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THERMAL-HYDRAULIC REPORT

SPENT FUEL STORAGE RACKS
INDIAN POINT UNIT NO. 3

8721-00-0104

PREPARED FOR

NEW YORK POWER AUTHORITY
123 MAIN STREET
WHITE PLAINS, N.Y. 10601
SPECIFICATION NO. FHS-01
REVISION NO. 00

Not A Design Change
REVISION 1
Jan. 1988

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2.0 Abstract

The thermal-hydraulic analysis for the new high density U.S. Tool & Die racks for Indian Point Unit No. 3 considers: (1) the decay heat for the spent fuel pool and the resulting pool temperatures, (2) the spent fuel pool heat-up rate and time until pool boiling following a loss of spent fuel pool cooling, (3) the natural circulation cooling of spent fuel and (4) the gamma heating and natural circulation cooling in the fuel assembly water boxes.

3.0 Introduction

3.1 The scope of the analysis covers Technical Specification (Section 8) and revisions of the New York Power Authority Specification FHS-01, Rev. 0 and includes the following:

a. Computation of decay heat loads for the spent fuel pool in accordance with the NRC's SRP 9.1.3 and SRP 9.2.5, Branch Technical Position ASB 9-2.

b. Temperature changes and heat-up rates for the loss of spent fuel pool cooling accident under normal refueling and full core off-load conditions.

c. Recirculation flow characteristics in the hottest and average stored spent fuel assemblies to determine local coolant temperature changes and peak clad temperatures. A local path (path 1) and an under-rack path (path 2) are examined.

d. Investigation of gamma heating in the water boxes of Region 1 containing a fuel assembly to assure adequate coolant flow exists and determination of the temperature distributions in the water box , and poison wall.

3.2 Methods and Assumptions used in the Thermal-Hydraulic Analysis

The methods used for analyzing the thermal-hydraulic aspects of the spent fuel pool involve relatively uncomplicated correlations for friction factors, loss coefficients, and heat transfer coefficients that make a detailed computer analysis unnecessary. Further simplifying but conservative assumptions reduce the mathematical complexity to the point where hand calculations or programmable calculators are all that are required.

3.3 The design criteria used for the thermal-hydraulic analysis of the spent fuel pool for Indian Point Unit 3 are in accordance with the NRC "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications", issued April 14, 1987. Additional conditions are given by Specification FHS-01, Rev. 0 for the Spent Fuel Storage Racks, NYPA's August 25, 1987 letter from C. Tessmer to J. Rhoden and NYPA's November 2, 1987 letter from C. Tessmer to F.E. Witsch.

Based on the NRC Position Paper, the specifications and the two letters, the following are established as design bases or requirements:

a. Decay heat loads for a pool are to be determined in accordance with the NRC Branch Technical Position ASB 9-2, "Residual Decay Energy for Light Water Reactors for Long-Term Cooling", Section 9.2.5 of the Standard Review Plan. Pool decay heat loads and temperatures are to be computed for the following six cases:

CASE	POOL FUEL LOAD	T^{MAX}_{POOL}, °F	TIME AFTER REACTOR SHUTDOWN FOR REFUELING BATCH OR FULL CORE ADDITION TO THE POOL
			SPECIFICATION
1	Regular Fuel Normal Refueling	140 150	120 hours 120 hours
2	Regular Fuel Core Off-Load	200	162 hours
3	Maximum Consolidated Normal Refueling	140 150	120 hours 120 hours
4	Maximum Consolidated Core Off-Load	200	162 hours

b. To ensure that adequate time exists for an alternate cooling method to be implemented in the event of a loss of spent fuel pool cooling capability accident, the heat-up rate is calculated and the time required for pool boiling to occur is determined.

c. Coolant flow rates, temperature increases, and peak clad temperatures are to be determined for worst case conditions (i.e. high pressure drops and low heat transfer conditions for fuel assemblies, high bundle decay heat, etc) to verify that boiling shall not occur.

d. The effect of gamma heating in the cell box and inter-cell spaces between fuel assemblies of Region 1 is to be analyzed to assure that gamma heating shall not cause boiling in these positions. Adequate flow must be established to preclude the possibility of trapping air or steam anywhere in the fuel racks.

e. Coolant flow paths and sparger locations affecting the analysis shall be identified.

3.4 As noted in the design bases, conservative assumptions are employed for evaluations of all coolant and clad temperatures. Some additional assumptions used for the thermal and hydraulic analysis of the spent fuel pool are as follows:

a. The thermal inertias of the concrete walls and the coolant and piping outside the pool boundaries are neglected in the transient heat-up analysis.

b. The pool surface is not assumed to mix to a lower pool bulk temperature in the heat-up analysis following the loss of spent fuel pool cooling accident. The time calculated to boil is based on the pool maximum bulk temperature and not a lower pool average temperature.

c. All decay energy is assumed to be absorbed in the fuel and surrounding coolant for the hot assembly natural circulation analysis. (In reality, some gamma radiation will be absorbed in the adjacent cell boxes and poison).

d. The gamma decay heat absorbed in the Region 1 cell box wall is taken to be proportional to the mass densities of the materials in the spent fuel pool. (In reality, most of the gamma radiation never leaves the fuel assembly due to strong uranium attenuation.) Gamma heating proportional to the mass fraction is roughly equivalent to the assumption of uniform gamma flux in the repeating unit cell.

e. A circulation flow path (path 2) from the South wall or downcomer to a position along the North wall is assumed for the hottest assembly. This derates the flow to the hottest assembly since flow down the three remaining walls is also possible.

f. The average hot assembly generates 1.84×10^5 BTU/HR based on refueling 120 hours after reactor shutdown. A conservative sinusoidal axial power distribution was assumed where the clad hot spot temperature rise factor is about 1.57 times the average temperature rise.

g. To conservatively predict the flow rates to the hot assembly the dominant pressure drops are over estimated by factors of 1.5 for the fuel assembly pressure loss and 2.0 for the under rack pressure losses.

h. Material properties (e.g. thermal conductivities, densities, and specific heats) are generally assumed to be independent of temperature and are evaluated at some specified (average, inlet, or surface) temperature.

3.5 The major areas of concern in the thermal-hydraulic analysis are to verify that the clad and coolant temperatures do not become high enough to cause boiling. In the event of the loss of spent fuel pool cooling accident, the heat-up rate must be slow enough to allow an alternate coolant system to be connected and operating before pool boiling occurs.

4.0 Detailed Analysis and Results

4.1 In this section, we present an analysis overview for the calculation summaries that follow. Decay heat fractions are computed according to NRC Branch Technical Position ASB-2 standards and are presented in Section 4.2. Total heat loads for the normal refueling and full core off-load conditions for intact fuel storage (Case 1 & 2) and consolidated fuel storage (Cases 3 & 4) are then calculated. The thermal inertia of the spent fuel pool (SFP) is computed in Section 4.3. Heat-up rates and the time taken for the pool water to reach 212°F following a loss of spent fuel pool cooling accident are found. Make-up rates at pool boiling are also determined in this section.

Section 4.4 contains the natural circulation cooling analyses. A local recirculation path (path 1) and a more complete under-rack path (path 2) are considered. Clad and coolant temperature distributions are determined for these worst case analyses. In Section 4.5 Region 1 gamma heating of the cell box walls and poison "Slabs" adjacent to the hottest assembly is investigated. The temperature distributions in the stainless steel cell wall, poison cell box interface are determined in Section 4.6.

4.2 Decay Heat Loads for the Spent Fuel Pool

The NRC Branch Technical Position ASB 9-2 is used to compute the decay heat fractions for the Indian Point Unit 3 spent fuel pool (SFP). For cooling times greater than 10^7 sec. (116 days), ASB 9-2 does not specify a fission product decay uncertainty factor, but SRP Section 9.1.3 recommends a value of 0.1 for times $> 10^7$ secs. and is used here.

Based on ASB 9-2 and the six cases outlined in the design bases, the decay heat fractions and the SFP decay heat

loads are as follows:

CASE	T^{MAX}_{POOL}, °F	HOURS AFTER REACTOR SHUTDOWN*	NO. OF FA'S	TOTAL, BTU/HR
1	140	324	1152*	13.99×10^6
	150	145	1152*	17.48×10^6
2	200	267	1345	35.00×10^6
3	140	648	2304*	13.99×10^6
	150	246	2304*	17.48×10^6
4	200	341	2497	35.00×10^6

*REQUIRED TO MAINTAIN 140°F, 150°F and 200°F POOL TEMPERATURES

**ROOM NEEDED FOR FULL CORE OFF-LOAD (193 cells)

The racks will provide storage for 1345 fuel assemblies. Allowing for the full core off-load capacity, 15 refueling discharges (or 25 years of spent fuel) can be safely accommodated before spent fuel relocation becomes necessary.

As applied here, a 25% margin of conservatism is expected through the use of ASB 9-2 and the pool histories used. This margin includes for uncertainties of approximately 10% (for curve fitting), 5% (for fission yields), and 10% (for PU^{239} uncertainties that are likely to introduce additional credits). Plus instant refueling is assumed as against some refueling rate which would lower the maximum thermal load in the pool.

4.3 Pool Thermal Inertia and Heat-Up Rates

In this analysis, the spent fuel pool (SFP) heat-up rate and time until pool boiling following the loss of spent fuel pool cooling accident is considered. The SFP thermal inertia for Indian Point Unit 3 is found. A full pool of consolidated fuel is assumed. For the initial license application intact fuel storage would only be used thus this approach is conservative. For simplicity (and also conservatism) we take cp = 60 BTU/ft^3/°F for water. Materials that possess some thermal inertia that are neglected included primary and secondary water piping exterior to the

liner plates, and the concrete walls of the spent fuel pool as well as the pool water fuel transfer canal. The total SFP thermal inertia under these simplifying but conservative assumptions are:

$$Ip = 2.39 \times 10^6 \text{ BTU}/^{o}\text{F}$$

In the heat-up analysis, the initial pool temperature is taken to be T_{HIN} - the inlet HX temperature. Pool mixing to a lower pool temperature is neglected. Following the loss of SFP cooling, the pool heat-up rates, times to reach 212°F, boil-off rates, and water level changes at boiling are determined. These are tabulated below:

TABLE 4.3

RESULTS OF HEAT-UP ANALYSIS FOLLOWING THE LOSS OF SPENT FUEL POOL COOLING

INDIAN POINT - UNIT 3

CASE	T^{MAX}_{POOL},°F	$\dot{q}$ BTU/HR	$\Delta T'$ °F/HR	(HRS) t to 212°F	$\dot{M}$ LB/HR	Q GPM	$\dot{h}$ FT/HR
1	140	13.99×10^6	5.84	12.3	14,422	29.9	.25
	150	17.48×10^6	7.30	10.6	18,020	37.4	.31
2	200	35.00×10^6	14.6	0.82	36,082	74.9	.63
3	140	13.99×10^6	5.84	12.3	14,422	29.9	.25
	150	17.48×10^6	7.30	10.6	18,020	37.4	.31
4	200	35.00×10^6	14.6	0.82	36,082	74.9	.63

DEFINITIONS:

$\dot{q}$ - TOTAL SFP DECAY HEAT LOAD, BTU/HR

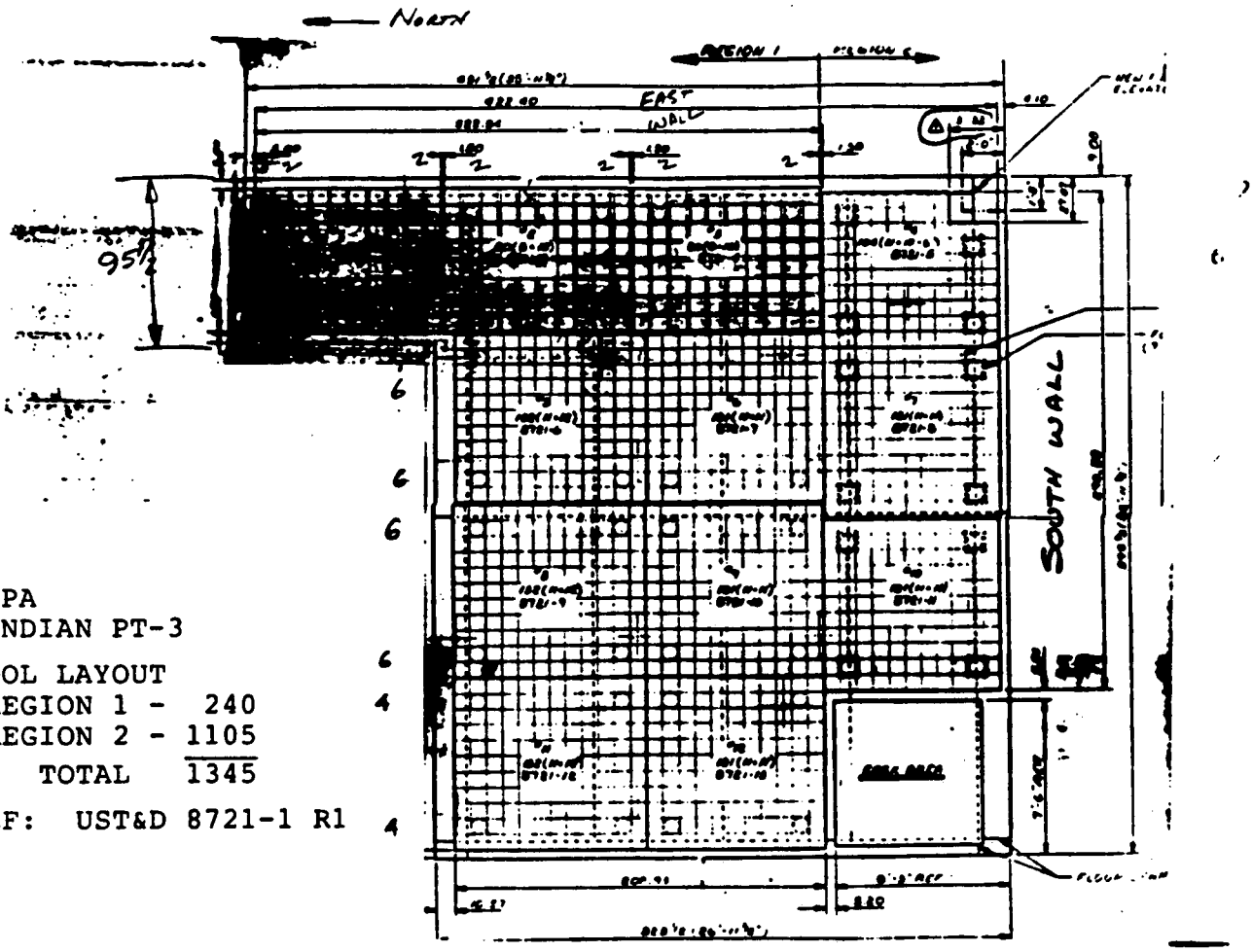
$\Delta T'$ - HEAT-UP RATE FOLLOWING LOSS OF SPENT FUEL POOL COOLING, °F/HR

t - TIME TO REACH 212°F FROM T_{HIN}, HRS

$\dot{M}$ - BOIL-OFF RATE @ 212°F, 14.7 psia, LB/HR

Q - MAKE-UP RATE (@ 60 LB/FT^3 WATER), GPM

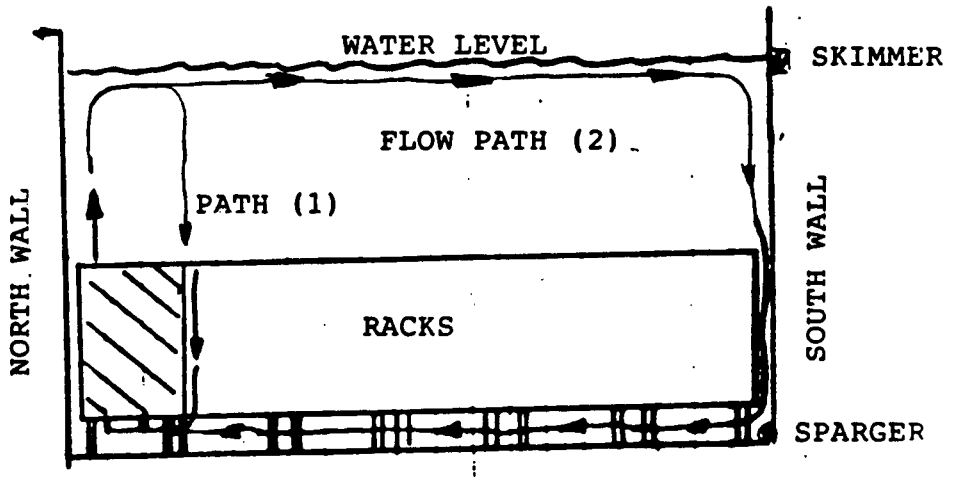
$\dot{h}$ - POOL HEIGHT REDUCTION RATE, FT/HR



NYPA
INDIAN PT-3

POOL LAYOUT
REGION 1 - 240
REGION 2 - 1105
TOTAL 1345

REF: UST&D 8721-1 R1



ELEVATION VIEW

LOCATION OF HOT FUEL ASSEMBLIES

5. For the local path (path 1) the fuel assembly inlet temperature is taken to be the hottest pool bulk temperature. For path (path 2) the inlet temperature is the average for the pool.

6. For the under-rack flow path (path 2), the assumed losses include:

   a. South wall friction losses

   b. 90° turn losses at South wall pool floor interaction

   c. Under-rack friction losses

   d. Expansion and contraction losses at the rack supports

   e. Inlet to rack cell losses

   f. Fuel bundle losses

   g. Branching - Momentum losses

   Flow branching/momentum losses are typically small (and recoverable) in comparison to the total losses and are neglected.

7. The derived driving pressure (caused by water density variations) is given by

$$\Delta P_d = \beta \frac{qL}{2C_pQ} \left(\frac{g}{g_c}\right)$$

where $\beta = -\frac{1}{\rho} \left.\frac{\partial \rho}{\partial T}\right|_p \left(\frac{1}{°F}\right)$ is the thermal coefficient of expansion for the water- only a mild function of pressure.

L is the heated (active) length (12 ft) and q is the decay heat rate. Fluid properties are evaluated at 125°F, a low temperature for the core off-load case. A 125°F, $\beta = 2.6 \times 10^{-4}$ °F^{-1} and at 150°F, $\beta = 3.1 \times 10^{-4}$ °F^{-1}. The driving pressure is then derated by approximately 20%.

A summary of significant results that apply to all six cases (for the purpose of determining peak coolant and clad temperatures) are as follows. These results bracket the temperatures to be expected in the pool.

TABLE 4.4-1

RESULTS RELATED TO PEAK COOLANT AND CLAD TEMPERATURES

		Path 1	Path 2
Volume flow rate in the hottest assembly	ft^3/sec	.0362	.0318
	(GPM)	16.2	14.2
Total pressure drop = ΔP_d - $\frac{lbf}{ft^2}$		2.20	2.50
Coolant ΔT - °F		22.9	26.0
Position of clad hot spot - ft		7.75	7.16
Difference ($T_{cladmax}$ - T_{IN}) - °F		37.2	38.8

Following assumption 5, the peak clad and coolant temperatures $T_{cladmax}$ and T_{OUT} are as tabulated below:

TABLE 4.4-2

PEAK CLAD AND COOLANT TEMPERATURES

Cases	T_{POOL}^{MAX}, °F	Path 1		Path 2	
		T_{OUT} (°F)	$T_{cladmax}$ (°F)	T_{OUT} (°F)	$T_{cladmax}$ (°F)
1	140	162.9	177.2	159.6	172.5
	150	172.9	187.2	168.1	180.9
2	200	222.9	237.2	210.1	222.9
3	140	162.9	177.2	159.6	172.5
	150	172.9	187.2	168.1	180.9
4	200	222.9	237.2	210.1	222.9

NOTE: T_{sat} = 241.8°F at top of racks (~26 psia)

T_{sat} = 245°F at peak clad T (~27 psia)

All cases are significantly subcooled and void fractions are negligible.

Since the actual flow paths will be complicated combinations of local and under-rack paths, the temperatures will not exceed those indicated. The indicated temperatures are much lower than the 700-800°F which is the lower bound for low temperature sensitization of austenitic stainless steel.

4.5 Gamma Heating of the Region 1 Fuel Box Walls, Poison & Water Box

In this analysis, gamma heating of the Region 1 fuel box walls, poison and water box is investigated.

Fission product decay accounts for virtually all residual heat in the spent fuel pool with minimum cooling times $t_s \cong 5$ days. At this time, a realistic but upper limit on the gamma fraction is 0.62, based on reference P1, the primary reference for the NRC position standard ASB 9-2, Reference N1.

A typical 1 MeV electron will travel approximately 0.016" in the UO_2 fuel. Thus, all β^- electrons will be stopped in the fuel or surrounding clad and coolant. A typical, but higher than average, 1 MeV gamma ray has a mean free path of approximately 0.56" in the UO_2 fuel - comparable to the pellet diameter .373". Therefore, the fuel will not stop all the gamma radiation emitted by the decaying fission products. It will then be conservative to assume the following when estimating the energy deposition in the fuel box, poison, and water box:

1. The fuel box is located within an infinite array of hottest assemblies - each generating 1.84×10^5 BTU/hr, 62% of which is gamma. This is based on decay heat level 120 Hours After Reactor Shutdown. Δ
2. The gamma energy absorbed in a unit cell comprised of one fuel assembly, one fuel box, and four Poison "Slabs" is proportional to a given material's mass fraction. This is roughly equivalent to the assumption of uniform γ-flux since ν/ρ is approximately constant for all materials at a given gamma energy.

3. Water in the water box must remove gamma heat due to energy depositions in the water box wall, and the water box itself.

   The flow rate in the water box is determined by equating the driving head to the loss head. The coolant temperature corresponding to this flow rate is then computed and is less than the boiling temperature of the coolant.

4.6 Poison Temperatures and Stainless Steel Temperatures

In this calculation, Region 1 temperature distributions in the poison are determined. Heat source terms (due to gamma irradiation) follow the same conservative assumptions used in Section 4.5. The maximum temperature of the poison is seen to be less than the boiling temperature of the coolant hence no boiling will occur in this region. Region 2 temperatures will be lower than Region 1 because of lower power fuel assemblies.

The temperature gradient in the stainless steel across the fuel box interface is modeled using a steady-state, 1-D, heat conduction equation. The maximum temperature difference is limited to less than 2.5°F and is of a magnitude which should produce negligible thermal stresses in the stainless steel walls of the rack for both Region 1 and Region 2.

5.0 Conclusions

The detailed thermal-hydraulic analyses described in Section 4 address the concerns, intent and design bases of the NRC's Position Paper "OT Position for Review and Acceptance of Spent Fuel Storage and Handling Applications" and the New York Power Authority Specification FHS-01, Rev. 0.

Based on these analyses, our professional staff has concluded that the spent fuel pool for Indian Point Unit 3 can be adequately cooled in accordance with the suggested regulatory standards of the Nuclear Regulatory Commission and comply with specifications outlined by New York Power Authority for spent fuel racks described in this report and loaded with spent fuel as indicated in this report.

6.0 Appendices

This section contains the four detailed thermal-hydraulic analyses attached as appendices. They are as follows:

Calc. 8721-01 --- "Decay Heat Loads for the Spent Fuel Pool"

Calc. 8721-02 --- "Pool Thermal Inertia and Heat-Up Rates"

Calc. 8721-03 --- "Natural Circulation Cooling of the Spent Fuel"

Calc. 8721-04 --- "Gamma Heating of the Region 1 Fuel Box Walls, Poison "Slabs", Water Box".

APPENDIX 8721-01

DECAY HEAT LOADS FOR THE SPENT FUEL POOL

IN THIS SECTION WE DETERMINE THE MAXIMUM EXPECTED DECAY HEAT LOAD FOR THE INDIAN POINT-3 PLANT SPENT FUEL POOL. NORMAL REFUELING AND FULL CORE OFF-LOAD CONDITIONS FOR BOTH REGULAR FUEL ASSEMBLIES AND CONSOLIDATED FUEL ASSEMBLIES WILL BE CONSIDERED ACCORDING TO THE METHOD OUTLINED IN REFERENCE (N1). A COMPUTER PROGRAM (PAGES 3 AND 4) IS USED TO CALCULATE THE DECAY HEAT FOR EACH REFUELING. VERIFICATION OF THE PROGRAM CALCULATION IS SHOWN ON PAGE 2. A COMPARISON OF TOTAL POWER OUTPUT FOR 10 BATCHES OF FUEL SHOWS AGREEMENT WITH REF (N1) COMPUTATIONS TO WITHIN 0.22% WHICH IS MORE THAN ADEQUATE.

THE POOL HISTORIES FOR THE FOUR CASES ARE OUTLINED AS FOLLOWS:

TEST CASE FOR PROGRAM VERIFICATION
EDITED SEPT 1, 1987
DATE(MON/DAY/YEAR): 9/ 1/87
TIME(HOUR:MIN:SEC): 14:16: 5

INITIAL POWER = .100E 01 (BTU/HR/FA)
100% OPERATING DAYS =1.000 TIMES RESIDENT TIME

BATCH #	#F/A	TO (DAYS)	TS (DAYS)	FF (FD/FO)	HE	Q (BTU/HR)
1	1	666.66	1.00	.4832E-02	.1139E-02	.5971E-02
2	1	666.66	2.00	.4038E-02	.8486E-03	.4886E-02
3	1	666.66	5.00	.3027E-02	.3506E-03	.3378E-02
4	1	666.66	10.00	.2447E-02	.8036E-04	.2528E-02
5	1	666.66	20.00	.1911E-02	.4222E-05	.1915E-02
6	1	666.66	25.00	.1724E-02	.9677E-06	.1725E-02
7	1	666.66	50.00	.1179E-02	.6122E-09	.1179E-02

TOTAL DECAY HEAT = .0000E+6 BTU/HR
TOTAL NO. OF F/AS STORED = 7

BATCH	FORTRAN PROGRAM, Q (BTU/HR)	ASB 9-2 FIGURE 9.2.5-11,12,13 (BTU/HR)	% DIFFERENCE
1	5.971×10^{-3}	5.9×10^{-3}	+ 1.2
2	4.886×10^{-3}	4.8×10^{-3}	+ 1.7
3	3.378×10^{-3}	3.4×10^{-3}	- .7
4	2.528×10^{-3}	2.65×10^{-3}	- 5.0
5	1.915×10^{-3}	1.95×10^{-3}	- 2.1
6	1.725×10^{-3}	1.73×10^{-3}	- .3
7	1.179×10^{-3}	1.20×10^{-3}	- 1.8
TOTAL	21.582×10^{-3}	21.63×10^{-3}	- .22

```
C DECAY HEAT PROGRAM FOR SPENT FUEL RACKS
C EDITED 26 MARCH 1986
      REAL ITS,ITO
      DIMENSION AN(11),SAN(11),ITO(40),ITS(40),NFA(40),ITITLE(41)
      DATA AN/ 0 558,1 65,3 10,3 87,2.33,1 29,0 462,0 328,
     . 0 170,0 0865,0 114/
      DATA SAN/1 772,0 5774,0 06743,0 006214,0 0004739,4 81E-5,
     1  5 344E-6,5 716E-7,1 036E-7,2 959E-8,7 585E-10/

      DELETE "FOR02 DA"
      CALL FOPEN(2,"FOR02 DA")
      CALL FOPEN(6,"CONS DA")
201   FORMAT(S80)
203   FORMAT(' 'S80)
      DO 1500 J=1,3
      READ(6,201) ITITLE(1)
1500  WRITE(2,203) ITITLE(1)
C  GET TIME AND DATE
      CALL FGDAY(IMON,IDAY,IYEAR)
      CALL FGTIME(IHOUR,IMIN,ISEC)
      WRITE(2,209) IMON,IDAY,IYEAR,IHOUR,IMIN,ISEC
209   FORMAT(' DATE(MON/DAY/YEAR). 'I2'/'I2'/'I2
     1/' TIME(HOUR MIN SEC)  'I2' 'I2' 'I2/)
      SUMF=0 0
      WRITE(10,202)
202   FORMAT('      ENTER BATCH NO  AND D A R S ')
      READ FREE (11)NBAT,IDARS
      READ FREE(6)FAPOW,REST
      READ FREE(6) (ITO(I),I=1,NBAT)
      READ FREE(6) (ITS(I),I=1,NBAT)
      READ FREE(6)(NFA(I),I=1,NBAT)
      WRITE(2,110)FAPOW,REST
110   FORMAT(1H ,'      INITIAL POWER ='E10 3,' (BTU/HR/FA)',/,
     1 '       100% OPERATING DAYS =',F5 3,' TIMES RESIDENT TIME',/)
      WRITE(2,100)
100   FORMAT(1H ,//////'  BATCH #F/A    TO        TS        FF
     :        HE          Q'/'      #          (DAYS)   (DAYS)
     2  FD/PO                      (BTU/HR)'/'  ************
     :*************************************************)

      TFA=0
      DO 4 JK=1,NBAT
      TFA=TFA+NFA(JK)
      ATS=ITS(JK )*IDARS
      TS=ATS*86400
```

```
      ATO=ITO(JK)
      T0=ATO*86400 *REST
      TSPTO=TS+T0
      SUM1=0 0
      SUM2=0 0
      DO 5 J=1,11
      SUM1=SUM1+AN(J)*EXP(-SAN(J)*TS)
      SUM2=SUM2+AN(J)*EXP(-SAN(J)*TSPTO)
5     CONTINUE
      AK=1 1
      IF(TS LT.1000 )AK=1 2
      FDPO=NFA(JK)*(AK*SUM1-SUM2)/200
      COEF=(1-EXP(-4 91E-4*T0))*(EXP(-4 91E-4*TS))
      PU239=0 00228*0 7*COEF
      COEF2=(1-EXP(-3 41E-6*T0))*EXP(-3 41E-6*TS)
      PN239=0 00217*0 7*(1.007*COEF2-0.007*COEF)
      FDPOHE=NFA(JK)*(PU239+PN239)
      POWER=FAPOW*(FDPO+FDPOHE)
      WRITE(2,101)JK,NFA(JK),ATO,ATS,FDPO,FDPOHE,POWER
101   FORMAT(3X,I2,4X,I3,2X,F7 2,2X,F7 2,3E11.4)
      SUMP=SUMP+POWER
4     CONTINUE
      SUMP=SUMP/1 E+6
      WRITE(2,102)SUMP
102   FORMAT('   ***************************************************
     :********'/'      TOTAL DECAY HEAT =',F9 4,'E+6 BTU/HR')
      WRITE (2,103)TFA
103   FORMAT('      TOTAL NO. OF F/AS STORED =',I5)
      STOP
      END
```

NOTE:

- Q - TOTAL POWER FROM REFUELING, $\frac{BTU}{HR}$
- NBAT - NUMBER REFUELINGS
- DARS - DAYS AFTER REACTOR SHUTDOWN
- FAPOW - FUEL ASSEMBLY POWER, $\frac{BTU}{HR}$
- REST - FRACTION OF RESIDENT TIME FOR FUEL IN REACTOR, 1.0= 100%
- NFA - NUMBER FUEL ASSEMBLIES PER REFUELING
- T_O - TIME FUEL ASSEMBLY OPERATES AT POWER IN REACTOR, DAYS
- T_S - TIME FROM REACTOR SHUTDOWN, DAYS
- FP - FISSION DECAY POWER
- HE - HEAVY ELEMENT DECAY POWER

CASE 1 REGULAR FUEL - NORMAL REFUELING:

ONE REFUELING, 76 ASSEMBLIES, EVERY 20 MONTHS PER REF (S1).

THERE ARE 1345 CELLS AVAILABLE, OF WHICH, 193 ARE RESERVED FOR CORE-OFF-LOAD. THAT LEAVES 1345 - 193 = 1152 AVAILABLE FOR FUEL.

THE NUMBER OF REFUELINGS WOULD BE

$$= \frac{1152}{76} = 15.15 \text{ REFUELINGS}$$

SINCE THE NUMBER OF REFUELINGS IS A FRACTION OVER 15, 12 ASSEMBLIES WILL BE ADDED AT THE FIRST REFUELING TO FILL THE POOL. THE INPUT TO THE COMPUTER PROGRAM IS AS FOLLOWS:

BATCH	NUMBER FUEL ASS'S	OPERATING TIME, DAY	TIME OUT OF REACTOR LESS SHUTDOWN TIME LAST REFUELING, DAYS
1	88	1050	8512
2	76	1050	7904
3	76	1050	7296
4	76	1050	6688
5	76	1050	6080
6	76	1050	5472
7	76	1050	4864
8	76	1050	4256
9	76	1050	3648
10	76	1050	3040
11	76	1050	2432
12	76	1050	1824
13	76	1050	1216
14	76	1050	608
15	76	1050	0

THE TIME AFTER REACTOR SHUTDOWN FOR LAST REFUELING IS 120 HOURS OR 5 DAYS.

CASE 2 REGULAR FUEL – CORE OFF-LOAD

THIS CASE HAS THE SAME POOL HISTORIES AS CASE 1 EXCEPT THE LAST NORMAL REFUELING OCCURS 36 DAYS BEFORE THE CORE OFF LOAD TAKES PLACE. ALSO THE CORE OFF-LOAD IS 162 HRS AFTER REACTOR SHUTDOWN. THE INPUT FOR THIS CASE IS: NOTE: PER REQUIREMENT (REF N1) OPERATING POWER SHOULD BE NO LOWER THAN 16000 EFPH ≡ 666 DAYS FOR ALL BATCH.

BATCH	NUMBER FUEL ASS'YS	OPERATING TIME, DAYS	TIME OUT OF REACTOR LESS LAST REFUELING SHUT DOWN TIME, DAYS	
1	88	1050	8548	
2	76	1050	7940	
3	76	1050	7332	
4	76	1050	6724	
5	76	1050	6116	
6	76	1050	5508	
7	76	1050	4900	
8	76	1050	4292	
9	76	1050	3684	
10	76	1050	3076	
11	76	1050	2468	
12	76	1050	1860	
13	76	1050	1252	
14	76	1050	644	
15	76	1050	36	LAST [illegible]
16	76	1050	0	
17	76	666	0	[illegible]
18	41	666	0	

CASE 3 CONSOLIDATED FUEL - NORMAL REFUELING

FOR CONSERVATISM IT WILL BE ASSUMED THAT THE POOL CAN BE COMPLETELY FILLED WITH CONSOLIDATED FUEL LESS ROOM FOR ONE CORE OFF-LOAD. ACTUALLY, THE LAST TWO TO THREE YEARS WORTH OF FUEL WILL NOT BE AVAILABLE FOR CONSOLIDATION BECAUSE THE DECAY HEAT PER ASSEMBLY WILL BE TOO HIGH.

THE NUMBER OF REFUELING REQUIRED TO FILL THE POOL LESS ONE CORE OFF-LOAD WILL BE:

$$(1945 - 193) \times \frac{2 \text{ FUEL ASSEMBLIES PER CELL}^*}{96 \text{ FUEL ASSEMBLIES/REFUELING}}$$

$$= 30.31 \text{ REFUELINGS}$$

AGAIN, SINCE THE NUMBER OF REFUELINGS IS NOT A WHOLE NUMBER, 0.31 REFUELINGS OR 24 FUEL ASSEMBLIES WILL BE ADDED TO THE FIRST REFUELING TO COMPLETELY FILL THE POOL LESS ONE CORE OFF-LOAD.

* IT'S ASSUMED THAT CONSOLIDATION WILL BE TWO FUEL ASSEMBLIES INTO ONE CELL.

U.S. TOOL & DIE, INC.

BY PCS DATE 9/15/87 SUBJECT NYPA IN0A-3 SECT. ____ SHEET 8 OF 17

CHKD. BY [illegible] DATE 11/24/87 DECAY HEAT PROJ. NO. 5721-01

CASE 3

BATCH	NUMBER FUEL ASSEMBLIES	OPERATING TIME, DAYS	TIME OUT OF REACTOR LESS LAST REFUELING SHUT DOWN TIME, DAYS
1	100	1050	17632
2	76	1050	17024
3	76	1050	16416
4	76	1050	15808
5	76	1050	15200
6	76	1050	14592
7	76	1050	13984
8	76	1050	13376
9	76	1050	12768
10	76	1050	12160
11	76	1050	11552
12	76	1050	10944
13	76	1050	10336
14	76	1050	9728
15	76	1050	9120
16	76	1050	8512
17	76	1050	7904
18	76	1050	7296
19	76	1050	6688
20	76	1050	6080
21	76	1050	5472
22	76	1050	4864
23	76	1050	4256
24	76	1050	3648
25	76	1050	3040
26	76	1050	2432
27	76	1050	1824
28	76	1050	1216
29	76	1050	608
30	76	1050	0
	2304 *		

DAYS AFTER REACTOR SHUTDOWN LAST REFUELING $= \frac{120 \text{ HRS}}{24 \text{ HR/DAY}} = 5 \text{ DAY}$

* NOTE: THIS IS AN UPPER LIMIT. LAST 3 OR 4 NORMAL REFUELINGS WOULD NOT BE CONSOLIDATED UNTIL LATER.

U.S. TOOL & DIE, INC

BY PSS DATE 9/22/87 SUBJECT NYPA INO PT-3 SECT. ____ SHEET 9 OF 17

CHKD. BY MMB DATE 11/24/87 DECAY HEAT PROJ. NO. 8721-0

CASE 4 CONSOLIDATED FUEL – CORE OFF-LOAD

BATCH	NUMBER FUEL ASSEMBLIES	OPERATING TIME, DAYS	TIME OUT OF REACTOR LESS DARS, DAYS
1	100	1050	17668
2	76	1050	17060
3	76	1050	16452
4	76	1050	15844
5	76	1050	15236
6	76	1050	14628
7	76	1050	14020
8	76	1050	13412
9	76	1050	12804
10	76	1050	12196
11	76	1050	11588
12	76	1050	10980
13	76	1050	10372
14	76	1050	9764
15	76	1050	9156
16	76	1050	8548
17	76	1050	7940
18	76	1050	7332
19	76	1050	6724
20	76	1050	6116
21	76	1050	5508
22	76	1050	4900
23	76	1050	4292
24	76	1050	3684
25	76	1050	3076
26	76	1050	2468
27	76	1050	1860
28	76	1050	1252
29	76	1050	644
30	76	1050	36
31	76	1050	0
32	76	666	0
33	41	666	0
	2497 *		

* NOTE: THIS IS AN UPPER LIMIT. LAST 30 OF 4 NORMAL REFUELINGS WOULD NOT BE CONSOLIDATED UNTIL THREE OR FOUR YEARS OUT OF REACTOR.

CASE 4 CONSOLIDATED FUEL CORE OFFLOAD

THIS CASE IS THE SAME AS CASE (3) EXCEPT FOR CORE OFF-LOAD. AGAIN, CORE OFF-LOAD OCCURS 162 HOURS AFTER REACTOR SHUTDOWN. THE LAST NORMAL REFUELING OCCURS 36 DAYS BEFORE REACTOR SHUTDOWN.

THE COMPUTER INPUT AND OUTPUT FOR THESE 4 CASES ARE SHOWN ON PAGES 11 THROUGH 14. FROM THE OUTPUT THE FOLLOWING RESULTS ARE TABULATED.

DECAY HEAT FOR SPENT FUEL POOL

CASE	CONDITION	TOTAL DECAY HEAT, BTU/HR
1	NORMAL REFUELING - REGULAR FUEL	18.58×10^6
2	CORE OFF-LOAD - REGULAR FUEL	40.88×10^6
3	NORMAL REFUELING - CONSOLIDATED FUEL	21.03×10^6
4	CORE OFF-LOAD - CONSOLIDATED FUEL	43.32×10^6

NEW YORK POWER AUTHORITY INDIAN POINT-3
CASE 1 NORMAL REFUELING
EDITED JULY 23, 1987
DATE(MON/DAY/YEAR): 9/17/87
TIME(HOUR:MIN:SEC): 9:53:17

INITIAL POWER = .535E 08 (BTU/HR/FA)
100% OPERATING DAYS =1.000 TIMES RESIDENT TIME

BATCH #	#F/A	TO (DAYS)	TS (DAYS)	FF (FD/FO)	HE	Q (BTU/HR)
1	80	1050.00	8517.00	.4779E-02	.0000E 00	.2556E 06
2	76	1050.00	7909.00	.4295E-02	.0000E 00	.2298E 06
3	76	1050.00	7301.00	.4470E-02	.0000E 00	.2391E 06
4	76	1050.00	6693.00	.4552E-02	.0000E 00	.2408E 06
5	76	1050.00	6085.00	.4841E-02	.0000E 00	.2589E 06
6	76	1050.00	5477.00	.5037E-02	.0000E 00	.2695E 06
7	76	1050.00	4869.00	.5242E-02	.0000E 00	.2804E 06
8	76	1050.00	4261.00	.5456E-02	.0000E 00	.2918E 06
9	76	1050.00	3653.00	.5680E-02	.0000E 00	.3038E 06
10	76	1050.00	3045.00	.5922E-02	.0000E 00	.3168E 06
11	76	1050.00	2437.00	.6215E-02	.0000E 00	.3324E 06
12	76	1050.00	1829.00	.6714E-02	.0000E 00	.3591E 06
13	76	1050.00	1221.00	.8154E-02	.0000E 00	.4362E 06
14	76	1050.00	613.00	.1430E-01	.0000E 00	.7648E 06
15	76	1050.00	5.00	.2349E 00	.2665E-01	.1399E 08

TOTAL DECAY HEAT = 18.5792E+6 BTU/HR
TOTAL NO. OF F/AS STORED = 1152

NEW YORK POWER AUTHORITY INDIAN POINT-3
CASE 2 CORE OFF-LOAD REGULAR FUEL
EDITED JULY 23, 1987
DATE(MON/DAY/YEAR): 9/17/87
TIME(HOUR:MIN:SEC): 10:13:23

INITIAL POWER = .535E 08 (BTU/HR/FA)
100% OPERATING DAYS =1.000 TIMES RESIDENT TIME

BATCH #	#F/A	TO (DAYS)	TS (DAYS)	FF (FD/FO)	HE	Q (BTU/HR)
1	80	1050.00	8554.75	.4767E-02	.0000E 00	.2550E 06
2	76	1050.00	7946.75	.4285E-02	.0000E 00	.2292E 06
3	76	1050.00	7338.75	.4459E-02	.0000E 00	.2385E 06
4	76	1050.00	6730.75	.4640E-02	.0000E 00	.2482E 06
5	76	1050.00	6122.75	.4829E-02	.0000E 00	.2583E 06
6	76	1050.00	5514.75	.5025E-02	.0000E 00	.2688E 06
7	76	1050.00	4906.75	.5229E-02	.0000E 00	.2797E 06
8	76	1050.00	4298.75	.5442E-02	.0000E 00	.2911E 06
9	76	1050.00	3690.75	.5666E-02	.0000E 00	.3031E 06
10	76	1050.00	3082.75	.5906E-02	.0000E 00	.3159E 06
11	76	1050.00	2474.75	.6193E-02	.0000E 00	.3313E 06
12	76	1050.00	1866.75	.6669E-02	.0000E 00	.3567E 06
13	76	1050.00	1258.75	.8000E-02	.0000E 00	.4279E 06
14	76	1050.00	650.75	.1355E-01	.0000E 00	.7246E 06
15	76	1050.00	42.75	.1027E 00	.3939E-06	.5491E 07
16	76	1050.00	6.75	.2142E 00	.1591E-01	.1231E 08
17	76	666.00	6.75	.2094E 00	.1591E-01	.1205E 08
18	41	666.00	6.75	.1129E 00	.8504E-02	.6501E 07

TOTAL DECAY HEAT = 40.8779E+6 BTU/HR
TOTAL NO. OF F/AS STORED = 1345

NEW YORK POWER AUTHORITY INDIAN POINT-3
CASE 2 CORE OFF-LOAD REGULAR FUEL
EDITED JULY 23, 1987
53.49E06.1.0
1050..1050..1050..1050..1050..1050..1050..1050..1050..1050..1050..
1050..1050..1050..1050..1050..666..666.
8548..7940..7332..6724..6116..5508..4900..4292..3684..
3076..2468..1860..1252..644..36..0.0,0.0,0.0
80.,76.,76.,76.,76.,76.,76.,76,76,76.,76.,76.,76.,76.,76.,76.,76.,41.

U.S. TOOL & DIE, INC

BY PSS DATE 9/23/87 SUBJECT NYPA IND Pt-3 SECT. ____ SHEET 13 OF 17

CHKD. BY [illegible] DATE 11/24/87 DECAY HEAT PROJ. NO. 8721-0

NEW YORK POWER AUTHORITY INDIAN POINT-3
CASE 3 NORMAL REFULLING CONSOLIDATED FUEL
EDITED JULY 23. 1987
DATE(MON/DAY/YEAR): 9/17/87
TIME(HOUR.MIN.SEC): 10:43: 2

INITIAL POWER = .535E 08 (BTU/HR/FA)
100% OPERATING DAYS =1.000 TIMES RESIDENT TIME

BATCH #	#F/A	TO (DAYS)	TS (DAYS)	FF (FD/FO)	HE	Q (BTU/HR)
1	100	1050.00	*******	.2900E-02	.0000E 00	.1598E 06
2	76	1050.00	*******	.2363E-02	.0000E 00	.1264E 06
3	76	1050.00	*******	.2459E-02	.0000E 00	.1315E 06
4	76	1050.00	*******	.2557E-02	.0000E 00	.1369E 06
5	76	1050.00	*******	.2663E-02	.0000E 00	.1424E 06
6	76	1050.00	*******	.2771E-02	.0000E 00	.1482E 06
7	76	1050.00	*******	.2884E-02	.0000E 00	.1542E 06
8	76	1050.00	*******	.3001E-02	.0000E 00	.1605E 06
9	76	1050.00	*******	.3123E-02	.0000E 00	.1670E 06
10	76	1050.00	*******	.3250E-02	.0000E 00	.1738E 06
11	76	1050.00	*******	.3382E-02	.0000E 00	.1809E 06
12	76	1050.00	*******	.3519E-02	.0000E 00	.1883E 06
13	76	1050.00	*******	.3663E-02	.0000E 00	.1959E 06
14	76	1050.00	9733.00	.3811E-02	.0000E 00	.2039E 06
15	76	1050.00	9125.00	.3966E-02	.0000E 00	.2122E 06
16	76	1050.00	8517.00	.4128E-02	.0000E 00	.2208E 06
17	76	1050.00	7909.00	.4295E-02	.0000E 00	.2298E 06
18	76	1050.00	7301.00	.4470E-02	.0000E 00	.2391E 06
19	76	1050.00	6693.00	.4652E-02	.0000E 00	.2488E 06
20	76	1050.00	6085.00	.4841E-02	.0000E 00	.2589E 06
21	76	1050.00	5477.00	.5037E-02	.0000E 00	.2695E 06
22	76	1050.00	4869.00	.5242E-02	.0000E 00	.2804E 06
23	76	1050.00	4261.00	.5456E-02	.0000E 00	.2918E 06
24	76	1050.00	3653.00	.5680E-02	.0000E 00	.3038E 06
25	76	1050.00	3045.00	.5922E-02	.0000E 00	.3168E 06
26	76	1050.00	2437.00	.6215E-02	.0000E 00	.3324E 06
27	76	1050.00	1829.00	.6714E-02	.0000E 00	.3591E 06
28	76	1050.00	1221.00	.8154E-02	.0000E 00	.4362E 06
29	76	1050.00	613.00	.1430E-01	.0000E 00	.7648E 06
30	76	1050.00	5.00	.2349E 00	.2665E-01	.1399E 08

TOTAL DECAY HEAT = 21.0263E+6 BTU/HR
TOTAL NO. OF F/AS STORED = 2304

NEW YORK POWER AUTHORITY INDIAN POINT-3
CASE 3 NORMAL REFUELING CONSOLIDATED FUEL
EDITED JULY 23. 1987
53.49E06.1.0
1050..1050..1050..1050..1050..1050..1050..1050..
1050..1050..1050..1050..1050..1050..1050..1050..
1050..1050..1050..1050..1050..1050..1050..1050..
1050..1050..1050..1050..1050..1050..
17632..17024..16416..15808..15200..14592..13984..13376..
12768..12160..11552..10944..10335..9728..9120..
8512..7904..7296..6688..6080..5472..4864..4256..
3648..3040..2432..1824..1216..608..0.0
100..76..76..76..76..76,76,76..76..76..76..76..76..76.
76..76..76..76..76..76..76..76..76..76..76..76..76..76..

NEW YORK POWER AUTHORITY INDIAN POINT-3
CASE 4 CONSOLIDATED FUEL CORE OFF-LOAD

BATCH #	#F/A	TO (DAYS)	TS (DAYS)	FF (FD/FO)	HE	Q (BTU/HR)
1	100	1050.00	*******	.2980E-02	.0000E 00	.1594E 06
2	76	1050.00	*******	.2357E-02	.0000E 00	.1261E 06
3	76	1050.00	*******	.2453E-02	.0000E 00	.1312E 06
4	76	1050.00	*******	.2552E-02	.0000E 00	.1365E 06
5	76	1050.00	*******	.2656E-02	.0000E 00	.1421E 06
6	76	1050.00	*******	.2764E-02	.0000E 00	.1479E 06
7	76	1050.00	*******	.2877E-02	.0000E 00	.1539E 06
8	76	1050.00	*******	.2994E-02	.0000E 00	.1601E 06
9	76	1050.00	*******	.3115E-02	.0000E 00	.1666E 06
10	76	1050.00	*******	.3242E-02	.0000E 00	.1734E 06
11	76	1050.00	*******	.3374E-02	.0000E 00	.1805E 06
12	76	1050.00	*******	.3511E-02	.0000E 00	.1878E 06
13	76	1050.00	*******	.3653E-02	.0000E 00	.1954E 06
14	76	1050.00	9770.75	.3802E-02	.0000E 00	.2034E 06
15	76	1050.00	9162.75	.3957E-02	.0000E 00	.2116E 06
16	76	1050.00	8554.75	.4117E-02	.0000E 00	.2202E 06
17	76	1050.00	7946.75	.4285E-02	.0000E 00	.2292E 06
18	76	1050.00	7338.75	.4459E-02	.0000E 00	.2385E 06
19	76	1050.00	6730.75	.4640E-02	.0000E 00	.2482E 06
20	76	1050.00	6122.75	.4829E-02	.0000E 00	.2583E 06
21	76	1050.00	5514.75	.5025E-02	.0000E 00	.2688E 06
22	76	1050.00	4906.75	.5229E-02	.0000E 00	.2797E 06
23	76	1050.00	4298.75	.5442E-02	.0000E 00	.2911E 06
24	76	1050.00	3690.75	.5666E-02	.0000E 00	.3031E 06
25	76	1050.00	3082.75	.5906E-02	.0000E 00	.3159E 06
26	76	1050.00	2474.75	.6193E-02	.0000E 00	.3313E 06
27	76	1050.00	1866.75	.6669E-02	.0000E 00	.3567E 06
28	76	1050.00	1258.75	.8000E-02	.0000E 00	.4279E 06
29	76	1050.00	650.75	.1355E-01	.0000E 00	.7246E 06
30	76	1050.00	42.75	.1027E 00	.3939E-06	.5491E 07
31	76	1050.00	6.75	.2142E 00	.1591E-01	.1231E 08
32	76	666.00	6.75	.2094E 00	.1591E-01	.1205E 08
33	41	666.00	6.75	.1129E 00	.8584E-02	.6501E 07

TOTAL DECAY HEAT = 43.3210E+6 BTU/HR
TOTAL NO. OF F/AS STORED = 2497

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NEW YORK POWER AUTHORITY INDIAN POINT-3
CASE 4 CONSOLIDATED FUEL CORE OFF-LOAD
EDITED JULY 23, 1987
53 49206.1.0
1050..1050 .1050 .1050 .1050..1050 .1050 .1050..
1050 .1050 .1050 .1050 .1050 .1050 .1050 .1050..
1050..1050..1050..1050..1050..1050..1050..1050..
1050 .1050..1050..1050..1050..1050..1050..666..666.
17660..17060 .16452..15844..15236..14628..14020.,13412..
12804 .12196 .11588 .10980..10372..9764..9156 .8548..7940.,
7332..6724 .6116..5508..4900..4292..3684..3076 .2468.,
1860 .1252 .644 .36 .0..0..0.
100 .76..76..76..76.,76.76.76..76..76..76..76 .76..76 .76
76 .76..76 .76..76 .76 .76..76..76..76 .76..76 .76..76 .76 .
76 .76 .41..
```

HEAT EXCHANGER ANALYSIS

GIVEN; FROM (FSAR)

$T_{H1} = 120°F$ $\quad T_{C1} = 100.0°F$

$T_{H2} = 112.8°F$ $\quad T_{C2} = 105.7°F$

$C_H = 1.1 \times 10^6$ LB/HR $\quad C_C = 1.4 \times 10^6$ LB/HR

$$\frac{T_{H_2} - T_{C_1}}{T_{H1} - T_{C_2}} = e^{-\left[1 - \frac{C_H}{C_C}\right] NTU}$$

REF: (T4)

SOLVE FOR NTU

$$\frac{112.8 - 100}{120 - 105.7} = e^{-\left[1 - \frac{1.1 \times 10^6}{1.4 \times 10^6}\right] NTU}$$

$$e^{-.2142\, NTU} = \frac{12.8}{14.3} = 0.8951$$

$$-0.2142\, NTU = -0.1108$$

$$NTU = 0.5172$$

THE HEAT EXCHANGER EFFECTIVENESS IS

$$\epsilon = \frac{1 - e^{-NTU(1 - C_{min}/C_{max})}}{1 - \frac{C_{min}}{C_{max}} e^{-NTU(1 - C_{min}/C_{max})}}$$

$$\epsilon = \frac{1 - e^{-0.5172(.2142)}}{1 - \left(\frac{1.1}{1.4}\right) e^{-0.5172(.2142)}} = \frac{1 - 0.8951}{1 - 0.70329}$$

$$\epsilon = 0.3535$$

PER NYPA AUGUST 25, 1987 LETTER, ϵ SHOULD BE DERATED TO 90% OR $\epsilon = 0.3535(.9) = 0.31815$

FROM HEAT EXCHANGER ANALYSIS

$$T_{POOL}^{MAX} = T_{C_1} + \frac{q}{C_m \epsilon}$$

$C_m = 1.1 \times 10^6$ $\epsilon = 0.31815$

$T_{C_1} = 100°F$ REF (S1)

THEN, $T_{POOL}^{MAX} = 100 + \frac{q}{1.1 \times 10^6 (.31815)}$

FOR THE FOUR CASES

CASE	q $\left(\frac{BTU}{HR}\right)$	T_{POOL} (MAX), °F
1	18.58×10^6	153.09
2	40.89×10^6	216.81
3	21.03×10^6	160.09
4	43.32×10^6	223.78

THE ABOVE POOL TEMPERATURES ARE HIGHER THAN ALLOWABLES.

A FURTHER ANALYSIS WAS UNDERTAKEN TO DETERMINE DATE AFTER SHUTDOWN TO MAINTAIN ALLOWABLE POOL TEMPERATURES.

THE RESULTS WERE AS FOLLOWS:

TO MEET 140°F MAXIMUM POOL TEMPERATURE THE ALLOWABLE POOL HEAT LOAD WOULD BE:

$$T_{POOL}^{MAX} = T_{in} + \frac{q}{C_c \epsilon}$$

OR

$$q = \left(T_{POOL}^{MAX} - T_{in}\right) C_c \epsilon$$

$$= (140-100)^{\circ F}\, 1.1 \times 10^6 \frac{BTU}{HR\,^{\circ}F} \times 0.31815$$

$$= 13.99 \times 10^6 \text{ BTU/HR}$$

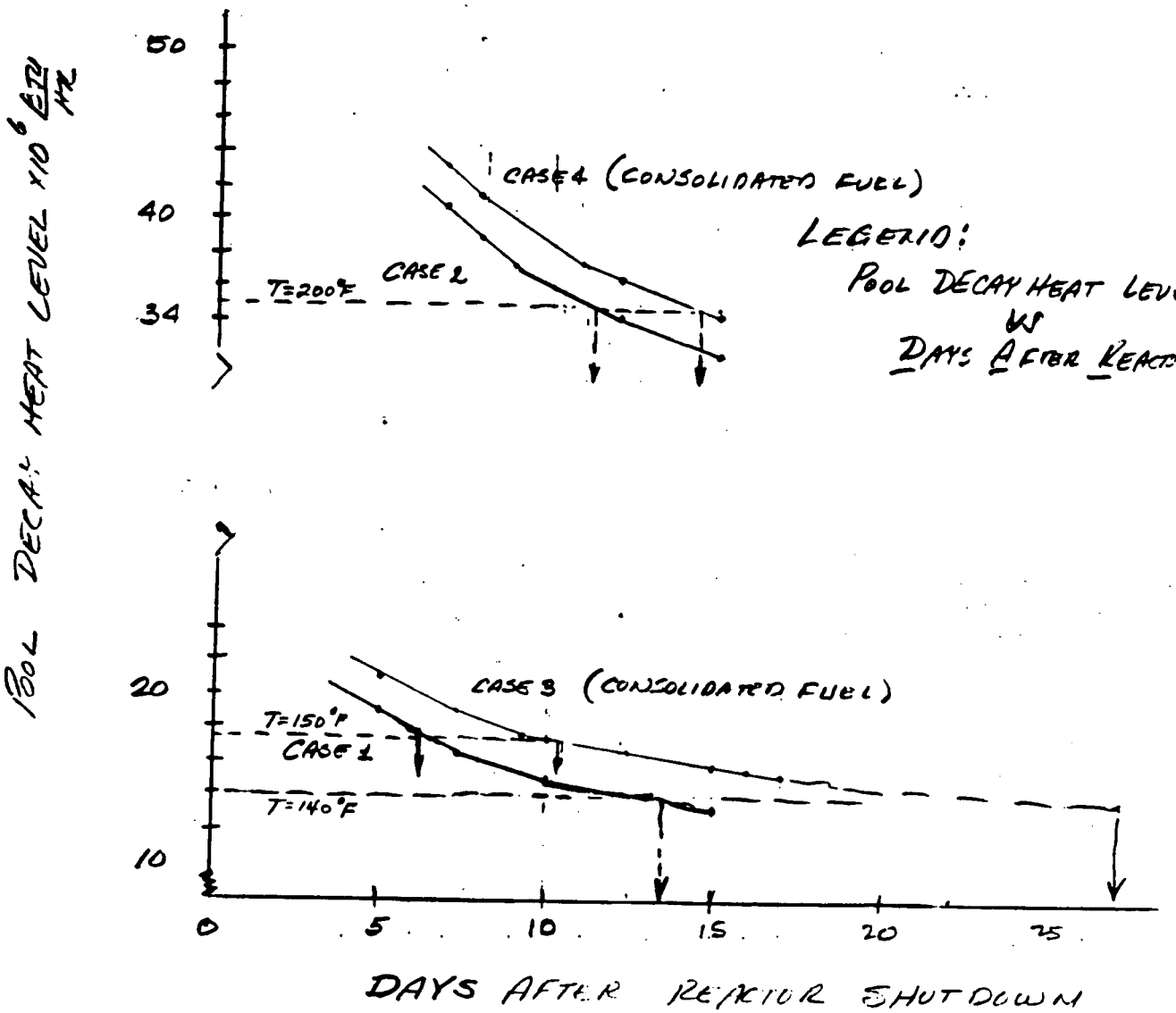
FOR 150°F:

$$q = (150-100) \times 1.1 \times 10^6 \times .31815 = 17.48 \times 10^6 \text{ BTU/HR}$$

FOR 200°F:

$$q = (200-100) \times 1.1 \times 10^6 \times 0.31815 = 35.10 \times 10^6 \text{ BTU/HR}$$

FROM THESE ALLOWABLE POOL LOADS THE DARS ARE DETERMINED FROM THE GRAPHS AND TABLES ON THE FOLLOWING PAGES



RESULTS

T_c	CASE	DARS	T_{pool}^{max}, °F	Q_{max} ALLOWABLE, BTU/H
100°F	1	13.50 (324 HRS)	140	13.99×10^6
		6.05 (145 HRS)	150	17.48×10^6
100°F	2	11.15 (267 HRS)	200	85.00×10^6
100°F	3	27.0 (648 HRS)	140	13.99×10^6
		10.25 (246 HRS)	150	17.48×10^6
100°F	4	14.22 (341 HRS)	200	85.00×10^6

RESULTS ARE VERIFIED ON FOLLOWING COMPUTER RUNS.

NEW YORK POWER AUTHORITY INDIAN POINT-3
CASE 1 NORMAL REFUELING
EDITED JULY 23, 1987
DATE(MON.DAY.YEAR) 11.12.87
TIME(HOUR.MIN.SEC) 9.35.31

INITIAL POWER = .535E 08 (BTU/HR/FA)
100% OPERATING DAYS =1.000 TIMES RESIDENT TIME

BATCH #	#F/A	TO (DAYS)	TS (DAYS)	FF (PD/PO)	HE	Q (BTU/HR)
1	88	1050.00	8525.50	.4777E-02	.0000E 00	.2555E 06
2	76	1050.00	7917.50	.4293E-02	.0000E 00	.2296E 06
3	76	1050.00	7309.50	.4467E-02	.0000E 00	.2390E 06
4	76	1050.00	6701.50	.4549E-02	.0000E 00	.2487E 06
5	76	1050.00	6093.50	.4838E-02	.0000E 00	.2588E 06
6	76	1050.00	5485.50	.5035E-02	.0000E 00	.2693E 06
7	76	1050.00	4877.50	.5239E-02	.0000E 00	.2803E 06
8	76	1050.00	4269.50	.5453E-02	.0000E 00	.2917E 06
9	76	1050.00	3661.50	.5677E-02	.0000E 00	.3037E 06
10	76	1050.00	3053.50	.5910E-02	.0000E 00	.3166E 06
11	76	1050.00	2445.50	.6210E-02	.0000E 00	.3322E 06
12	76	1050.00	1837.50	.6704E-02	.0000E 00	.3586E 06
13	76	1050.00	1229.50	.8119E-02	.0000E 00	.4343E 06
14	76	1050.00	621.50	.1412E-01	.0000E 00	.7553E 06
15	76	1050.00	13.50	.1737E 00	.2178E-02	.9408E 07

TOTAL DECAY HEAT = 13.9812E+6 BTU/HR
TOTAL NO. OF F/AS STORED = 1152

To maintain T_{pool}^{max} < 140°F, DARS = 13.50

NEW YORK POWER AUTHORITY INDIAN POINT-3
CASE 1 NORMAL REFUELING
EDITED JULY 23, 1987
DATE(MON/DAY/YEAR) 11/12/87
TIME(HOUR.MIN.SEC) 9.41. 4

INITIAL POWER = .535E 08 (BTU/HR/FA)
100% OPERATING DAYS = 1.000 TIMES RESIDENT TIME

BATCH #	F/A	TC (DAYS)	TS (DAYS)	IT (PD/PO)	HE	G (BTU/HR)
1	88	1050.00	8518.05	.4779E-02	.0000E 00	.2556E 06
2	76	1050.00	7910.05	.4295E-02	.0000E 00	.2297E 06
3	76	1050.00	7302.05	.4470E-02	.0000E 00	.2391E 06
4	76	1050.00	6694.05	.4651E-02	.0000E 00	.2488E 06
5	76	1050.00	6086.05	.4840E-02	.0000E 00	.2589E 06
6	76	1050.00	5478.05	.5037E-02	.0000E 00	.2694E 06
7	76	1050.00	4870.05	.5242E-02	.0000E 00	.2804E 06
8	76	1050.00	4262.05	.5456E-02	.0000E 00	.2918E 06
9	76	1050.00	3654.05	.5680E-02	.0000E 00	.3038E 06
10	76	1050.00	3046.05	.5922E-02	.0000E 00	.3167E 06
11	76	1050.00	2438.05	.6214E-02	.0000E 00	.3324E 06
12	76	1050.00	1830.05	.6713E-02	.0000E 00	.3591E 06
13	76	1050.00	1222.05	.8150E-02	.0000E 00	.4359E 06
14	76	1050.00	614.05	.1428E-01	.0000E 00	.7636E 06
15	76	1050.00	6.05	.2214E 00	.1956E-01	.1289E 08

TOTAL DECAY HEAT = 17.4755E+6 BTU/HR
TOTAL NO. OF F/AS STORED = 1152

To maintain T_{pool}^{max} < 150°F, DARS = 6.05

U.S. TOOL & DIE, INC.

BY PSS DATE 11/17/87 SUBJECT NYPA IND PT-3 SECT. SHEET 17C OF 17

CHKD. BY MRB DATE 11/24/87 INTACT FUEL - CORE OFF LOAD PROJ. NO. 8721-01

NEW YORK POWER AUTHORITY INDIAN POINT-3
CASE 2 CORE OFF-LOAD REGULAR FUEL
EDITED JULY 23, 1987
DATE(MON/DAY/YEAR): 11/12/87
TIME(HOUR MIN SEC): 9:29:57

INITIAL POWER = .3000E 08 (BTU/HR/FA)
100% OPERATING DAYS =1.000 TIMES RESIDENT TIME

BATCH #	F/A	TO (DAYS)	TS (DAYS)	FF (FD/FO)	HE	G (BTU/HR)
1	80	1050.00	8559.15	.4766E-02	.0000E 00	.2549E 06
2	76	1050.00	7951.15	.4283E-02	.0000E 00	.2271E 06
3	76	1050.00	7343.15	.4450E-02	.0000E 00	.2384E 06
4	76	1050.00	6735.15	.4639E-02	.0000E 00	.2481E 06
5	76	1050.00	6127.15	.4827E-02	.0000E 00	.2582E 06
6	76	1050.00	5519.15	.5024E-02	.0000E 00	.2687E 06
7	76	1050.00	4911.15	.5228E-02	.0000E 00	.2796E 06
8	76	1050.00	4303.15	.5441E-02	.0000E 00	.2910E 06
9	76	1050.00	3695.15	.5664E-02	.0000E 00	.3030E 06
10	76	1050.00	3087.15	.5904E-02	.0000E 00	.3158E 06
11	76	1050.00	2479.15	.6191E-02	.0000E 00	.3312E 06
12	76	1050.00	1871.15	.6664E-02	.0000E 00	.3565E 06
13	76	1050.00	1263.15	.7983E-02	.0000E 00	.4270E 06
14	76	1050.00	655.15	.1346E-01	.0000E 00	.7202E 06
15	76	1050.00	47.15	.9720E-01	.1077E-06	.5199E 07
16	76	1050.00	11.15	.1846E 00	.4352E-02	.1011E 08
17	76	666.00	11.15	.1798E 00	.4352E-02	.9852E 07
18	41	666.00	11.15	.9701E-01	.2348E-02	.5315E 07

TOTAL DECAY HEAT = 34.9774E+6 BTU/HR
TOTAL NO. OF F/AS STORED = 1345

TO MAINTAIN $T_{POOL}^{MAX} < 200°F$, DARS = 11.15

U.S. TOOL & DIE, INC.

BY PSS DATE 11/17/87 SUBJECT NYPA IND PT-3 SECT. SHEET 17-D OF 17

CHKD. BY JRB DATE 11/24/87 CONS. FUEL NORMAL REFUELING T_{POOL}^{MAX} < 140°F PROJ. NO. 8721-01

NEW YORK POWER AUTHORITY INDIAN POINT-3
CASE 3 NORMAL REFUELING CONSOLIDATED FUEL
EDITED JULY 23, 1987
DATE(MON/DAY/YEAR): 11/17/87
TIME(HOUR.MIN.SEC): 8.11:46

INITIAL POWER = .535E 08 (BTU/HR/FA)
100% OPERATING DAYS =1.000 TIMES RESIDENT TIME

BATCH #	#F/A	TC (DAYS)	TS (DAYS)	FF (PD/PO)	HE	Q (BTU/HR)
1	100	1050.00	17659.00	.2983E-02	.0000E 00	.1596E 06
2	76	1050.00	17051.00	.2359E-02	.0000E 00	.1262E 06
3	76	1050.00	16443.00	.2455E-02	.0000E 00	.1313E 06
4	76	1050.00	15835.00	.2555E-02	.0000E 00	.1367E 06
5	76	1050.00	15227.00	.2659E-02	.0000E 00	.1422E 06
6	76	1050.00	14619.00	.2767E-02	.0000E 00	.1480E 06
7	76	1050.00	14011.00	.2880E-02	.0000E 00	.1540E 06
8	76	1050.00	13403.00	.2997E-02	.0000E 00	.1603E 06
9	76	1050.00	12795.00	.3118E-02	.0000E 00	.1668E 06
10	76	1050.00	12187.00	.3245E-02	.0000E 00	.1736E 06
11	76	1050.00	11579.00	.3377E-02	.0000E 00	.1806E 06
12	76	1050.00	10971.00	.3514E-02	.0000E 00	.1880E 06
13	76	1050.00	10362.00	.3657E-02	.0000E 00	.1956E 06
14	76	1050.00	9755.00	.3806E-02	.0000E 00	.2036E 06
15	76	1050.00	9147.00	.3961E-02	.0000E 00	.2119E 06
16	76	1050.00	8539.00	.4122E-02	.0000E 00	.2205E 06
17	76	1050.00	7931.00	.4289E-02	.0000E 00	.2294E 06
18	76	1050.00	7323.00	.4463E-02	.0000E 00	.2388E 06
19	76	1050.00	6715.00	.4645E-02	.0000E 00	.2485E 06
20	76	1050.00	6107.00	.4834E-02	.0000E 00	.2586E 06
21	76	1050.00	5499.00	.5030E-02	.0000E 00	.2691E 06
22	76	1050.00	4891.00	.5235E-02	.0000E 00	.2800E 06
23	76	1050.00	4283.00	.5448E-02	.0000E 00	.2914E 06
24	76	1050.00	3675.00	.5672E-02	.0000E 00	.3034E 06
25	76	1050.00	3067.00	.5913E-02	.0000E 00	.3163E 06
26	76	1050.00	2459.00	.6202E-02	.0000E 00	.3318E 06
27	76	1050.00	1851.00	.6687E-02	.0000E 00	.3577E 06
28	76	1050.00	1243.00	.8063E-02	.0000E 00	.4313E 06
29	76	1050.00	635.00	.1385E-01	.0000E 00	.7408E 06
30	76	1050.00	27.00	.1308E 00	.4080E-04	.6993E 07

TOTAL DECAY HEAT = 13.7990E+6 BTU/HR
TOTAL NO. OF F/AS STORED = 2304

TO MAINTAIN T_{POOL}^{MAX} < 140°F, D HEE = 27.0

U.S. TOOL & DIE, INC.

BY RSS DATE 11/17/87 SUBJECT NYPA IndPt-3 SECT. SHEET 17-E OF 17

CHKD. BY MRR DATE 11/24/87 Cons. Fuel Normal Refueling $T_{POOL}^{MAX} = 150°F$ PROJ. NO. 8721-01

NEW YORK POWER AUTHORITY INDIAN POINT-3
CASE 3 NORMAL REFUELING CONSOLIDATED FUEL
EDITED JULY 23, 1987
DATE(MON/DAY/YEAR) 11/13/87
TIME(HOUR.MIN.SEC) 9.38.52

INITIAL POWER = .5355E 08 (BTU/HR/FA)
100% OPERATING DAYS = 1.000 TIMES RESIDENT TIME

BATCH #	F/A	TO (DAYS)	TS (DAYS)	TF (FD/FO)	HE	Q (BTU/HR)
1	100	1050.00	17642.25	.2907E-02	.0000E 00	.1597E 06
2	76	1050.00	17034.25	.2362E-02	.0000E 00	.1263E 06
3	76	1050.00	16426.25	.2458E-02	.0000E 00	.1315E 06
4	76	1050.00	15818.25	.2558E-02	.0000E 00	.1368E 06
5	76	1050.00	15210.25	.2662E-02	.0000E 00	.1424E 06
6	76	1050.00	14602.25	.2770E-02	.0000E 00	.1482E 06
7	76	1050.00	13994.25	.2883E-02	.0000E 00	.1542E 06
8	76	1050.00	13386.25	.3000E-02	.0000E 00	.1605E 06
9	76	1050.00	12778.25	.3122E-02	.0000E 00	.1670E 06
10	76	1050.00	12170.25	.3249E-02	.0000E 00	.1738E 06
11	76	1050.00	11562.25	.3381E-02	.0000E 00	.1808E 06
12	76	1050.00	10954.25	.3518E-02	.0000E 00	.1882E 06
13	76	1050.00	10346.25	.3662E-02	.0000E 00	.1959E 06
14	76	1050.00	9738.25	.3810E-02	.0000E 00	.2038E 06
15	76	1050.00	9130.25	.3965E-02	.0000E 00	.2121E 06
16	76	1050.00	8522.25	.4126E-02	.0000E 00	.2207E 06
17	76	1050.00	7914.25	.4294E-02	.0000E 00	.2297E 06
18	76	1050.00	7306.25	.4468E-02	.0000E 00	.2390E 06
19	76	1050.00	6698.25	.4650E-02	.0000E 00	.2487E 06
20	76	1050.00	6090.25	.4839E-02	.0000E 00	.2588E 06
21	76	1050.00	5482.25	.5036E-02	.0000E 00	.2694E 06
22	76	1050.00	4874.25	.5240E-02	.0000E 00	.2803E 06
23	76	1050.00	4266.25	.5454E-02	.0000E 00	.2917E 06
24	76	1050.00	3658.25	.5678E-02	.0000E 00	.3037E 06
25	76	1050.00	3050.25	.5920E-02	.0000E 00	.3166E 06
26	76	1050.00	2442.25	.6212E-02	.0000E 00	.3323E 06
27	76	1050.00	1834.25	.6707E-02	.0000E 00	.3588E 06
28	76	1050.00	1226.25	.8132E-02	.0000E 00	.4350E 06
29	76	1050.00	618.25	.1419E-01	.0000E 00	.7589E 06
30	76	1050.00	10.25	.1894E 00	.5674E-02	.1044E 08

TOTAL DECAY HEAT = 17.4600E+6 BTU/HR
TOTAL NO. OF F/AS STORED = 2304

TO MAINTAIN $T_{POOL}^{MAX} < 150°F$, DAYS = 10.25

APPENDIX 8721-02

THERMAL INERTIA AND HEAT-UP RATES

THE PURPOSE OF THIS CALCULATION IS TO DETERMINE THE SPENT FUEL POOL THERMAL INERTIA AND HEAT-UP RATES FOR THE EXPECTED MAXIMUM DECAY HEAT LOADS. THIS IS A REQUIREMENT SPECIFIED BY THE NRC IN THE POSITION PAPER "OT POSITION FOR REVIEW AND ACCEPTANCE OF SPENT FUEL STORAGE AND HANDLING APPLICATIONS."

THE THERMAL INERTIA OF AN OBJECT WITH DENSITY ρ, VOLUME V, AND SPECIFIC HEAT C_p, IS THE PRODUCT;

$$I = \rho C_p V$$

WITH UNITS OF BTU/°F WITH ρ IN LB/FT^3, C_p IN BTU/LB°F AND V IN FT^3. ON A UNIT VOLUME BASIS, THE SPECIFIC THERMAL INERTIAS FOR VARIOUS MATERIALS IN THE SFP ARE AS FOLLOWS:

MATERIAL	ρ (LB/FT^3)	C_p (BTU/LB°F)	ρC_p (BTU/°F FT^3)	REFERENCE
H_2O	61.2	1.0	61.2	@150°F E1
304SS	501	.12	60.1	E2, p105
UO_2	644	.055	37.4	E2, p78
ZIRCALLOY	409	.074	30.3	E2, p105
BORAL	169	0.22	37.2	E3, p3

SINCE THE FUEL ASSEMBLY MATERIALS UO_2 AND ZIRCALOY HAVE SMALLER THERMAL INERTIAS THAN THE WATER THEY DISPLACE, IT WILL BE CONSERVATIVE TO ASSUME A FULL POOL OF FUEL ASSEMBLIES. FOR SIMPLICITY AND A SLIGHT MARGIN OF CONSERVATISM, WE TAKE $\rho C_p = 60$ FOR WATER AND STAINLESS STEEL. IT IS THEN NECESSARY TO FIND THE THERMAL INERTIAS AND VOLUMES OF THE UO_2, ZIRCALOY AND POISON MATERIAL.

BORAL - THERMAL INERTIA

NUMBER POISON SLABS

	REGION 1	REGION 2
NUMBER	960	2210
LENGTH	133	136
WIDTH	7.5 (.98)	8 (.98)
THICKNESS	.075	.075

$$\text{BORAL VOLUME} = 960(133)(7.5)(.98)(.075) + 2210(136)(8)(.98)(.075)$$
$$= 70{,}383.5 + 176{,}729 \ in^3 = 143.0 \ ft^3$$

$$I_{BORAL} = 143.0 \ ft^3 \times 37.2 \ \frac{BTU}{ft^3 \, °F} = 5327 \ \frac{BTU}{°F}$$

UO_2 - THERMAL INERTIA

FROM REF (G1) THE FUEL ROD PELLET DIAMETER IS 0.3734. THERE ARE 204 FUEL RODS PER ASSEMBLY.

$$UO_2 \text{ VOLUME} = \frac{\pi}{4}\left(\frac{.3734}{12}\right)^2 (204)\ \frac{144''}{12''/FT} = 1.861\ FT^3/ASS$$

FOR (240 + 1105) × 2 ASSEMBLIES: CONSOLIDATION

$$I_{UO_2} = (1.861\ FT^3/ASS'Y)(644\ LB/FT^3)(0.058\ \tfrac{BTU}{LB\,°F})(2690\ \text{no ASS'Y}) = \underline{.187 \times 10^6}\ \tfrac{BTU}{°F}$$

ZIRCALOY - THERMAL INERTIA

ROD DIA = 0.422", CLAD THICKNESS = 0.0243"

$$\text{VOLUME OF ZIRC} = \frac{\pi}{4}(204)\left(\frac{.422^2 - .3734^2}{144}\right) \times \frac{154.34}{12}$$

$$V_{ZIRC} = 0.553\ FT^3/ASS'Y$$

$$I_{ZIRC} = 0.553\ FT^3/ASS'Y \times 2690\ ASS'Y \times 30.3\ \tfrac{BTU}{FT^3\,°F} = \underline{0.0451 \times 10^6}\ \tfrac{BTU}{°F}$$

TO DETERMINE THE WATER AND STAINLESS THERMAL INERTIA, WE MUST FIND THE WATER VOLUME DISPLACED BY THE FUEL RODS AND THE POISON.

ROD VOLUME $= \frac{\pi}{4} D^2 \times L \times N$

WHERE: D = DIA
L = LENGTH
N = NUMBER OF RODS

$$\text{ROD VOLUME} = \frac{\pi}{4}\left(\frac{0.422}{12}\right)^2 \times \frac{154.34}{12} \times 2690 \times 204$$

$$= 1220.8 \text{ FT}^3$$

POISON VOLUME = 143.0 FT^3

NON H_2O-SS VOLUME = 1363.8 FT^3

IN DETERMINING THE POOL THERMAL INERTIA, IT IS CONSERVATIVE TO NEGLECT THE CASK STORAGE PIT, TRANSFER CANAL, THE CONCRETE WALLS, AND ALL PRIMARY AND SECONDARY PIPING OUTSIDE THE POOL BOUNDARIES. TAKING ONLY THE WATER WITHIN THE LINER BOUNDARIES AS SHOWN ON DRWING REF (D1), THE SFP WATER SURFACE AREA IS

$$A_{SURF} = (431.5)(395.5) - \{(431.5-323.5)(395.5-95.5)\} - (90)(99) - (31.33)(27.07)$$

$$A_{SURF} = 1.285 \times 10^5 \text{ IN}^2 \text{ OR } 892.36 \text{ FT}^2$$

INCLUDING CASK AREA:

$$A_{SURF} \cong 892.36 + \frac{90 \times 99}{144} = 954.23 \text{ FT}^2$$

THE WATER DEPTH IN THE SFP IS FROM DRWG 9321-F-25143-7

WATER LEVEL = 93'8"

POOL LINER = 54'7 1/4"

POOL WATER DEPTH = 39' 3/4"

VOLUME H_2O + SS = (892.36)(39.06) − 1363.8

= 33494 FT^3

BASED ON CASK AREA ALSO = (954.23)(39.06) − 1363.8 = 35908 FT^3

HOWEVER, FSAR UPDATE TABLE 9.3-3 (REF 81) SPECIFIES VOLUME = 37,300 FT^3, THIS PROBABLY INCLUDES WATER ABOVE CASK REGION.

WE WILL USE FSAR DATA LESS VOLUME OF FUEL AND POISON

V_F = 37,300 − 1364 = 35936 FT^3

THIS CHECKS WITH ABOVE USING CASK AREA IN VOLUME CALCULATIONS.

THEN THE WATER + STAINLESS STEEL THERMAL INERTIA IS = (60)(35936) = 2.156×10^6 BTU/°F

THERMAL INERTIA TOTALS AND SUMMARY:

WATER PLUS STAINLESS STEEL	2.156×10^6 BTU/°F
UO_2 FUEL	$.187 \times 10^6$ BTU/°F
ZIRCALOY CLAD	$.045 \times 10^6$ BTU/°F
POISON	$.005 \times 10^6$ BTU/°F
TOTAL SFP THERMAL INERTIA	2.393×10^6 BTU/°F

POOL HEAT UP RATES

IF SFP COOLING IS LOST, THE POOL WILL HEAT-UP AT THE RATE;

$$\dot{T} = \frac{q}{\rho C_p I_p}, \quad {}^\circ F/HR$$

SINCE THE VALUES OF ρC_p DO NOT CHANGE SIGNIFICANTLY OVER THE RANGE OF TEMPERATURES OF INTEREST,

T_{POOL}^{MAX}		
140°F	$q_1 = 13.99 \times 10^6 \frac{BTU}{HR}$,	$\dot{T} = \frac{13.99 \times 10^6 \frac{BTU}{HR}}{2.393 \times 10^6 \frac{BTU}{°F}} = \underline{5.84}$ °F/HR
150°F	$q_1 = 17.48 \times 10^6$ BTU/HR,	$\dot{T} = \underline{7.30}$ °F/HR

FOR REGULAR FUEL, NORMAL REFUELING

200°F $q_2 = 35.0 \times 10^6 \frac{BTU}{HR}$, $\dot{T} = \underline{14.6}$ °F/HR

FOR REGULAR FUEL, CORE-OFF LOAD

140°F	$q_3 = 13.99 \times 10^6 \frac{BTU}{HR}$,	$\dot{T} = \underline{5.84}$ °F/HR
150°F	$q_3 = 17.48 \times 10^6$ BTU/HR	$\dot{T} = \underline{7.30}$ °F/HR

FOR CONSOLIDATED FUEL, NORMAL REFUEL.

200°F $q_4 = 35.0 \times 10^6 \frac{BTU}{HR}$, $\dot{T} = \underline{14.6}$ °F/HR

FOR MAXIMUM CONSOLIDATED FUEL, CORE-OFF LOAD

TIME TO REACH BOILING, 212°F

CASE 1

$$t = \frac{212 - T_{pool}^{MAX}}{\dot{T}_1}$$

$$= \frac{(212 - 140.0)\,°F}{5.84\ °F/HR} = 12.3\ HR.$$

FOR $T_{pool}^{MAX} = 150°F$

$$t = 10.6\ HR$$

CASE 2

$$t = \frac{212 - T_{pool}^{MAX}}{\dot{T}_2}$$

$$= \frac{(212 - 200.0)\,°F}{14.6\ °F/HR} = 0.82\ HR$$

CASE 3

$$t = \frac{(212 - T_{pool}^{MAX})}{\dot{T}_3}$$

$$= \frac{(212 - 140.0)\,°F}{5.84\ °F/HR} = 12.3\ HR$$

FOR $T_{pool}^{MAX} = 150°F$, $t = 10.6\ HR$

CASE 4

$$t = \frac{(212 - T_{pool}^{MAX})}{\dot{T}_4}$$

$$= \frac{(212 - 200.0)\,°F}{14.6\ °F/HR} = 0.82\ HR$$

THE POOL LEVEL HEIGHT REDUCTION RATE CAN BE APPROXIMATED AS

$$\dot{h} = \frac{\dot{Q}}{h_{fg} A_s \rho}$$ where; $h_{fg} = 970 \frac{BTU}{LB}$ FOR WATER AT 14.7 psia

A_s = Surface Area of pool $\cong 954\ FT^2$

ρ = Density of water $= 61.7 \frac{LB}{FT^3}$

CASE	T_{POOL}^{MAX}	$\dot{Q}$, BTU/HR	$\dot{h}$, FT/HR
1	140	13.99×10^6	0.251
	150	17.48×10^6	0.314
2	200	35.0×10^6	0.630
3	140	13.99×10^6	0.251
	150	17.48×10^6	0.314
4	200	35.0×10^6	0.630

BOIL-OFF RATES,

$$\dot{m} = \frac{\dot{Q}}{h_{fg}}, \ LB/HR$$

$$Q(GPM) = \frac{\dot{m}\ 7.48\ G/FT^3}{\frac{LB}{FT^3}\ 60 \times 60\ \frac{MIN}{HR}}$$

CASE	T_{POOL}^{MAX}	$\dot{Q}$ BTU/HR	$\dot{m}$, LB/HR	Q(GPM)
1	140°F	13.99×10^6	14,422	29.96
	150°F	17.48×10^6	18,020	37.44
2	200°F	35.0×10^6	36,082	74.97
3	140°F	13.99×10^6	14,422	29.96
	150°F	17.48×10^6	18,020	37.44
4	200°F	35.0×10^6	36,082	74.97

RESULTS OF HEAT-UP ANALYSIS FOLLOWING THE LOSS OF SFP COOLING, SUMMARY TABLE

CASE	T_{POOL}^{MAX}	q, BTU/HR	$\dot{T}$, °F/HR	Δt (to 212°F)	$\dot{m}$ (LB/HR)	Q (GPM)	$\dot{h}$ (FT/HR)
1	140	13.99×10^6	5.84	12.3	14,422	29.9	.25
	150	17.48×10^6	7.30	10.6	18,020	37.4	.31
2	200	35.00×10^6	14.6	0.82	36,082	74.9	.63
3	140	13.99×10^6	5.84	12.3	14,422	29.9	.25
	150	17.48×10^6	7.30	10.6	18,020	37.4	.31
4	200	35.00×10^6	14.6	0.82	36,082	74.9	.63

WHERE:

q = DECAY HEAT LOAD, BTU/HR

$\dot{T}$ = POOL HEAT UP RATE, °F/HR

Δt = TIME TO REACH BOILING, HRS

$\dot{m}$ = BOIL OFF RATE, LB/HR

Q = BOIL OFF RATE, GPM

$\dot{h}$ = POOL LEVEL REDUCTION RATE, FT/HR

APPENDIX 8721-03

NATURAL CIRCULATION COOLING OF SPENT FUEL

THE ULTIMATE PURPOSE OF THIS ANALYSIS IS TO DETERMINE THE PEAK COOLANT AND CLAD TEMPERATURES WITHIN THE HOTTEST FUEL ASSEMBLY. SINCE WORST CASE CONDITIONS ARE ASSUMED (i.e., HIGHEST FUEL ASSEMBLY HEATING AND HIGHEST RESISTANCE TO FLOW) WE SHALL USE OR ASSUME THE FOLLOWING:

1. A CORE OFF-LOAD SFP POPULATION WITH 193 HOT FUEL ASSEMBLIES (AND 1152 "COLD" ASSEMBLIES — TOTAL 1345 FUEL ASSEMBLIES) IS USED.
2. DECAY HEAT RATES FOR THE CORE OFF-LOAD WITH P/P_0 CHARACTERISTIC OF NORMAL REFUELING, 120 HOURS AFTER REACTOR SHUTDOWN ARE USED.
3. A SINUSOIDAL AXIAL HEAT FLUX DISTRIBUTION IS USED SO THAT THE PEAK CLAD HEAT FLUX IS $\pi/2 = 1.57$ TIMES THE AVERAGE CLAD HEAT FLUX

4. THE FUEL ASSEMBLY PRESSURE DROP CHARACTERISTICS ARE DERIVED FROM FSAR DATA REDUCED TO LAMINAR FLOW CONDITIONS.

5. TO MAXIMIZE THE WATER AND FUEL TEMPERATURE RISE IT IS ASSUMED THAT ALL OF THE DECAY HEAT IS DEPOSITED IN THE FUEL. 1

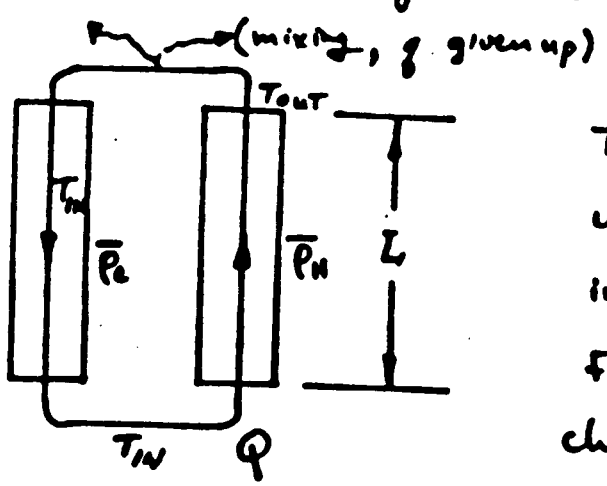
6. TO BE CONSERVATIVE AND UNDERESTIMATE THE ASSEMBLY FLOW RATES, THE DOMINANT FLOW RESISTANCES ARE OVERESTIMATED. 1

IN THE FOLLOWING ANALYSIS WE WILL CONSIDER TWO FLOW PATHS. THE FIRST (PATH 1) IS A LOCAL FLOW PATH (FLOW UP A HOT FUEL ASSEMBLY AND THEN DOWN AN ADJACENT COLD FUEL ASSEMBLY). THE SECOND (PATH 2) PATH IS A MORE COMPLICATED PATH DOWN THE SOUTH WALL AND UNDER THE RACK THEN UP THROUGH THE HOT ASSEMBLY THEN ACROSS THE TOP OF THE RACKS TO THE SOUTH WALL. IN THE ACTUAL OPERATION THE FLOWS WILL BE COMBINATIONS OF BOTH FLOW PATHS. THUS, THE CLAD TEMPERATURES WILL BE BRACKETED BY THIS ANALYSIS.

IN ALL CASES THE DRIVING PRESSURE WILL BE THE SAME.

Flow Driving Pressure

To model the driving pressure, let us consider two vertical channels of length L. We denote the average densities in these channels by $\bar{\rho}_c$ (cold channel) and $\bar{\rho}_H$ (heated channel). Let T_{IN} be the cold box temperature and the heated box inlet temperature. T_{OUT} will be the heated box outlet temperature. Mixing occurs above these channels so that the fluid enters the cold channel again at T_{IN}.



The driving pressure will be due to the imbalance of the fluid weights in these channels. Thus, the driving pressure can be written as

(1) $$\Delta P_d = \left(\bar{\rho}_c - \bar{\rho}_H\right) L \left(\frac{g}{g_c}\right)$$

The average temperature in the heated

box can be written as (for single phase flow)

(2) $$T_{AVG} = \frac{T_{OUT} + T_{IN}}{2} = T_{IN} + \frac{\Delta T_H}{2}$$

where $\Delta T_H = T_{OUT} - T_{IN}$.

Since ρ is only a mild function of pressure, we may write

(3) $$\frac{\partial \rho}{\partial T} \approx \frac{d\rho}{dT}$$

and expand $\bar{\rho}_H$ in a Taylor series about $\bar{\rho}_c$

(4) $$\bar{\rho}_H \approx \bar{\rho}_c + \frac{d\rho}{dT_{IN}}\left(\frac{\Delta T_H}{2}\right)$$

the driving pressure then becomes

(5) $$\Delta P_d = \left(\frac{-d\rho}{dT}\right)_{IN} \left(\frac{\Delta T_H}{2}\right) \left(\frac{g}{g_c}\right) L$$

where $(-d\rho/dT)$ is evaluated at the inlet temperature. Since ρ decreases as T increases, $\Delta P_d > 1$

Let Q be the volume flow rate into the heated assembly and q be the heat given up by the heated region. So

$$q = \rho C_p Q (\Delta T_H)$$

or

(6) $$\Delta P_d = \left(\frac{-d\rho}{dT}\right)_{IN} \left(\frac{g L q}{2 \rho C_p g_c}\right)\left(\frac{1}{Q}\right)$$

thus, the driving pressure is directly proportional to the heat rate q and inversely proportional to the volume flow rate Q.

If the heat is due to N fuel rods of diameter D, each with a heat flux distribution at the clad surface given by

$$(7) \qquad q'' = q''_{max} \sin\left(\frac{\pi z}{L}\right)$$

where z is the axial position measured from the inlet (bottom), then

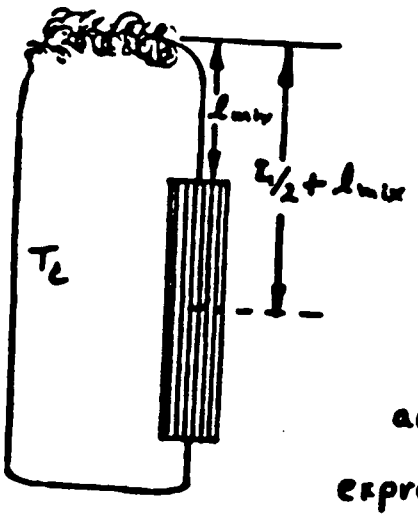
$$(8) \qquad q = q''_{max} \pi D N L \left(\frac{2}{\pi}\right)$$

[The factor $2/\pi$ is the average to peak ratio for the sine curve.]

The alternate form for the driving pressure for this special application can then be written as

$$(9) \qquad \Delta P_d = \left(\frac{-d\rho}{dT}\right)_{IN} \left(\frac{N D L^2 q''_{max}}{\rho\, c_p\, Q} \frac{g}{g_c}\right)$$

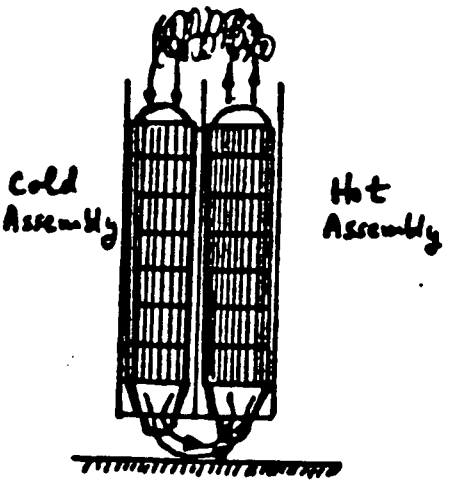
It should be noted that the driving pressure is simply the hydrostatic head difference around a closed path. For simplicity, we have neglected the differences in elevation from the fuel assembly outlet to the mixing region above the racks. We also assume a constant density in the cold leg. In a more refined calculation, L_1 would be replaced by $L_1/2 + l_{mix}$ where l_{mix} is the average distance to the mixing region. Typically there is ~20-25 ft of water on top of the racks, so $l_{mix} \sim 10$ ft.



But $L_1 \sim 12$ ft active fuel length, so $L_1/2 + l_{mix} \sim 15$ ft. Thus, using L_1 as the fuel active length in the preceeding expressions will conservatively derate the flow driving head.

Path 2 - Local Path.

For this path, we consider a flow path driven up the hot assembly. At the top of the racks, heat is given-up in the mixing region and flow continues by travelling downward in a "cold" batch fuel assembly. It will be conservative to take



$T_{IN} = T_{HIN}$ for this path — i.e. the hot temperature of the SFP water into the HX.

Since we assume

$$q_{HA} = 1.84 \times 10^5 \text{ BTU/hr}$$

and

$$q_{CB} \cong 4.6 \times 10^3 \text{ BTU/hr} \; (= .025 q_{HA})^{*}$$

such a path is possible and the cold batch heat can be neglected.

* 193 ASSEMBLIES @ 1.84×10^5 BTU/HR IS 35.5×10^6 BTU/HR. THE REMAINDER OF $(40.8 - 35.5) \times 10^6$ IS DIVIDED BY THE REMAINING CELLS (i.e., $5.3 \times 10^6 / 1152 \cong 4.6 \times 10^3$ BTU/HR/ASSY

FUEL ASSEMBLY HYDRAULIC PRESSURE DROP CHARACTERISTICS FOR INDIAN PT-3 WILL BE ESTIMATED FROM DATA SUPPLIED BY NYPA (REF B1) FROM THEIR FSAR. TO DO THIS, THE TOTAL HEAD LOSS AT FULL FLOW WILL BE DETERMINED. THE FORM LOSSES WILL BE CALCULATED AND ASSUMED INDEPENDENT OF REYNOLDS NUMBER (DATA REDUCTION OF G.E. FUEL ASSEMBLY PAGE 10A OF 30 JUSTIFIES THIS ASSUMPTION) AND THEN ADDED TO LAMINAR FLOW FRICTION LOSSES.

FULL FLOW

FROM TABLE 3.2-4 (REF B1)

ΔP = CORE PRESSURE DROP = 21.0 PSI

V = FLOW VELOCITY PASSED FUEL RODS = 14.9 FT/SEC

K = HEAD LOSS AT FULL FLOW $= K = \dfrac{\Delta P}{\frac{V^2}{2g}\rho}$

ρ = density
g = gravity constant 32.17 ft/sec²

$$K = \frac{(21.0\ LB/in^2) \times 144\ in^2/ft^2 \times 2g}{(14.9\ FT/SEC)^2 \times 44.6\ \frac{LB}{FT^3}}$$

$\rho = 44.6\ LB/FT^3$ @ 2250 psia SAT LIQ.

$K = 19.64$, VELOCITY HEADS

THE CORE REYNOLDS NUMBER CAN BE DETERMINED FROM;

$$Re = V D_e \frac{\rho}{\mu},$$ where; V = Velocity PASSED FUEL RODS
D_e = HYD. DIA OF FUEL ASSEMBLY
ρ = Density of water
μ = Viscosity of water

A_f = THE CORE FLOW AREA = $\dfrac{\text{TOTAL FLOW RATE } \frac{LB}{HR}}{\text{AVE MASS VELOCITY } \frac{LB}{HR\,FT^2}}$

$$= \frac{130.2 \times 10^6 \ \frac{LB}{HR}}{2.43 \times 10^6 \ \frac{LB}{HR\,FT^2}}$$

$$A_f = 53.58 \ FT^2$$

THE HYDRAULIC DIAMETER = $\dfrac{4\,\text{FLOW AREA}}{\text{WETTED PER}}$ = D_e

WETTED PERIMETER = $\dfrac{\text{HEAT TRANSFER AREA}}{\text{FUEL LENGTH}}$

$$= \frac{52{,}200 \ FT^2}{12 \ FT} = 4333 \ FT$$

THEREFORE;

$$D_e = \frac{4 \times 53.58 \ FT^2}{4333 \ FT} = 0.0494 \ FT$$

THE VISCOSITY OF WATER AT CORE CONDITIONS

$$\mu \cong 0.233 \ \frac{LB}{HR\text{-}FT}$$

THUS, THE CORE REYNOLDS NUMBER IS

$$R_e = \frac{14.9 \ \frac{FT}{SEC} \times 3600 \ \frac{SEC}{HR} \times 0.0494 \ FT \times 44.6 \ \frac{LB}{FT^3}}{0.233 \ \frac{LB}{HR\text{-}FT}}$$

$$R_e = 507{,}818$$

THE FRICTION FACTOR CAN BE APPROXIMATED FROM THE BLASIUS EQUATION (REF M1)

$$f = 0.325 \ R_e^{-.25}$$

THUS THE CORE FRICTION LOSSES CAN BE ESTIMATED AS:

$$\frac{fL}{D_e} = \frac{0.325\,(507{,}818)^{-.25} \times 150/12 \text{ FT}}{0.0494 \text{ FT}}$$

$$= 3.08 \text{ VELOCITY HEADS}$$

THE CORE FORM LOSSES (7 GRIDS) ARE

$$K_{FORM} = K_{TOTAL} - \frac{fL}{D_e}$$

$$= 19.64 - 3.08 = 16.55, \text{ VELOCITY HEADS}$$

THESE LOSSES WILL BE ASSUMED INDEPENDENT OF REYNOLDS NUMBER. (DATA REDUCTION OF DATA FROM G.E. FUEL BUNDLE SHOWS THIS EFFECT (NEXT PAGE)) AT LAMINAR FLOW CONDITIONS, $R_e = 2000$, THE FRICTION LOSSES WILL BE

$$\frac{fL}{D_e} = \frac{.325\,(2000)^{-.25} \times 150/12 \text{ FT}}{0.0494 \text{ FT}} = 12.29$$

SUMMARY, HYDRAULIC LOSSES FOR FUEL ASSEMBLY

REYNOLDS NO	FRICTION LOSSES	FORM LOSSES	TOTAL LOSSES
2000	12.29	16.55	28.84
507,818	3.08	16.55	19.64

THE ABOVE TOTAL LOSSES DATA WILL BE CURVE FITTED WITH A FORM:

$$K = A\,Re^{-n}$$

U.S. TOOL & DIE, INC.

BY PS² DATE 11/13/87 SUBJECT NYPA Two Pt-3 SECT. ____ SHEET 10A OF 30

CHKD. BY MRB DATE 11/24/87 GRID LOSS COEFFICIENT DATA PROJ. NO. 8721-03

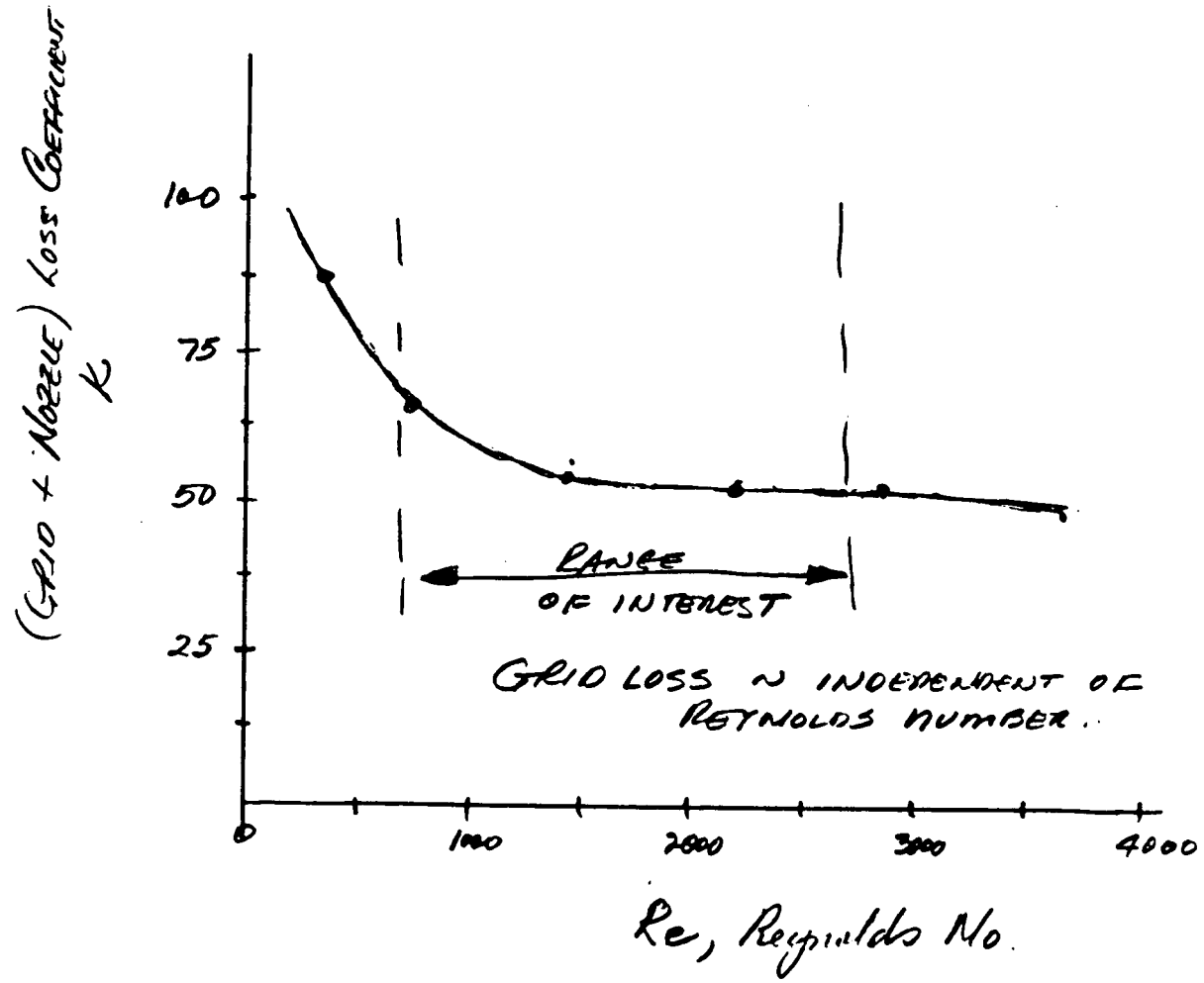
Rev(1) 1/12/88

GE DATA FOR SINGLE PHASE FLOW THRU 64 ROD BUNDLE (REPORT: GENERAL ELECT. TECHNICAL PAPER. 22A5866 REV 1)

m, lb/hr	ΔP (psi)	$\Delta P_{friction}$ psi	Re	ΔP_K, psi	$K = \frac{\Delta P_K}{V^2/2g\rho}$
1×10^3	.0015	.000548	360	.00095	87.3
2×10^3	.004	.00109	721	.0029	67.6
4×10^3	.012	.0022	1444	.0097	55.4
6×10^3	.025	.0033	2166	.0216	54.8
8×10^3	.042	.0044	2888	.0375	53.7
10×10^3	.070	.0055	3610	.0644	49.8

Ref (W1)

MRB



THIS WILL REPRESENT THE FUEL ASSEMBLY PRESSURE LOSS CHARACTERISTICS, i.e,

$$\Delta P_{FUEL} = A\, Re^{-n} \frac{V^2}{2g} \rho$$

FROM THE SUMMARY TABLE, THE TOTAL LOSSES FOR EACH REYNOLDS NUMBER IS:

$$28.84 = A(2000)^{-n}$$

AND; $19.64 = A(507818)^{-n}$

THESE EQUATIONS YIELD;

$$A = 48.72$$

$$n = 0.069$$

THEREFORE, THE FUEL ASSEMBLY PRESSURE DROP CHARACTERISTICS ARE:

$$\Delta P_{FUEL} = 48.72\, Re^{-.069} \frac{V^2}{2g} \rho$$

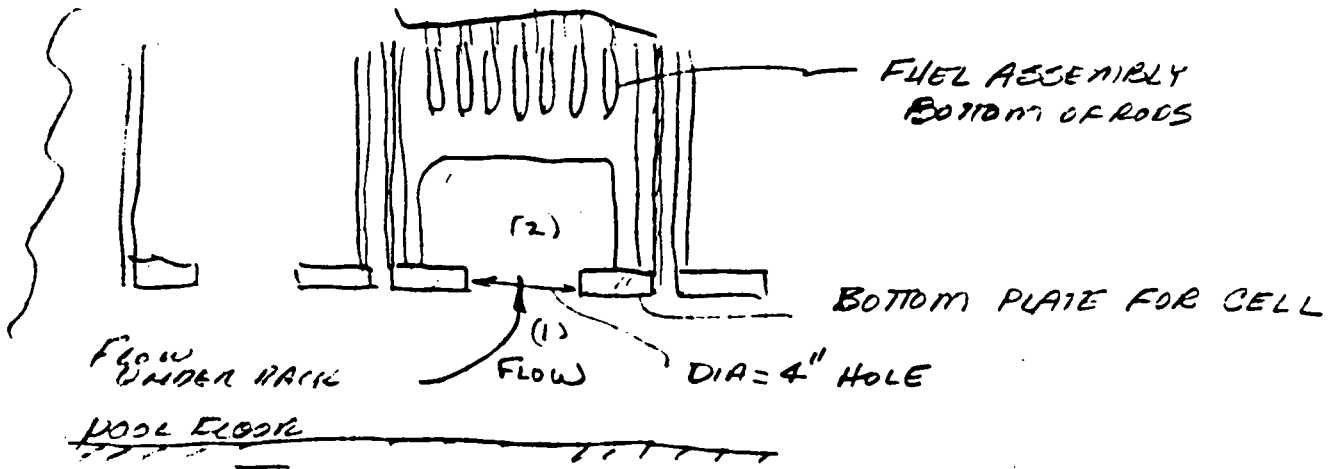
OR $\Delta P_{FUEL} = 48.72 \left(\frac{Q}{A_f} \frac{D_e \rho}{\mu}\right)^{-0.069} \frac{Q^2}{A_f^2} \frac{\rho}{2g}$

FOR; $D_e = 0.0494$ FT $\quad A_f = 53.58/193\ FT^2 = .277\ FT^2$

$$\Delta P_{FUEL} = 48.72 \left(\frac{0.0494}{.277 \times 5.8 \times 10^{-6}}\right)^{-.069} \frac{Q^{1.931}}{(.277)^2} \frac{61.2}{2 \times 32.2}$$

(10) $\Delta P_{FUEL} = 295.76\ Q^{1.931}$, (Q in FT^3/SEC)

PRESSURE LOSS INTO RACK CELL



THE PRESSURE LOSS INTO THE RACK CELL FROM (1) TO (2) WILL CONSIST OF A CONTRACTION PLUS AN EXPANSION LOSS; $K \approx 1.5$

$$\Delta P_{1-2} = 1.5 \frac{V^2}{2g} \rho \quad \text{OR} \quad 1.5 \left[\frac{Q^2}{A_f^2}\right] \frac{\rho}{2g}$$

THE HOLE DIAMETER IS 4"

THEREFORE; $A_f = \frac{\pi}{4}(4)^2 \cdot \frac{1}{144} = 0.08726 \text{ FT}^2$

$$\text{THEN } \Delta P_{1-2} = 1.5 \frac{Q^2}{(.08726)^2} \frac{61.2}{} \frac{1}{32.2 \times 2}$$

(11) $$\Delta P_{1-2} = 187.2\, Q^2, \text{ psf}, \quad Q = \text{FT}^3/\text{SEC}$$

THE TOTAL PRESSURE LOSS FOR FUEL AND CELL ENTRANCE IS;

(12) $$\Delta P_{TOTAL} = \underbrace{295.76\, Q^{1.931}}_{\text{(FUEL BUNDLE)}} + \underbrace{187.2\, Q^2}_{\text{(INLET TO RACK)}}$$

NOTE! RACK EXIT LOSSES ARE ASSUMED TO BE PART OF THE FUEL BUNDLE LOSSES.

FRICTION AND GRID LOSS IN THE BUNDLE SHOULD BE THE SAME FOR UP FLOW OR DOWN FLOW AND MAKE ALMOST ALL OF THE PRESSURE LOSS.

Assuming the flow down the fuel assembly has the same pressure loss as flow up the assembly, and the turn losses are small, we multiply eq. (12) by 2 and then 1.5 (to cover these uncertainties).

$$\Delta P_{LOSS} \cong 2(1.5)\{295.76\,Q^{1.931} + (187.2)\,Q^2\}$$

$$\Delta P_{LOSS} = 887.3\,Q^{1.931} + 561.6\,Q^2 \quad (Q \text{ in cfs}) \quad \text{psf}$$

(13)

From eq. (6), the driving pressure will be

(14) $$\Delta P_d = \beta \frac{\rho L}{2 c_p Q} (g/g_c)$$

where $\beta = -\frac{1}{\rho}\frac{d\rho}{dT}\Big|_{IN}$ is the coefficient of volume expansion at T_{IN}.

From ref E1, thermal properties of interest are tabulated below:

T (°F)	ρ ($\frac{lb}{ft^3}$)	c_p ($\frac{BTU}{lb\,°F}$)	ν ($\frac{ft^2}{sec}$)	K ($\frac{BTU}{hr\,ft\,°F}$)	β (1/°F)
125	61.7	1.00	5.8×10^{-6}	.376	2.6×10^{-4}
150	61.2	1.00	4.8×10^{-6}	.384	3.1×10^{-4}
175	60.7	1.00	3.9×10^{-6}	.390	3.6×10^{-4}

Using $T_{IN} \approx 125°F$ to be conservative (β denoted)

ORDER OF MAGNITUDE ESTIMATES OF PRESSURE LOSSES

NOTE: PATH (1) FLOW RATES ARE ONLY USED AS AN ORDER OF MAGNITUDE ESTIMATE FOR Re (etc).

1. SOUTH WALL FRICTION

BASED ON LOCAL PATH (1) CALCULATIONS FOR EACH HOT ASSEMBLY, $Q \sim .0362\ FT^3/SEC$

THEN, TOTAL FLOW RATE

$$Q_{TOTAL} \cong 193(.0362\ FT^3/SEC) = 6.98\ FT^3/SEC$$

THE VELOCITY BETWEEN THE RACKS AND SOUTH WALL WOULD BE

$$V = \frac{Q_T}{A_f} = \frac{6.98\ FT^3/SEC}{8.27\ FT^2} = 0.84\ FT/SEC$$

THE REYNOLDS NUMBER BASED ON 150°F WATER FLOWING

$$Re = \frac{VD_e}{\nu} = \frac{0.84\ FT/SEC \times 0.682\ FT}{4.8 \times 10^{-6}\ FT^2/SEC}$$

$$Re = 119{,}350 \quad \text{(TURBULENT FLOW)}$$

FROM MOODY CURVE REF (D2)

$$f \cong 0.184\ Re^{-1/5}$$

$$f = 0.184\ (119350)^{-1/5} = 0.0177$$

$$\text{THE PRESSURE DROP} = \frac{fL}{D_e}\frac{V^2}{2g}\rho = \frac{.0177 \times 176''/12}{.682\ FT} \times \frac{(.84)^2}{2g} \times 61.7$$

$$= 0.258\ psf$$

$$\therefore \Delta p = \frac{0.0177 \times 176/12}{.682} \times \frac{61.7}{2 \times 32.2}\left(\frac{193\ Q}{8.27}\right)^2 = 198.6\ Q^2\ psf$$

2. BEND LOSS AT BOTTOM OF SOUTH WALL

$$\Delta p = K \frac{V^2}{2g} \rho \qquad K = 1.0 \quad \text{REF (S2)}$$

$$V = 0.84 \text{ FT/SEC}$$

$$\Delta p = \frac{1.0 (.84 \text{ FT/SEC})^2}{2g} 61.7 \frac{\text{LB}}{\text{FT}^2} = 0.67 \text{ psf}$$

$$\Delta p = \frac{1.0 \times 61.7}{2g} \left(\frac{193\,Q}{8.27}\right)^2 = 521.8\,Q^2 \text{ psf}$$

3. EXPANSION CONTRACTION LOSSES UNDER RACKS.

VELOCITY BETWEEN RACK AND POOL FLOOR

$$V = \frac{Q}{A_f} = \frac{6.98 \text{ FT}^3/\text{SEC}}{21.7 \text{ FT}^2} = 0.32 \text{ FT/SEC}$$

TOTAL RESISTANCE FROM CONTRACTION AND EXPANSION BETWEEN SUPPORTS.

REF (T3) CHEMICAL ENG. DESKBOOK P174

FOR SUDDEN CONTRACTION

$$K_C = 0.5(1-\beta^2) \qquad \beta = \text{DIAMETER RATIO}$$

$$K_C = 0.5(1-.76) = 0.12 \qquad \text{FOR OUR CASE } \beta^2 = 0.76$$

FOR SUDDEN EXPANSION

$$K_E = (1-\beta^2)^2 = (1.0-.76)^2 = .0576$$

$$K_C + K_E = 0.12 + .0576 = 0.178$$

5. FLOW BRANCHING / MOMENTUM LOSSES

THESE TERMS ARE SMALL AND POSITIVE AND WILL BE NEGLECTED.

6. ENTRANCE TO CELLS

$$\Delta P = 187.2\,Q^2$$ FROM PATH(1) CALCULATION

7. FUEL ASSEMBLY

$$\Delta P = 295.7\,Q^{1.931}$$ FROM PATH(1) CALCULATION

FOR CONSERVATISM THE PATH(2) PRESSURE LOSSES WILL BE DOUBLED AND THEN THE TOTAL PRESSURE LOSSES WILL BE EQUATED TO THE DRIVING HEAD.

$$2\left(198.6\,Q^2 + 521.8\,Q^2 + 137.3\,Q^2 + 187.2\,Q^2\right) + 295.7\,Q^{1.931} = \frac{7.973\times10^{-2}}{Q}$$

OR $$2089.9\,Q^2 + 295.7\,Q^{1.931} = \frac{0.07973}{Q}$$

$$Q = 0.0318 \ \mathrm{FT^3/SEC}$$

CHECK $2.49 = 2.50$ OK

$$T_{OUT} = T_{IN} + \frac{1.84\times10^5 \ \frac{BTU}{HR} \times \frac{1}{3600} \ \frac{HR}{SEC}}{61.7 \ \frac{LB}{FT^3}\left(1.0 \ \frac{BTU}{LB\,°F}\right)\left(.0318 \ \frac{FT^3}{SEC}\right)}$$

$$T_{OUT} = T_{IN} + 26.0, \ °F$$

FOR THIS CASE, $T_{IN} = T_{HAVE}$, $T_{HOUT} = T_{Hin} - \frac{q}{C_H}$

$C_H = 1.1 \times 10^6$

CASE	T_{HIN}, °F	q, BTU/HR	T_{HOUT}, °F	$T_{HAVE} = \frac{T_{Hin} + T_{HOUT}}{2}$
1	140	13.99×10^6	127.2	133.6
	150	17.48×10^6	134.1	142.1
2	200	35.00×10^6	168.2	184.1
3	140	13.99×10^6	127.2	133.6
	150	17.48×10^6	134.1	142.1
4	200	35.00×10^6	168.1	184.1

THUS, $T_{OUT} = T_{HAVE} + 26.0$, °F

CASE	T_{POOL}^{MAX}, °F	T_{OUT}, °F
1	140	159.6
	150	168.1
2	200	210.1
3	140	159.6
	150	168.1
4	200	210.1

USING THE SAME HEAT TRANSFER COEFFICIENT DETERMINED FOR PATH (1), $h = 46.2 \frac{BTU}{HR\,FT^2\,°F}$

$$\frac{\pi}{L} z_{max} = \tan^{-1}\left(-\frac{2(1068)}{46.2(26)}\right) = \tan^{-1}(-1.778) = 1.87$$

$$z = \left(\frac{\pi}{12}\right)^{-1}(1.87) = 7.16 \text{ ft} > L/2$$

$$T_{CLAD\,MAX} = T_{in} + \frac{26.0}{2}\left(1 - \cos(1.87)\right) + \frac{106.8}{46.2}\sin(1.87)$$

$$= T_{in} + 16.8 + 22.0 = T_{in} + 38.8$$

CASE	T_{POOL}^{MAX}, °F	$T_{in} = T_{HAVE}$, °F	$T_{CLAD\,MAX}$, °F
1	140	133.6	172.5
	150	142.1	180.9
2	200	184.1	222.9
3	140	133.6	172.5
	150	142.1	180.9
4	200	184.1	222.9

FOR ALL CASES $T_{CLAD} < T_{SAT}$, HENCE THERE WILL NOT BE ANY LOCAL BOILING.

SUMMARY TEMPERATURES, °F

CASE	T_{POOL}^{MAX} °F	PATH (1) T_{OUT} °F	PATH (1) $T_{CLAD\,MAX}$, °F	PATH (2) T_{OUT} °F	PATH (2) $T_{CLAD\,MAX}$ °F
1	140	162.9	177.2	159.6	172.5
	150	172.9	187.2	168.1	180.9
2	200	222.9	237.2	210.1	222.9
3	140	162.9	177.2	159.6	172.5
	150	172.9	187.2	168.1	180.9
4	200	222.9	237.2	210.1	222.9

NOTE: PATH (2) FLOW RATES ARE LOWER THAN PATH (1) HOWEVER, T_{IN} TO THE HOT ASSEMBLY IS ALSO LOWER. THUS, THE CLAD MAXIMUM TEMPERATURE ARE LOWER FOR PATH (2) THAN PATH (1).

APPENDIX - 8709-04 -, GAMMA HEATING OF THE FUEL BOX WALLS, POISON SLABS, AND WATER BOX OF REGION 1

HIGH ENERGY GAMMA RAYS EMITED BY DECAYING SPENT FUEL MAY ESCAPE THE FUEL BOUNDARIES AND DEPOSIT ENERGY IN THE 304 SS BOX, POISON "SLAB", AND WATER BOX WATER. THE WATER BOX WATER MUST HAVE SUFFICIENT FLOW TO INSURE BOILING DOES NOT OCCUR. THIS ANALYSIS DEMONSTRATES THAT ADEQUATE FLOW (ADEQUATE COOLING) IS AVAILABLE.

AT COOLING TIMES TYPICAL FOR SPENT FUEL POOLS FISSION PRODUCT DECAY ACCOUNTS FOR MORE THAN 95% OF THE RESIDUAL HEAT. BASED ON REF. (P1) THE GAMMA FRACTION FOR THE MOST RECENT REFUELING DISCHARGE CAN BE TAKEN AS;

$$f_\gamma = 0.62,$$

ESSENTIALLY INDEPENDENT OF REACTOR POWER LEVEL AND OPERATING TIME. THE MAJORITY OF GAMMA HEATING IS DUE TO GAMMA RAYS WITH INITIAL ENERGIES LESS THAN 1 MeV. AS COOLING TIMES INCREASE

f_γ BECOMES SMALLER AND THE GAMMA SPECTRUM BECOMES SOFTER (LESS ENERGETIC).

THE FOLLOWING TABLE TAKEN FROM REF. E2, PP 135-136 SHOWS THE VARIATION OF μ AND μ/ρ (μ = LINEAR ATTENUATION COEFFICIENT, ρ = DENSITY OF MATERIAL) FOR MATERIALS OF INTEREST IN THE SPENT FUEL POOL FOR A GAMMA RAY OF ENERGY 1 MeV.

MATERIAL	ρ (gm/cm^3)	μ (cm^{-1})	μ/ρ (cm^2/gm)
WATER, H_2O	1.0	.0706	.0706
BORON CARBIDE, B_4C	2.6	.150	.060
STAINLESS STEEL, Fe	7.86	.470	.060
URANIUM, U	18.7	1.46	.078
ALUMINUM, AL	2.7	.16	.059

THE AVERAGE $\mu/\rho = 0.065$ cm^2/gm ($\pm$ 13% S.D) FOR THESE MATERIALS, INDICATING THAT IN A UNIFORM GAMMA FLUX, ABSORPTION IS NEARLY PROPORTIONAL TO A MATERIALS MASS FRACTION.

FOR THE UO_2 FUEL, $\rho = 644$ LB/FT^3 = 10.3 gm/cm^3 SO $\mu \cong 10.3(.067) = 0.7$ cm^{-1}. THE MEAN FREE PATH OF A 1MEV γ-RAY IN UO_2 FUEL WILL THEN BE $= \frac{1}{\mu} = 1.4$ cm $= 0.56$ in, COMPARABLE TO

THE RANGE OF A TYPICAL 1MeV ELECTRON IN UO_2 IS:

$$0.412 \frac{gm}{cm^2} \times \frac{cm^3}{10.3 gm} \times \frac{in}{2.54 cm} = 0.016''$$

WHICH IS << 0.373" = DIAMETER OF FUEL PELLET.

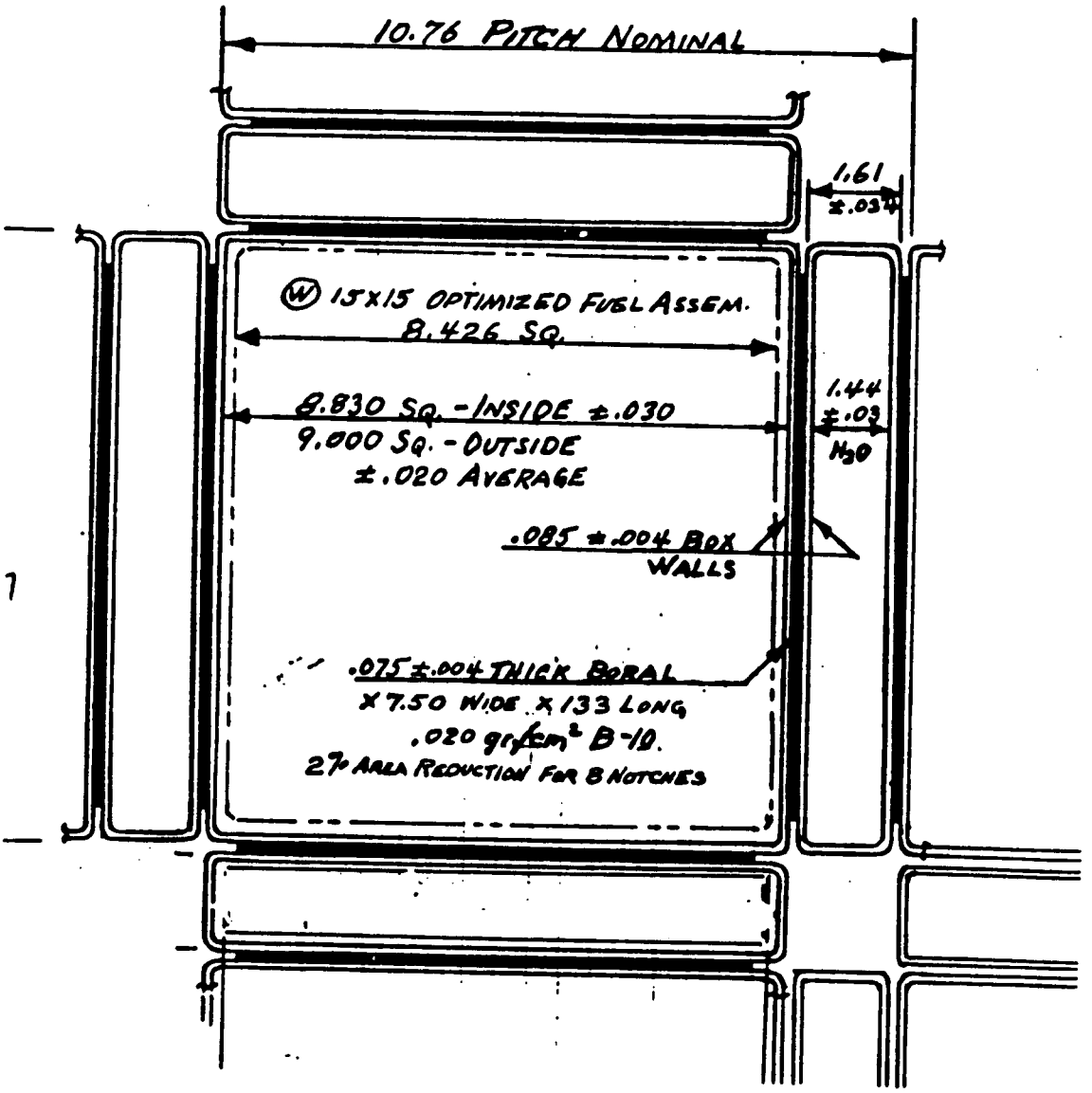
THUS, ALL e^- ARE EITHER STOPPED IN THE FUEL OR SURROUNDING CLAD. THEY HAVE A NEGLIGIBLE direct HEATING EFFECT ON MATERIALS OUTSIDE THE FUEL ASSEMBLY BOUNDRIES.

REGION 1 RACK CELL AND WATER BOX DETAILS ARE GIVEN IN THE FIGURE ON THE NEXT PAGE.

BOX MAT'L: 304SS, $\rho = 501$ LB/FT^3

POISON MATERIAL: BORAL, K = .128 BTU/HR-FT (KEF VI)

REGION 1
(FIXED POISON)



THE CROSS-SECTIONAL AREAS FOR THE UNIT CELL WILL BE AS FOLLOWS:

1. UO_2 FUEL

$$A_{UO_2} = 204 \frac{\pi}{4}(.373)^2 = 22.29\ in^2 = 0.154\ ft^2$$

2. ZIRCALOY CLAD

$$A_{ZR} = 15 \times 15 \frac{\pi}{4}\left(.422^2 - .373^2\right) = 6.88\ in^2 = 0.0478\ ft^2$$

3. RACK CELL WALL

$$A_{WALL} = 9.00^2 - 8.83^2 = 3.03\ in^2 = 0.021\ ft^2$$

4. WATER BOX WALLS - 2 BOXES

$$A_{WB_{WALL}} = \{(9.00)(1.61) - (8.83)(1.44)\}\,2 = 3.55\ in^2 = .0246\ ft^2$$

5. POISON SHEETS (4)

$$A_P = 7.5'' \times .075'' \times 4 = 2.25\ in^2 = 0.0156\ ft^2$$

6. WATER IN WATER BOX (2)

$$A_{H_2O} = 1.44''(8.83'')(2) = 25.43\ in^2 = 0.176\ ft^2$$

7. WATER IN RACK CELL

$$A_{H_2O} = 8.83^2 - \underset{(FUEL)}{22.29} - \underset{(CLAD)}{6.88} = 43.[illegible]\ in^2 = .[illegible]\ ft^2$$

CONTRIBUTION	ρ (LB/FT³)	A (FT²)	ρA (LB/FT)
UO_2 FUEL	644	.154	99.1
ZIRCALOY CLAD	409	.0478	19.5
FUEL CELL WATER	61.7	.339	20.9
RACK CELL S.S.	501	.021	10.5
BORAL POISON	169	.0156	2.63
WATER IN WATER BOX	61.7	.176	10.8
WATER BOX WALLS	501	.0246	12.3

TOTAL = 175.73

WE REQUIRE THE COOLANT IN THE WATER BOX TO REMOVE HEAT FROM THE WATER BOX WALLS, BORAL POISON, AND THE WATER IN THE BOX ITSELF.

THE FRACTION OF THE TOTAL HEAT WILL BE EQUAL TO THE MASS FRACTIONS OF MATERIALS MENTIONED

$$f_{HR} = \frac{12.3 + (2.63) + 10.8}{175.73} = 0.146$$

OR PER WATER BOX $= \frac{f_{HR}}{2} = 0.073$

IN LINE WITH ASSUMPTION 1 ON PAGE (3) THE WATER BOX WATER MUST REMOVE,

$$q = 0.073\ (0.62)(1.84\times10^5)\ \frac{BTU}{HR} = 0.6832\times10^5\ \frac{BTU}{HR}$$

FROM REF (T3)

CONTRACTION LOSS = $K_C = 0.5(1-\beta^2) = 0.5(1-.069) = 0.46$

EXPANSION LOSS = $K_E = (1-\beta^2)^2 = (1-.069)^2 = 0.866$

TOTAL LOSS = $K_T = K_C + K_E = .46 + .866 = 1.32$

ASSUME TOP LOSS SAME AS BOTTOM.

THEREFORE:

$$\Delta p = 2(1.32)\frac{V^2}{2g}\rho = 2(1.32)\left(\frac{Q}{.00613}\right)^2 \times \frac{61.7}{2g}$$

$$\Delta p = 6.746\times10^4\, Q^2$$

(3.) FRICTION LOSS – ASSUME SMALL NEGLECT

THUS, EQUATING LOSSES WITH DRIVING HEAD

$$1.73 + 6.746\times10^4 Q^2 = \frac{.0036}{Q}, \quad psf$$

$$Q = 0.00185 \ FT^3/SEC \qquad Ch\ 1.96 = 1.945$$

THE WATER TEMPERATURE RISE WILL BE:

$$\Delta T = \frac{\delta}{\rho c_p Q} = \frac{0.0832\times10^5 \frac{BTU}{HR}}{61.7\frac{LB}{FT^3}\times 1\frac{BTU}{LB\,°F}\times .00185\frac{FT^3}{SEC}\times 3600\frac{SEC}{HR}}$$

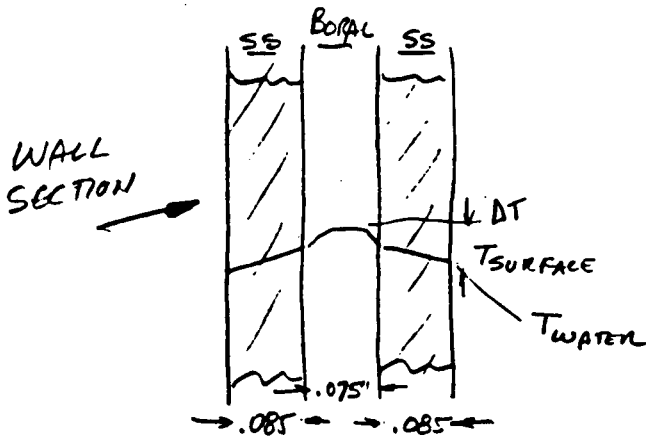
$$\Delta T = 20.2\ °F$$

$$T_{OUT} = T_{IN} + \Delta T$$

$$= 200 + 20.2 = 220.2\ °F \quad < T_{SAT} \cong 241\ °F$$

THEREFORE, THERE WILL BE NO BOILING IN THE WATER BOXES EVEN WITH 200°F INLET WATER (IE MAXIMUM POOL TEMPERATURE)

POISON TEMPERATURES - RACK WALL



THE FRACTION OF HEAT GENERATED IN THE POISON MATERIAL CAN BE ESTIMATED FROM THE MASS FRACTIONS ON PAGE (7) 8721-04

HERE THE BORAL AND SS WILL BE CONSIDERED

$$f_{HR} = \frac{2.63 + 10.5 + (12.3)(.5)}{175.73} = .109$$

PER BOX WALL THE HEAT GENERATED WOULD BE:

$$q = \frac{0.109(.62)(1.84\times10^5)\ \frac{BTU}{HR}}{4\ SIDES} = 3.129\times10^3\ \frac{BTU}{HR}$$

THE TEMPERATURE RISE THRU THE SS AND BORAL CAN BE APPROXIMATED AS:

$K_{BORAL} = 128$ PER (V1)

$$\Delta T \cong q\left[\frac{\Delta x_{SS}}{K_{SS}} + \frac{\frac{\Delta x}{2}_{BORAL}}{K_{BORAL}}\right]$$

$$\Delta T = 3.129\times10^3\left[\frac{0.085/12\ FT}{8\ \frac{BTU}{HR\,FT\,°F}} + \frac{\frac{.075}{2(12)}\ FT}{128\ \frac{BTU}{HR\,FT\,°F}}\right]\ \frac{BTU}{HR},\ °F$$

$$\Delta T \cong \underline{2.84\ °F}$$

THIS TEMPERATURE RISE WHEN ADDED TO THE WATER TEMPERATURE FROM PREVIOUS PAGE YIELD $T_{max} = 2.84 + 220.2 = 223°F \ll T_{SAT}$

THEREFORE, NO BOILING WILL OCCUR IN THE POISON MATERIAL

REGION 2

POISION TEMPERATURES.

REGION 2 POISON TEMPERATURES WILL BE LESS THAN REGION 1 BECAUSE OF

(1) LOWER FUEL ASSEMBLY POWER LEVELS

(2) HIGHER FLOW RATES THAN THE REGION 1 WATER BOX FLOW RATES.

THEREFORE, THERE SHOULD BE NO BOILING IN REGION 2 FUEL CELLS.

Question IV.1

Why the fuel assembly drop height is 20" instead of 36" normally used for such analysis?

Response to Question IV.1

It was determined that it would not be possible to raise a fuel assembly higher than approximately 19 inches over an installed maximum density rack. This determination was based on:

(a) Actual measurements with the manipulator crane in the full up position and a fuel assembly suspended from the fuel handling tool.

(b) The installed rack being at the minimum elevation permitted by the adjustable screw pedestals.

The 20 inch drop height used in the analysis is therefore a bounding condition.

Question IV.2

Since the analysis performed for a straight drop of a fuel assembly through an individual cell shows that the weld between the bottom plate and cell wall fails, provide calculations to demonstrate (a) the integrity of the pool liner to accommodate the dropped fuel assembly, and (b) the integrity of the fuel assembly under the impact.

Response to Question IV.2

(a)

The residual striking energy of the missile (fuel assembly) is much less than the strain-energy of the target (0.25" thick liner plate supported continuously on concrete). Confirmatory calculations were performed for the drop of the fuel assembly through water alone and showed that the liner plate thickness would not be penetrated. The Ballastic Research Laboratory formula for steel target thickness was used. This confirms that the integrity of the pool liner would be maintained if a fuel assembly is dropped on it.

(b)

Section 14.2.1 of the Indian Point 3 FSAR addresses fuel handling accidents, including analysis of dropping a fuel assembly vertically onto a rigid surface. This analysis indicated that the buckling load on the fuel rods was below the critical buckling load and stresses in the cladding were below yield. The loads induced by dropping a fuel assembly vertically through an individual rack storage cell would be less than the cas analyzed for the FSAR due to the kinetic energy absorbed by the cell bottom plate. The results of the previous FSAR fuel drop analysis, therefore, remain valid.

Question IV.3

Provide the basis or source for the buckling stress equation used for the fuel drop analysis. Also provide Reference #13 which developed the buckling coefficient K_C.

Response to Question IV.3

Attached please find a copy of Reference #13 as requested. The buckling stress equation used for the fuel drop analysis was derived from the classical theory of buckling of thin plates as may be found in "Theory of Elastic Stability" by S.P. Timoshenko and J.M. Gere. The adaptation of the classical theory for the present application is described in detail in the reference provided, under the chapter titled "Theoretical Buckling Predictions, Test Observations and Correlations."

APPENDIX B

FUEL BOX CRUSH TESTS

REPORT AND CALCULATIONS

907F17

-TYPICAL-

FOR FUEL STORAGE RACKS WHICH

HAVE CELL TO CELL CONSTRUCTION

WITH NO INTRACONNECTING STRUCTURE

JANUARY, 1980

SPENT FUEL STORAGE RACKS
FUEL BOX CRUSH TESTS
MAXIMUM DENSITY RACK DESIGN
TYPICAL PWR FUEL

REPORT AND CALCULATION 907F17

COVER SHEET: Conforming to WAI QA-8-75-100, No. Q-6 Section 4.0.

TRACEABLE DESIGN INPUT: ITLS Report No. 27852 (Attached).

PURPOSE: Determine fuel rack response to impact or external loading conditions based on box crush test results for two MDR box sections.

ASSUMPTIONS: Quasi-static compression tests on corner and fully loaded MDR box sections can be used to access the extent of damage in the actual rack array. In particular, some of the box crush test results must be predictable and the four requirements listed on page 2 should be justified.

REFERENCES: Den Hartog, Faupel, Shanley, and Roark listed on page 24.

CONCLUSIONS: The box crush test buckling forces are theoretically predictable within estimated undertainties and the four requirements of page 2 are justified. The actual rack array will buckle near allowable yield limits. The extent of vertical deformation in the rack array can be estimated conservatively high using a specific absorbed energy estimate of 1250 ft-lbs per inch deformation per square inch contact area. Lateral damage to boxes adjacent to the impacted one(s) will typically be small in comparison.

PREPARED BY: Eric R. Frantz 1/2/80
REVIEWED BY: C.W. Pierce 1/2/80
APPROVED BY: J.T. Kungener 1/2/80
This report is applicable to all PWR - MDR designs.

BY ERF 800102
CHKD. BY CWP
907F17

SPENT FUEL STORAGE RACKS
FUEL BOX CRUSH TESTS
MAXIMUM DENSITY RACK DESIGN
TYPICAL PWR FUEL

INTRODUCTION AND SUMMARY

The typical Wachter spent fuel storage box for a PWR maximum density rack (MDR) design measures 9" square and is fabricated from 1/8" thick (.120 - .125") 304 stainless steel sheet. To assess the extent of damage resulting from external impact loadings (e.g., fuel, cask, or gate drop accidents), two MDR box sections approximately 15½" in length were fabricated and compressed quasi-statically at deformation rates ranging from 25 to 200 mils/minute.

The first box was corner loaded over one quandrant and deformed 4" - equivalent to a total absorbed energy of 6000 ft-lbs. The second box was fully loaded in compression and deformed 3" - equivalent to a total absorbed energy of 11,000 ft-lbs.

The forces required for buckling are in good agreement with theoretical predictions and are nearly independent of the box length. Apart from inflections not exceeding 30%, the absorbed energy versus deflection curves are linear, defining an average force 60 to 65% of the buckling force.

TEST CONSIDERATIONS AND CONDITIONS

It is desired to have a relatively simple test that provides information on a variety of external loading

and impact conditions. Although testing of partial rack arrays could yield a considerable amount of information, a large number of arrays would need to be constructed and tested to cover uncertainties and varieties of accident loading conditions. In addition, tonnage capacities and test bed areas limit both the applied forces and the number of fuel boxes that can be tested.

It is recognized that for any postulated accident, the potential energy prior to impact is known or easily calculated. This energy is approximately mgh, where m is the mass of the impacting missile, g the acceleration due to gravity, and h is the height of the drop. Neglecting the bouyancy due to the water in the fuel pool, the rack must absorb this energy plus an additional $mg\delta$ where δ is the vertical deformation. Since δ is generally much smaller than h, it could be neglected as a first approximation and then determined with one iteration.

In order for the crush test results on single boxes to be used to estimate the extent of external loading damage in the actual rack array, some of the crush test results must be theoretically predictable. It is also desirable that the following be established:

1. The extent of damage is locally limited.
2. The absorbed energy is a known and simple (semi-empirical) function of axial deformation.
3. The test results and the actual rack array behavior are not strongly length dependent.
4. The test results can be used for accurate (or conservatively high) estimates of the actual rack array deformation.

A sketch of the two box modules used in the crush tests is illustrated in Figures 1(a) and 1(b). Prior to testing, both boxes measured 15½" in length with an average wall thickness of 0.122". The average outside face to face width was 9.206". The absolute uncertainties in these three dimensions are 0.5%, 1.2%, and 0.3%, respectively. Each box was fabricated from two 304 stainless steel plates (approximately 18" long) bent into half square C-shapes. The mating ends were seam welded over the entire box length. The inside corner radius ranged from 1/16 to 1/8". A 1/2" thick base plate 8.9" square was made to fit freely into the lower end of each box to prevent the sides from collapsing at the base during the compression tests.

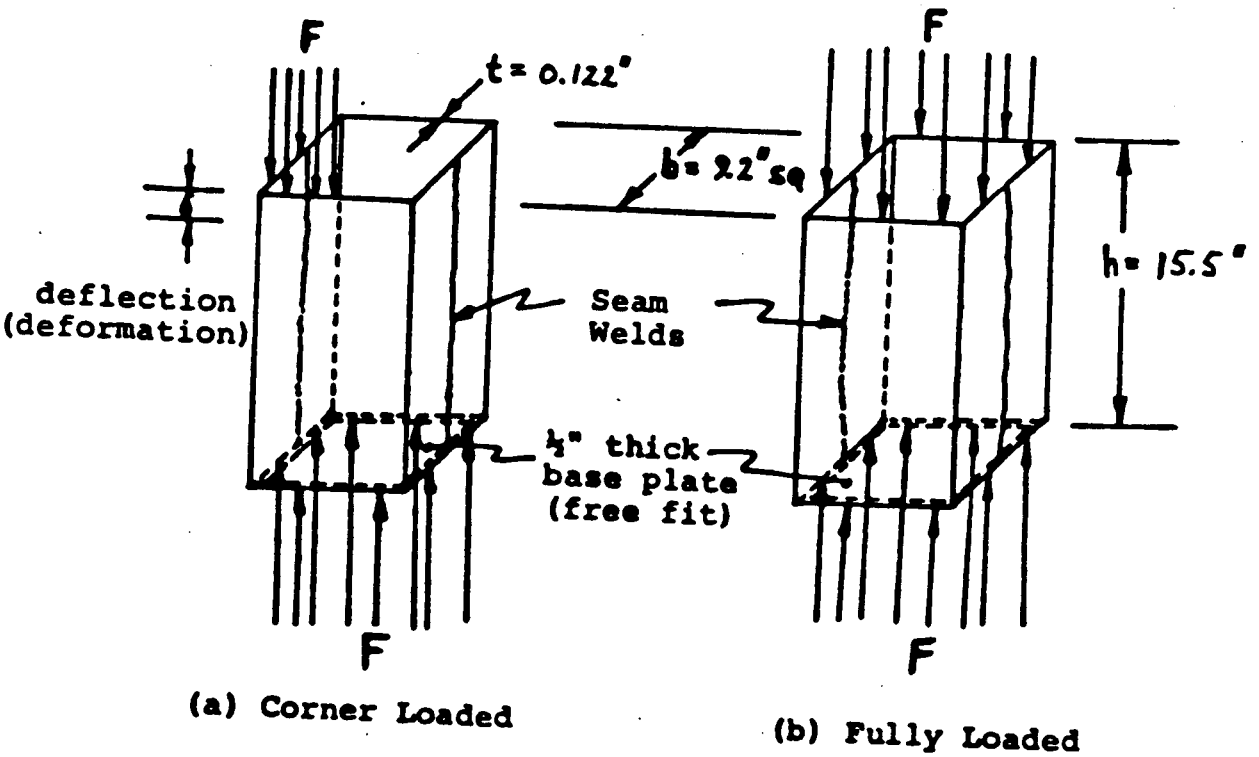


FIGURE 1
BOX AND LOAD CONFIGURATIONS FOR
COMPRESSIVE CRUSH TESTS

The first box was corner loaded in compression over one quandrant at the top of the box, approximately 4.5" along two adjacent sides. The base was fully supported in the direction of the applied force and constrained from bending laterally by the presence of the 1/2" thick plate. This box was axially deformed 4.0" over the loaded corner. The second box was supported at the base in a similar manner but fully loaded at the top. The contact area being four times larger for this box necessitated using a test machine with a larger tonnage capacity (up to 60 tons). This box was compressed 3.1". Deformation rates for both tests ranged from 25 to 200 mils/minute depending upon the frequency at which the force-deflection (or deformation) measurements were taken.

Both tests were conducted during the morning of Dec. 7, 1979 at Industrial Testing Laboratory Services, Inc., RIDC Park, Pittsburgh, PA 15238 under the supervision of Dr. G. T. Horne (ITLS). A Wachter Associates, Inc. (WAI) representative (E. R. Frantz) witnessed the test and independently recorded force-deflection data. The ITLS test reports are attached as an appendix to this report. The WAI data is given on the following two pages. The tabulated energies ($E = \int F d\delta$) were determined by numerically integrating the force-deflection (deformation) data following the trapazoidal method approximation. Plots of F <u>vs</u> δ and E <u>vs</u> δ for both tests are given on the three pages following.

WACHTER ASSOCIATES, INC.

MECHANICAL – NUCLEAR DESIG

BY ERF DATE 791210 SUBJECT

SHEET NO. 5 OF

CHKD. BY CWP DATE 1/2/80

PROJ. NO. 907F17

Data for Corner Loaded MDR Box – ITLS# 27852-1

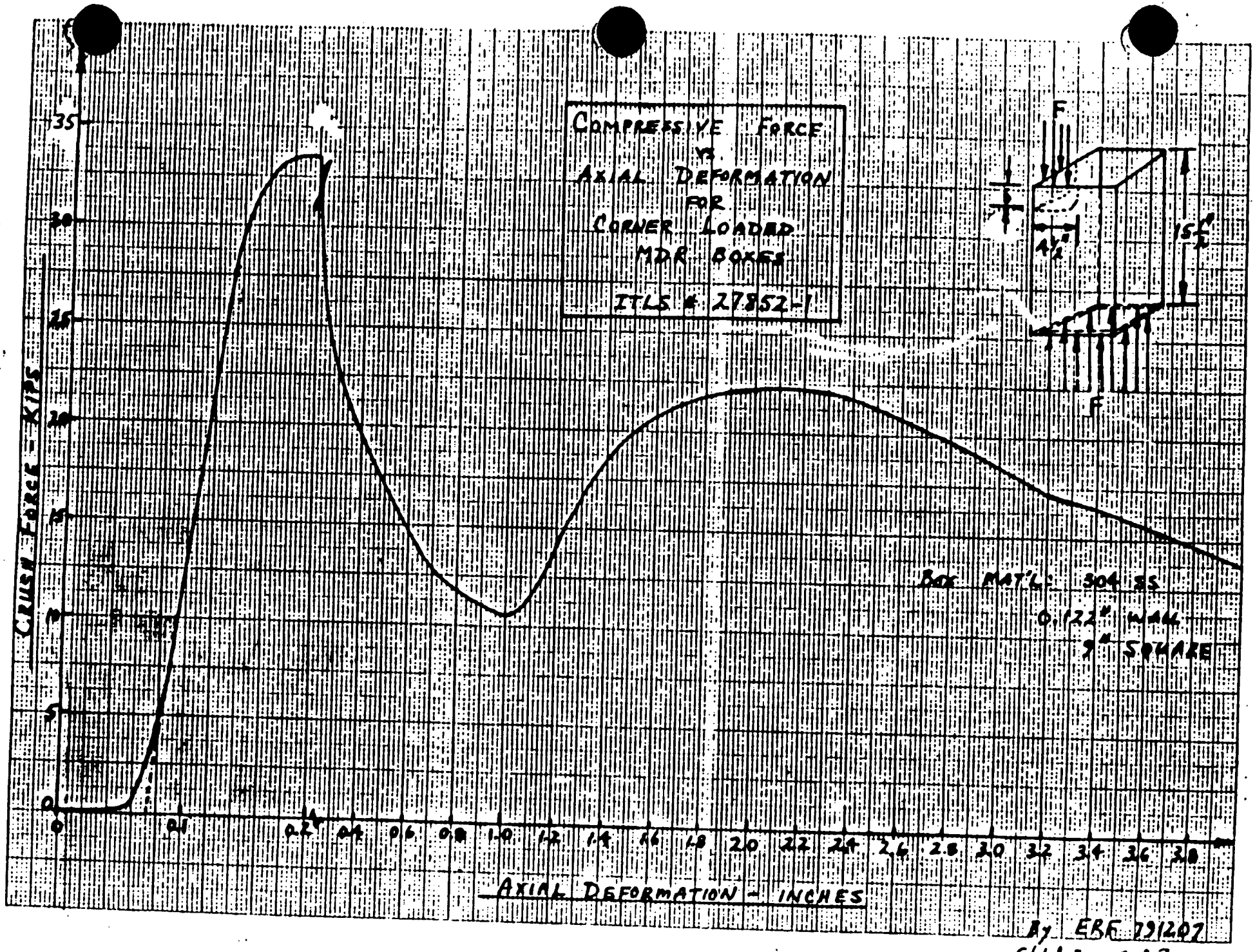
Deflection (in)	Force (lbs)	Energy (ft-lbs)
0.0	50	0.0
0.010	90	0.058
0.030	150	0.258
0.040	190	0.400
0.050	250	0.583
0.055	370	0.713
0.060	560	0.906
0.070	2280	2.090
0.080	5240	5.223
0.090	9000	11.16
0.100	13000	20.32
0.110	17750	33.14
0.120	22250	49.80
0.130	26400	70.07
0.140	29300	93.28
0.150	31200	118.5
0.160	32150	144.9
0.170	33000	172.0
0.180	33450	199.7
0.200	33500	256.
0.210	30500	282.
0.220	29750	307
0.230	27250	331
0.240	26250	353
0.250	25500	375
0.260	24850	396
0.270	24200	416
0.280	23650	436
0.290	23100	456
0.300	22700	475
0.310	22300	494
0.320	21900	512
0.330	21500	530
0.340	21200	548
0.350	20850	565
0.370	20100	599
0.390	19600	633
0.400	19400	649

Deflection (in)	Force (lbs)	Energy (ft-lbs)
0.430	18650	696
0.460	17900	742
0.490	17400	787
0.520	16700	830
0.550	16100	871
0.580	15300	910
0.600	14600	935
0.650	13600	994
0.700	12850	1049
0.750	12250	1101
0.800	11800	1151
0.850	11300	1200.
0.900	11050	1246
0.950	10800	1292
1.000	10500	1336
1.100	11800	1429
1.200	14200	1537
1.300	16400	1665
1.400	18600	1811
1.500	19800	1971
1.600	20700	2139
1.700	21400	2315
1.800	21800	2495
1.900	22000	2677
2.000	22100	2861
2.100	22200	3046
2.200	22200	3231
2.300	22100	3415
2.400	21750	3598
2.500	21350	3778
2.600	20800	3953
2.800	19750	4291
3.000	18500	4610
3.200	17250	4908
3.400	16550	5189
3.600	15700	5458
3.800	14750	5712
4.000	13800	5950

Data for Fully Loaded MDR Box – ITLS #27852-2

Deflection (in)	Force (lbs)	Energy (ft-lbs)
0.0	800	0.00
0.005	1600	0.500
0.010	2300	1.313
0.015	3500	2.521
0.020	4800	4.250
0.025	6100	6.521
0.030	9200	9.708
0.035	13900	14.52
0.040	19000	21.38
0.045	26900	30.94
0.050	37100	44.27
0.055	48500	62.10
0.060	60500	84.81
0.065	71700	112.4
0.070	80000	144.0
0.075	85700	178.5
0.080	89200	215.1
0.085	93100	253.2
0.090	96500	292.7
0.095	98700	333.
0.100	99900	375.
0.105	100800	417.
0.110	101400	459.
0.120	101700	543.
0.130	101000	628.
0.140	98400	711.
0.150	97100	792.
0.170	93500	951.
0.200	88100	1178.
0.225	84100	1358.
0.250	80800	1529.
0.275	77900	1695.
0.300	75500	1864.

Deflection (in)	Force (lbs)	Energy (ft-lbs)
0.350	71200	2160
0.400	67600	2449
0.450	64500	2724
0.500	61500	2987
0.600	56000	3476
0.700	51500	3924
0.800	47900	4339
0.900	45000	4726
1.000	42400	5090
1.100	40600	5436
1.200	38800	5766
1.300	37000	6082
1.400	35300	6384
1.500	33700	6671
1.600	32200	6946
1.700	30900	7209
1.800	29800	7461
1.900	28900	7706
2.000	28300	7944
2.100	29000	8183
2.200	30400	8431
2.300	32900	8694
2.400	34200	8974
2.500	35200	9263
2.600	36100	9560
2.700	37200	9866
2.800	38200	10180
2.900	39400	10503
3.000	40200	10835
3.100	41000	11173



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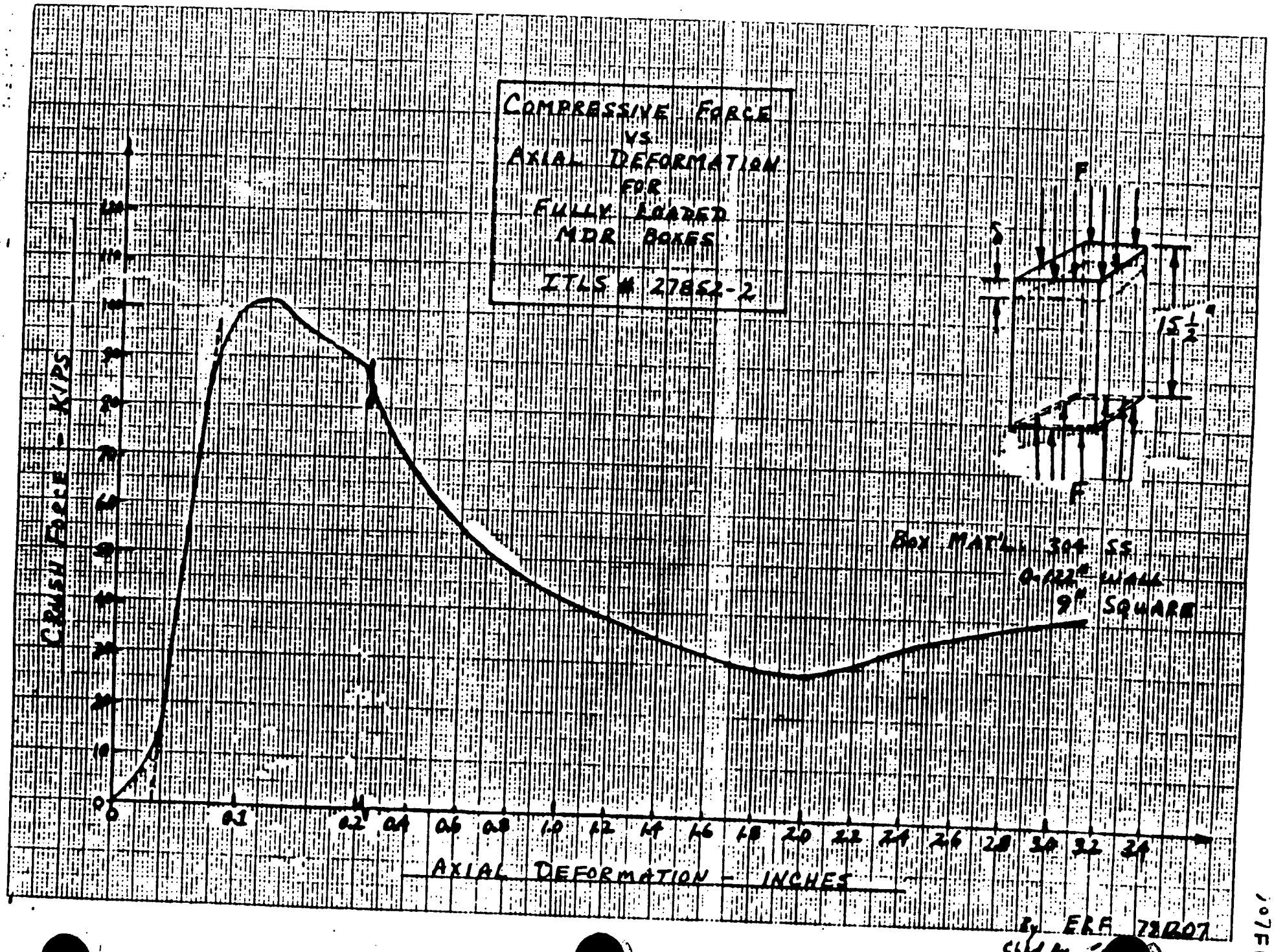
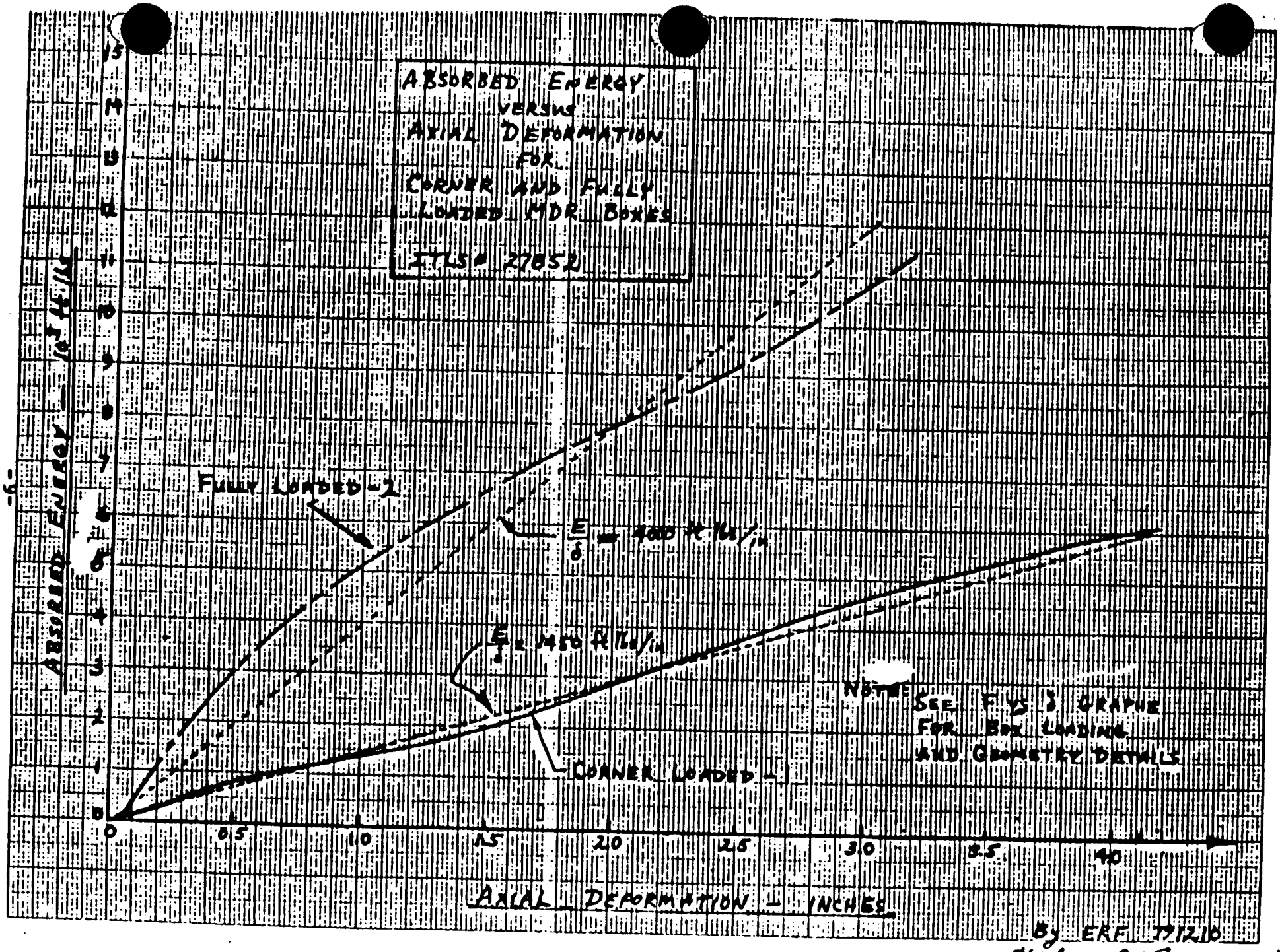
K-E 10 X 10 TO THE CENTIMETER 18 X 25 CM
KEUFFEL & ESSER CO. MADE IN U.S.A.


COMPRESSIVE FORCE
VS
AXIAL DEFORMATION
FOR
FULLY LOADED
MDR BOXES
ITLS # 27852-2
CRUSH FORCE - KIPS
AXIAL DEFORMATION - INCHES
0
10
20
30
40
50
60
70
80
90
100
110
120
0.1
0.2
0.4
0.6
0.8
1.0
1.2
1.4
1.6
1.8
2.0
2.2
2.4
2.6
2.8
3.0
3.2
3.4
F
δ
15 1/2"
Box MAT'L 304 SS
0.122" WALL
9" SQUARE
By ERF 78R07



By ERF 7/1/10
Chkd by CASP

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K-E 10 X 10 TO THE CENTIMETER 18 X 25 CM
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CHKD. BY CWD
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THEORETICAL BUCKLING PREDICTIONS, TEST OBSERVATIONS, AND CORRELATIONS

The exact physical behavior of the corner located box is difficult to justify accurately since its locally limited buckling lacks well-defined boundary conditions. Since the buckling behavior of the fully loaded box is better understood theoretically, the test results for it will be discussed first.

From the force-deformation curve for the fully loaded box, buckling is evident at a crush force of 80,000 lbs. With $A_c = 4tb = 4(.122)(9.206-.122) = 4.43\ in^2$, the corresponding buckling stress is 18,000 psi. This stress is approximately 40% of the yield stress*. This buckling stress or force is best explained by treating the test box as four thin plates, simply supported (i.e., fixed in position but free to rotate) at the edges. More complete treatments of the buckling theory to be presented can be found in reference 1 (pp. 301-305) and 2 (pp. 542-546).

The deflection of a thin plate of width b and height h, simply supported at the edges y = 0 and y = b and loaded in compression and simply supported (pinned) at x = 0 and x = h will deform according to

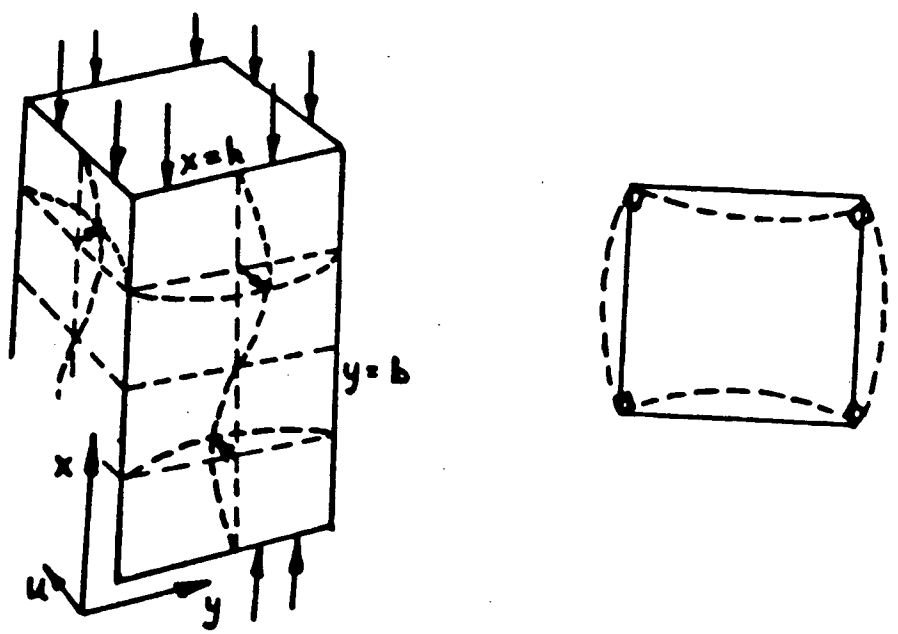
$$u(x,y) = \sum_{m,n} A_{mn} \sin\left(\frac{m\pi x}{h}\right) \sin\left(\frac{n\pi y}{b}\right) \qquad (1)$$

---

* Room temperature yield strengths for the 304 stainless steel sheet used for fuel box fabrication range from 40,000 to 50,000 psi. An average S_y = 45,000 psi is assumed when comparing the crush test results to theoretical predictions. Due to cold working, these yield strengths exceed the ASME III minimum S_y = 30,000 psi (at 100 F).

where u(x,y) is the deflection as measured normal to the plate surface. The mode numbers m and n, subject to the given boundary conditions, are restricted to the set of integers, with m, n = 0 giving a trivial solution. This m, n half-sine wave expansion, subject to u = 0 on the plate edges, forms a complete set of the normal modes.

For h > b, the modes n = ±1, m ≈ ± h/b dominate and a plate with h>>b will buckle in approximately square plates of dimensions b x b. The fully loaded box with b ≈ 9.1" and h = 15.5" can be expected to (and did) buckle predominately in the n = 1, m = 2 mode since h/b = 15.5/9.1 = 1.70 ≈ 2. A sketch of this buckling mode is illustrated in Figure 2.



(a) Face View

(b) Cross-Section View

FIGURE 2

The n = 1, m = 2 buckling mode characterizing the buckling behavior of the fully loaded fuel box module.

The stress required to buckle in the n=1, m mode is given by

$$S_{crit} = \frac{K_m \pi^2 Y}{12(1-\nu^2)} \left(\frac{t}{b}\right)^2 \qquad (2)$$

where

$$K_m = \left(\frac{mb}{h} + \frac{h}{mb}\right)^2 \qquad (2a)$$

Y is Young's modulus (= 28×10^6 psi), and ν is Poisson's ratio (= 0.3). The variation of K_m with the height to width ratio h/b is shown in Figure 3. At h/b = 1.70 and m=2, K_m = 4.1 and the m=2 mode of buckling is more likely than the m=1 mode (with K_m = 5.2). It should be noted that the smallest K_m = 4.0 when m = h/b. For h > b, the largest K_m occurs between h/b = 1 to 2. The m=1, m=2 critical stresses are equal (and largest) when h/b = $\sqrt{2}$ (=1.414) and $K_1 = K_2$ = 9/2 (=4.5). Thus, for h >b, the largest variation in S_{crit} will be 13%.

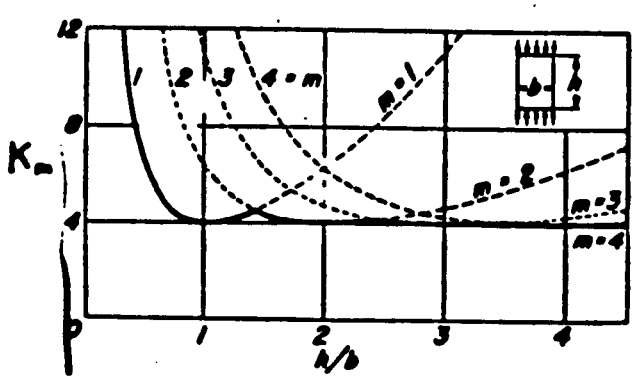


FIGURE 3

Variation of K_m with height to width ratio h/b.

For the fully loaded MDR box, the critical buckling stress predicted by equation (2) is

$$S_{crit} = \frac{(4.1)\pi^2 \ (28\times10^6)}{12[1-(.3)^2]} \left(\frac{.122}{9.1}\right)^2 = 18{,}600 \text{ psi}$$

equivalent to a load of $A_c S_{crit} = (4.43)(18{,}600) = 82{,}400$ lbs. These predictions agree with the test results within less than 4%. Uncertainties in comparing the test results to theoretical predictions exceed this uncertainty. In the force predictions, for example, the accuracy with which the buckling crush force can be established from the F vs δ curve is ~ 5/80 ~ 6%. The combined uncertainty in $A_c S_{crit} \sim K_m t^3 Y/b$ is of order

$$\left[\left(\frac{.05}{4}\right)^2 + 3\left(\frac{.001}{.12}\right)^2 + \left(\frac{1}{28}\right)^2 + \left(\frac{.1}{9}\right)^2\right]^{\frac{1}{2}} \approx 5\%$$

so 10% agreement would be good and 4% exceptional.

The n=1, m=2 mode of buckling was observed in the fully loaded test module. As buckling continued, a limit of 101,700 lbs. was reached at $\delta = 0.120$ inch deflection. After reaching this limit, the full sine wave (m=2) shape became distorted with the two peaks tending to move toward the inflection at $x = h/2$. Plastic flow of this type (i.e. F a decreasing function of δ) continued until $\delta = 2$ inches when the metal "curl over" became limiting. The force then became an increasing fraction of δ up to the test limit $\delta = 3.1$ inches. When the test was completed, the two faces without seam welds had bulged out to 12" at the top of the box approximately 2/3 up from the base and in to 6" near the mid-plane. The sides with seam welds bulged out to 11" at the base approximately 3/8 up from the base and in to 5" near the mid-plane of the box.

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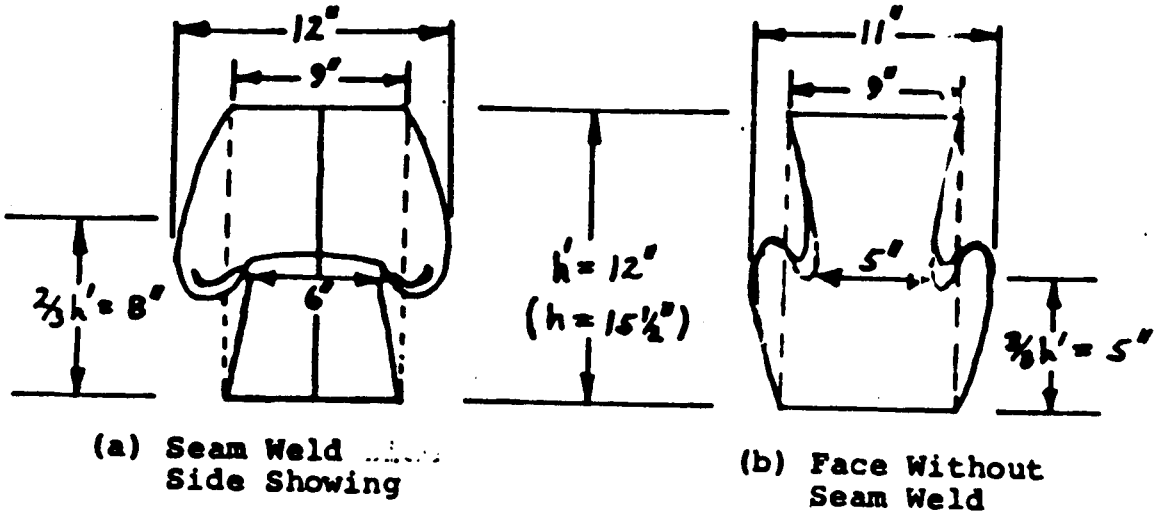

FIGURE 4
Fully Loaded Box Configurations
After Testing

The absorbed energy versus axial deformation curve for the fully loaded box is approximately linear. Apart from variations not exceeding 35%, E/δ = 4000 ft-lbs/in, the maximum percent errors occuring at small δ (~.25 to .5 inch). For δ > 1.5 inches, the proportional relation correlates within 10%. This straight line defines an average force F_{avg} = (4000)(12) = 48,000 lbs and F_{avg}/F_B = 48/80 = 0.60 defines the average force in terms of the buckling force.

The n=1, m=2 mode of buckling was observed on this fully loaded test module and the buckling stress was predictable within 4% by an expression of the form

$$S_{crit} = K_c Y \left(\frac{t}{b}\right)^2 \tag{3}$$

where

$$K_c = \frac{K_m \pi^2}{12(1-\nu^2)} \tag{3a}$$

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For h > b, the limiting $K_m = 4$ and does not exceed 9/2(4.5) assuming the plate edges are simply supported. Thus, with $K_m = 4$ and $\nu = 0.3$, the limiting K_c is

$$K_c \geq 3.62$$

for simply supported edges (free to rotate).

An expression of the form given by eq (3) holds for other boundary conditions. Figure 5 (taken from reference 3, p. 611) illustrates K_c as a function of h/b for fixed edges, simply supported edges, and one free-one supported edges.

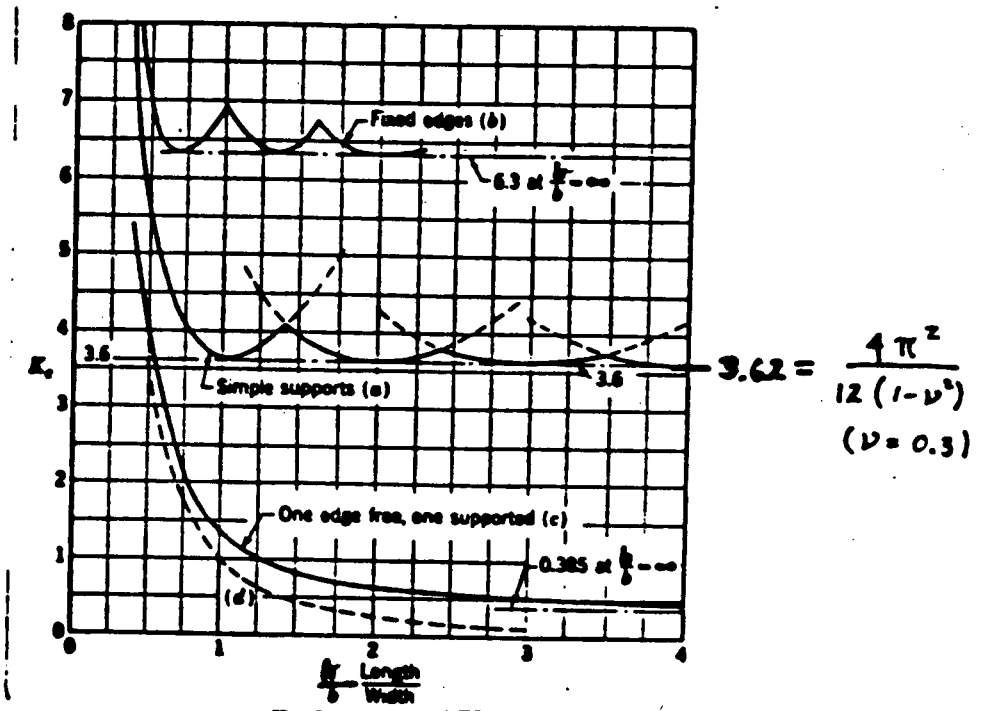


FIGURE 5

Buckling coefficients for flat plates under end compression (Loaded ends simply supported)

In the corner loaded box, the effective K_c is difficult to determine accurately since the load is eccentric and not directed over the full section area.

If the unloaded portion offers some support against rotation, then K_c will likely fall within the range 3.62 to 6.3, where the latter signifies fixed edges. The buckling stress will then fall between

$$(3.6)(28\times10^6)\left(\frac{.122}{9.1}\right)^2 < S_b < (6.3)(28\times10^6)\left(\frac{.122}{9.1}\right)^2$$

or

$$18{,}000 < S_b < 32{,}000 \text{ psi}$$

and since $A_c = bt = 4.43/4 = 1.11 \text{ in}^2$, the test buckling force should fall between

$$20{,}000 \text{ lbs} < F_b < 35{,}000 \text{ lbs.}$$

The observed proportional limit was approximately 27,500 lbs, approximately half way between these theoretical limits, defining an effective $K_c = 5.0$ for the corner loaded box.

As the box buckled, the loaded side with a seam weld deflected inward at the top of the box and the side without deflected outward. Although the seam welded sides in the fully loaded box also deflected inward at the top, there is no reason to suspect that this trend will always occur. In the fully loaded box, neither seam weld has failed. In the corner loaded box, the seam weld tore when $\delta = 1$ inch, far beyond the deflection $\delta = 0.2$ inch at the peak compressive force.

The F <u>vs</u> δ curve for the corner loaded box goes through two relative maxima and one minimum. At the second maximum, F = 22,000 lbs and $\delta = 2.0$ inches. The minimum

at δ = 1 inch is F = 10,500 lbs, approximately half as large. This variation was caused by the metal folding over or bending (which required little force to cause deflection) followed by a compressive loading on the flattened folds (which required larger forces). The force at this second relative maximum (= 22,000 lbs) was approximately 75% of the peak crush force (= 34,000 lbs at δ = 0.20 inch).

It is postulated that the second maxima exhibits characteristics of column buckling due to the eccentric loading since the final box configuration shows a net outward deformation superimposed on the inward-outward plate buckling deformation. The familar secant formula for column buckling (see e.g. Ref. 2, p. 510 or Ref. 4, p. 261) is used to predict this stress denoted S_c,

$$S_y = S_c\left[1 + \left(\frac{ec}{r^2}\right) \sec\left(\frac{h}{2r}\sqrt{\frac{S_c}{Y}}\right)\right] \tag{4}$$

where e is the equivalent eccentricity, c is the distance from the central axis to the position of peak stress, and r is the section radius of gyration. For the principal axes of the square section of side b with thickness $t \ll b$, $r = b/\sqrt{6}$. Taking $e = c = b/2$, the eccentric ratio $ec/r^2 = 3/2$. With S_c = 20,000 psi, Y = 28 x 10^6 psi, h = 15.5", and $r = b/\sqrt{6} = 9.1/\sqrt{6} = 3.7$ in, $\sec\left(\frac{h}{2r}\sqrt{\frac{S_c}{Y}}\right) =$ sec (.056) = 1.00; the critical stress $S_c = (2/5)\ S_y$ = (.4)(45,000) = 20,000 psi. The corresponding force is (20,000)(1.11) = 22,000 lbs, in agreement with the force near the second maxima. (This precise agreement, however, was not expected.)

The total axial deformation of the corner loaded box was 4.0 inches, equivalent to an absorbed energy of 5950 ft-lbs. A plan view of the top of this box in Figure 6 shows that most of this energy was absorbed in deforming the loaded corner with the adjacent unloaded corners laterally bending inward approximately 2". The corner opposite the applied load remained fixed, deflecting less than 1/16" toward the loaded corner. The local lateral deformation due to buckling was approximately 1½" inward and outward from either face, confined primarily to the upper 6" or 7" of the box length.

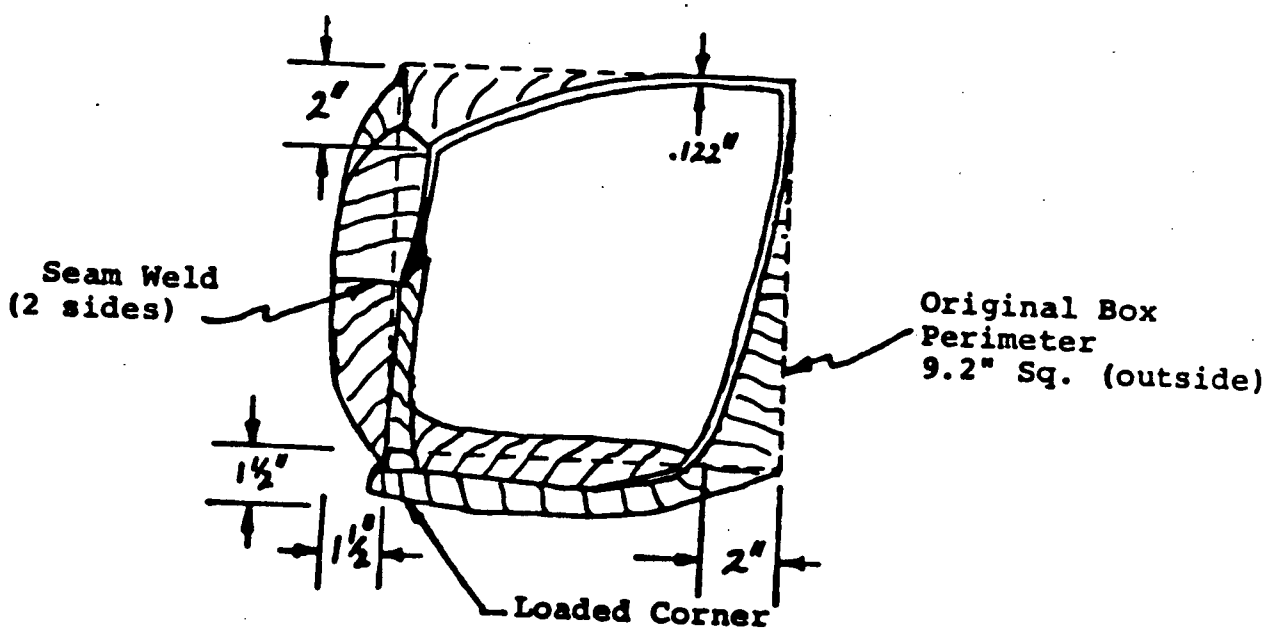


FIGURE 6
Top View Sketch of the
Corner Loaded Box

Despite the erratic behavior of the corner loaded box, the energy curve is very linear with axial deformation. The line E/δ = 1450 ft-lbs/in is accurate within 10% over the entire range of δ, defining an average force

$$F_{avg} = (1450)(12) = 17,400 \text{ lbs.}$$

For this corner loaded box, $F_{avg}/F_b = 17,400/27500 = 0.63$ is the ratio between the average force and the buckling force. (For the fully loaded box, this ratio was found to be 0.60).

Assuming column-like failure accounts for the secondary peaks in the F vs δ curves, the average force (E/δ) should be roughly proportional to the yield stress. The buckling stress, however, is proportional to (t^2/b^2) provided the length $h > b$. Thus, it is postulated that

$$\frac{F_{avg}}{F_b} = C\left(\frac{b}{t}\right)^2 \tag{5}$$

where F_b is the buckling force, F_{avg} is the absorbed energy per unit length (or average force), and the constant C is proportional to the ratio S_y/Y. For additional data in establishing the validity (or usefulness) of the correlation in eq (5), the ITLS tests on four corner loaded conventional PWR poison and non-poison fuel storage boxes are used. A tabulation of this data is given in Table I on the following page.

Box Type+	ITLS #	Loading Scheme	δ_{max} (in)	F_{avg} (lbs)	F_b (lbs)	$\frac{F_{avg}}{F_b}$	C x 100
MDR PWR	27852-1	Corner	4	17400	27500	.633	.1137
b = 9.1"							
t = 0.122"	27852-1	Full	3	48000	80000	.600	.1078
Conventional PWR	24406-1	Corner	0.9	12000	14000	.857	.09102
Poison or Non-Poison	24406-2	Corner	0.9	11000	12000	.917	.09695
b = 8.5"	26791-1	Corner	4	10100	13000	.777	.1004
t = 0.092"	26791-2	Corner*	4	12000	14500	.826	.1074

\+ All boxes 15-16" in length

* With C-channel supports

TABLE 1

Energy, Buckling, and Correlation Data for Box Crush Tests

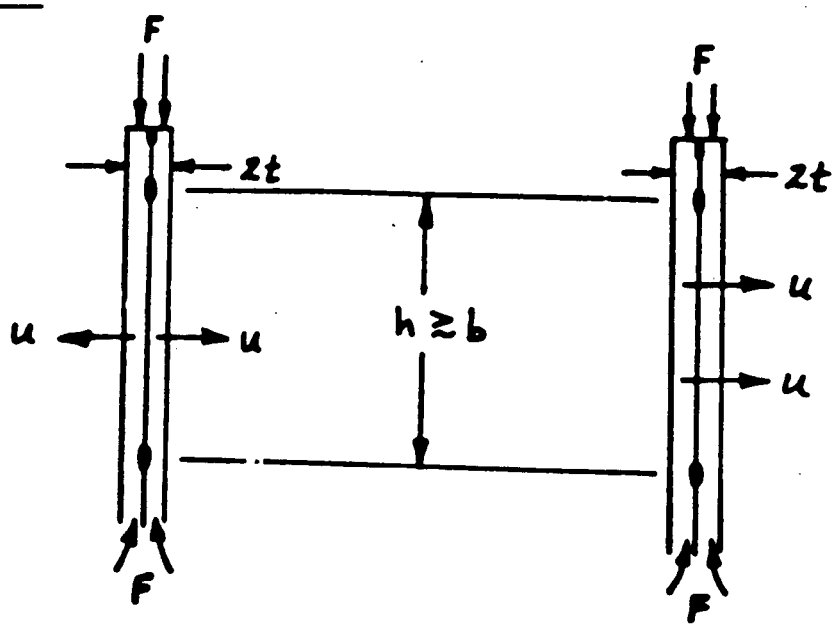
The data for C correlates well. In terms of a one standard deviation uncertainty, $C = .103 \times 10^{-3}$ (± 8%), or roughly $C = 0.10 \times 10^{-3}$ (± 20%).to cover the full range of uncertainties in the test data and graphical interpretation. This correlation may be useful when the effective buckling coefficient (K_c in eq (3) from Fig. 5) is known and the average force is not.

If the thickness to width ratio t/b becomes too large, the eq (5) correlation will likely break-down since the predicted buckling stress would exceed the yield stress. In the next section, the test results for single box crush tests will be used to analyze the actual rack array. On the basis of the single box, buckling stresses comparable to or less than the allowable yield stress are estimated making buckling more likely if actual yield stresses are used. However, forces transmitted to adjacent boxes are implicitly neglected making the extent of actual rack deformation conservatively over-estimated.

APPLICATIONS FOR ACCIDENTAL LOADINGS

A typical MDR fuel rack for PWR spent fuel consists of an array of 100 to 200 boxes, 1/2 to 3/4 of which may house fuel. Pairs of spot welds at 5 to 8 axial locations attach faces of neighboring boxes together. The first and last pairs of spot welds are approximately 2" from the top and bottom of the 13' long box, respectively. The remaining pairs distribute over the box length weighted near the base where stresses are highest. Additional strength is offered by continuous fusion welds around the top and bottom edges of each box interface.

Forces required for plate-like buckling in the upper two inches of the box (between the fusion weld and top pair of spot welds) are excessive since the effective h/b ratio is small. It is more likely that yield would occur there unless the force can be transmitted to other sets of spot welds. Since these sets of pairs are separated by distances h > b, buckling in an inter-rack position may be possible. The boundary conditions here, however, are different than those of the fully loaded test module and weld attached walls are more likely to deflect away from each other instead of moving together as a slab with effective thickness 2t. Referring to Figure 7, the following estimation will establish the reason for this.



(1) Plates deflect away (2) Plates deflect together

Figure 7
Comparison of Buckling Between
Pairs of Spot Welds

To deflect away, the corners at y = 0 and b must be fixed since adjoining walls do not allow for rotations (the slope of u must vanish at the corners). With this boundary condition,

$$S_1 = 6.3Y\left(\frac{t}{b}\right)^2 = (6.3)(28\times10^6)\left(\frac{.122}{9.1}\right)^2 = 32{,}000 \text{ psi} < S_y$$

In Figure 7(2), simply supported edges may be appropriate. With $K_c = 3.6$,

$$S_2 = 3.6Y\left(\frac{2t}{b}\right)^2 = (3.6)(28\times10^6)\left(\frac{.244}{9.1}\right)^2 = 72{,}500 \text{ psi} > S_1, S_y$$

Thus, Figure 7(1) may be physically possible ($S_1 < S_y$) but Figure 7(2) will not be. Note that S_1 is comparable to the allowed yield stress (ASME III, for 304 SS at 100°F allows S_y = 30,000); however, the actual yield stress is

near 45,000 psi, so buckling may occur before yield. Drawing a distinction here is inconclusive but also insignificant.

In the actual rack array, boxes adjacent to the impacted box as well as the impacting missile will absorb some energy, so using the single-box crush test results will lead to over-estimations of the actual deformation. The extent of lateral damage to boxes near the impacted box will be small since the corner opposite to the corner loaded test box deformed insignificantly (less than 1/16" when δ = 4"). Conservative and locally limited over-estimations of axial deformation will result then if $F_{avg}/F_b = 0.6$ for the MDR boxes is assumed. The corner loaded box will be the most limiting one with $K_c = 5$. (Inter-rack or side boxes with fixed boundaries have $K_c > 6$.) Under these assumptions,

$$\frac{F_{avg}}{A_c} = 0.6 \frac{F_b}{A_c} = (0.6)(5)(28\times10^6)\left(\frac{.122}{9.1}\right)^2 = 15{,}000 \text{ psi} = \frac{E}{\delta A_c}$$

or

$E/\delta A_c$ = 1250 ft-lbs/inch deformation/sq. inch contact area.

This relation may be used to conservatively estimate the axial deformation δ given the energy absorbed ($\sim$ mgh) and the impact area A_c. For example, a 2000 lb fuel assembly dropped vertically upon the rack from a height of 2 feet above the rack will penetrate a side or corner position in the rack (with $A_c = 1.11$ in^2) no more than (2)(2000)/(1250)(1.11) = 3 inches. For an inter-rack position, the penetration will not exceed 1.5 inches (edge loading) or 0.8 inch (intersection loading).

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INDUSTRIAL TESTING LABORATORY SERVICES CORP.
635 ALPHA DRIVE - R.I.D.C. PARK
PITTSBURGH, PA. 15238 PHONE 412-781-1080

cc: JFK
ERF

TEST REPORT

PURCHASE ORDER NO. 1381 DATE December 18, 1979

TO: Wachter Associates, Inc. ITLS REPORT NO. 27852
ADDRESS 5408 Wm. Flynn Highway
Gibsonia, PA 15044

ATTENTION Mr. E. R. Frantz/J. F. Klingener

RECEIVED
DEC 20 1979
WACHTER ASSOCIATES, INC.

REPORT OF: Crush Tests of Submitted Box Sections.

RESULTS

We received two (2) 9" square x 15½" long, 1/8" wall hollow sections.

We loaded the first longitudinally over two 4½" long legs of one corner, the other end of the box containing a restraining plate; loading was between parallel flats. We made measurements of load (in lbf) vs deflection (in mils). The loading device was a UTM (S/N 40900, calibrated 10 March 1979

Start 50 lbf (i.e., zero deflection).

Deflection (mils)	Load (lbf)	Deflection (mils)	Load (lbf)
10	90	270	24200
30	150	280	23650
40	190	290	23100
50	250	300	22700
55	370	310	22350
60	560	320	21900
70	2280	330	21500
80	5240	340	21200
90	9000	350	20850
100	13500	360	20500
110	17750	370	20100
120	22250	380	19800
130	26400	390	19400
140	29300	400	19000
150	31200	430	18650
160	32150	460	17900
180	33000	490	17300
190	33450	520	16700
200	33500 (local max.)	540	16100
210	30500	560	15300
220	29750	600	14600
230	27250	650	13600
240	26250	700	12850
250	25500	750	12250
260	24850	800	11800

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Deflection (mils)	Load (lbf)	Deflection (mils)	Load (lbf)
850	11300	2100	22200 (local max.
900	11050	2200	22200
950	10800	2300	22100
1000	10500 (local min.)	2400	21750
1100	11800	2500	21350
1200	14200	2600	20800
1300	16400	2800	19750
1400	18600	3000	18500
1500	19800	3200	17250
1600	20700	3400	16550
1700	21400	3600	15700
1800	21800	3800	14750
1900	22000	4000	13800
2000	22100		

(continued next column)

We loaded the second box over the full cross sectional area; again there was a restraining plate inside the bottom. The loading device here was another UTM (S/N 044-2002, calibrated 10 March 1979).

Start 800 lbf (i.e., zero deflection)

Deflection (mils)	Load (lbf)	Deflection (mils)	Load (lbf)
5	1600	130	101000
10	2300	140	98400
15	3500	150	97100
20	4800	170	93500
25	6100	200	88100
30	9200	225	84100
35	13900	250	80800
40	19000	275	77900
45	26900	300	75500
50	37100	350	71200
55	48500	400	67600
60	60500	450	64500
65	71700	500	61500
70	80000	600	56000
75	85700	700	51500
80	89900	800	47900
85	93100	900	45000
90	96500	1000	42400
95	98700	1100	40600
100	99900	1200	38800
✓110	100800	1300	37000
✓115	101400	1400	35300
120	101700 (local max.)	1500	33700
		1600	32200

(continued next column)

(continued next page)

Deflection (mils)	Load (lbf)	
1700	30900	
1800	29800	
1900	28900	
2000	28300	
	27800	(local minimum)
2100	29000	
2200	30400	
2300	32900	
2400	34200	
2500	35200	
2600	36100	
2700	37200	
2800	38200	
2900	39400	
3000	40200	
3100	41000	

Crushed boxes were taken by Wachter Associates.

INDUSTRIAL TESTING LABORATORY SERVICES

DR. G. T. HORNE

GTH/jd

✓ Indicate slight discrepancies with ERF data.
Differences insignificant.

Question V.1

Describe the sequence for the proposed modifications and fuel assembly storage arrangements. If this results in a partial loading of a rack or series of racks, then demonstrate the seismic adequacy for this arrangement in terms of stability of the rack(s).

Response to Question V.1

Described below is the proposed installation sequence for the new maximum density racks. During several steps when the existing racks are partially loaded and may not represent a seismically stable configuration, temporary seismic restraints will be installed. Revisions to the proposed installation during the actual installation may be necessary. However, at this time, the following is the Authority's planned installation/removal sequence.

INDIAN POINT 3 SPENT FUEL STORAGE RACKS
RACK HANDLING AND INSTALLATION INSTRUCTIONS

1. The maximum density spent fuel storage racks are to be installed in sequence with removal of the existing racks as delineated in these installation instructions. Figure 1 shows the existing rack array and stored fuel arrangement. Figure 2 through 23 shows the sequence of existing rack removal, UST&D rack installation and fuel movement.

2. Remove existing racks in this sequence, A-4, A-3, A-1, A-2 and A-5. See Figure 2. Install temporary seismic restraints on Racks A-7 and B-3.

3. Install UST&D Rack #1, locating it from the North and East wall. However, in the North-South direction the rack must line-up with the crane rails. See Figure 3.

4. Install the two floor plate combinations (See Figure 3), Group 1 and 2, from the South face of Rack #1 and from the East wall if the East wall lines up with the crane rails. If the wall and rails do not line up, compensation must be made for the difference. The racks must be lined up with the crane rails.

5. Move fuel from Rack A-6 to Rack #1 as shown on Figure 4.

6. Remove Rack A-6. See Figure 5.

7. Install UST&D Rack #2 in contact with Rack #1 at the bottom spacer blocks and the same distance from the East wall as Rack #1. However, the East face of Rack #2 must line up with the East face of Rack #1. See Figure 6.

8. Move fuel from Rack A-7 to UST&D Rack #1 as shown on Figure 7.

9. Remove temporary seismic restraints on Rack A-7 and remove the rack. See Figure 8.

Response to V.1 (Continued)

10. Install UST&D Rack #3 in contact with Rack #2 at the bottom spacer blocks and the same distance from the East wall as Rack #2. However, the East face of Rack #3 must line up with the East face of Racks #1 and #2. See Figure 9.

11. Install the two floor plate combinations (See Figure 9), (Group 3 and 4), located from Rack #3 and the East wall. If the wall and the crane rails do not line up, compensation must be made for the difference.

12. Install UST&D Rack #4 in contact with Rack #3 at the bottom spacer blocks and located from the East wall. See Figure 10. If the wall and crane rails do not line up, compensation must be made for the difference.

13. Move fuel from Racks B-1, B-2 and B-3 to Racks #1, #2, #3 and #4 as shown on Figure 11.

14. Install Rack #6 in contact with Rack #3 at the bottom spacer blocks and in contact with the cell in the last row on the West face of Rack #4, as shown on Figure 12.

15. Install the two floor plate combinations, Group 6 and 7, located as shown from Rack #4 and #6. Install the single floor plate combination, Group 5, located as shown from Rack #2 and #6. See Figure 12.

16. Install Rack #5 in contact with Rack #2 at the bottom spacer blocks and in contact with Rack #6. See Figure 13.

17. Move fuel from Racks B-1, B-2 and B-3 to Racks #1, #2, #3 and #5 as shown on Figure 14.

18. Install temporary seismic restraints on Racks B-4 and B-5 and remove seismic restraints on B-3.

19. Remove racks B-1, B-2 and B-3. See Figure 15.

20. Install Rack #7 in contact with Racks #4 and #6, as shown on Figure 16.

21. Move fuel from Racks B-4 and B-5 to Racks #3, #4 and #6 as shown on Figure 17.

22. Remove temporary seismic restraints from Rack B-4 and B-5 and remove racks. See Figure 18.

23. Install two floor plate combinations, Group 6 and Group 10, located as shown from Rack #7. Install the single floor plate Group 9, located as shown from Rack #6 and #7. See Figure 19.

24. Install Rack #9 in contact with Rack #6 and the cell in last row on the West face of Rack #7 as shown on Figure 20. Install Rack #10 in contact with Rack #7 and #9, as shown on Figure 20.

25. Install two floor plate combinations, Group 8 and Group 11 as shown, located from Rack #5 and #9. See Figure 20.

26. Install Rack #8 in contact with Rack #5 and #9. See Figure 21.

27. Install two floor plate combinations, Group 12 and 13 as shown, located from Racks #8 and #9. See Figure 22.

28. Install Rack #12 in contact with Rack #9 and the cell in the last row on the West face of Rack #10 as shown on Figure 23. Install Rack #11 in contact with Rack #8 and #12 as also shown on Figure 23.

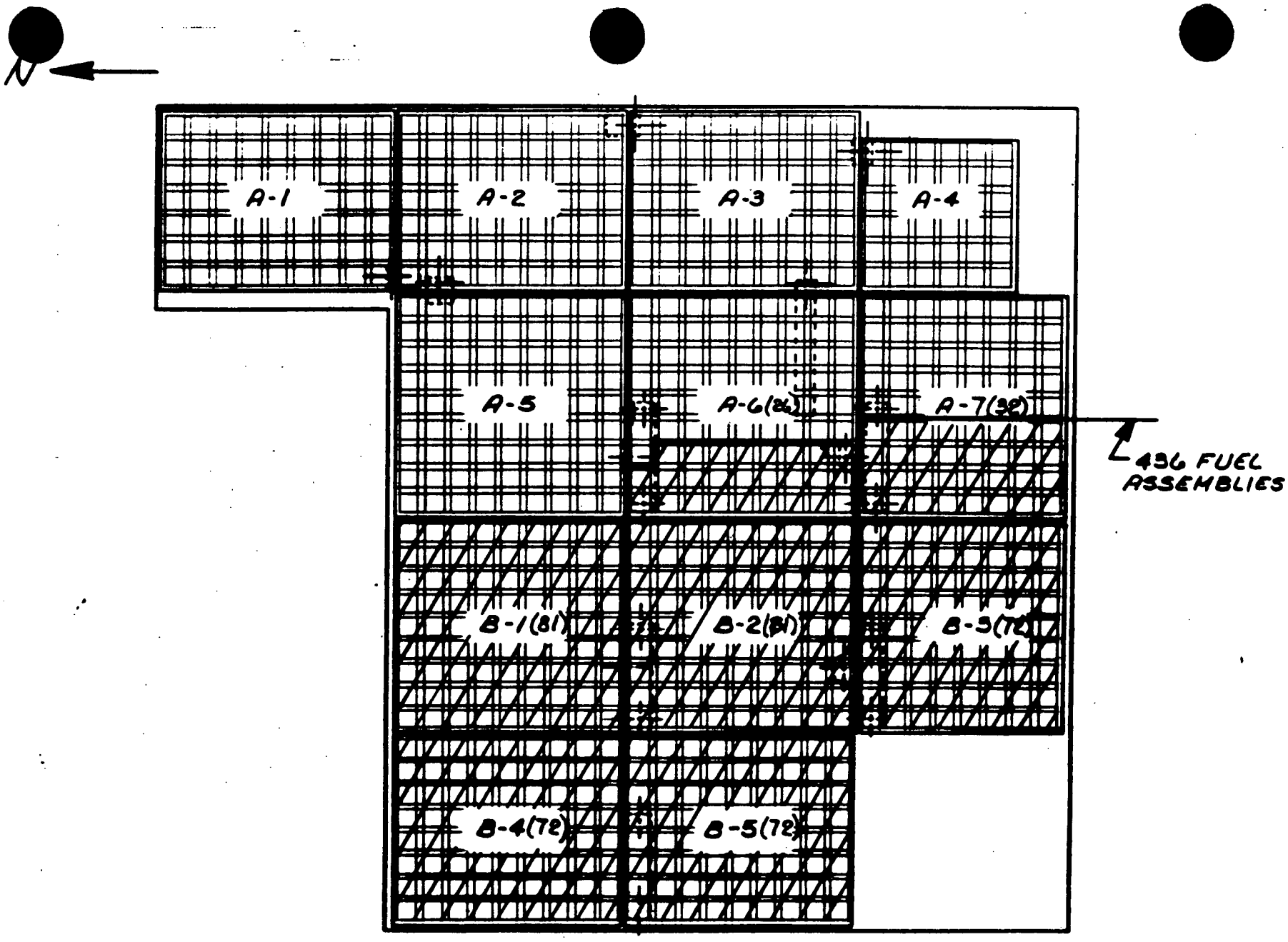

N
A-1
A-2
A-3
A-4
A-5
A-6(26)
A-7(32)
B-1(81)
B-2(81)
B-3(72)
B-4(72)
B-5(72)
436 FUEL ASSEMBLIES

FIGURE 1

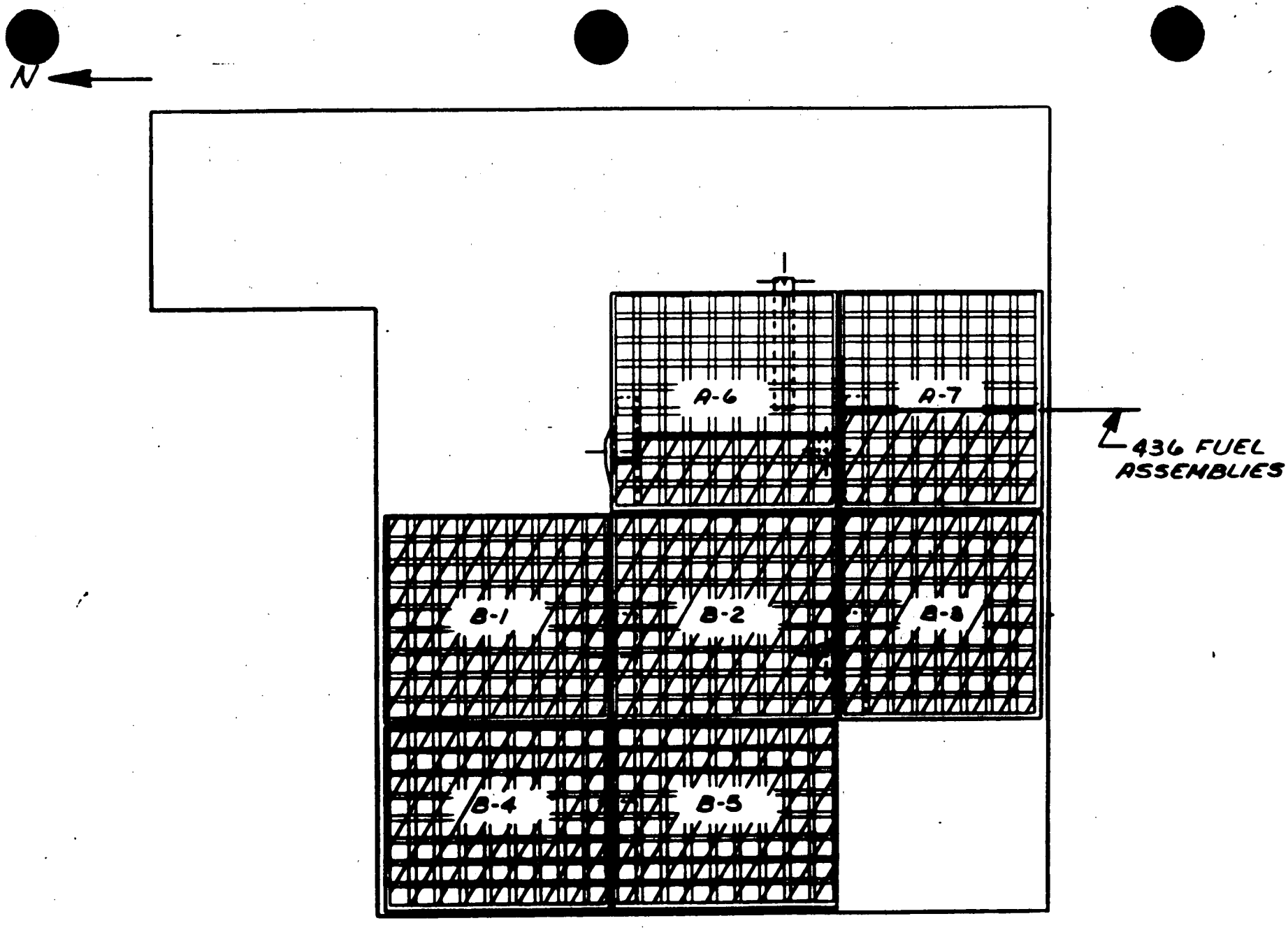


FIGURE 2

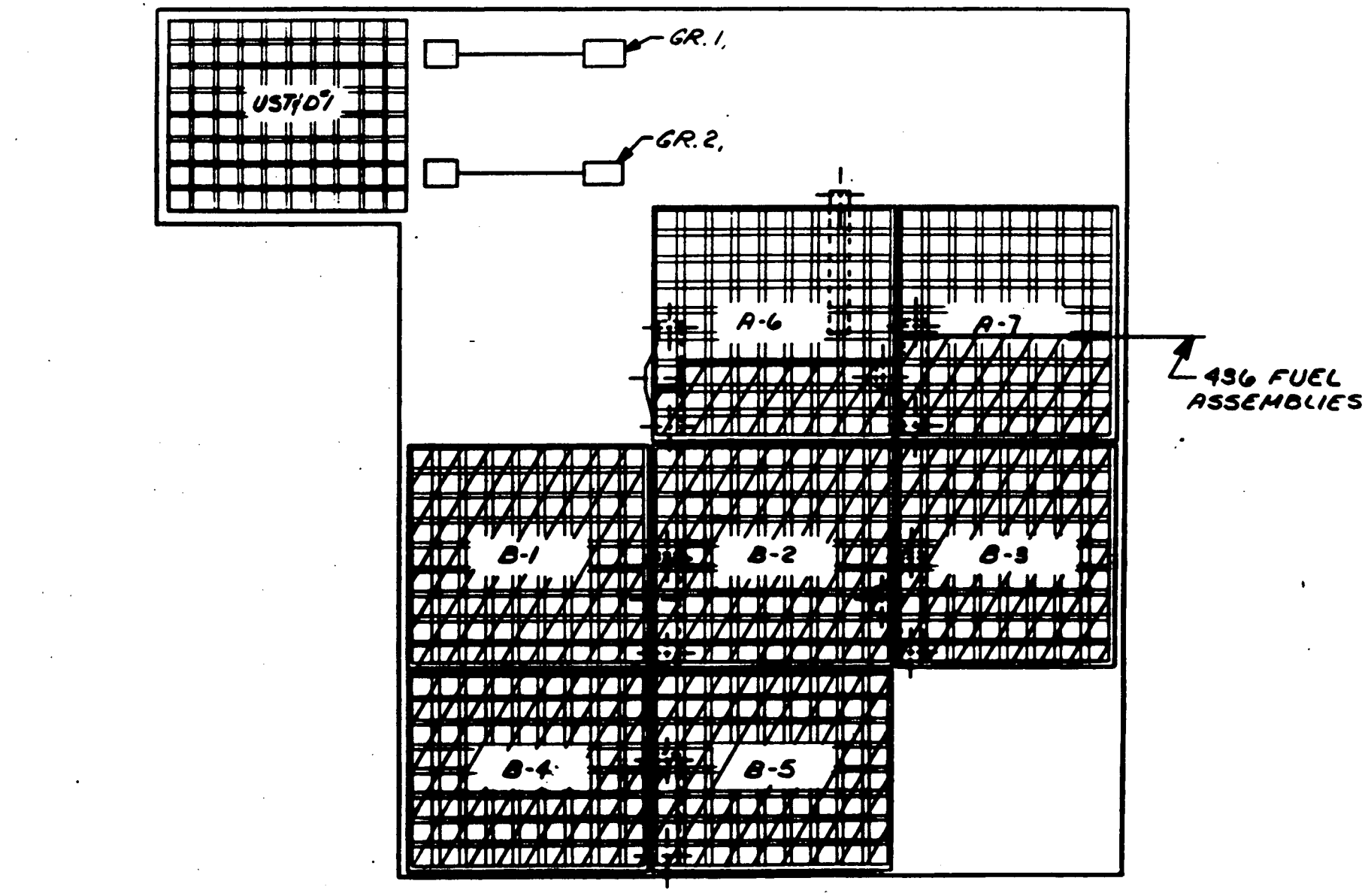


FIGURE 3

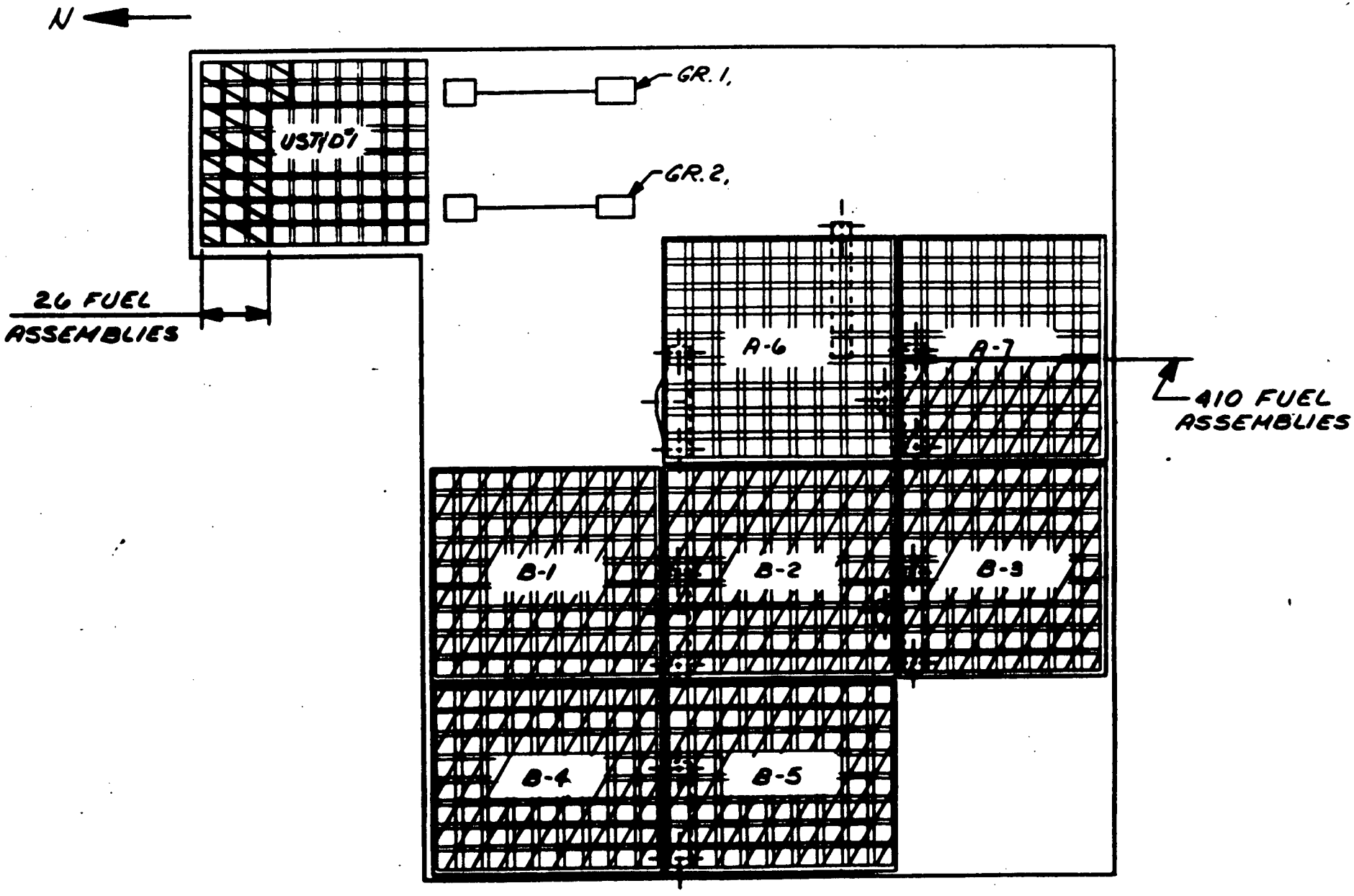

N
GR. 1,
GR. 2,
USTID 7
26 FUEL ASSEMBLIES
A-6
A-7
410 FUEL ASSEMBLIES
B-1
B-2
B-3
B-4
B-5

FIGURE 4

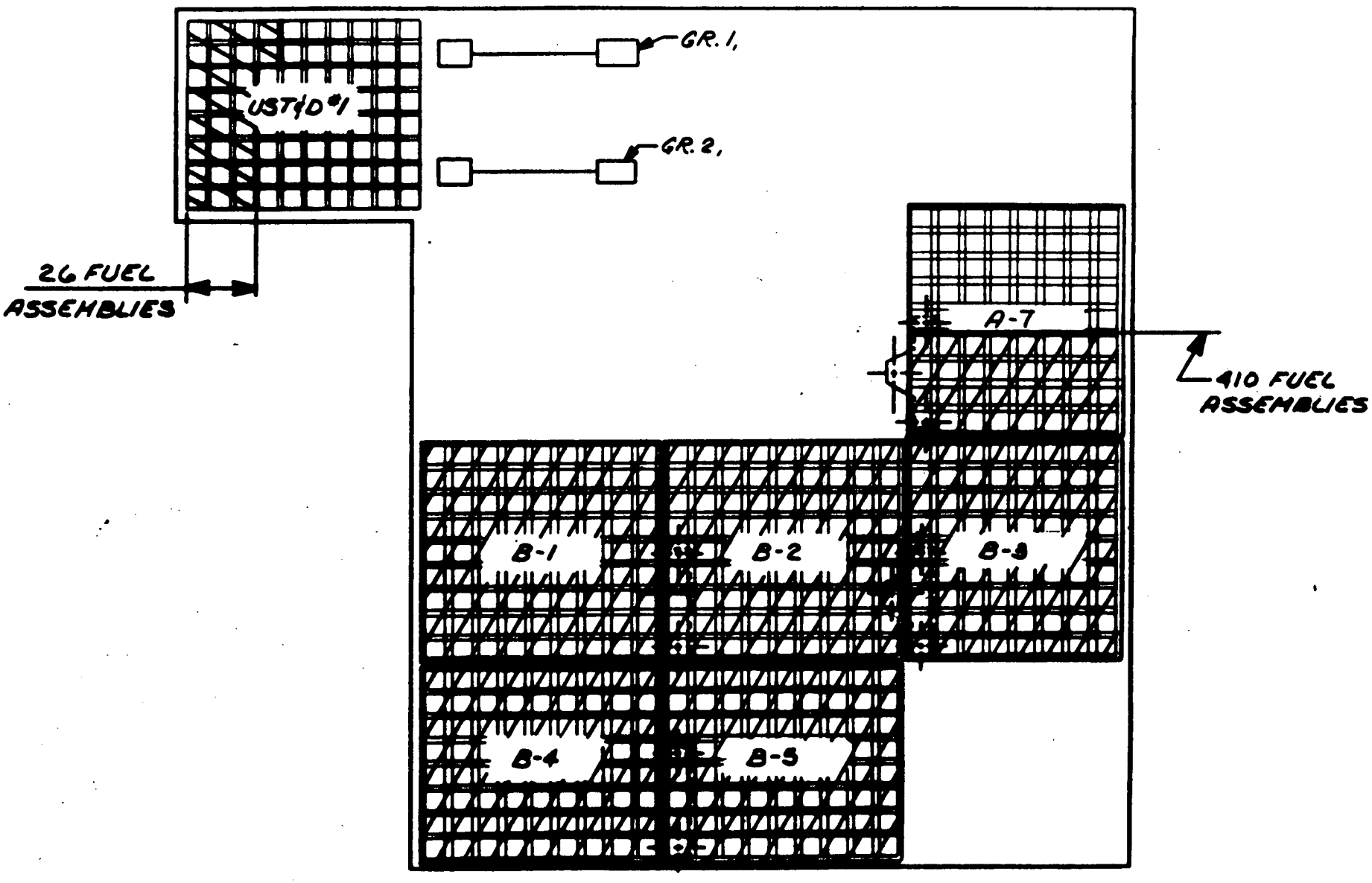

N
GR. 1,
GR. 2,
UST&D #1
26 FUEL ASSEMBLIES
A-7
410 FUEL ASSEMBLIES
B-1
B-2
B-3
B-4
B-5

FIGURE 5

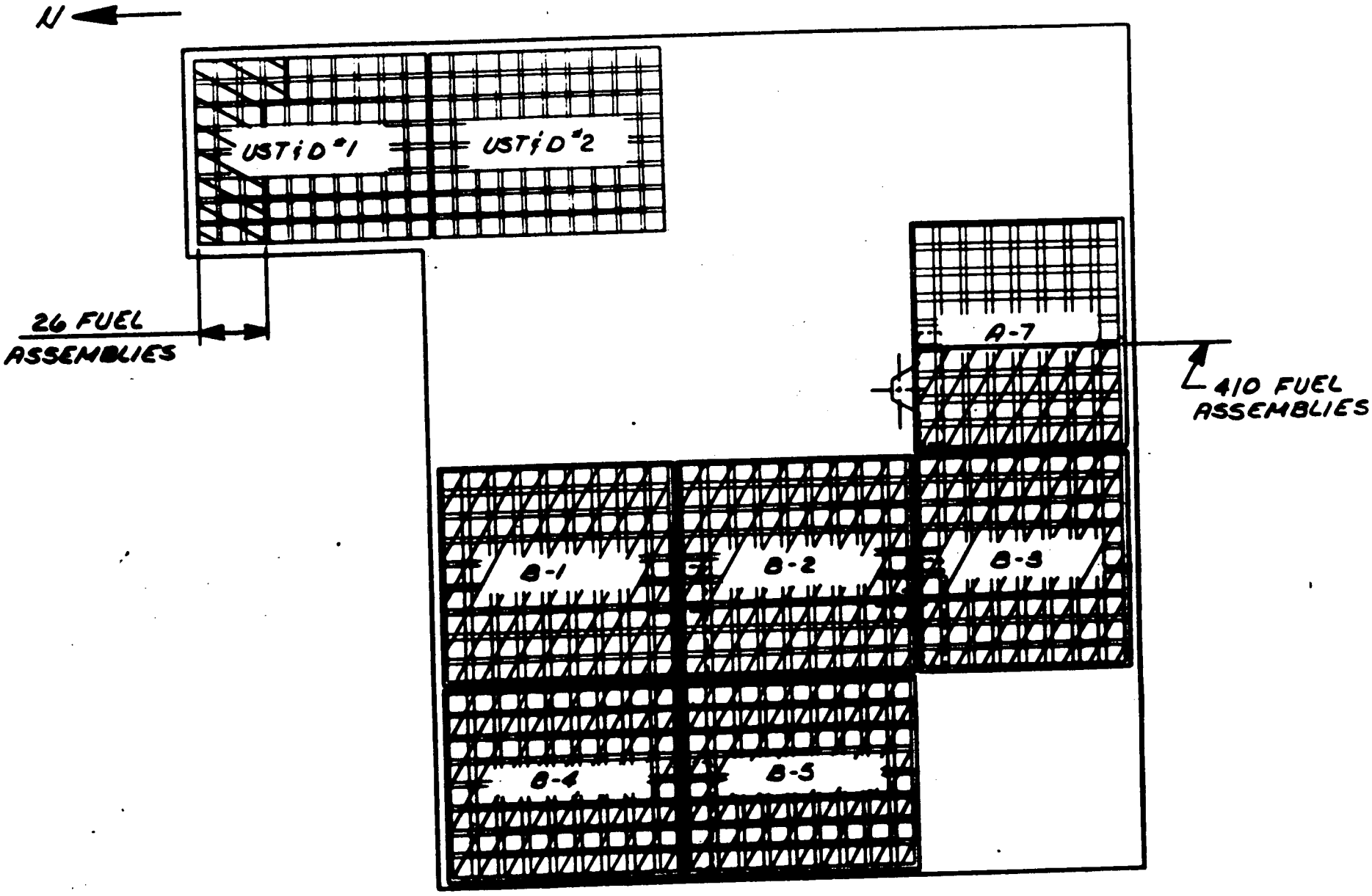

N
USTID #1
USTID #2
26 FUEL ASSEMBLIES
A-7
410 FUEL ASSEMBLIES
B-1
B-2
B-3
B-4
B-5

FIGURE 6

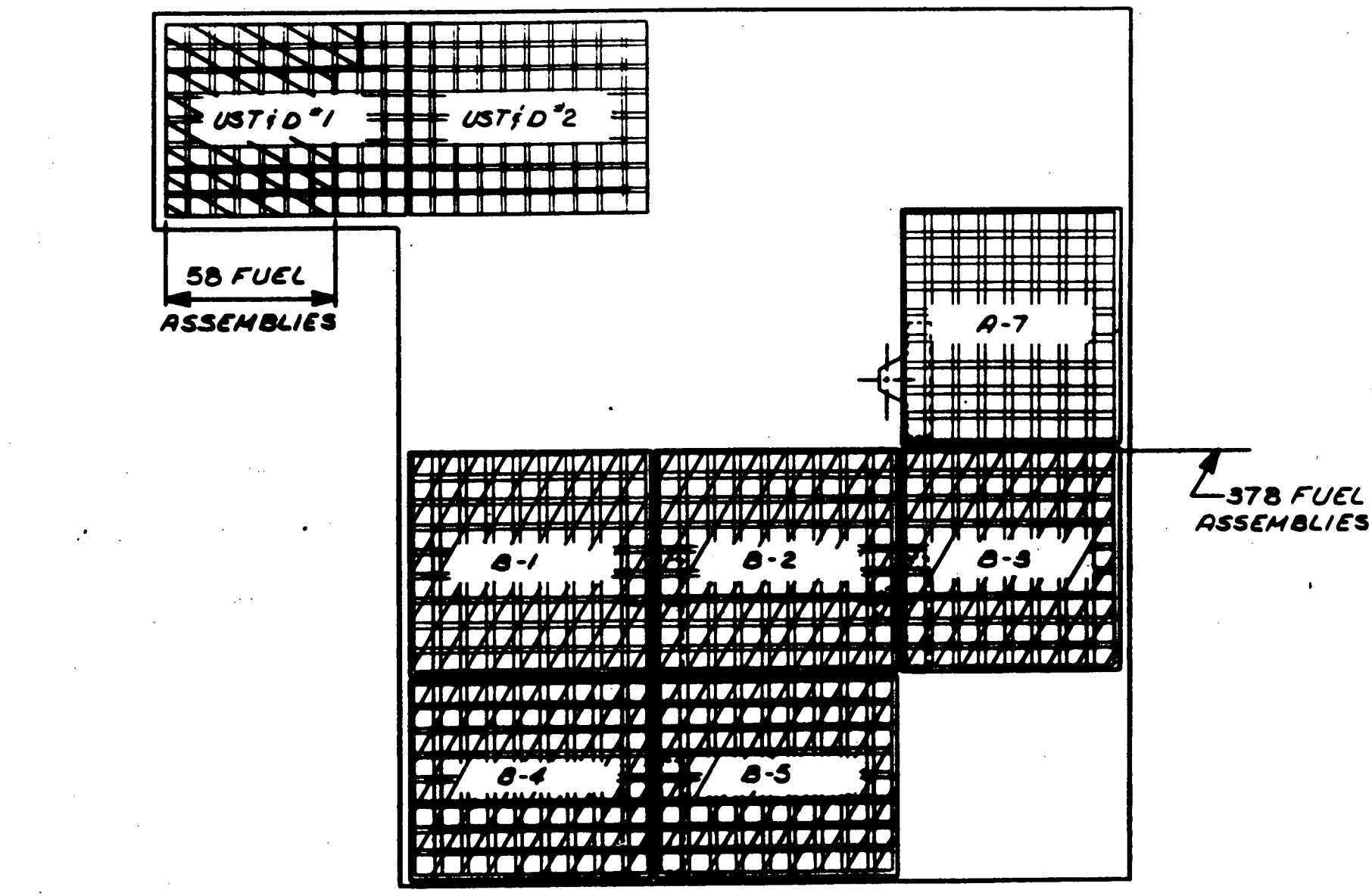

N
UST & D #1
UST & D #2
58 FUEL
ASSEMBLIES
A-7
378 FUEL
ASSEMBLIES
B-1
B-2
B-3
B-4
B-5

FIGURE 7

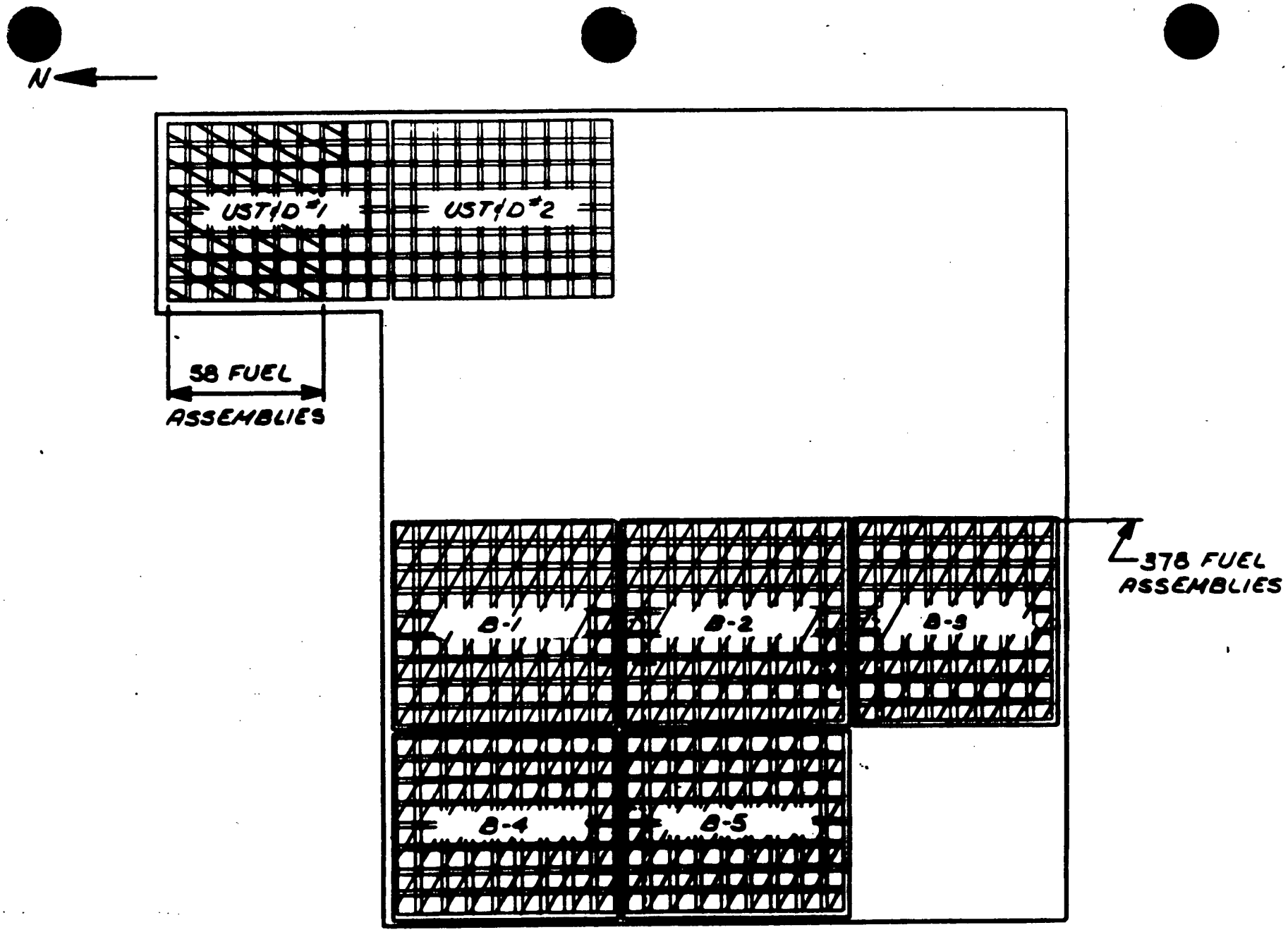

N
UST/D #1
UST/D #2
58 FUEL
ASSEMBLIES
B-1
B-2
B-3
B-4
B-5
378 FUEL
ASSEMBLIES

FIGURE 8

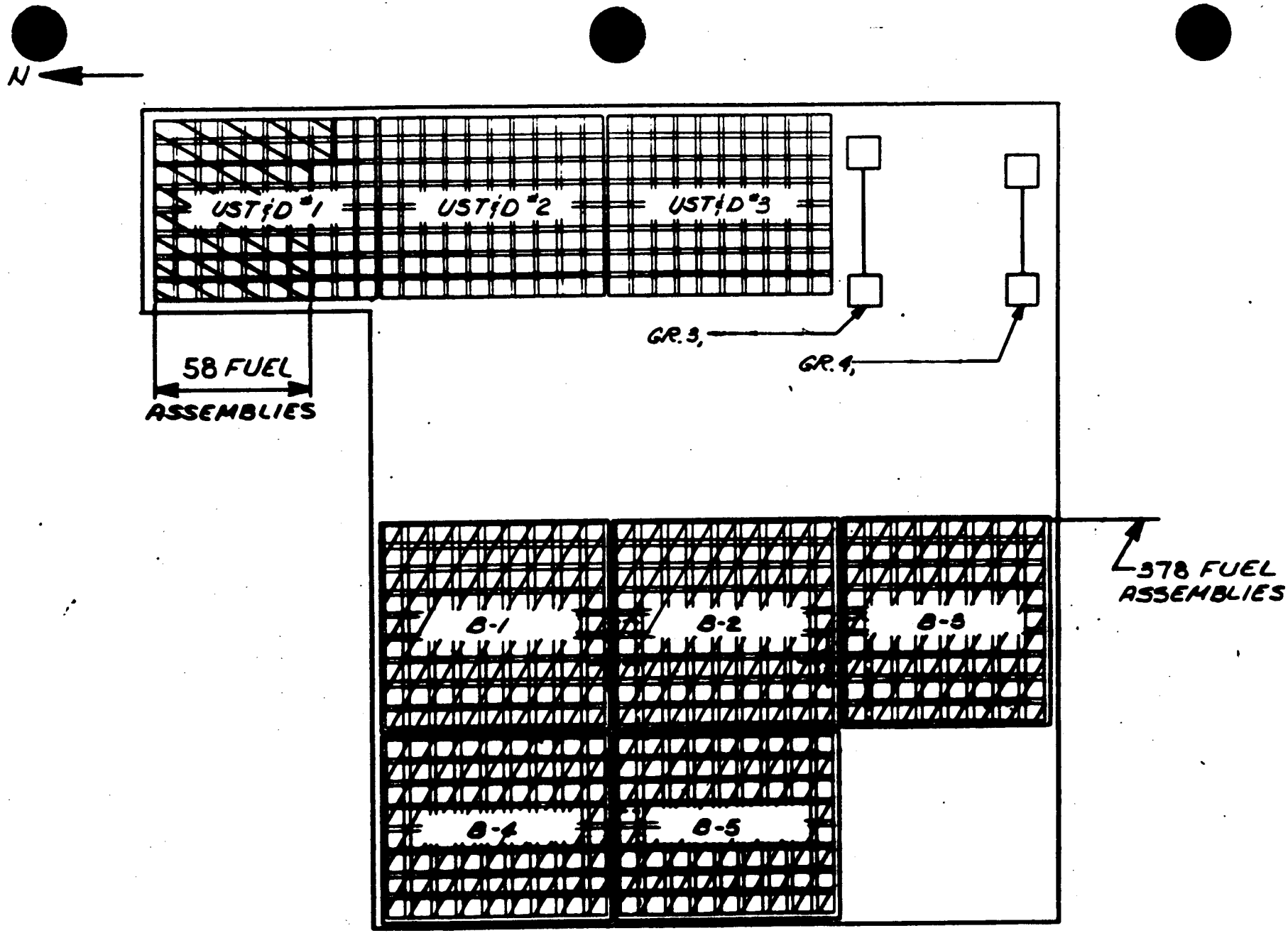

N
UST&D #1
UST&D #2
UST&D #3
GR. 3,
GR. 4,
58 FUEL
ASSEMBLIES
378 FUEL
ASSEMBLIES
B-1
B-2
B-3
B-4
B-5

FIGURE 9

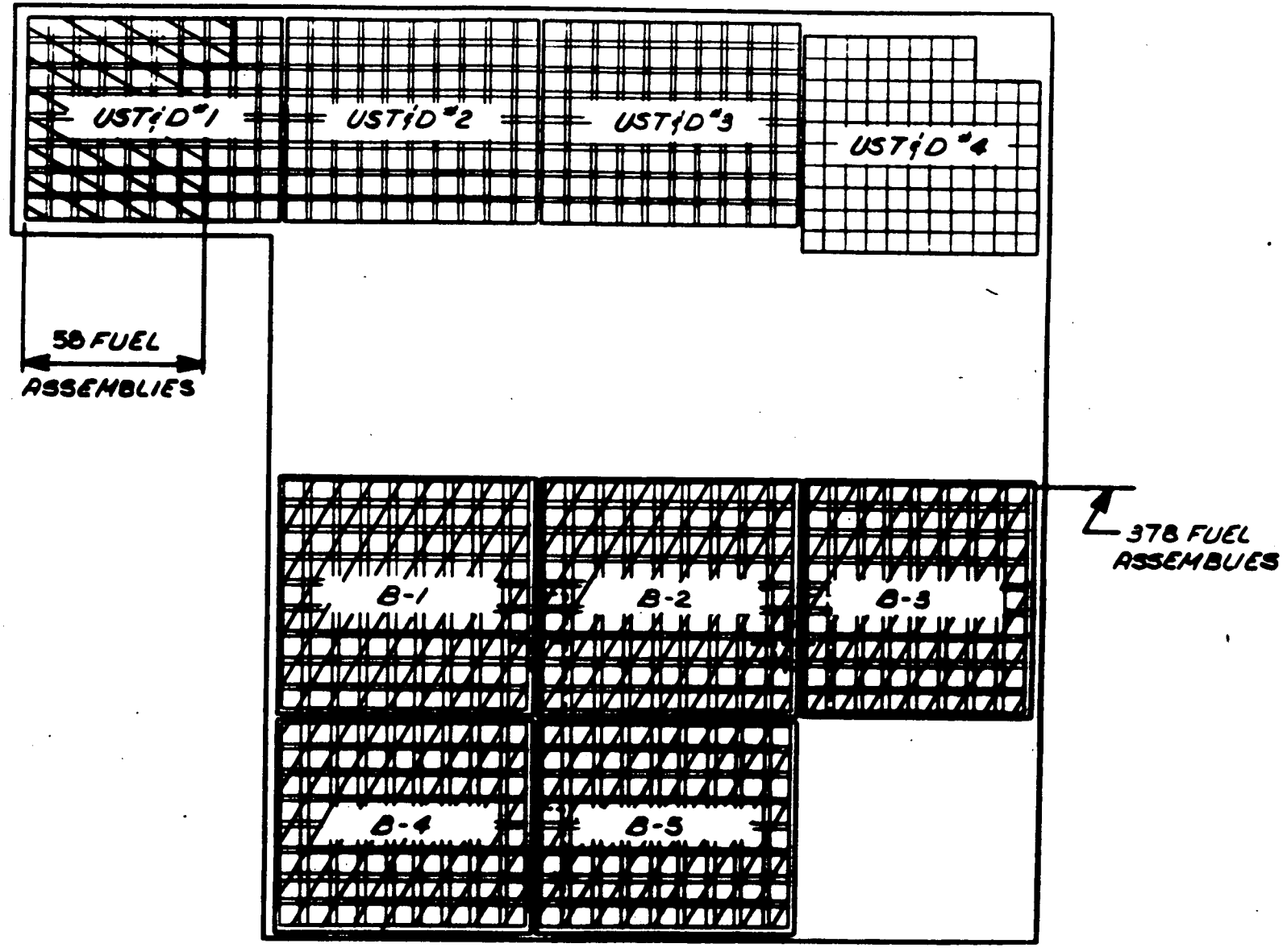

N
UST/D #1
UST/D #2
UST/D #3
UST/D #4
58 FUEL
ASSEMBLIES
B-1
B-2
B-3
B-4
B-5
378 FUEL
ASSEMBLIES

FIGURE 10

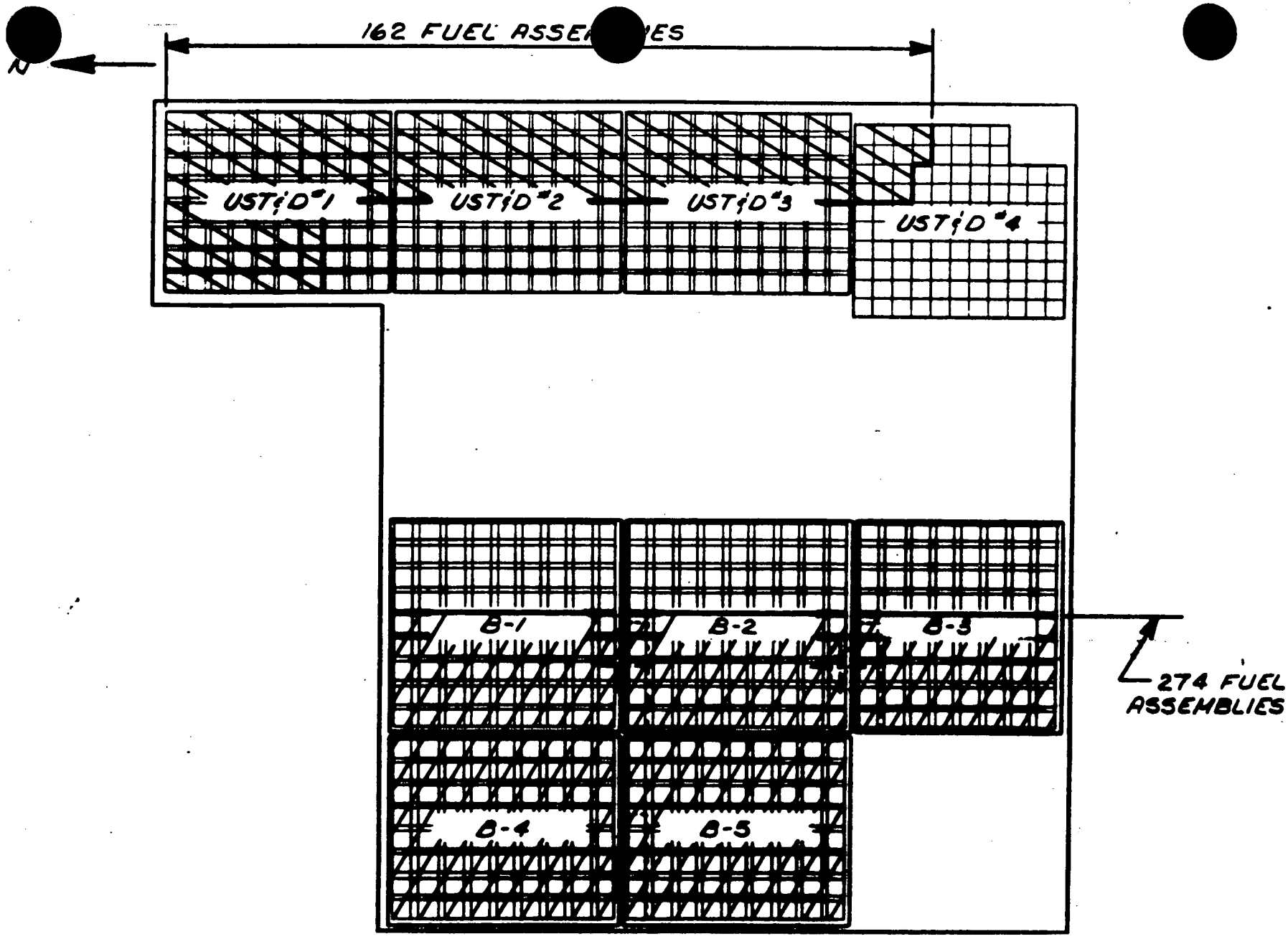

162 FUEL ASSEMBLIES
N
UST&D #1
UST&D #2
UST&D #3
UST&D #4
B-1
B-2
B-3
B-4
B-5
274 FUEL ASSEMBLIES

FIGURE 11

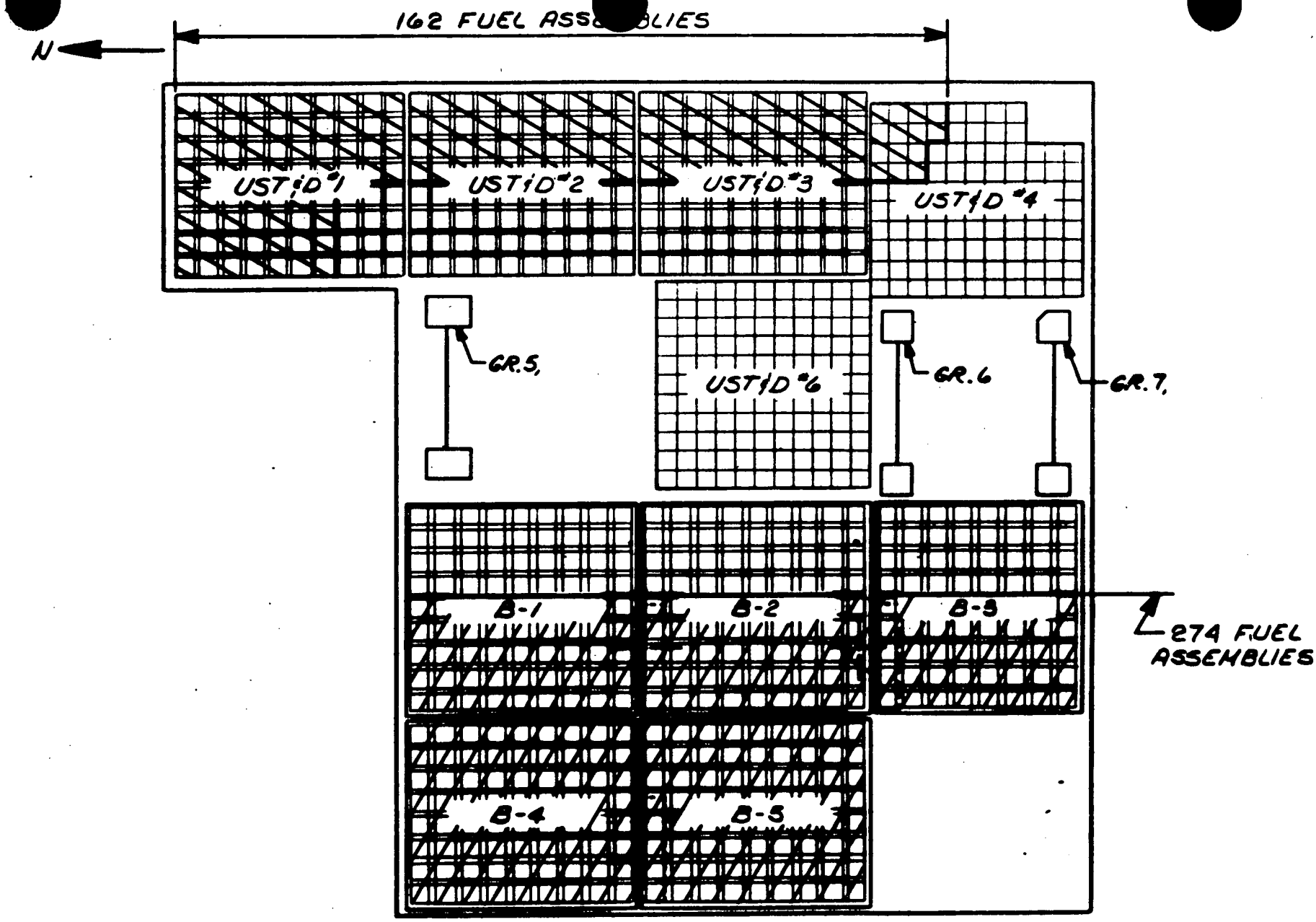

162 FUEL ASSEMBLIES
N
UST/D #1
UST/D #2
UST/D #3
UST/D #4
UST/D #6
GR.5,
GR.6
GR.7,
B-1
B-2
B-3
B-4
B-5
274 FUEL ASSEMBLIES

FIGURE 12

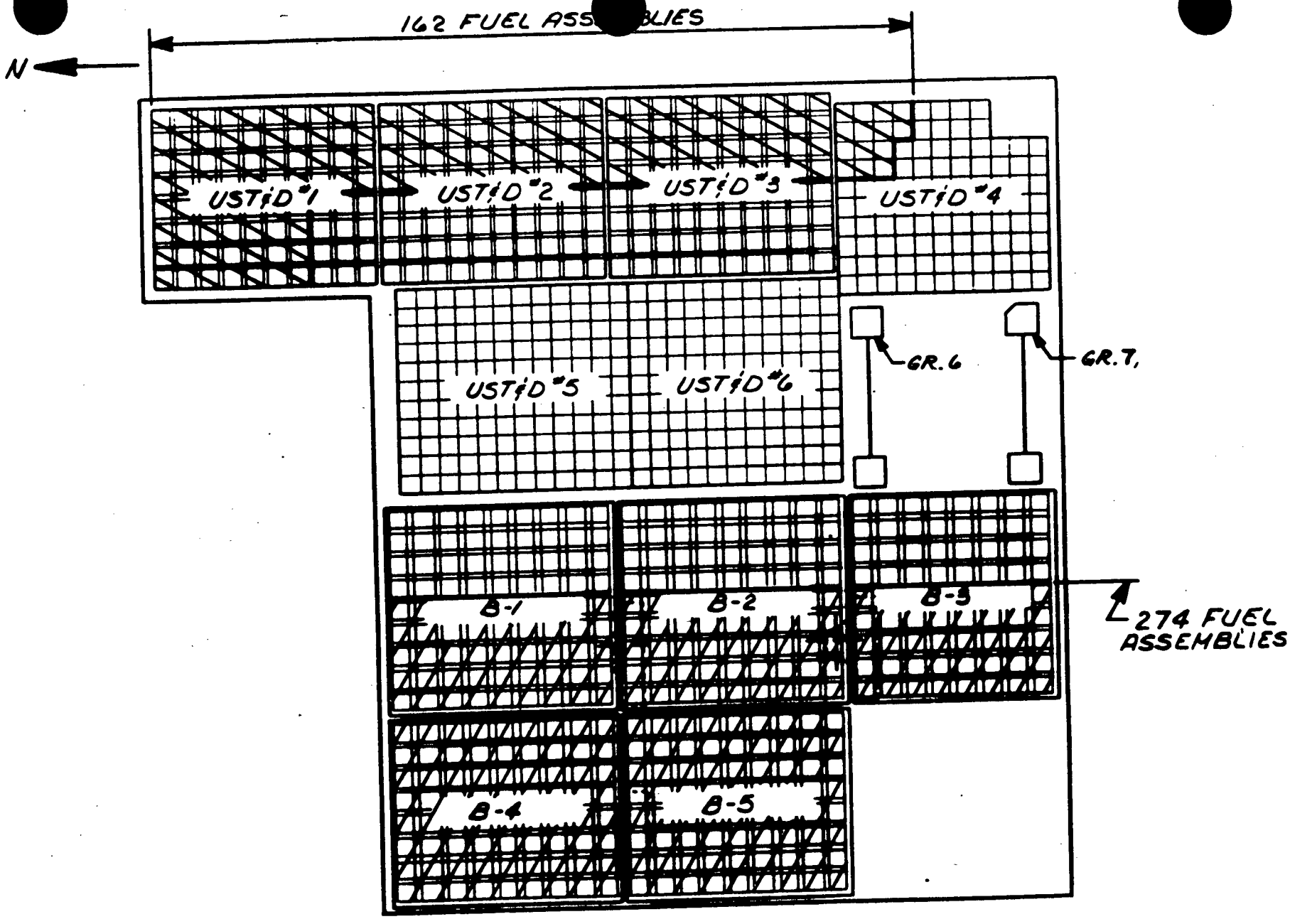

162 FUEL ASSEMBLIES
N
UST&D #1
UST&D #2
UST&D #3
UST&D #4
UST&D #5
UST&D #6
GR. 6
GR. 7,
B-1
B-2
B-3
B-4
B-5
274 FUEL ASSEMBLIES

FIGURE 13

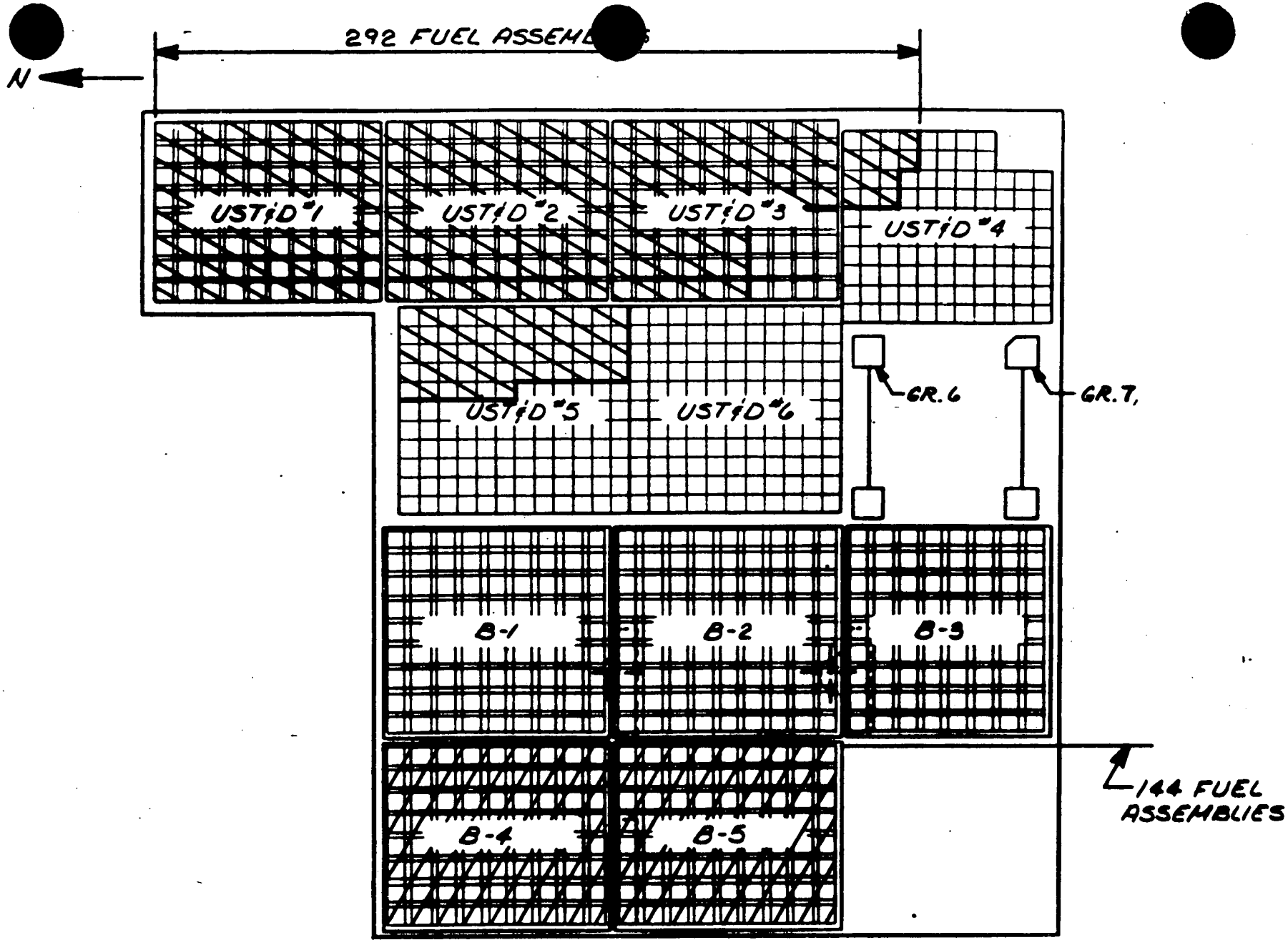

292 FUEL ASSEMBLIES
N
UST&D #1
UST&D #2
UST&D #3
UST&D #4
UST&D #5
UST&D #6
GR. 6
GR. 7,
B-1
B-2
B-3
B-4
B-5
144 FUEL ASSEMBLIES

FIGURE 14

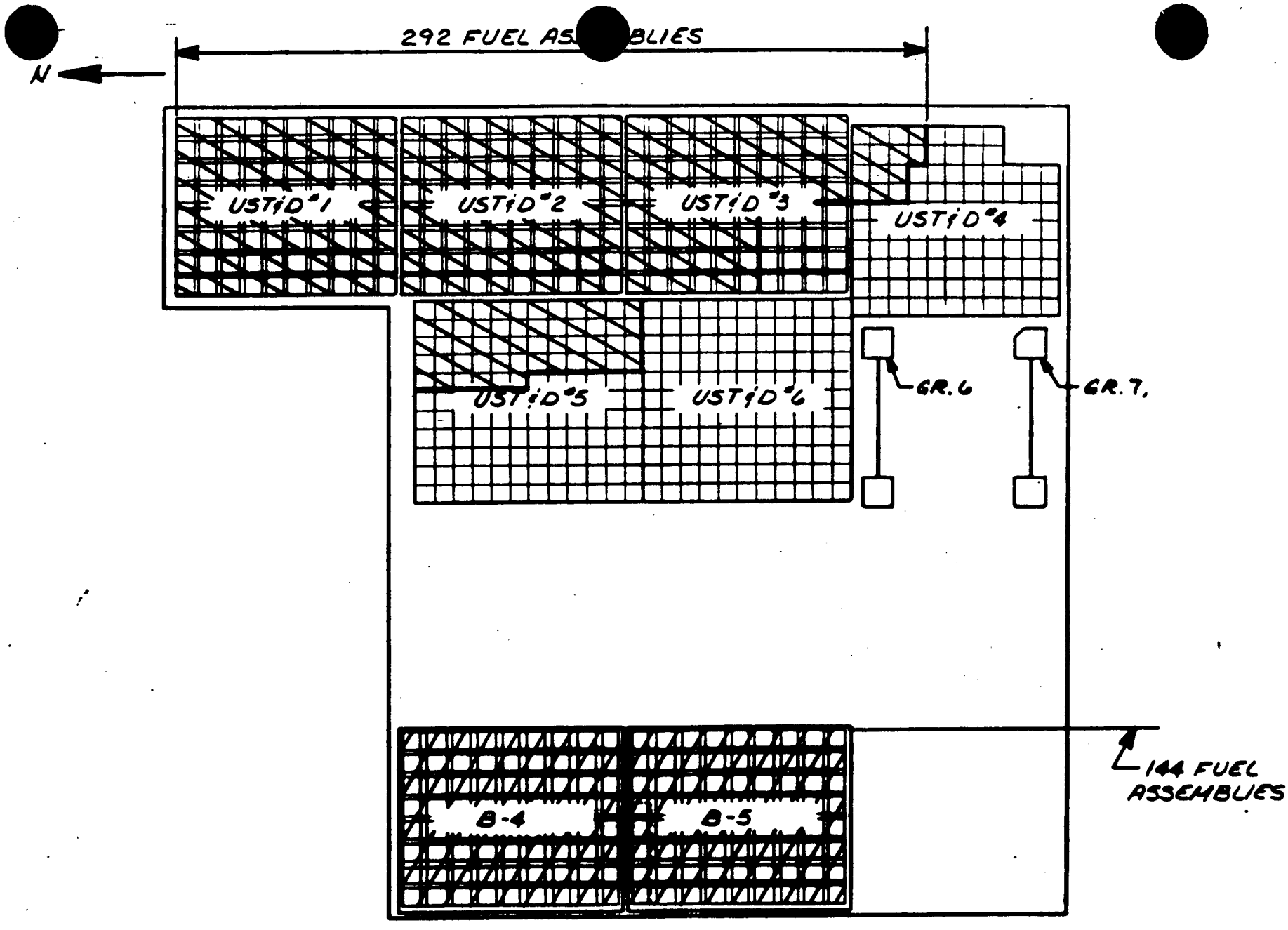

292 FUEL ASSEMBLIES
N
USTID #1
USTID #2
USTID #3
USTID #4
USTID #5
USTID #6
GR. 6
GR. 7,
B-4
B-5
144 FUEL ASSEMBLIES

FIGURE 15

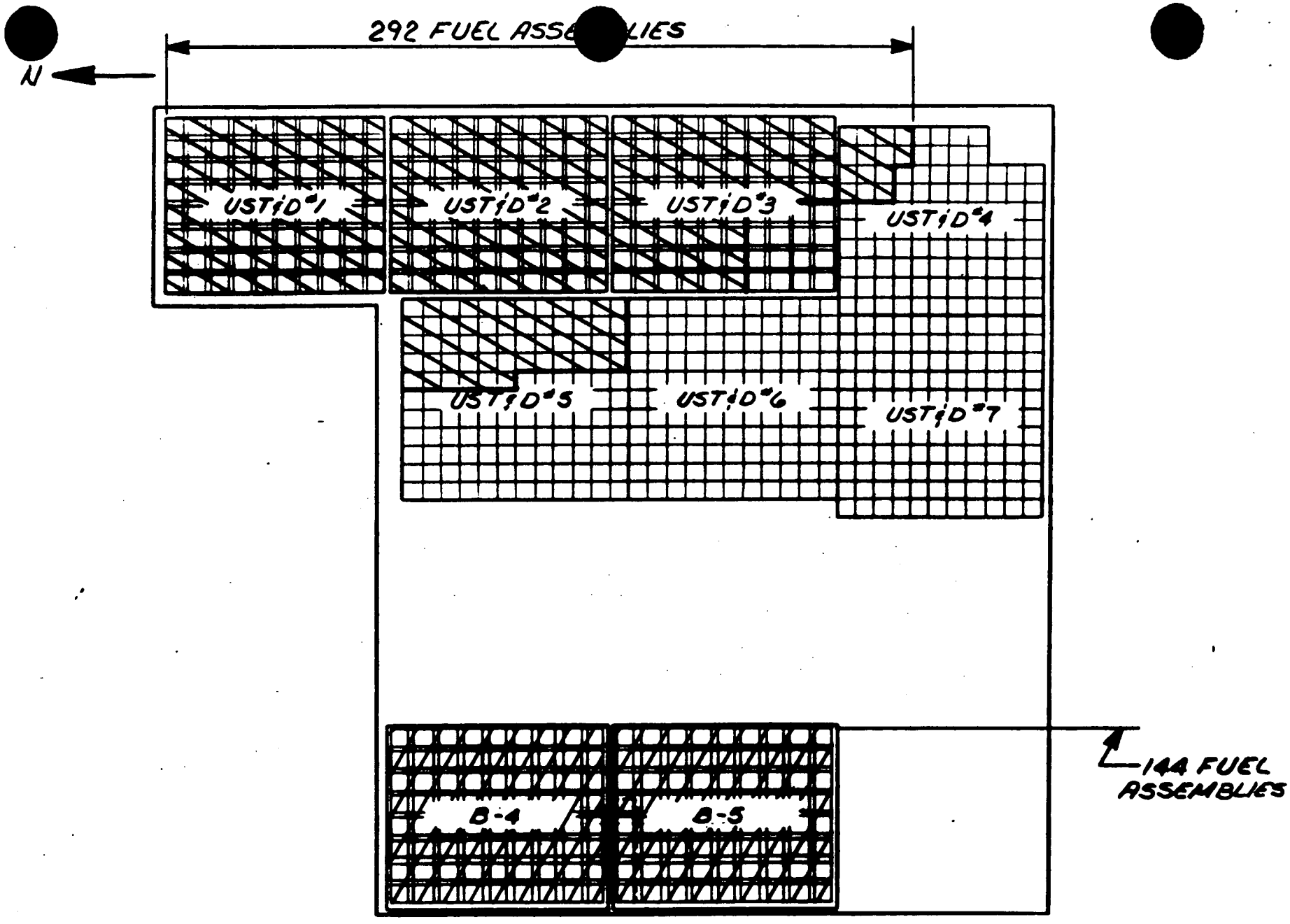

292 FUEL ASSEMBLIES
N
UST&D #1
UST&D #2
UST&D #3
UST&D #4
UST&D #5
UST&D #6
UST&D #7
B-4
B-5
144 FUEL ASSEMBLIES

FIGURE 16

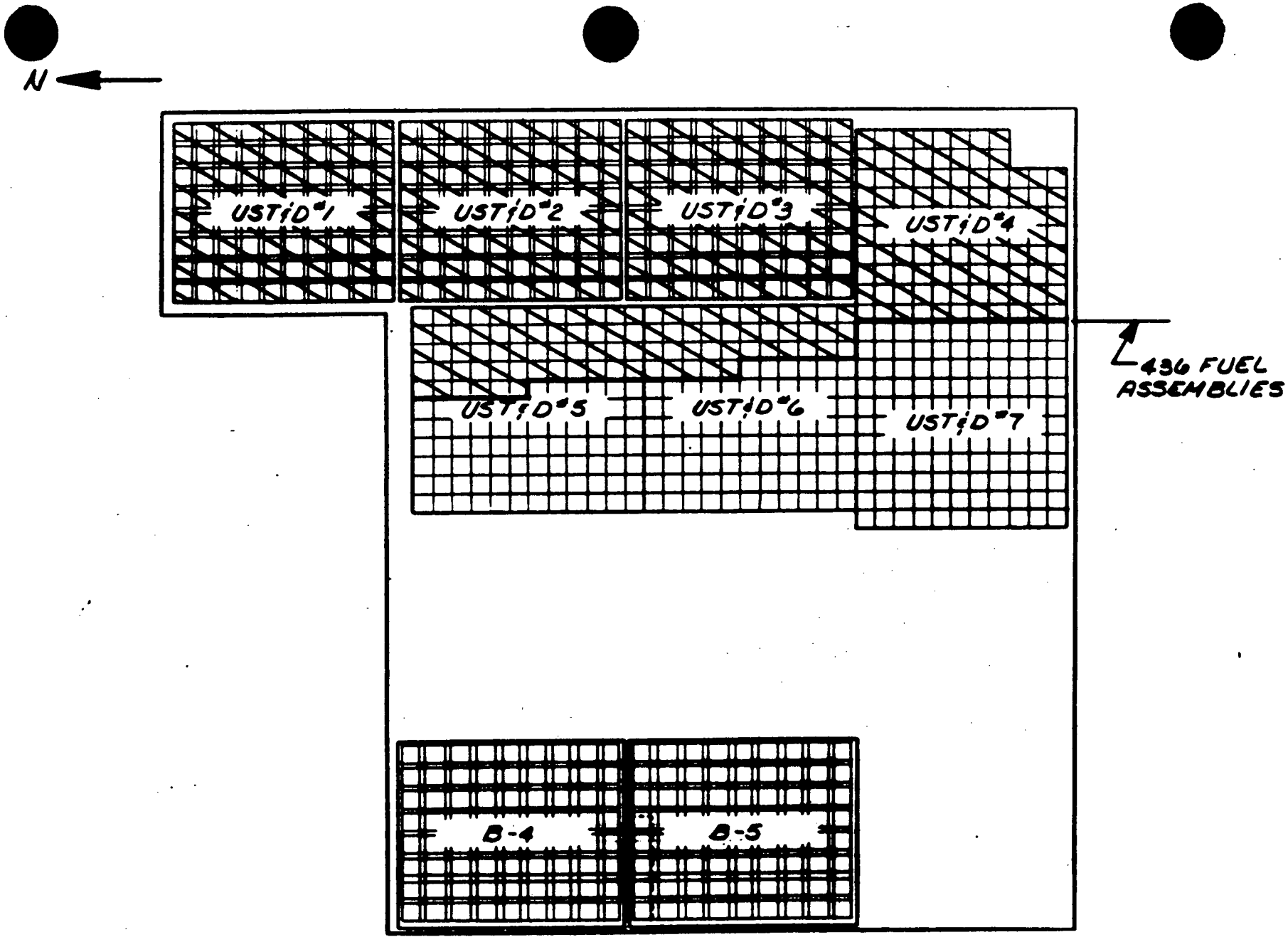

N
UST&D #1
UST&D #2
UST&D #3
UST&D #4
UST&D #5
UST&D #6
UST&D #7
436 FUEL ASSEMBLIES
B-4
B-5

FIGURE 17

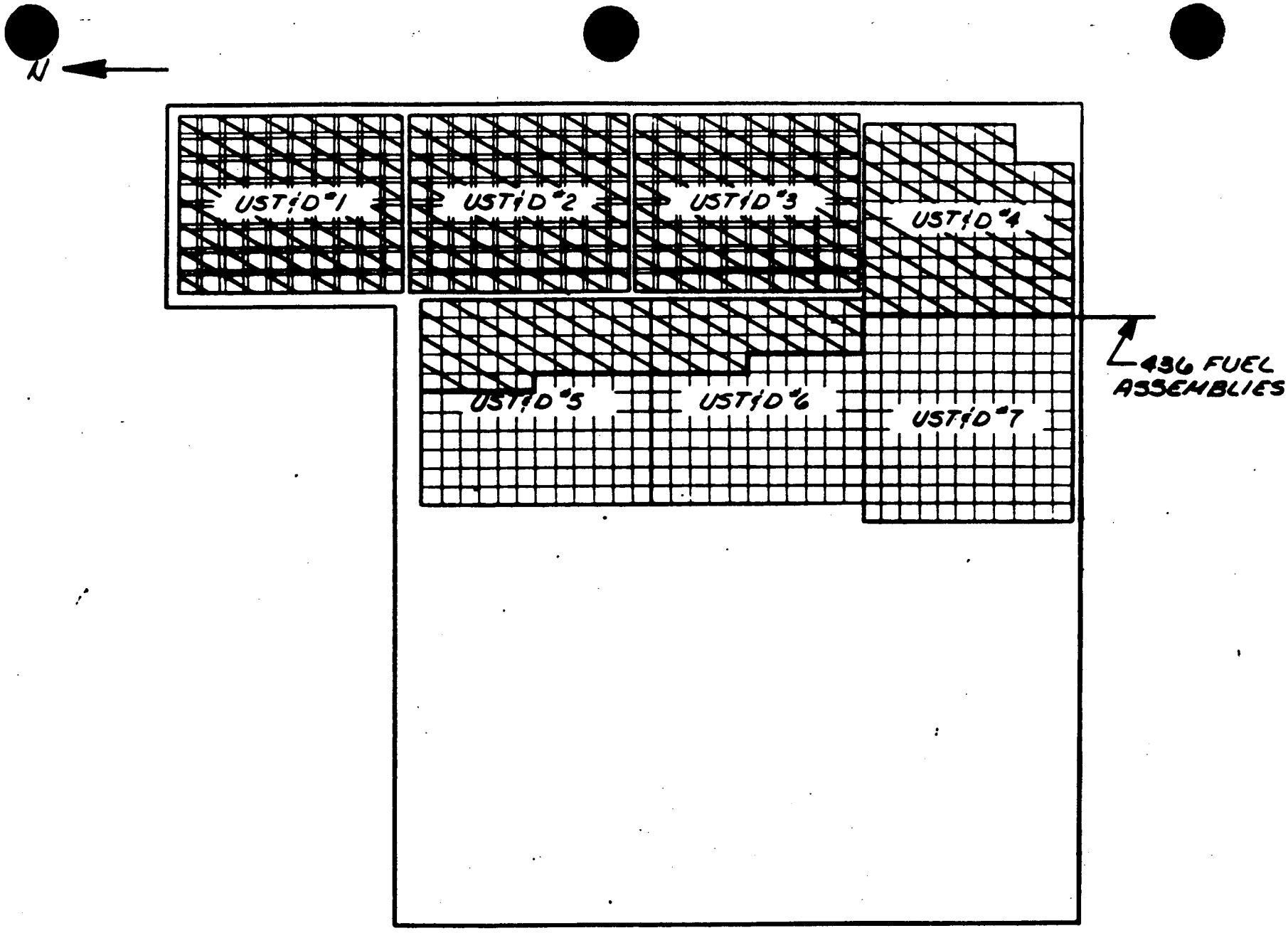

N
UST/D #1
UST/D #2
UST/D #3
UST/D #4
UST/D #5
UST/D #6
UST/D #7
436 FUEL ASSEMBLIES

FIGURE 18

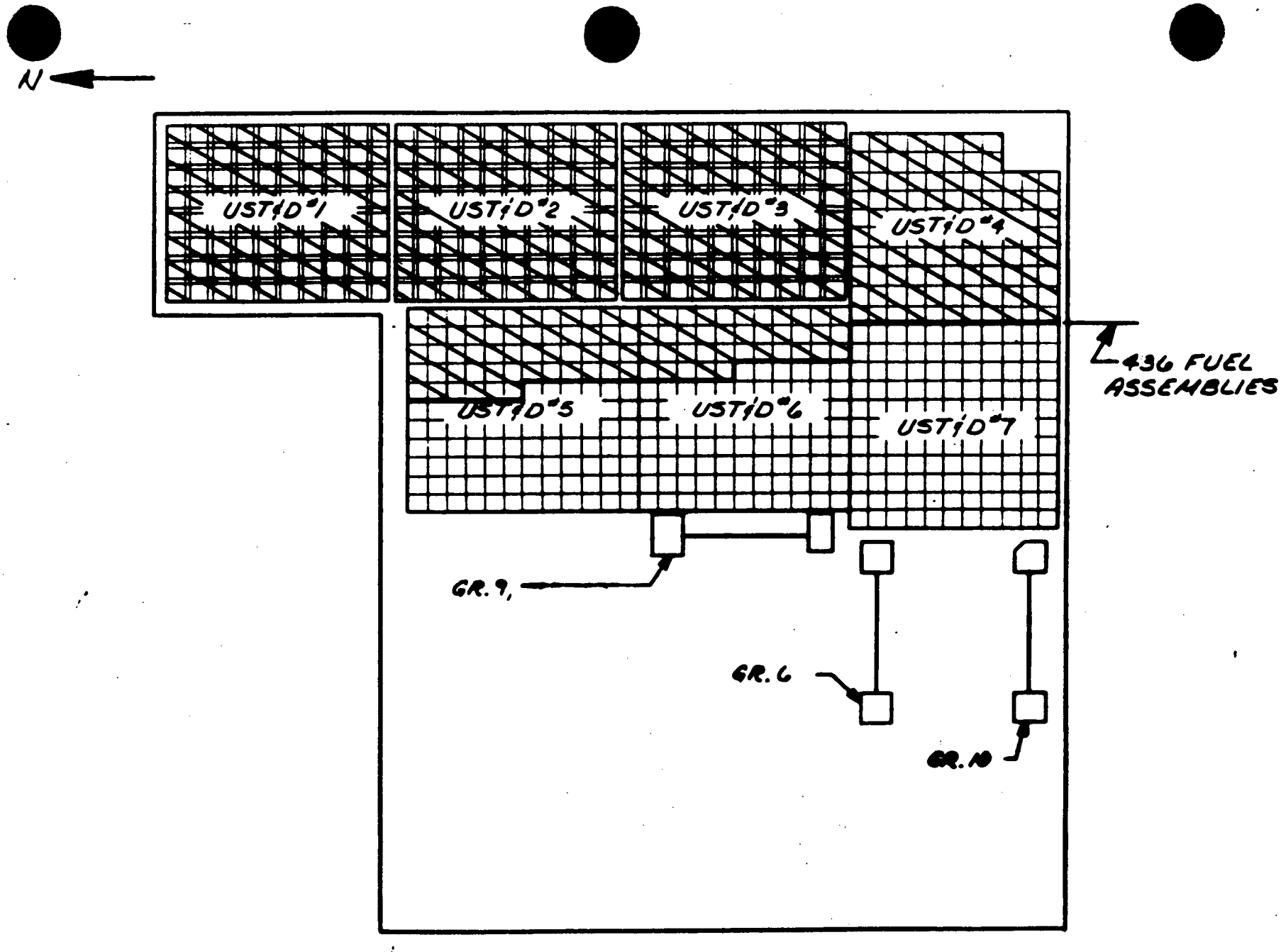

N
UST/D #1
UST/D #2
UST/D #3
UST/D #4
UST/D #5
UST/D #6
UST/D #7
436 FUEL ASSEMBLIES
GR. 9,
GR. 6
GR. 10

FIGURE 19

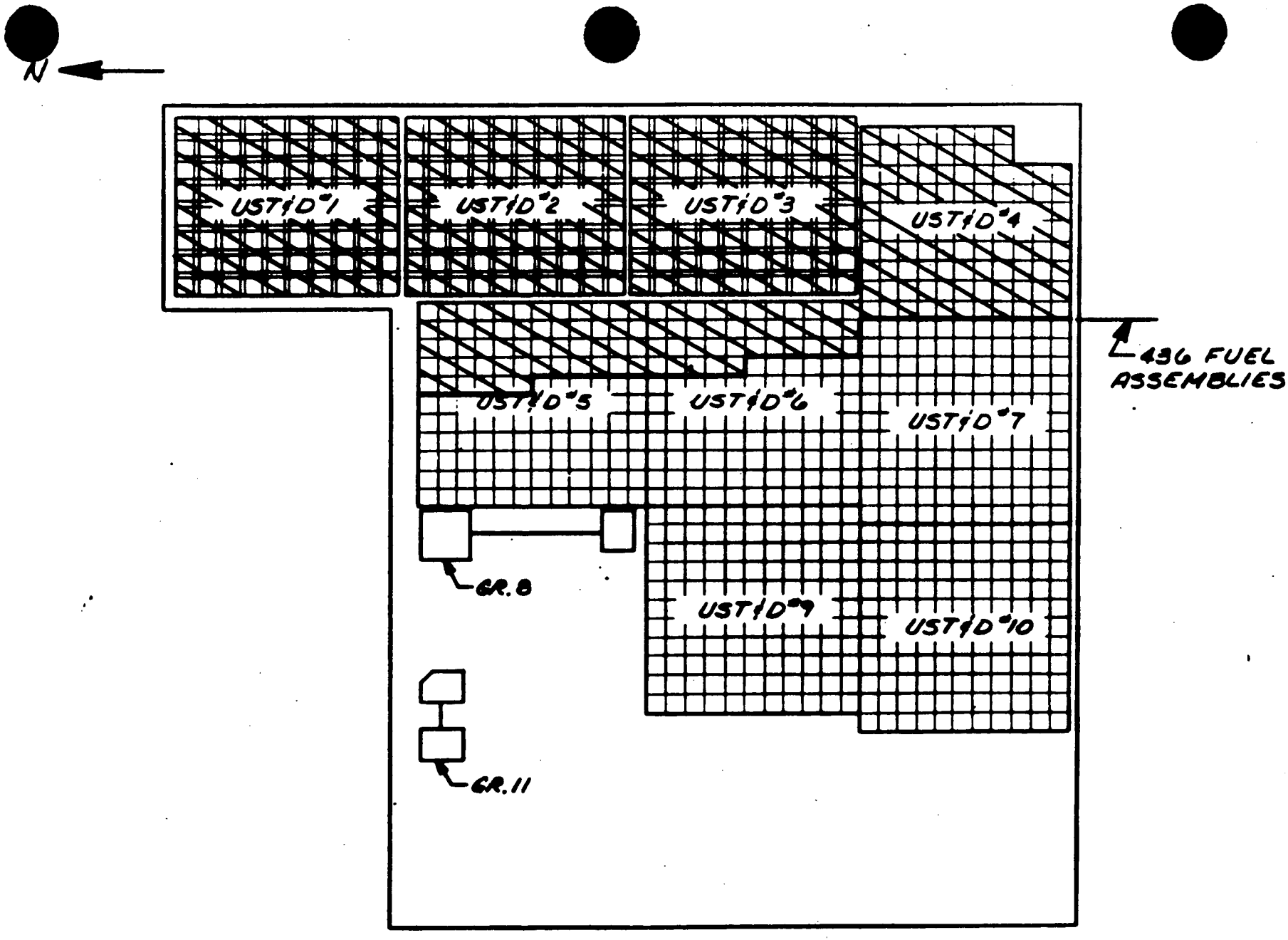

N
UST/D #1
UST/D #2
UST/D #3
UST/D #4
UST/D #5
UST/D #6
UST/D #7
UST/D #9
UST/D #10
436 FUEL ASSEMBLIES
GR. 8
GR. 11

FIGURE 20

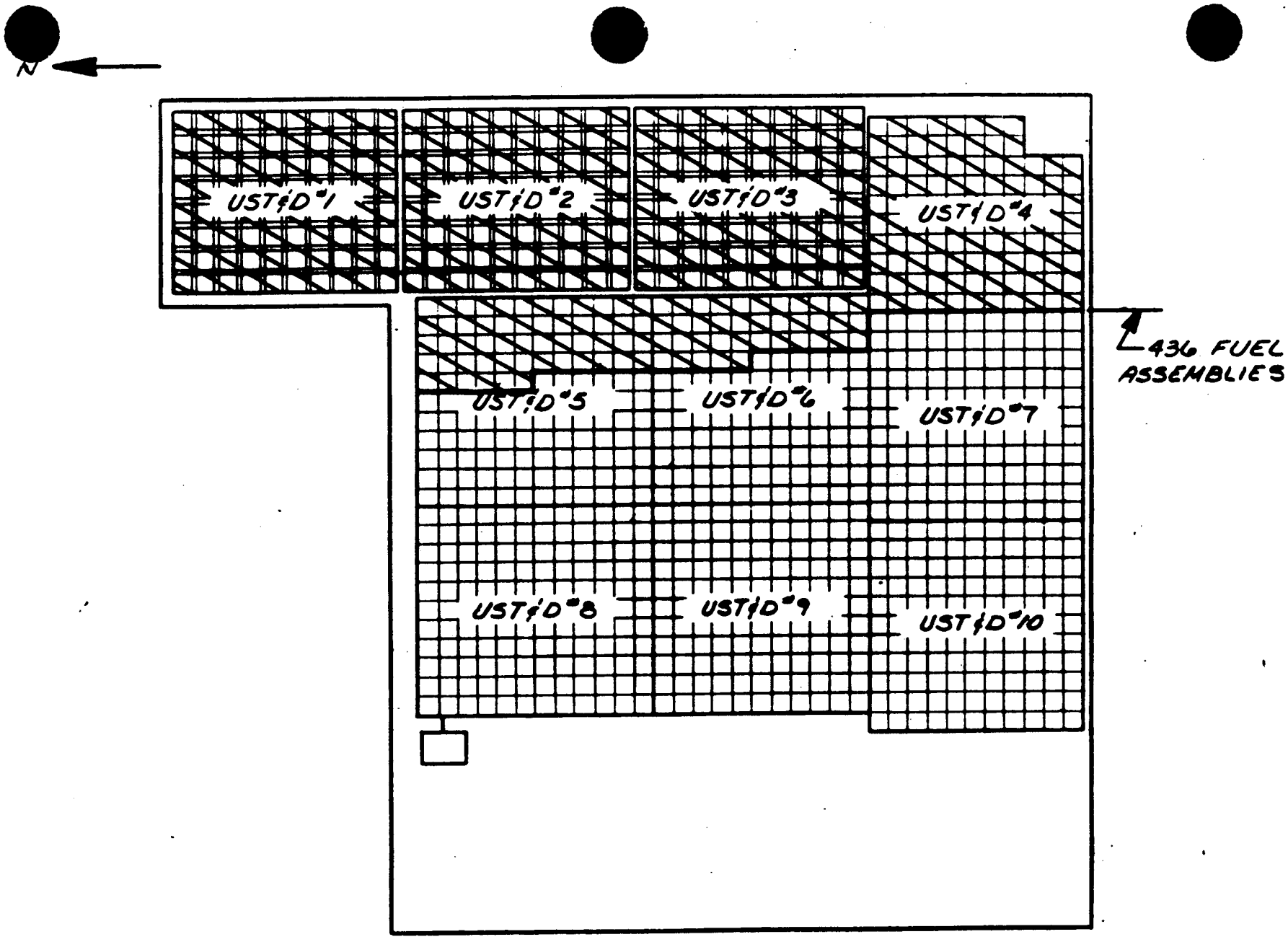

N
UST&D #1
UST&D #2
UST&D #3
UST&D #4
UST&D #5
UST&D #6
UST&D #7
UST&D #8
UST&D #9
UST&D #10
436 FUEL ASSEMBLIES

FIGURE 21

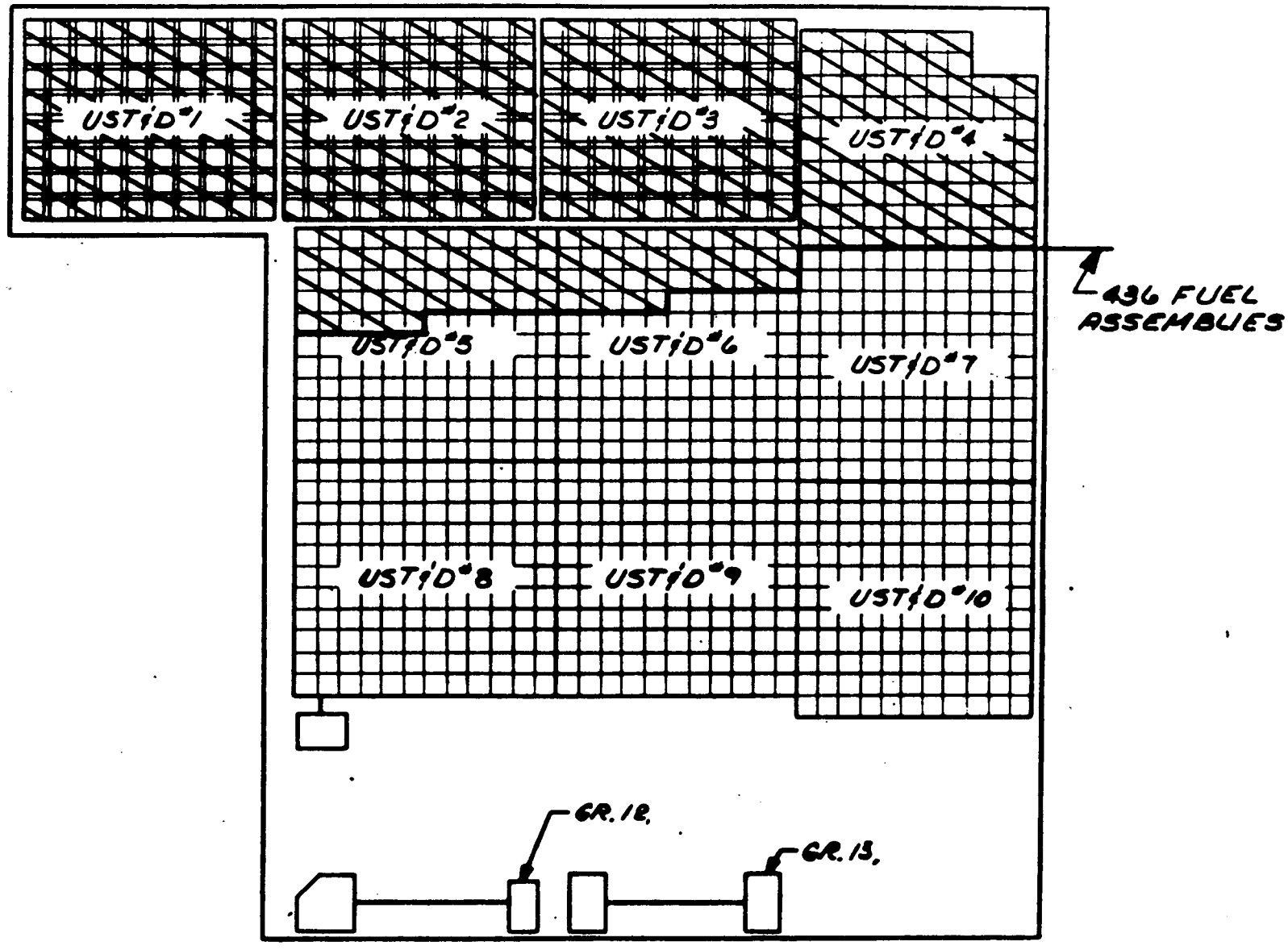

N
UST/D #1
UST/D #2
UST/D #3
UST/D #4
UST/D #5
UST/D #6
UST/D #7
UST/D #8
UST/D #9
UST/D #10
436 FUEL ASSEMBLIES
GR. 12.
GR. 13.

FIGURE 22

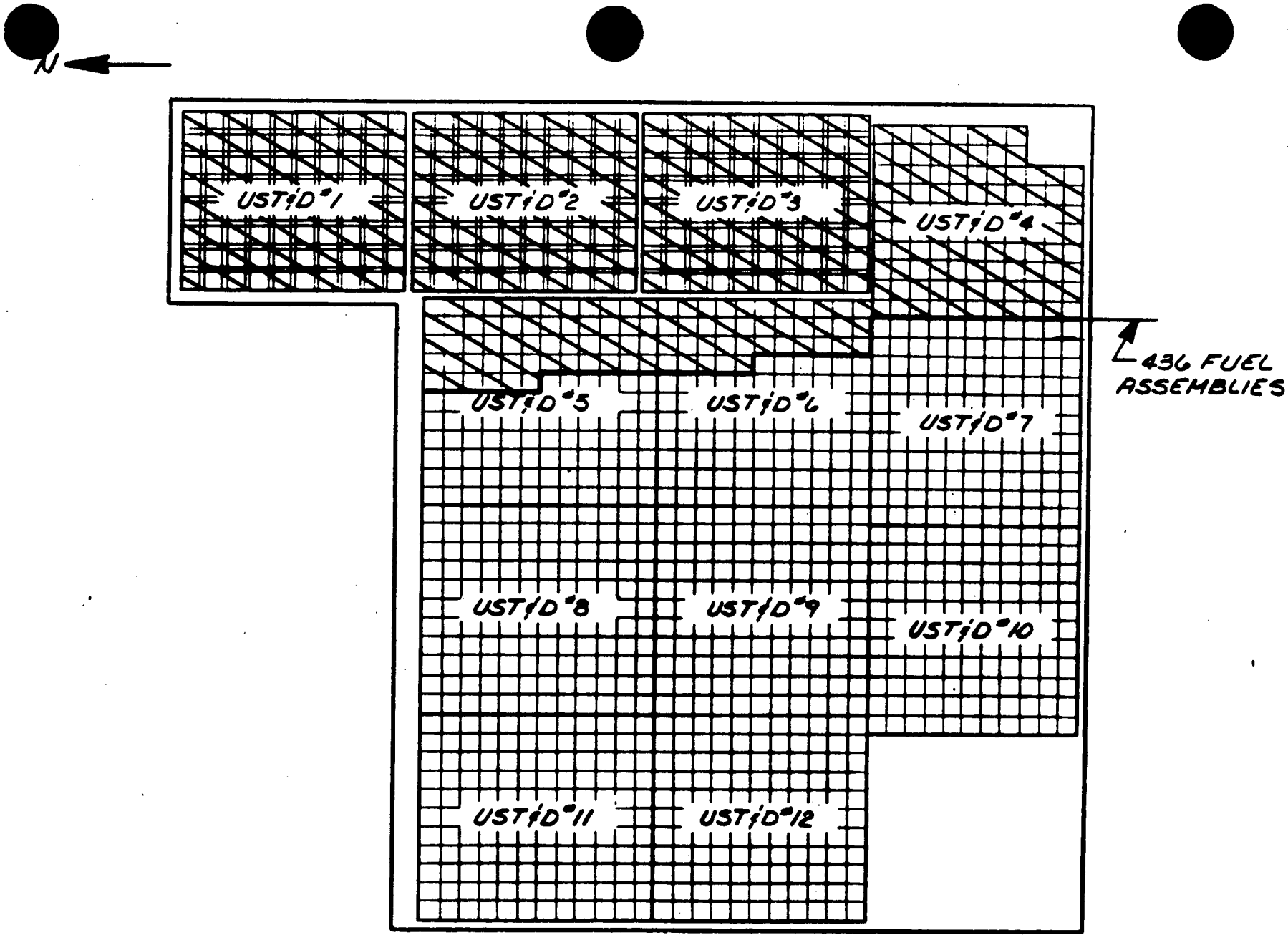

N
UST/D #1
UST/D #2
UST/D #3
UST/D #4
UST/D #5
UST/D #6
UST/D #7
UST/D #8
UST/D #9
UST/D #10
UST/D #11
UST/D #12
436 FUEL ASSEMBLIES

FIGURE 23

Question V.2

Provide details of the proposed installation procedure indicating how the elevations of the racks and designated gaps between the racks will be maintained and monitored.

Response to Question V.2

The detailed procedure for installing the new maximum density racks has not been developed pending selection of an installation contractor. An installation schedule, which will be used for bidding the installation contract, has been developed which provides the rack installation and removal sequence (see Response to Question V.I).

As stated in Attachment II of the Authority's May 9, 1988 letter, the racks are designed to be installed with essentially no gaps between racks. Each rack will be contacting the adjacent rack. The rack locations in the spent fuel pool are shown on UST&D drawing 8721-1, Rev. 3. The water space shown between Region 1 racks is maintained by spacers welded on the base of the racks and all Region 2 racks are installed nominally with no gaps between racks other than normal manufacturing tolerances.

Each rack is supported and leveled on four screw pedestals which bear directly on the pool floor. Where pedestals fall on pool liner plate welds, spreader plates will be used to bridge these welds.

The rack will be leveled within the pool using the racks adjustable pedestals and monitored using surveying instruments. Elevations between racks will be maintained within 1/2" total.

Question V.3

Provide a description of plant safety procedures as related to the spent fuel pool in case of:

a. fuel drop accident
b. a seismic event (OBE exceedance)
c. loss of water from the pool detected by leak chases or by low water level in the pool.

Response to Question V.3

a. Fuel drop accidents are covered under Site Procedure ONOP-RP-2 ("Irradiated Fuel Damage in Fuel Storage Building"). It addresses the indications to the Control Room Operators that would immediately indicate any associated rise in radiation fields and the protective actions that would occur as a result of these alarms. It then addresses the subsequent actions that may need to be taken in order to minimize the air and water activity levels in the Fuel Storage Building. Finally, it specifies the circumstances to consider in the evaluation of the damaged fuel assembly.

b. The operation, startup and shutdown of the seismic monitoring equipment is covered under Site Procedure SOP-S-1 ("Seismic Monitoring Equipment"). In the event of the occurrence of a seismic event exceeding the OBE, Site Procedure ONOP-S-1 ("Seismic Monitoring Equipment Actuation") provides the necessary operator response, depending upon the magnitude and location of the seismic event. This may include bringing the reactor to hot shutdown. The condition of the fuel and Spent Fuel Pit is covered under:

   " ... after a detailed engineering survey of the plant structures and components, an engineering task force will either (1) recommend to the Superintendent of Power that the plant can be returned to power, or (2) specify procedures and analytical work considered necessary before the plant can be returned to power."

c. Loss of water level in the Spent Fuel Pit is signaled in the Control Room by a pool water level alarm. This event is treated the same as if it were a loss of water level in the Reactor Cavity during refueling operations. The associated plant procedure, ONOP-RP-3, addresses Procedure SOP-SFP-1 ("Spent Fuel Pit Cooling and Purification System Operation") in order to provide borated make-up to the Spent Fuel Pit using various methods, e.g. through the Refueling Water Storage Tank or through the CVCS blender. Procedure SOP-SFP-1 also covers operation of the leak detection system in the Spent Fuel Pit.

   All ONOP's (Off-Normal Operating Procedures) address activation of the Site Emergency Plan as appropriate for the events they cover.

FROM CASE 1 DECAY HEAT AT 120 HR AFTER REACTOR SHUTDOWN THE MAXIMUM DECAY HEAT FROM A HOT FUEL ASSEMBLY WILL BE:

$$\gamma = \frac{0.1399E08 \ BTU/HR}{76 \text{ ASSEMBLIES}} = 1.840 \times 10^5 \ BTU/HR/\text{ASSEMBLY}$$

THUS, THE DRIVING HEAD FOR A HOT FUEL ASSEMBLY WILL BE:

$$\Delta P = 2.6\times10^{-4}\left(\frac{1}{°F}\right) \frac{\times 1.84\times10^{5} \ BTU/HR}{2 \times 1.0 \ BTU/LB°F \ Q \ FT^3/SEC \ (3600 \ SEC/HR)} \times 13 \ FT$$

(15) $$\Delta P = \frac{7.973\times10^{-2}}{Q} \ psf \ (\text{with } Q \text{ in cfs})$$

EQUATING THE DRIVING PRESSURE WITH THE PRESSURE LOSS FOR PATH (1) YIELDS:

$$887.3\, Q^{1.931} + 561.6\, Q^2 = \frac{7.973\times10^{-2}}{Q}$$

THEN; $Q = 0.0362 \ FT^3/SEC$ OR 16.2 GPM

OK! 2.197 = 2.20 OR

THEN; $$T_{OUT} = T_{IN} + \frac{1.84\times10^5 \ BTU/HR \times \frac{1}{778} \ HP/BTU/SEC}{(61.7 \ LB/FT^3)(1.0 \ BTU/LB°F)(0.0362 \ FT^3/SEC)}$$

(16) OR $T_{OUT} = T_{IN} + 22.9 \ °F$

FOR THE CASES CONSIDERED:

CASE 1	$T_{IN} = T_{HIN}$, °F	T_{OUT}, °F
1	140	162.9
	150	172.9
2	200	222.9
3	140	162.9
	150	172.9
4	200	222.9

AT THE TOP OF THE FUEL IN THE FUEL ASSEMBLY (13.28' FROM FLOOR) THE WATER PRESSURE IS APPROXIMATELY:

$$\Delta p \cong 14.7^{psia} + (39\tfrac{1}{4}'' - 13.28')\frac{61.7}{144}\ psia = 25.74\ psi$$

AT THIS PRESSURE THE SATURATION TEMPERATURE IS $T_{SAT} \cong 241.8\ °F$ (STEAM TABLES)

THEREFORE, THE MAXIMUM OUTLET WATER TEMPERATURE FROM THE HOTTEST FUEL ASSEMBLY IS WELL BELOW BOILING.

To determine the peak clad temperature, we need the clad heat transfer coefficient at the fuel rod surface:

(17) $$T_{clad} = T_{coolant} + \frac{q''}{h}$$

where h = local heat transfer coefficient.

q'' = clad heat flux.

we take

(18) $$q'' = q''_{max} \sin\left(\frac{\pi z}{L}\right)$$

which will be conservative for spent fuel since

$$q''_{max} = \frac{\pi}{2}\overline{q''} = 1.57\,\overline{q''}$$

$\overline{q''}$ being the average clad heat flux in the fuel assembly.

For the hottest assembly, we assume each of the fuel rods has a heat flux distribution given by eq. (18). For this assembly

$$q = 1.84\times10^5 \frac{BTU}{hr} = \overline{q''}\underbrace{\left[\pi\left(\frac{.422}{12}ft\right)\left(\frac{144}{12}ft\right)(204)\right]}_{A_{clad} = 270.4\ ft^2}$$

$$\overline{q''} = \frac{1.84\times10^5}{270.4} = 680.3\frac{BTU}{hr\cdot ft^2}$$

$$q''_{max} = \frac{\pi}{2}\overline{q''} = 1068.7\frac{BTU}{hr\,ft^2}$$

In the coolant of the fuel assembly,

(19) $$dq = \rho c_p Q\, dT_c$$

$$dq = q'' \pi D N dz = q''_{max} \pi D N \sin\left(\frac{\pi z}{L}\right) dz$$

$$\int_{T_{IN}}^{T_c(z)} dT_c = \frac{q''_{max} \pi D N}{\rho c_p Q} \int_0^z \sin\left(\frac{\pi z'}{L}\right) dz'$$

(20) $$T_c(z) = T_{IN} + \frac{N L D q''_{max}}{\rho c_p Q}\left(1 - \cos\left(\frac{\pi z}{L}\right)\right)$$

or

(21) $$T_c(z) = T_{IN} + \frac{\Delta T}{2}\left[1 - \cos\left(\frac{\pi z}{L}\right)\right]$$

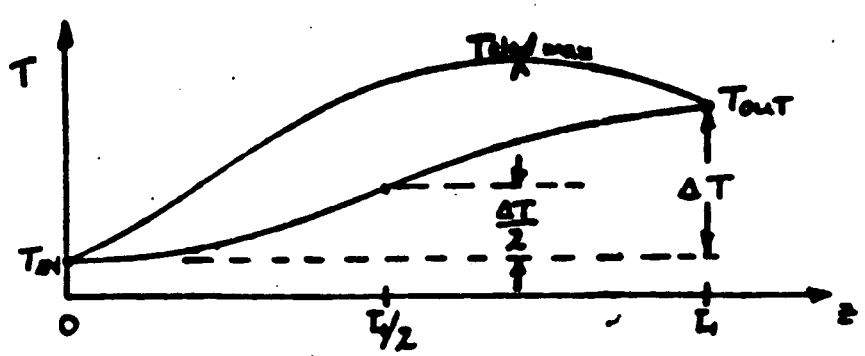
where $\Delta T = T_{OUT} - T_{IN}$ (= 122.9 °F here)

(Note $q = \bar{q}'' \pi N D L = q''_{max} \frac{N D L}{2} = \rho c_p Q \Delta T$)

the clad temperature is then

(22) $$T_{clad}(z) = T_{IN} + \frac{\Delta T}{2}\left[1 - \cos\left(\frac{\pi z}{L}\right)\right] + \frac{q''_{max}}{h} \sin\left(\frac{\pi z}{L}\right)$$

The axial profiles of the coolant and clad temperatures, eqs (21) and (22) look roughly as follows:



The location of $T_{clad\,max}$ can be found by solving $\frac{dT_{clad}}{dz} = 0$:

(23) $$z_{max} = \frac{L}{\pi} \tan^{-1}\left(-\frac{2\,q''_{max}}{h\,\Delta T}\right) \; ; \; \frac{L}{2} \le z_{max} \le L$$

Before determining h and $T_{clad\,max}$, we need to know the type of flow in the fuel assembly. This requires a more detailed knowledge of the flow areas in the fuel assembly.

Fuel Assembly — 15 X 15 fuel bundle.

$$\left\{ \begin{array}{l} D = .422'' \\ P = .563'' \end{array} \right\} \text{REF (G1)}$$

ref Dg { Fuel Box I. width = 8.830 inch

$$D_e = \frac{4\ \text{Flow Area}}{\text{Wetted Perimeter}}$$

$$= \frac{4\left\{\overline{8.83}^2 - (15)(15)\frac{\pi}{4}(.422)^2\right\}}{4 \times 8.83 + \pi(.422)(15)(15)} = 0.557\ \text{in}$$

OR 0.0464 FT

Flow area -

$$A_{ff} = (8.83)^2 - \pi(.225)(.422)^2 = 46.49 \text{ in}^2$$

$$A_{ff} = .323 \text{ ft}^2$$

flow velocity in assembly ($Q = .0362$ ft³/sec)

$$V = \frac{Q}{A_{ff}} = \frac{.0362}{.323}\frac{ft}{sec} = .112 \frac{ft}{sec}$$

$$Re \approx \frac{D_e V}{\nu} \approx \left(\frac{.557}{12}\right)\left(\frac{.112}{5 \times 10^{-6}}\right) = 1039$$

Since $Re < 2000$, the flow is laminar — so a free-convective laminar heat transfer coefficient is needed. Kays and London, ref. K1 gives the Nusselt number vs r^* plot shown on the following page for laminar flow in concentric annuli. We use Nu_{ii} for the inside tube (i.e. clad) heated:

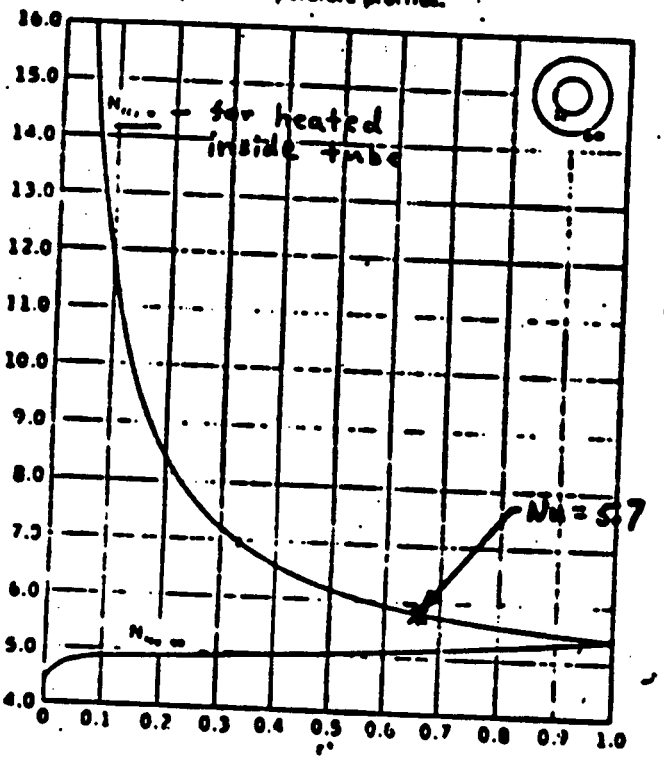
$$r_1 = D/2 = .422/2 = .211''$$

$$r_2 = P\left(\frac{2}{\sqrt{\pi}}\right) = .563\left(\frac{1}{\sqrt{\pi}}\right) = .317''$$

for the outside "tube" with equivalent area for the unit cell.

$$r^* = \frac{r_1}{r_2} = \frac{.211}{.317} = .664 \Rightarrow \underline{Nu = 5.7}$$

Fig. 6-5. Nusselt numbers [redacted] for laminar flow in concentric-circular-tube annuli for constant heat rate per unit of tube length and fully developed velocity and temperature profiles.



For the heat transfer coefficient,

(24) $$Nu = \frac{h D_e}{K} = 5.7$$

we use the <u>largest</u> D_e to obtain the smallest h. For the assembly, $D_e = .557''$, so at T=125 (low k),

$$h = 5.7 \frac{(.376)}{(.557)} (12) = 46.2 \frac{BTU}{hr\ ft^2\ °F}$$

FURTHER TO CALCULATE TEMPERATURE MIB NEED TO BOIL WE NEED TO ADD THE JENS & LOITES PARAMETER, Ref (McAdams HEAT TRANSFER)

$$\Delta T_{J\&L} = \frac{1.9\,(\phi)^{1/4}}{e^{P/900}}$$

FOR OUR CASE; $\phi = \phi_{MAX} \sin\left(2.03 \times \frac{180}{\pi}\right)$ $\boxed{1}$

OR $\phi = 1068 \frac{BTU}{HR\,FT^2} \times 0.89 = 956 \frac{BTU}{HR\,FT^2}$

AND $P = 27$ psia

THEN;

$$\Delta T_{J\&L} = \frac{1.9\,(956)^{1/4}}{e^{27/900}} \cong 10\,^\circ F$$

HENCE, WE ARE SUBCOOLED BY

$$\Delta T_{SUB\ COOLING} = T_{SAT} + \Delta T_{J\&L} - T_{CLADMAX}$$

$$= 245.6 + 10 - 237.2 = \underline{\underline{18.4\,^\circ F}}$$

EVEN FOR WORST CASE CONDITIONS,

PATH 2 - UNDER-RACK FLOW

UNDER-RACK FLOW IS MORE COMPLICATED DUE TO ADDITIONAL FLOW LOSSES AT RACK SUPPORTS, LOCATION OF THE HOT ASSEMBLIES, UNDER-RACK FRICTION LOSSES, AND A KNOWLEDGE OF THE MIXING REGION ABOVE THE RACKS. THE FOLLOWING SHOWS A SKETCH OF THE SPENT FUEL POOL AND THE ASSUMED LOCATION (DARK AREA) OF THE HOTTEST ASSEMBLIES. THE OPERATING SPARGER IS ASSUMED TO BE ON THE SOUTH WALL AND SKIMMING TANK LOCATED AT TOP OF THE SOUTH WALL.

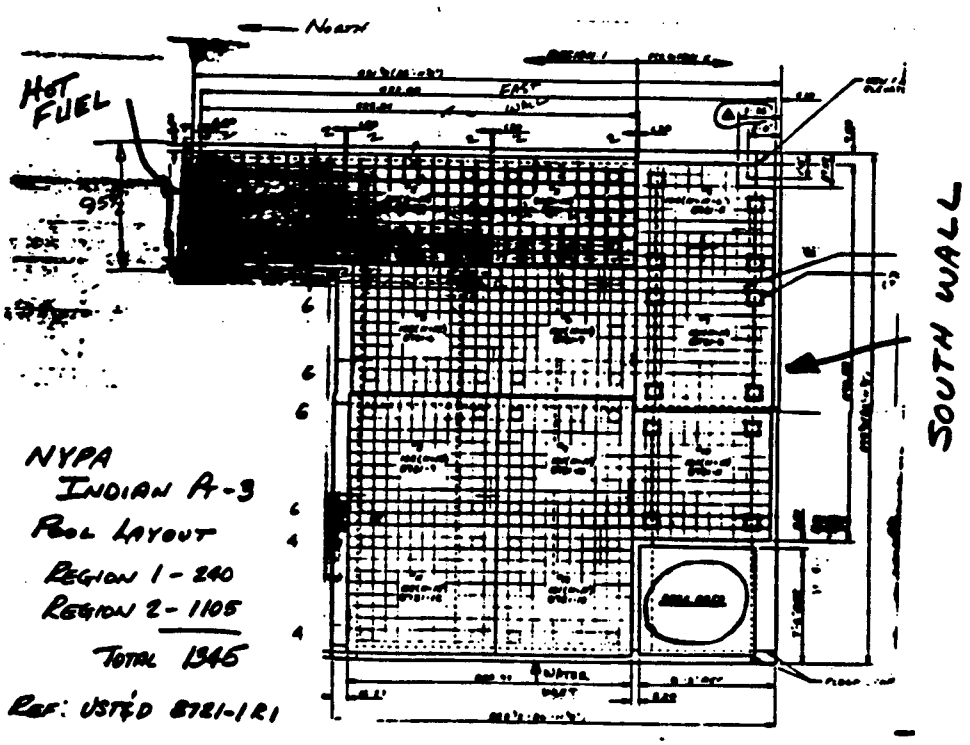
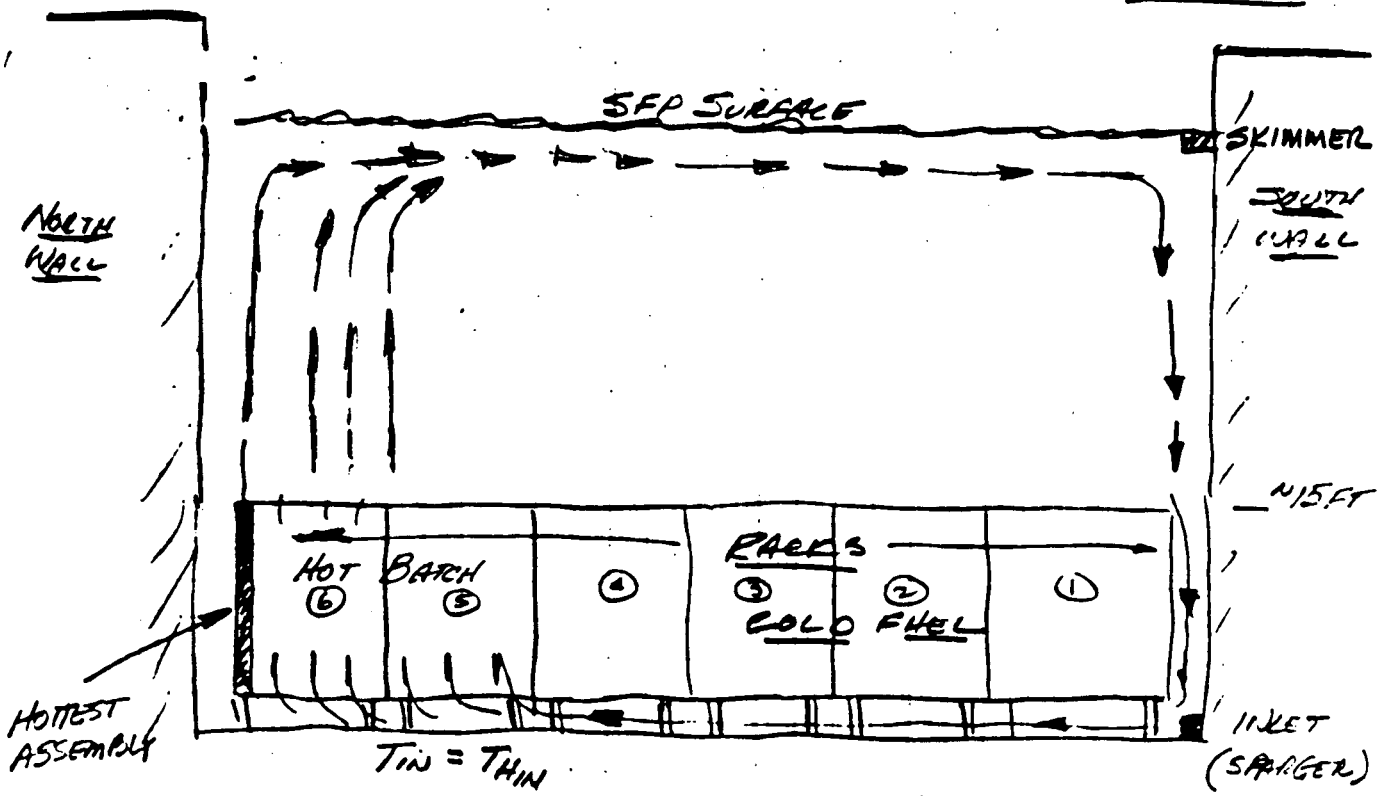


DIAGRAM OF FLOW (PATH 2) IN SPENT FUEL POOL



ELEVATION VIEW

→ ASSUMED FLOW (PATH 2)

SINCE THE ACTUAL PATH (WITH SOUTH SPARGER AND SOUTH SKIMMING TANK) WOULD BE SOUTHWARD AND UPWARD, WE MAY CONSERVATIVELY TAKE $T_{in} = T_{HAVE}$, THE AVERAGE HX TUBE SIDE OR SFP WATER TEMPERATURE.

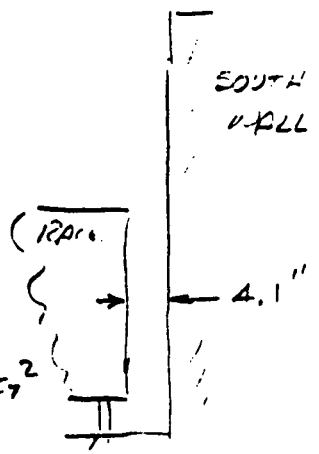
THE FLOW PATH (2) IS OVER THE RACK, DOWN THE SOUTH WALL, UNDER THE RACK AND THEN UP THROUGH THE HOT ASSEMBLY. NO CREDIT IS TAKEN FOR FLOW DOWN THE EAST OR WEST WALLS.

WE IDENTIFY THE FOLLOWING PRESSURE LOSSES IN PATH 2:

1. SOUTH WALL FRICTION $\left(\frac{fL}{D_e}\right)$ LOSSES
2. BEND LOSS AT BOTTOM OF SOUTH WALL.
3. EXPANSION AND CONTRACTION LOSSES UNDER THE RACKS AT THE RACK PEDESTALS.
4. UNDER RACK FRICTION $\left(\frac{fL}{D_e}\right)$ LOSSES
5. FLOW BRANCHING / MOMENTUM LOSSES
6. ENTRANCE TO RACK CELLS, CONTRACTION AND EXPANSION.
7. FUEL ASSEMBLY IN RACK LOSSES. (NOTE: RACK EXIT LOSSES ARE SMALL AND ARE ASSUMED AS A PART OF FUEL ASSEMBLY LOSSES.)

FLOW AREA CALCULATIONS

A. SOUTH WALL - RACK INTERFACE



RACK TO WALL SPACING = 4.10"

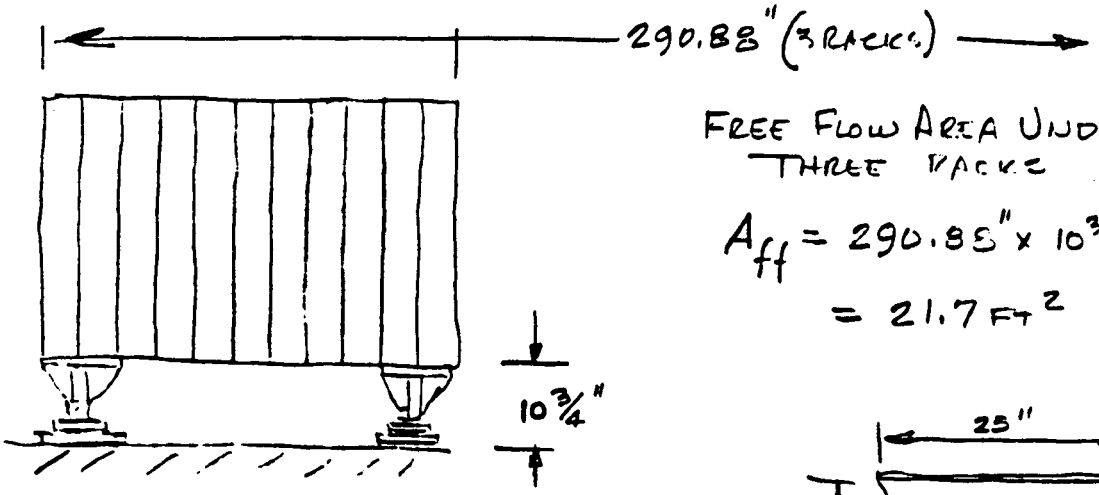
WIDTH OF THREE RACKS = 290.88"

FLOW AREA BETWEEN RACKS AND WALL $= \frac{4.1'' \times 290.8''}{144\ in^2/ft^2} = 8.27\ ft^2$

HYDRAULIC DIAMETER OF GAP BETWEEN WALL AND RACKS $= \frac{4 \times \text{FLOW AREA}}{\text{WETTED PERIMETER}}$

$= \frac{4(8.27)\ ft^2}{2(290.8'')/12''/ft} = 0.682\ ft$

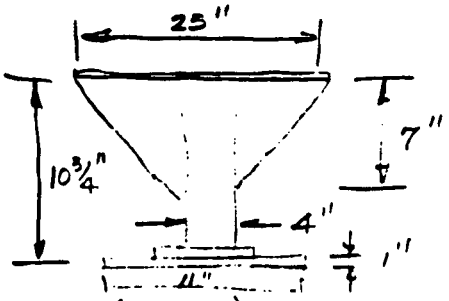
B. UNDER RACK FLOW AREA



FREE FLOW AREA UNDER THREE RACKS

$$A_{ff} = 290.88'' \times 10\tfrac{3}{4}'' \times \frac{1}{144 in^2/ft^2}$$

$$= 21.7 \text{ FT}^2$$



PROJECTED AREA OF PEDESTALS

$$A_{SUPPORTS} \cong \frac{25''}{2} \times 7'' + 4''(10.75-7) + 1''(8'' + 11'')$$

$$\approx 121.5 \text{ in}^2 \text{ per pedestal}$$

FOR 3 RACKS THERE ARE A TOTAL OF 6 PEDESTALS.

$$\text{TOTAL } A_{PEDESTALS} = \frac{121.5'' \times 6}{144} = 5.06 \text{ FT}^2$$

THEREFORE, THE FLOW AREA BETWEEN SUPPORTS $\approx$ 21.7 FT^2 − 5.06 FT^2

$$= 16.6 \text{ FT}^2$$

$$\text{THE FLOW AREA REDUCTION} = \frac{16.6 \text{ FT}^2}{21.7 \text{ FT}^2} = 0.7\%$$

PROP

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