

UNIVERSIDAD POLITÉCNICA DE MADRID
Escuela Técnica Superior de Ingenieros Industriales



**Application of Probabilistic Safety Assessment
to FLEX Strategies, Advanced Technology
Fuels and Design Extension Conditions**

DOCTORAL THESIS

Submitted for the degree of Doctor by:

Sergio Courtin Vega

Master in Nuclear Engineering by the Universidad Politécnica de Madrid

Mechanical Engineer by the Universidad de Chile

Madrid, 2024



UNIVERSIDAD POLITÉCNICA DE MADRID
Escuela Técnica Superior de Ingenieros Industriales

Doctoral Degree in Sustainable, Nuclear and Renewable Energy

**Application of Probabilistic Safety Assessment
to FLEX Strategies, Advanced Technology
Fuels and Design Extension Conditions**

DOCTORAL THESIS

Submitted for the degree of Doctor by:

Sergio Courtin Vega

Master in Nuclear Engineering by the Universidad Politécnica de Madrid
Mechanical Engineer by the Universidad de Chile

Under the supervision of:
Dr. César Queral Salazar (Supervisor)

Madrid, 2024

Title: Application of Probabilistic Safety Assessment to FLEX Strategies, Advanced Technology Fuels and Design Extension Conditions

Author: Sergio Courtin Vega

Doctoral Programme: Sustainable, Nuclear and Renewable Energy

Thesis Supervision:

Dr. César Queral Salazar, Professor of Nuclear Engineering at UPM (Supervisor)

External Reviewers:

Thesis Defense Committee:

Thesis Defense Date:

A mi padre, hace muchos años plantó una semilla que hoy da frutos.

A mi madre, compañera silenciosa de todas mis aventuras.

“A nuclear power plant is infinitely safer than eating because 300 people choke to death on food every year.”
-Dixie Lee Ray

Acknowledgement

En primer lugar, deseo expresar mi profundo agradecimiento al Profesor César Queral, no sólo por los conocimientos enseñados y el tiempo que dedicó a esta tesis, también por los momentos fuera del despacho y su gran calidad humana. Agradezco también al Consejo de Seguridad Nuclear por la financiación de este proyecto y, en particular, a Miguel Sánchez Perea y Enrique Meléndez Asensio, siempre dispuestos a colaborar y a entregar comentarios precisos y constructivos. A mis compañeros de despacho, especialmente a los que trabajaron conmigo en el proyecto SPAR-CSN: Julia Herrero, Marcos Cabezas y Alberto García Herranz, con quienes compartimos trabajo, copas y pochismo. A los profesores de la UPM Gonzalo Jiménez y Gonzalo García, quienes me dieron una oportunidad cuando más lo necesitaba. A mis amigos en Chile, Cristian y Vale, lejanos en distancia física, pero siempre cercanos en sus preocupaciones y apoyos, y a Marcelo Valenzuela, quejumbroso pero generoso como pocos. Finalmente, pero no menos importante, agradezco a mi pareja y compañera de aventuras, María Paz, por el apoyo, amor y cariño, sin ella, todo esto habría sido mucho más difícil.

Abstract

The Fukushima Dai-ichi accident left important lessons for nuclear safety. The first one was the importance of the challenge presented by the loss of safety-related systems after a Beyond Design-Basis External Event (BDBEE). In particular, Extended Loss of AC Power (ELAP) or Loss of Ultimate Heat Sink (LUHS) can severely compromise key safety functions associated with core cooling and containment, ultimately leading to reactor core damage. Because of that the so-called Diverse and Flexible Coping Strategies (FLEX) were designed. Also, nuclear power plants (NPPs) were required to demonstrate their reliability by increasing their safety provisions and equipment to mitigate accidental sequences. These requirements must cover equipment, procedure improvements, and regulatory development. One of the most important actions has been the introduction of the analysis of the Design Extension Conditions (multiple failure conditions outside the design basis envelope, DEC) into the design requirements of NPPs to further improve safety. And finally, several weak points in the conventional nuclear fuel designs were come into light, in response to this, the so-called Accident Tolerant Fuels (ATF) represents some potential to increase the capabilities of the NPPs against accidental scenarios that may lead to a core damage situation. On this aspect, ATF cladding materials, due to an improved oxidation resistance compared to the currently standardized Light Water Reactor fuels, allow to increase the available times to perform the human actions in different accidental conditions.

By using a Probabilistic Safety Assessment (PSA) generic model (3-loops, PWR-WEC) developed by the Universidad Politécnica de Madrid (UPM) in collaboration with the Spanish Nuclear Regulatory Body (CSN), SPAR-CSN, this work quantifies the impact of FLEX, these aforementioned FLEX strategies are useful to avoid core damage in sequences due to internal events. Therefore, these FLEX strategies should be incorporated in the Level 1 Probabilistic Safety Assessment (L1PSA) to analyze their impact on core damage frequency (CDF). As a result, a decrease of the CDF at power for internal events is observed and, therefore, indicates that these FLEX strategies have a positive impact on the plant safety not only for BDBEE but also for other accidental conditions.

Also, a novel methodology has been developed to identify sequences that are outside the design basis but not reaching fuel damage, DEC-A. Using this method, it has been possible to identify the DEC-A sequences with high relative risk by comparing their frequencies with their corresponding DEC-B sequences (with fuel damage). As shown in this study, the addition of post-Fukushima improvements highly reduces the relative risk of the DEC sequences.

Finally, the impact of the ATF implementation is analyzed by means of the same SPAR-CSN model, using the available time extension associated with the usage of the ATF. As an application of this model and supported by the correspondent thermal-hydraulic simulations and references, the enhanced available time of the main human actions were included in the model and then the reduction of the risk has been quantified in order to eventually evaluate the safety margins. The results compiled in this work show a reduction in the CDF.

Resumen

El accidente de Fukushima Dai-ichi dejó importantes lecciones para la seguridad de las centrales nucleares. La primera fue la importancia del reto que supone la pérdida de sistemas relacionados con la seguridad tras un suceso externo más allá de la base de diseño (BDBEE). En concreto, la Pérdida Prolongada de Potencia Exterior (ELAP) o la Pérdida del Sumidero de Último Calor (LUHS) pueden comprometer gravemente las funciones de clave de seguridad asociadas a la refrigeración y la contención del núcleo, lo que en última instancia puede provocar daños en el núcleo del reactor. Debido a esto, se diseñaron las llamadas Estrategias Diversas y Flexibles (FLEX). Asimismo, se exigió a las centrales nucleares que demostraran su fiabilidad aumentando sus dispositivos y equipos de seguridad para mitigar secuencias accidentales. Estos requisitos abarcan el equipamiento, la mejora de los procedimientos y el desarrollo normativo. Una de las acciones más importantes ha sido el desarrollo del análisis de las Condiciones de Extensión del Diseño (condiciones de fallo múltiple fuera de la base de diseño, DEC) en los requisitos de diseño de las centrales nucleares para mejorar aún más la seguridad. Por último, el accidente de Fukushima Dai-ichi también ha sacado a la luz varios puntos débiles en los diseños de los combustibles nucleares convencionales, en respuesta a lo cual, los denominados Combustibles Tolerantes a Accidentes (ATF) poseen el potencial para aumentar las capacidades de las centrales nucleares frente a escenarios que puedan conducir a una situación de daño del núcleo. En este aspecto, los materiales de revestimiento ATF, debido a una mayor resistencia a la oxidación en comparación con los combustibles de reactores de agua ligera actualmente utilizados, permiten aumentar los tiempos disponibles para llevar a cabo acciones humanas que permiten hacer frente a diversas condiciones accidentales.

Con el objetivo de analizar el impacto de estas mejoras se ha utilizado el modelo genérico de Análisis Probabilista de Seguridad (3 lazos, PWR-WEC, SPAR-CSN) desarrollado por la Universidad Politécnica de Madrid (UPM) en colaboración con el Consejo de Seguridad Nuclear (CSN), para cuantificar la implementación de la estrategias FLEX, dichas estrategias son útiles para evitar el daño al núcleo en secuencias accidentales debidas también a sucesos internos. Por lo tanto, estas estrategias FLEX deberían incorporarse al Análisis Probabilista de Seguridad de Nivel 1 (L1PSA) para analizar su impacto en la frecuencia de daño al núcleo (FDN). Como resultado, se observa una disminución de la FDN a potencia para sucesos internos y, por tanto, indica que estas estrategias FLEX tienen un impacto positivo en la seguridad de la central no sólo para BDBEE sino también para otras condiciones accidentales.

Asimismo, se ha desarrollado una nueva metodología para identificar secuencias que están fuera de la base de diseño pero que no alcanzan la condición de daño al combustible, DEC-A. Utilizando este método, ha sido posible identificar las secuencias DEC-A con alto riesgo relativo comparando sus frecuencias con sus correspondientes secuencias DEC-B (con daño en el combustible). Como se muestra en este estudio, la adición de mejoras posteriores a Fukushima reduce en gran medida el riesgo relativo de las secuencias DEC.

Por último, se ha analizado el impacto de la implantación de combustible ATF mediante el mismo modelo SPAR-CSN, utilizando la ampliación del tiempo disponible asociada al uso de este nuevo tipo de combustible. Como aplicación de este modelo y con el apoyo de las correspondientes simulaciones termohidráulicas y de un análisis de referencias, se ha incluido en el modelo la ampliación del tiempo disponible de las principales acciones humanas y se ha cuantificado la reducción del riesgo para finalmente evaluar los márgenes de seguridad. Los resultados recopilados en este trabajo muestran una reducción de la FDN.

Contents

Acknowledgement	iv
Abstract	v
Resumen	vi
List of Figures	xiii
List of Tables	xvi
Abbreviations	xvii
Publications	xix
1 Introduction	1
1.1 Thesis Objectives	3
2 Standardized Plant Analysis Risk Models	5
2.1 Description of the Generic SPAR-CSN Model	7
2.1.1 Event Trees for Internal Events at Full Power	7
2.1.2 Fault Trees	8
2.1.3 Human Reliability	16
2.1.4 Data Sources	18
2.1.4.1 Initiators Frequency	18
2.1.4.2 Basics Events	20
2.1.4.3 Common Cause Failures	24
2.1.4.4 Example of AFW system failures	27
2.2 Generic SPAR-CSN Model Results	28
3 Application of Probabilistic Safety Assessment to FLEX Strategies	31
3.1 FLEX Strategies and Portable Equipment	33
3.2 Extensive Damage Mitigation Guidelines and FLEX Support Guidelines in USA and Spain	38
3.3 FLEX Strategies Modeling	43
3.3.1 Event Trees Including FLEX Headers	46
3.3.2 Fault Trees Including FLEX Equipment	53
3.3.3 Data for FLEX Equipment	54
3.3.4 Human Reliability Analysis Applied to FLEX Strategies	55
3.3.4.1 SPAR-H	57
3.4 Impact of FLEX Strategies on the Risk Reduction	64
3.5 Sensitivity Analysis of the FLEX Strategies	66

4	Application of Probabilistic Safety Assessment for the Selection of DEC-A Sequences	69
4.1	DEC-A Identification Methodology and Implementation on the SPAR-CSN Model	75
4.2	Post-Fukushima Improvements in DEC-A Sequences	83
4.3	Risk Reduction of DEC-A Sequences Due to Post-Fukushima Improvements	87
4.4	Comparison with Other DEC-A Improvements Included in Several Technologies	89
5	Application of Probabilistic Safety Assessment to Evaluate the Impact of ATF	93
5.1	Advanced Technology Fuels	95
5.2	ATF Impact on L1PSA	98
6	Conclusions	103
6.1	SPAR-CSN Conclusions	104
6.2	FLEX Conclusions	105
6.3	DEC-A Conclusions	105
6.4	ATF Conclusions	106
	References	118
A	SPAR-CSN System Diagrams	119
B	SPAR-CSN Event Trees	141
C	SPAR-CSN Event Trees with FLEX Strategies	155
D	DEC-A Sequences Event Trees	161

List of Figures

2.1	LPSI simplified diagram of the SPAR-CSN generic model. Injection to CLs mode	10
2.2	LPSI simplified diagram of the SPAR-CSN generic model. Recirculation to CLs mode	11
2.3	Top Event of LPSI Fault Tree of the SPAR-CSN generic model (Cold Leg Injection and Recirculation modes)	12
2.4	Structure of the LH FT	13
2.5	One-line diagram of the SPAR-CSN model and its relation to the LPSI . . .	14
2.6	GT initiator event data [NRC, 2015c]	19
2.7	LOOP initiator event data [NRC, 2015c]	19
2.8	SGTR initiator event data [NRC, 2015c]	20
2.9	AFW MDP Fail to start (demand): MDP-FTS-AFW from NUREG/CR-6928	21
2.10	MDP-FTS-AFW modeling in RiskSpectrum	22
2.11	MDP-FTR-AFW (mission) from NUREG/CR-6928	22
2.12	MDP-FTR-AFW modeling in RiskSpectrum	23
2.13	MDP-AFW CCFs modeling in RiskSpectrum	26
2.14	Contribution of each IE to the overall CDF of the SPAR-CSN model, not considering human actions dependencies	30
2.15	Contribution of each IE to the overall CDF of the SPAR-CSN model, considering human actions dependencies	30
3.1	FLEX time-line	32
3.2	FLEX Enhances Defense-in-Depth [NEI, 2012]	34
3.3	FLEX equipment at the secure site of Vogtle NPP, USA [Southern Nuclear, 2021]	35
3.4	FLEX dome at North Anna NPP in Virginia [Breunig, 2017]	36
3.5	Portable diesel generator FLEX-DG for use in FLEX strategies [ANAV, 2013]	36
3.6	Submersible portable pump FLEX-SUP for use in FLEX strategies [ANAV, 2013]	38
3.7	Procedures and guidelines relationships in Spain [Queral and Alonso-Escos, 2023]	42
3.8	Diagram of FLEX-SGP injection to SG (FA-1)	44
3.9	Diagram of DDP-FPS injection to SG (FA-2)	44
3.10	Diagram of FLEX-MUP injection to RCS (FA-4/5)	45
3.11	Diagram of FLEX-HCP/DDP-FPS injection to RWST (FA-6/7)	45
3.12	LOOP-SBO ET with FLEX strategies	48

3.13	GT ET with FLEX strategies	49
3.14	SBLOCA ET with FLEX strategies	50
3.15	SGTR ET with FLEX strategies	51
3.16	MSLB ET with FLEX strategies	52
3.17	Top event FT corresponding to AF Header	53
3.18	Example of the FLEX equipment FT modeling	54
4.1	Flowchart of the PSA-based DEC-A identification method.	77
4.2	Amount of DEC-A sequences in the DEC-A identification method steps. . .	78
4.3	DEC-A and DEC-B sequences for GT	79
4.4	DEC-A and DEC-B sequences for SBLOCA	80
4.5	DEC-A and DEC-B sequences for GT with FLEX strategies	85
4.6	DEC-A and DEC-B sequences for SBLOCA with FLEX strategies	86
5.1	ATF roadmap to readiness [NRC, 2024]	95
5.2	PCT evolution under an SBO in a PWR (zirconium vs <i>FeCrAl</i> , MELCOR code) [Doncel et al., 2024]	98
A.1	SPAR-CSN diagram of HPSI: Working as CVCS in normal operation	120
A.2	SPAR-CSN diagram of HPSI: High Pressure Injection to Cold Legs	121
A.3	SPAR-CSN diagram of HPSI: High Pressure Recirculation to Cold Legs . . .	122
A.4	SPAR-CSN diagram of the AFW: Injection to SGs	123
A.5	SPAR-CSN diagram of the AFW: main steam extraction lines for the supply to the turbine of the AF-PT-PT turbo-pump	124
A.6	SPAR-CSN diagram of the Essential Service Water System (ESWS)	125
A.7	SPAR-CSN diagram of the CCWS	126
A.8	SPAR-CSN diagram of CCWS: LPSI cooling	127
A.9	SPAR-CSN diagram of CCWS: HPSI cooling	128
A.10	SPAR-CSN diagram of AIS: Injection of Accumulators to CLs	129
A.11	SPAR-CSN diagram of Air System Diesel Generator A	130
A.12	SPAR-CSN diagram of Fuel System Diesel Generator A	131
A.13	One-line diagram of the SPAR-CSN model and its relation to the DG	132
A.14	One-line diagram of the Alternating Current Safeguard System of the SPAR-CSN model	133
A.15	One-line diagram of the SPAR-CSN model 6.9 kV, 400 V (AC) Alternating Current Safeguard System	134
A.16	One-line diagram of the electrical connections in the SPAR-CSN model at SBO	135
A.17	SPAR-CSN diagram of the PORV-PZR	136
A.18	SPAR-CSN diagram of the RPS	137
A.19	SPAR-CSN diagram of the MS	138
A.20	SPAR-CSN diagram of the MS (Condenser relief function)	139
B.1	LBLOCA ET in SPAR-CSN Model	142
B.2	MBLOCA ET in SPAR-CSN Model	143
B.3	SBLOCA ET in SPAR-CSN Model	144
B.4	GT ET in SPAR-CSN Model	145

B.5	ATWS ET in SPAR-CSN Model	146
B.6	LC ET in SPAR-CSN Model	147
B.7	MSLB-DS ET in SPAR-CSN Model	148
B.8	SGTR ET in SPAR-CSN Model	149
B.9	LCWA ET in SPAR-CSN Model	150
B.10	LNSW ET in SPAR-CSN Model	151
B.11	LOOP ET in SPAR-CSN Model	152
B.12	LDC-A ET in SPAR-CSN Model	153
C.1	ATWS ET with FLEX strategies	156
C.2	LC ET with FLEX strategies	157
C.3	LCWA ET with FLEX strategies	158
C.4	LDC-A ET with FLEX strategies	159
C.5	MBLOCA ET with FLEX strategies	160
D.1	DEC-A Sequences: D1 sequences	162
D.2	DEC-A Sequences: D2 and D9 sequences	163
D.3	DEC-A Sequences: D3 sequence	164
D.4	DEC-A Sequences: D4 and D10 sequences	165
D.5	DEC-A Sequences: D5 sequences	166
D.6	DEC-A Sequences: D6 sequence	167
D.7	DEC-A Sequences: D7 sequences	168
D.8	DEC-A Sequences: D8 sequence	169

List of Tables

2.1	List of ETs of the SPAR-CSN generic model	8
2.2	List of system FTs of the SPAR-CSN Model	8
2.3	LPSI Dependencies matrix	15
2.4	Human Failure Events in the SPAR-CSN Model	17
2.5	Initiator frequency in SPAR-CSN Model	20
2.6	Failure modes for valves in the SPAR-CSN Model	23
2.7	Alpha factors distributions for MDP-FTS-AFW [NRC, 2015a]	25
2.8	Alpha factor and MGL Parameters for MDP-FTS-AFW	25
2.9	Alpha factor for MDP-FTS-AFW in SPAR-CSN Model	26
2.10	CCF for AFW System	27
2.11	AFW System equipment failure probabilities	28
2.12	CDF associated to every IE and overall CDF of the SPAR-CSN generic model. not considering human actions dependencies	29
2.13	CDF associated to every IE and overall CDF of the SPAR-CSN generic model. considering human actions dependencies	29
3.1	Portable equipment in PWR-WEC US and Spain NPPs	37
3.2	Entries to FSG procedures [Queral and Alonso-Escos, 2023]	42
3.3	FLEX Actions and relationship with FSGs and EDMGs (Spanish NPPs)	46
3.4	FLEX headers included in each ET (M: Modified; N: New)	47
3.5	FLEX equipment failure probabilities (also used in SPAR-CSN model)	55
3.6	PSFs and multipliers of SPAR-H [Gertman et al., 2005]	58
3.7	SPAR-H adapted for FLEX scenarios [Courtin et al., 2021] (1/2)	61
3.8	SPAR-H adapted for FLEX scenarios [Courtin et al., 2021] (2/2)	62
3.9	Comparison of HEPs for FLEX-DG	62
3.10	Comparison of HEPs for FLEX pumps	63
3.11	HEPs used in SPAR-CSN Model (modified SPAR-H)	63
3.12	CDF reduction for IE with FLEX strategies	64
3.13	SPAR-CSN MCS without FLEX strategies	65
3.14	SPAR-CSN MCS with FLEX strategies	65
3.15	Comparison of CDF reduction including FLEX strategies	66
3.16	FLEX strategies considered in the analysis	67
3.17	FLEX strategies contribution on the risk reduction	67
3.18	FLEX strategies contribution on the risk reduction for every IE	68

4.1	DEC-A sequences by country or design (1/2)	73
4.2	DEC-A sequences by country or design (2/2)	74
4.3	List of DEC-A sequences identified in the SPAR-CSN model	82
4.4	Dominant human actions associated to identified DEC-A sequences	83
4.5	Relationship between DEC-A Sequences and post-Fukushima improvements	84
4.6	Risk reduction in DEC sequences with FLEX strategies and SDS	88
5.1	Evolution of the main objectives pursued regarding ATF designs	96
5.2	Classification of ATF fuel rods	97
5.3	Increase in available time included in PSA model	99
5.4	Impact of ATF (Cr-Coated and <i>FeCrAl</i>) implementation on HEPs	100
5.5	CDF variation caused by ATF implementation (Cr-Coated and <i>FeCrAl</i>) . .	101
5.6	Model-wide CDF variation caused by the ATF application	101

Abbreviations

AFW	Auxiliary Feedwater System
ATF	Accident Tolerant Fuel
BDBEE	Beyond Design-Basis External Event
CD	Core Damage
CDF	Core Damage Frequency
CST	condensate storage tank
DEC	Design Extension Conditions
EDMGs	Extensive Damage Mitigation Guidelines
ELAP	Extended Loss of AC Power
EOPs	Emergency Operating Procedures
ET	Event Trees
F&B	Feed & Bleed
FLEX	Diverse and Flexible Coping Strategies
FSGs	FLEX Support Guidelines
FT	Fault Trees
HEP	Human Error Probabilities
HFE	Human Failure Events
HPSI	High-Pressure Safety Injection
HRA	Human Reliability Analysis
L1PSA	Level 1 Probabilistic Safety Assessment
LOCA	Loss of Coolant Accident
LOLA	Loss of Large Areas
LPSI	Low-Pressure Safety Injection
LUHS	Loss of Ultimate Heat Sink
NEI	U.S. Nuclear Energy Institute
NPP	Nuclear Power Plant

NRC U.S. Nuclear Regulatory Commission
PSA Probabilistic Safety Assessment
PWR Pressurized Water Reactor
PWROG Pressurized Water Reactor Owners Group
RCP Reactor Coolant Pump
RHR Heat Removal System
RWST Refueling Water Storage Tank
SBO Station Blackout
SDS Passive Thermal Shutdown Seals
SG Steam Generator
SPAR Standardized Plant Analysis Risk

Publications

The journal papers related to the doctoral thesis are:

1. **Sergio Courtin**, César Queral, Alberto García-Herranz, Risk reduction by means of FLEX strategies in pressurized water reactors, Nuclear Engineering and Design, Volume 414, 2023, 112656, ISSN 0029-5493, <https://doi.org/10.1016/j.nucengdes.2023.112656> JCR: Q1; SCi: Q1.
2. S. Martorell, I. Martón, A. Sánchez, **S. Courtin**, C. Queral, E. Meléndez, F. Olivar, A. García-Zamorano, R. Marín, P. Díaz, A. Serrano, A. Rísquez, B. Gonzalez, M. Saiz, C. Aramayo, C. Paris, R. Herrero, J.M. Martín-Valdepeñas, B. Gil, Spanish contribution to the development and application of probabilistic safety analysis: Experience and future challenges, Nuclear Engineering and Design, Volume 418, 2024, 112879, ISSN 0029-5493, <https://doi.org/10.1016/j.nucengdes.2023.112879> JCR: Q1; SCi: Q1.
3. **Sergio Courtin**, César Queral, Alberto García-Herranz, Miguel Sánchez-Perea, Enrique Meléndez, DEC-A identification and impact of post-Fukushima improvements on risk reduction, Nuclear Engineering and Design, Volume 428, 2024, 113476, ISSN 0029-5493, <https://doi.org/10.1016/j.nucengdes.2024.113476> JCR: Q1; SCi: Q1.

Conferences papers:

- “Applying the SPAR-H Human Reliability Methodology to FLEX Strategies”. **Sergio Courtin**, César Queral. 19th International Topical Meeting on Safety Assessment and Analysis (PSA 2025), June 15–18, 2025. Chicago, IL USA. **ACCEPTED**.
- “Risk-Informed Analysis of the Application of Accident Tolerant Fuels in the SPAR-CSN Probabilistic Safety Assessment Model”. **Sergio Courtin**, César Queral, Alberto García-Herranz, Jorge Sánchez Torrijos, David Canal, Julia Herrero-Otero. 18th International Topical Meeting on Safety Assessment and Analysis (PSA 2023), July 15-20, 2023. Knoxville, TN, USA.
- “Application of FLEX Strategies to Cope with a Station Blackout (SBO) Situation”, **Sergio Courtin**, César Queral, Marcos Cabezas, Alberto García-Herranz, Julia Herrero-Otero. International Conference Nuclear Energy for New Europe, September 12 - 15, 2022. Portoroz, Slovenia.
- “Analysis of the Impact of Post-Fukushima Improvements on Design Extension Conditions”, Alberto García-Herranz, César Queral, Marcos Cabezas, **Sergio Courtin**, Julia Herrero-Otero, Miguel Sánchez, Enrique Meléndez, Rafael Mendizábal. International Conference Nuclear Energy for New Europe, September 12 - 15, 2022. Portoroz, Slovenia.
- “Application of the Generic and Specific SPAR-CSN Probabilistic Safety Assessment Models”. Julia Herrero-Otero, Enrique Meléndez, Miguel Sánchez-Perea, César Queral, Marcos Cabezas, **Sergio Courtin**, Alberto García-Herranz, Carlos Paris. International Conference Nuclear Energy for New Europe, September 12 - 15, 2022. Portoroz, Slovenia.

- “Standardized Probabilistic Safety Assessment Models: Applications of SPAR-CSN Project”, Enrique Meléndez, Miguel Sánchez-Perea, César Queral, Julia Herrero-Otero, Marcos Cabezas, **Sergio Courtin**, Alberto García-Herranz, Carlos Paris, Rafael Iglesias. Probabilistic Safety Assessment and Management conference PSAM 16, June 26-July 1, 2022, Honolulu, Hawaii, USA.
- “On the Use of SPAR-CSN Model for Identifying Design Extension Conditions Type A Sequences. First Ideas”, César Queral, **Sergio Courtin**, Marcos Cabezas, Alberto García-Herranz, Julia Herrero-Otero, Enrique Meléndez, Miguel Sánchez, Rafael Mendizábal. Probabilistic Safety Assessment and Management conference PSAM 16, June 26-July 1, 2022, Honolulu, Hawaii, USA.
- “Standardized Probabilistic Safety Assessment Models: Status and First Results of SPAR-CSN Project”, César Queral, Carlos París, Enrique Meléndez, Marcos Cabezas, Santiago Arrazola, **Sergio Courtin**, David Castro, Rafael Iglesias, Miguel Sánchez. International Topical Meeting on Safety Assessment and Analysis (PSA 2021), November 7-12, 2021. Virtual meeting.
- “Aplicación de la Metodología de Fiabilidad Humana SPAR-H a Estrategias FLEX”. **Sergio Courtin**, César Queral, Carlos Paris, Rafael Iglesias. 46th Annual Meeting of the Spanish Nuclear Society, October 6-8, 2021. Granada, Spain.

Chapter 1

Introduction

Probabilistic Safety Assessment (PSA) are analysis techniques that come from technologies such as aeronautics and aerospace and which, in the 1970s, were adapted to nuclear power plant (NPP) safety studies as part of a research project of the Atomic Energy Commission (AEC) of the USA called Reactor Safety Study (RSS), published in 1975 by the U.S. Nuclear Regulatory Commission (NRC). Since its publication, the RSS has been the methodological reference for this type of safety analysis, although, naturally, most of its aspects have been refined over time. The Three Mile Island (TMI) accident in 1979 was the first major event that began to raise the suspicion that the traditional deterministic approach needed to be complemented in some way with approaches that took more explicitly into account the concept of accident probability, or risk, in short. The TMI accident was beyond the design basis, but it was a possible accident sequence detected and analyzed in the RSS. After this, the NRC intensified its research programs in the PSA and discussions began in the USA on the need to carry out these studies at all nuclear facilities. The Chernobyl accident in 1986, which produced the outcome that in risk analysis is considered the worst possible scenario, finally prompted the entry of this new vision of safety analysis around the world, as a need to complement the traditional deterministic approach.

The basic objective of a PSA is to estimate the risk of a facility. According to the definition, the first step towards this objective is to identify the possible accidental sequences, estimating their probability of occurrence, which could cause the damage to be prevented. This level is focused and emphasizes everything related to accident prevention, since, due to the nature of the analysis, it studies in depth the possibility of accidents and the reliability of the means and safety systems that a plant has to prevent them.

The PSA represent an overall assessment of the risk associated with a facility at a given point in its design and operation (the baseline configuration of the plant, in accordance with its current design and available operational data). Clearly, as modifications are made to the design and the population of operational data and results increases substantially, it is desirable to review the assumptions and baselines contemplated in the PSA to confirm that the results of the tool are still valid at the time of application for risk-informed regulatory

decision-making and other applications. Therefore, the PSA should undergo a process of maintenance and updating, so that at all times this tool represents as accurately as possible the situation of the plant.

Since the end of the 1980's, when the activities related to the development of PSA's started in Spain, the importance of PSA's for the use in the integral safety management in the licensing and authorization processes of the plants, in the framework of the Periodic Safety Reviews (RPS), as well as for the decision making in the regulatory framework of the Spanish Regulatory Body (CSN), as a fundamental tool of the Integrated Plant Supervision System (SISC), both for the operational indicators and for the risk significance of the findings, has become evident.

Revision 0 of Safety Guide GS-01.15 established and developed the aspects to be considered in the PSA updating and maintenance process and the requirements to be met in this process, all in accordance with the activities related to applications and the role that these played in the operation and licensing for each specific case, as well as with the type of applications that each plant wished to undertake or was already carrying out. With the issuing of Council Instruction IS-25, of 9th June 2010, "Criteria and requirements on the performance of probabilistic safety analysis and their applications to nuclear power plants", the need for adequate updating and maintenance became mandatory, both to incorporate the state of the art of the analysis methodologies and to incorporate plant modifications and operating experience that might affect the models, as a result of which the revision of this Guide is now necessary. Revision 1 of GS-01.15 develops the criteria and establishes the frequency of maintenance and updating of the PSAs, in accordance with the provisions of IS-25.

1.1 Thesis Objectives

The general objective of this work is to apply probabilistic safety analysis to FLEX strategies, advanced technology fuels, and design extension conditions in order to characterize the impact of these strategies and elements in reducing the associated risk. The specific objectives are:

- Define and select the most appropriate FLEX strategies for the available PSA model.
- Develop the event and fail trees associated to these FLEX strategies for their incorporation into the PSA model.
- Study the human reliability methodology for the FLEX strategies.
- Quantify the impact of selected FLEX strategies on the PSA model.
- Develop a methodology for design extension conditions sequences identification.
- Quantify the FLEX strategies impact on the identified design extension conditions sequences.
- Detect the human actions affected by the introduction of advanced technology fuels.
- Quantify the impact of the advanced technology fuels fuel on level 1 PSA model.

Chapter 2

Standardized Plant Analysis Risk Models

The regulatory activity requires the oversight of licensee performance to be made from an independent position. This position is better served when the regulatory body develops its own methodologies and tools. In particular, in the matter of probabilistic risk analysis, even if the licensees' analyzes are subject to peer-review and/or are reviewed by the regulatory body, it is very difficult to manage the large amount of hypotheses and assumptions behind the model. Thus, the development of a PSA model for regulatory use improves the knowledge of the NPP risks and can be seen as an enhancement of the regulatory practice.

On this regard, the CSN, in collaboration with the Universidad Politécnica de Madrid (UPM), has assembling its own generic standardized model (SPAR-CSN) for 3-loop pressurized water reactor (PWR) Westinghouse designs (WEC). The purpose of the project considered the elaboration of a standardized PSA model independent from the industry, providing a high-level view of risk in the evaluation of findings in Spanish NPPs, intended to be comparable in scope to NRC Standardized Plant Analysis Risk (SPAR) models [Watanabe et al., 2018, NRC, 2010a, NRC, 2010b]. To this end, the conclusions drawn from the comparison of the existing industry models were used to establish a common set of assumptions and standard modeling techniques to be used in CSN models. The SPAR-CSN models aim to:

- (a) Better understanding of the main contributors to risk in Spanish NPPs.
- (b) Prioritization in inspection and oversight tasks over Spanish NPPs through the analysis of systems and components importance.
- (c) Precursor analysis of operational incidents occurred in Spanish NPPs in order to determine closeness to a core damage scenario.
- (d) Assessment of inspection findings within the Spanish regulatory system including the systematic evaluation of the operation of the plants considered within the, “Integrated Plant Supervision System”, SISC, which incorporates novel supervision methods focused on the observation of the performance of the operating NPPs by means of operating

indicators and the assessment of the findings of the inspections performed by the CSN. The SPAR-CSN project utilized a methodology to standardize modeling criteria across various PSA aspects, achieving the project's objectives as detailed in [Meléndez et al., 2016]. This goal was realized by defining and applying a methodology that, through a comprehensive comparison of different PSA models, outlines the following high-level steps.

1. Comparison of the Event Trees (ET), success and failure sequences and success criteria of each header from Spanish PSA, SPAR-NRC and other models.
2. First proposal of standardized ETs and success criteria.
3. Comparison of the description and functional analysis of each considered system, of the modeling hypotheses (this allows to obtain a classification and discrimination of differences among PSA models), and of the Fault Trees (FT) models.
4. Comparison of Human actions (type 3/5) included in each ET of the analyzed PSA models, and identification of the Emergency Operating Procedures (EOPs) related with the sequences.
5. Comparison of the data base from the analyzed PSA models.
6. Proposal of headers, ETs and success criteria for the SPAR-CSN model.
7. Proposal for the standardization of FT models for each system, including a standardized block diagram for each system, common nomenclature of system equipment and logic elements of the FTs, first evaluation and proposal of hypotheses and modeling techniques, evaluation of the human actions and the operation conditions applicable, identification of data sources.
8. Proposal of ET models including the identification of the involved EOPs and applicable human actions.
9. Proposal of data base considered in FTs.
10. Modeling of ET and FT models in the quantification code (Risk Spectrum).
11. Verification of the SPAR-CSN model, through a high level comparison of the risk profile of the plant will be performed.

The following sections summarize some of the results obtained within the project. The main premises, scope and data sources of the generic SPAR-CSN model, with brief description of some representative examples of ET and FT, are depicted in Section 2.1. Section 2.2 summarizes the results obtained with the model. The conclusions are described in the Section 6.1 of Conclusions Chapter.

2.1 Description of the Generic SPAR-CSN Model

This section provides readers with a high-level description of the SPAR-CSN generic model, distributed as follows: subsections 2.1.1 and 2.1.2 depict the model scope regarding the Event Tree and Fault Tree tasks; subsection 2.1.3 summarizes the human reliability analysis task, focusing on the analysis methodology and its extent; finally, a list of the generic data sources used to quantify the probabilities of equipment failures is introduced in subsection 2.1.4.

It must be considered that the SPAR-CSN generic model has been devised as a standard base from which to develop the specific SPAR-CSN models of 3-loop PWR-WEC Spanish NPPs. Hence, both the modeling assumptions and safety systems diagrams have been chosen to be representative of a generic plant, leaving the design details and operational characteristics for the specific SPAR-CSN models.

The only post-Fukushima improvement included in the initial PSA model is the passive thermal shutdown seals (SDS). These are ring-shaped devices placed on the reactor coolant pumps (RCPs) designed to prevent leakage in case of loss of seal injection. The SDS have specific part which degrades and disappears at high-temperature allowing the movement of two annular pieces in contact that, thanks to the direction of water flow, are positioned in such a way that they completely seal the coolant leakage path through the pump. These passive seals act in situations such as loss of AC current, with consequent tripping of the pumps and no injection of coolant to seals, but cooling of the three steam generators (SGs) is still required to be maintained by means of the Auxiliary Feedwater system (AFW) turbine-driven pump (TDP-AFW). SDS deployment show a positive impact and a significant increase the margin to core uncover and reduce the core damage frequency (CDF) [Mena-Rosell et al., 2018]. Work is currently underway to incorporate containment systems for L2PSA applications.

2.1.1 Event Trees for Internal Events at Full Power

This section summarizes the scope of the SPAR-CSN model regarding the Event Tree delineation. In its first version, 14 main ETs were included (Table 2.1).

The plant design chosen for the SPAR-CSN generic model includes two service water systems: the Non-Essential service water system, which is normally operating, and the Essential service water system, which is normally in standby mode and starts when safety injection or LOOP condition occurs, or whenever the Non-Essential service water is lost. In the same way, the model also includes two component cooling water systems: CCW and NCCW. The general assumptions considered for the ET sequence delineation are:

- The mission time for the different accidental sequences mitigation strategies is 24 hours.
- After this time, it will be considered that the situation will lead either to core damage (Consequence CD) or to a safe stable state (Consequence S).
- A safe stable state is defined as follows in NUREG-2122 [NRC, 2013b]: “condition of

Table 2.1: List of ETs of the SPAR-CSN generic model

ID	Initiating event (IE)	ID	Initiating Event
LBLOCA	Loss of coolant accident (LOCA > 6 in.)	MSLB-US	Main steam line break upstream of MSIV
MBLOCA	LOCA (2in. 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 CCWS 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

the reactor in which the necessary safety functions are achieved”.

2.1.2 Fault Trees

This section summarizes the scope of the SPAR-CSN model regarding the FT task. In its first version, 18 safety systems are included (Table 2.2), with 539 FT pages and 1381 basic events. At present, also containment systems are being implemented.

Table 2.2: List of system FTs of the SPAR-CSN Model

ID	System	ID	System
AC	Emergency AC distribution system	IA	Instrumentation air system
AF	Auxiliar 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 system
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 assumptions considered for the FT construction are:

- All failures considered in the FTs occur during mission time. Situations where a system or equipment was already failed before initiating event due to any not detected spurious operation are not included in the model.
- A single unavailability basic event due to maintenance or testing is associated to each train of each safety system. These events include the contributions only from equipment that renders the train unavailable.

- A high-level representation of the failure of the automatic startup signals is modeled, which includes the potential loss of the required AC/DC power supply and a single undeveloped basic event representing the signal generation failure.
- The failure events for manual actions do not include the control room equipment failure, such as panels, levers, switches, etc.

A Fault Tree example is introduced in Figures 2.1 to 2.4, corresponding to the Low-Pressure Safety Injection system. The system diagram is depicted in Figure 2.1 and 2.2, where suction and discharge alternatives and the most important equipment are indicated. Additionally, the Fault Tree top event, corresponding to the cold leg injection and recirculation modes is represented in Figure 2.3. The LPSI FT structure is represented in Figure 2.4, indicating the correspondence between the FT pages and the different parts of the system diagram. Figure 2.5 and Table 2.3 show the LPSI dependencies with the power and signal systems.

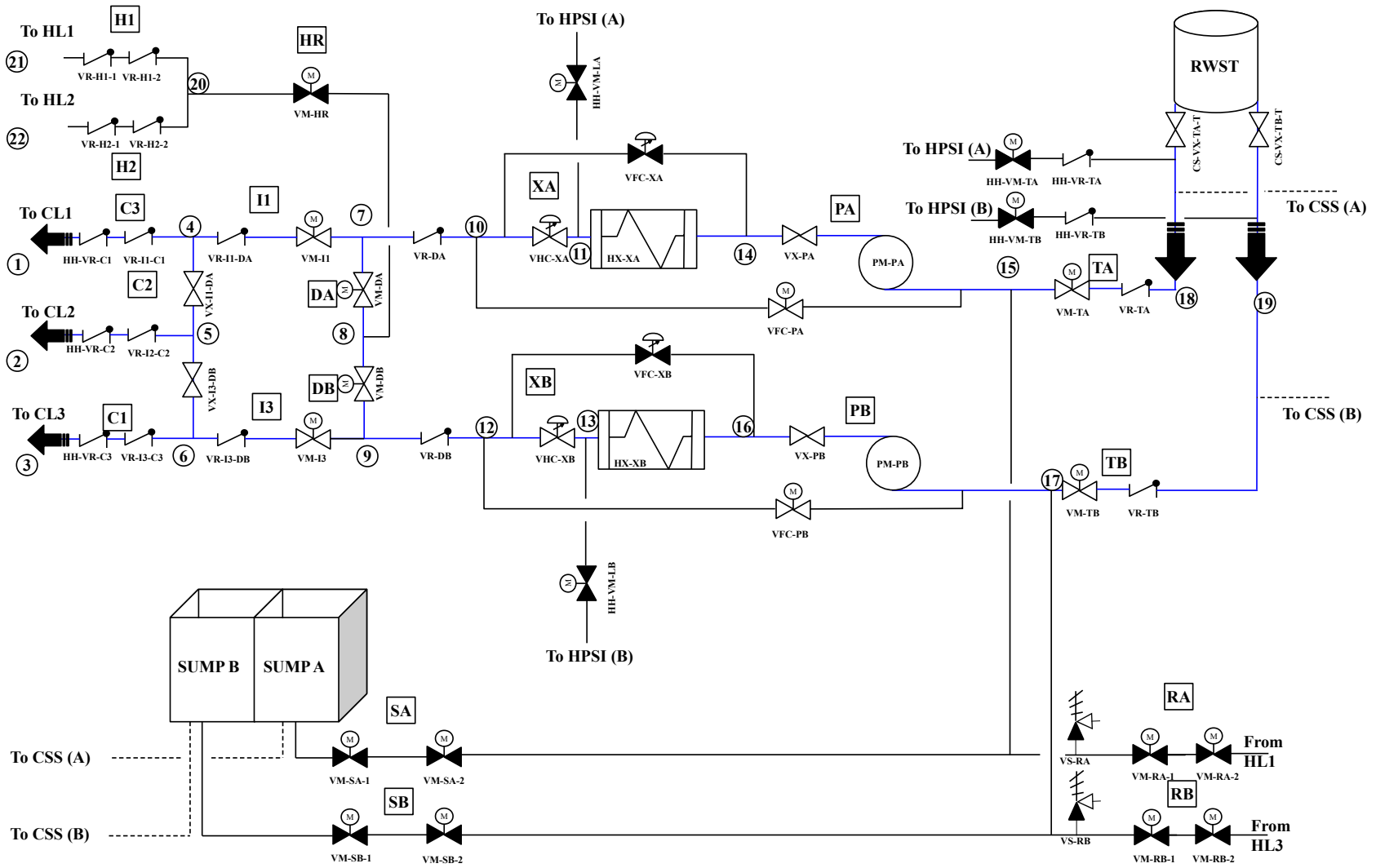


Figure 2.1: LPSI simplified diagram of the SPAR-CSN generic model. Injection to CLs mode

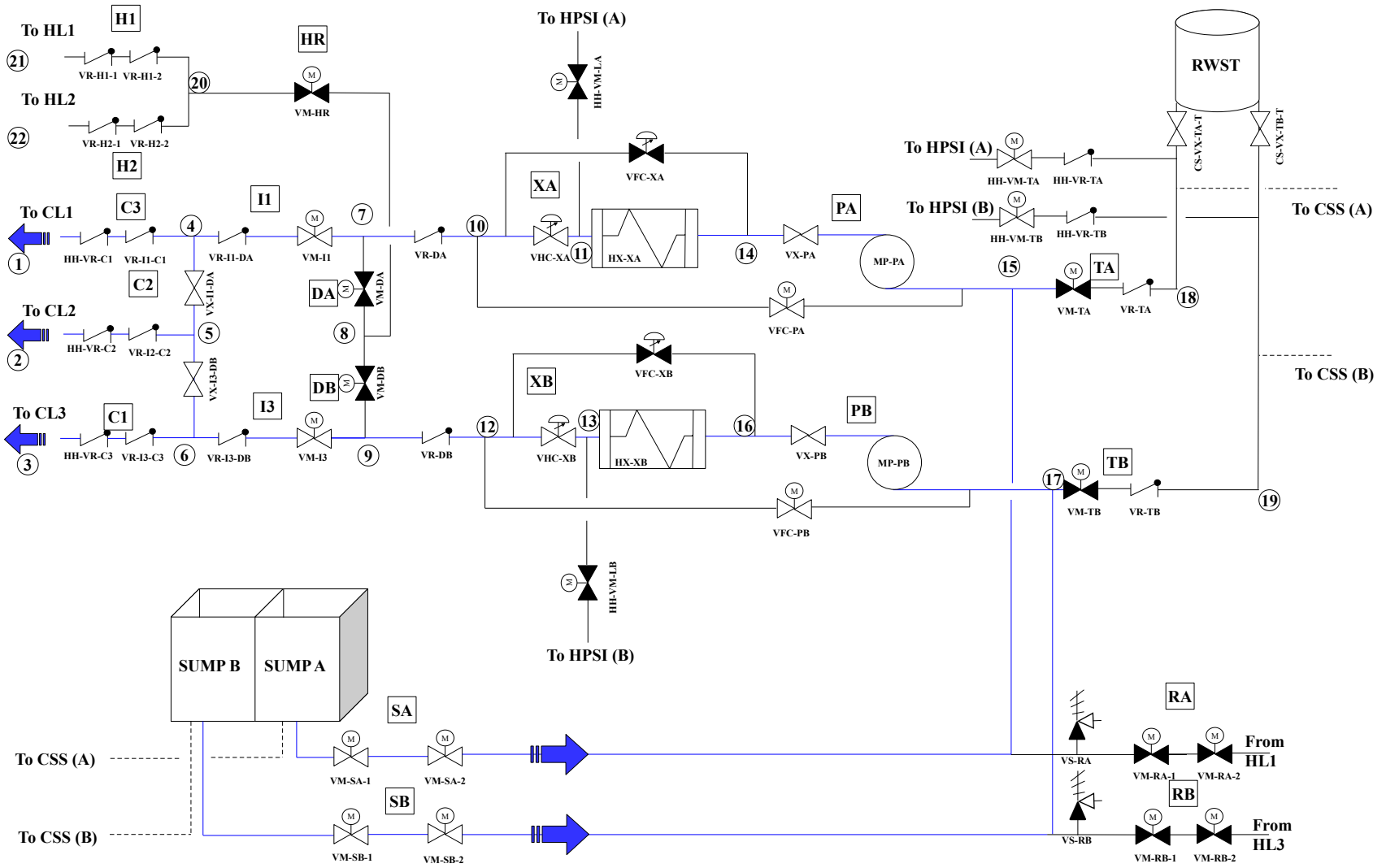


Figure 2.2: LPSI simplified diagram of the SPAR-CSN generic model. Recirculation to CLs mode

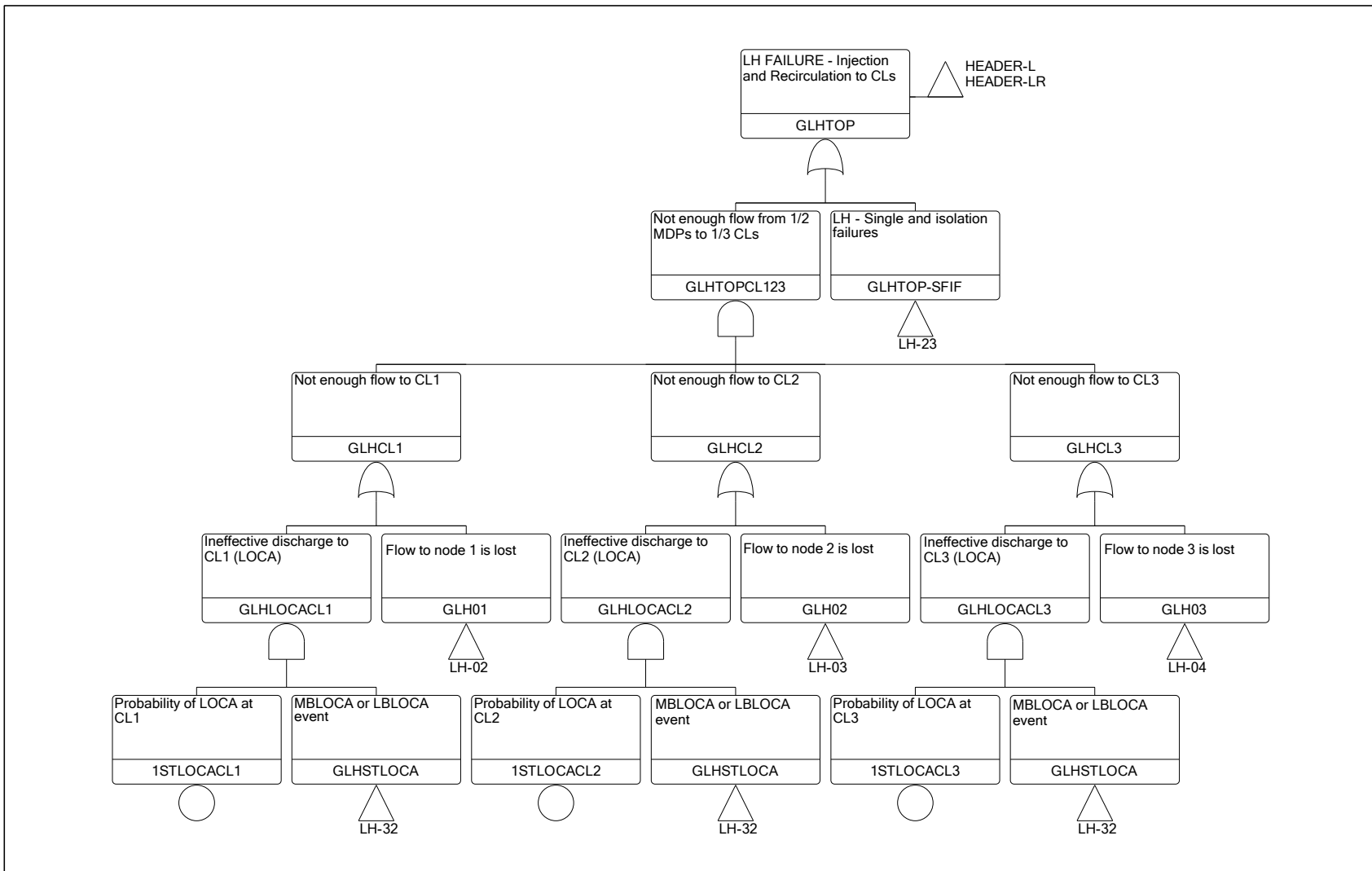


Figure 2.3: Top Event of LPSI Fault Tree of the SPAR-CSN generic model (Cold Leg Injection and Recirculation modes)

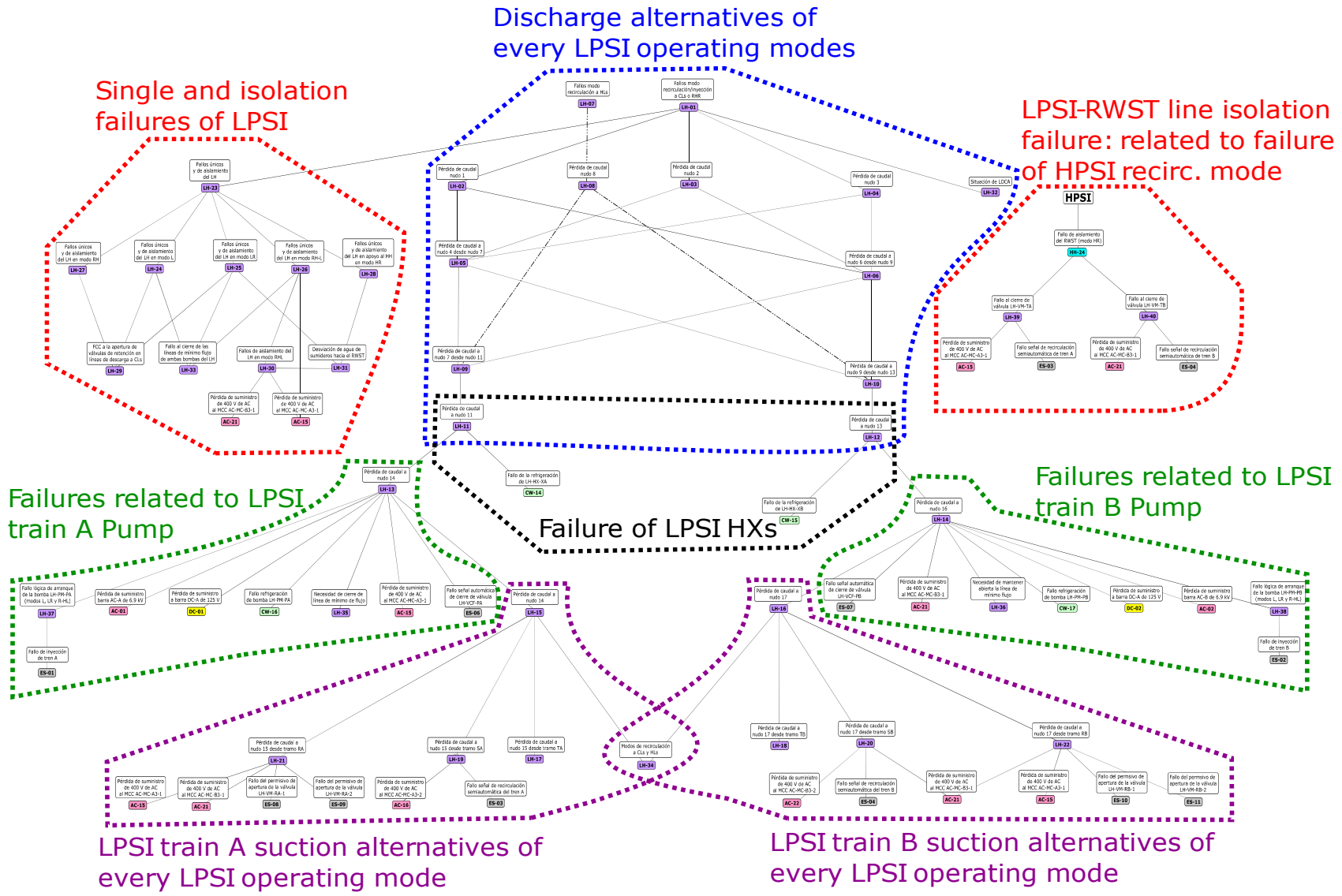


Figure 2.4: Structure of the LH FT

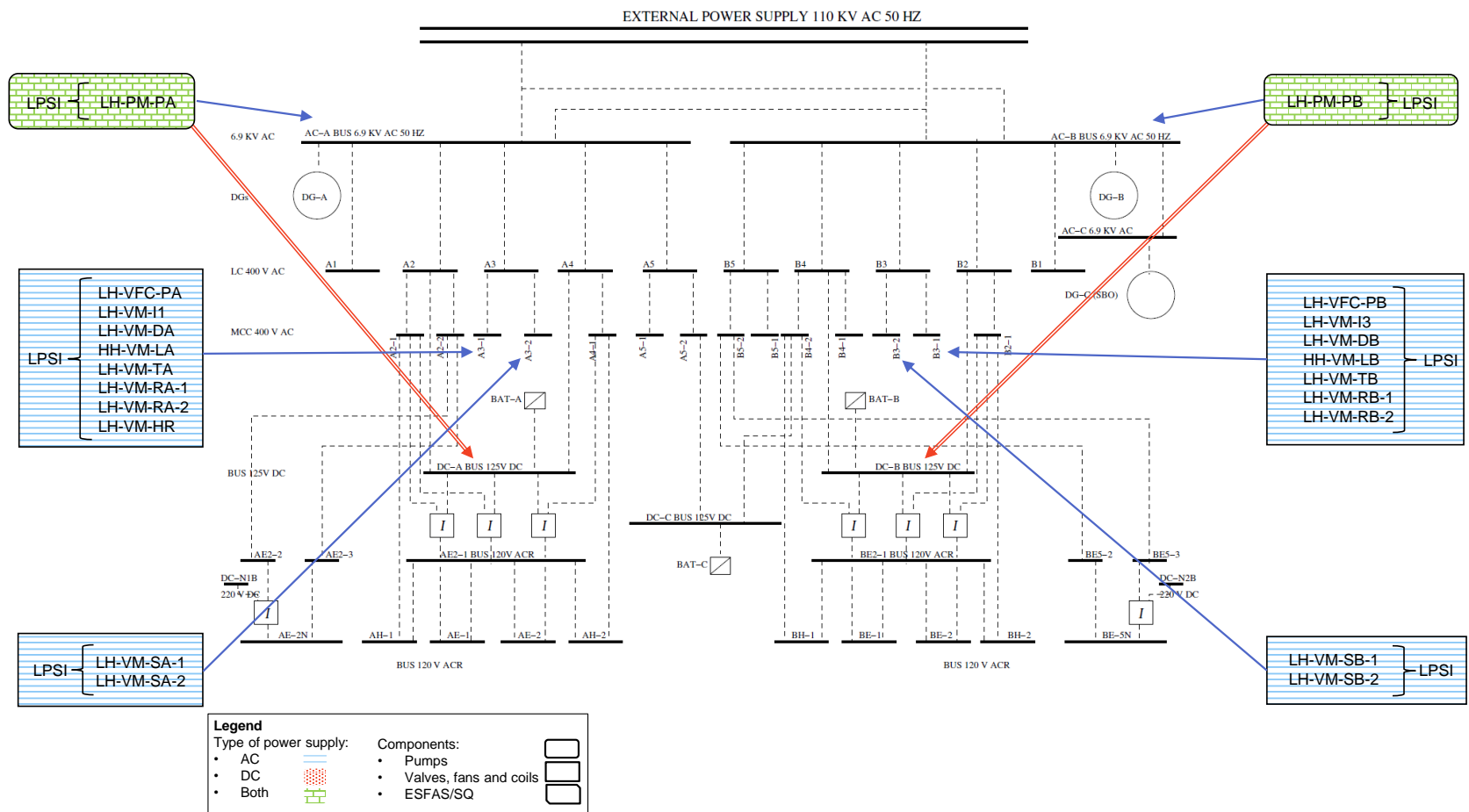


Figure 2.5: One-line diagram of the SPAR-CSN model and its relation to the LPSI

Table 2.3: LPSI Dependencies matrix

Mode L		AC-6.9kV		CWCS		AC-400V				DC-125V		ESFAS (SI)		ESFAS (SR)	
		AC-A	AC-B	CW-A	CW-B	CCM A3.1	CCM B3.1	CCM A3.2	CCM B3.2	DC-A	DC-B	SI-A	SI-B	SR-A	SR-B
LPSI Pumps	LH-PM-PA														
	LH-PM-PB														
Pumps valves	LH-VFC-PA														
	LH-VFC-PB														
Mode LR, R-HL, HR support		AC-6.9kV		CWCS		AC-400V				DC-125V		ESFAS (SI)		ESFAS (SR)	
		AC-A	AC-B	CW-A	CW-B	CCM A3.1	CCM B3.1	CCM A3.2	CCM B3.2	DC-A	DC-B	SI-A	SI-B	SR-A	SR-B
Motor-Valve	LH-VM-HR														
	LH-VM-I1														
	LH-VM-I3														
	LH-VM-DA														
	LH-VM-DB														
	HH-VM-LA														
	HH-VM-LB														
	LH-VM-SA-1														
	LH-VM-SB-1														
	LH-VM-SA-2														
	LH-VM-SB-2														
	LH-VM-TA														
LH-VM-TB															
Heat exchanger	LH-HX-XA														
	LH-HX-XB														
Motor-Valve (CCWS)	CW-VM-LA-1														
	CW-VM-LA-2														
	CW-VM-LB-1														
	CW-VM-LB-2														
Mode RH		AC-6.9kV		CW		AC-400V				DC-125V		ESFAS (SI)		ESFAS	
		AC-A	AC-B	CW-A	CW-B	CCM A3.1	CCM B3.1	CCM A3.2	CCM B3.2	DC-A	DC-B	SI-A	SI-B	SR-A	SR-B
Motor-Valve	LH-VM-RA-1														
	LH-VM-RB-1														
	LH-VM-RA-2														
	LH-VM-RB-2														

2.1.3 Human Reliability

Regarding the Human Reliability Analysis (HRA) task, the SPAR-CSN generic model uses the SPAR-H methodology [NRC, 2005] to quantify the error probability of type-3 human actions, just like SPAR-NRC models. However, compared to SPAR-NRC models, a larger number of different human failure events (HFE) have been included, 38 in total.

The available and required time for each human action have been obtained from a complete review, comparison and weighing of the Spanish NPPs PSA documentation. Human error probabilities (HEP) range between 1.00E-05 and 1.00E-01 (not considering dependencies between human actions).

To understand and justify the implementation of the SPAR-CSN type-3 human actions, each ET in the model has been represented identifying the EOP associated to each ET header. In addition, only few type 1 human actions have been included, specifically those related to crucial equipment in the plant (calibration error of Pressurizer relief and safety valves, calibration error of storage tanks level channels, etc.), and no type 5 human actions have been considered. Finally, a complete analysis of type 3 human actions dependency has been performed. The methodology consists of the following steps:

1. The probability value of each human error in the model is set to 1, so the contribution of human actions is maximized.
2. The CDF of each ET is calculated, identifying and choosing the combinations of human errors among the most important MCS.
3. The chosen combinations are studied following the SPAR-H dependency methodology to calculate the dependency level between the human actions.
4. Finally, a set of post-processing rules is created in the SPAR-CSN model, to change the probability value of the group of dependent human actions according to the dependency levels.

Table 2.4 shows the HFE, and their corresponding HEPs, included in the model SPAR-CSN model. Regarding the dependencies between human actions, a set of 19 combinations of 2 actions has been analyzed, identifying 3 combinations with a level of complete dependency (conditional probability value of the dependent action is set to 1), 9 with high dependency (conditional probability value of the dependent action around 0.5), 4 with moderate dependency (conditional probability value of the dependent action around 0.14) and 3 with low dependency level (conditional probability value of dependent action around 0.05). It must be clarified that no combinations consisting of 3 or more actions have been found to be significant.

Table 2.4: Human Failure Events in the SPAR-CSN Model

ID	ET	DESCRIPTION	HEP
1	Support	Operator failure to reconnect required MCCs	2.00E-03
2	LOOP	Operator failure to close breakers after external power recovery	[1.00E-04; 1.00E-03]
3	ATWS, SBLOCA, MBLOCA, MSLB, SGTR, GT, LC, LOOP	Operator failure at SG level control	5.00E-05
4	MSLB	Operator failure at SG level control	1.00E-02
5	MBLOCA, SBLOCA, MSLB, LCWA, LOOP, GT, LC	Operator failure in AFW start-up support	1.00E-04
6	MSLB	Operator failure in AFW start-up support	1.00E-01
7	SGTR	Operator failure in AFW start-up support	1.00E-04
8	SBLOCA-LOOP	Operator failure to start AFW motor pumps during AC recovery after SBO	1.00E-04
9	SBLOCA, MBLOCA	Operator failure to isolate or vent accumulators	1.00E-03
10	Support	Operator failure in support of CW auto start signals	2.00E-03
11	Support	Operator failure in manual system start during AC recovery after SBO	1.00E-03
12	LCWA	Operator failure in CCWS alignment to thermal barriers	1.00E-04
13	LOOP	Operator failure to reposition Train B chargers	1.00E-04
14	LOOP	Operator failure in support of DG-C generator startup	1.00E-04
15	Support, LOOP	Operator failure in support of DG-A/B diesel generator startup	1.00E-03
16	GT, LC	Operator Failure to Perform F&B Action	2.00E-03
17	MBLOCA, SBLOCA, MSLB, SGTR, GT, LC	Operator Failure to Switch to High Pressure Recirculation at CLs	1.00E-04
18	SBLOCA , MSLB	Operator failure to reduce safety injection flow rate	1.00E-04
19	SGTR	Operator failure to reduce safety injection flow rate.	2.00E-03
20	ATWS	Operator failure to start HH on ATWS (emergency boration)	2.00E-04
21	MBLOCA	Operator failure to support HH auto start signals (HI mode)	2.00E-02
22	SBLOCA, MSLB	Operator failure to support HH auto start signals (HI mode)	1.00E-04
23	SGTR	Operator failure to support HH auto start signals (HI mode)	1.00E-04
24	SBLOCA-LOOP	Operator failure on HH start-up when recovering AC after SBO	1.00E-05
25	MBLOCA	Operator failure at HH-VM-AB valve opening (by-pass BIT)	1.00E-03
26	SBLOCA, SGTR	Operator failure at HH-VM-AB valve opening (by-pass BIT)	1.00E-04
27	MSLB	Operator failure to open HH-VM-AB valve (by-pass BIT)	1.00E+00
28	LBLOCA	Operator failure in support of the semi-automatic recirculation signal	2.00E-03
29	MBLOCA	Operator failure in support of the semi-automatic recirculation signal	1.00E-04
30	SBLOCA	Operator failure in semi-automatic recirculation signal support	1.00E-05
31	SBLOCA	Operator failure in restarting the LH pumps	1.00E-04
32	LBLOCA	Operator failure in switching to recirculation to HLs.	1.00E-04
33	MBLOCA	Operator failure in support of LH start-up	1.00E-03
34	GT, LC	Operator failure to support closing of SG relief valves	1.00E-03
35	MSLB	Operator failure to isolate the AFWS	1.00E-04
36	SGTR	Operator failure to isolate main steam manually	1.00E-04
37	SGTR	Operator failure to open and close PORVs of SGs	1.00E-04
38	SGTR	Operator failure to open and close SG PORVs valves	1.00E-04
39	MBLOCA	Operator failure to open and control PORVs and SDs valves	1.00E-03
40	SBLOCA	Operator failure to open and control PORVs and SDs valves	1.00E-04
41	LCWA	Operator failure to open and control PORVs and SDs valves	1.00E-04
42	GT, LC	Operator failure to close relief paths	1.00E+00
43	ATWS, MSLB	Operator failure to close relief paths	1.00E-04
44	SGTR	Operator failure to close relief paths	1.00E-04
45	SBLOCA, MSLB, SGTR	Operator failure to open PORVs of the PZR	1.00E-04
46	LCWA	Operator failure to trip RCPs	1.00E-04
47	MBLOCA, GT, LC, SBLOCA, MSLB, SGTR, LOOP	Operator failure on reactor trip actuation	1.00E-03
48	LCWA	Operator failure in commissioning cooling and injection to RCP seals.	1.00E-03
49	LCWA	Operator failure in commissioning HTP to inject to RCPs seals	1.00E-04
50	Support	Operator failure at tower level control (SW)	1.00E-04
51	LNSW	Operator failure to support SW automatic start-up actions	1.00E-04
52	LBLOCA	Operator failure in SW automatic startup support actions	5.00E-02
53	Support	Operator failure at SW start-up in case of AC recovery after SBO	1.00E-04

2.1.4 Data Sources

Two main public data sources have been used to assign quantification models to basic events related to equipment failures, system unavailability and initiating events:

- NUREG/CR-6928 [NRC, 2015c]:
 - Unavailability Data Sheets (2015 update): for equipment failures
 - Initiating Event Data Sheets (2015 update): for initiating event frequencies
 - Train UA (2015 update): for system unavailability due to maintenance or testing
- NUREG/CR-5497 [NRC, 2015a]:
 - CCF parameter estimations (2015 update)

In addition, some other public references from SPAR-NRC models have been checked (e.g., [NRC, 2010a, NRC, 2013c]).

2.1.4.1 Initiators Frequency

When obtaining the frequency of the initiators, the Spanish NPPs use different sources:

- Generic data sources: NUREG, WASH, Nuclear Power Experiences (NPE), Experience at Westinghouse.
- Plant-specific data: When considering the expected experience with a particular initiator as your with a particular initiator as sufficient.
- Own simplified FT models.

In the case of SPAR-CSN, data from generic sources (NUREG/CR-6928 [NRC, 2015c] & NUREG/CR-5750 [NRC, 1999]) have been used, as well as data obtained from the Spanish NPPs analyzed: Ascó NPP, Vandellòs-2 NPP and Almaraz NPP. Figures 2.6, 2.7 and 2.8 show the industry-average frequency distribution for GT, LOOP and SGTR respectively, obtained from NUREG/CR-6928. Finally, Table 2.5 shows the frequency of every initiator considered in the SPAR-CSN Model.

2.1.2.3 Industry-Average Baselines

Table 2-4 lists the industry-average frequency distribution. This industry-average frequency does not account for any recovery.

Table 2-4. Selected industry distribution of λ for TRANS (PWR).

Source	5%	Mean	95%	Distribution		
				Type	α	β
EB/PL/KS	3.35E-01	6.76E-01	1.12E+00	Gamma	7.860	1.160E+01

Note – EB/PL/KS is an empirical Bayes analysis at the plant level with the Kass-Steffey adjustment. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

Figure 2.6: GT initiator event data [NRC, 2015c]

4.1.3 Industry-Average Baselines

Table 4-2 lists the industry-average frequency distributions for the four LOOP categories and total LOOP. These industry-average frequencies do not account for any recovery.

Table 4-2. Selected industry distributions of λ for LOOP.

Event	Source	5%	Mean	95%	Distribution		
					Type	α	β
PO.LOOP	JNID/IL	2.45E-02	3.11E-02	3.84E-02	Gamma	54.50	1.750E+03
PO.LOOP-GR	EB/PP/KS	1.10E-04	1.10E-02	3.94E-02	Gamma	0.61	5.530E+01
PO.LOOP-PC	JNID/IL	6.19E-04	2.00E-03	4.02E-03	Gamma	3.50	1.750E+03
PO.LOOP-SC	JNID/IL	9.22E-03	1.34E-02	1.83E-02	Gamma	23.50	1.750E+03
PO.LOOP-WR	EB/PL/KS	7.86E-04	5.08E-03	1.25E-02	Gamma	1.80	3.540E+02

Note – EB/PL/KS is an empirical Bayes analysis at the plant level with the Kass-Steffey adjustment. JNID/IL is a Jeffrey’s noninformative distribution at the industry level. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

Figure 2.7: LOOP initiator event data [NRC, 2015c]

1.2.3 Industry-Average Baselines

Table 1-12 lists the industry-average frequency distribution. Two different approaches to estimating the frequency for SGTR were discussed – the expert elicitation approach from Reference 5, and the data analysis using the IEDB. Because the expert elicitation process outlined in Reference 5 resulted in a mean frequency for SGTR (3.78E-3/rcry) which is higher than that obtained from optimizing the SGTR data from the IEDB (2.07E-03/rcry), the IEDB results were used. This industry-average frequency does not account for any recovery.

Table 1-12. Selected industry distribution of λ for SGTR.

Source	5%	Mean	95%	Distribution		
				Type	α	β
JNID/IL	3.82E-04	1.66E-03	3.69E-03	Gamma	2.500	1.500E+03

Note – JNID/IL is a Jeffrey's noninformative distribution at the industry level. The percentiles and the mean of the distribution have units of events/rcry. The units for β are rcry.

Figure 2.8: SGTR initiator event data [NRC, 2015c]

Table 2.5: Initiator frequency in SPAR-CSN Model

Initiator	Frequency	Source
LBLOCA	1.33E-06	NUREG/CR-6928
SBLOCA	5.10E-04	NUREG/CR-6928
MBLOCA	3.01E-04	NUREG/CR-6928
GT	6.76E-01	NUREG/CR-6928
MSLB-DS	1.80E-03	NUREG/CR-5750
MSLB-US	6.00E-04	NUREG/CR-5750
LC	4.82E-02	NUREG/CR-6928
SGTR	1.66E-03	NUREG/CR-6928
LNSW	2.00E-04	NUREG/CR-6928
LCWA	2.00E-04	NUREG/CR-6928
LOOP	3.11E-02	NUREG/CR-6928
LDC	1.00E-03	NUREG/CR-6928

2.1.4.2 Basics Events

Two types of component failures are considered within the SPAR-CSN model:

- Failures on demand (probability):

$$P = \frac{\text{Failures}}{\text{Demands}}$$

- Failures on mission (failure rate):

$$\lambda = \frac{\text{Failures}}{\text{Operating hours}}$$

The failure probability of mission failures is obtained by multiplying the failure rate by the mission time (typically 24hours in the SPAR-CSN model). The nomenclature for these failures in NUREG uses the acronyms of the failure description, FTR (Fail to run) and FTS (Fail to start). Two examples of the quantification of the probability of failure to start (failure to demand) and failure to operate (failure to mission) of an motor-driven pump (MDP) from AFW system, using NUREG/CR-6928 as the data source, are presented in Figures 2.9, 2.10, 2.11 and 2.12. The table 2.6 shows the different types of valve failures.

2.1.3 Industry-Average Baselines

Table 2-4 lists the selected industry distributions of p and λ for the MDP failure modes. These industry-average failure rates do not account for any recovery.

Table 2-4. Selected industry distributions of p and λ for MDPs.

Pooling Group	Failure Mode	Source	5%	Median	Mean	95%	Distribution		
							Type	α	β
Standby	FTS	EB/PL/KS	1.59E-04	6.78E-04	7.94E-04	1.83E-03	Beta	2.21	2.78E+03
Standby	FTR<1H	EB/PL/KS	3.94E-06	7.86E-05	1.22E-04	3.89E-04	Gamma	0.84	6.91E+03
Standby	FTR>1H	EB/PL/KS	1.35E-07	6.28E-06	1.15E-05	4.07E-05	Gamma	0.63	5.49E+04
-	ELS	EB/PL/KS	9.46E-09	1.87E-07	2.91E-07	9.22E-07	Gamma	0.85	2.92E+06
-	ELL	EB/PL/KS	2.18E-12	4.97E-09	2.04E-08	9.32E-08	Gamma	0.30	1.47E+07
Normally Running	FTS	EB/PL/KS	2.61E-04	9.52E-04	1.08E-03	2.37E-03	Beta	2.62	2.41E+03
Normally Running	FTR	EB/PL/KS	1.15E-06	3.43E-06	3.79E-06	7.68E-06	Gamma	3.42	9.01E+05
CCW	FTS	EB/PL/KS	5.86E-05	6.32E-04	8.78E-04	2.52E-03	Beta	1.12	1.28E+03
CCW	FTR	EB/PL/KS	2.87E-07	2.13E-06	2.76E-06	7.33E-06	Gamma	1.40	5.09E+05
SWS	FTS	EB/PL/KS	5.67E-05	5.56E-04	7.55E-04	2.14E-03	Beta	1.18	1.56E+03
SWS	FTS	EB/PL/KS	6.20E-05	6.17E-04	8.41E-04	2.38E-03	Beta	1.17	1.39E+03
SWS	FTR	EB/PL/KS	4.14E-07	4.70E-06	6.56E-06	1.91E-05	Gamma	1.09	1.66E+05
SWS	FTR	JNID/IL	3.43E-08	1.30E-07	1.50E-07	3.31E-07	Gamma	2.50	1.67E+07
NSW	FTR	EB/PL/KS	4.34E-07	4.85E-06	6.72E-06	1.96E-05	Gamma	1.10	1.63E+05
SWS-NE	FTS	EB/PL/KS	6.39E-05	6.18E-04	8.36E-04	2.36E-03	Beta	1.19	1.42E+03
SWS-NE	FTR	EB/PL/KS	8.93E-07	4.69E-06	5.67E-06	1.39E-05	Gamma	1.82	3.20E+05
CSW	FTR	JNID/IL	4.88E-06	6.90E-06	6.98E-06	9.41E-06	Gamma	25.50	3.65E+06

Figure 2.9: AFW MDP Fail to start (demand): MDP-FTS-AFW from NUREG/CR-6928

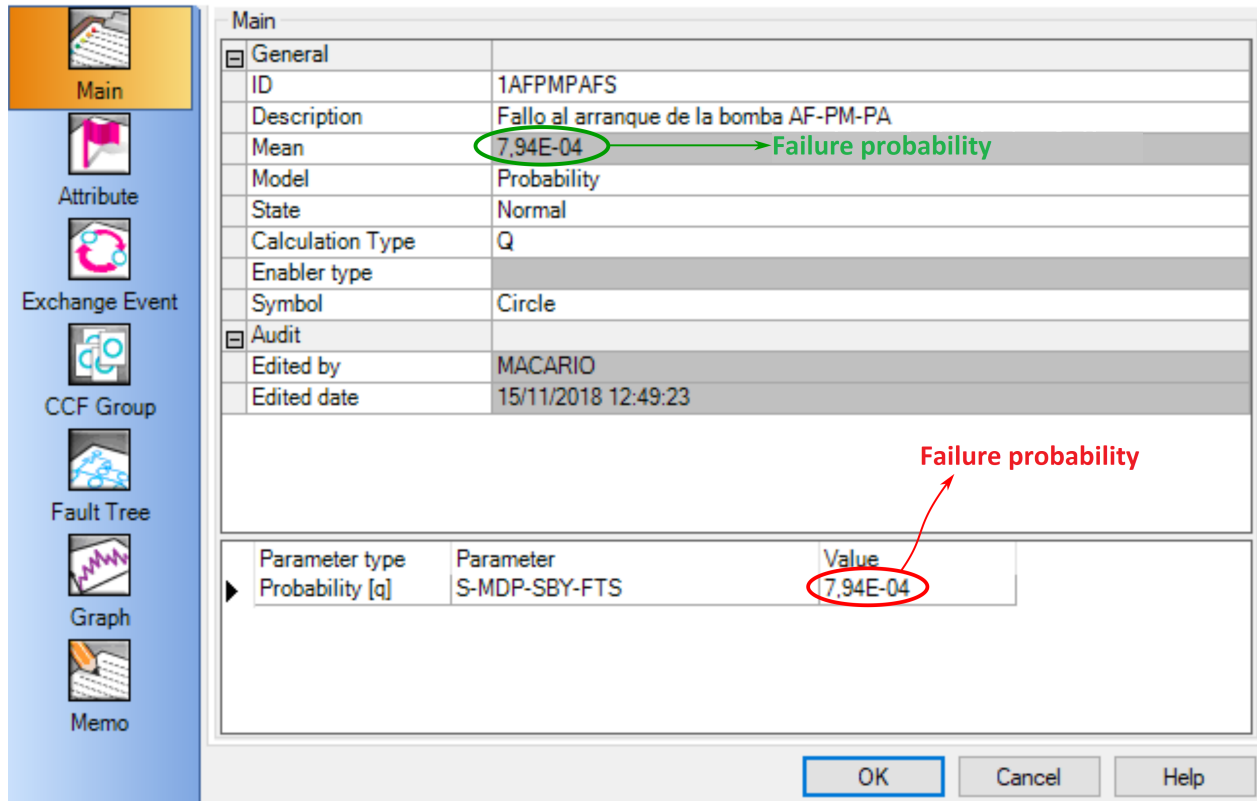


Figure 2.10: MDP-FTS-AFW modeling in RiskSpectrum

2.1.3 Industry-Average Baselines

Table 2-4 lists the selected industry distributions of p and λ for the MDP failure modes. These industry-average failure rates do not account for any recovery.

Table 2-4. Selected industry distributions of p and λ for MDPs.

Pooling Group	Failure Mode	Source	5%	Median	Mean	95%	Distribution		
							Type	α	β
Standby	FTR	EB/PL/KS	1.59E-04	6.78E-04	7.94E-04	1.83E-03	Beta	2.21	2.78E+03
Standby	FTR<1H	EB/PL/KS	3.94E-06	7.86E-05	1.22E-04	3.89E-04	Gamma	0.84	6.91E+03
Standby	FTR>1H	EB/PL/KS	1.35E-07	6.28E-06	1.15E-05	4.07E-05	Gamma	0.63	5.49E+04
-	ELS	EB/PL/KS	9.46E-09	1.87E-07	2.91E-07	9.22E-07	Gamma	0.85	2.92E+06
-	ELL	EB/PL/KS	2.18E-12	4.97E-09	2.04E-08	9.32E-08	Gamma	0.30	1.47E+07
Normally Running	FTR	EB/PL/KS	2.61E-04	9.52E-04	1.08E-03	2.37E-03	Beta	2.62	2.41E+03
Normally Running	FTR	EB/PL/KS	1.15E-06	3.43E-06	3.79E-06	7.68E-06	Gamma	3.42	9.01E+05
CCW	FTR	EB/PL/KS	5.86E-05	6.32E-04	8.78E-04	2.52E-03	Beta	1.12	1.28E+03
CCW	FTR	EB/PL/KS	2.87E-07	2.13E-06	2.76E-06	7.33E-06	Gamma	1.40	5.09E+05
SWS	FTR	EB/PL/KS	5.67E-05	5.56E-04	7.55E-04	2.14E-03	Beta	1.18	1.56E+03
SWS	FTR	EB/PL/KS	6.20E-05	6.17E-04	8.41E-04	2.38E-03	Beta	1.17	1.39E+03
SWS	FTR	EB/PL/KS	4.14E-07	4.70E-06	6.56E-06	1.91E-05	Gamma	1.09	1.66E+05
SWS	FTR	JNID/IL	3.43E-08	1.30E-07	1.50E-07	3.31E-07	Gamma	2.50	1.67E+07
NSW	FTR	EB/PL/KS	4.34E-07	4.85E-06	6.72E-06	1.96E-05	Gamma	1.10	1.63E+05
SWS-NE	FTR	EB/PL/KS	6.39E-05	6.18E-04	8.36E-04	2.36E-03	Beta	1.19	1.42E+03
SWS-NE	FTR	EB/PL/KS	8.93E-07	4.69E-06	5.67E-06	1.39E-05	Gamma	1.82	3.20E+05
CSW	FTR	JNID/IL	4.88E-06	6.90E-06	6.98E-06	9.41E-06	Gamma	25.50	3.65E+06

Figure 2.11: MDP-FTR-AFW (mission) from NUREG/CR-6928

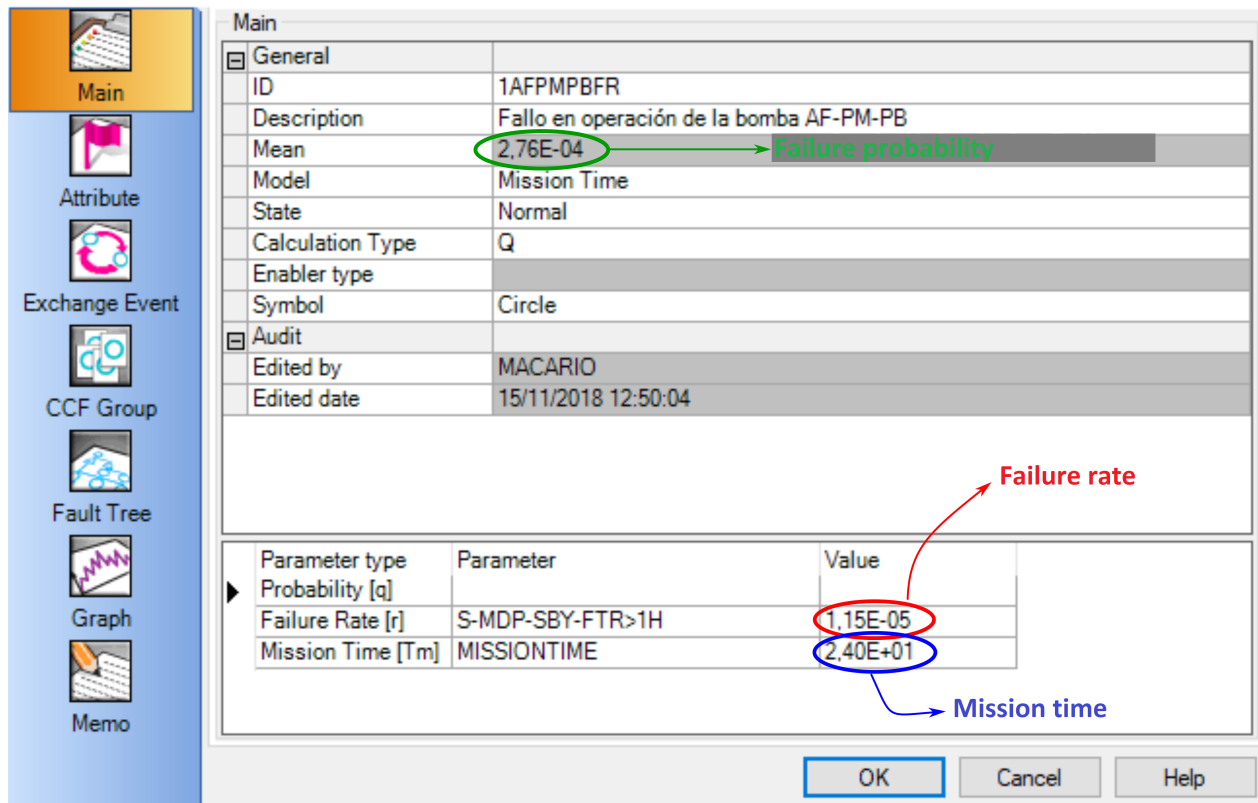


Figure 2.12: MDP-FTR-AFW modeling in RiskSpectrum

Table 2.6: Failure modes for valves in the SPAR-CSN Model

	Failure Mode	Parameter	Units	Description
Manual, check, motorized and pneumatic valves	FTO/C	p	-	Fail to open/Fail to close
	SOP	λ	1/h	Spurious operation (fail to remain open/close)
	ELS	λ	1/h	Small external leakage
	ELL	λ	1/h	Large external leakage
	ILS	λ	1/h	Small internal leakage
	ILL	λ	1/h	Large internal leakage
	FC	λ	1/h	Fail to control
	Failure Mode	Parameter	Units	Description
Safety valves	FTO	p	-	Fail to open
	FTC	p	-	Fail to close
	SOP	λ	1/h	Spurious opening
	FTCL	p	-	Fail to close after passing liquid

2.1.4.3 Common Cause Failures

Common-cause failures (CCFs) are among the most significant factors contributing to the loss of functionality in systems. The American Society of Mechanical Engineers (ASME) defines a CCF as “a failure of two or more components during a short period of time as a result of a single shared cause [ASME/ANS, 2015]. As outlined in the NUREG/CR-626 report, Common Cause Failure Database and Analysis System, Event Data Collection, Classification, and Coding [Wierman et al., 2007], a CCF event is characterized by the following four criteria:

- Two or more components fail or degrade within the same plant and system.
- The failures occur within a specified timeframe.
- The failures come from a single shared cause, linked by a coupling mechanism, in a way that implies other components in the group vulnerable to the same cause and failure mode.
- The failures are not triggered by equipment outside the defined component boundary.

The CCF Parameter Estimations document (NUREG/CR-5497) [NRC, 2015a] provides the necessary factors for CCFs. The SPAR-CSN model uses alpha factors (α). The alpha factors are divided for each component by Component Group Size (CCCG). In NUREG/CR-5497, they are organized by component system failure mode. The distribution function of the alpha factors is a beta-type distribution (β). Finally, the CCF probability is obtained by multiplying the α factor by the individual failure probability of the component under analysis.

$$P_{CCF} = \alpha \times P_{ind}$$

For example, for the motor-driven pumps of the Auxiliary feedwater with failure mode: Fail to Start. The Alpha factors are shown in Tables 2.7 and 2.8. The NUREG/CR-5497 considers:

- Date Range: 1997 through 2015
- Total Number of Independent Failure Events: 44.7
- Total Number of Common-Cause Failure Events: 2

Table 2.7: Alpha factors distributions for MDP-FTS-AFW [NRC, 2015a]

CCCG = 2							
Alpha Factor	5th%	Mean	Median	95th%	MLE	a	b
α_1	0.9069235	0.9604720	0.9665835	0.9930869	0.9607946	4.701E+01	1.935E+00
α_2	6.92E-03	3.95E-02	3.34E-02	9.31E-02	3.92E-02	1.935E+00	4.701E+01
CCCG = 3							
Alpha Factor	5th%	Mean	Median	95th%	MLE	a	b
α_1	0.9269559	0.9643765	0.9678764	0.9898157	0.9645327	8.394E+01	3.101E+00
α_2	2.36E-03	1.82E-02	1.47E-02	4.61E-02	1.33E-02	1.584E+00	8.546E+01
α_3	2.11E-03	1.74E-02	1.39E-02	4.49E-02	2.22E-02	1.517E+00	8.553E+01
CCCG = 4							
Alpha Factor	5th%	Mean	Median	95th%	MLE	a	b
α_1	0.9347529	0.9654366	0.9679465	0.9875420	0.9680758	1.182E+02	4.230E+00
α_2	2.86E-03	1.62E-02	1.36E-02	3.82E-02	1.01E-02	1.978E+00	1.204E+02
α_3	1.21E-03	1.15E-02	8.94E-03	3.04E-02	1.34E-02	1.404E+00	1.210E+02
α_4	2.28E-04	6.93E-03	4.49E-03	2.19E-02	8.40E-03	8.477E-01	1.215E+02

Table 2.8: Alpha factor and MGL Parameters for MDP-FTS-AFW

Alpha Factor	CCCG=2	CCCG=3	CCCG=4
α_1	9.61E-01	9.65E-01	9.68E-01
α_2	3.92E-02	1.33E-02	1.01E-02
α_3		2.22E-02	1.34E-02
α_4			8.40E-03
MGL	CCCG=2	CCCG=3	CCCG=4
1-Beta	9.61E-01	9.65E-01	9.68E-01
Beta	3.92E-02	3.55E-02	3.19E-02
Gamma		6.25E-01	6.84E-01
Delta			3.85E-01
Avg. Impact Vector	CCCG=2	CCCG=3	CCCG=4
Adj. Ind. Events	35.76	53.64	71.52
N_1	1.0000	0.7500	0.5000
N_2	1.5000	0.7500	0.7500
N_3		1.2500	1.0000
N_4			0.6250

In the SPAR-CSN model, the AFW system consists of 2 motor pumps, so $\mathbf{CCCG} = \mathbf{2}$ and since this is the probability of common cause failure of both pumps, then $\alpha_2 = \mathbf{3.95E-02}$ (Table 2.9 and Figure 2.13).

Table 2.9: Alpha factor for MDP-FTS-AFW in SPAR-CSN Model

CCCG = 2							
Alpha Factor	5th%	Mean	Median	95th%	MLE	a	b
α_1	0.9069235	0.9604720	0.9665835	0.9930869	0.9607946	4.701E+01	1.935E+00
α_2	6.92E-03	3.95E-02	3.34E-02	9.31E-02	3.92E-02	1.935E+00	4.701E+01

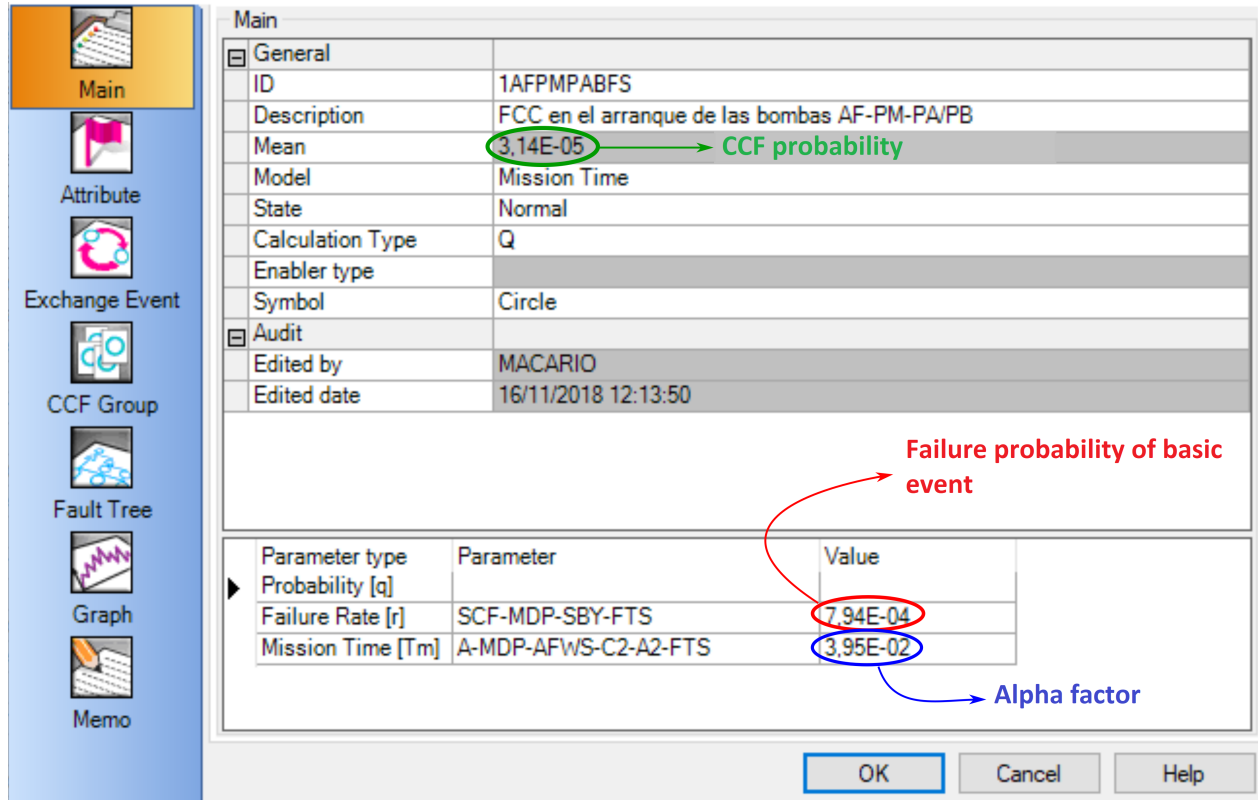


Figure 2.13: MDP-AFW CCFs modeling in RiskSpectrum

To analyze the case of CCF of 2 of the relief valves, it must be considered that the system has 3 SG-PORVs, so $CCCG = 3$. Factor alpha $\alpha_2 = 2.28E-02$ and SG-PORV fails to open: $P_{ind} = 3.00E-03$. So 2/3 SG-PORV fails to open by CCF:

$$P(2/3)_{CCF} = P_{ind} \times \alpha_2 = 3.00E - 03 \times 2.28E - 02 = 6.84E - 05$$

2.1.4.4 Example of AFW system failures

In order to exemplify the previous subsections, Tables 2.10 and 2.11 present CCF and failure probabilities of different components of the AFW system.

Table 2.10: CCF for AFW System

Common Cause Failures		Basic event	Alpha factor	SPAR-CSN
MOV	CCF to valve opening (MS-VM-T2/T3)	4.21E-04	1.52E-02	6.40E-06
AOV	CCF to pneumatic valves closing (GP-VN-G1/G2/G3)	3.63E-04	2.01E-02	7.30E-06
	CCF to pneumatic valve closing (GP-VN-G2/G3)	3.63E-04	3.12E-02	1.13E-05
	CCF to pneumatic valves closing (VN-PA/PB-R)	3.63E-04	2.35E-02	8.53E-06
	CCF for valve control (VN-G1/G2/G3/T1/T2/T3)	5.47E-06	1.90E-03	1.04E-08
	CCF for valve control (VN-PA/PB)	5.47E-06	3.05E-02	1.67E-07
	CCF for valve control (VN-T1/G1; VN-T2/G2; VN-T3/G3)	5.47E-06	6.34E-03	3.47E-08
CKV	CCF to valve opening (VR-G1/G2/G3/G3-C)	9.24E-06	8.18E-03	7.56E-08
	CCF to valve opening (VR-G2/G3)	9.24E-06	1.82E-02	1.68E-07
	CCF to valve opening (VR-PA/PB-S/D)	9.24E-06	3.43E-02	3.17E-07
	CCF to valve opening (MS-VR-T2/T3)	9.24E-06	2.10E-02	1.94E-07
	CCF to valve opening (VR-T1/G1; VR-T2/G2; VR-T3/G3)	9.24E-06	1.82E-02	1.68E-07
MDP	CCF at motor pump start-up	7.94E-04	3.95E-02	3.14E-05
	CCF at motor pump operation	2.76E-04	2.74E-02	7.56E-06
FAN	CCF at start-up of cooling units	6.52E-04	1.97E-02	1.28E-05
	CCF at cooling unit operation	4.76E-03	3.05E-02	1.46E-04

Table 2.11: AFW System equipment failure probabilities

Basic event		Failure	NUREG/CR-6928		SPAR-CSN
MOV	Motorized valve fails to open	DEMAND	4.21E-04	-	4.21E-04
	Motorized valve fails to remain open	MISSION	3.24E-08	24 h	7.78E-07
AOV	Pneumatic valve fails to open	DEMAND	3.91E-04	-	3.91E-04
	Pneumatic valve fails to close	DEMAND	3.63E-04	-	3.63E-04
	Pneumatic valve fails to remain open	MISSION	1.05E-07	24 h	2.52E-06
	Pneumatic valve fails to control	MISSION	2.28E-07	24 h	5.47E-06
CKV	Check valve fails to open	DEMAND	9.24E-06	-	9.24E-06
	Check valve fails closed	DEMAND	1.57E-04	-	1.57E-04
XVM	Manual valve fails to remain open	MISSION	5.08E-08	24 h	1.22E-06
MDP	Motor pump fails to start	DEMAND	7.94E-04	-	7.94E-04
	Motor pump fails to run	MISSION	1.15E-05	24 h	2.76E-04
TDP	Turbo pump fails to start	DEMAND	4.33E-03	-	4.33E-03
	Turbo pump fails to run (first hour)	MISSION	3.66E-03	1 h	3.66E-03
	Turbo pump fails to run (>1h)	MISSION	1.45E-03	23 h	3.28E-02
FAN	Cooling unit fails to start	DEMAND	6.52E-04	-	6.52E-04
	Cooling unit fails to run	MISSION	1.98E-04	24 h	4.76E-03
TNK	CST breakdown or collapse	DEMAND	6.26E-06	-	6.26E-06
UMT	Motor pump unavailability due to maintenance or testing	DEMAND	3.34E-03	-	3.34E-03
	Turbo pump unavailability due to maintenance or tests	DEMAND	5.24E-03	-	5.24E-03

2.2 Generic SPAR-CSN Model Results

In this section the main results related to the model overall CDF are presented and discussed. Tables 2.12 and 2.13 show the overall CDF obtained for the SPAR-CSN generic model and each ET contribution, considering and not considering dependencies between human actions, while Figures 2.14 and 2.15 show a sector diagram comparing the different ET contributions to CDF. The following remarks can be made:

- The CDF value is on the same order of magnitude of other SPAR models (e.g., [Watanabe et al., 2018] reports a CDF of 3.01E-05 for a 4-loop generic PWR plant, including human actions dependencies).
- The most important contribution to CDF comes from Generic Transients. This is a common result among other PSA models of PWR 3-loop NPPs. In particular, the main MCS consists of a Generic Transient with a human error controlling the SGs level followed by a human error in the Feed and Bleed (F&B) maneuver.
- The application of the human actions dependency analysis increases the CDF value by 2 orders of magnitude.

Table 2.12: CDF associated to every IE and overall CDF of the SPAR-CSN generic model. not considering human actions dependencies

IE ID	CDF (1/y)	IE ID	CDF (1/y)
LBLOCA	1.15E-08	MSLB-US	1.56E-09
MBLOCA	5.34E-08	MSLB-DS	3.17E-08
SBLOCA	1.01E-07	SGTR	1.51E-08
GT	2.07E-07	LCWA	7.11E-10
ATWS	4.93E-08	LNSW	5.57E-11
LC	1.45E-08	LOOP/SBO	1.21E-07
LDC-A	7.26E-09	LDC-B	9.98E-09
TOTAL			6.24E-07

Table 2.13: CDF associated to every IE and overall CDF of the SPAR-CSN generic model. considering human actions dependencies

IE ID	CDF (1/y)	IE ID	CDF (1/y)
LBLOCA	1.15E-08	MSLB-US	5.99E-08
MBLOCA	1.29E-09	MSLB-DS	7.62E-07
SBLOCA	1.45E-07	SGTR	3.45E-06
GT	7.39E-06	LCWA	7.12E-10
ATWS	5.02E-08	LNSW	2.18E-09
LC	5.15E-07	LOOP/SBO	4.52E-07
LDC-A	1.51E-08	LDC-B	1.60E-08
TOTAL			1.2871E-05

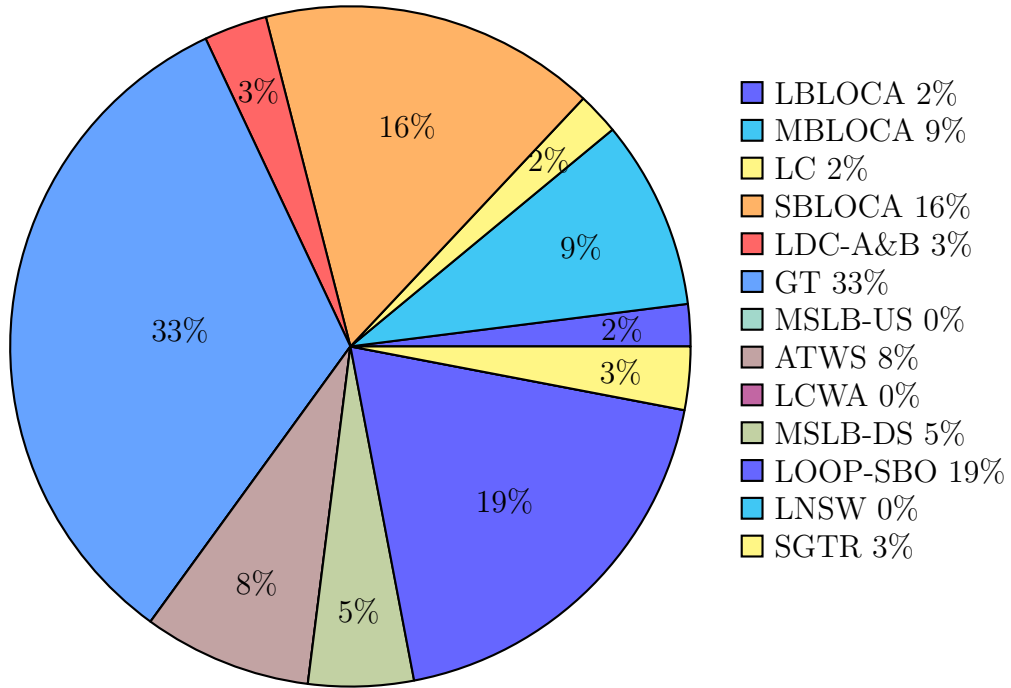


Figure 2.14: Contribution of each IE to the overall CDF of the SPAR-CSN model, not considering human actions dependencies

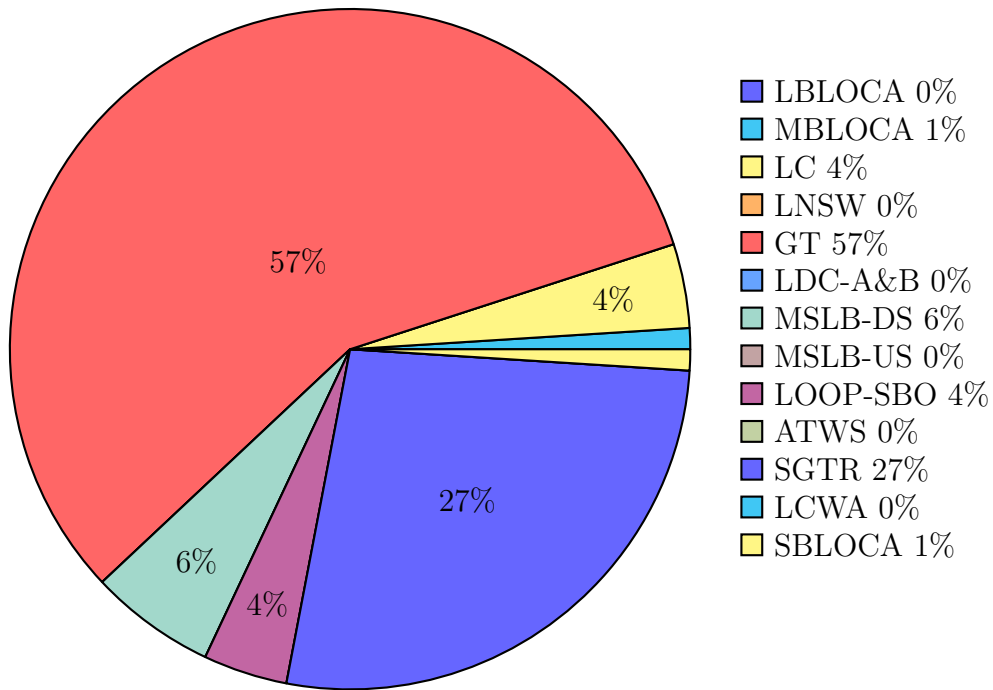


Figure 2.15: Contribution of each IE to the overall CDF of the SPAR-CSN model, considering human actions dependencies

Chapter 3

Application of Probabilistic Safety Assessment to FLEX Strategies

Following the Fukushima Dai-ichi accident in 2011, the efforts of the nuclear industry and regulatory bodies at the international level have focused on increasing the defense in depth of NPPs. The culmination of the joint work of different agencies, such as Western European Nuclear Regulators Association or the NRC and the nuclear industry, has resulted in the creation of FLEX Strategies to increase plant capability in the face of Beyond Design-Basis External Event (BDBEE) with the objective of avoiding significant reactor fuel degradation and ensuring the integrity of the fuel pool and containment. FLEX strategies must be able to counteract Extended Loss of AC Power (ELAP) and Loss of Ultimate Heat Sink (LUHS) conditions at the same time through alternative human actions and the use of portable equipment stored inside or outside the facility.

The FLEX strategies can be utilized also to prevent core damage caused by internal events sequences. As a result, incorporating these FLEX strategies into L1PSA allows for analyzing their influence on the CDF. Related with this issue, the U.S. Nuclear Energy Institute (NEI) developed the document NEI 16-06, Crediting Mitigating Strategies in Risk-Informed Decision Making [NEI, 2016b] provides an approach in crediting the use of portable equipment associated with various plant mitigating strategies to restore or maintain various safety functions during beyond design basis conditions and the loss of permanently installed plant equipment. Also, this guidance describes examples on the use of FLEX-specific portable equipment and strategies.

Regarding to the implementation of the FLEX equipment on PSA found in the literature, the following can be highlighted: Fujioka et al. [Fujioka et al., 2017] conducted a study to analyze the effectiveness of spent fuel pool (SFP) FLEX strategies in a LUHS scenario caused by a tsunami; Jabbari et al. [Jabbari et al., 2020] analyze the introduction of a portable air-cooled diesel generator for ELAP scenario mitigation in VVER-1000 and verified its effectiveness; Gjorgiev et al. [Gjorgiev et al., 2017] analyze the introduction of independent

water storage systems to enhance the safety of a typical PWR in a SBO scenario; Hakobyan et al. [Hakobyan and Nierode, 2017] analyze the implementation FLEX strategies in Surry power plant PSA; Zhang et al. [Zhang and Ma, 2020] study the impact of FLEX strategies in a dual-unit site experiencing LOOP; Kim et al. [Kim et al., 2021] include portable equipment in the PSA model for WH600, WH900, OPR1000 and APR1400 reactors in Korea obtaining the CDF reduction; Tabadar et al. [Tabadar et al., 2019a, Tabadar et al., 2019b] analyze the effectiveness of portable equipment in LUHS accidents for VVER-1000. Other researchers, such as Kim & Kim [Kim and Kim, 2021]; Lim [Lim, 2018]; Islam et al. [Islam and Lim, 2017]; Webster [Webster, 2017] and París et al. [París et al., 2019] have also employed the PSA method to study FLEX strategies impact for different scenarios and NPPs.

Also, Idaho National Laboratory has developed extensive work regarding FLEX strategies, including incorporating them into PWR models: [Ma et al., 2019, Shah et al., 2023] crediting FLEX in PSA, incorporating them to the generic LOOP/SBO and MBLOCA; and into BWRs [Ma et al., 2020b]; an approach on how to perform HRA for FLEX applications [Ma et al., 2020b] and later extending the EMERALD tool to FLEX dynamic HRA [Park et al., 2021b, Park et al., 2022]; and reviewed the development and use of weakly informed a priori distributions to obtain industry-wide probability and failure rate distributions [Gentillon et al., 2020]. Finally, the NRC also credits FLEX in their accident sequence precursor analysis [NRC, 2022a, NRC, 2023] and has already incorporated the FLEX strategies in their all 71 SPAR models [Ng, 2020]. Figure 3.1 shows a time-line of FLEX strategies.

While FLEX strategies are initially proposed to cope with an ELAP and LUHS following a BDBEE, this work is focused on modeling and analyzing the total impact of FLEX strategies on the L1PSA, regardless of the initiating event, at nominal power.

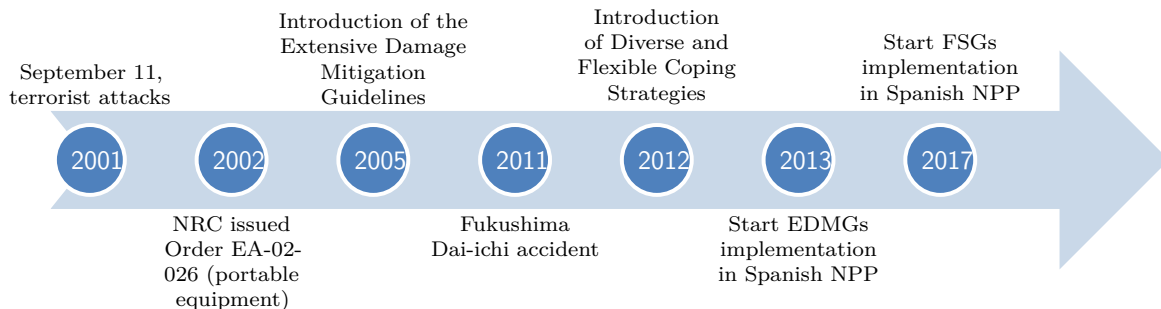


Figure 3.1: FLEX time-line

The present chapter is organized as follows: Section 3.1 resumes some post-Fukushima improvements, introduces the most common FLEX strategies used and how they are linked to the other mitigation strategies. Section 3.2 describes how the Extensive Damage Mitigation Guidelines and FLEX Support Guidelines were implemented in USA and Spain. Section 3.3 outlines the FLEX modeling and describes the specific actions and equipment considered

in the FLEX Strategies, also describes the modeling of event trees, fault trees, HRA and reliability data used for FLEX equipment. Section 3.4 summarizes the results obtained with the model and how the inclusion of FLEX strategies affects the risk. Finally, Section 3.5 presents a sensitivity analysis to evaluate the impact of each FLEX strategy separately.

3.1 FLEX Strategies and Portable Equipment

Following the Fukushima Dai-ichi accident, the culmination of the joint work of different agencies to increase the defense in depth, resulted in the creation of NEI 12-06, Diverse and Flexible Coping Strategies (FLEX) Implementation Guide [NEI, 2012], where the term “FLEX Strategies” is defined for the first time. This document defines FLEX as strategies to increase plant capability against of BDBEE with the objective of avoiding significant reactor fuel degradation and ensuring the integrity of the fuel pool and containment (Figure 3.2). FLEX strategies must be able to counteract ELAP and LUHS conditions at the same time, as a consequence of a BDBEE, through alternative human actions and the use of portable equipment, stored inside or outside the facility. The portable equipment was incorporated in US after September 11, 2001, but not in other countries, e.g., Spain. The guide NEI 12-06 proposes an implementation of FLEX strategies divided into three phases:

- Phase 1: Utilization of “pre-staged” fixed plant equipment, this equipment belongs to the safeguards systems or to other systems and can be realigned to cope with the accident situation (e.g. diesel driven-pump of the Fire Protection system, DDP-FPS; hydrostatic test pump).
- Phase 2: Transition from using fixed plant equipment to portable equipment or FLEX equipment located on site to maintain safety functions for a longer period.
- Phase 3: Obtaining additional mitigation capacity by transporting and deploying redundant FLEX equipment from a site outside the nuclear plant. This last stage allows safety functions, such as power supply or coolant injection, to be extended over a virtually unlimited period.

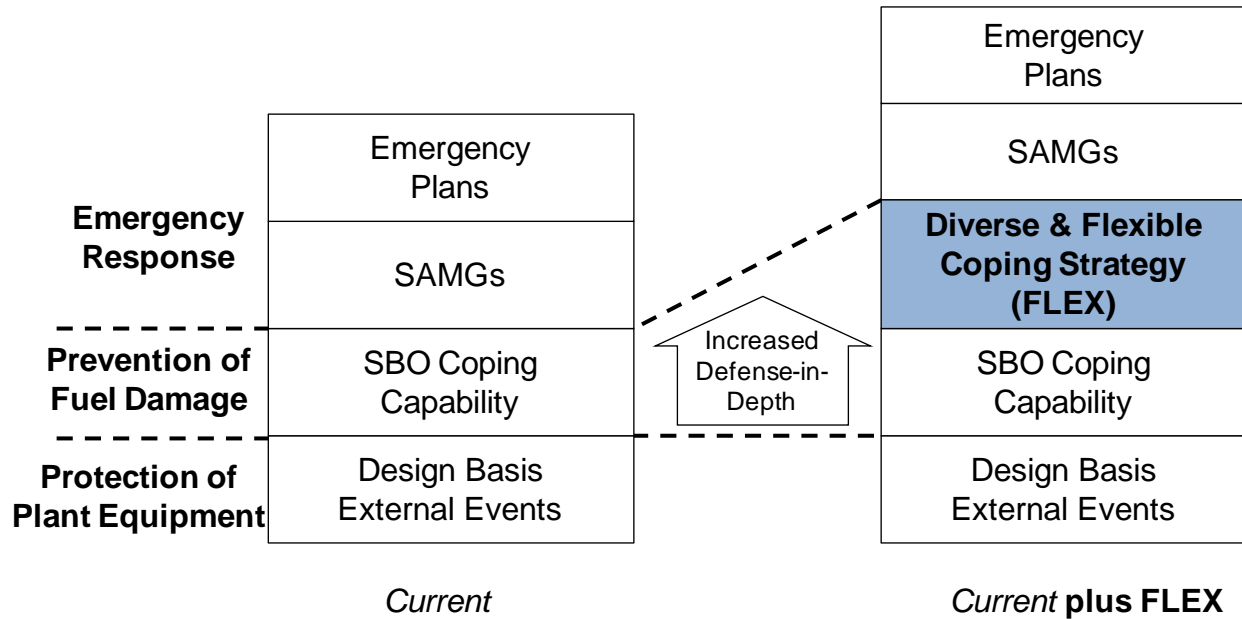


Figure 3.2: FLEX Enhances Defense-in-Depth [NEI, 2012]

The portable equipment most commonly incorporated in NPPs to implement FLEX strategies Phase 2 includes, Table 3.1:

1. FLEX-DG: Portable diesel generator for alternate AC supply.
2. FLEX-SGP: Low-pressure (LP) portable pump for alternative injection to SGs.
3. FLEX-MUP: High-pressure (HP) portable pump for alternative make-up and boration into the RCS.
4. FLEX-HCP: LP high-capacity portable pump for coolant drive from the ultimate heat sink.
5. FLEX-SUP: LP submersible pump for pumping coolant from the ultimate heat sink.

It is important to note that the mass-flow rate of some pumps vary greatly from plant to plant and, therefore, may not have the same effectiveness in FLEX strategies. On the other hand, it should be noted that the characteristics of the DDP-FPS are approximately 200 kg/s and 10 bar, maximum mass-flow rate and pressure head (FLEX strategies Phase 1).

Other devices are also mentioned: small portable containers of fuel; gas turbine-driven generator (GTG) for alternative AC supply or DC supply; hands-free battery powered portable lights; and portable instrumentation. In addition, the U.S. nuclear industry applies the $N+1$ approach, this implies that for every unit present on the site, additional equipment, connection points, or equivalent safeguard measures must be installed to ensure defense-in-depth [NEI, 2012]. The NEI-12-06 approach is not applied worldwide, e.g. South Korea applies the RObust Coping Strategy [Kim et al., 2016] and Multi-barrier Accident Coping

CHAPTER 3. APPLICATION OF PROBABILISTIC SAFETY ASSESSMENT TO FLEX STRATEGIES

Strategy (MACST) [Park et al., 2018]; Canada applies the Emergency Mitigating Equipment Guidelines [CNSC, 2015]; and France applies Hardened Safety Core [Xu and Zhang, 2021b]. Figures 3.3, 3.4, 3.5 and 3.6 show examples of FLEX equipment and FLEX facilities in different NPPs. Table 3.1 summarized the FLEX equipment available in Spanish and US PWR NPPs.



Figure 3.3: FLEX equipment at the secure site of Vogtle NPP, USA [Southern Nuclear, 2021]



Figure 3.4: FLEX dome at North Anna NPP in Virginia [Breunig, 2017]



Figure 3.5: Portable diesel generator FLEX-DG for use in FLEX strategies [ANAV, 2013]

Table 3.1: Portable equipment in PWR-WEC US and Spain NPPs

NPP	FLEX-DG P[kW]/U[V]	FLEX-SGP Q[kg/s]/P[bar]	FLEX-MUP Q[kg/s]/P[bar]	FLEX-HCP Q[kg/s]/P[bar]	FLEX-SUP Q[kg/s]/P[bar]
Braidwood [NRC, 2017]	350/480	30/28	2/103	91/10	-
Byron [NRC, 2013a]	500/480	18/21	2/103	66/15	-
Donald C. Cook [NRC, 2015b]	N/A/600 & N/A/480	36/28	1.6/107	20/21	-
Joseph M. Farley [Southern Nuclear, 2013]	N/A/600 & N/A/480	18/21	1.2/36	6/3	-
McGuire [NRC, 2016a]	500/600	18/28	N/A	91/N/A	6/N/A
North Anna [NRC, 2016b]	40/120-240 & 350/480	N/A	3/N/A	N/A	-
Point Beach [NextEra Energy, 2015]	404/480	20/28	4/N/A & 1/138	-	-
Robinson [Duke Energy, 2015]	N/A/480	18/69	4/138	36/5	-
Salem [NRC, 2014]	N/A/480	N/A	N/A	42/N/A	N/A
Seabrook [NRC, 2016c]	405/480 & 30/N/A	20/28	1/103	-	20/3
Sequoyah [NRC, 2016d]	3000/6900 & 225/480	-	2/41	302/10	302/2
Shearon Harris [NRC, 2016e]	830/480	20/25	2/110	181/10	-
South Texas [STP Nuclear, 2016]	1000/480	18/35	4/48	60/10	-
Summer [NRC, 2016g]	80/480 & 300/480	30/0.4 & 30/35	4/138	48/N/A	N/A
Surry [NRC, 2016f]	350/480 & 40/120	18/31	3/138	72/10	-
Vogtle [NRC, 2016h]	350/480	21/28	1/36	N/A	30/7
Almaraz [CSN, 2016, CSN, 2015a]	N/A	29/15	12/23	440/10	N/A
Ascó [CSN, 2013a]	550/N/A	N/A	19/28	370/11	N/A
Vandellós-II [CSN, 2014]	550/N/A	19/20	4/20	246/12	N/A



Figure 3.6: Submersible portable pump FLEX-SUP for use in FLEX strategies [ANAV, 2013]

3.2 Extensive Damage Mitigation Guidelines and FLEX Support Guidelines in USA and Spain

In response to the September 11, 2001, terrorist attacks, the NRC issued Order EA-02-026, known as the Interim Safeguards and Security Compensatory Measures (ICM) Order, in February 2002. This order imposed new requirements on licensees, specifically in Section B.5.b, which mandated the implementation of mitigation strategies using readily available resources to ensure the maintenance or restoration of core cooling, containment, and SFP cooling capabilities, with both, pre-stage and portable equipment [Caro, 2012]. These strategies were intended to address the potential Loss of Large Areas (LOLA) of the facility due to various types of large fires and explosions, including those caused by aircraft impacts beyond the design basis. As a result, NEI introduced in 2005 the Extensive Damage Mitigation Guidelines (EDMGs) [Xu et al., 2021]. In December 2006, the NRC endorsed Revision 2 of NEI 06-12 [NEI, 2009], as an acceptable method for developing the mitigation strategies required in Section B.5.b.1 of the ICM Order. Section 3.3 of NEI 06-12 guideline establishes enhanced site response strategies for PWRs:

- Makeup to refueling water storage tank (RWST)
- Manually depressurize SGs to reduce inventory loss
- Manual operation of turbine-driven (or diesel-driven) pump of Auxiliary Feedwater system (TDP-AFW)
- Manually depressurize SGs and use portable pump
- Makeup to condensate storage tank (CST)/AFW storage tank
- Containment flooding with portable pump
- Portable sprays

Later, in 2012, after the Fukushima accident, NEI released the first version of the NEI 12-06 guide [NEI, 2012]. To complement the guidelines in NEI 12-06, the Pressurized Water Reactor Owners Group (PWROG) in collaboration with Westinghouse started the project titled “Emergency Response to Extended Station Blackout Events” (PA-PSC-0965). The project objectives included creating universal FLEX Support Guidelines (FSGs), recognizing links to existing EOPs, identifying crucial plant instruments to be utilized during battery load-shedding approaches, and establishing a general framework for the timing and selection of portable equipment and techniques to respond to an ELAP [PWROG, 2012]. Westinghouse document WCAP-17601 presented the findings of the baseline thermal-hydraulic analyzes conducted to examine the response of the primary system and the cooling down of the plant during an ELAP [Stringfellow, 2017]. The analyzes aimed to provide a better understanding of the system’s behavior and trends in such events. Then, in 2014, the report WCAP-17792-P, addressed issues identified in WCAP-17601 [Stringfellow, 2017].

The FSGs should include clear instructions for operators on when and how to use the FLEX equipment. These guidelines should be tailored to the specific needs of each plant, considering the FLEX strategies and modifications adopted for that particular plant. The procedures must be implemented in a plant-specific manner to ensure their effectiveness. The list of standardized FSGs is as follows [NextEra Energy, 2015, Xu and Zhang, 2021a] :

- FSG-0: FLEX information and reference guide
- FSG-1: Long-term RCS inventory control
- FSG-2: Alternate AFW suction source
- FSG-3: Alternative LP feedwater
- FSG-4: ELAP power management
- FSG-5: Initial Assessment and FLEX Equipment Staging
- FSG-6: Alternate CST Makeup
- FSG-7: Loss of vital instrument or control power
- FSG-8: Alternate RCS boration
- FSG-9: Low decay heat temperature control
- FSG-10: Passive RCS injection isolation
- FSG-11: Alternate SFP makeup equipment deployment
- FSG-12: Alternate containment cooling (shutdown only)
- FSG-13: Transition from FLEX equipment
- FSG-14: Shutdown RCS makeup

In Spain the process followed was different from that in the USA. In 2011, the CSN issued Complementary Technical Instructions ITC-1 [CSN, 2011a] and ITC-2 [CSN, 2011b]. These

concentrate the main guidelines of the new extensive damage mitigation strategies, introducing portable equipment, alternative water sources and alternative power supplies. The ITC-2 was related to the NEI 06-12 guide and focused on the possibility of LOLA. As a result of the analyzes carried out by the plants, imposed by these technical instructions, the new EDMGs and the new portable equipment were implemented during 2012-2013 [Hernández et al., 2015, Queral and Alonso-Escos, 2023]. The way these strategies are implemented depends on the plant itself and, therefore, varies from plant to plant. A generic EDMGs list corresponding to the Spanish PWR NPPs is [CSN, 2013b, Gil-Rodríguez et al., 2016, Queral and Alonso-Escos, 2023]:

- EDMG-1: Instrumentation recovery
- EDMG-1.1: Manual operation of the TDP-AFW
- EDMG-1.2: Decrease pressure of SGs and flow control of AFW
- EDMG-1.3: Decrease pressure of SGs and flow control with portable HP pump
- EDMG-1.4: Decrease pressure of SGs and flow control with portable LP pump
- EDMG-1.5: Decrease pressure of SGs and flow control with DDP-FPS
- EDMG-1.6: Supply to the water tank to support the AFW system
- EDMG-2.1: Water supply to the RCS by means of hydrostatic test pump
- EDMG-2.2: Water supply to the RCS by portable HP pump through Residual Heat Removal (RHR) system
- EDMG-2.3: Water supply to RWST
- EDMG-3.1: Water supply to the SFP
- EDMG-3.2: Direct spraying of the SFP
- EDMG-3.3: Minimizing leakage in the SFP
- EDMG-4.1: Flooding of containment through containment spray lines
- EDMG-4.2: Spraying of containment leaks
- EDMG-4.3: Filling of the reactor cavity
- EDMG-4.4: Pressure control of the containment building
- EDMG-5.1: Installation of portable diesel generator and connection of loads
- EDMG-5.2: Portable LP pump installation and start-up
- EDMG-5.3: Portable HP pump installation and start-up
- EDMG-5.4: Refueling of portable equipment with tanker truck
- EDMG-5.5: Portable communication module installation and start-up
- EDMG-5.6: Portable meteorological station installation and start-up

In recent years, due to the development of the FSGs, implemented and updated in Spain from 2017 to 2022 [Queral and Alonso-Escos, 2023], the structure of the procedures and guidelines has been changed in Spanish NPPs. The present relationship between guidelines and procedures in Spain depends on the nature of the initiating event, Figure 3.7:

- The first possible scenario is that corresponding to L1PSA, where an incident is caused by an internal event, the management is done through the EOPs (at power) or the Abnormal Operating Procedures (AOPs, at shutdown). In the case that the strategies considered in the EOPs/AOPs are not effective, the FSGs that are referenced in these EOPs/AOPs will be applied, Table 3.2, and in the event that these are not effective either, the strategies of the EDMGs can also be used at the discretion of the Senior Reactor Operator. If despite the application of all the above strategies, the incident cannot be controlled, a severe accident situation will be reached and it will be necessary to apply the Severe Accident Mitigation Guidelines (SAMGs), this situation is part of the L2PSA.
- The second possible scenario is a BDBEE that causes catastrophic damage, in which case the Emergency Management Guideline (EMG) must be applied; these are alternative methods in events that could result in severe damage, such as the total loss of power and/or inability to operate from the main control room and the remote shutdown panel, or catastrophic failure of vital structures. In these situations, the expected command and control of the emergency may not be available. The EMGs are directly related to the EDMGs.

As a final remark, the relationship of EOPs, FSGs, EDMGs and SAMGs in Spanish NPPs is similar, although with certain differences, to the structure proposed by the NEI 14-01 guide [NEI, 2016a] adopted by the US NPPs.

In order to compare EDMGs and FSGs, it can be mentioned that EDMGs motivation and objectives were implemented to mitigate LOLA and maintain fuel cooling and containment capabilities. FSGs were implemented to mitigate BDBEE and also to maintain fuel cooling and containment capabilities. Regarding the strategic features, EDMGs propose the establishment of initial command and control in case the normal command and control is disrupted, and the use of temporary and portable equipment in a local and manual mode. In contrast, FSGs use the three-phase FLEX approach [Kim et al., 2016].

According to the scope of the generic PWR L1PSA model used (see Chapter 2), the FSGs that have been considered are FSG-1, FSG-3, FSG-4, FSG-5, FSG-7 and FSG-8. Also, are considered EDMG-1.1, EDMG-1.4, EDMG-1.5, EDMG-2.2, EDMG-2.3, EDMG-5.1, EDMG-5.2 and EDMG-5.3.

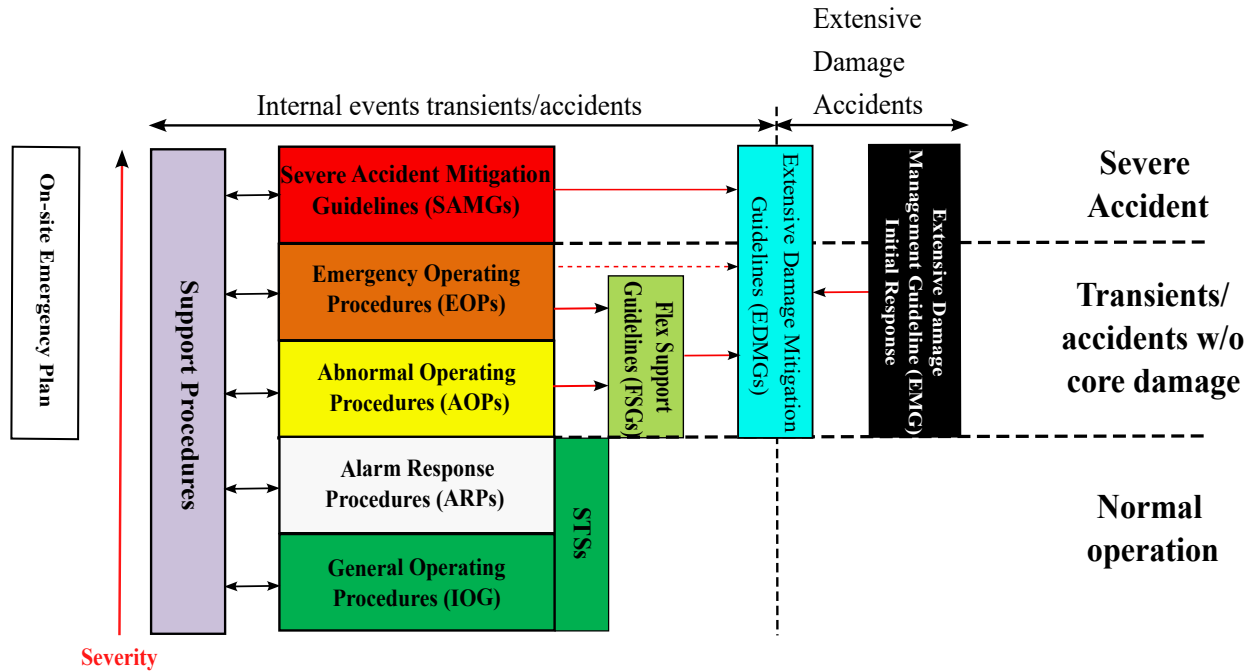


Figure 3.7: Procedures and guidelines relationships in Spain [Queral and Alonso-Escos, 2023]

Table 3.2: Entries to FSG procedures [Queral and Alonso-Escos, 2023]

FSG	From EOP	From AOP
FSG-1	ECA-0.0; FR-C.1; FR-C.2	
FSG-2	ECA-0.0	ARG-4
FSG-3	ECA-0.0; FR-H.1; FR-C.1	ARG-4
FSG-4	ECA-0.0	ARG-4
FSG-5	ECA-0.0	ARG-4
FSG-6	ECA-0.0	ARG-4
FSG-7	ECA-0.0	ARG-4
FSG-8	ECA-0.0; FR-S.1	
FSG-9	ECA-0.0	ARG-4
FSG-10	ECA-0.0	ARG-4
FSG-11		ARG-4
FSG-12	ECA-0.0	ARG-4
FSG-13	ECA-0.0	ARG-4
FSG-14		ARG-4

3.3 FLEX Strategies Modeling

The incorporation of FLEX strategies into the PSA model requires the modification of the ET, development of the FT corresponding to each new FLEX system and FLEX strategies; analysis of new human actions; and analysis of databases for FLEX equipment. The equipment considered within the modeling is: 1 DDP-FPS; 1 FLEX-SGP; 1 FLEX-MUP; 1 FLEX-DG; 1 FLEX-HCP. The inclusion of FLEX strategies in the L1PSA has been done considering a series of specific actions and equipment. These FLEX Actions (FA) are:

- FA-1: Alternative LP feedwater to the SGs by FLEX-SGP.
- FA-2: Alternative LP feedwater to the SGs by DDP-FPS.
- FA-3: Manual operation of TDP-AFW after DC loss.
- FA-4: Long term RCS inventory control by FLEX-MUP.
- FA-5: Alternative RCS boration by FLEX-MUP.
- FA-6: Alternative RWST inventory control by DDP-FPS.
- FA-7: Alternative RWST inventory control by FLEX-HCP.
- FA-8: Supply of AC power by FLEX-DG.
- FA-9: DC load shedding to increase battery life.

As previously commented, the use of portable equipment permits the recovery and/or support of safety systems and Figures 3.8, 3.9, 3.10 and 3.11 provides examples of these strategies: external pumps injecting water into the RPV or SGs [[Fernandez-Cosials et al., 2020](#)].

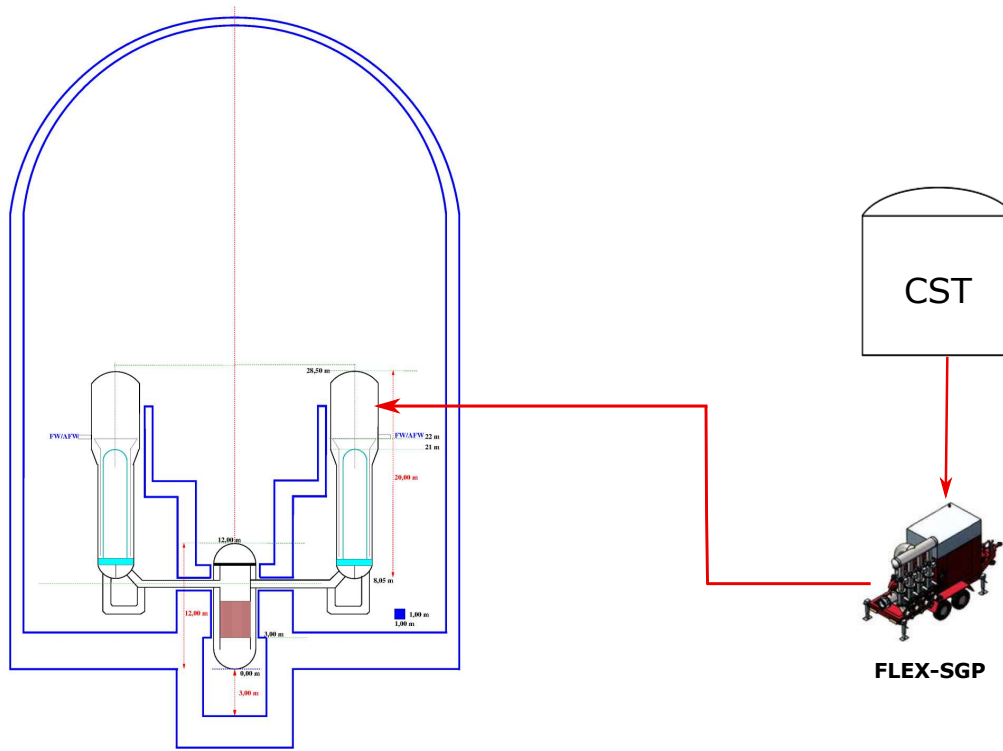


Figure 3.8: Diagram of FLEX-SGP injection to SG (FA-1)

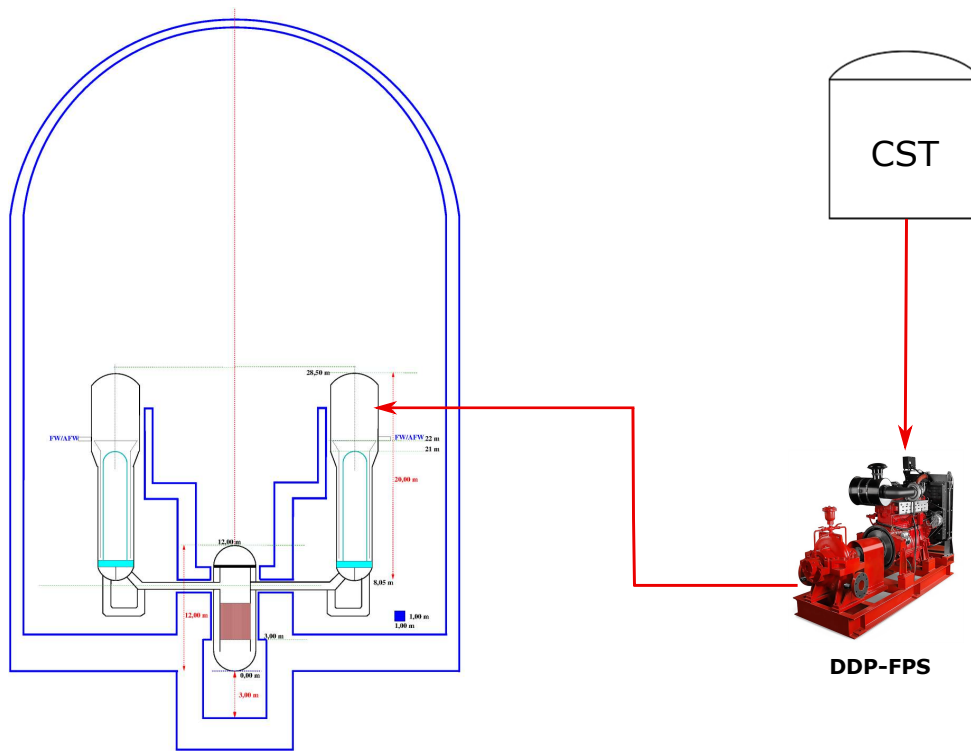


Figure 3.9: Diagram of DDP-FPS injection to SG (FA-2)

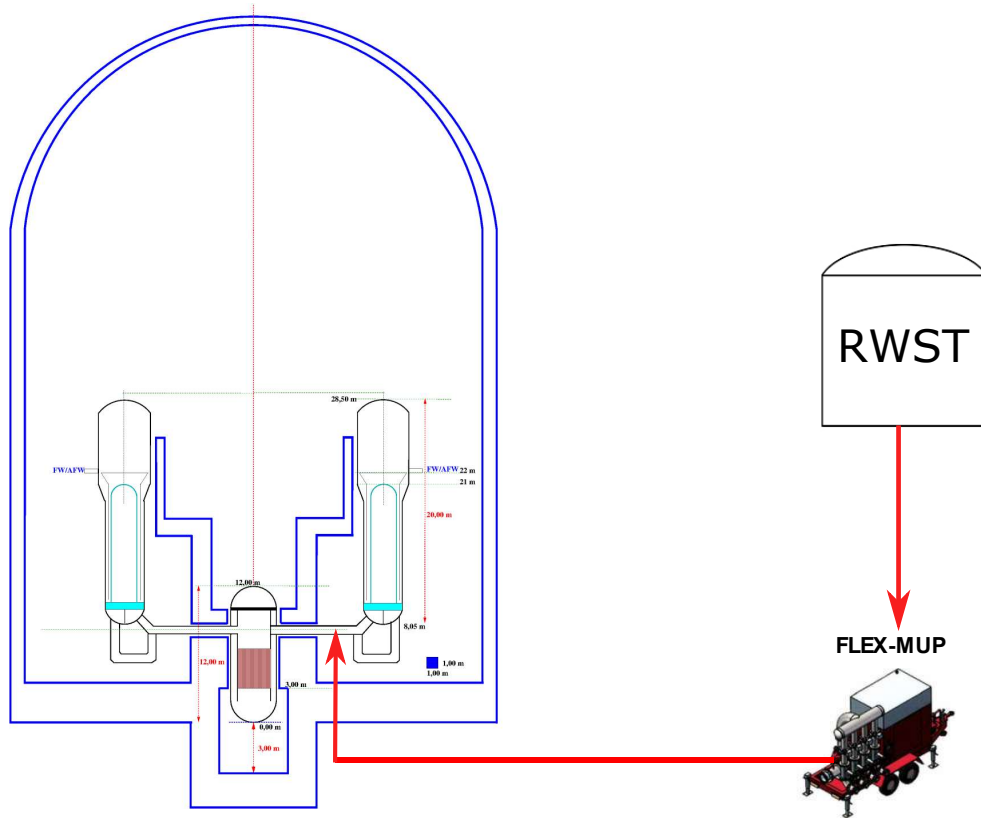


Figure 3.10: Diagram of FLEX-MUP injection to RCS (FA-4/5)

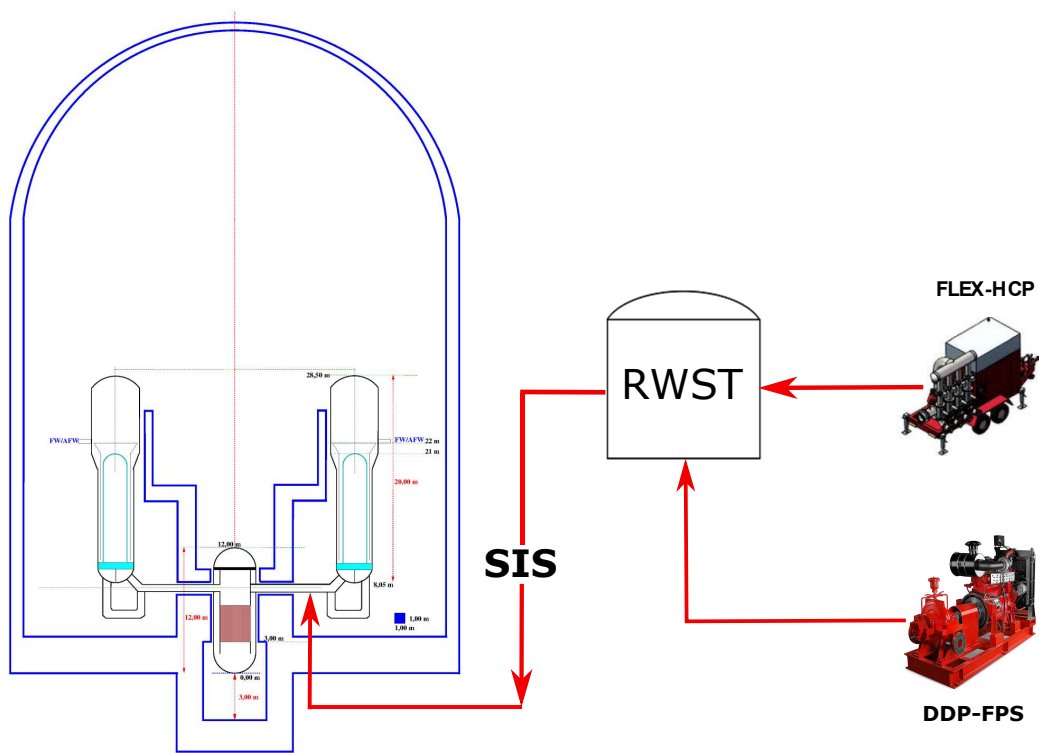


Figure 3.11: Diagram of FLEX-HCP/DDP-FPS injection to RWST (FA-6/7)

Table 3.3: FLEX Actions and relationship with FSGs and EDMGs (Spanish NPPs)

FLEX Action	FSG	EDMG	Equipment
FA-1: Alternative LP feedwater to the SGs by FLEX-SGP	FSG-3	EDMG-1.4 EDMG-5.2	FLEX-SGP
FA-2: Alternative LP feedwater to the SGs by DDP-FPS	FSG-3	EDMG-1.5	DDP-FPS
FA-3: Manual operation of TDP-AFW after DC loss	FSG-7	EDMG-1.1	TDP-AFW
FA-4: Long term RCS inventory control by FLEX-MUP	FSG-1	EDMG-2.2 EDMG-5.3	FLEX-MUP
FA-5: Alternative RCS boration.	FSG-8		FLEX-MUP
FA-6&7: Alternative RWST inventory control w/ diesel-driven pumps		EDMG-2.3	FLEX-HCP DDP-FPS
FA-8: Supply of AC power by FLEX-DG	FSG-4	EDMG-5.1	FLEX-DG
FA-9: DC load shedding to increase battery life	FSG-4		Batteries

It is important to note that each plant may have other FA depending on its own capabilities. These FA are applied according to the FSGs and EDMGs as shown in Table 3.3. The modeling of the FA is described in the following Subsections: Subsection 3.3.1 describes the new ETs including the FLEX headers. The modeling of FT and the corresponding new function events is presented in Subsection 3.3.2. The data sources used in the model are described in Subsection 3.3.3. Finally, Subsection 3.3.4 explains the HRA approach used.

3.3.1 Event Trees Including FLEX Headers

To include FLEX strategies in ET, it is necessary to identify which initiators and sequences can include these strategies. Then, the corresponding FLEX actions are introduced either within a pre-existing system, which implies modifying the header, or creating a new header if the FLEX action is not part of any previously defined system. In this process, thermal-hydraulic simulations are of great importance, as they give credibility to the FLEX actions in the model, especially in the quantification of the available times, given that the performance of many FLEX actions are not immediate and require certain time to be implemented. The addition of the above-mentioned FLEX Actions, Table 3.3, configures the new ETs of the model; this include the following new (or modified) headers with their corresponding function events, boundary conditions and success criteria. The modified headers (M), new ones (N) and their related FLEX actions are, see also Table 3.4:

- **AF-FLEX:** Residual heat removal through the secondary with diesel pumps FLEX-SGP or DDP-FPS or TDP-AFW (FA-1, FA-2).
- **FLEX-MUP:** Coolant injection into the RCS with the FLEX-MUP diesel pump (FA-4).
- **FLEX-MUP-RWST:** Alternate RCS boration with FLEX-MUP from RWST (FA-5).
- **LB-FLEX:** Power supply to loads (DC-B/C buses) from DG-SBO or FLEX-DG (FA-8).

- **LBS:** DC load shedding of non-essential loads from batteries (FA-9).
- **RWST-FLEX:** RWST inventory control with FLEX-HCP or DDP-FPS (FA-6, FA-7).
- **TDP-AFWS-MANUAL:** Manual operation of the TDP-AFW (FA-3).

Although not considered a FLEX strategy as such, the recovery of at least one emergency diesel generator (EDG) has also been modeled together with the recovery of the external AC power, even if the probability of recovery in the short term is low and depends on the EDG failure type. The modify AC recovery header is called R-EX. Figure 3.12 shows the new ET configuration for LOOP-SBO. In this new ET a new sequence can be observed: the injection of coolant into the RCS by means of the FLEX-MUP pump would produce a F&B type situation that would allow compensating the RCS inventory loss after the failure of the passive thermal seals. This pump may or may not be effective depending on the mass-flow rate it can provide. Figures 3.12, 3.13, 3.14, 3.15 and 3.16 show the new ETs for LOOP-SBO, GT, SBLOCA, SGTR and MSLB initiators respectively. The ETs for LC, LDC-A and LDC-B are similar to GT ET, the MBLOCA ET is similar to SBLOCA one. For LBLOCA and LNSW no FLEX actions have been considered.

Table 3.4: FLEX headers included in each ET (M: Modified; N: New)

ET	FLEX Headers							
	AF-FLEX	FLEX -MUP	FLEX-MUP -RWST	LB-FLEX	LBS	R-EX	RWST -FLEX	TDP-AFWS -MANUAL
ATWS	-	-	N	-	-	-	-	-
GT	M	-	-	-	-	-	N	-
LBLOCA	-	-	-	-	-	-	-	-
LC	M	-	-	-	-	-	N	-
LCWA	M	-	-	-	-	-	-	-
LDC-A	M	-	-	-	-	-	N	-
LDC-B	M	-	-	-	-	-	N	-
LNSW	-	-	-	-	-	-	-	-
LOOP	M	N	-	M	N	M	-	N
MBLOCA	-	N	-	-	-	-	N	-
MSLB-DS	M	-	-	-	-	-	N	-
MSLB-US	M	-	-	-	-	-	N	-
SBLOCA	M	N	-	-	-	-	N	-
SGTR	-	-	-	-	-	-	N	-

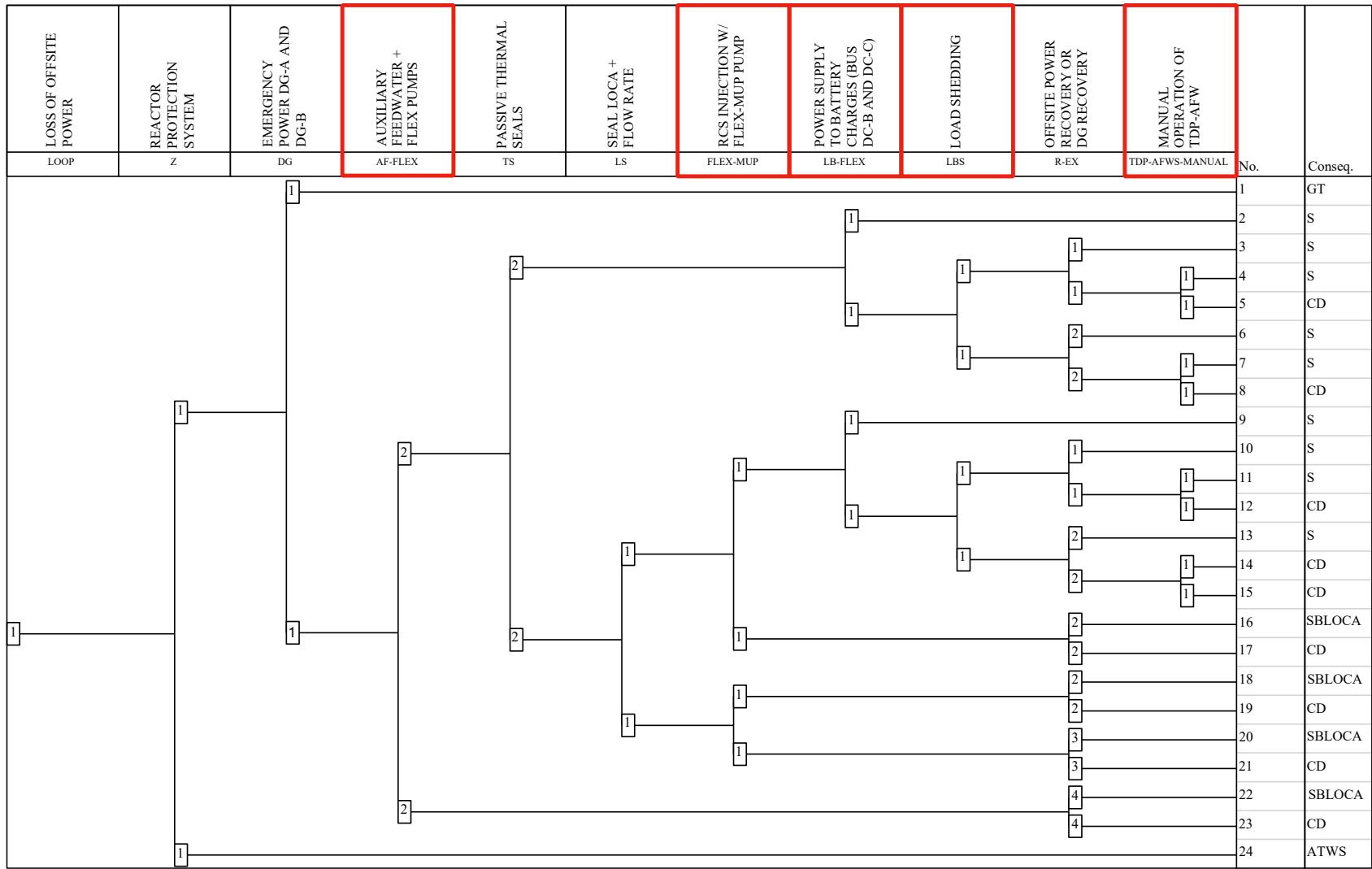


Figure 3.12: LOOP-SBO ET with FLEX strategies

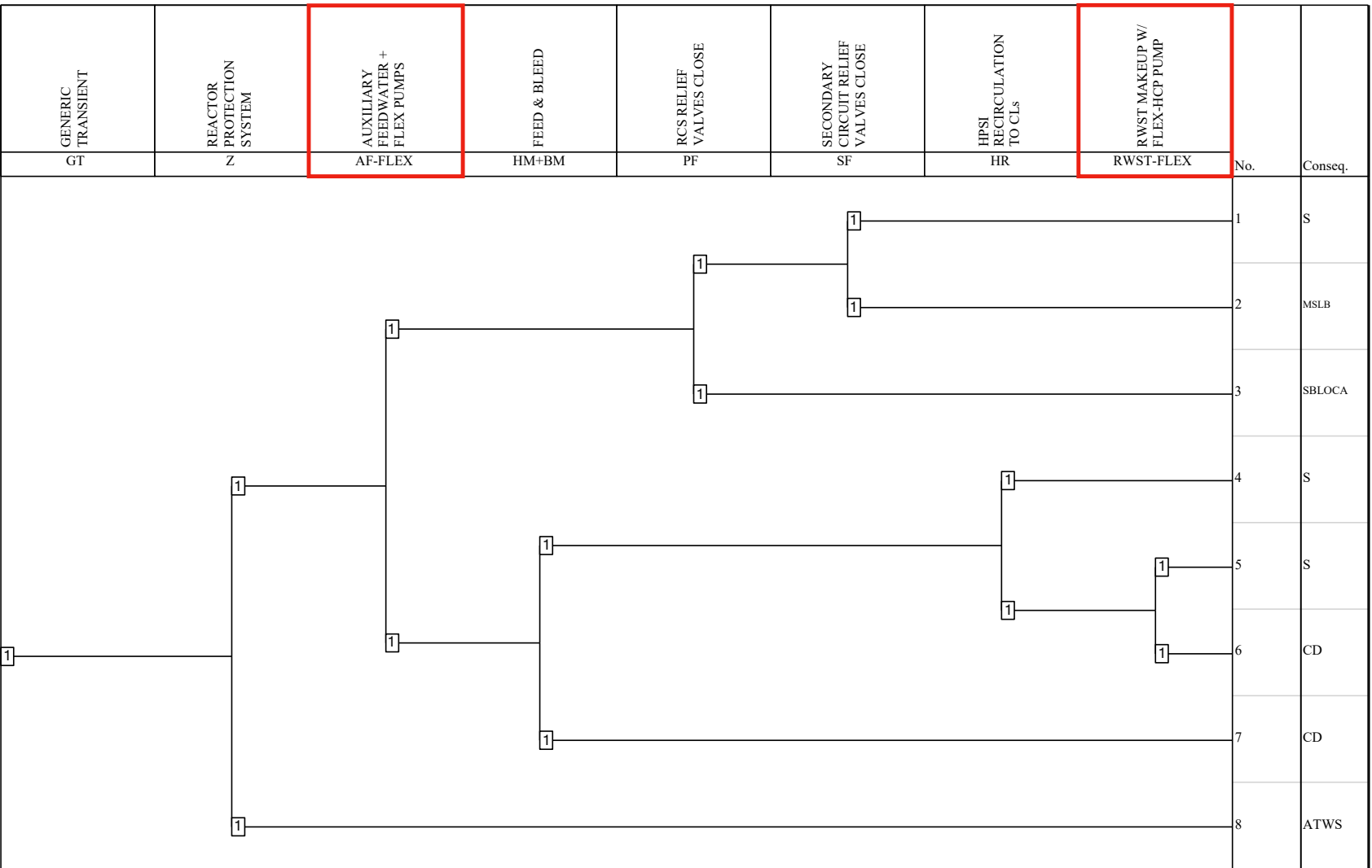


Figure 3.13: GT ET with FLEX strategies

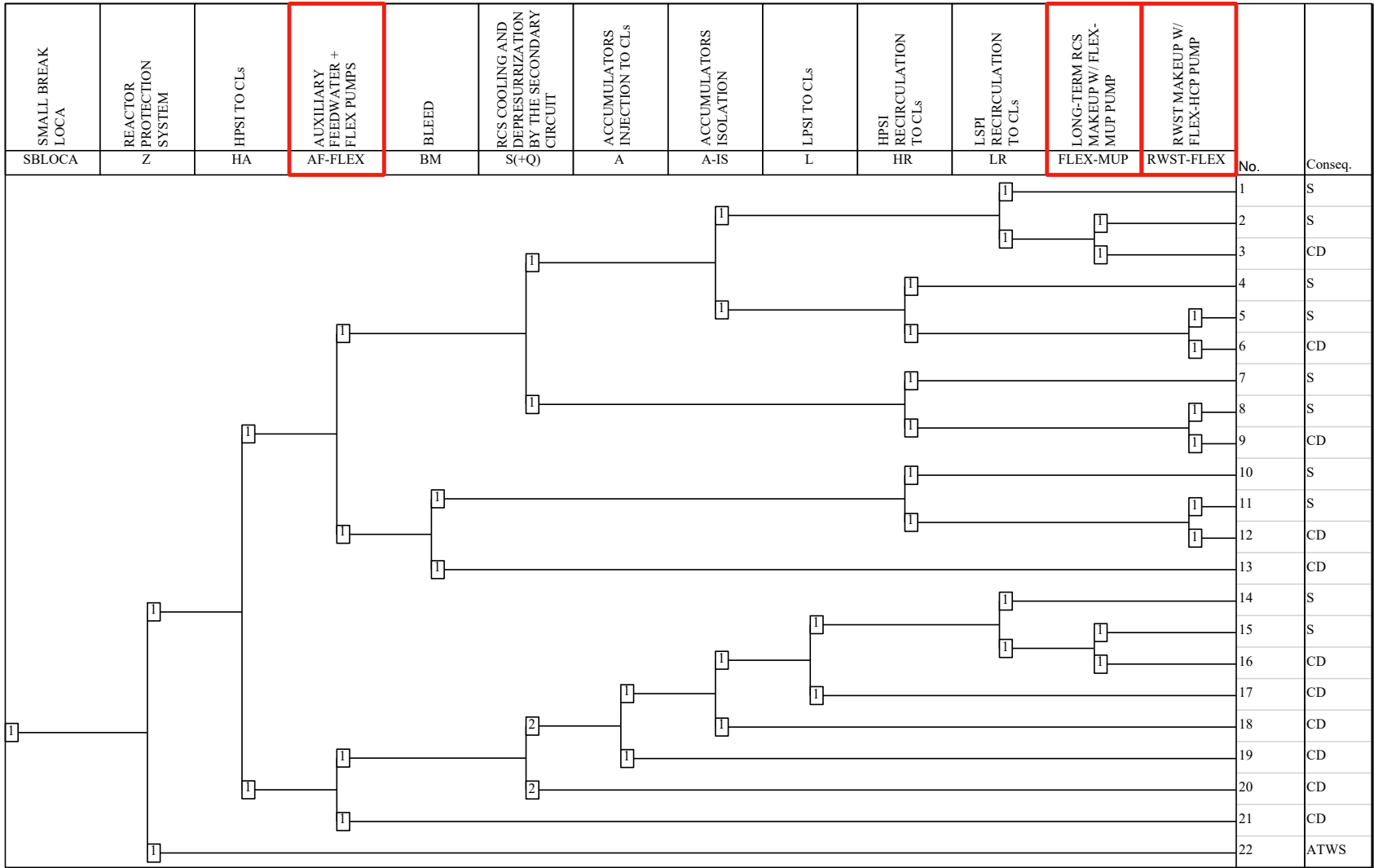


Figure 3.14: SBLOCA ET with FLEX strategies

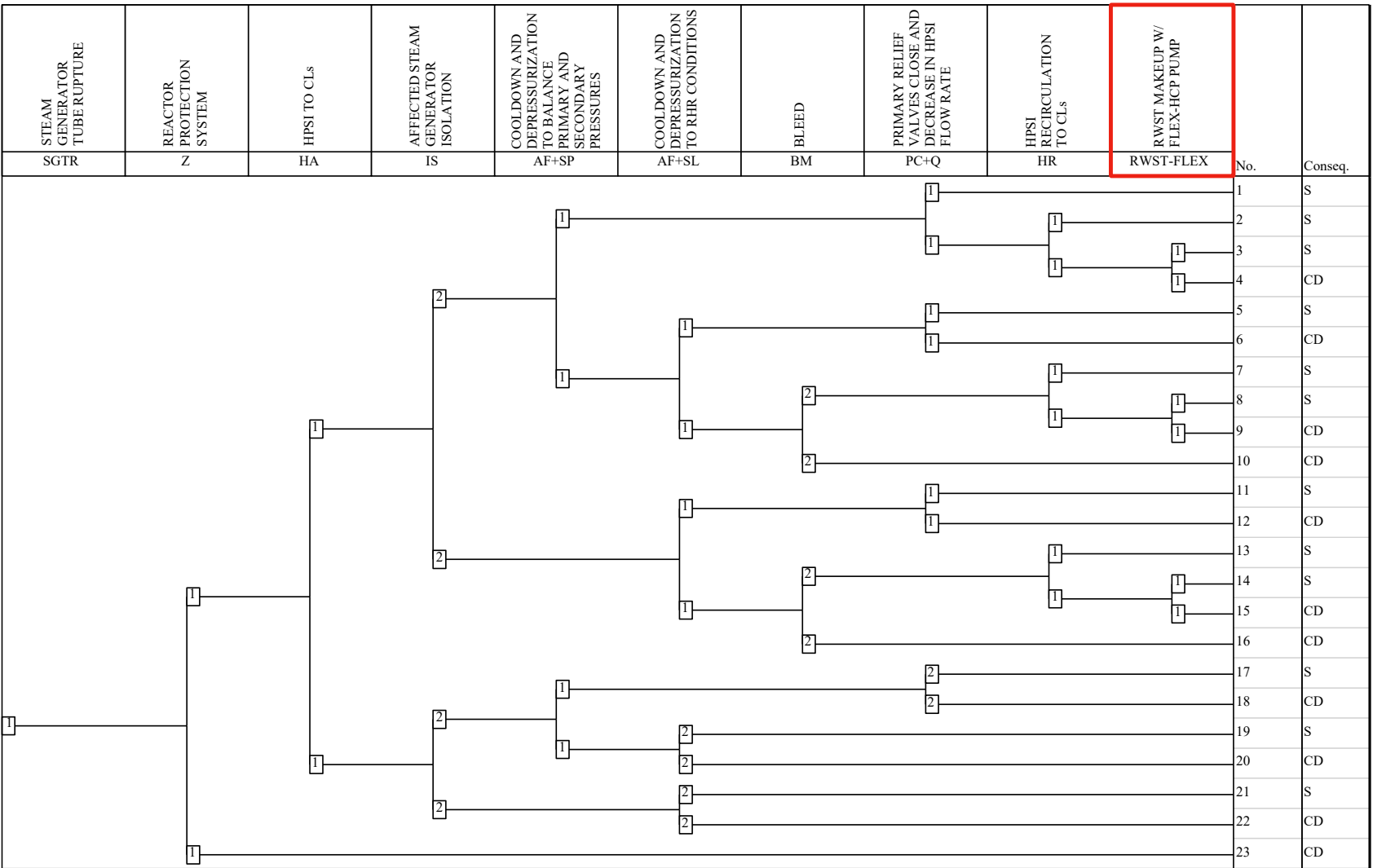


Figure 3.15: SGTR ET with FLEX strategies

RUPTURE OF ONE STEAM LINE UPSTREAM MSIV (in SG1)	REACTOR PROTECTION SYSTEM	INDUCED TUBE RUPTURE	STEAM GENERATOR ISOLATION	AUXILIARY FEEDWATER + FLEX PUMPS	AUXILIARY FEEDWATER (ATMOSPHERIC) + FLEX PUMPS	HPSI TO CLs	ALL CONTROL RODS INSERTED	BLEED	PZR-PORV AUTOMATIC OPENING	CLOSING OF PZR RELIEF PATHS AND SI DECREASE	HPSI RECIRCULATION TO CLs	RWST MAKEUP W/ FLEX-HCP PUMP	No.	Conseq.
MSLB-US	Z	TR	IS	AF-FLEX	AA-FLEX	HA	CR	BM	BA	Q+PF	HR	RWST-FLEX	1	S
													2	S
													3	S
													4	CD
													5	CD
													6	S
													7	CD
													8	S
													9	S
													10	CD
													11	CD
													12	CD
													13	S
													14	S
													15	S
													16	CD
													17	CD
													18	CD
													19	S
													20	S
													21	CD
													22	CD
													23	CD
													24	SGTR
													25	CD

Figure 3.16: MSLB ET with FLEX strategies

3.3.2 Fault Trees Including FLEX Equipment

FT modeling includes failures associated with portable equipment (FTS, FTR, and unavailability due to maintenance or testing) and HFE. The new AFW top event, including the modeling of FLEX strategies, is shown in Figure 3.17. A FT for a generic FLEX equipment (for example, FLEX-MUP, FLEX-DG and FLEX-HCP) is shown Figure 3.18. Also included are the new top events (FT) for each header associated with a FLEX strategy: Heat removal via the SG with FLEX; Emergency boration with FLEX; Supply of battery charges with FLEX-DG; RWST makeup with FLEX-HCP pump; Load shedding of non-essential loads from the batteries; Manual operation of the TDP-AFW; and RCS injection with FLEX-MUP pump.

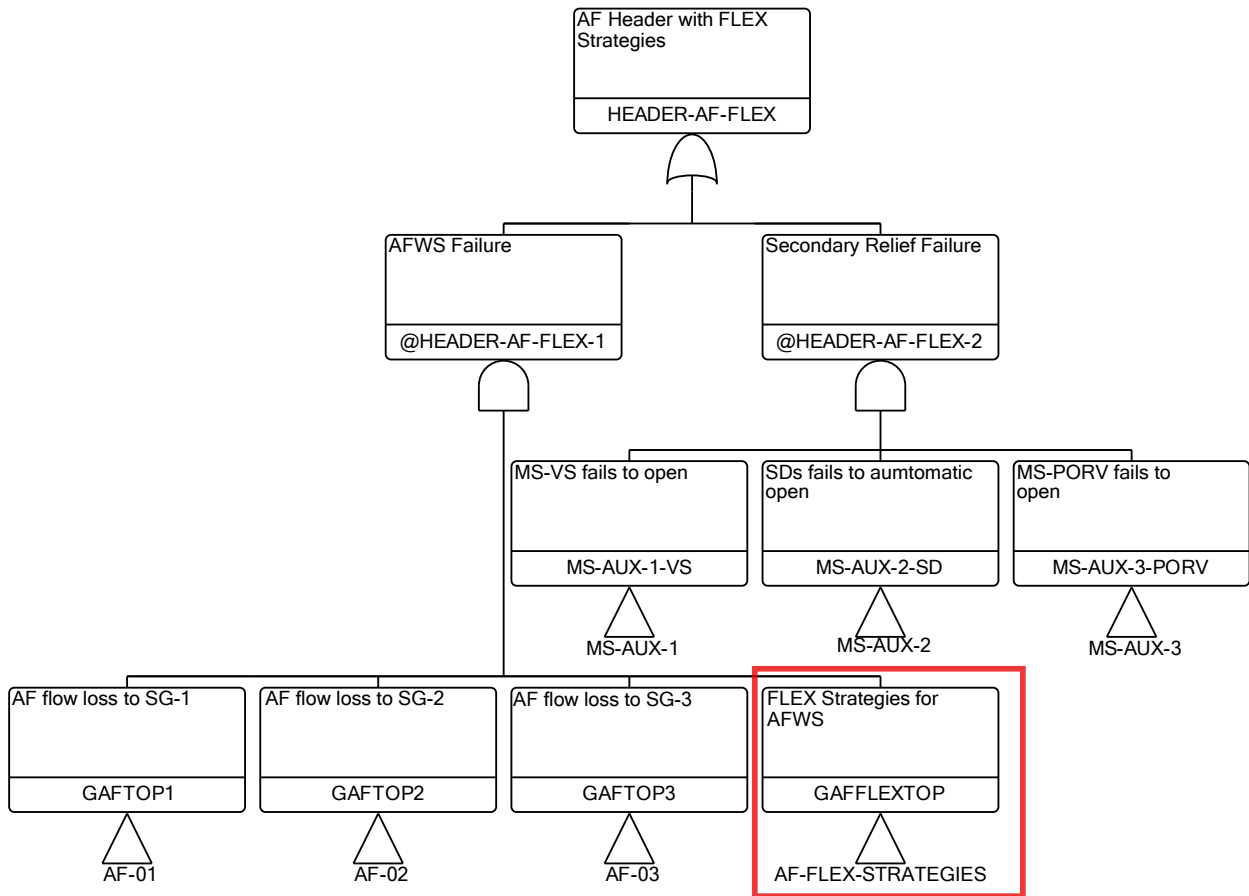


Figure 3.17: Top event FT corresponding to AF Header

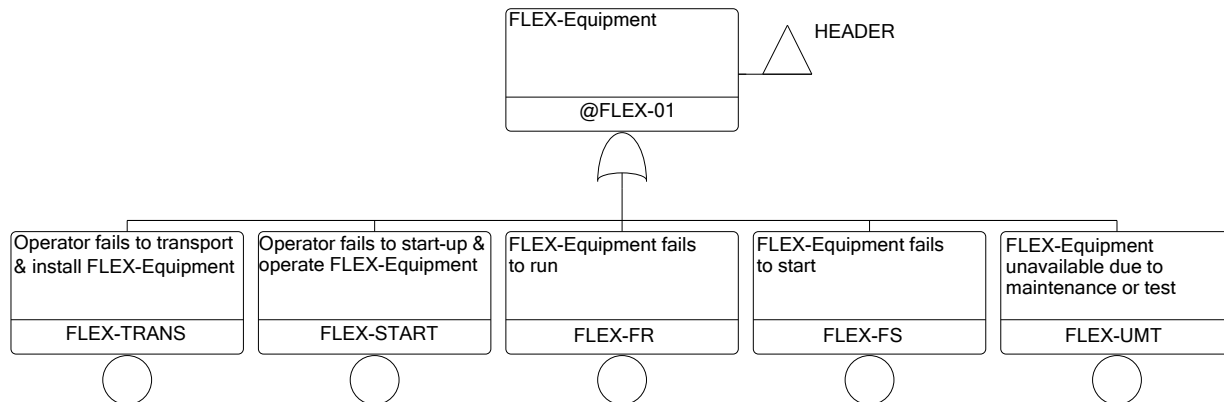


Figure 3.18: Example of the FLEX equipment FT modeling

3.3.3 Data for FLEX Equipment

Because the operating experience with FLEX equipment experience is practically non-existent (there are only equipment tests, but not in real conditions, and some integrated exercises), there was not a large collection of data on the failure rates of this equipment. Initially, guidance NEI 16-06 [NEI, 2016b] proposes that each site that includes portable equipment in its PSA model use engineering judgment regarding failure rates. Failure rates for permanently installed equipment were also used, and a sensitivity study was performed to increase these failure rates by a factor of 10; the NRC agreed that the use of engineering judgment is acceptable for estimating parameter values under certain conditions, but also noted that failure rates for permanently installed equipment cannot be used for portable equipment, even when sensitivity analysis are performed. The NRC emphasizes the need for up-to-date data for risk and safety assessment in SPAR models and suggests a possible path forward based on the Electric Power Research Institute (EPRI) data collection [Lane, 2017].

In recent years, initiatives have been developed that have made a first estimation of the failure parameters of FLEX components. In 2020, Idaho National Laboratory develops Evaluation of Weakly Informed Priors for FLEX Data [Gentillon et al., 2020], this report is based on FLEX Equipment Data Collection and Analysis, PWROG-18043-P Revision 0 to obtain industry-wide failure probability and rate distributions for portable FLEX equipment. In 2021, the PWROG issued Revision 1 of PWROG-18043 with proprietary information, which provides FLEX equipment reliability parameters for use in PSA. In 2022, The PWROG released a non-proprietary version of the report PWROG-18043, with generic failure probabilities for portable FLEX equipment, PWROG-18042-NP [Degonish, 2022]; the data included in this report are acceptable to the NRC to support risk-informed applications [NRC, 2022b] and is expected to be updated periodically to incorporate new operating experience data. This eliminates certain concerns that were associated with the: use of expert judgment; crediting spare equipment in lieu of equipment failure rates; use of permanently installed equipment failure rates; use of old data [NRC, 2022b]. The procedure to obtain the data was summarized by Linthicum et al. [Linthicum and Powell, 2022], Table 3.5. Uncertainties still remain

associated with maintenance or testing performed on FLEX equipment and CCF Parameters. The value of the parameters for unavailability due to maintenance or testing is highly variable and dependent on each NPP, and is in the range [1.00E-2, 1.00E-1]. A conservative value of 1.00E-1 has been implemented in the model.

Table 3.5: FLEX equipment failure probabilities (also used in SPAR-CSN model)

Equipment	Failure Mode		Ref.
	Fail to Run 24 hours	Fail to Start	
Portable diesel generator	2.47E-01	4.35E-02	[Degonish, 2022]
Portable combustion turbine	4.46E-01	3.30E-02	
Portable diesel-driven pump	3.72E-01	3.38E-02	
Portable motor-driven positive displacement pump	3.74E-01	7.35E-03	
Portable diesel/motor-driven air compressor	6.76E-01	2.46E-02	
Fire protection system diesel-driven pump	4.45E-02	2.17E-03	[Entergy, 2021]

In Spain, portable equipment is tested every three months and these tests are not included in the Technical Specifications. In addition, the values obtained in these tests are not available.

3.3.4 Human Reliability Analysis Applied to FLEX Strategies

Usual HRA analysis techniques have limitations for application to FLEX strategies. Aspects such as transportation or connection of portable equipment are not addressed in existing HRA methods. In this case, engineering judgment is generally used to find the unknown HEPs value based on generic data from other industries. However, with this method, the uncertainty of its results from subjective evaluation can be very high. The main proposal from NEI, NRC and EPRI are:

- NEI suggested a simplified approach to give credit for portable equipment, developed simple decision trees to estimate the HEPs of FLEX strategies in which the headers are four factors: time margin; command and control; environmental factors; and equipment availability. The final HEP can be calculated by multiplying the base HEP and the value of the four factors [NEI, 2015]. The base HEP is assigned as 1.00E-01 [Sandia, 2005]. Each factor is divided into two or three states to which multiplier values are assigned [NEI, 2015]. Then it was suggested to use Cause-Based Decision Tree Method (CBDTM) to analyze the diagnostic part and THERP for the action component [NEI, 2016b].
- The NRC proposed expert judgment to support the FLEX HRA [NRC, 2020a], it assumes two types of accident scenarios: not designed for FLEX and scenarios designed for FLEX, each is decomposed into several human activities, for which they estimate the corresponding HEP. In addition, they are working on developing a new HRA method for FLEX [NRC, 2020b]. In both cases they have consulted a panel of HRA analysts from the NRC and industry. Later, in 2022, The NRC presented IDHEAS-ECA [NRC,

2022c]. This methodology can be applied to external events, low power and shutdown (LPSD) events, and events that require the use of FLEX strategies. It makes use of five macrocognitive functions: detection, understanding, decisionmaking, action execution, and interteam coordination. To calculate the HEPs, IDHEAS-ECA defines 20 performance-influencing factors in four context categories.

- EPRI conducted a review of procedures related to FLEX-like actions at several plants in the U.S. [EPRI, 2014]. It discovered gaps in multiple tasks and approaches: Equipment transportation and installation; Environmental effects on execution (timing, execution, equipment availability and staffing); Organizational prioritization (Coordination, resource management); Procedures (detecting priorities); Synchronization (prioritization and availability of work teams); Execution in many steps (complex executions and decision making). As a result, it produced a reference document [EPRI, 2018] using CB-DTM, THERP and Integrated Human Event Analysis System (IDHEAS) methodologies. It is a guide based on examples and possible variations for the different scenarios: ELAP declaration; portable pump transportation, installation and actuation; and portable generator refueling. However, the authors highlight that questions remain about the suitability of HRA methods for addressing FLEX events [Ulrich et al., 2020]. EPRI also modified their HRA calculator to facilitate modeling HRA for FLEX actions [NRC, 2022b].

In addition, other groups have addressed the problem of HRA for FLEX strategies; Kim et al. [Kim et al., 2018, Kim, 2019] studied HRA for MACST. Hasan et al. [Hasan et al., 2018] calculated HEPs for the implementation of MACST during an extended SBO scenario in APR1400, and Duffey [Duffey, 2019] analyzed the dynamic human response and coping actions during a BDBEE with a time-dependent HRA approach. In addition to single-unit scenarios, Park et al. [Park et al., 2019] focused on human error issues in multi-unit HRAs. Park et al. [Park et al., 2021b] modeled FLEX human actions for an ELAP scenario using the EMERALD software and introduced two different HRA approaches, procedure-based modeling or PSA/HRA-based modeling. Later, Park et al. [Park et al., 2022] introduced PRIMERA-HRA method, which combines the two EMERALD modeling approaches. Finally, several authors have proposed the application of modified versions of SPAR-H approach to FLEX conditions and portable equipment [Park et al., 2021a, Park et al., 2019]. The HEPs obtained from these references are shown in Tables 3.9 and 3.10.

The SPAR-H methodology has been adapted by UPM to the conditions of FLEX strategies and then applied to the SPAR-CSN model incorporating FLEX strategies [Courtin et al., 2021]. This has been carried out considering previous work on adapting this approach for FLEX strategies [Park et al., 2021a, Park et al., 2019], along with other modifications related to other applications such as L2PSA and fire scenarios [Liu et al., 2016, Saint-Germain et al., 2016, Wang, 2013a, Wang, 2013b, Liu et al., 2018, Liu et al., 2019, Laumann and Rasmussen, 2016, Liu et al., 2020, Liu et al., 2021]. The obtained HEPs with the modified SPAR-H approach are shown in Table 3.11.

3.3.4.1 SPAR-H

Standardized Plant Analysis Risk-Human Reliability Analysis (SPAR-H) is the methodology used by the NRC to analyze human reliability in its SPAR models [Gertman et al., 2005]. It is a simple methodology for estimating the probabilities of human error associated with operator/staff (HA) actions and decisions. The structure of the SPAR-H methodology (Table 3.6) is characterized by: decomposing probability into contributions (diagnosis and action); using performance shaping factors (PSFs) and assigning dependencies to estimate the HEPs; and employing worksheets designed to ensure analyst consistency. SPAR-H identifies eight PSFs capable of influencing HA: Available Time; Stress/Stressors; Complexity; Experience/Training; Procedures; Ergonomics/HMI; Fitness for Duty; Work Processes.

SPAR-H not only considers adverse effects, but also assesses the potential benefit of these factors; the influence of some PSFs may reduce nominal failure rates, such as adequate experience and training. SPAR-H takes dependency into account; the model considers the negative influence of a human error on subsequent errors and is reflected in the HEP calculation.

Once the PSF levels have been assigned, the final HEP is the product of the nominal HEP ($NHEP_d = 1.00E-02$; $NHEP_a = 1.00E-03$) and the PSF multipliers. When the diagnosis and action are combined into a single HFE, the two HEPs are calculated separately and then summed to produce the composite HEP. If multiple PSFs have a negative impact on a HEP, then there is a possibility that the final individual HEP will be greater than one. For this case an adjusted PSF is calculated and the formula is used:

$$HEP_j = \begin{cases} NHEP_j \times PSF_c^j & \text{if less than 3 PSFs} > 1 \\ \frac{NHEP_j \times PSF_c^j}{NHEP_j \times (PSF_c^j - 1) + 1} & \text{if 3 or more PSFs} > 1 \end{cases} \quad (3.1)$$

where, $j=d, a$.

$$PSF_c = \prod_{j=1}^8 PSF_j \quad (3.2)$$

$$\boxed{HEP = HEP_d + PSF_a} \quad (3.3)$$

Table 3.6: PSFs and multipliers of SPAR-H [Gertman et al., 2005]

PSFs	PSF Level Diagnosis	Multiplier	PSF Levels Action	Multiplier
	NHEP=0.01		NHEP=0.001	
Available Time	Inadequate time	$P(\text{failure}) = 1$	Inadequate time	$P(\text{failure}) = 1$
	Barely adequate time ($\approx 2/3 \times$ nominal)	10	Time available is \approx the time required	10
	Nominal time	1	Nominal time	1
	Extra time (between 1 and $2 \times$ nominal and $> 30\text{Min}$)	0.1	Time available $\geq 5 \times$ the time required	0.1
	Expansive time ($> 2 \times$ nominal and $> 30\text{Min}$)	0.01	Time available is $\geq 50 \times$ the time required	0.01
	Insufficient information	1	Insufficient information	1
Stress/Stressors	Extreme	5	Extreme	5
	High	2	High	2
	Nominal	1	Nominal	1
	Insufficient information	1	Insufficient information	1
Complexity	Highly complex	5	Highly complex	5
	Moderately complex	2	Moderately complex	2
	Nominal	1	Nominal	1
	Obvious Diagnosis	0.1		-
	Insufficient information	1	Insufficient information	1
Experience/Training	Low	10	Low	3
	Nominal	1	Nominal	1
	High	0.5	High	0.5
	Insufficient information	1	Insufficient information	1
Procedures	Not available	50	Not available	50
	Incomplete	20	Incomplete	20
	Available but poor	5	Available, but poor	5
	Nominal	1	Nominal	1
	Diagnostic/symptom oriented	0.5		-
	Insufficient information	1	Insufficient information	1
Ergonomics/HMI	Missing/Misleading	50	Missing/Misleading	50
	Poor	10	Poor	10
	Nominal	1	Nominal	1
	Good	0.5	Good	0.5
	Insufficient information	1	Insufficient information	1
Fitness for Duty	Unfit	$P(\text{failure}) = 1$	Unfit	$P(\text{failure}) = 1$
	Degraded fitness	5	Degraded fitness	5
	Nominal	1	Nominal	1
	Insufficient information	1	Insufficient information	1
Work Processes	Poor	2	Poor	5
	Nominal	1	Nominal	1
	Good	0.8	Good	0.5
	Insufficient information	1	Insufficient information	1

Based on the abovementioned, some changes in the SPAR-H PSFs and nominal failure probabilities (NHEP) are proposed to adapt them to the needs of the FLEX strategies. Guidance on how the NHEPs and SPAR-H PSFs could be affected in such situations is presented below:

- Nominal HEPs: Due to the existing situation, the nominal stress would not be the same as in a Non-FLEX scenario. Therefore, are modified: $NHEP_d: 0.01 \rightarrow 0.05$; and $NHEP_a: 0.001 \rightarrow 0.005$
- PSF 1 Available time: The time available may be negatively affected by manual actuation outside of containment. In addition, mobilization of personnel between the storage location and the facility, and potential hazards between these locations must be considered [Saint-Germain et al., 2016]. Time constraints can affect the accuracy of plant condition diagnosis and decision making [Kang and Seong, 2020]. The “expansive time” level is eliminated and the multiplier of the “Barely adequate time” and “Insufficient information” levels is increased.
- PSF 2 Stress: High/extreme stress should be considered for external events [Saint-Germain et al., 2016]. “Threat stress” is generated when operators face the possibility of the catastrophe [Liu et al., 2021]; it could be included when FLEX strategies are needed. The shorter time available could also increase operators’ stress [Park et al., 2020]. In these conditions, the multiplier of the “Insufficient information” level is increased. In addition, information overload, (for example, multiple simultaneous alarms) can increase the stress level. [Ramos et al., 2018].
- PSF 3 Complexity: The more difficult and/or ambiguous the task, the greater the possibility of human error. Knowledge-based diagnostics and decision making often present more complexity and are associated with higher probability of error, so a FLEX strategy scenario should increase complexity. We specify “complexity” as “task complexity” [Rasmussen et al., 2015]. The multiplier for the “Insufficient information” level is increased.
- PSF 4 Experience/Training: It is very complex to have a high level of experience/training for human actions in FLEX. They are based on walkthroughs and simulations that do not have actual plant manipulations or equipment installation [Saint-Germain et al., 2016]. “Good” level is eliminated. Given the expected level of preparedness, it is very difficult for operators to respond with high probability correctly in front of a FLEX scenario [Laumann and Rasmussen, 2016]; the multiplier of the “Low” and “Insufficient information” levels is increased.
- PSF 5 Procedures: The assessment of FLEX procedures should be based on expected working conditions, not only on a validation of administrative requirements [Saint-Germain et al., 2016]. The minimum level is nominal. Overlap with other PSFs, 3 and 8, should be avoided in case of having incomplete or inadequate procedures.
- PSF 6 Ergonomics/HMI: Uncertain conditions in a FLEX scenario could negatively impact this PSF. There could be unforeseen physical challenges and inaccessible or hard-to-reach sites [Laumann and Rasmussen, 2016, Saint-Germain et al., 2016]; even the change to Alternate Center for Emergency Management could impact HMI [Bisio et al., 2020]. Therefore, a scenario is included in which the necessary tools are not available or the area is inaccessible and it is proposed that in this case $P(\text{failure})=1$. The multiplier for the “Insufficient information” level is also increased.

- PSF 7 Fitness for duty: Considers the fatigue and exhaustion to which the operator/team (Liu et al., 2016) [Liu et al., 2016] is subjected in a FLEX scenario (prolonged tasks, manual control of a portable pump or constant monitoring of a piece of equipment). Stress can manifest itself in the form of physical fatigue, terrain orography and distance traveled by personnel, potential multi-unit accidents [Saint-Germain et al., 2016] and physically demanding manual operations [Laumann and Rasmussen, 2016, Saint-Germain et al., 2016] should be considered.
- PSF 8 Work processes: Difficult to assess [Laumann and Rasmussen, 2016, Liu et al., 2020]. It is proposed to split the PSF “Work processes” into two new PSFs, [Laumann and Rasmussen, 2016]:
 - PSF 8A Teamwork: SPAR-H has no PSF that directly measures team cooperation and collaboration [Kang and Seong, 2020]. Therefore, it includes team communication, coordination and cooperation [Laumann and Rasmussen, 2016, Liu et al., 2020].
 - PSF 8B Adequacy of organization: In FLEX scenario the readiness of the plant team and their cultural perception of risk is critical [Kang and Seong, 2020]. It has two factors: Attitudes, safety and work behavior, attentiveness, care and other behaviors that contribute to a safe work environment; and Management Support, degree to which the operator has explicit support in performing tasks, [Laumann and Rasmussen, 2016].
- PSF 9 Working Conditions: Quantifies the specific working conditions during a FLEX scenario that affect performance, including environmental elements such as noise, heat and flooding, [Liu et al., 2020]. The “Very poor” level is added.
- PSF 10 Intensity of the external event: Quantifies the intensity of the earthquake and the frequency of possible aftershocks. It also considers the availability and integrity of off-site emergency personnel [Suh and Kim, 2019]. This PSF do not need to be included if there is no BDBEE.

The final proposal of SPAR-H adapted for FLEX scenarios is presented in Tables 3.7 and 3.8. Tables 3.9 and 3.10 show a comparison of HEPs found in the literature for actions related with FLEX-DG and FLEX pumps. The HEPs used in SPAR-CSN Model (modified SPAR-H) are shown in Table 3.11.

Table 3.7: SPAR-H adapted for FLEX scenarios [Courtin et al., 2021] (1/2)

PSF	ID	PSF Levels Action	SPAR-H L1PSA	FLEX
			Multiplier	Multiplier
Available Time	PSF 1	Inadequate time	P(failure) = 1	P(failure) = 1
		Barely adequate time	10	20
		Nominal time	1	1
		Extra time	0.1	0.1
		Expansive time	0.01	-
		Insufficient information	1	2
Stress/ Stressors	PSF 2	Extreme	5	25
		High	2	5
		Nominal	1	1
		Insufficient Information	1	5
Complexity	PSF 3	Highly complex	5	5
		Moderately complex	2	2
		Nominal	1	1
		Obvious Diagnosis	0.1	-
		Insufficient Information	1	2
Experience/ Training	PSF 4	Low	3	P(failure) = 1
		Nominal	1	1
		High	0.5	-
		Insufficient Information	1	2
Procedures	PSF 5	Not available	50	50
		Incomplete	20	20
		Available, but poor	5	5
		Nominal	1	1
		Diagnostic/symptom oriented	0.5	-
		Insufficient Information	1	1
Ergonomics/ HMI	PSF 6	Missing/Misleading	50	P(failure) = 1
		Poor	10	15
		Nominal	1	1
		Good	0.5	-
		Insufficient Information	1	15
Fitness for Duty	PSF 7	Unfit	P(failure) = 1	P(failure) = 1
		Degraded Fitness	5	5
		Nominal	1	1
		Insufficient Information	1	1
Work Processes	PSF 8	Poor	5	-
		Nominal	1	-
		Good	0.5	-
		Insufficient Information	1	-

Table 3.8: SPAR-H adapted for FLEX scenarios [Courtin et al., 2021] (2/2)

PSF	ID	PSF Levels Action	SPAR-H L1PSA	FLEX
			Multiplier	Multiplier
Teamwork	PSF 8A	Poor	-	5
		Nominal	-	1
		Good	-	0.8
		Insufficient Information	-	1
Adequacy of organization	PSF 8B	Poor	-	5
		Nominal	-	1
		Good	-	-
		Insufficient Information	-	2
Working Conditions	PSF 9	Very Poor	-	15
		Poor	-	10
		Moderate	-	5
		Nominal	-	1
Intensity	PSF 10	Extreme	-	20
		Very High	-	10
		High	-	5
		Moderate	-	2
		Nominal	-	1

Table 3.9: Comparison of HEPs for FLEX-DG

Author	Transport + Connection	Start-up + Operation	TOTAL HEP
[NEI, 2016b]	[1.0E-01, 5.0E-03]	-	-
[Son and Lim, 2017]	-	-	1.73E-02
[Shahinoor and Lim, 2017]	-	-	1.99E-02
[Hasan et al., 2018]	-	-	8.80E-03
[Kim et al., 2018]	1.37E-03	1.01E-03	6.23E-03
[Park et al., 2019]	2.00E-03	1.20E-02	1.40E-02
[Kim, 2019]	-	4.00E-02	-
[Ma et al., 2019]	3.00E-03	1.20E-02	1.50E-02
[Ryu et al., 2019]	-	-	[4.9E-01, 8.1E-01]
[NRC, 2022a]	3.00E-01	1.20E-01	4.20E-01
[NRC, 2022c]	1.00E-03	[1.0E-03, 1.0E-02]	1.00E-02
[Park et al., 2021b]	5.00E-02	-	-
[Park et al., 2021a]	3.01E-03	4.00E-04	3.42E-03
[Park et al., 2022]	7.40E-02	-	-

Table 3.10: Comparison of HEPs for FLEX pumps

Author	Transport + Connection	Start-up + Operation	TOTAL HEP
[Son and Lim, 2017]	-	-	6.16E-02
[Hasan et al., 2018]	-	-	8.90E-03
[Tabadar et al., 2019b]	1.39E-02	-	-
[Park et al., 2019]	2.00E-03	1.20E-02	1.40E-02
[Kim, 2019]	-	4.00E-02	-
[NRC, 2022a]	2.50E-01	1.40E-01	3.90E-01
[Park et al., 2021b]	5.00E-02	-	-

Table 3.11: HEPs used in SPAR-CSN Model (modified SPAR-H)

Equipment	Transport + Connection	Start-up + Operation	TOTAL HEP
FLEX-DG	4.70E-02	6.00E-03	5.30E-02
FLEX-SGP	4.70E-02	4.54E-01	5.01E-01
FLEX-MUP	4.70E-02	1.35E-01	1.82E-01
FLEX-HCP	4.70E-02	1.35E-01	1.82E-01
DDP-FPS	Align to SGs	Start-up + Operation	
	1.00E-01	5.00E-02	1.5E-01

3.4 Impact of FLEX Strategies on the Risk Reduction

The total impact of the strategies on the full model is quantified through the variation of the CDF. The new CDF value is $1.37\text{E-}06$ 1/y and, therefore, the global results show a reduction in the risk of situations leading to CD. This reduction (ΔCDF : -89.4%) is close to an order of magnitude, which shows the importance that this type of strategies can have in sequences that are not related to the BDBEE. The variation in CDF, for each initiator, is shown in Table 3.12. The minimal cutsets (MCS) obtained, with and without FLEX strategies, are presented in Tables 3.13 and 3.14. Also, a comparison of CDF results, including FLEX strategies, found in the literature is presented in Table 3.15. It should be noted that the decrease of the CDF depends on the FA implemented and the reactor design, which explains the wide range of values for the variation of the CDF shown in Table 3.15.

Table 3.12: CDF reduction for IE with FLEX strategies

L1PSA ET	Initiator Freq. (1/y)	No FLEX strategies		FLEX strategies		ΔCDF (%)
		CDF (1/y)	Total CDF	CDF (1/y)	Total CDF	
GT	6.76E-01	7.39E-06	57.14%	8.40E-07	61.54%	-89%
LBLOCA	5.91E-06	1.15E-08	0.09%	1.15E-08	0.84%	0%
LC	4.82E-02	5.14E-07	3.97%	6.08E-08	4.45%	-88%
LCWA	1.80E-03	7.08E-10	0.01%	6.21E-10	0.05%	-12%
LDC-A	5.00E-04	1.79E-08	0.14%	2.42E-09	0.18%	-86%
LDC-B	5.00E-04	1.50E-08	0.12%	1.78E-09	0.13%	-88%
LNSW	2.00E-04	2.19E-09	0.02%	2.32E-10	0.02%	-89%
LOOP	3.11E-02	4.45E-07	3.44%	3.57E-08	2.62%	-92%
MBLOCA	1.50E-04	1.21E-07	0.94%	2.90E-08	2.12%	-76%
MSLB-DS	6.32E-03	7.54E-07	5.83%	5.87E-08	4.30%	-92%
MSLB-US	3.01E-04	4.15E-08	0.32%	4.94E-09	0.36%	-88%
SBLOCA	4.01E-04	1.65E-07	1.28%	5.82E-08	4.26%	-65%
SGTR	1.66E-03	3.46E-06	26.74%	2.83E-07	20.73%	-92%
CDF Total		1.29E-05		1.37E-06		-89%

Table 3.13: SPAR-CSN MCS without FLEX strategies

No.	Frequency (1/y)	%	IE	Event 1	Event 2	Event 3
1	4.83E-06	37.4	GT	1OFAF-CNT-SG	1OFHHPR-F&B-D1	
2	3.32E-06	25.7	SGTR	1OFHH-RECIR-D6	1OFHH-REDUC-2	
3	1.69E-06	13.1	GT	1OFAF-CNT-SG	1OFHH-RECIR-D2	
4	6.32E-07	4.9	MSLB-DS	1OFHH-RECIR-D3	1OFHH-REDUC	
5	3.45E-07	2.7	LC	1OFAF-CNT-SG	1OFHHPR-F&B-D1	
6	2.22E-07	1.7	LOOP	1OFAF-CNT-SG	1OFHHPR-F&B-D1	
7	1.49E-07	1.2	GT	1MSPORVG3FC	1OFHH-RECIR-D3	1OFHH-REDUC
8	1.49E-07	1.2	GT	1MSPORVG1FC	1OFHH-RECIR-D3	1OFHH-REDUC
9	1.49E-07	1.2	GT	1MSPORVG2FC	1OFHH-RECIR-D3	1OFHH-REDUC
10	1.21E-07	0.9	LC	1OFAF-CNT-SG	1OFHH-RECIR-D2	
11	8.32E-08	0.6	SGTR	1OFHH-REDUC-2-D2	1OFPR-OPEN	
12	7.79E-08	0.6	LOOP	1OFAF-CNT-SG	1OFHH-RECIR-D2	
13	7.50E-08	0.6	MBLOCA	1OFHH-RECIR-D1	1OFMS-OP&CNT	
14	4.52E-08	0.4	MSLB-DS	1OFAF-CNT-SG	1OFPR-OPEN-D1	
15	4.01E-08	0.3	SBLOCA	1OFHH-RECIR-D3	1OFHH-REDUC	
TOTAL	1.29E-05					

Table 3.14: SPAR-CSN MCS with FLEX strategies

No.	Freq. (1/y)	%	IE	Event 1	Event 2	Event 3	Event 4
1 to 1	2.19E-07	16.1	GT	1OF-FLEX-SGP-START	1OFAF-CNT-SG	1OFDDP-ALIGN-FLXHOSE	1OFHHPR-F&B-D1
1 to 2	1.50E-07	11.0	GT	1FLEXSGPFR	1OFAF-CNT-SG	1OFDDP-ALIGN-FLXHOSE	1OFHHPR-F&B-D1
2 to 3	9.77E-08	7.2	GT	1OF-FLEX-SGP-START	1OFAF-CNT-SG	1OFHHPR-F&B-D1	1PCIEDPFR-23H
11 to 4	8.32E-08	6.1	SGTR	1OFHH-REDUC-2-D2	1OFPR-OPEN		
1 to 5	6.68E-08	4.9	GT	1FLEXSGPFR	1OFAF-CNT-SG	1OFHHPR-F&B-D1	1PCIEDPFR-23H
2 to 6	4.59E-08	3.4	SGTR	1FLEXHCPFR	1OFHH-RECIR-D6	1OFHH-REDUC-2	1PCIEDPFR-23H
1 to 7	3.60E-08	2.6	GT	1OF-FLEX-SGP-START	1OFAF-CNT-SG	1OFHHPR-F&B-D1	1PCIEDPUMT
1 to 8	2.46E-08	1.8	GT	1FLEXSGPFR	1OFAF-CNT-SG	1OFHHPR-F&B-D1	1PCIEDPUMT
27 to 9	2.41E-08	1.8	SGTR	1OFHH-REDUC-2-D1	1OFMS-ISOMSIV		
1 to 10	2.27E-08	1.7	GT	1OF-FLEX-SGP-TRANS	1OFAF-CNT-SG	1OFDDP-ALIGN-FLXHOSE	1OFHHPR-F&B-D1
2 to 11	2.063E-08	1.51	SGTR	1FLEXHCPFR	1OFDDP-ALIGN-FLXRWST	1OFHH-RECIR-D6	1OFHH-REDUC-2
2 to 12	2.00E-08	1.5	SGTR	1OF-FLEX-HCP-START	1OFHH-RECIR-D6	1OFHH-REDUC-2	1PCIEDPFR-23H
29 to 13	1.87E-08	1.4	GT	1RPCRODFTOP	1ST-MTC		
30 to 14	1.74E-08	1.3	MSLB-DS	1RPCRODFTOP			
2 to 15	1.69E-08	1.2	SGTR	1FLEXHCPFR	1OFHH-RECIR-D6	1OFHH-REDUC-2	1PCIEDPUMT
TOTAL	1.37E-06						

Table 3.15: Comparison of CDF reduction including FLEX strategies

Author	Δ CDF		Reactor type	FLEX strategies
	LOOP/SBO [MBLOCA]	Overall		
[Kim and Kim, 2021]	12%	9%	OPR1000	Portable DG to supply battery charger LP portable pump to supply feedwater to SG
[Kim and Kim, 2021]	73%	10%	APR1400	Portable DG to supply battery charger LP portable pump to supply feedwater to SG
[Hakobyan and Nierode, 2017]		75%	PWR	DC load shedding Portable DG DDP-FPS to supply inventory
[Lim, 2018]	(18-20)%		APR1400	Small mobile GTG for DC power supply Large mobile GTG for AC power supply
[Ng, 2020]	(5-75)%		PWR	DC load shedding. Portable DG Portable pump to supply feedwater to SG Manual control of TDP-AFW Portable pump to supply RCS
[Zhang and Ma, 2020]	64%		PWR	Portable DG. LP portable pump to supply feedwater to SG Manual control of TDP-AFW
[Jabbari et al., 2020]	99%		VVER-1000	Portable DG.
[Ma et al., 2019]	26%		PWR	Portable DG SG makeup pump
[Son and Lim, 2017]	77%		APR1400	Mobile GTG to supply power for FB Portable pump to supply feedwater to SG
[Shah et al., 2023]	90% [99%]		PWR	Portable DG Portable pump to supply feedwater to SG Portable pump to supply RCS
UPM (2023)	92%	89%	PWR	Alternative LP feedwater to the SGs by FLEX-SGP Alternative LP feedwater to the SGs by DDP-FPS Manual operation of TDP-AFW after DC loss Long term RCS inventory control by FLEX-MUP Alternative RCS boration by FLEX-MUP Alternative RWST inventory control by DDP-FPS Alternative RWST inventory control by FLEX-HCP Supply of AC power by FLEX-DG DC load shedding to increase battery life

3.5 Sensitivity Analysis of the FLEX Strategies

To study the impact of the FLEX strategies described in Section 2, it was decided to classify them into 4 groups, Table 3.16. Using this classification, the FLEX ET CDF has been calculated for each of the groups separately. The results are shown in Tables 3.17 and 3.18. The following conclusions can be obtained:

- Strategy 1, SGs coolant injection: concentrates a high-risk reduction. The inclusion of FLEX-SGP or DDP-FPS decreases the probability of AF-FLEX failure because there are two new independent options allowing injection to SGs. It must also be remarked that the better reliability of the DDP-FPS with respect to the FLEX-SGP is the main reason of this CDF reduction.

- Strategy 2, Coolant injection to RCS: does not represent a significant risk reduction in this scenario because the incorporation of SDS has greatly reduced the probability of seal leakage in the RCPs. However, it is still important in SBLOCA (Δ CDF: -27.9%) and MBLOCA (Δ CDF: -10.7%). It should be noted that in some plants this is not an effective strategy due to low mass-flow rate and will only be useful for the seal LOCA scenario.
- Strategy 3, RWST inventory control: also concentrates a high-risk reduction especially in sequences that require HP recirculation (MBLOCA, SBLOCA, MSLB or SGTR). If HP recirculation fails, this strategy allows to maintain High Pressure Safety Injection by means of an alternative RWST inventory control. In the long-term, containment venting and control of mass flow rate injection to RCS will be needed.
- Strategy 4, Alternate power supply and management: contributes to a lesser extent to risk reduction, and has only minimal importance in the LOOP-SBO scenario, since the probability of external power recovery is relatively high for the accident time evolution considered in this scenario. It may, however, be a very relevant strategy in a scenario without external power recovery, for example, an ELAP due to BDBEE.

It is important to note that these results are strongly dependent on HEPs and failure data for portable equipment, for both of which there are large associated uncertainties. Therefore, it is highly recommended to perform a sensitivity analysis on these uncertainties. In all of them, it is observed that the FLEX strategies considered have a significant impact on risk reduction.

Table 3.16: FLEX strategies considered in the analysis

ID	Strategy Description (associated equipment)	Headers
Strategy 1	SGs coolant injection (FLEX-SGP + DDP-FPS +TDP-Manual)	<ul style="list-style-type: none"> · AF-FLEX · TDP-AFWS-MANUAL
Strategy 2	RCS coolant injection: Long term RCS inventory Control /Alternate RCS boration (FLEX-MUP)	<ul style="list-style-type: none"> · FLEX-MUP · FLEX-MUP-RWST
Strategy 3	RWST inventory control (FLEX-HCP + DDP-FPS-RWST)	<ul style="list-style-type: none"> · RWST-FLEX
Strategy 4	Alternate power supply and management (FLEX-DG + LBS)	<ul style="list-style-type: none"> · LB-FLEX · LBS · R-EX

Table 3.17: FLEX strategies contribution on the risk reduction

ID	Strategy	TOTAL	
		CDF (1/y)	Δ CDF (%)
L1PSA	SPAR-CSN model without FLEX	1.29E-05	-
Strategy 1	SGs coolant injection	6.41E-06	-50.3%
Strategy 2	RCS coolant injection	1.28E-05	-0.5%
Strategy 3	RWST inventory control	6.28E-06	-51.4%
Strategy 4	Alternate power supply and management	1.29E-05	-0.1%
L1PSA-FLEX	SPAR-CSN model with FLEX	1.37E-06	-89.4%

Table 3.18: FLEX strategies contribution on the risk reduction for every IE

Initiator	Strategy 1		Strategy 2		Strategy 3		Strategy 4	
	CDF (1/y)	Δ CDF (%)	CDF (1/y)	Δ CDF (%)	CDF (1/y)	Δ CDF (%)	CDF (1/y)	Δ CDF (%)
GT	1.73E-06	-76.6%	7.38E-06	-0.1%	5.06E-06	-31.5%	7.39E-06	0.0%
LBLOCA	1.15E-08	0.0%	1.15E-08	0.0%	1.15E-08	0.0%	1.15E-08	0.0%
LC	1.06E-07	-79.4%	5.14E-07	0.0%	3.65E-07	-29.0%	5.14E-07	0.0%
LCWA	6.21E-10	-12.9%	7.13E-10	0.0%	7.13E-10	0.0%	7.13E-10	0.0%
LDC-A	2.97E-09	-83.4%	1.79E-08	0.0%	1.37E-08	-23.5%	1.79E-08	0.0%
LDC-B	2.58E-09	-82.8%	1.50E-08	0.0%	1.08E-08	-28.0%	1.50E-08	0.0%
LNSW	4.25E-10	-79.8%	2.10E-09	0.0%	1.48E-09	-29.5%	2.10E-09	0.0%
LOOP	8.40E-08	-80.3%	4.26E-07	-0.2%	3.01E-07	-29.5%	4.16E-07	-2.6%
MBLOCA	1.21E-07	0.0%	1.08E-07	-10.7%	4.17E-08	-65.5%	1.21E-07	0.0%
MSLB-DS	7.01E-07	-7.0%	7.54E-07	0.0%	9.80E-08	-87.0%	7.54E-07	0.0%
MSLB-US	3.40E-08	-18.1%	4.15E-08	0.0%	1.05E-08	-74.7%	4.15E-08	0.0%
SBLOCA	1.62E-07	-1.8%	1.19E-07	-27.9%	1.07E-07	-35.2%	1.65E-07	0.0%
SGTR	3.46E-06	0.0%	3.46E-06	0.0%	2.83E-07	-91.8%	3.46E-06	0.0%

Chapter 4

Application of Probabilistic Safety Assessment for the Selection of DEC-A Sequences

The Fukushima accident has brought, among others, an increment and improvement in the safety requirements for NPPs. One important requirement established in many countries is that the so-called Design Extension Conditions (DEC) need to be fully considered in the assessment of current and new advanced NPPs. Some countries have already included DEC sequences in their respective regulations:

- Belgium: Royal Decree SRNI-2011 [[SRNI, 2011](#)];
- Canada: Design of Reactor Facilities, REGDOC-2.5.2 [[CNSC, 2014a](#)]; Design Extension Conditions for Nuclear Power Plants, DIS-14-01 [[CNSC, 2014b](#)];
- China: Nuclear Power Plant Design Safety Regulations, HAF102-2016 [[NNSA, 2016](#)];
- Czech Republic: Safety Instructions BN-JB-1.7; BN-JB-2.2 [[SUJB, 2019](#), [SUJB, 2021](#)];
- Finland: GUIDE YVL B.3, Deterministic Safety Analyses for a Nuclear Power Plant [[STUK, 2019a](#)]; GUIDE YVL B.4, Nuclear Fuel and Reactor [[STUK, 2019b](#)];
- Japan: New Regulatory Requirements for Light Water Nuclear Power Plants [[NRA, 2013](#)];
- Korea: Regulations on the scope of accident management and detailed criteria for evaluating accident management capabilities [[NSSC, 2016](#)];
- Slovenia: Rules on radiation and nuclear safety factors - JV5 [[SNSA, 2017](#)];
- Spain: Nuclear Safety Council Instruction IS-37 [[CSN, 2015b](#)]; and
- UK: Safety Assessment Principles for Nuclear Facilities [[ONR, 2020](#)].

The DEC analysis were introduced by the International Atomic Energy Agency (IAEA) in the Safety Standards Series No. SSR-2/1 (Rev. 1), Safety of Nuclear Power Plants: Design

[IAEA, 2016], and comprise both conditions in events without significant fuel degradation and conditions in events with core melting [IAEA, 2020b]. The introduction DEC was undertaken with the objective of advancing the safety capabilities of NPPs. This enhancement aimed to enable NPPs to endure, without yielding unacceptable radiological consequences, accidents that surpass the severity outlined in design basis accidents (DBA) or involve additional failures [IAEA, 2020b]. According to IAEA [IAEA, 2020b] the analysis of DEC aims at:

- The selection of additional scenarios to be included in the design;
- The definition and design of features attributed to DEC with core melting, ensuring that these features meet the specified performance criteria in accordance with relevant safety requirements;
- Assistance in the establishment and validation of severe accident management guidelines;
- Provision of information on the environmental conditions under which systems required to address DEC must perform their intended functions effectively;
- Support in the safety classification of structures, systems, and components (SSCs) as deemed appropriate;
- Contribution of input for emergency preparedness and response measures.

The analysis of the DEC sequences focuses on the mitigation of complex accident sequences involving multiple failures of the safety systems, as included in current safety guides worldwide, e.g., as proposed by IAEA SSR 2/1 [IAEA, 2016]. Associated applications and practices begin to emerge, given that the topic of DEC is being advanced rapidly both nationally and internationally. Nevertheless, these practices are still not comprehensive regarding the interface with the plant design basis, its role in the Defense-in-Depth, selection of requirements, impact on operating limits and conditions, and/or selection of DEC sequences to be included in the analysis. Two DEC categories have been defined [WENRA, 2020]:

- DEC-A: prevention of severe fuel damage (core or spent fuel storage) can be achieved.
- DEC-B: severe fuel damage.

As for the selection of DEC-A sequences, most, if not all, the current practices/guidelines indicate that “the selection of events to be analyzed shall be justified on the basis of deterministic and probabilistic arguments and of engineering judgment” and that “the selection process shall take into account all those events or combinations of events that cannot be considered extremely unlikely with a high degree of confidence and that might give rise to accident conditions more severe than those considered in design basis accidents” [CSN, 2015b]. However, there is a notable absence of specific details on the quantitative application of this fundamental principle. Thus, there is a broad claim for the use of the PSA models as a key tool for the DEC identification process.

The development of the DEC analysis is progressing rapidly both nationally and internationally. Nevertheless, current practices are not yet fully consistent with other elements of design-basis event, and the process of identification and selection of these specific conditions is still under

discussion [IAEA, 2020b]. Current practices and guidelines indicate that the selection of events to be analyzed shall be justified through engineering judgment, deterministic assessments, and probabilistic assessments [CSN, 2015b, IAEA, 2016]. Nevertheless, there are few specific details about how this basic principle is quantitatively applied, and only some qualitative rationales can be found, in which PSA models are a key tool for the DEC identification process. Yang et al. [Yang et al., 2017] proposed a four-step identification process for EPR using a PSA model to assess the contribution to the risk of the features not credited in the DBA analysis. Viktorov & Harwood [Viktorov and Harwood, 2015, Harwood et al., 2017] state that the identification of DEC can be seen as a two-step process in which PSA would help to identify the dominant contributors to the CDF and large release frequency, as well as events that challenge the core and containment integrity. Kim & No [Kim and No, 2019] proposed a methodology to estimate the safety margin of the DEC and the conditional CDF by a combination of deterministic safety analysis (DSA) and PSA. Wang et al. [Wang et al., 2019] used a PWR PSA model to identify multiple failure events, identified as DEC-A sequences if the CDF is greater than $1.00\text{E-}08$ 1/y. Ma et al. [Ma et al., 2021] applied the same process to VVER-1200 with a screening value set to $1.00\text{E-}07$ 1/y. Havet et al. [Havet et al., 2023] proposed a DEC selection criteria based on: L1PSA and L2PSA internal events; internal and external hazards and a combination of events. Matejovic et al. [Matejovic et al., 2023] applied L1PSA with a focus on beyond DBA with higher contribution to CDF in Dukovany NPP.

New practices and guidance begin to emerge, given that the topic of DEC is rapidly advancing in the regulatory arena. Despite this, there is a certain consensus regarding the definition of DEC-A as sequences with multiple failures and conditions without significant fuel degradation, such as:

1. Initiating events that could lead to a situation beyond the capability of safety systems that are designed for a single initiating event, e.g., multiple SGTR in PWR.
2. Multiple failures, e.g., common cause failures in redundant trains, that prevent the safety systems from performing their intended function to control the postulated initiating events, e.g., MBLOCA/SBLOCA without actuation of High-Pressure Safety Injection (HPSI).
3. Multiple failures that cause the loss of a safety system while its system is used to fulfill the fundamental safety functions in normal operation, e.g., heat removal in accident conditions and during shutdown by means of a single system.

Nevertheless, these regulations do not normally identify frequency boundaries for DEC that would help to reduce the number of sequences to be analyzed. Recurrent reasoning for this lack of quantitative criteria refers to the difficulties in obtaining credible frequency values for low-frequency events, which may include multiple failures of equipment and human errors, as well as the inherent uncertainties. In the end, most common practices for identifying events to be considered as DEC inevitably fall back on the judgment of experts, which adds further notable uncertainties [CSN, 2015b, General Nuclear System Ltd., 2020a, Ma et al., 2021].

Given that the set of DEC is specific to the reactor technology and to design options, identification, and classification of events to be considered in design is the responsibility of the designer or the applicant for a license, and not to be imposed by the regulator who, however, is required to verify the results [Harwood et al., 2017]. A comparison of the different lists of DEC-A sequences by country and design (Tables 4.1 and 4.2) [CSN, 2015b, General Nuclear System Ltd., 2020a, General Nuclear System Ltd., 2020b, Havet et al., 2023, Kral, 2019, NNB Generation Company (HPC) LTD, 2017, Prošek and Uršič, 2019, Jung et al., 2021, STUK, 2019c, SUJB, 2019, Wang et al., 2019, Yang et al., 2017] has made it possible to summarize them as follows:

- ATWS;
- SBO;
- LUHS;
- Loss of CCWS (LCCW/LESW);
- Total loss of feedwater (TLFW);
- SGTR combinations (MSLB + SGTR; SGTR + PORV open; SGTR + Isolation failure);
- LOCA (including SBLOCA + CCWS/ESWS failure; LOCA + HPSI or LPSI failure);
- Loss of SFP cooling; uncontrolled boron dilution; LOCA at low power and shutdown (LPSD) conditions;
- Uncontrolled reactor level drop at LPSD conditions;
- Loss of RHR at LPSD conditions.

Table 4.1: DEC-A sequences by country or design (1/2)

	Belgium	Bulgaria	China (VVER-1200)	Czech Republic	Hungary	India	Japan
ATWS	ATWS + SORV	ATWS	ATWS	ATWS	ATWS		ATWS
SBO	SBO	SBO	SBO	SBO	SBO	SBO	SBO
LUHS			LUHS	LUHS (from secondary circuit)	LUHS		
LCCS/LESWS		LCCWS		LCCWS			LCCWS
TLFW (F&B)		TLFW	TLFW	TLFW	TLFW		TLFW
		MSGTR	MSGTR	MSGTR	MSGTR		
SGTR	SBL + SGTR		SBL + SGTR	SLB + SGTR	SLB + SGTR		
			SGTR + Isolation failure				SGTR + other failures
LOCA	SB-LOCA + LPSI Failure	LOCA + failure of one ECCS	SB-LOCA + HPSI Failure	LOCA + failure of ECCS	LOCA + failure of ECCS	LOCA + failure of ECCS (injection or recirculation)	LOCA + failure of ECCS (injection or recirculation)
					ISLOCA		ISLOCA
						LOCA+spray failure	
BORON	Uncontrolled boron dilution	Uncontrolled boron dilution		Uncontrolled boron dilution	Uncontrolled boron dilution		
SHUTDOWN		Uncontrolled reactor level drop or loss of circulation in regime with open reactor orduring refueling		Uncontrolled reactor level drop or loss of circulation in regime with open reactor orduring refueling	Uncontrolled reactor level drop or loss of circulation in regime with open reactor orduring refueling		
RHR (Shutdown)		Loss of core cooling due to failure of RHRS		Loss of core cooling due to failure of RHRS	Loss of core cooling due to failure of RHRS		Loss of core cooling due to failure of RHRS
Long term / Others		Loss of required safety systems in the long term		Loss of required safety systems in the long term After DBA	Loss of required safety systems in the long term		
		Break of the high pressure components				Other specific scenarios which are technology specific	
SFP		Loss of SFP cooling		Loss of SFP cooling	Loss of SFP cooling		

Table 4.2: DEC-A sequences by country or design (2/2)

	Slovenia	Slovakia	Spain	Sweden	UK EPR	UK HPR1000	
ATWS	ATWS	ATWS	ATWS	ATWS	ATWS	ATWS	
SBO	SBO	SBO	SBO	ELAP (Post-FD)	SBO	SBO	
LUHS	LUHS	LUHS	LUHS	LUHS (Post-FD)	LUHS for 100 hours (states A to C)	LUHS for 100 Hours	
LCCS/LESWS		LESWS	LCCWS-LESWS		Total loss of the cooling chain (TLOCC) with RCP SLOCA	TLOCC with RCP SLOCA	
						TLOCC with failure of EFW	
TLFW (F&B)		TLFW	TLFW		TLFW	TLFW	
SGTR		MSGTR	MSGTR			MSGTR (10 tubes)	
		SLB + SGTR				SLB + SGTR	
						SGTR + PORV OPEN	
LOCA		RCS leakage combined with ECCS failure (LH or HH system)	LOCA + total loss of an ECCS		LOCA up to 20 cm2 and failure of the SI signal	SB-LOCA with Failure of Medium Pressure Rapid Cooldown	
						LOCA up to 20 cm2 without MHSI	SB-LOCA with Total Loss of MHSI
						LOCA up to 20 cm2 without LHSI	SB-LOCA with Total Loss of LHSI
BORON	Uncontrolled boron dilution in reactor	Uncontrolled boron dilution in reactor	Uncontrolled boron dilution and in SFP with credit for dissolved boron		Homogenous boron dilution and failure of the operator's actions (states Cb and D)		
SHUTDOWN			Uncontrolled loss of level in PWR during midloop operation or refueling outages		Uncontrolled Level Drop and failure of the activation of the SIS	Uncontrolled Primary Water Level Drop without SI Signal	
		Loss of reactor coolant				SB-LOCA with Total Loss of LHSI	
						TLOCC	TLOCC
RHR (Shutdown)		Loss of core cooling due to failure of RHRS	Loss of core cooling due to failure of RHRS			Loss of core cooling due to failure of RHRS or Failure of Recovery of RHR after Loss of Offsite Power	
Long term / Others		Loss of required safety systems in the long term	Loss of required safety systems in the long term	Internals/external Hazards (Post-FD)			
	External flood	Break of the main pressure components		CCF sequences			
SFP	SFP leak	Loss of SFP cooling	Loss of SFP cooling		Loss of the two main trains of the SFP Cooling System during core refueling	Loss of all SFP Cooling and Treatment System Trains	

In conclusion, it seems that there is a need for an exhaustive and comprehensive method that, using a set of quantitative indicators (Figures of Merit, FoM), would be able to screen out the space of sequences, rank them, and truncate the search by means of suitable cut-off rules.

In addition to this Introduction Section, the present chapter is organized as follows: Section 4.1 outlines the DEC-A identification methodology proposed and its application to the PSA model. Section 4.2 describes how the post-Fukushima improvements impact on DEC sequences. Section 4.3 shows the impact of the implementation of post-Fukushima improvements in DEC sequences. Section 4.4 outlines a comparison with other DEC-A improvements included in several technologies. Finally, the conclusions drawn from this research are set out in Section 6.3.

4.1 DEC-A Identification Methodology and Implementation on the SPAR-CSN Model

This section shows the DEC-A sequence identification methodology developed by UPM and CSN, as well as its application to the PSA model of the SPAR-CSN project. The method, composed of 11 steps (see Figure 4.1), is described below:

- S1** Select an initiating event from the list of IEs covered in the PSA model.
- S2** Select Sequences from the event tree of the IE with at least 1 failed header (SEQ).
- S3** Identify SEQ leading to a safe state (S):
 - (a) These are candidates to be DEC-A sequences (i.e., sequences where multiple failures occur but lead to a safe state).
 - (b) Tag each identified sequence with **DEC-A** in the code (RiskSpectrum™) for the quantification process.
- S4** From each sequence tagged as DEC-A, identify all the sequences with additional failures that lead to CD:
 - (a) These are candidates to be DEC-B-related sequences of the tagged DEC-A sequence.
 - (b) Tag each identified sequence with **DEC-B** in the code (RiskSpectrum™) for the quantification process.
- S5** Quantify the frequency of every DEC-A tagged sequence, $DEC-A_{freq}$. This step will provide a sorted list, which is helpful to the subsequent steps of the method.

Recurrent application of steps **S1** to **S5** allows obtaining a complete and sorted list of DEC-A sequences of the PSA model in conjunction with their related DEC-B, i.e., sequences originated from each DEC-A sequence after subsequent failures, resulting in core damage.

- S6** Apply a DEC-A frequency truncation criterion for sequence list reduction. The truncation criterion depends on the analysis target: a lower frequency limit requires the analysis of a larger set of DEC sequences.

S7 Quantify the accumulated frequency of the DEC-B sequences related to each DEC-A sequence, $DEC-B_{freq}$ (4.1). The method also postulates the use of the Conditional Core Damage Probability of each DEC sequence, $CCDP_{DEC}$ (4.2), as FoM for every element of the collapsed list.

$$DEC-B_{freq} = \sum_{i=1}^n \text{Frequency}_i^{DEC-B} \quad (4.1)$$

$$CCDP_{DEC} = \frac{DEC-B_{freq}}{DEC-A_{freq} + DEC-B_{freq}} \quad (4.2)$$

- S8** Group similar DEC-A sequences and remove low-risk ones, $DEC-B_{freq} < p_0$ or $CCDP_{DEC} < p_1$, and sequences covered by the DSA. The outcome would be a collapsed list of the most important DEC-A sequences by risk and their accumulated DEC-B-related frequency.
- S9** Add new sequences in terms of other eventual criteria not considered up to now (e.g., large early release frequency, releases, regulations, etc.)
- S10** Apply steps **S1** to **S9** to other PSA models such as fire protection, SFP, external events, LPSD, etc.
- S11** Add new sequences in terms of other sources of DEC-A identification, such as dynamic PSA, operating experience, expert judgment, etc.

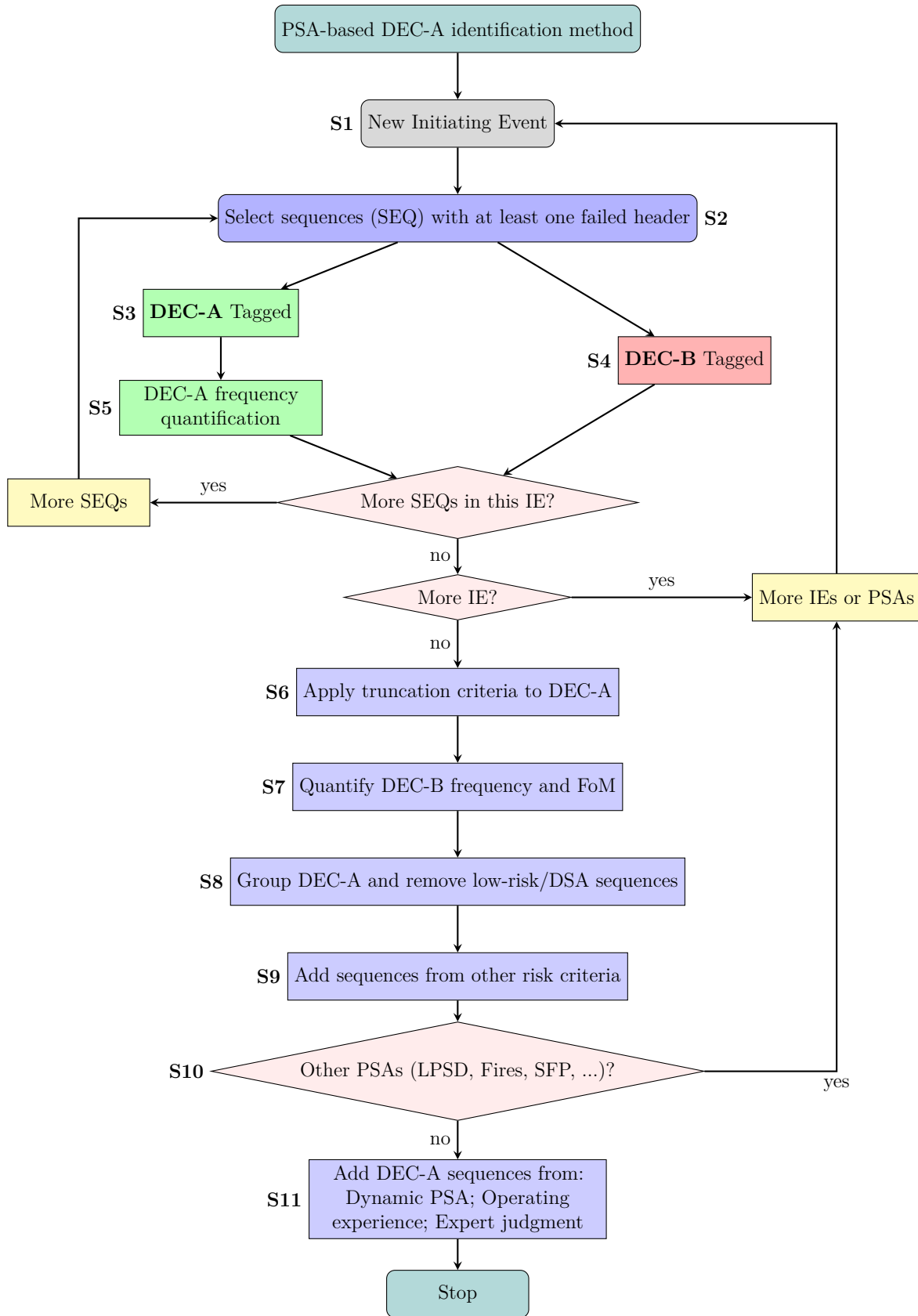


Figure 4.1: Flowchart of the PSA-based DEC-A identification method.

The application of the methodology to the SPAR-CSN PSA model resulted in a tentative list of DEC-A sequences with their frequency and FoM which represent their relative risk. Following the description of the methodology, the steps S1 to S11 of the method have been implemented through the following stages, Figure 4.2:

1. Steps **S1** to **S5** yield 182 DEC-A sequences as sequences with multiple failures which lead to a safe state.
2. In a subsequent step, the initial list of 182 sequences was reduced to 69 sequences after the application of a frequency truncation criteria of $1.00E-08$ 1/y (Step **S6**).
3. Following Step **S7**, the DEC-B sequences related to the reduced list of DEC-A have been quantified to obtain the FoM $DEC-B_{freq}$ and $CCDP_{DEC}$. Step **S8**, related to the elimination and grouping of sequences, was applied to the reduced list of 69 elements. In this process, 7 sequences were eliminated because they were analyzed in the DSA and additional 3 low-risk sequences ($DEC-B_{freq} < 1.00E - 09$ 1/y and $CCDP_{DEC} < 1.00E - 04$) sequences were eliminated, providing a list of 59 DEC-A sequences. Finally, some of the remaining sequences were grouped by similarity, resulting in a new collapsed list of 10 DEC-A sequences.
4. Through steps **S9** to **S11**, five sequences were added, all of them by expert judgment. The reason for the addition of sequences has been the lack of other criteria or consideration in the PSA.

As examples, Figures 4.3 and 4.4 show the DEC-A and DEC-B sequences for GT and SBLOCA initiators, respectively.

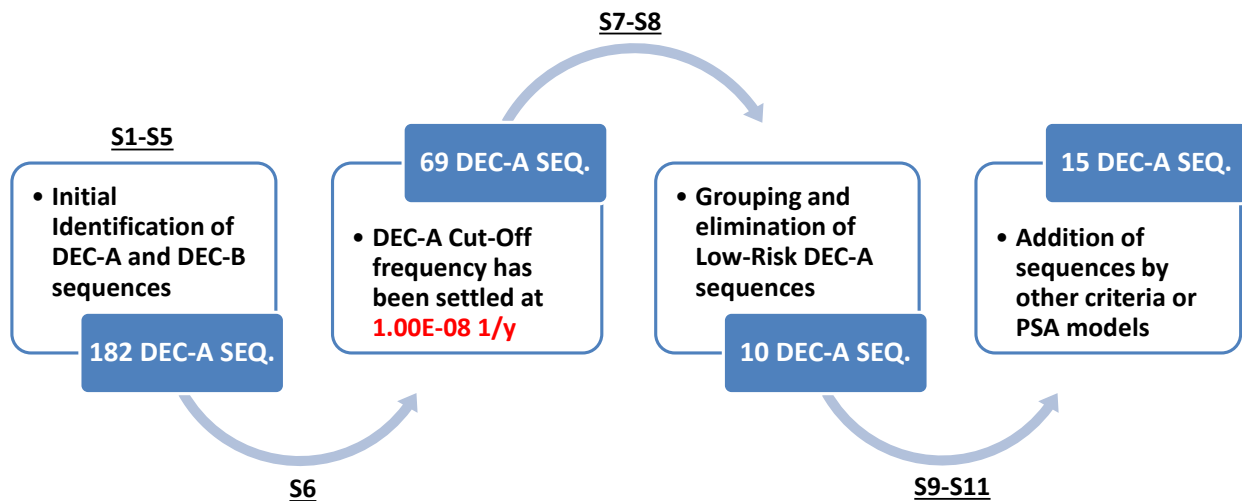


Figure 4.2: Amount of DEC-A sequences in the DEC-A identification method steps.

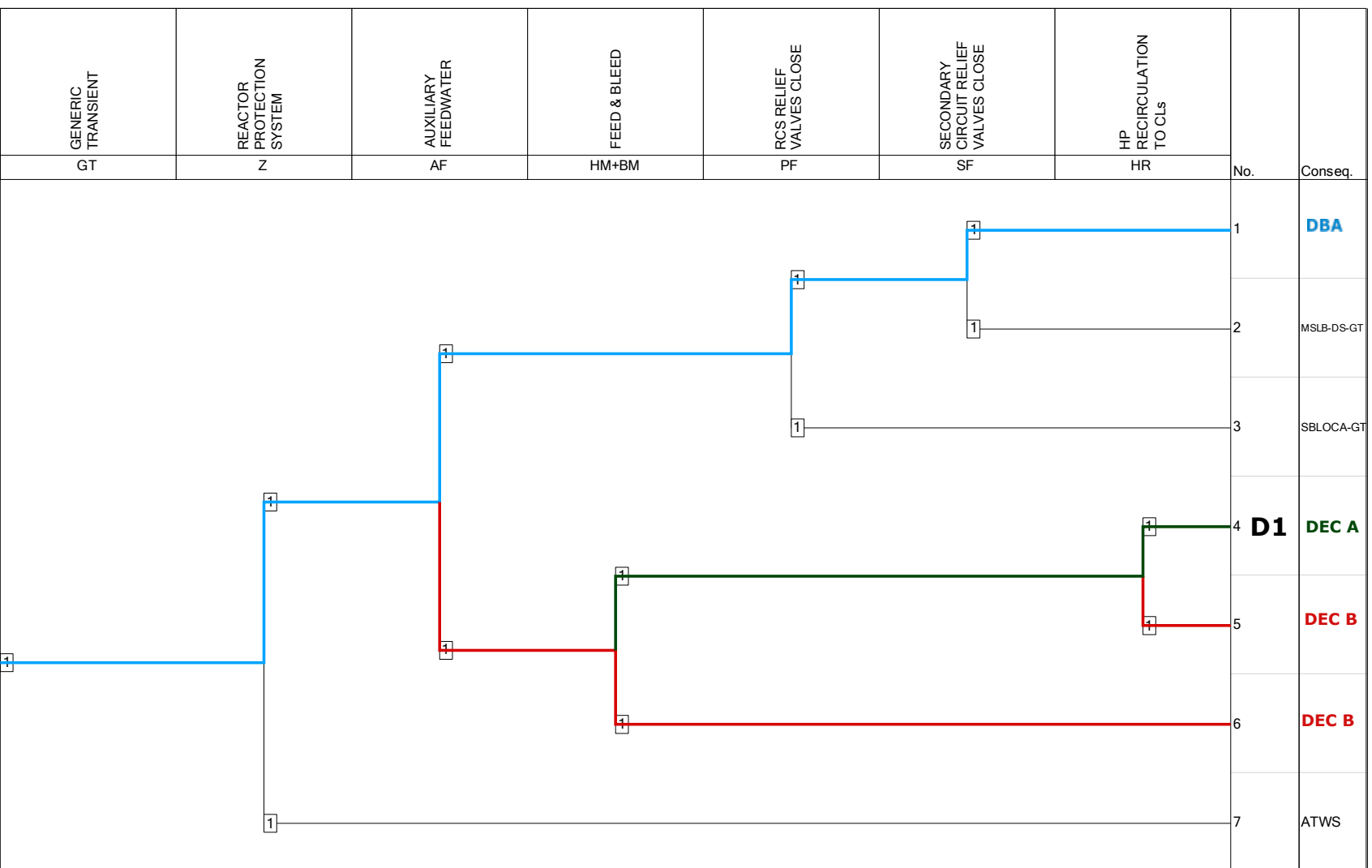


Figure 4.3: DEC-A and DEC-B sequences for GT

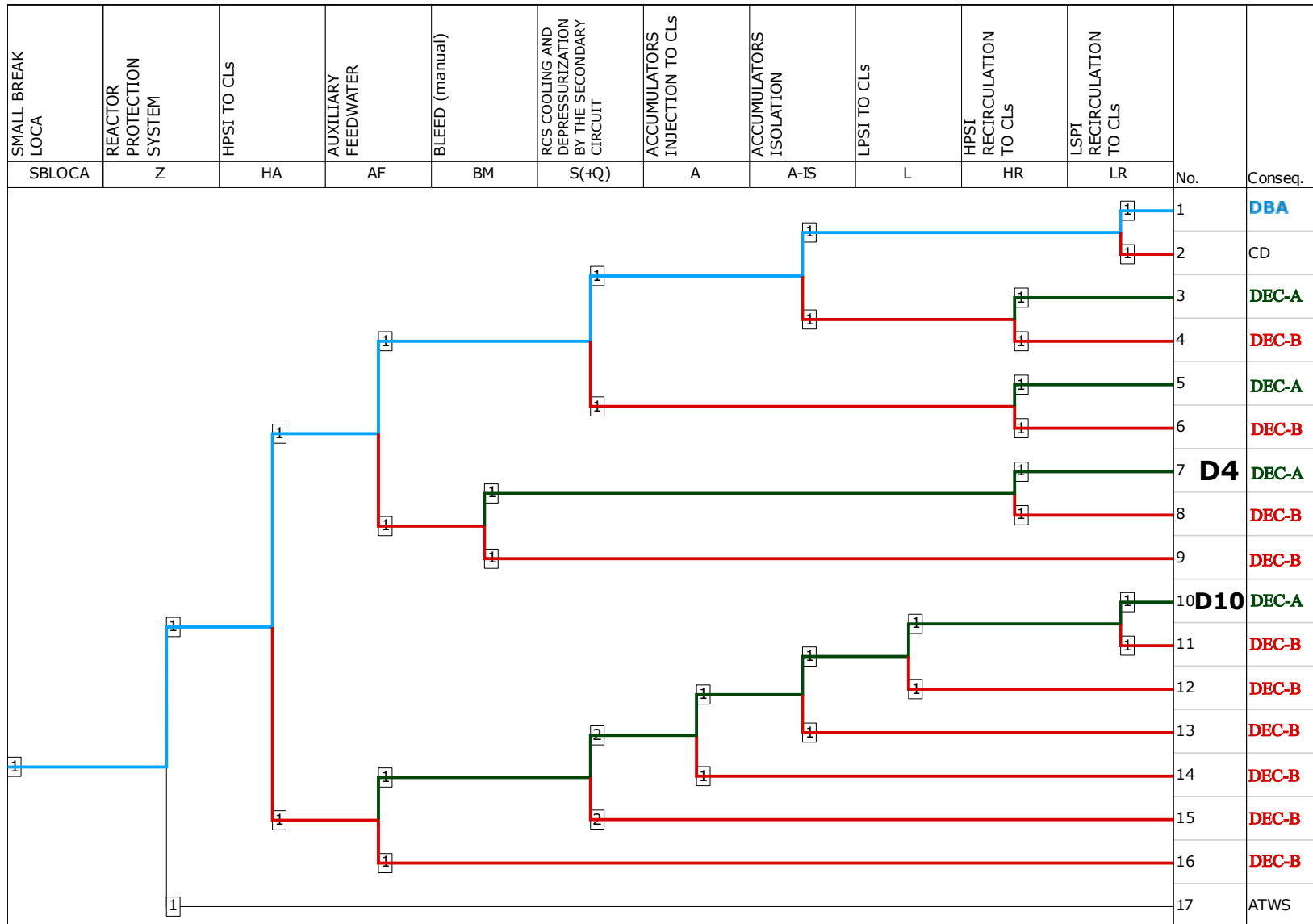


Figure 4.4: DEC-A and DEC-B sequences for SBLOCA

The outcome of the final stage is Table 4.3, which shows 15 DEC-A sequences sorted by cumulative DEC-B frequency, $DEC-B_{freq}$. To make the phenomenology of the sequences in Table 4.3 clearer, a more in-depth description of the sequences is included below:

- D1** TLFW with successful F&B and high-pressure (HP) recirculation.
- D2** SGTR with a stuck-open pressurizer relief valve (SORV), and HPSI actuation (both in injection and recirculation modes).
- D3** Steam line break (SLB) with SORV, and HPSI actuation (both in injection and recirculation modes).
- D4** SBLOCA/MBLOCA with failure to cooldown and depressurize the RCS, and HPSI actuation (both in injection and recirculation modes).
- D5** SBO with offsite power recovery.
- D6** SLB with induced SGTR, HPSI actuation, and successful RCS cooldown and depressurization.
- D7** LCCW with successful reactor coolant pump (RCP) seals injection by the hydrostatic test pump (HTP).
- D8** ATWS.
- D9** SGTR with steam generator (SG) isolation failure, HPSI actuation, closing of RCS relief paths, and decrease of safety injection flow rate.
- D10** SBLOCA/MBLOCA with HPSI actuation failure, and successful cooldown and depressurization of RCS.
- D11** Multiple SGTR (MSGTR).
- D12** Boron dilution.
- D13** Loss of the RHR.
- D14** Loss of SFP cooling.
- D15** Two simultaneous SLB by airplane crashing.

It is important to remark the comparative purpose of the last column of Table 4.3; this represents the appearance of each DEC-A sequence in other current DEC lists found in the literature.

Table 4.3: List of DEC-A sequences identified in the SPAR-CSN model

No.	Sequence	DEC-A_freq (1/y)	DEC-B_freq (1/y)	$CCDP_{DEC}$	List?
D1	TLFW	5.75E-05	7.76E-06	1.19E-01	Yes
D2	SGTR + SORV	3.38E-06	3.31E-06	4.94E-01	Yes
D3	SLB + SORV	2.12E-05	1.36E-06	6.05E-02	No
D4	SBLOCA/MBLOCA + RCS cooldown and depressurization failure	1.57E-06	2.62E-07	1.43E-01	Yes
D5	SBO (without SDS)	7.30E-05	1.33E-07	1.82E-03	Yes
D6	SLB + SGTR	4.80E-05	1.10E-07	2.29E-03	Yes
D7	LCCW (without SDS)	1.94E-07	6.42E-08	2.49E-01	Yes
D8	ATWS	1.90E-06	5.67E-08	2.90E-02	Yes
D9	SGTR + SG isolation failure	1.77E-07	2.41E-08	1.20E-01	Yes
D10	SBLOCA/MBLOCA + failure of HPSI	3.30E-08	1.82E-08	3.55E-01	Yes
D11	MSGTR	Engineering Judgment			Yes
D12	Boron dilution				Yes
D13	Loss of RHR				Yes
D14	Loss of SFP cooling				Yes
D15	Two simultaneous MSLB				Yes

DEC-A sequences with the highest relative risk ($CCDP_{DEC} > 1.00E - 02$ and $DEC-B_{freq} > 1.00E - 07$ 1/y) of the SPAR-CSN model are highlighted in **red** in Table 4.3. The high-values of $DEC-B_{freq}$ and $CCDP_{DEC}$ for sequences, **D1**, **D2**, **D3**, and **D4** are due to the dependencies between human actions, modeled as part of the human reliability analysis of the PSA model. Sequence **D3** is not included in any published DEC list, and, despite being one of the four DEC-A sequences of the model with the highest relative risk. This reveals the potential of the use of PSA-based methodology to identify DEC-A sequences missed by DSA.

Finally, the relative risk of sequences **D8**, **D9**, and **D10**; included in Table 4.3, may be reduced by implementing other modeling assumptions (evolution of the moderator temperature coefficient between the beginning of the cycle and end of the cycle) or implementation of new human actions or equipment currently included at NPPs (implementation of motor-operated RV-line isolation valve, “fast cooldown” beyond 55 K/h).

It should be noted that the DEC-A sequences, in general, have associated human actions that are dominant in their potential risk. The dominant human actions corresponding to sequences **D1** to **D10** are listed in Table 4.4.

Table 4.4: Dominant human actions associated to identified DEC-A sequences

No.	Sequence	Dominant Human Action	Fig
D1	TLFW	F&B	D.1
D2	SGTR + SORV	Manual steam line isolation; Balance primary and secondary pressures; Change HPSI to recirculation mode	D.2
D3	SLB + SORV	Change HPSI to recirculation mode	D.3
D4	SBLOCA/MBLOCA + RCS cooldown and depressurization failure	Change HPSI to recirculation mode	D.4
D5	SBO	Recovery of AC power supply	D.5
D6	SLB + SGTR	Reduce HPSI flow	D.6
D7	LCCW	HTP alignment to the RCP seals	D.7
D8	ATWS	Emergency boration by means of HPSI	D.8
D9	SGTR + SG isolation failure	Manual bleed; Change HPSI to recirculation mode	D.2
D10	SBLOCA/MBLOCA + failure of HPSI	RCS cooldown and depressurization by means of SG	D.4

4.2 Post-Fukushima Improvements in DEC-A Sequences

As stated in the Chapter 3 events that lead to the loss of safety-related systems, like ELAP or LUHS, can severely compromise key safety functions associated with core cooling and containment, ultimately leading to reactor CD. The FLEX Strategies were designed to mitigate such accidents but they are also useful to avoid CD in sequences due to internal events. Therefore, they could be incorporated in the L1PSA to analyze their impact on CDF and to quantify the plant risk reduction. This work quantifies the impact of FLEX strategies on DEC sequences by using the generic model introduced in Chapter 2 and using the methodology provided in Courtin et al. [Courtin et al., 2023]. The equipment considered within the modeling FLEX strategies includes:

- FLEX-DG
- FLEX-SGP
- FLEX-MUP
- FLEX-HCP
- DDP-FPS

The FLEX Actions considered are the same introduced en Section 3.3. The incorporation of FLEX strategies requires the identification of the corresponding DEC-A sequences in the ETs incorporating these strategies (see Figures 4.5 and 4.6).

Another of the improvements introduced in NPPs since the Fukushima-Daiichi accident has been the installation of SDS. This equipment is especially useful for SBO, or ELAP type accidents and loss of safety support systems related to cooling of the RCP seals. In

these conditions, when the RCPs seals heat up due to the loss of the coolant injection, SDS actuate to prevent inventory losses and the subsequent release of radioactive material into the containment. For this reason, the model for SDS has been introduced in LOOP/SBO and LCCW ETs. The post-Fukushima improvements: FLEX strategies (FA + equipment) and SDS, that affect each DEC-A sequence are shown in Table 4.5.

Table 4.5: Relationship between DEC-A Sequences and post-Fukushima improvements

No.	Sequences	FLEX Action	Equipment
D1	TLFW	FA-1/2: Alternative LP feedwater to the SGs FA-6/7: Alternative RWST inventory control	DDP-FPS FLEX-SGP FLEX-HCP
D2	SGTR + SORV	FA-6/7: Alternative RWST inventory control	FLEX-HCP
D3	SLB + SORV	FA-1/2: Alternative LP feedwater to the SGs FA-6/7: Alternative RWST inventory control	DDP-FPS FLEX-SGP FLEX-HCP
D4	SBLOCA/MBLOCA + RCS cooldown and depressurization failure	FA-1/2: Alternative LP feedwater to the SGs FA-6/7: Alternative RWST inventory control	DDP-FPS FLEX-SGP FLEX-HCP SDS
D5	SBO	FA-1/2: Alternative LP feedwater to the SGs FA-3: Manual operation of TDP-AFW FA-4: Long term RCS inventory control FA-8: Supply of AC power by FLEX-DG FA-9: DC load shedding	DDP-FPS FLEX-SGP TDP-AFW FLEX-MUP FLEX-DG SDS
D6	SLB + SGTR	FA-6/7: Alternative RWST inventory control	FLEX-HCP
D7	LCCW	FA-1/2: Alternative LP feedwater to the SGs	DDP-FPS FLEX-SGP SDS
D8	ATWS	FA-5: Alternative RCS boration	FLEX-MUP
D9	SGTR + SG isolation failure	-	-
D10	LOCA + HPSI failure	FA-1/2: Alternative LP feedwater to the SGs FA-4: Long term RCS inventory control	DDP-FPS FLEX-SGP FLEX-MUP

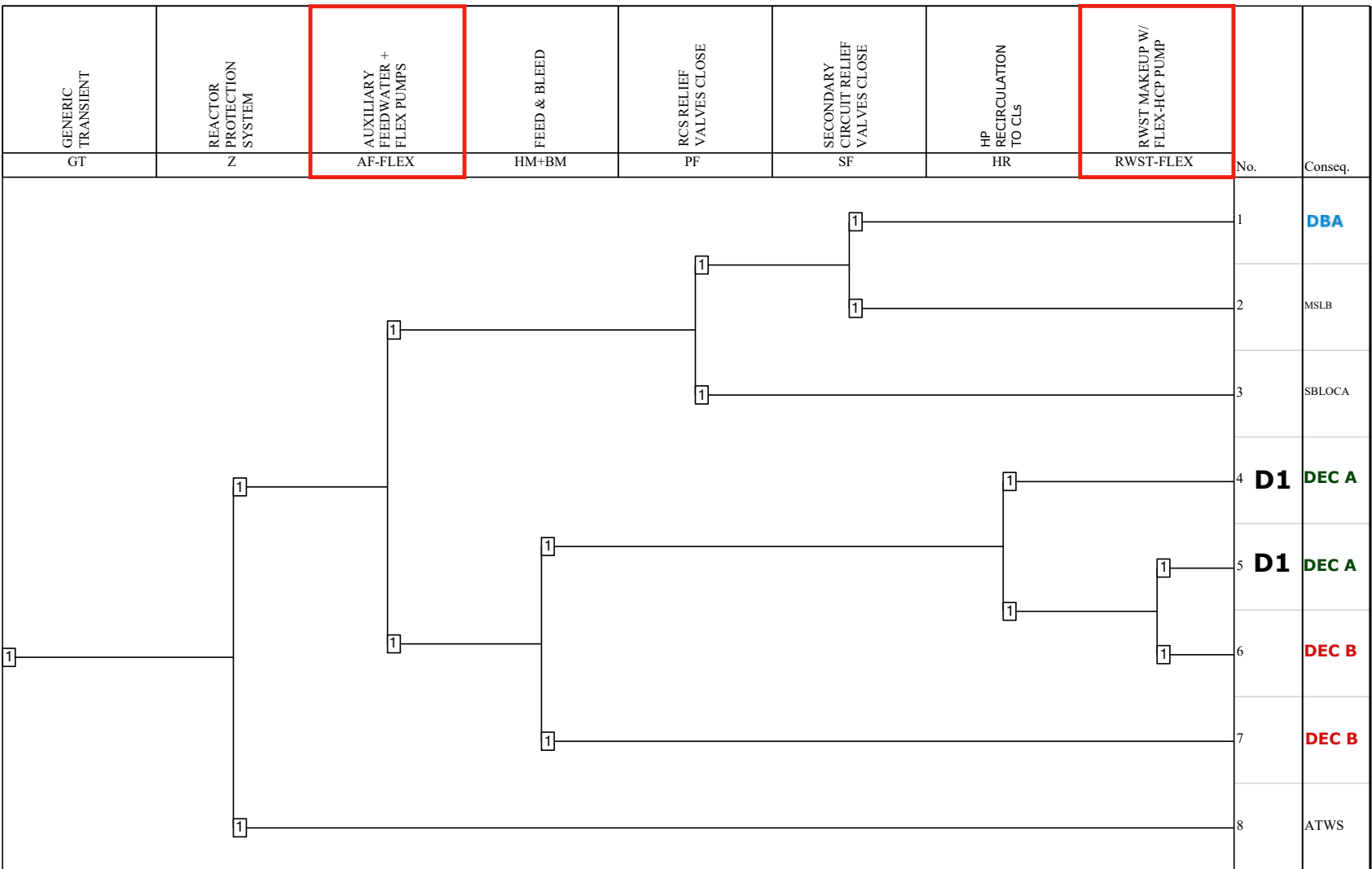


Figure 4.5: DEC-A and DEC-B sequences for GT with FLEX strategies

SMALL BREAK LOCA	REACTOR PROTECTION SYSTEM	HPSI TO CLs	AUXILIARY FEEDWATER + FLEX PUMPS	BLEED	RCS COOLING AND DEPRESSURIZATION BY THE SECONDARY CIRCUIT	ACCUMULATORS INJECTION TO CLs	ACCUMULATORS ISOLATION	LPSI TO CLs	HP RECIRCULATION TO CLs	LP RECIRCULATION TO CLs	LONG-TERM RCS MAKEUP W/ FLEX-MUP PUMP	RWST MAKEUP W/ FLEX-HCP PUMP	No.	Conseq.
SBLOCA	N	HA	AF-FLEX	BM	S(+Q)	A	A-IS	L	HR	LR	FLEX-MUP	RWST-FLEX		
													1	DBA
													2	DEC A
													3	DEC B
													4	DEC A
													5	DEC A
													6	DEC B
													7	D4 DEC A
													8	D4 DEC A
													9	DEC B
													10	DEC A
													11	DEC A
													12	DEC B
													13	DEC B
													14	D10 DEC A
													15	D10 DEC A
													16	DEC B
													17	DEC B
													18	DEC B
													19	DEC B
													20	DEC B
													21	DEC B
													22	ATWS

Figure 4.6: DEC-A and DEC-B sequences for SBLOCA with FLEX strategies

4.3 Risk Reduction of DEC-A Sequences Due to Post-Fukushima Improvements

Considering the previous post-Fukushima improvements, the PSA model is quantified again, Table 4.6. A decrease in the frequency of DEC-B sequences is observed, but also a decrease in the frequency of DEC-A sequences, as a net result of the implementation of FLEX strategies. The main contributors to these risk reduction results shown in Table 4.6 are due to:

- **D1** (TLFW): The largest reduction is due to the FA-1/2 ($\sim 70\%$) avoiding the loss of the secondary-side heat removal by means of an alternative feedwater to the SGs. Then, if HP recirculation fails, FA-6/7 allow to maintain HPSI injection by means of an alternative RWST inventory control, this accounts for 30% of risk reduction. In the long-term, containment venting and control of mass flow rate injection to RCS will be needed (long-term measures). In addition, the inclusion of the FLEX strategies affects both DEC-A and DEC-B, decreasing the frequency of both.
- **D2** (SGTR + SORV): The main contributors to the CDF reduction are FA-6/7 ($\sim 90\%$) that allow to maintain HPSI injection by means of an alternative RWST inventory control, with corresponding long-term measures.
- **D3** (SLB + SORV): The largest reduction ($\sim 80\%$) is due to the FA-6/7, if HP recirculation to CLs fails. These FA allow to maintain HPSI injection by means of alternative RWST inventory control, also with long-term measures. Besides, FA-1/2 can avoid the loss of secondary-side heat removal by means of alternative feedwater to the SGs ($\sim 20\%$).
- **D4** (SBLOCA/MBLOCA + failure to cooldown and depressurize the RCS): The largest risk reduction is due to FA-6/7 ($\sim 90\%$) maintaining HPSI injection by means of an alternative RWST inventory control, with corresponding long-term measures. Also, there is a reduction in the relative risk of this sequence due to the reduction in the frequency of a seal LOCA following another IE, such as LOOP, obtained thanks to SDS actuation ($\sim 10\%$).
- **D5** (SBO): The dominant contributors to the CDF decrease are FA-1/2 ($\sim 70\%$) avoiding the loss of secondary-side heat removal. Also, SDS avoid reaching an SBLOCA situation, taking $\sim 30\%$ of the relative risk reduction. All other FA have a minor impact on this scenario.
- **D6** (SLB + SGTR): The largest reduction is due to the FA-6/7 ($\sim 90\%$) allowing to maintain HPSI injection by means of alternative RWST inventory control, with long-term measures.
- **D7** (LCCW): The SDS are the most important improvement in these sequences, avoiding a LOCA situation (together with RCP seals injection by the HTP, $\sim 99\%$).
- **D8** (ATWS): The unique strategy that affects this sequence is FA-5 to supply boron injection to RCS by means of the FLEX-MUP, taking borated water from RWST, and is responsible for a 12% decrease in risk.

- **D9** (SGTR + SG isolation failure). There appears to be no risk reduction in this sequence, but this is due to the conservatism of the model. Strategies such as FA-6/7 or FA-4 could be considered to reduce the risk of this sequence.
- **D10** (SBLOCA/MBLOCA + HPSI actuation failure): The dominant failure in the DEC-B sequences related to their DEC-A sequences is “RCS cooling and depressurization by the secondary circuit”, which prevents the FA-3 from supplying the RCS effectively, which explains the small impact seen of post-Fukushima improvements in the results.

It is observed that overall, SDS are well suitable for reducing the risk of sequences D4, D5, and D7, although it causes a relative risk reduction (Δ CDF) of only 3% in the CDF of the entire L1PSA model; from 1.33E-05 1/y to 1.29E-05 1/y. The main reason for the low absolute Δ CDF value is the fact that the main risk contribution comes from human actions and the dependency modeled between them. It may, however, be a very relevant modification in a scenario without external power recovery, for example, an ELAP due to BDBEE.

Table 4.6: Risk reduction in DEC sequences with FLEX strategies and SDS

No.	Sequences	No FLEX - No SDS		W/FLEX - W/SDS		DEC-B Δ Freq (%)
		DEC-A (1/y)	DEC-B (1/y)	DEC-A (1/y)	DEC-B (1/y)	
D1	TLFW	5.75E-05	7.76E-06	8.40E-06	7.96E-07	-89.7%
D2	SGTR + SORV	3.38E-06	3.31E-06	3.38E-06	3.01E-07	-90.9%
D3	SLB + SORV	2.12E-05	1.36E-06	2.13E-05	1.25E-07	-90.8%
D4	SBLOCA/MBLOCA + RCS cooldown and depressurization failure	1.57E-06	2.62E-07	8.24E-07	1.43E-08	-94.5%
D5	SBO	7.30E-05	1.33E-07	6.29E-07	8.22E-09	-93.8%
D6	SLB + SGTR	4.80E-05	1.10E-07	4.80E-05	1.14E-08	-90.0%
D7	LCCW	1.94E-07	6.42E-08	2.56E-07	6.22E-10	-99.0%
D8	ATWS	1.90E-06	5.67E-08	1.90E-06	5.00E-08	-11.8%
D9	SGTR + SG isolation failure	1.77E-07	2.41E-08	1.77E-07	2.41E-08	-0.0%
D10	LOCA + HPSI failure	3.30E-08	1.82E-08	3.30E-08	1.81E-08	-0.3%
	Total		1.31E-05		1.35E-06	-89.7%

4.4 Comparison with Other DEC-A Improvements Included in Several Technologies

This section shows additionally some improvements implemented due to DEC-A in other reactor designs.

HPR1000 (General Nuclear System Ltd., 2020b; Xin, 2018) [[General Nuclear System Ltd., 2020a](#), [Xin, 2018](#)]:

- Containment Heat Removal System (EHR): During Total Loss of Cooling Chain (TLOCC), SBO, SBLOCA with loss of LPSI and loss of RHR or failure of recovery of RHR after LOOP, EHR limit the In-containment Refueling Water Storage Tank (IRWST) water temperature to avoid the cavitation of LPSI pumps.
- Extra Cooling System (ECS): During TLOCC/SBO the ECS is used to cool the EHR. In the case of SBO in SFP, the ECS is actuated to cool Fuel Pool Cooling and Treatment System and remove the decay heat in spent fuel.
- Diverse Actuation System: designed to deal with ATWS sequences due to signal failure. For the uncontrolled RCS level drop without Safety Injection signal, a signal is designed to actuate the Safety Injection System.
- Emergency Boration System (EBS): designed to provide sufficient boration and to ensure subcritical conditions in ATWS.
- Safety Chilled Water System (SCWS): In the case of TLOCC/SBO with LCCW/LESW, the SCWS can offer a diverse cooling chain to cool the LPSI pump motors so as to ensure the normal operation of LPSI pumps.
- Manual F&B: In the case of TLOFW or SBLOCA with failure of Medium Pressure Rapid Cooldown or total loss of RHR with failure of secondary side heat removal, manual F&B can remove the decay heat.
- Low Pressure Full Cooldown: designed to depressurize the primary rapidly for the efficient injection of LPSI. Used in case of SBLOCA with total loss of Medium Pressure Safety Injection (MPSI).
- Secondary Passive Heat Removal System (SPHRS): designed to remove the core decay heat when the secondary active heat removal system fails. In addition, the SPHRS tank is used as the water source to feed the EFWS tank in the long term of the SBO or LUHS accident. It is also used as the water source to feed the SFP during the loss of three Fuel Pool Cooling and Treatment System Trains.
- Standstill seal of RCPs to avoid inventory leakage in case of SBO.

CPR1000 (Hui et al., 2013) [[Hui et al., 2013](#)]:

- Fast backup between LPSI and Containment Spray system (CSS) pumps: In case of LOCA or TLFW fast backup between LPSI and CSS pumps is applied by adding adequate lines and valves.

- Portable equipment: in case of SBO+LUHS portable pumps can supply makeup and injection to RCS; makeup to EFWS tank; and injection to SGs. Also, in case of loss of SFP cooling, mobile makeup means is added.

Dukovany NPP, VVER-440/V213 (Matejovic et al., 2023)[[Matejovic et al., 2023](#)]:

- Installation of new pressurizer relief valve.
- Adoption of F&B strategy into EOPs.
- Seismic reinforcement of all structures, systems and components.

Belgian PWRs (Havet et al., 2023)[[Havet et al., 2023](#)]:

- EBS: during SLB with LCCW/LESW or SBLOCA with LCCW/LESW, EBS serves as a backup of HPSI.

UK-EPR (NNB Generation Company (HPC) LTD, 2017)[[NNB Generation Company \(HPC\) LTD, 2017](#)]:

- EBS: To mitigate the rod seizure, an ATWS signal is generated by the Protection System and this leads to automatic fast boration by the EBS.
- Diverse Reactor Trip: in case of failure of the automatic reactor trip, the diverse automatic reactor trip signal is activated to switch off the power supply to the Control Rod Drive Mechanisms.
- Ultimate Diesel Generators (UDGs): In case of SBO, the two UDGs will supply electrical power to the emergency supply system.
- Manual F&B: in case of total loss of the SG feedwater system when the reactor is at power.
- Automatic switchover of the cooling of the LPSI pumps: in case of SBLOCA as consequence of TLOCC with failure of the RCP seals, the automatic switchover of the cooling of the LPSI pumps to their diverse cooling system takes place.
- Diverse Automatic Activation of Safety Injection and Partial Cooldown: in case of LOCA and failure of the Safety Injection, the diverse signal activates safety injection and partial cooldown. The primary pressure drops allowing the MPSI pumps to inject into the RCS.
- Manual Activation of Fast Secondary Cooldown: in case of SBLOCA and total failure of the MPSI, the RCS pressure has to decrease quickly to permit accumulator and LPSI pump injection into the cold legs. RCS depressurisation is achieved through fast secondary cooldown.
- Cooling of the IRWST by the Containment Heat Removal System (CHRS): in case of SBLOCA and total failure of the LPSI, removal of residual heat and RCS makeup are initially ensured by the SG and injection of water into the RCS via the MPSI. In the long term, the MPSI pumps could fail due to the high temperature of the IRWST. The operator should activate the CHRS to cool the IRWST.

- **Diverse Automatic Activation of the MPSI Pumps:** in case of LOCA and failure of Safety Injection signal, the Safety Automation System ensures the automatic activation of the MPSI.
- **Manual Switchover of the Cooling for the LHSI Pumps 1 and 4 to their Diverse Cooling System:** in case of TLOCC, the manual switchover of LPSI pumps to their diverse cooling system (SCWS) permits injection by the LPSI.

Olkiluoto 3, EPR (STUK, 2019c)[[STUK, 2019c](#)]:

- Increase of the RCS cooling rate from 100 K/h to 250 K/h in case of LOCA.
- The automatic functions necessary for primary-secondary leak management were modified to ensure that under no circumstances does water without reactivity poison flow in excess from the damaged SG to the RCS.
- RCP static seal system is available for a period of 30 hours.

Krško NPP, PWR-WEC 2-loop (SNSA, 2020)[[SNSA, 2020](#)]:

- Installation of additional pumps in the bunkered building for injecting non-borated water into SGs and RCS, as well as indirectly to the containment spray system.
- Installation of additional RHR pump and dedicated heat exchangers.

Some of the improvements introduced in other technologies for DEC-A sequences are similar to those implemented in the Spanish NPPs after the Fukushima accident, including the use of portable equipment [[Hui et al., 2013](#)] and SDS [[Xin, 2018](#)]. Also, manual F&B and fast secondary cooldown that were already included in the EOPs of the Spanish NPPs prior to Fukushima accident.

Finally, it should be noted that although no other analyses of post-Fukushima improvements have been found, it is evident that these improvements, which have been implemented in most NPPs worldwide, have an impact on the risk of DEC-A/B sequences that should be considered.

Chapter 5

Application of Probabilistic Safety Assessment to Evaluate the Impact of ATF

As it is well-known, the Fukushima Daiichi NPP in Japan was unable to deal with the consequences of being hit by a giant tsunami in 2011. One of most important consequences in terms of improving the nuclear safety drawn from the lessons learned from the commented accident was the appearance of the “Accident Tolerant Fuel” (ATF) idea. Afterward, the assessment of the possible usage of new materials was proposed for the different components of the nuclear fuel and several concepts were presented, [OECD-NEA, 2018]. At the beginning, the main aim pursued with the implementation of the ATF concepts in real cores was based on obtaining an enhanced response of the plant against certain accidental conditions such as SBO or LOCA scenarios. However, it was early noted that that aim should be extended to the operational framework of the NPPS in order to compensate the additional costs of implementing ATFs. At that moment, the definition of ATF was changed to “Advanced Technology Fuels”. Therefore, ATF concept should meet the requirements focused on improving the capabilities to tackle accidental scenarios as commented, but they should also perform better than the conventional fuels under normal operating conditions allowing increasing the margins of the plant.

Within the context of the US-DOE Light-Water Reactor Sustainability program, specifically the Risk-Informed Safety Analysis pathway (LWRS-RISA), ongoing research activities are underway at the Idaho National Laboratory (INL). These activities focus on the development of tools and methodologies designed to assist the nuclear industry in quantifying the benefits associated with the introduction of ATF [Parisi et al., 2020]. Ma et al. [Ma et al., 2018] analyze two accident scenarios, SBO and BLOCA, to evaluate the risk impact brought by the *FeCrAl* and Cr-coated ATF designs for a typical PWR, using SAPHIRE PRA model to identify and group accident scenarios to be analyzed by RELAP5-3D and RAVEN risk-analysis code. As a results, they found 4% and 2% CDF reductions for the *FeCrAl* and Cr-coated designs, respectively.

Parisi et al. [Parisi et al., 2019a] analyze LOFW (7 cases) and SGTR scenarios using RELAP5-3D simulations for ATF for a Westinghouse three-loop PWR. The results show that the delay of time to core damage due to the ATF designs is less than 20 minutes for most LOFW scenarios and therefore the risk-benefit brought by the ATF designs would be very small. For the SGTR accident the sequences based on the PRA model do not reach CD within 48 hours. The report [Parisi et al., 2019b] complements the previous work by adding SBLOCA with ATWS scenarios, two types of GT scenarios, and MSLB scenarios for ATF (*FeCrAl* and Cr-coated) also using SAPHIRE PSA model and RELAP5-3D for thermal hydraulic analysis, the results show relatively small increases of the time to CD and therefore the risk-benefit brought by the ATF designs would be very small.

Parisi et al. [Parisi et al., 2020] developed risk-informed methodology (LWRS/RISA) that combines deterministic Best Estimate tools like RELAP5-3D code, and Probabilistic Risk Analysis tools like RAVEN and SAPHIRE codes. The ultimate goal of the LWRS/RISA methodology is to provide advanced safety analyzes using a risk-informed approach. The methodology was applied to three-loops pressurized water reactor (PWR) for SBO and LBLOCA scenarios and considering near-term ATF (*FeCrAl* and Cr-coated clads). The results revealed that, in the case of SBO, there is only a marginal increase in coping time. However, for the LBLOCA scenario, the most notable differences are observed in cases with very low probability. Notably, ATF demonstrated superior performance, particularly when considering hydrogen production. Finally, Shah et al. [Shah et al., 2021] uses dynamic probabilistic risk assessment (DPRA)-based (RAVEN coupled with RELAP5 and FAM) to quantify the ATF (Silicon carbide, *SiC*; and Zr-based) benefits for a generic 1000 MWe PWR. The results show that *SiC* is likely to provide additional coping time compared to other ATF candidates. Subsequently, INL has also replicated these analysis for Boiling Water Reactors [Ma et al., 2020a, Zhang et al., 2021b].

In parallel, the NRC has developed a project plan to for the licensing of ATF, including consideration of fuel fabrication, fresh fuel transport, in-reactor requirements, and spent fuel storage and transportation, in this way, considering the complete fuel cycle [NRC, 2021]. To complement this ATF project plan a roadmap was and issued in 2024 [NRC, 2024], Figure 5.1.

In the present work, the benefits associated with the implementation of the evolutionary ATF (e-ATF) cladding designs, Cr-Coated zirconium and *FeCrAl* are analyzed. In order to perform this analysis, the increase in available times to cope with accidental scenarios in a PSA model are considered and supported with public information and thermal-hydraulic calculations performed using an in-house modified version of TRACE. By doing so, it was possible to consider the Cr-Coated zirconium and *FeCrAl* performances under high temperatures, especially, with regard to the oxidation-related phenomena.

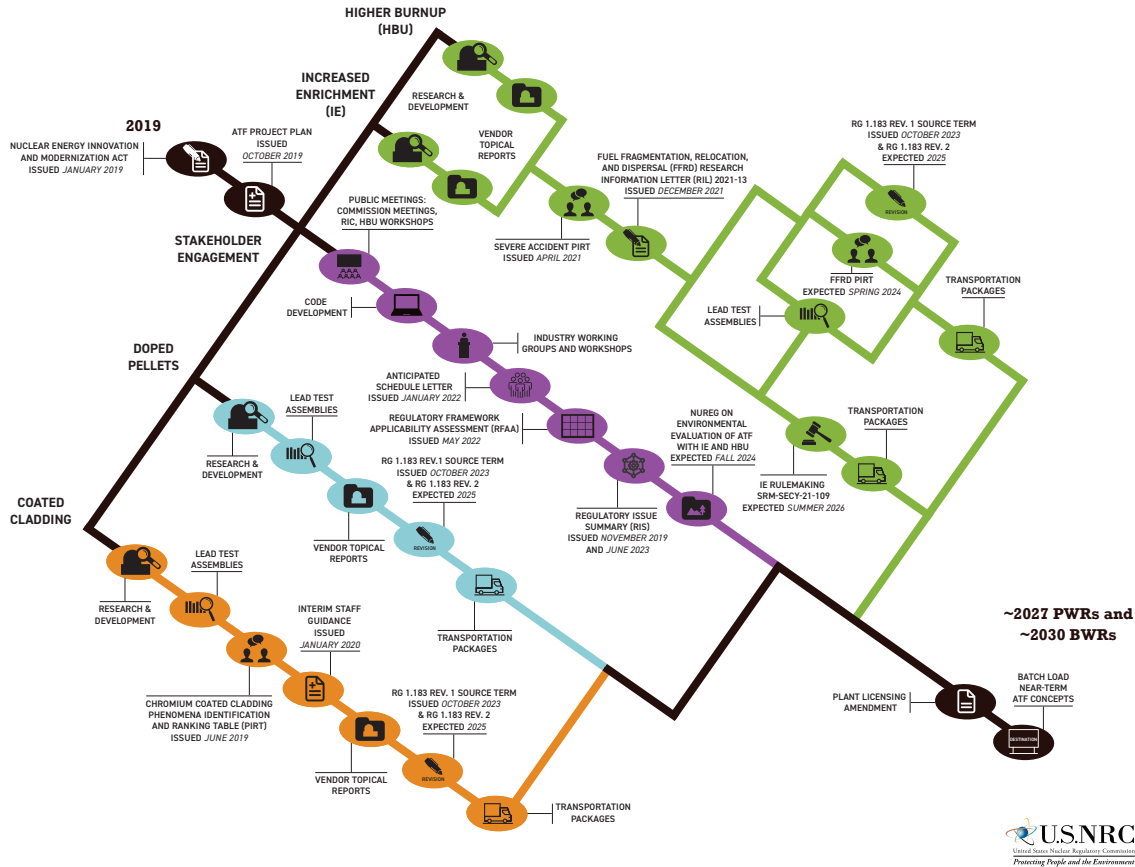


Figure 5.1: ATF roadmap to readiness [NRC, 2024]

The present chapter is organized as follows: First, Section 5.1 is devoted to describing briefly the main characteristics of the e-ATF design focusing on the usage of Cr-Coated zirconium and $FeCrAl$ cladding for the simulation. The impact of the ATF implementation on the L1PSA model and the results are depicted in Section 5.2. Finally, the conclusions drawn from this work are presented at Section 6.4.

5.1 Advanced Technology Fuels

In this section, a brief review of the currently relevant ATF concepts is performed. According to the NRC: “ATF is a set of new technologies that have the potential to enhance safety at NPPs by offering better performance during normal operation, transient conditions, and accident scenarios”. At the very beginning, main efforts were focused on materials tolerant to LOCA and SBO conditions or, at least, with the capability to increase the available time to deal with severe conditions. Currently, ATF should provide with sufficient advantages also under operation conditions to justify the increased expenses (Table 5.1).

Table 5.1: Evolution of the main objectives pursued regarding ATF designs

Initial approach	Current approach
Related to safety issues	Related to safety & operational issues
ATF: Accident Tolerant Fuels focused on: <ul style="list-style-type: none"> • Enhancing fuel rod behavior under accident conditions. • Increasing of safety margins under accident conditions. • Mainly devoted to deal with LOCA and SBO accidents. 	ATF: Advanced Technology Fuel aiming at: <ul style="list-style-type: none"> • Enhancing fuel rod behavior under normal, DBA and BDBA. • Increasing safety margins considering also normal operation conditions. • Reaching higher enrichments and burnups. • Improving fuel cycle economics.

Also, the ATF concepts can be classified as evolutionary and revolutionary (Table 5.2) in accordance with the technology readiness and the possible term of its implementation [OECD-NEA, 2018]. The ATF concepts with a higher technology readiness level (TRL) are the ones referred to as *evolutionary* because those designs are based on improving the existing fuel rods without performing major changes in materials or fuel rod design. Those are the advanced stainless steels called *FeCrAl* or the chromium coatings (for conventional zirconium cladding) for the cladding combined with fuel pellets made with chromium doped UO_2 or conventional UO_2 . Currently, one of the main challenges regarding the implementation of those e-ATF designs in actual NPPs is based on achieving a complete characterization of their thermomechanical performance under high-temperature conditions allowing to eventually defining the corresponding failure criteria for the fuel rods along with the proper values for the acceptance criteria. The main characteristics of the e-ATF are:

- Good performance has been demonstrated by *FeCrAl* cladding and Cr-coatings under normal operation conditions.
- *FeCrAl* cladding and Cr-Coatings are capable to reduce significantly the impact of the HT oxidation reaction on the oxide thickness, hydrogen generation and energy source term in comparison with zircalloy cladding.
- It has been demonstrated that coatings has no impact on mechanical Zr properties. In fact, as Cr is a very strong material, the Cr-coating could provide additional protection against debris or fretting.
- *FeCrAl* cladding presents a better mechanical behavior, but their penalty come from the high neutron capture cross-section comparing to Zr alloys.

On the other hand, *revolutionary* ATF concepts are based on completely new materials or even completely new fuel rod designs, and are consequently associated with a lower TRL [OECD-NEA, 2018]. In this sense, the most remarkable materials identified on the fuel side are the uranium silicide (U_3Si_2) or uranium nitride (UN_x) due to their highly improved thermal conductivity compared to the conventional UO_2 pellets and, on the clad side, the *SiC* for its much improved corrosion resistance, especially at high temperatures, and a higher melting point than conventional zirconium cladding. The main features of the revolutionary ATF are:

- Very promising results have been achieved regarding advanced fuel materials such as U_3Si_2 .
- UN_x fuels has shown good behavior, but the manufacturing process base on separating $N15$ from natural N is still too expensive.
- Water and steam corrosion resistance and the fact that U_3Si_2 and UN fuels react with water and steam at operating conditions and high temperatures are the most important drawbacks of those materials.
- It has been demonstrated the SiC cladding properties are quite interesting, but commercial production still presents great difficulties.
- Further investigation and development is needed regarding the possible implementation of the revolutionary materials in nuclear fuel rods.

Table 5.2: Classification of ATF fuel rods

ATF	CLADDINGS	FUELS
EVOLUTIONARY (Near-term)	New Zr alloys Coated Zr cladding Advanced steels	Doped UO_2 High Thermal Conductivity UO_2 Dual-cooled fuel
REVOLUTIONARY (Long-Term)	Refractory Metals (Lined Mo-alloy) SiC Composite	High density fuels (Nitride, Carbide, Silicide, Metallic Fuels) Microencapsulated fuels (TRISO-SiC composite pellets)

In this work, the benefits obtained from the use of Cr-Coated zirconium and $FeCrAl$ cladding are evaluated in terms of the reduction of the CDF associated with accidental sequences (e.g. LOCA and SBO events). For this purpose, the acceptance criteria applied in the thermal-hydraulic simulations corresponding to the accidental sequences is the temperature at which the eutectic formation between Cr and Zr occurs (1603.7 K) for Cr-Coated zirconium cladding, and the melting temperature in the case of $FeCrAl$ cladding (1773 K) [Zhang et al., 2021a, Geelhood and Luscher, 2019]. Other authors have applied slightly different values for the commented acceptance criteria (e.g. 1804 K for both cases Cr-Coated zirconium and $FeCrAl$ cladding) [IAEA, 2020a, Gurgun and Shirvan, 2018].

Therefore, ATF cladding increases the time margins for two reasons, the rate of cladding temperature increase is lower when ATF cladding is present; and second, as mentioned above, the temperature acceptance criteria are greater for ATF cladding (Figure 5.2).

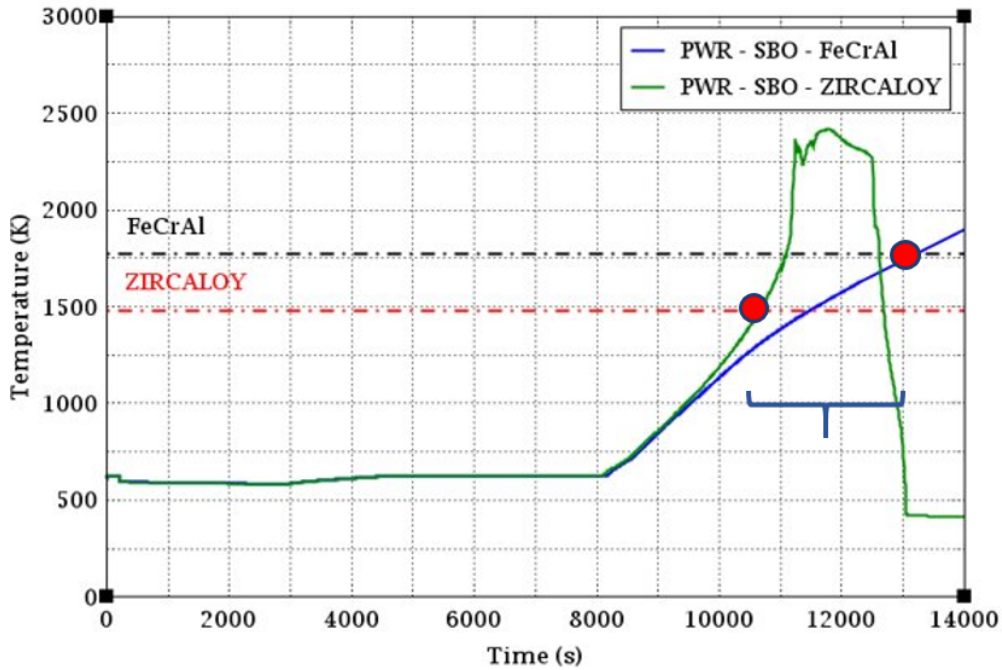


Figure 5.2: PCT evolution under an SBO in a PWR (zirconium vs *FeCrAl*, MELCOR code) [Doncel et al., 2024]

5.2 ATF Impact on L1PSA

The benefits of ATF on the DSA are quite clear, modifying the success criteria and thus increasing the safety margin for several scenarios. Nevertheless, their impact on the PSA and how to evaluate their implementation are still under discussion. The present paper proposes the evaluation of ATF by means of the increase in the available time to perform several human actions included in the L1PSA model.

ATF implementation improves the ability of fuel rods to withstand the very high temperatures reached under certain accidental due to their improved performance and the possibility to consider a higher value of the cladding temperature acceptance criteria (1477 K for the conventional zirconium cladding). Considering this, the time to reach a core damage condition must be larger for ATF cases, increasing the time to perform the actions required to avoid this condition. Accordingly, the ATF allow to enhancing the time window of several human actions of the PSA model, and decreasing their failure probability.

The methodology followed to evaluate the reduction of the HEP begin with the identification of the HFE which led to core damage in the L1PSA model and are whose failure probabilities are closely related to available time. From 38 type-3 human actions, eight of them meet these criteria. These actions are mainly associated with RCS depressurization, HPSI and LPSI startup/control, recovery of the AC or F&B performance.

In order to obtain the time increment, first, a literature review was performed [IAEA, 2020a, Ma et al., 2018, Parisi et al., 2019b, Parisi et al., 2019a, Parisi et al., 2020, Shah et al., 2021]. In addition, to compare and confirm the data collected, thermal-hydraulic simulations were performed utilizing the TRACE and MELCOR codes. The increase of available times is shown in Table 5.3. These data is used to re-quantify the HEPs, the results are shown in Table 5.4. By using the new HEPs, the CDF is quantified for each ET and the full model, for Cr-Coated and *FeCrAl* cladding. The results of this analysis show a risk reduction in specific scenarios, Table 5.5. The methodology used can be summarized as follows:

- Identification of the HFE which can reduce their HEP due to the incorporation of ATFs:
 - 8 HFE meet these criteria.
 - Mainly associated with RCS depressurization, HPSI and LPSI startup/control, recovery of the AC or F&B performance.
- In order to obtain the time available increment:
 - First, a literature review was performed.
 - In addition, to compare and confirm the data collected, thermo-hydraulic simulations were performed utilizing the TRACE and MELCOR codes.
 - New acceptance criteria for ATF cladding (*there is no consensus*): Cr-coated (1605.15 K); *FeCrAl* (1773 K)
- These data is used to re-quantify the HEPs.
- By using the new HEPs, the CDF is quantified for each ET and the full model.

Table 5.3: Increase in available time included in PSA model

Time to CD (min)	Increase in time (min)	
	Cr-Coated	<i>FeCrAl</i>
50-80	2	6
130-160	5	10
300-350	10	21
500-550	12	50

Table 5.4: Impact of ATF (Cr-Coated and *FeCrAl*) implementation on HEPs

Description	Probability Zry	Probability Cr-Coated	Probability <i>FeCrAl</i>
Operator failure to support HPSI automatic startup (MBLOCA)	2.00E-02	1.12E-03	2.35E-04
Operator failure to perform F&B	2.00E-03	1.55E-03	1.10E-03
Operator failure to close circuit breakers after AC recovery (LOFW case)	1.00E-03	9.91E-04	9.74E-04
Operator failure to open Boron Injection tank by-pass valve	1.00E-03	7.30E-04	5.94E-04
Operator failure to support LPSI startup	1.00E-03	7.75E-04	7.73E-04
Operator failure to open/control SG-PORVs and SD valves (MBLOCA)	1.00E-03	7.74E-04	7.70E-04
Operator failure to close circuit breakers after AC recovery (>480 gpm RCP leakage)	1.00E-03	9.77E-04	9.53E-04
Operator failure to support HPSI automatic startup (SGTR)	1.00E-04	9.85E-05	9.55E-05
Operator failure to open Boron Injection tank by-pass valve (SBLOCA and SGTR)	1.00E-04	9.76E-05	9.52E-05
Operator failure to support HPSI automatic startup (SBLOCA and MSLB)	1.00E-04	9.63E-05	9.25E-05
Operator failure to open/control SG-PORVs and SD valves (SBLOCA)	1.00E-04	9.67E-05	9.33E-05
Operator failure to open SG-PORVs and SD valves (SBO and loss of CCWS A train)	1.00E-04	9.67E-05	9.33E-05
Operation failure to open PZR-PORVs	1.00E-04	9.67E-05	9.33E-05
Operator failure to close circuit breakers after AC recovery (<480 gpm RCP leakage)	1.00E-04	9.94E-05	9.87E-05
Operator failure to close circuit breakers after AC recovery (battery depletion)	1.00E-04	9.96E-05	9.91E-05
Operator failure on HPSI startup after SBO	1.00E-05	1.00E-05	1.00E-05

Table 5.5: CDF variation caused by ATF implementation (Cr-Coated and *FeCrAl*)

IE	CDF w/o ATF (1/y)	CDF w/ Cr-Coated (1/y)	Δ CDF (%)	CDF w/ <i>FeCrAl</i> (1/y)	Δ CDF (%)
LBLOCA	1.15E-08	1.15E-08	0.00	1.15E-08	0.00
MBLOCA	3.81E-08	3.81E-08	-0.05	3.80E-08	-0.05
SBLOCA	1.01E-07	1.01E-07	0.00	1.01E-07	0.00
GT	1.98E-07	1.76E-07	-11.11	1.54E-07	-22.22
LC	1.51E-08	1.35E-08	-10.60	1.19E-08	-21.19
MSLB-DS	2.42E-08	2.42E-08	-0.04	2.42E-08	-0.04
MSLB-US	2.70E-09	2.70E-09	0.00	2.70E-09	0.00
LCWA	7.06E-10	7.06E-10	0.00	7.06E-10	0.00
SGTR	1.51E-08	1.51E-08	-0.13	1.51E-08	-0.13
LNSW	5.85E-11	5.20E-11	-11.11	4.54E-11	-22.39
LDC-A	1.29E-08	1.27E-08	-1.55	1.26E-08	-2.33
LDC-B	9.86E-09	9.73E-09	-1.32	9.59E-09	-2.74
LOOP	1.14E-07	1.12E-07	-1.75	1.11E-07	-2.63

ATF implementation causes a risk reduction ($\sim 10\%$ Cr-Coated, $\sim 20\%$ *FeCrAl*) in GT, LC and LNSW scenarios; mainly because of the new values of HEP to perform F&B (Table 5.4). In addition, other scenarios have experienced a small CDF reduction ($< 1\%$ Cr-Coated, $< 3\%$ *FeCrAl*) as well. On the other hand, several scenarios show non-significant CDF reduction which is an expected result due to the smaller difference between the previous HEPs and the new ones, and the lack of presence of critical HFE which lead to CD condition in these scenarios. It is important to remark that the obtained results for both ATF alternatives (Cr-Coated and *FeCrAl*) considered in the present study are similar by order of magnitude to the ones found in the literature [Ma et al., 2018, Parisi et al., 2019b, Parisi et al., 2019a, Parisi et al., 2020, Shah et al., 2021].

After the analysis of each accidental scenarios covered by the L1PSA model, an analysis of full model was performed in order to quantify the full impact of each ATF in the PSA model, Table 5.6.

Table 5.6: Model-wide CDF variation caused by the ATF application

CDF Zry (1/y)	CDF Cr-Coated (1/y)	Δ CDF Cr-Coated (%)	CDF <i>FeCrAl</i> (1/y)	Δ CDF <i>FeCrAl</i> (%)
5.44E-07	5.18E-07	-4.82	4.92E-07	-9.56

Chapter 6

Conclusions

The SPAR-CSN models (generic and specific) are valuable tools to understand and evaluate risk associated with operation of Spanish NPPs. The standardized construction of these models in terms of modeling assumptions, level of detail, data, HRA methodology, etc., provides a consistent approach from which to evaluate individual plant risk impacts as well as industry-wide issues. The aim of CSN is that these SPAR-CSN models contribute to a better understanding of the main risk drivers in Spanish NPPs and be used as a tool for the different PSA applications that are currently implemented at CSN (e.g., precursor analysis, prioritization in inspection and oversight tasks, assessment of inspection findings). With that aim, the intent is to extend the present task to cover other Spanish NPPs, and improve the model in detail, accuracy and breadth of coverage to meet the emerging needs of CSN and to reflect advancements in the state of the art of the PSA technology.

The addition of post-Fukushima improvements highly reduces the relative risk. In the case of the implementation of FLEX strategies, a decrease of the CDF at power for internal events is observed and indicates that these have a positive impact on the plant safety not only for BDBEE but also for other accidental conditions. For DEC-A sequences, a novel methodology has been developed to identify them, using this method, it has been possible to identify the DEC-A sequences with high relative risk by comparing their frequencies with their corresponding DEC-B sequences. It is important to note that the results obtained in this work are only valid for internal events at nominal power for a generic 3-loop PWR-WEC design. It is highly recommended to extend the study of the impact of FLEX strategies to other different and/or more complex scenarios, such as LPSD conditions; floods; fires and external events.

Although a minor reduction in the CDF or coping time under accidental conditions has been obtained in the present work for ATF Cr-Coated and *FeCrAl* claddings, the reader should be aware of the fact that these results represent a partial assessment of the whole impact of the ATF implementation on NPPs safety. In fact, ATF implementation could also play an important role in the extension of the current limitation in the discharge-burnup due to their improved performance, specially, regarding the oxidation resistance under normal

operating conditions. In addition, revolutionary materials, such as SiC claddings, seem to be able to present an even better behavior, with CDF reductions of up to 40%. Therefore, further investigation is recommended regarding two main lines. Firstly, the characterization of the advantages in nuclear safety and economics associated with the ATF implementation in current NPPs should be completed and, secondly, the assessment of the possible advantages of reaching higher burnup conditions using e-ATFs materials.

The specific conclusions are summarized in the following Sections 6.1, 6.2, 6.3, 6.4.

6.1 SPAR-CSN Conclusions

The main conclusions drawn up to now are summarized within the following points:

- The methodology delineated for the development of SPAR-CSN models has demonstrated the feasibility to standardize ETs, success criteria, FTs and human actions between the Spanish NPPs tackled in this project.
- Standardization features allow to identify and model the specific design differences between the plants and departures from the generic model; some effort is being invested for that purpose within the current project.
- The method has allowed to identify and to assess relevant differences between the Spanish NPPs PSA models. These differences regard ETs (e.g., transfers between ETs, requirements for the containment safety systems, classification of LOCA break size categories), FTs (e.g., system operating modes and main failures distribution in the FT, transfer gates use, impossible failure combinations removal techniques, etc.) and modeling hypotheses, as well as Human reliability analysis results and Data sources for equipment failure probabilities.
- All the identified and relevant differences, either regarding modeling decisions or adopted hypotheses, have been evaluated by CSN PSA experts, and a common practice established based on their experience. In case of large differences, the scope, premises and practices of the SPAR-USNRC have been used as reference for the generic SPAR-CSN model.
- The overall CDF value obtained from the SPAR-CSN generic model, as well as the related main MCS, reasonably agree with those obtained from other classic and SPAR PSA models.
- The next steps of the SPAR-CSN project include a comparison with other SPAR models and with the Spanish NPP PSA models. On this regard, the project researchers intend to determine the safety significance of a set of inspection findings using the SPAR-CSN generic model, which could be compared to the results obtained using other models.

6.2 FLEX Conclusions

The main conclusions drawn from the study are summarized in the following points:

- New ETs have been created in the generic SPAR-CSN model to model the post-Fukushima FLEX strategies, including human actions and equipment, making it possible to quantify the risk reduction.
- The quantification of the new ETs, including the FLEX strategies, shows a decrease in the CDF close to 90% (i.e., an order of magnitude). This reduction is directly related to the new or modified headers related to these FLEX strategies implemented in the SPAR-CSN generic model with the associated data.
- The sensitivity study of the FLEX strategies has determined that, for L1PSA, the strategies related to the injection of coolant into the SGs and RWST inventory control have the greatest contribution in terms of relative risk reduction.
- The incorporation of FLEX strategies in the L1PSA models is recommended, including uncertainty and sensitivity analysis.

6.3 DEC-A Conclusions

The main conclusions drawn up to now are summarized in the following points:

- The present methodology, developed in the framework of the SPAR-CSN project, allows to standardize the identification of DEC-A sequences in L1PSA models and quantify their risk by representative FoM ($DEC-B_{freq}$ and $CCDP_{DEC}$).
- The application of the methodology to the SPAR-CSN model has revealed 15 DEC-A sequences (10 from the PSA model and 5 by expert judgment) with quantitative relevance. Two sequences from this list are not included in other DEC lists of other countries.
- The implementation of SDS has a high impact in SBO and LCCW sequences, reducing the expected CDF.
- The incorporation of FLEX strategies has a high impact on most DEC-A sequences decreasing their own frequency (**D1**, **D4**, **D5** and **D7**), and considerably decreasing the frequency of their respective DEC-B sequences. The total frequency of DEC-B sequences is reduced by almost 90%.
- Strategies related with alternatives with LP feedwater (FA-1/2) and alternative RWST inventory control (FA-6/7) are the most important FLEX strategies to reduce the risk of DEC-A sequences

6.4 ATF Conclusions

The main conclusions are summarized in the following points:

- ATF represents an alternative to deal with accidental scenarios such as SBO or LOCA, due to the increase in the available time to perform actions before fuel reaches its failure criterion (melting or eutectics production, depends on ATF). In order to analyze their potential, a safety assessment of the evolutionary designs, Cr-Coated and *FeCrAl* cladding, have been performed using a L1PSA model.
- With the ATF implementation, the L1PSA CDF decreases near to 5% for Cr-Coated cladding and 10% for *FeCrAl* cladding.
- ATF technology represents an interesting alternative to conventional Zry. Nevertheless, the lack of a consensus about the failure criteria between ATF technologies and the absence of enough experimental data to support the models may lead to changes in time to reach CD condition, not considered in the present work.

References

- [ANAV, 2013] ANAV (2013). Vandellós II NPP information committee. Technical report, Ascó-Vandellós Nuclear Association.
- [ASME/ANS, 2015] ASME/ANS (2015). Standard for level 1/large early release frequency probabilistic risk assessment for nuclear power plant applications, ASME/ANS RA-Sa-2009.
- [Bisio et al., 2020] Bisio, R., Fernandes, A., and Blackett, C. (2020). A framework to analyze human performance outside the control room. In *Proceedings of the 30th European Safety and Reliability Conference and the 15th Probabilistic Safety Assessment and Management Conference*, Venice, Italy.
- [Breunig, 2017] Breunig, J. (2017). FLEX – making plants safer than ever before. <https://www.xceed-eng.com/flex-nuclear-safety/>.
- [Caro, 2012] Caro, R. (2012). Extensive damage emergency response. In *38th Annual Meeting of the Spanish Nuclear Society*, Cáceres, Spain. (In Spanish).
- [CNSC, 2014a] CNSC (2014a). Design extension conditions for nuclear power plants, DIS-14-01.
- [CNSC, 2014b] CNSC (2014b). Design of reactor facilities, REGDOC-2.5.2.
- [CNSC, 2015] CNSC (2015). Accident management regulatory document REGDOC-2.3.2, version 2. Technical report, Canadian Nuclear Safety Commission (CNSC).
- [Courtin et al., 2023] Courtin, S., Queral, C., and García-Herranz, A. (2023). Risk reduction by means of FLEX strategies in pressurized water reactors. *Nuclear Engineering and Design*, 414.
- [Courtin et al., 2021] Courtin, S., Queral, C., Paris, C., and Iglesias, R. (2021). Application of the SPAR-H human reliability methodology to FLEX strategies. In *46th Annual Meeting of the Spanish Nuclear Society*, Granada, Spain (In Spanish).
- [CSN, 2011a] CSN (2011a). Complementary analysis and improvements to be implemented as a result of the stress tests performed by Spanish nuclear power plants, ITC-1 (In Spanish).
- [CSN, 2011b] CSN (2011b). Supplementary technical instructions on the loss of large areas, ITC-2 (In Spanish).
- [CSN, 2013a] CSN (2013a). Inspection report, CSN/AIN/AS0/13/1013 (In Spanish).

- [CSN, 2013b] CSN (2013b). Inspection report, CSN/AIN/VA2/13/829 (In Spanish).
- [CSN, 2014] CSN (2014). Inspection report, CSN/AIN/VA2/14/869 (In Spanish).
- [CSN, 2015a] CSN (2015a). Inspection report, CSN/AIN/AL0/15/1060 (In Spanish).
- [CSN, 2015b] CSN (2015b). Nuclear Safety Council Instruction IS-37: on the analysis of design basis accidents at nuclear power plants.
- [CSN, 2016] CSN (2016). Inspection report, CSN/AIN/AL0/16/1086 (In Spanish).
- [Degonish, 2022] Degonish, M. (2022). FLEX equipment data collection and analysis, PWROG-18042-NP, Rev. 1.
- [Doncel et al., 2024] Doncel, N., Martínez, L., Aragón, P., Feria, F., Herranz, L., Queral, C., Fernandez-Cosials, K., Ruiz-Hervias, J., Cristobal-Beneyto, M., Castro, E., Sanchez-Torrijos, J., and Soler, A. (2024). R&D in advanced technology fuels (ATFs) in Spain. *Nuclear Engineering and Design*, 424:113246.
- [Duffey, 2019] Duffey, R. B. (2019). The risk of extended power loss and the probability of emergency restoration for severe events and nuclear accidents. *Journal of Nuclear Engineering and Radiation Science*, 5.
- [Duke Energy, 2015] Duke Energy (2015). H. B. Robinson, compliance letter and final integrated plan in response to commission order modifying licenses with regard to requirements for mitigation strategies for beyond-design-basis external events (order number EA-12-049).
- [Entergy, 2021] Entergy (2021). Arkansas Nuclear One, Unit 1 and Unit 2; license amendment request.
- [EPRI, 2014] EPRI (2014). Incorporating flexible mitigation strategies into PRA models: Phase 1: Gap analysis and early lessons learned. 3002003151.
- [EPRI, 2018] EPRI (2018). Human reliability analysis (HRA) for diverse and flexible mitigation strategies (FLEX) and use of portable equipment. 3002013018.
- [Fernandez-Cosials et al., 2020] Fernandez-Cosials, K., Queral, C., Robledo, F., and Sánchez-Perea, M. (2020). Analysis of a LBLOCA with FLEX actuations in a PWR-W. *Nuclear Engineering and Design*, 370.
- [Fujioka et al., 2017] Fujioka, B., Hirokawa, N., and Taniguchi, D. (2017). Probabilistic assessment of countermeasures for loss of ultimate heat sink to spent fuel pool. In *25th International Conference on Nuclear Engineering*, Shanghai, China.
- [Geelhood and Luscher, 2019] Geelhood, K. and Luscher, W. (2019). Degradation and failure phenomena of accident tolerant fuel concepts: Chromium coated zirconium alloy cladding, PNNL-28437. *Pacific Northwest National Laboratory*.
- [General Nuclear System Ltd., 2020a] General Nuclear System Ltd. (2020a). UK HPR1000 GDA: Pre-construction safety report, chapter 13 design extension conditions and severe accident analysis.
- [General Nuclear System Ltd., 2020b] General Nuclear System Ltd. (2020b). UK HPR1000

- GDA: Pre-construction safety report, chapter 4 general safety and design principles.
- [Gentillon et al., 2020] Gentillon, C., Atwood, C., Mack, A., and Ma, Z. (2020). *Evaluation of Weakly Informed Priors for FLEX Data, INL/EXT-20-58327*. Idaho National Laboratory.
- [Gertman et al., 2005] Gertman, D., Blackman, H., Marble, J., Byers, J., and Smith, C. (2005). The SPAR-H human reliability analysis method. Technical Report NUREG/CR-6883, US Nuclear Regulatory Commission, Washington DC, USA.
- [Gil-Rodríguez et al., 2016] Gil-Rodríguez, M., Martínez, P., Tanarro, A., Fornós, J., and Hernández, H. (2016). Modification to SAMGs at ANAV, revision 2. In *42th Annual Meeting of the Spanish Nuclear Society*, Santander, Spain (In Spanish).
- [Gjorgiev et al., 2017] Gjorgiev, B., Volkanovski, A., and Sansavini, G. (2017). Improving nuclear power plant safety through independent water storage systems. *Nuclear Engineering and Design*, 323:8–15.
- [Gurgen and Shirvan, 2018] Gurgen, A. and Shirvan, K. (2018). Estimation of coping time in pressurized water reactors for near term accident tolerant fuel claddings. *Nuclear Science and Engineering*, 337:38–50.
- [Hakobyan and Nierode, 2017] Hakobyan, A. and Nierode, C. (2017). Implementation of FLEX strategies in Surry PRA. In *International Topical Meeting on Probabilistic Safety Assessment and Analysis, PSA 2017*, volume 2, pages 1159–1164, Pittsburgh, Pennsylvania, USA.
- [Harwood et al., 2017] Harwood, C., Viktorov, A., and Lei, Q. (2017). *Design Extension Conditions: Concept and its Application to Operating Reactor in Canada*, pages 263–275. Springer Singapore.
- [Hasan et al., 2018] Hasan, T., Sengee, S., Pamungkas, S., Bello, M., and H-K, L. (2018). HEPs calculation for FLEX strategies in response to APR1400 extended SBO for risk-informed decision making. In *Transactions of the Korean Nuclear Society Autumn Meeting*, Yeosu, South Korea.
- [Havet et al., 2023] Havet, M., Haedens, M., Schneidesch, C., and Zhang, J. (2023). A commensurate industry approach to DEC A safety analysis. *Nuclear Engineering and Design*, 407.
- [Hernández et al., 2015] Hernández, H., Portela, J. M., Morena, J., and Ibor, J. (2015). Program for updating operating procedures and guidelines as a result of the post-Fukushima actions at ANAV. In *41th Annual Meeting of the Spanish Nuclear Society*, La Coruña, Spain (In Spanish).
- [Hui et al., 2013] Hui, T., Gong, A.-C., and Feng, Z. (2013). Study on modifications of CPR1000 nuclear plants for design extension conditions.
- [IAEA, 2016] IAEA (2016). Safety of nuclear power plants: Design. Specific safety requirements No. SSR-2/1 (Rev. 1).
- [IAEA, 2020a] IAEA (2020a). Analysis of options and experimental examination of fuels

- for water cooled reactors with increased accident tolerance (ACTOF). Number IAEA-TECDOC-1921.
- [IAEA, 2020b] IAEA (2020b). Current approaches to the analysis of design extension conditions with core melting for new nuclear power plants, IAEA-TECDOC-1982.
- [Islam and Lim, 2017] Islam, M. G. S. and Lim, H.-K. (2017). Comparative study of FLEX strategies for Extended Station Blackout (SBO) using PRA. *International Journal of Engineering Research & Technology*, 6:158–164.
- [Jabbari et al., 2020] Jabbari, M., Hadad, K., and Pirouzmand, A. (2020). The safety improvement of VVER-1000 NPP against SBO accident using portable air-cooled diesel generator. *Annals of Nuclear Energy*, 139.
- [Jung et al., 2021] Jung, M. S., Jung, D. H., and Lim, H. K. (2021). A review on the scope of accident management in the accident management plan of nuclear power plant.
- [Kang and Seong, 2020] Kang, S. and Seong, P. (2020). Performance shaping factor taxonomy for human reliability analysis on mitigating nuclear power plant accidents caused by extreme external hazards. *Ann. Nucl. Energy*, 145:107533.
- [Kim and No, 2019] Kim, B. and No, H. C. (2019). Application of Wilks’ formula and concept of state change time to integrated deterministic and probabilistic safety assessment for evaluation of the safety margin of DEC accidents. *Nuclear Engineering and Design*, 352.
- [Kim, 2019] Kim, J. (2019). Status on SAMG and FLEX/MACST HRA methods in KAERI. In *U.S.NRC HRA Data Workshop*. U.S.NRC HRA Data Workshop.
- [Kim et al., 2018] Kim, J., Jung, W., and Park, J. (2018). Human reliability analysis of the flex/macst actions deploying portable equipment. In *Transactions of the Korean Nuclear Society Autumn Meeting*, Yeosu, South Korea.
- [Kim et al., 2016] Kim, J., Park, S. Y., Ahn, K. I., and Yang, J. E. (2016). IROCS: Integrated accident management framework for coping with beyond-design-basis external events. *Nuclear Engineering and Design*, 298:1–13.
- [Kim and Kim, 2021] Kim, J.-S. and Kim, M.-C. (2021). Development of level 1 probabilistic safety assessment incorporating portable equipment. In *Transactions of the Korean Nuclear Society Autumn Meeting*, Virtual Meeting.
- [Kim et al., 2021] Kim, J.-S., Kim, M.-C., Son, K., Heo, G., Park, G., Kim, J., Hwang, M., and Jae, M. (2021). Modeling portable equipment in probabilistic safety assessment of nuclear power plants. In *International Topical Meeting on Probabilistic Safety Assessment and Analysis*, Virtual Meeting.
- [Kral, 2019] Kral, P. (2019). Analysis and implementation of design extension conditions (DEC).
- [Lane, 2017] Lane, J. (2017). FLEX data needs for PRA and SPAR models. (Issue ADAMS Accession Number ML17320A232). Office of Nuclear Regulatory Research.
- [Laumann and Rasmussen, 2016] Laumann, K. and Rasmussen, M. (2016). Suggested im-

- provements to the definitions of Standardized Plant Analysis of Risk-Human Reliability Analysis (SPAR-H) performance shaping factors, their levels and multipliers and the nominal tasks. *Reliability Engineering & System Safety*, 145:287–300.
- [Lim, 2018] Lim, H.-K. (2018). A conceptual comparative study of FLEX strategies to cope with extended station blackout (SBO). In *14th Probabilistic Safety Assessment and Management PSAM 14*, Los Angeles, California, USA.
- [Linthicum and Powell, 2022] Linthicum, R. and Powell, M. (2022). FLEX equipment reliability data. In *16th Probabilistic Safety Assessment & Management conference*, Honolulu, Hawaii, USA.
- [Liu et al., 2016] Liu, J., He, X., Tian, X., and Liu, K. (2016). Application of SPAR-H method in level 2 PSA-HRA. In *13th Probabilistic Safety Assessment and Management*, Seoul, South Korea.
- [Liu et al., 2019] Liu, K., He, X., Tian, X., and Liu, X. (2019). A study on the method of fire human reliability analysis. In *International Conference on Nuclear Engineering, Proceedings, ICONE-27*, volume 2019-May, pages 1–5, Tsukuba, Ibaraki, Japan.
- [Liu et al., 2018] Liu, K., Tian, X., He, X., and Liu, X. (2018). Application of SPAR-H method in fire human reliability analysis. In *14th Probabilistic Safety Assessment and Management*, Los Angeles, California, USA.
- [Liu et al., 2020] Liu, P., Qiu, Y., Hu, J., Tong, J., Zhao, J., and Li, Z. (2020). Expert judgments for performance shaping factors’ multiplier design in human reliability analysis. *Reliability Engineering & System Safety*, 194.
- [Liu et al., 2021] Liu, P., Zou, Y., Wang, W., Zhang, L., Qing, T., Zheng, T., and Ding, Q. (2021). A study on assigning performance shaping factors of the SPAR-H method for adequacy human reliability analysis of nuclear power plants. *International Journal of Industrial Ergonomics*, 81.
- [Ma et al., 2021] Ma, C., Ma, Y., and Du, J. (2021). Study on PSA application in VVER NPP design extension condition identification.
- [Ma et al., 2019] Ma, Z., Parisi, C., Davis, C., Park, J., Boring, R., and Zhang, H. (2019). Risk-informed analysis for an enhanced resilient PWR with ATF, FLEX, and passive cooling, INL/EXT-19-53556.
- [Ma et al., 2018] Ma, Z., Parisi, C., Zhang, H., Mandelli, D., Blakely, C., Yu, J., Youngblood, R., and Anderson, N. (2018). Plant-level scenario-based risk analysis for enhanced resilient PWR - SBO and LBLOCA. Technical Report INL/EXT-18-51436, Light Water Reactor Sustainability Program, USA.
- [Ma et al., 2020a] Ma, Z., Zhang, S., Zhang, H., Park, J., Yu, J., Blakely, C., Ulrich, T., and Boring, R. (2020a). Risk-informed ATF and FLEX analysis for an enhanced resilient BWR under design-basis and beyond-design-basis accidents. Technical Report INL/EXT-20-59906, Light Water Reactor Sustainability Program, USA.
- [Ma et al., 2020b] Ma, Z., Zhang, S., Zhang, H., Park, J., Yu, J., Blakely, C., Ulrich, T., and

- Boring, R. (2020b). Risk-informed ATF and FLEX analysis for an enhanced resilient BWR under design-basis and beyond-design-basis accidents, INL/EXT-20-59906.
- [Matejovic et al., 2023] Matejovic, P., Barnak, M., Bachraty, M., Vranka, L., Spacek, J., and Hofmann, E. (2023). Backfitting of Dukovany NPP for design extension conditions. *Nuclear Engineering and Design*, 407.
- [Meléndez et al., 2016] Meléndez, E., Sánchez-Perea, M., Queral, C., Mula, J., Hernández, A., and Paris, C. (2016). Standardized probabilistic safety assessment models: First results and conclusions of a feasibility study. In *13th International Conference on Probabilistic Safety Assessment and Management (PSAM 13)*, Seoul, South Korea.
- [Mena-Rosell et al., 2018] Mena-Rosell, L., Queral, C., Ruiz-Zapatero, M., Sanchez-Perea, M., Jimenez, G., Gomez-Magan, J., Hortal, J., and Sanchez-Espinoza, V.-H. (2018). Analysis of pwr sbo sequences with rcp passive thermal shutdown seals. *Journal of Nuclear Science and Technology*, 55:649–662.
- [NEI, 2009] NEI (2009). NEI 06-12, B.5.b phase 2 & 3 submittal guideline, revision 3.
- [NEI, 2012] NEI (2012). NEI 12-06, diverse and flexible coping strategies (FLEX) rev.0 implementation guide.
- [NEI, 2015] NEI (2015). Streamlined approach for crediting FLEX in risk-informed decision making. Revision 0.
- [NEI, 2016a] NEI (2016a). NEI 14-01, emergency response procedures and guidelines for beyond design basis events and severe accidents. Rev.1.
- [NEI, 2016b] NEI (2016b). NEI 16-06, crediting mitigating strategies in risk-informed decision making.
- [NextEra Energy, 2015] NextEra Energy (2015). Point Beach, LLC’s notification of full compliance with order EA-12-049 modifying licenses with regard to requirements for mitigation strategies for beyond-design basis external events and submittal of final integrated plan, NRC 2015-0072 1.
- [Ng, 2020] Ng, C. (2020). Incorporating FLEX into NRC SPAR models. In *Regulatory Information Conference, NRC*.
- [NNB Generation Company (HPC) LTD, 2017] NNB Generation Company (HPC) LTD (2017). HPC PCSR3: Chapter 14 reference plant design basis, sub-chapter 14.5 design extension conditions a (DEC-A). Multiple failures analysis.
- [NNSA, 2016] NNSA (2016). Nuclear power plant design safety regulations, HAF102-2016 (In Chinese).
- [NRA, 2013] NRA (2013). New regulatory requirements for light water nuclear power plants.
- [NRC, 1999] NRC (1999). Rates of initiating events at U.S. nuclear power plants: 1987 - 1995. Technical Report NUREG/CR-5750.
- [NRC, 2005] NRC (2005). The SPAR-H human reliability analysis method. NUREG/CR-

6883.

- [NRC, 2010a] NRC (2010a). Risk assessment of operational event. handbook. Volume 3 - SPAR model reviews, 2nd edition.
- [NRC, 2010b] NRC (2010b). SPAR model development program.
- [NRC, 2013a] NRC (2013a). Byron station, units 1 and 2 - interim staff evaluation relating to overall integrated plan in response to order EA-12-049 (mitigation strategies).
- [NRC, 2013b] NRC (2013b). Glossary of risk-related terms in support of risk-informed decision-making. Technical Report NUREG-2122.
- [NRC, 2013c] NRC (2013c). Risk informing emergency preparedness oversight: Evaluation of emergency action levels - a pilot study of Peach Bottom, Surry and Sequoyah. Technical report NUREG/CR-7154, Vol. 2.
- [NRC, 2014] NRC (2014). Salem nuclear generating station, unit 1 and 2 - interim staff evaluation and audit report relating to overall integrated plan in response to order EA-12-049 (mitigation strategies) (Tac Nos. MF0868 and MF0869).
- [NRC, 2015a] NRC (2015a). Common cause failure parameter estimations, NUREG/CR-5497.
- [NRC, 2015b] NRC (2015b). Donald C. Cook, units 1 and 2 - safety evaluation regarding implementation of mitigating strategies and reliable spent fuel instrumentation related to orders EA-12-049 and EA-12-051.
- [NRC, 2015c] NRC (2015c). Industry-average performance for components and initiating events at u.s. commercial nuclear power plants. Technical Report NUREG/CR-6928.
- [NRC, 2016a] NRC (2016a). McGuire nuclear station, units 1 and 2 - safety evaluation regarding implementation of mitigating strategies and reliable spent fuel pool instrumentation related to orders EA-12-049 and EA-12-051 (Cac Nos. MF1160, MF1161, MF1062, and MF1063).
- [NRC, 2016b] NRC (2016b). North Anna, units 1 and 2 - safety evaluation regarding implementation of mitigating strategies and reliable spent fuel pool instrumentation related to orders EA-12-049 and EA-12-051.
- [NRC, 2016c] NRC (2016c). Seabrook station, unit 1 - safety evaluation regarding implementation of mitigating strategies and reliable spent fuel pool instrumentation related to orders EA-12-049 and EA-12-051 (Cac Nos. MF0836 and MF0837).
- [NRC, 2016d] NRC (2016d). Sequoyah nuclear plant, units 1 and 2 - safety evaluation regarding implementation of mitigating strategies and reliable spent fuel pool instrumentation related to orders EA-12-049 and EA-12-051 (Cac Nos. MF0864, MF0865, MF0794, and MF0795).
- [NRC, 2016e] NRC (2016e). Shearon Harris nuclear power plant, unit 1 - safety evaluation regarding implementation of mitigating strategies and reliable spent fuel pool instrumentation related to orders EA-12-049 and EA-12-051 (Cac Nos. MF0874 and MF0792).

- [NRC, 2016f] NRC (2016f). Surry power station, units 1 and 2 - safety evaluation regarding implementation of mitigating strategies and reliable spent fuel pool instrumentation related to orders EA-12-049 and EA-12-051 (Tac Nos. MF0980, MF0981, MF0976, and MF0977).
- [NRC, 2016g] NRC (2016g). Virgil C. Summer, unit 1, safety evaluation regarding implementation of mitigating strategies and reliable spent fuel pool instrumentation related to orders EA-12-049 and EA-12-051.
- [NRC, 2016h] NRC (2016h). Vogtle electric generating plant, units 1 and 2 - safety evaluation regarding implementation of mitigating strategies and reliable spent fuel pool instrumentation related to orders EA-12-049 and EA-12-051.
- [NRC, 2017] NRC (2017). Braidwood station, units 1 and 2 - safety evaluation regarding implementation of mitigating strategies and reliable spent fuel pool instrumentation related to orders EA-12-049 and EA-12-051.
- [NRC, 2020a] NRC (2020a). Applying HRA to FLEX - expert elicitation. volume 1.
- [NRC, 2020b] NRC (2020b). Applying HRA to FLEX - using IDHEAS-ECA. Volume 2.
- [NRC, 2021] NRC (2021). Project plan to prepare the U.S. nuclear regulatory commission for efficient and effective licensing of accident tolerant fuels. Technical Report ML21243A298.
- [NRC, 2022a] NRC (2022a). ASP analysis accident sequence precursor, Waterford steam electric station (unit 3), loss of offsite power due to hurricane Ida.
- [NRC, 2022b] NRC (2022b). Crediting FLEX in probabilistic risk assessments used to support risk-informed applications. In June 21, 2022, US Nuclear Regulatory Commission public meeting with industry to discuss items pertaining to diverse and flexible coping strategies (FLEX) data analysis (Issue Adams Accession Number ML22222A204).
- [NRC, 2022c] NRC (2022c). Integrated human event analysis system for event and condition assessment (IDHEAS-ECA).
- [NRC, 2023] NRC (2023). ASP analysis accident sequence precursor, Surry power station (unit 1). Failure of unit 1 EDG results in an operation or condition prohibited by TS.
- [NRC, 2024] NRC (2024). Update to the accident tolerant fuel roadmap to readiness. Technical Report ML23353A163.
- [NSSC, 2016] NSSC (2016). Regulations on the scope of accident management and detailed criteria for evaluating accident management capabilities (In Korean).
- [OECD-NEA, 2018] OECD-NEA (2018). State-of-the-art report on light water reactor accident-tolerant fuels. Technical report.
- [ONR, 2020] ONR (2020). Safety assessment principles for nuclear facilities, 2014 edition, revision 1 (January 2020), ONR CM9 Ref 2019/367414.
- [Parisi et al., 2019a] Parisi, C., Ma, Z., Davis, C., Dailey, R., Wang, J., Zhang, H., Zhang, S., and Corradini, M. (2019a). Evaluation of the benefits of ATF, FLEX, and passive cooling system for an enhanced resilient PWR model. Technical Report INL/EXT-19-56215, Light

- Water Reactor Sustainability Program, USA.
- [Parisi et al., 2019b] Parisi, C., Ma, Z., Davis, C., Park, J., Boring, R., and Zhang, H. (2019b). Risk-informed analysis for an enhanced resilient PWR with ATF, FLEX, and passive cooling. Technical Report INL/EXT-19-53556, Light Water Reactor Sustainability Program, USA.
- [Parisi et al., 2020] Parisi, C., Ma, Z., Mandelli, D., Anderson, N., and Zhang, H. (2020). Risk-informed safety analysis for accident tolerant fuels. *Nuclear Science and Engineering*, 194:748–770.
- [Park et al., 2021a] Park, G., Arigi, A., and Kim, J. (2021a). Application of SPAR-H human reliability analysis method to the use of portable equipment in nuclear power plants. In *Transactions of the Korean Nuclear Society Virtual Autumn Meeting*, Virtual Meeting.
- [Park et al., 2019] Park, J., Arigi, A., and Kim, J. (2019). Treatment of human and organizational factors for multi-unit HRA: Application of SPAR-H method. *Annals of Nuclear Energy*, 132:656–678.
- [Park et al., 2022] Park, J., Boring, R., and Ulrich, T. (2022). An approach to dynamic human reliability analysis using the EMERALD dynamic risk assessment tool. In *16th Probabilistic Safety Assessment and Management*, Honolulu, Hawaii, USA.
- [Park et al., 2020] Park, J., Jung, W., and Kim, J. (2020). Inter-relationships between performance shaping factors for human reliability analysis of nuclear power plants. *Nuclear Engineering and Technology*, 52:87–100.
- [Park et al., 2021b] Park, J., Ulrich, T. A., Boring, R. L., Zhang, S., Ma, Z., and Zhang, H. (2021b). Modeling FLEX human actions using the Emerald dynamic risk assessment tool. In *2021 International Topical Meeting on Probabilistic Safety Assessment and Analysis*, Virtual Meeting.
- [Park et al., 2018] Park, J. K., Lee, J.-D., Jang, H.-S., and Park, H.-S. (2018). Analysis of mobile diesel generator operation to cope with extended loss of all AC power in nuclear power plant. In *53rd International Universities Power Engineering Conference*, Glasgow, UK.
- [París et al., 2019] París, C., Queral, C., Mula, J., Gómez-Magán, J., Sánchez-Perea, M., Meléndez, E., and Gil, J. (2019). Quantitative risk reduction by means of recovery strategies. *Reliability Engineering & System Safety*, 182:13–32.
- [Prošek and Uršič, 2019] Prošek, A. and Uršič, M. (2019). Review of design extension conditions experiments and analyses for non-degraded core. *Journal of Energy*.
- [PWROG, 2012] PWROG (2012). PWR owners group executive committee/NRC meeting March 12th, 2012.
- [Queral and Alonso-Escos, 2023] Queral, C. and Alonso-Escos, J. (2023). Lecture 15: Post-fukushima improvements in spanish npps. impact in severe accident phenomenology. In *Severe Accident Phenomenology short course*, Madrid, Spain. ETSII - Universidad Politécnica de Madrid.

- [Ramos et al., 2018] Ramos, M., Utnea, I., and Mosleh, A. (2018). On factors affecting autonomous ships operators performance in a shore control center. In *14th Probabilistic Safety Assessment and Management*, Los Angeles, California, USA.
- [Rasmussen et al., 2015] Rasmussen, M., Standal, M. I., and Laumann, K. (2015). Task complexity as a performance shaping factor: A review and recommendations in standardized plant analysis risk-human reliability analysis (spar-h) adaption. *Saf Sci*, 76:228–238.
- [Ryu et al., 2019] Ryu, H.-S., i Park, J., and Lim, H.-K. (2019). HEPs calculation for MACST equipment. In *Transactions of the Korean Nuclear Society Autumn Meeting*, Ilsan, Korea.
- [Saint-Germain et al., 2016] Saint-Germain, S., Boring, R., Banaseanu, G., Akl, Y., and Xu, M. (2016). Modification of the SPAR-H method to support HRA for level 2 PSA. In *13th Probabilistic Safety Assessment and Management*, Seoul, South Korea.
- [Sandia, 2005] Sandia, N. L. (2005). Good practices for implementing human reliability analysis (HRA), NUREG-1792. Technical report.
- [Shah et al., 2021] Shah, A., Christian, R., Kim, J., Kim, J., Park, J., and Kang, H. (2021). Dynamic probabilistic risk assessment based response surface approach for FLEX and accident tolerant fuels for medium break LOCA spectrum. *Energies*, 14:2490.
- [Shah et al., 2023] Shah, A. U. A., Kim, J., Christian, R., and Kang, H. G. (2023). Sensitivity analysis of effectiveness of FLEX strategies using a CDF-based importance measure under accident conditions. *Nuclear Science and Engineering*.
- [Shahinoor and Lim, 2017] Shahinoor, I. and Lim (2017). Comparative study of FLEX strategies for extended station blackout (SBO) using PRA. *International Journal of Engineering Research & Technology*, 6.
- [SNSA, 2017] SNSA (2017). Rules on radiation and nuclear safety factors - JV5. *Official Gazette of the Republic of Slovenia, No. 74/16 and 76/17 – ZVISJV-1*.
- [SNSA, 2020] SNSA (2020). The report of the republic of slovenia on nuclear safety second national report of the republic of slovenia according to article 9.1 of council directive 2009/71/euratom establishing a community framework for the nuclear safety of nuclear installations amended by council directive 2014/87/euratom.
- [Son and Lim, 2017] Son, H.-J. and Lim, H.-K. (2017). Study of the FLEX effectiveness of strategies under the long-term SBO by PRA. *International Journal of Engineering Research & Technology (IJERT)*, ISSN, 6.
- [Southern Nuclear, 2013] Southern Nuclear (2013). Joseph M. Farley, units 1 and 2 notification of full compliance of required action for nrc order EA-12-049 mitigation strategies for beyond-design-basis external events), NL-16-1431.
- [Southern Nuclear, 2021] Southern Nuclear (2021). National safety month: FLEX strategy. <https://www.southernnuclear.com/news-center/education/2021-0609-national-safety-flex.html>.
- [SRNI, 2011] SRNI (2011). Royal decree of 30 november 2011 on the safety requirements for

- nuclear facilities. Article 21. Kingdom of Belgium (In French).
- [STP Nuclear, 2016] STP Nuclear (2016). South Texas project, units 1 and 2 docket no. STN 50-498, STN 50-499 notification of full compliance with order EA-12-049 for mitigation strategies for beyond design basis external events and update for order.
- [Stringfellow, 2017] Stringfellow, J. (2017). The PWR owners group role in Fukushima response. In *IAEA Technical Meeting on Operational Experience with Implementation of Post-Fukushima Actions in Nuclear Power Plants*.
- [STUK, 2019a] STUK (2019a). Guide YVL B.3, deterministic safety analyses for a nuclear power plant.
- [STUK, 2019b] STUK (2019b). Guide YVL B.4, nuclear fuel and reactor.
- [STUK, 2019c] STUK (2019c). Olkiluoto 3 - FSAR review report.
- [Suh and Kim, 2019] Suh, Y. and Kim, J. (2019). Current status and future plans on estimation of human error probabilities of flex/macst actions. In *Transactions of the Korean Nuclear Society Spring Meeting*, Jeju, South Korea.
- [SUJB, 2019] SUJB (2019). Deterministic analysis extended design conditions without severe fuel damage (DEC A). Nuclear Safety BN-JB-2.2 (Rev. 0.0). (In Czech).
- [SUJB, 2021] SUJB (2021). Deterministic safety analyses of postulated initiating events and scenarios DEC B. Nuclear Safety BN-JB-2.3 (Rev. 0.0). (In Czech).
- [Tabadar et al., 2019a] Tabadar, Z., Ansarifar, G. R., and Pirouzmand, A. (2019a). Probabilistic safety assessment of portable equipment applied in VVER-1000/V446 nuclear reactor during loss of ultimate heat sink accident for stress test program development. *Progress in Nuclear Energy*, 117.
- [Tabadar et al., 2019b] Tabadar, Z., Ansarifar, G. R., and Pirouzmand, A. (2019b). Thermal-hydraulic modeling for deterministic safety analysis of portable equipment application in the VVER-1000 nuclear reactor during loss of ultimate heat sink accident using RELAP5/MOD3.2 code. *Annals of Nuclear Energy*, 127:53–67.
- [Ulrich et al., 2020] Ulrich, T., Mortenson, T., Boring, R., and Prescott, S. (2020). Dynamic modeling of field operators in human reliability analysis: An emerald and goms-hra dynamic model of flex operator actions.
- [Viktorov and Harwood, 2015] Viktorov, A. and Harwood, C. (2015). Design extension conditions concept and its application to operating reactors in Canada.
- [Wang, 2013a] Wang, Y. (2013a). The human reliability analysis in fire PSA using SPAR-H method. In *21st International Conference on Nuclear Engineering*, Chengdu, China. ASME.
- [Wang, 2013b] Wang, Y. (2013b). The human reliability analysis in level 2 PSA using SPAR-H method. *Advanced Materials Research*, 608:848–853.
- [Wang et al., 2019] Wang, Y., Feng, C., Liu, M., and Li, L. (2019). Study on identification of design extension conditions without significant fuel degradation and safety features of

nuclear power plant.

- [Watanabe et al., 2018] Watanabe, H., Sancaktar, S., and Dennis, S. (2018). Generic PWR SPAR model white paper.
- [Webster, 2017] Webster, B. (2017). Modeling FLEX in Surry power station internal events and flooding PRA. In *International Topical Meeting on Probabilistic Safety Assessment and Analysis*, Pittsburgh, PA, USA.
- [WENRA, 2020] WENRA (2020). Safety reference levels for existing reactors.
- [Wierman et al., 2007] Wierman, T., Rasmuson, D., and Mosleh, A. (2007). Common-cause failure database and analysis system: Event data collection, classification, and coding, NUREG/CR-6268.
- [Xin, 2018] Xin, T. (2018). Safety approach and safety assessment of hualong one. Hualong Pressurized Water Reactor Technology Corporation, Ltd.
- [Xu and Zhang, 2021a] Xu, H. and Zhang, B. (2021a). A review of the FLEX strategy for nuclear safety. *Nuclear Engineering and Design*, 382.
- [Xu and Zhang, 2021b] Xu, H. and Zhang, B. (2021b). A review on the defense-in-depth concept and the FLEX strategies in different countries after Fukushima accident. *Nuclear Technology and Radiation Protection*, 36:116–127.
- [Xu et al., 2021] Xu, H., Zhang, B., and Liu, Y. (2021). New safety strategies for nuclear power plants: A review. *International Journal of Energy Research*, 45:11564–11588.
- [Yang et al., 2017] Yang, Z., Chong, Y., Li, C., and Zhang, J. (2017). Preliminary study on the identification of DEC’s without significant fuel degradation in new NPP designs.
- [Zhang and Ma, 2020] Zhang, S. and Ma, Z. (2020). Incorporating FLEX strategies in multi-unit probabilistic risk assessment. In *28th International Conference on Nuclear Engineering Joint with the ASME 2020 Power Conference*, Anaheim, California, USA.
- [Zhang et al., 2021a] Zhang, S., Zhang, H., Park, J., Liu, T., Ulrich, T., Boring, R., Ma, Z., and Jiang, H. (2021a). Risk-informed analysis for enhanced resilient nuclear power plant with initiatives including ATF, FLEX, and advanced battery technology, INL/EXT-21-64546.
- [Zhang et al., 2021b] Zhang, S., Zhang, H., Park, J., Liu, T., Ulrich, T. A., Boring, R. L., Ma, Z., and Jiang, H. (2021b). Risk-informed analysis for enhanced resilient nuclear power plant with initiatives including ATF, FLEX, and advanced battery technology. Technical Report INL/EXT-21-64546, Light Water Reactor Sustainability Program, USA.

Appendix A

SPAR-CSN System Diagrams

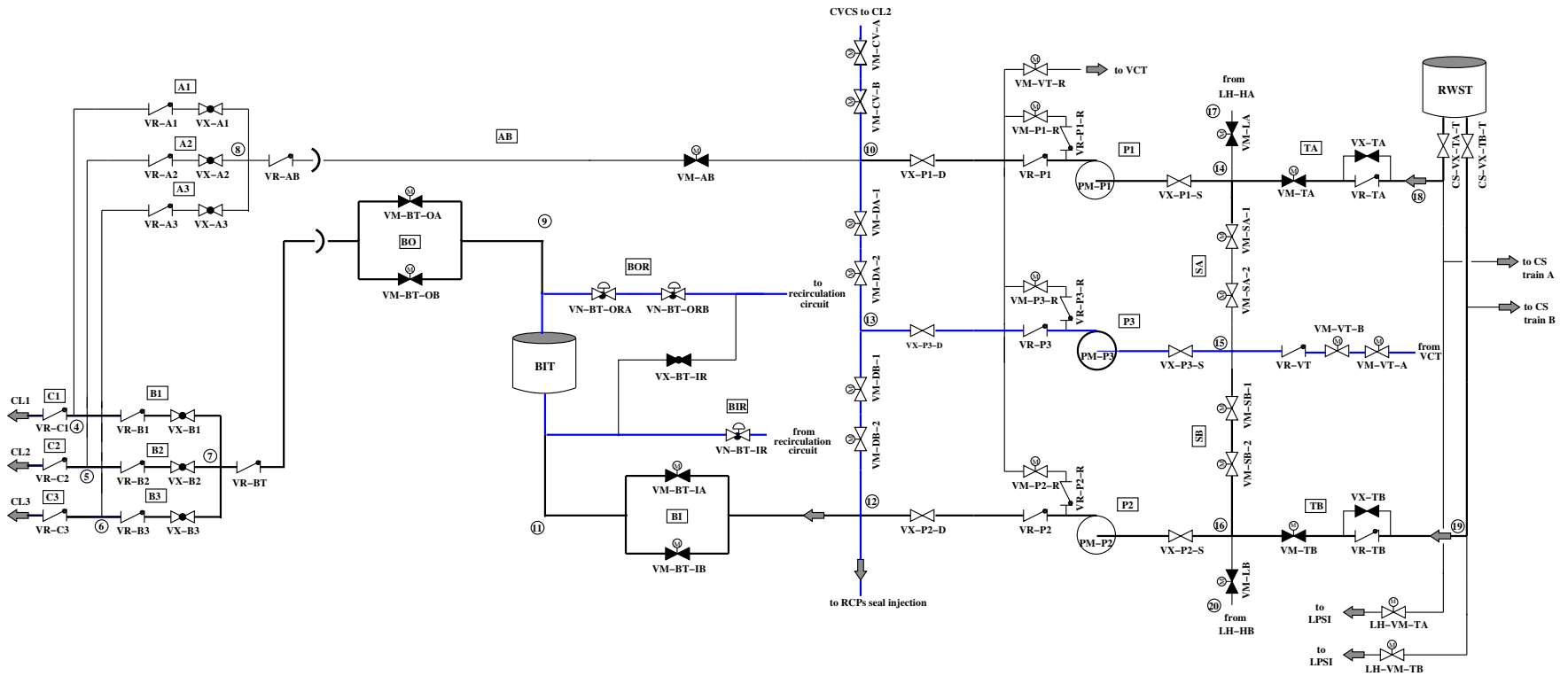


Figure A.1: SPAR-CSN diagram of HPSI: Working as CVCS in normal operation

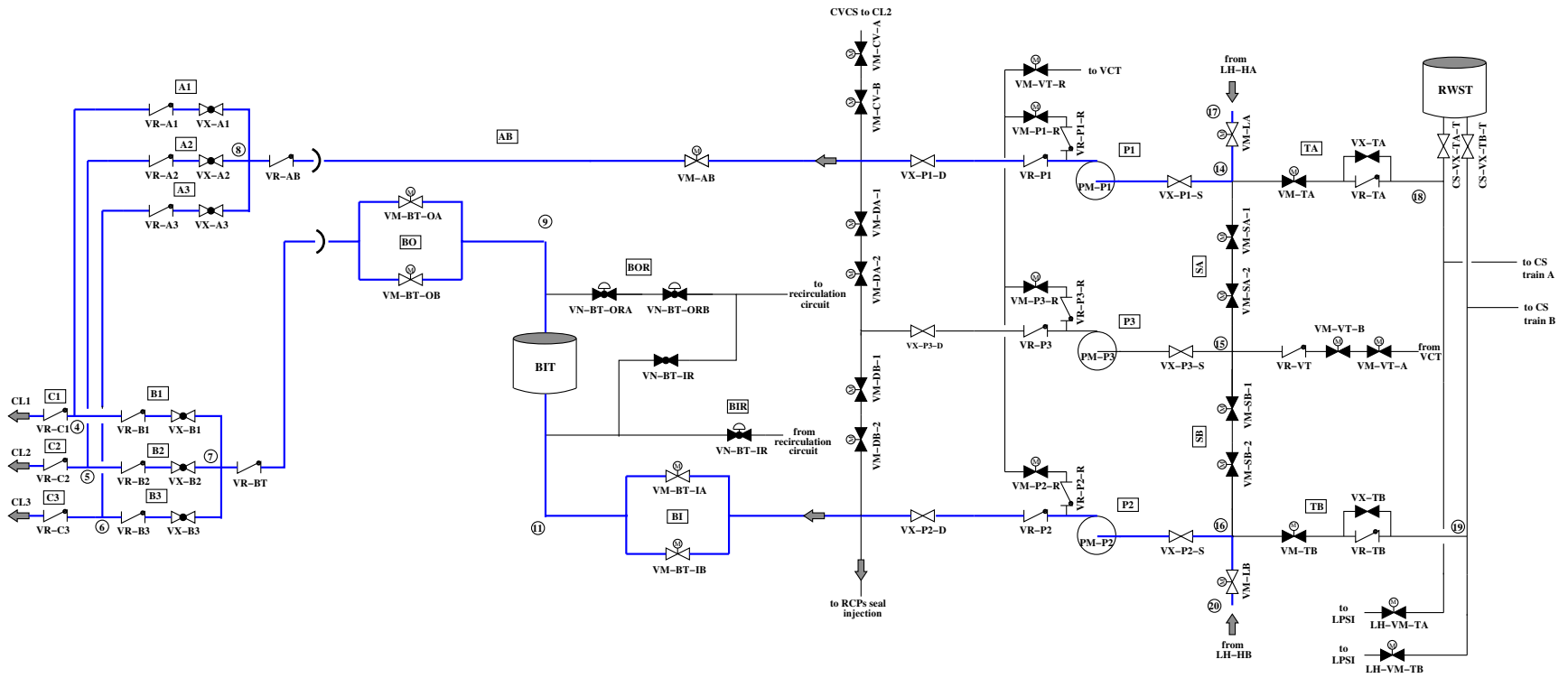


Figure A.3: SPAR-CSN diagram of HPSI: High Pressure Recirculation to Cold Legs

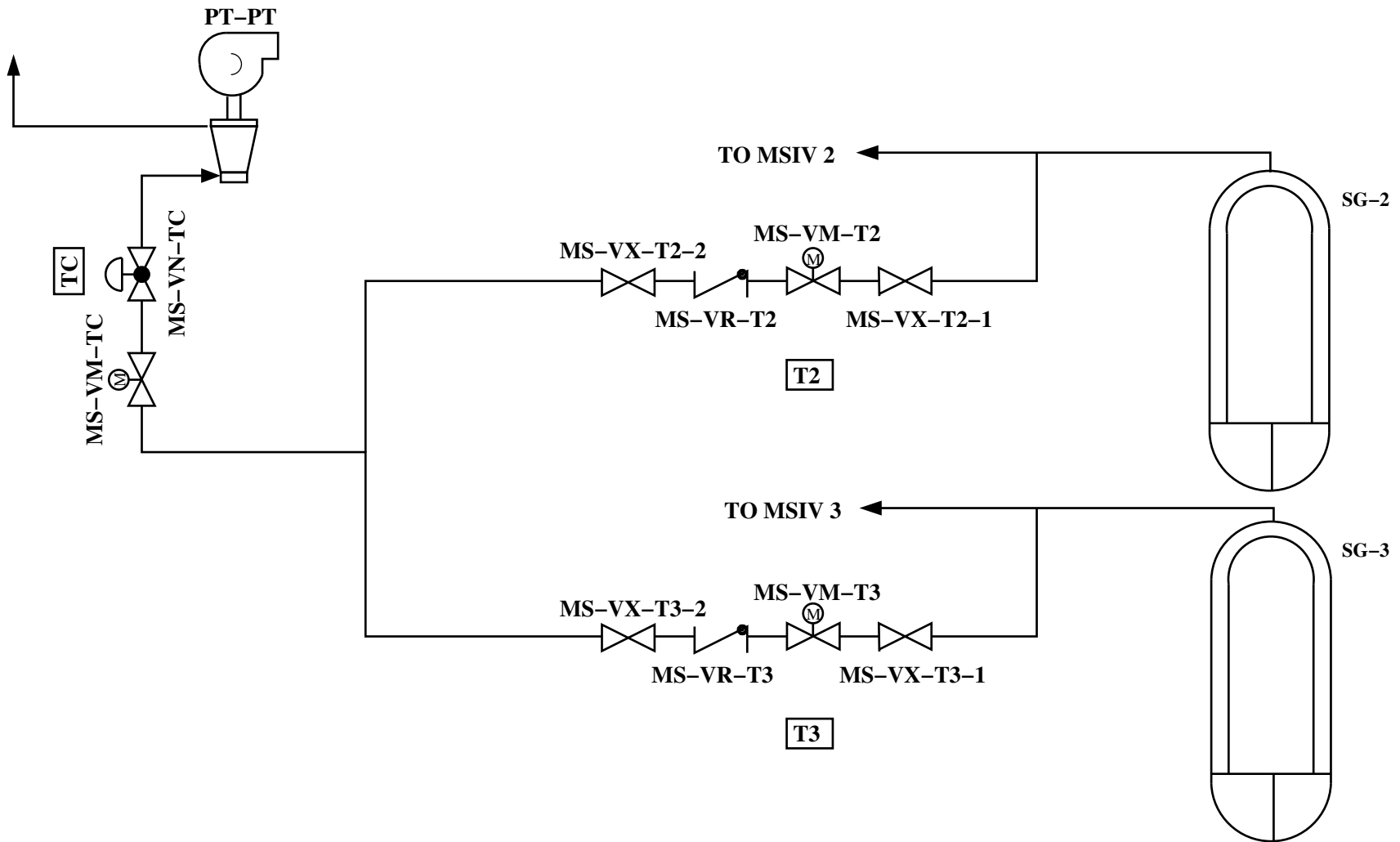


Figure A.5: SPAR-CSN diagram of the AFW: main steam extraction lines for the supply to the turbine of the AF-PT-PT turbo-pump

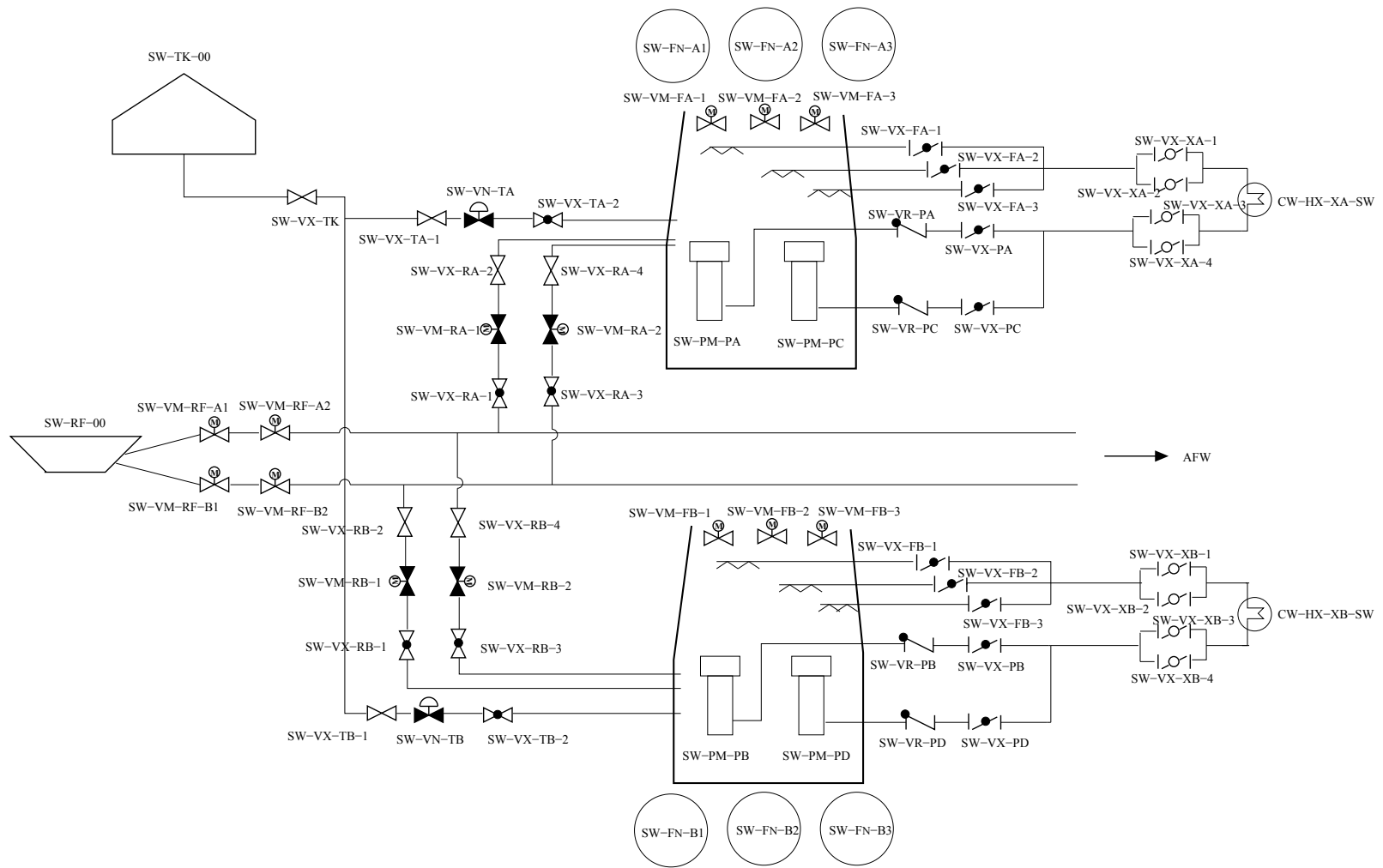


Figure A.6: SPAR-CSN diagram of the Essential Service Water System (ESWS)

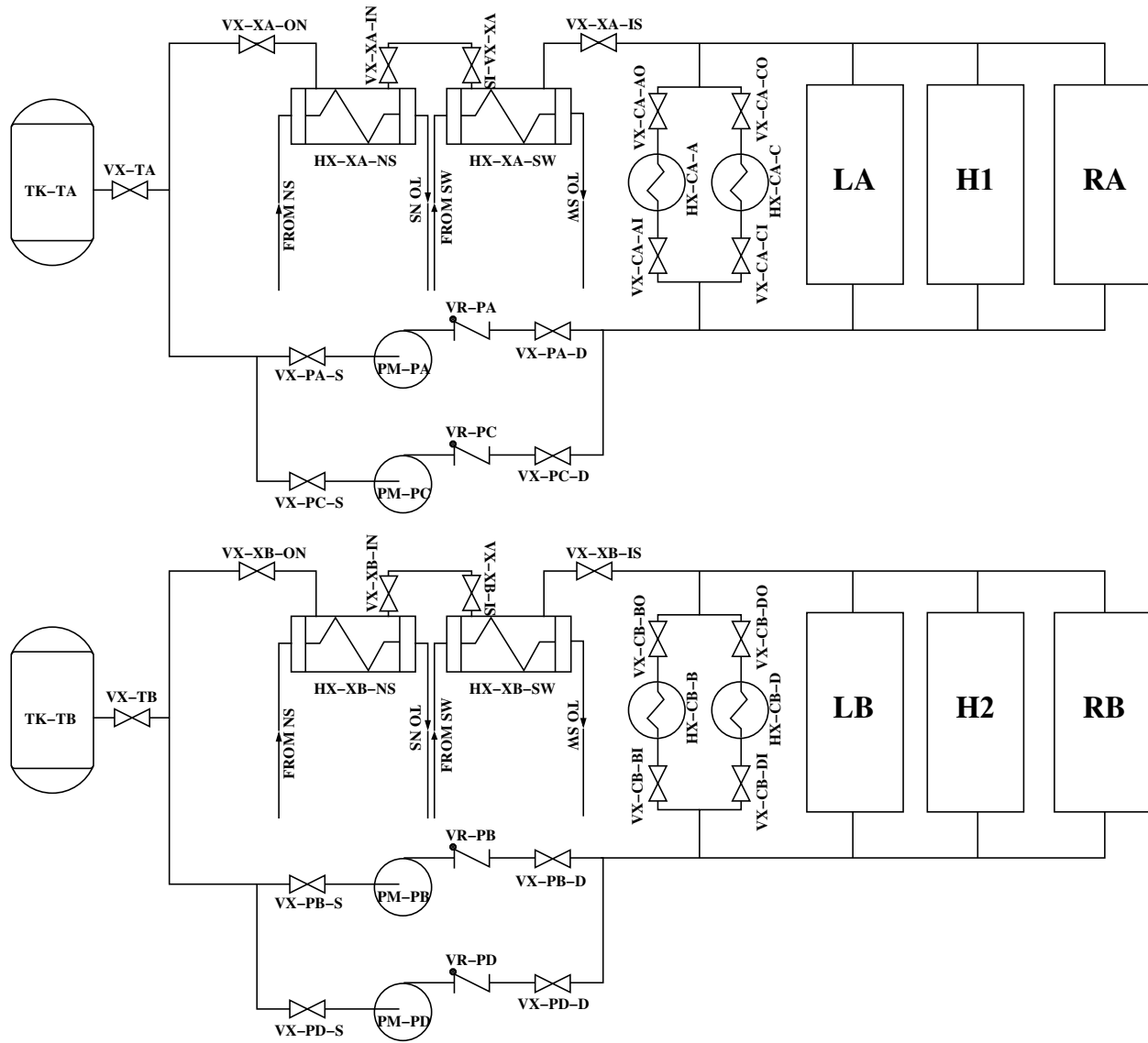


Figure A.7: SPAR-CSN diagram of the CCWS

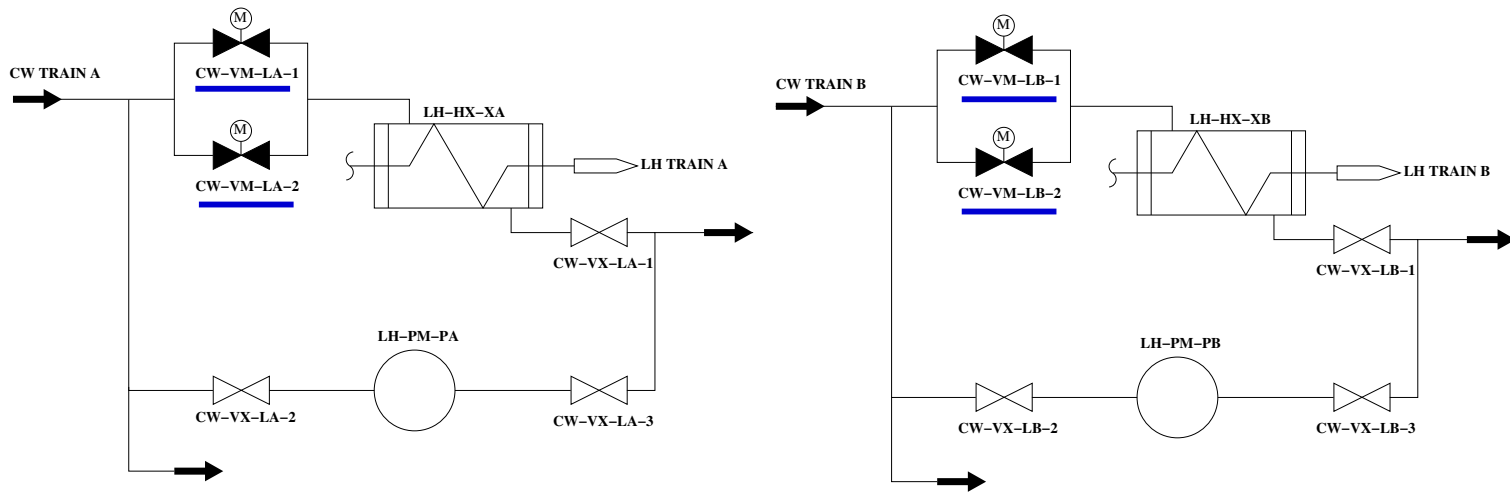


Figure A.8: SPAR-CSN diagram of CCWS: LPSI cooling

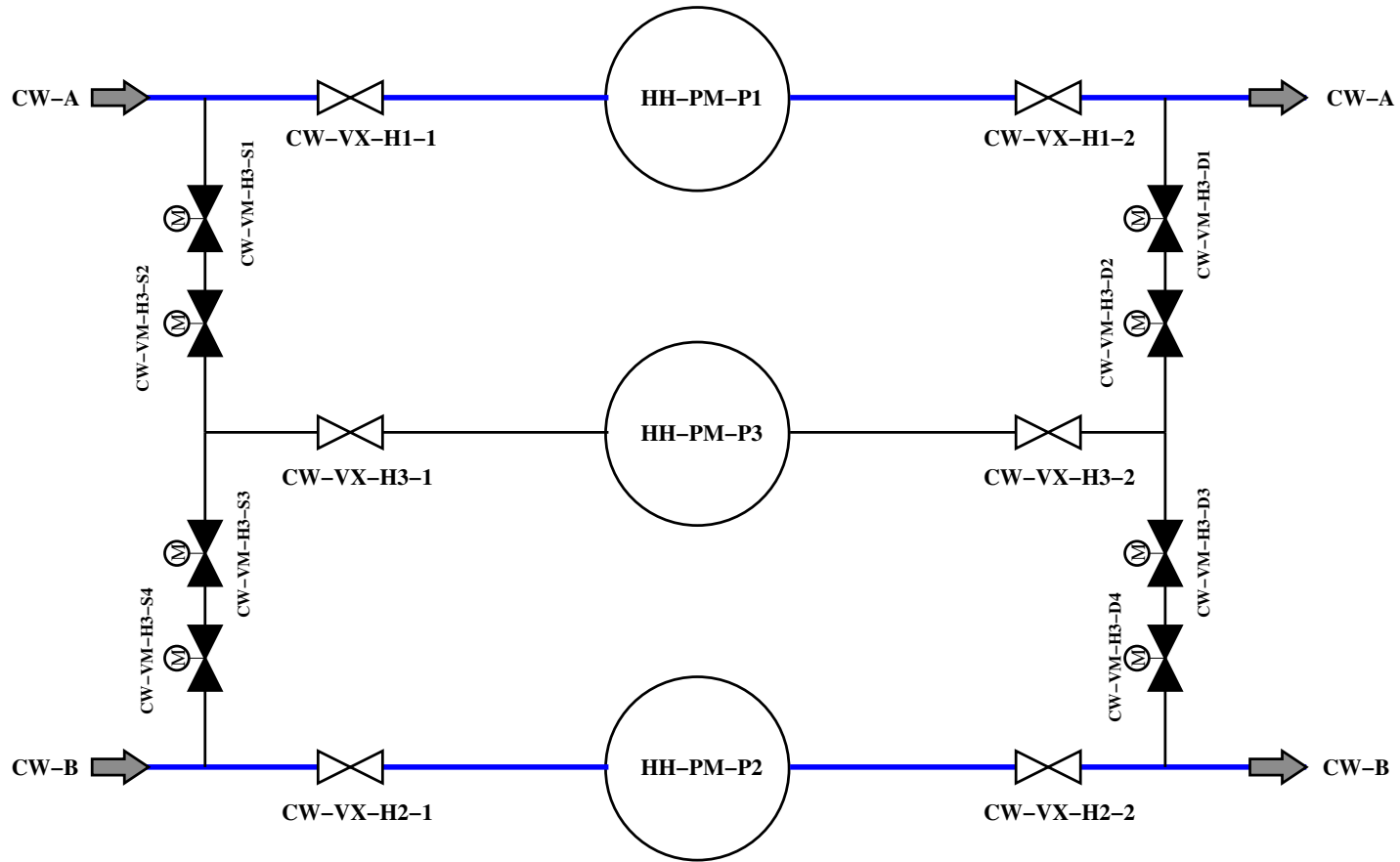


Figure A.9: SPAR-CSN diagram of CCWS: HPSI cooling

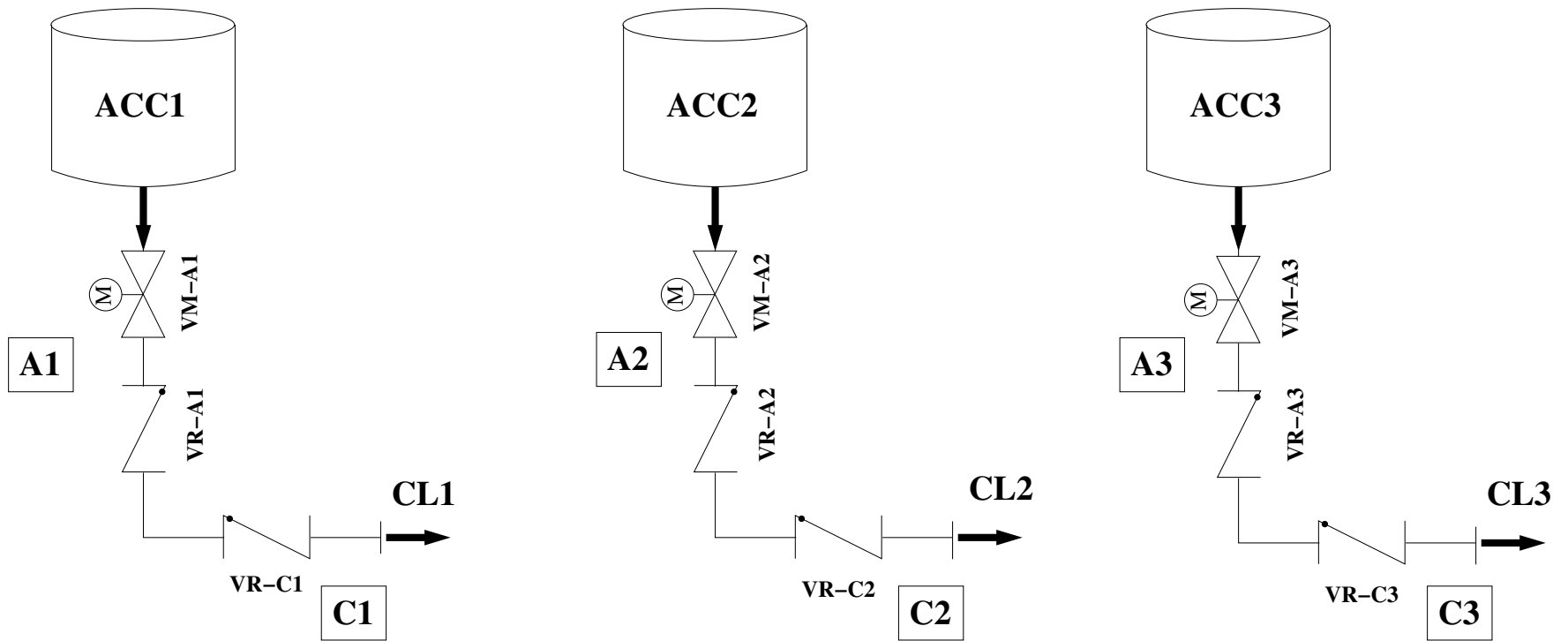


Figure A.10: SPAR-CSN diagram of AIS: Injection of Accumulators to CLs

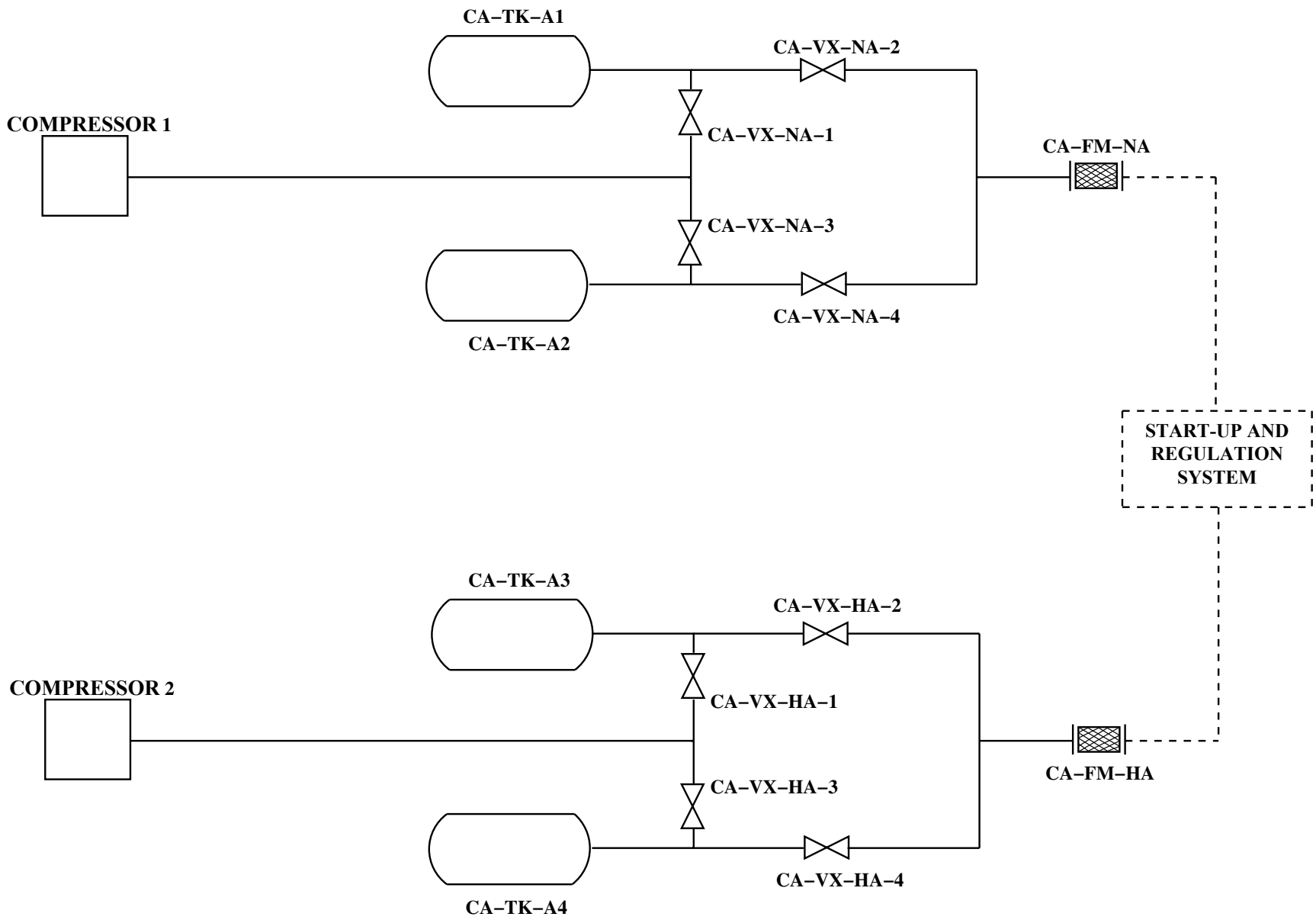


Figure A.11: SPAR-CSN diagram of Air System Diesel Generator A

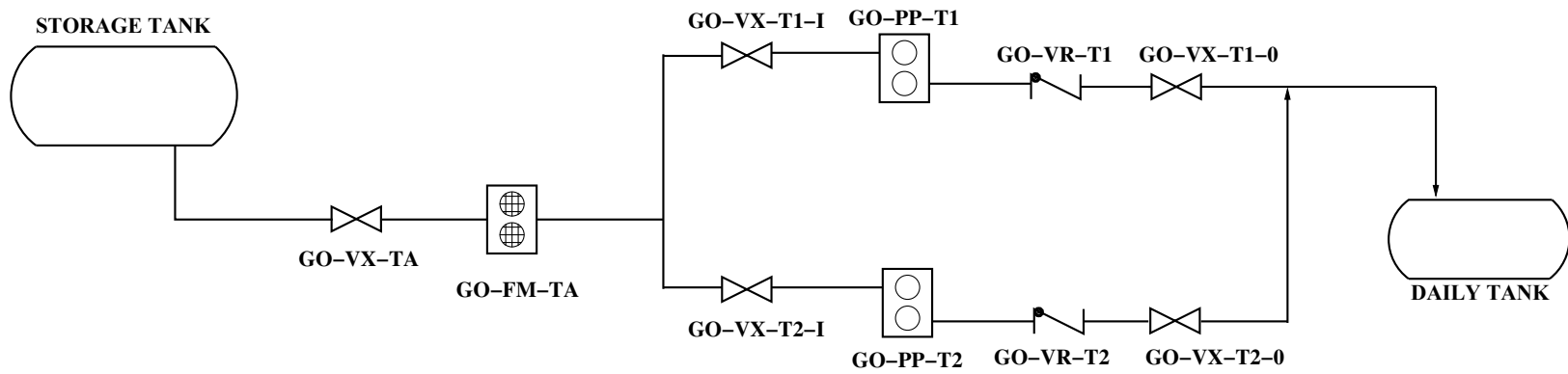


Figure A.12: SPAR-CSN diagram of Fuel System Diesel Generator A

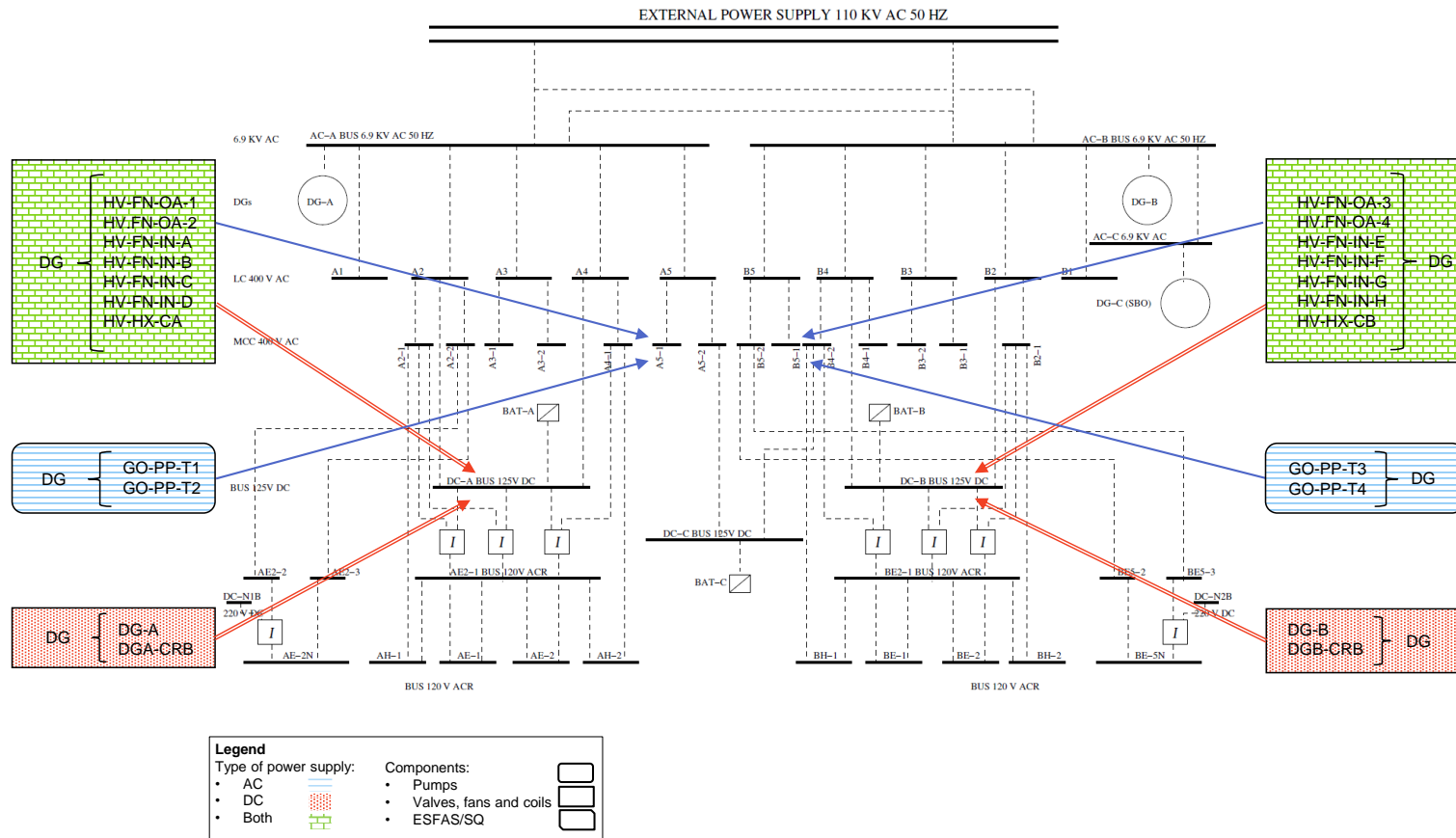


Figure A.13: One-line diagram of the SPAR-CSN model and its relation to the DG

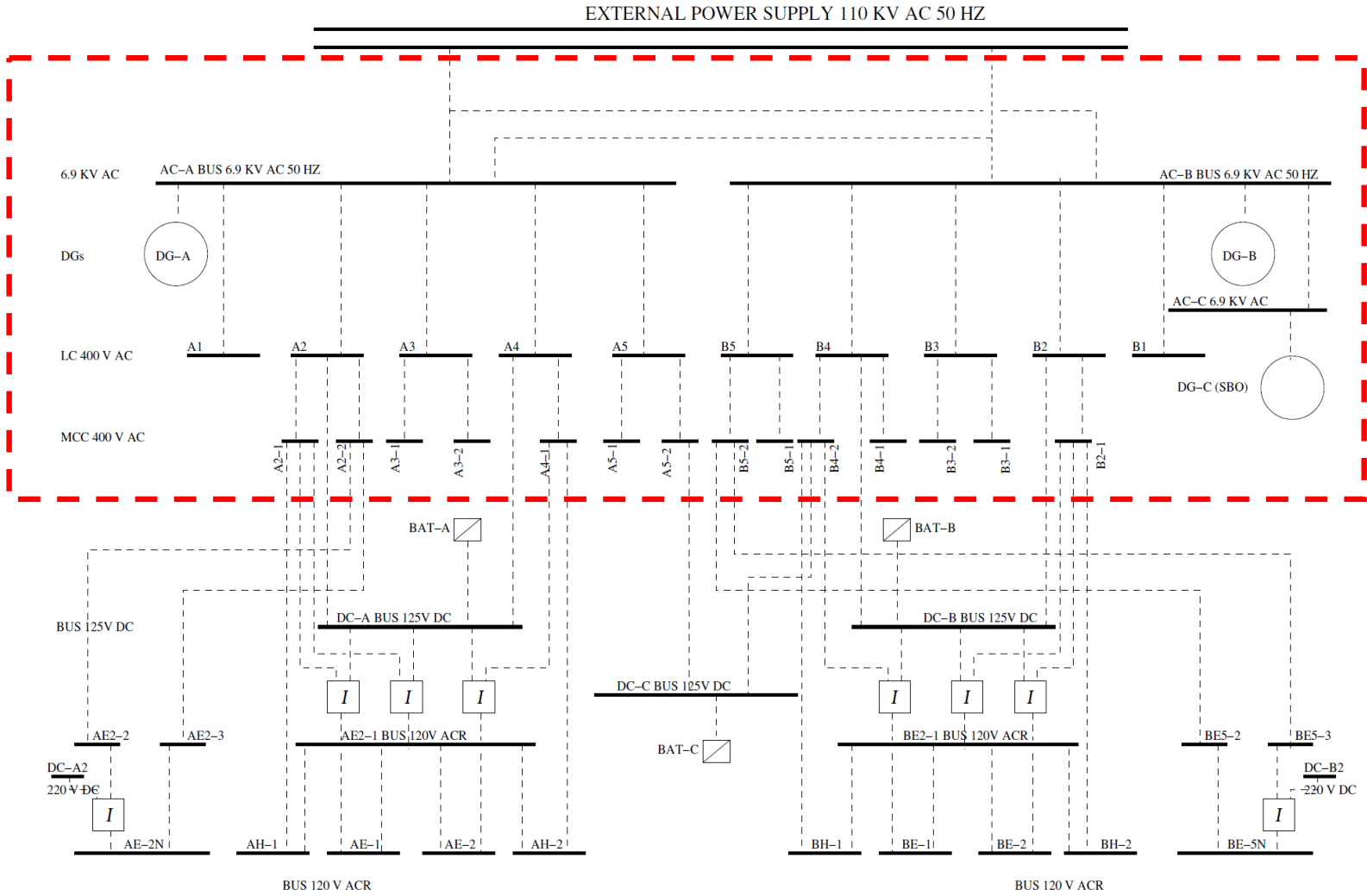


Figure A.14: One-line diagram of the Alternating Current Safeguard System of the SPAR-CSN model

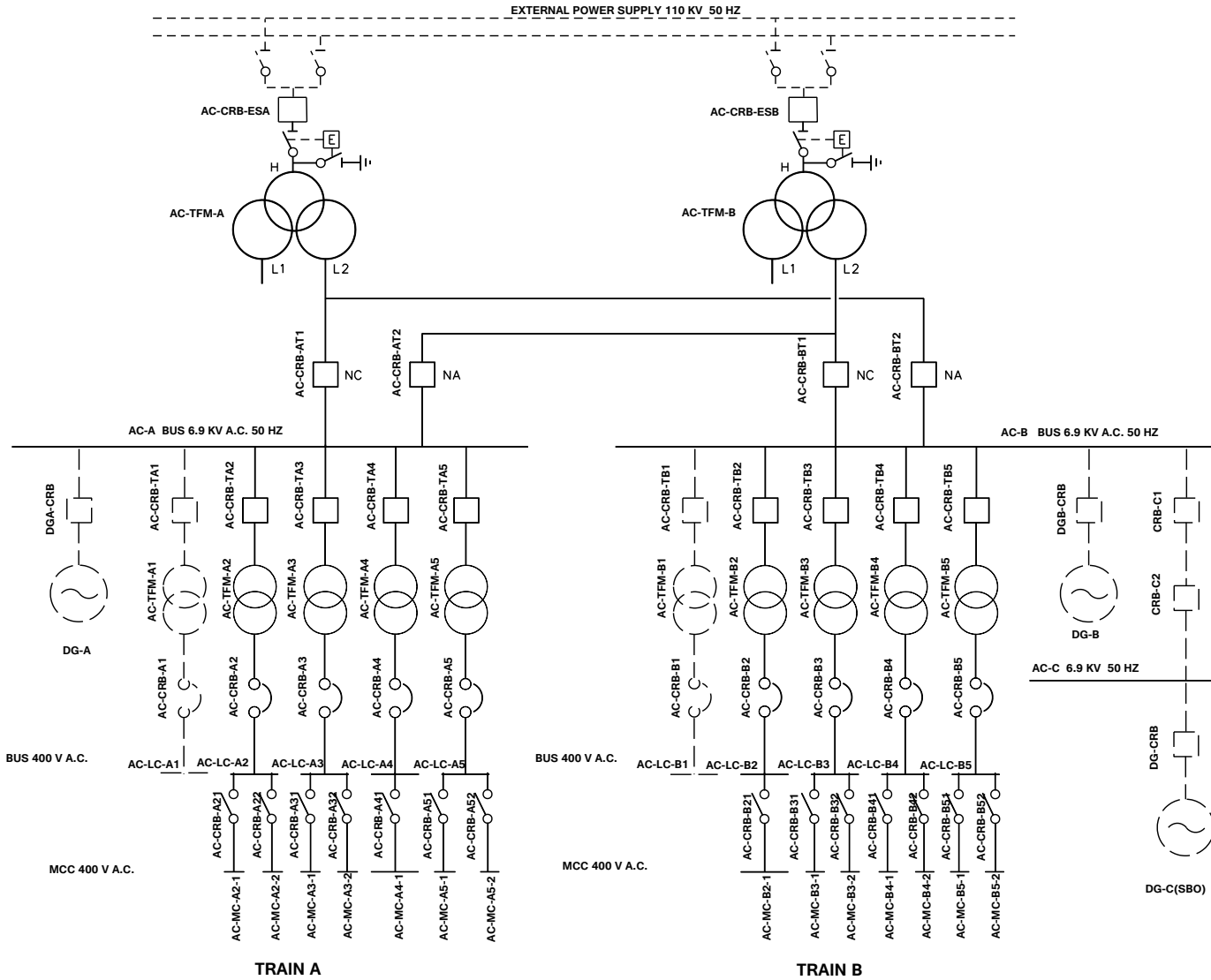


Figure A.15: One-line diagram of the SPAR-CSN model 6.9 kV, 400 V (AC) Alternating Current Safeguard System

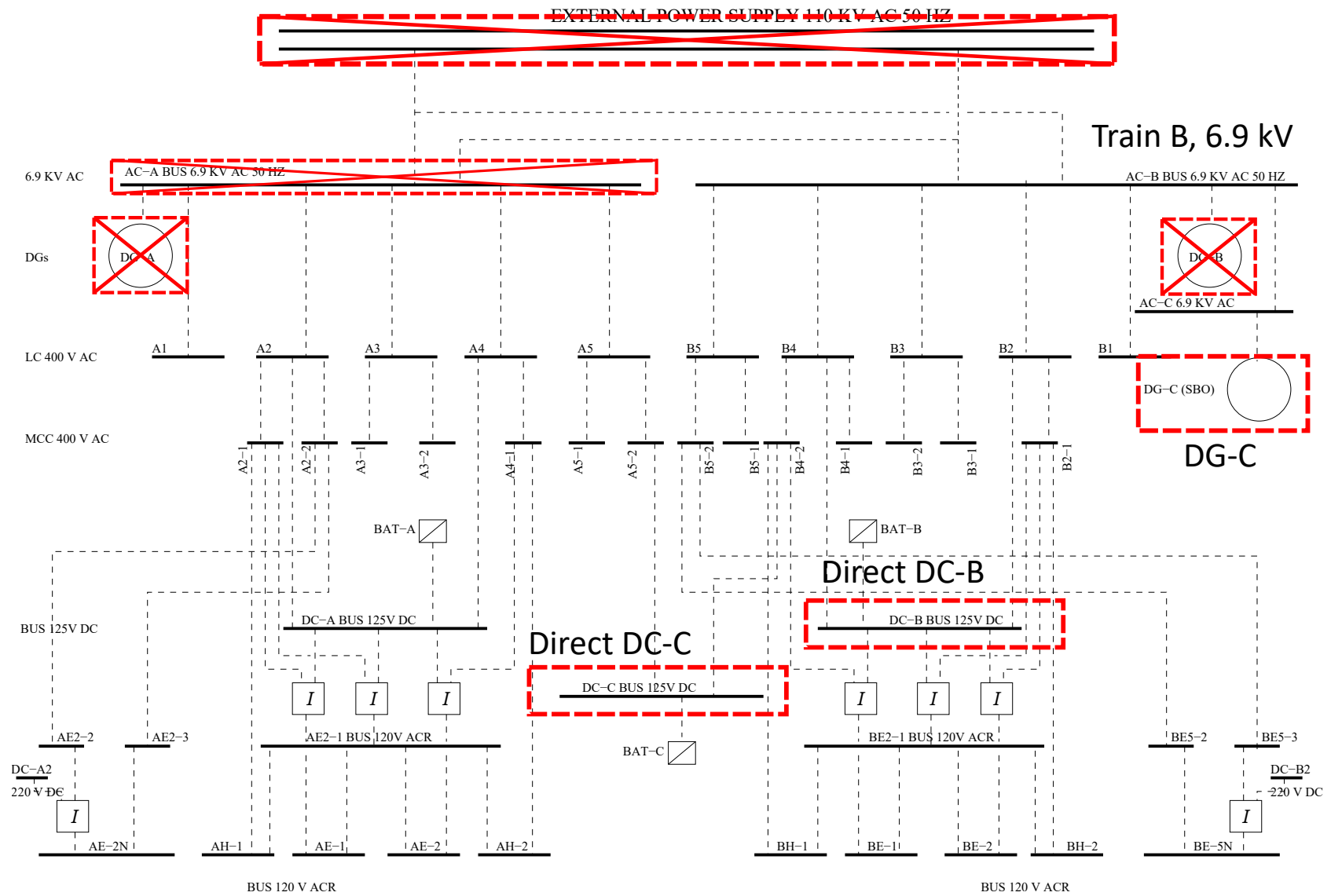


Figure A.16: One-line diagram of the electrical connections in the SPAR-CSN model at SBO

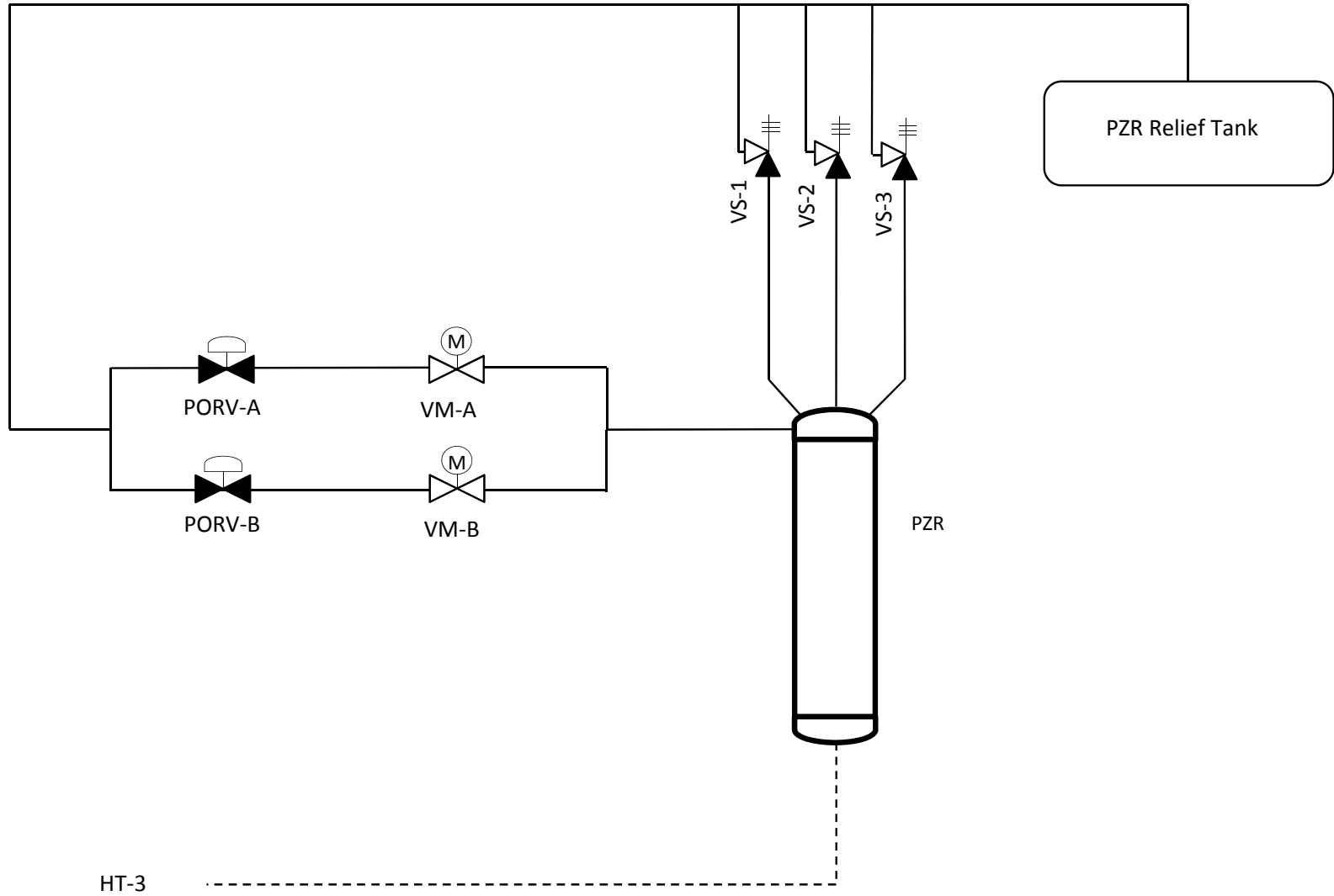


Figure A.17: SPAR-CSN diagram of the PORV-PZR

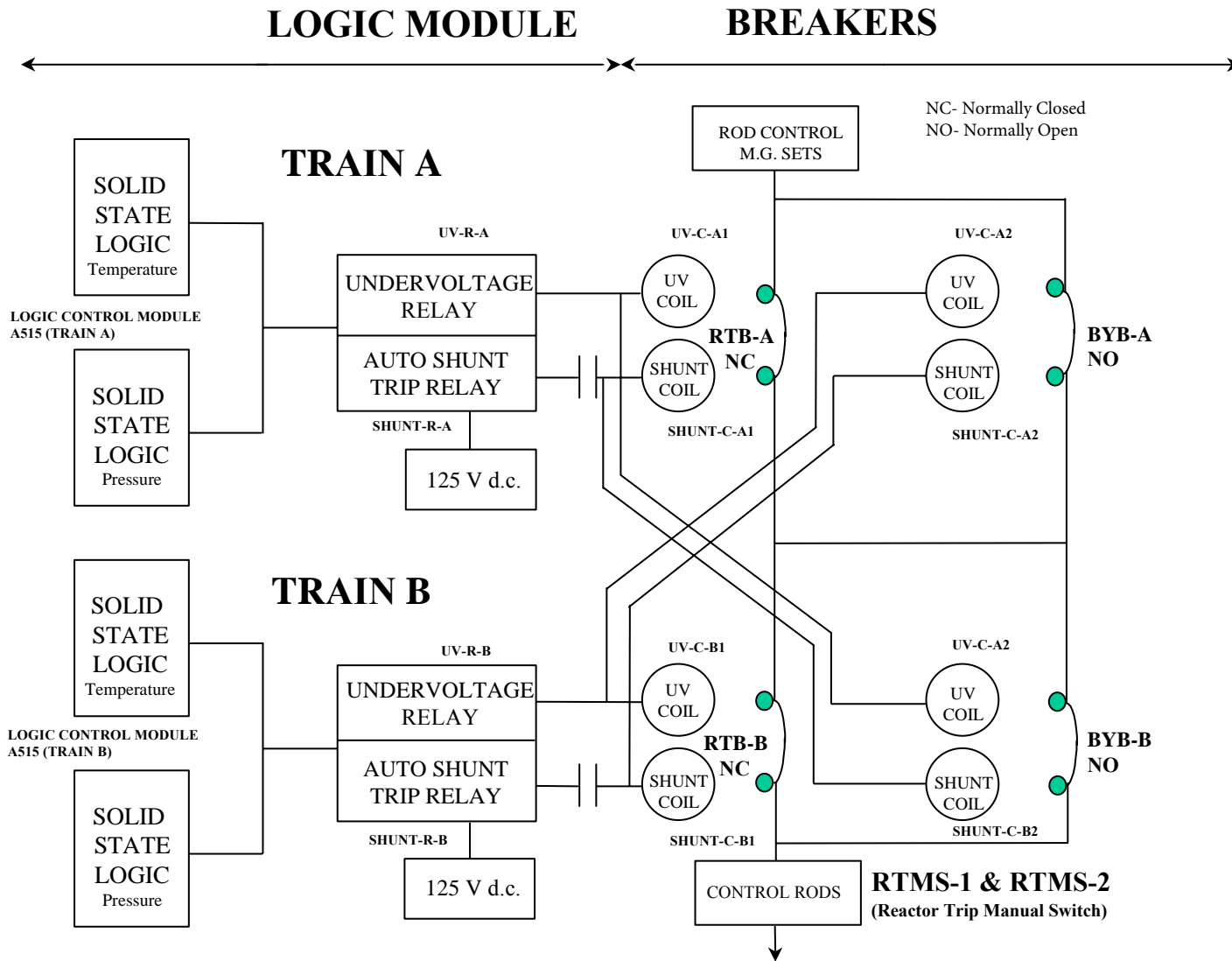


Figure A.18: SPAR-CSN diagram of the RPS

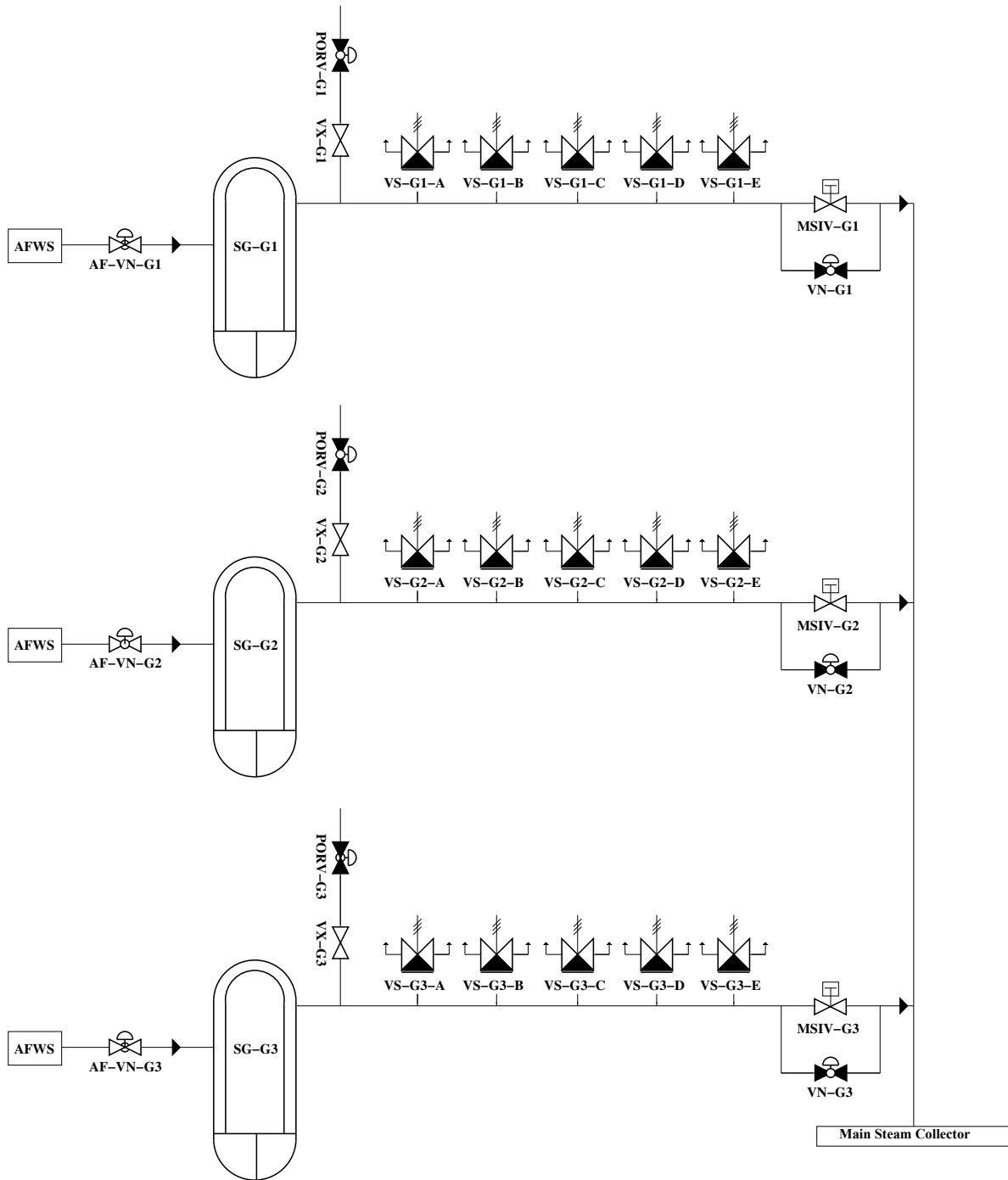


Figure A.19: SPAR-CSN diagram of the MS

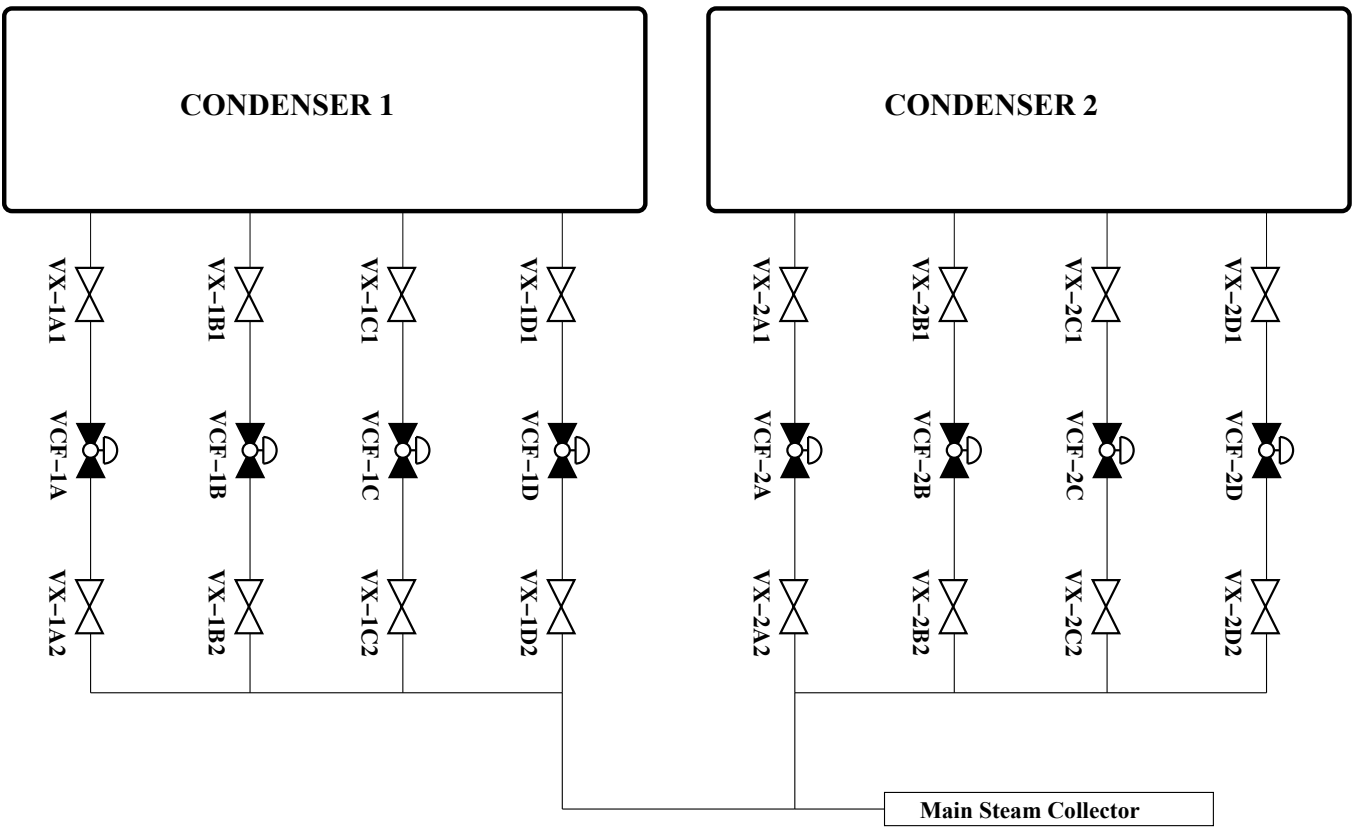


Figure A.20: SPAR-CSN diagram of the MS (Condenser relief function)

Appendix B

SPAR-CSN Event Trees

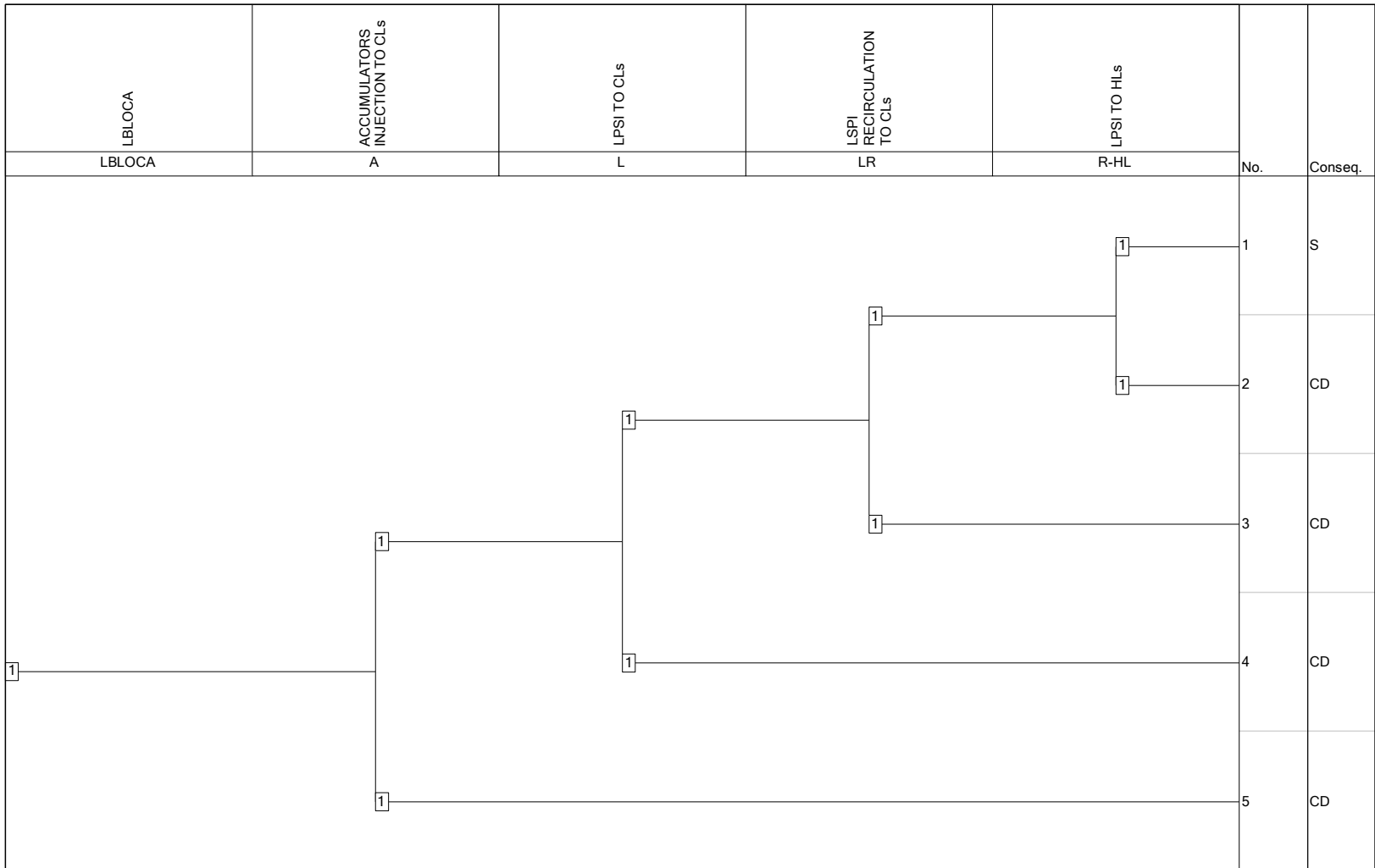


Figure B.1: LBLOCA ET in SPAR-CSN Model

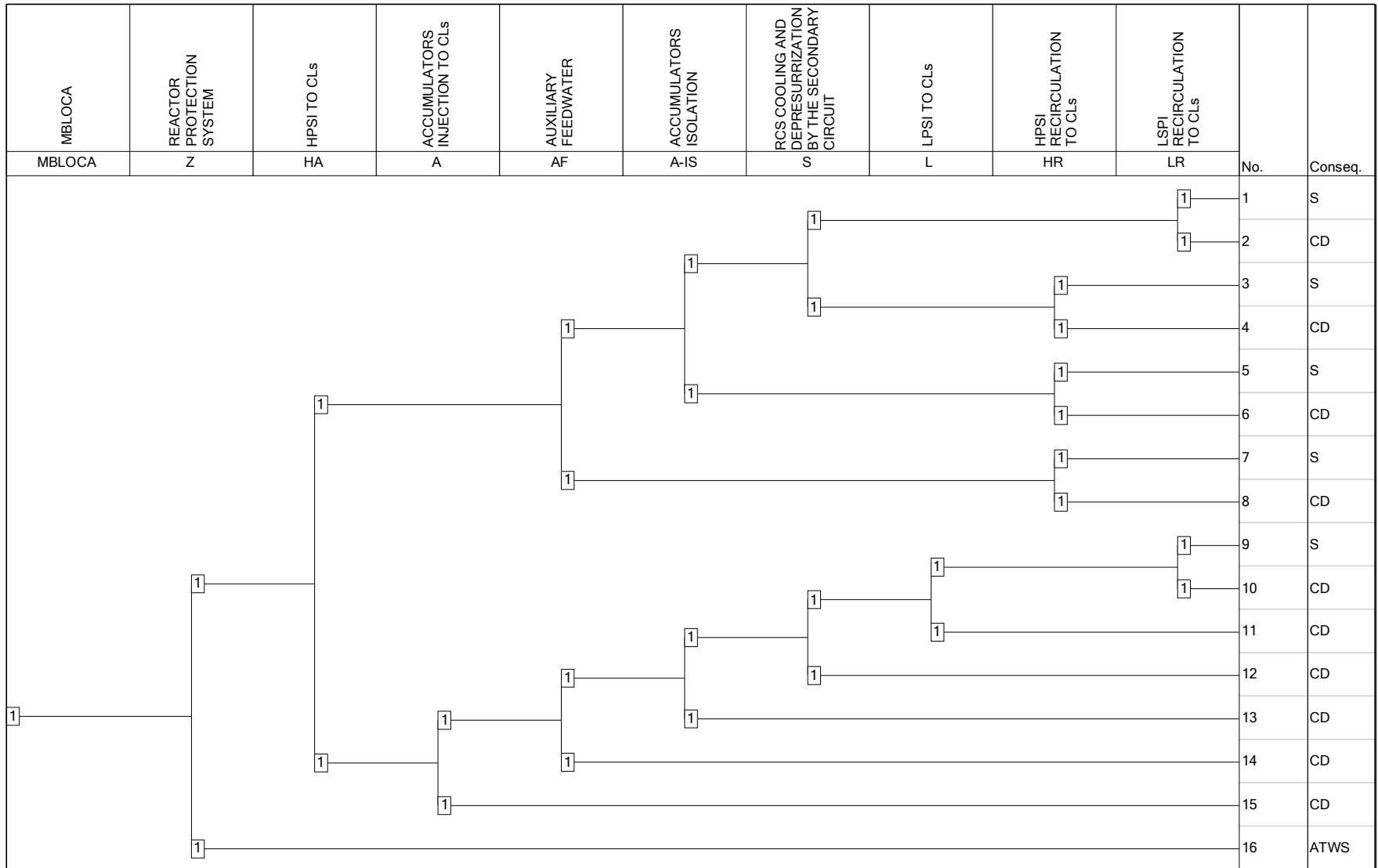


Figure B.2: MBLOCA ET in SPAR-CSN Model

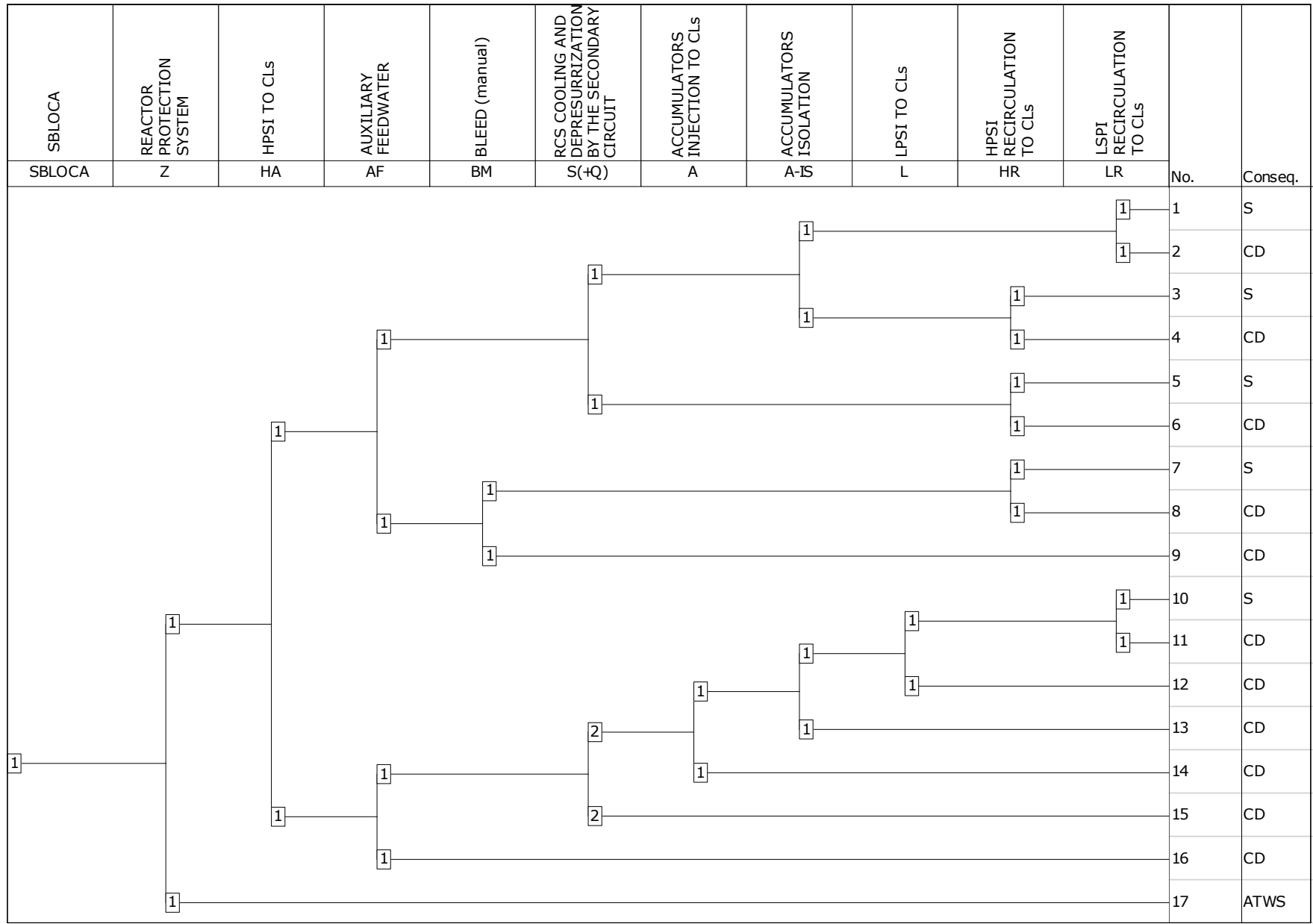


Figure B.3: SBLOCA ET in SPAR-CSN Model

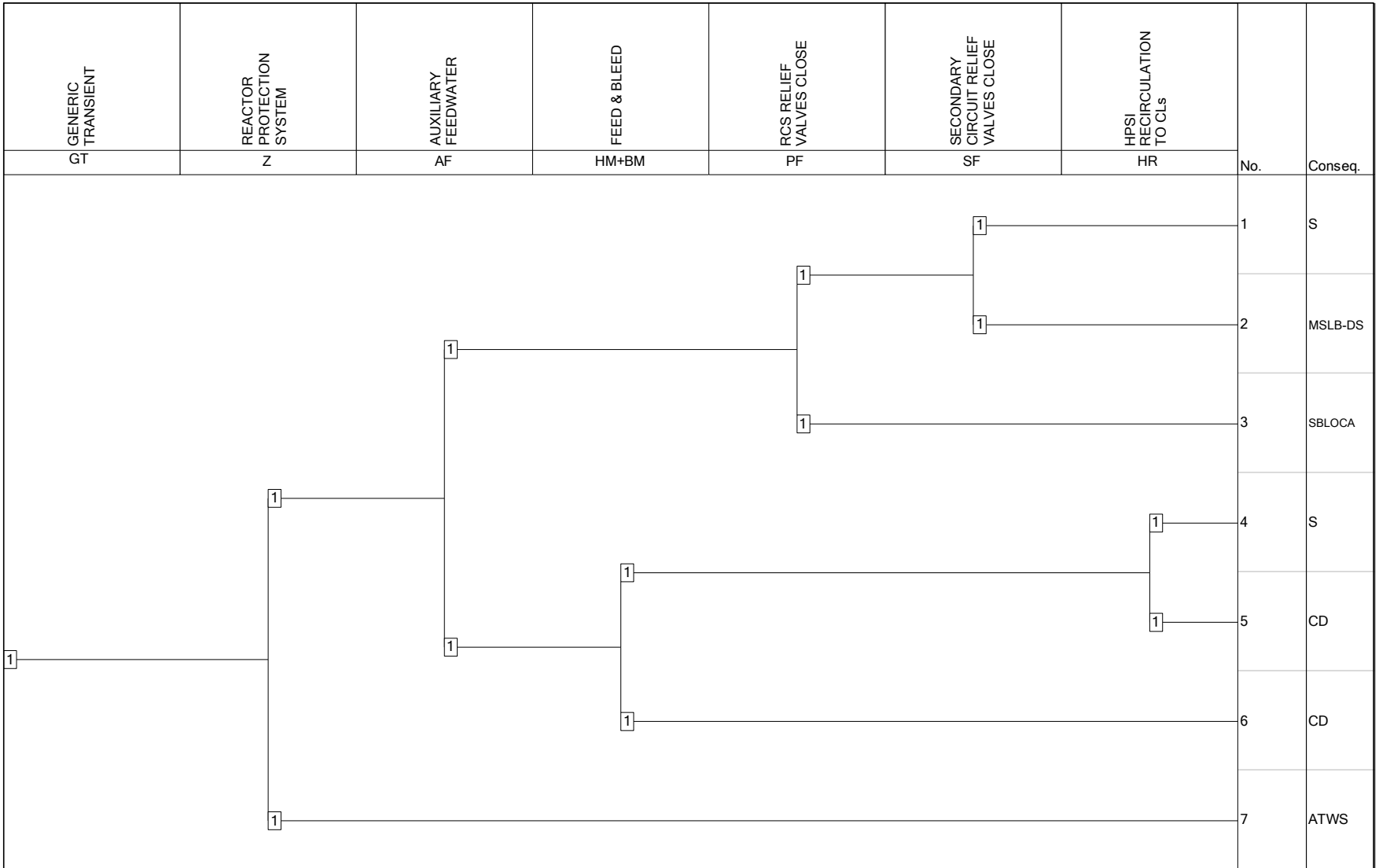


Figure B.4: GT ET in SPAR-CSN Model

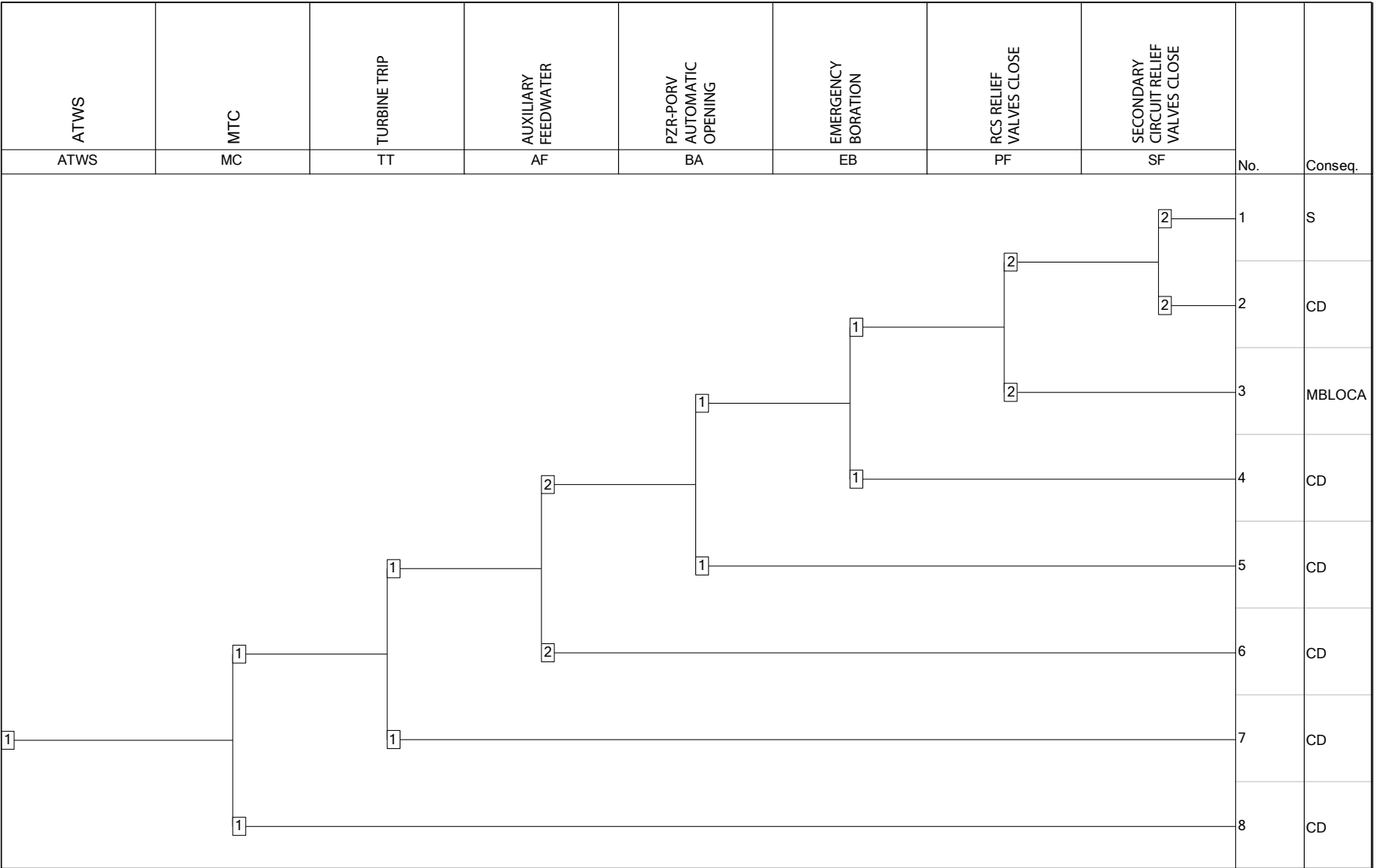


Figure B.5: ATWS ET in SPAR-CSN Model

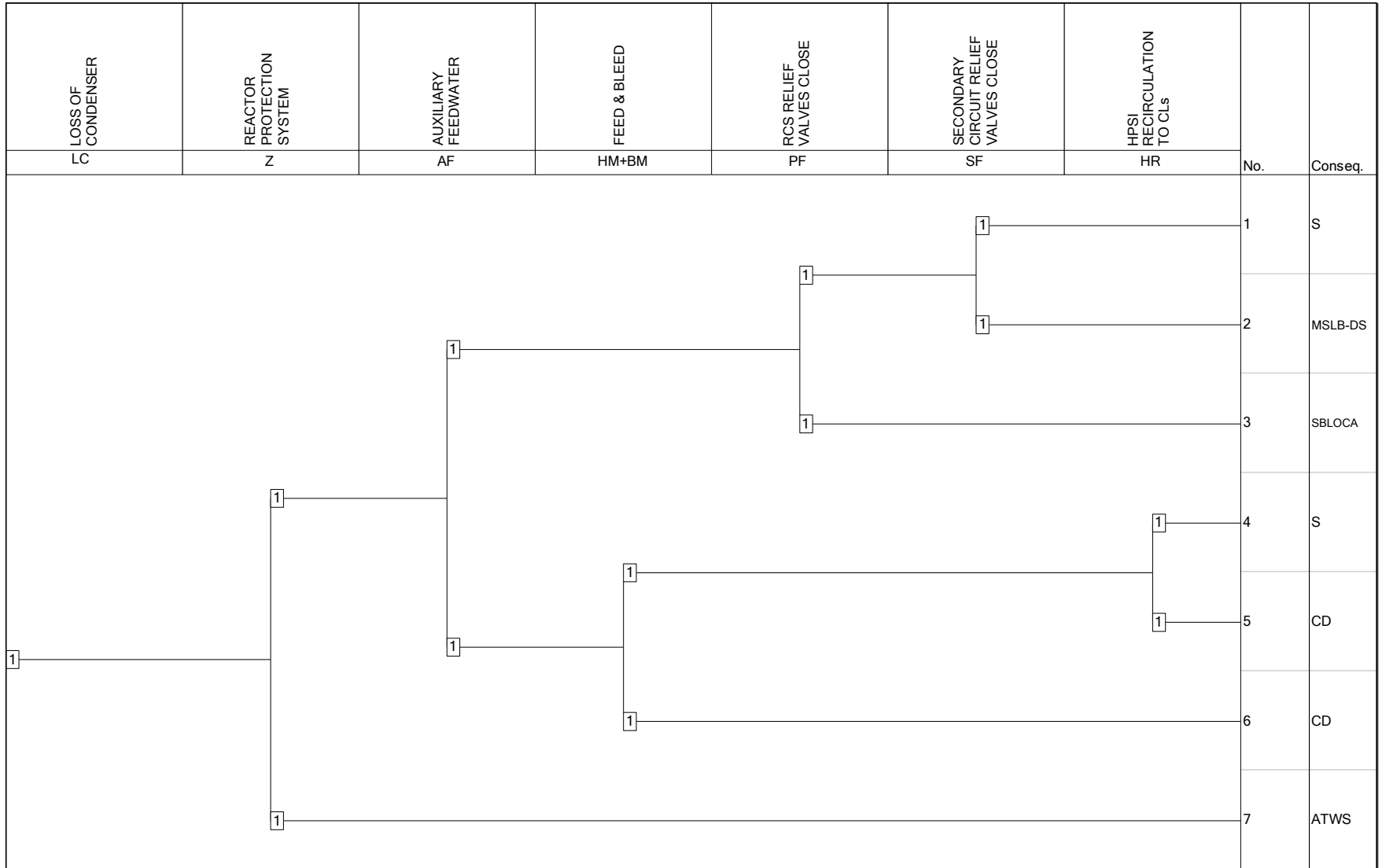


Figure B.6: LC ET in SPAR-CSN Model

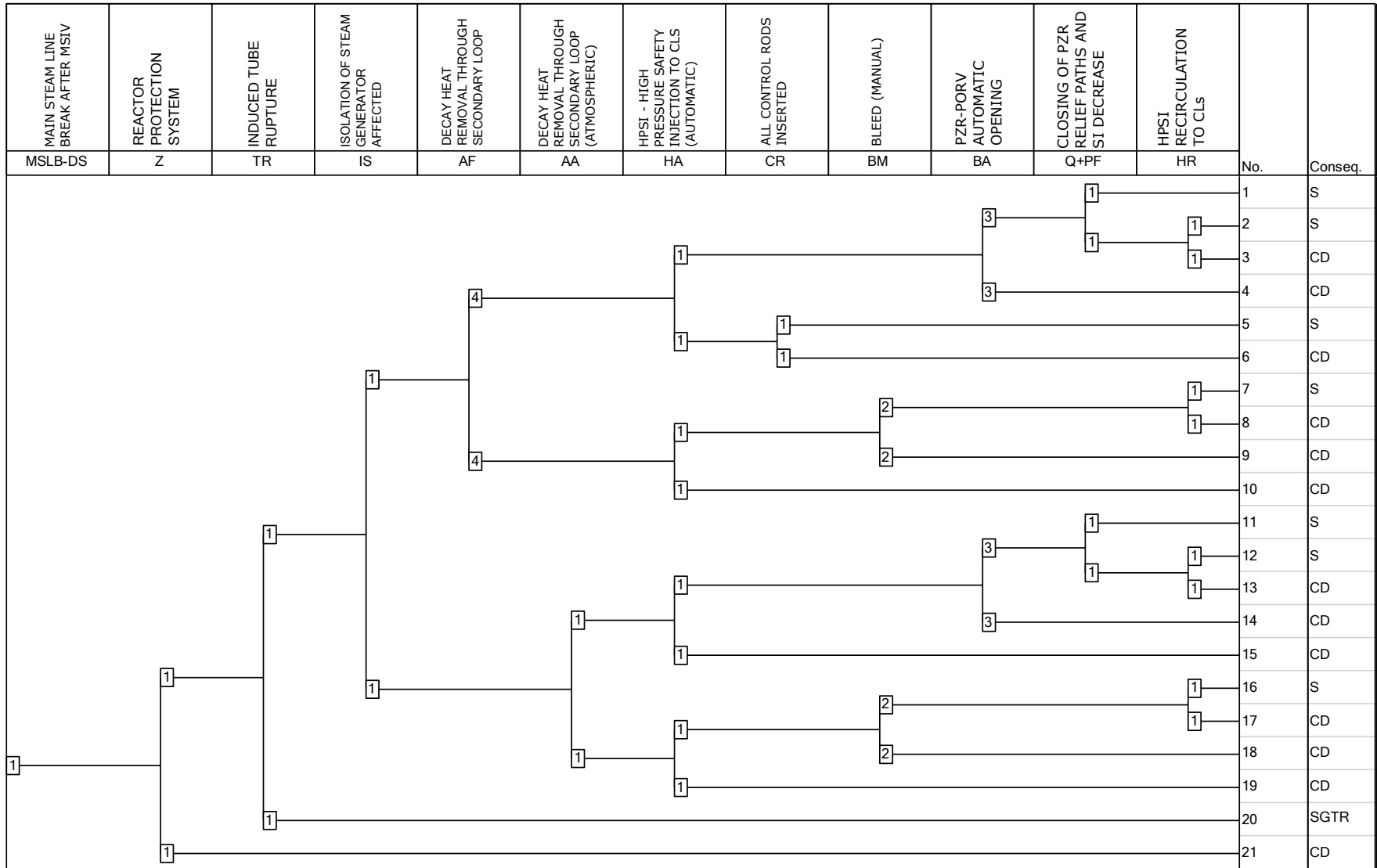


Figure B.7: MSLB-DS ET in SPAR-CSN Model

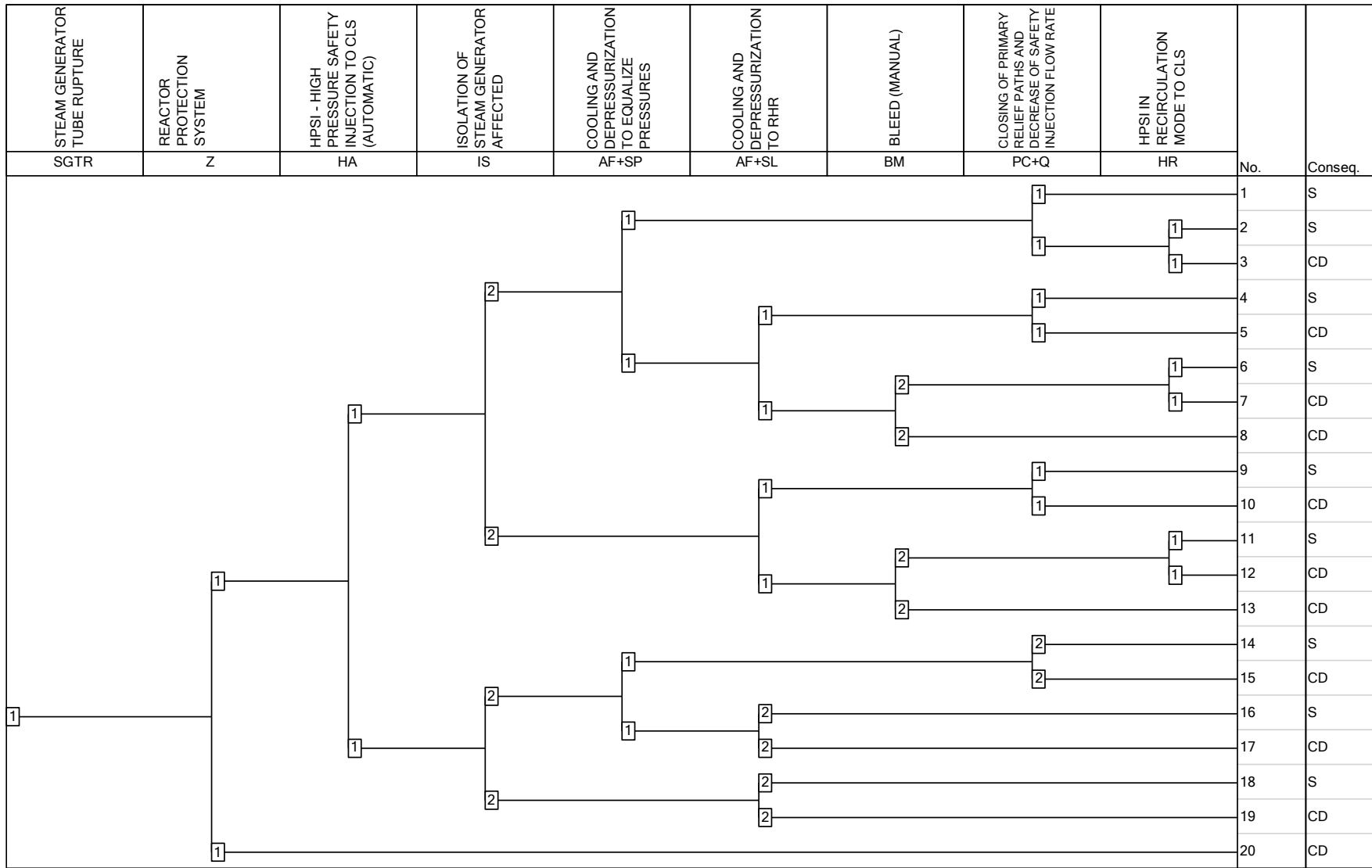


Figure B.8: SGTR ET in SPAR-CSN Model

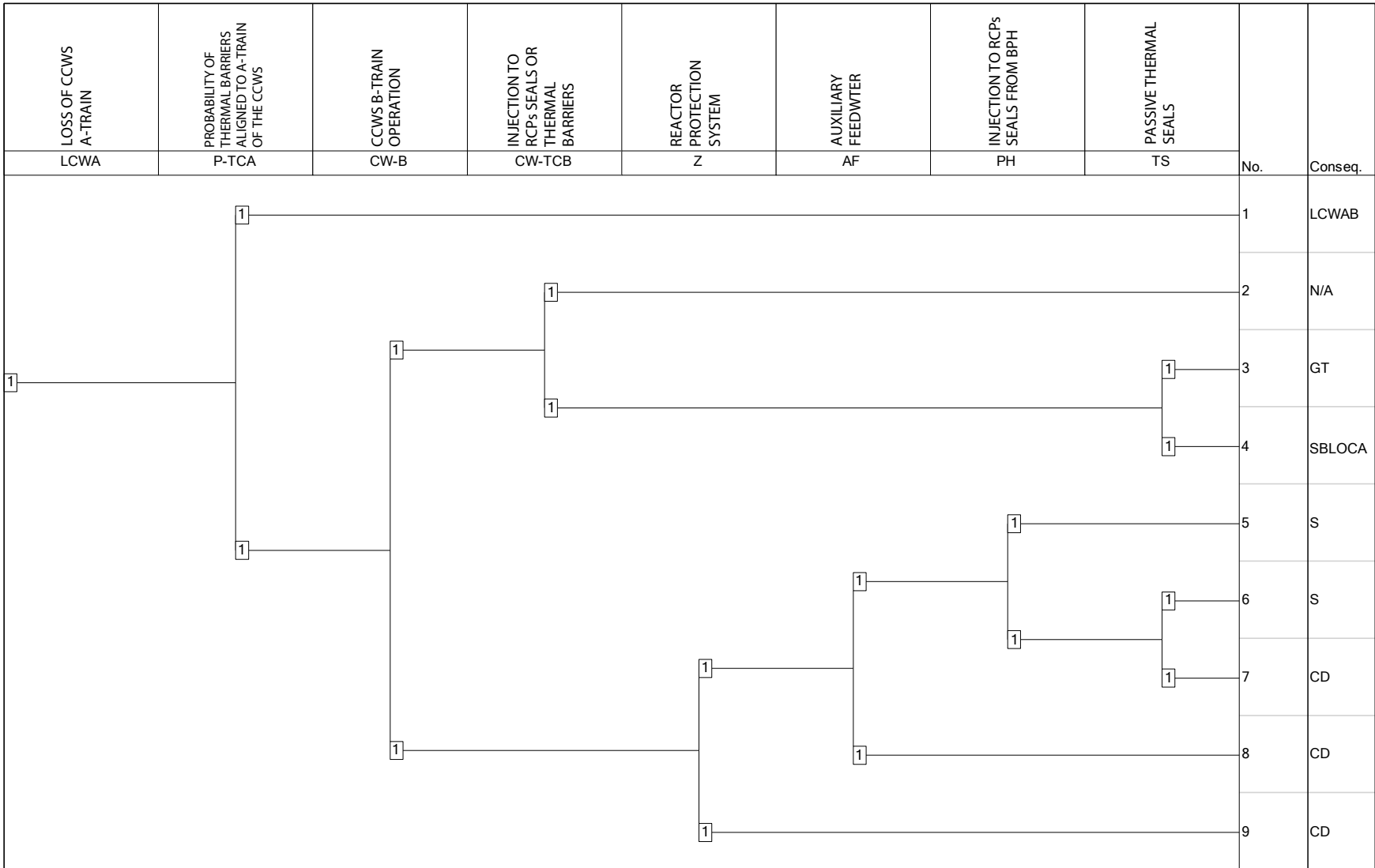


Figure B.9: LCWA ET in SPAR-CSN Model

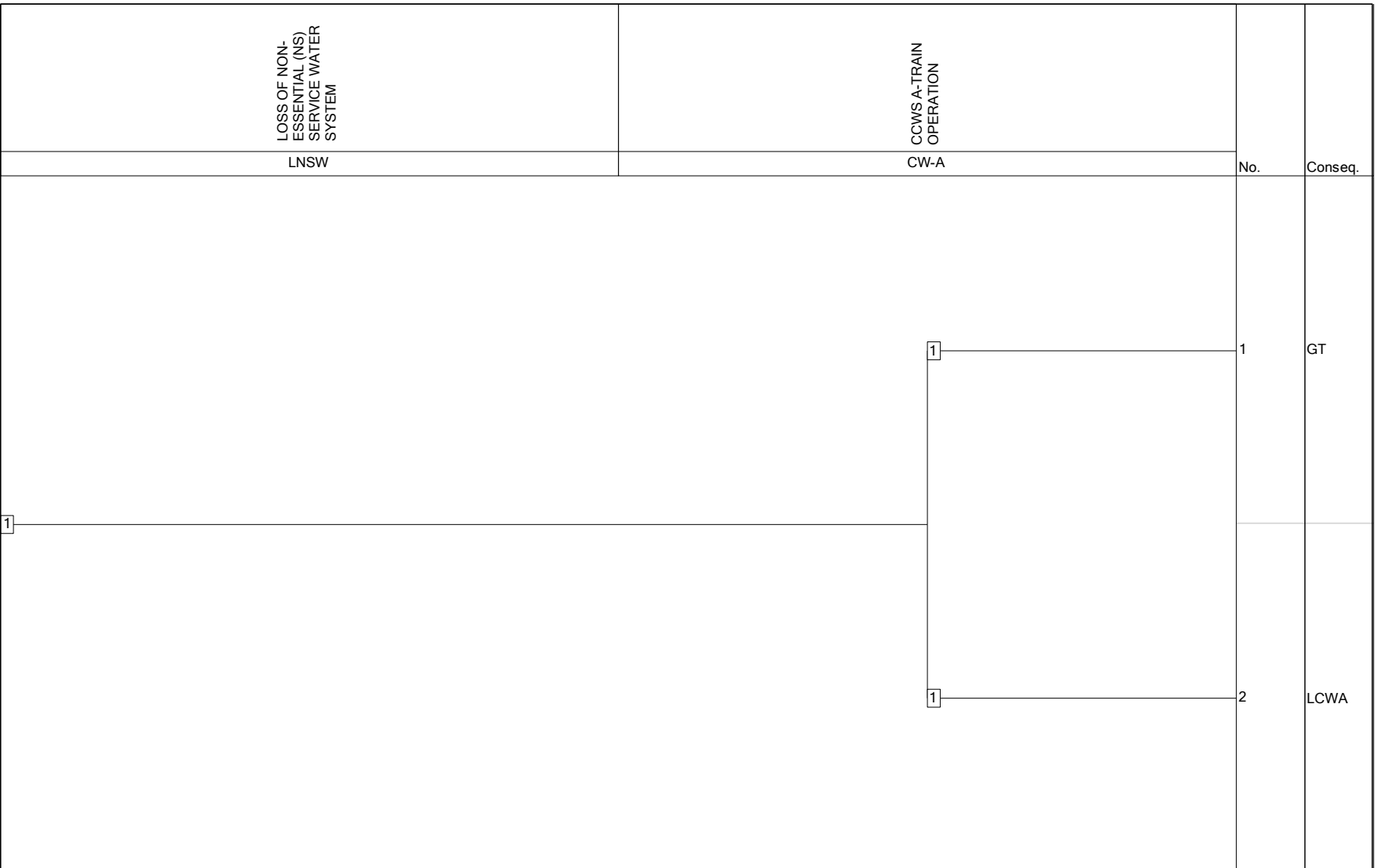


Figure B.10: LNSW ET in SPAR-CSN Model

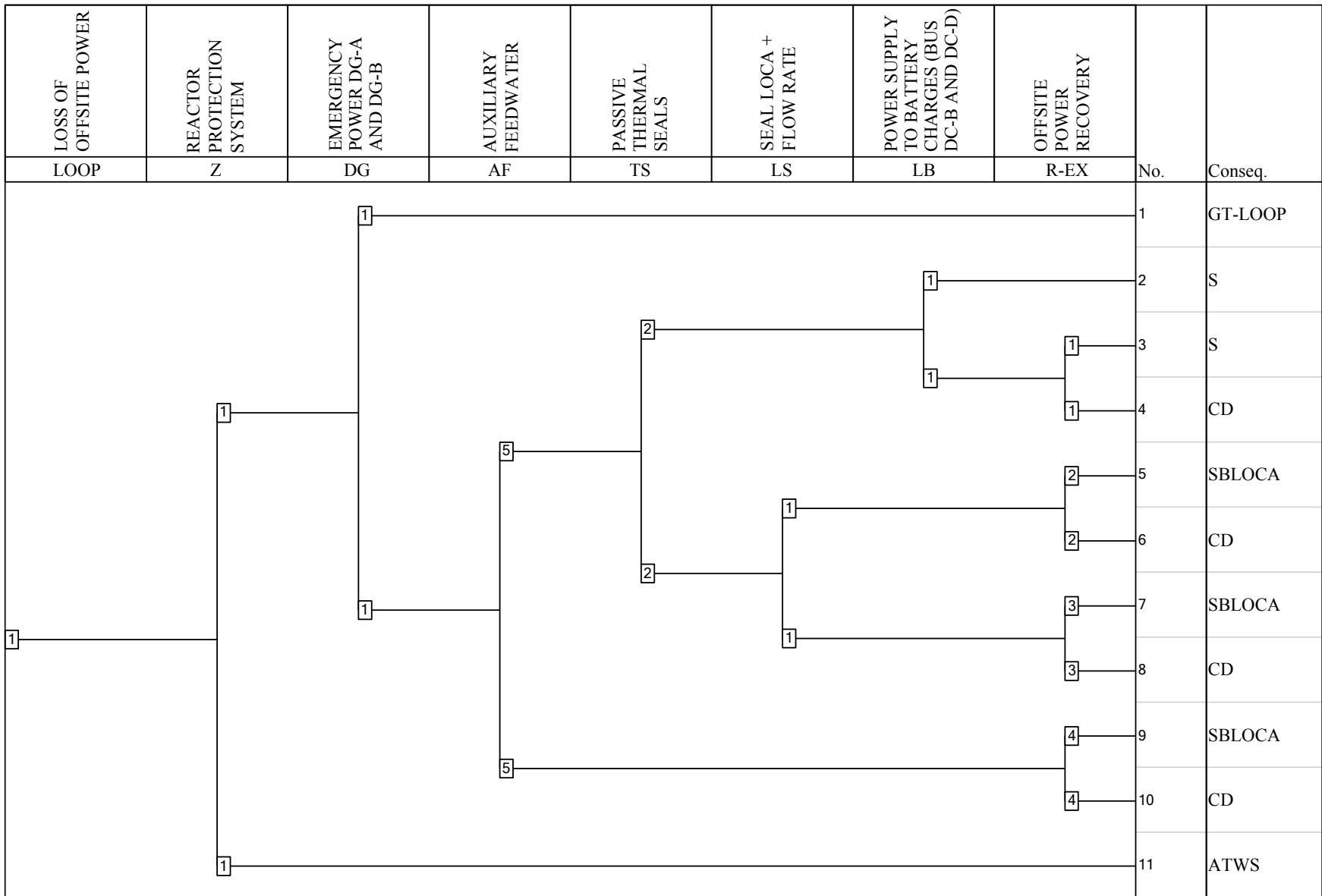


Figure B.11: LOOP ET in SPAR-CSN Model

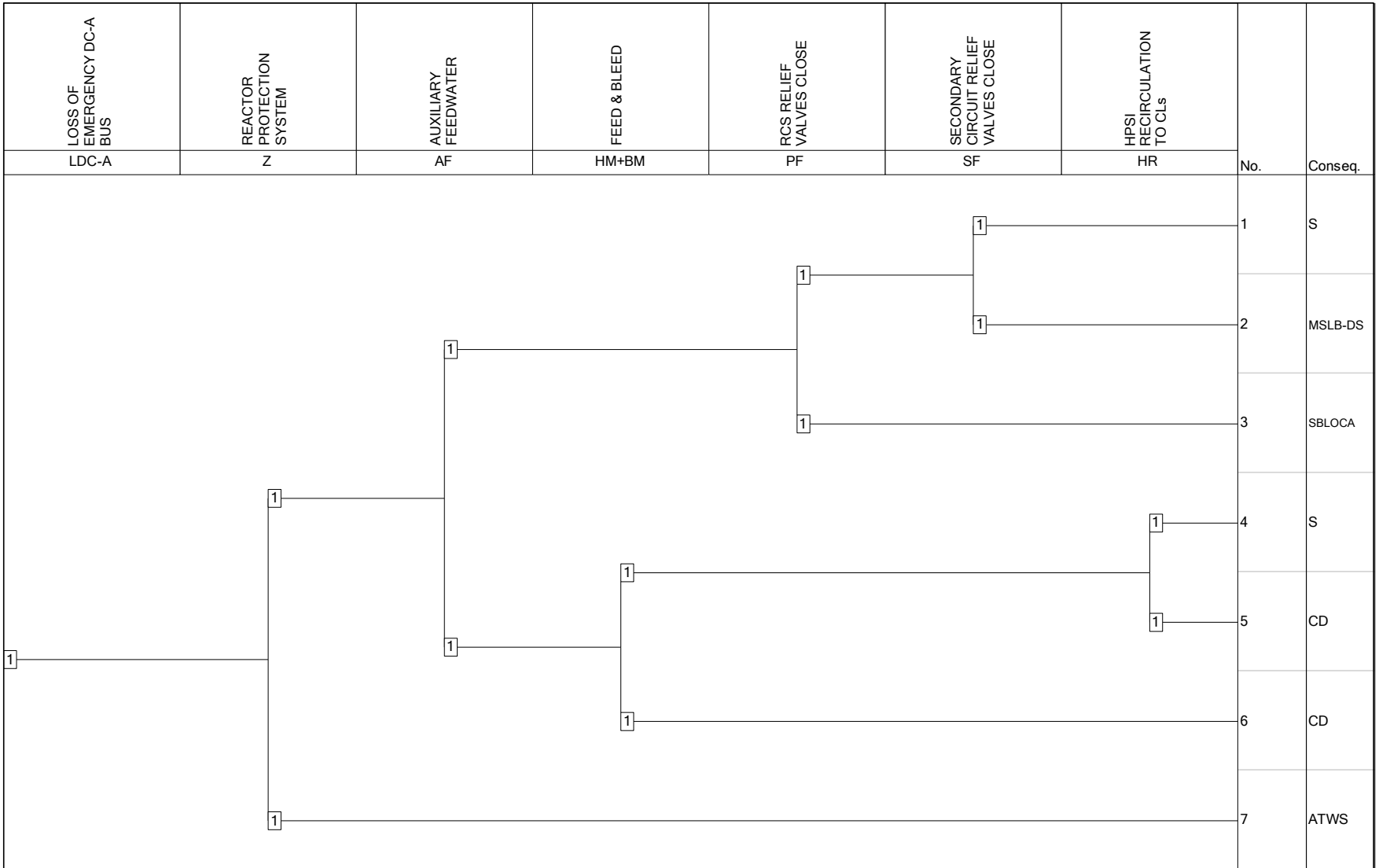


Figure B.12: LDC-A ET in SPAR-CSN Model

Appendix C

SPAR-CSN Event Trees with FLEX Strategies

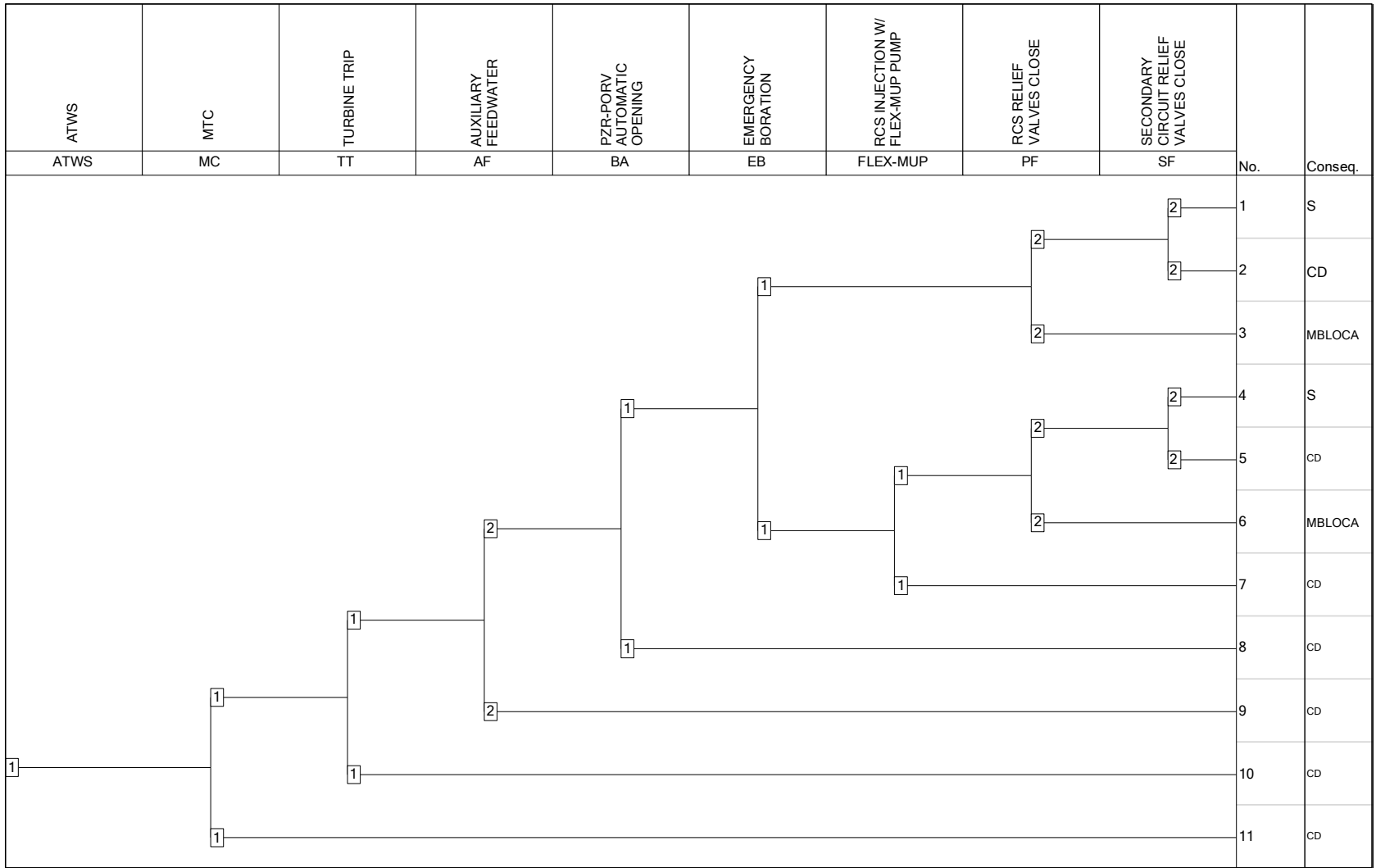


Figure C.1: ATWS ET with FLEX strategies

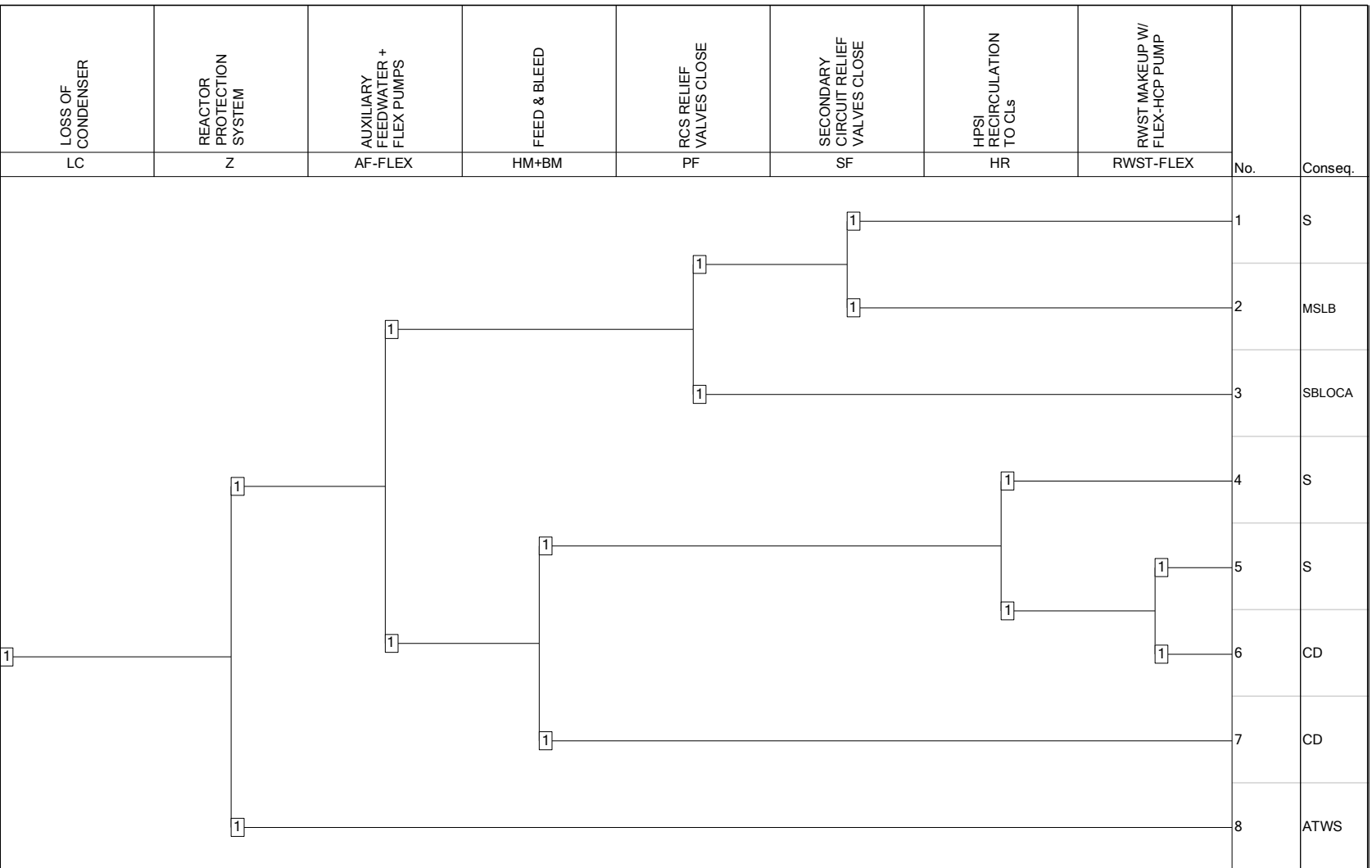


Figure C.2: LC ET with FLEX strategies

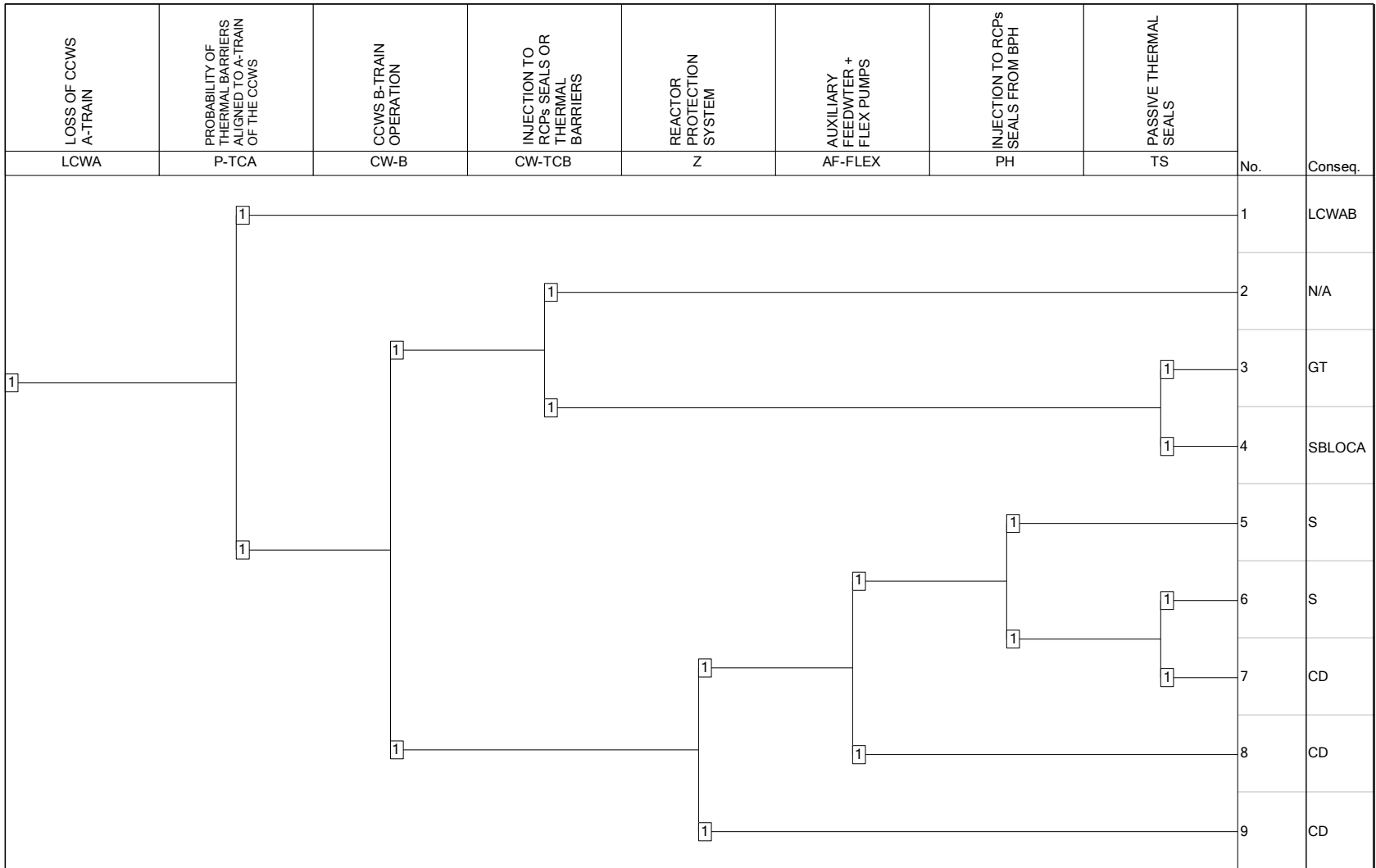


Figure C.3: LCWA ET with FLEX strategies

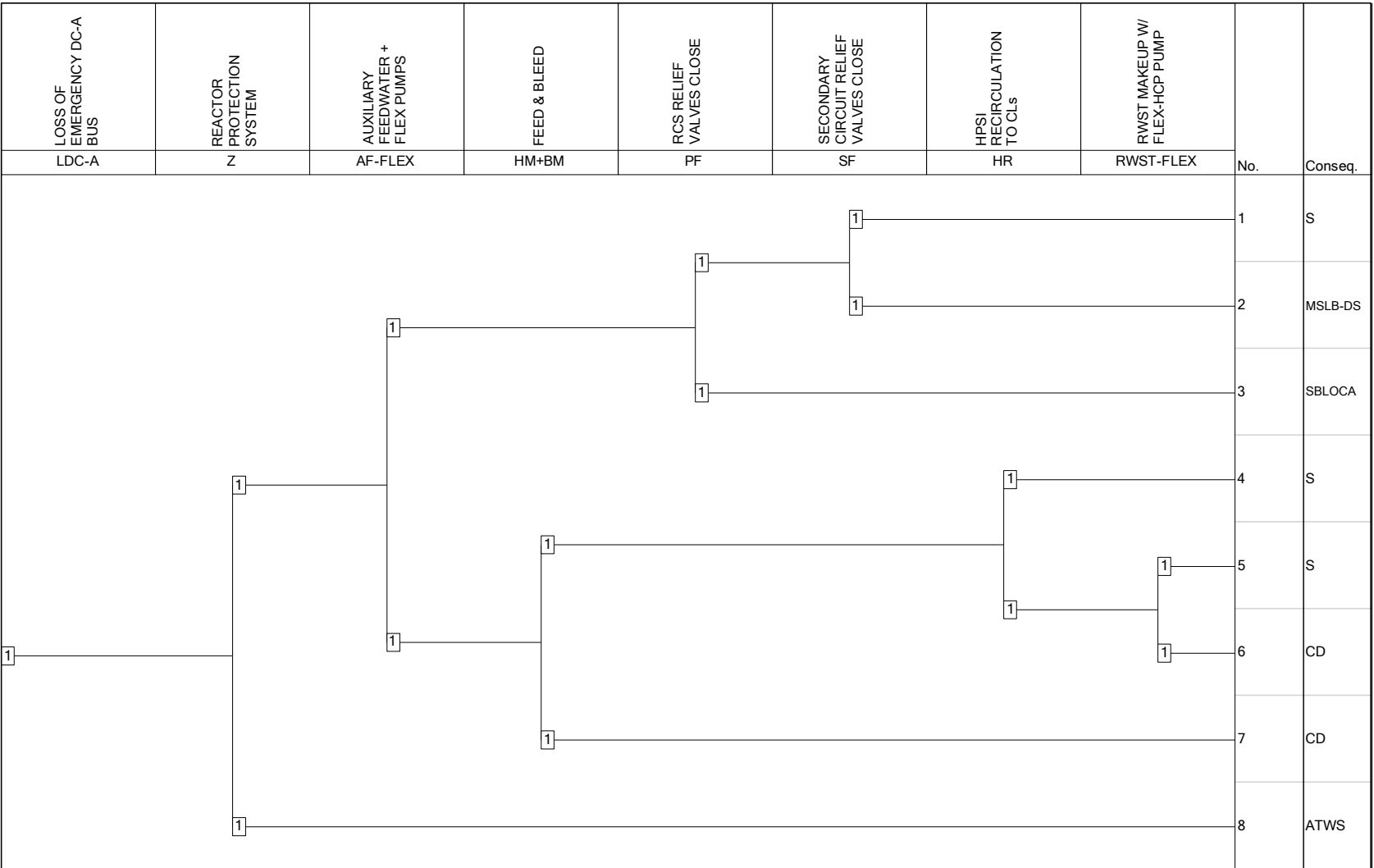


Figure C.4: LDC-A ET with FLEX strategies

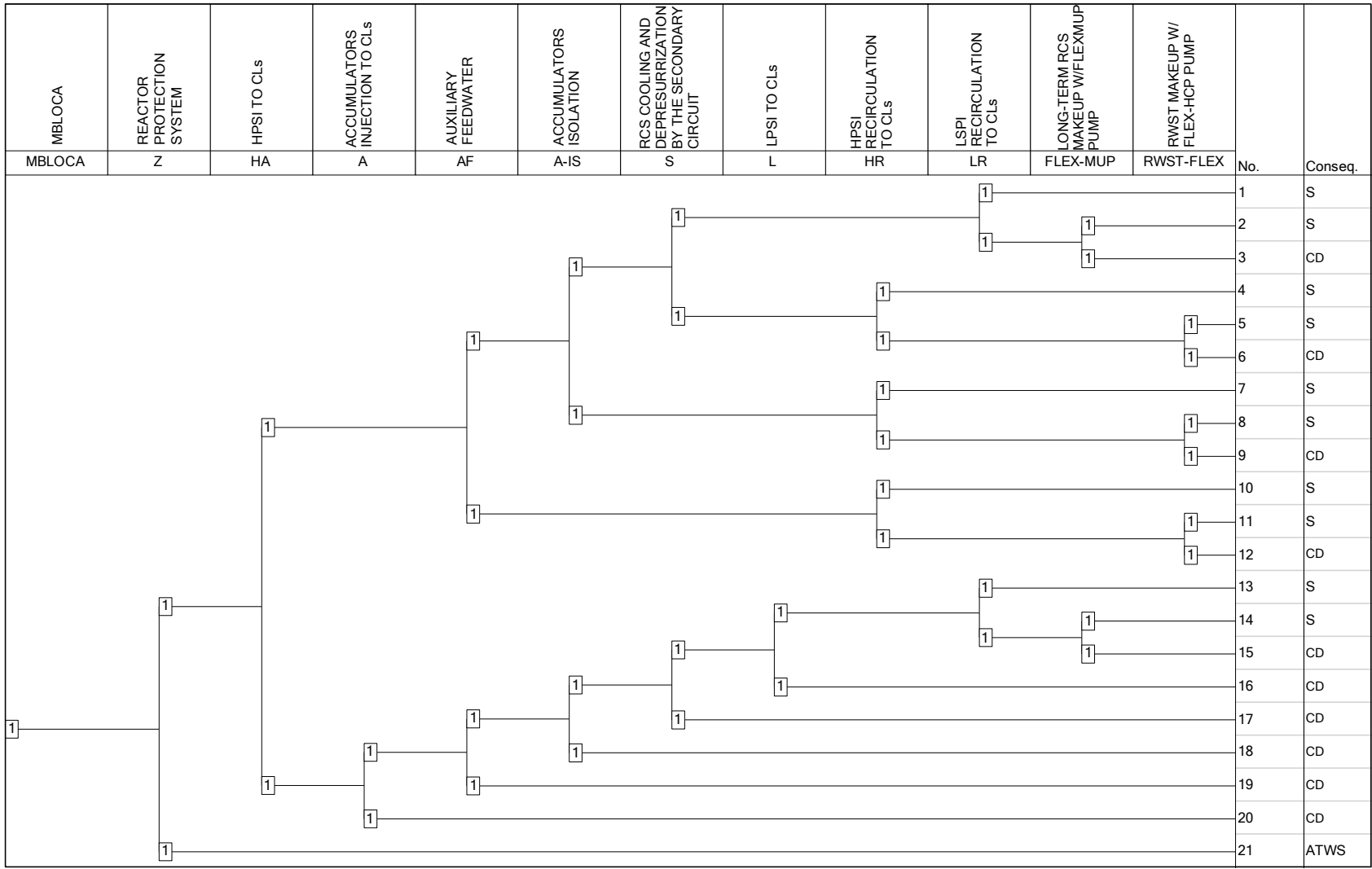


Figure C.5: MBLOCA ET with FLEX strategies

Appendix D

DEC-A Sequences Event Trees

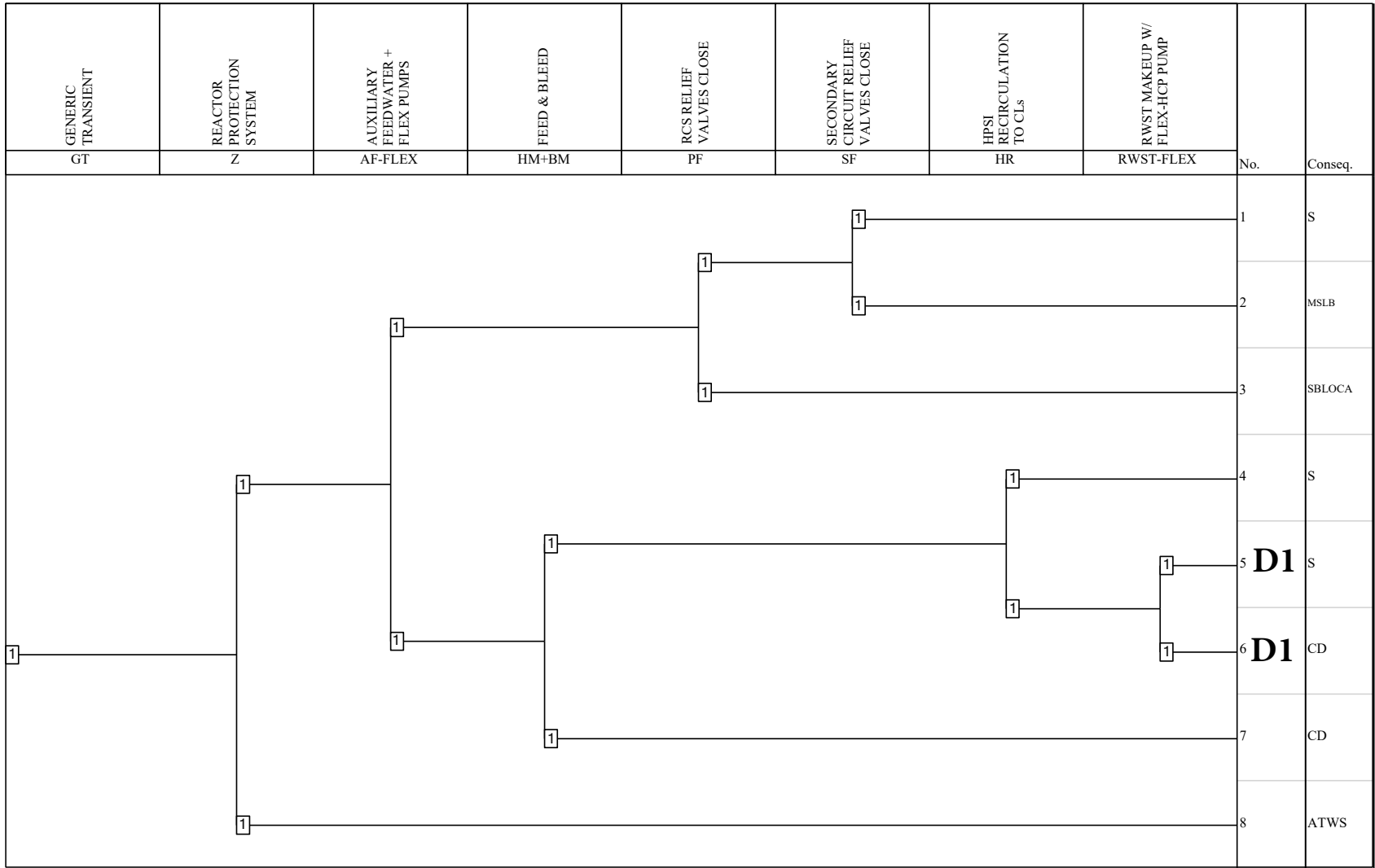


Figure D.1: DEC-A Sequences: D1 sequences

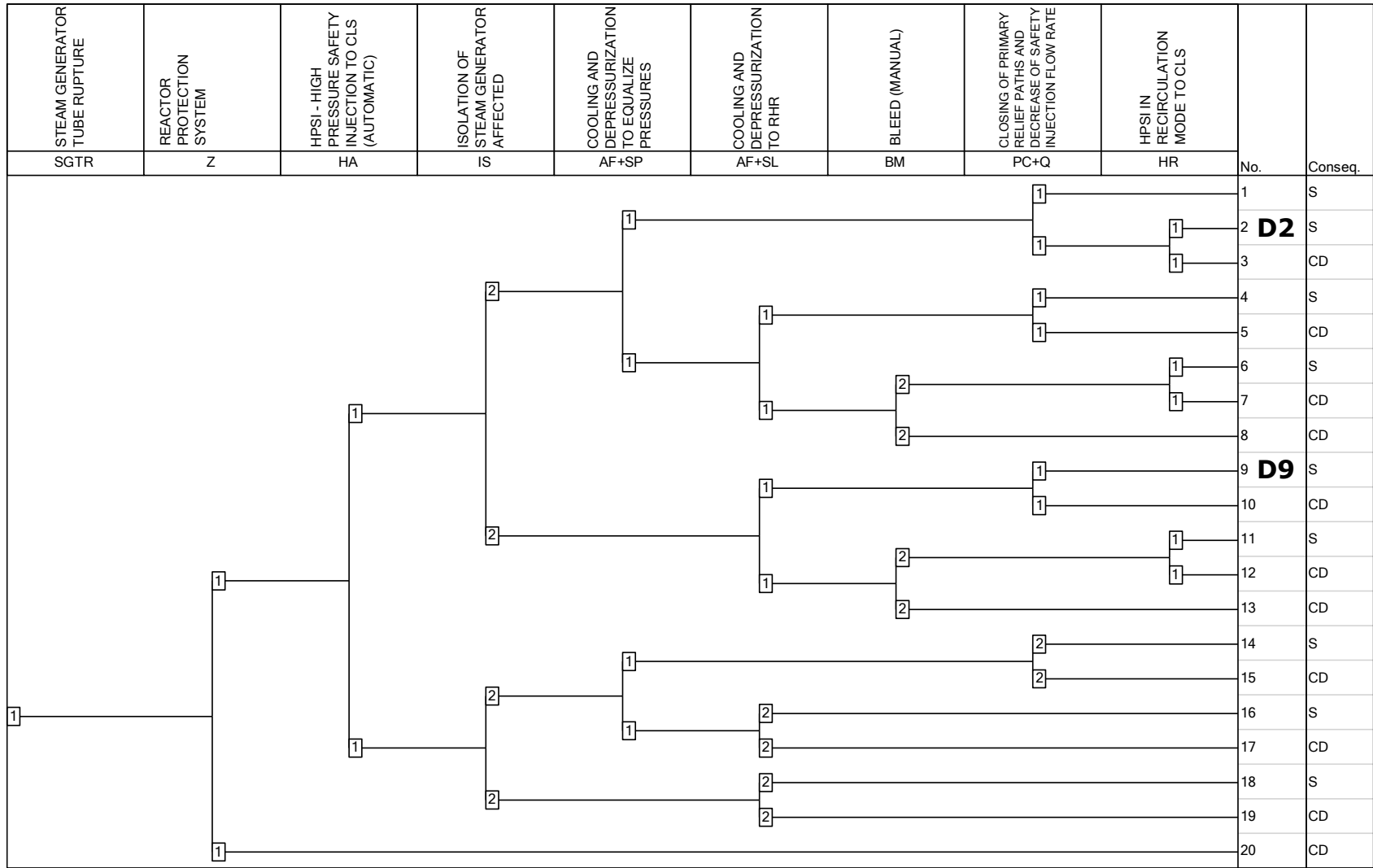


Figure D.2: DEC-A Sequences: D2 and D9 sequences

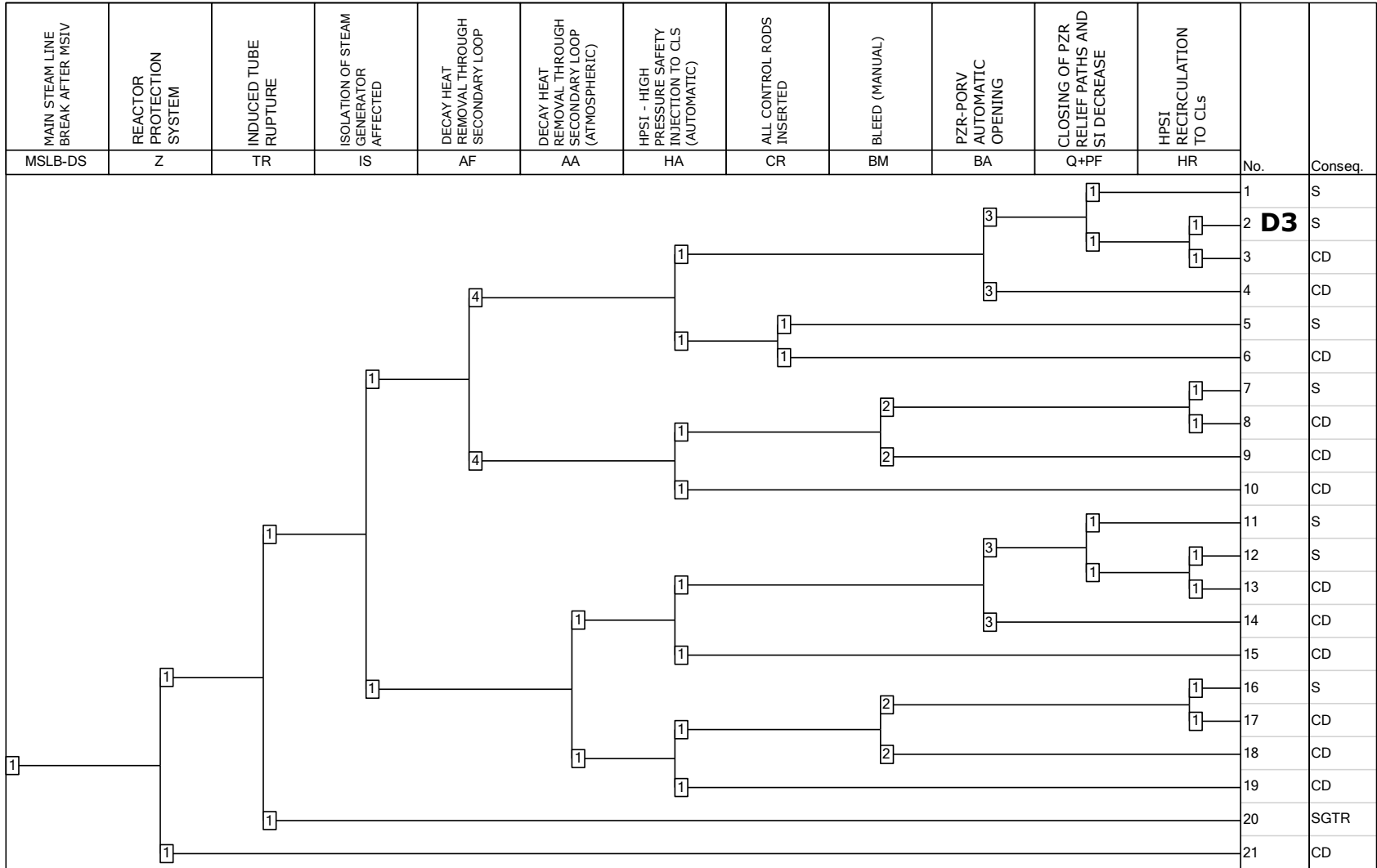


Figure D.3: DEC-A Sequences: D3 sequence

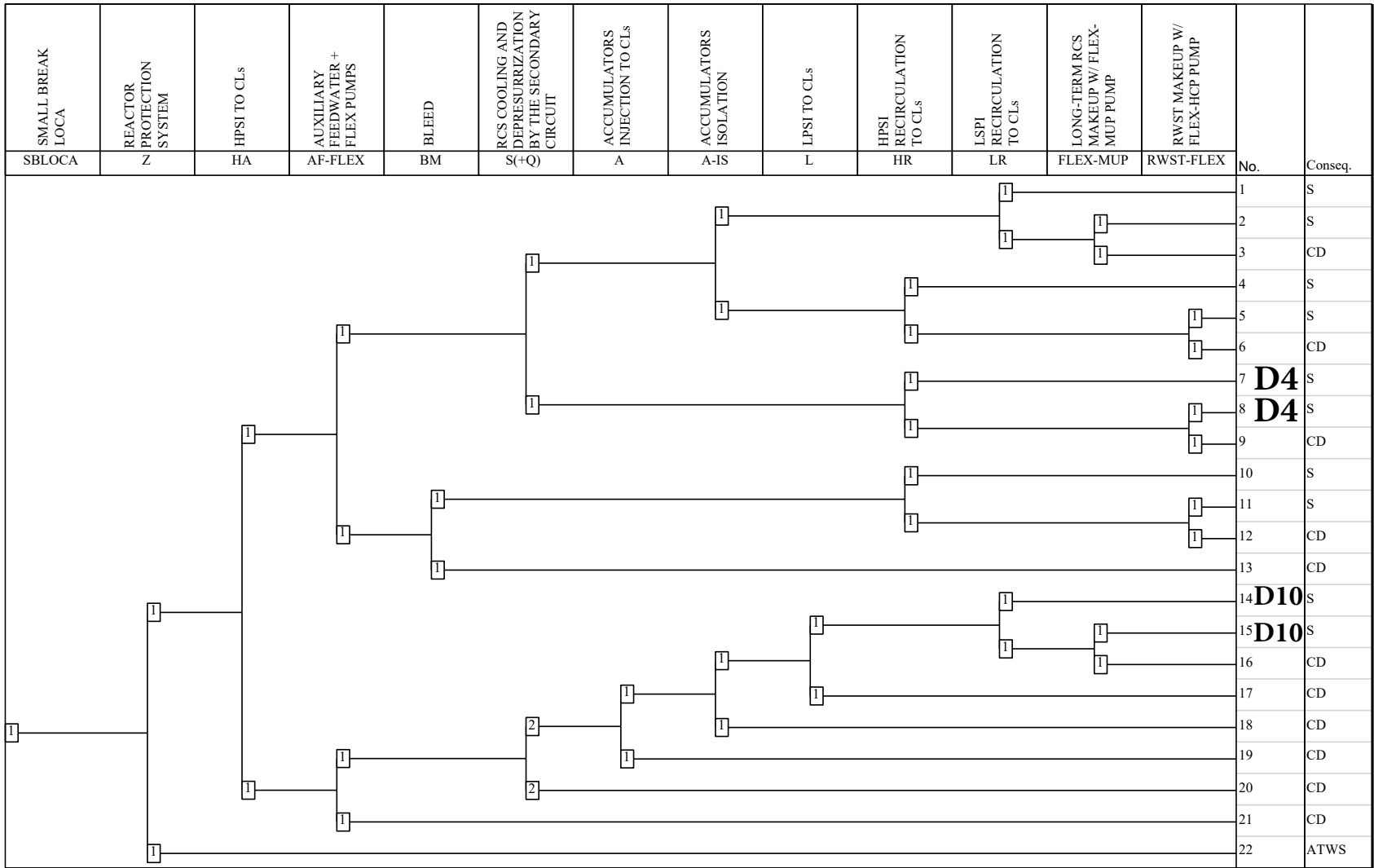


Figure D.4: DEC-A Sequences: D4 and D10 sequences

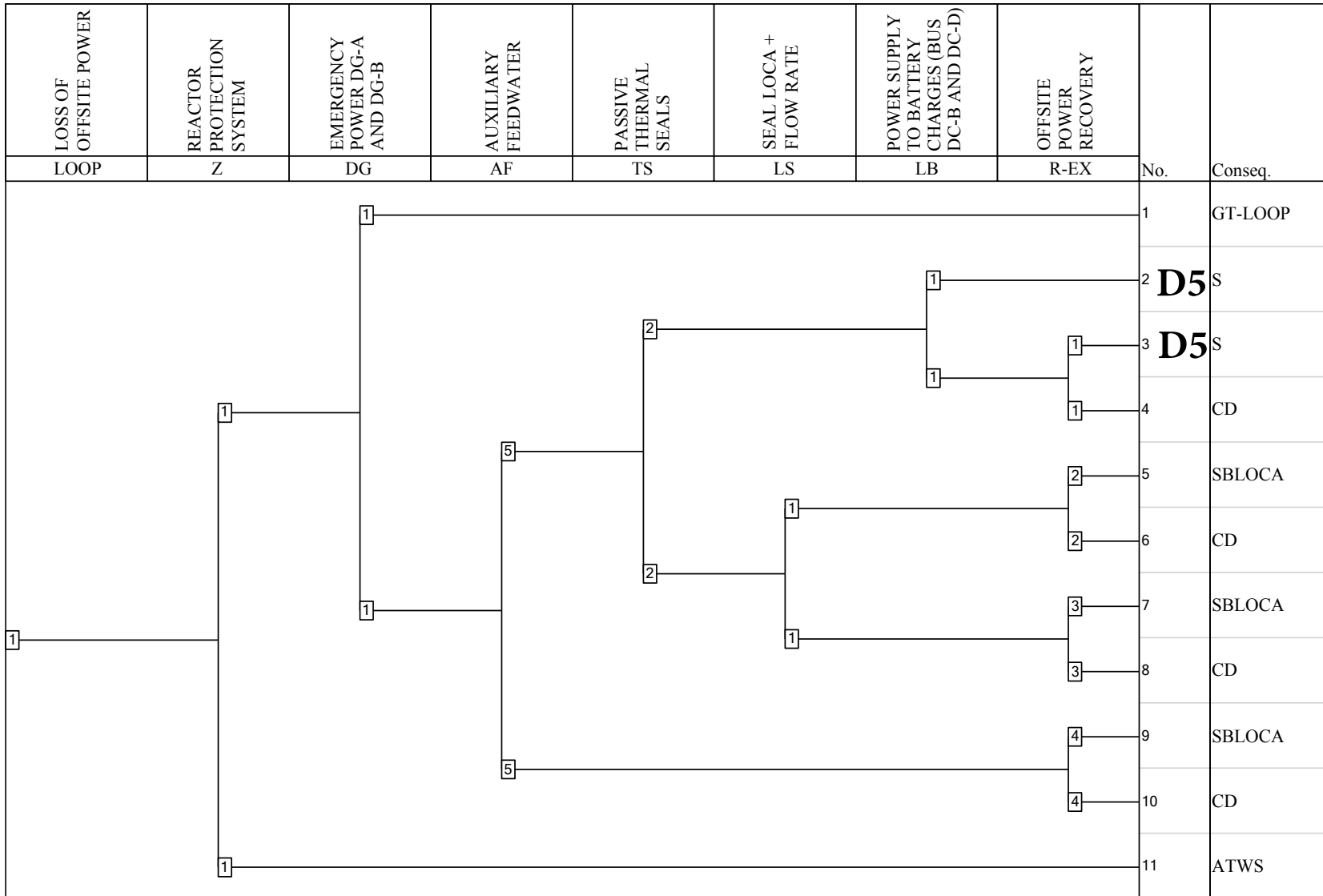


Figure D.5: DEC-A Sequences: D5 sequences

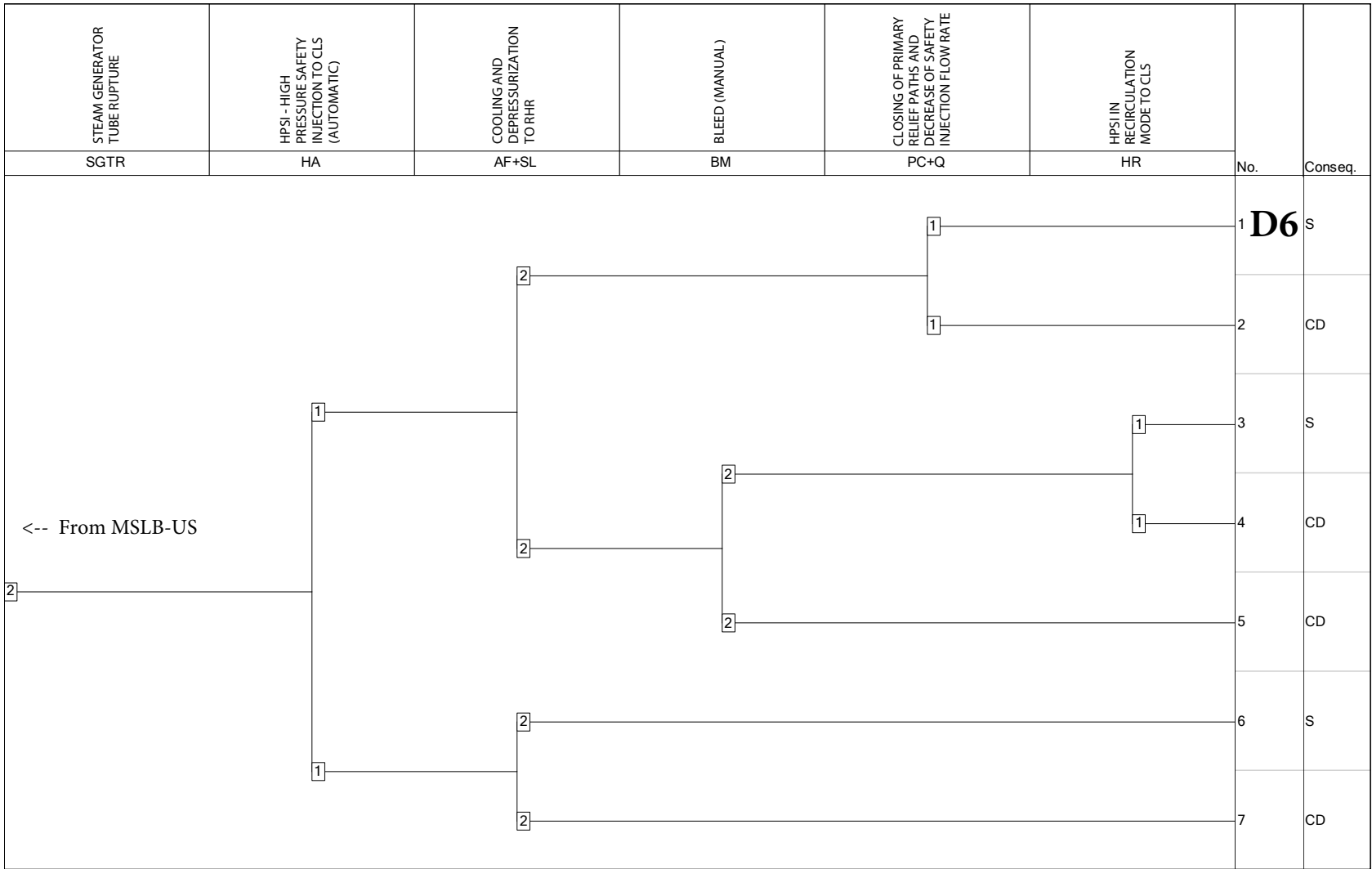


Figure D.6: DEC-A Sequences: D6 sequence

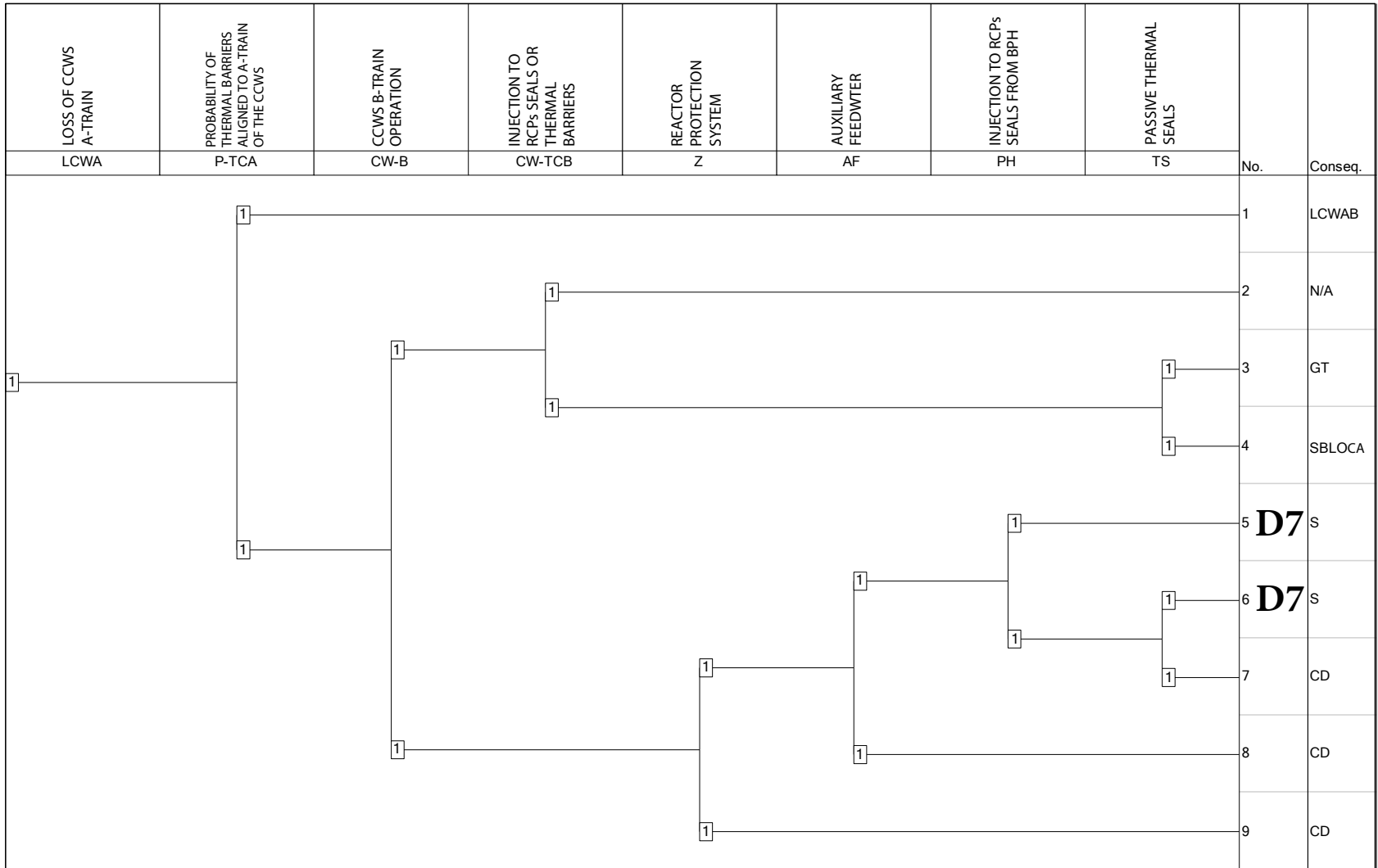


Figure D.7: DEC-A Sequences: D7 sequences

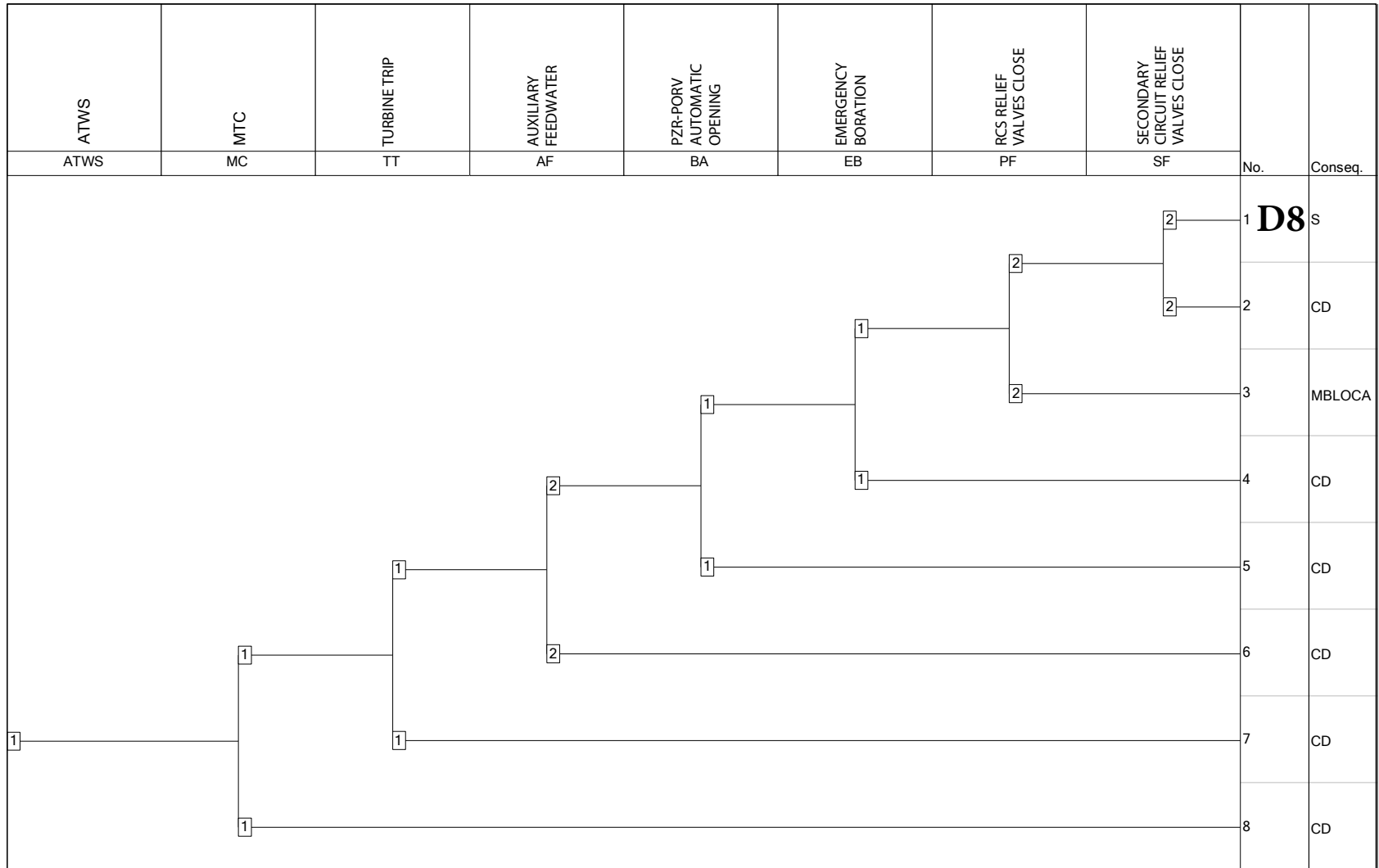


Figure D.8: DEC-A Sequences: D8 sequence