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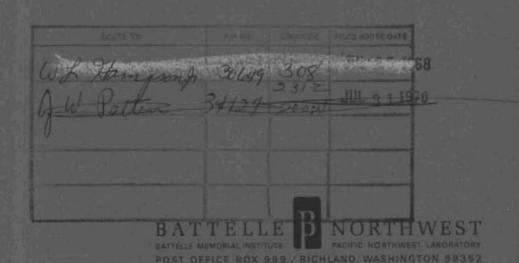
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NITRIDE FUELS FOR FAST BREEDER REACTORS: FUEL CYCLE CONSIDERATIONS

J. F. FLETCHER
J. GREENBORG

FEBRUARY 1968

AEC RESEARCH & DEVELOPMENT REPORT



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NITRIDE FUELS FOR FAST BREEDER REACTORS: FUEL CYCLE CONSIDERATIONS

By

J. F. Fletcher and J. Greenborg

Engineering Analysis Section
Engineering Development Department

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NITRIDE FUELS FOR FAST BREEDER REACTORS: FUEL CYCLE CONSIDERATIONS

ABSTRACT

The nitrides of uranium and plutonium appear technically attractive as potential fuels for fast breeder reactors. To examine the potential economic status of the nitrides in commercial power production, a study was made to develop comparative fuel cycle costs for nitride and carbide fuels in a 1000 MW_e sodium-cooled fast breeder reactor concept previously developed by Combustion Engineering, Incorporated, in an AEC-sponsored study. Results of the study indicate that nitrides would be competitive with carbides if low-cost processes for conversion of fuel materials to the nitrides were developed. The nitride fuels appear to contribute significantly toward a negative sodium-voiding coefficient.

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NITRIDE FUELS FOR FAST BREEDER REACTORS: FUEL CYCLE CONSIDERATIONS

J. F. Fletcher and J. Greenborg

INTRODUCTION

The nitrides of uranium and plutonium are dense refractory compounds of high thermal conductivity, with favorable combinations of physical and neutronic properties which make them potentially of great interest as reactor fuels. Previous studies of the use of plutonium as a fuel for fast compact reactors indicated that the nitrides had potential technical advantages as fuel materials for reactors of this type. (1) Much development would be needed on the technology of the nitrides, however, to realize these advantages fully.

The nitride fuels should also be technically attractive in commercial power reactors. For such use however, they must be able to compete not only technically but economically with the more conventionally considered oxide and carbide fuels. The nitrides are materials whose technology remains for the most part to be developed, whose properties are only imperfectly known, and which have been prepared only in small laboratory quantities. Therefore, any investigation of the technical and economic standing of the nitrides in a given reactor system must depend largely on projected development of nitride technology and on comparisons of that technology with that of the more highly developed oxide and carbide fuels.

To make a preliminary assessment of the potential of nitride fuels in commercial power reactors, during 1965-66 a study was performed on the

behavior and the fuel cycle costs of mixed plutonium-uranium nitride fuels, using a 1000 MW, fast breeder reactor concept as a model for the study.

This document reports the results of that study.

SUMMARY

The goal of the study of the fuel cycle aspects of a nitride-fueled fast breeder reactor was the determination of the characteristics of such a reactor and the estimation of the economic potential of uranium and uranium-plutonium mononitrides as fuel for a typical fast breeder power reactor concept. Because of the expected similarity in behavior of nitride and carbide fuels, the study was based on a reactor model conceived by Combustion Engineering, Incorporated in an AEC-sponsored study of a 1000 MW fast breeder reactor.

(2) An evaluation of this and similar fast breeder reactor studies has been published by the Atomic Energy Commission.

(3) Use of this reactor model permits a ready comparison of the nitride and carbide fuels; however, since this reactor was developed for carbide-fuel evaluation, the comparison might be expected to be biased slightly in favor of the carbides.

To provide a valid comparison of the behavior of nitride and carbide fuel, nuclear data for the carbide reactor model were recalculated using techniques identical to those used for the nitride-fueled reactor.

A summary of the calculated nuclear characteristics of the nitrideand carbide-fueled reactors is given in Table I, together with comparison data for carbide reactors from the Combustion Engineering study and the AEC evaluation.

Fuel-cycle costs were calculated for both the nitride- and carbidefueled reactor models, assuming private ownership of the fuel. For these

TABLE I

NUCLEAR CHARACTERISTICS SUMMARY:

CARBIDE AND NITRIDE FUELED REACTORS

| | Ca | arbide Fue | 1 | Nitride Fuel | | |
|---|----------------------|----------------------|----------------------|----------------------|--|--|
| | $_{C.E.}$ (a) | AEC(b) | BNW(c) | BNW(c) | | |
| Enrichment Required (%Pu, equilibrium core) | | | | | | |
| Inner Core Zone Outer Core Zone Core Average | 14.1 18.3 16.2 | 15.3 19.9 17.6 | 15.3 19.8 17.6 | 16.0 20.6 18.4 | | |
| Sodium Void Coefficient, %Ak, for Sodium Loss from: | | | | | | |
| Total Reactor Core Only | 0.0 +2.4 | +1.0 | -0.8 +1.5 | -1.4 +1.0 | | |
| Doppler Coefficient | | | | | | |
| $T \frac{dk}{dt} \times 10^3$ | 5 | 5.4 | 3.7 | 3.5 | | |
| Peak-to-Average Power Ratio | | 1.55 | 1.60 | 1.66 | | |
| Breeding Characteristics | | | | | | |
| Core Conversion Ratio Total Breeding Ratio Doubling Time, Years | 0.83 1.42 6.9 | 0.76 1.41 7.2 | 0.81 1.39 8.4 | 0.75 1.33 11.7 | | |

- (a) Per Combustion Engineering calculations, Reference 2, based on initially loaded fuel metal.
- (b) Per AEC evaluation, Reference 3, based on equilibrium (U + Pu) content.
- (c) Per Battelle-Northwest calculations, this study, based on equilibrium (U + Pu) content.

calculations, an interest rate of 10% per year (non-depreciating) was assumed on the working capital invested in the fuel. Charges on depreciable capital investment in the plant were based on a 30-year plant life at a 6% interest rate, with an added 6% allowance for insurance and taxes. Plant efficiency was assumed at 40%, a plant utilization factor of 80% was used, and the value of plutonium was assumed to be \$10.00/g fissile isotopes as nitrate.

Since the original AEC-sponsored study by Combustion Engineering had been based on leasing of fuel from the AEC, fuel-cycle cost calculations were repeated on that basis to provide a comparison between this study and that performed by Combustion Engineering. The leased-fuel costs were performed using the same ground rules as set by the AEC for the series of 1000 MW fast breeder studies, which included the Combustion Engineering study. These ground rules assumed the same operating factors and plutonium values as for the privately-owned fuel case, but assumed a 4.75% use charge on the leased fuel. In the leased-fuel calculations, a capitalization option used by Combustion Engineering (capitalizing the entire first core over the 30-year reactor life) was adopted for consistency in cost comparisons. For both privately-owned and leased-fuel calculations, fuel reprocessing costs were based on published reprocessing charges for the Nuclear Fuel Services, Inc. (NFS) reprocessing plant in West Valley, New York, rather than on those for the standard AEC conceptual plant as used in the earlier studies. To provide a valid comparison of fuel-cycle costs, the reprocessing costs reported in the Combustion Engineering study were recalculated to reflect NFS reprocessing charges.

Fuel-cycle costs for the nitride-fueled reactor were calculated for two cases: the first assumed conversion of the fuel materials to nitrides through carbon reduction of the oxides in a nitrogen atmosphere, and the second case assumed reduction of the fuels to metallic form followed by nitridation. Although neither conversion has been developed beyond laboratory scale, approximate estimates of fabrication and conversion costs have been made assuming the processes to be developed to a full-plant usage. (4) These cost estimates were used in calculating the fuel-cycle costs.

A summary of the fuel-cycle costs for the two assumed nitride cases is given in Table II. Costs are also given for the carbide-fuel cycle, based on Battelle-Northwest and Combustion Engineering reactor nuclear analyses.

TABLE II

FUEL CYCLE COST SUMMARY

| | "Non-Metal" NITRIDE(a) (BNW) mills/kW-hr | "Metal" NITRIDE(b) (BNW) mills/kW-hr | CARBIDE(c) (BNW) mills/kW-hr | CARBIDE(d) (CE) mills/kW-hr |
|-------------------------|--|---|------------------------------------|-----------------------------------|
| Privately Owned Fuel | 0.818 | 1.006 | 0.750 | |
| AEC Leased Fuel | 0.549 | 0.730 | 0.480 | 0.411 |

- (a) Per Battelle-Northwest calculations, nitride conversion through non-metallic intermediates.
- (b) Per Battelle-Northwest calculations, nitride conversion through metallic intermediates.
- (c) Per Battelle-Northwest calculations.
- (d) Per Reference 2, recalculated to reflect NFS processing charges.

Fuel-cycle costs calculated for nitride-fuel cores prepared by use of metallic intermediates are 0.18 to 0.19 mill/kW-hr higher than for those prepared via carbothermic reduction of the oxides. This cost difference is due almost entirely to the high cost of reduction of oxide to metal, principally to the approximately \$1.50 per gram required for plutonium reduction. On the other hand, differences in fuel-cycle cost between the "non-metallic" (carbothermic-reduction) nitride case and the BNW-calculated carbide case are due primarily to the greater plutonium credit in the carbide cycle, resulting from the higher breeding ratio calculated for the carbide-fueled reactor. The small difference in the fuel-cycle costs for these cases (about 0.07 mills/kW-hr difference was calculated) indicates that with suitable development of an economic nitride conversion process, nitrides should be competitive with carbides as fuel for fast breeder reactors.

Comparison of the calculated nuclear characteristics of nitride and carbide fuels as listed in Table I shows the two fuels to be quite similar in their behavior. The nitride-fueled reactor requires slightly more enrichment and has a lower breeding ratio, slightly lower Doppler coefficient, and slightly higher median fission energy than its carbide-fueled counterpart. Gas generation in the fuel is approximately 15% higher due to (n,α) and (n,p) reactions in nitrogen. However, the calculated contribution to the reactor sodium void coefficient is significantly more negative for the nitride fuel than for the carbide, as a result of the increasing capture cross section of nitrogen as the neutron spectrum is hardened by loss of the sodium. The apparent ability of the nitride fuel to provide an inherent negative contribution to the sodium void coefficient of a reactor could be of importance to

safeguards considerations in fast breeder reactor design. If core designs are constrained by sodium void characteristics, the optimum nitride design could well be superior in fuel cycle cost to a carbide design. No attempt was made in this study to compare the fuels on the basis of such a constraint. DISCUSSION

Power reactor fuel development work has historically been concentrated almost exclusively on the oxides and carbides of uranium and plutonium, by far the best known and most developed refractory compounds of those elements. More recently, attention has been given to uranium and plutonium mononitrides as potential reactor fuels. The nitrides are highly refractory compounds with melting points approaching or exceeding that of UO2 (with suitable nitrogen atmosphere for the nitrides) and exceeding those of the plutonium oxides or of uranium and plutonium carbides. These compounds present metal densities greater than those of the carbides or oxides, and have thermal conductivities somewhat above those of the carbides and far exceeding those of the oxides. In addition, they appear not to be highly susceptible to radiation damage, and tests have indicated their compatibility with a variety of cladding materials and with liquid metal coolants at temperatures of interest in power reactor technology. On the other hand, they exhibit a tendency toward volatilization and/or decomposition at temperatures near their melting points, and because of the (n, a) and (n, p) reactions of nitrogen they generate somewhat more gas under irradiation than do the carbides and oxides.

Development work on uranium and plutonium nitrides has, for the most part, been performed only recently, and many uncertainties remain about the

properties and behavior of these materials. However, the work performed on these compounds to date indicates considerable potential for the nitrides as reactor fuel. Uranium nitride was a candidate fuel material for the SNAP-50 (PWAR-20) reactor reference design. The potential of uranium and plutonium nitrides as fuels has been considered in several studies of fast compact reactor concepts, (1,5) and their possible future application as fast breeder power reactor fuels has attracted interest. Programs at several sites in the United States and Europe have been directed toward development of the nitrides and determination of their properties as fuel materials.

Scope of Study

The primary goals of this study were the prediction of the behavior of the mononitrides of uranium and plutonium as fuels in a fast breeder power reactor; calculation of the costs associated with a nitride fuel cycle; and identification of those portions of a nitride fuel cycle wherein costs differ significantly from the corresponding costs for oxide or carbide fuels, and of areas where development would be needed to provide a nitride fuel and fuel cycle technically and economically competitive with other fast breeder power reactor fuels.

The nitrides have physical and neutronic properties similar to those of the carbides and might be expected to behave similarly as reactor fuels.

Therefore, a forthright evaluation of the nitride fuels should be obtainable by direct comparison of nitride and carbide fuel cycles using reactor models of essentially identical design. With proper selection of a model compatible with both fuels, this procedure should provide a valid comparison of carbide and nitride fuels without unduly penalizing either fuel by failing to optimize

reactor design or fuel management based on the behavior of the fuels. Use of an existing carbide-fueled reactor concept model, if suitable for both fuels, would permit a simplification of the scope of the study and would permit concentration of efforts on the nitride fuel cycle study proper by eliminating the need to develop a separate reactor concept model.

Projected fuel cycle costs for oxide and carbide fuels in sodium cooled, 1000 MW_e fast breeder power reactors were investigated in a series of four studies undertaken by different contractors at the request of the Atomic Energy Commission. The results of these studies⁽³⁾ provide a convenient reference for comparison of nitride fuel performance. One of these studies, performed by Combustion Engineering, Inc., ⁽²⁾ utilizing a carbide fuel cycle in a fast breeder reactor of high specific power, appeared particularly suitable as a reference model for nitride-carbide comparisons. Therefore, the nitride fuel cycle study was performed based on the Combustion Engineering reactor model and on the fuel management scheme used in the Combustion Engineering study, with essentially no changes in either the reactor design or the fuel handling scheme. Subsequent nuclear and thermal hydraulic analyses confirmed the suitability of this model as a basis for the nitridefuel cycle studies.

Nitride-fuel cycle studies were based on a modification of the ground rules set by the AEC⁽³⁾ for the series of four 1000 MW_e reactor design studies; whereas these ground rules assumed the use of AEC-leased fuel, the primary calculations in this study were based on privately-owned fuel. To provide a valid comparison between the present study and that performed by Combustion Engineering, the fuel-cycle cost calculations were repeated

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assuming the use of AEC-leased fuel. The ground rules as adapted for this study included the following assumptions:

- (1) The fuel cycle is at equilibrium with respect to isotopic compositions of the plutonium and uranium.
- (2) For privately-owned fuels, 10% interest is charged on the non-depreciating "working capital" invested in fuel; for AEC-leased fuel, a 4.75% use charge is applied.
- (3) Plutonium value is 10.00/g (Pu²³⁹ + Pu²⁴¹) as nitrate.
- (4) Thermal efficiency of the plant is 40%, and the plant utilization factor is 80%.

Within the framework of these ground rules, the nitride-fuel cycle study paralleled the Combustion Engineering study. Where differences in calculation techniques or source data appeared to affect the nitride-carbide comparison-principally in the reactor nuclear analysis--calculations for the carbide system were repeated using the same techniques as for the nitride calculation to provide a normalizing factor between the present study and that by Combustion Engineering.

The fuel cycle used in the Combustion Engineering study, and followed in this study, assumed that all required plutonium enrichment of the fuel was supplied by recycling of produced plutonium; excess plutonium was sold. Each fuel batch was assumed to utilize all-new, depleted (0.3%) uranium; recovered uranium from the reprocessed fuel was assumed to be transferred, at no cost, to the supplier furnishing new uranium. The cycle was based upon reprocessing in a remote, central facility and upon conversion/fabrication facilities supplying three similar reactors and combining facilities for both uranium

and mixed uranium-plutonium fuels operations. Variations in the fuel cycle and the potential effect of alternate schemes on cycle costs were included. However, the only alternate schemes included in the study were those whose effects were significantly dependent on fuel type. For example, a close-coupled, integrated "nuclear plant" concept which combines reactors, reprocessing and fabrication facilities at a single site was not included as a variable in the study. While such a scheme may show potential cost benefits in large installations, these benefits would apply equally to a nitride or carbide (or oxide) cycle. On the other hand, different conversion processes could significantly affect nitride fuel cycle costs, or certain fabrication techniques may be uniquely applicable to carbide or to nitride fuels and thus may introduce cost differentials between fuel types. Such variables were included as process alternates.

Properties of Nitride Fuels

Although the preparation of uranium nitride was first reported in 1842, (6) and plutonium nitride was prepared in 1944, (7) the properties of these compounds remained largely unknown until recently. Interest in the last few years in their potential use as fuel materials has engendered considerable effort on development of the nitrides and has resulted in increased understanding of their properties and behavior. Still, the properties of the nitrides are considerably less well defined than are those of the more familiar oxides and carbides of uranium and plutonium.

Uranium and plutonium nitrides may be prepared by several processes.

The methods used almost exclusively for preparation of laboratory quantities of pure nitrides consists of hydriding or superficially hydriding finely-

divided uranium or plutonium metal, followed by heating to about 600 to 800 °C in a nitrogen atmosphere. The resulting compound has an approximate formula $UN_{1.6}$ to $UN_{1.75}$, lying in the single-phase solid solution region which exists in the uranium-nitrogen system between the sesquinitride U_2N_3 , and the dinitride UN_2 .

This compound is subsequently reduced to the mononitride by heating to about 1300 °C in a vacuum or an inert atmosphere. With plutonium, on the other hand, the nitriding reaction produces the mononitride PuN directly.

Other processes for the preparation of the nitrides of uranium and plutonium include reduction of the oxides with carbon in a nitrogen atmosphere ("carbothermic reduction"); reaction of the halides with nitrogen or ammonia; or by the arc melting of uranium or plutonium metal in a high-pressure nitrogen atmosphere. Those processes which do not require prior reduction to the metal appear economically attractive, primarily because of the high costs currently associated with plutonium metal-reduction processes. In particular, investigation of the carbon-nitrogen reduction of the oxides has been conducted at several sites.

(8,9,10)

Both this process and the nitridation of uranium and plutonium halides have been investigated at Battelle-Northwest.

(10,11)

The preparation of uranium nitride by conversion from the carbide was also investigated by Atomics International

as one step of a proposed fuel reprocessing scheme involving alternate conversions to nitride and to carbide.

Plutonium nitride and uranium-plutonium nitride mixtures have been investigated at the Columbus Laboratory (13) and the Pacific Northwest

Laboratory $^{(10,14)}$ of the Battelle Memorial Institute, at Los Alamos Scientific Laboratory, (15) and at Fontenay, France (16) The mononitrides of uranium and plutonium both have face-centered cubic (NaCl-type) crystal structure with quite similar lattice spacings. These compounds are freely miscible, forming a complete range of solid solutions. The higher nitrides of uranium exhibit either a body-centered cubic $(\mathrm{U_2N_3})$ or a fluorite-type face-centered cubic (UN_2) structure, and are not miscible with PuN. Anselin was unable to prepare a uranium-plutonium nitride of uniform composition by nitridation of superficially hydrided metal even when uranium-plutonium alloy was used as the source material; however, when the uranium phase was reduced to the mononitride by heating, a rapid inter-diffusion of the uranium and plutonium mononitrides was noted. Investigators at Battelle-Columbus (17) prepared a homogeneous PuN-85 wt% UN composite by hot isostatic pressing of a powder blend of mononitrides for 3 hours at 1550 °C and 10,000 psi, followed by heat treatment for 91 hours at 1720 °C. The hot pressing step alone did not result in complete alloying; no data for intermediate points were reported. In Battelle-Northwest investigations of nitride preparation through carbon-nitrogen reduction of mixed plutonium and uranium oxides, a slightly inhomogeneous uranium-plutonium nitride, somewhat contaminated with carbide, was observed as a direct reaction product. (10) diffusion of the nitrides to form a homogeneous solid solution is apparently rapid, although additional data are needed on the actual diffusion rates, variables affecting these rates, and possible thermal-gradient effects which might promote plutonium migration in a fuel element.

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The mononitrides of uranium and plutonium readily form solid solutions with the monocarbides. Workers at Harwell, (9) investigating the preparation of uranium nitrides and carbonitrides through carbon-nitrogen reduction of the oxides, were unable to produce a mononitride of greater purity than $U(N_{0.96} C_{0.04})$. At Battelle-Northwest, (18) the carbon content of an impure uranium carbonitride was successfully reduced from 4M% to 0.6% by heating in a hydrogen atmosphere, demonstrating the feasibility of producing essentially pure nitrides through the carbon-nitrogen reduction process.

The basic physical properties of uranium and plutonium mononitrides are shown in Table III; the thermal conductivity of uranium mononitride is shown in Figure 1. The properties of the monocarbides are shown for comparison where applicable. The similarity in properties of the nitrides and carbides is apparent from this comparison. For the properties most pertinent to fuel behavior, the nitrides appear equal to or slightly superior to the carbides, although their tendency to vaporize and/or dissociate at temperatures near the melting point would require thorough analysis of transient thermal effects in any specific reactor application.

The thermal conductivity of plutonium nitride has not been reported as experimentally determined; however, it has been calculated (at 180 °C) as being very close to that of uranium nitride. (13) The conductivities of plutonium and uranium nitrides might be expected to be similar in view of the near-identity of their crystal structures. At any rate, the conductivity of a dilute (15-20% PuN) uranium-plutonium nitride should not differ appreciably from that of UN; for the purposes of this study the thermal conductivity values for UN were used.

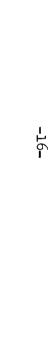
TABLE III PROPERTIES OF NITRIDE AND CARBIDE FUEL MATERIALS

| Material | Density (g/cm3) | Metal Density (g/cm ³) | Melting Point (°C) | Thermal Expansion $(\Delta L/L/^{\circ}C)$ |
|-------------------------------|-----------------|--|--------------------------|--|
| UN ^(19,20,21) | 14.32 | 13.52 | 2850 ^(a) | 9.3 x 10 ⁻⁶ (20-1000°C) |
| UC ⁽²²⁾ | 13.63 | 12.97 | 2560 | 11.6 x 10 ⁻⁶ (25-950°C) |
| PuN (13,23) | 14.52 | 13.4 | 2627 ^(b) | 11.2 x 10 ⁻⁶ (25-1000°C) |
| PuC ^(13,24,25) | 13.6 | 13.0-13.1 | 1654 | 11.0 x 10 ⁻⁶ (25-900°C) |
| UN-20% PuN ^(13,26) | 14.30 | 2.72 ^(c) | 2630 ^(d) | 9.8 x 10 ⁻⁶ (25-1000°C) |
| UC-20% PuC (13,24,25,27) | 13.3 | 2.59 ^(c) | 2430 | ll.l x 10 ⁻⁶ (25-900°C) |

- (a) Congruent melting under 2.5 atm N_2 . (b) Dissociation temperature under 1 atm N_2 .
- (c) Pu metal density.
- (d) Dissociation temperature of UN under 1 atm N2.

Compatibility studies indicate that both UN and PuN are compatible with several potential cladding materials, including stainless steel, zirconium alloys, and refractory metals over long exposure periods and at temperatures of interest to power reactors. (13,28,29) In addition, UN is compatible with NaK. (29) and PuN exhibits compatibility with sodium, NaK, and lithium, up to 800-825 °C, (13) although PuN was attacked by sodium at 950 °C. (14)

Test irradiations of uranium and plutonium nitrides in thermal neutron fluxes under varying conditions of temperature and for varying exposures (10,29)



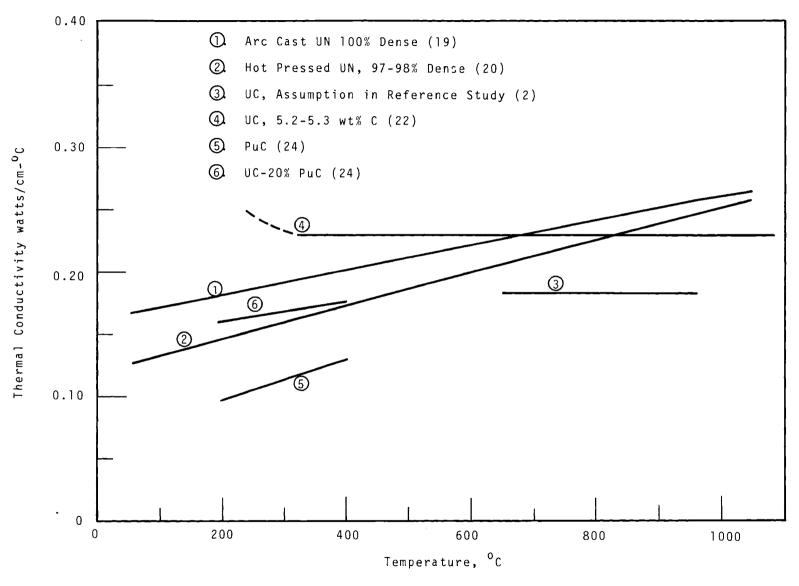


FIGURE 1 Thermal Conductivities of Carbides and Nitrides

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indicated satisfactory behavior to relatively high burnups; capsule exposures in excess of 50,000 MWd/MT were reported. (10) The nitrides appear to be superior to the carbides in resistance to radiation damage.

Because of the (n, α) and (n, p) reactions on nitrogen, the nitrides generate more gas under irradiation than do the carbides. In a fast neutron spectrum, the gas generation by these mechanisms has been estimated at less than 15% of fission gas generation. (29) Limited experimental data (30) indicate that fission gas release from sintered UN is comparable to that of sintered UC of the same density.

In summary, the properties of the mononitrides, their compatibility with potential cladding and coolant materials, and the limited data on their behavior under irradiation, indicate that they should be equal to or superior to the carbides as fuel for fast breeder power reactors. However, more data are needed on the properties and behavior of these materials and on the behavior under irradiation at prototypical reactor conditions of the nitrides and of fabricated nitride fuel elements. In particular, more data are required as to the behavior of nitride fuel elements at temperatures around the melting points of the nitrides; although these temperatures (2750-2850 °C) are considerably above those normally experienced in a reactor, the phase relationships and temperature-nitrogen pressure relationships of nitrides confined in a fuel element would require evaluation for determination of transient response of the fuel for reactor safeguards studies.

Fuel Cycle Study Model

The reactor model developed by Combustion Engineering, as adapted for the nitride fuel cycle study, is a sodium-cooled fast breeder reactor -18- BNWL-606

operating at 2500 MW_t and 1060 MW_e (rounded off to 1000 MW_e for purposes of this study). The reactor features a two-zone core, 30 inches in height and 86.6 inches in diameter. Axial blankets 18 inches thick are located above and below the core; surrounding the core and axial blankets is a radial blanket having an outside diameter of 120 inches. The core contains 157 fuel assemblies of 169 rods each; 79 of these assemblies are in the inner core zone, and 78 in the outer zone. The outer (top and bottom) sections of these rods are filled with depleted UN to form the axial blankets.

The core and axial blanket fuel rods are 0.300 inch in outer diameter, and are clad with type 19-9DL stainless steel with a clad thickness of 0.011 inch. The rods in the axial blanket are 0.45 inch in diameter with a clad thickness of 0.016 inch; again, 19-9DL stainless steel is the clad material.

The design data for the reference reactor model are shown in Table IV.

The Combustion Engineering model included 12 control rod followers, which were fueled with depleted uranium carbide and extended through the core and axial blankets. To simplify the reactor model for physics calculation in the present study, the followers were assumed to be divided into "core" and "axial blanket" sections, and the depleted uranium fuel in each section was "smeared" into the core or axial blanket inventory. Since an entire fuel batch was assumed to be processed through the fuel cycle as a unit, no appreciable difference in fuel cycle inventories or costs should arise from this simplification.

The fuel management scheme used by Combustion Engineering was adopted for this study. In this scheme, the reactor is shut down for fueling operations each 82 days. During each refueling period, one-eighth of the core-and-

| General | Reactor | Data |
|---------|---------|------|
|---------|---------|------|

Net Electrical Power, MWe

Average Core Burnup, MWd/T

Thermal Power, MW

| Core Equivalent Diameter, in. Core Height, inches Axial Blanket Height (top & bottom, Radial Blanket Diameter, in. (O. D. Radial Blanket Height, in. | 30 , each) in. 18 | |
|--|---|--|
| Fuel Loading (No. Assemblies x Ro- Core, inner zone Core, outer zone Axial blanket Control rod followers Radial blanket | ds per Assembly) 79 x 78 x (integral with core f 12 x 156 x | 169 uel) 127 |
| Fuel Rod Data | Core ^(a) | Radial Blanket |
| Fuel Composition Clad Material Fuel Diameter, in. Sodium Bond Thickness, in. Clad I. D., in. Clad Thickness, in. Clad O. D., in. | UC-PuC ^(b) 19-9 DL Stainless Steel 0.259 0.010 0.279 0.011 0.301 | UC (Depl.) 19-9 DL Stainless Steel 0.396 0.011 0.418 0.016 0.450 |
| Volume Percentages Fuel Sodium Stainless Steel | 25.6 66.5 7.9 | 45 43 12 |

1,000

2,500

100,000

Core Thermal Hydraulic Data

| Average Specific Power, kW Linear Power, kW/ft of Fuel | | (۱ | 210 | | |
|---|------------------|----------|-------------|----------------------|---------|
| Maximum | 1 111,0, | | 43. | 18 | |
| Average | | | 28. | | |
| Linear Power, kW/ft of Fuel | Assembly | ,(c) | 20. | | |
| Maximum | | | 7297 | | |
| Average | | | 4761 | | |
| Heat Flux, Btu/hr-ft ² x 10 ⁻⁶ | • | | | | |
| Average | | | 1. | 22 | |
| Maximum | | | 1. | 87 | |
| Average Coolant Velocity, ft | /sec | | 20 | | |
| Coolant Outlet Temperature, | ⁰ F | | | | |
| Average | | | 1120 | | |
| Maximum | | | 1202 | | |
| Maximum Clad Temperature | , ^o F | | 1400 | | |
| Maximum Fuel Temperature | , ^o F | | 2600 | | |
| Hot Channel Factors: | | | | | |
| Coolant | | | | 11 | |
| Surface Film | | | | 11 | |
| Clad | | | | 12 | |
| Sodium Gap | • | | | 15 | |
| Fuel | | | 1. | 07 | |
| | Inner | Outer | Control Rod | Axial ^(d) | Radial |
| Fuel Inventory | Core | Core | Followers | Blanket | Blanket |
| • | | | | | |
| Total Fuel, kg(U+Pu) | 4485 | 4428 | 1140 | 10, 696 | 27,886 |
| Feed Pu Enrichment wt% | | | | | |
| Total | 13.88 | 19.88 | | | |
| Fissile | 10.68 | 15.28 | | | |
| Equilibrium Pu Enrichment, wt% | | | | | |
| Total | 14.10 | 18.30 | 4,44 | 1.73 | 2.10 |
| Fissile | 10.35 | 13.23 | 4.17 | 1.68 | 1.91 |
| | 10, 22 | 4.J. L.J | 7. 11 | 1.00 | 4. / 1 |
| Discharge Pu Enrichment, wt% | | | | | |
| Total | 13.99 | 17.08 | 7.32 | 3.09 | 3.86 |
| Fissile | 9.81 | 11.80 | 6.50 | 2,93 | 3.51 |

TABLE IV Carbide Fueled Reactor Data: Combustion Engineering Reference Model (2)

Physics Parameters

Breeding Ratio, Equilibrium Inner Core (incl. co Outer Core (incl. cor Axial Blanket Radial Blanket Total Reactor

Breeding Ratio, Based on Ma

Reactivity Worth, % ∆k/k

Maximum Fresh Fuel
All Control Rods
Max. Worth, Any On
Sodium Voiding Effec
Sodium Remove

Sodium Temperature

Doppler Coefficient, a

(a) Axial blanket integral wi

(b) Depleted UC in axial blar

(c) Linear power data not pr herein were calculated f

(d) Data originally given for single zone for this stud

| Core | Thermal | Hydraulic | Data |
|------|---------|-----------|------|
| | | | |

| | 000 500 | Average Specific Power, k Linear Power, kW/ft of Fue | W/kg(U+Pu | (۱ | 210 | | |
|--------------------------------------|-------------------------|---|---|-------|-------------|----------------------|---------|
| | ,000 | Maximum | ,1 1 111 | | 43. | 18 | |
| | | Average | | | 28. | 17 | |
| 86 | 5.56 | Linear Power, kW/ft of Fue | el Assembly | ,(c) | | | |
| 30 | | Maximum | ,,,,,,,,,,,,,,,,,,,,,,,,,,,,,,,,,,,,,,, | | 7297 | | |
| 18 | | Average | | | 4761 | | |
| 120 | | Heat Flux, Btu/hr-ft ² x 10 | -6 | | | | |
| 54 | | Average | | • | 1. | 22 | |
| | | Maximum | | * | | .87 | |
| nbly) | | Average Coolant Velocity, | ft/sec | | 20 | | |
| | : 169 | Coolant Outlet Temperatur | | | | | |
| | (169 | Average | •, | | 1120 | | |
| aral with core | | Maximum | | | 1202 | | |
| , | (127 | Maximum Clad Temperatur | re OF | | 1400 | | |
| 156 x | | Maximum Fuel Temperatur | | | 2600 | | |
| | · | Hot Channel Factors: | | | | | |
| re ^(a) | Radial Blanket | Coolant | | | 1. | 11 | |
| _ | | Surface Film | | | | 11 | |
| | UC (Depl.) | Clad | | | | 12 | |
| ainless Steel | 19-9 DL Stainless Steel | Sodium Gap | | | | 15 | |
| ,,,,,,,,,,,,,,,,,,,,,,,,,,,,,,,,,,,, | 0.396 | Fuel | | | | 07 | |
| | 0.011 | 1 401 | | | | | |
| | 0.418 | | Inner | Outer | Control Rod | Axial ^(d) | Radial |
| | 0.016 | Fuel Inventory | Core | Core | Followers | Blanket | Blanket |
| | 0.450 | · | | | | | |
| | o. 150 | Total Fuel, kg(U+Pu) | 4485 | 4428 | 1140 | 10, 696 | 27,886 |
| | | Feed Pu Enrichment wt% | | | | | |
| | 45 | Total | 13.88 | 19.88 | | | |
| | 43 | Fissile | 10.68 | 15.28 | | | |
| | 12 | Equilibrium Pu Enrichment, wt% | | | | | |
| | | | 14.10 | 18 30 | 4.44 | 1.73 | 2.10 |
| | | Total Fissile | 10.35 | 13.23 | 4.17 | 1.68 | 1.91 |
| | | LI22116 | 10.33 | 17.27 | 4.11 | 1.00 | 1. 71 |
| | | Discharge Pu Enrichment, wt% | | | | | |
| | | Total | 13.99 | 17.08 | 7.32 | 3.09 | 3.86 |
| | | Fissile | 9.81 | 11.80 | 6.50 | 2.93 | 3.51 |
| | | | | | | | |

TABLE IV Carbide Fueled Reactor Data:
Combustion Engineering Reference Model(2)

| Breeding Ratio, Equilibrium Reactor | |
|--|---------------------|
| Inner Core (incl. control followers) | 0.392 |
| Outer Core (incl. control followers) | 0.256 |
| Axial Blanket | 0.346 |
| Radial Blanket | 0.427 |
| Total Reactor | 1.421 |
| Breeding Ratio, Based on Materials Inventory | 1.372 |
| Reactivity Worth, %∆k/k | |
| | |
| Maximum Fresh Fuel Element | 0.33 |
| Maximum Fresh Fuel Element All Control Rods | 0.33 4.0 |
| | |
| All Control Rods Max. Worth, Any One Rod | 4.0 |
| All Control Rods Max. Worth, Any One Rod Sodium Voiding Effects | 4.0 |
| All Control Rods Max. Worth, Any One Rod | 4.0 0.36 |
| All Control Rods Max. Worth, Any One Rod Sodium Voiding Effects Sodium Removed from Core Only Sodium Removed from Entire Reactor | 4.0 0.36 +2.4 |
| All Control Rods Max. Worth, Any One Rod Sodium Voiding Effects Sodium Removed from Core Only | 4.0 0.36 +2.4 |

- (a) Axial blanket integral with core fuel rods (b) Depleted UC in axial blanket section
- (c) Linear power data not presented in Reference 2; data presented herein were calculated from heat flux data in Reference 2
- (d) Data originally given for 2-zone axial blanket; combined to single zone for this study.

axial-blanket assemblies and 1/24 of the radial blanket assemblies are discharged and fresh assemblies installed. Although these quantities do not equate to integral numbers of fuel assemblies, they may be validly considered as time-averaged values. With this refueling scheme, and with an 80% plant operating factor, core fuel receives an average 100,000 MWd/ton burnup.

The fuel element assumed in the Combustion Engineering study was based on cast carbide pellets of 100% theoretical density with a 0.010-inch sodium-filled gap for thermal bonding to the clad. For the present study, the nitride fuel elements were assumed to be fabricated by vibratory compaction of uranium nitride or uranium-plutonium nitride within the cladding, followed by isostatic hot pressing of the clad element in a high-temperature, high-pressure gas autoclave. A fuel density of about 90% of theoretical was assumed, to maintain approximately the same fuel-metal density (U + Pu) in the reactor as with the carbide fuel, while eliminating the sodium bond. This fabrication technique should permit a slight reduction in fabrication cost as compared to the carbide fabrication process, and eliminates the necessity of head-end treatment to remove sodium prior to processing the irradiated fuel.

Thermal contact between fuel and clad was assumed to be by mechanical bonding, augmented by a nitrogen diffusion bond between the fuel and the clad. The conductance of the fuel-to-clad bond was assumed at 5000 Btu/(hr)(ft²)(°F).

The integrity of such a thermal bond would require considerable experimental verification, but the magnitude of this heat transfer coefficient appears reasonable for bonding of this nature. Temperature capabilities of the

nitrides appear adequate to accommodate even a large error in the assumed bond conductance, although fission gas release at higher temperatures would require careful evaluation. Finally, even if a mechanical bond should prove unreliable, the use of a denser nitride with sodium-bond thermal path could be resorted to as a backup design with a small penalty in added costs. The nitrides appear quite compatible with sodium at temperatures in the bonding region.

The fuel cycle utilized in this study assumes that all scrap material (from fabrication, etc.) is internally recycled; that plutonium is internally recycled, with only the excess removed for sale; and that reprocessed uranium, as UNH, is returned to the supplier at no charge, with new, depleted (0.3%) uranium being supplied to the conversion-fabrication step of the fuel cycle. This cycle, based on use of the NFS processing plant, is depicted in flowsheet form in Figure 2. During the course of the study, cursory surveys were made of variations in this scheme, including consideration of onsite rather than central plant processing; the effects of multiple-reactor installations with onsite processing; use of ion-exchange separation processes, and various nonaqueous processes, as alternates to the solvent-extraction process as represented by the NFS plant; and complete recycling of uranium rather than the use of new depleted uranium for each batch. However, although such variations appeared to have some minor effect on fuel cycle costs for both carbide and nitride systems, the only factor investigated which appeared to affect significantly the comparison between nitride and carbide fuel cycle costs was the choice of process used for conversion of the fuel materials to the nitrides.

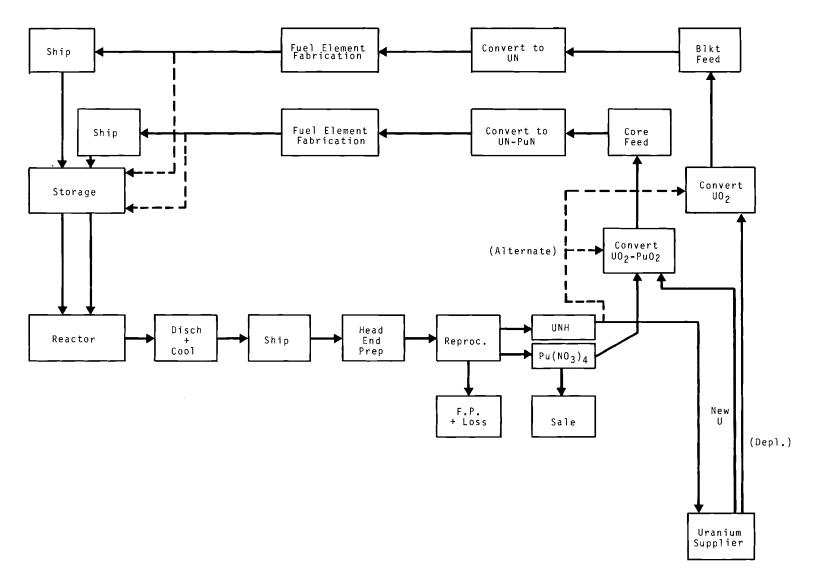


FIGURE 2
Assumed Fuel Cycle Scheme for Study

-2¹4- BNWL-606

Nuclear Analysis

Analysis of the nucleonic behavior of a nitride-fueled fast breeder reactor entailed the use of calculation techniques, computer programs, and cross section data which differed from those used by Combustion Engineering for analysis of their carbide-fueled fast breeder reactor concept used as a model in this study. Consequently, in order to provide a valid comparison between the nitride and carbide fueled reactors, it was necessary to recalculate the carbide-fueled reactor behavior using the same techniques employed for the nitride fuel calculations.

Reactor Physics Model

For purposes of the physics calculations, the reactor was divided into five regions as indicated in Figure 3. These regions include two concentric zones into which the core is subdivided; an axial blanket above and below the core; a radial blanket; and a small region at the center of the core. The last region was included specifically for the study of reactivity trends during the early phases of a sodium-voiding accident. For calculations not concerned with sodium voiding, this zone was merged with the central core region.

The elemental composition of the five regions, expressed as atom densities in an equilibrium core, are indicated in Table V. Compositions are given for both a nitride- and a carbide-fueled model as used in Battelle-Northwest analyses. Uranium and plutonium isotopic densities for the core regions are those computed to be required for criticality margins in the equilibrium core, and were obtained by iteration of the criticality and burnup calculations described below. Total fuel loadings for each region, and

| Core #1 (Region #1) | height Radius | 76.2 76.2 | | in.) |
|--|---------------------------|----------------|------------------|-----------------|
| Core #2 (Region #2) | height Radius | 76.2 109.2 | | in.) |
| Radial Blanket (Region #3) | height Radius | 167.6 152.4 | cm (66 cm (60 | in.) in.) |
| Axial Blanket (one on each end- Region #4) | height Radius | 45.7 109.2 | cm (18 cm (43 | in.) .2 in.) |
| Core Center (Region #5) | he i ght Radius | 5.0 16.0 | | |

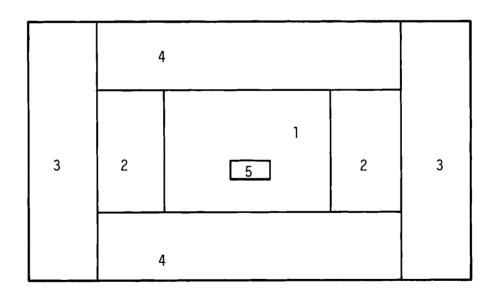


FIGURE 3
Reactor Physics Model

TABLE V

ELEMENTAL COMPOSITION OF REACTOR MODEL AT EQUILIBRIUM BURNUP

Atom Densities, atoms/cm³ x 10²⁴

| | Nitride | BNW Carbide (a) |
|---|---|---|
| Region 1 and 5(b) Pu239 Pu240 Pu238 Fe Cr Ni Na N or C F.P. (Pairs) | 9.34 x 10 ⁻¹ 4 3.14 x 10 ⁻¹ 4 6.50 x 10 ⁻³ 4.68 x 10 ⁻³ 1.48 x 10 ⁻³ 7.90 x 10 ⁻¹ 1.43 x 10 ⁻² 8.30 x 10 ⁻³ 5.50 x 10 ⁻¹ | 8.92 x 10 ⁻¹⁴ 2.93 x 10 ⁻¹⁴ 6.58 x 10 ⁻³ 1.48 x 10 ⁻³ 7.90 x 10 ⁻¹⁴ 1.43 x 10 ⁻² 8.30 x 10 ⁻³ 5.30 x 10 ⁻¹⁴ |
| Region 2(b) Pu239 Pu240 Pu238 Fe Cr Ni Na N or C F.P. (Pairs) | 1.17 x 10 ⁻³ 4.03 x 10 ⁻⁴ 6.27 x 10 ⁻³ 4.68 x 10 ⁻³ 1.48 x 10 ⁻³ 7.90 x 10 ⁻⁴ 1.43 x 10 ⁻² 8.30 x 10 ⁻³ 4.59 x 10 ⁻⁴ | 1.13 x 10 ⁻³ 3.81 x 10 ⁻⁴ 6.33 x 10 ⁻³ 4.68 x 10 ⁻³ 1.48 x 10 ⁻³ 7.90 x 10 ⁻⁴ 1.43 x 10 ⁻⁴ 8.30 x 10 ⁻³ 4.55 x 10 ⁻⁴ |

(Blanket Regions: Nitride and Carbide)

| Region 3(c) | | Region 4(c) | |
|-------------------|-------------------------|---------------------|-------------------------|
| Pu ²³⁹ | 2.75 x 10 ⁻⁴ | P11 ² 39 | 1.46×10^{-4} |
| Pu ²⁴⁰ | 2.70×10^{-5} | Pu ²⁴ 0 | 6.00×10^{-6} |
| _Մ 238 | 1.37×10^{-2} | _U 238 | 8.11×10^{-3} |
| Fe | 7.12×10^{-3} | Fe | 4.68×10^{-3} |
| Cr | 2.25×10^{-3} | Cr | 1.48×10^{-3} |
| Ni | 1.20×10^{-3} | Ni | 7.90×10^{-4} |
| Na | 9.25 x 10 ⁻³ | Na | 1.43×10^{-2} |
| N or C | 1.46 x 10 ⁻² | N or C | 8.30 x 10 ⁻³ |
| F.P. (Pairs) | 1.50×10^{-4} | F.P. (Pairs) | 6.50 x 10 ⁻⁵ |

- (a) "BNW Carbide" represents the Combustion Engineering reactor model recalculated at Battelle-Northwest by same techniques used for nitride fuel model.
- (b) Core region densities iterated for convergence at equilibrium burnup.
- (c) Blanket region densities assumed identical to those of Combustion Engineering model.

therefore the fuel densities for each region, were assumed identical to those of the Combustion Engineering carbide-fueled reference model; plutonium enrichment of the core regions was varied to satisfy criticality requirements.

For purposes of the model, the plutonium content of the blanket regions was assumed to be identical to that of the Combustion Engineering reactor model. Subsequent burnup calculations indicated slight differences in equilibrium blanket composition between the nitride and carbide models. While these differences were taken into account in fuel materials balance calculations, the effect on reactor criticality was so slight that correction of the equilibrium-reactor model was not deemed to be warranted.

Calculation Techniques

Reactor Statics - For calculations of reactor criticality, neutron flux spectrum, and radial power distribution, the one- and two-dimensional transport theory codes DTK and DDK⁽³¹⁾ were employed. For calculations using the one-dimensional code DTK, a 26-group Russian cross section set⁽³²⁾ was used. For use with the two-dimensional DDK, these cross sections were collapsed to an equivalent 4-group set, using the zero-dimensional fundamental mode processing codes, RED CROSS⁽³³⁾ and CRUNCH.⁽³⁴⁾ Tables VI and VII compare the energy-lethargy structures of the original 26-group set and of the collapsed 4-group set.

As an additional check on criticality calculations for the nitrideand carbide-fueled reactor models, further calculations were made using the DTK code and also the HFN (35) diffusion-theory code. For these calculations a nitrogen-augmented Yiftah-Okrent-Moldauer (YOM) cross section set was used. The nitrogen cross section data for inclusion in this set were compiled -28- BNWL-606

TABLE VI
26-GROUP ENERGY-LETHARGY STRUCTURE

| Group No. | Lower Energy | Higher Lethargy | Fission Neutron Fraction |
|---------------|------------------------|--------------------|-----------------------------|
| | 10.5 Mev (upper limit) | | |
| 1 | 6.5 | 0.48 | 0,020 |
| 2 | 4.0 | 0 ູ 96 | 0.098 |
| 3 | 2.5 | 1.44 | 0,190 |
| 3 4 | 1.4 | 2.01 | 0.268 |
| | 0.8 | 2.58 | 0.196 |
| 5 6 | 0 . 4 | 3.27 | 0.135 |
| 7 | 0.2 | 3.69 | 0.058 |
| 7 8 | 0.1 | 4.65 | 0.022 |
| 9 | 46.5 keV | 5.42 | 0.009 |
| 10 | 21.5 | 6.19 | 0.003 |
| 11 | 10.0 | 6. ₉₆ | 0.001 |
| 12 | 4.65 | 7.73 | 0 |
| 13 | 2.15 | 8 - 50 | 0 |
| 14 | 1.0 | 9.27 | ¹ O |
| 15 | 465.0 eV | 10.04 | 0 |
| 16 | 215.0 | 10.81 | 0 |
| 17 | 100.0 | 11.58 | 0 |
| 18 | 46.5 | 12.35 | 0 |
| 19 | 21.5 | 13.12 | 0 |
| 20 | 10.0 | 13.89 | 0 |
| 21 | 4.65 | 14,66 | 0 |
| 22 | 2.15 | 15,43 | 0 |
| 23 | 1.0 | 16.20 | О |
| 24 | 0.465 | 16.97 | 0 |
| 25 | 0.215 | 17.74 | 0 |
| 26 | Thermal | | 0 |

TABLE VII
4-GROUP ENERGY-LETHARGY STRUCTURE

| Group No. | Lower Energy | Higher <u>Letharg</u> y | Fission Neutron Fraction |
|-----------|------------------------|----------------------------|--------------------------|
| | 10.5 MeV (upper limit) | | |
| 1 | 1.4 | 2.01 | 0.576 |
| 2 | 0.1 | 4.65 | 0.411 |
| 3 | 215.0 eV | 10.81 | 0.013 |
| 4 | Thermal | | . 0 |

from the Hanford Basic Data Tape (37) with the processing code HRG. The energy structure for this cross section set is that of the original YOM grouping; Combustion Engineering had used a modified YOM energy structure in their calculations. (2)

Sodium Removal Coefficients - Sodium removal coefficients (sodium void coefficients) were calculated with the DDK code, using the full 5-region reactor model. Separate calculations were made for cases assuming voiding of the entire reactor or of specific regions of the reactor model. Included in these calculations was a case assuming the voiding of sodium from only the small central zone within the core (Region 5), to estimate the initial trends in reactivity at the start of a voiding incident.

Eurnup - Reactor burnup analyses were performed with the zero-dimensional code PYRE, (39) using the 26-group Russian cross section set. This code operates on a time-step basis and is designed for computation of one region fuel burnup in fast spectrum reactors. At each time step the neutron spectrum for an infinite reactor is generated, based on the atom densities and cross sections of the various elements present in the region under study, and on the buckling required for criticality. Spectrum-averaged, 1-group macroscopic cross sections are then generated for each isotope of uranium or plutonium (or other burnable species) and used to calculate new concentrations of these isotopes at the end of the time period covered by the calculation. These steps are repeated to completion of the irradiation. The burnup calculations were run at constant flux, as computed from the power-normalized DTK-DDK equilibrium fuel cases. Since this is a region-independent code, it was necessary to repeat the calculations for each region of the reactor model.

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Prior to the burnup analysis, region equilibrium fuel densities were calculated based on criticality, power distribution and plutonium isotope ratio requirements. To perform the burnup analysis, initial cold-clean reprocessed fuel densities for each region were assumed based on Combustion Engineering data, adjusted for required enrichment, and burned for a fuel life of 100,000 MWd/ton. Mid-term fuel was then examined for correct equilibrium fuel densities. This process was repeated with progressively corrected initial fuel densities until correct mid-term fuel composition was obtained.

Doppler Coefficients - Doppler coefficients were calculated using resonance self-shielded data generated with the RED CROSS program, which uses temperature-corrected resonance self-shielding factors from the Russian cross section data compilation. Self-shielding factors for fission, neutron capture, transport, and elastic scattering are included in the compilation and thus enter into the Doppler coefficient calculations. The RED CROSS code uses these self-shielding factors to calculate multigroup, temperature-corrected mixed cross sections for the reactor loading at an assumed temperature and finally calculates a criticality factor for the reactor. Criticality factors were calculated for several assumed temperatures at constant buckling;

Results of Analyses

System Criticality - Because of the variation in results which can occur in criticality calculations when different cross section compilations and different computational techniques are used, it was necessary that the primary DTK-DDK criticality calculations be backed up with calculations performed by other techniques in order that a valid comparison could be made

of the BNW results with those of Combustion Engineering (2) and of an AEC-sponsored review (3) of several fast breeder reactor concepts. To establish this comparison, calculations for nitride- and carbide-fueled reactors were performed using several computation techniques, assuming the same fuel densities in each case as were used by Combustion Engineering and the AEC review team. Results of these calculations are summarized in Table VIII.

<u>TABLE VIII</u>

<u>k</u>eff <u>COMPARISONS FOR CARBIDE</u> <u>AND NITRIDE</u> <u>FUELED REACTORS</u>

| Cross Section Set | Nitride Fuel | Carbide Fuel | | | |
|-----------------------------------|----------------------|----------------------|----------|-----------|--|
| and Reactor Code | (BNW)(a) | (BNW)(a) | (AEC)(b) | (C.E.)(c) | |
| 26-group Russian (DDK-DTK) | 0.9237 | 0.944 | | | |
| 16-group YOM (DTK) | 0.9682 | | | | |
| 16-group Hansen (DDK) | | | 0.9304 | | |
| 16-group Modified YOM (HFN, FAIM) | 1.066 ^(d) | 1.085 ^(d) | | 1.000 | |
| 22-group ANL (CRAM) | | | 0.9235 | | |

- (a) Per BNW calculations, this study.
- (b) As reported in Reference 3.
- (c) As reported in Reference 2.
- (c) Original YOM energy-lethargy structure used in Battelle-Northwest calculations.

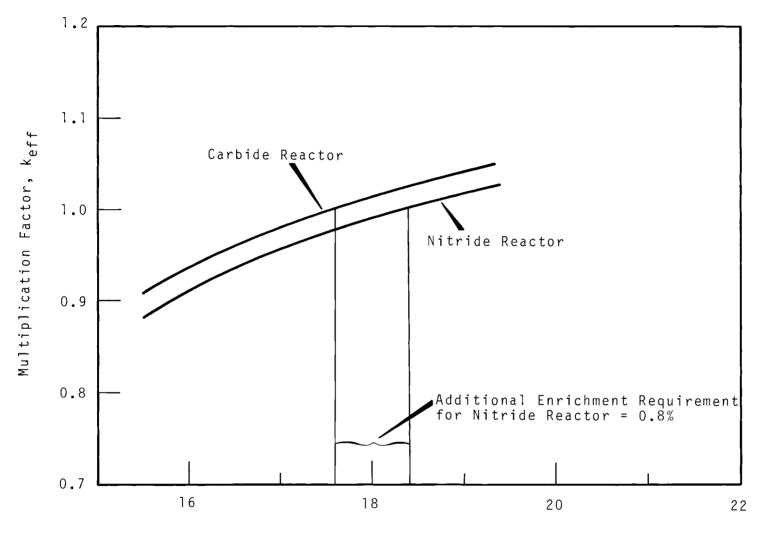
These results graphically illustrate the scatter resulting from the use of different compilations of cross section data and different computer programs. Comparison of the results shows that the DDK-DTK calculations, while somewhat more pessimistic than the results obtained by Combustion Engineering, agree quite closely with those of the AEC review. They also indicate the

slightly lower k of a nitride-fueled reactor as compared with one fueled with carbide, for equivalent fuel densities.

Following the establishment of the correlations shown in Table VIII, the remainder of the criticality calculations for this study were performed using the DTK and DDK codes with the Russian cross section set or its collapsed equivalent.

The variation of multiplication factor k_{eff} with core enrichment, for nitride- and carbide-fueled reactors at equilibrium conditions is shown in Figure 4. These factors were calculated using the DTK code; enrichment is expressed as total plutonium content, averaged over the two-region core. Plutonium content of the blankets was assumed to be the equilibrium content quoted by Combustion Engineering for those regions. The nitride-fueled reactor model apparently requires a slightly higher core plutonium content than the carbide-fueled model. The required plutonium content of the equilibrium nitride core, expressed as weight percent total plutonium in the fuel metal (excluding fission products), is 18.4% as compared with 17.6% for carbide fuel. Based on total metal weights as initially loaded--and thus including fission products in the equilibrium fuel weight compilation--these plutonium contents are 17.1% and 16.3% respectively. The original Combustion Engineering study indicated a required equilibrium plutonium content of 16.2%, which is based on total initially loaded metal weights.

With the reactor fuel at an assumed average temperature of 1300 °C, the nitride-fueled reactor would exhibit a calculated criticality factor of 0.9960 by DTK calculations or of 0.9795 by DDK calculations.



Total Pu Content (Equilibrium Core), Percent

FIGURE 4
Equilibrium Nitride and Carbide Reactors - Multiplication Factor Versus
Enrichment

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Neutron Spectra - Neutron spectra as calculated for the equilibrium nitride reactor are shown in Figure 5. Spectra are shown for the two core regions (Regions 1 and 2 of the model) and for the radial blanket. The curves in this figure have been normalized such that the area under each curve equals unity. These curves show the spectrum in the blanket region to be considerably softer than those in the core region. Likewise, the outer core region (Region 2), which is more highly enriched than the inner core, has a slightly harder spectrum.

A comparison of inner-core (Region 1) spectra for nitride- and carbide-fueled reactors is shown in Figure 6. The spectrum obtained by Combustion Engineering for the corresponding region of their reactor model is also shown. Figure 6 indicates the spectrum for the nitride-fueled reactor to be slightly harder than that for the (BNW) carbide-fueled model. The harder spectrum with the nitride fuel can be attributed in part to the somewhat higher enrichment in the nitride core and to the slightly lesser thermalization of neutrons by scattering with nitrogen than with carbon.

Comparison of the Battelle-Northwest and Combustion Engineering spectra for carbide cores reveals more fundamental differences. The Battelle-Northwest spectrum has a lesser peak-flux value than the Combustion Engineering spectrum; it is displaced slightly toward higher neutron energies, and exhibits a slight dip in the 3 keV sodium resonance-capture region which is not shown in the Combustion Engineering spectrum. These differences are attributable both to differences in calculation techniques and to differences in the cross section data used.

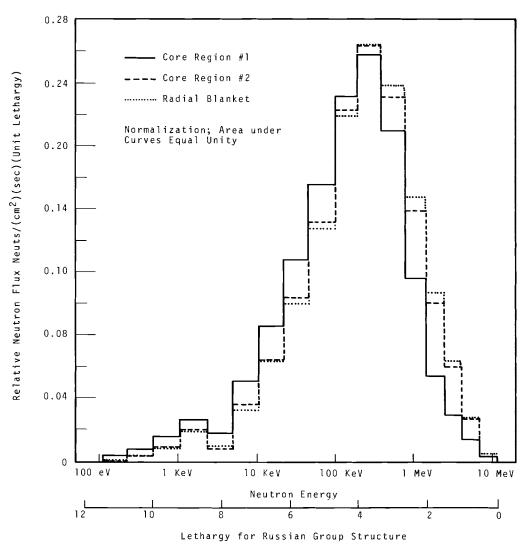


FIGURE 5
Nitride Reactor Spectra



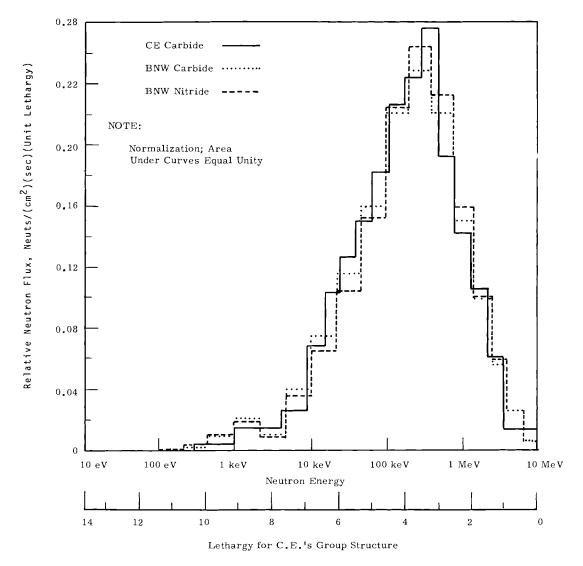


FIGURE 6 Comparison of CE Carbide, BNW Carbide, and BNW Nitride Reactors - Core I Spectra

By using different combinations of DTK and HFN computer codes with the Russian and YOM cross section sets, alternate calculations at Battelle-Northwest produced the spectra shown in Figure 7. The combination of the HFN code with the YOM cross section set produces a spectrum which nearly duplicates that reported by Combustion Engineering. However, the Russian cross section set apparently provides a more correct accounting for the 3 keV sodium resonance, as indicated by the spectra developed by use of this set; it is also believed to give a more accurate accounting for nitrogen in the nitride fuel evaluations. The apparent improvement in handling of sodium resonances with the Russian cross section set, incidentally, indicates that calculations of sodium voiding effects by use of this set may be more accurate than those using the YOM cross section set.

The median absorption and fission energies as calculated for the various regions of the nitride- and carbide- reactor models, along with corresponding values reported by Combustion Engineering in their study, are given in Table IX. These figures reflect the somewhat harder spectra calculated at Battelle-Northwest as compared with the Combustion Engineering calculations, and also the harder spectrum of the nitride-fueled model as compared with a carbide-fueled reactor. Particularly noteworthy are the calculated median absorption energies of the sodium-voided cores. The nitride-fueled model apparently exhibits a considerably greater increase in absorption energy on voiding of sodium than does the carbide-fueled one. As discussed in subsequent sections of this report, this difference is significant in its effect on the reactivity changes occurring upon voiding of sodium.

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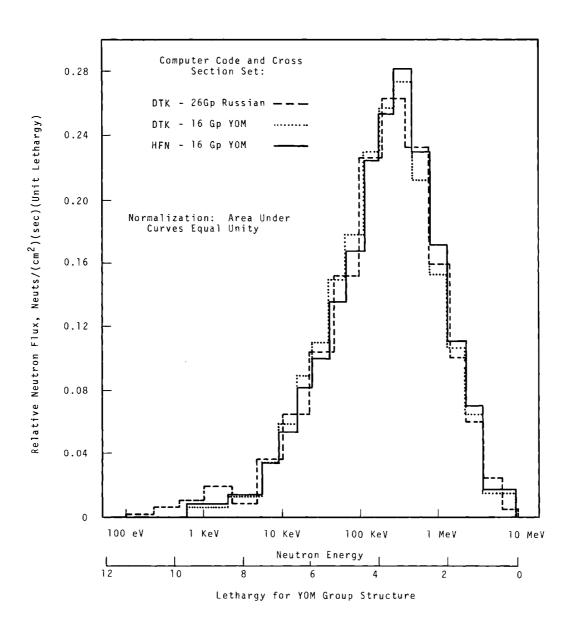


FIGURE 7
Comparison of Nitride Core I Spectra Calculated with Russian and YOM Cross Sections

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TABLE IX

MEDIAN ABSORPTION AND FISSION ENERGIES

| | Absorption Energy, keV | | Fis | gy, keV | | |
|-----------------------------|------------------------|-----------------------|-------------------|-----------------------|----------------|------------------|
| Region | BNW <u>Nitride</u> | BNW <u>Carbide</u> | CE Carbide (a) | BNW <u>Nitride</u> | BNW Carbide | CE Carbide(a) |
| Inner Core | 150 | 115 | 92 | 440 | 410 | 302 |
| Inner Core - Sodium Void | 370 | 250 | | | | |
| Outer Core | 185 | 132 | 110 | 450 | 420 | 311 |
| Radial Blanket | 80 | 65 | 52 | 580 | 570 | 354 |
| Axial Blanket | 43 | 38 | 34(Avg) | 690 | 680 | 448(Avg) |

(a) Combustion Engineering carbide values as reported in Reference 2.

Fuel Managent, Materials Inventory, and Breeding Ratio - The fuel management scheme assumed in this study for the nitride- and carbide-fueled reactor models was briefly described previously in this report. This scheme is essentially identical to that used by Combustion Engineering in their study; like that scheme, it is based on a graded cycle with average core fuel burnup of 100,000 MWd/ton.* Under this scheme, partial refueling of the reactor is performed each 82 days (an exposure interval of 12,500 MWd/ton for core fuel, based on 80% plant factor). One-eight of the core (and axial blanket) elements and 1/24 of the radial blanket elements are discharged and replaced

^{*} In the Combustion Engineering study fuel burnup values were based on the 2000-pound ton rather than the metric ton; to permit direct comparison of the results, the same units were used in the present study.

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with new elements at each refueling. Radial blanket fuel thus receives an exposure in the reactor equivalent to 300,000 MWd/ton core fuel exposure.

The frequency of refueling in this scheme implies that conditions in each fuel region will not vary greatly from equilibrium conditions at any time; thus the neutron flux in each region will be approximately constant. Fuel burnup calculations were made assuming constant flux in each region.

The region-independent PYRE analysis used in this study gave results for the core regions which were quite comparable with those of the Combustion Engineering study⁽²⁾ and the AEC-sponsored fast breeder reactor design review.⁽³⁾ However, for the blanket regions the PYRE analysis indicated an unduly high buildup of plutonium. The tendency of PYRE to produce overoptimistic results in blanket-region calculations had been noted in previous studies. Therefore, the blanket plutonium concentrations calculated by PYRE were normalized to the results quoted by Combustion Engineering. The calculated total plutonium contents of the blanket regions of the carbide reactor were reduced to agree with the Combustion Engineering results. The concentrations calculated for the nitride-fueled reactor were then reduced by the same proportion for normalization. The isotopic content of the plutonium in each case was retained as calculated.

Tables X and XI show the feed, equilibrium and discharge core plutonium concentrations and compositions for the BNW nitride and carbide reactors. Also included are the blanket discharge plutonium concentrations and compositions. The blanket fuel material is depleted uranium. As might be expected, plutonium buildup in the blanket regions is approximately proportional to the blanket exposure.

TABLE X

NITRIDE REACTOR PLUTONIUM CONTENT

CORE PLUTONIUM CONCENTRATION AND COMPOSITION

| | Pu Concentration, wt% | | Pu Isotopic Content, | | | nt, wt% |
|-----------------|-----------------------|----------------|----------------------|-------------------|-------------------|-------------------|
| | Total | <u>Fissile</u> | Pu ²³⁹ | Pu ²⁴⁰ | Pu ²⁴¹ | Pu ²⁴² |
| Inner Core Zone | | | | | | |
| Feed | 15.64 | 12.04 | 73.3 | 22.2 | 3.6 | •9 |
| Equilibrium | 14.93 | 11.13 | 70.2 | 24.3 | 4.4 | 1.1 |
| Discharge | 14.30 | 10.39 | 67.6 | 25.9 | 5.0 | 1.5 |
| Outer Core Zone | | | | | | |
| Feed | 20.89 | 16.04 | 73.3 | 22.2 | 3.6 | .9 |
| Equilibrium | 18.92 | 14.05 | 69.9 | 24.7 | 4.3 | 1.1 |
| Discharge | 17.43 | 12.57 | 67.1 | 26.5 | 5.0 | 1.4 |

BLANKET PLUTONIUM CONCENTRATION AND COMPOSITION AT DISCHARGE

| | Pu Concentration, wt% | | Pu Isotopic Content, w | | |
|----------------|-----------------------|----------------|------------------------|-------------------|--------------------|
| | Total | <u>Fissile</u> | Pu ²³⁹ | Pu ²⁴⁰ | Pu ²⁴ 1 |
| Axial Blanket | 2.96(a) | 2.83 | 95.5 | 4.3 | 0.2 |
| Radial Blanket | 3.66 ^(a) | 3.43 | 93.3 | 6.3 | 0.4 |

(a) Normalized to Combustion Engineering reactor by comparison to Battelle-Northwest carbide reactor.

Table XII summarizes the core and blanket fuel residence time and flux exposure rate. The region average neutron fluxes presented in Table XII were calculated based on the total reactor output power and the unit fission neutron normalized fluxes calculated by DTK and DDK.

TABLE XI

BATTELLE_NORTHWEST CARBIDE REACTOR PLUTONIUM CONTENT

| | CORE PLUTONIUM CONCENTRATION AND COMPOSITION | | | | | |
|-----------------|--|----------------|-------------------|-------------------|-------------------|-------------------|
| | Pu Concent | tration, wt% | Pu Is | otopic | Content | , wt% |
| | Total | <u>Fissile</u> | Pu ²³⁹ | Pu ²⁴⁰ | Pu ²⁴¹ | Pu ²⁴² |
| Inner Core Zone | | | | | | |
| Feed | 14.25 | 10.93 | 73.3 | 22.2 | 3.6 | 0.9 |
| Equilibrium | 14.20 | 10.64 | 70.6 | 24.0 | 4.3 | 1.1 |
| Discharge | 13.94 | 10.21 | 68.3 | 25.4 | 4.9 | 1.4 |
| Outer Core Zone | | | | | | |
| Feed | 19.74 | 15.21 | 73.3 | 22.2 | 3.6 | 0.9 |
| Equilibrium | 18.09 | 13.48 | 70.3 | 24.4 | 4.2 | 1.1 |
| Discharge | 16.83 | 12,20 | 67.6 | 26.2 | 4.9 | 1.3 |

BLANKET PLUTONIUM CONCENTRATION AND COMPOSITION AT DISCHARGE

| | Pu Concentration, wt% | | Pu Isotopic Cont | | ent, wt% | |
|----------------|-----------------------|----------------|-------------------|-------------------|-------------------|--|
| | Total | <u>Fissile</u> | Pu ²³⁹ | Pu ²⁴⁰ | Pu ²⁴¹ | |
| Axial Blanket | 3.09 ^(a) | 2.95 | 95.5 | 4.3 | 0.2 | |
| Radial Blanket | 3.86 ^(a) | 3.61 | 93.2 | 6.4 | 0.4 | |

⁽a) Battelle-Northwest reactor results normalized to Combustion Engineering reactor results.

It is interesting to examine the progress of the region dependent burnup, specifically the change in overall breeding ratio, the change in $k_{\mbox{eff}}$ and the change in bundle power with respect to progress of the fuel burnup. These data are summarized in Table XIII. In this table, values

TABLE XII

CORE AND BLANKET FUEL RESIDENCE TIME

AND FLUX EXPOSURE RATE (0.8 PLANT FACTOR)

| | Residence Time | Average New | utron Flux |
|----------------|----------------|--------------------------|-------------------------|
| | (days) | Nitride Reactor | Carbide Reactor |
| Core Region #1 | 657 | 1.027 x 10 ¹⁶ | 1.05 x 10 ¹⁶ |
| Core Region #2 | 657 | 6.74 x 10 ¹⁵ | 6.97 x 10 ¹⁵ |
| Radial Blanket | 1971 | 1.43 x 10 ¹⁵ | 1.48 x 10 ¹⁵ |
| Axial Blanket | 657 | 2.54 x 10 ¹⁵ | 2,61 x 10 ¹⁵ |

given for the instantaneous breeding ratio are typically higher in the carbide reactor, expecially in core regions I and II. The blanket breeding ratios in both reactors are very nearly the same, but slightly higher in the nitride reactor. In the nitride reactor for Core I and Core II, the instantaneous breeding ratio tends to rise with progressing fuel exposure. In Core I of the carbide reactor, the instantaneous breeding ratio tends to decrease slightly with increased fuel exposure. In Core II of this reactor, the instantaneous breeding ratio rises slightly with respect to increased fuel exposure. In the blankets of both the nitride and carbide reactors, the instantaneous breeding ratio for new fuel (new, depleted uranium) is very high; however, it soon falls off as transuranium elements burn in and competing reactions begin to take place.

The bundle powers given in Table XIII are approximate relative to each other and absolute for each given fuel bundle (assuming the constant flux approximation to be correct). One should also note the significant change in bundle power for the term of exposure of a radial blanket fuel element.

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TABLE XIII

FUEL BREEDING RATIO, WORTH AND POWER

VS BURNUP FOR BNW-CALCULATED REACTORS

Nitride Reactor

| Region | Fuel Condition | Breeding Ratio (Instantaneous) | k _{eff} (Infinite Bundle) | Bundle Power, MW (Approx.) |
|-----------------------|----------------------------------|--------------------------------------|--|----------------------------------|
| Core I | Feed Equilibrium Discharge | .828 .861 .878 | 1.05 .989 .939 | 15.4 14.2 13.2 |
| Core II | Feed Equilibrium Discharge | .575 .650 .713 | 1.05 .970 .905 | 12.8 11.4 10.3 |
| Radial (a) Blanket | Feed Equilibrium Discharge | 30.48 3.32 2.08 | .280 .680 .860 | .86 2.25 3.00 |
| Axial Blanket(a) | Feed Equilibrium Discharge | 30.6 5.06 3.06 | .248 .487 .640 | . 34 . 65 . 93 |
| | | Carbide Reactor | | |
| Core I | Feed Equilibrium Discharge | . 946 . 926 . 914 | 1.05 1.01 .977 | 14.6 14.0 13.3 |
| Core II | Feed Equilibrium Discharge | . 625 . 695 . 7 51 | 1.05 .975 .915 | 12.7 11.4 10.4 |
| Radial Blanket(a) | Feed Equilibrium Discharge | 30.47 3.25 2.03 | . 282 . 713 . 898 | .88 2.41 3.22 |
| Axial Blanket(a) | Feed Equilibrium Discharge | 31.08 5.03 3.02 | . 253 . 506 . 666 | .35 .71 .98 |

⁽a) Unnormalized data.

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It is also interesting to note the k_{∞} for these various fuel bundles. In both the nitride and carbide reactors the core bundles have criticality factors very near 1. The blanket bundles have criticality factors substantially less than 1. These numbers are intimately connected with the way the region burnup calculations were done. In the case of the core regions a k_{eff} was used. Based on the neutron balance calculated within PYRE, a buckling necessary to maintain $k_{\text{eff}} = 1$ is calculated at each time step and the cross sections are weighted by the buckling adjusted spectrum. This operation simulates a controlled reactor. Also in these core cases reactivity vs time was calculated by specifying an initial $k_{\text{eff}} = 1.05$ and then evaluating k_{eff} at each time step using the initial buckling. These values of k_{eff} are given in Table XIII. The blanket burnup calculations were done at a constant geometric buckling of 1 x 10⁻¹⁴; the criticality factor was calculated as a function of fuel exposure. The cross sections in the blanket burnup were not adjusted for criticality at each time step.

With the equilibrium condition instantaneous breeding ratio values given in Table XIII, it is possible to calculate a total reactor instantaneous breeding ratio. In Table XIV reactor breeding ratios for the equilibrium reactor burned to 50,000 MWd/ton are given for the BNW nitride, BNW carbide, and Combustion Engineering carbide reactors. The region internal breeding ratios are weighted by the corresponding zone fuel throughputs to give a contribution to the total breeding ratio. Contributions to the total breeding ratio are then summed up to give the total instantaneous or total average breeding ratio for the reactor. The BNW carbide reactor has a slightly

lower breeding ratio than the Combustion Engineering reactor. The breeding ratio of the nitride reactor is lower than that of the BNW carbide reactor.

TABLE XIV

REACTOR BREEDING RATIOS AT EQUILIBRIUM EXPOSURE

(50,000 MWd/ton)

| | Battelle-Northwest Nitride Reactor | | Battelle-Northwest Carbide Reactor | | Combustion Engineering Carbide Reactor | |
|--------------------|---------------------------------------|------------|---------------------------------------|----------|--|----------|
| | RIBR* | C to TBR** | RIBR | C to TBR | RIBR | C to TBR |
| Inner Core Zone | 0.861 | . 376 | 0.926 | . 394 | 0.937 | 0.392 |
| Outer Core Zone | 0.650 | . 235 | 0.695 | . 251 | 0.708 | 0.256 |
| Axial Blankets | 4.650++ | .314 | 4.608† | 。325 | 4.608 | 0.346 |
| Radial Blankets | 3.005++ | .401 | 2.942† | .422 | 2.942 | 0.426 |
| TOTAL | | 1.326 | | 1.392 | | 1.421 |

^{*} Region internal breeding ratio.

The region internal breeding ratios for the BNW nitride and carbide reactor core zones are those given in Table XIII. For the blanket zone the region internal breeding ratios are normalized values. For the BNW carbide reactor the Combustion Engineering breeding ratios are assumed; the nitride blanket breeding ratios are obtained by comparing the relative magnitudes of breeding ratios calculated for the BNW nitride and carbide reactors.

^{**} Contribution to total breeding ratio.

t Normalized to Combustion Engineering results.

tt Normalized to BNW carbide reactor.

The data of Tables X and XI are the actual output of the PYRE burnup calculations. However, these data are not used directly; they are normalized to the batch loading of the reactor, specifically the amount of metal loaded into the reactor during a fuel charge-discharge operation. The region dependent fuel loads are broken down into the uranium and plutonium fractions and by using the data of Tables X and XI these fractions are normalized to actual kilograms of metal discharged. These calculations are summarized in Tables XV and XVI for the nitride and carbide reactors, where the total reactor inventory shown is based on plutonium-uranium feed weights. These data are given at equilibrium reactor conditions and again at discharge composition.

The calculated decrease in uranium and plutonium within the core implies an average core burnup of 12.1% for the nitride reactor case, and 12.0% for the carbide case, as compared with actual burnup of 11% corresponding to 100,000 MWd/ton of fuel metal charged to the core. The apparent discrepancy represents the formation of higher nuclides from Pu²⁴² and, to a lesser extent, the formation of U²³⁶ from U²³⁵. These reactions were included in the PYRE calculations, but the reaction products were not included in the summation of fuel components.

The unduly high rate of plutonium generation in the blankets as calculated by PYRE, which necessitated normalization of blanket calculations to the Combustion Engineering results, may be due in part to the relatively hard spectrum calculated for the blanket regions by PYRE. Figure 8 shows the spectra for the radial blanket as calculated by PYRE and by DTK. For the

TABLE XV

BNW NITRIDE REACTOR: FEED, EQUILIBRIUM AND DISCHARGE FUELS INVENTORY

| | Inner Core | Outer Core | Axial Blanket | Radial Blanket | TOTAL |
|--|-----------------------------|------------------------------|------------------------------|------------------------------|--------------------------------|
| Reactor Inventory, kg metal | 4,560 | 4,862 | 10,696 | 27,886 | 48,004 |
| Equilibrium Enrichment % Pu % Fissile (Pu + U ²³⁵) | 14.93 11.26 | 18.92 14.21 | 1.63 1.84 | 2.29 2.43 | |
| Fuel Batch Load Fraction of Region Wt. Metal | 1/8 5 7 0.000 | 1/8 607.750 | 1/8 1337。000 | 1/24 1161.917 | 3676.667 |
| Feed Composition kg U kg Pu kg Fissile Pu | 480.869 89.131 68.610 | 480.805 126.945 97.509 | 1337.000 | 1161.917 | 3460.591 216.076 166.119 |
| Equilibrium Composition kg U kg Pu kg Fissile Pu | 445.073 85.078 63.432 | 458.056 115.013 85.386 | 1310.764 21.851 21.333 | 1130.098 26.611 25.573 | 3343.991 248.553 195.724 |
| Discharge Composition kg U kg Pu kg Fissile Pu | 411.828 81.513 59.220 | 436.109 105.943 76.404 | 1286.374 39.521 37.821 | 1105.396 42.538 39.855 | 3239.707 269.515 213.300 |

TABLE XVI

BNW CARBIDE REACTOR: FEED, EQUILIBRIUM AND DISCHARGE FUELS INVENTORY

| | Inner Core | Outer Core | *Axial Blanket | *Radial Blanket | TOTAL |
|--|-----------------------------|------------------------------|------------------------------|------------------------------|--------------------------------|
| Reactor Inventory, kg metal | 4,560 | 4,862 | 10,696 | 27,886 | 48,004 |
| Equilibrium Enrichment % Pu % Fissile (Pu + U ²³⁵) | 14.20 10.77 | 18.09 13.65 | 1.71 1.91 | 2.41 2.55 | |
| Fuel Batch Load Fraction of Region Wt. Metal kg | 1/8 570,00 | 1/8 607.750 | 1/8 1337.000 | 1/24 1161.917 | 3676.667 |
| Feed Composition kg U kg Pu kg Fission Pu | 488.750 81.251 62.317 | 487.778 119.972 92.447 | 1337。000 | 1161.917 | 3475.445 201.223 154.764 |
| Equilibrium Composition kg U kg Pu kg Fissile Pu | 450.479 80.961 60.654 | 463.384 109.919 81.939 | 1309.634 22.822 22.280 | 1128.165 28.057 26.940 | 3351.662 241.759 191.813 |
| Discharge Composition kg U kg Pu kg Fissile Pu | 415.303 79.444 58.192 | 439.932 102.267 74.140 | 1284.146 41.276 39.490 | 1101.161 44.850 41.964 | 3240.542 267.837 213.786 |

^{*} Battelle-Northwest carbide results normalized to Combustion Engineering reactor.



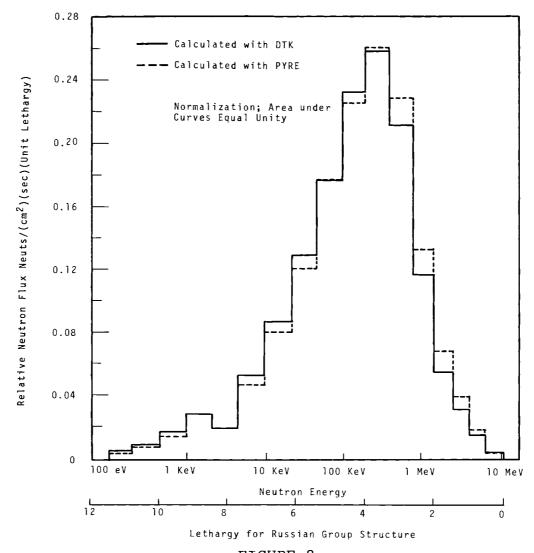


FIGURE 8 Comparison of the Nitride Reactor Radial Blanket Spectrum Calculated with DTK and PYRE

core regions the PYRE and DTK spectra are nearly identical. The harder spectrum of PYRE in the blanket regions would tend to accentuate U²³⁸ fission events and also to increase the neutron yield per fission; thus, plutonium generation rates would tend to be greater.

Enrichment Zoning and Power Distribution - A brief study was completed to determine if power distribution across the reactor core could be varied by enrichment zoning. It was found that the power distribution could be varied significantly by varying fuel densities within the zones specified for the Combustion Engineering reactor, while still maintaining the core critical. Because of the limited scope of this study, it was impossible to examine all the ramifications of fuel zoning, such as the effect on system breeding ratio, thermohydraulics and fuel burnup. Thus, to simplify the analysis, the fuel enrichment values given by Combustion Engineering were scaled upwards in proportion to overall increased enrichment requirements calculated for the carbide and nitride reactors. The power distribution was then calculated for the final equilibrium fuel densities presented herein and it was found that for both the BNW nitride and carbide reactors the power distribution did not vary significantly (See Table I) from the power distribution presented by Combustion Engineering. The radial power distribution was calculated with DTK by calculating the product of flux and fission cross section vs radius.

Sodium Removal Coefficients - When sodium is removed from the core, there are two major opposing reactivity effects: (1) a positive effect from the hardening of the neutron spectrum yielding more fission neutrons per neutron absorbed, and (2) a negative effect from the increased neutron

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leakage. A smaller positive effect results from the absence of absorption in the removed sodium. The hardening of the neutron energy spectrum is important because the fission cross section for ${\tt U}^{238}$ above 1 MeV rises with increasing energy. Also, ${\tt v}$ for ${\tt Pu}^{239}$ increases with energy. For the typical fast reactor the positive effect outweighs the negative effect and the reactor has a positive sodium void coefficient.

The sodium void coefficient of a fast reactor may be made less positive, or even negative, by variations in the core design. For example, the reactor can be designed to have a large surface area—such as a "pancake" design—to promote leakage of neutrons. Also, enough moderator can be added to degrade the reactor neutron spectrum to a range where resonance absorption becomes important; a large negative Doppler coefficient in a core of this type would tend to counter the positive sodium void coefficient. However, the mitigation of sodium void effects by such means is obtained at the expense of increased enrichment requirements and reduction in breeding ratio, with a resultant increase in fuel cycle costs.

In the Combustion Engineering reactor design, the sodium void coefficient is reported to be approximately zero for voiding the entire reactor; it is positive for voiding various regions of the reactor. It appears that no particular emphasis was placed by Combustion Engineering on modifying the reactor design to produce an inherently negative sodium void coefficient.

In the course of these studies, it was determined that nitrogen provides a small but inherent negative effect on sodium voiding. This is because of the (n, p) and (n, α) reactions that occur at high energy with nitrogen. The cross sections for these reactions are reported in AEC document BNL-325, "Neutron Cross Sections." (41,42)

The sodium void coefficients calculated in this study for nitrideand carbide-fueled reactors are given in Table XVII. Calculations for carbide-fueled reactors as performed by Combustion Engineering and the AEC
review committee are shown for comparison. The sodium removal coefficient
for core voiding as calculated in this study is less positive than that calculated by Combustion Engineering but more positive than the value obtained
by the review committee. For voiding of the complete reactor a negative
coefficient (-0.8%) was obtained in this study; Combustion Engineering reported
a net coefficient of zero. In the present study, the calculated sodium
voiding coefficients for the nitride reactor were consistently about 0.6%
more negative than those for the carbide reactor. The more negative coefficient results from high-energy neutron capture by nitrogen, and to a lesser
extent from the slightly higher enrichment of the nitride fuel.

TABLE XVIT

SODIUM VOID COEFFICIENTS: COMPARISON OF CALCULATIONS

| | | Carbide Fuel | | |
|--|--------------|-----------------|--------------|-------------|
| Sodium Void Coefficient, % Ak, for Sodium Loss from: | Nitride Fuel | <u>C.E.</u> (2) | AEC (3) | BNW |
| Total Reactor | ≖1 .⁴ | 0.0 | | -0.8 |
| Core & Axial Blanket | =·1 , 0 | | | |
| Core Only | +1.0 | +2.4 | + 1.0 | 41.5 |
| Inner Core Only | +2 .9 | | | |
| Small Volume at Core Center (% Ak/liter) | ÷0.031 | | | |

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The nitride-fueled reactor was calculated to have a sodium-void coefficient of -1.4% for total voiding of sodium from the reactor. For voiding of the core and axial blanket only, a coefficient of -1.0% was calculated, while for voiding of the inner core alone the change is +2.9%. A small-volume reactivity coefficient of +0.031% $\Delta k/liter$ of void core was calculated, which indicated the initial trend in reactivity as voiding of the core begins.

The sodium void calculations were done by comparing the DDK basic equilibrium fueled reactor to DDK cases run for the same reactor with sodium removed from various regions. The model used is that given in Figure 3.

Figure 9 shows the reactor absorption spectra for both normal and sodium-voided nitride and carbide cores. The normal and sodium void absorption spectra for the nitride core are harder than for the carbide core.

More important is the fact that the change in spectra on voiding the nitride core is greater than the corresponding change in spectra on voiding the carbide core. This effect can also be observed in Table XVIII where for Core I sodium voiding, the effect on neutron balance between U²³⁸ fission capture and parasitic nitrogen (or carbon) capture is shown. The numbers presented are the total absorption rate with respect to one reactor neutron produced per second. In terms of total flux, a Core I neutron is equivalent to approximately one-half of a reactor neutron. Thus, the capture rates given are relative to a production rate of 1/2 neutron per second within Core I. In each case the capture rate (or fission rate) is the integral of the Core I neutron flux and the corresponding energy-dependent cross section.

When sodium is voided from Core I of the reactor, the fission rate of ${\tt U}^{238}$ within the core increases by 0.0029. At the same time, the rate of





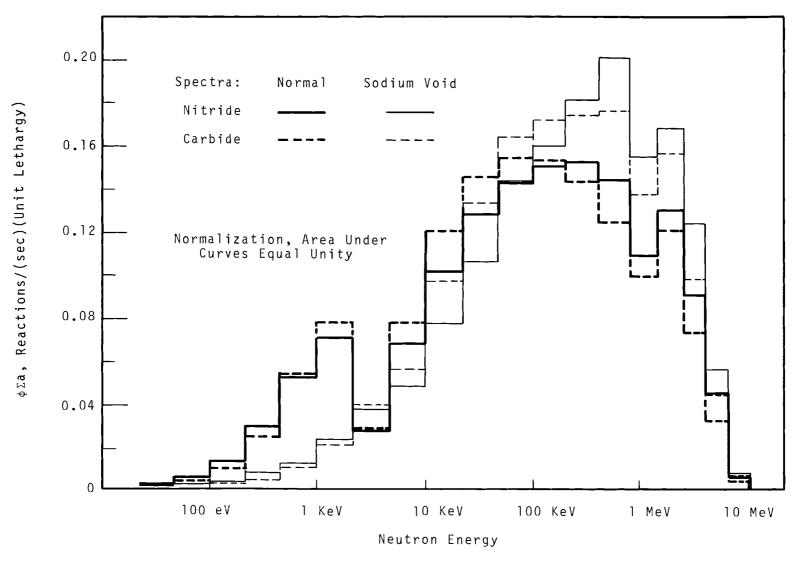


FIGURE 9
Reactor Absorption Spectra

TABLE XVIII

CORE I SODIUM VOIDING: EFFECT ON NEUTRON BALANCE BETWEEN U²³⁸ FISSION CAPTURE AND PARASITIC NITROGEN (OR CARBON) CAPTURE

Capture Rate, Relative to 1/2 Neutron Per Second Production Rate in Core I With Without Voiding Effect Sodium Sodium on Reactor Nitride Reactor Fission rate* of U^{238} 0.0277 0.0306 +0.0029 Capture rate** in 0.0175 nitrogen 0.0155 +0.0020 +0.0009 Net Effect in Reactor Carbide Reactor Fission rate of U^{238} 0.0282 0.0319 +0.0037 Capture rate in carbon 0.00000295 0.00000483 +0.00000188 +0.0037 Net Effect in Reactor Voiding Effect Comparison -0.0028 Nitride to carbide reactor - Core I only $-0.6\% \Delta k$ Nitride to carbide reactor - Total System

*
$$\int \phi_{N} \epsilon_{f}^{(U238)} dE$$

**
$$\int \phi_N \epsilon_a^N dE$$

neutron capture in nitrogen increases by 0.0020. The net effect is a relative contribution of ± 0.0009 toward increased neutron production. This positive effect, of course, is opposed by other negative effects such as increased leakage. For the carbide reactor, the relative fission rate in U^{238} increases by ± 0.0037 . The increase in rate of capture by carbon is negligible; thus, for this reactor the net contribution is ± 0.0037 —considerably higher than for the nitride reactor.

It is possible that the magnitude of the negative void coefficient due to nitrogen absorption may have been underestimated in this study. The reason is pointed out in Table XIX, where the nitrogen absorption cross section is listed for the first eight neutron energy groups as given in the Russian set and used in the DTK-DDK analysis. These data are compared to the sum of the (n, p) and (n, a) cross sections extracted from BNL-325 $^{(h_1, h_2)}$ and group-averaged by the authors. It is seen that these cross sections are 25 to 35% higher than those given in the Russian set. Thus, it is recommended that future analysis include a close examination of high-energy absorption reaction in nitrogen and a re-evaluation of the nitrogen effect on sodium void coefficient.

The calculations on sodium voiding performed in this study indicate that the nitride reactor is inherently more negative with respect to total sodium voiding than the carbide reactor. It is not negative, however, with respect to voiding of a small portion of the reactor. The analysis methods used in these calculations are considered to be the best available at the time they were performed and should be a reliable indicator of sodium void effect in these reactors.

TABLE XIX

COMPARISON OF MeV NITROGEN ABSORPTION

CROSS SECTIONS FROM THE RUSSIAN SET AND FROM BNL-325

| Group No. | Russian Data ⁽³²⁾ oa ^N , barns | BNL-325 Data (41,42) |
|--------------|---|----------------------|
| 1 | . 250 | .262 |
| 2 | 。220 | _。 299 |
| 3 | .200 | .318 |
| 4 | 。085 | .131 |
| 5 | 。030 | 。020 |
| 6 | .045 | .025 |
| 7 | ٥002 | .002 |
| 8 | .001 | .001 |

Doppler Coefficients - Doppler coefficients were obtained for both the BNW nitride and carbide reactors for two conditions: (1) raising fuel temperature from 600 to 1533 °K (Case I), and (2) raising the fuel temperature from 1533 to 1811 °K (Case II). Table XX summarizes the results of these calculations. Case I presents the Doppler coefficient that is associated with the normal temperature defect of the reactor. This case yields a Doppler coefficient that is characteristic with normal temperature rise of the entire reactor system. Case II gives a Doppler coefficient which is characteristic of the accident condition, that is, the prompt rise in fuel temperature without a corresponding rise in clad or sodium temperature. It is seen that the nitrides and carbides have similar Doppler coefficients, with the nitride reactor being

slightly lower. The lower Doppler coefficient in the nitride reactor is to be expected due to the harder neutron spectrum and higher enrichment.

TABLE XX
SUMMARY OF DOPPLER COEFFICIENT CALCULATIONS

| Case I(a) | Reactor | Initial | Final |
|--------------------------------------|------------|-----------------|-----------------|
| | Components | Temperature, °K | Temperature, °K |
| vase 1 | Fuel | 600 | 1533 |
| | Structure | 500 | 977 |
| | Sodium | 400 | 798 |
| Case II(b) | | | |
| | Fuel | 1533 | 1811 |
| | Sodium | 977 | 977 |
| | Structure | 798 | 798 |
| Doppler Coefficients | | Case I(a) | Case II (b) |
| $-T \frac{dk}{dt} \times 10^{\circ}$ |)3 | | |
| Nitride Re | | 3.5 | 1.9 |
| Carbide Re | | 3.7 | 2.1 |

- (a) Normal temperature rise in reactor system.
- (b) Accident conditions; temperature rise in fuel only.

As indicated in the paragraph on methods and analysis, Doppler coefficients in this study were only calculated with one method and no intercomparison of methods was used. Thus, the overall magnitude of the Doppler coefficient may be questioned. However, the relative differences between nitride and carbide reactors are valid.

Reactivity Coefficients, Control Requirements, and Safety Margins - No attempt was made to calculate reactivity coefficients (in addition to those Doppler coefficients indicated above), control requirements or safety margins.

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These values should remain essentially unchanged from those given for the Combustion Engineering carbide reactor system. As indicated previously, the overall control system of the reactor remains unchanged.

Gas Generation - Calculations were performed to estimate the amount of gas generated in nitride fuel due to (n, p) and (n, α) reactions in nitrogen. The indicated generation rate in Core I is 1.717 x 10^{12} molecules/cm³-sec, consisting of 1.173 x 10^{12} molecules/cm³-sec of helium and 0.544 molecules/cm³-sec $(1.088 \times 10^{12} \text{ atoms/cm}^3\text{-sec})$ of hydrogen. The fission gas simultaneously generated in this fuel was estimated at 1.075 x 10^{13} molecules/cm³-sec. Thus there is a 15.9% increase in gas generation rate due to production of hydrogen and helium in the nitrogen. These calculations were done by calculating group-averaged cross sections from the basic data of BNL-325 $^{(41,42)}$ and then summing the product of ϕ Σ_a . The sum of the (n, p) and (n, α) cross sections obtained from BNL-325 are given in Table XIX.

It is seen that the additional gas generation due to nitrogen reactions is of little consequence in the overall reactor design. Typically, the hydrogen will diffuse through the fuel clad, but the helium will be retained; consequently, the actual gas retained is less than the 15.9% indicated—perhaps nearer 11%.

Thermal Hydraulic Analysis

Since the reactor model had been extensively investigated by Combustion Engineering, no extensive thermal hydraulic analyses of the model were required. Temperature distribution calculations were performed, however, to estimate the operating characteristics of the nitride fuel elements. Calculations were performed both for a sodium-bonded, fully dense nitride element and for a 90%

dense nitride mechanically bonded to the clad. Calculations were made with the STHTP⁽⁴³⁾ computer program, a three-dimensional steady state program employing relaxation techniques. The calculated fuel and clad temperatures were modified by application of the same hot-channel factors used by Combustion Engineering for the carbide-fueled reactor model.

The thermal conductivity of polycrystalline uranium nitride as shown in Figure 1 was assumed as that of the nitride fuel; for the mechanically bonded fuel element, the conductivity was adjusted for the lower fuel density.

Conductivity data for uranium nitride do not extend above 1000 °C (1832 °F). For fuel temperatures above this, the fuel conductivity was assumed to remain constant at the 1000 °C value. Since the data indicate the conductivity would increase for higher temperatures, this assumption should tend to predict temperatures near the centerline of the fuel slightly higher than would actually occur.

Results of the temperature calculations for both the sodium-bonded and the mechanically-bonded nitride fuel elements are shown in Figure 10.

For the carbide reference model, Combustion Engineering selected a maximum allowable clad temperature of 1400 °F, based on the 10,000-hr stress-to-rupture value for 19-9 DL^(2,44) and on calculated stresses caused by fission-gas release from the carbide fuel. Limited data on fission gas release at very low exposures⁽³⁰⁾ indicate that the release rate from sintered UN is comparable to that from sintered hyperstoichiometric UC, but greater than that for cast UC. Differences in initial structural form should tend to become less important at higher exposures. Thus, although extrapolation from very low to very high exposures is extremely risky, fission gas release from

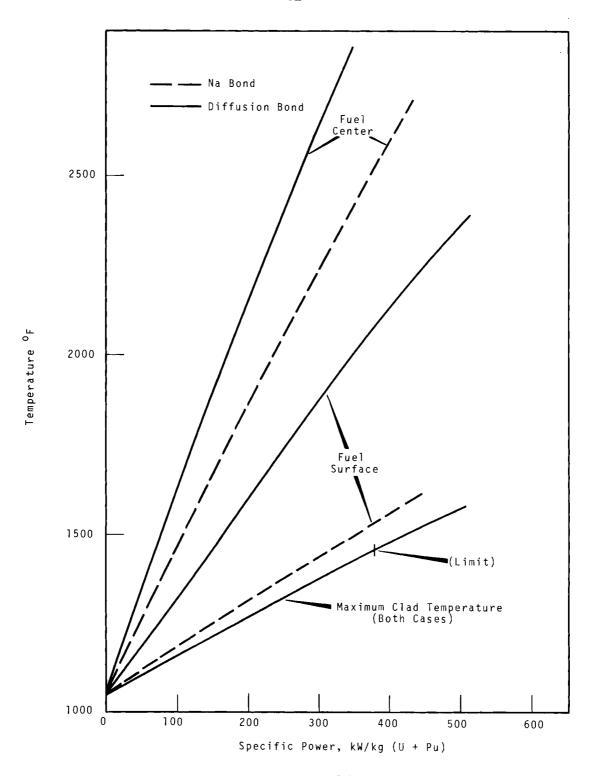


FIGURE 10 Nitride Fuel Temperature Comparisons

the UN fuel is assumed to be not greatly different from the UC fuel under reactor conditions. Helium from the (n,α) reaction on nitrogen would add about 10% to the gas generation, and if it equilibrates with the fission gases, would add an equivalent percentage to the released gases. Hydrogen from the nitrogen (n,p) reaction would largely escape by diffusion through the fuel cladding.

Therefore, under comparable conditions gas release from the nitrides would be expected to be slightly higher than from the carbides. On the other hand, embrittlement of the cladding by carburization (recognized by Combustion Engineering as a problem in the use of 19-9DL cladding with carbide fuel) would not occur with the nitrides. On the basis of compatibility, tests of type 304 stainless and uranium nitride, with exposure up to 5000 hrs at 1000 °C, (28) nitriding of the clad would appear not to present a problem (experimental verification would, of course, be required).

On the basis of comparative estimates of fission gas release and of fuel-clad interactions, therefore, the 1400 °F limit for 19-9DL chosen for the carbide-fueled model appears equally valid for the nitride-fueled element, and was chosen as a maximum allowable clad temperature for this study.

The calculated fuel and clad temperatures shown in Figure 10 indicate that the specific power capability of the nitride-fueled element will be limited by the 1400 °F maximum cladding temperature; a similar situation was estimated by Combustion Engineering for carbide-fueled elements. The figure indicates that, for a fuel rod diameter of 0.259 in. as used in this reactor design, the 1400 °F limiting temperature will be reached at a specific

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power of about 330 kW/kg (U + Pu). This result agrees well with the maximum specific power of 324 kW/kg (U + Pu)* for the Combustion Engineering model. Under these conditions the calculated maximum fuel temperatures are 2800 °F (1538 °C) for the mechanically-bonded element, and 2350 °F (1288 °C) for the sodium-bonded element. In either case the fuels are well within their temperature capabilities. For Combustion Engineering's carbide-fueled case, the corresponding fuel temperature was 2600 °F (1427 °C).

The results of the temperature calculations agree with the findings of Combustion Engineering that the primary limit on specific power in the fuel elements is the limitation on clad temperature. In both cases, the fuel should be capable of withstanding considerably higher temperatures, although the fission gas release rate would increase. For example, if fuel centerline temperature were the only limitation to power, a 2900 °F limit would permit specific powers of 350-500 kW/kg for the element design considered in the study; a 2400 °C (4350 °F) limit would under these assumptions allow specific powers of 700-900 kW/kg, if higher surface heat transfer rates could be maintained or if smaller fuel rod sizes were economically attractive.

Development of a superior high temperature clad material, or of a stress relieving mechanism such as venting of the fuel element, would permit either operation at higher specific powers or use of a larger-diameter fuel rod. Either option would potentially result in a decrease of fuel cycle costs. Effects of Variations in Fuel Cycle

Investigations were carried out in the course of this study on the effects of varying procedures in certain steps of the reactor fuel cycle, of varying

^{*} Based on Combustion Engineering data of average inner core heat flux of 222 kW/kg, and maximum/average ratio of 1.46 for the inner core. (2)

fuel management techniques, and the like. For the most part, these studies were cursory, although a more intensive study was made of fuel fabrication and conversion methods, which were found to bear importantly on fuel cycle costs. Brief descriptions of the investigations into fuel cycle variations, and the estimated effects of these variations on fuel cycle cost are given in the following paragraphs.

Fuel Conversion and Fabrication

The mononitrides of uranium and plutonium have been prepared to date only on a small-scale laboratory basis. For most cases the desired product was a pure nitride for testing or properties measurement. For these purposes, the nitrides have been prepared primarily by the direct nitridation of finely divided uranium and plutonium metal, usually with a superficial hydriding treatment prior to the nitriding step.

Considerable progress has been made toward the development of alternate processes for nitride conversion; a typical process, and one which appears to show great promise, is the carbon-nitrogen reduction of uranium or plutonium oxides. These processes appear to have considerable potential for nitride conversion on a large-scale reactor load basis, and costs for nitride conversion by these processes when developed should not be greatly different than projected costs of carbide preparation (e.g., by direct carburization of the oxides). However, these processes for nitride conversion require extensive development before they could be considered for use on a commercial, reactor-scale basis.

The costs of nitride conversion by nitridation of uranium and plutonium metals are not well tabulated since nitride preparation to date has

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been performed primarily by laboratory methods. The costs of nitride conversion as used in this study were arrived at by assuming a scaling-up of these methods to full plant-scale procedures and by drawing analogies to similar processes where applicable. The carbothermic-reduction process for nitride conversion, for example, is quite similar to a process for conversion of UO₂ to UC proposed by Combustion Engineering in the reference carbide study, and a more or less direct comparison of the processes could be made to estimate conversion process costs. (4) Such a comparison indicates that the carbothermic reduction process for nitride conversion, when fully developed, should be capable of providing nitride fuel materials at costs competitive with the carbides.

With nitride conversion processes which require the use of metallic uranium or plutonium as intermediates, process costs will be influenced strongly by, and in some cases dictated by, costs of reduction of the source materials to the metallic state.

Conversion costs for uranium metal preparation generally tend to be higher than the corresponding oxide costs, although the metal reduction costs will vary sharply both with plant throughput and with enrichment. For the conversion plant size considered in this study, of sufficient capacity to serve three similar fast-breeder reactors, metal conversion costs might be expected to run from \$10 to \$30/kg uranium, based on interpolation of published cost comparisons. (45) For offsite purchase from a large-scale producer, costs of \$10/kg or less might be realized; for such large-scale production the cost of U metal could approach that of UO₂. For purposes of this study, the cost of uranium metal for use in nitride conversion was assumed to be 10% greater than the cost of UO₂.

For the reduction of plutonium to the metal, however, criticality considerations require small batch sizes, and safeguards required by the toxicity of plutonium act to increase the unit costs. The current cost quoted in the Cost Evaluation Handbook (46) for plutonium reduction is \$1.50/g plutonium. For the bomb-reduction process currently used, conversion costs are not likely to differ much from this. Alternate processes still under development, such as a continuous electrolysis process being developed at Hanford, (47) may effect an appreciable reduction in conversion costs when developed. For purposes of this study, however, plutonium conversion costs of \$1.50/g plutonium were used.

The vibratory-compacted, mechanically bonded, isostatically hotpressed fuel element assumed for the nitride-fueled reactor appears to be
slightly less expensive to fabricate than the skull-cast, sodium-bonded,
carbide fuel element, based on a cursory cost comparison of the two processes. (4) The apparent cost difference, however, is negligible compared to
cost uncertainties in either the carbide or nitride fabrication processes.

A comparison of conversion-plus-fabrication costs for the nitride fuels, for the alternate carbothermic reduction and metallic intermediate conversion processes assumed for the study, is shown in Table XXI. The increased conversion-fabrication cost for the route involving metallic intermediates for nitride-conversion--some 60% higher than the costs using the carbothermic reduction process--is due virtually entirely to the assumed \$1.50/g charge for plutonium metal preparation. This is the most significant difference in fuel cycle costs developed in this study, and illustrates the

TABLE XXI

FABRICATION COST COMPARISONS: CARBIDE AND NITRIDE FUELS

| | Fabrication Costs Including Conversion Costs For: Nitride Fuel, Using Nitride Fuel, Using | | | | | |
|-------------------------|--|-------------|-------------------------------------|-------------|------------------|-------------|
| Fuel Component | Carbothermic-Reduction cent Conversion | | Metallic-Intermediate Conversion | | Carbide Fuel (a) | |
| | \$/core kg(b) | mills/kW-hr | \$/core kg(b) | mills/kW-hr | \$/core kg(b) | mills/kW-hr |
| Radial Blanket | 81 | 0.0605 | 84.15 | 0.0628 | 81 | 0.0605 |
| Control Rod Followers | 16 | 0.0119 | 16.34 | 0.0122 | 16 | 0.0119 |
| Core and Axial Blanket | 257 | 0.1920 | 455.83 | 0.3405 | 273 | 0.2040 |
| Total Fabrication Costs | 354 | 0.2644 | 556.32 | 0.4155 | 370 | 0.2764 |

- (a) Carbide fuel fabrication costs based on unit costs quoted in Reference 2.
- (b) "Dollar per core kilogram" is the ratio of the fabrication cost of a fuel component to the weight in kilograms of the (U + Pu) metal contained in the core portion of a fuel batch.

dependence of an economic nitride fuel cycle on the development of a nitride conversion process, such as the carbothermic reduction process, whose costs would compare with carbide conversion costs.

Fuel Reprocessing

Reprocessing costs for spent reactor fuel constitute an important fraction of the total fuel cycle cost. Furthermore, a wide choice of alternate processes for treatment of the discharged fuel is potentially available, although few of these processes have actually been developed.

Conventional solvent-extraction processes, carried out in large central plants, are the only ones for which definitive cost information is available, since the projected costs for processing power reactor fuels by these processes is based on operating experience in existing government-owned plants. Reprocessing costs for the reference study conducted by Combustion Engineering were based on the standard AEC conceptual reprocessing plant. For the present study the proposed NFS reprocessing cost schedule was used, since this is believed to give a somewhat more realistic projection of future reprocessing costs. For consistency in cost comparisons, the fuel cycle costs developed in the Combustion Engineering study were modified by recalculating reprocessing costs to reflect the proposed NFS schedule. Since neither cost schedule is arranged to accommodate high-plutonium fuel, the approximation used by Combustion Engineering was adopted; fuel of a given plutonium concentration was assumed equivalent, in limiting the size of processing units in the plant, to a U²³⁵-enriched fuel of 1.5 times the fissile content of the plutonium-containing fuel (e.g., 10% fissile plutonium content is equivalent, for setting processing unit size, to 15% U²³⁵ content). When the carbide reactor model of Combustion Engineering is recalculated using the same computer programs and cross section sets used for the nitride-fueled case, reprocessing costs for the nitride and carbide models are essentially identical. The nitride fuel may have a slight cost advantage, since head end treatments may be simpler than would be required for the sodium-bonded carbide fuel; however, no credit for simplified treatment was taken in assessing nitride reprocessing costs.

In the reference carbide study, Combustion Engineering surveyed several alternate processing methods to estimate their potential for application to a fast breeder reactor fuel cycle, as central plants or as smaller scale, onsite reprocessing plants. Use of an onsite reprocessing facility would eliminate shipping charges for irradiated fuel, and would presumably permit economical operation of a process having somewhat higher unit costs. Processes surveyed in the carbide study included a small scale, onsite solvent extraction process, a carbide-oxide pyrochemical process, a liquid-metal-solvent process, and an anion exchange process. The conclusions reached in that study were that of those processes, only the anion extraction process showed promise for application to a carbide fast breeder reactor fuel cycle. Based on cost estimates developed by the General Electric Company, (49) an ion exchange process might well be competitive, either for onsite or central plant application, as a fuel reprocessing method for such a cycle.

In the present study, the alternate processes considered by Combustion Engineering were resurveyed briefly; in addition, a proposed carbide-nitride pyrochemical process $^{(50)}$ and the Hanford-developed Salt Cycle process $^{(51,52)}$ were briefly reviewed. The goal in these reviews was to

determine whether any process considered might be uniquely favorable to nitride fuels, or conversely whether nitrides might be incompatible with certain processes or might require extra treatment and therefore engender higher processing costs. Since the emphasis on this study was on cost differentials between fuels, no detailed cost comparisons between processes were attempted. However, the results generally agree with those of Combustion Engineering that an ion-exchange process is the most promising process of those reviewed as an alternative to conventional solvent extraction central plant processing.

For none of the processes reviewed was a significant cost difference apparent for processing carbides or nitrides. For the most part the various processes were equally adaptable to carbides or nitrides, or required equivalent head end treatment (e.g., conversion to oxides) for both. For the aqueous processing methods, and particularly for standard solvent extraction processes, the use of nitride fuels may be slightly advantageous economically as compared to carbides. For the particular nitride fuel design selected for this study, the absence of a sodium bonding layer would eliminate the necessity for head end sodium removal which would be required for the sodium-bonded carbide fuel. Furthermore, nitrides on dissolution would tend to form ammonia rather than hydrocarbons as byproducts. If ammonia or ammonium salts can be tolerated in the process, or removed during dissolution, the nitride fuels may be amenable to straightforward dissolution, eliminating the conversion to oxide which is foreseen as being required for carbide fuels.

However, any cost savings due to sodium elimination would be ascribed to differences in element design rather than fuel material (a non-sodium bonded carbide element could show comparable savings); and the savings

in reduced pre-dissolution treatment would require experimental verification.

Therefore, although such savings in reprocessing costs may well be real,

they were not factored into the reprocessing costs used for this study.

Fuel processing costs were calculated on the same basis as was used by Combustion Engineering, except that the proposed price schedule for the NFS processing plant was used, with a \$23,500 charge per processing unit. The plant was assumed capable of handling the high-plutonium fuels considered in the fast breeder reactor studies. The limits on processing rates as published for the NFS plant are based on U²³⁵ enrichment of the fuel. For the plutonium-enriched fuel of the study, the limit on the size of a processing unit for material of given fissile plutonium content was assumed to be the limit for fuel containing a 50% higher concentration of U²³⁵. Processing charges were based upon mixing of the core and axial blanket elements at the head end of the plant and processing the combined fuel as a single entity of 5.75% fissile plutonium content. Separate processing of core and axial blanket elements would entail higher processing charges because of the high plutonium content of the core. However, the higher-grade plutonium available from the radial blanket may at some future date command a premium price as a reactor fuel (perhaps for use in compact, special purpose reactors), making separate processing of core and blanket economical.

Incremental processing costs for waste storage and perpetual maintenance were not included. These costs were not considered in the reference study on carbide fuels; they would be only minor additions and would apply equally to the carbide and nitride cycles.

The processing costs calculated for the nitride fuel amounted to \$97.65/kg of metal processed, or 0.2277 mills/kW-hr; essentially identical costs were indicated for the carbide fuel.

Uranium Fuel Management

In the fuel cycle used as the basis for the Combustion Engineering study, (2) all plutonium enrichment required for the reactor was assumed to be supplied by recycling of produced plutonium; excess plutonium over that required was sold. Each fuel batch was assumed to utilize all-new, depleted uranium with a U²³⁵ content of 0.3%. The same cycle was followed in this study, with recovered uranium from the reprocessing plant assumed to be disposed of, at no cost, to the supplier furnishing new uranium.

Under this scheme, the uranium contained in each new fuel batch contains some 10.2 kg U²³⁵ in the nitride fueled reactor, or 10.3 kg for the carbide case. Of these amounts some 4.05 kg or 4.2 kg, respectively, are destroyed during in-reactor residence of the fuel. Since the major portion of the U²³⁵ depletion is due to fission, this fuel component has a significant effect on net plutonium generation, and thus both on doubling time and on plutonium credit as applied to fuel cycle costs.

If the management scheme were changed to provide for total recycle of the uranium as well as plutonium, only about 275-290 kg of makeup uranium would be required for each batch. At 0.3%, the U^{235} content added would be 0.83-0.88 kg; under equilibrium conditions an equivalent amount would be destroyed during irradiation. The U^{235} content of a feed-fuel batch would then be some 2.1 kg, rather than 10.2 kg.

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The decrease in U²³⁵ content, and consequently in U²³⁵ fission, would require the fission of approximately 2 kg of plutonium per batch, at a cost per batch of some \$20,000. Savings in uranium purchases, on the other hand, would reduce fabrication costs by about \$13,000 (net savings of about 3200 kg uranium @ \$4/kg). Thus, net additions to the fuel cycle cost on the order of \$7000 per batch, or some 0.004 mills/kW-hr, could be expected from such a scheme as compared to the fuel management scheme assumed for the study.

In schemes such as the foregoing, involving total recycle of uranium as well as plutonium, additional complications may arise owning to the buildup of $\rm U^{236}$ and other uranium isotopes. However, no investigation of such effects was made in this study.

Fuel Cycle Cost Estimate

Fuel cycle costs for the nitride- and carbide-fueled reactor models were calculated based on private ownership of fuel materials, and recalculated on a leased-fuel basis, in basic accordance with the AEC ground rules prescribed for the four studies of 1000 MW_e fast breeder reactors, (3) as previously discussed in the section describing the fuel cycle study model.

A materials balance summation for one fuel batch over the entire fuel cycle is given for nitride fuel in Table XXII. Table XXIII gives the equivalent data for carbide fuel. Fuel cycle cost calculations were based on these material balances and on the reactor loading data given in Tables XV and XVI. Except where otherwise noted, the methods of calculation followed those used by Combustion Engineering in their study of a 1000 MW_e fast breeder reactor.

TABLE XXII

MATERIALS BALANCE - SINGLE FUEL BATCH

Nitride Fuel

| | | U | <u>Pu</u> | <u>Fissile Pu</u> | U + Pu |
|------------|---------------------------------------|-----------|-----------|-------------------|-----------|
| | Reactor Charge, Wt | 3460.591 | 216.076 | 166.119 | 3676.667 |
| | Reactor Discharge, Wt | 3239.707 | 269.515 | 213.300 | 3509.222 |
| | Reprocessing Feed, Wt | 3239.707 | 269.515 | 213.300 | 3509.222 |
| | -(Reprocessing Loss - 1% of Feed) | (32.397) | (2.695) | (2.133) | (35.092) |
| | Reprocessing Product* | 3207。310 | 266.820 | 211.167 | 3474.130 |
| | -(Excess Pu) | | (48.343) | (43.202) | (48.343) |
| | +(New U) | 3499.042 | | | 3499.042 |
| - | Scrap | 384.510 | 24.008 | 18.458 | 408.518 |
| | Conversion Feed | 3883.552 | 242,485 | 186,423 | 4126.037 |
| recycle | -(Conversion Loss - 1% of Product) | (38.451) | (2.401) | (1.846) | (40.852) |
| rec | Conversion Product | 3845.101 | 240.084 | 184.577 | 4085.185 |
| | Fabrication Feed | 3845.101 | 240.084 | 184.577 | 4085.185 |
| L | | (384.510) | (24.008) | (18.458) | (408.518) |
| | Fabrication Product | 3460.591 | 216.076 | 166.119 | 3676.667 |
| | Reactor Charge, Wt | 3460.591 | 216.076 | 166.119 | 3676.667 |

^{*} Uranium returned to supplier.

TABLE XXIII

MATERIALS BALANCE - SINGLE FUEL BATCH

BNW Carbide

| | | U | <u>Pu</u> | <u>Fissile Pu</u> | U + Pu |
|---------|---------------------------------------|----------|-----------|-------------------|----------|
| | Reactor Charge, Wt | 3475.445 | 201.222 | 154.764 | 3676.667 |
| | Reactor Discharge, Wt | 3240.542 | 267.837 | 213.786 | 3508.379 |
| | Reprocessing Feed, Wt | 3240.542 | 267.837 | 213.786 | 3508.379 |
| | -(Reprocessing Loss - 1% of Feed) | (32.405) | (2.678) | (2.138) | (35.083) |
| | Reprocessing Product* | 3208.137 | 265.159 | 211.648 | 3473.296 |
| | -(Excess Pu) | | (61.701) | (55.164) | (61.701) |
| | +(New U) | 3514.061 | | | 3514.061 |
| ľ | Scrap | 386.161 | 22.358 | 17.196 | 408.519 |
| | Conversion Feed | 3900,222 | 225.816 | 173.680 | 4126.038 |
| recycle | -(Conversion Loss - 1% of Product) | (38.616) | (2.236) | (1.720) | (40.852) |
| rec | Conversion Product | 3861.606 | 223.580 | 171.960 | 4085.186 |
| | Fabrication Feed | 3861.606 | 223.580 | 171.960 | 4085.186 |
| Į | (Scrap - 10% of Feed) | 386.161 | 22.358 | 17.196 | 408.519 |
| | Fabrication Product | 3475.445 | 201.222 | 154.764 | 3676.667 |
| | Reactor Charge, Wt | 3475.445 | 201.222 | 154.764 | 3676.667 |

^{*} Uranium returned to supplier.

The fuel cycle schedule assumed by Combustion Engineering for their study was adopted for the present study, with the time required for reprocessing increased from 11 to 16 days as required by the NFS formula. In order to maintain the same total cycle time and thus permit closer comparison of the results of this study with those of Combustion Engineering, the time allotted for pre-irradiation storage and preparation of a fuel batch was reduced from 30 to 25 days. The fuel cycle schedule as used in this study is shown in Table XXIV. Under this schedule, which assumes a plant operating factor of 80%, a batch of fuel is discharged from the reactor, and a new one charged each 82 days.

TABLE XXIV

FUEL CYCLE SCHEDULE

| In-Reactor Exposure(a) | Days 657 |
|-----------------------------------|-------------|
| Post-Irridation Cooling | 120 |
| Shipping to Reprocessing Plant | 20 |
| Reprocessing | 16 |
| Shipping to Fabrication Plant (b) | 10 |
| Fabrication | 219 |
| Storage and Preparation | 25 |
| Total Cycle Time | 1067 |

⁽a) Core-and-axial blanket elements. Radial blankets remain in-reactor three times as long.

⁽b) Includes time for shipment from fabricator to reactor site.

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Inventory Charges

For calculations of inventory charges involving privately-owned fuel, the entire fuel metal inventory--uranium and plutonium--involved in the fuel cycle was computed, and its value was calculated based on \$10/g fissile isotopes for plutonium, and \$4/kg for uranium. This value was assumed to represent nondepreciable, working capital; the inventory charge was taken as a 10% annual interest charge on this capital.

Based on the fuel cycle schedule in Table XXIV, each fuel batch spends 410 days of its cycle outside the reactor. With a batch discharged each 82 days, at any one time there are thus five batches undergoing various stages of the cycle outside the reactor. A spare-fuel reserve equaling 0.1 batch was also assumed. The five batches were apportioned among the various steps of the fuel cycle in proportion to the time spent in each step; the fuel inventory at each step—and in the spare-fuel reserve—was calculated from the materials balances in Tables XXII and XXIII. To this was added the fuel content of one entire reactor load under equilibrium conditions. The inventory charge was calculated as annual interest on the aggregate value of this inventory.

For nitride fuel, this inventory charge was calculated at \$2,991,700 per year, or 0.4269 mills/kW-hr. For carbide fuel, this charge was \$2,943,200 per year, or 0.4200 mills/kW-h4.

Inventory charges for the leased-fuel case were calculated in a manner identical to that used by Combustion Engineering. The fuel cycle time of 1067 days was divided into 657 days irradiation time; 156 days post-irradiation time (cooling, shipping, and reprocessing): and 254 days pre-irradiation

time (shipping, fabrication, and storage). For these periods the fuel batch was assumed to be, respectively, at equilibrium, discharge, and feed compositions as given in Tables XV and XVI. Using the plutonium value of \$10/g of Pu²³⁹ and Pu²⁴¹ isotopes, use charges of 4-3/4% per year were apportioned among these three periods of the cycle based on the batch plutonium content during each period. The sum of the use charges for these three periods consituted the inventory charge portion of the fuel cycle cost. For the nitridefueled reactor these charges totaled \$266,115 per batch, or 0.1688 mills/kW-hr. For the carbide fuel case the charges amounted to \$259,118 per batch, or 0.1643 mills/kW-hr, reflecting the slightly lower enrichment required for the carbide fuel.

Reprocessing Costs

Reprocessing costs were calculated based on the published price formula for the NFS reprocessing plant, (48) with a base charge of \$23,500 per day. Incremental charges for plutonium content, head end treatment, waste storage, and the like were not included. In calculating the equivalent U^{235} content of a fuel batch, one gram of fissile plutonium was assumed to be equivalent to 1.5 grams U^{235} ; the actual residual U^{235} content of the fuel was added to this calculated equivalent content, and the equivalent U^{235} enrichment was computed for use in the pricing formula. For nitride fuel, reprocessing charges amounted to \$97.65/kg processed, or 0.2277 mills/kW-hr. For carbide, the charges were \$94.78/kg or 0.2257 mills/kW-hr.

Plutonium Loss

Process losses of plutonium were assumed to equal 1% of the reprocessing feed plus 1% of the fuel conversion product prior to fabrication, as

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detailed in the materials balance sheets. The same percentage losses were assumed for nitride and carbide fuels, and for alternate methods of conversion to nitride. These are the same percentage losses assumed by Combustion Engineering, although they noted in their study that actual process losses would be considerably less than this.

The calculated costs for plutonium loss in the nitride fuel cycle are \$39,790 per batch, or 0.0252 mills/kW-hr; for carbide fuel, these costs are \$38,580 per batch, or 0.0245 mills/kW-hr.

Shipping Costs

Shipping costs were estimated on the basis used by Combustion Engineering: \$16/kg for irradiated fuel; \$1/kg for processed and decontaminated fuel, or new fuel materials; and \$3/kg for fabricated fuel elements, with all weights expressed as total (U + Pu) fuel metal content. Costs were identical for nitride and carbide fuels, totaling \$73,570 per fuel batch or 0.0467 mills/kW-hr.

Fabrication

The basis for fabrication cost estimates was detailed previously in this report, in the section describing the fuel cycle study model. For nitride fuel prepared via the carbothermic-reduction nitridation process, fabrication and conversion costs were estimated at \$416,920 per batch of fuel, or 0.2644 mills/kW-hr. If the nitridation step involves the preparation of metallic intermediates, costs of \$655,210 per batch, or 0.4155 mills/kW-hr, are indicated. Carbide fuel fabrication costs were estimated at \$435,770 per batch, or 0.2764 mills/kW-hr.

Capitalization

Capitalization charges for the first core were calculated in the same manner as that reported for the Combustion Engineering study. The first core of the reactor was assumed to have the same plutonium content as the equilibrium core. Fabrication charges for this core were estimated at \$4.86 million, and plutonium losses associated with fabrication were estimated at \$219,000. Total capitalization for the first core then was calculated as the sum of: (1) interest charges of 10% per year on the value of the fuel material (privately owned fuel), or use charges for the contained plutonium at 4-3/4% per year (leased fuel), during the pre-startup period; (2) fabrication charges for the core; (3) charges for plutonium losses; and (4) shipping charges for the fuel, estimated to total \$192,000. Annual capitalization charges were based on capital recovery over a 30-year period at 6% interest (capital recovery factor of 0.07265), plus a 6% allowance for taxes and insurance.

Annual capitalization charges calculated for privately-owned fuel were: for nitride fuel using carbothermic-reduction conversion, \$883,000 per year, or 0.1261 mills/kW-hr; for nitrides using metallic intermediates for conversion, \$1.14 million per year, or 0.1633 mills/kW-hr; and for carbide fuels, \$902,600 per year or 0.1288 mills/kW-hr.

For leased fuels, the computed annual charges amount to \$786,000, or 0.1122 mills/kW-hr for nitride fuel using the carbothermic reduction nitridation process; for processes using metallic intermediates, this cost was \$1.018 million annually, or 0.1452 mills/kW-hr. Carbide fuel capitalization charges were estimated at \$805,700 per year, or 0.1150 mills/kW-hr.

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Plutonium Credit

Credit for plutonium produced in the fuel cycle was based on the "ground rule" plutonium value of \$10/g fissile isotopes (Pu²³⁹ + Pu²⁴¹), as nitrates at the reprocessing plant. Since plutonium losses during reprocessing were previously taken into account in following the calculation methods used by Combustion Engineering, plutonium credit was based on net production in the reactor rather than on actual excess plutonium delivered. The same overall results would be obtained by basing credits on the actual quantities of excess plutonium delivered, and deleting the charge for plutonium loss during reprocessing.

For nitride fuels the calculated net generation of fissile Pu isotopes was 47.18 kg per batch (although actual delivery was 43.20 kg); the value credited was thus \$471,800 per batch or 0.2992 mills/kW-hr. For carbide fuel the estimated net generation was 59.02 kg (delivery: 55.16 kg); credit was \$590,200 per batch, or 0.3743 mills/kW-hr.

Net Fuel Cycle Costs

A summary of estimated fuel cycle costs based on private ownership of fuel is given in Table XXV; the corresponding summary based on leasing of fuel from the AEC is shown in Table XXVI.

For both the privately-owned and leased-fuel cases, cost summaries are shown for nitride fuels using both carbothermic-reduction and metallic intermediate processes for nitride conversion, and for carbide fuel, in a reactor model as calculated by DDK-DTK-PYRE techniques as a part of this study. For the leased-fuel case, summaries are also given for carbide fuel as reported

TABLE XXV

FUEL CYCLE COST SUMMARY

PRIVATE OWNERSHIP OF FUEL

Interest charge 10% per year on non-depreciating working capital representing total uranium and plutonium value in fuel system.

| | (| Cost (mills/kW-hr) | |
|----------------|-----------------------|-----------------------|------------------|
| | Nitride (nonmetal) | Nitride (metal)(b) | BNW (carbide) |
| Inventory | 0.4269 | 0.4269 | 0.4200 |
| Reprocessing | 0.2277 | 0.2277 | 0,2278 |
| Pu Loss | 0.0252 | 0.0252 | 0.0245 |
| Shipping | 0.0467 | 0.0467 | 0.0467 |
| Fabrication | 0.2644 | 0.4155 | 0.2764 |
| Capitalization | 0.1261 | 0.1633 | 0.1288 |
| Gross Cost | 1.1170 | 1.3053 | 1.1242 |
| Pu Credit | 0.2992 | 0.2992 | 0.3743 |
| Net Cost | 0.8178 | 1.0061 | 0.7499 |

- (a) Nitride fuel utilizing carbothermic reduction process or equivalent process for conversion to nitrides.
- (b) Nitride fuel utilizing metallic intermediates in conversion to nitrides.
- (c) Carbide fueled reactor as calculated in this study by same procedures and techniques used for nitride fuel calculations.

by Combustion Engineering, as recalculated to reflect use of the NFS fuel reprocessing plant. The Combustion Engineering results were not recalculated for private ownership.

TABLE XXVI

FUEL CYCLE COST SUMMARY

LEASED FUEL

Use Charge: 4-3/4% per year on value of plutonium.

| | Cost (mills/kW-hr) | | | |
|----------------|--------------------------------------|--|--------------------------------------|------------------------------|
| | Nitride ^(a) (nonmetal) | Nitride ^(b) <u>(metal)</u> | _{BNW} (c) <u>Carbide</u> | CE ^(d) Carbide |
| Inventory | 0.1688 | 0.1688 | 0.1643 | 0.1562 |
| Reprocessing | 0.2277 | 0.2277 | 0.2278 | 0.2257 |
| Pu Loss | 0.0252 | 0.0252 | 0.0245 | 0.0235 |
| Shipping | 0.0467 | 0.0467 | 0.0467 | 0.0477 |
| Fabrication | 0.2644 | 0.4155 | 0.2764 | 0.2614 |
| Capitalization | 0.1122 | 0.1452 | 0.1150 | 0.1086 |
| Gross Cost | 0.8478 | 1.0291 | 0.8547 | 0.8231 |
| Pu Credit | 0.2992 | 0.2992 | 0.3743 | 0.4125 |
| Net Cost | 0.5486 | 0.7299 | 0.4804 | 0.4106 |

- (a) Nitride fuel utilizing carbothermic reduction process or equivalent process for conversion to nitrides.
- (b) Nitride fuel utilizing metallic intermediates in conversion to nitrides.
- (c) Carbide fueled reactor as calculated in this study by same procedures and techniques used for nitride fuel calculations.
- (d) Carbide fueled reactor as reported in Reference 2; costs recalculated to reflect fuel processing at NFS facility.

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Costs for the carbide fuel cycle (leased fuel) as calculated by BNW techniques are slightly higher than those reported by Combustion Engineering--0.480 vs 0.411 mills/kW-hr--reflecting the higher enrichment requirements predicted by the BNW calculations. As was previously discussed, physics calculations for the carbide-fueled reactor were performed in this study principally to provide a valid comparison between nitride and carbide fuel costs, by eliminating the rather large uncertainties introduced through varying calculational techniques.

Private ownership of fuel, as shown in Table XXV, increases fuel cycle costs for all cases studied by about 0.27 mills/kW-hr as compared to leased-fuel costs. The cost differences among the various fuels are essentially the same for either privately-owned or leased fuel. Since the major effect of private ownership is the substitution of a (probably substantially higher) charge for interest on working capital in place of the fuel-use charge, the inventory and first-core capitalization charges are the only portions of the fuel cycle cost affected by private ownership versus leasing of the fuel.

For nitride fuel produced through the carbothermic reduction process, fabrication costs are estimated to be slightly lower than for carbide fuel. Total net fuel cycle costs, however, are slightly higher than the carbide fuel cycle costs, by some 0.069 mills/kW-hr. This increase in cost is due primarily to the somewhat higher enrichment required for the nitride-fueled reactor.

The higher conversion cost of nitride fuel produced through metallic intermediates results in a significant increase in the fuel cycle cost--

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0.181-0.188 mills/kW-hr higher than for carbothermic-reduction nitride, or 0.250-0.256 mills/kW-hr higher than for carbide fuel. The only difference in the two nitride cases is in the conversion costs, which affect costs of fabrication and of first-core capitalization. This large difference in fuel cycle cost illustrates the importance of development of economic conversion processes-such as the carbothermic reduction method-to establishment of nitrides as economically competitive fuels.

The indicated difference of 0.068-0.069 mills/kW-hr between fuel costs for "carbothermic" nitride and for carbide is small enough to indicate that, assuming development of the carbothermic reduction process or an equivalent process, nitride fuels could be considered competitive with carbides. The indicated cost differential could presumably be reduced in a reactor design optimized for the use of nitride fuels. Furthermore, it was indicated previously in this report that the nitrogen component of the nitride fuel apparently provides a significant negative contribution toward the sodium voiding coefficient. If this negative contribution could be realized in an actual reactor design, it could result in important benefits from design simplification and improved reactor operation. These benefits could considerably outweigh the relatively small difference in fuel cycle costs indicated in this study between nitrides and carbides.

Areas Requiring Development

During the course of this study it became evident that the nitrides of uranium and plutonium are attractive as potential fuels for commercial fast breeder power reactors, which are expected to come into use in the late 1970's

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or early 1980's; but that further development work on these materials would be required to establish their role as reactor fuels. Among the more crucial areas requiring development are the following.

Nitride Conversion Processes

The importance to a nitride fuel cycle of the development of an economical process for conversion of fuel materials to the nitrides was discussed at length previously in this report, and the need for such development is reiterated here primarily for emphasis. The development of a suitable and economical conversion process, such as the carbothermic reduction process assumed for this study, appears to be crucial to the establishment of nitrides as fuels for power reactors.

Materials Data

Uranium and plutonium nitrides have attracted interest only in recent years as potential fuels for nuclear reactors. Consequently, although considerable information on the properties of these materials has been developed within the past few years, much additional information on these materials is required. Included in the needed information is that in the following categories:

Physical and Chemical Properties - Considerable additional information is needed on the properties of uranium nitride and also on the uranium-plutonium nitride mixtures envisioned for use as fast breeder reactor fuels. Particularly needed are data pertinent to behavior of these materials at temperatures which might be postulated during a power excursion of a reactor.

<u>Compatibility</u> - The small amount of data available on compatibility with potential cladding materials and with liquid metal coolants indicate

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that nitride fuels would be compatible in reactor service with these materials.

Additional data are required, however, to ascertain compatibility for long periods of time, under varying temperature conditions, and under radiation exposure.

Irradiation Behavior - Existing information on the behavior of the nitrides under irradiation has been obtained largely with small-scale capsule irradiations. Considerable additional work is needed to expand the presently available information, and to extend it to higher temperatures and higher exposures with sufficient confidence that integrity of the fuel under operating conditions may be assured.

Gas Generation and Retention - Currently available data, although quite limited, indicate that the fission gas retention characteristics of the nitrides should not differ greatly from those of the carbides. Additional information is needed to corroborate this behavior, to extend knowledge of retention characteristics to higher temperatures and to higher fuel burnup, and to verify calculated rates of hydrogen and helium generation due to (n, p) and (n, α) reactions with nitrogen.

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D. R. deHalas
D. E. Deonigi

R. L. Dillon

K. Drumheller

E. A. Eschbach

 $\texttt{E}_{\, \circ} \ \, \texttt{A}_{\, \circ} \ \, \texttt{Evans}$

P. L. Farnsworth

S. R. Fields

J. F. Fletcher (5)

J. C. Fox

M. D. Freshley

B. C. Fryer

S. Goldsmith

W. O. Greenhalgh

W. L. Hampson

K. M. Harman

H. Harty

B. R. Hayward

J. W. Helm

R. J. Hennig

R. M. Hiatt

C. W. Higby

P. L. Hofmann

M. T. Jakub

B. M. Johnson

J. N. Judy

D. D. Lanning

C. E. Leach

R. C. Liikala

C. W. Lindenmeier

W. W. Little

L. L. Maas

D. E. Mahagin

W. B. McDonald

R. W. McKee

M. A. McLaughlin

C. E. McNeilly

E. T. Merrill

W。I。Neef

R. E. Nightingale

L. T. Pedersen

R. E. Peterson

A. M. Platt

W. A. Reardon

J. J. Regimbal

W. E. Roake

R. K. Robinson

C. A. Rohrmann

L. C. Schmid

G. L. Simmons

D. E. Simpson

G. L. Tingey

G. H. TINEC)

J. C. Tobin

R. E. Turley

E. E. Voiland

E. T. Weber

J. H. Westsik

R. G. Wheeler

K. R. Wise

F. W. Woodfield

J. R. Worden

H. H. Yoshikawa

F. R. Zaloudek

Technical Information (3)

Tech Pubs. (2)