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SENSITIVITY ANALYSIS OF FUEL ROD PARAMETERS IN STEADY STATE CONDITION USING TRANSURANUS CODE

Amanda A. Aguiar[1], Alfredo Abe[1] and Cláudia Giovedi[2]
[1] Instituto de Pesquisas Energéticas e Nucleares (IPEN / CNEN - SP)
Av. Professor Lineu Prestes 2242
05508-000 São Paulo, SP
amanda.abati.aguiar@gmail.com.br
ayabe@ipen.br

[2]Analysis, Evaluation and Risk Manager Laboratory (Poli / USP)
Av. Prof. Mello Moraes, 2231
05508-000 São Paulo, SP, Brazil
claudia.giovedi@labrisco.usp.br

ABSTRACT

In this paper, a simulation of steady state conditions using TRANSURANUS code applied to Arkansas Nuclear One Unit 2 (PWR) fuel rod is presented. The fuel rod considered in this work was exposed to a peak rod average burnup of 64 GWd/TU, which corresponds to a batch-average exposure of about 53 GWd/TU. TRANSURANUS code offers two different approach for sensitivity analysis: Numerical Noise Analysis and Monte Carlo. In this work, sensitivity analysis using Monte Carlo approach was considered in the range of fuel rod manufacturing parameters, such as internal and external radius of the cladding, external radius of the fuel, and filling gas pressure of the fuel rod, in order to verify some existing correlation with fuel centerline temperature, internal cladding temperature, average tangential stress in the cladding, average permanent tangential strain in the cladding, internal pressure, and fission gas release.

1. INTRODUCTION

The use of fuel performance codes to simulate the fuel behavior in different irradiation conditions is an important tool to evaluate new fuel rod design and materials as part of the safety analysis.

TRANSURANUS [1] [2] [3] is a computer program for the thermal and mechanical analysis of fuel rods in nuclear reactor. The TRANSURANUS code consists of a clearly defined mechanical-mathematical framework into which physical models can easily be incorporated. The mechanical-mathematical concept consists of a superposition of a one-dimensional radial and axial description (the so called quasi two-dimensional or 1½ - D model). The code was specifically designed for the analysis of a whole rod.

TRANSURANUS code was utilized to perform sensitivity analysis, the code has capability to perform calculations under normal operating conditions, transients and accidents in the same simulation, as well as sensitivity analysis with deterministic and statistical approach [4].

Sensitivity analysis must be carried out to address the effect of design parameters and adopted models in the fuel rod performance. Sensitivity analysis applying Monte Carlo technique allowed statistical variations of a large number of input quantities to be simulated according to many types of input distributions (Normal, Uniform, Log-Normal, Cauchy) allowing user-defined lower and upper bounds of the input quantities to be set, the code input options cover the fuel rod geometry at beginning of life, all prescribed time-dependent quantities (e.g. linear heat rate and coolant or cladding outside temperatures) as well as all material properties (e.g. thermal conductivity, creep) that are applied in the code for fuel, cladding and coolant [5].

TRANSURANUS statistics postprocessor has estimated 5% / 95% percentiles of all simulated output quantities. It reflects the representation of the fuel rod as a stack of axially symmetric slices that vary in initial geometry, composition as well as time- dependent boundary conditions, and thus entails an analysis of the output's dependence on time and axial position [5].

This work aims to present the results obtained in the study of sensitivity analysis using the TRANSURANUS code [4] considering the Monte Carlo statistical approach for Nuclear Unit of ARKANSAS (ANO-2) reactor fuel. The following manufacturing parameters of the fuel rod were considered: fuel pellet diameter; internal and external cladding radius; and fuel rod filling gas pressure in order to evaluate the resulting effect on the analyzed output parameters.

2. METHODOLOGY

The manufacturing parameters distribution were considered random with normal distribution with average values as design data and deviation as the tolerance of the manufacturing processes usually considered in the fuel design of conventional western PWR reactors, as described in Table 1, according to the reference [6].

The values of the considered variable were randomly selected with a confidence interval of 95% (±2σ).

Table 1: Definition and description of fuel rod manufacturing parameters analyzed [6]

Input Variable Name*	Variable Description	Limitations and/or Variable Pattern	Unit	Nominal Value ± 2σ	Upper Value	Lower Value
RAB	fuel pellet radius	Input data	[mm]	4.13 ± 0.02	4.15	4.11
RIH	Inner cladding	Input data	[mm]	4.22 ± 0.08	4.30	4.14

Input Variable Name*	Variable Description	Limitations and/or Variable Pattern	Unit	Nominal Value ± 2σ	Upper Value	Lower Value
	radius					
RAH	Outer cladding radius	Input data	[mm]	4.85 ± 0.1	4.95	4.75
PI0EIN	Filling gas pressure of fuel rod	Input data	[MPa]	$2.62 \pm^{0}_{0.70}$	2.62	1.93

*TRANSURANUS input variable names.

For the sensitivity analysis, 200 data were assigned for each fuel rod manufacturing parameter according to adopted tolerance interval.

The TRANSURANUS code simulations was performed considering 200 different data and following parameters were analyzed: fuel centerline temperature, internal cladding temperature, tangential stress and strain in the cladding, internal pressure in the fuel rod and average fission gas release.

2.1 Test Case Description and Boundary Conditions

The sensitivity analysis performed in this study with the TRANSURANUS code used as test case the fuel rod of Arkansas Nuclear One Unit 2 (ANO-2) reactor located near Russellville, Arkansas in a irradiation cycle of 1,697 days or 4.6 years, which data are available in the open literature [6]. This reactor is a 2815 MWth Pressurized Water Reactor (PWR) owned and operated by Arkansas Power and Light. It is the first Combustion Engineering unit to utilize 16 X 16 fuel.

The configuration of the simulated uranium dioxide fuel pellet geometry contains concavity and shoulder with 3.48% enrichment level. The initial pressure of the fuel rod filling helium gas is 2.62 MPa at 25°C. The coolant pressure is 15.5 MPa. The inner temperature of the coolant is 290°C and the evaluation of the average neutron flux of the fuel rod as function of time is shown in the graph of Figure 1.

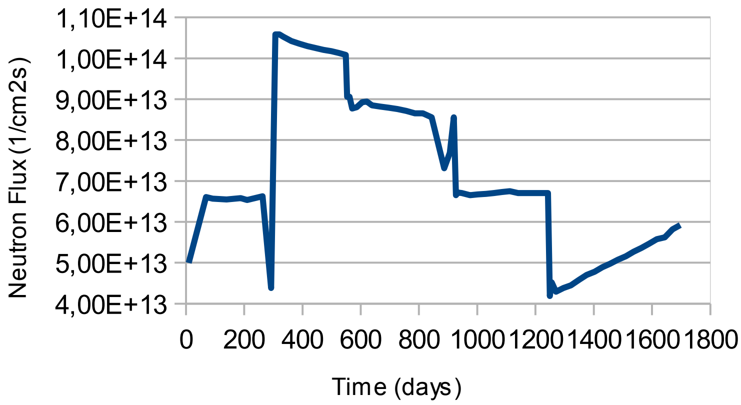

Figure 1: Neutron Flux versus Irradiation Time for the Fuel Rod Test Case

The fuel rod was axially sectioned in 12 parts along its active length and the power profiles in the beginning of life (BOL), middle of life (MOL), and end of life (EOL) were used as shown in the graphs of Figure 2. Its noticed that the profiles are uniform.

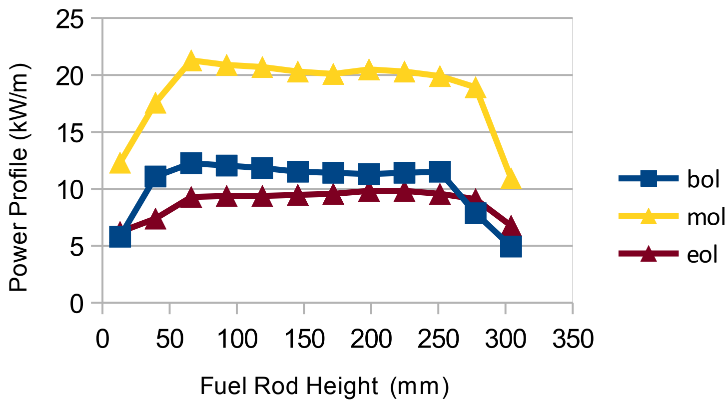

Figure 2: Power Profile (BOL, MOL, and EOL) versus Fuel Rod Height

3. RESULTS AND DISCUSSION OF THE SENSITIVITY ASSESSMENT

Table 2 presents the average, absolute deviation and percentage deviation (%) of fuel centerline temperature, internal cladding temperature, average tangential stress in the cladding, average permanent tangential strain in the cladding, internal pressure in the fuel rod, and fission gas release. These results were obtained by varying all fuel rod manufacturing parameters simultaneously: fuel pellet diameter; internal and external cladding radius and fuel rod filling gas pressure (RAB, RIH, RAH and PI0EIN).

The fuel centerline and inner cladding temperatures were less sensitive among others studied parameters, the tangential stress and strain, internal pressure, and fission gas release are more sensitive to fuel manufacturing parameters.

Table 3 shows the percentage deviation (%) of the output parameters (average tangential stress in the cladding, average permanent tangential strain in the cladding, internal pressure and fission gas release) that are more sensitive to variations in manufacturing tolerances, obtained from Table 2. These results were obtained considering only one fuel rod manufacturing parameter at the time, in order to verify their contribution on the selected output parameters.

It can be seen from Table 3, that the fuel rod manufacturing parameter inner radius of the fuel rod cladding (RIH) contributes considerably to the average tangential stress, average permanent tangential strain in the cladding, and fission gas release.

The outer fuel pellet radius (RAB) and the outer cladding radius (RAH) influence in the mean permanent tangential strain.

The fuel rod filling gas pressure (PI0EIN) influences the fission gas release.

Table 2: Sensitivity Analysis Results Obtained Using TRANSURANUS Code

Parameter	Average	Absolute Deviation	Percentage Deviation (%)
Fuel Centerline Temperature (°C)	714.1	15.9	2.22
Internal Cladding Temperature (°C)	309.7	1.89	0.6
Average Tangential Stress in the Cladding (MPa)	- 48.37	14.08	29
Average Permanent Tangential Strain in the Cladding (%)	- 0.71	0.39	55
Internal Pressure (MPa)	4.66	0.38	8.1
Fission Gas Release (%)	1.90	0.49	26

Table 3: Percentage Deviation (%) for each output parameter calculated by the TRANSURANUS code varying one fuel rod manufacturing parameter at the time

Output Parameter	Fuel Rod Manufacturing Parameters			
	(RAB) fuel peller radius	(RIH) Inner cladding radius	(RAH) Outer cladding radius	(PI0EIN) Filling gas pressure
Mean Tangential Stress in the Cladding	2.2%	31.3%	3%	4%
Mean Permanent Tangential Strain in the Cladding	10.8%	57%	15.3%	6%
Internal Pressure	0.21%	1.4%	0.5%	8.2%
Fission Gas Release	1.85%	18%	0.07%	0.05%

3. CONCLUSIONS

The sensitivity analysis performed using TRANSURANUS code considering as test case the Nuclear Unit of ARKANSAS (ANO-2) fuel rod applying Monte Carlo approach as shown that the tangential stresses and strains in the cladding, the internal pressure in the fuel rod and the fission gas release are sensitive to manufacturing parameter associated to fuel cladding inner radius. Moreover, there is an intrinsic limitation in TRANSURANUS geometric modelling: one dimensional, plane and axisymmetric schematization characterized by plain strain condition, inability to model local geometry variation (i.e. ridges) [7].

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