BWR PERFORMANCE IMPROVEMENT FEASIBILITY USING HYDRIDE FUEL

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ABSTRACT

The feasibility of improving the neutronic characteristics of Boiling Water Reactors (BWR) by using U-Zr hydride fuel is studied. Several modified BWR fuel assembly designs are considered. These include designs in which hydride fuel rods replace only water rods, replace water rods and a fraction of the oxide fuel rods, replace oxide fuel in the upper half of all the fuel rods, and replace all the oxide fuel in the assembly. It is found that replacement of at least half of the oxide fuel rods in the fuel assembly by U-$ZrH_{1.6}$ fuel might simultaneously improve the performance of BWR in three ways: (a) Increasing the energy extracted per fuel assembly and the cycle length by up to ~10%. (b) Reducing the uranium ore and SWU requirements by approximately 10%. (c) Reducing the negative void coefficient of reactivity by, at least, 50%. It is also found that replacement of all the oxide fuel by hydride fuel opens interesting new options for the design of BWR fuel assemblies. The net result might be simplified assembly designs that can generate significantly more energy while featuring small negative void coefficient of reactivity. U-ThH_2 fuel appears to be even more promising than U-$ZrH_{1.6}$. For the potential benefits form hydride fuel to be realized, a clad material that is not permeable to hydrogen and is not as neutron absorbing as stainless steel need be developed.

1. INTRODUCTION

Due to the boiling of their coolant, the cores of Boiling Water Reactors (BWRs) are undermoderated, primarily at their upper part. The oxide fueled BWR is also characterized by a large degree of heterogeneity across the fuel assembly; contemporary designs call for more than 10 different fuel rod compositions in one assembly. This is due, to a large extent, to the non-uniform moderator distribution across the assembly; a significant fraction of the moderator is located in the water gap between the fuel assemblies, making the assembly center

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undermoderated. This undermoderation and heterogeneity penalize the performance of BWRs in a number of ways that affect their economics and safety.[1]

Many measures for alleviating the undermoderation in BWRs have been proposed throughout the years;[1-6] most of them call for replacement of part of the fuel by either water (water rods) or solid hydride (such as ZrH_x). Although improving the moderation, all of these measures penalize the BWR core performance in a different way: by reducing the overall length of fuel rods in the core, and/or by reducing the total weight of fuel in the core. Another design approach was recently proposed by one of the authors[7]. It calls for use of hydride fuel instead of water rods and, possibly, instead of oxide fuel in fuel assembly regions which could benefit from additional moderation. A specific hydride fuel proposed is U-$ZrH_{1.6}$, the type of fuel used in TRIGA research reactors[8]. It is anticipated that such hydride fuel is compatible with BWR operating conditions; a significant volume of such fuel has been irradiated at higher power densities and for higher burnups than uranium oxide fuel in BWRs. This is illustrated in Table I that compares selected characteristics of hydride fuel of the high power Rumanian TRIGA reactor[9] with those of a commercial BWR. A unique feature of this newly proposed approach is that it enables to practically eliminate all the undermoderation in BWRs without having to reduce the total length of fuel rods in the core. Another unique feature of this approach is that it enables a more uniform distribution of the moderator across the fuel assembly.

The purpose of the present work is to explore the feasibility of improving the performance of BWRs by designing them to include hydride fuel in selected locations of their fuel assemblies. The approaches for incorporation of hydride fuel within the BWR fuel assembly considered in this work are described in section 2. Section 3 describes the computational tools used for this study and their benchmarking. The results of this study are summarized in sections 4, 5 and 6 while the conclusions and suggestions for future work are presented in section 7.

2. APPROACHES FOR INCORPORATION OF HYDRIDE FUEL IN BWR

2.1 HYDRIDE FUEL FORM AND PROPERTIES

The U-Zr hydride composition used for the TRIGA fuel has, typically, 1.6 hydrogen atoms per Zr atom, i.e., it is U-$ZrH_{1.6}$.[8] The Medium Enriched Uranium (MEU) fuel developed by General Atomics for TRIGA reactors contains 45w/o uranium[8]. This corresponds to a U/Zr atom ratio of 0.31. The hydride fuel we are proposing for BWRs and referring to throughout this research is to have the same elemental composition. The uranium enrichment is a design variable.

As illustrated in Table I, TRIGA fuel has been operating under conditions that meet or exceed BWR fuel performance requirements. Relative to UO_2 fuel in BWR, the TRIGA fuel operates at close to twice the average linear-heat-rate, and reaches more than twice discharge burnup. The fuel rods are of very similar radius but the TRIGA fuel has only approximately half of the U loading per unit length and uses stainless steel rather than Zy clad. The water in TRIGA reactors is at a significantly lower temperature than in BWR, and it does not boil.

The U-$ZrH_{1.6}$ fuel is reported[8] to be stable at steady-state temperatures up to 700C. Under transient conditions the U-$ZrH_{1.6}$ fuel can withstand temperatures as high as 1000C. Although these temperatures are significantly lower than the maximum permissible operating temperatures of UO_2 fuel, they enable to operate the U-$ZrH_{1.6}$ fuel in BWR with linear-heat-ratings that are comparable to those the UO_2 fuel is designed to operate at. This is due to the significantly higher thermal conductivity of the U-$ZrH_{1.6}$ fuel. Table II compares the temperature in the two types of fuel under identical BWR operating conditions.

The atomic density of hydrogen per unit volume of U-$ZrH_{1.6}$ is approximately 4.7×10^{22} cm^{-3}. This is very close to the hydrogen atomic density of 4.8×10^{22} cm^{-3} in liquid water at the BWR operating temperature of about 280C. As in a typical BWR the volume of fuel in a fuel rod is nearly 62% of the volume of water which surrounds the fuel rod, the hydride fuel significantly increases the hydrogen content in the fuel assembly. Therefore, the reactivity of the U-$ZrH_{1.6}$ fueled lattice should be insensitive to the void volume fraction.

Zircaloy is not a compatible clad material for hydride fuel, as the hydrogen of the fuel may hydride the clad. Stainless steel is compatible with hydride fuel, but may undergo stress corrosion in boiling water at the BWR operating temperatures. It thus appears that a special clad will have to be developed for hydride fuel in BWR. A number of approaches have been proposed[10] for the design of proper clad for hydride fuel. Most of them involve adding a hydrogen permeation barrier to the inner side of a Zy clad. Such barriers may have a negligible negative effect on the neutron balance of the hydride fuel lattice. Consequently most of the present feasibility study assumes that the hydride fuel is clad with Zy. A number of calculations were performed in order to check how sensitive the neutron balance is to replacement of Zy by SS.

2.2 FUEL ASSEMBLY DESIGN CONCEPTS CONSIDERED

Five approaches have been considered for the incorporation of hydride fuel in the BWR fuel assembly: (1) Hydride fuel rods replacing water rods. (2) Hydride fuel rods replacing water rods and a number of oxide fuel rods. (3) Hydride fuel rods are used throughout the assembly. (4) Hydride fuel replaces oxide fuel in the upper half of the fuel assembly. (5) Like approach "3" but using U-ThH_2 fuel instead of U-$ZrH_{1.6}$.

Except for approaches "3" and "5", it is assumed that the U-$ZrH_{1.6}$ fuel will be manufactured in the form of pellets, similar in size to the UO_2 pellets in the reference BWR. The hydride fuel pellets will be loaded into hydride-only fuel rods or into mixed oxide-hydride fuel rods. In case of a mixed oxide-hydride fuel rod, it is assumed that the hydride fuel pellets are separated from the oxide fuel pellets by a barrier that is not permeable to hydrogen. The outer dimensions of all the fuel rods are assumed to be identical. Similarly, except for approaches "3" and "5", the geometry and dimensions of the BWR fuel assembly is that of the reference BWR. A better performance might have been achieved by modifying the design of hydride fuel containing assemblies.

Different fuel assembly designs were considered for approaches "3" and "5". As these approaches introduce plenty of hydrogen within the fuel, it is not necessary to add additional

volume of water in the gap between fuel assemblies. Consequently, the water gap was reduced to 1 mm in the two sides of the assembly that are control blade free, and to 2 mm to the two sides that need to accommodate the control blade. The number of fuel rods was kept at 8x8 and the coolant cross section area was the same as in the reference design. These design modifications enabled to increase the hydride fuel pellet diameter by 26%, corresponding to 58.9% increase in the HM loading.

2.3 REFERENCE FUEL ASSEMBLY AND ASSUMPTIONS

A typical BWR 8x8 fuel assembly has been chosen as the reference assembly for this study. It includes fuel rods with 10 different uranium enrichment levels, burnable poison in 8 fuel rods and 4 water rods at the center-most sites. The radius of the fuel pellet, inner clad and outer clad is, respectively, 0.52197 cm, 0.53213 cm and 0.61341 cm. The fuel rod lattice pitch is 1.6256 cm and the fuel assembly box and water gap dimensions are given in Table III. The UO_2 density is 10.06435 g/cm^3 and its average temperature is assumed to be 1000K. The average clad and moderator temperatures are assumed to be 580K and 560K. When used, the $UZrH_{1.6}$ density is assumed to be 8.256 g/cm^3 and its average temperature is 785K.

Although the reference fuel assembly uses gadolinium burnable poison, we did not account for it in the present analysis. All the analysis was done for a single fuel assembly assumed to be in an infinite array of similar assemblies. Most of the studies assumed infinite axial dimensions with axially independent composition. An exception is the studies described in Sec. 5; they considered the actual assembly length and axially varying composition.

The attainable burnup was taken to be the upper limit of the integral of k_∞ with burnup that makes the value of this integral zero. The calculations assumed an average rod linear heat rate of 21 KW/m and expanded fuel pellet to neglect the radial gas gap between the fuel pellet and the inner surface of clad.

3. COMPUTATIONAL TOOLS

The utility program MOCUP[11] developed by Idaho National Engineering and Environmental Laboratory is used for most of this research. It is a linkage code that couples the MCNP generalized-geometry point-energy Monte Carlo transport code[12] and the ORIGEN2 isotope generation and depletion code[13].

Before applying MOCUP to our BWR fuel assembly studies, we checked it against two well-documented benchmarks – one thermal[14] and the other fast[15]. The description of these benchmarks, of certain parameters of using MOCUP and of the results of our comparison is given in reference 16. Following is a brief summary of our findings.

For the PWR unit cell burnup benchmark[14] our MOCUP results show the following trends: (a) They are in agreement with the measured values of the actinide concentrations to within 11%; most deviations are less than 5%. (b) They are in agreement with the measured concentration of all the fission products studied to within less than 10%. An exception is Sm-149 the deviation of

which may be up to 50%. (c) All the MOCUP isotopic concentration results fall within the range of the results calculated by different sets of codes. That is, the relative errors of MOCUP results with the average calculated values are less than the standard deviation of MOCUP. Several isotopes of Am and Sm are an exception, for certain burnup values. (d) The k_∞ values calculated by MOCUP deviate from the benchmark values by less than 0.012.

For the Na-cooled MOX fuelled fast reactor burnup benchmark[15] our MOCUP results show the following trends: (a) Our calculated BOL k_{eff}, neutron balance components (production, absorption and leakage) and their decomposition by isotope fall within the range of the results calculated by the benchmark participants. (b) The results of reactivity loss with burnup and of inner core and outer core isotopic composition at the end of life are also in the ranges of corresponding results given in the benchmark.

4. FULL-LENGTH U-$ZrH_{1.6}$ FUEL RODS

A number of possibilities for using full-length U-$ZrH_{1.6}$ rods in the fuel assembly were investigated. These include the following: (a) Replacing only the 4 water rods. (b) Replacing the 4 water rods and some of the oxide fuel rods at either the central or the peripheral locations of the fuel assembly. (c) Fueling an entire assembly of modified dimensions with uniform enrichment U-$ZrH_{1.6}$ fuel rods.

The design variables include the number and location of the hydride fuel rods, the uranium enrichment and, for possibility "c", the assembly and fuel rods dimensions. The performance characteristics considered are the peak-to-average linear heat rate across the assembly, the void coefficient of reactivity, the attainable burnup, the amount of thermal energy that could be extracted from the fuel assembly, and the needed natural uranium feed and separative work units (SWU).

4.1 U-$ZrH_{1.6}$ FUEL RODS REPLACING WATER RODS

A preliminary study[17] examined the effect of replacing the 4 water rods of the reference fuel assembly by hydride fuel rods. The coolant water void volume fraction considered was 40% -- representing a core average void fraction. The following conclusions were reached[17]: (a) Replacement of 4 water rods by hydride fuel rods reduces the peak reactivity swing by approximately 2%. (b) The reactivity of the assembly at high burnups tends to be slightly higher with the hydride fuel replacing the water rods. (c) Both of the above effects are insensitive to the enrichment level of the hydride fuel uranium. This indicates that the main reactivity effect of the hydride fuel is due to the hydrogen it adds and not to the addition of ^{235}U. As it increases the heavy metal (HM) loading of the fuel assembly, use of hydride fuel instead of water rods enables to increase by few percents the total energy extracted from a fuel assembly and, correspondingly, the time between refueling and hence the capacity factor. (d) The hydride fuel rods offer a slightly more negative void coefficient than water rods, the difference, however, is very small; most likely of no safety significance. (e) The power peaking factor is higher by about 6% for the assembly with hydride fuel using 3% enriched uranium. Re-optimization of the oxide fuel across the assembly can likely remove this negative effect.

Based on the above findings we concluded that if hydride fuel is to make a significant improvement in the BWR performance, more than 4 hydride fuel rods need be incorporated in the fuel assemblies.

4.2 U-$ZrH_{1.6}$ RODS REPLACING WATER RODS AND SOME UO_2 RODS

From here on the reference assembly is assumed to have one large water rod instead of the 4 small ones considered in section 4.1. The water rod is replaced by four hydride fuel rods having the same dimensions as the oxide fuel rods. In addition, hydride fuel rods are used at certain locations of the assembly instead of the oxide rods of the reference assembly. The number and uranium enrichment of the hydride fuel are design variables.

Table IV and Fig. 1 compare the beginning-of-life reactivity and void coefficient of reactivity of fuel assemblies having different number of hydride fuel rods that are in hot and clean (no fission products) condition. Let us first focus on Cases 3 through 9 (Table IV); the enrichment of the oxide in these cases is identical to that in the reference assembly. It is observed that the void coefficient of reactivity is less negative with more hydride fuel rods and with lower enrichment of the hydride fuel. The void coefficient sensitivity to the hydride fuel enrichment is very small. Lower void coefficient is favorable from reactor control and reactor stability considerations.

Table V shows the dependence of the uranium ore and SWU requirements on the number of hydride fuel rods used per fuel assembly. The hydride fuel uranium is enriched to 7.5%. Replacing the water rod by 4 hydride fuel rods increases both the uranium ore and SWU requirements. However, replacing oxide fuel by hydride fuel involves a reduction in both the uranium ore and SWU requirements; the reduction in the uranium loading going from oxide to hydride fuel more than compensates for the effect of higher enrichment of the hydride fuel uranium. The fuel assembly having 64 hydride fuel rods referred to in Table V (and in Fig. 1) is of the reference dimensions and uses fuel rods of standard dimensions.

The next question considered is what can be the performance of hydride fuel containing assemblies, if the oxide fuel enrichment distribution is optimized. A couple of optimal enrichment mixed hydride – oxide fuel assemblies have been designed. They are described in Fig. 2b and 2c and referred to in Table IV as Case 10 and Case 11. The assembly of Fig. 2b has 16 hydride fuel rods of 7.5% enrichment at its central locations. The assembly of Fig. 2c has 28 hydride fuel rods of 5% enrichment at its periphery. The enrichment distribution of the oxide fuel rods is optimized to flatten the power distribution while keeping the BOL k_∞ within 2% of the reference assembly (Case 2, Table IV).

The BOL pin-wise power distribution in the two optimized mixed oxide-hydride fuel assemblies is compared in Fig. 3 with that of the reference assembly. Their void coefficient of reactivity is shown in Table IV. It is observed that whereas the void coefficients of Case 10 are comparable to those of the reference assembly (Case 2), the void coefficients of Case 11 are approximately one-half. They are even somewhat smaller than the void coefficients of Case 9 that has 32 equal enrichment hydride rods at the center of the assembly. It is concluded that for hydride fuel to

significantly reduce the void reactivity coefficient of BWRs, it needs to occupy at least half of the fuel rods in the assembly.

Figure 4 compares the burnup dependent k_∞ of the two optimized mixed hydride-oxide fuel assemblies with that of the reference oxide fuel assembly. The k_∞ decline rate of the hydride fuel containing assemblies is smaller. This implies that the assemblies having hydride fuel can reach higher burnups than the reference assembly. Table VI compares the attainable burnup and other characteristics of the three fuel assemblies. It is observed that relative to the reference fuel assembly, the mixed hydride-oxide fuel assemblies are characterized by the following: (a) Reduced uranium loading by 9.7% or 19.5%. (b) Reduced uranium ore feed material by 8.4% or 10.7%. (c) Reduced SWU requirements by 9.5% or 8.2%. (d) Increased average burnup by 18.7% or 37.9%. (e) Increased amount of energy that can be extracted from the fuel assembly by 7.3% or 11.0%. The assembly residence time (or time between reloads) will be increased by the same amount. (f) Reduced void coefficient of reactivity at BOL by more than a factor of 2 and at EOL by more than a factor of 3. (g) Increased EOL peak-to-average pin power ratio by 3% or 8%. So far we did not attempt to optimize the mixed hydride-oxide fuel to have minimum EOL power peaking. We expect that such an optimization will eliminate the disadvantage of the mixed hydride-oxide fuel assemblies.

It is concluded that by incorporating a sufficiently large number of hydride fuel rods within the fuel assembly it might be possible to improve the performance of BWR in three ways: (a) Increasing the energy extracted per fuel assembly and the cycle length. (b) Reducing the uranium ore and SWU requirements. (c) Reducing the void coefficient of reactivity. The first two attributes might have a significant economic benefit. The realization of this benefit depends on whether or not the hydride fuel will be able to reach burnups as high as 160 GWD/tHM.

4.3 SIMPLIFIED ALL-HYDRIDE FUEL ASSEMBLIES

The simplified all-hydride fuel assembly has a larger cross section area obtained by eliminating the water gap between the assemblies – except for the space required for the control rod blades. Keeping the number of fuel rods per assembly at 64 and the total cross section area for coolant flow as in the reference assembly (See Sec. 2.2 for details) we were able to increase the fuel pellet diameter by 26 %. The corresponding increase in the uranium inventory per assembly is 58.9 %. This is also the relative increase in the energy that can be extracted from a given assembly for a given maximum hydride fuel burnup. All the fuel rods use uranium enriched to 4%.

Figure 5 shows the relative pin-wise power distribution in the simplified all-hydride fuel assembly and in an all-oxide fuel assembly. This oxide fuel assembly is identical to the reference assembly (Case 2, Table IV) except for the fuel composition: a uniform enrichment of 3.44 % (average enrichment of the reference assembly) is assumed for the present comparison. It is observed that the hydride power density distribution is flatter. By reducing the enrichment level of one corner rod and less so, for two rows of fuel rods adjacent to the control rod gap of the all-hydride fuel assembly it is possible to get a close to uniform pin-wise power distribution. That is, three enrichment levels are likely to give highly uniform power distribution in the all-hydride fuel assembly.

Figure 4 shows the burnup dependent k_∞ of the all-hydride fuel assembly. It is significantly flatter than that of the reference assembly even though the average enrichment of the hydride assembly is higher. However, as the BOL k_∞ of the hydride fuel is significantly smaller, its attainable burnup is also smaller. The attainable burnup and other characteristics of the all-hydride assembly are summarized in Table VI. The all-hydride fuel assembly has a small void-coefficient of reactivity but it can deliver significantly less energy than the reference assembly. However, by increasing its uranium enrichment it might be possible to design an all-hydride fuel assembly that can operate up to significantly higher burnups so as to generate comparable or larger amount of energy than the reference oxide fuel assembly.

5. U-$ZrH_{1.6}$ FUEL PELLETS REPLACING OXIDE-FUEL PELLETS IN THE UPPER PART OF THE CORE

We consider a single three-dimensional fuel assembly. The assembly dimensions in the x-y plane are the same as of the reference BWR fuel assembly considered in section 4.2. The assembly length, 370.8 cm, and the axial void distribution were taken from reference 18. The average assembly void fraction is 40.8%. Reflective boundary conditions are assumed in the x and y directions and a 20 cm thick reflector consisting of 25% SS and 75% water is used at the bottom and top of the fuel assembly. When used, the hydride fuel section of the assembly was taken to have no water rods. It contained 64 hydride fuel rods of uniform composition and same pellet dimensions as of the oxide fuel pellets. The clad of all the fuel rods is taken to be 0.4mm thick stainless steel.

First we searched for the optimal enrichment of the hydride fuel pellets in the upper half of the assembly that will minimize the axial peaking factor. It was found that the axial power shape is very sensitive to the hydride fuel enrichment; it can shift from peaking at the lower part of the core to peaking at the upper part of the core as the hydride fuel enrichment is being changed from below 5% to above 6%. The BOL optimal hydride fuel enrichment is 5.4%. Figure 6 shows the optimal axial power shape in comparison with the power shape of all-oxide fuel assemblies. The axial power peaking factor of the optimal mixed oxide-hydride fuel assembly is only 1.41. This is to be compared with 2.15 of the reference oxide fuel assembly. It is even smaller than that of the reference assembly with no void, 1.55. However, it was found that similar axial flattening of the power density shape can be obtained by increasing the enrichment of the uranium oxide in the upper half of the fuel assembly. For example, figure 6 shows that increasing the enrichment from the reference 3.44% to 5% at the upper half of an all oxide fuel rod shifts the power density peak from the lower part to the upper part of the core.

Next we studied the void coefficient of reactivity. Table VII compares the reactivity effect of changing the density of the water used as coolant from liquid density (zero power) to reduced density corresponding to the reference reactor axial void distribution[18] (full power). It is found that use of hydride fuel at the upper half of the core significantly reduces the negative reactivity effect due to voiding. The reduction can be by a factor between 2 to 3, depending on the hydride fuel enrichment. This is likely to improve the stability and safety of BWRs.

As Zircaloy is not a compatible cladding material for U-Zr hydride fuel, stainless steel 316 was assumed to be the clad material in this section and in section 4.3. The BOL reactivity effect of replacing 0.81 mm thick Zircaloy clad by 0.4 mm thick SS-316 clad in all the fuel rods in the assembly was calculated to be –10% for simplified all hydride fuel assembly and –7.3% for 3D hydride-oxide fuel assembly. However, a significant number of alternatives to SS clad have been identified[10]. If found practical, one of these alternatives might significantly reduce this reactivity penalty associated with the use of hydride fuel.

6. URANIUM-THORIUM HYDRIDE

According to Dr. M. T. Simnad[19] U-ThH_2 fuel is even more stable than U-$ZrH_{1.6}$ fuel and can operate at higher temperatures. As thorium is a fertile material, the HM density in the U-ThH_2 fuel can be very high. Had we loaded the simplified fuel assembly described in section 4.3 with U-ThH_2 fuel, the HM loading of this assembly would be just about twice that of the reference oxide fuel assembly. If the U-ThH_2 fueled core could be designed to reach the same burnup as the reference oxide fuel assembly, it could generate twice the amount of energy and operate for twice as long as the reference fuel assembly. This can have a significant economic benefit. In fact, it can have a significant economic benefit even for PWRs. Consequently we undertook a preliminary evaluation of U-ThH_2 as a fuel for BWR.

Figures 7 and 8 compare the k_∞ and initial-conversion-ratio (ICR) values of our simplified assembly fueled with U-ThH_2 with those of the U-$ZrH_{1.6}$ and reference oxide fuel assemblies. The change in the water-to-fuel volume ratio is done by adjusting the fuel rod diameter; the lattice pitch is kept constant. The water-to-fuel volume ratio in figure 8 is fixed at 1.10203. The uranium contents in the U-ThH_2 fuel is assumed to be $25^{w}/_{o}$. The uranium enrichment is uniform across the assemblies; its value in figure 7 is 3.44%, 4%, and 20% for, respectively, the oxide fuel, $ZrH_{1.6}$ fuel and ThH_2 fuel. In figure 8 the enrichment is the design variable. The coolant void fraction is 40%.

Figure 7 shows that the thorium-hydride based fuels are more reactive than the zirconium-hydride fuels. Of the thorium-hydride fuels, that using ^{233}U is the most reactive; it is significantly more reactive than the reference oxide fuel.

Figure 8 shows that for the same k_∞, the ^{233}U fuelled thorium-hydride fuel offers a significantly higher ICR than all the other fuel types considered. Hence, it is expected that the ^{233}U containing ThH_2 fuel assembly will give the flattest k_∞ versus burnup dependence, and hence, the largest amount of energy extracted per fuel assembly and the longest cycle time.

The void coefficient of reactivity of ThH_2 fuel was found to be somewhat more negative than for $ZrH_{1.6}$ fuel. For example, the average void coefficients between 0% void and 40% void are -5.7777×10^{-4}/(% void), -4.7114×10^{-4}/(% void) and -4.6256×10^{-4}/(% void) for, respectively, 6%, 15% and 20% enriched ^{233}U. The values of void coefficient of reactivity for the $ZrH_{1.6}$ and oxide fuel assemblies are given in Table IV and Table VI. This trend is probably due to the fact that the atomic density of hydrogen per unit volume of U-ThH_2, 4.0×10^{22} cm^{-3}, is smaller than that of U-$ZrH_{1.6}$, 4.7×10^{22} cm^{-3}, and of water, 4.8×10^{22} cm^{-3}.

7. CONCLUSIONS

Replacing water rods by hydride fuel rods can enable a few percent increase in the amount of energy that can be extracted per fuel assembly but will have a very small effect on the void reactivity coefficient.

For hydride fuel to have a significant effect on the BWR void coefficient of reactivity, it has to occupy at least half of the fuel rods in the fuel assembly. Doing so might simultaneously improve the performance of BWR in three ways: (a) Increasing the energy extracted per fuel assembly and the cycle length by up to ~10%. (b) Reducing the uranium ore and SWU requirements by approximately 10%. (c) Reducing the void coefficient of reactivity by, at least, 50%. The first two attributes might have a significant economic benefit. The realization of this benefit depends on whether or not the hydride fuel will be able to withstand burnups as high as 160 GWD/tHM. The third attribute is expected to improve the stability and safety of BWR and simplify their control requirements.

Replacement of all the oxide fuel by hydride fuel opens interesting new options for the design of BWR fuel assemblies. The net result might be simplified assembly designs that can generate significantly more energy while featuring small negative void coefficient of reactivity.

Replacing the oxide fuel in the upper part of a BWR fuel assembly by U-$ZrH_{1.6}$ fuel pellets can significantly reduce the magnitude of the negative void coefficient of reactivity, and can significantly flatten the axial power density distribution. However, by increasing the enrichment of the oxide fuel in the upper half of the core it is possible to achieve a comparable power flattening effect.

U-ThH_2 fuel appears to be more promising than U-$ZrH_{1.6}$ fuel for BWR. In may also be attractive for PWR as it features a significantly higher heavy-metal inventory per unit fuel volume than oxide fuel.

Use of SS clad in the entire assembly has a significant reactivity penalty of up to 10%. A significant number of alternatives to SS clad have been identified. If found practical, one of these alternatives might significantly reduce this reactivity penalty associated with the use of hydride fuel.

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REFERENCES

1. A. Weitzberg, “Boiling Water Reactor Fuel Bundle,” United States Patent No. 4,591,479, May 27, 1986.

2. D.H. Imhoff, “Boiling Nuclear Reactor and Fuel Element Therefor,” United States Patent No. 3,145,149, Aug. 18, 1964.

3. J.D. Gylfe, “Fuel-Moderator Element for a Nuclear Reactor and Method of Making,” United States Patent No. 3,145,150, Aug. 18, 1964.

4. P. K. Doshi, “Segmented Fuel and Moderator Rod,” United States Patent No. 4,664,88, May 12, 1987.

5. J. F. Wilson, P. K. Doshi and S. Cerni, “Light Water Moderator Filled Rod for a Nuclear Reactor,” United States Patent No. 4,664,878, May 12, 1987.

6. S. Uchikawa and M. Aoyama, “Fuel Assembly,” United States Patent No. 4,652,427, Mar. 24, 1987.

7. Greenspan, “BWR Fuel Assembly Having Oxide and Hydride Fuel,” United States Patent No. 5,349,618, Sept 20, 1994.

8. M. T. Simnad, “The U-ZrH_x Alloy: Its Properties and Use in TRIGA Fuel,” General Atomic Report GA-A16029, Aug. 1980.

9. E. Greenspan, Information received from General Atomics (1993).

10. E. Greenspan, Personal Communication with Dr. Massoud Simnad and with Prof. Don Olander, 1998.

11. R. L. Moore at al.,"MOCUP: MCNP-ORIGEN2 Coupled Utility Program," Idaho National Engineering Laboratory Report INEL-95/0523, 1995.

12. J.F. Briesmeister, Ed., “MCNP – A General Monte Carlo N-Particle Transport Code”, Los-Alamos National Laboratory Report LA-12625-M, Version 4B (1997).

13. A.G. Croff, “ORIGEN2 - A Revised and Updated Version of the Oak Ridge Code,” Oak Ridge National Laboratory Report ORNL/TM-7175, 1980.

14. D. DeHart, M. C. Brady and C. V. Parks, “OECD/NEA Burnup Credit Calculational Criticality Benchmark Phase I-B Results,” Oak Ridge National Laboratory Report ORNL-6901, June 1996.

15. The Working Party on the Physics of Plutonium Recycling of the NEA Nuclear Science Committee, “Physics of Plutonium Recycling, Volume 4, Fast Plutonium-Burner Reactors: Beginning of Life,” OECD Document, ISBN 92-64-14703-9, 1995.

16. K. Wang, T-P. Lou, E. Greenspan, J. Vujic, “Benchmarking and Validation of MOCUP,” to be published in the Proceedings of this Conference.

17. E. Greenspan, K. Wang, T. Postma and T.S. Yu, “Feasibility Study of U-Zr Hydride Fuel,” Technical Report, Department of Nuclear Engineering, University of California, Berkeley, CA, Feb 15, 1999.

18. GE Nuclear Energy, “Study of Plutonium Disposition Using Existing GE Advanced Boiling Water Reactors,” NEDO-32361, June 1994.

19. M. T. Simnad, “Uranium Thorium Hydride Nuclear Fuel”, United States Patent No. 4,493,809, May 27, 1986.

Table I. Characteristics of U-$ZrH_{1.6}$ Fuel of the Rumanian TRIGA Reactor vs. Characteristics of a BWR Fuel

Characteristics	TRIGA	BWR
Fuel O.D. (cm)	1.27	1.25
Cladding		
Material	SS	Zy
Thickness (mm)	0.40	0.86
Fuel loading (Kg U/m)	0.42	0.88
Av. linear-heat-rating (kW/m)	37	21
Discharge burnup (MWD/Kg)	80	30
Energy extracted from fuel (MWD/m)	68.7	28.7

Table II. Selected Temperatures in U-$ZrH_{1.6}$ and UO_2 Fuel Rods in BWRs

Characteristic	UO_2 fuel		U-$ZrH_{1.6}$ fuel	
Coolant temperature (C)	290		290	
Maximum linear heat rate, q_{max} (kW/m)	45		45	
q_{av}/q_{max}	0.45		0.45	
Temperature increase (C)	Max./Av.		Max./Av.	
Clad-coolant	8	4	8	4
Cladding	82	37	82	37
Gap [a]	226	102	113	51
Fuel	1279	576	204	92
Fuel center temp. (C)	1885	1009	697	474

[a] Heat conductance through the gap of TRIGA type fuel was assumed, after GA, to be twice that for oxide fuel.

Table III. Dimensions (in cm) of Reference Assembly and All-Hydride-Fuel Assembly

Dimension identification	Reference assembly	All-hydride-fuel assembly
Fuel rod		
Diameter of fuel pellet	1.04394	1.34158
Inner diameter of fuel clad	1.06426	1.34158
Outer diameter of fuel clad	1.22682	1.42158
Pitch of fuel rods	1.62560	1.73433
Water rod (large)		None
Inner diameter of water clad	3.20040	
Outer diameter of water clad	3.40360	
Assembly box		
Inner width	13.40612	14.18736
Thickness	0.20320	0.20320
Cross control blade		
Thickness	0.79248	0.79248
Length of one wing	12.45870	12.85470
Gap between assemblies		
Thickness of water gap between surfaces of assembly box and control blade	0.55626	0.20000
Half thickness of water gap between control blade free surfaces of assemblies	0.47498	0.05000

Table IV. Initial Reactivity and Void Coefficient of Different Assembly Composition

Case	Assembly type [a]	k_∞			Void coefficient	
		0% void	40% void	70% void	0% - 40% void	40% -70% void
1	Reference assembly with 4 small water rods	1.38117 [c] (0.00154)	1.35551 (0.00173)	1.31278 (0.00174)	-0.00034	-0.00080
2	Reference assembly with one large water rod	1.38008 (0.00163)	1.35884 (0.00172)	1.32041 (0.00170)	-0.00028	-0.00071
3	4 Hydride rods, 7.5% enrichment	1.38368 (0.00176)	1.35058 (0.00170)	1.30107 (0.00176)	-0.00044	-0.00094
4	8 Hydride rods, 7.5% enrichment	1.38585 (0.00189)	1.35700 (0.00187)	1.31281 (0.00188)	-0.00038	-0.00083
5	16 Hydride rods, 7.5% enrichment	1.40327 (0.00172)	1.37338 (0.00179)	1.33426 (0.00177)	-0.00039	-0.00071
6	16 Hydride rods, 5.0% enrichment	1.37992 (0.00133)	1.35927 (0.00182)	1.32010 (0.00158)	-0.00028	-0.00073
7	16 Hydride rods, 2.5% enrichment	1.35261 (0.00162)	1.33561 (0.00157)	1.29881 (0.00167)	-0.00024	-0.00071
8	32 Hydride rods, 7.5% enrichment	1.41581 (0.00147)	1.40120 (0.00167)	1.37428 (0.00177)	-0.00018	-0.00047
9	32 Hydride rods, 5.0% enrichment	1.37304 (0.00167)	1.36167 (0.00153)	1.33869 (0.00175)	-0.00015	-0.00042
10	16 Hydride rods, 7.5% enrichment, optimized [b]	1.37208 (0.00182)	1.35565 (0.00163)	1.31847 (0.00180)	-0.00022	-0.00069
11	28 Hydride rods at perimeter, 5% enrichment, optimized	1.36463 (0.00151)	1.35632 (0.00165)	1.33994 (0.00177)	-0.00011	-0.00030

[a] This column defines the number and type of water rods and the number, location and enrichment of the hydride fuel rods replacing the water or oxide fuel rods in the reference assembly. The water rods and/or hydride fuel rods are in the central part of the assembly unless specified otherwise. The geometry of the reference assembly is not changed and the clad is Zircalloy.

[b] "optimized" means that the enrichment distribution of oxide fuel rods is changed in order to make the power distribution across the fuel assembly flatter.

[c] The number in parentheses is the estimated standard deviation of k_∞ as obtained from MCNP.

Table V. Uranium Ore and SWU Requirements as a Function of the Number of Centrally Located Hydride Fuel Rods per Assembly [a]

Number of hydride fuel rods	0	4	8	16	32	64
Uranium ore (kg)	317.8	336.1	331.3	319.3	296.4	293.1
SWU (kg)	265.5	285.6	283.5	279.2	271.1	303.4

[a] The enrichment of hydride fuel is 7.5%.

Table VI. Comparisons of Some Performance Characteristics

Type of fuel assembly	Reference oxide fuel assembly	Optimal assembly with 16 central hydride fuel rods	Optimal assembly with 28 perimeter hydride fuel rods	Simplified all-hydride-fuel assembly with 4.0% enrichment
Fuel load(kgU/m)	49.7	44.9	40.0	32.6
Initial k_∞	1.35884 (0.00172)	1.35565 (0.00163)	1.35632 (0.00165)	1.21336 (0.00159)
Feed needed (kg)	317.8	291.1	283.9	242.4
SWU needed(kg)	265.5	240.2	243.6	213.3
Burnup (GWd/TU) Total energy (GWD/m)	59.4 2.95	70.6 3.17	81.9 3.28	50.4 1.64
Max. burnup (GWd/TU)	69.8	UO2: 68.9 U-$ZrH_{1.6}$: 152.3	UO2:70.7 U-$ZrH_{1.6}$: 168.9	68.1
BOL peak-to-average power ratio	1.11	1.07	1.23	1.40
EOL peak-to-average power ratio	1.43	1.55	1.48	1.33
BOL void coefficient	0-40%: -0.00028 40-70%: -0.00071	0-40%: -0.00022 40-70%: -0.00069	0-40%: -0.00011 40-70%: -0.00030	0-40%: -0.00010 40-70%: -0.00054
EOL void coefficient	0-40%: -0.00089 40-70%: -0.00140	0-40%: -0.00063 40-70%: -0.00149	0-40%: -0.00019 40-70%: -0.00043	

Table VII. Effect of Hydride Fuel in the Upper Half of the Core on the Void Reactivity

Fuel in upper half of assembly		Voided?	k_∞	Reactivity effect of voiding
Type	Enrichment	(yes/no)		(%)
UO_2	3.44%	yes	1.35081(0.00063)	
	3.44%	no	1.37826(0.00058)	-2.0
$UZrH_{1.6}$	4%	yes	1.20715(0.00060)	
	4%	no	1.21595(0.00057)	-0.7
$UZrH_{1.6}$	5.4%	yes	1.28075(0.00060)	
	5.4%	no	1.29508(0.00057)	-1.1

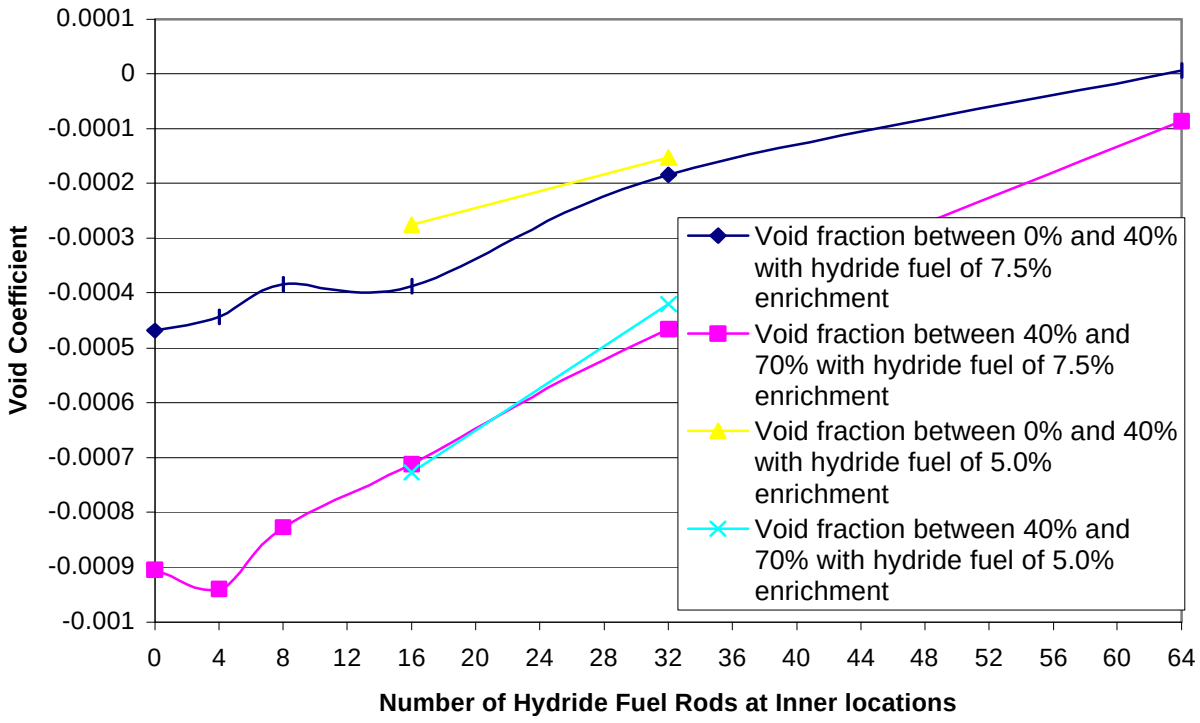

Figure 1. Effect of Number of Hydride Fuel Rods at the Fuel Assembly Center and of Hydride Fuel Uranium Enrichment on the Void Coefficient of Reactivity.

							2.8
						3.95	3.8
					3.95	3.95	3.95
				0.0	3.95	3.95	3.95
			0.0	0.0	3.95	3.95	3.95
		3.95	3.95	3.95	3.95	3.95	3.8
	2.8	3.6	3.95	3.95	3.95	3.8	3
1.6	2	2.4	2.8	2.8	2.8	2.6	2.2

(a)

							0.94
						0.91	1.03
					0.87	0.87	1.00
				0.0	0.94	0.87	0.96
			0.0	0.0	0.99	0.91	1.00
		1.10	1.01	1.04	0.96	0.95	1.06
	1.05	1.11	1.10	1.07	1.05	1.06	1.01
0.96	0.97	1.02	1.05	1.02	1.03	1.02	0.99

(a)

							2.8
						3.95	3.6
					7.5	3.95	3.95
				7.5	7.5	3.95	3.6
			7.5	7.5	7.5	3.95	3.6
		7.5	7.5	7.5	7.5	3.6	3
	2.4	2.8	3	3	3	3	2.6
1.6	2	2.2	2.4	2.4	2.4	2.4	2.2

(b)

							1.01
						0.99	1.05
					0.80	0.99	1.07
				0.90	0.86	0.98	1.01
			0.97	0.92	0.88	1.03	1.06
		1.03	0.99	0.96	0.91	1.03	0.99
	1.02	1.04	1.07	1.02	0.99	1.01	1.01
1.02	1.05	1.04	1.04	1.02	1.00	1.04	1.07

(b)

							5
						3	5
					3.95	3.6	5
				3.95	3.95	3.6	5
			3.95	3.95	3.95	3.6	5
		3.95	3.95	3.95	3.95	3	5
	2	2.6	3	3	3	2.4	5
4	5	5	5	5	5	5	5

(c)

							1.11
						1.08	0.93
					0.95	1.06	0.83
				0.86	0.91	1.01	0.79
			0.90	0.87	0.94	1.05	0.84
		1.14	1.01	0.99	1.06	0.99	0.89
	0.95	1.03	1.04	1.03	1.07	0.99	1.02
1.16	1.20	1.06	0.99	0.96	1.02	1.11	1.23

(c)

Figure 2. BOL Enrichment Distribution Across a Diagonally Symmetric Fuel Assembly.

Figure 3. BOL Normalized Power Distribution Across a Diagonally Symmetric Fuel Assembly.

(a) The reference oxide fuel assembly; (b) The optimized fuel assembly with 16 hydride fuel rods in inner locations; (c) The optimized fuel assembly with 28 hydride fuel rods at perimeter.

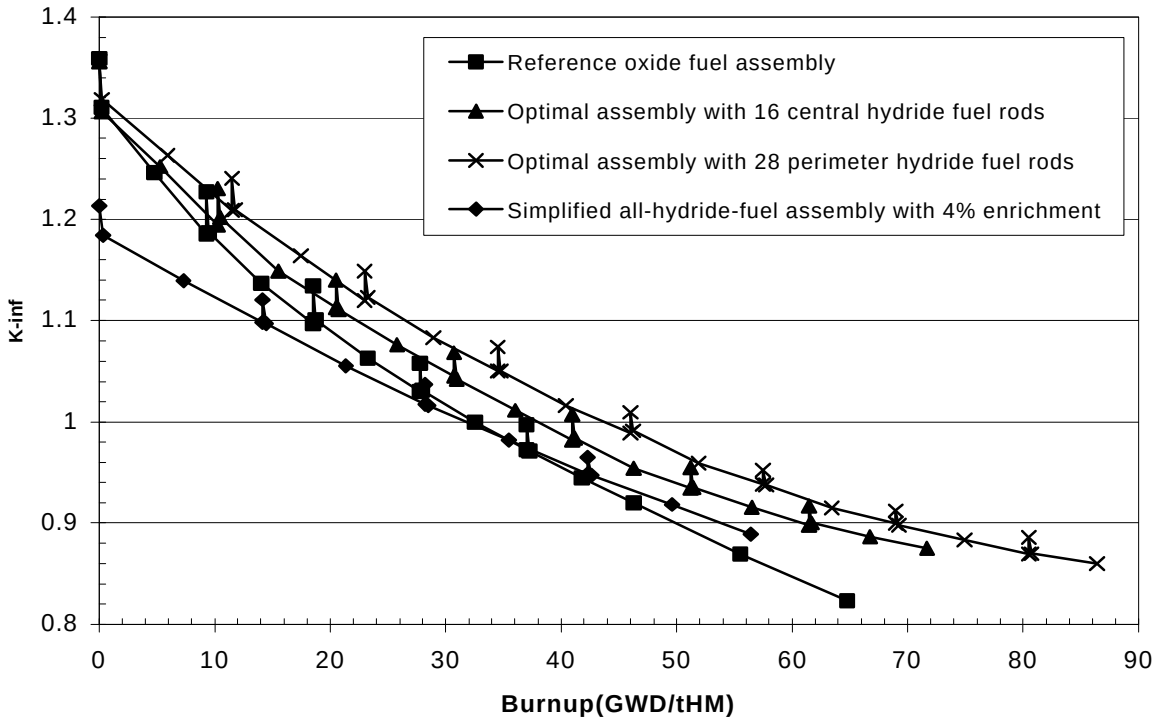

Figure 4. k_∞ Variation with Burnup for Different Fuel Assemblies.

				1.06			
				0.82	0.95		
			0.81	0.82	0.91		
			0.0	0.89	0.84	0.89	
		0.0	0.0	0.91	0.84	0.94	
		0.9	0.96	0.95	0.86	0.86	0.99
	1.06	0.97	0.94	0.93	0.91	0.96	1.11
1.61	1.3	1.18	1.13	1.14	1.15	1.21	1.34

(a)

				0.92			
				0.92	0.93		
			0.89	0.91	0.92		
			0.89	0.90	0.89	0.92	
		0.92	0.89	0.89	0.93	0.93	
		0.95	0.94	0.93	0.92	0.95	0.96
	1.09	1.01	0.99	1.00	0.98	0.99	1.03
1.40	1.27	1.18	1.14	1.15	1.14	1.15	1.18

(b)

Figure 5. BOL Normalized Power Distribution Across A Diagonal Symmetric Assembly. (a) Reference Oxide Fuel Assembly, Uranium Enriched to 3.44%; (b) All Hydride Simplified Fuel Assembly, Uranium Enriched to 4 %;

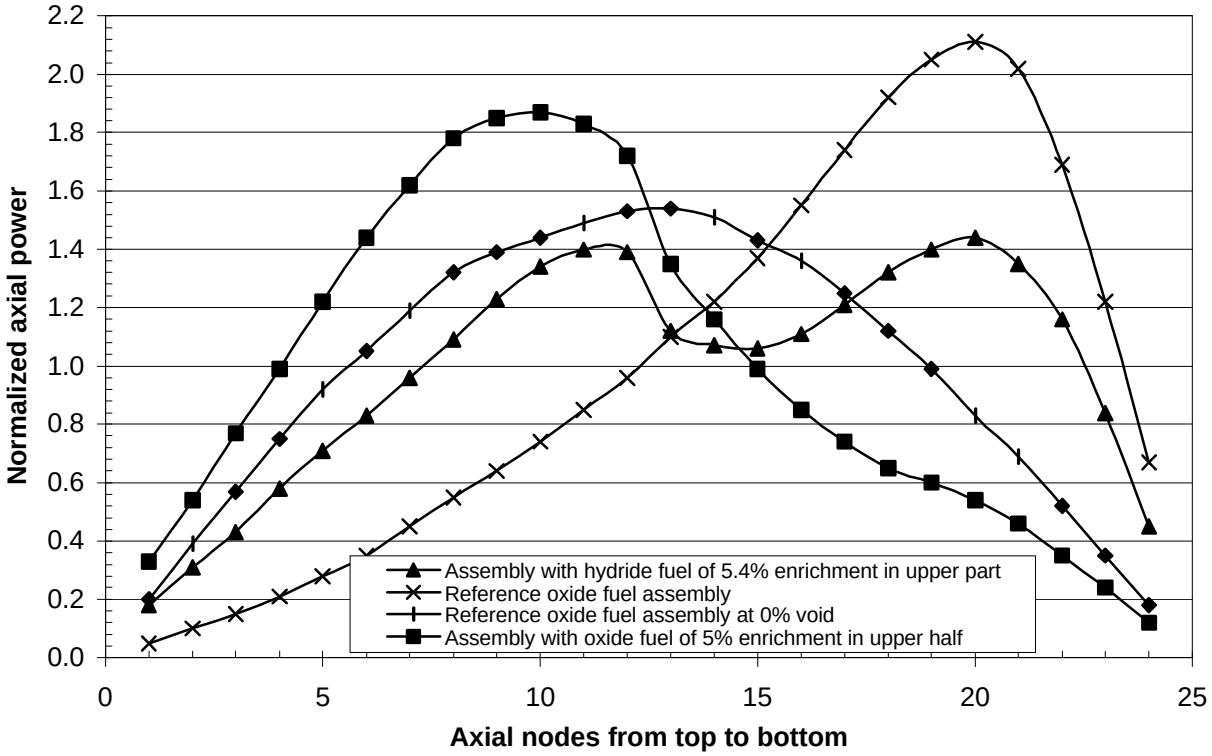

Figure 6. Normalized Axial Power Distribution of Different Fuel Assemblies

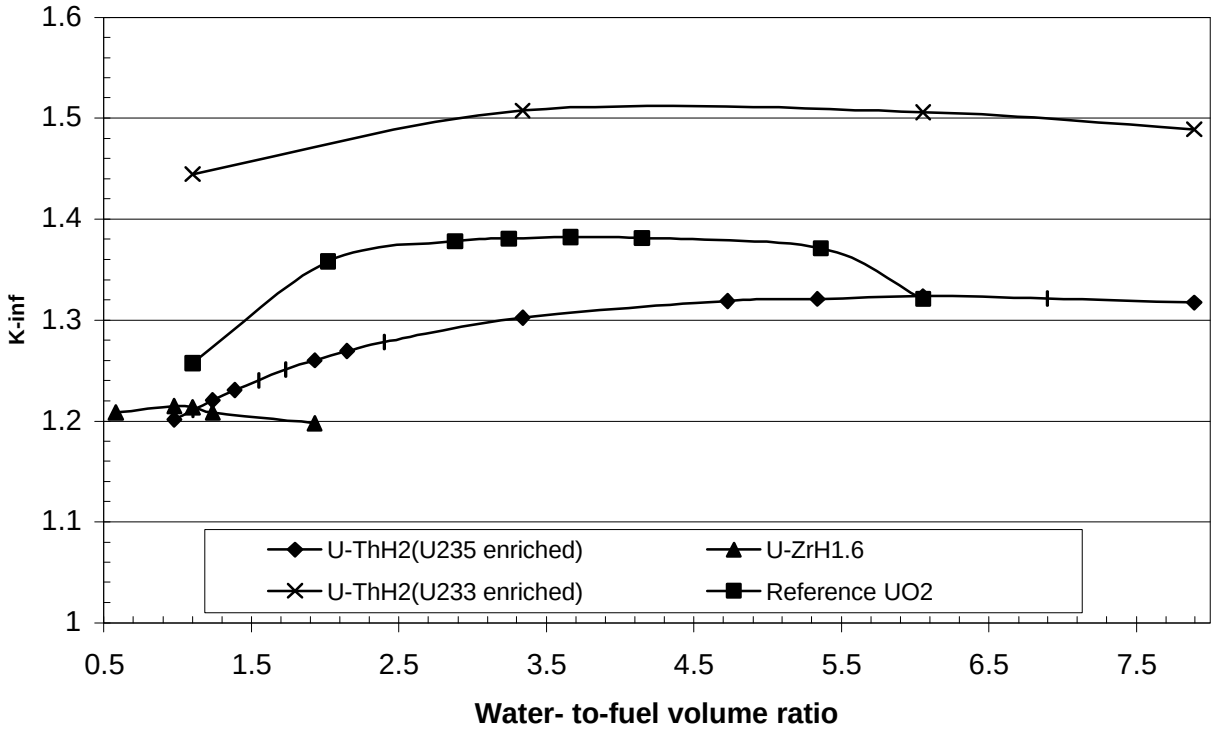

Figure 7. Effect of Fuel Rod Diameter on k_∞ of Several Fuel Assemblies. Uranium Enrichment is 4% for U-$Zrh_{1.6}$, 20% for U-Thh_2 and 3.44% for UO_2 Fuel. Coolant Void is 40 %.

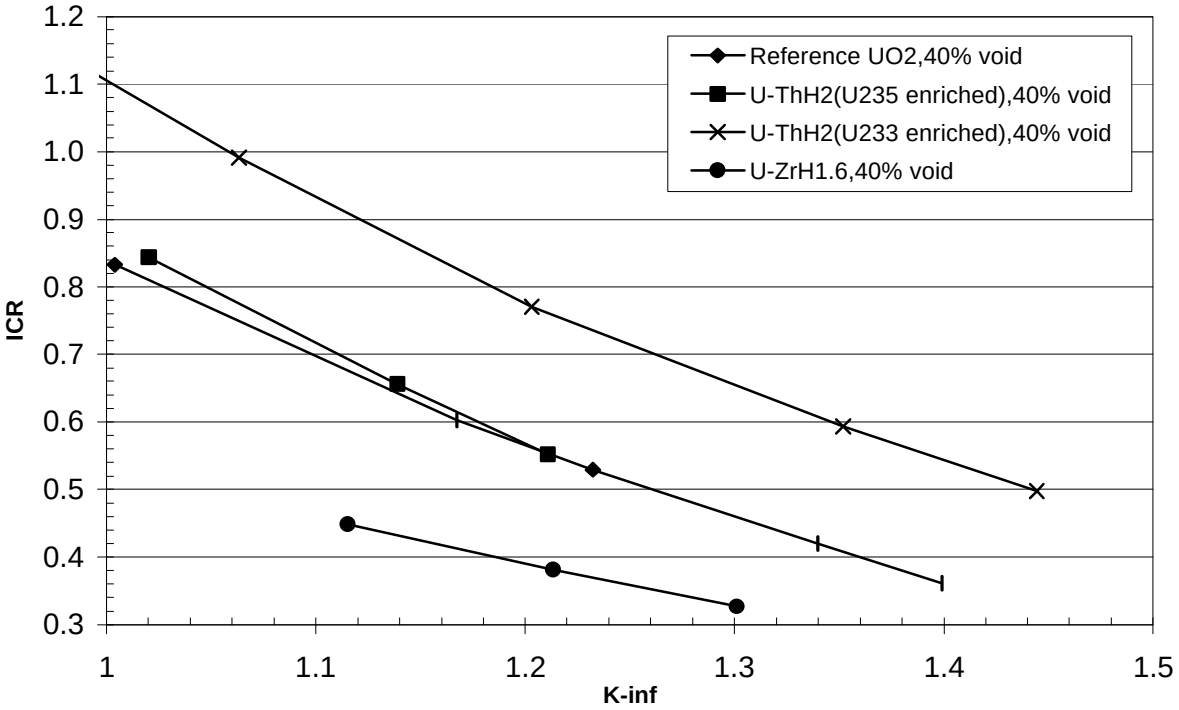

Figure 8. Initial Conversion Ratio Versus k_∞ of Several Fuel Assemblies. Water-to-Fuel Volume Ratio is 1.10203. Fuel Enrichment is Varied.