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IMPORTANCE OF UNCERTAINTY MODELLING FOR NUCLEAR SAFETY ANALYSIS

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ABSTRACT

The U.S. Nuclear Regulatory Commission (NRC) reviewed the 10CFR50.46c regulations regard the loss-of-coolant-accident (LOCA), and emergency core cooling system (ECCS). In this planned rulemaking named as 10CFR50.46c. New LOCA criteria included the integration of models used to the hydrogen uptake changes equivalent cladding react (ECR), coupled with peak cladding temperature (PCT). This rule inserts the embrittlement mechanism considering the hydrogen buildup as a pre-transient condition, reducing a loss of operational margin. 10CFR50.46c criteria should combine the effects produced from different fields, such as neutronic analysis, thermal-hydraulic, with fuel performance codes. Besides, it should contemplate Best-Estimate Plus Uncertainty (BEPU) practices. Consequently, increases the challenges to safety analysis because of nuclear power plants run for extended periods than planned initially. In these circumstances, nuclear units need to operate on extended life cycles based on safety margins. With a lifespan of 60 years or more, we reviewed the behavior of the structural material on accident scenarios. This work showed the importance of uncertainties created by physical models such as the fission gas release, thermal conductivity, and loss of ductility caused by hydrides.

INTRODUCTION

The advanced frameworks provide wide horizons to safety analysis based on uncertainty quantification (UQ), and sensitivity analysis (SA). The nuclear simulation could combine several knowledge fields with improving reactor safety analysis. In this study, we pretended to explore state-of-the-art regarding uncertainty treatment produced by several concurrent concepts that can help establish realistic safety margins for transient conditions. Recently, a few Multiphysics systems comprise a means of advancing simulations of nuclear plants. Today, regulatory agencies have faced the problem for the aging of the nuclear plant many reactors operating for over 30 years.

Then the U.S. Department of Energy (DOE) started programs like the light water reactor sustainability (LWRS) plan. The LWRS pretend to extend the operational lifetime power reactor beyond 60 years. In these circumstances were necessary to create an in-depth description and quantification of safety margin to can operate the LWRs for a long-term lifetime over 30 years. Besides, other plans also started like Risk-Informed Safety Margin Characterization (RISMC) [1] The RISMC are strategies, which can determine the risk associated with life extension coupled with nuclear power additions [2]. Also, it introduced the Reactor Analysis and Virtual control Environment (RAVEN) a framework able to measure probabilistic risk assessment. Following occurs the introduction of an integrated system or only a single system divided in a fee modulus able to realize all calculation with lesser uncertainties spread [3-4].

Now had the Idaho National Labs (INL) develop the LOCA Toolkit for US light water reactors (LOTUS) [5]. Combinations of tasks reduce spreader uncertainties from the neutron kinetics, thermal-hydraulic with the fuel performance codes and increase the efficiency. Fuel codes calculate at peak power the fuel centerline temperature, and the gap conductance. The system permits an easy analysis produced from multiple nuclear codes.

1.1. Review of Conservative Rules

In 1970 years, reactor-grade of zirconium showed breakaway oxidation at temperatures around 350 °C. The breakaway is one of the most prominent features of the kinetic oxidation of zirconium. However, zirconium alloys exhibit a similar weight gain rate for temperature ranged of 1100-to 1500 °C. In 1973, suggested in Appendix K of 10CFR50.46 specified Baker-Just correlation for calculation of heating rate, hydrogen generation, and Effective Cladding Reacted (ECR) [6]. Experiments performed with zircaloys showed the embrittlement of cladding around half temperatures of melting point. At high temperature the zircaloy suffers transitions of the alpha-to-beta, speeding up the embrittlement process. Also, are empirical rules the oxidation limits of 17% and the temperature of 1204 °C.

The peak cladding temperature (PCT) of 1204 °C extensively related arose in connection with the 17% maximum cladding oxidation value. Initially, empirical rules used to PCT had similar limits, such as Westinghouse, Combustion Engineering agreed on 1371 °C, and Babcock and Wilcox advised a more conservative 1315.6 °C as the peak calculated the temperature. Besides, guideline §50.46 recommended that in the ruptured region may involve the calculation of double-sided oxidation, and the amount of oxidation defined as the 17%. The embrittlement Criteria used to zircaloys specify a fixed temperature coupled with oxidation limits referred in CFR §50.46. Rulemaking agreeing on a ductility criterion that could apply to any cladding alloy at any burnup level. During 1980 and 1990 decades, Nuclear Power Plant (NPP) licensing, accident transient analysis focuses on Large Break Loss Of Coolant Accidents (LB-LOCA). In the 1990 decade, focused on the Basis Accident (BDBA) scenarios, including the long-term loss of power [6]. Following this, it raises designs describing the passive cooling system, improved interactions between components, compact design, and different fluids used to primary loop coolants. After the Fukushima accident, 2001 focused on more tolerant materials to replace zirconium-based alloys. In 1988 United States Nuclear Regulatory Commission U.S. NRC approved the revised rule for ECCS criteria as results including uncertainty analysis what is by following the regulatory guide, RG 1.157 published on September 16, 1988.

1.2. Best Estimate Models

During the period between 1974 and 1988, regulatory agencies sponsored a groundwork enough to create a more realistic safety analysis method. Figure 1 shows a rapid comparison between safety margin representation for BEPU models and current requirements form 10 CFR§50.46c [7].

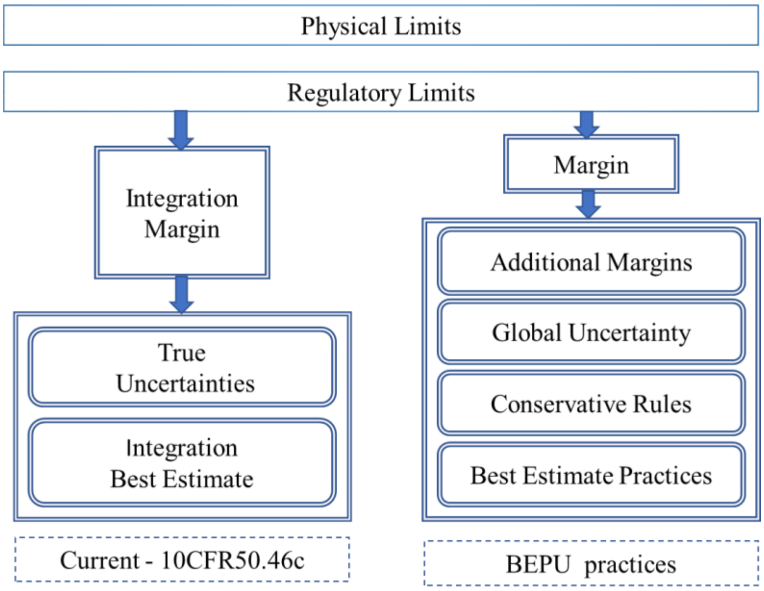

Figure 1: Safety margin representation for BEPU models and 10CRF§50.46c

It can be observed that a load distribution that represents input uncertainties shows a wide range of variability from input parameters. According to nonparametric formulations, it must obey the limit criteria to (95/95). Figure 2 shows a representation of safety margins.

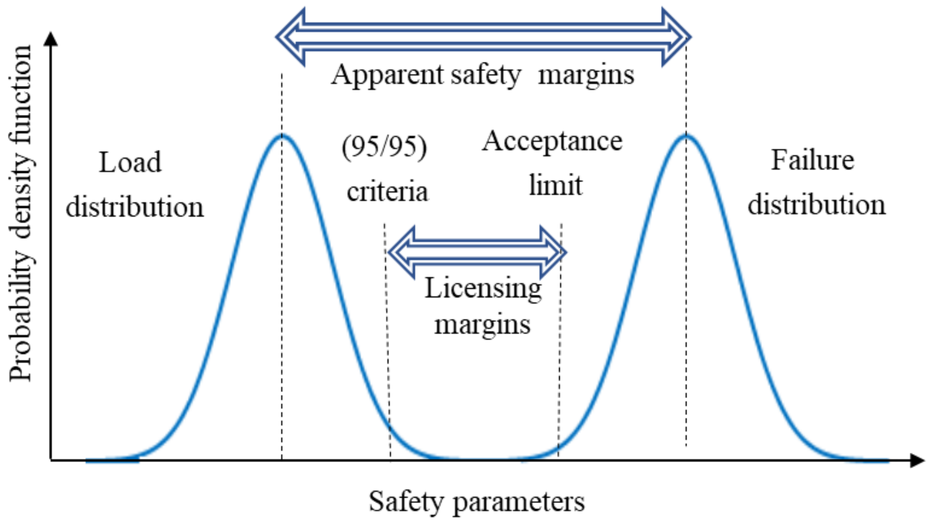

Figure 2: Statistical representation of nuclear safety margins

These limits forced by samples help to define the probability level of 95%, covered with a confidence level of 95%, which represents fewer uncertainties than the acceptance criterion. The BEPU models spread for all THC systems and fuel performance codes (FPC). In 2005, a significant limitation showing by the deterministic approach was unrealistic safety margins. Best Estimate Plus Uncertainty (BEPU) methods may provide more realistic safety margins working with statistical methods. BEPU procedures evolved into the Evaluation Model Development and Assessment Process (EMDAP), used on Regulatory Guide 1.203. The standard methods used to BEPU need to follow several steps such as uncertainty quantification (UQ) process, at least one hundred simulation using licensing codes and producing sensitivity analysis (SA).

However, BEPU focused on the small-break LOCA (SBLOCA) providing to the audit process. Then, arisen other approaches like Best Estimate Methods – Uncertainty, and Sensitivity Evaluation (BEMUSE). The BEMUSE method used LOCA benchmarks to test. The motivation re features of nonlinear systems of core reactor and multi model's interaction using in part deterministic rules and probabilistic tools.

The efforts to produce uncertainty quantifications resulting in a cooperative contribution the existing code scaling, applicability, and uncertainty (CSAU) method, 1996 [8]. The CSAU practices used two-loop pressurized water reactor for the analysis and occurred a rapid proliferation of BEPU method. The safety analysis based on BEPU obeys all uncertainty need of uncertainty quantification and statistic order to define sample sizes. BEPU method has induced the CSAU evaluation method in the last two decades. Therefore, the source of this investigation is the rule 10CFR50.46c established in 2014. The nuclear community provided many comments about the requirements incorporated the cladding embrittlement criteria.

1.2 Statistical Analysis

Techniques used to Sensitivity Analysis (SA) govern the sensitivity of output to each of the pertinent input uncertainties. Also, SA may explain the physical features of the input and output relation such as linearity, coupled interactions between inputs. Figure 3 displays a block diagram of sensitivity analysis used for integrations between licensing codes.

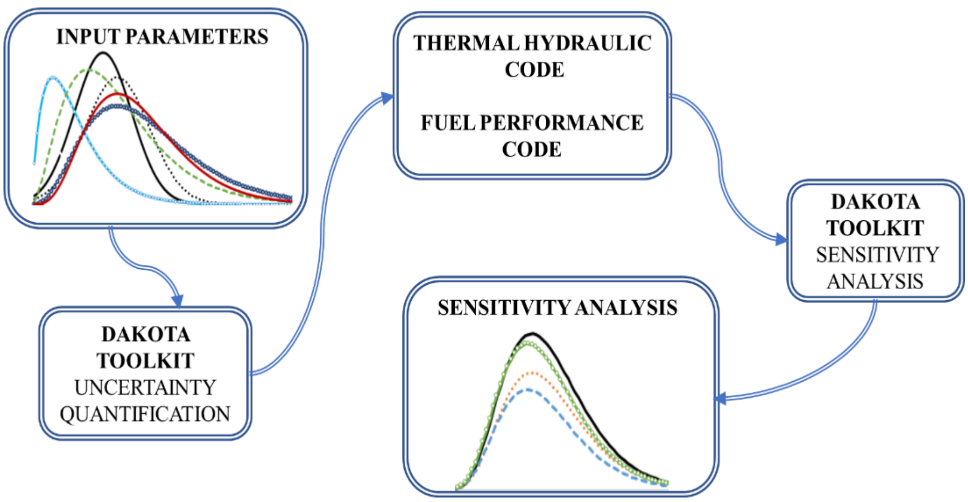

Figure 3: Sensitivity analysis models incorporated into THC and FPC.

Methods of high fidelity appoint to a specific output, with a series of input data are essentials, while impactful of each information can contribute to reported models are a top priority. The assumptions used must identify the nonparametric models, which can apply to free distributions to express uncertainties. Then it can conclude that the concepts recommended to 10CFR§50.46c should offer potential solutions to LOCA/ECCS analysis. Since 2014, occurs a rapid spread of concepts proposed on the risk-informed safety margin or RISMC. The safety margin pathways offer a framework that can support the reactor supplies introducing the Multiphysics system LOTUS. The licensing code uses BEPU to measure the uncertainty treatment, statistical tools promoting uncertainty, and sensitivity analysis. The evolution of BEPU practices, coupled with thermal-hydraulic codes (THC).

1.3. LOTUS Framework

The framework LOTUS still being developed and will permit an in-depth investigation to the LB-LOCA cases, also aiming at supporting compliance with the 10CFR§50.46c coupled with Emergency Core Cooling System (ECCS) performance during LOCA. We produced simulations of PWR under quasi-steady-state conditions, else during accident scenarios such as loss of coolant accident (LOCA). Safety analyses, including system analysis, fuel performance, and neutron analyses. The SA module focuses on coolant systems investigating thermal-hydraulics behavior. The fuel performance (FP) modulus analyzes the physical responses such as fuel centerline temperature, cladding deformation, radiation effects, and fission gas release (FGR) of the fuel rods. Neutronics deals with the generation, absorption, and scattering of neutron fluxes within of core reactor. The LOTUS framework concept contains uncertainty quantification (UQ) and sensitivity analysis (SA). Risk assessment (RA) and core design automation (CD-A). The significant advantage of LOTUS environment, where several modules can be called upon like a traditional subroutine. This framework concept is superior to the time-intensive, error-prone method of manually creating and changing input files. Figure 4 illustrates the LOTUS framework concept.

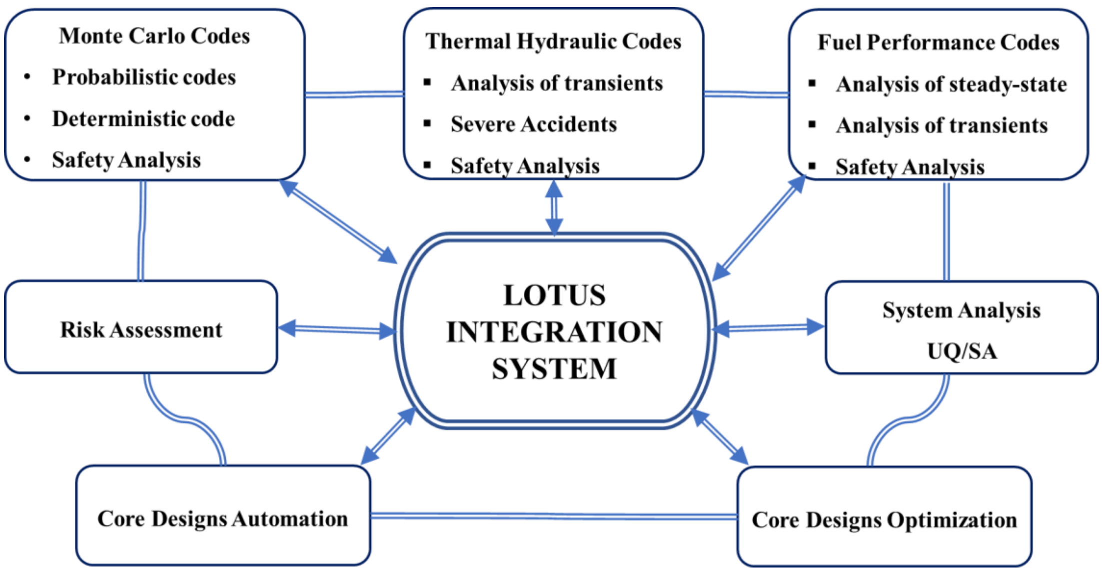

Figure 4: Lotus integration system

The neutron-physics analyses codes used to PWRs are widely knowledgeable in the open literature having many similar capacities. In the last fifth years arisen many systems codes have been able to simulate core reactors. The most popular is the Monte Carlo N-Particle Transport Code (MCNP). The Serpent code shows several concurrent activities with MCNP. The suite for nuclear safety analysis and design supported by Oak Ridge National Laboratory SCALE helps to review fuel response during the burn cycle. Today, several Multiphysics systems that show powerful version containing fully capacities and able to interface with other systems. Idaho National Laboratory (INL) created a core analysis tool integrated with LOTUS framework named PHYSICS [9-11].

Over the years, it observed that accumulated several thermal-hydraulic systems to foresee of the reactor core behavior, analyzing the primary loop under steady-state and operational transient. Thermal-hydraulic analyses use system codes like reactor excursion, and leak analysis program (RELAP) can investigate steady-state and transient response. The RELAP system applies diverse ways for homogenization, fluid dynamics, and heat conductions coupled with cladding oxidation models. Early, thermal-hydraulic codes do not show statistic tools integrating but after of BEPU models occur sturdy interfacing with uncertainty quantification and sensitivity analysis models.

The transition to the proposed 10 CFR§50.46c should offer potential solutions to LOCA/ECCS analysis. In 2015, the Risk-Informed Safety Margin Characterization (RISMC) Pathway, initialed a set of demonstration activities to support the industry introducing LOTUS framework. The LOTUS framework shows integrated functions from RELAP and FRAPCON contained on fuel clad performance (FCP), form FRAPCON, core design optimization (CD-O) and automation (CD-A), also risk assessment modulus containing BEPU tools.

2. MATERIALS AND METHODS

2.1. Evaluation Model

The aim of the 10 CFR 50.46c, is to ensure adequate core cool ability under LOCA conditions. Section §50.46 is a regulation that addresses the rules to distinguish ECCS requirements defined to cool function during and after LOCA scenarios [12]. The NRC suggests a similar cladding reacted (ECR) criterion, which depends on the hydrogen content of the pre-transient phase. Including features comprised high burnup on cladding performance considering complementary phenomena and technical issues. Also, these issues may allow an alternative for a safe treatment for debris effects on long-term core cooling. Research programs must investigate the effects produced for long irradiation cycles and fuel high burnup levels.

In 10CFR50.46c does not change related to long-term cooling requirements. Related to zirconium-based alloys, followed the guidelines to determine the analytical limits on peak cladding temperature, also the integral time and at a temperature that corresponds to the assessed ductile-to-brittle transition. The 10 CFR 50;46c induce the integration to perform a Wilks formulation used to UQ on the ratios of equivalent cladding reacted (ECR) and peak cladding temperature (PCT) to their corresponding cladding hydrogen content-based limits.

2.2. FRAPCON Generalized Uncertainty Models

Nuclear fuel performance codes use several empirical models to describe complex thermal and mechanical phenomena. The reliability of their predictions suffers from the effects of uncertainty in parameters used in the models. There are several difficulties in modeling uncertainty because the variables of physical models are often impossible to measure. Interpreting uncertainty translated into no inherent physical observations but depended on expert judgment to confirm. The fuel licensing code, FRAPCON, contains eight models for uncertainty calculations; including fuel thermal conductivity, thermal fuel expansion, fuel swelling, fuel fission gas release, cladding creep, cladding growth, cladding corrosion, and cladding hydrogen uptake [13-16]. Four of the uncertainty models relate to the properties of the uranium oxide fuel material.

The models used to calculate fuel thermal conductivity is more critical to safety analysis. Thermal conductivity is a fundamental material property that describes a material's ability to conduct heat. Equation 1 displays the thermal conductivity model for UO_2.

$$k_{95} = \frac{1}{A + a.gad + BT + f(Bu) + (1 - 0.9\exp(-0.04Bu))g(Bu)h(T)} + \frac{E}{T^2}\exp\left(-\frac{F}{T}\right) \qquad (1)$$

where $k95$ is thermal conductivity of 95% *TD* fuel (W/m-K); *T* is temperature (K); *Bu* is burnup (GWd/MTU); *f(Bu)* is the effect of fission products in a crystal matrix; *g(Bu)* is the effect of irradiation effects; *h(T)* is the temperature dependence of annealing on irradiation defects; *Q* is the temperature dependence parameter; *A* is 0.0452 (m-K/W); *a* is 1.1599; *gad* is a weight fraction of gadolinia; *B* is 2.46e-04 (m-K/W/K); *E* is 3.5e+09 (W-K/m), and *F* is 16361 (K).
Fuel porosity, estimated for fresh fuel at 5%, increases with the fission process and consequently decreases thermal conductivity. The statistical distribution of porosity shows reduced dependence with fuel centerline temperature. Several mechanisms involve the FGR model with slight variability in fission gas rates that shows dependency. Next, the models used for fission also insert uncertainties as diffusion coefficients because of the fragmentation of fuel following relocation. There are many factors creating uncertainties in fuel temperatures, coupled with internal models.

Thermal expansion has a dependence on temperature and fuel density. The fuel density has a dependence on the lattice parameter of uranium dioxide, considered 5.47127 ± 0.00008 nm at 20°C. The thermal expansion model used to predict the fuel pellet behavior reported significant fuel geometry changes produced by high-temperature thermal stresses and fuel dilatation. The bias of thermal fuel expansion is around 15%, and cladding exhibits a bias of 30% for diametral thermal expansion. Equation 2 expresses a thermal expansion model for temperature-related strain in UO_2 fuel, given in NUREG/CR-7024.

$$\frac{\Delta L}{L_0} = K_1 T - K_2 + K_3 \exp\left(-\frac{E_D}{kT}\right) \qquad (2)$$

where $\Delta L/L_0$ represents the linear strain caused by thermal expansion (equal to zero at 300K);

T is temperature (K); ED is the energy of formation of a defect equal to 1.32e-19 J; k is Boltzmann's constant equal 1.38e-23 (J/K); K_1 is 9.8e-06 K; K_2 is 2.61e-03, and K_3 is 0.316.

Uncertainties in the FGR model in FRAPCON result from the bias applied to the gas diffusion coefficient. At the temperatures experienced by the fuel in the pin cell problem, even where other uncertain variables affect temperature, it does not activate thermal gas diffusion through this mechanism. Equation 3 describes the Massih model used in FRAPCON.

$$\frac{dC}{dt} = D(t)\Delta_r C(r,t) + \beta(t) \tag{3}$$

where $C(r,t)$ represent gas concentration; $\beta(t)$ is gas production (assumed uniform within grain); Δr is a Laplacian operator in spherical geometry; $D(t)$ is the diffusion coefficient.

Both models used for fuel swelling and gaseous fission products are results of semiempirical models, which show dependence with temperature and burnup level. There is equation that can describe a practical method to calculate fuel swelling. Equation 4 expresses the empirical relationship used in the fuel swelling model in FRAPCON, valid for burnup less than 80 GWd/MTU, while Equation 5 expresses the swelling model for burnup over the limit of 80 GWd/MTU.

$$soldsw = 2.315\times10^{-23} bus(1+0.08 sigsw) \tag{4}$$

$$soldsw = 3.211\times10^{-23} bus(1+0.16 sigsw) \tag{5}$$

where $soldsw$ represents the fractional volume change because of solid fission products (m^3/m^3); *sigsw* represent the incrementally changes error; *bus* express the fuel burnup during a time step; *bu* is burnup at the end (MWs/kgU); bu_l is burnup at the end at the time-step (MWs/kgU);

FRAPCON code shows two types of creep rates: the first type is creeping correlations from effects caused by fast neutron flux, also called irradiation-induced creep; while, the second type is a thermal creep model. Equation 6 describes the thermal creep model. Thermal creep shows two complementary models. The first model includes both primary and secondary creep, while the second method describes only secondary creep. However, there is a vital impact on the mechanical properties of zirconium-based alloys, which is embrittlement caused by hydrogen release. Equation 7 expresses the formulation used in irradiation-induced creep.

$$\varepsilon_{th} = A\frac{E}{T}\left(\sinh\frac{a_i \sigma_{eff}}{E}\right)^n \exp\left(-\frac{Q}{RT}\right) \tag{6}$$

$$\varepsilon_{irr} = C_0 \phi^{C_1} \sigma_{eff}^{C_2} f(T) \tag{7}$$

An uncertainty model of axial cladding growth results from irradiation. However, growth is not an important mechanism that changes the fuel rod performance for the cladding material used. The uncertainty model of cladding hydrogen pickup expresses the irradiation effects of hydrogen accumulation that contributes to cladding embrittlement.

2.3. Wilks Non-Parametric Method

The preliminary work proposed for the Wilks presented in 1941 is a method that finds the minimum sample size required to determine tolerance limits. Wilks formulation finds estimate the 95th percentile of a single output, and it helps first to express the probability β that at least one instance of sample size N is above an upper limit below which portion γ of the population resides. Equation 8 shows the method of Wilks. The nuclear industry considers the formulation created by Wilks as a method based on order statistic, which helps in determining the minimum number of samples necessary to reach (β=95%) of confidence, and (α=95%) of probability. Equation 9 shows the relation between parameters used in Wilks formulation. It used distribution functions are used to represent the input parameters created by the DAKOTA toolkit using the Latin Hypercube Sampling (LHS) method [17-20]. The LHS is a method to build robust random sampling. Equation 10 expresses the generalization of the method proposed by Wilks.

$$\sum_{j=0}^{n-k} \binom{n}{j} \alpha^{j} (1-\alpha)^{(n-j)} \geq \beta \tag{8 }$$

$$1-\alpha^{n} \geq \beta \tag{9}$$

$$1-\alpha^{n} - n(1-\alpha)\alpha^{(n-1)} \geq \beta \tag{10}$$

where α is the quantile level, β is the confidence level for the upper bound of the quantile, and N is the number of simulations.

2.4. Uncertainty Quantification

UQ comprises perturbing the inputs of a code, executing the code for each perturbation, and analyzing the resulting sample of outputs. The most popular measurement used to UQ is the confidence interval so-called 95/95, which is a set of bounds that comprise 95% percentile of the population with 95% of confidence level or certainty. It is not reasonable to perturb all inputs for most codes. Early, quantification procedures use a Phenomena Identification, and Ranking Table (PIRT) included all related inputs with their corresponding range.

2.5. Sensitivity Analysis

Sensitivity analysis helps to explain the relation between input and output uncertainty from a system of interest. SA methods comprise practices able to filter the most impactful contributors from a list of potential candidates.

There are at least two types of SA, the local analysis, and global approach. Restrict to a single model is the SA local like the one-at-a-time (OAT) method and global sensitivity analysis (GSA) method. There are various practices to test sensitivity measures, such as a Pearson product-momentum coefficient (PMMC), Spearman rank correlation (SRC), and variance analyses. The assumptions used THC and FPC identified as nonparametric models, which apply to free distributions. Nonparametric cases must use rank correlations between the related variables. The purpose of the SA is to investigate the connection between the inputs and responses. By performing the SA applied to fuel codes, it is possible to detect priority input parameters with higher relevance.

2.6. Analysis of Physical Models

Critical parameters used on guidelines comprise PCT and ECR in testing fuel safety under LOCA scenarios. PCT is the highest cladding temperature permitted for transient circumstances. Extreme PCT leads to the cladding failure coupled with higher cladding oxidation. The ECR is another parameter investigated that comprises the maximum limit of corrosion on coating. Cladding oxidation is an exothermic reaction that increases the oxidation rate for higher temperatures. The oxidation kinetics at temperatures over 1000 K show the potential to generate vast amounts of hydrogen that lead to blast scenarios.

Using FRAPCON code developed a sensitivity study on gap conductance can perform for a pin at core-average conditions. Also, it had boundary conditions such as inlet mass flow rate and temperature and axial power distributions representative of a PWR during a 60 GWd/MTU depletion. FRAPCON analysis using finite axial nodes distributed along the rod length, and when to occur the gap closure has a higher gap conductance.

On the beginning burn cycle, the gap is open between the fuel and cladding for all nodes, and the thermal conductivity refers to helium into the gap. The gap conductance is 5000 W/m^2-K. However, average fuel temperatures reach the order of 1000 °C, and fuel pellets suffer thermal expansion. Also, occur the fuel swelling that shows a dependence with burnup level and begin fuel fragmentation and relocation. Then occur the gap closure, and gap conductance reaches around 200,000 W/m^2-K.

At the temperature of 1204 °C, the Baker-Just correlation over-predicts weight gain and zirconium by as much as 30%. In 1989, Regulatory Guide 1.157 allowed the use of a best-estimate correlation using Cathcart-Pawel correlations for temperatures higher than 1078 °C. Equation 11 exhibits the corrosion rate of the zirconium-based alloys. The Arrhenius equivalence is accurate and straightforward, a formulation for the temperature dependence of a chemical reaction. Equation 12 expresses the Arrhenius formulation used to oxidation shows the temperature dependence of the chemical reaction.

$$k = Ae^{-Ea/RT} \tag{11}$$

.where k represents rate constant in (g^2/cm^{4}s), T is the temperature in Kelvin, Ea is the activation energy for the reaction in (J/mole), R is universal gas constant in (J/mole-K), and A is constant for specific reactions.

$$w = \sqrt{Ate^{-Ea/RT}} \tag{12}$$

where the weight gain of the samples (ω) in high-temperature steam in (g/cm^2)

3. SIMULATION UNDER NORMAL CONDITIONS

The experiments performed was choice comprises fuel rod CIP0-1 with widely documentations Irradiated in the Vandellos reactor up to 74.8 GWd/MTU. Figure 5 illustrates CIP0-1 under normal irradiation cycle.

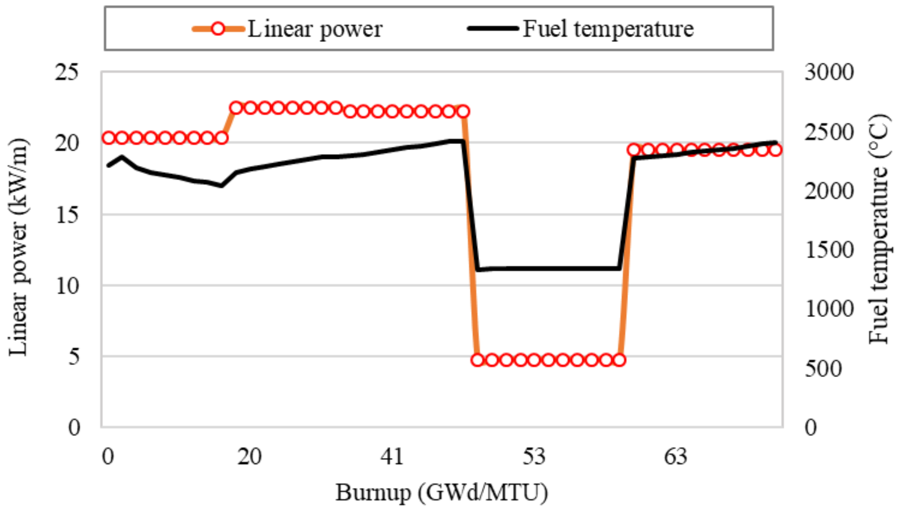

Figure 5: Linear power and average fuel temperature of CIP0-1.

On test occurred an important zirconia layer of 75 µm average also showed significant axial azimuthal variations. For steady-state simulations, use finite difference methods to solve the heat conduction equation in the radial direction. The power profile used in FRAPCON allows change linear heat rates in the axial direction. Table displays manufacture tolerances of rod fuel.

Table 1: Engineering tolerances of fuel rod test CIP0-1.

	mean	Upper bound	Lower Bound	Input
Cladding outside diameter (mm)	9.5	9.6425	9.3575	dco
Cladding thickness (mm)	0.571	0.579565	0.562435	thkcld
Fuel mean roughness (mm)	0.002	0.00203	0.00197	roughc
Cladding mean roughness (mm)	0.005	0.005075	0.004925	roughf
Fuel pellet length (mm)	9.83	9.97745	9.68255	hplt
Fuel pellet dish depth (mm)	0.239	0.242585	0.235415	dish
Fuel pellet true density(%)	95.4	96.831	93.969	Td
Enrichment U-235 (%)	4.5	4.5675	4.4325	enrichment
Plenum length (mm)	36.5	37.0475	35.9525	cpl
Plenum spring diameter (mm)	8.19	8.31285	8.06715	dspg
Plenum spring wire diameter (mm)	1	1.015	0.985	dspgw
Fuel rod pitch (mm)	12.3	12.4845	12.1155	pithc

The uncertainties used came from physical models, boundary conditions, and manufacture tolerances. The simulation carried out used FRAPCON to execute 59 run-codes, according to the method proposed by Wilks. Latin hypercube sampling creates a Gaussian distribution of input data. The SA process comprises the Pearson correlation between input parameters and fuel responses during normal operations.

Table 2: Input variables for uncertainty analysis of FRAPCON

Physical models	Parameter	Bias
Fuel thermal conductivity	sigftc	8.80%
Fuel thermal expansion	sigftex	10.30%
Fission gas release	sigfgr	100%
Fuel swelling	sigswell	0.08% - 0.16%
Cladding creep	sigcreep	14.50%
Cladding axial growth	siggro	20.30%
Cladding corrosion	sigcor	15.30%
Cladding hydrogen pickup	sigh2	94

Pearson coefficient is a statistical measure that can describe relationship, or association, between two continuous variables. Pearson with values over 0.30 appointed linear relations. Table 3 describe Pearson correlation between physical models' uncertainties and fuel response under steady-state.

Table 3: Pearson correlation between fuel response and physical models of FRAPCON

Fuel response	sigftc	sigftex	sigfgr	sigswell	sigcreep	siggro
Fuel stack axial extension	-0.15	0.05	0.15	0.81	-0.28	-0.39
Plenum gas temperature	-0.30	-0.31	-0.32	0.49	-0.31	-0.41
Plenum pressure	-0.67	-0.22	0.04	0.61	0.26	-0.05
Fuel average temperature	-0.95	-0.20	0.06	0.29	0.33	0.55
Fission gas release	-0.83	0.07	0.32	0.24	0.46	0.74
Cladding axial strain	-0.01	0.16	0.13	0.69	-0.27	-0.42
Cladding hoop strain	-0.47	-0.04	-0.09	0.81	-0.14	-0.15
Cladding radial strain	0.18	-0.03	-0.20	-0.83	0.33	0.20
Cladding permanent strain	-0.44	0.06	-0.23	0.59	-0.03	-0.06
Gas-gap pressure	-0.65	-0.23	0.04	0.61	0.26	-0.09
Fuel surface temperature	-0.24	-0.86	-0.49	-0.03	-0.44	-0.23
Fuel centre temperature	-0.94	-0.22	0.08	0.25	0.34	0.60
Energy fuel	-0.95	-0.20	0.07	0.28	0.33	0.56

3. CONCLUSIONS

The computational experiments appointed significant variations observed on fuel responses related to slight variability of input parameters. The method used to integrate the fuel code with the DAKOTA toolkit.

The aim of simulations consists of verified uncertainties regarding corrosion of zirconium-based alloys, amount of gaseous release, and cladding deformations. Table 2 showed uncertainties form physical models. The fuel rod represents steady-state conditions and directly combining the SA using the Pearson index. We built the statistical distributions used to implement the model using DAKOTA, based on the mean values and standard deviations of the variables. FRAPCON build testing and analysis. In similar research, FRAPCON was used to develop the SA with internal models for uncertainty propagation. The sampling was performed based on the effects due to the manufacturing tolerances combined with the boundary conditions. The manufacturing parameters were observed to have the most significant influence.

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