Edwin I. Hatch Nuclear Plant Units 1 and 2 Docket Nos. 50-321 and 50-366

Spent Fuel Pool Modification

Georgia Power

2104 095

7907170073

EDWIN I. HATCH NUCLEAR PLANT

UNITS 1 AND 2

SPENT FUEL POOL MODIFICATION

2104 096

TABLE OF CONTENTS

2104 097

1.0 INTRODUCTION

This design report and safety evaluation considers the installation of high density, poisoned fuel storage racks in the existing spent fuel pools of Edwin I. Hatch Nuclear Plant Units 1 & 2.

The Hatch 1 and 2 spent fuel pools currently contain racks that can hold 840 and 1120 fuel assemblies, respectively. It was originally assumed that about one quarter of the core would be discharged annually and that spent fuel would be removed from the plant for reprocessing within approximately a year after discharge from the reactor. Because the reprocessing option is not available at this time, the storage capacity of the spent fuel pools must be expanded by replacing the existing spent fuel storage racks with high density, poisoned racks.

It is desirable to have enough capacity in reserve to allow for a full core offload. Such capacity will not exist in the Unit 1 spent fuel pool subsequent to its 1979 refueling cycle. Storage space in the Unit 2 spent fuel pool must then be used to allow for a full core discharge from Unit 1. The high density spent fuel storage racks will provide 3171 storage spaces in Hatch 1 and 2755 in Hatch 2. The modification will provide storage capacity up to the year 1997 with a full core reserve, assuming annual quarter reloads. Installation of the high density storage modules is scheduled to commence in March 1980, first in the Hatch 2 spent fuel pool and then in Hatch 1.

This report describes the design of the high density fuel storage racks to be installed and contains a discussion of the environmental and radiological considerations of the installation. The information contained herein has been prepared based on the recommendations provided in "Operating Technical Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" which was issued by the Nuclear Regulatory Commission (NRC) on April 14, 1978 and later amended on January 18, 1979.

General Electric Company will design and supply the high density, poisoned spent fuel storage racks that will be installed at Plant Hatch. Similar storage racks have previously been reviewed and approved by the NRC on the Monticello and Browns Ferry Nuclear Plants.

2104 098

2.0 OVERALL DESCRIPTION

The location of the spent fuel storage pools within the plant is shown in Figures 2-1 through 2-10 (Unit 1 FSAR Figures 12.1-4 through 12.1-8 and Unit 2 FSAR Figures 3.8-28 through 3.8-32, respectively). The arrangement of the high density fuel storage system for the pools is shown in Figures 2-11 and 2-12.

The high density racks are a base-supported modular design that will replace the existing fuel storage and control rod storage racks. Control rod storage will be provided by supplying a minimum of twenty storage hangers in each of Units 1 and 2. There will be ten extra positions in each pool for storage of defective fuel.

The high density module provides storage spaces for fuel bundles, which do not include the flow channels, or fuel assemblies, which do include the flow channels (see Note 1), on approximately 6.5 inch center to center spacing. Six basic configurations of the basic module are contemplated, containing 13 x 11, 13 x 13, 13 x 15, 13 x 17, 13 x 19, and 15 x 19 storage cells. The combined pool capacity of 5926 fuel assemblies stored in high density fuel racks is made up as shown below:

	Module Configuration	Module Capacity	Quantity	Fuel Assemblies
Unit 1	13 x 11	143	4	572
	13 x 13	169	4	676
	13 x 15	195	5	975
	13 x 17	221	3	663
	15 x 19	285	1	285
				3171
Unit 2	13 x 15	195	8	1560
	13 x 17	221	3	663
	13 x 19	247	1	247
	15 x 19	285	1	285
				2755
		TOTAL		5926

An additional 80 spaces are included in the Unit 2 pool for spent fuel storage by using four of the existing storage racks. Together with the ten defective fuel locations in each pool, the maximum combined pool storage capacity is 6026 (3181 in Unit 1 and 2845 in Unit 2).

Each free standing fuel storage module is fabricated from fuel storage tubes, made by forming an outer tube and an inner tube of 304 stainless steel with an inner core of Boral (see Note 2) into a single tube. The outer and inner tubes are welded together after being sized to the required dimensional tolerances. The completed storage tubes are fastened together by angles welded along the corners and attached to a base plate to form storage modules. Figure 2-13 shows the 13 x 13 module schematically. This module is approximately 7 feet square

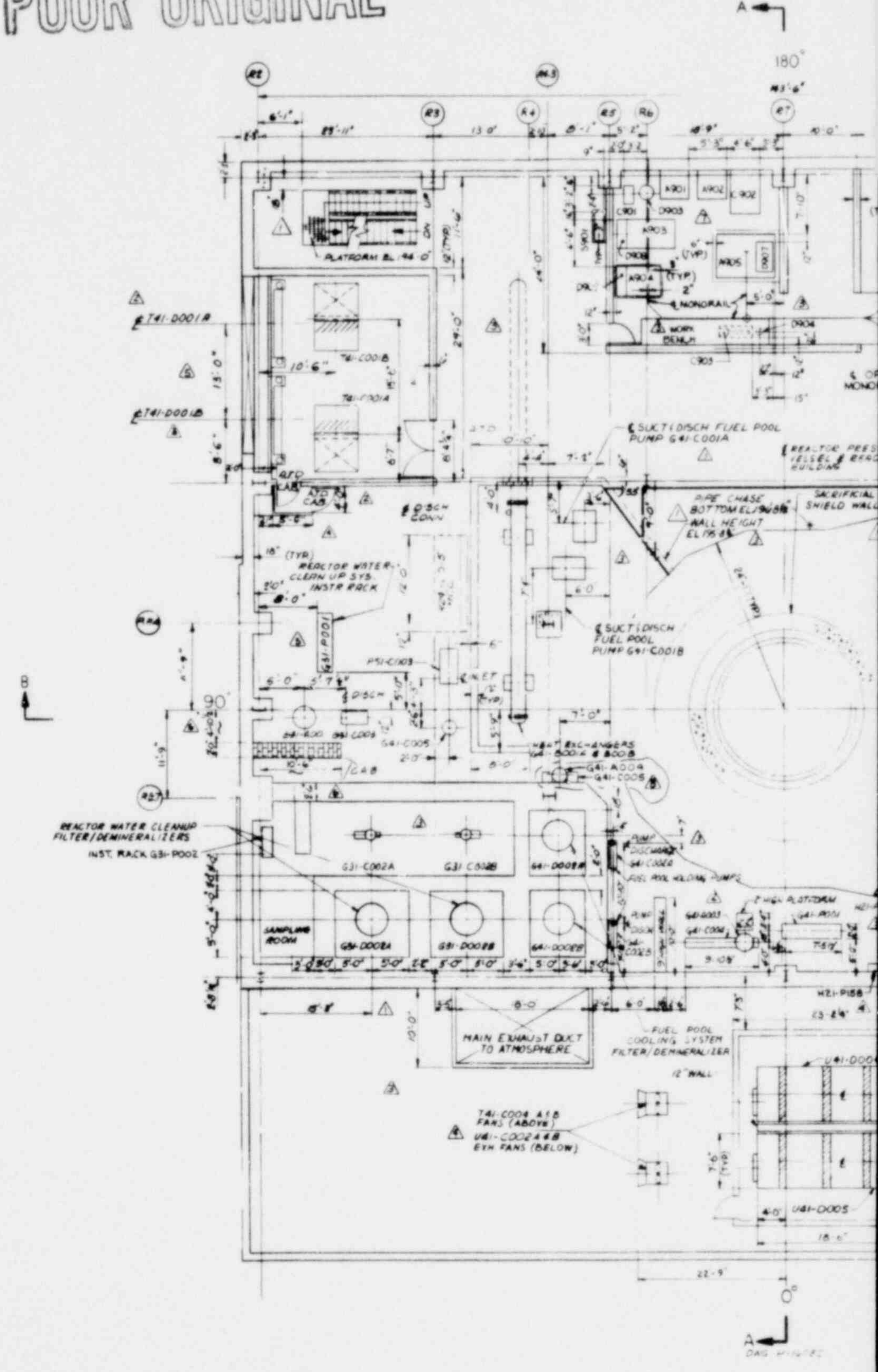
2104 099

and 14 feet high. Figures 2-14 and 2-15 provide additional information pertaining to the arrangement plan of the pools.

The base plate of each module is supported on all four corners by 2-inch thick foot pads. The foot pads rest on 6-inch thick corner-support pads which in turn rest on the fuel pool floor liner. This raises the base plate of the module a minimum of 8 inches above the floor of the fuel pool, allowing sufficient clear area to permit natural circulation of cooling water to the modules without taking credit for sources of forced cooling.

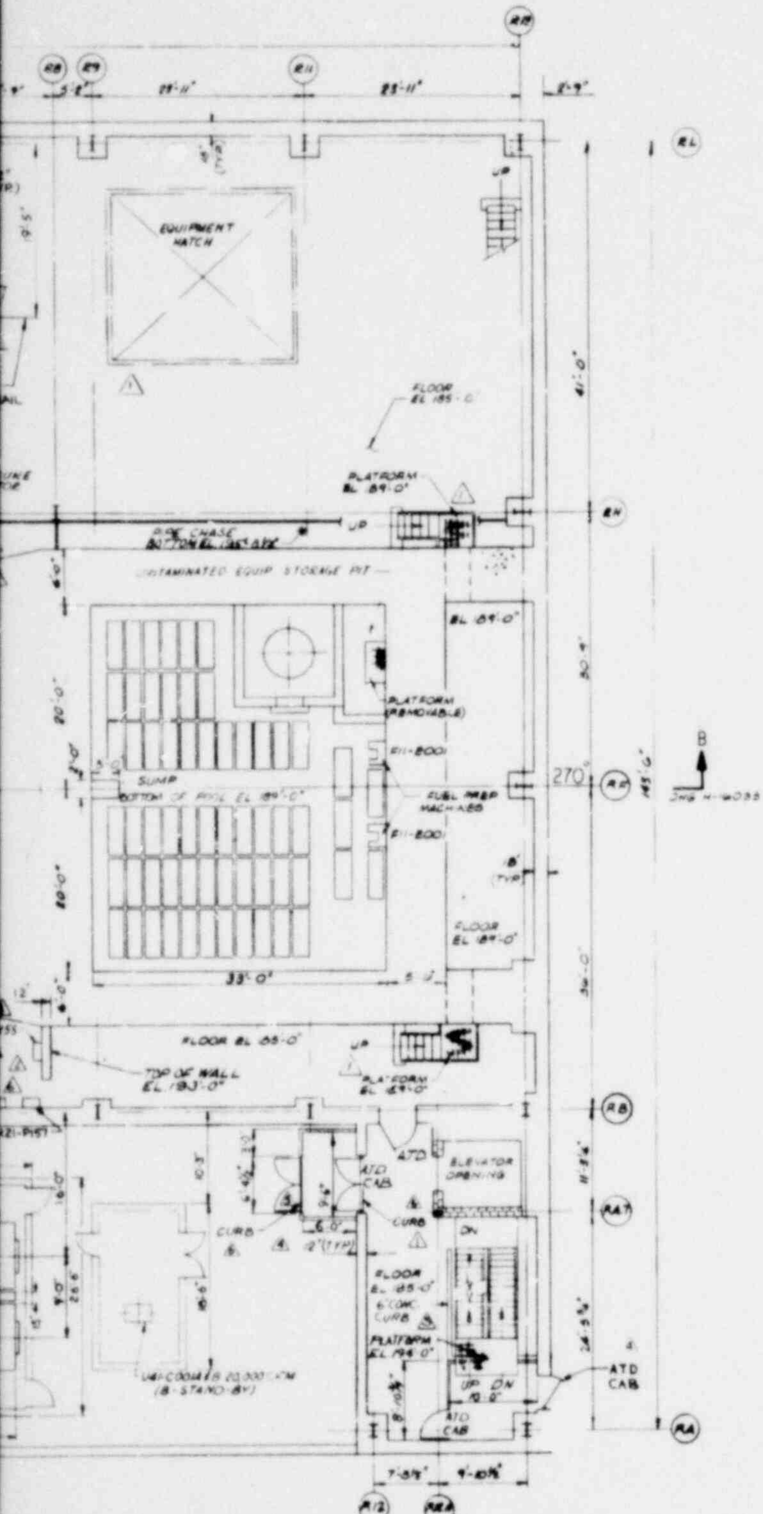
2104 100

POOR ORIGINAL



FLOOR EL. 185'-0"

2104 101



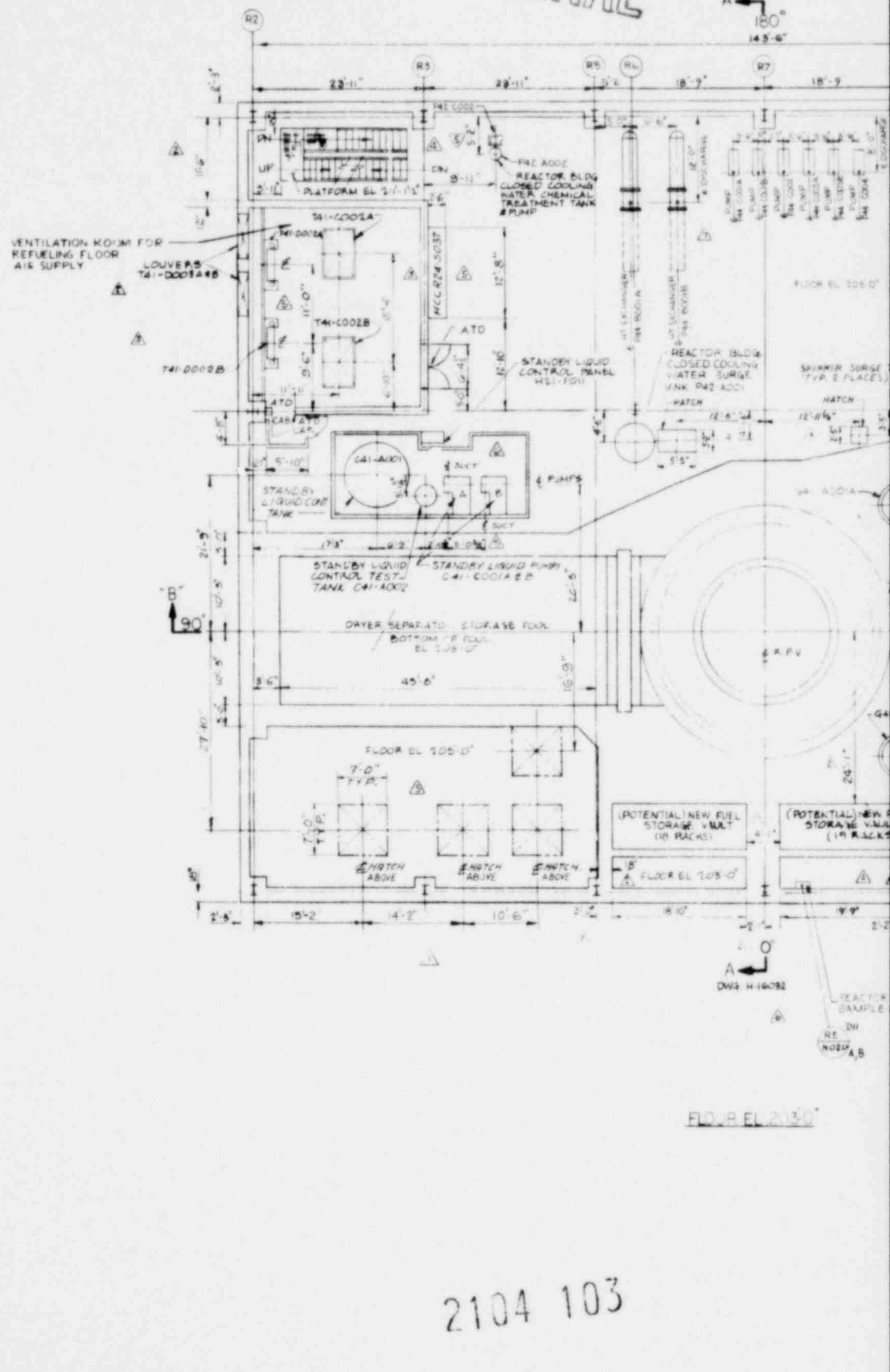
N

POOR ORIGINAL

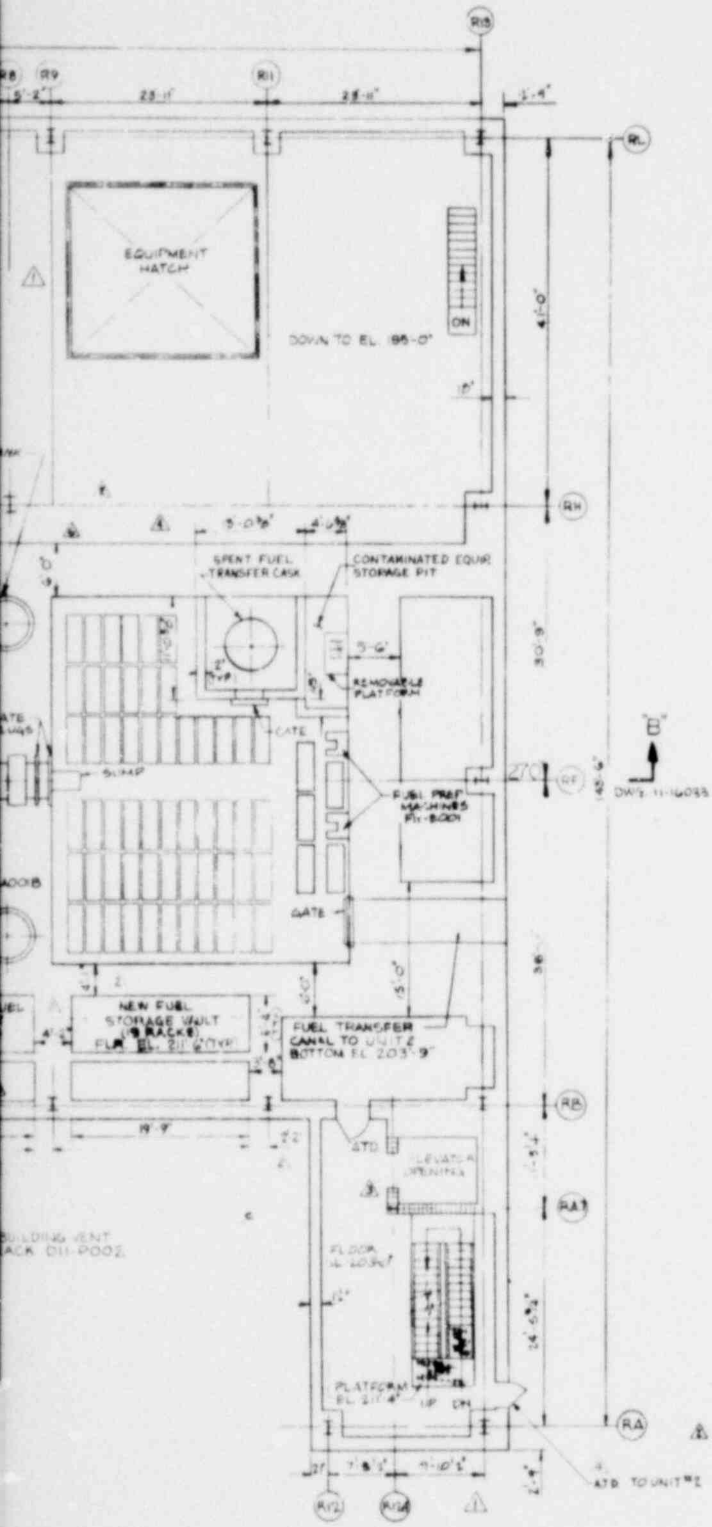
2104 102

Figure 2-1 Unit 1 Reactor Building Equipment Locations, El. 185'-0"

POOR ORIGINAL



2104 103

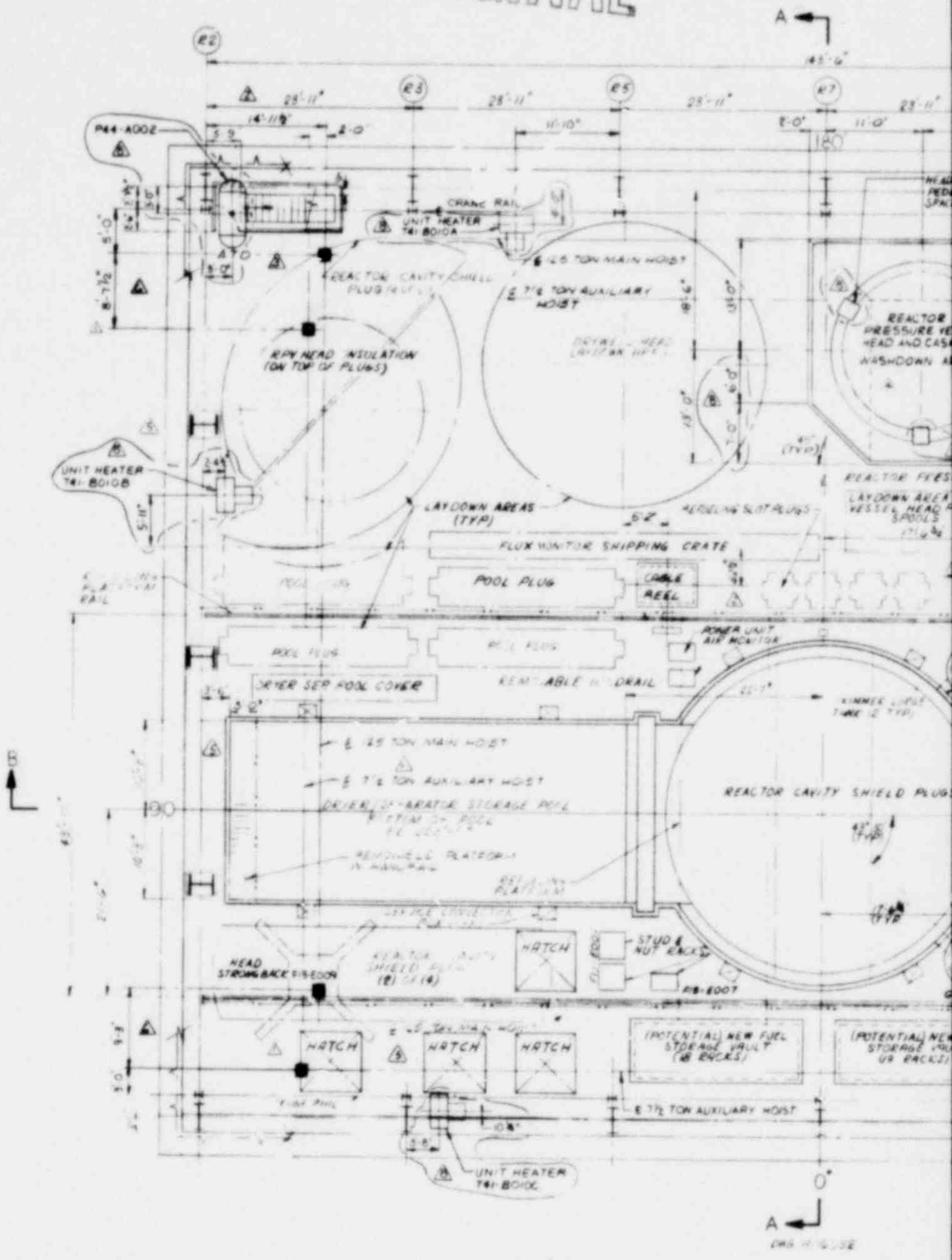


POOR ORIGINAL

2104 104

Figure 2-2 Unit 1 Reactor Building Equipment Location, El. 203'-0"

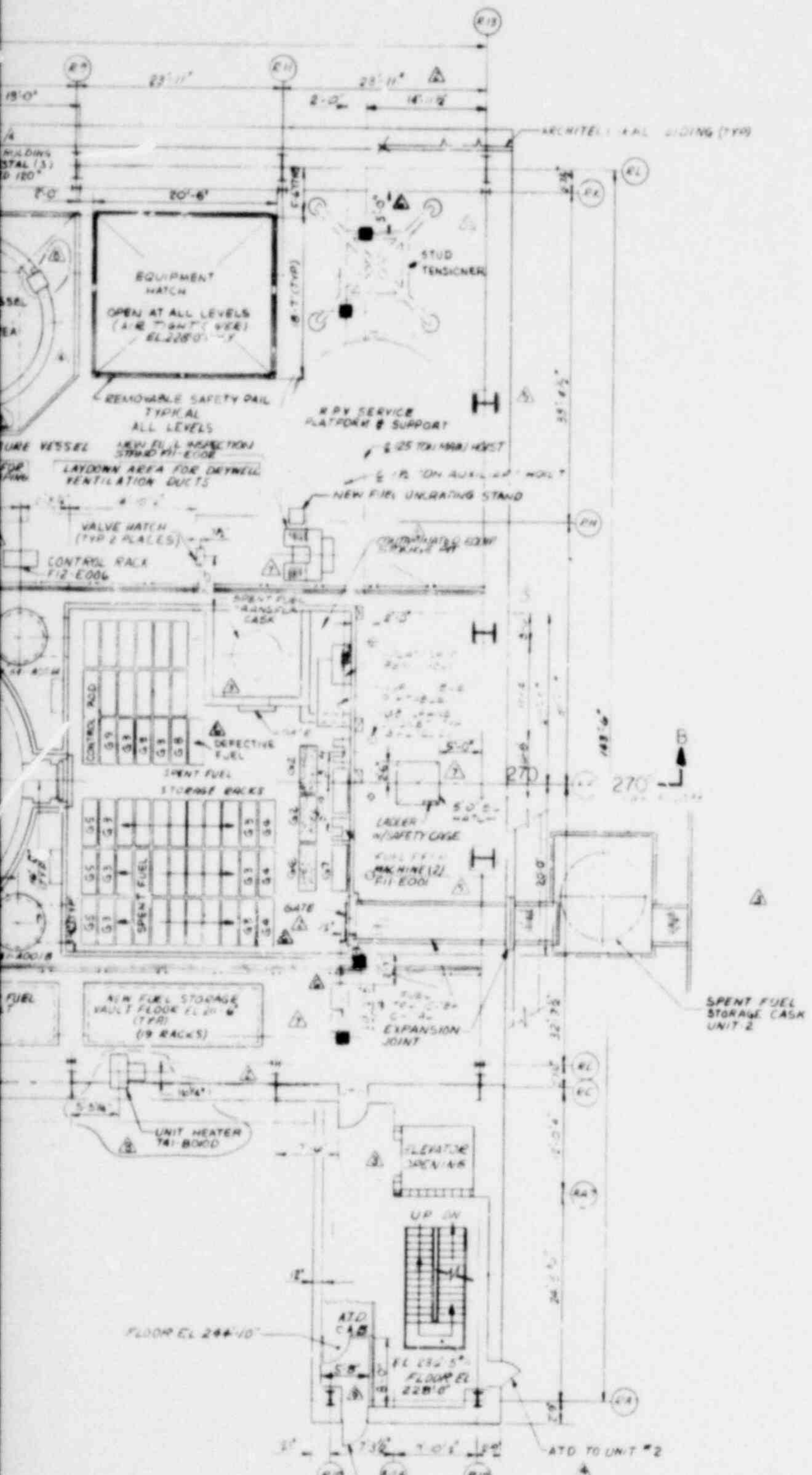
POOR ORIGINAL



FLOOR EL 228'-0"

2104 105

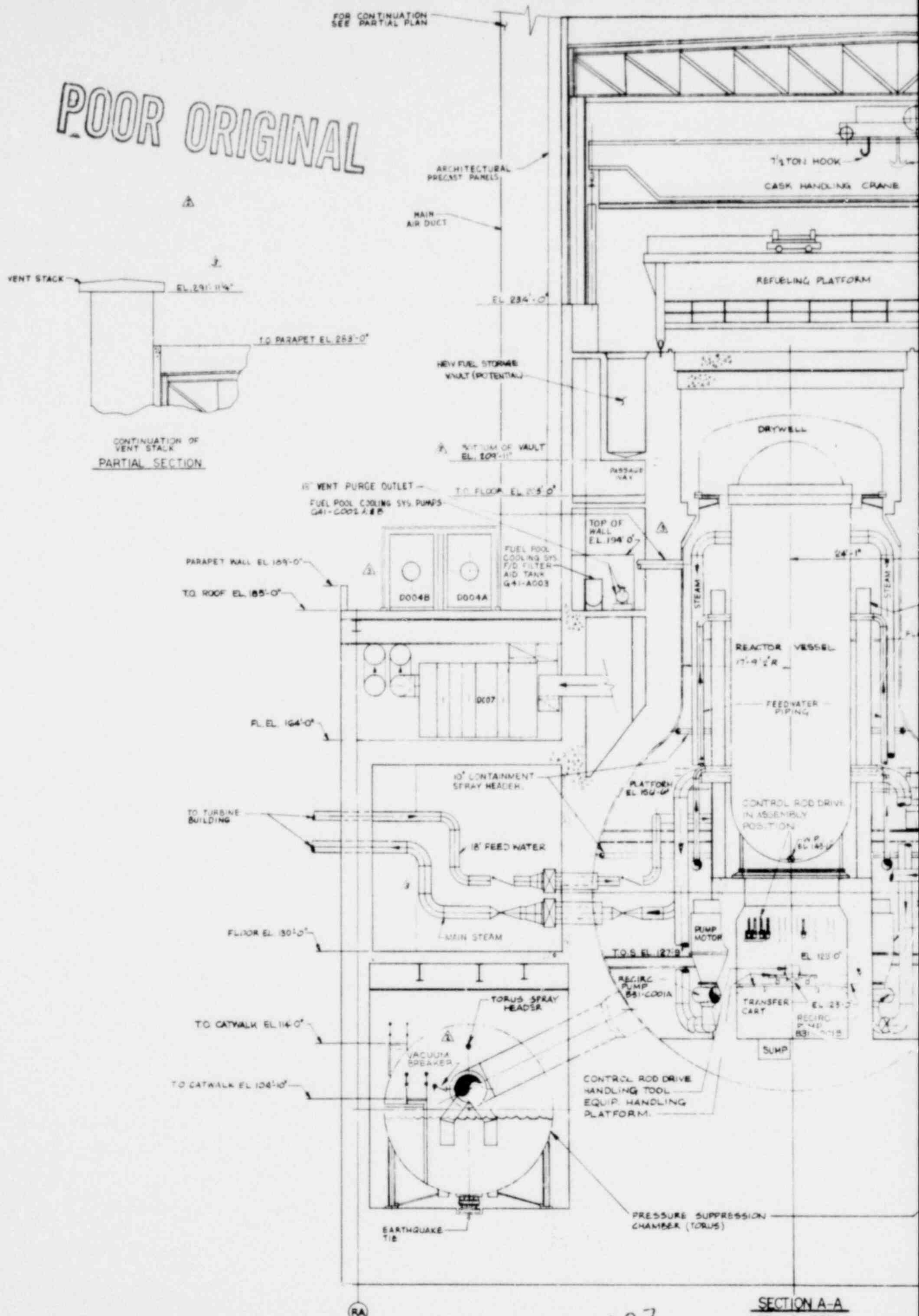
POOR ORIGINAL



2104 106

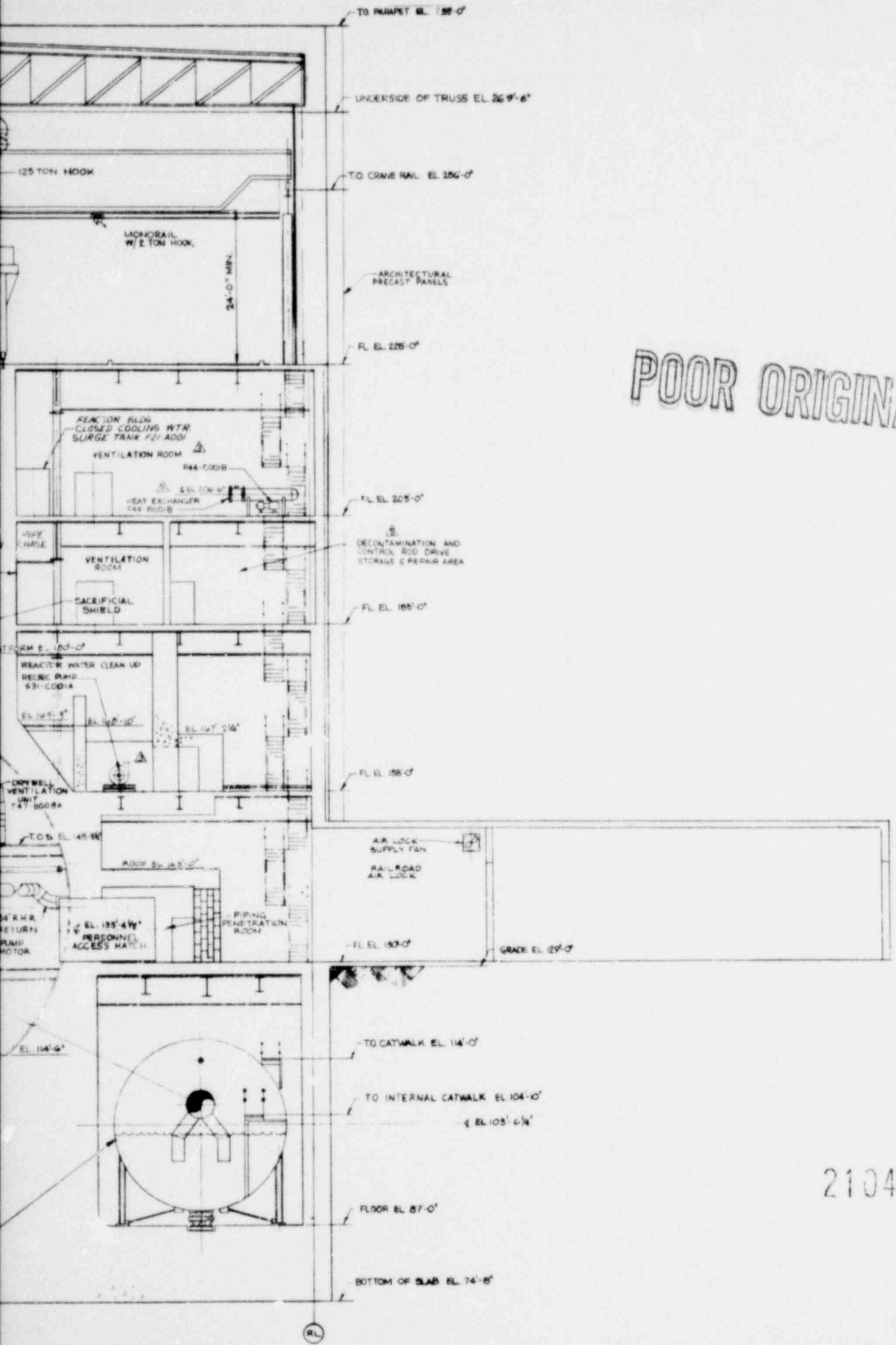
Figure 2-3 Unit 1 Reactor Building Equipment Locations, El. 228'-0"

POOR ORIGINAL



SECTION A-A
LOOKING NORTH

2104 107

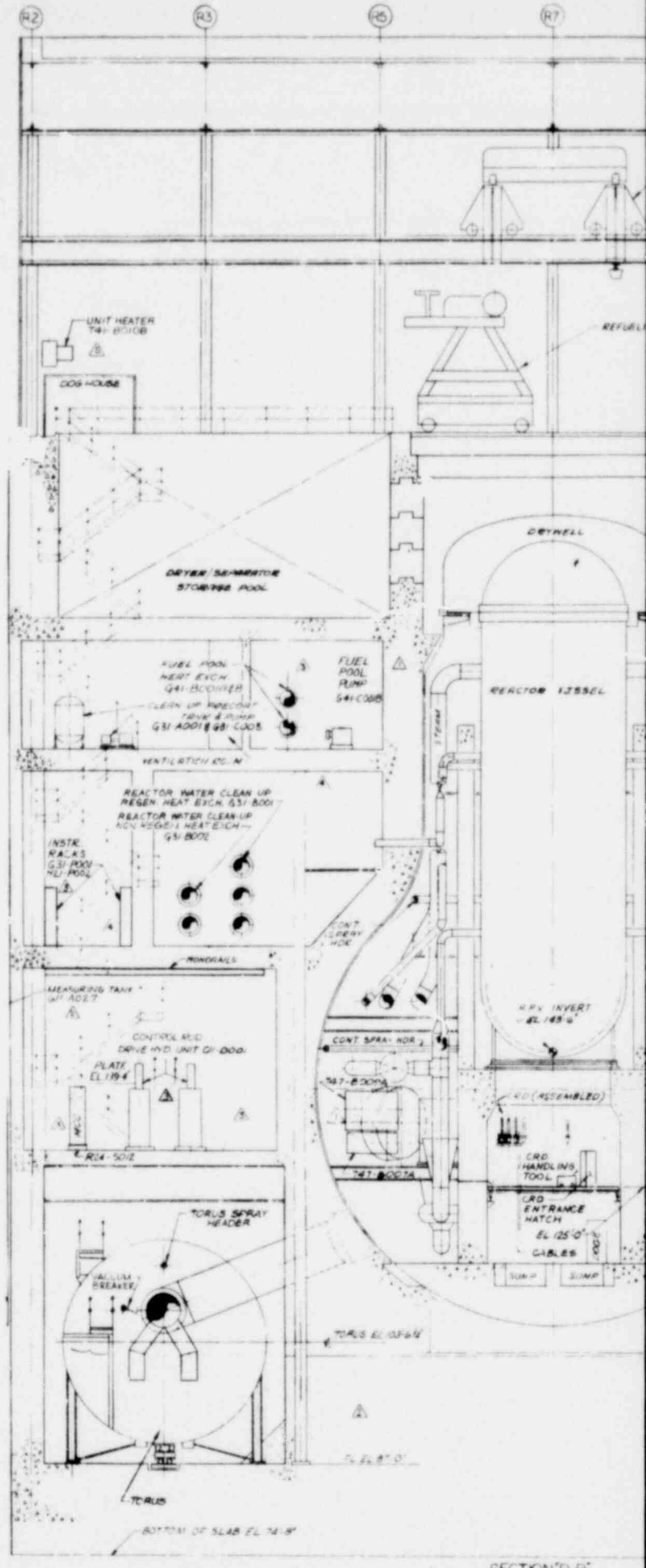


POOR ORIGINAL

2104 108

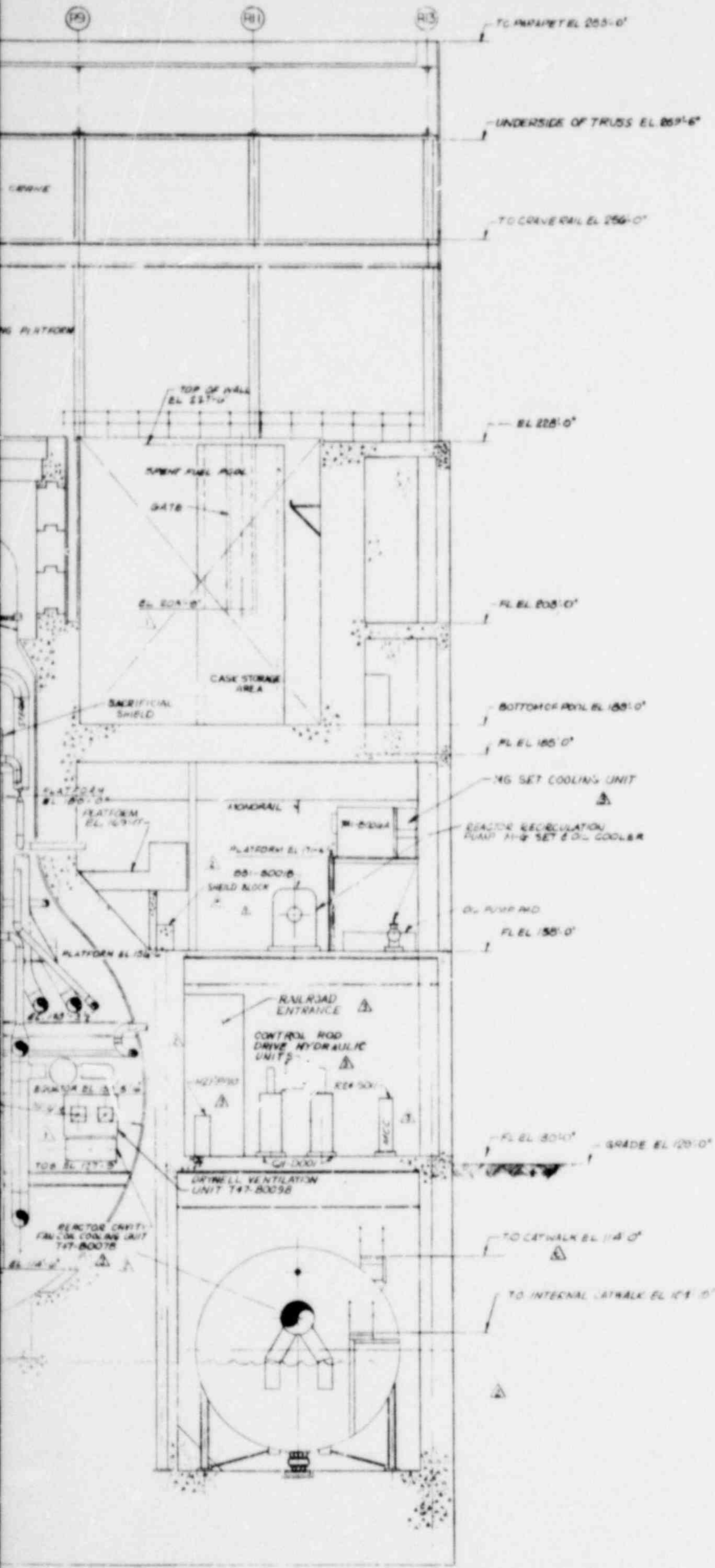
Figure 2-4 Unit 1 Reactor Building, Section A-A of Figure 2-1

POOR ORIGINAL



SECTION "B-B"
LOOKING EAST

2104 109

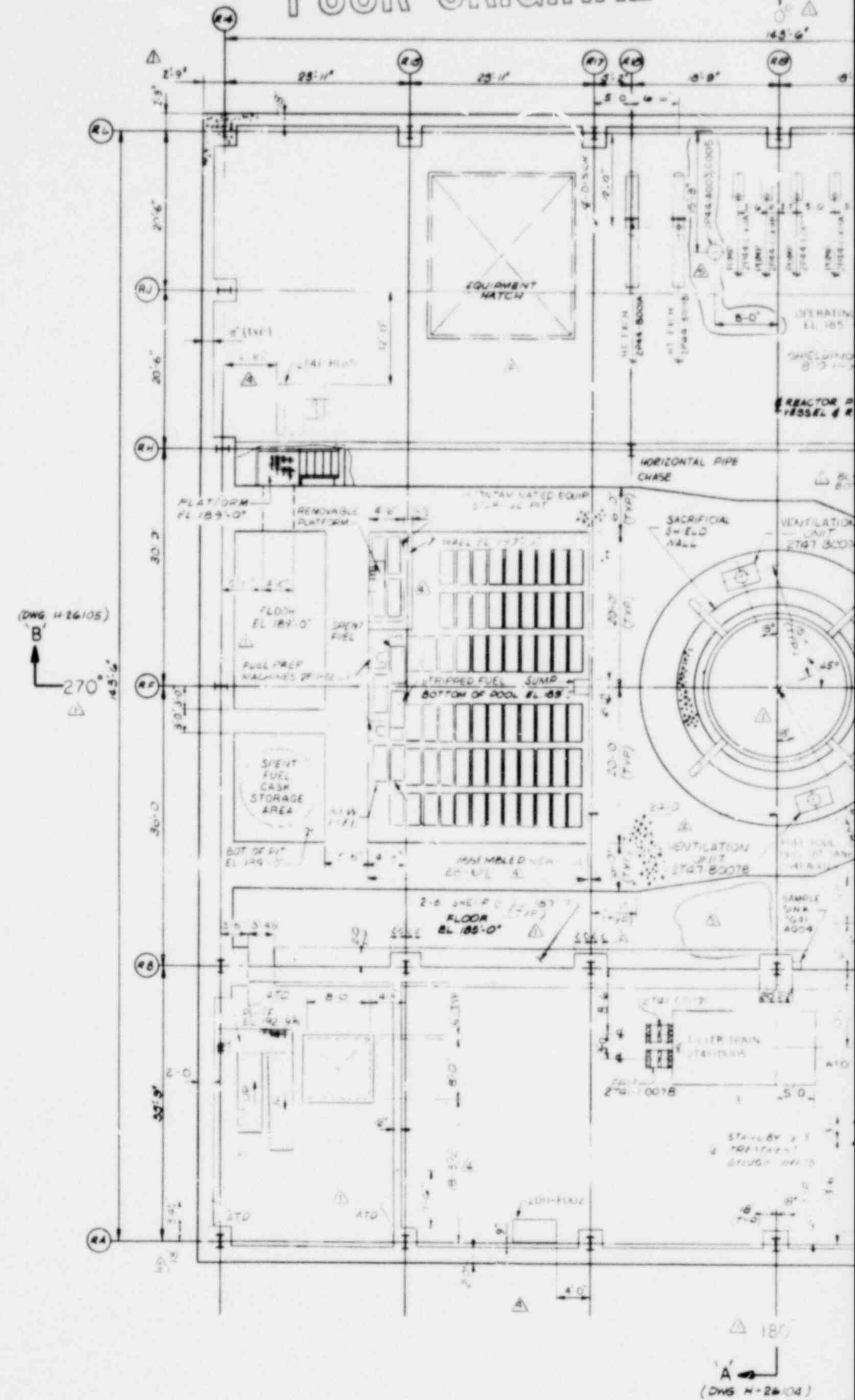


POOR ORIGINAL

2104 110

Figure 2-5 Unit 1 Reactor Building, Section B-B of Figure 2-1

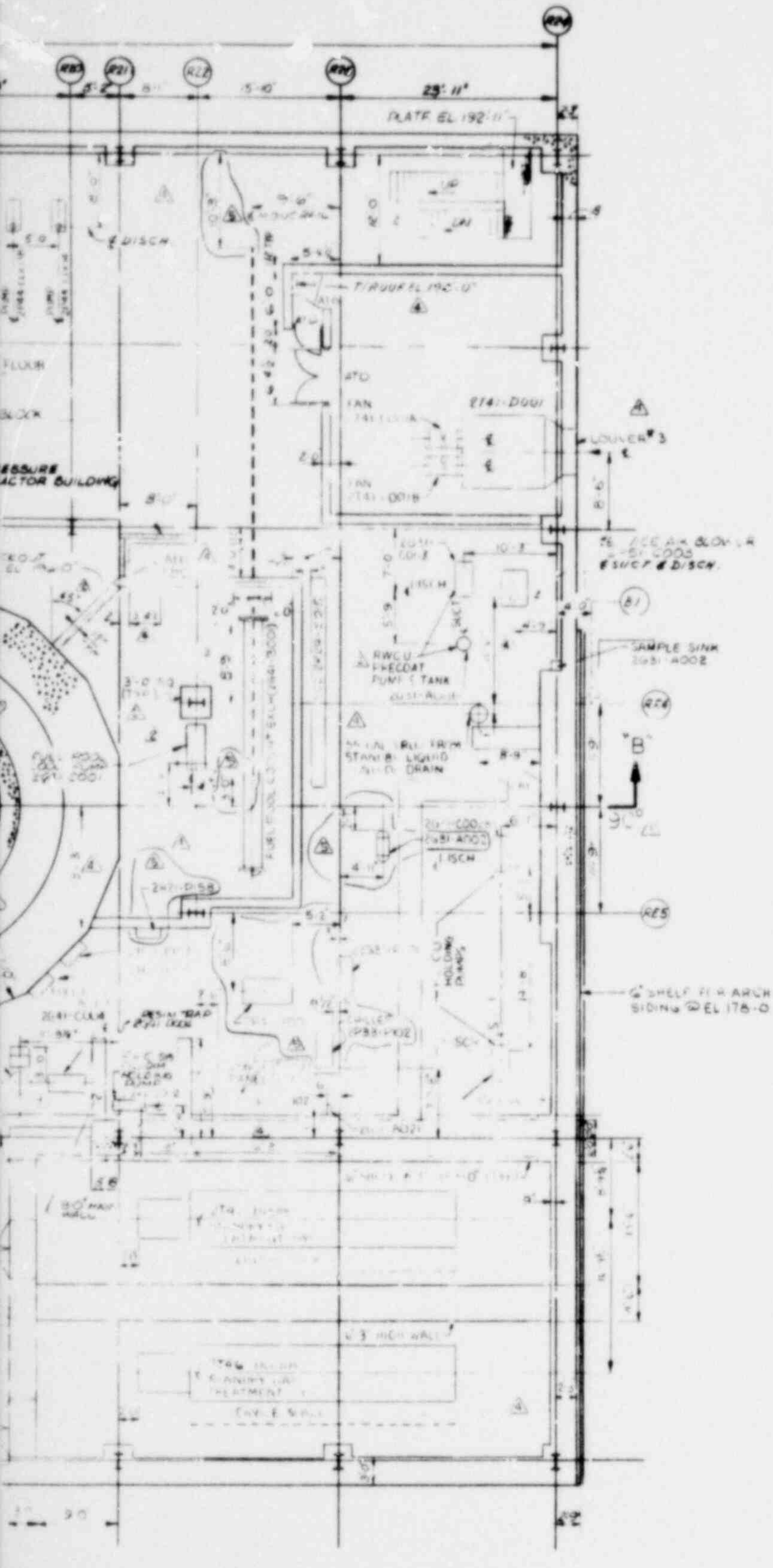
POOR ORIGINAL



REACTOR BUILDING

FLOOR EL 185'-0

2104 111

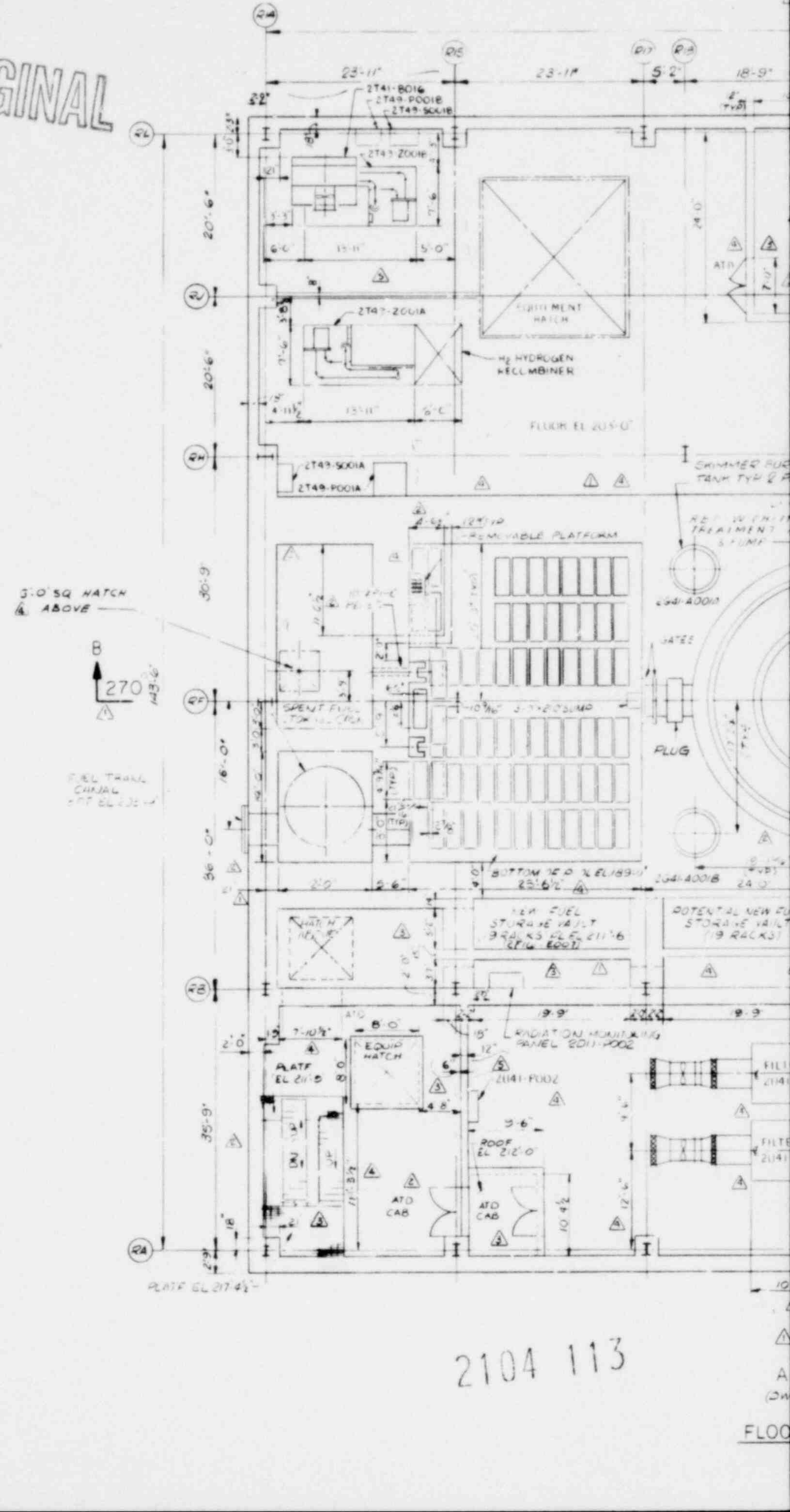


POOR ORIGINAL

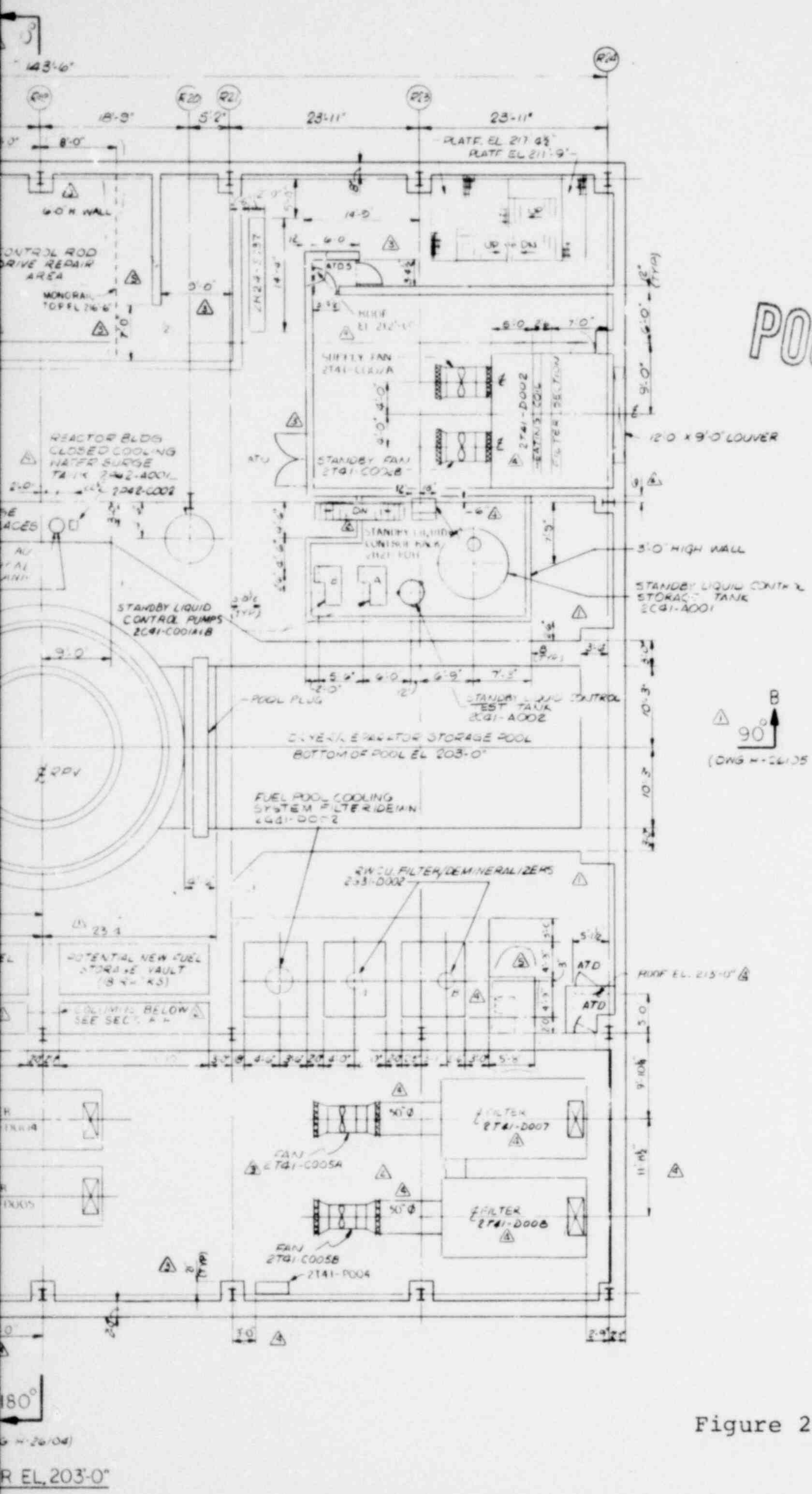
2104 112

Figure 2-6 Unit 2 Reactor Building Equipment Locations, El. 185'-0"

POOR ORIGINAL



2104 113

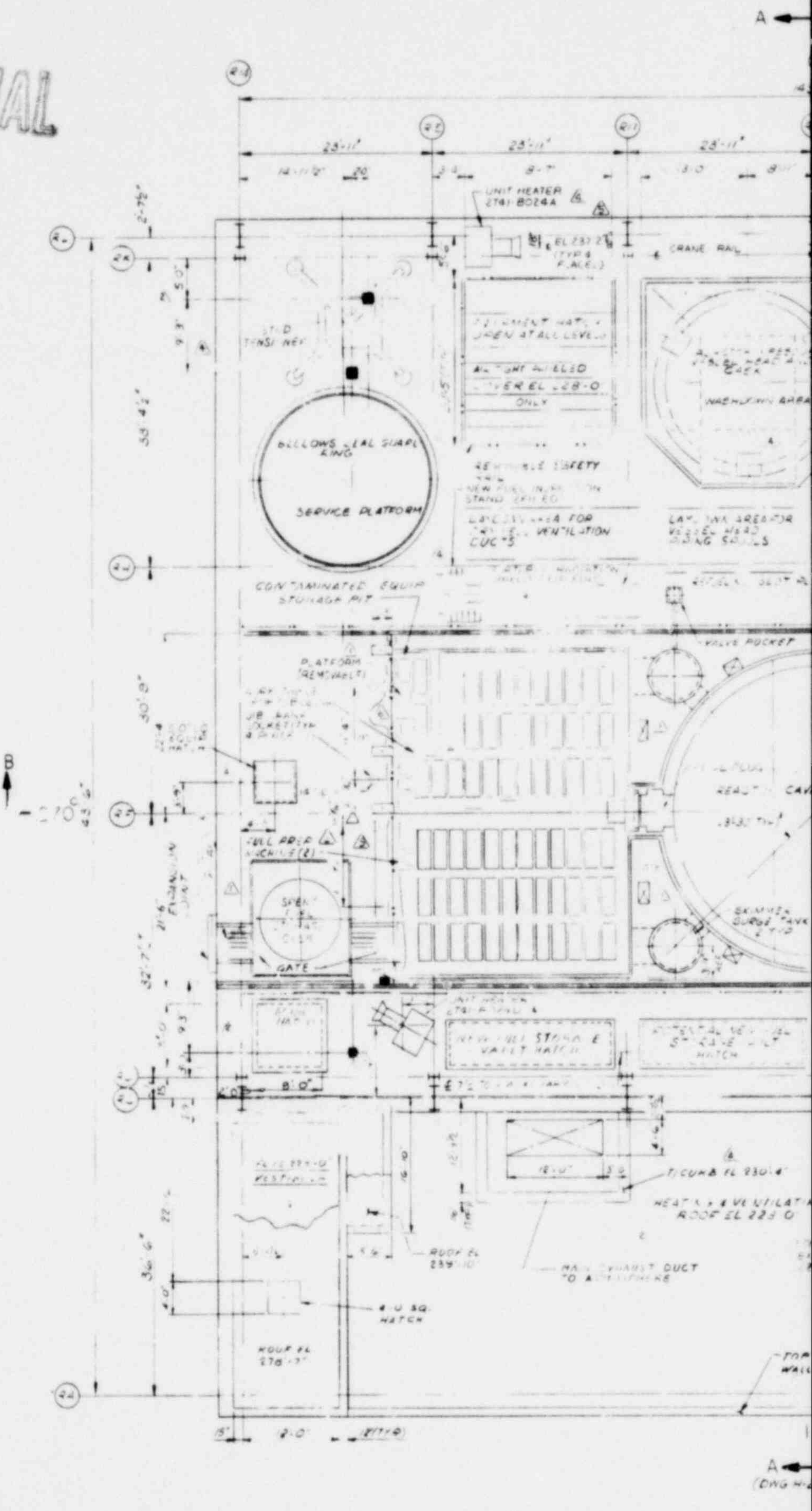


POOR ORIGINAL

2104 114

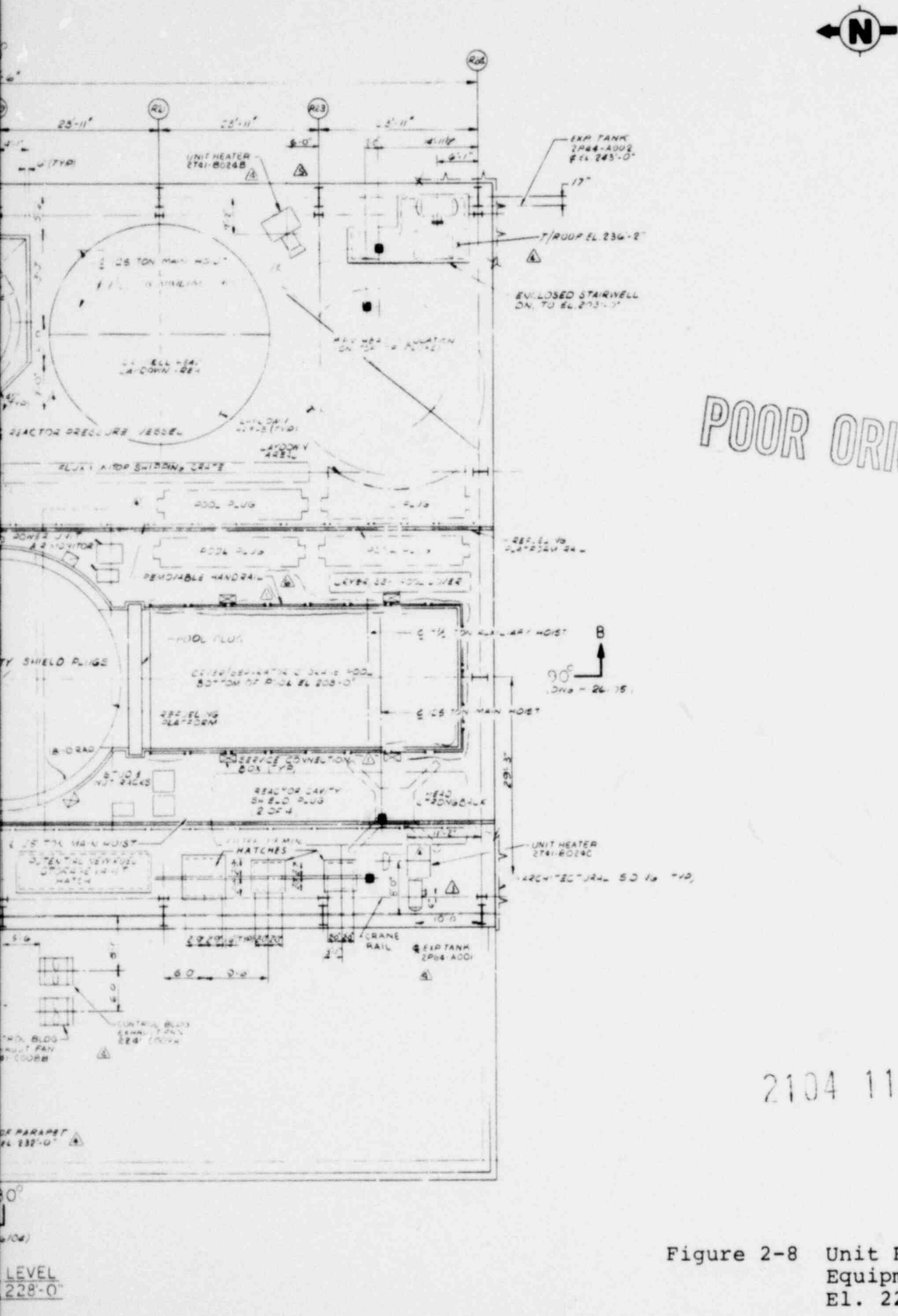
Figure 2-7 Unit 2 Reactor Building Equipment Locations, El. 203'-0"

POOR ORIGINAL



2104 115

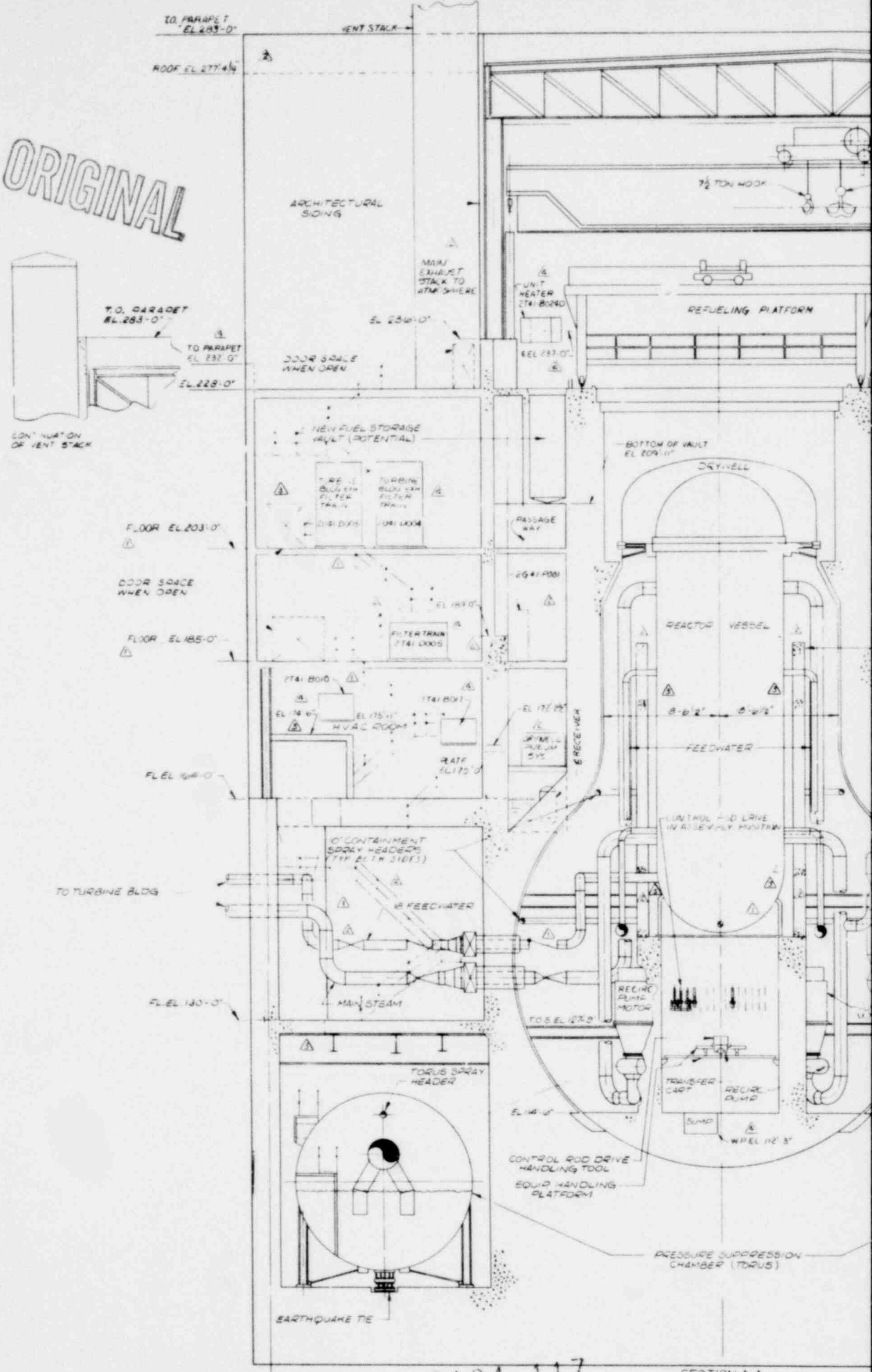
REFUELING
FLOOR EL



POOR ORIGINAL

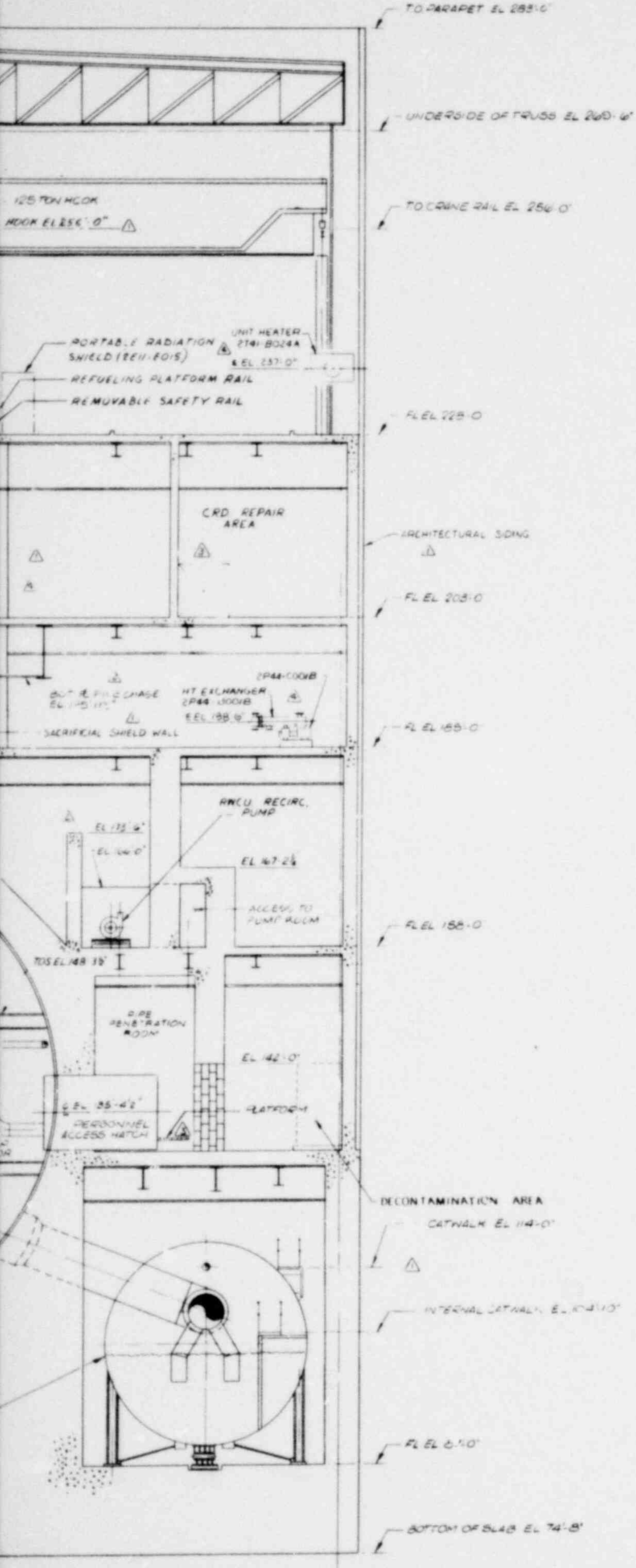
2104 116

Figure 2-8 Unit Reactor Building Equipment Locations, El. 228'-0"



2104 117

SECTION A-A
LOOKING NORTH

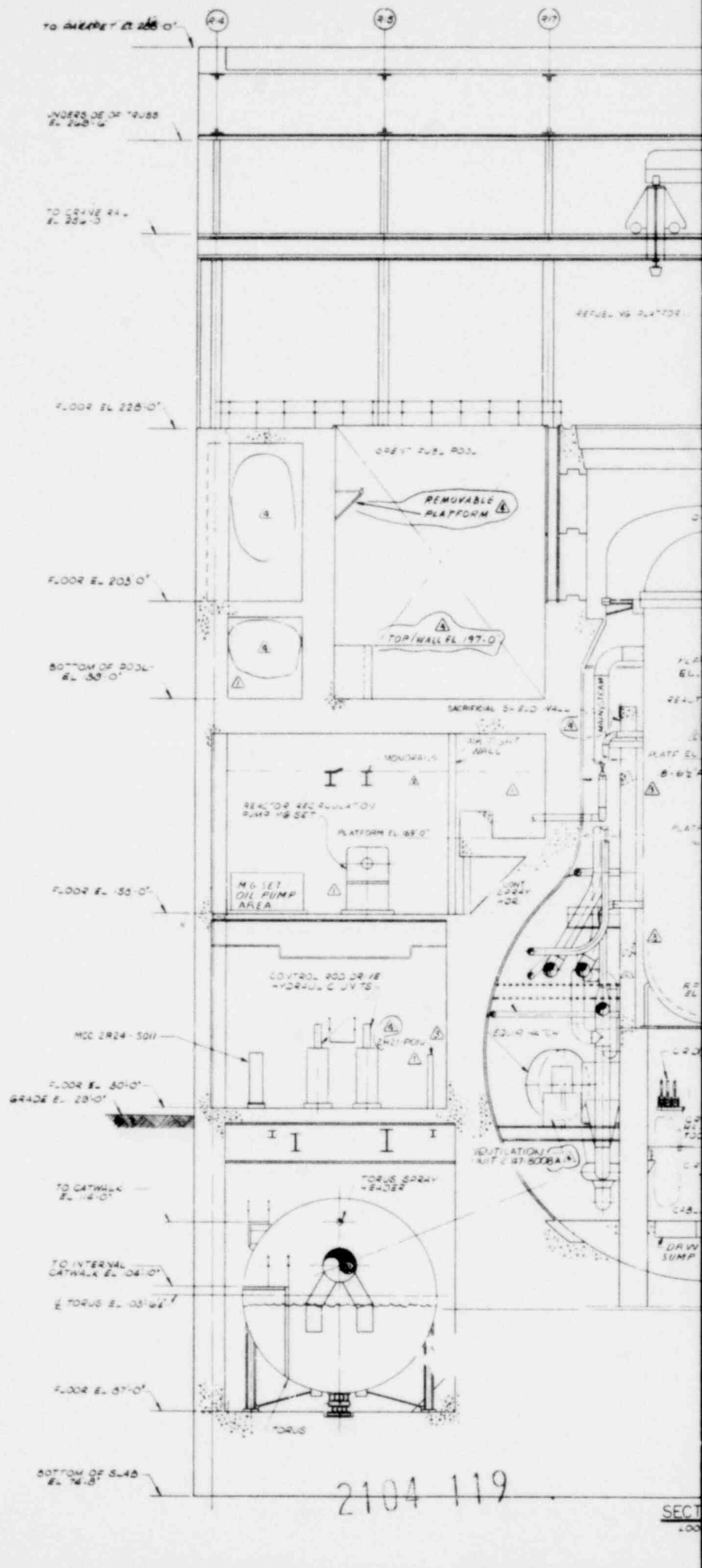


POOR ORIGINAL

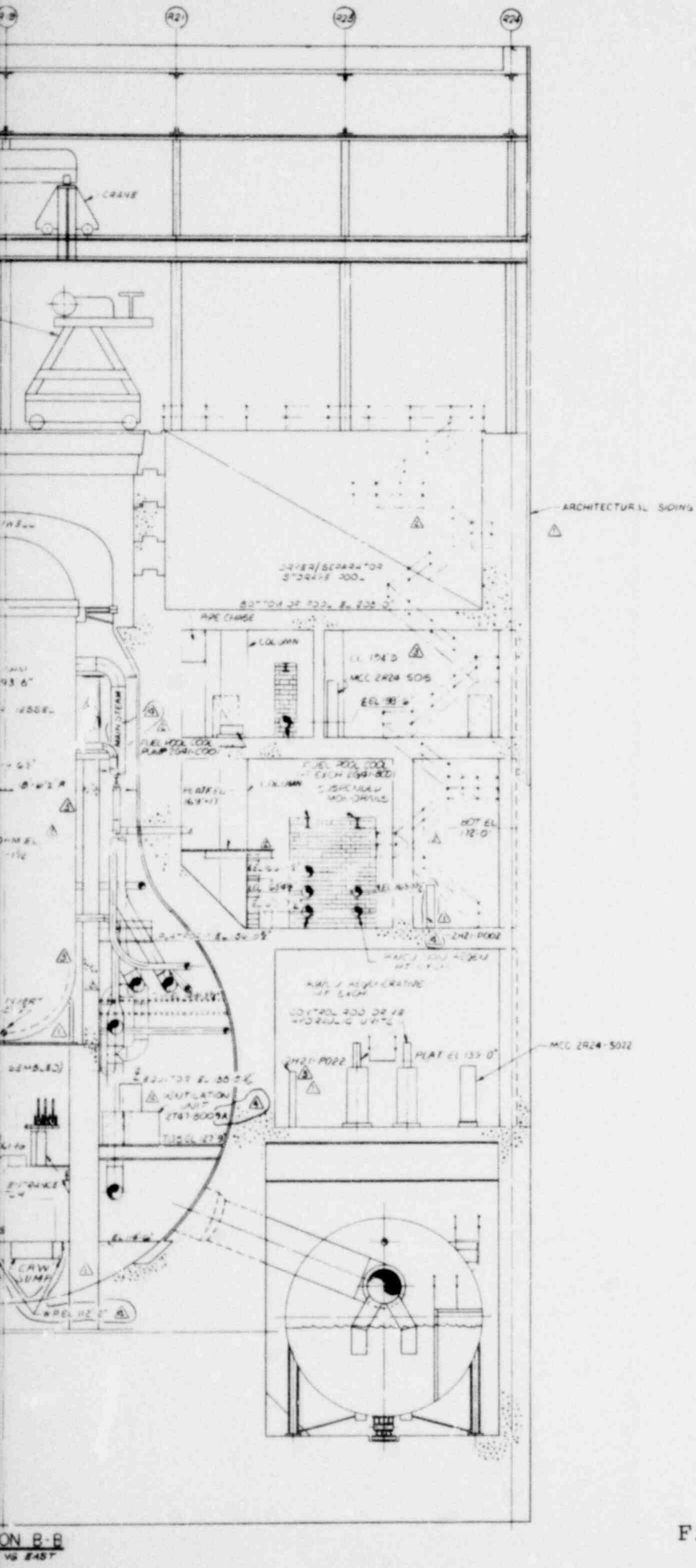
2104 118

Figure 2-9 Unit 2 Reactor Building, Section A-A of Figure 2-6

POOR ORIGINAL



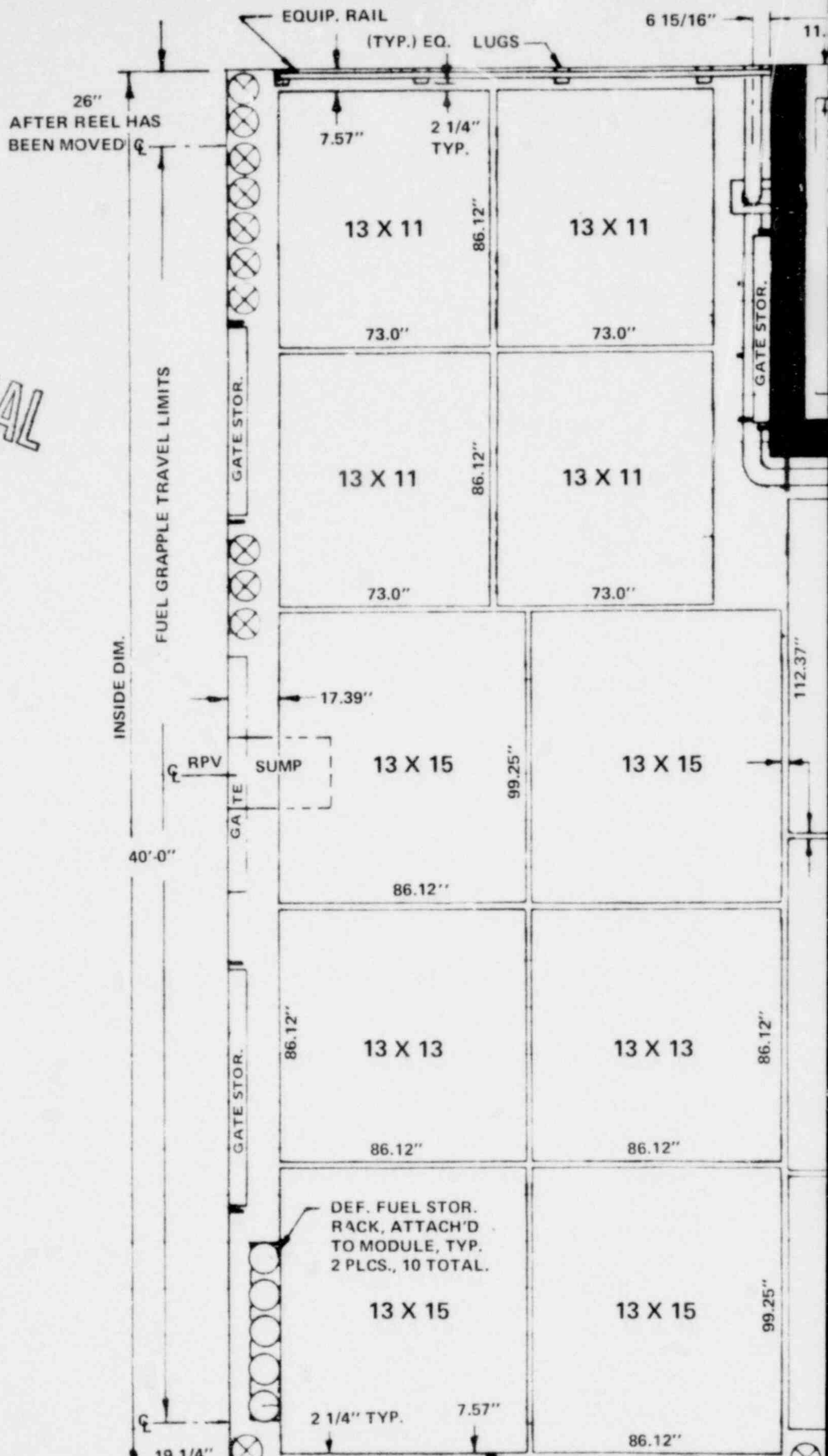
2104 119



POOR ORIGINAL

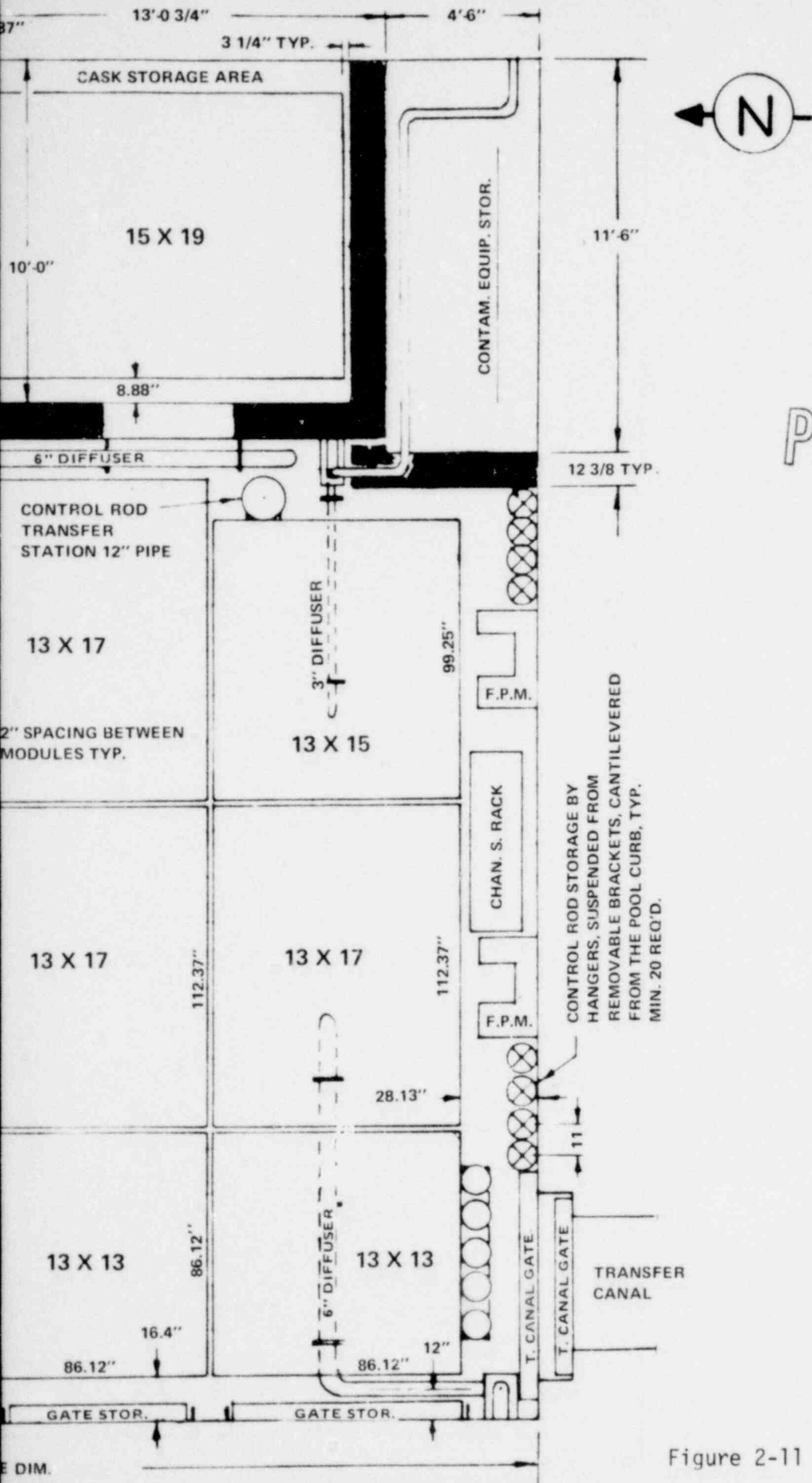
2104 120

Figure 2-10 Unit 2 Reactor Building, Section B-B of Figure 2-6



EQUIP. RAIL
(TYP.) EQ.
LUGS
6 15/16"
11.
26"
AFTER REEL HAS BEEN MOVED
7.57"
2 1/4" TYP.
13 X 11
13 X 11
86.12"
73.0"
73.0"
GATE STOR.
13 X 11
13 X 11
86.12"
73.0"
73.0"
FUEL GRAPPLE TRAVEL LIMITS
INSIDE DIM.
40'-0"
17.39"
112.37"
RPV
SUMP
GATE
13 X 15
13 X 15
99.25"
86.12"
GATE STOR.
13 X 13
13 X 13
86.12"
86.12"
86.12"
86.12"
DEF. FUEL STOR. RACK, ATTACH'D TO MODULE, TYP. 2 PLCS., 10 TOTAL.
13 X 15
13 X 15
99.25"
2 1/4" TYP.
7.57"
86.12"
19 1/4"
EQUIP. SUPT. LUG, TYPE.
EQUIP. SUPT. RAIL
33'-0" INSID

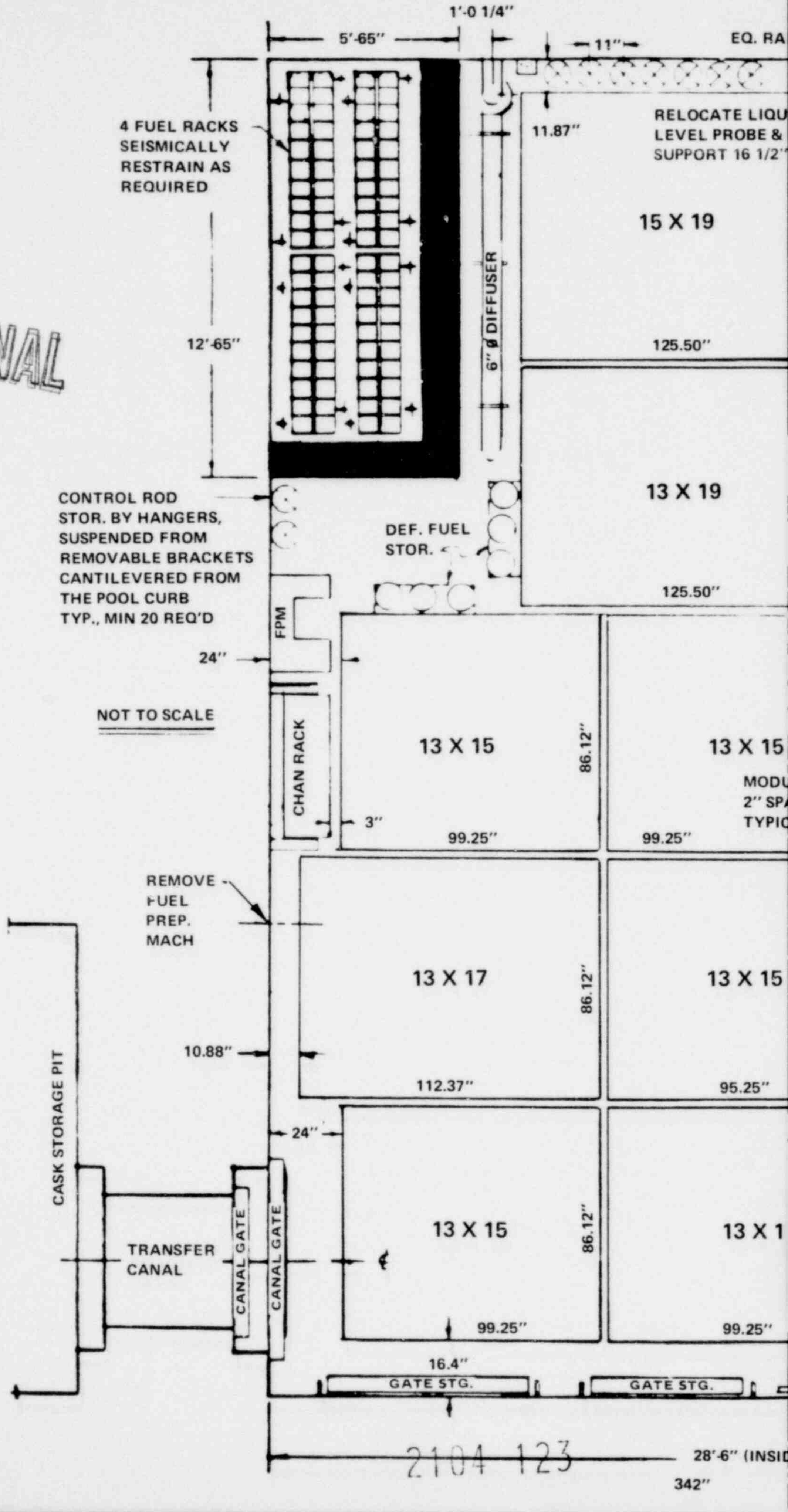
2104 121



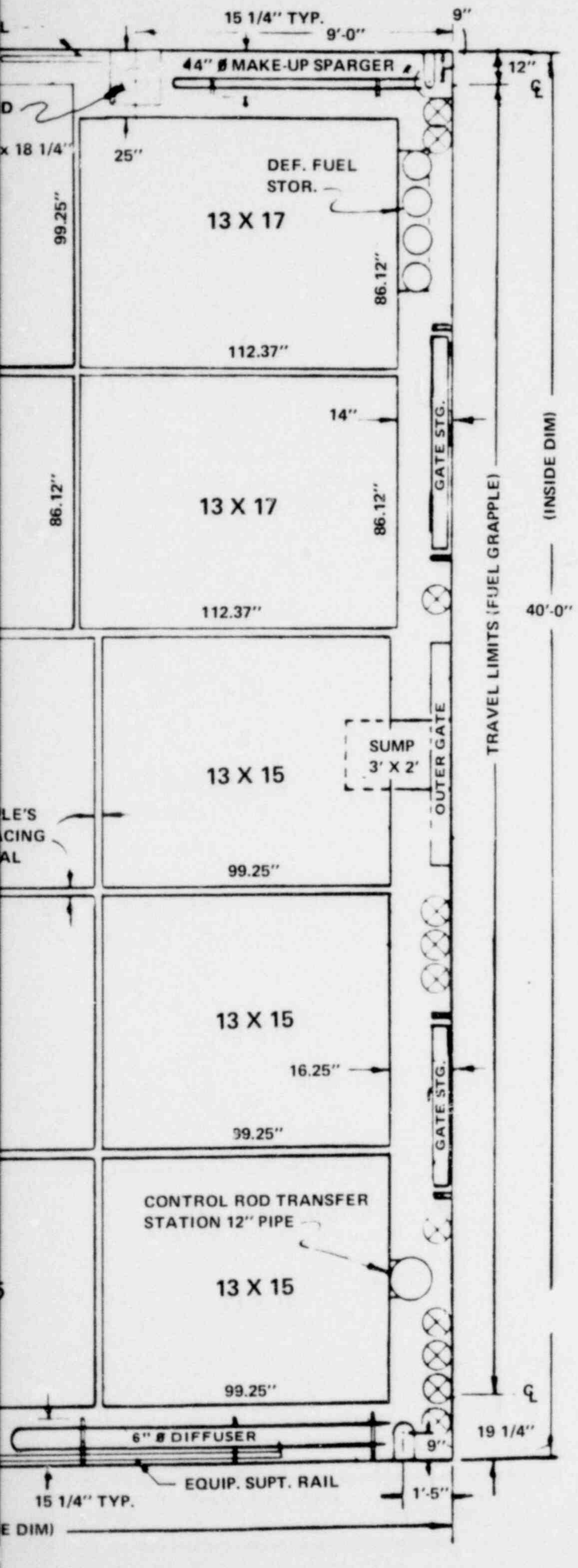
POOR ORIGINAL

2104 122

Figure 2-11 Hatch 1 High Density Fuel Storage System Arrangement


1'-0 1/4"
5'-65"
11"
EQ. RA
4 FUEL RACKS SEISMICALLY RESTRAIN AS REQUIRED
11.87"
RELOCATE LIQU
LEVEL PROBE &
SUPPORT 16 1/2"
15 X 19
12'-65"
6" Ø DIFFUSER
125.50"
13 X 19
CONTROL ROD STOR. BY HANGERS, SUSPENDED FROM REMOVABLE BRACKETS CANTILEVERED FROM THE POOL CURB TYP., MIN 20 REQ'D
DEF. FUEL STOR.
125.50"
FPM
24"
NOT TO SCALE
CHAN RACK
13 X 15
86.12"
13 X 15
MODU
2" SP
TYPIC
3"
99.25"
99.25"
REMOVE FUEL PREP. MACH
13 X 17
86.12"
13 X 15
10.88"
112.37"
95.25"
CASK STORAGE PIT
24"
TRANSFER CANAL
CANAL GATE
CANAL GATE
13 X 15
86.12"
13 X 1
99.25"
99.25"
16.4"
GATE STG.
GATE STG.
28'-6" (INSI
342"

2104 123



N

2104 124

Figure 2-12 Hatch 2 High Density Fuel Storage System Arrangement

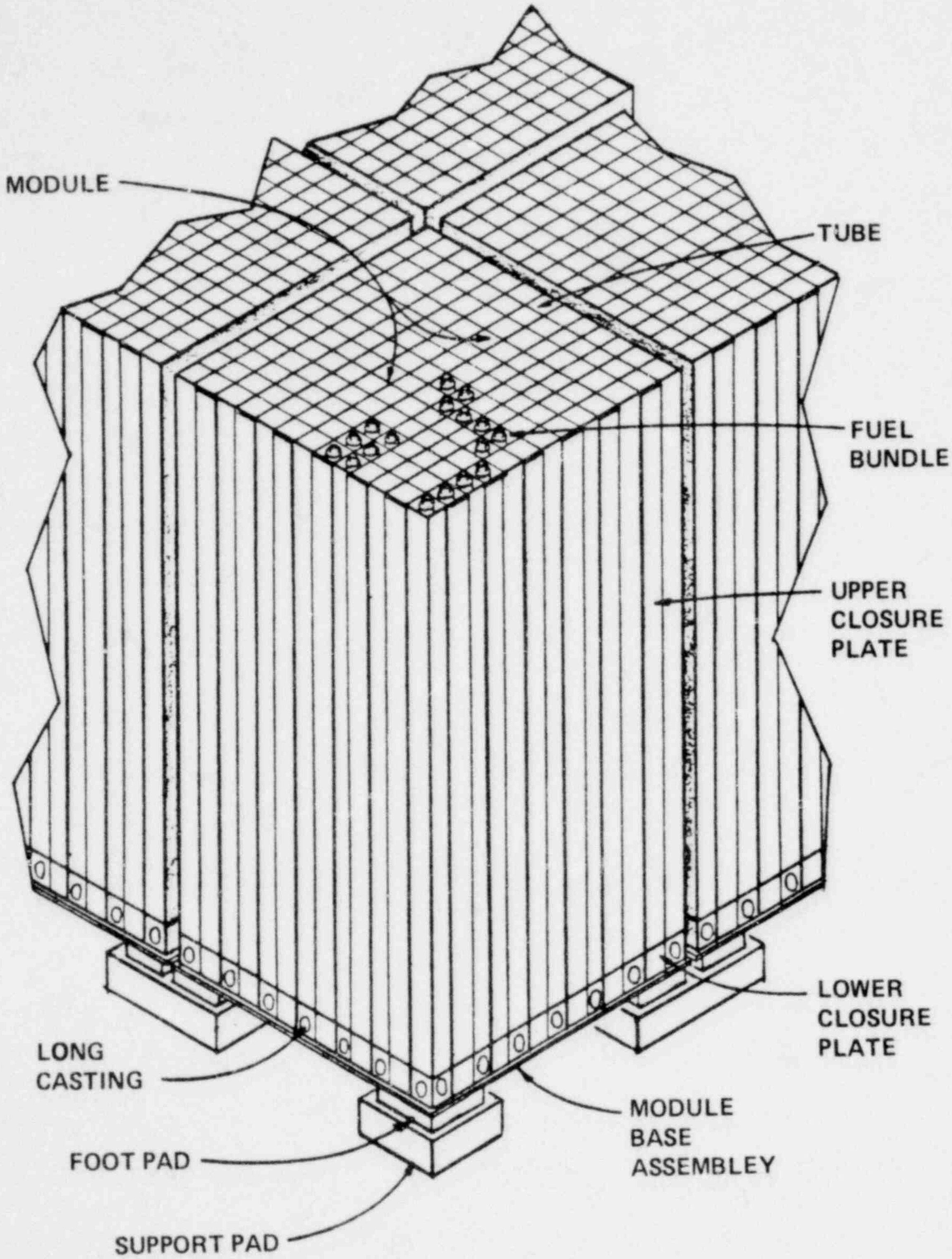


Figure 2-13 Isometric View of Modules

2104 125

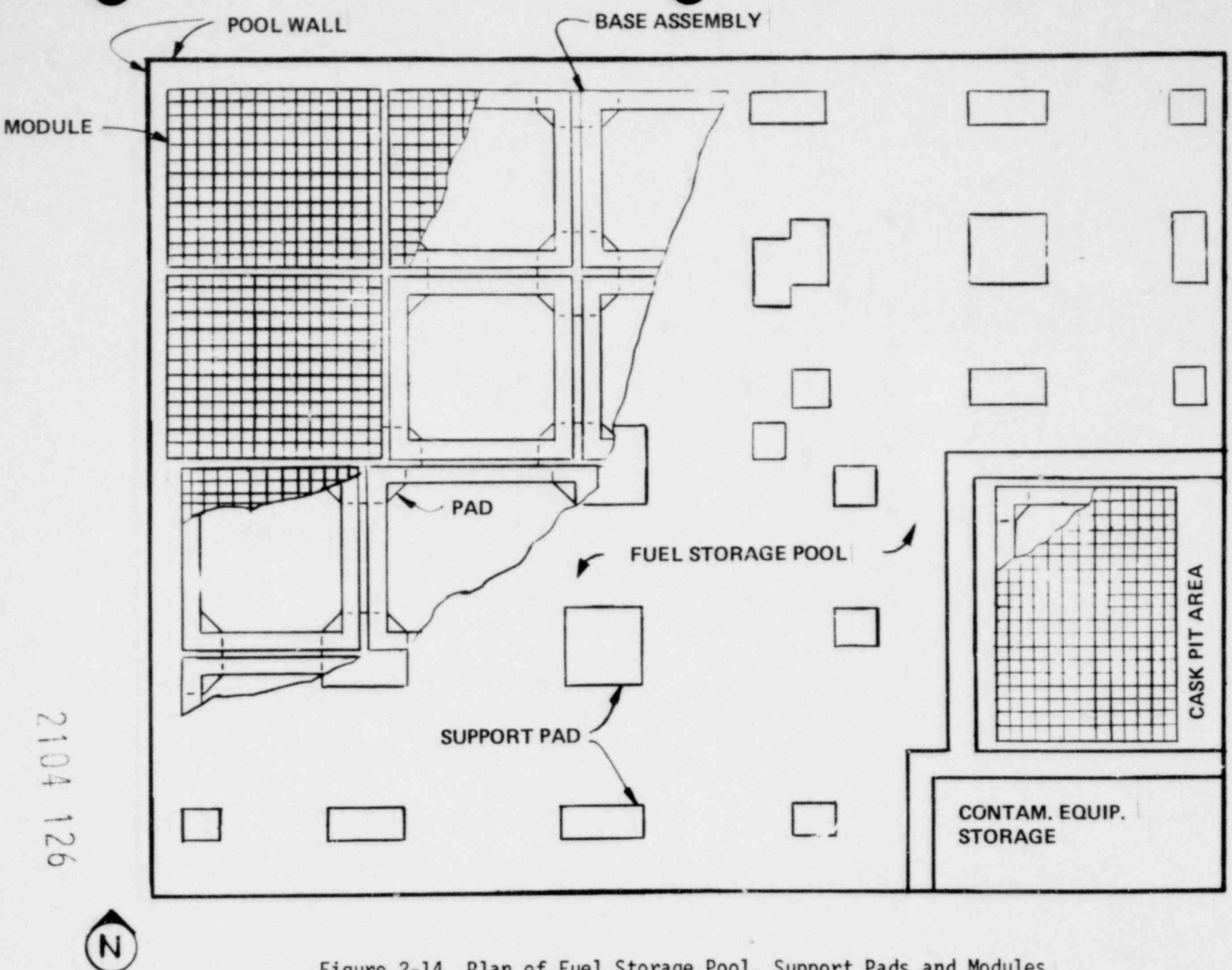


Figure 2-14 Plan of Fuel Storage Pool, Support Pads and Modules

2104 126

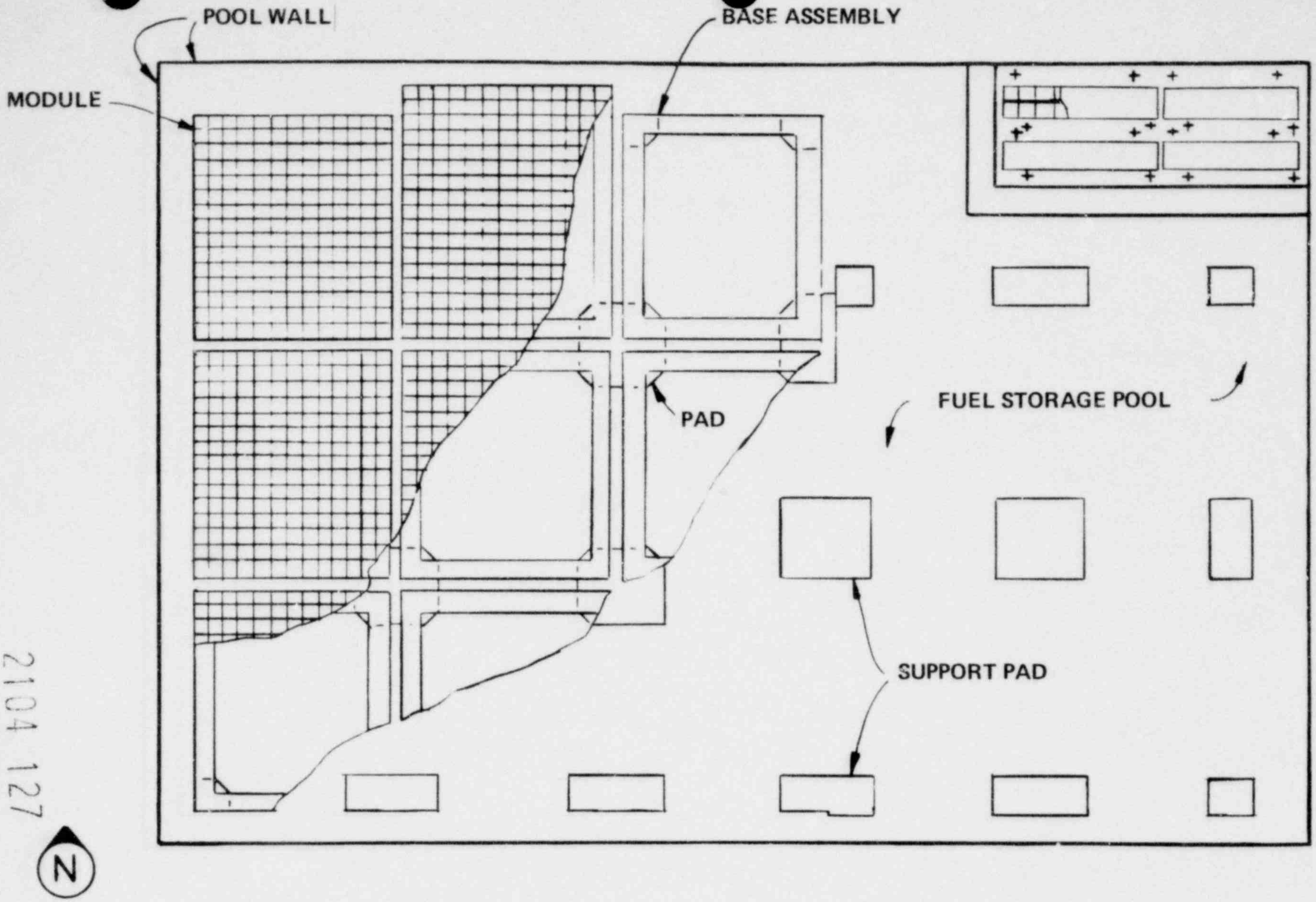


Figure 2-15 Plan of Fuel Storage Pool, Support Pads and Modules

2104 127

3.0 DESIGN BASES

The new spent fuel storage system was designed to conform to the applicable provisions of the following codes, standards, and regulations:

1. General Design Criterion 2 (per 10CFR50, Appendix A) as related to components important to safety being capable of withstanding the effects of natural phenomena.

2. General Design Criterion 3 as related to protection against fire hazards.

3. General Design Criterion 4 as related to components being able to accommodate the effects of and to be compatible with the environmental conditions associated with normal operation and postulated accidents.

4. General Design Criterion 62 as related to the prevention of criticality by physical systems.

5. Regulatory Guide 1.13 as it relates to the fuel storage facility design to prevent damage resulting from the SSE and to protect the fuel from mechanical damage.

6. Regulatory Guide 1.29 as related to the seismic design classification of facility components.

7. Regulatory Guide 1.92 as related to combination of loads for seismic analysis.

8. 10CFR20.

9. ASME Section III.

10. Branch Technical Position ASB 9-2 contained in the Standard Review Plan.

11. Light-Gage Cold-Formed Steel Design Manual, 1961 Edition, American Iron and Steel Institute.

12. 10CFR100

2104 128

4.0 MECHANICAL AND STRUCTURAL CONSIDERATIONS

The high density fuel storage system (HDFSS) module has been analyzed for both operating basis earthquake (OBE) and safe shutdown earthquake (SSE) conditions. A detailed stress analysis was then performed to check the design adequacy of the module against calculated loads. Results indicated that the HDFSS module design is adequate for the postulated combined loading conditions.

4.1 Seismic Analysis

The HDFSS module has been analyzed for both OBE and SSE conditions. Critical damping ratios of 2 percent were used in the analysis for the SSE condition and 1 percent for the OBE condition. The design floor acceleration response spectra are given in Figures 4-1 through 4-6. These spectra are based on Hatch 2 which bounds the spectra for Hatch 1. Combination of the modal response and the effect of the three components of an earthquake were performed in accordance with the applicable provisions of US NRC Regulatory Guide 1.92.

The seismic analysis was performed in several steps. First, the hydrodynamic effect, which represents the inertial properties of the fluid surrounding the submerged modules, was calculated to obtain the hydrodynamic virtual mass terms based on the module and pool configuration. Three-dimensional end effects and leakage between modules are accounted for by modifying the calculated hydrodynamic mass.

Figures 4-7 and 4-8 show the plan view of the two-dimensional model of the modules and pool used in the hydrodynamic virtual mass analysis. The model consisted of two rigid bodies: the modules and the pool walls. Water finite elements fill the spaces in between the walls and the modules. The total mass matrix of each module for the analysis is equal to its structural mass matrix plus the hydrodynamic mass matrix. Conservative structural damping values of 1 percent for the OBE and 2 percent for the SSE are applied without any added damping from fluid effects.

The WATER-01 computer program, GE-proprietary, was used to determine the hydrodynamic mass of one rectangular body inside another rectangular body. This program has been design reviewed and meets NRC-QA requirements. The methodology in calculating hydrodynamic mass has been presented in Reference 1.

Second, the derived total mass of the module was used to perform dynamic analysis for the OBE and SSE. As seen in Figure 4-9, for a typical 13 x 13 module, when the added-mass terms from the hydrodynamic mass effect were included, the fixed base frequency decreased.

Third, both finite-element and lumped-mass models of a module were then developed to provide a basis for selecting simplified module models to be used in the module and support system analysis and module sliding analysis. The finite-element model also was used to obtain the distribution of shear forces in the module plate elements.

Fourth, an eleven-node lumped-mass model was then developed by lumping the tributary module mass to the corresponding node point and initially selecting the stiffness properties based on beam theory. The stiffness properties of this model were based on matching the natural frequencies of the finite element model.

In the nonlinear analysis to calculate the amount of sliding and tilting, a two-node lumped mass model was found to adequately represent the module and support system analyses since the response to the module and support system was shown to be primarily a rigid body motion. Both the first mode and rigid body dynamic properties were simulated by this model. The effects of the corner supports were added to the model by including base springs and the final model was used in the sliding analysis. The horizontal spring represents the stiffness of the support pad and the vertical spring represents the stiffness of the fuel support plate, the foot pad, and the support pad.

The mechanism for controlling the shear force in each module is the limiting of the coefficient of friction between the module and the support pad by the selection of a non-galling, corrosion-resistant material with a low coefficient of friction to be used as the module foot pads which are in contact with the stainless-steel support pads. The range of friction coefficient for the selected materials was found to be between 0.145 and 0.203. The friction coefficient between the stainless-steel support pads and the stainless-steel liner is at least 0.349. This difference ensures that sliding will occur between the foot pad and the support pad, and not between the support pad and the floor liner.

The sliding analysis was done using the two-dimensional, non-linear DRAIN-2D and SEISM computer codes. DRAIN-2D was originally developed at the University of California at Berkeley; SEISM was developed by GE. Both computer codes have been design reviewed and meet NRC-QA requirements. Sliding and overturning of the module were studied for the SSE and OBE conditions. All of the modules were found to be stable under the worst postulated seismic loading conditions, and the minimum 2-inch clearance between modules precludes contact during a seismic event.

4.2 Stress Analysis

The HDFSS module has been designed to meet Seismic Category I requirements. Structural integrity of the rack has been demonstrated for the load combinations below using linear elastic design methods.

Analysis was based upon the criteria and assumptions contained in the following documents:

a. ASME Boiler and Pressure Vessel Code Section III, Subsection NF.

b. USNRC, Regulatory Guide 1.92, Combining Modal Responses and Spatial Components in Seismic Response Analysis

c. Hatch 2 Final Safety Analysis Report, Seismic Design Criteria.

OBE - Operating Basis Earthquake

SSE - Safe Shutdown Earthquake

d. Light-Gage Cold-Formed Steel Design Manual, 1961 Edition, American Iron and Steel Institute.

Acceptance criteria were based on:

a. Normal and upset (OBE) Appendix XVII, ASME, Section III.

b. Faulted (SSE) Paragraph F-1370, ASME Section III, Appendix F.

c. Local buckling stresses in the spent fuel storage tubes were calculated according to "Light-Gage Cold-Formed Steel Design Manual" of American Iron and Steel Institute in lieu of Appendix XVII, ASME, Section III, because of its applicability to these light-gage tubes. Only the strength of the outer wall thickness of 0.090 inch nominal is considered in the stress calculations.

The applied loads to the rack are:

a. Dead loads which are weight of rack and fuel assemblies, and hydrostatic loads.

b. Live loads - effect of lifting an empty rack during installation.

c. Thermal loads - the uniform thermal expansion caused by pool temperature changes from the pool water and stored fuel.

d. Seismic forces of OBE and SSE.

e. Accidental drop of a fuel assembly from the maximum possible height.

f. Postulated stuck fuel assembly causing an upward force of 1000 pounds.

The load combinations considered in the rack design are:

a. Live loads.

b. Dead loads plus OBE.

c. Dead loads plus SSE.

d. Dead loads plus fuel drop.

In accordance with ASME Section III, Subsection NF, Paragraph NF-3230, thermal stresses are not considered. Furthermore, thermal loads were not included in combinations because the design of the rack makes them negligible; i.e., the rack is not attached to the structure and is free to expand or contract under pool temperature changes.

Stress analyses were done for both OBE and SSE conditions, based upon the shears and moments developed in the finite-element dynamic analysis of the seismic response. These values were compared with allowable stresses referenced in ASME Section III, Subsection NF (Table 4-1). Values given in Table 4-1 are based on the limiting module size. Stresses for the other module configurations are lower, and therefore, are not given here. Additional analyses were then performed to determine the dynamic frequencies, earthquake loading reactions, and internal forces in critical module and support system locations. Those values are summarized in Table 4-2.

The force path in the module caused by a horizontal earthquake is shown schematically in Figure 4-10. This figure shows the path of the horizontally induced earthquake fuel element inertial forces from the fuel element to the module support pads. Part of the fuel bundle inertial forces induced by the motion of the module are transferred either through the water or directly to the tube walls perpendicular to the direction of motion (Point 1 in Figure 4-10). These walls then transfer the forces to the side tube walls, which carry the forces down the walls and into the fuel support plates (Point 2). The portion of the fuel bundle load which is not transferred to the fuel tube walls is transferred directly to the fuel support plate at the point where the lower end fitting of the fuel bundle is supported vertically (Point 3). The fuel support plates, acting as a relatively rigid diaphragm, transfer the in-plane shear forces to the long casting which then transfers the shear forces to the module base assembly plate (Point 4). The forces are carried in the module base assembly (Point 5) until they are ultimately transferred to the foot pad and to the support pad and the pool slab (Point 6).

The vertical forces caused by earthquake and gravity loads become axial forces in the foot pads. The critical location for the compression forces from the foot pads is in the long castings and tubes directly above the foot pads. For stress analysis purpose, these compression forces are considered to be resisted by four fuel tubes sitting directly above the support pad.

Fuel assembly drop accidents were analyzed. The results are summarized in Table 4-3. The HDFSS design does not require any different fuel handling procedures from those discussed in the Unit 1 and Unit 2 FSAR.

The loads experienced under a stuck fuel assembly condition are less than those calculated for the seismic condition and have therefore not been included as a load combination.

2104 132

TABLE 4-1

Comparison of Calculated Stress vs. Allowables (psi)

Location/Type	OBE Condition		SSE Condition	
	Calc Stress	Allowables[1]	Calc Stress	Allowables[1]
Tube wall bending	Will be pro-	20,630	Will be pro-	41,250
Tube wall shear	vided by	11,000	provided by	22,000
Tube wall tension	July 31, 1979	14,880	July 31, 1979	29,760
Tube weld throat shear		11,000		22,000
Angle, weld throat shear		11,000		22,000
Casting bending		20,630		41,250
Casting wall shear		11,000		22,000
Casting wall compression		16,500		33,000
Fuel support plate bending		20,630		41,250
Support plate weld throat bending		20,630		41,250
Closure plate bending		20,630		41,250
Closure plate shear		11,000		22,000
Closure plate weld bending		20,630		41,250
Closure plate weld shear		11,000		22,000
Corner tube local compressive - stress check for local buckling		-		17,224

[1]Allowable stresses referenced in ASME Section III, Subsection NF

2104 133

TABLE 4-2

Will be provided by July 31, 1979

Table 4-3

High Density Spent Fuel Storage System Assembly Drop Accident

Case Summary

No.	Case Description	Effect on Reactivity
1.	A fuel assembly drops 27 inches vertically and impacts the top of a fully loaded HDFSS module. The dropped assembly comes to rest horizontally on top of the HDFSS.	Analysis indicates that localized tube damage or fuel support member damage will occur, but neutron absorber material will not be removed from its position between adjacent fuel assemblies. A fuel assembly resting horizontally atop the HDFSS does not increase the system reactivity because the reactivity assumes an infinite vertical length of fuel (no neutron leakage in the vertical dimension). $k_{eff} < 0.90$
2.	A fuel assembly drops from 27 inches above the HDFSS, enters an empty storage position, and falls to the bottom of the storage position.	Structural analysis indicates that localized tube damage will occur and one neutron absorber plate may be damaged. A reactivity analysis of this case, with the neutron absorber plate between two fuel assemblies totally missing, shows that k_{eff} remains less than 0.90.
3.	A fuel assmebly drops from 27 inches above the HDFSS and strikes a tube wall at an oblique angle.	Same as Case 2
4.	A fuel assembly drops from 27 inches above the top of a fully loaded module and strikes the upper tie plates of 2, 3, or 4 fuel assemblies in storage.	It is not possible for a fuel assembly drop of 27 inches to drive four stored assemblies through the bottom of the module. Even so, the reactivity effect of this postulated event was calculated as a limiting value. An 18-inch section of fuel in four bundles in an unpoisoned square array was found to have a k_{eff} approximately equal to that of the system. There would be no increase in the overall reactivity $k_{eff} < 0.90$.
5.	A fuel assembly drops from 27 inches above the HDFSS, falls outside of the loaded HDFSS, and lodges adjacent and parallel to an unpoisoned, occupied fuel storage position.	This case was analyzed for normal handling conditions; $k_{eff} < 0.90$.

2104 135

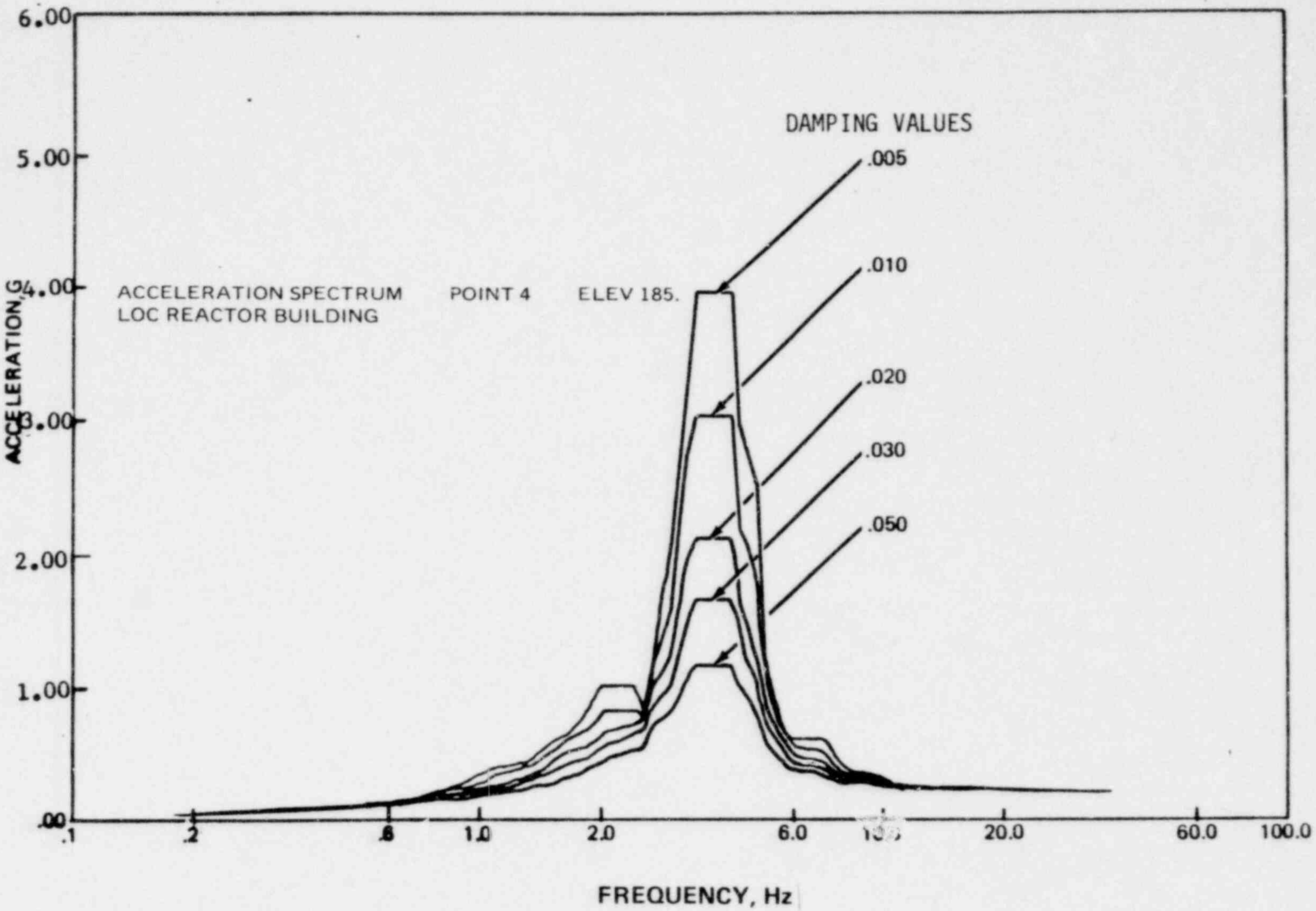


Figure 4-1 Reactor Building North-South OBE CE 921 Analysis

2104 136

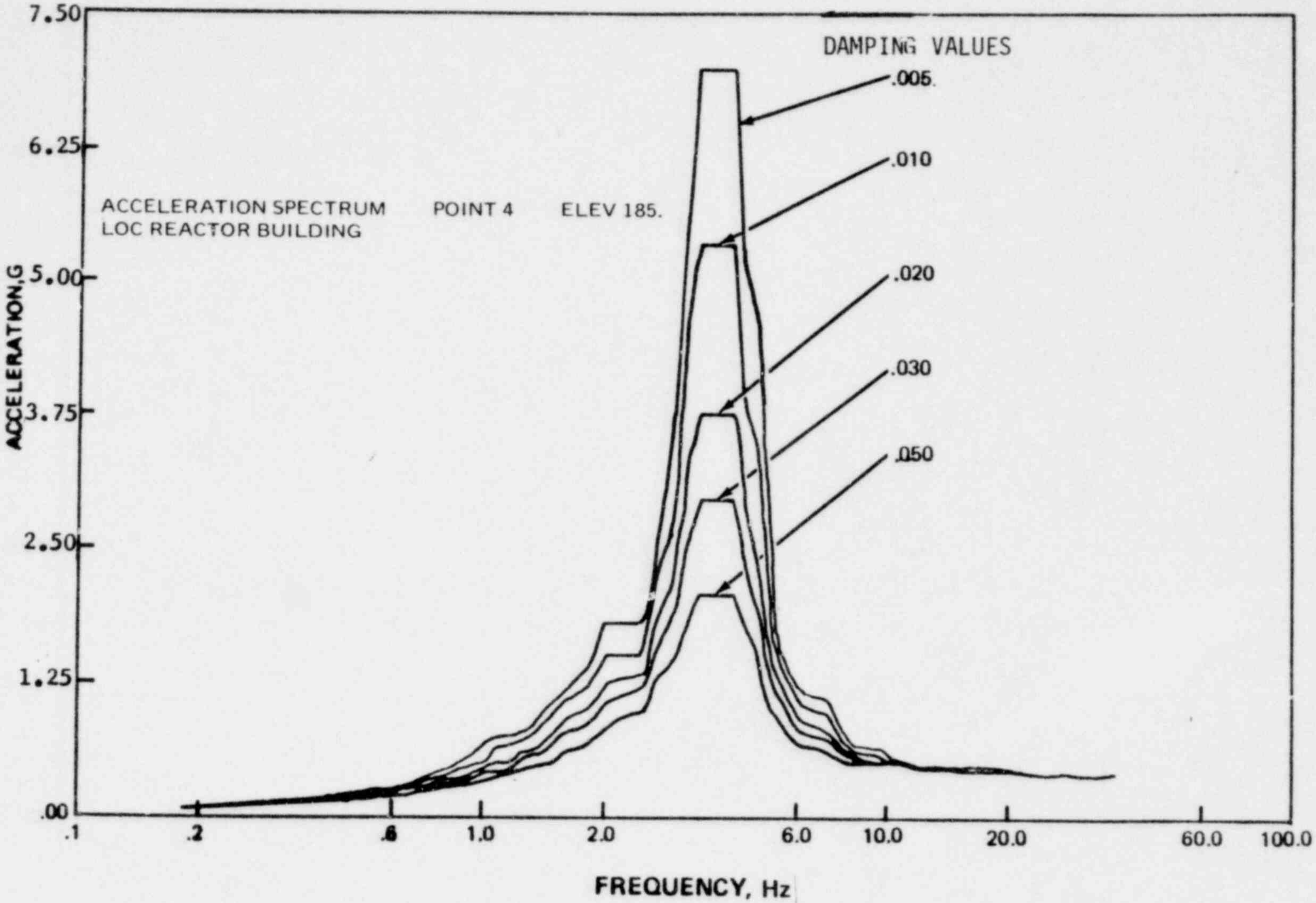


Figure 4-2 Reactor Building North-South SSE CE 921 Analysis

2104 137

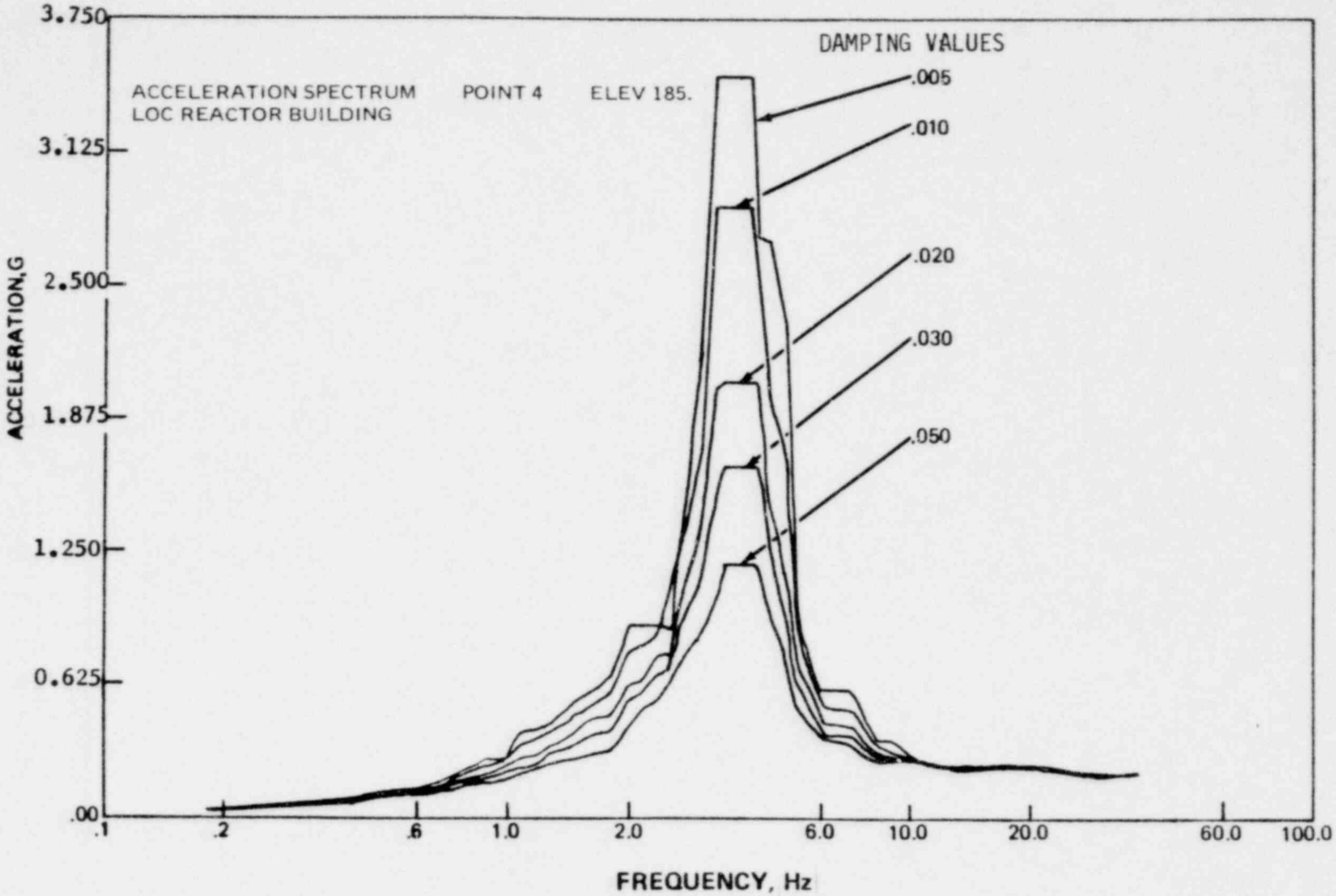
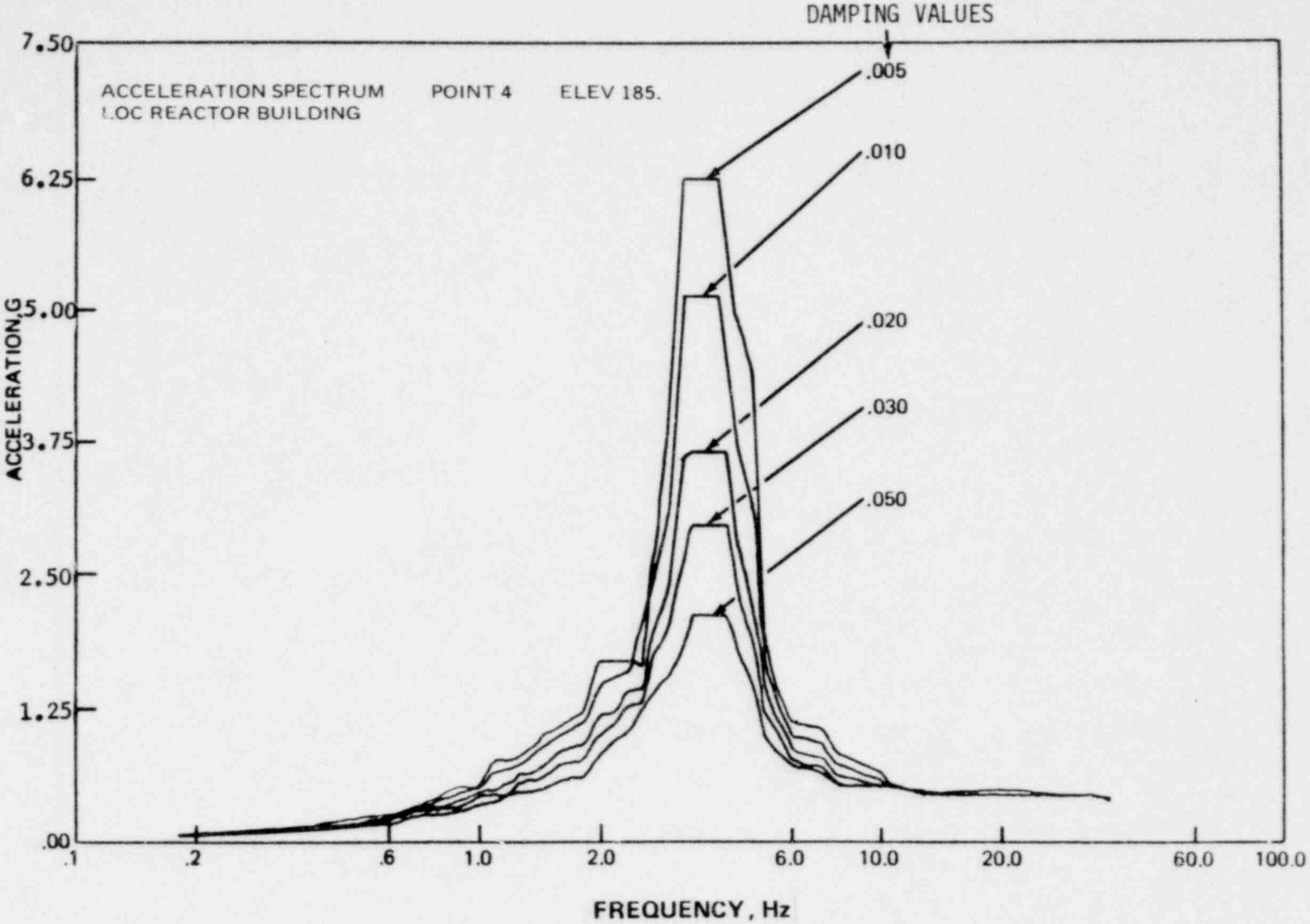


Figure 4-3 Reactor Building East-West OBE CE 921 Analysis

2104 138



2104 139

Figure 4-4 Reactor Building East-West SSE CE 921 Analysis

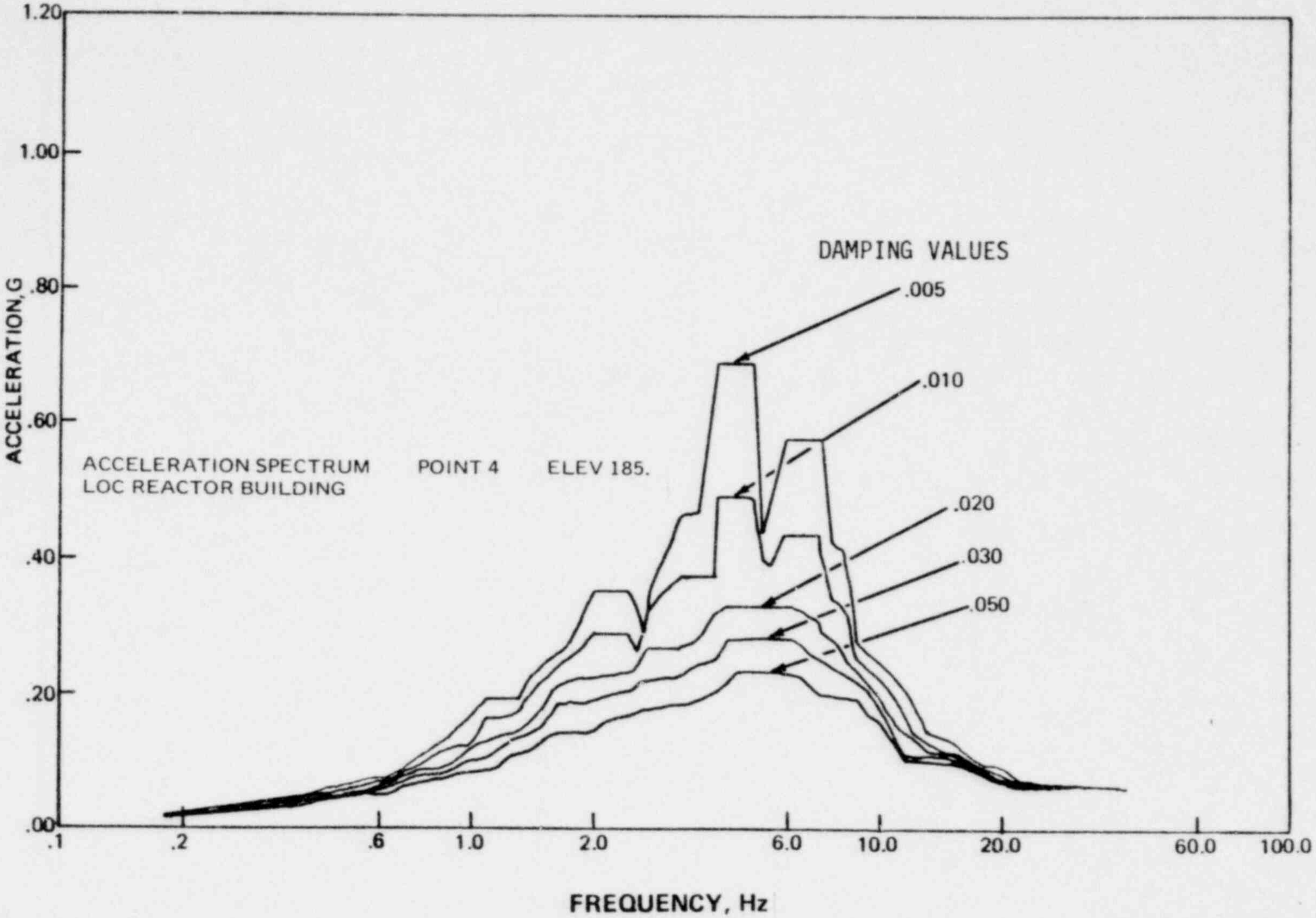
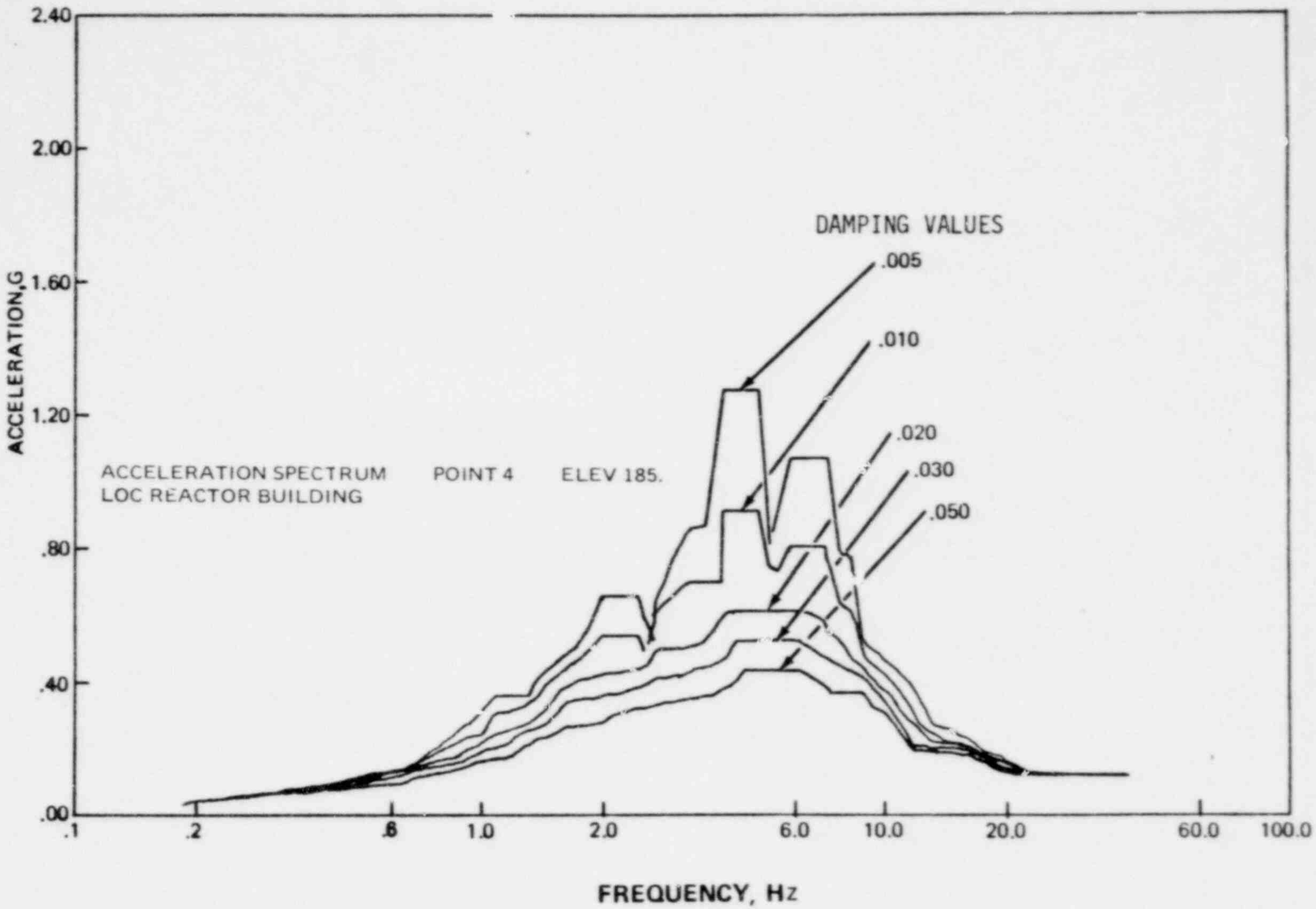


Figure 4-5 Reactor Building Vertical OBE CE 921 Analysis

2104 140



2104 141

Figure 4-6 Reactor Building Vertical SSE CE 921 Analysis

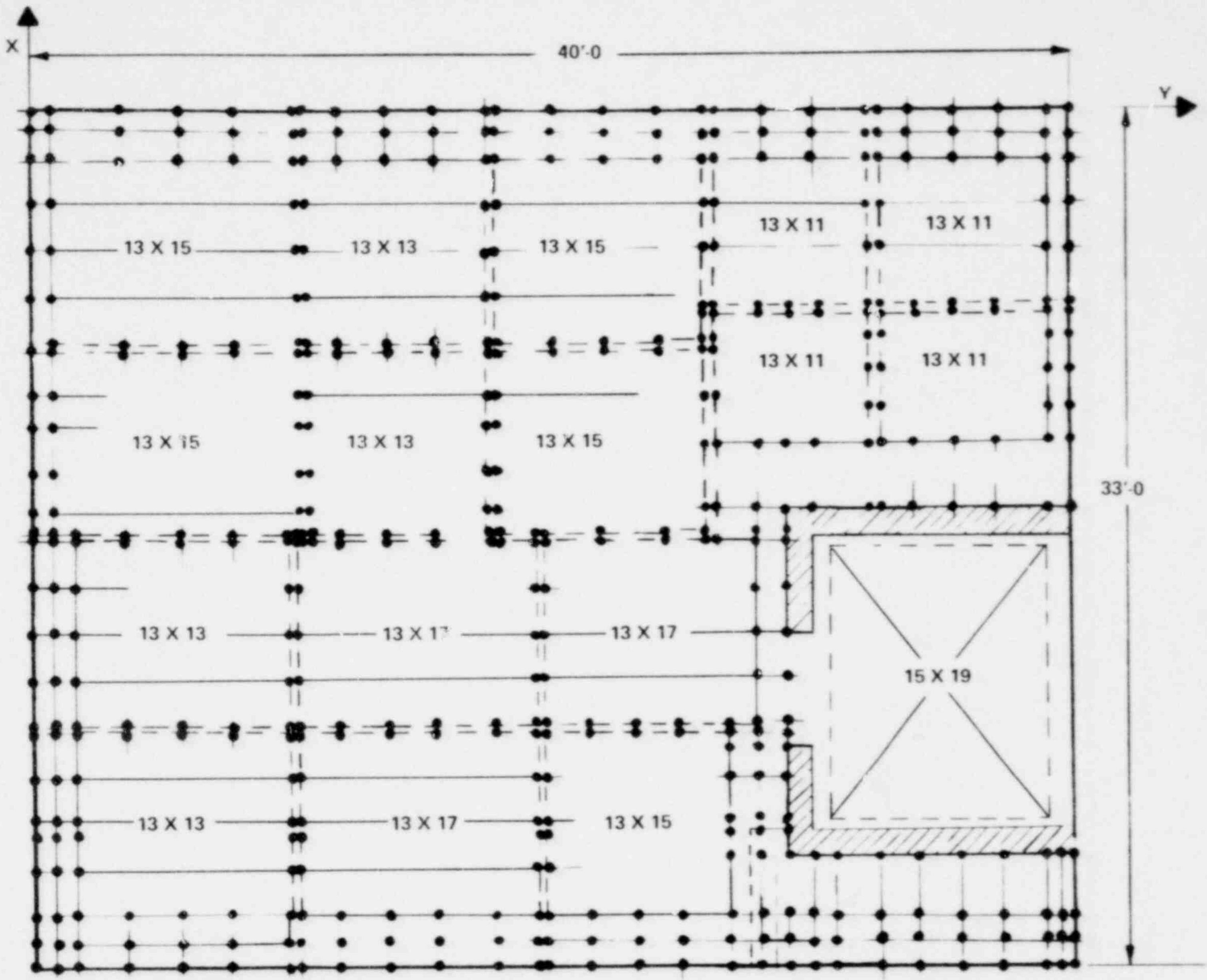
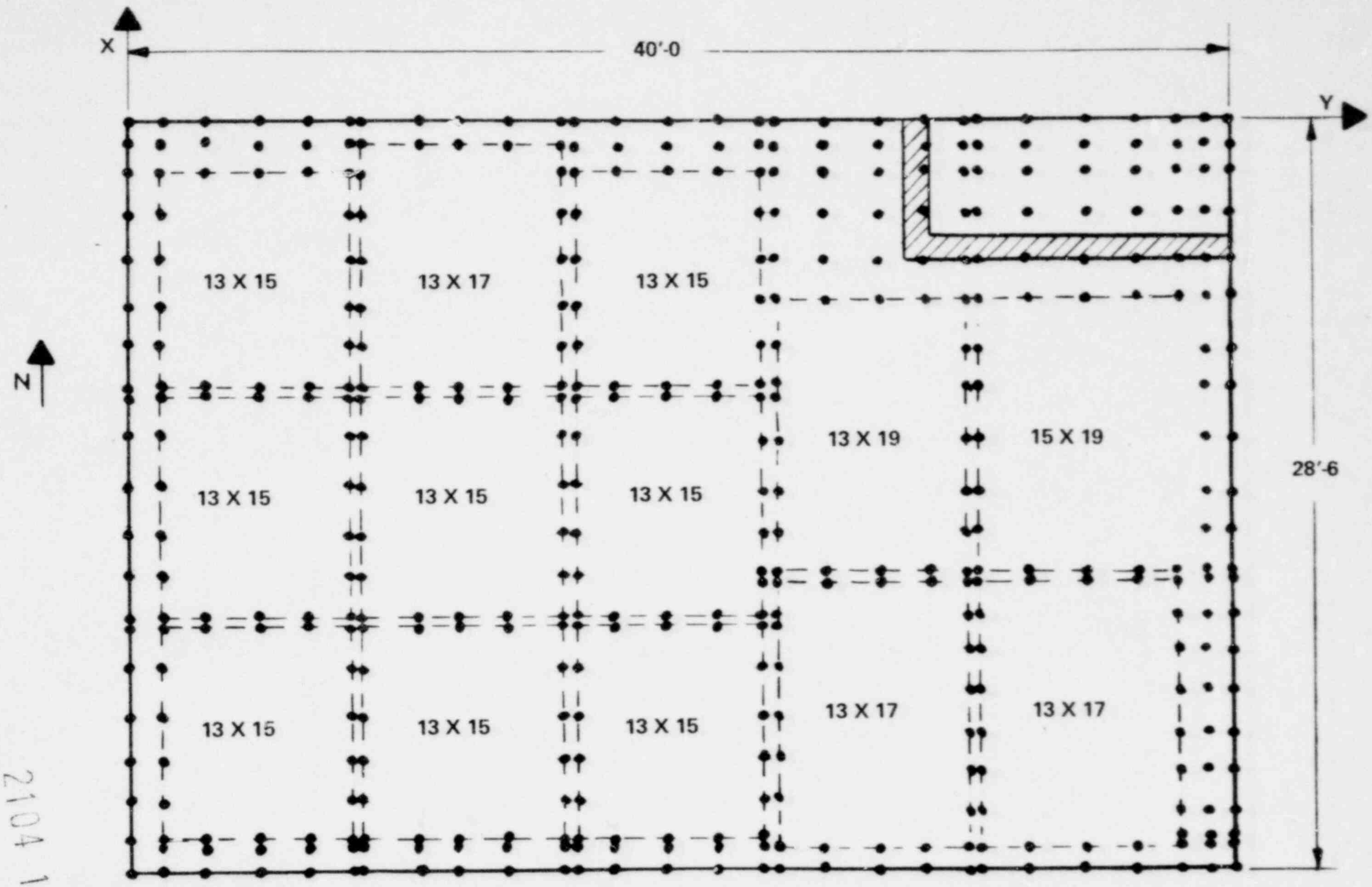


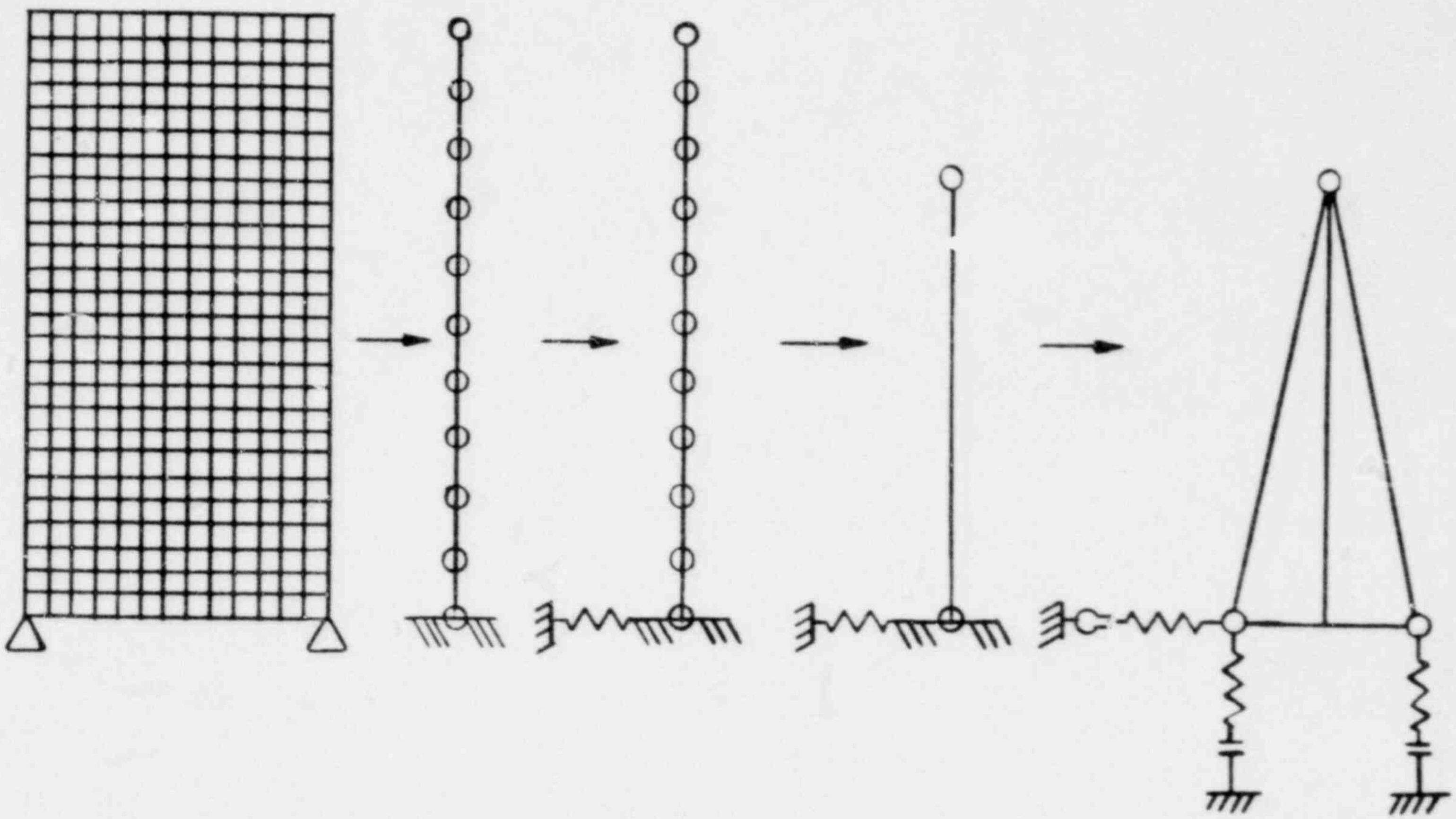
Figure 4-7 Modules and Pool Model for Hydrodynamic Virtual Mass Analysis Hatch 1

2104 142



2104 143

Figure 4-8 Modules and Pool Model for Hydrodynamic Virtual Mass Analysis Hatch 2 f* = 16.0 Hz | f = 15.8 Hz | f = 11.7 Hz | f = 11.8 Hz | f = 9.7 Hz



Finite Element Interior Section Without Added Mass | Fixed-Base 11 Lumped-Mass Model Without Added Mass | Fixed-Base 11 Lumped-Mass Model With Added Mass | Fixed-Base 2 Lumped-Mass Model With Added Mass | Module with Support System

*f = Fundamental frequency

Figure 4-9 Typical 13 x 13 Sequence of Module Modeling

2104 144

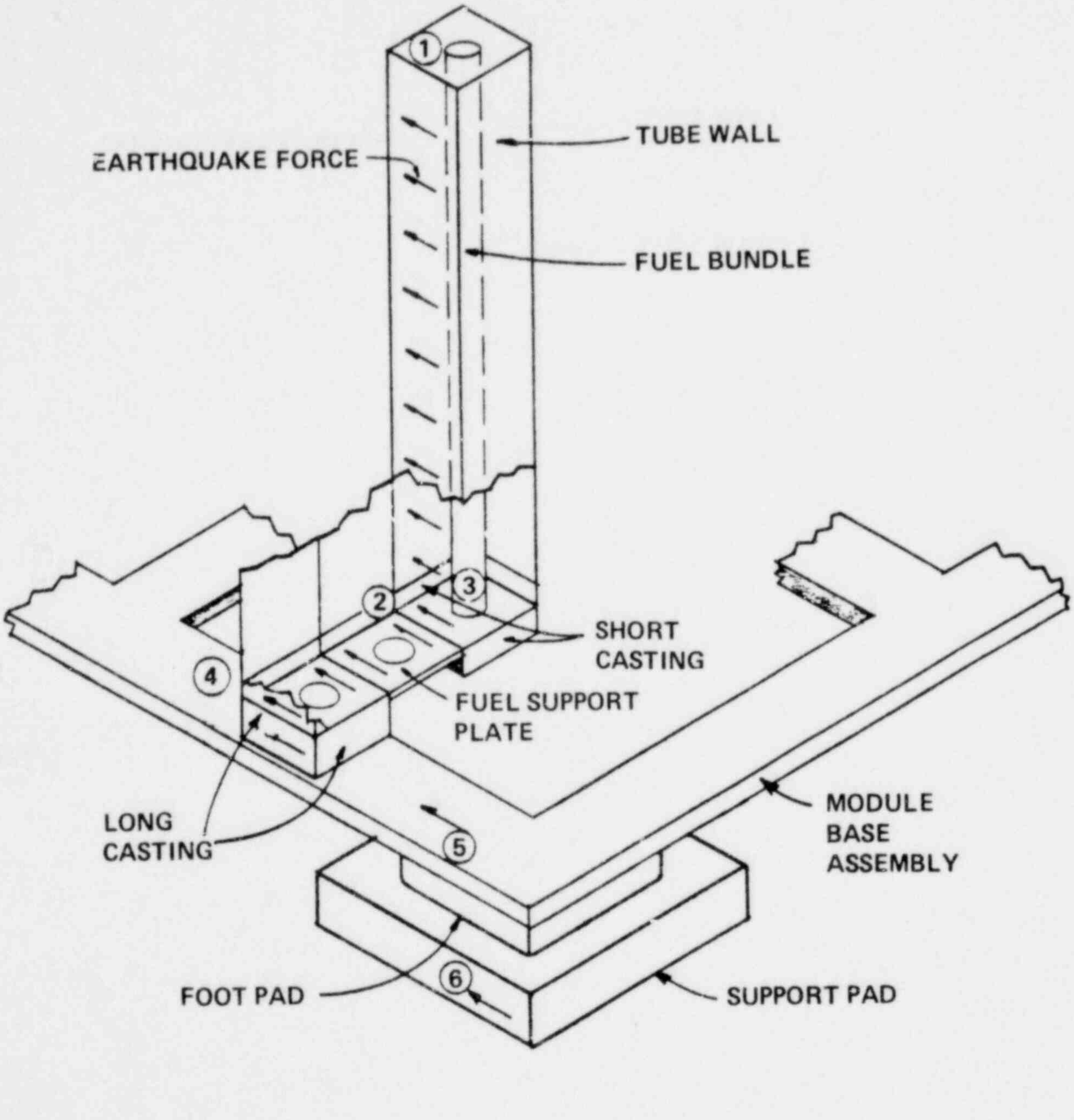


Figure 4-10 Path of Earthquake Horizontal Forces in Module

5.0 MATERIAL CONSIDERATIONS

Most of the structural material used in fabrication of the new HDFSS is type 304 stainless steel. This material was chosen because of its corrosion resistance and its ability to be formed and welded with consistent quality. The only structural material employed in the structure that is not 304 stainless steel is a special low-friction material used as a foot pad between the module and the support pad. Boral plates, used as a neutron absorber, are an integral non-structural part of the basic fuel storage tube. These plates are sandwiched between the inner and outer wall of the storage tube and are not subject to dislocation, deterioration, or removal. The inner and outer walls of the storage tube are welded together at each end for mechanical rigidity. Small openings are formed in the top and bottom of each tube assembly by leaving gaps in the weld to allow for the venting of the envelope between the inner and outer tube walls. At normal pool water operating temperature there is no significant deterioration or corrosion of stainless steel or Boral.

Specifications were developed specifically for the HDFSS which impose quality control requirements during the design, procurement, fabrication, installation, and testing of the storage system. Periodic audits of the various facilities and practices are performed by certified quality assurance personnel to ensure that these QA/QC requirements are being met. All welding and nondestructive examination (NDE) is done in accordance with the applicable provisions of the ASME Boiler & Pressure Vessel Code, Section IX, and the American Society for Nondestructive Testing (ASNT).

Storage module components are assembled and welded in special fixtures to maintain close control of dimensional tolerances. Each storage position is checked with full length gauges to assure proper clearance between stored fuel bundles and storage tube walls.

To provide assurance that specification Boral sheet is used in tube fabrication, a special quality control program is in effect at the manufacturer's facility. The concentration and distribution of the neutron absorbing material (B_4C) are verified by chemical analyses and/or neutron transmission tests, and each sheet is dimensionally inspected. Before each piece of Boral is inserted into a tube assembly successful performance of the required inspections is verified.

The presence of the neutron absorber material in the fabricated fuel storage module will be verified at the reactor storage pool site by scanning each storage tube in the modules with a neutron source and neutron detectors. The recorded results provide a comparison between neutron absorption through each Boral sheet and neutron absorption measured through the stainless steel without Boral. A significant increase in neutron absorption verifies the presence of Boral.

2104 146

Boral's corrosion resistance is similar to that of standard aluminum sheet. Corrosion data and industrial experience confirm that aluminum and Boral are acceptable (Reference 2) for the proposed application. Although experience indicates that it is unnecessary, an inservice test program will be conducted, consisting of periodically removing and examining samples of Boral plate which will be suspended in the storage pool.

Pool water quality will be maintained as specified in the Hatch 2 FSAR, Section 9.1.3.2.4. No changes to water quality are expected as a result of the planned modification to the spent fuel storage capacity (see Section 10-1 of the Radiological Evaluation).

2104 147

6.0 INSTALLATION

The HDFSS modules are a free-standing, bottom-supported design, resting on support pads placed onto the floor of the fuel storage pool. The installation program will consist of removing the low-density aluminum racks in the pools, placing the new support pads into prescribed positions, and lowering the new modules into position on their respective support pads.

The initial installation will be in the Hatch 2 pool with the pool wet or dry. Special load-tested lifting fixtures, designed with a minimum safety factor of 3, are used to handle the support pads and the storage modules to minimize dropping any materials. The single-failure-proof reactor building crane will be used to remove the old racks and to lower the new equipment into place.

The Hatch 1 pool, which is filled with water and contains spent fuel, will be reracked similarly after the initial installation of modules in Hatch 2 has been completed. Stored fuel may be transferred to the Hatch 2 pool through the transfer canal to empty the Hatch 1 pool, or may be concentrated away from the rerack work locations. The sequence of the rerack work will be such that no heavy equipment will be transferred over stored spent fuel at any time. The installation equipment is designed to allow installation of modules and pads into a water-filled pool. Following the installation, verification of neutron absorbers will be completed.

2104 148

7.0 NUCLEAR CONSIDERATIONS

7.1 Neutron Multiplication Factor

The criticality analysis calculations were performed with the MERIT (Reference 3) computer program, a Monte Carlo program which solves the neutron transport equation as an eigenvalue or a fixed source problem including the effects of neutron shielding. This program is especially written for the analysis of fuel lattices in thermal nuclear reactors. A geometry of up to three space dimensions and neutron energies between 0 and 10 MeV can be handled. MERIT uses cross sections processed from the ENDF/B-IV library tapes.

The qualification of the MERIT program rests upon extensive qualification studies including Cross Section Evaluation Work Group (CSEWG) thermal reactor benchmarks (TRX-1, -2, -3, -4) and B&W UO_2 and PuO_2 criticals, Jersey Central experiments, CSEWG fast reactor benchmarks (GODIVA, JEZEBEL), the KRITZ experiments, and in addition, comparison with alternate calculational methods. Boron was used as solute in the moderator in the B&W UO_2 criticals, and as a solid control curtain in the Jersey Central experiments. The MERIT qualification program has established a bias of 0.005 ± 0.002 (1σ) Δk with respect to the above critical experiments. Therefore, MERIT underpredicts κ_{eff} by approximately 0.5 percent Δk.

The storage space (cell) infinite multiplication factor (k_∞) was calculated for the high density fuel storage system as defined by the assumptions below and the exact geometry specifications.

7.2 Input Parameters

a. Standard BWR fuel configurations

b. Maximum BWR fuel bundle multiplication factor (k_∞) of 1.35 in standard-core geometry at 20°C. The use of a maximum fuel k_∞ as a criticality base eliminates the need to analyze the multiplicity of U^{235} enrichment and burnable poison combinations.

c. Storage space pitch of 6.563 in.

d. Minimum allowable boron (B^{10}) concentration equivalent to a homogeneous areal concentration of 0.013 grams B^{10}/cm^2.

e. Analysis conservatively performed using 2-dimensional infinite lattice (X,Y) model (no credit taken for axial or radial neutron leakage).

f. Credit taken for double wall stainless steel tubes that separate fuel bundles.

The results of the calculations for several cases are in Table 7-1. The model geometry, bias, and uncertainity for each of the cases is described below.

7.3 Geometry, Bias, and Uncertainty

The repeating cell geometry in Figure 7-1 is the exact geometry model, with the exception of squared corners, used in cases 1, 2 and 3 of Table 7-1. This model has the minimum allowable corner gap (storage cells touching), using the nominal dimensions shown in Figure 7-2. No geometry bias is associated with this model. The MERIT program bias is 0.005 ± .002 (1σ) Δk.

The same basic geometry model, but with the maximum axial average gap as shown in Figures 7-2 and 7-3, was used for case 4 of Table 7-1. The pitch was increased to 6.8324 in., resulting in a gap spacing of 0.381 in. Note that this gap can occur only along one diagonal of the module with all storage tubes bowed at a maximum. This model has the same bias as the above; i.e., no geometry bias and MERIT program bias of 0.005 ± 0.002 (1σ) Δk.

An approximate geometry model, shown in Figure 7-4, was used for case 5 in Table 7-1. The model geometry bias relative to the exact model for the same conditions was 0.0087 ± 0.0050 (1σ) Δk. The MERIT program bias remains the same at 0.005 ± 0.002 (1σ) Δk. In all cases the reported value of k_∞ includes the sum of all biases and the root-mean-square of the uncertainties.

The maximum k_∞ of a storage cell occurs at 20°C with the fuel bundles centered and no flow channels present. Any variation, such as increasing the cell pitch, eccentric bundle positioning, reducing moderator density, and increasing the temperature to 65°C decreases the K_∞. Table 7-2 shows the maximum k_∞ of the storage cell broken down into contributing bias and uncertainty values.

The sensitivity of the cell k_∞ to decreasing moderator density is shown graphically in Figure 7-5. Since the cell is under-moderated, the optimum k_∞ occurs at 1.0 g/cc.

The design of the HDFSS has alternating spaces on the periphery of each module fabricated with unpoisoned closure plates. The unpoisoned locations are also directly opposite each other on adjacent modules. The effect of the partially unpoisoned storage locations is small and insensitive to the inter-module water gap, as shown in Table 7-3. The maximum module k_∞ occurs at the minimum possible water gap (1.244 in.) and is less than that of an infinite array of storage cells with no water gap.

The module calculations in Table 7-3, were done with the model shown in Figure 7-6. Some of the details in the exact model were homogenized and simplified to reduce the input preparation in the module calculations. The model geometry bias was determined from an infinite array of simplified storage cells (Figure 7-7) relative to the exact geometry model. The module geometry model bias was determined to be 0.0017 ± 0.0051 (1σ) Δk. The same MERIT program bias applies.

For all calculations the fuel bundle was discretely represented by fuel pellets, cladding, water rods, channels (when present), and fuel rod enrichment and burnable poison distributions within the bundle. Fuel pin spacers were not included (a conservative exclusion). The nominal bundle dimensions were used for all cases.

The HDFSS includes defective fuel storage spaces attached externally to some of the storage modules. The geometric layout is shown in Figures 2-11 and 2-12. Analyses have demonstrated the HDFSS $k_{eff}<0.95$ with all defective fuel storage locations occupied with fuel.

The sensitivity of k_∞ analyses to various changing parameters is implied above. More specific relationships are as follows:

a. Bundle Reactivity (percent U^{235}) - Calculations are based on maximum k_∞ thereby obviating enrichment sensitivity considerations.

b. Stainless steel thickness - Neutron absorption by the two layers of stainless steel comprising the storage tube was included in the criticality calculations using the nominal thicknesses of 0.0355 and 0.090 inch for the inner and outer tubes, respectively. The nominal inner tube thickness has been reduced to 0.0300 inch, and Monte Carlo calculations shown that the change in k_∞ is within the statistical uncertainty of the calculation (Case 2, Table 7-1).

c. Water density - Figure 7-5 shows the variation of k_∞ with moderator (water) density. Since the cell is under-moderated the optimum k_∞ occurs at 1.0 g/cc.

d. Storage lattice pitch - An analysis was done using a minimum fuel pitch, represented by the storage tubes touching. Material tolerances in the tubes were taken to maximize the k_∞ of the storage lattice. The result of this analysis is given as Case 6 in Table 7-1. The results in Table 7-1 show that the nominal pitch (Case 2) has a higher k_∞ result than the minimum pitch case (Case 6).

e. The HDFSS and the BWR fuel to be stored in it are designed and fabricated to prevent significant quantities of air or other gas from being entrapped. Thus, no areas of reduced effective moderator density are created. But even if air were trapped, the effect of reduced density on the under-moderated fuel bundles is to reduce the k_{eff} of the system.

7.4 Postulated Accidents

Several fuel assembly drop accidents have been analyzed. The results are summarized in Table 4-3. The handling of heavy objects in the spent fuel pool area is addressed in Section 11.0 of the accident evalution.

A tornado-generated missile model has been used for the Hatch spent fuel pools (refer to the response to Question 130.19 in the Unit 2 FSAR) that could result in impacting the storage module. The angles in the structural grill system associated with the reactor building tornado relief vent openings have been postulated as a secondary missile source resulting from impact of a plank missile. A maximum of three angles could be generated as secondary missiles with a maximum energy of 2000 ft-lb each. Analysis shows that the HDFSS module can withstand such impact energy without resulting in a nuclear hazard.

Loss of all cooling in the spent fuel pool, resulting in boiling of the pool water, is an accident that has been analyzed in Section 8.4. The effect of such boiling on the undermoderated fuel bundle is to reduce the system k_{eff}. No criticality accident will result.

The fuel storage module design has been evaluated for the acceptability of stresses from several combined loads, including earthquake-induced loads, as discussed in Section 4.2. Resultant stresses are within allowable limits, assuring the integrity of the modules under the combined loading. This precludes a criticality accident resulting from an earthquake.

2104 152

TABLE 7-1

Single Cell High-Density Fuel Storage Criticality Results

Case	Description	$K_\infty (\pm 2\sigma)$ [1]
1	Nominal Rack Dimensions [2] With Flow Channel @20°C	0.8668 ± 0.0075
2	Nominal Rack Dimensions Without Flow Channel @20°C	0.8674 ± 0.0086
3	Same as Case 2 except @65°C	0.8561 ± 0.0084
4	Increased Pitch without Flow Channel @20°C	0.8364 ± 0.0106
5	Same as Case 2 but with Eccentric Bundle Position	0.8276 ± 0.0123
6	Minimum Pitch [3] without Flow Channel @20°C	0.8650 ± 0.0088

---

[1] k_∞ includes MERIT Program Bias and Uncertainty

[2] 6.563 inch Pitch with Nominal Material Thickness

[3] 6.503 inch Pitch with Minimum Storage Tube Material Tolerances to Maximize k_∞

2104 153

TABLE 7-2

Bias and Uncertainty Components for Maximum k_∞ of a Storage Cell

k_∞		0.8624		
Calculational Convergence	Δk		$\pm$	0.0038
MERIT Bias and Uncertainty	Δk	0.005	$\pm$	0.002
Model Bias and Uncertainty	Δk	None		
Total		0.8674	$\pm$	0.0086 (2σ)

2104 154

TABLE 7-3

HDFSS Criticality Analysis

Module Interaction

Description	k_∞ ($\pm 2\sigma$)
Minimum gap between modules (2A = 1.244 in.)	0.8593 $\pm$ 0.0131
Intermediate gap between modules (2A = 2.100 in.)	0.8579 $\pm$ 0.0130
Nominal gap between modules (2A = 2.967 in.)	0.8506 $\pm$ 0.0134

2104 155

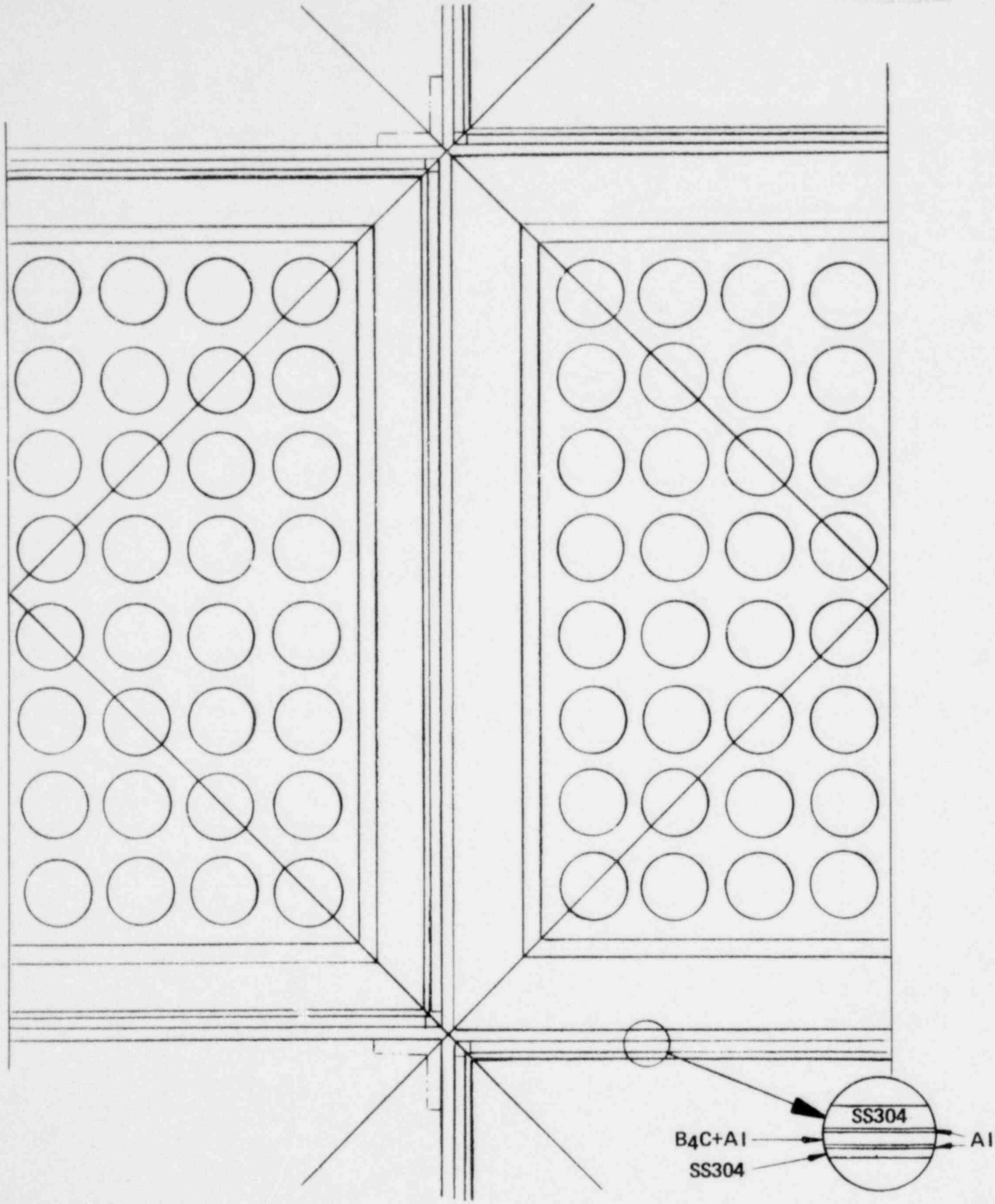
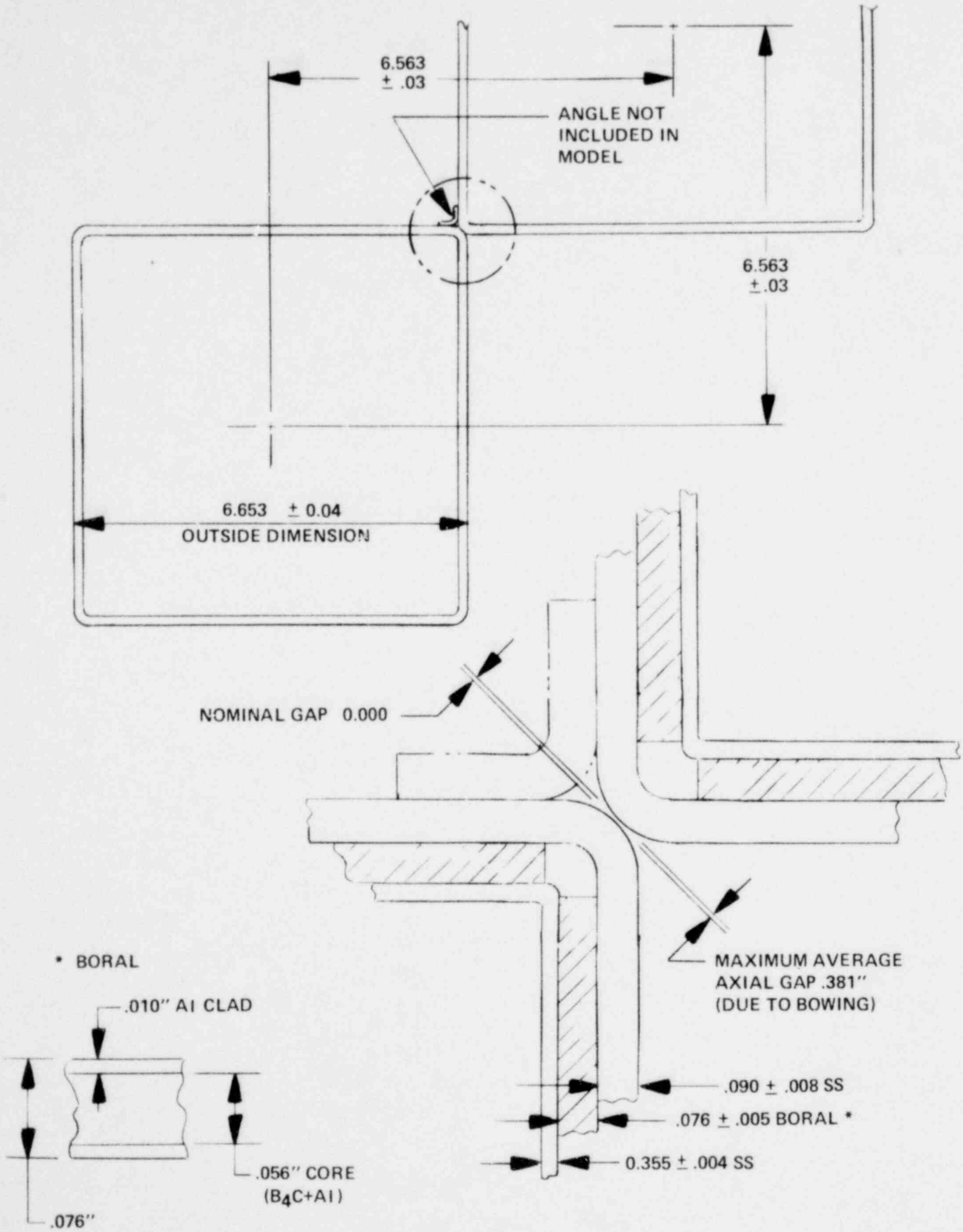


Figure 7-1 Exact Geometry Model

2104 156



2104 157

Figure 7-2 Storage Cell Dimensions

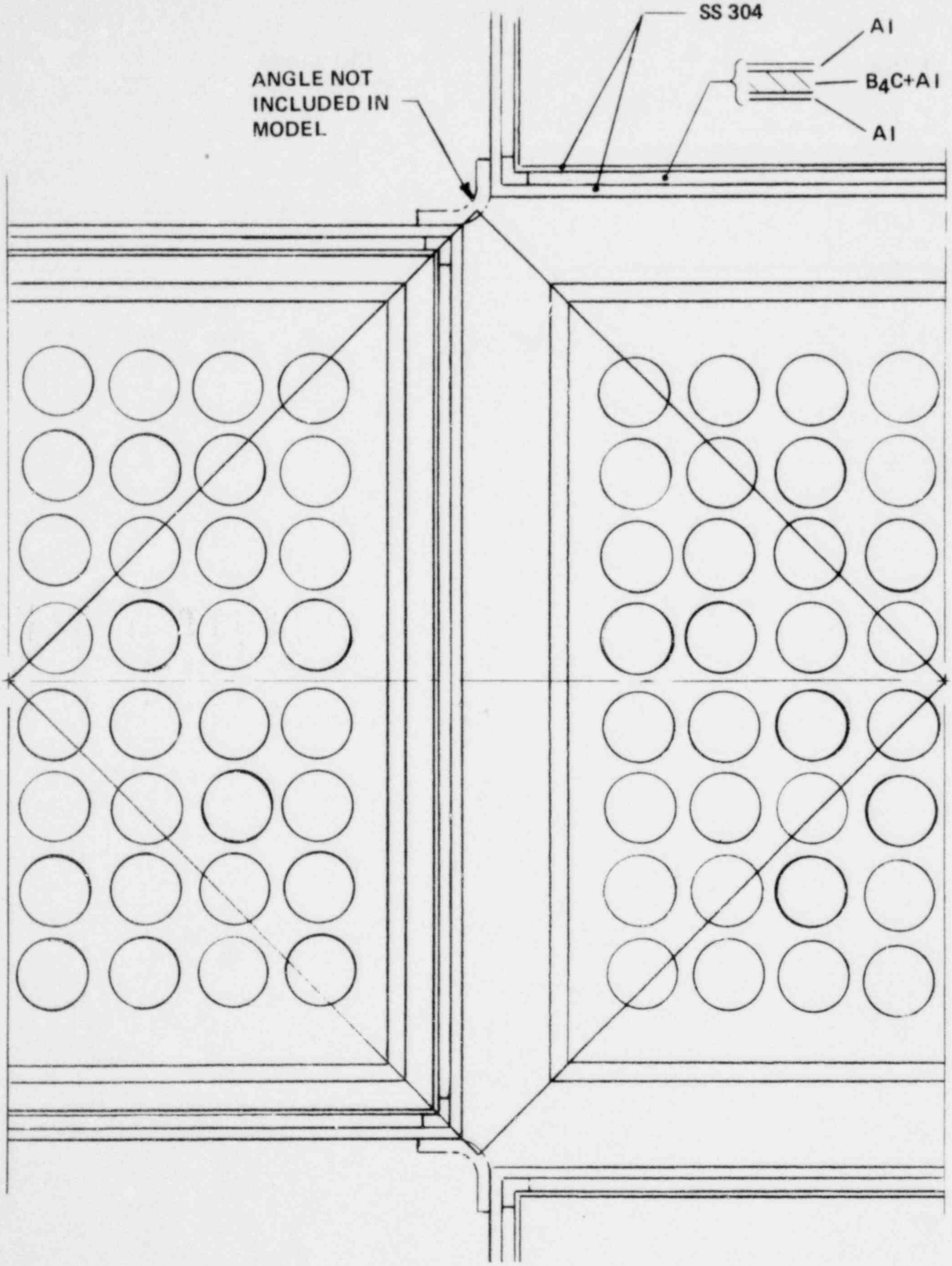
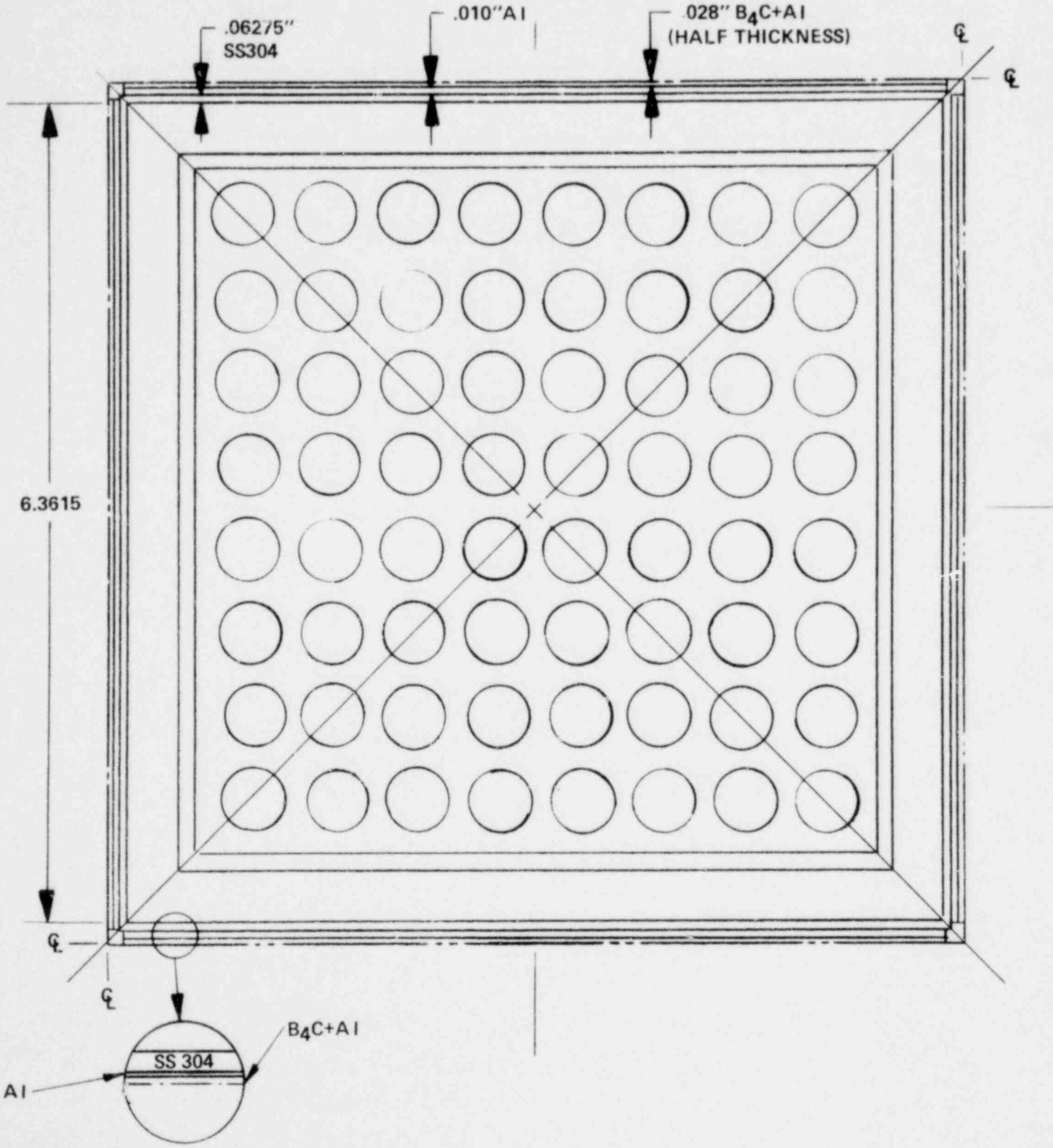


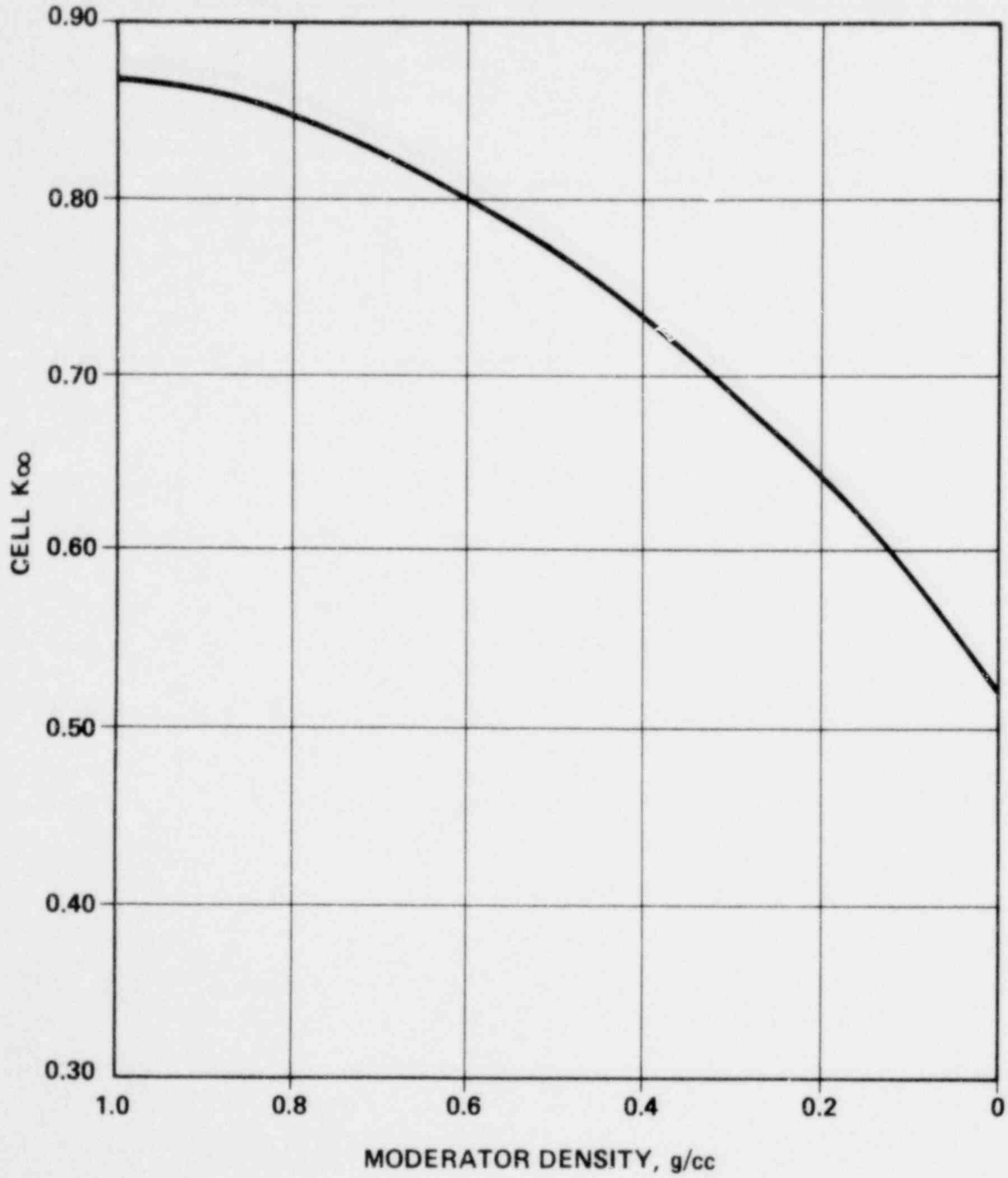
Figure 7-3 Wide Gap-Exact Geometry

2104 158



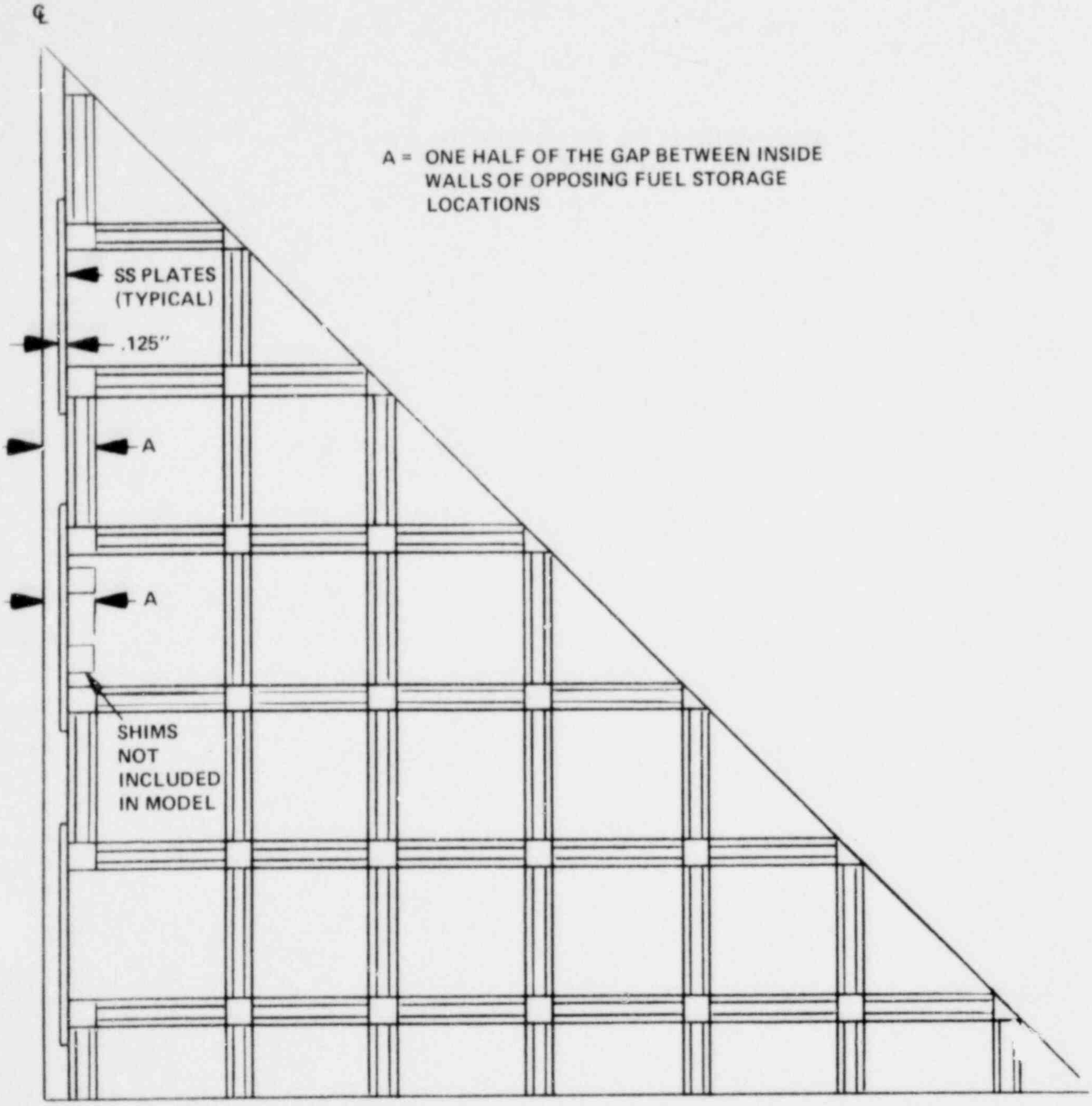
2104 159

Figure 7-4 Approximate Cell Model



2104 160

Figure 7-5 Cell K∞ as a Function of Moderator Density



2104 161

Figure 7-6 Module Model

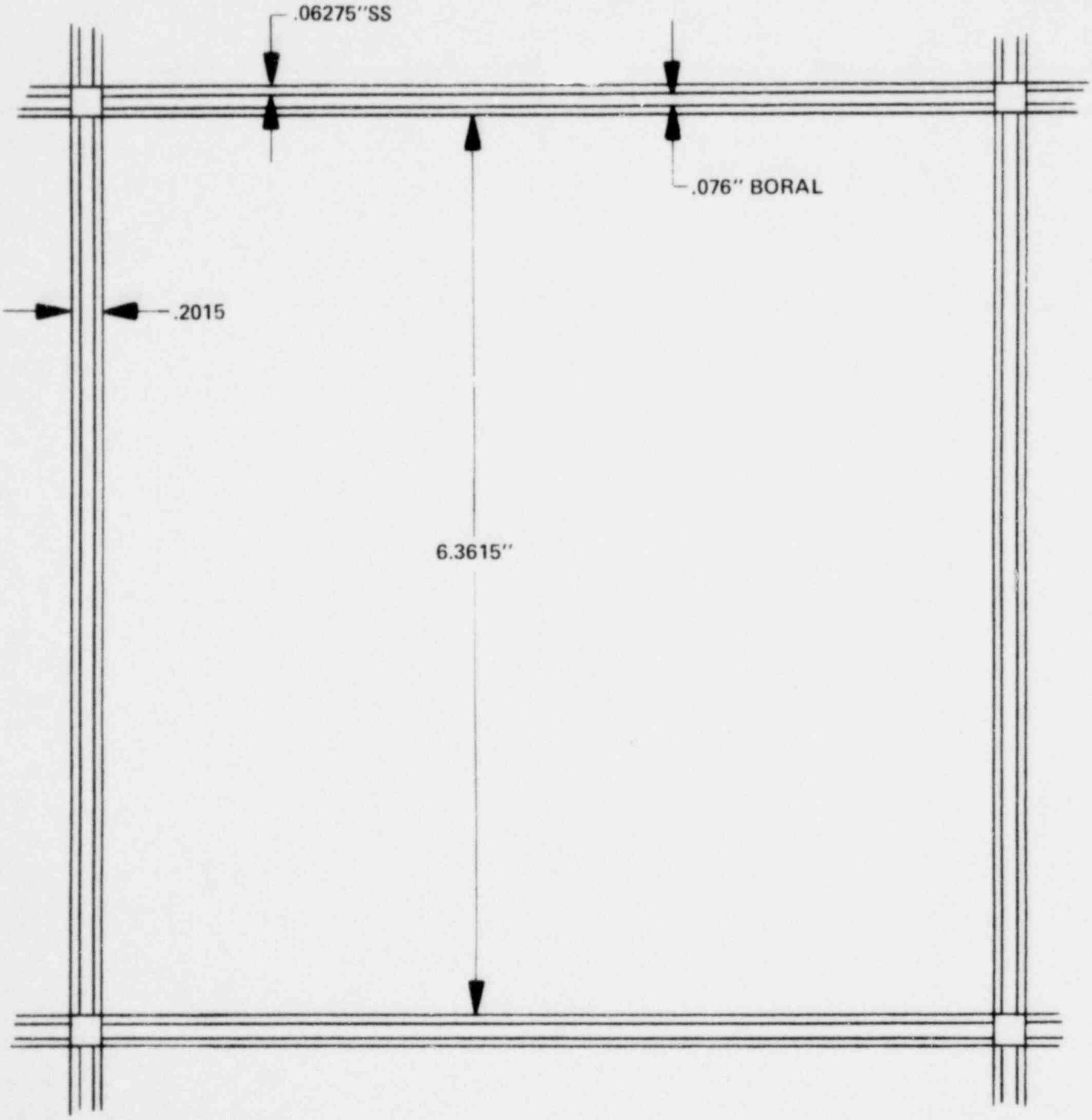


Figure 7-7 Simplified Cell Model

2104 162

8.0 THERMAL HYDRAULIC CONSIDERATIONS

8.1 Description of the Spent Fuel Pool Cooling System

The Spent Fuel Pool Cooling (SFPC) systems are described in detail in Hatch Unit 1 FSAR Section 10.4 and Unit 2 FSAR Section 9.1.3. The Hatch Unit 1 SFPC system includes two 4.25×10^6 Btu/hr capacity cooling trains. Unit 2 has a single 4.25×10^6 Btu/hr cooling train. One of the two Unit 1 trains is devoted to Unit 1 cooling and the other functions as a standby swing cooling train which is designed to operate as part of either the Unit 1 or the Unit 2 SFPC system. Normal make-up water to the spent fuel pool is provided from each unit's condensate storage tank. Plant service water provides a manually initiated backup Seismic Category I make-up source to each spent fuel pool. The SFPC system is not designed to meet Seismic Category I requirements; however, this design was justified, reviewed in detail, and approved by the NRC prior to issuance of the Unit 2 Operating License.

Interconnection of the Residual Heat Removal (RHR) system to the SFPC system is possible and provides a Seismic Category I, 31.3×10^6 Btu/hr capacity backup cooling system; i.e., the RHR system and the SFPC system piping exposed to RHR flow comprise a Seismic Category I method of cooling. Only a fraction of the capacity of the RHR system is required in this mode of operation such that a restricting orifice is provided to limit the amount of water delivered to the fuel pool for cooling. This flow of approximately 1700 gpm uses one RHR pump and one RHR heat exchanger to maintain the pool water under 150°F for maximum heat load conditions when the entire core is discharged to the pool. Under normal conditions, a closed valve and a blind flange provide dual isolation to the inlet and outlet lines connecting the SFPC system to the RHR system.

8.2 Heat Loads and Pool Temperatures for Present Storage Capacity

Three design conditions were postulated for the design of the SFPC system:

1. Normal Condition

   The water in the pool is held to 125°F or less with a heat load of 4.25×10^6 Btu/hour generated by stored fuel consisting of a 25 percent core that was unloaded from the reactor 30 days before and a 25 percent core that has been in storage for one year from a previous refueling operation. Thirty days after unloading the fuel to the pool, the rate of heat generation from the spent fuel is assumed to approach a constant level. It is also assumed that the 25 percent core unloaded at each refueling outage has had a maximum residence time in the reactor of four years. Under normal conditions, a single cooling train with a capacity of 4.25×10^6 Btu/hour will be sufficient to maintain the pool water at or below 125°F.

2104 163

2. Refueling Condition

The pool water is held at 125°F or less with a heat load of 8.5×10^6 Btu/hr generated by stored fuel consisting of a 25 percent core that has decayed for 150 hours since reactor shutdown plus a 25 percent core in storage for one year from a previous refueling operation. The minimum projected time after reactor shutdown to accomplish cooling and opening of the reactor vessel and completion of transferring the spent fuel to the pool is 150 hours. With the assistance of the standby swing cooling train, a combined cooling capacity of 8.5×10^6 Btu/hr is available to cope with the heat generated by newly unloaded fuel and to hold the pool water at or below 125°F.

3. Maximum Condition

The pool water is held to 150°F or less with a heat load of 31.3×10^6 Btu/hr generated by stored fuel consisting of a 100 percent core unloaded from the reactor plus a 25 percent core held over for one year from a previous refueling. The 125 percent core load is assumed to have undergone the following exposures:

25 percent of core:	4 year exposure + 1 year decay
25 percent of core:	4 year exposure + 150 day decay
25 percent of core:	3 year exposure + 150 hour decay
25 percent of core:	2 year exposure + 150 hour decay
25 percent of core:	1 year exposure + 150 hour decay

Under the maximum condition postulated, it is assumed that approximately 150 hours after reactor shutdown the entire core in the reactor will have been transferred to the pool. Thus, the RHR system will be free for cooling the large fuel load in the pool. With the full core offload plus one quarter core remaining from a previous refueling, a single train of the RHR system, without the assistance of SFPC, will maintain the spent fuel pool temperature at or below 150°F 150 hours after the shutdown.

Operating experience with Hatch Unit 1 has indicated that calculated spent fuel pool heat loads and temperatures for the design basis are conservative and the actual heat loads have been approximately 15 percent less than the heat loads calculated.

8.3 Heat Loads and Pool Temperatures for Increased Storage Capacity

8.3.1 To re-evaluate the Plant Hatch spent fuel pool cooling capabilities with the enlarged storage capacity, the decay heat loads were calculated using methods described by Branch Technical Position ASB 9-2 of the Standard Review Plan.

8.3.2 The pool capacity for the increased storage capacity heat load evaluation is assumed to be 5.83 cores. The 5.83 core capacity is arrived

2104 164

at by assuming 1/4 core yearly offloads to the spent fuel pool up to 5-1/2 cores (22 batches) plus an additional batch (batch 23) of 1/3 core. All batches are assumed to have operated at full power for 90 percent of their four-year exposure time. The three design conditions postulated in Section 8.2 are similarly evaluated below.

8.3.2.1 Normal Condition

The heat load analysis for the normal operating condition assumed that there were 22 batches in the pool that had decayed from 1 to 22 years, and the latest batch (23) decayed for 30 days. A single spent fuel pool cooling system train was used for decay heat removal.

The analysis showed that the heat load was 7.24×10^6 Btu/hr and bulk pool water temperature was at or below 139°F. Heat loads and pool temperatures as a function of refueling batches are shown in Figure 8-1.

8.3.2.2 Refueling Condition

The assumptions for the refueling mode analysis were the same as those for the normal mode except that the latest batch was assumed to have decayed for only 150 hours and two spent fuel pool cooling trains were in service.

The analysis showed the heat load was 11.57×10^6 Btu/hr and the bulk water temperature at or below 133°F. Heat loads and pool temperatures as a function of refueling batches are shown in Figure 8-2.

8.3.2.3 Maximum Condition

The analysis for the heat load following full core discharge assumed that the pool already had 19 quarter core batches in storage that had decayed from 1 to 19 years. The calculated heat load from the 19 batches was 2.39×10^6 Btu/hr. The additional decay heat load at 150 hours after shutdown for a full core offload was calculated to be 26.3×10^6 Btu/hr. Therefore, the cumulative heat load in the pool at 150 hours after shutdown is 28.69×10^6 Btu/hr. With a single train of the RHR system aligned for fuel pool cooling duty without the assistance of the SFPC system, the system will maintain pool water temperature at or below 145°F. Figure 8-3 shows the heat load as a function of time after shutdown for the full core discharge.

As an alternative to aligning the RHR system to the spent fuel pool for a full core offload, the fuel may be allowed to decay in the reactor vessel until the heat load of the core has decreased to a point where the SFPC system can maintain a temperature less than the design maximum temperature. A waiting time of 500 hours (approximately 21 days) is required in this case prior to full core offload. After this time, two fuel pool cooling trains can maintain the pool water temperature at or below 150°F (i.e., a heat removal capability of 18.77×10^6 Btu/hr).

8.3.3 For each design condition analyzed in 8.3.2, completely utilizing the expanded spent fuel pool storage capacity, the present SFPC systems or a single train of the RHR (for the full core offload condition) are capable of maintaining pool water temperatures less than the design maximum temperature of 150°F. Considering the conservative assumptions used in the calculations and past operating experience, the actual temperatures for each condition are expected to be lower than those calculated and described above.

8.4 Loss of Spent Fuel Pool Cooling

The consequences of a loss of the SFPC systems has been evaluated for the following two conditions:

1. Concurrent loss of the SFPC systems.

2. Maximum heat load.

8.4.1 Concurrent Loss of SFPC Systems

Both spent fuel pools are assumed to be loaded as delineated in Section 8.3.2. Unit 1 and Unit 2 are assumed to be shut down for refueling 21 days apart, with Unit 2 being shut down first. Also, 21 days is assumed to be the minimum time required to complete a refueling operation. Therefore, Unit 2 will be operating while Unit 1 is shut down. Subsequently, both units' SFPC systems are postulated to be lost 150 hours after Unit 1 is shut down.

Calculations using pool water volumes of 38,640 ft^3 each indicate that the time to boil for the Unit 1 pool is 14.7 hours and that the time to boil for the Unit 2 pool is 22.8 hours. The makeup water requirement following boiling was calculated to be 24 gpm for the Unit 1 pool and 15 gpm for the Unit 2 pool. During transition to boiling, no credit is taken for evaporative heat losses. Water level is maintained by the Seismic Category I Plant Service Water system. Conservatisms are included in the analysis by assuming that all decay heat is rejected to the pool water and none is rejected to the structures. Also, the heat capacity of the makeup water is neglected.

After approximately 150 hours following Unit 1 shutdown, the decay heat contributed by 2/3 core in the Unit 1 reactor pressure vessel has decreased enough to allow aligning one train of the RHR to provide spent fuel pool cooling and reactor pressure vessel cooling. With the reactor vessel head and the spent fuel pool gates removed, the RHR system can be aligned for spent fuel pool and reactor pressure vessel cooling by installation of two spectacle flanges and operation of four isolation valves. The time required for realignment is estimated to be 8 hours.

Subsequent to loss of the SFPC systems, Unit 2 will be brought to cold shutdown. A radiological analysis has been performed assuming that both pools boil simultaneously. The consequences are presented in Section 8.6.

8.4.2 Maximum Heat Load

A full core offload creates the highest heat load in the spent fuel pool. However, with no fuel in the reactor pressure vessel, the RHR system is available for unrestricted spent fuel pool cooling. The redundant Seismic Category I design of the RHR system provides a high degree of assurance that it operates satisfactorily in the spent fuel pool cooling mode.

8.5 Local Fuel Bundle Thermal Hydraulics

The bounding thermal-hydraulic conditions were calculated for fuel stored in a HDFSS module in the Hatch pools. Bases for the calculations for typical current generation fuel were the following:

Maximum bundle burn-up	35,300 MWD/MTU	
Specific Power	36.6 kW/kgU	20% of time
	48.3 kW/kgU	60% of time
	60.0 kW/kgU	20% of time

The ORIGEN Code (Reference 4) was used to calculate the decay heat for the bundle defined by these bases. The result was:

Actinide Contribution	9,500 W/MTU
Fission Product Contribution	152,000 W/MTU
TOTAL	161,500 W/MTU

With the bulk water temperature of the spent fuel storage pool constant at 140°F, the maximum fuel cladding temperature will be 186.1°F. The maximum water temperature associated with the hottest fuel bundle will be 163.2°F. These temperatures and the maximum storage tube wall temperature of 157.5°F are low relative to structural integrity or corrosion limiting temperatures of the structural components of the storage system and fuel.

A second set of calculations bracketed the thermal hydraulic conditions expected in potential future fuels. The bases used for these calculations were:

Maximum bundle burn-up	44,000 MWD/MTU	
Specific Power	20.3 kW/kgU	20% of time
	40.2 kW/kgU	60% of time
	60.0 kW/kgU	20% of time

2104 107

The ORIGEN Code was used as before, but the initial U^{235} content was adjusted to 3.6 weight percent to correspond to the higher burn-up value. The decay heat calculated was:

Actinide Contribution	10,700 W/MTU
Fission Product Contribution	155,000 W/MTU
TOTAL	165,700 W/MTU

These values result in a maximum fuel cladding temperature of 186.6°F. The maximum water temperature will be 163.6°F and the maximum storage tube wall temperature will be 157.7°F. There is no thermal-hydraulic problem presented by potential future high burn-up fuels.

Continuing efficiency of the exchange of heat from the spent fuel to the pool water depends on the convection flow of water through the storage tube and flow channel, if present, encompassing a fuel bundle. The floc-like crud that adheres to the surfaces of the spent fuel bundles was studied to determine whether it is a potential mechanism for blocking flow through the channel. The floc was found to be extremely fine; pieces that spall off of the aggregate are not disposed to settle, but will flow upward with the convection current. Additionally, the floc is so fine that some of it will pass through conventional laboratory filter papers. Growth of crud in fuel storage conditions has not been observed in commercial facilities. The potential for channel plugging by se[illegible] or by blockage of flow passages is therefore negligible.

8.6 Radiological Impact of Spent Fuel Pool Boiling

The radiological impact of spent fuel pool boiling is maximized by assuming simultaneous failures of the SFPC systems for both Units 1 and 2 as described in Section 8.4.1.

A radiological analysis has been performed to determine the thyroid dose at the site boundary/LPZ, assuming that the pools boil and that there has been an iodine spike in the pools.

The assumptions used are as follows:

1. The time to reach boiling is 14.7 hours for Unit 1 and 22.8 hours for Unit 2.

2. Boiling rate of the pool water is 11,955 lb/hr for Unit 1 and 7700 lb/hr for Unit 2.

3. Volume of water in each pool is 38,640 ft^3.

4. All failed fuel rods of the full core (average 1 percent of the core) are present in the 1/3 core discharged to each pool.

5. The normal I-131 release rate coefficient for leaking rods in the Unit 1 pool is 4.6×10^{-10} sec^{-1} at 150 hours and 1×10^{-10} sec^{-1}

2104 168

for leaking rods in the Unit 2 pool at 27.25 days (21 days + 150 hours) using the methods described in Reference 5. These release rate coefficients are conservatively assumed to be constant during the heatup and boiling periods.

6. The above release rate coefficient is spiked by a factor of 100 to simulate the heatup conservatively.

7. The decontamination factor for I-131 during boiling is conservatively assumed to be unity.

8. No credit is taken for iodine plate-out or filtration by the standby gas treatment system.

9. Conservative ground level accident X/Q values are assumed for the dose calculation.

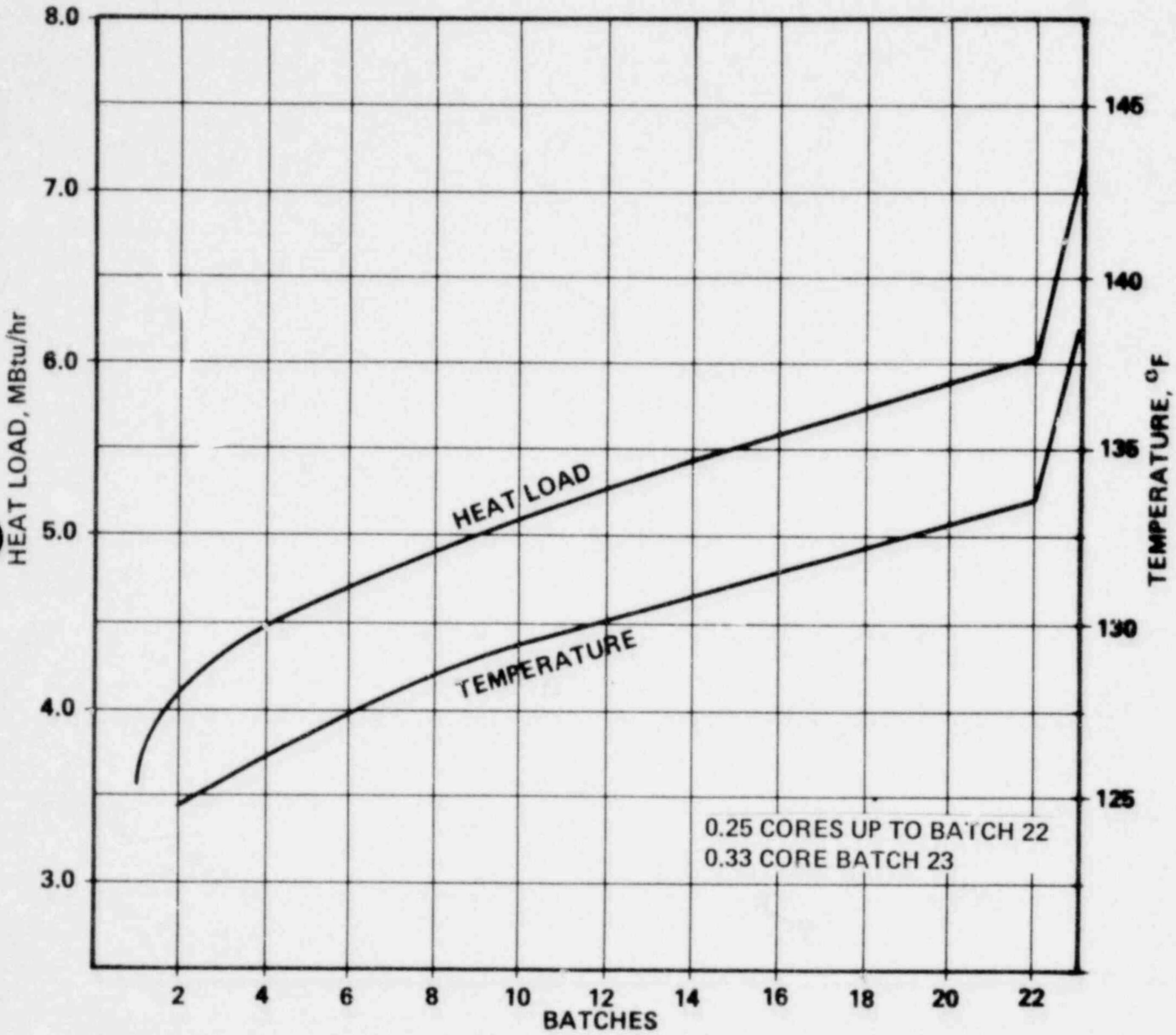
The results are summarized below:

Site boundary/LPZ thyroid dose (0-2 hrs.) 1.5 rem

Site boundary/LPZ thyroid dose (0-4 days) 9.3 rem

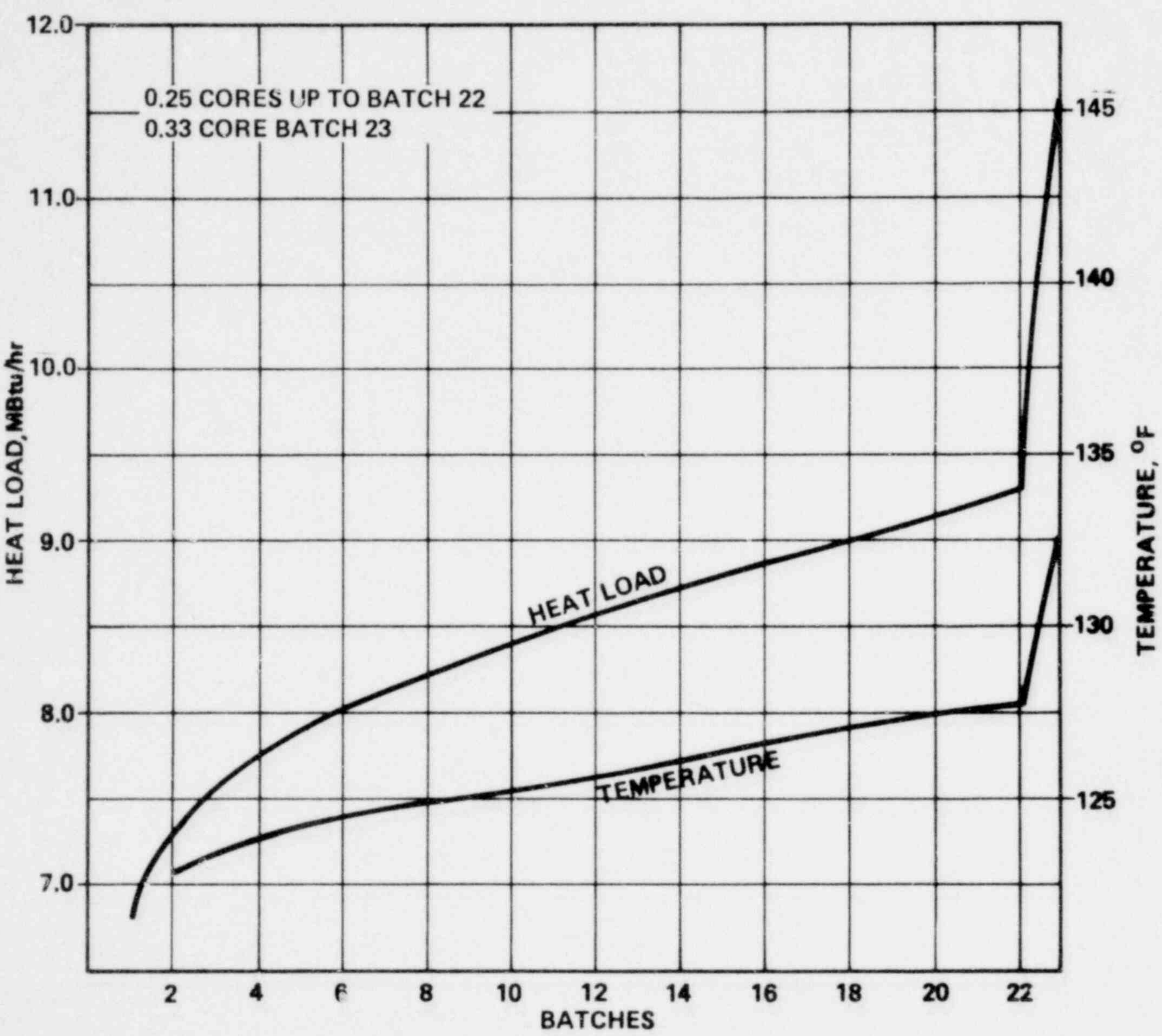
The above results, which are based on boiling of both Unit 1 and 2 pools, compared to the results presented in the Hatch 2 FSAR (response to Question 20.20 - 1.3 rem for 0-2 hours and 8.3 rem for 0-4 days), which are based on boiling of the Unit 2 pool only, support the Applicant's position that the SFPC system need not be upgraded to meet Seismic Category I design requirements.

2104 169



2104 170

Figure 8-1 Fuel Pool Heat Load & Temperature 30 Days After Shutdown



2104 171

Figure 8-2 Fuel Pool Heat Load & Temperature 150hr After Shutdown

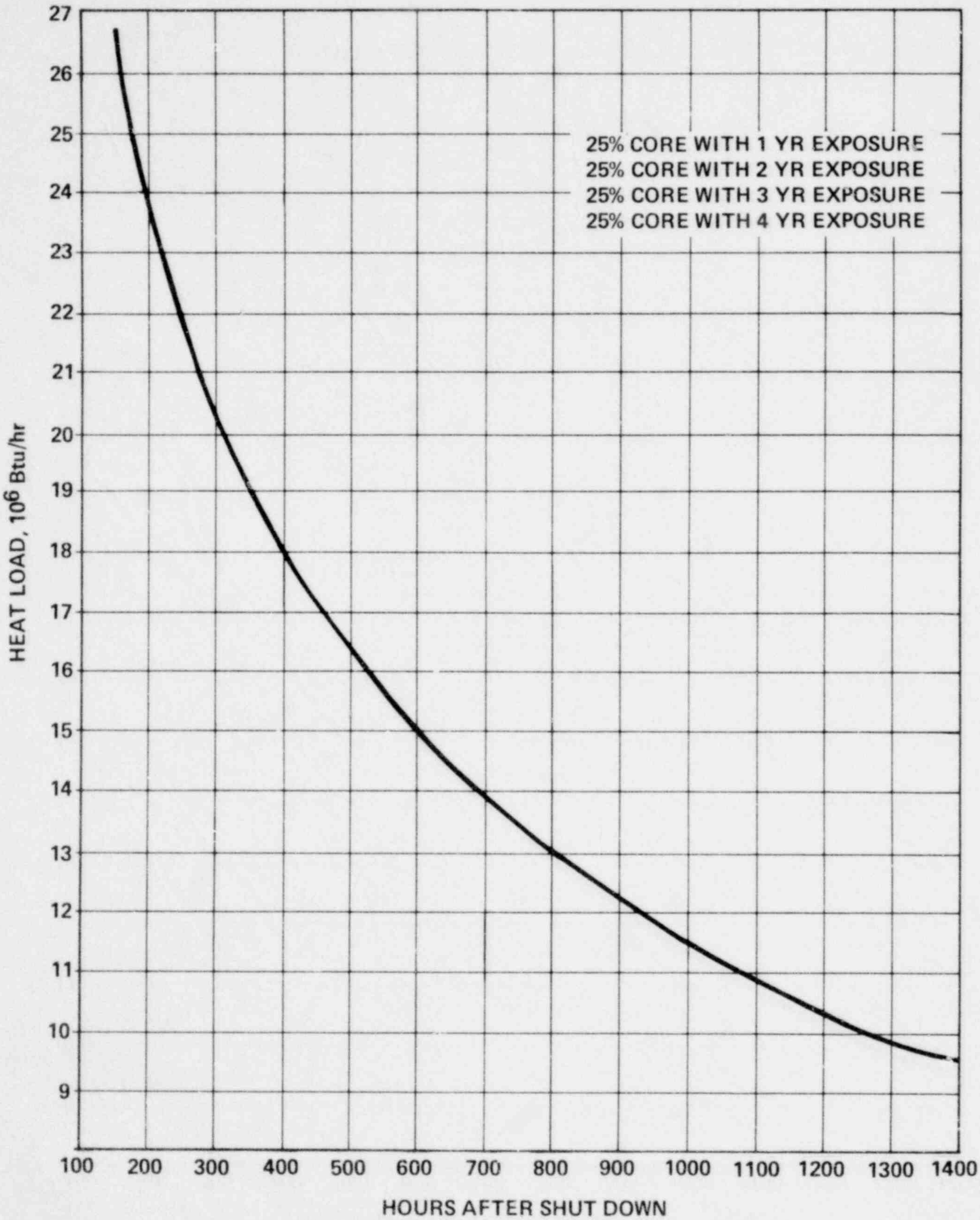


Figure 8-3 Full Core Heat Load As a Function of Time After Shutdown

9.0 COST BENEFIT ASSESSMENT

9.1 Need for Increased Storage Capacity

The present spent fuel storage facilities at Hatch Units 1 and 2 were designed for temporary storage of spent fuel until the fuel had decayed enough for safe transport to a reprocessing facility. The absence of activity in the construction of new fuel reprocessing facilities and the cessation of operation of exisiting reprocessing facilities have created the need for increased on-site storage of spent fuel to permit long-term power plant operation.

The terms of Georgia Power Company's 1968 fuel supply contract with General Electric provide for the buy back and removal of the first two cores of Hatch-1 fuel by General Electric. Georgia Power and General Electric have agreed to defer removal of this fuel and to temporarily store this fuel at Plant Hatch. Georgia Power also entered into a reprocessing contract with General Electric covering spent fuel discharged for reprocessing through 1983. In 1974, General Electric informed Georgia Power that its Morris, Illinois, reprocessing facility was inoperable and that the contract was being terminated.

The anticipated spent fuel discharge schedule for the Hatch Nuclear Plant is described in Table 9-1. A review of the schedule indicates that, with the present storage rack configuration, full core storage reserve capability will be lost in 1983 and all storage capacity will be expended in 1985. This prediction is based on maintaining reserve storage for a single core using the combined storage capacities of both spent fuel pools. This is possible because Unit 1 and Unit 2 share a common refueling floor and a transfer canal which connects the two spent fuel pools.

Expansion of the storage capacity in both pools by using the General Electric designed high-density, poisoned storage racks will provide enough reserve storage capacity for off-loading a full core until 1997 and will provide spent fuel storage without a full core reserve until 1999.

Presently, the Hatch spent fuel pools contain the following items in addition to the fuel and fuel racks:

2 control rod assemblies (Unit 1 pool)

8 control rod blade guides (Unit 1 pool)

140 control rod storage locations, Unit 1

40 control rod storage locations, Unit 2

Test weights for the fuel handling bridge, Unit 1 and Unit 2 (Unit 2 weights to be removed)

2104 173

Underwater vacuum cleaner (Unit 1)

Miscellaneous other equipment such as fuel sipping canisters which are temporarily located in the pool for outage work.

9.2 Alternative to Increasing Storage Capacity

9.2.1 Several alternatives to the expansion of the storage capacities of the Hatch Unit 1 and Unit 2 spent fuel pools to alleviate the spent fuel storage space storage were considered.

In summary, the alternatives were:

a. shipment to a fuel reprocessing facility.

b. shipment to an independent spent fuel storage facility.

c. shipment to another reactor site.

d. shutting down the reactor.

9.2.1.1 Shipment to a Fuel Reprocessing Facility

There are currently no commercial spent fuel reprocessing facilities in operation in the United States. In April 1977, the President of the United States announced a spent nuclear fuel policy which included the indefinite deferral of commercial reprocessing in the U.S. nuclear power program. Reprocessing of spent fuel is not a viable alternative to the expansion of the Hatch spent fuel pools. Storage of the Hatch spent fuel at the existing (although not operating) reprocessing facilities is also not a viable alternative to the expansion of the Hatch spent fuel pools since the facility owners are not offering to provide comparable storage capacity.

9.2.1.2 Shipment to a Storage Facility

Spent fuel storage at a private or government operated independent spent fuel storage facility is not currently available. The alternative of constructing a facility to serve Plant Hatch would not be economically viable. The Department of Energy has estimated that construction of a 5000 MTU independent spent fuel storage facility would cost $200,000,000 (DPE/ET-0055 "Preliminary Estimates of Charge for Spent-Fuel Storage and Disposal Services", July 1978) or about $40/kg. A smaller facility designed to serve Plant Hatch would be expected to have a higher cost per kg. These costs are significantly larger than the estimated cost of the increased storage capacity which will be obtained by expanding the present reactor pools (approximately $12.5/kg).

9.2.1.3 Shipment to Another Reactor Site

The only available reactor site which could be used as an alternative for Plant Hatch spent fuel storage facility within Georgia Power

Company is Vogtle Nuclear Plant (a PWR) Unit 1 which has an expected inservice date of November 1984. This schedule cannot prevent Plant Hatch from losing its full core reserve capacity in 1983; neither can it alleviate the Plant Hatch spent fuel storage problem until the back-end-of-fuel-cycle problems are resolved. However, even if Plant Vogtle were used as an alternative site for Plant Hatch spent fuel storage, the estimated cost would be greater than that of expanding the Hatch pools, as shown below. The costs do not reflect the loss of storage space at Plant Vogtle.

1.	Cost of BWR spent fuel storage racks	\$1,300/assembly
	Installation (9%)	120
	Contingencies (10%)	130
	Engineering, supervision, and overhead (including licensing) (20%)	250
		\$1,800/assembly
2.	Cost of transportation (with cask rental)	\$1,200/assembly
3.	Total Cost	\$3,000/assembly (approximately \$16/kg)

9.2.1.4 Plant Shutdown

Shutdown of the Hatch Nuclear Plant would require the purchase of power from substitute sources and/or production from less economical sources within the system. The figures shown in Table 9-2 are the increased production costs (actual year dollars) to the Southern electric system for replacement power if Unit 1 and Unit 2 are closed after the 1983 refueling. These figures do not include any capital (fixed) cost dollars that still would have to be amortized 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.

9.3 Capital Costs

Costs incurred by expanding the spent fuel storage capabilities at the Hatch Plant are summarized on Table 9-3. These costs represent the current prediction of the total project costs, including the installation of the high density spent fuel storage racks and disposal costs of the presently installed racks. Indirect capital costs other than those specified have not been considered.

The overall scope of the project will include the following:

2104 175

a. Design feasibility study.

b. License amendment preparation and submittal.

c. Engineering studies to support license amendment including nuclear analysis, seismic analysis, and thermal-hydraulic analysis.

d. Installation preparation, including removal and disposal of original racks, hold-down clips, seismic restraints, etc..

e. Installation of new racks.

f. Development and implementation of poison verification procedures.

9.4 Resource Commitment

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 consumed by the proposed modification are listed below.

Material	Hatch Modification Quantity (lb)
304 Stainless Steel	5.8×10^5
Boron Carbide	1.4×10^4
Aluminum	5.1×10^4

9.5 Environmental Impact of Expanded Spent Fuel Storage

An analysis of the Hatch Unit 1 spent fuel pool heat load when filled to the present 1.5 core capacity, 30 days after the last refueling shutdown, indicates that the bulk spent fuel pool temperature will be approximately 127.5°F. The bulk temperature of the Unit 2 spent fuel pool when filled to its 2.0 core capacity will be approximately 128°F. For the proposed expanded capacity, assuming that the spent fuel pools are filled to their expanded capacity 30 days after the last refueling shutdown, as previously discussed, each reactor building closed cooling water (RBCCW) heat exchanger inlet temperature can be expected to rise less than 1.5 degrees. The total evaporation rate of the two spent fuel pools can be expected to increase by 340 lb/hr.

Each unit has a once-through refueling floor ventilation system with a 30,000 cfm capacity for a combined ventilation capacity of 60,000 cfm. The increased evaporation rate will have negligible effect on the refueling floor ventilation systems and, therefore, no effect on the environment.

2104 176

Assuming that all additional heat transferred to the RBCCW system is ultimately transferred to the plant service water system and assuming that no heat is lost through piping or components, the plant service water discharge temperature will be increased by approximately 0.6°F.

Therefore, under normal conditions, the spent fuel pool storage expansion will have negligible effect on the operation of installed plant components and negligible impact on the environment as a result of increased heat loads.

2104 177

Table 9-1
Estimated Spent Fuel Discharge Schedule

	Year	Annual Discharge Schedule (No. of Assemblies) Unit 1	Unit 2	Combined	Cumulative Discharges (No. of Assemblies)
	1977	92		92	92
	1978	168		168	260
	1979	164		164	424*
	1980	140	168	308	732
	1981	140	140	280	1012
	1982	140	140	280	1292
(1)	1983	140	140	280	1572
	1984	140	140	280	1852
(2)	1985	140	140	280	2132
	1986	140	140	280	2412
	1987	140	140	280	2692
	1988	140	140	280	2972
	1989	140	140	280	3252
	1990	140	140	280	3532
	1991	140	140	280	3812
	1992	140	140	280	4092
	1993	140	140	280	4372
	1994	140	140	280	4652
	1995	140	140	280	4932
	1996	140	140	280	5212
(3)	1997	140	140	280	5492
	1998	140	140	280	5772
(4)	1999	140	140	280	6052
	2000	140	140	280	6332

*Presently in storage

(1) Existing storage capacity - loss of full core reserve
(3) Existing storage capacity - filled
(3) Expanded storage capacity - loss of full core reserve
(4) Expanded storage capacity - filled

2104 178

Table 9-2

Replacement Power Costs in Actual Year Dollars

Year	Differences In Cost If Generated In The Southern Electric System x $1,000	+ [Difference In Emergency Energy Purchased* (GWH)	X Combustion Turbine Generation ($/MWH)	X 112%**]	= Total Cost Difference x $1,000
3/83-12/83	$150,805.00	157.0	80.86		$165,023.00
1984	207,193.00	72.1	86.46		214,175.00
1985	197,139.00	57.0	92.57		203,049.00
1986	108,872.00	171.3	98.59		227,787.00
1987	213,569.00	236.4	105.39		241,473.00
1988	226,721.00	231.1	112.93		242,665.00
1989	226,721.00	473.6	120.25		290,505.00
1990	239,770.00	563.8	128.97		321,209.00

*This energy would be purchased outside the Southern electric system, if available.
**Price is assumed to be 112% of Georgia Power's most expensive combustion turbine generation.

2104 179

Table 9-3

Capital Costs

New Fuel Storage Racks for Spent Fuel Pool	$6,100,000
Installation (including Disposal Costs)	$ 552,000
Contingencies (10%)	$ 665,000
Engineering, Supervision, and Overhead (including Licensing and Legal Fees) (20%)	$ 1,330,000
Subtotal:	$ 8,647,000

Assuming expenditures of 25 percent in 1979, 50 percent in 1980, and 25 percent in 1981, escalation should result in the following additional changes:

1979 Escalation (10%)	$ 648,000
1980 Escalation (10%)	$ 432,000
1981 Escalation (10%)	$ 216,000
SUBTOTAL:	$1,296,000

Allowances for funds used during construction are calculated on a cumulative percentage basis and result in the following additional charges:

1979	$ 67,500
1980	$275,500
1981	$484,000
SUBTOTAL	$ 827,000

Adding all of the above costs results in a total budget projection for the project of:

$10,770,000

2104 180

10.0 RADIOLOGICAL EVALUATION

10.1 Spent Resin Waste

The fuel pool filter-demineralizer units are designed to maintain a water conductivity of less than 0.5 micro mho/cm. The units are backwashed when either the differential pressure across the demineralizers is greater than 10 psi or the effluent conductivity is greater than 5 micro mhos.

Hatch Unit 1 experience indicates that the filter-demineralizer was backwashed 41 times during 1978. Each backwash cycle amounts to 2.5 cubic feet of spent resin. The dose attributed to handling of the spent fuel pool resin in the radwaste system is approximately 0.3 man-rem/yr.

The increase in the spent fuel pool storage capacity is not expected to appreciably affect the annual amount of solid radwaste or the annual man-rem dose.

10.2 Noble Gases

Krypton-85 is released to the pool water and subsequently to the refueling floor atmosphere from the leaking fuel assemblies. For normal operating conditions, most of the krypton comes from the most recently discharged batch of fuel. After the most recent batch has cooled in the pool for 12 months, the pressure buildup in a fuel pin which causes the release of krypton has become very small. Thus, the increase in krypton-85 activity attributed to the increase in spent fuel pool storage capacity will be small compared to the total quantity of all noble gases released form the pools and negligible when compared to the annual plant noble gas releases. Despite the presence of some defective fuel bundles in the Unit 1 pool, krypton-85 activity levels in the refueling floor ventilation exhaust are below the minimum detectable level of approximately 10^{-8} μCi/cc.

10.3 Gamma Isotopic Analysis for Pool Water

Hatch Unit 1 has undergone three refuelings. Typical radioactive isotope concentrations in the Unit 1 spent fuel pool water are presented in Table 10-1 at various dates.

10.4 Dose Levels Over and Along Sides of Pool

Dose surveys at Hatch Unit 1 indicate that after every refueling outage the radiation field over the pool surface has returned to an apparent equilibrium of approximately 1 mr/hr. Local areas show 4 mr/hr (e.g., around the fuel grapple).

Measurements taken during the May 1979 refueling outage show that the radiation levels along the sides and center of the pool are essentially the same (approximately 2 mr/hr). This indicates there has been no significant crud build up around the sides of the pool and the radiation levels are as low as reasonably achievable.

2104 181

10.5 Airborne Radioactive Nuclides

Air samples taken from the Unit 1 refueling floor atmosphere during and after each refueling showed activity levels below the lower level of detection. Storage of additional fuel is not expected to increase the airborne activity on the refueling floor since the major contribution of airborne activity is attributed to the most recent batch of spent fuel that is placed in the pool.

10.6 Radiation Protection Program

The Radiation Protection Program is described in Section 12.5 of the Hatch 2 FSAR. This program will be adhered to during the removal of the old racks and installation of the new racks.

10.7 Disposal of Present Spent Fuel Racks

There are at present 42 aluminum racks in the Unit 1 pool and 56 in Unit 2. Each rack weighs about one ton. Presently, there is no fuel stored in the Unit 2 spent fuel pool. The racks removed from Unit 2 will be prepared and stored in the warehouse for future sale or use. The racks from the Unit 1 pool will be decontaminated, crated and shipped offsite to a licensed burial location. A reasonable effort will be made to limit personnel exposures to as low as reasonably achievable during this work.

2104 182

Table 10-1

Radioactive Isotopic Concentrations in the Spent Fuel Pool Water

Isotope	Fuel Pool Activity (μCi/cc)			
	7/11/78	1/15/79	5/8/79	5/29/79
I-131	LLD*	LLD	6.76E-5	LLD
Xe-133	LLD	LLD	6.06E-5	LLD
Mo-Tc-99m	LLD	LLD	1.25E-5	LLD
Cr-51	LLD	LLD	6.15E-4	LLD
F-18	LLD	LLD	3.49E-5	LLD
Cs-134	1.03E-5	5.73E-6	1.74E-4	3.58E-5
Cs-137	1.89E-6	7.64E-6	1.46E-4	3.72E-5
Zr-95	LLD	LLD	1.40E-4	7.4E-6
Nb-95	LLD	LLD	1.55E-4	9.7E-6
Co-58	4.19E-6	8.29E-7	5.0E-5	1.09E-5
Mn-54	LLD	LLD	6.77E-5	7.33E-6
Fe-59	LLD	LLD	5.68E-5	LLD
Zn-65	3.00E-5	4.1E-5	5.65E-4	6.94E-5
Co-60	5.1E-6	5.31E-6	1.58E-4	8.88E-6

*Lower level of detection

2104 183

11.0 ACCIDENT EVALUATION

The spent fuel shipping cask drop analysis is described in the Hatch Unit 1 FSAR Question 10.3.4 response. The referenced drop analysis is applicable to Unit 2. Since the cask will not be handled over or in the immediate vicinity of either the Unit 1 or the Unit 2 spent fuel pool, the consequences of the cask drop are not affected by the installation of additional spent fuel storage capacity.

Protection against the cask drop is afforded by the licensed single failure proof crane described in Hatch Unit 1 FSAR Section 10.20, by the single failure proof cask yoke described in Hatch Unit 2 FSAR Subsection 9.1.4.2.2, and by the interlocks and administrative controls described in the same subsection which limit the cask height over the refueling floor during cask handling operations.

The Hatch Nuclear Plant design also incorporates several levels of protection against the drop of other crane loads into the spent fuel pool and onto stored spent fuel.

The overhead crane is interlocked to prohibit operation over the spent fuel pool. The interlocks can be overridden, but only under strict administrative controls. The only postulated loads which would require bypassing the interlocks which prohibit movement over the spent fuel pool are the handling of the spent fuel pool plugs (9 tons) and gates, and removal and installation of the old and new spent fuel racks, respectively, as discussed in Section 6.0. The spent fuel pool gates and plugs will be handled only under strictly controlled administrative procedures. Additional information pertaining to the control of heavy loads near spent fuel has previously been discussed in Reference 6.

If unanticipated load handling should occur, the size of the load that can be handled over stored spent fuel, by any means, is limited to 1600 pounds by Hatch 2 Technical Specification 3/4.9.7. A proposed change to the Unit 1 Technical Specifications will be submitted to incorporate this same requirement.

2104 184

12.0 CONCLUSIONS

The information contained in this document to support the proposed modification satisfies the necessary applicable regulatory requirements to allow NRC approval for Georgia Power Company to rerack the Plant Hatch Units 1 and 2 spent fuel pools and demonstrates that the proposed modification can be safely accomplished. This proposed modification is the most cost effective and desirable alternative, and is in the best interest of the public. The proposed modification does not significantly change or impact any previous determinations which are documented in the Hatch 1 and 2 Safety Evaluation Reports and Final Environmental Statements, and therefore precludes the need for preparation of an environmental impact statement.

2104 185

13.0 NOTES AND REFERENCES

Notes:

1. For the purposes of this report the term "fuel bundle" will imply configuration either with or without flow channels unless the term "fuel assembly" is specifically and distinctly intended.

2. Boral is a product of Brooks and Perkins, Inc., consisting of a layer of boron carbide-aluminum (B_4C-Al) matrix bonded between two layers of aluminum.

References:

1. L. K. Liu, "Seismic Analysis of the Boiling Water Reactor," Symposium on Seismic Analysis of Pressure Vessel and Piping Component, First National Congress on Pressure Vessel and Piping, San Francisco, California, May 1975.

2. U.S. NRC Safety Evaluation for Yankee Rowe, dated December 29, 1976, Page 4, Structural and Material Considerations.

3. C. M. Kang and E. C. Hanson, ENDF/B-IV Benchmark Analysis with Full Spectrum Three-Dimensional Monte Carlo Models, ANS Meeting, November 1977.

4. M. J. Bell, "ORIGEN Code - The ORNL Isotope Generation and Depletion," ORNL-4628.

5. N. Eickelpasch and R. Hock, "Fission Product Release After Reactor Shutdown," IAEA-SN-178/19.

6. Letter from W. E. Ehrensperger, Georgia Power Company, to U. S Nuclear Regulatory Commission, dated July 24, 1978.

2104 186