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FUEL ELEMENTS CONFERENCE

Held at
GATLINBURG, TENNESSEE
MAY 14-16, 1958

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Division of Research, AEC
Division of Reactor Development, AEC
and
Oak Ridge National Laboratory

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FOREWORD

The Fuel Element Conference held in Gatlinburg, Tennessee, May 14 - 16, 1958, under the joint sponsorship of the Atomic Energy Commission Divisions of Research and Reactor Development and the Oak Ridge National Laboratory, provided a favorable medium for presentation and discussion of recent developments in the field of solid fuel elements. Such topical conferences were designed to replace the more general Metallurgy Information Meetings held annually in the past.

The specific objective of the conference was to exchange classified information between contractors to stimulate the over-all development of fuel element technology at AEC sites. The scope of the meeting embraced the design, fabrication, and performance of fuel elements. Materials problems and developments which are of concern and interest in reactor operations were also discussed.

At six formal technical sessions, thirty-six papers on developmental research in fuel element technology were presented under the following categories:

Research Reactors	Mobile Reactors
Production Reactors	Radiation Damage
Power Reactors	

Also a panel discussion was conducted on the "Future Trends of Fuel Materials." All these proceedings are included in this volume.

In addition, two films were presented: "Nichrome Fuel Element Production," produced by the General Electric Company - Aircraft Nuclear Propulsion Department, and "Technology of Enriched Fuel Plate Elements for Research Reactors," produced by the Oak Ridge National Laboratory.

Mr. J. E. Cunningham, Assistant Director, Metallurgy Division, Oak Ridge National Laboratory served as general chairman of the conference, which was attended by approximately 200 representatives of the AEC prime contractors.

The cooperation of the authors in preparing manuscripts suitable for direct reproduction by photo-offset lithography is greatly appreciated.

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†These papers are classified and have been published in TID-7559(Pt.3).

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*These papers are classified and have been published in TID-7559(Pt.2).

†These papers are classified and have been published in TID-7559(Pt.3).

Paper 1

FUEL CYCLE DEVELOPMENT

CLARK GOODMAN

Division of Reactor Development, U. S. Atomic Energy Commission,
Washington, D. C.

Nuclear power costs can be divided into three general areas: (a) capital costs, (b) operating costs, and (c) fuel cycle costs. Everyone in attendance at this conference is well aware that we must reduce the costs in all of these areas before nuclear power can enter the promised land of competitive costs with conventional fuels.

CAPITAL COSTS

We generally measure capital costs in dollars per kilowatt of capacity. We are just learning to predict with fair reliability the true capacity and cost of nuclear power plants. Most of our previous predictions have been shown to be ultra-conservative. The Experimental Boiling Water Reactor (EBWR) at the Commission's Argonne National Laboratory is an outstanding example of how much more power we may be able to get out of a nuclear plant than was originally expected. As laid out on the drafting board, the EBWR was a 20,000 thermal kilowatt reactor. As you know, the engineers at Argonne have been able to operate this plant at more than three times this rated capacity. What is more, they think with only a few changes the power can be boosted even more, possibly even to 100,000 thermal kilowatts. A factor of three increase in capacity does not mean a reduction to one-third of the capital costs. More or larger turbo-generators must be installed to take full advantage of this higher power. Hence, as a rough estimate in this case the dollars per kilowatt would be one-half of the design estimate. Similar increases in capacity have also been demonstrated in the Vallecitos Boiling Water Reactor of the General Electric Co. Sizable increases in capacity are also anticipated for pressurized water reactors.

Other important reductions in capital costs will come from (a) simplification in design, (b) increased use of standardized components, (c) adoption of commercial specifications and fabrication techniques, and (d) from multiple production of components. Unit capital costs will also be reduced by building larger plants, i. e., in the 200,000 to 500,000 kilowatt (electrical) range.

OPERATING COSTS

Because of lack of operating experience for civilian nuclear powerplants, estimates of operating and maintenance costs are particularly uncertain. Present plants or plants about to be placed in operation may require relatively large operating and maintenance staffs to meet contingencies. Operation and maintenance will also be complicated and relatively costly due to the necessity for controlling radiation. Experience is expected to reduce these costs to the range of those for conventional plants.

FUEL CYCLE COSTS

The fuel cycle costs of a reactor must include all of the expenses for every atom of fuel from the "cradle to the grave," that is from the mining of the uranium ore to the final disposal of the fission product wastes. The major components of these fuel costs are: (a) the costs of the fissionable material and other ingredients in the fuel elements, (b) the costs of fabricating the materials into fuel elements, (c) the charges for chemically processing the fuel elements after removal from the reactor, and (d) the costs of waste processing and disposal. There are other fuel cycle costs, such as inspection, transportation and losses during processing but these are relatively minor in importance and offer little opportunity for significant fuel cycle cost reduction.

It is clear from the program of this conference and from the talks presented before the Atomic Industrial Forum Conference held in Florida last week that a great deal of work is being done in fuel element development by both government and industry. However, I am sure that after three days of meetings we will arrive at much the same general impression as did our colleagues in Florida, namely that there must be a major expansion in fuel cycle development for civilian nuclear power plants.

I am pleased to report that the Atomic Energy Commission has for some time been aware of this need and has already taken a number of steps to initiate an ambitious expansion in fuel cycle development which extends from materials and process development through the testing of multi-ton batches of developmental fuel elements in full scale power reactors. To accomplish this, it is recognized that the present Commission program which includes general materials development and testing, chemical processing and waste disposal work must be expanded considerably and oriented toward the development, fabrication and testing of a number of fuel element types suitable for test operation in the various power reactors now in operation or under construction, including those owned by industry as well as those owned by the government.

FUEL CYCLE DEVELOPMENT PROGRAM

I will outline the general features of this program, and then discuss in some detail a few of the important problems that we hope this program will be able to solve.

Although the Commission is supporting a \$17,000,000+ Fuel Technology Center at Argonne under the able leadership of Frank Foote we have concluded that a centralized, government-owned and government-operated fuel element manufacturing facility is neither feasible nor desirable. Fuel element development represents an integral part of the individual reactor programs of a number of the Commission laboratories and contractors. We plan to expand and accelerate those developments along the lines which offer the greatest promise, both short and long range. It is also considered extremely important that various industrial concerns take as large a role as possible in order to continue the Commission's policy of encouraging the growth of the nuclear fuels industry. However, in order to focus our attention on the specific objectives, we plan to keep the Commission's fuel development as a separate program.

We plan to concentrate on oxides, carbides and other ceramic or ceramet materials containing uranium, thorium and plutonium. We will work primarily toward lower fabrication and materials costs and longer lifetime of fuels suitable for full-scale power reactors. The initial work will result in the development of prototype fuel elements which can subsequently be produced in large numbers for tests in full-scale power reactors. Consideration will be given to improved manufacturing techniques and inspection and testing procedures which offer promise of lowering the costs in the subsequent production of the various fuel elements of major interest.

Obviously this program will require more test space than is now available in either the Materials Testing Reactor (MTR) or the Engineering Test Reactor (ETR). In accord with its announced policy, the Commission hopes to secure this additional irradiation space through contracts with the operators of privately-owned test reactors. Fortunately, there are a number of such reactors now built, building or planned. However, more are needed, particularly test reactors of a size intermediate between the MTR and the ETR. One of the pre-requisites for getting on with this part of the program is that the Commission be given specific authority to enter into long-term agreements with test reactor operators for the provision of this necessary test space to meet the development schedule.

The accelerated program for fuel cycle development requires that increased effort be devoted to chemical processing development. The emphasis on oxides, carbides and other ceramics and ceramets necessitates expansion of the work on these materials that has already begun. The fine work of Floyd Culler and his group at the Commission's Oak Ridge National Laboratory (ORNL) on aqueous chemical process development should be accelerated. Increased effort should be placed on volatility work at both Oak Ridge and Argonne. A significant increase should be made in the level of effort being applied to pyrometallurgical process development now under way mainly at Atomics International.

In keeping with the goal of lowered fuel cycle costs, increased effort must be devoted to waste treatment development. We need to expand what we are already doing and try and find some bright new ideas that will drastically reduce the present excessive storage costs. We would also hope to include an aggressive program aimed at developing the industrial potential of fission by-products, both the radiation and the heat released.

PROBLEMS TO BE SOLVED

Now I will summarize some of the problems we expect this program to solve. First and foremost is the development of proven long-lived fuel elements. High burn-up of the fuel in a reactor, especially in an enriched uranium reactor, is essential for good economics. Our immediate goal is an average of 10,000 megawatt days per ton - proven, not postulated.

As you know, when four atoms of U-235 are fissioned, one of the eight fission-product atoms formed either is or soon becomes an atom of rare gas, xenon or krypton. This means that when only 1% of the U-235 atoms in a cubic centimeter of the metal has fissioned, some 4.4 cc of these gases (at standard conditions) have been produced. These gas atoms must be accommodated within the cc of metallic fuel. Yet these rare gases are practically insoluble in uranium. Hence, the xenon and krypton are in effect under very high pressure within the lattice structure of the solid fuel.

As a result of the brilliant work done by John H. Kittel and his colleagues at Argonne National Laboratory, it has been dramatically demonstrated that:

- (1) The accumulation of these gases results in a decrease in density of the metal that is essentially independent of prior fabrication history and irradiation temperature.
- (2) Fine grained, highly oriented natural uranium metal disintegrates under irradiation by a process of transverse fractures, whereas coarse grained, randomly oriented material fails by progressively deeper surface damage.
- (3) Natural uranium metal which has been rolled at 300°C and beta quenched should withstand 2 at. % burn-up (17,000 Mwd/ton) without disintegrating if the temperature is kept below about 200°C.

We certainly want to see the effects of grain size and orientation tested under operating conditions. However, there is little hope that these methods are sufficient at elevated temperatures.

About 300° to 400°C uranium metal is so poor in strength, and diffusion of the fission gases is so large, that gas bubbles are formed which cause serious swelling. This swelling imposes both burn-up and temperature limitations on metallic uranium fuels, especially homogeneous metal fuels such as pure uranium or solid solution alloys of uranium or plutonium.

There are several promising methods of overcoming these limitations. One is to use a dispersion type fuel consisting of a fuel phase dispersed in a ductile metallic matrix containing essentially no fuel - for example, UAl_4 in aluminum. The other is to use high density ceramic materials, notably UO_2 . Another approach is to use a sufficiently strong cladding to prevent by mechanical means the swelling of the fuel.

All of these approaches are under study at a number of laboratories both here and abroad. In fact, this is an area where many of the extensive investigations being made on military reactors are directly applicable to the civilian program. I have in mind the characteristically thorough fuel element development being carried on at the Westinghouse Bettis Laboratories under the general guidance of Admiral H. G. Rickover.

A second important area in fuel technology which we hope to see accelerated is that of the use of man-made fissionable materials as fuel in power reactors. The use of U-233 and Pu-239 as fuels has been delayed both by the difficulties inherent in handling these toxic isotopes and by the emphasis on their military applications. The latter is a policy matter which appears to have been settled in favor of fuel development. The Commission is solidly behind the plutonium recycle and thorium - U-233 fuel cycle programs. I think the thorium conference held at Brookhaven National Laboratory on January 9 and 10, 1958, and the plutonium conference held at Hanford Engineering Works on April 22 and 23, 1958, were very helpful in bringing together all of the major activities in each of these two areas of nuclear fuel development.

There are several important reactor development projects going forward which employ thorium as fertile material and require U-233 as the fissionable material. There is a general need for a variety of critical experiments to be carried out in these programs over the next few years. Although some of the needed basic physics data are available from which to make approximate reactivity calculations, critical experiments are essential for each specific reactor type in order to confirm the assumptions in the calculations, especially at elevated temperatures. A study recently completed indicates that 30 kg of U-233 are needed this coming year and 80 kg in 1960 to support the minimum thorium - U-233 development program. A really comprehensive program would require more than ten times this amount by 1960. Considerations are now being given to these needs. I have confidence that the commission will produce sufficient U-233 to satisfy these important requirements.

To date, no reactor has been operated on U-233. Hence, the actual feasibility of its use as fuel is not yet established. As a first step we are hoping to be able to operate the MTR with a full set of U-233 fuel elements. As now planned the fabrication and use of these fuel elements will follow the already announced plan to operate the MTR on plutonium. These integral experiments should serve as valuable checks on the basic nuclear data as well as demonstrate in a limited way the feasibility of fabricating fuel elements containing these toxic materials. During a recent visit to Hanford, I was pleased to see the progress being made by General Electric on this difficult problem.

The operation of the MTR on 20% enriched fuel was completed successfully last fall as a demonstration of the practicality of using fissionable material of this enrichment. As you know, until recently our bilateral agreements limited the enrichment of exported uranium to this level for all except test reactors.

Operation of the MTR on plutonium fuel is scheduled to begin in June of this year. The Plutonium Recycle Test Reactor (PRTR) is under construction at Hanford. However, operation of this facility is not scheduled until 1960. Likewise, the Los Alamos Molten Plutonium Reactor Experiment (LAMPRE) and the plutonium loading in the second core of the Experimental Breeder Reactor No. 2 (EBR-2) are not likely to provide operating experience with plutonium fuel before late 1960 or early 1961. This means that there is a serious hiatus in the plutonium fuel development program. I hope that by calling attention to this situation some of you will be stimulated to come in with ideas for how we can get operating experience with plutonium fuels in power reactors more quickly.

Another important area in our fuel development program is that of providing fuel elements for foreign power reactors under co-operative arrangements such as that now being discussed with the EURATOM group. In order for a foreign utility to estimate with any reasonable reliability the cost of the nuclear power to be produced, it is essential that there be some assurance that the fuel can be carried to the predicted burn-up. The proposed co-operative program with EURATOM must provide some form of government guarantees of fabrication costs and physical integrity of fuel elements. One possibility is that the guarantees would be in the form of U. S. Government assistance to American manufacturers rather than to foreign utilities. In this way, the foreign fuel program would prove of direct value in lowering the cost of nuclear power in the United States as well as in Europe.

SUMMARY

In brief, then, I think all of us at this conference can take encouragement from the interest of the Commission in an expanded and accelerated fuel element development program. There are many problems to be solved among the most important of which are the fuel lifetime, the reduction in cost of chemical processing and waste disposal, the reduction in cost of fuel fabrication and the development of plutonium and U-233 as fuels for power reactors.

In addition there are two other areas in which we would like to see more work being done. Both of these are aimed at increasing the efficiency of nuclear power plants through the attainment of higher temperatures. The very comprehensive studies on gas-cooled, graphite moderated reactors made by Kaiser Engineers together with ACF Industries, and paralleled at ORNL with the assistance of the TVA, were recently reported to the Joint Committee on Atomic Energy. These two groups independently recommended taking a giant step in this field - that is to design for exit gas temperatures of 1000°F and above. Time does not permit discussing this further, but it is clear that if this could be done it would be a major breakthrough in gas-cooled reactor development. To accomplish this advance will require some really imaginative effort. The same kind of thinking is required to achieve the other objective I wish to mention, namely nuclear superheat. Until we can get superheated steam from nuclear power reactors without resorting to chemical fired superheaters, we are not able to take full advantage of the progress that has been made during the past 10 - 20 years in the development of high efficiency fossil fuel systems. The immediate need for this new knowhow is not only in our own domestic program but in the programs of our foreign friends as well. We have a lot of work ahead and this conference is a good way of focussing attention on the major problems and of bringing co-ordinated efforts to bear on the prompt solution of these problems.

DISCUSSION

MR. A. R. KAUFMANN (Nuclear Metals, Inc.): I didn't quite get one remark you made. It sounded as though you were saying that all future work would be done on compounds -- carbides, oxides, etc. Is that what you said?

MR. GOODMAN: No, I don't think I said that. I think I said that the expansion of the present program would be in that direction. At least that is what I intended to say.

Part of the problem in this, taking my hair down a little, is that Ken Fields is today reporting to the Joint Committee on the expanded budget, that is, the supplementary budget which was approved just last week by the Bureau of the Budget. So some of the things I said I could not have said two days ago.

Now you also realize that this question of the direction that the program should take that I have mentioned is essentially as I see it I should say -- I won't even say as we see it in Washington -- but as I see it, which of necessity has a certain amount of blinders involved in it. But I have attempted to say what I think and this has been approved by Ken Davis and a number of others, but it has not as yet had a chance, because of the time element, to be discussed with Jules Simmons, Carson Dalzell, Mal Whitman, or even Glen Wensch here. I haven't had a chance to talk it over with him; so I would say this: If I said anything that this conference really disagrees with then let it come out in the discussion. We are open-minded about this but this would be the direction I would like to see it go, as I see the picture right now. Does that help any?

MR. E. L. CURRIER, JR. (ACF Industries): I am interested in your remarks about government guarantees on integrity and burn-up of foreign fuel elements and how that affects a commercial U. S. reactor designer who is engaged in selling foreign reactors. What is your thinking about how the AEC would participate in guaranteeing the integrity?

MR. GOODMAN: This is a very complicated problem and, as I said, the particular wording which I put in here went through a lot of hands before it got in its present form, and rather than answer your question, let me tell you the delicacy of it. We are right now in the midst of negotiating this very point with the Euratom people. The big hurdle which apparently has been made with the Euratom group is the question of inspections and we think we have that pretty well under control. It has to be settled first. It is a co-operative program and we are trying to get the Euratom people to tell us what they need. At the same time we are not trying to subsidize their program but to support it only to the extent that it is necessary to make it become a reality.

To do this the really key issue is the question of guarantees on fuel. This has not been approved by the Commission or by the Joint Committee, nor has it been accepted by the Euratom bodies. If we take the figure which I mentioned -- and this is the figure which is being talked about -- of 10,000 Mw days/ton of enriched fuel, there should be some way that the fuel element manufacturers in the United States who supply that fuel for Europe can say to Europe or Euratom, "We will guarantee you as a minimum 10,000 Mw days/ton of this fuel. If the fuel for some reason or other does not meet that, okay, we make it good."

Now the question of how you make something good. Let's say it has gone now 9,900 Mw days instead of 10,000. Well, a battery-type guarantee does not get you much if it is a battery-type guarantee. The fuel element manufacturer, on the other hand, although there is some question about this, especially from what we hear on the Senate proposals, would probably not be willing to guarantee that if he didn't feel that he had Uncle Sam behind him if this thing got very serious.

As I said, I am not really answering your question. I am only telling you some of the delicacy of the actual answer, and the truth of the matter is it has not been worked out in detail. We have a program worked out in detail which is being discussed but it has not been agreed upon by anybody yet.

MR. CURRIER: There is a lot of confusion in my mind concerning the extent to which enriched uranium, perhaps in all degrees of enrichment, can actually be used to make economical power, and many of the programs that you talked about were aimed in the direction of enriched uranium. I have heard the European people say that they don't see how they can make economic power this way. They don't have any interest in it, and I am wondering whether somebody can make a definite statement that enriched fuel will produce economical power.

MR. GOODMAN: Let me say that really is a broad question. The more we look into the question of economics, the more we are convinced that you are not going to do it any other way than with enriched uranium. For example, the ACF-Kaiser Engineers and Oak Ridge-TVA people have looked at this problem and I think have looked at it very objectively as to whether we should or should not build gas-cooled modified reactors with natural uranium compared with enriched uranium. There are reports this high, one of which is a study of a prototype natural uranium and a full-sized natural uranium reactor and a prototype enriched uranium and a full-sized enriched uranium reactor. When you put them together the capital costs -- at least in the American frame of reference of capital costs -- the increase in capital costs in using natural uranium more than offset the somewhat, but not very large, increase in cost of fuel provided you make this assumption with respect to burn-up: 3,000 Mw/ton on natural uranium -- 10,000 Mw/ton on enriched uranium. By the way, the uncertainty of getting natural uranium to 3,000 Mw days/ton is probably just as great as getting enriched uranium, say in oxide form, to 10,000 Mw days/ton.

Now I see Mr. Herron, of the American-Standard here in the audience. They have studied this thing far more than I feel qualified to say. Do you have any comments to add to what I have just said, in capsule form anyway?

MR. D. P. HERRON (American-Standard Atomic Energy Division): I am going to talk on this tomorrow, so I don't want to scoop myself. We have studied enriched uranium in both the Calder-Hall reactor and the heavy water-type reactor, and we find that, as he states, the longer burn-up you can get with the enriched uranium, because of the increased reactivity, more than compensates for the slight cost of enrichment. Enriched uranium is one of the best bargains today in the world over at Oak Ridge, and every case that we have studied shows that using slightly enriched is desirable. If you go up much above 1-1/2 % in enrichment then the burn-up cost does get pretty extensive, but if you can stick below 1-1/2 % enrichment that range is almost always economical.

MR. CURRIER: One more question. In most reactors there seems to be emphasis on high specific power -- getting as much power as possible out of a small region. This helps to reduce the size of the core, the pressure vessel, and everything else. Now on the other hand, people keep talking about the importance of getting to higher temperatures in order to get higher thermodynamic efficiency. I am under the impression that what you are really accomplishing is going, say, from a 25% efficiency to a 30 or 35% efficiency. The question is whether this increase in thermodynamic efficiency really offsets the fact that you are getting much lower specific power and you are faced with a much bigger plant and everything else. In other words, it is demonstrated that the most important thing to do is to get higher temperatures or is it just a matter of keeping the power plant engineers happy?

MR. GOODMAN: Again, I think Mr. Herron will have something to say about that, but let me just give my impression. I don't think any of us would think

that higher thermodynamic efficiency in itself is the objective. When the increased thermodynamic efficiency allows you to use more efficient power machinery which has been developed for these high temperature-high pressure plants, if that decreased cost/kilowatt is reflected back in the over-all analysis as an over-all decrease in cost as compared to the more expensive reactor that you would have to have in order to get these -- if the net gain is negative, obviously we don't want to do it.

The other thing is -- and I think this is really the basic point -- maybe with a little more courage we can get this without any increase in capital cost, or with very little increase in capital cost, by running these plants at higher specific powers. But the group that studied the gas-cooled reactor, if I may use that as an example, really struck a happy balance on this thing. They didn't make these little, tight, compact plants in order to take all the saving in capital cost and put all the burden over onto the metallurgist in terms of temperature and the chemical engineer in terms of heat transfer. Instead these are quite sizeable plants and they do have high capital costs, and it is admitted that they are not the optimum in terms of thermodynamic characteristics in order to keep the capital costs down. In other words, if you do the job right -- and the trouble in doing it right is that there are so many uncertainties in the calculation -- but if you knew all of the factors involved, then you would obviously include this as one of the factors in the optimization of the cost and that optimization is exceedingly difficult.

MR. HERRON: In going to high temperature you get the higher temperature and you save two ways: In the burn-up cost because you are utilizing more of the heat to produce power and in the cost/capital equipment. I think you are right that the savings in the burn-up cost, particularly if you use only slightly enriched fuel, is not as important as the savings in capital equipment cost because of the fact that you can use a smaller heat exchanger.

One of our men was working on a fuel study at Oak Ridge and they found with the high temperature and the good thermal conduction of the fuel and the liquid sodium the total heat-transfer area was really no greater than for a straight water-cooled reactor, because of the fact that the temperatures were higher and the heat-transfer fluids were very good. So you can keep your size very good by using high temperature, and that is just as important as the saving of the burn-up cost of the fuel, which I say again is fairly cheap.

Paper 2

NEW DEVELOPMENTS IN RESEARCH REACTOR TECHNOLOGY

J. A. LANE

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ABSTRACT

At the present time, 105 research and test reactors are in operation, being built, or planned in the United States, compared to 22 in operation by the end of 1955. The most popular types are the low power training reactors and the high flux pool or tank type. This is due to the relatively low cost of the training reactors (~ \$100,000) and the flexibility and high fluxes of the tank type reactors. The thermal and fast fluxes in such reactors are of the order of 1×10^{13} per Mw. In the flux-trap reactors herein described, fluxes of $10\text{-}20 \times 10^{13}$ per Mw may be possible. Examples of this type of reactor, which consists of an annular core with a moderating "island" and reflector, are the Thermal Test Reactor at Knolls Atomic Power Laboratory, the Argonaut at Argonne National Laboratory, and the USSR 500 kw research reactor described at Geneva. Application of the flux-trap concept to ultra-high flux reactors results in reactor designs capable of achieving fluxes in the range of $5\text{-}8 \times 10^{15}$ neutrons/cm²(sec). Application of the concept to a High Flux University Reactor results in thermal and fast fluxes of the order of $2\text{-}3 \times 10^{14}$ at a 5 Mw power level. The cost of such a reactor is estimated to be approximately \$1.5 million by the Internuclear Company. The characteristics of the Internuclear design, described in this paper, are ideal for a high level nuclear research and testing program.

GROWTH OF THE UNITED STATES RESEARCH REACTOR BUSINESS

A review of the status of reactor technology in 1956¹ indicated a growing interest in those types of research reactors capable of achieving highest thermal neutron fluxes. This is shown in Fig. 1 reproduced from this previous review. Here it is seen that the growth rate of tank type or CP-5 type reactors has been greater than that of the natural-uranium fueled research reactors or the lower flux swimming pool or water boiler types. By the end of 1955, there were 21 reactors in operation in the United States and six more had been built, operated, and dismantled. In June 1956, it was predicted that 54 research and test reactors would be in operation in the United States by 1960 and 50 more in countries outside of the United States. The present count for the United States as shown by Table 1 indicates that these early predictions were too conservative. This table, summarizing the results of a survey by the Atomic Energy Commission in December 1957, indicates a total of 105 research and test reactors in operation, being built, or planned in the United States. This number does not include 69 critical experimental facilities and 23 research reactors built or planned by United States firms for installation in foreign countries. Adding these to the total makes a grand total of almost 200 such facilities built or planned. The distribution according

Fig. 1—World research reactor construction trends (as of Dec. 31, 1955).

Table 1
U. S. RESEARCH AND TEST REACTORS*

USES	OPERATING	BEING BUILT	PLANNED	TOTAL
Research and Training	28	20	29	77
General Testing	2	1	2	5
Special Testing	<u>13</u>	<u>4</u>	<u>6</u>	<u>23</u>
TOTAL	43	25	37	105

* As of December 31, 1957

to type and use is shown in Table 2. The tank type reactors no longer head the list, but have been replaced by solid-homogeneous training reactors. The large number of solid-homogeneous reactors shown in this table, most of which have been built or planned by Aerojet General Nucleonics, is a tribute to the salesmanship of this company or is due to the fact that such reactors cost in the neighborhood of a mere \$100,000 and fall comfortably within the amount the Atomic Energy Commission grants to universities for nuclear training facilities. Other types of training reactors, available from American-Standard, ACF Industries, AMF Atomics, and others, though designed for higher power levels (10 kw compared to 0.1-1 kw for the AGN reactors), have been less popular because of their higher price (\$150,000-\$250,000).

Table 2
TYPES OF U. S. RESEARCH REACTORS*

<u>Reactor Type</u>	<u>Primary Uses</u>	<u>No. Operating or Planned</u>
Solid Homogeneous	Training and Research	45
Pool	Research and Testing	18
Tank	Research and Testing	14
Liquid Homogeneous	Research and Training	10
Graphite	Research and Testing	8
Heavy Water	Research	4
Graphite-Water	Training and Research	3
Pressurized Water	Testing	<u>3</u>
	TOTAL	105

* As of December 31, 1957

Table 3
COST FOR TEN REPRESENTATIVE RESEARCH
AND ENGINEERING TEST REACTORS

<u>Reactor</u>	<u>Type</u>	<u>Power kw</u>	<u>Total Costs in Millions of Dollars</u>
Armour Research	Homogeneous	50	0.70
Penn. State Reactor	Pool	100	0.31
Battelle	Pool	1,000	0.76
University of Michigan	Pool	1,000	0.85
Oak Ridge Reactor	Pool	20,000	3.8
Materials Testing Reactor	Tank	30,000	18.0
Engineering Test Reactor	Tank	175,000	17.2
Chicago Pile No. 5	Heavy Water	1,000	2.4
Oak Ridge Graphite	Graphite	3,800	5.2
Brookhaven	Graphite	2,800	19.7

The popularity of the pool or tank type reactors as shown by Table 3 is due to the sound technology developed for such reactors and their general utility for research, training, and general or specialized testing. The suitability of the tank type reactors for high flux research accounts, in part, for the large number in operation or being built.

Fig. 2—Total cost vs reactor power for six basic reactor designs.

Fig. 3—Average thermal neutron flux to power ratio for six basic reactor designs.

COST FACTORS^a

Total costs vs. the reactor power level for various types of research reactors are shown in Fig. 2. Here it is seen that costs increase linearly with power level; however, for a given power level, pool or tank type reactors cost about one-half as much as the water boilers or heavy water reactors and about one-third as much as the graphite piles. The average thermal neutron flux per kw is shown in Fig. 3. Since values are inversely proportional to the total fission cross section, water boilers and D₂O moderated reactors, which operate with minimum amounts of U-235, achieve unit flux levels of about 2×10^{10} neutrons/cm²(sec) kw. Values for pool type reactors are about 1×10^{10} neutrons/cm²(sec) kw. In flux-trap type research reactors, to be discussed later, unit thermal fluxes of $10\text{-}20 \times 10^{10}$ neutrons/cm²(sec) kw may be possible. Because of the very low unit fluxes of the natural uranium reactors, consideration is being given to the possibility of fueling these reactors with enriched uranium to gain about a factor of ten in flux.

The average fast neutron flux per kw in various reactors is shown in Fig. 4. As seen in this Fig., highest fast fluxes per kw are obtained with reactors with the smallest core volume and minimum macroscopic total cross section for fission neutron energy. The average fast flux per kw for the best cases is about $1\text{-}2 \times 10^{10}$ whereas in flux-trap reactors maximum values of 5×10^{10} fast neutrons/cm²(sec) kw may be possible. The characteristics of these reactors and their costs are summarized in Table 3. The cost of supplementary facilities such as radiation laboratories and hot cells are not included. These costs may add from \$1.5-2.0 million to the cost of the installation.

The fuel cost will, of course, depend on the type of fuel and the power level. For MTR type fuel elements annual fuel costs per kw, excluding in-

Fig. 4—Average fast neutron flux to power ratio for six basic reactor designs.

^aTaken from Internuc-22, "High Flux Reactors for University Research."

ventories, amount to \$7.50 for fabrication, \$6.00 for burn-up and \$10.00 for processing. Thus, annual fuel costs for a 10 Mw reactor are in the neighborhood of \$240,000.

DEVELOPMENTS IN RESEARCH REACTOR TECHNOLOGY

It is evident from the foregoing that the most significant developments in the research reactor field during the past few years have been (1) the development of reasonably inexpensive training reactors that can be mass produced (AGN series), (2) the design and construction of a very high power, high fast-flux testing reactor (the ETR), and (3) the construction of a reactor with approximately MTR fluxes at appreciably lower capital cost (the ORR). These are noteworthy achievements; however, they do not represent advances in technology measured in terms of power density, specific power, or neutron flux per unit of power since none of the research or testing reactors built to date appreciably surpasses the MTR in these respects.

In the following paragraphs some developments are discussed which may result in reactors with fluxes very much higher than the ETR at ETR costs, or MTR fluxes at costs and power levels lower than those for the ORR. These developments are all based on a type of reactor known as a "flux-trap reactor," which consists of a spherical shell or cylinder of fuel surrounded on all sides by moderator. With such an arrangement, the thermal neutron flux peaks in the center region (or island) and in the reflector, while the fast neutron flux peaks in the fuel annulus. In this manner the fast and thermal neutron fluxes are separated, which is highly desirable for doing neutron research. Before describing specific reactors, however, let me review the origin of the flux-trap concept.

HISTORICAL BACKGROUND OF FLUX TRAP REACTORS

The idea of separating the fuel and moderating regions to provide thermal flux peaking in the latter was first suggested by H. Newson in about 1947. This early concept involved slabs of fuel with adjacent moderator. The flux peaking in the moderator or reflector was recognized as the main advantage of such an arrangement. The idea was applied to the design of the MTR to obtain a factor of two increase in thermal flux. This occurs in the beryllium reflector outside of the core.

The first reactor to incorporate an island of moderator within the fuel region was the Thermal Test Reactor (TTR), a 10 kw reactor at Knolls Atomic Power Laboratory designed for detection work in nuclear physics research.⁽²⁾ Provision for an island was also incorporated in the specifications for the Livermore Pool Test Reactor (LPTR) issued in May 1955.⁽³⁾ The arrangement is also used in the Argonaut,⁽⁴⁾ a 10 kw training and research reactor recently constructed at Argonne National Laboratory, and in a 500 kw research reactor described by the Russians at Geneva.⁽⁵⁾ In the Russian reactor a 60-70 cm D₂O island was introduced into a graphite reflected, D₂O cooled core. In the TTR and Argonaut, both the island and reflector were graphite and the core cooled with H₂O.

Thus, the idea of a flux-trap is not new; however, in none of the reactors mentioned has the principle been thoroughly exploited. Parametric studies of optimum dimensions of island, fuel annulus and reflector for various combinations of moderating material in these regions indicate that considerably higher flux peaking can be achieved by proper design than previously thought possible. These parametric studies, carried out at Argonne, Oak Ridge,⁽⁷⁻¹⁰⁾ Internuclear Company,⁽¹¹⁾ and others,^(12,13) were initiated in connection with the design of an Advanced Engineering Test Reactor for producing ultra high fluxes ($> 2 \times 10^{15}$).⁽⁶⁾

The approach to the design of an Advanced Engineering Test Reactor at Oak Ridge was made in an ORSORT study⁽⁸⁾ and by the Reactor Experimental Engineering Division. In the ORSORT study, the idea was to extend MTR technology as far as possible to achieve maximum fluxes. The resulting design, however, looked nothing like the MTR but involved an annular fuel region cooled with D₂O, with D₂O in the island and reflector. An independent study by Argonne National Laboratory resulted in a similar arrangement, both reactors achieving $\sim 5 \times 10^{15}$ thermal fluxes at power levels up to 500 Mw. The design proposed by Internuclear for the AETR involving an H₂O island, H₂O cooled annular core, and D₂O reflector produced an unperturbed thermal flux of 1×10^{15} in the test hole at 35 Mw. A parallel study of a High Flux Homogeneous Reactor⁽¹⁴⁾ indicated that fluxes of about 6×10^{15} could be achieved in such a reactor at a 500 Mw power level. Thus, it became evident that the value of $\theta_{\max}/P$ (maximum thermal flux per megawatt) decreased from about 3.5×10^{13} at 35 Mw to about 1×10^{13} at 500 Mw. On this basis, further studies were undertaken at the Oak Ridge National Laboratory and by Internuclear to find the conditions for maximizing the flux per unit of power. Both spherical and cylindrical systems were investigated as follows:

SPHERICAL SYSTEMS

W. K. Ergen⁽¹⁵⁻¹⁷⁾ has investigated spherical flux-trap reactors consisting of thin shells of fuel in an infinite moderator. The results indicated that for each moderator the maximum flux per Mw occurs when the radius of the island is approximately equal to $2\sqrt{\tau}$, where $\sqrt{\tau}$ is the slowing down length. Fluxes per Mw from this study are summarized in Table 4 for the various moderators considered. It is noted from these results that highest fluxes per Mw are obtained with beryllium or beryllium oxide. H₂O is not given since such a reactor cannot be made critical.

Table 4

IDEALIZED FLUX-TRAP REACTOR WITH RADIUS
ADJUSTED TO GIVE MAXIMUM CENTRAL FLUX

Moderator	Density	Required Multiplication Factor	Radius, cm	$\theta_{\max}/\text{Mw}$
D ₂ O	1.1	1.474	23.3	3.8×10^{13}
Be	1.85	1.944	19.2	7.2×10^{13}
BeO	3.0	1.833	20.4	6.9×10^{13}
C	2.0	1.780	30.5	3.2×10^{13}

The studies made by Ergen involve the same moderator in reflector and island. Studies of spherical systems with varying combinations of H₂O and D₂O in the island and fuel annulus and D₂O in the reflector were carried out by T. B. Fowler.⁽¹⁸⁾ Some of the results of this study are plotted in Fig. 5. As indicated, the highest values of $\theta_{\max}/P$ are obtained with small H₂O islands and thin fuel annuli. Large H₂O islands, however, are less attractive than the large D₂O islands. For the very high power, super flux reactors, therefore, an all D₂O system may be required.

CYLINDRICAL SYSTEMS

Studies of cylindrical flux-trap reactors have been carried out at the Oak Ridge National Laboratory⁽¹⁹⁾ and by Internuclear⁽¹¹⁾⁽²⁰⁾ for both D₂O and H₂O islands. Figure 6 shows values of $\theta_{\max}/\theta_{\text{fuel}}$ and $\theta_{\max}/P$ for cylindrical D₂O reactors in all regions as a function of the U-235 loading based on data from

Fig. 5— ϕ_{max}/P for spherical reactors ($K_{\text{eff}} = 1.0$).

Fig. 6—Flux peaking vs fuel annulus dimensions and U^{235} loading.

Reference 19. The two numbers given represent the radius of the island and fuel annulus thickness in cm, respectively. Thus, each of the points shown in Fig. 6 represents a different reactor geometry. Values of $\theta_{\max}/\theta_{\text{fuel}}$ for varying amounts of U-235 in the reactor may be obtained by assuming a straight line relationship identified by a line passing through the point indicating the dimensions of the island and fuel annulus and the point shown on the ordinate at $M_{25} = 0$. Values of $\theta_{\max}/P$ may be obtained by interpolating between dotted lines. For example, for a reactor with a 30-cm radius island and 20-cm thick fuel annulus loaded with 10 kg of U-235, $\theta_{\max}/\theta_{\text{fuel}} \approx 4.7$ and $\theta_{\max}/P = 1 \times 10^{13}$. For a 20-kg U-235 loading with the same reactor, $\theta_{\max}/\theta_{\text{fuel}} \approx 8.3$ and $\theta_{\max}/P = 8.9 \times 10^{12}$. Thus, it is seen that the thermal flux peaking ($\theta_{\max}/\theta_{\text{fuel}}$) increases with increasing U-235 loadings but the unit thermal flux per Mw ($\theta_{\max}/P$) decreases. The amount of U-235 required will depend on the desired K_{eff} and on the radius of the island and fuel annulus thickness. U-235 requirements for a 30-cm island as a function of K_{eff} and fuel annulus thickness (R_2) are shown in Fig. 7.

SUPER FLUX REACTORS

In order to achieve thermal neutron fluxes significantly above 10^{15} neutrons/cm²(sec), a reactor must operate at power densities and specific powers considerably above those in existing reactors. This means thinner fuel elements, higher coolant velocities, and higher operating temperatures. Calculations indicate⁽⁸⁾ that in a 3-ft long core, power densities up to 4×10^8 Btu/hr (cu ft) or 4100 kw/liter can be achieved with 30-mil thick fuel plates spaced 45 mils apart and a water velocity of 35 ft/sec. Such fuel elements, if constructed of aluminum, may, however, not be strong enough for satisfactory performance. Increasing the fuel plate thickness to 45 mils and the spacing to 50 mils will reduce the achievable power density to 3×10^8 Btu/hr (cu ft). This is used as a basis for estimating the maximum power removable from a given fuel annulus shown in Fig. 8. Allowing a $\frac{\Delta K}{K}$ of 16.6% to compensate for temperature coefficient, equilibrium xenon, samarium, fission product poisons, and experiments (or in other words a $K_{\text{eff}} = 1.20$), calculations were made of the maximum thermal flux as a function of power level for cylindrical D₂O reactors with varying size islands. The results shown in Fig. 9 indicate that a 30-cm radius island gives the maximum flux of 8.8×10^{15} at power levels in the range of 500-700 Mw. Additions of fuel to compensate for burn-up over a 5-10 day fuel cycle will reduce this flux at the start of the cycle to about $6.5-7 \times 10^{15}$. Thus, with an all D₂O reactor, thermal fluxes of 10^{16} cannot be achieved without further increases in power density. One possibility is to shorten the length of the fuel elements to permit higher water velocities. Another possibility for increasing the maximum thermal flux is to use an H₂O island in combination with an H₂O or D₂O coolant and D₂O coolant and D₂O reflector to increase the flux peaking.

HIGH FLUX UNIVERSITY RESEARCH REACTOR

Since the flux per Mw increases considerably as the dimensions of the island and fuel annulus decrease, flux-trap reactors are particularly adaptable to research reactors in the 5 Mw range. The design of such a reactor has been sponsored by the National Science Foundation and carried out by the Internuclear Company for Purdue University.⁽²⁰⁾ The proposed reactor has an annular cylindrical core cooled with H₂O with an H₂O island and a D₂O reflector. The arrangement of the core is shown in Fig. 10. As seen in this figure the core is very small, 5.4-in. ID and 9.5-in. OD, and uses uniquely formed fuel elements. These are 24-in. long and curved to maintain a uniform perpendicular gap, and at the same time completely fill the annulus. Shim and regulating plates are curved to fit the space between the core and reflector.

A plan view of the reactor tank is given in Fig. 11 which shows the location of experimental facilities and the size of the core relative to the reactor tank. Figure 12 shows a vertical section through the tank.

Fig. 7— K_{eff} vs fuel annulus thickness and U^{235} loading.

Fig. 8—Maximum heat power vs fuel annulus dimensions.

Fig. 9— $\phi_{\max}$ in burned out core ($K_{\text{eff}} = 1.20$) cylindrical D_2O - D_2O - D_2O reactors.

Fig. 10—HFUR core mid-plane cross section.

Fig. 11 — HFUR plan view.

Fig. 12 — HFUR sectional elevation.

The neutron distribution for the proposed arrangement is shown in Fig. 13 which indicates a maximum thermal flux of 3×10^{14} in the island and 1.5×10^{14} in the reflector. The fast flux in the fuel annulus is 2.5×10^{14} at the 5 Mw level.

A plan view of the reactor building is shown in Fig. 14, and a vertical section in Fig. 15. In general respects the design is very similar to the ORR. Estimated costs of the reactor building and associated radiation laboratory are summarized in Table 5. It is seen in this table that, exclusive of the radiation laboratory, the cost of the 5 Mw reactor is in the neighborhood of \$1.5 million. Although this is not cheap, it opens the possibility of universities carrying out high level nuclear research at relatively low costs.

Table 5

ESTIMATED COST OF 5Mw FLUX-TRAP REACTOR
(Internuclear Company Design)

Reactor	\$ 680,000
Reactor Auxiliaries	260,000
Reactor Building	600,000
Radiation Laboratory	<u>1,100,000</u>
TOTAL	\$ 2,640,000

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Fig. 13—Radial neutron flux distribution in reactor at 5 Mw.

Fig. 14—HFUR building plan view.

Fig. 15—HFUR building, longitudinal section.

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DISCUSSION

MR. E. L. CURRIER, JR. (ACF Industries): In these island reactors or island cores you get a flux peaking. Is this of value for doing experiments when you have a large absorber in this region or, in other words, when you introduce an absorber, such as a full-scale fuelelement or a loop? Does this not almost negate the peaking effect that you showed on the curve?

MR. LANE: The effect of absorbers in the central region has been investigated. Although the light water islands are less sensitive to the effect of absorbers, a depression in the flux does occur, depending on the neutron cross section and the size of the absorber. The central hole which was shown is only 5 in. in dia. If you put in a stainless steel rod 1-1/2 in. in dia, it would depress the flux by a factor of three. A solid stainless steel rod represents possibly more absorption than one would normally put in such a system. There are many applications which don't use large samples. Of course, beam-hole-type experiments are less sensitive to the effect of neutron absorption by the sample. Although I did not mention it, the particular design shown has six major beam holes, some 4 and others 6 in. in dia. It is possible to design the large, heavy water islands so that the flux depression is of the order of seven-tenths of the maximum flux. By making the island larger the depression gets less. The particular design shown, however, is applicable only for rather small sample irradiations.

MR. CURRIER: Such as capsule type, I presume.

MR. W. H. WILSON (USAEC, Washington): I would be interested in your comment on the role of the fast flux in test reactors. I think, from a practical fuelelement test standpoint, interest is not only in the fission rate by the thermal flux, on which most of your remarks were made, but also in the possible fast flux application in test reactors.

MR. LANE: You will notice that, because of the very high power density in the core, one gets almost as much peaking of fast flux in the core region as thermal flux in the island. If you really want to build a reactor for fast flux irradiations, however, you take out the moderating region in the island which gives a flat fast flux across that region. Such a design was proposed by Babcock and Wilcox and by Lockheed for the Engineering Test Reactor. But for certain small irradiations again, as in the case of thermal flux exposures, the design described is very applicable for fast flux irradiations.

MR. C. D. GOODMAN (USAEC, Washington): Your fuel elements to a certain extent depart from Eugene Wigner's way of getting away from the vibration of the plates due to the motion of the water. I wonder if any analysis of the behavior of the constant curvature plates in high velocity water has been made.

MR. LANE: The water velocity for this particular 5-Mw reactor is not high. It is 20 ft/sec and should not cause the plates to vibrate.

MR. GOODMAN: How does that compare with MTR?

MR. LANE: The water velocity in the MTR is 30 ft/sec. The proposed fuel elements are probably as rigid as the MTR fuel plates. In both cases the amount of curvature is about the same. No analysis has been made of the hydrodynamics of the flow in the gap, however. Although the distance between adjacent plates is constant in the perpendicular direction to the gap, it changes in the circumferential direction.

MR. GOODMAN: It would be difficult to calculate, I would think.

MR. LANE: I feel that the proposed fuel elements would work very satisfactorily, since the heat removal from this particular high flux reactor is not great. As a matter of fact, the power density is less than the MTR. This was done in order to keep the heat removal cost at a minimum.

MR. R. J. BEAVER (Oak Ridge National Laboratory): Those plates that you are referring to, are they 30 mils thick in design?

MR. LANE: No, in this particular design, the fuel plates are 50 mils thick with a 98-mil water gap. In other words, the design is just standard technology. It is only when you go to the superflux reactor that it is necessary to reduce the plate thickness and the gap width and in such a case you have to push as far as you can go. I think 30 mils may be stretching the technology a little. However, one idea suggested by C. E. Winters and probably by others is to emboss the plates by one of the embossing procedures such as is used for metal cloth. This gives them rigidity without increasing the amount of metal. Although the embossing is, say, only 5 mils thick, it would increase the rigidity and permit one to use a thinner plate.

MR. BEAVER: I might mention that in some slides for the talk I am presenting we have some data on the strength of fuel elements that have 36-mil plates.

MR. LANE: Has anyone looked at these embossed plates?

CHAIRMAN CUNNINGHAM: Not yet.

MR. LANE: Another nice thing about embossing the plates is that you probably get very much lower film drops and higher heat-transfer coefficients. The uniform ripple on the surface would cut down the film resistance. It might increase the pressure drop, however.

CHAIRMAN CUNNINGHAM: You were talking about getting a high fast and high thermal flux using light and heavy water -- do you recall what the ratio is in the MTR?

MR. LANE: Well, normally you get a very low fast flux from a heavy water-moderated reactor. In the case of the light water MTR the virgin fast flux is about one-half as much as the thermal flux.

CHAIRMAN CUNNINGHAM: Yet in all the experimental reactors mentioned you said the designs were going to use heavy water. Is this because they are not interested in the fast flux?

MR. LANE: The designs do not all use only heavy water. I said that in the studies that are directed toward the superflux reactors, the problem is to make the fuel annulus large enough to get the heat out. With light water in such a case the neutrons don't leak into the island and you don't get much peaking. Thus you have to use heavy water coolant and reflector or some other very good reflector. The trend in the research reactors that are being built these days has been in the direction of light not heavy water. As a matter of fact, the pool- and tank-type reactors are increasing in number more rapidly than the heavy water types due to their lower cost.

Paper 3

FUEL ELEMENT PROBLEMS ASSOCIATED WITH OPERATIONS OF THE ENGINEERING TEST REACTOR AND MATERIALS TESTING REACTOR

M. H. BARTZ

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MATERIALS TESTING REACTOR FUEL ELEMENT EXPERIENCE

Since 1952 the MTR has operated with no fuel element failure that can be directly attributed to the changes in properties due to the uranium burnout or to general radiation damage. Since its conception, the MTR has consumed 1300 fuel elements with an average burn-up of 25 to 30%. During this period the original 140-g - 18 plate fuel element has given way to a 200-g - 19 plate fuel element. Probably most of you are familiar with it but very briefly the element is a three by three, 24-in. fuel length of curved plates. The core consists of a highly enriched uranium in a matrix of high purity aluminum. The cladding and side plates are 1100 aluminum and, of course, the end boxes are a cast aluminum alloy.

From the start-up in 1952 until the summer of 1954 all of the fuel elements functioned properly. The original 140-g assembly was intended for a reasonably short burn-up i.e., one reactor cycle of three weeks. However, during the 1952-1954 period many developments were made and many changes tried. A 140-g element, a 168-g element and a 200-g element were tested for five to ten cycles with a total burn-up of 75 to 85% of the fuel. All of these elements performed properly as far as the metallurgy was concerned. In June of 1954, however, a gradual increase in the fission product activity in the process water was noted and was finally attributed to the fuel element.

Figure 1 shows one of the fuel elements that was examined. An elliptical-shaped blemish, dark in the center and light gray around the outside, is apparent an inch or so from the lower end of the element and extending two or three inches. The next Fig. (2) shows a close-up of this blemished area. A smear taken of this area gave evidence of fission product activity and indicated that the fuel core had been exposed.

Figure 3 shows some outward buckling of the concave fuel plate in the direction of the adjacent fuel element and in the same area of the blemish. It is evident that these two fuel plates had been in contact and that local overheating then caused the corrosion of the aluminum cladding and the release of the fission products.

Figure 4 summarizes some of the changes that took place during this two-year period and that later were adjudged to be the cause of the fission breaks.

Fig. 1—MTR fuel element showing area of fission break.

Fig. 2—MTR fuel element showing close-up of fission break area.

Fig. 3—Buckling of concave plate on MTR fuel element.

LOWER END BOX

ORIGINAL

a-0.060
b-0.060
c-0.1875
A-3.7 SQ. IN.

INTERIM

.050
.050
.106
3.2 SQ. IN.

FINAL

0.0625
0.050
0.1875
3.7 SQ. IN.

Fig. 4—Development changes in MTR fuel element.

Originally the fuel element was composed of all 60-mil fuel plates, the outside as well as the inside plates; the side plates being 187.5 mils. The cross section at the right shows the bottom end box. Originally the cross section area of this end box was 3.7 in.² In the transition that took place as many changes were tried and a fuel element evolved with 50-mils-thick fuel plates and 106-mils side plates. In addition the cross section area of the bottom end box was reduced to 3.2 in.² Thinner fuel and side plates, of course, resulted in a weaker element and the reduced cross-section area of the end box caused some restriction in flow with corresponding increase in internal pressure. After it was determined that the fuel element design changes had indeed caused the fission breaks in the reactor, MTR fuel elements were stabilized at 50-mil inside plates, 62-1/2-mil outer plates, and 187-1/2-mil side plates. The end boxes were cut out to obtain the original 3.7-in.² cross section.

Figure 5 presents somewhat the same picture in terms of changes that occurred in the lateral pressures. It is observed at the left, that there was approximately 1 psi differential pressure across the outside fuel plate in 1952 (9.1 minus 8.1). In 1954, on what we might term the interim fuel element design, the pressure across the outside fuel plate was 10 psi. Static pressure tests made at that time indicated that the strength of the 50-mil outside fuel plate was only about 9 psi before some permanent buckling occurred.

After return to what was essentially the old design, the lateral pressure differential returned to approximately 2 psi. Tests on this design showed that the element was capable of withstanding 23 psi before any buckling occurred. Since that time the MTR has been operating at 40 Mw with no fuel element trouble.

ENGINEERING TEST REACTOR FUEL ELEMENT EXPERIENCE

The original ETR fuel element was approximately 3 in. x 3 in. x 54 in. containing 19 flat fuel plates each 50-mils thick by 37-in. long separated by a 110 mil channel. The element was supported in a grid by a cylindrical bottom end box. All of the first elements received were within the tolerance specified. However, the very early out-of-pile hydraulic tests indicated a structural weakness in the fuel element. These tests demonstrated the elements' inability to withstand the hydraulic forces accompanying the design coolant velocity of 35 ft/sec. A full core loading in the reactor verified the out-of-pile test results and indicated that the ETR would not be able to operate at designed power on these early elements. In addition, a velocity distribution measurement indicated that some channels would be as low as 25% below the average velocity through the nominal channel. Therefore, it was decided that the ETR would start up at approximately 80 Mw, equivalent to about 18 ft/sec of the coolant flow. Two cores, using the standard fuel element as designed, were operated at this 80 Mw level.

During the period of these core loadings a considerable amount of development work was performed on the fuel elements in an attempt to achieve the design power of 175 Mw. Four types of fuel elements in particular were tried; all assembled by mechanical means. GE-APED produced an element in which the fuel plates were roll-swaged to the side plates. This resulted in a good element with flat plates and straight channels. A second type was produced by Babcock and Wilcox based upon a pinning procedure. Figures 6 and 7, show the fuel plates assembled into the side plates by 70-mil pins running through the edges of the side plates at approximately 1-in. intervals. A temporary expedient of cutting slots into the side plates of brazed fuel elements, was tried by preparing modified slotted elements in the ETR shops. These slots were 4-in. long, each about 30-mils wide.

Fig. 5—Lateral pressure changes accompanying development changes in MTR fuel element.

Fig. 6—ETR fuel element showing pinned assembly.

Fig. 7—ETR fuel element close-up of pinned assembly.

Figure 8 shows this modification as later produced by Babcock and Wilcox. The slots in this case are 2-in. long, 3/16-in. wide and staggered so that all the plate channels are covered.

Figure 9 shows a close-up of the same thing except that in this case the element is slotted and also assembled by pinning. The pins are about 4-in. apart, going through the unslotted area. The object of these experimental elements was to eliminate the brazing cycle which results in a lower strength element than one can obtain by retaining the cold work of the last passes of the rolling procedure. An 1100 aluminum alloy fuel element as-brazed has a yield strength of about 5,000 psi. If a 15% reduction in thickness of the plate is obtained in the cold rolling, it is possible to reach yield strengths of 14,000 to 15,000 psi. The object of the slots was merely a means of relieving any hydraulic lateral stresses as they occurred.

The results of the hydraulic tests on these elements indicated that all were capable of withstanding 140% of the design flow, (roughly 48 to 49 ft/sec with approximately 100 psi pressure drop across the element). Accordingly, a core loading of solid side-plate pinned elements was purchased and operated for 2900 Mw days for a burn-up of approximately 20 or 25%; most of this at 175 Mw. A core loading of slotted side-plate pinned fuel elements is available for the next loading.

As mentioned earlier, the velocity distribution also served to limit the maximum power. A velocity profile taken of five channels indicated a variation from perhaps 80 to 105% of the average value at the center of the channel while the corners might be as low as 70%. In order to improve the velocity distribution, changes were made in the end box to decrease the flow restrictions for the outer channels. The end box was shortened, bosses were reduced, and slots inserted. These changes increased the minimum velocity to 90% of average but disturbed the distribution of water through the grid.

The spacing between the fuel plates was next varied to reduce the flow variation. Originally all the plates were spaced about 110 mils. The outer three channels were increased to 120 mils, reducing the remaining plate spacing to 105 mils. In addition, the spacing between adjacent fuel elements was increased; originally the specifications called for 90 to 150 mils. After the tests, a change in the specifications to 105 to 170 mils was recommended.

All of these changes gave some reduction in the variation of the velocity; the average velocities varying not more than approximately 5%.

The final problem that plagued the early ETR fuel element was the control of burnable poisons. As originally specified, the boron content in the core was to be 1.55 ± 0.2 I g. The received elements varied from 0.6 g to 2.5 g; part of the variation being in the determination. A neutron scan proved not very successful. The chemical procedures available at the time were not adequate. Satisfactory methods were developed by the analytical group using a core sample prior to cladding. A method was worked out in the critical facility in which a boron-free element was measured and then known amounts of boron taped to the plates.

Presently the use of sinter-cast is being investigated. This is a boron-carbide-aluminum material for use in the side plates. Ten elements incorporating this material have been ordered to be tried in the reactor.

This report has been a brief summary of some of the fuel element operating problems in the MTR and ETR. Some of these problems, particularly in the ETR, still continue. The development of an improved fuel element, however, can now proceed while the reactor operates at design power.

Fig. 8—ETR experiment fuel element showing slotted side plates.

Fig. 9—ETR experimental fuel element close-up of slotted-pinned assembly.

DISCUSSION

W. C. FRANCIS

MR. R. J. BEAVER (Oak Ridge National Laboratory): Do you have any values in mind of the uncertainty in the procedure for analyzing the boron concentration?

MR. FRANCIS: I just don't recall what they are, but this procedure has been written up as a standard by the chemical analytical people and they have quoted values. This can be made available to you. It has been made available to the manufacturer.

MR. J. H. KITTEL (Argonne National Laboratory): I wonder if you have any information as to how well the cold work in the plates holds up during irradiation.

MR. FRANCIS: The proof is probably in the first core, which was being unloaded as I left Idaho Falls. The elements were inspected, however, after every shutdown and we could not detect any weaknesses.

CHAIRMAN CUNNINGHAM: Since aluminum recrystallizes around 350°C generally for heavy cold work, they are probably all right.

MR. C. W. CONANT (National Advisory Committee for Aeronautics): Are all these later elements flat plates or curved?

MR. FRANCIS: They are all flat plate. It would be a major change-over to curved plates with rods and reflector being designed for flat plates.

MR. L. R. WEISSERT (Babcock and Wilcox Company): I have two questions about the MTR. First of all, if I understand correctly, your shim-control rods had fuel sections in them.

MR. FRANCIS: That is right.

MR. WEISSERT: I have a question about what the average burnout of these fuel sections is. Secondly, I would like to get some elaboration on these high burn-out test plates -- specifically their composition and how they are examined after irradiation.

MR. FRANCIS: To answer the first question on the control rods, the four rods in the center of the core are fuel-cadmium rods. We do have some safety rods that are beryllium-cadmium. The fuel-cadmium rods are normally removed when they no longer have sufficient reactivity control, and I would guess that the fuel burn-up is of the order of magnitude of that for the fuel elements. We are going through a development program on our shim rods -- on the fuel shims -- and have hopes of increasing their reactor lifetime considerably.

To answer the second question on the examination of these plates, most of the examinations, particularly during the time of the fission breaks, were made in the canal. This, of course, is not a very precise method. Then, after three to four months of cooling, some of the fuel elements were examined in the hot cell. This is where some of our details came from.

MR. WEISSERT: This includes the high burn-up test plates?

CHAIRMAN CUNNINGHAM: These are not test plates. These are regular elements that went through the loading. They were tagged and requested to be taken to excessive burn-ups, and they just held them in the core of neutrons, but nevertheless they used full size units -- regular production fuel elements.

MR. WEISSERT: And they were examined in the hot cells?

CHAIRMAN CUNNINGHAM: Some of them were. They sectioned plates out, but this was a long time ago and there was not a great deal of work done. It was more of a confidence test than it was a test for radiation damage.

MR. FRANCIS: Actually most of our core tests, unless we have a full core loading, are just that -- confidence tests -- does the element stand up for three cycles, five cycles, or some other value in the core.

MR. A. R. KAUFMANN (Nuclear Metals, Inc.): You spoke of a roll-swaging process for assembling the boxes. What do you mean by that?

MR. FRANCIS: I understand it is a matter of putting in one plate at a time and actually crimping or forcing the side plate metal down onto the fuel plate. Mr. Beaver is going to be talking about some methods of assembly and perhaps he is more familiar with this than I am.

CHAIRMAN CUNNINGHAM: They are assembled just like the regular elements are and put in a milling machine where they run a burnishing tool down and force the lip on the side plate down so it locks the plate in place. It is a rolling effect to give you a burnishing. It is not a cutting operation; it is a deformation operation.

MR. J. E. MINOR (Hanford Atomic Products Operation): I am just wondering if you can translate a couple of figures. You talked about per cent burn-up. Was that per cent burnout of total atoms?

MR. FRANCIS: I don't have the equivalent in atom per cent. My figures for the MTR example start with a 200-g element and burn the fuel down, say to 150 g. We say this is 25% burn-up.

CHAIRMAN CUNNINGHAM: Twenty to 25% burn-up is about 1% of the total atoms in the meat.

We see now that we have a fuel element that apparently is working fairly well. We try to optimize it to get lower metal-to-water ratio, to increase the fueling to get longer life, to take out burnout poisons, etc., and we run into problems and then we have to back off. This is the type of work we are going to get into with a lot of our older and more standard-type fuel elements.

Paper 4

RECENT DEVELOPMENTS IN ALUMINUM-BASE FUEL ELEMENTS FOR RESEARCH REACTORS

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I. INTRODUCTION

The purpose of this report on the subject of aluminum-base fuel elements for research reactors is essentially two-fold. First, an attempt will be made to dispel the common belief that the service requirements of Al-clad fuel plate elements in pool- and tank-type reactor applications are essentially the same by pointing out major differences in operating conditions and by discussing the limiting or life-determining factor in each case. Secondly, the recent and significant developments in the field of aluminum-base fuel elements designed for research reactors will be described. It is hoped that by making designers, fabricators, and reactor operators aware of the important differences between these two applications, as well as the new developments, fuel element failures such as have occurred in pool-type reactors during the past few years can be avoided in the future.

The aluminum-base fuel element of curved-plate design was the product of an Oak Ridge National Laboratory program aimed at advancing nuclear research and development. The component was first utilized in December of 1950 to power the Bulk Shielding Reactor, a light water-cooled, -moderated and -reflected reactor of 1 Mw thermal capacity and of pool-type design. The component was next used in the Materials Testing Reactor, a tank-type reactor which employs forced-circulation of the coolant to promote efficient dissipation of the 30 Mw of heat developed in the reactor core. The performance and reliability of this fuel element stimulated its popularity throughout the subsequent years to the extent that the element as originally constructed has approached a contemporary status. More than 30 reactors, mostly of the pool-type variety are fueled with this type of element, and it is estimated that since 1950 approximately 3,000 of these components have been manufactured.

As illustrated in Fig. 1, the more popular version of the pool-type element is an array of 18 composite roll-clad plates which are spaced at 117-mil intervals and rigidly fixed by brazing to a pair of spacer plates with an 11.6% Si-Al alloy. The plates are clad with 20 mils of type 1100 aluminum to protect the underlying 14% U-Al alloy. The MTR Mark XI element differs in that it consists of 19 composite plates, each containing an 18.5% U-Al alloy in the core or fuel-bearing section. The 17 inner plates are 50 mils in over-all thickness with a clad-core-clad thickness of 15-20-15 mils. The outer plates, however, are 65 mils in over-all thickness clad with 22.5 mils of aluminum.

Fig. 1—Over-all view of BSR fuel element.

II. LIFE-DETERMINING FACTORS

Pool-Type Reactor Applications

Because of the differences in application, the pool-type and tank-type research reactor fuel elements are not alike and the design changes, although slight, are significant. This distinction cannot be over-emphasized. Fabricators, and especially designers, should be aware that the limiting condition with pool-type elements is corrosion of the aluminum. The negligible corrosion rate in low temperature and relatively quiescent water is misleading. The most significant effect observed during examination of irradiated pool elements as well as out-of-pile specimens has been pitting of the aluminum. This condition is attributed mainly to improper quality of the water. If pitting is to be held in check, it is desirable that demineralized water, with a pH between six and seven and a resistivity of 300,000 ohm-cm be maintained.¹ Experiments with sodium and potassium dichromate additions to the demineralized water appear effective in reducing pitting corrosion; however, sodium and potassium both present activity problems and the chromate ion must be maintained to be effective.² Alodizing also acts to retard deleterious corrosion effects and experience indicates that it extends the life of the element at least six months.³ For the past eight years, all fuel elements for the ORNL Bulk Shielding Reactor have been alodized. Figure 2 compares two non-alodized BSR pool elements which were in pool water for approximately two years. The lower element was in water of less than 150,000 ohm-cm. The upper element was exposed to demineralized water of 200,000 to 300,000 ohm-cm. The difference is striking. Granted, appreciable pitting occurs along the brazed joint, but significantly, pits greater than 20 mils deep were observed in the cladding of the element exposed to water of low quality.⁴

Tank-Type Research Applications

The limiting factors in tank-type reactors are reactivity lifetime and burn-up rate of the U-235 atoms rather than corrosion. In an effort to extend lifetime, the U-235 content of the MTR fuel element has progressed from the orig-

inal 140 g to the present 200 g unit. The life of the present element is generally less than six weeks. In this type of research reactor, the mechanical strength of the fuel element is also an important factor. At the required water velocities, potential pressure differentials may occur which cause a deleterious permanent deformation of the fuel element. Such a condition did in fact occur during operation of the MTR with an element modified to improve the metal-to-water ratio. Increasing the thickness of the outside fuel plates and side plates corrected the problem. The substitution of high strength aluminum alloys for type 1100 aluminum not only should add a good margin of safety but offers the potential of thinner plates — thus a more favorable metal-to-water ratio. Operation of SPERT "B" in 1955 with a special loading of fuel elements manufactured at ORNL demonstrated the structural advantages of substituting air-hardened type 6061 for type 1100 aluminum alloy.⁵ The tensile strength of the type 6061 aluminum in this condition averaged 35,000 psi.

Recent experiments were conducted in which high-strength aluminum alloys were substituted for type 1100 aluminum in MTR-type elements and the elements tested under differential pressure conditions. The results, shown graphically in Figs. 3 and 4, reveal the improvement with the substitution of the high strength alloys.

III. UTILIZATION OF 20% ENRICHED FUEL

When the limitation of 20% enrichment was established for export of uranium under the Atoms-for-Peace Plan, it soon became apparent that a problem of major significance was created. Such a specification automatically necessitates a five-fold increase in uranium content in the fuel element and generally requires

Fig. 2—Comparison of full-size BSR fuel element exposed approximately two years to good and poor water conditions. Pitting greater than 20 mils in depth was observed in the bottom element which was exposed to the poor quality water.

increasing the concentrations of uranium from 14/18 wt % U to 35/50 wt % U range. Such alloys contain nearly 50% brittle intermetallic compounds. High-uranium concentration requirements suggest powder-metallurgy practices and a program was launched not only to develop aluminum-uranium alloys containing 40 to 50 wt % U, but also dispersions of uranium carbides and oxides containing up to 45 vol % of the fissile compound. The program was directed toward the utilization of the conventionally accepted manufacturing practice which includes brazing of the fuel elements. Thus, compatibility of the fuel compounds with aluminum at 600°C was mandatory.

Fig. 3—Deflection of top and bottom plates of brazed curved plate aluminum fuel elements after designated differential air pressurization. Age-hardening alloys solution treated, air quenched, and aged.

Fig. 4—Permanent set of top and bottom plates of brazed curved plate aluminum fuel elements after designated differential air pressurization. Age-hardening alloys solution treated, air quenched, and aged.

Development of High-Uranium Content Alloys

In the development of 40 to 50 wt % U-Al alloys, a significant contribution was made when silicon was used to completely suppress the UAl_4 inter-metallic compound, which results in a material consisting of $U(Si, Al)_3$ in a eutectic matrix. A comparison of the effect on microstructure is vividly portrayed in Figs. 5 and 6. Figure 5 shows the high concentration of the UAl_4 phase in the microstructure of the 48% U-Al alloy. Figure 6 illustrates the marked improvement when 3% silicon is added, demonstrating the well defined UAl_3 compound in the eutectic matrix. Mechanical tests of the silicon-modified alloy showed a slight gain in ductility but a 25% decrease in yield strength which significantly relieved the fabrication problems. The disadvantage associated with the use of this alloy is mainly the low yield ($\sim 35\%$) from casting to finished product and the resulting high recycle of scrap. Considerable difficulty has also been experienced in brazing composites containing this alloy into fuel assemblies within the generally accepted 10% tolerance allowed for the plate spacings.

Development of High-Uranium Content Dispersions

Investigations have been conducted with dispersions of UC_2 , U_3O_8 and UO_2 in aluminum. One of the main requirements is that the fissile compound be compatible with aluminum at 590°C to 610°C which are the conventional fabrication and heat-treatment temperatures used in manufacturing plate-type aluminum fuel plates. An important criteria for determining compatibility is dimensional stability of the composite plate. The normal length of time the plates are subjected to such temperatures during processing is three to four hours. It is also important that the fuel compound be homogeneously dispersed throughout the active section of the plate and the dispersion must be relatively corrosion resistant if inadvertently exposed to the water coolant.

UC_2 Powder Dispersion

Results of investigations with dispersions of uranium carbides in aluminum revealed that carbides containing approximately 9% C were more compatible with aluminum at 620°C than carbides containing approximately 4% C. It was also shown, by thermodynamic calculations supported by experimentation, that blisters in the fabricated and heat-treated composites containing 18 vol % dispersions of UC_2 in aluminum could be traced to the reaction of uranium carbide with hydrogen to form methane. Although the blister problem was eliminated by vacuum degassification of the fuel cores prior to roll-cladding and it was demonstrated that fuel elements containing dispersions of UC_2 in aluminum could be manufactured, corrosion tests in 60°C and 90°C water revealed that this type of element was risky as a fuel material for water-cooled research reactors. Figure 7 illustrates the remains of a deliberately defected specimen containing a dispersion of 16 vol % UC_2 in aluminum, after 24 hr in 60°C demineralized water. The catastrophic nature of attack is apparent. In comparison, the same type of specimen, containing 48% U-3% Si-bal Al alloy showed slight evidence of attack after six weeks in 90°C distilled water. Specimens containing dispersions of U_3O_8 in aluminum appeared unaffected.

U_3O_8 Powder Dispersion

Of the three fissile compounds considered, U_3O_8 appears most attractive for the conventional manufacturing practice. As shown in Fig. 8, experiments with hydrothermal U_3O_8 have revealed that full size plates, containing dispersions of 56 wt % U_3O_8 with a particle size range of 44 to 149 μ and those with less than 44 μ , are dimensionally stable after as long as 160 hr at 600°C. Latest experiments with relatively inexpensive U_3O_8 recently developed at Y-12 have indicated that this material is comparable to the hydrothermal grade of U_3O_8 in its dimensional stability under heat treatment at 600°C. In contrast, plates

Fig. 5—Microstructure of 40 wt.% U-Al alloy after 75 per cent reduction in thickness at 600°C from casting to plate.

Fig. 6—Microstructure of 48 wt.% U-3% Si-Al alloy after 75 per cent reduction in thickness at 600°C from casting to plate.

Fig. 7—Catastrophic attack of exposed dispersion of UC_2 in aluminum matrix of Al-clad fuel plate after 24 hr in $60^\circ C$ demineralized water.

Fig. 8—Growth characteristics of 52.3 wt % UO_2 and 55.7 wt % U_3O_8 aluminum fuel plates.

containing 20 vol % hydrothermal UO_2 , as well as those with fused UO_2 , show a significant increase in volume after four hours. In addition, recent experience with composite plates containing Mallinkrodt high-fired UO_2 (with an 0.3% TiO_2 addition), a reportedly good but expensive oxide for this specific purpose, revealed a 4% volume increase after four hours at 600°C and 1% after one hour. It is obvious that if present brazing practices are to be followed, high-priced UO_2 is required, and still has marginal compatibility. This high cost of oxide places such a fuel element in an unfavorable economical position.

IV. MECHANICAL JOINING OF FUEL ELEMENTS

Mechanical joining of elements has been considered for some time as a promising substitute for brazing. Preliminary work at ORNL indicates that it is quite feasible to mechanically join the composites into side plates with joint strengths of $\sim 2,000$ psi. A cross section of mechanically produced joint in a recent experimental fuel element is shown in Fig. 9. Independent efforts at Babcock and Wilcox, Lynchburg, Virginia, resulted in an acceptable pinned element for the Engineering Test Reactor⁶ and during the same period General Electric at San Jose, California, developed a mechanically joined element.⁷ Eliminating the rather tedious and expensive brazing method offers three distinct advantages: (1) complete elimination of lithium entrapment which is characteristic of brazed fuel elements, (2) greater flexibility in heat treatment of fuel plates, and (3) lower fabrication costs. Properly designed, mechanically-joined fuel elements may be the ultimate for both tank-type and pool-type reactors. In pool-type reactors, where life is dependent upon corrosion, the innovation of a mechanically joined element must be treated with caution because of potential crevice corrosion. A recent crevice corrosion test conducted at ORNL for six weeks in highly purified distilled water of 1,000,000 ohm-cm resistivity revealed that significant pitting occurred.⁸

V UTILIZATION OF A BURNABLE POISON

The burnable poison, boron, has been considered as an element for controlling excess reactivity and flux perturbations in high-flux aluminum research reactor fuel elements. In fact, the ETR is presently using a plate-type aluminum fuel element in which boron is incorporated in the uranium-aluminum alloy.⁹ The results of experimentation at ORNL revealed that the major problem in alloying boron is the uncertainty in determining the exact boron content because of unpredictable losses and segregation during melting and casting of the alloys.¹⁰ An evaluation of various boron-bearing aluminum alloys is presented in Fig. 10. It can be observed that utilization of nickel boride as the additive results in alloys with the better ratings. Although no actual experimentation was conducted on the dispersion-type fuel element in which B_4C and U_3O_8 are incorporated in aluminum by powder-metallurgical practices, it is apparent that by such processing very accurate predictions of the boron content is possible.

VI. SUMMARY

In relatively quiescent water, typical of pool-type reactors, aluminum fuel elements will corrode at a relatively rapid rate if the water purity is not maintained. It is recommended that the water be demineralized and that resistivity and pH of the water be controlled between 300,000 to 500,000 ohm-cm and six to seven, respectively.

In tank-type reactors, the resistance of the fuel element to pressure differentials is important. Substitution of high strength aluminum alloys significantly improves the strength of the presently designed tank-type reactor fuel element.

At the present time, fuel elements containing 20% enriched uranium can be manufactured by utilizing a 35 to 50 wt % U-Al alloy modified with silicon. The

Fig. 9—Cross section of four joints in mechanically joined experimental fuel element.

THE EFFECT OF PREPARATION ON RATING OF BORON-BEARING ALUMINUM ALLOYS AND DISPERSIONS				
Boron Addition Agent	Composition Alloy or Dispersion, wt. %	% Recovery	Index of ⁽¹⁾ Homogeneity	Rating ⁽¹⁾
AlB ₁₂	17 U - 0.22 B-Al	46	56	53
5% B-Al	17 U - 0.22 B-Al	74	76	75
11.5% B-Ni	17 U - 0.22 B-Al	89	88	88
5% B-Al	0.10 B-Al ⁽²⁾	74	90	85
11.5% B-Ni	0.10 B-Al ⁽²⁾	90	88	88
2 to 5% B-Al	0.10 B-Al ⁽²⁾	99 ⁽³⁾	94 ⁽³⁾	95 ⁽³⁾
B ₄ C Powder	20 U ₃ O ₈ - 0.28 B ₄ C-Al	99.9	100	99.9

⁽¹⁾ 100 represents maximum.

⁽²⁾ Fuel plate cladding.

⁽³⁾ Based on 66% casting-to-plate yield.

Fig. 10—The effect of preparation on rating of boron-bearing aluminum alloys and dispersions.

element suffers an economical disadvantage compared to highly enriched fuel elements because of additional expense in fabrication.

Dispersions of uranium dicarbide in aluminum can be utilized in manufacturing pool- or tank-type fuel elements. However, in the event of an inadvertent defect in the cladding, the fuel material will corrode catastrophically when exposed to the water coolant.

Dispersions of uranium dioxide are marginal for applications in which composite plates are to be roll-clad and processing temperatures of 600°C are specified. Under these conditions, this type of dispersion is also economically unattractive.

Fuel elements containing dispersions of U_3O_8 in aluminum appear to be the most economical fuel elements, in which 20% enriched uranium is required.

Mechanical joining is a promising replacement for brazing fuel elements. The chief advantage is elimination of the nuclearly poisonous lithium. A cautious approach should be taken for a pool-type reactor application until crevice corrosion has been fully evaluated.

Alloys of aluminum containing both uranium and boron, or boron alone, are difficult to prepare because of boron losses and segregation during melting and casting operations. Incorporating the burnable poison and fissionable fuel as B_4C and U_3O_8 powders in aluminum by powder-metallurgical processing offers an attractive method to circumvent the problems associated with boron-bearing aluminum alloys.

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DISCUSSION

R. J. BEAVER

MR. J. L. KLEIN (Nuclear Metals, Inc.): It is not clear to me how you achieved mechanical bonds to the side plates.

MR. BEAVER: As I said, we are still in the preliminary development state, but we have made 12-plate elements. We are working on 18-plate elements. We did this simply by applying pressure to the edges of the side plates.

MR. KLEIN: Do you mount it?

MR. BEAVER: Yes, we assemble each plate individually and then apply pressure.

MR. S. SIEGEL (Metals and Controls, Inc.): Would you care to make any general comments about the yield, particularly with respect to blisters on these alloy aluminum-clad plates as compared with the standard aluminum-clad plates?

MR. BEAVER: We really did not fabricate on a production basis. We did this more on the research and development scale. As far as blister rejections were concerned, they were probably not any greater than for the regular highly enriched alloy with 1100-type aluminum. One thing that is troublesome with this particular alloy is the segregation pattern. The segregation of these alloys is markedly inferior to the segregation or the homogeneity, you might say, of highly enriched alloys. It is also necessary to recycle quite a bit of material, and I estimated that our casting to core yield was in the order of 40% compared with about 70% yield with the highly enriched material. Now in the case of the dispersion type, nearly a 100% yield from input to finished cores is possible. If you are careful in fabrication, the rejects can be very low, and we have indications that we don't have to flux anneal the oxide-bearing plates.

MR. KLEIN: To pursue this last remark of yours a little further, do you have to flux anneal the other plates?

MR. BEAVER: You have to flux anneal unless you vacuum cast your alloys. The more conventional technique in the field has been to air melt these alloys.

CHAIRMAN CUNNINGHAM: And to roll them in ambient atmosphere rather than completely evacuating. This is all open air -- more commercial.

MR. KLEIN: Does UO_2 react with aluminum? Is that why you use U_3O_8 ?

MR. BEAVER: All types of UO_2 studied will react with aluminum at 600°C , forming Al_2O_3 and the U-Al intermetallics. The relative reactivity of the UO_2 depends upon its method of preparation. Sylvania has reported, for example, that UO_2 types made with CaF_2 or TiO_2 additions are less reactive than other types. We did most of our work with UO_2 prepared by the reduction of $\text{UO}_3 \cdot \text{H}_2\text{O}$. We consider it to be marginal, however. We have shown that after as little as 4 hr at 600°C we get a 4% vol increase in our plate. We feel that a 1% vol increase makes the

material marginal. The dispersion containing UO_2 with the TiO_2 in it shows about a 1% increase and we consider that marginal. We have not yet looked at any of the CaF_2 types.

MR. KLEIN: What happens to the UO_2 that forms from the reaction of U_3O_8 with aluminum?

MR. BEAVER: In the time period that one would ever think about for fabrication, all that happens is that a certain amount of U_3O_8 converts to UO_2 , but we have never detected an intermetallic compound which one might presume to occur as the result of the reaction of this UO_2 with aluminum. We have found that after a time the UO_2 does, in fact, react to form U-Al_4 , but this is after longer times at 600°C .

MR. R. C. WAUGH (Oak Ridge National Laboratory): Fuel plates containing 55.7% U_3O_8 -aluminum, with the U_3O_8 being -100 + 325 mesh and -325 mesh, have been heat treated for extended times at 600°C . After 600 hr the -325 mesh U_3O_8 has completely converted to UO_2 and a large portion of this reaction-product UO_2 has reacted in turn to form UAl_4 . Similar x-ray analysis of the -100 + 325 mesh U_3O_8 fuel plate reveals that the U_3O_8 has completely converted to UO_2 , but no U-Al intermetallic compounds were detectable. These data demonstrate the importance of the respective surface/vol values.

MR. BEAVER: It is very interesting. Another thing with UO_2 is that it is necessary to use elevated heat treatments -- and some of the people tell me that fabricators are required to blister-anneal plate at 600°C to demonstrate bonding. If you have to do this with UO_2 you may be in marginal operation. In order to use UO_2 you have to utilize an expensive UO_2 .

CHAIRMAN CUNNINGHAM: Mr. Beaver only summarized this work because it was presented in even more detail at the Paris meeting. Of course, when you use the high concentration alloy you have the problem of segregation. You have the mechanical deformation problem. You have to circumvent that problem. When you use the oxides then you are concerned about the chemical reaction of the UO_2 with aluminum. Of the two, the UO_2 and U_3O_8 , we prefer to work with the U_3O_8 because we have a longer margin of time to work. It is stable over a much longer period.

MR. KLEIN: I have one other question. I noticed you got fragmentation and void formation associated even with the 3% silicon-modified 48% uranium alloy. This is the same sort of thing we notice when we extrude that alloy. I wonder if you have any trouble with fission gases collecting in these cracks in the compounds and in the voids just behind or around the compounds?

CHAIRMAN CUNNINGHAM: We had one full-sized regular element run through the MTR, but it was never examined in the detailed sense that you are talking about. We looked at it and measured the plate spacing and the gapping and it was right within our experimental error that we took on the measurement before the fuel went in -- before the aluminum went in. We have to do failure analysis on the materials to look for that sort of thing, which has not been done.

Paper 5

DEVELOPMENTAL EFFORT TO PRODUCE A TUBULAR FUEL ELEMENT WITH VARIABLE FUEL LOADING

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I. INTRODUCTION

Tubular fuel elements which can be joined into bundles without external supports offer many advantages over conventional rod and plate-type assemblies. Some inherent and hoped for advantages of such a system are structural integrity and compactness, low metal-to-coolant ratio, a high-heat-transfer area, and less critical pressure drops.

Utilizing a tubular fuel element and incorporating into each individual element a varying fuel load which can give an over-all non-uniform fuel distribution throughout the reactor core may offer other advantages. These are higher power output, longer core life and power flattening, uniform burnout, and lower maximum fuel element temperature.

A stainless steel tubular fuel element containing highly enriched UO_2 dispersed in stainless steel has been developed for power applications. Several thousand elements were manufactured for a full core loading. Each was 26 in. long, with an 0.375-in. OD and a 0.030-in. wall. The wall was 0.006-in. type 349 stainless steel clad, with 0.018-in. 24.5 wt % UO_2 in stainless steel.

An aluminum tubular fuel element containing 20% enriched UO_2 dispersed in aluminum has been developed for research applications. Several hundred elements, each 25.75 in. long with an 0.302-in. OD and a 0.037-in.-thick wall that consisted of 0.010-in. 2S aluminum clad, and 0.017-in. 65 wt % UO_2 in aluminum powder, were manufactured for assembly into Materials Test Reactor type (MTR) boxes for evaluation in that reactor. Table 1 shows the general specifications of the two types of elements being discussed.

The process for making aluminum elements has been extended to make experimental fuel tubes with a variable fuel loading along the length. Development efforts are also continuing on developing a stainless steel element with a variable fuel loading.

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TABLE 1
Typical Finished Specifications

	Aluminum Fuel Element (in.)	Stainless Fuel Element (in.)
Outer Diameter	0.302	0.375
	Wall 0.037	Wall 0.030
Inner Diameter	0.228	0.315
Clad Thickness	0.010	0.006
Meat Thickness	0.017	0.018
Meat Length	23.750	23.00
Over-All Length	25.750	26.25
Wt % UO ₂	65%	24.5%
Cladding	2S	349 stainless steel

II. STAINLESS STEEL FUEL ELEMENTS

The fabrication of a tubular fuel element is essentially one of bonding three concentric tubes to form a composite. Figure 1 is an artist's concept of the tubular element. The cladding is a commercially wrought material, while the center tube is a high-density powder-metallurgy product containing UO₂ dispersed throughout a metal matrix. This tube is referred to as the core tube and consists of the "meat" or fueled portion and "dead-end" material.

Fig. 1—Tubular fuel element assembly.

There are several methods for making the powder-metallurgy core tube with utilization of each depending on the desired tube diameter, tube thickness, material, and metal-to-UO₂ ratio.

A. Fabrication

The process for manufacturing the stainless steel elements is as follows: A blended powder mixture of UO₂ in pre-alloyed austenitic stainless steel is cold shaped through vertical feed rolls into a 0.030-in.-thick strip. This

is cut and sintered at 1175°C for one hour in dry hydrogen. In Fig. 2, a 12-ft-long-green strip formed in the vertical feed rolling mill is shown. Thirty-three per cent cold reduction in thickness in two passes with an intermediate anneal, resulted in a 0.020-in.-thick strip. After shearing and machining, the strip was ~ 1.250 in. wide by 20 in. long. Wrought stainless strips are then resistance welded to each end and act as end closures. The 0.020-in. fuel sheet is then cold formed into a tube, cleaned, and assembled between two wrought type 349 stainless steel tubes of 0.007-in. wall thickness. Figure 3 shows the stages in forming a tube from a core strip. The three concentric tubes are drawn tightly over a steel mandrel by cold drawing to an 18% reduction in area. The final step is a diffusion bonding treatment at 1150°C for one hour in dry hydrogen.

During the fabrication process, the cladding is reduced ~ 0.001 in. in thickness. This results in an extremely uniform interface and core-to-clad ratio. Figure 4 shows a transverse section of a finished element, and Fig. 5 a longitudinal view of an element. The fabrication process minimizes stringering and oxide particle break-up, as is seen in the micrographs. A highly magnified transverse view (Fig. 6) of an interface shows a high degree of metallurgical bond with grain growth across the original tubing interface.

B. Testing

Radiography and Measuray are used to determine uniformity of fuel dispersion. Ultrasonics, thermal shock, peel tests, autoclaving, porosity tests, and metallography are used to determine bonding and over-all integrity of the elements.

A three-inch prototype element has been irradiated in the MTR for 12 cycles and a 41 at. % U-235 burn-up, without any deleterious effects was attained.

III. ALUMINUM FUEL ELEMENTS

The use of thin-plate fuel components for service in high-performance research reactors has proven quite successful. Attempts to determine whether thin-walled tubular elements offered the sought for advantages lead to the development of an aluminum element for similar applications.

Since it is also desirable to incorporate high UO_2 -to-aluminum ratios in the element in order to utilize low-enrichment fuels, an alternate method for making the powder-metallurgy core tube was used. In addition, since this technique lends itself more readily to making variable-loaded fuel elements, it will be discussed in greater detail than the previous section.

A. Fabrication

The five essential steps in the process of making tubular aluminum fuel elements, specifically those designed for MTR applications, are as follows:

1. core fabrication,
2. component assembly,
3. vacuum-degassing,
4. vacuum drawing, and
5. diffusion bonding.

Blended powders of 65 wt % UO_2 , 5 wt % silicon-aluminum eutectic, 0.5 wt % stearic acid and 29.5 wt % 2S aluminum, are cold compacted into thin-walled cermet bushings with 0.317-in. outside diameters (OD) and wall thicknesses. Pressure is maintained at ton/in.² and sintering is done in vacuo for 15 min. at 625°C.

Fig. 2 — Powder rolled "green" strip.

Fig. 3—Core strip formed into tube.

Fig. 4—Transverse section, stainless steel element (Mag. 125X).

Fig. 5—Long section, stainless steel element (Mag. 100X).

Fig. 6—Transverse section, high magnification of interface (Mag. 400X).

The density of the sintered bushings averages 4.70 g/cm^3 or $\sim 86 - 90\%$ of theoretical density. The aluminum cladding and wrought dead-end tubing which are 3 in. long (0.320 in. OD x 0.020 in. wall thickness) are chemically cleaned prior to the next operation. The assembling of the components requires extreme care in order to avoid contamination of the surfaces which would inhibit bonding later on. First the inner cladding (0.265-in. OD x 0.014-in. wall thickness) is slipped over a mandrel lubricated with graphite. Dead-ends and "meat" bushings (24 g) are then slipped over the inner clad with the dead-ends crimped to prevent the separation of the bushings during subsequent drawing operations. The outer clad (0.375-in. x 0.013-in. wall) is slipped over the core tube and the completed assembly is tipped by swaging 2.5 in. of one end through a 0.300-in. swaging die and drawing another inch through a 0.300-in. drawing die in order to form a vacuum-tight seal for the next operation. To prevent blistering of the tubes during the bonding operation, they are vacuum-degassed. The fuel tube is slipped into a stainless steel chamber which is evacuated to 50 μ . While still under a dynamic vacuum, the stainless steel chamber with the element in it is suspended vertically in a furnace at 620°C for about 1-1/2 hr until the pressure falls to 25 μ . After cooling to room temperature the element is transferred from the vacuum chamber in such a way that the vacuum is maintained on the element during the drawing operation.

The fuel assembly is then given a 35% reduction in cross-sectional area in order to bring the cladding, core and dead-ends into intimate contact, and to work harden the material sufficiently so that recrystallization will occur across the tubing interfaces. The steps in this process are as follows: The fuel assembly, still under a static vacuum is lubricated and drawn, (using wire or tube drawing dies) in one pass to 0.303-in.-OD; the core length is checked either visually or by radiography.

If the "meat" is within tolerances, the mandrel is removed by radiography. If it is short of tolerances, the fuel assembly is annealed in air at 620°C , cooled to room temperature, and redrawn to 0.302 in. or 0.301-in. OD before removing the mandrel. Using this technique, $\sim 90\%$ of the fuel tubes manufactured met core length tolerances during the MTR production.

After the tubes are drawn and the mandrels withdrawn they are bonded by heating above the recrystallization temperature of aluminum, and holding it at these elevated temperatures until sufficient recrystallization has taken place across the tube interfaces. Bonding is accomplished by suspending the rough trimmed tubes vertically in a forced-air furnace at 620°C for one hour. They are then cooled to room temperature, radiographed to locate the fuel, checked for defects, and trimmed to final length, which is $25.75 \text{ in.} \pm 1/32$. After trimming to length, the tube ends are fusion welded to further insure its integrity and the welding is then followed by radiography to check for porosity. All welded tubes ends are then machined to ± 0.005 -in. tolerance, and straightened in order to facilitate assembly into fixtures for dip brazing into bundles. Following this the brazed bundles are then welded into the MTR boxes. A transverse view of a bonded aluminum element is seen in Fig. 7, while Fig. 8 shows a higher magnification micrograph of a longitudinal section taken under polarized light.

B. Brazing

The MTR fuel element assembly, or bundle, can be described as an array of tubes inside a segment of a cylinder $25\text{-}3/4$ in. in length. The segment of the cylinder, almost square in shape, is three inches on each side with two sides straight and parallel, and the other two sides having a radius of curvature of 5.5 in. and parallel. The tubes are arranged in rows of eight across and eight down for a total of 64 tubes brazed such that the tubes are separated from each other by four spacer pins at each end. A photograph of a brazed bundle is seen in Fig. 9.

Fig. 7—Macrograph view of aluminum-65 wt.% UO_2 element.

Fig. 8—Long view, aluminum element, polarized light (Mag. 240X).

Fig. 9—Braided bundle.

Fig. 10—Joining cermet bushings into tube.

Before brazing the tubes into a bundle it is necessary to pre-clean the ends in a dilute nitric acid solution.

The tubes and spacer pins are aligned in a header plate or jig with a 0.050-in. spacer or shim used to separate the tubes in the center of the bundle. Once aligned, the assembly is bound with wire in order to maintain the alignment prior to and during the brazing operations. The aligned fuel tubes are then dipped in a braze slurry of 18 parts aluminum to 12 parts Alcoa No. 718 braze powder and the excess braze removed. The assemblies are heated to 538°C and held for 30 min. then dipped in a bath of chloride and fluoride salts at 598°C for five minutes. After cooling to room temperature, the salt flux is removed from the assemblies by dipping in boiling water and scrubbing the insides of the tubes with brushes. Chemical cleaning with nitric acid and alkaline solutions is followed by a water rinse. The side plates and end adapters are welded in place and the machining is done as the final operation.

C. Evaluation

In order to select only those elements that most nearly meet the required specifications as to bond, core and cladding dimensions, concentration and dispersion of fuel, and over-all integrity of the element, a testing and inspection program was specified. The following is a breakdown of the tests performed on fuel tubes during and after fabrication.

1. Nondestructive

1. Visual examination	4. Crimp test on trimmed sections
2. Radiography	5. Boroscope examination
3. Porosity	6. Blister test

2. Destructive

1. Peel test	5. Blister test
2. Hacksaw test	6. Metallographic examination
3. Bend test	7. Chemical analysis
4. Thermal shock	

Autoclaving was performed on all good elements prior to brazing in order to put a protective oxide film on the surface.

D. Discussion

As a result of the fabrication and evaluation program, several weak areas became quite apparent. One of the most serious defects was the aluminum-to-aluminum bond in the dead-end area. By increasing the reduction in the cross-sectional area of the dead-ends to an excess of 50% and/or using M-388, a nickel-containing alloy, as the dead-end material instead of 2S, it was possible to improve the aluminum-to-aluminum bond.

The final diffusion bonding step at 620°C left the fuel tube in an annealed condition which proved to be a serious handicap in subsequent assembling and handling. By changing the process so that a small amount of cold work can be introduced into the tubes after bonding, the tubes can be handled with ease, are more rigid, and in addition, a rotary straightening operation can be performed.

Another serious problem is the density control of bushings while using a hand-operated press. Density fluctuations directly affect the elongation of the core during fabrication. By installing a pressure regulator, density variations were minimized. The use of a fully automatic press would eliminate the problem.

IV. VARIABLE LOADED FUEL TUBES

A. Aluminum Fuel Elements

The technique of using bushings as the "meat" of the core tube is quite amenable to preparing variable loaded fuel tubes. Sintered bushings of different weight percentages of UO_2 have been assembled and fabricated into fuel tubes. Fuel tubes with continuous sections containing 30, 45, and 70 wt % UO_2 along its length have been fabricated. This technique can be used to make any reasonable combination of fuel loadings per tube.

B. Stainless Steel Fuel Elements

In order to make stainless steel tubes with a variable fuel loading, a similar approach was undertaken. Bushings for preparing variable loaded stainless steel elements are cold compacted at 50 tons/in.² sintered in dry hydrogen at 1300°C and repressed at 60 tons/in.² to ~ 85% of theoretical density. The working characteristics of the sintered stainless steel cermet bushings are such that they separate during the drawing operation, therefore, another operation is required prior to the fabrication of the complete fuel tube. It is necessary to fabricate an integral tube of the many cermet bushings. The bushings are joined into a tube by assembling on a spring-loaded mandrel and subjecting the entire assembly to a one-hour heat treatment at 1300°C in a dry hydrogen atmosphere. The joined bushings are then swaged twice for densification and achievement of final dimensions. These steps are depicted in Fig. 10. The core tubes are then assembled and fabricated into an element by the techniques previously described.

This method has been used to prepare stainless fuel tubes with loadings of 14, 26, 30 wt % of UO_2 along its length. Such an element is seen in Fig. 11.

C. Seamless Cores

In order to overcome the problems in preparing bushings and joining them into tubes, a new approach was undertaken whereby a high density cermet seamless core tube was fabricated. A number of seamless cores of various UO_2 concentrations have been fabricated by loading the blended powders between a centered mandrel and a sacrificial clad. The assembly is then tipped and drawn in order to compact the powders sufficiently so that they can be handled easily. The loading operation is seen in Fig. 12. The sacrificial clad is then stripped off, and the cores with the mandrels still in them are sintered. After sintering the mandrels are withdrawn by stretching them on the drawbench. Further densification and dimensioning is done by drawing or swaging the core tube. The core tube is then bonded to its cladding by using the techniques described.

Several seamless cores of different lengths and UO_2 percentages have been joined to make variable loaded fuel elements. Also, by varying the fuel concentration during loading, seamless joint-free cores containing non-uniform fuel concentrations along the length have been made.

The development effort described has been performed on stainless steel and aluminum UO_2 dispersions.

Fig. 11 — Radiograph of variable loaded stainless steel- UO_2 fuel element.

Fig. 12—Loading operation for preparing seamless fuel cores.

DISCUSSION

D. L. EATON

MR. S. J. PAPROCKI (Battelle Memorial Institute): Did you encounter any reaction between your aluminum and your UO_2 ?

MR. EATON: Yes, we did. During our heat-treating process ~ 10% of the UO_2 reacted with the aluminum. We also subjected some samples to extensive heat treatment. Mr. Waugh, of Oak Ridge, performed the analysis. After we subjected the elements to about 80 hr, we got about an 80% reaction. However, we did not notice any gross volume changes with this reaction.

MR. PAPROCKI: Do you have any idea what the minimum temperature was when you began to get reaction between the aluminum and the UO_2 ?

MR. EATON: I believe it was around 600°C. Below that you don't get much reaction.

CHAIRMAN CUNNINGHAM: It has been observed down to 400°, I think.

MR. R. J. BEAVER (Oak Ridge National Laboratory): I would like to point out one thing which we observe about UO_2 reaction with aluminum in that as-pressed compacts heat treated at 600°C did not exhibit nearly as much reaction as did heat-treated fabricated plates. Apparently this is related to the oxide film on the aluminum in as-compacted form.

CHAIRMAN CUNNINGHAM: Could you make just one comment on the main advantages of these elements? What I am thinking of primarily is the amount of heat transfer per surface you get and the mechanical rigidity.

MR. EATON: With regard to heat transfer, I am not too familiar with that. Perhaps Mr. Maxwell, of the Materials Section, can help me out on that a little.

Mr. W. A. MAXWELL (The Martin Company): Heat transfer is a subject which I wish to avoid today if I can. It does have an advantage in high surface/g of fuel.

MR. EATON: With regard to the rigidity of the fuel tubes, these tubes were in the dead-annealed condition when sent to the MTR. Since that time, though, we have made an improvement by giving them a slight amount of cold working. This does increase the rigidity markedly.

MR. A. R. KAUFMANN (Nuclear Metals, Inc.): I would like to ask how extensive were your evaluations of the bond in the case of the aluminum tubes. The photomicrograph did seem to show some lines of demarcation. I wonder whether you made a lot of fracture tests?

MR. EATON: We performed a number of fracture tests or a test which we called the peel test. During this test the outer cladding and the inner cladding were forcibly pulled apart from the core. When testing in this manner we not only pulled the cladding apart from the core but we also pulled some of the UO_2 and aluminum from the core, indicating, we think, a good bond, when you fracture it, not at the interface but in the core itself. This contained 65 wt %, I might add, and the core was a lot weaker than the bond itself.

MR. R. PUGH (E. I. du Pont de Nemours and Company): Are these tubes and the MTR tubes in contact with each other?

MR. EATON: No, they are not. They are separated by 0.050-in. spacers on the ends.

MR. PUGH: How is the center supported?

MR. EATON: There is no support in the center. The tubes are supported on the ends only.

MR. J. B. CONWAY (G. E. Aircraft Nuclear Propulsion Department): In your MTR exposure of this single tube what was the surface temperature and how did you measure it?

MR. EATON: I am not too familiar with this MTR test. Perhaps Mr. Maxwell can help me out again on this.

MR. MAXWELL: We have no direct information on the details of that test.

MR. J. L. KLEIN (Nuclear Metals, Inc.): Did you get bonding between the core and the dead-ends?

MR. EATON: On the aluminum?

MR. KLEIN: On either.

MR. EATON: Yes, we did.

MR. KLEIN: On both?

MR. EATON: On both of them.

CHAIRMAN CUNNINGHAM: I have one general comment. What work are you directing now to try to get away from this chemical cleaning, because this looks like one disadvantage of this technique, particularly if you are trying to control the fuel, and if you use a burnout to control it, your accountability procedure gets a little messy when you start to put the fuel-bearing material into the acid solution?

MR. EATON: Actually, after the preliminary development was done on the aluminum fuel element we found that we did not have to clean the core itself to get a bond. On the stainless steel element, however, there is a slight etching of the UO_2 that is necessary to promote bonding, and I believe they weighed the core tube after cleaning to determine how much UO_2 was lost.

MR. J. J. CADWELL (Hanford Atomic Products Operation): I want to ask about the elements with the inner plate welded on the end. How did you succeed in not contaminating the surface during the welding operation?

MR. EATON: You are speaking of the aluminum bundle, I assume.

MR. CADWELL: It was one of them to which you welded an inner plate. I believe it was the steel one.

MR. EATON: The core there was first bevelled and then the UO_2 was leached out right on the very ends where the resistance weld was made. This provided a satisfactory bond without the UO_2 contaminating the dead-ends. Actually, I believe this dead-end was about an inch long, so that the weld itself was inside the tube and not on the exposed end.

Paper 6

FABRICATION AND PROPERTIES OF URANIUM-ZIRCONIUM HYDRIDE FUEL ELEMENTS FOR TRIGA REACTORS

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BRIEF DESCRIPTION OF THE TRIGA REACTOR

Recognizing the need for an economical but useful nuclear reactor by academic institutions and industry, General Atomic Division of General Dynamics Corporation has developed a 10-30 kw reactor, which has operated up to 115 kw for short periods. This type of reactor was specially designed for research and training and isotope production.

An inherent safety feature is built into the reactor; i.e., if the fuel elements begin to heat up because the reactor has excess reactivity, the increase in fuel temperature leads immediately to a decrease in reactivity. As there is no time delay in moderator heating, the average energy of the thermal neutrons follows promptly the instantaneous temperature of the fuel elements. Hence, an increase in fuel temperature leads immediately to a decrease in reactivity because of the decrease in fission cross-section in U^{235} as the neutron energy increases. The presence of appreciable amounts of U^{238} in the core contributes a prompt decrease in reactivity because of the greater parasitic absorption of neutrons brought about by the Doppler broadening of the U^{238} resonances.

The critical mass is less than 2 kg of U^{235} and the maximum excess reactivity is 0.5%. This reactor produces a thermal neutron flux of 10^{11} to 10^{12} neutrons/cm²-sec at 10-kw operating power.

One version of the reactor is located underground in a water-filled tank sunk into a pit approximately 21 ft deep and 6 ft 6 in. in diameter. Shielding is provided by approximately 16 ft of water above the core. This amount of water provides sufficient shielding of reactor radiation, allows removal of specimens while the reactor is operating, and acts as coolant for the core. The water also allows visual observation of the core and control rods during reactor operation and provides large volumes of irradiation space. A graphite reflector 12 in. thick surrounds the reactor core and 4-in.-long graphite sections in the top and bottom of the fuel elements form the top and bottom reflector.

The reactor is also available in an alternate design in which the water tank is above ground and surrounded by concrete shielding. Additional facilities such as beam holes and a thermal column can then be provided.

The boron-carbide control rods in the core are used to regulate the power level and to shut off the reactor. These rods are actuated by drive mechanisms located on a steel grating near the top of the tank and can be operated by hand or by a servo-control system.

TRIGA provides three main irradiation facilities in the reactor core and reflector for nuclear science studies--the "Lazy Susan," a "rabbit" tube, and a "glory hole." The Lazy Susan is a rotary specimen rack which holds 40 specimen containers. The pneumatically operated rabbit tube is used for producing isotopes with very short half-lives. The glory hole is a 1-in.-ID tube located in the center of the core, at the point of maximum flux and maximum statistical weight.

THE FUEL

The core consists of approximately 55 fuel elements, each of which contains a solid, homogeneous mixture of 8% uranium, 91% zirconium, and 1% hydrogen, which is equivalent to an atom ratio of U:Zr:H of 0.03:1:1. The uranium is 20% enriched in U^{235} ; the hydrogen is present as zirconium hydride (ZrH); thus the fuel (uranium) and the moderator (hydrogen) are intimately mixed in the fuel rod, resulting in a more efficient system than one in which the fuel and moderator are separated physically.

The active part of each fuel element is 1.42 in. in diameter by 14 in. long. Aluminum discs containing burnable samarium poison are placed at each end of the fuel rod and 4-in. sections of graphite are inserted above and below the fuel. The entire assembly is jacketed within a 0.030-in. 1100 aluminum can. The over-all fuel-element length is about 27 in. (Fig. 1). The samarium is burned up at a rate approximately equal to the samarium poison produced in the fuel as a fission product during operation of the reactor. This results in fuel elements with a reactivity life of about 10 years if the reactor is operated continuously at 10 kw.

FABRICATION OF THE FUEL RODS

Zirconium sponge is mechanically mixed with uranium pieces and the mixture is pressed into rods 1 in. by 1 in. square by 12 in. long. A pres-

Fig. 1—Fuel element.

sure of approximately 30 tons per square inch is applied and the resulting compact has about 90% of theoretical density.

Several of these compacts are welded together in an argon atmosphere to form a long electrode (Fig. 2). This electrode is inserted into an arc furnace (Fig. 3) and melted down to form an ingot. The mold is made of copper, water cooled. The ingot is removed from the mold and machined to clean up the surface.

Two of these ingots are welded together in an atmosphere box and the melting procedure repeated. Upon this double melting, the uranium is fairly uniformly distributed throughout the ingot.

After the second melting, the ingot is machined down to 3.8 in. diameter to clean up the surface, placed into a copper container which is evacuated and sealed, heated to 1400°F and extruded into a long rod, approximately 1.6 in. in diameter. The copper is removed by pickling in nitric acid and the clean rod is transferred to the machining room. The long rod is cut into the required lengths, machined to size and hydrided to ZrH. Figure 4 shows the hydriding room.

The hydriding process has to be carefully controlled. The fuel-rod hydriding apparatus is shown in Fig. 5. Too low a temperature gives rise to severe cracks and fissures in the fuel element, and too high a hydriding temperature results in the appearance of "pimples" on the surface of the element, which consist of uranium-rich particles.

The hydrided rods are assembled into an aluminum tube together with the graphite reflector rods and the samarium discs. The tube is sized onto these rods in a drawbench and the end plugs are welded (Figs. 6 and 7). The welds are tested for leaks by a helium mass-spectrograph leak detector and by means of a cold autoclave test.

METALLOGRAPHY

Figure 8 illustrates the macrostructure of a cross section of a fuel element magnified to 3X. It will be observed that the structure is uniform from center to edge, which illustrates that the uranium and hydrogen are fairly uniformly distributed throughout the cross section. Although the various phases have not been identified completely, we believe that the light structure is alpha zirconium and that the blue and tan structures are gamma zirconium hydride, of various hydrogen concentrations. The uranium is probably present in two phases: alpha and epsilon; the latter phase is an intermetallic compound composed approximately of 50 wt-% zirconium, whereas the former phase is alpha zirconium containing a small amount of uranium in solid solution. Hence, the uranium does not exist as such structurally, but is combined with zirconium to form two different phases. The zirconium hydride also exists in two phases: a gamma phase (face-centered cubic) with a zirconium-to-hydrogen ratio of 1.5 ($ZrH_{1.5}$) and an alpha phase (hexagonal); the latter phase is alpha zirconium with a small amount of hydrogen in solid solution. Although four phases are described above, actually only three probably exist: gamma zirconium-hydride phase ($ZrH_{1.5}$), epsilon phase (50% uranium plus 50% zirconium), and alpha zirconium phase, containing hydrogen and uranium in solid solution.

Figure 9 illustrates a series of microstructures of variable hydrogen concentrations in the zirconium-hydrogen system. The presence of uranium appears to impede the rate of hydriding.

Fig. 2—Arc welding of U-Zr electrodes in atmosphere box.

Fig. 3—Arc welding furnace.

Fig. 4—Hydriding room.

Fig. 5—Fuel-rod hydriding apparatus.

Fig. 6—Macrostructure of weld (28X).

Fig. 7—Top end of fuel element and cross section through cap (actual size).

Fig. 8—Macrostructure of cross section of fuel element (3x).

Fig. 9—Zr-H diagram based on X-ray diffraction. (Data by D. A. Vaughn and J. R. Bridge, Journal of Metal, May 1956).

PROPERTIES

Figure 10 illustrates the stress-strain characteristics of the U-Zr and the U-Zr-H alloys. It will be observed that U-Zr alloy before hydriding has a much higher tensile strength than the alloy after hydriding; also, a plastic range exists for the former, whereas the latter exhibits only elastic properties. The modulus of elasticity for the alloy before hydriding is slightly greater than the value after hydriding.

Thermal conductivities as functions of temperature for Zr, U-Zr and U-Zr-H are illustrated in Fig. 11.⁽¹⁾ The values for electrical resistivity at room temperature are 45 microhm-cm for zirconium, 73 microhm-cm for Zr-U alloy, and 53 microhm-cm for U-Zr-H alloy.

FUEL EVALUATION

The criteria of good fuel elements are (1) resistance to catastrophic failure due to pin holes in the cladding and (2) resistance to failure by radiation damage. To assure that the TRIGA fuel elements will perform satisfactorily, many and varied tests on fuel materials and fuel assemblies have been performed. Figure 12 illustrates a section of a fuel element with a 1/16 in. hole drilled through the cladding. Figure 13 illustrates this same section after 500 cycles from 200°C (furnace) to 20°C (water); no evidence of corrosion or undercutting was observed.

The resistance of the fuel material to thermal shock has been tested. A 1-in.-diameter rod of U-Zr-H alloy was heated in a furnace to 800°C and end-quenched in water; no deleterious effects were observed. Also a 1/4-in.-diameter rod of U-Zr-H alloy was heated electrically to about 800°C and a rapid stream of water was sprayed on it; no reaction was observed.

Also, other small and large samples were heated to various temperatures up to 900°C and quenched in water; the only effect observed was a slight discoloration on the samples. Finely divided powder of U-Zr-H has been heated to 300°C and quenched in 80°C water; no reaction was observed.

Four samples of U-Zr-H alloy have been exposed to neutron fluxes in the MTR. One sample of $ZrH_{1.0}$ and one sample of $ZrH_{1.5}$ were exposed to 1% burn-up of U^{235} ; no changes in dimensions were observed. Two other samples, as per above, were exposed to 5% burn-up of U^{235} ; no changes in physical dimensions were observed. Hence, since the lifetime burn-up of TRIGA fuel elements is only 0.003 atom-%, the results of these four tests indicated that the fuel elements will resist damage by the irradiation climates in TRIGA reactors for their reactivity lifetime of 10 years under continuous operation at 10 kw.

ACKNOWLEDGMENTS

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Fig. 10—Stress-strain characteristics of the U-Zr and the U-Zr-H alloys.

Fig. 11—Thermal conductivities.

Fig. 12—Section of a fuel element with a $\frac{1}{16}$ in. hole drilled through the cladding.

Fig. 13—Same as figure 12 after 500 cycles from 200°C (furnace) to 20°C (water). No evidence of corrosion or undercutting was observed.

DISCUSSION

W. P. WALLACE

MR. A. R. KAUFMANN (Nuclear Metals, Inc.): Do you know what state the uranium is in after the hydriding?

MR. WALLACE: We believe that it is in the ϵ phase (uranium-zirconium alloy), but we have not confirmed this by x-ray analysis.

MR. KAUFMANN: No uranium hydride?

MR. WALLACE: No uranium hydride.

MR. C. F. FRANCIS (Phillips Petroleum Company): Would you care to comment on the burn-up of the samarium? How does this compare with boron or other materials?

MR. WALLACE: This was done by our physicists. I am not a physicist -- I am more of a blacksmith. We have the head of our metallurgy department here. Does he care to comment?

MR. M. T. SIMNAD (General Dynamics Corporation): I think I can classify myself somewhat as a blacksmith, too. I believe that the radiation damage is greater with boron carbide dispersions because of helium production, whereas the burn-up of samarium is accompanied by γ radiation.

MR. WALLACE: One comment on that point might be that we produced this poison that I mentioned by mixing aluminum and samarium oxide, placing this into a container and extruding it into rod, rolling it into strip, and then cutting the strip into circular discs. One of these is placed at the end of each 14-in. length of fuel element.

MR. H. JAFFE (Aerojet-General Nucleonics): We are using samarium in the gas-cooled reactor experiment, and the samarium addition on the basis of nuclear considerations is dictated by the lifetime that we hope to achieve; namely, by the burn-up that we hope to achieve, so that at the end of this particular irradiation time, the samarium is essentially consumed and we have no excess reactivity or excess uranium there that is critical. If you define the lifetime of the burn-up, you can choose the poison that meets this, based on the cross section of the burn-up poison, and boron having a much lower cross section than samarium becomes ideal for this process for very long-lived systems. In our case we have a reactor of a few megawatts. Samarium was chosen on the basis of about 4,000-operating core length.

MR. SIMNAD: I agree that this is one of the important considerations, since our reactor fuel is expected to last about 10 years on continuous operation.

MR. H. F. TURNBULL (Westinghouse Atomic Power Division): I would like to ask a question along the blacksmith line. You mentioned mixing zirconium with uranium and melting. Will you please elaborate? How large were your melts and how did you arc-melt it? Was it vacuum or atmosphere?

MR. WALLACE: The first core of this was produced for us by Oregon Metallurgical Corporation and Nuclear Metals. We received the uranium from Oak Ridge in the form of 5/8-in.-dia rods. These were cut into small pieces by chopping and then loaded into the compacting die with zirconium sponge. The uranium and zirconium were pre-weighed and loaded into this die. A 2 x 2 x 15-in. electrode was produced. These were welded together and inserted into an arc-melting furnace in an atmosphere of argon.

MR. TURNBULL: How much did these electrodes weigh?

MR. WALLACE: The ingot itself weighed about 50 lb.

MR. E. GORDON (Olin Mathieson Chemical Corporation): How did you get the right amount of hydrogen?

MR. WALLACE: The hydriding furnace contains a ceramic tube; this tube is evacuated. First, the alloy is weighed, inserted into the ceramic tube, the temperature is raised, and then the hydrogen is metered into the system; after a certain time, the furnace is cooled down and the hydrided alloy is weighed again. This is one check and metering of hydrogen is another check. The third check is that we then spot-check analyze for hydrogen on the diameter as well as the length.

MR. JAFFE: I have two questions. The first one is how do you dissipate the heat through the system, and the second question is what temperature do you use?

MR. WALLACE: The heat from the hydriding system?

MR. JAFFE: No, from the 10 kw operation.

MR. WALLACE: This reactor operates at about 40 to 45°F, and there is a cooling system. The other question was?

MR. JAFFE: Well, the temperature.

MR. R. J. BEAVER (Oak Ridge National Laboratory): I would like to get back to that samarium question. It is my understanding that both in samarium and gadolinium there are some isotopes that haven't been measured that can significantly affect the utility of these either as burnable poisons or control material. Would anybody care to comment on that?

MR. JAFFE: Aren't you thinking of europium?

MR. BEAVER: No, europium has a higher cross section.

MR. JAFFE: As I recall, the most significant capturing isotopes of both samarium and gadolinium go to products that have no significant cross section from the point of view of burnout, so that to the best of my knowledge these are both single-shot affairs.

MR. FRANCIS: I think that part of the answer here is the use that the thing is being put to now. In our place we think of cadmium and boron and perhaps even of praseodymium as rapid burnouts. Praseodymium is not as fast as the others, and samarium and europium are slower burnout materials. Perhaps this is a matter of the flux at which the reactor is operating. We think of the latter ones as control-rod materials. Scandium, for example, we equate with xenon. The boron we equate with the fuel burnout.

Paper 7

NONDESTRUCTIVE TESTING OF REACTOR FUELS

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ABSTRACT

Nondestructive Testing is a general name given to all methods and techniques which permit testing or inspection of raw material or fabricated components without impairing the future usefulness of the material or part. In this paper the application of nondestructive testing to nuclear reactor fuel elements will be discussed. Nearly every form of energy and nearly every property of material have been made a basis for some type of nondestructive test, however, these test methods may be arbitrarily divided into the following groups: Visual, Pressure and Leak, Penetrant, Thermal, Radiography, Acoustic, Magnetic, Electrical, and Electrical Magnetic. All of these various methods are being applied to a variety of testing and inspection problems.

The essential parts of any nondestructive test are, 1) APPLICATION of the testing or inspection medium, 2) MODIFICATION of the testing or inspection medium by defects or variation in structure, 3) DETECTION of this change by a suitable detector, 4) CONVERSION of this change into a form suitable for interpretation, and 5) INTERPRETATION of the information. In some cases a sixth step may be necessary that is to convince the supplier, fabricator or engineer that the interpretation is the most important. The principles of all the methods are relatively simple but it is the application and interpretation of the results that are important and essential. Most nondestructive test results are indirect measurements, for example, variations in density of an X-ray film, deflection on a recorder trace, the appearance of a pip or change in the pattern of an oscilloscope trace or a meter deflection. Consequently, it is essential that the interpretation of such data be made by an experienced person. In order to aid in the interpretation of such results it is necessary to do considerable destructive testing, especially during the initial stage of a new testing program. During the testing program itself, it is necessary to make periodic destructive tests in order to ascertain that the equipment is operating at the desired sensitivity and resolution.

Typical defects which can be found by nondestructive tests include: surface and sub-surface cracks, tears, laminations, voids, flakes, laps, seams, inclusions, blow holes, porosity, pipe, incomplete fusion in welds, and lack of bond. Metallurgical variables such as grain size, grain orientation, and heat treatment can be measured using nondestructive testing techniques. Variations in physical and chemical properties can sometimes also be found by use of nondestructive tests. Thickness and density variations can be measured to a high degree of precision using radiation.

PRINCIPLE OF OPERATION

Fig. 1—Principles of pulse echo technique.

Nondestructive testing of fuel elements for heterogeneous reactors and of other components for a reactor system such as heat exchanger tubes differs from ordinary industrial nondestructive testing in that the integrity required in the nuclear field is much greater than that which have been used until recently in other industries. The sensitivity and resolution required of the tests are in excess of those that can be justified in most other industrial applications.

* * * *

The first testing method to be discussed is the use of ultrasonics. Three different ultrasonic testing techniques are in common use today: pulse echo, resonance and transmission. In the pulse echo technique a pulsed ultrasonic beam is sent into the specimen. The beam is reflected from the opposite side of the specimen as well as from any discontinuity or defect as shown in Figure 1. In a variation of this technique the pulsed ultrasonic beam is projected into the specimen at an oblique angle to the surface. The beam travels in a zig-zag path as shown in Figure 2 between outside and inside wall of the specimen and the echoes picked up as above. Three types of data presentation are used in echo or reflection testing, the A, B, and C scan. The A scan is a one dimensional presentation in which the signal intensity is presented as a function of the time or distance traveled by the ultrasonic wave. The B scan gives a cross-section view of the specimen as the transducer moves along a given line. This type of presentation has the advantage since both the length of the flaw and its depth below the surface are indicated. The C scan is obtained from an A scan presentation made at a number of points over the surface of a specimen, and if sufficient points are plotted a contour of the flaw

Fig. 2—Principle of shear-wave testing.

is obtained. In the C scan only the reflections from the flaw are used. These three types of data presentation are shown in Figure 3. Inspection of the test specimen can be carried out by either having the transducer contact the specimen through a thin film of the couplant - contact scanning, or by having the transducer and specimen each immersed in the couplant but separated from each other - immersed scanning.

The ultrasonic transmission technique is finding wide use and acceptance for testing of reactor fuels components and completed fuel assemblies. In the transmission technique a pulsed ultrasonic beam is sent through the specimen and the transmitted energy received by the second transducer as shown in Figure 4. The energy lost by the beam due to scattering and reflections reveals existing flaws and their extent. In this technique, the frequency and the cross-section of the incident beam are important considerations, for example, the higher the frequency the more nearly rectilinear the propagation of the beam. The cross-sectional area of the transmitted beam has a direct bearing on the size of the flaw that can be detected. The size of the transmitted beam can be controlled by use of 1) small crystals, 2) masks, and 3) lenses. A mask is an acoustic absorbent material which covers the transducer. A hole through the mask provides the beam aperture. Figure 5 shows the ultrasonic scanning and recording system developed at Argonne National Laboratory. This general principal is used by commercially produced systems. The transducers can move longitudinally and vertically independently of each other, thus the composite motion of vertical and horizontal displacements permits the simultaneous inspection of a flat surface both in width and length. The intensity of the transmitted beam at any point on the specimen is recorded on current or voltage sensitive recording paper. Figure 6 shows a typical recording made of a uranium alloy casting using the Argonne equipment. Figure 7 shows the structure of the casting in and adjacent to the light area on the recording. The ultrasonic energy is attenuated much more in the porous area producing a white region on the recording.

It is possible to use the transmission technique to determine the relative degree of bonding of zirconium clad specimens. By proper selection of the amplification and recording level, the relative darkening of the paper gives a relative measure of the bond strength. Bond strengths were determined also by tensile testing. Zirconium studs, $\frac{1}{4}$ inch in diameter were percussion welded to the specimen, as shown in the upper portion of Figure 8. The area around each stud was machined through the cladding leaving only the bonded area under the stud. The specimen was then put in a tensile testing machine and pulled. Comparison of the tension test with the relative darkening of the recorder trace gave a good correlation between bond strength and darkening of the paper. Figure 9 shows a special transducer holder which was made for inspecting the bond of fuel elements already welded into an experimental boiling water reactor fuel assembly.

In the resonance test, a mechanically driven tuning condenser sweeps the frequency of an electronic oscillator through its tuning range. The oscillator in turn drives the transducer. There are momentary increases in the energy drawn by the transducer whenever the frequency of the oscillator coincides with a thickness resonant frequency of the specimen. The increase in energy can be indicated on a cathode ray tube or a meter. A change in resonant frequency which cannot be accounted for as a change in thickness is usually an indication of a defect.

The third phenomenon which may occur at an interface is mode conversion. When an ultrasonic wave either longitudinal or transverse impinges on an interface between two media at an angle other than at right angles, a wave transformation or mode conversion may take place. The transformation which takes place may best be visualized by means of Figure 10, which shows a longitudinal

(Text continues on p. 94.)

LAMINATION IN PLATE

"A" SCAN OF LAMINATION

"B" SCAN OF LAMINATION

"C" SCAN OF LAMINATION

Fig. 3—Three types of data presentation.

Fig. 4—Principle of transmission technique.

Fig. 5—Ultrasonic scanning and recording system.

Fig. 6—Electrosensitive-paper recording of uranium-alloy casting.

Fig. 7—Macro of casting showing porous areas.

Fig. 8 — Zirconium-clad plates, electro-sensitive-paper record and percussion welded studs.

Fig. 9—Transducer holder for inspecting plates in subassembly.

Fig. 10—Mode conversion.

Fig. 11—Shear-wave transducer.

wave impinging at an angle (θ) on an interface. In the most general case, four separate waves are produced, a wave of each type is reflected and a wave of each type is refracted. Since the velocity of the two types of waves are different they have different angles of reflection and refraction as shown by Figure 10. The different types of waves also have different critical angles which makes it possible to totally reflect one type of wave and have only one refracted wave entering the second medium, Figure 11. This permits the use of either longitudinal or transverse waves for ultrasonic testing.

Ultrasonic waves can be propagated over the surface of the solid as the surface or Rayleigh waves as they are sometimes called, are propagated over the surface, they are attenuated. This attenuation depends upon the material, the surface conditions of the material and the frequency. Any surface on which surface waves are to be generated should be as free as possible of dirt, grease, since such substances damp or absorb the waves appreciably in some instances produce direct reflections of the beam. For the same reason the couplant should be confined to the area directly beneath the transducer. The use of surface waves for testing an irregular surface is shown in Figure 12. Surface waves will show the presence of surface inclusions in inclusions just under the surface of the material. As can be seen in the figure, the surface waves follow the curvature of the part and are able to locate defects on the opposite side of the specimen from the transducer. Since the defects often have a random orientation it is advisable to send the waves in two directions on the surface of the part. Defects only 0.009 inch to 0.010 inch in diameter can be detected as far away as twelve feet. Surface waves can be used to determine if discontinuities exist just below the specimen surface, even though the surface itself is free of defects.

Besides the longitudinal, shear and surface waves which have been discussed, there is another important and less well known class of vibrational modes that is of great potential value in nondestructive testing. These waves were theoretically discovered by Horace Lamb in 1916, who reported them in a paper to the Royal Society of London. In essence, Lamb mathematically proved that under the conditions that thickness of the specimen was less than the wave length, an infinite number of vibrational waves could travel in an elastic plate. Lamb grouped the infinite number of modes in which a plate can vibrate into two main types, according to the direction of particle displacement, which he called symmetrical and asymmetrical. However, each type appears to be of equal value from testing standpoint. The important feature is that the various modes in each type travel along the plate with a velocity that is dependent upon the thickness of the plate, the frequency of the wave, the order of the mode, and the material of the plate. The velocity being dependent upon these factors is in striking contrast to longitudinal, shear, and surface waves whose velocities are determined by the properties of the medium in which they are traveling. The velocity with which each mode propagates in a given plate can be calculated from Lamb's equations as a function of frequency, plate thickness, and material.

Ultrasonic testing with the single crystal pulse-echo method is limited in detecting flaws very close to the surface of a metal because of the difficulty in providing interrogating pulses sufficiently short in time duration. Lamb waves offer the possibility of considerably extending the effective inspection zone closer to the surface. It is found that these waves can be generated in the metal overlying shallow defects as shown in Figure 13 provided the defects are oriented such that one edge is near parallel to the surface. The metal covering a flaw of this type appears to the Lamb waves as though it were part of a thin plate. Lamb waves can also be generated in cylindrical objects so that they can be used for testing pipes and tubing for laminar flaws since laminar flaws located in the tube wall effectively reduces the wall thickness. The tube could be scanned with the ultrasonic beam impinging upon the outside surface, with

Fig. 12—Surface waves.

ULTRASOUND BARRIER
BLOCKS SURFACE ECHO
FROM RECEIVE TRANSDUCER

FLAW

NO FLAW

Fig. 13—Lamb waves.

the frequency and incident and exit angles chosen and held fixed to generate and detect a suitable Lamb wave mode. With the incident angle set to excite the first symmetrical mode, a flaw, in reducing the effective depth, would destroy the Lamb waves, and hence be identified by the resulting loss of signal reaching the receiver crystal.

It was previously mentioned that metallurgical variables or such as grain size, grain orientation, and other physical properties and chemical properties of special significance in reactor design, construction, and operation can be found by use of nondestructive tests. For example, ultrasonics can be used to measure grain size. When the size of the grains in the metal approaches the wavelength of the ultrasonic waves, the scattering increases rapidly. Consequently, by proper choice of frequency, differences in grain size can easily be determined.

This brief review of the use of ultrasonics in nondestructive testing indicates that the development and uses of these techniques has just only begun, especially the use of surface waves and Lamb waves. Many possibilities and applications of these techniques will suggest themselves to the nondestructive testing engineer as more knowledge and experience with the various types of waves is accumulated. It has been my experience that where new nondestructive testing techniques have been developed, there are problems awaiting solution. However, in fairness to the nondestructive testing engineer, I would like to ask that he be given time for sufficient development and application to each particular problem.

* * * *

Radiography, although one of the oldest nondestructive testing techniques, continues to find wide application in the field of fuel element inspection. Radiography is used on a variety of inspecting problems, for example, the detection of voids and inclusions in cladding material as well as in core material. The location of the fuel, the detection of inhomogeneities in uranium alloys and the examination of weldments for porosity, lack of penetration. The problems in the nuclear reactor field are similar to many found in the metals industry but the solution is complicated by the necessity for better than normal radiography sensitivities and the high density of materials involved. The high atomic number of the materials used in nuclear reactors also makes the radiography more difficult. For example, zirconium has a density slightly lower than that of steel, but because of its high atomic number, zirconium has a higher absorption coefficient than steel.

In the field of nuclear energy the quality of the radiographs has to be better than that accepted in routine industrial radiography. Various techniques make such improvements possible, they are for instance, the utilization of vacuum, cassettes in order to attain an intimate contact between the intensifying screen and the film, increasing object-to-film distance in order to obtain an enlarged radiograph, therefore improving the accuracy made on the film and at the same time combating specimen scatter by applying the inverse square law, and the technique of radiographing a radiograph in order to increase the latitude. The Los Alamos Scientific Laboratory has used this latter technique in radiographing zirconium coated with graphite. A conventional radiograph is first prepared but because of the difference in the atomic number of the materials, the graphite will not be delineated. However, the film in the emulsifier has been effected and by making a low kilovolt exposure of the developed radiograph, the graphite can be resolved. When radiographing materials such as uranium, unless certain precautions are given the specimen can badly fog the film or completely obliterate the film. Using a two film technique, a fast and slow film in the same cassetts, it is possible with one exposure to show the location of the fuel in the element and inhomogeneities in the fuel itself.

In testing for defects in closure welds of fuel elements often times the region being radiographed varies in thickness, especially on cylindrical fuel elements as shown in Figure 14. A double film technique can be used to an advantage in this case, a fast film being used to record the inner edge (thick) of the weld where the transmitted intensity is low and a slow film used to record the outer (thin) edge. To determine the sensitivity and resolution of such a technique specimens of the same size and shape as the components to be tested should be made. Holes of various sizes and depths can be drilled into these samples. Some of the holes should be filled with a lead oxide paste or similar material and others left unfilled. These filled and unfilled holes then serve as voids and inclusions of known dimensions.

The natural radioactivity of the uranium core was used to measure the cladding thickness of the Experimental Boiling Water Reactor fuel plates. One of the daughters of U^{238} emitted a 2.3 Mev. beta ray. This radiation is attenuated in passing through the clad. By using a scintillation (anthracene) crystal the intensity of the transmitted radiation was measured. For the measurement of zircaloy, cladding in the range of 0 to 20 mils, this equipment gave a 10 per cent change in the counting rate per mil change in thickness.

The decay spectrum of uranium is rather complex, but a scintillation counter analysis shows the principle activity to be concentrated in two regions. A low energy region corresponding to approximately 90 kilovolts and a high energy region corresponding to 80 kilovolts. These two energies are being used to determine how much uranium²³⁵ there are in completed fuel plates or in cores to be used in fuel plates. Corrections must be made to the counting rate to account for geometrical, self absorption and counting losses. Figure 15 shows a schematic of the system used.

The procedure used is as follows. The 180 KV gamma from U^{235} was counted differentially with a 2% window width giving a counting rate of about 10,000 counts per second. Each core was counted for five minutes. The standard core was counted after each five cores. The cores counted between standards were compared to the average of the counts for these two standards. A core of the same weight and giving the same number of counts was analyzed to give us the weight of U^{235} in the standard. Two other cores with U^{235} content less than and greater than the standard were also analyzed to give three points which were plotted on a graph. The graph of the weight of U^{235} against the counting rate was linear over this limited range. Therefore the change in the counting rate was proportional to the change in U^{235} content, the slope of the graph being the constant of proportionality. Using this information the counting rate for each core was compared to the counting rate of the standard to get the U^{235} content. To experimentally determine the accuracy of the method, six cores were counted again after pickling. These cores showed a loss of U^{235} of from 100 - 300 milligrams which agreed with what was calculated from the weight of uranium metal lost in the pickling process.

The Argonne Low Power Reactor cores had a uranium content of approximately forty (40) grams. The uranium content could be determined to ± 0.1 gram. Some of the above mentioned fuel element cores were rolled into fuel plates approximately thirty (30) inches long and then scanned. A scanning device was assembled which carried the fuel element in front of the detecting scintillator. Heavy lead shielding was provided so that the detector could view only a section of the fuel element at a time and was completely shielded from the rest of the element. The fuel element passed under the sodium-iodide detector in about eight (8) minutes. The speed of the traverse must be carefully regulated because the accuracy of the method depends upon the accuracy of the speed of this motor. Good correlation was obtained between these results and the results obtained during the five (5) minute counting interval.

Radiation transmission gages are used for measuring the uniformity of distribution of uranium in two component fuel plates such as uranium-graphite,

Fig. 14 —Radiographic set-up for end weld areas.

Fig. 15—Schematic diagram of a scintillation counting system.

Fig. 16—Backscatter of β particles.

uranium-stainless steel, and uranium aluminum. Conventional X-ray machines as well as gamma ray sources have been used as radiation sources. The Los Alamos Scientific Laboratory reports that for (1) uranium-graphite plates with a thickness of 0.5 in. and a loading of uranium of 0.100 g/cc the uranium content can be determined to 1%. (2) Uranium-stainless steel plates with a thickness of 0.040 in. containing 25% by weight of uranium, the uranium content can be determined to 1%, (3) uranium-aluminum plates with a thickness of 1 in. and containing 15 wt. % uranium, the uranium content can be determined to 0.4%.

The X-radiation gages have the advantages of good geometrical resolution, because of the use of a small beam diameter and high speed. The major disadvantages are the limited penetration of material possible with well controlled X-ray generators and the difficulties involved in making a theoretical interpretation of results. The gamma gage has the advantages of much greater penetration of material and of more straightforward theoretical interpretation of results when an essentially monochromatic source is used. The main disadvantages are the lower signal level, leading to a need for a larger beam diameter and a longer measurement time, and the inability to gain a very appreciable separation of the two components in, for example, the aluminum-uranium plates mentioned above.

When beta rays impinge on material, a fraction of the electrons are reflected or back scattered. The intensity of the back scattered radiation increases with increasing thickness of material until the thickness equals approximately one half the range of the electrons. Therefore, increase in the thickness does not increase the intensity of the back scatter radiation. At saturation or infinite thickness, the most energetic beta particles do not possess sufficient energy to penetrate the thickness of the material and be reflected to the detector. The ability of materials to back scatter is also function of the atomic number of the material as shown in Figure 16. If a second material having a different atomic number is placed upon the first material the counting rate will change. The counting rate will increase if the second material has a higher Z and will decrease if the Z is lower. The greater the difference in atomic number between the two materials, the greater the difference in counting rate and consequently sensitivity of the measurement. Beta ray back scatter can be used to measure the thickness of cladding on fuel plates. At Argonne National Laboratory we have investigated this technique and found that in 30 mil aluminum cladding, 1 mil variations can be determined. Helium gas used in connection with commercially available mass spectrometer helium leak detectors gives a very sensitive test. Commercial helium leak detectors can detect the presence of one part of helium in 100,000 parts of air. Only one of the numerous helium leak techniques will be used. In applying the latter technique, the test specimen is first placed in a helium pressurizing vessel and exposed to a helium atmosphere. The sample is removed from the pressurizing vessel and transferred to a second chamber which is connected to a vacuum pump and helium leak detector, as shown in Figure 17. This chamber is first evacuated to a pressure of approximately 150 microns, then the throttle valve on the leak detector opened, and the relative quantity of helium released read on the leak indicating meter. Table I shows typical results of helium leak results on a group of fuel elements. The two samples having excessive response were found to have defective welds. Dirt, oil, and scale on the test object cause large response when using this technique, but samples that have had an acid dip are sufficiently clean for testing. A large open cavity (such as cracks in welds which are visible to the eye) does not produce a noticeable response because helium is not retained long enough after the fuel element is removed from the helium pressurizing vessel.

When a coil carrying alternating current is placed near a metal specimen eddy currents are indicated in the metal as shown in Figure 18. The induced

FIG. 1
HELIUM LEAK DETECTOR
TEST APPARATUS

Fig. 17—Helium leak detector test apparatus.

Fig..18 —Eddy currents induced in specimen.

TABLE I.

HELIUM LEAK TEST RESULTS

<u>Response (Arbitrary Units)</u>	<u>No. of Slugs in Range</u>
0.0 - 0.5	48
0.6 - 1.0	5
1.1 - 1.5	18
1.6 - 2.0	19
2.1 - 3.0	30
3.1 - 4.0	3
4.1 - 5.0	3
12	1
14.0	1

eddy currents set up a magnetic field which opposes the original magnetic field. The impedance of an exciting coil or a pick-up coil in close proximity to the specimen is affected by the presence of the induced eddy currents. If the path of the eddy current is distorted by a defect or structural variations exist, the apparent impedance of the test or pick-up coil is different than when such variations are not present. This change of impedance can be measured and used to indicate the presence of a defect or variations in physical structure. In general, two types of test coils can be used, a concentric coil and a point probe. The concentric coil encircles the specimen and interrogates an annulus of the specimen having width equal to the effective width of the coil. A point probe interrogates an area essentially equal to the cross sectional area of the probe. The test coils can be either resonant or driven.

Figure 19 shows how the impedance of a probe may vary in proximity with a plane specimen of nonferromagnetic metal. Curve A is typical of the type of curve which may be obtained as the probe-to-sample spacing is varied, the axis of the probe being maintained perpendicular to the plane of the metal. The ordinate is the fraction of the free space inductance obtained when the metal is in proximity to the probe. The abscissa is the incremental change in resistance from the free space value divided by a constant, the free space reactance. Curve B is typical of the type of curve obtained when the probe-to-sample spacing is maintained constant and the conductivity is varied.

The application of eddy current methods to the problems of nondestructive testing has always been hindered by the difficulty of separating the useful information produced by the test from the unnecessary information which is always present. For example, in testing reactor fuel elements, it is usually necessary to suppress the effect of variable spacing between test coil and metal over a useful range of spacings. A number of methods have been used to accomplish this.

Variations in probe-to-metal spacing can often be controlled mechanically by using a Teflon tape spacer between the probe and test specimen or by using a servo-actuated probe carrier. However, in some applications surface contour variations and geometric limitations may cause difficulties, so that electronic circuits must be designed to provide additional discrimination against probe movement. In one system, eddy currents of two separate frequencies are induced in the specimen to be tested. One frequency (low frequency channel) is used to obtain the information required, the other frequency (high frequency channel) is used to compensate for the effect of variations in test coil-to-metal spacing and other undesirable effects. A block diagram of the equipment is shown in Figure 20. One application of this technique was the measurement of the thickness of the zircaloy-2 clad on a uranium-2% zirconium core. The

Fig. 19—Impedance plane curves.

Fig. 20—Dual-frequency test system.

cladding thickness was nominally 0.020 in. The resistivity of the core material was approximately 72 microhm cm while that of the cladding material was about 52 microhm cm. It was expected that the cladding thickness would always range between 0.015 and 0.025 in. Experiments showed that frequencies between 25 and 50 kilocycles per second (kc) - low frequency - were suitable for the measurements of the clad thickness and that eddy currents of a frequency of 260 kc - high frequency - were relatively ineffective for the measurement of clad thicker than 0.015 in. Thus, the high frequency channel output was essentially independent of the cladding thickness. The probes used in this application were wound on 0.250 in. or 0.1875 in. cylindrical ferrite forms. The winding consisted of a single layer of 0.003 in. diameter enameled wire. This dual frequency eddy current system has also been used to measure the zirconium cladding thickness of uranium molybdenum alloy fuel rods. The nominal cladding thickness was 0.004 in. and the nominal outside diameter of the fuel rods was 0.158 in. The low frequency channel was operated at a frequency of 256 kc. This instrument measures the 0.004 in. cladding within $\pm$ 0.0002 in. The area of a measurement is about 0.003 square in.

A sodium-bonded, pin type fuel element of an alloy of uranium and zirconium contained in a thin walled (0.008 inch) stainless steel tube has been selected for use in the EBR-II. The fuel element is approximately 17 inches long and 0.175 inches in diameter. Normally there is a 0.006 inch annulus between the uranium rod and the wall of the tubing. The presence of voids or gas pockets in the sodium bonding layer between the uranium rod and the tubing wall will impair the heat transfer characteristics of the fuel element. The fact that the fuel element may not be symmetrically located in the stainless steel can is not considered to be objectionable. Likewise, variations in the inside diameter or the outside diameter of the stainless steel cladding are not considered objectionable.

A number of EBR-II fuel elements have been tested for void or gas pockets using eddy currents. Both circular type coils and point probe type coils were used. Some results are shown in Figure 21. The top trace was obtained using the circular coil and the lower four traces were obtained using the point probe which had a diameter of $\frac{1}{4}$ inch. Each successive test was made with the fuel element rotated 90 degrees from its previous position starting at an arbitrary zero degree point. This figure shows that the point probe was able to detect a void on one side of the fuel element from positions not only directly above the void but also with the probe at angles as much as 180 degrees away, measuring around the fuel element circumference. The voids in these traces are the short peaks and valleys. The long rises and falls are caused by variations in the thickness of the sodium layer under the probe due to the uranium rod being warped or off center. This effect is partially cancelled when the circular test coil is used due to its symmetry. At frequencies in the neighborhood of 25 kilocycles per second, the eddy currents flowing in the stainless steel have a very minor effect compared to those flowing in the sodium because of the thinness of the can and the much greater conductivity of the sodium. The test coil effectively "sees" through the can.

Some of these fuel elements were stripped by the simple expedient of cutting the can open at one end and then peeling off the stainless steel in a spiral, using a pair of long nose pliers. It was necessary to identify and photograph the voids quickly as they appeared since the sodium oxidized very rapidly and voids quickly became unidentifiable or changed in appearance. Because of the crudeness of the stripping procedure, it was difficult to determine if all the voids extended from the can to the uranium, but apparently the majority did. Figure 22 shows typical traces.

As can be seen from Figure 22, there is a definite correlation between the amplitude of the void indication and the length of the void around the

Fig. 21—Test results using point probe.

Fig. 22—Correlation of stripping and test results.

circumference, and naturally, a correlation between the length of the void indication on the recorder trace and the actual length of the void along the rod axis. As usual, the finite size of the test coil and its area of influence made interpretation of the recorder traces difficult for the smaller voids, and blurred the edges of the larger ones as shown by the testing instrument. These recordings were made using the circular type coil.

* * * *

Generalizations about the selection and application of nondestructive tests are not of much value since each actual case is a specific problem and undoubtedly has some features about it which make it unique and different from other applications. Some of the factors which must be considered when searching for or selecting the best test method are as follows: type of material, method of fabrication, shape and thickness of material, processing treatments, surface conditions, location and size of possible or expected defects, the sensitivity and resolution required or desired and how standards and specifications are to be determined. The various nondestructive tests are not competitive but rather supplement each other. An understanding of the basic principles of each method of testing is essential if the possibilities, advantages and limitations of each method are to be appreciated. It should be emphasized that destructive tests must be made during the development of a new testing technique to determine precisely what kind of defects or structural variations are being found. Although most of the tests are simple in principle it is the application of the techniques and interpretation of the test results that require skill and experience.

The science of nondestructive testing is developed to a state that it is now accepted as a useful and dependable tool. The results are reproducible and do give information about the condition of material both in the raw and fabricated states. However, as nondestructive testing is primarily used today, it gives qualitative results. What is really needed is qualitative and quantitative information. For example, ultrasonic transmission can be used to determine lack of bond. However, the question to be really answered is how good or how bad is the bond - what is the bond strength. Eddy current techniques can be used for locating cracks in uranium rod, but does not tell how the cracks effect the tensile strength of the rod. Radiography will detect porosity but does not tell the effect on the mechanical properties of the weld.

In the field of nuclear reactor components, especially fuel elements, nondestructive tests have been used as a guide to improve the uranium metal, both castings and rolled, to improve the cladding and techniques for canning uranium. During these development programs, old methods and techniques have been improved, new techniques have been developed and special instruments evolved. Particular efforts have been made to increase the sensitivity and resolution.

In order to fulfill the objectives and potentials of nondestructive testing it must be made part of the fuel element cycle. It is advisable to test the various components, then the finished product. Earlier in this conference, we heard that in order to achieve economical nuclear power we must have long fuel element life. In order that nondestructive testing may contribute to achieving this goal, there is a need for not only more nondestructive testing but also the need for better nondestructive testing.

Paper 13

THE ECONOMICS OF NUCLEAR POWER GENERATION

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ABSTRACT

Capital costs and fuel-cycle costs for nuclear power are examined to determine the costs ^{presently} existing today and the probable future trend. Particular emphasis is given to the effects of enrichment, fuel type and cladding, core design, and reactor type.

INTRODUCTION

From the economic standpoint, the introduction of nuclear power will depend upon its cost relative to other means of generating electricity, such as by hydroelectric or fossil-fueled plants. In countries such as England, France, and Italy, the economic need for nuclear power is much more pressing than it is in the United States. Coal in these countries is becoming increasingly scarce and expensive. Oil must be imported, primarily from the Middle East, where its supply is likely to be cut off at any time by the uncertain political situation.

The urgency for nuclear power development in Great Britain, for example, may be illustrated by Table I,¹ which compares the delivered cost of coal for a recent seven-year period in Britain and the United States. In Britain, coal has risen in price over 50% (to 47 cents per million Btu), while in the United States coal prices have been almost steady. Since coal is over half again as expensive in Britain as in the United States, a cheaper source of fuel such as is promised by nuclear power is attractive.

Table II shows a comparison of large central station plants, both conventional and nuclear, in the United States and Britain. It is important to note the pairs of figures in parentheses, which represent the capital charge and load factor for each case. In Britain, conventional power facilities are government-owned, so that the interest and amortization charges for invested capital are only 6.2% per year. Thus the capital charges are low, and, in spite of the high fuel cost (5.0 mills/kwh), the total generating cost is only 7.2 mills/kwh.²

The capital costs of conventional power plants in the United States have been rising somewhat (as have most construction costs) and are now believed to be \$160/kw or above for large power plants. The capital charge rate for private utility plants in the United States is high (about 15% per year for each dollar of investment), but the low fuel cost of 2.6 mills/kwh results in a total generating cost in the range of 6.5 to 7.0 mills/kwh.

The difference in capital-charge rates between the United States and Britain has a decided effect upon nuclear power economics in the two countries. Although the British gas-cooled reactor is expensive (about \$360/kilowatt for construction in Britain) the low capital charge rate and cheap fuel element give estimated nuclear power costs only about one mill higher than Britain's conventional power costs for plants under construction today.

TABLE I
DELIVERED COST OF COAL - U.S. AND BRITAIN

Year	Britain		U. S.	
	Price/Ton	<u>¢/Million Btu</u>	Price/Ton	<u>¢/Million Btu</u>
1950	\$ 7.11		\$ 6.38	
1951	7.87		6.42	
1952	8.26		6.54	
1953	8.77		6.52	
1954	9.05		6.25	
1955	10.00		6.01	
1956	10.86	47	6.29	30

TABLE II
POWER-GENERATING COSTS FROM CONVENTIONAL AND FROM NUCLEAR PLANTS

	Conventional Plants		Nuclear Plants (Early 1960's)			
			Britain		U. S.	
	<u>Britain</u>	<u>U. S.</u>	<u>(Gas-Cooled)</u>		<u>(Boiling Water)</u>	
	\$/kw					
	168	180	360		225+	
<u>Mills/kwh:</u>					\$12 Pu Credit	\$30 Pu Credit
Capital	1.7 (6.2-70)	3.4 (15-80)	4.8 (8.7-75)	5.8 (15-75)		5.8
Fuel	5.0	2.6	2.9	3.0		1.8
Opg	<u>0.5</u>	<u>0.5</u>	<u>0.6</u>	<u>0.9</u>		<u>0.9</u>
	7.2	6.5	8.3	9.7		8.5

In the United States, using quite optimistic nuclear generating costs as taken from a recent study by Cohen and Zebroski,³ nuclear power costs are two or three mills higher than conventional power costs, depending on the credit used for plutonium. Other studies are even less optimistic about United States-built nuclear power plants, giving figures in the range of 11 to 14 mills/kwh for large plants based upon today's technology.

With their greater need for nuclear power and their more favorable financing terms, it is not surprising that European nations have planned more intensive nuclear power plant construction programs than has the United States. Table III

TABLE III

INTERNATIONAL PLANS FOR NUCLEAR POWER DEVELOPMENT

	<u>Planned EMW by 1965</u>	<u>% of Total Generating Capacity</u>	<u>% of Total Electricity Generated</u>
United States	1600	1.6	2.7
France	850	3	5-6
Russia	2,000-3,000	8	12
United Kingdom	5,000-6,000	15	25

shows the announced plans of several countries.⁴ The less ambitious construction plans of the United States compared to France, Russia, and England are evident.

CHARGES FOR PLANT INVESTMENT

The capital cost of nuclear plants is expressed most conveniently in \$/kw of net plant output. Figure 1 is a recent AEC chart indicating the effect of output on plant cost.⁵ The vertical arrows show the drastic rise in estimated capital cost as designs have become more firm. (It is interesting to note that every revision has been upward). Early nuclear plants will cost at least double a conventional plant of the same output, although the nuclear plants appear to be closing the gap as the output increases. The unit capital cost of nuclear plants in \$/kw appears to vary to the -0.4 power of the output. Put another way, total nuclear plant cost increases as the 0.6 power of the output. This same relation is often used in the chemical industry also to approximate the effect of output on plant cost.

To determine the annual capital charge against the power produced, the unit capital cost in \$/kw must be multiplied by the annual capital charge rate which includes interest, return on investment, taxes on profit, and plant amortization, and divided by the plant innage factor. Figure 2 shows the wide ranges of capital charges in mills/kwh which may be obtained depending upon the assumptions used. The top line (15% capital charge, 80% plant factor) is commonly employed in the United States for nuclear plant evaluations.

FUEL CYCLE COSTS

In most nuclear power plant designs, the important factors in the fuel-cycle cost are:

1. The exposure level achievable
2. The enrichment of U-235 employed
3. The fabrication cost for the fuel

These factors are interdependent. The enrichment and the fabrication cost are components of the total cost per kilogram of fuel, while the exposure level determines the heat available for power generation per unit weight of fuel. Figure 3 shows this relation. Since the exposure level appears in the denominator of the expression for mills/kwh, the curves are hyperbolic in shape.

If zirconium or stainless steel is used for cladding, the fuel-cycle cost including fabrication and burn-up costs is presently in the neighborhood of \$100/kg. Figure 3 indicates that it is absolutely essential at this high cost to achieve exposure levels of over 6000 Mwd/ton. This is quite difficult with natural uranium, and hence enrichment is desirable.

Fig. 1—Capital cost of nuclear powerplants.

Fig. 2—Fixed charges for power generation.

Fig. 3—Effect of exposure level on fuel-cycle cost.

Burn-up Cost

Since the heat and power output of most thermal reactors varies almost directly with the grams of U-235 consumed, the burn-up cost in mills/kwh is in turn proportional to \$/g U-235 consumed. Figure 4 shows this latter cost for various burn-ups as a function of the enrichment of the fresh fuel.

This relation can be derived directly from the published AEC price scale for enriched uranium. The inner scale on the ordinate also shows the approximate burn-up cost in mills/kwh, assuming typical values for the thermal efficiency of the power plant and for the heat output from plutonium fission. The upper set of curves is for no plutonium credit, while the lower curves are based upon 0.6 g of plutonium recovered per gram of U-235 consumed, using a plutonium credit of \$12/g.

There are several significant aspects of these curves:

1. Above about 2% initial enrichment, the burn-up cost is almost independent of the enrichment and of the exposure level.
2. Below 2% initial enrichment, a given burn-up cost may be achieved by employing natural or low-enriched uranium for a moderate exposure level or by achieving longer exposure levels through somewhat more enrichment. If exposure level is limited by radiation damage, then at any given exposure important savings can be achieved by lowering the enrichment (which might be accomplished by employing zirconium instead of stainless steel cladding or by modifying the core design).
3. For slightly enriched reactors, the recovery or recycle of plutonium and probably also of spent uranium will be economic.

The savings made possible by reducing the enrichment are illustrated by a recent study of Benedict.⁶ Benedict concluded that zirconium is economically attractive as a substitute for stainless steel in reactors employing natural uranium or only slight enrichment, such as the Canadian NPD reactor, the sodium-graphite reactor, and (marginally) the Yankee reactor. Zirconium is not as attractive for reactors employing higher enrichments, such as the power plant for the nuclear ship, Savannah, or the fully enriched Vallecitos reactor. This conclusion would be expected, since as shown in Figure 4 the reduction in initial enrichment made possible by zirconium results in large savings in burn-up cost at low enrichments but hardly any savings at the higher enrichments. Benedict also found that zirconium was more attractive at higher burn-ups, which would be expected from the larger slopes of the curves for burn-up cost at the longer exposure levels. It was also concluded that a reactor using zirconium for permanent core parts and stainless steel for the fuel-element cladding might be desirable even at fairly high enrichments. The reason for this is that zirconium for permanent core parts is a one-time cost and hence may permit a net reduction in the cost of burn-up and uranium inventory.

Fuel Fabrication Cost

As mentioned above, a major component in the fuel-cycle cost today is the preparation and cladding of the fuel, particularly when zirconium is employed.

The part of the fuel-cycle cost assigned to fuel-element^a fabrication is generally assumed to include all materials, processing, pelleting, assembly, inspection, and scrap recovery costs from the uranium ingot or UO₂ powder stage

^a A fuel element is defined here as the entire assembly or bundle of fuel-bearing plates, tubes or rods (and accompanying support structure, sheathing, or shroud) which is removed from the core during refueling.

Fig. 4—Burn-up cost of uranium fuel.

to the finished element. Use charges and conversion costs are not included since they are itemized separately in the over-all fuel-cycle cost account.

Figure 5 is a graph of fabrication cost as a function of the zirconium to uranium ratio for several zirconium-clad elements.⁷ Metallic plate, UO_2 pellet, and concentric UO_2 tubular-type elements are included. It is apparent that the cost is strongly influenced by zirconium content regardless of configuration. The correlation should not be interpreted too literally, however, since type, enrichment, quantity, specifications, tolerances, and other factors all influence the final cost. In many cases the cost estimates themselves are quite nebulous. For example, a spread of 300% in the cost was obtained when five commercial fuel element manufacturers were asked to supply engineering estimates for a typical element. Variations of this order of magnitude are frequently encountered, particularly for elements with which there has been little or no previous experience.

The dependence of cost on zirconium content is due to the high cost of Zircaloy and the fact that this material is very difficult to fabricate. As the ratio of zirconium to uranium increases, the cost of zirconium (per pound of uranium) and the actual cost of fabrication (per pound of uranium) both increase. Zircaloy costs will generally represent about 30% to 50% of the total fabrication costs.

In civilian power reactor fuel elements, stainless steel is used primarily as a canning material for oxide pellets. Highly enriched oxide dispersions are used in stainless steel elements of the plate or tubular type for military applications (APPR) and for one or two experimental reactors (SRE, EBR-II), which employ metallic rods canned in stainless steel tubes. With these exceptions, and a few undeveloped conceptual designs, however, stainless steel for water-cooled reactors is generally considered only as a cheap and reliable substitute for the Zircaloy tubes normally used to clad oxide pellets. The substitution results in about a ten-fold reduction in materials costs and a substantial reduction in actual fabrication charges. Pelleting costs (\$7 - \$12 per pound) are, of course, unchanged.

The unit cost of fabricating elements consisting of oxide pellets in stainless steel tubes is not strongly influenced by the stainless steel-uranium ratio because of the much lower material costs. A relationship similar to that in the zirconium clad cases may exist, but the effect is minor in the range of tube diameters and cladding thickness generally considered.

Figure 6 is a graph of fabrication cost as a function of pellet diameter. It is apparent that elements with pellet diameters less than one-quarter inch are very expensive on a unit cost basis. The increase in cost with smaller diameters is due primarily to the increasingly smaller weight of uranium per inch of tubing. In this small-diameter range, the stainless steel-uranium ratio increases rapidly, tubing costs increase, and fabrication costs become prohibitive.

Plutonium Credit

The U. S. AEC has agreed to pay at least \$30/g for plutonium until the early 1960's, and the longer-range "fuel cost" of plutonium is believed to be about \$12/g. As a rough rule of thumb, a plutonium credit of \$12/g reduces the fuel-cycle cost by about one mill/kwh, while a credit at \$30/g reduces the fuel-cycle cost by somewhat more than two mills/kwh.

Over-all Fuel-Cycle Costs

Table IV shows the major components of the fuel-cycle cost (excluding burnup) for normal and slightly enriched uranium and two ANL plate-type metal elements used in a recent study of heavy-water reactors.¹ This tabulation illustrates the

Fig. 5—Effect of zirconium content on fuel element fabrication costs.

Fig. 6—Effect of pellet diameter on fabrication cost of stainless steel- UO_2 fuel elements.

TABLE IV

FUEL CONVERSION AND FABRICATION COSTS*

	\$ / kg U	
	Normal	Enriched
	U	U
Ship UF ₆	\$ -	\$ 0.03
UF ₆ → Metal	0.90	1.50
Manufacture Fuel Elements:		
Element A	57.00	57.00
Element B	75.00	75.00
Ship Elements	0.03	0.03
Ship Irradiated Elements	5.00	5.00
Chemical Processing	20.00	20.00
Ship Uranium Nitrate	0.04	0.04
UN → UF ₆	5.60	5.60
Ship UF ₆	0.14	0.14
Subtotal		
Element A	\$ 88.71	\$ 89.34
Element B	106.71	107.34

*Not including burn-up cost, plutonium credit, inventory charge, or conversion of plutonium nitrate to metal.

importance of zirconium fabrication costs. Although the burnup costs are not shown, they are quite low because of the low enrichment needed in the heavy-water reactor.

The effect of the principal fuel-cycle variables can be illustrated by Fig. 7, which shows graphically some calculations performed on a heavy-water reactor core of given diameter and lattice spacing. Each of the circled points indicates an enrichment, exposure level, and fuel-cycle cost in \$/kg for a particular core design. The dotted lines radiating out from the origin are lines of constant fuel-cycle cost in mills/kwh. As enrichment is increased, both the attainable exposure level and the per-kilogram fuel-cycle costs will increase. The core design giving minimum fuel-cycle cost in mills/kwh is seen to be at the upper right-hand corner of the figure at an enrichment of slightly over 1.3% U-235 and a fuel-cycle cost of about 2.2 mills/kwh.

As an example of an element which has a lower fabrication cost but higher burnup cost, Table V shows a recent tabulation of a stainless steel fuel element cost employing 3% U-235 as the initial fuel.⁸ The difference between the upper and lower limits for the fissionable material costs is due to the change in assumed plutonium credit from \$12 to \$30 per gram. The other cost differences are simply due to optimism or pessimism.

Fig. 7—Effect of enrichment on fuel-cycle cost.

Plutonium Recycle

Particularly in Europe, there is interest in reactor designs which may require slight enrichment for the initial loading but can subsequently be operated on natural uranium and recycled plutonium. Preliminary studies have been made⁹ which indicate that in the Calder Hall type of a gas-cooled reactor it will probably be about as economic to recycle plutonium in this reactor (and thereby obtain higher exposure levels) as to sell the plutonium produced at \$12/g.

TABLE V

Fuel Cost Breakdown for Hypothetical PWR
(Stainless Steel, UO_2 , 7,500 MWD/Ton, 140 EMW)

	Upper Limit			Lower Limit		
	Total Cost Per Core (\$ x 10 ⁶)	$\frac{c}{10^6 \text{ Btu}}$	$\frac{\text{Mills}}{\text{kwh}}$	Total Cost Per Core (\$ x 10 ⁶)	$\frac{c}{10^6 \text{ Btu}}$	$\frac{\text{Mills}}{\text{kwh}}$
Fissionable Material	2.55	16.6	2.0	0.97	6.3	0.8
Fabrication	2.00	13.0	1.6	0.30	2.0	0.2
Reprocessing	0.46	3.0	0.4	0.28	1.9	0.2
Shipping Spent Fuel	0.22	1.4	0.2	0.16	1.0	0.1
Misc. Processing	1.25	8.1	1.0	0.41	2.7	0.3
Inventory Charge	0.83	5.4	0.7	0.83	5.4	0.7
Totals	7.31	47.5	5.9	2.95	19.3	2.3

TABLE VI

DESIGN HEAT FLUXES FOR PRESENTLY PROPOSED REACTORS

	Average Heat Flux, Btu/hr-sq ft
<u>Boiling</u>	
EBWR	44,000 - 100,000
Dresden	120,000
Elk River	74,000
<u>Pressurized</u>	
PWR (Blanket)	65,000 - 100,000
Yankee	100,000
Consolidated Edison	169,000
Belgian Thermal Reactor	96,000
Canadian D ₂ O NPD Initial Design	100,000

The Thorium Cycle

The foregoing remarks have applied to the uranium-plutonium cycle in thermal reactors. There is also considerable interest in the thorium - U-233 cycle, particularly since conversion ratios in the neighborhood of 1.0 appear possible in thermal heterogeneous or homogeneous reactors.

The economics of the thorium cycle are quite sensitive to the conversion ratio, since this affects not only the U-233 make-up cost but also the attainable exposure level. In a recent appraisal of this cycle,¹⁰ it was stated that remote or semi-remote fabrication of both the recycled thorium and U-233 might be necessary, but that this can hopefully be accomplished at a sufficiently low cost to make this cycle attractive.

FEATURES OF CURRENT REACTOR DESIGNS

In an over-all look at current power reactor designs, the following points are worth noting:

1. Most designs are quite conservative in certain key respects. As an example, the design average heat fluxes in several reactors are tabulated in Table VI. These average fluxes are only about one-fifth or less of burnout heat fluxes and can probably be appreciably exceeded as operating experience is gained. Already in the case of the EBWR and the Vallecitos reactor these conservative design heat fluxes have been bettered.

2. Because of the present high cost of fuel conversion and fabrication high exposure level is essential, and this means enrichment. As an example, some results from a parametric study of heavy-water uranium metal cores are shown in Table VII.⁷ Attainable reactivity exposure levels for different core diameters, lattice spacings, and enrichments were calculated assuming a "one-batch" loading method and a continuous charge-discharge method advocated by the Canadians for their NPD reactor.¹¹ The calculated fuel-cycle costs show the beneficial effect of large core diameters and lattice spacings and of slight enrichment on the fuel-cycle cost. (However, large core diameters adversely affect pressure-vessel cost, and large lattice spacings reduce heat transfer surface, so that an optimization must be carried out).

If continuous charge-discharge can be accomplished at reasonable cost, or if conversion and fabrication costs can be reduced, then the natural uranium reactor may compete economically with the slightly enriched reactors.

3. There is no clearly superior power reactor type. Apparent superiorities are often due to a judicious selection of the ground rules, so that there may well be "different winners in different races," depending upon the set of economic assumptions most appropriate to a given situation. As an example, Table VIII shows cost estimates for small (90 EMW) Calder Hall and PWR reactor types as they would compare under U. S. utility accounting practices and under the British Central Electricity Authority practices.¹² It is evident that the two reactor designs appear almost equally attractive under the U. S. rules, but that for the same technical designs the British rules minimize the high capital cost penalty of the Calder Hall type and make it appear definitely more economic than the PWR.

A further comparison of reactor types of larger power output is given in Table IX, based on U. S. economic ground rules. These figures show the small variation of power costs from different reactor designs. (The small 20 EMW heavy-water reactor design was included to illustrate the drastic effect of reactor output on the power cost).

TABLE VII

EFFECT OF CORE DESIGN ON EXPOSURE LEVEL AND FUEL-CYCLE COST IN
HEAVY WATER REACTORS

Fuel Element: Nine Zr clad plates, each 0.12 in. thick.
Total fuel plate width 30 in. Coolant channel width 0.35 in.

	Lattice Spacing (in.)	Uranium Enrichment % U-235	BATCH LOADING		CONTINUOUS CHARGE DISCHARGE	
			Maximum Exposure Level, MWD/Metric Ton	Fuel-Cycle Cost Mills/KWH	Maximum Exposure Level, MWD/Metric Ton	Fuel-Cycle Cost Mills/KWH
<u>9-ft Diameter</u>	7	0.71	Subcritical	5.46	Subcritical	3.22
		0.8				
		1.3				
	8	0.71	4,020	4.18	7,720	2.07
		0.8				
<u>11-ft Diameter</u>	7	0.71	Subcritical	4.23	Subcritical	2.27
		0.8				
		1.3				
	8	0.71	5,370	3.34	10,600	1.68
		0.8				
<u>13-ft Diameter</u>	8	0.71	6,800	2.90	Exposure extremely high	
		0.8				

TABLE VIII

EFFECT OF ECONOMIC GROUND RULES
ON NUCLEAR POWER COSTS

Mills/kwh:	Case 1		Case 2	
	Construction of reactor in the U. S. Capital charges based on U. S. private utility accounting practices.		Construction of reactor in Britain. Capital charges based on British Electricity Authority accounting practices.	
	Calder Hall	PWR	Calder Hall	PWR
<u>Invested Capital</u>				
Construction Cost	12.8	7.4	4.8	2.8
Capital Charges				
During Construction	1.2	0.8	-	-
Startup Cost	0.1	0.1	-	-
Investment in First Core	<u>0.3</u>	<u>1.1</u>	<u>0.1</u>	<u>0.4</u>
Total	14.4	9.4	4.9	3.2
<u>Production Costs</u>				
U-235 Burnup Cost	1.4	2.0	1.4	2.0
Plutonium Credit	<u>1.4</u>	<u>0.5</u>	<u>1.4</u>	<u>0.5</u>
Net Burnup Cost	0.0	1.5	0.0	1.5
Rental of Fuel	0.4	0.2	0.5	0.2
Fuel Fabrication and Processing	<u>1.6</u>	<u>7.5</u>	<u>1.6</u>	<u>7.5</u>
Total	2.0	9.2	2.1	9.2
Site Operating Cost	<u>1.5</u>	<u>1.0</u>	<u>1.0</u>	<u>0.7</u>
Total Power Cost	17.9	19.6	8.0	13.1

TABLE IX

U. S. POWER COSTS FROM VARIOUS REACTOR TYPES

	<u>PWR</u>	<u>BWR</u>	<u>Calder Hall</u>	<u>D₂O Pressurized</u>	
Net Power Output (MW)	134	180	150	250	20
Plant Cost (\$ MM)	39.2	45.0	-	75.5	13.9
\$/kw	342	250	450	302	695
Enrichment (% U-235)	2.6	1.5	0.71	1.1	1.1
<u>Power Cost (mills/kwh)</u>					
Capital charge (15%)	7.3	5.3	9.6	6.4	14.9
Fuel-Cycle Costs	2.7	3.1	1.5	1.9	1.9
Total Inventory Charge	1.0	1.0	0.4	0.4	0.5
D ₂ O Losses	-	-	-	0.4	1.7
D ₂ O Inventory	-	-	-	0.8	3.0
Operating and Maintenance	<u>1.0</u>	<u>1.0</u>	<u>1.0</u>	<u>1.0</u>	<u>1.8</u>
Total Power Cost	12.0	10.4	12.5	10.9	23.8

DUAL-PURPOSE REACTORS

A reactor type which may be of interest is a large "convertible" reactor which is built with government funds and is operated for a period of years to produce weapons-grade plutonium. The reactor is then converted by appropriate changes in fuel element dimensions, cladding, pressure, and temperature and by the addition of a turbogenerator plant to produce fuel-grade plutonium (worth \$12/g) and electric power for an additional period of dual-purpose operation. The economics of such a reactor can be compared against both the straight production reactor and a reactor operated to produce both plutonium and power from the date of startup. Figure 8 shows the calculated unit cost of the weapons-grade plutonium from a water-cooled, graphite-moderated reactor as a function of the date at which the type of operation is modified, assuming a power credit of 7 mills/kwh.¹³

As would be expected, in the dual-purpose case with private operation in the second phase, the unit cost of weapons-grade plutonium decreases steadily as a function of the amount made (which is proportional to the years of operation to produce this plutonium grade). On the other hand, the results for the convertible reactor (which is operated with no power output during its period of weapons-grade plutonium production) depend upon whether the plant is operated as a power producer by the government or by a private utility. For the government, with its low interest rate and no taxes, operation as a dual-purpose reactor is profitable, and hence the sooner that the reactor is converted to power production the lower will be the net unit cost which must be charged for the weapons-grade plutonium. A private utility, however, has a higher capital charge rate and cannot afford to pay very much for the government-owned plant at the end of the weapons-grade plutonium phase. Hence the net cost charged to each unit of weapons-grade plutonium produced is higher the earlier that weapons-grade production is terminated and production of fuel-grade plutonium plus power is initiated.

Fig. 8—Dual-purpose reactors. Initial operation to produce weapons-grade Pu, perhaps power. Subsequent operation to produce fuel-grade Pu plus power.

NUCLEAR POWER COSTS IN THE FUTURE

Based upon the available evidence of today, there are certain conclusions which can perhaps be drawn:

1. Present concepts of heterogeneous water-cooled reactor in the United States operating on the uranium-plutonium cycle have a long and difficult road to travel before they can compete with conventional power plants unless a substantial credit is paid for plutonium. In Europe, because of higher conventional generating costs and a lower capital charge rate, the water-cooled reactors may become competitive in a relatively few number of years.
2. There are significant improvements possible in present-day water reactors such as nuclear superheat, cheaper fuel claddings, the use of the thorium cycle, and the elimination of control rods and costly precision in core dimensions by using moderator level to control reactivity.
3. The really basic breakthroughs needed to make central station atomic power economically attractive in the U. S. have not been demonstrated. There are a number of reactor concepts which in the longer run may outstrip the water-cooled reactor:
 - a. The high temperature gas-cooled reactor
 - b. The homogeneous reactor
 - c. The liquid metal or fused salt reactor
 - d. The fast breeder
4. In the turbine part of the plant, a significant improvement in efficiency may be possible by utilizing the mercury turbine cycle rather than the steam cycle. Figure 9 shows the over-all thermal efficiency possible with a binary mercury-water cycle as recently estimated by Thompson Products, Inc.¹⁴

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DISCUSSION

D. P. HERRON

MR. M. BENDER (Oak Ridge National Laboratory): You commented concerning the higher fuel burn-up costs for highly enriched reactors. Do you feel it is worthwhile to concentrate on reducing the burn-up cost at this time or should the concentration be on reducing capital costs and ignoring the burn-up charge which is, roughly, 1 to 2 mills, until such time as the capital cost charges are down close to steam-plant current costs?

MR. HERRON: I think in the heterogeneous reactors, where the fabrication costs are always appreciable, reducing enrichment is quite important because you have to get the whole fuel cycle cost down roughly to 2 mills in order to make atomic power attractive. I know that in the homogeneous reactor, highly enriched fuel is almost mandatory and the fabrication is non-existent, and I think that in that type of reactor you can go ahead with the high enrichment. It is difficult for me to see in the heterogeneous thermal reactor that enrichment above 1-1/2% is really going to be economic in the long run, but I think in other reactors it could possibly be so.

MR. BENDER: This depends somewhat on the fuel fabrication costs.

MR. HERRON: Right, it does, but they are all pretty high.

MR. BENDER: And if you have a long fuel lifetime?

MR. HERRON: Figure 4 of the paper showed that, if you use highly enriched material, a long fuel lifetime does not help you much in the burn-up cost.

MR. BENDER: I agree with that but all I am saying is that until you get to the point where the capital charge is down very close to that of fossil fuel plants there is no point in concentrating on reducing burn-up charges.

MR. HERRON: Well, I think most people feel that going along both routes is probably the best way to do it. It is true that in the homogeneous reactor you don't worry about this higher burn-up cost because you hope that the capital cost will be lower.

MR. BENDER: How do you reconcile the relatively low construction costs of steam plants in this country compared with Great Britain? The report is that fabrication costs in Britain are something like 50% higher.

MR. HERRON: In a Construction Cost Index the capital cost items, labor costs, and construction overhead are weighted. Labor costs are considerably lower in Great Britain, and the capital cost for a given design in this country is considerably higher than it is in Great Britain.

MR. BENDER: Why isn't this reflected in their conventional steam plant construction?

MR. HERRON: Well, I think the main reason is that the British so far have not gotten up to our operating conditions. They use smaller plants with lower steam conditions. Although I don't recall the exact figures, up to the present time at least most of the British plants are just not as efficient as our plants. I think if they built a comparable design in Great Britain their total construction cost ought to be a little cheaper than ours.

MR. S. SIEGEL (Metals and Controls, Inc.): I should like to throw this out for comment. I plotted a curve, that was inspired by one of your remarks, of the cost of fabrication of a fuel element as a function of the experience of the vendor. These are arbitrary units over here. This is fuel cost. This is one and this is two and this is three, and this is the experience of the vendor in years, zero, one, two, three, and four. I think it has been our experience that the curve goes something like this, with the initial portion having some variation that may include a maximum point. In the initial bid the price may be fairly low. This is a function of how clearly one understands the specifications and how daring the management is and how much the people decide that this is going to be a growing business and therefore, they must get into it. It is during the intermediate period that the full appreciation of the specifications and the requirements begins to sink in. As described in his paper, Mr. McGonnagle begins to find nondestructive tests which don't destroy the product but destroy the business and you begin to realize that you must have 99% confidence that the fuel elements are good. About this point you have a clear understanding of what you are making and your customer has a clear understanding of what he wants. About this time you know what your fuel costs are. You don't have much experience in your plant yet, your people are still learning, and you have not done much to cut these costs.

It is from this point on that you begin to get a real saving in these fabrication costs both because of your own experience in fabrication and because your customer has had called to his attention the areas in which he can effect a saving without losing the quality of the product that he requires. I think the reason for your three-fold variation that you mentioned is just this situation right here.

MR. A. R. KAUFMANN (Nuclear Metals, Inc.): When you figure the cost of power with various enrichments do you take into account the fact that it costs more to handle enriched uranium? For example, somebody has to pay for the uranium that disappears. It costs more to recycle it, to reprocess it, etc., and if you get into highly enriched material you get into criticality. Have you taken that factor into account?

MR. HERRON: We took that factor into account. It shows up, I think in the UF_6 - UO_2 reduction. That is an expensive step. It is a very sensitive function in enrichment. I think that is another argument that is often not taken into account enough in considering the benefit of only slight enrichment.

MR. KAUFMANN: We recently made some cost estimates on a fuel that was about 30% enriched and it turned out that just the cost of the uranium that disappeared was greater than the whole cost of doing the job.

Paper 14

PLATE-TYPE, UO_2 -BEARING FUEL ELEMENTS FOR THE PWR CORE 2 BLANKET*

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ABSTRACT

Plate-type fuel elements, containing UO_2 fuel are of special interest for the PWR (Pressurized Water Reactor) Core 2 blanket because of their increased thermal performance compared to the rod-shaped elements used in PWR Core 1 and their potential for reduced core manufacturing costs. Since the UO_2 fuel is not bonded to the relatively thin Zircaloy-2 cladding, provision must be made for the lower mechanical rigidity of plates versus rods and for possible deflection of the cladding caused by release of fission products during irradiation. The required rigidity has been achieved by compartmentation of the unbonded fuel. Release of fission products has been restricted to tolerable levels by employing UO_2 fuel platelets cold-pressed and sintered to densities of 96% of theoretical. A method of bonding the cladding components, which involves eutectic-diffusion-bonding and uses nickel, copper or iron as the bonding agent is described in detail. This method of bonding provides considerable design flexibility and permits the use of fuel and cladding components which are fabricated to finished dimensions by inexpensive and established commercial operations prior to assembly for bonding. Information is presented on mechanical properties and corrosion resistance, which are of particular concern for PWR application of such fuel elements.

INTRODUCTION

Included in the design objectives for the second PWR Core are 1) higher thermal performance (100 MW gross electrical power output), 2) increased life (10,000 full power hours for the seed) and 3) lowered fabrication costs. To achieve these objectives a radically new type of fuel element for the PWR Core 2 blanket is under development. This element has a plate-type geometry designed to provide a larger blanket heat transfer surface area than is available for the PWR Core 1 rod blanket. High density uranium dioxide, pressed and sintered into a plate geometry, is used as the fuel to take advantage of its high irradiation stability.^{1, 2} As in the Core 1 blanket rods, Zircaloy-2 is used as the fuel cladding. Because the ceramic uranium dioxide fuel is not metallurgically bonded to the cladding, a plate-type fuel element has inherently less mechanical rigidity than a rod-type fuel element. It

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must be provided, therefore, with supports to prevent excessive deflection of the cladding caused by possible fission product release during irradiation. A method of fabricating such a plate-type fuel element, involving compartmentation of the ceramic fuel and joining of cladding components by a novel and low cost process, is described in this paper. Methods of evaluating such elements and properties of significance for reactor application are also discussed.

DESIGN OF PLATE-TYPE, OXIDE-BEARING FUEL ELEMENT

Figure 1 illustrates schematically the Zircaloy-2 cladding and fuel components of a proposed design of compartmented oxide, plate-type fuel element. It will be noted that the cladding components consist of three members, i.e. a central fuel receptacle plate, which is compartmented to receive the fuel, and two relatively thin and solid cover plates. After assembly of the fuel and cladding components the latter are joined together, as will be described later, into a completed fuel element of high integrity in which all fuel compartments are gas tight.

No dimensions are included in Fig. 1 since final sizes of both fuel and cladding components have not been established. These will depend for the most part on design and property requirements such as required heat transfer surface area, allowable deflection of cladding over fuel areas, effects produced by the presence of intentional cladding defects during irradiation in a hot water environment, etc. It is apparent, however, that there is considerable flexibility of design in an element of this type since fuel geometry and size may be varied over wide limits without materially affecting fabrication techniques.

METHOD OF FABRICATION

The cladding and fuel components are fabricated to final dimensions and inspected for dimensions and quality prior to assembly and joining into the completed fuel element. The cladding components are prepared from rolled sheet which are cut to size. The central fuel receptacle plate is then compartmented by die punching to receive the fuel components. The fuel components are fabricated by cold pressing and sintering to a density in excess of 96% of theoretical using manufacturing procedures similar to those described for the PWR Core 1 rod-shaped pellets. A significant difference in the manufacture of the plate-shaped fuel components is the higher density compared to Core 1 pellets which result in greater retention of fission gas.² Another significant difference is the elimination of expensive sizing operations such as grinding. The critical thickness dimension of the platelets can be maintained within ± 0.001 in. by pressing and sintering using automatic equipment. Length and width dimensions are not critical since it is desirable to maintain some void space (in the order of 5% in each fuel compartment). These dimensions can be held within desired tolerances for geometries of interest without difficulty. Thus, all of the fuel element components are fabricated using well established, low cost commercial practices.

The development of manufacturing procedures for bonding the cladding components into a high integrity, leak tight fuel element with dependable, strong and ductile bonds received major attention. Several possible methods of bonding were investigated. Among these were 1) fusion welding, 2) solid state diffusion bonding, 3) brazing and 4) a new procedure developed expressly for the application and identified as eutectic-diffusion-bonding. As the

Fig. 1—Schematic representation of cladding and fuel components for a compartmented oxide, plate-type fuel element.

work progressed it was apparent that the first three methods, while showing promise, involved serious manufacturing problems which required considerable time and effort for their successful solution. Eutectic-diffusion-bonding, however, was found to be relatively simple, inexpensive and gave maximum assurance of strong bonds and fuel compartment integrity in the finished product. This method, which is described below, was selected for continued development.

Eutectic-diffusion-bonding, as applied to the joining of the Zircaloy-2 cladding components of the plate-type, compartmented oxide fuel element, involves the formation of a molten eutectic alloy film between the solid mating surfaces to be joined. This eutectic alloy film is formed by diffusion of a preplaced thin layer of pure metal, which serves as a bonding agent, on the surfaces of the fuel receptacle plate during heating of the assembled components to the bonding temperature. On reaching the melting point of the eutectic alloy, the latter melts and effects bonding of the solid and higher melting point cladding components. Holding the assembly at the bonding temperature for an extended period of time then results in solidification of the molten film and diffusion of the bonding agent throughout the mating cladding components. Thus, no solid eutectic alloy layer remains in the bond interface as is the case for brazing.

Important features of the eutectic-diffusion-bonding process involve:

- 1) selection of the bonding agent, 2) method of preplacing the bonding agent, 3) method of assembly of components to ensure intimate contact of surfaces to be bonded and 4) the actual bonding cycle. A brief discussion of these features follows:

Nickel, copper or iron are possible bonding agents. The lowest melting point eutectics which these elements form with zirconium melt at 961, 885 and 934°C, respectively. These melting points are well below the melting point of zirconium, about 1850°C. The limit of solid solubility of these elements in beta zirconium (1.9, 3.8 and 5.5 weight percent, respectively) will permit adequate solution and diffusion of the relatively small amount required to effect bonding. Of these elements nickel and iron are alloy constituents of Zircaloy-2. Their use as bonding agents, therefore, do not introduce foreign elements into the cladding components. However, the lower melting point of the copper-zirconium eutectic and its relative ease of preplacement on Zircaloy-2 are factors in its favor. The final selection of the bonding agent will be based on thorough evaluation of bonded samples in out-of-pile and in-pile tests.

Chemical replacement techniques have been found most satisfactory for preplacing nickel, copper and iron bonding agents on the surfaces of the Zircaloy-2 fuel receptacle plates. This process involves solution of zirconium in an acid bath containing dissolved nickel, copper or iron salts and simultaneous replacement of the dissolved zirconium with the desired bonding agent. This method is preferred over electroplating or other methods because of its simplicity, because no hydrogen evolution and contamination is involved, and because uniform thicknesses of coatings are obtained despite variations in compartment geometries. Thicknesses of coatings can be controlled within narrow limits (± 0.02 mils) by proper adjustment of plating time, bath temperature and bath composition. Desirable coating thicknesses are in the range of 0.15 to 0.25 mils. A minimum coating thickness, consistent with achieving satisfactory bonds is preferred to avoid large changes in cladding composition which might adversely affect corrosion resistance in high temperature water. Copper deposits can be easily applied using a room temperature bath. Nickel and iron plating, however, is best done in elevated temperature baths (90-100°C). This method of plating is more fully described in reference 4.

Because very thin deposits of bonding agents are employed, it is imperative that intimate contact be maintained between the cladding surfaces to be joined during the bonding operation. This is done by sealing the assembled fuel element components by fusion welding the outside periphery and applying a light, but positive, inert gas pressure (15 to 30 psig) to the external cladding surfaces during the eutectic-diffusion-bonding operation. The assembly may be edge welded in air using an argon or helium shielded arc to prevent contamination of the cladding with nitrogen and oxygen. Any small amount of air which may be entrapped in void spaces within the element will not effect the bonds since it will be gettered by the cladding material on heating during bonding. During bonding, therefore, a vacuum will be present within the sealed element. The positive inert gas pressure on the external surfaces of the element, combined with the lowered yield strength of the cladding material at bonding temperatures, will ensure intimate contact of mating surfaces to be joined.

Bonding conditions can be varied somewhat depending on the eutectic melting points of the bonding agents employed and their relative diffusion rates in Zircaloy-2. In general, however, a bonding temperature of about 1000°C and time of several minutes to several hours have produced strong bonds as will be indicated in the following section. Heating and cooling rates have a significant effect on results. It is found that a fast heating rate is necessary to avoid excessive diffusion of the bonding agent into the cladding before the eutectic melting point is reached. A fast cooling rate (in excess of 30°C per minute) is necessary to minimize precipitation and redistribution of alloying constituents in the cladding. This occurrence, on slow cooling, adversely affects the hot water corrosion resistance of the cladding. Fast heating rates can be achieved without difficulty by transferring the fuel element assembly from a cold zone to a hot zone of an evacuated and inert gas pressurized furnace. This procedure can be reversed for fast cooling. A schematic drawing of such a furnace is shown in Fig. 2.

RESULTS AND DISCUSSION

Numerous plate-type, compartmented oxide fuel element samples have been fabricated using the eutectic-diffusion-bonding method described above. Out-of-pile evaluation tests are underway. Some of the significant results obtained thus far are discussed below.

Appearance and Metallographic Structure

The general appearance of fuel element samples after bonding is indicated in Figs. 3 and 4. It will be noted that the cladding has been depressed over the fuel area due to the differential pressure during bonding. The amount of cladding depression depends, of course, on the difference in thickness between fuel and fuel receptacle plate. The fact that this depression does occur is a good indication that the cladding components are well bonded. If it did not occur, it would indicate that the element had not been properly sealed and no differential pressure existed during bonding.

Fig. 5 and 6 illustrate the metallographic structure of bonds obtained in the solid edge margin sections and in the compartment rib areas when using copper as the bonding agent. It will be noted that no layer of undiffused bonding agent or eutectic is present in the joints. The structure is essentially continuous except for the appearance of a composition gradient. Such gradients are more apparent for copper and iron bonded samples since these elements have a somewhat lower diffusion rate than nickel in Zircaloy-2 at the bonding temperature employed.

Fig. 2—Schematic representation of sample and furnace arrangement for eutectic-diffusion-bonding of assembled plate-type oxide-bearing fuel element.

Fig. 3—General surface appearance of eutectic-diffusion-bonded sample (0.19 mils Cu, bonded at 1000°C for 30 min under 15 psig external pressure). Surface has been vapor blasted and etched. (Approx. 0.75X)

Fig. 4—Cross section of a eutectic-diffusion-bonded sample showing compartments and fuel. Sample was bonded at 1000°C for 30 min at 15 psig pressure using 0.15 mils of copper. No copper was used on inside surfaces of compartments. (2X)

Sample
Surface

Bond at
Sample edge
margin

Fig. 5—Metallographic structure at edge margin bond for cross-sectioned sample shown in Figure 4. (100X)

Sample
Surface

Bond at
Compartment
Rib

Fig. 6—Metallographic structure at compartment rib for cross-sectioned sample shown in Figure 4. (100X)

Some small, isolated voids are present in the bond interface. Bond line voids are formed to some degree regardless of the bonding agent used. They appear to be related to contamination of the mating Zircaloy-2 surfaces during the etching and plating operation. When proper control is maintained over these operations, the bond line voids are small (up to about 0.005 in. long and 0.001 in. wide), isolated, and do not adversely affect bond strength or compartment integrity. (A more detailed discussion of this subject is given later.)

Interaction of fuel and cladding has been noted in bonded samples. This interaction, which is illustrated in Fig. 7, results from the reduction of UO_2 by zirconium and subsequent diffusion of reaction products, oxygen, and uranium into the cladding. Uranium penetration of the cladding is quite superficial, while oxygen penetration, as indicated by the white, alpha-zirconium layer, is appreciable. The extent of this solid-state interdiffusion is a function of the length of time the fuel and cladding are held in intimate contact at the bonding temperature. For short bonding periods (30 min at 1000°C and 15 psig pressure), oxygen from the UO_2 -zirconium reaction diffuses about 0.0025 in. into the cladding. This reaction zone is of importance only in the event that a defect exists in the cladding and communicates to the outside surface. In such a case, some corrosion of internal compartment surfaces would occur in a hot water environment. The extent of this corrosion effect has been determined and is discussed later. This reaction can be eliminated by the placement of a thin barrier layer of graphite between the fuel and cladding. This layer is applied to the fuel by spraying. Metallic or non-metallic coatings such as niobium, molybdenum, inert oxides, or graphite over the fuel are not effective in eliminating this type of reaction.

A second type of reaction with the fuel has been observed. This is caused by contact between the fuel and molten eutectic at the edges and corners of compartments during bonding. However, if the bonding agent is applied only on the mating cladding surfaces and not on the internal edges of fuel compartments, this type of fuel-clad interaction is eliminated as indicated in Fig. 8. Figure 9 illustrates fuel-clad interaction resulting from contact of fuel with molten eutectic when plated deposits are not removed from compartment edges.

Integrity of Bonds

Bond integrity was determined by internally pressurizing intentionally defected compartments after bonding. Holes, 0.040 in. diameter, were machined through one of the cladding cover plates into the fuel of each compartment. The compartments were then individually pressurized to 700 psig of helium (below the cladding yield stress) and leaks through the bonds of adjacent compartments detected by escaping helium. As indicated in Table I, 535 compartments from 74 separate samples bonded under a variety of conditions were found, without exception, to be gas tight.

Bond strength, was determined by two destructive tests. In the first test a qualitative indication of strength was obtained in cross-sectioned samples by attempting to tear the cladding cover plate off the receptacle plate. If failure occurred through the bonded areas, bonds were considered weak. If failure occurred through the unbonded cladding over the fuel, the bonds were considered stronger than the cladding itself. In the case of nickel-bonded samples, Table I indicates that bond strength is not reliable if the thickness of bonding agent employed is less than about 0.18 mils. This appears to be related to increased bond porosity the lower the nickel thickness employed. For copper-bonded samples deposits as thin as 0.09 mils appeared satisfactory but bonding temperatures less than about 980°C prevented adequate diffusion of copper and resulted in weak bonds. For iron-bonded samples, deposit thickness and bonding temperatures similar to those for copper appear satisfactory. However, bonding times in excess of about 2 hr result in enlargement of bond voids and decreased bond strength. Thus,

Sample
Surface

Fuel-clad
interaction
zone
Fuel

Fig. 7—Extent of fuel-clad interaction at fuel surface for cross-sectioned sample shown in Figure 4. (100x)

Fig. 8—Metallographic structure in compartment rib area of cross-sectioned sample shown in Figure 4. (40X)

Fig. 9—Extent of interaction between molten eutectic and fuel on sample copper-diffusion-bonded at 1000°C for 15 min under 15 psig when 0.13 mils of copper deposit is not removed from compartment edges. (75X)

TABLE I

INTERNAL INTEGRITY OF FUEL COMPARTMENTS AND BOND STRENGTH AS AFFECTED BY BONDING CONDITIONS

<u>Bonding Agent</u>	<u>Bonding Agent Thickness, Mils</u>	<u>Bonding Conditions</u>			<u>Compartment Integrity</u>			<u>Bond Strength</u>	
		<u>Temp. °C</u>	<u>Time, Hr</u>	<u>Press. psig</u>	<u>Samples Tested</u>	<u>Compt's Tested</u>	<u>% Compt's Gas Tight</u>	<u>Failure Location*</u>	<u>Burst Press., psi</u>
Ni	0.09-0.17	1010	1/6-2	60	9	-	-	Bond	2400-2600
Ni	0.18-0.22	1010	2	60	23	155	100	Clad	2650-4500
Ni	0.21	1010	4	60	1	10	100	Clad	3600-3800
Cu	0.09-0.20	980-1054	1/4-3	15-30	21	165	100	Clad	2750-4300
Cu	0.13	925	1/4	15	1	10	100	Bond	- -
Cu	0.19	925	1/4	15	1	10	100	Bond	- -
Cu	0.25-0.39	980-1054	2-3 1/2	30-60	8	110	100	Clad	2880-3700
Cu	0.26	925	2	30	1	15	100	Bond	- -
Fe	0.12-0.22	980-1010	1/4-1/2	15-30	7	40	100	Clad	3300-3400
Fe	0.25	1025	2	30	1	10	100	Clad	3000-3780
Fe	0.25	980	3	30	1	10	100	Bond	- -

Note: Each fuel element sample had overall dimensions of 2.375 x 11.25 x 0.120 in. and contained 15 fuel compartments, 0.25 x 3.0 x 0.080 in., separated by 0.080 in. wide ribs.

* Failure location determined using qualitative cladding tear test. All failures occurred through the cladding at compartment edges on internal pressure burst tests.

optimum deposit thickness, bonding temperature and bonding time are dependent on the particular bonding agent employed.

A second type of test for bond strength involves internal pressurization of compartments until failure occurs. It will be noted from Table I that bursting pressures for the 0.25 x 3.0 x 0.08 in. compartments are all in excess of 2400 psi. Failure occurs at the long edge of compartments adjacent to the bonded ribs where the stress is a maximum. It will be noted also that relatively high bursting pressures are obtained even for samples which displayed weak bonds in the qualitative tear test. Only when extremely poor bonding occurs, due to too low a deposit thickness or improper bonding conditions, does failure occur through the bonds rather than the cladding on internal pressure burst tests.

The bursting pressure, elastic and plastic deflection which occurs on internal pressure testing depends on the geometry and size of the fuel compartment. This is illustrated in Fig. 10. Cladding deflection is recorded at the center of the outside surface of the compartment being pressurized. Maximum deflection and maximum bursting pressure is realized with the small square or narrow rectangular compartments and increases, as expected, with compartment size. Elastic deflections correlate well with calculated theoretical deflections. At elevated temperature (600°F) the bursting pressure does not change appreciably but the onset of plastic yielding occurs at lower internal pressures. This, however, does not appear to be of major significance since the fuel element would be subjected to 2000 psi external coolant pressure during reactor operation.

Strength and Ductility of Bonded Zircaloy-2

Zircaloy-2, as normally employed in cladding applications, has an equiaxed alpha grain structure which provides a good combination of strength and ductility. During eutectic-diffusion-bonding, however, the cladding is heated above the alpha-beta transformation temperature (about 1500°F). On subsequent cooling the characteristic Widmanstätten structure of prior beta grains is retained. This structure, which cannot be reconverted to equiaxed alpha without unreasonably long time heating in the alpha phase field, is stronger and somewhat less ductile than the equiaxed structure. The ductility of the beta treated Zircaloy-2 is reduced from 32 to 24% at 500°F. However, its yield strength is increased from 18,000 to 24,000 psi. Strength and ductility of beta-treated Zircaloy-2, therefore, appear more than adequate for cladding applications.

To establish the effect of introducing bonding agent into the cladding by diffusion during bonding, 0.025 in. thick sheets of Zircaloy were nickel, copper or iron diffusion bonded and their tensile properties determined. These are listed in Table II. The results indicate that the strength properties are equal to or somewhat higher than unbonded, beta-treated Zircaloy-2 while the ductility is reduced only slightly at 650°F. Thus, the strength properties of eutectic-diffusion-bonded Zircaloy-2 are superior to those of alpha material. The small reduction in ductility which results does not appear to be a cause for concern.

The introduction of excess nickel, iron or copper into the cladding during bonding does not result in any age-hardening-effects. Long time aging in the temperature range of 650 to 750°F shows no measurable hardness increase. The same is true for aging temperatures in the range of 1350 to 1475°F. At the latter temperature agglomeration and spheroidization of the excess metallic constituent occurs in about 20 hr, but no appreciable hardness or strength property changes occur even with several days exposure in this temperature range.

Fig. 10—Effect of compartment size and geometry on cladding deflection and bursting pressure.

TABLE II

TENSILE PROPERTIES OF BONDED AND UNBONDED ZIRCALOY-2

Bonding Agent	Deposit Thick. Mils	Bonding Conditions			Test	Tensile Properties		
		Temp. °C	Time, Hr	Press. psig	Temp. °F	0.2% Yield Str. psi	Ult. Str. psi	Elong. %
Equiaxed alpha Zircaloy-2 (annealed 8 hr/1450°F)					75	40,500	68,900	27
Beta Quenched Zircaloy-2 (air cooled from 1850°F)					75	47,200	61,200	16
Ni	0.20	1010	2	60	75	48,600	70,000	16
Cu	0.15	990	2	15	75	60,200	73,500	12
Cu	0.20	1000	1/2	15	75	50,000	68,200	10
Cu	0.20	990	2	15	75	52,000	68,500	14
Fe	0.20	1000	1/4	15	75	50,000	64,700	12
Equiaxed alpha Zircaloy-2 (annealed 8 hr/1450°F)					500	18,100	38,600	32
Beta Quenched Zircaloy-2 (air cooled from 1850°F)					500	23,500	33,700	24
Ni	0.20	1010	2	60	650	22,200	32,400	20
Cu	0.15	990	2	15	650	22,500	32,500	18
Cu	0.20	990	2	15	650	22,000	33,900	15
Cu	0.20	1000	1/2	15	650	24,000	33,500	17
Fe	0.20	1000	1/4	15	650	21,400	29,700	17

Sample gauge dimensions: 1.00 in. x 0.150 in. x 0.050 in. (sheet type)

Behavior of Bonded Elements in Out-of-Pile Water-logging Tests

Since the compartmented oxide fuel is not bonded to the cladding it would be anticipated that the formation of a defect which penetrates the cladding of any given compartment after bonding would result in filling of the compartment void space with the water coolant during reactor operation. On subsequent rapid heating the water inside the compartment would flash to steam, exert a high internal pressure on the unsupported cladding and result in swelling of the cladding over the fuel in the defected compartment. The extent of this swelling would be dependent on the size of the defect, the size and geometry of the compartment, the amount of water entrapped in the compartment and the rate of heating or steam formation. To obtain some idea of the severity of such an occurrence bonded samples containing 0.25 x 3.0 x 0.08 in. compartments (0.025 in. thick cladding cover plates) with compartment void volumes in the order of 5%, were defected with 0.005 and 0.040 in. diameter holes through the cladding. The defected compartments were then waterlogged in 650°F water under 2200 psi pressure. After water-logging the samples were rapidly immersed in 650°F molten lead and the amount of compartment swelling recorded.

It was found that swelling due to rapid steam formation does not occur when the defect is 0.040 in. in diameter. Apparently, this size of defect is large enough to permit escape of steam as fast as it is formed. However, the small diameter defect does result in cladding deflection. This is indicated in Table III. In most cases more than one water-logging cycle is required to cause cladding deflection. As the cycles are repeated, however, deflection does occur. Eventually, the cladding is severely deformed and it fractures at the compartment edge where stress is a maximum. The variability in amount of swelling noted in Table III for each water-logging cycle is attributable to variations in the size of actual defects. The fuel for these samples were not graphite coated. Therefore, fuel-clad interaction occurred and it was difficult to machine clean-cut, 0.005 in. diameter holes through the hard interaction layer.

Behavior of Bonded Elements in Out-of-Pile Thermal Cycling Tests

Thermal cycling of individual fuel elements or welded assemblies of fuel elements has not presented any distortion or dimensional change problems. Sample elements, having overall dimensions of 12 in. x 2.5 in., with thickness varying from 0.080 to 0.120 in. and with square and rectangular fuel compartments of various sizes, have been thermally cycled between 1100°F and 75°F. Rapid heating by immersion in a carbonate salt bath and rapid cooling in an air blast for as many as 200 cycles (20 minutes at each temperature) did not result in any measurable dimensional changes.

Corrosion Resistance of Bonded Elements

The oxide film formed on the external cladding surfaces of eutectic-diffusion-bonded fuel elements on exposure to 680°F water or 750°F steam is not unlike that of unbonded Zircaloy-2 which has been described elsewhere.⁵ Weight changes are essentially the same as for alpha or beta Zircaloy-2 provided the beta treated material and the bonded elements are rapidly cooled (at a rate in excess of 30°C/minute) from the beta phase field.

Of particular interest is the corrosion behavior of diffusion-bonded samples which have 0.040 in. diameter defects machined through the compartment cladding so that the internal compartment surfaces and fuel are exposed to the corrosive environment. Tests in 750°F degassed steam, 680°F degassed water and 680°F degassed water with a hydrogen over pressure of 300 psi,

TABLE III

BEHAVIOR OF INTENTIONALLY DEFECTED COMPARTMENTS
IN WATER-LOGGING TESTS

Bonding Agent	Deposit Thickness, Mils	Maximum Cladding Deflection, mils Per Water-Logging Cycle									Total Deflection, Mils
		1	2	3	4	5	6	7	8	9	
Cu	0.16	0	0	0	0	0	5	0	0	0	5
		0	0	0	0	0	0	0	0	0	0
		0	0	0	0	0	0	0	0	0	0
		0	0	0	0	7	16	0	0	0	23
		0	0	0	0	0	0	0	0	0	0
		0	0	0	0	0	0	0	0	0	0
		0	0	0	0	0	0	0	0	0	0
		0	0	0	0	0	0	0	0	0	0
		13	0	0	0	0	0	0	0	0	13
		0	0	0	0	0	0	0	0	6	0
Ni	0.15	0	0	25	0	0	0	13	0	0	38
		0	0	0	0	0	0	14	8	36	58
		0	21	4	0	0	0	2	0	22*	49
		0	11	0	16	0	0	81	0	5	113
		0	0	0	0	5	0	0	0	0	5
		0	0	0	0	3	0	54	0	0	57
Ni	0.20	0	0	0	0	0	0	13	0	7	20
		0	0	0	0	0	4	8	0	0	12
		0	0	0	0	0	6	8	9*	-	23
		0	0	0	0	0	17	7	0	12	36
		0	0	0	0	0	6	26	0	0	32
		0	0	22	0	12	0	0	0	17	51
		0	0	0	0	2	0	13	0	7	22
		0	0	9	0	13	0	3	0	0	25
		0	0	10	0	33	0	5	0	0	48
		0	0	0	0	13	0	11	0	20	44

Notes: 1. All compartments are 0.25 x 3.0 x 0.080 in., contain approximately 5% void volume, and are defected with a 0.005 in. diameter hole.

2. Cladding thickness is 0.025 in.

3. Bonding agent was not removed from compartment edges before samples were bonded.

* Cladding fractured at compartment edge.

listed in Table IV, indicate that the exposed bonds do not display accelerated corrosion effects. The cladding on some compartments swelled a maximum of about 15 mils due to corrosion of fuel-clad interaction products. Since the bonding agent was not removed from the compartment edges of these samples this swelling was anticipated. The extent of swelling, however, was relatively small despite the build-up of corrosion products. It was apparent that once the corrodible material was spent no further swelling occurred. Internal pressure tests performed on the individual compartments at 700 psi after the indicated days of exposure confirmed that the compartments did not inter-communicate and bond areas did not corrode through.

Corrosion tests in 680°F degassed water with 300 psi over pressure of hydrogen are not included in Table IV for samples in which nickel was used as the bonding agent. The high hydrogen environment, about 6000 cc/kg of water, in this test resulted in accelerated absorption of hydrogen by the cladding of the nickel-bonded samples. As a result, the cladding was rapidly embrittled and cracked. This occurred in the peripheral edge weld and, more extensively, in compartment rib areas where the nickel concentration is highest. Fig. 11 illustrates this effect in a sample exposed 7 days in this environment. This embrittlement does not occur in copper or iron-bonded samples even in longer time simultaneous exposure to the same environment as is illustrated in Fig. 12.

Analysis of nickel, copper and iron-bonded samples for hydrogen absorption during corrosion indicates that introduction of nickel into the cladding markedly accelerates hydrogen absorption while iron and copper do not. Table V, for example, indicates that beta treated Zircaloy-2 will absorb only a small fraction of the hydrogen produced as a part of the corrosion process of zirconium in water. The initial hydrogen concentration of the original material, 50 to 75 ppm, rises only a small amount on exposure for 19 days to 680°F water. Nickel-bonded samples, however, pick-up relatively large amounts of hydrogen in both 750°F degassed steam and in 680°F water containing an over pressure of 300 psi of hydrogen. The amounts absorbed are higher, the higher the nickel concentration as evidenced by the thickness of the nickel bonding agent and the variation in hydrogen concentration in the weld and compartment rib areas. The latter areas contain a higher concentration of bonding agent because the latter was deposited on both the mating cladding surfaces as well as on unbonded compartment edges. While hydrogen pick-up in unbonded Zircaloy-2 is not influenced by molecular hydrogen dissolved in water, it is apparent that the nickel-bonded samples will pick-up not only corrosion produced hydrogen but also hydrogen present in the corrosion environment.

Copper and iron-bonded samples apparently react like unbonded Zircaloy-2 with reference to hydrogen pick-up during corrosion. The hydrogen concentration in weld areas correlates with anticipated pick-up from the corrosion reaction. This is true also for rib areas. The higher hydrogen concentrations in these areas can be explained by a somewhat increased rate of corrosion resulting from some fuel-clad interaction due to contact of molten eutectic with the fuel during bonding. This effect, as previously explained, is largely eliminated when the bonding agent is removed from fuel compartment edges prior to bonding.

SUMMARY

A method of fabricating plate-type fuel elements containing compartmented ceramic fuel has been described. This method, which involves eutectic-diffusion-bonding of the Zircaloy-2 cladding components has been shown capable of producing fuel elements of considerable design flexibility at relatively low cost. The method permits efficient utilization of uranium

TABLE IV

CORROSION OF EUTECTIC-DIFFUSION-BONDED FUEL ELEMENTS AFTER
DEFECTING INDIVIDUAL COMPARTMENTS

Sample No.	Bonding Agent	Deposit Thick. Mil	Bonding Conditions			Corrosion Test Conditions		Exp. Days	Max. Comp. Swelling (in.)
			Temp. °C	Time Hrs	Press psig	Environment			
H38-5	Ni	0.20	1010	2	60	680°F degassed water		53	.005
H45-1	Ni	0.21	1010	2	60	"		53	.005
H46-1	Ni	0.21	1010	2	60	"		53	.005
H47-1	Ni	0.18	1010	2	60	"		46	.005
H53-1	Ni	0.21	1010	2	60	"		46	.005
H55-1	Ni	0.19	1010	2	60	"		46	.010
H62-1	Ni	0.26	1010	2	60	"		39	.010
H64-5	Ni	0.22	1010	2	60	"		36	.005
H65-1	Ni	0.26	1010	2	60	"		36	.005
H63-1	Cu	0.25	1010	2	60	"		36	.010
H67-1	Cu	0.25	1010	2	60	"		36	.010
H82-1	Cu	0.13	1010	1/4	15	"		17	.010
H93-1	Cu	0.18	980	1/2	30	"		28	.005
H95-1	Cu	0.18	980	2	30	"		28	.005
H105-1	Fe	0.12	1000	1/2	30	"		14	.005
H63-5	Cu	0.25	1010	2	60	680°F water +300 psi H ₂		39	.005
H80-5	Cu	0.12	1010	1/4	15	"		25	.005
H82-5	Cu	0.13	1010	1/4	15	"		25	.005
H84-5	Cu	0.19	1010	1/4	15	"		25	.005
H93-5	Cu	0.18	980	1/2	30	"		7	.005
H95-5	Cu	0.18	980	2	30	"		7	.005
H76-5	Fe	0.22	1000	1/2	30	"		25	.010
H105-5	Fe	0.12	1000	1/2	30	"		7	.005
H39-5	Ni	0.19	1010	2	60	750°F degassed steam		55	.005
H41-5	Ni	0.19	1010	2	60	"		55	.005
H45-5	Ni	0.21	1010	2	60	"		55	.005
H46-5	Ni	0.21	1010	2	60	"		55	.005
H53-5	Ni	0.21	1010	2	60	"		48	.005
H83-1	Cu	0.13	1010	1/4	15	"		28	.005
H85-1	Cu	0.19	1010	1/4	15	"		28	.005
H87-1	Fe	0.19	1010	1/4	15	"		21	.015

- Notes:
1. Five compartments (0.25 x 3.0 x 0.08 in.) in each sample
 2. All compartments were defected with a 0.040 in. diameter hole through the cladding.
 3. No intercommunication of compartments was observed at 700 psi internal pressure test after corrosion testing.

Fig. 11—Structure of edge margin and compartment rib areas of a nickel-diffusion-bonded sample after exposure for 7 days to 680°F degassed water with an over pressure of 300 psi of hydrogen. (18X)

Fig. 12—Structure of edge margin and compartment rib bonds of iron-diffused-bonded sample after exposure for 11 days to 680°F degassed water with an over pressure of 300 psi of hydrogen. (30X)

TABLE V

HYDROGEN ABSORPTION DURING CORROSION TESTING OF SECTIONED EUTECTIC-DIFFUSION-BONDED ELEMENTS

<u>Type of Sample</u>	<u>Days</u>	<u>Environment</u>	<u>Hydrogen Concentration,</u> ppm	
			<u>Weld Area</u>	<u>Rib Area</u>
Beta Quenched Zircaloy-2	19	680°F Degassed Water	-	56
Beta Quenched Zircaloy-2	15	680°F Degassed Water with 300 psi H ₂ over pressure	-	81
Bonded with 0.20 mil Ni	11	680°F Degassed Water with 300 psi H ₂ over pressure	523	837
Bonded with 0.22 mil Fe	11	680°F Degassed Water with 300 psi H ₂ over pressure	91	112
Bonded with 0.13 mil Cu	11	680°F Degassed Water with 300 psi H ₂ over pressure	106	185
Bonded with 0.18 mil Ni	34	750°F Degassed Steam	285	434
Bonded with 0.26 mil Ni	27	750°F Degassed Steam	423	2150
Bonded with 0.22 mil Fe	14	750°F Degassed Steam	88	146
Bonded with 0.12 mil Cu	14	750°F Degassed Steam	67	70

Note: Original H₂ concentration of cladding was in the range of 50-75 ppm

dioxide as a fuel in a plate-type configuration to take advantage of its high irradiation stability in a fuel element of high thermal performance.

Out-of-pile evaluation of eutectic-diffusion-bonded fuel elements has indicated that strength and ductility are quite adequate. Nickel has been shown to be unsatisfactory as a bonding agent because it results in excessive absorption of hydrogen and embrittlement of the cladding during corrosion testing. Copper and iron, in the amounts employed, do not adversely affect corrosion resistance and appear to be satisfactory bonding agents. Judicious application of the bonding agent only to surfaces to be bonded prevent undesirable reactions with the fuel. Some solid state fuel-clad interaction is to be expected during bonding but the extent of this reaction is small and can be minimized by proper choice of bonding conditions. Compartmentation of unbonded ceramic fuel can lead to water-logging type failures if a cladding defect communicates with the outside surface of the element. Proper design of fuel compartment size and geometry as well as proper spacing of fuel elements in assemblies are means of restricting damage which might occur from this source.

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Paper 15

MANUFACTURING METHODS FOR THE EXPERIMENTAL BREEDER REACTOR MARK I AND MARK II FUEL LOADINGS

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ABSTRACT

This paper describes the EBR-II fuel cycle, the reactor core subassembly and fuel element. The proposed methods for manufacturing the U-235 fueled Mark I loading and the uranium-plutonium fueled Mark II loadings are described in some detail. The initial loadings will be made using a simulated equilibrium fissium alloy. These will be manufactured by non-remote methods but using the same operational sequences as the recycle material. The recycle loadings will be pyrometallurgically processed, refabricated, and reassembled by remote control in a special facility under construction adjacent to the reactor. A brief description of this plant and the remote control process is given.

The Experimental Breeder Reactor II, scheduled to be completed in 1960, will be fueled first with an enriched uranium-fissium alloy and later with an enriched uranium-plutonium-fissium alloy. Both fuel loadings will complete a number of reactor-reprocessing-refabrication cycles. The enriched uranium-fissium alloy is referred to as the Mark I loading while the plutonium containing fuel is designated as the Mark II loading.

The EBR-II has been designed not only to test a complete power cycle including reactor plant and power plant, but to include a complete fuel cycle with facilities for reprocessing and refabricating the reactor fuel and reassembling it into fuel elements for return to the reactor. Remote controlled, non-aqueous methods will be used for reprocessing and refabricating the fuel without decontaminating it to the levels required for conventional fabrication. These unconventional methods are made necessary by the short reactor cycle, relatively high burn-up and intense radioactivity of the fuel, which would result in a prohibitively large fuel inventory if conventional complete decontamination by aqueous methods were employed.

Figure 1 shows the EBR-II reactor in cut-away section. The reactor assembly, primary coolant pumps and heat exchangers, fuel cooling, storage racks, and equipment for charging fuel elements will operate submerged in a 20,000 gallon tank of primary coolant sodium. Figure 2 shows the reactor assembly and the positions of the fuel core and the inner and outer blanket assemblies within the neutron shield. These are 61 core subassemblies containing fuel elements and 66 inner blanket subassemblies contained in a central,

Fig. 1 — EBR-II reactor.

Fig. 2—EBR-II reactor shield, tank, blanket, and core assembly.

high velocity coolant channel. The core subassemblies are interchangeable allowing the fuel section to be expanded or contracted and giving flexibility to the core configuration. Five hundred ten outer blanket subassemblies are contained in a separate annular coolant channel. The blanket subassemblies contain depleted uranium as the fertile material. Control of the reactor is by twelve movable fuel subassemblies which are similar in design to the stationary subassemblies. Two similar core subassemblies which may be ejected from the core act as the safety rods.

The core subassemblies are as shown in Fig. 3. Each subassembly contains 19 upper and 19 lower depleted uranium blanket elements and 91 fuel elements. These are enclosed in a hexagonal stainless steel tube. A forked plug, welded to the lower end of the tube, engages the reactor support grid. The coupling at the top serves as the handle for the fuel loading manipulator socket. Openings at the bottom and at the top of the subassembly provide for coolant flow.

The fuel elements are as shown in Fig. 4. The fuel is an equilibrium alloy pin which measures 0.144 in. diameter by 14.22 in. long. This pin is contained in a stainless steel tube 0.174 in. outside diameter by 0.009 in. wall thickness. A 0.006 in. annulus between the pin and the tube is sodium filled to provide a thermal bond. A spiral wire on the outside of the tube serves to space the fuel elements in the core subassemblies. The forked clutch at the lower end of the tube engages miniature rails on a grid subassembly holding the fuel element in place. A restrainer-cap is welded to the upper end of the fuel element in final assembly. This serves as a closure, provides a void space into which the sodium may expand and it also holds the fuel pin below the surface of the sodium.

The following requirements were established for the EBR-II fuel:

1. The concentration and total quantity of the fissile material must be sufficient for the reactor to become critical and to operate at the required power level over the reactor operating cycle. With the fuel occupying approximately 32% of the core volume and by simulating the distribution of stainless steel, sodium and fission products, it was determined that fuel alloy containing 49 wt % of U-235 would be required in the reactor. Less information on the composition of plutonium plus U-235 is available at this time for the Mark II loading; however, 20% plutonium appears to be approximately the maximum for metallurgical considerations. Uranium-235 will be added to the concentration to bring the reactor to criticality. It appears at this time that approximately 13 wt % of U-235 will be needed in addition to the 20 wt % of plutonium.
2. The core should exhibit a negative thermal coefficient of reactivity to produce a stable control situation. The fuels should possess a positive coefficient of expansion and should not undergo phase changes causing a rapid volume contraction during either power increase or decrease.
3. The fuel must be resistant to irradiation damage within the burn-up levels required for practical and economic operation of the reactor. It must be dimensionally stable and should resist swelling. Two percent of total atom burn-up, giving a 90-day reprocess cycle at full power, was arbitrarily selected as an achievable goal for the EBR-II fuel.
4. The fuel must be capable of operation at the design temperature of the reactor and must have a sufficiently high coefficient of thermal conductivity that the central metal temperatures are kept within the materials limitations. With an effluent coolant temperature of 900°F, the central

Fig. 3 — EBR-II core subassembly. (55 subassemblies needed for typical EBR-II reactor.)

Fig. 4—EBR-II fuel element.

metal temperature of the Mark I loading is expected to be between 1100°F to 1190°F. Figure 5 shows the thermal conductivity of uranium, uranium-fissium and uranium-plutonium fissium. Because the thermal conductivity of the Mark II fuel is lower than the Mark I fuel, it may become necessary to operate the reactor with slightly lower coolant temperatures in order to keep the central metal temperature within satisfactory limits.

5. The fuel element must be of such design that it can be processed, fabricated and reassembled by remote control processes.

FUEL ALLOYS

Fissium is a name that has been applied to the alloys of uranium, plutonium and the fission products resulting from the pyrometallurgical processing of the irradiated reactor fuels. The pyrometallurgical refining process was selected rather than the more conventional chemical separations process. It consists of holding the molten metal in a fluxing or drossing crucible to oxidize the more reactive elements and at sufficient temperatures to volatilize the high vapor pressure elements. The pyrometallurgical process generally removes the non-metallic and rare earth elements. The metallic elements niobium, silver, cadmium, and antimony are partly removed while zirconium, molybdenum, technetium, ruthenium, and rhodium remain with the fuel ingot. With the unfortunate exception of zirconium, these elements produce a beneficial effect in the fuel alloy. Zirconium removal may be effected by the introduction of carbon to the melt. This unfortunately produces a bulky, difficult to handle dross and attempts have been made, rather unsuccessfully to date, to utilize the high zirconium material.

After repeated like cycles to irradiation and reprocessing, the fission alloys are expected to reach an equilibrium composition determined by the amount of new fuel added for each cycle. Assuming the addition of 5% new material per cycle, Burris and Dillon¹ have computed the Mark I fuel composition and Dunworth² has computed the Mark II fuel composition as shown in Table II. Radiation effects upon these alloys are discussed in a paper by Kittel.³ Fortunately, the equilibrium fissium alloy was found to be comparatively resistant to irradiation damage without the addition of other constituents.

THE INITIAL FUEL LOADINGS

The initial fuel charges for the Mark I and Mark II loadings will be manufactured from non-irradiated materials. An effort will be made to simulate the irradiated material as closely as possible using natural elements to stand in for the fission isotopes. Approximately 0.5 wt % molybdenum and 0.5 wt % ruthenium have been added to the composition shown in Table II to stand in for the technetium which, of course, is not available for the initial loadings. The processes used for fabrication of the initial loadings will duplicate those of the recycle loadings as closely as possible except that they will be manual rather than remote control operations.

The fabrication process was developed to be equally adaptable to the Mark I and Mark II loadings. The plutonium containing alloy is somewhat less impact resistant and must be handled with more care than the Mark I loading. The Mark II core will be initially fabricated in a plant designed to protect against plutonium hazards. This plant, the Fuel Fabrication Facility, is nearing completion at Argonne National Laboratory.⁴

RADIATION LEVELS OF THE RECYCLE MATERIAL

The fuel elements, at the time they are taken from the core, will generate about one percent of the total reactor power as a result of fission product

Fig. 5—Thermal conductivity curves uranium, uranium fissium, and uranium-plutonium fissium.

TABLE I
COMPARATIVE DATA, EBR I AND EBR II FUEL ELEMENTS

	EBR I	EBR II
Maximum Power, Megawatts	1.4	62.5
Maximum Fuel Temperature °F	643	1189
Core Length, Inches	9.0	14.22
Equivalent Core Dia Inches	7.93	19.04
No. of Subassemblies	-	61
No. of Fuel Elements Per Subass'y	-	91
Total No. of Fuel Elements	217	5551
Fuel Tube Thickness	0.015"	0.009"

TABLE II
CALCULATED EQUILIBRIUM COMPOSITION OF FISSIUM
BASED UPON 5 WT % PROCESS/ADDITION CYCLE

ELEMENT	ALLOY A ⁹	ALLOY B10
Zirconium	0.10	0.08
Niobium	0.01	0.01
Molybdenum	3.42	3.15
Technetium	0.99	0.96
Ruthenium	2.63	3.48
Rhodium	0.47	0.60
Palladium	0.30	1.35
Total Fission Elements	7.91	9.67
Balance		
Uranium	92.09	70.33
Plutonium	-	20.00

decaying heating. This heating falls off rapidly in the first few days as shown in Figure 6. The fuel elements must remain in the primary coolant tank for about 15 days before they may be cooled sufficiently by gas cooling to prevent their melting. Although this cooling period reduces the intensity of irradiation by a factor of 6, it is estimated that each fuel element will be equivalent to a 24 kilocurie source. The core subassemblies will still overheat in a static atmosphere making it necessary to force cool them by blowing an atmosphere through the coolant channels until after disassembly and dispersion of the fuel elements.

THE REACTOR FUEL CYCLE

The reactor fuel cycle is shown in the flow diagram, Figure 7. The sequence of manufacture may be broken down into six essential operations.

1. Disassembly of the core assemblies.
2. Fuel element disassembly.
3. Pyrometallurgical processing of the fuel.
4. Refabrication and fuel pin inspection.
5. Fuel element reassembly and testing.
6. Reassembly of the core subassembly.

These operations will be carried out in a special remote control process plant to be constructed adjacent to the reactor. The process plant will be as shown diagrammatically in Figure 8. The core subassemblies will be transported from

Fig. 6—Fission product decay heating as calculated for EBR-II fuel.

Fig. 7—EBR-II fuel cycle flow sheet.

Fig. 8—EBR-II fuel process plant.

the reactor plant to the process plant in a transfer coffin equipped with a gas coolant system. Reprocessed fuel elements will be returned by the same coffin.

The shielded facility is divided into two cells: a main process cell, which is circular in plan, and an adjoining rectangular air cell. The main process cell is controlled from a series of operating and viewing stations around the outside of the shield wall and from an operating turret at the center. The shield walls are five feet thick. Electrically operated, general purpose manipulators, special purpose manipulators, and crane hoists will operate from a series of radial cranes which sweep about a central control and power distribution column. The cell is sealed by a welded metal skin and will contain an argon gas atmosphere. Once in operation, it is thought that the area will become radioactive beyond possible decontamination to biological tolerances. Provision must, therefore, be made for the remote maintenance of all process equipment. All equipment must be capable of remote controlled replacement.

A shielded rectangular cell adjoins the process cell. This cell will be used for the disassembly and reassembly of the reactor core subassemblies and for transfer to and from the process cell. Transfer between cells will be made by means of vacuum locks.

All glass viewing windows were selected because of the high radiation levels encountered. These were designed so that the inner 9 in thick window slab may be changed by remote control from inside the cell. A steel shutter will be provided to close the window from the inside to protect it from radiation when it is not being used. Close viewing will be by a sealed magnifying periscope designed to fit into a standard window opening. To prevent radiation damage, all optical elements exposed to direct radiation will be reflective. The manufacturing process will be as follows.

FUEL ELEMENT DISASSEMBLY

The core subassemblies will be sheared apart in the air cell and the fuel elements transferred through a special vacuum lock into the process cell where they will pass to the decanning station. Here the stainless steel tubes will be removed by a longitudinal slitting machine developed by the Argonne National Laboratory Remote Control Division.

PYROMETALLURGICAL PROCESSING

The pyrometallurgical process developed by the Chemical Engineering Division of ANL² is an oxidative drossing process which consists of holding the fuel alloy in an oxide crucible at a temperature just above the melting point for a period long enough to allow volatilization of the volatile elements and selective oxidation of the more reactive elements. Some of these oxides are absorbed by the crucible in a manner analogous to the cupellation refinement of the noble metals while others pass into a floating dross or oxide layer. The drossing operation is carried out in a tilting induction furnace which is covered by a metal bell jar. Volatile metallic materials are collected on a cooled crucible cover. The other volatile materials will be flushed away by a flow of purified argon gas and by evacuation.

REFABRICATION

The Metallurgy Division of ANL was made responsible for the development of suitable refabrication methods. A number of possible methods of remote control refabrication were suggested to convert the processed fuel back into fuel pins but none of the conventional methods such as wire fabrication, powder metallurgy, or extrusion appeared to be adaptable to the production of the long, slender, relatively brittle, highly radioactive, and pyrophoric fuel pins. All of the

conventional processes, other than casting, produced pins by metal working which produced a preferred orientation of the structure reducing radiation stability. Because of the non-adaptability of conventional equipment and processes, it seemed desirable to develop new fabrication methods for the EBR-II fuel pins.

Several preliminary requirements were set down. It was desirable that the process have a minimum number of operations. The equipment should be of componentized design so that subassemblies can be replaced as units, requiring a minimum of adjustment and assembly work within the process cell. Manipulators should not be used for routine operation of the equipment but to perform service upon the equipment and non-routine operations. The operation, total replacement, and maintenance of the equipment should be within the limited capabilities of the manipulators and the cranes within the process cell. The construction materials should be chosen to withstand the radiation levels encountered and radiation sensitive equipment should be shielded from direct exposure to the highly radioactive fuel. Precision casting was selected as the method of greatest promise for development because it produced the required nonoriented structure and because it promised a simple sequence of manufacturing operations.

The method proposed was a simplified non-mechanical gas pressure injection precision casting process. Figure 9 shows a schematic of the injection casting machine. The fissium alloy is induction melted under a vacuum in a refractory coated graphite crucible to a temperature of 1375 to 1425 °C. The molds in the form of precision bore, high silica, glass tubes coated on the inside with a thin thoria wash and closed at the upper end are suspended open end down above the crucible. The assembly of mold tubes is heated by resistance windings within the vacuum chamber. When the molten alloy has reached the selected casting temperature, the vacuum pumping system is isolated and the ram upon which the crucible is mounted is actuated causing the crucible to rise, submerging the open tube ends. After a two second time delay to allow quieting of the metal surface, the furnace is opened to an inert gas pressurized accumulator tank causing a rapid increase in the furnace pressure. The pressure difference across the wall of the submerged tubes causes the alloy to rise into the tubes where it freezes. The crucible containing the remainder of the molten charge is lowered while the metal in the crucible is still molten but after solidification of the castings.

Because of the self-heating from fission decay, the casting array will cool to an equilibrium temperature determined by the arrangement and number of castings made at a time. This necessarily limits the total number of pins which may be cast at a time without forced cooling. An analysis showed that fewer than 60 pins could be cast at a time if a multi-concentric solid lattice were used. If a larger diameter crucible is used and the center pins are removed, it appears that 100 or more pins may be cast in a single operation. The final number of pins cast will be determined during operation.

A mold stripping and casting cropping machine has been built in which the glass molds are pulverized away from the casting by feeding the mold (with the casting) between roughened concaved rolls. The casting then passes between two pairs of contrarotating wire brushes where it is cleaned of adhering glass and mold coating refractory. From these brushes, the casting is pushed into a double shear which simultaneously crops both ends of the casting leaving a pin of 14.22 in. length. This machine is as shown in Figure 10.

The pins will be visually inspected by means of a magnifying periscope which is sealed through the wall of the cave and then will be transferred by an argon powered pneumatic pusher to an inspection device similar to that shown in Figure 11. Diameters will be checked as the pin is fed through argon jet gauges. An induction coil for eddy current inspection is located adjacent to and in line with the gauges. This will be used to sense internal flaws or surface irregularities. The pin will be pushed through the gauges

Fig. 9—Injection casting machine.

Fig. 10—Demolding and shearing machine.

Fig. 11 — Fuel pin inspection machine.

onto a semiautomatic balance for weighing. The drop of the balance pan will be sensed by a microformer with two self-bucking windings to minimize reactive forces. After weighing, the pin will be picked up by pneumatic cylinders and pushed between the anvils of a length gauge. The deflection of the gauge will be sensed by a microformer. After gauging, the pin will again be dropped into the V-channel of the balance and pushed back onto the inspection-transfer tables. The reject pins will be broken into 1-1/2 in lengths for return to the process, while good pins will be transferred to an indexing device upon which is mounted a rotary fuel tube magazine. The pins will be pushed into the fuel jackets which have been previously prepared outside the process cell with a measured quantity of sodium loaded into them.

The casting variables including furnace heating rates, degree of superheat of the alloy, mold outgassing and preheat temperature, mold coatings, crucible coatings, furnace vacuum, pressurizing rate, final furnace pressure and cooling period have all been optimized on the 1/2 scale furnace described by Yaggee et al.⁶

Satisfactory injection casting conditions were determined to be as shown in Table III. Approximately 500 pins have been produced using the optimum values determined above. Table IV shows diameter hardness, density and weight variations produced by the method. Figure 12 shows a magnified view of pin surfaces produced by the process.

The optimized conditions have been applied to full scale operating equipment which is now under test. This equipment is designed to handle 10 KG of feed material or about 120 fuel pins per batch. The equipment is being tested on cold materials under remote control conditions.

TABLE III

MELTING AND CASTING CONDITIONS GAS PRESSURE
INJECTION CASTING OF EBR-II FUEL PINS

	Range Explored	Satisfactory Range
Heating Rate, °C/min	5.5 to 26.6	10 to 26.6
Casting Temperature, °C	1275 to 1425	1325 to 1350
Vacuum, μHg	40 to 0.2	Less than 4
Mold Temperature °C	70 to 600	250 to 600
Pressurizing Rate, psi/sec	4 to 30	18 to 22
Final Pressure, psia	15 to 60	15 to 30

TABLE IV

DIAMETER, WEIGHT, DENSITY AND HARDNESS OF 5 WT %
FISSION ALLOY INJECTION CAST PINS

	Number of Observations	Average Value	Standard Deviation
Diameter, inches	659	0.1446	0.0006
Weight, grams	93	67.929	0.507
Density	109	17.939	0.016
Rockwell "A" Hardness	1414	69	--

Fig. 12—Surfaces of injection cast fuel pins.

FUEL ELEMENTS ASSEMBLY

The fuel element assembly procedures were developed by the Reactor Engineering Division of ANL. All parts of the fuel element received at the assembly station will be finished in the inspected and tested condition. The tubes loaded with fuel in the previous operation will be heated in a vacuum furnace which will melt the sodium so that the fuel pin settles into it. The top restrainer cap will be installed and welded to the tube. The assembly will then be transferred to the air cell where it will be centrifuged to assure complete filling of the annulus between the pin and the tube.

The core subassemblies will then be assembled. Preassemblies of the lower adapter plugs and lower blanket sections containing 19 depleted uranium blanket rods will be positioned in the core assembly machine. The fuel section will be assembled one element at a time by manipulator on the key rail grid which positions and anchors each element in its respective position in the core subassembly. The rather tedious manipulator operation will be assisted by a multiple slot loading fixture which assures that the elements be inserted in their proper orientation. The T-slot in each fuel element is guided into the corresponding key-rails by means of a wire guide. An automatic electrical scoreboard will be used by the assembler to assure that each of the 91 positions in the core section is filled in the required sequence. Because the assembly will now be heated by fission product decay radiation, it is necessary that it again be cooled by a forced flow of gas as the assembly operation progresses. The assembled bundle of fuel elements will then be clamped in their hexagonal close-packed positions. The hexagonal outer tube preassembled, with the 19 upper blanket elements positioned on their supporting grids and the upper coupling head, will be mechanically slipped down over the core section and the lower blanket section. The assembly will then be rotated on its axis and a profile welding head will weld the hexagonal tube to the bottom plug adapter by the inert gas shielded arc weld process.

Current status of the EBR-II fuel manufacturing program is as follows:

1. Sufficient experimental production and prototype machines have been constructed that purifying and manufacturing processes have been fixed.
2. Second stage, more completely mechanized equipment has been constructed and is under test or is under construction. This equipment may or may not be satisfactory for remote operation, but will presumably be satisfactory for the initial fuel loadings.
3. From the test and modification of this equipment, it is expected that the final equipment for remote operation may be designed.

Sufficient experience has been concluded to date to make the remote controlled pyrometallurgical reprocessing refabrication and reassembly of the EBR-II fuel appear a practical possibility and the reactor fuel cycle has been based upon the processes described.

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DISCUSSION

MR. J. E. MINOR (Hanford Atomic Products Operation): Mr. Shuck made one remark that I want to find out about. He said that there was an indication that zirconium is harmful.

MR. SHUCK: There will be further discussion of the effect of zirconium in the fissium equilibrium alloys in a paper by Mr. Kittel at a later session. We have generally found that zirconium added to the equilibrium fissium alloy causes extreme embrittlement. Microscopic inspection indicates the formation of a brittle compound in the grain boundaries. There is also indication that the γ -stabilizing effect of the fissium alloy is impaired.

MR. A. R. KAUFMANN (Nuclear Metals, Inc.): I wonder whether you can give us some idea of the degree of perfection that you get in these castings. I am thinking of the secondary type of imperfection: cold shuts on the surface, scabs, and that sort of defect.

MR. SHUCK: We have had a certain amount of difficulty with the type of surface condition mentioned. Our first attempts to cast directly into the silica glass produced an extremely poor surface caused by the reaction of the metal with the glass and most of the experimental work was done on coatings which could be applied to the glass to improve these conditions. We feel now that we have the casting conditions fairly well under control through control of the temperature of the molds, gas pressure, rate of pressurization, and the delay period before pressurization of the furnace. Density, dimensional control, and hardness are within rather close limits as shown by Table IV. We are presently producing about 80-85% acceptable pins. All of the pins that we have produced from the non-radioactive metal have been radiographed and the ratio of acceptable pins has been brought up from about 30% to the present 80-85%. We hope that, with production-sized batches, this proportion of good pins may be increased to over 90%.

MR. C. E. WEBER (Knolls Atomic Power Laboratory): Will you give me an idea of the approximate cost you figure for recycling pyrometallurgically versus chemical decontamination?

MR. SHUCK: We do not have sufficient information on production-batch lots to give an accurate cost figure at this time. Certainly the cost figures on our small experimental batches are currently much too high.

MR. E. P. HAMMOND (Los Alamos Scientific Laboratory): Have you given any thought to the type of blanket-processing scheme that will be used eventually?

MR. SHUCK: A number of processes for processing the blanket elements are under investigation or development by the Argonne National Laboratory Chemical Engineering Division. Because I have not been directly associated with these investigations, I am not prepared to report in any detail upon them. The liquid magnesium-liquid uranium extraction process in which plutonium is dissolved from the uranium phase into the magnesium phase, the magnesium poured off and distilled away leaving a plutonium concentrate, has been reported by Winsch and Burris.¹ The Pyrozinc Process has been under study for recovery of residue and blanket materials. This process has been reported by Feder.²

MR. J. DAVIDSON (ACF Industries): What do you plan to do with the 5% by-pass for makeup?

MR. SHUCK: Approximately 5% of the process flow leaves the cycle as residues and goes into slag and other non-melttable scraps. This material is rich in fission elements and will be stored for a sufficient cooling time that it can be recovered by conventional methods or by a pyrometallurgical method such as the Pyrozinc Process.

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Paper 16

DESIGN LIMITATIONS ON METALLIC URANIUM FUEL ELEMENTS FOR A GAS-COOLED REACTOR

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ABSTRACT

The use of natural uranium in a graphite moderated gas-cooled power reactor for operation at a gas-outlet temperature of 750 - 800 F imposes severe restrictions upon the fuel element design.

Experience with the irradiation behavior of uranium metal at useful surface temperatures has shown that at temperatures above 400 C (752 F), a phenomenon called "swelling" must be considered. This phenomenon is associated with the expansion of the fission gases in the weakened metal lattice and results in serious increases in volume of the uranium. The design of a fuel element must provide for this "swelling." Since the low multiplication factor possible in the natural uranium-graphite system dictates that the fuel be restricted to high density uranium, strengthening of the base metal by introducing alloying agents is limited to very low weight percentages of additives. The low multiplication factor also requires the "lumping" of the uranium in bodies of approximately 1 in. diameter. The practical results of these considerations are that a cylindrical metal rod of unalloyed or low alloy uranium metal must be considered as the basic fuel element type.

The low reactivity of the natural uranium system also restricts the choice of structural and canning materials which can be used in the reactor core. The only metals that can be present in any appreciable amounts in such a system (in addition to the uranium) are aluminum, magnesium, beryllium, and zirconium.

The bases for the design of fuel elements using natural uranium metal under gas-cooled power reactor conditions are discussed, and several possible fuel assembly designs are described.

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Kaiser Engineers and ACF Industries have been studying both natural uranium graphite moderated power reactors and enriched uranium graphite moderated power reactors. The purpose of this paper is to report on the design study of the fuel assembly for a natural uranium system, so there are included discussions of the fuel assembly materials, selection of the design bases for the application of these materials to specific fuel element designs, and descriptions of some of the fuel element designs which were developed.

The natural uranium graphite moderated gas-cooled reactor system imposes severe restrictions on the design of the fuel element. For example, the low infinite multiplication factor requires high uranium density and concentration, and only small alloy additions can be tolerated. Furthermore, the fuel must be in the shape of approximately one-inch diameter rods. For these reasons, the general fuel element type selected is that of a cylindrical metal rod. The low multiplication factor also restricts the choice of canning and structural materials since very few nuclear poisons can be tolerated in the reactor system.

Having selected uranium metal as the standard fuel material, radiation damage becomes one of the most important problems in the design of the fuel element. There appear to be three ways to design around this problem. One is to provide a void to accommodate the swelling which occurs. A second is to restrain the fuel and direct the swelling. A third is to strengthen the uranium by alloying, thereby reducing the swelling.

In addition to these restrictions on the fuel element design, the operating philosophy of the plant presents some problems. Specifically, the charging and discharging of the fuel is of interest. The reactor plant envisioned in this study has a machine for charging, axially rearranging, and discharging fuel while the reactor is operating at full power. A similar machine for servicing control rods, performing various core measurements and inspection functions, and for manipulating faulty fuel assemblies is also employed. The charge-discharge machine is located in a subreactor room, and the service machine is located overhead. If the decision were made to charge fuel from the top, the assemblies would have to be designed for satisfactory handling when in tension. However, in charging from the bottom, the assembly is designed for handling under compressive load. Operation of the charge-discharge machine with the reactor at full power permits considerable flexibility in programming the fuel so that approximately uniform burnup can be achieved.

Having selected the fuel material, the next step was to assess the radiation damage problem. At temperatures below 400 C the usual anisotropic dimensional changes are the chief problem. However, with temperatures above 400 C a different effect becomes evident. This is the swelling phenomenon which is isotropic. This swelling is presumably due to expansion of the fission gases within the metal structure. The only apparent means of preventing this damage is to make the uranium stronger by alloying. Unfortunately, no significant effect has been obtained with alloy additions of less than 5 to 10 weight per cent. Such relatively large alloy concentrations are not suitable for the natural uranium graphite moderated reactors.

The available data on the irradiation behavior of uranium metal and low additive alloys were assembled through discussion with fuel element personnel at the various United States reactor development sites. Discussions were also held with representatives of the United Kingdom. These data have been reduced to "R" values, the per cent density decrease per per cent atom burnup. Figure 1 summarizes

Fig. 1—Uranium "R" values vs irradiation temperature.

these data. The curve, A-B, represents the relationship used to establish the radiation damage design basis. It will be noted that the swelling becomes more pronounced above 400 C. Two observations of general interest made in the preparation of this curve are: (1) there are few data available on the irradiation behavior of uranium and low additive alloys for conditions of high surface temperature and high burnup, and (2) the bulk of the data available are U.K. measurements. The U.S. data which were available usually showed greater swelling for a given burnup and irradiation temperature, and this fact was weighted rather heavily in drawing the curve. Almost all of the high temperature U.K. data points fall well below the curve. Some additional problems are found when the fuel is operated above 400 C. For volume expansions greater than 15 per cent, the British report that fission gas is released from the metal structure. In addition, at high burnups a further volume increase occurs due to cracking of the metal.

Having surveyed the data which were available on the irradiation behavior of uranium, the design criteria were specified. The first choice was that of the material. Since the irradiation performance was not appreciably improved by very low alloy additions, unalloyed metallic uranium was selected. Having made this choice, it was necessary to decide how much volume increase could be tolerated. A volume increase of 10 per cent was specified in order to avoid the fission gas release problem.

Table I—FUEL SURFACE TEMPERATURE-EXPOSURE RELATIONSHIP

Uranium surface temperature		Exposure for 10% volume, Increase Mwd/metric ton
°C	°F	
400	752	4000
425	797	3000
450	842	2000
475	887	1500
500	932	1000

Having summarized the irradiation performance of the metal, a surface temperature-fuel exposure relationship was developed. This relationship of surface temperature to burnup for 10 per cent volume increase is given in Table I. At 400 C it is expected that an exposure of 4,000 megawatt days per metric ton can be achieved; at 425 C, 3,000 MWD/MT; at 450 C, 2,000 MWD/MT; at 475 C, 1,500 MWD/MT; and at 500 C, only 1,000 MWD/MT.

With this information, it was possible to establish temperature limitations for fuel element operation. A maximum surface temperature of 850 F was specified, since this would permit exposure to 2,000 megawatt days per metric ton. Anything less than that is of little interest for power reactor operation. The central temperature limitation is 1200 F; this was specified to avoid the uranium alpha-beta phase transformation.

With these limits established, the fuel geometry was next determined. It has already been indicated that a metal cylinder of uranium is required. For this element it was decided to use an annular cylinder. This was chosen for two reasons: first, to provide an internal void to accept the swelling which will occur; and secondly, to reduce the mean temperature in the piece and, therefore, the thermal stresses that it must endure. Only the outer surface of this element is cooled, however.

The diameter of the rod is determined from a physics optimization of rod diameter, coolant channel diameter, and lattice pitch. For the most feasible lattice pitch (the closest lattice pitch which can be tolerated), namely, 8 inches, a solid cylinder of 1.1 inch diameter is the optimum fuel rod diameter.

In order to maintain the physics characteristics approximately the same as in the solid cylinder, the annular piece was sized so that the same uranium volume per unit length was obtained as with a 1.1 inch diameter solid rod. The actual ratio of inner to outer diameter was selected as 0.323. This was based upon achieving a reasonable compromise between providing volume to accept the swelling and having sufficient reactivity.

Table II shows the effect of the inside to outside diameter ratio on the fuel element performance. The center column gives the ratio of the temperature difference between the center and the outer surface for an annular slug to that for a solid rod. The last column gives the ratio of the temperature difference between the volume mean temperature of the fuel and the fuel surface temperature in the annular slug to that in the solid rod. The basis for calculation of these values is equal power generation per unit length of fuel and equal fuel volume. For the design case slug, the ΔT is about 75 per cent of that in the solid piece, and the difference between the average uranium temperature and the surface temperature is about 83 per cent of that in the solid rod.

Table II—COMPARISON OF SOLID AND CORED SLUGS

[With equal fuel volume (cross-sectional area)
and equal total power generation]

$U_i = \frac{R_i}{R_o}$	$\Delta T_c / \Delta T_s$	$\frac{\bar{T}_c - T_o}{\bar{T}_s - T_o}$
0.000	1.00	1.00
0.323	0.74	0.83
0.400	0.65	0.75
0.500	0.54	0.61

Figure 2 shows the temperature profile along the central coolant channel in the core. It should be noted that the peak metal temperature occurs above the midplane of the reactor, approximately two-thirds of the way up the channel. Also note that the average surface temperature is about 750 F. By means of the charge-discharge machine, the assemblies can be rearranged within the channel so that the element which is initially positioned at the maximum surface temperature point does not remain for its entire life in the reactor in that position. This axial rearrangement of fuel permits an approach to higher burnups than would be predicted on the basis of exposure at maximum surface temperature for the entire irradiation period.

Having fixed the radii of the rod, it was necessary to specify a length for the element. Again, because of the charge-discharge machine and the desire to program the fuel, a short length was required. Also, a short length minimizes the structural requirements which must be placed on both the cladding and the fuel. Furthermore, a two-foot length could be processed at Hanford without special handling, while greater lengths would require some sort of sectioning operation prior to dissolution and feed

Fig. 2—Fuel element and coolant temperature profiles along coolant channel.

preparation. A two-foot fuel length greatly simplifies the design of the charge-discharge machine because the short element permits a shorter radius of curvature in the curved tubes which lead from the machine to the bottom of the core.

In most reactors the purpose of the cladding is to prevent reaction between the fuel and the coolant and to prevent escape of the fission gases into the coolant. Furthermore, in the gas-cooled reactor, because of the nature of the heat transfer properties of the coolant, it is necessary to provide extended surface on the fuel rods to promote heat transfer.

In selecting cladding materials then, the main criteria are neutron economy, because of the low multiplication factor, and strength. The British have proposed a useful technique for comparing materials for this application. The index they propose is the ratio of the absorption cross section to tensile strength. This ratio has been computed for candidate cladding materials and is presented in Table III. It is quickly seen that the only cladding materials which are of great interest are beryllium, aluminum, zirconium, and magnesium.

Table III—COMPARISON OF CANNING MATERIALS AT 400°C

Material	Melting point, °C	Thermal conductivity, cal/sec/cm ² /°C/cm	Macroscopic absorption cross-section, cm ⁻¹	Tensile strength, psi	$\frac{\Sigma a \times 10^6}{S}$
Be	1284	0.320	0.00123	32,000	0.038
Mg	650	0.380	0.00254	1,230	2.060
Magnox C	650	(0.320)	0.00261	2,240	1.070
Zr	1860	0.040	0.00765	16,800	0.456
Al	659	0.530	0.01300	1,950	6.670
18/8 SS	1500	0.036	0.22700	68,000	3.340
Fe	1528	0.180	0.20600	45,000	4.580
Cb	2415	0.130	0.06000	33,000	1.820

The gas-cooled reactor employs carbon dioxide coolant and, therefore, zirconium must be excluded because of corrosion considerations. Beryllium would be ideal, but at present it is not possible to fabricate extended surface shapes for fuel element cans. Aluminum reacts with metallic uranium to form inter-metallic compounds, so if it is used a diffusion barrier must be provided between the cladding and the fuel. This barrier would impose additional resistance to heat transfer. Aluminum also has a considerably higher neutron absorption cross section than the magnesium. Thus, by a process of elimination, magnesium was chosen as the cladding material.

In considering the various magnesium alloys available, the British Magnox alloys appear to be most suitable. These alloys have acceptable strength at the operating temperature desired, exhibit good ductility, and the beryllium addition to the magnesium raises the ignition temperature so that the safety characteristics of the reactor are improved.

The mechanical considerations in cladding design are considerably minimized by the use of a short length of fuel. Minimum structural requirements are placed upon the

cladding and the fuel, and, in the case of the cladding, only three problems appear to be significant.

The first problem is the compressive load on the bottom end cap of the can, since the fuel element is supported vertically in the reactor; the second is to provide sufficient strength in the cladding material to support the extended heat transfer surface, so that the latter is not displaced during its life in the reactor due to the flow of the coolant; the third is the necessity of ensuring that the can can plastically deform to accommodate swelling of the fuel. In addition, the prime emphasis on improving the heat transfer in this reactor requires that the extended surface shape be dictated primarily by the heat transfer considerations rather than manufacturing considerations.

Figure 3 shows a number of fuel element cans that were considered. The top can with longitudinal fins was designed for the standard size uranium slug. The second can from the top, also longitudinally finned, was designed to clad smaller diameter slugs. The third can, which is also for the smaller diameter slug, has helical fins which give somewhat better thermal performance than the longitudinal fins. At the bottom is an unfinned can which is discussed in a following section of this paper. The can design which is preferred for the full size slug is not shown, but is similar to the top can with the exception that the fins are helical rather than longitudinal.

It was important to minimize the amount of magnesium present in the can because of the low multiplication factor of the system.

It was then necessary to also consider the design of the fuel assembly support structure since the canned elements must be positioned within the vertical channels in the graphite moderator stack. For this purpose a sleeve arrangement was selected. The functions of the sleeve are to carry the load of the uranium, to center the fuel element within the coolant channel, and to provide a short, simple shape which can be readily handled by the charge-discharge machine. Furthermore, this arrangement was designed to permit easy disassembly prior to reprocessing. Another advantage of the sleeve concept is the fact that a number of different types of fuel elements can be used without modification of the graphite moderator stack. For example, either the longitudinally finned element or the helically finned element can be used within the same sleeve.

To actually support these elements within the sleeve, a top and a bottom spider are employed. Graphite was selected as the material for the sleeve, since it has good strength at operating temperatures when loaded compressively, it has low neutron absorption characteristics, and it is relatively cheap. Other possible materials would be beryllium and silicon carbide. The bottom spider is simply a three-armed arrangement which looks like an automobile steering wheel, although it does not have an outer rim; it will be fabricated from beryllium. The three spokes are fastened into the graphite sleeve near the bottom of the sleeve. With respect to the individual fuel elements, this is the chief load-bearing member of the fuel assembly. The graphite sleeve itself transmits the load of the assemblies above it in the channel to those below and in turn to the bottom latch mechanism which supports the whole channel load in the reactor. The top spider has no major structural requirement placed upon it and only positions the element in the center of the sleeve. For this reason graphite has been specified for its fabrication.

Having developed these design criteria, the next step was the development of specific fuel assembly designs. Figure 4 shows the preferred assembly. Actually, as was pointed out previously, a helically finned can was preferred to a can with longitudinal fins. The illustration shows the graphite sleeve, longitudinally finned can, and top

Fig. 3—Fuel element cans.

Fig. 4—Preferred fuel assembly.

Fig. 5—Three-element cluster fuel assembly.

Fig. 6—Finned graphite fuel assembly.

spider. The bottom spider is located at the extreme left and is fastened into three holes which pass through the thickness of the sleeve near its lower end. Good reactor thermal performance has been predicted for this assembly, and it appears to be reasonably inexpensive to fabricate.

Figure 5 shows a three-element cluster with the gently spiraled finned cans, and a more complex top spider. This design has been modified recently to employ an individual spider on each element. Again the bottom spider is attached to the bottom of the sleeve at the left. This configuration gives better heat transfer performance, but it is more expensive to fabricate and it also has slightly poorer nuclear characteristics.

Figure 6 shows a novel fuel element assembly which was developed in an attempt to apply the principle of restraint of the fuel to direct the swelling. This embodies an outer graphite sleeve and an inner graphite sleeve positioned concentrically within the outer sleeve by a number of longitudinal graphite webs which are an integral part of the inner sleeve. This fuel can does not have any extended surface and is therefore considerably less expensive to fabricate. The graphite webs function as the extended heat transfer surface. Again, good heat transfer data for the analysis of the performance of this design concept are not available, so the potential of this assembly will have to await further developmental work.

In summary then, the natural uranium graphite moderated gas-cooled reactor must be fueled with high density uranium, either as metallic uranium or low diluent alloys, and the fuel must be in the form of approximately one-inch diameter rods. With the design bases which have been established, and which were based on the available radiation performance data, and with the fuel programming scheme which is permitted by the charge-discharge mechanism, it is expected that an average exposure of approximately 3,000 megawatt days per metric ton of uranium can be obtained.

DISCUSSION

R. S. MILLER

MR. W. T. WARNER (USAEC, Fernald): You mentioned and showed in your temperature distribution chart an exterior and an interior cladding. You also mentioned a cored slug.

MR. J. L. KLEIN (Nuclear Metals, Inc.): The inner cladding temperature is essentially the same as that of the uranium surface temperature.

MR. WARNER: The inner surface of the cladding?

MR. KLEIN: You are just pouring the hot center out of the side.

MR. WARNER: There is then a cavity on the inside of your slug?

MR. MILLER: That is correct.

MR. WARNER: We saw a picture of the cluster as well as of the single element?

MR. MILLER: That is correct.

MR. WARNER: What is the diameter of that?

MR. MILLER: It is 0.375 in. in the case of the design slug. It has an OD of 1.6 in.

MR. WARNER: What is your method of plugging the end of the hole?

MR. MILLER: It is not plugged. There are end caps on the cladding can which are welded in.

MR. KLEIN: Will you draw a picture on the board?

MR. MILLER: We take a section through the element (see sketch). The uranium piece looks like this, and the cladding can is then like this. The end caps are welded in here -- they have positioning knobs which fit into the spider.

QUESTION: What is the ID of the cluster element?

MR. MILLER: The cluster element has an OD of 0.75 and an ID of 0.4 in. The cluster element has the same uranium volume in the three pieces as the preferred element does in the single piece.

MR. R. C. DALZELL (USAEC, Washington). Your numbers must be wrong. They are not 0.4 and 0.75. That would be awfully thin.

MR. MILLER: If you square the ID and the OD and take the difference, I think it comes out to 1.21 in.² which is the same as the 1.1-in.-dia solid rod. Remember, there are three of these.

MR. DALZELL: I know, but the 0.4 ID and the 0.75 OD leaves a wall of only 0.18. That is a really thin wall too.

MR. MILLER: I think those numbers are correct, aren't they?

MR. J. K. DAVIDSON (ACF Industries): 0.75 OD is correct, and I think that 0.4 is right for the ID -- a little over 3/8 in.

MR. J. H. KITTEL (Argonne National Laboratory): I wonder if you considered slugs with circumferential fins in your studies. These are very good for heat transfer.

MR. MILLER: Do you mean transverse fins?

MR. KITTEL: Well, circumferential fins around the slug rather than longitudinal.

MR. MILLER: As I pointed out, the preferred element has a helical fin. It approaches a transverse fin.

MR. D. P. HERRON (American Standard Atomic Energy Division): In the last element you showed, how was the transfer made from the smooth can to the inner graphite ring? Was that just a close fit?

MR. MILLER: Expansion at the operating temperature will provide an interference fit. We are depending upon that to give us the contact there.

MR. P. E. STEIN (Union Carbide Nuclear Company, Y-12 Plant): Will you use enriched UO_2 fuel gas cooled?

MR. MILLER: As I pointed out early, we have made studies of both enriched and natural uranium and this paper describes only the natural uranium element. We are in the process of preparing a report on enriched uranium fuel elements.

MR. B. W. DUNNINGTON (Battelle Memorial Institute): I am interested in your table of temperature vs burn-up. I realize that these numbers are something like you say you get by going around and talking to people. I wonder what your own feeling is now that you have talked to a lot of people about it. You chose a temperature there and referred to a burn-up. If you were 80% confident, say, of your remarks, what variation in the burn-up would you have to include to be 80% optimistic in thinking you are going to be right?

MR. MILLER: As I pointed out, this survey was made early in the study. There has not been very much additional information to come to our attention. I am not trying to dodge what the 80% confidence point would be. However, I believe that recently the British have reported that for temperatures between 450 and 600°C they can achieve 3,000 Mw days/metric ton burn-up with only 10% swelling. I think it is over 3,000 in fact. So that says that our relationship is fairly conservative.

Also as I pointed out, by rearranging the fuel, no fuel element stays at the maximum temperature point for its entire period of radiation.

Paper 17

A BULK UO_2 -STAINLESS STEEL FUEL ELEMENT FOR A GAS-COOLED REACTOR

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INTRODUCTION

Recently the Oak Ridge National Laboratory was requested by the Atomic Energy Commission to design a gas-cooled graphite-moderated power reactor prototype. Due to time limitations, the reactor was to be designed using existing technology in order that construction would be immediately initiated if desired.

The results of this design study have been published¹ and it is the intent of this paper to briefly review the basic reactor concept and to present the highlights of the Materials Section of the report dealing with fuel elements.

REACTOR SYSTEM

Figure 1 is the schematic diagram of a gas-cooled reactor circuit combined with a boiler and steam turbine for power production. The steam circuit consists of four steam generators operating at 950°F and 950 psi constructed in much the same manner as those found in conventional power plants. The reactor core consists of a helium-cooled graphite-moderator containing type 304 stainless steel fuel capsules. The capsules are loaded with 2% enriched uranium oxide slugs of 95% density. The helium coolant is circulated at 300 psi and enters the reactor core at 460°F where it is heated to 1000°F. In order to obtain the 1000°F exit gas temperature, the fuel capsule wall temperature is maintained at 1200°F.

The reactor is contained in a 3-in. thick pressure vessel fabricated from ASTM type SA 212 B steel. The pressure vessel is designed to operate at 300 psi and 460°F. A schematic of the pressure vessel and a table of the reactor parameters are presented in Fig. 2.

REACTOR FUEL

The Oak Ridge National Laboratory gas-cooled reactor will require ~ 131 tons of UO_2 fuel. The fuel will be enriched to ~ 2% U_{235} and will be in the form of bushings 0.750 OD x 0.320 ID x 0.500 high with a density of 95% - 1% of the theoretical density of UO_2 (10.95 g/cc).

The significant advantages of a process described by Curtis and Johnson² for producing UO_2 from ammonium diuranate were recognized and utilized by the ORNL Ceramics Laboratory.

PRESSURIZED WATER

LIQUID METAL FUELED

GAS COOLED — GAS TURBINE

GAS COOLED — STEAM TURBINE

Fig. 1—Several basic reactor systems.

PRESSURE VESSEL

FUEL ELEMENTS

Reactor core diameter (graphite moderated)	30 ft.
Active length of channel	20 ft.
Fuel element hole diameter	3.25 in.
Lattice pitch	8 in.
Fuel channels in graphite	1,597
Fuel elements per channel	6
Fuel elements per loading	9582
Fuel capsules per element	7
Fuel capsules per loading	67,100
Tubing length per fuel capsule	37-1/2 in.
Total tubing per loading	215,260 ft
Number of UO_2 slugs per fuel capsule *	75
Number of UO_2 slugs per loading	5.17×10^6
Total weight UO_2 per loading	approx. 171 tons

* 0.750 in. OD x 0.320 in. ID x 0.500 in. length

Fig. 2—Gas cooled reactor GCR-2.

The process described by Curtis and Johnson involves reducing the UF_6 to metal and then processing with HNO_3 and NH_4OH to produce ammonium diuranate. By modifying the above process the ORNL Ceramics Laboratory has developed a method for producing ammonium diuranate which is less costly, since no reduction to metal is involved. A simplified flow sheet is shown in Fig. 3, wherein UF_6 is dissolved in water and precipitated from the solution by the addition of aqueous ammonia. The precipitate is then filtered and redissolved in nitric acid. An extremely fine ADU is then precipitated and filtered from solution. The ADU cake is dried and calcined to U_3O_8 and reduced to UO_2 in a hydrogen atmosphere.

Figure 3 also shows the typical compacting procedure for an ADU oxide and for a conventional oxide. It will be noted that the proposed GCR-2 oxide requires fewer steps in fabrication, which should result in an over-all cost reduction for the fabricated fuel slug.

Figure 4 shows the two types of UO_2 derived from ammonium diuranate and fabricated by the same process. The GCR-2 oxide when compacted at 10 tsi had a density of 40% and could be sintered to a density of 93%, while the coarse-grained oxide when compacted at 10 tsi had a density of 58% and could only be sintered to a density of 67%. This points out the fact that it would be necessary to crush the coarse-grained oxide and compact at higher pressures to reach the desired density.

From the above description, it can be seen that the process developed by the ORNL Ceramics Laboratory is less complicated and more economical than some other UO_2 fabrication procedures. Additional economies are realized since no grinding is required to meet dimensional tolerances.

FUEL ELEMENT

The fuel element consists of seven stainless steel capsules with six located at the corners of a hexagon and one centrally located as shown in Fig. 5. The capsules are supported in tension by a hanger which in turn is supported by ledges machined in the graphite moderator. The method of assembly is shown in Fig. 6. Loading and unloading of the fuel element is accomplished by sliding the element down grooves machined in the sides of the graphite channels. The unit is secured in position by rotating the assembly to engage the hanger in the circumferential recesses.

The fuel capsules are fabricated from type 304 stainless steel 0.80 in. OD x 0.020 in. wall x 38.5 in. long. These are filled with 75 UO_2 slugs, 2% enriched, which are 0.750 in. OD x 0.320 in. ID x 0.500 in. high. The capsule is sealed by fusion welding hemispherical caps to both ends. After sealing, the capsule has an internal atmosphere of helium. The end caps have attachments for hanging and positioning the capsule. A magnesium oxide slug is placed in each end of the fuel capsule to provide thermal insulation and to distribute the weight of the UO_2 evenly over the bottom end cap.

Several designs of end caps may be considered. The hemispherical design is advantageous since inspection by radiography is simplified and since the caps could be fabricated by stamping from sheet material with the support pin attached by brazing.

The fuel capsule hangers could be fabricated by forging or casting or could be a brazed composite constructed from several stamped pieces. Figure 6 illustrates a hanger fabricated by the latter method.

A spacer is required at the bottom of the fuel element to maintain the hexagonal arrangement. Figure 6 also illustrates a spacer fabricated from

Fig. 3—A, proposed flow sheet for producing UO_2 for GCR-2 and fabrication process proposed for GCR-2 fuel; B, fabrication process for UO_2 of a different type than that selected for GCR-2.

Fig. 4—Two types of UO_2 derived from ammonium diuranate and fabricated by the same process. Each unit shown weighs 25 g.

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Fig. 6—Gas cooled reactor fuel element.

stamped pieces by brazing. A more economical method for producing parts of this type would be by casting or stamping.

While accumulating fuel element cost estimate data for presentation to private industry, it was felt that it would be worthwhile to fabricate several fuel elements with the facilities and materials at hand. A supply of 3/4 in. OD x 0.020 in. wall 304 stainless steel tubing was available and several fuel elements were fabricated as shown in Fig. 7. A cutaway shows simulated UO_2 bodies.

The inspection procedure developed for the above type of element is shown as a flow sheet in Fig. 8. The percentages shown on the flow sheet indicate the probable amount of material which will be rejected at each stage of the operation. These quantities are based on previous experience on other reactor projects. Although the values appear quite pessimistic, they are thought to be realistic for obtaining a product having a high degree of integrity.

Inspection techniques have recently been improved through the incorporation of ultrasonic and eddy-current inspection, and the rigid inspection required can probably be carried out at lower cost than is described in this report.

As mentioned previously, the intent of this study was to obtain a realistic picture of the fuel element cost. The data gained from previous reactor projects and from the fabrication of several sample fuel elements were submitted to 30 companies for preparation of cost estimates for supplying a reactor loading. Each company was also urged to submit their design alternatives in order to accumulate the best fabrication procedures in the limited time available. A response was received from 25 companies. Of these, 20 estimated the entire job and many submitted modified designs for the fuel capsules.

The estimates from all companies were analyzed and adjusted to current reactor parameters, since there were modest changes made in the interim. A composite of the costs is presented in Table 1. It was interesting to observe that at least 15 companies were within \$5.00 of these figures. Since several of the companies have considerable experience in fuel fabrication, it is felt that the estimate presents a realistic picture of costs for this type of element.

Table 1. FUEL CAPSULE COST ESTIMATE

Operation	Costs (\$)	
	Per Capsule	Per Reactor
Conversion of UF_6 to UO_2	19.01	1,275,600
Fabrication of UO_2 Slugs	17.44	1,170,200
Fabrication of Capsule	19.59	1,314,500
Hangers and Spacers	0.93	62,400
Transportation (300 mi. and Return)	0.16	10,700
Assembly of Element at Site	0.46	30,800
Development	3.06	205,300
Capital Investment for Production	2.15	144,200
Totals	\$ 62.80	4,214,000

Fig. 7—Fuel elements.

Fig. 8—GCR fuel element inspection flow sheet.

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DISCUSSION

P. PATRIARCA

MR. R. B. Gordon (Atomics International): Have you done any testing or worrying about fuel redistribution down the hollow center?

MR. PATRIARCA: Tests to determine the amount of spalling are being planned. If there is any abnormal spalling, we propose to put a barrier in the center, such as MgO or graphite.

MR. L. R. WEISSERT (Babcock and Wilcox Company): Do I understand correctly that you have a helium atmosphere inside those rods?

MR. PATRIARCA: There is a nominal pressure of one atmosphere of helium inside the tubes.

MR. WEISSERT: No pressure above atmosphere?

MR. PATRIARCA: No. However, as the temperature increases to the operating conditions the pressure increases accordingly.

MR. W. E. ROAKE (Hanford Atomic Products Operation): If I understood you correctly you said the steel cladding will collapse down about the UO_2 during operation without any wrinkling.

MR. PATRIARCA: Yes, our experiments verify that a 20-mil wall tube will collapse uniformly without wrinkling on the UO_2 bodies if the clearance between the tube and the UO_2 is 10 mils or less. If the clearance is any greater than this, you get wrinkling, which would naturally be objectionable. Perhaps we will end up pre-collapsing the fuel element at a higher temperature and some nominal pressure before we put them into service rather than to rely on the phenomenon happening by creep deformation in service.

MR. ROAKE: Does this manifest itself in wall thickening or tube extension?

MR. PATRIARCA: I cannot answer that as specimens for measurement have not yet been prepared.

Paper 18

IRRADIATION BEHAVIOR OF URANIUM-FISSIUM AND URANIUM-PLUTONIUM FISSIUM FAST REACTOR FUELS

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ABSTRACT

The fuel alloys to be used in EBR-II will contain uranium, plutonium, and a group of fission-product elements known as "fissium." Most of the irradiation studies to date have been made on uranium-fissium alloys, without plutonium additions. These alloys are similar in irradiation behavior to uranium-molybdenum alloys, in that if sufficient fissium is present, the desirable gamma phase can be retained so that excellent stability can be achieved. Uranium-plutonium-fissium alloys are also highly stable under irradiation, although there has been some indication that the plutonium alloys may be subject to a lower swelling temperature.

INTRODUCTION

In the pyrometallurgical reprocessing scheme developed at ANL for fuel elements discharged from fast-breeder reactors, studies have shown that a group of fission-product elements will not be completely removed and that these impurities will, therefore, tend to build up to an equilibrium amount after several successive recycles of the fuel. If the reprocessing procedure is not varied, the ratio of these elements among themselves is expected to remain fairly constant. However, the level to which they accumulate in the fuel can be controlled by varying the amount of burn-up and the efficiency of the process. The residual-fission products are referred to collectively as "fissium." The amount of fissium which accumulates in the fuel and the composition of the fissium will also depend upon whether U^{235} or Pu^{239} is used as fuel.

In order to feed to the EBR-II reprocessing plant spent fuel which is not constantly changing in composition in the early stages of reactor operation, it has been decided to fuel the reactor at the start with a synthetic fissium alloy. Specimens made from synthetic fissium alloys are being irradiated in NaK capsules, so that the irradiation behavior of various fissium alloys could be determined before actual reactor operation. Also, several full-length fissium-alloy fuel elements have been irradiated in loops.

RESULTS AND DISCUSSION

Work at ANL by the Physical Metallurgy Group has shown that fissium additions to uranium or uranium-plutonium alloys markedly alter the transforma-

tion kinetics. The fissium alloys are characterized by sluggish transformations, so that if enough fissium is present, the gamma phase is retained even with slow cooling. As might be expected, these characteristics strongly influence the irradiation behavior of the alloys. As will be shown, the performance of the alloys under irradiation is quite similar to the more familiar uranium-molybdenum alloys.

Figure 1 shows compositions of typical fissium alloys which have been irradiated. Alloys of both uranium-fissium and uranium-plutonium-fissium are shown. Because of the fact that the first loading of the reactor will be a uranium-fissium alloy rather than uranium-plutonium-fissium, and because at present our fabrication facilities for plutonium fuels are of limited size, most of the irradiation studies to date have been confined to straight uranium-fissium alloys.

Element	U ²³⁵ Fissium	Pu ²³⁹ Fissium
Uranium	95.0	69.2
Plutonium	-	20.0
Zirconium	0.2	0.5
Molybdenum	2.5	2.8
Ruthenium	1.5	4.3
Rhodium	0.3	0.7
Palladium	0.5	2.5

Fig. 1—Typical reference fissium alloys.

Uranium-Fissium Alloys

The first irradiations of uranium-fissium alloys represented various combinations of uranium, fissium, and extra additions of molybdenum.¹ Figure 2 shows typical irradiated specimens from a uranium-3.2 wt % fissium alloy. It is evident that furnace cooling from the gamma phase is much more effective in stabilizing the material than is a quenching treatment. In this respect the behavior of the alloy is very similar to that shown by uranium-rich zirconium and zirconium-niobium alloys.^{2,3}

Figure 3 shows typical irradiated specimens of uranium-2.5 wt % molybdenum alloy which was irradiated for purposes of comparison. The behavior of these specimens is very similar to that shown by the fissium alloy in Fig. 2. In other words, the alloy is most stable under irradiation when it has been allowed to fully transform from the gamma phase.

As can be seen in Fig. 4, the alloy is much less sensitive to heat treatment when the fissium content is raised to 5%. The uranium-5 wt % fissium composition shown in this figure is the alloy which is being most seriously considered for the first loading in EBR-II. The fuel will be in the as-cast condition, so the excellent performance of the as-cast specimens, is typified by the specimen shown at the extreme left, has given encouragement that the alloy will prove to be a satisfactory fuel material at least up to a temperature of ~ 500°C.

Fig. 2—Effect of irradiation on uranium-3.2 wt. % fissium alloy.

Fig. 3—Effect of 0.46 at. % burnup at approximately 500°C on uranium-2.5 wt. % molybdenum alloy.

	AS CAST	CAST (WQ 850°C)	CAST (FC 850°C)	WROUGHT (WQ 850°C)	WROUGHT (FC 850°C)

	a	b	c	d	e
BURNUP (%)	0.32	0.21	0.19	0.51	0.66
CENTRAL METAL TEMPERATURE (°C)	470	295	265	249	324

Fig. 4—Effect of irradiation of uranium-5 wt. % fissium alloys. (Mag. 2X)

Figures 5 and 6 show irradiated specimens of the same alloy but with the addition of an extra 2.5 wt % and 7.5 wt % molybdenum, respectively. The appearance of these specimens was completely unaffected by the exposure they received. The surfaces of the specimens retained their bright mirror-like appearance throughout the irradiation period.

Irradiations have also been made on full-length (14.22 in.) EBR-II fuel pins in an air-cooled loop.⁴ It was found that at a temperature of 570°C the uranium-5 wt % fissium alloy shows large volume increases, presumably because of the onset of swelling. Additional experiments now in progress at controlled temperatures are expected to provide information on the maximum permissible operating temperature of the uranium-fissium alloys.

Uranium-Plutonium-Fissium Alloys

As mentioned earlier, irradiations of plutonium alloys containing fissium have been more limited in scope. The first irradiations of these alloys were on cast uranium-20 wt % plutonium alloys with additions of 5 wt % Mo, 5.4 wt % Fs, and 10.8 wt % Fs.⁵ These alloys all showed excellent performance under irradiation, and were equal to the straight uranium-fissium alloys described earlier. However, the specimens were irradiated at relatively low temperatures, not exceeding about 300°C.

More recently, a group of cast specimens with the composition uranium-20 wt % plutonium-10 wt % fissium were irradiated at higher temperatures and to higher burn-ups.⁶ These specimens were all found to have swelled under irradiation. Figure 7 shows typical irradiated specimens of the plutonium alloys which have been studied thus far. It can be seen that high irradiation temperature is particularly damaging. The irradiated specimens in this group presented a difficult handling problem, as they were found to be highly pyrophoric in the cave-air atmosphere. Two of the specimens ignited spontaneously while they were being examined and porous material from another specimen suddenly exploded shortly after it was exposed to air. These incidents spread considerable fission product and alpha activity throughout the cave facility. However, no alpha activity was detected on the discharge side of the CWS filters in the cave ventilation system.

CONCLUSIONS

In conclusion, it can be stated that uranium-fissium and uranium-plutonium-fissium alloys are very stable under irradiation, particularly when in the as-cast condition. The alloys have definite temperature limitations, however, above which swelling will occur. Irradiation experiments now in progress are expected to furnish information as to just what these maximum permissible operating temperatures are.

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Fig. 5—Effect of 0.15 at. % burnup at approximately 300°C on uranium-5 wt. % fissium-2.5 wt. % molybdenum alloy.

Fig. 6—Effect of 0.13 at. % burnup at approximately 300°C on uranium-5 wt. % fissium-7.5 wt. % molybdenum alloy.

Fig. 7—Effect of irradiation on uranium-20 wt. % plutonium castings with molybdenum or fissium additions.

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DISCUSSION

J. H. KITTEL

MR. J. E. MINOR (Hanford Atomic Products Operation): Is the information on the effect of zirconium based on observations of transformation times or on irradiation results?

MR. KITTEL: No, it is based on x-ray studies, metallography, electrical resistivity, and all the other means that are available for studying whether the material is in the α or the γ phase. We have under irradiation now a group of specimens with high zirconium and low zirconium, and we hope that this will furnish a clear indication one way or the other as to which way we should go.

MR. J. L. KLEIN (Nuclear Metals, Inc.): Going back to your uranium-fissium, you show that water quenching of the low-fissium alloy gives poor resistance to radiation damage but presumably you have retained the γ .

MR. KITTEL: Not entirely. Some α will also be present because in the low-fissium alloys you cannot entirely retain γ . The low-fissium alloys behave like certain uranium-zirconium alloys where some mixed type of structure is produced on quenching which is neither simple α nor γ .

MR. KLEIN: What you are saying is that if you cannot retain the γ you are better off slowly cooling, but if by rapid cooling you can retain the γ then this is even more desirable.

MR. KITTEL: That is true. Another thing I wish to mention is that we are very interested in learning whether these alloys show the phase-reversal phenomenon that has been noted in uranium-molybdenum and uranium-niobium alloys, because if they do, this can be expected to have a marked effect on their irradiation behavior. In other words, they will retain γ during irradiation with less fissium than would be indicated from pre-irradiation studies of the equilibrium diagram and transformation kinetics.

MR. E. L. FOSTER (Battelle Memorial Institute): I wonder if you would like to comment on the metallographic structure of the as-cast alloys.

MR. KITTEL: The as-cast alloys are generally characterized by an equiaxed grain structure, particularly as the amount of fissium is increased.

MR. FOSTER: Does it exhibit coring problems?

MR. KITTEL: We have not been troubled with coring problems in these alloys when they are injection-cast. Arc-cast buttons, on the other hand, evidently pass through the liquidus more slowly and coring is often observed.

MR. E. F. LOSCO (Westinghouse Atomic Power Division): I would like to ask about the relatively high-molybdenum concentrations that were used in some of the

alloys. Is it necessary for the retaining of the γ phase? Secondly, what does this do in so far as physics go in affecting the resonance absorption?

MR. KITTEL: We think that the additions of molybdenum certainly caused more of the alloy to be in the γ phase during irradiation. Even 5% fissium alloy is not entirely γ , and it may be that the addition of the 2-1/2% molybdenum put the structure entirely into the γ phase. With regard to your second question, fast reactors are relatively insensitive to poisoning or absorption effects, so that extra molybdenum has a minor effect on the physics of such reactors.

MR. LOSCO: Does ruthenium act the same as molybdenum?

MR. KITTEL: Yes, it does.

MR. LOSCO: Is ruthenium more effective?

MR. KITTEL: Per cent for per cent, it appears to have about the same effect as molybdenum in retaining γ .

MR. H. PEARLMAN (Atoms International): Have you analyzed your irradiated specimens for retained sodium?

MR. KITTEL: Yes, we have considered the possibility that entrained sodium may have made the irradiated specimens pyrophoric. However, one specimen ignited two days after it had been first exposed to air. This particular specimen had been thoroughly cleaned. It had been left in water for a long period of time, and there was no indication that there was any sodium in the fuel alloy.

MR. PEARLMAN: We cleaned ours thoroughly also and yet when we dissolved them up and analyzed them we found sodium and potassium in them. Don't blame it all on the plutonium.

MR. KITTEL: No, we are not, because we have also had several uranium alloy specimens burn after exposure to air.

CHAIRMAN FOOTE: I can see one reason why Mr. Shuck maintained that he is going to put argon atmosphere in this reprocessing cycle.

MR. KITTEL: There have been some difficult cleanup problems resulting from spontaneous ignition of irradiated plutonium alloys, so that the use of inert gas atmospheres is quite desirable.

Paper 23

GAS-PRESSURE BONDING OF FUEL ASSEMBLIES

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ABSTRACT

A technique of bonding fuel assemblies by utilizing high gas pressure at elevated temperatures is being investigated for the preparation of Zircaloy-clad zirconium-uranium fuel assemblies. The components to be bonded are fabricated to final size and then assembled and sealed in an evacuated container. The sealed assembly is then subjected to a high gas pressure and temperature in a high-pressure cold-wall autoclave. Plastic flow brings the components into intimate contact, and bonding and cladding occur simultaneously to form an integral assembly. Excellent results have been obtained by bonding 5 hr at 1550 F with a helium pressure of 10,000 psi.

INTRODUCTION

During the past year, the use of a gas-pressure-bonding process has been investigated for the preparation of Zircaloy-clad zirconium-uranium flat-plate assemblies. The objective of this study has been the development of a process for preparing integral assemblies in one operation at a reduced cost.

The gas-pressure-bonding process is ideally suited to the bonding of integral assemblies. In this process, the components are fabricated to final size and inspected prior to cladding and bonding. The dimensional control achieved during the bonding operation is governed by the dimensional tolerances imposed on the individual components.

There is one major problem related to this process and to any other process which produces an integral assembly; it is the problem of evaluating the bonded assembly. Various nondestructive tests can be conducted on individual fuel plates; however, most of these tests cannot be feasibly applied to the evaluation of the fuel elements of an integral assembly. Previous investigations concerned with the development of nondestructive tests

for evaluating fuel elements of an integral assembly have shown that although it is a difficult problem nondestructive testing of assemblies is feasible.¹

DESCRIPTION OF PROCESS

The gas-pressure-bonding technique involves the simultaneous cladding and bonding of a complete assembly in one operation. The components to be bonded are enclosed in a metallic container which is evacuated and sealed. This is essentially the same technique that is employed for roll bonding of fuel elements. Since the pressure-bonding process is accomplished with a very small amount of deformation, the container is a low-cost item prepared from thin-gage stainless steel sheet.

The sealed evacuated assembly is subjected to a high external gas pressure at an elevated temperature in an autoclave. A uniform external pressure is transmitted through the container, exerting sufficient pressure to bring the components into intimate contact. The pressure and temperature are maintained for a sufficient length of time to achieve bonding at all interfaces of the components.

Since the only deformation is that required to fill void space and force the components into intimate contact, the components to be bonded are fabricated to the final desired dimensions prior to bonding. The final dimensional control that is achieved is governed by the dimensional tolerances of the individual components prior to assembly and bonding.

There are advantages in a process in which the components are fabricated to final size prior to cladding. All of the components can be fully inspected prior to bonding. Distortions occurring during conventional fabrication such as core and cladding thickness are avoided. There are also some obvious disadvantages, almost all of which relate to the machining and handling of longer length components.

Equipment

The autoclave equipment used for pressure bonding is unique, and for this reason it appears desirable to discuss it. The gas-pressure-bonding technique was initially used for the pressure bonding of pin-type Zircaloy-clad fuel elements. The assembled pins were loaded into a sealed high-pressure tube. This tube was pressurized with 2000 lb of helium gas and inserted into a muffle furnace. Excellent bonding was achieved; however, at this relatively low pressure, a bonding cycle of over 24 hr was required to deform the Zircaloy tubes and achieve intimate contact with the uranium-zirconium core.

From this initial study, it became obvious that a higher gas pressure was needed to decrease the time required to obtain bonding. A 2-in.-ID hot-wall autoclave was procured. However, the hot-wall autoclave has a

limited use, being restricted to a relatively low combination of temperature and pressure; the one obtained was limited to a maximum pressure of 5000 psi at a temperature of 1500 F. In order to operate at high pressures and temperatures, several cold-wall autoclaves were assembled.

The pressure vessels were procured from the Autoclave Engineering Company. A schematic of a cold-wall autoclave is shown in Figure 1. The vessel is sealed at both ends by use of a modified Bridgeman closure. The unit has openings for electrical and thermocouple leads, pressure inlet, vacuum outlet, pressure-control outlet, and rupture safety disk.

A resistance heater is incorporated into the center of the autoclave. The specimens to be pressure bonded are inserted in the center of this heater. Ceramic insulation is tightly packed around the sides, top, and bottom of the heater. The furnace is constructed of semicylindrical refractory units wound with helical coiled Chromel A wire. A stainless steel tube is positioned within the furnace liner and serves as a loading container.

Three cold-wall autoclaves are available. Two of these are 9-in.-ID autoclaves and the other is a 14-in.-ID unit. Specimens up to 5 by 5 by 48 in. can be pressure bonded in the large autoclave. All of the units have been designed for operation at 10,000-psi gas pressure with an autoclave wall temperature of 700 F. The units have been operated at temperatures up to 2000 F and pressures of 10,000 psi. Under these conditions, it was possible to operate with a maximum wall temperature of 300 F without the use of coolant in the wall liner.

Satisfactory temperature control is difficult to achieve in these autoclaves. Convective heat transfer increases rapidly with pressure, and at 10,000 psi, would be hundreds of thousands of times greater than at atmospheric pressure. Thus, it is necessary to prevent convective heat transfer paths to the autoclave wall. To do this, all of the void space between the autoclave wall and the heater must be tightly packed with a dense insulation. Excellent results have been obtained using Fibre Frax and Micro Quartz fiber for insulation.

Any inert gas is satisfactory, although helium has been used in this work. It possesses a high heat capacity and high thermal conductivity in comparison with the other inert gases. This is a disadvantage from a heat-loss standpoint; however, it is an advantage for attaining uniform temperatures and good heat transfer between the heater and the specimens.

The use of zone furnaces and the development of techniques for minimizing convective flow have resulted in relatively uniform heat zones over a length of 20 in.

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Fig. 1—Sectional view of a high-pressure cold-wall autoclave.

Preparation of Components for Pressure Bonding

The method employed for the surface finishing of the Zircaloy components has a significant influence on the resultant bond quality. This is to be expected as the only deformation that takes place during the bonding operation is the amount required to bring the components into intimate contact. Bonding studies have been made using Zircaloy surface finished by pickling, belt abrading, machining, shot peening, grit blasting, and cold rolling. After surface finishing, the Zircaloy components were degreased and washed with detergent prior to assembly for bonding. The best results have been obtained with materials whose surfaces have been finished by an abrading operation. The current technique involves the removal of from 2 to 3 mils of surface from cold rolled or annealed Zircaloy by a belt-abrading process. A 60-grit silicon carbide abrasive belt is used at a surface speed of 2500 fpm. The Zircaloy is fed under this rotating belt at a speed of approximately 4 fpm. This is a relatively low-cost operation, comparable in cost to pickling.

A sulfonated oil was selected as the coolant in the machine belt-abrading operation. This oil resulted in the least amount of overheating and embedding of grit of several different types of oil that were investigated. However, during one stage of the work, a continuous increase in bond-line contamination was observed. It was discovered that the removed zirconium oxide and surface metal did not separate by gravity in the viscous oil. As the oil became heavily saturated with this material, it was embedded in the Zircaloy during the abrading process. The installation of a centrifugal filter and periodic oil changes have minimized this surface contamination.

After belt abrading the Zircaloy and zirconium-uranium core components were degreased and cleaned in the usual manner as outlined below:

- (1) Scrubbed in cold alcohol
- (2) Degreased for 15 min in hot trichloroethylene
- (3) Washed in cold Alconox
- (4) Rinsed in cold water
- (5) Scrubbed in hot Alconox solution (180 F)
- (6) Rinsed in cold water
- (7) Rinsed in hot water (180 F)
- (8) Rinsed in cold water
- (9) Rinsed in hot water (200 F)
- (10) Dried with paper towels.

A typical bond that was obtained with surfaces pickled by an HNO_3 -HF solution is shown in Figure 2. It is apparent that no grain growth occurred across the original bond interface. Possibly, a film was formed during pickling which interferes with the diffusion process, though no evidence of contamination can be seen in the electron microscope. One might interpret the results of Gulbransen and Andrew during the hydriding of pickled zirconium² as resulting from a film. Use of other pickling solutions and a combination treatment consisting of acid pickling followed by a caustic rinse resulted in bonds possessing partial grain growth across the interface; however, it has not been possible to obtain consistent results. Many specimens possessing a bond with incomplete grain growth have been evaluated by destructive methods and tested in 680 F water. They apparently possess good corrosion resistance and relatively good bond strength. However, this type of bond is considered to be unsatisfactory as there is a potential plane of weakness.

A bond between as-cold-rolled Zircaloy without any subsequent surface preparation is shown in Figure 3. This bond also possessed incomplete grain growth across the interface.

A bond obtained between Zircaloy specimens surface finished by a belt-abrading process is shown in Figure 4. This is an excellent bond possessing complete grain growth across the interface. Frequently, the bond line can be detected by a line of contamination although there is complete grain growth across the interface. This contamination is believed to be introduced during the surface finishing or subsequent cleaning operations. Specimens with this type of bond have also been destructively evaluated. The bonds appear sound and cannot be separated. They also possess excellent corrosion resistance in 680 F pressurized water.

Ti-Namel, an enameling grade of steel, was used as spacer material. The Ti-Namel spacers were plated with chromium to prevent reaction with the Zircaloy that would result in diffusion of iron into the Zircaloy. The spacers were also degreased and cleaned in the same manner.

After the cleaning operation, the components were assembled in a container that was sealed and evacuated. The components for the pressure bonding of a flat-plate assembly were assembled in the manner shown in Figure 5. Zircaloy strips were used to form the picture frame of the individual fuel plates. The use of Zircaloy strips saves on both machining and material costs. The components used for the pressure bonding of a 2 by 2 by 6-in. assembly are shown in Figure 6.

Results of Pressure-Bonding Tests

The evacuated assemblies were placed within the heater in the high-pressure autoclave. The autoclave was sealed and temperature and gas pressure were applied simultaneously. The bonding cycle for a 2 by 2 by 6-in. assembly is shown in Figure 7.

Fig. 2—Zircaloy-to-zircaloy bonded specimen, cold rolled and pickled with HNO_3 -HF. (Pressure bonded at 1550°F at 10,000 psi for 4 hr.)

Fig. 3—Zircaloy-to-zircaloy bonded specimen, cold rolled. (Pressure bonded at 1550°F at 10,000 psi for 4 hr.)

Fig. 4—Zircaloy-to-zircaloy bonded specimen, cold rolled and belt abraded. (Pressure bonded at 1550°F at 10,000 psi for 4 hr.)

Fig. 5—Assembly of components for pressure bonding.

Fig. 6—Components of a zirconium-clad zirconium-uranium flat-plate assembly.

Fig. 7—Gas-pressure bonding cycle for a zircaloy-clad zirconium-uranium flat-plate assembly.

After pressure bonding, the container was stripped and the assembly was treated in nitric acid to dissolve the steel spacers. Sixty hours was required to dissolve the Ti-Namel spacers of a 2 by 2 by 20-in. assembly in a 50-50 nitric acid solution at 180 F. Small amounts of urea were added to decrease pickling time by decreasing the passivity of the steel.

Ultrasonic agitation was tried in an attempt to speed the pickling process, but no apparent improvement was obtained.

Chromium-plated copper and composite steel-copper spacers are also being investigated. Such spacers could be dissolved in a fraction of the time that is required to dissolve the Ti-Namel spacers.

A 2 by 2 by 6-in. bonded assembly, prepared from the components shown in Figure 6, is illustrated in Figure 8. The bonding cycle for this assembly is presented in Figure 7. The sides of the pressure-bonded assembly are relatively smooth and can probably be utilized without any subsequent machining. A cross section of this assembly is shown in Figure 9. The Zircaloy pins can be seen in the channels bonded to the adjoining fuel plates. These pins were used to demonstrate the flexibility of the process. It is feasible to incorporate channel supports of almost any shape. This will permit the use of thinner fuel plates, with struts for stiffening.

The assembly was corrosion tested in 680 F water and then sectioned to determine bond quality by metallography and destructive chisel and bend tests. The bonds were found to be comparable in strength and corrosion resistance to those obtained by conventional co-reduction fabrication operations. A typical Zircaloy-to-Zircaloy bond and Zircaloy-to-zirconium-22 w/o uranium bond is shown in Figures 10 and 11. Complete grain growth was obtained across the original Zircaloy interfaces.

A pressure-bonded 2 by 2 by 6-in. assembly incorporating 16 fuel plates is shown in Figure 12. This assembly was bonded 5 hr at 1550 F and 10,000-psi helium pressure. As-rolled components were used; consequently, several areas were observed with incomplete grain growth across the original interface. However, after a 28-day corrosion test in 680 F water, no bond-line attack was observed. Attempts to separate the specimen at the original Zircaloy-to-Zircaloy bond interface were unsuccessful. Excellent dimensional control was achieved in the pressure bonding of this assembly. The variations in channel thickness were within ± 0.001 in., the width of the channels increased 0.003 in., and the over-all assembly height varied ± 0.003 in. over the length.

Recently, 20-in.-long assemblies have been made. A 2 by 2 by 20-in. assembly is shown in Figures 13, 14, and 15. The components for this assembly were surface finished by a belt-abrading technique. Metallographic examination of sections taken from both ends of this assembly re-

Fig. 8—Pressure-bonded flat-plate assembly incorporating eight pin supports in each of the ten coolant channels. (Specimen pressure bonded at 10,000 psi for 5 hr at 1550°F. Zirconium-22 wt. % uranium core clad with Zircaloy-2.)

Fig. 9—Transverse section of pressure-bonded assembly. (Specimen pressure bonded at 10,000 psi for 5 hr at 1550°F.)

Fig. 10—Zircaloy-to-zircaloy bond from section of pressure-bonded assembly. (Specimen pressure bonded at 10,000 psi for 5 hr at 1550°F.)

Fig. 11—Typical section of a zircaloy-to-zirconium-22 wt. % uranium bond. (Specimen pressure bonded at 10,000 psi for 5 hr at 1550°F.)

Fig. 12—Transverse section of pressure-bonded 2- by 2- by 6-in. assembly with 16 fuel plates. (Specimen pressure bonded at 10,000 psi for 5 hr at 1550°F.)

Fig. 13—Flat-plate 2- by 2- by 20-in. zircaloy-clad zirconium-uranium assembly prepared by pressure bonding.

Fig. 14—Transverse section of pressure-bonded 2- by 2- by 20-in. assembly.

Fig. 15—Side view of pressure-bonded 2- by 2- by 20-in. assembly.

vealed that very good Zircaloy-to-Zircaloy bonds were obtained. Excellent corrosion resistance was observed in a brief 3-day 680 F water corrosion test.

This assembly was bonded with no external jiggling. It is slightly bowed; however, this is no indication that external jiggling will be required as the length of the assemblies is increased. Considerable difficulty was experienced in obtaining a uniform heat zone across the length of the assembly. Another 20-in. assembly pressure bonded with a relatively uniform heat zone did not exhibit any significant bow.

Advantages of the Gas-Pressure-Bonding Process

One of the most important advantages of the gas-pressure-bonding technique is the decrease in cost over the current techniques employed for fabricating assemblies. It is obvious that the simultaneous cladding and bonding of a complete subassembly in one operation eliminates many steps of the conventional fabrication processes which involve roll cladding of individual fuel plates followed by welding of a composite assembly. Another advantage is the over-all close dimensional control that can be achieved. The cores are not rippled and can be prepared with squared ends. It is also possible to extend the width of the cores to the full width of the channels.

Aside from these advantages, perhaps the greatest long-range advantage is related to the versatility of the process. It will be possible to prepare assemblies of a design that cannot be feasibly prepared by current processes. Some of the possibilities for combining a burnable poison into flat-plate assemblies are shown in Figure 16. The burnable poison in the form of a zirconium-boron alloy can be incorporated into the cladding, support members, side plates or cores of the assembly. It is also possible to vary the core-alloy composition across the width and along the length of the fuel elements by use of strip components. Hexagonal-shaped assemblies and assemblies with channels for control rods can also be prepared.

SUMMARY AND CONCLUSIONS

Flat-plate Zircaloy-clad zirconium-uranium assemblies can be prepared in a one-step bonding operation by the gas-pressure-bonding technique. Problems associated with the pressure-bonding equipment and methods of component preparation and assembly must be solved before full-scale assemblies can be prepared; however, the feasibility of the process has been completely demonstrated.

By proper surface conditioning of the cladding and core components prior to bonding, bonds possessing excellent mechanical and physical properties can be obtained. Since the components are fabricated to final size and inspected before bonding, excellent dimensional control is achieved.

In Cladding

In Side Plates

In Channel Support

In Channel Supports and Core Inserts

Zirconium-uranium core

Zirconium-boron burnable poison

Fig. 16—Feasible designs for combining zirconium-boron burnable poison in pressure-bonded flat-plate assemblies.

Assemblies can be prepared by the gas-pressure-bonding technique at a much lower cost than by conventional fabrication operations. The process is extremely versatile. Assemblies of various shapes incorporating burnable poisons and variations in fuel composition can be readily pressure bonded.

ACKNOWLEDGMENT

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DISCUSSION

S. J. PAPROCKI

MR. H. F. TURNBULL (Westinghouse Atomic Power Division): You got all of these fellows ready to go out and pressure bond, but you did not tell them the disadvantages. You cannot tell whether the plates are bonded in the center. You have to develop a way to guarantee the bonding as well as the integrity of the fuel.

MR. PAPROCKI: A nondestructive test capable of detecting small bond imperfections is not available. It should be possible to detect relatively large-bond imperfections by use of the potentiometric test developed at Battelle, or the test discussed by Mr. McGonnagle.

MR. E. GORDON (Olin Mathieson Chemical Corporation): You mentioned that they are considerably cheaper. Have you any idea of the order of magnitude?

MR. PAPROCKI: We have been running some cost studies with co-operation from Bettis. They are preliminary, and I do not want to give any actual dollar figures. The cost is ~ 50% of the conventional fabrication process currently in use.

MR. GORDON: One other question. What confidence do you have in being able to get the channel spacing just right?

MR. PAPROCKI: Do you mean the dimensions?

MR. GORDON: Yes.

MR. PAPROCKI: We feel confident that very close dimensional tolerances can be met. In the large 20-in. assembly, the slight bowing resulted from a relatively extensive thermal gradient between the two ends of the assembly. This was the first 20-in. assembly that was prepared, and it is evident that the internal heater must be redesigned. In all previous specimens bonded with a uniform heat zone, there was no distortion and excellent dimensional control was achieved.

MR. GORDON: Have you made multiple runs of these things?

MR. PAPROCKI: Yes, we have bonded a relatively large number of specimens. We have also bonded stainless steel, molybdenum, niobium, and aluminum specimens.

MR. F. C. ROBERTSHAW (G. E. Aircraft Nuclear Propulsion Department): You mentioned belt abrading using a sulfonated oil. I wonder what kind of cleaning you used following the belt abrading?

MR. PAPROCKI: Essentially, all of our components were cleaned by the usual technique used for Zircaloy - degreased with trichloroethylene and alcohol followed by the hot and cold-water rinses - the usual ritual that people go through in preparing Zircaloy components.

MR. W. L. MUDGE (Westinghouse Atomic Power Division): What temperatures have you used for this Zircaloy bonding.

MR. PAPROCKI: All of the Zircaloy bonding has been conducted at 1550°F. For other work, the flat-plate oxide element for PWR, we are investigating temperatures as low as 1350°F. We have obtained bonds as low as 1400°F. In the pressure bonding of subassemblies, we want to develop a process for preparing full-scale assemblies. After a process is established, it may then be desirable to investigate lower temperatures.

MR. S. SIEGEL (Metals and Controls, Inc.): Do you find that it is necessary to cold roll or to use cold-rolled components?

MR. PAPROCKI: No, at one time we believed that the annealed components resulted in better bonds than cold-rolled components; however, we used the same abrading technique for both. By use of a different process for the cold-rolled components, it is also possible to obtain satisfactory bonding.

MR. T. DREIER (Knolls Atomic Power Laboratory): It was pointed out that ultrasonics is not a reliable test for the strength of the bond; so that in cases of this kind to make sure that you have a good bond everywhere, you have to do a rather exhaustive amount of destructive testing. I wonder whether you have done that on a sufficient number to be sure. You showed that you got complete grain growth across the bond, but did you get it uniformly?

MR. PAPROCKI: Yes, in fact, these assemblies were sectioned, and there has been an extensive evaluation of the bonds. The problem is that you cannot tell how well an assembly is bonded without this examination. An extra half inch can be left on the ends of each assembly, and if this area is well bonded, assume that the center was good; however, this still would not guarantee that the bonds in the center are good.

CHAIRMAN KAUFMANN: I have been wondering whether the function of the abraded surface may not be that you get a thin cold-worked layer which recrystallizes and you have a much better chance of getting grain growth across the interface. I am under the impression that any abraded surface will have poor corrosion resistance and that it is attributed to impurities. Now you are arguing that the abraded surface is cleaner than any other surface.

MR. PAPROCKI: We have obtained the best bonds with belt-abraded material. Evidently, bonding is related to the surface roughness, as we have not been able to achieve good bonding with as-rolled material.

CHAIRMAN KAUFMANN: If you took one of these abraded surfaces after cleaning and exposed it to high-temperature water, would it corrode?

MR. PAPROCKI: I do not know.

QUESTION: I wonder if you have ever belt abraded and pickled?

MR. PAPROCKI: Yes, we have belt abraded and pickled and did not obtain a good bond. We have also taken a machined surface and cold rolled it 10%. This material also did not bond well.

MR. DREIER: Isn't it possible that abrading gives you intimacy of contact uniformly? If it is cold rolled then there may be parts of the surface that are not completely in contact simply because they get in the way of each other. I know that in our welding work, in the bonding that we have been working on at KAPL, the minute we obtain a completely frosted surface then diffusion bonding will occur without any pressure, but without that, we cannot achieve bonding.

MR. PAPROCKI: I think this is true. We have looked at all of the different surfaces under high magnification. We have found that a pickled and cold-rolled surface is relatively smooth, whereas a machined or abraded surface is rough. This, of course, depends on the machining operation. It may be possible to obtain a smooth surface by machining and experience trouble in obtaining good bonding.

Paper 24

MANUFACTURE OF THE FUEL PLATES AND FUEL SUBASSEMBLIES FOR THE ARGONNE LOW-POWER REACTOR

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Before I begin my discussion on the Argonne Low-Power Fuel Elements, I would like to give credit to David Walker of Argonne, who has been a colleague of mine for many years and who in a great measure is responsible for having done and done well most of the work on which I will report to you today.

The Argonne Low-Power Reactor, ALPR, is a light water-cooled and -moderated boiling reactor. It is rated at 3 Mw thermal output, a figure which is achieved by operating with a bulk coolant temperature of 216°C maximum at 300 psi gage. The water in the reactor is high-purity water of ~ 1 -megohm resistivity and the water purity is maintained by continuous re-purification through a mixed resin bed. The production of useful power and heat are the sole purposes of the reactor. Fissionable material is supplied by highly enriched uranium in an aluminum-17.5 wt % uranium-2.0 wt % nickel alloy. This material is clad on all sides with aluminum-1 wt % nickel alloy (X8001) cladding, metallurgically bonded to the core material and to itself at all interfaces.

The reactor core grid plate can accommodate up to 59 parallel plate-type fuel assemblies, each of which contains nine fuel plates spot welded in an MTR array to aluminum side plates. The fuel plates are spaced 0.310 in. apart. All of the exposed surfaces on the fuel assembly including the top and bottom end fittings are wrought aluminum-1 wt % nickel alloy, the same alloy as that used for cladding each of the individual fuel plates. Figure 1 gives an idea of the size of the fuel assembly. Not shown in Fig. 1 is an aluminum-0.4 wt % boron¹⁰ poison strip which will be attached to the outer face of one of the side plates of a number of fuel assemblies. These are 0.020-in. thick and are made by extruding (hot) a mixture of X8001 aluminum powder and high-purity elemental boron¹⁰ powder into an oversize plate (0.250) which is next rolled to size and sheared to dimension.

One of the requirements that the fuel element is required to meet is that it survive a service period of three years minimum without failure for any reason, a fairly rigorous requirement.

Figure 2 is a drawing of an ALPR fuel plate in finished form and gives the over-all and internal dimensions of the fuel. A total of 531 of these plates will be required for a full reactor-core loading. Each of them in addition to meeting the drawing requirements, must demonstrate adequate resistance to corrosion by high-purity water at 290°C for two weeks, and must also exhibit no detectable unbonded areas as indicated by examination for blisters after

Fig. 1—ALPR fuel element.

Fig. 2—ALPR fuel plate.

annealing at 550°F for one hour and by ultrasonic transmission inspection. The ultrasonic-transmission test is considered to be by far the more rigorous of the two bond tests mentioned.

Figure 3 is a photograph of the plate components. The outer dimensions of the two cladding plates and the picture frame are identical being 6-in. wide by 9-7/16-in. long. The cores and the matching rectangle in the picture frame are 3-5/16-in. wide by 6-7/8-in. long and are obtained by blanking both picture frames and cores on the same punch and die. Each of the components at this stage is four times the finished thickness, thus the cladding stock is 136-mils thick, and the picture frames and cores are 0.200 mils in thickness. Bonding of the plate components is not achieved entirely if at all by hot rolling, but is instead achieved by a process developed at ANL by Noland and Walker called silicon bonding. This is an extension of the puncture bonding method which is adapted to aluminum as follows: first, the articles to be joined are coated on one or both sides of the interface with elemental silicon powder. The coated faces are next brought into intimate contact with each other and are placed under pressure in press-mounted dies maintained at a temperature above the melting point of the aluminum silicon eutectic; in other words above 577°C. Pressures of the order of 1000 to 2000 psi are exerted on the work until it reaches the desired temperature which is generally in the neighborhood of 595 - 600°C. After being at temperature a minute or so, the work is cooled under pressure to about 550°C, the press is opened and the work removed. Figure 4 schematically illustrates the mechanism.

In Fig. 5 is shown a photomicrograph of a section which was deliberately underheated which tends to support the proposed mechanism.

After the work in the dies has exceeded the melting point of the eutectic formed by solid-phase processes, the alloy formed, being under pressure, is largely squeezed out of the joint. This is indicated by the photomicrograph next shown in Fig. 6 in which you will notice a nearly complete absence of second-phase silicon in the joint area.

Flow sheets showing the process steps (somewhat simplified) in the making of ALPR plates are shown in Figs. 7 and 8.

The manner in which the components are assembled after silicon coating the cladding pieces is shown in Fig. 9.

The hot pressing set-up is shown in Fig. 10. A 60-ton capacity tensile testing machine is used for this operation. The top and bottom dies are identical, both being stainless steel blocks heated by cartridge-type heaters. Each die is equipped with an automatic temperature control which is wired into a control system whose function is to control the process sequence automatically.

After the hot pressing operation the compacts are processed as indicated in the flow sheets. This consists of hot rolling to near size, cold finishing, blister annealing, radiographic-core location, shearing, sonic-bond inspection, etc.

The plates are fixed in fuel subassemblies in the same way as followed in manufacturing the fuel elements for the Borax-III and Borax-IV reactors, both of which have been described in the past.¹ For those who may be unfamiliar with the method employed, it involves first forming 3/8-in.-wide flanges at right angles to the plate surface at both long edges of each plate. After forming, both flanges are milled to reduce their thickness to the order of 55 mils, and the plates are assembled one at a time with side plates and are spot welded in place. Figures 11 and 12 illustrate this procedure.

To date we have made about 100 practice plates to develop the bonding procedure parameters. Normal uranium rather than enriched uranium was used

Fig. 3—Plate components.

Fig. 4—Mechanism: Aluminum oxide perforation in silicon bonding (Idealized).

Fig. 5—Silicon bonding: Unreacted silicon particles surrounded by rejected silicon in underheated section.

Fig. 6—Typical microstructure at core to clad interface.

Fig. 7—ALPR fuel plates: Preparation of compact components.

Fig. 8—Process steps.

Fig. 9—Assembly of components after silicon coating the cladding pieces.

Fig. 10—Hot pressing set-up.

Fig. 11—Schematic illustration of assembly method.

Fig. 12—ALPR: Fuel plate forming die.

in the cores of these. The last 70 of the practice plates were satisfactory in all respects. We are just beginning our production run using enriched cores, thus we have no production figures at present. We hope the production plates will be as good as the last 70 practice plates were.

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DISCUSSION

MR. W. C. THURBER (Oak Ridge National Laboratory): I have two questions with regard to your bonding process. First is the reason for using this silicon bonding practice that the oxide film on the 8001 alloy is more tenacious than with ordinary aluminum; and second, about how thick is this silicon layer that you apply to the shoulder plates?

MR. NOLAND: Well, I will answer the second question first. Minus 325 mesh powder has maximum particle size of about 1.7 mils; so that the layer is a monolayer of powder. Its thickness then will be of that order.

I don't believe I know the answer to your first question, which was whether I believe that the oxide film on the 8001 is more tenacious than the ordinary. I rather doubt that it is. I think what you are really seeking may be an explanation as to why we would use a process like this. I think the reason is that we were trying to provide a fuel plate for a reactor employing aluminum-clad fuel at temperatures higher than such fuel had ever been subjected before. We felt that to do this we had to have a process which at least in theory could produce a 100% bond. We thought this might be the method.

MR. L. D. SCHAFFER (Oak Ridge National Laboratory): How do you incorporate your burnable poison core into the fuel element?

MR. NOLAND: We incorporate it by manufacturing a material by a powder metallurgical technique. We mix first X 8001 powder and elemental boron powder. We put this in an aluminum can and direct extrude to a strip about 1/4-in. thick. This is cold finished to size (0.020 in.) with intermediate anneals in the process.

MR. SCHAFFER: Where do you put it?

MR. NOLAND: We tack weld a strip on one side plate of each fuel subassembly that requires it. It is not in the fuel. It is attached instead to the subassembly.

MR. H. PEARLMAN (Atomics International): Why is it that the boron isn't in the meat?

MR. NOLAND: The reason that the boron is not in the meat is not a physics reason. It is not in the meat because we felt that in the time we had in which to do this work we did not want to introduce a process variable of which we knew very little.

CHAIRMAN KAUFMANN: Why do you have nickel in the meat?

MR. NOLAND: We have nickel in the meat to improve its corrosion resistance.

MR. S. SIEGEL (Metals and Controls, Inc.): You recycle, I presume.

MR. NOLAND: Yes.

Paper 31

DESIGN OF OXIDE FUEL ELEMENTS

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ABSTRACT

It is assumed that at moderate temperature and burn-up the primary limitation to the use of oxide fuel is release of fission gas by diffusion. The methods of calculation developed by Lustman et al.,¹ are extended and are applied to several sample cases. Some special relationships among time, temperature, burn-up, and fission-gas release are pointed out.

INTRODUCTION

Uranium dioxide has been shown to have highly desirable properties as a nuclear fuel. The major initial application has been in the Pressurized Water Reactor² and the success here has led to actual or planned use in a majority of new power reactors. Design problems arising in new applications require improved understanding of the performance of the material as a fuel element.

BACKGROUND

Extensive background exists on the performance of experimental elements for the PWR.¹ The major problem in irradiation of UO_2 at moderate burn-up is the evolution of fission gas, presumably by diffusion, increasing as temperature increases. The thermal conductivity of UO_2 is low, so that low-power densities and thin sections must be used to prevent overheating and the evolution of large amounts of gas. One approach is to make the section in plate form and so thin that gas evolution is very small. This technique is being developed for possible use in the second loading of the PWR.² One advantage is the use of low-cost sheet for cladding material, rather than the relatively high-cost tubing currently used for rod-type elements.

An alternative approach is to use rod-type elements permitting moderate fission-gas release, and to accommodate the fission gas under pressure inside the clad. This method is being widely adopted, following the lead of the PWR element. The desirable goal in this case is to design the element so that a known and acceptable amount of fission gas is released from the fuel pellets. Once it is assumed that the fission gas is released by diffusion, it is possible to write valid mathematical relationships between the fission-gas release and the geometry, properties, and environment of the fuel and clad. Numerical solutions to these relationships will then provide design data that will permit greatly improved economy in the fuel cycle, by use of conditions that are closer to the limit of performance.

Unfortunately, a straightforward application of design principles does not now lead to an element of known performance because the important properties of the fuel are not known with sufficient precision and are probably subject to wide variation in current manufacturing processes. There is also the strong possibility that oxide elements made by different manufacturing processes or subjected to higher burn-up and higher temperature than the PWR elements will behave quite differently. An example of the disturbing surprises that may be in store in future applications of ceramic elements is given in a recent KAPL report.³ The UO_2 in cermet elements appeared to show swelling on irradiation above 9 at. % burn-up at temperatures somewhat above 1000°F. Although this burn-up is greatly in excess of that currently planned in thermal reactors, it is of the same order of magnitude of that planned for fast breeders. Also, the existence of bloating in high-burn-up oxide elements makes it necessary to exercise extreme caution when extrapolating the results of oxide elements at lower temperatures and burn-ups.

Despite these limitations, it is useful to carry out design analyses of the type initiated by the PWR workers. Such analyses aid in selecting optimum conditions and serve to indicate the most productive areas for future development. Although the design analyses have been developed on the basis of circular pins, the equations have been kept in a general form and can easily be adapted to odd geometrical shapes.

Both rod and plates can be designed to permit a moderate release of fission gas which can be contained under pressure inside the clad. This approach is more adaptable for pins than plates because of the much higher pressure which can be contained by the pins. For example, in one case a study of two comparable core designs indicated that the plates could contain gas under a pressure of 10 atm while the pins could contain gas under a pressure of 150 atm. The allowable pressure in either design can be increased to any desired value by modifications in design, but there is a limit beyond which such design modifications are uneconomical.

DISCUSSION

Consider a simple pin element with some void space above the oxide. An extensive treatment of the appropriate relationships was recently given by Lustman et al.¹ with specific application to PWR pins. Some similar material is covered here to clarify additional parametric analyses.

Release of Fission Product Gas

For moderate burn-ups, the production of fission gas is proportional to the burn-up times the fraction of fission products that are gaseous,

$$n = K_1 = Qt = K_2 B \quad (1)$$

(Note: all symbols used in text are defined in Appendix I)

It is convenient to have n expressed both in terms of Q and B because it will be seen later that the rate of release is variable with Q for constant B .

The actual amount of fission product gas released is equal to the total produced times a fraction dependent on diffusion rate, diffusion distance, and time. Fick's second law has been solved for the appropriate boundary conditions

by Frank et al.,¹ and by Booth.⁴ The solution considers the ceramic body to be a number of small spheres of perfect density, and with a radius chosen such that the effective diffusion path is equivalent to an actual fuel element with interconnected porosity. Apparently this criterion can be satisfied by choosing the effective radius such that the ratio of surface area to volume is equal to that measured experimentally. The solution is given for the practical case where release by diffusion is taking place at the same time that the concentration is being built up by fission. The fractional release is expressed as a function of time of irradiation t , the diffusion coefficient D , and the effective diffusion path c . If the term b is taken as D/c^2 , then the fraction released, F , is a simple function of bt , provided $\pi^2 bt \leq 1$, which is always the case in practical fuel elements:

$$F = 4\sqrt{\frac{bt}{\pi}} - 1.5 bt \quad (2)$$

For values of bt smaller than 10^{-3} , the second term of this expression can be dropped with a resulting error of about 2%. Table I gives a numerical evaluation of F over a wide range of values of bt .

Since the amount of fission gas in the fuel gradually increases as fission takes place, the total release represented by a given F also increases. Figure 1 shows the relationship between total amount produced, amount released, and time. It is clear that the amount released at a given burn-up will be greater for smaller burn-up rates because the time of diffusion is greater.

The total amount of gas released is

$$U = F n \text{ mols/cc in time } t \quad (3)$$

but

$$U = \left(4\sqrt{\frac{bt}{\pi}} \right) K_1 Q t \quad (4)$$

$$\text{or} \quad U \propto Q t^{3/2} D^{1/2} / c \quad (5)$$

$$U \propto B t^{1/2} D^{1/2} / c \quad (6)$$

The importance of the rate of burn-up on the maximum gas released at any total burn-up is emphasized because it is a factor that may be overlooked in extrapolating experimental results.

One of the most difficult to determine of these terms is the diffusion coefficient, D . A great deal of work has been carried out, primarily at Bettis, and the results give reasonable agreement between calculated and observed gas release. A plot of D vs $1/T$ is given in Fig. 2, using the data given in WAPD-183. The dotted lines show very approximately the probable error of the curve.

It is apparent that at the higher temperatures where the diffusion rate is increasing rapidly, the steep slope combined with the probable error may result in an error of estimate of nearly two orders of magnitude. More data on the diffusion rate should reduce this error, but it is unlikely that accurate enough data can be obtained to permit use of small factors of safety in design studies.

TABLE I. NUMERICAL EVALUATION OF bt VERSUS F

bt	F
10^{-19}	7.136×10^{-10}
10^{-18}	2.257×10^{-9}
10^{-17}	7.136×10^{-9}
10^{-16}	2.257×10^{-8}
10^{-15}	7.136×10^{-8}
10^{-14}	2.257×10^{-7}
10^{-13}	7.136×10^{-7}
10^{-12}	2.257×10^{-6}
10^{-11}	7.136×10^{-6}
10^{-10}	2.257×10^{-5}
10^{-9}	7.136×10^{-5}
10^{-8}	2.257×10^{-4}
10^{-7}	7.135×10^{-4}
10^{-6}	2.256×10^{-3}
10^{-5}	7.121×10^{-3}
10^{-4}	2.242×10^{-2}
10^{-3}	6.986×10^{-2}
10^{-2}	2.107×10^{-1}
10^{-1}	5.636×10^{-1}
10^0	9.333×10^{-1}
10^1	9.933×10^{-1}
10^2	9.993×10^{-1}
10^3	9.999×10^{-1}

Using the relationships described above, it is interesting to consider the case of a cylindrical element with a constant rate of heat production across the radius. A series of terms are plotted versus the radius in Fig. 3 for a particular case. These results show that release rates increase rapidly as the temperature rises above 1000°C , due to the sharp rate of increase of the diffusion coefficient. Small increases in power of the element may cause large increases in gas release, because of the exponential relationship between rate of burn-up and rate of gas release. Note that since the amount of gas released per incremental radius is proportionate to F/rdr , this term approaches zero as r approaches zero.

Fig. 1—Relationship between amount of fission gas produced, amount released, and time.

A specific arbitrary problem was solved to derive the curves in Fig. 3. The problem used is given below:

Surface temperature of oxide	500°C
Heat flux at surface	300,000 Btu/ft ² /hr
Uniform heat generation through pellet	
Coefficient of thermal conductivity (avg)	1.5 Btu/hr ft °F
Pellet dia	1/2 in.
Diffusion coefficient	See Fig. 3

$$D = 60 e^{-\frac{115,000}{RT}} \text{ for } T > 1000^{\circ}\text{C}$$

$$D = 3.2 \times 10^{-18} e^{-\frac{850}{RT}} \text{ for } T < 1000^{\circ}\text{C}$$

Effective radius, c	10 ⁻¹ cm
Time, t	100 days
Burn-up rate and burn-up calculated from surface heat flux and time.	

The expedient of a particular case was used to aid in visualizing the relationships derived in the general case. In the derivation given below, the curves in Fig. 3 are explained.

The temperature at any point on the radius is

$$T = T_s + \frac{Q}{4k} (a^2 - r^2) \quad (7)$$

if it is assumed that k , the coefficient of thermal conductivity, does not vary with temperature. Later improvements in this kind of study should incorporate k as a function of temperature. This leads to the usual parabolic relationship of T with r as shown in Fig. 3. The diffusion coefficient D has an exponential relationship with T as shown in Fig. 2. Use of the specific relationships listed in the problem above results in a sudden increase in the rate of increase of D above 1000°C. The result is relatively large values of D toward the center of the element.

Fig. 2—Plot of log D vs $1/T$ for UO_2 .

Recall that

$$F = 4 \sqrt{\frac{bt}{\pi}} - 1.5 bt \quad (8)$$

where $b = D/c^2$ so that the curve of F vs r has about the same shape as the curve for D vs r . Observe here the value of using a specific example, in this case with time fixed, to permit a graphic representation of an increasingly complex series of relationships. The total amount of fission gas released per unit length is obtained by integrating over the cross section. Thus

$$U_T = 2\pi \int_0^a F n \, r dr = 2\pi \int_0^a U \, r dr \quad (9)$$

In the simplified case (i.e., for fixed time), the shape of the $U(r)r$ function is given by $F \times r$ because of (9). This function is shown in Fig. 3. Graphical integration of the area under the curve gives the total gas released. However, in general design studies, it is desirable to retain t as a variable. The general expression for U_T , equation (9), is, therefore, expanded below.

For the values of F and n given in equations (1) and (2)

$$U_T = 2\pi K_1 \int_0^a Q t \left\{ \frac{4}{c} \sqrt{\frac{t D_0 \exp(-E_0/RT)}{\pi}} - \frac{1.5 t D_0 \exp(-E_0/RT)}{c^2} \right\} r dr \quad (10)$$

Fig. 3—Relationship between T, D, F, Fr, and the radius.

In a case such as that cited earlier, the diffusion coefficient may have discrete expressions in different temperature regions. Let

$$D_0 = D_1; E_0 = E_1 \quad \text{for } T > T^* \quad (11)$$

$$D_0 = D_2; E_0 = E_2 \quad \text{for } T < T^* \quad (12)$$

It is convenient to break the quadrature (10) into separate terms. Let d be the radius at which $T = T^*$. The temperature is expressed in equation (7). Then

$$U_T = H_1 \int_0^d J_1 r dr + H_2 \int_d^a J_2 r dr + H_3 \int_0^d J_3 r dr + H_4 \int_d^a J_4 r dr \quad (13)$$

where
$$H_1 = \frac{8}{c} \sqrt{\pi t D_1} Q t K_1 \quad (14)$$

$$J_1 = \exp \left\{ - \frac{E_1}{2R \left[T_s + \frac{Q}{4k} (a^2 - r^2) \right]} \right\} \quad (15)$$

The integration of equation (13) is expressed

$$U_T = H_1 I_1 + H_2 I_2 + H_3 I_3 + H_4 I_4 \quad (16)$$

where
$$I_1 = - \frac{2k}{Q} \left\{ - \frac{E_1}{T^* e^{2RT^*}} - T_c e^{-\frac{E_1}{2RT_c}} + \frac{E_1}{2R} \left(\log \frac{T_c}{T^*} + \sum \frac{(-1)^n \left(\frac{E_1}{2RT^*} \right)^n \left[1 - \left(\frac{T^*}{T_c} \right)^n \right]}{n \cdot n!} \right) \right\} \quad (17)$$

$H_2 I_2$ is similar to $H_1 I_1$ with the substitutions of T_s for T^* , T^* for T_c , E_2 for E_1 and D_2 for D_1 . (18)

I_3 and I_4 are similar to I_1 and I_2 respectively, with the substitution of R for $2R$. (19)

H_3 and H_4 are similar to H_1 and H_2 respectively, with the substitution of $3\pi t D_0 / c^2$ for $8\sqrt{\pi t D_0} / c$. (20)

This permits solution for total gas released in the general case, but it is too cumbersome for convenient use without machine aid.

However, several very important conclusions are derived from this solution. Notice first that all direct occurrence of size (a) has dropped out. This is equivalent to the conclusion used by Lewis,⁵ where the gas releasing property of oxide elements is assumed to be constant for constant power/unit length (P). Note that

$$P = Q\pi a^2 \quad (21)$$

and

$$T = T_s + \frac{Q}{4k} (a^2 - r^2) \quad (22)$$

so

$$\Delta T \propto P \quad (23)$$

and it is known from the earlier statements that if other factors are constant,

$$\Delta T \propto U \quad (24)$$

On the basis of considerations of this sort, it is possible to make estimates of the allowable burn-up of new elements if the power/unit length is the same as other elements already tested, and this is done by Lewis.⁵ The caution mentioned earlier is repeated here; this relationship is only approximate because for constant P, Q may vary as a varies [see equation (21)], and, therefore, for constant burn-up, the time also varies ($B = \frac{K_1}{K_2} Qt$), and, therefore, F and U vary.

The conclusion is that calculation of the amount of gas released from an element requires use not only of P and B, but also of at least one of the terms Q, a, or t. It is desired to arrive at a graphical expression of the design and environmental parameters that uniquely define the amount of gas released. Curves of this sort have not yet been prepared. Lines of constant B/F(r) could be plotted on a grid of P versus U_T . From plots of this sort, it will then be possible to equate U_T with the maximum gas retention of a particular design, as discussed later.

A simplified solution to these equations can be obtained by the use of graphical integration procedures. In this manner, approximate results can be obtained with a minimum of calculation, and machine aid is not necessary.

If an effective diffusion path c is estimated and the time t is known, the data given in Table I and Fig. 2 can be used to obtain an equation of the form

$$\log F = \frac{A}{T} + B \quad (25)$$

Since there are two equations for the diffusion coefficient, there must also be two expressions as listed above for the temperature regions $T > 1000^\circ\text{C}$ and $T < 1000^\circ\text{C}$. F as a function of temperature is shown as Fig. 4 for a time of one year and an effective diffusion path of 0.1 cm.

The temperature at any distance from the center of the fuel element can be calculated by standard heat-transfer equations for the particular geometrical shape being used. The temperature is then related to a particular parameter in such a manner as to give the effective area over which heat is being transferred at that temperature. For example, in a circular pin the radius is used with the result that $T_r = T_o - Kr^2$ and the effective area/unit length is $2\pi r$. The function F is obtained for temperature T_r from Fig. 4 and as shown in Fig. 3 when $F \cdot 2\pi r$ is plotted vs r , the area under the curve is the total fraction of gas released. About 0.2 mols of gas are produced for each mol of U^{235} fissioned. Consequently, the internal gas pressure can be readily computed for any burn-up and enrichment.

The following example will serve to show how the method can be applied to flat plates:

Core width	4.126 in.
Core thickness	0.062 in.
Void space	30 vol %
Temperature at center of core	1766 K
Temperature at surface of core	811 K
Core length	20 in.
Void length	7.2 in.
UO_2 assumed to be 95% dense	
U^{235} to U^{238} ratio	0.312

Since the heat flow will be identical in each quarter section of the cross section, the heat-transfer equation is calculated for one quarter section. On the horizontal centerline axis $T_x = T_o - K_x X$ and on the vertical centerline axis $T_y = T_o - K_y Y$. In these equations K_x and K_y include a small but constant area dA . Along the diagonal which has the equation $y = 0.015X$, the temperature may be obtained by first integrating along the Y axis and then along the X axis which results in the equation $T_D = T_o - 0.03K_y X + 7.28 \times 10^{-3} K_y X^2$. The shape of the isothermal lines connecting the points on the x and y axes with the diagonal change is shown in Fig. 5. The area/unit length bounded by these lines is equal to $0.06x - 0.022x^2$ or for the entire cross section $0.24x - 0.088x^2$ with the x value being used to locate any point on the diagonal at T_D . The value of F is obtained from Fig. 4 for T_D and $F(0.24x - 0.088x^2)$ is plotted vs x to give the curve shown in Fig. 6. These results are approximate since the actual shape of the isothermal area is a curved surface, and since the method of obtaining the isothermal lines is also approximate.

Graphical integration yields a total of 2.14×10^{-4} mols which creates a pressure of 0.421 atm at the surface temperature on the basis of 100% burn-up with an enrichment of 31.2%.

Containment of Fission Product Gas

The gas that is released causes the pressure of the gas inside the fuel element to increase. If the original amount of gas is small compared to gas released, then the pressure p inside the element is

$$p = \bar{U} \frac{V}{V} RT \quad (26)$$

Fig. 4—Effect of temperature on fraction of gas released in a one-year period. (Effective diffusion path, 0.1 cm.)

Fig. 5—Section of flat plate showing approximate temperature gradations across core.

Fig. 6—Total fraction of gas released by flat plate element.

where v = vol of fuel/element
 V = vol of void/element

so $\frac{V}{v}$ = vol of void/unit vol of fuel = H , which will be called the "void coefficient."

The maximum internal pressure that can be tolerated in a pin-type element can be calculated from the strength and geometry of the cladding. If it is assumed that the hoop stress, σ , is the criterion for failure, then the maximum pressure p_1 is found from

$$\sigma = \frac{pa}{f} \quad (27)$$

where f is clad thickness, or

$$p_1 = \sigma f/a = S, \text{ "strength coefficient"} \quad (28)$$

Maximum allowable release occurs when (27) = (28), or

$$\bar{U} = \frac{SH}{RT} \quad (29)$$

The allowable $\bar{U}$ is thereby simply expressed in terms of the pin design. If H is expressed in terms of void coefficient/unit length of fuel element, then U_T can be substituted for $\bar{U}$.

CONCLUSIONS

It is assumed, following the work of Lustman et al., that the maximum burn-up of oxide fuel elements is limited by the release of fission product gas by diffusion, increasing the pressure inside the cladding. When the internal pressure equals the maximum safe pressure of the cladding, the element must be discharged.

Some relationships are developed to facilitate new design applications. These relationships emphasize the need for the most accurate possible information regarding the properties of the oxide. Normal variation within a given manufacturing process and introduction of new manufacturing processes can be expected to cause important variations in the properties of the oxide.

New low-cost testing procedures are needed to check critical properties such as thermal conductivity, surface area-to-volume ratio, and diffusion constants. If such tests can be developed, a higher quality oxide can be produced by directly controlling the manufacturing process to yield oxide of optimum properties.

Assuming that the design methods available are valid, it is possible to select the optimum element design. However, there are uncertainties in the performance of the element associated with uncertainties in the properties of the oxide.

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APPENDIX I

List of Notations Used in Report

n = mols of fission product gas/cc of fuel
Q = power density in w/cc of fuel
B = burn-up in % of metal atoms in the oxide
 K_1, K_2 = constants containing appropriate conversion factors and fission yield of gas
t = time
D = diffusion coefficient
c = effective diffusion path
b = D/c^2
F = fraction of fission product gas released
T = temperature, °K
 U_T = total fission gas released, per unit length of a cylindrical element
U = total fission gas released, per unit vol of fuel
 T_s = temperature at the surface
r = variable distance on the radius
a = radius of the fuel pin
k = coefficient of thermal conductivity
 T^* = 1000°C (or any other temperature at which the rate of increase of log D with $1/T$ changes)
p = power generation/unit length of fuel element
 T_c = temperature at center

DISCUSSION

B. W. DUNNINGTON

MR. A. R. KAUFMANN (Nuclear Metals, Inc.): I get the impression that the diffusion distance you use is not the physical dimension of the thing but some much smaller distance. Now, what is this distance?

MR. DUNNINGTON: I made the same assumption that the WAPD people did. In their analysis the problem is treated as a series of spheres. The radius of this little imaginary sphere is what I referred to as the diffusion distance. It is chosen so that the surface area-to-volume ratio is equal to the same surface area-to-volume ratio that you can physically measure on the solid oxide fuel. This is the treatment that WAPD used and it worked in applying the data. In other words, if you have a fuel of some moderate density, 90 to 95%, it has interconnected porosity, and you can represent this increased surface area by a series of spheres.

MR. KAUFMANN: But then you further assume that the gas that is liberated within the bulk of the material actually finds its way out.

MR. DUNNINGTON: This is a logical corollary of the initial assumption that these porosities are interconnected and therefore, that you can accurately represent the problem by a series of imaginary spheres. It is one of possible limitations, but this is the analysis that WAPD has made and it influences all of our thinking. If we go to higher density oxide, 99%, we may get a completely different physical problem with no amenable solution.

MR. H. JAFFE (Aerojet-General Nucleonics): Have you tried to introduce into your analysis the decay properties of the fission gases, so that you crank out at the back end the curies that are coming out rather than the cubic centimeters of gas?

MR. DUNNINGTON: Yes.

MR. JAFFE: Is this part of the report that covers this or the report that will eventually come out?

MR. DUNNINGTON: As currently planned our report does not include this material. A primary solution published in the WAPD report treats that problem in a simplified form, simply because they ran a fuel element, they counted the radioactivity of the fission gas and therefore, they needed the solution to get the answer; the kind of answer that I got is arrived at by taking the limit of that case.

However, we are engaged in a much more general solution than this type, and again we are attempting to reduce that solution to a series of numerical tables that can be immediately available to the designer. This we certainly will be able to do, but it is a matter of how simple we can make it.

The problem with regard to the latter part of the question is how many curies of activity come out. This gets involved with some details of questions that are too complex to deal with analytically at this time and we need more experimental data.

Mr. W. V. JOHNSTON (Knolls Atomic Power Laboratory): I am very pleased that you paid quite a bit of attention to the assumptions that we are using here, because I think that if you will go back to the data that you are basing them on, you will find that the burn-ups involved here are extremely small and the burn-ups are not included in this. Also some of your assumptions as regards distance of diffusion I think are open to discussion, let us say, in view of the slides that Mr. Weber showed where you can see the porosity on a much finer scale

than the particle size on your interconnected porosity. Further, as regards the high density of the UO_2 that we have been talking about here (95% or better) the claim is made that the porosity is not interconnected but is internal. So for your UO_2 you don't have your interconnected pores to get it out, and I feel that your gas diffuses a distance, only a μ or so, and ends up in one of these pores, and the distance that any fraction of gas can ever diffuse is just the short distance between the bubble closest to the surface and out. I should like your comment on this.

But one last remark, apropos of this. Some post-irradiation heating was done by Dr. Wisney at KAPL, that is, UO_2 powders of various burn-ups were simply heated up rather quickly, and the fission gas was observed to come out in bursts, more like rupture of these bubbles as you go up in temperature, and not the nice square root type of diffusion you are forced to assume in your mathematical work.

MR. DUNNINGTON: I share your concern that work of this sort be applied with caution. It should be applied by engineers who in every application will consider exactly the sort of thing that you are talking about. For example, I am very pleased to hear people call attention to the fact that UO_2 will swell just like metal. However, my feelings are that if one deals with the low burn-ups and the moderate temperatures that are experienced in water-cooled thermal reactors, the restrictions on the basic assumptions that are behind this kind of analysis are not too great. That is, there are areas of application. If we had not felt that there were areas of application, it would not have been worthwhile to perform these labors. Time will tell.

MR. JOHNSTON: A last comment. Could I put in the plug that later on this afternoon I will show some pictures of bubbles on low-temperature irradiation? You simply have to go to a much higher magnification to find them. They seem to be there.

Paper 32

IRRADIATION TESTING OF NOVEL URANIUM DIOXIDE FUEL ELEMENTS

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During the past eighteen months the Hanford Laboratories has been developing uranium oxide fuels for evaluation in the Plutonium Recycle Test Reactor.

The Plutonium Recycle study at Hanford is "designed to demonstrate the self-sustaining plutonium fuel cycle and to optimize the economic aspects of the plutonium recycle technology." Versatile loadings of uranium-bearing fuel elements are being developed for use in the test reactor, with emphasis on technology directly applicable to the plutonium development work being carried out in another component of Hanford Laboratories. New types of fuel elements have been conceived at Hanford, and elsewhere, which will be applicable to plutonium-enriched ceramics. Irradiation testing has followed development of fabrication techniques.

This paper discusses irradiation testing of four types of fuel elements: internally-and-externally cooled, sintered uranium dioxide tubes; swage compacted oxide powders; isostatic-formed fuel elements; and low-density, non-sintered uranium dioxide.

Cladding costs can be reduced by increasing the core-to-clad weight ratio. This can be accomplished by increasing the diameter of a cylindrical fuel element. However, unless the heat generation per unit length is kept constant, a core temperature limitation is soon encountered. The next obvious extrapolation is to remove heat from the center by making the core an internally-and-externally cooled tube. Theoretically, it can be argued that the hotter fuel core will expand away from the unbonded inner cladding, thus losing all benefit of internal cooling. However, it is difficult to theorize about the cracking and the high temperature plasticity of sintered uranium dioxide. The two tubular elements shown in the Fig. 1 were irradiated to determine quickly the extent of the problem. They are Zircaloy-clad, and are 1.44-in. OD, 0.395-in. ID, and 4.4-in. long. They contain uranium dioxide sintered to 92% theoretical density and ground to fit the Zircaloy cans, with radial clearances of 0.002 in. They were irradiated in the GEH-4 facility in the MTR. This facility is a non-recirculating, pressurized, 40°C water loop. The elements were exposed for one cycle, generating heat at a maximum rate of 50 kw/ft, which is 24 W/g. The calculated maximum core temperature was 1123°C. The maximum heat flux in the hole was 550,000 Btu/hr ft². One of these elements is being re-irradiated, whenever convenient, to accumulate exposure. The other was sectioned after an exposure of 145 Mwd/T. Figure 2 shows a cross section of this element after irradiation. This picture reveals

Fig. 1—Zircaloy clad, internally and externally cooled UO_2 fuel elements.

Fig. 2—Post irradiation view of interior of tubular UO_2 test element.

the radial cracking which has been found characteristic of this fuel configuration. A roughly cylindrical crack occurs at about the calculated adiabatic surface. This irregular, cylindrical crack extends the full length of the element. There is no evidence of any gross expansion of the core away from the internal cladding. We must assume that any gap which was formed was effectively relocated to the hotter interior by the cracking which occurred.

W. S. Figg (HAPO) devised a method of successive approximations to calculate the maximum core temperature of the tubular fuel element. In essence, several adiabatic radii are assumed and the heat flow is calculated in both directions from the adiabatic surface. The correct radius is, of course, the one which gives the same core temperature, calculated from both directions. To test these methods of core temperature calculation a thermocouple was inserted along the adiabatic surface of a tubular element. Two more thermocouples were installed in the irradiation facility for the measurement of the heat generation. The expected core temperatures for the experiment were between 850 and 950°C. The measured values varied between 850 and 950°C. However, the later measurements disclosed that an unexpectedly low thermal neutron flux had existed in our position during that cycle and that the power generation had been only 32 kw/ft, 65% of that expected in the experiment. This meant that a low interface conductance probably existed due to the lack of sufficient expansion of the oxide to pressurize the core-clad interface. To account for the measured temperature it was necessary to assume an interface conductance of approximately 500 Btu/hr ft² °F. This is not unreasonable for an interface between sintered UO₂ and anodized aluminum. Ex-reactor interface conductance measurements are to be made and the thermocoupled test element measurements are scheduled for repetition.

Other elements of this tubular geometry have been irradiated, clad in the M-388 1% nickle-aluminum alloy. Because of its lower structural strength, this material does not resist distortion as well as Zircaloy. In one case a slight bowing of an 8-in.-long fuel element was observed. Presumably, this was caused by the high flux gradient across the GEH-4 position in the MTR which causes roughly 40% greater power generation to occur on one side of the test element than on the other.

Because these irradiation tests provided the desired answers, and because fabrication studies have gone well, the development of the nested tubular fuel element for the Plutonium Recycle Test Reactor has proceeded. This fuel element is shown in exploded and cutaway views in Fig. 3.

This element, 8 ft 4 in. long, consists of two concentric Zircaloy-clad uranium oxide tubes and an axial rod. The element is 3 in. over-all OD and is designed to fit into a 3-1/4-in. ID, Zircaloy, vertical pressure tube. This element has several advantages when compared with an earlier proposed bundle of 19 half-inch diameter rods. These include 28% more uranium per unit length, a higher fuel to cladding ratio, and more uniform coolant channels. Physics tests have shown that this element has a greater reactivity, a smaller reactivity increase on loss of coolant, a 2% greater thermal utilization, and a 3-4% greater conversion ratio.

Possible disadvantages may include expansion of the fuel away from the large diameter inner cladding. The long, uncompartmented fuel zone may be subject to ratcheting of the fractured UO₂, and to disastrous expansion of the cladding in the event of failure and water-logging. To investigate this latter possibility, a Zircaloy-clad uranium oxide test piece similar to the ones already illustrated was purposely defected by drilling a 0.013-in. hole through the midpoint of the cladding. This test element was thoroughly waterlogged and subjected to rapid power increase to a maximum of 27 kw/ft. The calculated maximum core temperature was 1000°C. The expected release of fission gases to the water coolant occurred, of course, but there were no measurable dimensional changes in the element. The element originally contained nearly 1 cc of water.

Fig. 3—PRTR nested tubular fuel element.

Two-foot long test assemblies are being made for irradiation in a horizontal tube. Study of the possibility of ratcheting of the thermally fractured oxide in a vertically oriented fuel element must await completion of the HAPO-ETR test facilities.

Another approach to fuel element simplification and versatility is in the use of non-sintered uranium dioxide. It is true that by comparison with sintered UO_2 , there is a loss in uranium density, a lower thermal conductivity, and poorer fission gas retention. However, there are economic advantages in the fewer manufacturing steps which in some cases combine powder compaction and cladding into one operation. Such elements are easily adaptable to plutonium enriched fuel materials and have the added major advantage that the plutonium compound can be blended at the last moment before closure and so reduce many-fold the quantity of fuel material which must be manipulated in total enclosures.

The first application of non-sintered UO_2 was in the swage compacted fuel rods which were discussed at the Oak Ridge Ceramic Information Meeting in October, 1956. In this method of fabrication, loose powder is loaded into a cladding tube and compacted by reducing the over-all diameter by rotary swaging. There are no length limitations and reproducible UO_2 densities can be easily achieved. Depending on the nature of the oxide and of the cladding material, UO_2 densities 80 to 90% of theoretical are easily obtained by cold swaging; 85 to 94% is obtained by swaging at 600 to 800°C.

One question immediately arises. What happens in the reactor in this situation where there are no cold clearances between core and cladding. Does the hotter oxide expand thermally and seriously distort the cladding? In effect, there are two simultaneous and competing phenomena. The core does expand thermally, but as operating temperatures are reached, the core also shrinks, because of sintering. Sintering in the core also improves the heat-transfer characteristics and fission gas retention. Internal pressures generated by release of fission gases can be controlled by incorporation of an inert reservoir at the end of the fuel rod.

Studies of fission gas release from compacted powders have been made and are continuing. An equation is being developed which will describe fission gas release in terms of neutron flux, an exponential function involving specific surface area of the powder, and two as yet unidentified functions of enrichment and irradiation time. Packing density appears to be unimportant.

The first irradiation tests of swaged UO_2 involved four and five inch long capsules exposed in the MTR. Zircaloy-clad UO_2 powder was swaged to 0.380 in. diameter to achieve densities of 85 and 90% of the theoretical. Figure 4 shows a cross sectional view of one which generated heat at the rate of 15 kw/ft or 110 watts/g UO_2 . The core temperature was estimated to be 1700°C.

No measurable increase of length or diameter was observed. The cylindrical crack was caused by sintering shrinkage of the hot core.

The upper photograph shows a three-rod cluster assembly which was irradiated to approximately 600 Mwd/T in the MTR. The maximum heat generation was 13 kw/ft per rod, or 32 W/g UO_2 . Bain and Robertson at Chalk River have irradiated similar swaged UO_2 capsules to greater than 2500 Mwd/T with no dimensional changes, within their limits of measurement. Several clusters of swaged rods prepared for irradiation in one of the Hanford pressurized, recirculating water loops are displayed in Fig. 5. These are scheduled to go to 2000 Mwd/T for study of dimensional stability and fission gas release.

Because there are occasional failures of cladding, during swaging, which would be inconvenient were the fuel to contain plutonium, the simultaneous UO_2 compaction and clad forming by isostatic pressing was investigated.

A

B

Fig. 4—A, uranium dioxide fuel element fabricated by swaging; B, swaged fuel element sectioned after irradiation.

Fig. 5—Four rod cluster, swaged UO_2 fuel elements.

Fig. 6—Deformation of aluminum alloy cladding by isostatic pressing.

A typical container is a 1% Ni-Al alloy extrusion consisting of two concentric tubes separated by six integral ribs. A cross sectional view is shown in Fig. 6. One end of such a tube is capped and the six channels filled with UO_2 powder. The proper atmosphere or vacuum is provided and the container sealed. The whole assembly is then pressed isostatically at 50,000 psi. The cladding collapses between the ribs and compacts the UO_2 to about 60 per cent of theoretical density. The cross section after pressing is shown in the other view. This element can be considered as a modified six rod cluster. Its advantages include the use of aluminum alloy, and the heat transfer paths formed by the integral ribs. The finished element comes automatically equipped with ribs to properly define the exterior coolant channels. A possible disadvantage may be dimensional instability as the UO_2 sinters during operation.

Figure 7 reveals an isostatically pressed element prepared for irradiation in the MTR. The element, containing 2.4% enriched UO_2 , is eight inches long and 1.4 in. OD. Power generation was calculated to be 48 kw/ft or 70 w/g UO_2 . The core temperature was estimated to be 2100°C. Figure 8 shows a cross sectional view after irradiation. No exterior dimensional changes occurred. Notice how the oxide has sintered and cracked. Hollow cores developed in some sectors and shrinkage away from the ribs is evident in all sectors. An interesting experiment will be to irradiate a similar element in a loop pressurized to somewhat over 1000 psi to determine if the cladding will follow the oxide as it shrinks -- much like hot pressing of UO_2 in a die.

A popular "ghost" is the fear of melting of UO_2 fuel element cores. No reactor operator is currently considering running his machine with melted UO_2 fuel elements, because of concern about fission gas release, possible reactions between molten UO_2 and cladding, and lack of any measurements of UO_2 expansion on melting. A small, but intensive, high temperature research program is underway at HAP0. The behavior of UO_2 near its melting point is being investigated by both ex-reactor and in-reactor experiments. Some interesting things have been observed. For instance, UO_2 melts only in an inert atmosphere. In a vacuum it sublimates. We were interested in what importance could be attached to sublimation as a mechanism of material transport in an operating fuel element.

Two one-inch diameter Zircaloy cans containing 3.2% enriched UO_2 powder were irradiated in the MTR, under conditions in which a large portion of the core was molten. Originally, the powder was packed to approximately 40% density. One can contained a helium atmosphere, the other a vacuum. Maximum heat generation was 60 kw/ft, or 60 w/g UO_2 . During operation the material migrated to the cooler end of the test elements, regardless of which end was up. A cross section of the helium-filled test assembly after irradiation is illustrated in Fig. 9. The limit of the molten core was 0.3 in. in diameter. The annular volume between the molten core and the cylindrical crack is sintered to an average density greater than 90% of theoretical. Note the radial columnar grain formation in this area. Similar grain structures have been interpreted in the past to be caused by solidification from the melt. An attempt to back calculate the radial temperature distribution in this element confirmed that the limit of melting was as described.

A later test element identical to the one described developed a cladding leak while operating with a molten core. The cladding failure appears to have been caused by insufficient localized heat transfer to the coolant. Other than the expected fission gas release there was no evidence of swelling or other catastrophic occurrence.

In summary, the aim of the irradiation test program is to provide answers to the questions of heat transfer and mechanical stability of sintered UO_2 tubes; to questions of fission gas release, thermal conductivity, core temperature limitations, and material relocation of non-sintered UO_2 powders.

Fig. 7—Isostatic formed UO_2 irradiation test element.

Fig. 8—Post irradiation view of interior of isostatic formed UO_2 test element.

Fig. 9—Irradiated UO_2 powder.

DISCUSSION

MR. J. P. HAMMOND (Oak Ridge National Laboratory): This concerns your hot swaging of these fuel elements. I have noticed in some British report that in warm pressing of UO_2 they got good results when they had a high oxygen-to-uranium ratio. Not so with lower ratios. I wonder if you found any such thing to hold in the hot swaging of your samples. If you did investigate this sort of thing could you tell me what your oxygen-to-uranium ratio was?

MR. ROAKE: No, we have not investigated the high O/U ratios for a number of reasons, one of which is that we are interested in high heat fluxes which lead to high temperatures in the core. We are concerned about oxygen release at high core temperatures. This, in turn, gives rise to the problem of loose oxygen floating around in the fuel. Where does it go? It combines with the cooler uranium oxide and with the zirconium cladding.

MR. HAMMOND: What was your ratio?

MR. ROAKE: It was very close to stoichiometric.

MR. KAUFMANN (Nuclear Metals, Inc.): How much is the power per foot or per pound of uranium or something of that sort that you get out of these oxide elements compared with what you get out of your uranium metal elements of similar diameter?

MR. ROAKE: Well, this depends on the number of fissionable atoms/unit volume. Uranium dioxide has roughly half the uranium atom density as does uranium metal.

MR. HAMMOND: Uranium metal is self-shielding.

MR. ROAKE: That also strongly affects the heat rate. The flux depression in the center of an oxide element is not nearly as great as in a similarly shaped piece of uranium metal.

MR. KAUFMANN: Just take the per foot of length. How does the metal compare with the oxide?

MR. ROAKE: I think that the best way I can answer you is to recall the six clusters I showed you on the slide. These were prepared for an irradiation test. They contain 1.6% enriched UO_2 and they are designed to generate the same heat/unit length as identical natural uranium metal elements.

MR. J. W. ULLMAN (Oak Ridge National Laboratory): Have you observed any reaction to UO_2 ?

MR. ROAKE: No, we have not; we looked for it specifically in the case of the isostatically pressed elements. There we have UO_2 in aluminum. Apparently sufficient heat was carried through those ribs to prevent any solid state reaction. I have to qualify this by saying that we have not looked at it under metallographic magnification. This element was cut open just last week. Visibly there is no evidence of reaction. (No evidence of reaction observed on later microscopic examination. -- W.E.R. (7-21-58))

MR. E. L. CURRIER, JR. (ACF Industries): In that swaged oxide can does the wall thickness decrease by swaging and how uniform is the wall after it is swaged?

MR. ROAKE: The thickness of the cladding increases when you first begin to swage. Then you reach a breakover point -- the thickening stops and the cladding

begins to thin out. When you swage an empty tube, depending, of course, on the die design, the entrance angle, etc., the tube wall thickness increases. This is essentially what happens when the tube is filled with UO_2 , until the oxide is sufficiently dense to act as a mandrel. From that point you are swaging the cladding tube over a mandrel and this makes it grow longer and thins it out.

Paper 33

IRRADIATION BEHAVIOR OF $\text{ThO}_2\text{-UO}_2$ FUELS

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ABSTRACT

$\text{ThO}_2\text{-UO}_2$ fuels have been studied because of certain advantages possessed by this type of fuel over UO_2 . Pellets of $\text{ThO}_2\text{-UO}_2$ have been found to be highly stable dimensionally under irradiation. Various experimental assemblies which have been irradiated have included aluminum, Zircaloy, and stainless steel cladding, with the annulus surrounding the pellet filled with air, helium, NaK, or lead. A full-scale core of pellets lead-bonded to aluminum tube-plates has been used successfully in the Borax-IV reactor. Preliminary defect tests have shown that solid fission products are retained and that only the gaseous products, Xe^{138} and Kr^{88} , are detectable in the reactor coolant system.

INTRODUCTION

Most of the ceramic fuel development at Argonne National Laboratory has been concerned with thorium oxide in which has been dissolved various amounts of enriched uranium oxide. Our interest in this type of fuel came about because of two advantages possessed by thorium oxide-base fuels. First, thorium oxide-uranium oxide bodies can be sintered in air, whereas uranium oxide bodies must be sintered in vacuum or a protective atmosphere. Second, the use of thorium atoms in the fuel presents the possibility of constructing a thermal breeder based on conversion of natural thorium to the fissionable isotope U^{233} .

This paper will discuss results which have been obtained in experimental irradiations of $\text{ThO}_2\text{-UO}_2$ fuel specimens, and will conclude with a brief description of reactor operating experience on these fuels.

RESULTS ON EXPERIMENTAL SPECIMENS

The densities of the $\text{ThO}_2\text{-UO}_2$ specimens we have studied have been in the range of 85 to 90% of theoretical. Our first irradiations were made on specimens with the composition $\text{ThO}_2\text{-2-1/2 wt } \% \text{ UO}_2$.¹ The specimens were in the form of pellets, $\sim 1/4$ in. in dia by $3/4$ in. long. The specimens were irradiated in NaK capsules. Some of the specimens were irradiated bare and others were clad in Zircaloy-2 or in stainless steel, with the bond between the fuel and cladding consisting either of NaK or air. Figure 1 shows bare specimens after irradiation. It can be seen that the appearance of the pellets was not affected by the irradiation they received. Dimensional changes in the pellets, if any, were less than 0.0002 in. The clad pellets also showed no changes resulting from irradiation.

Fig. 1—Effect of irradiation ThO_2 -2 $\frac{1}{2}$ wt. % UO_2 ceramic specimens.

The next specimens that were irradiated in NaK capsules were made from the composition ThO_2 -10 wt % UO_2 .¹ These were in jackets of aluminum-1 wt % nickel alloy. One-half of the specimens contained a mixture of helium and argon in the annulus between the pellets and the cladding, and the other half contained

Total Metal			
Atom Burn-up, %	0	0.20	0.88
Irrad. Temp. °C	-	115	450

lead in the annulus. Figure 2 shows typical specimens of each type after irradiation and illustrates that the cladding failed on the gas-bonded specimen, whereas no damage was found in an identically irradiated lead-bonded assembly.

Figure 3 shows the same two specimens after partial removal of the cladding. Here it may be noted that the lead has held the fuel fragments in place in the lead-bonded specimens, whereas in the gas-bonded assembly the pieces have moved about sufficiently to damage the jacket and finally cause local failure.

Figure 4 shows a transverse section of a gas-bonded assembly which operated at a high enough temperature to melt the aluminum alloy cladding. Also a central void can be noted in the ceramic pellet. This is believed to have resulted from sintering or possibly melting of the oxide.

In order to irradiate experimental specimens at high heat fluxes without overheating the aluminum cladding, it was decided to directly water-cool small assemblies in the MTR process water. Figure 5 shows the specimen shape which was selected. It consists of a plate 6 in. long containing two tubes of pellets lead-bonded to the aluminum alloy cladding. In order to satisfactorily weld the end of the assembly, after the lead had been added, it was necessary to leave a short space free of lead, as shown to the left in the horizontal section. Figure 6 shows a cross section of the specimen and illustrates the lead bond. Figure 7 shows the specimen in an aluminum holder which can be fitted into an X-basket at the MTR. With this arrangement, the plates are cooled by a 1/16-in. annulus of water flowing at about 25 ft/sec.

Lead-Bonded

Gas-Bonded

Fig. 2— ThO_2 -10 wt. % UO_2 irradiation specimens in Al-1 wt. % Ni alloy cladding after 0.64 per cent metal atom burn-up at 425,000 Btu/hr/sq ft.

Lead-Bonded

Gas-Bonded

Fig. 3— ThO_2 -10 wt. % UO_2 specimens irradiated to 0.64 per cent metal atom burn-up at 425,000 Btu/hr/sq ft after partial removal of aluminum alloy cladding.

Fig. 4—Evidence of central melting or sintering in ThO_2 -10 wt. % UO_2 specimen in gas-bonded Al-1 wt. % Ni alloy cladding after 1.25 per cent metal atom burn-up at 840,000 Btu/hr/sq ft.

Top View

Horizontal Section

Fig. 5—Lead-bonded ThO_2 -6.36 wt. % UO_2 irradiation specimens.

Fig. 6—Cross section of ThO_2 -6.36 wt. % UO_2 irradiation specimen.

Fig. 7—Lead-bonded ThO_2 -6.36 wt. % UO_2 specimen in irradiation holder.

To date we have examined only one of these specimens after irradiation. This particular plate received a burn-up of 1.0% of the metal atoms, or 7500 Mwd/T of oxide. The heat flux from the aluminum was 190,000 BTU/hr-ft². The only change which developed during irradiation was a 5.5% increase in dia of the aluminum tubes at the end of the specimen where the flux was highest. Other specimens which just arrived from the MTR and which are awaiting examination in our hot labs have received higher exposures under more severe conditions. Burn-ups in these specimens are estimated to have reached about 4.3% metal atom burn-up, or 32,000 Mwd/T of oxide. The maximum heat flux from the aluminum cladding in these specimens was on the order of 700,000 BTU/hr-ft².

REACTOR FUEL RESULTS

The excellent performance of ThO₂-UO₂ fuel in irradiation tests led to its use as fuel in the Borax-IV reactor. The composition selected was ThO₂-6.36 wt % UO₂. Tube plates of the type shown in Fig. 8 were extruded from aluminum-1 wt % nickel alloy and the oxide pellets were loaded into each tube with a lead bond.^{2,3} The plates were spot welded into a box-type subassembly shown in Fig. 9. The subassembly is ~ 4 in.² and 32 in. long. Originally, 59 subassemblies were used to make up the core of the reactor. The reactor first achieved criticality in December 1956, and for more than a year was operated with no difficulties. During this period of time the reactor was operated up to 14.5 Mw with a peak heat flux near 300,000 BTU/hr-ft².

The reactor was shut down for 10 weeks beginning in December 1957, to increase the core size to 69 elements and to remove the boron poison strips. When the reactor was started in February, it was found that gaseous fission products were present in the condensate return. By sampling the air ejector flow these gases were found to be mainly Xe¹³⁸ and Kr⁸⁸. The amount of short-lived xenon present relative to the longer-lived krypton was less than would be predicted from fission yields. It appeared, therefore, that before the gases escaped from the defective fuel element to the steam they were required to diffuse some distance. The activity in the reactor water itself showed no substantial increase over normal operating conditions. This indicated that if water were contacting the fuel pellets, they were not releasing activity through corrosion or leaching. Location of the defective element or elements has proved to be unexpectedly difficult, so that as yet no information is available concerning either the nature of the defect or its location.

In order to study the effect of irradiation on a defected ThO₂-UO₂ fuel assembly under more controlled conditions, R. E. Bailey and R. F. S. Robertson recently conducted an experiment in EBWR in which several fuel pellets were irradiated in a stainless steel tube with a 0.010-in. hole drilled in the end of the tube. The defected tube was operated in the reactor for eight days at various reactor-power levels, including 30 hr at full power. During this run the heat flux from the cladding surface of the defected assembly was ~ 330,000 BTU/hr-ft². The pressure in the reactor vessel and the power generation were varied during the test so that the defect "breathed" reactor water in and out to repeatedly waterlog the pellets.

The gaseous fission products Xe¹³⁸ and Kr⁸⁸ were detected in the gases from the air ejector, just as they had been at Borax-IV. At full power, for example, Xe¹³⁸ was released at about 2.4 curies/day. Slight traces of only one solid fission product, I¹³¹, were found in the reactor water. The results from these preliminary studies may be compared to those reported by Westinghouse investigators⁴ for defected UO₂ rods. In their loop experiments in the NRX reactor, release of not only xenon and krypton was encountered, but also numerous solid fission-product activities were found in the water. Since essentially only gaseous activities have been found in the Borax-IV and in the EBWR defect tests, there is some indication that defected ThO₂-UO₂ fuel ele-

Fig. 8—Cross section of Borax-IV fuel plate.

Fig. 9—Borax IV fuel subassembly.

ments may be somewhat more resistant to leaching of solid fission products by hot water than are defected UO_2 fuel elements.

CONCLUSIONS

In conclusion, it can be stated that irradiation experience to date has shown that $\text{ThO}_2\text{-UO}_2$ fuels are highly stable under irradiation. Fuel elements containing this ceramic fuel have shown excellent performance in arrangements where the pellets are lead-bonded to aluminum alloy cladding. Preliminary tests indicate that essentially only gaseous fission products are released from a defected fuel element.

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DISCUSSION

MR. J. P. HAMMOND (Oak Ridge National Laboratory): How do you put the fuel material in the element? Do you use silicon bonding as was described for the ALPR?

MR. KITTEL: Where the ends are bonded, yes.

MR. HAMMOND: Is this also an M-388 aluminum alloy?

MR. KITTEL: Yes.

MR. HAMMOND: How does this silicon interface affect the corrosion resistance of this aluminum alloy under pressurized water conditions? Have you investigated this?

MR. KITTEL: These elements have been subjected to numerous corrosion and autoclave tests, and to my knowledge no problem has existed at the bond line where they might be expected to be less corrosion resistant.

MR. J. C. DANKO (Westinghouse Atomic Power Division): Have you done any work on the UO_2 end of the diagram with regard to ThO_2 additives and sinterability in air?

MR. KITTEL: Yes, the ceramists have been able to add as much as 78% UO_2 to ThO_2 and still successfully sinter the pellets in air. We are now irradiating material which has as much as 26% UO_2 in ThO_2 and this also was sintered with no difficulty in air.

CHAIRMAN BILLINGTON: I should like to ask whether you attempted to measure thermal conductivity of those original pellets?

MR. KITTEL: No, we did not. For thermal conductivity we have relied on published data for ThO_2 .

Paper 36

URANIUM FUEL

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The fuels development program at Hanford supports the present reactors, future production reactors, and the plutonium recycle reactor.

For the present reactors, our effort involves, for example: the development of geometries with higher surface-to-volume ratio such as the internally and externally cooled slug; the development of improved core and cladding materials; and the development of alternate jacketing methods such as hot press, cold closure, and extrusion cladding.

For future production reactors the research and development program entails new designs with extended surface suitable for safe, long exposure operation in high-temperature water or organic coolant. The elements are intended for operation in horizontal pressure tubes, and both aluminum and Zircaloy cladding are included in the program. Designs being developed include a seven-rod cluster element and a hollow cylindrical element possibly with a solid rod in the center. Alternate fabrication techniques such as direct casting of uranium into zirconium components are being studied.

The plutonium recycle program covers many facets of reactor technology. Included among them are uranium fuel development and plutonium fuel development. In the field of uranium fuels, UO_2 has been selected for the start-up element. Two geometries are under consideration, the first being a massive tubular element fabricated by more or less conventional techniques and the second being the 19-rod cluster wherein the individual rods are fabricated by the direct swaging of UO_2 powder in Zircaloy jacketing. Future studies will be directed toward reducing fuel cost as well as exploring the ultimate limitations with respect to temperature, specific power, and exposure.

The plutonium fuel group at Hanford will be called upon to fuel some 20 to 30 channels in the Plutonium Recycle Test Reactor with plutonium-enriched elements. The fuel under development at the present time is the 19-rod cluster employing plutonium-aluminum alloy. The completion of the plutonium-fabrication pilot plant now under construction will permit development work employing other metal and ceramic fuel materials.

As you can see, new reactors have caused a re-alignment of the metallurgy program to emphasize higher temperatures, power, and exposure than those previously included in Hanford studies.

Month by month our basic metallurgy program is accumulating data which permit an understanding of the phenomena which occur in uranium and structural materials under irradiation. We believe this understanding will permit prediction of, control of, and allowance for irradiation damage which occurs in both metallic and ceramic fuels.

Paper 37

URANIUM ALLOY FUELS

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My instructions with regard to subject matter for this talk were that it should be concerned with alloys for fuel elements which would be useful for production of economical power. The exclusion of reactors for producing mobile power or for other special purposes severely restricts the discussion on alloys. For stationary power plants the only material of interest at the present time is pure uranium or uranium-rich alloys. There are two reasons for being interested in alloying, namely, improved corrosion resistance and improved radiation stability.

A great deal of work has been carried out aimed at finding an alloy of uranium that would make it corrosion resistant to high-temperature water or air. At the present time there is little reason to believe that such an alloy will be found. Good corrosion resistance is obtained only when alloying has been carried to the extent that the uranium is no longer the major constituent. Some uranium-rich alloys which are more corrosion resistant than pure uranium by a factor of 100 or 1,000 have been found. Such alloys are uranium-12% molybdenum, uranium-10% niobium-5% zirconium, and uranium-4% silicon. The first two of these alloys are corrosion resistant only when the γ phase is retained and the alloy additions are in solid solution, while with the silicon alloy it is necessary to develop an intermetallic compound by heat treatment to insure that no orthorhombic uranium is left. It is not generally realized that these three alloys contain only about 70 to 75 at. % uranium. They hardly qualify as uranium-rich alloys and, with the exception of the silicon alloy, it is doubtful that they can be used in a reactor aimed at producing economical power. At the present time there is little reason to believe that uranium-rich alloys can be made adequately corrosion resistant.

Concern with radiation damage arises from two different phenomena, namely, anisotropic growth and swelling at elevated temperatures. At the present time distortion due to anisotropic growth is no longer a matter of major concern to reactor builders. The reason for this is that satisfactory behavior can be obtained by a β heat treatment which produces a more or less random orientation of the crystals and a fairly fine grain size. Small additions of alloying elements, such as 2% zirconium or 3% molybdenum, do give improved resistance to anisotropic growth. With such alloys it is still necessary to obtain a random orientation by heat treatment, and it is probable that the only function of the alloying addition is to produce a fine grain size. It is now believed that anisotropic growth more or less vanishes at temperatures above about 500°C and, therefore, this subject is not of great concern for high-temperature operation such as is required in power reactors.

Swelling appears at temperatures above about 450°C in pure uranium. Swelling is believed to be due to accumulation of gaseous fission products and, therefore,

it obviously becomes worse at high burn-ups. It appears that an alloy of uranium-10% molybdenum is definitely better than pure uranium in resisting swelling. It is not known at the present time how much improvement, if any, is obtained by smaller alloying additions. It is clear that with any material, including uranium oxide, trouble will develop as burn-up or operating temperature is increased. The best that can be hoped for is that certain materials will lead to a significant improvement in the operating characteristics which may be great enough to justify their use. It is possible that future research may demonstrate that small alloy additions to uranium under optimum conditions of heat treatment and use will enable these materials to go to five or ten times the burn-up and perhaps twice as high a temperature as is possible with pure uranium.

Paper 38

BULK OXIDE FUEL

E. F. LOSCO

Westinghouse Atomic Power Division, Pittsburgh, Pa.

The use of bulk UO_2 as a fuel in long-life, high-power reactors has been emphasized by the several papers so far presented in this conference and is receiving more and more attention.

Perhaps the principal characteristics of UO_2 which make it so attractive might be listed as: (1) its high-irradiation stability; its relative freedom from dimensional changes, due to its ability to retain large fractions, in the order of 99.9% of the fission gases formed, and also its freedom from irradiation or temperature-induced phase transformations makes it a highly attractive fuel. (2) Another reason for its attractiveness is, of course, its chemical inertness to high-temperature water. (3) A third advantageous feature is its high-melting point, in the order of 2750°C . From the standpoint of fuel-operating characteristics, it is theoretically possible to operate up to the melting point of UO_2 . This, in fact, has been done at least on an experimental scale.

There are, however, certain features of UO_2 which are actual disadvantages when they are compared with metallic fuels, and which should receive more attention in future development. Among these are, for example, its very low thermal conductivity which is about $1 \text{ Btu/hr/ft}^2/^\circ\text{F}$. Recent data obtained at Bettis, and I believe also confirmed at Argonne National Laboratory, suggests that under irradiation, conductivity may be about one-third of the out-of-pile value. This needs more work for confirmation since the results may be affected by gross fracturing of the pellets and may not be caused by irradiation-induced property effects.

Another feature of UO_2 that requires additional work is its low thermal shock resistance. This may actually be a factor in the observed low thermal conductivity values obtained so far in-pile. The material, of course, is quite brittle, which means that the methods which one can use to fabricate clad fuel elements are limited. Thus, there is less flexibility for fuel-element fabrication which may affect realization of the full potential of UO_2 unless means of circumventing the brittleness problem are developed.

Another factor which limits flexibility in UO_2 application is that there is presently no known method of metallurgically bonding this material to the cladding to get better heat-transfer characteristics.

Perhaps the encouraging thing about UO_2 is that we are at a stage now which permits recognition of its advantages and disadvantages so that we can do something about the latter in order to take advantage of its attractive properties as much as possible. In the meantime I think that methods of overcoming some of its shortcomings should be investigated. Some suggested items for investigation might be:

1. Means of improving thermal conductivity and thermal shock resistance through small additions of inert ceramics such as beryllium oxide. Work along these lines I think is already in progress at several laboratories and should bring some fruitful results.
2. Means should also be developed for using U-235 - UO_2 as a seed material, so that we can get increased life of the seeds compared to what we expect to receive in the way of natural blanket material. The dilution of enriched UO_2 with an inert ceramic, such as cerium oxide or zirconium oxide, should be fruitful. Work so far performed along these lines indicates that the diluent should be in solid solution. Ceramic dispersions, such as $\text{Al}_2\text{O}_3\text{-UO}_2$, do not seem to have the irradiation stability shown by solid solutions, such as $\text{ZrO}_2\text{-UO}_2$.
3. Another problem that should receive attention is means for incorporating a burnable poison in UO_2 . Work at Bettis with B-10, either in the elemental form or as a compound, has not been successful thus far, apparently due to volatilization of boron during sintering of the UO_2 in a hydrogen atmosphere. More work should be done along these lines and it should be expanded to include other types of burnable poisons.
4. Some work should be directed toward solving the UO_2 -clad bonding problem. Perhaps it is not as formidable as it at first appears to be.
5. Work should be directed also toward a better understanding of irradiation behavior with very high burn-up, in the order of 100,000 Mw days/ton. This, of course, should be done in conjunction with evaluation of cladding materials, since it may very well be that the cladding rather than the fuel may limit realization of the full potential of the fuel element. In this connection phase-diagram and thermodynamics of zirconium-hydrogen-water systems should be investigated.
6. More effort should be directed toward the studying of the kinetics of fission gas release in bulk-powder UO_2 and UO_2 -inert ceramic combinations as affected by such things as non-stoichiometric uranium-oxygen ratio.
7. Composition in the case of combined ceramics, grain size, burn-up temperatures, separation of fission gas release resulting from recoils and that resulting from the diffusion are also fields of interest which merit further attention.

Paper 39

DISPERSION FUELS

C. E. WEBER

Knolls Atomic Power Laboratory, Schenectady, N. Y.

To attempt to predict the future of dispersion fuels, let us look at today. Dispersion systems consisting of UO_2 -stainless steel, UO_2 -nichrome, U_3O_8 -aluminum, and UO_2 -aluminum have been studied, and a lot of work has been done on dispersed boron compounds. One might ask what were the reasons for the initiating work on these particular systems. In my opinion work on the UO_2 -stainless steel system was first started to capitalize on the properties of the matrix and cladding; namely, high strength and the corrosion resistance to high-temperature air. U_3O_8 and UO_2 was dispersed in aluminum to achieve higher loadings than could be achieved with UAl_4 . Boron dispersions were investigated on the assumption that performance of boron as a fissionable metal would be similar to that of the uranium compounds in dispersion systems.

Where are we today? Without debating whether the dispersion theory preceded practical application, the theory that a dispersion is better than an alloy has been proven true, proving that the particular criteria to be used to define improvement is properly chosen. Improved post-irradiation ductility through dispersion of the fissile phase is a proven fact. We will agree that the corrosion resistance of a dispersion is determined by the rate of attack of undamaged matrix metal. If you had a cladding and should a crack develop into the fuel core, only a small fraction of the total fuel is exposed to the coolant and progressive attack should not occur.

On the other hand, consider fission product induced swelling; is a dispersion inherently more stable? One predicts, and this has been partially verified, that the swelling would be worse for dispersion fuel elements than in alloys of the same over-all composition, such as an alloy of uranium-zirconium relative to a dispersion of large particles of UZr_2 in zirconium. Since on a constant volume of fuel core the fission products are much more concentrated in the dispersed particle than is the case when the matrix metal is used as a diluent, then if the change in property being measured increases more than linearly as the fission products are concentrated into a localized zone, the dispersion will be more damaged than the alloy. In this manner a gain in ductility may be realized at the expense of stability with respect to other properties such as swelling resistance.

I hope that this example demonstrates where we are today. The question is, from a fuel-element design basis, how do all the variables such as ductility and resistance to swelling add up in a fuel element? It is the sum total result such that over-all performance is improved through dispersing the fissile phase, which is the most important variable. Each of us has his opinion, but we need a much better understanding before this question can be answered.

Another question is that, in a dispersion element, a variable degree of stringering and fragmentation may be present. The real possibility exists that

an optimum amount of such stringering and fragmentation may exist and performance may be better than that of high density, randomly mixed and stringer-free dispersions. Theory has yet to be proven for this case.

To look to the future, one of the first things to do is to answer today's questions. Future applications will probably capitalize on the unique advantages of dispersion elements as applied to a specific design situation. An example previously mentioned is that of using stainless steel matrix and cladding in gas-cooled reactor application where high strength and corrosion resistance to gas is required.

Secondly, I feel that the chemical form of the fissile material is an important design variable. I feel that UO_2 is a far superior material with respect to both normal expansion due to fission product accumulation and with respect to high temperature stability to fission gas diffusion and agglomeration because of the ionic state of the atoms in UO_2 and its high melting point.

Further, in the dispersion elements, you can achieve high surface areas, reach high heat fluxes, still holding to a low particle temperature compared to straight ceramic cores. I think this is an advantage of the dispersion element.

In the future, we will probably find broader ranges of second phase material being used. At the present time, dispersion loadings up to 25 or 30 volume % of the fuel core are the vogue. We may see, for example, UO_2 loadings up to 80% with 20% of an interlacing metallic phase to give better thermal conductivity and other improved properties. The uranium-magnesium element that Hanford has worked on is well-behaved and an example that may be cited. Perhaps UO_2 or similar ceramic systems will behave in a similar fashion.

To conclude, incorporation of internal porosity in the dispersion element to provide space to accommodate expansion due to fission product accumulation may improve performance at very high exposure levels even though low burnup data from studies on the PWR elements indicates porosity is not desirable.

Paper 40

FUELS FOR FAST BREEDER REACTORS

F. G. FOOTE

Argonne National Laboratory, Lemont, Ill.

As all of you are aware, of course, most of the discussion at this conference has been on the thermal reactors. A fair amount of our effort at Argonne is, however, on fast breeder reactors and in a general sort of way we are dealing now with a considerably different set of problems; a set of problems which are based primarily upon the fact that a fast breeder is a machine with a larger critical mass. Even a small breeder, a very concentrated core breeder such as the EBR-I, has a critical mass of somewhere around 50 kilos of U-235.

A breeder which is diluted to the point where it can produce appreciable amounts of power will have a critical mass of several hundred kilos of fissionable material. This has a number of unfortunate corollaries. One very obvious one, of course, is that such a machine has a very high inventory of fissionable materials and, therefore, the carrying charges on the fissionable material are high. This in turn puts a high premium on a fast turnover of fuel. It is difficult to imagine that we can afford to have in the fuel cycle several critical masses. It puts a very high premium, therefore, upon the development of a rapid reprocessing method. It puts a high premium upon keeping the amount of fissionable material committed to a particular reactor to the minimum that you can get away with.

Of course, it is quite impossible to operate a reactor system with only one critical mass, but one can hope that one can operate the cycle with perhaps 1-1/4 critical masses or 1-1/2 critical masses, and thereby keep the amount of fissionable material committed to the reactor down to a minimum.

Since in general the aqueous processing methods require a fair amount of cooling time, the time when the fissionable material is out of the reactor is not doing you any good. It is not working. We have felt for a long time that we must provide a more rapid reprocessing method, and this has led to the development of the pyro-metallurgical processes which were described rather briefly this morning. In this way we cut ourselves loose from problems concerned with the aqueous dissolution, the acid dissolution of the fuel material, of the disposition of the water and the acid.

The pyro reprocessing method, therefore, can work on very fresh material primarily because there is nothing much to be decomposed by the radiation activity associated with the material. In working out a reprocessing method, the very simple oxidative dissolving method seems to be quite applicable to fast reactor fuels, and the fissium, which was described this morning by Messrs. Lowe, Sharp and Kittel, is an artificial alloy defined primarily by what the reprocessing method will or will not remove.

It is true, therefore, that the compositions of the fast breeder fuels are set not so much by the metallurgical conditions as by the reprocessing considerations, that is, by what the particular reprocessing method chosen for the given reactor

will leave in the material. It is obviously true that if the reprocessing method that is chosen for a particular reactor is one which was not tested for fission, then the composition of the fission changes and, therefore, the composition of the fuel with which you must deal also changes.

While the method that we have presently chosen, oxidative dissolving, is very simple and a very obvious way of cleaning up the fuel, it is not necessarily the only process which we could use. We find ourselves, therefore, with a set of metallurgical problems which arise not from metallurgical considerations but primarily from the processing considerations, and I do not doubt that in the future we will be dealing with a whole series of odd alloys merely because the reprocessing methods which are in vogue then are different from those that are presently used and which will therefore contain different elements, in different amounts, and different ratios than those that we are presently dealing with.

Another feature of the fast reactor is, of course, the fact that it is a metal-cooled reactor and it is, therefore, operable at high temperature without development of excessive pressures, and the premium that is placed also on very high burnouts. In these fast breeder fuels we have run into the rather obvious problem of fission gas swelling. Whether or not we will be able to achieve the temperatures and the burnups which we would like to achieve may very well be limited for some time to come by this problem of fission gas swelling, and in so far as we are concerned, this is still one of the major problems in the development of a wholly suitable, really attractive fast reactor fuel.

The oxide fuels, while they are of some interest in fast breeders, are still not very well developed and in our view mostly because they do not lend themselves to a fast turnover cycle. We have ideas as to how such fuels could be handled by pyrometallurgical processes, but at the moment the techniques by which one would attempt to reprocess spent oxide fuels is not very well known. I think perhaps something of this sort can be worked out in the future.

CLADDING AND CANNING MATERIALS

J. E. CUNNINGHAM

Oak Ridge National Laboratory, Oak Ridge, Tenn.

As this and other conferences in the past have borne out, the selection of fuel cladding or canning material for nuclear power generation is often a difficult and controversial task. It is complicated by virtue of the fact that the solid fuel element is exposed to the most intense radiation, the greatest burn-up, the severest corrosion, and the highest thermal stress of any of the components in the heterogeneous reactor system. Final selection depends on proper evaluation of a host of interrelated factors associated with economics, design, fabrication, and performance.

The basic or underlying aim is quite clear; that is, to produce power at the lowest possible net electrical cost. If one looks at the economic picture in terms of cladding and canning materials that will give dependable or reliable performance, some general trends are evident:

1. In the case of gas-cooled reactors, where the direction is aimed at utilization of UO_2 as the fuel and higher thermodynamic efficiency, stainless steel holds a decided edge over other potential candidates for the canning material. Low-melting metals such as aluminum and magnesium do not possess the necessary strength for high-temperature operation, while the principal disadvantage of zirconium-base materials is the lack of chemical inertness in contact with the coolant. The fuel component which looks most attractive for service in gas-cooled reactors in the immediate future, therefore, is low-enrichment UO_2 canned in stainless steel. Materials which should be extensively investigated for use in advanced concepts of this reactor type in the future include: beryllium, impervious graphite, beryllium oxide and other ceramic coatings.

2. In the case of the pressurized-water or boiling-water reactor concept, the decision on which material to employ is not quite so obvious. Both zirconium and stainless steel have been used with considerable success, so that the choice between these two materials should generally be made on a pure economic basis. Before one can seriously consider the use of a new material, however, technical feasibility must first be established.

The economic advantages of zirconium and stainless steel for various fuel-element applications were compared recently in a study made by Benedict et al., under contract to an industrial supplier of zirconium. The report recommends zirconium for application employing normal or slightly enriched uranium and stainless steel for highly enriched uranium systems such as the APPR and STR. Of the two materials, zirconium, of course, offers the greatest potential for material cost reduction, but it is difficult to see how the cost of zirconium will ever drop below the prevailing high price for titanium sponge because of the hafnium

separation step required. On the other hand, stainless steel imposes a severe nuclear penalty because of its relatively high cross section for thermal neutrons and is subject to poison-forming tendencies.

The chief advantage offered by stainless steel is its amenability to simple and inexpensive fabrication practice. This factor alone far outweighs material cost and is generally not properly weighed in most economic studies. Stainless-steel components can be processed both metallurgically and chemically at a very small fraction of the cost entailed in the processing of similar components of zirconium.

In the future when it is possible to employ either material, I believe that stainless steel will prove to be the superior material for the construction of reactor-fuel components. The radiochemical transport problems associated with cobalt-60 in the primary loop can be minimized by employing a low-cobalt grade of stainless steel. Zirconium is inherently anisotropic and is subject to hydrogen embrittlement during radioactive service in contact with water. These advantages will tend to become more pronounced as we attempt to reduce fuel-cycle costs by extending reactivity lifetime.

Vigorous efforts should be made on the zirconium program to reduce the cost of mill products to improve fabrication schedules, so as to eliminate the presence of stringers, and to develop cheaper techniques for fuel-element manufacture. In addition I firmly believe that we need more basic studies aimed at gaining a better understanding of the mechanism by which zirconium picks up interstitial atoms and the influence of these constituents on mechanical properties. In the case of stainless steel, more work should be directed toward minimizing the radiochemical transport problem and investigating the potential of the straight iron-chromium alloys for fuel-element applications.

The last item I want to talk about is the development of the M-388-type aluminum alloys, which possibly brings a third candidate into the picture. Thus far, aluminum-base alloys of this type look promising in small-scale laboratory tests but require very careful control of the water pH. Phosphoric acid addition must be made to reduce the solution pH level to ~ 3.5 . Such a solution we would expect to be rather corrosive to steel. Another disadvantage detected during the evaluation of the alloy is that it corrodes rather rapidly under dynamic or flowing-water conditions.

In the event, however, that full-scale engineering tests demonstrate that the M-388-type alloys can compete corrosion-wise, then stainless steel and zirconium will be hard pressed to compete as cladding and canning materials in fuel-element construction. Current work to further improve the corrosion resistance by alloy addition should definitely be pursued, and dispersion hardening should be investigated as a possible means for improving creep resistance.

At the present time I see all of these materials as offering potential advantages in specific areas, and would recommend that we not pursue any one at the expense of another.

Paper 42

CHEMICAL REPROCESSING

J. W. ULLMANN

Oak Ridge National Laboratory, Oak Ridge, Tenn.

I am not going to say anything about the trends in metallurgy because I am not qualified. If you gentlemen would follow up on Mr. Cunningham's suggestion and make an aluminum clad I don't believe we would have anymore complaints. You might ask what is really the function of chemical reprocessing of a fuel element? Why bother at all?

The first thing is that so far as we have not developed a power reactor element which you can burn 100 per cent; therefore, our job in reprocessing is to recover the residue that is left. Second, we have a distinct function in the assistance that we can give the reactor in the periodic discharging of fuel to remove fission products and to improve the nuclear behavior of the system. Finally, if we get into the promised land of breeding, we will sooner or later have to remove the breeding gain in order to sell it.

I have made up a list of a few trends that I can see in chemical processing, and it is surprising that a number of them are intimately tied to metallurgical and fuel element fabrication questions.

First of all, in much the same manner as Mr. Foote sees it, I see a strong pressure on the chemical processing people to recover fuel elements with very short cooling periods. If we assume the current AEC lease charge of 4 % per year on fissionable material, this means that every day that we hold the fuel element we have to pay the equivalent of 0.01% loss. If you take the typical cooling times we are used to, of 120 days or 150 days, this is costing us the same as 1% loss. If we are talking about a fast reactor system where the burnup may be perhaps 4% or so, then we will require many processing cycles for full burnup, and if we lose either physically or in cost 1% every time, we have wiped out the advantage of the breeder.

Along these lines I don't think the situation is at all hopeless. Mr. Foote has mentioned the development of new high temperature methods which give relatively limited separation from fission products but which are highly immune to radiation damage. These methods are already well along in development for metals and work is starting now on this type of treatment for oxides.

A second method which has just recently been put through its initial pilot plant tests is the use of the volatility process. This is a particularly interesting method because it has the advantage of not suffering from radiation damage and yet it is capable of high separation from fission products.

The third hope that we have is that we can develop new organic reagents which are more immune to radiation damage than the solvents we are now talking

about. There are quite a few candidates and some of them appear perhaps five to ten-fold superior both in radiation resistance and in extraction performance to tributyl phosphate which has been our standby.

We are going to require, primarily from the safety angle rather than pure economics, a great deal of work in the field of waste disposal and waste treatment. For most power reactor fuels we can tolerate the cost of putting our waste into a tank, but this is not a satisfactory solution for storing waste for the centuries we will have to do it.

We have a great deal of work to be done in the field of maintenance of radio-chemical plants. At the present time we have two basic philosophies. One is to build a plant as cheaply as possible and then decontaminate it, and maintain it directly. The second is to put in expensive equipment so that we do not have to decontaminate the plant, but we pay a premium for this in the initial stage in the plant.

Typical of some of the techniques that we are looking at are underwater maintenance, the possibility of remote welding, and the possibility of what we call modular construction whereby an entire section of a plant could be removed and remotely replaced.

In the field of trying to reduce the cost of operation, two developments appear to be of a great deal of interest. One is the use of more inline instrumentation whereby we can continuously keep tab on what is going on in the process. This can result in a great reduction of manpower in both the plant and in the analytical laboratory.

Another field where we are putting an effort is the question of how to increase plant capacity. In the last analysis the unit cost of reprocessing is most sensitive to the amount of material you put through. We are doing two things of particular interest along this line. One is we are trying to see if it is feasible to introduce a new concept of criticality control in the plant -- purposely adding poison materials. The other is that we are looking at the use of far more efficient equipment, particularly high speed contactors for liquid-liquid extraction. If this works out, we will have gone a long way toward reducing the radiation damage to our reagents.

One final word. We have a program going on cooperatively between our Metallurgy and Chemical Technology Divisions and we think this is distinctly a step in the right direction, toward understanding each other's problems and perhaps arriving at more optimal economic solutions.

Paper 43

RADIATION DAMAGE

D. S. BILLINGTON

Oak Ridge National Laboratory, Oak Ridge, Tenn.

I can only talk about fuel-element technology from a very limited viewpoint and that is radiation damage. I will try to do a little bit of anticipation, however. I suggest, for instance, that some day a reactor will be built for the specific purpose of testing fuel elements. I also have some ideas about how this reactor will be built. It will have a range of fast flux, let's say from 10^{10} or 10^{15} for those people who need fast flux, and it will have the same range of thermal flux for those people who want thermal flux. Furthermore, those who are using the thermal flux will not get in the way of those who are using the fast flux.

We will have no gamma heat. We will have lots of room. We will be able to do a lot of conventional metallurgical tests in-pile that we would like to do. It will be nice to be able to put a regular machine inside the reactor.

Well, surprisingly enough, it is possible to build such a reactor, and this is a hollow-sphere graphite with uranium on the inside surface. The only thing wrong is, of course, the minute you put a fuel element in it goes haywire, but other than that it would be a very good reactor.

Well, seriously though if I can compare just for a moment our position today with what it was ten years ago, I think that we have every reason to feel very optimistic, at least from the standpoint of radiation damage and the future of fuel-element technology. Ten years ago we had very primitive hot laboratories. We had practically no reactors other than using, say, a production reactor or a reactor that was not designed for radiation damage testing. Today we have several very good reactors.

Ten years ago there were not many trained personnel in the field. Today we have hundreds of people. They are very cognizant of the problems that are facing us.

And so we extrapolate on this basis. I think that on the basis of physical facilities, the reactors available, and the personnel we can look forward to very great advance in the fuel-element technology.

The other trend that I should like to suggest will happen is that we will do more generalized study of potential fuel-element systems. That instead of picking a certain fuel element and looking at the specific temperature and the specific confirmation, uranium, etc., more studies of a general nature will be made. For instance, we will look at the effect of temperature of irradiation, and we will do this from the very low temperatures to the very high temperatures, so that for any one system it will be well understood. Well, there is a tendency toward

this. The brief comment that Mr. Beard made this afternoon on the KAPL program indicates that they are beginning to study dispersion fuels very systematically.

Well, the other thing that I should like to anticipate is that some day there will be a fuel-element conference and radiation damage will be the first thing on the program, not the last thing.

DISCUSSION OF TALKS BY PANEL MEMBERS

MR. J. DAVIDSON (ACF Industries): I should like to direct a question to Mr. Foote and to Mr. Ullmann. It is about fast reactors and the question has to do with the emphasis on the short processing. I wonder if I could go into a little of the reason that I have and see if you agree with me, and then maybe we can have an answer to my question.

Isn't it true that what we are really interested in is a relatively small amount of material in the outside cycle compared to the amount in the inside of the reactor? In other words, we don't want to have two cores in the system. We only want 1-1/4 or 1-1/2. Am I in turn so far? Then isn't it true also that it isn't really the time that it takes on the outside cycle, but the relative time between the inside cycle and the outside cycle? In other words we want a short period outside compared to the time inside, which means that if we have a short time inside we have to have a short time outside, but if we have a very long time inside we can have a pretty long time outside. The point is that if I have a high burn-up fuel element, not 2 or 4% but maybe 50%, I can cool it for 30 days or 60 days and there is no strain. This is not really a question, but I am interested in it.

MR. FOOTE: Yes, I think your argument is perfectly sound. Because of the high investment in fissionable material/machine we must work this material as hard as we possibly can. This means that you operate the machine at as high a specific power as you possibly can. At the present moment we do not believe that the fuels that we have available can operate to more than about 2% burn-up. With any reasonable specific power it means that the fuel turns over in about three months' time. In other words, the residence time in the reactor under presently envisioned burn-up limitations is relatively short.

This is the thing that basically puts a premium on a fast external turn-over. Now if we could go to 10% burn-up at the same specific power, leave the material in the reactor for something around a year, then of course, one could afford a somewhat longer external cycle. But I think you have put your finger on one of the basic problems, that it is the ratio between the on-stream time and the off-stream time that really makes the difference.

MR. DAVIDSON: At KAPL we did some work with on-site fuel which showed that we hoped to get one year's residence time in the reactor. In this case 30 days cooling did not look too bad and the aqueous process was used.

MR. ULLMANN: I would agree with everything that has been said. I feel that my point is correct, that something like 90 days cooling is equivalent in dollar value to 1% loss of material processed. I think that what Mr. Davidson is pointing out is that with a high burn-up fuel 1% loss of material processed may be a very few mills/kw hr of electricity.

MR. W. C. THURBER (Oak Ridge National Laboratory): A question for Mr. Losco. I wonder if you could possibly comment a little more specifically on your experience with the addition of burnable poison to UO_2 ceramics.

MR. LOSCO: There has not been a great deal of work on this subject. What little work has been done has been directed toward the incorporation of B-10 in UO_2 as a burnable poison. Actually, this was done in $\text{Al}_2\text{O}_3\text{-UO}_2$ and $\text{ZrO}_2\text{-UO}_2$ fuels. It was found that during the sintering cycle B-10 volatilized in hydrogen atmosphere so that we ended up with essentially no boron in the fuel. An attempt was made to get around this problem by incorporating the boron as a carbide or as a boride. I don't remember just which particular ones were employed, but they were not a solution to the problem. The boron was lost in those too.

We have not done any work beyond that at this point, but I feel that work of this nature demands attention because design engineers are asking for burnable poison in fuels. It might mean in the case of UO_2 that we would have to go to some other type of burnable poison, perhaps rare-earth oxide, but I think in this respect that physicists today at least feel that they do not know sufficient about these rare earths in so far as their burn-up characteristics are concerned to permit them to calculate their burn-up rates and self-shielding information. I think that the metallurgists have to supply this information to them before they can make use of these materials.

MR. W. L. ALBRECHT (Tennessee Valley Authority): As a non-metallurgist I would conclude that, for low enrichments at least, uranium is a four letter word and that uranium dioxide is not quite so dirty. I have heard also that increased thermal conductivity would be a very nice thing for uranium oxide.

My question is directed to anyone who might be able to answer it. How much promise is there for some additive, or some other means, to increase the thermal conductivity of uranium dioxide?

MR. LOSCO: I think there is certainly some potential in the way of improving the conductivity of the UO_2 . I was speaking to someone who indicated that he has been doing some work in the way of combining uranium dioxide with a small percentage, in the order perhaps of 3 or 4% of beryllium and as a result of this he has seen appreciable improvement in thermal conductivity as well as thermal shock resistance. I don't expect that there would be marked changes, that is, factors of ten improvements or something of that nature, but I think that any improvement in the thermal conductivity of UO_2 , which is quite low now, would be an advantage.

MR. WEBER: This is the point I was trying to make previously. I think that the distinction between bulk- UO_2 elements and the relatively low UO_2 -content dispersion elements will fade as the loadings approach each other using an inter-lacing metal, somewhat along the line of the uranium - magnesium work at Hanford. With 10 vol % of metal, we can perhaps make a factor of two improvements in conductivity. This appears to be a fruitful direction for future work.

MR. J. H. KITTEL (Argonne National Laboratory): Along these lines I might mention that we have sponsored some work at Armour Foundation, and $\text{ThO}_2\text{-UO}_2$ bodies were developed that contained metallic fibers to enhance the thermal conductivity. The metallic fibers that were used were niobium and molybdenum. We have a number of these under irradiation at the present time, and the thermal conductivity of these materials was measured by Armour, and it was found that by putting about 10 wt % molybdenum fibers into these ThO_2 and UO_2 bodies the thermal conductivity was increased by a factor of three and this factor of three developed at high temperatures where we think it will be most useful. So we are quite hopeful that these composite types of ceramic bodies will certainly enable us to go to higher heat fluxes from a given size of pellet.

MR. B. RUBIN (University of California Radiation Laboratory): I should like to address a question to Mr. Billington. I may be way off base here. It is my understanding that in the early days of irradiation damage there was quite a bit of consideration given to some of the theoretical interpretation of irradiation damage and that a lot of spiking was being done, but it has apparently turned out that these effects are not very important, at least in the metallic fuel elements up to the fluxes that we have gone to so far. But then the question comes up as the fluxes continue to go up to 10^{15} and even 10^{16} , and things of this sort are being talked about, does not this whole thing come up again and are we now going to have to worry about a very serious radiation effect on the metallic element too.

MR. BILLINGTON: With regard to thermal spiking, I think that you have to accept this as a reality in uranium alloys.

MR. RUBIN: What I was referring to was other metals.

MR. BILLINGTON: If you are talking about non-fissile metals, I think it is true that in general you can just wipe out thermal spiking as a concept usually because you can always find a temperature dependence in this radiation effect in non-fissile metal. In most cases we think this is a good enough argument that the thermal spiking is important. This does not apply, however, to non-metal or to uranium-metal materials.

There is just one point on non-fissile materials and the subject of creep which was of serious concern many years ago. A number of experiments done in-pile showed that there was an effect on in-pile. There was a decrease. On the other hand, if you were to take reactors up to fluxes of the order of 10^{15} or 10^{16} then you might introduce vacancies at such a rate that it would begin to equal the thermal equilibrium of vacancies at quite high temperatures in which case then you might start to see acceleration of the creep rate on the argument that vacancies contribute to dislocation time which is a part of the creep process at elevated temperatures.

MR. L. W. KATES (Sylvania-Corning Nuclear Corporation): I should like to make a couple of comments, if I may. The first is in connection with the thermal conduction of UO_2 . There is another approach which we are working on in some of the engineering-development work, and that is improved thermal conduction with minor additions of other ceramic materials. I am not enough of a ceramist to know exactly what happens when you do this, but apparently there is some promise here. We have not made the conduction measurements. We are just starting to do them now, so I cannot report any results.

But the second point is directed to Mr. Losco again. UO_2 seems to be a popular subject tonight. Assuming that there is such a thing as manufacturing and selling fuel elements which a lot of people talk about, but there does not seem to be too much of it going on, what kind of a warranty do you think you could give say on the PWR type of element? In other words, could you guarantee burn-up? Could you tell a customer that you would guarantee that the thing was going to go 15,000 Mw days or something like that? How do you feel about that?

MR. LOSCO: This my own opinion. I think essentially that when a customer says to a contractor, "Build a reactor to certain specifications," that is tantamount to guaranteeing that the reactor will meet those specifications or he will make it good if it does not. I believe that with the greater confidence and knowledge and tests that are coming about and the greater experience there should not be much question that you can warrant a given burn-up for a fuel element or a reactor.

MR. KATES: Would you hazard a guess as to what would be a typical or an acceptable warranty on a PWR element, for example a blanket element?

MR. LOSCO: Well, I believe that seed elements designed for 3,000 equivalent full-power hours are presently expected to operate for as much as 5,000 or 6,000 full-power hours, and the blanket elements are expected to go at least two seed lives. There is indication that they will probably be good for three seed lives. I see no reason for doubting that they will make this.

MR. KATES: Could you translate them? Mr. Kittel has asked me to translate them in to Mw days/ton.

MR. LOSCO: Well, in the case of blanket elements this would amount to something of the order of 10,000 Mw days/ton.

MR. T. DREIER (Knolls Atomic Power Laboratory): So far there haven't been any fuel contracts with a burn-up guarantee in them, have there?

MR. WEBER: Warranty of a new type of fuel element under new operating conditions does not appear feasible without getting reactor-core operating experience. After all, each core usually contains a large number of fuel elements and core-operating experience is needed to give a reasonable statistical basis for warranting performance.

The second point is that the warranty will depend on both the intrinsic ability of the element to stand up under radiation damage, and independently on the ability of the manufacturer to build and inspect in quality, a problem which I know you are quite familiar with. Both are important, intrinsic stability and built-in quality, the latter raising the question how high a quality level is needed and how much should the consumer specify and pay for.

MR. DAVIDSON: Well, just to answer that point, you would be amazed or maybe you are aware of the number of people who are in the process of designing reactors, for example, foreign power stations and the like, with no operating experience. As a matter of fact, the darned things have not been designed in many cases. There are just some line drawings of a fuel element or something else that he is supposed to manufacture and is interested in having a warranty on burn-up.

It seems to me that perhaps one of the types of data that we ought to be talking about for the future is this kind of data that can be translated into something that is commercial.

This really gets to the third point that I was going to remark on here, too. That is I think that Mr. Cunningham and Mr. Ullmann are to be congratulated for bringing the matter of economics into the discussion here, because I think most of you will agree with me that the matter of economics is the factor that is holding up the nuclear energy field today. I wonder if there should not be more thought given to the whole matter of economics in some of the future work. I mean that we have heard a lot of discussion of the technical advancements that should be made and I certainly am sympathetic with them, but what about some work on the economics? Can I get some comments from the group?

CHAIRMAN SIMMONS: I think that the whole purpose of these meetings is to attempt to understand the economics. I don't believe that it is a very well defined field, at least not in the nuclear business. If anybody has any ideas on the economics, we will be glad to hear them.

MR. W. S. PELLINI (Naval Research Laboratory): I am wondering what has happened to the fuel element for liquid metals for heat-transfer units and if anyone on the panel would say a word or two regarding its future.

MR. FOOTE: Well, we are a little uncertain as to whether the question relates to liquid-metal fuels or to solid fuels cooled with liquid metals?

MR. PELLINI: Cooled with.

MR. FOOTE: It is certainly true that the fast breeders are set up with liquid-metal cooling, and the favorite coolant, as a matter of fact the only coolant considered, is liquid sodium. In these fast reactors, where we must get a very high specific power, we must use extremely efficient coolant, as efficient a coolant as one could imagine, and therefore, the whole fast reactor program -- the whole plumbing system is based upon sodium technology. This is going ahead and I hope in a satisfactory manner.

There is one feature, of course, of liquid sodium vs metallic fuels worth noting and that is that we have very little in the way of a corrosion problem. That is one of the reasons we stick to this combination of liquid-metal coolants and metallic fuels. Simply because we do feel that the fuels and the coolants are compatible. It is also true, of course, that uranium oxide is compatible with sodium, so we have the possibility of using a compatible liquid sodium oxide system also.

I think this is about all we need to say, that liquid sodium systems are used and will be used as an integral and, I believe, necessary part of the fast reactor program.

CHAIRMAN SIMMONS: Dr. Kaufmann, would you like to comment on this one, too?

MR. KAUFMANN: I am afraid to comment on that question, but something else has been running through my mind. That is, it seems to me that we ought to talk about a fuel element, not about a core material or a cladding material separately. In my opinion the cladding can be a very important part of the fuel element. It can determine whether the fuel element will function or not. For example, all this concern that we have about whether the corrosion resistance of uranium can be made as good as that of stainless steel is based only on the fear that the corrosive environment is going to get through the cladding. If people had the courage to believe that cladding can be made entirely reliable and foolproof, one could forget about whether the fuel is corrosive resistant or not.

Likewise all this concern about swelling and distortion. It is entirely conceivable that the cladding can be made strong enough to restrict the swelling and growth, and in this case again the concern about radiation damage and what happens to the fuel becomes secondary.

There should be more concern then about what the cladding can do to make the fuel element successful rather than worrying only about what the fuel wants to do and acting as though the cladding is merely an isolation medium to separate the fuel from the rest of the world.

MR. B. W. DUNNINGTON (Battelle Memorial Institute): I should like to ask a question with regard to the use of oxide fuel at very high burn-up. I believe that Mr. Foote mentioned that the oxide element appeared attractive in the fast breeder except possibly from the standpoint of the recovery afterward; and I commented from the floor that there have been some studies by KAPL which indicated something very similar. Recently all of us, I believe, think and talk about oxide fuel burn-up on the assumption that it works very well except that fission products are released by diffusion during cycling.

Recently there was a report published by KAPL which indicates, at least, that oxide fuel, UO_2 , may exhibit swelling just like a metal at high temperature and high burn-up.

MR. WEBER: Since I am from KAPL, I would like to comment and to also mention that I will discuss this question. One answer is -- come around tomorrow in the opening session tomorrow afternoon.

We have irradiated UO_2 to exposure levels equivalent to about 500,000 Mw days/ton of UO_2 , both as dispersions in stainless steel and as 65%

dense UO_2 powder in annularly loaded fuel elements. The UO_2 powder sinters into a porous solid. Depending on the fuel temperature, 10 to 50% of the gas will diffuse out and is present in the pores in the UO_2 . The material acts in a plastic manner and with room to accommodate the gases.

Some distributed internal porosity also developed in the UO_2 dispersions. If you think back to Mr. Beard's comment this afternoon, the test data indicate that more gas escapes at low burn-up with increasing porosity. I believe, however, that there may be a good chance that at higher burn-ups porosity is actually advantageous. The advantages probably offset the disadvantage of increased gas diffusion.

As far as swelling is concerned, I think that one of the intrinsic properties of the UO_2 is that it is much more able to accommodate the fission products. Uranium and some of the alloys expand, say 4% in volume at 1% burn-up at a low temperature. Based on KAPL data, UO_2 expands as little as 1/2% for 1% burn-up. This may be taken as a lower limit, and 0.7% used as an upper limit. So this material basically does not expand as much as alloys. Also it is a high melting material. It is not as sensitive to temperature, and a much higher temperature may be reached before diffusion of the gas atoms leads to catastrophic swelling. The best data indicate that UO_2 may be heated to about 1000°C before appreciable fission gas diffusion occurs. At most, operating temperatures with a little void present, as has been the case in our dispersion elements (~5%). The elements were very stable dimensionally.

I mentioned before the question of stringing and porosity and fragmentation. You may actually want a little porosity in the material to give better dimensional stability than you get with really high density material, although this remains to be proven.

MR. LOSCO: I think so far as swelling or release of fission gas from bulk UO_2 is concerned, the data indicates that you will get release of fission gases, but the release, as Mr. Weber indicated, is not significant below about 1000°C. Of course, if your temperatures are up so high that you get melting, I believe that essentially you release all of the fission gas. Our experience has been that release of fission gas is related directly to the density and the opened and closed porosity in the bulk fuel. Above about 95% theoretical density you have zero open porosity, and the release of fission gas under these conditions appears to be mainly diffusion controlled. Above 1000°C it will occur but at a rate that is still low and that increases gradually with temperature.

I do not know whether I have completely answered the question, but in so far as swelling is concerned, I might say that the data that has been obtained, at least by WAPD, has not been on the bulk UO_2 itself but on the fuel element clad with Zircaloy. Measurements were taken really on the change of dimensions of the completed elements, and under these conditions even at high burn-ups, as high as we have gone, which have been in the order of 25,000 Mw days/ton, there has been no noticeable increase in volume.

Along these same lines, there is some very recent data which has been obtained on plate-type oxide-bearing fuel elements which have been burned up in capsules to the extent of about 30,000 Mw days/ton, and here if there were appreciable gas evolution we would expect very severe cladding deflections. Actually no cladding deflection has occurred. So here again, while the temperature has been relatively low in this case, it appears that fission gas released per se under the conditions that you would normally consider using for plate-type elements, at least, is not as serious a problem as it would at first appear.