

IMPROVING PERFORMANCE WITH ACCIDENT TOLERANT-FUELS

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ABSTRACT

After the Fukushima reactor accident, efforts to improve risk management in nuclear operations have included the intensification of research on accident-tolerant fuels (ATFs). In this investigation, the physical properties of recently developed ATFs were compared with those of the current standard fuel, $\text{UO}_2\text{-Zr}$. The goals for innovative fuel design include a rigorous characterization of the thermal, mechanical, and chemical considerations. The intentions are to lengthen the burnup cycle, raise the power density, and improve safety. Fuels must have a high uranium density—above that supported by UO_2 —and possess a coating that exhibits better oxidation resistance than Zircalloys. ATFs such as U_3Si_2 , UN, and UC contain a higher uranium density and thermal conductivity than UO_2 , providing significant benefits. The ideal combination of fuel and cladding must increase performance in a loss-of-coolant accident. However, U_3Si_2 , UN, and UC have a disadvantage; their respective swelling rates are higher than that of UO_2 . These ATFs also have thermal conductivities approximately four times higher than that of UO_2 . A study was conducted investigating the hydrogen generated by the oxidation of zirconium alloys in contact with steam using cladding options such as Fe-Cr-Al and silicon carbide. It was confirmed that ferritic alloys offer a better response under severe conditions, because of their mechanical properties as creep rate. The findings of this study indicate that advanced fuels should replace $\text{UO}_2\text{-Zr}$ as the fuel system of choice.

1. INTRODUCTION

Following the Fukushima Daiichi accident in 2011, programs directed towards the development of accident tolerant fuels (ATFs) were initiated. These advanced fuels have the potential to reduce the disaster risk for the next generation of reactors. The ATF campaign has included studies for advanced fuels in the current phase between Gen-III and Gen-IV reactors. Innovative fuels should improve fuel management, keep nuclear plants safer, and reduce costs. ATFs permit extended fuel cycles and reduce waste, while improving efficiency and safety. These advanced fuels are designed to be suitable alternatives to current system that is based on $\text{UO}_2\text{-Zr}$ alloy. High thermal gradients are produced inside UO_2 pellets, weakening their structure and increasing incident risk. For standard fuel systems that work with a peak linear power rate of 45 kW/m, fuel centerline temperatures can reach more than 1400 °C. Several existing ATFs, such as U_3Si_2 , UC, and UN, have thermal conductivities approximately four times higher than UO_2 .

In addition, fuel cladding that avoids hydrogen explosion during a loss of coolant must be designed. Currently, it being studied options to replace zirconium alloys are silicon carbide (SiC), representing ceramic composites, and ferritic materials such as iron-chromium-aluminium (FeCrAl) alloys [1].

In this investigation, works with the licensed code FRAPCON (A Computer Code for the Calculation of Steady-State, Thermal-Mechanical Behavior of Oxide Fuel Rods for High Burnup) [2]. The fuel code was implemented after it underwent code modification to simulate advanced fuels, including U_3Si_2 , UN, and UC [3]. Advanced versions of U_3Si_2 , UN and UC were developed that comprised all thermal properties, swelling models, and creep rate equations [4]. FeCrAl alloy was also incorporated to make the cladding more resistant to steam oxidation. By expanding the code's ability to simulate advanced fuels, such as U_3Si_2 , UN, and UC, their performance can be compared with that of UO_2 .

2. ADVANCED FUEL CONCEPT

2.1. Progress Accident Tolerant Fuel for Light Water Reactors

Next generation nuclear reactors will need high-density uranium owing to their need for higher power output, but must also include the addition of accident-tolerant features. New materials must work at high temperatures, and improved options must replace the fuel systems of future power reactors. Innovative new fuels should be easy to manufacture, contribute to a high uranium loading, and show a low swelling rate when subjected to neutron irradiation. Concept fuels can work with up to 15% lower fuel enrichment than UO_2 , but their success depends on the materials used as cladding. The U_3Si_2 , because of its elevated thermal diffusivity, lower capacity for swelling, and compatibility with water, is an ideal candidate [5].

A system formed by U_3Si_2 -FeCrAl is a desirable option because it displays considerable improvement in fuel performance. However, only a small amount of data was collected in 1960 for U_3Si_2 , primarily in monolithic form. An ATF plan must consider the mechanical responses of the fuel during accident conditions [8]. Tolerant fuel must include a lower creep rate, and acceptable swelling rate for licensing of the fuel. UC and UN have poor chemical stability in the presence of water, reducing their popularity, while nitride fuels utilize a protection layer produced using U_3Si_5 to avoid water corrosion in the event of cladding failure.

2.2. Advanced Fuel Design Proposal

Since the mid-1950s, UO_2 has been used as nuclear fuel. The ceramic pellets show poor thermal conductivity and medium uranium density. Uranium enrichment is limited to 5%, which can be reduced by employing higher-density fuels. Cost reduction and safety improvements are the primary targets for next generation fuels. Pressurized water reactors (PWRs) have high fuel temperatures and a reduced burn cycle. High-pressure water accelerates cladding corrosion, producing free hydrogen during a postulated transient.

Currently, many proposed fuel concepts reduce accident risk by using UO_2 -BeO, U_3Si_2 , UC, and UN. However, UC and UN have a complex manufacturing process and are more suitable

for fast reactors working with a gas cooled system. U_3Si_2 has a lower melting point than UO_2 , but this is partially compensated for by an improved thermal conductivity which produces average fuel temperatures approximately 400 °C less than those of UO_2 [9] Ceramic materials, such as UN, were used in fast reactors and are being investigated as an option for the space program. The thermophysical properties of nuclear fuel, and cladding candidates are summarized in Table 1, [10].

Table 1: Fuel and cladding thermophysical properties.

Advanced materials	Melting point (°C)	Density (g/cm ³)	Thermal conductivity (W/m-K)	Heat capacity (J/Kg-K)	Linear expansion coefficient	Young's Modulus (GPa)
Accident tolerant fuels						
UO_2	2850	10.96	8.68	235	9.76	200
U_3Si_2	1665	12.2	16.3	202	15.2	77.9
UN	2365	14.33	13.0	190	7.52	199
UC	2507	13.5	25.3	200	10.0	172
Advanced cladding						
Zircaloy	1850	6.56	21.5	285	6.0	99.3
FeCrAl	2732	7.25	11.0	480	12.4	220
H-Nicalon	2800	2.65	1.5	670	4.66	250
CVD-SiC	2700	3.21	9.5	640	2.2	466

3. PHYSICAL PROPERTIES OF ADVANCED FUELS

The physical properties of tolerant fuel are factors that define the better performance of fuels investigated. In general, these properties are as well as knowledge of open literature. The material references, also used fuel systems that support many options of fuels, and cladding as BISON [11]. In next year's the fuel performance system will support ATF concept planned to FEMAXI (Finite Element Method in AXIs-symmetric system), [12].

3.1 Fuel Density

U_3Si_2 fuels have a higher load of uranium than UO_2 , enabling them to support long irradiation cycles. Silicide fuel has nearly 17% more uranium than UO_2 for the same volume U_3Si_2 [13]. This superior uranium loading allows for the possibility of either extending cycles, or working with reduced enrichment. Empirical correlations have been developed for fuel density through polynomial fitting. The following empirical correlations describe the density below melting in g/cm³, given as functions of temperature in K:

$$\rho_{UO_2} = -2.785 \times 10^{-11} T^3 + 4.102 \times 10^{-8} T^2 - 3.358 \times 10^{-4} T + 11.05, \quad (1)$$

$$\rho_{UN} = 1.058 \times 10^{-12} T^3 - 5.36 \times 10^{-8} T^2 - 2.717 \times 10^{-4} T + 14.42, \quad (2)$$

$$\rho_{UC} = -5.257 \times 10^{-12} T^3 - 2.658 \times 10^{-8} T^2 - 4.465 \times 10^{-4} T + 13.63. \quad (3)$$

3.2 Thermal Conductivity

Pellets with lower thermal conductivities experience reduced fuel centerline temperatures. The following empirical correlations were used to calculate the thermal conductivities of each fuel type as a function of temperature:

$$k_{\text{UO}_2(\text{TD}-95\%)} = 161.1T^{-0.4974} - 1.70, \quad (4)$$

$$k_{\text{UO}_2(\text{TD}-99\%)} = 184.1T^{-0.4943} - 2.152, \quad (5)$$

$$k_{\text{UO}_2} = 2.245 \times 10^{-13}T^4 - 2.028 \times 10^{-9}T^3 + 7.56 \times 10^{-6}T^2 - 0.01315T + 10.8, \quad (6)$$

$$k_{\text{U}_3\text{Si}_2} = 0.0151T + 6.004, \quad (7)$$

$$k_{\text{UN}} = 1.457T^{0.3865} - 0.1587, \quad (8)$$

$$k_{\text{UC}} = 1.014 \times 10^{-9}T^3 - 6.073 \times 10^{-6}T^2 + 0.01749T + 8.47, \quad (9)$$

where T represents temperature in K and k represents the thermal conductivity given in W/m-K. Each empirical correlation has been validated from 298 K to the melting point of the corresponding fuel. Unless otherwise stated, the porosity and theoretical density (TD) were 0.05% and 95%, respectively. It is worth noting that the thermal conductivity of U_3Si_2 is twice that of UO_2 at 25 °C was considered as room temperature in this study [14].

3.3 Thermal Expansion Coefficient

Pellet-to-cladding gaps change owing to linear expansion in the fuel pellet. Thermal expansion is a key parameter for calculating the gap heat transfer, and thus the stored energy. Furthermore, pellet strain is caused by thermal expansion, densification, and swelling – making thermal response a key factor when performing mechanical analysis. The thermal stress generates diametrical cracks in the pellet, resulting in fuel pulverization during long burn cycles. The following empirical correlations express the coefficient of thermal expansion as functions of temperature in K for UO_2 , UN, and UC, respectively:

$$\alpha_{\text{UO}_2} = 1.873 \times 10^{-10}T^3 - 2.29 \times 10^{-6}T^2 - 2.343 \times 10^{-5}T + 10.38, \quad (10)$$

$$\alpha_{\text{UN}} = 1.298 \times 10^{-11}T^3 - 5.086 \times 10^{-8}T^2 + 1.466 \times 10^{-3}T + 7.08, \quad (11)$$

$$\alpha_{\text{UC}} = 5.164 \times 10^{-12}T^3 - 2.477 \times 10^{-8}T^2 + 1.203 \times 10^{-3}T + 9.74, \quad (12)$$

$$\alpha_{\text{U}_3\text{Si}_2} = 15.7 - 0.002 \times (T - 273.15) \times 10^{-6}. \quad (13)$$

UC and UN exhibit thermal expansion coefficients higher than that of UO_2 , which increase with temperature. U_3Si_2 has a high thermal conductivity - approximately twice that of UO_2 at room temperature

3.4 Heat Capacity

The specific heat capacity of UO_2 at room temperature is 235 J/kg-K, and reaches 792 J/kg-K just below its melting point at 2847 °C. Advanced fuels exhibit reduced heat capacity compared to UO_2 . The following empirical correlations express the heat capacity in J/kg-K as functions of temperature in K, from room temperature, at 25 °C to below the melting point:

$$c_{P,\text{UO}_2} = -1.363 \times 10^{-11} T^4 + 1.5 \times 10^{-7} T^3 - 4.136 \times 10^{-4} T^2 + 0.4547 T + 136.5, \quad (14)$$

$$c_{P,\text{UC}} = -8.111 \times 10^{-12} T^4 + 5.676 \times 10^{-8} T^3 - 1.275 \times 10^{-3} T^2 + 0.1565 T + 171.6, \quad (15)$$

$$c_{P,\text{UN}} = 6553 \times 10^{-15} T^4 + 6859 \times 10^{-12} T^3 - 6859 \times 10^{-12} T^2 + 1368 \times 10^{-4} T + 159.1, \quad (16)$$

$$c_{P,\text{U}_3\text{Si}_2} = 0.02582 T + 140.5. \quad (17)$$

3.6 Fuel Enthalpy

Fuel enthalpy is obtained by integrating the heat capacity from 298 K to the temperature analyzed. The specific enthalpy of UO_2 , measured as $h(T) - h(298.15 \text{ K})$, varies from zero at room temperature to 11143 kJ/kg at 2847 °C. Fuel enthalpy is a metric applied for the analysis of the design basis for a reactivity-initiated accident. During a reactivity transient, the threshold for fuel failure (in terms of the enthalpy) is primarily defined as 280 J/kg for fresh fuels. The following empirical correlations express fuel enthalpy in J/kg as functions of temperature in K:

$$h_{\text{UO}_2} = 2.858 \times 10^{-12} T^5 + 3.885 \times 10^{-8} T^4 - 1.428 \times 10^{-4} T^3 + 0.2352 T^2 + 131.6 T - 56325 \quad (18)$$

$$h_{\text{UC}} = -1.6222 \times 10^{-12} T^5 + 1.419 \times 10^{-8} T^4 - 4.25 \times 10^{-5} T^3 + 0.0783 T^2 + 171.6 T - 57074 \quad (19)$$

$$h_{\text{UN}} = 1.3106 \times 10^{-12} T^5 + 1.7148 \times 10^{-9} T^4 - 2.367 \times 10^{-5} T^3 + 0.0684 T^2 + 159.4 T - 53592 \quad (20)$$

$$h_{\text{U}_3\text{Si}_2} = 0.0129 T^2 + 140.5 T - 4303.7 \quad (21)$$

3.6 Fuel Swelling Model

The strain produced during irradiation cycles, combined with volumetric swelling, is the basis of the correlation adopted for U_3Si_2 . The developed swelling correlation utilizes polynomial fitting performed on data from fuel plates. The following empirical correlations were adopted for swelling for each of the advanced fuels:

$$Sw_{\text{U}_3\text{Si}_2} = 3.8808 B u^2 + 0.79811, \quad (22)$$

$$Sw_{\text{UN}} = 4.7 \times 10^{-11} \times T^{3.12} \times B u \times \rho^{0.5}, \quad (23)$$

$$Sw_{UC} = 1.5 + (6.412 - 0.0198T + 1.52 \times 10^{-5} T^2) \left(\frac{Bu}{Bu_0} - Ratio \right), \quad (24)$$

where Bu is the burnup given in FIMA units, T is the temperature in K, and $Ratio$ is the O/M ratio, which equals 2.0. The irradiation swelling trends of UC and UN are considered superior to that of UO_2 . The swelling rate of UN ($\Delta V/V=0.9$ vol%/ %Bu) is lower than that of UC ($\Delta V/V=1.5$ vol% / %Bu).

3.7 Fuel Creep Models

The creep model in FRAPCON code is defined for UO_2 and divided into two sources: 1) thermal creep with temperature dependence, and 2) irradiation creep as a function of fission density and stress. The creep equations used for UN and UC, implemented as functions of pressure, porosity, and fission rate are:

$$\varepsilon_{UN} = \left[2.054 \times 10^{-3} \sigma^{4.5} e^{-39369/T} \times \frac{0.987 e^{-8.65P}}{(1-P)^{27.6}} \right] + \left[1.81 \times 10^{-26} (1+1250P^2) \sigma F \right], \quad (25)$$

$$\varepsilon_{UC} = \left[1.45 \times 10^{10} \sigma^{2.44} \times e^{-63000/T} \right] + \left[3.6 \times 10^{-22} \times F \sigma \right], \quad (26)$$

where P is the fraction of porosity in the pellet, F represents the fission rate (fissions/cm³-s), and ε represents the irradiation creep rate measured in s⁻¹. The thermal creep for UO_2 is valid in the range of 750 °C–1200 °C. In general, thermal creep is negligible for UN and UC at temperatures below 1000 °C. Advanced fuels have lower creep rates than UO_2 . The key parameters used are fission density (ranging from 8.4×10^{17} – 1.18×10^{20} fission/m³-s), and stress (ranging from 10–50 MPa).

3.8 Cladding Models

For the FeCrAl class of ferritic alloys, which resist steam oxidation at high temperature, the original model alloys contained 10–20 wt% Cr and 3–6 wt% dissolved Al. The steam oxidation resistance rises with the chromium and aluminum content. FeCrAl exhibits an oxidation rate that is half that of Zircaloy. Iron alloy can be treated with small amounts of molybdenum, niobium, and silicon to improve thermal stability. FeCrAl alloys may avoid the onset and extent of ballooning and burst, as well as potentially reducing heat release and hydrogen generation due to oxidation [15]. Empirical equations have been developed for the thermophysical properties of cladding material. The thermal properties of Zr-4, are expressed in the equations (27), (28), and (29). The thermal conductivity in (W/mK), and T is temperature (K) is:

$$k_{(Zr4)} = 7.51 + 2.09 \times 10^{-2} T - 1.45 \times 10^{-5} T^2 + 7.67 \times 10^{-9} T^3, \quad (27)$$

The specific heat capacity of Zr-4 in J/kg-K is:

$$Cp_{(Zr4)} = 245.11 + 0.1558T - 3.3414 \times 10^{-5} T^2, \quad (28)$$

Young's modulus of Zr-4 in Pa is:

$$E_{(Zr4)} = [9.900 \times 10^5 - 566.9 \times (T - 273.15)] \times 9.8067 \times 10^4, \quad (29)$$

The thermal conductivity of FeCrAl in W/m-K is:

$$k_{\text{FeCrAl}} = 0.01638T + 5.236, \quad (30)$$

The heat capacity of FeCrAl in J/kg-K is:

$$c_{p,\text{FeCrAl}} = -0.0001912T^2 + 0.5848T + 322.2, \quad (31)$$

The thermal expansion coefficient of FeCrAl in $\mu\text{m/K}$ is:

$$\alpha_{\text{FeCrAl}} = 0.01638T + 5.236, \quad (32)$$

Young's modulus in GPa is:

$$E_{\text{FeCrAl}} = -1.928 \times 10^{-5}T^2 - 0.06041T + 237, \quad (33)$$

In the first generation, FeCrAl alloys exhibited tensile creep up to 800 °C, with a composition of Fe-10–15Cr-5–6Al +Y₂O₃ (+Zr, Ti, Hf). The second generation employed an oxide dispersion-strengthened (ODS) FeCrAl alloy that had a higher mechanical strength and better radiation tolerance. These alloys contain metallic Yttrium (Fe-13Cr-4.5Al-0.05Y) in weight percent. ODS alloys provides better protection against accident conditions up to temperatures in the range of 1200 °C to 1400 °C.

3.9 Advanced Cladding Candidates

Iron-chromium-aluminum alloys are strong candidates that can replace zirconium alloys. Based on the studies of model FeCrAl alloys are better concerning to manufacture process, mechanical properties, oxidation resistance. Silicon carbide has received renewed interest in the development of a ceramic fiber called Nicalon, containing excellent strength and a high corrosion resistance. SiC fiber has a lower cross-section for thermal neutrons and a reduced creep rate at 1400 °C, exhibiting meager oxidation rates at 1700 °C, and working at temperatures exceeding 2500 °C.

4. RESULTS AND DISCUSSION

The standard UO₂-Zr fuel system was compared with the advanced fuel concept UN, UC and U₃Si₂ working with FeCrAl as cladding. In simulation assuming a standard fuel rod definition for a PWR 17 × 17 reactors, where the fuel response analyzed correspond to a normal operation. The simulation performed used the FRAPCON code with Zircaloy cladding, initially. In the simulations were performed for three increasing power rates of 32.8 KW/m, 49.12 KW/m, and 65.62 KW/m. The burn cycle was of 300 effective full power days. In this phase, the fuel response from the steady state. Because of the thermal parameters regarding the reduction in the centerline fuel temperature, that show a significant decrease for 65.62 KW/m. Table 2, describes the PWR 17 × 17 properties.

The fuel load per fuel assembly presents differences due to high uranium density of ATF candidates for the UO_2/Zr system employed 530 Kg, must reach 590 Kg for $\text{U}_3\text{Si}_2/\text{FeCrAl}$ and 692 Kg UN/FeCrAl .

Table 2: PWR 4-Loop core properties

Parameters	Values
Plant type	PWR
Plant electric output (MWe)	1150
Reactor thermal power (MWth)	3520
Inlet coolant temperature ($^{\circ}\text{C}$)	287.9
Coolant flow rate (kg/s)	1.74×10^4
Nominal coolant pressure (MPa)	15.51

The fuel parameters defined for PWR 4-loop reactors are shown in Table 3, for standard system UO_2/Zr . For FeCrAl alloys were changed the cladding wall thickness using 73% of zirconium alloys due to neutron penalty. The gap between fuel and pellet is the same, but pellet outside diameter must increase to compensate thickness of cladding.

Table 3: Input parameters of fuel rod

Fuel parameters	
Cladding outer diameter (mm)	9.4996
Cladding inner diameter (mm)	9.3345
Fuel pellet diameter (mm)	8.255
Fuel pellet height (mm)	10
Theoretical density of fuel (%)	95
Stack length (m)	3.6576
Plenum length (cm)	25.4
Dish radius (mm)	2.0066
Dish depth (mm)	0,28702
Fuel U-235 enrichment (%)	4.5
Fill gas pressure He (MPa)	2.41

The FRAPCON modeling parameters were defined for a postulated fuel rod divided into 20 axial nodes. Pellet fuel utilized 25 radial nodes. The mechanical model adopted was FRACAS. The fission gas model was standard Massih method, using 30 radial nodes for the fission gas models. Table 3 shows fuel properties used for simulation. In Table 4, describes the fuel changes used to simulate advanced system using FeCrAl as cladding. The gap diameter used was 0.166 mm, with a wall thickness of 0.57 mm. For the $\text{UO}_2\text{-Zr}$ system, the value of the hydraulic diameter was a pitch of 12.59 mm. Table 4, presents the key parameters adapted for simulations In Fig. 1, express the fuel centerline temperature, where can demonstrate the effect of high thermal conductivity of advanced fuels coupled with advances cladding.

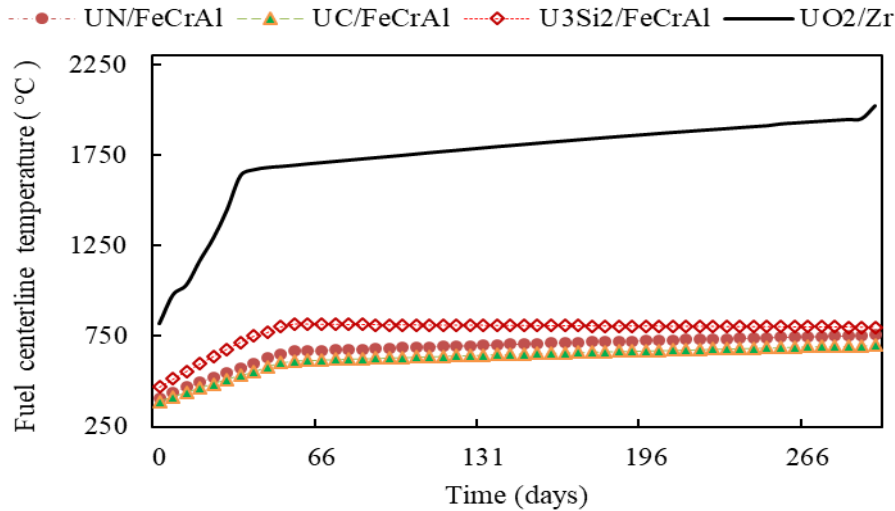


Figure 1: Centerline temperature for UN, UC, U₃Si₂ and UO₂-BeO, using FeCrAl

TABLE 4 : Performance indicator form simulation as FRAPCON versions

Advanced fuel concept	UO ₂ /Zr	UC/FeCrAl	UN/FeCrAl	U ₃ Si ₂ /FeCrAl
Cladding outside diameter (mm)	9.4996	9.4996	9.4996	9.4996
Cladding thickness (mm)	0.0826	0.6047	0.6047	0.6047
Gap thickness (mm)	0,5397	0,5397	0,5397	0,5397
Pellet outside diameter (mm)	8,2550	8.2992	8.2992	8.2992
Burn cycle of 300 days with average power 32.81 KW/m				
Maximum fuel centerline T, (° C)	1258	495	534	569
Maximum fuel average T, (° C)	791	423	445	463
Maximum pellet outside T, (° C)	370	345	346	348
Maximum cladding average T, (° C)	356	334	334	37
Burn cycle of 300 days with average power 49.21 KW/m				
Maximum fuel centerline T, (° C)	2023	701	672	696
Maximum fuel average T, (° C)	1186	555	538	542
Maximum pellet outside T, (° C)	399	382	376	359
Maximum cladding average T, (° C)	363	360	359	359
Burn cycle of 300 days with average power 65.62 KW/m				
Maximum fuel centerline T, (° C)	2570	624	757	800
Maximum fuel average T, (° C)	1514	509	591	602
Maximum pellet outside T, (° C)	419	376	383	382
Maximum cladding average T, (° C)	364	359	360	360

The chemical properties of silicide fuel are reminiscent of carbides and nitrides. Uranium nitride has the most attractive combination of properties of the advanced fuel candidates. The nearest competitor, UC, has the disadvantage of reacting with all refractory metals. UN as a nuclear fuel contains a worse chemical stability and must isolate using U₃Si₂ for example. Advanced fuel as U₃Si₂ and UC show temperature dependence with fission rate and gas swelling.

5. CONCLUSIONS

ATF concepts could provide a large incentive regarding safety performance. Advanced materials and manufacturing are the key factors to research and development for ATF success. The higher thermal conductivity of the UC, UN, and U_3Si_2 keeps the pellet at a lower temperature than UO_2 , causing less thermal expansion. For the same linear average power, the centerline temperature of UC is 400 K to 500 K less than UO_2 . Due to the high thermal conductivity of advanced fuels, the heat stress across the pellet must be not high enough to cause the fuel to relocate. An important factor is swelling rate demonstrated by fuels as UN and UC. The radial displacement could reach values that differ by two orders of the magnitude verified with the UO_2/Zr system in irradiation cycles of 60 GWd/MTU. ATF has a density of fissile material that reaches 30% higher than the UO_2 as UN. The thermal properties of the UN are better than UO_2 , the thermal conductivity about 70% greater than UO_2 , also has reduced specific heat to 70% of UO_2 , that must reduce the stored energy during a transient condition. UN and UC can produce higher fuel swelling rate and no desirable chemical reactions in contact with water/steam at high temperature. Advanced fuel as U_3Si_2 and UC have temperature dependence with fission rate and gas swelling.

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