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MIT LMFBR BLANKET RESEARCH PROJECT

Quarterly Progress Report

October 1, 1980 - December 31, 1980

Principal Investigator

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1. General

This quarterly progress report covers work initiated, completed or underway in the period October 1, 1980 through December 31, 1980, since the prior progress report (1).

(1) MIT LMFBR Blanket Research Project, Quarterly Progress Report, July 1, 1980 - September 30, 1980, DOE/ET/37241-44, January 9, 1981.

2. Heterogeneous Assembly Evaluation

2.1 Introduction

The investigations of the neutronics of internally moderated and/or blanketed fast breeder reactor assemblies have been nearly completed as regards beginning-of-cycle snapshot cases. This report will document results for a number of internally moderated, unmoderated, blanketed, and both moderated and blanketed assemblies, all based upon the well-benchmarked LCCEWG-III driver assembly design (1). Two classes of heterogeneous assemblies have been analyzed: first, the "ringed" Moderator design (figure 2.1) described in our previous quarterly report ² and second the "Flux Trap" design (Figure 2.2) which utilizes a central moderator cluster immediately surrounded by a ring of depleted natural uranium pins (U-28 content 99.8%), all placed within the center of a Pu-enriched LCCEWG driver assembly. In all cases, the MIT 30° symmetry version of 2DB was used for the calculations.

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- (1) N.C. Horning, M.B. Parker and R.P. Omberg, "Proposed Third Benchmark Problem for the LCCEWG," (Second Revision) HEDL-8051932, May 1980.
 - (2) MIT LMFBR Blanket Research Project, Quarterly Progress Report, July 1, 1980 - September 30, 1980, DOE/ET/37241-44, Jan. 9, 1981.
 - (3) D.B. Lancaster and M.J. Driscoll "An Assessment of Internal Blankets for Gas-Cooled Fast Reactors" MITNE-237, Oct. 1980.

2.2 Ringed Moderator Description

The Ringed Moderator design will be briefly reviewed. An infinite sea of such assemblies is modelled in 30° symmetry (1/12 assembly) by imposing a reflective boundary condition on all sides. Out of 271 total pins, 12 contain $ZrH_{1.6}$ in a ringed arrangement (zone 3, Fig. 2.1); in unmoderated cases these are replaced by mixed oxide pins. Two lattice model types, referred to as discrete and smeared, are considered separately. In the discrete lattice representation a number density is assigned to each isotope which appears within the superposed mesh triangles as the mesh literally dictates (Fig 2.3), so that some triangles contain only sodium. The smeared lattice steps back to a coarser viewpoint and averages the number densities of neighboring mesh triangles (Fig 2.3) almost as if the whole core were being examined and relatively few mesh triangles per assembly could be assigned. However, in the present case, the geometric mesh remains the same, and no triangles contain only sodium.

2.3 Flux Trap Description

The Flux Trap design (Fig 2.2) is intended to improve on the ringed moderator design by first protecting the mixed oxide pins against excessive power peaking by means of a ring of blanket pins, and second by using these blanket pins to trap moderated neutrons and hopefully thereby to lower the sodium void worth. The base-case design of Fig 2.2 uses 12

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FIG. 2.1
RINGED MODERATOR
ASSEMBLY DESIGN

- TRIANGLE DESIGNATION:
- 1 FUEL, HARD FLUX
 - 2 FUEL, MODERATED FLUX
 - 3 MODERATOR
 - 4 SODIUM, HARD FLUX
 - 5 SODIUM, MODERATED FLUX
 - 6 DUCT, INTERDUCT SODIUM
- INDICATES PIN LOCATION
 — INDICATES REFLECTIVE
 BOUNDARY CONDITION
 X TRIANGLE ROW INDEX

FIG. 2.2

FLUX TRAP
ASSEMBLY DESIGN

TRIANGLE DESIGNATION:

- 1 FUEL, HARD FLUX
- 2 FUEL, MODERATED FLUX
- 3 MODERATOR
- 4 SODIUM,
- 5 SODIUM, MODERATED FLUX
- 6 DUCT, INTERDUCT SODIUM
- 8 DEPLETED URANIUM,
MODERATED FLUX

- PIN LOCATION
- REFLECTIVE BOUNDARY
CONDITION
- x TRIANGLE ROW INDEX

FIG. 2.3

DISCRETE AND
SMEARED MODELS

depleted U28 pins surrounding 7 $\text{ZrH}_{1.6}$ pins at the assembly center, and the remaining outer pins are all enriched mixed oxide. This design has only been studied in the discrete lattice to date. Variations in the number of moderator and blanket pins as well as in the stoichiometry of the moderator (n in ZrH_n) have been modelled.

2.4 Cross Section Generation

The SPHINX code was used to collapse several sets of cross sections. A description of the assumptions built into SPHINX and other details relevant to this study is provided by Lancaster (3). The results labelled "Preliminary" in the previous report (2) were used as the basis for a second iteration. All cross-sections were self-shielded and collapsed (From the 50 group LIB-IV library to the 14 group structure described previously).

Several notes on the cross section sets employed are in order. First, separate sets were not calculated for sodium-voided cases. This was considered unjustified since only 10% of the sodium is removed to calculate the void worth (full voiding was not examined because the discrete lattice contains sodium-only triangles). The errors encountered should be negligible, and any case-to-case bias should be consistent. Second, the moderated cross section set is based on 12 $\text{ZrH}_{1.6}$ pins in a 271 pin assembly, so that small errors will be implicit in any calculation which greatly varies the fuel-to-moderator

ratio (which we will characterize henceforth in terms of the hydrogen-to-heavy-metal (H/HM) ratio). Third, in the moderated cross section set, two fuel pin sets were generated: one in a hard spectrum (for the region physically apart from the moderator pins) and another in a soft spectrum (physically zoned with the moderator pins). The soft set was employed for fuel and blanket pins close to ZrH_m pins, and the hard set was used for all other pins.

2.5 Ringed Moderator Results, Base Case

Three closely linked tables contain the results of comparing moderated versus unmoderated and discrete versus smeared cases.

Table 2.1 presents the base case set of calculations. All cases are in pairs of flooded (sodium-in) and voided (10% sodium removed) runs. An enrichment search was used to determine the critical heavy metal number densities for which $k_{eff} = 1.0000$. These searched-to densities were then used for a calculation of k_{eff} with 10% sodium removed. All the runs in Table 2.1 use $B^2 = 2.8-4 \text{ cm}^{-2}$, the axial buckling calculated by BNL for the UNIMOD central-island-of-moderator design upon which the 12 ZrH_m pin design is loosely based.

Table 2.2 is identical to Table 2.1, except that results listed are for $B^2 = 0.0 \text{ cm}^{-2}$.

Table 2.3 is very similar to the previous two tables, however, a buckling search with fixed enrichment, $\epsilon = 14.75\%$ as in

TABLE 2.1
BASE CORE RESULTS
RINGED MODERATOR DESIGN

$B^2 = 2.8-4 \text{ cm}^{-2}$		NO MODERATION	MODERATED (12-ZrH _{1.6} pins)
DiSCRETE	f, ϵ 8.49% (1)		f, ϵ 8.49% (3)
	v, k 0.99864 (2)		v, k 0.99827 (4)
	Δk_{Na} -0.00136		Δk_{Na} -0.00173
SMEARED	f, ϵ 8.27% (5)		f, ϵ 8.27% (7)
	v, k 1.00189 (6)		v, k 1.00095 (8)
	Δk_{Na} +0.00189		Δk_{Na} +0.00095

KEY

f = Flooded: Na-in

v = Voided: 10% Na removed

ϵ = Enrichment; k = 1,0000

k = k_{eff}

$\Delta k_{Na} = k_{eff} \text{ (voided)} - k_{eff} \text{ (flooded)}$

= Held constant for all table entries

(xx) Run ID number

TABLE 2.2
BASE CASE RESULTS
RINGED MODERATOR DESIGN
ZERO BUCKLING

$B^2 = 0.0 \text{ cm}^{-2}$		NO MODERATION	MODERATED (12-ZrH _{1.6} pins)
DISCRETE	f, ϵ 7.54% (9)	f, ϵ 7.55% (11)	
	v, k 1.00263 (10)	v, k 1.00178 (12)	
	Δk_{Na} +0.00263	Δk_{Na} +0.00178	
SMEARED	f, ϵ 7.54% (13)	f, ϵ 7.55% (15)	
	v, k 1.00277 (14)	v, k 1.00185 (16)	
	Δk_{Na} +0.00277	Δk_{Na} +0.00185	

KEY

f = Flooded: Na-in

v = Voided: 10% Na removed

ϵ = Enrichment; k = 1,0000

k = k_{eff}

$\Delta k_{Na} = k_{eff} \text{ (voided)} - k_{eff} \text{ (flooded)}$

= Held constant for all table entries

(xx) Run ID number

TABLE 2.3
BASE CASE RESULTS
RINGED MODERATOR DESIGN
BUCKLING SEARCH

$\epsilon = 14.75\%$	NO MODERATION	MODERATED (12-ZrH _{1.6} pins)
DISCRETE	f, B^2 1.837-3 (17)	f, B^2 1.824-3 (19)
	v, k 0.98375 (18)	v, k 0.98447 (20)
	Δk_{Na} -0.01625	Δk_{Na} -0.01553
SMEARED	f, B^2 2.375-3 (21)	f, B^2 2.369-3 (23)
	v, k 0.99622 (22)	v, k 0.99627 (24)
	Δk_{Na} -0.00378	Δk_{Na} -0.00373

Key

f = Flooded: Na-in

v = Voided: 10% Na removed

B^2 = Buckling, cm^{-2} , $k = 1.000$

$k = k_{\text{eff}}$

$\Delta k_{Na} = k_{\text{eff}} (\text{voided}) - k_{\text{eff}} (\text{flooded})$

= Held constant for all table entries

(xx) = Run ID number

the LCCEWG driver, was used to find the value of B^2 which causes $k = 1.000$. A k_{eff} calculation using the searched-to B^2 was then done for 10% sodium removed, and Δk_{Na} determined.

2.6 Discussion, Base-Case Results, Ringed Moderator Design

Comparing entries (1) to (3) and (5) to (7) in Table 2.1, the enrichment required for criticality is identical for moderated and unmoderated cases, therefore the heavy metal loading is decreased with moderation (259 pins are fueled vs 271). This result is not observed unless separate hard and soft zone cross sections are employed. The sodium void worth is negative for all discrete cases and positive for all smeared cases. While no concrete explanation for this effect has yet been advanced, Tables 2.2 and 2.3 show that a major factor is neutron leakage. The moderated void worth is consistently lower (more negative or closer to zero) than the unmoderated void worth.

When $B^2 = 0.0$, as in Table 2.2, ϵ becomes identical for discrete/smeared case pairs, and differs by only + 0.01% when moderation is added. Thus, again, moderation decreases the required heavy metal loading. The elimination of axial leakage equalizes the discrete/smeared pairs. Void worth is greater for all Table 2.2 entries than the corresponding Table 2.1 entries because leakage contributes a negative reactivity addition. Void worth is still consistently lower for the moderated cases, though consistently positive once axial leakage is removed.

The opposite extreme is seen when buckling is searched for a critical assembly: Void worth is then consistently negative (Table 2.3), and the unmoderated cases have a more negative void worth than the moderated cases. The relative B^2 values are consistent with Table 2.1 results for ϵ (higher B^2 with lower ϵ). Void worth is unusually large for the discrete cases.

These results are difficult to condense into firm conclusions, because of the unusual effects observed due to problem representation. Tentatively,

1. The dominant contributor to sodium void worth in these assemblies is axial leakage. Only in the smeared cases, Table 2.1, does spectrum hardening overcome axial leakage.
2. The combined effects of spectrum hardening, loss of absorption and increased leakage are not truly separable or additive, even for small (10%) changes in sodium number density.
3. The low positive, or negative void worth in the discrete cases is probably due to preferential leakage in the sodium channels, and possibly due to an inexact treatment of the diffusion coefficient in the basic cross section set preparation and collapse.

2.7 Effects of Heterogeneous Moderation

Four configurations test the effect on k_{eff} of the distribution of the moderator:

1. Discrete/clumped. The base case, Figure 2.1. Search ϵ to critical, then void with searched-to number densities, $\text{ZrH}_{1.6}$ is contained in moderator pins. There are "Na only" triangles in the lattice.
2. Discrete/Dispersed. There are no moderator pins; $\text{ZrH}_{1.6}$ is distributed evenly in all 271 fuel pins (Fig 2.4). The amount of $\text{ZrH}_{1.6}$ distributed is equal to that contained in the 12 moderator pins of the base case, as is the amount of heavy metal. There are "Na only" triangles in the lattice.
3. Smeared/Clumped. $\text{ZrH}_{1.6}$ is contained in moderator pins (Fig 2.3). Fuel pins and their surrounding sodium are smeared together.
4. Smeared/Dispersed. Fuel pins (containing $\text{ZrH}_{1.6}$ as in 2. above) and surrounding sodium are smeared together (Fig 2.5).

Table 2.4 summarizes the results, the same buckling and enrichment are used in all cases. It can be concluded that:

1. Heterogeneously distributed (clumped) moderator is more advantageous than homogeneously distributed (dispersed) moderator, in so far as k_{eff} is concerned.
2. Result 1. is independent of the assembly representation (discrete or smeared).
3. Sodium void worth is barely sensitive to moderator distribution, negligibly so for the smeared representation.
4. Comparing cases (27) and 28) against (7) and (8) of Table

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FIG. 2.4

DISCRETE/DISPERSED
ASSEMBLY MODEL

TRIANGLE DESIGNATION:

- 2 FUEL AND MODERATOR
- 5 SODIUM, MODERATED FLUX
- 6 DUCT, INTERDUCT SODIUM

FIG. 2.5

SMEARED / DISPERSED
ASSEMBLY MODEL

TRIANGLE DESIGNATION:

1 FUEL AND MODERATOR

2 FUEL AND MODERATOR

6 DUCT, INTERDUCT SODIUM

TABLE 2.4
Runs Pertinent to Heterogeneity of
Moderator Ringed Moderator Design

$\epsilon = 8.49\%$ $B^2 = 2.8-4\text{cm}^{-2}$	MODERATOR DISTRIBUTION		Δk_{HET}
	CLUMPED	DISPERSED	
DISCRETE	f 1.0000 (1)	f 0.99789 (25)	f - 0.00211
	v 0.99827 (2)	v 0.99626 (26)	v - 0.00201
	Δk 0.00173	Δk 0.00163	Δk - 0.00010
SMEARED	f 1.01386 (27)	f 1.01173 (29)	f - 0.00213
	v 1.01483 (28)	v 1.01273 (30)	v - 0.00210
	Δk +0.00097	Δk +0.00100	Δk - 0.00003

Key

f = Flooded k_{eff} : Na in

v = Voided k_{eff} : 10% Na removed

Δk = k_{eff} (voided) - k_{eff} (flooded)

= Held constant for all table entries

(xx) = Run ID number

2.1 proves the equivalence of the search/void and flood/void approaches. In other words, void worth is insensitive to small changes in enrichment, all other parameters being equal.

Since the fabrication of clumped moderator assemblies could be simpler than for dispersed moderator assemblies, and no neutronic advantages are inherent in either type, it can be concluded that moderated assemblies should be heterogeneously moderated.

2.8 Sodium Void Worth Breakdown

The spatial components of the sodium void worth for a moderated, radially reflected assembly can be distinguished by selectively voiding individual zones of the assembly. Figure 2.6 shows that sodium can be selectively removed from 1. The fuel pin triangles, 2. The Na-only triangles, 3. The duct/interduct gap triangles, and combinations of the above. The results for $B^2 = 2.8-4 \text{ cm}^{-2}$ and $B^2 = 0.0$ are collected in Table 2.5.

Comparing lines 4. and 5., the absence of leakage causes the individual worths of sodium-only and fuel pin triangles to be negative, while the combined effects result in a positive worth. Some tradeoffs between absorption and spectral changes must cause this. In contrast, with $B^2 = 2.8-4 \text{ cm}^{-2}$, the individual worths do sum to the combined worth. Therefore the major void worth component is leakage, which does sum within

FIG. 2.6

MESH TRIANGLE
LAYOUT FOR
SODIUM VOID WORTH
BREAKDOWN

TABLE 2.5
Sodium Void Worth Breakdown
Ringed Moderator Design

$\epsilon = 8.49\%$	$B^2 = 2.8-4 \text{ cm}^{-2}$	$B^2 = 0.0$
DISCRETE	Δk_{Na}	Δk_{Na}
1. PIN (Fuel)	-0.00225 (31-3)	-0.00393 (35-34)
2. NA only	-0.00203 (32-3)	-0.00313 (36-34)
3. DUCT	+0.00012 (33-3)	+0.00052 (37-34)
4. (1.) + (2.)	-0.00428	-0.00706
5. PIN + NA	-0.00433 (38-3)	+0.00142 (39-34)
6. (1.) + (2.) + (3.)	-0.00416	-0.00654
7. WHOLE ASSEMBLY	-0.00173 (4-3)	+0.00199 (40-34)
8. (3.) + (5.)	-0.00421	+0.00194

Key

PIN (FUEL): 10% Na void, Na in fuel pin triangle only
 NA ONLY: 10% Na void, Na in Na triangle only
 DUCT: 10% Na void, Na in duct/interduct gap triangle only
 PIN + NA: 10% Na void fuel pin and Na only triangles together
 WHOLE ASSEMBLY: 10% Na void all triangles
 (a.) + (b.): SUM OF RESULTS, LINES a AND b
 (xx-yy): k (voided case xx) - k (flooded case yy)
: Held constant for all table entries

this range of sodium number density. Additive superposition breaks down when the duct worth is added in: comparing lines 6 and 7, voiding the entire buckled assembly results in a less negative worth than does addition of the separate effects. This discrepancy could be due to the duct wall reflective (white) boundary condition; noting line 3, voiding the duct only will harden the spectrum and decrease absorption. Finally, comparing the total worths for the zero buckling cases: lines 6 and 7 are not equal but lines 7 and 8 are, indicating that once enough sodium is voided, voiding still more will add worth in the same direction.

2.9 Flux Trap Configuration - Further Description

Four types of flux trap assemblies have been studied to date. First, the unmoderated but blanketed assembly (Fig 2.7), which consists of 19 depleted U28 pins centrally placed in an LCCEWG driver assembly. Second, the moderated flux trap, of which there are several variations: The basic 7-ZrH_{1.6}, 12-U28 design introduced above (Fig 2.2), a version with 1-Zr_{1.6} (in the center), and 18-U28 pins (Fig 2.8), and a 19-ZrH_{1.6}, 18-U28 design (Fig 2.9). The different designs will hereafter be designated by FT(m,n) where m is the number of moderator pins and n is the number of blanket pins (both of which displace fuel pins in the 271-pin driver assembly).

The standard search enrichment/remove 10% sodium procedure employed earlier has been used to compare void worth among the various flux trap assemblies described. In addition, the FT(7,12) design was used to study the effects of varying the ZrH_n stoichiometry between ZrH_{1.6} and ZrH_{0.4}. Finally, the FT(19,18) design was used to study the differences between the discrete and smeared lattice representations.

The fissile inventory varies, at fixed enrichment, among these assemblies because the number of driver pins varies. Also, the power produced by each assembly differs slightly, though the specific power (kw/ft) of the driver pins should be held constant. In order to equalize the comparisons, a "power factor" may be defined relative to the base case; this will allow burnup

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FIG. 2.7

FT (0,18)
0 - ZrH
18 - U28
FLUX TRAP
ASSEMBLY

TRIANGLE DESIGNATION:

- 1 FUEL, HARD FLUX
- 4 SODIUM, HARD FLUX
- 6 DUCT, INTERDUCT SODIUM
- 8 DEPLETED URANIUM (BLANKET)

O PIN LOCATION
- REFLECTIVE BOUNDARY
CONDITION

X TRIANGLE ROW DESIGNATION

FLUX TRAP
ASSEMBLY

TRIANGLE DESIGNATION:

- 1 FUEL, HARD FLUX
- 3 MODERATOR
- 4 SODIUM, HARD FLUX
- 5 SODIUM, MODERATED FLUX
- 6 DUCT, INTERDUCT SODIUM
- 8 DEPLETED URANIUM (BLANKET)
- O INDICATES PIN LOCATION
- INDICATES REFLECTIVE BOUNDARY CONDITION
- X TRIANGLE ROW INDEX

FIG. 2.9

FT (19, 18):

19 - ZrH

18 - U28

FLUX TRAP
ASSEMBLY

TRIANGLE DESIGNATION:

- 1 FUEL, HARD FLUX
- 2 FUEL, MODERATED FLUX
- 3 MODERATOR
- 4 SODIUM, HARD FLUX
- 5 SODIUM, MODERATED FLUX
- 6 DUCT, INTERDUCT SODIUM
- 8 DEPLETED URANIUM (BLANKET)
- 0 PIN LOCATION
- REFLECTIVE BOUNDARY
CONDITION
- 2 TRIANGLE ROW INDEX

FIG. 2.10

24 MESH/HEX
SMEAR ZONES
IN ONE
ASSEMBLY

TRIANGLE DESIGNATION:

1, BASE FUEL, HARD FLUX

1, PT(19,18) FUEL, MODERATOR,
AND BLANKET

2 FUEL, HARD FLUX

ALL TRIANGLES CONTAIN SODIUM

O PIN LOCATION

-- 24 MESH/HEX ZONE BOUNDARY

— REFLECTIVE BOUNDARY CONDITION

x TRIANGLE ROW DESIGNATION

cases to be treated equivalently. Moreover, the reciprocal of the power factor is the ratio of the number of assemblies in a core composed only of FT(m,n) assemblies to the number of assemblies in a core composed only of base case assemblies, given that both cores produce the same amount of energy. Thus, dividing the assembly loading, KGHM, by the power factor results in an adjusted assembly loading which compensates for the core size effect. On the assembly level the flux trap may require less HM, but the entire core may require enough extra assemblies to actually increase the HM loading.

The power factor is computed for each assembly based on the number of driver pins, which produce the same specific power (Kw/ft) in all assemblies, and the number of blanket pins, whose power output varies according to the spectrum. Defining f_D and f_B as the fractions of assembly power produced by Driver and Blanket pins, and D and B as the number of such pins, the assembly power relative to the base can be written

$$\frac{P}{P_O} = f_D D + f_B B \quad (1)$$

Since $P/P_O = 1$ when $D = 271$, $f_D = (271)^{-1}$. Since $P/P_O =$ (blanket pin power)/(driver pin power) $\equiv \gamma$ when $B = 271$; $\gamma =$ (fast fissions in U28)/(all driver fissions), $f_B = \gamma (271)^{-1}$. For each type of assembly,

$$\gamma \approx \frac{\Sigma_f^{28}}{\Sigma_f^{\text{Driver}}} \quad (2)$$

and the Σ_f^i are composed of one-group cross sections collapsed in the appropriate spectrum. The power factor finally becomes

$$\frac{P}{P_0} = \frac{D}{271} + \frac{\gamma B}{271} \quad (3)$$

This expression ignores any contribution due to gamma heating in the moderator pins, which will be relatively negligible.

2.10 Flux Trap Results

Table 2.6 summarizes the enrichment search/10% sodium void results for the flux trap assemblies described above. As expected, all discrete, $B^2 = 2.8-4 \text{ cm}^{-2}$ cases exhibit negative void worth while all $B^2 = 0.0$ cases exhibit positive void worth. FT(7,12) and FT(19,18) appear interesting because Δk_{Na} is less than that for the base case, even with $B^2 = 0.0$.

A comparison of just the enrichment and void worth between these assemblies is misleading because of the variation in fissile inventory. Table 2.7 reports the enrichment, loading, breeding ratio, and power factor (defined previously) for each assembly. The base case and FT(10,19) actually require the same mass of Pu per assembly, but to construct cores of equal power the FT(10,19) assembly requires more plutonium. Despite the equality in loading and breeding ratio, FT(0,19) has a higher positive void worth (when $B \equiv 0.0$). Hence this form of flux trap, with no moderation does not achieve the desired effect. The results for FT(1,19) show that by adding moderator this excess void worth is diminished, though the breeding ratio also declines.

TABLE 2.6
FLUX TRAP COMPARISON

DISCRETE	$B^2 = 2.8-4 \text{ cm}^{-2}$	$B^2 = 0.0$
F BASE L FT (O FT (1,18) O FT (7,12) D FT (19,18)	$k = 1.0000$ $\epsilon = 8.49\% (1)$ 9.21% (41) 9.08% (43) 8.85% (45) 8.82% (47)	$k = 1.0000 (9)$ 1.06956 (49) 1.06841 (51) 1.06439 (53) 1.05946 (55)
10% BASE V FT (0,19) O FT (1,18) I FT (7,12) D FT (19,18)	$k = 0.99864 (2)$ 0.99882 (42) 0.99843 (44) 0.99820 (45) 0.99829 (48)	$k = 1.00263 (10)$ 1.07268 (50) 1.07130 (52) 1.06663 (54) 1.06138 (56)
Δk BASE FT (FT (1,18) N FT (7,12) A FT (19,18)	-0.00136 -0.00118 -0.00152 -0.00180 -0.00171	+0.00263 +0.00312 +0.00289 +0.00224 +0.00192

key

ϵ : driver (fissile inventory)/(fissile plus fertile)

Flooded: Na-in

10% voided: 10% Na removed

Δk Na: $k_{eff}(\text{void}) - k_{eff}(\text{flood})$

FT(m,n): m moderator pins, n blanket pins, 271-m-n driver pin

BASE CASE: 271-pin LCCEWG driver assembly

= Held constant for table entries

The lower void worths of FT(7,12) and FT(19,18) come at great cost when the adjusted Pu inventories are compared. Moreover, neither assembly has a breeding ratio comparable to the base case. These flux traps work as intended to lower the void worth, but are economically disadvantageous due to the increased loading requirements.

A consistent trend characterizes the FT assemblies compared to the base case: As moderation increases (read left to right in Table 2.7, and top to bottom in the Δk_{Na} list of Table 2.6) void worth, driver pin enrichment, breeding ratio, and the power factor all decrease. However, the adjusted Pu inventory always remains above that of the LCCEWG driver assembly.

2.11 The effects of Stoichiometric Variation in the Moderator

The stoichiometry of the moderator was varied from $ZrH_{1.6}$ to $ZrH_{1.2}$, $ZrH_{0.9}$, and $ZrH_{0.6}$ in the FT(7,12) assembly. Table 2.8 indicates that k_{eff} declines uniformly in both flooded and voided cases, though the difference, Δk_{Na} , does not. $ZrH_{1.6}$ is more advantageous with respect to void worth than the other stoichiometries. It is important to note that as n decreases in ZrH_n , Zr is added at the expense of H, so that absorption is substituted for moderation. The number of neutrons downscattered by the moderator is reduced, though the mean lethargy increment remains constant (insofar as scattering from hydrogen goes). k_{eff} declines in all cases as absorption increases. It is possible that Δk_{Na} would be affected more by a change in the mean lethargy increment

TABLE 2.7

COMPARISON OF FLUX TRAP VERSIONS

DISCRETE	BASE CASE	FT (0,19)	FT (1,18)	FT (7,12)	FT (19,18)
U25 + U28, kg	7.271-2	7.272-2	7.244-2	7.088-2	6.790-2
ALL Pu, kg	8.809-3	8.809-3	8.773-3	8.543-3	7.900-3
ASSEMBLY TOTAL HM, kg	8.152-2	8.152-2	8.122-2	7.942-2	7.581-2
ϵ , DRIVER PIN	8.49%	9.12%	9.08%	8.85%	8.24%
BREEDING RATIO	1.3158	1.3158	1.2921	1.2077	1.100
POWER FACTOR	1.0000	0.9422	0.9418	0.9377	0.8744
ADJUSTED Pu, kg	8.809-3	9.349-3	9.315-3	9.111-3	9.035-3
ADJUSTED HM, kg	8.152-2	8.652-2	8.624-2	8.470-2	8.670-2

key

ϵ : Enrichment of driver pins

All Pu: P49 + P40 + P41 + P42

TOTAL HM: Sum of U and Pu inventory = total Heavy Metal

POWER FACTOR: (Power produced by this assembly/Base assembly power)

FT (m,n): m moderator pins, n blanket pins, 271-m-n driver pins

BASE CASE: 271-pin LCCEWG driver assembly

TABLE 2.8
FLUX TRAP MODERATION:
STOICHIOMETRIC VARIATION

<div style="border: 1px solid black; padding: 2px; display: inline-block;"> $\epsilon = 8.85\%$ 7 ZrH pins 12 U28 pins DISCRETE </div>		ZrH _n	$B^2 = 0.0$	$B^2 = 2.8-4 \text{ cm}^{-2}$
F	n = 1.6		k = 1.06439 (53)	k = 1.0000 (45)
L	1.2		1.06270 (57)	----
O	0.8		1.06053 (59)	0.99491 (63)
O	0.4		1.05811 (61)	----
D				
E				
D				
10%	n = 1.6		k = 1.06663 (54)	k = 0.99820 (46)
V	1.2		1.06551 (58)	----
O	0.8		1.06304 (60)	0.99341 (64)
I	0.4		1.06083 (62)	----
D				
Δk	n = 1.6		+0.00224	-0.00180
	1.2		+0.00281	----
Na	0.8		+0.00251	-0.00150
	0.4		+0.00272	----

key

Flooded: Na-in

10% voided: 10% Na removed

Δk Na: $k_{\text{eff}}(\text{void}) - k_{\text{eff}}(\text{flood})$

ZrH_n: Subscript n indicates stoichiometry

: Held constant for all table entries

xx: Run ID number

imposed by the moderator, rather than by a change in the number of neutrons experiencing this increment. This can be accomplished by using light atoms other than hydrogen, such as the beryllium used in the FMSR, in the moderation zone. A harder driver zone spectrum, as would be generated using metal or carbide fuel in place of the oxide fuel used here, would also be expected to have a beneficial effect.

2.12 The Effects of Lattice Representation

One flux trap assembly, FT(19,18), was chosen to compare the effects of the lattice representation with that of the base case. A consistent smeared representation is obtained by calculating number densities as if a whole-core analysis were to be employed, modelling each assembly with 24 mesh intervals (as BNL used in analyzing the UNIMOD core). Figure 2.10 shows the smeared zones resulting from this approach as applied to the base case and FT(19,18) assemblies. In the diffusion calculations, the base case number density set, $\epsilon = 8.49\%$, was used and k was searched for the smeared base cases; the flux trap number densities were searched to critical. These smeared results are collected in Table 2.9.

The Δk_{Na} values are listed in Table 2.10, comparing the discrete and smeared representations. As expected, all smeared Δk_{Na} values are positive. Further, when $B^2 = 0.0$, Δk_{Na} is greater in the smeared cases, paralleling the results of Table 2.2. Again, the Δk_{Na} values for the smeared, $B^2 = 2.8-4$ cases

TABLE 2.9

FLUX TRAP 24 MESH/HEX
SMEARED REPRESENTATION

SMEAR 24 MESH/HEX	$B^2 = 0.0$	$B^2 = 2.8-4 \text{ cm}^{-2}$
BASE CASE	ϵ 8.49% f 1.06897 (65) v 1.07221 (66)	f 1.01699 (67) v 1.01911 (68)
FT (19,18)	8.76% f 1.04293 (69) v 1.04468 (70)	f 1.0000 (71) v 1.00100 (72)

key

ϵ = driver pin enrichment

f = flooded: Na-in

v = voided: 10% sodium removed

BASE CASE: 271-pin LCCEWG driver assembly

FT (m,n): m moderator pins, n blanket pins, 271-m-n driver pins

: held constant all cases

(xx) run ID number

SMEAR 24 MESH/HEX number densities calculated as if a whole-core analysis modelled each assembly with 24 mesh intervals.

TABLE 2.10
COMPARISON OF FLUX TRAP REPRESENTATIONS
 Δk_{Na} VALUES

REPRESENTATION	B^2, cm^{-2}	LCCEWG BASE CASE	FT (19,18)
DISCRETE	2.8-4	-0.00136	-0.00171
	0.0	+0.00263	+0.00192
SMEARED 24 MESH/HEX	2.8-4	+0.00212	+0.00100
	0.0	+0.00324	+0.00175

are lower than those for the discrete, $B^2 = 0.0$ cases. Note that the base case, smeared void worths differ among Tables 2.1, 2.2, and 2.10 because ϵ is different in each case; the change is consistent in that worth increases with ϵ .

Based on this comparison, the smeared flux trap assembly will outperform the base case, over the range of B^2 values examined in terms of void worth.

2.13 Conclusions

The overall results obtained so far indicate that the insertion of a strong moderator into an otherwise conventional LMFBR assembly offers only modest advantages in terms of less positive sodium void reactivity. This is achieved at the expense of an increased fissile loading requirement and a decrease in the breeding ratio. It is possible, however, that the advantages would be enhanced if the initial spectrum were harder (metal or carbide fuel) and/or the moderator were weaker, so that colliding neutrons lose less energy (e.g., replacing $ZrH_{1.6}$ by BeO) -- as indeed is the case in the Fast Mixed Spectrum Reactor. The optimal endpoints (if any) of this spectral shift need to be pinned down.

The results also suggest that there is little advantage to intra-assembly heterogeneity: apparently inter-assembly heterogeneity is required to realize substantial modifications in core behavior, as in common heterogeneous core designs. We will, therefore, investigate the LCCEWG-III core in which moderator is added to the depleted uranium blanket assemblies -- i.e., moving closer to Atefi's original breed/burn configurations.

3.0 Engineering Compatibility of Heterogeneous Assembly Designs

As reported in the previous progress report, a 30° sector (1/12 core) of a LCCEWG III assembly with a 271 pins uniform assembly (1) has been modelled using COBRA-IV-I to make a thermal hydraulic analysis. The original modelling was done using wire wrap on the pins, but it was determined that COBRA-IV-I cannot handle wire wrap on partial rods which fall on the lines of symmetry, as shown in Fig 3.1. When the wire wrap is removed the program works. Running a full duct of 271 pins with wire wrap would solve this problem; however it would be too expensive for the number of runs of current interest. Alternatively the COBRA-IV-I code would have to be modified if 30° sectors (or any partial geometry having partial rods on lines of symmetry) are to be analyzed with wire wrap, which is quite a formidable task.

An effort has been made to simulate the effect of wire wrap on a 7-pin, full duct sample problem. Pressure drop across the core can be matched by increasing the friction factor in the subchannels. Several computer runs were made to determine a friction factor which matches the pressure drop.

The friction factor relation used in COBRA for the actual problem modelled, as well as for the 7-pin wire-wrapped bundle sample problem, was:

$$f = \frac{0.316}{0.25^{0.25}} \quad (1)$$

(Re)

FIG. 3.1 ROD AND SUBCHANNEL LAYOUT

where,

f = dimensionless friction factor

R_e = Reynolds number

The above empirical formula given by Blasius is good for smooth-pipe in turbulent flow up to $R_e = 100,000$.

There is a considerable discrepancy in friction factor data for rod bundles with $\frac{P}{D}$ greater than one. In general, there is a tendency towards increasing friction factors with increasing $\frac{P}{D}$. This is probably due to cross flow between channels in the rod bundle and turbulent sheer stress at fluid-fluid interfaces (2).

For tests on a 37-rod bundle of 3 foot heated length with 0.625 inch diameter rods arranged on a triangular lattice with $\frac{P}{D} = 1.46$ the friction factor correlated as (3)

$$f = \frac{0.296}{(R_e)^{0.2}} \quad (2)$$

This gives values of the friction factor about 60 percent higher than the Moody curve for smooth pipes.

Experiments on a 6x6 rod bundle of 11.5 foot length with 0.59 inch diameter rods with $\frac{P}{D} = 1.3$ gives (4)

$$f = \frac{0.337}{(R_e)^{0.24}} \quad (3)$$

In order to simulate the pressure drop effect across the core both Eqs (2) and (3) and several other versions were tried on the 7-pin sample problem. A very close match for the pressure drop was obtained by using

$$f = \frac{0.296}{(Re)^{0.22}} \quad (4)$$

To match the coolant velocity in the case of no wire wrap, clad thickness was increased according to the relation,

$$(D_c)_{new} = \left[(D_c)_{old}^2 + (D_w)^2 \right]^{\frac{1}{2}} \quad (5)$$

where, $(D_c)_{old}$ = old clad diameter

D_w = wire wrap diameter

$(D_c)_{new}$ = new clad diameter

The increased-clad-thickness approach matches temperature, enthalpy and velocity in all subchannels very closely.

The only phenomenon which cannot be simulated or matched is the cross flow between adjacent subchannels, regardless of the value used for the turbulent mixing parameter. Fortunately, cross flow is not of much concern as long as pressure, temperature, enthalpy and velocity in the subchannels are otherwise reasonably well matched. Attempts are currently being made to model wire wrap by omitting it only on the partial rods on the lines of symmetry, but including it on interior rods.

The results for the representative subchannels 5, 25, and 48 for the problems modelled without wire-wrap are presented in Table 3.1.

During the coming quarter several internally heterogeneous assemblies will be modelled and an analysis made of the thermal-hydraulic penalties associated with such designs.

TABLE 3.1

LCCEWG BENCHMARK 3

RESULTS FOR CONFIGURATION I - UNIFORM
ASSEMBLY, NO WIRE WRAP

CODE USED: COBRA-IV-I

CHANNEL 5

DISTANCE (IN.)	DELTA-P (PSI)	ENTHALPY (BTU/LB)	TEMPERATURE (°F)	VELOCITY (FT/SEC)
0.0	33.0952	349.76	650.00	24.9816
4.0	30.2693	351.56	655.87	25.4456
8.0	27.4348	356.82	673.02	25.7625
12.0	24.6183	365.10	700.03	25.9650
16.0	21.8146	375.86	735.09	26.1340
20.0	19.0221	388.43	776.11	26.3035
24.0	16.2444	401.88	819.94	26.4770
28.0	13.4859	415.32	863.79	26.6484
32.0	10.7497	427.91	904.83	26.8095
36.0	8.0368	438.67	939.91	26.9521
40.0	5.3429	446.94	967.00	27.0737
44.0	2.6628	452.15	984.32	27.1570
48.0	0.0	453.91	990.19	27.1848

LCCEWG BENCHMARK 3

RESULTS FOR CONFIGURATION I - UNIFORM
ASSEMBLY, NO WIRE WRAP

CODE USED: COBRA-IV-I

CHANNEL 25

DISTANCE (IN.)	DELTA-P (PSI)	ENTHALPY (BTU/LB)	TEMPERATURE (°F)	VELOCITY (FT/SEC)
0.0	33.1106	349.76	650.00	24.9816
4.0	30.2661	351.56	655.85	25.4854
8.0	27.4326	356.80	672.96	25.7838
12.0	24.6173	365.07	699.92	25.9749
16.0	21.8141	375.81	734.94	26.1383
20.0	19.0219	388.38	775.91	26.3051
24.0	16.2443	401.81	819.73	26.4771
28.0	13.4859	415.26	863.59	26.6477
32.0	10.7498	427.87	904.68	26.8084
36.0	8.037	438.66	939.87	26.9510
40.0	5.3428	446.98	967.13	27.0739
44.0	2.6629	452.26	984.68	27.1573
48.0	0.0	454.10	990.83	27.1862

LCCEWG BENCHMARK 3

RESULTS FOR CONFIGURATION I - UNIFORM
ASSEMBLY, NO WIRE WRAP

CODE USED: COBRA-IV-I

CHANNEL 48

DISTANCE (IN.)	DELTA-P (PSI)	ENTHALPY (BTU/LB)	TEMPERATURE (°F)	VELOCITY (FT/SEC)
0.0	33.1352	349.76	650.0	24.9817
4.0	30.2616	351.67	656.21	21.3622
8.0	27.4298	357.94	676.66	19.6333
12.0	24.6161	368.36	710.66	18.9921
16.0	21.8136	382.01	755.16	18.8495
20.0	19.0217	397.80	806.64	18.9132
24.0	16.2442	414.33	860.55	19.0538
28.0	13.4858	430.47	913.17	19.2152
32.0	10.7499	445.15	961.05	19.3689
36.0	8.0371	457.22	1001.18	19.4675
40.0	5.3426	465.94	1030.20	19.5296
44.0	2.6629	470.68	1045.99	19.5568
48.0	0.0	471.04	1047.19	19.5344

References:

1. N.C. Horning, M.B. Parker and R.P. Omberg "Proposed Third Benchmark Problem for the LCCEWG," (Second Revision) HEDL 8051932, May 1980.
2. Rust, J.H., Nuclear Power Plant Engineering, Haralson Publishing Company, Georgia, 1979.
3. Romander, C.M., Schwer, L.E. and Cagliostro, D.J., Response of water-filled thin-walled pipes to pressure pulses: Experiments and analysis, ASME Paper 78-PVP-53 (1978).
4. A. Moneim, M.T. and Chang, Y.W., Comparison of ICEPEL code predictions with straight flexible pipe experiments, Nuclear Engn. and Design 49, 187-196 (1978).

4. Administrative Matters

The monthly financial statement for January is attached:
it also includes FY totals to date.

As of December 31, 1980 the project staff was as follows:

Staff

M.J. Driscoll, Principal Investigator

C. Whelan, Senior Staff Assistant

Students

M. Plys, Research Assistant, SM thesis student

S. Ahmad, Half-time Research Assistant, SM thesis student

AS OF JANUARY 31, 1981

MONTHLY STATEMENT

PAGE 1

SUPERVISOR M J DRISCOLL

EFFECTIVE 10-01-80

ACCOUNT 89957

NW13-200

EXPIRING 02-28-81

DEPARTMENT OF ENERGY

DEPARTMENT 068000

AUTHORIZED TOTAL

3 REFERENCE -PNDGPENDING
FAST REACTOR BLANKET INVESTIGATIONS

RESTRICTIONS	CURRENT MONTH	FISCAL YEAR TO DATE	CUMULATIVE
EXPENDITURES	Jan '81	As of 1/31/81	
SALARIES AND WAGES			
FAC SAL-TEN	766.67	3,066.67	3,066.67
RESEARCH STAFF-ON.	122.50	122.50	122.50
HOURLY PERSONNEL-ON		42.02	42.02
SECRETARIAL & CLERICAL-ON	110.98	340.60	340.60
TOTAL SUBJECT TO E. B. .00	1,000.15	3,571.79	3,571.79
RESEARCH ASSISTANT-ON	1,498.75	6,294.75	6,294.75
TOTAL SALARIES AND WAGES .00	2,498.90	9,866.54	9,866.54
EMPLOYEE BENEFITS ON OFF			
BILLING RATES 27.00% 30.60%	270.04	964.39	964.39
OPERATING EXPENSES			
TOTAL OPERATING EXPENSES .00	69.17	891.57	891.57
COMPUTATION EXPENSES			
E.D.P. EXPENSE	158.28	988.68	988.68
I.P.C. CHARGES	1,061.94	2,910.64	2,910.64
TOTAL COMPUTATION EXPENSES .00	1,220.22	3,899.32	3,899.32
INDIRECT EXPENSES ON OFF			
BILLING RATES 70.50% 13.50%	1,423.79	5,416.91	5,416.91
TOTAL EXPENDITURES	5,482.12	21,038.73	21,038.73

MASTER

Distribution

MIT LMFBK Blanket Research Project

Quarterly Progress Report

Department of Energy Research and Development
Contract DE-AC02-76ET37241.A004
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