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# COMPUTATIONAL ASSESSMENT USING SENSITIVITY ANALYSIS OF MORE TOLERANT FUELS

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***Abstract.*** *The secure operation of nuclear technology, including the avoidance of nuclear accidents and the optimal use of fuel cycles, poses a significant challenge. As of 2020, a total of 442 nuclear reactors are in operation in 30 countries across the world, accounting for over 13% of the global electricity demand. Over the last 50 years, light water reactors have operated using uranium dioxide as fuel and zirconium alloys as cladding. However, extensive research on more tolerant fuels has been conducted with high priority since the Fukushima disaster in 2011. New tolerance ideals seek the replacement of the standard fuel system with state-of-the-art options soon. Such investigations have predominantly focused on the responses and physical properties of new candidates for tolerant fuels, which are superior to those of standard fuel systems, based on close time-advanced alloys manufactured by substituting zirconium alloys with FeCrAl for enhanced corrosion resistance. Further, more tolerant fuels exhibit high uranium densities, with better thermal conductivities than uranium dioxide. This study proposes a fuel-licensing code as safety criteria collaborating with stochastic models and conservative rules. It also incorporates a sensitivity analysis using* $U_3Si_2$ *as fuel and FeCrAl alloys as cladding.*

***Keywords:*** *uranium silicide, FeCrAl alloy, uncertainty analysis, sensitivity analysis, FRAPCON*

## 1. INTRODUCTION

In 2011, it recorded an earthquake in the northeastern region of Japan, followed by a tsunami immediately before the nuclear accident at the Fukushima Daiichi nuclear plant. The public consternation regarding the Fukushima disaster's harmful impact ushered in a concentrated global effort to avoid similar accident risks and the leak of isotopes in the future. Current several plans are in progress to implement more tolerant fuels under the International Atomic Energy Agency (IAEA) sponsorship and the Organisation for Economic Co-operation and Development (OECD). Multiple agencies are collaborating to identify materials with higher resistance under adverse conditions.

This study follows the initiatives undertaken by the United States Department of Energy (USDOE) to develop accident-tolerant fuels (ATFs) (Zinkle et al., 2014). Since 2015, USDOE has supported the Fuel Cycle Research and Development program, coordinating research and development tasks about the Advanced Fuels Campaign (AFC) (Cappia et al., 2019). One of the AFC objectives is to identify and select advanced light water reactor (LWR) fuel concepts toward lead test rod testing by 2022. Initiatives such as ATF and AFC involve national laboratories, fuel manufacturers, and even international collaborations between Korea, France, Japan, China, Russia, and Europe. The most accident-resistant fuels identified thus far exhibit certain standard features. ATF fuels have been observed to exhibit high thermal conductivities, combined with a high density of fissile materials, higher than those shown by $UO_2$, which reveals a density of 10.96 g/cm$^3$.

The ATF program created an international collaboration steady to produce a substitute for uranium dioxide ($UO_2$). The replacement candidates investigated are uranium silicide ($U_3Si_2$), uranium mononitride (UN), uranium carbide (UC), and uranium diboride ($UB_2$) (Petrie et al., 2018). In 2015, Westinghouse started an irradiation test with UN-$U_3Si_2$ nuclear fuel using Zirlo as cladding to measure oxidation effects in aqueous environments.

Zirconium alloys (zircaloys) exhibit considerable oxidation resistance, but their contact with steam at temperatures above 850 °C induces hydrogen release, causing an explosion. Iron-chromium-aluminum (FeCrAl) alloys provide the most viable alternatives for zirconium alloys. In particular, Kanthal APMT provides continuous service up to 1250 °C (Rebak et al., 2016).

Nuclear reactors are complex systems with about 100,000 inputs and 10,000 output parameters on average, each containing uncertainties (D'Auria et al., 2008). Over the years, it arises many computational tools have been independently employed to improve neutron transport, thermal and hydraulic systems, and fuel performance. The programs defined during the research of advanced fuels require an adaptation method to simulate different compositions (Ott et al., 2014).

We varied the fuel performance codes to assess the performance of $U_3Si_2$/FeCrAl as a fuel. The uncertainty treatment compares the safety of operations using $U_3Si_2$/FeCrAl with those using standard $UO_2$/Zr fuels (Terrani et al., 2018). The simulation occurs in the context of a loss-of-coolant accident (LOCA) in a system using $U_3Si_2$ as a tolerant fuel and FeCrAl as cladding.

This study proposes a framework to compensate for the deficiencies inherent to current computational tools solely based on options licensed by the United States Nuclear Regulatory Commission (USNRC). The primary obstacle to implementing versions capable of simulating the most resistant candidates is the stochastic treatment of fuel uncertainty. Both the fuel codes, FRAPCON and FRAPTRAN, correspond to options based on standard fuel series (Luscher et at., 2011). The framework proposed in this study allows the fuel response to combine uncertainty and sensitivity analyses. The framework enables the use of performance codes to identify the factors responsible for reducing safety margins during the investigation of accident-tolerant fuels.

### 1.1 Uncertainty analysis

The nuclear system is using conventional risk analysis methods based on hybrid methodologies comprising deterministic rules and statistical models. Classical analyses estimate the safety margins for steady-state and accident scenarios based on conservative assumptions and interpretations. Acceptance criteria for emergency core cooling systems (ECCS) in LWRs (10 CFR 50.46) must satisfy the three following primary criteria: (1) The peak cladding temperature (PCT) should be less than 1204.5 °C, (2) The local maximum clad oxidation should be less than 17%, (3) The core-wide oxidation should be less than 1% (Frepoli, Cesare, et al., 2016). The ECCS criteria partially establish margins based on deterministic factors, such as the primary 17% equivalent clad reacted limit (ECR) and the secondary safety parameter limit of PCT of 1204.5 °C used for LOCAs.

Uncertainty treatment improves the power output of the reactors while simultaneously reducing the initial investment. Licensing codes possess various input parameters, such as physical models, mechanical tolerances, and boundary conditions (Gomes and Teixeira, 2017). Uncertainties about thermal properties induce a broad spectrum of variability in fission gas release, thermal expansion, heat transfer, and mechanical models. The ascertainment of the contribution of manufacturing tolerances of fuel rods is essential because even slight variations in the initial conditions of the system induce significant alterations in the system (Gomes and Giovedi, 2019)

In 1999, the regulatory rule, 10 CFR 50.46, was amended to implement realistic physical models to predict uncertainties. At that time, the safety margins used conservative empirical rules to determine risk limits, which justified best-estimate (BE) approaches. In 2017, the USNRC proposed an additional control, called 10 CFR 50.46c, to apply to ECCS in LOCA scenarios, offering an equivalent cladding oxidation criterion based on the hydrogen content (Zhang, Hongbin, et al., 2018).

Since then, it has significant technological progress using best estimate plus uncertainty (BEPU), which requires uncertainty treatment based on the stochastic analysis. BEPU analysis, combined with uncertainty propagation methods, has become the standard for new nuclear power plants (Zhang, Jinzhao, et al., 2019). The use of combinations of deterministic and statistic approaches is a consequence of the amendment of the code scaling, applicability, and uncertainty (CSAU) method introduced by Westinghouse in 1996 by the USNRC. Over the years, the continued success of CSAU can be attributed to several risks-management approaches used to control uncertainty, including uncertainties induced by material properties, external loads, boundary conditions, and initial conditions. The University of Pisa proposed a framework for the assessment and uncertainty evaluation of thermal-hydraulic codes, called the Uncertainty Methodology Based-on-Accuracy Extrapolation (UMAE), which operates along with the Best-Estimate Plus Uncertainty (BEPU) method, highlighting its preferred model for transient calculations.

The CSAU and BEPU methodologies use Wilks formulation to determine the minimum sample size corresponding to the desired lower and upper tolerance limits (Guba et al., 2003). The models can support multivariate statistics used in applications of BEPU. The order statistic is a nonparametric technique that defines the minimum number of samples required to attain the desired significance and confidence levels (Sun et al., 2020). The order statistic method (OSM) describes an ideal-free distribution. Table 1 presents the minimum number of trials for given quantiles and confidence levels up to the third-order using the Wilks method.

Table 1: Sample sizes concerning different quantiles and confidence levels up to the third-order obtained using Wilks formula

| $\alpha$ (%) | $\beta$ (%) | 1$^{nd}$ $N_1(\alpha, \beta)$ | 2$^{nd}$ $N_2(\alpha, \beta)$ | 3$^{rd}$ $N_3(\alpha, \beta)$ |
|---|---|---|---|---|
| 90 | 90 | 45 | 77 | 105 |
| 95 | 95 | 59 | 93 | 124 |
| 99 | 99 | 90 | 130 | 165 |

Uncertainty analyses offer several benefits. We use probability distribution models to represent the uncertainty, which is then propagated from the input system to the physical models of the code. The propagation principle of the uncertainties is ascertained based on Monte Carlo simulations, which use sample sizes obtained via the Wilks method. Sensitivity analyses estimate the contributions of various uncertainty sources to global uncertainty in a mathematical model. Global sensitivity analysis (GSA) predicts fuel behavior with realistic safety margins.

Sensitivity analysis records the impact of multiple uncertainties propagated from the input model on the output model results. Statistical correlation measures such as the Pearson product momentum correlation (PPMC), the Spearman rank correlation coefficient (SRCC), and the Kendall rank correlation coefficient (KRCC) are metrics for the strength of monotonic relationships between input and output parameters (Ikonen, Timo, and Ville Tulkki. 2014). Manufacturing uncertainties are modeled using random distributions defined by their standard deviations and means. Following Wilk's theory, a minimum of 59 samples must attain the desired quantiles and confidence levels (95/95) for the first order. Further, the limited number of samples should represent all possible physical tolerances. Sensitivity treatment estimates the effects of variations on each item of interest.

### 1.2 Accident-tolerant fuel

The method proposed in this study uses paired models combined with the acceptance criteria described by 10CFR50.46. Likewise, the proposed substitute for zirconium-based alloys exhibits high resistance to oxidation, which prevents the accumulation of hydrogen released by steam corrosion.

The Los Alamos National Laboratory and Oak Ridge National Laboratory implemented plans to perform experiments using Kanthal alloy as cladding (Field et al., 2017). The nuclear community expects discontinuing of use of zirconium soon. FeCrAl alloys or silicon carbide fiber/silicon carbide matrix composites (SiC/SiC) should replace zircaloys in the next future (Field et al., 2017).

UC and UN fuels exhibit deficient chemical stability in the presence of water and require a protective layer that prevents unwanted contact with water (Nelson et al., 2013). However, uranium carbides and nitrides are more suitable for fast-breeder reactors that use sodium as a coolant with restricting contact with water. Reactors that use the fuels mentioned earlier are more compact than LWR reactors, exhibit steel linings, and are not water-cooled. Nuclear space exploration programs have presented an in-depth study of UN fuels. UN fuels exhibit more desirable properties than UC fuels in the context of fast reactors.

Experiments have revealed that $UO_2$ doped with 0.5 wt% $Cr_2O_3$ exhibits less swelling than the reference fuel of pure $UO_2$. A reduced swelling occurs because of the dopant, $Cr_2O_3$, minimizes pellet-cladding interactions, enhances the gas diffusion rate, and reduces grain boundary surface energy in $UO_2$-doped Cr2O3 (Che et al., 2018). Innovative fuel concepts such as $UO_2$ matrices dispersed with nanoparticles, such as carbon nanotubes and silicon carbide (SiC), have also been investigated. The spark plasma sintering method employs $UO_2$-SiC sintering and benefits from the high thermal conductivity of SiC (273.6 W/mK at 373 K). Further, the $U_3Si_2$/FeCrAl system has been reported as an integrated option because it strikes an appropriate balance between security and fuel performance. Ferritic alloys exhibit a higher penalty for thermal neutrons. However, the high density of fissile isotopes can compensate for this effect while maintaining enrichment limits of up to 5%. Nowadays, many fuel options like the (Th-Pu)$O_2$ fuel based on plutonium content should produce the same total energy release per fuel rods using the pure $UO_2$ fuel. The composition used contains 8% of $PuO_2$ and 92% of $ThO_2$

he chemical compositions and properties of FeCrAl alloys allow them to exhibit wide variability and share similarities with commonly used Fe- and Ni-base alloys. Several variants of the alloy comprising 15–22% of chromium performed inner programs had shown excellent performance.

Further, FeCrAl alloys exhibited higher tensile strengths of up to 800 °C, corresponding to a composition incorporating molybdenum, titanium, and hafnium. In the second phase are using FeCrAl steel containing reinforced oxide dispersion (ODS) of the type (Fe–12Cr–6Al-ODS) exhibit higher corrosion resistance and better tolerance to neutron radiation. ODS alloys containing metallic Yttrium offer better protection under accident conditions corresponding to temperatures in the range between 1200 °C and 1425 °C. Kanthal APMT, a ferritic alloy (Fe-21Cr-6Al-2Mo) with high creep strength, was selected as the cladding material. Unfortunately, FeCrAl alloys suffer from the disadvantage of high neutron capture, which reduces the cladding thickness and slightly increases fuel enrichment.

In 2018, it was tested a more tolerant coating at Georgia Power's Hatch reactor, i.e., Abrasion Resistant More Oxidation Resistant (ARMOR), an iron-coated zirconium cladding solution designed to provide considerable oxidation resistance and superior material response over various situations compared to previous offerings.

### 1.3 Fuel performance code

FRAPCON is a system code used to estimate the mechanical behavior in steady-state systems that simulate a single fuel rod up to 62 GWd/MTU (Geelhood et al., 2015). The fuel-rod analysis program transient (FRAPTRAN) is a Fortran-based system used to calculate the transient response of fuel rods in LWRs during accident conditions. Both FRAPCON and FRAPTRAN are products of the Pacific Northwest National Laboratory under the sponsorship of

USNRC (Geelhood and Luscher, 2016). Besides, neutron kinetic codes show a long track of work rendered, calculating the assembly-averaged neutron flux and power distributions to evaluate transient conditions. Transient modeling explores all dependence on stored energy and the thermal properties of the fuel.

## 2. MATERIAL AND METHODS

One of the primary challenges to ensuring nuclear safety is identifying materials that exhibit adequate radiation resistance as $UO_2$ and zirconium-alloys. Irradiation degrades the lattice structure of fuel crystals. The thermal conductivity of $UO_2$ decreases during the irradiation cycle because heat transfer depends on the atomic diffusion of the synthesized fission products. At least five thermal properties used predict $UO_2$ response suffers from irradiation within the reactor core, such as thermal conductivity, thermal expansion, density, heat fusion, enthalpy, specific heat, and the melting point being affected in particular.

The thermal and mechanical properties are the most relevant characteristics for prospective replacements of the classic $UO_2$-Zirconium fuel-alloy combination. Ideally, they should exhibit superior responses concerning the properties mentioned earlier than the standard option during transient conditions.

The library of materials (MATPRO) summarizes all physical properties obtained via studies conducted over the years that have been accepted by regulatory agencies. BISON is a finite element-based nuclear fuel performance code maintained by the Idaho National Laboratory that incorporates the relatively more accident-tolerant $U_3Si_2$-FeCrAl combination as an option. All computer systems used to estimate fuel performance are expected to be updated to support more accident-tolerant fuels shortly, pending approval in 2026 following the scheduled completion of the ATF program. $U_3Si_2$ exhibits specific desirable physical properties, including a high density that is 17% higher than that of $UO_2$, a higher thermal conductivity of 8.5 W/m-k at 25 °C and a melting point of 1665 ºC (Metzger et al., 2014).

### 2.1 Thermal response of more accident-tolerant fuels

Thermal diffusivity of material establishes a formulation where the thermal conductivity divided by the product of its density and its specific heat capacity under constant pressure. It represents the rate of heat transmission through a material. The thermal conductivity of fuel determines the rate of conversion of heat produced via fission into electricity. Generally, more accident-tolerant fuels exhibit better thermal conductivity, and it shows lower temperature variations over their volumes. In ceramic oxides, the diffusivity decreases with an increase in temperature. The thermal conductivity of $U_3Si_2$ is assumed to be twice that of $UO_2$ at room temperature in this study. Figure 1 depicts the thermal conductivity curves of the most accident-tolerant fuels.

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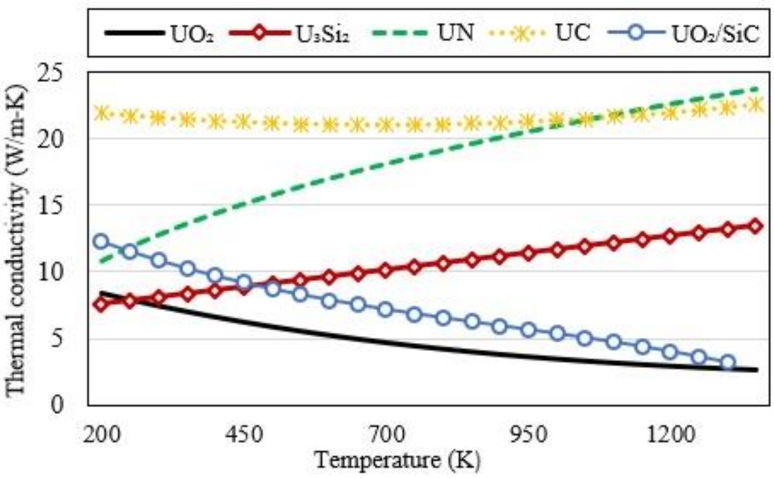


Figure 1. Thermal conductivity curves of the tolerant fuels, $UO_2$, $U_3Si_2$, UN, UC, and $UO_2$/SiC.

The thermal conductivity of $U_3Si_2$ is ~8.5 W/m-K at 300 K and increases with an increase in temperature, while $UO_2$ decreases with an increase in temperature. In contrast, $U_3Si_2$ increases with an increase in temperature. Figure 2 depicts the thermal diffusivity curves of the fuels, $U_3Si_2$, $UO_2$, UN, UC, and $(UPu)O_2$.

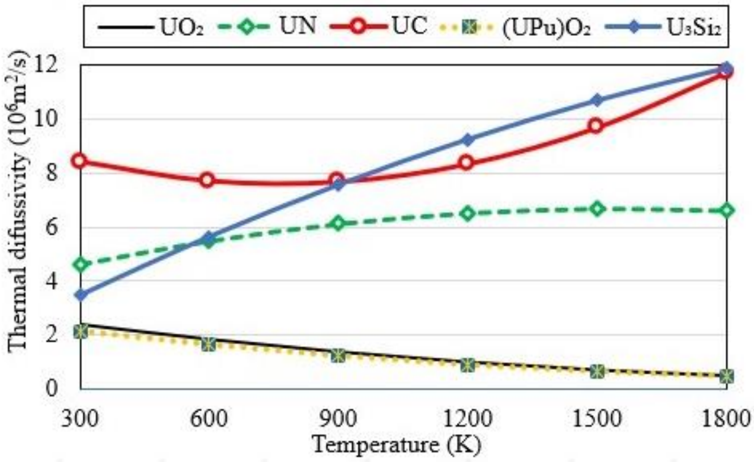


Figure 2. Thermal diffusivity curves of the fuels, $U_3Si_2$, $UO_2$, UN, UC, and $(UPu)O_2$.

## 2.2 Thermal expansion

Thermal expansion is a crucial parameter in calculating the heat transfer between the fuel pellet and the coating. A physical model was adopted in this study to determine the thermal expansion strains based on temperature. Volumetric deformations induced by thermal expansion and swelling make the thermal response a crucial factor during mechanical analysis. $U_3Si_2$ exhibits high thermal conductivity—it is twice that of $UO_2$ at room temperature. These considerations are essential as diameter cracks, which results from thermal stress, leading to fuel spraying over long firing cycles. Figure 3 depicts the thermal expansion curves for the candidates investigated in the ATF and AFC projects compared to $UO_2$.

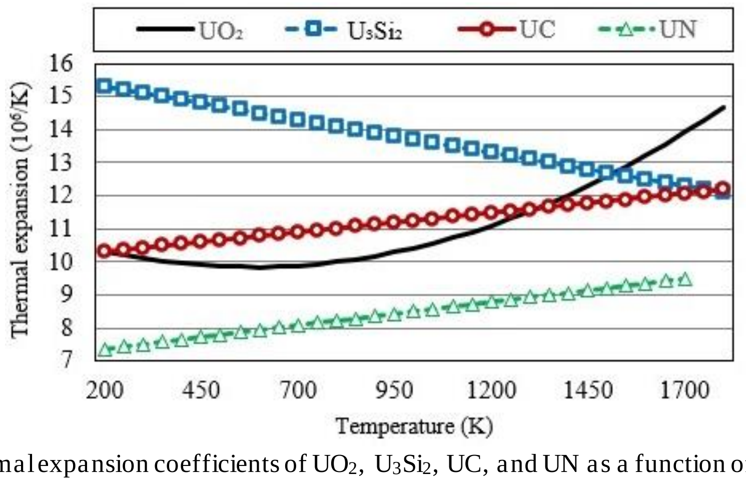


Figure 3. Thermal expansion coefficients of $UO_2$, $U_3Si_2$, UC, and UN as a function of temperature

## 2.3 Specific heat capacity

The specific thermal capacity of $UO_2$ is 235 J/kg-K at room temperature and 792 J/kg-K, just below its melting point of 2847 °C. The specific enthalpy of $UO_2$, defined as $H$ (T)–$H$ (298.15 K), ranges between zero at room temperature and 11143 J/kg at 2847 °C. Fuel enthalpy is a metric for reactivity accidents in design basis accidents. During a reactivity transient, the fuel enthalpy assumes the safety limit of 280 J/kg to fail used to fresh fuels. Ferritic alloys, such as stainless steel, exhibit values of 500 J/kg-K for specific heat, and zircaloys have specific heat in comparison to FeCrAl alloys. A higher heat capacity should improve fuel safety due to the slowing of the transient response. For cladding temperatures under steady-state or stable operation temperature varying around 350 °C, zircaloys have a specific heat varying from 306 to 317 J/kg-K, while FeCrAl exhibits values in the range of 574 to 609 J/kg-K in the same temperature conditions. The fuel enthalpy results from integrating the heat capacity function over the interval between 298 K and the analysis temperature. $U_3Si_2$ exhibits a lower specific heat than $UO_2$, as depicted in Fig. 4.

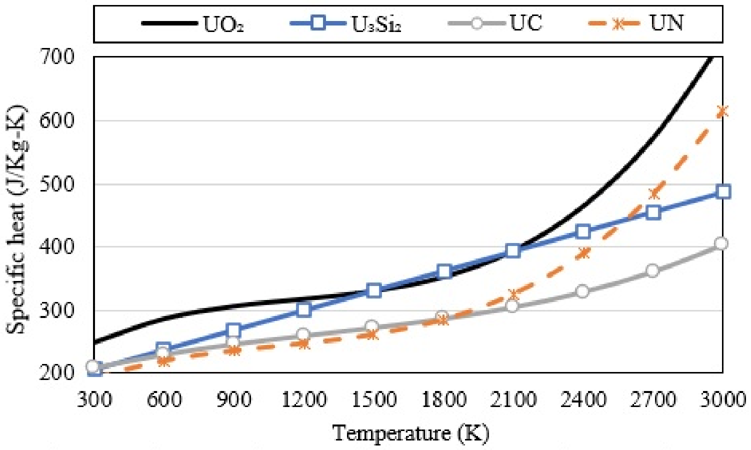


Figure 4. Specific heat capacities of $U_3Si_2$, UC, UN compared to the standard $UO_2$.

## 3. COMPUTATIONAL SIMULATION

An instrumented rod conducted in a controlled environment consisting of IFA-650-5 was chosen in this study as the scenario for the simulation. The Halden reactor research program (HRP) performed the IFA-650 series under the sponsorship of the OECD. This project improved the knowledge of fuels by conducting a series of LOCA tests. The program was conducted in Halden, Sweden, and comprised a total of 15 trials. The primary aim of this series was to research the behavior of fuels obtained from commercial suppliers under accident conditions.

Manufacturing tolerance accounts for variations in geometrical dimensions induced by industrial processes such as random sampling or normal distributions. A confidence interval of 95% is used for the input variables due to the central limit theorem. Table 2 illustrates the obtained uncertainties as a Gaussian distribution.

Table 2. Fuel parameters used as inputs with the corresponding uncertainties, mean, and upper and lower limits

| Fuel parameters | Fuel code parameter | μ (mean) | Upper (μ+σ) | Lower (μ-σ) |
|---|---|---|---|---|
| Cladding outer diameter | dco (mm) | 10.75 | 10.84 | 10.64 |
| Fuel theoretical density | den (%) | 94.80 | 96.69 | 92.91 |
| Cladding thickness | thkcld (mm) | 0.725 | 0.730 | 0.720 |
| Cold plenum length | cpl (mm) | 81.79 | 83.42 | 80.15 |
| Gap thickness | thkgap (mm) | 0.0851 | 0.090 | 0.080 |
| Cladding roughness | roughc (mm) | 0.0005 | 0.000510 | 0.000490 |
| Cold work | cldwks(dimensionless) | 0.5 | 0.51 | 0.49 |
| Pellet length | hplt (mm) | 11 | 11.22 | 10.78 |
| Pellet dish | hdish (mm) | 0.2794 | 0.28498 | 0.2738 |
| Pellet shoulder | dishsd (mm) | 1.20 | 1.22 | 1.17 |
| Fuel roughness | roughf (mm) | 0.002. | 0.00.04 | 0.00196 |
| Pellet sinter temperature | tsint (°C) | 1599.44 | 1631.79 | 1567.10 |
| Enrichment | enrch (%) | 3.50 | 3.57 | 3.43 |
| Cladding texture | catex (dimensionless) | 0.05 | 0.051 | 0.049 |
| Outer spring diameter | dspg (mm) | 8.59 | 8.79 | 8.41 |
| Center to center distance | pitch(mm) | 14.30 | 14.075 | 3.916 |
| Initial gas pressure | fgpav (MPa) | 4.00 | 4.075 | 3.916 |
| Coolant system pressure | p2 (MPa) | 15.50 | 15.82 | 15.52 |
| Coolant inlet temperature | tw(°C) | 301.67 | 308.05 | 295.28 |
| Coolant mass flux | go (kg/s-m$^2$) | 3539.76 | 3610.56 | 468.97 |

The IFA-650-5 experiments reveal an extension of the irradiation cycle to 1994 days using average linear powers of 37.5, 28.0, 22.0, 20.0, 18.0, and 18.0 kW/m. The outer diameter is of 9.132 mm, and the wall thickness is 0.721 mm. The plenum volume lies near 15 cm$^3$. A balanced mixture of 90% Ar and 10% used as the filling gas, pressurized at 4.0 MPa. The pellets use 3.5% enriched using uranium dioxide as fuel and Zircaloy-4 as cladding. The coating exhibited embrittlement following irradiation owing to a hydrogen uptake of 650 ppm. The fuel rod, IFA-650.5, undergoes cycles of 83.4 GWd/MTU. Under transient conditions during the heating phase, the temperature increases abruptly from 200 °C to 1100 °C over 178 s. The fuel reached a temperature of 1010 °C after initiating the blowdown.

The swelling model exhibits a dependence on temperature, burn-up level, and porosity. The effect of the transmission of fission products on physical models is significant. Fission gas release models increase the pressure and decrease the thermal conductivity of the pellet-gap interface. Table 3 presents the uncertainty models of FRAPCON. .Figure 5 illustrates the sensitivity analysis of ECR concerning input parameters, the outer cladding diameter, springer diameter, cold-work, and coolant mass flux.

Table 3: Model uncertainty variables for sensitivity analysis

| Physical models | Bias variable | Deviation range |
|---|---|---|
| Fuel swelling model | sigsweel | ± 1.5σ |
| Fission gas release | sigfgr | ± 1.5σ |
| Thermal conductivity | sigftc | ± 1.5σ |
| Cladding creep model | siggreep | ± 1.5σ |
| Fuel thermal expansion | sigftex | ± 1.5σ |
| Cladding corrosion model | sigcor | ± 1.5σ |
| Cladding axial growth model | siggro | ± 1.5σ |
| Cladding hydrogen pickup | sigh2 | ± 1.5σ |

FRAPCON shows the parameterization of uncertainty in physical models includes thermal conductivity, mechanical models, FGR, cladding oxidation, hydrogen content, and heat transfer models. Figure 6 depicts the statistical correlation index concerning the failure time of each fuel.

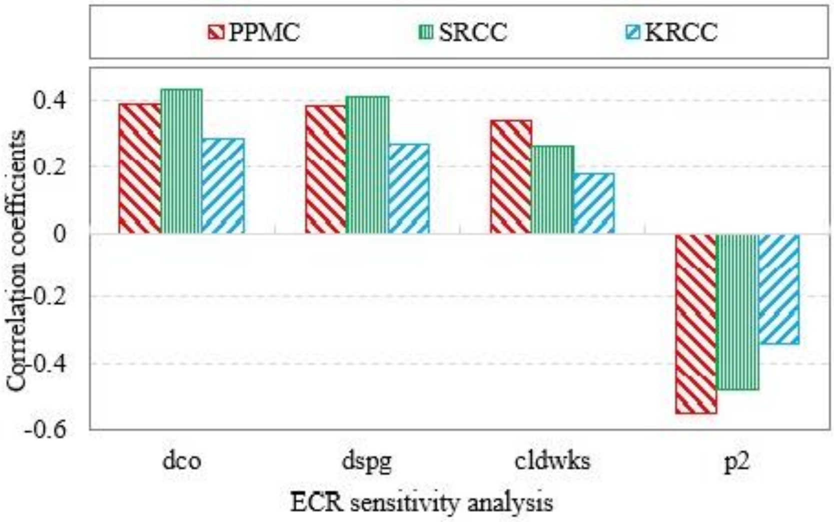


Figure 5. Sensitivity analysis of the input parameters for the oxide layer thickness

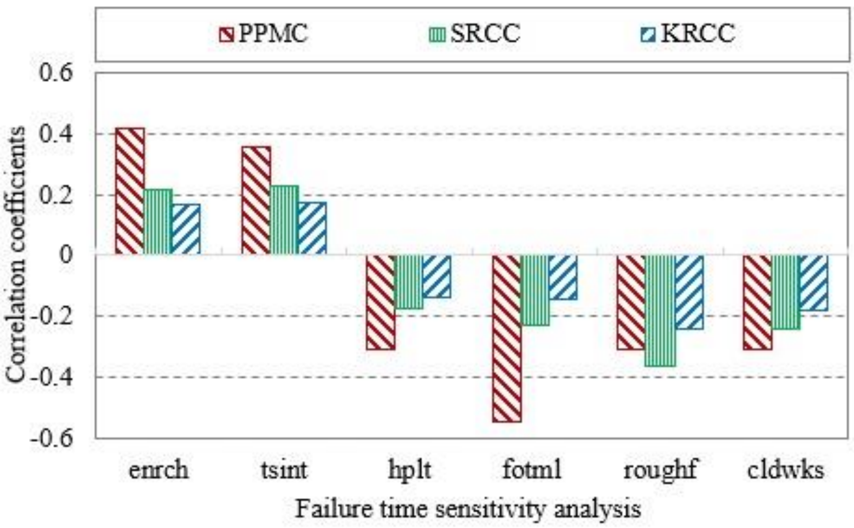


Figure 6. Sensitivity analysis of input parameters concerning the failure times

The internal standards of FRAPCON enable the performance of constructed sensitivity analyses based on thermal conductivity, thermal fuel expansion, FGR, fuel swelling, irradiation creep, cladding thermal expansion, and cladding corrosion via cladding hydrogen release. Following the execution of 96 run codes, had significant variations. However, better identification of cause and effect is obtained via the sensitivity analysis. When correlation coefficients are significantly greater than 0.30, the priority variables correspond to the statistical index. Figure 7 illustrates the statistical indices of the various fuels concerning the outer cladding temperature. The data in Table 4 reveals considerable variability in the safety parameters.

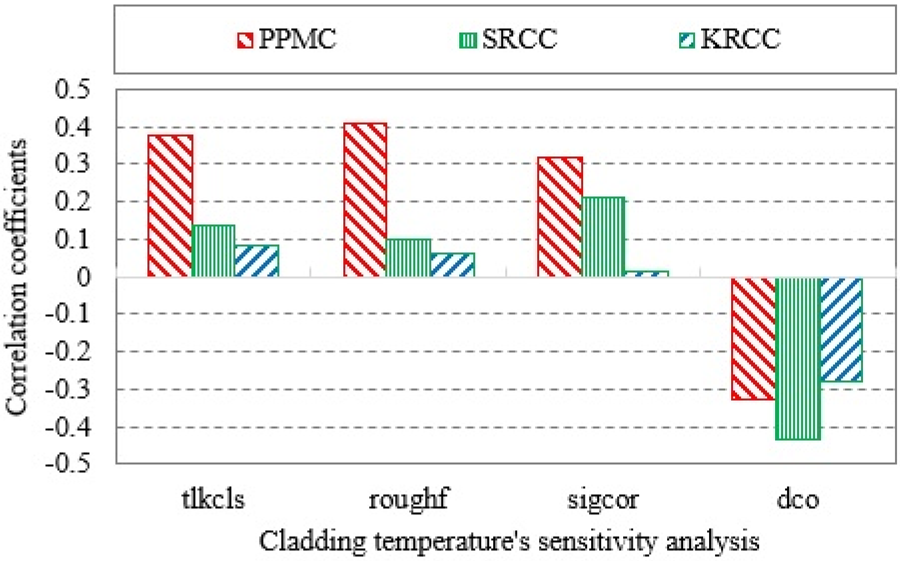


Figure 7. Sensitivity analysis of input parameters for the outer cladding temperature

In the simulation, it has a fast evolution regarding the causes and effects generated by the uncertainties. All deviations from the initial cycle represent the variability introduced into the boundary and initial conditions.

Table 4. Variability of safety parameters after uncertainty propagation over 96 run codes in $U_3Si_2$/FeCrAl fuel

| Safety parameters | Mean (μ) | Sigma (σ) | Variability (%) |
|---|---|---|---|
| Failure time (s) | 339 | 36.41 | 10.74 |
| Gas gap pressure (MPa) | 5.36 | 2.33 | 43.47 |
| Cladding inside temperatures | 763.55 | 6.30 | 0.85 |
| Cladding outside temperatures | 766.83 | 20.66 | 2.69 |
| Fuel average temperature | 956.27 | 13.16 | 1.38 |
| Fuel pellet surface temperature | 933.40 | 40.79 | 4.37 |
| Fuel centerline temperature | 956.24 | 45.75 | 4.78 |
| Cathcart-Pawel (CP) correlation (%) | 8.78 | 1.62 | 18.50 |
| Equivalent cladding reacts (%) | 8.57 | 1.34 | 15.70 |
| Cladding true stress (MPa) | 27.35 | 10.74 | 39.26 |

Also, it has reported approximately 100% variation in the diffusion coefficient. The fuel-rod corrosion model is primarily a function of time and temperature and has shown effects from the burn cycle extension. FRAPCON-4.0 is used to analyze the effect of the partial evolution of the physical model based on uncertainty quantifications and propagations without any alterations to the code. This framework exhibits results that enable the utilization of conservative rules coupled with stochastic approaches. Initially, defining the uncertainties introduced into FRAPCON, which its spreader into FRAPTRAN, combining boundary conditions. When the number of simulations increasing the output, parameters exhibit strong tendencies to increase the uncertainties. The calculated statistical correlation coefficients indicate the importance of uncertainty sources. The fabricated tolerances also exhibit a considerable influence on IFA-650.5

## 4. DISCUSSION

The results collected in this study establish that the $U_3Si_2$/FeCrAl system is a potential replacement of the standard $UO_2$/Zr combination owing to its myriad advantages. For example, it exhibits slightly higher enrichment due to the higher density of $U_3Si_2$. However, the thermal and physical properties of $U_3Si_2$ require further complementary research. The $U_3Si_2$/FeCrAl combination also indicates a decreased fuel centerline temperature. Further, the high thermal conductivity of $U_3Si_2$ reduces the fuel temperature during the steady-state. The combined effect of the thermal expansion of $U_3Si_2$ and the higher thermal expansion of Kanthal APMT induces a delayed gap-closure time.

The amount of released fission gas is lower than that in the standard $UO_2$/Zr system because the lower fuel temperature also decreases plenum pressure. Therefore, in accident scenarios, $U_3Si_2$/FeCrAl serves as a more tolerant fuel, exhibiting a behavior comparable to the classical $UO_2$/Zr system. The failure time of the fuel rod observed is to lie within the range of the standard deviation. Kanthal exhibits higher mechanical strength than zircaloy and better corrosion resistance, slightly improving IFA-650-5. Based on FRAPCON and FRAPTRAN, it also performed a sensitivity analysis in this study. The uncertainty levels are determined to be a consequence of manufacturing tolerances with boundary conditions dictated by uncertainty sources. The Wilks method shows that increasing the sample size increases the uncertainties and makes the sensitivity analysis more complicated.

## 5. ACKNOWLEDGEMENTS

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