Application for Amendment
to
License No. NPF-3
for
Davis-Besse Nuclear Power Station
Unit No. 1

Enclosed are forty-three (43) copies of the requested changes to the Davis-Besse Nuclear Power Station Unit 1 Technical Specifications, Appendix A to License No. NPF-3 together with the report "Safety Evaluation of the Spent Fuel Storage Capacity Modification for Davis-Besse Nuclear Power Station Unit 1" which states the reasons for the requested changes and contains details of the design, design analysis, and safety evaluation of the proposed modification to increase spent fuel storage capacity.

By Lowell E. Roe
Vice President, Facilities Development

Sworn to and subscribed before me this 19th day of December, 1977.

Fred W. Germain
Notary Public

FRED W. GERMAIN
Notary Public — State of Ohio
My Commission Expires Oct. 30, 1982

8001310543

DESIGN PRESSURE AND TEMPERATURE

5.2.2 The reactor containment building is designed and shall be maintained for a maximum internal pressure of 40 psig and a temperature of 264°F.

5.3 REACTOR CORE

FUEL ASSEMBLIES

5.3.1 The reactor core shall contain 177 fuel assemblies with each fuel assembly containing 208 fuel rods clad with Zircaloy -4. Each fuel rod shall have a nominal active fuel length of 144 inches and contain a maximum total weight of 2500 grams uranium. The initial core loading shall have a maximum enrichment of 3.0 weight percent U-235. Reload fuel shall be similar in physical design to the initial core loading and shall have a maximum enrichment of 3.3 weight percent U-235.

The first cycle fuel loading shall contain 68 burnable poison rod assemblies with each assembly containing up to 16 burnable poison rods of sintered AL_2O_3-B_4C clad with Zircaloy-4.

CONTROL RODS

5.3.2 The reactor core shall contain 53 safety and regulating and 8 axial power shaping (APSR) control rods. The safety and regulating control rods shall contain a nominal 134 inches of absorber manterial. The APSR's shall contain a nominal 36 inches of absorber material at their lower ends. The nominal values of absorber material shall be 80 percent silver, 15 percent indium and 5 percent cadmiun. All control rods shall be clad with stainless steel tubing.

5.4 REACTOR COOLANT SYSTEM

DESIGN PRESSURE AND TEMPERATURE

5.4.1 The reactor coolant system is designed and shall be maintained:

a. In accordance with the code requirements specified in Section 5.2 of the FSAR, with allowance for normal degradation pursuant to applicable Surveillance Requirements.

b. For a pressure of 2500 psig, and

c. For a temperature of 650°F, except for the pressurizer and pressurizer surge line which is 670°F.

VOLUME

5.4.2 The total water and steam volume of the reactor coolant system is 12,110 ± 200 cubic feet at a nominal T_{avg} of 525°F.

5.5 METEOROLOGICAL TOWER LOCATION

5.5.1 The meteorological tower shall be located as shown on Figure 5.1-1.

5.6 FUEL STORAGE

CRITICALITY

5.6.1 The new fuel storage racks are designed and shall be maintained with a nominal 21 inch center-to-center distance between fuel assemblies placed in the storage racks to ensure a k_{eff} equivalent to ≤ 0.95 with the storage pool filled with unborated water. The K_{eff} of ≤ 0.95 includes a conservative allowance of 1% Δk/k for uncertainties as described in Section 9.1 of the FSAR.

The spent fuel storage racks are designed and shall be maintained with a rectangular array of stainless steel cells spaced 12 31/32 inches on centers in one direction and 13 3/16 inches on centers in the other direction. Fuel assemblies stored in the spent fuel pool shall be placed in a stainless steel cell of 0.125 inch nominal thickness or in a failed fuel container such that a K_{eff} equivalent to ≤ 0.95 is maintained with the storage pool filled with unborated water. The K_{eff} of ≤ 0.95 includes a conservative allowance of 1% Δk/k for calculational uncertainty.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below 9 feet above the top of the fuel storage racks.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 735 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

5.7.1 The components identified in Table 5.7-1 are designed and shall be maintained within the cyclic or transient limit of Table 5.7-1.

The Toledo Edison Company
and

The Cleveland Electric Illuminating Company

Safety Evaluation of the Spent Fuel Storage Capacity Modification

for

Davis-Besse Nuclear Power Station Unit 1

Docket No. 50-346
License No. NPF-3

December 5 , 1977

Table of Contents

1.0 Introduction

This report is prepared and submitted in support of Toledo Edison's request, on behalf of itself and Cleveland Electric Illuminating, to amend the Davis-Besse Nuclear Power Station Unit 1 Facility Operating License No. NPF-3, to reflect an increase in spent fuel storage capacity.

Due to the present and expected near-term shortage of reprocessing facilities and other uncertainties associated with the tail end of the LWR fuel cycle, it is necessary to replace the existing Davis-Besse Unit 1 spent fuel storage racks (260 assembly capacity) with high capacity spent fuel storage racks (735 assembly capacity). This modification will be necessary before the first refueling, in order to replace the racks with no spent fuel in the pool and to maintain the ability to discharge spent fuel while maintaining full core (177 fuel assemblies) off-load capability.

The Toledo Edison Company is responsible for the overall design, construction, and operation of Davis-Besse Unit 1. Toledo Edison has contracted with Nuclear Energy Services, Incorporated, to design, manufacture, deliver and install the proposed high capacity spent fuel storage racks.

2.0 Design of the Spe Fuel Storage Racks

2.1 Design Bases

The high capacity spent fuel storage racks are designed to provide storage locations for up to 735 fuel assemblies and are designed to maintain the stored fuel, having an equivalent uranium enrichment of up to 3.3 weight percent U-235 in UO_2 in a safe, coolable, and subcritical configuration during normal and abnormal conditions.

2.2 Storage Rack Description

The spent fuel storage rack consists of a 16 x 45 array of 720 square stainless steel cells spaced on a pitch of 13-3/16 inches in the 16 cell direction and 12-31/32 inches in the 45 cell direction, plus 15 locations for failed fuel containers located adjacent to one side of the 16 x 45 cell arrangement. To achieve the 720 cell configuration, six 7 x 8 array and six 8 x 8 array modules are employed. The storage rack is shown in the general arrangement drawing, Figure 1.

Each storage rack module has two basic components: the support structure and the fuel storage cell as shown in Figure 2. The support structure consists primarily of the four corner storage cells which are secured to two levels of grid members which maintain the horizontal position and vertical alignment of the remaining inner storage cells. The support structures are supported from the spent fuel pool floor pads through the four corner cells. Diagonal bracing is provided on the structure principally to accommodate the loads imposed by rack installation, by fuel handling and by seismic events.

Each storage cell is basically a stainless steel box 9.25 in. square (OD) by 166 in. long with 0.125 in. walls. The cells are flared at the top to simplify insertion of the fuel assembly into the cell. Attached to the bottom of each cell are four stainless steel posts which support the weight of the cell and its contents. The posts attached to the inner cells rest directly on the spent fuel pool floor and space the cells off the pool floor a sufficient distance to assure adequate area for cooling flow. To accommodate any pool floor liner irregularities, the rack is designed to permit the inner cells to move vertically within the rack structure (± ½ in. motion is provided). The cells, however, are positively locked into the support structure so that they cannot be inadvertently lifted out of the rack.

Horizontal seismic loads are transmitted from the rack structure to the walls surrounding the spent fuel pool through seismic bracing extending from the perimeter of the 16 x 45 array to the pool wall. Bracing is provided at each of the two grid levels (approximately 25 in. and 145 in. above the spent fuel pool floor). Vertical dead weight and seismic loads are essentially transmitted directly to the pool floor by each storage cell.

2.3 Storage Rack Safety Evaluation

2.3.1 Nuclear Criticality Analysis

A detailed nuclear analysis has been performed to demonstrate that for all anticipated normal and abnormal configurations of fuel assemblies within the fuel storage racks, the k_{eff} of the system is less than 0.95 as confirmed by transport theory calculations.

The following conservative assumptions have been used in the criticality calculations:

1. The pool water has no soluble poison.

2. The fuel assemblies have no burnable poison.

3. The fuel is fresh and of a specified enrichment (3.3 w/o) which is higher than that of any fuel currently scheduled for use in Davis-Besse Unit 1.

4. The rack configuration is infinite in all three dimensions.

5. No credit is taken for structural material poison other than the stainless steel fuel rack cell.

6. All fuel cells are assumed to be 0.125 in. thick, the minimum allowable thickness.

The normal configurations considered in the nuclear analysis included the reference configuration (fuel assembly centrally positioned within cell having nominal rack dimensions), the eccentric positioning of fuel within the fuel storage cells, and the variations permitted in fabrication of the principal storage rack dimensions (fuel storage cell pitch and cell wall thickness). A transport theory analysis of the reference configuration was done to establish a diffusion to transport theory bias.

The abnormal configurations analyzed were: a variation from the maximum water density at near 40° F to 260° F including a partly voided situation at 212° F, and a configuration where a complete cell and fuel assembly were displaced due to failure of the retaining clips. Inadvertant placement of a fuel assembly adjacent to a fuel rack was not analyzed since structure is provided on the peripheral racks where required to maintain a center-to-center spacing in excess of 17 inches, a clearly acceptable value (k_{eff} less than 0.90).

The principal calculational method used for the criticality analysis was diffusion theory using HAMMER (a multigroup integral transport theory code) and EXTERMINATOR (a 2-D multigroup diffusion theory code). Verification calculations were dor by transport theory using GGC-3 (a consistent Bn or P1 coda for the calculation of fast neutron spectra and associated multigroup constants) and DOT (a 2-D multigroup discrete ordinates transport theory code).

The following diffusion theory results were obtained for the normal configurations:

Description	k_{eff}
13.078 in. average pitch, 68° F	0.8874
13.016 in. average pitch, 68° F	0.8929
13.078 in. average pitch, 68° F, fuel displacement	0.8944
Δk due to reduced pitch	+0.0055
Δk due to eccentric fuel placement	+0.0070

The worst case normal configuration k_{eff} is obtained by statictically combining (square root of sum of the squares) the effects of the normal variations. The result is k_{eff} = 0.8874 ± 0.0089.

The abnormal configurations analyzed gave the following results:

Δk due to increased pool temperature	+0.0104
Δk due to can displacement	+0.0009

The worst case abnormal configuration combines the worst case normal configuration with the most adverse abnormal condition (pool temperature rise).

Worst case abnormal configuration k_{eff} = 0.8874 $^{+.0193}_{-.0089}$

The reference case was also calculated using transport theory to establish a diffusion to transport bias of 0.0342. This results in:

Worst case normal configuration k_{eff} = 0.9216 ± .0089 (with bias)
Worst case abnormal configuration k_{eff} = 0.9216 $^{+ .0193}_{- .0089}$ (with bias)

Using a criticality calculational uncertainty factor of .01 combined statistically with the uncertainties due to normal variations produces a worst case k_{eff} = 0.9454. This value meets the criticality criterion of a k_{eff} of less than about 0.95 (reference USNRC Standard Review Plan 9.1.2).

2.3.2 Structural and Seismic Analysis

The Davis-Besse Nuclear Power Station Unit 1, high density spent fuel storage racks have been designed to meet the FSAR requirements for Seismic Category I structures. Detailed structural and seismic analyses of the high density storage racks have been performed to verify the adequacy of the design to withstand the loadings encountered during installation, normal operation, the severe and extreme conditions of the operating basis and safe shutdown earthquakes and the abnormal loading condition of an accidental fuel assembly drop event.

2.3.2.1 Applicable Codes, Standards and Specifications

The following design codes and regulatory guides have been used in the design/analysis of spent fuel storage racks.

1. A.I.S.C. Manual of Steel Construction, Seventh Edition, 1970.

2. USNRC Regulatory Guide 1.61, "Damping Values for Seismic Design of Nuclear Power Plants", October, 1973.

3. USNRC Regulatory Guide 1.92, "Combination of Modes and Spatial Components in Seismic Response Analysis, Rev. 1, February, 1976.

4. USNRC Standard Review Plan, Section 3.8.4.

2.3.2.2 Loads and Load Combinations

The following load cases and load combinations have been considered in the analysis in accordance with the requirements of USNRC Standard Review Plan, Section 3.8.4.

Load Cases

Load Case 1 - Dead Weight of Rack Plus Corner Fuel Assemblies, D + L (Normal Load)

Under normal operating conditions the rack is subjected to the dead weight loading of the rack structure itself plus the loads resulting from four fuel assemblies stored in the four structural corner cells. The loads resulting from the individual storage cells and contained fuel assemblies transmit their load directly to the pool floor and not through the structure.

Load Case 2 - Dead Weight of Rack Plus 1 g. Vertical Installation Load, D + I.L. (Normal Load)

During installation the rack is subjected to the loading resulting from its own structural weight plus a 1 g. vertical load resulting from a suddenly applied crane load.

Load Case 3 - Dead Weight of Rack Plus Uplifting Load, (D + U.L.) (Abnormal Load)

The possibility of the fuel handling bridge fuel hoist grapple getting hooked on a fuel storage cell was considered. The uplift force considered for this load case was 500 pounds which conservatively exceeds the maximum load of 2750 lbs. allowed by the hoist load limit cell and the fuel assembly, storage cell and hoist grapple combined weight of ~2500 lbs.

Load Case 4 - Operating Basis Earthquake, E (Severe Load)

The rack, fuel assemblies, and virtual water mass react to the simultaneous loading of the horizontal and vertical components of the seismic response acceleration spectra specified for the Operating Basis Earthquake. The seismic loading is applied to the fully loaded rack.

Load Case 5 - Safe Shutdown Earthquake, E' (Extreme Load)

Same as Load Case 4 except that the seismic response acceleration spectra corresponding to the Safe Shutdown Earthquake was used in the analysis.

Load Case 6 - Assembly Drop Impact Load, (Abnormal Load)

The possibility of dropping a fuel assembly on the rack from the highest possible elevation during spent fuel handling was considered. A 1685 pound weight was postulated to drop on the rack from a height of 24 inches. This height was determined based on an assumed minimum water cover of 8 feet to be maintained during fuel assembly handling. Two cases were considered: 1) a direct drop on top of a single storage cell and 2) a subsequent tipping of the assembly onto surrounding storage cells.

Thermal Loading, T (Normal Load)

The stresses and reaction loads due to thermal loadings are insignificant since clearances are provided between racks to allow unrestrained growth of the racks for the maximum expected temperature differential based on a maximum pool temperature of 185° F.

Load Combinations

(a) For service load conditions, the following load combinations are considered using elastic working stress design methods of AISC:

(1) D + L (1a) D + L + T

(2) D + I.L.

(3) D + L + E (3a) D + L + T + E

(b) For factored load conditions, the following load combinations are considered using elastic working stress design methods of AISC:

(4) D + L + T + E'

(5) D + T + U.L.

2.3.2.3 Design and Analysis Methods

Static Analysis

The response of the rack structure to specified static loading conditions has been evaluated by means of linear-elastic analysis using the finite element method. The rack was mathematically modeled as a three-dimensional finite-element structure consisting of discrete three-dimensional elastic beams. Six degrees of freedom (three translations and three rotations) were permitted at each nodal point. Appropriate boundary conditions were assumed for each load case.

Dynamic Analyses

The response of the rack structure to specified seismic loading conditions has been evaluated by mathematically modeling the storage rack as a lumped mass, multi-degree-of-freedom system. Masses are lumped so as to represent the dynamic characteristics of the storage racks. The eigenvalues and eigenvectors (frequency and mode shapes of vibration) of the lumped mass model have been calculated using the Householder-QR technique.

The Seismic Response Analyses are then performed using response spectrum modal superposition methods of dynamic analysis, using the Davis-Besse Unit 1 Amplified Response Spectra and appropriate damping for welded steel structures in accordance with Regulatory Guide 1.61. Individual modal responses of the system are combined in accordance with Section 1.2.1 of Regulatory Guide 1.92. The maximum response of the system for each of the three orthogonal spatial components (two horizontal and one vertical) of an earthquake has been combined on a square root of the sums of square (SRSS) basis (Regulatory Guide 1.92).

The sloshing effects of water on the fuel racks have been evaluated using the analytical methods given in ASCE's "Structural Analysis and Design of Nuclear Plant Facilities". The "rattling" effects of the fuel inside the cell have been accounted for by using suitable impact factors.

The static, seismic and stress analyses for the fuel storage racks were performed utilizing the STARDYNE computer code.

The assembly drop load case (Load Case 6) was performed with linear and non-linear analysis techniques using energy-balance methods.

2.3.2.4 Structural Acceptance Criteria

The following allowable limits constitute the structural acceptance criteria used for each of the loading combinations presented in Section 2.3.2.2.

Load Combinations	Limit
1, 2, 3	S
1a, 3a	1.5S
4, 5	1.6S

S is 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. The yield stress value for stainless steel is taken as 30.0 ksi.

The acceptance criteria for Load Case 6, the accidental fuel assembly drop onto the rack, is that the resulting impact will not adversely affect the leak tightness integrity of the fuel pool floor and liner plate and that the deformation of the impacted storage cells will not adversely affect the value of k_{eff}.

2.3.2.5 Results of the Analysis

The results of the static, seismic and stress analysis of the rack structure show that the stresses and deflections for each load combination are nominal and within the applicable acceptance criteria.

The evaluation of the water sloshing effects in the pool resulting from a seismic event indicates that water sloshing will have insignificant effects on the fuel storage rack due to the depth of the pool relative to the rack height.

The results of the analysis for Load Case 6 indicates that the drop of a fuel assembly onto a fuel storage cell will not collapse or buckle the cell, thereby precluding a significant change in rack geometry. The analysis shows that the external kinetic energy of the dropped fuel assembly is absorbed in the local deformation of the flare at the top of the fuel storage cell, in the partial shearing of the cell leg weld, in the local crumbling of the pool floor concrete and in the minor deformation of the pool floor liner plate under each leg. The leak tightness of the fuel pool, however, will be maintained since the deformation is insufficient to result in a tear or puncture of the liner.

It has, therefore, been concluded from the results of the seismic and structural analysis that the deflections and/or stresses in the rack structure resulting from the various loadings meet the deflection and stress acceptance criteria for Seismic Category I structure.

2.3.3 Storage Rack Thermal-Hydraulic Analysis

The adequacy of natural circulation flow to cool the spent fuel assemblies in the rack matrix was verified by establishing for the East/West rack row with the maximum number of assemblies, a thermal-hydraulic balance between the driving head produced by decay heat generation and the pressure losses existing in the natural circulation flow path. Pressure losses in the downcomers, in the rack inlet plenum, and along the fuel assemblies were explicitly considered in the analysis. Cross-flows in the inlet plenum area were conservatively neglected.

The results of the thermal-hydraulic analyses indicate that even with the most conservative assumptions, the natural circulation in the spent fuel pool is adequate to preclude local boiling by a substantial margin. The maximum temperature increase in the assembly with the minimum flow is less than 21° F. This results in a maximum outlet temperature of less than 206° F for the maximum heat load (assumed maximum bulk temperature of 185° F), which is still below the pool water saturation temperature of 239° F at the highest assembly elevation.

3.0 Spent Fuel Pool Structural and Seismic Analysis

The spent fuel pool, located inside the fuel handling area in the auxiliary building, is a reinforced concrete pool lined with ¼ inch thick stainless steel. The auxiliary building, as well as the storage pool, is a seismic Class I structure which is designed to withstand seismic, tornado, and thermal loads as discussed in Sections 3.7 and 3.8 of the Davis-Besse Unit 1 FSAR.

The spent fuel pool structure has been analyzed to determine the effects due to the proposed design for the high capacity spent fuel storage racks. The spent fuel racks interact with the spent fuel pool structure through thermal stress loads, dead weight loads, and seismically induced loads.

As discussed in Section 2.3.2.3, thermal stress loads are insignificant.

As stated in Section 2.2, vertical seismic and dead weight loads are essentially transmitted directly to the pool floor by each storage cells. The four corner cells of each module transmit the load of the grid structures and bracing, and thus transmit somewhat higher loads than interior cells. All vertical seismic and dead weight loads are within design limits for the pool structure.

Horizontal seismic loads are transmitted to the pool walls from seismic bracing attached to each of the two levels of the rack grid structure. Figure 1 shows a plan view of the locations for the bracing. The horizontal seismic loads for the Operating Basis Earthquake (OBE) and Safe Shutdown Earthquake (SSE) were analyzed and found to be well within design limits for the 5½ foot walls on the north, east and south sides of the pool. The loads on the west wall, the 3 foot wall between the spent fuel pool and the fuel transfer tube pit, were found to be well within design limits for the OBE but were found to exceed design limits (0.9 of yield strength) for the SSE. To maintain a conservative margin of safety for this wall, the Applicant will modify the pool structure by providing adequate struts spanning the fuel transfer tube pit to transmit a portion of the loads to the 5½ foot wall on the west side of the fuel transfer tube pit (see Figure 1). The struts will be in place whenever there is; (1) spent fuel stored in the pool and (2) the fuel transfer tube pit is not filled with water. With the fuel transfer tube pit filled with water, the hydrostatic forces acting on the 3 foot wall between the spent fuel pool and fuel transfer tube pit are equalized. Under these conditions all affected walls are well within design limits for SSE loadings without the additional bracing. The fuel transfer tube pit is filled with borated water during refuelings and normally drained otherwise.

4.0 Spent Fuel Pool Cooling System Evaluation

The spent fuel pool cooling system is designed to maintain the borated spent fuel pool water quality and clarity and to remove the decay heat from the stored fuel in the spent fuel pool. A detailed discussion of the system is presented in Section 9.1.3 of the Davis-Besse Unit 1 FSAR (see Appendix A of this document for applicable changes to this material).

The capability of the existing spent fuel pool cooling system to handle heat loads resulting from the expanded spent fuel storage has been calculated for both the normal and the emergency heat load cases. Decay heat generation is calculated according to NUREG-75/087, Section 9.2.5, Branch Technical Position APCSB 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling", 11-24-75. The values used in this analysis include recommended uncertainty factors and contributions of actinides (U-239 and Np-239). For the decay heat calculations it was conservatively assumed that the average discharge batch was 60 fuel assemblies and the batch underwent an equivalent continuous operating period of 3 years at a core thermal power of 2772 Mwt with an average cooling time before discharge to the spent fuel pool of 150 hours. Refuelings were assumed to occur on an annual basis.

The maximum normal heat load results with the pool filled with one freshly discharged batch in addition to 11 batches from previous refueling outages. Under these conditions, the decay heat generated is calculated to be 12.4×10^6 Btu/hr. With this heat load the existing spent fuel pool cooling system, with both pumps and both heat exchangers operating, is capable of maintaining the spent fuel pool at 125° F or less. With one pump and two heat exchangers operable the pool can be maintained at 140° F or less, and with one pump and one heat exchanger operable the pool can be maintained at 155° F or less under the maximum normal heat load conditions.

The maximum abnormal heat load would result when one entire core (177 assemblies) is discharged 150 hours after shutdown, 65 days after the last of 9 batches from previous refueling outages. Under these conditions the decay heat generated is calculated to be 29.5×10^6 Btu/hr. The station is designed such that the decay heat removal system (see Section 6.3 of the Davis-Besse Unit 1 FSAR) is used to remove the decay heat from the spent fuel pool under full core discharge conditions and serves as back-up system to the spent fuel pool cooling system under normal conditions. Each of the two decay heat removal trains is designed for a heat removal capacity of 30×10^6 Btu/hr. at an inlet temperature of 140° F.

The seismic Class I decay heat system is permanently connected to the Class I boundary of the spent fuel pool cooling system. The decay heat system thus will serve to make up the spent fuel pool water by supplying the borated water from the BWST to the fuel pool to prevent uncovering of the fuel should the need ever arise.

Based on the discussion above, it is concluded that the existing spent fuel pool cooling system provides adequate cooling capability to safely handle the additional heat loads caused by the expanded storage capacity. The designed availability of the decay heat removal system as a back-up heat removal and makeup system for the spent fuel pool provides assurance of maintaining the pool in a safe thermal condition.

5.0 Radiological Consequences

The radiological consequences of increased fuel pool storage capacity were evaluated based on a conservative refueling scenario. Spent fuel equal to 1/3 core was assumed to be deposited in the fuel pool at one year intervals for a period of 9 years to simulate normal refueling capacity. This was assumed to be followed by a final full core unload. Fuel was assumed to be stored in the high density spent fuel racks.

The following doses of interest were evaluated.

1. The dose obtained directly from the stored fuel assemblies, assuming the technical specification requirement of 23 ft. of water above the stored fuel assemblies.

2. The dose obtained from those radionuclides which become suspended or dissolved in the fuel pool water.

3. The dose obtained from those radionuclides which escape the fuel pool into the fuel building atmosphere.

Direct Gamma Dose From Stored Fuel

The buildup of fission products within the spent fuel assemblies was based upon one year of full power operation. These products were assumed to decay for a period of 72 hours before being transferred to the spent fuel pool to account for the technical specification required 72 hour decay period. The dose rate after a final full core unload was calculated at the fuel pool water surface. Previously stored fuel from normal refuelings was assumed to have decayed for a maximum of one year. In accordance with the above assumptions, the maximum direct gamma dose rate at the pool water surface from stored spent fuel is on the order of 10^{-7} mrem per hour.

The gamma dose rate through the side walls of the spent fuel pool will be greatly dependent upon the loading arrangement of the spent fuel in the racks. Use of proper radiation control measures in the affected areas will minimize the potential for any additional occupational exposure due to the increased number of spent fuel assemblies.

The dose rate through the bottom of the pool will approximately double for a period of one month under the one third of core just removed from the reactor. However, much of the area under the pool is already an "E" zone, the rest is a "C" zone which can be temporarily posted and barricaded if necessary for such a short period. (Reference Chapter 12 of the DavisBesse Unit 1 FSAR.)

Dose From Dissolved Radionuclides

Increasing the number of assemblies in the pool has not led to increased concentrations of radionuclides in the spent fuel pool water at other operating plants. Measurements taken of the activity in the pool water before and after refueling have indicated essentially no change in concentrations. As a result, it is not expected that there will be more frequent changing of demineralizer resin or filter cartridges. Therefore, little increase in annual-man rem is expected from either the radionuclides in the pool water or accumulated on the resin or filters.

Dose From Airborne Radionuclides

Almost all of the leakage of fission products from the fuel will occur for each batch of fuel within a few months of when it is removed from the core. Therefore, most of the inhalation and submersion doses will be due to the latest batch of fuel placed in the pool, the increase in dose due to the additional fuel stored in the pool with several years decay will be negligible, (I-131, Xe-133, and other shorter half life iodines and noble gates will have decayed, so that Kr-85 will be the only volatile fission product left in the fuel).

As discussed previously, there will be a slightly increased heat load on the pool due to the storage of additional fuel with several years decay. The incremental increase in the temperature of the pool water from 120°F to 125°F due to this increased heat loading will lead to a slightly higher tritium concentration in the air due to increased evaporation. This increase in temperature above the original design of 120°F will only be for a period of approximately 10-15 days after a refueling. After that time the newly discharged spent fuel will have decayed sufficiently to reduce the heat load such that the spent fuel pool cooling system can maintain the pool less than 120°F. Calculations show that the resulting small increase in tritium concentration in the air will be well within the MPC limit in 10 CFR 20 Appendix B, Table I, Column I and will allow normal occupancy in the spent fuel pool area.

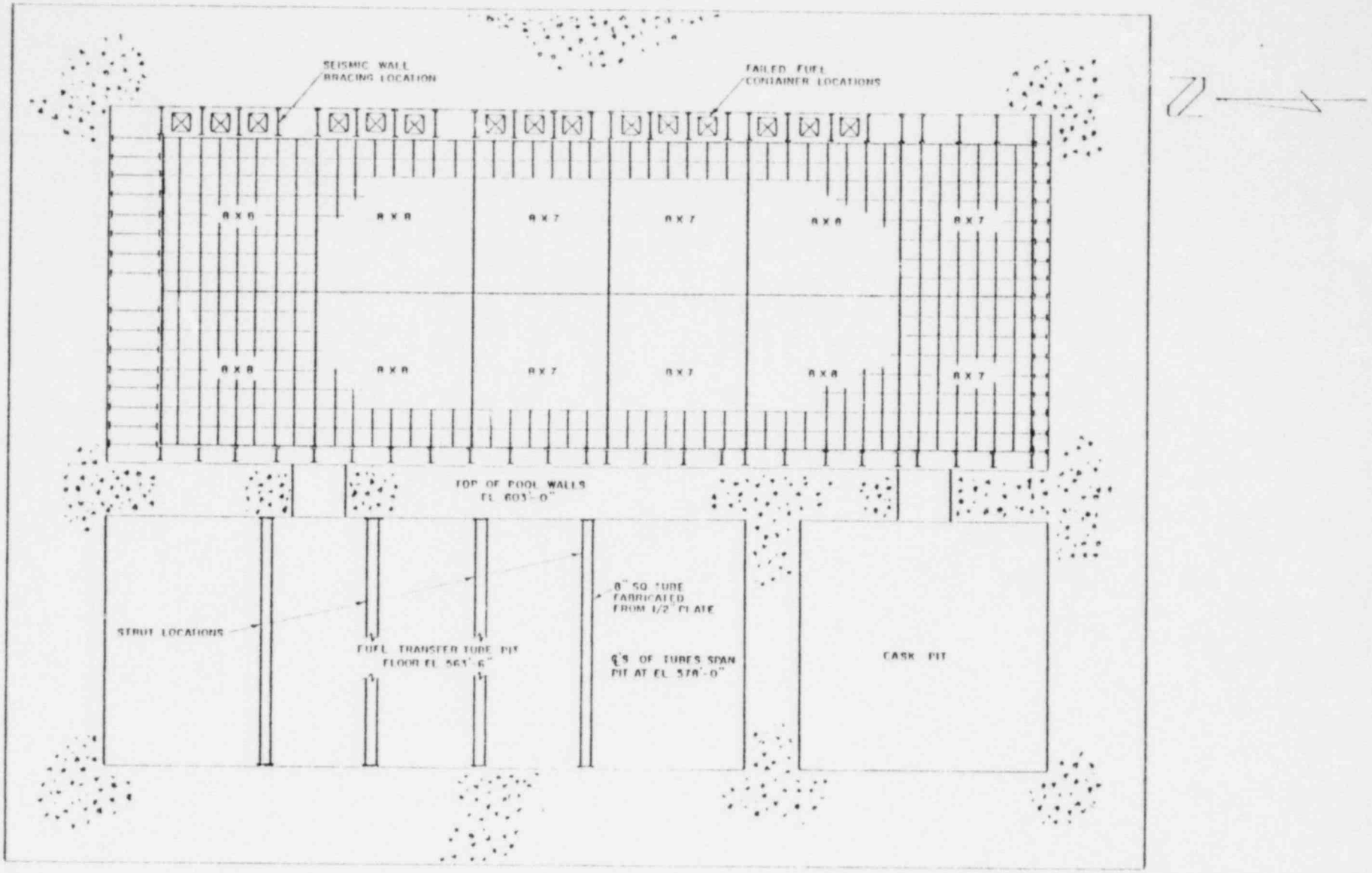
Liquid and Gaseous Releases

The storage of additional spent fuel assemblies in the spent fuel pool will not result in any additional liquid release from the plant.

To determine the potential for gaseous releases due to the spent fuel stored in the pool, the activity of important iodine and noble gas isotopes were calculated and compared to the Davis-Besse Unit 1 FSAR analyses. Almost all of the releases occur from each one third of a core within a short period after it is removed from the reactor. Very little of the releases will be from the older fuel which has been stored for several years. The increase in the dose to an individual at the site boundary will be insignificant.

Radiation Exposure During Modifications

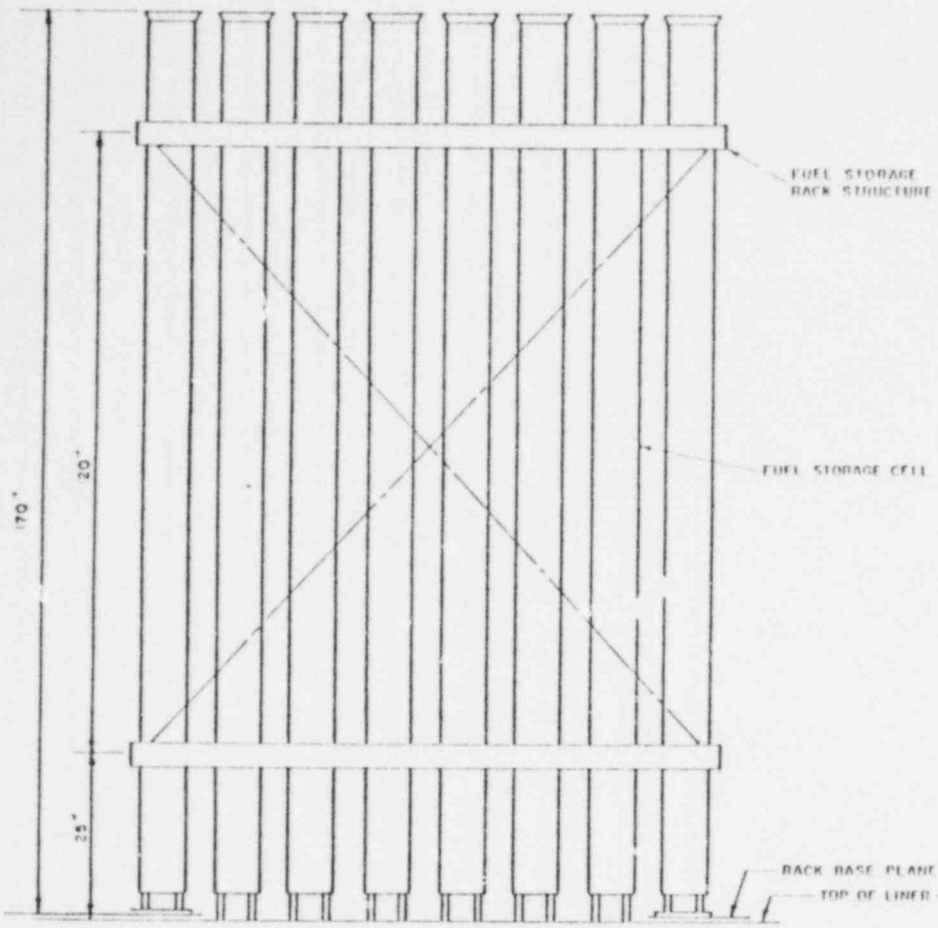
This modification will require the disposal or storage of the existing spent fuel racks which are not expected to be contaminated, since they will be replaced before the first refueling.



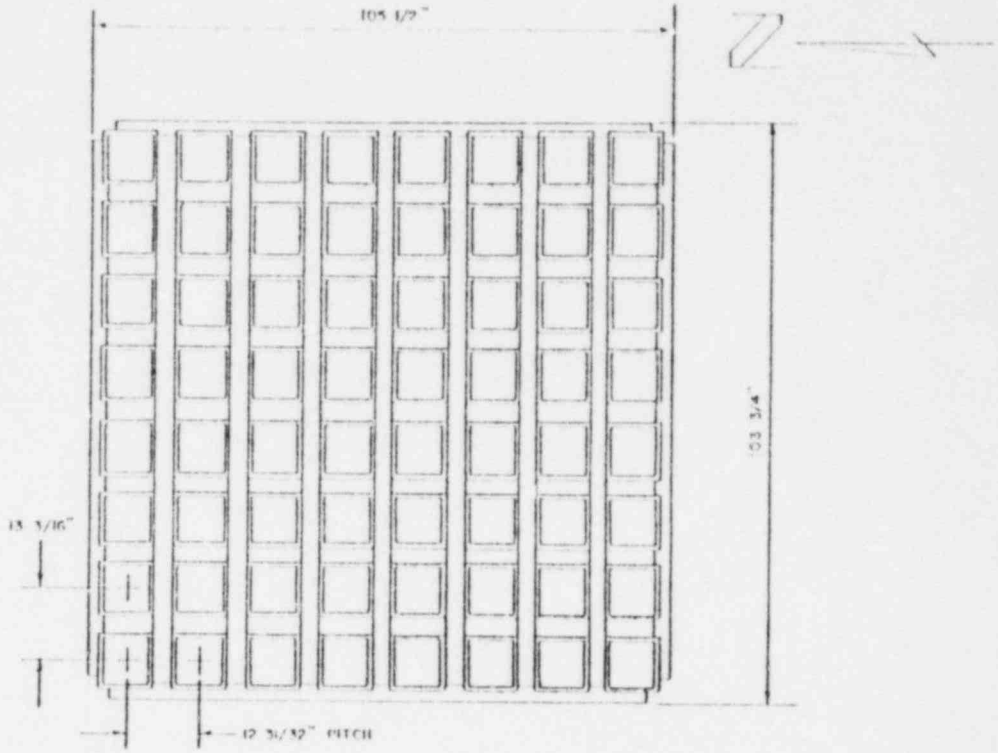
PLAN

SCALE 3/16"=1'-0"

DAVIS-BESSE NUCLEAR POWER STATION
SPENT FUEL STORAGE RACK
GENERAL ARRANGEMENT
FIGURE 1



SIDE VIEW



TOP VIEW

DAVIS-BESSE NUCLEAR POWER STATION
SPENT FUEL RACK STORAGE MODULE

FIGURE 2

Appendix A

Davis-Besse Nuclear Power Station Unit 1

Modified FSAR Material

1.2.8.2.2 Chemical Addition System

Chemical addition operations are required to alter the concentration of various chemicals in the reactor coolant and auxiliary systems. The system is designed to add boric acid to the reactor coolant system for reactivity control, lithium hydroxide for pH control, and hydrazine for oxygen control.

1.2.8.2.3 Cooling Water Systems

The cooling water systems remove heat from the station equipment to permit a sustained operation and safe shutdown of the station.

Condenser Circulating Water System

The condenser circulating water system is sized to handle the maximum condenser heat loads and consists of a closed system utilizing a hyperbolic neutral draft cooling tower and the associated circulating water pumps, piping and valves. Fill and makeup water is taken from Lake Erie through the intake water system and intake structure. Four circulating water pumps with suction from the cooling tower discharge channel pump through the condenser and back to the cooling tower.

Service Water System

The service water system takes Lake Erie water from the intake structure pump suction pit after the traveling screens. This system supplies cooling water to the component cooling water system, ECCS pump room coolers, containment air coolers, the turbine plant cooling water systems, and is a source of water to the auxiliary feed pumps. It also provides a source of makeup water to the cooling tower.

Component Cooling Water System

This system is a closed loop system which provides cooling water to the nuclear and engineered safety features systems and also acts as an intermediate barrier between the radioactive system and the service water system. The system consists of three circulating pumps, three heat exchangers, a surge tank, associated valves, piping, instrumentation, and controls.

Turbine Plant Cooling Water System

The recirculated closed loop system furnishes purified and treated cooling water to main turbine and turbine plant pump oil coolers, various pump seals, generator hydrogen equipment auxiliaries including generator hydrogen coolers and stator liquid cooler, isolated phase bus, air compressor jackets and coolers, and turbine plant sample coolers.

The engineered safety features equipment is not dependent on the turbine plant cooling water system.

1.2.8.2.4 Spent Fuel Pool Cooling System

The spent fuel pool cooling system is designed to maintain the borated spent fuel pool water at 125 F or less with heat load based on removing decay heat from

1/3 core, which is assumed to have undergone irradiation for an equivalent continuous operating period of 3 years at full core thermal power and to have been cooled for 150 hours, and the decay heat from 11 batches of spent fuel from previous refueling outages.

The unit is, however, designed such that if it becomes necessary at some time to off-load an entire core into the spent fuel pool, the cooling capacity can be provided by the decay heat removal system.

During normal operation, both pumps and both heat exchangers are in continuous operation; as the decay heat emitted by the spent fuel decreases one pump and one heat exchanger can be shut down. The spent fuel water temperature is normally maintained at 125 F or less.

During cold shutdown and refueling conditions, the reactor refueling canal is filled with water from the borated water storage tank.

1.2.8.2.5 Decay Heat Removal System

The normal function of this system is to remove reactor decay heat during the latter stages of cooldown and maintain reactor coolant temperature during refueling.

1.2.8.2.6 Sampling System

The sampling system provides samples for laboratory analyses which serve to guide the operation of the reactor coolant system, the makeup and purification system, the chemical addition system and the power conversion steam system. These samples flow to central locations in the auxiliary and turbine buildings; access to the containment vessel for this purpose is not required during power operation. Typical of the analyses performed on such samples are reactor coolant boric acid concentration, pH, fission product activity levels, dissolved gas content, corrosion product concentration and activity and main steam gross activity. Analytical results are used for regulating boron concentration adjustments, evaluating the integrity of fuel rods and the performance of the demineralizers, and regulating chemical addition to the the reactor coolant.

1.2.8.2.7 Station Ventilation Systems

The heating, ventilating, and air-conditioning systems for the station are designed to provide a suitable environment for equipment and personnel with equipment arranged in zones so that potentially contaminated areas are separated from clean areas. The path of ventilating air in the auxiliary building is from areas of low activity toward areas of progressively higher activity. Conditioned air is recirculated in clean areas only.

The containment air recirculation system is used to circulate air within the containment vessel. The emergency ventilation system ventilates the shield building and penetration rooms.

9.1.2 SPENT FUEL STORAGE

9.1.2.1 Design Bases

The spent fuel storage is designed to store the irradiated fuel assemblies under water for decay prior to shipment offsite for reprocessing. The storage pool is sized to store 735 irradiated fuel assemblies which includes storage for 15 failed-fuel containers. The spent fuel storage cells are installed in parallel rows with center-to-center spacing of 12 31/32 inches in one direction, and 13 3/16 inches in the other orthogonal direction. This spacing and "flux trap" construction, whereby the fuel assemblies are inserted into neutron absorbing stainless steel cans, is sufficient to maintain a K_{eff} of 0.95 or less. Shielding and seismic classification are discussed in subsection 9.1.2.2.

The design of the spent fuel storage area closely follows the intent of Safety Guide 13.

9.1.2.2 Description

After removal from the reactor, the spent fuel is stored under water within the spent-fuel storage pool. The storage pool is a reinforced concrete pool lined with 1/4-inch-thick stainless steel. It is located inside the fuel-handling area in the auxiliary building. The auxiliary building, as well as the storage pool, is a seismic class I structure which is designed to withstand seismic, tornado, and thermal loads as discussed in sections 3.7 and 3.8. The spent-fuel storage racks are also seismic class I structures which are designed to withstand seismic loadings. The mass model is shown in figure 9-25a. The fuel-handling area is also protected against tornado-generated missiles and other potential missiles.

Adequate shielding is provided for station personnel by the 5-1/2-foot-thick concrete walls and borated water in the pool. The radiation zones around the spent fuel pool are shown in figures 12-2 and 12-3.

The spent-fuel racks (not including the failed fuel container locations) are arranged in a 16 X 45 array constructed of six 7 X 8 modules and six 8 X 8 modules. The arrangement is shown in figure 9-3A and 9-3B. The location of the storage pool within the station complex is shown in figures 1-6 and 1-7.

A separate space is provided for loading the spent fuel shipping cask. The spent fuel cask pit is independent of and separated from the spent fuel pool by a 3-foot-thick concrete wall. The only communication between the spent fuel pool and the cask pit is through the 24-inch-wide slot opening provided for the transfer of the spent fuel assemblies from storage to the shipping cask. This opening is provided with a watertight bulkhead which can isolate the spent fuel pool when needed. Following sufficient decay, the spent fuel assemblies can be removed from storage and loaded into the spent fuel shipping cask under water for removal from the site. Casks up to 140 tons in weight can be handled by the spent fuel cask crane.

A cask-wash-and-decontamination area is also provided adjacent to the cask pit. In this area, outside surfaces of the cask can be decontaminated before shipment.

9.1.2.3 Safety Evaluation

The spent fuel storage facility is designed for noncriticality by use of adequate spacing and "flux trap" construction whereby the fuel assemblies are inserted into neutron absorbing stainless steel cans. The spent fuel storage racks are designed to prevent accidental insertion of a fuel assembly in other than the prescribed locations,thereby ensuring a safe geometric array.

All spent fuel assembly transfer operations are conducted under a minimum of 9-1/2 feet of borated water above the top of the active fuel assembly. All piping penetrations into the spent fuel pool penetrate at least 9 feet above the top of the fuel assemblies to avoid any possibility of completely draining the pool in case of a pipe rupture. Isolation valves are provided on the pipes penetrating the pool. These valves are located as close to the concrete wall as practicable to minimize the possibility of pipe failure between the isolation valves and the pool.

The spent fuel pool water is cooled by the spent fuel pool cooling system as discussed in subsection 9.1.3.

The spent fuel cask crane is electrically interlocked to prevent the crane from traveling over the spent fuel pool while any load is hanging on the main hook. This interlock can be bypassed only with a key. Even upon bypassing this interlock, the main hook stays inoperative; only the auxiliary hook can be used.

The cask pit is separated and isolable from the pool to preclude the possibility of draining the spent fuel pool in case of damage to the cask pit by an accidental drop of a cask in the pit. The base of the cask pit is solid concrete extending down to the foundation. Thus, a cask drop is not postulated to do any significant damage to the structure.

The storage racks are designed to eliminate any possibility of fuel assembly sticking in the racks. All projections and corners are properly tapered and rounded off. The spent-fuel assemblies are placed into, and removed from, the racks by the spent-fuel handling bridge crane. Since the fuel assemblies make free contact with the storage cells, there would be no uplift force exerted on the racks. The spent-fuel-handling bridge crane is provided with an overload interlock on the hoist which shuts off the power to the hoist any time the load on the hoist exceeds 2700 pounds. The racks are designed to withstand this uplift force.

9.1.3 SPENT FUEL POOL COOLING AND CLEANUP SYSTEM

9.1.3.1 Design Bases

The spent fuel pool cooling system is designed to maintain the borated spent fuel pool water quality and clarity and to remove the decay heat from the stored fuel in the spent fuel pool. It is designed to maintain the spent fuel pool water at approximately 125 F, with a heat load based on removing the decay heat generated from 1/3 of the core fuel assemblies which are assumed to have undergone infinite irradiation and to have been cooled in the reactor for an average of 150 hours prior to being stored in the pool, plus the decay heat generated by the previous 11 batches from prior annual refuelings.

The decay heat removal system described in section 6.3 serves as a back-up system to the spent fuel pool cooling system under normal conditions and is used to remove the decay heat from the spent fuel pool should it be necessary to off-load the entire core into the spent fuel pool.

In addition to its primary function, the spent fuel pool cooling system provides for purification of the spent fuel pool water, the fuel transfer canal water, and the contents of the borated water storage tank to remove fission and corrosion products and to maintain water clarity.

The radiation level and shielding are described in Chapter 12.

9.1.3.2 System Description

The spent fuel pool cooling system is shown in figure 9-1. It consists of two half capacity recirculating pumps and two half capacity heat exchangers, associated valves, piping and instruments. A bypass system consists of a demineralizer and a filter.

System performance data are shown in table 9-2. Major components of the system are briefly described below.

9.1.3.2.1 Spent Fuel Pool Heat Exchangers

The spent fuel pool heat exchangers are designed to maintain the temperature of the spent fuel pool water noted in section 9.1.3.1.

9.1.3.2.2 Spent Fuel Pool Pumps

The spent fuel pool pumps take suction from the spent fuel pool and recirculate the water back to the pool after it passes through the heat exchangers, deminerizer and/or filter in various combinations, depending on conditions.

9.1.3.2.3 Spent Fuel Pool Demineralizer

The spent fuel pool demineralizer can remove approximately fifty percent of the fission products contained in the spent fuel pool water in 34 hours.

9.1.3.2.4 Spent Fuel Pool Filter

The spent fuel pool filter is designed to remove particulate matter from the spent fuel pool water. The filter is sized for the same flow rate as the demineralizer (100 gpm).

9.1.3.2.5 Borated Water Storage Tank Recirculation Pump

The borated water storage tank recirculation pump recirculates water from the borated water storage tank through the spent fuel pool cleanup system for demineralizing and filtering. The pump may also be used for demineralization and filtering the water in the fuel transfer canal during a transfer of fuel.

The pump will be used for draining a portion of the refueling canal, fuel transfer pit and cask pit after completion of the fuel transfer operation. During the winter, the pump will also serve to maintain the borated water storage tank temperature by circulating water through an external heater to prevent the tank water from freezing.

9.1.3.2.6 Spent Fuel Pool Skimmers

Surface skimmers are provided in the spent fuel pool to facilitate the removal of accumulated particulate matter from the surface of the spent fuel pool water.

9.1.3.3 Modes of Operation

9.1.3.3.1 Normal Operation

The spent fuel pool cooling system serves two main functions. The first is to remove the decay heat generated by spent fuel stored in the pool as a result of normal refueling conditions and, the second function is to provide purification of the spent fuel pool water for clarity during fuel handling operations.

The first function is accomplished by recirculating spent fuel pool cooling water from the spent fuel pool through the pumps and heat exchangers and back to the pool. The spent fuel pool pumps take a suction from the pool and deliver pool water through the tubeside of two heat exchangers arranged in parallel back to the pool. The maximum normal heat load results with the pool filled with one freshly discharged batch in addition to 11 batches from previous refueling outages. With this heat load and both pumps and heat exchangers operating, the spent fuel pool cooling system is capable of maintaining the spent fuel pool at 125 F or less. With one pump and two heat exchangers operable the pool can be maintained at 140 F or less and with one pump and one heat exchanger operable the pool can be maintained at 155 F or less under the maximum normal heat load conditions.

The second function is accomplished by providing a bypass purification system. The bypass loop branches off from the spent fuel pool pump discharge cross-connect line, bypassing the heat exchangers. After demineralizing and filtering, the bypass flow is directed into the normal line downstream of the heat exchanger and returns to the pool.

The purification will also be utilized to purify the water in the borated water storage tank following refueling, and to maintain clarity in the fuel transfer canal during refueling. Water from the borated water storage tank or fuel transfer canal can be purified by using the borated water storage tank recirculation pump.

9.1.3.3.2 Abnormal Conditions

The maximum abnormal heat load would result when one entire core (177 assemblies) is discharged 150 hours after shutdown, 65 days after the last of 9 batches from previous annual refueling outages. The unit is designed such that under these conditions or other full core discharge conditions, the decay heat removal system is used to remove the decay heat from the spent fuel pool. One decay heat train alone is capable of maintaining the spent fuel pool at about 140 F or less under the maximum abnormal heat load described.

9.1.3.4 Reliability Considerations

The spent fuel pool cooling system provides adequate capacity and component redundancy to ensure the reliable cooling of spent fuel stored in the spent fuel pool. Ample time is available to ensure that cooling can be restored even in the unlikely event of multiple component failures or complete cooling loss. The system is so arranged that no uncontrolled, complete loss of water from the pool is possible by piping or component failures. The system performs no emergency functions and is not directly connected to the reactor coolant system.

The decay heat removal system, which has a higher heat removal capacity, serves as a back-up system to the spent fuel pool cooling system.

9.1.3.5 Codes and Standards

Each component of this system is designed to the code or standard, as applicable, as noted in table 9-1.

9.1.3.6 Fuel Leakage Considerations

If a leaking fuel assembly is transferred from the refueling canal to the spent fuel pool, a small quantity of fission product activity may enter the spent fuel pool cooling water, even though the assembly's cladding temperature is lowered, and leakage should be minimized. The purification loop removes these fission products and other contaminants from the pool water. Radiological evaluation is presented in Chapter 11 and Chapter 12.

removed and a thorough visual inspection is made of the assembly. The fuel assembly is then placed into the new fuel elevator. The new fuel elevator is provided to lower the new fuel assembly under water, thereby eliminating the need to lower the crane hook into water. After the new fuel assembly has been lowered to the bottom of the fuel transfer pit, it is picked up by the fuel grapple on the spent fuel handling bridge crane and placed into the fuel transfer mechanism. The tilting mechanism on the transfer mechanism rotates the fuel assembly from a vertical to a horizontal position. The transfer carriage transfers the fuel assembly through the transfer tube to the inside of containment. Inside the containment a second tilting mechanism rotates the fuel assembly back to the vertical position. The main fuel handling bridge inside the containment removes the fuel assembly from the transfer carriage and places it in the reactor.

The procedure for the removal of the spent fuel from the reactor is similar to the one above in reverse order. The spent fuel assembly is removed from the transfer carriage in the auxiliary building by the spent fuel handling bridge crane and is placed into the spent fuel storage racks for decay prior to off-site shipment.

Once refueling is completed, the refueling canal water is drained and pumped to the borated water storage tank.

9.1.4.3 Shipping Spent Fuel

The spent fuel assemblies will be stored in the spent fuel pool prior to their shipment offsite.

The spent fuel shipping cask can be received at the site either by truck or railroad.

Upon arrival, the cask, on the railroad car (or truck), is inspected for any evidence of physical damage. The cask is then unloaded from the railroad car (or truck) with the spent fuel cask crane and placed in the cask wash area. The cask is washed, scrubbed, and steam cleaned to remove all road dirt and grime. After thorough cleaning, the lid on the cask is unbolted, removed and stored. The cask is lifted from the wash area (Figure 9-28) and lowered into the cask pit. If the cask pit is empty to start with it is filled with the borated water from the borated water storage tank to elevation 601 feet 5 inches. The bulkhead between the spent fuel pool and the cask pit is removed to establish communication between the two. The spent fuel is now picked up from the storage racks by the spent fuel bridge crane and placed into the cask. Depending on the size of the cask, as many as 10 spent fuel assemblies may be shipped in one cask. When the cask is fully loaded, still in the cask pit, the lid is placed on top of the cask to provide shielding when the cask is lifted out of the water. When the cask is partially out of water, two or three bolts are loosely installed to keep the lid in place. The cask is now lifted out of the pit and placed in the cask wash area. The cask is connected to a cooling system for the removal of decay heat from the fuel assemblies. After all of the head bolts are installed and properly torqued, the cask is washed and decontaminated, and the surface radiation level is checked. When it is below the Department of Transportation limits specified in 49 CFR Part 171-178, it is ready for shipment. The

cask is then placed on the railroad car (or truck) and connected to its cooling system and shipped offsite to the reprocessing plant.

9.1.4.4 Safety Provisions

Safety provisions are designed into the fuel handling system to prevent the development of hazardous conditions in the event of component malfunctions, accidental damage, or operational and administrative failures during refueling or transfer operations. A mechanical lock prevents disengagement of the fuel assembly grapple latches as long as a fuel assembly weight is suspended from the grapple mechanism. Bridge and trolley controls are interlocked to prevent movement until the fuel assembly has been completely withdrawn into the protective mast tube.

The new and spent fuel assembly storage facilities are designed for noncriticality by use of adequate spacing and, in the case of the spent fuel racks, by use of a stainless steel "flux trap" design. The new and spent fuel storage racks are designed to prevent insertion of a fuel assembly in other than the prescribed locations, thereby ensuring a safe geometric array. A safe condition is ensured even if new fuel is immersed in unborated water. Under these conditions, a criticality accident during refueling or storage is not credible.

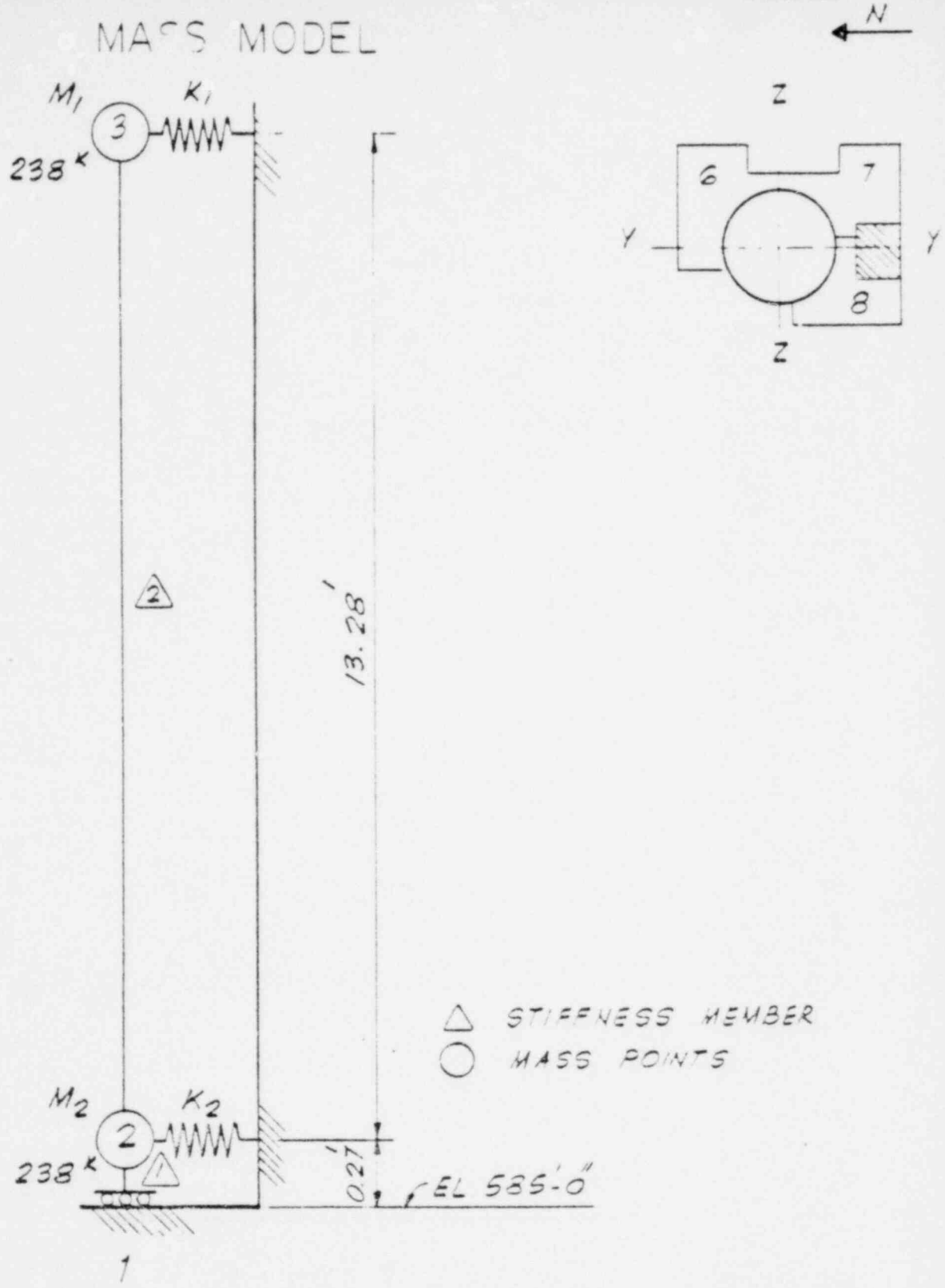
All spent fuel assembly transfer operations are conducted under water. The water level in the refueling canal provides a minimum of 9-1/2 ft of water over the top of the active fuel in the spent fuel assemblies during movement from the core into storage. The depth of the water over the fuel assemblies, as well as the thickness of the concrete walls of the refueling canal, is sufficient to limit the maximum continuous radiation levels in the working area to values consistent with the radiation zoning described in Chapter 11.

The spent fuel storage pool water is cooled by the spent fuel cooling system as described in section 9.1.3. A power failure during the refueling cycle will create no immediate hazardous condition owing to the large water volume in both the refueling canal and spent fuel storage pool.

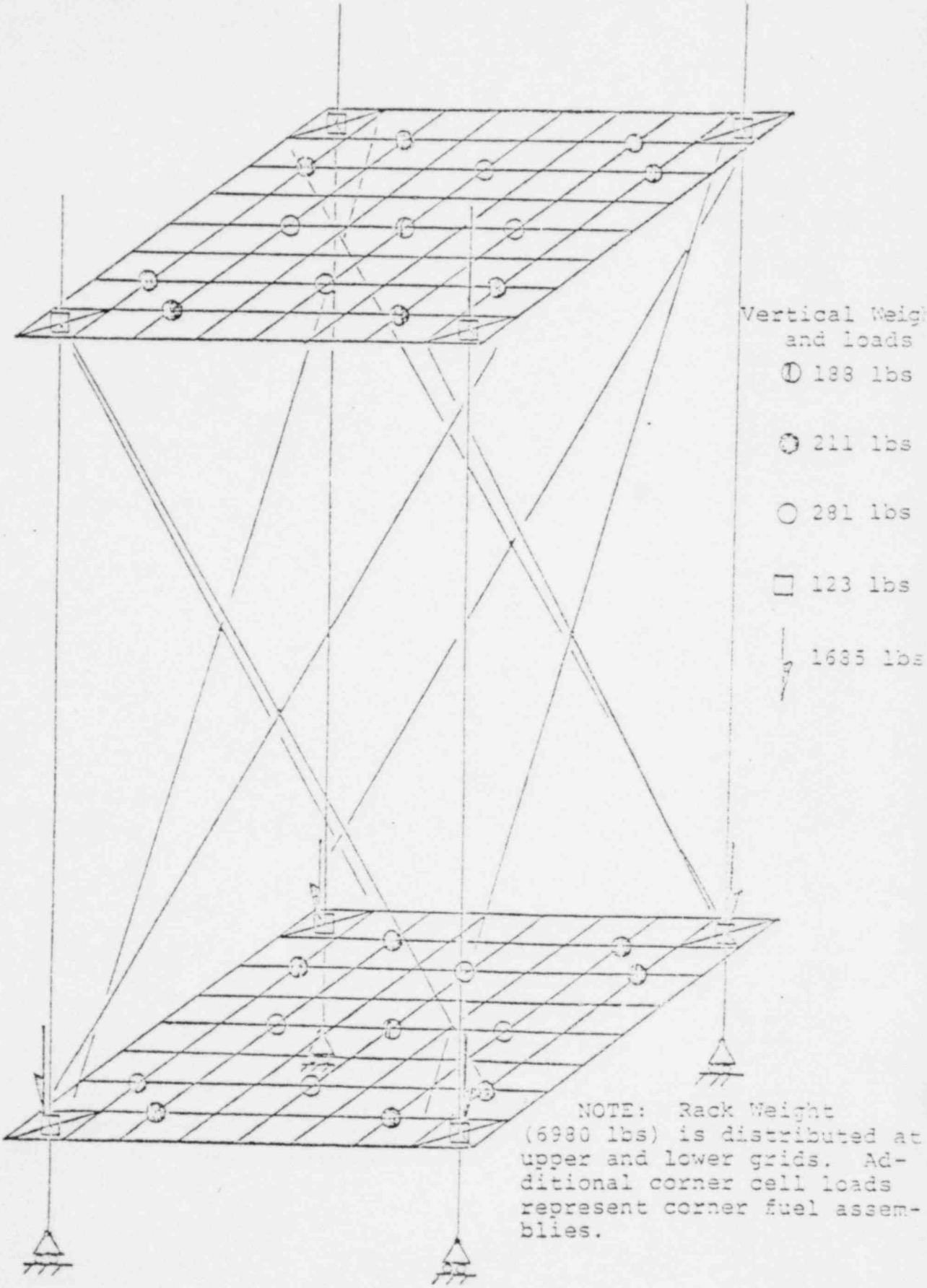
During the refueling period the water level in both the refueling canal and the spent fuel storage pool is the same, and the fuel transfer tube valve is continuously open. This eliminates the necessity for an interlock between the fuel transfer carriage and fuel transfer tube valve operations except to verify full open valve position.

The simplified movement of a transfer carriage through the horizontal fuel transfer tube minimizes the danger of jamming or derailing. All operating mechanisms of the system are located in the fuel handling area for ease of maintenance and accessibility for inspection before the start of refueling operations.

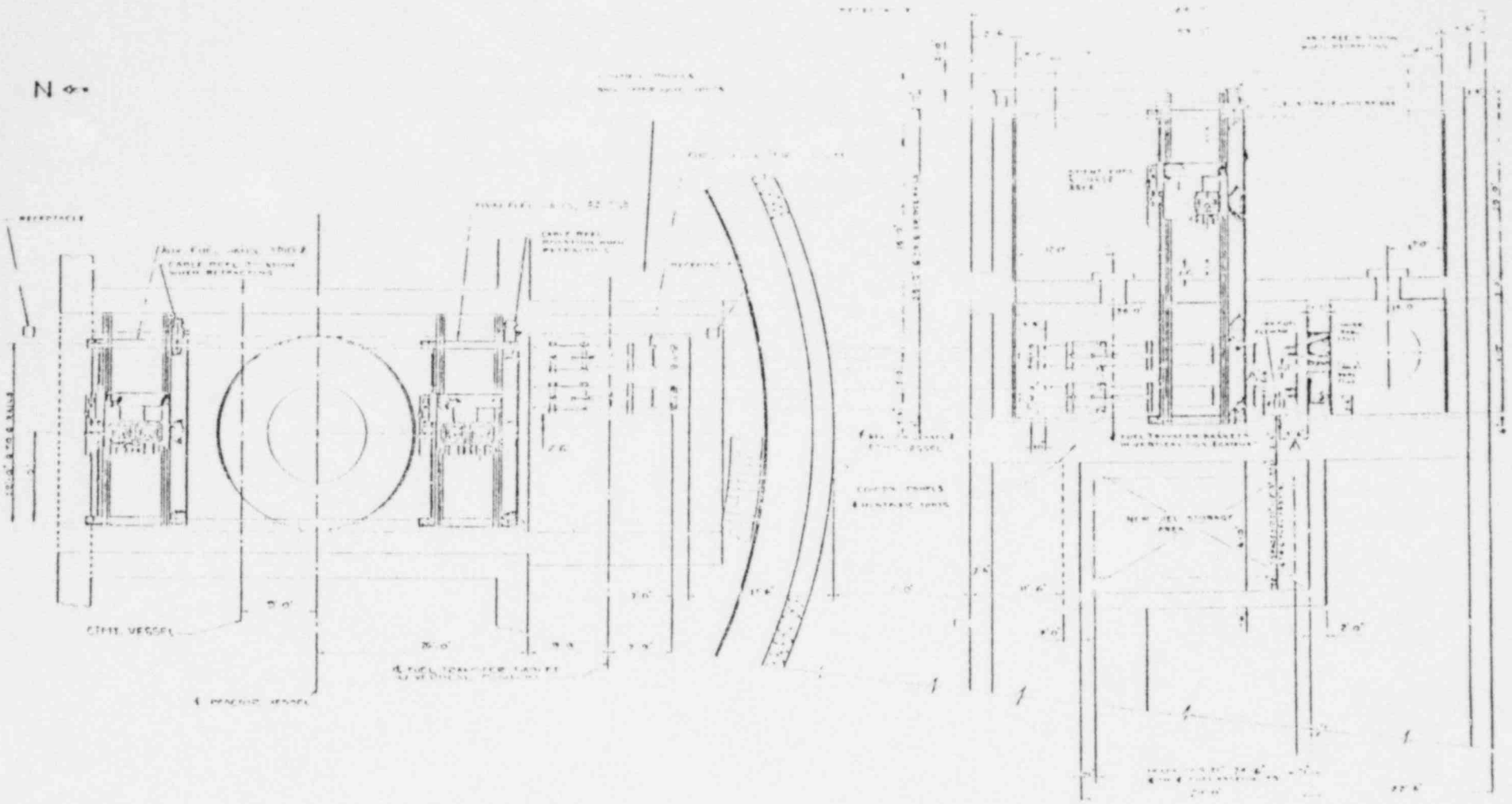
During reactor operation, a bolted closure plate and gasket on the containment vessel flange of the fuel transfer tube and the fuel transfer tube valve on the fuel handling area end of the tube provide containment vessel isolation as described in section 6.2.4. Both the spent fuel storage pool



DAVIS-BESSE NUCLEAR POWER STATION
MASS MODEL FOR NEW FUEL
STORAGE RACKS
FIGURE 9-25



Davis-Besse Nuclear Power Station
Mass Model for Spent Fuel Storage Racks
Figure 9-25a



DAVIS BESSE NUCLEAR POWER STATION
FUEL HANDLING SYSTEM
(PLAN)
FIGURE 9.26

REVISION 1