Annals of Nuclear Energy 104 (2017) 28–41

Contents lists available at ScienceDirect

Annals of Nuclear Energy

journal homepage: www.elsevier.com/locate/anucene

# Uncertainty and sensitivity analysis for the simulation of a station blackout scenario in the Jules Horowitz Reactor

Alberto Ghione [b,*], Brigitte Noel [a], Paolo Vinai [b], Christophe Demazière [b]

[a] Commissariat à l'Énergie Atomique et aux énergies alternatives, CEA, DEN/DM2S/STMF/LATF, 17 rue des Martyrs, Grenoble, France
[b] Chalmers University of Technology, Division of Subatomic and Plasma Physics, Department of Physics, Gothenburg, Sweden

ARTICLE INFO

*Article history:*
Received 14 October 2016
Received in revised form 3 February 2017
Accepted 11 February 2017

*Keywords:*
Jules Horowitz Reactor
Station blackout
CATHARE
Uncertainty propagation
Sensitivity analysis
URANIE

ABSTRACT

An uncertainty and sensitivity analysis for the simulation of a station blackout scenario in the Jules Horowitz Reactor (JHR) is presented. The JHR is a new material testing reactor under construction at CEA on the Cadarache site, France. The thermal-hydraulic system code CATHARE is applied to investigate the response of the reactor system to the scenario. The uncertainty and sensitivity study was based on a statistical methodology for code uncertainty propagation, and the 'Uncertainty and Sensitivity' platform URANIE was used. Accordingly, the input uncertainties relevant to the transient, were identified, quantified, and propagated to the code output. The results show that the safety criteria are not exceeded and sufficiently large safety margins exist. In addition, the most influential input uncertainties on the safety parameters were found by making use of a sensitivity analysis.



## 1. Introduction

The safety analysis of nuclear power plants relies on simulations of operational and accidental scenarios. The simulation of the reactor system behavior under these conditions is usually performed with Best-Estimate (BE) system codes, such as CATHARE (Geffraye et al., 2011), TRACE (U.S. Nuclear Regulatory Commission, 2007) and RELAP (Information Systems Laboratories, Inc., 2001). For licensing purposes, BE codes in combination with conservative hypothesis have been extensively employed. However, this kind of approach may lead to unrealistic predictions where important safety issues may also be masked due to the high degree of conservatism. Therefore, in the past few decades, there has been an increasing interest in the use of BE codes with realistic assumptions, complemented with an uncertainty analysis (i.e. BEPU, Best Estimate Plus Uncertainty) (International Atomic Energy Agency, 2008). The evaluation of the impact of the uncertainties that can arise from the boundary and initial conditions, material properties, reactor operating conditions, code models, etc. becomes a crucial step because of the more realistic character of the simulations. The BEPU approach allows a reliable calculation of the safety margins, and avoids unnecessary conservatism. Several methodologies for uncertainty analysis have been developed, like the CSAU (Boyack et al., 1990), the GRS (Glaeser, 2008) and the CIAU (Petruzzi and D'Auria, 2008) ones. The GRS methodology is employed in this work. It is based on the statistical propagation of selected input uncertainties throughout the simulation, so that it is possible to determine uncertainty bands for the results, with a certain probability and degree of confidence. Furthermore, the results of the uncertainty propagation can be used for a sensitivity analysis to identify the most influential parameters.

The aim of the study is to apply a BEPU methodology for the analysis of a station blackout scenario in the Jules Horowitz Reactor (JHR), as a possible alternative to the modeling based on conservative assumptions (as described in Pegonen et al. (2014)).

The JHR (Iracane, 2006) is a material testing reactor under construction at CEA on the Cadarache site, France. It is a 100 MWth pool-type light water reactor, where fast and thermal neutron fluxes can reach high values (about $5 \cdot 10^{14}$ neutrons/cm$^2$/s). The installation will be suitable for the development and qualification of materials and nuclear fuel related to GEN-III and IV nuclear reactors, and for the production of medical radioisotopes.

In the context of this work, the GRS statistical methodology for uncertainty and sensitivity analysis is applied to the simulation of a Station Black-Out (SBO) scenario, making use of the thermal-hydraulic BE system code CATHARE 2 V25_3 mod5.1 and the 'Uncertainty and Sensitivity' platform URANIE (Gaudier, 2010).

* Corresponding author.
*E-mail addresses:* ghione@chalmers.se (A. Ghione), brigitte.noel@cea.fr (B. Noel), vinai@chalmers.se (P. Vinai), demaz@chalmers.se (C. Demazière).

http://dx.doi.org/10.1016/j.anucene.2017.02.008
0306-4549/

**Nomenclature**

| | |
|---|---|
| $A$ | flow area, $m^2$ |
| $c_p$ | specific heat capacity, J/kg/K |
| $D_{hydr}$ | hydraulic diameter $D_{hydr} = \frac{4A}{P_{wet}}$, m |
| $g$ | acceleration of gravity $g = 9.8066$, $m/s^2$ |
| $G$ | mass flux $G = \frac{\dot{m}}{A}$, $kg/m^2/s$ |
| $Gr$ | Grashof number $Gr = \frac{g\beta\rho^2 D_{hydr}^3}{\mu^2}\lvert T_W - T_l \rvert$, – |
| $h$ | heat transfer coefficient, $W/m^2/K$ |
| $i$ | specific enthalpy, J/kg |
| $i_{lg}$ | latent heat of vaporization, J/kg |
| $\Delta i_{sub}$ | liquid sub-cooling $\Delta i_{sub} = i_{l,sat} - i_l$, J/kg |
| $k$ | thermal conductivity, W/m/K |
| $l_{heat}$ | equivalent heated width $l_{heat} = \frac{P_{heat}}{2}$, m |
| $\dot{m}$ | mass flow rate, kg/s |
| $Nu$ | Nusselt number $Nu = \frac{hD_{hydr}}{k}$, – |
| $p$ | pressure, Pa |
| $Pe$ | Peclet number $Pe = RePr$, – |
| $Pr$ | Prandtl number $Pr = \frac{\mu c_p}{k}$, - |
| $P_{heat}$ | heated perimeter, m |
| $P_{wet}$ | wetted perimeter, m |
| $Ra$ | Rayleigh number $Ra = Gr\ Pr$, – |
| $Re$ | Reynolds number $Re = \frac{GD_{hydr}}{\mu}$, – |
| $S_{heat}$ | heated surface, $m^2$ |
| $T$ | temperature, °C |
| $\Delta T_{sat}$ | wall superheat $\Delta T_{sat} = T_w - T_{sat}$, °C |
| $z$ | axial distance, m |

*Greek symbols*

| | |
|---|---|
| $\beta$ | volumetric expansion coefficient, 1/K |
| $\lambda$ | Laplace length $\lambda = \frac{\sqrt{\sigma}}{\sqrt{g(\rho_l - \rho_g)}}$, m |
| $\mu$ | dynamic viscosity, kg/m/s |
| $\rho$ | density, $kg/m^3$ |
| $\sigma$ | surface tension, $kg/s^2$ |
| $\phi$ | heat flux, $W/m^2$ |

*Subscripts*

| | |
|---|---|
| $g$ | gas |
| $heat$ | heated |
| $l$ | liquid |
| $sat$ | saturation |
| $sub$ | sub-cooled |
| $w$ | wall |

The paper is organized as follows. In Section 2, the CATHARE modeling of the JHR is presented; in Section 3 the Station Black-Out transient is described along with the relevant safety criteria; in Section 4 the methodology for uncertainty and sensitivity analysis is discussed; in Section 5 the relevant uncertainties included in the study and the results of the analysis are commented; in Section 6 conclusions are drawn.

## 2. CATHARE modeling

In this section, the modeling of the JHR in the system code CATHARE is presented, together with the thermal-hydraulic correlations used for the simulations.

### 2.1. The Jules Horowitz reactor

The JHR is a pool-type light water reactor and the core is located in a pressurized tank at the bottom of the reactor pool (whose depth is approximately 10 m). The core has a diameter of 710 mm, and it is surrounded by a beryllium reflector. As shown in Fig. 1, the nuclear Fuel Assemblies (FAs) consist of a set of curved plates, arranged in eight concentric rings fixed with stiffeners. The narrow channels between the curved fuel plates (with an average gap size equal to 1.95 mm) are cooled by upward forced convection of water. In normal operations, high velocities of the coolant (up to 15 m/s) are needed due to the high core power density (approximately 460 kW/l) and the high heat fluxes (up to 5.5 $MW/m^2$). The central position of the FA can host either a control rod or an experimental test device (called test device 1 in the figure). Up to 37 fuel assemblies can be loaded in the core, although other experimental test devices (type 2 in the figure) can be included instead of the fuel assemblies.

In the current investigation, a core configuration with 34 FAs (27 of which have a control rod and 7 have an experimental device of type 1) and 3 experimental devices of type 2 is considered. For the purpose of the safety analysis, one of the fuel assemblies is assumed to be at a higher power and is named as 'hot fuel assembly'. The other 33 fuel assemblies, which are at the same lower power, are named as 'mean fuel assemblies'. The CATHARE

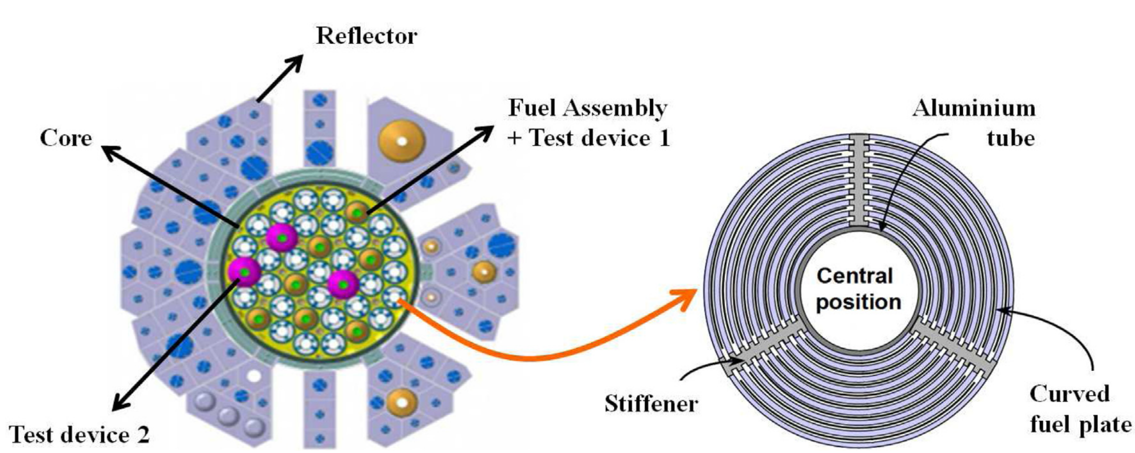


**Fig. 1.** Schematic of the JHR core and fuel assembly. Courtesy of CEA.