CONF-9404227--1

Paper for the 14th TRIGA Users Conference

Results of the MCNP Analysis of 20/20 LEU Fuel for the Oregon State University TRIGA Reactor

by

B. Dodd, A. C. Klein, B. R. Lewis, P. A. Merritt

Abstract

The Monte Carlo Neutron/Photon (MCNP) code has been used to perform the neutronics analysis required to support revision of the Oregon State University TRIGA Reactor (OSTR) Safety Analysis Report (SAR). The SAR revision is a necessary part of the preparation of the application for authorization to convert the OSTR core from High Enriched Uranium (HEU) FLIP fuel to a Low Enriched Uranium (LEU) fuel. Before MCNP was applied to LEU-fueled cores, it was first validated by comparing MCNP calculations on FLIP cores to historical, measured values for these cores. The LEU fuel considered was the 20 wt%, 20 % enriched (20/20) TRIGA fuel approved by the Nuclear Regulatory Commission (NRC) in NUREG 1282.

The results show that the 20/20 fuel is much more reactive than FLIP fuel. A just-critical OSTR FLIP core contains 65 elements, while a just-critical 20/20 core only needs 51 elements. Similarly, the current operational FLIP core consists of 88 elements, whereas a 20/20 core giving the same core excess only requires 65 elements. This presents a significant problem for the OSTR because of potentially significant neutron flux loss in experimental facilities. Further analysis shows that to achieve a full size operational core of about 90 LEU elements the erbium content of the LEU fuel would need to be increased from 0.47 wt% to about 0.85 wt%.

FG02-92ER75747

Results of the MCNP Analysis of 20/20 LEU Fuel for the Oregon State University TRIGA Reactor

by

B. Dodd, A. C. Klein, B. R. Lewis, P. A. Merritt

Introduction

The OSTR core has a circular grid pattern as shown in Figure 1. The grid locations are referenced by letter and number with the letter designating the ring and the number giving the position in the ring. The number of positions in each ring is as given below.

A - 1	E - 24
B - 6	F - 30
C - 12	G - 36
D - 18	

Grid locations may be filled with fuel elements, control rods, graphite reflector elements, aluminum plugs, experimental apparatus such as sample-holding dummy elements, pneumatic transfer termini, or simply left vacant in which case water will be in that position.

Surrounding the core grid is an aluminum-encased graphite reflector. This reflector has a number of penetrations and cut-outs, including those for some of the beam ports and the rotating rack. The whole core structure is submerged in a tank of light water.

Calculational Methods and Verification

Monte Carlo Codes

Most of the neutronics analysis work was performed using the Monte Carlo program MCNP version 4.2. Monte Carlo methods are very different from deterministic transport methods. Deterministic methods solve the transport equation for the average particle behavior. By contrast, a Monte Carlo code does not solve an explicit equation, but rather obtains answers by simulating individual particles and recording some aspects (tallies) of their average behavior. The average behavior of particles in the physical system is then inferred from the average behavior of the simulated particles. Not only are Monte Carlo and deterministic methods very different ways of solving a problem, even what constitutes a solution is different. Deterministic methods typically give fairly complete information, whereas Monte Carlo supplies information only about specific tallies requested by the user.

The MCNP Code

MCNP is a well known, general-purpose, continuous-energy, generalized-geometry, time-dependent, coupled neutron/photon Monte Carlo transport code. It solves neutral particle transport problems and may be used in any of three modes: neutron transport only, photon transport only, or combined neutron/photon transport, where the photons are produced by neutron interactions. The neutron energy regime is from 10^{-11} MeV to 20 MeV, and the photon energy regime is from 1 keV to 100 MeV. The capability to calculate k_{eff} eigenvalues for fissile systems is also a standard feature.

Pointwise cross-sections data are used in MCNP (i.e. there is not the classic reduction of data into a small number of energy groups). For neutrons, all reactions given in a particular cross-section evaluation (such as ENDF/B-V) are considered. Thermal neutrons are described by both the free gas and $S(\alpha,\beta)$ models.

The user creates an input file that is subsequently read by MCNP. This file contains information about the problem in areas such as: the geometry specification, description of the materials, which cross-section evaluations to use, the location and characteristics of the neutron or photon source, the type of answers or tallies desired, and any variance reduction techniques used to make the problem rum more efficiently.

The OSTR MCNP Model

The nature of MCNP enables the core geometry to be described in as great a level of three dimensional detail as desired or as necessary to achieve the purposes of the specific calculation. The near complete 3-D core model developed for the OSTR is shown in Figures 2 and 3. Each grid position is modeled in its exact location in the grid plate. All components which fit into the core grid plate have the exact dimensions of their constituent materials. The geometry specification for each fuel element includes the central zirconium rod, the fuel meat, and the cladding as well as the top and bottom graphite reflectors and end pieces. The data used are shown in Table 1. Control rods, fuelled-followers and graphite elements can also be placed in their exact locations. The reflector region around the core is treated as concentric cylinders of aluminum, graphite, and lead.

Once the model was developed then it was a relatively simple process to assign different materials to each grid location. In this manner, it was possible to determine the number of elements required for criticality, and the variation of excess reactivity with various cores.

Table 1 Dimensional and Density Data		
Property	FLIP	20/20
Zr rod radius (cm)	0.3175	0.3175
Fuel/graph. radius (cm)	1.82245	1.82245
Fuel/Zr length (cm)	38.10	38.10
Cladding radius (cm)	1.87325	1.87325
Top graph length (cm)	8.738	8.738
Bottom graph length (cm)	8.814	8.814
FFCR Fuel radius (cm)	1.665	1.665
FFCR Cladding radius (cm)	1.7234	1.7234
$g^{235}U$/element	134.279	99.0
gU/element	191.880	502.538
Enrichment (%)	70.0	19.7
Fuel density (g/cm^3)	5.999	6.656
SS density (g/cm^3)	7.86	7.86
Graphite density (g/cm^3)	1.60	1.60
Zirconium density (g/cm^3)	6.4	6.4
Aluminum density (g/cm^3)	2.7	2.7

Atom densities for each of the constituent materials were determined from TRIGA data sets provided by the manufacturer General Atomics (GA) or from standard references for common materials. Tables 2 and Table 3 show the atom densities used in the analysis.

Table 2 Atom Densities (x 10^{24} atoms/cm^3)		
Property	FLIP	20/20
Control Rod Atom Densities		
Boron	0.1075	0.1075
Carbon	0.02687	0.02687
304 Stainless Steel Cladding Atom Densities		
Carbon (0.08 wt%)	0.00031519	0.00031519
Chromium (19 wt%)	0.017290	0.017290
Nickel (10 wt%)	0.0080622	0.0080622
Iron (70.92 wt%)	0.060088	0.060088
Total Stainless	0.085755	0.085755
Other Atom Densities		
Graphite	0.080193	0.080193
Zirconium	0.042234	0.042234
Aluminum	0.06027	0.06027
Water Atom Densities		
Hydrogen	0.0668	0.0668
Oxygen	0.0334	0.0334

Table 3 Fuel Material Atom Densities (x 10^{24} atoms/cm^3)							
Fuel Atom Densities							
Erbium (wt%)	^{235}U	^{238}U	^{167}Er	^{160}Er	Zr	H	Total
FLIP Fuel							
1.44 (-10%)	0.000892797	0.000378151	0.0000713912	0.000239682	0.0351300	0.0562079	0.0929199
1.552 (-3%)	0.000892797	0.000378151	0.0000769439	0.000258324	0.0350864	0.0561382	0.0928308
1.6 (Nominal)	0.000892797	0.000378151	0.0000793236	0.000266313	0.0350677	0.0561083	0.0927926
1.68 (+5%)	0.000892797	0.000378151	0.0000832897	0.000279629	0.0350366	0.0560585	0.0927289
20/20 Fuel							
0.423 (-10%)	0.000658232	0.00264923	0.0000232679	0.0000781173	0.0345417	0.0552667	0.0932173
0.4559 (-3%)	0.000658232	0.00264923	0.0000250776	0.0000841931	0.0345275	0.0552440	0.0931883
0.47 (Nominal)	0.000658232	0.00264923	0.0000258532	0.0000867970	0.0345214	0.0552343	0.0931758
0.4935 (+5%)	0.000658232	0.00264923	0.0000271459	0.0000911368	0.0345113	0.0552180	0.0931551

Erbium Variation

A special cross-section set was generated which consisted of the ENDF/B-V data modified to include the erbium poison that exists in TRIGA fuel elements. Erbium has six stable isotopes, two of which are present only in small abundances. Because of the difficulty of obtaining cross section data sets for all of these isotopes, in the current calculations the erbium was considered to be 22.95% ^{167}Er with the rest being ^{166}Er. It should be noted that core excess and criticality calculations are extremely sensitive to the erbium atom density in the fuel. Therefore, it is important to know the erbium atom density as accurately as possible, or at least know how much it can vary. Discussions with the fuel manufacturer revealed that the erbium tolerances at each level of manufacturing and supply were as follows:

Individual Fuel Meat	+10%	-15%
Fuel Element (3 meats for OSTR)	+5%	-10%
Overall Core	+0%	-3%

This explains why Table 3 has the different erbium atom densities listed. Some core calculations were performed using all four of the atom densities given; i.e. with -10% of the nominal erbium content, with -3% of nominal, with nominal (+0%) and with +5% of the nominal erbium content. The significance of the uncertainty of the erbium concentration and the impact of the erbium loading for the HEU-LEU conversion are discussed in detail later.

Core Number Identification

Cores considered in this analysis have been identified by the use of up to three numbers. The first number is the nominal ^{235}U enrichment in %, the second is the nominal core excess in dollars, and the third is the number of graphite elements surrounding the core. As an example core 70-0-21 is the just-critical FLIP core with 21 graphite elements in the outside ring.

Verification

The validity of the use of MCNP for the TRIGA reactivity analyses was confirmed by performing calculations on the FLIP core and comparing the results to those measured during the initial commissioning tests of this core. These calculations were performed by two independent groups (identified as Lewis and Merritt) using very similar geometry and the data set given in Tables 1-3. This provided some further validation of the method. A compilation of these data can be seen in Table 4. This table shows the initial just-critical core ($0.12), one core loaded above critical ($3.35) and the first operational core ($7.17). It can be seen that there is good agreement between calculated and measured values of the effective multiplication factor.

The error range indicated for the calculated values is based on the statistics associated with Monte Carlo methods and is part of the results, while the error range given for the measured values is an estimation of the accuracy of the core excess measurements based on experience with the repeatability of control rod worth measurements.

Table 4 MCNP-Calculated and Measured Values of k_{eff} for FLIP Cores			
Core No.	No. Elements (including fuel-follower control rods)	Calculated	Measured
		k_{eff}	k_{eff}
70-0-21	65	0.992 ± 0.0026[1] 1.0004 ± 0.0014[2]	1.0008 ± 0.0007
70-3-21	76	1.0197 ± 0.0025[1] 1.0317 ± 0.0013[2]	1.0235 ± 0.0007
70-7-21	85	1.0454 ± 0.0025[1] 1.0567 ± 0.0014[2]	1.0502 ± 0.0007

[1]Lewis and [2]Merrit represent two independent calculations using the same data set, but slightly different levels of detail in geometry.

An indication of the magnitude of the variation in k_{eff} due to the potential manufacturing tolerances can be seen in Table 5 . It is clear that this is a significant factor in the uncertainty of k_{eff}.

Table 5 Effect of Erbium Variation on k_{eff} for Core 70-7	
Erbium (% Difference from the Nominal 1.6 wt%)	k_{eff}
-10	1.0741 ± 0.0015
-3	1.0583 ± 0.0013
0	1.0567 ± 0.0014
5	1.0450 ± 0.0014

It is felt that because MCNP can calculate the information for the FLIP core within the level of accuracy verifiable by actual measurements, then it can do so for the LEU cores. Merely changing the atom densities of some of the isotopes in the fuel elements does not affect the validity of the calculations.

Critical Mass

Several runs were made varying the number of fuel elements in the core until the minimum number required for criticality was determined. The results of these runs are given in Table 6. Here it can be seen that it takes a minimum of 51 of the 20 wt% fuel elements to go critical. This calculation is somewhat validated by the fact that the Bangladesh TRIGA reactor went critical with 50 fuel elements of 20/20 fuel. It should be noted that the Bangladesh TRIGA has a hexagonal grid core; however, the lattice pitch is fairly similar to that of the OSTR and, therefore, the results are comparable. Plans of the just-critical 20/20 core can be seen in Figure 4. For comparison, the just-critical core for the original FLIP fuel was 65 elements.

Operational Cores

Past experience has shown that a cold, clean core excess of about $7 is needed to allow for all aspects of the OSTR operation. Approximately $3 is needed for xenon, temperature and power effects, while the additional reactivity is needed to account for the cadmium-lined in-core irradiation tube, samples in the rotating rack and experimental apparatus. For these reasons, and in order to directly compare the 20/20 fuel with FLIP fuel, two above-critical 20/20 cores were evaluated: one with about $3 core excess and the other with about $7 core excess.

Core Sizes

Sequential MCNP runs, made by adding fuel elements to the reactor core, provided the results shown in Table 6 and Figures 5 and 6. It can be seen that a 20/20 core with about $3 core excess would have 57 elements, while the initial operational core with about $7 core excess would consist of only 65 fuel elements. This is approximately 20 elements less that the equivalent FLIP core. Such a situation presents a significant problem for the OSTR in that this results in a core which has almost two whole vacant rings between the core and the rotating-rack (Fig. 6). Therefore the neutron flux in the rotating rack would be significantly lower with the 20/20 fuel than it is with the FLIP fuel. Since the rotating rack is the major irradiation facility for the OSTR, such a situation is unacceptable.

Table 6 MCNP-Calculated Values of k_{eff} for 20/20 Cores			
Core No.	No. Elements (including fuel-follower control rods)	Lewis	Merritt
		k_{eff}	k_{eff}
20-0	51	1.0067 ± 0.003	1.0018 ± 0.0015
20-3	57	1.0269 ± 0.003	1.0227 ± 0.0014
20-7	65	1.0499 ± 0.0013	1.0496 ± 0.0014

20/20 Fuel with Increased Erbium Content

Because the currently approved 20/20 TRIGA fuel is an unacceptable replacement for the FLIP fuel in the OSTR, a number of scoping MCNP runs were made to determine a more suitable fuel composition. In particular, the erbium weight percent was varied to determine its value for a 20/20 core of about 90 elements with a core excess of about $7. A 90 element core (including the fuelled-followers) would fill the F ring, leaving the G ring for graphite elements. The results indicate that fuel which has 0.85 wt% erbium would provide the necessary core size. Clearly, further calculations would need to be made to determine if the core lifetime reactivity and other effects would be acceptable for a fuel with this composition. In addition, such a fuel would have to be approved by the USNRC prior to manufacture and use.

Conclusions

The results of this analysis show that the currently approved 20/20 fuel is much more reactive than FLIP fuel. A just-critical OSTR FLIP core contains 65 elements, while a just-critical 20/20 core only needs 51 elements. Similarly, the current operational FLIP core consists of 88 elements, whereas a 20/20 core giving the same core excess only requires 65 elements. This presents a significant problem for the OSTR because of potentially significant neutron flux loss in experimental facilities. Further analysis shows that to achieve a full size operational core of about 90 LEU elements the erbium content of the LEU fuel would need to be increased from 0.47 wt% to about 0.85 wt%. It would appear that fuel with a different erbium concentration than that given in NUREG 1282 would have to be approved by the USNRC.

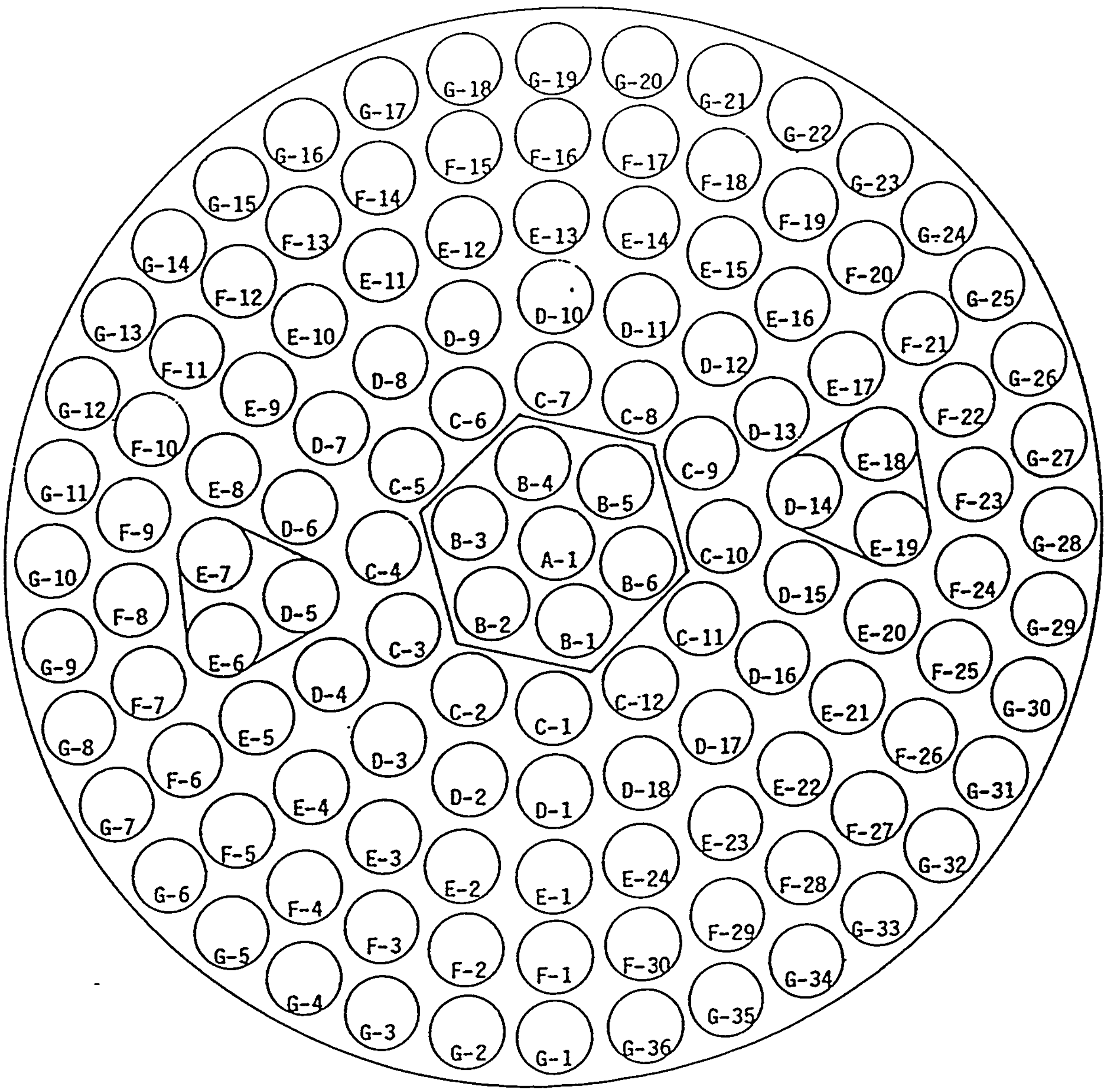

Figure 1 - OSTR Core Grid

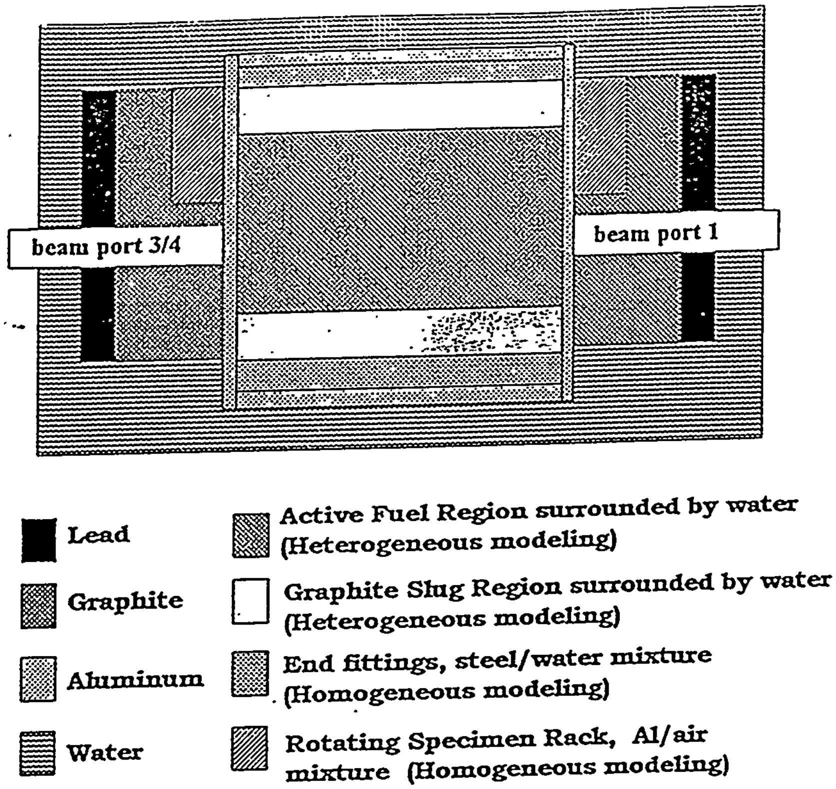

Figure 2 - MCNP Model of the OSTR Core

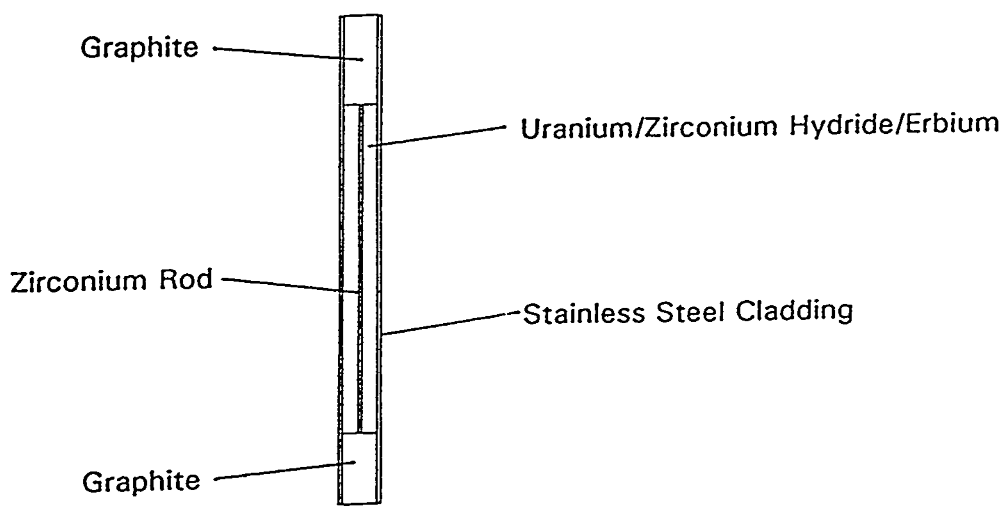

Figure 3 - MCNP Model of a Fuel Element

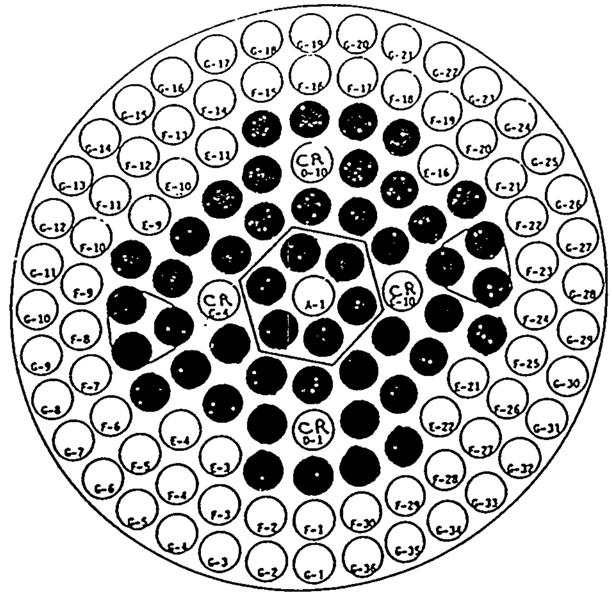

Figure 4 - Just Critical 20/20 Core

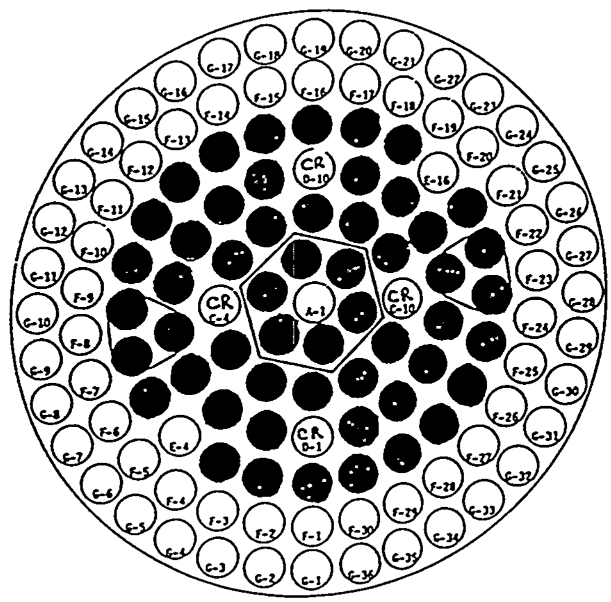

Figure 5 - $3 Excess 20/20 Core

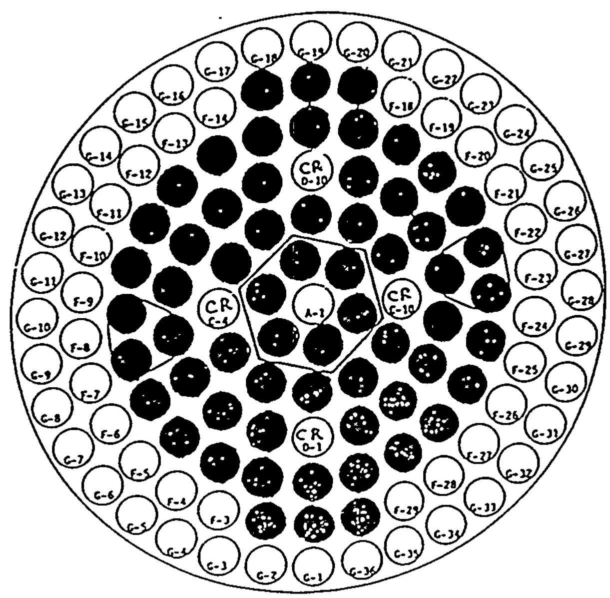

Figure 6 - $7 Excess 20/20 Core