IV. NEUTRONICS OF LASER FISSION-FUSION SYSTEMS.

IV.1. OBJECTIVES.

Before planning experimental work in the field of Laser driven fission-fusion microsystems several theoretical tasks have been initiated, in order to acquire a set of codes to allow simulation of physical processes involved.

Many different subjects have to be considered because of the complexity of the phenomena, which are strongly connected among them. So, hypotheses and simplifications made upon the theoretical basis to obtain a calculation method must be reviewed even with numerical stimation of errors, if possible.

From a neutronic point of view two main subfields can be defined:

i) The imploding process of the micropellet, including the external neutron shoot to induce the fission burst, the burst itself, the bootstrapping mechanism of fusion neutrons, Tritium reproduction, and so on, that is, the microexplosion.

ii) The neutron (and gamma) transport from the point in which the burst has happened to the coolant, structure, and blanket. This second part is directly related to the plant economy (1), due to the interest in getting an adequate temperature distribution in coolant, and in getting high rates of reproduction to fissile and fusionable materials.

It's obvious that the analytical study of the first subject is more difficult than that of the second one. It involves to deal with:

- Laser absorption and ablation process.

- Fission ignition.

- Thermodynamic evolution of the micropellet with internal generation of energy.

- Fusion burst and "in situ" Tritium reproduction. Boots-trapping of fusion neutrons.

- Follow-on study of the neutronic and thermomechanic of the pellet till the total disassembly.

The final computational procedure will be made-up from different modules resulting from subjects which can be studied "almost" separately. This modular structure will enable to improve the different topics being developed, in a stagewise manner.

IV.2. NEUTRONIC ANALYSIS.

The two neutronic aspects quoted above have to be studied with time - dependent transport theory. Nevertheless, neutron transport through the coolant and external blankets can be performed with other less sophisticated means.

In this sense, two different calculational approaches have been attempted:

i) The first one is the use of discrete ordinates steady state codes, mainly DTF-IV (2).

The system is simulated in one-dimensional spherical geometry by describing approximately the zones: inner cavity, inner wall, coolant (lithium), main wall and so on.

The problem is solved with an isotropic source at the center of the sphere, and the flux is determined in several groups.

Adiabatic or/quasi-static calculations are done, with short time steps at the beginning to allow the representation of the microburst. As the interval between consecutives pulses is rather long compared with the length of the microexplosion, not too much steps are necessary to reproduce the transient. Once one of them has been carried out it is possible to calculate the interesting reaction rates, determining some magnitudes which have a great influence upon the economics of the plant.

ii) The another method involves the use of explicit time - dependent S_N codes (3). TDA (4) has been used to a little extent, but meaningful results haven't been obtained so far. It is still under implementation and improvement.

It is expected to have available the TIMEX (5) code in a short time, and an own version of TDA or/and TIMEX will be developed to deal with specific cases.

Results on this area haven't been published yet because further analysis is needed. Anyway, it seems that using steady state codes is adequate for many problems.

On the other hand, performing of the whole microexplosion calculation needs a correct treatment in the subjects listed above. Most of them are not directly related with neutron transport, but they affect upon isotopic concentrations and temperature distribution.

Assuming the phenomena to be mechanically isotropic it will be possible to calculate it in one-dimensional geometry. So, the neutronic module might be a code such as TDA or TIMEX or any other similar. At first, TDA and TIMEX differ between them on the way in which the time dependence is put into difference equations (and, of course, in many other aspects). The non-iterative character of TIMEX could be an error source, mainly for long time steps. However, computing economy with it could be very interesting

if a very short step has to be used due to other requirements, such as rapid burnup or thermomechanic evolution.

In any case, the neutronic module can't be set up merely by using one of those codes. There are other points to be analysed which are as important as the transport equation solution itself. These problems also appear in the external propagation of the neutron burst, but they have a greater incidence in calculating the microexplosion.

i) Time mesh, and interaction with the spatial mesh. This wellknown problem comes out from differencing the continuous dependence of the flux upon space and time, and it can be difficult to treat in wave-front situations. This will be probably the case at early times of the microexplosion (depending on the neutronic shoot and the design of the micropellet). A pseudo-analytical procedure could be suitable in some instances and it had to be compared with the solutions offered by codes.

   Using different time meshes for each group can also solve - or mitigate - the problem, but it generates a new one for calculating the "outer source" of a group at a given moment. It would be possible to interpolate in fluxes of upper groups, but it would be a hard question to extrapolate in the lower ones, both for consistency and reliability.

ii) Flux corrections due to moving boundaries.

   The thermodynamics evolution of the system will induce a non-steady geometry which will affect to the neutron transport. This point can be taken into account in two ways:

   - Moving the spatial mesh according to the mechanical evolution of the matter, i.e., describing it in lagrangian coordinates.

- Keeping the mesh fixed in the laboratory system and correcting the isotopic concentrations according to the current of matter at the inner and outer boundary of each interval.

In this last case the flux has not to be corrected in every time step. However, fission and scattering sources for every group have, because of the change in macroscopic cross sections.

In the first one, fluxes have to be expressed in the new system, what induces an extra effort to "conservate neutrons". This is really easy to do, at first, but it's not possible to maintain the shape because the flux is not known within the interval. Moreover, macroscopic cross sections have to be corrected, too, because of burnup and temperature.

Just a little work has been done in this field, but it seems the problem will be solved according to the specifications of other - thermomechanical - modules.

iii) Generation of effective cross sections.

Reliability of neutronic solutions will depend upon the group constant as in any other problem, but in this case special considerations can be posed:

- Weighting flux within group. Due to the hard spectrum being in the fissile zone, fission spectrum could be suitable, and better than that of collision density constant, even for intermediate energies. However, reflected neutrons from the fusionable zone are strongly moderated and the latter will be more adequate.

- Isotopic concentrations rapidly varying within a spatial interval, due to thermomechanical evolution.

- Background cross sections varying with time, specially

during the burst. This point is related to the burnup study and simulation of fission products.

A code is being developed to generate cross section using a revised shielding factor formalism. It's still under implementation, but it has been used for the calculations quoted below.

It reads nuclear data from "ff" codes (ETOX (6), MINX (7)) and performs 1D transport calculations in many groups. Ligh isotopes are treated in a different way than heavy ones, giving special attention to their moderating properties.

It's expected this code will have to be a module of the global program if the effective cross sections have to be re-calculated as fast as it's guessed now.

iv) Burnup calculations.

To evaluate the total yield of energy from fission several problems arise from the production and distribution of fission products. However, a first stimation seems to lead to only one pseudo-product.

v) Bootstrapping mechanism and neutron interaction between fissile and fusionable zones. It includes the study of Tritium reproduction, essential for calculating fusion rates.

vi) Performances of different designs.

Influence of the design (8) upon the neutronic behaviour of micropellets is a major aspect to be analysed. Bi-layere and multi-layered hollow and solid spheres, with the fissi material inside or outside might be performed. Of course, micropellet configuration also affects the other thermome-chanical fenomena.

There are many other problems which are involved in neutronic calculation. For instance, upscattering suffered by neutrons below a few kevs due to the high temperature of the matter.

IV.3. NEUTRONIC CALCULATIONS.

Some calculations have already been made, but many of them are not meaningful because the codes used are intermediate versions under development.

Regarding to performances of coolant and blanket any special effort hasn't been done, because they are really similar to fast reactors and critical assemblies calculations, except for the explicit dependence on time, which is not as strong and important as in the microexplosion.

Related to the latter some cases have been carried out, but the attention has been focused to the analysis of the final state at the end of the implosion. Because the global code is not yet working, that state was defined merely by assuming hypothetical but logical values for the major thermodynamic magnitudes. By changing those ones, it also was possible to make sensitivity studies which gave some information upon the influence of those magnitudes on neutronic parameters. These calculations will be reported in the next future, after completing their interpretation.

To compare our results with other published data a case similar to those referenced above has been chosen (9). It's a Plutonium (95 % ^{239}Pu, 5 % ^{240}Pu) micropellet surrounded by ^{6}LiD, which is compressed to a supercritical state by means of a laser pulse. This state is defined by:

Inner (Pu) radius	0.216 mm
Outer radius	0.256 mm
Pu density	4667.14 $g.cc^{-1}$
Tamper density	3416.17 $g.cc^{-1}$

Results quoted in the bibliography (10) are

$$Keff : 1.25$$
$$\Lambda : 0.0204 \text{ ns}$$

So, using the equation

$$\alpha = \frac{Keff - 1}{Keff \cdot \Lambda} \qquad |1|$$

Λ beign the prompt-neutron mean generation time, it's obtained:

$$\alpha = 9.8 \text{ ns}^{-1}$$

This calculation was done (11) using transport static codes to evaluate Keff and

$$\Lambda = \frac{\langle \frac{1}{v} \rangle}{\langle \nu \Sigma f \rangle} \qquad |2|$$

Reproducing the calculation for different times it's possible to get Keff(t) and Λ(t) which yield the α(t) value to be used in point kinetics theory.

Instead of repeating the sequence our atention was mainly focused onto the "final state". We used different static codes, and in different ways, to evaluate the three parameters given above. Sometimes the values were obtained directly from the code, and sometimes were from the application of some formula. Besides it, a simple two-groups time dependent calculation was done with the TSN (12) code.

Results are listed in Table I, which also presents the equations used in each calculation.

Before commenting anything upon them it's necessary to explain in a greater detail what was done in each one.

Cross-sections were generated from ENDF/B-III through ETOX-1DX (13) (except for TIMOC (14), that CODAC was used). A 28-group library of $\Delta u=0.5$ was specified, which was collapsed into two groups for TSN.

DTF calculations were of three types, evaluating k, α and $k(\gamma)$ respectively. $k(\gamma)$ means the k obtained with a poison seeding the micropellet with a macroscopic absorption cross section (γ/v_g), v_g being the group speed.

MonteCarlo static calculation was done with TIMOC, using a 5000 secondary neutrons sampling.

$\Lambda^* (\phi_i, \phi_j^+)$ means a bilinear weighting to determine Λ, using S_N (N=12) fluxes from DTF (direct, adjoint). The subscript obviously indicates the kind of calculation performed.

It's really fair the agreement between DTF and TSN in determining α. A 6 % is not too much if one remembers that the two ways differ strongly in the number of groups and the theory to obtain it.

On the other hand, discrepancy between DTF and TIMOC in k is rather small. Again, remember the so different theories involved and the fact that cross sections were precessed with different codes.

However, Λ values do not agree in a similar manner. It can be observed one group of calculations which yield a number very close to that of the bibliography, i.e., 0.0204 ns. In fact, DTF (γ) and TIMOC agree very well with it, and DTF(k) is slightly above.

On the other hand, another group can be formed with $\Lambda^*(\alpha,k^+)$ $\Lambda^*(\alpha,\alpha^+)$ and DTF(α,k), that is, those calculations which, in a certain sense, take into account the importance of neutrons.

Values from $\Lambda^*(k,k^+)$ and DTF(α) are in the middle of the two, what is not surprising by two reasons:

TABLE I

Calculation	Directly			Through Equations			Equations
	$\alpha(ns^{-1})$	k	Λ(ns)	$\alpha(ns^{-1})$	k	Λ(ns)	
DTF(k)		1.2595		8.627		.0239	$\Lambda=\langle\frac{1}{v}\rangle/\langle\nu\Sigma_f\rangle$; $\alpha=(k-1)/k.\Lambda$
DTF(α)	16.961					.0175	$\Lambda = \langle 1/v\rangle/\langle\nu\Sigma_f\rangle$
DFT(γ)		1.2595		10.051		.0205	$\Lambda= \lim_{\gamma\to 0}(k(0)-k(\gamma))/\gamma.k(\gamma)$
TIMOC		1.2569	.02114	9.668			
TSN	16.023						$\alpha = \frac{\langle \partial\phi/\partial t\rangle}{\langle\phi\rangle}$
$\Lambda^*(\phi_\alpha,\phi_k^\pm)$	16.961	1.2595				.01216	$\Lambda = \langle\phi^+\frac{1}{v}\phi\rangle / \langle\phi^+ \nu\Sigma_f \phi\rangle$
$\Lambda^*(\phi_\alpha,\phi_\alpha^+)$	16.961					.01300	"
$\Lambda^*(\phi_k,\phi_k^+)$		1.2595				.01628	"
DTF(α,k)	16.961	1.2595				.01214	$\Lambda = (k-1)/k.\alpha$

- The system is highly supercritical (k=1.25). So, the real flux will have a shape closer to the α-solution than to the k-solution, and $\Lambda^{*}(k,k^{+})$ will not be a too good bi-weighting, at least because of the real part.

- DTF(α) don't consider by any means the importance of neutrons, but just -probably- the flux shape in a better way than DTF(k), as it was said inmediately above.

The foregoing considerations are meaningful as far as those parameters represent integral values, which can be compared with experimental measurements. For instance, α is a tipical magnitude to evaluate because it's a measure of the energy yield.

Regarding to the results shown above some last comments can be added. First, the exact value of α is probably 16 ns^{-1}. TSN and DTF evaluation can be considered as good ones, and they agree. Second, all the methods yield the same k value: 1.25, and all those methods have been tested many times in k calculations. But, third, which is the value of the prompt-neutron mean generation time?

If the α and k exact values are used whit equation |1| -which is not a point kinetic formulae but it comes out from integrating over energy, space and direction- the value 0.012 is obtained.

This agrees with methods which take into account the neutron importance doing bilinear weighting, what can induce to think this is the exact value. Moreover, bilinear weighting cancels the errors in some extent, as it's demonstrated by variational procedures.

IV.4. DESCRIPTION OF CODES.

Codes used in the foregoing calculations are listed below, quoting their main features.

DTF-IV

Developed in Los Alamos, USA (1965) it was incorporated into the reactor calculation scheme of the JEN some years ago.

It's a steady state neutron transport code, with multigroup, finite difference equations and S_N techniques, allowing general anisotropy and inhomogeneous sources.

It solves α and K eigenvalue problems, criticality searches, and source calculations, both for the direct and the adjoint flux.

ETOX

From Batelle NWL, USA (1969).

It generates cross-sections in the selfshielding factor format, from the ENDF/B. Specially suitable for fast spectrum systems.

1DX

Also from Batelle NWL, USA (1969).

It reads nuclear data from ETOX and calculates nuclear libraries for specified isotopic mixtures and temperatures.

It uses the neutron flux obtained by solving a diffusion problem for a one-dimensional case to collapse cross-sections to produce few-groups libraries.

TIMOC

Obtained from Euratom (Italy), (1970).

It's a Monte Carlo code which solves the steady state transport equation, using a multigroup structure, with several optional sampling techniques. It provides integral parameters, as the effective multiplication constant and the mean production time.

TSN

Developped by Atomics International, USA (1967-68). It solves the time-dependent transport equation in 1-D by the discrete-ordinates method (based on ANISN), coupled with a thermodynamic calculation at constant pressure. Restrictions: two energy groups, no delayed neutrons. Oriented toward experimental cylindrical cores at fast transients.

IV.5.- REFERENCES

(1) L.A. BOOTH, "Central Station Power Generation by Laser - Driven Fusion". Nucl.Eng.Design. 24, 263-313 (1973).

(2) K.D.LATHROP, "DTF-IV, a Fortran-IV Program for Solving the Multigroup Transport Equation with Anisotropic Scattering" LA-3373 (1965).

(3) G.E.BOSLER, "Energy Deposition Rates in a Laser-Fusion Reactor". LA-UR-75-973 (1975).

(4) S.A.DUPREE et al, "Time-Dependent Neutron and Photon Transport Calculation Using the Method of Discrete Ordinates". LA-4557 (1971).

(5) T.R. HILL and Wm.H. REED, "TIMEX: a Time-Dependent Explicit Discrete Ordinates Program for the Solution of Multigroup Transport Equations with Delayed Neutrons". LA report to be published. Also: Wm.H. REED. LA-4800 (1972).

(6) R.B.KIDMAN et al, "ETOX, a Code to Calculate Group Constants for Nuclear Reactor Calculations". BNWL-1002 (1969).

(7) D.R.HARRIS et al, "MINX, a Modular Code System for Processing Multigroup Cross Sections from Nuclear Data in ENDF/B Format" LA-UR-1766 (1973).

(8) W.SEIFRITZ and H. NAEGELE, "Trans.Am.Nucl.Soc., 21, 18 (1975)

(9) W.SEIFRITZ and J.LIGOU, "Laser Induced Thermonuclear Micro-explosions using Fissionable Triggers". Trans.Am.Nucl.Soc., 18, 18 (1974).

(10) W. SEIFRITZ and J. LIGOU, "Laser-Induced Thermonuclear Micro-Explosions Using Fissionable Triggers". Nucl.Sci.Eng., 56, 301 (1975).

(11) W. SEIFRITZ and J. LIGOU, Private Communication . (1975).

(12) W.A. RHOADES and J.C. MORGAN, "The TSN Computer Program for Spatially Dependent Reactor Kinetics Calculations". NAA-SR-MEMO-12319 (1967).

(13) R.W.HARDIE and W.W.LITTLE, Jr., "1DX, a One-Dimensional Diffusion Code for Generating Effective Nuclear Cross Sections". BNWL-954 (1969).

(14) H. RIEF and H. KSCHWEND, "TIMOC, a General Purpose Monte Carlo Code for Stationary and Time Dependent Neutron Transport", EUR 4519 e (1970).