UNITED STATES
NUCLEAR REGULATORY COMMISSION
WASHINGTON, D. C. 20555

MAR 1 9 1986

Docket No. 50-416

LICENSEE: Mississippi Power & Light Company (MP&L)

FACILITY: Grand Gulf Nuclear Station, Unit 1

SUBJECT: SUMMARY OF FEBRUARY 5, 1986 SITE MEETING REGARDING SPENT FUEL POOL STORAGE FACILITIES

The purpose of the meeting was to observe the high density spent fuel racks and associated shielding walls and pools and to discuss calculations of occupational dose rates. Enclosure 1 is a list of participants. Enclosure 2 is a handout prepared by the licensee.

The licensee described the high density spent fuel racks in the spent fuel pool in the auxiliary building and in the upper containment pool in the containment building. The upper containment pool is for temporary storage during refueling. (Enclosure 2, Sheets 3, 4, 5 and 6).

The licensee described the calculations of occupational radiation doses during storage of the spent fuel. (Enclosure 2, Sheets 7 through 20). Calculated doses are based on placing freshly discharged fuel (with the highest radioactivity) adjacent to the pool walls. Calculated doses on the outside of the spent fuel pool walls are higher than NRC ALARA guideline values. The licensee plans to preclude loading freshly discharged spent fuel adjacent to the walls, as indicated on Sheet 9, Enclosure 2. The licensee will calculate doses for a specific spent fuel storage pattern which will bring calculated doses outside the pool walls within ALARA guideline values. During refueling and after spent fuel is loaded into the pools, the licensee will measure radiation doses to confirm that calculated values are conservative.

The calculated doses outside the upper containment pool are highest on the west wall (Sheet 10 of Enclosure 2). This wall is close to the containment inside wall, however, and the high radiation area is not accessible without placing long ladders or scaffolds in this space. Doses calculated for the south wall, which is easily accessible, are within ALARA valves of 2.5 mR/hr. (Sheet 10, Enclosure 2).

The NRC staff toured the facility to see the spent fuel pool, the upper containment pool and the areas near the spent fuel transfer tube. Areas that are expected to be the highest radiation areas will be monitored during transfer of spent fuel through the tube to assure doses are ALARA.

The licensee said it would provide a fuel storage pattern to maintain doses in easily accessible areas below 2.5 mR/hr. and its plans for surveys and administrative controls during refueling in a letter by March 15, 1986.

Original Signed by

L. L. Kintner, Project Manager
BWR Project Directorate No. 4
Division of BWR Licensing

Enclosures:
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NRC Participants

PD#4/PM	PD#4/D
LKintner:lb	WButler
03/19/86	03/19/86

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Enclosure 1

Attendees

February 5, 1986 Meeting with MP&L

NRC	MP&L
L. L. Kintner	Rod Elms
Mike Lamastra	L. K. Daughtery
J. L. Caldwell	M. L. Crawford
R. C. Butcher	S. E. Thomas
	C. B. Franklin
	D. L. Marshall
	J. C. Vincelli
	F. W. Rosser
	A. S. McCurdy

NRC-MP&L Meeting to Discuss High Density Spent Fuel Racks Installation

Date: February 5, 1986
Time: 9:00 a.m.
Place: PLS Conference Room

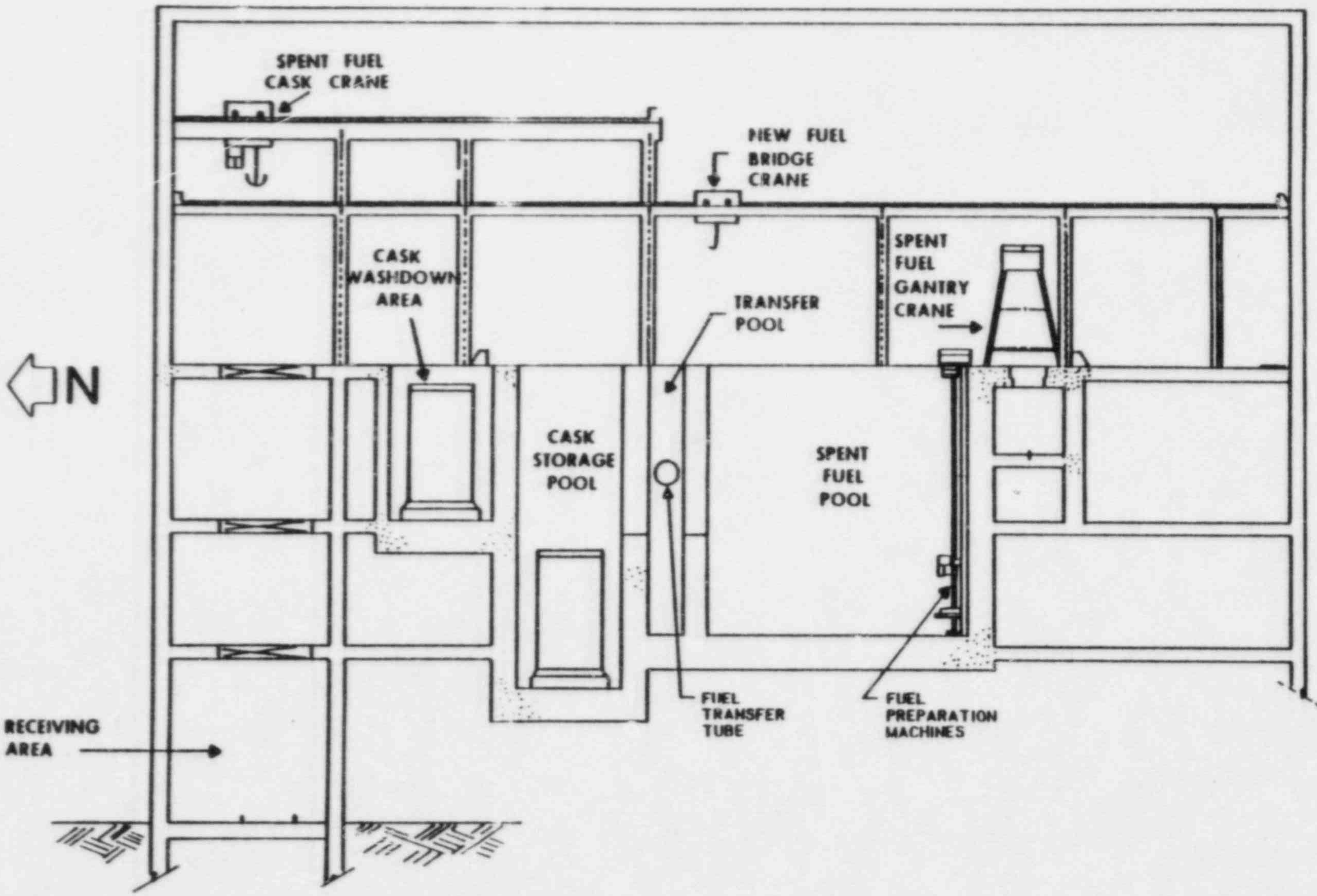
I. Introduction/Purpose	M. Crawford
II. HDSFR Configuration	J. Elms
- Spent Fuel Pool (SFP)	
- Upper Containment Pool (UCP)	
III. Occupational Dose Rate Calculations	B. Franklin
- Methodology	
- Results	
IV. Proposed Administrative Controls	
- Task Force (ALARA, possible changes, i.e. ARMS)	S. Thomas
- H. P. Surveys during refueling outage	J. Vincelli
- Refueling Procedures	A. McCurdy
V. Questions/Answers	MP&L/NRC
VI. Tour	J. Elms

/

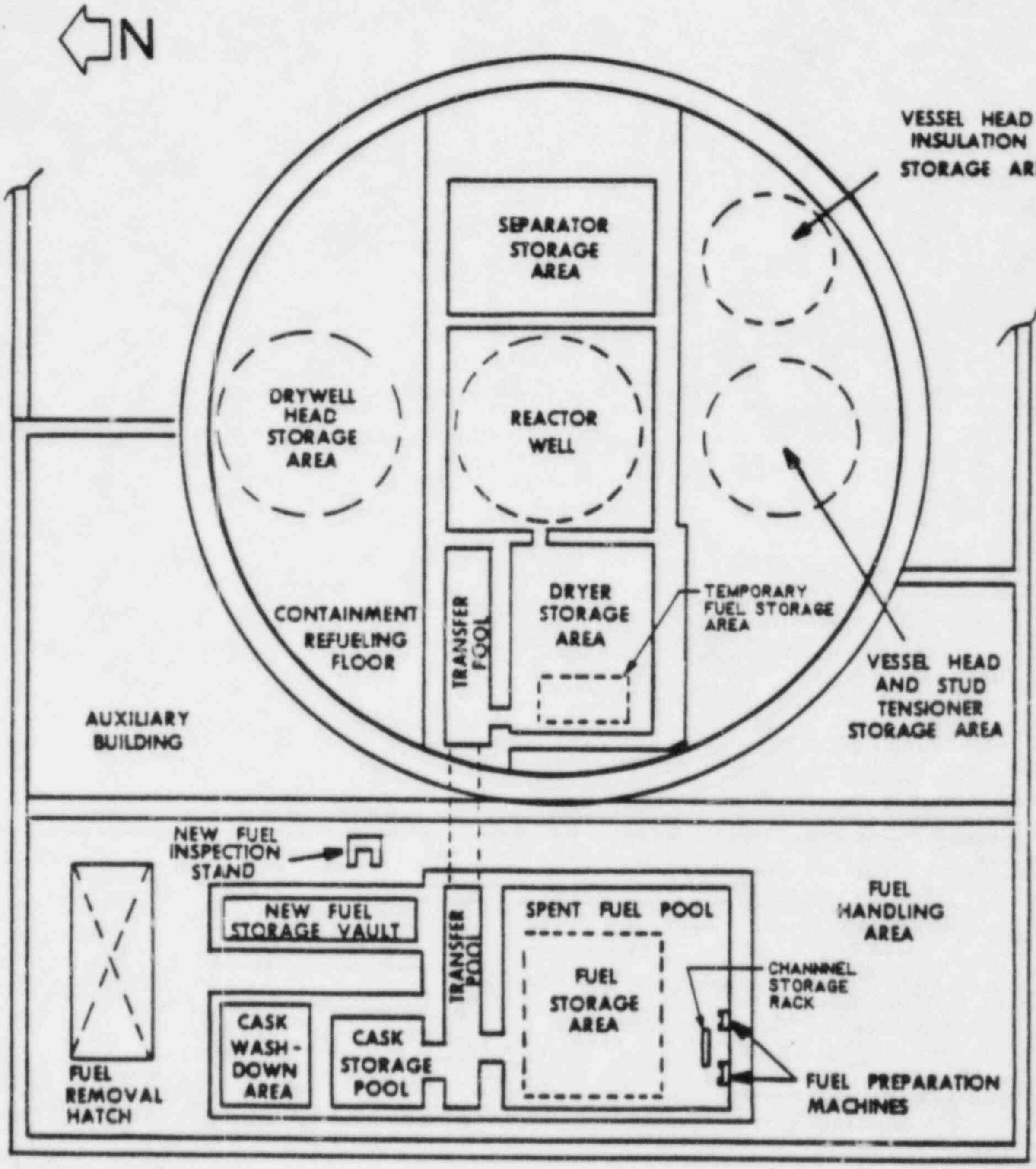
HDSFR CONFIGURATION

Spent Fuel Pool (SFP)

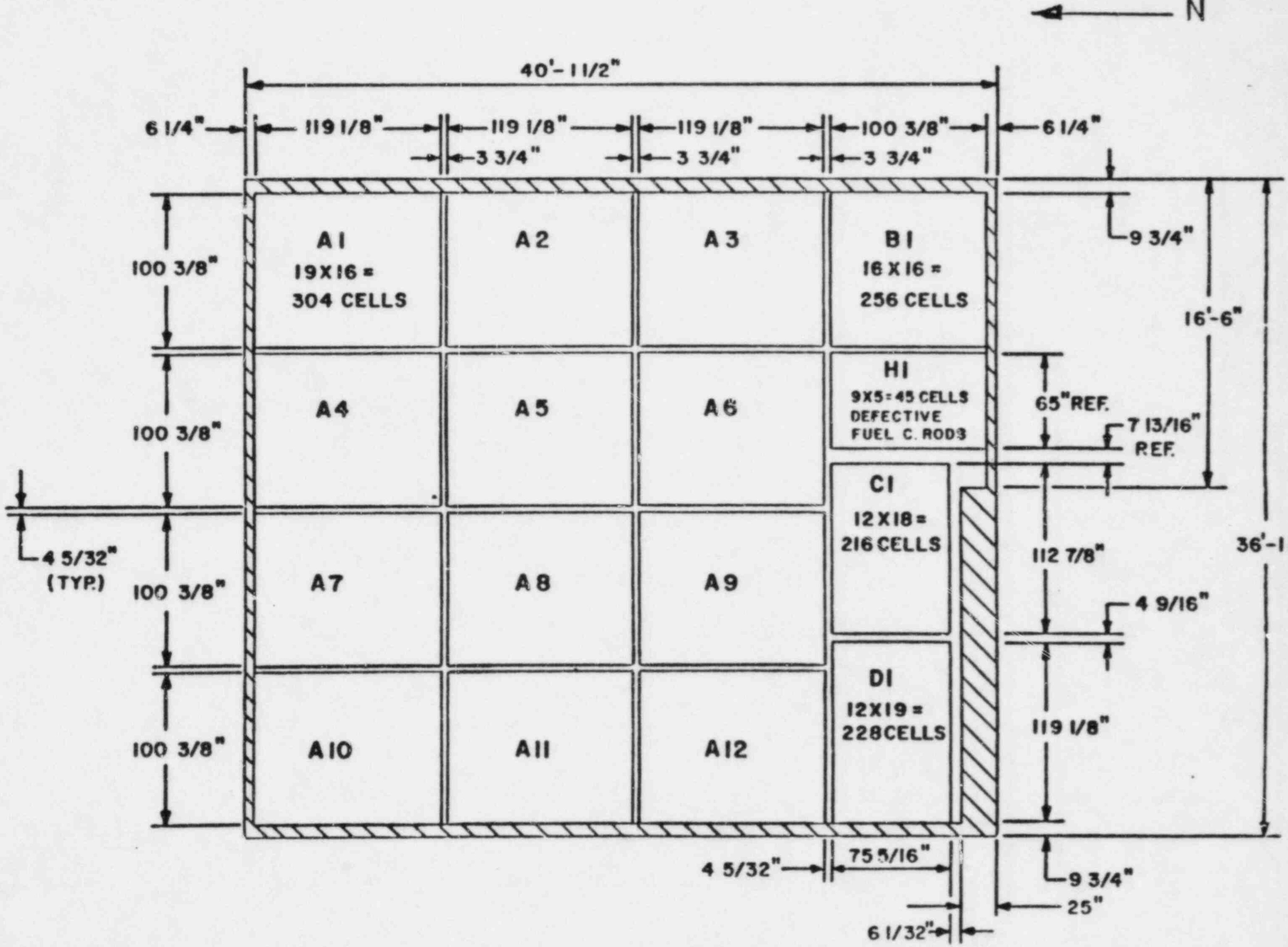
Upper Containment Pool (UCP)



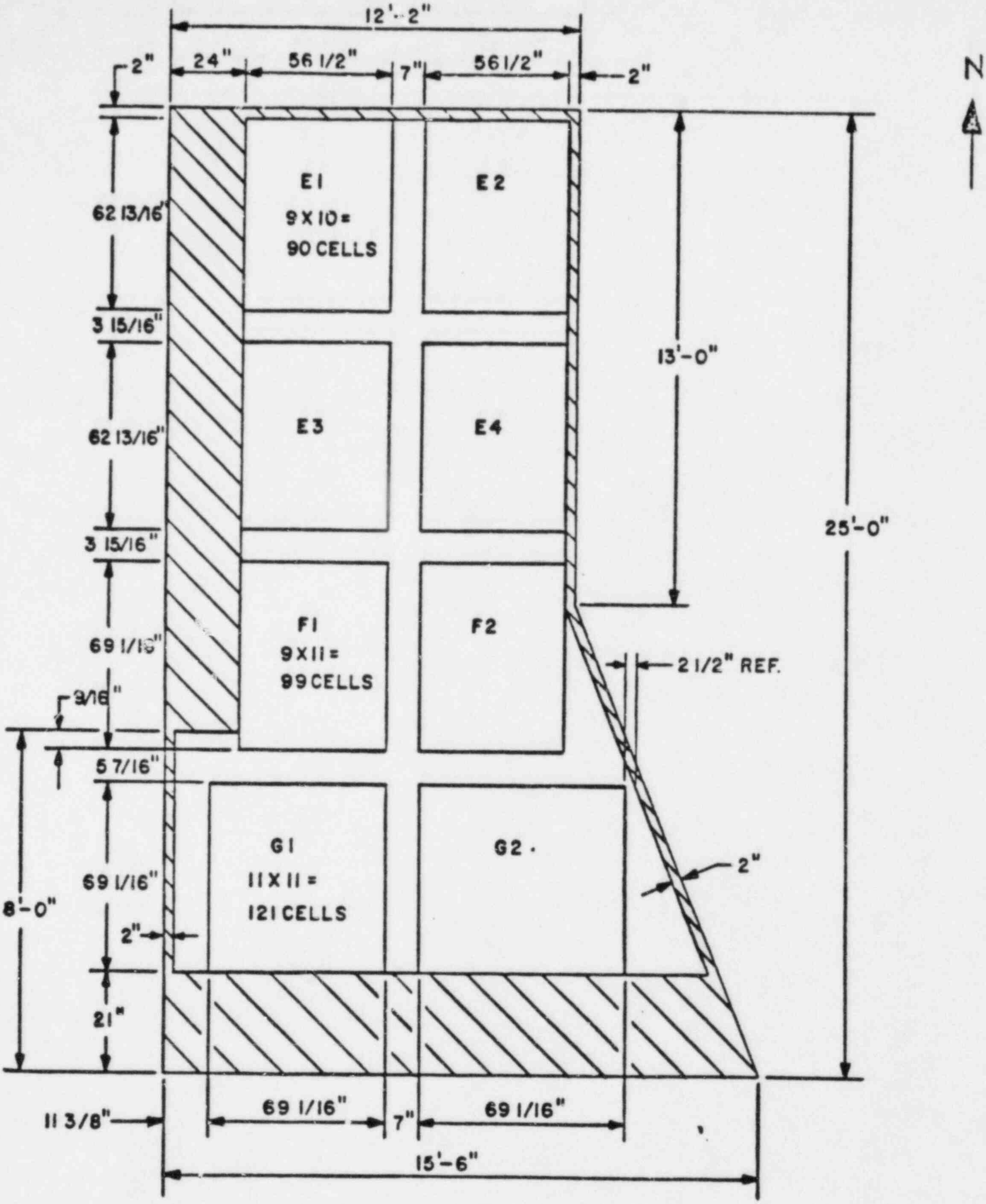
Fuel Handling Area



Fuel Pool Arrangement



RACKS ARRANGEMENT IN SPENT FUEL POOL



RACKS ARRANGEMENT IN CONTAINMENT POOL

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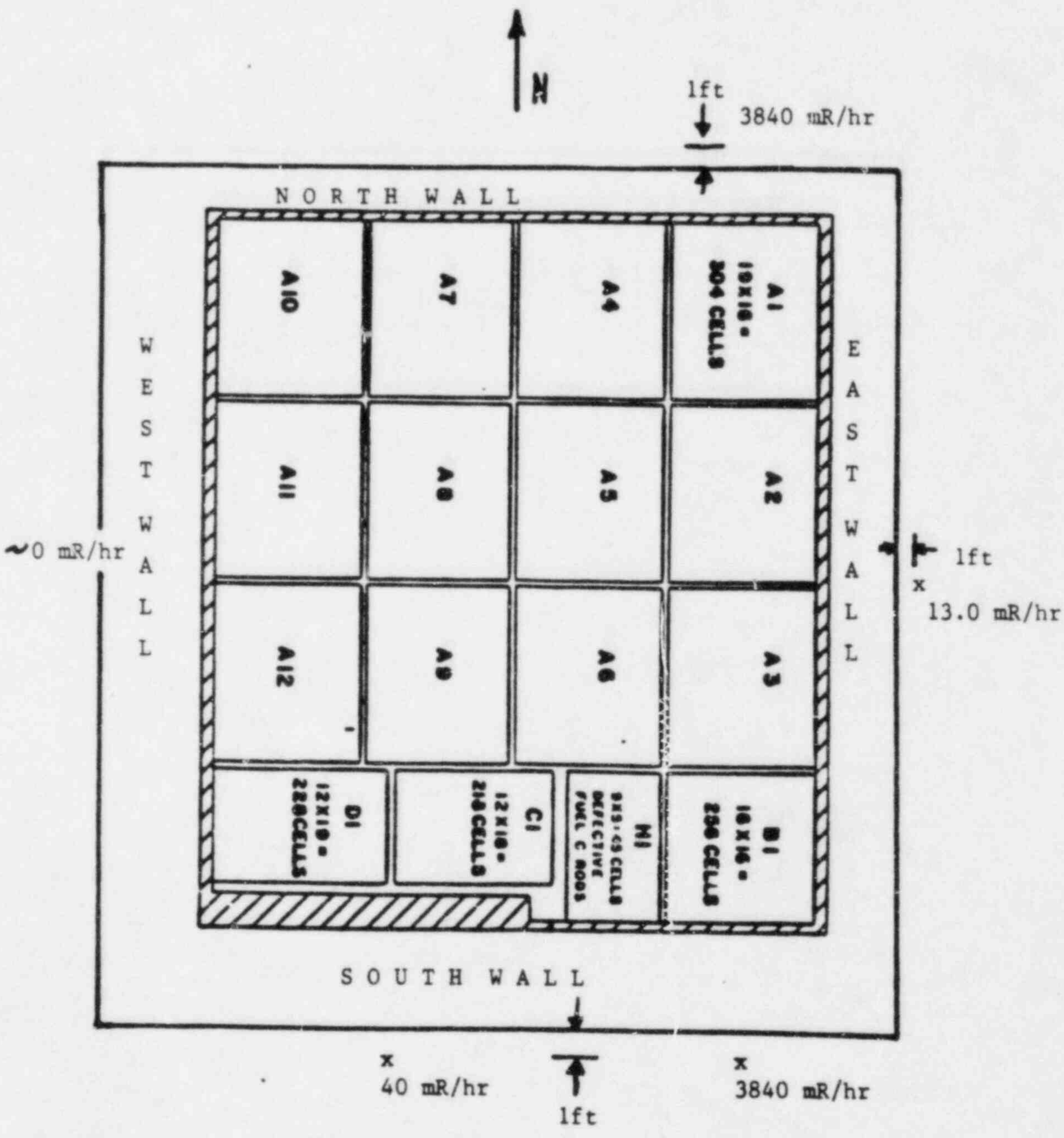
OCCUPATIONAL DOSE RATE CALCULATIONS

Methodology

Results

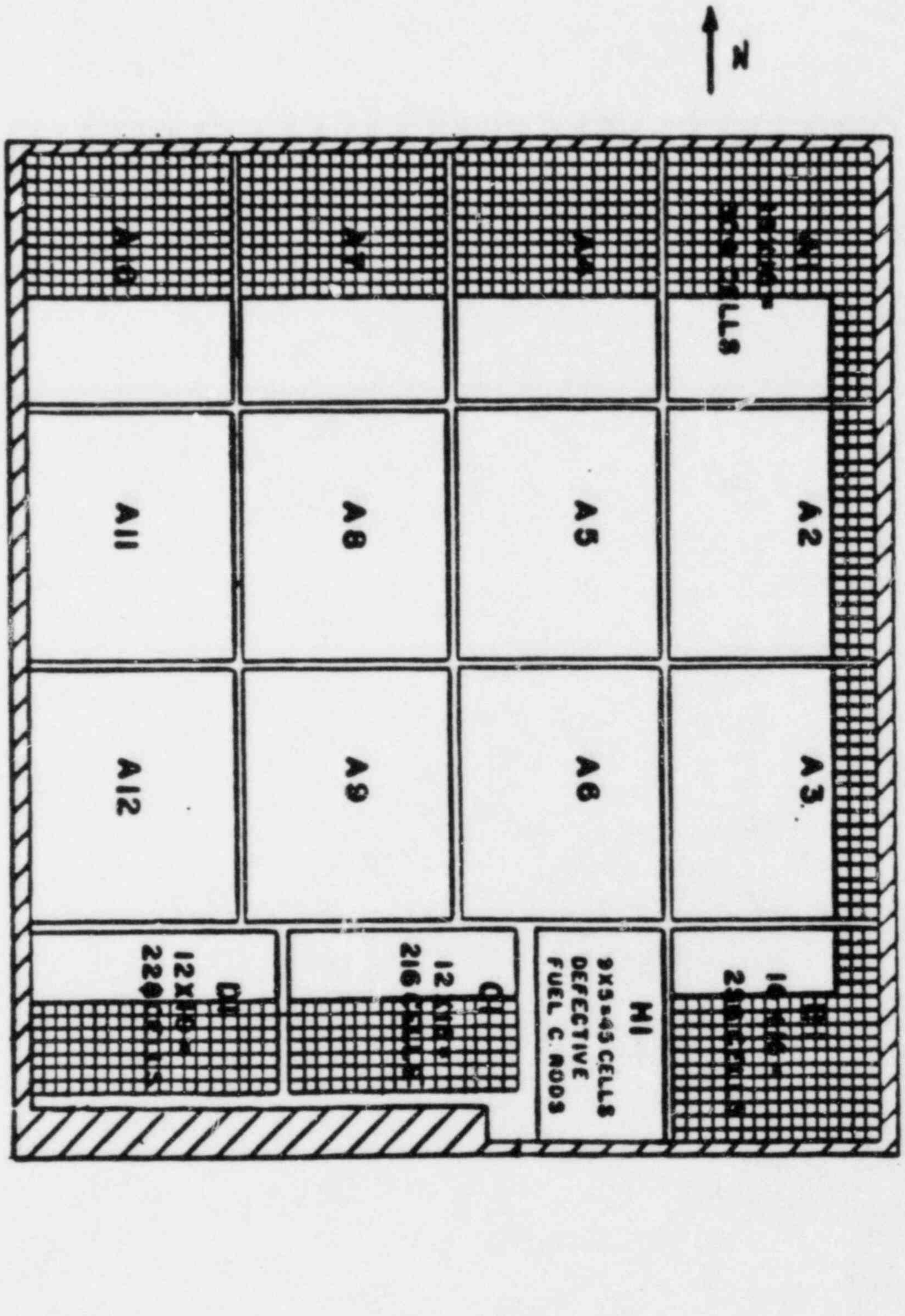
**DOSE RATES AROUND THE SPENT FUEL POOL
WITH FRESHLY DISCHARGED FUEL**



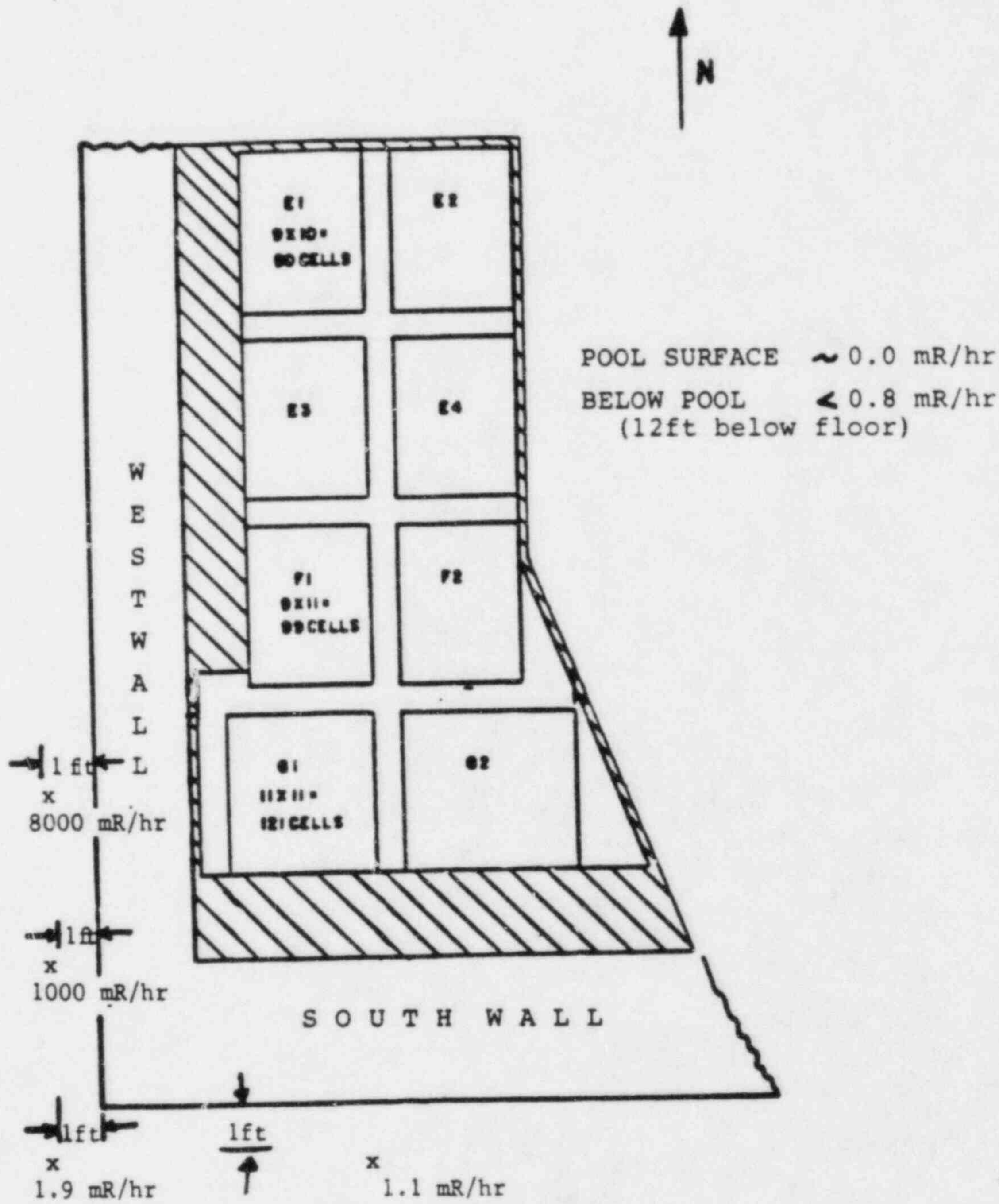
POOL SURFACE ~ 0.0 mR/hr

BELOW POOL ≤1.0 mR/hr
(1ft below floor)

PROHIBITED LOCATIONS IN THE SPENT FUEL POOL
FOR LOADING FRESHLY DISCHARGED FUEL



DOSE RATES AROUND THE UPPER CONTAINMENT POOL WITH A FULL CORE DISCHARGE



NEW CALCULATIONS

* UCP capacity 800 assemblies

* Assumptions

1. Assemblies irradiated to full power equilibrium fission product inventory (> 1000 hours)
2. Assemblies have "cooled" for 42 hours
3. The UCP is fully loaded
4. Shielding by uranium in the assemblies is included.
5. Source strength same as Bechtel.
6. Each assembly is treated as a set of uniform volumetrically weighted point sources

* Data

1. Reactor Power = 3833 MWt
2. Gamma Source Strength taken from Reactor Handbook, Vol III.
3. Pool Geometry - Plant drawing, C-1095 Unit 1 Containment
4. Rack Geometry - Plant drawing, D7371, Rev 5, Layout for Containment and Spent Fuel Pools.

* Model

1. Geometry is as shown
2. QADMOD-G code used.

* QADMOD-G

- Calculates γ-ray fluxes, dose rates or heating rates at discrete locations within a complex source-geometry configuration
- method
  - represent a volume-distributed source of radiation by a number of point isotopic sources
  - compute the line-of-sight distances traversed through each region from the point source to the desired receiver point
  - energy-dependent exponential attenuation factors and energy-dependent buildup factors are applied for each material region traversed.
  - the gamma-ray dose with buildup represents the direct dose plus the scattered dose.
  - the gamma-ray energy spectrum is input
- material description: energy-dependent attenuation coefficients

$$\mu_i(E_k) = \sum_{j=1}^{J} \rho_{ji}\,\sigma_j(E_k)$$

where ρ_{ji} = the density of the j^{th} material in the i^{th} composition (gm/cm^3)

$\sigma_j(E_k)$ = the attenuation coefficient of the j^{th} material for gamma rays in the k^{th} energy interval.

- Source Description
  - the radiation source is described by a series of volumes, each containing a point isotropic source with a defined source strength for each source subvolume.
  - option selected computes volume source internally for a uniformly-distributed source within the source volume.
  - the source power for each point source is equal to the fraction of the total source volume that is occupied by the subvolume containing the point source times the total source strength
  - Source spectrum

E-bin (MeV)	$\dot{E}_\gamma(E)$ (Mev/watt·sec)	$\dot{E}'_\gamma(E)$ (Mev/sec·assembly)	$S_v(E)$
0.4	1.100 E9	0.527 E16	0.1005
0.8	6.500 E9	3.114 E16	0.5941
1.3	0.030 E9	0.014 E16	0.0028
1.7	3.000 E9	1.437 E16	0.2741
2.2	0.060 E9	0.029 E16	0.0055
2.5	0.250 E9	0.120 E16	0.0230
2.8	0.000 E9	0.000 E16	0.0000
>2.8	0.000 E9	0.000 E16	0.0000
TOTAL	10.940 E9	5.242 E16	1.0000

where $\dot{E}'_\gamma(E) = \dot{E}_\gamma(E) \cdot 3833 \times 10^6 \text{ watts} / 800 \text{ assemblies}$

$$A_0 = \text{Source Strength} = \sum_i \dot{E}'_\gamma(E_i)$$

$$S_v(E_i) = \frac{\dot{E}_\gamma(E_i)}{A_0}$$

Gamma-Ray source

For each source energy group, i, the uncollided gamma-ray dose rate is computed as a point kernel numerically integrated throughout the source region:

$$D_{u\gamma_i} = \sum_{\substack{\ell=1\\ m=1\\ n=1}}^{L,M,N} \frac{E_i K_i S_{\ell,m,n}}{4\pi\rho^2_{S_{\ell,m,n}}} e^{-x_i} \tag{18}$$

where

$$x_i = \sum_{m_o=1}^{M_o} \sum_{h=1}^{H} \rho_h \theta_{m_o h} \Sigma_{m_o i} \tag{19}$$

for each (ℓ,m,n). The terms in Eqs. (18) and (19) have the definitions:

$D_{u\gamma_i}$ = detector response without buildup

K_i = a suitable conversion factor

E_i = source strength of the ith energy group

$S_{\ell,m,n}$ = the volume weighted power associated with the (ℓ,m,n) source point

$\rho_{S_{\ell,m,n}}$ = the distance from the (ℓ,m,n) source point to the detector point

ρ_h = the path length through the hth composition along the source-detector line

$\Sigma_{m_o i}$ = the gamma attenuation coefficient of the m_oth material for the ith energy group

$\theta_{m_o h}$ = the fractional density or absolute density of material m_o in composition h depending upon the units of $\Sigma_{m_o i}$.

- Buildup Factor

$$B_i = B_0 + B_1 x_i + B_2 x_i^2 + B_3 x_i^3$$

where

$B_i \equiv$ the buildup factor for the i^{th} energy group at x_i attenuation mean free paths

B_0, B_1, B_2, B_3 = energy-dependent coefficients

– used internal coefficients

- The total gamma ray dose from the i^{th} energy group is

$$D_{\gamma i} = B_i \cdot D_{4\gamma i}$$

so that the total gamma ray dose is

$$D_\gamma = \sum_i D_{\gamma i}$$

CONSERVATISMS:

- SOURCE SPECTRUM - 42 HOURS DECAY
  FASTEST POSSIBLE 120 HOURS ~10%

- RADIATION FIELD - 8000 mR/hr UNIFORM
  ACTUALLY VARIABLE ~40%

- CONTAINMENT WALL - FLAT
  ACTUALLY CURVED ~20%

- DIRECT DOSE - SAME AS AT DP1 ~ 5%

- TOTAL ESTIMATED CONSERVATISM

$$(0.9)(0.60)(0.80)(0.95) = 0.27$$

FIGURE 2-1
GEOMETRY OF THE UPPER CONTAINMENT POOL
X,Y DIMENSIONS

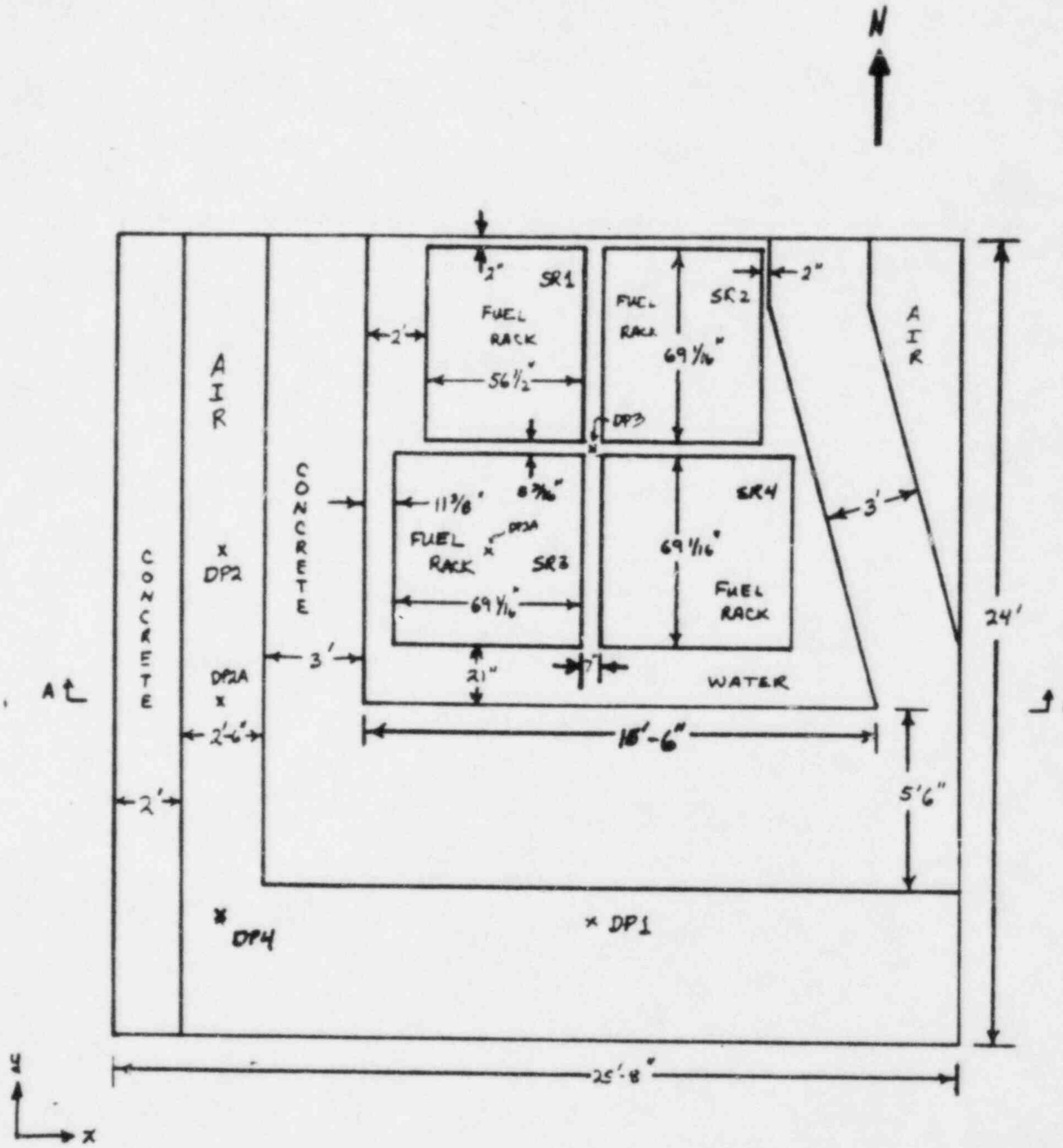


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FIGURE 2-2
GEOMETRY OF THE UPPER CONTAINMENT POOL
X,Z DIMENSIONS

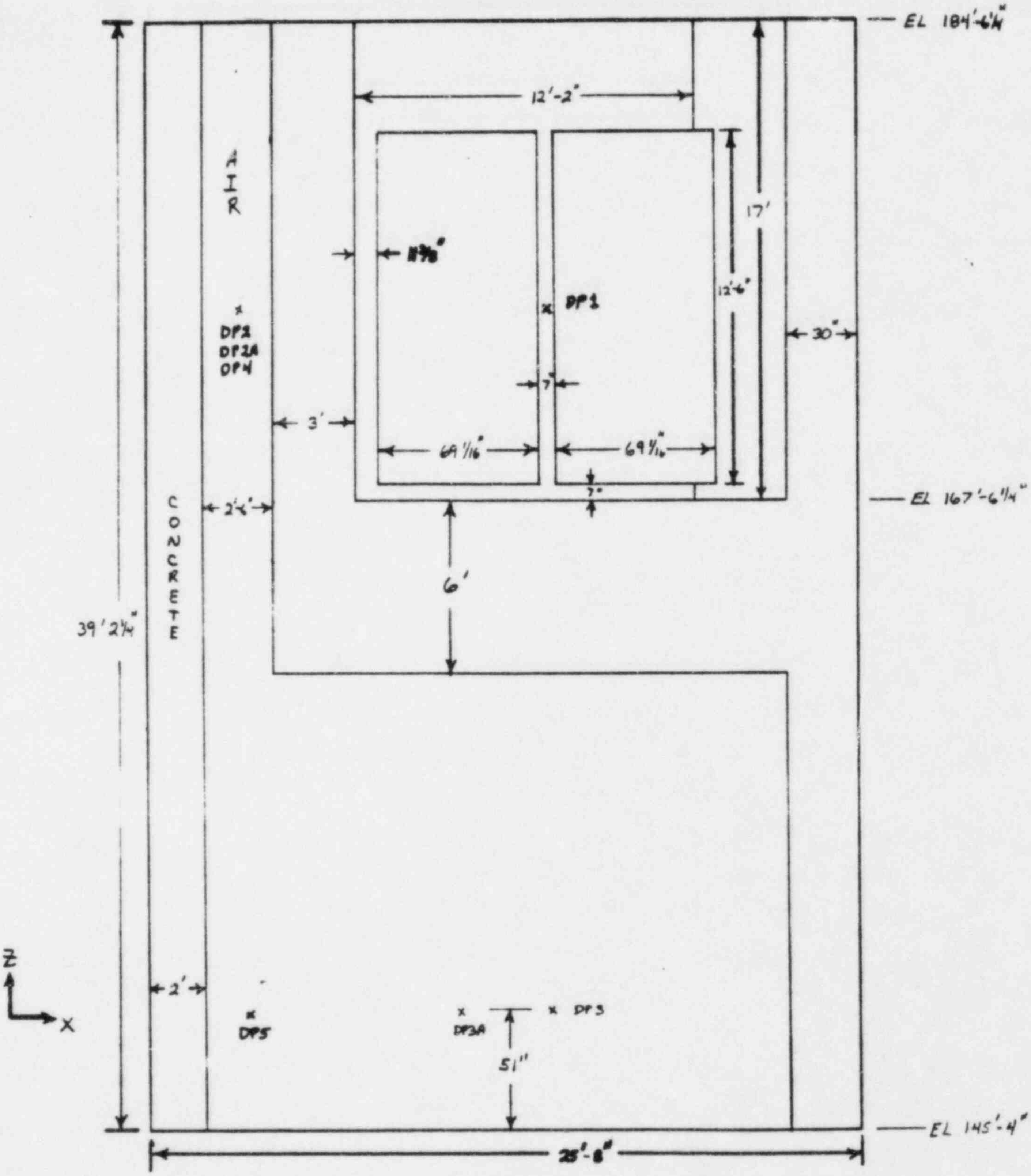

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FIGURE 2-1
GEOMETRY OF THE SPENT FUEL POOL
X,Y DIMENSIONS

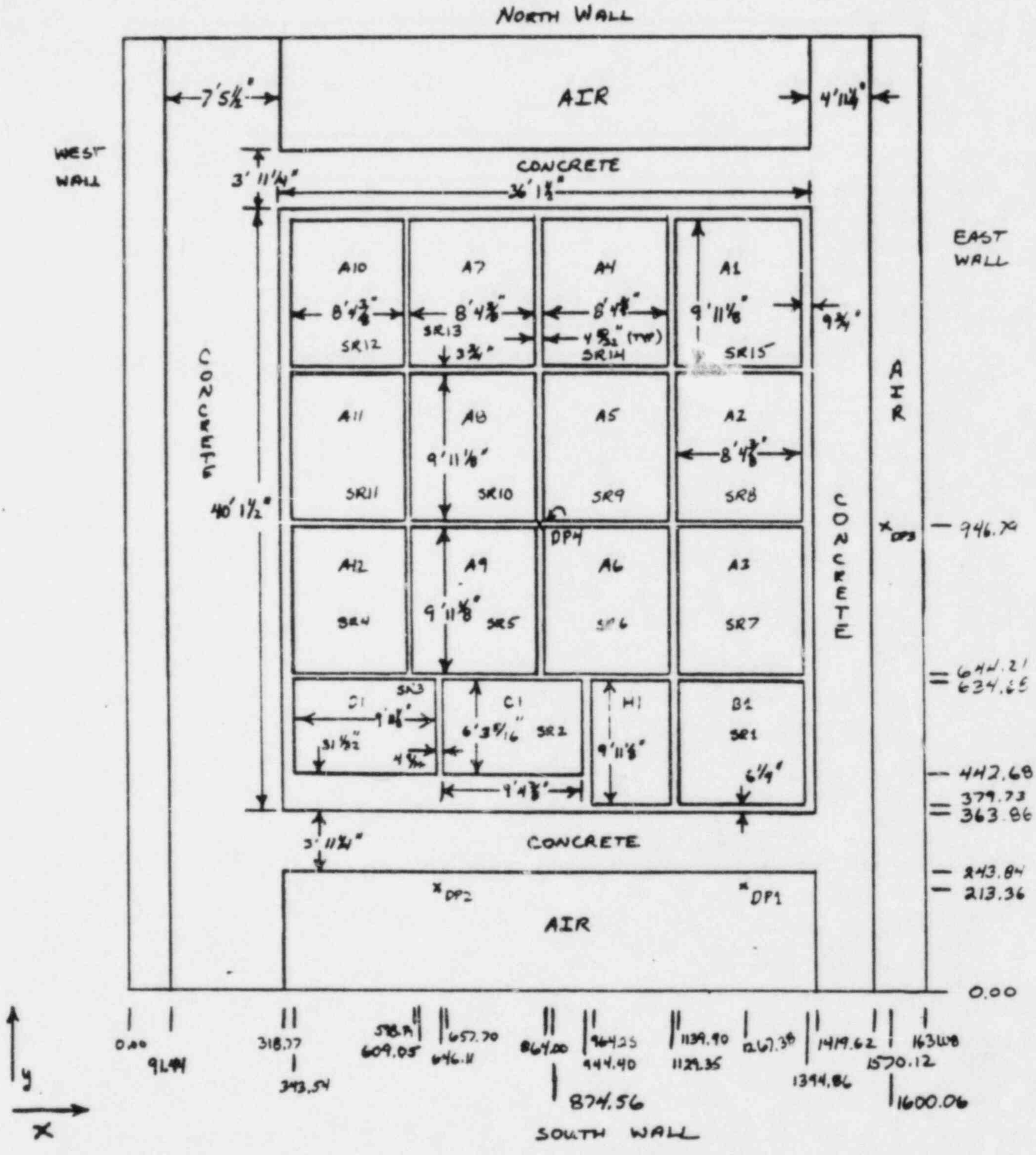

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FIGURE 2-2
GEOMETRY OF THE SPENT FUEL POOL
X,Z DIMENSIONS