Technical University of Denmark

# Computer modelling of radioactive source terms at a tokamak reactor

**Forskningscenter Risø, Roskilde**

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# COMPUTER MODELLING OF RADIOACTIVE SOURCE TERMS AT A TOKAMAK REACTOR

Progress report by
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ABSTRACT. The Monte Carlo code MCNP has been used to create a simple three-dimensional mathematical model representing 1/12 of a tokamak fusion reactor for studies of the exposure rate level from neutrons as well as gamma rays from the activated materials, and for later estimates of the consequences to the environment, public, and operating personnel. The model is based on the recommendations from the NET/INTOR workshops.

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## INTRODUCTION

Early recognition of safety problems with a tokamak fusion reactor will provide the opportunity to reduce these problems by improving design and materials.

Using a computer model of a fusion reactor is one of the ways to evaluate these problems, and to calculate the consequences to the environment, public, and operating personnel. With a computer model it will be reasonably easy to recalculate the consequences when the design and/or the materials are modified.

The Monte Carlo code MCNP version 3, has been used to create a three-dimensional mathematical model for studies of the exposure rate level from neutrons as well as gamma rays from the activated materials, and for later estimates of radiation doses to the personnel that might have to enter the reactor hall after an accidental event.

A simple model representing 1/12 of a tokamak fusion reactor has been developed. The model is based on the recommendations from the INTOR workshops.

## CHOICE OF COMPUTER CODE

In order to make the necessary calculations of neutron and photon transport we got the IBM versions of two Monte Carlo radiation transport codes: MORSE-CG (developed at Oak Ridge National Laboratory), and MCNP version 3 (developed at Los Alamos National Laboratory), the former of which we received late in 1983, and the latter in February 1984. During the spring 1984 both codes were implemented on an IBM 3033 computer at the

Northern European Computer Center at the Technical University of Denmark in order to run some test problems to decide which of the codes was most suited to our application.

One of the differences between the two codes is the design of the geometrical input. The combinatorial geometry of MORSE-CG must be described by a combination of nine geometric bodies, i.e. rectangular parallelepiped, box, sphere, right circular cylinder, right elliptical cylinder, truncated right angle cone, ellipsoid, right angle wedge, and convex polyhedron, each of which may be arbitrarily oriented in space. They are combined into volumes by three operators denoting combination, exclusion, and overlap. The basic units of the MCNP geometry are cells. A cell is created from regions of space bounded by first- and second-degree surfaces and certain fourth-degree surfaces (tori). The cells can then be combined with the binary Boolean operations of intersection, union, and complement.

One of the reasons for the final choice of MCNP version 3 was that we are provided with more flexibility in defining a complex geometry from the available surfaces in the code instead of being limited to a small number of predefined geometrical bodies as in the combinatorial geometry of MORSE.

During the autumn 1984 MCNP version 3 was made operational at the computer at Risø National Laboratory, a Burroughs 7800.

The code treats an arbitrary three-dimensional configuration of materials in geometric cells which are defined as mentioned above. It can be used for neutron, photon or coupled neutron-photon transport. For neutrons, all reactions are accounted for. For photons, the code takes account of incoherent and coherent scattering, the possibility of fluorescent emission following photo-electric absorption in pair production with local emission of annihilation radiation.

A computer graphics system, RIGS, developed at Risø has been implemented in the code in order to make it easy to check the geometrical input on the plotting devices available to us.

The Radiation Shielding Information Center at Oak Ridge National Laboratory, Tennessee, who distribute the code announced in September 1984 that MCNP had to be updated to make changes correcting conditions which in rare instances would give wrong answers, and in more frequent instances the program would abort; to correct all of these possibilities numerous changes were made. In December 1984 we received the new version of the code which is now being made operational at our computer.

In addition to the thirteen test problems enclosed in the code files we have run the code on some simple experimental arrangements performed at Risø's laboratories, and we found good agreement with the experimental results (≤ 4 % deviation). The tests will be reported in detail later because the present report deals only with computer modelling of a simple tokamak fusion reactor section.

## DESCRIPTION OF THE MODEL

The design of the tokamak fusion reactor section as shown in figure 1-4 is based on the NET/INTOR specifications as given in the report from INTOR Phase I workshop (April 1982). The dimensions of the toroidal field coil is in accordance with the remote concepts given in the report from INTOR Phase IIa workshop (June 1984, Summary Document for Critical Issue Group D, table 5.2 and figure 5.4).

The section represents 1/12 of the complete torus. The sides of the section are supposed to reflect neutrons so that transport calculations using the model can be scaled to give the result for the complete reactor.

The section is placed in a coordinate system so that the Z-axis represents the center line of the torus, and the Y-axis is the horizontal radius of the section.

As described previously a complete three-dimensional geometry is defined by means of cells which are bounded by surfaces. The way to specify cells in MCNP is to list the cell number, material mixture number, and material density followed by a list of surfaces that bound the cell.

Each surface divides all space into two regions, one with positive sense with respect to the surface and the other with negative sense. A cell is then described as the intersection, union, and/or complement of the listed regions.

Figure 1 shows an elevation view of the section with the different parts marked as in an ordinary technical drawing. The figures 2-4 show the section seen in three planes perpendicular to each other. All the figures are drawn by means of the plotter routine in MCNP.

The necessary division of the different parts of the tokamak into several cells is seen in the figures 2-4. In total 53 cells and 73 surfaces are used to describe this first simple model. The many cells can be used in the calculations to divide the system into regions of different importance for variance reduction. This is called geometry splitting and is performed in the following way: When a particle of a certain weight passes from a cell of importance, x, to one of importance, x+1, the particle is split into a number of identical particles of lower weight. Especially in deep penetration shielding problems the number of particles diminishes to almost nothing, but geometry splitting improves the efficiency of the calculations. Several adjacent cells may contain the same mixture of materials but can have different importance.

## FINAL REMARKS

The first primitive three-dimensional model representing 1/12 of a tokamak fusion reactor is seen on the figures 1-4. The model will enable us to study the neutron activation of parts of the reactor.

During 1985 the model will be refined, and calculations will be done on the neutron activation products that are generated mainly in the structural materials by the high energy neutrons produced in the deuterium-tritium reaction. The majority of the activations will take place in the first wall, the blankets, and the shielding.

The divertor and the pumping systems will be added later, and streaming will be taken into account in the calculations.

The materials to be used in the calculations will be chosen from the NET reports, and we will keep in touch with the NET team for the latest trends and development in design and material specifications.

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MORSE-CG. General Purpose Monte Carlo Multigroup Neutron and Gamma-Ray Transport Code With Combinatorial Geometry. Oak Ridge National Laboratory, Tennessee, U.S.A. 1975. 412 p.

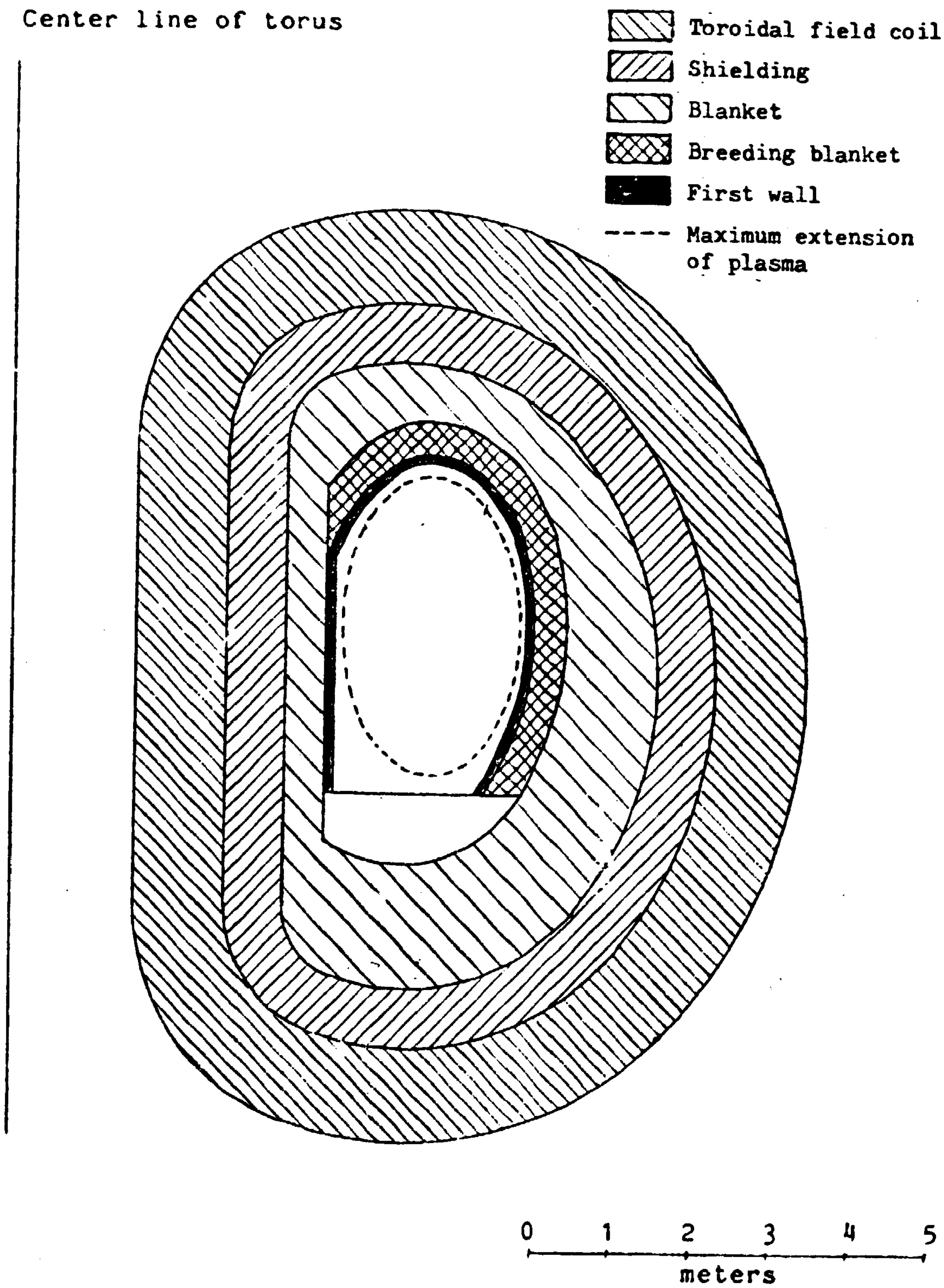


Figure 1. Elevation view of section.

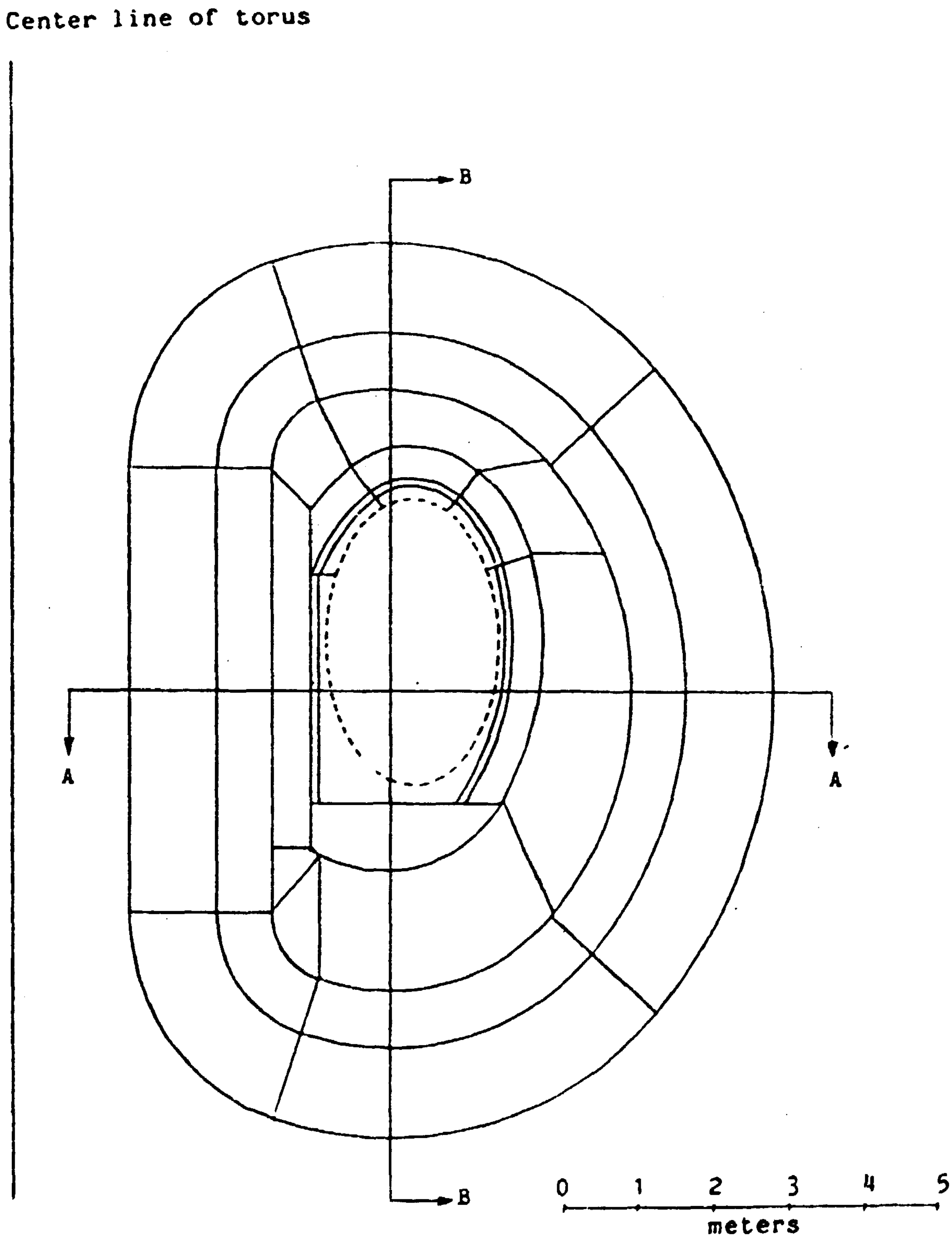


Figure 2. Elevation view of section. Cross section view A-A, see figure 3. Cross section B-B, see figure 4. All cell boundaries are shown even if they do no separate different materials.

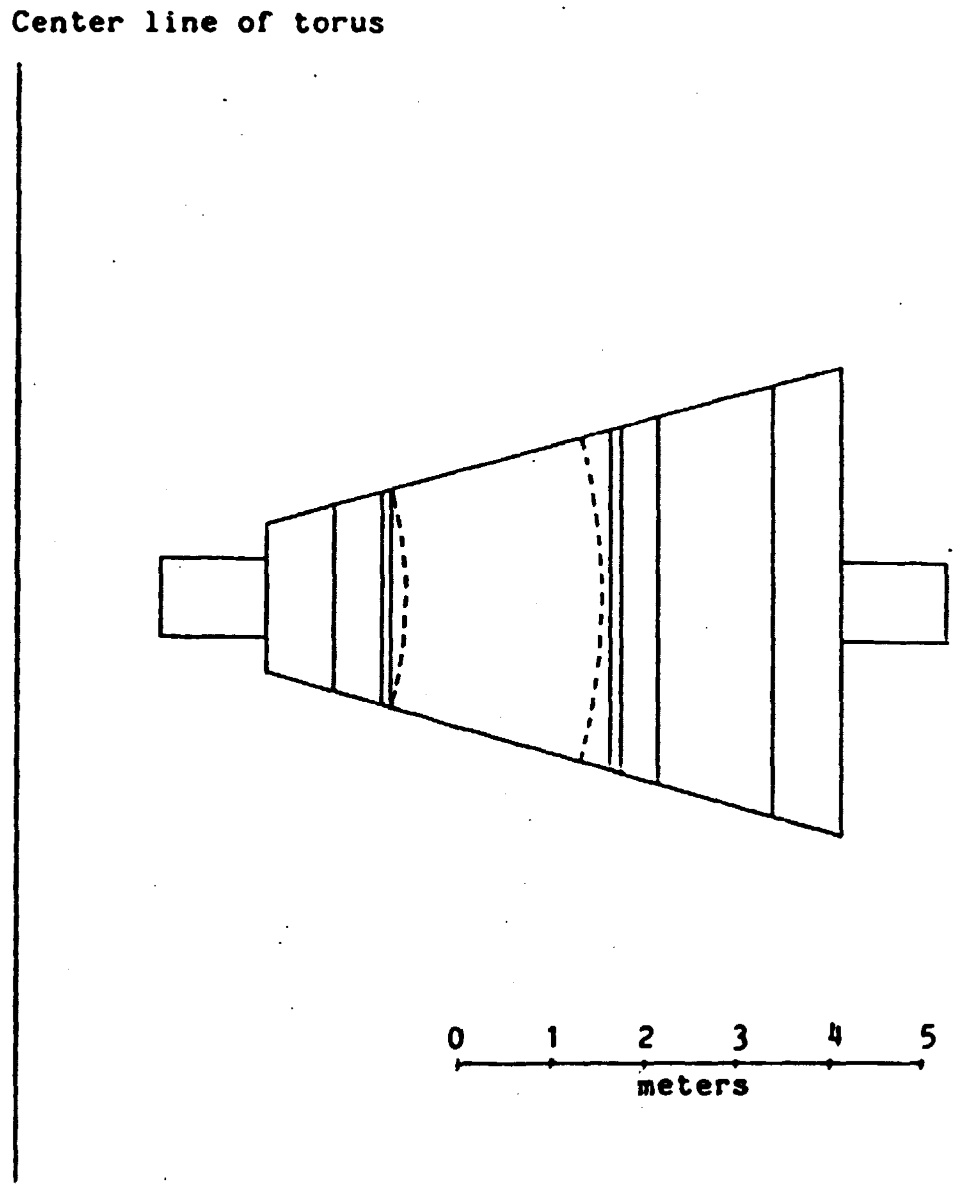


Figure 3. Cross section A-A of section.

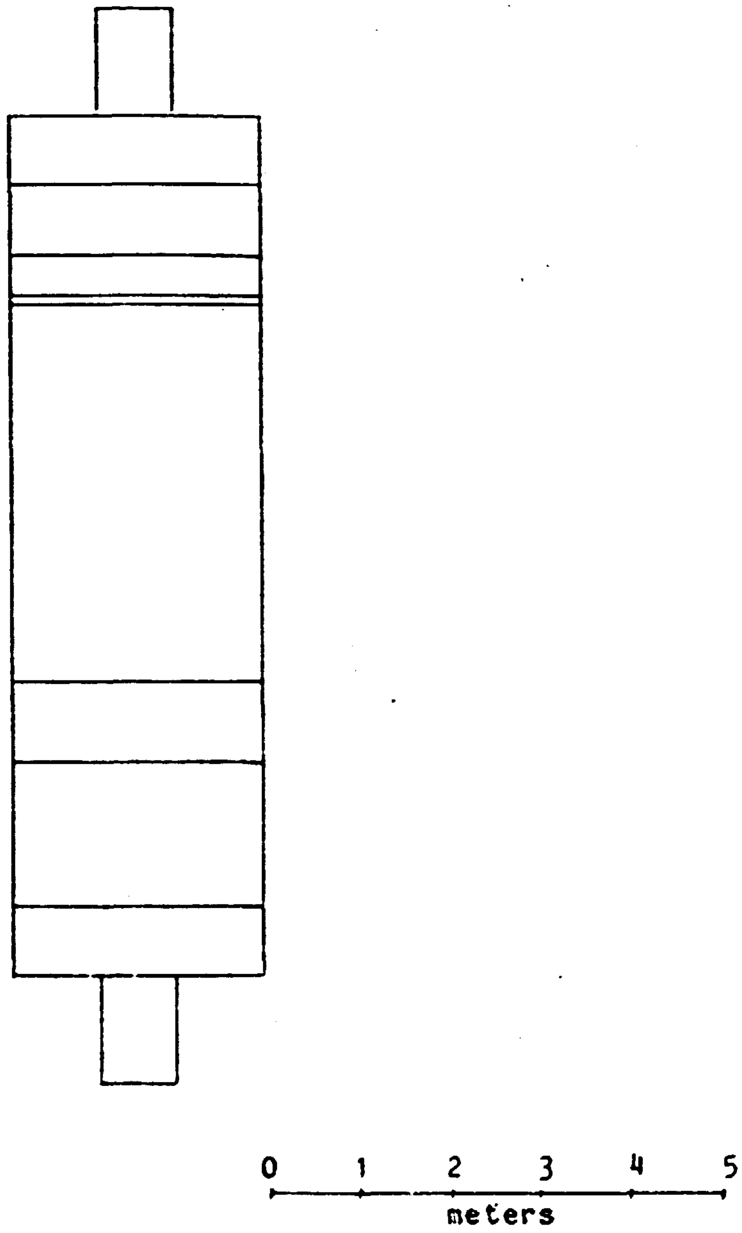


Figure 4. Cross section B-B of section.

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Group's own registration number(s)

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Abstract

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