SUMMARY OF PROPOSED MODIFICATIONS TO THE SPENT FUEL STORAGE POOL ASSOCIATED WITH INCREASING STORAGE CAPACITY

FOR

NORTH ANNA POWER STATION

UNIT NOS. 1 AND 2

DOCKET NOS. 50-338
50-339

LICENSE NOS. NPF-4
SNM-1801

APRIL, 1978

352 286

VIRGINIA ELECTRIC AND POWER COMPANY

TABLE OF CONTENTS

352 287

PREFACE

The purpose of this document is to summarize the proposed changes to the spent fuel storage pool at the North Anna Power Station, Unit Nos. 1 and 2, associated with increasing the storage capacity of the pool. It is intended that the document fully describe and analyze the proposed modification, as well as describe the installed systems which may be affected by the modification. Operating experience and data are presented where appropriate to substantiate the conclusions made regarding systems performance and environmental impact.

The information contained herein is also intended to provide pertinent technical details associated with the modification and its environmental impact, as well as to identify the reasons for the change.

This document should provide the Nuclear Regulatory Commission Staff with sufficient information to review the proposed change in accordance with regulatory requirements.

1.0 INTRODUCTION

Virginia Electric and Power Company's North Anna Power Station, Unit No. 1 was issued Operating License No. NPF-4 on November 26, 1977. The unit will have a thermal generation rate of 2775 megawatts. The operating license for Unit No. 2 is expected in December of 1978, with commercial operation tentatively scheduled for March 1979. The initial fuel cycle for Unit No. 1 is presently scheduled to end in November 1979. After each fuel cycle, at an average design burn-up of 33,000 MWD/MTU, approximately one-third of the fuel elements are discharged permanently from the core and are stored in the spent fuel storage pool. The present storage capacity of the spent fuel pool is 400 fuel assemblies, or approximately 2 1/2 cores.

It is prudent engineering practice and the policy of the Virginia Electric and Power Company (Vepco) to reserve storage space in the spent fuel pool to receive an entire reactor core (157 fuel assemblies), i.e., full core off-load, should unloading of the core be necessary or desirable because of operational considerations. Virginia Electric and Power Company has a policy and commitment to its customers to supply reliable and economic electric service, which requires that its nuclear power stations be operated reliably and continuously. To ensure continued operation, the ability to discharge spent fuel must be maintained. This, together with the fact that alternative spent fuel storage, reprocessing, or permanent disposal facilities cannot assuredly be available to Vepco when first needed leads to the conclusion that an increase in the spent fuel storage capability is necessary to accommodate both subsequent spent fuel discharges and to maintain the entire core off-load capability.

Vepco has evaluated the available alternatives for ensuring that the above objectives are realized. The decision to install additional spent fuel storage capacity provides the most favorable solution considering commitment of resources and availability. It is intended to be a near term solution which will provide storage capacity until 1987, at which time it is planned to ship the spent fuel to an offsite storage facility and then to a reprocessing or permanent disposal facility subject to the implementation of the Department of Energy's (DOE) spent fuel management policy.

To accommodate both subsequent spent fuel discharges until 1987, and maintain the full core off-load capability, a modification is planned to increase the spent fuel storage capacity by installing new spent fuel storage racks with a reduced center-to-center spacing of the fuel assemblies while maintaining subcriticality under all conditions. The planned modification will result in a maximum storage capacity of 966 fuel assemblies.

352 289

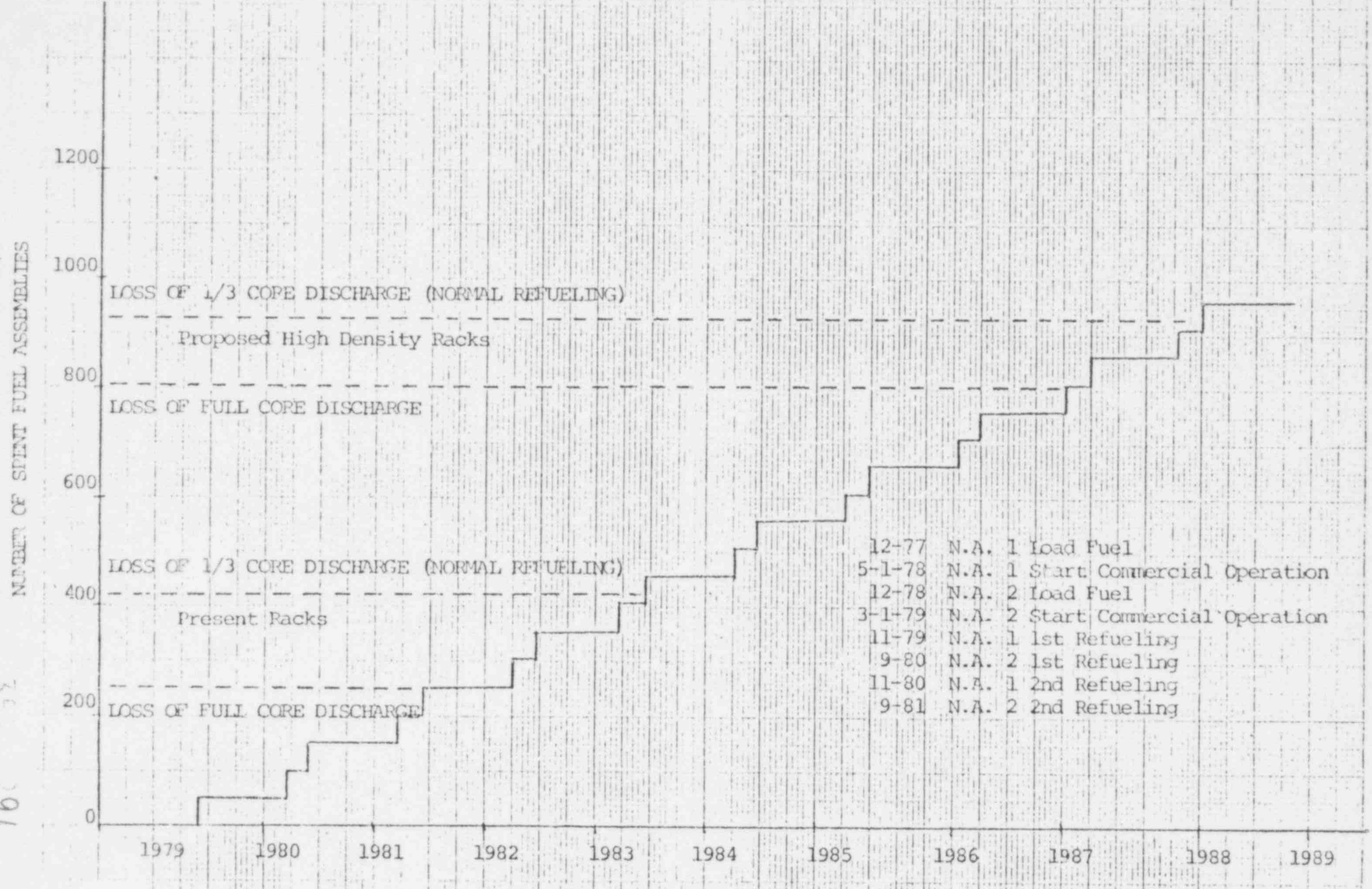
In summary, to avoid unnecessary unit shutdowns when an entire core off-load capability is not available for normal and emergency operating conditions, and to provide for additional spent fuel discharge storage capacity until 1987, given the uncertainty of alternative capabilities, an increase in the spent fuel storage capacity at the North Anna Power Station, Unit Nos. 1 and 2, is necessary.

Figure 1-1 diagrammatically shows the buildup of spent fuel assemblies in the spent fuel pool.

352 290

46 1513

FIGURE 1-1 BUILDUP OF SPENT FUEL ASSEMBLIES AT NORTH ANNA



392 291

2.0 REASONS FOR THE MODIFICATION

The basic reason for the planned modification is to maintain full core off-load capability and to provide additional onsite spent fuel storage capacity until such a time when adequate offsite interim storage, reprocessing, or disposal facilities are operational and available to the U.S. commercial nuclear power industry.

The existing design of the pool was predicated on being able to ship spent fuel offsite for processing after about 150 days resident time in the pool for decay of the short lived radioactive fission products. However, the President has recently suggested that commercial reprocessing of spent fuel will be indefinitely deferred until the international problems associated with the proliferation of nuclear weapons are resolved. Consequently, spent nuclear fuel will have to be disposed of as wastes or stored until reprocessing or an alternative use of spent fuel is implemented. The [illegible] has recently announced its spent fuel management policy whereby the federal government will take title to utilities' spent fuel and provide interim storage facilities by 1983 and radioactive waste disposal facilities by 1988. However, the capacity and location of these facilities have yet to be defined. Furthermore, a policy has yet to be implemented on a firm basis through the enaction of legislation and other necessary actions. In addition, considering the anticipated large accumulation of spent fuel inventory in the U.S. by 1983, it is not known when Vepco would be able to ship spent fuel to the facility once in operation. This condition necessitates that the majority of spent fuel discharged from operating reactors be stored until at least the late 1980's.

With the present number of fuel racks available at the North Anna fuel pool, Vepco anticipates loss of full core discharge in the Winter of 1981. The ability to discharge one third core for a normal refueling will be lost in the Fall of 1983. In order to avoid a problem with spent fuel storage capacity at North Anna Power Station Units 1 and 2 (which results from insufficient storage, reprocessing, or disposal facilities), it is desirable to replace the existing fuel racks with the high density fuel racks before any spent fuel is stored. This will minimize radiation exposure to installation personnel. Therefore, Vepco plans to complete the high density fuel rack installation as outlined in this report.

Therefore, the most prudent course of action to satisfy the necessity for storage of our spent fuel until the late 1980's is to complete the modification as described herein.

352 292

3.0 PROPOSED ACTION AND SCHEDULE

The Virginia Electric and Power Company has contracted with NUS Corporation to design, engineer, and manufacture the new spent fuel storage racks.

The following are key dates associated with the proposed modification:

Date	Description
July, 1976	Placed order with vendor
April, 1978	Submit report to NRC
October, 1978	Receive NRC approval
October, 1978	Commence rack installation
November, 1978	Complete rack installation

In summary, the additional storage capacity will provide VEPCO with additional operating flexibility which is desirable even if adequate offsite storage facilities hereafter become available.

352 293

4.0 ALTERNATIVES TO THE PROPOSED MODIFICATION

The proposed modification has been chosen after an evaluation of the possible alternatives to delay a possible shortage of spent fuel storage capacity. The following alternatives were considered:

1. Expand the spent fuel storage capacity at the nuclear station
2. Ship the spent fuel to reprocessors or commercial storage facilities.
3. Reduce the unit rating by operating them at less than 100 percent power
4. Shutdown the operating units when existing storage capacity is depleted
5. Build new storage pools
6. Ship fuel to Vepco's Surry Power Station
7. Store spent fuel at Department of Energy (DOE) Facilities
8. Ship fuel to other utilities
9. Physical Expansion of Existing Pool
10. Storage at North Anna Units 3 and 4

Each of the above alternatives is discussed below.

4.1 Increase Storage Capacity of the North Anna 1 and 2 Spent Fuel Pool

The storage capacity of the spent fuel pool can be increased by replacing the existing racks with racks of reduced center-to-center spacing resulting in an increased storage for a total storage of 966 fuel assemblies. This modification will require a minimum of physical changes to the pool structure. The racks will be manufactured offsite and will be shipped to the station for installing; thereby reducing disruption of normal operation.

The proposed modification will not alter the external physical geometry of the spent fuel pool or require additional modifications to the spent fuel pool cooling or purification system. The proposed modification does not affect in any manner the quantity of uranium fuel utilized in the reactor over the anticipated operating life of the facility. The rate of spent fuel generation and the total quantity of spent fuel generated during the anticipated operating lifetime of the station and stored in the spent fuel pool remains unchanged as a result of the proposed expansion. The modification will increase the

352 294

number of spent fuel assemblies stored in the spent fuel pool as well as th length of time that some of the fuel assemblies will be stored . he pool.

The approximate cost of the spent fuel racks is $2,600,000, exclusive of installation. The cost of installing the racks is estimated to be $100,000, which yields an estimated total cost of the modification to be $ 2,700,000. Based on the increased storage capacity of the spent fuel storage pool from 400 to 966 fuel assemblies, the approximate cost of the modification per added fuel assembly is $4,770. No additional operating costs will be incurred as a result of the modification.

Based on economic and operational considerations, as well as existing conditions, the expansion of spent fuel storage capacity of the existing pool provide the most feasible alternative to delay a possib shortage of spent fuel storage capacity. Therefore, this alternative was selected.

4.2 Shipment to Reprocessors or Commercial Storage Facilities

The shipment of spent fuel from North Anna Units 1 and 2 to a commercial fuel reprocessing or storage facility cannot be relied on as a viable alternative at this time.

On April 7, 1977, the President issued a statement outlining his policy on continued development of nuclear energy in the United States. The President stated that: "We will defer indefinitely the commercial reprocessing and recycling of the plutonium produced in the U.S. Nuclear Power Programs." As a direct result of the current policy on reprocessing and recycling as stated above, these services are not expected to be available to Vepco in the near future.

Presently, three private companies have unused storage space. Allied-General Nuclear Services (AGNS) has an unused storage space of approximately 400 MTU as spent fuel at its Barnwell Fuel Reprocessing Plant in Barnwell, South Carolina, and Nuclear Fuel Services (NFS) has an unused storage space of approximately 95 MTU as spent fuel at its plant in West Valley, New York. However, NFS is not willing to accept additional quantities of spent fuel for storage, and, under existing conditions, AGNS cannot be considered realistically to be a source of spent fuel storage capability at this time. Even if NFS and AGNS would receive spent fuel for storage, the associated shipments would be uneconomical because of additional handling and shipping costs. Assuming appropriate shipping vehicles are available, it is estimated that the cost of shipping the additional 566 assemblies, which could be stored at North Anna with the use of high density racks, to the Barnwell Plant would be in excess of $6,500,000 (in 1977 dollars) which is equivalent

352 295

to $11,485 per added fuel assembly. This cost is based on preliminary cost data with no firm contractual arrangements. Charges for storage at Barnwell and subsequent transportation from the AGNS plant to a potential Federal storage facility or another commercial facility would be additional.

The third company, the General Electric Company (G. E.), also cannot reasonably be expected to provide commercial storage. G. E.'s Midwest Fuel Recovery Plant near Morris, Illinois, is presently capable of storing approximately 700 MTU as spent fuel with an application before the NRC to expand this capacity to approximately 1,800 MTU. However, all of this space is considered by G.E. to be reserved for other utilities. Even if such storage space were made available to Vepco, additional handling and shipping costs make this alternative economically unacceptable. A similar estimate of the cost of shipping 550 assemblies from North Anna to Morris, Illinois, based on preliminary cost data with no firm contractual arrangements, results in costs exceeding $9,500,000 (in 1977 dollars) or $16,785 per added fuel assembly.

Therefore, these alternatives are unacceptable because of economic, operational, and availability considerations. However, should storage be required beyond the time allowed by using high density racks, any of these alternatives may be implemented.

4.3 Reduce Output of the Two Units

The amount of spent fuel to be shipped could be reduced by lowering the units' output, thereby extending the life of the fuel. The obvious discrepancy in this alternative is that the unit could not be operated to the extent possible and the amount of electrical power generated would be reduced. This alternative is not viable because it does not effectively use the resources available and would result in a significant increase in fuel costs to Vepco customers for replacement power from either Vepco owned facilities or electrical purchases from other utilities.

4.4 Shutdown of the Units

Assuming that the storage capacity of the pool remains the same and no offsite shipments are made, the units would have to be shutdown in late-1983. This is clearly not a viable or practical alternative. The generation provided by the nuclear units is necessary to supply customer requirements at the lowest cost possible. An economic evaluation would show that in a matter of several days, at approximately $250,000 per unit per day, the replacement cost of North Anna generation would exceed that of the proposed modification.

352 296

4.5 Build New Storage Pools

Additional storage capacity could be made available by building a new storage pool, either on or offsite. A detailed evaluation has not been performed because of the obvious large cost associated with such a facility. The current estimate is approximately $25,000,000 (in 1977 dollars) or about $22,007 per added fuel assembly. Another cost resulting from this alternative is the cost associated with double handling the fuel. Also such a facility would require four to six years to design, license, and construct; therefore, it does not satisfy our near term requirement of completing the modification before the first refueling of Unit 1 which would result in spent fuel being stored in the pool or our intermediate term requirement of avoiding a possible shutdown of the units in 1983.

This alternative is unacceptable because of economic, operational, and availability considerations. However, should the completion of the aforementioned DOE and commercial storage programs be significantly delayed, this alternative may constitute a possible long term solution.

4.6 Shipment of Spent Fuel to Surry Power Station

Unit 1 of Vepco's Surry Power Station, located approximately 110 miles (road mileage) SE of North Anna, has been in operation since 1972. Surry Units 1 & 2 will be equipped with high density racks, with a capacity of 1,044 spent fuel assemblies. Shipment of spent fuel between North Anna and Surry could be carried out; however, it has several distinct disadvantages.

Shipment of spent fuel from North Anna to Surry would be a very short-term solution. Assuming a full-core discharge capability is maintained and the spent fuel from North Anna 1 and 2 (in excess of its existing design capacity) and Surry 1 and 2 is stored in the Surry pool, the full-core discharge capability of the pool would be exceeded in 1983. If full-core discharge capability is not maintained, the pool's capacity would be exceeded in 1984. Thus, the storage capacity of the Surry pool would be exceeded before the expected availability date of DOE's aforementioned storage facility. Taking into consideration the critical need for these facilities at the time they become available, as well as the possibility of additional delays, it is not prudent to depend primarily on intra-system shipments to alleviate the North Anna storage problem.

Therefore, although it is a viable alternative in the very near term, intra-system shipment of spent fuel is not advantageous for the reasons outlined above. In the event of an emergency, intra-system shipments could be made.

352 297

4.7 Shipment to and Storage of Spent Fuel at DOE Facilities

As previously stated, the DOE's spent fuel management policy calls for the DOE to provide interim storage capacity to utilities. The DOE could do so by utilizing any of the three following alternatives: (1) present commercial storage and reprocessing facilities; (2) present domestic Federal facilities; and (3) new Federal facilities. However, the use of any of these facilities would involve substantial transportation and double handling costs thereby making this alternative economically unattractive at this time. In addition, it does not seem prudent at this time for Vepco to rely solely on the availability of DOE facilities because of the preliminary nature of the DOE spent fuel program.

4.8 Shipment to and Storage at Other Utility Storage Facilities

This alternative is mentioned only for completeness and is not considered to be a viable alternative. Because of the increasing lack of alternative storage, reprocessing, or permanent disposal capabilities, all utilities are faced with the same storage problem. Even if other utility pools were available, the economics of such shipments would be unfavorable. Again double handling would be required and would be similar to the other alternatives discussed.

4.9 Physical Expansion of Existing Pool

The alternative of physical expansion of the spent fuel pool would consist of removing one wall of the fuel pool and expanding the fuel pool in the direction of the removed wall. At present the fuel pool is bounded on four sides by existing structures necessary for the operation of Units 1 and 2 at North Anna. Those structures are on the north side the auxiliary building, on the west side the unit two containment, on the east side the unit one containment, and on the south side the decontamination building, the waste solidification area, the waste gas decay tanks and the primary water storage tanks. The work, time, and money involved including the moving of the structures on any side of the expansion of the fuel pool would be in excess of building a new fuel pool, which is discussed in Section 4.5. This would have to be done with no spent fuel in the pool and the power station would have to shut down during the construction.

4.10 Storage at North Anna Units 3 and 4

North Anna Units 3 and 4 are expected to be complete in the mid to late 1980's. This is too late to avert a loss of full core discharge capability in 1981 or a loss of refueling discharge capability in 1983. It is difficult to accelerate the completion of the fuel building because of its early stage of construction and because of dependence on the service wate. and component cooling water systems which will run throughout the facility. Licensing and construction problems make this solution unreliable at best.

4.11 Summary

Based on a review of the possible alternatives, the proposed modification is the best and has been selected to be implemented pending NRC approval.

5.0 EXISTING FACILITIES

The existing spent fuel storage pool is common to both units and has a total storage capacity of 400 fuel elements. Figures 5-1A and 5-1B show the arrangement of the building.

5.1 Fuel Building

The fuel building is a Class I seismic structure and is supported by a reinforced concrete mat on bedrock.

The spent fuel pool is a reinforced concrete structure with a 1/4 in. stainless steel (304) liner. The pool is designed for the underwater storage of spent fuel assemblies and control rod assemblies after their removal from the reactor. It is designed to accommodate a total of 400 fuel assemblies and a spent fuel shipping cask. These assemblies are stored in 25 fuel racks, each containing 16 fuel assemblies each. In the currently installed racks, fuel assemblies are placed in vertical cells, grouped in parallel rows having a minimum center-to-center distance of 21 in. in both directions. Restraining clips, which are welded to the floor embedment pads, prevent lateral motion of the spent fuel storage racks.

5.2 Fuel Pool Cooling and Purification System

The spent fuel pool is equipped with a spent fuel pool cooling system to remove decay heat and a purification system for maintaining fuel pool water quality. These systems are shown schematically in Figure 5-2.

5.2.1 Design Basis

The fuel pool cooling and purification system is designed to:

a. Remove the residual heat produced by one-third of an irradiated core 150 hr after reactor shutdown while maintaining the spent fuel pit water temperature at or below 140 F with one fuel pit cooler and associated pump with 105 F component cooling water (i.e., normal condition).

b. Remove the residual heat produced by one irradiated core 150 hr after shutdown and one-third irradiated core 45 days after shutdown, while maintaining the spent fuel pit water at a temperature of 170 F or less with one pump and two coolers with 113.2 F component cooling water (i.e., abnormal condition).

c. Remove soluble and particulate impurities from the water in the spent fuel pit,

352 300

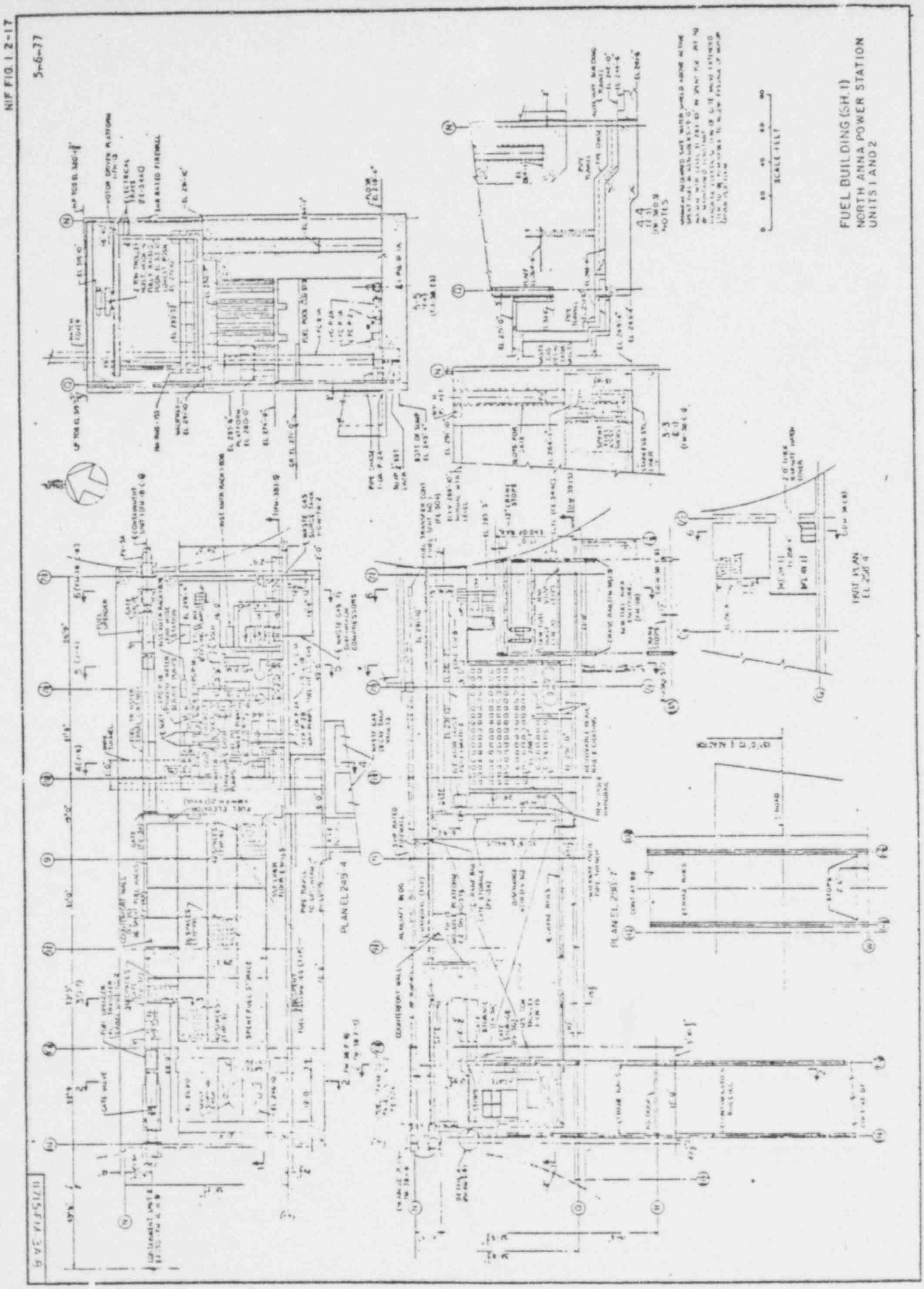


Figure 5-1A

352 301

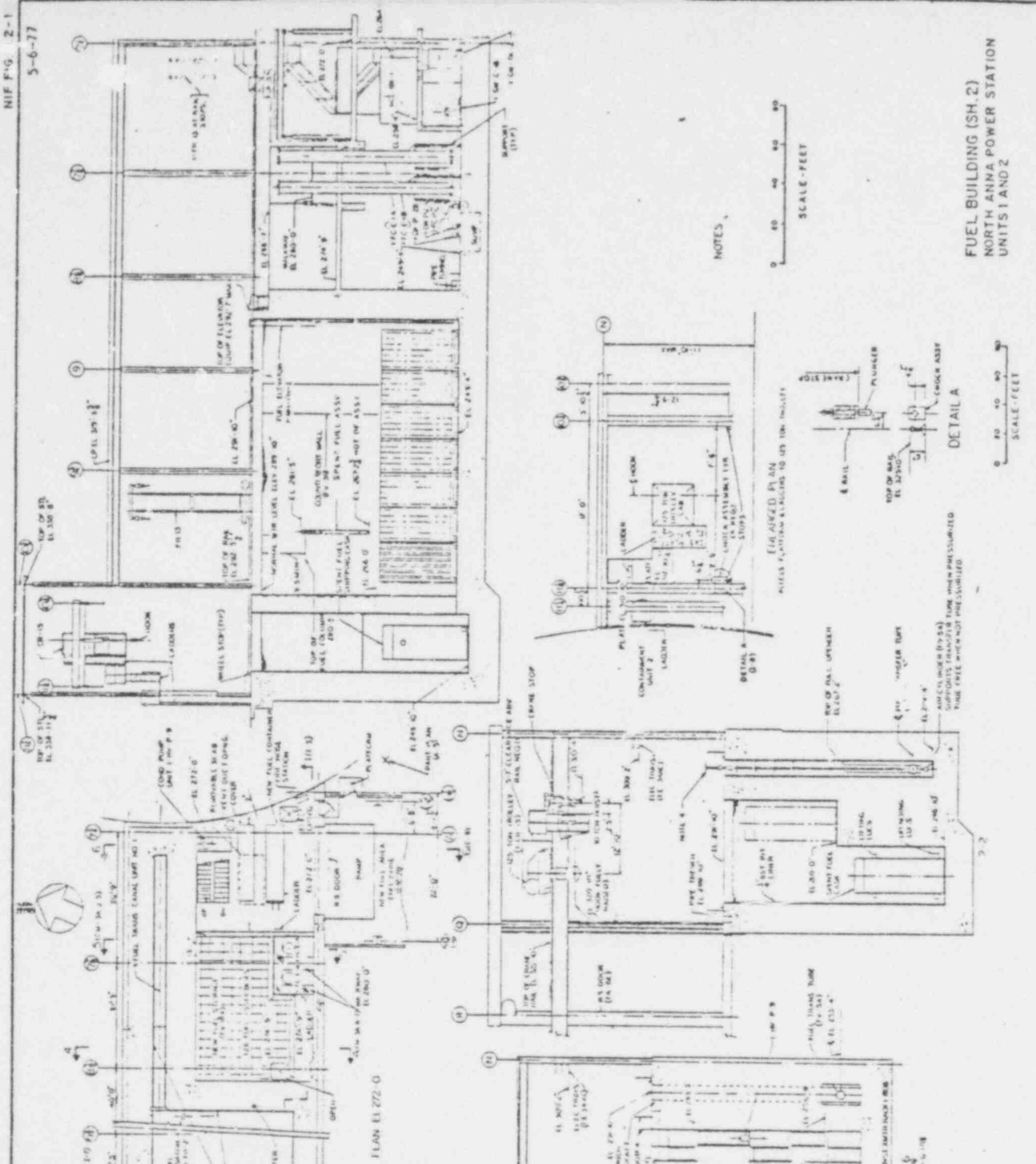


Figure 5-1B

POOR ORIGINAL

352 302

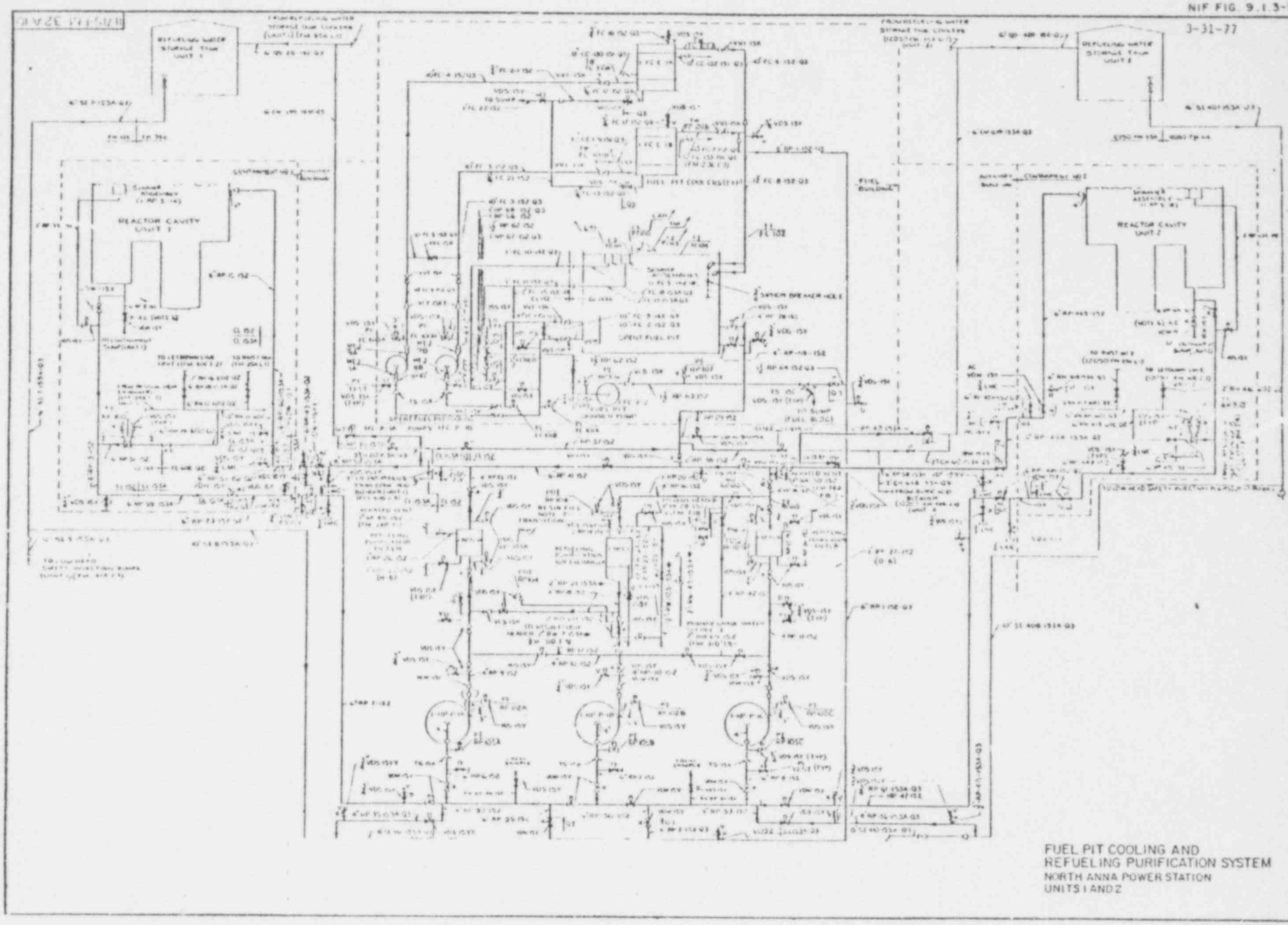


Figure 5-2

POOR ORIGINAL

352 303

either reactor refueling cavity, and either refueling water storage tank, to maintain the cavity water optically clear and radiation levels within acceptable limits.

5.2.2 Description

The fuel pool cooling and purification system has two shell and tube heat exchangers, two circulating pumps, and three 100 percent capacity purification pumps, all located in the fuel building. The heat exchangers and pumps are arranged for cross-connected operation. The heat exchangers are cooled with component cooling water with service water available as an emergency supply of cooling water.

The purification pumps take suction at the outlet of the fuel pool coolers and pump pool water to a demineralizer and filters located in the auxiliary building. The demineralizer or the filters can be bypassed if not required. The water returns to the fuel pool at the end of the pool opposite the suction point to assure mixing. The purification system is run independently of the cooling system whenever purification is required. The surface of the water is kept clear of floating matter by two permanently installed skimmers connected to the suction of the spent fuel pit cooling pumps. The fuel pool skimmers are also provided with a pump which allows the skimmers to be operated when the fuel pool cooling pumps are not in operation.

The spent fuel pool cooling and purification system is designed as a Class I seismic system. All piping, valves, and components which come in contact with the fuel pit water are austenitic stainless steel. Design data for components of the system are summarized in Table 5-1.

5.2.3 System Operation

The existing spent fuel pool cooling system is equipped with two circulating pumps and two fuel pool heat exchangers. Redundant piping is provided from the fuel pit through the pumps and coolers to the main return header located above pool water level.

The 400 gpm filtering rate of the purification system results in a clean-up half life of 27 hr and maintains suspended solids at a low concentration for optical clarity. The skimmer

TABLE 5-1

FUEL PIT COOLING SYSTEM COMPONENT DESIGN DATA

Fuel Pit Coolers

Number		2
Design duty, Btu/hr each (with tube inlet 210 F and shell inlet 105 F)		56,800,000
	Shell	Tube
Fluid flowing	Component cooling water or service water	Fuel pit water
Design pressure, psig	150	100
Design temperature, F	150	212
Operating pressure, psig	110	45
Material	Carbon steel	Stainless steel type 304
Design code	ASME VIII, Div. 1-1968	ASME VIII, Div. 1-1968

Spent Fuel Pit Pumps

Number	2
Type	Horizontal centrifugal
Motor horsepower, hp	100
Seals	Mechanical
Capacity, gpm	2,700
Head at rated capacity, ft	80
Design pressure, psig	125
Design temperature, F	250

TABLE 5-1 (Cont'd)

Materials	
Pump casing	Stainless steel type 316
Shaft	Stainless steel type 316
Impeller	Stainless steel type 316
Refueling Purification Pumps	
Number	3
Type	Vertical centrifugal
Motor horsepower, hp	20
Pump capacity, gpm	400
Seals	Mechanical
Head at rated capacity, ft	99
Design pressure, psig	185
Design temperature, F	200
Materials	
Pump casing	Stainless steel type 316
Shaft	Stainless steel type 316
Impeller	Stainless steel type 316
Refueling Purification Filter	
Number	2
Retention size, microns	3
Filter element capacity, gpm at 5 psid, normal/max.	400/440
Material	Stainless steel type 304
Design pressure, psig	150
Design temperature, F	250

TABLE 5-1 (Cont'd)

Refueling Purification Ion Exchanger

Number	1
Active volume, cu ft	45
Design pressure, psig	200
Design temperature, F	250
Demineralizer resin	50/50 cation-anion
Materials	Stainless steel type 316L
Design flow rate, gpm	200
Design code	ASME VIII, Div. 1-1968

Skimmer Assemblies

Number	4-2 in spent fuel pit, 1 in each reactor cavity
Debris basket	1/8 in. x 1/4 in. openings
Design temperature, F	210
Flow rate, gpm each	25 (min) to 55 (max)
Material	
Housing, base, and deck	High impact grade cycolac
Plate frame	(Acrylonitrile-Butadien-Styrene)
Debris basket	Polypropylene
Gasket	Buna-N
Screws and springs	Stainless steel 300 Series 18-8

Fuel Pit Cooling Piping and Valves

Materials	Austenitic stainless steel
Design code	ANSI B31.7-1969 and ANSI B31.1-1967

filter removes particles which fall and float on the water surface thus reducing the amount of impurities entering the water and reducing surface refraction.

5.3 Fuel Building Ventilation System

The fuel building is equipped with a ventilation system to provide high-efficiency filtration, heating to inhibit the buildup of condensation, and excess exhaust flow to maintain a negative pressure in the building to prevent outward leakage.

5.3.1 Description

The fuel building ventilation system has two supply fans, one to serve the spent fuel pit area and one for the remote equipment space at El. 249.33. Both take suction from a common plenum fitted with a combination roll and high-efficiency filter (95 percent atmospheric dust spot efficiency) and steam coils for air tempering and space heating. The exhaust fans discharge through the ventilation vent and are arranged for selective bypass through the auxiliary building filter bank. The area of the remote equipment room subject to radioactive contamination is exhausted by a branch from the decontamination building exhaust system.

The design provides (1) sufficient air at a temperature that will inhibit condensation on the overhead structure to avoid drippage into the pool, (2) high-efficiency supply air filtration to minimize dust clouding of the surface, and (3) supply air distribution to avoid ruffling the surface. The dual exhaust combined with two-speed supply fan arrangement provides step capacity control and protection against a single failure. The exhaust is continuously vented through the ventilation vent, with the capability to bypass through the auxiliary building iodine filter bank. The exhaust is filtered continuously during irradiated fuel-handling operations to prevent the spread of any possible airborne contamination through the exhaust air system.

The fuel building exhaust also discharges air entering the fuel building from the tunnel between the fuel building and the waste disposal building.

5.4 Instrumentation Applications

Instrumentation provided gives local indication in the fuel building and the auxiliary building and remote

indications and alarms in the main control room. Unit 1 control board indication and alarms include:

1. Fuel pit temperature indication

2. Spent fuel pit temperature at >140 F and >170 F

3. Spent fuel pit high/low water level with the low-level alarm 6 in. below normal water level (El. 289.33)

4. Start/stop switch for spent fuel pit cooling pumps with run indication on both Units 1 and 2 main control boards

5. High differential pressure alarm for the refueling purification filters

Local indications include various flows, temperatures, pressures, and differential pressures.

The system instrumentation, including the spent fuel pit level and temperature instrumentation, are calibrated on a periodic basis.

352 309

5.5 Operating Experience at Surry Power Station

Based on the operating experience VEPCO has received at Surry Power Station, Unit Nos. 1 and 2, similar results are expected at North Anna since both are of similar design. The information that follows has been extracted from information from Surry Power Station.

The operating history of Unit Nos. 1 and 2, Surry Power Station, has been reviewed in light of the fuel pool cooling and purification system, ventilation system, personnel exposures, etc. The purpose of this review was to confirm the satisfactory operation of the systems and to provide baseline data for estimating the effect the proposed modification may have. Unit Nos. 1 and 2 have operated for about 45,480 and 42,360 hr, respectively, and have generated a gross electrical output of approximately 43,179,352 megawatt-hours as of February 28, 1978.

The operating experience at Surry Power Station has been grouped into four general areas for purposes of discussion:

1. Performance of the spent fuel pool cooling and purification system

2. Environmental conditions in the fuel building

3. Ventilation system

4. Radiation exposure

5.5.1 Performance of Spent Fuel Pool Cooling and Purification System at Surry Power Station

Operating experience with the spent fuel pool cooling system has been excellent to date. As of March 1, 1977, 292 fuel assemblies were stored in the pool. The pool water temperature has been maintained at about 95 F all year round. This temperature has been maintained using only one train of cooling, i.e., one pump and one heat exchanger. During normal operation, both trains have not been operated simultaneously since one train maintains proper temperature. This system has performed satisfactorily.

The normal flow rate for the purification system is about 110 gpm and remains in operation continuously to maintain a clean and clear pool. The maximum allowable differential pressure across the filter is 15 psi. The maximum allowable differential pressure across the demineralizer is 25 psi. If the pressure drop across the filter or demineralizer exceeds the

352 310

allowable value, the filter is replaced or the resin is replenished, respectively.

The radiation levels at the demineralizer are usually from 1 R/hr to 4 R/hr. The filters are normally changed because of high pressure drop and usually have radiation levels of about 100 mR/hr. The filters are normally changed prior to each refueling, i.e., twice per year assuming two units are operating. In changing filters an individual receives an exposure of about 150 mR. In replacing the resins in the demineralizer, approximately 55 mR is received. This exposure is divided among three individuals.

5.5.2 Pool Environment Conditions

The spent fuel pool purification system removes both radioactive and nonradioactive particulates from the pool water. The purity of the pool water is normally maintained between 0 to 0.3 ppm, with a maximum particulate concentration of about 0.4 ppm. This purity level provides sufficient optical clarity for refueling operations.

Based on samples taken since station start-up, the following major isotopes have been detected in the pool water in the approximate concentrations indicated below.

Isotope	Concentration (micro Curies/ml) Normal	Maximum	Minimum
Cs-134	0	1.2×10^{-4}	0
Cs-137	10^{-4} to 10^{-5}	1.3×10^{-4}	0
Co-58	10^{-3} to 10^{-4}	1.5×10^{-3}	6.6×10^{-4}
Co-60	10^{-3} to 10^{-4}	1.1×10^{-3}	4.8×10^{-4}
I-131	0	6.5×10^{-5}	0
Gross Activity	10^{-3} to 10^{-5}	1.1×10^{-3}	5.0×10^{-5}

Crud buildup along the sides of the spent fuel pool has not significantly affected the radiation levels on the edge of the pool. Smear samples from the sides of the pool had the following activities:

352 311

Isotope	Concentration (micro Curies/cm²)
Cs-134	1.53×10^{-4}
Cs-137	5.60×10^{-10}
Co-58	1.47×10^{-3}
Co-60	2.54×10^{-3}

The crud buildup on the sides of the pool is removed with hydrogen peroxide (H2O2).

5.5.3 Ventilation System Experience at Surry

The ventilation system in the fuel building has maintained the levels of radioisotopes in the atmosphere in the building at acceptable concentrations. During normal station operation, i.e., refueling is not in progress, the gross activity above the pool water is about 1×10^{-11} to 1×10^{-10} micro Ci/ml. The principle isotopes noted are Co-58, Co-60, Cs-134, and Cs-137. During refueling operations, I-131 levels of 5×10^{-11} to 5×10^{-10} micro Ci/ml have been noted. Other isotopes, Co-58, Co-60, Cs-134, and Cs-137, have been noted in the concentration of 1×10^{-10} to 1×10^{-9} micro Ci/ml. Tritium H-3 and Kr-85 have not been detected in the fuel building. As stated, I-131 has only been detected in the fuel building during refueling operations. During refueling, the fuel building ventilation is directed through the auxiliary building charcoal filters, with a decontamination factor of about 100. The fuel building exhaust contributes about one-half of one percent (0.5%) of the total I-131 released from the ventilation vent during the refueling period.

5.5.4 Radiation Exposure at Surry

Based on 208 fuel assemblies stored in the fuel pool, individuals would receive the following exposures for the locations indicated:

Exposure (mR/hr)	Location
3.5 to 5.0	Six in. above surface of water
0.7 to 1.5	Waist level at the edge of the pool
0.5 to 1.0	Fuel bridge

352 312

Based on working in the fuel building for a 10-hr day, an individual would receive approximately 15 mR.

5.5.5 Radioactive Waste Experience at Surry

The spent fuel purification system generates certain radioactive waste which must be shipped off-site for disposal. When the filters are changed, they are placed in 55 gal drums for shipment as "low specific activity solid waste." Two 55 gal drums of solid radioactive waste (approximately 15 ft^3) are generated annually. Skimmer filter changes on the average produce an additional 15 ft^3 of solid waste annually. The spent fuel pool ion exchanger resin is replaced twice a year producing about 90 ft^3 of solid radioactive waste. Thus, the total radioactive waste generated annually by the spent fuel purification system is approximately 120 ft^3.

Approximately 60 ft^3 of solid radioactive waste is associated with spent fuel storage for each refueling. A typical refueling outage produces about 1,000 to 1,500 ft^3 of radioactive waste.

352 313

6.0 DESIGN OF NEW HIGH DENSITY SPENT FUEL STORAGE RACKS

6.1 Introduction

This section presents a detailed description of the modified fuel racks and comprehensive analyses of these racks which show their capability to safely store spent fuel for an extended period of time.

6.2 General Description

6.2.1 Present Design

The spent fuel pool currently has a capacity of 400 fuel assemblies of the Westinghouse 15 by 15 or 17 by 17 array design. The present storage racks consist of vertical cells on a center-to-center spacing of 21 in., which are fastened together to form 4 by 4 array racks. The racks are designed for peak site earthquake accelerations of .24g in the horizontal direction and .16g in the vertical direction. The spacing of fuel bundles in the spent fuel storage pool is designed to maintain keff less than 0.90, even under accident conditions.

6.2.2 High Density Storage Rack Design

6.2.2.1 Design Basis

The following design bases apply to the high density spent fuel storage rack design. Other design bases for the spent fuel storage facility remain unchanged.

a. Fuel storage space is provided in the spent fuel storage pool for the 966 fuel assemblies.

b. The center-to-center spacing between stored fuel assemblies in a fully loaded rack is sufficient to maintain a keff equal to or less than 0.95 for the normal wet condition and for all abnormal and accident conditions. This design basis is met even with fresh fuel of up to an equivalent 3.50 w/o U-235 enrichment (44.2 grams of U-235 per axial cm.), a conservative water temperature of 68 F, and no credit for either fixed poison in the fuel assembly or soluble boron in the fuel pool water (this design basis is discussed in Section 6.4).

c. The modified rack design precludes storage of a fuel assembly other than where intended, in the racks.

d. The modified racks are classified seismic Category I and are designed to withstand the effects of the Design Basis Earthquake (DBE) and yet remain functional and maintain subcriticality.

352 314

e. The modified racks are designed to withstand either a dropped fuel assembly or the upward force of a stuck assembly without loss of function (structural analysis is discussed in Section 6.5 and accident analysis in Section 6.7).

f. The racks are designed to allow adequate cooling of the stored spent fuel assemblies (this is discussed in Section 6.6).

6.2.2.2 Design Description

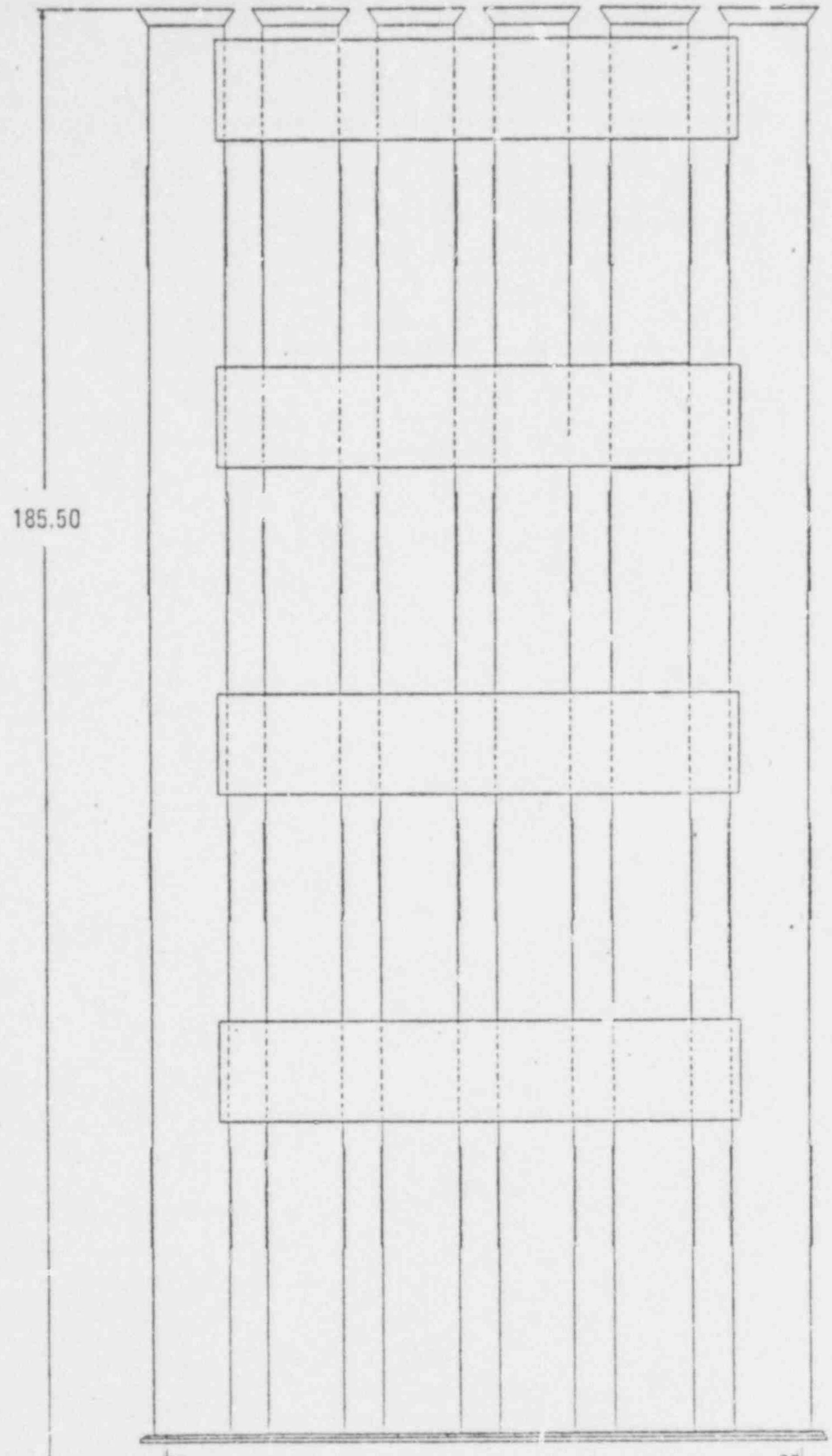
The high density spent fuel rack design utilizes a center-to-center spacing between storage cells of 14 in. to increase the storage capacity. Each storage location consists of an austenitic, Type 304, stainless steel square tube, which is separated from adjacent tubes and held firmly in place by steel plates welded to the sides of the tubes at several elevations. The tubes are grouped in this manner to form upper rack assemblies, which are welded to a pre-assembled welded base. The base consists of beams and plates with an opening at each storage location to receive the fuel assembly nozzle and allow cooling water to enter the assembly. The racks are free-standing but restrained from lateral motion under seismic excitation by restraints welded to embedment plates in the pool floor. The modified rack design and spent fuel pool layout of the revised racks are shown in Figures 6.3-1 and 6.3-2, respectively. The mechanical design is discussed more fully in Section 6.3.

6.3 Mechanical Design

6.3.1 Fuel Rack Description

The modified spent fuel storage racks, shown in Figure 6.3-1, consist of square stainless steel tubes of 1/8 in. thick Type 304 austenitic stainless steel. They are spaced at 14 in. center-to-center by Type 304 stainless steel plates. The plates, which are also 1/8 in. thick, are welded to the sides of the square storage tubes at four elevations. The tubes are flared at the top to permit easy storage and retrieval of the stored fuel assemblies, and to be compatible with the fuel handling equipment. Three different rack cell arrays are utilized to maximize use of the available fuel storage space in the pool. The upper rack structure is welded to an elevated base which is a system of welded beams and stiffeners. The base serves to support the weight of the fuel assemblies and to distribute the load on the pool floor. The base, which contains an opening at each fuel assembly storage location to permit coolant flow, accommodates the fuel assembly bottom nozzle. Natural circulation of pool water flows down between the storage tubes and up through the bottom nozzle and the fuel assembly to remove decay heat. The storage cells are designed to provide lateral support for stored assemblies of the Westinghouse 15 by 15 or 17 by 17 array design and other assemblies with the same external dimensions and similar lower nozzle design.

352 315



352 316

Figure 6.3-1 Rack Elevation

TABLE II			
RACK TYPE	CELLS / RACK	NUMBER	TOTAL
A (IT. 1)	36	18	648
B (IT. 2)	24	1	24
C (IT. 3)	42	7	294
			966

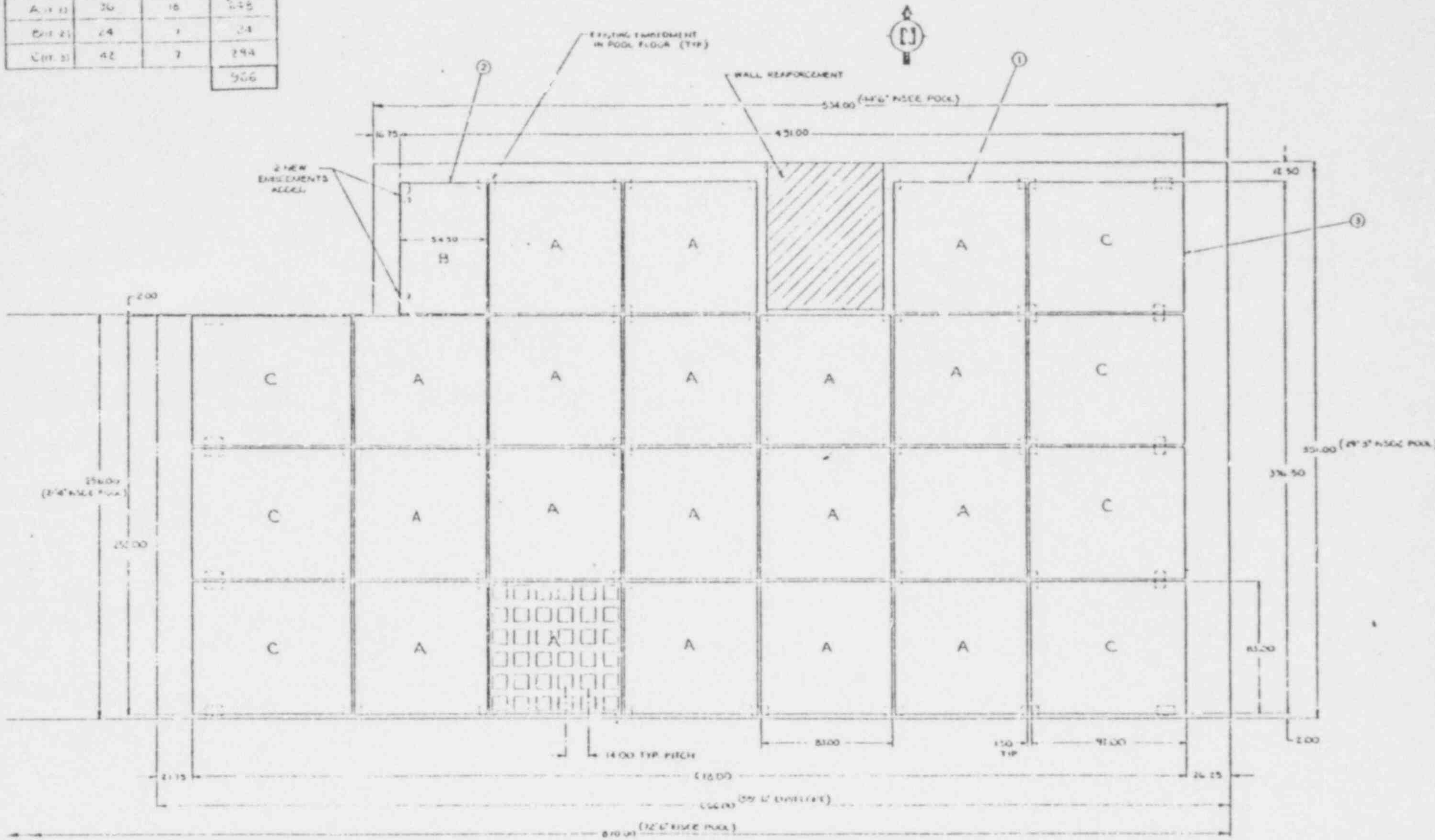


Figure 6.3-2 — SPENT FUEL POOL ARRANGEMENT

352 317

To prevent lateral movement of the racks, particularly from seismic excitation, the rack legs are restrained on their outer faces by right-angle plates which are welded to embedment plates in the spent fuel pool floor. The angle plates are positioned to allow for thermal expansion. The legs, which raise the racks off the floor to allow natural circulation flow, are square with gussets for reinforcement. Shims are used to level the racks during installation.

Each corner of the one 6 by 4 row storage rack has a floor bolt which restrains the rack tipping motion during seismic excitation. The bolts are positioned to allow the rack to slide horizontally and to be engaged only if a tipping motion occurs. During lateral movement, the rack legs will make contact with the aforementioned angle plates rather than engaging the bolts. All of the other racks are larger (6 by 6 or 6 by 7 rows) and do not require floor bolts.

The racks, including the base structure, the angle plates, embedment plates, and spent fuel pool liner are all stainless steel. Thus, galvanic corrosion is not a problem. Stainless steel has also been shown to be compatible with spent fuel pool water and the stored assemblies.

6.3.2 Modified Fuel Rack Codes and Standards

The design and fabrication of the high density spent fuel storage racks will be performed in accordance with applicable portions of the following codes and standards.

6.3.2.1 Design Codes

a. AISC Manual of Steel Construction, 7th Edition, 1970

b. ASME Boiler and Pressure Vessel Code, Section III, Nuclear Power Plants Components, 1974, including Addenda through Winter, 1974 (Used for allowable stress values, coefficients of thermal expansion, and moduli of elasticity)

c. AISC "Specification for the Design, Fabrication, and Erection of Structural Steel for Buildings," February 10, 1969, and Supplements 1 through 3 (Supplement 3 was effective 6/12/74.)

6.3.2.2 Material Specifications

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

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

352 318

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

6.3.2.3 Welding Codes

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

b. Regulatory Guide 1.31, Rev. 1, "Control of Stainless Steel Welding," June 1973 (as modified by Branch Technical Position MTEB 5-1, November 24, 1975)

6.3.2.4 Quality Assurance, Cleanliness and Packaging Requirements as Applicable to the Spent Fuel Racks

a. S&W and VEPCO requirements

b. Regulatory Guide 1.38, "Quality Assurance Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water-Cooled Nuclear Power Plants," March 1973

c. ANSI N45.2.1-1973, "Cleaning of Fluid Systems and Associated Components During Construction"

d. ANSI N45.2.2-1972, "Packaging, Shipping, Receiving, Storage and Handling of Items for Nuclear Power Plants"

In addition, the racks are designed in accordance with the applicable portions of the following:

ANSI N210-1976 "Design Objectives for Light Water Reactor Spent Fuel Storage Facilities at Nuclear Power Stations"

6.4 Criticality Analyses

In order to verify that the design basis presented in Section 6.2 pertaining to criticality is met, analyses were performed for the normal, wet storage of spent fuel and for accidents pertaining to spent fuel storage. The accident analyses are presented in Section 6.7. The analysis of normal, wet storage of fuel assumes nominal conditions plus geometric material and calculational uncertainties. A parametric study is also presented to show the sensitivity of the results to variations in the assumed parameters. The analyses are based on storage of Westinghouse 17 by 17 fuel. In addition, an adjustment has been made to account for a higher reactivity due to the presence of Westinghouse 15 by 15 fuel in the event such fuel should be transferred to the North Anna Power Station for interim storage.

6.4.1 Assumptions

The calculations were based on the following conservative assumptions:

352 319

a. Fresh fuel of an equivalent enrichment of 3.50 w/o U-235 (44.2 grams of U-235 per axial cm)

b. Water temperature of 68 F

c. Fuel racks are infinite in three dimensions.

d. Fixed neutron poisons in the fuel assembly are neglected.

e. No credit is taken for neutron absorption by structural materials in the rack system, except the stainless steel tubes.

f. No soluble neutron poison in the pool water

A uniform array of 3.50 w/o enriched fuel is also assumed, although calculations have shown that this assumption is conservative compared to the more realistic one of distributed enrichments within the array.

6.4.2 Methods of Analysis

The majority of the calculations were performed with methods commonly used in light water reactor design; i.e., 4-group diffusion theory cell calculations using PDQ-07. Cross sections for these calculations are generated with NUMICE, the NUS version of LEOPARD. This code uses the same cross-section library tape and calculational techniques as LEOPARD.

Selected cases were checked and the final design multiplication factors were verified with Monte Carlo calculations using KENO with 123-group-cross-sections. The 123-group-cross-section library is generated from a basic GAM-THERMOS library using two subroutines, NITAWL and XSDRNPM in the AMPX code package. Both the PDQ code with cross-sections based on the LEOPARD library, and the KENO code using the 123-group-cross-section library have been benchmarked. The geometry of the fuel rack used in the calculations is shown in Figures 6.4-1 and 6.4-2.

6.4.3 Results of the Analyses

Under nominal conditions, a rack assembly as shown in Figure 6.4-1 with a center-to-center spacing of 14 in. between two adjacent storage locations and 1/8 in. Type 304 stainless steel storage tubes, results in a keff of 0.889 based on KENO analyses. Considering maximum variation in the position of fuel assemblies within the storage rack, engineering tolerances, seismic induced deflections, and calculational uncertainty resulted in a keff of 0.923 with a confidence level of 95 percent. This keff still meets the design basis to maintain keff equal to or less than 0.95 for the normal wet conditions. The above uncertainties are discussed further below and summarized in Table 6.4-1.

352 320

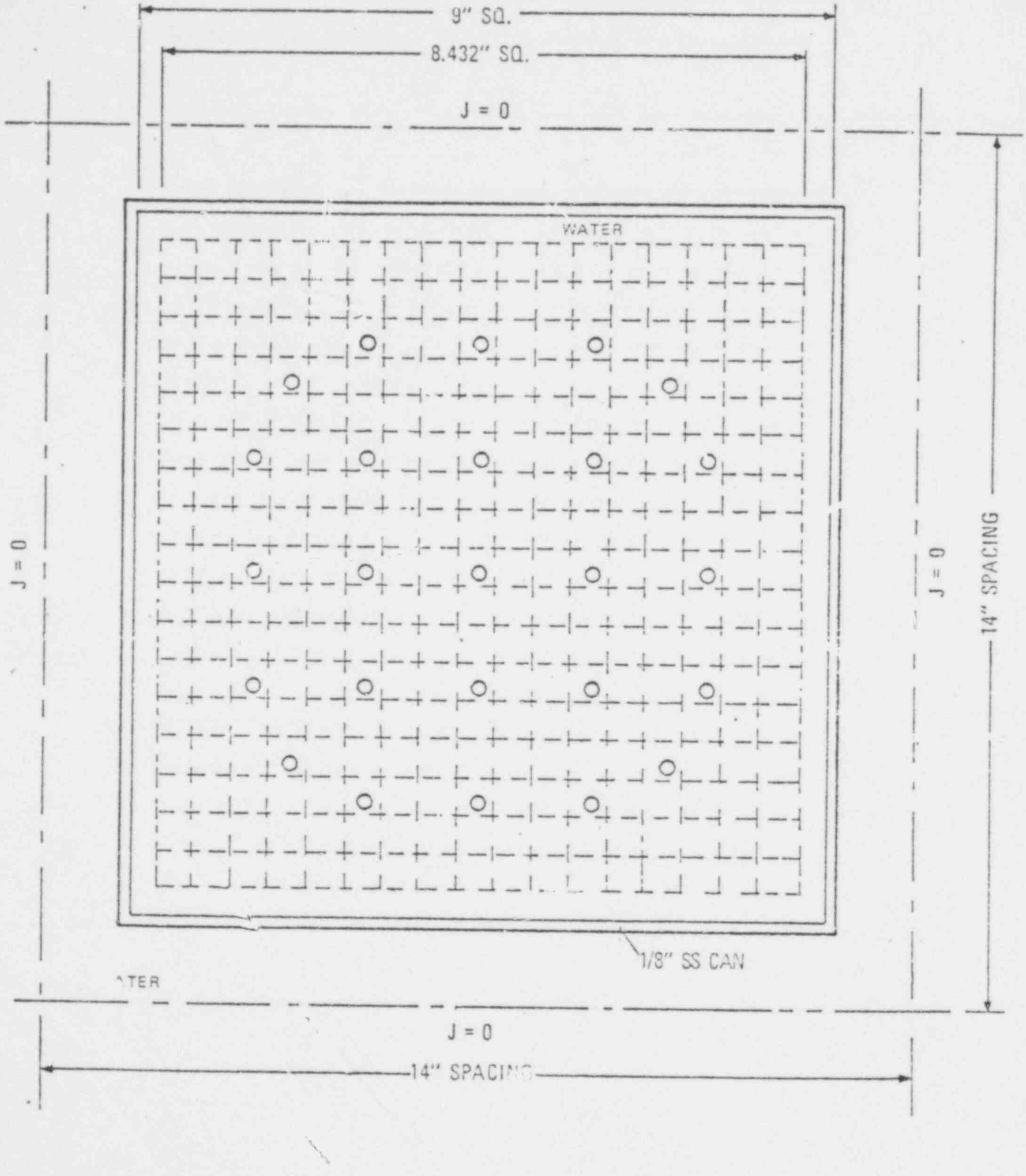


Figure 6.4-1
Storage Lattice Cell

352 321

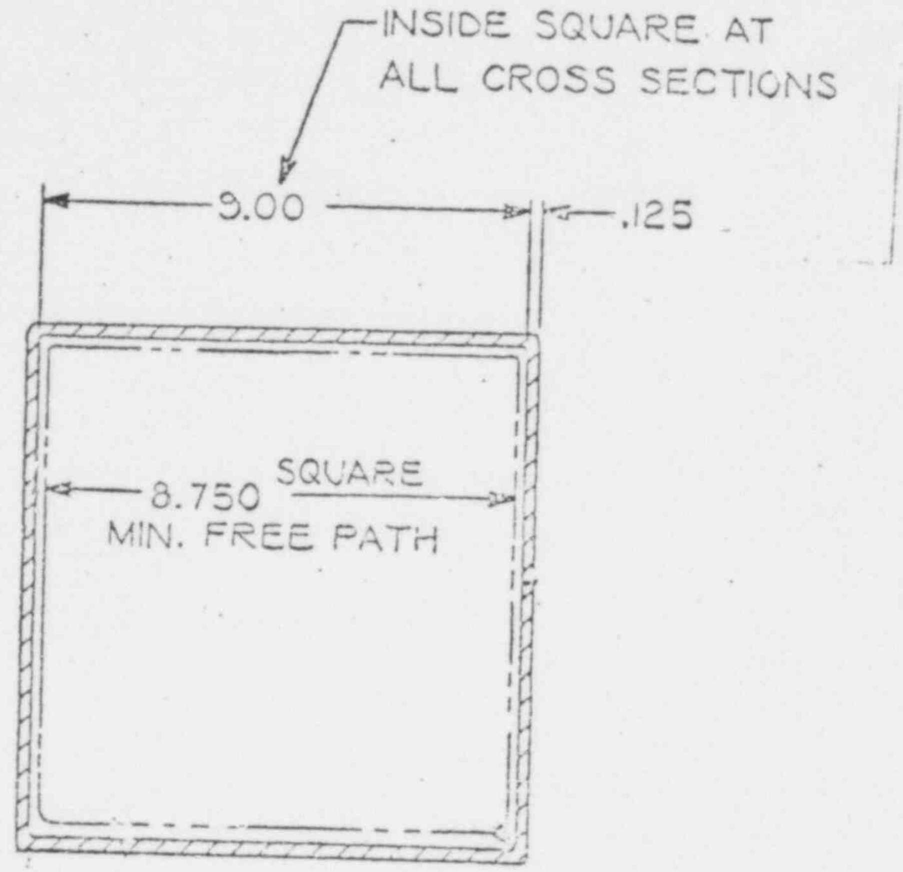


Figure 6.4-2
Stainless Steel Tube

352 322

6.4.3.1 Calculational Uncertainties

In general, the 4-group PDQ diffusion calculations produce keff values comparable to the Monte Carlo calculations. Calculational uncertainties in the use of PDQ with cross sections based on the LEOPARD library have been obtained by comparing the results of a series of benchmark calculations with critical experiments. These comparisons (Reference 1) have shown that the average difference between the calculations and experimental results was 0.009 delta-k. Thus, the LEOPARD-PDQ calculation can be used with confidence in predicting reactivity effects in this type of lattice.

However, to establish the absolute value of k, the KENO code was used to establish the uncertainty appropriate for this method. The KENO code using the 123-group GAM-THERMOS cross-section library has been benchmarked. For a series of ten experiments reported in Reference 2, the average keff as calculated using KENO and 123-group-cross-sections was 0.9914 ± .0020. Using the same method, NUS has performed another benchmark on one of the Yankee critical experiments (Reference 3, page 82) with Ag-Cd cruciform control rods banked at 26.37 cm from the bottom of the fuel. The calculated keff was 1.0077 ± 0.0034. On the basis of the above comparisons with criticals a calculational uncertainty of + .0086 delta-k was assigned to the KENO calculations.

A statistical analysis of the Monte Carlo runs results in a standard deviation of ±0.0055 giving an uncertainty in k-infinity of 0.0110 delta-k at a 95 percent confidence level. Adding this to the previously identified uncertainty of 0.0086 in the KENO value of k results in 0.0196 delta-k as a conservative uncertainty to be applied to the KENO results.

References

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

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

3. YAEC-94 "Yankee Critical Experiments" by P.W. Davison et al., Westinghouse Electric Corporation, April 1959.

6.4.3.2 Geometric and Material Variations

The most adverse geometric and material criticality condition was obtained by using the maximum tolerances for the positioning of the fuel assemblies within the storage can as well as the relative can-to-can positioning. The racks are toleranced on an overall rack width basis, such that cumulative tolerances between can-to-can positioning are not possible. Thus, simultaneous

352 323

inclusion of all these tolerances provides added conservatism to the analyses.

The tolerances considered in the analysis are for center-to-center spacing between storage locations, storage can inside dimension and wall thickness, and fuel assembly location in the storage can. These tolerances are: a reduction in center-to-center spacing from 14.00 to 13.94 in., a maximum positive tolerance on the inside dimension of the stainless steel can of 0.06 in., and a minimum wall thickness of 0.120 in. vs. the nominal 0.125 in. With respect to assembly location, every four assemblies were assumed to shift such that they are clustered in a common corner, giving the highest potential reactivity increase from fuel assembly location in the rack. Adding the above reactivity effects gives a reactivity uncertainty due to mechanical spacing and tolerances of 0.0083 delta-k.

Fuel enrichment uncertainty has a direct effect on reactivity. Assuming a very conservative uncertainty in enrichment of +2 percent, i.e. 3.57 w/o rather than 3.50 w/o, results in a reactivity increase of +0.0029 delta-k.

Variation in stainless steel composition also affects reactivity. An arbitrary reduction in the Fe, Cr and Ni content of the 304 stainless steel in the storage tubes to the lowest limit allowed by ASME Specification SA-240 would cause an increase in reactivity of +0.0031.

Spent fuel storage pool temperature variations are expected to be above 68 F. However, for conservatism, the reactivity increase due to a decrease in pool temperature to 39 F (4 C), the temperature of maximum water density, is included in the uncertainties. This reactivity is +0.0004 delta-k.

6.4.3.3 Sensitivity Study

The preceding sections discussed the reactivity additions due to worst-case tolerances and calculational uncertainties which are included in the criticality analysis of the racks. These analyses indicate to some extent the sensitivity of the multiplication factor to these parameters. This section further explores the sensitivity of keff to variations in parameters, in particular, spent fuel pool water temperature, center-to-center spacing, and storage can wall thickness.

a. Variation in Pool Water Temperature

The water temperature of 68 F used in the criticality analyses is lower than expected in the storage pool and is considered conservative. Using PDQ, analysis of the lattice of storage cells results in the following temperature reactivity effects:

352 324

Water Temperature, F	delta-k
39	+0.0004
68 (Base case)	0
100	-0.0012
212 (no void)	-0.0090

b. Variation in Center-to-Center Spacing

In the discussion of uncertainties, a possible 0.06 in. reduction in center-to-center spacing was included. To further investigate the effect of varying center-to-center spacing, a ±0.125 in. variation was considered. Using PDQ, the results are as follows:

Center-to-Center Spacing, In.	delta-k
13.875	+0.0034
14.0 (base case)	0
14.125	-0.0032

c. Variation in Wall Thickness of Storage Can

The minimum thickness of 0.120 in. was considered in the uncertainty discussions. As seen below, the effect of variation in thickness as calculated using PDQ is the same magnitude for ±0.005 in. variations.

Wall Thickness, In.	delta-k
0.120	+0.0014
0.125 (base case)	0
0.130	-0.0014

6.4.3.4 Summary of Results

The results are summarized in Table 6.4-1. The most adverse reactivity increase due to geometric and material variations is 0.0147 delta-k, which, added to the calculational uncertainty of 0.0196 and the nominal keff of 0.8893, results in a maximum keff of 0.9236 for the normal storage condition. It should be noted that an algebraic sum overestimates the effect of combining the reactivity effects of geometric and material variation. The root mean square of all effects excluding spent fuel pool water temperature, plus the pool water temperature effect, is more appropriate. This yields a combined reactivity for these effects of 0.0044 rather than 0.0147, and a maximum keff of 0.9133 rather than 0.9236.

6.5 Structural Analysis

6.5.1 Loads and Loading Criteria

The spent fuel storage racks are classified Seismic Category I. Structural integrity of the fuel racks when subjected to normal, abnormal,

352 325

TABLE 6.4-1

SUMMARY OF CRITICALITY ANALYSES

Nominal Conditions	
Enrichment: 3.50 w/o	
Mechanical Spacing: 14 in.	
Pool Temperature: 68 F	
keff	0.8865
15 by 15 in Place of 17 by 17 Fuel (delta-k)	0.0028
Calculational Uncertainties	
KENO Benchmarks (delta-k)	0.0086
Statistics	0.0110
Total, Calculational Uncertainties (delta-k)	0.0196
Geometric and Material Variations	
Variation of Enrichment from 3.5 to 3.57 w/o (delta-k)	0.0029
Mechanical Spacing and Tolerance (delta-k)	0.0083
Possible Variation in S. Composition (delta-k)	0.0031
Pool Temperature (delta-T=29 F) (delta-k)	0.0004
Total, Geometric and Material Uncertainties	0.0147
Maximum keff	
Nominal keff	0.8893
Calculational Uncertainties	0.0196
Geometric and Material Variations	0.0147
Maximum keff	0.9236

352 326

and seismic loads is demonstrated with respect to the NRC Standard Review Plan Section 3.8.4. In accordance with the Review Plan, the following loads, load combinations, and structural acceptance criteria are considered.

6.5.1.1 Loads

a. Normal Loads

i. Dead Loads - deadweight of rack and fuel assemblies and buoyancy

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

iii. Thermal Loads - thermal gradient between adjacent storage locations of 39 F

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

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

d. Accidental drop in water of spent fuel assembly from height consistent with fuel handling operations which is 3 ft-1 in. above the top of the spent fuel racks.

e. Postulated stuck fuel assembly which causes an upward force on the rack of 4,000 lb, which is the refueling bridge crane limit.

6.5.1.2 Load Combinations

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

a. Dead Loads Plus Live Loads

b. Dead Loads Plus OBE

c. Dead Loads Plus Thermal Loads Plus OBE

d. Dead Loads Plus Thermal Loads Plus DBE

e. Dead Loads Plus Thermal Loads Plus Fuel Assembly Drop

f. Dead Loads Plus Thermal Loads Plus Stuck Fuel Assembly

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

352 327

6.5.1.3 Structural Acceptance Criteria

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

Load Combination	Strength Limit
a.	1.0 S
b.	1.0 S
c.	1.5 S
d.	1.6 S
e.	1.6 S (except as noted below)
f.	1.6 S

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

6.5.2 Seismic Analysis

The seismic loading of a fuel rack module is determined from a response spectrum modal dynamic analysis in which the stiffness of the fuel assembly is neglected. However, the mass of the fuel assemblies and an effective mass of water are considered to be uniformly distributed along the storage tubes. The appropriate response spectra for the OBE and the DBE as taken from structural analysis presented in the FSAR, Section 3.7.2 are employed. The STARDYNE computer program is used to perform the structural analysis of the racks. Racks are modeled in detail using beam and plate finite elements. The three-dimensional finite element model for a spent fuel rack is shown in Figure 6.5-1.

To determine the earthquake response, STARDYNE is first run to determine the natural frequencies and participation factors. For frequencies with significant modal participation, mode shapes and modal loads are calculated. Closely spaced modes are combined directly and then combined in an SRSS manner as defined in Regulatory Guide 1.92 with other significant modes. The results of horizontal and vertical earthquakes are combined in an SRSS fashion.

In the general seismic/structural analysis of the fuel racks, the mass of a fuel assembly is assumed to be uniformly distributed along the length of each of the fuel storage cans. This assumption is conservative in that lower rack fundamental frequencies are calculated which, due to the relatively stiff rack design, result in higher seismic amplified acceleration loading on the rack. To account for the submergence of the racks in the analysis, the mass of a volume of liquid was added to the

352 328

mass of a structure giving a total "virtual" mass. The structure can then be analyzed as though it stood in air. This approach is discussed in Section 6.4 of Fundamentals of Earthquake Engineering by Newmark and Rosenbleuth. The added mass of water outside of the can was calculated based on a cylinder with a diameter of the width of the can (9.25 in.) and length, that of the can (170.25 in.). The added mass of water inside the can was based on the inside volume of the can minus the water displaced by the fuel.

The fundamental frequencies of a typical (6 by 6) rack in the two lateral directions, north-south and east-west, and in the vertical direction are calculated to be 15.2 Hz, 14.1 Hz, and 70.7 Hz, respectively. The high frequency (>40 Hz) in the vertical direction allowed for a static analysis for this direction.

Since a gap on the order of 1/4 in. exists between the sides of a fuel assembly and the can, the fuel will actually move within the can during a seismic event and cause impact loads to be transmitted to the fuel rack. The effects of this fuel-can interaction are analyzed by utilizing the ANSYS computer program. A nonlinear dynamic analysis of a single can and fuel assembly is performed to determine the shear force and bending moment which may occur at critical sections of the can as a result of the fuel assembly impacting the can at the maximum velocity.

The can and fuel assembly are modeled by finite elements separated by nonlinear gap elements as shown in Figure 6.5-2. The can has stiffness characteristics representative of a can within a rack. The fuel, which is assumed to be pinned at its base (by friction), is given an initial velocity relative to the can. This initial velocity is equal to the SRSS of the floor velocity and the velocity of the rack with respect to the floor.

6.5.3 Structural Adequacy

Using the previously listed loads and load combinations, stresses are calculated at critical sections of the racks. The results of the structural and seismic analyses demonstrate that the fuel racks are structurally adequate and meet the design criteria. Critical stresses together with locations and margins to allowable are given in Table 6.5-1.

6.5.4 Pool Floor Loading

The structural adequacy of the pool for floor loading is discussed in Section 7.1.

6.6 Stored Fuel Assembly Thermal - Hydraulic Analysis

The fuel rack base is elevated above the floor to assure adequate flow under the rack to each fuel assembly. The spacing of the fuel assemblies also permits adequate downflow within the rack to each storage location. Analyses have been performed and show

352 329

TABLE 6.5-1

SUMMARY OF CRITICAL STRESS RESULTS

Location	Limiting Load Combination[1]	Calculated Stress (Psi)	Allowable Stress (Psi)	Margin
Support Foot Welds	d	22,680	27,600	1.22
Fuel Can to Base Welds	d	9,700	18,400	1.90
Fuel Cans	d	12,740	27,600	2.17
Shear Plate to Fuel Can Welds	d	16,990	18,400	1.08
Base Beam Welds	b	7,580	11,500	1.52

[1] See Section 6.5.1.2 for definitions.

352 330

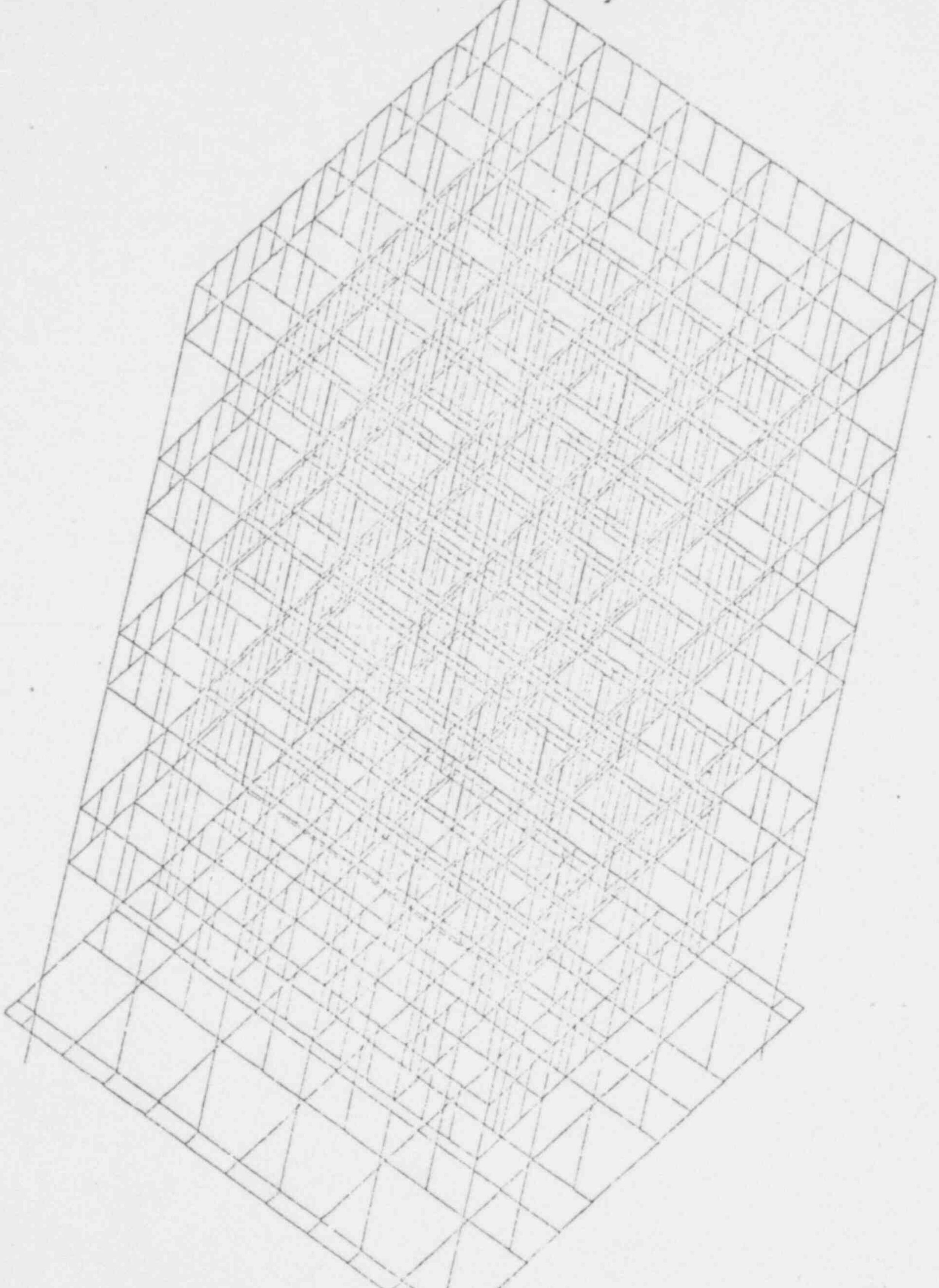


Figure 6.5-1
FINITE ELEMENT MODEL

352 331

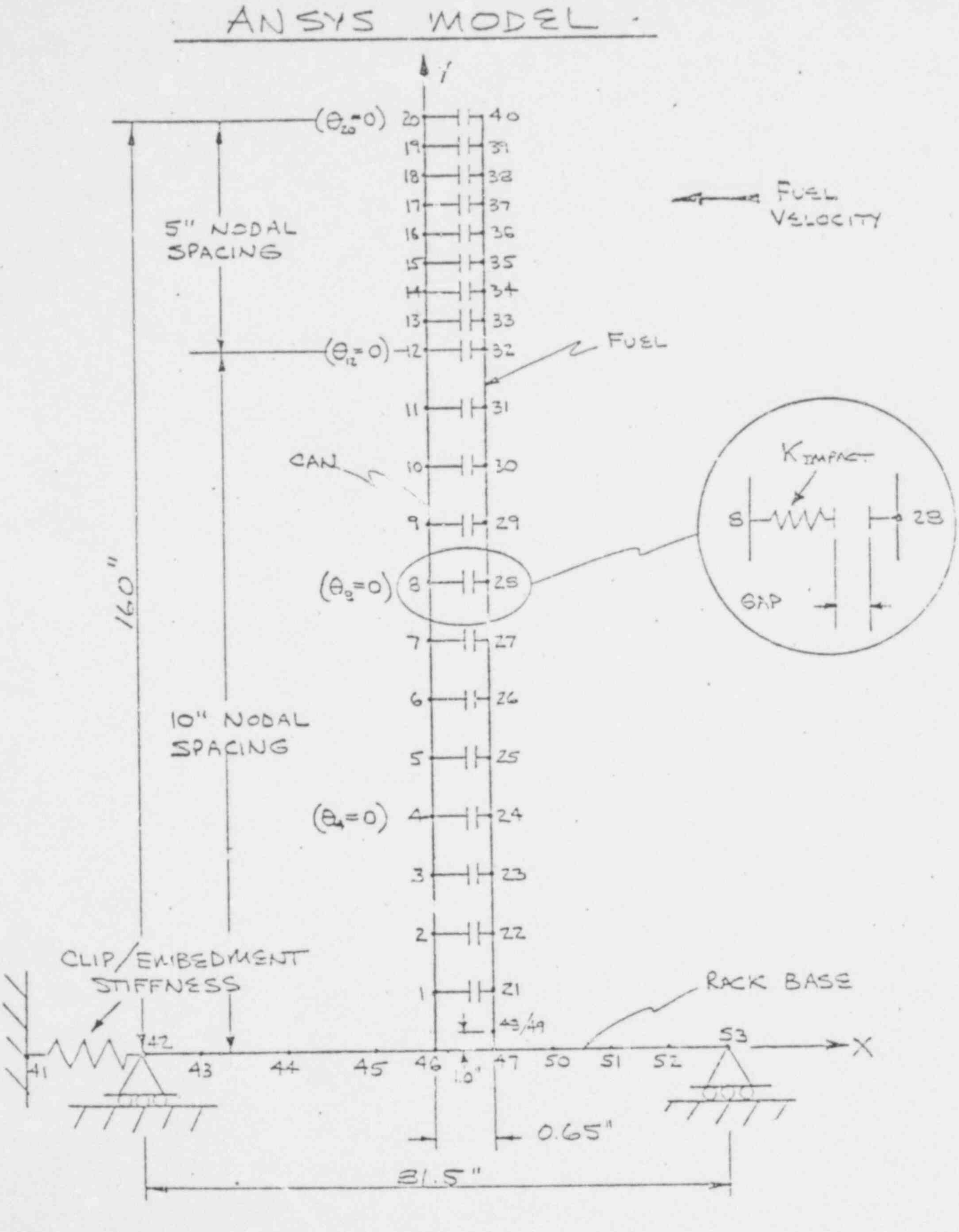


Figure 6.5-2

352 332

that sufficient flow is induced by natural convection to preclude local boiling in the hottest storage location. This location is established by a 15 by 15 Westinghouse fuel element which was determined to be worse than for a 17 by 17 fuel element.

The analyses were based on the following assumptions:

a. The element inlet temperature is the mixed hot temperature of the pool.

b. A hot assembly peaking factor of 1.55 is applied to the core average assembly energy release rate of 2.00×10^5 Btu/hr, (corresponds to 150 hr after shutdown).

c. The maximum local peaking factor is 2.22, giving a maximum local heat flux of 1,642 Btu/hr - ft^2.

d. A film coefficient of 35 Btu/hr - ft^2 - F is based on pure conduction through a stagnant boundary layer at the fuel rod surface.

e. The downcomer region is established by 40 percent of the flow area between the corner assemblies in the rack since these have the greatest hydraulic resistance.

f. One dimensional fluid flow analysis used.

g. The Rohsenow pool boiling correlation is used to indicate no local boiling.

h. The maximum local heat flux is conservatively applied at the exit of the hot channel.

During full-core offload with the bulk pool temperature at 170 F, the maximum temperature of the water exiting from the hottest storage location, i.e., with a 15 by 15 Westinghouse assembly, is less than 197 F. This is 44 F below the local saturation temperature of 241 F. With a less restrictive 17 by 17 Westinghouse assembly in which the hot spot temperature is lower because of more heat transfer area, the maximum bulk water temperature exiting from the hottest assembly is no greater than 198 F.

Under design operating conditions, with the bulk pool temperature at 140 F, the fuel rod surface temperature calculated on the basis of the heat flux and film coefficient defined above is below the local saturation temperature and thus precludes local boiling. Assuming a maximum bulk pool temperature of 170 F, the fuel rod surface temperature is at least 4 F below the nucleate boiling temperature evaluated using the Rohsenow pool boiling correlation and therefore no local boiling is predicted.

352 333

6.7 Safety Evaluation

6.7.1 Potential Criticality Accidents

To determine the potential for criticality during an accident, two possible but highly unlikely postulated events were analyzed using the analytical methods described in Section 6.4.2. The first case is a fuel assembly, in transport in a vertical position, which accidentally drops into the water channel between a rack and the pool wall. The second case is a fuel assembly which drops and falls to a horizontal position on top of a loaded storage rack. The analyses include the assumptions of all the calculational uncertainties and geometrical and material variations discussed in Section 6.4.3. The design enrichment of 3.50 w/o is assumed for both the dropped and stored fuel assemblies and all fuel assemblies are assumed to be fresh assemblies.

In the first case, the concern is assumed to be the dropping or accidental lowering of a fuel assembly so that it is parallel to and at the same level as the stored fuel in rack assemblies (see Figure 6.7–1). The resulting keff including uncertainties is 0.9239 which is only slightly higher than the maximum keff for the normal storage situation (Section 6.4.3) and is less than the design basis value of 0.95. Mechanical restriction will be provided to prevent an unprotected fuel assembly from being brought closer than 5 in. to the side of any rack assembly in side water channels.

For a dropped assembly assumed to lay across the top of a fuel rack, the fuel assembly end fittings on the top of each fuel assembly provide a spacing between the dropped assembly and the active fuel in the storage racks of approximately 10 in. This is a significantly less reactive condition than that analyzed above in which the fuel assembly is much closer to the fuel in the rack. Thus, this accident has not been analyzed further.

352 334

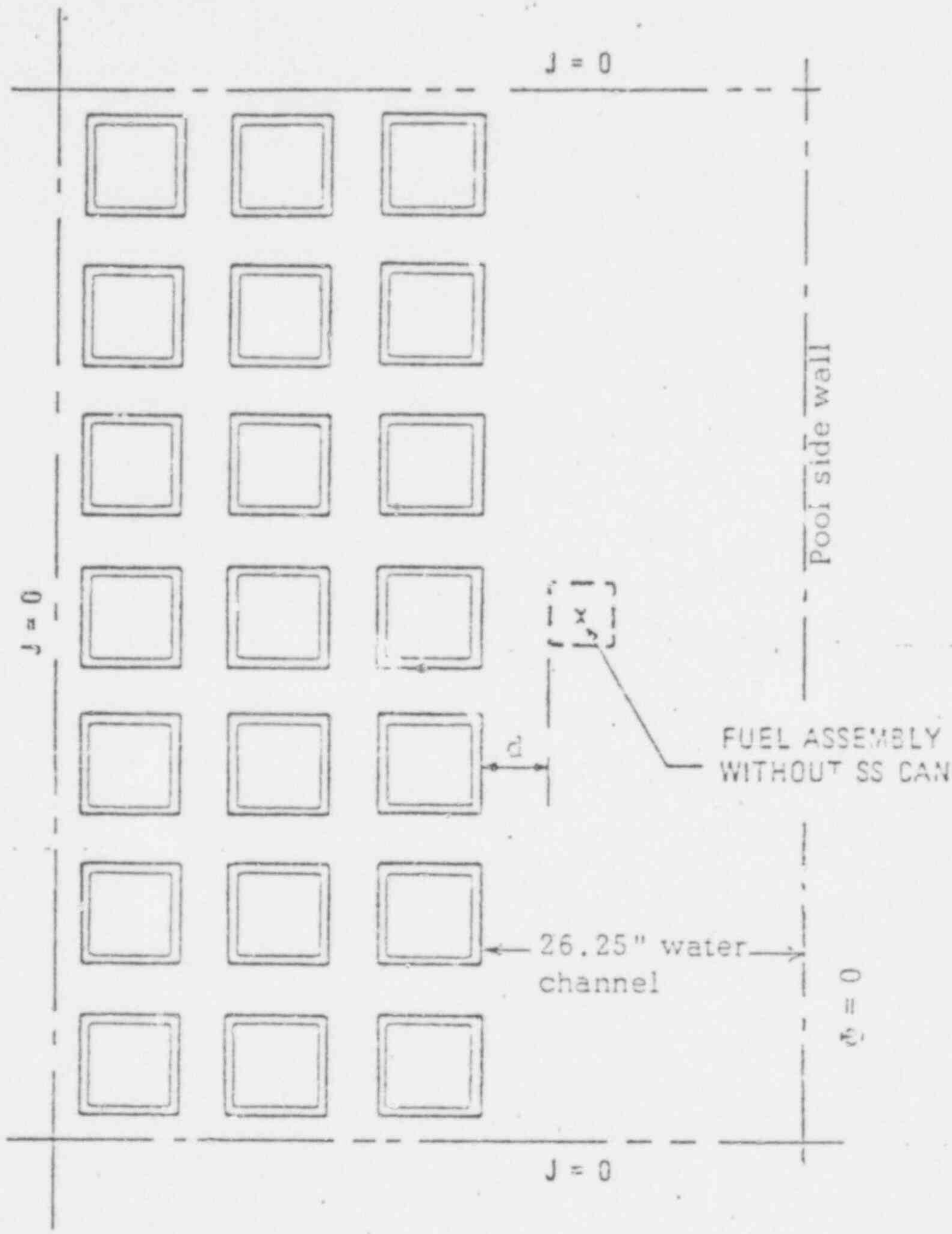


Figure 6.7-1

GEOMETRY OF THE WORST ACCIDENT CASE

352 335

7.0 ANALYSIS OF EXISTING FACILITIES AND SYSTEMS AFFECTED BY THE PROPOSED MODIFICATION

The proposed modification does not change the physical configuration of the spent fuel pool and only requires the addition of two floor pads. The existing spent fuel pool cooling and purification system and fuel building ventilation system will not require any modifications. The primary effect of installing new spent fuel storage racks will be to increase the amount of spent fuel which may be stored in the pool; thereby, increasing the weight to be supported by the pool floor. The additional spent fuel stored in the pool will slightly increase the amount of decay heat which must be removed by the spent fuel cooling system. The effect of the proposed modification on the purification system will be minor. These and other effects are discussed below.

7.1 Structural Considerations

The spent fuel pool structure has been analyzed to determine the effect of the additional weight of the new racks and the stored fuel will have on the structure under static and dynamic conditions. The analysis also included consideration of the thermal effects due to increasing the storage capacity. The existing structure has sufficient design margin which permits the installation of new storage racks without any additional structural modifications to the pool concrete or superstructure except for the addition of two new floor pads required to accommodate a high-density rack in an area where presently no rack is located.

In the original design of the spent fuel storage racks, the seismic response of the rack system was calculated taking into account 2% damping due to rack submergence in water. Additional response calculations were made eliminating the additional damping and the resulting load on the floor embedment pads were well within the calculated allowable values.

7.2 Fuel Pool Cooling System

The installed spent fuel pool cooling system was analyzed in view of the expanded fuel storage capacity. Table 7-1 summarizes the cooling system performance for both the normal and abnormal (full core discharge) conditions.

The design basis heat load was determined using the following assumptions:

1. The irradiation times used were 272, 544, and 816 EFPD which correspond to a one, two, and three year fuel cycle, respectively, with a load factor of 85 percent and an annual 45 day refueling outage.

2. Back to back refuelings 45 days apart.

3. Uranium decay heat from NRC Branch Technical Position 9-2.

4. All fuel to be moved into the pool is done instantaneously 150 hr after shutdown except for the full core discharge case when fuel is moved from the

352 336

reactor to the pool at a rate of .20 min per assembly starting 150 hr after shutdown.

5. Stretch rating of 2900 MW is used for full power.
6. Maximum of 966 storage locations in pool.
7. Service water at its design maximum of 110°F yielding a component cooling water temperature of 113.2°F.

The resulting fuel pool temperatures are found to be within the limits of 140 deg for the normal case and 170 deg for the abnormal case if one fuel pool cooling system pump and two coolers are utilized.

These fuel pool temperatures are calculated based on very conservative and worst case assumptions and are valid for establishing a design basis. Actual operating temperature experienced by Surry Power Station, Unit Nos. 1 and 2, which is of similar design have been significantly lower than the calculated temperature, and in fact has been maintained at about 95 deg F during winter and summer conditions utilizing only one pump and one cooler.

The additional heat load due to the increased storage capacity is compared in Table 7-2 to the design duty of the component cooling water system, the design duty of the service water system, and the total station thermal discharge to the environment.

A failure analysis of the spent fuel pool cooling system is summarized in Table 7-3. The analysis confirm that boiling is prevented even in the event of a postulated failure.

The analyses presented in the FSAR related to spent fuel pool makeup capability is not affected by the proposed modification and is not discussed herein.

7.3 Fuel Pool Purification System

As previously mentioned in Section 5.5.2, based on the experience of Surry Power Station, no significant effect on the system is expected due to prolonged storage of spent fuel assemblies. The maximum load on the purification system occurs during refueling operations when fuel is being moved. Therefore, there will be no significant increase on the purification system load due to the modification since the number and frequency of refueling operations will not change.

7.4 Fuel Building Ventilation System

Since the added fuel storage represents longer term storage of well-cooled fuel, the escape of gaseous or volatile fission products from even defective fuel is expected to be negligible. Much of the iodines and the xenon has decayed after 100 days cooling time. Sinc most of the tritium in the water is

352 337

formed primarily as a product of the neutron irradiation of boron in the primary coolant, the contribution of fission product tritium is minor. There is no mechanism for particulate fission products to become airborne. Because of its long half life, Kr-85 levels remain in older fuel. However, the thermal driving force required to cause its diffusion in defective fuel is not present. Samples from the ventilation filter area at Surry do not show Kr-85 at detectable levels, and it is not expected to become significant as fuel storage increases. Therefore, increased fuel storage will have essentially no impact on concentrations of radioactivity in the air of the fuel building.

Since the FSAR pool temperature limits of 140 F (normal case) and 170 F (abnormal case) will not change with the modification, there will be no effect on the design evaporation rate of the pool.

TABLE 7-1

SPENT FUEL POOL COOLING SYSTEM HEAT LOAD AND OPERATING TEMPERATURE WITH THE INCREASED STORAGE CAPACITY

	Decay Heat MBtu/hr	Fuel Pool Temperature, Deg F 1 Train 1P-1Clr	1P-2Clr	2 Train 2P-2Clr
Normal	19.4	147.8	135.4	130.4
Abnormal	35.9	176.9	154.2	144.9

TABLE 7-2

HE... LOAD COMPARISON

	Heat Load, Mbtu/Hr Present System (400 Assemblies)	Heat Load, Mbtu/Hr With Modification (966 Assemblies)
Fuel Pool Cooling System Design Duty		
Normal Case	13.8	19.4
Abnormal Case	33.2	35.9
Component Cooling Water System Design Duty (1)		
Normal Operation (2P-2HXR)	103.1	108.7
Abnormal Operation (4P-4HXR) (2)	402.1	407.6
Service Water System Design Duty (1)		
Normal Operation (2 Pumps)	103.1	108.7
Abnormal Operation (4 Pumps) (2)	402.1	407.6
Total Heat Discharged to the Environment (1) Normal Operation	13,713	13,719

NOTES:

1. Total for both units

2. Maximum abnormal operating duty for the component cooling water system and the service water system occurs during simultaneous fast cooldown of both reactor units. This does not occur concurrent with the maximum abnormal fuel pool heat load which is the off loading of one full core starting 150 hr after shutdown.

352 340

TABLE 7-3

SPENT FUEL POOL COOLING SYSTEM MALFUNCTION ANALYSES

Component	Malfunction	Comments and Consequences
Spent Fuel Pit Cooling Pumps	Pump fails to start or fails during operation	The standby pump will be started manually.
		If the operating pump should stop, over 1 hr exists to start the standby pump before the pit heats up 10 F at the maximum abnormal heat load.
		Pool temperature will not exceed 170 F with only one pump and two coolers in service for the maximum abnormal heat load.
Fuel Pit Coolers	Loss of Function	The standby exchanger will be used. More than 1 hr exists to realign the piping system because of the slow heatup rate of the pool. The re-alignment is effected by operating manual valves. During the design basis conditions with only one pump and one cooler in service the maximum calculated temperature is 176.9 F which is above the administrative limit of 170 F but is still below the temperature at which the structural analysis was performed.

352 341

8.0 INSTALLATION AND REMOVAL OF SPENT FUEL STORAGE RACKS

The modification will be limited to the confines of the fuel building and will not involve any changes in the safety-related systems which are necessary for the station.

The fuel pool cooling and purification equipment is located in the area of the fuel building which is physically isolated by concrete walls and ceilings and is not subject to damage while the old racks are removed and the new racks installed.

Since the old spent fuel storage racks have not been exposed to radiation, and therefore are not contaminated, it is at this time planned to dispose of them as scrap.

352 342

9.0 ANALYSIS OF THE SAFETY IMPLICATIONS OF THE PROPOSED MODIFICATION

The proposed modification will not change the safety analyses which have been performed and reported in the Final Safety Analysis Report, Section 15. The proposed expansion of the spent fuel storage capacity could affect the offsite radiological consequences of an incident because of the additional increment of long-lived radioactive fission products stored in the pool. The effect of this amount of additional radioactive products on normal station operation is discussed in Section 9.5 of this report and its effect on the spent fuel handling accident is discussed in Section 9.4.

The following discussion summarizes the potential effects which the proposed modification may have on the safety of the station and the public.

9.1 Loss of Spent Fuel Pool Cooling Capability

As discussed in Section 7.2, the proposed modification will increase the amount of heat energy which is added to the pool water which must be removed by the spent fuel pool cooling system. The existing cooling system has sufficient design margin to remove the additional heat load when uranium fuel is stored in the pool. As indicated by the failure analysis presented in Section 7.2, cooling capacity could be restored quickly in the event of a component failure.

In the unlikely event that the spent fuel pool cooling system was to become completely inoperable, installed station systems could provide sufficient makeup water to cool the fuel and to maintain sufficient water shielding over the pool. There are several sources of makeup water readily available in the event it is required. These sources are:

1. primary grade water system
2. fire protection system
3. boron recovery system
4. refueling water storage tank

These sources could be utilized by either changing valve lineups or implementing certain temporary measures, such as the use of temporary pumps or hoses.

352 343

In summary, sufficient heat removal capacity is installed to assure that the pool temperature remains below the boiling point. As additional backup a number of installed station systems could provide makeup and cooling water if required.

9.2 Fuel Pool Leakage Control and Shielding

The proposed modification will not affect the leakage and shielding requirements contained in the FSAR. The lowest level of pipe penetration through the fuel pool structure is at El. 285 ft-5 in., which provides a minimum water level of over 21 ft of water above the stored fuel to provide shielding and cooling.

The proposed modification will not require any additional piping penetrations; therefore, there are no safety implications associated with spent fuel pool leakage control or shielding.

9.3 Earthquake and Tornado Protection

The proposed modification will not require any structural changes; therefore, it will not affect the ability of the structure to withstand the effects of an earthquake or tornado as stated in the FSAR. The new spent fuel storage racks and pool structure have been analyzed to ensure that the racks can be accommodated by the structure during a seismic event. These analyses are discussed in detail in other sections of this report.

In summary, the seismic and tornadic provisions stated in the FSAR are not changed as a result of the proposed modification; therefore, there are no safety implications associated therewith.

9.4 Fuel Handling Accidents

Section 15 of the FSAR describes the fuel handling accidents which have been analyzed, including the case where a fuel assembly is dropped onto the floor of the spent fuel pool. The proposed modification will not affect the consequences of the accidents analyzed in the FSAR because the analysis assumes that only one fuel assembly, the one being installed, is damaged. Thus, the consequences of the accident are independent of the number of spent fuel elements stored in the pool.

The high density spent fuel racks have been reviewed in regard to:

1. dropping a fuel assembly on the racks

2. a fuel assembly becomes stuck in the spent fuel rack

352 344

3. dropping a fuel assembly next to the rack

While minor damage may be incurred to the rack if an element is dropped on it, the stored fuel will not be affected and subcriticality will be maintained. The amount of force applied to a stuck fuel assembly is limited by the capacity of the crane. While damage may be incurred by the stuck fuel assembly, the weight of the fuel rack is sufficient to prevent any motion of the rack itself. The surrounding stored fuel assemblies will not be damaged and subcriticality will be maintained.

Mechanical barriers are provided on the outside of the rack to prevent a dropped fuel assembly from being brought too close to the rack in order to maintain subcriticality as discussed in Section 6.

In summary, the safety implications of the proposed modification as related to fuel handling accidents remain the same as previously analyzed in the FSAR.

9.5 Personnel Radiation Exposure

Storing additional spent fuel in the pool will increase the amount of corrosion and fission product nuclides introduced into the pool water. The proposed modification will approximately double the amount of fuel to be stored in the pool. Dependent upon the disposition of the reprocessing capability, the fuel could be stored in the pool for about 10 years.

During the storage of spent fuel under water, both volatile and nonvolatile radioactive nuclides may be released to the water from the surface of the assemblies or from defects in the fuel cladding. Most of the material released from the surface of the assemblies consists of activated corrosion products such as Co-58, Co-60, Fe-59 and Mn-54 which are not volatile. The radionuclides released through defects in the cladding, such as Cs-134, Cs-137, Sr-89 and Sr-90, are predominantly nonvolatile and, as with the activated corrosion product nuclides, the primary effect is their contribution to radiation levels to which workers near the spent fuel pool would be exposed. As noted in Section 5, the four primary isotopes noted in the pool water at Surry have been Cs-134, Cs-137, Co-58 and Co-60.

Based on measured data at Surry Power Station, an individual continuously working around the pool would receive about 1.5 mR/hr, based on approximately 208 fuel assemblies stored in the pool. This exposure will probably slightly increase when additional fuel assemblies are stored; however, because of the current storage pattern, the increase is not expected to be significant. Even if the exposure is doubled, about 3 mR/hr, this exposure is a relatively minor contribution to the overall exposure at the station.

352 345

The installed purification system described in Section 5.2 will be used to remove the nonvolatile corrosion and fission product nuclides. The removal of these nuclides will assure that the radiation exposure to personnel will be maintained at low levels.

The volatile fission product nuclides of most concern that might be released through defects in the fuel cladding are the noble gases (Xenon and Krypton), tritium and iodine isotopes. Since short-lived noble gases will decay to negligible amounts, the only significant noble gas isotope which could remain in the spent fuel pool and attributable to storing additional assemblies for a longer period of time would be Krypton-85. It is not expected that increasing the spent fuel storage capacity will increase the Krypton-85 release rate, since the fuel discharge will continue on a 1/3 core per year per unit rate and the release of Krypton-85 is most likely to occur during the initial year of storage.

Iodine 131 releases will not be significantly increased by the expansion of the fuel storage capacity since the I-131 inventory in the fuel will decay to negligible levels. Operation experience at Surry to date indicates negligible levels of I-131 in the pool water.

The pool water temperature will be maintained below the current design temperature; therefore, it is not expected that there will be any significant change in evaporation rates and the release of tritium. Operating experience at Surry to date has not indicated the presence of tritium in the fuel building.

As discussed in Section 5.5, the purification filters are normally changed because of high differential pressure; therefore, it is not expected that the proposed modification will significantly increase personnel radiation exposure during filter changes. Based on experience at Surry Power Station, the radiation exposure is relatively low, approximately 140 mR/hr. The demineralizer resins are currently changed about twice a year resulting in personnel exposure of about 110 mR/hr. The proposed modification is not expected to significantly increase this value.

In summary, the proposed modification will not significantly increase personnel radiation exposure during normal and refueling operations.

352 346

10.0 ENVIRONMENTAL IMPACT OF THE PROPOSED MODIFICATION

The proposed modification would increase the amount of decay heat produced in the spent fuel pool, increase the amount of radioactivity stored in the pool and result in a small commitment of metal resources.

The environmental impact of the proposed modification is insignificant. The environmental impact has been reviewed in light of the current Final Environmental Statement, the Final Safety Analysis Report, 10CFR50, 40CFR1500.6 and guidance contained in Federal Register Notice (40 F.R. 42801) dated September 16, 1975. Based on this review it has been concluded that the proposed modification will not significantly affect the quality of the human environment.

10.1 Federal Register Notice (40 F.R. 42801)

The Federal Register (FR) Notice directs that in consideration of licensing actions to ameliorate a possible shortage of spent fuel storage capacity, five specific factors should be applied, balanced and weighed in the context of the required environmental appraisal. Each of the five specific factors are addressed below.

10.1.1 Independence of the Action

The FR Notice states: "Is it likely that the licensing action here proposed would have a utility that is independent of the utility of other licensing actions designed to ameliorate a possible shortage of spent fuel capacity?"

Based on the information contained in this report, it has been concluded that a need for additional spent fuel storage capacity exists at the North Anna Power Station, Unit Nos. 1 and 2, which is independent of the utility of other licensing actions designed to delay a possible shortage of spent fuel storage capacity.

10.1.2 Commitment of Resources

The FR Notice states: "Is it not likely that the taking of the action here proposed prior to the preparation of the generic statement would constitute a commitment of resources that would tend to significantly foreclose the alternatives available with respect to any other licensing actions designed to delay a possible shortage of spent fuel storage capacity?"

The proposed modification will require the utilization of 3.22×10^5 lb of stainless steel. The amount of stainless steel used annually in the United States is about 2.8×10^{11} lb. The amount of stainless steel required for the racks is a small percentage of this resource consumed annually in the United

352 347

States and is insignificant. No other significant material resources will be required because the design of the fuel pool will remain unchanged. The land area now used for the fuel pool will be used more efficiently by reducing the spacing among fuel assemblies.

The existing installed spent fuel racks will be disposed of as scrap since they have not been exposed to radiation. This should pose no problem.

The proposed expansion of the storage capability of the existing fuel pool is only a measure to allow for continued operation and to provide operational flexibility at the station and will not affect similar licensing actions at other nuclear power stations. It is concluded that the increase of the spent fuel storage capacity at North Anna prior to the preparation of the generic statement does not constitute a commitment of either material or nonmaterial resources that would tend to significantly foreclose the alternatives available with respect to any other individual licensing action designed to delay a possible shortage of spent fuel storage capacity.

10.1.3 Cumulative Environmental Effects

The FR Notice states: "Can the environmental impacts associated with the licensing action here proposed be adequately addressed within the context of the present application without overloading any cumulative environmental effects?"

The additional capacity of the spent fuel pool is proposed for North Anna Power Station, Unit Nos. 1 and 2, only; therefore, the environmental impacts can be assessed within the context of the application. Based on the information contained herein, it has been shown that the environmental impact due to the installation and operation of an expanded spent fuel pool storage capacity is insignificant. It is concluded that the cumulative environmental impacts associated with the expansion of the spent fuel pool will not result in radioactive effluent releases nor occupational radiation exposure nor thermal effluent releases that significantly affect the quality of the human environment during either normal operation of the expanded fuel pool or under postulated fuel handling accidents.

10.1.4 Technical Issues

The FR Notice states: "Have all technical issues which have risen during the review of this application been resolved within that context?"

The technical issues associated with the proposed modification are addressed in this report. There is reasonable assurance that the proposed modification can be carried out as described herein with no adverse effects on the health and safety of the public.

352 348

10.1.5 Need for the Action

The FR Notice states: "Would a deferral or severe restriction on this licensing action result in substantial harm to the public interest?"

As stated in Section 4.0, a number of alternatives have been considered. The modification described herein provides the most economically feasible solution to ameliorate the potential shortage of spent fuel storage capacity. If the proposed modification is not implemented, the alternative of ceasing operation of the facility would be much more expensive than the proposed action because of the need to provide fossil fuel replacement power. Deferral or severe restriction of the proposed modification would result in substantial harm to the public interest.

10.2 Final Environmental Statement

The proposed modification will not significantly alter the evaluations contained in the Final Environmental Statements. The proposed modification will create a slight additional heat load on the station water system; however, because of the minor amount of additional heat to be added to the service water system, no discernable temperature difference in the thermal effluent from the station is expected.

10.3 Final Safety Analysis Report

The descriptive information contained herein is intended to supplement the material contained in the FSAR. The design criteria specified in the FSAR have been used as the basis for the proposed modification and have been supplemented as appropriate for the new spent fuel storage racks. The modification does not substantially change the analyses and descriptions in the FSAR.

352 349

11.0 CONCLUSIONS

Based on the information contained herein, it is concluded that:

1. The proposed modification to increase the storage capacity of the spent fuel pool is necessary to maintain the capability of a full core discharge and to assure adequate storage space for normal refuelings.

2. The use of fuel racks with reduced center-to-center spacing provides the most economical and feasible alternative to delay the potential shortage of storage capacity.

3. The installed fuel pool cooling and purification system and the fuel building ventilation system are adequate without any modifications.

4. The fuel building structure is adequate and does not require modification.

5. The availability of alternate storage, reprocessing, or permanent disposal capabilities cannot be relied upon in the near future to preclude the necessity of the proposed modification.

6. The proposed modification will not affect the health and safety of the general public.

7. The proposed modification will not significantly affect the quality of the environment.

352 350

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of	)	
	)	Doc. Nos. 50-338SP
VIRGINIA ELECTRIC AND POWER COMPANY	)	50-339SP
	)	Proposed Amendment to
(North Anna Power Station, Units 1 and 2)	)	Operating License NPF-4

AFFIDAVIT OF ROBERT W. CALDER

My name is Robert W. Calder. A true statement of my professional qualifications is attached to this affidavit.

I do not expect that storing 966 instead of 400 spent fuel assemblies in the North Anna 1 and 2 spent fuel pool will materially increase the corrosion of the fuel cladding, the spent fuel storage racks, or the pool liner. The amount of additional radiation to which these materials would be exposed in the pool is insignificant compared to the levels in the reactor core during power operation. The materials (stainless steel and zircaloy) were chosen because of their low susceptability to corrosive attack in a nuclear environment, that is, under exposure to high temperature, high pressure, water and radiation. Because the fuel pool water temperature of 140°F (normal condition) and 170°F (abnormal condition) will still be maintained, I would not expect the corrosion or stress on the fuel, the racks, or the liner to materially increase due to heat.

Mr. A. B. Johnson, Jr., Staff Scientist, Corrosion Re-

search and Engineering, Battelle Pacific Northwest Laboratories, has reported that "fuel handling experience in the U. S., going back to 1959, has not revealed any instance where Zircaloy-clad uranium oxide fuel has undergone observable corrosion or other chemical degradation in pool storage. This favorable experience is corroborated by experience in other countries with the following maximum pool residence time for Zircaloy-clad fuel as of late 1977: Canada, 14 years; United Kingdom, 11 years; Belgium (MOL), 10 years; Japan, 9 years; Norway, 9 years; Karlsruhe, Germany (WAK), 7 years; Sweden, 5 years."

Finally, I would not expect the additional storage capacity to make the eventual removal from the pool of the spent fuel assemblies any more difficult due to corrosion of the fuel cladding.

Robert W. Calder

Robert W. Calder

DATED: May 11, 1979

Signed and sworn to before me by Robert W. Calder this [illegible] day of May, 1979.

[illegible]

Notary Public

My commission expires

[illegible]

STATEMENT OF PROFESSIONAL QUALIFICATIONS
OF ROBERT W. CALDER
SUPERVISOR - ENGINEERING SERVICES
Vepco

Mr. Calder's technical experience includes 9 years in the field of materials engineering, of which 7 years have dealt with nuclear power plants.

Mr. Calder joined Martin Marietta Research Institute for Advanced Studies in 1969 and conducted basic research in the areas of physical and mechanical metallurgy of high temperature alloys.

Mr. Calder joined Westinghouse Bettis Atomic Power Laboratory in 1973. As a metallurgical engineer he worked in the construction of Naval Nuclear Power Plants. His work included destructive and nondestructive evaluation of zircaloy-clad nuclear fuel elements. In 1975, he transferred to the Naval Reactors Facility where he qualified as a nuclear plant engineer and was responsible for plant testing and maintenance.

Mr. Calder joined Virginia Electric and Power Company in 1977 as an engineer in the Nuclear Engineering Services Department. His present title is Supervisor - Engineering Services.

EDUCATION

B.S., Metallurgical Engineering, Grove City College
M.S., Material Science, University of Maryland
M.B.A., Business Administration, University of Pittsburgh

352 353

UNITED STATES OF AMERICA
NUCLEAR REGULATORY COMMISSION

BEFORE THE ATOMIC SAFETY AND LICENSING BOARD

In the Matter of VIRGINIA ELECTRIC AND POWER COMPANY (North Anna Power Station, Units 1 and 2)	Doc. Nos. 50-338SP 50-339SP Proposed Amendment to Operating License NPF-4

AFFIDAVIT OF H. STEPHEN McKAY

My name is H. Stephen McKay. An accurate statement of my professional qualifications is attached to this Affidavit. I am the Project Engineer responsible for the design and installation of the high-density spent fuel racks for North Anna 1 and 2. I am familiar with the design and technical analyses of those racks that have been done.

Attached to this Affidavit is a copy of the document that Vepco has submitted to the NRC in support of its application to install and use the high-density racks. It is entitled "Summary of Proposed Modifications to the Spent Fuel Storage Pool Associated with Increasing Storage Capacity." Certain portions of this Summary have recently been amended; the amended portions are indicated by a vertical line in the right-hand margin. I am familiar with the contents of this Summary. It is true and correct to the best of my knowledge

and belief. I should like to supplement that Summary in the following respects. All references below are to the paragraph numbers of "Vepco's Statement of Material Facts as to which There is No Genuine Issue to be Heard."

The fuel pool heat exchangers and circulating pumps are located in the fuel building (¶ 3).

Each of the spent fuel pit pumps is connected to a separate, independent Class I power supply (¶ 7).

It will require a maximum of 12 gpm of evaporation to dissipate the additional heat discharged to the environment because of the proposed modification (¶ 15).

To determine the design basis heat load, a stretch rating of 2900 MWt for full power was assumed (¶ 18e).

The failure analysis of the spent fuel cooling system confirms that boiling and any adverse effects are prevented even in the event of a postulated failure of a spent fuel cooling pump or a spent fuel pool heat exchanger (¶ 27).

The component cooling water temperature could get as high as 113.2°F in the unlikely event of a LOCA in Unit 3 or 4, but the pool temperature would still be less than 177.5°F, the temperature that was used for the structural analysis of the spent fuel pool (¶ 33).

Unit 1 control board instruments and alarms include spent fuel pit temperature alarms at greater than 140°F and greater than 170°F (¶ 38b).

The spent fuel pool heat exchangers transfer the heat from the spent fuel pool water to the component cooling water (or, in an emergency, to the service water), and the component cooling water transfers its heat to the service water (¶ 12).

The service water, in turn, goes to the Service Water Reservoir, where the heat is transferred to the atmosphere (¶ 13).

Acceptable and appropriate engineering techniques were used to calculate the fuel pool temperatures (¶ 19).

The second heat exchanger would be required for only a period of 4-5 days, and only if a highly unlikely sequence of events were to occur:

a. Unit 1 refueled 45 days before the event;
b. Unit 2 just defueled;
c. Unit 3 or 4 loss-of-coolant accident (LOCA); and
d. Other unit cooldown (¶ 22).

The spent fuel pit is a reinforced concrete, seismic Class I structure lined with stainless steel plate a minimum of 1/4 inch thick. If the integrity of the 1/4 inch thick stainless steel liner were violated, water could enter the channels behind the liner. These channels are connected to a common drain point, which is the fuel building sump. In the event of a leak into one of these channels, water would rise in the fuel building sump, the sump pump would go on, and an alarm would sound in the control room. If the puncture were at a point other than the channels and the fuel pit water were somehow to pass through the liner and reinforced concrete, it would reach the foundation material below, which is virtually impenetrable (¶¶ 43-47).

Analysis of the thermal-hydraulic characteristics of the high-density racks have been performed using techniques that are generally accepted in the engineering community (¶ 50).

During the full core offload (abnormal case) with the bulk pool temperature at 170°F, the maximum temperature of the water exiting from a storage location is less than 197°F, which is 44°F below the local saturation temperature (boiling point) of 241°F (¶ 53).

Because of the long half-life of Krypton-85 (10.76 years), Kr-85 levels remain in older fuel; however, the thermal driving force required to cause its diffusion in defective fuel is greatly reduced (¶ 67).

Three 100% capacity purification pumps take suction at two permanently installed skimmers and pump water to a demineralizer and filters located in the auxiliary building (¶ 72).

The radiation levels of the demineralizers (which are shielded in cubicles) are usually from 1 to 4 R/hr (¶ 77). The filters (which are similarly shielded) are normally changed because of high pressure drop and usually have radiation levels of about 100 mR/hr (¶ 78).

Any increase in the liquid or gaseous radioactive emissions from North Anna 1 and 2 resulting from the proposed modification are expected to be negligible (¶ 85). They will not violate NRC regulations, either during normal operation of the expanded fuel pool or under postulated fuel handling accidents (¶ 86).

Vepco's seismic analysis of the new spent fuel storage racks and the pool structure shows that the racks can be accommodated by the structure during a seismic event (¶ 92).

The techniques used by Vepco for analyzing the structural integrity of the fuel racks under normal, abnormal, and seismic loads are generally accepted in the engineering community (¶ 97).

Criticality calculations show subcriticality maintained with a fuel assembly lying across the top of a rack or next to a rack (¶ 108).

With the normal concentration of boric acid in the pool water, criticality cannot be attained with any possible array of fuel assemblies (¶ 109).

The accident defined as the dropping of a spent fuel assembly onto the spent fuel pit floor and the resultant rupture of the cladding of all the fuel rods in the assembly has been analyzed in ¶ 15.4.5.1 of the FSAR (¶ 111). The analysis was done in accordance with NRC Safety Guide 25 (¶ 112).

The analysis of the fuel-drop accident shows that the accident would not result in excessive radiation exposure at

the site boundary, that is, in exposures exceeding the guidelines of 10 CFR Part 100 (¶ 113). Assuming as a worst case that the cladding of all rods in one entire fuel assembly fails, the offsite doses would not exceed the limits of 10 CFR Part 100 (¶ 114).

An analysis of the effect of a small tornado missile has been performed in § 15.4.5.2.4 and § 9.1.2 of the FSAR using accepted engineering techniques (¶ 115). Stored fuel in the spent fuel pit is protected from horizontal missiles by the thick reinforced concrete walls of the pit, which extend 20 feet, 10 inches above grade (¶ 116). The building geometry protects the fuel elements from direct impact of missiles with angles of approach up to approximately 45°F above the horizontal (¶ 117).

An analysis of the risk of turbine missiles has been done and is described in § 10.2.1 of the FSAR (¶123). The FSAR turbine missile analysis was done with appropriate and sound calculational techniques (¶ 124). The FSAR analysis shows that the risk of unacceptable damage to the fuel building is 0 for low-trajectory turbine missiles, 1.3139×10^{-13} per unit per year for high-trajectory missiles at design overspeed, and 1.3235×10^{-10} per unit per year for high-trajectory missiles at destructive overspeed (¶ 125). The turbine missile analysis is not changed by the proposed modification (¶ 126).

The amount of additional radiation to which the fuel would be exposed is insignificant in comparison to the levels in the reactor core during power operation (¶ 128).

The zircaloy of the fuel and the 304 stainless of the fuel racks and pool liner are the same material no matter whether the high-density racks or the low-density racks are used (¶ 132). The proposed modification is not expected to make the eventual removal from the pool of the spent fuel assemblies any more difficult; to the contrary, after extended

storage the radiation levels and therefore the heat generated will have decayed to lower levels, so the handling and shipment of the assemblies will be easier (¶ 133).

The spent fuel pool purification system is adequate to remove any potential incremental impurities resulting from the proposed modification; most of the impurities are released during refueling, and the number of fuel movements will be no greater with the high-density racks than with the low-density racks (¶ 134).

The exposure to radiation at Surry will probably slightly increase when additional fuel assemblies are stored; however, because of the age of the earlier-stored fuel by the time the additional fuel reaches the pool, the increase should not be significant (¶ 142).

Even if the exposure were doubled, to about 3 mR/hr, the exposure is a relatively minor contributor to the overall exposure at the station (¶ 143).

Iodine-131 releases will not be significantly increased by the expansion of the fuel storage capacity, because the inventory of the I-131 (which has a half-life of 8 days) in the fuel will decay to negligible levels (¶ 149).

Experience at Surry indicates negligible levels of I-131 in the pool water; I-131 is noted only during refueling (¶ 150).

Based on experience at the Surry Power Station, the radiation exposure is relatively low, approximately 150 mR for a filter change (¶ 153).

There have been no overexposures associated with the Surry fuel pool (¶ 156).

The pool walls and the pool liner cannot be reworked with spent fuel in the pool (¶ 171). There is not enough time to expand the spent fuel pool before the first refueling, and so the spent fuel would have to be transferred to other storage until the work was done (¶ 172). This would require finding

another licensed storage facility and double handling the fuel (¶ 173).

The North Anna 3 and 4 fuel pool would have to be licensed by the NRC before it could be used to store spent fuel (¶ 178).

The high-density fuel racks have already been fabricated and are at North Anna waiting to be installed (¶ 179).

In the event of a leak from the spent fuel pool, several station systems are capable of providing make water until the leak could be repaired. Because the wate vel in the pool would be maintained despite the leak, there would be no temperature increase in the pool. Instead, there would probably be a decrease due to the addition of cooler make-up water.

The building geometry protects the fuel elements from direct impact of missiles with angles of approach up to approximately 45° above the horizontal (¶ 117). According to technical papers by D. R. Miller, W. A. Williams, and T. L. Doan, large missiles such as utility poles and automobiles (which are the design tornado missiles for North Anna 1 and 2) lack sufficient lift or velocity to clear a height of 25 feet (¶ 118). These could not, therefore, strike the fuel elements (¶ 119).

The spent fuel elements would be protected from lighter missiles by the water covering the storage racks in the pool (¶ 120). According to the paper by T. L. Doan, small fast-moving missiles traveling downwards would impact only one fuel assembly (¶ 121). A tornado missile impacting the spent fuel would not result in radiation doses that exceed the limits of 10 CFR Part 100 (¶ 122).

H. Stephen McKay

H. Stephen McKay

DATED: May 11, 1979

352 360

Signed and sworn to before me by H. Stephen McKay this 11th day of May, 1979.

Notary Public

My commission expires January 28, 1981

352 361

STATEMENT OF PROFESSIONAL QUALIFICATIONS OF H. STEPHEN McKAY

My name is H. Stephen McKay. My business address is P.O. Box 26666, Richmond, Virginia 23261. I am an Associate Engineer with Virginia Electric and Power Company. I am presently Project Engineer for the North Anna 1 and 2 high-density spent fuel storage rack project. Formerly I was the Project Engineer for the high-density spent fuel storage project at the Surry Power Station, Units 1 and 2.

I hold a B.S. degree in physics from Moravian College, Bethlehem, Pennsylvania (1972), an M.S. degree in physics from Indiana University of Pennsylvania, Indiana, Pennsylvania (1974), and a Master of Engineering degree in nuclear engineering from the University of Virginia, Charlottesville, Virginia (1976).