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BEPU ANALYSES FOR THE POSTULATED LOSS-OF-COOLING ACCIDENT SCENARIO IN THE SPENT FUEL POOL AT THE FUKUSHIMA DAIICHI UNIT 4

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Abstract

The aim of the paper is to present the results obtained from the study devoted on the Best Estimate plus Uncertainty (BEPU) Methods application. To this end, the simulation of the Loss of cooling accident (LOCA) scenario in the Fukushima Daiichi spent fuel pool was carried out. Special attention was paid on the uncertainty analysis with the domain fission products release. The work presented herein was carried out by using the ASTECv2.2b/SUNSET in the coupling mode. A brief description of the ASTECv2.2b/SUNSET coupling is provided and a short discussion concerning the BEPU methodology is also presented.

Key words: fission products, severe accident conditions, BEPU

Introduction. BEPU [1] is an approach used in the evaluation of the safety of Nuclear Power Plants (NPPs). This approach combines the best estimate code results with the uncertainty of the code predictions. Thus, BEPU provides a more realistic estimation of the plant response during the accident conditions. The present work deals with the application of this approach from the point of view of the fission products release during the postulated accident scenario at the

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spent fuel pool. For the purposes of the analysis, the ASTECv2.2b code [2] alone and coupled with the uncertainty quantification tool (SUNSET [3]) was used to simulate the postulated accident scenario at the Fukushima Daiichi unit 4 Spent Fuel Pool (FD SFP) [4,5] in order to apply the BEPU approach.

Short description of the Fukushima Daiichi spent fuel pool. In the past, the spent fuel pool (SFP) severe accidents have been considered highly improbable since the accident progression is slow (in comparison with reactor core accidents) and allow time for corrective operator actions. However, the accident at the Fukushima Daiichi Nuclear Power Plants has highlighted the vulnerability of nuclear fuels that are stored in spent fuel pools in case of prolonged loss of cooling accidents. The SFPs are large structures equipped with storage racks designed for temporary storage of the irradiated nuclear fuel removed from the reactor core [6].

The Fukushima Daiichi unit 4 spent fuel pool is the most accurately described spent fuel pool due to the strong interest from the research community after the accident in March 2011 and also as a subject of the international investigations in the area of severe accidents. The size of the pool is 12.2 m $\times$ 9.9 m $\times$ 11.5 m (Length $\times$ Width $\times$ Height), with a total number of fuel assemblies (FAs) in the pool of 1535. The fuel assemblies are from the recently unloaded core (548 FAs), longer stored spent fuel (783 FAs) and fresh fuel (204 FAs). The fuel of the recently unloaded core was assumed with a total fission power of 1.9 MW and the longer stored fuel assemblies with a total fission power of 0.5 MW [4].

The ASTECv2.2b code summary, a SFP model description and main hypothesis. The aim of the integral code ASTECv2.2b is to simulate the postulated severe accident conditions starting from the initiating event and ending with the discharge of the radioactive material (so-called "source term") out of the containment [7]. The main applications of the integral ASTEC code are: assessment of fission product (FP) behaviour [8], determinations of source term, Probabilistic Safety Assessment, level 2 (PSA2) studies, accident management studies and analyses of the severe accident phenomenology [2]. ASTECv2.2b code is a modular code system, where a part of the phenomenology is implemented in an appropriate module.

The input model for FD SFP is developed by the Institute of Nuclear research and Nuclear energy (INRNE) for the ASTECv2.2b and was used as a base for the presented coupling calculations. For the calculations, some of the modules tailored in the ASTECv2.2b were activated: CESAR module, ICARE module coupled with the ELSA module, SOPHAEROS, DOSE, and CPA. All ASTEC modules have been used in a "coupled mode".

The Fukushima Daiichi unit 4 Spent Fuel Pool (FD SFP) accident scenario initial and boundary conditions are as follows: Water level (collapse level) – 4.5 m; Pressure in the SFP building – 1 bar; Water temperature – 100 °C; Temperature of the concrete wall of the pool – 100 °C; Atmosphere temperature – 80 °C; Atmosphere composition – 100% (relative humidity); Temperature of the building wall

– 80 °C; Temperature in the environment – 20 °C; Heat transfer outside of concrete wall – Adiabatic; Heat transfer coefficient from building wall to environment – not specified.

The nodalization scheme for FD SFP unit 4 is presented in Fig. 1. The SFP is modelled by VESSEL, PRIMARY and CONTAINM structures (the "stru" VESSEL is modelled with four channels and three rings; the "stru" PRIMARY contains 3 volumes: two volumes are located above the vessel and one connecting bottom with the containment; the "stru" CONTAINM contains two zones: SFP-HALL and ENVIRONMENT. A connection between the HALL and the Environment is organized).

The Fukushima Daiichi unit 4 Spent Fuel Pool (see Fig. 1) is modelled as a cylinder, enclosed in a steel liner and concrete. The fuel and the racks are modelled in the ICARE/CESAR module with 15 axial and 3 radial nodes. The lower plenum "LOWERPLE" is connected to the 4 channels located above it. The initial and boundary conditions previously described are used. Furthermore, as a criterion to stop the computation, we choose for the presented analysis – 1% dissolving of UO_2. Additional features used when modelling the SFP, are: Fuel assembly (FA) model in axial direction is subdivided into the following nodes: 3 for unheated lower part of FA, 10 for heated part of FA, and 2 nodes for unheated

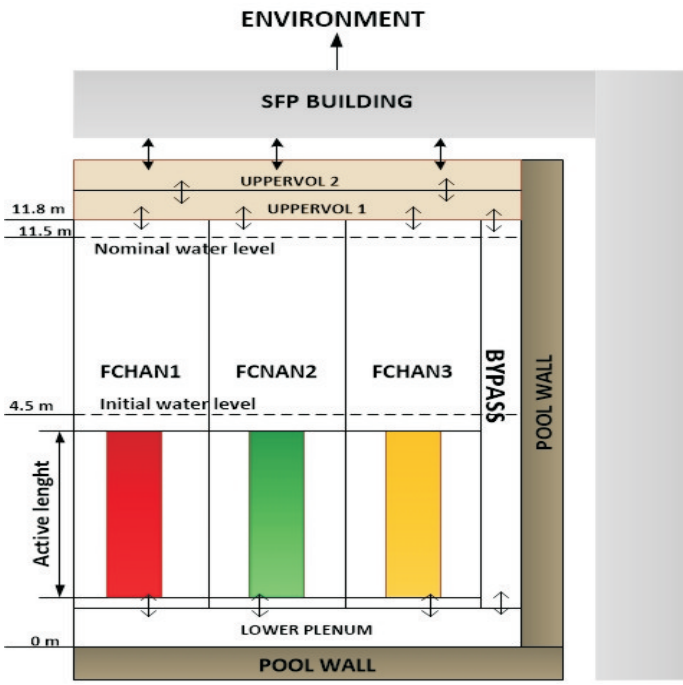

Fig. 1. Schematic view of the ASTECv2.2b nodalization SFP Fukushima Daiichi unit 4

upper part of FA. The volume above fuel assemblies is divided in 10 nodes (from unheated upper part of FA to 11.8 m). For these analyses the water level is reduced to the level of upper part of the fuel assemblies. The rack region is divided into 3 radial rings: ring 1 contains the hottest 548 Fuel Assemblies (FAs) (1.9 MW); ring 2 contains the less hot 783 FAs (0.5 MW); and ring 3 contains the fresh 204 FAs.

The "Lower plenum" (volume located between bottom of SFP and bottom of the fuel assemblies) is modelled with a "STRU" LOWERPLE. The main supporting plate is located in a lower plenum and is used as a base of racks and is modelled with a "STRU" MACR. Below the main support plate there is a small space of around 0.2 m and below that space is located a lower wall of a SFP with a concrete material.

The SUNSET tool brief description. The SUNSET (Statistical UNcertainty and Sensitivity Evaluation Tool) software [3] developed by IRSN, is a statistical tool designed for uncertainty and/or sensitivity analysis of mathematical or physical models like computer codes. It is composed of two parts: a set of statistical computing and interface functions, based on C++ language; and a Graphical User Interface allowing the call of statistical functions, in Java language. In the framework of severe nuclear accidents SUNSET has been coupled to ASTEC code. The construction and execution of a SUNSET data deck is performed through the SUNSET GUI [9,10]. The general sketch of uncertainty analysis, discussed in this paper is presented in Fig. 2.

The coupling between the SUNSET tool and the ASTEC code comprises of four main steps, as follows [9]: preparation of the ASTEC runs/data deck; SUNSET Pre-processing; launching the calculations/ASTEC running; SUNSET post-processing.

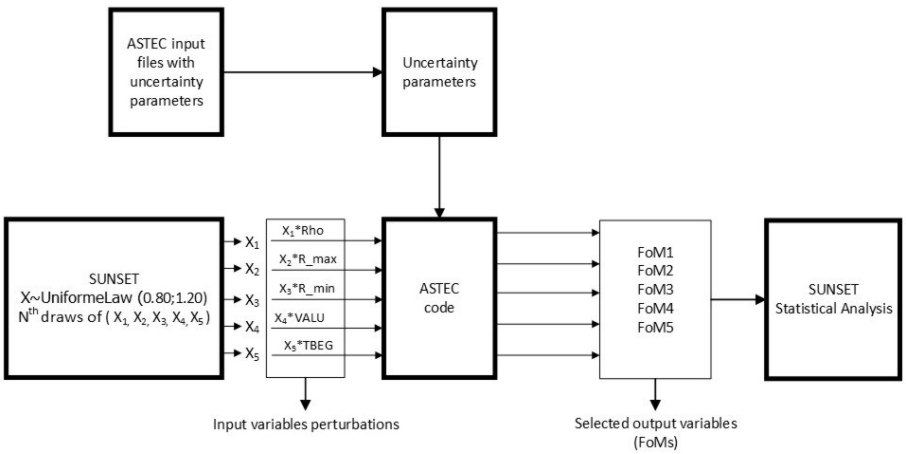

Fig. 2. Sketch of the ASTEC/SUNSET coupling calculation scheme applied in the uncertainty analyses (Source: Figure 3 from Ref. [7] was adapted)

Results and discussions. This section describes the basic steps in the process of the application of the BEPU approach for the selected accident scenario at the spent fuel pool. These steps are: characterization of the scenario, the selection of the code, the nodalization process, the uncertainty quantification process and the final step – application of the uncertainty analysis. According to [1], the uncertainty analysis consists of identification and characterization of relevant input parameters (input uncertainty) as well as of the methodology to quantify the global influence of the combination of these uncertainties on selected output parameters (output uncertainty). Within the SUNSET tool, the uncertainties are evaluated using the propagation of input uncertainties. The propagation of input uncertainties means that the uncertainty is obtained following the identification of uncertain input parameters with specified ranges of these parameters, and performing calculations varying these parameters.

Prior to the uncertainty analysis, a base case analysis has been performed. The base case input deck for the studied accident scenario is used for the preparation of the ASTECv2.2b/SUNSET coupling data deck.

The SUNSET statistical tool coupled with the severe accident code (ASTECv2.2b) was used to carry out the present study devoted to the application of the BEPU approach for the analyses of the accident scenario in the SFP at Fukushima Daiichi NPP, Unit 4. For this purpose, five uncertain parameters with their distribution function were chosen in order to study its effect at the five output parameters (FoMs). The list of the input uncertain parameters is as follows: "Rho" – particle mean density [kg/m^3]; "R_max" – particle maximum geometrical radius [m]; "R_min" – particle minimum geometrical radius [m]; "VALU" – thickness for loss of clad integrity [µm]; "TBEG" – oxidation [K]. We choose to model the uncertainty (see Fig. 2) by 5 uniform distribution (X1 to X5) between (0.80; 1.20).

The selected FoMs dealing with the fission products behaviour used in the calculations as a plot file in the ASTECv2.2b/SUNSET input deck are as follows: FoM1 – mass of "I" released from the fuel, [kg] (see Fig. 3); FoM2 – mass of "Cs" released from the fuel, [kg]; FoM3 – "I" released from the fuel as a % of the initial inventory (i.i); FoM4 – "Cs" released from the fuel as a % of the i.i.; FoM5 – Integral mass of 6 important FPs released into the environment (Σ FP: I, Cs, Sr, Ru, Ce, Ba), [kg]. A set of a random sample of input decks is generated by a random combination of the uncertain parameter over their probability distribution function. The number of samples is determined by using the Wilks' formula [1.3] according to the desired probability content and confidence level. In the present study, a random sample of 50 input decks is generated to get a 95% probability content and a confidence level of 90% of a one-sided statistical tolerance limit. The used sampling method is a Latin Hypercube Sample (LHS). This method of Latin hypercube sampling (LHS) ensures that each variable is represented once in the entire domain of interest.

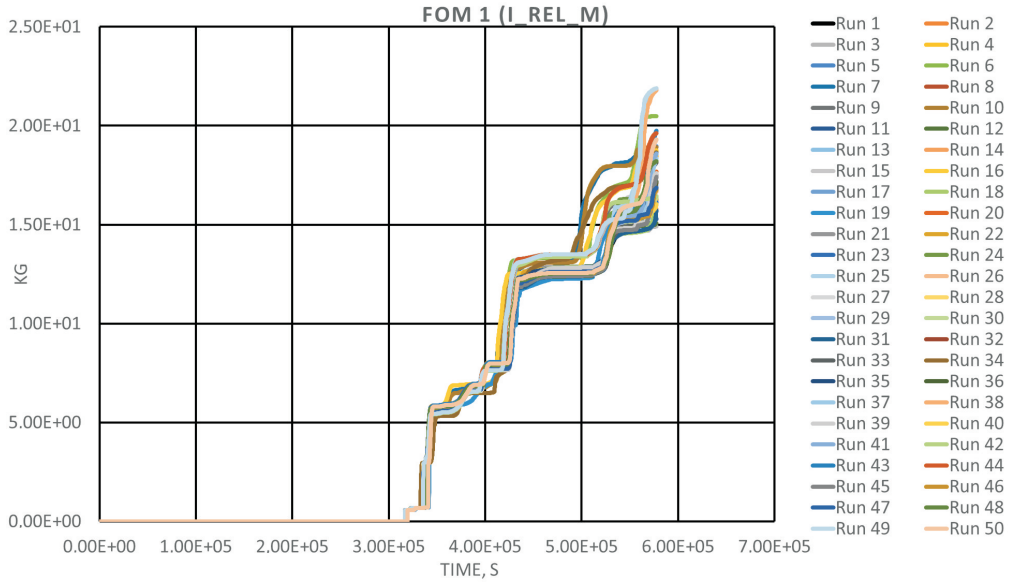

Fig. 3. Results for the FoM1 ("I" released from the fuel)

The results of the uncertainty analysis are summarized in Table 1. For each of the five studied output variables, the information for the average value, the standard deviation (std), the minimum (min), and the maximum (max) values is presented.

T a b l e 1

Results from the classic statistical analysis of the selected FoMs

	average	std	min	max
'Outputs_variable#1'(FoM1)	17.77645	1.719583	14.9091	21.8861
'Outputs_variable#2'(FoM2)	277.9908	24.70221	236.801	337.026
'Outputs_variable#3'(FoM3)	39.20067	3.792027	32.8776	48.2632
'Outputs_variable#4'(FoM4)	41.24174	3.664737	35.131	50.0001
'Outputs_variable#5'(FoM5)	121.8886	9.650967	110.041	144.577

Summary and conclusions. An application of the BEPU approach in the area of severe accidents research for spent fuel pool (SFPs) is presented. The main steps for the coupling of the severe accident code ASTEC and the uncertainty tool SUNSET are outlined.

The report demonstrated the calculation of some important fission products (FPs) released from the fuel in a case of a severe accident in the SFP and assessed their uncertainties based on an application of the SUNSET tool. A total of 50 ASTEC calculations were performed for this purpose.

The main statistical quantities – the average values and their standard deviations, as well as the minimum and the maximum values of investigated FPs (FoMs) were calculated and evaluated as a result of a study based on the Best Estimate plus Uncertainty (BEPU) methodology.

According to the Wilks' formula, the received results are with a 95% probability and 90% confidence.

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