VERMONT YANKEE

SPENT FUEL STORAGE RACK

REPLACEMENT REPORT

APRIL 1986

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VERMONT YANKEE NUCLEAR POWER STATION

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REPLACEMENT REPORT

1.0 INTRODUCTION

This report is submitted in support of Vermont Yankee Nuclear Power Corporation's (VYNPC) application to amend the Vermont Yankee Nuclear Power Station License DPR-28 for the purpose of increasing the storage capacity of the existing Vermont Yankee Spent Fuel Pool (SFP).

The SFP at Vermont Yankee currently contains aluminum storage racks for 1,680 Boiling Water Reactor (BWR) fuel assemblies with a center-to-center spacing of 7.0 inches. The facility's license, as amended in 1977, allows for storage of up to 2,000 assemblies. This report describes the design of high density storage racks which will increase the storage capacity of the SFP to 2,870 assemblies. The new racks are freestanding stainless steel modules containing a neutron absorber (Boral) and have a center-to-center spacing of 6.218 inches. The new racks will provide storage capacity until at least the year 1999 with a full core reserve.

This report contains a discussion of the nuclear, thermal-hydraulic and structural safety of the proposed installation, as well as a discussion of environmental and radiological considerations. The information has been prepared based on the guidance provided in "Operation Technical Position for Review and Acceptance of Spent Fuel Storage and Handling Applications," which was issued by the Nuclear Regulatory Commission (NRC) on April 14, 1978 and later amended on January 18, 1979.

The design of the spent fuel racks and the rack structural analysis were performed by Nuclear Energy Services, Inc. (NES) of Danbury, Connecticut and reviewed by Yankee Atomic Electric Company (YAEC) and VYNPC. The radiological, environmental, nuclear criticality, thermal-hydraulic, bulk pool heatup, and ambient pool temperature analyses were performed by YAEC. The Reactor Building structural analysis was performed by Bechtel of Ann Arbor, Michigan and reviewed by YAEC.

The racks will be fabricated by a subcontractor selected by NES and reviewed by YAEC and VYNPC. VYNPC will remove the existing spent fuel racks and install the new racks.

1.1 NEED FOR INCREASED SPENT FUEL STORAGE

1.1.1 Current Storage Capacity

The original design of the Vermont Yankee Nuclear Power Station provided storage for 600 spent fuel assemblies. This design allowed for the storage of the normal quantity of spent fuel assemblies from two refueling operations (184) and the concurrent storage of a full core discharge of fuel assemblies (368). It was assumed that spent fuel would be removed from Vermont Yankee on the basis that a fuel cycle would be in existence that would only require storage of spent fuel for a year or two prior to shipment to a reprocessing facility. Once it became clear that reprocessing would not be available, VYNPC took appropriate steps to provide additional on-site storage capability.

In 1976, VYNPC applied to the NRC for a license change to install new, freestanding spent fuel racks to increase the spent fuel storage capacity at the Vermont Yankee plant to 2,000 assemblies.[2] These new racks were intended to accommodate spent fuel until 1990, assuming full core discharge capability was maintained. The U.S. Department of Energy (USDOE) had indicated that the projected operational date for a geological waste repository under the National Waste Terminal Storage (NWTS) Program was 1985. Also, following President Carter's ban on reprocessing in 1977, the Administration recognized that storage problems at reactors existed, and that additional storage away from reactors would be necessary until a repository became operable. Consequently, in October 1977, the USDOE announced a program whereby spent nuclear fuel would be stored at Federal Away-From-Reactor (AFR) sites after 1983 until a repository was available.

In September 1977, Amendment No. 37 to the Vermont Yankee Operating License[3] was granted by the NRC allowing installation of new racks to accommodate 2,000 spent fuel assemblies. The currently installed capacity of the SFP is 1,680 assemblies.

The USDOE, however, experienced considerable delays in finalizing plans for a Federal waste repository, and the initial planned startup date for a repository gradually slipped. In addition, the AFR Program did not receive congressional authorization.

In December 1982, the Nuclear Waste Policy Act of 1982 (the Act) was passed by Congress.[4] The Act establishes a detailed program for radioactive waste disposal, and includes a schedule to achieve implementation of the program. The Act requires the USDOE to begin accepting and disposing of spent nuclear fuel from civilian power reactors by no later than January 31, 1998. The Act also sets up a schedule for the establishment of a monitored retrievable storage facility, as well as permanent waste repositories.

The Act includes provisions for the Federal government to provide up to 1,900 MT of AFR storage, but only for utilities determined by the NRC to be unable to reasonably provide adequate spent fuel storage capacity at the reactor site through the use of high-density storage racks, fuel rod compaction, construction of additional on-site spent fuel storage facilities or other means.

Due to the small amount (1,900 MT) of AFR storage to be available, and due to the severe restrictions likely to be placed on utilizing the AFR storage, this additional storage capacity is not likely to be available to Vermont Yankee.

1.1.2 Projected Spent Fuel Generation Rate

For the eleven cycles of operation to date at Vermont Yankee, a total of 1,312 fuel assemblies were discharged into the SFP. The average discharge

fuel burnup was less than 25,000 MWd/Mtu. On an annual basis, the fuel management plan discharges 92 assemblies per year to the SFP. Table 1-1 shows projected future inventories assuming this plan. Based on this plan, the current number of spent fuel assemblies stored in the SFP, and the current number of available spent fuel storage locations, it is estimated that full core discharge capability would likely be lost in 1987.

An alternate fuel management plan is also under consideration for future use at Vermont Yankee. The alternate plan involves increasing fuel enrichment to approximately 3.13 wt% (from the current 2.89 wt%), and increasing burnup to 31,000 MWd/Mtu which results in eighteen-month operating cycles (128 assemblies per cycle would be discharged). By increasing both enrichment and burnup, fewer assemblies will be discharged to the SFP. However, it is not expected that this plan could be implemented before mid-1987; therefore, the date of full core reserve discharge loss would not be affected. In addition, this plan would require a change to the Vermont Yankee Operating License to allow the use of fuel with >2.89 wt% enrichment.

1.1.3 Alternatives for Increasing Spent Fuel Storage

Section 131 of the Nuclear Waste Policy Act states that, "The persons owning and operating civilian nuclear power reactors have the primary responsibility for providing interim storage of spent nuclear fuel from such reactors, by maximizing, to the extent practical, the effective use of existing storage facilities at the site of each civilian nuclear power reactor, and by adding new on-site storage capacity in a timely manner where practical."[4]

Furthermore, Section 132 of the Act states that, "The Secretary (USDOE), the Commission (USNRC) and other authorized Federal officials shall each take such actions as such official considers necessary to encourage and expedite the effective use of available storage, at the site of each civilian nuclear power reactor..."[4]

In view of the above provisions of the Act, the following alternatives to increasing spent fuel storage capacity at Vermont Yankee were considered:

1. Shipment to another reactor site.

2. Modifying the plant fuel management plan to reduce spent fuel generation rate.

Shipment to another reactor site is not a viable alternative to increasing spent fuel storage at Vermont Yankee. The Vermont Yankee Nuclear Power Station is the only nuclear plant operated by VYNPC. The only other reactor plants in the Northeast region with compatible spent fuel storage racks currently operating are Boston Edison's Pilgrim 1 and Northeast Utility's Millstone 1. It is extremely unlikely that either these or any other utility would agree to accept Vermont Yankee spent fuel since it would only reduce their available capacity.

As was discussed in Section 1.1.2, the proposed fuel management plan at Vermont Yankee is to increase design fuel burnup beginning in 1988, thereby slightly decreasing the number of spent fuel assemblies discharged per year to the SFP. However, this plan would not alleviate the lack of storage capacity in the current racks on a timely basis.

The option of shipment of fuel to an AFR storage facility was considered but not pursued due to: 1) the unavailability of an AFR storage facility and 2) the provision of the Act that requires utilities to maximize on-site storage before gaining access to an AFR. Likewise, the provision of the Act setting a target date of 1998 for operation of a waste repository precludes any consideration of shipping Vermont Yankee spent fuel to a repository instead of increasing on-site storage.

In addition to the above alternatives to considering off-site storage of spent fuel, the following alternate methods for increasing on-site storage were examined:

1. Pin consolidation.

2. Independent dry (cask, concrete silo) storage.

3. Construction of an independent wet storage pool.

4. Independent air-cooled vault storage.

5. Reracking with high density storage racks.

With the exception of reracking, the above alternatives have not previously been fully licensed for commercial power plants by the NRC. Since additional spent fuel storage has to be in place at Vermont Yankee by 1987, it is not considered prudent to select a storage option that has not been previously licensed due to uncertainties in the ability to license such methods and uncertainties concerning the licensing schedule. In addition, the above options have, in general, not been demonstrated on other than a theoretical or prototype basis, further adding to the uncertainty concerning the schedule for design and construction. Also, the Act only requires that reactor licensees utilize previously licensed technologies for maximization of on-site storage.

In view of the above considerations and schedular constraints, increasing on-site storage capacity by replacing existing racks with a proven design to allow closer spacing of the fuel assemblies was concluded to be the only practical alternative.

1.2 GENERAL DESCRIPTION OF STORAGE MODIFICATION PLANS

This section provides a general description of the spent fuel storage modification plans for Vermont Yankee. A detailed description is contained in Chapters 3 and 5.

1.2.1 Additional Storage Capacity

The method to be used to increase spent fuel storage at Vermont Yankee will be to install new high-density storage racks containing a neutron absorbing material (Boral) in the existing SFP. Figures 1-1 and 1-2 show the location of the SFP.

The new racks will be freestanding and will be arranged as shown in Figure 1-3. A total of 2,870 storage spaces will be provided.

Figure 1-4 shows a typical spent fuel storage rack. The rack cells are constructed by forming Type 304L stainless steel sheets into square cells, seam-welded centrally along one side. The inside dimension of each cell is 5.922 inches. The cells are fastened together by Type 304L stainless steel tie plates welded intermittently along their full length. The cells in turn are welded to a stainless steel base plate which contains cooling flow orifices for each cell. The center-to-center spacing of the cells is 6.218 inches.

The construction of the storage cells provides four vented (open to the SFP) compartments in which neutron absorber elements are placed. The racks are designed to accommodate fresh fuel at an average bundle enrichment of up to 3.25 wt% with a $k_{eff} \leq 0.95$, including all uncertainties.

The rack support feet raise the racks above the SFP floor to the height required to provide an adequately sized cooling water supply plenum. The support feet contain remotely adjustable jackscrews (accessible from the top of the spent fuel rack) to facilitate and ensure proper load distribution and leveling of the racks.

The storage cell structure, acting with the rack base and the rack support feet, provides the structural strength and stiffness for the rack to meet all required loading combinations. No wall bracing or attachments are required to support the fuel racks under any design condition. Sufficient space is provided between adjacent spent fuel racks to preclude impact in the event that sliding or rocking occurs during a seismic event.

The new racks will be installed in conjunction with the removal of the existing spent fuel racks and the two existing 6-inch diameter Fuel Pool Cooling and Demineralizer System (FPCDS) discharge sparger lines from the SFP.

1.2.2 Projected Increase in Allowed Operation

Utilizing the projected spent fuel inventory in Table 1-1, it is estimated that the installation of additional storage capacity in the SFP will allow operation of the plant with a full core discharge capability until 1999. Full-core discharge capability is lost with the proposed new fuel racks when the number of spent fuel assemblies in storage exceeds 2,502 (2,870 minus 368 assemblies) following the assumed annual refueling. The Vermont Yankee Operating License, DPR-28, expires on December 11, 2007.

No credit is taken in Table 1-1 for spent fuel transfers to the USDOE, which are assumed to begin in 1998. In addition, the numbers of occupied spaces specified in Table 1-1 do not include items other than fuel assemblies, as these are not stored in spent fuel rack locations at Vermont Yankee.

VYNPC intends to maintain a full core discharge capability throughout the life of Vermont Yankee in order to ensure reliable plant operation.

TABLE 1-1

Projected Spent Fuel Storage Inventory

Following Refueling Year	Number of Assemblies Discharged*	
	Per Year	Total
1986 (current outage)	92	1,312
1987	92	1,404
1988	92	1,496
1989	92	1,588
1990	92	1,680
1991	92	1,772
1992	92	1,864
1993	92	1,956
1994	92	2,048
1995	92	2,140
1996	92	2,232
1997	92	2,324
1998	92	2,416
1999	92	2,508

*Assumes annual fuel cycles

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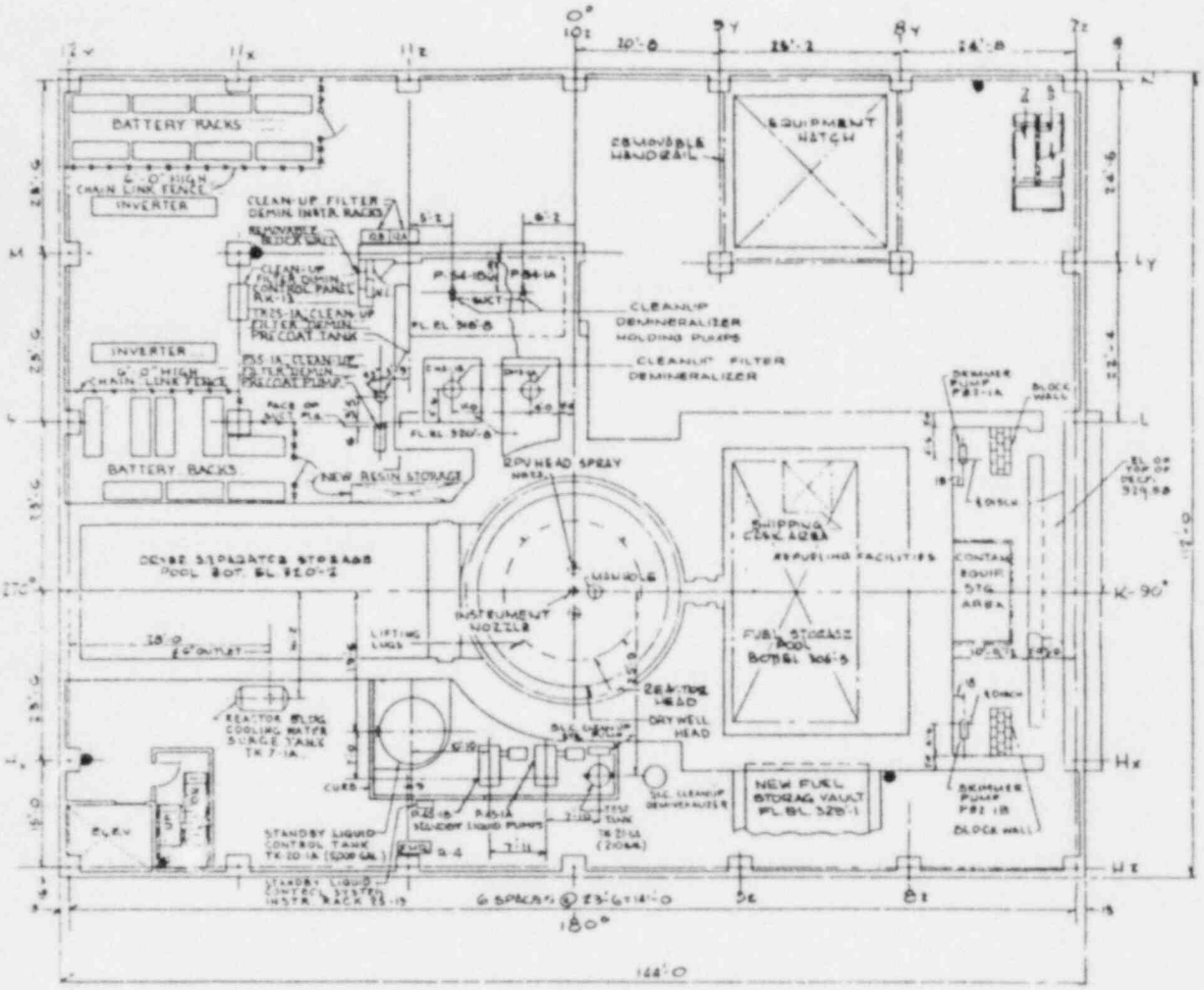
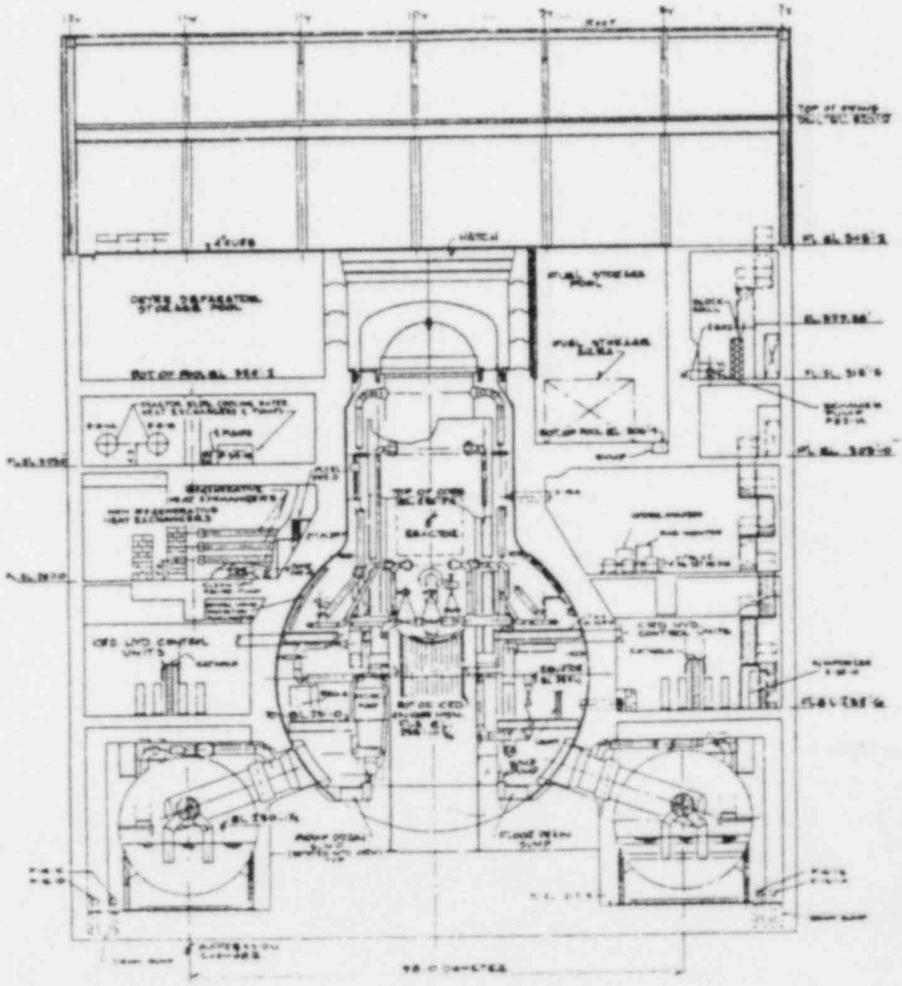


FLOOR EL. 318'-8

VERMONT YANKEE NUCLEAR POWER STATION
REACTOR BUILDING PLAN EL. 318'-8 FIGURE 1-1

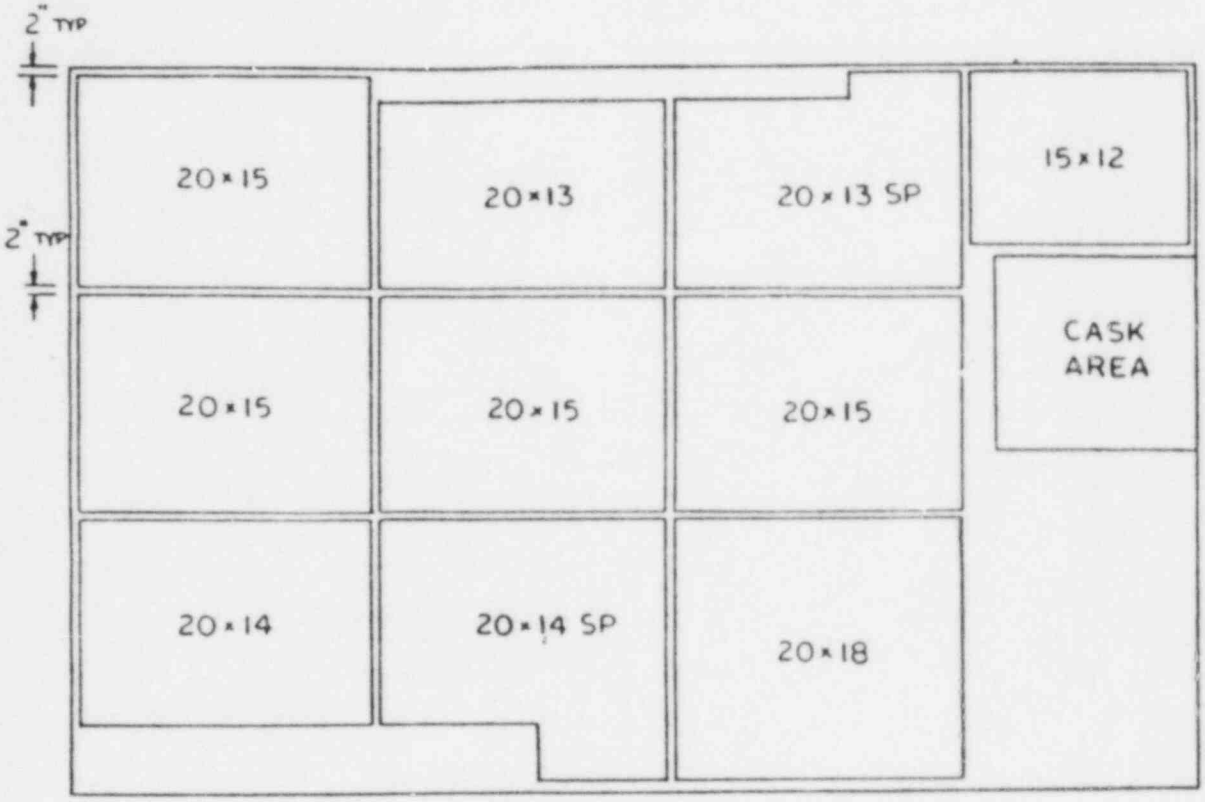
REV. 1



VERMONT YANKEE
NUCLEAR POWER STATION

REACTOR BUILDING
SECTION A-A

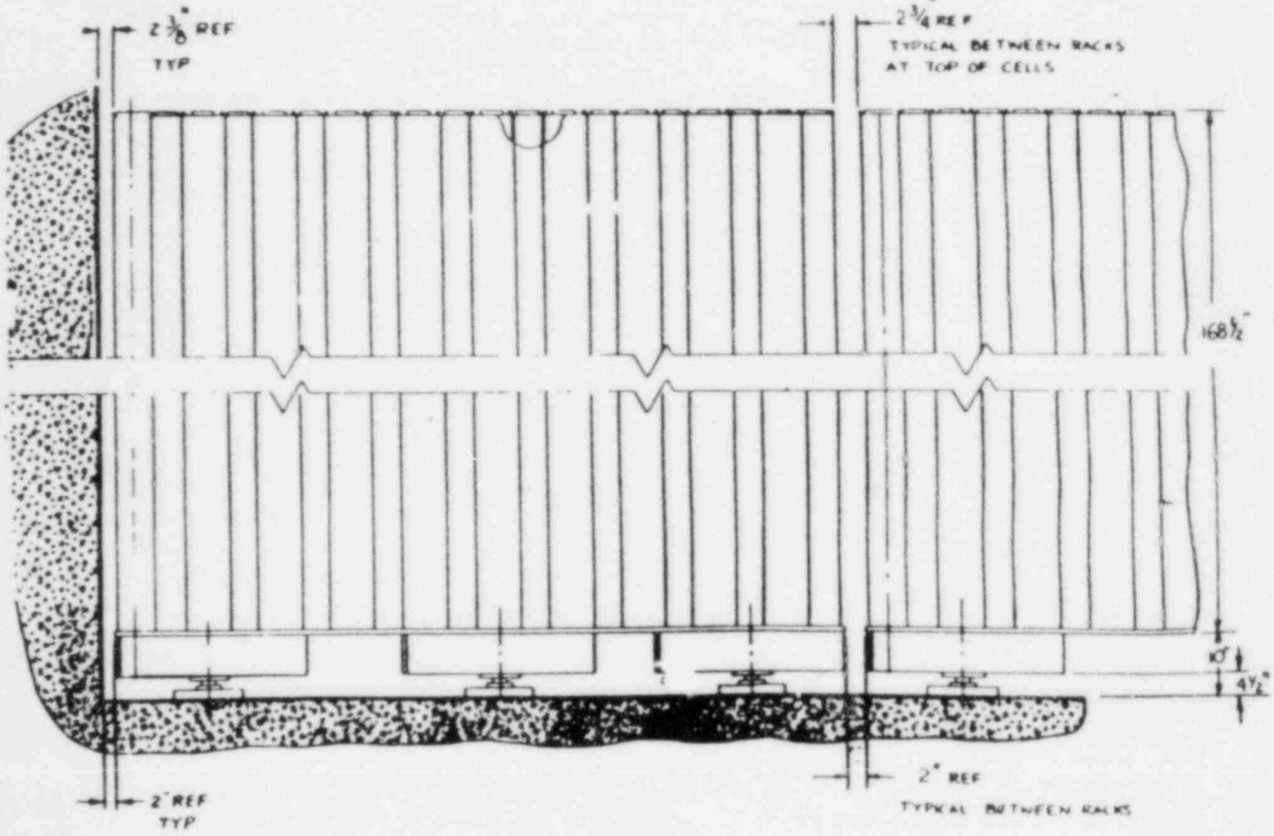
FIGURE 1-2



PLAN

FUEL STORAGE RACK ARRANGEMENT

FIGURE 1-3



SPENT FUEL STORAGE RACK SCHEMATIC

FIGURE 1-4

2.0 DESIGN CRITERIA

2.1 NUCLEAR DESIGN CRITERIA

2.1.1 Fuel Assembly Design Criteria

To assure that all NRC fuel storage criticality requirements are met, the high density storage racks shall be designed to accommodate any fuel bundle with a maximum infinite lattice reactivity less than or equal to 1.35 ($k_{inf} \leq 1.35$). This $k_{infinity}$ shall be based on the fuel bundle arranged in the reactor core geometry at 68°F in an uncontrolled state.

Accordingly, the YAEC design basis fuel assembly for spent fuel rack criticality analysis is an 8 x 8 array of fuel rods at a uniform fuel enrichment of 3.25 w/o U-235. No credit is taken for burnable poison or exposure. This results in a design bundle reactivity ($k_{inf} = 1.35$) which provides generous margin to any of our present or expected fuel designs.

2.1.2 Criticality Safety Analysis

The criticality safety analysis shall demonstrate that the storage rack design maintains $k_{eff} \leq 0.95$, including all uncertainties under all conditions in accordance with Reference 6.

All variations in rack dimensions, neutron absorber parameters, fuel parameters and fuel location permitted by fabrication tolerances shall be included in the analysis. Uncertainties and abnormalities under the above conditions shall be included in the analysis.

Calculational methods used in the criticality analysis shall conform to the guidance provided in References 6 and 7.

2.1.3 Neutron Absorber Material

The neutron absorber material to be used in the storage racks shall be previously licensed by the USNRC. The neutron absorber material shall have no significant loss of neutron absorption, creep-down, redistribution of neutron absorber material, accumulation of voids, swelling, cracking or powdering over a 40-year design life in the SFP.

2.2 THERMAL-HYDRAULIC DESIGN CRITERIA

The SFP and its auxiliary systems are designed to remove the decay heat from the spent fuel, to purify and maintain the system water inventory. There are several SFP thermal conditions, for both 18-month and 12-month refueling cycles, under which the SFP and its associated systems must perform their intended function. Three design conditions, however, bound all others, and these are described below:

Condition 1:

The SFP contains the maximum number of fuel assemblies (fuel pool full), including the 136 assemblies from one 18-month cycle discharge, 10 days after shutdown.

Condition 2:

The SFP contains the maximum number of fuel assemblies (fuel pool full), including a full core discharge (368 assemblies) with annual cycle reloads, 10 days after shutdown.

Condition 3:

Identical to Conditions 1 and 2 with the exception that all forced cooling is lost. Cooling occurs by SFP boiling (212°F) at the surface.

Design criteria for SFP, associated cooling and purification systems, and components can generally be classified into two categories: A) bulk SFP criteria and B) local or component criteria.

A. The thermal-hydraulic bulk SFP criteria are:

1. The Spent Fuel Pool Cooling and Demineralizer System (SFPCDS) shall be capable of maintaining the bulk SFP water temperature $<150^{\circ}F$ for Condition 1.

2. A single train of the RHR Cooling System shall be capable of maintaining the bulk SFP water temperature $<150^{\circ}F$ for Condition 2.

3. For Condition 3, adequate time shall be available to provide makeup water to the SFP should a loss of cooling water occur.

B. The following design criterion regarding local thermal-hydraulic conditions was developed to preclude a loss of structural integrity of either the spent fuel assembly or rack components.

1. For Conditions 1 and 2, the coolant temperature shall not exceed the local boiling point. The analysis shall consider the hottest fuel assembly located in the area of highest flow resistance to natural circulation flow.

A conservative peaking factor was applied to the decay heat generated by each assembly to evaluate local fluid conditions. The decay heat values were derived utilizing the ASB 9-2 standard[24]. The thermal analyses shall also consider the effects of manufacturing tolerances of the racks and uncertainties of fuel assembly positions in the rack.

2.3 MECHANICAL AND STRUCTURAL DESIGN CRITERIA

2.3.1 Mechanical Design Criteria

2.3.1.1 General Description

The racks shall be of proven design and shall be contained in the envelope of usable area shown in Figure 2-1. Minimum distance between racks or other obstacles shall have adequate clearance to eliminate contact under worst case loading conditions. All racks shall be designed to facilitate leveling at the time of installation in the pool. Adequate lifting provisions shall be provided to facilitate installation in the fuel pool and compliance with heavy load control requirements. Adequate provisions for cooling water flow shall be provided.

Each storage location shall have adequate clearance and lead-in capability to permit fuel assembly insertion and removal without damaging the fuel assembly (see Figure 2-2). Each storage location shall preclude insertion of a spent fuel assembly between the normal positions of assemblies in the rack.

2.3.1.2 Fuel Assembly

The new racks shall be designed to store boiling water reactor fuel assemblies as described in Section 3.3.1.4.

2.3.1.3 Control Rods

The racks shall be designed to store only boiling water reactor fuel assemblies. Control rods will be stored in the existing control rod storage racks shown in Figure 2-1.

2.3.1.4 Fuel Handling Equipment

The racks shall be designed to interface with the fuel handling equipment described in Sections 3.3.1.6 and 3.3.1.7.

2.3.2 Structural Design Criteria for New Fuel Racks

2.3.2.1 General Description

The structural design of the spent fuel racks shall be in accordance with NRC Standard Review Plan Section 3.8.4[9], Appendix D, "Technical Position on Spent Fuel Racks," and shall ensure that the spent fuel assemblies are maintained in a safe configuration for all postulated loading conditions. In addition, deflections experienced under any loading condition shall be limited to less than those which would prevent removal of spent fuel assemblies from the rack.

The racks shall be designed to be freestanding, with no connection of the racks to SFP walls or floor. Vertical loads shall be transferred by bearing, and horizontal loads shall be transferred by frictional resistance between the rack support feet and the SFP liner and concrete foundation slab.

2.3.2.2 Applicable Codes, Standards, and Specifications

Construction materials for the racks shall conform to the ASME or equivalent ASTM specifications as required by the Specification for High Density Spent Fuel Storage Racks[25]. All rack materials shall be compatible with the SFP environment. The acceptance criteria for rack design and analysis are based on the applicable sections of the NRC Position Paper on fuel storage racks[6], Standard Review Plan Section 3.8.4[9], and the ASME Code, Section III, Subsection NF[10].

2.3.2.3 Design Loads

Loads to be used in the design of the spent fuel racks are as follows:

D = Dead load of the racks and all attachments

L = Live load includes the weight of any combination of fuel assemblies placed in the most adverse configurations

Weight of fuel assembly plus channel = 670 lbs

T_o = Thermal loads due to temperature gradients and local heating during normal conditions

T_a = Thermal loads due to temperature gradients and local heating during accident conditions

E = Seismic loads due to the operating basis earthquake (OBE)

E' = Seismic loads due to the safe shutdown earthquake (SSE)

P_f = Load from crane uplift on a stuck fuel assembly; the uplift load shall be taken as 4000 lb

F_d = Effects due to hypothetical load drop accidents, or effects due to postulated missiles generated during a design basis tornado

OBE design response spectra based on Vermont Yankee FSAR Appendix A[11] and applicable to the SFP slab are shown in Figures 2-3 through 2-5. SSE accelerations are two times the OBE acceleration values. Seismic excitation in accordance with these spectra shall be applied along three orthogonal axes simultaneously. For dynamic analysis by the response spectrum technique, the structural responses shall be combined by the square root of the sum of the squares (SRSS) method in accordance with Regulatory Guide 1.92[12]. When the time-history method of dynamic analysis is utilized, statistically independent time-histories, compatible with the response spectra in each of the three orthogonal directions, shall be used.

Damping ratios based on Vermont Yankee FSAR Section 12.2.1.2[11] are given in Table 2-1. No increase in damping ratio due to submergence effects is permitted unless documentation demonstrating the basis for acceptability is provided. Other hydrodynamic effects, such as trapped water mass and fluid coupling, shall be considered in the analysis.

Loads generated during a seismic event by the impact of the fuel assemblies on the walls of the rack cells shall be considered both in terms of local as well as overall effects on the rack structure.

Postulated missiles generated during a design basis tornado should be addressed. Impact from dropping a fuel assembly a distance of 12 inches from the top of the rack shall be investigated.

2.3.2.4 Load Combinations and Acceptance Criteria

The spent fuel racks shall be designed in accordance with the following load combinations and acceptance criteria:

Load Combination, Linear Elastic Analysis	Acceptance Limit[a]
$D + L$	Level A service limits
$D + L + T_o$	Level A service limits
$D + L + T_o + E$	Level A service limits
$D + L + T_a + E$	Level B service limits
$D + L + T_o + P_f$	Level B service limits
$D + L + T_a + E'$	Level D service limits
$D + L + F_d$	The functional capability of the fuel racks shall be demonstrated (k_{eff} <0.95)
Load Combination, Plastic (Limit) Analysis	
$1.7\ (D + L)$	ASME Code Section III, Appendix XVII, Paragraph 4000 for all load combinations
$1.3\ (D + L + T_o)$	
$1.7\ (D + L + E)$	
$1.3\ (D + L + E + T_o)$	
$1.3\ (D + L + E + T_a)$	
$1.3\ (D + L + T_a + P_f)$	
$1.1\ (D + L + T_a + E')$	

(a) Acceptance limits are in accordance with ASME Section III, Division 1, Subsection NF for Class 3 component supports.

For load combinations which include fuel assembly impact on the rack cells, it shall be demonstrated that rupture of fuel pin cladding is precluded.

If structural behavior is inelastic, ductility ratios shall be quantified and the basis for acceptability provided. Both local and overall structural ductility demands shall be addressed.

Factors of safety against overturning and sliding shall be shown to be acceptable, or effects of geometrical constraint and basis of acceptability shall be demonstrated.

For conditions which will exist during installation of new spent fuel racks and removal of existing racks, hypothetical accidents shall be considered, such as dropping a rack on another rack containing spent fuel, dropping a rack on the SFP liner and floor slab, or instability during a seismic event of new and existing rack configurations where spent fuel damage is possible. The basis upon which such accidents are considered precluded, or the basis upon which the consequences of such accidents are considered acceptable, shall be provided.

2.3.3 Structural Design Criteria for Existing Structure

The existing SFP structure shall be shown to remain adequate to resist all loading conditions and effects imposed by the new racks. Such loads and effects shall include new rack dead and live loads, loads transmitted during seismic events, and thermal effects due to bulk SFP heating.

2.3.3.1 Codes and Standards

A. ACI 349-80, Code Requirements for Nuclear-Safety Related Concrete Structures and Commentary

B. NRC Regulatory Guides and Documents

1. NUREG-0800, Standard Review Plan
   - Section 3.8.4 and Appendices, Revision 1, July 1981
   - Section 9.1.2, Revision 3, July 1981

2. NRC's approval of Vermont Yankee's Spent Fuel Storage and Handling License Amendment request, dated September 15, 1977 (Amendment No. 37).

3. 1.92 - Combining Modal Responses and Spatial Components in Seismic Response Analysis

C. ANS-57.2, Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at NPS

D. ACI ASME Section III, Division 2, Code for Concrete Reactor Vessels and Containments, Subsection CC, 1977 Edition, dated July 1, 1977

E. ASME Section III, Division 1, Appendix I, Design Stress Intensity Values, Allowable Stresses, Material Properties, and Design Fatigue Curves (July 1980)

2.3.3.2 Material Properties

The following symbols are used to describe the material properties used in structural analysis:

f'_c = Compressive strength of concrete

γ = Specific weight

μ = Poisson's ratio

α = Coefficient of linear expansion

E = Modulus of elasticity

f_y = Yield stress

A. Concrete

f'_c = 4000.0 psi*

γ = 150.0 pcf

μ = $\sqrt{f'_c}/350$ = 0.181

E = 3.834×10^3 ksi*

α = 5.5×10^{-6} per ^oF

B. Reinforcing Steel

γ = 490 pcf

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* For the fuel pool floor the in place concrete test data shown in Table 2-2 may be used.

f_y = 40.0 ksi

μ = 0.3

E = 29×10^3 ksi

α = 6.5×10^{-6} per ^oF

C. Water

γ = 62.5 pcf

D. Grout

γ = 138 pcf

E. Stainless steel liner plates are conforming to ASTM A240, Type 304. The following material properties are independent of temperature values:

γ = 490 psf

μ = 0.3

E = 28×10^6 psi

α = 9.4×10^{-6} per ^oF

However, the stainless steel yield stress varies for various temperatures and shall be interpolated from the following data (Table I-2.2 of ASME Section III, Division 1, Appendix I)[10]:

Temperature (^oF)	f_y (ksi)
100	30
200	25
300	22.5

F. Structural angles and wide flange sections used as stiffeners for pool liner plates are conforming to ASTM A36.

γ = 490 psf

μ = 0.3

E = 29×10^6 psi

f_y = 36 ksi

2.3.3.3 Design Loads

A. Dead Load (D)

Dead loads consist of the dead weight of concrete, grout, steel liner, fuel racks, fuel, and any equipment permanently attached to the pool.

Hydrostatic pressure loads acting on walls and floor shall be included in this category. It should be noted that the hydrostatic load acting on the north wall of the pool is dependent on the operation condition.

Dead loads distributed from the contiguous floors and beams to the pool walls shall be considered in the analysis. An 80 psf intensity on all floors connected to the pool shall be used to account for piping weights as documented in Reference 13.

B. Live Load (L)

Live loads are random, temporary load conditions during maintenance and operation. They shall include the following loads:

o Weight of fuel cask

o Weight of control rod storage racks and stored control rods

o Weight of refueling bridge and service platform

Live loads distributed from the contiguous floors and beams shall be calculated based on the following load densities:

o El. 345.17' 500.0 psf

o El. 318.67' 200.0 psf

o El. 303.00' 200.0 psf

Values of load densities are obtained from Reference 13.

C. Normal Operating Thermal Load (T_o)

These thermal loads are generated under normal operating or shutdown conditions. The following temperature data shall be used in the calculation of thermal gradients through walls and floor:

o Inside drywell temperature $135^{o}F$

o Pool water temperature $150^{o}F$

o Room temperature $60^{o}F$

o Outside ambient temperature:

Summer $100^{o}F$

Winter $0^{o}F$

D. Accident Thermal Loads (T_a)

These thermal loads are due to the thermal conditions generated by the postulated accidents.

The thermal accident temperature for the spent fuel pool is 212°F throughout the whole pool.

E. Seismic Loads (E, E')

E = Seismic loads generated by Operating Basis Earthquake (OBE)

E' = Seismic loads generated by Safe Shutdown Earthquake (SSE)

The SSE seismic loads are assumed to be twice that of the OBE seismic loads. The maximum accelerations for OBE are given in Figures 2-6 through 2-8. The vertical seismic effects shall be considered either upward or downward to result in the worst loading in the load combinations.

The hydrodynamic loads of pool water acting on pool walls shall be calculated in accordance with Chapter 6 of Reference (14).

The vertical seismic loads due to the refueling bridge and service platform and the vertical seismic loads distributed from the contiguous floors and beams shall be included in the analysis.

The seismic loads due to equipment mounted on pool walls shall be calculated using 150 percent of the peak floor response spectra.

The effects of three components of earthquake shall be combined by the SRSS method. Amplified response spectra provided by YAEC shall be used to determine the appropriate acceleration values for computing seismic loads.

The effect of rack impact on the pool floor during a seismic event should also be considered in the analysis. These loads should be combined with the loads due to vertical seismic floor acceleration.

2.3.3.4 Load Combinations

The following load combinations shall be considered in the spent fuel pool analysis (Paragraph 3.0, Section 3.8.4 of NUREG-0800)[9]:

A. Concrete Pool Structure

1. Service Load Conditions

a. $1.4D + 1.7L$

b. $1.4D + 1.7L + 1.9E$

c. $(0.75)\ (1.4D + 1.7L + 1.7T_o)$

d. $(0.75)\ (1.4D + 1.7L + 1.9E + 1.7T_o)$

e. $1.2D + 1.9E$

2. Factored Load Conditions

a. $D + L + T_o + E'$

b. $D + L + T_a$

c. $D + L + T_a + 1.25E'$

Where any load reduces the effects of other loads, the corresponding coefficients for that load should be taken as 0.9 if it can be demonstrated that the load is always present or occurs simultaneously with other loads. Otherwise, the coefficient for that load should be taken as zero.

B. Liner and Liner Anchors

1. Service Load Conditions

a. $D + E + T_o$

2. Factored Load Conditions

a. $D + E' + T_a$

2.3.3.5 Structural Acceptance Criteria

The stresses and strains resulting from the load combinations described in Section 2.3.3.4 shall satisfy the following acceptance criteria:

A. Reinforced Concrete Pool Structures

The design stress limits described in ACI 349-80[31] and 80R will be used for the evaluation of the spent fuel pool. The capacity of all sections will be computed based on the ultimate design strength.

B. Liner and Liner Anchors

The acceptance criteria for the liner and liner anchors will be in accordance with the requirement specified in Paragraph CC-3720 and CC-3730 of ACI ASME Section III, Division 2, Subsection CC[38].

2.4 RADIOLOGICAL DESIGN CRITERIA

Radiological design criteria ensure that the radiological consequences of increasing spent fuel storage at the Vermont Yankee Nuclear Power Station do not exceed plant operational limits or NRC regulatory limits.

2.4.1 Dose Rates

The dose rates at the surface of the SFP from spent fuel assemblies, as well as dose rates at the outside surface of the walls of the SFP, shall not exceed the maximum radiation zone levels for those areas specified in Vermont Yankee FSAR Section 12.3.4.

2.4.2 Fuel Handling Accident

A fuel handling accident involving the dropping of a single spent fuel assembly in the SFP from its maximum attainable height shall not result in off-site radiation doses to the public exceeding the values calculated in Section 14.6.4 of the Vermont Yankee FSAR.

2.4.3 Radiological Effluents

Radiological effluents from the SFP shall not result in radiological effluents exceeding Plant Radiological Technical Specification limits.

2.5 QUALITY ASSURANCE

All design, fabrication and related items shall comply with the requirements of the Title 10, Code of Federal Regulations, Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Processing Plants."

VYNPC, YAEC and NES shall establish and maintain effective quality assurance programs which comply with this code. NES's quality assurance program shall also conform to the general requirements and guidance of ANSI N45.2-1977, "Quality Assurance Program Requirements for Nuclear Facilities," (Regulatory Guide 1.28, Revision 2). The program shall also comply with the following ANSI N45.2 "daughter" standards:

ANSI N45.2.2-1978	-	"Quality Assurance Requirements for Packaging, Shipping, Receiving, Staging and Handling of Items for Water-Cooled Nuclear Power Plants" (Regulatory Guide 1.38, Revision 2).
ANSI N45.2.6-1978	-	"Qualifications of Inspection, Examination and Testing Personnel for Nuclear Power Plants," (Regulatory Guide 1.58, Revision 1).
ANSI N45.2.11-1974	-	"Quality Assurance Requirements for the Design of Nuclear Power Plants," (Regulatory Guide 1.64, Revision 2).
ANSI N45.2.13-1976	-	"Quality Assurance Requirements for Control of Procurement of Equipment Materials and Services for Nuclear Power Plants," (Regulatory Guide 1.123, Revision 1).

VYNPC/YAEC's Quality Assurance Program shall conform to the requirements of the YAEC Operational Quality Assurance Manual (YOQAP-1-A)[15].

The YOQAP-1-A follows the requirements and guidelines of ANSI N18.7, "Administrative Controls and Quality Assurance for the Operations Phase of Nuclear Power Plants."

All design, procurement, fabrication and testing shall be subject to surveillance, inspection or audit by VYNPC or YAEC to assure conformance with the procurement documents, NES' Quality Assurance Program or NES' VYNPC/YAEC-approved design documents and procedures.

TABLE 2-1

FSAR Damping Values[11]

Type of Structure	Percent of Critical Damping
Reinforced Concrete Structure	5.0
Steel Frame Structure	2.0
Welded Assembly (Equipment and Supports)	1.0

TABLE 2-2

In-Place Concrete Test Data

(Bottom of Fuel Pool Floor)

Soils Engineering Inc.

P.O. Box 884
Charlestown, New Hampshire 03603
John F. Kennedy, P.E.
603/826-5873

Client: Vermont Yankee	Address: Vernon, Vermont		Phone No.
Project Name: Floor Reactor BLDG	Project Location: Vernon, Vermont		Job Phone No.
Job Number: 2587-85	Date: 12-6-85	Test Date: 12-6-85	By: John F. Kennedy, P.E.

Remarks:

Approximate Locations of Probes

AMERICAN CONCRETE INSTITUTE 214-65 STANDARDS OF CONCRETE CONTROL TABLE 2				
Class of operation	Coefficients of variation for standard control concrete			
	Excellent	Good	Fair	Poor
General Construction	Below 10.0%	10.0 - 15.0%	15.0 - 20.0%	Above 20.0%

Test Area	Type	Agg.	Moh's No.	Cure Days	Probe & Power Load Certification Number	Ht. of Probe Gage (Av. - 3 Probes)	Comp. Strength (psi.)
1	PRS-01	Gravel	6.4	YR 1968 +-	287193	A=2.25	7,210
2	"	"	"	"	287189	A=2.225	7,010
3	"	"	"	"	287195	A=2.175	6,590
4	"	"	"	"	287177	A=2.225	7,010
						Avg. Comp. Strength	6,955 psi
LOCATION:		SEE SKETCH ABOVE					
		Coefficient of Variation = 2.9%					

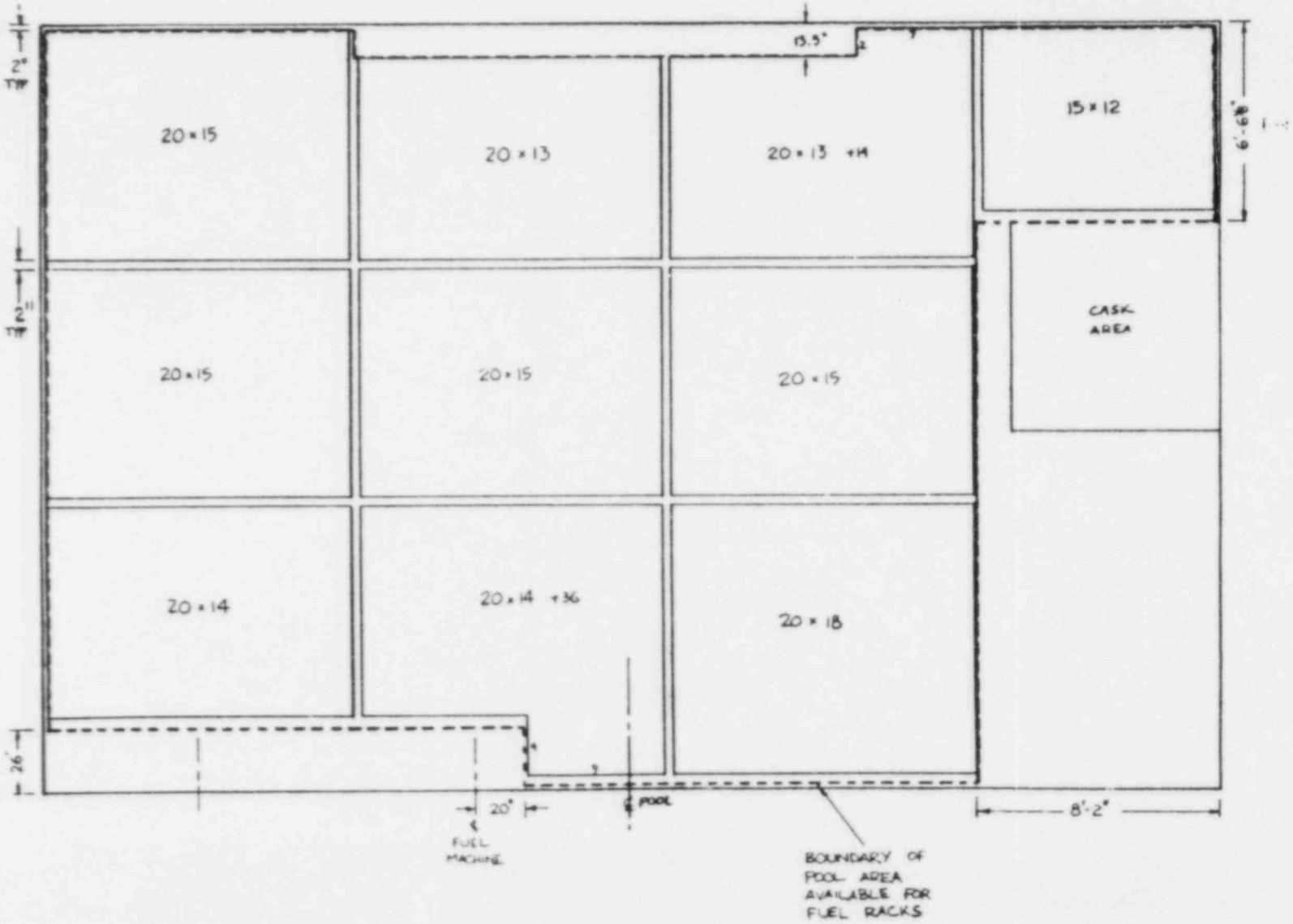
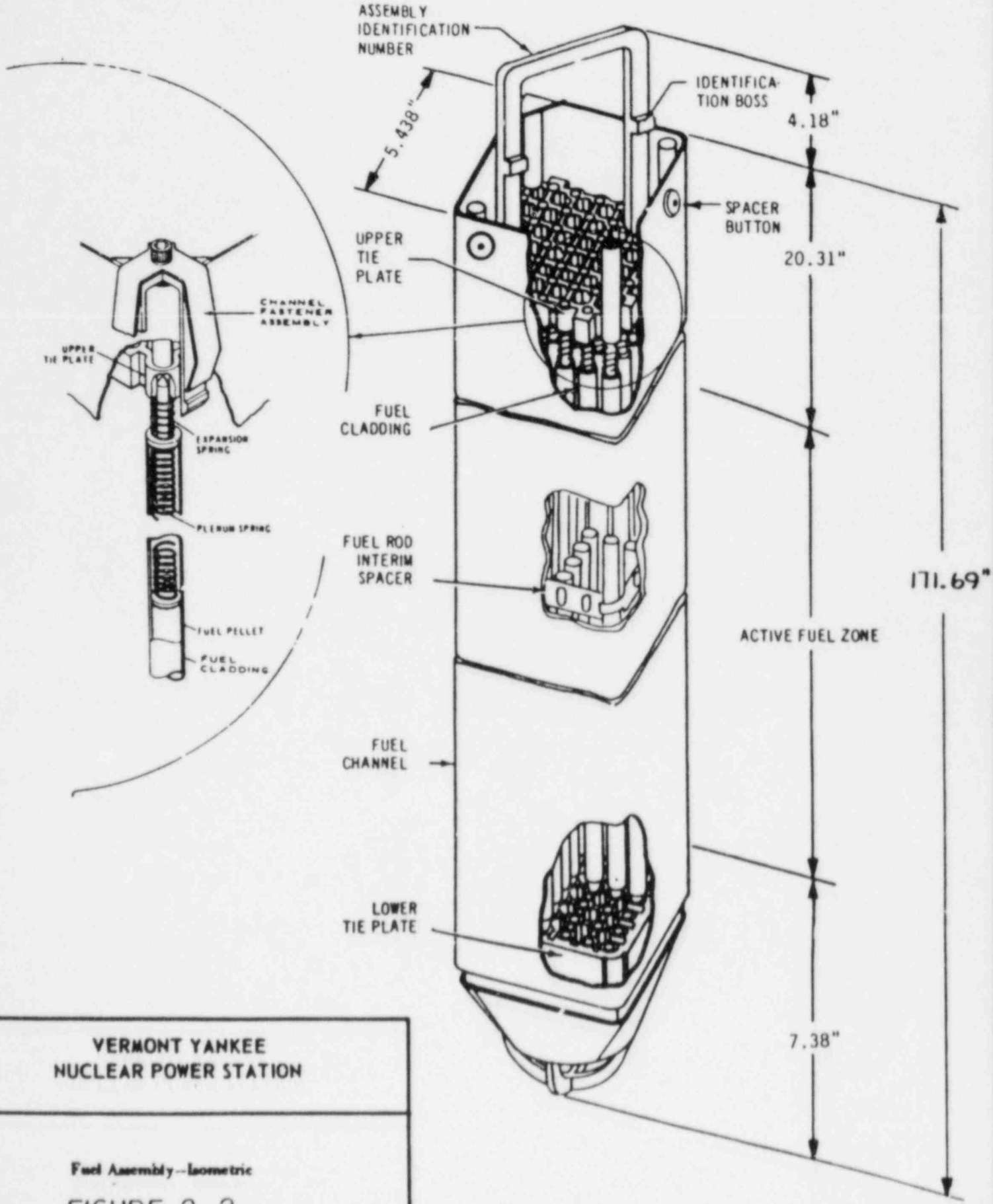


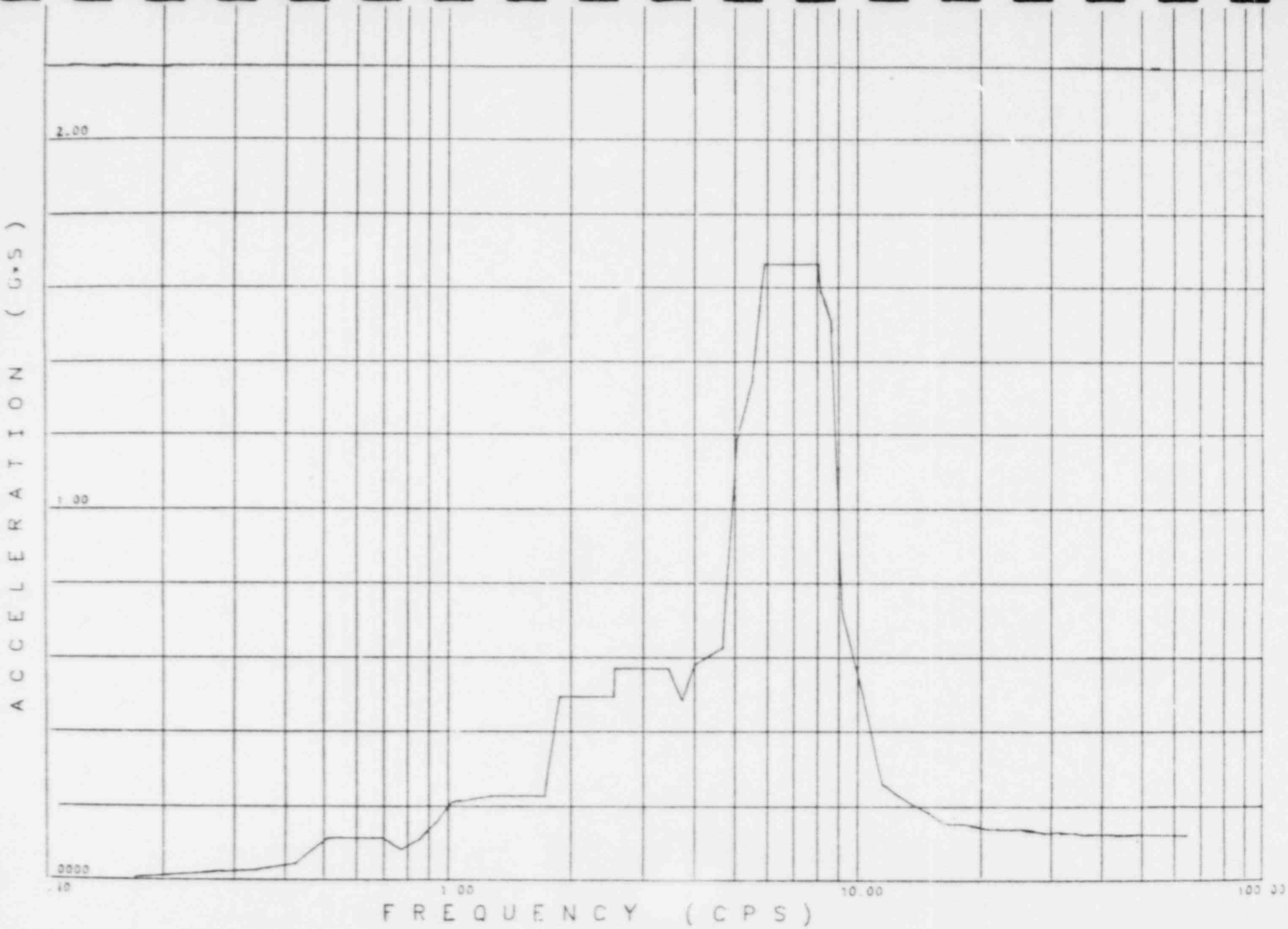
FIGURE 2-1 ENVELOPE OF USABLE AREA



VERMONT YANKEE NUCLEAR POWER STATION

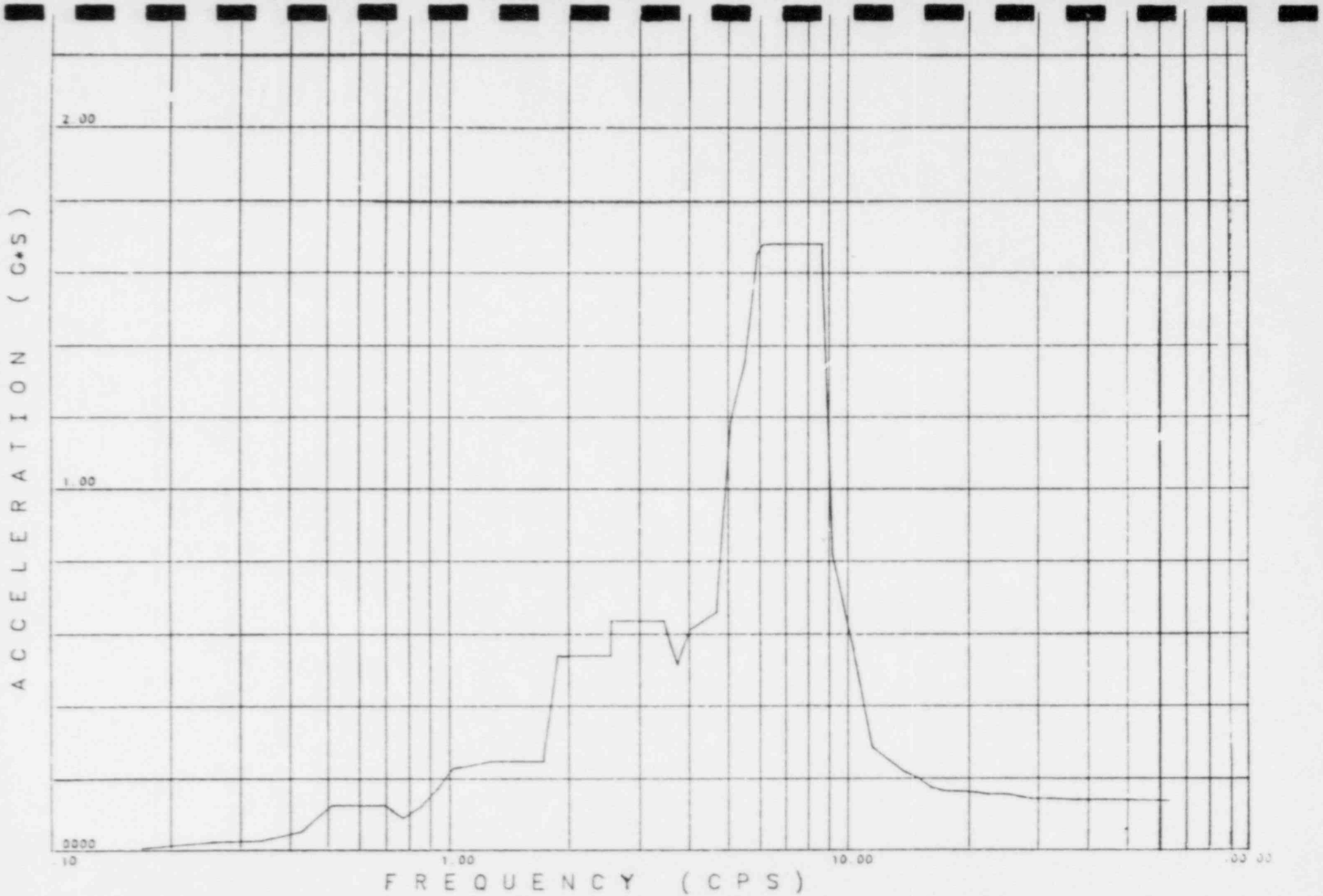
Fuel Assembly--Isometric

FIGURE 2-2



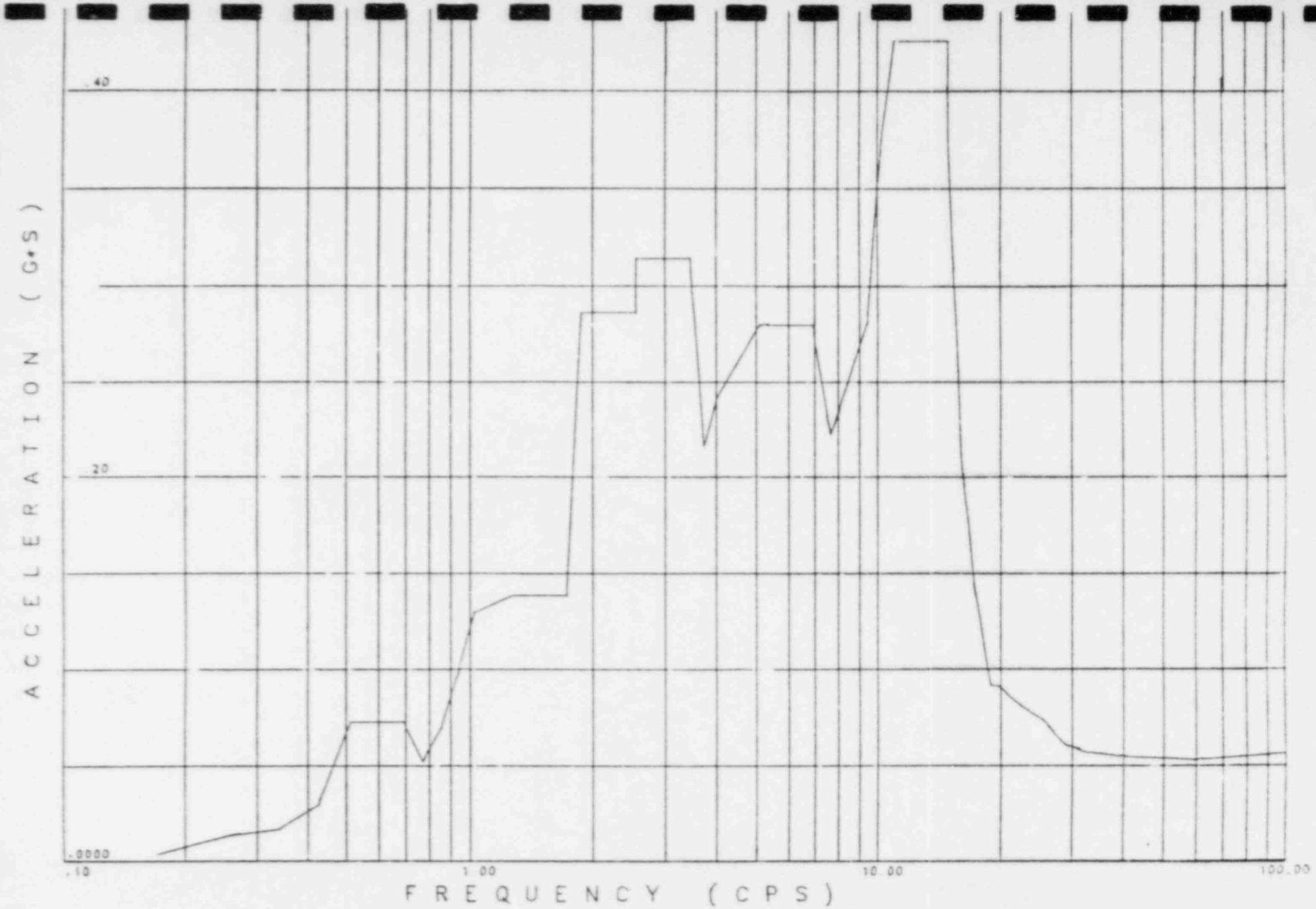
ENVELOPE - Y.A.E.C. REACTOR BLDG ARS (N-S) TAFT, OBE , EL. 303.0' D=1.0%

FIGURE 2-3



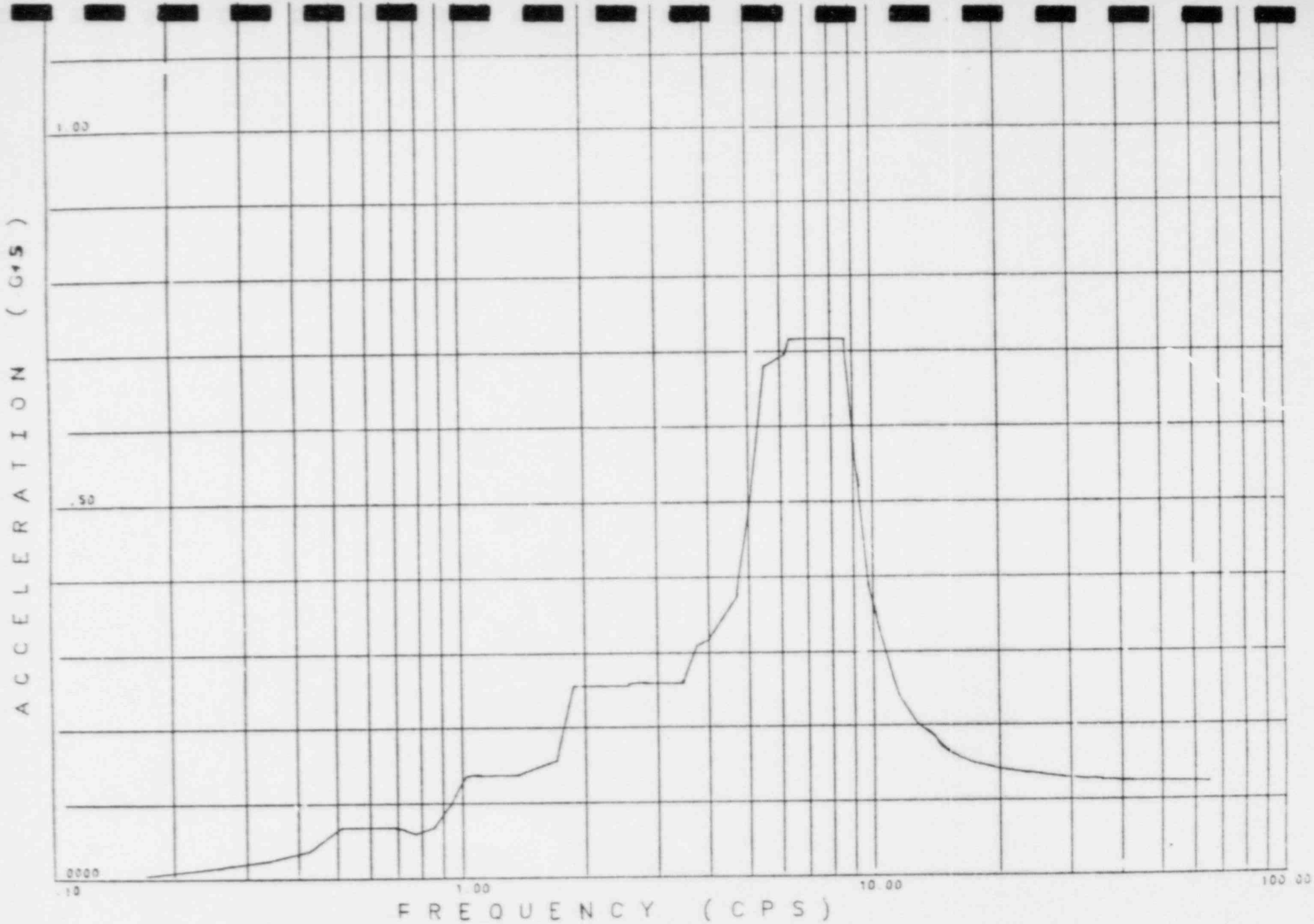
ENVELOPE - Y.A.E.C. REACTOR BLDG. ARS (E-W) TAFT, OBE, EL. 303.0' D=1.0%

FIGURE 2-4



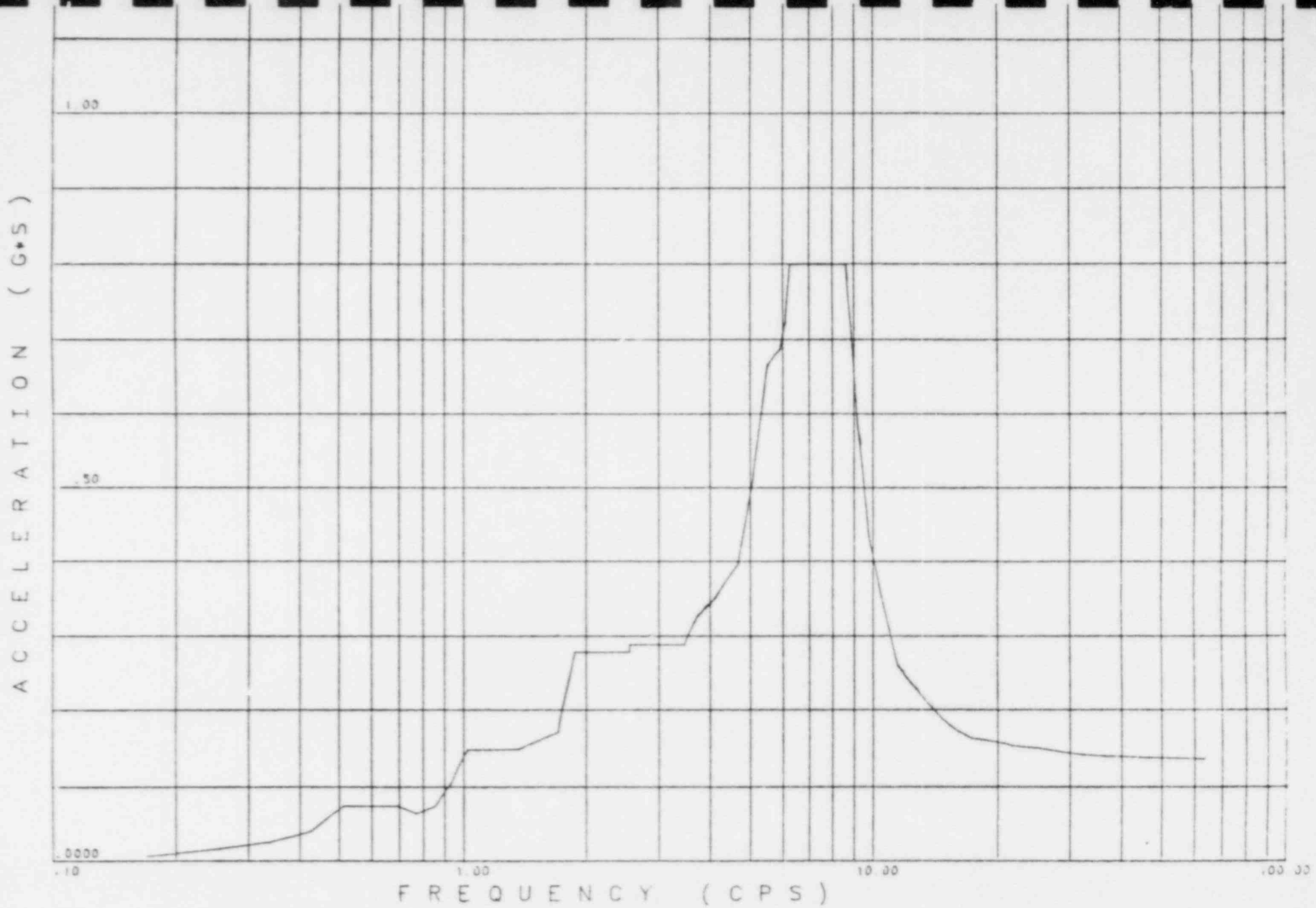
ENVELOPE - Y.A.E.C. REACTOR BLDG. ARS (VT.) TAFT, OBE, EL. 303.0' D=1.0%

FIGURE 2-5



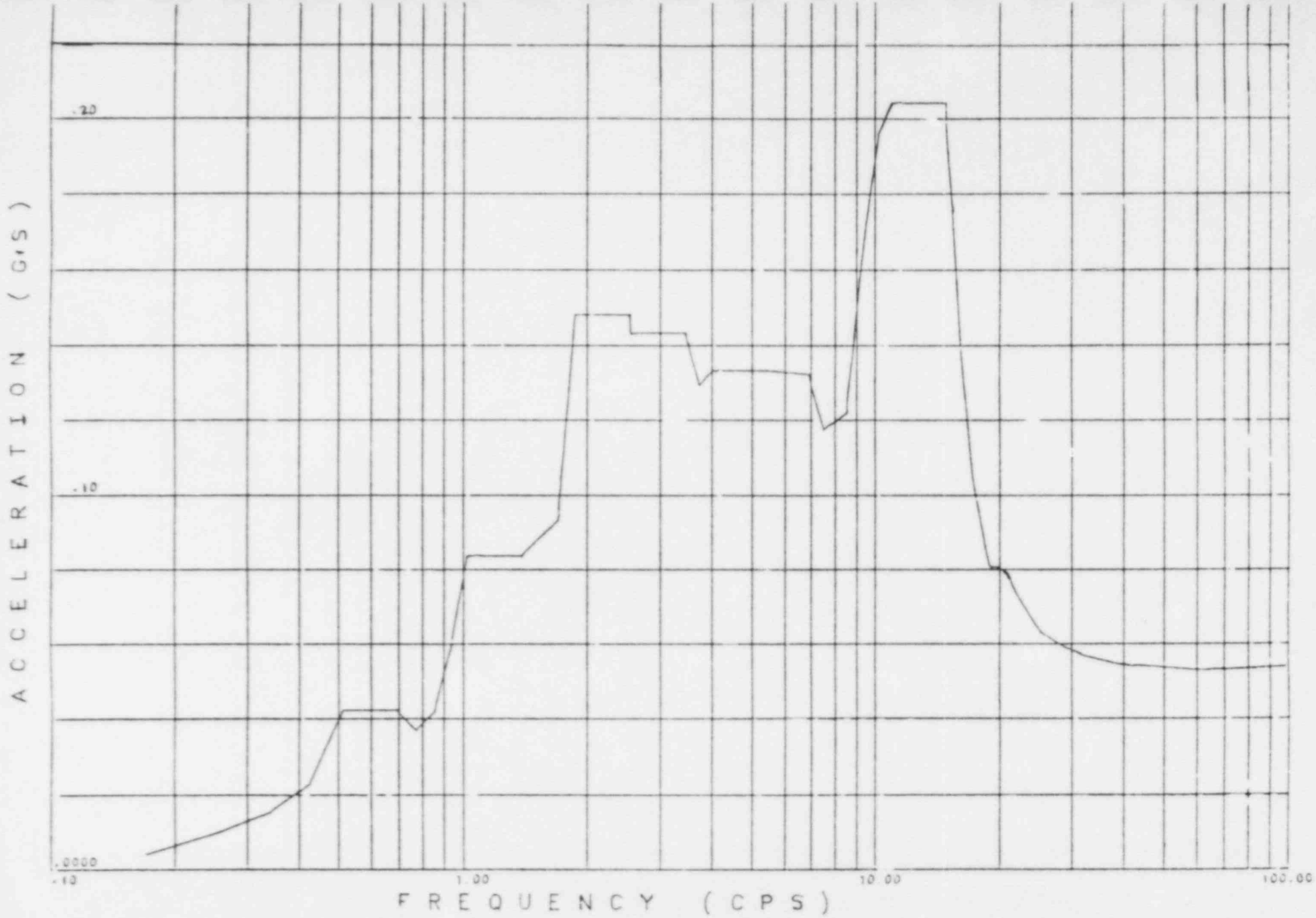
ENVELOPE - Y.A.E.C. REACTOR BLDG. ARS (N-S) TAFT. OBE, EL. 303.0' D=5.0%

FIGURE 2-6



ENVELOPE - Y.A.E.C. REACTOR BLDG. ARS (E-W) TAFT, OBE, EL. 303.0' D=5.0%

FIGURE 2-7



ENVELOPE - Y.A.E.C. REACTOR BLDG. ARS (VT.) TAFT, OBE, EL. 303.0' D=5.0%

FIGURE 2-8

3.0 SAFETY EVALUATION

3.1 NUCLEAR ANALYSIS

3.1.1 Criticality Model

The criticality analysis was performed by Yankee Atomic Electric Company (YAEC) using the analytical methods discussed in Section 3.1.2. The new spent fuel storage racks are described in Section 3.3.1.3. YAEC verified the design adequacy of the racks by calculating the k_{eff} for the reference configuration under normal and abnormal spent fuel pool (SFP) conditions. This analysis demonstrates that for all conditions, the k_{eff} of the racks is less than the criticality criterion of 0.95.

3.1.1.1 Calculational Assumptions

The following conservative assumptions have been used in the criticality calculations performed to verify the adequacy of the rack design:

1. The fuel is unirradiated and has a uniform average enrichment of 3.25 w/o.

2. The two water rods were assumed to be filled with fuel for a total of 64 fuel rods.

3. All burnable poison in the fuel assemblies is ignored.

4. The reference configuration conservatively contains an infinite square array of storage locations spaced 6.218 inches on center.

5. The neutron absorption of minor structural members is ignored.

6. Axial buckling is ignored.

7. No soluble boron exists in the pool.

3.1.1.2 Normal Conditions

Normal conditions include variations such as "worst case" eccentrically positioned fuel assemblies, SFP water temperature variations, dimensional variations permitted by fabrication tolerances, fuel enrichment variations and variations in the boral neutron absorber.

3.1.1.2.1 Reference Configuration

The reference configuration consists of an infinite array of storage cells having the nominal dimensions shown in Table 3-1. Each cell contains a fresh 3.25 w/o, 8 x 8 fuel assembly centrally located within the storage cell. The water temperature within the rack is 68°F.

3.1.1.2.2 Eccentric Configuration

It is possible for a fuel assembly to not be positioned centrally within a storage cell or resultant cell because of the clearance allowed between the assembly and the cell wall.

Calculations have been performed to determine the effects of eccentrically located fuel. In these calculations it was assumed that in an array of nine fuel assemblies, the center fuel assembly was displaced within a resultant cell as far as possible. This results in a decrease in k_{eff}.

3.1.1.2.3 Fuel Assembly Tolerance

The major fuel assembly parameter determining k_{eff} is the ratio of the amount of U-235 to that of water. The amount of U-235 per assembly is controlled to within a few tenths of a percent by weighing pellet stacks as the fuel is built and by using a known enrichment.

This analysis assumes a conservatively high average fuel enrichment of 3.25 w/o and no burnable poison, which produces a k_{inf} considerably higher than that of any fuel currently available or envisioned. Thus, changes in enrichment need not be considered as normal variations.

The fuel assembly parameters that determine the volume of water in an assembly are the clad outside diameter and the fuel rod pitch. These parameters are closely controlled to typical tolerances of ±0.4%. The effects of these fuel assembly tolerances on k_{eff} have been determined to be negligible on the basis of simple k_{inf} cell calculations. Consequently, fuel assembly tolerances were not considered further in this analysis.

3.1.1.2.4 Fuel Design Variation

Calculations were performed to determine the sensitivity of k_{eff} to variations of fuel enrichment from the base enrichment of 3.25 w/o. The criticality configuration used for these calculations was that of the reference configuration, with the exception of fuel enrichment which was varied from 2.5 to 4.0 w/o. These calculations do not affect the final k_{eff} for the fuel rack because, as mentioned above, the reference enrichment is higher than actual. These calculations were performed as a matter of interest only, to investigate the correlation between design assembly k_{inf} and spent fuel rack k_{eff}.

3.1.1.2.5 Fuel Rack Pitch Variation

Calculations were performed to determine the sensitivity of k_{eff} to changes in pitch, i.e., the center-to-center spacing between storage and resultant cells. Variations in pitch are limited by rack fabrication tolerances. The calculational impact of pitch variations on k_{eff} is provided in Section 3.1.3.2.

3.1.1.2.6 Cell Wall Thickness Variation

Calculations were performed to determine the sensitivity of k_{eff} to changes in the cell wall thickness. The reference configuration cell wall thickness was changed to represent variations in canister dimensions as supplied by NES. In a similar manner, the neutron absorber retaining envelope thickness was varied to determine the effect on k_{eff}. The calculational impacts of thickness variations are provided in Section 3.1.3.2.

3.1.1.2.7 Neutron Absorber Boron Concentration Variation

The manufacturer of the neutron absorbing material (Boral) provides documentation with the material which indicates that a 95% probability exists at the 95% confidence level that a minimum boron loading of 0.027 grams B^{10}/cm^2 is present[17]. This 95/95 confidence level is met for the minimum thickness allowable by manufacturing tolerances. Therefore, no changes in the boron concentration or thickness of the neutron absorber were considered in this analysis.

3.1.1.2.8 Neutron Absorber Width Variation

The effect on k_{eff} due to changes in neutron absorber width was determined. The neutron absorber width was varied according to the maximum allowable tolerances of the material as supplied by the manufacturer. The calculational impact of neutron absorber width variations is provided in Section 3.1.3.2.

3.1.1.2.9 Water Temperature Variations

The SFP water temperature was varied within the range of 68°F to 260°F to determine the effect of this parameter on k_{eff}. Results are provided in Section 3.1.1.3.3.

3.1.1.2.10 Fuel Stored in Zircaloy Channels

The storage of fuel within the Zircaloy channels is considered a normal condition and its effect on k_{eff} was calculated. The calculational impact of the Zircaloy channels is provided in Section 3.1.3.2.

3.1.1.2.11 "Worst Case" Normal Configuration

The "worst case" normal configuration considers the statistically combined adverse effects of eccentric fuel positioning, fuel assembly tolerance and design variations, cell wall thickness tolerances, poison

concentration and dimensional tolerances, storage and resultant cell inside dimension tolerances (i.e., pitch) and the presence of Zircaloy shrouds. Calculational uncertainties (see Section 3.1.1.2) on k_{eff} are then added directly to yield the final worst case normal k_{eff} as given in Section 3.1.3.2.

3.1.1.3 Abnormal Conditions

The abnormal conditions analyzed considered the effects of a variation of SFP temperature due to a loss of normal cooling with water temperature up to 260°F (212°F at the SFP surface), inadvertent placement of a fuel assembly adjacent to a loaded fuel rack, a single fuel assembly lying horizontally across the top of loaded racks, tornadoes and earthquakes (see Section 3.3.3) and load drop accidents (including the effects of impacted fuel in a storage cell due to a dropped fuel assembly).

3.1.1.3.1 Single Storage Cell Displacement

It is not possible to separate one storage cell from another because they are continuous (see Figure 1-4); thus, single storage cell displacement is not possible and will not be considered.

3.1.1.3.2 Fuel Handling Error

Accidental placement of fuel adjacent to an unpoisoned location at the fuel rack edge is possible and the effect upon k_{eff} was calculated in detail.

It is also credible that an assembly could be laid across the top of a fuel rack. In this case, the distance between the top of the active stored fuel and the bottom of the misplaced fuel will be greater than 12 inches, a distance that effectively "decouples" the two groups of fuel. No increase in k_{eff} will result from this incident.

3.1.1.3.3 Pool Temperature Variation

Calculations were performed to determine the sensitivity of k_{eff} for the reference configuration to variations in the spent fuel pool temperature. The pool temperature was varied from 68°F to 260°F. These calculations verify that 68°F is the most reactive SFP temperature.

3.1.1.3.4 Fuel Drop Incident

A dropped fuel assembly could impact directly on the fuel stored in a cell. The effect of this type of fuel drop incident was dismissed from criticality considerations in that no credit was taken for axial buckling. The net effect is that the rack is neutronically modeled as infinite in the axial direction.

3.1.1.3.5 Heavy Object Drop

It has been concluded that k_{eff} would decrease if a heavy object were dropped onto the storage rack with sufficient impact to cause structural deformation. The basis for this conclusion is that the principal effect of dropping a heavy object would be to displace water from the rack. Depletion of water has been shown to lead to decreases in k_{eff}.

It would not be possible for a dropped heavy object to eject the poison material from the rack; the crushing effect of the heavy object could only act to compress the fuel and poison together.

3.1.1.3.6 Load Drops and Tornado Missiles

The structural analysis indicates that load drop accidents and tornado missiles (as defined in Section 2.3.2) do not cause structural deformation of the rack (see Section 3.3.3). The effect upon k_{eff} of these accident sequences was not considered further in this analysis.

3.1.1.3.7 Seismic Event

Vibration of the fuel storage racks will cause only random eccentric fuel positioning that has already been taken into account under normal variations and thus will not be considered further. The maximum rack sliding displacement of 0.31 inch (Section 3.3.3.7.5) was determined to have no effect upon k_{eff}.

During the phased installation of the new racks, a minimum separation distance of 2-5/8 inches will be maintained between all racks of the current design and the proposed new rack modules. This will ensure adequate margin to maintain k_{eff} less than 0.95 for any transition pool configuration.

3.1.1.3.8 "Worst Case" Abnormal Configuration

The "worst case" abnormal configuration considers the effect of the most adverse abnormal condition in combination with the worst case normal configuration as given in Section 3.1.1.2.11.

3.1.2 Analytical Methods

3.1.2.1 Method of Analysis

The nuclear criticality analyses of the new spent fuel storage rack design will be performed by two methods:

1. NITAWL/KENO method with the 123-group XSDRN library, and

2. CASMO/PDQ method in two neutron energy groups.

Method 1 employs the NITAWL (Reference 16) subroutine of the AMPX code package to perform resonance self-shielding calculations and collect the selected cross-section data from the XSDRN (Reference 16) library into the format required by KENO-IV (Reference 18). The KENO-IV code then solves the neutron transport equation by means of a Monte Carlo technique. This will be

used as the principal calculational method for determining reference reactivities. The NITAWL/KENO methodology has been benchmarked by YAEC against reference critical experiments and the application of this methodology is described in detail in Reference 19.

Method 2 uses the lattice code CASMO-2 (Reference 20) to produce two-group macroscopic cross-sections in PDQ format. With the aid of the G- and H-factor calculations in CASMO, these two-group resultant cross-sections maintain the reaction rates in the boral as determined by the CASMO 25 neutron group transport calculation. The two-group macroscopic cross-sections from CASMO are then configured into HARMONY tableset format by the code CHART-2 (Reference 21) for direct input to PDQ. The PDQ-7 code (Reference 22) then solves the diffusion equation by means of a finite difference technique.

Method 2 will be used only to predict trends in order to reduce the number of necessary KENO calculations as well as calculate the reactivity contribution, if any, due to abnormal configurations.

The spent fuel rack criticality analysis methods are summarized in Figure 3-1.

3.1.2.2 Uncertainty and Benchmark Calculations

In order to comply with the requirements of ANSI N16.9-1975[23], YAEC has previously benchmarked their spent fuel rack design calculations against experimental results obtained by Babcock and Wilcox (B&W) and Battelle Northwest Laboratories (BNL). These experiments provide the most geometrically representative criticality benchmarking data presently available for spent fuel rack design. The KENO-IV[18] calculational tool, a Monte Carlo code, has become the industry standard in criticality studies due to its ability to analyze large problems with great detail in both energy spectrum and geometry. However, the nature of the results, namely, a k_{eff} with an applicable statistical uncertainty, makes the correct combination of problem uncertainties imperative to achieve a final k_{eff} which may be compared to the desired limit. At YAEC, worst case mechanical tolerances will always be

assumed to be implicit in the problem. Therefore, the only uncertainties which are required to be considered are the calculational bias and the KENO statistical uncertainty. The total uncertainty is then calculated as simply:

$$\text{Total Uncertainty} = \sqrt{(\text{Calculational Uncertainty})^2 + (\text{KENO Uncertainty})^2}$$
$$= \Delta k_u$$

where the calculational uncertainty was conservatively set at 0.008 from benchmark results[19] and the KENO uncertainty equaling the 95/95 limit of statistical uncertainty in the particular problem under examination. These uncertainties are independent, which therefore allows combination in the above manner.

With the total uncertainty thus defined, the final "worst case abnormal spent fuel rack k_{eff}" given in Section 3.1.3.3 to a 95% probability at a 95% confidence level is:

$$\text{worst case abnormal } k_{eff} = k_{ref} + \Delta k_u + \Delta k_s + \Delta k_c$$

where:

k_{ref} = The computed effective multiplication factor for the reference configuration case design calculation;

Δk_u = The total calculation uncertainty as described above;

Δk_s = The total Δk due to normal variations which are statistically independent. Δk_s is calculated using the "square root sum of the squares" method with all uncertainties at the 95% probability level with a 95% confidence value;

Δk_c = The total Δk due to correlated abnormal effects which must be combined additively;

and the worst case abnormal k_{eff} must be shown to be less than or equal to 0.95.

3.1.3 Results of Criticality Calculations

3.1.3.1 Reference Configuration

The reference configuration k_{eff} value for the Vermont Yankee spent fuel storage racks was calculated to be 0.9046 with a total uncertainty of 0.0141 Δk.

3.1.3.2 "Worst Case" Normal Conditions

The k_{eff} for the worst case normal configuration results from the statistical sum of the Δk's due to normal variations added to the k_{eff} for the reference configuration plus calculational uncertainty.

k_{eff} of reference configuration	0.9046
Δk due to uncertainty (total)	0.0141
Δk due to poison thickness variation	0.0060
Δk due to poison concentration variation	0.0038
Δk due to variation in pitch	0.0033
Δk due to Zircaloy shrouds	0.0016
Δk due to poison width variation	0.0006
Δk due to stainless steel thickness variation	0.0001
Δk due to cladding thickness variation	0.0001
Δk due to eccentric fuel positioning	0.0000

$$\text{Worst Case Normal } k_{eff} = k_{ref} + \Delta k_u + \Delta k_s$$

$$= 0.9046 + 0.0141 + 0.0080$$

$$= 0.9267$$

3.1.3.3 "Worst Case" Abnormal k_{eff}

The effects of single cell displacement, fuel handling incidents, pool temperature variations, fuel and heavy object drop and seismic incidents have either zero or negative effects on k_{eff}. Since none of the abnormal

variations are positive, the worst case abnormal k_{eff} is equal to the worst case normal k_{eff}, 0.9267.

3.1.4 Conclusion

The k_{eff} of the spent fuel rack, for all anticipated normal and abnormal conditions is 0.9267 which is less than the required 0.95 limit.

3.2 THERMAL-HYDRAULIC ANALYSIS

This section describes and evaluates the methods used to provide cooling for the spent fuel assemblies stored in the SFP. The existing SFP cooling systems and the proposed modification to the cooling water return spargers are discussed in Section 3.2.1 and 3.2.1.1, respectively. The adequacy of the existing systems to remove decay heat for the design conditions (listed in Section 2.2) are evaluated in Section 3.2.2. Section 3.2.3 analyzes the natural circulation cooling capability of an individual fuel assembly to demonstrate adequate cooling.

3.2.1 Spent Fuel Cooling System Description and Operation

The Fuel Pool Cooling and Demineralizer System (SFPCDS) is shown in Figures 3-2 and 3-3. Specifications for the SFPCDS are provided in Table 3-2. The system cools the fuel storage pool by transferring the spent fuel decay heat through a heat exchanger to the Reactor Building Closed Cooling Water System. Water purity and clarity in the storage pool, reactor well and dryer-separator storage pit are maintained by filtering and demineralizing the pool water through a filter-demineralizer, which is shown in Figure 3-3.

The system consists of two circulating pumps connected in parallel, two heat exchangers, two filter-demineralizers and the required piping, valves and instrumentation. Each pump has a design capacity equal to the system design flow rate and is capable of simultaneous operation. Two filter-demineralizers are provided, each with a design capacity equal to the design flow rate. The pumps circulate the pool water in a closed loop, taking suction from the spent

fuel storage pool, circulating the water through the heat exchangers and filters, and discharging it through diffusers at the bottom of the fuel pool and reactor well. Section 3.2.1.1 discusses the proposed removal of the fuel pool spargers.

The fuel pool pumps and heat exchangers are located in the Reactor Building below the bottom of the fuel pool. The fuel pool filter-demineralizers, which collect radioactive corrosion products, are located in the Radwaste Building. The fuel pool concrete structure, metal liner and spent fuel storage racks are designed to withstand earthquake loads of Class I seismic intensity.

The pool is filled and makeup to the pool is supplied from the Condensate Transfer System and Demineralizer Water Transfer System. Water is removed from the pool via the fuel pool pumps through the fuel pool filter-demineralizer units to the condensate storage tank.

Fuel pool water is continuously recirculated except during the period when the reactor well and dryer-separator pit are being drained. The heat exchangers are designed to remove the decay heat load of the normal discharge batch of spent fuel. The operating temperature of the fuel pool is permitted to rise up to 125°F alarm, approximately 25°F above the normal operating temperature, when the circulating flow is interrupted to drain the reactor well or when larger than normal batches of fuel are stored. The heat exchangers in the Residual Heat Removal System (RHR) can be used in conjunction with the Fuel Pool Cooling and Demineralizer System to supplement pool cooling in the event that a larger than normal amount of fuel is stored in the pool. The RHR System also provides a seismically qualified fuel pool cooling capability that can be relied upon should the normal SFPCD System require backup.

When fuel storage racks of increased capacity were previously installed[2,3], calculations of expected decay heat loads from normal refuelings and from a full core discharge with previous cycles of spent fuel in the racks were performed. Examination of data from these analyses shows

that while fuel pool heat exchanger capacity at most limiting conditions is exceeded, the backup capability of the RHR System is more than sufficient.

Two small skimmer pumps are provided which take suction from the top of the pool to remove surface debris. Water is pumped through the cartridge filters then back to the pool through the service boxes located around the pool.

Pool water clarity and purity are maintained by a combination of filtering and ion exchange processes. The filter-demineralizer maintains total heavy element content (Cu, Ni, Fe, Hg, etc.), at 0.1 ppm or less, with a pH range of 6.0 to 7.5 for compatibility with fuel racks and other equipment. Particulate material is removed from the circulated water by the pressure precoat filter-demineralizer unit in which a finely divided disposable filter medium is supported on permanent filter elements. A post-strainer is provided in the effluent stream of the filter-demineralizer to limit the migration of the filter material. The filter-demineralizer units are located separately in shielded rooms. Each room contains only the filter-demineralizer and piping. All inlet, outlet, recycle, vent, drain and other valves are located on the outside of one shielding wall of the room, together with necessary piping and headers, instrument elements and controls. Penetrations through shielding walls are located so as not to compromise radiation shielding requirements.

The fuel pool filter-demineralizers are also used to process liquid radioactive wastes. This is discussed in Chapter 9 of the Vermont Yankee FSAR[11].

The system instrumentation is provided for both automatic and remote manual operations. Instrumentation and controls are provided to detect, control and record pump operation, pool temperature and system flow. A Pool Leak Detection System has been provided to monitor leakage and thus indicate pool integrity.

The pumps are controlled locally. Pump low suction pressure automatically turns off the pumps. A pump low discharge pressure alarm indicates in the Main Control Room and in the Pump Room. The controls for the remote controlled valves which discharge the fuel pool water to the condenser hotwell, condensate storage tank and waste surge tank, are located on a panel in the Control Room. The open or closed condition of each of these valves is indicated by lights in the Control Room.

The flow rate through each of the filter-demineralizers is indicated by a flow indicator on the Pump Room panel. The flow indicators can be seen by the operators from the vicinity of the Fuel Pool Cooling System control valves.

A high rate of leakage through the refueling bellows assembly, drywell to reactor seal or the fuel pool gates is indicated by lights on the operating floor instrument racks, and is alarmed in the Main Control Room.

The filter-demineralizers are controlled from a local panel in the Radwaste Building. Differential pressure and conductivity instrumentation are provided for each filter-demineralizer unit to indicate when backwash is required. Suitable alarms, differential pressure indicators and flow indicators are provided to monitor the condition of the filter-demineralizers.

3.2.1.1 Removal of Sparger Lines

Installation of the new SFP racks will entail removal of the existing sparger lines which currently distribute the cooling water return near the SFP floor. The two existing discharge lines to the SFP will be removed just above the racks so that cooling flow will be directed downward. The downward momentum and negative buoyancy of the cooler discharged water will carry the flow downward. The analysis of the spent fuel cooling capability (Section 3.2.3) takes no credit for cooler water introduced in the lower plenum and is, therefore, conservative and not affected by the removal of the spargers.

3.2.2 Spent Fuel Cooling System Adequacy

The ability to cool the SFP was evaluated to assure that the decay heat can be removed during all operating conditions and that acceptable SFP temperatures are maintained. Three design conditions (Section 2.2) were chosen to bound all expected operating modes:

- The SFPCDS is used for decay heat removal following a normal refueling outage. Condition 1 was chosen to bound this operating procedure.

- The RHR System is aligned to cool the SFP if a full core is discharged during a refueling. Condition 2 was chosen to bound this operating procedure.

- In addition to the cooling system evaluation, the time to reach bulk SFP boiling and the water make-up rates for boiling are calculated for the unlikely event of a total loss of SFP cooling. Condition 3 was chosen to bound this condition.

The maximum decay heat loads for the SFPCDS and RHR System are calculated in Section 3.2.2.1. The method for predicting the new SFP temperatures used in Sections 3.2.2.2 and 3.2.2.3 assume the same heat exchanger effectiveness and conservative design fouling factors used in the original design. Section 3.2.2.4 conservatively assumes adiabatic SFP heatup should all SFP cooling be lost.

3.2.2.1 Analytical Methods

The fission product decay heat power is calculated to find the total heat load on the SFPCDS and on the RHR System for the conditions defined in Section 2.2. Two projected discharge fuel schedules were analyzed to bound anticipated variations in SFP heat loads. These scenarios are: (1) An annual cycle of quarter core reloads (92 assemblies); and (2) An 18-month cycle of approximately one-third core reloads (136 assemblies). The discharge

schedules for these two scenarios were devised conservatively to predict a maximum decay heat load bounding the various refueling schedules which may realistically take ce. The most limiting condition (Condition 2) is a full core discharge (annual cycle), 10 days after shutdown that fills the SFP to its full capacity. The proposed SFP capacity is 2,870 assemblies.

The decay heat power analysis assumed that each 18-month cycle accumulated burnup at 100 percent power for an exposure time of 456 days per 18-month period. This is equivalent to a cycle capacity factor of 83%. The most limiting Condition 1 heat load assumes that all 136 assemblies are discharged simultaneously to the SFP, 10 days after shutdown. A similar assumption is made for the most limiting Condition 2 heat load on the SFP, discharging the full core (annual cycle) simultaneously. In realistic operation, some time is needed to move the assemblies to the SFP. In the Condition 2 case, the heat load was calculated assuming that all assemblies discharged had received a maximum burnup, i.e., a burnup equivalent to 4 annual cycles. The actual burnup should be less for a full core discharge. The above assumptions are conservative relative to actual operation.

Decay heat is found using the NRC Branch Technical Position ASB 9-2[24]. The correlations in ASB 9-2 are quite conservative for SFP decay heat calculations. These correlations include a 20% uncertainty for shutdown times $0 \leq t_s < 10^3$ and a 10% uncertainty for $10^3 \leq t_s < 10^7$ where t_s is shutdown time in seconds.

The decay heat power for both Conditions 1 and 2 with the 18-month and annual cycles is summarized in Table 3-3. The limiting Condition 1 heat load is 6.84×10^6 Btu/hr for an 18-month cycle, 10 days after shutdown and the maximum Condition 2 for a full core discharge heat load is 15.63×10^6 Btu/hr for an annual cycle, 10 days after shutdown.

3.2.2.2 Results for Condition 1

Based on an 18-month refueling schedule (worst case), the maximum heat load following a refueling outage with 136 assemblies discharged is 6.84×10^6 Btu/hr at 10 days after shutdown. At this heat generation rate,

the SFPCDS will maintain the SFP temperature below 125°F if both SFPCDS trains are used and below 150°F if one train is used with a maximum hot weather Reactor Building Closed Cooling Water (RBCCW) temperature of 85°F. The decay heat power drops fairly rapidly as short-lived fission products decay. After several days, the decay heat power drops below 3.79×10^6 Btu/hr, which allows the SFPCDS to maintain a SFP temperature <125°F even with one SFPCDS train and maximum RBCCW temperatures. Given the conservatisms built into the decay heat generation calculation and the conservative heat exchanger effectiveness assumed, it is concluded that the SFP temperature will not exceed 150°F for Condition 1.

3.2.2.3 Results for Condition 2

Based on a 12-month refueling schedule (worst case), the maximum heat load following a full core discharge is 15.63×10^6 Btu/hr at 10 days after shutdown. At the above heat generation rate (i.e., 10 days after shutdown), one RHR train will maintain the calculated SFP water temperature below 150°F.

After several days, the decay heat power drops below 12.1×10^6 Btu/hr, which allows both SFPCS trains to maintain the calculated SFP temperature below 150°F, even with a maximum RBCCW temperature of 85°F.

Due to the conservatisms built into the decay heat calculation (particularly the assumption that the entire core is placed into the SFP with only 10 days decay), along with the conservative heat exchanger effectiveness assumed, the actual maximum SFP water temperature should always be <150°F.

The impact of a full core discharge placed in the SFP 150 hours after shutdown was also considered. In this case, a maximum heat load of 2.5×10^7 Btu/hr is discharged to the SFP. This is well within the cooling capacity of one operating RHR train (5.75×10^7 Btu/hr rated capacity) thus maintaining SFP water temperature <150°F.

In any event, since the SFP water temperature is continuously monitored and alarmed in the Control Room (Section 3.2.1), appropriate actions can be taken, such as initiating RHR cooling, should the SFP water temperature approach 125°F at any time during refueling operations. RHR availability, therefore, restricts startup until the SFPCDS is capable of maintaining the pool temperature below 125°F.

3.2.2.4 Results for Condition 3

This condition assumes the loss of all SFPCDS cooling subsequent to a seismic event. Under maximum heat load conditions, calculations show that fuel pool surface boiling will begin in approximately 6 hours (150 hours after shutdown). This allows sufficient time for operators to enter the Reactor Building and establish RHR to fuel pool cooling makeup water at greater than the expected boiloff rate and make necessary repairs. Additionally, the SFP is designed such that no failure, in a seismic event, can drain the SFP to a water level less than 10 feet above the spent fuel racks assuming a total loss of makeup water.

3.2.3 Spent Fuel Cooling Capability

3.2.3.1 Analytical Methods

The underlying assumptions for the fuel cooling capability analysis are described below. Figure 3-4 shows the location in the SFP of the assembly used in the analysis. The row selected is representative of the most critical area in terms of cooling for the assemblies in the SFP. A bulk pool temperature of 150°F was assumed. This temperature is the design temperature for Vermont Yankee SFP and provides a conservative bounding assumption. No credit is taken for cooling water being directed to the lower plenum, or for the lower plenum temperatures being lower than the bulk pool temperature. Instead, the model assumes that the only driving force for flow is the buoyant force generated by the removal of decay heat. The model only considers flow in one direction to the highest resistance assembly, when actually crossflow will occur which will lower flow resistance in the lower

plenum. All the assemblies considered were assumed to have operated at peak power. The peak assembly power is obtained by multiplying the average assembly power by a radial peaking factor of 1.45. The decay power of an assembly was conservatively evaluated by multiplying peak assembly power with the normalized decay power using the ASB 9-2[24] correlations at a shutdown time of 150 hours. The 150 hour reactor shutdown time was considered because it conservatively bounds realistic SFP heat loadings. The reactor operation time was conservatively assumed to be 16,000 hours at full power, bounding future operations. Pressure losses from the downcomer region between the racks and the wall, the inlet plenum and the fuel assemblies are explicitly considered in the analysis. Downflow in the vacant spaces between racks is conservatively ignored in the analysis. The resistance to the downflow is conservatively maximized by using the smallest rack to pool wall clearance even though a larger clearance is available at certain elevations. The results of the analysis are bounding given the above conservative assumptions.

3.2.3.2 Results

For both Conditions 1 and 2, using 150 hours after shutdown, a conservative analysis was performed. The results indicate that the water in the most limiting fuel assembly will have a temperature increase of 71°F, resulting in an exit temperature of 221°F. This is below the boiling point of 237°F at fuel exit and indicates that natural circulation provides adequate cooling for the assembly in the worst location. This demonstrates that the rack, downcomer and plenum designs are adequately sized to ensure sufficient flow.

3.3 MECHANICAL, MATERIAL AND STRUCTURAL ANALYSIS

3.3.1 Mechanical Analysis

3.3.1.1 Description of Spent Fuel Pool

The Spent Fuel Pool (SFP) is a reinforced concrete structure supported by the Reactor Building walls at Elevation 303'. The SFP is 26 feet-0 inches

wide by 40 feet-0 inches long by 39 feet-3/4 inch deep and is completely lined with seam welded ASTM A240 Type 304 stainless steel. The floor plate is 1/4-inch thick and the wall plate is 3/16-inch thick. The storage pool volume is approximately 41,600 $feet^3$.

The design of Vermont Yankee's spent fuel pool is such that no fuel in the spent fuel storage racks can be uncovered in the event of a failure of the reactor cavity seal or the failure of piping associated with the spent fuel storage system or the reactor vessel.

The Fuel Pool Cooling and Demineralizer System cools the fuel storage pool by transferring the spent fuel decay heat through a heat exchanger to the Reactor Building Closed Cooling Water System (RBCCW). Water purity and clarity in the storage pool is maintained by filtering and demineralizing the pool water through a filter demineralizer.

3.3.1.2 Spent Fuel Pool Chemistry

The spent fuel pool water clarity and purity is maintained by a combination of filtering and ion exchange processes (see Section 3.2.1, Spent Fuel Cooling System Description and Operation). The filter-demineralizer maintains total heavy element content (Cu, Ni, Fe, Hg, etc.) at 0.1 ppm or less. The pH range of the water is 6.0 to 7.5 which is compatible with the new stainless steel racks along with other equipment in the pool.

Particulate material is removed from the circulated water by the pressure precoat filter-demineralizer unit in which a finely divided disposable filter medium is supported on permanent filter elements. The filter medium is replaced when the pressure drop is excessive or the ion exchange resin is depleted.

3.3.1.3 Description of New Racks

The freestanding high density neutron absorber spent fuel storage racks are designed to provide a maximum storage capacity of 2,870 locations in the spent fuel pool. The fuel storage rack arrangement will contain storage racks with array sizes shown in Figure 1-3.

Each rack consists of a welded assembly of individual storage cells in a staggered checkerboard array. The storage cells are comprised of Type 304L stainless steel boxes (5.922 inches square ID) welded to each other with corner angles to maintain a pitch of 6.218 inches. Each storage cell has an interior height of 168 inches. The construction of the storage cells provides four vented (open to the pool) compartments in which B_4C neutron absorber elements are placed for criticality control. The neutron absorber elements are positioned on the side of the storage cell at an elevation corresponding to the fuel region of a spent fuel assembly placed within the cell. The bottom of each storage cell sits on, and is welded to, the rack base plate which provides the level seating surface required for each fuel assembly and also contains the openings necessary for adequate cooling flow. Figure 1-4 shows a schematic drawing of a typical rack.

The rack support feet raise the racks above the pool floor to the height required to provide an adequately sized cooling water supply plenum. The support feet contain remotely adjustable jackscrews (accessible from the top of the spent fuel rack) to facilitate and ensure proper support for the vertical loads and achieve the required levelness.

The storage cell structure, acting in concert with the rack base and the rack support feet, provides the structural strength and stiffness characteristics required for the rack to accommodate the applicable seismic excitation for the Vermont Yankee Nuclear Power Station. No wall bracing or attachments are required to support the fuel racks under any design condition. Sufficient space is provided between spent fuel racks to preclude impact/collision in the event that two adjacent racks slide toward each other during a seismic event.

3.3.1.4 Fuel Assembly

The fuel assembly parameters used to design and analyze the rack are listed in Table 3-1.

3.3.1.5 Control Rods

The weight of control rods and control rod storage racks used in the analysis of the SFP floor is listed in Table 3-1.

3.3.1.6 Reactor Building Bridge Crane

The Overhead Crane Handling System for VYNPS consists of an overhead, bridge-type crane, spent fuel cask lifting devices, and controls. The Overhead Crane Handling System is used during plant operation for lifting and transporting the spent fuel shipping cask between the spent fuel pool and the cask decontamination/shipping area. The overhead crane is located indoors in a controlled environment and has a main hoist rated at 110 tons. The Overhead Crane Handling System has been designed to minimize the potential of a spent fuel cask drop accident which could result in an unacceptable release of radioactive materials. Further, the crane was designed in accordance with the Electric Overhead Crane Institute (EOCI) Specification Number 61, and with the minor exceptions previously reviewed and approved by NRC (Reference 30), meets all requirements of the Crane Manufacturers Association of America (CMAA) Specification Number 70.

Redundant limit switches, of different types, are provided to prevent overhoisting, and a load indicating/limiting device prevents overloading. An overspeed switch is provided on each load path to prevent runaway lowering. Operating power and control for all crane motions are provided by a General Electric max-speed control system which incorporates a torque limiter on the main hoist for additional overload protection.

NRC's previous review of Vermont Yankee's Overhead Crane Handling System determined that the cask drop accident was adequately resolved by VYNPC and concluded that fuel cask handling with the Reactor Building crane under the surveillance and testing requirements of the Technical Specifications was acceptable[29].

3.3.1.7 Spent Fuel Refueling Platform

The refueling platform is wheel-mounted on tracks extending along each side of the reactor well and fuel pool. The platform supports the refueling grapple and auxiliary hoists will be used to transport fuel assemblies underwater from the existing racks to the new racks. The hoist travel is designed to limit the maximum lift of a fuel assembly to a safe shielded

depth. A single operator is capable of controlling all the motions of the platform required to handle the fuel assemblies.

3.3.2 Materials

3.3.2.1 Rack Materials

All structural members of the spent fuel racks are ASTM A240 Type 304L for sheets and ASTM A479 or A276 Type 304L for round bars. Intergranular corrosion tests shall be performed on all austenitic stainless steel in accordance with ASTM A262, Practice E.

The threaded rods attached to the leveling pads are SA-564, Type 630 17-4 pH-hardened stainless steel. The threaded rods are heat-treated, chemically cleaned and chrome-plated to assure functionability. Chrome plating is per ASTM B-254, "Hard Chrome Plating," with a thickness of 0.0003 to 0.0005 inches.

The neutron absorber material is Boral with a minimum areal density of 0.027 B^{10} g/cm^2.

The compartments containing the Boral are not watertight. This precludes the potential for a pressure buildup within the compartment due to radiolysis of entrained water and any Boral off-gassing that may occur.

3.3.2.2 Prohibited Materials

The materials used in the construction shall be compatible with the storage environment of the racks, and shall not contaminate the fuel assemblies with foreign particles or matter that would compromise the integrity or function of a fuel assembly during its lifetime, nor shall they contaminate the water in the storage pool. Materials that shall not be a part of fabrication processes or the final product are:

- Any material that contains halogens in excess of 200 ppm, including chlorinated cleaning compounds
- Lead

- Mercury
- Sulfur
- Phosphorus
- Zinc
- Copper and copper alloys except for backing strips used during fabrication in accordance with approved welding procedures
- Cadmium
- Tin
- Antimony
- Bismuth
- Misch metal
- Carbon steel, e.g., carbon steel contaminated wire brushes
- Magnesium oxide, e.g., insulation
- Neoprene or other similar gasket materials made of halogen-containing elastomers
- Viton
- Saran
- Silastic Ls-3
- Rubber-bonded asbestos
- TFE (Teflon) containing more than 0.075 percent total chlorine (glass-filled) and TFE films containing more than 0.05 percent total chlorine
- Nylon containing more than 0.07 percent total chlorine
- Stellite or Cobalt containing materials

3.3.2.3 Effects of Spent Fuel Pool Environment

The austenitic stainless steel to be used in the rack fabrication will have a maximum carbon content of 0.03% by weight for 304L which will minimize the sensitization in weld heat-affected zones. All austenitic stainless steel shall have intergranular corrosion tests performed in accordance with ASTM A262, Practice E.

The heat treatment scale on the 17-4 PH-threaded stainless steel rod will be chemically removed to prevent possible stress corrosion, cracking, pitting, and sludge formation.

Based on the above, the design requirements for the racks, the program to be implemented to minimize the possibility of sensitization in rack materials and the SFP temperature control requirements the effects of corrosion, including corrosion cracking over the lifetime of the spent fuel racks are expected to be insignificant.

Boral, the neutron absorber to be used, has considerable industry experience as a spent fuel pool poison. In addition, the compartments containing the boral are not water-tight, precluding the potential for a pressure buildup within the compartments.

Testing done at the University of Michigan has shown that at a high radiation level, the boral remains unaffected. Additionally, General Electric has estimated the service life of vented boral in deionized water to be much greater than 40 years. GE has also estimated the corrosion along the edge of boral exposed to deionized pool water to be minimal.

3.3.3 Structural Analysis

3.3.3.1 Description of Existing Structures

The spent fuel pool and Reactor Building are Seismic Category I structures designed in accordance with FSAR Section 12.2[11]. The spent fuel pool is located in the south end of the Reactor Building. Its dimensions are 26.0 ft wide x 40 ft long and 39 ft - 3/4 in deep. The normal pool water level is at El. 342 ft - 2 in. The bottom of pool elevation is 306 ft - 5 in.

The pool is connected with floor slabs at El. 303.00', 318.67' and 345.17'. It is primarily supported by two north-south run and one east-west run concrete girders at El. 303.00'. The north-south run girders are in turn supported by the reactor shield wall and the south exterior wall of the building.

The interior of the pool is constructed with stainless steel plates anchored to concrete walls and floor by stiffeners. The liner is designed as a leaktight membrane and is not relied upon as a structural member. The leaktightness of the liner is monitored by the leakchase system described in Section 12.3.4 of the FSAR[11]. The plate thickness of the floor liner is 1/4 inch and the plate thickness of the wall liner is 3/16 inch. The spacing for horizontal and vertical stiffeners on wall liners is 6 feet-6 inches and 1 foot-8 inches, respectively. On the floor liner, the stiffeners are only in the north-south direction having a spacing of 4 feet-5 inches or 5 feet-7 inches. Underneath the floor liner there is an 11-inch thick grout on top of the rough concrete surface at El. 305.50'.

A fuel cask storage area at the bottom of the pool is provided. The liner of the storage area is reinforced by a 7/8-inch thick plate.

On the refueling floor (El. 345.17'), the refueling bridge and service platform is supported by the east and west walls of the pool. The platform has a weight of about 25.0 kips (Reference 13).

During normal operation conditions, the reactor cavity above the steel drywell head is dry. However, this area is flooded during refueling/maintenance activities.

3.3.3.2 Description of the New Rack Structure

The new Vermont Yankee spent fuel storage racks have been designed to provide a storage capacity of 2,870 fuel locations. The arrangement of the storage racks in the fuel pool is shown in Figure 1-3. The storage rack arrangement consists of the following seven types of storage rack arrays:

Four 20 x 15 rack modules
One 20 x 14 rack module
One 20 x 18 rack module
One 15 x 12 rack module
One 20 x 13 rack module
One 20 x 14SP rack module
One 20 x 13SP rack module

Each rack consists of a welded assembly of individual storage cells in a staggered checkerboard array. The storage cells are comprised of Type 304L stainless steel boxes (5.922 inches square ID) welded to each other with corner angles to maintain a pitch of 6.218 inches. Four poison sheets are place between the pockets formed by the 0.075-inch cell wall and the 0.045-inch poison retainer of each storage cell. The unitized construction of the racks provides the strength and stiffness required to accommodate the seismic excitation.

The bottom of each storage cell sits on and is welded to the rack base plate. The base is a 1/2-inch thick stainless steel plate. It provides the level support surface required for each fuel assembly and contains a cooling flow orifice for each cell. Each storage cell will have a total interior height of 168 inches to ensure that the stored fuel assembly does not project above the top of the storage cell.

The storage racks are positioned on the pool floor so that adequate clearances are provided between racks and between the racks and pool structures to avoid impacting of the racks during seismic events. The only horizontal seismic loads transmitted from the rack structure to the pool floor are those associated with friction between the rack structure and the pool liner. The vertical deadweight and seismic loads are transmitted directly to the pool floor by the support feet.

3.3.3.3 Applicable Codes, Standards and Specifications

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

1. ASME Code, Sections II, III and IX, 1980, with Addenda through Summer 1981

2. ASTM, Annual Book of ASTM Standards

3. USNRC Standard Review Plan, Section 9.1.2, July 1981

4. USNRC "Position Paper for Review and Acceptance of Spent Fuel Storage and Handling Applications," April 14, 1978 and Supplement, January 18, 1979

5. Yankee Atomic Electric Company (YAEC) Specification VY-ME-S1, "Specification for High Density Spent Fuel Storage Racks"

6. AISC, Manual of Steel Construction, Eighth Edition, 1980

7. AISI, Stainless Steel Cold-Formed Structural Design Manual, 1974 Edition

8. USNRC SRP 3.8.4, "Other Category I Structures"

9. USNRC Regulatory Guide 1-122, "Development of Floor Design Spectra," 1978

3.3.3.4 Loading Conditions and Acceptance Criteria

3.3.3.4.1 Loading Conditions

The following load cases have been considered in the analysis in accordance with the requirements of USNRC Standard Review Plan, Section 3.8.4 (Reference 9) and the USNRC Position Paper (Reference 6) and Vermont Yankee requirements (Reference 25):

Normal Loads

Deadweight (D)	Loading due to rack base structure and storage cells.
Live Load (L)	Loading due to the weight of the fuel assemblies.

Thermal Load (T_o)	Loading due to thermal gradients during normal operating conditions. Thermal effects are negligible as clearances are provided for unrestrained expansion of racks.

Severe Environmental Load

Operating Basis Earthquake (E)	Horizontal acceleration time history data furnished by Bechtel Corporation (Reference 26) and maximum vertical seismic acceleration were applied simultaneously. The seismic time-histories were calculated by Bechtel as conservative envelopes to the FSAR floor design response spectra shown in Figures 2-3 and 2-4. Maximum vertical acceleration was taken from the applicable vertical spectra at the fundamental vertical frequency of the rack.

Extreme Environmental Load

Safe Shutdown (E')	Horizontal acceleration time-history data furnished by Bechtel Corporation (Reference 26) and maximum vertical seismic acceleration were applied simultaneously. The seismic time-histories were calculated by Bechtel as conservative envelopes to the FSAR floor design response spectra shown in Figures 2-3 and 2-4. Maximum vertical acceleration was taken from the applicable vertical spectra at the fundamental vertical frequency of the rack.

Abnormal Loads

Uplift Load (P_f)	4000 lb minimum net upward load at critical location of rack.
Fuel Assembly Drop (F_d)	Fuel assembly dropping from 12" above top of the rack.
Thermal Load (T_a)	Loading due to highest temperature associated with the postulated abnormal design conditions. Thermal effects are negligible as clearances are provided for unrestrained expansion of the racks.

The existing tornado missile analysis (Reference 37) for Vermont Yankee demonstrated that the water above the stored fuel pool combined with the high fuel pool location in the Reactor Building provide sufficient protection against tornado effects. The new rack design will not affect the basis of this calculation. Therefore, the previous analysis for tornado effects is not affected.

3.3.3.4.2 Structural Acceptance Criteria

The following load combinations and load limits constitute the structural acceptance criteria, in accordance with requirements of References 6 and 9:

Load Combination	Acceptance Limit*
1. D + L	Level A service limits
2. D + L + E	Level A service limits
3. D + L + T_o	Level A service limits

4.	$D + L + T_o + E$	Level A service limits
5.	$D + L + T_a + E$	Level B service limits (1.33S or S_y)
6.	$D + L + T_o + P_f$	Level B service limits
7.	$D + L + T_a + E'$	Level D service limits (1.6S or S_y)
8.	$D + L + F_d$	The overall structural integrity of the racks shall be demonstrated.

Where S is the required section strength based on the elastic design methods and the allowable stresses defined in ASME Code, Section III, Subsection NF. The yield stress value (S_y) for Type 304L stainless steel is taken as 25.0 ksi from the ASME Code Appendices. Allowable concrete bearing stresses were obtained from ACI 318 for 4000 psi concrete strength.

NOTE: * The acceptance criteria are based on the applicable sections of the NRC Position Paper on Fuel Storage Racks, SRP 3.8.4, and the ASME Code, Section III, Subsection NF.

3.3.3.5 Method of Analysis

The basic rack seismic stress analysis steps are outlined below:

1. Detailed finite element models of the racks were developed using beam and plate elements.

2. Three-dimensional beam-type nonlinear models of the combined rack and contained fuel assemblies were developed using data from the detailed finite element models.

3. Time-history analyses were performed on the nonlinear models of Step 2 assuming fully loaded racks and a high value for the friction coefficient between the racks and pool floor. This step yielded pool floor load time-histories and data for the rack stress analysis.

4. Detailed rack stresses were determined using the models of Step 1 and loads from Step 3.

3.3.3.5.1 Detailed Finite Element Models (For Static Analysis)

In order to account for various storage rack configurations, three critical storage racks were selected for detailed stress analysis. These three storage rack types are 15 x 12, 20 x 18, and 20 x 14SP storage rack modules as they bound all other arrays in size and configuration.

These rack modules were mathematically modeled as a finite element structure consisting of discrete three-dimensional elastic plate elements interconnected at a finite number of points using the ANSYS[27] computer program. The stiffness characteristics of the structural members were related to the plate thickness, cross sectional area, effective shear area and moment of inertia of the element section. Proper boundary conditions were provided at the bottom of the rack modules. Six degrees of freedom (three translations and three rotations) were permitted at each nodal point. The static live loads were applied as pressure on the face of the appropriate plate elements. The deadweight of the rack was accounted for by specifying an appropriate density, resulting in proper distribution.

3.3.3.5.2 Mathematical Formulation of the Static Analysis

The static analysis of the finite element model has been performed using the direct stiffness method of structural analysis. If the force displacement relationship of each of the discrete structural elements is known (the element stiffness matrix), then force displacement relationship for the entire structure can be assembled using standard matrix methods as shown below.

For each element:

$$ku = f$$

where:

k = Element stiffness matrix

u = Element nodal displacement vector

f = Element nodal force vector

For the detailed idealized system, the equation of equilibrium may be written in matrix form as follows:

$$KU = F$$

3.3.3.5.3 Nonlinear Dynamic Analysis Models

From the detailed finite element models described earlier and using the super element feature of the ANSYS computer program, equivalent three-dimensional models of the rack were generated. The super element, which is a mathematical matrix representation of the rack structure, is generated by running a super element generation pass in which the desired degrees of freedom associated with a super element were specified. The super element consists of five lumped cantilever beams supported on a rigid base, which, in turn, interfaces with the ground at each support foot location. Each cantilever beam represents the tributary storage cells in its vicinity.

The masses of the rack and fuel cantilevers are connected by means of three-dimensional gap elements and hydrodynamic coupling elements. The support feet locations of the super element are connected to the floor by means of interface frictional elements. The stiffness characteristics of the interface element adequately account for the vertical stiffness of the storage rack, storage cell base plates and the reinforced concrete floor of the storage pool. The nonlinear three-dimensional model, therefore, adequately represents the effects of impacting, friction and hydrodynamic coupling.

Three conditions of fuel storage were analyzed full, half full and empty. The half full condition was included to account for the eccentricity of load distribution and the potential rotation of the rack.

Two extreme cases of coefficient of friction between the rack and the pool have been postulated, 0.2 and 0.8. The fully loaded storage rack condition has been analyzed for these two friction coefficients. The empty and partial rack conditions have been analyzed only for the low coefficient of friction. The lower coefficient gives the maximum sliding distances, while the higher coefficient gives maximum tipping, vertical fall back and shear forces transferred to the pool floor.

In each analysis, the rack was simultaneously subjected to two horizontal SSE time-histories. The effects of vertical acceleration were considered by conservatively applying the maximum vertical acceleration in the appropriate direction. The maximum vertical acceleration was taken from the vertical spectra at the fundamental vertical frequency of the rack.

3.3.3.5.4 Mathematical Formulation of Nonlinear Dynamic Analysis

Considering only translational degrees of freedom and assuming viscous (velocity proportional) damping, the equation of motion in matrix form can be expressed as follows:

$$M(\ddot{U}_t + \ddot{U}_{gt}) + C\dot{U}_t + KU_t = 0$$

where:

$\ddot{U}_t$ = Relative acceleration time-history vector

$\ddot{U}_{gt}$ = Ground acceleration time-history vector

M = Mass matrix

C = Damping matrix

K = Stiffness matrix

$\dot{U}_t$ = Velocity time-history vector

U_t = Relative displacement time-history vector

Rearranging:

$$M\ddot{U}_t + C\dot{U}_t + KU_t = -M\ddot{U}_{gt} = F_t$$

The above equations of motion are solved using step-by-step numerical integration methods. The numerical integration techniques used in the analysis are based on the modified Houbolt method. The straightforward numerical integration of the equation of motion has the advantage that nonlinear effects, such as variation in K, M or C due to closing or opening of gaps, sliding, large deflection and plasticity, can be readily included in the analysis.

In the numerical integration procedure:

$$U = f\ (U_{t-3},\ U_{t-2},\ U_{t-1}\ U_t)$$

$$\dot{U} = g\ (U_{t-3},\ U_{t-2},\ U_{t-1},\ U_t)$$

$$\ddot{U} = h\ (U_{t-3},\ U_{t-2},\ U_{t-1},\ U_t)$$

For the third order integration used in the analysis, f is a cubic function and the acceleration is a linear function across the time interval. The equation is solved at each time point in the dynamic transient. Since mass (M), damping (C) and stiffness (K) are recalculated at each time point, they can vary with time or can be functions of the displacement.

3.3.3.5.5 Fuel Assembly Impact Loads

Clearances are provided between fuel assemblies and the storage cells to avoid interferences during fuel storage and removal operations. The storage cell/fuel assembly clearance results in the impacting of the fuel assembly and storage cell during a seismic event. The fuel storage racks have been analyzed using the nonlinear time-history method of dynamic analysis, in which the effect of impacting masses has been directly accounted for by inclusion of gapped impact elements in the nonlinear model. Additionally, the effect of hydrodynamic coupling due to entrapped water is included by employing proper elements, as described earlier.

3.3.3.5.6 Dropped Fuel Assembly Analysis

Linear and nonlinear analysis techniques using energy balance methods were used to evaluate the structural response resulting from impact on the rack due to dropped objects or tornado missiles.

3.3.3.5.7 Water Sloshing Analysis

Water sloshing analysis has been performed using the analytical methods given in TID-7025, "Nuclear Reactors and Earthquakes," prepared by Lockheed Aircraft Company and Holmes and Narver, Inc., for the Atomic Energy Commission, Washington, D.C., (Reference 14).

3.3.3.6 Analysis of Existing Structures

3.3.3.6.1 Global Effects

The global effects of the added mass associated with the increased fuel load are considered negligible compared to the mass of the structure at this elevation and were therefore, not investigated in detail.

3.3.3.6.2 Fuel Pool Structural Analysis

The spent fuel pool floor and walls were analyzed to determine the combined rack and fuel load (in terms of kips per square foot) that the fuel

pool structure could accommodate and remain within the acceptance criteria specified in Section 2.3.3. This capacity analysis is based on a concrete strength of 6,400 psi at and below the floor slab level.

The data from these tests are summarized in Table 2-2. Above the fuel pool floor level the design compressive strength value of 4,000 psi was used. A design yield strength value of 40 ksi was used for all reinforcing steel.

The basic approach in determining fuel rack load capacity was to isolate the controlling element in the load path and then determine the load capacity (in terms of ksf fuel rack floor load) with respect to this element. The strength design method for reinforced concrete was used in conjunction with conventional structural analysis procedures to determine capacities. Section strengths were determined using the methods and procedures contained in Reference 31. The allowable limit loads were converted to actual rack loads using the load equations contained in Reference 9.

The impact loading associated with predicted rocking motion of the racks was included as part of the seismic loading conditions. Determination of impact energy of the fuel racks was based on maximum predicted gaps between the fuel rack support legs and the supporting surface of the fuel pool floor. Gap data were obtained from Table 3-8, which considered an equivalent fuel rack load of 2.87 ksf. This data was a direct product of the NES Report identified as Reference 34. The impact energy applied to the slab was determined using vectorial partitioning of the total racking energy with respect to normal impact on the slab (impact energy based on velocity component normal to the slab).

The kinetic energy transmitted into the slab during impact and the resulting structural response was determined using the methods and procedures contained in References 31, 32, and 33.

To account for the nonconcurrent aspects of the impacts from different racks (Reference 34), the slab kinetic energies from each impact were combined using the square-root-sum-of-squares (SRSS) method to determine the maximum concurrent kinetic energy of the slab.

Kinetic energy was then equated to strain energy to determine the maximum structured response. This enabled determination of the differential uniform load capacity required for impact effects.

The fuel rack load capacity was then determined from the controlling load equation considering the fuel rack dead and seismic (with impact) loads to be proportional to the mass of the fuel racks.

3.3.3.6.3 Liner Plate Assessment

The increased fuel rack load increases the load transferred to the liner plate by sliding frictions. This load is transferred through the plate and into the concrete (by friction) with essentially no increase in membrane stresses in the plate (steel-to-steel friction factor for rack-to-plate sliding is less than the steel-to-concrete friction factor for plate-to-concrete). Hydrostatic pressure further increases the force-to-resistance margin by additional steel-to-concrete friction resistance. The design for other loadings considered in previous analyses (such as Reference 13) remains essentially unchanged and requires no specific investigation with respect to increased rack loads.

3.3.3.7 Results of Analysis

3.3.3.7.1 Rack Structural/Stress Analysis

The results of the rack structural/stress analysis, which included rack and fuel assembly impact, are summarized in Table 3-4. Table 3-4 presents the maximum stresses in the critical rack structural member for the various load combinations developed in accordance with the NRC Standard Review Plan, Section 3.8.4 and compares them with the allowable values as specified in the acceptance criteria of Section 3.3.3.4.2. It can be seen that the maximum stresses in various structural members of the rack are nominal and within the allowable limits. Partially and fully loaded rack as well as empty rack conditions were analyzed. The stresses due to partially loaded and empty rack conditions are smaller than that for the fully loaded condition. Therefore, only the results for the fully loaded condition have been presented.

3.3.3.7.2 Spent Fuel Pool Floor Loads

The maximum reaction loads transmitted to the pool floor resulting from the deadweight, live loads, thermal effects and seismic loadings are presented in Table 3-5. The method used in the analysis of the existing structure is discussed in Section 3.3.3.6.

3.3.3.7.3 Water Sloshing Effects

Detailed calculations evaluating the effects of sloshing water on the storage racks indicate that the sloshing water mass will exert small convective forces on the storage racks. The maximum convective forces on the 20 x 18 storage racks resulting from the north/south and east/west safe shutdown earthquake are calculated to be less than 0.17 kips and 0.69 kips, respectively. These convective forces are significantly smaller than the impulsive water forces resulting from the effects of constrained water mass on the storage racks. The effects of the impulsive water forces have been considered in the seismic analysis by consideration of hydrodynamic mass effects. It has been concluded that the sloshing water mass will have insignificant effects on the canister storage racks.

3.3.3.7.4 Fuel Assembly Drop Analysis

The results of the fuel assembly drop analysis are presented in Table 3-6. For a vertical drop on top of the storage cell, the external kinetic energy will be absorbed by a series of springs, representing the local bending, crushing and tearing of the impacted area, bending of the top of the cell, local plate buckling of the impacted cells, elastic/inelastic deformation of the nonbuckled portion of the impacted cells and elastic deformation of the adjacent cells. Each spring will absorb the energy based on their load deformation characteristics and their proximity to the impact location. The results of the analysis indicate that there will be a total deformation of 1.24 inches if the impact is in such a fashion that there is tearing of the cell wall. For the case in which tearing is not initiated, the total cell deformation is 0.16 inches. Analysis indicates that the external kinetic energy will be absorbed in the local deformation of the impacted

cells. However, due to the honeycomb structural configurations, the overall integrity of the rack and the cell-to-cell pitch will not be adversely affected.

For the case of vertical impact at the top of the rack and subsequent tipping of the impacting fuel, the maximum kinetic energy per storage cell due to the tipping is less than the kinetic energy for the vertical impact alone. The resulting damage is therefore less severe.

From Table 3-6 it can be seen that for a drop through an empty storage cell, the cell base plate will not be perforated and the bearing stress on the concrete floor will be within the allowable bearing stress of 4000 psi concrete. Therefore, it can be concluded that for this drop case, there will be local damage to the cell base plate and the weld between the cell wall and base plate; however, because of the honeycomb type construction of the rack, the overall integrity of the rack and cell-to-cell pitch will not be adversely affected.

The drop of a fuel assembly on a storage cell already occupied by another fuel assembly is bounded by the case described above, as the kinetic energy will be absorbed by a combination of cushioning effects of the cell wall material and the stored fuel.

3.3.3.7.5 Fuel Storage Rack Stability Analysis

The stability evaluation of the fuel storage rack using the results of the nonlinear dynamic analysis indicate the following:

1. The storage racks will remain stable during OBE and SSE events. During the SSE event, the maximum displacement of the racks is 0.62 inches, including siding, tilting, flexure and rotation. Maximum sliding is 0.31 inches. These results, as summarized in Table 3-7, clearly preclude any impact between racks or between racks and pool walls.

2. For the SSE event, the maximum impact load that will be generated during recontact with the pool floor has been calculated to be 376.1 kips for the 20 x 18 rack. The maximum impact load will act as an impulse load on the pool floor. It should be noted that all racks will not recontact the pool floor simultaneously. The maximum reaction forces for OBE and SSE are summarized in Table 3-5.

3. A maximum reaction load of 139.7 kips will be developed in any single foot of the rack for an SSE event (Table 3-5).

4. Displacement as well as force time-histories for all three rack models have been stored on magnetic tape for evaluation of the existing structure. A summary of the rack displacements is shown in Table 3-8.

3.3.3.7.6 Existing Structure Analysis

3.3.3.7.6.1 Spent Fuel Pool Structure

The capacity limiting element in the load path was determined to be the fuel pool floor, in that all the supporting walls have more load capacity than can be transferred to them by the fuel pool floor slab (with all elements remaining within the design limits specified in Section 2.3.3). The capacity of the floor slab was determined to be 12.63 ksf (total uniform load) as limited by shear in accordance with Section 11 of Reference 31. This compares to a slab flexural capacity of 12.73 ksf (as limited by reinforcing steel) with moment redistribution in accordance with Section 8.4 of Reference 31.

The equivalent impact load associated with the 2.87 ksf rack load (used in Reference 34 to determine rocking rack displacements) was found to be 0.475 ksf. Considering the dead and seismic-plus-impact fuel rack loads to be proportional to the fuel rack mass, and using the controlling load combination (Number 2 of Section 3bi(b) of Reference 9; 1.4D + 1.7L + 1.9E), the allowable fuel rack load was determined to be 4.25 ksf.

3.3.3.7.6.2 Global Effects

The increase in rack load from 2 ksf to 4.25 ksf constitutes an increase in mass at this elevation in the structure of 1,734 kips. Compared with a mass of 12,000 kips for this elevation of the structure and a total structured mass of 130,000 kips (References 35 and 36), the increase in dead load and the corresponding seismic response would be on the order of 1-1/2 percent, which is within the design margin of the structure.

3.3.3.7.6.3 Liner Plate

The increased fuel rack load would not affect the integrity of the liner plate. The loads transmitted to the liner plate would be limited by the steel-to-steel friction factor (between the racks and the plate). Liner plate motion would be resisted by liner plate membrane stresses plus friction between the liner plate and the concrete. The steel-to-steel friction factor is lower than the steel-to-concrete friction factor; therefore, the liner plate would have more resistance to motion than the force applied through rack sliding friction considering rack-only loads. The resistance-to-load margin is further increased when considering the increased friction resistance afforded by the hydrostatic pressure forcing the plate against the concrete.

Other loading conditions accounted for in the previous design justification analyses (such as Reference 13) remain essentially unchanged and, therefore, require no further investigation.

3.3.3.8 Analysis of Installation Conditions

The seismic load stability of existing racks during installation was demonstrated during the previous reracking of the SFP[2,3] and no unstable configurations of existing racks will exist during the replacement sequence. The new fuel racks are individually stable since they are designed to be freestanding.

Procedures shall ensure adequate clearance between new racks and existing racks to preclude any impacting between racks during phased installation.

3.3.3.9 Conclusions

3.3.3.9.1 Fuel Racks

1. The results of the seismic and structural analysis indicate that the stresses in the rack structure resulting from the loadings associated with the normal and abnormal conditions are within allowable stress limits for Seismic Category I structures.

2. Sloshing of pool water in a seismic event will have insignificant effects on the fuel storage racks.

3. The analysis of the dropped fuel assembly indicates acceptable local structural damage to the storage cells with no buckling or collapse, and no adverse effects on the cell or cell pitch.

4. The nonlinear time-history analysis indicates that the rack will maintain its stability, and the space provided between racks and between racks and pool walls is adequate to preclude any impact during a seismic event.

5. It is concluded that the design of the Vermont Yankee spent fuel storage racks is adequate to withstand the loadings of normal and abnormal conditions.

3.3.3.9.2 Existing Structure

The existing fuel pool structure is capable of carrying fuel rack loads up to 4.25 ksf over the entire floor without exceeding the design requirements of Section 2.3.3 or compromising the integrity of the fuel pool liner plate.

The area beneath the new racks is the most densely loaded area of the fuel pool. The fully loaded rack dead load is less than 3 ksf, well below the 4.25 ksf floor capacity.

TABLE 3-1

Fuel Assembly, Control Rod, Fuel Channel and Storage Cell Parameters

Fuel Assembly:

Fuel type	Typical/Design Bundle*
Fuel enrichment, wt%	2.89/3.25
Fuel rod pellet OD, in.	0.411/0.411
Fuel rod cladding OD, in.	0.483/0.483
Zircaloy-2 cladding thickness, in.	0.032/0.032
Fuel rod pitch, in.	0.640/0.640
Number of fuel rods/assembly	62/64
Number of water rods/assembly	2/0
Numbers of grids per assembly (nominal)	7/0
Total assembly weight with channel (dry), lbs.	670/670
Control Rod Rack Weight (fully loaded), lbs.	6263

Channels:

Outside dimensions, in.	5.438
Zircaloy-4, channel thickness, in.	.080

Rack Cell:

Cell pitch, in.	6.218 in.
Cell ID, in.	5.922 in.
Cell wall thickness, mils	75
Neutron absorber material	Boral
Neutron absorber B^{10} loading, gms/cm^2	.027 (Min.)
Neutron absorber thickness, in.	.087 (Min.)
Narrow neutron absorber width, in.	4.375
Medium neutron absorber width, in.	4.875
Wide neutron absorber width, in.	5.375
Neutron absorber length, in.	146" (Min.)
Neutron absorber chamber width, in.	4.5," 5.0" and 5.5"
Cover sheet thickness, mils	45

* Conservative 1.35 k_{inf} bundle used for criticality analyses.

TABLE 3-2

Fuel Pool Cooling and Demineralizer System - System Specifications

System Function	System Specification
Total pool, well, and pit volume	81,500 ft^3
Fuel storage pool volume	41,600 ft^3
System design:	
Pump characteristics	450 gpm, 225 ft TDH, 25 ft NPSH
Heat exchanger - rated capacity*	2.23×10^6 Btu/hr, with 11.8°F LMTD
Filter-demineralizer	267 square ft, 450 gpm, 250 psi maximum p (dirty)

* Note - Actual capacity will increase with increased temperature differentials.

TABLE 3-3

Decay Heat Loads

Condition 1	Condition 2
18-Month Refueling Discharge	Full Core Discharge (Annual Cycle)
10 Days After Shutdown	10 Days After Shutdown
6.84×10^6 Btu/hr	15.63×10^6 Btu/hr

TABLE 3-4

Results of Structural/Seismic Analysis Member Stresses (KSI)

Load Combinations	Maximum Stress (in storage cell wall)	Allowable Stress
15 x 12 Rack		
1. $D + L$	3.07	16.7
2. $D + L + T_o$	3.07	16.7
3. $D + L + T_o + E$	11.59	16.7
4. $D + L + T_a + E$	11.59	16.7
5. $D + P_f$	2.96	22.2
6. $D + L + T_a + E'$	12.67	25.0
20 x 18 Rack		
1. $D + L$	4.09	16.7
2. $D + L + T_o$	4.09	16.7
3. $D + L + T_o + E$	10.50	16.7
4. $D + L + T_a + E$	10.50	25.0
5. $D + P_f$	2.96	22.2
6. $D + L + T_a + E'$	14.00	25.0
20 x 14 Rack		
1. $D + L$	5.55	16.7
2. $D + L + T_o$	5.55	16.7
3. $D + L + T_o + E$	10.61	16.7
4. $D + L + T_a + E$	10.61	25.0
5. $D + P_f$	2.96	22.2
6. $D + L + T_a + E'$	14.20	25.0

TABLE 3-5

Maximum Floor Load Summary

	Maximum Horizontal Reaction (K)		Dry Weight K	Maximum Vertical Reaction Load (K)		
	OBE	SSE		D + L[1]	D + L + OBE[2]	D + L + SSE[2]
15 x 12 Rack						
Per Rack Module	125.7	124.6	139	120.7	222.2	218.4
Per Support Pad	83.3	85.8		30.2	116.6	124.7
20 x 18 Rack						
Per Rack Module	208.9	211.1	277	239.1	282.4	376.1
Per Support Pad	71.0	107.5		40.3	103.3	137.8
20 x 14SP Rack						
Per Rack Module	180.2	221.1	243	210.3	294.8	304.4
Per Support Pad	70.0	101.9		54.6	104.4	139.7

(1) Buoyancy Force included.

(2) Buoyancy Force plus Zero Period Acceleration Force added to dynamic loads.

TABLE 3-6

Results of Fuel Drop Analysis

Straight Drop on Top of Storage Cell	Calculated Value	Allowable Value
Weight of Fuel (K)	0.670	--
Impact Velocity (ft./sec.)	8.03	--
Kinetic Energy of Impact (in. k)	8.04	--
Total Cell Deformation with Tearing of Cell Wall (in.)	1.24	--
Total Cell Deformation without Tearing of Cell Wall (in.)	0.16	--
Transmitted Reaction Load (K)	49.9	99.8*
Bearing Stress on Concrete Floor (ksi)	0.58	4.76
Straight Drop Through the Storage Cell		
Impact Velocity (ft./sec.)	31.07	--
Kinetic Energy of Impact (in. k)	120.6	--
Maximum Plate Thickness that may be Perforated (in.)	0.262	0.5
Transmitted Reaction Load (K)	106.9	--
Bearing Stress on Concrete Floor (ksi)	1.23	4.76

* Euler buckling load for one storage cell is 105.7K.

TABLE 3-7

Summary of Sliding Analysis Results

	Fully Loaded	Partially Loaded	Empty
15 x 12 Rack			
Maximum Rack Sliding Displacement[1](in.)	0.23	-	-
Maximum Rack Combined Displacement[2](in.)	0.62	-	-
20 x 18 Rack			
Maximum Rack Sliding Displacement (in.)	0.14	0.23	0.01
Maximum Rack Combined Displacement (in.)	0.35	-	-
20 x 14SP Rack			
Maximum Rack Sliding Displacement (in.)	0.11	0.31	-
Maximum Rack Combined Displacement (in.)	0.37	-	-
Rack to Wall Clearance[3](in.)	2"		
Rack to Rack Clearance (in.)	2"		

(1) Displacement relative to initial floor location.

(2) Maximum tipping and sliding are conservatively added even though they do not occur at the same time. Combined displacement includes sliding, tipping, flexure, and rotation.

(3) Nominal clearance at the base plate elevation.

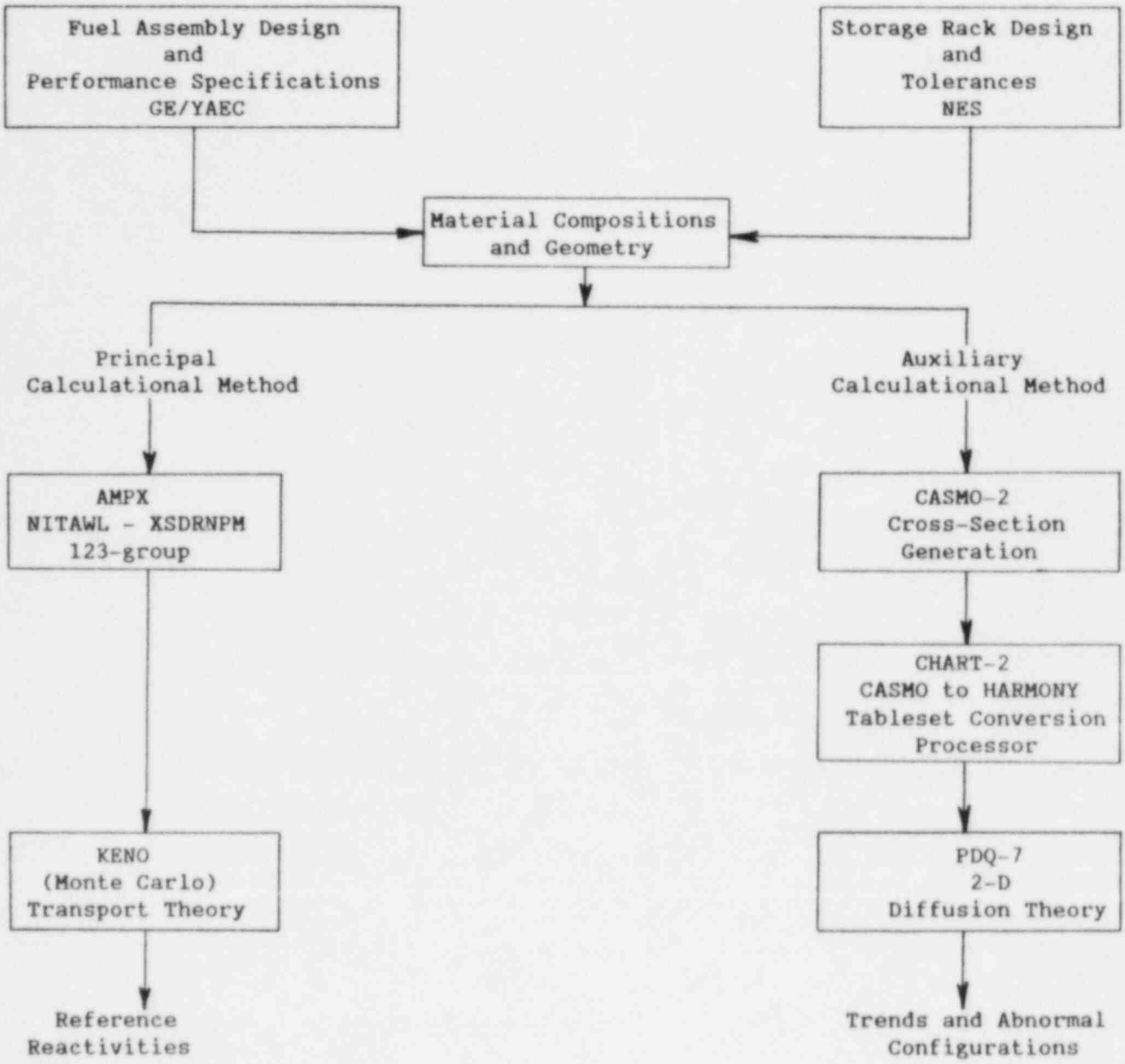
TABLE 3-8

Summary of Vertical Rack Displacement
During Seismic Events
(inches)

	Event	Rack 15 x 12	Rack 20 x 18	Rack 20 x 14 sp
OBE	Initial Displacement	-.00597	-.00958	-.00852
	Maximum Uplift	+.04023	+.00324	+.01713
SSE	Initial Displacement	-.00562	-.00902	-.00802
	Maximum Uplift	+.05215	+.02932	+.04895

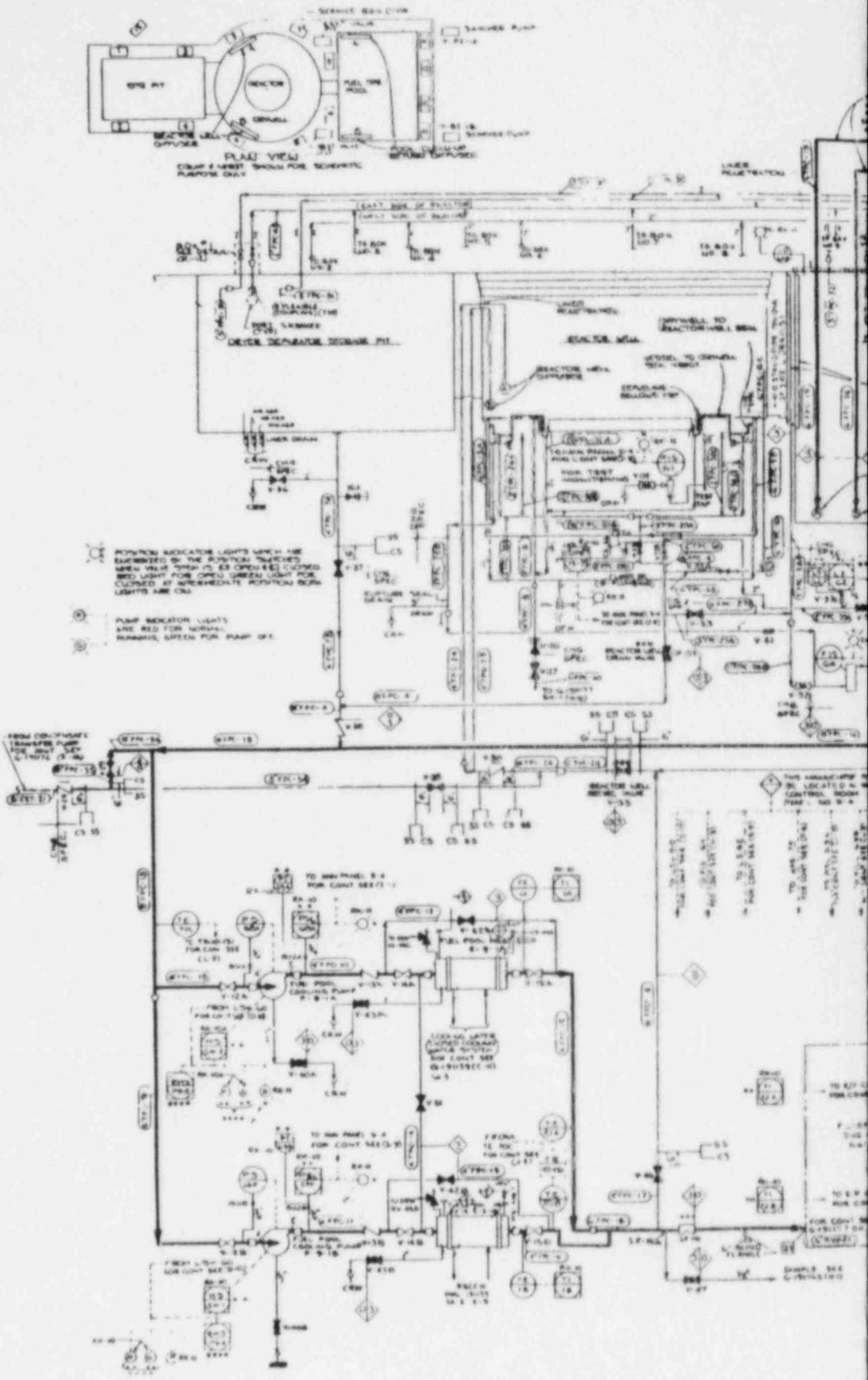
FIGURE 3-1

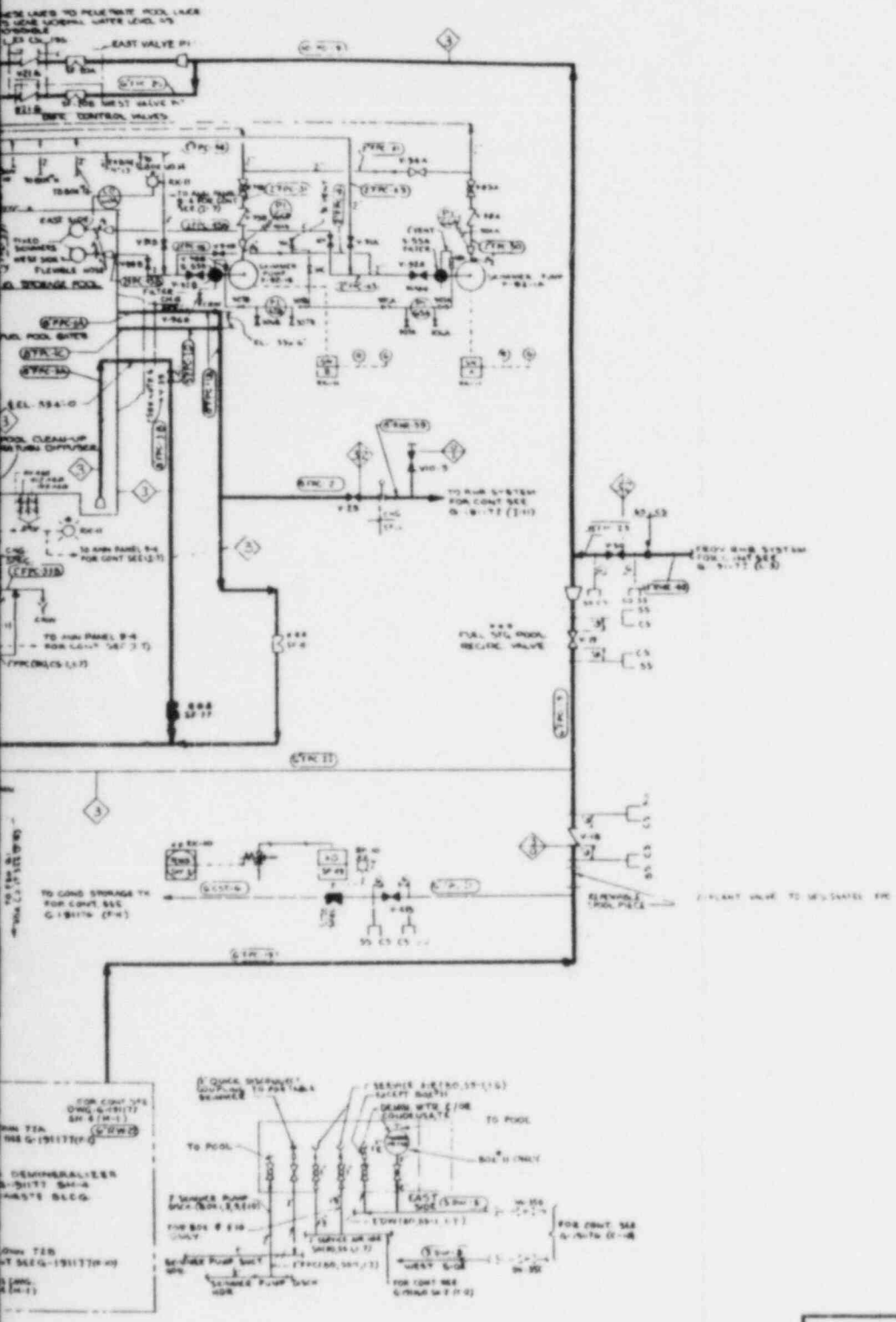
Spent Fuel Rack Criticality Analysis



Note: Benchmark (KENO) calculations have the following bias:

$$\text{bias} = 0.000 + [(.008)^2 + \Delta k^2_{calc}]^{1/2} \qquad \text{(Reference 19)}$$


PLAN VIEW
REACTOR WELL
DRYWELL
REACTOR WELL DIFFUSER
DRYWELL TO REACTOR WELL SEAL
VESSEL TO DRYWELL SEAL VENT
REACTOR WELL DIFFUSER
DRYWELL RELIEF VAP
POSITION INDICATOR LIGHTS WHICH ARE EXERCISED BY THE POSITION SWITCHES WHEN VALVE TRAVELS ON OPEN (RED) CLOSED (RED) LIGHT FOR OPEN GREEN LIGHT FOR CLOSED AT INTERMEDIATE POSITION BOTH LIGHTS ARE ON
PUMP INDICATOR LIGHTS ARE RED FOR NORMAL RUNNING, GREEN FOR PUMP OFF.
FUEL POOL COOLING PUMP
FUEL POOL HEAT EXCHANGER
COOLING WATER CLOSED COOLANT WATER SYSTEM



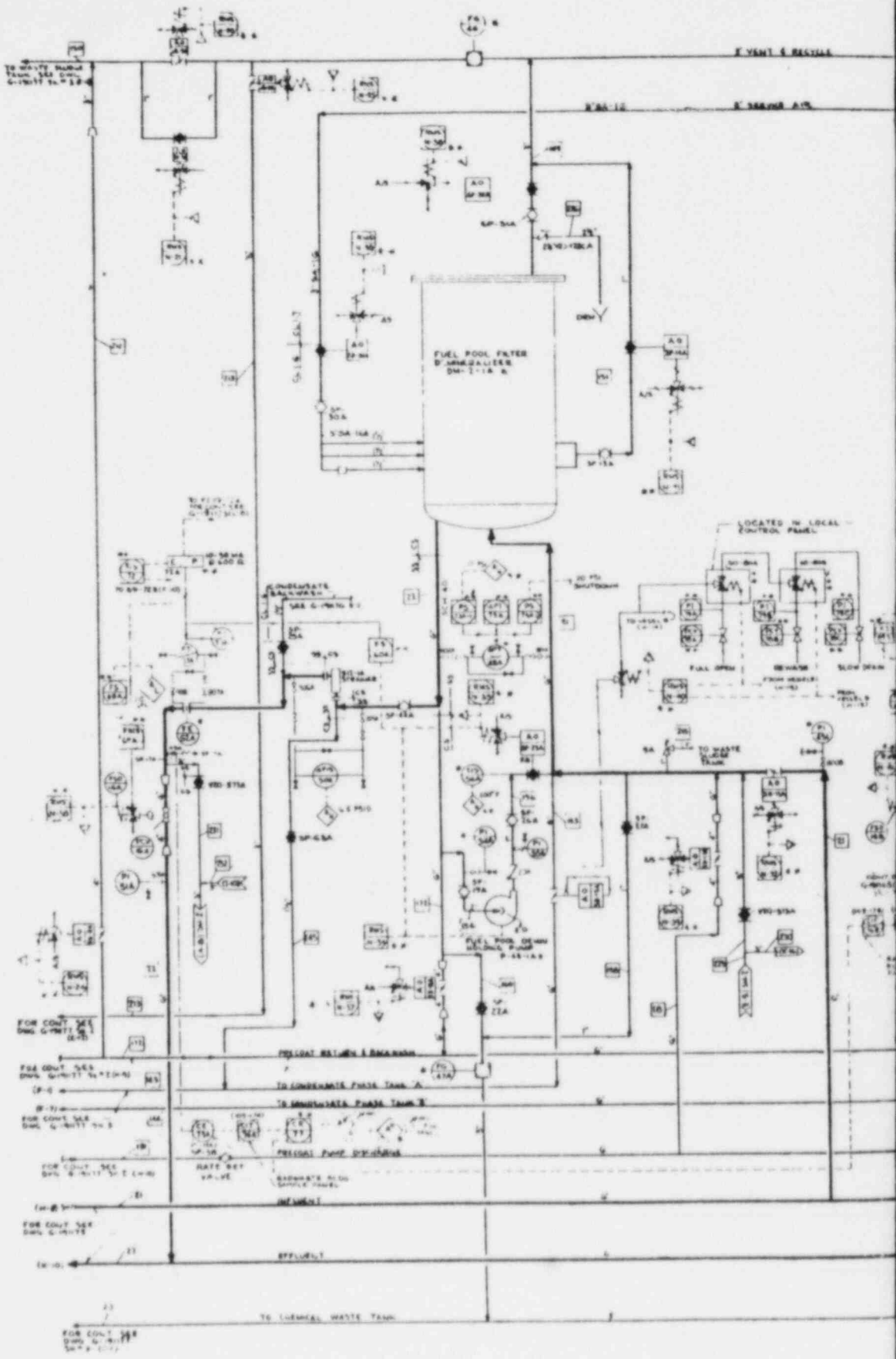
TYPICAL SERVICE BOX DIAGRAMS
(EXCEPTIONS AS NOTED)

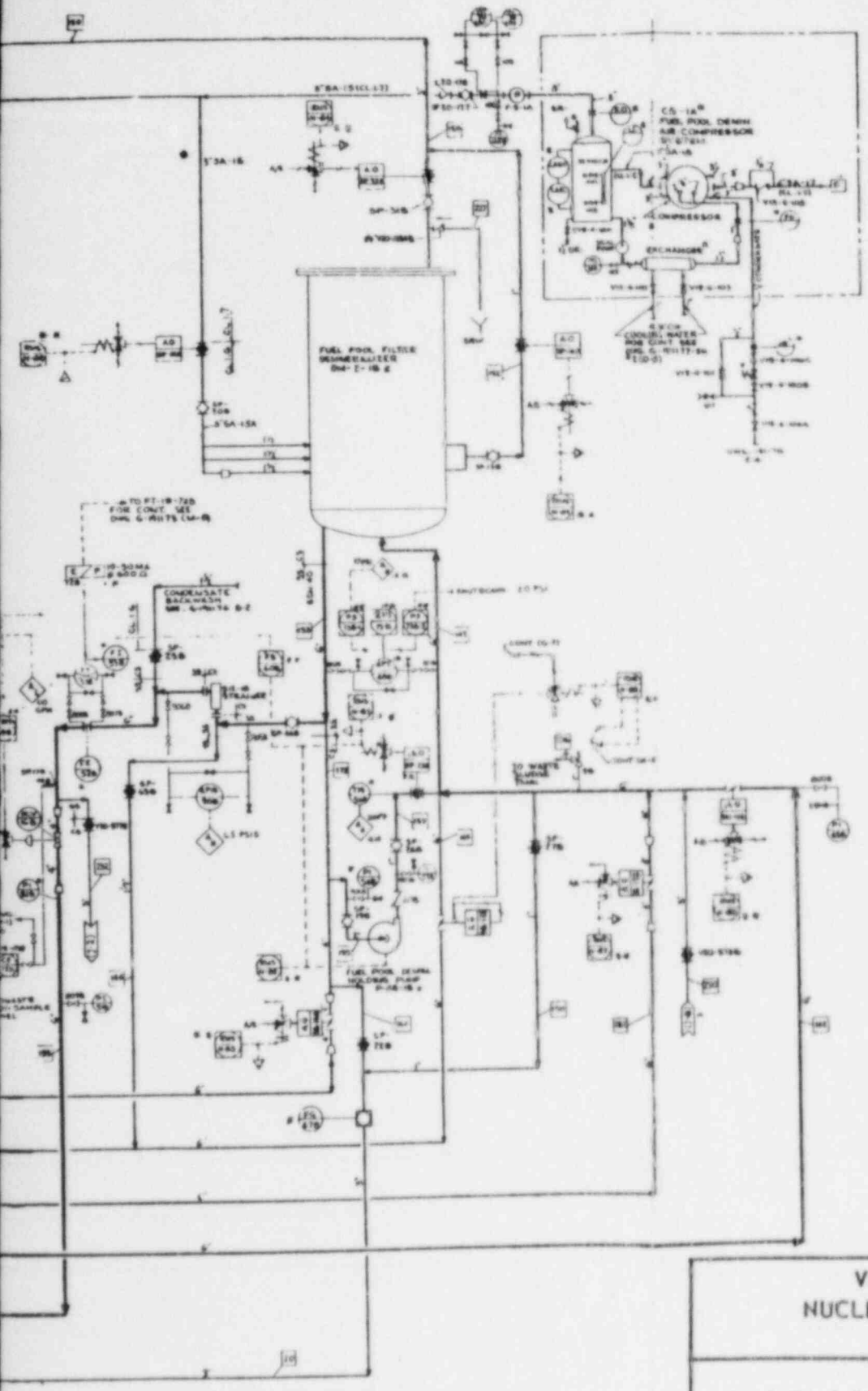
VERMONT YANKEE
NUCLEAR POWER STATION

Fuel Pool Cooling System

FIGURE 3-2

8605010052-01


FUEL POOL FILTER
LOCATED IN LOCAL CONTROL PANEL
TO CONDENSATE PHASE TANK A
TO CONDENSATE PHASE TANK B
INFLUENT
EFFLUENT
TO CHEMICAL WASTE TANK

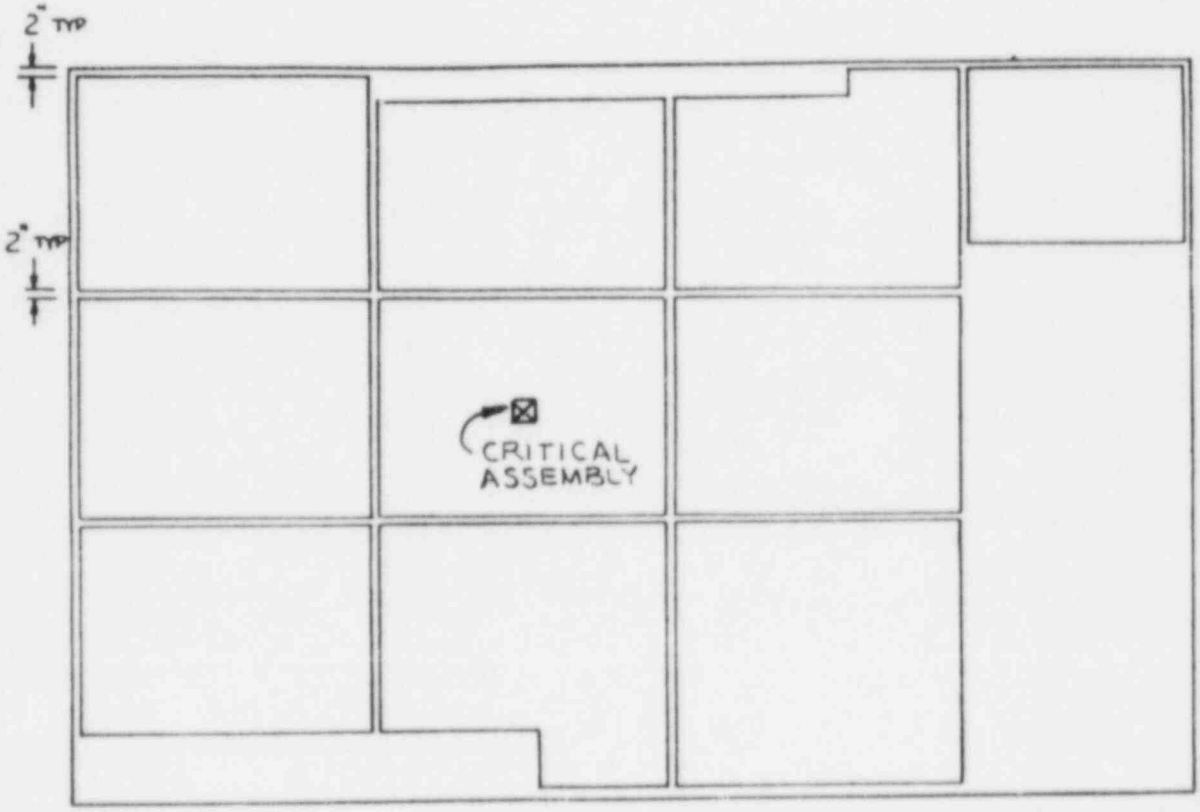


VERMONT YANKEE
NUCLEAR POWER STATION

Fuel Pool Filter Demineralizer System

FIGURE 3-3

8605010052-02



CRITICAL ROW / ASSEMBLY FOR
SPENT FUEL ASSEMBLY COOLING ANALYSIS
FIGURE 3-4

4.0 ENVIRONMENTAL ASSESSMENT

4.1 NONRADIOLOGICAL EFFECTS

4.1.1 Thermal Effects

The increased heat load to the SFP resulting from this proposed SFP expansion will occur when the total number of spent fuel assemblies stored in the SFP exceeds the currently licensed capacity of 2,000[3]. This heat load will be a maximum when a full core off-load of 368 assemblies with full discharge exposure is assumed to completely fill all available fuel storage locations in the SFP at 10 days after shutdown. For this condition, 15.6×10^6 Btu/hr is discharged to the SFP.

The Fuel Pool Cooling System transfers the spent fuel decay heat to the Reactor Building Closed Cooling Water (RBCCW) System through two heat exchangers. The RBCCW in turn is cooled by the station service water through two 100% capacity heat exchangers rated at 20×10^6 Btu/hr each. From there, the heat is discharged into the Connecticut River or to the mechanical cooling towers, or a combination of both. The total heat load from the Circulating Water System to the river or to the towers normally is in the range of 3×10^9 Btu/hr, which is of a magnitude that the amount of heat released to the environment by the increased storage in the spent fuel pool is insignificant.

4.1.2 Chemical Discharges

Increasing the spent fuel pool capacity will not change the chemistry of the pool. Normally the spent fuel pool water is removed for purification by the fuel pool filter-demineralizer and stored in the condensate storage tank for future reuse. Any discharge from the condensate is monitored before being discharged to the environment.

4.2 RADIOLOGICAL EVALUATION

4.2.1 Source Terms

4.2.1.1 Radioactivity in Spent Fuel

Increasing the amount of stored spent fuel assemblies from 2,000 to 2,870 as a result of future discharge batches of Vermont Yankee spent fuel will not appreciably increase the total spent fuel source term. Each newly discharged batch will continue to be the controlling source term while all previous batches will have undergone significant radioactive decay.

4.2.1.2 Radioactivity in Spent Fuel Pool Water

The following table identifies the principal radionuclides and their respective concentrations in the spent fuel pool water found by gamma isotopic analysis during a typical refueling operation. During refueling, the barrier which separates the spent fuel pool and the reactor cavity is removed so that the spent fuel pool water is at that time essentially the same as the reactor cavity water. The data presented is, therefore, considered to be representative of the maximum radionuclide concentrations in the spent fuel pool.

Detectable Concentrations of Radionuclides in the Spent Fuel Pool at Vermont Yankee Power Station

Isotope	uCi/ml Soluble	uCi/ml Insoluble
Mn-54	3.1 E-05	1.4 E-05
Co-58	7.2 E-05	1.0 E-05
Fe-59	---	4.6 E-06
Co-60	1.3 E-04	6.5 E-05
Zn-65	---	3.7 E-05
Zr-95	---	3.0 E-06
W-187	3.9 E-05	---

Vermont Yankee's SFPCDS has the ability to handle the contaminants produced by the present inventory of stored assemblies, including those with defective fuel rods. The proposed higher density spent fuel storage racks will only increase the capacity of the spent fuel pool and not the frequency or amount of newly discharged fuel to be stored per fuel cycle. The SFPCDS has been determined to be capable of handling greater than expected levels of defective fuel without causing unacceptable dose rates in the vicinity of SFPCDS components.

Further, the Vermont Yankee Liquid Radwaste System is able to handle any potential leakage from the SFPCDS. The Liquid Radwaste System coupled with the Radioactive Effluent Monitoring System and sampling procedures, ensures that any potential releases from the SFP during normal operation will not exceed the limits specified in the Vermont Yankee Operating License.

4.2.2 Environmental Releases

4.2.2.1 Gaseous Releases

Under normal conditions, levels of airborne radionuclides in the SFP area are below detectable levels. It is expected that increases, if any, in airborne activity due to increasing the number of stored assemblies, will be less than minimum detection levels both in the SFP area and at the site boundary.

4.2.2.2 Liquid Releases

The Vermont Yankee Nuclear Power Station does not routinely release liquid effluents. The SFP expansion proposed herein will not change the operational practice of filtering and demineralizing process water for recycle.

4.2.2.3 Solid Wastes

The amount of solid wastes generated by the SFPCDS in a five-year period was less than 330 cubic feet. The volume of the filter-demineralizer

in the SFP Purification System is 5.5 cubic feet. This filter-demineralizer is replaced each month and the basis for replacement is typically chemical exhaustion of the demineralizer resin. Since the basis for replacement of the filter-demineralizer is chemical exhaustion, it is expected that there will be no significant increase in the amount of solid wastes from the facility due to the SFP modification.

4.2.3 Radiation Doses

4.2.3.1 Occupational Exposures

The dose rate above the spent fuel pool at Vermont Yankee is typically on the order of 1 to 3 mrem/hr. This dose rate is attributed to radionuclides in the pool water, not to direct radiation from the spent fuel assemblies. The activity from the pool water is governed by the operation of the two spent fuel pool filter-demineralizers. Since the basis for changeout of the filter-demineralizers is chemical exhaustion, it is not expected that there will be any significant increase in replacement frequency, fuel pool activity concentration, or dose rates in the fuel pool area as a result of this proposed SFP expansion. The replacement of the filter-demineralizer resins for the SFPCDS at Vermont Yankee is accomplished by remote control and, therefore, doses associated with the routine monthly changeouts are minimal.

4.2.3.1.1 Rack Installation

A detailed exposure estimate for the rack replacement work will be developed when the work definition is explicitly determined. The use of divers is planned only for the removal of the sparger lines discussed in Section 3.2.1.1. The rack replacement project is not expected to present any difficult health physics problems. Certainly, the experience Vermont Yankee gained in the first rack replacement, as well as the extensive spent fuel pool work at other Yankee plants, will be of benefit in maintaining exposures from the rack replacement work as low as reasonably achievable.

4.2.3.2 Off-Site Exposures

As discussed in Section 4.2.2.1, increasing the storage capacity of the SFP from 2,000 to 2,870 assemblies will not measurably increase gaseous effluent releases (and, therefore, off-site doses) from the SFP.

4.2.4 Accident Analysis

The design basis fuel handling accident involves damage to one recently discharged assembly. Adding additional storage capacity will not impact the basis. Therefore, the design basis spent fuel handling accident, as outlined in Vermont Yankee FSAR[11] Section 14.6.4, will not be affected by an increase in the amount of spent fuel stored in the SFP.

5.0 INSTALLATION AND OPERATION

5.1 INSTALLATION PROCEDURES

The new racks will be installed in phases and replacement of the existing racks will be in accordance with written procedures to ensure that no rack module will be moved over spent fuel assemblies. Further, the area to be used for a spent fuel cask will be unchanged by these modifications, and as before, the spent fuel cask will be moved without passing over spent fuel or fuel racks. Thus, there are no new considerations associated with the proposed reracking effort which would alter NRC's 1977 Safety Evaluation (Reference 30) conclusion that, "... the Overhead Crane Handling System provided for moving shielded casks in the area of the SFP is provided with a sufficiently high degree of redundancy that the probability of a cask and/or heavy load handling accident which can damage the pool water-tight integrity is small enough to preclude consideration of the event."

5.1.1 Removal of Existing Racks

Prior to removal of the existing racks, the spent fuel will be shifted as necessary from the racks being removed to facilitate rack replacement. The existing spent fuel storage rack lifting rig or equivalent and the Reactor Building bridge crane will be used to remove the existing racks. The removal of existing racks will be proceduralized in such a way as to minimize fuel moves and eliminate any lifting of existing racks over spent fuel storage racks containing fuel assemblies. The SFP cooling sparger lines and associated supports will be shortened or removed from the pool.

5.1.2 Installation of New Racks

The installation of the new racks shall be accomplished using Vermont Yankee approved procedures by trained personnel. These procedures shall preclude carrying the new racks over any new or existing racks which contain fuel assemblies. The procedure shall also ensure that there is the required minimum spacing between new racks in their final configuration and between new and existing racks during the phased installation sequence.

5.1.3 Special Tools and Procedures

5.1.3.1 Special Tools

Lifting and handling rigs for new racks, including a rack up-ender, will be designed by NES and fabricated by the rack manufacturer. The lifting rig for the existing racks was designed by Programmed and Remote Systems Corporation (manufacturer of the existing racks).

The remote leveling tool for the racks has been designed by NES and supplied with the racks for use in adjusting the rack feet from above the spent fuel pool surface.

Vermont Yankee will develop and implement special remote handling tools, where practical, to facilitate removal of the existing racks and sparger line and to minimize exposure to personnel involved with the modification.

5.1.3.2 Special Procedures

Vermont Yankee will develop and implement special procedures to assure that racks and other heavy loads are not carried over stored spent fuel. Vermont Yankee will prepare specific procedures, where necessary, to minimize exposure to divers involved with the modifications.

5.2 DISPOSITION OF EXISTING RACKS

The existing racks shall be removed from the SFP and stored in the separator-dryer pit for initial decontamination. The racks will be packaged and stored on-site until disposal at an available storage site.

5.3 TESTS AND INSERVICE SURVEILLANCE

5.3.1 Neutron Absorber Verification

5.3.1.1 Fabrication

Each neutron absorbing sheet shall be visually inspected by the Fabricator immediately before incorporation into a storage cell to assure that full size undamaged sheets are being used. In addition, the rack fabricator will be required to assure that no Boral misloading (incorrect width) occurs during fabrication of the cells. The inspection required by the fabricator shall be documented and shall verify that no damage to the material has occurred during shipping and that the sheets are free of seams, welds, oil and cracks.

The Boral manufacturer provides documentation with the material which states that a 95 percent probability exists at the 95 percent confidence level that a minimum boron loading of 0.027 grams B^{10}/cm^2 is present. The 95/95 confidence level is met for the minimum thickness allowable by manufacturing tolerance.

5.3.2 Rack Fabrication

The new high density spent fuel storage racks will be fabricated by a subcontractor selected by NES and reviewed by VY and YAEC.

5.3.2.1 Material Certifications

Certified material test reports and heat treatment reports shall be obtained for all materials used in the fabrication of the spent fuel storage racks. For bolts, washers and studs, a Certificate of Compliance shall be acceptable in lieu of certified material test reports.

Certified material test reports shall be obtained from the neutron absorber manufacturer upon delivery.

5.3.2.2 Rack Inspections

Component fit-up, rack assembly and final inspection at the fabrication facility will be performed by certified Quality Control personnel in accordance with the design criteria contained in Reference 25 and approved procedures.

Additionally, a final inspection of the racks and a verification of all neutron absorber sheets shall be done upon delivery at Vermont Yankee using approved procedures.

Weld examination methods shall be performed in accordance with written procedures which comply with the requirements of Section V of the ASME Code.

5.3.3 Drag Test

Each of the storage cells will be tested at the fabrication facility using an inspection mandrel (resembling a fuel assembly) and load cell to verify the drag force does not exceed 50 pounds.

6.0 REFERENCES

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