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Application Of The Generic And Specific SPAR-CSN Probabilistic Safety Assessment Models

Julia Herrero-Otero
Universidad Politécnica de Madrid
C. Alenza, 4
28040, Madrid, Spain
julia.herrero.otero@alumnos.upm.es

Enrique Meléndez, Miguel Sánchez-Perea
Consejo de Seguridad Nuclear (CSN)
C. Pedro Justo Dorado Dellmans, 11
28040, Madrid, Spain
ema@csn.es, msp@csn.es

César Queral, Marcos Cabezas, Sergio Courtin, Alberto García-Herranz, Carlos París
Universidad Politécnica de Madrid
C. Alenza, 4
28040, Madrid, Spain
cesar.queral@upm.es, sergio.courtin@upm.es

ABSTRACT

Oversight of licensee's performance should be done from an independent position. This position is best achieved when the regulatory body develops its own methodologies and tools. In particular, in the case of probabilistic safety analysis (PSA), the large number of hypotheses and assumptions behind the model makes it very difficult to perform sound regulatory analyses using licensee models. Therefore, the development of a PSA model to be used by the regulator provides a better understanding of the risks of nuclear power plants and an improvement of the regulatory practice, at the same time enhancing consistency of the risk perspective across regulatory actions.

In this context, the Spanish Regulatory Body (CSN), in collaboration with the Universidad Politécnica de Madrid (UPM), has been developing its own models: first a standardized generic Standardized Plant Analysis Risk model (SPAR-CSN) for a PWR-WEC, 3-loop design that serves as the basis for the development of a specific model of one of the Spanish PWR-WEC, 3-loop design. The development of both SPAR models has been applied to the analysis of two actual incidents, from a risk point of view. The results obtained have led to the identification of safety-critical systems and components.

1 INTRODUCTION

The Spanish regulatory body (CSN) together with the Universidad Politécnica de Madrid (UPM), have developed standardized PSA (SPAR-CSN) models for PWR-WEC 3-loop reactor designs. Firstly, a generic standard model to be used as the basis for subsequent specific models, of which one for a particular Spanish plant, PWR-WEC 3-loop has been completed. The project aims to have a standardized plant-independent PSA model to obtain a high-level view of the

risk and to be used for regulatory purposes. The intended scope is to be comparable to USNRC SPAR models ([1], [2]). This type of model improves the understanding of the risks of nuclear power plants and is considered an improvement of regulatory practice. The SPAR-CSN model aims to:

- Understand the main risk factors of the different nuclear power plants.
- Help define the areas of inspection and supervision tasks by using measures of the importance of the systems and components.
- Evaluate inspection performed within the Spanish regulatory system, SISC.
- Perform precursor analyses of events occurring at NPPs in Spain.
- Evaluate and identify the differences between the PSA models of the Spanish nuclear power plants, to help in the assessment and verification of those models.

The following sections summarize the results obtained with both models within the SPAR-CSN project. Section 2 briefly describes the two models, and their similarities and differences. Then, in section 3, we explain in detail two applications to event analysis, describing the reported event and the assumptions that have been used to model it. Finally, section 4 shows the results obtained in the applications.

2 DESCRIPTION OF THE SPAR-CSN MODELS

As mentioned in the introduction, the generic SPAR-CSN model has been developed to be the basis for the specific SPAR-CSN models of the Spanish plants with a PWR-WEC 3-loop design. Therefore, the configuration of the different systems and components as well as the assumptions taken for the modeling is a composite chosen to be representative of most of the plants. On the other hand, the specific SPAR-CSN model has been derived based on the generic model, making the pertinent changes to fit the design of a specific Spanish nuclear power plant. Both models have the same 14 main event trees (ETs, Table 1):

Table 1: List of ETs of the SPAR-CSN models

ID	Initiating event	ID	Initiating Event
LBLOCA	LOCA (> 6 in.)	**MSLB-US**	Main Steam Line Break upstream of MS isolation valves (MSIV)
MBLOCA	LOCA (2 in. to 6 in.)	**MSLB-DS**	Main Steam Line Break downstream of MSIV
SBLOCA	LOCA (3/8 in. to 2 in.)	**SGTR**	Steam Generator Tube Rupture
GT	General Transient	**LCWA**	Loss of a CW train
ATWS	Anticipated transient without scram	**LNSW**	Loss of Non-essential Service Water System
LC	Loss of Condenser	**LOOP/SBO**	Loss of Offsite Power/ Station Blackout
LDC-A	Loss of emergency DC-A bus	**LDC-B**	Loss of emergency DC-B bus

To define the different sequences of each ET, a mission time of 24 hours and two different consequences, success (S) or core damage (CD), are taken into account.

The two models also share the 18 systems (Table 2), but the modelling, (fault tree, FT: logic gates, basic events, human actions, dependencies, etc.) in some cases is different to fit the specific design.

Table 2: List of system FTs of the SPAR-CSN models

ID	System	ID	System
AC	Emergency AC distribution system	**IA**	Instrumentation Air System
AF	Auxiliary Feedwater System	**LH**	Low Pressure Injection System
AI	Accumulators Injection System	**MS**	Main Steam System
CW	Component Cooling Water System	**NC**	Non-essential Component Cooling Water
DC	Emergency DC distribution system	**NS**	Non-essential Service Water System
DG-A/B	Emergency Diesel Generators	**PR**	RCS Pressure Relief System
DG-SBO	SBO Diesel Generator	**RP**	Reactor Protection System
ES/SQ	ESFAS/Sequencer	**SC**	RCS Seal Injection System
HH	High Pressure Injection System	**SW**	Essential Service Water System

The main difference between the generic and the specific model is in the auxiliary feed water (AFW), the Table 3 shows the most important differences.

Table 3: Main differences between generic and specific AFW system

GENERIC MODEL	SPECIFIC MODEL
Individual injection lines for the tubine driven pump (TDP-AFW).	Interconnecting lines connecting the three pumps of the AFW.
Failure due to maintenance or testing of individual components	Failure due to maintenance or testing of sections and system trains
Human actions concerning the connections between lines are not included	Two new human actions are included concerning the connections between lines
Control and isolation valves: air-operated valve and manual valve	Control and isolation valves: two motor-operated valves

Regarding human reliability, both models use the SPAR-H methodology [3] to quantify the probability of the different type 3 human errors, following the methodology of the SPAR-NRC models. The generic SPAR-CSN model includes a total of 38 human actions and the specific model adds 4 additional ones, making a total of 42. The choice of the different human actions was based on the Emergency Operating Procedures (EOP's) of the different Spanish nuclear power plants. The different human reliability parameters have been obtained from actual data from Spanish plants.

To obtain the reliability parameter data for the different basic events related to system unavailability, equipment failures, and initiating events, public data have been used as a source: NUREG/CR-6928 [4], NUREG/CR-5497 [5].

3 SPAR-CSN MODEL APPLICATIONS: PRECURSOR ANALYSIS

3.1 Application 1: LOOP in a Twin-Unit NPP.

Following an earthquake near the site, both reactors tripped. The standby station service transformers (RSST) tripped, which caused a loss of off-site power (LOOP). The four DGs and the DG-SBO started automatically. All four DGs were correctly aligned to their respective safety bus. At the instant of the event, the TDP-AFW of Unit 1 was unavailable due to a surveillance test. In the course of the post-trip SCRAM actions, the operators stopped the test, restarted, and aligned the TDP-AFW to SG-A. 49 minutes after the LOOP occurred, DG 2H tripped due to a coolant leak, triggering the need to align the DG-SBO to Unit 2's 2H bus. Three hours later, one of the standby station service transformers was energized and returned to service to feed one of the Unit 1 buses. Nine hours after the initiating event, outside power was

restored to all 4 safety buses. Both units were safely shut down and stabilized in hot shutdown conditions [6].

Modelling assumptions for Unit 1:

- The probability of the LOOP initiator is set to 1 and the rest of the initiators to zero.
- Because of the unavailability of the TDP-AFW due to surveillance testing, a new basic event has been added: unavailability due to test or maintenance of the TDP-AFW during LOOP, with a probability equal to 1.
- After recovering the TDP-AFW it is necessary to restart and align it to the steam generators. To model this, a new basic event has been created: operator failure to restart and align the TDP-AFW with steam generators (SGs).
- The basic event of the TDP-AFW test unavailability model has been set as FALSE.
- Recovery of the external current before 180 minutes has been assumed.
- DG-SBO is not available.
- Battery depletion time in the model: 5.2 hours.
- The mission duration in the model is 24 hours.

3.2 Application 2: Failure of the TDP-AFW.

While the plant was operating at 100% power, the reactor protection system actuated automatically due to an electronic card failure. During this event, the auxiliary feed water was automatically started, but the TDP-AFW stopped due to overspeed and it was not possible to start the TDP-AFW again.

Modelling assumptions for Unit 1:

- When the reactor protection system is activated automatically, only the generic transient initiator is analyzed, for which its probability has been set as 1 and the rest of the initiators as 0.
- SCRAM reactor succeeds.
- Due to the failure of the TDP-AFW caused by overspeed, the basic events: failure in the operation of the TDP-AFW (first hour) and failure in the operation of the TDP-AFW (>1h) have been set to TRUE.
- According to the Standard Technical Specifications (STS), the reactor was shut down after 72 hours because the TDP-AFW could not be recovered in time, To model this, a new basic event has been created: operator failure on TDP-AFW restart on time.

4 COMPARISON OF THE RESULTS

An analysis of the two events with both models has been carried out below.

4.1 Application 1

The dominant sequence for both models corresponding to this reported event is number 10, contributing 90.62% in the generic model (conditional core damage probability, CCDP=1.63E-04) and 87.31% (CCDP=2.90E-04) in the specific model. Figure 1 shows the event tree of the LOOP initiating event, where this sequence is represented:

- Initiating event: LOOP
- Reactor protection system successful (Z)
- Failure of power supply from DG-A and DG-B (DG)

- Failure of the auxiliary feed water system (AFW)
- Failure of external electrical power recovery (R-EX)

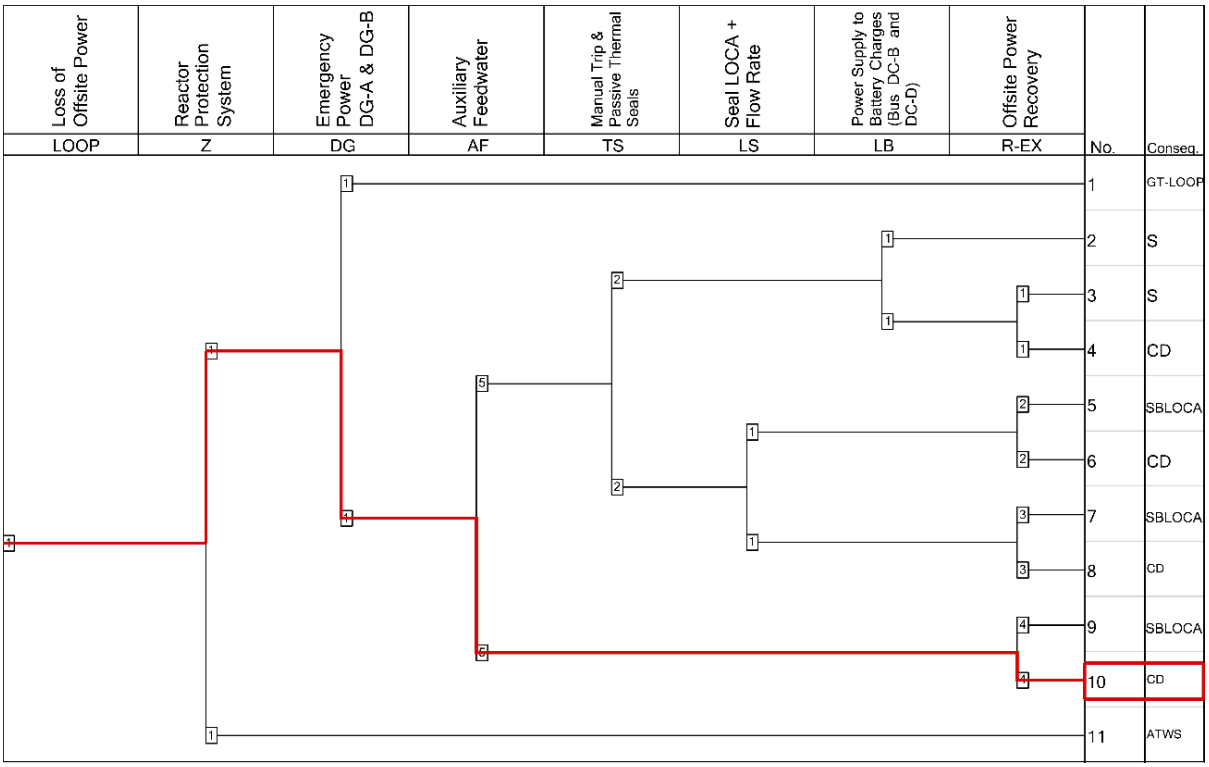

Figure 1: LOOP event tree. Dominant sequence in bold.

Table 4 shows the minimum dominant fault sets obtained with the generic model for the LOOP, with a total CCDP of 1.66E-04. First MCS corresponds to sequence 1 of transfer to GT-LOOP. The remaining MCS correspond to the dominant sequence (S10).

Table 4: Dominant minimal cut-set (MCS) for LOOP, Generic Model

Nº	**CCDP**	**%**	**Events**
1	7.15E-06	4.3	The operator fails to control the SGs level; the operator fails to initiate Feed & Bleed
2	6.91E-06	4.2	TDP-AFW operation failure (>1h); and common cause failure (CCF) in the operation of DG-A and DG-B
3	6.65E-06	4.0	TDP-AFW operation failure (>1h); unavailability for maintenance or testing of the DG-A; and DG-B operation failure
4	6.65E-06	4.0	TDP-AFW operation failure (>1h); DG-A operation failure; and unavailability for maintenance or testing of the DG-B
5	6.65E-06	3.7	TDP-AFW operation failure (>1h); DG-A and DG-B operation failure
6	4.87E-06	2.9	CCF at the opening of circuit breakers; and TDP-AFW operation failure (>1h)
7	4.21E-06	2.5	Unavailability for testing or maintenance TDP-AFW during LOOP; CCF in the operation of DG-A and DG-B; and Operator failure in restarting and aligning TDP-AFW with SGs
8	4.06E-06	2.5	Unavailability for testing or maintenance TDP-AFW during LOOP; DG-A operation failure; Unavailability for maintenance or testing of the DG-B; and Operator fails in restarting and aligning TDP-AFW with SGs
9	4.06E-06	2.5	Unavailability for testing or maintenance TDP-AFW during LOOP; Unavailability for maintenance or testing of the DG-A; DG-B operation failure; and Operator fails in restarting and aligning TDP-AFW with SGs
10	3.76E-06	2.3	Unavailability for testing or maintenance TDP-AFW during LOOP; DG-A operation failure; DG-B operation failure; and Operator fails in restarting and aligning TDP-AFW with SGs

The data obtained with the specific model are different than the generic model, a total CCDP of 3.32E-04 has been obtained, the minimum failure sets are shown below, Table 5:

Table 5: Dominant MCS for LOOP Specific Model

Nº	CCDP	%	Events
1	2.51E-05	7.5	Operator fails to control the SGs level; the operator fails to initiate Feed & Bleed
2	1.31E-05	4.0	TDP-AFW operation failure (>1h); unavailability for maintenance or testing of room cooler; and unavailability for maintenance or testing of room cooler.
3	9.71E-06	2.9	TDP-AFW operation failure (>1h); unavailability for maintenance or testing of room cooler; and unavailability for maintenance or testing of DG-B.
4	9.71E-06	2.9	TDP-AFW operation failure (>1h); unavailability for maintenance or testing of room cooler; and unavailability for maintenance or testing of DG-A.
5	8.99E-06	2.7	TDP-AFW operation failure (>1h); DG-B operation failure; and unavailability for maintenance or testing of room cooler.
6	8.99E-06	2.7	TDP-AFW operation failure (>1h); DG-A operation failure; and unavailability for maintenance or testing of room cooler.
7	8.00E-06	2.4	Unavailability for testing or maintenance of TDP-AFW during LOOP, unavailability for maintenance or testing of room cooler and unavailability for maintenance or testing of room cooler, and operator failure to restart and align TDP-AFW with SGs.
8	6.91E-06	2.1	TDP-AFW operation failure (>1h); and CCF in the operation of DG-A and DG-B
9	6.65E-06	2.0	TDP-AFW operation failure (>1h); unavailability for maintenance or testing of the DG-A and DG-B operation failure
10	6.65E-06	2.0	TDP-AFW operation failure (>1h); DG-A operation failure and unavailability for maintenance or testing of the DG-B

As can be seen in the table, the specific model results in a higher value than the generic model, but both are in the same order of magnitude, Table 6.

Table 6: CCDP LOOP comparison between models.

CCDP	Generic	Specific
LOOP	1.66E-04	3.32E-04

The reason for this difference is that the specific model includes the control room air-conditioning unit failure (room cooler) which considerably increases the probability value, while the generic model does not include it.

4.2 Application 2

The dominant sequence corresponding to this reported event is number 6, contributing 69.29% (CCDP=8.49E-06) in the generic model and 87.75% (CCDP=2.68E-05) in the specific model. Figure 2 shows the event tree of the GT initiating event, where this sequence is represented:

- Initiating event: GT
- Reactor protection system successful (Z)
- Failure of the auxiliary feed water system (AFW)
- Feed & Bleed failure (HM+BM)

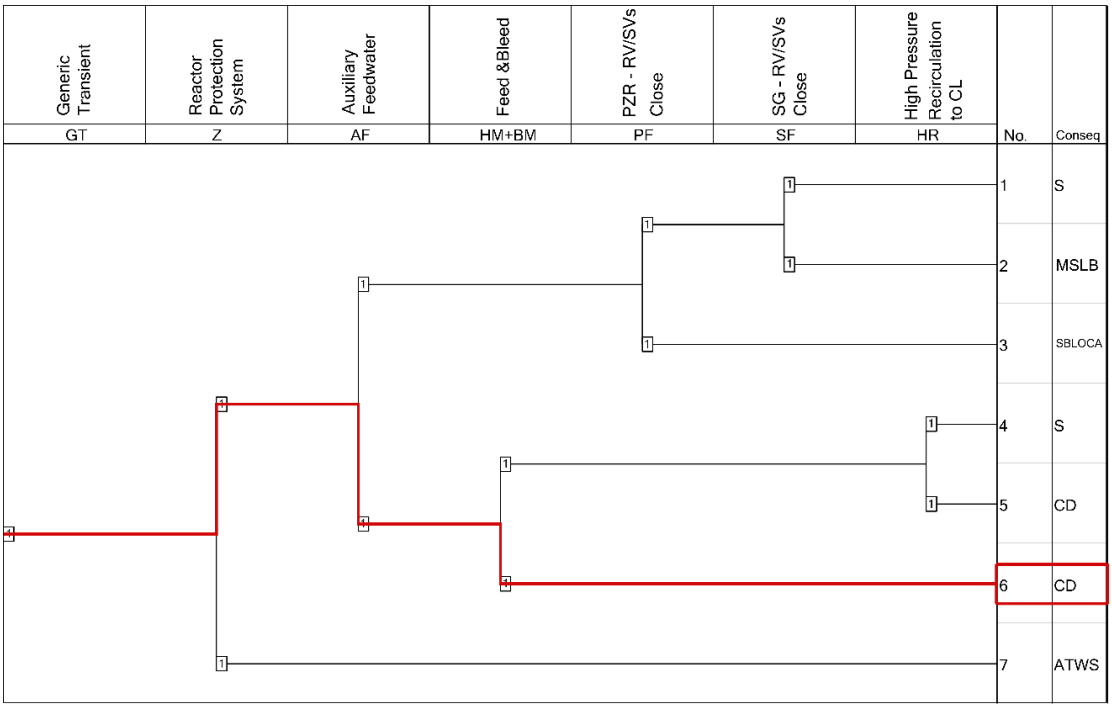

Figure 2: GT event tree. Dominant sequence in bold.

Table 7 shows the minimal cut sets obtained with the generic model for the GT, for the second application, with a total CCDP of 1.22E-05.

Table 7: Dominant MCS for GT, Generic Model.

No.	CCDP	%	Events
1	7.15E-06	58.7	The operator fails to control the SGs level; the operator fails to initiate Feed & Bleed
2	2.51E-06	20.6	The operator fails to control the SGs level; the operator fails to initiate the high-pressure cold leg (CL) recirculation
3	2.92E-07	2.4	CCF of pump cooling units; and operator fails to initiate Feed & Bleed
4	2.21E-07	1.8	PORV-1 fails to close; the operator fails to initiate the high-pressure CL recirculation; the operator fails to reduce the safety injection flow
5	2.21E-07	1.8	PORV-3 fails to close, the operator fails to initiate the high-pressure CL recirculation and the operator fails to reduce the safety injection flow
6	2.21E-07	1.8	PORV-2 fails to close, the operator fails to initiate the high-pressure CL recirculation and the operator fails to reduce the safety injection flow

As in the previous application, the specific model obtained different results from the generic model, Table 8, giving a total CCDP of 3.92E-05.

Table 8: Dominant MCS for GT, Specific Model.

No.	CCDP	%	Events
1	2.51E-05	81.9	The operator fails to control the SGs level; the operator fails to initiate Feed & Bleed
2	2.51E-06	8.2	The operator fails to control the SGs level; the operator fails to initiate the high-pressure CL recirculation
3	2.92E-07	1.0	CCF of pump cooling units; and operator fails to initiate Feed & Bleed

In this case, as can be seen in Table 9, the specific model gives a higher result than the generic model, as in the previous case.

Table 9: CCDP GT comparison between models.

CCDP	Generic	Specific
GT	1.22E-05	3.06E-05

This is because the value of the probability of failure in dependent operator actions is higher in the specific model than in the generic model.

5 CONCLUSIONS

After these analyses, the main conclusions obtained are as follows:

- The standardized models allow modeling of different plant designs starting from the specific SPAR-CSN model.
- The generic SPAR-CSN model has shown that it is feasible to standardize the ETs, FTs, success criteria, and human actions for the different Spanish nuclear plants in a single model.
- The use of both SPAR-CSN models allows for identifying the main differences between the generic model and the specific model. These differences are found in the ETs (different success criteria), the FTs (different modeling of the systems), and the modeling assumptions, as well as the results of the human actions dependence analysis.
- Both models have been applied to the analysis of two incidents. The results obtained in both models are similar and are in agreement with each other and with other SPAR models.

The SPAR-CSN models are very important tools for evaluating and understanding the different operating risks of the Spanish nuclear power plants. The standardized construction of these models in terms of modeling provides a consistent model on which to base the assessment of the risk impacts of individual plants. The CSN's future objective for the SPAR-CSN models is for them to serve as tools for other types of applications, such as precursor analysis (as seen in this article), prioritization in inspection and surveillance tasks, or the evaluation of inspection results. In fact, recently the SPAR-CSN model has been applied as well to design a comprehensive methodolgy suitable to identify the so-called DEC-A sequences (i.e., Design Extension Conditions without core damage) based on PSA quantitative information [7].

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