Enrico Fermi Atomic Power Plant

REPORT ON CORE AND BLANKET DESIGN AND TECHNOLOGY

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Prepared for Fuel Bidders Conference

by

C. M. Ladd
R. J. Hennig
O. E. Homeister
D. O. Leeser
R. A. MacLeod
R. G. Rateick
L. P. VanMaele

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I. SCOPE

This report contains general and specific information to assist the fuel and blanket element fabricator to understand the objectives of, and requirements for, the initial core and blanket loading of the Enrico Fermi Atomic Power Plant. The introduction includes a brief description of the private industrial organizations engaged in the demonstration of this reactor plant concept, a description of general features of the plant, and its operating conditions.

A summary of the information developed to date on the core and blanket fuel materials is presented, as well as brief resume of present and future development programs to increase the fund of knowledge on fast reactor fuel cycles. The specification requirements for the core and blanket subassembly fabrication are discussed from the standpoint of their need and the cost of variations therefrom in plant operation costs to the Power Reactor Development Company.

A discussion on the special materials accountability and shipping problems is presented.

This report should be read in conjunction with APDA Specification No. 30-8 and the core and blanket subassembly drawings included as a part of that specification.

II. INTRODUCTION

Power Reactor Development Company (PRDC) is a non-profit membership corporation, the members of which consist of substantial public utility and industrial corporations in the United States. It is engaged in the construction of a full scale developmental fast neutron breeder reactor in Monroe County, Michigan. This reactor is being constructed on land leased from The Detroit Edison Company, which is constructing adjacent turbine-generating facilities for the production of electricity, and purchasing the steam required for these facilities from PRDC. The entire plant will be known as the Enrico Fermi Atomic Power Plant.

Atomic Power Development Associates, Inc. (APDA) is a non-profit membership corporation engaged in research, experimental and developmental activities in the fast breeder reactor field. APDA and PRDC are separate and distinct companies; APDA has contracted to furnish PRDC substantial design, research and development services, and to transfer to it certain equipment and devices which are to be incorporated in the PRDC reactor after non-nuclear testing by APDA. PRDC also has a contract with the AEC under the Power Demonstration Reactor Program.

PRDC will, with private resources, finance, design, build, own and operate the nuclear plant which incorporates the APDA design. The AEC will carry on important research work in support of this design and construction program. Most of the AEC-supported research will be carried on by the Argonne National Laboratory.

The fast breeder reactor was selected by APDA and PRDC as the reactor having the most promise of becoming competitive with conventional methods of producing power and the means by which America could best maintain its supremacy in atomic power development. It also has the unique ability to make more nuclear

fuel than is consumed while in the process of producing heat energy for steam-electric power generation. The nuclear fuel produced by the fast breeder reactor is plutonium -- a man-made fuel that is also of great value to national defense. When the Enrico Fermi plant is completed in 1960, private industry will have made another big contribution to the nation's expanding nuclear technology and atomic leadership.

The PRDC fast breeder reactor is similar to other types of nuclear reactors in that it releases heat energy through controlled nuclear fission. It is unlike most other reactors in that it utilizes high energy (fast) neutrons without the use of a moderator, and it produces more fissionable material than it consumes. Three reactor power plants of this type, but having lower power production capacity, are due for operation prior to operation of the PRDC reactor plant.

EBR-I was operated by Argonne National Laboratory from the fall of 1951 to November 1955. During a reactor excursion test, the core melted and had to be removed. A new reactor core has been installed and loaded with a "rigid core" type of design using fuel elements similar to those specified for the Fermi reactor, and the reactor resumed operation in November 1957. Normal power production level of this reactor plant is 200 Kw electrical. The first core element loadings of this reactor were similar to the PRDC reactor blanket elements using stainless steel tubes containing the uranium rods and sodium for thermal bonding.

The Dounreay fast breeder reactor is scheduled for operation on July 1, 1958, by the United Kingdom Atomic Energy Authority. Rated electric power output of this plant is 20 MW.

EBR-II is scheduled for operation by Argonne National Laboratory about March 1, 1960. The electrical power output of this plant is 20 MW at rated capacity. The core fuel elements of this reactor are also of similar design to

the PRDC blanket elements using a uranium alloy canned in stainless steel tubes with a sodium heat transfer bond, but using smaller diameter elements.

In addition to the above fast reactors, the SRE built by North American Aviation has core elements nearly identical to the PRDC blanket elements.

The PRDC fast breeder reactor is cooled by removing heat from the core and blanket regions with liquid sodium and transporting it to the intermediate heat exchanger, as shown by Figure 1. As the molten sodium is pumped through the reactor, it is heated from 550 F to 800 F through contact with the hot surfaces of the many closely-spaced fuel and blanket elements. In the intermediate heat exchanger, the heat is transferred to a second liquid sodium system and transported to a once-through type steam generator where water is transformed directly to superheated steam. The steam is passed through a conventional steam turbine driving the electric generator to produce electricity. The second liquid sodium loop is used for safety reasons to prevent radioactive sodium contacting water in the boiler in the event of a tube failure.

The major components of the core region of the reactor vessel are shown in Figure 2. Ninety per cent of the heat and 29 per cent of the plutonium is produced in the core region of the fuel subassemblies.

The core is an assembly of thin elements, enclosed in subassemblies, containing enriched uranium, the atoms of which will fission. When the control rods are properly adjusted, a chain reaction is sustained and fissioning takes place within the uranium atoms. Large amounts of heat energy are released at a controlled level as the fissioning process goes on.

Surrounding the core is a 2-foot thick volume containing blanket subassemblies enclosing the lower grade uranium elements of fertile (non-fissionable) material. In these subassemblies, 71 per cent of the plutonium is produced and 10 per cent of the reactor power is produced. The end sections of core subassemblies also contain blanket elements.

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For its first power production, the Enrico Fermi Atomic Power Plant will have 91 pin-type fuel subassemblies and 572 radial blanket subassemblies. These subassemblies are identical in outward appearance measuring 98-5/8 inches long and 2-5/8 inches square. As illustrated in Figure 3, which shows a pin-type core subassembly, the core area of each subassembly is composed of 144 pins 30-1/2 inches long. They are fabricated of a partially enriched uranium and clad with a thin covering of zirconium to contain the fission products. On each end of the core section is an axial blanket containing elements similar in design and function to the radial blanket. These blanket elements each contain cylindrical rods of depleted uranium thermally bonded with static sodium to the stainless steel cladding. Sodium is pumped through the core and blanket subassemblies to pick up the heat generated by the nuclear reaction. Provision is made to insure adequate space around each rod for flow of the coolant sodium as it passes upward through the reactor. The large surface area provided by dividing the uranium fuel into small elements and the use of a liquid metal capable of rapidly transferring the heat generated insures maximum operating cycle efficiency.

In order to minimize critical mass and reactor fuel inventory, core volume is kept small. This, in turn, makes core design parameters very limited in flexibility. Manufacturing tolerances must be small in order to avoid excessive fuel hot spots during operation and to prevent large variations in reactivity worth of each subassembly. Similarly, fuel material must be homogeneous in regard to alloy content and isotopic enrichment in order to maintain reactor safety with a small amount of reactivity control, to minimize reactor plant downtime for fuel subassembly loading, and to prevent the occurrence of local fuel hot spot temperatures which would reduce the irradiation stability of the fuel material.

The present status of reactor plant construction is shown in Figure 4.

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The installation of all mechanical and control equipment required for Test Facility operation will be completed in December 1958. Non-nuclear testing will proceed until shortly before the reactor critical date, September 1, 1960. By that date, the second and third coolant system loops will have been installed and the core and blanket subassemblies placed within the reactor vessel.

III. CORE PIN ELEMENTS

A. Description

The fuel pin element is a right cylinder .158-inch outside diameter, 30-1/2 inches long, with a clad .004-inch thickness as extruded, swaged and cut to length. The finished pin design is shown on APDA Drawing 5XN-1722. The fuel meat is U-10 w/o* 110 alloy with an enrichment of approximately 27 per cent. The clad material and one end cap are reactor⁽¹⁾ grade zirconium. The lower end cap is Type 304 stainless steel. The end caps are shown on APDA Drawing 2XN-1704.

B. Fuel Material Quality

The uranium fuel material is supplied by PRDC in the following enrichments for mixing to final enrichment specified -- approximately 30 per cent and natural uranium. The approximate composition of enriched and natural uranium supplied by PRDC is as follows:

B - 3 ppm	O - 200 ppm (maximum)
C - 100 ppm (maximum)	Be - 50 ppm
Cu - 20 ppm	Cd - 0.4 ppm
H - 3 ppm (maximum)	Li - 5 ppm
Fe - 200 ppm (maximum)	Al - 15 ppm
Ni - 50 ppm	Ca - 10 ppm (maximum)
N - 50 ppm (maximum)	Si - 20 ppm (maximum)

The history of this material before shipment of the special materials is as follows: The AEC processes and delivers enriched and depleted uranium to PRDC as UF_6 , which is reduced directly to UF_4 with hydrogen. By "bomb" type reduction, the UF_4 is reduced to uranium metal with either calcium or magnesium.

* Per cent by weight.

(1) To be changed in specification requirements.

An alternate process for reducing UF_6 to uranium metal by going through several steps from UF_6 to UO_2 to UF_4 to metal probably will not be used. The uranium raw materials are shipped in the form of "derbies," which is a term used to describe the first metal reduction form. These derbies may vary from a few pounds to 300 pounds in weight.

C. Typical Fabrication Process Outline

The information shown in Figure 5 is intended to guide the fabricator in setting up a preliminary fabrication process for estimating purposes, and the operations portrayed are not those which the fabricator may desire to use. No statements in this description are intended to be a part of the specification.

Special nuclear material will be received by the fabricator from PRDC in appropriate shipping containers. An initial check on composition and weight of uranium received will undoubtedly be desired for accountability. Before melting, the metal surfaces should be cleaned and the proportions of enriched and diluting uranium and molybdenum for each batch accurately weighed and possibly stored. From the development programs of APDA on small melts, indications are that vacuum induction melting may be used for alloying and adjusting the enrichment in small batches, possibly up to 200 pounds of 27 per cent enriched alloy fuel. Batches above 200 pounds weight should be considered with care because of safety requirements. The fabricator may select a convenient size for ingot casting.

Before hot reduction, it may be desirable to inspect the cast ingot for imperfections, purity, molybdenum content or

Figure 5
FLOW DIAGRAM FOR CORE PIN
FABRICATION

enrichment variation, or this may be done later as experience dictates. Other operations may be desirable at this point also, such as cutting off ingot ends or surface finishing. Hot reduction to about 0.9 inches may be done by rolling, extrusion or forging. The resulting billet is then cut into convenient length for coextrusion. Surface conditioning may be necessary before coextrusion. Since this is the last point in the fabrication cycle where bare uranium metal is handled, an accountability weight check will be required of the billets going into extrusion.

The finished bare uranium billet is inserted into a Zr tube, and the other extrusion billet components are assembled; the assembled billet is welded tight at the ends after evacuation. Other operations may be required during coextrusion billet assembly such as: degreasing of uranium and zirconium surfaces, oxidizing internal surfaces of steel sheaths to facilitate disassembly, and insertion of Zr spacer discs at ends between the uranium and steel. The entire billet is preheated to approximately 1700 F and coextruded to an approximate outside diameter of 0.25 inches on the zirconium clad. The steel end plug and billet sheath are removed by machining and/or pickling. The coextruded rods are cold swaged to finished pin diameter and cut to length. Each pin is straightened, and the ends are cold swaged to shape for end capping. The end caps are now mechanically locked on the pin by cold swaging to pin diameter. Each pin is annealed in a vacuum furnace for one hour at about 1475 F and furnace cooled. The finished pins are inspected non-destructively for acceptance and stored for assembly.

D. Fabrication Scrap Recovery

It should be recognized by the fabricator that the cost of recovering enriched uranium from non-remeltable fabrication scrap may be appreciable. With this fact in mind, a fabrication process which minimizes non-remeltable uranium scrap will afford the bidder a competitive advantage. Our only source of information on the cost of fabrication scrap recovery today is the Baker Company, with Mallinckrodt preparing to enter this business as well. Costs are rumored to be as high as \$1.50 per gram of contained U-235 for converting miscellaneous types of contaminated scrap to uranium metal melt stock. If this figure were used as a basis of cost of scrap recovery for a process which produced non-remeltable scrap in the volume of 10 per cent of the finished product, we would have 50 Kg of U-235 scrap material per core fabricated which would cost \$75,000 to return to uranium melt stock or over \$5.00 per core element fabricated chargeable to this operation.

Any process techniques which can be developed to reduce this indirect fabrication cost will be advantageous. In a careful examination of process operations, the fabricator may determine that certain types of his non-remeltable uranium scrap can be "cleaned up" by simple inexpensive operations in his own facilities. Your attention is called to the fact that our impurity specification permits 0.2 w/o Zr content in the fuel alloy, allowing the fabricator to remelt some Zr clad scrap.

E. Discussion of Specification Requirements

This portion of the report is intended to afford the fabricator a further understanding of our specification requirements as

stated in APDA Specification No. 30-8. Because of the special characteristics of fast breeder reactor design and fuel cycle costs, certain areas of our specification require greater than normal precision and manufacturing control. This is primarily due to the high specific power and heat transfer fluxes within the reactor core volume which necessitates obtaining core and blanket structures capable of thermal performance and irradiation stability exceeding that obtainable at this time in other reactor types already completed or under construction.

The APDA fuel engineering research and development programs completed and planned for the future have given confidence that end products conforming to this specification will meet our performance requirements. These development programs are described in detail in the Appendices. APDA intends to continue these development programs to answer any remaining questions on the performance of the core and blanket subassemblies. The fabricator is not responsible for performance of subassemblies fabricated to our specifications.

The following is a discussion of the specification requirements which appear to justify elaboration and explanation in the order that they appear in the specification. Wherever it is possible to do so, we have assigned a dollar value to tolerance limits to assist the fabricator in evaluating the cost of exceptions or waivers to the specification requirements.

1. Part I - Page 2: Fabrication Process - General

Casting - The fabricator may choose the size ingot casting to suit his available equipment and for the maximum

economic benefit in the fuel process he proposes. The approximate limit from a criticality standpoint for the PRDC fuel alloy is 200 pounds in one cast ingot, and the maximum size requires AEC approval. Crucible and mold materials used are limited only by the impurities they may release to the fuel material. Our experience on small melts indicates that adjustment of enrichment and alloying can be accomplished in a single vacuum induction melting operation with a zirconia-lined graphite crucible.

Initial Reduction of Casting - A hot reduction operation is required after the homogenization heat treatment of the cast ingot. This operation may be performed by any normal method such as rolling, forging or extruding. This operation is required to refine the grain size after the heat treatment. We have not specified the amount of reduction required, but it must be sufficient to make a fine-grained structure suitable for coextrusion with the zirconium clad sleeve. Other factors that may be considered in selecting the type of hot reduction operation are the resulting surface contamination and quantity of scrap created in machining the billet to cylindrical shape for coextrusion. The diameter of the hot reduced billet may be approximately 0.9 inches or larger but could vary depending upon the process of the fabricator.

Coextrusion - The hot reduced billets may be cut to any length to produce any multiple of core elements per extrusion. It should be noted that larger billet sizes will require fewer non-destructive test specimens as specified by the

specifications. The uranium alloy billet is encapsulated in a close-fitting zirconium sleeve and other billet components designed by the fabricator.

Extrusion is specified as the method of co-reduction because it is the only method that has been demonstrated in APDA development programs to produce a good metallurgical bond between fuel alloy and zirconium clad with resulting irradiation stability. The amount of area reduction necessary to achieve a sound metallurgical bond from our experience is 7 to 1 or higher. A metallurgical bond of high integrity is necessary because of the high heat transfer fluxes in this reactor type. A poor bond between fuel alloy and clad may cause a temperature rise as high as 1000 F above normal at that point with resulting decrease in irradiation stability and possible further complications. Satisfactory elements with good metallurgical bonds have been produced in development programs at The Babcock & Wilcox Company, Alliance, Ohio, (Appendix A) with an area reduction as low as 7 to 1. The elements produced in development programs at Nuclear Metals, Inc. (Appendix B) have utilized an area reduction ranging between 15 and 20.

The coextruded rod diameter has been specified between 50 per cent and 75 per cent greater than finished rod diameter for reasons given in the next paragraph.

Reduction of Coextruded Rod - Reduction to final diameter is specified by cold swaging. This operation was selected because it resulted in good surface finish, accurate control of finished element outside diameter, and it seems to be a

good test of the metallurgical bond integrity as extruded. Development experience has indicated that, in at least some cases of poor bonding, the cold swage operation will peel the thin Zr clad from the element.

End Capping - The swaged rod is cut to element length and pointed on the ends by further cold swaging. The fuel pin end caps shown in APDA Drawing 2XN-1704 are assembled onto the element ends and swaged down to element diameter, at the same time locking in place on the pin ends. The end caps serve both as structural support pieces and as a means of limiting fission gas leakage to the coolant sodium.

The method of attaching end caps to the core elements as specified is the reference design. APDA is presently pursuing development work to perfect the method of end cap attachment. Of all the methods tested to date, the present design offers the best promise of limiting leakage of fission gases recoiled from the exposed end surface of each element under the cap. This is of considerable importance to maintain tolerable activities in the external reactor coolant system.

2. Part I - Page 3: Materials

Initial Uranium Metal - The impurity content of the enriched and depleted metal supplied to the fabricator shall be the responsibility of PRDC. Present source of supply of enriched uranium, of course, is the Atomic Energy Commission, and analysis of typical broken metal buttons indicates impurity contents satisfactory to permit the fabricator to meet PRDC requirements in the cast ingot analysis, with the possible

exceptions of boron and hydrogen. Carbon is usually present in about 50-200 ppm. We are informed that carbon content can be held to 50 ppm by careful control in the metal reduction process.

Enrichment - The enrichment tolerances specified are limited by the control margin available in the PRDC reactor and the margin of the hot spot metal temperatures. The present tolerance of $\pm 1/4$ per cent U-235 enrichment amounts to a variation of $\pm 1-1/2$ subassemblies at the edge of a full core loading. In one refueling cycle, this may amount to ± 5 cents of reactor control margin out of a total available control margin of 30 cents per unloading cycle. If sufficient control margin is not available, the time between reactor refueling shutdowns is shortened, decreasing the plant load factor and increasing plant fixed charges per unit power produced. Because of the experimental nature of early operation of the PRDC reactor plant with the first core loading, we cannot easily determine the cost of such additional reactor downtime, but the fabricator should be cognizant that a waiver of the enrichment tolerances specified will involve some cost to PRDC in plant operation.

Uranium Alloy Composition - The tolerance on molybdenum alloy content specified is primarily due to the reactor control margin available and fuel cycle economics. The $10 \pm 1/2$ w/o Mo content permits a variation in U-235 loaded per core of $\pm$ approximately $4-1/2$ Kg, which is about the content of a full subassembly. In the initial core loading, a ± 0.5 w/o variation

in Mo content makes an uncertainty of ± 2 subassemblies in core size. On later reloadings when spent subassemblies are replaced with fresh subassemblies, ± 0.5 w/o Mo requires as much as ± 7 cents out of a total available 30 cents of reactivity available as control margin. This control margin must accommodate several other tolerances as well as that due to Mo.

Impurity contents specified are largely limited by metallurgical considerations and irradiation stability. Zirconium has been specified as 0.2 per cent to permit the fabricator the possibility of remelting fuel alloy scrap which has zirconium cladding attached thereto and because this amount does not affect irradiation stability of the fuel alloy. In the initial core loading, a variation from zero to 0.2 per cent zirconium content will require an extra 1-1/2 subassemblies to compensate for fuel alloy volume occupied by the zirconium, if enrichment has not been increased accordingly. Subsequent reloadings of fresh subassemblies would cost 5 cents of reactor control margin if containing 0.2 per cent zirconium when none was expected. In nearly all cases, the impurity content specified is not rigid, and suggestions from the fabricators are solicited if any of these are particularly difficult to meet.

Zirconium Cladding Composition - The impurity limits specified were set slightly above those normally found in reactor grade zirconium and should cause the fabricator no difficulty.

3. Part I - Page 5: Heat Treatment

The vacuum specified for heat treat operations is considered reasonable but not limiting. If the fabricator feels that an increase in the vacuum pressure is economically desirable without affecting the quality of the finished product adversely, PRDC will consider relaxation of this requirement.

The heat rate and cooling rate specified for heat treating initial castings is limited by thermal stresses generated in the material, and we consider this a reasonable rate for the heat treatment specified. The hold temperature specified has been found to be satisfactory for this alloy for a 24-hour hold period.

The heat treatment specified for completed core elements is one that has produced good irradiation stability for this type of core element. APDA tests have not indicated how great a tolerance can be used on the temperature and time for this heat treatment, and we have specified what are considered reasonable limits for temperature control.

4. Part I - Page 6: Finished Element

Dimensions - The element outside diameter tolerance specified is considered attainable by the process operations specified.

An increase of tolerance by .001 inch has the following effects on reactor operation: If the element outside diameter is smaller for only a few elements, this results in a slight temperature change of sodium coolant compared to adjacent coolant flow channels and a negligible effect on reactor costs; if the average outside diameter of all elements in the core is smaller, all fuel

cycle costs that are proportional to core life are increased by 1.2 per cent and this would mean an increase in Fermi reactor operation costs initially between \$5,000 and \$10,000 per year; if the element outside diameter is larger for only a few elements, irradiation stability of these elements is reduced because fuel alloy temperature increases and the fuel cycle is economically affected, but we cannot evaluate this since the relationships between irradiation stability and alloy fuel temperature are not known; if the average outside diameter of all elements in the core is greater, coolant pressure drop increases, coolant and fuel alloy temperatures increase, and irradiation stability is decreased; in addition to these direct effects of tolerance variation, there are indirect effects such as assembly interference, vibration of loose assemblies and effects on reactor control margin.

The average zirconium thickness tolerance is important economically, as it affects the amount of fuel alloy in the core loading, fuel cycle costs and reactor control margin. A variation of ± 0.0005 -inch clad thickness combined with a ± 0.001 -inch variation in pin outside diameter represents a total possible variation of $\pm$ approximately 9 Kg of U-235 per core. In the initial loading, this represents an uncertainty of ± 3 sub-assemblies at the edge of the core. During refueling, this variation is similar to a Mo content variation in its effect on control margin requirements, but it is twice as large, requiring as much as ± 13 cents of available control margin. One-third of this variation is due to the ± 0.0005 -inch clad tolerance and

two-thirds due to the ± 0.001 -inch pin outside diameter tolerance.

The minimum clad thickness at any point as specified is limited by the amount of fission product and uranium diffusion that can be tolerated through the cladding. This requirement has been extended to allow a minimum of .002 inch. The maximum clad thickness at any point is not a crucial tolerance or cause for rejection.

The element length tolerance specified is not rigid and can be relaxed if the fabricator considers that this tolerance results in higher costs.

The angle of element ends after pointing should be reasonably close to match the end caps to be attached. APDA development programs have not yet indicated what effects the tolerance variations on element point and end cap socket will have on heat transfer characteristics and integrity of the end cap.

Straightness - The test for permissible straightness is considered a reasonable limitation that can be met with the fabricated product. We find it difficult to pin-point the exact straightness required for a long thin pin, since it can be deflected elastically a considerable amount during assembly, resulting in a slight increase of locked-in stress in the element as loaded into the reactor core. Straightness difficulties arise largely from the swaging operation.

Surface Finish - The roughness limits for local areas are determined by the affect on coolant flow over the local surface, heat transfer from the element, and by the possibility of

increasing local stresses leading to splits in the cladding.

The average surface roughness is limited by pressure drop of coolant flowing past the elements at an average velocity of about 30 feet per second. This limitation is quite rigid because the installed capacity of our coolant pumps is fixed. AFDA flow tests indicate that a roughness of 100 micro-inches results in 75 psi pressure drop across the core, whereas 25 micro-inches corresponds to 65 psi drop. Therefore, with a fixed pressure drop across the core of 65 psi, higher average surface roughness values would result in an increase in coolant temperature rise through the core, higher thermal shocks when the reactor scrams, higher fuel alloy temperature and lower irradiation stability.

Flaws and lapping in the cladding are not acceptable because they act as stress raisers and traps for Na_2O build-up which could lead to splits in the zirconium cladding.

Weight - A permissible variation in gross weight of an element has been set at ± 1 per cent. This is an overriding tolerance established to prevent the effects of other tolerances accumulating in one direction. For example, without a weight tolerance, it is possible to obtain a large batch of elements with maximum Mo content, maximum Zr content, and minimum element diameter. A substitution of a number of subassemblies made of these pins for those with an opposite combination would require twice the control margin presently available for this and several other purposes in the reactor. This situation, as well as many more likely combinations of tolerances, will require extra

shutdown time to obtain necessary reactivity increases by edge loading.

By requiring a weight tolerance of ± 1 per cent, it is expected that the uranium content of an element will vary probably be within ± 1.5 per cent, thus limiting control margin requirements for Mo, Zr and size tolerances to a total of approximately ± 10 cents in reactivity when as many as 10 spent subassemblies are replaced with fresh subassemblies. (± 1 cent per average fresh subassembly). From the standpoint of reactor control, there would be a saving in shutdown time if this tolerance could be altered as follows: Require that each group of 144 elements selected for a subassembly be within ± 0.25 per cent of nominal gross calculated weight, using sorting and selection methods. PRDC would consider increasing the tolerance on the weight of the individual elements if a smaller tolerance were used on the total weight of the elements in the core subassembly.

5. Part I - Page 7: Cleanliness

Cleanliness of billet components necessary for good process control will be the fabricator's responsibility. We do not wish the oxide coating removed from the finished Zr clad element surface because cleaning agents would reduce initial clad thickness, and special shipping and storage procedures would be required to prevent new oxide formation on the zirconium surface.

6. Part I - Page 9: Tests - Preliminary

The fabrication of 500 depleted uranium alloy elements at PRDC expense has been specified to assist PRDC and the fabricator

to evaluate technical details of the fabrication process. It is also PRDC's intention to develop non-destructive tests during this period for use in inspecting the enriched elements. Our present thoughts in these tests are described in Appendix G.

The fabrication of a batch of 1000 enriched elements can then proceed, and the tests on these are performed to indicate satisfactory manufacturing process control for the remaining core elements. Again it is our intent to adjust process variables with a minimum expense of materials consumed. Close coordination of these two preliminary sets of tests will be maintained by PRDC so as to minimize downtime of the fabricator's equipment when set up for this production job. No long delays are expected in proceeding from fabricating the 500 depleted core elements, to the initial 1000 core elements fabricated with enriched materials, to the fabrication of the remaining enriched core elements, providing the fabricator produces the test information requested and the elements fabricated meet the specifications satisfactorily.

7. Part I - Page 10: Tests on Final Product

The table shown on this page is essentially self-explanatory as a summary of written description of tests in the Specification, Pages 11-16, Part I. This table lists only tests performed on finished core elements or on specimens taken after coextrusion and swaging. Tests for Mo content, impurity content, and enrichment isotopic analysis are performed on samples removed from the cast ingot as described in Part I, Pages 3 and 4, of the Specification. It has been the intent of PRDC to allow the

fabricator to select the method of testing for the various requirements listed in the Specification.

In the columns under Elements - 500 Depleted and First 1000 Enriched opposite the four items described as Specimens, the numbers of specimens listed are based on a coextrusion billet which produces two finished core elements. If a coextrusion billet size is such as to produce a different number of finished core elements, the number of specimens will change accordingly.

8. Part I - Page 11: Tests - Non-Destructive

Specimens - Any defects which are larger than .001-inch diameter in specimens or finished elements are considered cause for rejection on the basis that they are stress raisers likely to cause eventual core element failure in operation. The size imperfection selected is believed to be reasonable, and it is believed that the fabricator can meet this requirement.

Elements - The check for surface flaws is specified as a dye penetrant or alternate to be selected by PRDC and the fabricator. Care should be taken that no hydrogenous residue remains on the element surface after testing. The use of dye penetrant may leave these and other impurities imbedded in the Zr surface that are undesirable in the reactor. PRDC intends to examine other methods of surface testing besides dye penetrant.

The Specification did not include a requirement on concentricity of the element pointed ends or the hole in the end cap to receive the element end. A tolerance specification will be made on this, but the only apparent requirements of concentricity are such that the completed element will pass the straightness

test and that the anchor bar may be inserted through the lower end cap.

9. Part I - Page 16: Tests - Destructive

Each section of an element, selected for destructive testing, which contains an end cap must be examined after a longitudinal cut for (a) effect of the mechanical lock on the element (b) position of the end cap, (c) gap between the end cap and the element, and (d) cracks in the cap, cladding or fuel alloy. A basis for acceptance under such examination was not detailed in the Specifications but shall be determined at a later date by development programs under APDA. These Specifications will be written to insure that the end of the element will have irradiation stability and provide adequate heat transfer.

IV. BLANKET ROD ELEMENTS (Axial and Radial)

A. Description

The blanket elements are composed of one or more depleted U-2-3/4 w/o molybdenum alloy slugs, sodium for thermal bonding, stainless steel clad tubes, stainless steel end plugs, and an inert gas space above the sodium level. The axial blanket elements are shown on APDA Drawing 6XN-1719; the radial blanket elements are shown on APDA Drawing 6XN-1724. It should be noted that the radial blanket elements also include a spiral wire wound around the clad tube surface and attached at the end plugs.

The axial blanket elements as shown on APDA Drawing 6XN-1719 contain a total length, in one or more slugs, of 14 inches of U-2.75 w/o molybdenum alloy with a finished diameter of .415 inch. The stainless steel tube cladding is 16-3/4 inches long, .443-inch outside diameter and .010-inch nominal wall thickness; the stainless steel end plugs are dimensioned on the drawing. The sodium level in an assembled blanket element shall be 1 inch above the end of the slug at the top of the assembled element. The remaining space between the top of the sodium level and the end cap shall contain argon gas at atmospheric pressure.

The radial blanket slugs are composed of uranium alloyed with 2.75 w/o molybdenum fabricated to an outside diameter of .415 inch and a total length of one or more slugs making up each radial blanket element of 65 inches. The stainless steel cladding tube for each radial blanket element is 71-1/2 inches long with an outside diameter of .443 inch and a nominal wall thickness of .010 inch. The stainless steel end plugs are fabricated as shown on APDA Drawing 6XN-1724.

This drawing also shows the completed and assembled radial blanket element and all of its components. The sodium level in the assembled radial blanket element shall be 3 inches above the top of the blanket alloy slugs. The remaining space between the top of the sodium level and the top end plug shall contain argon gas at atmospheric pressure. A .039-inch diameter wire is spirally wound around the clad tube surface on a 9-inch pitch and attached by welding to the end plugs as shown.

B. Uranium Material Quality

PRDC will supply to the fabricator the depleted uranium necessary for the manufacture of blanket element slugs. The fabricator is responsible for procurement of the alloy material and stainless steel components. At the present time, the Atomic Energy Commission is the only source of depleted uranium metal in a broken button form suitable for melting and subsequent fabrication. However, PRDC will obtain the depleted uranium metal from a commercial source (Mallinckrodt Chemical Works, Davison Chemical, Baker Company, Nuclear Materials and Equipment or others) at the time it is needed for blanket slug fabrication.

Following is a typical impurity content of depleted uranium metal as supplied by Oak Ridge at the present time:

B - 2 ppm	Al - 50 ppm
Be - 0.5 ppm	Ca - 100 ppm
Cd - 1 ppm	Si - 300 ppm
Li - 5 ppm	C - 100 ppm

C. Typical Fabrication Process Outline

Figure 6 is a flow diagram which is considered typical for blanket element fabrication starting with the raw materials for the end caps, the stainless steel tubing, the uranium-molybdenum alloy slugs, the sodium and the argon inert gas. The typical flow diagram shown is not

FIGURE 6
FLOW DIAGRAM FOR BLANKET ELEMENT
FABRICATION

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for the purposes of binding the fabricator to a process but rather one which is suggestive of the process that may be used. The Specification requirements for detailed operations in the process allow the fabricator complete freedom in selecting ingot size for casting, fabrication, breakdown operations and assembly procedure.

The only limitation in his process is that it shall be one that will meet the purity requirements of the Specification. The assembly procedure is left as the complete responsibility of the fabricator, and the sodium bonding techniques used are also his responsibility. Welding operations on the end plugs to the stainless steel clad and of the spiral wire to the end plugs shall be according to APDA Specification. Procedures for sodium bonding which have been shown to be adequate to produce sodium bonds between uranium slugs and stainless steel tubes are described in Appendix I to this report. This appendix describes the development experience with sodium bonding as developed by The Babcock & Wilcox Company under contract to APDA. Argonne National Laboratory has also done extensive work to develop a sodium bonding technique for the fabrication of EBR-I Mark I and Mark II loadings and for the fabrication later of EBR-II core elements. North American Aviation has done extensive development of techniques for NaK bonding for the SRE fuel elements. The techniques developed by The Babcock & Wilcox Company are considered to be simple and inexpensive and have been shown to produce satisfactory bonds in 100 per cent of the cases tested. This technique is similar to that used by Argonne National Laboratory in fabricating EBR-I core elements.

D. Fabrication Scrap Recovery

The fabricator has the responsibility for recovering or disposing of non-remeltable depleted uranium scrap resulting from his fabrication process. Even though this material has a low dollar value compared to the enriched core fuel material, the fabricator will recognize that he will incur large inventory dollar losses if his fabrication process is wasteful, since the fabricator must reimburse PRDC for depleted uranium not returned in metal form. Some types of non-remeltable scrap may be reclaimed simply and inexpensively on the fabricator's premises. This is done in the case of bare uranium slug fabrication at the Fernald plant of the Atomic Energy Commission. This plant is operated by the National Lead Company. The fabricator must use his judgment in balancing fabrication process control expense against the cost of recovering and/or disposing of non-remeltable uranium scrap.

E. Discussion of Specification Requirements

1. Part II - Page 4: Materials

The specification for molybdenum alloy content is shown as 2.75 per cent $\pm$ 1/4 per cent by weight. The minimum molybdenum content as shown by irradiation tests must be 2.5 w/o to meet PRDC requirements. The average Mo content in the entire blanket should be as close to this as possible, but above 2.5 w/o; in order to maintain maximum breeding ratio. We have considered 2.75 w/o to be the average. If this tolerance of Mo content, 2.5 to 3.0 w/o, were increased to 2.5 to 3.5 w/o such that the average Mo content was 3.0 w/o, PRDC would incur a loss in plutonium production of \$35,000 in the first five years (which is approximately

equal to the average blanket element life). As a result, the fabricator must maintain the average molybdenum content of blanket uranium alloy as close to 2.5 w/o as possible without reducing the minimum content below 2.5 w/o, since this would affect the irradiation stability.

The impurity content limits are determined by metallurgical considerations largely and are considered reasonably attainable. If any of these element contents specified are not considered practical, the fabricator should so specify so that he and PRDC may mutually agree on the economic balance between reactor operation cost and fabrication process control loss.

Defects such as porosity, shrinkage cavities, voids, or cracks are limited to 1/32 inch maximum dimension from the standpoint of stress raisers. This limit is considered reasonable for the fabricator to attain.

The stainless steel components specified are commercially available.

Sodium purity has not been specified in the APDA Specification, but it must be clean enough to assure thermal bond integrity in all blanket elements fabricated, inspected and accepted. It is suggested that sodium be obtained in sealed drums under cover of inert gas such as argon or helium. Some difficulty has been experienced from sodium shipped under nitrogen cover gas due to nitriding of uranium surfaces after sodium bonding.

2. Part II - Page 6: Heat Treatment

The heat treatment procedure specified is tentative, pending test results from irradiation programs. Irradiation tests have

shown poor results from samples that were quenched from 850 C. Specimens which are slow cooled have shown good irradiation stability up to 1/2 a/o burn-up, which is the operating requirement for the PRDC reactor. The precise heating and cooling rates will be determined at a later date, but it is expected that the sum total of heat treatment time specified will not change appreciably from 19 - 20 hours.

3. Part II - Page 7: Detailed Requirements for Slugs

The diameter and length tolerances specified are not critical to reactor plant operation. If any of these can be relaxed to save fabrication cost appreciably, the fabricator should so state.

Straightness is only limited by problems of assembly to prevent damage to uranium alloy slug or stainless steel tube surfaces.

The above tolerances are considered practical with a fabrication process as shown in Figure 6.

The weight variation tolerance specified is believed to be reasonable and is the controlling tolerance on the slug. The density of the depleted U-2-3/4 w/o Mo alloy has not been determined yet, but will be supplied by PRDC at a later date.

4. Part II - Page 9: Detail Requirements for Finished Element

The dimensional tolerances for element outside diameter, element over-all length and spacer wire diameter are critical in order to permit assembly of the finished elements into the sub-assembly structure.

The finished element straightness tolerance set is considered a practical limit to permit assembly without damage to finished elements.

Surface finish limit specified is based on pressure drop considerations for coolant flow around the elements in the reactor. Development work has shown that the limit is easily attainable.

The sodium bonding procedure to be used is only limited to that necessary for heat transfer integrity and to produce the tolerance on sodium level above the uranium alloy slugs in the finished blanket element. Appendix I contains a report of results and practical considerations in a sodium bonding process developed by The Babcock & Wilcox Company for APDA. This development program is completed, and the results were 100 per cent successful with the procedure employed. This procedure is similar to that which was used by Argonne National Laboratory for sodium bonding elements for the first two loadings of EBR-I cores. It is simple, essentially fool-proof, and should give a high yield of satisfactory sodium bonds, if not 100 per cent yield.

5. Part II - Page 15: Addendum - Description of Fabrication Process

The following comments are offered to be of assistance to the fabricator:

Molybdenum particle sizes are important in obtaining alloy homogeneity. Fine particles tend to float to the surface of the melt where the tendency to alloy is less likely. Crushed particles, wire clippings and chips readily melt to result in good alloy homogeneity.

The use of an induction furnace for melting and alloying will help in obtaining good mixing and subsequent alloy homogeneity.

Zirconia crucibles or liners in crucibles have been used satisfactorily in development work with minimum impurity pick-up.

Heated graphite molds with zirconia washed surface treatment produce satisfactory castings. Chill casting into copper or steel molds may result in better surface condition of the cast ingot and may produce better homogeneity depending on the casting size.

V. CORE AND BLANKET SUBASSEMBLY STRUCTURES AND ASSEMBLY

A. General

The core subassembly structures are shown in final assembly on APDA Dwg. 6XN-1716 and 6XN-1732 and in detail on APDA Dwgs. 6XN-1721, 6XN-1720, 6XN-1717, 6XN-1718, and 6XN-1730. Details of the cartridge type wire spacer shown on APDA Dwg. 6XN-1730 are considered an alternate to the reference design cartridge type wire spacer shown on APDA Dwg. 6XN-1718.

The spacers designated as Item C on APDA Dwgs. 6XN-1716 and 6XN-1727 and shown in detail on APDA Dwgs. 6XN-1721 and 6XN-1728 are to be omitted from the present subassembly fabrication operations. Current development programs will determine how these spacers may be provided and what tolerances can be held. At a later date PRDC will negotiate the cost of adding these spacers with the fabricator.

The radial blanket subassemblies consist of two types: the inner row radial blanket subassembly "A" shown in final assembly on APDA Dwgs. 6XN-1727 and 6XN-1731, and in detail on APDA Dwgs. 6XN-1728 and 6XN-1729; the radial blanket subassembly "B" shown in final assembly on APDA Dwgs. 6XN-1723, and in detail on APDA Dwgs. 6XN-1725 and 6XN-1726.

The materials and manufacture of components is well described on the above listed drawings, all of which were included with the APDA Specification No. 30-8. The tolerances required are in most cases normal commercial practice. All of the subassembly operations have been performed in APDA development programs by various contractors, as described in the following paragraphs.

To assist the fabricator in bid preparation of the subassembly structure components which he may not be familiar with, we list the following sources of supply, which APDA has used in the past, that are familiar with our requirements:

1. The square tube housing for the core and blanket subassemblies was the most difficult item to locate, because the required tolerances are much closer than normal mill practice. Standard Tube Company, 24400 Plymouth Road, Detroit 39, Michigan has supplied a sample order of about 50 tubes, all of which meet tolerances required.

2. The cartridge type wire spacers have been manufactured for a similar design to that on APDA Dwg. 6XN-1718 by the K-J Metal Products Company, 3840 Beaver Ave., Detroit, Michigan. A former officer of a company that also did this work, but is no longer in business, is Mr. Charles J. Nagler, president of Metallurgical Consultants, Inc., 14513 Puritan, Detroit, Mich., and he indicates an interest in performing work of this type.

3. APDA is purchasing a complete loading of dummy subassemblies (core and blanket) for use in the APDA Test Facility operation prior to nuclear fueling of the reactor. These subassemblies are identical to the actual core and blanket subassemblies in every way except that they contain no internal structures or core or blanket elements. The following companies submitted lowest bids on these subassemblies:

Purolator Products, Inc., 970 Brunswick, Rahway, N. J.
Misco Fabricators, Inc., 2420 Wills Ave., Marysville, Mich.
Congress Tool and Die Div., Jann Corp., 3300 Denton Ave., Detroit, Mich.
Jackson and Church Co., 321 N. Hamilton St., Saginaw, Mich.
Brown-Hutchinson Iron Works, 1831 Clay, Detroit, Mich.

Two other firms, that have indicated an interest in this type of work, and have performed similar jobs for APDA, are:

Kay Industries, Inc., 1401 Twelfth St., Detroit, Mich.
Trimmer Tool and Die Co., 3110 Dewitt Road, Lansing, Mich.

B. Assembly Procedures1. Core Subassembly "C" - APDA Dwg. No. 6XN-1716 and 6XN-1732

All parts are to be thoroughly cleaned according to APDA Spec. No. 10-16. Place sixteen axial blanket pins (5 on 6XN-1719) in position on upper blanket retaining grid (9 on 6XN-1720), and slide into lower end of upper sub-assembly (8 on 6XN-1721) while holding all parts vertical. Push upper blanket retaining grid all the way up to the top, reaching down between the upper cam blades of the handling head to guide the axial blanket pins and the four corner wires into place on their spacing grid which is fastened to the lower side of the upper cam blades. When all parts are in position, the four corner wires are bent into position over the upper spacing grid and welded in place (see Section A-A on Sheet 6XN-1716).

The nozzle should next be assembled as follows:

Slide spring (4 on 6XN-1717) over lower inner subassembly (2 on 6XN-1717). Next slide lower outer subassembly (3 on 6XN-1717) over spring and lower inner subassembly and compress spring by loading with approximately 390 lbs. Flare the lower end of the lower inner subassembly over the nozzle tip of the lower outer subassembly. Spring length should then be not less than 5.313 inches and the lower inner subassembly should not bind on the lower outer subassembly.

The 16 lower axial blanket pins (5 on 6XN-1719) should next be assembled on top of the nozzle and the lower blanket retaining grid (1 on 6XN-1720) placed in position on top of the pins. The four corner wires of the lower blanket retaining grid should then be welded to the corners of the bottom support grid as shown in section G-G on 6XN-1716.

The fuel element bundle is next assembled as follows: Slide 144 fuel elements (6 on 5XN-1722) into the cartridge type wire spacer (7 on 6XN-1718) placing the slots in the pins to the bottom or anchor end of the wire spacer. Hold in place with the 12 locking bars (10 on 6XN-1718) which go into the slots of the wire spacer. Be sure that the X mark on the anchor bars is placed toward the upper end. Place the two locking plates (11 on 6XN-1718) in position to hold the anchor bars in place, being sure that the X mark is on the inside, and weld to the corner wires of the wire spacer. (See section F-F on 6XN-1716.)

Next place the fuel element bundle on top of the nozzle assembly so that the lower lugs of the cartridge type wire spacer and the locking plates rest on the lower blanket retaining grid. Bend the four corner wires of the cartridge type wire spacer slightly so that they are in position beside the four corner wires of the lower blanket retaining grid and weld in place. See Section H-H on 6XN-1716.

Clean all welds made during assembly before making final assembly.

Slide fuel pin bundle, lower axial blanket and nozzle assembly into the upper subassembly and align in jig to maintain concentricity requirements before making weld between upper subassembly and lower inner subassembly. Grind weld smooth and flush.

Give finished subassembly final cleaning before packaging for shipment.

2. Inner Row Radial Blanket Subassembly "A"

APDA Dwg. 6XN-1727 and 6XN-1731.

All parts are to be thoroughly cleaned (see APDA Spec. No. 10-16). Assemble nozzle first. Slide spring (4 on 6XN-1729) over lower inner subassembly (2 on 6XN-1729). Next slide lower outer subassembly (5 on 6XN-1729) over spring and lower inner subassembly and compress by loading with approximately 390 lbs. Flare the lower end of the lower

inner subassembly over the nozzle tip of the lower outer subassembly. Spring length should not be less than 5.313 inches and the lower inner subassembly should not bind on the lower outer subassembly.

Place 25 radial blanket elements (3 on 6XN-1724) on top of nozzle with the end caps marked "T" at the top and the slots on the bottom cap fitted over the support bars on the nozzle. Note that the helical spacer wires should all be oriented in the same direction. Slide the blanket elements into the upper subassembly (1 on 6XN-1728) and align in jig to maintain concentricity requirements before making weld between upper subassembly and lower inner subassembly. Grind weld smooth and flush.

Give finished subassembly final cleaning before packaging for shipment.

3. Radial Blanket Subassembly "B" - APDA Dwg. 6XN-1723.

All parts are to be thoroughly cleaned (See APDA Spec. No. 10-16)

Place 25 Radial Blanket Elements (3 on 6XN-1724) on top of lower subassembly (2 on 6XN-1726) with the end caps marked "T" at the top and the slots on the bottom cap fitted over the support bars on the lower subassembly. Note that the helical spacer wires should all be oriented in the same direction. Slide the blanket elements into the upper subassembly (1 on 6XN-1725) and align in jig to maintain concentricity before making weld between upper and lower subassembly. Grind weld smooth and flush.

Give finished subassembly final cleaning before packaging for shipment.

VI. GENERALA. Accountability Requirements for Special Nuclear Materials

The only accountability requirements set forth in the original specifications were contained in the cover letter on Commonwealth Associates, Inc. letterhead and in Addenda I to the General Conditions to Competitive Bid Contracts. It was specified that the fabricator is required to maintain his accountability records to 0.05 per cent by weight accuracy.

The fabricator will be responsible for determining and meeting any accountability requirements, in force at the time, of the Atomic Energy Commission. The latest information APDA has on these requirements is that the AEC will require an accounting in materials or dollars for Special Nuclear Materials delivered to a licensee and returned to the AEC after use. In other words, the licensee must reimburse the AEC for any material not returned. Since PRDC is the licensee in this case, PRDC will require this same accountability of the fabricator. The difference between the value of uranium melt stock supplied to the fabricator and uranium metal returned by him to PRDC shall be chargeable to the fabricator. The AEC may be expected to reserve the right to enter the fabricator's premises at any time to determine that SNM is being handled and controlled in such a way that no public health hazard is created.

PRDC then requires only that the fabricator be responsible for the dollar value of SNM lost while in his possession for fabricating fuel elements for PRDC. To determine these losses, it is necessary to accurately weigh all material received by the fabricator from PRDC and then by weighing and estimating determine the quantity delivered to PRDC either as finished fuel elements or remaining uranium metal melt stock or recycled uranium metal from chemical recovery.

The uranium broken melt stock received by the fabricator should be unpacked, weighed in the as-received condition and a sample or number of samples taken for analysis. If there is surface film or non-metallic particles on the surfaces, the fabricator should clean these surfaces to remove the undesirable material and once again weigh the material received. From these weight checks and the sample analyses, PRDC and the fabricator shall agree on the amount of SNM delivered to him.

The weight of enriched uranium metal contained in finished core fuel elements cannot be exactly determined. To estimate this amount, PRDC will require the fabricator to weigh the enriched uranium alloy and zirconium material going into coextrusion billets and to weigh the finished elements just prior to end capping. Samples taken from the cast ingots for determination of uranium isotope content, Mo content, and impurities content will be satisfactory for determining the weight of U-235 and U-238 in the uranium alloy assembled into coextrusion billets and in the finished core elements. Specimens taken from the ends of coextruded and swaged billets shall be used as a basis for determining average clad quantities on finished core fuel elements.

The quantity of depleted uranium delivered to PRDC in blanket elements shall be determined by weighing fabricated bare slugs before assembly into elements and by analysis of specimens taken from the cast ingots. The above is a general description of procedures to be used for accountability requirements of PRDC. Detailed measurements and samples for analysis may vary from those described, but the detailed procedure shall be agreed upon between PRDC and the fabricator at a later date when both have greater familiarity with the practical problems of the production process.

The fabricator will undoubtedly prefer to set up his own procedures for material control in process and scrap recovery because of the high

cost for losses of material in his possession. Our understanding of the AEC requirements and PRDC requirements as described above leaves the fabricator free to determine his own procedures beyond those specified.

Since accountability requirements were not set forth in the original specifications for core and blanket elements, the fabricator should describe in his bid proposal what accountability operations his bid price includes.

B. Use Charges

Under the present PRDC-AEC contract, use charges for SNM in the possession of PRDC will be waived during the period, including fabrication of the first core loading. Therefore, the fabricator should not include an item in his bid proposal for such use charge, but PRDC requires a detailed statement of the length of time the fabricator requires any quantity of PRDC enriched uranium in his possession. This is necessary because the AEC requires PRDC to list use charges on material in PRDC possession and ask for a waiver of these charges in their contract. This waived charge is then considered an AEC contribution to PRDC as part of the Power Reactor Demonstration Program.

C. Fabricator's Research and Development Responsibility

Extensive developmental programs have been completed, are in progress, and are planned by APDA to prove the integrity of the core and blanket fuel elements. These programs are described in detail in the Appendices to this report. The titles listed at the head of each Appendix in the Index are self-explanatory.

The fabricator is required to assume responsibility for performance of developmental work necessary to establish the production operations to

manufacture PRDC core and blanket subassemblies. In some phases, such as subassembly structures, production processes have already been established. In other phases, such as coextrusion and swaging of finished core fuel elements, it will be necessary for the fabricator to use his production experience to bridge the gap from laboratory experimental or semi-production work to that of full scale production. APDA and PRDC will cooperate in assisting the fabricator in the performance of this work. The fabricator will have free access to all development work on core and blanket elements to be performed by APDA or PRDC for the Enrico Fermi Breeder Power Reactor.

D. Shipment of Completed Subassemblies

The vendor will be completely responsible for the shipment of the core and blanket subassemblies to the Fermi plant site near Monroe, Michigan.

The shipping containers must meet the following requirements:

1. A critical mass formation should be impossible even if the containers are placed side by side.
2. They must be approved by the AEC.
3. Although a low level of alpha radiation will be present, normal packaging should offer sufficient shielding.

A courier is not mandatory for civilian fuel shipments, but it is recommended that a guard be used because of the monetary value of the material.

The NEPIA (Nuclear Energy Property Insurance Associates) will insure nuclear fuel in transit.