FUELS FOR CANADIAN RESEARCH REACTORS

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INTRODUCTION

Originally, when I was requested to attend the meeting, it was to be as an observer. Last Friday David Stahl asked if I would make an informal presentation on the Canadian situation.

The remarks I will make will be of a general nature and should not necessarily be construed as official policy of AECL. They will be personal observations on several aspects of the program. Although my 22 years experience in the nuclear field ranges from reactor operations, fuel development, and now designing of remotely operated fuel plants for gamma active ^{233}U-Th fuels, I am not expert in the fields of reactor physics, reactor safety, and the political implications of changing the enrichment in our two large research reactors.

So what I would like to do this morning is:

- say a few words on the uranium silicide fuels for which we have significant fabrication, irradiation, and defect performance experience.

- describe the two Canadian high flux research reactors which use high enrichment uranium (HEU) and the fuels currently used in these reactors.

- comment on the limited fabrication work we are doing on Al-U alloys to uranium contents as high as 40 wt%. This work is aimed at our fast neutron program. I will then try and apply this experience in general terms to the NRX and NRU designs of fuel.

U_3Si PROGRAM

For a period of about 10 years AECL had a significant program looking into the possibility of developing U_3Si as a high density replacement for the UO_2 pellet fuel in use in CANDU power reactors. The element design consisted of a Zircaloy-clad U_3Si rod containing suitable voidage to accommodate swelling. We found that the binary U_3Si could not meet the defect criterion for our power reactors, i.e., one month in 300°C water with a defect in the sheath and no significant damage to the element. Since U_3Si could not do the job, a new corrosion resistant ternary U-Si-Al alloy was developed and patented. Fuel elements containing this alloy came close to meeting the defect criterion and showed slightly better irradiation stability than U_3Si. Shortly after this, the program was terminated for other reasons.

We have made much of this experience available to the Low Enrichment Fuel Development Program and will be glad to supply further data to assist this program.

THE NRX AND NRU RESEARCH REACTORS

The Canadian high flux research reactors (1,2) play a vital role in the development of the peaceful uses of atomic energy. They provide outstanding facilities for fuel development, research in neutron physics, and isotope production. Two of these high flux reactors, the NRU and NRX reactors at Chalk River, are fuelled with 93% enriched Al-U alloy rods. The rods are taken to very high burnup, typcially 65% for NRX and 80% for NRU.

Both these reactors have a very heavy experimental load and are important not only to the Canadian program but to the US program as well. The US has occupied several in-reactor loop positions in the NRX reactor for twenty-five years. These reactors are somewhat unique in that they are amongst the few reactors in the world where a full length prototype US design of power reactor fuel element can be tested in a high-temperature, high-pressure loop facility.

NRU is Canada's largest and most versatile research reactor. The prototypical designs and production fuels for the successful CANDU type of heavy water power reactor have all been tested in NRU's two large loops, U-1 and U-2. These loops have heat removal capacities of 8 and 12 MW, a maximum thermal flux of 3×10^{18} $n/m^2 \cdot s$, a pressure tube inner diameter of 100 mm and a useful length of over 2.5 metres. The reactor also has a large number of horizontal beam holes (3×10^8 $n/m^2 \cdot s$) which are used for basic research in physics. The Commerical Products Division of AECL is one of the major world suppliers of radioisotopes such as Mo-99, Co-60, etc., and most of these are produced in NRU and NRX. Large quantities of these isotopes are shipped to the US and other countries. A total of 9 different isotopes are produced in NRX and NRU on a commerical basis.

The NRX reactor is the oldest medium flux (1×10^{18} $n/m^2 \cdot s$ max.) research reactor still in use in the world, having been originally commissioned in 1947. The reactor core is contained in a cylindrical aluminum vessel 2.7 m in diameter and 3.2 m high. The experimental facilities consist of six experimental loops for testing fuel elements up to about 2.5 m long and for corrosion tests on materials, several creep facilities, a neutron radiographic facility, and vertical holes for isotope production. Recently, $1 M was spent to rebuild one of the loops to handle extended irradiations of defect fuel tests and the capability of blowing down the loop for LOCA tests.

I would like to now describe the fuel designs used in these reactors. Unfortunately they are unique to these reactors and are not used by any other reactor in the world at present.

The NRX fuel design (Slide 1) consists of Al-28 wt% U cores (6.3 mm dia. by 2.7 m long) extrusion clad in aluminum with four fins. Seven elements are assembled into one fuel rod inside an outer flow tube.

NRU fuel (Slide 2) consists of Al-21 wt% U cores (5.5 mm dia. by 2.7 m long) extrusion clad in aluminum with six fins. The elements are arranged with three in an inner circle and nine in an outer circle inside a flow tube.

In addition to the reactor fuels which are fabricated at Chalk River, we have an on going program to develop fast neutron rods for irradiating non-fissile metallurgical specimens such as zirconium alloys. Slide 3 shows the Mark 6 FN rod (1.2 m long) which is currently in use in NRX. This rod is produced by coextruding a tubular billet which consists of an Al-40 wt% U fuel core clad internally and externally with aluminum. These rods have been in operation for about six years with limited success. The main problem in fabricating coextruded fuel with high uranium content is the mismatch in coextrusion properties between core and cladding and the brittleness of the alloy. This results in irregularities in the sheath thickness and dogboning.

As an alternative to the Mark 6 FN rod we are working on the Mark 7 FN rod shown in Slide 4. The rod consists of two Zircaloy sheaths separated by a thin (0.5 mm) annulus of 93% enriched UO_2 powder. The UO_2 powder is compacted to 90% TD using vibratory compaction followed by a small swaging reduction. The first 1.2 m long prototype rod should be ready for irradiation shortly. This design of fast neutron rod is easier to make than the coextruded rod, has better cladding quality, and should have a longer reactor life. We believe that the defect behaviour of the oxide design of rod will be as good as or better than the U-Al design both from the viewpoint of defect detection and fuel lost to the system.

The question of changing the enrichment in the fuels for these reactors from 93% to a lower enrichment is a complex one involving physics, safety, fuel design and fabrication, engineering, reactor utilization, and costs. I do not feel that I am qualified to talk on the practicality of the situation at this time. These high performance, highly utilized reactors are essential to the Canadian nuclear program, essential to the world supply of diagnostic radioisotopes, and important to the US nuclear program. Downgrading of their capabilities would not be sensible or acceptable.

Although these reactors are superficially similar, they are in fact quite different and the evaluation and fuel development would have to be done for each. For example, refuelling of NRX is done with the reactor shudown while NRU has on-power fuelling. The NRX is D_2O moderated and light water cooled while NRU is cooled and moderated by D_2O. The data obtained in any evaluation would not be directly applicable to other reactors except in a generic way, since our reactors do not belong to a large family like the MTR's. Also costs of the evaluation could not be spread over many similar reactors.

So rather than commenting directly on the feasibility or practicality of changing enrichment, I would like to talk more on the implications of using higher Al-U alloys in rod type designs similar to the NRX and NRU fuels.

Keep in mind that each pencil in the rod is about three metres long and has a diameter of about 6 mm. The rods are extruded, drawn to size, and then extrusion clad in aluminum.

REFERENCES

1. R. E. Manson and H. E. Smyth, "The NRX Reactor A General Description", AECL-2692 (1968).

2. D. G. Hurst et al, "Utilization of Canadian Research Reactors", Proc. Int. Conf. on Peaceful Uses of Atomic Energy - Geneva (1964).

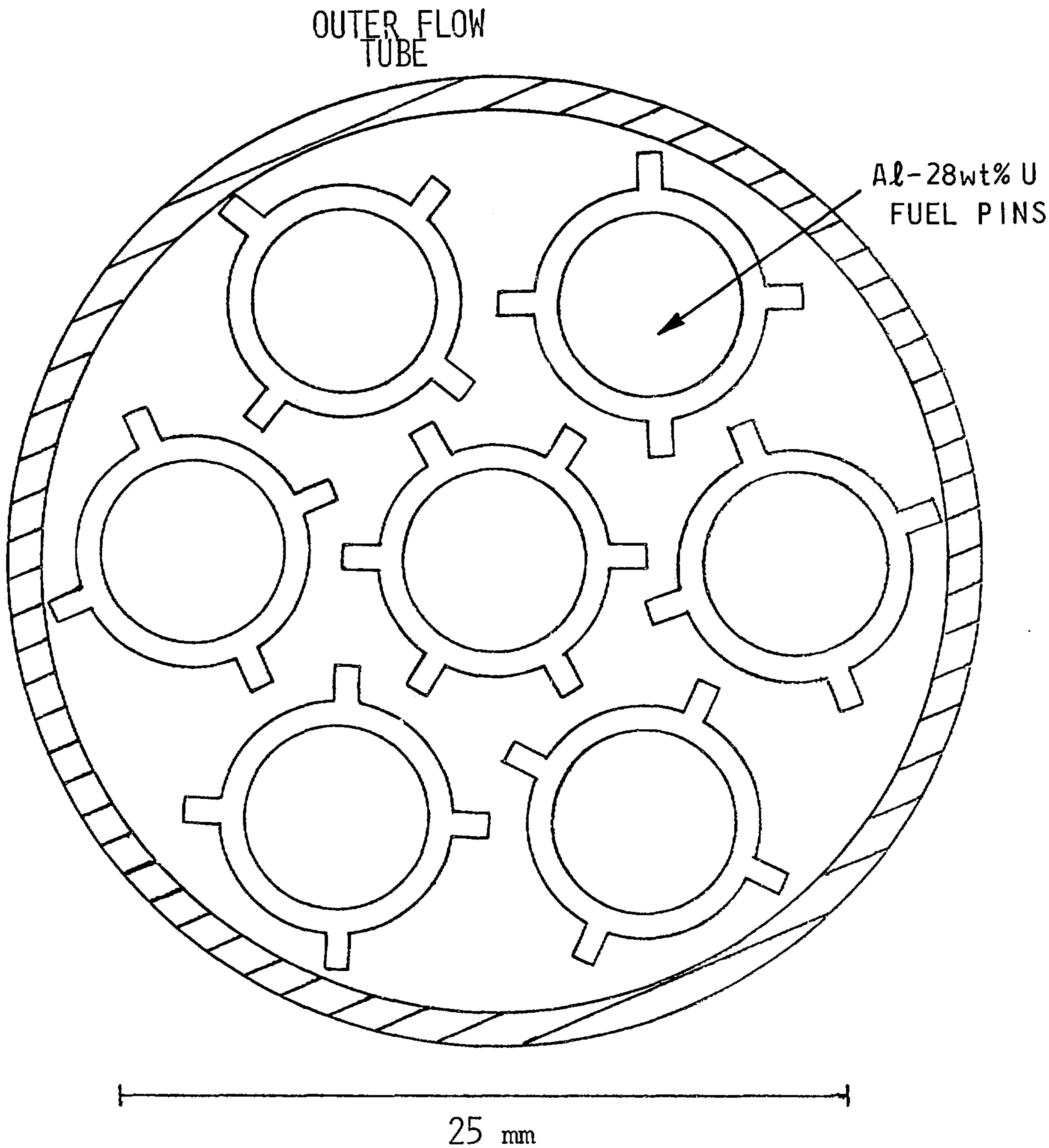
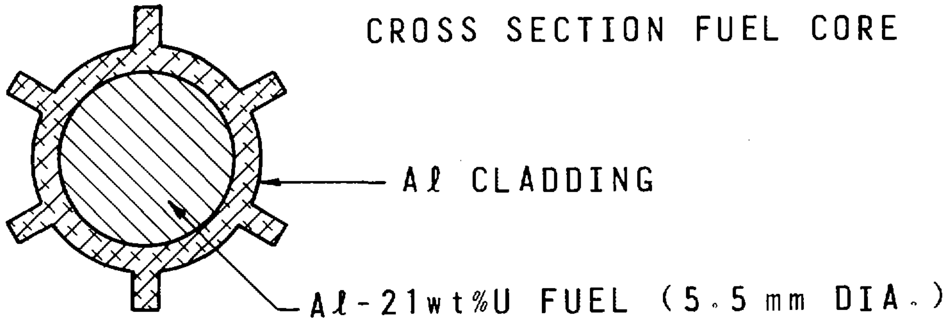


Fig. 1 Seven Element U-Aℓ Rod Design for NRX



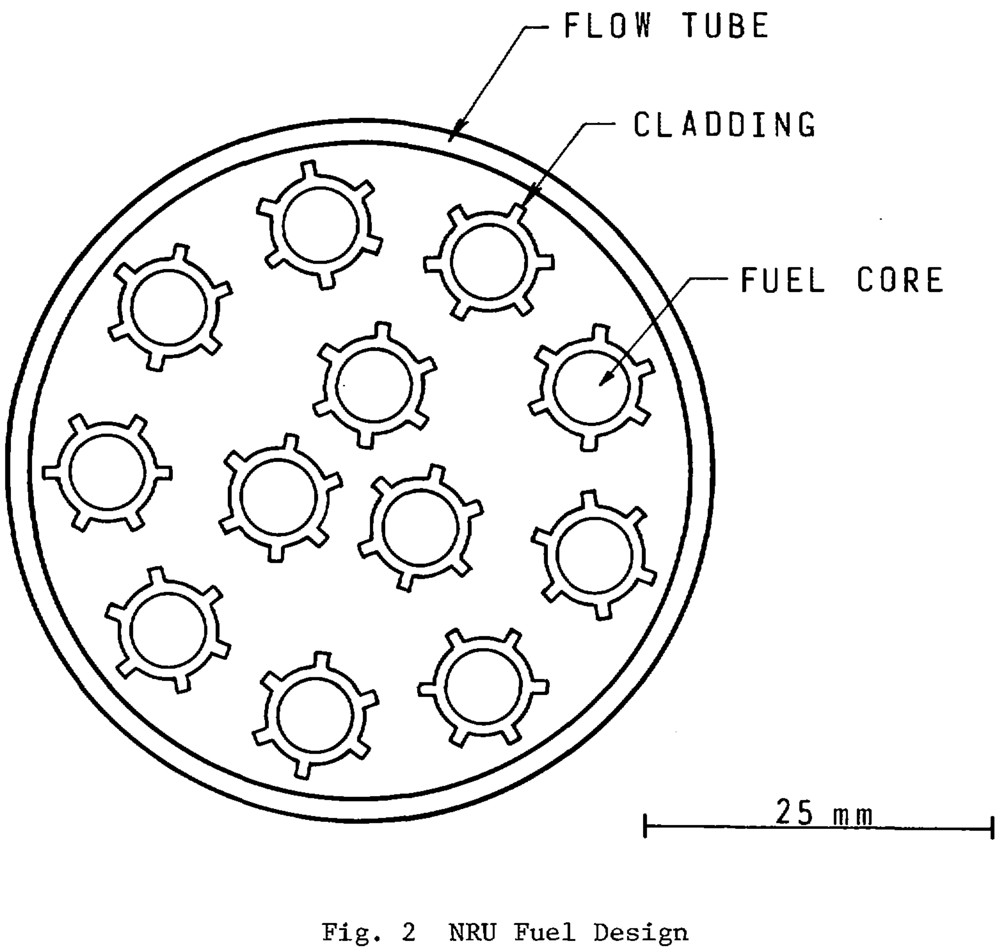


Fig. 2 NRU Fuel Design

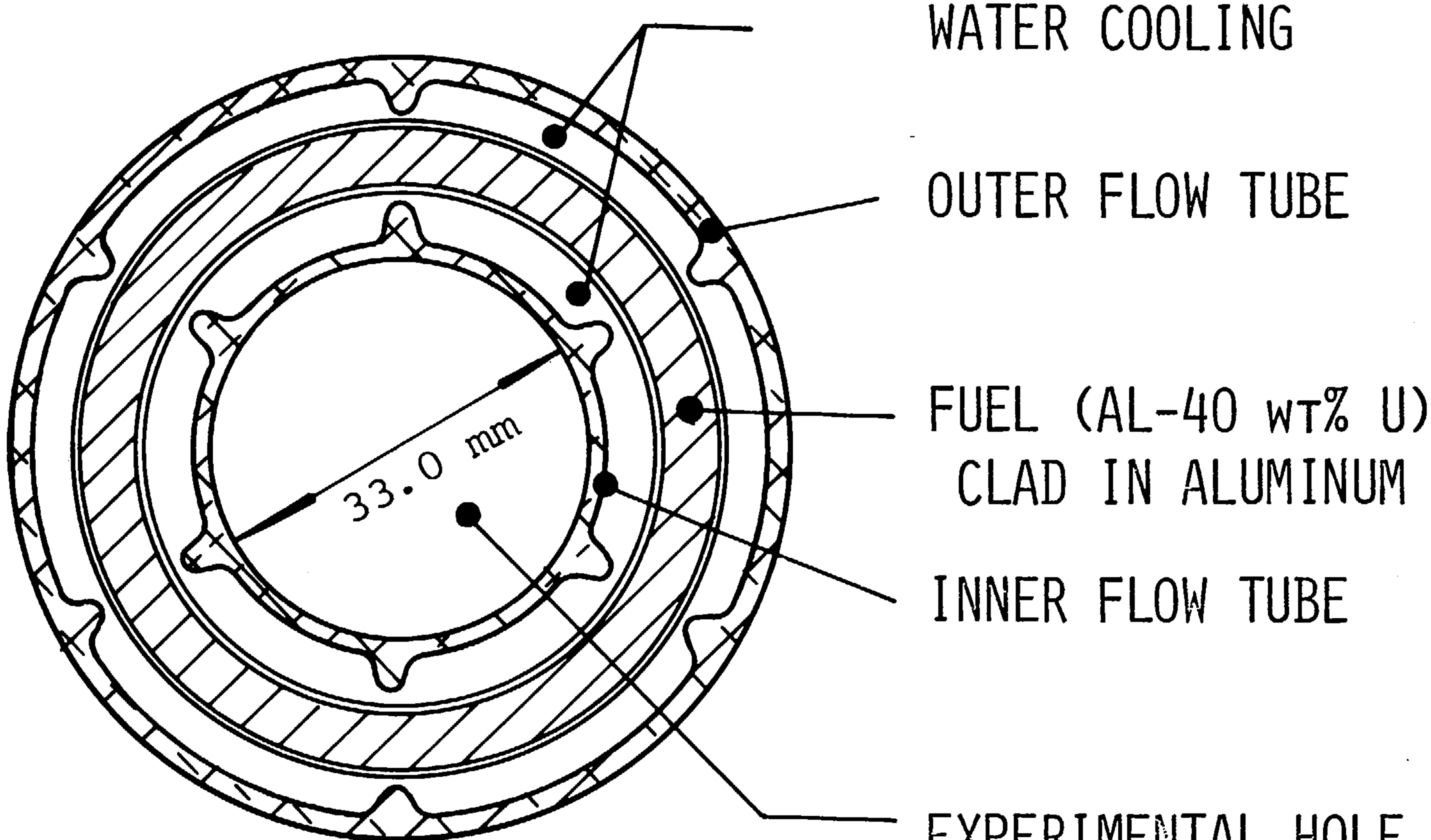


Fig. 3 Mark 6 Fast Neutron Assembly (NRX)

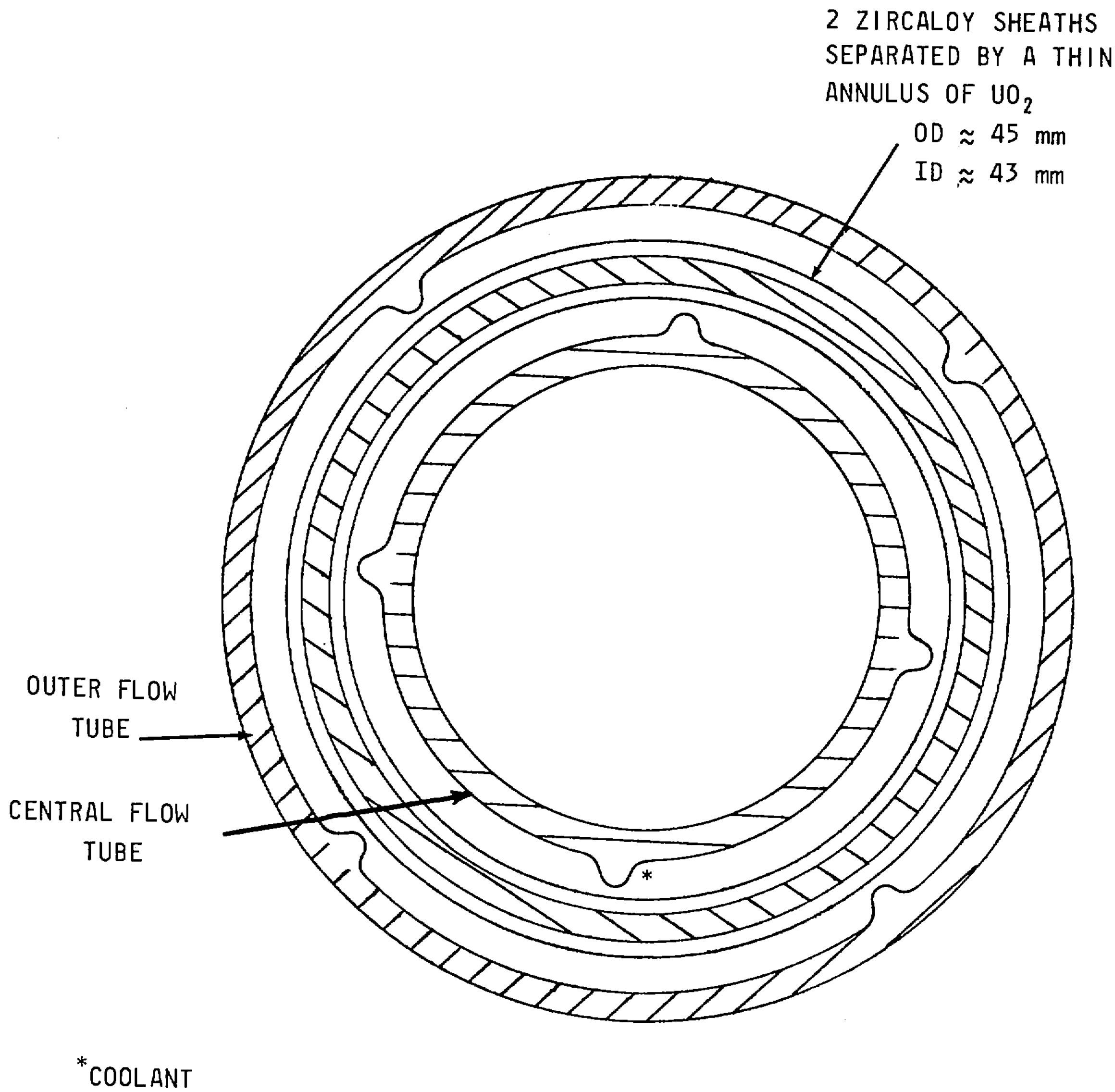


Fig. 4 Mark 7 (UO_2) Fast Neutron Rod for NRX Reactor

DISCUSSION

THRESH (ANL): You talked about the defect experience, of course, on the U_3Si, which you've done a lot of work on. Do you have any comparable defect experience in your reactor on the straight uranium aluminum alloys?

FERADAY: Yes, in some of the defects in the 40 wt% fast neutron rods. The problem with any metal fuel is that obviously you don't get a defect signal until the amount of fuel exposed to the coolant has reached the state where (about 2 centimeters) your defect equipment can pick up the signal. In some of these, the rods were rather badly eaten away before we realized they were defective. If there was a gradual creep up in the activity in the reactor then they suspected it was on one of these elements. On the other hand if you had a UO_2 type of fast neutron rod, as soon as you get a small defect, fission gases are going to be available much more readily and you will see them much more quickly.

CUNNINGHAM (ORNL): I have a suggestion. A number of years ago when we were looking at going to higher-uranium-content uranium aluminum alloys, we were trying to supress this formation of certain compounds in favor of others and also help out the thermal stability and reduce the brittleness of these alloys (they get very brittle as you add uranium). As I recall, there are two additives that look very good, one is silicon and the other is zirconium. The additives are in a range of 1%-3%. I don't know if you can tolerate those additives, but if you can it helps out a great deal on the fabricability of these alloys.

FEREDAY: We had a look at these. We were aware of the silicon, but I wasn't aware ot the zirconium contribution, and, yes, I think that certainly is a possibility.

TRAVELLI (ANL): I'm very much interested in that annular UO_2 design that you showed. We have been listening to a Pulstar type of fuel and to a caramel fuel, and it seems like that type of design of yours has essentially something in common with both of the other fuels, but you seem to achieve a thickness in your annulus which is about 1/3 of the caramel thickness, and I wonder if you could comment a little bit about how you achieve that thickness and whether or not this type of fuel holds a promise for becoming essentially some type of very thin UO_2 fuel that could be used for high density uranium fuel reactors.

FEREDAY: We had done a little fabrication work on this type of fuel and I think it offers a great deal of promise even for the very thin plates. The loading of it with suitable designing of jigs is no problem. We had loaded annuli as narrow as 6/1000 and 10/1000. The time element is not a significant problem. We have done it up to 2 ft in length on the thicker annulus and I think it certainly has a lot of application to some of the designs of the U_3O_8 type of fuel plate. And I think the advantage is that the webb is so thin that you need to only reduce either by swaging or by rolling depending on the shape. If you have a 20/1000 thick laboratory compacted gap, 60% density then to roll it up to 90% you would only need down to 10/1000 or 15/1000. I think that it certainly is an interesting idea and I think that it would be worth taking a look at. The advantage of the UO_2 is that you do get a pretty good defect signal. Once again, this might be an embarrassment to some of the university type reactors, I don't know. It depends on the level of the fission gases that are released. I certainly think it would be worth looking at.