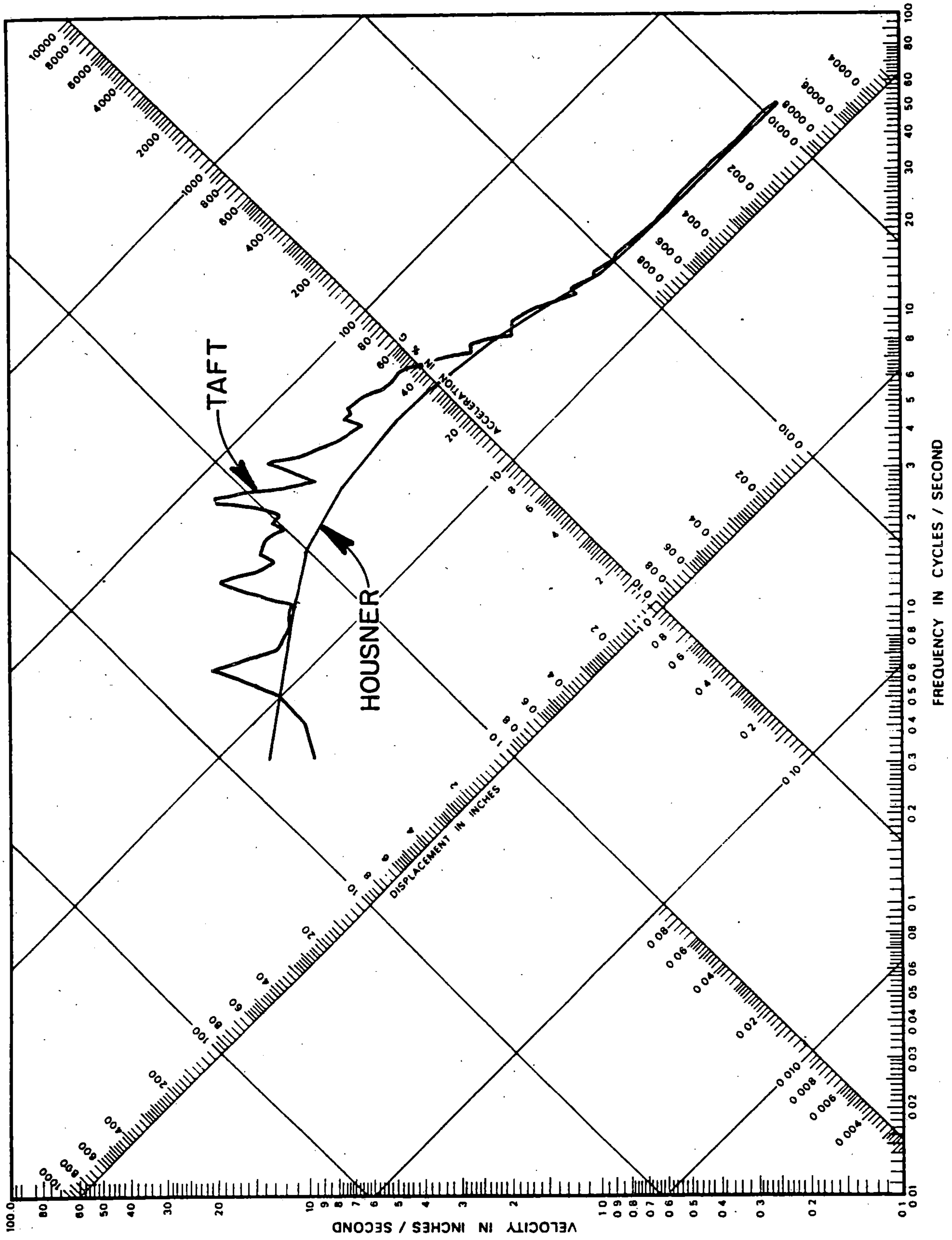

Figure 5-3. Palisades Plant Taft and Housner Spectra at 4% Damping

importantly, the computed response spectrum envelopes the design response spectrum at frequencies in excess of 9 hertz. Since the fundamental frequency of the racks is greater than 9 hertz, the use of the Taft time history to generate floor response spectra is conservative and suitable for use in the design of the spent fuel racks.

The STARDYNE computer program was used to perform the structural analysis of the racks. Storage racks were modeled in detail using beam and plate finite elements. The three-dimensional finite element model for the 8 x 8 rack in the main pool is shown in Figure 5-1. A similar model was used for the tilt pit rack. Two racks were analyzed in detail, an 8 x 8 rack in the main pool and the 5 x 10 rack in the tilt pit pool. These were considered representative of the rack design.

In the general seismic/structural analysis of the fuel racks, the mass of a fuel assembly was assumed to be uniformly distributed along the length of each of the fuel storage cans. This assumption was conservative in that lower rack fundamental frequencies were calculated which, due to the relatively stiff rack design, result in higher seismic amplified acceleration loading on the rack. Since a gap on the order of 1/8" will exist between the sides of a fuel assembly and the can, the fuel will move within the can during a seismic event. The effect of this motion, termed fuel-can interaction, was analyzed using the ANSYS computer program. A nonlinear dynamic analysis of a single can and fuel assembly was performed to determine the maximum shear force and bending moment which might occur at critical sections of the can as a result of the fuel assembly impacting the can at maximum velocity. The can and fuel assembly were modeled by beam finite elements and are separated by nonlinear gap elements. The can was restrained at the upper and lower grid elevations. The fuel, which was assumed to be pinned at its base, was given an initial velocity relative to the can. Impact loads were determined as a function of time.

5.3 Structural Adequacy

Using the previously listed loads and load combinations, stresses will be calculated at critical sections of the rack. The results of the structural and seismic analyses will be used to demonstrate that the spent fuel racks will be structurally adequate and will meet the design criteria.

5.4 Pool Wall and Floor Loading

The ability of the fuel pit and tilt pit floors and walls to withstand the loads imposed by the new fuel racks will be detemined as follows. The seismic loads and sloshing pressures will be combined with the static, live, and thermal loads in accordance with the load combinations given in Appendix A to the Palisades FSAR. Moments and shears will be calculated and compared to yield capacities based on a cracked section analysis using the material properties and yield capacity reduction factors as shown in the Palisades FSAR.

6.0 COOLING CONSIDERATION

6.1 General

As discussed below, analyses were performed of the existing spent fuel cooling and fuel pool cleanup systems. It was concluded that the present installed systems provide sufficient capacity and redundancy to accommodate the proposed increase in spent fuel loading. Table 6-1 defines the performance of the system under various single active failure conditions.

6.2 Cooling System Performance

The adequacy of the cooling system was analyzed in view of the expanded fuel storage capacity. The decay heat generation rates due to the spent fuel were calculated using the ORIGEN computer code developed at Oak Ridge National Laboratory. ORIGEN is a point depletion code which solves the equations of radioactive buildup and decay for large number of isotopes.

Two design conditions were evaluated; (1) normal refueling of 1/3 of a core, and (2) full core offload.

The heat load for the normal refueling case was based on the following projected sequence of fuel movement:

1. 205 fuel assemblies in the pool as of January 1976.
2. 456 additional fuel assemblies accumulated in successive refuelings by 1984.
3. 68 fuel assemblies (1/3 core) offloaded 36 hours after reactor shutdown in 1985.

The above refueling description results in the total of 729 fuel assemblies in the pool which effectively fills all available

TABLE 6-1

PALISADES PLANT - SPENT FUEL POOL COOLING SYSTEM (SFPCS) SINGLE ACTIVE FAILURE ANALYSIS

Component	Failure Mode	Consequence: Normal Refueling Heat Load 16.9×10^6 BTU/hr	Consequence: Full Core Offload Heat Load 26.4×10^6 BTU/hr
Spent Fuel Cooling Pump	Mechanical Failure	Second pump is operational. Complete inventory of spares available for rapid pump repairs, if needed. However, maximum pool temperature will reach 118°F.	Second pump is operational. Shutdown cooling system is available and can be put into operation by manual connections between the two systems. Without the shutdown system, the pool temperature will reach 134°F. With the shutdown cooling system also in operation, pool temperature will be reduced to 103°F.
Low Pressure Safety Injection (LPSI) Pump (Shutdown Cooling Pump)	Mechanical Failure	Shutdown system is not required for normal cooling. No effect on SFPCS.	The second LPSI pump and two shutdown heat exchangers, and the SFPCS are available for cooling. Pool temperature will be less than in the case of ESP Bus 1c failure.
Component Cooling Water (CCW) Pump	Mechanical Failure	No effect on SFPCS, as two 2/3 capacity CCW pumps are available.	No effect on SFPCS or on shutdown cooling system, as two 2/3 capacity CCW pumps are available.
Offsite Power	Electrical Failure	Emergency Power is available. Manual starting of SFP and CCW pumps is possible. Hence, no effect on SFPCS performance.	Emergency power is available. Manual starting of SFP pumps, CCW pumps and LPSI pumps is possible. Hence, no effect on SFPCS and on shutdown cooling systems.
Offsite Power and Diesel Generator No. 1 (ESF Bus 1c)	Electrical and/or Mechanical	Diesel generator No.2 is available. Manual starting of one SFP pump and one CCW pump are possible. Second SFP pump may be started later, if tie-in breakers can be operated. Pool temperature will not exceed 118°F.	Diesel generator No. 2 is available. One LPSI pump, one CCW pump, and one SFPC pump are available for manual startup. Pool temperature will not exceed 109°F.
Air Operated Valve CV-0944 (On CCW inlet to SFPC heat exchangers)	Air Failure	Single failures in the air system other than rupture of air piping to valve cannot cause valve closure as redundant compressors, cross connects and air storage tanks are available for valve operation.	Both LPSI pumps and heat exchangers are available.
CV-3055 (At inlet to both shutdown heat exchangers on LPSI/shutdown system	Air Failure	No effect on SFPCS.	Single failures in the air system other than rupture of air piping to valve cannot cause valve closure as redundant compressors, cross connects, and air storage tanks, are available for valve operation.

storage locations except for the two 50 element racks located in the main pool cask laydown area.

The heat load for this sequence of events is 16.9 x 10^6 BTU/hr. The pool temperature, assuming both spent fuel pool cooling pumps are operational, is predicted to be no greater than 116°F. Postulating a single active failure in the SFPC system the maximum expected temperature is 118°F. This compares to the FSAR design basis for normal refueling of 20 x 10^6 BTU/hr and a 125°F pool temperature and 128°F for the single failure condition. The difference in heat loads is attributed to the use of more advanced techniques for calculating decay heat generation as compared to the earlier methods used in the design of the Palisades plant.

The analyses treated the main pool and the tilt pool as one pool. Although they are interconnected, the bulk temperature in the pit is expected to be higher due to the fact that the cooling flow into the tilt pit is low and taken directly from the pump discharge prior to the SFPCS heat exchanger. Therefore, the temperature of the cooling water entering the tilt pit is the bulk mixed temperature of the main pool which under single failure conditions is 118°F. Based on 100 gpm cooling flow into the tilt pit, the bulk temperature will be 145°F under single failure normal refueling conditions. The temperature is based on a stored fuel decay time of one year. Cooling water to the tilt pit is supplied through an existing 4" fill line.

The heat load for the core offload case was based on the following projected refueling sequence:

1. 205 fuel assemblies in the pool as of January, 1976.

2. 388 additional fuel assemblies accumulated in successive refuelings through 1983.

3. 204 fuel assemblies (full core) offloaded 7 days after the refueling shutdown the following year (1984).

The above refueling description results in a total of 798 fuel assemblies in the pool. The maximum heat load for this sequence of events is 26.4×10^6 BTU/hr. The maximum pool temperature expected under single failure conditions and utilizing the shutdown cooling system is 103^oF.

6.3 Fuel Element Heat Transfer

The fuel rack base is elevated above the floor to assure adequate flow under the rack to each fuel assembly. Analyses have been performed which shown that sufficient flow is induced by natural convection to preclude local boiling in the hottest storage location.

The analyses were based on the following assumptions:

1. The element inlet temperature is the mixed hot temperature of the pool. This temperature is 118^oF in the main pool and 145^oF in the tilt pit and applies to the thermally limiting single failure condition.

2. A hot assembly peaking factor of 1.94 is applied to a limiting batch average assembly energy release rate. For the main pool the batch average assembly energy release rate is 1.83×10^5 BTU/hr corresponding to 36 hours after shutdown of the 2200 Mwt core. For the tilt pit the assembly average energy release rate is 1.28×10^4 BTU/hr, corresponding to 1 year after shutdown, with the applicable assembly peaking factor of 1.68.

3. The maximum local peaking factor is 3.62 (for a 2200 MWt core) giving a maximum local heat flux of 2629 BTU/hr/ft^2 in the main pool and 184 BTU/hr-ft^2 in the tilt pit.

4. A film coefficient of 36 BTU/hr-ft^2-^{o}F is based on pure conduction through a stagnant boundary layer at the fuel rod surface.

5. A one-dimensional fluid flow analysis is used.

6. In the main pool the downcomer region on the periphery of the pool feeds 8 assemblies in a row, each assumed to be generating the maximum heat rate defined in Assumption 2. Similarly in the tilt pit the downcomer region feeds 3 assemblies in a row (see Fig. 2-1).

7. During full core offload, the shutdown cooling system is interconnected with the spent fuel pool cooling system.

With a single failure in the spent fuel pool cooling system, the bulk pool temperature will not exceed 118^{o}F for a 36 hour normal offload. For this condition the maximum surface temperature of a fuel rod is less than 230^{o}F providing more than 9^{o}F margin to local boiling. The margin to bulk boiling is greater than 85^{o}F. This represents the limiting thermal condition in the pool.

The tilt pit normally will contain only fuel which has decayed at least one year. Under limiting single failure conditions with a pit coolant flow rate of 100 gpm, the maximum bulk water temperature in the tilt pit may reach 145^{o}F. However, because of the low heat generation rate in the tilt pit assemblies, the maximum fuel element surface temperatures will be substantially lower than that of the main pool and thus the margin to local boiling will be greater.

In summary, with a single active failure, the hottest fuel rod surface temperature is below the local saturation temperature and thus precludes local boiling. Both present (2200 Mwt) and stretch power (2650 MWt) cores and their applicable design peaking factors have been considered in establishing the limiting thermal conditions. Thus no local boiling is predicted in the main pool or the tilt pit under single failure as well as normal operation.

6.4 Spent Fuel Pool Chemistry Control

Water chemistry and clarity are maintained by the existing fuel pool cooling and cleanup system. The fuel pool recirculation booster pump takes suction from the skimmer outlets located at the surface of the fuel pool, circulates the water through a filter and demineralizer before discharging back to the pool. In this way, floating debris is removed from the surface and water quality is maintained. The filters are of a replaceable cartridge type with a rating of 25 microns. The demineralizer is of a mixed bed cation-anion resin. Both filter elements and resins are changed on a maintenance schedule or when the differential pressure across either filter or demineralizer becomes excessive.

This equipment has been demonstrated during operating service to be effective in maintaining water purity and clarity within the existing limits during storage periods and during refueling operations. The radioactive contaminant levels in the pool are primarily a function of activated corrosion products, failed fuel fraction and reactor operating level and are highest during and shortly following refuelings. The concentration of impurities is controlled by removal in the demineralizer and filter and by natural radioactive decay. The installed systems and equipment are considered adequate for maintaining water purity and clarity with the expanded pool storage capacity.

7.0 RADIOLOGICAL CONSIDERATIONS

7.1 Fuel Building Dose Rates

7.1.1 Pool Surface Dose

The additional spent fuel assemblies in the pool will result in an increase in dose rates in the spent fuel pool area due to a buildup of radionuclides in the pool water. To determine the amount of increase, a calculational model was devised that considered the presence of activated corrosion products, leakage of the isotopes from the fuel to the pool, the decontamination factor and flow rate of the pool purification system, the isotopic half-lives, and the decay time of the fuel. Using this model, the pool's activity was predicted for the present pool capacity (272 assemblies), and for the proposed capacity (798 assemblies). On the refueling platform, five feet above the center of the pool, the dose rate increased from 2.17 mrem/hr for 272 assemblies to 3.24 mrem/hr for 798 assemblies. At poolside, one foot from the pool wall and five feet above the surface, the dose rate increased from 1.58 mrem/hr to 2.34 mrem/hr. The increase in the pool capacity has a negligible effect on personnel exposure. Assuming an occupancy time of 504 man-hours per year at the refueling platform and 1134 man-year poolside for refueling operations, and an additional 52 manhours per year poolside for routine operations, the total incremental dose due to the expansion of pool capacity from 272 to 798 assemblies is 1.43 man-rem per year.

7.1.2 Airborne Doses

The water evaporation rate, and hence tritium release to the environment around the spent fuel pool, is expected to change as a result of the following factors.

a. Lower calculated water temperatures in the spent fuel pool than those evaluated previously in the FSAR.

b. Higher water temperatures in the north tilt pit area relative to the main pool.

c. Increased water surface area due to utilization of the north tilt pit.

Calculations show that the overall evaporation rate will increase approximately 9% over that stated in the Palisades FSAR. It can be concluded that the anticipated airborne dose rate due to tritium will increase no more than 9% in areas above the pool and tilt pit.

7.1.3 General Area Doses

The adequacy of the spent fuel pool and tilt pit shielding was analyzed with the QAD and ANISN computer codes, to take into account storage of additional spent fuel according to the schedule provided in Section 6.2. The QAD computer code is a point kernel computer program, developed by LASL, designed for calculating the effects of gamma rays that originate in a volume distributed source. The ANISN code is a one dimensional, multi-group transport program developed by ORNL, which solves the Boltzmann transport equation by the method of discrete ordinates.

Preliminary analyses have shown that the existing shielding is generally adequate to reduce effectively neutron and secondary gamma radiation in all expected areas of occupancy surrounding the pool. However, three areas in which fission product gamma dose rates have exceeded the FSAR radiation zoning criteria have been identified. These are (1) outside the north wall of the north tilt pit, (2) outside the north wall of the existing spent fuel pool, and (3) in the space directly below the spent fuel pool cask loading area.

When the north tilt pit is used to store fuel which has decayed for at least one year, it has been calculated that the expected gamma dose rate on the north wall of the tilt pit, which is 2 feet thick, will be approximately 14 R/hr. Initial studies show that approximately 7 inches of lead equivalent will be required in addition to the 2 foot thick concrete wall to achieve dose rates consistent with the FSAR radiation zoning criteria. Assuming that the spent fuel pool will be used to store fuel which has decayed for at least 36 hours, it has been calculated that the expected gamma dose rate on the North wall of the pool will exceed 10 mrem/hr. Assuming the cask loading area will be used to store fuel which has decayed for at least 36 hours, it has been calculated that the gamma dose rate under the pool floor adjacent to the cask loading area will exceed 200 mrem/hr. Additional studies are being performed to develop an adequate shield design to ensure compliance with the radiation zoning criteria as prescribed in the Palisades FSAR.

7.2 Heavy Objects Over Spent Fuel Pool

The pathway to the containment equipment access hatch traverses the spent fuel pool and an evaluation of the potential for and consequences of dropping large equipment onto the spent fuel elements stored in the spent fuel pool has been performed. This evaluation is based on an envelope that an object may impact on the stored fuel rather than an identification of specific pieces of equipment or components that may be handled over the pool. For conservatism and ease of analysis it is assumed that all the fuel impacted by an object fails, i.e. no attempt is made to take credit for the structural strength of the storage racks or the fuel assembly itself to resist deformation. The analysis addressed both the offsite radiological doses as well as nuclear criticality.

The offsite radiological dose was assessed based on the criteria and assumptions set forth in Safety Guide 1.25 "Assumptions Used for Evaluating the Potential Radiological Consequences of a Fuel

Handling Accident in the Fuel Handling and Storage Facility for Boiling and Pressurized Water Reactors," dated March 23, 1972, and Standard Review Plan 15.7.5, "Spent Fuel Cask Drop Accidents," dated November 24, 1975. Analyses show that the number of fuel assemblies decayed for 30 days after shutdown which may be damaged prior to exceeding 10CFR100 limitations for accidents is 361. It should be noted that a full core consists of 204 assemblies. This analysis assumes that the release occurs 30 days after the shutdown. During this accident a negative pressure is maintained in the fuel building. This analysis considers iodine removal by the charcoal filters. The χ/Q value utilized is 2.56×10^{-4} sec/m^3 at a distance of 667 meters, as discussed in FSAR Section 14.22 (page 14.22-27, revised May 26, 1971).

A worst case physically attainable configuration in the pool is: (1) 204 assemblies representing a full core offload, (2) 68 assemblies (1/3 core) removed at the previous refueling, and (3) 416 assemblies stored in the pool with more than one year decay time. This fills to capacity the 688 storage locations available in the main pool. Conservatively assuming that the full offloaded core has reached full design burnup and that all assemblies have decayed 30 days, the offsite dose due to failure of 272 assemblies is approximately 75% of the 10CFR100 thyroid criteria dose which is limiting. The dose due to the 416 fuel assemblies with a minimum of one year in-pool decay time is due only to Kr-85 and is less than 0.1 Rem. Therefore, even if the dropping of an object of sufficient cross sectional area and weight to fail all the fuel contained in the main pool--an incredible event--is postulated, the 10CFR100 criteria are not exceeded.

The above accident, but assuming fresh fuel, was also evaluated to determine the effects on nuclear reactivity. In this analysis credit is taken for the dissolved boron in the spent fuel pool water. The reactivity analysis was performed with NUMICE/PDQ (see Section 4.0). The following worst situations were postulated as a result of this possible deformation.

Case A

1. The fuel rods buckle to the most reactive pitch.

2. The poison cans buckle so that they touch each other (i.e., all the water between the cans is lost).

3. The poison can inside dimension does not change.

Case B

1. The fuel rods buckle to the most reactive pitch.

2. The poison cans buckle so that they touch each other (i.e., all the water between the cans is lost).

3. The poison cans collapse inward so that they touch the peripheral fuel rods of the assembly at the most reactive pitch (i.e., there is water only between the fuel rods at the most reactive pitch).

The worst fuel rod pitch at 1720 PPM boron in the water was determined to be 0.490 inch. For comparison, the nominal pitch is 0.550 inch. Preliminary results indicate that the calculated infinite multiplication factor for either accident will be less than the nominal value of 0.872.

The situation of complete fuel rod collapse to form a homogeneous mixture of UO_2, water and Zircaloy is not worse than cases A or B above, because the k_∞ of this homogeneous "slurry" is lower than the heterogeneous fuel lattice k_∞ as a result of the increased U-238 resonance absorption.

In conclusion, an accident involving the failure of a major portion of the fuel in the pool is not considered credible because (1) all large objects are handled by a highly reliable crane as discussed in Appendix J of the Palisades FSAR, (2) there is no object handled over the pool of the size required to impact 704 assemblies. The analysis was done to envelop the possible effects of this occurrence and on a conservative basis shows no adverse consequences.

8.0 CONCLUSION

Based on the above analyses and description, CPCo concludes that the described modification can be accomplished without undue hazard to the health and safety of the public and conforms to applicable regulations.

Docket No. 50-255

CONSUMERS POWER COMPANY
PALISADES NUCLEAR GENERATING STATION
SPENT FUEL POOL MODIFICATIONS
ENVIRONMENTAL IMPACT EVALUATION

November 1976

TABLE OF CONTENTS

1.0 INTRODUCTION

1.1 Purpose of Environmental Impact Evaluation

Subject to the approval of the Nuclear Regulatory Commission, Consumers Power Company (CPCo) intends to increase the capacity of the spent fuel storage pool at the Palisades Nuclear Power Generating Station. CPCo is taking this action in order to assure the continued availability of electrical power to its service area. In view of the present uncertain future of the fuel reprocessing industry, which is extensively documented in both technical and popular literature, CPCo considers the only prudent course of action to be to increase its own capacity to retain spent fuel. This Environmental Impact Evaluation was prepared, therefore, to evaluate the impact of the modification of the spent fuel racks to allow such an increase.

This Environmental Impact Evaluation describes the history and need for proposed modifications. The proposed modification is described in Section 2.0. Section 3.0 evaluates the environmental effects of the normal operation of the modified facility while Section 4.0 addresses the environmental effects of accidents. Section 5.0 describes and evaluates the several alternative actions considered to alleviate the anticipated shortage of spent fuel storage capacity. The summary of the several alternative actions and cost-benefit analyses is presented in Section 5.0.

1.2 History and Need for the Proposed Modification

Palisades Plant received its provisional Operating License, DPR20, in May 1972. At present there are 205 spent fuel assemblies stored in the spent fuel storage pool. These assemblies were removed at less than full burn-up in order to replace them with prepressurized fuel design. The present total storage capacity the Palisades spent fuel racks is 276 assemblies or approximately 1 1/3 full cores.

It is prudent engineering practice and the policy of CPCo to reserve storage space in the spent fuel pool to receive an entire reactor core ("core offload") should unloading of the core be necessary or desirable because of operational considerations. This, together with the fact that spent fuel reprocessing facilities cannot assuredly be available to CPCo prior to the mid-1980's at the earliest (and, therefore, no additional spent fuel can be shipped for reprocessing), leads to the conclusion that an increase in the spent fuel storage capability is necessary. A modification is planned to increase the spent fuel storage capacity by installing new spent fuel racks utilizing a neutron absorbing material ("poison"). These new racks will maximize the storage capability by decreasing the center-to-center spacing of the fuel assemblies while maintaining subcriticality under all conditions. The planned modification will result in a maximum storage capacity of 798 assemblies.

2.0 GENERAL DESCRIPTION

2.1 Present Design

As described in the FSAR, the spent fuel storage pool is located in the auxiliary building adjacent to the containment. The pool contains four spent fuel storage racks with a capacity of 276 spent fuel elements and one rack for storage of control rods. It is lined with stainless steel and has reinforced concrete walls and floor 4 1/2 to 6 feet thick.

The present fuel racks are stainless steel with a center-to-center spacing of 11 1/4 inches. There are two 1/4 inch stainless steel plates between each pair of fuel assemblies. At design temperature with no credit taken for soluble boron in the pool water, the maximum K_{eff} is less than 0.95. A recessed area is provided in the pool for a spent fuel shipping cask.

The fuel pool cooling system is a closed loop system consisting of two half-capacity pumps, a full-capacity heat exchange unit consisting of two heat exchangers in series, a bypass filter, a bypass demineralizer, a booster pump, piping, valves, and instrumentation.

The spent fuel pool cooling system has a heat removal capability of 23 x 10^6 BTU/hr. The spent fuel cooling system is conservatively designed to maintain pool average temperature at less than 125°F with 1/3 core of fully burned up fuel in the pool, 36 hours after reactor shutdown. A single failure of the cooling system would increase pool temperature by only 3°F. The water in the spent fuel pool is normally borated to 2000 ppm. The entire fuel pool cooling system is tornado-protected and is located in a Seismic Class 1 structure.

Fuel pool makeup water is supplied from the Safety Injection and Refueling Water (SIRW) tank. A secondary backup supply of water is available from the fire system. This would be utilized to replenish the fuel pool water content in the event of considerable loss of pool water.

The clarity and purity of the water in the spent fuel pool are maintained by passing a portion of the flow through the bypass filter and/or demineralizer. Skimmers are provided in the spent fuel pool to remove accumulated dust from the pool.

Connections are provided for a temporary tie-in to the shutdown cooling system to provide for additional heat removal in the event that a full core has to be unloaded into the pool. These connections also provide a backup capability for the fuel pool heat exchangers.

The fuel pool cooling system is connected by valved piping to the reactor refueling cavity for additional cooling of the reactor cavity water during spent fuel transfer.

Two fuel tilt pits are located in the fuel building adjacent to the spent fuel pool and connected to it by canals which are closed off by dam blocks. One tilt pit is used for normal fuel transfer activities. The second tilt pit was provided to accommodate an additional unit then being considered. This second pit was not lined at the time of construction. Presently, it is being lined with stainless steel to permit its use for additional spent fuel storage.

2.2 Proposed Modification

The proposed fuel rack modification will involve removing the existing fuel and control rod racks and replacing them with new racks with smaller center-to-center spacing. Each individual storage location consists of two concentric 1/8" austenitic type 304 stainless steel square cans with the annular space occupied by B_4C neutron absorber plates to ensure subcriticality.

A rack assembly consists of a rectangular array of storage cans with a minimum 10.25 inches center-to-center spacing of the fuel assemblies. The array size of each rack was chosen to optimize use of pool space. The expanded spent fuel storage capacity is 798 assemblies.

The new racks are seismic Category I and are restrained to the pool wall at the top and bottom of each rack to prevent excessive movement of the racks under postulated seismic accelerations. Provisions are made in the design to accommodate thermal expansion.

The present cask laydown area will contain two 50-element racks which will be used to store fuel during full core offloads only. These two racks may be removed to allow placement of the spent fuel shipping cask or to allow the use of fuel inspection and repair equipment, etc. At the time that fuel shipment resumes, the two racks in the cask area may be removed and the caks anti-tipping device (described in Appendix J to the FSAR) will be installed in the pool. This device will provide anti-tipping protection and will act as a seismic restraint for the remaining racks.

The presently unused tilt pit will be used for spent fuel, and control rod storage and as an alternate cask laydown area. Control rods and fuel stored in the rack in this tilt pit use a rack design with slightly larger cans than those used in the other racks. A jib crane will be added to facilitate fuel handling in the tilt pit area. To minimize heat generation in the tilt pit, normally only fuel decayed for at least one year will be stored there.

The additional heat load from the increased number of spent fuel assemblies can be accommodated by the spent fuel pool cooling system without modification except for addition of a direct cooling water supply to the tilt pit.

2.3 Schedule for Proposed Modification

The schedule for the proposed installation of the spent fuel racks is presented in Table 2.1. The installation is scheduled to start in July 1977. Since about five months are required for fabrication, the spent fuel pool modification is required to begin in early February 1977. In order that procurement may begin in a timely manner, it will be necessary to obtain NRC review and approval by January 31, 1977.

Table 2-1

SCHEDULE FOR PROPOSED MODIFICATION

Conceptual Design Submitted to NRC	September 8, 1976
Discussions with NRC and Preliminary Concept Approval Received	September 16, 1976
Formal Application to NRC for Spent Fuel Storage Modification Approval	November 1, 1976
NRC Approval Received	January 31, 1977
Fabrication and Delivery of the New Racks	February 1, 1977 - December 1, 1977
Rack Installation:	
Phase 1 - Cask Area and Spare Tilt Mechanism Pool	July 1, 1977 - August 1, 1977
Phase 2 - Remainder of Spent Fuel Pool	Commencing December 1, 1977

3.0 ENVIRONMENTAL EFFECTS OF INCREASED STORAGE

This section discusses the environmental effects (heat, radiological, chemical) resulting from the design change, above those previously reported.

3.1 Heat Dissipation Effects

As spent fuel assemblies will be accumulating from each refueling, the maximum heat load on the spent fuel pool cooling system, and, in turn into the ultimate heat sink will be slightly larger than previously evaluated.

The heat load was calculated using the computer code ORIGEN developed at Oak Ridge National Laboratory. ORIGEN is a point depletion code which solves the equations of radioactive buildup and decay for large numbers of isotopes with arbitrary coupling. The heat load was determined for the first case based on the accumulation of assemblies through the normal refueling cycle through 1985.

The maximum heat load resulting from a normal refuleing of 69 assemblies is 14.8×10^6 BTU/hr. The incremental increase in heat load due to older fuel as the pool is filled to its enlarged storage capacity is 0.19×10^6 BTU/hr. This additional heat load of 0.19×10^6 BTU/hr to be discharged to the environment is negligible when compared to the present station heat rejection rate 4.78×10^9 BTU/hr based on operation at the present power level of 2200 Mwt.

3.2 Radiological Effect

This section describes the radiological effects from the proposed modification of the spent fuel pool, including the effects of increased tritium release as a result of increasing pool storage capacity.

3.2.1 Airborne Activity

The water evaporation rate, and hence tritium release to the environment around the spent fuel pool, is expected to change as result of the following:

a. Lower calculated water temperatures in the spent fuel pool than those evaluated previously in the FSAR

b. Higher water temperatures in the tilt pit area relative to the main pool

c. Increased water surface area due to utilization of the north tilt pit.

The concentration of tritium in the fuel building is calculated to increase from a maximum value of 0.86×10^{-6} μ Ci/cc (based on FSAR information) to 0.94×10^{-6} μ Ci/cc, or about 19% of the occupational 10 CFR 20 limit. Even with the higher calculated potential evaporation rate from the spent fuel pool, the incremental offsite dose expected from the modification is a small fraction of the total plant tritium release (approximately 2%).

3.2.2 Direct Local Dose

The direct dose rate from the fuel measured above the pool surface decreases when the number of fuel assemblies in the pool increases from 272 to 798, due to the shielding effect of the spent uranium in the additional assemblies. However, there is an increase in the dose rates above the pool and in the personnel exposure from the increased concentration of radionuclides in the pool water. A calculational model was developed to estimate the pool's activity for the pool filled to its present capacity of 272 assemblies and for the pool filled to the proposed capacity of 798 assemblies. The model considered the leakage of the isotopes from the fuel to the pool, the decontamination factor and flow rate of the pool purification system, the isotopic half-lives, and the decay time of the fuel. The activities thus calculated were then used in the QAD computer program to obtain the dose rates. On the refueling platform, 5 feet above the center of the pool, the dose rate increased from 2.17 to 3.24 millirem per hour when the additional assemblies were added. At poolside, 1 foot from the pool wall and 5 feet above the water surface, the dose rate increased from 1.58 to 2.34 millirem per hour. Assuming that one man occupies the refueling platform and three men occupy the poolside area during refueling, the incremental dose due to the increased capacity of the pool is 1.39 man-rem per year. The refueling exposure is based on man-ours per year. Routine exposure in the fuel building is assumed to be 52 man-hours. Routine exposure results in another 0.04 man-rem/year, a total 1.43 man-rem per year as the total incremental dose due to the pool enlargement.

3.3 Chemical Discharges

No chemical discharge is anticipated as a result of this modification. The only consumable or dischargeable materials associated with the spent fuel storage system are the resin and filter elements. These are packaged as radioactive waste and shipped to an approved burial site.

3.4 Resources Committed

Construction of the high density spent fuel storage racks for the storage of 798 fuel assemblies will involve the commitment of stainless steel and boron carbide in the amounts shown below. The annual U.S. consumption is included for comparison.

Item	Amount Used	Annual US Consumption
Stainless Steel	350,000 lb.	2.82×10^{11} lbs.
Boron Carbide	30,700 lbs.	900,000 lbs.

As seen, only a small fraction of these resources will be used.

3.5 Summary of Environmental Effects

The concludion to be drawn from the above analyses is that increasing the spent fuel pool storage capacity as proposed will have a negligibly small increased effect on the environment over that previously evaluated.

4.0 ENVIRONMENTAL EFFECTS OF ACCIDENTS

This section discusses the changes in environmental effects of postulated accidents which involve the handling of spent fuel in the spent fuel pool. These accidents were previously analyzed in the Environmental Report (ER). In addition, the Spent Fuel Cask Drop was discussed in a series of letters, the most recent to Mr. Robert A. Purple from Mr. D. A. Bixel dated January 9, 1976 and in Appendix J (Amend. 29) to the Palisades FSAR transmitted August 9, 1974.

4.1 Fuel Assembly Drop in Fuel Storage Pool

This accident was analyzed by the AEC (NRC) in Final Environmental Evaluation Statement, Palisades Nuclear Generating Plant, Docket 50-255, June 1972, Table VI-2, Section 7.1. The FES stated the would result in a site boundary dose of 0.7% of 10 CFR 20 limits and a 50 mile radius population dose of 0.30 man-rem. Examining the assumptions used for calculating this accident as given in Appendix 5, USNRC Regulatory Guide 4.2 (Rev 1, Proposed Annex to Appendix D 10 CFR 50) it is concluded that nothing in the proposed modification would cause a change in the stated results.

4.2 Heavy Object Drop Onto Fuel Rack

Similarily, the FES states the dose resulting from this accident as 3% of 10 CFR 20 at the site boundary and as 1.2 man-rem for the 50 mile radius population dose. The proposed modification would not change the Appendix 5 assumptions and values used in calculating the resulting dose and therefore the stated results are still valid.

4.3 Fuel Cask Drop

The analysis of the spent fuel cask drop accident was presented in the August 9, 1974 letter previously discussed. This evaluation is currently under NRC Staff review. If Annex 5 assumptions are used, there is no change in dose.

4.4 Summary of Environmental Effects of Accidents

The environmental effects of accidents, as a result of the proposed modification of the spent fuel racks, were described in the preceding three sections. The effects of any of the accidents result in no increase in the environmental impact previously evaluated.

5.0 ALTERNATIVE ACTIONS

In reaching the conclusion to increase the spent fuel storage capacity of Palisades, CPCo considered several alternatives to the action. These are storage at independent commercial facility, storage at an independent CPCo facility, storage at a reprocessing facility and storage at other nuclear plant sites. Each alternative was evaluated on a cost-benefit bases and compared with the proposed storage increase and the consequences of reactor shutdown.

The total cost of the revised spent fuel storage rack design including restraints and pool modifications is approximately $3,800 per storage location in today's dollars. This also includes engineering, contingencies, financing, and other peripheral costs. The benefit is the capability to operate through the 1980's with no additional transportation charges.

5.1 Storage at an Independent Commercial Facility

The cost of storage in a commercial storage facility has been investigated with the conclusion that the cost would be in the range of $2,000-$3,000 per storage location per year, with increases to account for escalation allowed. The present worth cost of commercial storage arrangements reviewed is in the range of $15,000 to $19,000 per storage location, which does not include the cost of shipping the fuel to the storage facility.

5.2 Storage at an Independent CPC Facility

The economic and technial feasibility of a reprocessing plant, in which utilities were a participant, has been considered by other utilities. A spent fuel receiving pool of 500-1500 MTU capacity would cost $20-50 million with increments of 1000 MTU adding about $30-40 million to the cost.

5.3 Storage at a Reprocessing Facility

The spent fuel storage pools at all of the reprocessing facilities currently in existence are filled. If these pools are modified to increase capacity then the cost of storage would be in the range of $2,000-3,000 per storage location per year.

This estimate is based on the assumption that reprocessing and storage will be accomplished at the same facility. If the fuel has to be reprocessed at another reprocessing facility, then the cost impact for additional transportation is noted in 5.4 below.

5.4 Storage at Other Nuclear Plant Sites

With the universality of the spent fuel storage problem, the short term nature of the solution,and the possible multiple transportation costs involved, this is clearly more expensive than high density storage at Palisades. Credible cost estimates are not available, but the additional transportation costs alone could be $1,000-3,000 per fuel assembly. The costs are dependent on the location of the other nuclear plant site, on the location of the eventual reprocessor and the schedule, and on transportation.

5.5 Reactor Shutdown

Shutdown would result in levelized annual expense of roughly $176.6 million with no revenue. Considering a fuel discharge of roughly 68 assemblies per year, the cost per fuel assembly to be stored is roughly $2.6 million.

5.6 Summary of Cost-Benefit Analyses

Table 5-1 summarizes the costs and benefits of each alternative. The benefit to be derived from four of the alternaties is continued generation of electrical energy. Reactor shutdown has no benefit associated with it and storage at other nuclear plants is not

TABLE 5-1

SUMMARY OF COST-BENEFIT

Alternative	Cost	Benefit
Pool Expansion	$3,800 per assembly	Continued Operation and Energy Generation
Storage at Independent Commercial Facility	$15,000-$19,000 per assembly	Continued Operation and Energy Generation
Storage at Independent CPCo Facility	$14,000 per assembly	Continued Operation and Energy Generation
Storage at Reprocessor's Facility	$15,000-$19,000 per assembly	Continued Operation and Energy Generation
Storage at Other Nuclear Plants	----	None - Not Feasible
Reactor Shutdown	$2.6 million per assembly	None

possible at this time or in the foreseeable future. Therefore, there is no associated cost or benefit. As seen from the table, the pool storage capacity increase is clearly the preferred choice.