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Some Results on the Development of a Fast Reactor Fuel Element

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Contents:

1. INTRODUCTION

Within our Fast Breeder Project a major effort has been started some years ago in order to develop fuel elements suitable for large fast power reactors with different types of coolant. While primarily three coolant mediums, namely sodium, helium and steam, were considered the project has been concentrated now to the two lines of sodium or steam cooling, respectively. In the first step of development also different possible fuel types, that is metal, oxide and carbide were evaluated with the conclusion that only ceramic fuel may meet the economic requirements which are highly emphasized in the project. Within the different ceramic fuels mechanically mixed oxide was finally chosen as the fuel for the prototype reactors. This decision is mainly based on the available technical background and experience in the ceramic field. Although our present initiative is concentrated only on oxide fuel we may try a second approach in the future with other ceramics.

The present paper describes shortly the logical structure of our works, the experimental activities including some available results and the theoretical background we try to establish for the whole fuel element development.

2. OBJECTIVES AND PROGRAM

2.1 Requirements

The requirements for our fuel element development are based upon the fast breeder reactor design studies for a 1000 MWe sodium cooled reactor and for a 1000 MWe steam cooled version. Although the basic parameters of both designs have a good deal of similarity there are two different development lines. Some of the lay-out parameters are exactly equal for both types while other requirements - for instance corrosion performance or mechanical features of the cladding - are quite different. In any case a high burnup (up to 100 000 MWd/ton) is required. The maximum linear rod power in the present reactor designs lies between about 400 watts/cm (steam cooling) and 600 watts/cm (sodium cooling).

2.2 Fundamental Working Hypotheses

In order to profile a reasonable development program it is necessary to establish some working hypotheses, especially for the fuel parameters.

These assumptions are step by step "streamlined" following the evaluation of the own development work and the results in the published literature. They deal with the following groups of fuel pin parameters:

A) Fuel density, expansion and swelling
B) Fuel temperature distribution
C) Fuel composition and fuel-clad interaction
D) Pressure build-up and cladding design

The detailed working assumptions are as follows:

1 (A) The basic requirements in view of solid fission product swelling and slumping danger are met in the density region 80 - 90 % th.d. (smeared), presumably at about 85 % th.d.

2 (A) The thermal expansion of porous fuel ceramic is similar to 100 % dense material; therefore we take the (conservative) 100 %-values for calculating the necessary gap width.

3 (A) The model for radial swelling assumes an initially low swelling rate increasing continuously with burnup.

4 (B) With He-bonding and a diametral gap of less than 150 µ the thermal gap conductance is not worse than 1 watt/cm^2 $^\circ$C. A corresponding temperature rise is also applied to vibratory compacted fuel, thus counterbalancing the lower conductivity of powder fuel in the cooler regions.

5 (B) The thermal conductivity integral (from 0 $^\circ$C to melting point) for the 100 % dense mixed oxide is not lower than 95 watts/cm. For the reduction due to porosity a conservatively modified Loeb-equation is applied.

6 (B) At local fuel temperatures above 1600 $^\circ$C migration of voids is assumed. The central hole volume thus formed will be not larger than about 50 % of the original porous volume.

7 (C) Concerning the stoichiometry question we assume that stoichiometric or only slightly substoichiometric oxide can reliably be fabricated and do not change the composition during burnup in an amount that a larger adverse influence to related parameters (as thermal conductivity) must be expected.

8 (C) The necessary microscopic homogeneity in the UO_2-PuO_2-mixture can properly be reached by mechanical mixing. Provided that the maximum excursions have ramp rates below 15 dollars/sec - as evaluated in our Na1 reference design study /¯1_7 - theoretical calculations show that the largest highly Pu-concentrated oxide spots should be smaller than 150 μ in diameter /¯2_7. A slight macroscopic segregation trend - for instance induced by diffusion in a thermal gradient - will not be harmful to fuel behaviour or core physics. Also some few PuO_2-agglomerates above 150 μ are insignificant.

9 (C) The compatibility between oxide fuel and the proposed can material categories (stainless steel and Ni-base alloys) will raise no major problems if only He-bonding (or vacuum) is applied.

10 (D) For calculation of fission gas pressure the fission gas release is (conservatively) assumed to be 100 % thus counterbalancing some possible drawback coming from the doubt whether the gas can always migrate completely free to the plenum.

11 (D) The most important characteristic property of the can material is the long term creep behaviour which has to face the superposition of gas pressure stress and thermal stress. The yield point may be surpassed only in the first few thermal cycles.

12 (D) Properly evaluated single alloys of the chosen can material categories suffer no major detrimental effects of high fast neutron integrated flux to their overall mechanical behaviour.

A lot of these hypotheses are highly conservative or to some extent founded by experimental results while others are only theorized and need practical proof.

2.3 Structure of Development Program

The whole development program began with the evaluation of the specific requirements and led in a first step to some fundamental decisions concerning pin type and fuel, see the formal scheme in Fig.1. On this working basis a network of development works concerning the fuel, can material and fuel pin production was established followed by an irradiation program concentrated not only on fuel and single cladding materials irradiation but also on performance tests of prototype fuel pins. In the last step all the experimental results will be used to formulate the concept, design and specification of prototype fuel element for both lines of development.

3. EXPERIMENTAL DEVELOPMENT WORK

3.1 Test Production of Fuels

In preliminary steps the proposal to increase the thermal conductivity of the oxide ceramic by some metal additives was experimentally evaluated. Some encouraging results were reached by Mo-addings of 10 to 15 w/o, uniformly distributed or as particle coatings. The normally low thermal conductivity at temperatures above 1000 °C jumped to doubled or threefold values. In view of this not too large progress compared to the additional production difficulties, but especially in view of the detrimental influence of any Mo in the core to the fast reactor physics (Doppler- and coolant void-coefficient, breeding ratio) we have dropped this fuel side line completely.

All our effort is focussed now to pure oxide where we distinguish 3 simulation steps, namely UO_2, UO_2-CeO_2 and UO_2-PuO_2, both mixtures mechanically mixed. Due to some lack in Pu-facilities in the first phase of development we had to gain the preliminary know-how with simulated fuel. Since one year however the experience is transfered now step by step to the handling of Pu-containing oxide.

In all our work the two basic fuel types, that is pellet fuel and compacted powder, are examined in parallel lines. For the pellet type we realized the production of small diameter pellets (5 to 7 mm) with densities in the range 89 to 92 % th.d. We produced rather large series in the order of 10 000 pieces, mostly without dishing. Thus we got some production routine with such small pellets. The homogeneity as reached by mechanical mixing is sufficient. Large series of pellets can be sintered with such precision that they need no final grinding procedure if a diameter tolerance of ± 50 µ is specified.

For the vibration powder production a new method was evaluated avoiding the large and costly feedback of sintered oxide refuse. In this new process the ceramic grade material is pressed to large green pellets and milled before sintering to particles. After that the green particles are rounded, screened to a proper vibration characteristic and sintered. We attained with such powder at 5.5 to 7 mm fuel diameter a density up to 87 % of theoretical routinewise. We are earnestly considering also the powder production process by melting in the direct current, which was developed by Nukem recently [3]. This is a very easy way to produce high dense particles on large scale. The vibration results are very encouraging, too. It seems that no major difficulties for such a routine production with Pu may arise.

These different oxide lines are rather promising especially from an economic standpoint. Our present effort aims to verify this conclusion - based on experience with UO_2 and UO_2-CeO_2 - also for Pu-containing mixed oxide. Therefore we are about to enlarge the laboratory scale with UO_2-PuO_2 to a real test production scale.

3.2 Cladding Materials

The strong can concept which is exclusively pursued in this stage of development asks for cladding materials with good creep behaviour at high temperatures. A characteristic maximum temperature (including hot spot conditions) for Na-cooling is 650 °C, for steamcooling 750 °C. At these temperatures a long term mechanical strength of at least 15 kg/mm^2 should be available. This goal can be envisaged - from a principal standpoint - with different classes of alloys. Of course the commercial advanced Fe-base and Ni-base alloys have to be considered first while other promising classes as particularly vanadium and its alloys need a long range development.

We have investigated in screening tests a lot of the different stainless steels and the available Ni-alloys of the Inconel type. Our present selection is mainly concentrated to a 16Cr13Ni stainless steel for Na-cooling and Inconel 625 for steam cooling. For V-base alloys we have started some years ago a metallurgical development. The already finished first stage of experimental work investigated the possible alloying components and the mechanical properties of such compositions. Now the production studies for tubes of small diameters are underway, it was possible to fabricate tubing of V-10Ti and V-10Ti-10Nb. The development of more advanced Fe-base alloys, especially of the Incoloy type, is considered to be very promising. While the now available Incoloy 800 has still a mechanical strength not high enough we expect a marked improvement by some additives. We are stimulating such development lines and hope to get further can material for steam and sodium cooled projects.

On all this potential cladding materials the most important property data are rechecked by own out-of-pile testing. The mechanical experiments deal mainly with short and long time creep rupture tests (up to 25 000 hours) at high temperatures. Also some corrosion experiments in sodium and steam were carried out. The preliminary conclusion is that for sodium cooling 16Cr13Ni stainless steel and for steam cooling Inconel 625 meet the mechanical and corrosion requirements best.

3.3 Test Production of Fuel Pins

This is the final step of the experimental development work. It uses fuel produced according to the production routine described in 3.1 and cladding materials of the selection of 3.2. There are two different objectives. On the one side we have to establish and to try a realistic pin production on a larger scale in order to supplement some more technical experience to the pure laboratory knowledge. On the other side there are to be produced irradiation specimens for the fuel irradiations program of section 4.3.

The large scale pin production uses as simulation fuel natural UO_2. The pin design is according to one of the reactor reference designs, see Fig.2. The production comprises about 200 pins. Each set of about 25 pins is slightly different from the others, because the different fuel types should become experienced. The results of this test production show whether the established specifications can be met and give also some valuable indication concerning the production costs.

The fabrication of irradiation specimens is a more delicate task, because the specifications are stronger than for routine production. About one hundred specimens were fabricated up to now. Presently the pins for the irradiation in the EFFBR are in preparation. As Fig.3 demonstrates they show all important features (including the dimensions) of real oxide breeder fuel pins.

3.4 Economic Consideration

Our test productions of fuels and pins were used to establish economic outlines for a future real large scale production. The incentive hereby was expressively not to figure out the present situation but the extrapolated state, say, at a continuous production of about 20 tons of fuel per year after having overcome initial difficulties in a startup period. We are presently calculating the pin production costs (per kg of fuel) in oxide fuel type. We shall deal with the results of this investigation in a later publication.+)

4. IRRADIATION EXPERIMENTS

4.1 Experimental Questions

The fundamental question is directed to the relation between obtainable burnup of a fuel pin on the one side and production specifications and operation conditions on the other side. In this connection the fuel density and the linear rod power are most important. The definition of the expression "obtainable burnup" has at least two increments. We define a special burnup to be obtainable if

on the outside of the pin no major change, geometrical or else, has been evolved which prohibits or limits normal function

in the inside of the pin no conditions are being established or developed which influence adversely normal function or which has to be considered as a potential danger for reactor safety.

Within this framework, all special questions which can be treated by irradiation experiments are included, e.g. change of temperature distribution, central hole formation, radial and axial swelling, migration of fuel components and fission products. Above that a special emphasis is justified for the question whether detrimental changes and influences are developed continuously or whether there exist critical periods in the life of a fuel pin. One such critical period is the startup of newly produced fuel. Therefore the startup behaviour of a pin requires additional attention. Another such critical period is an intermediate swelling stage at which the fission gas remains trapped in the fuel.

+) see KFK-576

4.2 Scope of the Irradiation Program

The whole irradiation program is distributed in different test series. Some of them deal only with the irradiation of can material specimens in a hard neutron spectrum and constitute the material irradiation program. Parallel to this the fuel irradiation program which contains short fuel specimens and complete fuel pins is carried out, most of it in a thermal neutron spectrum, a very important part however in a fast flux. The Tables I and II show a compilation of the whole program and give also the most characteristic features of each test series.

Naturally the irradiation space with a real fast flux is very limited. Therefore it is highly important that also in thermal reactors many experiments give valuable results for a fast reactor development. In this situation the question arises how to profile the experiments and to distribute them on to the available irradiation space. We have answered for our program with the following scheme:

a) As the mechanical properties of can materials are highly influenced by fast neutrons the materials irradiation take place in a high flux region of a hard spectrum. Thus integrated fast fluxes up to 5×10^{22} nvt are obtainable within reasonable time.

b) Most of the fuel and fuel pin irradiations can be undertaken in thermal reactors as there should not be larger differences between thermal and fast flux in the fuel behaviour.

c) Finally one series of pin irradiations is carried out in a real fast reactor as a performance test and as a real proof for the assumptions.

4.3 Fuel Irradiations

The fuel irradiations of our program have three large sections:

A) The capsule irradiations in the FR2 reactor
B) The irradiations in the FR2-Helium-Loop
C) The irradiations in a EFFBR high flux position

<u>Ad A.</u> For irradiations on normal fuel element positions of the FR2 reactor in Karlsruhe a special capsule was developed. In the last design the heat transfer between fuel and cooling water (D_2O) is performed by

a twin layer of liquid metals, namely Na and Pb-Bi-alloy. In such a capsule 4 "short specimens" with a length of about 17 cm each can be installed or 1 "long specimen" which has good similarity to realistic dimensions. The surface temperature of the specimens is controlled by thermocouples. The construction allows a linear rod power of up to 750 watts/cm. With this device 5 main test groups are specified. Test group 1 contained 5 capsules and 20 short specimens. The initial irradiation of this type was carried out at a rod power of 450 watts/cm maximum up to about 7000 MWD/ton burnup. Test group 2 with 8 capsules is irradiating 32 specimens with reduced diameter at somewhat higher rod power. Test group 3 - now underway - will still contain short specimens in a rather large amount, it constitutes a screening test which compares different fuel variants at different conditions up to very high burnups. Test group 4 which repeats a similar screening with Pu-containing fuel is a very important supplement to test group 3. Test group 5 now will bring the first long specimen irradiations in the capsules. The pins will be designed similar to the EFFBR-pins, see below. Further test groups with capsules are planned but not yet specified in detail. They are reserved to additional fuel variants and new concepts.

Ad B. For the central channel of the FR2 a He-loop with two irradiation devices was designed and constructed. The "long term irradiation rig" operates at rod powers up to 700 watts/cm at 8.6 mm fuel diameter. The irradiations now underway shall lead up to 100 000 MWD/ton burnup under these gas cooling conditions. With the "short term irradiation rig" the test samples can be introduced into the reactor core and removed during reactor operation. Thus short term changes of the fuel, especially the startup behaviour, can be observed. The short term irradiation program comprises about 60 specimens which shall be irradiated short times between minutes and days at rod powers between 500 and 1000 watts/cm.

Ad C. Under an APDA-Euratom-contract there will be available one high flux position in the Fermi reactor in the USA for our program. The now far advanced preparations envisage the irradiation of two or three batches of 12 pins each. For the pin design see Fig.3. Each batch will contain pins with different cladding materials (16Cr13Ni SS and Inconel 625) and different oxide fuels (pellets and vibrational compacted). The rod power shall be about 600 watts/cm at a maximum sodium temperature of

550 °C. The pin design is based on the assumptions for 100 000 MWD/ton burnup but due to contractually limited irradiation time we shall not reach more than about 60 000 MWD/ton.

4.4 Examination of Irradiated Fuel

For the post irradiation examination of fuel specimens and pins we follow a working scheme which contains all important steps, see Fig.4. Especially the microprobe technique is included [4]. The single investigation is accompanied by a thorough calculation concerning the obtained accuracy in the light of the experimental possibilities.

4.5 Material Irradiations

As most of the fuel irradiations are not fully realistic concerning the cladding behaviour the material irradiation program must try to simulate the requirements step by step. It has three major sections:

A) Irradiation of tensile specimens in a thermal reactor with high fast flux component at low and high temperatures (Mol 1, Mol 3, DIDO M)

B) Irradiation of pressurized tubes in a thermal reactor with high fast flux component at high temperatures (Mol 2)

C) Irradiation of tensile specimens in a fast reactor (Fermi M)

Ad A. The cylindrical and flat specimens of the test series Mol 1 are irradiated in the reactor BR2 at Mol, Belgium, at temperatures below 100 °C in direct contact to the cooling water. In this very simple parameter constellation it is already possible to study some of the most important features of the irradiation behaviour of structural materials. At first the influence of fast neutrons onto the low temperature embrittlement becomes evident. Secondly the high temperature embrittlement induced by (n,α)-reactions can be evaluated. Finally the annealing of low temperature damage, especially the radiation annealing temperature, can be determined in detail applying different annealing schedules after irradiation. We have carried out up to now many tests of this type, see compilation in Table III. As Fig.5 for some of the investigated materials demonstrates the low temperature embrittlement after an integrated fast flux of about 3×10^{20} nvt (>0.1 MeV) can be fully annealed. The results of another test however show that this is not completely possible at a higher irradiation dose. Examples for high temperature embrittlement are demonstrated in Fig.6. The details of these investigations are published elsewhere [5]. Also some related studies dealing specifically with high temperature embrittlement [6] and its theoretical background [7] and with the irradiation behaviour of precipitation hardened alloys [8] were carried out. The Mol 1 series is being continued, recent additional results concerning the irradiation of V-base alloys are presently compiled [9]. The post irradiation examination of further tests shall also include long term creep experiments after the proper creep devices are installed in our hot cell facilities. The Mol 3- and the DIDO M-test series (in the reactor FRJ2, Jülich) work with the same type of specimens as Mol 1 but at high temperatures of 650 °C in different contact media as inert gas and sodium. Having the right temperature (and possibly the real cooling medium) at the specimen's surface these experiments simulate the reality by some steps more. In addition to the results of the Mol 1 tests an exact and final judgement of the high temperature embrittlement needs an experimental evaluation of the influence of the irradiation temperature to this embrittlement. After irradiation the Mol 3 specimens shall mainly be examined in creep tests up to 3000 hours in accordance to the long term stress in a fuel pin.

<u>Ad B.</u> The pressurized tubes bring remarkable additional steps of simulation. At the Mol 2 experiments the temperatures (600 and 700 °C) and the pressures (up to several hundreds of atmospheres) are exactly controlled and recorded. Thus any fission gas pressure at high burnup and the wall temperature of a fuel pin can be simulated but not the - nevertheless rather important - temperature gradient across the wall. The whole equipment for this series of irradiations was specially designed and is presently installed at the BR2 site. The pressures are pre-determined following two points of view. A part of the tubes will be pressurized such that they - hopefully - should burst in pile after some time. Hereby clear creep rupture results are wanted. The other part of the specimens is lower pressurized in order to study the in pile secondary creep of thin walled tubes. We can reach within reasonable time an integrated fast flux of 2×10^{21} nvt, this order of magnitude being still too

low for simulation of real fast reactor fuel pins.

Ad C. In the "Materials Surveillance Subassembly" of the EFFBR there are presently 32 materials specimens under irradiation at 350 °C in direct contact to the Fermi sodium. This first set of tensile specimens (16Cr13Ni, Inconel 625, V-5Ti-20Nb, V-20Ti-20Nb) will be removed in summer 1966 having then an integrated fast flux of about 5×10^{21} nvt. It will immediately be replaced by a second set of specimens which will hopefully get an integrated flux far above 10^{22} nvt. The post irradiation examination will be the same as in the Mol 1 series. As we have here an integrated flux representative for fast reactor conditions this experiment is the missing link in the chain and allows to control and prove the extrapolation of former results. Furthermore the possible influence of fast (n,α)-reactions which become evident only with such high integrated fluxes (because of their low neutron cross sections) can be evaluated.

5. CONCLUSIONS

In view of the investigations and available results our preliminary conclusions with reference to the requirements are as follows:

1 With oxide fuel the required linear rod power can easily be achieved below central melting. Special provisions as Na-bonding, metallic additives, substoichiometry and highest fabricable densities for improving the thermal conductance and conductivity are not necessary.

2 The fuel pins with the proper fuel geometry and density can routinewise be fabricated. The specific requirements have no important influence to pin fabrication economy.

3 The necessary U-Pu-homogeneity in the fuel can be achieved by mechanical mixing of the oxide components. The upper limit for PuO_2 particle size of about 150 µ raises no fabrication problem.

4 At a pin type for sodium cooling no special problems arise with respect to the required cladding temperature of 650 °C provided that a fission gas space (calculated for 100 % fission gas release) is available which limits the maximum pressure to about 100 atm. Also in view of the thermal stresses austenitic steels of the type 16Cr13Ni are sufficient.

5 At steam cooling the necessary maximum temperature of up to 750 °C and the high external pressure cause additional problems. Ni-base alloys as Inconel 625 seem just to meet the requirements if also enough fission gas space is available. A special danger to the stability of the can may arise from creep-buckling. As we consider the ceramic fuel to give no internal support to the wall we might have to provide an extra internal gas pressure.

6 The irradiation damage to can materials of the selected types at doses up to 2×10^{21} n/cm^2 (achieved up to now) are not harmful to mechanical properties. An extrapolation to higher doses seems possible so that there might be no limitations for very high pin burnups.

7 Considering the fuel the required high burnup seems to be achievable provided that no major mechanical fuel clad interaction - due to swelling - takes place. In order to adjust the internal geometry to the expected swelling a high free volume of about 15 % in the fuel region is necessary. A too low fuel density however might increase the danger for fuel slumping. A high fission gas release is considered to be advantageous as it lowers the swelling rate.

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Table I Fuel Irradiation Program

	Test Group	Number of Specimens	Fuel Diameter (mm)	Specimen Length (mm)	Fuel Type Pellets + Vibro-compacted Powder	Cladding Temperature (°C)	Linear Rod Power (watts/cm)	Maximum Burnup (MWD/ton)
Capsule Irradiations in FR2	1	20	10.0	240	UO_2, UO_2-Mo, UO_2-CeO_2	500	450	7000
	2	32	8.6	240	UO_2, UO_2-CeO_2	500	500	9000
	3	64	6.4	173	UO_2, UO_2-CeO_2	520 - 580	700	80000
	4	32	6.4	173	UO_2-PuO_2	520 - 580	700	100000
	5	8	5.55	1300	UO_2-PuO_2	520 - 580	650	100000
Loop Irradiations in FR2	1	24	8.6	114	UO_2	550	700	5000
	2	26	10.0	160	UO_2, UO_2-CeO_2	600	1000	-
	3	26	10.0	160	UO_2-PuO_2	600	1000	-
	4	20	8.6	114	UO_2-PuO_2	550	700	50-100000
Pin Irradiations in EFFBR		36	5.55	1286	UO_2-PuO_2	550	600	60000

Table II Material Irradiation Program

Specimen Type and Reactor	Test Designation	Materials	Number of Specimens	Irradiation Conditions			
				Contact Medium	Temperature (^{o}C)	Maximum Integrated Flux Thermal (nvt)	>0,1 MeV (nvt)
Tensile Specimens in BR2 and FRJ2	Mol 1	16Cr13Ni;15Cr25Ni; 20Cr25Ni;Inconel 102,600,625,718, X750;Incoloy 800; V-alloys; TZM;	473	H_2O	<100	1.5×10^{21}	1.4×10^{21}
	Mol 3	16Cr13Ni;Inconel 625,718;V-alloys	72/50	He/Na	600	5×10^{21}	5×10^{21}
	DIDO M	16Cr13Ni;Incoloy; Inconel 625,718; V-alloys	96	He	600	1×10^{21}	5×10^{20}
Pressurized Tubes in BR2	Mol 2	16Cr13Ni;20Cr25Ni; Incoloy 800; Inconel 625,718, X750;V-alloys	112	He	600/700	2.5×10^{21}	2.2×10^{21}
Tensile Specimens in EFFBR	Fermi M	16Cr13Ni; Inconel 625; V-alloys	64	Na	350		5×10^{22}

Table III Specimens Irradiated in the Mol 1 - Series

Test	Material	Number of Specimens	Average Integrated Flux in 10^{20} nvt	
			Thermal	> 0.1 MeV
Mol - 1A	16Cr13Ni 15Cr25Ni Inconel X750	16 15 30	5.5	4.8
Mol - 1B	16Cr13Ni 20Cr25Ni Inconel 600 Inconel X750	15 15 15 16	3.0	2.7
Mol - 1C	16Cr13Ni 20Cr25Ni Inconel 600 Inconel X750	15 15 15 16	13.0	11.0
Mol - 1D	Inconel 102 Inconel 625 Inconel X750 Incoloy 800 V-base alloys	8 3 24 1 45	16.5	15.0
Mol - 1E	V-base alloys TZM	30 11	17.0	15
Mol - 1F	Incoloy 800 Inconel X750	31 50	17.6	16
Mol - 1G	16Cr13Ni Inconel 625 Inconel 718	30 29 28	In Preparation	

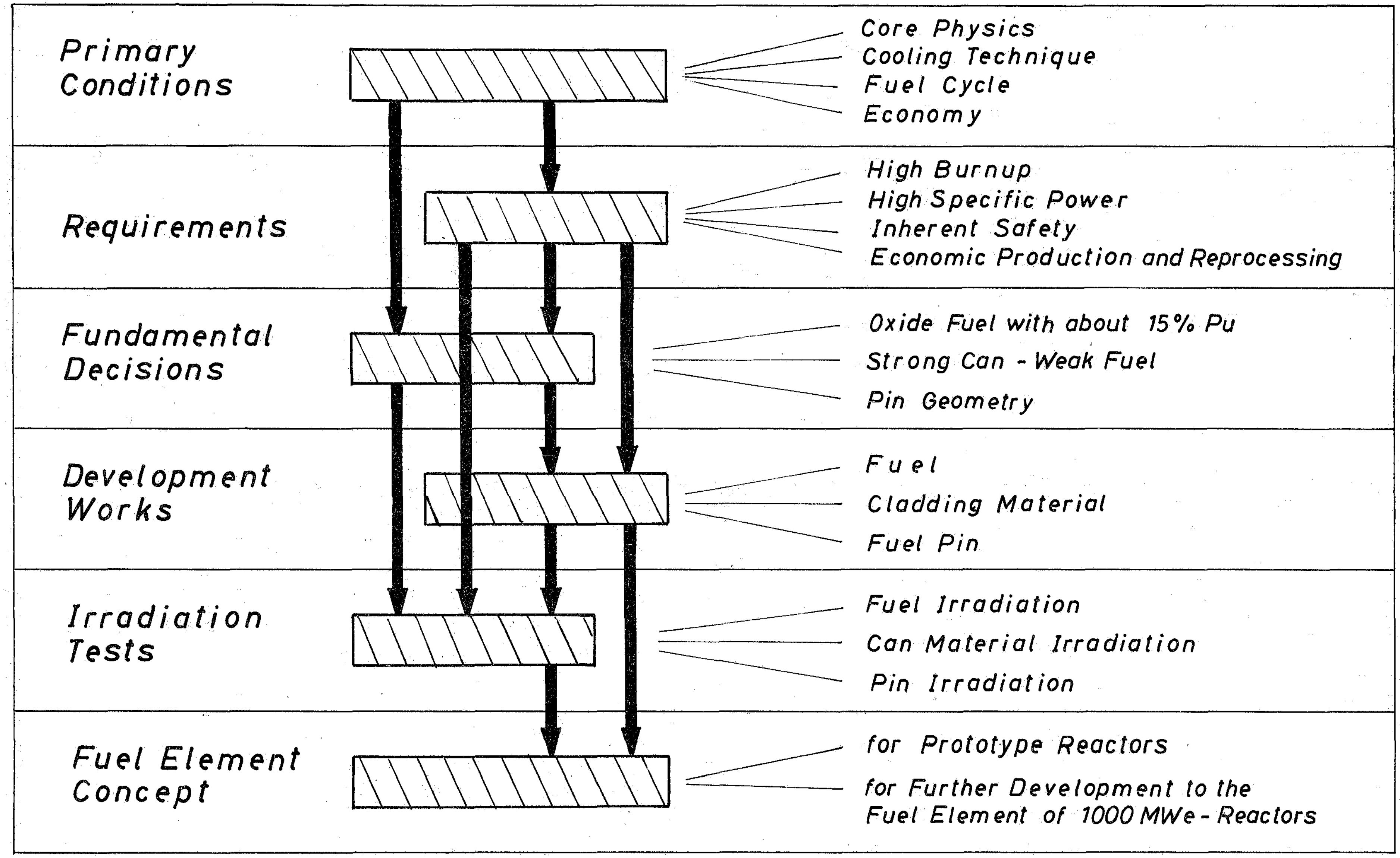


Fig. 1 STRUCTURE OF FUEL ELEMENT DEVELOPMENT

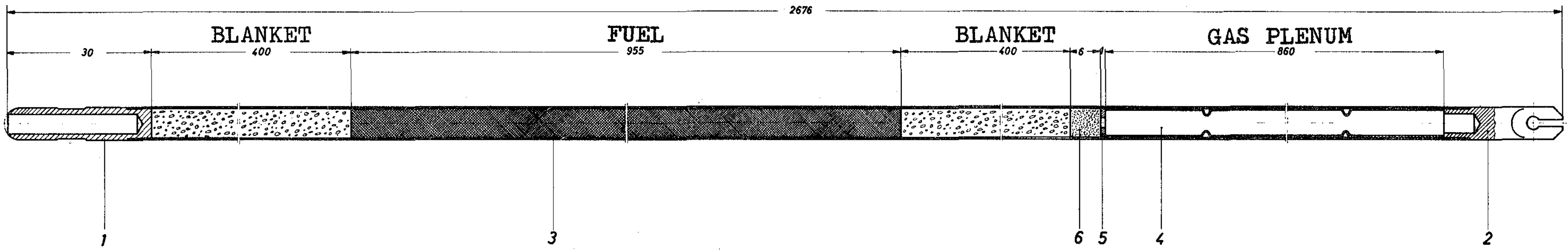


Fig. 2 PIN DESIGN FOR TEST PRODUCTION

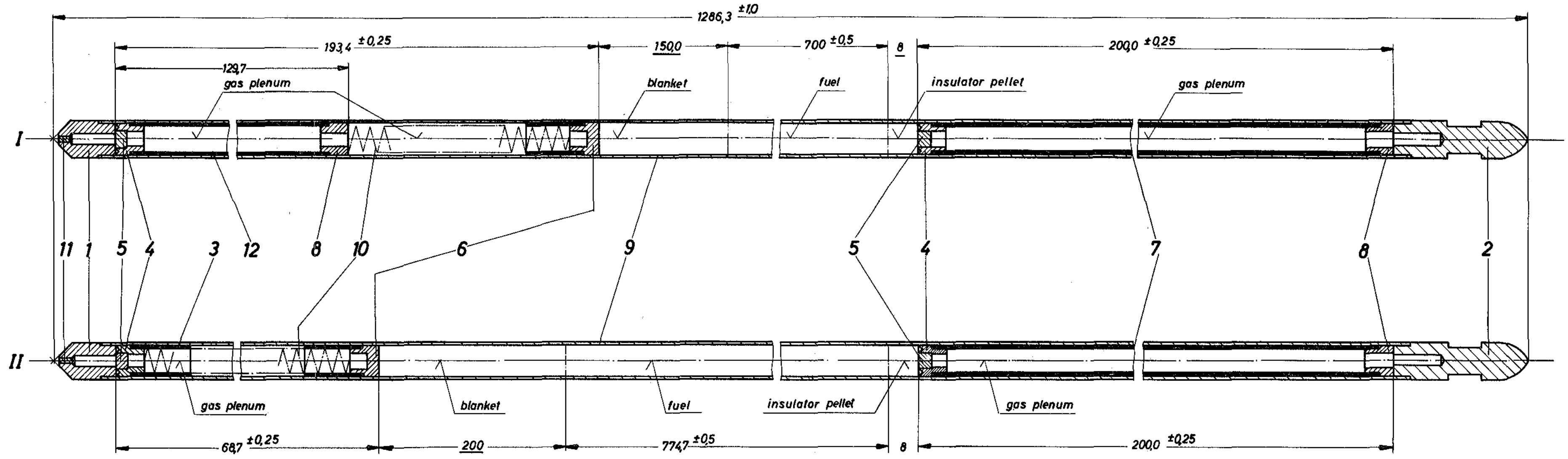


Fig. 3 PIN FOR EFFBR IRRADIATION (TYPE I AND II)

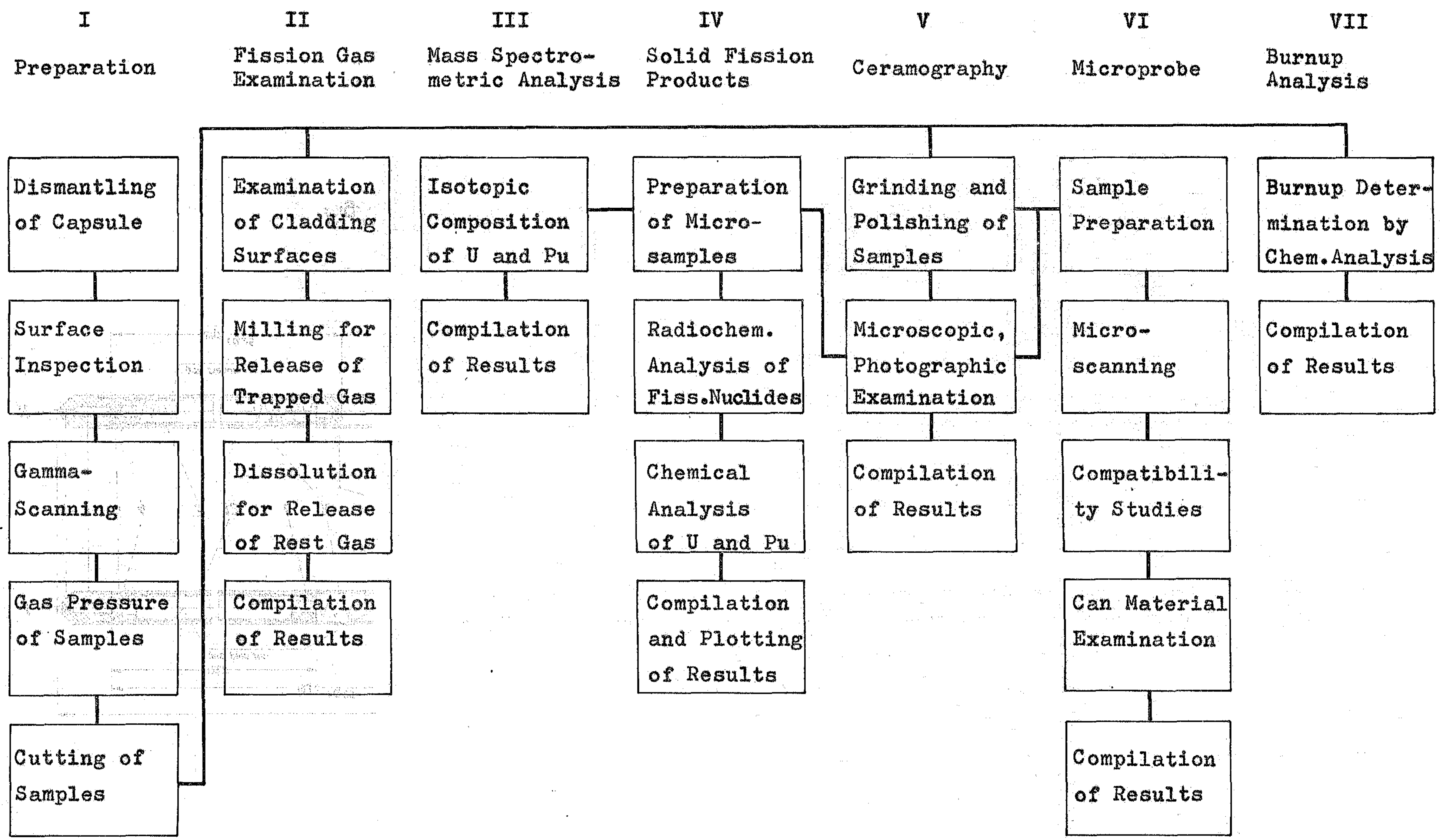
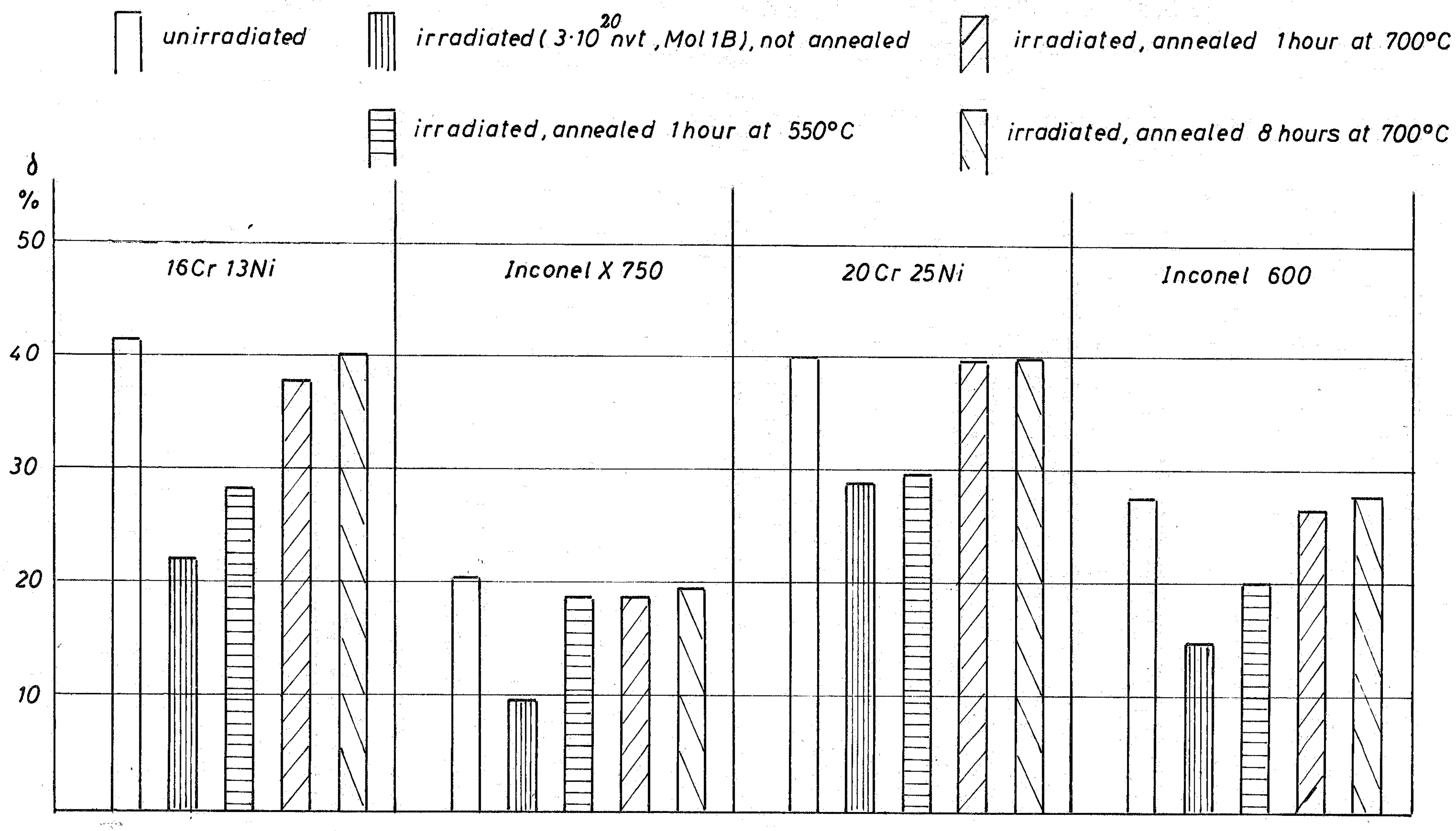


Fig. 4 Scheme for Post Irradiation Examination



LOW TEMPERATURE EMBRITTLEMENT: TOTAL ELONGATION AT ROOM TEMPERATURE

Fig. 5

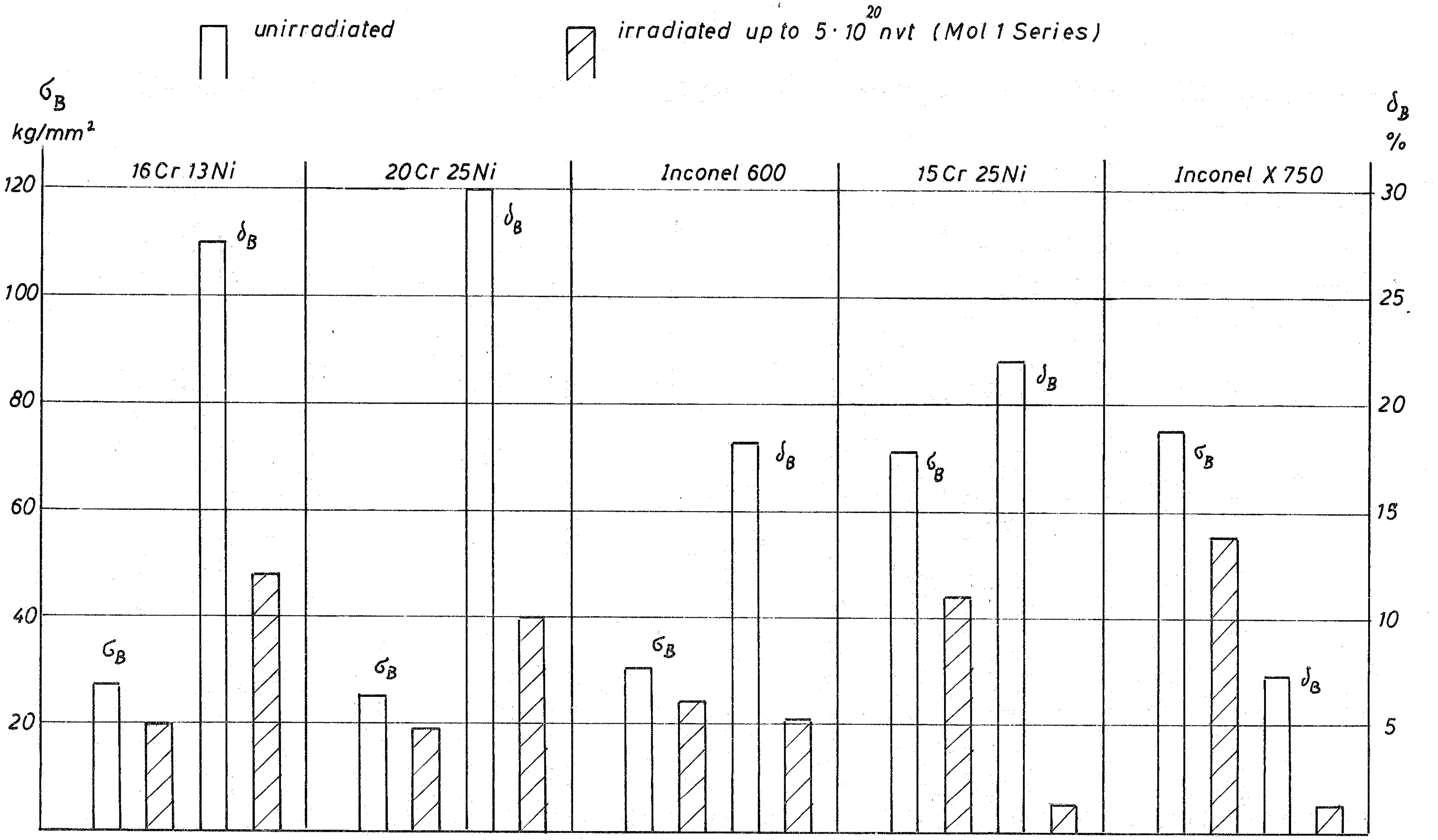


Fig. 6 HIGH TEMPERATURE EMBRITTLEMENT :
TENSILE STRENGTH AND TOTAL ELONGATION AT 750° C