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CRITIQUE - NPR FUEL

by

S. H. Bush

This Document consists of

52 Pages No. 7 of

7 Copies Series A

INTRODUCTION

The current status of the NPR fuel fabrication and testing program has been reviewed; suggestions have been made concerning areas requiring more information.

This report is divided into a series of sections, each of which is a complete entity so a specific problem area can be considered apart from the entire NPR Fuel process. The final section incorporating conclusions and recommendations reaffirms the factors of major importance as well as evaluating those areas which are interrelated. Process problems have been considered from the aspect of manufacturing, not engineering.

Major sections considered include:

1. Components
2. Billet assembly
3. Extrusion
4. Quality of Coextruded Product
5. Cleaning and Copper Removal
6. Closures
7. Supports
8. Heat Treatment
9. Straightening
10. Autoclaving
11. Testing and Inspection
12. Irradiation
13. Future Development
14. Conclusions and Recommendations
15. Appendices

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HEP FUEL COMPONENTS - STATUS AND PROBLEMS

The three major components for the fuel are: (1) uranium core, (2) Zircaloy sheath, and (3) copper jacket. In addition, there are items of Zircaloy hardware for end caps, fittings, supports, etc., as well as steel supports.

Uranium Billets

Two possible paths in the fabrication of uranium billets are the casting of ingots 11 inches in diameter and the use of Dingo. The second appears technically feasible but is fraught with criticality problems in the handling of enriched green salt. A major decision would be required by the AEC concerning the reduction and casting of quantities of material where a criticality incident is possible, and in the authorization of another dingo line.

The problem with 11-inch ingots is one of inadequate experience. The number of castings produced to date is very limited; there is no production experience; metal quality specifications have not been established; the coring concept to obtain high quality metal has not been examined; the incidence of ingot cracking during triple beta heat treatment is still unknown; and there is little experience in alloying and in degree of homogeneity of alloys. These are some of the problem areas. Essentially, the problem is one of lack of experience with this size casting, and there is very limited time to obtain the necessary information. Of major importance is the establishment of realistic metal quality specifications as soon as possible.

The study of alloys should be carefully evaluated to establish the ultimate value of this approach. There is little to justify such work other than in grain refinement and possible improvement in fabrication and quality of extruded product. Insofar as corrosion resistance or improved reactor performance are concerned, the extensive work at several sites carried out

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with alpha alloys has not revealed much improvement in these areas. Alloying results in limited or no improvement insofar as in-reactor corrosion, swelling, or dimensional stability are concerned. Under no circumstances should the alloy work be permitted to interfere with the studies directly pertaining to the billet studies for the first charge because there is so much to do and so little time remaining.

An evaluation of a limited number of billets indicates the triple beta-heat-treat route followed by alpha extrusion is preferable to an initial alpha extrusion of the cast ingot with final triple beta heat treatment. Potentially, a more fruitful route would be (1) alpha extrusion, (2) triple beta heat treatment, and (3) alpha extrusion. This would require a 13-16 inch ingot, and substantial development work, and does not appear to be justified unless the 11-inch ingot proves unsatisfactory. If it becomes necessary to go to the larger ingot, the dingot route or alloy additions should be reconsidered.

Some work should be done on obtaining a more satisfactory structure in the area adjacent to the hole through the uranium billet. Two possible approaches are alpha piercing of the casting and casting the billets with a core to produce the hole. The alpha piercing introduces severe folds and laps which can yield a poor interface unless the billet is overbored to remove the doubtful zone. Some work should be done in this area, but there is no assurance that alpha piercing will produce a desirable structure.

Essentially, no development has been done on casting cored billets. Therefore, it is not possible to evaluate the process. However, based on success in similar fields, it is quite probable that it will work with uranium.

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A major problem which requires immediate action and close integration with NLO or NEV is in the establishment of quality specifications for the current material. The work at HAPU in the evaluation of metal quality has been limited or non-existent, and there is no reason to believe that it has been much better at NLO. Chemical analyses are not enough. A carefully designed sampling program is required, incorporating non-destructive testing and metallography. One possibility is the extrusion of 11-inch castings to 7-inch billets followed by rolling to fabricate I&E cores. These could be evaluated at NLO using stringer counts to establish metal quality.

Zircaloy-Components

There are several potential problems in the use of Zircaloy-2. These include (1) product variability, (2) lack of corrosion resistance, (3) hydriding and embrittlement, (4) water penetration through stringers in the end caps, (5) sensitivity to contamination during fuel processing, and (6) high costs with concomitant necessity to minimize scrap losses.

Variability in the composition and mechanical properties can be due to variations in sponge quality or to the process used in fabrication. If the hot strength changes much, the quality of the extruded product may also change. Variations in hot strength will appear as rough interfaces, striated surfaces, localized thinning of the cladding or even as tearing of the cladding. To date, this has not been experienced at HAPU, but it has been experienced elsewhere. This can be minimized by a suitable specification and adequate quality control.

Corrosion resistance is a function of composition and of impurity levels. This should not be a problem with good product control, but it should be pointed out that several EMI constructions failed to pass the autoclave tests.

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In some cases, the contamination could be traced to poor quality Zircaloy-2 components or to a leaking billet. In other cases, the failure to pass the corrosion test can be ascribed to lack of control during the processing of the fuel elements. Potentially, this is a very serious problem.

Severe hydriding and resultant embrittlement of the Zircaloy is a definite problem. Zircaloy is quite susceptible to hydrogen pickup and to selective distribution of hydride in the cold zone which will be the outermost layer of the cladding. This hydride will act as a stress riser and site of failure. Major sources of hydrogen are autoclaving and irradiation, but the components do have a finite amount in the as-received form. If this problem becomes serious, it may be necessary to shift to an alloy such as Zircaloy-4. The hydriding experience with constructed Zircaloy-2 clad fuel elements has not been bad, but the numbers tested are still quite limited and hydriding must be considered to be a definite possibility.

Little or no attention has been devoted to the quality of the end caps used in the fuel tubes. The least expensive way of fabricating such caps might be to section them from pieces of heavy walled tubing of the requisite size, using a parting tool. A potential problem here is the existence of stringers in the metal, serving as a direct path for water to the core. This can be minimized by forge or press upsetting the caps to break up the stringer path. An alternate route worth investigating is to cast the rings and upset them to final dimensions. With recycle of scrap, this might prove to be an inexpensive method for producing end hardware.

Great care must be exercised in handling Zircaloy-2 to prevent contamination and ultimate rejection in the autoclave step. The impurity limits

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should be established as soon as possible to permit an adequate specification to be prepared for manufacturing.

The cost of Zircaloy-2 is a major factor so every care should be taken to minimize losses in all steps of the operation.

No major problems are visualized in the Zircaloy strip stock used for supports.

Copper Jacket

The appropriate sizes of copper tubing of commercial grade, plus copper plate for end hardware, can be readily obtained. Providing reasonable care is observed in controlling metal quality, no problems should occur. The major source of difficulty would be impure copper containing inclusions which would mar the Zircaloy by the hard intermetallic inclusion penetrating into the softer cladding.

Steel Supports

Too little is known concerning the optimum composition of the steel outer tube supports. Assuming mechanical keying of the steel to Zircaloy, no problems can be visualized. Roll bonding and the strength of the bond after irradiation are two major unknowns which must be investigated if this approach appears to be justified.

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(2) BILLET ASSEMBLY

The benefits of good quality control in the billet components, uranium, zirconium, and copper can be negated by careless assembly procedures or lack of control in the various assembly steps.

Sufficient information is available to permit a production assembly process to be outlined, but there is too little information relating to the handling of large quantities of components in the cleaning, assembling, and evacuation operations.

Inadequate cleaning or leaky billets will result in poor bond quality, dirty interfaces, and, possibly, in localized cladding defects. There is definite evidence of a lack of pickling solution quality control based on an evaluation of the bond characteristics of FPD extrusions in the 30-40 series. This can be attributed to solution depletion and dirty baths due to cation contamination. The answer is obvious: better quality control. Substantial work is required to fix the permissible limits on cleaning solutions as functions of component throughput and cation contamination. Until this is done, the production cleaning process cannot be established.

A gettering disk of uranium or zirconium in the billet assembly should be a valuable addition to minimize component contamination during heating and to improve bond quality. The degree of improvement resulting from such a getter disk can be fixed by a series of controlled tests.

Tungsten inert gas welding of the copper sheath poses no production problems, providing there is adequate edge preparation and sufficient care is taken to minimize blowholes and inclusions. This step should lend itself well to the use of a relatively automated and semi-skilled labor operation.

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Billet evacuation should pose no problems, providing the components are adequately cleaned and then dried. Tests should be made for leaky billets. The added cost of such testing is more than offset by the recovery of such leaky billets. Once they are extruded, the fuel is scrap if there is inadequate bonding.

Reasonable care in the assembly operations and good quality control in all stages should minimize losses at this stage.

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(3) FUEL EXTRUSION

The principal problem in the optimization of the extrusion operation is lack of experience. Substantial progress has been made, but it will be necessary to extrude many more billets to obtain answers to several of the problems. Specifically, better results must be achieved in smoothness and freedom from striations in the outer cladding, elimination of local cladding defects such as indentations, chevron-like defects, deep scratches, patches of roughness and local clad thinning.

A major problem is variability in cladding thickness over the entire fuel length; however, there appears to be some solution to this in control of operations in uranium billet fabrication.

Bond quality can be related to reduction ratio as well as to component cleanliness. Results are too limited to permit an optimization of reduction ratios. It is desirable to go to the maximum reduction ratio commensurate with press limitations, billet size, and yields.

The billet heating operation requires a great deal more work. Minimum heating times have not been fixed, nor has the effect been determined of an inert heat treating atmosphere on the copper flow characteristics. There is limited evidence to indicate that minimization of the oxide on the copper may improve the surface finish of the Zircaloy-2 sheath.

There is a great deal more work required before a process is available, covering all stages of the charging and extrusion operation, which can be taken over by the manufacturing group. This is directly attributable to lack of experience and too few extrusions.

Potential problem areas which have not caused difficulty include eccentricity, lack of concentricity, and undue bowing. Unless there is a marked

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change in extrusion behavior, it appears that the press alignment is good and control during the extrusion step is sufficient to essentially eliminate these problem areas.

A major effort must be made in improvement of surface quality. To date, only one extrusion has been made where the external surface could be non-destructively tested. This is a formidable problem since bond quality, cladding thickness, and uranium quality depend on accurate non-destructive test measurements. The only answer to this problem appears to be a painstaking investigation of such variables as copper quality, billet heating, uranium quality and prior fabrication history, zirconium fabrication history, and extrusion conditions coupled with careful destruction and non-destructive testing. Work in the correlating of destructive and non-destructive test results has been grossly inadequate and is a major area requiring additional work.

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(5) QUALITY OF COEXTRUDED FUEL

An examination of Appendices A and B, dealing with the quality of fuel coextruded by NMI and FPD, reveals that the problems are numerous and of considerable magnitude. A great deal of work is required to improve the processes before a coextruded product of reactor quality is achieved.

The material in Appendix A, dealing with NMI coextrusions, reveals how much work must be done. Their experience in the coextrusion of reactor fuels exceeds all other sites, yet the fuel yields, based on what appear to be realistic criteria, are quite low. Many factors are responsible for these low yields. It is important to realize that the yield is an inverse function of the amount and care of the examination procedure. The more careful and the more extensive the tests, the lower the yield. Typical problem areas include: (1) poor quality uranium, (2) internal cladding defects, (3) copper breakthrough during coextrusion with severe galling and ellipticity, (4) unduly thin cladding, (5) excessive variations in cladding thickness, (6) rough, uneven and striated surfaces, (7) inclusions imbedded in the cladding, (8) pits, nounds, chevron defects, and gouges in the cladding, (9) excessive "dirt" at the core-clad interface, (10) segregation and pronounced banding in the U-2 w/o Zr core material, (11) unbonded or poorly bonded fuel, and (12) Zircaloy-2 fails to pass corrosion test.

The preceding is a rather imposing list resulting from inferior quality components, inadequate control during assembly, and non-optimum extrusion conditions.

It is obvious that control of all the factors is necessary. Even so, one problem is paramount. The external surfaces of the fuel are quite uneven

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and the amount of material produced in any one size that has a reasonable surface is so limited and occurs so infrequently that no answers are available as to how a good product can be achieved.

An evaluation of the data in Appendix B, pertaining to FPD coextrusions, is made up of equal parts inadequate information and information indicating the fuel is of non-reactor quality with a slight leavening of hopeful signs that the situation will improve.

Problem areas are: (1) rough non-uniform surfaces which cannot be non-destructively tested, (2) bond strengths ranging from good to no bond at all, (3) excessive variations in the thickness of cladding, (4) large grains adjacent to the core-clad interface, and (5) cladding defects including dimples, bumps, chevron marks, scratches, and strias.

A major deficiency is the lack of data pertaining to the quality of the core. Very little information is available so this area, which resulted in an extensive rejection of BMI stock, cannot be assessed. Still another lack is a good characterization of the surface quality of the cladding and too little information concerning corrosion resistance of the cladding. It is imperative that data be collected in the preceding areas so that the FPD coextrusions can be adequately analyzed. A major effort has been made to improve the uniformity of the internal cladding and to minimize cladding thickness variations on both inner and outer cladding. This is noteworthy, but the problems of cladding defects and core quality are equally important.

An extensive sampling program incorporating existing tests and adding metallographic evaluation of the core and careful visual examination of the internal and external surfaces must be coupled with careful nondestructive

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tests. These tests will permit the quality of the constructions to be determined and indicate problem areas requiring more attention.

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(5) CLEANING AND COPPER REMOVAL

Experience is available pertaining to the removal of copper after construction, vapor blasting, and chemical cleaning of the Zircaloy-2 surfaces. Unfortunately, experience is limited to small batches, and the critical factors in a production operation still must be evaluated. These factors include solution concentration, optimum bath temperature, depletion rate, effects of bath contamination, cleaning rates for large batches, and variation in cleaning rate from piece to piece in any one batch. Little or no work has been done to fix these factors.

A specific problem requiring an answer is the incomplete removal of imbedded copper due to surface passivation. A foolproof method for removal must be developed since residual copper will result in rejection during the autoclave operation. It may be necessary to use more than one bath to remove the copper.

Another problem is the optimization of the vapor blasting operation. As the process exists now, it is very dependent on operator skill. Too long a time results in an excessive removal of cladding; too short a time leaves a residuum of oxide which is not removed in a nitric-hydrofluoric pickle and results in rejection during the autoclave operation. Grit concentration and particle size must be fixed. This one step may result in a high percentage of the fuel recycled to scrap or for recovery. This does not mean the fuel cannot be recovered, but it will influence costs markedly.

Too little has been done in establishing a production process for the pickling of the Zircaloy-2 surfaces. Positioning in the bath, agitation,

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temperature, and bath compositions must be fixed before a specification can be written.

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(6) CLOSURES

Only two types of fuel element closures can be considered seriously for the first NFR charge. These are the brazed and welded closure, and, if irradiation performance does not warrant the charging of brazed closures, the unbrazed welded closure can be used.

Specific operations required in the brazed closure include: (1) chemical milling, possibly in conjunction with mechanical milling to remove the uranium from the ends, (2) chemical cleaning after milling, (3) conducting the brazing operation, (4) machining to remove excess braze metal, and (5) welding the end cap. In the case of the unbrazed end closure, the brazing step and the subsequent machining step are eliminated.

Chemical Milling

Some work has been completed on the chemical milling of unalloyed uranium with considerable variation in the results. The type and degree of chemical attack changes with the depth of metal removed. For example, a process satisfactory for one-tenth of an inch may be totally unacceptable for one-quarter inch. The newer solutions being used (sulfuric-nitric-copper sulfate) perform more satisfactorily than hydrochloric, but nothing is known concerning the behavior in batches approaching the numbers to be handled in the manufacturing process. Until the information is available, it must be assumed that the operation is manual rather than automatic with the rate of attack of every tube having to be checked regularly.

The following problem areas still require answers: (1) rate and uniformity of attack from tube to tube in a batch, (2) depletion rate on solutions and method of solution makeup, (3) degree of control required on bath temperature and agitation, (4) uniformity of uranium surface after milling

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(this will probably be concave with some possibility of bond line attack, and mechanical milling or controlled removal by electrolytic polishing will be required), and (5) a major problem area will be acid milling of alloys due to variable rates of attack resulting from inhomogeneities in the core. Very little work has been done here.

The whole area of acid milling requires a great deal of work both on and off site to establish the necessary parameters. Major limitations are the quantity of fuel material available and the lack of work in batch lots.

The cleaning step following acid milling should not be a problem area other than in the handling of the uranium-zirconium rich solutions after substantial cleaning.

Brazing

The brazing area represents the process where the least is known and where many problems must be solved as soon as possible. Principal problem areas are: (1) too few brases have been made, and these vary in quality, (2) the brase material poses a health hazard due to the beryllium content, and this will result in a more costly process, (3) the brase operation adversely affects the properties of the Zircaloy-2 cladding and of the uranium core due to the high temperatures required, (4) the stability of the uranium-brase bond has not been established; a failure in this area might be worse than an unbonded fuel element, and (5) the lack of irradiation data is a major problem.

Substantially more work must be done to establish the uniformity of the brase process from piece to piece. The eutectic Zr-Be alloy is difficult to fabricate at eutectic composition, which will adversely influence the brazing process control.

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The requirements for the brase should be carefully reviewed. Outstanding corrosion resistance does not appear to be a controlling factor, and relaxation here might permit an alloy free from beryllium to be used; this simplifying the process. In any event, some attention should be paid to the ternary Zr-Fe-Be alloy since its lower melting point is a definite advantage which might offset the increased brittleness. A decrease in brase temperature from 1035 to 950 C would be worthwhile.

The major deficiency is the lack of any irradiation experience on brazed closures. The obtaining of such data should be a top priority item. The behavior of the brase bond cannot be predicted and it is conceivable that a broken bond will represent the worst possible condition due to the existing thermal incompatibility between uranium and zirconium. It is quite possible that a failed bond would be followed by a circumferential cap-core split, exposing the uranium to the coolant.

Another major problem area is in the development of inspection techniques to detect unbonded brase-uranium interfaces. Very little is being done here in the development of such tests. Possible approaches should include: (1) drop tests followed by non-destructive and destructive testing, (2) hot end tests using infra-red scanners to detect non-bonded areas, and (3) thermal cycling to establish bond integrity.

Some consideration should be given to initiation of offsite contracts to develop alternate brase alloys. This should be done only after the requirements for such a brase are carefully considered.

An analysis of the brase process itself indicates some potential problem areas. Included are: (1) voids below the cap, (2) tipping of the cap on pressurizing, (3) excessive amounts of brase material required with subsequent loss on machining, (4) requirement for careful, pre-brase cleaning to eliminate the hydride layer, and (5) the bulge developed in the diameter on

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bracing.

The effect of brazing on the zirconium is to form the beta structure which differs markedly in properties from the alpha, having less ductility and some tendency for diffusion of hydrogen to the grain boundaries. These changes are accompanied by a substantial, though not necessarily critical, loss of bond strength.

The development of columnar grains in the uranium is also a problem until it is certain that beta heat treatment will destroy this structure. There are no data available on this area as yet.

Welding

Some satisfactory tungsten inert gas welds have been made on brazed pieces, but several of the welds have not been of reactor quality. The problem is principally one of lack of experience. More work should be done so that the process can be proven to be reproducible. This is not the case at present. The effect of lack of welding control will be seen in the rejects from autoclaving.

The unbonded closure still requires welding development. Too little has been done to fix this process as a backup.

Alternate Closure Processes

An alternate closure process, hot heading, has been tested in-reactor with good results, but it is 1-3 years away insofar as extensive pilot plant or production testing is concerned. A major problem will be the substantial capital investment required in heading and welding equipment, but the process should have a higher throughput and require less labor than is the case for the brazed closure. Continued development work in this process is advised.

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Several other closure processes have been investigated, including seam welding, ultrasonic welding of zirconium discs to uranium, stitching a cap to uranium with an electron beam, pressure bonding a powder metallurgy cap, and explosive bonding. None of these appear to be sufficiently developed to replace brazing, and work on these currently underway should be carefully reviewed to be sure it does not interfere with the required work on the brazed closure.

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(7) SUPPORT DEVELOPMENT

Problems in support selection involve: (1) attachment, (2) design to minimize pressure drop, (3) selection of compatible materials to minimize tube-support galling during charging and discharging. None of the problems represent a major area of work, but none of them has been solved either. Timing is the most critical item if the projected goals are to be met.

Attachment can best be done by resistance spot welding based on current available information. Ultrasonic welding should also work, but even less work has been done with this process than with resistance welding. Inspection will probably be difficult since only a small percentage of the contact area is bonded. Use of a weld analyzer may prove to be the best solution. Corrosion under the support does not appear to be a problem based on autoclave, ex-reactor and in-reactor loop tests.

Support design to minimize pressure drop is relatively advanced and answers should be available. The choice of having supports fabricated off-site or onsite must be made. Presumably, stamping them from sheet is the logical solution and a small fast-acting press could deliver the requisite number. Die design and die costs will prove to be items of concern, and little has been done in these areas.

The most critical problem is the selection of a support attachment of a metal that will not unduly mar or gall the Zircaloy-2 process tube. The final selection must be made soon, and sufficient choices are available in the low carbon steels to permit this selection. The method of attachment to the Zircaloy-2 support or to the cladding is not settled. Any attempt to weld to the cladding is fraught with problems of brittle bonds, alloying of

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the cladding and probable loss in corrosion resistance. A logical solution is to mechanically attach to a specially shaped Zircaloy-2 support. Work is progressing in these areas and answers should be available. The difficulties may arise in converting a concept to a workable manufacturing process; however, no real difficulties are visualized. Time and some hard work should give the answers in a few months.

Loss of ductility during irradiation should not be a problem with the Zircaloy-2. There will probably be a substantial loss in impact strength of the steel supports but the conditions of loading should not be so severe as to result in a major discharge problem, providing the support contains no severe stress risers and the discharge load is nominal.

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(8) BETA HEAT TREATMENT

Beta heat treatment is necessary to randomize the oriented uranium structure so that the fuel element will be dimensionally stable during irradiation. The heat treatment must be done after the brase closure operation to break up the columnar grains produced during the brazing. So long as the brase closure is used, the heat treatment must follow closure. This would also be true for a hot headed fuel element. Only in the case of an unbonded fuel element can heat treatment precede closure.

Two possible processes for heat treatment are induction and salt bath. Induction heat treatment is not preferred now because of inadequate experience, lack of knowledge on the control of the heat treatment parameters, and high capital costs. Possible advantages are elimination of salt cleaning step and better control during the cooling stage. The induction process has sufficient promise to warrant some degree of backup effort but not to the extent that the collection of necessary information in the salt bath heating is neglected.

There is considerable experience in salt bath heat treating of uranium and coextruded Mircaloy-2 clad uranium; however, very little of the data on the coextruded fuel are under production conditions. The preferred orientation information on the fuel is adequate with the exception of the behavior of the area immediately adjacent to the brase closure. This area consists of large columnar grains extending along the axis of the tube and produced in the brazing process. More work is required to establish if the beta heat treatment breaks up this structure effectively. This work is underway and answers should be available in one to two months.

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Another problem area requiring an answer soon is the type of quench to be used. Oil will work, but requires two additional cleaning steps, one to remove the oil, the other to rinse the element. Air is a possibility, but a non-uniform retained salt film may lead to uncontrolled cooling, residual stresses, and some preferred orientation. Tests are required to determine the behavior. Quenching in a controlled temperature water bath is another possibility. The problem here may be one of process control since nominal variations in water temperature change the quench rate markedly.

The ultimate test of the beta heat treatment must be irradiation. This should take place as soon as possible.

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(9) POST HEAT TREATMENT STRAIGHTENING

Very limited data are available concerning the degree of bowing introduced in the fuel element due to fabrication, particularly during heat treatment. Even less information has been collected on the straightening of bowed fuel elements. The third area requiring more information is in the bowing of fuel elements in-reactor due to asymmetrical residual stresses or to localized areas of preferred orientation.

Optimistically, the bow developed in the relatively heavy NPH outer and inner tube sections in lengths of two feet or less will be sufficiently small so that no correction will be required. There is some possibility this will be the case, but much more work is required to confirm this.

Work must be conducted in the straightening of the fuel on the assumption that such straightening will be required. It is advisable to warm straighten to minimize residual stresses and multi-line support appears the most desirable. This can best be achieved with a three roll straightener supporting the fuel along its entire length. Care must be taken to prevent surface contamination of the Zircaloy-2 by the steel rolls since such contamination may cause pitting or excessive corrosion. One solution may be an alloyed zirconium used as the roll. If the number of fuel elements with excessive bowing is nominal, press forming should prove satisfactory and relatively inexpensive.

The only solution to the bowing problem in-reactor is to collect sufficient information prior to irradiation and to irradiate as many fuel elements as possible. Bowing, if it occurs, may represent a major limitation in permissible burnup since a bowed fuel element will fail by hot spot attack after undue restriction of the annulus.

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(10) AUTOCLAVING

Optimization of the autoclaving process is well advanced and has adequate documentation. Continuing tests should resolve areas still in doubt. The HLO program has been quite timely and should assist in the development of an excellent autoclave installation. Problem areas remaining include: (1) establishing the desirability of a combined water-steam cycle to obtain a better quality oxide layer; (2) necessity for hydrogen alarm systems on the autoclaves to minimize oxidation of fuel after failure; and (3) desirability of training personnel in this area as soon as possible. R. L. Dillon has volunteered to set up an arrangement where personnel could work with persons in his group to learn as much as possible about the autoclaving process. This appears worthy of consideration.

Alternate approaches to steam autoclaving include air oxidation of the fuel surface. This is a potentially hazardous operation in that nitrogen contamination may result in accelerated failure in-reactor, it will not serve to screen out fuel elements containing pinhole size defects, and there is no assurance that the defect structure of the oxide film produced in air is similar to that in steam or water. The difference may be sufficiently marked to preclude such a process. In any event, once the autoclaves are installed, the incentive to go to another process requiring substantial capital investment is essentially eliminated.

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(11) TESTING AND INSPECTION

A production operation devoted to the fabrication of precision components must depend to a marked degree on non-destructive testing. This is true of the HPR fuel elements. Work is underway to establish adequate tests, but too little has been done. There is inadequate collaboration between fabrication and testing groups, and fuel quality is not good enough to permit a meaningful inspection.

There must be greater rapport between fabrication and testing components if equipment is to be set up and tested. Standards are not available and existing equipment consistently accepts fuel which is known to contain gross defects based on destructive tests and visual examination.

This problem is one of testing enough material of known history so that the equipment can be calibrated.

A major effort is required to collect all relevant information on the constructions as soon after they fabricated as is possible. Key areas include: (1) careful visual examination of the cladding before and after removal of the copper (very little of this has been done to date), (2) a destructive examination of cladding variability, interface quality, and bond strength similar to tests now conducted, (3) a selected metallographic study of transverse and longitudinal sections to establish uranium quality and cladding quality, and (4) one hundred percent non-destructive testing to permit correlation with the other tests.

The lack of information in this field represents one of the major deficiencies in the current program.

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(12) STATUS OF IRRADIATION PROGRAM

A major limitation in the optimization of the NFR fuel element is the lack of meaningful irradiation data. Principal deficiencies include: (1) no irradiations where the fuel operates under proposed NFR conditions, (2) no irradiations of the proposed NFR size fuel element, (3) no irradiations of fuel fabricated with the proposed process of brass closure, (4) lack of tests to goal exposure under irradiation either prototypical or non-prototypical, and (5) little hope that there will be sufficient irradiation information prior to NFR startup to permit adequate evaluation of the proposed fuel fabrication concept.

The factors enumerated in the preceding paragraph represent the greatest single lack in the development of a satisfactory fuel element. This lack, and the efforts to alleviate it by crash programs of irradiation testing, mask another potentially serious deficiency in that no work is under way on Phase II or Phase III fuels. This problem is not severe now, but it will become so in the next year due to the very long lead times necessary in developing a new fuel concept. Work initiated now will not have a payoff in less than 3-5 years so every month lost now means a serious delay in the future.

The status of relevant irradiations, including tests at HAPC, ETR-MTR, ETR, VETR, and SFP, is presented in Appendices C, D, and E. Only general conclusions are given here.

Examination of the table in Appendix C dealing with HAPC irradiations in the ETR loops reveals how limited the information is pertaining to tubular geometries. Only four tests have reached exposures of greater than

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1000 MW/T. Of ten production tests, seven have been terminated by failures in the fuel, two are still operating, and one reached a goal exposure of 250 MW/T. Failures have included fuel with definite swelling, cracking in the uranium, cladding splits by localized yielding, and cap circumferential splits, severe hot spot attack in conjunction with crud buildup, and severe warp coupled with hot spot attack. Perhaps the most serious form of failure is caused by localized cladding thinning. Hopefully, this may be corrected through better control of the construction process and by going to a thicker cladding where variations in thickness become less critical. A potentially serious problem is one of poor quality uranium. Failures to date cannot be attributed to this effect, but they may begin to occur once other deficiencies are corrected.

An unfortunate limitation in the irradiation tests conducted is the lack of pertinent metallurgical data on the fuel material in the unirradiated state. This represents a severe deficiency in that an analysis of the relatively limited pre-irradiation data available reveals a preponderance of construction stock of poor quality.

An evaluation of irradiations of constructed rods in B&K filled capsules is of interest since the test conditions more closely approach NFR conditions with the obvious exception of the environment. The post-irradiation information is similar to that of the tubular irradiations. The uranium was cracked; there were several cladding failures, an end-cap was separated by a circumferential crack; and there was definite swelling. Of 16 tests, six appeared to suffer little or no damage, seven suffered cladding splits, one had extensive cracking in the uranium, and four have not been examined. This is not a very optimistic picture, considering the cladding was 20 or 30 mils on a 0.630-inch rod.

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Tests at the ETR and MTR are not too meaningful. A limited quantity of coextruded IAE stock was irradiated in the single pass loop of the MTR, and suffered end cap failures by circumferential splitting, possibly as a result of thermal incompatibility. The ETR irradiations have been limited to controlled rupture tests on previously unirradiated fuel or on test specimens previously irradiated in the KAPG reactors. These tests told little about fuel behavior, but they did give some indication of the rate of reaction between fuel and coolant.

Most of the Du Pont irradiation information on tubular fuel is not particularly relevant; however, some trends can be observed. These data, presented in Appendix D, cover irradiation of 18 fuel elements at SRP, HRU, and VVER.

The tests in cold water at SRP contain two types of fuel - unalloyed uranium and uranium containing 2 w/o Zr; of eight natural uranium tubes, one failed by swelling and sticking in the tube. The others reached exposures as high as 5200 MWD/T average at surface temperatures of 88-112 C.

The U-2 w/o Zr tubes (8) were irradiated to burnups as high as 3400 MWD/T average. Three failed by cracking or swelling, one developed an excessively rough surface, and the other four were relatively unchanged.

A natural uranium tube irradiated in the HRU bowed sufficiently that it could not be recharged after 1000 MWD/T exposure.

The VVER test, which has accumulated about 1100 MWD/T, displays localized swelling amounting to 3.5%. The test is continuing.

Sections of coextruded U-2 w/o Zr rod have been irradiated in lead-filled capsules at SRP. The principal variable was cladding thickness which included values of 5, 10, 15, and 25 mils. A short-time (800 MWD/T) series of tests suffered no damage. Exposures to 2000 MWD/T resulted in three

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cladding failures of eight tested.

A few sections of Zircaloy-2 clad coextruded U_3Si rod have been irradiated in the NRU by ANCL personnel. Of seven specimens irradiated, one failed immediately due to defective cladding; the remainder were irradiated to about 1000 MWD/T where two were unchanged, while the remaining four swelled markedly and two of these suffered cladding splits.

On evaluation of the preceding information, it is obvious that there is a universal problem of cladding splits, both longitudinal and circumferential. This problem has not been solved and is further complicated by two different failure mechanisms, ductile and brittle. Substantial work is required in this area to establish the conditions of such failures, influence of thickness and variability of cladding, and the relation of fuel swelling and cladding strain on failure.

It is probable that the NFR must start up with many of the problem areas in the irradiation field still unsolved. In essence, the first charge will represent a major production test. In the limited time remaining, the effort should be concentrated on tests under prototypical NFR conditions, tests aimed at evaluating the brazed closure, and tests to establish the effect of thicker cladding. A test should be made in the NFR on an NFR sized fuel element since the current schedule for retubing the NFR loops to handle the NFR outer tube is several months away. These tests should have major priority with sufficient backup from stock coextruded at HAPG to keep the loops filled.

Serious consideration should be given to diverting most of the effort in the Fuel Design Operation to advanced fuel element concepts. This will become very important in the next 12 months.

The attached table summarizes irradiation experience of concern to the NFR fuel program.

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NUMBER OF CONTAINERS FOR INFORMATION

Test Information	Number of Experiments	Facility	Radiation Conditions			No. Tests to Qualification	Test Results Information					Total Information
			Normal DO C	Low TID C	Proposed Dose Rate		Successful	Partial	Failure	Other		
DOF Thermal Fuel Tests	20	DOF	20	10	2000-7000	1	0	2	1	4	0	7
Tests in DOE Capabilities	20	DOF, DOE, DOF	16	10	2000-7000	16	6	6	1	1	6	0
DOF Tests DOE-DOF	2	DOF-DOF	2	2	2000	0	0	0	1	1 (Success)	0	0
DO Fuel Thermal Fuel Tests - Qualification	8	DOF	0	2	2000-7000	6	1	0	0	1	0	0
DO Fuel Thermal Fuel Tests - 2 up to 20	6	DOF	0	7	2000-7000	5	3	2	0	2	0	0
DO Fuel Thermal Fuel Tests - High Temp.	2	DOF, DOE	2	2	1	0	0	0	0	1	0	1
DO Fuel Tests in DOE Capabilities	22	DOF	22	12	2000-7000	22	9	3	2	0	0	0
DOF Containers Up to 20	6	DOF	6	6	2000	0	2	1	1	2	0	0

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(13) FUTURE DEVELOPMENT

A major deficiency in the NFR fuel program is the absence of work on alternate fabrication processes or on other fuel materials. The assumption has been made that the coextrusion process must and will work, and there is some assurance that this will be the case. However, it is quite possible that other fabrication processes may yield a better and less expensive fuel element. Some thought should be given to work in these areas and this should begin relatively soon since any work started now will require 3-5 years to reach fruition.

Work in any substantial variation of the current process such as pressure bonding, improved coextrusion techniques, a shift in cladding material, or a pronounced change in fuel core composition or geometry represent possible areas of effort. Some of these approaches will become more promising if the demand for plutonium decreases and the NFR is used for the production of power. In this case, alternate fuel routes such as thorium or UO_2 become promising.

As FPD expands its activities into the fuel irradiation field, it would be advisable to shift some Fuel Design effort to advanced fuel concepts in the metallic area.

Information pertaining to uranium-oxide fabrication and testing can be obtained directly from work in the FRTR.

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(2) CONCLUSIONS AND RECOMMENDATIONS

1. A major deficiency is the lack of correlation and interpretation of the data being collected. Unless this is done, problem areas cannot be defined.
2. Too little effort is being expended in the solution of the problems of immediate concern and too much effort goes to peripheral problems.
3. A program should be established to evaluate the quality of 11-inch uranium castings; a portion of this program should be an evaluation of cored castings.
4. Careful consideration should be given to the advantages and disadvantages of alloying uranium before deciding to go to alloyed cores. There is definite reason to believe that little or nothing will be gained through such alloying.
5. A major problem which requires immediate action and close integration with HLO is in the establishment of quality specifications for the current material. The work at HAPC in the evaluation of metal quality has been limited or non-existent, and there is no reason to believe that it has been much better at HLO. Chemical analyses are not enough. A carefully designed sampling program is required, incorporating non-destructive testing and metallography.
6. There are several potential problems in the use of Zircaloy-2. These include: (1) product variability, (2) lack of corrosion resistance, (3) hydriding and embrittlement, (4) water penetration through stringers in the end caps, (5) sensitivity to contamination during fuel processing, and (6) high costs with concomitant necessity to minimize scrap losses.

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7. Billet assembly is a very critical area where quality control will be of major importance. Poor bond quality, dirty interfaces, and localized cladding defects may result from inattention during cleaning or to lack of care in closing the billets. This process requires additional work before it can be covered with a specification. Operator attitude will be a key factor here.
8. Definite progress has been made in controlling some of the extrusion parameters. Even so, most factors have not been fixed and a reactor quality product is yet to be produced. Work is required in optimizing the heating cycle, improving smoothness of the cladding, eliminating local defects in the cladding, and in optimizing the reduction ratios.
9. An evaluation of coextruded product produced by EMI and FPD reveals the number and magnitude of the problems requiring solution. Typical problem areas include: (1) poor quality uranium, (2) internal cladding defects, (3) copper breakthrough during coextrusion with severe galling and ellipticity, (4) untuly thin cladding, (5) excessive variations in cladding thickness, (6) rough, uneven and striated surfaces, (7) inclusions imbedded in the cladding, (8) pits, wounds, chevron defects, and gouges in the cladding, (9) excessive "dirt" at the core-clad interface, (10) segregation and pronounced banding in the U-2 w/o Zr core material, (11) unbonded or poorly bonded fuel, and (12) Zircaloy-2 fails to pass corrosion test. These problem areas all require solution before an adequate coextrusion process is ready for manufacturing.
10. An extensive sampling program incorporating existing tests and adding metallographic evaluation of the core and careful visual examination of

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the internal and external surfaces must be coupled with careful non-destructive tests. These tests will permit the quality of the constructions to be determined and indicate problem areas requiring more attention.

11. The steps dealing with the removal of copper from the constructions and Kircaloy-2 cleaning may result in a greater loss of coextruded fuel than any other operation based on problems in the existing process. Complete removal of copper is unduly time consuming and often incomplete. Vapor blasting is dependent on operator skill and lacks adequate control. Surface cleaning is uncontrollable. All these factors will result in losses during autoclaving.
12. Additional work must be done in chemical milling to optimize the process. Definite difficulties will occur with alloys including decreased rates of attack and non-uniform attack and attack rates.
13. A major problem area is the brazed closure. Too little work has been done and this has not been evaluated; beryllium toxicity is a deterrent to its use in a braze alloy; brazing adversely affects the uranium cladding and bond; and no irradiation information is available.
14. The requirements for the braze should be carefully reviewed. Outstanding corrosion resistance does not appear to be a controlling factor, and relaxation here might permit an alloy free from beryllium to be used; this simplifying the process. In any event, some attention should be paid to the ternary Zr-Fe-Si alloy since its lower melting point is a definite advantage which might offset the increased brittleness. A decrease in braze temperature from 1035 to 950 C would be worthwhile. Some consideration should be given to the offsite development of alternate braze alloys.

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15. There are no inspection techniques to detect unbonded brase-uranium interfaces. Several approaches should be investigated since an unbonded interface may lead to rapid failure due to thermal incompatibility between cladding and core.
16. The work on developing a welding process for unbonded fuel elements is not sufficient to permit a process specification to be prepared. This should be completed as soon as possible since this is the only backup process capable of being used for the first charge.
17. Supports do not appear to be a major problem; however, information is required on the behavior of steel attachments during irradiation since they may become notch brittle.
18. Additional work is required in salt bath beta heat treating to establish quenching conditions and to determine if the columnar structure produced during brazing can be eliminated. Induction heat treating appears to have no economic incentive on short fuel lengths since it requires more costly equipment, is slower, and will require more maintenance.
19. Straightening after heat treatment may not be a problem since the short lengths of tubular elements do not tend to bow very much. However, if such straightening is required, more work must be done to minimize Zircaloy-2 contamination during the straightening operation.
20. Autoclaving processes are well established; however, certain areas should be investigated such as establishing the desirability of a steam-water cycle to improve the oxide layer and pros and cons of installing hydrogen alarms.
21. Non-destructive testing processes have not been developed to the extent required for AFR fuels. Greater rapport is required between fabrication

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and testing components to permit maximum utilization of information obtained. This is not being done at all adequately.

22. A major limitation in the optimization of the NFR fuel element is the lack of meaningful irradiation data. Principal deficiencies include: (1) no irradiations where the fuel operates under proposed NFR conditions, (2) no irradiations of the proposed NFR size fuel element, (3) no irradiations of fuel fabricated with the proposed process of braze closure, (4) lack of tests to goal exposure under irradiation, either prototypical or non-prototypical, and (5) little hope that there will be sufficient irradiation information prior to NFR startup to permit adequate evaluation of the proposed fuel fabrication concept.
23. A major deficiency in the NFR fuel program is the absence of work on alternate fabrication processes or on other fuel materials. The assumption has been made that the construction process must and will work, and there is some assurance that this will be the case. However, it is quite possible that other fabrication processes may yield a better and less expensive fuel element. Some thought should be given to work in these areas and this should begin relatively soon since any work started now will require 3-5 years to reach fruition.

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APPENDIX A

ANALYSES OF NRI EXTRUSIONS

NRI INNER - NATURAL UNALLOYED

Approximate Footage - 74

Extrusion Nos. (20459, 50, 21145, 46, 47, 27487, 88, 89, 90, 91, 92)

Quantity Not Evaluated - 43 (3' left)

Rejected After Extrusion Step - 19

Rejected During Fabrication - 9 - to date

Amount as Fuel - 3

Comments - Rejected material due to oxide stringers, tears, internal clad defects, Cu breakthrough with severe galling and ellipticity (27489), cladding as low as 16 mils after Cu removed (final 10-12 mils). Acceptable material had relatively severe surface striations and roughness, as well as fine oxide inclusions in cladding. Rejects in progress due to U in brase and pinholes in welds in brase fuel.

Maximum Percent Yield - 62% assuming all unevaluated is good and none is lost in processing.

NRI INNER - NATURAL ALLOYED 2 w/o Zr

Approximate Footage - 20

Extrusion Nos. (21148, 49, 50, 51)

Quantity Not Evaluated - 12 (0' left)

Rejected After Extrusion Step - 4-1/2

Rejected During Fabrication - 0

Amount as Fuel - 3

Comments - Limited amount evaluated, mostly used as charge-discharge tubes. One severe cladding tear extending through to U. Some unbonded areas.

Maximum Percent Yield - 75% assuming unevaluated is good.

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APPENDIX A (Cont'd.)

KEK OUTER TUBE 1.47% ENRICHED

Approximate Footage - 79

Extrusion Nos. (21138, 39, 40, 21167, 68, 69, 21177, 70, 79)

Quantity Not Evaluated - 44-1/2 (35 stock)

Rejected After Extrusion Step - ?

Rejected During Fabrication - 21

Amount as Fuel - 13-1/2

Comments - Rejects due to white oxide in autoclave step either due to inadequate cleanup during vapor blasting or to poor clad quality; warped fuel elements due to heat treatment. Some cladding quite thin after final operations (8 mils). Inadequate metallurgical history available.

Maximum Percent Yield - 73% assumes unevaluated is good.

KEK OUTER TUBE 1.6% ENRICHED 2 w/o Zr

Approximate Footage - 72

Extrusion Nos. (21073, 74, 75, 76, 21115, 16, 17)

Quantity Not Evaluated - 30 (40 in stock)

Rejected After Extrusion Step - (72% see Comments)

Rejected During Fabrication - 4-1/2

Amount as Fuel - 10-1/2 to date

Comments - Essentially all of this material examined was not of reactor quality. The uranium was very dirty; microstructure was extremely variable with islands of inclusions, bond line dirt, segregation and pronounced banding. Some blistering was observed. External appearance good, a few small dimples. Used in-reactor in spite of history.

Percent Yield - 0 or 5%.

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APPENDIX A (Cont'd.)

NER OUTER TUBE - NATURAL

Approximate Footage - 43

Extrusion Nos. (21672, 73, 74, 75)

Quantity Not Evaluated - 0

Rejected After Extrusion Step - 21

Rejected During Fabrication - (as hot head setup 10')

Amount as Fuel - 12

Comments - A very complete history. All extrusions of poor uranium, containing slag and inclusions resulting in very poor interfaces. Used in spite of this for reactor tests. Carbon 750 ppm. Small dimples in inner bore.

Wall $\pm$ 4 mils variation.

Percent Yield - 0 or 28.

NER INNER - 1.47% ENRICHED

Approximate Footage - 44

Extrusion Nos. -(21171, 72, 21180, 81, 82, 83)

Quantity Not Evaluated - 8 (10-1/2)

Rejected After Extrusion Step - 0

Rejected During Fabrication - 26

Amount as Fuel - 10-1/2

Comments - Relatively poor metallurgical history on this batch. Nine feet rejected for pre-post heat-treat warp. Six feet rejected for oxide appearance after autoclaving. Some of this had thin cladding. U quality not known.

Percent Yield - 42 max. assumes 8' is good.

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APPENDIX A (Cont'd.)

KER INNER - 1.6% ENRICHED. 2 w/o Zr

Approximate Footage - 40

Extrusion Nos. (21130, 31, 32, 33, 34)

Quantity Not Evaluated - 0

Rejected After Extrusion Step - 14 (actually all poor)

Rejected During Fabrication - 0 to date, 12 in process or scrap

Amount as Fuel - 13-1/2

Comments - One length severely scratched. Defective bond and clad (swiles) in some areas. Chemistry okay. U quality poor; bending, inclusions, non-uniform dimensional changes on heat treatment; chevron defects in clad; rough and striated on O.D. 18-27 mils cladding.

Percent Yield - 0 or (2 maximum.

KER INNER - NATURAL

Approximate Footage - 59

Extrusion Nos. (21736, 37, 38, 21873, 74, 75)

Quantity Not Evaluated - 29

Rejected After Extrusion Step - 10

Rejected During Fabrication - 12

Amount as Fuel - 6

Comments - Marginal quality; cladding as low as 17 mils after Cu removal (10-12 final); surface poor; clad poor, severe galling, dirty bonds.
Percent Yield - 62 maximum; assumes all material remaining good.

APPENDIX B
ANALYSIS OF FPD EXTRUSIONS

KER OUTER

Reactor Quality	Extrusion Number	O.D. Range Max.	I.D. Range Min.	Wall Range Min.	Outer Clad Range Min.	Inner Clad Range Min.	Peel Test Q, Q, F, P		Interface Cleanliness Range Average	Uranium Grain Size I.D. O.D.
							Outer	Inner		
No	10	5 1825	7 1449	6 182	* 7 019	*10 020	Q-Q	Q	5-8 6.5	*130 065
No	11	6 1826	6 1450	7 181	4 020	*13 018	Q-F	Q	4-8 6.0	*130 090
+Grey	12	5 1826	3 1453	8 170	5 020	* 8 018	Q-Q	Q	6-8 7.3	055 035
Possible Grey	13	* 8 1826	5 1450	*11 176	5 021	* 8 019	Q-Q	Q	4-8 6.9	055 045
No	14	5 1827	6 1449	7 180	5 021	* 9 017	Q-Q	Q	4-8 6.8	055 045
++Possible	26	7 1822	3 1452	13* 179	5 021	5 020	Q-F*	Q-F	6-8 6.3	040 045
No	27	5 1826	5 1451	20* 177	5 025	8* 018	F	FF*	5-7 6.2	075 075
No	28	5 1827	5 1451	16* 178	5 021	12* 016	Q-F	F*	5-8 6.4	065 045
Possible Grey	29	6 1827	4 1451	10 182	5 021	8* 019	F	Q	3-8 6.3	075 055
No	30	7 1826	3 1453	8 181	5 021	12* 016	F	F*	5-7 5.8	090 075
No	31	2 1826	6 1451	11* 181	3 022	9* 019	F	F*	4-7 5.1	065 065

* Cause for rejection.

+ Grey indicates of doubtful quality but not sufficiently bad to reject.

++ Possible - Mediocre but usable.

KER SINGLE

No	24	7 1744	6 1055	9 336	5 026	18* 018	Q	Q	5-8 6.8	075 070
No	25	8 1744	5 1056	13 335	5 026	17* 018	Q	Q	3-8 7.2	075 065
Yes	32	9 1752	3 1048	12 344	5 027	8 023	Q	Q	4-8 6.7	090 090
Possible	33	11 1754	5 1048	13 344	7 027	15 023	Q	Q	6-8 6.8	110 075
No	35	14* 1755	5 1049	12 344	4 027	5 024	Q	Q	6-8 7.1	210* 065
Grey	37	2 1754	3 1049	22* 340	4 027	7 024	Q	Q	5-7 5.3	075 065
Grey	38	10 1758	5 1050	14 343	7 026	6 023	Q-Q	Q	6-7 6.3	210* 065
Grey	40	7 1760	4* 1056	17 341	3 027	8 022	Q-F*	Q	6-7 6.5	250* 075

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APPENDIX B

ANALYSIS OF FPD EXTRUSIONS

FPD OUTER

Reactor Quality	Extrusion Number	O.D. Range	I.D. Range	Wall Range	Outer Clad Range	Inner Clad Range	Fuel Test		Interface Cleanliness	Uranium Grain Size
		Max.	Min.	Min.	Min.	Min.	O.Q.F.P.		Range Average	I.D. O.D.
No	1	22 2466	6 1844	11 296	---	---	WT-P	WT-P	2-6* 4.3	300* 130
Grey	2	13 2465	9 1841	10 298	4 020	*11 017	0	*Q-F	4-7 6.2	210* 045
No	3	14 2466	10 1839	11 299	5 018	*18 015	0-Q	Q	5-8 5.8	130* 110
No	4	8 2459	7 1838	15 299	6 015	*14 016	Q	*Q-F	2-8 6.4	130* 130
No	5	12 2463	8 1836	14 298	*11 016	*12 013	0-Q	*Q-F	2-8 6.4	065 075
No	6	13 2461	8 1833	10 301	*11 014	*19 016	0-Q	*Q-F-P	3-8 6.7	090 090
No	7	17 2465	9 1834	18 298	*12 015	*26 016	0-Q	Q-F	1-8 7.1	110 075
No	8	9 2459	6 1840	14 297	*12 018	*14 018	*Q-F	*Q-F	1-8* 5.2	065 065
No	9	13 2461	11 1836	11 301	*19 018	*21 015	*Q-F	*Q-F	1-8 6.5	210* 090
Grey	15	8 2457	9 1840	8 298	5 019	5 019	*Q-F	*Q-F	7-8 7.8	130* 060
Possible	16	*7 2449	8 1834	14 295	7 018	7 018	0-Q	Q	7-8 7.8	130* 045
Possible	17	7 2454	9 1840	17* 293	3 020	3 020	0-Q	0-Q	6-8 7.3	075 055
---	18	*11 2447	4 1838	4 296	3 020	3 020	0-Q	WT	4-7* 5.3	110 090
No	19	14 2466	13 1835	9* 306	*9 014	*9 014	0	WT	5-7* 5.9	110 075

FPD INNER

No	21	6 1426	7 512	22* 440	5 021	15* 015	0	Q	6-8 7.4	360* 055
No	22	13 1431	9 514	16 440	4 022	7 019	Q	Q	3-7 5.6	210* 035
Grey	23	14 1428	4 511	36* 434	7 022	11 019	Q-F*	Q	5-8 7.0	1301 060
No	34	9 1423	4 511	16 443	2 023	8* 018	Q	Q	5-7 6.4	220* 065
No	36	5 1426	2 515	13 441	61 033	8* 017	Q	Q	5-7 6.4	250* 075
No	39	5 1422	2 510	16 446	4 021	6 018	P*	Q	1-7 5.0	300* 050

ROUGH DRAFT

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APPENDIX B (2)

URANIUM HISTORY - FPD EXTRUSIONS

Extrusion No. & Type	Heat No.	Ingot Size Inches	Billet Fabrication					Comments
			Step 1	Step 2	Step 3	Step 4	Step 5	
1	32914TB	11"	Alpha Pierce	Alpha Extrude	Rough Machine	3 Beta Quenches	Finish Machine	
2	32920T							
3	32922TB							
4	32915T							
5	32955B							
6	32956T							
7	32964T							
8	32964B							
9	32958B							
10	2160A	16"	Gamma Pierce & Extrude to 9" x 2-5/8"	Alpha Extrude	Rough Machine	3 Beta Quenches	Finish Machine	Dingot 270 ppm Fe 160 ppm Si
11	Not Known							
12	32986T2	11"	Alpha Pierce	Alpha Extrude	Rough Machine	3 Beta Quenches	Finish Machine	
13	32987T3							
14	32988T1							
15	32954T							
16	32958T							
17	32954B							
18	32985T							
19	32990T							
20								
21	32973T							

Added 3 Beta
quenches at
end.

22	1					
23	32972B1					
24	?					
25	32995B2					
26	32988B2					
27	32986T3					
28	32988B1					
29	32987B3					
30	32987B2					
31	32987B1					
32	33992B1					
33	32986B1					
34	33171T	11"	3 Beta quenches	Alpha Pierce	Alpha Extrode	Pinch Machine
35	33167B					
36	33170					
37	33164					
38	33167					
39	33170					
40	33164					
41						
42						
43						
44						
45						
46						
47						
48						
49

DOZEN DRIFT

MEASUREMENTS

Capsule No.	Chest Size	Core Size	Core Area	Exposition Conditions				Exposure Time	Date
				Latitude	Longitude	Altitude	Direction		
3-31	30	1100	800	905	255	295	215	1100	3/5/4
3-32	30	900	635	370	300	290	230	3900	7/7/4
3-37	30	1300	875	400	340	305	265	1900	10/30/
3-43	30	2150	635	570	435	420	330	1600	11/21/
3-53	30	900	625	610	750	450	420	2400	12/17/
546	30		470		320		555	2000	One 77
547	30		510		330		850	2000	
3010	20		100		315		250-	2000	7/
2015	20		560		400		340	2075	
5416	30		530		410		335	2070	1/
4420	30		575		330		380	1780	
4422	30		460		255		375	1720	One 77
542	30		500		300		270	1815	
5412	20		400		275		245	1560	9/
2027	20		510		300		230	1420	
447	20		500		230		285	1890	2/
5415	30		410		305		210	1270	
4421	30		400		250		195	1700	

0.630" O.D. x 1"

• Measured Core Temp.

2470

APPENDIX C

FORM (REV. 12) USED IN RAK CAPTURED BY R-22

Post Irradiation Testing (Wils)

Test	L	Boils	Wils	Remarks	Post-Irradiation History	Post Irradiation Note
1 - 6/1/50	7	30	8.5	-2.84	WRT - H ₂ O (g)	Nothing unusual - WRT
2 - 6/1/50	0		8.5		WRT - H ₂ O (g)	In WRT - cap gap fill
3 - 5/17/50	-10 (2" more)	22	8.7	-1.96	WRT - Air Cool (2)	Cap gap filled in - 3
4 - 6/1/50	7	40	30		WRT - Air Cool	Cap gap filled in - 3
5 - 6/14/50	0	48	39		WRT - Air Cool	Circumferential crack
6 - 6/1/50					WRT - Air Cool	No information - WAPC
7 - 6/1/50	-7	20	5		WRT - Air Cool	Crack split: brittle
8 - 6/1/50	-7	20	13		WRT - Air Cool	Crack split: - brittle
9 - 6/1/50					WRT - Air Cool	No information - WAPC
10 - 6/1/50					WRT - Air Cool	No information - WAPC
11 - 6/1/50	-32	10	1		WRT - Air Cool	Nothing unusual - WAPC
12 - 6/1/50					WRT - Air Cool	No information - WAPC
13 - 6/1/50	-7	10	5		WRT - Air Cool	Nothing unusual - WAPC
14 - 6/1/50	7	14	6		WRT - Air Cool	Nothing unusual - WAPC
15 - 6/1/50	-32	24	14		WRT - Air Cool	Split - brittle failure
16 - 6/1/50	-32	15	8		WRT - Air Cool	Brittle longitudinal
17 - 6/1/50	-32	6	4.5		WRT - Air Cool	Nothing unusual - WAPC
18 - 6/1/50					WRT - Air Cool	No information - WAPC

Spec 3.6" long

- (1) Both Heat Treat & Water Quench
- (2) Both Heat Treat & Air Cool

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EXAMINATION REPORT

Nothing unusual - NTR.

In NTR - cap gap filled in; clad split - longitudinal through U.

Cap gap filled in - NTR. Severe cracking in U - not through clad.

Cap gap filled in - NTR.

Circumferential crack at end cap - split from fuel - NTR.

No information - NTR.

Clad split: brittlely longitudinally - NTR.

Clad split - brittle - longitudinal - NTR.

No information - known to split - NTR.

No information - NTR.

Nothing unusual - NTR.

No information - NTR.

Nothing unusual - NTR.

Nothing unusual - slight working in cladding - NTR.

Split - brittle failure - considerable lumping - longitudinal splits - NTR.

Brittle longitudinal split - NTR.

Nothing unusual - NTR.

No information - NTR.

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ENG-201	ENG-202	ENG-203	ENG-204
Geometry	ENG	ENG & TET	TET
Enrichment	Enr.	Enr. - 25 En	Enr. & Enr. 2 w/o En
Dimensions (in.)			
O.D.	1.705	1.000	1.70
I.D.	1.375	0.800	1.45
Length	35	35	35
Cladding	.050	.050	.050
Loop No.	1	1	1
Time in Loop	11/17/78 - 7/11/79	7/15/79 - 6/20/80	6/20/79 - 7/11/79
No. of Elements	2	2 (2 & 1)	4 (2 & 2)
Power M/ft.	73	---	60
Heat Flux	---	---	.15 - .23 $\times 10^6$
Temperature (R)			
Bulk Outlet	280	263	270
Surface	260	---	270 - 290
Core	---	---	---
Coolant pH	4.3	10	10
Core Exposure			
MAY/T	2700	270	2700
Actual Exposure	1900	250	250
Reactor Performance	Cool Failure	Non-DET tube deformed extensively but did not fail.	Failure vacuum DET tube.

Purpose of Test	Observe tubular fuel element behavior.	Determine low exposure behavior of fuel.	Determine jacket, bond and U behavior at high exposure.
Reference Doc.	?	20371 (OT)	20155 (OT-A) 20336 (IT-A) 20370 (OT) 20150 (IT)
Material Quality	No history?	No history	No history
Test Lab Results	35 Δ V tube internally cracked and distorted.	---	Vacuum DET bugged tail; radial cracks in end & severe warp.

APPENDIX C

HISTORY OF CONTINUED TUBULAR FUEL MANUFACTURED IN THE 1960s

ITEM	ITEM	ITEM	ITEM
1.475	1.47 & 1.6 & 2 w/o 20	1.47 & 1.6 & 2 1/2 20	20A
1.02 1.05 1.45 .50 35 35 .020 .020 3	1.02 1.05 1.02 1.05 1.45 .50 1.45 .50 25 25 35 35 .020 .020 .020 .020 4	1.02 1.05 1.45 .30 25 25 .020 .020 2	1.02 1.05 1.45 .30 25 25 .020 .020 4
9/7/59 - 9/30/59	2/20/60 - 4/24/60	1/11/60 - 7/3/60	4/25/60
3	3 (2 & 1)	4 (2+2)	6
20	---	133	---
.24 = .41 ± 10 ⁶	.27 = .45 ± 10 ⁶	.27 = .45 ± 10 ⁶	.11 = .30
222 = 203 230 = 210	200 ---	270	203
4.3	10	400 Calc.	10
5000	2000	3500	2500
310	1070	3800	
Failure by heavy coral building and severe pitting in low pH water.	Failure by warp and hot spot in alloy element, slight buckling.	Failure unalloyed element. Blister and severe cracking of U.	St112 type

High power high exposure
test.

21172, 71, 60 (IT)
21132, 65, 63 (OT)

No history

High power high exposure
test.

21131, 34 (IT-A)
21073, 75 (OT-A)
21163 (IT)
21179 (OT)

Severe buckling, dirty U,
barely passable.

High power high exposure
test.

21177 (OT)
21162 (IT)
21073 (OT-A)
21131 (IT-A)

Some history, some in
the defective bond only
fair.

Investigate
bonded set

21512, 74
21736, 37.

Fair to p
interfuse
clad.

Not made
TIC value

	307 & 308-A	312	309, 309 & Supp. A 302-A, 313
	HTP-Inner	ILR	HTP-Inner
	Mat. & Mat. -0% 20	Mat.	
2 1.05	1.000		
3 .70	0.600		
18		6	
30 .000	3	3	2
1/60	7/2/60	1/12/60 - 1/13/60	8/4/60 - 8/4/60
	4 (1+3)	2	5 (2+2+1)
	---	---	
- .30 $\times 10^5$	.01 - .02 $\times 10^5$	---	
	250	---	253
	10	10	10
0	2500	2500	2500
		0	5
11 operating	Still operating	Mechanical failure of wall within 10 hours of startup; probably cup pipped off.	Failure indication aft 3 days; not at full po
Investigate beh- viour concept.	Determine the behavior of fuel under conditions approximating HTP.	Observe effect of various heat treatments on be- havior of ILR fuel.	Investigate trace elem and irradiated rupture specimens.
2, 74 (OT) 15, 17, 38 (IT)	27452 21150		27491, 57, 21150
to poor U; poor surface variable	No history. Companion pieces good in some instances.	Good to very good bond interface and core. Outer surface rough.	Generally reactor quali minor inclusions.
heated fuel IT walled outer.			

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HW-67121

307 & Supp. A

312

**309, 309 & Supp. A
312-2, 313**

HTF-Inner

IIS

HTF-Inner

Mat. & Mat. -07, 22

Mat.

**1.250
0440**

6

3

3

2

7/1/60

5/14/60 - 5/15/60

6/4/60 - 6/4/60

4 (1+3)

5

5 (2+2+1)

.01 - .02 x 10⁴

250

253

20

10

10

2500

2500

2500

0

5

Still operating

**Mechanical failure of
vial within 10 hours of
startup; probably cap
popped off.**

**Failure indication after
3 days; not at full power.**

**Determine the behavior
of fuel under conditions
approximating HTF.**

**Observe effect of various
heat treatments on be-
havior of IIS fuel.**

**Investigate trace elements
and irradiated rupture
specimens.**

**HTF
21150**

27490, 27, 21150

**No history. Compression
pieces good in some
instances.**

**Good to very good
bond interface and core.
Outer surface rough.**

**Generally reactor quality;
minor inclusions.**

October 14, 1960
 W-67222

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APPENDIX C
 HIR-SEN IRRADIATIONS WITH CO-EXTENDED
 IRRADIATION TIME CORRECTION

TIME	LOSS	Spec. Pwr.	Pressure	Temperature	Surface Heat Flux	History
Operation of Prototype Elements in the HIR Reactor 6 x 9 Range	6 x 9	67.5 w/gm	2000 psi	240°C bulk 300°C ann. Clad Temperature	1.25×10^5 BTU/hr-ft ² max. on inner surface of inner tube.	Discharged after 12 hours (12.5 hr/T) after Phillips Petroleum Co. operations ac- cidentally drained the process water from around the element. Post irradiation revealed no apparent damage to the element.
GEN-4-70, 71; Sr-8 Clad 125 Gasol Fuel Element Irradiation	GEN-4	83.1 w/gm	180 psi	110°F bulk 117°F film 221°F clad	670,000 BTU/hr-ft ² - outer surface 1,120,000 BTU/hr-ft ² - inner surface	Post irradiation examination disclosed the clad cap on 4-70 element to be cracked at the periphery of both external and internal welds.
GEN-10-08, 29, 30, 31 4th Rupture Experiment	5 x 3	21.2 w/gm peak outer tube, 1.2 w/gm peak inner tube.	5000 psi	707°F bulk 524°F at clad surface. 591°F at clad surface of inner tube of inner tube.	Outer tube (peak ϕ) 1. 211,000 BTU/hr-ft ² outer surface. 2. 217,000 BTU/hr-ft ² inner surface. Inner tube (peak ϕ) 1. 213,000 BTU/hr-ft ² outer surface. 2. 695,000 BTU/hr-ft ² inner surface.	Element irradiated for 10 hours (5 hr/T) before intentional rupture initiated.

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Tube No.	Description and Dimension	Heat Treatment	Infiltration Conditions			
			Temp	Atmosphere	Pressure	Time
<u>Unalloyed U Cases</u>						
6930	a	b	160 270	120 180	88 112	12 9
6937	a	b	160	180	88	9
6938	a	b	150	115	84	16
6931	a	b	150	115	84	14
9131	a	b	270	180	112	6
9132	a	b	270	180	112	9
6933	a	b	360	205	134	4
72	c	f	400	130	210	9
<u>U-C w/a Br Cases</u>						
13	a	e	355	205	130	12
17	a	e	411	207	136	9
29	a	e	411	220	125	24
30	a	e	411	220	125	22
<hr/>						
11	a	f	350	205	125	15
26	a	e	350	191	125	27
22	a	e	265 350	155 215	97 130	31
34	b	i	375	215	135	17
53 (Det. U)	c	j	400	205	119	16
41	b	f	405	300	300	12

- (a) 2.95" O.D. x 2.44" I.D. x 14" L., 30 mil Br clad natural U case 28".
 (b) Heat treatment 750 C - 10 min. Salt, air cool - 8 min., H₂O (g).
 (c) 2.060" O.D. x 1.467" I.D. x 11", 30 mil Br clad natural U case 2.9".

APPENDIX 2

HISTORY OF CONTAINER DATA & FUEL INFORMATION BY DEL FINE

11/1/79		Fuel Information		Temperature	
DEL	MO/T	Range	Date	DEL	MO/T
570	1310	1.14 - 1.20	12/1/55 - 3/8/56	60	11
700	2050	2.11 - 2.25	11/15/56 - 1/27/57		
970	650	1.13 - 1.21	12/1/55 - 1/17/56	NO INFORMATION	
000	670	0.69 - 1.15	3/3/56 - 5/20/56	NO INFORMATION	
120	580	0.71 - 1.20	3/3/56 - 5/20/56	NO INFORMATION	
550	5250	1.00 - 2.31	10/18/56 - 7/16/57	15	0
600	4500	1.20 - 2.37	10/18/56 - 4/20/57	0	0
140	3075	3.00 - 3.56	3/4/59 - 6/12/59	22	6
955		22.9/T	4/13/60 - 5/25/60	90	3.9 1.2 -0.2 -60
1350	1200	1.9 - 2.00	7/17/58 - 6/23/58	07	8 0 2 -1
1400	1350	1.8 - 2.30	9/19/58 - 12/31/58	370	30 -6 12 -05
1450	2300	2.2 - 2.18	1/3/59 - 3/10/59	---	35 9 0 -14
1210	2075	2.2 - 2.15	1/3/59 - 2/16/59	370	30 -3 0 -10
1900	1720	2.1 - 2.29	4/26/59 - 6/12/59	---	24 0 0 -9
1350	2250	1.9 - 2.15	6/16/59 - 8/16/59	---	15 -5 Not available
1175	2325	1.25 - 2.36	8/21/59 - 12/5/59	NOT AVAILABLE	
1750	1750	0.3 - 0.39	8/21/59 - 10/2/59	NOT AVAILABLE	
1830	1750	1.7 - 1.85	8/21/59 - 10/12/59	15	-60 Not available
1200	1100	2.00 - 230/T		See Appendix D-3	

- (d) 2.070" O.D. x 1.467" I.D. x 11", 15 mil 20-4 clad, 30° core.
 (e) 600 C - 7 hr. H_2O (g) in container.

- (f) Same as (e) except built as (200
 (g) Same as (e) + spherulizing 2.5
 (h) Same as (d) except 6" (34° core

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Specimens (W1s)			Comments	
Run	W	W	Pre-irradiation	Post-irradiation
---		0	Clad thickness 19-22 mils	Surface unchanged - SFP
		0	Clad thickness 19-25 mils	Surface unchanged - SFP Surface unchanged - SFP Surface unchanged - SFP
		0		Post-irradiation heating - no visual cre C.D. changed 0-10 mils to 0-15 mils - SFP Cladding grooved, but no failure - SFP Spotted - no noticeable roughness - SFP
-10	1.5	30-40 to 200	Undulated surface. Clad varies 20-30 mils.	Slight surface rippling in thinner clad ; No stud. Severe bowing in recharge. 10 → 200 mil (in SWI).
-1			Clad varies outer = 2-15 inner = 10-15	Unchanged - SFP
-05			Clad varies outer = 10-17 inner = 12-15	Unchanged - SFP
-14			Clad varies outer = 12-15 inner = 11-15	Failed - longitudinal crack 30" from end
-13			Clad varies outer = 12-15 inner = 11-17	Unchanged - SFP
-9				Unchanged - SFP
-21a				Rough except 2" (from spotwelding S.T.) Failed - SFP Failed - SFP Failed - heavy surface - stuck - SFP Still being test. 3.5% a, 575,000 BW/hr

as (END C - 1/2 hr.)
using S.T. 620 C - 72 hr.
30° conv).

- (1) as extracted.
- (2) Same as (b) except air cool 400 C & 2.7 (a).
- (3) 2.05" C.D. x 1.175" I.D. x 14" x 15 mils S-T.

POST # 17 2000120

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October 14, 1980
NSA-6722 RD

APPENDIX E

EXTRACT OF THERMAL DATA (2-2 w/o 2-1) FROM THE TEST CONDUCTED AT SRI

Sling No.	Cord Thickness (in.)	Radiation Temperature - F. ± 10°				Conduction		Time	Fast Irradiation Results (in.)				Fast Irradiation Remarks
		Surf.	Surf.	Interface	Surf.	Surf.	Surf.		1.	2.	3.	4.	
22	25	690	970	390	340	3000	615	11/14/78 - 2/16/79	2	1.5	0.20	1.5	No change - SW.
13	25	745	980	400	370	2070	655	11/14/78 - 2/16/79	13	1.7	0.28	1	No change - SW.
7	20	630	730	375	375	2000	840	11/14/78 - 2/16/79	40	1.2	0.13	7.5	Slight bump over-end cap interface - SW.
3	5	670	905	365	310	2070	680	11/14/78 - 2/16/79	79	1.1	0.09	30	Slight bumping - SW.
19	25	670	905	365	310	2130	2020	11/14/78 - 2/16/79	12	2.2	1.25		Long cracks 15" all sides - 1" long near top. Long bump over end. Slight bump over-end cap interface - SW.
24	25	680	875	380	300	2000	2020	11/14/78 - 2/16/79	47	1.7	0.21		Slight bump at over-end cap interface - SW.
6	20	705	860	325	300	2130	1770	11/14/78 - 2/16/79	49	1.7	1.49		Slight separation at over-end cap interface - SW.
2	5	670	860	380	320	2320	2000	11/14/78 - 2/16/79	---	1.5	1.25		Top cap separated from core at interface. Cracks in closure weld at bottom cap - SW.
23	25	875	110	270	275	2605	---	3/4/79 - 3/15/80					All values estimated on irr.-SW
15	25	820	885	295	270	2380	---	3/4/79 - 2/1/80					All values estimated on irr.-SW
9	20	825	830	195	250	2375	---	3/14/79 - 2/1/80					All values estimated on irr.-SW
4	5	865	770	260	255	2065	---	3/14/79 - 3/15/80					All values estimated on irr.-SW

All slings 0.75" dia. x 12.5" long with 7" core 520 C - 7 hr. - 4L.

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APPENDIX D

SWELLING OF DU PONT U-25 IR TUBE IN VVER

Fuel Dimensions

Cladding	0.015 inch Ir-2
O.D.	2.064
I.D.	1.474
Length	44 inches
Enrichment	3.07% U-235

Irradiation Conditions

Pressure	1000 psi	Coolant	285 C
NV	4.66×10^{13}	Core-clad interface	350 C
Specific Power	200-230 KW/T	Core	446 (425 reached)

Heat Flux
BTU/hr/ft²

I.D. Avg. - 450,000

O.D. Avg. - 396,000

I.D. Max. - 575,000

O.D. Max. - 505,000

Post Irradiation Measurements

Per Cent Increase in Cross Sectional Area
At Various Positions on Tube

Burnup (Avg.)

MWD/T

150

772

1100

	1	4	8	12	14	16	20
	0.1	0.1	0.2	0.2	0.2	0.1	0.2
	0.1	0.3	0.5	0.7	0.7	0.4	0.1
	0.1	0.7	1.5	3.2	3.5	2.0	0.3

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October 14, 1960
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APPENDIX E

IRRADIATION BEHAVIOR OF ENRICHED U_3Si ELEMENTS OPERATED IN Zr-2 L. E. MOORE, CORNET-904, 1/60 (U)

U_3Si 1.9 w/o U-235 irradiated X-1 and X-2.

0.8" diameter rods constructed Zr-2 clad 4" long.

Sample	Watts/cm	BTU/ft ² /hr.	Max. Core Temp. (C)	W-1650/Tonne	
HM	600	287,000	490	290	No damage.
HN	680	333,000	585	330	Definite swelling.
JC	750	365,000	618	370	1-3/8" axial split- definite swelling.
JD	790	386,000	637	390	Definite swelling.
HS	830	408,000	658	410	Circ. split at fuel- and cap interface, al swollen markedly.
HP	870	417,000	666	420	No damage.

HO - Failed on startup due to clad defect.

HP

Sample	Core Temp. (C)	Initial Diameter	2 Mile Burner	3 Mile Burner	4 Mile Burner
HS	658	.814	4	524	26 1200 *
JD	637	.815	4	498	23 1133
JC	618	.812	2	450	20 1074 *
HN	585	.813	2	574	12 965
HP	666	.814	6	396	
HM	490	.812	5	---	* Failed
HO	680	.818	Failed startup - brittle failure fuel & clad.		

DATE

FILMED

9/22/94

END