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POTENTIAL OF DUPLEX FUEL IN PREBREEDER, BREEDER, AND POWER REACTOR DESIGNS: TESTS AND ANALYSES

(AWBA Development Program)

SEPTEMBER 1982

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BETTIS ATOMIC POWER LABORATORY

PITTSBURGH, PENNSYLVANIA 15122-0079

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(AWBA Development Program)

MASTER

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FOREWORD

The Shippingport Atomic Power Station located in Shippingport, Pennsylvania was the first large-scale, central-station nuclear power plant in the United States and the first plant of such size in the world operated solely to produce electric power. This program was started in 1953 to confirm the practical application of nuclear power for large-scale electric power generation. It has provided much of the technology being used for design and operation of the commercial, central-station nuclear power plants now in use.

Subsequent to development and successful operation of the Pressurized Water Reactor in the Atomic Energy Commission (now Department of Energy, DOE) owned reactor plant at the Shippingport Atomic Power Station, the Atomic Energy Commission in 1965 undertook a research and development program to design and build a Light Water Breeder Reactor core for operation in the Shippingport Station.

The objective of the Light Water Breeder Reactor (LWBR) program has been to develop a technology that would significantly improve the utilization of the nation's nuclear fuel resources employing the well-established water reactor technology. To achieve this objective, work has been directed toward analysis, design, component tests, and fabrication of a water-cooled, thorium oxide-uranium oxide fuel cycle breeder reactor for installation and operation at the Shippingport Station. The LWBR core started operation in the Shippingport Station in the Fall of 1977 and will finish routine power operation on October 1, 1982. After End-of-Life core testing, the core will be removed and the spent fuel shipped to the Naval Reactors Expended Core Facility for detailed examination to verify core performance including an evaluation of breeding characteristics.

In 1976, with fabrication of the Shippingport LWBR core nearing completion, the Energy Research and Development Administration, now DOE, established the Advanced Water Breeder Applications (AWBA) program to develop and disseminate technical information which would assist U. S. industry in evaluating the LWBR concept for commercial-scale applications. The AWBA program, which is concluding in September, 1982, has explored some of the problems that would be faced by industry in adopting technology confirmed in the LWBR program. Information already developed includes concepts for commercial-scale prebreeder cores which would produce uranium-233 for light water breeder cores while producing electric power, improvements for breeder cores based on the technology developed to fabricate and operate the Shippingport LWBR core, and other information and technology to aid in evaluating commercial-scale application of the LWBR concept.

All three development programs (Pressurized Water Reactor, Light Water Breeder Reactor, and Advanced Water Breeder Applications) have been conducted under the technical direction of the Office of the Deputy Assistant Secretary for Naval Reactors of DOE.

Technical information developed under the Shippingport, LWBR, and AWBA programs has been and will continue to be published in technical memoranda, one of which is this present report.

ACKNOWLEDGEMENTS

The authors wish to recognize Mr. T. H. Alff for his careful review of and comments on this report, Dr. L. A. Waldman for his effort in correlating the rod bowing data to axial strains and in comparing the length changes between LDR and PBIT test rods, Mr. C. D. Sphar for his assistance in understanding the irradiation data, Mr. R. J. Kushner for the rod performance analyses during LOCA transient, Dr. R. Lloyd for the dissolution tests on UO_2 and ThO_2-UO_2 , Mr. M. J. Smithnosky for the new pellet stacking concept, and Mr. J. Dunyak for his assistance in understanding the results from the CYGRO-5 analyses.

TABLE OF CONTENTS

	<u>Page</u>
ABSTRACT	1
I. INTRODUCTION	1
II. AWBA FUEL AND FUEL ROD IRRADIATION TESTS	5
A. Short Rods	6
1. D-1 Test	6
2. ALT-1 Test	6
3. ALT-2 Test	6
B. Long Rods	6
1. LDR Test	6
2. NLDR Test	7
3. PBIT Test	7
III. ADVANTAGES OF DUPLEX PELLET FUEL RODS	7
A. Performance	7
1. Reduction in Fuel Temperature	7
2. Improvement in LOCA Performance	8
3. Reduction in Rod Elongation	10
4. Reduction in Fuel Rod Bowing	18
5. Reduction in Axial Gap	18
6. Reduction in Rod Diameter Changes	20
7. Reduction in Cladding Tensile Stresses	21
8. Reduction in Fission Gas Release at Low Depletion	23
B. Extraction of U-233 from Spent Prebreeder Fuel	24
IV. CONCERNS AND DISADVANTAGES WITH DUPLEX FUEL RODS	25
A. Performance	26
1. Cladding Ridging and Hydriding in Defected Rods	26
2. Fragility of Annulus under Axial Load	29
3. Aggravation of Waterlogging in Defected Rods	29

TABLE OF CONTENTS (Cont'd)

	<u>Page</u>
4. More Fuel Oxidation in Defected Rods	31
5. High Fission Gas Release Rate at High Depletion	32
6. Effect of High Depletion on Fuel Structure	33
7. An Annulus Chip Lodging Between Center Pellets	33
8. Fuel Redistribution by Evaporation and Condensation	33
B. Manufacturing	34
V. SUMMARY AND CONCLUSIONS	35
A. Advantages	35
B. Concerns and Disadvantages	36
REFERENCES	37

LIST OF FIGURES

<u>Figure</u>	<u>Title</u>	<u>Page</u>
1	Schematic Diagram of a Duplex Pellet	4
2	Transverse Metallographic Samples from ThO ₂ -UO ₂ D-1 Rods 97-5 and 97-17 with Maximum Peak Powers of 15.2 and 17.8 kw/ft respectively	9
3	Comparison of Calculated Cladding Surface Temperatures between Duplex and Solid Pellet Rods during A LOCA Transient	11
4	Comparison of Overall Length Changes between LDR Duplex and PBIT Solid Pellet Rods with SRA Cladding - Fluence	14
5	Comparison of Overall Length Changes between LDR Duplex and PBIT Solid Pellet Rods with SRA Cladding - Depletion	15
6	Comparison of Overall Length Changes between LDR and LWBR Test Rods: Duplex versus Solid Pellet Rods	17
7	Peak Power Span Creep Bow versus Peak Power Span Elongation	19
8	Comparison of Peak Diameter Changes between LDR and LWBR Test Rods: Duplex versus Solid Pellet Rods	22
9	Intentionally Defected D-1 Rod 97-48D External Appearance	27
10	Schematic Diagram of Variable Diameter Duplex Pellets	28
11	Typical Neutron Radiograph of A Section of UO ₂ Duplex Fuel Pellet Stack of LDR Rod 97-53	30

LIST OF TABLES

<u>Table</u>	<u>Title</u>	<u>Page</u>
1	Summary of AWBA D-1, ALT-1, and ALT-2 Rod Length and Diameter Changes and Rod Test Parameters	13

Dual region fuel pellets, called duplex pellets in this document, are comprised of an outer annular region of relatively high uranium fuel enrichment and a center pellet of fertile material with no enrichment. UO_2 and ThO_2 are the fissile and fertile materials of interest. Both prebreeders and breeders are discussed as are the performance advantages of duplex pellets over solid pellets in these two pressurized water reactor types. Advantages of duplex pellets for commercial reactor fuel rods are also discussed. Both irradiation test data and analytical results are used in the comparisons. Performance advantages stem primarily from the lower pellet temperature and also from the pellet configuration. Advantages include lower cladding tensile stresses for the majority of life, lesser length and diameter changes, better response during LOCA, better bowing performance, shorter axial gaps in the fuel stack, and enhanced chemical separation as required in prebreeding. Satisfactory performance of duplex pellet fuel rods has been demonstrated by irradiation tests. However, irradiation testing did identify cladding ridging and hydriding adjacent to cold thoria spacer pellets in intentionally defected fuel rods as concerns. Manufacturing of duplex fuel is discussed.

POTENTIAL OF DUPLEX FUEL IN PREBREEDER, BREEDER,
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I. INTRODUCTION

Fuel rods containing dual region pellets have demonstrated good performance characteristics based on data from in-pile testing and analytical calculations. These data and calculations indicate that for a given power level and a given set of pressurized water reactor environmental conditions dual region pellet fuel rods, hereafter called duplex pellet fuel rods, exhibit the following

characteristics as compared to single region pellet fuel rods: small diameter and length changes, a delayed pellet-cladding interaction (PCI) strong enough to induce significant tensile stresses in the cladding, an improved up-power transient capability, and a superior response to a loss-of-coolant accident (LOCA). All of these advantages result either from the inherently low temperatures of duplex pellets or from the separation of the two fuel regions (a highly enriched fuel annulus and a fertile center pellet of no enrichment) by a finite diametral gap. Other noteworthy characteristics, throughout most of core lifetime, are a lower fission gas release, lower steady-state cladding stresses, better bowing performance, and an enhanced capability for separation of fissile fuel bred in the light water prebreeder reactor system.

While the data reported in this document and the analyses used to interpret the data derive from U-235 and thorium fuel systems for the light water breeder concepts, the performance advantages of duplex pellets are applicable to commercial UO_2 fuel systems. Small diameter and length changes, delayed strong PCI, lower stored energy at the instant of a LOCA, and an increased capability in up-power transients are all highly desirable in commercial cores. The observed performance advantages of duplex pellets are consistent with the British and Danish experience with duplex fuel as reported in Reference 1.

In the light water breeder reactor conceptual systems, reactor cores are divided into two categories: the prebreeders and the breeders. Prebreeders use a combination of enriched U-235 fuel and fertile thorium to produce U-233, while generating power. U-233 does not occur naturally in any significant quantity and must be produced by transmutation from thorium. The U-233 produced is extracted from spent prebreeder fuel at the end of the fuel cycle. This fissile U-233 together with fertile thorium is used to fuel breeders which generate power while producing enough U-233 from thorium to replace the U-233 fuel used. Breeders are self-sustaining in that each core produces enough U-233 to fuel a successor core, allowing for 1% loss during the reclamation process.

An important consideration in the case of prebreeders is the ability to extract U-233 chemically from spent fuel without contamination by U-235 and higher mass uranium isotopes. Prebreeder designs using separate rods for fertile (ThO_2) and fissile (UO_2) materials achieve the desired separation but do not meet the desired core power densities when the thorium fraction in the core

is high enough for efficient breeding. Consequently, alternative approaches that achieve performance characteristics comparable to a pure UO_2 system while providing U-233 uncontaminated by U-235 and higher isotopes have been pursued.

One method of achieving this objective is the use of fuel rods containing duplex pellets. This system provides both the desired core power densities and the needed separation of U-233, since the UO_2 annulus can be chemically separated from the $\text{ThO}_2\text{-U}^{233}\text{O}_2$ center pellet. A more detailed discussion of the system is given in Section III in this report.

Figure 1 shows the main features of a duplex pellet. As noted earlier the annulus is a region of relatively high uranium enrichment while the center pellet is a region of fertile material with no enrichment. The low temperatures experienced in these pellets are accounted for by the fact that the power producing fissioning occurs primarily in the annulus which is closer to the coolant, the heat sink, than is the case for conventional solid* pellet fuels. Figure 1 also shows an initial fuel annulus-to-center pellet gap. Based on the test data and the parallel analytical work, it has been observed that the order of gap closure (annulus-cladding and annulus-center pellet) is an important attribute of duplex pellets. It is postulated that the annulus-cladding gap first closes as a function of initial gap size, power level, and annulus swelling rate. After this gap has closed the annulus begins to swell inward and eventually closes the annulus-center pellet gap. Only after both gaps have been closed is there evidence of strong PCI in the sense that there is an increase in rod diameter. Because these initial gaps can be controlled during manufacturing and assembly of the fuel rods the strong PCI can be delayed until late in the core lifetime or, perhaps, until the end of core lifetime.

The purpose of this report is to highlight advantages of duplex pellet fuel rods demonstrated by irradiation testing and analysis and to present an assessment of concerns and possible problems associated with duplex pellet fuel rods.

*A solid pellet is defined as a simple pellet with a radially uniform composition in this report. It may be possible to make a solid pellet with ThO_2 in the center and fuel such as UO_2 on the outside. For the purpose of nomenclature, this configuration would still be defined as a duplex pellet.

Figure 1. Schematic Diagram of a Duplex Pellet

A number of performance concerns relating to the duplex pellet configuration have been postulated. These concerns are fragility of the annulus under axial load, aggravation of waterlogging in defected rods, increased fuel oxidation due to more high temperature surfaces in defected rods, high fission density in the annulus and its effects on gas release and fuel structure, consequence of an annulus chip lodged between center thoria pellets, and fuel redistribution by evaporation and condensation. Irradiation test results, to a burnup of 16.7×10^{20} f/cc of compartment ($\sim 33.4 \times 10^{20}$ f/cc of annulus) in undefected rods and 11.2×10^{20} f/cc of compartment ($\sim 22.4 \times 10^{20}$ f/cc of annulus) in defected rods, have indicated satisfactory performance for duplex pellet rods relative to these concerns. However, irradiation testing did identify cladding ridging and hydriding adjacent to cold thoria spacer pellets in defected rods as a potential concern. The spacer pellets were used to maintain axial registry of center pellets with respect to fuel annuli. This effect can be eliminated by using a fuel stack design which does not need thoria spacer pellets. Performance concerns and potential problems are discussed in Section IV. In addition, complexity in manufacturing of duplex fuel, a disadvantage relative to manufacturing of solid pellet, is discussed in Section IV.

AWBA fuel and fuel rod irradiation tests, described in Section II, were designed for two major purposes: one, to explore basic responses and mechanisms of duplex pellet fuel and fuel rods; two, to be prototypic of AWBA prebreeder and breeder designs. For those reasons test parameters were not selected to afford a rigorously direct comparison with solid pellet fuel rods of comparable power capability, depletion, and fast neutron fluence. However, as noted in this report, although comparisons are not always rigorous, a body of data and trend of performance strongly supports the superior performance of duplex pellet fuel rods.

II. AWBA FUEL AND FUEL ROD IRRADIATION TESTS

AWBA fuel and fuel rod irradiation tests can be divided into two general groups: short rods and long rods. The short test rods had an overall length of 11 inches and an outside diameter of 0.3 inch. The long test rods had an overall length of 90 to 110 inches and an outside diameter of 0.3 to 0.37 inch. To facilitate data discussion, a brief description of these tests is given in this section.

A. Short Rods

1. D-1 Test - The Duplex Pellet Rod Test - No. 1 (D-1) screening irradiation test, described in Reference 2, consisted of forty-four fuel rods. Each rod contained an 8-inch long pellet stack. The prime design variable was the fuel annulus composition. Four compositions were included: UO_2 , $\text{ZrO}_2\text{-UO}_2$, $\text{ThO}_2\text{-UO}_2$, and $\text{ZrO}_2\text{-UO}_2\text{-CaO}$.
2. ALT-1 Test* - The Alternate Fuel System - No. 1 (ALT-1) screening irradiation test consisted of twenty-two fuel rods. The ALT-1 test was designed to extend the D-1 duplex pellet rod test series to include (1) high power operation of solid $\text{ThO}_2\text{-UO}_2$ and UO_2 pellets, (2) alternating fissile and fertile pellet stack, and (3) up-power transient testing of candidate fuel systems.
3. ALT-2 Test* - The Alternate Fuel System - No. 2 (ALT-2) screening irradiation test consisted of eighteen fuel rods. The ALT-2 test extended the AWBA screening test program to include (1) up-power transient testing of $\text{ThO}_2\text{-UO}_2$ solid pellet rods, (2) steady-state testing of rods with small fuel-cladding gaps operated at high power levels, and (3) defect operation of candidate fuel systems not covered in D-1 and ALT-1 tests.

B. Long Rods

1. LDR Test - The Long Duplex Rod (LDR) irradiation test, described in Reference 3, consisted of six duplex pellet fuel rods. The overall lengths of these rods ranged from 90 to 92 inches but the active fuel length was 60 inches of which 48 inches was irradiated. The rod diameter was 0.3 inch. This test was designed to determine performance capability of duplex pellets in long rods and thus complements the D-1 short rod screening test. Test variables included different fuel annulus compositions (UO_2 , $\text{ThO}_2\text{-UO}_2$, and $\text{ZrO}_2\text{-UO}_2\text{-CaO}$), cladding material condition, pre-pressurization level, fuel stack arrangements with and without thorium spacers, intentional defect operation, and graphite coating on the internal cladding surface.

*ALT test data were obtained from Mr. C. D. Sphar.

2. NLDR Test - The New Long Duplex Rod (NLDR) irradiation test, described in References 4 and 5, consisted of six duplex pellet fuel rods tested two at a time in the NRX reactor at Chalk River, Canada. The overall rod length was 110 inches and the rod diameter was 0.3 inch. The principal objectives of this test were (1) to compare the performance of UO_2 , $\text{ZrO}_2\text{-UO}_2\text{-CaO}$, and $\text{ThO}_2\text{-UO}_2$ duplex pellet fuel systems at power levels adjusted to reflect relative melting points, (2) to compare the performance of a long UO_2 fueled rod with 3.6-mil fuel-cladding diametral gap with another UO_2 fueled rod with 6.2-mil gap, (3) to compare the performance of a long rod containing one plenum spring at the top of the pellet stack with another rod containing two plenum springs, one at the top and the other at the bottom, and (4) to assess the effect of duplex pellets with center thorium pellets as long as or longer than their associated fuel annuli.
3. PBIT Test - The Prebreeder Bundle Irradiation Test (PBIT) consisted of twenty rods fueled with solid pellets of $\text{ThO}_2\text{-UO}_2$ or UO_2 . The overall rod length was 94 inches but the active fuel length was 60 inches. The diameters of these test rods were about 0.37 inch, which is larger than the 0.3-inch rod diameter used in all the above tests. The purpose of the PBIT irradiation test was to demonstrate fuel rod performance capability with the $\text{ThO}_2\text{-UO}_2$ solid pellet fuel system and compare this system with a UO_2 commercial type solid pellet fuel system.

III. ADVANTAGES OF DUPLEX PELLET FUEL RODS

A. Performance

The performance advantages of fuel rods containing duplex pellets compared with fuel rods containing solid pellets, based on results from irradiation tests and analytical evaluations, are as follows:

1. Reduction in Fuel Temperature

In normal reactor operation, a duplex fuel pellet operates with lower peak fuel temperature than that of a solid fuel pellet at the same linear power with the same fuel-cladding gap. For example, calculated peak fuel temperatures for a duplex and a solid pellet fuel rod are 2625°F and 4650°F, respectively, at

15.2 kw/ft at BOL (beginning-of-life) and 2760°F and 3580°F at 9.6 kw/ft at EOL (end-of-life) for a depletion of 12×10^{20} f/cc of compartment. The fuel annulus to center pellet volume ratio was about 1:1 in this duplex pellet design. The solid pellet was taken as UO_2 , and the duplex pellet had a fuel annulus of UO_2 and a center pellet of ThO_2 . Typical transverse metallographic samples from irradiated duplex pellet fuel rods are shown in Figure 2. At BOL, the ThO_2 center pellet does not contribute to power generation but an appreciable power contribution is made at EOL. ThO_2 melts at about 1000°F higher temperature than UO_2 , and thus there is a large margin to melting. These temperatures were calculated using the CYGRO-5 fuel rod analysis computer program (Reference 6) and using the power history of a typical commercial power reactor core design. The significant reduction in the calculated peak fuel temperatures is caused by the close proximity of the heat generating fuel annulus to the heat conducting cladding.

A reduction in peak fuel temperature in duplex fuel rods has three direct advantages: larger power margin to fuel coring and melting, improvement in LOCA performance, and delayed and/or reduced pellet-cladding interaction (PCI). Other performance advantages which are discussed later in this section are also indirectly related to the reduced fuel temperature in duplex pellets.

2. Improvement in LOCA Performance

Fuel pellet temperature at power has a large effect on cladding temperature during a LOCA transient due to the stored energy in the pellet and the physical properties of the pellet. For the same power level and the same fuel-cladding gap, a duplex pellet operates at a lower temperature than does a solid pellet, as discussed above. This implies that a duplex pellet would have a lower stored energy at power, and, therefore, the cladding of a duplex pellet fuel rod would be expected to experience a lower temperature during a LOCA than that of a solid pellet rod. For example, the calculated energies stored in a duplex and a solid pellet are 51 and 81 Btu/ft, respectively, at 15.2 kw/ft at BOL and 45 and 55 Btu/ft at 9.6 kw/ft at EOL. These are the energies stored above a 650°F reference temperature. The fuel rods used in the calculation had 0.38-inch OD, 0.33-inch cladding ID, and 0.008-inch fuel-cladding gap.

Rod 97-5

Depletion - 1.4×10^{20} fissions/cc-compartment

Rod 97-17

Depletion - 9.3×10^{20} fissions/cc-compartment

Note: Notch in cladding is identification mark made during destructive examination.

Figure 2. Transverse Metallographic Samples from $\text{ThO}_2\text{-UO}_2$ D-1 Rods 97-5 and 97-17 with Maximum Peak Powers of 15.2 and 17.8 kw/ft, respectively

LOCA analyses with the FLASH-6 computer program (Reference 7) confirmed the expected decrease in cladding temperature of a duplex pellet fuel rod. For example, at 15.6 kw/ft the peak cladding temperature of a duplex pellet rod during a LOCA is predicted to be significantly lower than that of a solid pellet rod during blowdown (1220°F vs. 1550°F) as shown in Figure 3. This is mainly due to lower stored energy in duplex pellets because of lower fuel temperature. Some improvement also exists during refill (1790°F vs. 1880°F) as shown in Figure 3. The inside diameter of the reactor vessel used in the calculation was 180 inches. For breeder and prebreeder designs with a larger reactor vessel (250 to 279 inches I.D.), the stored energy in fuel rods is removed during a longer blowdown period. Therefore, the advantage of duplex pellets during a LOCA would not be as great. However, performance of duplex pellets during a LOCA would always be superior to that of solid pellets.

3. Reduction in Rod Elongation

Length changes measured from irradiation test rods indicate a significant reduction in rod elongation for duplex pellet fuel rods compared with solid pellet fuel rods. The major feature which results in smaller length changes for duplex pellet rods is associated with the duplex pellet design. This annulus-center pellet arrangement results in lower fuel temperatures and therefore a smaller axial fuel-cladding interaction to stretch the cladding. Because of the large number of test parameters, a rigorous comparison between individual test rods is not feasible. Therefore, a summary comparison of rod elongation measurements is presented in this section by grouping test parameters into ranges.

A smaller rod elongation has the beneficial effect of a smaller axial gap in the fuel pellet stack and less rod bowing, as discussed in Sections III.A.4 and III.A.5.

a. Short Rod Tests

In the fuel screening test program, eighty-four short rods were irradiation tested in the D-1, ALT-1, and ALT-2 series of tests. Overall rod length was 11 inches and the fuel stack length was 8 inches. The nominal rod diameter was 0.3 inch. Test parameters that were varied among the test rods are as follows:

Zircaloy-4 cladding - highly cold worked and either stress relief annealed (SRA) or recrystallization annealed (RXA)

Figure 3. Comparison of Calculated Cladding Surface Temperatures between Duplex and Solid Pellet Rods during A LOCA Transient

Pellet type - duplex and solid
Pellet stacking arrangement - thickness and frequency of spacer pellets
Rod power
Fuel depletion
Fast neutron fluence
Fuel-cladding gap
Fuel annulus-center pellet gap
Cladding thickness
Rod pre-pressurization
Fuel composition
Intentionally defected cladding
Graphite coating at cladding internal surface

The ranges of these parameters are given in Table 1. Also given in the table are the ranges of measured rod length and diameter changes. Rod diameter changes are discussed in Section III.A.6.

Table 1 shows that the ranges of measured rod elongation are 10.9 to 25.6 mils and 17.1 to 32.8 mils for the duplex and solid pellet rods, respectively, with SRA cladding and 6.8 to 16.9 mils and 10.7 to 22.8 mils with RXA cladding. For both SRA and RXA cladding, the solid pellet rods as a group elongated more than the duplex pellet rods: about 28% more for SRA clad rods and 35% more for RXA clad rods, calculated with the upper range elongations.

b. Long Rod Tests

Rods with SRA Cladding - Comparison of length increases for duplex pellet LDR test rods with those for solid pellet PBIT test rods is shown in Figures 4 and 5. These rods operated with similar power and fuel-cladding gaps. The PBIT test rods were fabricated with SRA cladding. Thus, LDR rods 97-53 and 97-58, both with SRA cladding, are used for comparison. Length changes in Figure 4 are plotted against fast fluence, which is considered to be more significant in the case of axial strains produced mostly by neutron flux induced growth. Since test depletion could influence cladding elongation in the presence of pellet-cladding interaction, such as observed in the diametral traces of rods 97-53 and 97-58, the data are also plotted as a function of depletion in Figure 5. The data for the test rods are connected by straight lines since intermediate data were not obtained. Figures 4 and 5 show that the length changes for duplex pellet LDR rods are substantially less than the values for solid pellet PBIT rods.

TABLE 1 - SUMMARY OF AWBA D-1, ALT-1, and ALT-2 ROD LENGTH AND DIAMETER CHANGES AND ROD TEST PARAMETERS

	SRA Zircaloy-4 Cladding		RXA Zircaloy-4 Cladding	
	Duplex	Solid	Duplex	Solid
Peak Power, kw/ft	12-22	12-22	12-18	6-18
Fluence, 10^{20} n/cm ²	13.4-25.0	10.3-25.2	7.5-25.0	10.7-25.1
Depletion, 10^{20} f/cc-comp	4.88-8.33	2.91-8.38	2.7-8.33	1.67-8.22
Fuel-Cladding Diametral Gap, mils	2.0-6.0	2.7-5.0	2.0-8.4	2.5-15
Fuel Annulus-Center Pellet Gap, mils	2.0-10	-	2.0-10	-
Rod Length Increase, mils	10.9-25.6	17.1-32.8	6.8-16.9	10.7-22.8
Rod Diameter Change, mils	-1.26 to +0.85	-0.55 to +2.60	-0.58 to +0.28	-0.70 to +0.92

NOTE: SRA: 50% cold worked and stress relief annealed at 925°F for 4 hours
 RXA: 50% cold worked and recrystallization annealed at 1225°F for 4 hours

Number of rods tested: 84

Rod diameter: 0.3 inch

OD/t ratio: 13.7-16.3

Number of rods with graphite coating: 26

Rod pre-pressurization: 0, 100, 300, and 500 psig

Fuel system: UO_2 , ThO_2-UO_2 , ZrO_2-UO_2 , ZrO_2-UO_2-CaO , and ThO_2 (center thoria pellet in duplex pellet)

Overall rod length: 11 inches

Pellet Stack length: 8 inches

			BOL PEAK POWER (KW/FT)	FISSILE FUEL	INTERNAL PRESSURE (PSIG)	FUEL- CLADDING GAP (MILS)
PBIT TEST	○	ROD 97-145	15.4	ThO ₂ -UO ₂	400	7.3
SOLID PELLETT	□	ROD 97-147	15.2	UO ₂	400	7.9
LDR TEST	▲	ROD 97-53	13.9	UO ₂	100	5.2-6.3
DUPLEX PELLETT	◆	ROD 97-58	16.9	ThO ₂ -UO ₂	500	4.9-5.7

Figure 4. Comparison of Overall Length Changes between LDR Duplex and PBIT Solid Pellet Rods with SRA Cladding - Fluence

			BOL PEAK POWER (KW/FT)	FISSILE FUEL	INTERNAL PRESSURE (PSIG)	FUEL- CLADDING GAP (MILS)
PBIT TEST	{	○ ROD 97-145	15.4	ThO ₂ -UO ₂	400	7.3
SOLID PELLETT	{	□ ROD 97-147	15.2	UO ₂	400	7.9
LDR TEST	{	▲ ROD 97-53	13.9	UO ₂	100	5.2-6.3
DUPLEX PELLETT	{	◆ ROD 97-58	16.9	ThO ₂ -UO ₂	500	4.9-5.7

Figure 5. Comparison of Overall Length Changes between LDR Duplex and PBIT Solid Pellet Rods with SRA Cladding - Depletion

LDR rods 97-53 and 97-58 had graphite coating on the cladding inside surface whereas the solid pellet PBIT rods were not graphite coated. While graphite coating could contribute to the difference in length change observed between the duplex and solid pellet rods, it would not be expected to produce the large difference in length changes, presented above, based on the effect of graphite coating on length change noted in Reference 3.

The difference in rod diameters between the duplex and solid pellet rods (0.3 versus 0.37 inch) would not be expected to produce a large effect on the observed length changes.

Rods with RXA Cladding - Measured length changes of duplex pellet LDR rods 97-54, 97-57, and 97-52, as functions of fuel depletion and fast neutron fluence, are compared in Figure 6 with those of similar long rods, which contained solid pellets and were irradiated in the LWBR test program. All these LDR and LWBR test rods had RXA cladding, an outside diameter of 0.3 inch, and an OD/thickness ratio of about 14. The values of other test parameters for these test rods are listed in Figure 6.

The data in Figure 6 indicate that the duplex pellet rods at comparable fuel depletion had about half or less the rod elongation of the solid pellet rods. At comparable fast neutron fluence, the duplex pellet rods elongated less than or about equal to the solid pellet rods but the duplex pellet rods had a 33 to 61% higher linear power output.

The LDR rods had higher power than the LWBR rods and also two of the LDR rods had about 3 mils less fuel-cladding gap. These would be expected to aggravate the duplex pellet rod elongation. However, the LDR rods experienced lower fast neutron fluence and were pre-pressurized. In addition, one of the LDR rods had a graphite coating on the inner cladding surface. Pre-pressurization and graphite coating would tend to reduce the duplex pellet rod elongation. The net result of the compensating effects on rod elongation cannot be determined from the test data.

In addition, the LWBR test assembly was accidentally dropped 7.8 inches vertically in water, after the second interim examination, and the holder nosepiece contacted the bottom of the shielded transfer cask. The effect of this drop on subsequent rod elongations is not known.

	ROD	PEAK POWER KW/FT	INITIAL PRESSURE PSIG	FUEL- CLADDING GAP, MILS	GRAPHITE COATING	PEAK FLUENCE 10^{20} n/cm ²	PEAK DEPLETION 10^{20} f/cc
LWBR TEST	△ 79-621	10.5	0	8.4	NO	101	10.0
	◇ 79-619	10.5	0	8.3	NO	100	9.8
	○ 79-630	10.5	0	8.0	NO	97	9.9
LDR TEST	● 97-54	15.0	100	5.3	YES	18	6.0
	■ 97-57	14.0	100	5.5	NO	39	11.0
	▲ 97-52	16.9	500	8.4	NO	25	7.9

Figure 6. Comparison of Overall Rod Length Changes between LDR and LWBR Test Rods: Duplex vs. Solid Pellet Rods

Although the net amount of reduction in rod elongation due to the use of duplex pellets cannot be determined quantitatively because of the above differences in test parameters, the data indicate a significant reduction in elongation in duplex pellet rods for a comparable fuel depletion, fast neutron fluence, linear power, and fuel-cladding gap. This also is in agreement with that duplex pellets operate at lower temperature than do solid pellets.

4. Reduction in Fuel Rod Bowing

Bow data from LDR duplex pellet, PBIT solid pellet, and LWBR solid pellet rod tests indicate a correlation between measured bow and measured elongation in the peak power span. This is shown in Figure 7. The ranges of peak rod powers are 14 to 16.9 kw/ft for the LDR rods, 12.7 to 14.8 kw/ft for the PBIT rods, and 10.2 to 10.5 kw/ft for the LWBR rods. The peak power span lengths were 14 inches for the LDR and LWBR rods and 18.3 inches for the PBIT rods. However, the effect of the span length difference on rod bowing would be expected to be offset by that of the difference in rod diameters (0.3 versus 0.37 inch). Although the data have a large scatter, the trend of correlation is strong. In Figure 7, the band between the two dashed lines specifies the 95% confidence interval of the mean, i.e., the best fit line. A similar correlation is also indicated in the lower power spans.

Because the data indicate that duplex pellet fuel rods elongated less than solid pellet fuel rods at comparable depletion and fluence, duplex pellet fuel rods are expected to bow less. This reduces the propensity for duplex pellet fuel rods to touch in service.

5. Reduction in Axial Gap

The axial gap which develops at the top of a fuel stack or which potentially can develop between two fuel pellets is a function of both cladding and fuel stack length changes. Data from LDR duplex pellet rods indicate no fuel stack shrinkage but rather, a growth. For instance, fuel stack length increases of 0.18% at a peak fuel depletion of 1.1×10^{20} f/cc of compartment for Rod 97-54, 0.54% at 7.9×10^{20} f/cc for Rod 97-52, and 0.46% at 11.7×10^{20} f/cc for Rod 97-53 have been indicated, Reference 3. Also, as discussed above (Table 1 and Figures 3 through 5), rods with duplex pellets had a smaller length increase. Thus, duplex pellet fuel rods are expected to have smaller axial gaps. The calculated

Figure 7. Peak Power Span Creep Bow Versus Peak Power Span Elongation

axial gap for a duplex pellet fuel rod is only about 62% of that for a solid pellet fuel rod using the CYGRO-5 computer program and the power history of a typical commercial power reactor core design.

A shorter axial gap reduces the propensity for cladding to collapse into the gap under the system coolant pressure. In addition, pre-pressurization of fuel rods is needed to prevent cladding from deforming into the axial gap. A longer axial gap requires more pre-pressurization. Thus, a reduction in axial gap implies that less pre-pressurization is needed to prevent cladding collapse into an axial gap. A lower internal pressure results in an improvement in fuel rod performance during a LOCA transient.

6. Reduction in Rod Diameter Changes

A smaller expansion of fuel rod diameter produces a lower cladding hoop tensile stress during reactor operation. This phenomenon is described in Section III.A.7. Also a smaller shrinkage of fuel rod diameter prior to complete fuel-cladding contact improves the rod-to-support grid spring contact capability and this minimizes the potential for loose rods at support grid cells which can aggravate cladding wear.

a. Short Rod Tests

Measured diameter changes of the same short test rods discussed in Section III.A.3.a are grouped in Table 1. The data indicate significant reduction in diameter expansion for duplex pellet rods compared with solid pellet rods. The diameter decreases shown in the table occurred before complete fuel-cladding contact. The ranges of measured diameter changes are -1.26 to +0.85 and -0.55 to +2.70 mils for duplex and solid pellet rods with SRA Zircaloy-4 cladding, respectively, and -0.58 to +0.28 and -0.70 to +0.98 mils with RXA Zircaloy-4 cladding. Based on the upper range values, reductions in diameter expansions for the duplex pellet rods with respect to solid pellet rods were about 67% for SRA clad rods and about 70% for RXA clad rods. These smaller diameter expansions are identified with the greater flexibility of the fuel annulus than that of a solid pellet and possibly with less fuel expansion due to reduced fuel temperature.

b. Long Rod Tests

Rod diameter changes of the long LDR duplex and long LWBR solid pellet test rods, which were discussed in Section III.A.3.b in regard to length changes, are compared in Figure 8. These rods had large fuel-cladding gaps, which required a long time of operation before complete fuel-cladding contact occurred, and thus diameter shrinkages, rather than expansions, were measured. The measured shrinkages are plotted against both peak fuel depletion and peak fast neutron fluence. As shown in Figure 8, the diameter changes of LWBR rods were measured only at the terminal fuel depletion of about 10×10^{20} f/cc. Solid curves which represent estimated diameter shrinkage histories were calculated with the CYGRO-4 computer program (Reference 8). CYGRO-4 was qualified for calculation of diameter changes of fuel rods with RXA cladding by using 11 LWBR long test rods. The two plots in Figure 8 show that duplex pellet rods have significantly less diameter shrinkage than solid pellet rods at comparable fuel depletion and fast neutron fluence (50% less diameter shrinkage at least). A mechanism for less shrinkage in duplex pellet rods is the low fuel-cladding axial interaction force which stretches the cladding less. This can be inferred from the small elongation of the duplex pellet rods as discussed in Section III.A.3.b. For example, CYGRO-4 analyses of LWBR rod 79-621 whose diameter shrinkage is shown in Figure 8 indicated that about one-third of the measured diameter shrinkage could be attributed to fuel-cladding axial interaction which was accounted for calculationaly by assuming pellet chips in the fuel-cladding gap. However, part of the reduction in diameter shrinkage was due to the effect of pre-pressurization in the duplex pellet rods (about 9% reduction for 100 psig pressure from CYGRO calculation as shown in Figure 8) and, for LDR Rods 97-54 and 97-57, to the smaller as-built fuel-cladding gaps (see Figure 6).

Although the precise amount of rod shrinkage reduction due to duplex pellets is not known, the data are consistent with a reduction which is identifiable with low fuel-cladding axial interaction force.

7. Reduction in Cladding Tensile Stresses

CYGRO-5 analyses of a duplex and a solid pellet fuel rod using the power history of a typical commercial power reactor core design indicate a substantial reduction in cladding hoop tensile stresses in the duplex pellet rod early in core life. For example, at the beginning of fuel cycle 2 (4.5×10^{20} f/cc of

Figure 8. Comparison of Peak Diameter Changes between LDR and LWBR
Test Rods: Duplex vs. Solid Pellet Rods

compartment), the calculated steady-state hoop tensile stresses are 24,000 psi for the solid pellet rod and zero for the duplex pellet rod, i.e., the calculated cladding stress of the duplex pellet rod is still compressive due to system coolant pressure. The calculated corresponding stresses at EOL (12×10^{20} f/cc of compartment) are 26,400 psi for the solid pellet rod and 25,500 psi for the duplex pellet rod. At EOL, the reduction in calculated cladding stresses is small because complete fuel-cladding and fuel annulus-center pellet contact has occurred. The lower stresses are consistent with the diameter expansion data, shown in Table 1, which indicate less diameter expansion for duplex pellet rods than for solid pellet rods.

The reduction in cladding hoop tensile stresses is due both to lower duplex fuel temperature and to the existence of a fuel annulus-center pellet gap, which allows the annular fuel to grow inward after fuel-cladding contact.

A lower cladding hoop tensile stress means a larger margin to the limit at which stress corrosion cracking occurs and thus a greater up-power transient and reactivity insertion transient (RIT) capability. This will at least exist at those times in life before both fuel-cladding and fuel annulus-center pellet gaps close.

8. Reduction in Fission Gas Release at Low Depletion

Less fission gas release implies less release of aggressive gaseous fission products, such as iodine, from fuel pellets. The reduction in fission gas release improves stress corrosion cracking performance of the cladding.

The reduction in fuel temperature in duplex pellet rods, discussed in Section III.A.1, leads to less fission gas release until a high specific depletion of the fuel annulus is reached. Although measurements of the fission gas releases from duplex and solid pellet test rods with comparable power histories, e.g., ALT-1 rods 97-69 (duplex) and 97-75 (solid), have not been made, the fission gas releases calculated with the qualified CYGRO-5 computer program (Reference 6) are 9.2% and 53.6% of the total fission gas generated for rods 97-69 (duplex) and 97-75 (solid), respectively, for a compartment depletion of about 14×10^{20} f/cc. Both rods used UO_2 fuel and operated with similar power history of about 17 kw/ft. It should be noted that fission gas release is

extremely sensitive to power history and therefore one cannot compare two rods with different power histories purely on the basis of depletion.

Fission gas release is related to two prime attributes: fuel temperature and fuel depletion. At low fuel depletion, fuel temperature has dominance in fission gas release, and at high fuel depletion, fuel depletion has dominance. For the same power and the same compartment fuel depletion, a duplex pellet operates at a lower temperature than does a solid pellet but the local depletion in the annulus of the duplex pellet is about twice that in the solid pellet because of the high concentration of fissile fuel in the annulus of the duplex pellet. For this reason, a duplex pellet reaches the depletion-dominance region for fission gas release in about half the time, and therefore the total fission gas release from a duplex pellet may overtake that from a solid pellet at high depletion. For example, at about 11 to 12×10^{20} f/cc (~ 22 to 24×10^{20} f/cc of annulus), data from the D-1 test indicate a sharp increase in the rate of fission gas release. This upswing behavior is reported in Reference 11. Concerns regarding high fission gas release rate at high depletion are addressed in Section IV.A.5. However, the high level of depletion at which the total fission gas release from a duplex pellet exceeds that from a solid pellet is expected to be beyond the current extended burnup fuel design of 12 - 14.5×10^{20} f/cc.

B. Extraction of U-233 from Spent Prebreeder Fuel

In the development of prebreeder fuel, one objective is to prevent the U-233 produced from contamination by U-235 and higher mass uranium isotopes. Use of separate fissile (U-235) and fertile (thorium) fuel rods could achieve this objective. However, the drawback is that to provide the desired core power capability, the thorium fraction in the core must be low. This fuel system does not provide the core power densities comparable with that of commercial cores when the thorium fraction in the core is high enough for efficient breeding. Three core concepts for producing U-233 are presented in Reference 9. Use of solid binary (ThO_2 - UO_2) fuel rods could achieve the desired power densities, but the drawback is that the U-233 produced cannot be extracted without contamination by U-235 and higher isotopes.

Duplex fuel, which consists of an outer annulus of fissile fuel and an inner center portion of thorium, provides both the desired core power densities and the needed U-233 source with little or no contamination by U-235 and higher isotopes. It has been demonstrated that the UO_2 annulus of the spent fuel can be chemically dissolved using nitric acid, which does not attack ThO_2 containing a small quantity of U^{233}O_2 . The ThO_2 - U^{233}O_2 center portion thus remains as a comparatively pure source of U-233.

A bench mark test was conducted at Bettis using unirradiated fuel to demonstrate the ability to dissolve the UO_2 annulus leaving the center ThO_2 - UO_2 pellet. For this test the spent annular fuel was represented by sintered pellets of pure UO_2 , while the end-of-life fertile fuel was represented by sintered binary pellets of ThO_2 -1.14 w/o UO_2 . Since the rate of solution of sintered ThO_2 in HNO_3 (nitric acid) is not changed appreciably by the addition of UO_2 up to 10%, the results obtained with the binary pellets of this test are representative of ThO_2 - UO_2 fuel compositions of up to several percent UO_2 . Pellets of UO_2 fuel and of ThO_2 - UO_2 fuel were comminuted to 8-10 mesh, weighed separately, and then combined to yield a mixture simulating the spent composite fuel. Results indicate that the HNO_3 dissolved 100% of the UO_2 fuel but less than 0.2% of the ThO_2 - UO_2 fuel. Tests also demonstrated that the residual ThO_2 - UO_2 dissolves slowly in a conventional thorex solution (13M HNO_3 + 0.04-0.06M HF) boiling at atmospheric pressure with a residue or dissolves rapidly in the same solution at 392°F (under 125 psig pressure) even in the presence of Zircaloy cladding, which inhibits the reaction at lower temperature.

Similar possibilities exist for a ZrO_2 - UO_2 or ZrO_2 - UO_2 -CaO annulus due to its different chemical constitution although specific techniques have not been developed.

IV. CONCERNS AND DISADVANTAGES WITH DUPLEX FUEL RODS

The duplex pellet configuration, a fuel annulus and a center pellet, caused concerns because the fuel annulus appears to be more fragile than a solid fuel pellet and because the fissioning rate in the fuel annulus is about twice that in a solid fuel pellet. One of the objectives of the AWBA irradiation tests is to assess the performance characteristics with respect to the concerns. This

section presents an assessment of the concerns using the results of irradiation testing coupled with analytical calculations.

Concerns and potential disadvantages with duplex pellet fuel rods lie in two general areas: performance and manufacturing. In performance, several concerns have been postulated with duplex pellet fuel rods. In addition, irradiation testing of defected fuel rods did identify that ridging can be produced by hydrogen migration to the Zircaloy cladding over the cold thoria spacer pellets. In manufacturing, the disadvantage is the complexity in fabrication and assembly of the composite (duplex) fuel pellets.

A. Performance

1. Cladding Ridging and Hydriding in Defected Rods

Cladding ridging was observed in intentionally defected duplex pellet test rods. Figure 9 shows the external appearance of an irradiated intentionally defected test rod. This ridging occurred at thoria spacer pellets which were used to ensure that the annulus and the center pellet of each duplex pellet remained in axial registry over the rod length. The cause of ridging has been identified as a high concentration of hydrogen in cladding adjacent to the cooler thoria spacer pellets. This is based on metallographic examination results reported in Reference 2. This condition may not rule out the use of thoria spacer pellets since the hydriding is not different in substance from hydriding observed in other defected rods with solid pellets, although it occurred at different locations. For example, an AWBA test rod using solid pellets (ALT-2 rod 97-121) broke into two pieces due to massive hydriding after defecting. This rod is suspected of having had a cladding penetration near the top end as a result of a transient test, and subsequently the rod broke over the bottom active pellet. Similar phenomena have occurred during the course of the LWBR test program where solid pellets were used.

Use of alternative fuel pellet stacking arrangements to eliminate the need for thoria spacer pellets to maintain fuel axial registry is possible. For example, axial registry of center pellets can be maintained by varying the annulus thickness of adjacent duplex pellets in the stack. A schematic diagram of this stacking arrangement is shown in Figure 10.

Figure 9. Intentionally Defected D-1 Rod 97-48D
External Appearance

Figure 10. Schematic Diagram of Variable Diameter Duplex Pellets

No adverse performance aspects have been observed in duplex pellet test rods without thoria spacer pellets, References 2 and 3. For example, LDR rod 97-52 which used no spacer pellets was irradiated to a depletion of 8.2×10^{20} f/cc of compartment without incident, although an axial gap of 0.14 inch developed in service between two center thoria pellets (Reference 3).

2. Fragility of Annulus under Axial Load

Under a uniform axial compressive load (applied to the end of an annulus), the crushing strength of the annulus of a duplex pellet is not significantly different from that of a solid pellet. However, the annulus cracks are more extensive because of the presence of a gap between the fuel annulus and center pellet.

Irradiation data indicated a smaller rod elongation in duplex pellet rods than in solid pellet rods, as discussed in Section III.A.3. This leads to the conclusion that duplex pellet rods experienced a lower fuel-cladding axial interaction force which stretches the cladding. Consequently, the compressive load on the duplex fuel annuli also would be lower. This is an apparent built-in compensating feature of duplex pellets. In addition, post-irradiation neutron radiographs of LDR rods 97-52, 97-53, 97-54, and 97-55D showed that annuli do stay intact, i.e., they do not crumble from axial compressive load during irradiation testing, Reference 3. A typical neutron radiograph for a section of LDR rod 97-53 is shown in Figure 11, after a depletion of 11.7×10^{20} f/cc of compartment. No crumbling of the annuli is indicated. Cracks are similar to those of solid pellets. Thus, the concern about fragility of the annulus is not substantiated by the observed behavior of duplex pellet fuel rods.

3. Aggravation of Waterlogging in Defected Rods

The presence of water/steam (fluid) in the fuel annulus-center pellet gap of a duplex pellet is similar to that of water/steam in the end dish region between two solid pellets with respect to high fluid temperature. Fluid in the fuel annulus-center pellet gap region is connected to the fuel-cladding gap through the pellet-pellet interfaces. Thus, rod internal pressure buildup

Figure 11. Typical Neutron Radiograph of A Section of UO₂ Duplex Fuel Pellet Stack of LDR²Rod 97-53

during a power increase is expected to be the same in fuel annulus-center pellet and fuel-cladding gaps. Bursting of the annuli due to a high pressure differential across the annulus wall is therefore not expected to occur. However, because of the existence of more water at high temperature in duplex pellet rods, a reactor startup rate equal to about 83% of the LWBR startup rate will produce the same internal rod pressure for power below 63% of rated power. Above 63% of rated power, the allowable startup rates are about equal, since steam, which is much more compressible than water, covers nearly the entire fuel stack length above this power level. Thus, no severe restriction on the startup rate is required if duplex pellets are used instead of solid pellets. The LWBR startup rate was set at 2.1% of rated power per minute from 0 to 60% power and 4.2% per minute from 60% to full power. This rate is significantly faster than the rates used for commercial reactors.

Six intentionally defected duplex pellet rods in the AWBA irradiation test program have shown no unexpected results with respect to waterlogging performance through a multitude of startups and transients. Each of these intentionally defected rods had a drilled hole of 5 mils in diameter to simulate a rod defected in service. One of them, LDR defected rod 97-55D, showed a 2.0% diameter swelling in the plenum region, Reference 3, after a depletion of 11.2×10^{20} f/cc of compartment. This is less than the 3.8% diameter swelling of LWBR solid pellet defected rod 79-433D, reported in Reference 10, after a depletion of 7.93×10^{20} f/cc.

4. More Fuel Oxidation in Defected Rods

A duplex pellet has more surfaces at higher temperatures than does a solid pellet, which has only an outer surface and two end surfaces. Through the surface areas, oxygen can be absorbed and diffused. The degradation in fuel thermal conductivity and melting point due to fuel oxidation would reduce the power margin to melting of the duplex fuel to no less than 80% of that which would occur in pure UO_2 in undefected rods. This conclusion is based on kinetic calculation of oxygen migration. This level of degradation can be tolerated since additional thermal margin exists in duplex fuel by virtue of its lower peak fuel temperature, as discussed in Section III.A.1. This calculation is

supported by the lack of evidence of columnar grain growth or melting of duplex pellets from destructive examination of D-1 defected rod 97-48D. Columnar grain growth or melting would indicate high fuel temperature caused by excessive oxidation. The calculated maximum fuel temperature for this rod is about 3000°F at 15.2 kw/ft while the melting temperature is about 5700°F for the fuel annulus of $\text{ThO}_2\text{-UO}_2$ used in this rod.

Another concern is that depletion of oxygen in coolant inside the rod (due to reaction with UO_2) would lead to a higher hydrogen concentration and could reduce the time to a massive hydriding of cladding. However, massive hydriding has been observed in LWBR defected test fuel rods containing comparatively stable $\text{ThO}_2\text{-UO}_2$ binary fuel. The conclusion is that hydrogen generation associated with oxidation of the Zircaloy cladding is alone sufficient to cause massive hydriding, while the fuel system plays a secondary role.

5. High Fission Gas Release Rate at High Depletion

A possible concern of duplex pellet rod designs arises from the high depletion attained in the fuel annulus compared with a solid pellet operating at the same power. Data from the D-1 irradiation test show that an increase in fission gas release rate occurred after high depletions are reached ($\sim 11 \times 10^{20}$ f/cc of compartment; 22×10^{20} f/cc of annulus), Reference 11. Prior to this depletion, measured fission gas release from the D-1 duplex pellet rods was low ($\sim 2.2\%$ of total fission gas generated), due to the low duplex fuel operating temperature. At a depletion of about 12×10^{20} f/cc of compartment, the measured maximum gas release was about 11.8% of the total fission gas generated for UO_2 annulus fuel. However, this value is still significantly lower than the calculated maximum gas release ($\sim 26\%$) at a depletion of 11×10^{20} f/cc for solid UO_2 pellets at a comparable power. As discussed in Section III.A.8, the primary mechanisms for high gas release rate are high depletion in duplex fuel annuli and high temperature in solid pellets, and one effect must be traded off for the other. A duplex pellet has no disadvantage compared with a solid pellet with respect to total gas release up to a depletion of at least 14×10^{20} f/cc of compartment based on CYGRO-5 calculations, as discussed in Section III.A.8. This compartment depletion of 14×10^{20} f/cc is about equal to the current extended burnup fuel design of $12\text{-}14.5 \times 10^{20}$ f/cc.

6. Effect of High Depletion on Fuel Structure

Data from PWR UO_2 fuel wafers show a change in fuel grain structure at a depletion of about 25×10^{20} f/cc, where the initial grain boundaries became indistinguishable and fine sub-structure was observed, Reference 12. This behavior was also observed in D-1 rod 97-24 which had a depletion of about 24×10^{20} f/cc of annulus and a fission gas release of 11.8%. The fracture strength of unirradiated fuel is known to improve as grain size decreases, Reference 13. Although the corresponding behavior of irradiated fuel is not known, the appearance of a sub-structure within the grains in irradiated fuel at high depletion does not necessarily imply a loss of fuel fracture strength.

Another concern is that an accumulation of large fission gas bubbles which would be expected at intermediate temperatures ($\sim 2500^\circ\text{F}$) might have a harmful effect. However, irradiation data for a depletion of up to 16.7×10^{20} f/cc of compartment ($\sim 33.4 \times 10^{20}$ f/cc of annulus) from LDR and NLDR duplex pellet rods gave no indication of abnormal fuel expansion or shrinkage, References 3 through 5. This implies that fuel annuli were apparently performing satisfactorily.

7. An Annulus Chip Lodging between Center Pellets

The temperature of a small UO_2 annulus chip (50-mil diameter by 50-mil height) lodged in an axial gap larger than 50 mils between two center pellets is predicted to be well below melting (3283°F at 17.9 kw/ft compared with a 5064°F melting temperature). However, if a 50-mil space above the lower center pellet in the axial gap is completely filled with UO_2 material, melting of UO_2 is predicted to occur. In reality, this will not occur unless an annulus is completely crushed and also there exists an axial gap adjacent to the crushed annulus. Post-irradiation radiographs have not indicated any such crushing of the annuli, although a 0.14-inch axial gap was observed between two center thoria pellets in one rod (LDR Rod 97-52), which contained no thoria spacers, Reference 3.

8. Fuel Redistribution by Evaporation and Condensation

Mass transport by evaporation and condensation at elevated temperature in oxide fuel rods is a potential concern because of the possibility of building up

at some location a thick layer of fuel that could melt at high power. Metallographic examinations of D-1 rods 97-17 ($\text{UO}_2\text{-ThO}_2$) and 97-24 (UO_2) showed no such evidence of fuel redistribution. Thus, fuel redistribution at normal rod power is not a primary concern for fuel rods with duplex pellets, since fuel operating temperatures are even lower than those existing when solid pellets are used.

B. Manufacturing

Using standard fuel pellet manufacturing procedures, duplex pellet fabrication requires at least two separate manufacturing lines: one to make the fuel annulus and the other to make the center thorium pellet. If spacer pellets are employed for axial registry of center pellets, a third line is required. In the AWBA test program, fuel annulus and center pellet were cold pressed, sintered, and ground to specific diameters, Reference 14. The annulus and center pellet were separately compacted using double acting pressing, that is, the punch travel was adjusted so that an equal load is applied to each end of each component. The midpoint of the pressed annulus or center pellet is the region of minimum density and shrinks in diameter the most during sintering. This results in an hourglass distortion in sintered components. After sintering, the outside diameters of both annulus and center pellet are ground to remove sintering induced distortions and to provide specific gap sizes between the annulus and the cladding and between the annulus and the center pellet.

The separately made components must be individually mated during rod loading to form the composite (duplex pellet) with a specific radial gap between them. Production of duplex pellets is therefore less economic than production of solid pellets. During rod loading, special provisions must be used to prevent mismatches in overall lengths of annulus and center pellet stacks due to stack-up of tolerances in individual component lengths. Provisions used in the AWBA irradiation test program to control overall stack length included insertion of disc-like spacer pellets between alternate duplex pellets or between short stacks of duplex pellets. However, an alternative design, not tested, would use annuli and center pellets with two different annulus I.D.'s and two different

center pellet O.D.'s. The large and small I.D. annuli (mated with center pellets with corresponding diameters) alternate in the fuel rod to prevent slippage of the center pellet stack relative to the fuel annulus stack. A schematic diagram of this design is shown in Figure 10.

A simplified manufacturing process was sought wherein both annulus and center pellet could be compacted together and would remain together through all subsequent manufacturing operations. This process would eliminate the need for a double manufacturing line. A technique, described in Reference 15, has been developed which permits co-compaction of two coaxial cylinders. The outer annular cylinder is compacted first, followed by a separate powder fill and independent compaction of the center pellet. The as-pressed (green) density of the center pellet can be made less than that of the annulus either by reducing the powder fill or by reducing the recompaction pressure. Reducing density of the center pellet increases the shrinkage during sintering. This causes the center pellet to draw away from the annulus. Thus, a controlled gap size is produced between the two. The two components stay together either by forming a solid state diffusion bond along a line interface or by mechanical locking due to slight component distortion during sintering.

As reported in Reference 15, the feasibility of the above co-compaction technique for both single oxide system, i.e., annulus and center pellet regions of the same oxide, and two oxide systems, e.g., a fuel annulus of $\text{ThO}_2\text{-UO}_2$ and a central pellet of ThO_2 , has been demonstrated.

Fabrication of duplex fuel and fuel rods for AWBA irradiation tests has indicated no technical process problem which would inhibit commercial application of this concept. However, higher fuel fabrication costs are expected. A co-compaction technology would reduce fabrication costs significantly.

V. SUMMARY AND CONCLUSIONS

A. Advantages

Significant performance advantages of duplex pellet fuel rods are indicated compared with solid pellet fuel rods though rigorous comparisons cannot always be made. Irradiation data and analyses show that duplex fuel produced less cladding deformation in both circumferential and axial directions than did solid fuel for a comparable operating history. This result is attributable to the

significantly lower temperature and greater flexibility of the fuel annulus. When the fuel annulus interacts with the cladding, the gap space between the fuel annulus and the center pellet provides a region into which the annular fuel may swell, rather than increasing the cladding diameter.

The benefit of reduced cladding burden is to (1) increase margin for up-power maneuvers and transients, including consideration of stress corrosion cracking, (2) reduce cladding maximum temperature during a LOCA transient, (3) reduce fuel rod bow, and (4) reduce axial gaps in the fuel stack, therefore providing improved margin against cladding collapse or enabling thinner cladding to be used.

In prebreeder design, duplex fuel offers the potential for both the desired core power densities and a comparatively pure source of U-233 by sequential chemical dissolution of the annulus and the center pellet.

B. Concerns and Disadvantages

Although several performance concerns have been postulated, the only potential problem identified by irradiation testing was the presence of cladding ridging and hydriding adjacent to thorium spacer pellets in intentionally defected rods. The spacer regions acted as cold traps for hydrogen. To correct this disadvantage, thorium spacer pellets can be eliminated by using duplex pellets with different annulus wall thickness between adjacent duplex pellets in the fuel stack to maintain axial registry of center pellets.

Though no technical fuel fabrication problem was indicated in fabricating test rods, for commercial application of the duplex fuel concept, the more complex fabrication process used will be less economic for producing a large quantity of duplex pellets. A co-compaction technique, which simplifies the fabrication process by making fuel annulus and center pellet together as a unit, reduces the complexity in duplex pellet fuel rod fabrication.

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