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On the Development of Fast Breeders

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K E R N F O R S C H U N G S Z E N T R U M K A R L S R U H E

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*Chairman paper, session on fast breeders, The American Nuclear Society International meeting, Washington, November 1968.

ON THE DEVELOPMENT OF FAST BREEDERS*

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I. Introduction

The principle of breeding is as old as the development of nuclear reactors. Fermi and Zinn have had their first design work for a fast reactor as early as 1944 [1]. From that time on until the years around 1960 in the United States, Great Britain and Russia fast reactors have been developed which we now classify to be the first generation of fast breeders (see table 1) [2] - [8]. EBR I, EBR II, EFFBR, the DFR and BR 5 are the more prominent reactors of this first generation. The principal fuel there was metal, the sodium temperatures were modest, the cores small, the breeding external and all the attention went to the doubling time [9], [10]. Economical considerations however shifted this attention from doubling time to the fuel cycle [11]. It became apparent that a high burn up is a condition sine qua non for a sound fuel cycle and economical attractiveness. The paper of Sampson and Luebke [12] opened the era of the second generation of fast breeders in evaluating the potential and features of fast breeders with PuO_2/UO_2 as fuel. Following that it were particularly the groups of GE [13] and Karlsruhe [14] that were actively going into this new field. After the Vienna Conference of 1961 [15] this ceramic fast reactor scheme received world wide attention. At first it was the Doppler coefficient that

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was in the center of interest [16], [17]. This subject led to major undertakings, among others the SEFOR project of SAEA, AEC, GE and Karlsruhe together with EURATOM [18], [19]. But shortly thereafter most of the attention went to the Na void coefficient after having recognized, that large ceramic reactors might have a positive, partial Na void effect [20]. Following the somewhat too general idea that all power coefficients should be negative, a number of large design studies were initiated in the U.S. [17], [21]. In order to do this another important parameter had to be fixed: the target size of these reactors. That in turn required a first assessment of a possible time scale for development. During 1963/64 world wide understanding was obtained that fast reactors should and could be economically available by the end of the 70's. That in turn made it clear, that 1000 mWe is the target size. So the above mentioned 4 major studies in the U.S., namely that of GE, Westinghouse, B+W and Combustion Eng. were conducted. They all showed distorted core configurations in order to depress the Na void effect. This sometimes hurts the breeding potential very much, but emphasis had shifted from breeding to economy [22]. Also in Great Britain, France and Germany such large design studies were conducted, however they all did not show distorted cores. At the Argonne Conference of 1965 [23] the reactors of the existing studies were analyzed as a system and it became apparent, that the Na void comes into action only if a major accident of an incredible nature takes place [24]. The chain of events that have to happen for such an accident were analyzed and in so doing the two phase Na flow phenomena together with the capability of subassembly boxes to withstand pressure peaks became the center of interest [23], [25], [26], [27]. But that meant the departure of a so far observed route of development: the reactor design should be placed on a consistent and conceivable chain of accidental events, forming the design basis accident; the ad hoc postulated maximum (and incredible) accident was therefore at that time no longer the design guidance [23]. Most reactor groups of the world decided then, to have an early prototype reactor of something like 200 - 300 MWe. This was the case for the GE, Westinghouse and the AI group, for France and for Germany. The British and Russians were obviously already

ahead, they were already actively engaged in their PFR and respectively BN 350 prototype reactor designs. The basic idea was and still is to have a round of fast breeder prototype reactors that can be compared to the Yankee, Dresden, Indian Point round of thermal prototype reactors in the 50's. The function of such prototype reactors would be the demonstration of technical feasibility on a commercially significant scale, somewhat in combination with providing a test bed for commercially developed better fuel and reactor components. More recently the U.S. groups call these prototype reactors therefore demonstration plants. The European countries continue to follow that line whereas the USAEC shows concern about the necessity of excessive testing of reactor components including fuel and instrumentation and therefore is considering a more stretched out timing. This is deeply interrelated to the question of safety and the operational availability of Na cooled reactors and the question which way to go has been heavily debated ever since.

But there is no debate that an early prototype requires an intermediate step, more general experience has to be collected before such undertakings take place. In the U.S. this point is well taken care of by the EBR II, EFFBR and more specifically and recently by the SEFOR reactor. In Great Britain this intermediate step is DFR, in France the recently and so successfully started RAPSODIE reactor fullfills this requirement. Germany is somewhat late, but the KNK reactor of Interatom which is in its final construction stages at Karlsruhe has engineering wise this function, its first core is a thermal but highly compact core, and the second core will be a fast core, leading to the fast, 20 MWe, KNK II reactor by 1971. The Japanese started recently a larger fast reactor program and, remarkably enough, among their first project goals is to have by 1972 a 100 MWth fast reactor which in a way lines up with EBR II, DFR and the above mentioned reactors. This line of experimental reactors gives particularly overall engineering experience.

II. Fast reactor physics

One of the main reasons for distinguishing between the first and the second fast breeder generation is their physics. As this session does not allow for a special paper on fast reactor physics, some more detailed remarks on that field may be reasonable. Fast breeder reactors of the first generation have metallic fuel and were small. Therefore they have a hard spectrum the lower end of which does not really touch the resonance region. The important contributions to the power coefficient come from fuel thermal expansion. The second generation of fast breeders has a soft spectrum (see fig. 1), which definitely covers the resonance region. Therefore the physics of such reactors have to deal in detail with that resonance region and is therefore the most difficult part of all reactor physics. The first effect to be explored was the Doppler coefficient. The calculation of it was first done by Goertzel [28], but the investigation was really taken up by Nicholson [29], still for a somewhat hard spectrum, but later extended and refined by the Argonne, GE and Karlsruhe groups [30], [31], [32] and others. Today one can say that the calculation techniques are established for that [33], [34]. To measure the Doppler coefficient two principally different approaches are possible: One approach is to measure the Doppler coefficient in action when it terminates a power excursion. According to basic ideas of GE and Karlsruhe the already mentioned SEFOR project is designed and executed along these lines [18], [19]. The other approach is to measure the Doppler effect in a fast zero power critical facility by sample heating. For some time the proper interpretation of that created some difficulties, but particularly after a contribution of Storrer [35], but also others, this problem was solved. Today there exists a great variety of Doppler measurements in criticals with satisfactory accuracies. The expected SEFOR results will enlarge the domain of Doppler measurements into higher temperatures and operational conditions thus leading to a generally satisfactory situation [36].

The investigation of the Na void coefficient was both experimental and theoretical and turned out to be difficult in both areas. The principal tool

of exploring the Na void coefficient is the fast, zero power, critical facility. Practically all existing such facilities make use of platelets, mostly 2 inch × 2 inch, a typical thickness being 1/8 of an inch. In the portion of the spectrum, where $\frac{d\phi^+}{dE}$ governs the physics of this effect, there are large heterogeneities which influence the absolute size of this effect and large computer programs on the basis of a fairly detailed and cumbersome theoretical analysis is necessary to extrapolate from measured values to values which are significant for a fast power reactor design. But it seems that today this technique is at hand [37], [38]. The theoretical prediction of the Na void coefficient has led into the more refined schemes of calculating large fast reactors having the soft spectrum of the second generation. Particularly for calculating the Na void effect it is necessary to have a two dimensional code which gives the proper weighing spectrum for group constants in the various parts of the core and the blanket. Then a very careful preparation of the group constants is required. It turns out that it is not so much the calculational procedure that creates uncertainties, rather it is the sensitive dependence on microscopic input data and the process of preparing group constants that is of strong influence. Departing from the old idea of having universal sets of group constants [39], [40] there esist today extensive and very detailed programs for preparing a set of group constants for each particular reactor case, that is the MC^2 program of ANL [41], the Galaxy program of UKAEA [42] and the Migros program of Karlsruhe [43]. These programs take into account the self shielding effect of the prevailing potential scattering cross section of the core, the temperature broadening of the fission and absorption resonances, the influence of scattering resonances and the influence of weighing spectra . In order to make all that possible a tremendous input of microscopic cross section data is needed. Due to the activities of groups like EANDC [44] and evaluating and judging compilations of groups like that of Brookhaven [45] and Karlsruhe [46] the knowledge and availability of these data has improved considerably during the past few years. The most striking event in the area of microscopic input data was the Schomberg measurement [47] of the capture to fission α ratio of Pu 239. Schomberg's data would have hurt the breeding ratio by 7 - 9 points and therefore the fuel cycle economy by as much as

0.1 $\frac{\text{mills}}{\text{kWh}}$. It would have further increased the Na void coefficient by 40 %, which is due to the sharper raise of ϕ^+ in the respective spectrum areas. Theoretical indications by Pitterle and Barre [48], [49], together with recent measurements of Gwin seem to indicate that the damage is only half as large. This has been confirmed more recently also by a Doppler experiment [50]. Other remarkable trends in this area are the downward trend of the U 235 (n, f) standards and particularly the downward trend of U 238 (n, γ) data [51], [52], [53]. (See fig. 2 - 5)

A remark should be made on the interconnection of extended reactor physics calculations and computer capabilities. The Na void calculations in particular and the calculation of large fast second generation breeders in general require as of today three dimensional calculations, in the case of the Na void effect for instance two space dimensions and one energy dimension. This just fits the calculational capability of todays computers, say IBM 360/65 (or better 360/91) or CDC 6600. Some years ago all reactor problems were treated only in two dimensions (1 space, 1 energy dimension or 1 space, 1 time dimension). It is probable that, as the art develops, four dimensions can and have to be handled and this requires the next generation of big computers. It should be realized how strong this interlink is (see table 2). By and large the reactor theory situation is satisfactory within certain limits. Further improvements can be expected as microscopic input data become better, computer capabilities increase and results of large critical Pu experiments become available. Here also the situation has improved significantly as large Pu facilities become operativ. The first was ZEBRA in Great Britain (1962) then it was Masurca in France and SNEAK in Germany. They became critical in the same night (15.12.66), and the ZPPR in the U.S. that goes into operation about now. It is almost certain, that with the large Pu experiments to come the art of reactor physics will improve further and significantly. Among other things it will be particularly the verification of group constant sets for certain classes of reactors. Last not least attention should be focused on the Dutch STEK (KRITO) experiment, which is designed to measure the reactivity influence of true fission product samples [54]. This is important, because both differential and integral data on

fission products are missing and of increasing importance as longer burn up's become realistic. A list of existing zero energies facilities is attached to this paper. (See table 3)

III. The fuel element

A burn up in the order of 100 000 $\frac{MWd}{to}$ is mandatory for fast reactor fuel elements in order to burn something like 100 % of the original fissionable content, a condition to make the fuel cycle economic [22]. With metal fuel elements being only capable for burn up's of the order of 10 000 $\frac{MWd}{to}$ a principle switch to another concept typical for the second generation of fast breeders was necessary. UO_2/PuO_2 fuel is such another fuel concept provided it has a design that allows for high fission gas release. Low density and high fuel operating temperatures are the principal steps to achieve that [13], [55]. This is different from a UO_2/PuO_2, or only UO_2, fuel design for a thermal reactor where high densities and low fission gas release are required. The intended high fission gas release of fast reactor fuel requires a plenum above or below the axial blanket for collecting the released fission gases. This in turn requires the fission gases to travel over long axial distances. The cladding of such a fast reactor fuel pin must be designed to be free standing and for being strong enough to withstand the eventually large internal pressure which might be as high as 100 atm. It is very remarkable that today all fast reactor groups are virtually in agreement on the specifications of such a fast reactor fuel pin. They are roughly as follows: 450 $\frac{W}{cm}$ maximum linear rod power, a pin diameter of 6 mm being consistent with a fuel rating of 1 $\frac{MWth}{kg\ fiss}$, an active core length of 100 cm, the cladding a 16/13 stainless steel of 0.35 mm wall thickness and as the most salient point: a smear density in the neighborhood of 80 %. In the past some debate was going on whether to provide for axial restraining of the fuel or not. The APDA group always favoured this concept of axial restraining in order to avoid any reactivity movement whatsoever, but many other groups don't count on, but allow for axial fuel creeping and/or expansion. There is strong indication that fuel under high neutron fluxes and

high temperatures is creeping much more readily and easily than under the same mechanical and temperature conditions out of pile [56], thus enhancing the idea not to restrain the fuel axially and to avoid radial swelling therefore. In the EBR II the present loading with test oxide pins of a type as described above is 105 pins, formerly a similar number has passed the reactor after having reached a burn up of 6 % of heavy atoms [57]. According to published data [58] a somewhat smaller, but still significant number of such oxides pins of the British project has passed the DFR having reached even higher burn up's. At present there is a Franco - German - Benelux subassembly of 77 pins in the DFR aiming for 50 000 MWd/to and in parallel, 7 + 19 pins are, or respectively will be, irradiated in the Belgian reactor BR 2 under adjusted fast flux conditions. This is at present the basis of fuel experience for the above mentioned round of 300 MWe fast oxide breeder prototypes. This basis is kind of narrow, it is sufficient, but not satisfactory. A broader fuel test experience is among the missing and most urgent developmental requirements.

The cladding material considered for the fuel of these early prototypes is in all cases stainless steel, a 16/13 CrNi version. It seems to be impossible to pass with the hot spot, midwall, max. cladding temperature the 700 °C mark. This limits as a main factor at present the sodium outlet temperature. Extended irradiation tests with cladding material specimens have been executed by practically all fast reactor groups. In the early years one was only concerned about displacement damages. The on that time completely new phenomenon of (n, α) reaction with the related He bubble formation changed the picture and the phenomenon of high temperature embrittlement came up; that was 1963 [59], [60]. More recently the stainless steel swelling due to void collapsing beyond doses of 10^{22} n/cm^2 and at high fast neutron fluxes came into the picture [61]. Today one finds the following situation: Doses of 10^{22} n/cm^2 have been obtained at three different places (DFR, EBR II, BR 2), doses beyond that and up to 10^{23} only at DFR, partly also at EBR II (see fig. 6). According to these results the applicable tangential creep rupture strengths under irradiation are 40 - 50 % lower than without irradiation,

the associated tangentials strains there are between 0.3 and 1.5 % (see fig. 7 [62]). The influence of the radiation dose on the ductility, that is the possible tangential strains is unclear. Particular concern existson the influence of multiaxial stress configuration to the picture of radiation damage to the ductility as compared with that of uniaxial stress. For present fuel pin designs one therefore has to design for creep strains as low as 0.5 - 1 %. The influence of the recently detected stainless steel swelling on the pin, subassembly and core design is still unclear. There exists single DFR data, that indicate a swelling as high as 10 % at doses of 10^{23} n/cm^2 and higher. This phenomenon does not only depend on the neutron dose but also on the neutron flux as it is the flux that creates a supersaturation of point defects, also the temperature influences the phenomenon. Figure 8 illustrates the present picture of this void collapsing [63]. The swelling of stainless steel,if confirmed, would be an awkward and not yet fully understood phenomenon. During the next one or two years a great deal of attention will probably go into it. In the light of the above remarks the following conclusion is of great importance: All existing fast neutron test reactors are of low flux. The reactors in operation have a flux as low as $\approx 2 \cdot 10^{15}$ n/cm^{2}sec, the EFFBR (200 MWth, core A) would be higher by a factor of 2.5, but it is not in operation. The required neutron flux of the 300 MWe prototype reactors is at $8 \cdot 10^{15}$, four times as high as the presently available neutron flux (see table 4). With respect to the fuel and the required burn up's during fuel pin performance tests it is possible to artificially enrich the UO_2 of the UO_2/PuO_2 mixture and therefore to compensate for the lower fluxes by higher over all microscopic fission cross sections. But with respect to the applicable cladding neutron doses no such trick is possible. The bad thing is, that the missing factor 4 might be decisive as one realizes that void collapsing appears to a reasonably extend only beyond 10^{22} n/cm^2, not to mention the dependence of this phenomenon on the flux level. One can further speculate that at even higher doses and fluxes still other phenomena might occur, - UFO-s, "unforseen flux obstructions". It seems to be possible however to stretch the existing possibilities for irradiation tests and to verify the present approach for having a 16/13 stain-

less steel as cladding material for the envisaged 300 MWe early prototypes. This will be particularly so, if the EFFBR could come quickly into operation again. But independent of building the early 300 MWe prototypes it is now very clear, that a fast neutron, high flux, materials test reactor is a necessity. This test reactor must have a flux of $1.5 \cdot 10^{16}$ n/cm^{2}sec or more in order to have a certain flux margin for the present day fuel and to allow therefore for shorter irradiation periods consistent with a considered dose rate. Looking at the possible time schedule of such a test reactor, one realizes that such a fast MTR cannot go into operation before, say 1975. On that time, not only the cladding doses and todays fuel pins have to be tested but also the then interesting high performance fuels, particularly the carbides. Up to now the irradiation experience with the carbides is not very encouraging, but there is indication that the hot fuel concept with linear rod powers of such a carbide pin of 1250 - 1500 $\frac{W}{cm}$ will lead to acceptable swelling rates. The higher breeding ratio of a carbide fueled core is well known. But it is important, that besides doubling time, also the fuel rating alone is of great significance because it determines the first core requirements. Up to the year 2000 the first core requirements of fast breeders will largely influence the over all natural uranium consumption of a compound converter|breeder economy [64]-[65]. Also the far reaching question of the required and properly chosen capacity for isotope separation plants is interconnected to that. Therefore high performance fuel with a rating of at least 2 $\frac{MWth}{kg\ fiss}$ will come undoubtedly into the picture, thus leading to flux levels of $1.5 \cdot 10^{16}$ n/cm^{2}sec. A modern fast neutron test reactor should not have a flux lower than that.

IV. Fast reactor safety

Fast reactor safety has been a subject for exploration since the beginning. Originally it was the short neutron lifetime that caused concern. But it is now clear that the short neutron lifetime is an advantage provided the instantaneous power coeficient is negative. In that case the first power

peak is terminated within a short time scale. One recalls that this time scale is given by $\sqrt{\frac{\ell}{a}}$, if ℓ is the neutron lifetime and a the ramp rate. Also the inserted energy in that peak is smaller if the neutron lifetime is smaller as this energy under the first peak is proportional to $\sqrt{\ell \cdot a}$ [66]. The second concern for fast reactor safety stemmed from the EBR I melt down accident [67]. A partial but instantaneous power coefficient, the bowing effect of fuel elements due to thermal gradients, has been positive there. From that time on attention was focussing on the power coefficients. The phenomenon of the bowing effect became transparent [68], but other coefficients came in as the second generation of fast breeders started. As mentioned before it was firstly the Doppler coefficient. The above mentioned measurements in fast criticals, the elaborate calculations of that effect and the SEFOR program to come clarify the size and sign of that Doppler coefficient. The role of the Doppler effect in an accidental sequence of events has not always been entirely clear. It is two fold: First, it terminates the first power peak of a fast excursion and decreases the energy that can be pumped into it and therefore gives time to the shut off system to react. By the same token it helps to establish inherent operational stability. Second, the Doppler coefficient influences also strongly the energy release figures of a Bethe Tait calculation and makes these Bethe Tait results less sensitive against the data of the equation of state of the involved reactor fuel [69], [70]. It were the years between 1962 and 65 when this became fully apparent. The discussion of the partially positive Na void coefficient started 1963 and is still going on so far as its role in an accidental sequence of events in concerned. As mentioned before, such voiding of the inner core zones which have positive Na void contributions really has to take place. For that, either initiating reactivity ramp rates are necessary which can happen only if there is a complete failure of the shut off system. It is debatable whether this is a reasonable assumption. Or, one has to experience the blockage of a subassembly from the coolant flow. This can lead to sodium ejection from this one subassembly. If the sodium ejection is preceeded by superheat of the fluid, concern exists that this ejection is so violent that propagation of this subassembly failure to other subassemblies takes place [25]. In these cases and only in these

cases the Na void coefficient is of significance [24]. Depressing the Na void coefficient by distorting the cores, e.g. to make them extremely flat to enhance leakage, hurts the breeding capability considerably and the question comes up whether it is reasonable to depress breeding through all the operation of a fast breeder because of accidents that are considered unrealistic. The situation is further complicated because the Bethe Tait codes which predict the accidental energy release are essentially codes which start from a homogeneous core model. But the ejection of Na from cooling channels which acts as kind of a ramp rate multiplication (or initiation, respectively) makes reference to the pin - cooling channel geometry and is therefore a strong feature of heterogeneity. Due to this circumstance it is difficult to predict with real confidence the energy release of a Bethe Tait event. Finally one has to realize that the after melt down decay heat of large 1000 MWe reactors is very large (10 - 100 MWth) and one has to take it away from a configuration that has experienced such a hypothetical accident. This requires active engineered safeguard measures. If one goes into details, one realizes that active and therefore engineered safeguard measures are indeed unavoidable. If, now, that is so, one should concentrate on avoiding Na ejection by engineered safeguard measures. These are among others the following: Instrumentation of each subassembly in the core, diversification of the rods of the shut off system, a second and completely independent and different shut off system, avoiding superheat by properly designed cooling channels and pin surfaces, avoiding damage propagation from subassembly to subassembly by a proper design. I refrain from elaborating a consistent scheme for such safety measures because that would take too much time and would be outside the scope of this chairman paper. But I make the point that the present trend is to let aside the Na void effect because one has to rely on engineered safeguards anyway. More generally one can conclude from that, that the new art of quality control, reliability control and the probabilistic assessment of failure in a network of possible events [71] has to be built up much more than it has been the case in the past, although it is clear that in so doing one has to establish a certain equilibrium with the previous way of safety evaluations. I am personally convinced that reactor designers can learn a lot here from missile designers.

V. Heavy Na components

Mastering the Na technology requires first of all large and reliable sodium pumps. Present Na reactors have pumps up to 3000 $\frac{m^3}{h}$ (≈ 12 000 gpm). The mechanical pump type is now prevailing, EM pumps are used today only for special purposes. Most 300 MWe prototype reactors will have pump sizes of 5000 $\frac{m^3}{h}$ (≈ 20 000 gpm), but present studies for pumps go far higher. There is general agreement that extended pump tests are required.

Besides of Na pumps it is specifically the steam generator that requires special attention. Large sodium component test rigs have to be brought up, to develop and test these engineering components. The large 35 MWth sodium component test installation at Santa Susanna came into operation 1965, in the U.K. the test rig for Na pumps came into operation 1964/65 and these rigs provided the first significantly large test experience. Now larger test rigs are under construction in the U.S., in Russia, in France, in the Netherlands and in Germany. It is with high confidence that one can expect the test results which are necessary for the commitment on the construction of the 300 MWe prototypes. The attached table 5 lists the more important and prototype oriented test rigs. All the existing fast breeder designs provide a primary and a secondary Na circuit and as a third circuit the steam generating turbine circuit. Sometimes it was debated to let away the intermediate circuit because of the related capital cost burden. It is only recently that the Belgonucleaire of Belgium proposed a CO_2 gas turbine circuit as a second circuit thus eleminating the intermediate Na circuit. [72] This scheme is feasible because CO_2 gas turbines require not so high temperatures for a considered thermal efficiency, it is more the high gas pressure that is required. And this scheme promises lower capital costs because of the elimination of the intermediate Na circuit and the much smaller size of a CO_2 gas turbine if compared with the corresponding steam turbine. In the framework of the German-Benelux fast breeder project a study is going on to see whether such a CO_2 circuit of limited size shall be added to their prototype.

VI. Other coolants

Fast reactors of the second generation are large and therefore of low enrichment, they make use of Pu instead of U which further lowers the fraction of fissile atoms in the fuel and they make use of UO_2/PuO_2 (or other ceramics) as fuel and such fuel has only half the density of the former metallic fuel. This gives cooling densities of only 300 - 500 $\frac{\text{kWth}}{\text{liter core}}$ and such cooling densities allow for other coolants than Na. This was discussed at the Vienna Conference of 1961. [14] Then Helium [73], [74] and steam [75], [76], [77], [78] were considered as coolants. Detailed studies of Karlsruhe and GE and more recently an ENEA study for comparing Na, steam and He as coolants have been executed. It is now a fairly well established fact, that steam cooling has the same economical potential as the Na, oxide fueled fast breeder which is now under development at many places. However, there is one heavy set back: The high external coolant pressure of steam requires the fuel element scheme of a partially collapsed cladding using either Incoloy 800 or the 12RX72 swedish steel (or a similar material). For satisfactorily testing out such a fuel element with collapsed cladding, dry steam and high external pressure, a fuel test bed is required, or in other words, a small experimental reactor, that plays for steam the role that has been played by EBR II, DFR, RAPSODIE and KNK for Na. The GE proposal for an experimental steam cooled fast ceramic reactor (ESCR) was exactly this kind of a reactor. In Germany the intended remodelling of the HDR reactor (Heißdampf Reactor of AEG at Großwelzheim, Germany) into a coupled fast thermal STR reactor was meant to have the same function. Both projects did not materialize for financial reasons, but in addition to these financial problems there was the other problem, namely that of timing. An ESCR or STR would have become operative not earlier or not much earlier, than the 300 MWe Na prototypes. Steam cooled fast breeders would be late therefore instead of being earlier than Na cooled fast breeders. The breeding potential of steam cooled breeders is limited and there is no foreseeable potential to go to a high performance version as it is the case with the carbide fuel for Na cooled reactors. These features are not detrimental if the steam cooled breeder comes early. They become real concerns however if

the timing of steam cooled breeders is delayed. Technically the gas cooled breeder is in the same position. Also there high external coolant gas pressures are applied to the fuel and undoubtedly a large scale fuel development with the associated tests either in a fast neutron material test reactor or in a EBR II, ESCR type reactor with gas as coolant is mandatory. This necessarily results in a different time scale, it seems to be impossible that gas cooled fast breeders could be commercially available before 1985. But the gas cooled, fast breeder has two features that makes it even with such a time scale attractive: the potential for high gain breeding, the potential for very high temperatures with the associated direct gas turbine cycle and, related to that, the potential for very low energy production costs coming from both the low capital costs and the low fuel cycle costs. This makes this reactor scheme attractive in spite of the fact that this reactor belongs to the third and not to the second generation of fast breeders.

VII. The present fast breeder reactor projects

At present there are the following Na fast breeder projects:

a) In the USSR there is the BN-350 prototype reactor at the Kaspian sea in the advanced stages of construction. This reactor is designed for 150 MWe and for 200 MWe equivalent for sea water desalination. The reactor is expected to become ready by 1969.

b) In Great Britain there is the PFR, a 250 MWe fast breeder prototype reactor, which is supposed to become ready by 1971.

c) In France there is the PHENIX reactor, a 250 MWe fast breeder prototype reactor, which shall be ready by 1973.

d) In the U.S. a study is going on by GE together with the ESADA group on a 310 MWe prototype plant, this reactor might go into operation by 1975. Westinghouse is conducting a similar study, also with utili-

ties as partners, the contemplated size is 212 MWe and the reactor might go into operation also by 1975.

Atomics International considers together with the GPU group a 500 MWe plant, the time scale is similar to that of GE and Westinghouse.

e) Germany together with Belgium, the Netherlands and Luxemburg is designing a 300 MWe SNR prototype plant. The construction shall start 1970, 1974 is the date of completion, Germanys share is 70 %, that of Belgium and the Netherlands 15 % each.

f) Japan. A 200 - 300 MWe prototype is envisaged. It is contemplated to have its start up around 1976.

Besides of these prototype reactor projects, the following projects are going on:

a) The SEFOR project. SEFOR will have its start up in a few month from now, it is a 20 MWth experimental reactor of SAEA, AEC, GE and Karlsruhe together with EURATOM.

b) The FFTF project of the USAEC. This shall be a 400 MWth test reactor, the expected neutron flux is $7 \cdot 10^{15}$ n/cm^{2}sec, 5 big loops are envisaged.

c) Italy decided to build the PEC reactor, a 130 MWth test reactor the expected neutron flux is $3.8 \cdot 10^{15}$ n/cm^{2}sec, 3 test loops are provided.

d) Interatom of Germany will convert the KNK reactor into the KNK II reactor, a fast reactor with 20 MWe, 60 MWth. 1971 is expected to be the date for operating this KNK II reactor, the thermal KNK reactor will go into operation next year.

e) The BOR reactor of Russia. This is a test reactor with 60 MWth, which in a way is the extension of the BR 5 reactor line. Its date of start up is 1969.

f) A 100 MWth experimental fast reactor is under design in Japan. It is expected to go into operation by 1972.

A more elaborate table with much more data is attached to this paper (see tables 6 and 7) [7], [8], [79] - [85].

VIII. Concluding remarks

Fast breeders have a strong and short range economic incentive. This is particularly so bacause the present generation of thermal power reactors produce large amounts of Pu which can be used meaningfully only in fast breeders. And they have the long range potential of breeding and therefore really making use of the existing uranium resources and more than that, of the ever increasing vast amounts of depleted uranium [86], [87]. Beyond the year 2000 breeding is a necessity and one should bear in mind that there are only 30 years left, that is the life span of only one power station. Therefore there is no doubt that fast breeders are the ultimate solution of the problem of providing nuclear energy. The international fast reactor community is well under way to accomplish their tasks.

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Most of the references are examples for the literature of the referred subject. It is not the intention to give a complete literature survey, this would be outside the scope of this paper.

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Table 1 First Generation Fast Breeder Reactors

		USA				USSR			UK	France
		CLEMENTINE	EBR-I	EBR-II	EFFBR	BR-1	BR-2	BR-5	DFR	RAPSODIE
Reactor Power										
Thermal	MWth	0.025	1.2	62.5	200	0	0.1	5	72	20
Electrical	MWe	0	0.2	20	66	0	0	0	15	0
Core										
Fuel		Pu-Metal	U-Metal	U-Metal	U-Metal	Pu-Metal	Pu-Metal	PuO_2	U-Metal	PuO_2/UO_2
Core Volume	liters	2.5	6	65	420	1.7	1.7	17	120	54
Fuel Rating avg.	MWth/kg fiss	0.0016	0.02	0.3	0.37	0	0.008	0.1	0.24	0.14
Power Density avg.	MWth/liter	0.01	0.17	0.8	0.45	0	0.06	0.3	0.5	0.32
Linear Rod Power max.	W/cm	(av.50)	300	450	250	0	150	200	(av.320)	(av.210)
Neutron flux max.	n/cm^2sec	(av.$5\cdot10^{12}$)	$1.1\cdot10^{14}$	$3.7\cdot10^{15}$	$4.7\cdot10^{15}$	$5\cdot10^{10}$	$1\cdot10^{14}$	$1\cdot10^{15}$	$2.5\cdot10^{15}$	$1.8\cdot10^{15}$
Primary Heat Transfer System										
Coolant		Hg	NaK	Na	Na	-	Hg	Na	NaK	Na
Coolant Temperature										
Core Inlet	°C	40	230	370	290	-	30	375	200	410(450)
Core Outlet	°C	120	320	470	430	-	60	450(500)	350	500(540)
Coolant Mass Flow	m^3/h	0.6	80	2200	5500	-	6	240	1800	800
Number of Coolant Loops		1	1	2	3	-	1	2	24	2
Time Schedule										
Design		1945	1945					1956		1958
Construction		9/1946	1949	1957	8/1956			1957	3/1955	1962
First Criticality		11/1946	8/1951	10/1961	8/1963			6/1958	11/1959	1/1967
Full Operation		3/1949	12/1951	4/1965	8/1966	1955	1956	7/1959	7/1963	3/1967
Shutdown		6/1953	1963	-	-	1956	1957	-	-	-

To Table 1 First Generation Fast Breeder Reactors

	USA				USSR			UK	France
	CLEMEN-TINE	EBR-I	EBR-II	EFFBR	BR-1	BR-2	BR-5	DFR	RAPSODIE
Remarks	1. fast reactor, 1. Pu-fueled reactor	1.nuclear electricity generation Pu-Core since 1962	Reactor Plant with integral fuel processing facility	Since 10/1966 out of operation			UC-Core since 1965		(Rapsodie is not really a reactor of the first generation, it belongs to a large extent to the second generation)

Table 2 Fast Reactor Calculations and Computer Capabilities

Generation	Year	Typical Computer	Maximum Number of Dimensions for Parameter Studies	Maximum Number of Dimensions for Single Calculations
1	1954-1960	Univac I and II IBM 650	1 (e.g. zerodimensional multigroup calculations)	2 (one-dimensional multigroup diffusion calculations)
2	1960-1965	IBM 7090 Univac 1107	2 (1-D diffusion calculations)	3 (1-D burn up calculations, 2-D diffusion calculations)
3	1965-1968 from 1968 on	CDC 6600 IBM 360/91 IBM 360/85 CDC 7600	3 (1-D burn up calculations, 2-D diffusion calculations)	4 (3-D diffusion calculations)

Table 3 Fast Critical Assemblies

	Location	year first critical	Short description	Fissile Material	Typical Core Size, liters (for average reflector thickness)
ZPR-III	Argonne, Idaho	1955	horizontal split-table machine	U 235, Pu 239 (600 kg)	600
ECEL	Atomics International, California	1960	horizontal split-table; thermal driver	U 235, 25 kg of U 233 were used in some assemblies	100 (test zone)
VERA	Aldermaston, UK	1961	vertical, split-table	U 235, Pu 239 (40 kg)	400
BFS	Obninsk, USSR	1961	vertical, fixed	U 235	1800
ZEBRA	Winfrith, UK	1962	vertical, fixed	U 235, Pu 239 (400 kg)	3000
ZPR-VI	Argonne, Illinois	1963	horizontal, split-table	U 235	3000
ZPR-IX	Argonne, Illinois	1964	horizontal, split-table	U 235	3000
FRO	Studsvik, Sweden	1964	vertical, split-table	U 235	65
MASURCA	Cadarache, France	1966	vertical, fixed	U 235, Pu 239 (200 kg)	3000
SNEAK	Karlsruhe, F.R.of Germany	1966	vertical, fixed	U 235, Pu 239 (200 kg)	3000
FCA	Tokai-mura, Japan	1967	horizontal, split-table	U 235, Pu 239 planned	3000
ZPPR	Argonne, Idaho	1968	horizontal, split-table	U 235, about 3000 kg of Pu 239 planned	3000

Table 4 Neutron Flux of Fast Reactors

	EXISTING REACTORS				PROJECTED		PROTOTYPES	
	EFFBR CORE A	EBR II	DFR	RAPSODIE	RAPSODIE FORTISSIMO	EFFBR OXID-CORE	PHENIX PFR	GFK-Na2
Reactor-Power Mwth	200	42	60	22	40	430	530	730
Total-Neutron Flux $\overline{/}$ x 10^{15} n/cm^{2}sec$\overline{/}$	4.73	2.07	2.5	1.8	3.0	8.2	7.0	8.7
Neutron Flux above 0.11 Mev $\overline{/}$ x 10^{15} n/cm^{2}sec	3.51	1.77	2.22	1.6		5.0	4.1	4.8

Table 5 Heavy Sodium Component Test Facilities

	Facility	Purpose	Technical Data	Time Schedule
USA	35 MW Sodium Component Test Installation-SCTI	Testing of different steam generators and intermediate heat exchanger	Na-Na-Steam System Na:max. 650°C (700°C) Steam: 560°C/170 at	1965 Preoperational Test 1966 Operation
	Sodium Pump Test Facility - SPTF	Testing of pumps	Pump capacity up to 32000 m^3/h, Temp. max. 650°C	1969 Construction 1971 Operation
USSR	3 MW Sodium Test Loop	Investigation of steam generator and intermediate heat exchanger models		1960 Operation
	Sodium Pump Test Facility	Testing of BN-350 pumps		1966 Construction
UK	Sodium Pump Test Facility	Testing of sodium pumps	Pump Capacity 1620 m^3/h	1964/65 Operation
France	5 MW Grand Quevilly	Investigation of steam generator and intermediate heat exchanger models	Na-NaK-Steam System Na: max. 600°C (625°C) Steam: 545°C(565°C)/130at	1964 Operation
	50 MW EdF Test Facility	Testing of steam generator	Na-Steam System Na: max. 650°C	1967 Construction 1969 Operation
Germ. Benelux	5 MW INTERATOM Test Facility	Investigation of special aspects of steam generators	Na-Na-Steam System Na: max. 560°C Steam: 500-540°C/200 at	1963 Construction 1965 Operation for KNK 1969 Operation for SNR
	INTERATOM Sodium Pump Test Facility	Testing of pumps	Pump Capacity 5000 m^3/h (15000 m^3/h)	1967 Construction 1969 Operation
	50 MW NERATOOM Sodium Component Test Facility	Testing of 50 MW steam generator 70 MW intermediate heat exchanger	Na-Na-Steam System Na: max. 650°C Steam: 600°C/215 at	1968 Construction 1970 Operation
Japan	2 MW Sodium Test Facility	Investigation of various characteristics of Sodium components	Na-System, max. 650°C	1969 Operation

Table 6 Second Generation Fast Breeder Reactors

	USA			USSR	UK	France	Germany
	GE	Westinghouse	AI	BN-350	PFR	PHENIX	SNR
Reactor Power							
Thermal MWth	750	540	1250	1000	600(670)	600	730
Electrical MWe	310	212	500	350	250(275)	250	300
Core (Reference)							
Fuel	PuO_2/UO_2	Carbide	PuO_2/UO_2	PuO_2/UO_2 or UO_2	PuO_2/UO_2	PuO_2/UO_2	PuO_2/UO_2
Core Volume	2000	1000	3000	1900	1320	1150	1750
Fuel Rating avg. MWth/kg fiss	0.82	0.9	1.0	0.96	0.7	0.8	0.76
Power Density avg. MWth/liter	0.31	0.5	0.37	0.5	0.4	0.5	0.37
Linear Rod Power max. W/cm	500	1000	500	470	450	430	420
Breeding Ratio	1.2	1.45	1.3	1.5	1.2	1.1-1.3	1.33
Burn up MWd/to	100000	100000	75000		70000	>50000	55000
Primary Heat Transfer System							
Type	Pool	Loop	Loop	Loop	Pool	Pool	Loop
Coolant	Na	Na	Na	Na	Na	Na	Na
Number of Coolant Loops	3	1	3	5	3	3	3
Pump Capacity m^3/h	5000	16000	12000	3200	5000	4800	4670
Coolant Temperature							
Core Inlet °C	425	410	425	300	400-420	405(420)	380
Core Outlet °C	590	540	570	500	560-585	565(590)	560

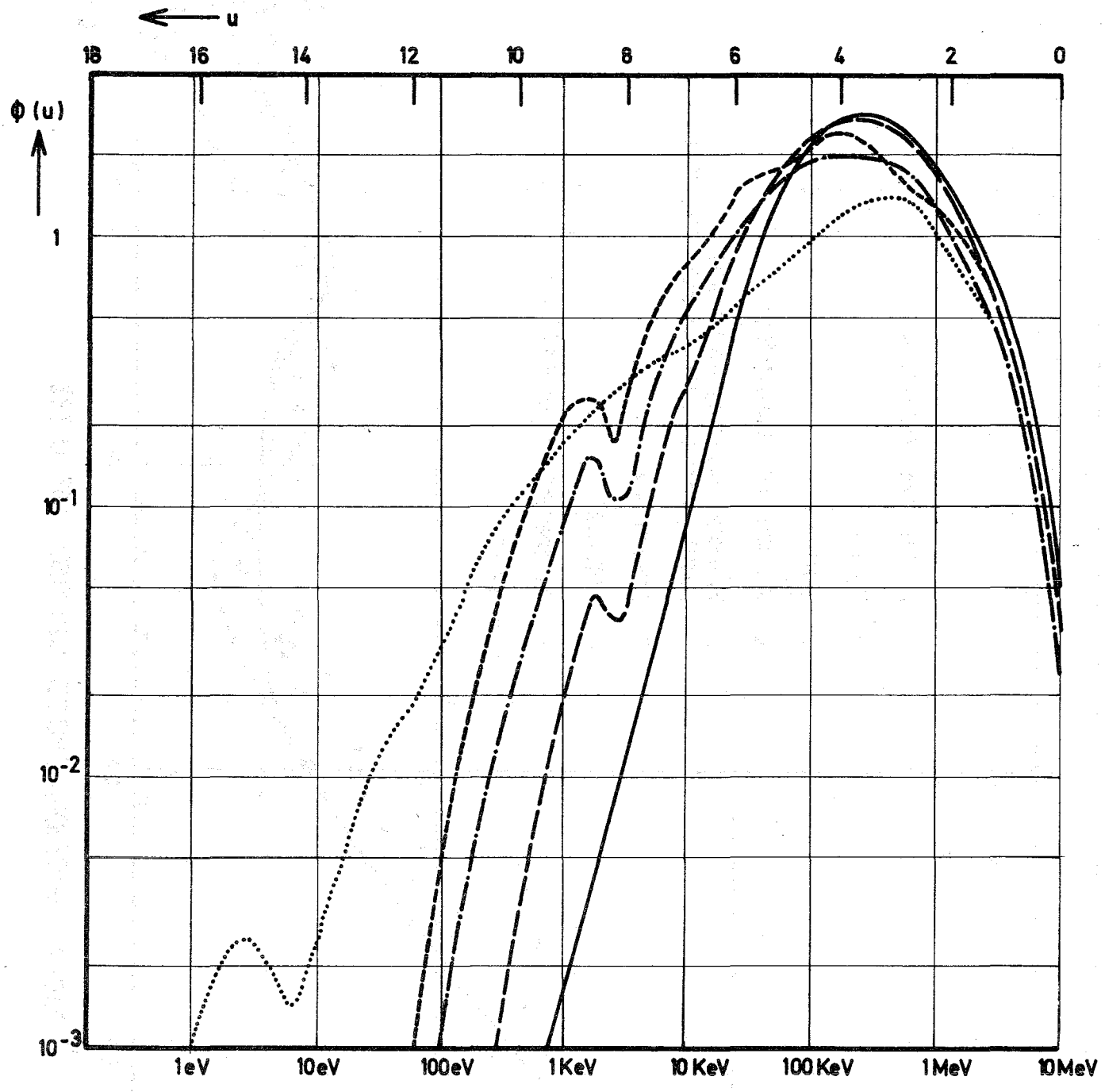
To Table 6 Second Generation Fast Breeder Reactors

	USA			USSR	UK	France	Germany
	GE	Westinghouse	AI	BN-350	PFR	PHENIX	SNR
Steam System							
Steam Generator							
Number		1	3	5	3	3	9
Typ		Pool	Modular			Modular	Pool
Materials Evaporator			Ferritic		Ferritic	Ferritic	Ferritic
Materials Superheater + Reheater			Austentic		Austentic	Austentic	Ferritic
Steam conditions							
Temperature °C	510	480	480	435	510-540	540	505
Pressure at	170	170	170	50	162	163	165
Date of Operation	1975	1975	1975	1969	1971	1973	1974/75

Table 7 Second Generation Experimental Fast Reactors

		USA		USSR	Germ.	Italy	Japan
		SEFOR	FFTF	BOR-60	KNK-II	PEC	JEFR
Reactor Power							
Thermal	MWth	20	400	60	58	130	92
Electrical	MWe	0	0	12	20	0	0
Core							
Fuel		PuO_2/UO_2	PuO_2/UO_2	PuO_2/UO_2 or UO_2		UO_2	PuO_2-UO_2
Core Volume	liter	500	950	53		420	240
Fuel Rating avg.	MWth/kg fiss	0.06	0.7			0.34	0.4
Power Density avg.	MWth/liter	0.04	0.45	1		0.28	0.35
Linear Rod Power max.	W/cm	650	500	590	420	400	430
Neutron Flux max.	n/cm^2sec	$6 \cdot 10^{14}$	$7 \cdot 10^{15}$			$4 \cdot 10^{15}$	$4 \cdot 10^{15}$
Primary Heat Transfer System							
Type		Loop	Loop		Loop	Loop	Loop
Coolant		Na	Na	Na	Na	Na	Na
Number of Coolant Loops		1		2	2	2	1
Pump Capacity	m^3/h	1360		600	670	1300	2400
Coolant Temperature							
Core Inlet	°C	370	260	360-450	410	375	370
Core Outlet	°C	430	420	600	560	525	500
Date of Operation		1968/69	1973	1969	1971		1972

FIG. 1 NEUTRON FLUX SPECTRA OF FAST REACTORS



				σ_f of Pu-239, b
———	EFFBR			1.61
— — —	1000 MWe Breeder	Metal Fuel	Na cooled	1.61
—·—·—	1000 MWe Breeder	Carbide Fuel	Na cooled	1.71
------	1000 MWe Breeder	Oxide Fuel	Na cooled	1.78
.....	1000 MWe Breeder	Oxide Fuel	Steam cooled	2.01

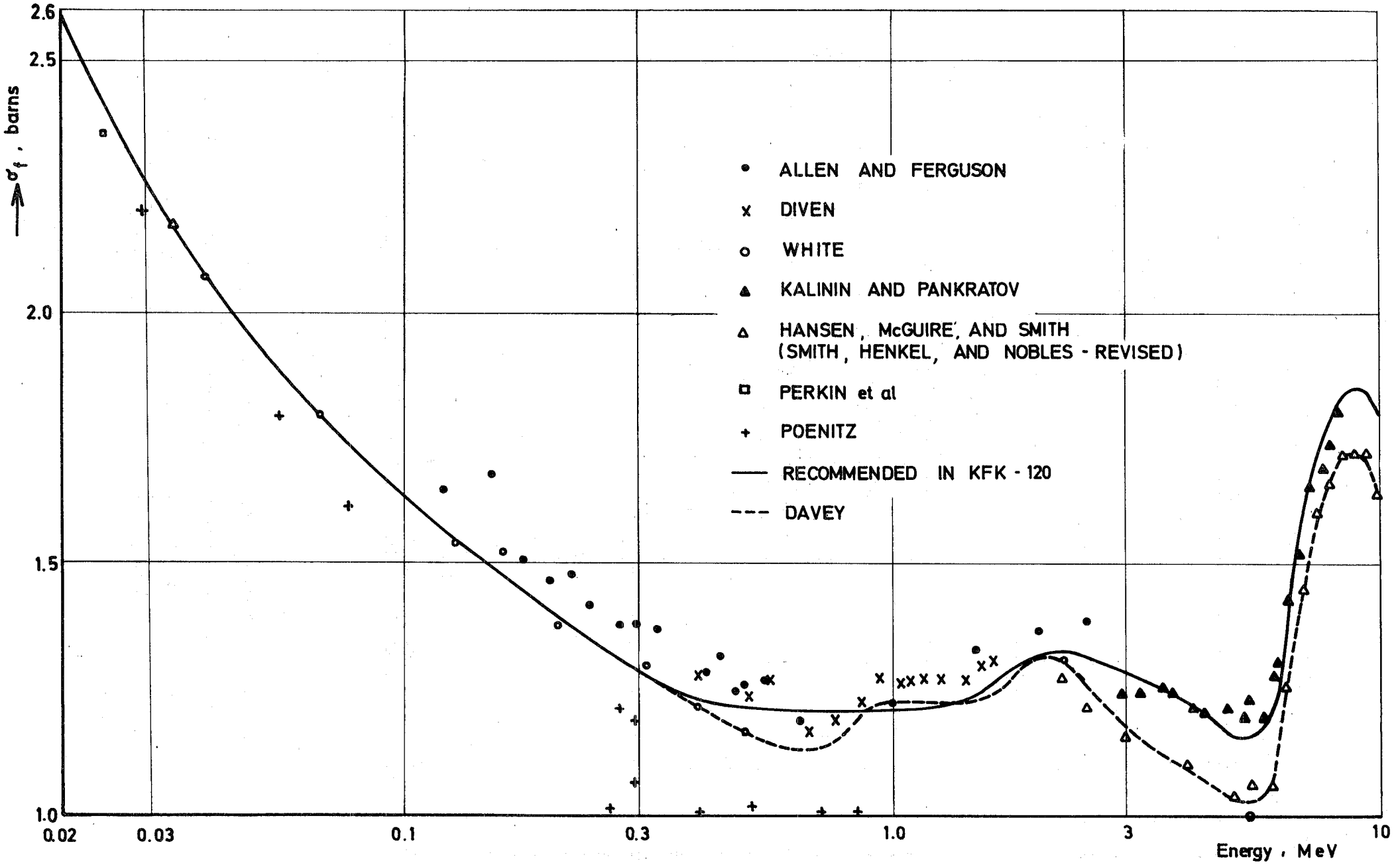
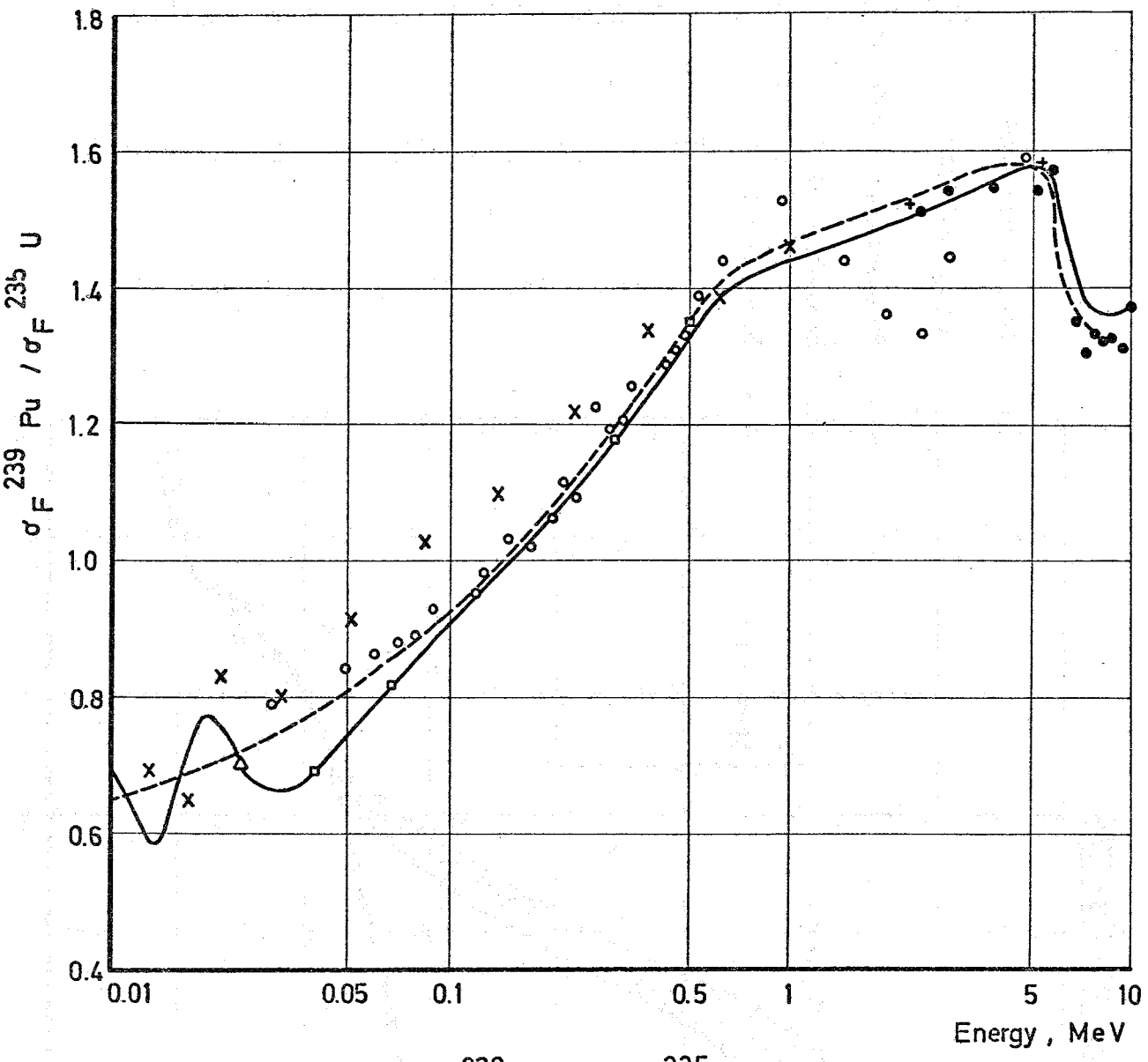


FIG. 2 FISSION CROSS SECTION OF U 235

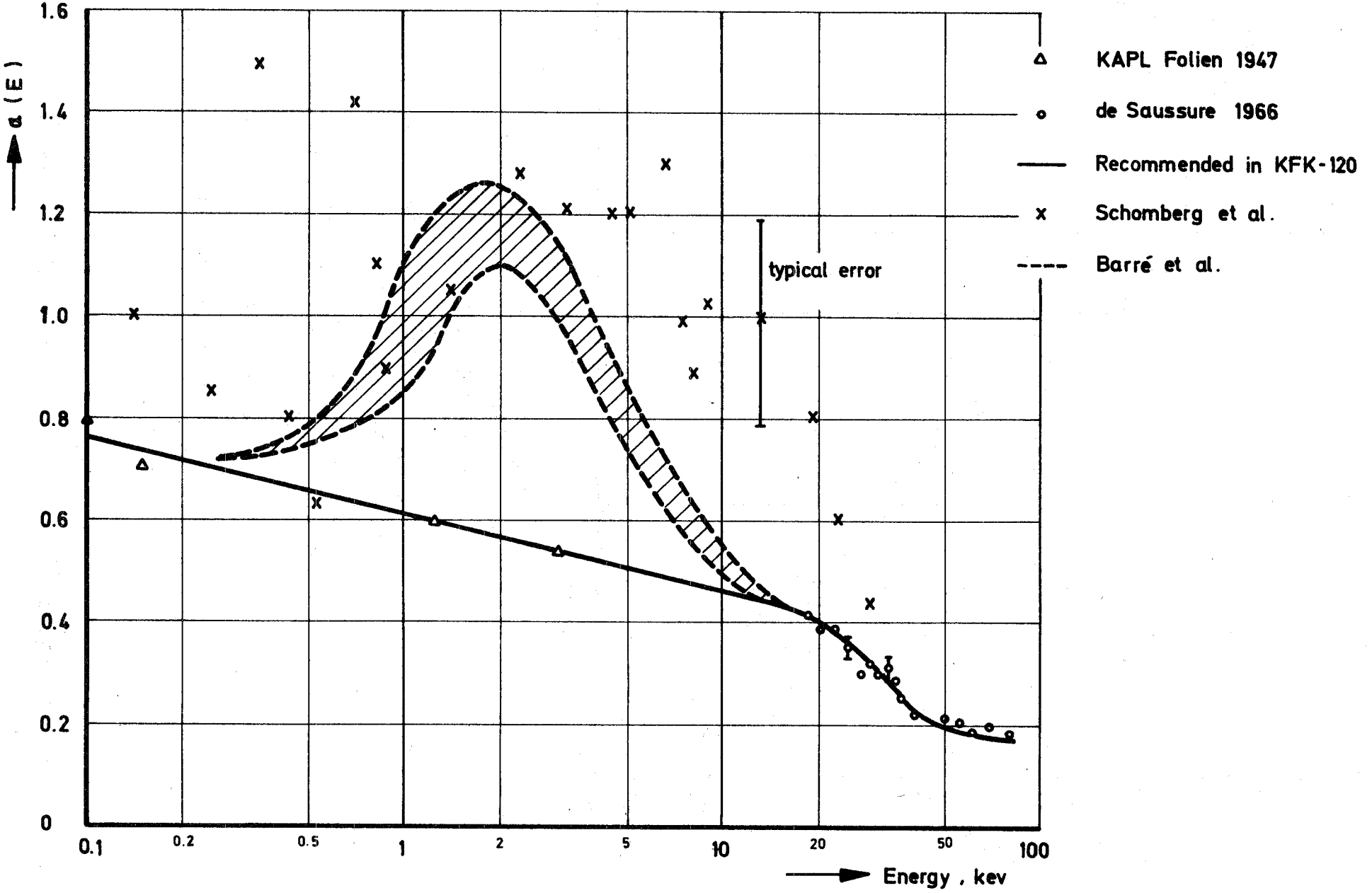


$\sigma_F^{239}{}_{Pu} / \sigma_F^{235}{}_{U}$ RATIOS

- ALLEN AND FERGUSON
- SMITH, HENKEL, NOBLES
- DUBROVINA, SHIGIN
- WHITE , HODKINSON, WALL
- PERKIN et al
- WHITE , WARNER
- RECOMMENDED IN KFK-750
- DAVEY

FIG. 3 RATIO OF Pu^{239} and U^{235} FISSION CROSS SECTION

Fig. 4 α (E) of Pu-239



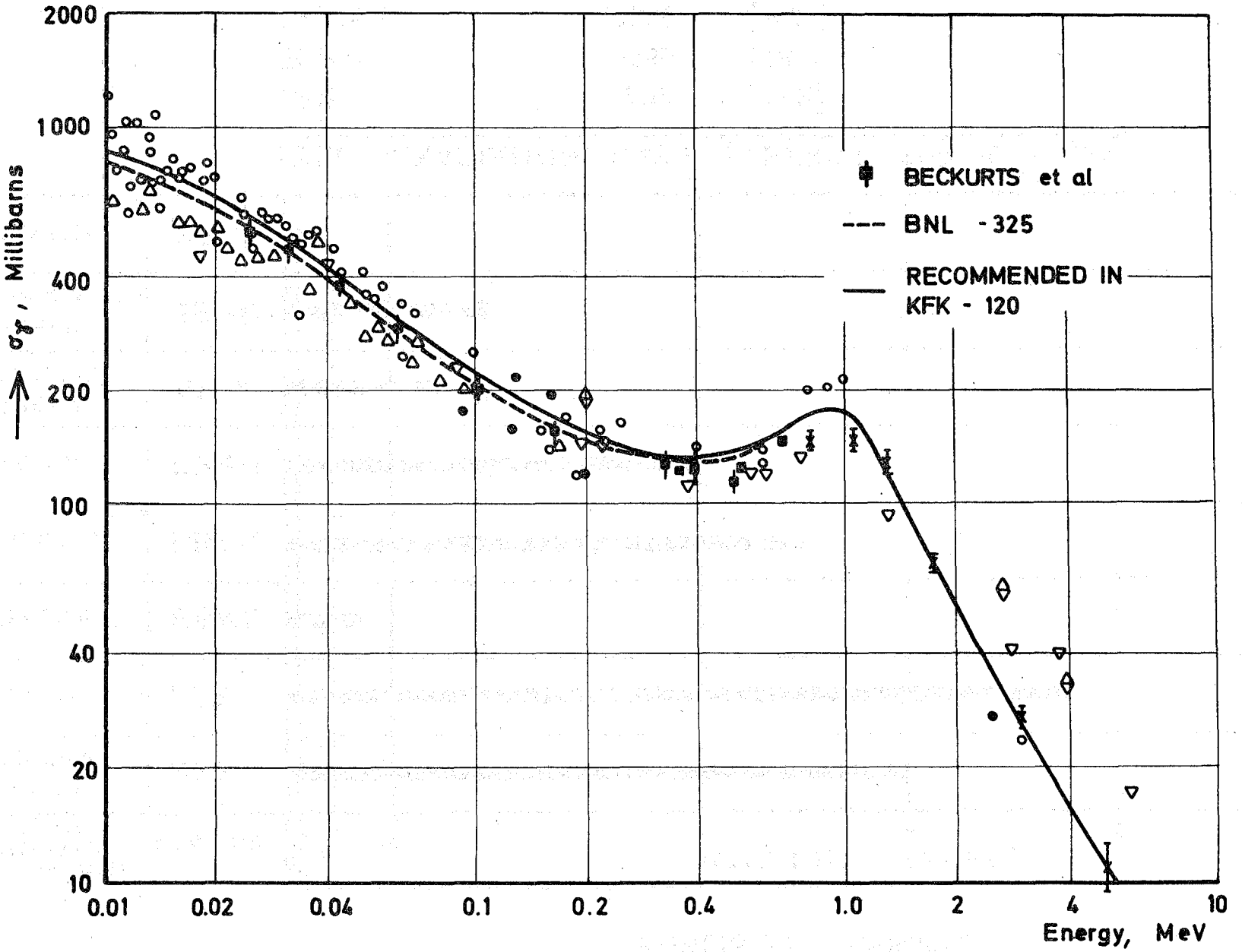
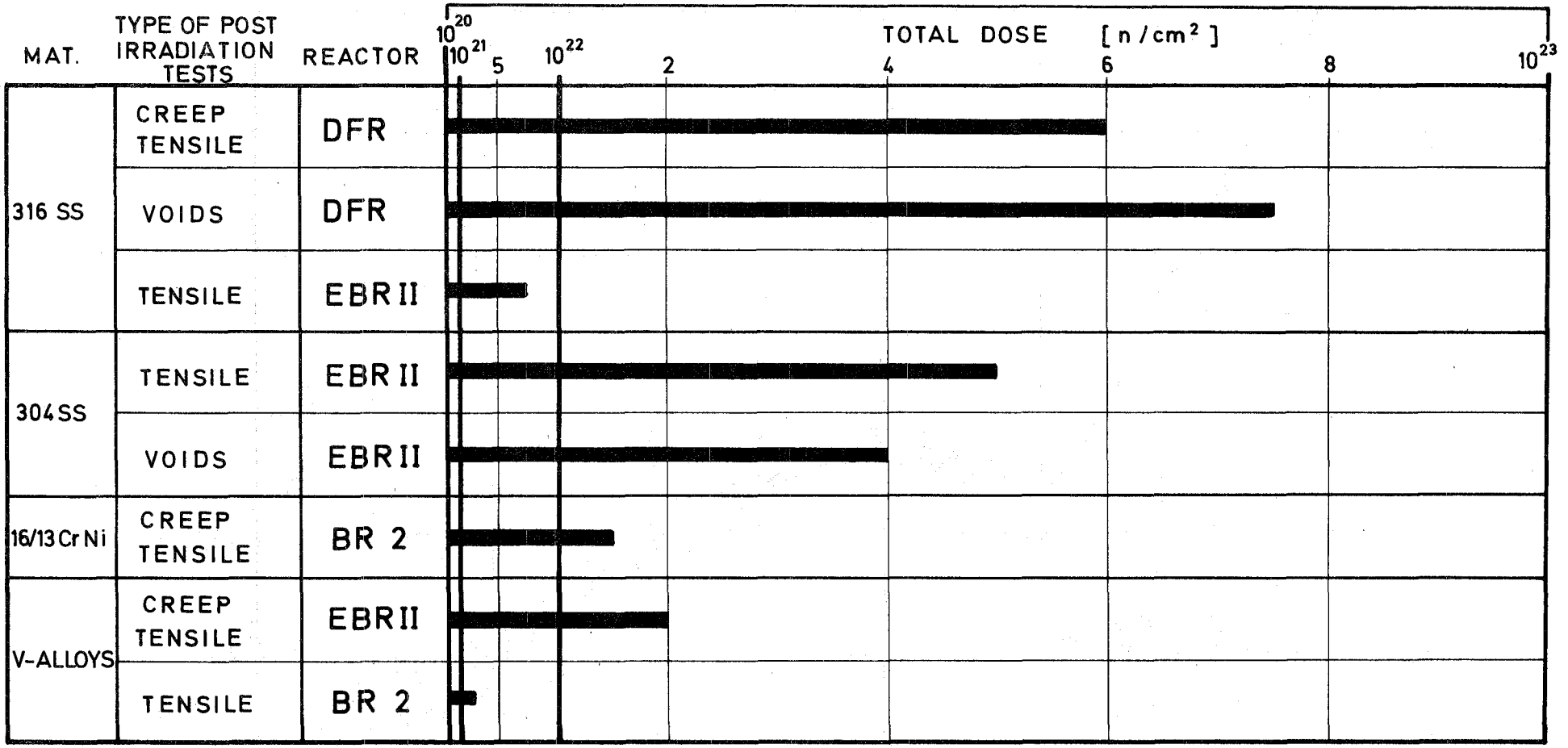


FIG. 5 CAPTURE CROSS SECTION OF U 238

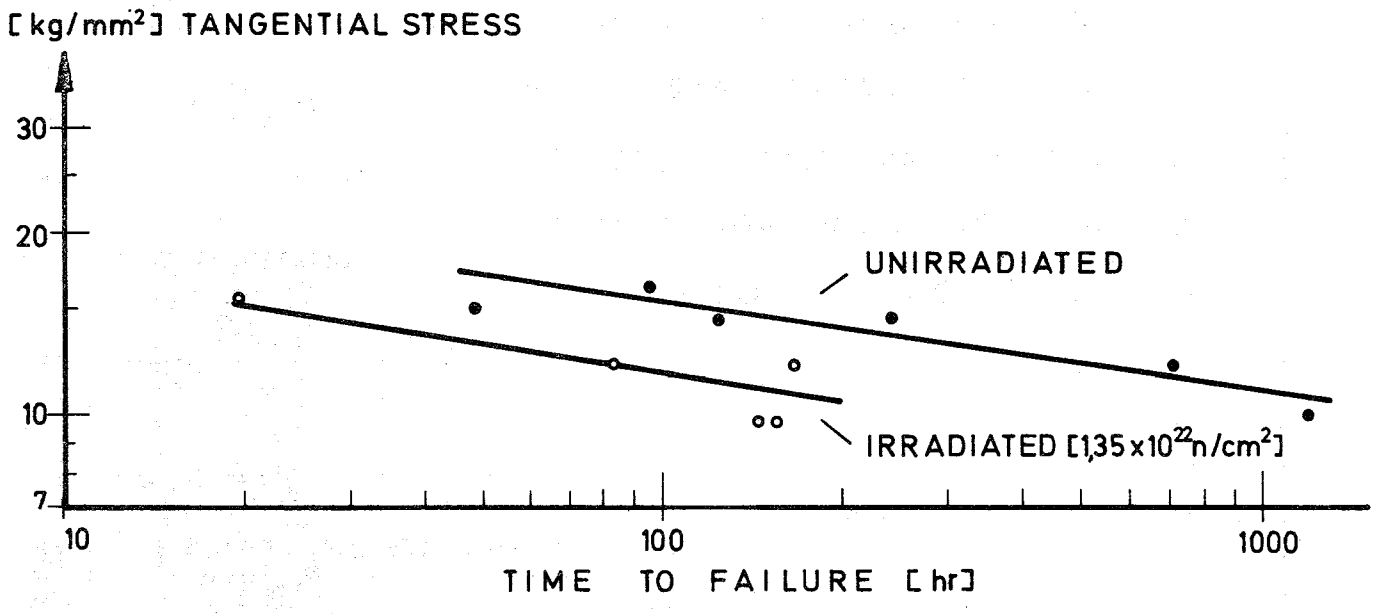
FIG. 6 OBTAINED DOSES OF CLADDING - IRRADIATIONS

[PUBLISHED RESULTS]



FULL IRRADIATION - DAYS [YEARS] FOR 10^{23} n/cm^2 :

DFR :	520	[1,42]
EBR II :	660	[1,81]
BR2/MOL:	2900	[8,0]



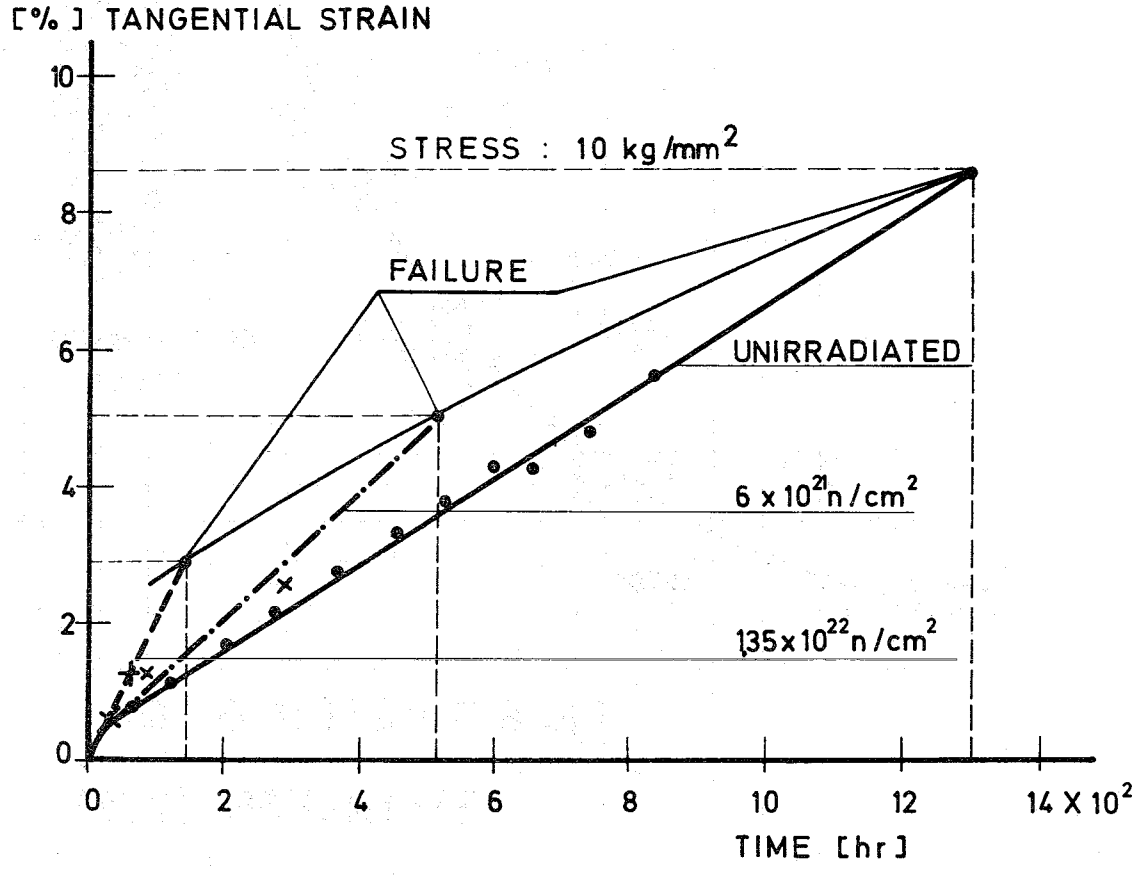


FIG. 7 EFFECT OF IRRADIATION ON THE CREEP AND STRESS-RUPTURE STRENGTH OF PRESSURISED 316 L TUBES AT 650°C [from I. P. BELL et al [62]]

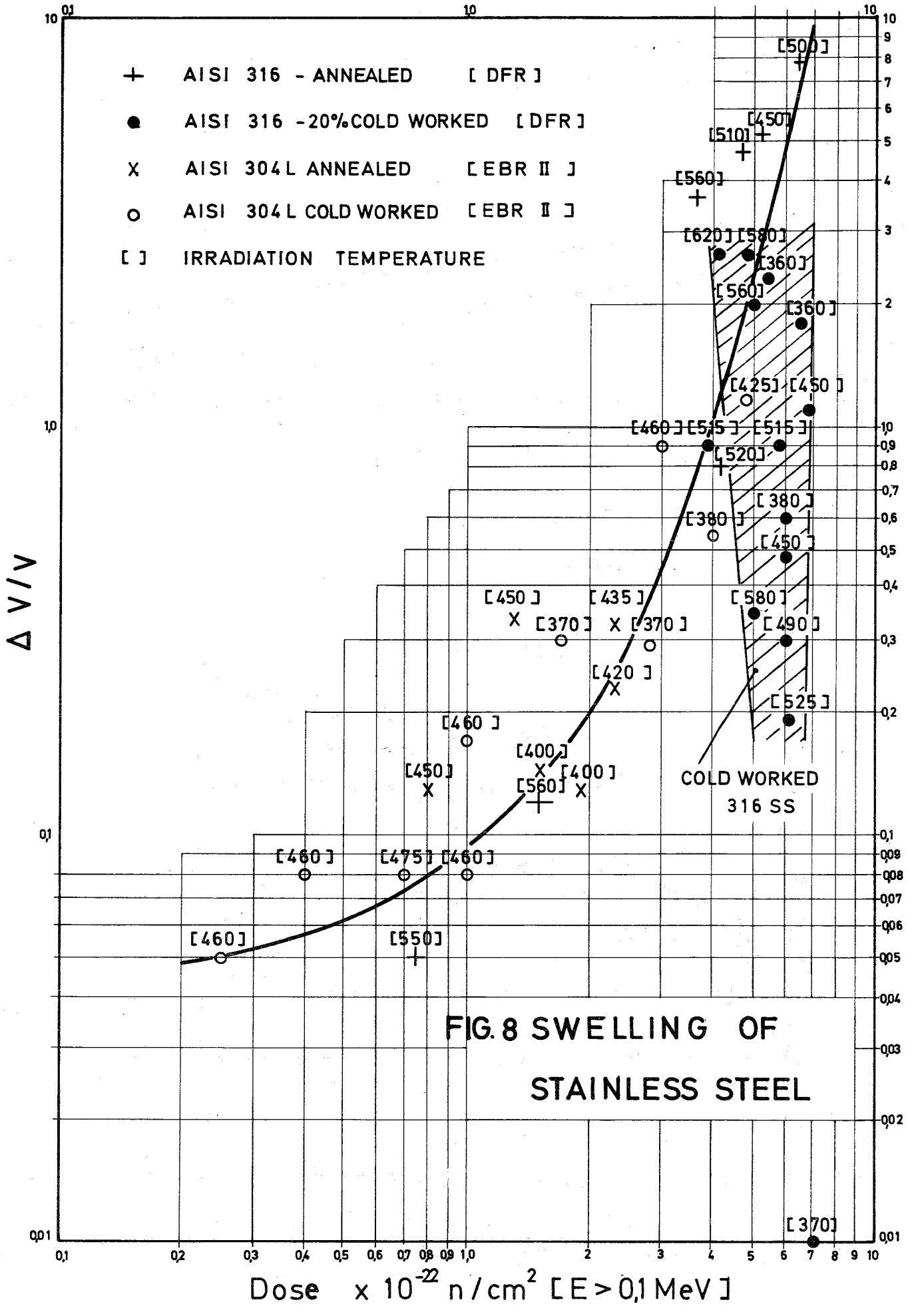


FIG. 8 SWELLING OF STAINLESS STEEL