JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 2

DOCKET NO. 50-364

SPENT FUEL POOL
MODIFICATION

December 1981

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I. INTRODUCTION

This report provides information required by the Nuclear Regulatory Commission (NRC) in support of an application for installation of high-density poison spent fuel storage racks at the Joseph M. Farley Nuclear Plant-Unit 2. The report has been prepared using the guidance of the NRC position paper entitled "OT Position for Review and Acceptance of Spent Fuel Pool Storage and Handling Applications" dated April 14, 1978, as amended by NRC letter dated January 18, 1979. Sections III through V of the report are consistent with the section/subsection format and content of the NRC position paper, sections III through V. The extent of information provided involves an overall description of the spent fuel rack system and addresses disciplines such as nuclear and thermal hydraulics, mechanical, material, structural, and environmental.

The overall description (section II) provides a detailed description of the high-density poison racks and their location relative to other systems.

The nuclear and thermal hydraulic aspects of the report (section III) address the neutron multiplication factor, considering normal storage and handling of spent fuel as well as postualted accidents with respect to criticality and the ability of the spent fuel pool cooling system to maintain sufficient cooling.

Mechanical, material, and structural aspects (section IV) involve the capability of the fuel assembly, storage racks, and spent fuel pool system to withstand effects of natural phenomena and other service loading conditions.

The environmental aspects of the report (section IV) concern the thermal and radiological release from the facility under normal and accident conditions. This section also addresses the occupational radiation exposures, generation of radioactive waste, need for expansion, commitment of material and nonmaterial resources, and a cost-benefit assessment.

In conclusion, this report documents the compliance of the Farley-Unit 2 high-density poison spent fuel rack installation with the NRC requirements, as specified in the position paper referenced above.

II. OVERALL DESCRIPTION

Low-density spent fuel racks designed by Westinghouse, with a large pitch (13 inches center-to-center), are presently installed in the Farley-Unit 2 dry, clean spent fuel pool. Due to an increased demand for spent fuel assembly storage space Alabama Power Company has recognized a need to more efficiently utilize the available space in the existing pool. This can be accomplished by re-racking the Unit 2 pool prior to the scheduled November 1982 refueling outage with high-density poison racks, designed by PaR Systems, with a smaller pitch (10.75 inches center-to-center). This will increase storage capacity from 675 to 1407 cells.

Plan and elevation views of the containment and auxiliary building, containing the spent fuel pool, are shown in figures II-1 and II-2. The plan view and general arrangement of the high-density storage system is shown in figure II-3, and storage racks details are illustrated in figures II-4 and II-5.

The spent fuel rack modules are free-standing and free to move on the pool liner floor during a seismic event. The module is composed of poison cannisters with a bottom grid. Except for the neutron absorber (vented boraflex) and threaded foot (17-4 PH alloy) all other rack materials are fabricated using 304 stainless steel alloy.

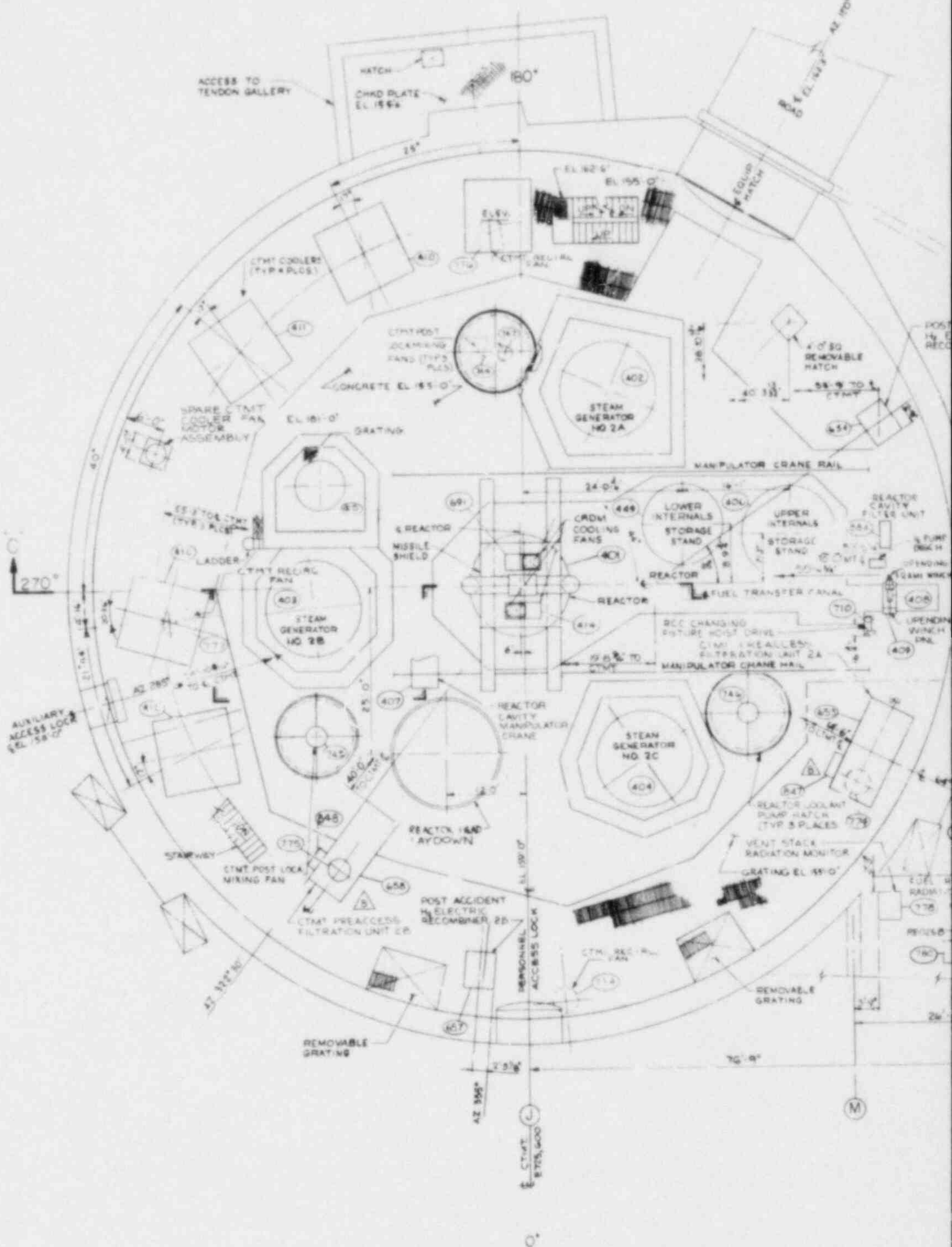
Three basic module configurations are planned. Their dimensions are 6 x 7, 7 x 7, and 7 x 8. The combined capacity will provide 1407 cells with the following breakdown:

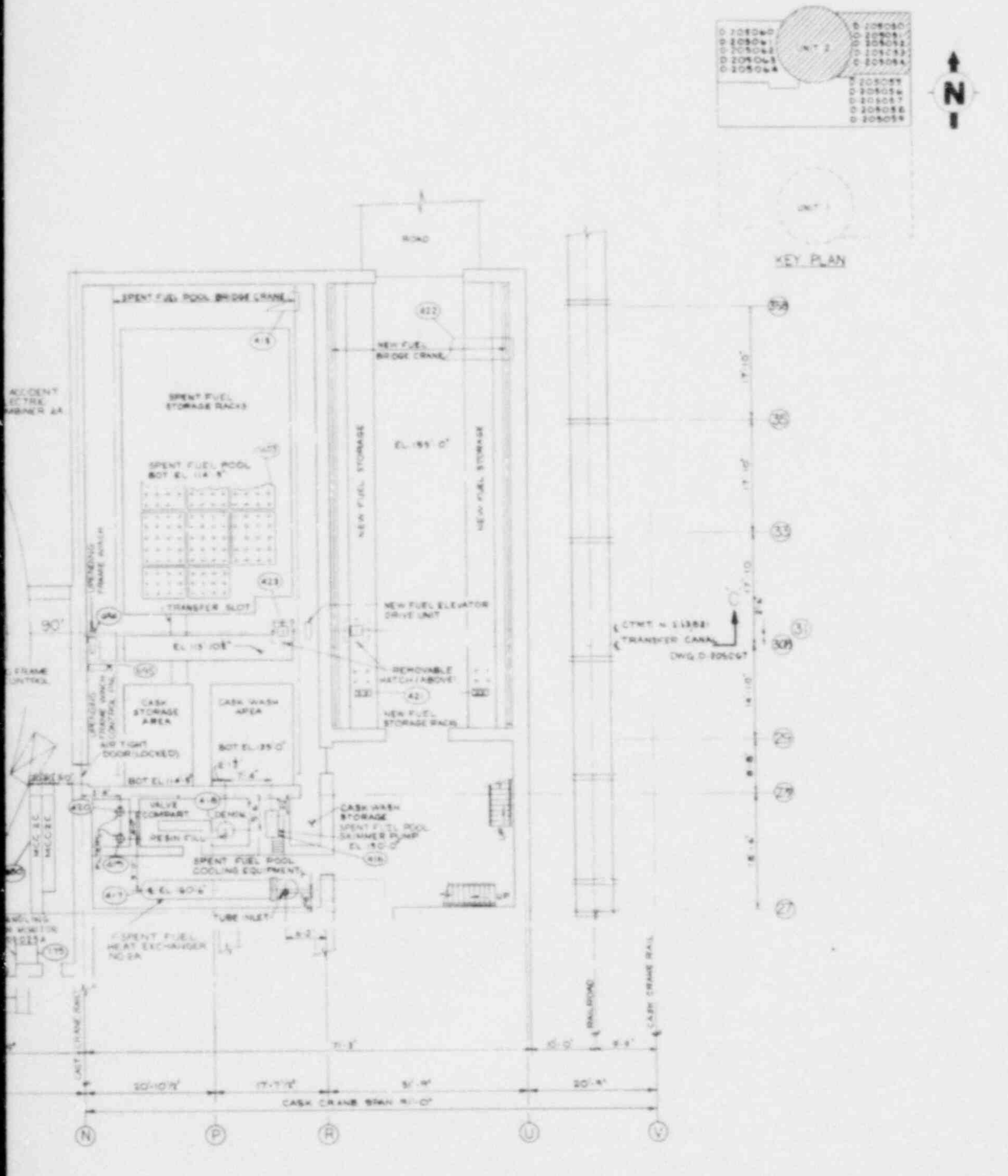
Module				
Configuration	Capacity	Quantity	Total Cavities	Individual Weight
6 x 7	42	2	84	14,300
7 x 7	49	19	931	16,700
7 x 8	56	7	392	19,040
		Total:	1,407	

All weld techniques and processes will produce clean, spatter-free welds with good penetration and no slag formation. The outer canister wrapper will be gas tungsten arc spot welded (TIG) together and also TIG spot-welded onto the canister. The indivudual cavities will be welded on special fixtures to maintain required squareness with dimensional tolerance. The module cavities and bottom grid are then welded together using a special fixture to assure the assembly is square, properly aligned, and the center-to-center spacing is accurate.

After functional testing (dummy bundle drag test), the racks will be cleaned and completely wrapped with reinforced plastic and skidmounted. The racks will be covered with tarp and shipped by motor freight to the plant site. The spent fuel racks are designed to withstand shipping, handling, normal impact loads (impact and dead loads of fuel assemblies) as well as safe shutdown earthquake (SSE) and operation basis earthquake (OBE) seismic loads meeting ASME Section III, Appendix XVII which is equivalent to AISC Section 5 requirements. The racks are also designed to meet Category I seismic requirements of Regulatory Guide 1.13.

In summary, spent fuel will be stored in the spent fuel pool in 10.75 inch center-to-center racks. The racks are composed of vented boraflex between individual austenitic stainless steel canisters, which are fastened together in a free-standing module. The module, maintaining a 10.75 inch center-to-center cell spacing with the neutron absorbing material, is sufficient to maintain a subcritical array.


ACCESS TO TENDON GALLERY
HATCH
CHKD PLATE EL 155'6"
180°
EQUIP HATCH
EL 162'6"
EL 155'-0"
ELEV.
CTMT COOLERS (TYP 4 PLCS)
CTMT POST LOCA MIXING FANS (TYP 3 PLCS)
CONCRETE EL 155'-0"
STEAM GENERATOR NO 2A
4'-0" SQ REMOVABLE HATCH
SPARE CTMT COOLER FAN MOTOR ASSEMBLY
EL 181'-0"
GRATING
MANIPULATOR CRANE RAIL
CRDM COOLING FANS
LOWER INTERNALS STORAGE STAND
UPPER INTERNALS STORAGE STAND
REACTOR CAVITY FILTER UNIT
℄ REACTOR
MISSILE SHIELD
LADDER
CTMT RECIRC FAN
270°
STEAM GENERATOR NO 2B
REACTOR
FUEL TRANSFER CANAL
RCC CHANGING FIXTURE HOIST DRIVE
CTMT PREACCESS FILTRATION UNIT 2A
MANIPULATOR CRANE RAIL
UPENDING WINCH PNL
AUXILIARY ACCESS LOCK EL 158'-0"
REACTOR CAVITY MANIPULATOR CRANE
STEAM GENERATOR NO. 2C
REACTOR COOLANT PUMP HATCH (TYP 3 PLACES)
VENT STACK RADIATION MONITOR
GRATING EL 159'0"
REACTOR HEAD LAYDOWN
STAIRWAY
CTMT POST LOCA MIXING FAN
CTMT PREACCESS FILTRATION UNIT 2B
POST ACCIDENT H2 ELECTRIC RECOMBINER 2B
PERSONNEL ACCESS LOCK
CTMT RECIRC FAN
REMOVABLE GRATING
REMOVABLE GRATING
J
M
0°



Alabama Power

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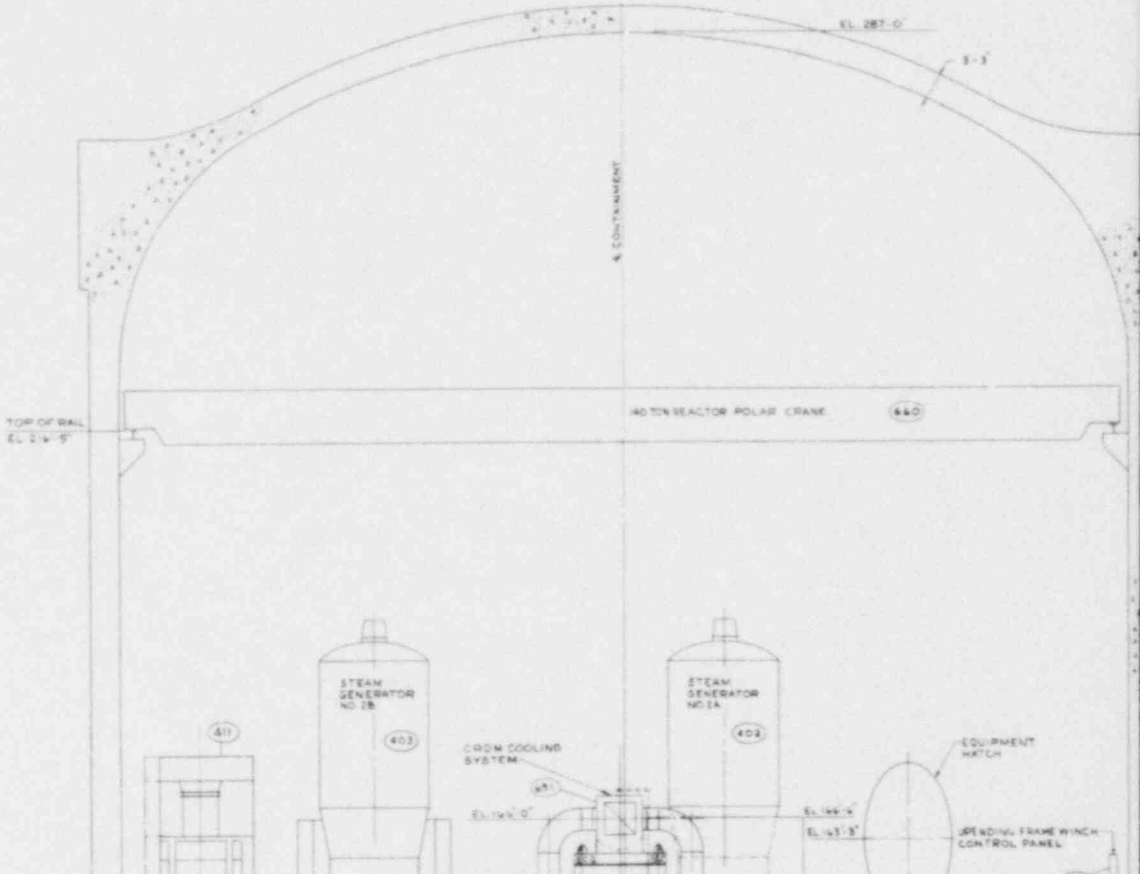
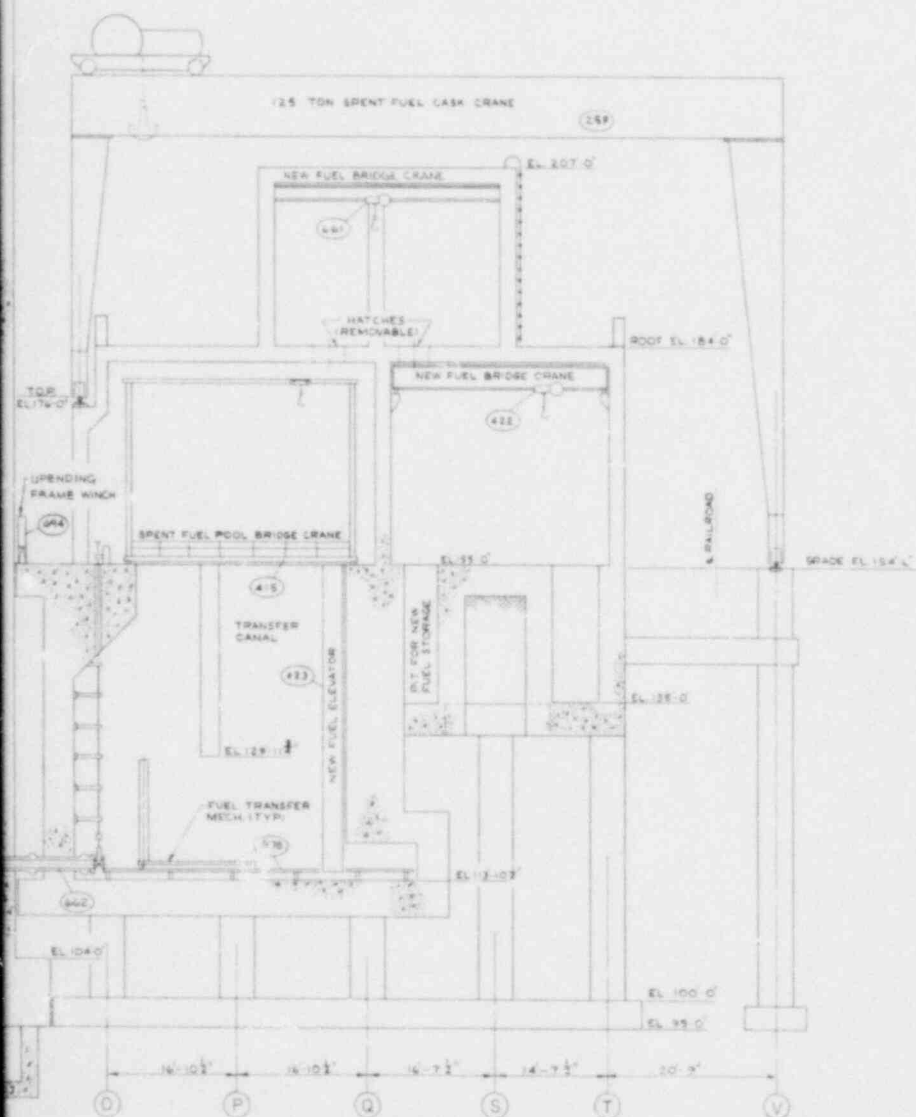
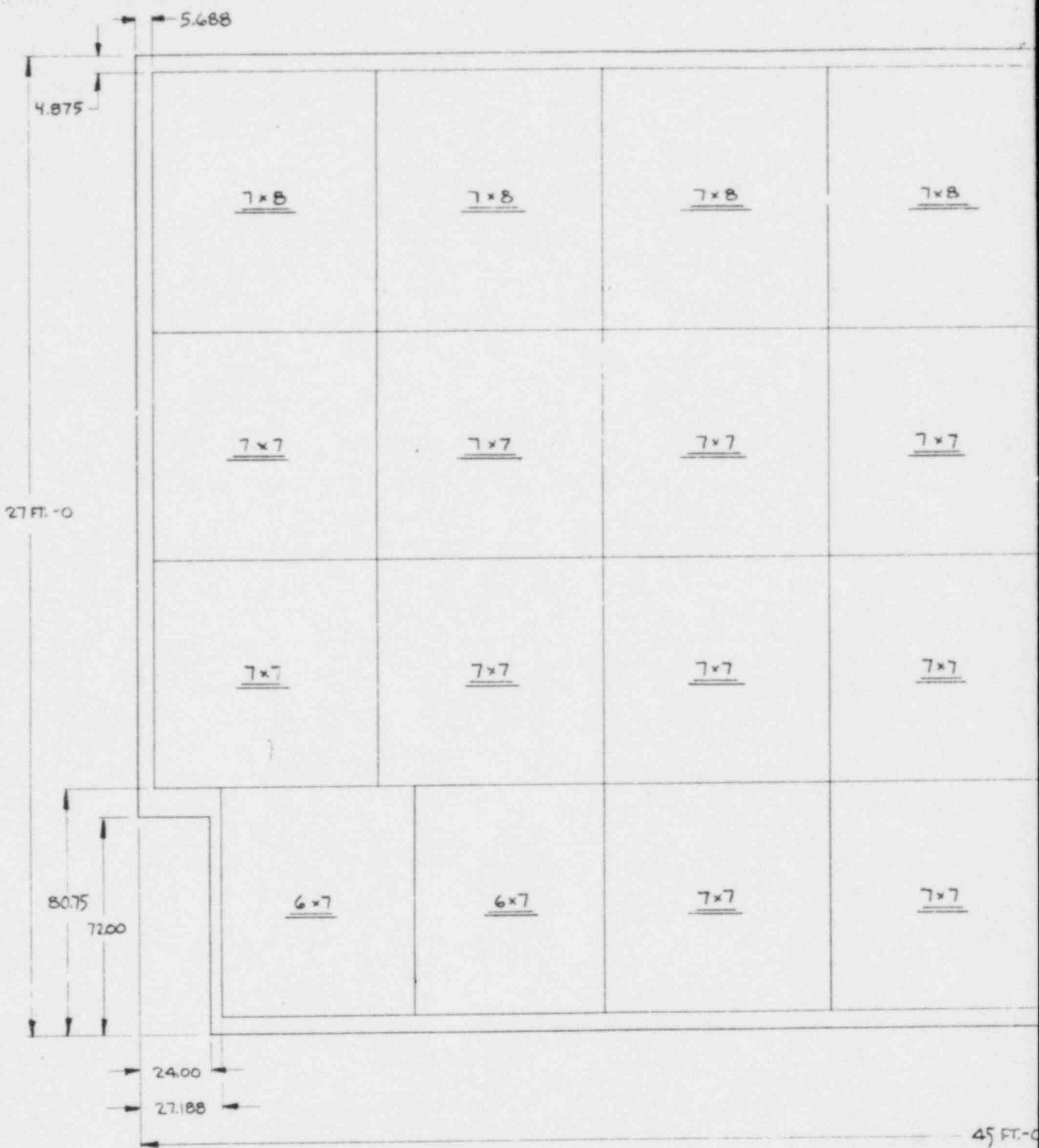
EQUIPMENT LOCATION PLAN
EL. 155 FT.

FIGURE II-1

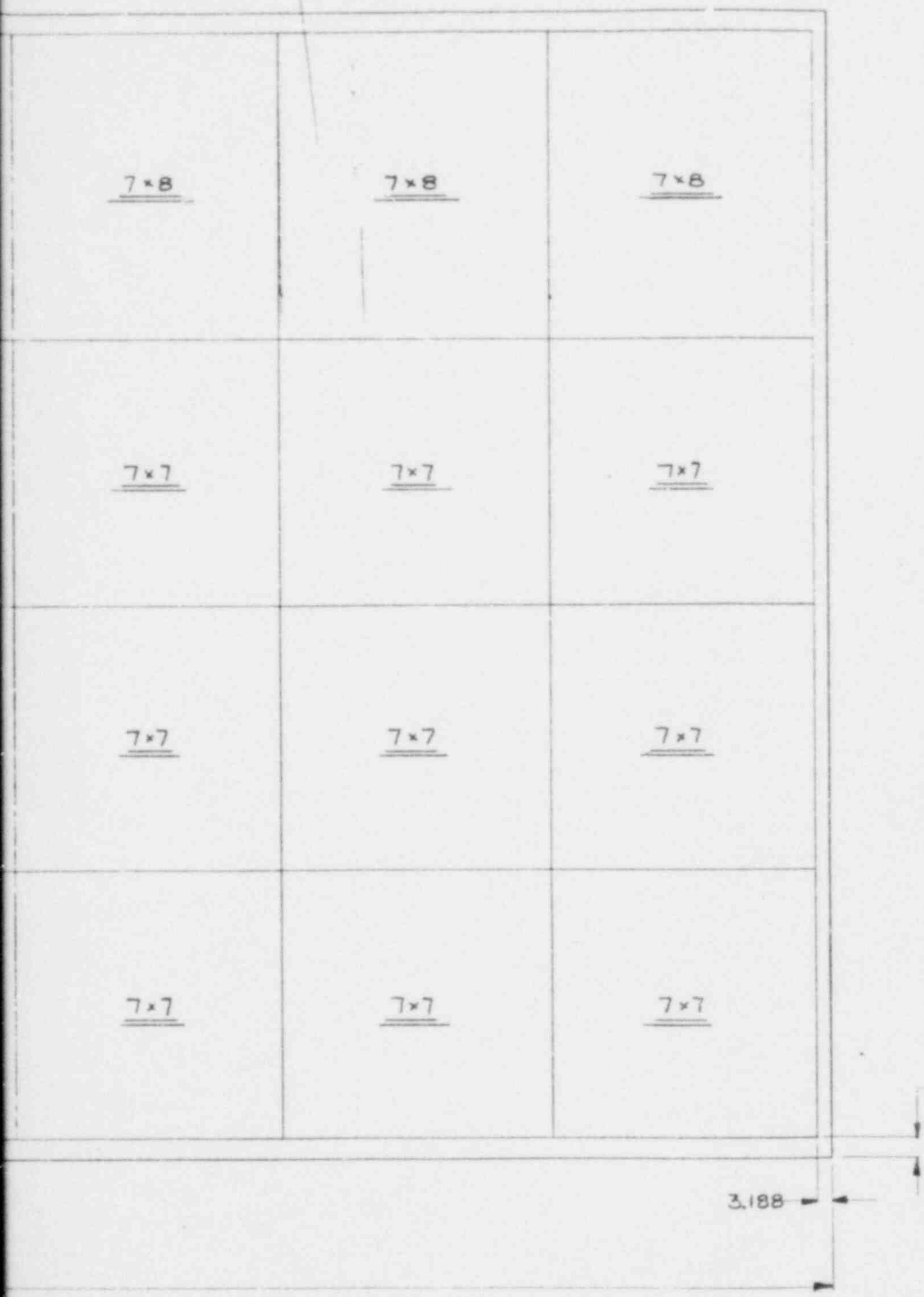

EL. 287'-0"
140 TON REACTOR POLAR CRANE
TOP OF RAIL
STEAM GENERATOR NO. 2B
STEAM GENERATOR NO. 1A
CRDM COOLING SYSTEM
EQUIPMENT HATCH
UPENDING FRAME WINCH CONTROL PANEL
OPERATING FLOOR EL. 155'-0"
ACCUMULATOR NO. 2B
REGENERATIVE HT EXCH
REACTOR COOLANT PUMP NO. 2B
REACTOR
EL. 129'-0"
FUEL TRANSFER TUBE
PASSAGE WAY
FINISHED FLOOR EL. VARIES
PRESSURIZER RELIEF TANK
REACTOR COOLANT DRAIN TANK
TENDON ACCESS GALLERY
EL. 87'-0"
VALVES
SECTION "C-
LOOKING NOR

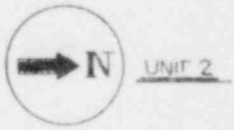


Alabama Power	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 2 SPENT FUEL POOL MODIFICATION	EQUIPMENT LOCATION. SECTION C-C FIGURE II-2



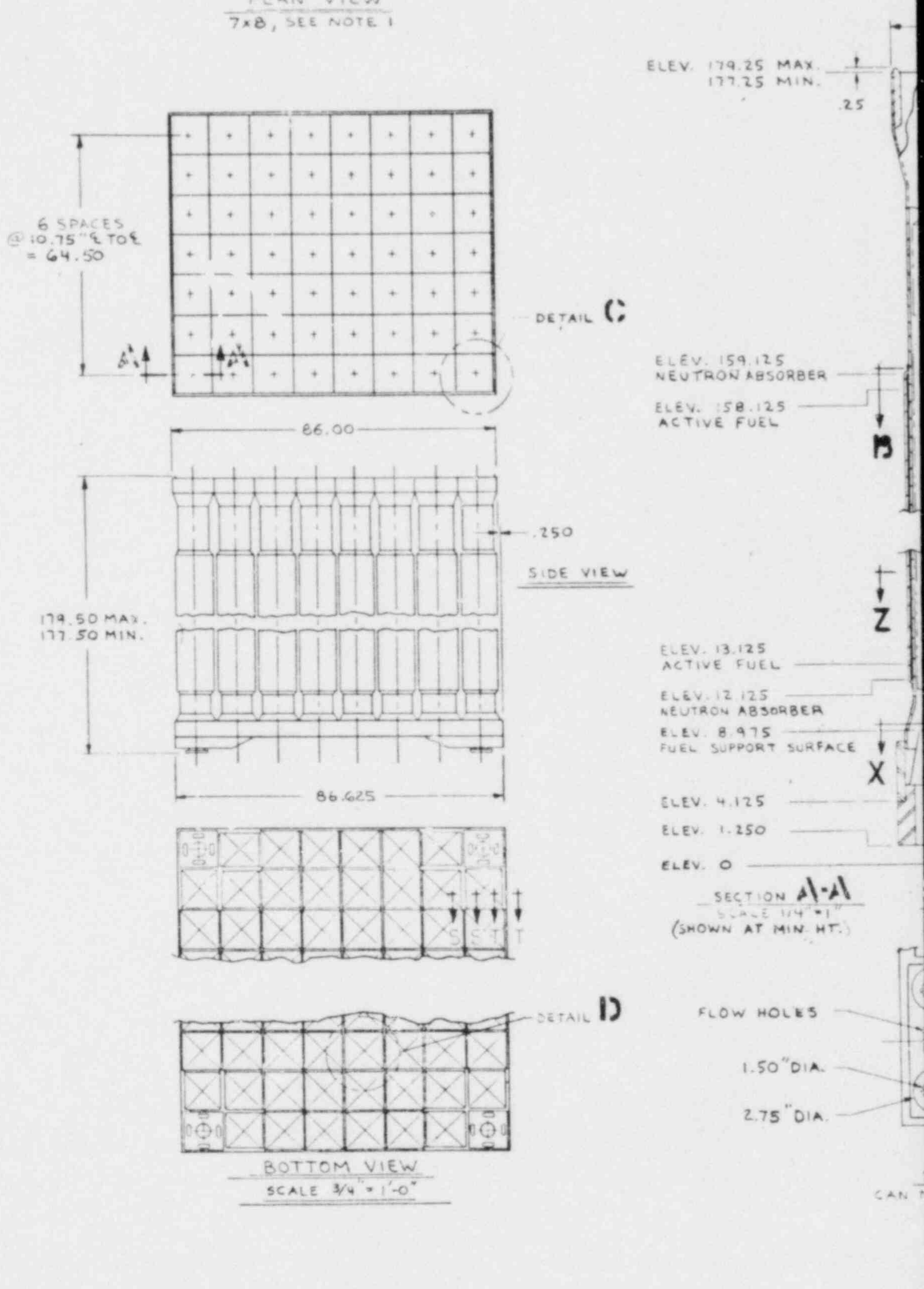
FULL POOL		
MODULE	NO.	CAVITIES
7 × 8	7	392
7 × 7	19	931
6 × 7	2	84
TOTAL	28	1407

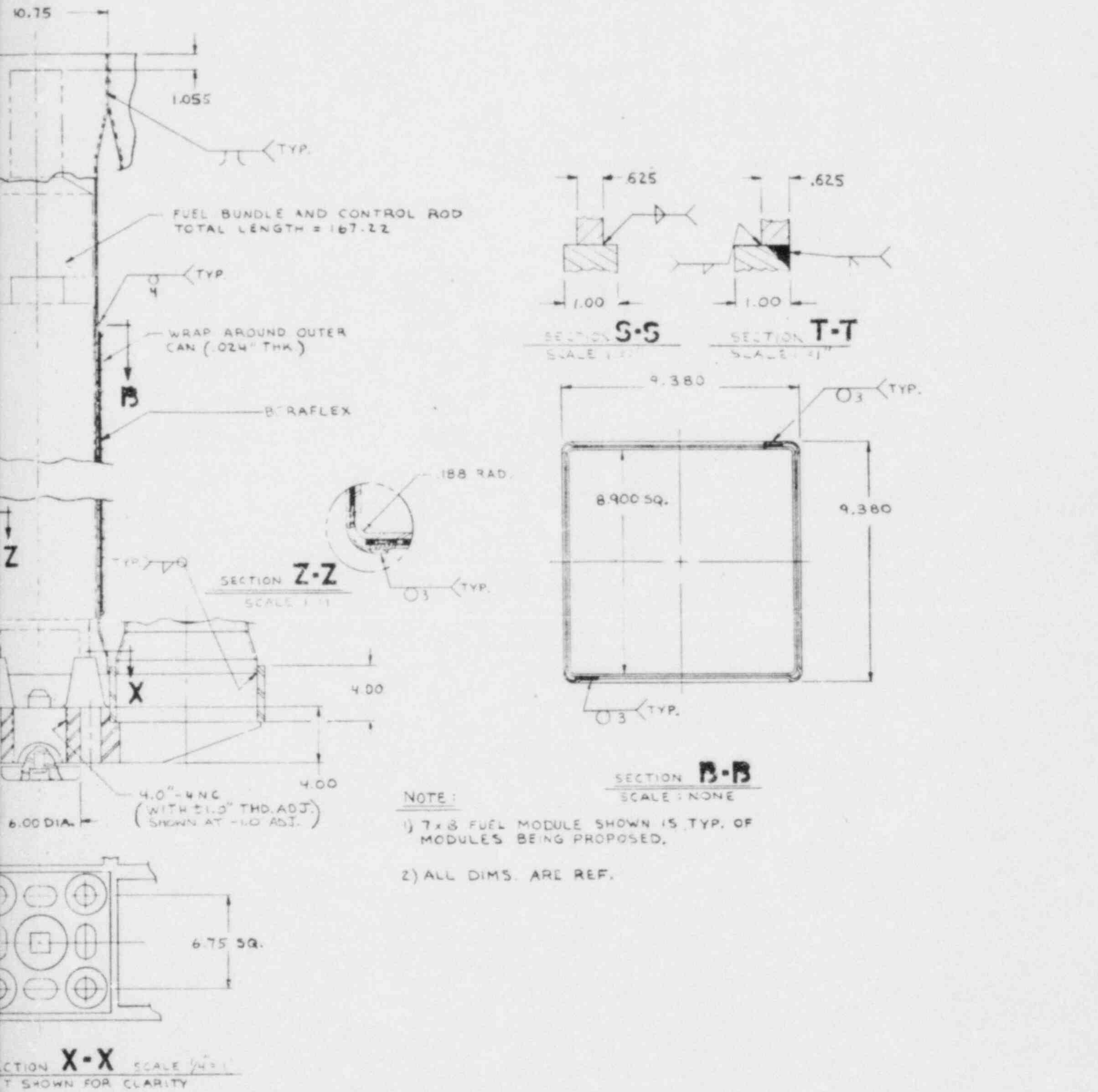




NOTES:
1. CAVITY PITCH = 10.75"
2. 5/8" GAP BETWEEN TOP OF MODULES
3. LAYOUT APPLICABLE TO UNITS 1 OR 2.

Alabama Power	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 2 SPENT FUEL POOL MODIFICATION	POOL LAYOUT
		FIGURE II-3


PLAN VIEW
7x8, SEE NOTE 1
6 SPACES
@ 10.75" ℄ TO ℄
= 64.50
DETAIL C
A-A
86.00
.250
SIDE VIEW
179.50 MAX.
177.50 MIN.
86.625
S S T T
DETAIL D
BOTTOM VIEW
SCALE 3/4" = 1'-0"
ELEV. 179.25 MAX.
177.25 MIN.
.25
ELEV. 159.125
NEUTRON ABSORBER
ELEV. 158.125
ACTIVE FUEL
B
Z
ELEV. 13.125
ACTIVE FUEL
ELEV. 12.125
NEUTRON ABSORBER
ELEV. 8.975
FUEL SUPPORT SURFACE
X
ELEV. 4.125
ELEV. 1.250
ELEV. 0
SECTION A-A
SCALE 1/4" = 1"
(SHOWN AT MIN. HT.)
FLOW HOLES
1.50" DIA.
2.75" DIA.

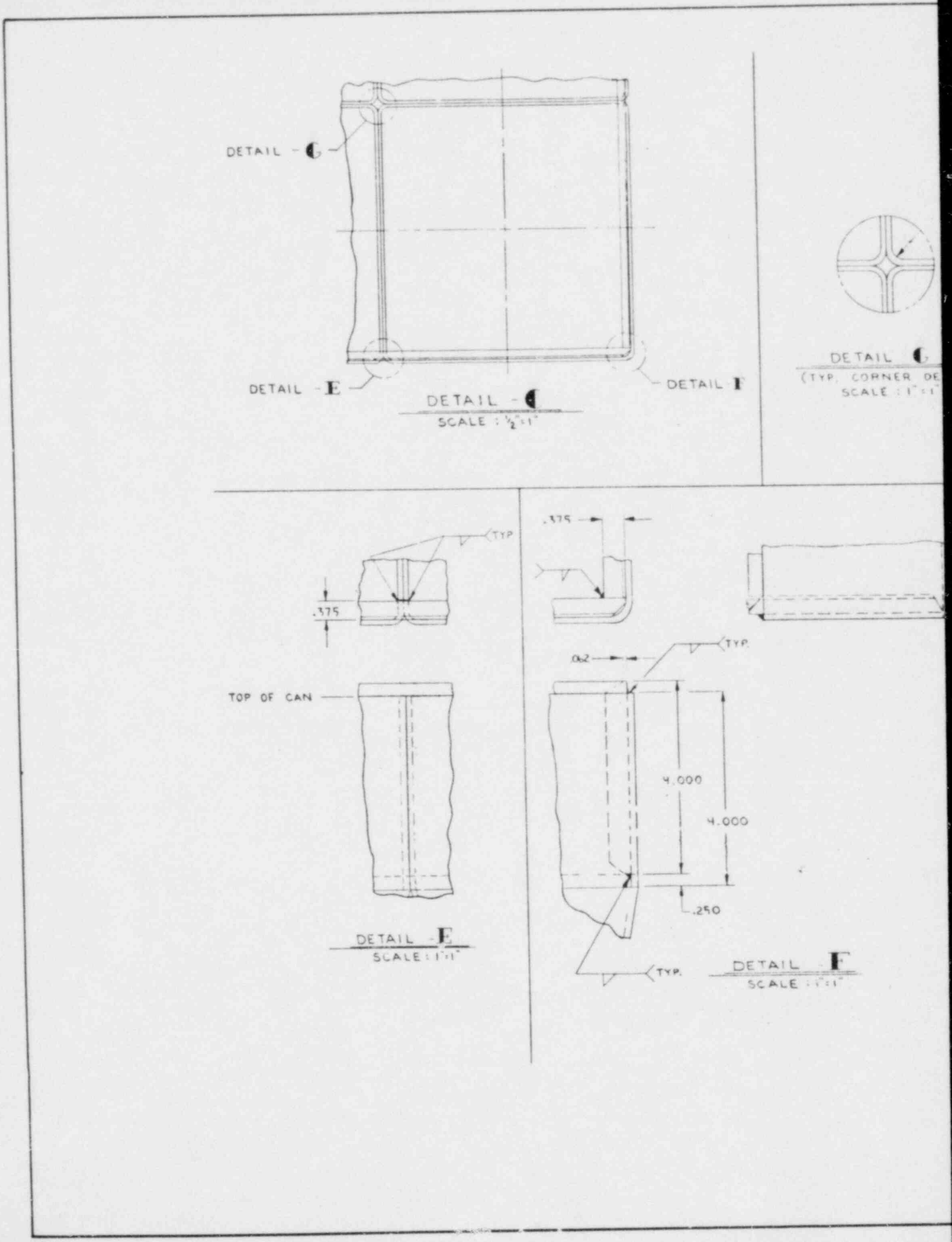


NOTE:

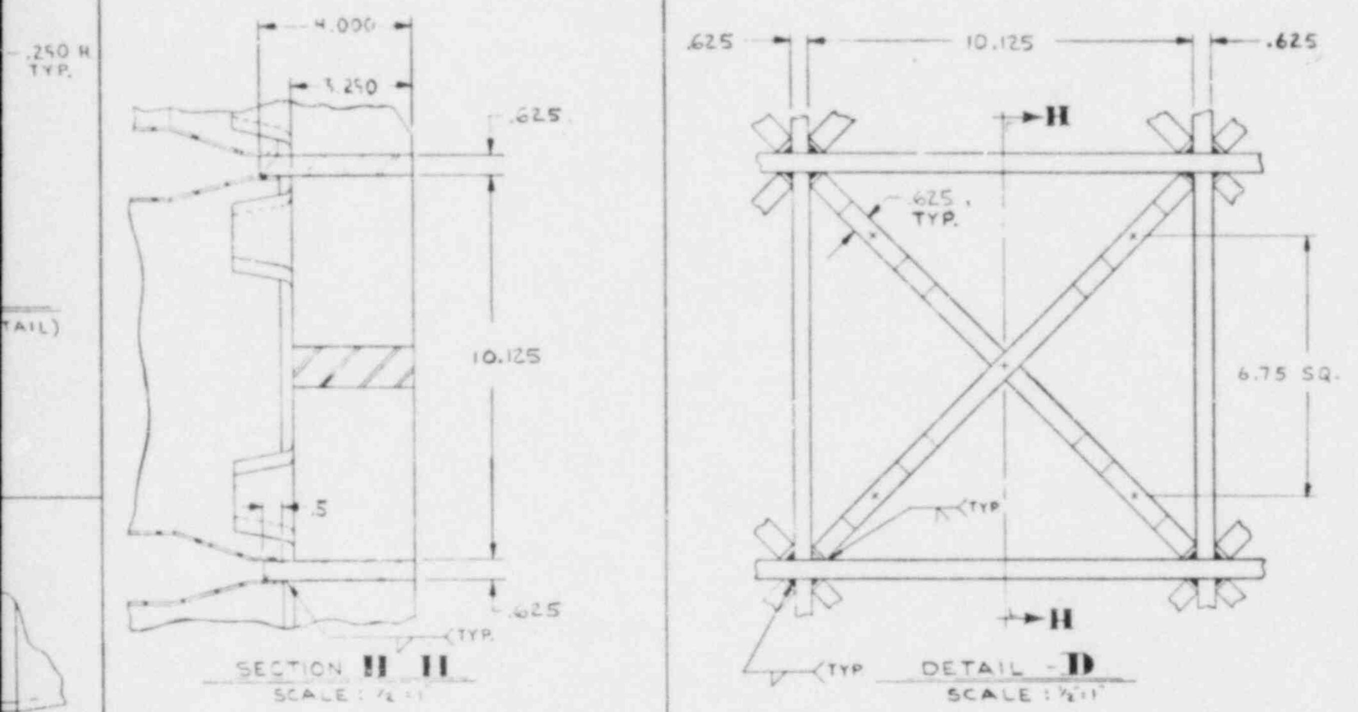
1) 7×8 FUEL MODULE SHOWN IS TYP. OF MODULES BEING PROPOSED.

2) ALL DIMS. ARE REF.

Alabama Power	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 2 SPENT FUEL POOL MODIFICATION	FUEL MODULE
		FIGURE II-4


DETAIL - G
DETAIL - E
DETAIL - F
DETAIL - C
SCALE : 1/2"=1"
DETAIL G
(TYP. CORNER DE
SCALE : 1"=1"
.375
TYP
TOP OF CAN
DETAIL - E
SCALE : 1"=1"
.375
TYP.
.062
4.000
4.000
.250
TYP.
DETAIL - F
SCALE : 1"=1"

4440-1



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FUEL MODULE

FIGURE II-5

III. NUCLEAR AND THERMAL-HYDRAULIC CONSIDERATIONS

III.1. Neutron Multiplication Considerations

The effective neutron multipliation factor (k_{eff}) is calculated, assuming the following conditions in the spent fuel pool.

III.1.1 Normal Storage

III.1.1.a The Farley high-density spent fuel racks are designed to store Westinghouse 17 x 17 fuel assemblies with a maximum U-235 enrichment of 4.3 weight percent without control rods or any noncontained burnable poison. The fuel is assumed to be fresh and is at the most reactive point of its life.

III.1.1.b The moderator used in the criticality analysis is pure water at the temperature within the spent fuel pool limits which yields the maximum reactivity.

III.1.1.c In the nominal case analysis, the rack array is assumed to be infinite in lateral extent.

III.1.1.d Sensitivity studies have been performed to obtain the reactivity effect of the mechanical tolerances.

III.1.1.e The analysis has taken credit for the neutron absorption by the control poison material (Boraflex), the stainless steel structural material of the spent fuel racks, and some of the structural materials of the fuel assemblies.

III.1.2 Postulated Accidents

The double contingency principle of ANSI N16.1-1975, which stipulates that two unlikely, independent, concurrent events are required to produce a criticality accident, is used.

III.1.2.(1)

a) Single assembly dropped on top of rack

   No adverse reactivity effect is expected from dropping a fuel assembly on top of a fully loaded storage rack during fuel handling because of the large water thickness (about 10 inches) existing between the top of the assemblies already inside the cavities and the dropped assembly resting on top of the rack. Moreover, the calculational model assumes an infinite fuel length in the axial direction.

b) Single assembly next to rack

The dropping of an assembly outisde the rack is a possible event because of the unobstructed water area existing between the periphery of the storage racks and the side walls of the pool. A conservative analysis was performed to evaluate this situation. The results indicate that with the presence of soluble boron in the pool water, as permitted by the double contingency rule, the dropping of fuel assembly next to the rack proper does not raise the k_{eff} value of the racks, with all uncertainties and biases included, to above 0.950.

III.1.2.(2)

Protection against a cask drop is assured by the Seismic Category I, CMAA Specification No. 70, Class A1, single failure-proof outdoor spent fuel cask crane, by the single failure-proof lifting device, and by the interlocks and administrative controls described in the Farley FSAR subsection 9.1.4.

A cask drop or tip into the spent fuel pool is also prevented by permanently installed rail stops and mechanical bumpers which prohibit cask crane travel over or into the vicinity of the spent fuel pool. The cask crane hook approach to the spent fuel pool, as shown in Farley FSAR figures 1.2-2 and 1.2-10, is limited to approximately 12 feet. Since the cask will not be handled in the vicinity of the spent fuel pool, the consequences of the cask drop are not affected by the increased storage of the spent fuel pool.

The spent fuel bridge crane, located inside the spent fuel pool building, is used for refueling operations. The spent fuel bridge crane is the only crane capable of handling objects over the spent fuel pool. When fuel assembies are stored in the spent fuel pool, the size of the load that can be handled over the spent fuel pool is limited to 3,000 pounds by Farley Unit 2 Technical Specification Section 3/4 9-7. Use of the spent fuel bridge crane is discussed in Farley FSAR subsections 9.1.2 and 9.1.4.

Subsection III.1.1.(1) of this submittal discusses the dropping of a fuel element on top of the racks or any other achievable abnormal location of a fuel assembly in the pool.

III.1.2.(3)

The exterior walls and roof slab of the spent fuel area are designed for tornado wind, differential pressure, and missile loadings on the basis that the area is fully enclosed. Farley FSAR subsection 3.3.2.1 specifies tornado design criteria for fully enclosed Category I structures. A tornado will not have an effect on the deformation and relative position of the fuel racks since the building structure will not be modified for the re-rack program, and all existing building analyses will remain valid and unchanged.

The design of the racks is seismically qualified, therefore, the earthquake effect on criticality is of no concern due to the structural acceptance of the rack.

III.1.2.(4)

Loss of all cooling systems will not result in criticality. In addition, the spent fuel pool cooling system is Seismic Category I and meets the single failure criteria.

III. 1.3 Calculation Methods

The criticality analysis employed two independent models to validate the results of the nominal geometry calculations.

The base calculational method employes the KENO-IV/AMPX model. The basic neutron cross-section data comes from the master library of AMPX - a 123 group GAM-THERMOS neutron library prepared from ENDF/B version II data. The NITAWL module of the AMPX program is used to perform a Nordheim integral treatment of the U-238 resonsances accounting for self-shielding effect. The working library produced by this process retains the 123 group energy structure and is used directly by KENO-IV. The KENO-IV/AMPX model has been benchmarked against the critical experiment data measured by Battelle Pacific Northwest Laboratories.

The diffusion theory model, which uses three codes, namely, CHEETAH-P, CORC-BLADE, and PDQ-7 is used for the validation process and is also used to perform all the sensitivity calculations. The model has been extensively tested through benchmarking calculations of measured criticals, as well as through core physics calculations for several operating reactors.

The final k_{eff} value for the Farley spent fuel racks is obtained by summing the k_{eff} value calculated by the KENO-IV/AMPX model, the calculational bias which was obtained from the benchmark work and the total uncertainty which was obtained by a statistical combination of the calculational and mechanical uncertainties. The calculational uncertainity is such that the true multiplication factor (k_{eff}) will be less than the calculated value with a 95 percent probability at a 95 percent confidence level.

III.1.4 <u>Rack Modification</u>

The spent fuel storage racks being supplied to Alabama Power Company are of new construction with no modifications to the existing racks or pool liner being required. The existing racks will be partially removed prior to installation of the new racks.

III.1.4.(a) The overall fuel assembly parameters for the Farley PWR 17 x 17 fuel assemblies are as follows.

Fuel assembly dimension	8.426 in. x 8.426 in.
Storage cell pitch	10.75 in.
Percent of total cell area occupied by a fuel assembly	61.4

III.1.4.(b) This section of the NRC position paper is not applicable since the new high-density racks utilize a neutron absorbing poison rather than stainless steel flux traps.

III.1.4.(c) Refer to III.1.4.(b) above.

III.1.4.d.(1) There are two poison plates separating every two adjacent fuel assemblies stored in the racks. The poison plates have a B-10 loading of (later) g/cm^2.

III.1.4.d.(2).a The analysis is based on an average enrichment of 4.3 weight/percent of U-235. The fuel loading is calculated to be 54.63 gram of U-235 per axial centimeter of fuel assembly. The reactivity sensitivity of enrichment is calcualted to be (later) delta k per gram U-235 change around the nominal fuel loading.

III.1.4.3.(2).b The nominal storage lattic pitch is 10.75 in. The pitch reactivity sensitivity is calculated to be (later) delta k per 0.1 in. change in pitch around the nominal value.

III.1.4.d.(2).c The B-10 loading in the poison plates is (later) g/cm^2. The reactivity sensitivity due to B-10 loading variation is calculated to be (later) delta k per 0.005 g/cm^2 change around the nomial loading.

III.1.5 Acceptance Criteria For Criticality

The acceptance criteria for criticality calculations is that k_{eff} be less than or equal to 0.95 including all uncertainties.

III.1.5(1) Neutron Absorber Verification

PaR Systems, operating under a Quality Assurance Program which meets 10 CFR 50 Appendix B, requires the poison manufacturer to produce his product under a program which also meets 10 CFR 50 Appendix B. A detailed specification is part of the purchase order. This specification covers the neutron absorber sheet requirements, material requirements, quality assurance program requirements, documentation requirements, etc. PaR requires the manufacturer to sumit his quality assurance program manual and operating procedures for approval before the start of production. PaR also audits the quality assurance program at the maufacturing facility at least once a year (or before the first order). After receipt of material, PaR reviews all documentation for conformance before incorporating the poison into the spent fuel racks. PaR maintains traceability of the poison material throughout the rack manufacturing process. Alabama Power Company or its agent is committed to periodically perform quality audits and inspection of the above described quality program.

III.1.5(2) Decay Heat Calculation for the Spent Fuel

The calculations for the amount of thermal energy that will have to be removed by the spent fuel pool cooling system are made in accordance with Branch Technical Position APCSB 9-2 entitled, "Residual Decay Energy for Light Water Reactors for Long Term Cooling." This Branch Technical Position is part of the Standard Review Plan (NUREG 78/087).

III.1.5.(3) Thermal Hydraulic Analysis of Spent Fuel Cooling

The computer code HPOOL is used to analyze the natural circulation cooling of the spent fuel under normal cooling conditions. HPOOL is a proprietary program of Nuclear Associates Incorporated (NAI). HPOOL calculates the pressure loss through a fuel assembly for a given flow rate. This pressure loss is compared with the buoyant head resulting from the difference between the average density of the fluid in the fuel channel and the average density of the fluid in the downcomer. The downcomer is the space between the wall of the pool and the racks. If the density difference results in a buoyant head greater than the pressure loss, the flow rate through the fuel assembly is increased and a new average density of the fluid is determined. This iterative process is continued until the buoyant head and pressure loss in the fuel assembly are equal. Using this flow rate, HPOOL determines the fuel temperature.

The computer code BPOOL is used to analyze the natural circualtion cooling of the spent fuel in the event of a loss of all external means of cooling for the spent fuel pool. BPOOL is a proprietary program of NAI. The code is based on the assumption that boiling takes place near the top of the fuel channel. BPOOL evaluates the saturation properties of the coolant on the basis of the static pressure at the top of the storage racks. These properties include water density, temperature, and steam density. The steam is assumed to separate and flow out of the pool. The water at the saturation temperature corresponding to the pressure at the top of the racks flows downward to the inlet of the storage racks. The static pressure at this location is higher than the pressure at the top of the storage racks and as a result the fluid is subcooled as it enters the fuel assembly. The fluid becomes less dense as it passes up the fuel channel. Near the top of the fuel channel the fluid reaches saturation conditions and net boiling occurs. The computer code, BPOOL, assumes a loss of all external means of cooling, but it should be noted that the Farley spent fuel pool cooling system is redundant and single failure-proof.

Voiding in the space between fuel assemblies is not possible since these spaces contain poison plates.

III.1.5.(4) Potential Fuel and Rack Handling Accidents

The high-density poison racks are of a free-standing design, utilizing bottom support pads, resting on the floor of the spent fuel pool. The installation of the high-density racks will include removal of the existing clean and uncontaminated 13-inch center storage racks. The high-density racks will be installed dry since there is no fuel in the storage pool.

The following is a sequence of events for installing the high-density poison racks.

Phase I Install and test a temporary crane for handling the existing racks and the high-density racks. The spent fuel bridge crane is a 4,000-pound capacity crane and is not of adequate capacity for the re-rack modification.

Phase II Remove a portion of the existing 13-inch center racks, leaving enough racks intact for one emergency core offload into the spent fuel pool. The number of intact rack modules will be seismically qualified.

Phase III Remove interferences between new racks and existing floor studs by removing a portion of each stud, where required.

Phase IV Install the high-density poison racks into the pool areas left vacant by the removal of the 13-inch center racks. This work may be done gradually over a period of time due to delivery schedules of the highdensity racks Storage capability for one emergency core offload will be available at all times during the spent fuel pool re-rack program.

Phase V When one core offload storage capacity is achieved with new racks, remove any remaining 13-inch center racks and anchor studs from the spent fuel pool.

Phase VI Install the balance of high-density racks into the spent fuel pool to complete rack installation.

Phase VII Remove temporary crane from the spent fuel pool area.

These phases of work will require support work (i.e. leveling of new racks, testing, etc.) to complete the re-racking program.

The outdoor spent fuel cask crane will be used to bring the high-density poison racks from the delivery vehicle into the spent fuel cask area. The racks will then be moved from the spent fuel cask area, by the temporary crane, into the spent fuel pool. The reverse sequence will be performed to remove the existing 13-inch center storage racks from the spent fuel pool.

The installation of the high-density poison racks will not increase the potential for a fuel and rack handling accident for the following reasons:

- The spent fuel pool is dry and does not contain any spent fuel.

- The temporary crane, as with the spent fuel bridge crane, can carry loads over the spent fuel cask area and the spent fuel pool only. There is not any safe shutdown equipment located in these areas. Therefore, there will not be any damage to safe shutdown equipment should a rack drop into these areas.

- The spent fuel cask crane, used to bring the racks into and out of the spent fuel cask area, is a single failure proof crane as described in subsection III.1.2.(2).

Protection against a rack drop is assured since the cask crane is single failure-proof, and a dual point attachment will be used between the spent fuel pool cask crane main hook and the lifted spent fuel rack module.

In addition, the racks can follow the cask load path into and out of the cask area. By following this path, the racks will not pass over any safety-related equipment except the spent fuel pool cooling system. Since no fuel is presently stored in the spent fuel pool, the spent fuel pool cooling system is not presently required for a safe shutdown of the plant.

III.1.5.(5) Technical Specifications

To insure against criticality, the following technical specifications are proposed in figure III-1 on spent fuel storage in the high-density poison racks.

III.1.5.(5).1 Paragraph 5.6.1.1 of the proposed revision to the Farley Unit 2 Technical Specifications requires that the spent fuel storage racks be designed and maintained such that the neutron multiplication factor (k_{eff}) in the fuel pool shall be less than or equal to 0.95 when flooded with unborated water. This represents the most conservative pool condition from a criticality standpoint.

III.1.5.(5).2 In addition, paragraph 5.6.1.1 of the proposed revision to the Farley Unit 2 Technical Specifications also specifies a maximum enrichment of 4.3 weight percent U-235 (which equates to 54.63 grams per axial centimeter of the fuel assembly) for fuel loading in the fuel assemblies. This limit is consistent with the design of the high-density poison racks to preclude criticality in the fuel pool.

5.6 FUEL STORAGE

CRITICALITY

5.6.1.1 The spent fuel storage racks are designed and shall be maintained with:

a. A K_{eff} equivalent to less than or equal to 0.95 when flooded with unborated water, which includes conservative allowances for uncertainties and biases based on a maximum enrichment of 4.3 weight percent U-235.

b. A nominal 10.75 inch center-to-center distance between fuel assemblies placed in the storage racks.

5.6.1.2 The new fuel pit storage racks are designed and shall be maintained with a nominal 21 inch center-to-center distance between new fuel assemblies such that K_{eff} will not exceed 0.98, based on a maximum enrichment of 3.5 weight percent U-235, assuming aqueous foam moderation.

DRAINAGE

5.6.2 The spent fuel storage pool is designed and shall be maintained to prevent inadvertent draining of the pool below elevation 149.

CAPACITY

5.6.3 The spent fuel storage pool is designed and shall be maintained with a storage capacity limited to no more than 1407 fuel assemblies.

5.7 COMPONENT CYCLIC OR TRANSIENT LIMIT

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

IV. MECHANICAL, MATERIAL, AND STRUCTURAL CONSIDERATIONS

IV.(1) Description of Spent Fuel Pool and Racks

Alabama Power Company has purchased high-density spent fuel storage racks for the Joseph M. Farley Nuclear Plant Unit No. 2. The spent fuel pool can contain sufficient high-density racks to safely store 1407 spent fuel assemblies. Figures IV-1 through IV-7 show the plans and sections of the spent fuel pool with figure II-3 showing the arrangement of the spent fuel storage racks.

The high-density (poison) spent fuel storage racks are of stainless steel welded construction. They consist of three basic components. (See figures II-4 and II-5.)

1. Bottom grid.

2. Neutron absorber canister (poison cans).

3. Adjustable foot assembly.

The neutron absorber canister, hereafter called poison can, consists of a stainless steel wrapper holding the Boraflex firmly against the inner can. The neutron absorber (Boraflex) is comprised of a polymeric silicone encapsulant entraining and fixing fine particles of boron carbide in a homogeneous stable mixture. The outer wrapper is comprised of two "L" shaped sheets which are firmly pressed against the Boraflex and inner can and then spot welded along the canister length at diagonally opposite corners.

Only the inner tubes of the poison cans act as structural elements. The upper ends of the inner tubes are expanded by die forming to provide lead in surfaces for the fuel. The upper ends of adjacent cans are welded together to form the module top grid. The lower die formed ends of the inner tubes are also welded to the bottom grid. The fuel support surfaces and rack support feet.at integral to th bottom grid. Large leveling screws are located at the bottom grid feet to adjust for variations in pool floor level.

IV.(1).a <u>Support of Spent Fuel Racks</u>

The spent fuel storage racks are freestanding and are thus free to slide or rock on the pool floor during a seismic event. The only interface with the pool floor are the four stainless steel pads per module, attached to the rack leveling screws. In areas where interference with a pool floor weld seam or embedment occurs, a stainless steel plate will be placed to bridge the obstruction. This plate will either be grooved for placement over a weld seam or have a through hole for locating over an embedment bolt(s). This method of bridging will allow a smooth, unobstructed surface for the sliding of the rack foot stainless steel pad. This type of restraint system has the following advantages.

1. Uplift loads are eliminated from any pool floor embendments.

2. Horizontal loads are eliminated from pool walls.

3. Horizontal forces on the pool floor are reduced relative to a vertically restrained rack.

4. Each rack is individually self-supporting and can be removed or installed with minimal effort.

The racks are individually installed with the bottom grids of adjacent racks butting to one another leaving a nominal 5/8 in. gap at the top. Since the racks are not tied together there is potential for rack interaction along with sliding and rocking. These three concerns are all considered in the analysis and discussed further in paragraph IV.(5).

The pool layout (figure II-3) leaves ample clearance for seismic displacement and coolant flow betwee the parameter racks and the spent feul pool walls.

IV.(1).b <u>Fuel Handling</u>

The spent fuel storage rack is designed to withstand the following fuel handling and heavy load impact conditons.

1. Fuel bundle drop from 42 in. above the rack impacting on the middle of the top grid.

2. Fuel bundle frop from 42 in. above the rack impacting on the corner of the top grid.

3. Fuel bundle drop from 42 in. above the rack free falling through an empty cavity and impacting the bottom grid.

4. Inclined fuel bundle drop on top of the rack.

5 Gate drop from 9 in. above the rack impacting on the top of the rack.

Refer to figures IV-7 and IV-8 for details of the fuel handling system.

IV.(2) Applicable Codes, Standards, and Specifications

The spent fuel storage racks are a welded structure consisting of materials of U.S. origin. The following chart presents material specifications and alloys used in the rack assembly.

Material Spec.	Description	Alloy
ASTM-A240 or ASTM-A276	Bottom Grid	304SS
ASTM-A240	Outer Wrapper	304SS
ASTM-A666 Gr. B	Inner Tube	304SS
ASTM-A564	Threaded Foot	Type 630 H-1100
ASTM-A743	Bottom Grid Foot	CF-3

The design of the spent fuel storage rack is as in Section 5 of the AISC Steel Construction Manual with fabrication welding as in ASME Section IX.

The racks were also designed and fabricated to meet and utilize the applicable portions of the following regulatory guides, safety review plan sections, published standards, and computer programs.

1. United States Nuclear Regulatory Commission (USNRC)

   a. Reg. Guide 1.13 — Spent Fuel Storage Facility Design Basis, Rev. 1, Dec. 1975.

   b. Reg. Guide 1.29 — Seismic Design Class, Rev. 2, Feb. 1976.

   c. Reg. Guide 1.92 — Combination of Modes in Seismic Analysis, Rev. 1, Feb. 1976.

   d. Reg. Guide 1.38 — "QA Requirements for Packaging, Shipping, Receiving, Storage, and Handling of Items for Water Cooled Nuclear Power Plants", Rev. 2, 1977.

   e. Reg. Guide 1.60 — Design Response Spectra for Seismic Deisgn of Nuclear Power Plants, Rev. 1, Dec. 1973.

f. Reg. Guide 1.61 — Damping Values for Seismic Design of Nuclear Power Plants, Oct. 1973.

g. Reg. Guide 1.31 — Control of Ferrite Content in Stainless Steel Weld Metal, Rev. 3, April 1978.

h. SRP 3.7 — Seismic Design, 1975.

i. SRP 3.8.4 — Seismic Category I Structures, 1975.

j. SRP 9.1.2 — Spent Fuel Storage, 1975.

k. NRC Guidance on Spent Fuel Pool Modifications, Review and Acceptance of Spent Fuel Storage and Handling Applications (April 14, 1978 revised Jan. 18, 1979).

2. Industry Codes and Standards

a. ASME — Boiler & Pressure Vessel Code Section IX and Section III, Appendix I, XVII, and Article NF-4000, 1980 Edition (American Society of Mechanical Engrs.).

b. AISC — Steel Construction Manual AISC (8th Edition, Dec. 1980 (American Institute of Steel Construction).

c. ACI 318-71 — Building Code Rquirements for Reinforced Concrete. (American Concrete Institute.)

d. ASTM — ASTM Standards: A240, A276, A666, A564, A743.

e. ANSI N45.2 — "Quality Assurance Program Requirements for Nuclear Facilities", 1977.

f. ANSI N45.2.2 — "Packaging and Shipping, Receiving Storage and Handling of items for Nuclear Power Plants", 1972, except Para. 2.4 and 2.6.

g. ANSI N210 — Design Objectives for High Water Reactors Spent Fuel

		Storage Facilities at Nuclear Power Stations, 1976.
h.	ANSI N45.2.10	"Quality Assurance Terms and Definition", 1973.
i.	ANSI N18.2	Nuclear Safety Criteria for the Design of Stationary Pressurized Water Reactor Plants, 1973.
j.	SNT-TC-1A	Recommended Practice for Personnel Qualificationand Certification in Nondestructive Testing, American Society for Nondestructive Testing, 1975.
k.	ANSI N16.9	Validation of Calculation Methods for Nuclear Criticality Safety", 1975.

3. Federal Specifications (Standards)

a.	10 CFR 10	Code of Federal Regulations, Title 10, Part 50 (Appendix A and B).
b.	10 CFR 73.55	Requirements for Physical Protection of Licensing Activities in Nuclear Power Reactor against Industrial Sabotage.
c.	10 CFR 20	Standards for Protection against Radiation.
d.	10 CFR 21	Reporting of Defects and Nonconformances.

4. Computer Programs

a.	ANSYS	Computer Program "Engineering Analysis System" Swanson Analysis Systems, Inc.
b.	SIMQKE	Computer Program, digitized time histories are generated [illegible]ally using "SIMQKE [illegible] was developed under the auspices of the National Science Foundation.

c.	RC[illegible] 1	Computer program for considering nonlinear rocking sliding motion of submerged eccentrically loaded fuel racks. It also considers simultaneous horizontal and vertical time history acceleration.
d.	SPECT	Computer Program, a subroutine of SIMQKE for the computation of spectra from time histories digitized at equal time intervals.
e.	CHEETAH-B/CORC-B/ PDQ-7	Computer Programs for Criticality Analysis (diffusion theory model).
f.	KENO-IV-AMPX-123	Computer Programs for Criticality Analysis (Monte Carlo model).
g.	HPOOL	Computer Program for Spent Fuel Pool cooling water flow and heat transfer under normal condition.
h.	BPOOL	Computer Program for Spent Fuel Pool cooling water flow and heat transfer under boiling condition.

IV.(3) Seismic and Impact Loads

The floor response spectra and damping values are 2 percent (OBE) and 5 percent (SSE).

For a detaile description of analysis method and the parameters involved see paragraph IV.(5).

IV.(4) Loads, Load Combinations, and Structural Acceptance Criteria

The following contains the loads, load combinations, and design allowable stress to which the racks are designed.

A. Load Definitions

D = Dead load of racks

L = Live load due to the weight of fuel assemblies which shall be considered as varying from zero to full load and loadings corresponding to varying placement of the fuel assemblies in the rack shall be considered so that the most critical loads are obtained.

T = Thermal loads for water temperature of 150°F. The minimum water temperature is 40°F.

P = Lifting force of 4000 pounds applied to the top of the rack at any fuel bundle location.

H = Horizontal force of 1000 pounds applied to the top of the rack at any fuel bundle location.

E = Loads generated by the operating basis earthquake (OBE) resulting from ground surface horizontal acceleration and vertical ground surface acceleration acting simultaneously.

E^1 = Loads generated by the safe shutdown earthquake (SSE) resulting from ground surface horizontal acceleration and vertical ground surface acceleration acting simultaneously.

T^1 = Thermal loads for loss of coolant condition corresponding to pool surface temperature of 212°F (240°F at rack elevation).

I = Impact load resulting from the following conditions:

Condition 1 - fuel drop from 42 in. above the rack impacting on the middle of the top grid.

Condition 2 - fuel drop from 42 in. above the rack impacting on the corner of the top grid.

Condition 3 - fuel drop from 42 in. above the racks free falling through an empty cavity and impacting the bottom grid.

Condition 4 - inclined fuel bundle drop on the top of the rack.

Condition 5 - gate drop from 9 in. above the rack impacting on the top of the rack.

B. Load Combinations

The follwoing load combinations shall be satisifed:

Load Combinations	Stress Limit
1. D + L + T + P	F_s
2. D + L + T + H	F_s
3. D + L + T + E	F_s
4. D + L + T + I	
Condition 1	1.6 F_s (1)
Condition 2	1.6 F_s (1)
Condition 3	1.6 F_s (1)
Condition 4	1.6 F_s (1)
Condition 5	1.6 F_s (1)
5. D + L + T^1 + E^1	

NOTES

(1) Local failure of the fuel support or the top grid impact interface is allowed. However, overall member stresses shall be limted to 1.60 F_s and resulting rack deformation shall not cause the fuel configuration to reach a K_{eff} of 0.95.

C. Design Allowable Stresses

F_s = Allowable working stress

f_s = Calculated stress

F_y = Yield stress

D. Allowable Stresses (For Stainless)

The allowable stresses are in accordance with ASME Boiler and Pressure Vessel Code Section III Appendix XVII. This is interpreted as being identical to the AISC Steel Construction Manual (Section 5).

The one third increase in allowable stress for emergency condition is not allowed. The increase in allowable stress is defined by the preceeding paragraph B.

IV.(5) Design and Analysis Procedures

The following is a brief description of the methods used to structurally analyze the spent fuel storage rack design. This freestanding rack design was structurally qualified by a detailed time history and static analysis.

Simplified time history analysis were done at both 0.2 and 0.8 coefficients of friction (μ) conditions with 0, 1/4, 1/2, 3/4, and full eccentric fuel loading conditons. The low coefficient (0.2) was used to define maximum credible sliding displacement, and the higher coefficient (0.8) was used to define the worst loading conditon on a rack. These simplified analyses were done using PaR proprietary computer program RCKN1 and are further explained in paragraph A of this section.

A detailed time history analysis using the ANSYS computer code was then performed for the worst loading condition (previously defined) on a three-dimensional double rack model (6 x 7 and 7 x 8). This detailed analysis defines the combined dead, live, and seismic stresses on a 6 x 7 and 7 x 8 rack. The model and analysis method are further explained in paragraph B of this section.

A static analysis was done on each detailed ANSYS finite element mdoel of a 6 x 7 and 7 x 8 rack for the following static load cases:

a. Rack dead load ("D" loading).

b. Fuel load ("L" loading).

c. Impact load ("I" loading).

d. Fuel handling ("P", "H" loading).

This model and method of analysis are further described in paragraph C of this section.

The stresses resulting from the static analysis were combined as in load combinations 1, 2, and 4 of paragraph IV.(4) and stresses from the time history analysis were combined as in load combinations 3 and 5 of paragraph IV.(4).

The resultant member stresses from the above load combinations satisfy the allowable stress limits as stated in paragraph IV.(4).

A. <u>RCKN1 Time History Model Description</u>

This model (figure IV-9) was used to determine the worst loading conditons based on the aforementioned eccentric fuel loading conditons and varying coefficients of friction. Fuel rattling and rack/rack interaction were not considered in this analysis. In this program the rack is idealized as a vertical beam connecting to a riding base via a torsional spring. This spring/beam is sized to match the lowest horizontal rack frequency. In addition, this beam may be located eccentrically on the base to account for eccentric fuel loading. At each corner of this base, vertical gap springs are located. These springs take only compression loads allowing for rack uplift and rocking. At the lower left corner of the base a horizontal slider spring is located. This element allows for sliding when the horizontal force exceeds μ times the total gap spring force. Fluid coupling forces are also included in the equations motion. The fluid coupling forces are assumed to be either in-phase or out-of-phase with the support motion. The model is excited using simultaneous horizontal and vertical support accelerations.

All the springs have dashpots associated with them to represent structural damping.

This model has been benchmarked against a comparable ANSYS computer model.

B. ANSYS Time History Model Description

To consider the effects of module rocking, sliding, and interaction, the double rack three-dimensional ANSYS model was used and is shown on figure IV-10. For illustration purposes this model is beam representation (Section 1) of an 6 x 7 and 7 x 8 rack. However, this model was generated directly from the detailed rack models, described later in paragraph C, using the super element capabilities of ANSYS. This time history model was then verified by comparing its fundamental natural frequencies to the detailed finite element mdoels frequencies.

Section No. 2 of this model represents the mass and stiffness of all the fuel assemblies extending the height of the rack. It is pinned at the bottom of the rack and is allowed to impact at the top, top-quarter, and mid-point. A gap at the top-grid, top-quarter, and mid-point represents fuel-to-can clearance. These clearances are represented in the model by ANSYS gap spring elements. Note, also, that this model conservatively assumes that all fuel assemblies are in phase and move together at all times.

The rack is connected to the floor by use of three-dimensional interface elements. This elements represents two plane surfaces which may maintain or break physical contact and slide relative to each other. At each time step, the program determines if tensile forces exist in the element (e.g. allowing for uplift and rocking). In addition, the program checks the horizontal force in the interface element to see of sliding occurs.

The double rack model include module interaction or potential for banging with other racks in the pool. Gap springs are located at the top grid elevation and initially have a 0.625 in. gap. This model assumes that the largest interaction occurs for a pair of racks because their rocking motion away from each other is unconfined by no adjacent modules.

A single vertical degree of freedom spring represents the pool floor under the racks. The spring rate is calcualted based on the floor stiffness and using the mass of both racks.

A structural damping of 5 percent (DBE) and 2 percent (OBE) for welded steel frame structures was used. All internal water entrapped within the rack envelope is added to the horizontal mass. The external hydrodynamic water mass determination is based upon a paper by R. J. Fritz entitled, "The Effects of Liquids on the Dynamic Motions of Immersed Solids," Journal of Engineering for Industry, February 1972. Both racks for this analysis are assumed to be full of fuel.

The model accounts for in-phase fluid coupling with the pool water by use of a fluid dynamic coupling element. This element is used to represent dynamic coupling between two points. The coupling is based on the dynamic response of two points connected by a constrained mass of fluid, as described in the aforementioned paper by R. J. Fritz.

The impact damping of 10 percent was used for all gap elements. The repetitive impacting of these elements dissipates substantial amounts of energy. Consequently, there is a higher damping within the structure than would exist if there were no gaps.

C. Finite Element Model

The computer program ANSYS was used to analyze the detail finite element model.

Figure IV-11 delineates the computer model. The spent fuel rack is idealized as a three-dimensional detail finite element model of nodal points, consisting of shell and beam elements representing the poison cans and bottom grid.

The poison cans are flared at the top and welded to each other to form the top grid and fuel lead-in surfaces. The cans are also welded into a bottom grid. The super element capability of ANSYS is used to model the cans and grid. (See figure IV-12.) Each of the poison cans are modled with the same super element. The can super element consists of over 80 quadrilaterial shell elements. This allows proper modeling of the flared can section and weld connections. The bottom grid super element is modeled with beam, solid, and shell elements.

The rack static model is then made by assembling the can and grid super elements, by coupling the top can nodes to each other, and by coupling the bottom can nodes to the bottom grid.

IV.(6) Structural Acceptance Criteria

Refer to paragraph IV.(4) for the acceptance criteria used for the design of the racks. This criteria is consistent with the applicable sections of the Standard Review Plan 3.8.4 which is to be used in conjunction with AISC code.

IV.(7) Materials, Qualtiy Control, and Special Construction Techniques

The materials used in the fabrication of the spent fuel storage racks are per the chart in paragraph IV.(2). These materials have Certified Material Test Reports which include actual chemical and physical test results. The weld filler materials also have, as a minimum, certificates of compliance and ferrite data.

The fabrication of the racks is in accordance with the following QA/QC procedures:

a. Manufacturing and inspection plan.

b. Cleaning procedure.

c. Liquid penetrant inspection procedure.

d. Visual weld inspection procedure.

e. Welding procedures.

f. Packaging, shipping, and handling procedure.

g. Final inspection/test procedure.

h. Procurement specification for Boraflex.

i. Procurement specification for castings.

j. Weld repair procedure.

k. QA documentation checklist.

l. Fabrication and inspection procedure.

Installation of the spent fuel storage racks wll be in a dry pool with no spent fuel being stored. The spent fuel racks to be installed are independently self supporting and can be installed in any sequence. Each rack is equipped with a leveling screw (±1.0 in. adjustment) in each corner which allows a rack to be individually leveled after installation. This method of adjustment compensates for variations in the pool floor. A storage location system will also be installed around the perimeter of the rack layout which allows for individual cell identification.

IV.(8) Testing and Inservice Inspection

Refer to BISCO reports 748-10-1 and 748-30-1 which contain irradiation studies and performance data for the neutron absorbing material Boraflex.

Poison test coupons are supplied for an inservice surveillance program of the neutron absorber (Boraflex) used in the high-density fuel storage racks. The coupons duplicate the condition of the Boraflex which is encased in the poison canisters. Ten coupons are provided; they are to be removed and analyzed at intervals of 1, 2, 4, 7, 11, 15, 20, 25, 30, and 35 years after installation.

The poison coupons are to be hung alongside the high-density fuel racks, to be subjected to the maximum neutrons, gamma and heat flux.

The procedure for fabrication and testing of the coupons is as follows.

1. Boraflex samples are cut to size and conditoned in normal atmosphere at 20°C to 30°C and 30 percent to 70 percent R.H. for 3 days.

2. Each sample has the following measurements taken at predetermined points on the samples:

    1. Dimensions including thickness, length, and width.

    2. Hardness on the Shore A scale.

    3. Neutron Attenuation at 0.06 eV.

3. Each sample is then fabricated into a coupon and this coupon is installed in the pool.

4. A coupon is to be removed per the above schedule.

5. Carefully open the coupon without damaging the neutron absorber (Boraflex) sample. Condition the samples in accordance with Step 1 of this procedure.

6. Remeasure the samples as indicated in Step 2, assuring that the measurements are made in the same location and by the same procedure as originally performed.

7. Visually examine the surface for change. Take photographs of the surface and any suspect areas.

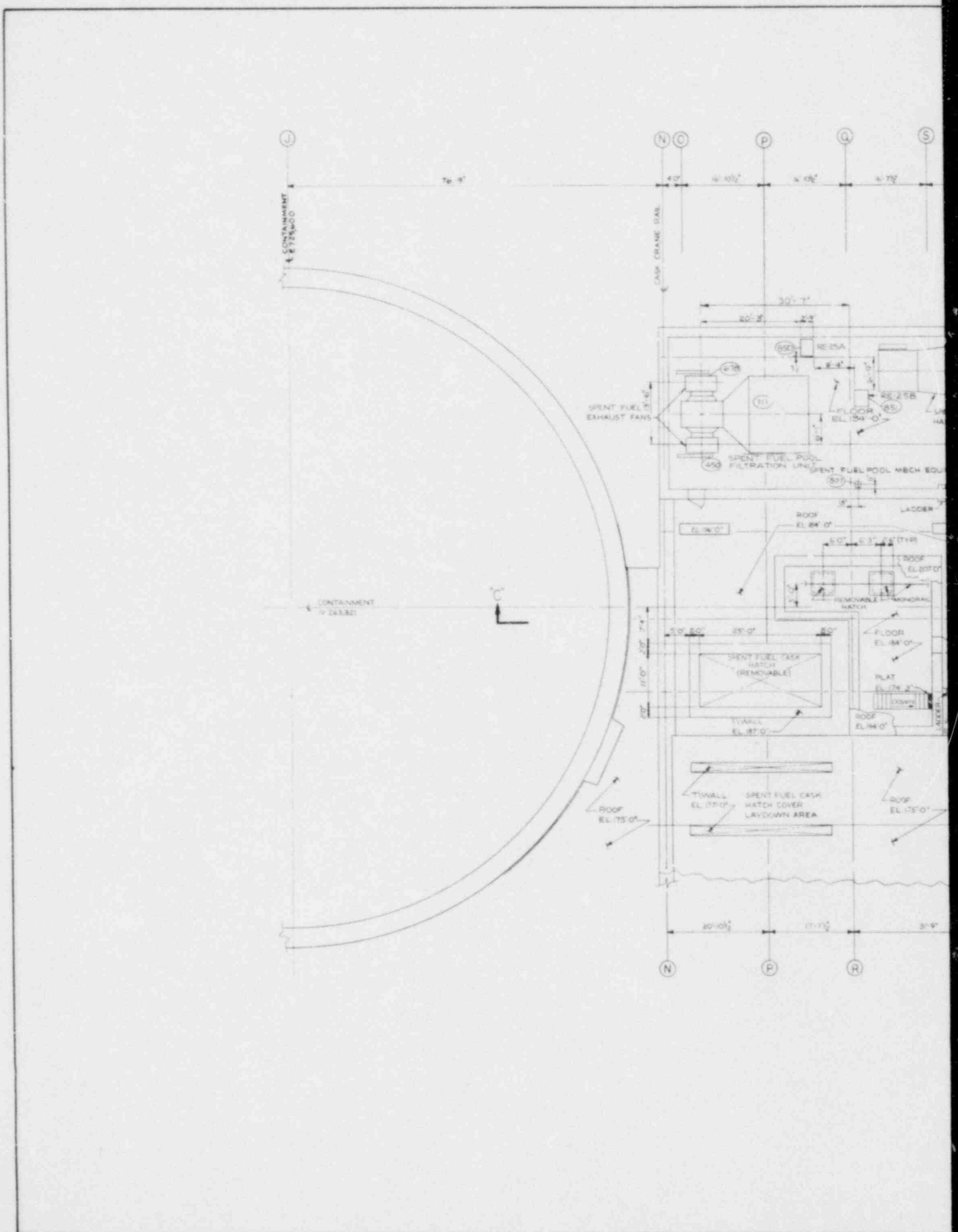
8. Prepare report of sample test results and observations.

9. Should any adverse conditions be detected, the samples may be subject to further neutron transmission studies.

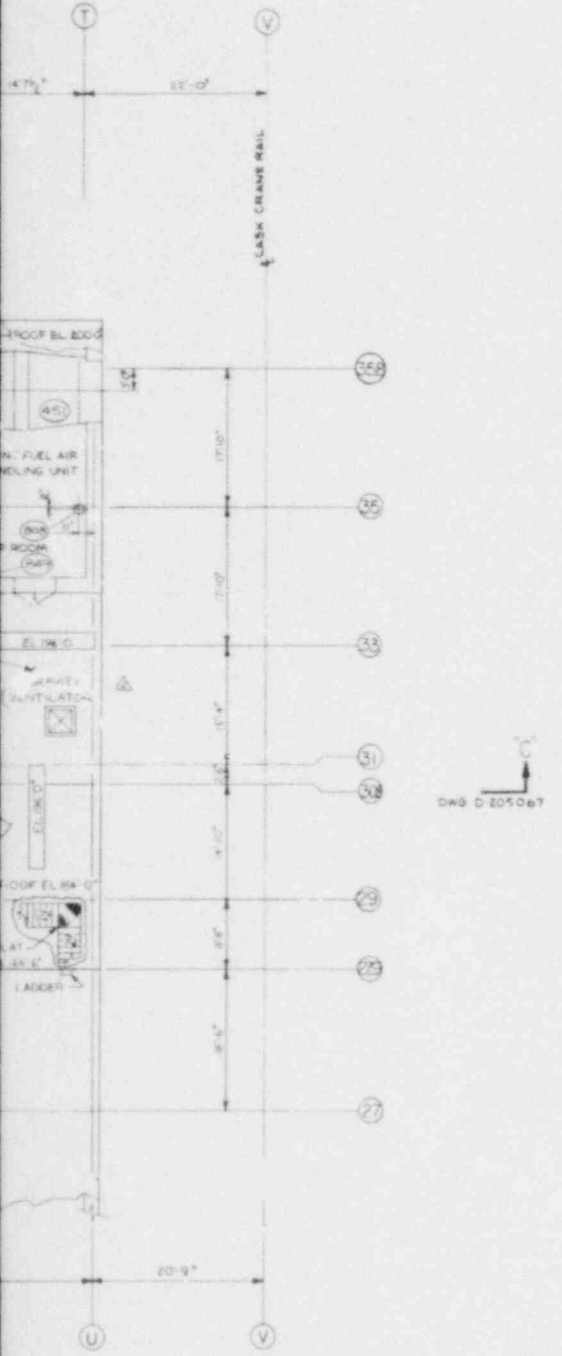
10. Retain samples.

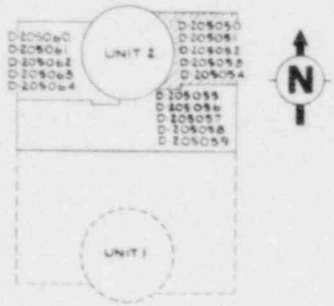
NOTE:

A reference neutron absorber sample will be provided for each coupon. This reference sample is from the same strip of neutron absorber as the coupon sample. This reference sample can be used for future comparison to the installed sheets in the test coupons.


CONTAINMENT
CASK CRANE RAIL
SPENT FUEL EXHAUST FANS
SPENT FUEL POOL FILTRATION UNIT
SPENT FUEL POOL MECH EQUI
RE-25A
RE-25B
FLOOR EL 184'-0"
ROOF EL 184'-0"
LADDER
REMOVABLE HATCH
MONORAIL
SPENT FUEL CASK HATCH (REMOVABLE)
T/WALL EL 187'-0"
ROOF EL 194'-0"
PLAT
T/WALL
SPENT FUEL CASK HATCH COVER LAYDOWN AREA
ROOF EL 175'-0"
"C"

4440-1





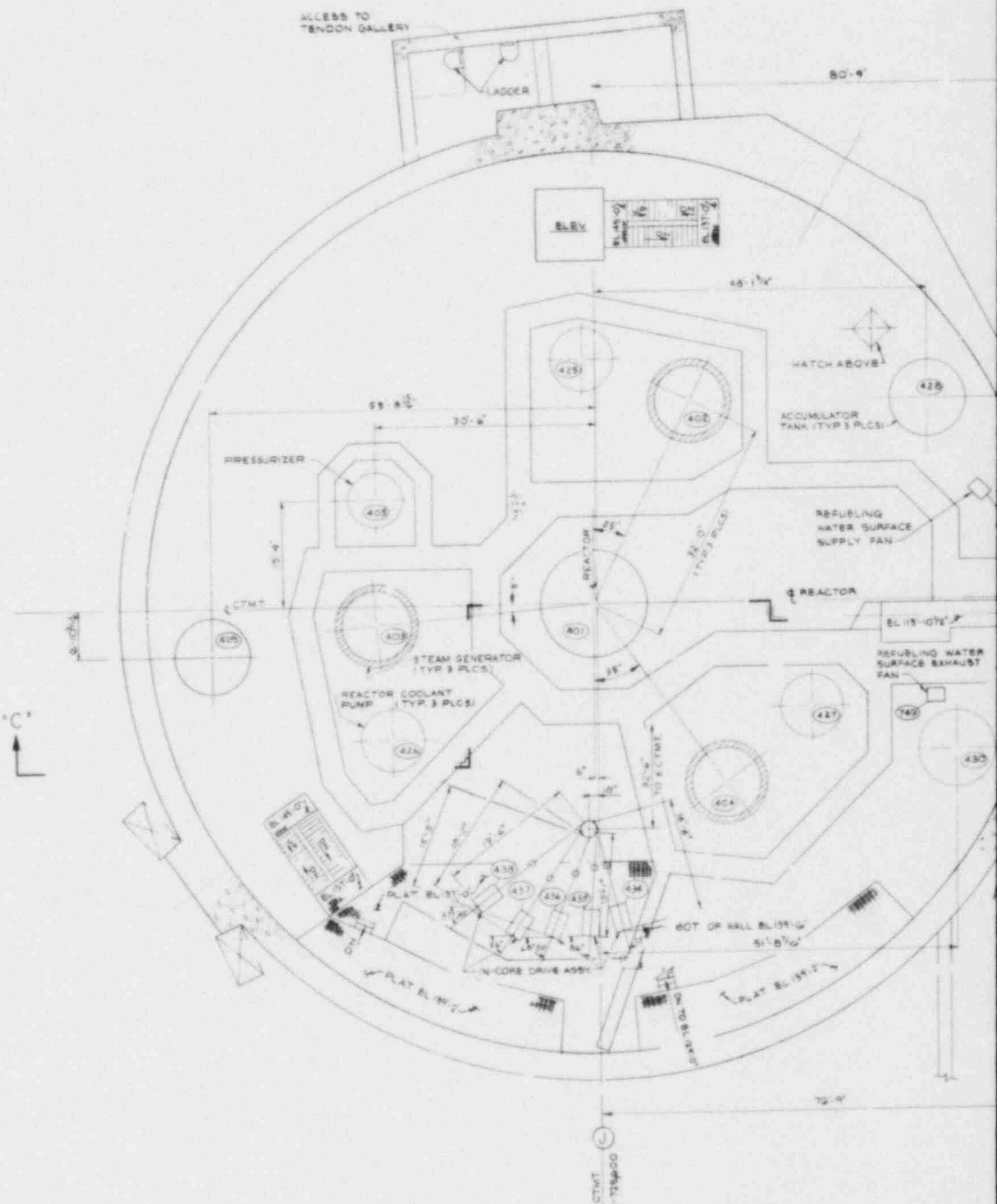
KEY PLAN

Alabama Power

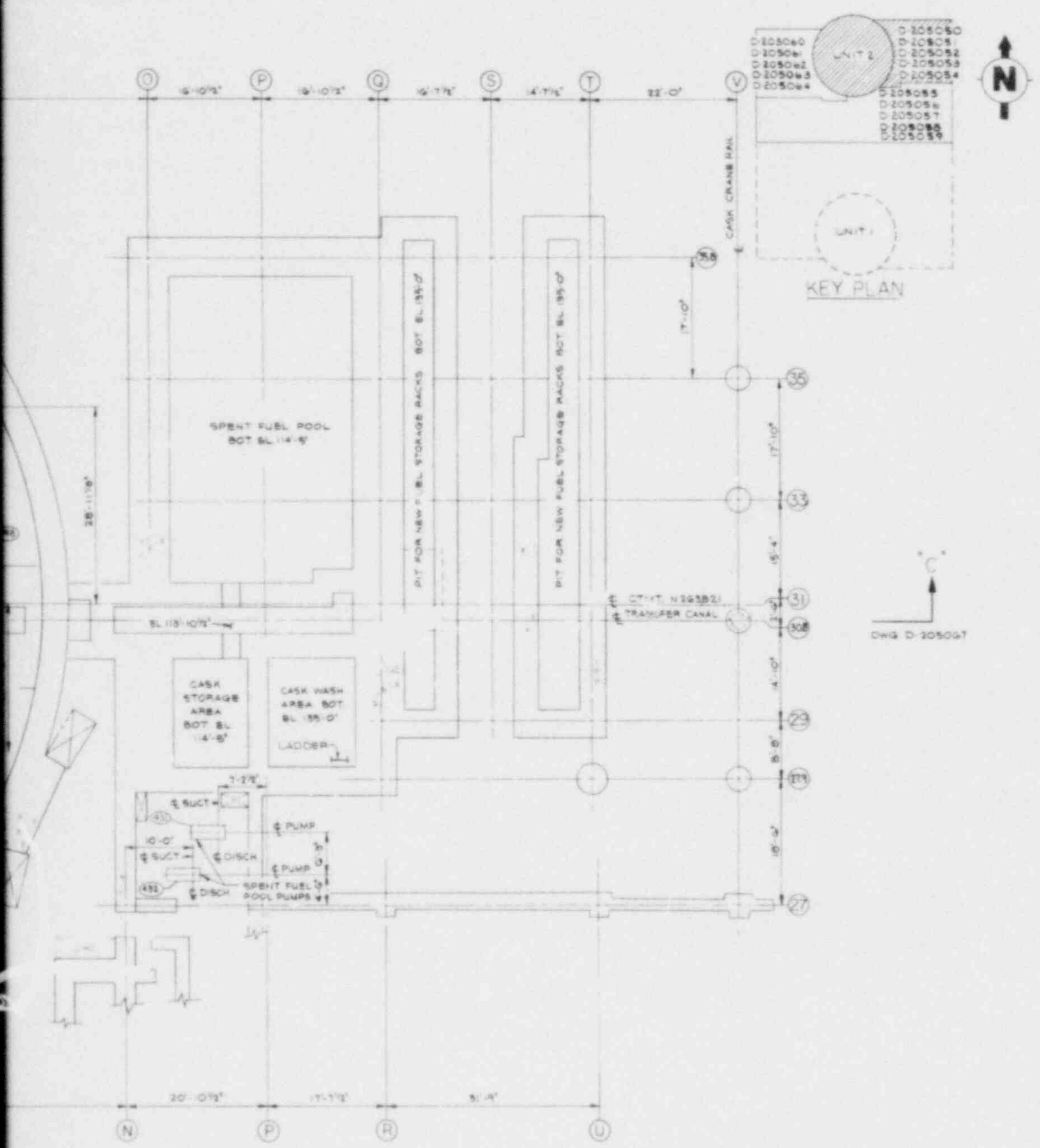
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 2
SPENT FUEL POOL
MODIFICATION

EQUIPMENT LOCATION PLAN.
EL. 175 FT.

FIGURE IV-1



ACCESS TO TENDON GALLERY
LADDER
80'-9"
ELEV
HATCH ABOVE
ACCUMULATOR TANK (TYP 3 PLCS)
PRESSURIZER
REFUELING WATER SURFACE SUPPLY FAN
REACTOR
℄ REACTOR
STEAM GENERATOR (TYP 3 PLCS)
REACTOR COOLANT PUMP (TYP 3 PLCS)
REFUELING WATER SURFACE EXHAUST FAN
"C"
IN-CORE DRIVE ASSY
PLAT EL 139'-2"
J

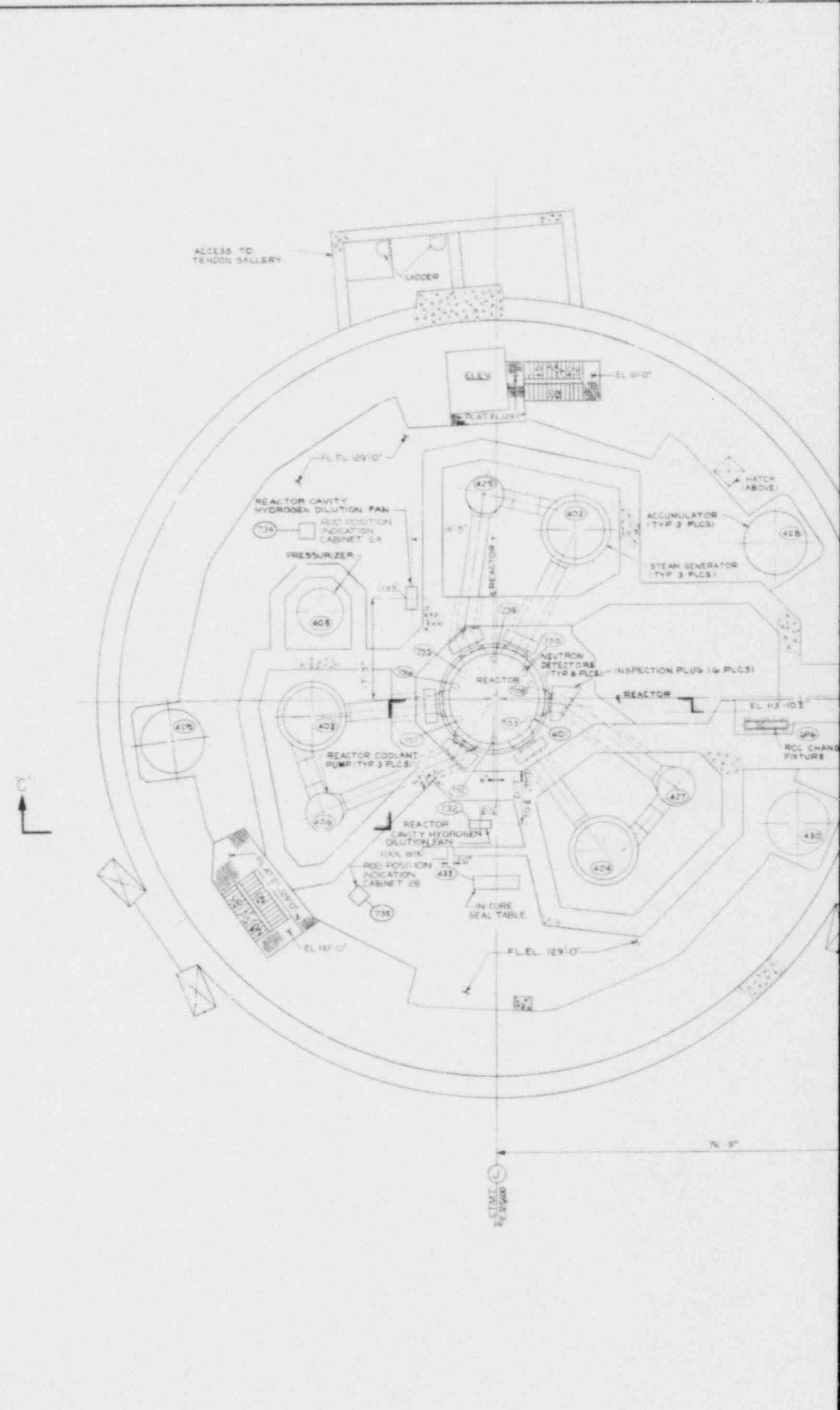


Alabama Power

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 2
SPENT FUEL POOL
MODIFICATION

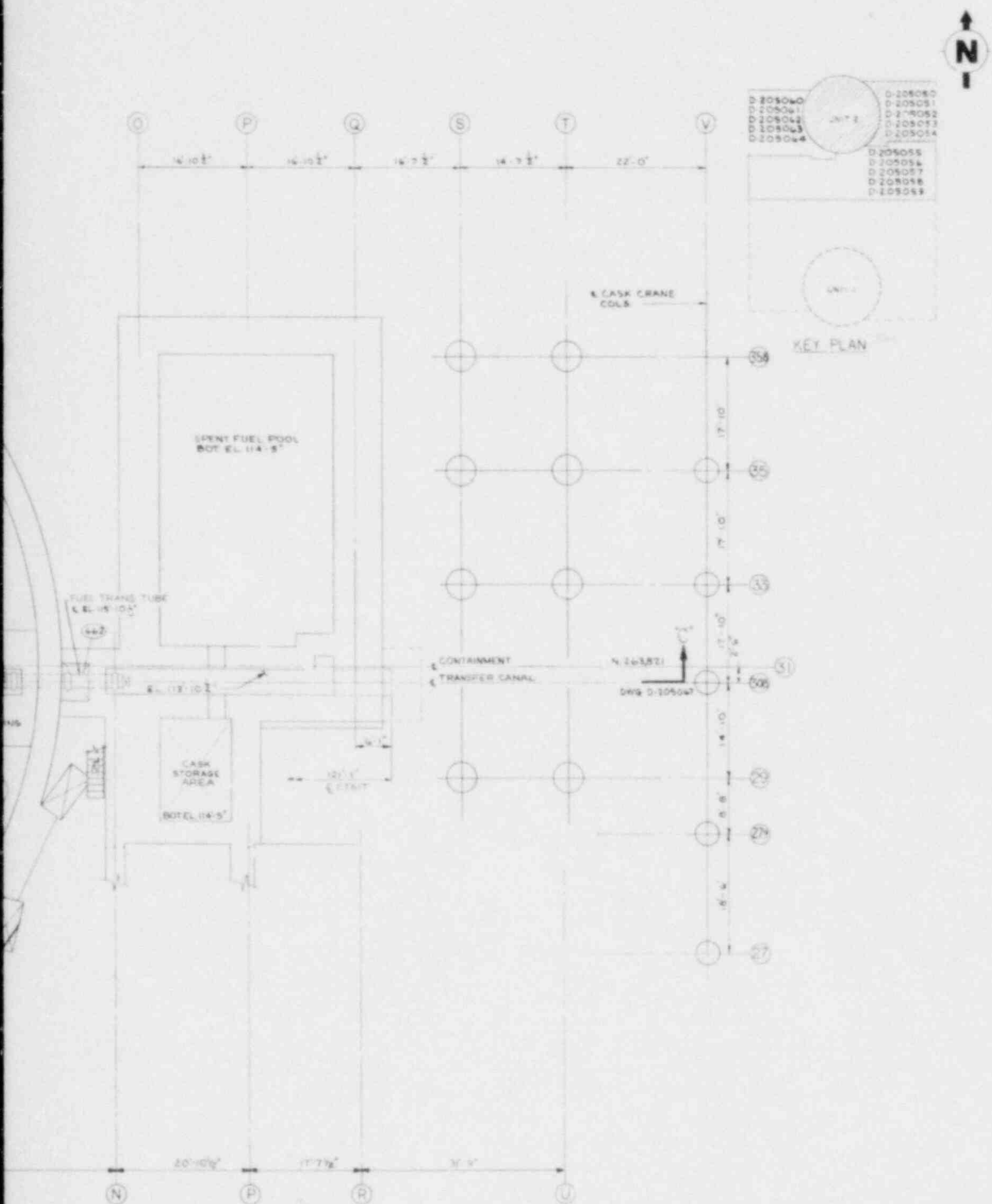
EQUIPMENT LOCATION PLAN.
EL. 139 FT.

FIGURE IV-2



ACCESS TO TENDON GALLERY
LADDER
ELEV
EL 101'-0"
PLAT EL 129'-0"
FL EL 129'-0"
HATCH (ABOVE)
REACTOR CAVITY HYDROGEN DILUTION FAN
ROD POSITION INDICATION CABINET 2A
ACCUMULATOR (TYP 3 PLCS)
STEAM GENERATOR (TYP 3 PLCS)
PRESSURIZER
NEUTRON DETECTORS (TYP 8 PLCS)
INSPECTION PLUG (4 PLCS)
REACTOR
REACTOR COOLANT PUMP (TYP 3 PLCS)
REACTOR CAVITY HYDROGEN DILUTION FAN
ROD POSITION INDICATION CABINET 2B
IN-CORE SEAL TABLE
FL EL 129'-0"

4440-1

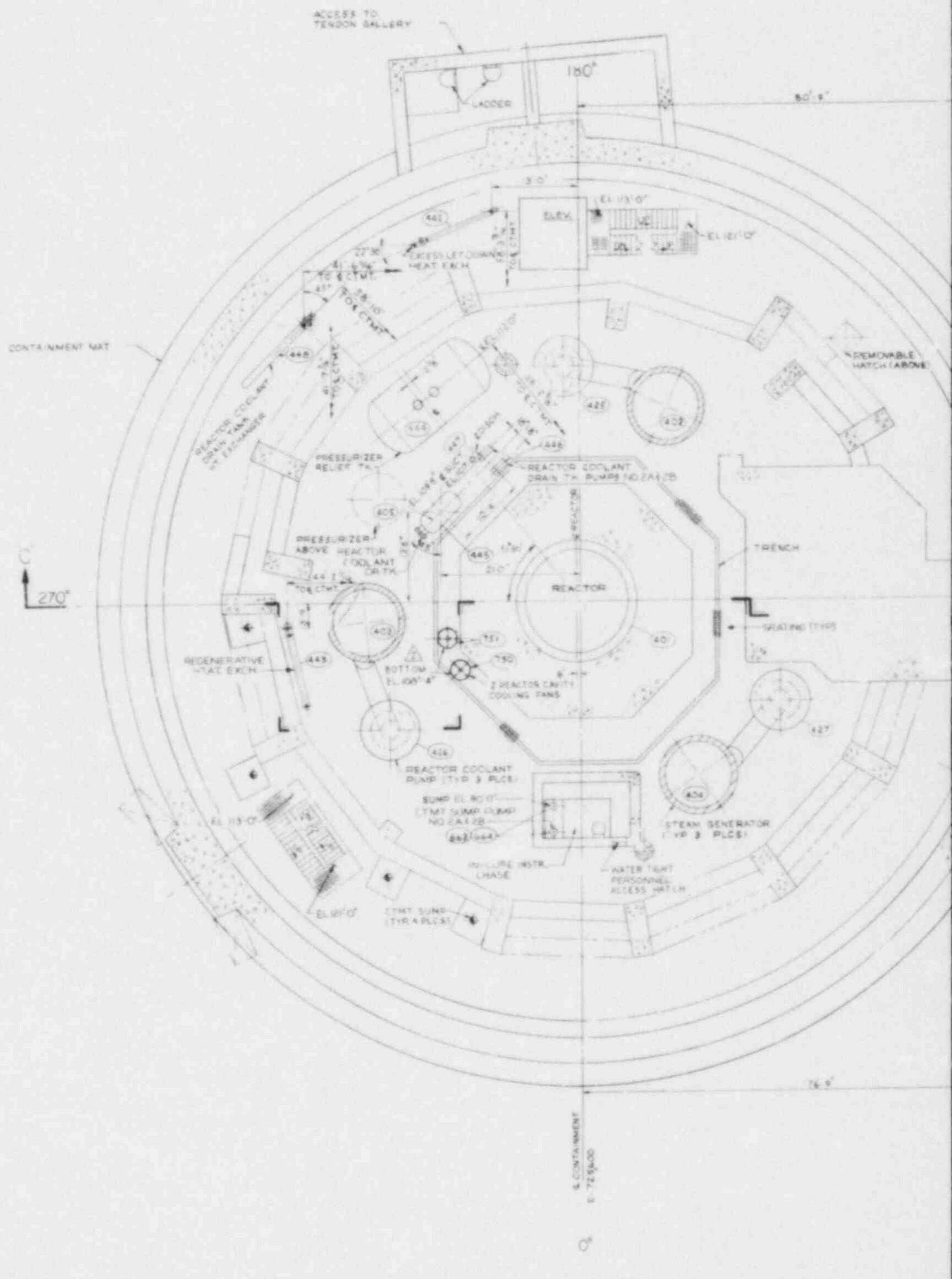


Alabama Power

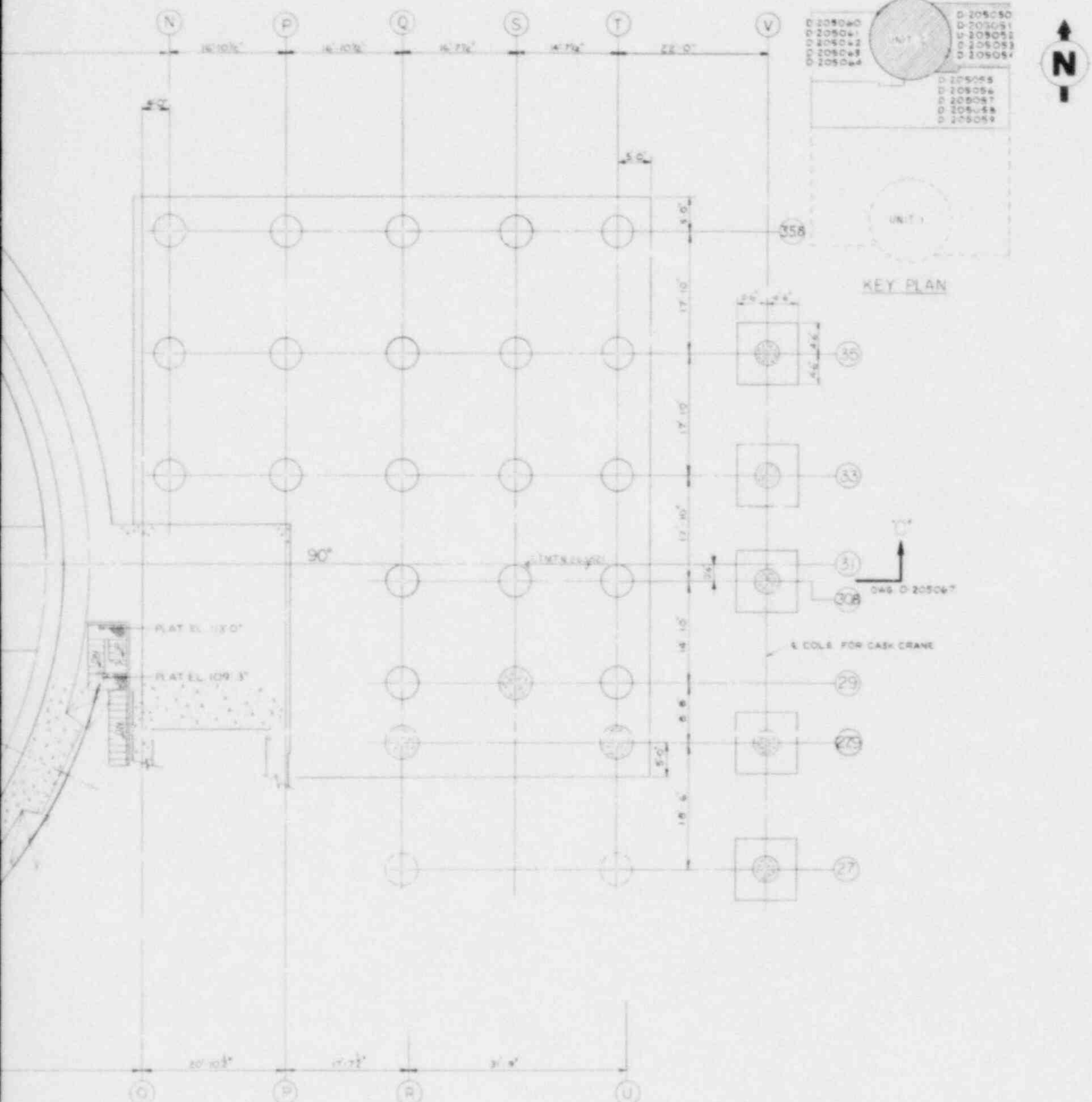
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 2
SPENT FUEL POOL
MODIFICATION

EQUIPMENT LOCATION PLAN.
EL. 129 FT.

FIGURE IV-3



ACCESS TO TENDON GALLERY
LADDER
180°
CONTAINMENT MAT
REMOVABLE HATCH (ABOVE)
REACTOR COOLANT DRAIN TANK HT. EXCHANGER
PRESSURIZER RELIEF TK.
REACTOR COOLANT DRAIN TK. PUMPS NO. 2A & 2B
PRESSURIZER ABOVE
REACTOR COOLANT DR. TK.
TRENCH
REACTOR
270°
SEATING (TYP)
REGENERATIVE HEAT EXCH
BOTTOM EL. 108'-4"
2 REACTOR CAVITY COOLING FANS
REACTOR COOLANT PUMP (TYP 3 PLCS)
SUMP EL. 90'-0"
CTMT SUMP PUMP NO. 2A & 2B
STEAM GENERATOR (TYP 3 PLCS)
IN-CORE INSTR. CHASE
WATER TIGHT PERSONNEL ACCESS HATCH
CTMT SUMP (TYP 4 PLCS)
ELEV.
C
0°

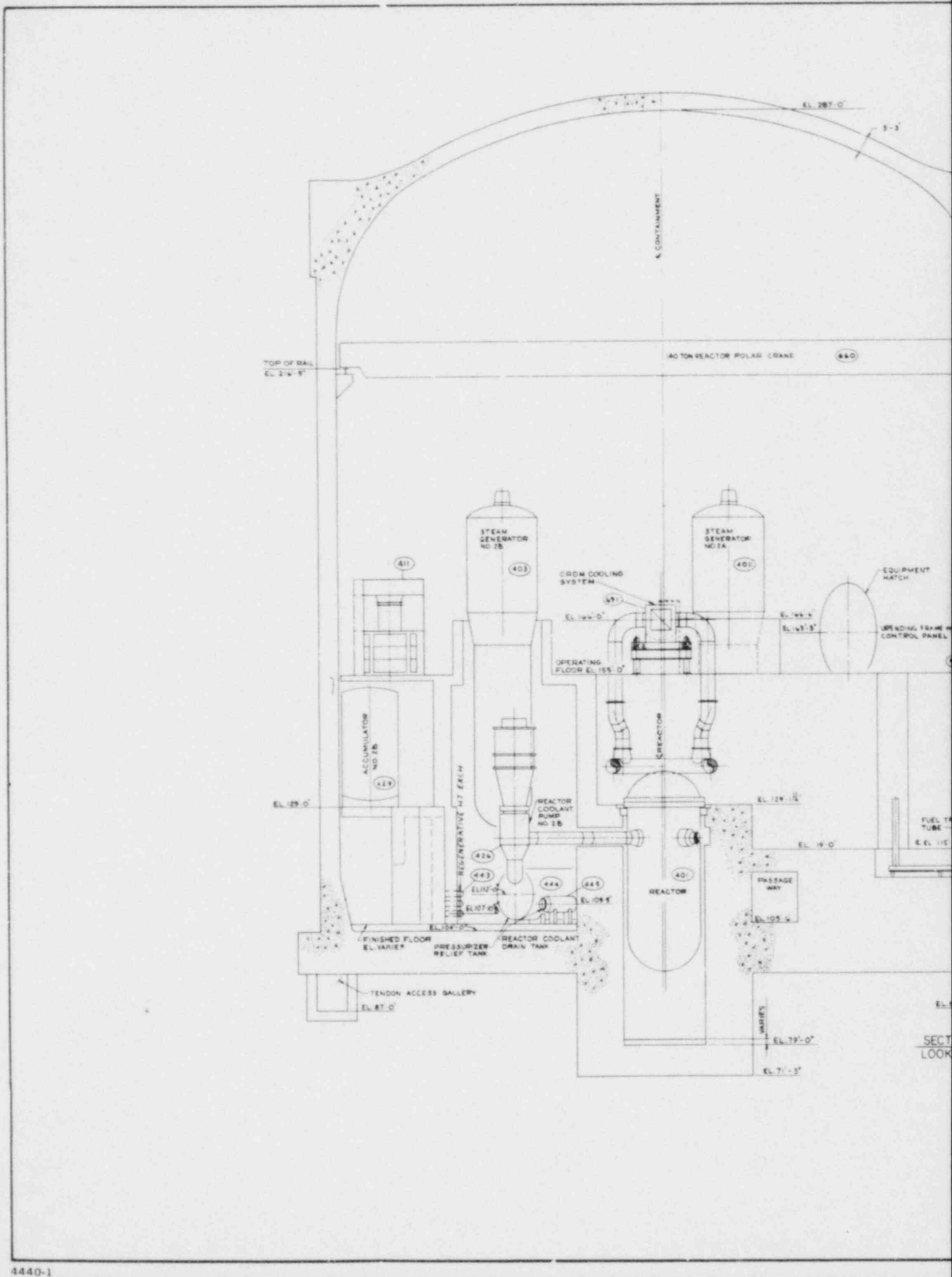


Alabama Power

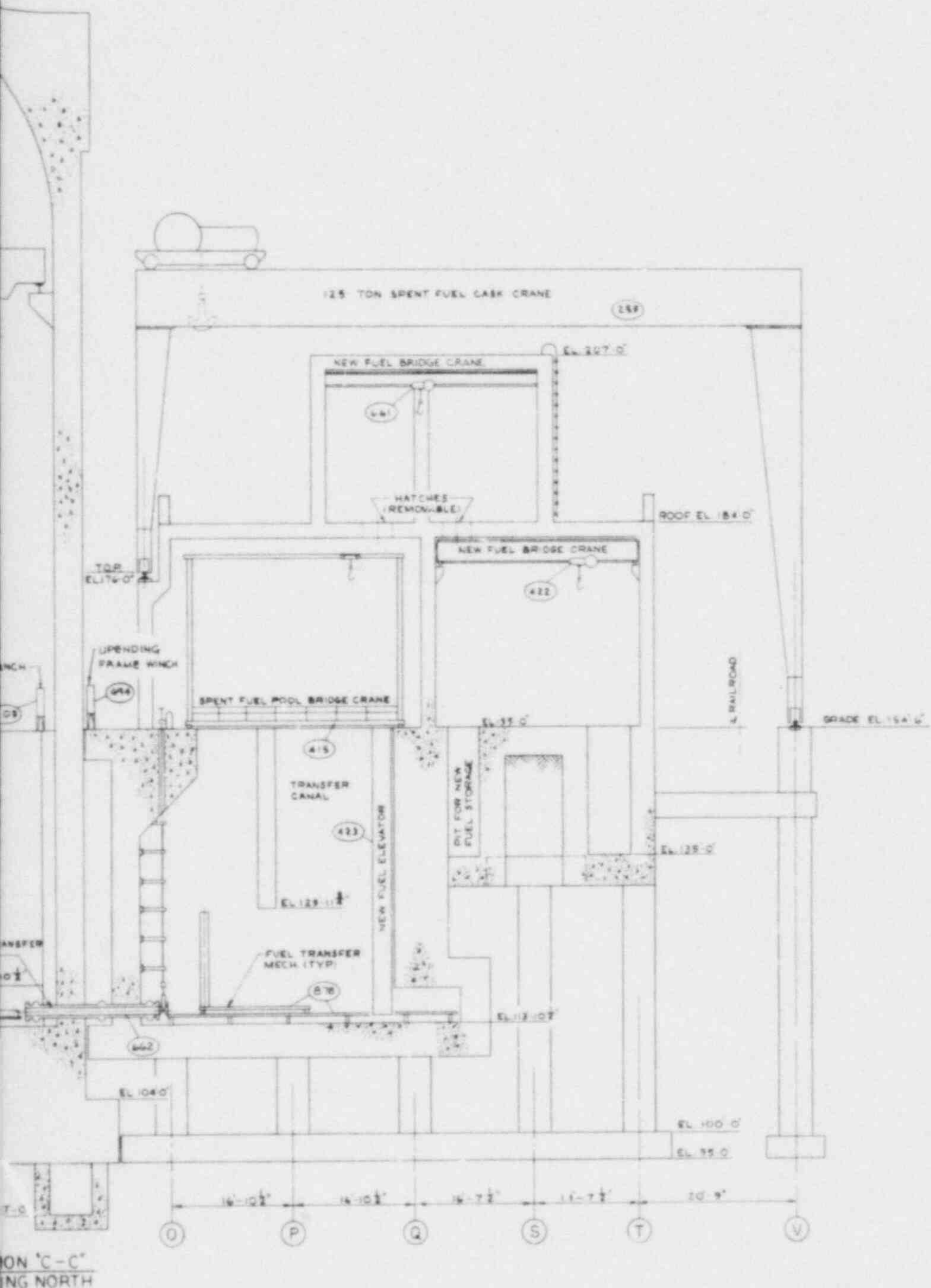
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 2
SPENT FUEL POOL
MODIFICATION

EQUIPMENT LOCATION PLAN
EL. 105 FT. 6 IN.

FIGURE IV-4



EL. 287'-0"
3'-3"
℄ CONTAINMENT
TOP OF RAIL
EL. 216'-9"
140 TON REACTOR POLAR CRANE
660
STEAM GENERATOR NO. 2B
403
STEAM GENERATOR NO. 2A
402
411
CRDM COOLING SYSTEM
491
EL. 166'-0"
EQUIPMENT HATCH
OPERATING FLOOR EL. 155'-0"
UPENDING FRAME W/ CONTROL PANEL
ACCUMULATOR NO. 2B
429
REGENERATIVE HT EXCH
℄ REACTOR
EL. 129'-0"
REACTOR COOLANT PUMP NO. 2B
424
443
444
445
401
REACTOR
PASSAGE WAY
EL. 105'-6"
FINISHED FLOOR EL. VARIES
PRESSURIZER RELIEF TANK
REACTOR COOLANT DRAIN TANK
TENDON ACCESS GALLERY
EL. 87'-0"
VARIES
EL. 79'-0"
EL. 71'-3"
SECT
LOOK

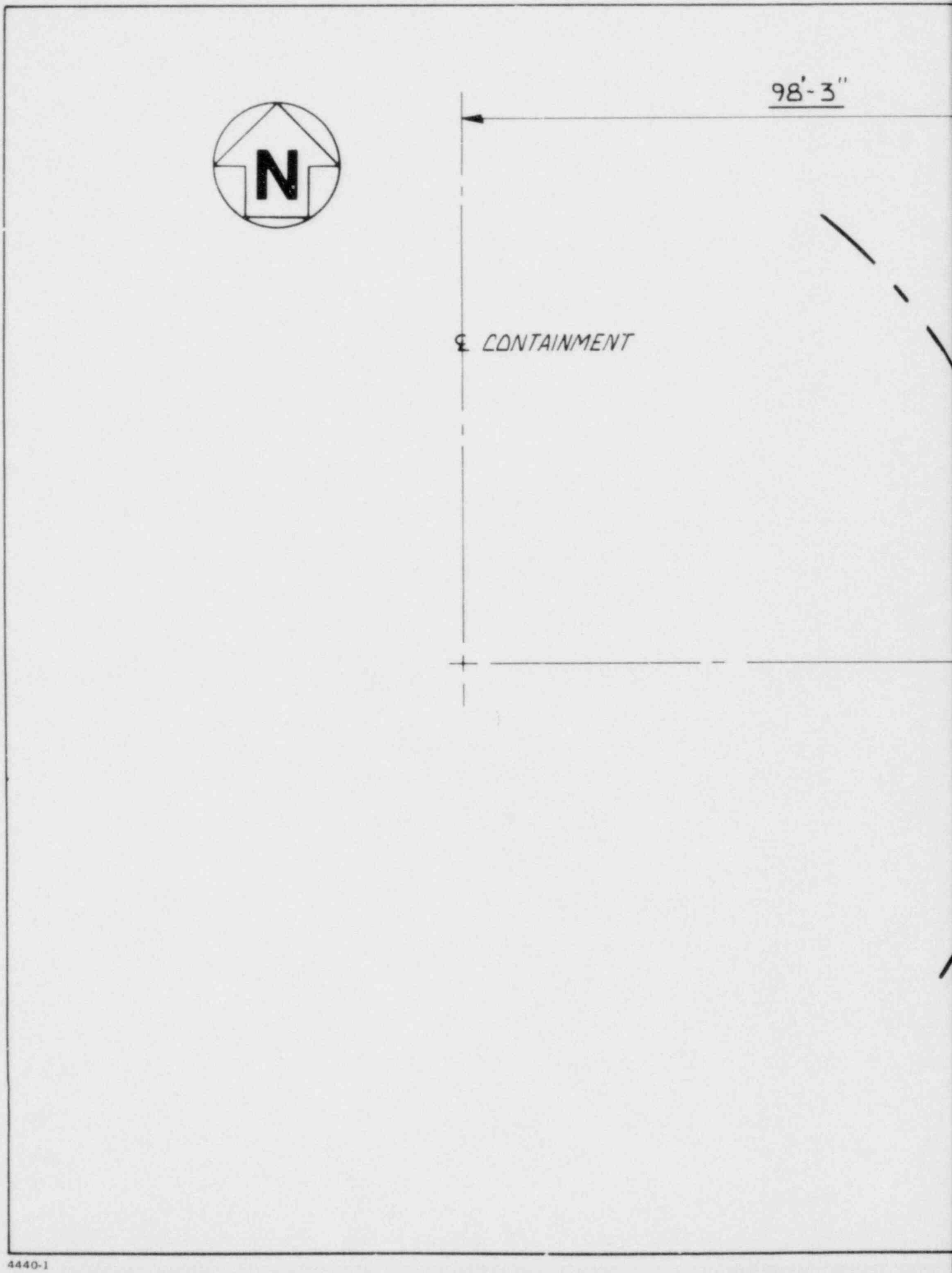


Alabama Power

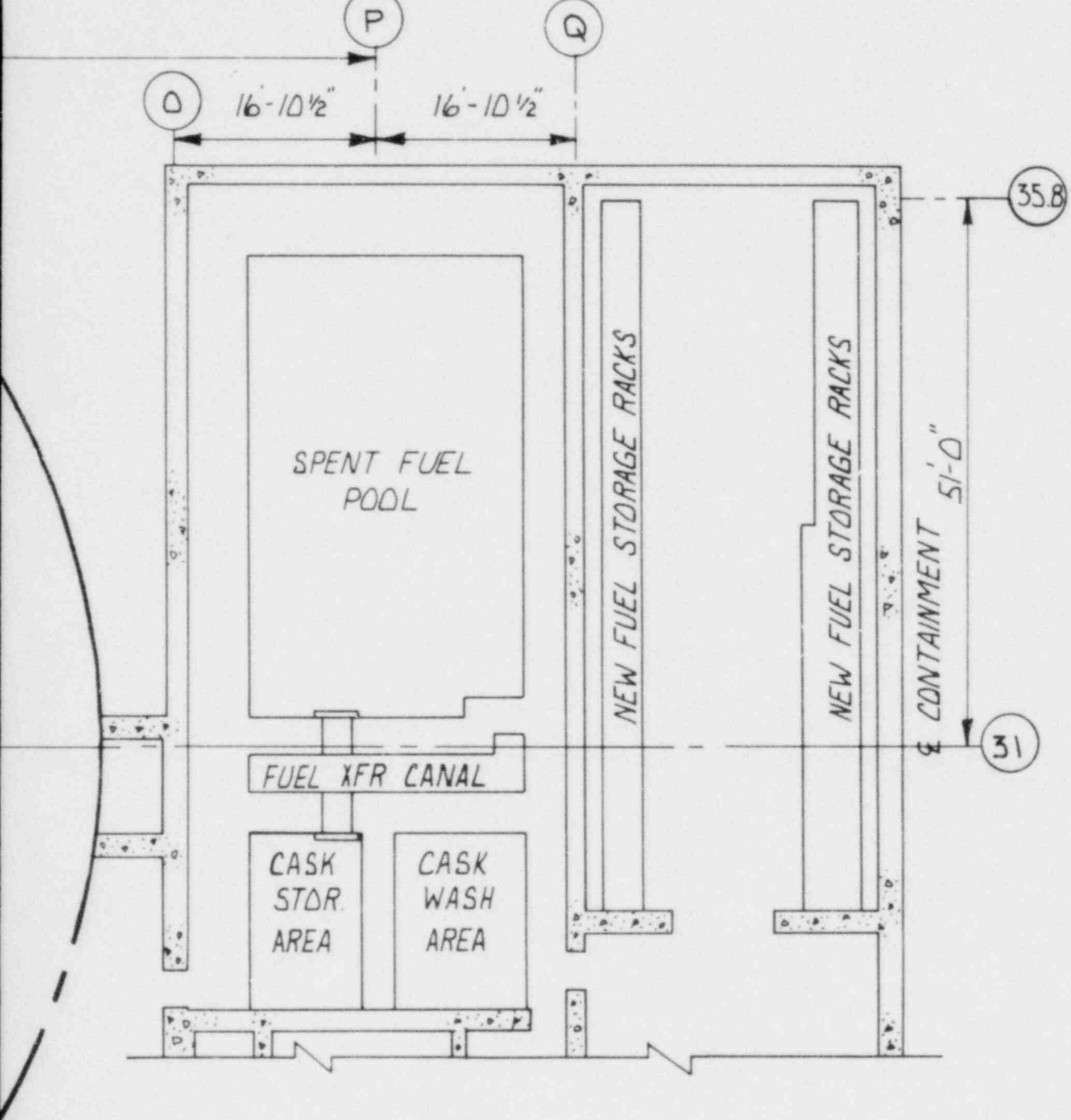
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 2
SPENT FUEL POOL
MODIFICATION

EQUIPMENT LOCATION.
SECTION C-C

FIGURE IV-5



4440-1



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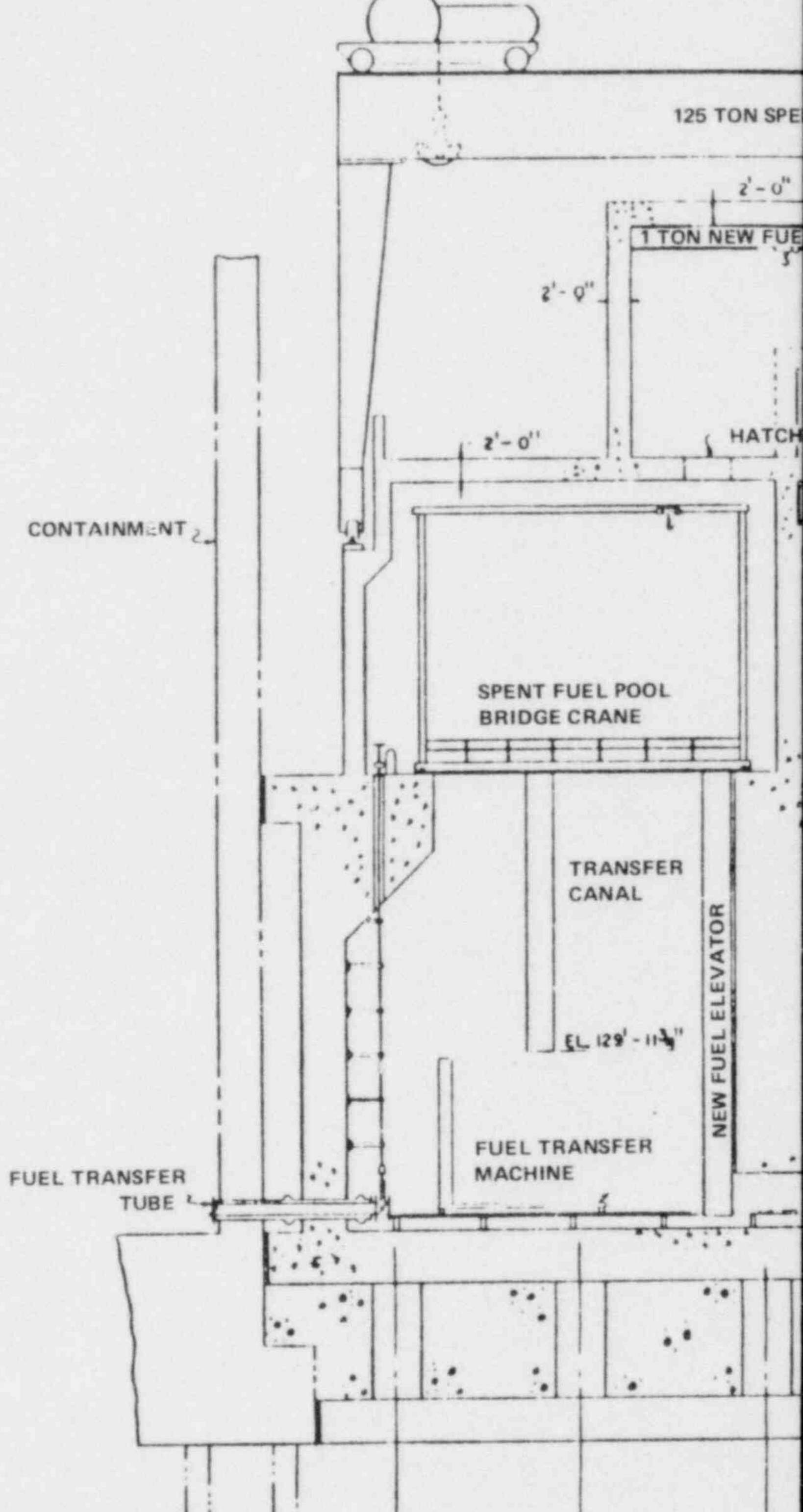
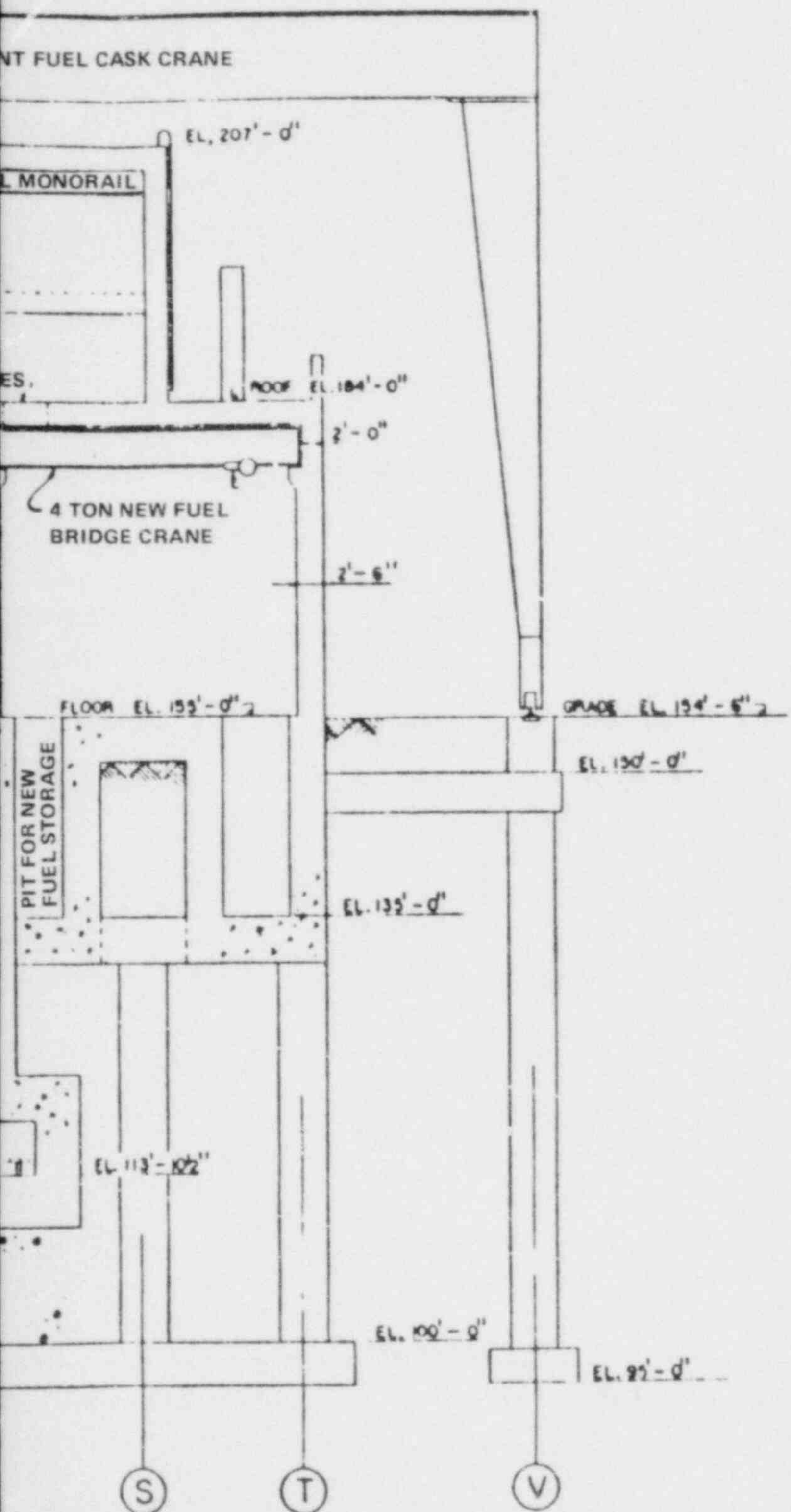
NUCLEAR PLANT
UNIT 2
SPENT FUEL POOL
MODIFICATION

AUXILIARY BUILDING PLAN

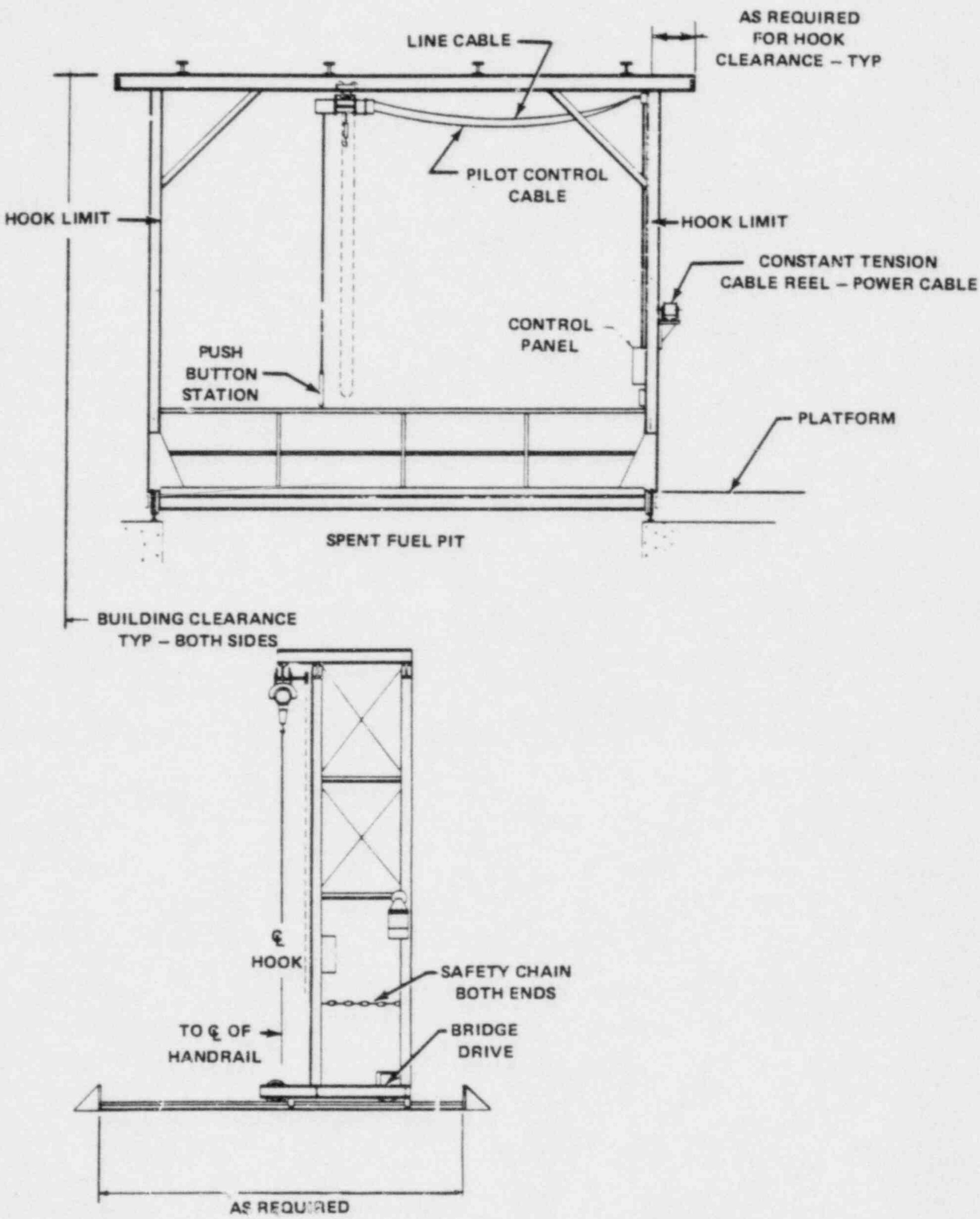
FIGURE IV-6


125 TON SPE
2'-0"
1 TON NEW FUE
2'-0"
HATCH
2'-0"
CONTAINMENT
SPENT FUEL POOL
BRIDGE CRANE
TRANSFER
CANAL
NEW FUEL ELEVATOR
EL 129'-11¾"
FUEL TRANSFER
MACHINE
FUEL TRANSFER
TUBE
O
P
Q

4440-1



Alabama Power	JOSEPH M. FARLEY NUCLEAR PLANT UNIT 2 SPENT FUEL POOL MODIFICATION	AUXILIARY BUILDING. SECTION B-B
		FIGURE IV-7



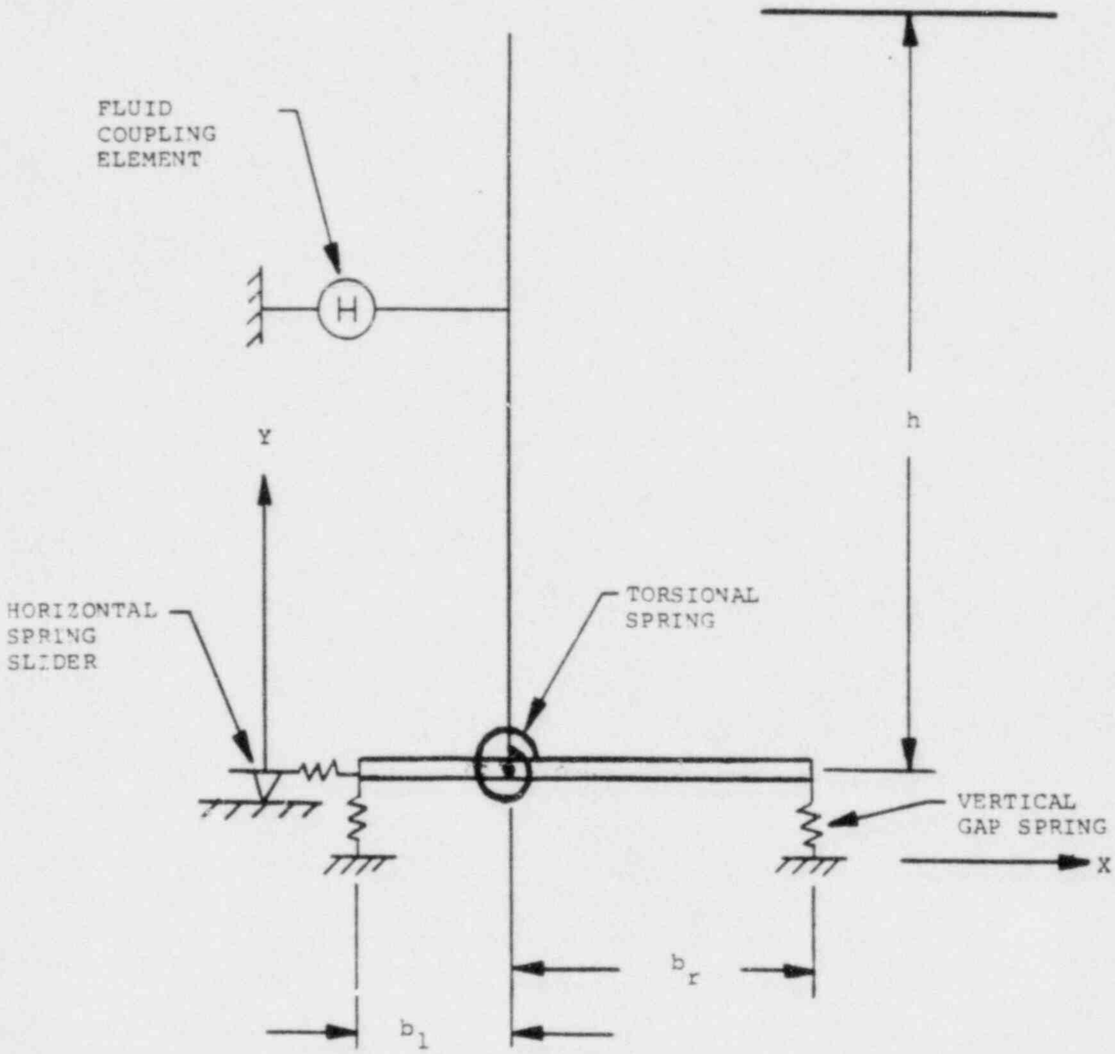
Alabama Power

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 2
SPENT FUEL POOL
MODIFICATION

SPENT FUEL POOL BRIDGE CRANE

FIGURE IV-8

4440-1



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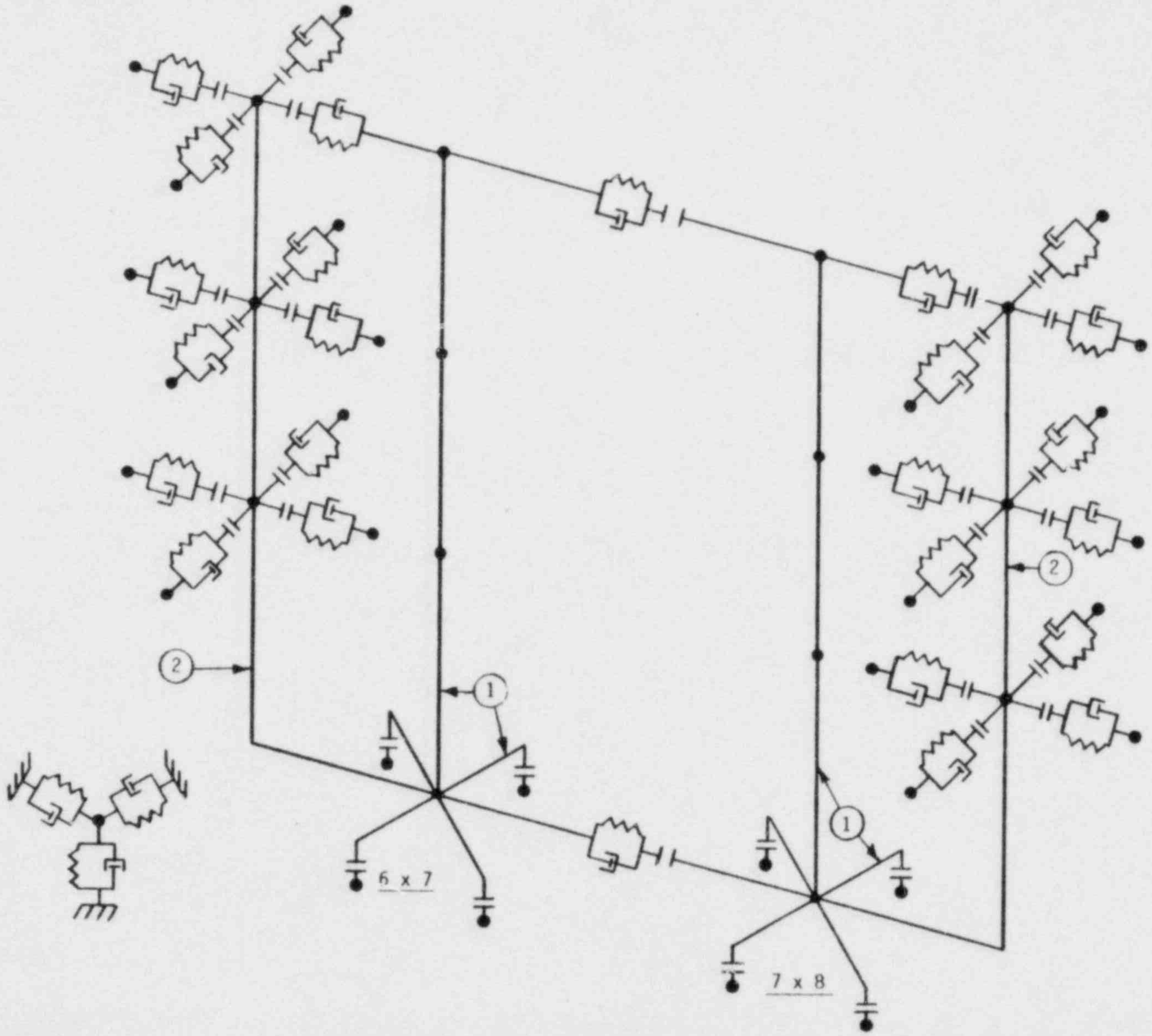
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 2
SPENT FUEL POOL
MODIFICATION

RCKN1 MODEL

FIGURE IV-9

4440-1



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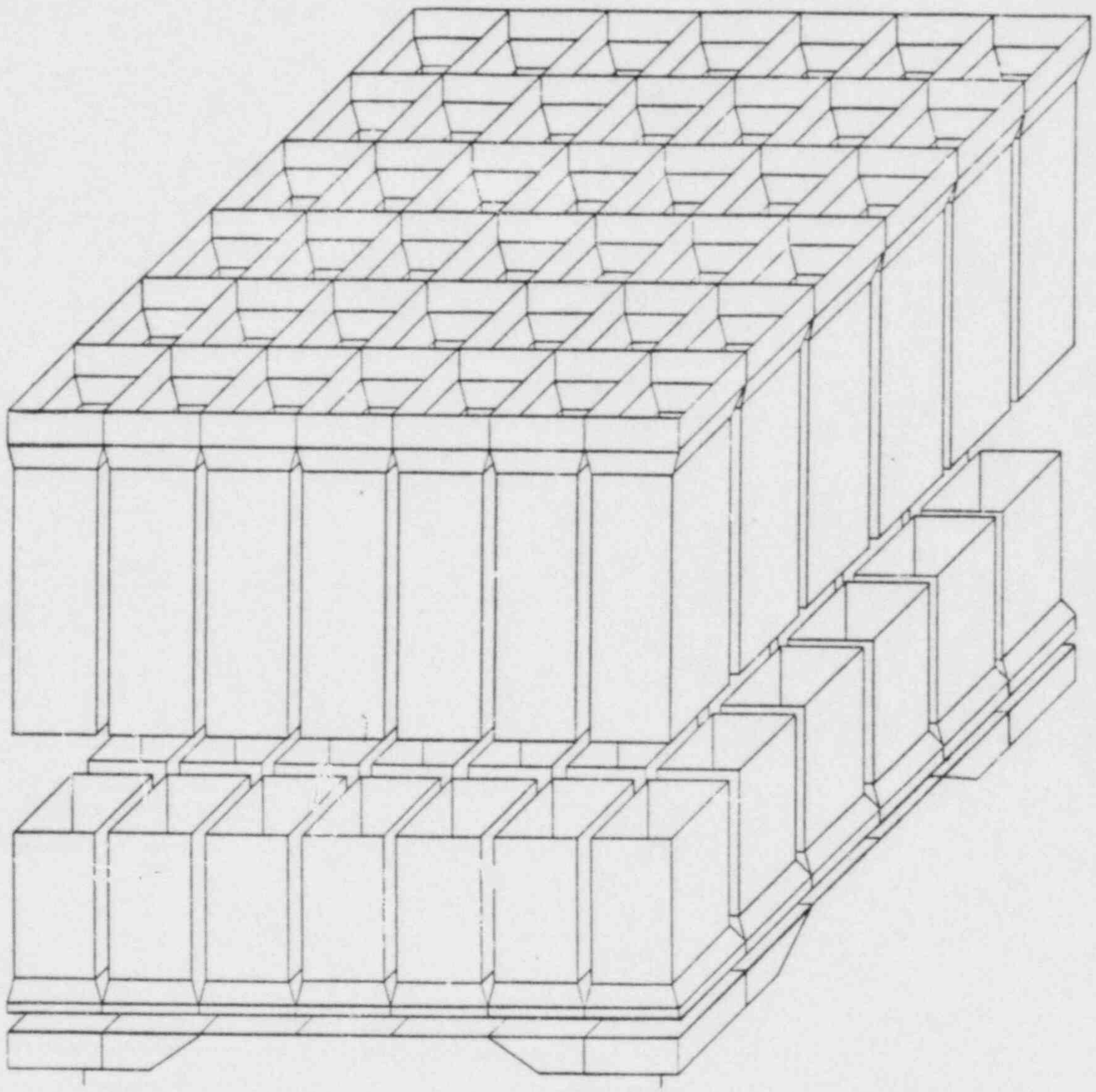
JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 2
SPENT FUEL POOL
MODIFICATION

ANSYS TIME HISTORY
DOUBLE RACK MODEL

FIGURE IV-10

4440-1



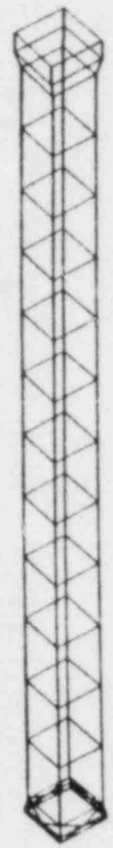
Alabama Power

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 2
SPENT FUEL POOL
MODIFICATION

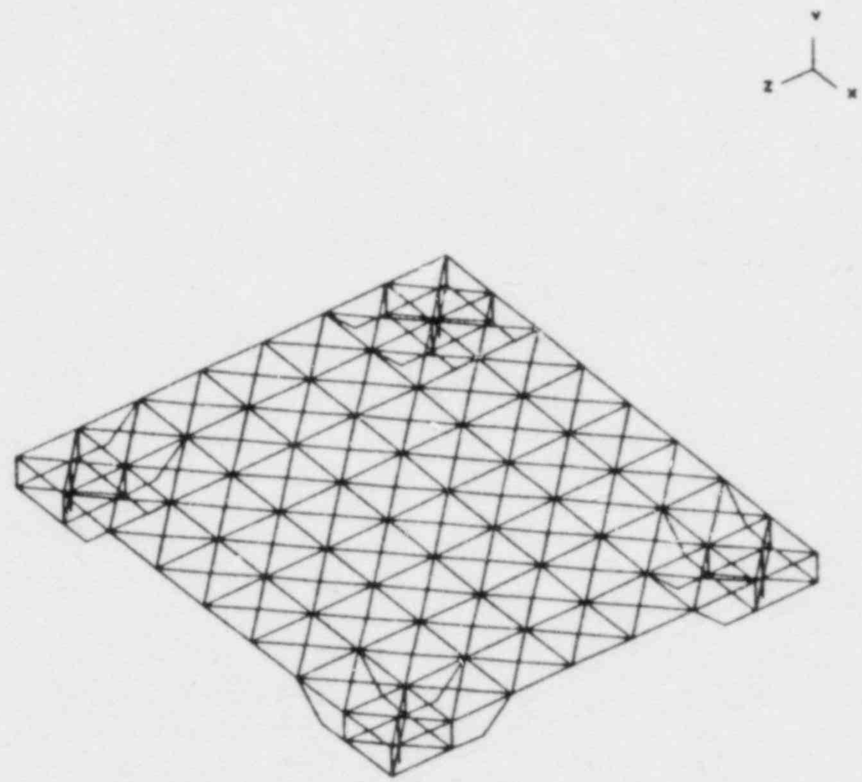
ISOMETRIC VIEW OF
FINITE ELEMENT MODEL

FIGURE IV-11

4440-1



9020 FARLEY .135 CAN GENERATION



9020 FARLEY 7X8 RACK GRID GENERATION

Alabama Power

JOSEPH M. FARLEY
NUCLEAR PLANT
UNIT 2
SPENT FUEL POOL
MODIFICATION

ANSYS SUPERELEMENT GENERATION

FIGURE IV-12

4440-1

V COST/BENEFIT ASSESSMENT

V.1 Environmental Assessment

Following is the information required for the environment cost/benefit assessment.

V.1.1 Special Ne ds for Increased Storage Capacity

The present spent fuel storage facility for Farley Unit-2 was designed for temporary storage of spent fuel until the fuel heat output had decayed sufficiently for transport to a reprocessing facility. The absence of activity in the construction of fuel reprocessing facilities and the cessation of operation of existing reprocessing facilities have created the need for increased on-site storage of spent fuel to permit continued power plant operation.

V.1.1(a) Alabama Power Company currently has no contracts for reprocessing.

V.1.1(b) The anticipated spent fuel discharge schedule for Farley Unit-2 is described in table V-1. The existing spent fuel pool has a capacity of 675 storage locations. When that capacity is filled to less than 157 storage locations remaining then full core storage reserve capability is lost. Under the schedule described in V-1, full core storage reserve capability is lost with the 1991 discharge and all storage capacity would be expended with the 1994 discharge.

V.1.1(c) At present no spent fuel assemblies are stored in the Farley Unit-2 spent fuel pool.

V.1.1.(d) At present, the spent fuel pool is dry and has never contained any spent fuel. There are no control rod assembles or other components stored in the spent fuel pool.

V.1.1(e) Under the proposed expansion, the spent fuel pool capacity would be increased by 732 storage locations which would provide approximately 15 years of additional storage capacity, based on the schedule shown in table V-1.

V.1.1(f) With the expanded capacity full core storage reserve capability would be lost with the 2006 discharge and all storage capacity would be expended with the 2009 discharge. (See table V-1.)

V.1.2 Total Construction Associted with Proposed Modification

Alabama Power will be removing the existing non-poison racks in Farley Unit 2 and replacing them with the new poison racks which are being fabricated by PaR Systems. A temporary crane will be installed for this purpose. There will be no modifcations to the spent fuel pool liner plate. Some of the existing studs in the floor will be ground down to accomodate the new rack modules. These studs are welded to the floor liner plate and embedded in concrete. In order to maintain the liner integrity, no stud will be ground down to the heat affected zone of the stud to liner weld. The spent fuel pool lighting system will be slightly modified to accomodate the new racks.

Listed below are the costs associated with procuring fabricated racks, removing the existing racks, and licensing and installing the new racks.

Progress payment

Six months	- $ 255,160
Nine months	- 1,275,800
10 1/2 months	- 765,480
13 1/2 months	- 255,160

Total of progress payments $2,551,600

Cost of money for progress payments at 20 percent interest. A normal payment date of 11 months after placement of the order is used.

Six months	- $ 21,161
Nine months	- 42,522
10 1/2 months	- 6,376
13 1/2 months	- (10,633)
Net cost of money for progress payments	$ 59,527
Total for manufacturing racks	$2,611,127
Estimated cost of engineering and licensing (other than that provided by the rack supplier)	$ 247,000
Estimated cost of actual installation and testing	$ 724,450
Estimated total costs associated with re-racking	$3,582,577

The existing racks in Farley Unit 2 have never held spent fuel and therefore no decontamination costs are anticipated. At the present time, Alabama Power is negotiating for the sale of the existing racks to another utility. Because these negotiations are incomplete, no credit is taken for salvage value.

Payments to the vendor and installation costs will be incurred entirely in 1982. The remaining engineering and licensing costs will be incurred over a period from June 1982 to December 1982. Because these expenditures will occur over such a short period of time, no adjustment has been made to reduce future expenditures to a present cost.

V.1.3 Alternatives to Increased Storage Capacity

Several alternatives to the expansion of the Farley Unit-2 spent fuel pool to alleviate the spent fuel storage space shortage were considered.

V.1.3.(a) There are no commercial spent fuel reprocessing facilities in operation in the United States. Therefore shipment to a reprocessing facility is not a viable alternative to the expansion of the Farley Unit-2 spent fuel pool.

V.1.3(b) Spent fuel storage at a private or government operated independent spent fuel storage facility to serve Farley unit-2 is not currently available. The alternative of constructing a facility to serve Farley Unit-2 would not be economically viable. Table V-2 shows some selected alternative storage facilities to spent fuel pool expansion. The lowest unit cost alternative is the 10,000 MTU open field drywell storage facility at $48/KgU in construction costs. A facility sized down for the storage needs of Farley Unit-2 would be expected to cost more per kilogram. These costs are significantly higher than the estimated cost of the increased storage capacity which will obtained by expanding the present spent fuel pool (approximately $11/KgU construction costs).

V.1.3.(c) The only other reactor site within Alabama Power Company which could store Farley Unit-2 fuel is Farley Unit-1. That unit is expected to exhaust its present spent fuel storage capacity by about 1992. Storing Unit-2 fuel in the Unit-1 pool would advance the date of loss of storage capacity at Unit-1. Storing Farley Unit-2 fuel at the expense of the spent fuel pool capacity for Farley Unit-1 is not a viable alternative.

V.1.3.(d) The shutting down of the Farley Nuclear Plant-Unit 2 would require either purchase of replacement power outisde of the Alabama Power Company system or construction of new generating capacity. The installation of the new racks will allow an additional 15 years of operation at a cost of

$3,582,577 (approximately $19,000,000 in 1995 dollars). The purchase of replacement power is not a viable option as no long-term commitment may be obtained for the purchase of 863 MW from either within or outside of the Southern Company System. All current Alabama Power long-term generation expansion planning uses standardized 818 MW fossil units. The costs (1995 dollars) of placing an 818 MW size fossil plant in operation in 1995 (old fuel racks filled to capacity in 1994) are as follows:

1.	Capital Cost	$2,396,400,000
2.	Scrubber Capital Cost	$ 398,000,000
3.	Fixed O & M	$ 26,123,200/year
4.	Variable O & M	$ 11,839,000/year
5.	Environmental Cost	$ 21,130,100/year
6.	Fuel Cost	$ 319,606,100/year

It is readily seen that the cost of the increased spent fuel pool storage is insignificant compared to the replacement cost of the Farley Unit-2 generation. These figures do not include any capital (fixed) cost dollars for Farley Nuclear Plant-Unit 2 that still would have to be amoritized whether the plant is operating or not. Also, not included is the cost of maintaining the plant in a shutdown condition and maintaining site security.

V.1.4 Resource Committment

The relatively small quantities of material resources that would be committed to the proposed modification would not significantly foreclose the alternatives available with respect to any other licensing actions designed to ameliorate a possible shortage of spent fuel storage capacity. The material resources that would be comsumed by the proposed modification are listed below.

Material	Quantity (lb)	Dollars/lb
304 Stainless Steel	4.5×10^5	$1.72
Boraflex	2.9×10^4	$2.31

V.1.5 Maximum Water Temperature in the Spent Fuel Pool

Scoping calculations have shown that the heat load will increase in the spent fuel pool as a result of the increased storage capacity of the spent fuel pool. Maximum temperature in the pool for worst case heat loading conditions will be less than the design basis temperature of the spent fuel pool, with one cooling train in operation. Section III.1.5.(3) of this submittal provides additional information on the increased heat load of the spent fuel pool. Results of the final calculations will be provided later.

TABLE V-1

ESTIMATED SPENT FUEL DISCHARGE SCHEDULE

FARLEY UNIT-2

	Year	Annual Discharge Schedule (No. of Assemblies)	Cumulative Discharge (No. of Assemblies)
	1982	52	52
	1983	52	104
	1984	52	156
	1985	52	208
	1986	52	260
	1987	52	312
	1988	52	364
	1989	52	416
	1990	52	468
(1)	1991	52	520
	1992	52	572
	1993	52	624
(2)	1994	52	676
	1995	52	728
	1996	52	780
	1997	52	832
	1998	52	884
	1999	52	936
	2000	52	988
	2001	52	1040
	2002	52	1092
	2003	52	1144
	2004	52	1196
	2005	52	1248
(3)	2006	52	1300
	2007	52	1352
	2008	52	1404
(4)	2009	52	1456

1. Existing storage capacity - loss of full core reserve.
2. Existing storage capacity - filled.
3. Expanded storage capacity - loss of full core reserve.
4. Expanded storage capacity - filled.

TABLE V-2

SUMMARY OF ALTERNATIVE SPENT FUEL STORAGE FACILITIES

	Description	Capital Cost	$/KgU
1.	500 MTU Water Basin Storage Facility (DOE/SR-0006 "Department of Energy Report on Fee for Spent Nucle.r Fuel Storage and Disposal S rvices," October, 1980)	$265,000,000	53
2.	10,000 MTU Open Field Drywell Storage Facility (pp 5-24,25; WI:JBW:81-176 "Conceptual Design and Evaluation Study for an Interim Off-Site Spent Fuel Storage Installation," performed by Westinghouse Electric Corp. for Edison Electric Institute, Tennessee Valley Authority, and U.S. Department of Energy; August, 1981)	$477,000,000	48
3.	10,000 MTU Tunnel Drywell Storage Facility (pp. 5-41, 42; Ibid.)	$1,313,900,000	131

V.2 Radiological Evaluation

V.2.1

The spent fuel pool has never had spent fuel stored in it. The racks are dry, clean, and uncontaminated. Therefore, there have been no radioactive wastes generated from the spent fuel pool.

No significant increase is expected in the wastes generated by the spent fuel pool clean up system over those generated for less dense storage racks. Most of the crud is released soon after fuel is removed from the reactor. Fuel stored for an additional number of years will not contribute significantly to crud loading in the water. Operating plant experience with high density fuel storage has not indicated any noticeable increase in the solid radioactive wastes generated by the increased fuel storage.

Since the spent fuel pool has never been used for spent fuel storage, there will be no radioactive waste generated in replacing the existing 13-inch center storage racks by the high-density storage racks.

V.2.2

Since the spent fuel pool has never had spent fuel stored in it, there have been no Kr-85 releases from the spent fuel pool. Operating plant experience has not indicated any significant release of Kr-85 from spent fuel pools regardless of storage density.

V.2.3.(a)

Since the spent fuel pool has never had any spent fuel stored in it, there has been no radioactive water in it. The nuclides found in spent fuel pools have been predominantly Co-58 and 60 at operating plants.

V.2.3.(b)

There has never been any spent fuel in the spent fuel pool. However, operating experience shows dose rates of a few mr/h either above the center or at the edge of a typical spent fuel pool regardless of fuel storage density.

V.2.3.(c)

Since the spent fuel pool has never had any spent fuel stored in it, there have been no releases of airborne radioactivity from it. Generally, the only airborne radionuclide found in significant concentration in spent fuel pool areas is tritium.

V.2.3.(d)

Operating plant experience with dense fuel storage has shown no noticeable increases in airborne radioactivity above the spent fuel pool or at the site boundary.

V.2.3.(e)

As stated in the response to subsection V.2.1 of this submittal and based on operating experience with dense fuel storage racks, there is no increase in the radwaste generated by the spent fuel pool clean up system. This is because operating experience has shown that with high density storage racks there is no significant increase in the radioactivity levels in the spent fuel pool water. Operating experience with high-density storage racks has shown no increase in the annual man-rem due to the increased fuel storage including the changing of spent fuel pool cooling system resins and filters.

V.2.3.(f)

Operating experience with dense storage of spent fuel has shown that there is no increase in dose rates anywhere above the pool due to increased storage capacity of the spent fuel pool.

As stated in subsection V.2.1 of this submittal, most of the crud released to the pool and clean up system is released soon after fuel is removed from the reactor, not as a result of the increased storage capacity.

V.2.3.(g)

There is no access around the sides or underneath the spent fuel pool. The depth of the water above the fuel is sufficient so there will be no measurable increase in dose rates above due to radiation emitting directly from the fuel. As stated in subsections V.2.1 and V.2.3.(e) of this submittal, operating experience with high-density racks has shown no increase in the processing of radioactive waste or the man-rem associated with it.

The response to subsection V.2.3.(f) of this submittal indicated that operating experience with dense racks has shown no increase in radioactivity levels in the water or dose rates above the pool. As discussed in subsections V.2.3.(c) and V.2.3.(d), operating experience has shown no increased airborne radioactivity for high-density storage racks. Operating experience has also shown no increase in the man-rem due to the increased fuel storage with high-density racks. Therefore, no increase in the annual man-rem is expected at Farley as a result of the increased storage capacity of the spent fuel pool.

V.2.4

The spent fuel storage rack modules that will be removed from the spent fuel pool weigh approximately 16,000 pounds each. Presently, there is no fuel stored in the spent fuel pool. The sale of the racks to another utility is being investigated and, since the racks are clean and uncontaminated, there will be no adverse radiological consequences.

V.3 Accident Evacuation

V.3.1.(a)

Protection against a cask drop is assured by the Seismic Category I, single failure-proof, outdoor spent fuel cask crane, by the single failure-proof lifting device, and by the interlocks and administrative controls described in the Farley FSAR subsection 9.1.4.

A cask tip or drop into the spent fuel pool is also prevented by permanently installed rail stops and mechanical bumpers which prevent cask crane travel over or into the vicinity of the spent fuel pool. The cask crane approach to the spent fuel pool, as shown in Farley FSAR figures 1.2-2 and 1.2-10, is limited to approximately 12 feet. Since the cask will not be handled in the vicinity of the spent fuel pool, the consequences of the cask drop are not affected by the increased storage capacity of the spent fuel pool. The re-rack program will not alter the cask handling procedures described in Farley FSAR subsection 9.1.4.2.3.

V.3.1.(b)

The spent fuel cask crane is not physically capable of traveling over or into the vicinity of the spent fuel pool. Permanently installed rail stops and mechanical bumpers prevent the cask crane from traveling over the spent fuel pool. Since the cask will not be handled over or in the vicinity of the spent fuel pool, the consequences of the cask drop are not affected by the increased storage capacity of the spent fuel pool. A complete cask crane component description, cask handling description, and cask crane design evaluation are described in Farley FSAR subsections 9.1.4.2.2, 9.1.4.2.3, and 9.1.4.3.2, respectively.

V.3.2

The only crane capable for cask movement is the spent fuel cask crane. The spent fuel cask crane is a Seismic Category I, single failure-proof outdoor crane. A cask drop or tip into the spent fuel pool is prevented by permanently installed rail stops and mechanical bumpers which prohibit cask travel over or into the vicinity of the spent fuel pool. The accident aspects of review establish acceptability with respect to subsections V.3.1.(a) and V.3.1.(b) of this submittal.

V.3.3.(1).(a)

The accident aspects of review establish acceptability with respect to subsection V.3.1.(b). Also, the design evaluation of the spent fuel cask crane, as discussed in Farley FSAR subsection 9.1.4.3.2, will not be affected as a result of the re-rack program.

V.3.3.(1).(b)

See response to V.3.3.(1).(a).

V.3.3.(2)

See response to V.3.3.(1).(a).

V.3.4

The cask drop/tip analysis is provided in subsection 3.1.(a).

V.3.5

The maximum weight of loads which may be transported over the spent fuel is not substantially in excess of a fuel assembly. Farley Unit 2 Technical Specification Section 3/4 9-7 limits the size of the load that can be handled over spent fuel to 3,000 pounds.

V.3.6

Since the spent fuel cask will not be handled over or into the vicinity of the spent fuel pool, the proposed modification does not significantly change or impact any previous determinations of Farley Unit 2 Safety Evaluation Reports.

Since there will be a negligible change in radiological conditions due to the increased storage capacity of the spent fuel pool, no change is anticipated in the radiation protection program. Therefore, there will be no change or impact to any previous determinations of Farley Unit 2 Final Environmental Statements.