*2021 International Nuclear Atlantic Conference – INAC 2021*
*Virtual meeting, Brazil, November 29 – December 2, 2021*

ISBN: 978-65-594-1256-3

# Steady State Studies on Accident Tolerant Fuels for Pressurized Water Reactor

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## 1. Introduction

Severe accidents, such as those at the Three Mile Island and Fukushima Daiichi have shown that under such extreme conditions, nuclear fuel will fail and the high temperature reactions between zirconium-base alloys and water will lead to generation of hydrogen, with the potential for explosions to occur, damaging the plant further [1]. Then, after the Fukushima accident, the nuclear industry focused the activities for the developing of nuclear reactor fuels with enhanced accident tolerance, that is accident tolerant fuel (ATF), mainly to widen the existing safety margin for nuclear plants.

The LWR (Light Water Reactor) fuel cladding is currently made from Zirconium-based alloys and it has been used for over 50 years. The Zirconium based alloys enables a combination of desirable properties, e.g., high mechanical strength, high melting point (2125 K), high corrosion resistance and a low absorption cross-section for thermal neutrons. The latter characteristic is the key advantage of zirconium over most other metals, since it translates directly into economic terms, i.e., the Fuel Cycle Cost [3]. However, it could not ignore the fact that when Zirconium alloys are submitted to high operation temperatures and burnup, there is an oxidation resistance reduction and increased hydrogen formation [2], [3], [4], [5]. Therefore, it is fundamental the developing of ATFs to provide better efficiency in the conversion of energy in nuclear reactors and capability to reduce the possible damages in fuel and claddings in accident situation are needed. Lately several different fuel-cladding systems have been proposed including Mo cladding, SiC cladding, UxSix fuel, and others [6].

Neutronic studies about SiC reinforced with Hi-Nicalon type S fibers (SiC HNS) as nuclear fuel cladding [6, 7, 8, 9, 10] showed that it would be possible to replace Zircaloy for SiC HNS without loss of neutronic performance. In addition, a preliminary TH study of SiC HNS [8, 9] and FeCrAl (KANYHAL) [11, 12] as nuclear fuel cladding was carried out with good results.

The aim of this work is to advance in the TH study of the SiC HNS and FeCrAl (KANTHAL) claddings as well as the fuel $U_3Si_2$, through comparison of thermal response between these proposed materials and the conventional $UO_2$/Zircaloy 4 materials, at steady-state condition. Firstly, a verification of Angra 2 model to RELAP5 MOD3.3 code was presented. Next, a study of reactor thermal response when the proposed nuclear fuel and fuel claddings are used was carried out. In this study, the $U_3Si_2$ was used in the simulations replacing the $UO_2$ nuclear fuel. The compounds based in uranium-silicon have been investigated and utilized as nuclear reactor fuels. The high $U_3Si_2$ uranium density compared with uranium dioxide ($UO_2$) density has made the first attractive to ATF in LWR fuels research [13]. Also the thermophysical properties of these high uranium content silicides ($U_3Si_2$ and $U_3Si$) make them an attractive material from both an economic and safety perspective as a replacement for $UO_2$ [14]. A main advantage of $U_3Si_2$ over $UO_2$ fuel is the higher thermal conductivity. Thermal conductivity of $UO_2$ is ~2-5 W/m/K whereas the thermal conductivity of $U_3Si_2$ is ~15-30 W/m/K. This large difference results in lower fuel centerline temperatures and lower temperature gradients in the fuel pellet. The high uranium density of $U_3Si_2$ (11.3 g-U/cm3 versus 9.7 g-U/cm3 for $UO_2$) is economically attractive since it may enable higher burnup and longer cycle length [15].

## 2. Methodology

The Pressurizer Water Reactor (PWR) Angra 2 is located in Rio de Janeiro, Brazil, and it is the reactor considered in this work. It has generating capacity of 1350 MWe and thermal power of 3765 MWth. The reactor was modeled in RELAP5-3D code [16], developed in previous work [18], and has been adapted to the

RELAP5 MOD3.3 code [17]. The core of the previous nodalization was completely replaced by a RELAP5 previous core nodalization [23, 24]. In this new nodalization the radial power distribution for BOL e BOC conditions given by the Angra 2 Final Safety Analysis Report (FSAR) [22] was considered. Then, 189 hydraulic components and 41 heat structures were used. The reactor core was modeled using 37 thermal-hydraulic channels (THC) each one with 34 axial volumes. The 193 fuel assemblies were grouped into 37 heat structures (HS) components and associated with 37 corresponding THC. A not heated channel represents the bypass. In the Figure 1 is represented the 37 TH regions. The small numbers inside each region represent the radial power factor of each core fuel assembly. The thermal-hydraulic validation of a RELAP5 nodalization implicates that the model reproduces the measured steady-state conditions of the system with acceptable margins [19].

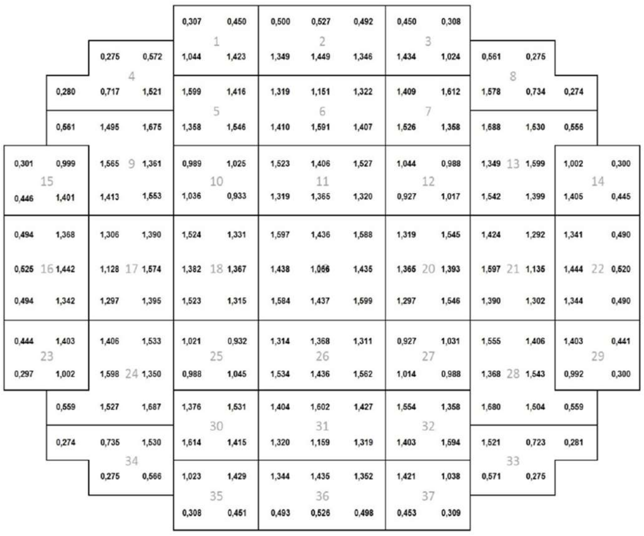


Figure 1: Thermal hydraulic regions modeled in RELAP5 MOD3.3 according to Angra 2 core.

## 3. Results and Discussion

The RELAP5-3D and RELAP5 MOD3.3 steady-state calculation were performed considering the reactor operating at full power, begin of life (BOL), cycle 01. The calculated TH parameters were compared with the plant nominal values [22] as it is shown in Table I. The obtained errors are into the range of the maximum acceptable error suggested by the RELAP5 users.

Table I: Comparison between reference and calculated values.

| Parameter | Reference Values [22] | Calculated Values RELAP5-3D – 10 THC | Suggests**/ Error (%) * | Calculated Values RELAP5 – 37 T HC | Suggests**/ Error (%) * |
|---|---|---|---|---|---|
| Coolant | Temperature(K) | | | | |
| RPV inlet | 565.25 | 566.45 | 0.5 /0.39 | 562.51 | 0.5/0.49 |
| RPV outlet | 599.25 | 601.60 | 0.5 /0.43 | 599.62 | 0.5/0.12 |
| Core inlet | 564.45 | 566.70 | 0.5 /0.39 | 562.52 | 0.5/0.35 |
| Core outlet | 601.25 | 601.90 | 0.5/0.11 | 601.64 | 0.5/0.07 |
| Mass Flow | (kg/s) | | | | |
| RPV | 18800 | 18702.85 | 2.0/0.52 | 18948.95 | 2.0/0.80 |
| Eff. in core | 17672 | 16933.37 | 10/4.18 | 17841.60 | 10/0.95 |

*Error = 100 $x$ (Calculated - Experimental)/Experimental
**List or requirements for the steady-state qualification of a nodalization [18].

The next steps were to consider the reactor TH studies using FeCrAl (KANTHAL) and SiC HNS as fuel cladding material. Simulations using $UO_2$ and o $U_3Si_2$ as fuel were also carried out. The Zircaloy 4 thermal properties were substituted in the nodalization by SiC HNS and then FeCrAl alloy properties to both fuels under study. The differences between the thermal properties of the Zircaloy-4, SiC HNS and FeCrAl (KANTHAL) influence the heat transfer from fuel to water. The fuel temperature when using FeCrAl as cladding showed no significant difference when Zircaloy 4 was used; however, the fuel temperature when using the SiC HNS cladding presented substantially lower. The thermo-physical properties as volumetric heat capacity and thermal conductivity, for claddings, have been provided in [20] for the Zircaloy-4, and in [21] for the SiC HNS and FeCrAl (KANTHAL) and the fuel$U_3Si_2$.

The calculated center fuel temperatures to the heat structure 614 (peripheral) and 618 (center) at 18$^{th}$ axial are shown in Table II. As expected, at same power, the cladding outer face temperature and the coolant temperature to the three simulated cladding types present the same value. The calculated coolant temperature at outlet of the 14 and 18 THC are 599.62 K and 601.18 K, respectively.

Table II: Comparison between RELAP5 MOD3.3 calculated values of center fuel temperatures at the central (18) and peripheral (14) heat structures on 18$^{th}$ axial level.

| Fuel Material | Heat Structure | Center Fuel Temperature (K) | | | | |
|---|---|---|---|---|---|---|
| $UO_2$ | | Zircaloy 4 | SiC HNS | FeCrAl | Cladding | Coolant |
| | 18 | 2101.75 | 2038.62 | 2100.37 | 629.53 | 590.96 |
| | 14 | 1369.16 | 1330.96 | 1368.8018 | 608.35 | 589.43 |
| $U_3Si_2$ | | Zircaloy 4 | SiC HNS | FeCrAl | Cladding | Coolant |
| | 18 | 1150.70 | 1113.51 | 1149.83 | 629.53 | 590.96 |
| | 14 | 938.13 | 914.23 | 937.91 | 608.35 | 589.43 |

## 4. Conclusions

The aim of this work was to study the thermal response of nuclear reactors when ATF fuel and fuel claddings are used. Neutronic studies of SiC HNS and FeCrAl alloys used as nuclear fuel cladding indicated reasonable behaviour and encouraged studies continuity. A previous TH nodalization of the nuclear reactor Angra 2 to RELAP5-3D code was adapted to RELAP5 MOD3.3 code; the initial reactor core was replaced and then the new nodalization was verified. The obtained results in steady-state condition were compared with FSAR data and were presented. They are in a good agreement.

After this stage, the next step was to simulate the reactor response when new fuel and fuel claddings are used. The SiC HNS and FeCrAl (KANTHAL) claddings were considered in the place of the traditional Zircaloy 4. The obtained results have shown that the substitution of Zircalloy 4 by FeCrAl (KANTHAL) or S iC HNS alloys presented respectively similar and better thermal response.

The $U_3Si_2$ density compared with uranium dioxide ($UO_2$) density, among others properties has made the $U_3Si_2$ attractive to accident-tolerant LWR fuels research. So in another simulations $UO_2$ fuel has been replaced by $U_3Si_2$ fuel. The obtained results showed lower temperatures at fuel center than the temperatures obtained through calculations using $UO_2$ as nuclear fuel.

## Acknowledgements

The authors are grateful to the *Coordenação de Aperfeiçoamento de Pessoal de Nível Superior* (CAPES), *Fundação de Amparo à Pesquisa do Estado de Minas Gerais* (FAPEMIG), *Conselho Nacional de Desenvolvimento Científico e Tecnológico* (CNPq) and *Comissão Nacional de Energia Nuclear* (CNEN) for the support.

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