AN APPROACH TO MERGE PSA-BASED PREMISES AND BEPU METHOD IN ORDER TO DEVELOP THE EBEPU METHODOLOGY

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In the past, nuclear industry has trusted on the concept of defense in depth and safety margins to deal with the uncertainties associated with the design and operation of nuclear facilities. This approach suggests making extensive use of redundancy, diversity and large margins to guarantee plant safety in a conservative way.

For example, IAEA (International Atomic Energy Agency) produced guidance on the use of DSA (Deterministic Safety Analysis) for the design and licensing of NPP (Nuclear Power Plants) in "DSA for NPP Specific Safety Guide, No. SSG-2". This guide proposes four options for the application of DSA.

Options 1 and 2 are conservative, which have been used since the early days of civil nuclear power, and are still widely used today.

Option 3 involves the use of best-estimate codes and data together with an evaluation of the uncertainties, the so called BEPU methodologies. Several BEPU approaches have been developed in a scope that are accepted by the regulator authorities nowadays. Most of them are based on propagation of in-put uncertainties and make use of the Wilks' –based methods to determine the number of calculations of the output (usually safety-related parameters) needed to verify compliance of acceptance criteria with "standard tolerance levels (STL)" (typically 95/95) in accordance with current regulatory practice.

Option 4 establishes the framework for combining insights from probabilistic and deterministic analysis with an aim of combining the use of well-established BEPU methods and probabilistic assumptions on safety system availability.

In this paper, an approach is proposed to build an Extended BEPU methodology that follows the principles of Option 4 of the above IAEA SSG-2 guide. The so-called EBEPU methodology proposed in this paper merges PSA-based models and data with well-established BEPU-DSA. An example of application is provided to demonstrate how this approach performs.

I. INTRODUCTION

Both deterministic and probabilistic safety analyses are performed with an aim to achieve regulatory approval of NPP (Nuclear Power Plant) design and operation according to well-established licensing basis (Ref. 1).

What concerns DSA (Deterministic Safety Analy-sis), the International Atomic Energy Agency (IAEA) produced guidance on the use of deterministic safety analysis for the design and licensing of nuclear power plants (NPPs): ''Deterministic Safety Analysis for Nuclear Power Plants Specific Safety Guide'', Specific Safety Guide No. SSG-2 (Ref. 2, hereinafter referred to as SSG-2). SSG- 2 addresses four options for the application of DSA.

Options 1 and 2 are conservative and they have been used since the early days of civil nuclear power, and are still widely used today. However, the desire to utilize current understanding of important phenomena and the availability of reliable tools for more realistic safety analysis without compromising plant safety has led many countries to use option 3.

Option 3 involves the use of best-estimate codes and data together with an evaluation of the uncertainties, the so-called BEPU (Best Estimate Plus Uncertainty) methodologies. Several BEPU approaches have been developed in a scope that are accepted by the regulator authorities nowadays.

Nowadays, there is a growing interest aimed at developing methods for using probabilistic safety analysis (PSA) results into requirements and assumptions in deterministic safety assessment (DSA), and vice versa, to provide a more comprehensive and realistic measure of reactor safety (Ref. 3).

In Ref. 4, it is proposed an approach for combining insights from probabilistic and deterministic analysis in Option 4 based on the IAEA SSG-2 guide (Ref. 2), which can be used for deterministic safety analysis of current NPP designs.

A medium-term challenge for implementing the fundamentals of Option 4 in Ref. 2 is to merge the use of well stablished BEPU methods and probabilistic-based assumptions on systems availability to build an Extended BEPU methodology, the so-called EBEPU methodology.

This paper adopts the approach proposed in Ref. 5 to integrate probabilistic assumptions on the availability of safety systems into DSA and adopts the fundamentals of BEPU methods to build a novel EBEPU approach that follows the principles of Option 4 of the above IAEA SSG-2 guide, which improves the approach proposed in Ref. 6. The EBEPU methodology proposed in this paper integrates PSA-based models and data into well-established BEPU-DSA. This paper presents the main results obtained using the novel EBEPU approach as compared to the results found using the traditional BEPU one facing on an accident scenario corresponding to the initiating event “Loss of Feed Water (LOFW)” for a typical three-loops Pressurized Water Reactor (PWR) NPP.

II. BEPU AND EBEPU APPROACHES

BEPU approaches for the analysis of a particular design basis accident (DBA) assume the uncertainty in the safety outputs, i.e. the figures of merit (FOM) involved in the acceptance criteria of the DSA analysis, derives from the uncertainties in the inputs to the calculations (initial and boundary conditions) and those arising from the computational model.

These FOM are usually extreme values of safety variables during the transient linked to the DBA accident (e.g. CHF, PCT, …). They describe the degree of challenge to the physical barriers that confines radioactivity in a NPP.

Current BEPU methodologies mainly rely on a probabilistic description of the uncertainty and on the use of statistical techniques to estimate it. In this framework, the uncertainty of a FOM can be identified with its probability distribution.

Regulatory requirements impose acceptance criteria on these FOM and the BEPU analysis must demonstrate compliance of FOM with acceptance criteria.

Most of BEPU approaches accepted by the regulatory authorities are based on propagation of input uncertainties and make use of the Wilks’ –based methods to determine the number of calculations of the output, i.e. FOMs, needed to verify compliance of acceptance criteria with “standard tolerance levels (STL)”, typically 95/95 in accordance with current regulatory practice. Accordingly, the value of the FOM that is compared with the corresponding acceptance criterion is often an upper or lower tolerance limit with level 95/95 instead of the FOM probability distribution. For example, it is often used one-side tolerance interval of FOM based on the use Order Statistics (OS) of first order with STL=95/95, which requires a sample size of N=59 runs.

The main advantage of using first order statistics (FOS) based on Wilks’ formulae to derive the STL is that it provides always a conservative result with a few runs of the computer code. This way, the computational cost is kept practicable since the simulation of the evolution of the plant transient for each sample of inputs using complex TH (Thermal Hydraulic) models of NPP is very expensive in terms of computational cost.

The EBEPU approach proposed in this paper merges traditional BEPU methods and PSA-based assumptions on the availability of safety systems in a simple and natural way, the later making use of the approach proposed in Ref. 5, which consist of the following steps:

1. Selection of the accident scenario.
2. Selection of the safety criteria linked to the accident scenario under study and the FOM involved in the acceptance criteria.
3. Identification and ranking of relevant physical phenomena based on the safety criteria.
4. Selection of the appropriate TH (Thermal Hydraulic) parameters to represent those phenomena.
5. Identification of relevant safety-related systems involved in the accident scenario.
6. Selection of relevant components/trains of the above redundant safety systems that are responsible for performing the intended safety function to mitigate accident consequences.
7. Development of the TH computer model of the accident scenario, e.g. develop an input for TRACE (Ref. 7).
8. Association of PDF (Probability Density Functions) for each selected TH parameter.
9. Identification of relevant system configurations based on the availability of safety components/trains and association of a probability of occurrence for each configuration. Ref. 5 proposes an approach to accomplish this step.
10. Random sampling of the selected TH parameters and plant configurations. Sample size (N) will depend on the particular statistical method and the acceptance criterion adopted to verify compliance of safety criteria. Perform N computer runs to obtain FOM for each run.
11. Processing the results of the multiple computer runs (N) to estimate either the probability distribution of the FOM, or rather some descriptor of this distribution, such as for example a percentile of the FOM, or a tolerance level of FOM with STL using OS, etc.
12. Verify compliance of acceptance criterion for each FOM

The main difference between the BEPU and EBEPU approach is the incorporation of steps 6 and 9 to account for realistic and PSA-based assumptions on safety systems availability under the EBEPU approach. In addition, step 10 must be updated to account also for random sampling of safety systems configurations in addition to TH parameters. At last but not at least, the TH computer model must be developed in step 7 at appropriate level of detail at component/train in a coherent manner with step 6 in order to make it possible to address the particular configuration of the safety systems required for each computer run in step 10.

BEPU approaches focuses only on an enveloping sequence representing a conservative progression of the accident scenario (step 1) from an initiating event. Thus, for such an enveloping accidental sequence, it is adopted a conservative assumption on the availability of safety systems (steps 5 and 7), so that steps 6 and 9 are not necessary.

III. CASE STUDY

III.A. Description of the Initiating Event

The case study focusses on an accident scenario corresponding to the initiating event “Loss of Feed Water (LOFW)” for a typical three-loops Pressurized Water Reactor (PWR) NPP.

The group LOFW includes those transients involving total loss of main feed water to steam generators (SG), which reduce water level of SG and consequently reduction of its capacity to extract heat from the reactor coolant system (RCS). In particular, this group includes initiating events of category 16 and 24 in EPRI/NP-2230 (Ref. 8).

Table I shows the safety functions required following the occurrence of LOFW and corresponding success criteria of such functions. It shows two alternative ways to remove heat from the RCS once the Reactor Protection System (RPS) is successful to perform SCRAM. One way involves the injection of water to SG by the Auxiliary Feed Water System (AFW) and evacuation of heat through steam-dump valves (SD), relief valves (RV) or safety valves (SSV). Eventually, in case of RCS pressurization, there may be a need to reduce pressure by means of the PORV valves (and safety valves SV when required) of the pressurizer (PRZ). Second alternative involves removing heat from the RCS by means of “Feed and Bleed”, i.e. extracting warm water opening PORV valves manually and injecting cold water using the high-pressure injection system (IHI). In addition, it is needed recirculation of water from the RCS using the same system under recirculation operational mode (IHR) in order to keep a safe operational state of the NPP in the long term.

Figure 1 shows a typical event tree for the LOFW transient taken from the level 1 PSA available. Several accidental sequences follow the initiating event.

TABLE I. Safety functions and success criteria required for LOFW

Initiating Event	Reactivity Control	Inventory Control	Pressure Control	RCS Heat Removal	Long Term Heat Removal
LOFW	RPS	N/A	1/2 PORV or 1/3 SV	1/3 AA & (1/8 SD or 1/3 RV or 1/15 SSV)	N/A
	RPS	1/3 IHI	1/2 PORVm	N/A	1/3 IHR & 1/2 ILA

III.B. Safety Criteria, FOMs and Safety Limits

The document “Acceptance criteria and related safe-ty margins” developed by the Task Group on CSNI Safety Margins Action Plan (SMAP) of the OECD/NEA (Ref. 9) provides not only safety criteria and corresponding FOM, but also an entire methodology to classify them into Categories ac-cording to plant states and transient frequencies.

The PSA available for this case study corresponds to a typical three-loops PWR NPP, which uses a frequency of the initiating event LOFW F(LOFW)= 2,66 10^{-1} r^{-1} yr^{-1}. Based on this frequency, the initiating event LOFW belongs to Category 2 according to the above document. It also proposes several FOM representing safety criteria, their corresponding safety limits and acceptance criteria for Category 2.

III.C. Safety Criteria

Once the category has been selected, it is required identify the safety variables of interest and set the acceptance criteria marked for their respective limits.

There are different barriers in the plant in order to guarantee the public safety.

The first barrier is the fuel, and the safety variables of interest include the critical heat flux (CHF) in the rods, the fuel temperature at the centerline of the rod (FUELT), and the peak clad temperature (PCT).

Related to the CHF (Ref. 10), in this work the Departure from Nucleate Boiling Rate (DNBR) is calculated, which is the ratio of the heat flux needed to cause departure from nucleate boiling to the actual local heat flux of a fuel rod. This DNBR cannot be lower than 1.17 at 0% of the rods for category 2. Our safe-ty parameter related with the DNBR will be DNB_MIN that is the minimum of the DNBR at any location and at any time of the transient.

The fuel temperature safety variable FUELT has as related safety parameter the FUELT_MAX, which is the maximum fuel temperature at any location and at any time of the transient, whose limit of acceptance for category 2 is 2863 K.

In the same way, the safety variable PCT have as related safety parameter PCT_MAX, and its limit of acceptance for category 3 is 923 K.

Next barrier is the primary system of the plant and in order to assure its integrity the safety variable of interest considered herein is the pressurizer pressure (PRZP). The safety parameter associated is the maximum pressurizer pressure (PRZP_MAX) and the limit of acceptance for category 2 is 18.9 MPa.

Last barrier considered is the containment and the safety variable is the leakage at the containment (CONTL), where the corresponding safety parameter is the maximum containment leakage (CONTL_MAX) and the limit is 0.2% for categories 2.

Table II shows safety parameters and acceptance criteria established for categories N=2.

TABLE II. Safety parameters (FOM) and limits (acceptance) required for Categories 2

Barrier	Safety Variable	Safety Parameter (FOM)	Acceptance Criteria Category 2
Fuel	CHF	DNB_MIN	1.17 at 0% rods
Fuel	FUELT	FUELT_MAX	2863 K
Fuel	PCT	PCT_MAX	923 K (limit proposed for sceneries Category 3)
Primary	PRZP	PRZP_MAX	100% design Press (1.89E+7 Pa)
Containment	CONTL	CONTL_MAX	0.2% of primary inventory

III.D. Selection of Accidental Sequences

For sake of simplicity in the exposition of the integration of the PSA-based configurations of systems availability into BEPU studies, sequence #1 in figure 1 is the only one considered in the remaining of this case study. Other sequences should be studied in a similar way as proposed for this sequence #1.

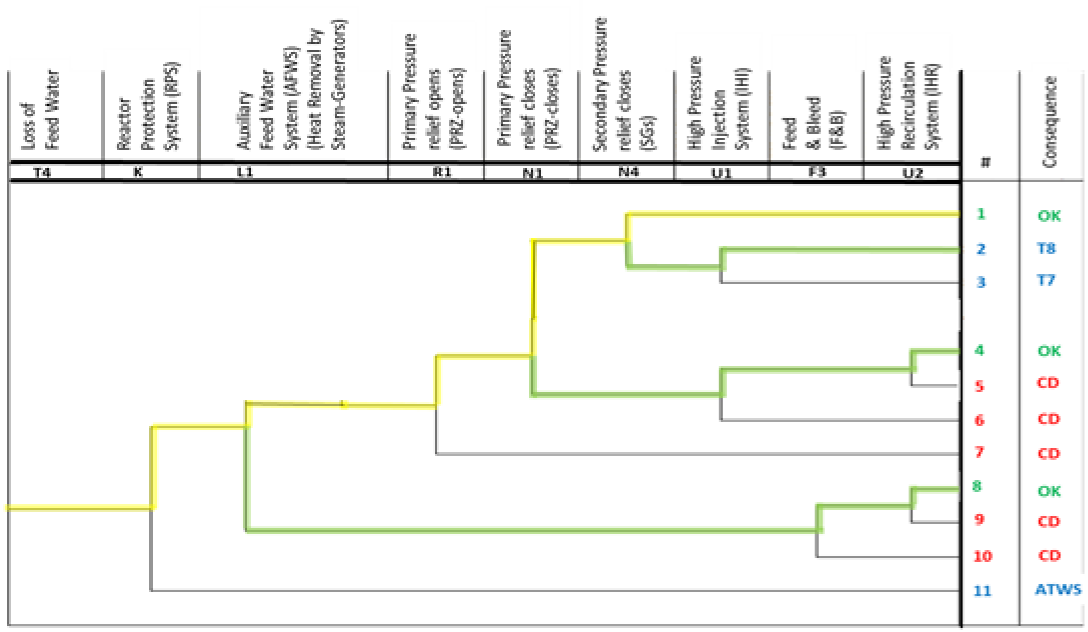

Fig. 1. Event Tree for the LOFW transient (T4.

III.E. Model Description

A typical 3-loops PWR- NPP has been modeled for TRACE code and ran with version V5.0 Patch 4 using the SNAP suite to simulate the transient corresponding to sequence #1. The model consists of two thermal hydraulic systems linked: primary and secondary systems (Ref. 6).

The primary system includes a tri-dimensional component type VESSEL, which represents the re-actor pressure vessel including the core, three independent reactor coolant loops, three steam generators (3 SG), one pressurizer (PRZ) and three pumps.

Reactor core consists of fuel elements of 15x15 rods. The elements are divided into two groups. One group models the average power and another group, which is the most unfavorable fuel element, models the peak power of the hot channel. Average power elements are associated to the coolant cells of the VESSEL element (in radial sector 1, axial levels 3 to 8), while hot channel is modeled apart through a unidimensional PIPE component connected to the en-trance and exit of the core.

In addition, safety systems supporting the primary system have been modeled: three trains corresponding to the high-pressure injection and recirculation system (3 IHI and 3 IHR respectively) connected to the RCS (reactor coolant system), and 2 PORV valves and three safety valves (3 SV) connected to the PRZ.

The secondary system includes steam generators (3 SG) associated each with the corresponding cooling loops in the primary system and PIPES representing the main feed water system (3 AP). It also includes turbine group and steam dump valves. In addition, safety systems supporting the secondary system have been modeled: auxiliary feed water sys-tem (3 AA), main steam isolation valves (3 MSIV), relief valves (3 RV) and safety valves (3 SSV groups of 5 valves).

III.F. Availability of Safety Systems: Configurations and Probabilities

Figure 1 shows accidental sequence #1 involves simultaneous occurrence and without exceptions of the following events: Occurrence of initiating event LOFW, success of RPS (K), success of the AFWS (L1), success to open of the primary pressure valves connected to the PRZ (R1) and success to close completely these valves (N1).

The PSA available includes the above event tree (Figure 1) and the fault trees required to represent the availability (and unavailability) of the safety sys-tem trains that take part of the safety functions involved in this event tree.

In (Martorell et al., 2015a) it is proposed an approach to obtain the set of realistic configurations of availability of the safety trains taking part of a given accident scenario and the corresponding occurrence probability. This approach has been used in this section.

The following set (vector) is used to represent the availability of safety function trains for this case of application: x = {AA1, AA2, AA3, PORV1c, PORV2c, SV1c, SV2c, SV3c, PORV1o, PORV2o, SV1o, SV2o, SV3o}

Vector x encodes the generic configuration of the availability of the trains taking part of the accident scenario. One realization of vector x will represent a particular a possible configuration.

Table III summarizes the list of results found only for the 20 most relevant configurations and probabilities using the procedure proposed in Ref. 5 with application to accidental sequence #1, which represent a cumulative probability of 0,967. There, every configuration is represented by a set of “1” and “0” values, each for the corresponding safety train encoded in vector x. A value “1” means the train, e.g. PORV1o, is available while a value “0” means it is unavailable.

On the other hand, under the BEPU approach only the most conservative configuration is considered, which is represented in Table IV for the accidental sequence #1. Note this represents the worst, where it is assumed the failure to open of the steam dump and relief valves of the secondary side, and that only one safety valve of the secondary side opens (very conservative). This very conservative configuration is associated a very low occurrence probability (Pr) as shown in Table IV as compared with those more realistic configurations considered under the EBEPU approach, which have associated the highest occurrence probabilities as shown in Table III.

TABLE III. EBEPU Safety systems configurations and probabilities for accidental sequence #1

Header	K	L1			R1					N1					N4													
Comp	-	AA1	AA2	AA3	PORV1a	PORV2a	SV1a	SV2a	SV3a	PORV1c	PORV2c	SV1c	SV2c	SV3c	SD1c	SD2c	SD3c	SD4c	SD5c	SD6c	SD7c	SD8c	MSIV1c	MSIV2c	MSIV3c	Sec	Pr	Pr Acc
Config #																												
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0,7831	0,7831
2	1	1	0	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0,0611	0,8442
3	1	0	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0,0251	0,8692
4	1	1	1	0	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0,0249	0,8941
5	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0,0174	0,9115
6	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0,0102	0,9218
7	1	1	1	1	1	1	0	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0,0049	0,9266
8	1	1	1	1	1	1	1	0	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0,0049	0,9315
9	1	1	1	1	1	1	1	1	0	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0,0048	0,9364
10	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0	0	0	1	0,0038	0,9402
11	1	1	1	1	1	0	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0,0034	0,9436
12	1	1	1	1	0	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0,0034	0,9470
13	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0	1	1	1	1	1	1	1	1	1	0,0025	0,9495
14	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0	1	1	1	1	0,0025	0,9520
15	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0	1	1	1	1	1	1	0,0025	0,9546
16	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0	1	1	1	1	1	0,0025	0,9571
17	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0	1	1	1	1	1	1	1	1	1	1	1	0,0025	0,9595
18	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0	1	1	1	1	1	1	1	0,0025	0,9620
19	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0	1	1	1	1	1	1	1	1	0,0025	0,9645
20	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0	1	1	1	1	1	1	1	1	1	1	0,0025	0,9670

TABLE IV. Worst case safety systems configuration for accidental sequence #1

Header	K	L1			R1					N1					N4												
Comp	-	AA1	AA2	AA3	PORV1a	PORV2a	SV1a	SV2a	SV3a	PORV1c	PORV2c	SV1c	SV2c	SV3c	SD1c	SD2c	SD3c	SD4c	SD5c	SD6c	SD7c	SD8c	MSIV1c	MSIV2c	MSIV3c	Sec	Pr
Config #																											
1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	1	0,7831

III.G. Initial and boundary conditions

Table V shows initial and boundary conditions for primary and secondary loops and, if applicable, their corresponding uncertainty parameters.

TABLE V. Initial and boundary conditions for the primary and the secondary loops

Parameter	**Units**	**Ref. Value**	**Uncertainty**
Thermal Power	MW	2686.0	INPOW
Hot Channel Peak Power	(-)	+20%	PEAKF
Pressurizer Pressure	MPa	15.6	-
Hot Channel Temperature	K	618.0	-
Hot Leg Temperature	K	604.8	-
Cold Leg Temperature	K	564.3	-
Average Temperature	K	584.54	-
Pressurizer Temperature	K	618.43	-
Rod Outside Diameter	mm	9.48E	-
Pressure Loss in the core	kPa	250	-
Mass Flow Rate in the primary	kg/s	12700	-
Pressurizer Level	M	7.22	-
Steam Generators Pressure	MPa	6.86	-
Steam Generators outlet Temperature	K	557.60	-
Steam Generators inlet Temperature	K	499.00	-
Steam Generators Pressure Loss	kPa	20	-
Steam Generators Level (NR)	%	50.60	-
Mass Flow Rate by Steam Generator	kg/s	475.00	-
Fuel Thermal Conductivity	W/mK	Table vs Temp.	UO2TC

In addition, Table VI shows relevant operational conditions for the transient and, if applicable, their corresponding uncertainty parameters.

TABLE VI. Relevant conditions for the transient

Parameter	Units	Ref. Value	Uncertainty
Level set point in SGs for SCRAM signal	%	17.6	SCRSG
Delay to reactor SCRAM	S	0	SCRTO
Residual power multiplier	MW	Table vs time	RPOWM
Delay to start AFWS pumps	s	0	AAATO
AFWS flow temperature	K	293.15	AAATI
AFWS flow rate	kg/s	24.28	AAAQI
Delay to open Steam Dump valves	s	0	-
Pressure set point secondary RV valves	MPa	7.7	-
Pressure set point secondary SSV valves	MPa	8.1	-
Pressure set point primary PORV valves	MPa	16.03	PRPRV
Pressure set point primary SV valves	MPa	17.13	PRPSV
Delay to close PORV and SV valves	s	0	VCLTO

Table VII shows the description of the uncertainty parameters considered in this case study (Ref. 11).

TABLE VII. Description of uncertainty parameters

Parameter	Parameter type	Distribution type	Min	Max	Mean	Std. Dev.
INPOW	Multiplicative	Normal	0.98	1.02	1	0.01
UO2TC	Multiplicative	Normal	0.9	1.1	1	0.05
PEAKF	Multiplicative	Normal	0.95	1.05	1	0.025
SCRSG	Multiplicative	Uniform	0.95	1.05	NA	NA
SCRTO	Additive	Uniform	0	27	NA	NA
RPOWM	Multiplicative	Normal	0.92	1.08	1	0.04
AAATO	Additive	Uniform	0	197	NA	NA
AAATI	Additive	Uniform	-2	2	NA	NA
AAAQI	Multiplicative	Normal	0.95	1.05	1	0.025
PRPRV	Additive	Uniform	-0.2	0.2	NA	NA
PRPSV	Additive	Uniform	-0.2	0.2	NA	NA
VCLTO	Additive	Uniform	0	180	NA	NA

IV. RESULTS

This section summarizes the results of the application of the BEPU and EBEPU approaches proposed in Section 2 using the TRACE code and the models and data introduced in previous Section 3. It en-compasses with steps 10 to 12; i.e. random sampling of system configurations and TH parameters (59 samples to obtain the one-sided STL 95/95), 59 computer runs to simulate accidental scenario #1 to obtain the evolution of the FOMs and verification of acceptance criteria for the safety parameters. For this case, we have chosen one FOM for each barrier, selecting the PCT for the first barrier, as is the most representative FOM for the fuel in this configuration.

Simulation of the accidental scenario #1 starts with the coast down of the steam generators main feed water pumps at t=100s, diminishing the capability of the secondary system to remove primary heat. Consequently, the water level of the steam genera-tors abruptly plunges activating the SCRAM signal. In addition, auxiliary feed water pumps start restoring the capability of the secondary system to remove primary heat after SCRAM. However, primary pressure starts increasing early in the transient. Thus, primary pressure may increase up to reach PORVs valves rated pressure that opens (alternatively, SVs must open in case of PORVs failure). Once pressure has decreased latter on during the transient, primary relief or safety valves closes and the plant goes to OK state.

Figures 2 to 4 show the simulation of the evolution of the three FOMs selected under the BEPU approach, while Figures 5 to 7 show the evolution of the same FOMs under the EBEPU approach.

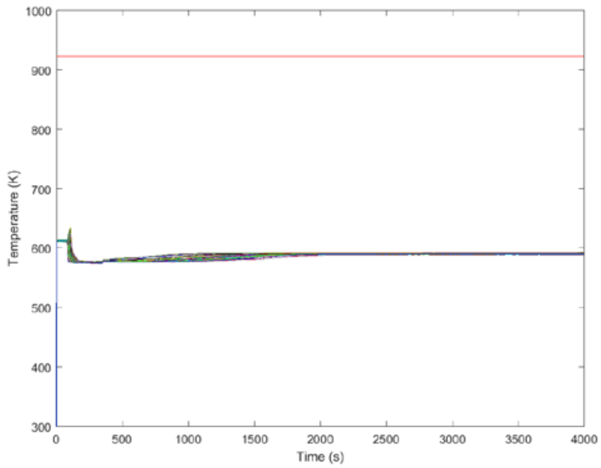

Fig. 2. PCT in seq #1 for BEPU approach.

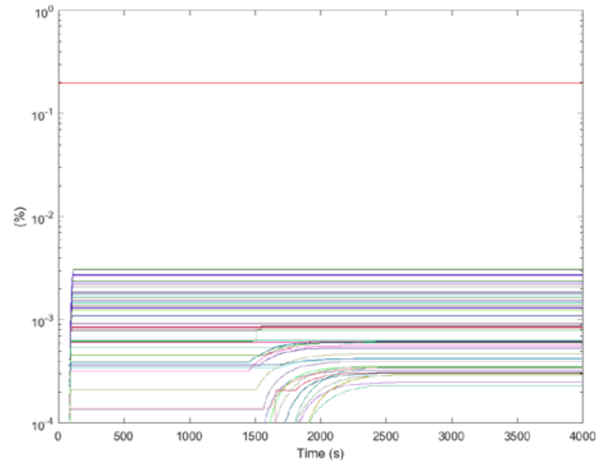

Fig. 3. CONTL in seq #1 for BEPU approach.

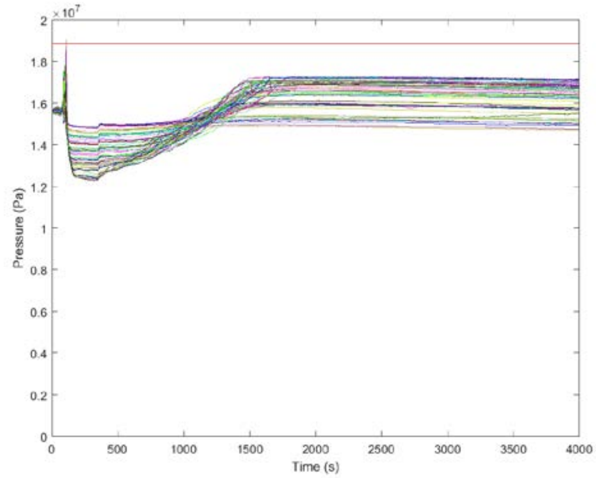

Fig. 4. PRZP in seq #1 for BEPU approach.

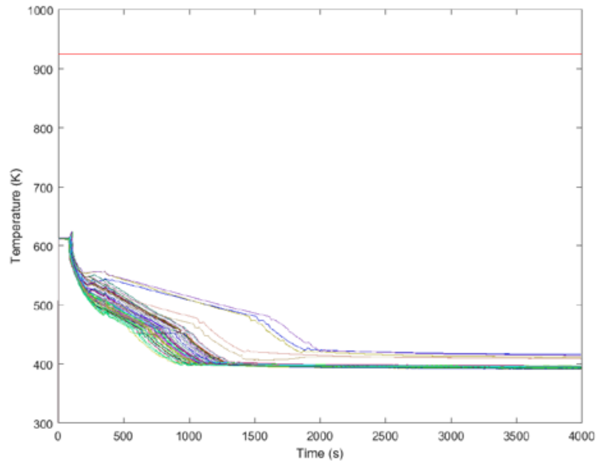

Fig. 5. PCT in seq #1 for EBEPU approach.

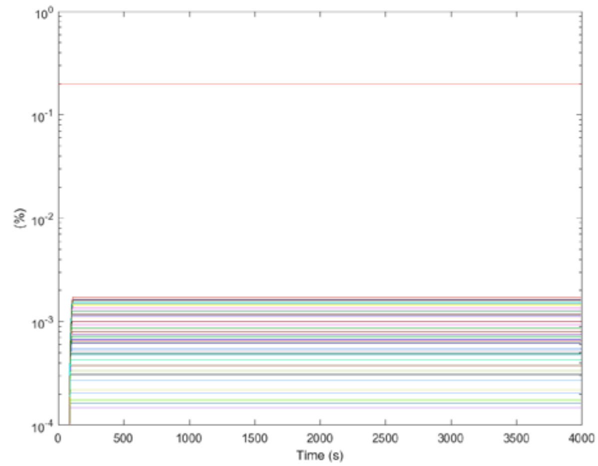

Fig. 6. CONTL in seq #1 for EBEPU approach.

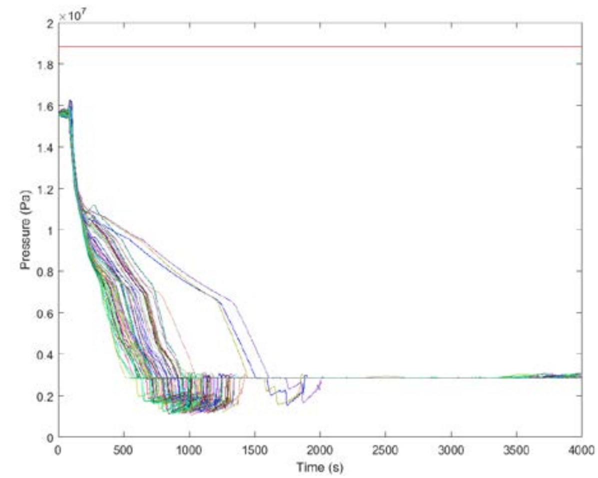

Fig. 7. PRZP in seq #1 for EBEPU approach.

Adopting the EBEPU approach, all FOMs fulfil the acceptance criteria, and therefore the safety criteria are.

However, under the BEPU approach, the primary pressure goes above the acceptance criteria in 1 out of 59 simulations, i.e. PRZ_MAX is above the acceptance criteria in one case (see Figure 4).

Comparing Figure 7 with its corresponding Figure 4, the former shows the evolution of the primary sys-tem pressure (PRZ) increase up to reach the set point of PORVs (SVs) valves, which open. After that, primary pressure drops abruptly, and therefore the PRZ_MAX is below the safety limit in all cases

V. CONCLUSIONS

The results of the application as proof of concept come to show that it is possible to address analysis of extension of design conditions of NPP departing from the results available from the NPP Safety Analysis (deterministic and probabilistic analyses), which can be integrated in a natural way adopting the EBEPU procedure proposed in Section 2.

Comparing the results of the safety analysis following option 4 (EBEPU) against those obtained under option 3 (BEPU) we can observe that EBEPU approach is more realistic than BEPU approach.

Main difference is observed what concerns PRZP_MAX, however, safety criteria are met in both cases anyway.

Note the probability of the only safety systems configuration considered under the BEPU approach is well below the probability of the realistic configurations of safety systems considered under the EBEPU approach as a result of adopting a very conservative assumption in the former case.

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