XA04C1901

INIS-XA-C--036

International Meeting on Reduced Enrichment for Research and Test Reactors (RERTR), Petten, The Netherlands, 14-16 October 1985

REDUCED ENRICHMENT ACTIVITIES AT GKSS

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ABSTRACT

The GKSS research centre is participating since 1980 in the national and international efforts on enrichment reduction to clarify the needs on safety and licensing issues, to decide on fuel tests, to perform fuel test and conversion calculations! These efforts are made within the IAEA working groups, the German AF-program and GKSS R & D program.

Especially within the fuel test GKSS has successfully tested 10 fuel elements, 23 plates, 45 %, UAl_x, = 1,41 g U/cc up to 64 % burnup. Today tests are going on for

- 7 fuel elements, 23 plates, 20 %, U_3O_8, = 3,1 g U/cc
- 4 fuel elements, 23 plates, 20 %, U_3Si_2 = 3,7 g U/cc
- 5 miniplates, 5 cm x 18 cm, 20 %, U_3Si_2, = 4,75 g U/cc, fine grain 25 %.

Some of the fuel elements are instrumented with up to 8 thermocouples to perform loss of flow measurements. GKSS has done some reactivity, microflux and temperature measurements.

Considerations are made for the conversion of our 5 MW-reactor FRG-1 and the 15 MW-reactor FRG-2 from 93 % to 20 % enrichment.

1. Present situation and future needs

The GKSS research centre Geesthacht GmbH operates the two MTR-type swimming pool research reactors FRG-1 (5 MW) and FRG-2 (15 MW). These reactors in operation since 27 resp. 22 years are

used for fundamental and applied research, mainly in materials testing and development. Main efforts are made (e.g. installation of a cold neutron source /1/, backfitting /2/) to have today and in future high utilization, availability and safety standard. Both reactors are operated in one reactor hall in a connecting pool system with ca. 800 MWd/a resp. 3200 MWd/a. The fuel elements used are identical: 23 plates, UAl_x, standard geometry, 93 % enriched and 180 g U-235/element, meat dimensions 0,51 x 62,8 x 600 mm^3, canning thickness 0,38 $\pm$ 0,08 mm, canning material AlMgl.

The 5 MW FRG-1 will be used in future with higher availability and to a greater percentage for beam tube experiments. Therefore the installation of a cold neutron source in one of the beam tubes will be supported by reducing the core size. Reducing the core size, enlarging the burnup and fuel cycle length can be made only if the fuel loading will be increased and the control rod design will be changed from old oval type to fork type absorbers.

The 15 MW-FRG-2 is and will be used today and in future mainly for materials testing. The present program (testing of large specimens of pressure vessel steel) require a core size 7 x 7 position. Increase in fuel loading can and will reduce fuel cycle cost significantly and (or) enables to reduce core size due to future changing experimental needs. Therefore the tendency is to increase fuel loading and to change control rod design. As our two reactors are operated in one pool system with the same fresh and spent fuel storage we intend to have the same fuel elements for both reactors in the future, too. This makes design and decisions a little more complicate.

2. Fuel testing

After discussions with our independent experts (TÜV* = consultants of the licensing authority) and the licensing authority it was agreed to have a test procedure for fuel development and testing showing five steps and starting from no knowledge (step 1) and finishing with the conversion of the reactor (step 5) (s. table 1). Only tests of prototype fuel elements should be done in the reactor under consideration /3/. If these prototype fuel element tests are not performed in the reactor under consideration, the conversion procedure will be extremely stretched as operation has to be interrupted by fuel element inspection.

Fuel tests in the FRG-2 reactor were and are made within the German AF-program (RERTR) and for own conversion needs: qualification of high density fuel plates, investigation of fabricational limits and qualification of fuel elements.

*) TÜV = Technischer Überwachungsverein

a. Qualification of fuel plates
It was considered and licensed (Nov. 1981) to test fuel plates up to 4,75 g U/cc in a fuel element. Due to the progress of fuel element development believed to exist at that time it was decided in spring 1982 not to start with such test.

b. Investigation of fabricational limits
Swelling depends on the amount of fine grain (40 µ). 5 fuel plates U_3Si_2 with fine grain content up to 40 % and a U-density of 4,75 g U/cc are under irradiation. Goal burnup is 60 % and will be achieved in summer 1986. Fission product release is measured on line. No defects till today.

c. Qualification of fuel elements (details see table 2)
- 10 fuel elements, 45 % enrichment, UAl_x, = 1,41 g U/cm³, 280 g U-235 were burned up to 64 %, test reason: backup variant
- 7 fuel elements, 20 % enrichment, U_3O_8, = 3,1 g U/cm³, 270 g U-235 are burned up to 55 %, test reason: fabricational limits and backup variant.
- 4 fuel elements, 20 % enrichment U_3Si_2, = 3,7 g U/cm³, 323 g U-235 are burned up to 36 %, test reason: prototype for conversion.

3. Experimental program

For and during the qualification tests of fuel elements especially the following experiments are performed: reactivity measurements (comparison with present fuel), "micro"thermal neutron flux distribution, fuel plate temperature at normal operation and during loss of flow experiments, flux mapping and fission product release.

- the "micro"thermal neutron flux distribution parallel to fuel plate direction was measured many times for HEU/MEU/LEU* fuel elements with different loadings. Max/Min thermal neutron flux in horizontal plane in one water channel parallel to fuel plate direction was determined between 1,15 - 1,18 for the 3 fuel elements in chapter 2c.

- fuel plate temperature and water temperature at top and buttom of the fuel element were measured before and after stop of primary coolant flow to investigate critical cooling conditions and differences in thermal conductivity for the different meat materials. Typical results are presented in fig. 1

*) HEU = High enriched uranium — ca. 90 %
MEU = Medium enriched uranium — ca. 45 %
LEU = Low enriched uranium — below 20 %

showing the temperature peak in the fuel plate shortly after stop of primary coolant flow and before effective natural convection, and the temperature conditions between 100 % forced and natural convection.

- flux mapping in mixed core with 6 U_3O_8 and 4 U_3Si_2 fuel elements (see chapter 2c) were performed. Differences between experimental determined and calculated /4/ peaking factors are below $\pm$ 10 % for 90 % of the fuel elements. The agreement is sufficient.

- sipping tests are periodically made during the tests of new fuel elements as there were made some unlucky experiences with fission product release in the past. All fuel elements show excellent results.

4. Conversion of the FRG-reactors

The conditions from the reactors are: same fuel elements for FRG-1 and FRG-2, unchanged geometry (23 plates), reduction of fuel cycle cost /5/ and increase of excess reactivity (higher U-235 loading, Be-reflector, fork type absorbers), increase of neutron flux for FRG-1 (reduce core size). Status is that all decisions are made and all nuclear calculations are performed.

To license new fuel as a minimum: results of fuel plate and fuel element PIE's must be available (same fuel, higher density, higher burnup), test of prototype fuel must be finished, the fabrication process must be well qualified, the ORR whole core demonstration must be licensed and first results available, the future use of the selected fuel by others too, has to be undoubtedly clear for economical reasons.Qualification of U_3Si_2 up to 3,7 g U/cc for the conversion of the FRG-reactors is mainly done but there are some remaining questions: PIE reports, all fabricators fully (!?) qualified, prototype test is in progress, ORR reports are missing. Hopefully all results will be positive and available latest beginning of 1986.

Licensing demands are qualification of fuel (a few remaining questions), nuclear calculations (finished), safety calculations (in progress), safety report (will be prepared in spring 1986, if above mentioned points give positive results) and a guarantee for reprocessing or fuel disposal (no special agreement for a limited number of fuel elements).

If (in spring 1986) the safety report is submitted to the licensing authority, (consultants check, additional questions and/or

demands (?), approval) it will last probably as a minimum one year before the license will be granted. Therefore it is believed that the conversion procedure will not start before mid 1987 with a mixed core procedure for FRG-2 and a "reduced" mixed core procedure for FRG-1. "Reduced" = More 20 % fuel elements to achieve within a short time the desired small core configuration. The total mixed core operation time will be of the order of two years. The time schedule described here is the optimistic approach.

Literature

/1/ W. Krull: Design considerations for cold neutron sources, Seminar on the use of research reactors in fundamental and applied sciences, Tripolis, 16. - 20. September 1984. See also IAEA-SR-119/27 (1985)

/2/ W. Krull: Upgrading of the research reactors FRG-1 and FRG-2, Seminar on research reactor operation and use, IAEA-SR-77/27, Jülich, Sept. 14. - 18., 1981

/3/ W. Krull: Remarks on the demands for the qualification of high density fuel, GKSS/E/40 (1984)

/4/ A. Strömich: private communication 1985

/5/ W. Krull: Remarks on the influence of enrichment reduction on fuel cycle cost, RERTR meeting, Argonne, 15. - 18. Oct. 1984

Table 1: Conversion steps

Step	Knowledge	Test procedure	Number	PIE*
1	nothing	irradiation of miniplates in rigs	few	yes
2	something	miniplates in pool water	many	yes
3	advanced	test fuel elements	1 - 2	yes
4	excellent	prototype fuel elements (for conversion)	5 - 10	no
5	all main questions answered	conversion	unlimited	no

*) PIE = Post irradiation examination

Table 2: Fuel Elements Qualification

	I	II	III	IV
Weight U-235 (g)	180	280	270	323
Weight U (g)	193, 200	622	1357	635
Enrichment (%)	90, 93	45	20	20
Meat material	UAl_x	UAl_x	U_3O_8	U_3Si_2
Density (g U/cc)	0,44 (0,45)	1,41	3,1	3,7
Canning material	AlMg1	AlMg1	AlMg2	AlMg2
Number		10	7	4
Goal average burnup (%)	45	60	65	65
Burnup MWd	64,8	134,4	140,4	168,0
Peak burnup 10^{21} fissions/cc (ex Pu)	0,48	0,95	0,99	1,18
Test status	present fuel	finished	test	test
Test finished	open	April 84	Spring 86	Autumn 86
Max peak burnup* 10^{21} fissions/cc (ex Pu)	0,66	1,01	0,68-0,86	0,61-0,70
Mean U-235 burnup* (%)	45	58,9-64,0	42,7-55,4	31,5-36,0

*) 15. October 1985

Fig.1 : L O F A fuel and water temperatures