CESAR AUGUSTO GABE

Methodology for Risk Assessment of Blackout on Marine Based Nuclear Reactors

São Paulo

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Dissertation presented to the Polytechnic School of Universidade de São Paulo to obtain the degree of Master of Science.

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REVIEWED VERSION

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Advisor:
Prof. Dr. Gilberto Francisco Martha de Souza

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This work is dedicated to my daughter Júlia, my son Pedro and my wife Byanka, and to my parents who taught me the meaning of hard work and perseverance.

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“The errors of the great geniuses are more instructive than the truths of the mediocre intelligences”.

(Arthur Graf)

ABSTRACT

Nuclear power can contribute significantly to maritime transport. However, the economic and regulatory issues intimidate the deployment of nuclear powered commercial shipping. There are several discussions to present the economic feasibility of marine based small modular reactors (SMR), but no nuclear powered commercial ship will be deployed if the acceptable level of safety is not demonstrated to regulatory bodies. Risk-informed approach would be appropriate for licensing marine based small modular reactors, providing a country neutral method for reviewing safety plans. The risk-informed approach to blackout accidents permits exploring the state-of-art on dynamic reliability best estimate modeling as well as a dose based consequence analysis. There are SMRs empowered with passive and inherent safety features, but their application in the maritime context still not suitable. Moreover, the large feedback and technological readiness of Light Water Reactors (LWR) based on active safety systems can significantly reduce the duration of deployment and licensing. This work proposes a methodology to assess the risk of blackout on a marine based nuclear power plant in early design stage. The methodology is based on probabilistic safety analysis and dose exposure analysis in post-accident scenario. The methodology is applied to a hypothetical pressurized water reactor. The results are compared with a representative Generation II LWR (Surry). The methodology estimates a frequency of core damage frequency by long station blackout in 2.24×10^{-5} reactor.year for the hypothetical reactor. NUREG-1150 estimates the Surry long station blackout core damage frequency in 8.2×10^{-6} reactor.year. Regarding the environmental dose exposure, the total effective dose for the whole body at 3.2 Km (in 24 hours) is 0.34 and 8.9 Sieverts for hypothetical and Surry respectively, considering no containment failure and no early large releases. The higher likelihood of blackout on lower marine based SMR with lower redundancy level has been balanced by lower radiological dose exposition.

Keywords: Nuclear powered commercial shipping. Blackout. Risk-informed approach. Hypothetical reactor.

RESUMO

A energia nuclear pode contribuir significativamente para o transporte marítimo. No entanto, as questões econômicas e regulatórias intimidam o desenvolvimento de navios comerciais de propulsão nuclear. Existem estudos que apresentam a viabilidade econômica de reatores modulares para aplicação marítima, mas nenhum navio comercial de propulsão nuclear será comissionado se um nível aceitável de segurança não for demonstrado aos órgãos reguladores. A abordagem baseada em risco é apropriada para o licenciamento de reatores modulares para aplicação marítima, fornecendo um método internacionalmente reconhecido para revisão dos relatórios de análise de segurança. A abordagem baseada em risco para acidentes de falta de energia permite explorar o estado da arte em modelos de melhor estimativa de confiabilidade dinâmica, bem como uma análise de consequência baseada em dose radiológica. Existem reatores modulares intrinsicamente seguros ou dotados de sistemas de segurança passivos, mas a sua aplicação no contexto marítimo ainda não é adequada. Além disso, a experiência e disponibilidade tecnológica dos reatores de água leve baseados em sistemas de segurança ativos podem reduzir significativamente o tempo de obtenção e licenciamento. Este trabalho propõe uma metodologia para avaliação de risco de falta de energia em uma planta nuclear marítima, aplicável à fase de projeto conceitual. A metodologia é baseada em análise probabilística de segurança e análise de exposição à dose radiológica em cenários acidentais. A metodologia é aplicada a um reator hipotético de água pressurizada. Os resultados são comparados com uma planta representativa de Geração II (Surry). A metodologia estima uma frequência de dano ao núcleo por falta de energia em longo prazo em $2{,}24 \times 10^{-5}$ reator.ano para o reator hipotético. A NUREG-1150 estima a frequência de danos no núcleo por falta de energia em longo prazo para a planta Surry em $8{,}2 \times 10^{-6}$ reator.ano. Em relação à exposição radiológica, a dose efetiva total de corpo inteiro a 3,2 Km (em 24 horas) é de 0,34 e 8,9 Sieverts para a planta hipotética e Surry, respectivamente, considerando falha de contenção e grande liberação prematura. A maior probabilidade de falta de energia em reatores modulares marítimos é equilibrada pela menor exposição à dose radiológica.

Palavras-chave: Navios comerciais de propulsão nuclear, Falta de Energia, Abordagem baseada em risco, Reator hipotético

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LIST OF ACRONYMS

AC	Alternate Current
ALARP	As Low as Reasonable Practical
ANS	American Nuclear Society
ANSI	American National Standard Institute
ATWS	Antecipated Transient without SCRAM
AVR	Automatic Voltage Regulator
BWR	Boiled Water Reactor
CNEN	Comissão Nacional de Energia Nuclear
CSV	Comma Separated Values
CTMC	Continuous Transition Markov Chain
DC	Direct Current
DG	Diesel Generator
DWT	Deadweight Tonnage
ECCS	Emergency Core Cooling System
EDE	Effective Dose Equivalent
eDSPN	Extended Deterministic and Stochastic Petri nets
EPA	Energy Protection Agency
EPZ	Emergency Planning Zone
FSAR	Final Safety Analysis Report
GFR	Gas-cooled Fast Reactors
GSPN	Generalized Stochastic Petri Net
HTGR	High Temperature Gas-cooled Reactors
IAEA	International Atomic Energy Agency
ICRP	International Commission for Radiological Protections
IE	Initiating Event
iPWR	Integrated Pressurized Water Reactor
LCP	Local Control Panel
LMFR	Liquid Metal Fast Reactors
LOCA	Loss of coolant accident
LTSBO	Long Term Station Blackout
LWR	Light Water Reactor

MSR	Molten Salt Reactors
MTBCF	Mean Time Between Critical Failure
MTBF	Mean Time Critical Failure
MTTR	Mean Time to Repair
NMSPN	Non-Markovian Stochastic Petri Nets
NRC	United States Nuclear Regulatory Commission
NPP	Nuclear power plant
NUREG	Nuclear Regulatory Guide
PN	Petri Net
PRA	Probabilistic Risk Assessment
PRS	Partially Repairable Systems
PSA	Probabilistic Safety Analysis
PWR	Pressurized Water Reactor
PZR	Pressurizer
RASCAL	Radiological Assessment System for Consequence Analysis
RCS	Reactor Cooling System
RHR	Residual Heat Removal
SBO	Station Blackout
SCALE	Standardized Computer Analyses for Licensing Evaluation
SCRAM	Safety Control Rod Axe Man (Reactor trip)
SG	Steam Generator
SMR	Small Modular Reactors
SOARCA	State-of-Art Reactor Consequence Analysis
SPN	Stochastic Petri Net
RHR	Residual Heat Removal
TEDE	Total Effective Dose Equivalent
TG	Turbine Generator
ULCC	Ultra-large crude carriers
UPS	Uninterruptable Power Supply
US	United States
ZR	Zircalloy

CONTENTS

1. INTRODUCTION

The nuclear fission of uranium produces several radioactive elements. In a nuclear reactor, these fission products are stored on the nuclear fuel pellets within the cladding of the fuel elements. The biggest hazard from a nuclear plant to society and the environment is the uncontrolled release of these radioactive fission products from the reactor core.

After the reactor shutdown, significant energy release from the nuclear core continues for a long time. This residual heat is a result of the fission product's radioactivity and must be properly dissipated, otherwise core heat up can damage the fuel cladding, resulting in fission products release.

The fission products can leak somehow to the environment and subsequently be dispersed, especially by the air according to meteorological and geographic conditions. This dispersion can carry these fission products to inhabited locations, exposing the population to radiological doses and contamination. According to the dose magnitude, different consequences levels to human health can be expected. However, a large radioactive release to the environment will only occur in case of containment failure, the last physical barrier to contain radioactive elements.

Long-term exposure is a result of the ingestion of contaminated water, foodstuffs, milk and agricultural produce. The released fission products can also contaminate soil, water, crops and animals. Due to the slow rate of accrual of ingestion doses, immediate action is not required to protect the population from these hazards. Decisions to implement protective actions, such as relocation and food restrictions, would be based on the results of extensive radiation and contamination monitoring.

Emergency plans for accidents involving radiological releases are focused on gamma irradiation and inhalation of radionuclides. These dose expositions have a direct pathway to humans and more confidence in quantification methods.

To avoid radionuclides release on pressurized water reactors there are three physical barriers: The nuclear fuel cladding, the reactor coolant pressure boundary and the containment. To ensure the integrity of such barriers, three main groups of safety functions must be available: control of reactivity, removal of heat from the reactor and confinement of radioactive.

Most of nuclear reactor designs conceives passive control of reactivity, i.e., independent of electrical power, through reactor trip from bars falling by gravity. The confinement of radioactive elements is performed by physical structures. The heat removal is ensured by normal and by safety systems, in many cases composed of active equipment supplied by electrical energy. The outage of electric power leads to the unavailability of active safety systems, which can result in potential damage to the nuclear core due to heat-up. The confinement of radioactive material could not be ensured in case of heat is not properly dissipated.

The electrical power supply is a key function to ensure the safety of nuclear power plants, in special to plants with a high dependency on active safety systems.

1.1 NUCLEAR POWER ON COMMERCIAL SHIPPING

Nuclear power generation is stable in terms of availability and cost when compared to other renewable energy sources, and is relatively cheap (FREIRE; ANDRADE, 2015). It also presents historically fewer accidents, environmental disasters and human health problems (STRUPCZEWSKI, 2003) and (RASHAD; HAMMAD, 2000). Several studies emphasize the use of nuclear energy to reduce climate change and environmental impacts created by energy generation (HONG, 2013, 2014a, b, 2015).

Maritime transport is a major source of air pollution, according to (GRAVINA, 2012), it is responsible in 2007 for 2.7% of CO_2, 4 to 9% of SO_x and 15% of NO_x of global emissions (EYRING, 2010). Also, 95% of international trade is maritime (ROYAL ACADEMY OF ENGINEERING, 2013). In addition, 35% of the cost of the sea freight rate for a merchant ship of 10,000 dwt over a 20-year service life is due to fossil fuel consumption (UNCTAD, 2012).

Given the relevance of maritime transport and the advantages of nuclear power, it is natural the idea of using nuclear power to propel merchant ships, a solution widely used in military vessels. Despite of anti-nuclear feeling, there is currently a positive experience of about 700 naval reactors-years worldwide (ROYAL ACADEMY OF ENGINEERING, 2013) for military use, of which more than 200 reactors are still operating in 2012 (GRAVINA et al., 2012). It is also important to note that US Navy nuclear warships are welcome at 150 ports in 50 countries around the world (O'ROURKE, 2010).

Unfortunately, nuclear-powered civil merchant ships have not developed beyond a few experimental ships. The N.S. Savannah commissioned in 1962, is evidence of the technical feasibility of nuclear power applied on merchant ships. N.S.

Savannah was considered expensive to operate as a merchant ship. The German-built Otto Hahn, a cargo ship and research facility operated for over 10 years without any technical problems. It was again considered too expensive to operate and was converted to diesel. The Japanese Mutsu was dogged by technical and political problems. Sevmorput was a Russian carrier with icebreaking that operated successfully since commissioned in 1988. These ships suffered from the capital costs of specialized infrastructure only for them. As the experiments were unique, sustaining a costly infrastructure for one ship of the class was not economically viable. A larger fleet could share fixed costs, reducing operating costs. But the most important learning from the historical development is that these experimental ships constitute the technical feasibility proof of marine nuclear power application.

1.2 MARINE BASED REACTOR

Nuclear power plants can be classified according to design evolutions and technology. (HIRDARIS, 2014) presented the following classification largely recognized:

- *Generation I* are the early prototypes and first-of-a-kind reactors built in the 1950s and 1960s;
- *Generation II* reactors have been built from 1960s - 1990s, they utilize low enriched uranium fuel with light water as coolant and moderator. These plants are therefore designated as Light Water Reactors (LWR).
- *Generation III* are advanced LWR type, but still dependent on active safety systems;
- *Generation III*+ reactors that add incremental improvements with enhanced levels of safety and security. Passive safety features are present in these reactors;
- *Generation IV* reactors are different from current designs, they include reactors cooled by lead, sodium, molten salt, supercritical water and helium. These advanced reactors use various nuclear fuel types including oxide, nitride, carbide, and metal, and can be based on uranium, plutonium, and thorium.

In recent years, there has been widespread interest in small modular reactors (SMR). SMRs offer the advantage of lower initial capital investment, scalability, and siting flexibility at locations unable to accommodate traditional larger reactors. These small reactors generate up to 300 MWe of electricity, which is enough power even for ultra-large crude carriers (ULCC) ships. (CARLSON et. Al., 2011) studied the nuclear propulsion for different ships and the maximum installed power has been estimated at 78MWe for a ULCC with a capacity over 1,000,000 DWT at a speed of 16 knots.

There are a very wide variety of SMR designs with distinct characteristics that are being developed, several of them of interest in the marine context. Complete report information on SMR development status can be found on (ARIS, 2020).

The SMR designs are mainly *Generation III+* and *Generation IV* reactors. PWRs and iPWRs Small Modular Reactor are considered *Generation III+* reactors. The *Generation IV* reactors are the High Temperature Gas-cooled Reactors (HTGRs), Molten Salt Reactors (MSRs) and fast reactors type Liquid Metal Fast Reactors (LMFRs) and Gas-cooled Fast Reactors (GFRs).

In *Generation III+* nuclear reactors, safety systems independent of electric power and taking advantage of natural circulation are already developed. However, the integration of some passive features might not be suitable for marine applications. The characteristic of nuclear plants that let nuclear reactors unusable for ships are mainly the large quantity of water for passive removal of decay heat, the height of the reactor and coolant water circulation interruption by the rolling motions of a ship (HASS, 2014). The passive safety features of Small Modular Reactor (SMR) like NuScale pressurized water reactor or the Toshiba 4S liquid metal-cooled reactor might not be suitable or economical for ships (HAAS, 2014).

(HIRDARIS, 2014) also investigated the practical marine applications for small modular reactors. The research intended to produce a concept tanker-ship design, based on a 70 MWt reactor. (HIRDARIS, 2014) reviewed past and recent work in the area of marine nuclear propulsion and describe a preliminary concept design study for a 155,000 DWT. The Gen4Energy power module, a small fast-neutron reactor, able to operate for ten full-power years before refueling, and in service last for a 25-year operational life of the vessel.

(HOQUE, 2018) make a comparative analysis of small modular reactors for nuclear marine propulsion. The main experience in operating nuclear power plants has been in nuclear naval propulsion, mainly aircraft carriers and submarines (HOQUE, 2018). The technical feasibility of *Generation II* reactors for marine application had been evident in the naval reactors already built and operated. *Generation II* reactors have technological readiness and large feedback, these features will significantly reduce the expected duration for licensing those SMRs. HTGRs and fast neutron reactors are under design more recently. Although the more interesting and attractive features, they also have much more difficulties and uncertainties than the traditional LWR designs due to their innovativeness.

Table 1 lists all the marine based water cooled SMR designs with the applicable technology, the major design characteristics and the development status. Some *Generation IV* SMR for marine applications are also in development. However, PWRs are the best technology readiness for marine application. Most of the safety systems of reactors use active safety system, especially in long term scenarios that challenge passive safety systems. From the large reactors perspective, in the last two decades relatively few reactors have been built globally, with most investors (mainly in South Korea, Japan and China) using proven designs like the large *Generation II* reactors (LOCATELLI et al., 2014).

Table 1: Marine based water cooled SMR design characteristics and status (ARIS, 2020).

Design	MW(e)	Type	Fuel enrichment (%)	Safety systems	RPV height/diameter (m)	Country	Status
KLT-40S	35	PWR 4-loop forced circulation	18.6	Active (partially passive)	4.8 / 2.0	Russian Federation	In Operation
RITM-200M	50	iPWR forced circulation	< 20	Combined (active and passive)	8.6 / 3.45	Russian Federation	Under Development
ACPR50S	50	PWR 2-loop forced circulation	< 5	Passive	7.2 / 2.2	China	Conceptual Design
ABV-6E	6 - 9	iPWR natural circulation	<20	Passive	6 / 2.4	Russian Federation	Final design
VBER-300	325	iPWR forced circulation	4.95	Combined (active and passive)	9.3 / 3.9	Russian Federation	Licensing Stage
SHELF	6.6	iPWR forced and natural circulations	19.7	Combined (active and passive)	3 / 1.2	Russian Federation	Detailed Design

1.3 JUSTIFICATION AND CONTRIBUTION

(HAAS, 2014) presented two major issues for the deployment of nuclear-powered commercial shipping, economic and regulatory issues. From an economic point of view, commercial success with nuclear powered merchant ships still needs to be achieved. (HAAS, 2014) provided a qualitative discussion about strategies for the success of nuclear-powered commercial shipping. (ONDIR, 2018) discussed the economic feasibility of NPP for merchant ships, proposing an economically competitive solution of nuclear power for their propulsion.

There are several discussions to present the economic feasibility of marine applications of SMR, however, nuclear-powered ships will not be deployed if the acceptable level of safety cannot be demonstrated to nuclear regulatory bodies. In addition, the safety demonstration also means capital cost of investment. A risk-based safety demonstration would alleviate certain economic burdens coming from the current licensing model of land based NPP.

This work contributes to the safety aspects of marine application of nuclear power, presenting a risk-informed methodology for risk assessment on marine nuclear power plants, and contributing to the feasibility analysis of such application from the point of view of nuclear safety.

In comparison to an active safety marine based nuclear reactor, the stationary nuclear power plants have more electric power sources for several layers of the defense-in-depth and safety classes on the design. The possibility to account with non-safety related mobile Diesel-Generator constitutes an important advantage to land-based reactors for accident management. Therefore, it can be postulated that the electric power outage on a nuclear-powered ship at sea has a greater occurrence frequency than on stationary nuclear power plants. In the case of a nuclear-power merchant ship at sea, the electric power recovery must be reached by onboard means, otherwise the station blackout rest through long term. It can take several days up to external support can be provided for distressed ships far from the coast. Therefore, it can be postulated that the blackout accidental scenario has a greater occurrence frequency on *Generation II* and *III* marine reactors when compared to land-based reactors of the technology. However, the lower radioactive inventory (due to small reactor power) and operation away from populated areas most of the time, reduce the accident consequences to the public. This balance on risk will be largely explored in this work.

The traditional safety analysis of large nuclear power plants must be performed in each reactor submitted to design certification. The LWR accident like large LOCA and ATWS is almost eliminated in the current design that integrates the steam generator, reactor and control rod drive mechanism in the same vessel, the technology used on several marine based reactors design. Therefore, the blackout accident, the most core damage contributor in reactors up to *Generation III*, constitutes an important analysis.

Another interest about the blackout safety analysis is that it permits to explore risk-informed approach through best-estimate dynamic reliability modeling, taking into account the reparability actions and systems reconfigurations. It permits to complement the systems design approach of “redundancy, segregation and independency” with “maintainability, maintenance and operational procedures” approach, reducing capital costs and keeping safety.

The blackout risk is analyzed on a hypothetical *Generation II* 100MWt marine based reactor. A comparison with Surry nuclear power plant was done. Surry is a typical PWR well documented in safety studies NUREG-1150 and NUREG-1935, the State-of-the-Art Reactor Consequence Analyses (SOARCA).

In addition, this work can provide guidelines to regulatory bodies for evaluating marine based nuclear reactors risk, providing reasonable order of magnitude in the results. Such a method can be applied to both existing and future reactor designs. The accident dose level at the exclusion and the low population areas are important to subsidy the decision of nuclear-powered ships acceptance on ports of the world and can be obtained by the methodology proposed too.

1.4 OBJECTIVES

This work has the main objective of contributing to Probabilistic Risk Assessment during the conceptual design phase by proposing a methodology to quantify the risk of the blackout event on nuclear-powered merchant ships as and comparing it with stationary nuclear power plants.

The main objective is breakdown into the following complementary objectives:

- Study a blackout event on a generic marine nuclear electric power system and postulate the accidental scenarios involving blackout on such plants;
- Define a dynamic reliability model of emergency electric power sources for best-estimate analysis;
- Perform a structural reliability analysis of the containment;
- Implement a simplified thermal transient algorithm for time to core uncover calculation in case of a blackout;
- Implement a simplified algorithm to calculate the source term according to core temperature;

- Obtain the RASCAL code dose calculation results to define the dose to an individual of public, according to its location and exposure time duration;
- Compare the results with stationary nuclear power plants in order to assess the risk level; and
- Determine areas of exclusion and low population based on the blackout accident.

1.5 LIMITATIONS OF THE RESULTS

First, the compendium of data for the hypothetical case study does not represent a design itself. Therefore, the results obtained do not attest the risk of any specific plant. Nevertheless, the case study was useful to attest the proposed methodology.

A general remark regarding probabilistic analysis must also be highlighted. Even if adequate modeling is applied, the accuracy of results depends on the input data. The reliability of the equipment in this work is obtained by failure databases not only from the nuclear area but also from the industrial fields as a whole.

Probabilities of initiating events are adapted from the (NUREG/CR-5750, 1999), which is a compendium of historical information of initiating events on US stationary NPP. Naval reactors would be a more indicated source of data, but they are not public.

Complete consequence analysis of radiological dose to the public should include the number of people exposed to radiation, and the number of latent cancer and deaths. In the case of a mobile plant, there is no specific geographical point of release, indeed the number of people exposed would depend on the demography of the release point. The consequence considered in this work was the dose as a function of distance from the release point. Meteorological and geological data influence the dispersion of radionuclides. They were set to standard values, giving the uncertainty of release position characteristics.

1.6 OVERVIEW OF CONTENT

Chapter 2 briefly reviews the safety analysis typically applied to nuclear power plants. History, definitions and the state-of-art of reliability analysis are also described in such chapter.

Chapter 3 presents the proposed methodology for blackout risk assessment. The main input data, processes and results are illustrated through a flowchart and described subsequently.

Chapter 4 performs the risk assessment on a hypothetical marine NPP. It describes in a detailed way the input data, hypothesis and activities, always exemplifying it based on the case study. The results of probability and consequence are joined to obtain the blackout risk, and compared to stationary PWR, performing a theoretical assessment of risk. Moreover, a discussion of the main risk contributors and risk management measures are also presented in this chapter.

Chapter 5 presents the conclusion.

2. BLACKOUT SAFETY ANALYSIS

Firstly, the definition of blackout event on mobile nuclear plants is proposed. Then, possible scenarios of blackout are discussed, and their related levels of nuclear core damage are presented. Consequences to the public in terms of radioactive dose are expected if the fission products from the nuclear core are leaked to the environment. Therefore, paths of radiological releases are commented on.

To understand how the blackout phenomena are dealt on safety studies, a brief overview of nuclear safety and licensing is presented. The chapter is concluded with the *"Risk-Informed"* approach for nuclear safety demonstration, explaining how such approach can be useful in the case of marine nuclear power plants, which have a different risk profile compared to large land-based nuclear power plants, especially in blackout accidents.

2.1 ELECTRIC POWER SYSTEM OF MOBILE REACTORS

Station blackout *means the complete loss of alternating current (ac) electric power to the essential and nonessential switchgear buses in a nuclear power plant (i.e., loss of offsite electric power system concurrent with turbine trip and unavailability of the onsite emergency ac power system). Station blackout does not include the loss of available ac power to buses fed by station batteries through inverters or by alternate ac sources as defined in this section, nor does it assume a concurrent single failure or design basis accident* (USNRC, 10 CFR50, 1988).

Station blackout is here called simply blackout. Stationary nuclear power plants usually adopt segregated power transmission lines and auxiliary lines, to ensure an uninterruptable power supply in case of loss of the power transmission line. On a mobile nuclear power plant, the uninterruptible power supply of essential loads can be performed by a bank of batteries. In this case, the batteries are usually capable to supply energy for a safe shutdown of the reactor and maintain so for the short term, supplying the safety systems responsible to remove the residual heat. The DG sets will supply the safety-related systems in long term.

Since batteries can be dimensioned to supply safety functions for a short term on mobile reactors, the safety function loss takes place only after the depletion of batteries. This battery supply time can be used for accident management. During battery power supply, measures to recover the lost power sources can be carried out. The recovery capacity was largely explored in this work (for example the diesel-generators repairs before batteries depletion) as accident management for risk reduction. Therefore, the blackout event on a marine nuclear plant can include the outage of station batteries, and planning accident management procedures in this situation, to risk reduction.

Post-Fukushima insights indicate the need for investigation of NPP electrical systems coping with offsite and onsite power loss, as shown on (GEISSLER, 2015). If the plant does not account for passive core cooling systems, only batteries can supply emergency bus bars to remove residual heat. Therefore, Post-Fukushima analyses indicate the necessity to consider batteries and other power sources to ensure residual heat removal in SBO scenarios. Such indication reinforces the analysis performed herein.

Figure 1 presents examples of typical architectures of stationary and marine electric power distribution networks.

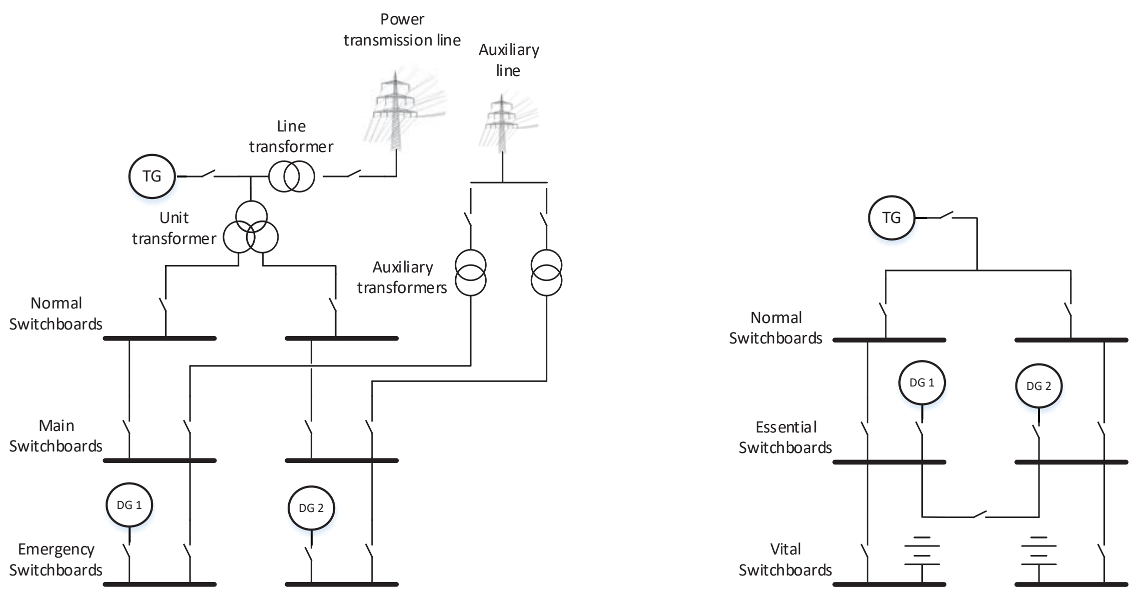

Figure 1: Unifilar diagram of hypothetical stationary and marine nuclear power electric distribution networks.

In a normal operating condition of a marine power system, the electrical energy is supplied by turbo-generators, and the batteries are kept in the floating state. In accidental situations, the batteries will supply with no power interruption, at least vital and essential consumers up to start the diesel-generators. The diesel-generators are the emergency electrical sources providing the energy for safety functions in a long term. They must supply the electrical needs (including the recharge of the batteries that may be discharged) of the ship until the reaching of a harbor or ship's external support.

In a first approach, DG redundancy level 2 will be studied, and focus on reparability and battery dimensioning, complementing the systems design approach with *"maintainability, maintenance and operational procedures"* issues. The frequency of occurrence of an electric power outage can be decreased throughout diesel-generators high maintainability and provisioning of adequate maintenance resources, optimal battery capacity and DG reparability level to decrease risk. Such a strategy leads to compactness and costless solutions. In the case of the risk assessment demonstrating the necessity of decreasing the blackout probability, increasing DG redundancy level could be a more evident solution, if volume limitations can be overcome.

A nuclear-powered ship at sea counts only on the diesel-generators to perform long term emergency electric power supply. If the normal power supply is not recovered and diesel generators are lost, core damage could take place after battery depletion. Due to the absence of alternative power sources, it can be postulated that the blackout frequency on mobile nuclear plants is greater than on stationary NPP.

2.2 CORE DAMAGE ON BLACKOUT SCENARIO

The blackout is considered a severe accident since such event can progress to nuclear core damage (CARVALHO, 2004). Nuclear power plants have two main specific characteristics that differentiate them from other power generation plants. The nuclear reactor accumulates a large number of fission products on the core. The quantity of fission products increases by fuel burning (since more nuclear fission products are accumulated), increasing the heat produced by decay after SCRAM.

The probability of core damage by blackout on NPP usually is lower than 10^{-5} reactor-year, considering the results of (NUREG-1150, 1990). Chapter 4 presents the probability calculation of a blackout on a marine nuclear plant. In PSAs of large reactors, the loss of electrical sources is the major contributing family to core melts frequency. Moreover, the NUREG-1935 study, the State-of-the-Art Reactor Consequence Analyses (SOARCA), presented greater fission products release fraction on station blackout than the ones defined in Large LOCA NUREG-1465 study.

The residual heat of the nuclear core must be removed, otherwise, the core temperature will increase up to core damage. The RHR is a nuclear safety system responsible to remove the residual heat. Such a system will be unavailable in case of an electric power outage since the RHR pumps will not be power supplied.

With RHR unavailable, either temperature and pressure will increase on the RCS. At a certain level of pressure, the safety and/or Relief valves of the pressurizer will actuate, decreasing the RCS pressure, at the cost of also reducing the inventory of coolant, due to relief of steam to the pressurizer relief tank. If the accidental scenarios continue for long time, the relief still up to a point that the relief tank disk rupture takes place due to overpressure, releasing the RCS coolant inventory directly to containment. The decrease of coolant inventory continues up to a point the nuclear core is uncovered by liquid coolant. The core increases its temperature abruptly, beyond cladding material capacity, causing cladding failure. At this time a low quantity of fission products is leaked on the primary circuit since most of them are retained on the fuel pallets. However, if no measure is taken to stop the core temperature increase, the core can be damaged to a level where the fission products cannot be retained anymore on the fuel pallets. At this point, a considerable quantity of fission products is released first to the pressure barrier and after to the containment atmosphere.

Figure 2 presents the core temperature evolution and the different level of core damage (IAEA, 2015).

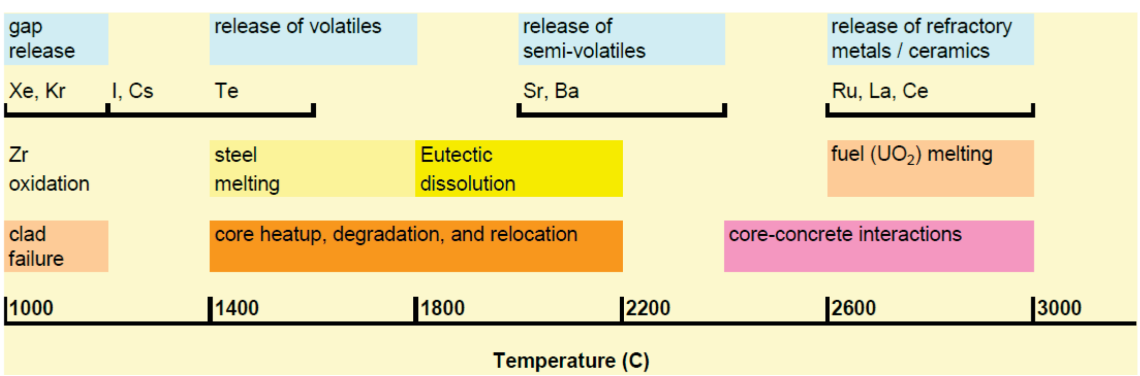

Figure 2: Fission products release, fuel degradation and severe accident phenomena as a function of temperature (IAEA, 2015).

Along with the blackout probability, the consequence of the release of fission products to the environment must be analyzed to obtain the risk of such an event. The accident severity is defined by the level of radiological dose of an individual of public, as a function of the distance from the local point of release. An assessment of the risk results must be performed to decide if nuclear power on ships has an acceptable level of safety.

2.3 SOURCE TERM

Source term refers to the magnitude and mix of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release (USNRC, 10 CFR50, 1988).

The people, environment and property must be protected from large releases of these products. The leakage of the fission products contained on the primary circuit up to the environment is not trivial and can happen by several paths. The fission products in the containment atmosphere would be released to the environment in case of containment failure. There are several scenarios of containment failure, based on the studies of (WASH-1400 NUREG-75/014):

- Mode α: Containment rupture due to reactor vessel steam explosion;
- Mode β: Containment failure resulting from inadequate isolation of containment openings and penetrations.
- Mode γ: Containment failure due to hydrogen burning/explosion;
- Mode δ: Containment failure due to overpressure;
- Mode ε: Containment vessel melt-through.

In case of no contained failure, at least a normal leakage is considered, the design leakage rate. The containment failure modes will be explored in Chapter 4. For now, it will be exemplified how the δ containment failure mode is considered in this work. The containment pressure will increase due to vaporized coolant relief to containment, in long term blackout scenarios. The emergency core cooling system will not operate (by sprays to condensate steam) giving the unavailability of electric power to pumps. This increase in containment internal pressure could lead to containment failure by collapse or fracture failure of the metal structure. The fracture failure can occur due to flaws in the structure.

The containment is the last barrier to fission products reaching the environment. Once released to the atmosphere, the fission products will be dispersed according to geographical and meteorological conditions. Such dispersed radionuclides would irradiate individuals of the public, and according to the time of exposition (mainly defined by the time to evacuate the affected region), the society receives a certain level of dose.

2.4 ACCIDENT RADIOLOGICAL CONSEQUENCES

There are distinct pathways of human radiological exposition. The first is the direct gamma radiation from radionuclides confined the in the containment, which is a localized but strong source of gamma radiation. The risk of this kind of dose is dependent of radiation shielding provided by the containment, and not directly related to the emergency planning zones. Such pathway is not explored in this work since it is a design dependent, and related to occupational exposed workers, and not for the individuals of public. The fission products release to the environment due to containment failure or by the normal containment leakage is a potential radiological exposure pathway to society, especially when the ship is close to litoral areas. The leaked fission products are dispersed by the air, according to meteorological and geographic conditions. The radioactive plume exposure pathways considered for emergency plans are the following:

- Whole body external exposure to gamma radiation from the plume and from deposited material in the ground; and
- Inhalation exposure from the passing radioactive plume.

According to the dose magnitude that an individual of public is exposed to, consequences to human health can be expected. Collective dose provides a measure of the societal consequences in terms of the number of health effects, which may appear over the ensuing lifetime of the surrounding population. The collective dose will not be explored in this work since it is a harbour or a port specific analysis, when the ship is moored to harbour.

To dealt with the geographical location of the ship in the accident consequences analysis, the figure of merit estimated is the radiological dose of a hypothetical individual of society at a distance of the point of release and a specified time duration of exposure. This result of individual dose must be transformed to collective dose according to each specific port, to support the definition of emergency planning zones for a ship. In case of a ship far from coast, only occupational exposure takes place, and no consequences to public is expected.

2.5 LICENSING CONSIDERATIONS

In the case of NPP licensing in the U.S., the normative set of the U.S. NRC establishes several prescriptive requirements to be met. Among the requirements, it is quoted a plant response analysis to the postulated baseline events, like the ones specified in (ANSI / ANS 51.1, 1983), documented in an FSAR. This analysis simulates the integrated response of the plant to the events, in order to confirm if structures, systems and components reach the nuclear safety criteria as well as the acceptance requirements of the safety analysis. (ANSI / ANS 51.1, 1983) also specifies, for each category of operational situation, the occurrence frequencies of the events that lead the plant to such situations, and the radiological consequences considered acceptable. Recent studies identified that the use of prescriptive requirements lets the licensing process inflexible, leading sometimes to over dimensioning, and not surely guaranteeing the safety of the plant. In cases where compliance to prescriptive requirements is not met, however other solutions oriented to risk reduction are applicable, the Risk-informed approach could be applicable. The Risk-informed is a flexible but robust methodology for safety demonstration.

2.5.1 Risk-informed on licensing

A merchant ship can perform the shipping of goods to the most diverse countries. The demonstration or licensing of stationary NPP may vary between countries. Therefore, it is necessary to carry out a study applicable in most cases. The *"Risk-Informed"* approach may be an option, minimizing the need to comply with prescriptive requirements, that may vary according to the normative basis of the different countries. In this approach, the risk is quantified and informed using state-of-the-art probabilistic and physicochemical-biological modeling to assess consequences. The risk-informed does not exclude the use of standards-based analyzes and prescriptive requirements. The results of risk are formally presented to the nuclear authorities for licensing.

The greater reason to adopt Risk-informed approach in the case of SMR is that such plants have really different risk profiles when compared to large land based NPP Travis (2019), especially marine based SMRs with a quite lower radiological inventory. (Prochazkova et al. 2021) show SMR risk sources and indicate a risk-based method for the design of such plants. The risk-informed approach enables design decisions driven by the risk level, comprehending exhaustive probabilistic analysis. A trustworthy and worldwide auditable method to analyze probabilistic analysis is key feature for risk-informed approach success, in special for maritime context.

The U.S. NRC policy for implementing risk-informed regulation was expressed in the (SECY-00-0213, 2000) policy statement for the use of PRA methods in nuclear regulatory activities. The policy statement mainly defines: The use of PRA technology should be increased in all regulatory matters to the extent supported by the state-of-the-art in PRA methods and in a manner that complements the NRC's deterministic approach and supports the NRC's traditional defense-in-depth philosophy.

NRC has generally regulated the NPP based on a deterministic approach. The deterministic approach considers a series of challenges to safety and determines how those challenges should be addressed in design. A probabilistic approach to regulation enhances and extends this traditional deterministic approach, by:

- Allowing consideration of a broader set of potential challenges to safety;
- Providing logical means for prioritizing the challenges based on risk significance;
- Allowing consideration of a broader set of resources to defend against these challenges.

The US NRC normative base for licensing reactors and some Regulatory Guides supporting the normative base provide guidance concerning probabilistic considerations, for instance, the (SECY-00-0213, 2000). PRA methods have been applied successfully and have proved to be a valuable complement to the deterministic approach. For example, PRA methods were used effectively during the ATWS and SBO rulemaking and supported the generic issue prioritization and resolution process.

The emergency protection zones are viewed as one of the most pressing regulatory issues. The methodology can be used to establish EPZs in dose-based risk-informed approach. In 2011 the NRC concluded EPZ designation could employ a technology-neutral, dose-based, risk-informed approach for SMRs (NRC, 2011). A risk and performance-based approach to licensing would be appropriate for SMR, especially in marine applications which have risk profiles different from land based plants. The same approach has been applied to Australia to decide about the acceptance of U.S. naval nuclear propulsion plants in their ports. Such a study can be found on (ARPANSA, 2000), which identified the EPZ around these naval reactors, and concluded that they are feasible on some country harbors.

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3 METHODOLOGY

The main contribution of this work is the methodology proposed to develop a risk assessment of the NPP in early design phases, and identify issues related to nuclear safety. Assessing safety issues early in design is extremely useful to address solutions when they are easier to be taken into account.

The proposed methodology emphasizes the plant overview of accidental scenarios more than specific studies of safety. The data and the methods discussed are not those applied in many specialized disciplines devoted to the in-depth study of safety but are those required for overall, first approximation assessments, like the ones applicable in the conceptual and preliminary design phase. Such assessments are the most useful ones for the detection of many safety-related issues in a plant and drafting a complete picture of safety issues. Accurate and precise methods are considered essential in licensing and optimizations of plant design.

The methodology proposed herein is based on five main steps:

- Plant data acquisition;
- Blackout and containment failure probability quantification;
- Source term quantification;
- Dose to individuals of public calculation;
- Risk Assessment.

It is firstly presented in Figure 3 the methodology flowchart providing a macroscopic overview of the activities. The next subchapters provide a detailed explanation of each activity shown in Figure 3.

As a brief overview of the methodology, it is necessary to acquire input data of the reactor, electric power circuit and containment. The probabilistic analysis of the electric circuit and the containment is performed to define probabilities of the accidental paths of the event tree. The accidental paths have related source terms associated that are defined by the use of thermo-hydraulic transient associated with the relative volatile model. These source terms have they consequence dose calculated with the aid of RASCAL software. The associated probabilities and dose of each accidental path jointly define the risk, which is compared to reference plants to support the assessment.

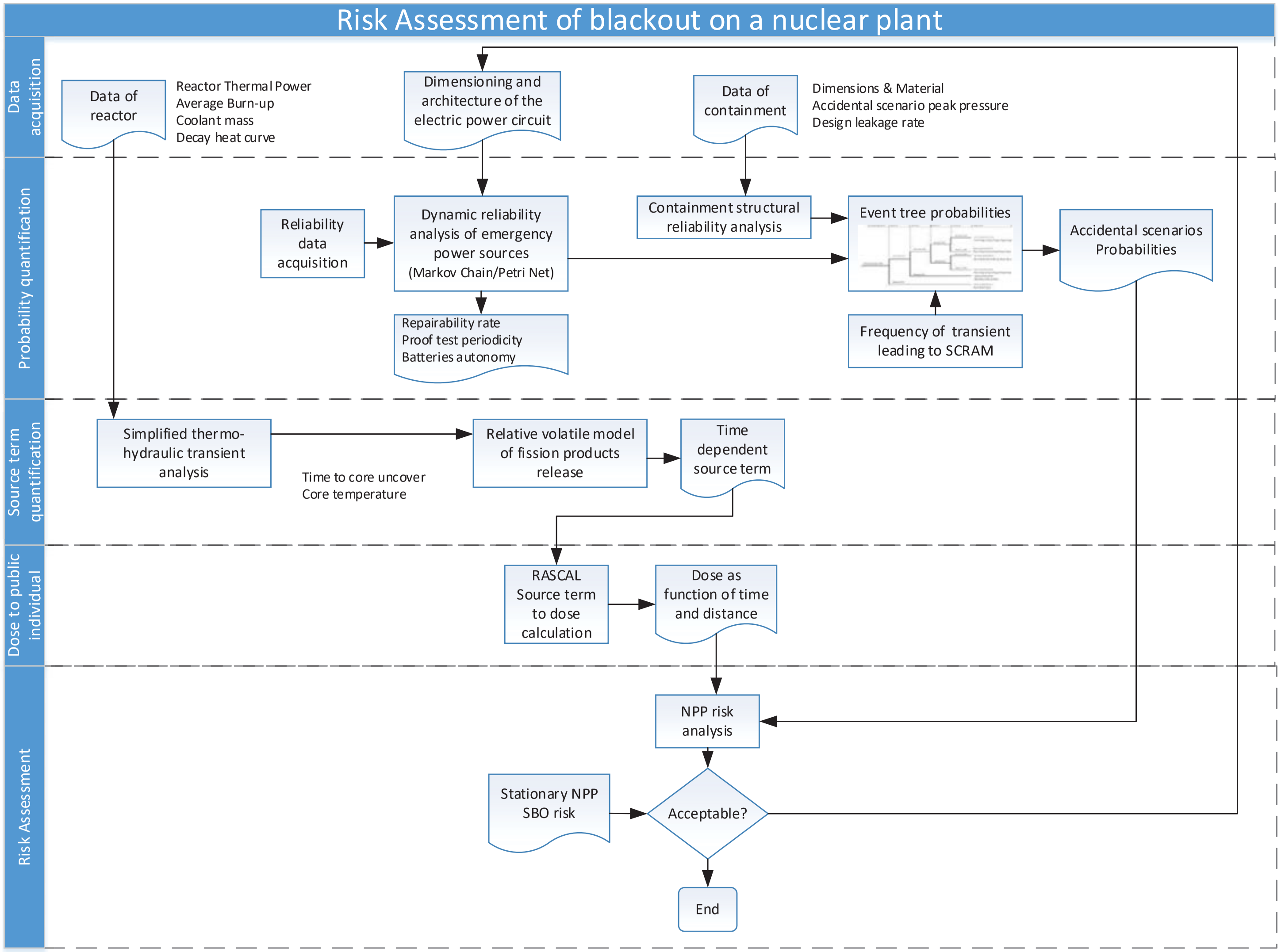

Figure 3: Methodology to assess the risk of blackout on a marine nuclear power plant.

3.1 DATA ACQUISITION

The data adopted on the definition of risk level is coherent with a conceptual design phase as demonstrated in the description below. Hereinafter is presented an explanation of the necessary input data to perform a blackout risk assessment based on the methodology of Figure 3.

3.1.1 Data of reactor

Reactor thermal power and coolant mass, i.e. the inventory of coolant on the RCS, are the main dimensioning figures of a NPP for the study, usually defined in a conceptual design phase. The reactor power is mandatory to calculate the decay heat energy. If specific nuclear fuel data are available, for instance the ones calculated by codes like ORIGEN-S, decay heat can be calculated by a precise method of (ANSI/ANS-5.1, 2005). Otherwise, a generic decay heat curve can be used with considerable precision for time to core uncover calculation. (KNIEF, 1992) proposes the formula (1) to be used as the first approximation of the heat decay curve

$$\frac{P(t)}{P_{SCRAM}} = 6.6 \times 10^{-2}\left(t^{-0.2} - (t + t_0)^{-0.2}\right) \qquad (1)$$

Where

$P(t)$ is the decay heat power as a function of time after SCRAM;

P_{SCRAM} is the reactor thermal power at the instant of SCRAM;

t_0 is the cumulated operating time of the nuclear core, since divergence;

t is the time after SCRAM.

The coolant inventory is also necessary for the calculation of time to core uncover, jointly with the decay heat curve. The reference (ANSI/ANS-18.1, 1999) defines the concentration of a radionuclide by unit of coolant mass. Based on such concentration and total volume of coolant, the radiological consequence of accidents without core damage, i.e., only release of coolant, can be estimated.

For accidents with core damage, it is necessary to define the amount of fission products present at the core in the instant of the accident. If specific nuclear fuel from codes like ORIGEN-S is not available, the (NUREG-1940, 2012) specifies generic core radionuclides inventory. The (NUREG-1940, 2012) estimated inventory based on calculations made by the U.S. NRC staff in 2003 using the SAS2H control module of SCALE (Standardized Computer Analyses for Licensing Evaluation). The calculations were done for a fuel core at a burnup of 38585 MW.days per metric ton of uranium (MWd/MTU). The calculated core had a power level of 3479 megawatts thermal (MWt) and enrichment of 4% of uranium-235. A linear adjustment is performed in the inventory of radionuclides that have half-lives that exceed 1 year to account for burnup, through the formula (2),

$$\boldsymbol{I_{ACTUAL}} = \boldsymbol{I_{38585}} \times \frac{\boldsymbol{BURNUP_{ACTUAL}}}{\mathbf{38585}} \times \frac{\boldsymbol{P_{th}}}{\mathbf{3479}} \quad \mathbf{(2)}$$

Where

I_{38585} is the radionuclide total activity in the nuclear fuel with burn up of 38585 MWd/ton,

$BURNUP_{ACTUAL}$ is the nuclear fuel current burn up in MWd/MTU; and

I_{ACTUAL} is the radionuclide activity at the instant of $BURNUP_{ACTUAL}$.

This method recommended by (NUREG-1940, 2012) permits to define an order of magnitude of the inventory of radionuclides at the instant of an accident.

3.1.1.1 Reactor Dead Time

A curious behavior occurs to the xenon after the reactor shutdown. Although its normal production from nuclear fission ceases, it continues to build up as a result of the decay of its iodine parent decay. Therefore, the concentration of xenon increases after shutdown. Since its cross section for neutrons is high, it absorbs neutrons and avoid the reactor from being restarted for a period of time denoted as *"reactor dead time"*. (RAGHEB, 2011) used Bateman's solution to calculate the iodine and xenon concentrations as a function of time. Such a solution result has been used to identify a reactor dead time. It estimates the minimum time to recover the normal power supply.

3.1.2 Dimensioning and architecture of the electric power circuit

The electric power distribution architecture is fundamental to drafting the model for onsite power loss probability. The electric sources' nominal power in comparison to emergency system power consumption is important as well. The level of redundancy in electric systems is defined considering the largest power consumption scenario in an accidental scenario and the number of diesel generators necessary to satisfactorily supply it. For instance, on a fully redundant emergency power system, one DG can supply the safety systems necessary on the largest power consumption design basis accident.

3.1.3 Data of Containment

The main dimensions of containment are indispensable to identifying the order of magnitude of the probability of structural collapse or fracture due to peak pressure during an accidental scenario. Overpressure can be caused by long blackout scenarios when the relief of steam to containment by the reactor cooling system is inevitable. Another possibility is the leakage of coolant from the primary circuit by a break on it, for instance a LOCA. In front of a core damaged, hydrogen is produced by the chemical reaction of cladding alloys with water at high temperature. Severe overpressure can be caused by hydrogen concentration increase and subsequent explosion. Hydrogen explosion can be avoided by engineered safety features like passive hydrogen burning systems. These features can be independent of electrical power, and so are not considered in the blackout accident pathways defined herein.

Even no containment failure takes place, a low quantity of containment contaminated atmosphere is expected to be leaked to the environment. An admissible leakage rate is considered since the structure cannot be validated as 100% leakage proof. Therefore, even if no containment failure takes place, radiological consequences to society are possible.

3.2 PROBABILITY QUANTIFICATION

The probability analysis can involve several tools for probability evaluation, since the most typical, like fault tree, up to state of art on dynamic reliability calculation by Markov chains and/or Stochastic Petri Nets.

The traditional analyses of systems reliability like Reliability Block Diagrams and Fault Trees are consolidated methods used on probabilistic analysis, but are based on static basic events and do not consider complex scenarios of dynamic reliability, leading to conservative results. For instance, components' behavioral interdependencies demand system states orientation modeling. The Markov Chains have been used so far for it. However, Markov Chains demand the modeling of systems directly on their states, which is cumbersome and error prone in large systems. A formal method to define the states diagram is recommended. The modeling through Petri Net provides several characteristics to ensure model verification, such as net properties verification, invariants analysis, reachability graph, simulation, etc.

Petri Nets permit us to explore the state-of-art reliability best estimate modeling, including system reconfigurations, human intervention and permissibility of short outage periods, which is useful to designers. It also enables an easier and formal method to audit probabilistic analysis review of safety reports, through several methods of modeling verification, which is useful to regulatory bodies.

This methodology recommends to model the blackout probability through Generalized Stochastic Petri Nets (GSPN), a Markovian class of Petri Net with analytical and numerical steady state and transitory analysis relatively simple. GSPN has been considered the ideal tradeoff between modeling powerful and quantitative analysis simplicity. Availability for fully repairable systems is calculated through the well explored GSPN steady state analysis. Transitory analysis of GSPN with absorbing states, applied for reliability calculation, is used to model the blackout probability. Shukla and Arul (2020) used Monte Carlo simulation to quantify probabilities of a wider class of PN like eDSPN, but the simulation of really low probability events is cumbersome and challenging to perform parametric and sensitivity analysis.

Thanks to a powerful modeling tool, this work proposes to largely apply the reliability and maintainability studies on the design of safety systems, which certainly provide insights to decrease the risk. The analysis of the level of reparability on accidental scenarios should be performed, even in a marine context of operation. Actions performed by the crew have been valorously implemented on Dynamic Probabilistic Safety Assessment (PSA) software platform of (Diaconeasa and Mosleh, 2018). Disregarding the possibility of repairing failed equipment during an accident scenario leads to a conservative estimate (Bouissou et al., 2020). When repairs are considered on reliability evaluation, a Partially repairable systems model of reliability must be adopted, according to (SOUZA; GABE, 2017), resulting in less conservative results of failure probability. A feasible level of reparability in an accidental scenario is defined by identifying reparable components and their contribution to the global failure rate, according to the method proposed by (SOUZA; GABE, 2017).

A qualitative safety analysis is recommended to identify hazards and related safety functions. It also identifies several procedures and corrective measures. Those insights are important to ensure good modeling. This step is part of the methodology mainly to identify features not evident, like components behavioral interdependencies, system reconfigurations, repairs and operational procedures, etc. The maintenance engineering aspects must also be deeply explored, since they can contribute significantly to risk, like repair actions performed by the crew.

The use of Petri nets (PN) to model system reliability is not a recent idea. Schneeweiss (2001) shows many models of reliability through PN in a didactic way. (Dosda and Brandeletb, 2021) reinforced currently the advantages of the use of PN to overcome the limitation of static reliability tools PSA.

Starting the modeling by PN enables the formalism and analysis to ensure model correctness. When a PN model is adequately constructed, the system states can be obtained through the reachability graph analysis. Modeling the reliability of large system directly on its Markov chain is laborious and error-prone. Therefore, each time complex scenarios are not modeled by combinatorial models like fault trees, a system reliability model through PN is recommended. There is a package of properties and analyses to be performed on a Petri Net to formally ensure model correctness and performance measuring.

The Stochastic Petri Nets (SPN) is a Petri Net where each transition has a firing delay probabilistically distributed. GSPN is characterized due to its capacity to accept two types of transition, timed exponential and immediate, this last used to represent a logical control and action that delay is negligible, expanding the representativeness capability of a SPN. Wider classes of SPN can model other types of transition firing delays, such as deterministic or generally distributed firing delays. However, the stochastic process behind wider SPN classes is non-Markovian, which is solved by discrete time approximation of the stochastic behavior process of the marking process Horváth et al (2001). This solution cannot be simply calculated analytically or numerically and is usually solved by computational tools. Therefore, non-Markovian Stochastic Petri Nets (NMSPN) turns the analytical solution presented hereinafter impossible. Monte Carlo simulations on events with quite low probability are challenging and usually avoided. It is preferable to keep simply using an approximation where deterministic of other than exponential distribution firing delays are approximated by exponentially distributed firing delays and ensuring the applicability of the stochastic Markovian process.

The reachability graph of a GSPN is isomorph of a Markov chain. In other words, when all firing delays are exponentially distributed, the stochastic process behind the

Petri Net is a Markovian process, and the performance or quantitative analysis of SPNs can be carried out straightforwardly by analyzing the corresponding Markovian process. The Markov chain state space is obtained by the reachability graph of the PN with the initial marking M_0. The transition rate from state M_i to state M_j is given by

$$\boldsymbol{q_{ij}} = \begin{cases} \sum_{k \in H_{ij}} \boldsymbol{I_k} \\ -\sum_{k \in H_i} \boldsymbol{I_k} \end{cases} \tag{3}$$

Where

H_i is the group of all transitions enabled by the M_i marking,

H_{ij} is the group of all transitions enabled by the M_i marking, whose firing leads to M_j marking.

Supposing an ergodic chain, where the initial marking can be recovered for any state of the reachability graph, i.e. a cycling net free of traps and deadlocks, the steady state probability vector $\pi = (\pi_1\, \pi_2 \ldots \pi_s)$, where "s" is the number of states, can be calculated, through the resolution of the following equations,

$$\boldsymbol{\pi * Q = 0} \tag{4}$$

$$\sum_i \boldsymbol{\pi_i} = \mathbf{1} \tag{5}$$

Where

Q is the transition matrix.

The steady state probability vector represents the probability of the system being found in each state. Associating the probabilities of all states that present the system available, the system availability can be found.

A small modification to transitory analysis is that q_{ij} is equal to 1-q_i. An initial state probability vector π_0 must be introduced. $\pi(t)$ represents the instantaneous state probability vector. The time dependent behavior of the Continuous Transition Markov Chain (CTMC) can be described using the Kolmogorov differential equation

$$\dot{\boldsymbol{\pi}}(\boldsymbol{t}) = \boldsymbol{\pi}(\boldsymbol{t})\boldsymbol{Q} \tag{6}$$

The instantaneous availability of the system can be found by the same association described previously. The transient solution demands interactive calculation to find each next instantaneous state probability vector. The transient analysis has not the constraint of cycling nets and supports absorbing states, i.e., deadlocks. Therefore, the system reliability can be calculated as the association of all states' instantaneous probability that represents a system free of failures, i.e., the probability of being out of absorbing states representing a system in failure.

3.2.1 Reliability data acquisition

There are at least four sources of reliability data, in the following order of priority:

- Supplier information or In-service failure data of the concern equipment;
- In-service failure data of similar equipment or similar context of use;
- Reliability data bases from the same context of use or from applicable industrial context;
- Expert judgement by feedback.

Reliability data from databases may be the only source of data to perform reliability analysis at the conceptual and preliminary design phases. This work used reliability data from open databases like (EIReDA,1998). The supplier or in-field data of reliability is considered the most reliable data. However, to draft an order of magnitude of failure probability, especially at the conceptual or preliminary design phase, databases or feedback experience can be used for the calculation to obtain an order of magnitude of probability.

A detailed study has been done regarding the maintainability data. Firstly, not all failures can be modeled as repairable, since the ship far from the coast doesn’t account on external support for maintenance and the necessary resources for it. Only limited onboard stock of spares and tools are available. In front of this issue, (SOUZA; GABE, 2017) proposes a reparability factor to model Partially Repairable Systems. In merchant ships on cruise, repairs must be performed by crew members with onboard resources. On mechanical systems, reasons for limited maintainability are dismantling to access some components and the need for handling heavy components. In order to define an approximate repair rate consistent with the context of operation studied, it must be considered a breakdown structure of the equipment up to the spares level.

Afterward, it is necessary to know the contribution of each component to the equipment's global failure rate. Such detailed information is not easy to be acquired. (OREDA, 2002) proposes a generic breakdown structure of equipment and a list of the contribution of each item to the global failure rate. Data from (OREDA, 2002) can be applied to this study for the reparability rate factor.

3.2.2 Dynamic reliability analysis of emergency power sources

Figure 1 presents an architecture of the hypothetical marine based reactor electric power system with level two of redundancy. The Station Batteries are dimensioned to support safety-related functions to achieve and maintain safe shutdown at short term.

During battery autonomy to supply safety-related electrical buses, a temporary redundancy to Diesel-Generators is ensured. Moreover, the possibility to recover at least one Diesel-Generator while batteries supply power constitutes an important accident management procedure. Core damage due to the blackout scenario does not take place while the batteries have energy.

A reliability modeling of the electric power circuits on a NPP can be very exhaustive since many failure contributors are involved and several load feeding configurations are possible. The following approximation has been proposed, without compromising the precision of the results:

- The scope of the modeling includes only the main failures contributors of the electric power circuits, for instance, the Diesel-Generators and their auxiliary systems;
- It has been chosen to neglect failure on bus bars, connectors, cables, etc… since such equipment are passive, which is not usually considered in reliability analysis because they have high reliability, are not a complex technology, almost no mechanical movement, and independence of external ancillaries to operate;
- Circuit breakers have not been considered because some provisions can be taken to drastically decrease their contribution to failure probability, for instance the manual operation, redundancy and adoption of control strategies to decrease the number of switching to emergency power distribution configuration;

- For DG auxiliaries, generic architectures proposed on (EPRI/NP-5924, 1988) can be considered on a system equivalent failure rate and reparability factor. Since there is no real DG and auxiliary system, a typical reference must be considered; and
- The battery reliability modeling considers:
  $R_{bat}(t) = 1$ for $0 < t \leq T_{blackout} + T_{autonomy}$; and
  $R_{bat}(t) = 0$ for $t > T_{blackout} + T_{autonomy}$.
  Batteries are also considered as passive elements, therefore failures are not considered on probabilistic analysis, only their temporary redundancy before depletion.

Such approximations can be useful, and necessary on a generic plant to illustrate the methodology proposed herein. Reducing the scope of reliability modeling to batteries, diesel-generators and their auxiliaries is useful in early design stages to anticipate safety-related issues. However, for licensing process, modeling exhaustiveness must be demonstrated.

Analytical and numerical modeling for standby redundant partially repairable proposed by (SOUZA; GABE, 2017) should be applied as deep as possible, to account for repair actions on risk figure. Partially repairable equipment is considered to have three states: available, unavailable (under repair) or lost (not possible to be repaired). The model is based on Continuous Markov Chains, where all state transitions must be exponentially distributed. The battery autonomy to supply power is a deterministic event since, after enough time to discharge, the electric power supply certainly stops. The GSPN does not model timed deterministic transitions, only exponential and immediate transitions. The Markov Chains PRS model can be calculated analytically but limits the transitions to exponentially distributed events. It is preferable to keep simply using an approximation where deterministic of other than exponential distribution firing delays are approximated by exponentially distributed firing delays and ensuring the applicability of stochastic Markovian process.

The failure rate of Diesel-generators is considered constant, and then the transitions are exponentially distributed. Such a hypothesis is realistic since the equipment operates in stand-by state, not subjected to continuous operation wear. Moreover, preventive maintenance can keep equipment integrity against aging.

Chapter 4 shows a model of Generalized Stochastic Petri Nets of 2 stand-by redundant DG with repairs count on reliability for the generic case study of this work. This model also considers the battery temporarily redundant with DG, which provides important risk reduction, since repairs could recover a DG before a blackout.

3.2.3 Containment structural reliability analysis

The containment performance consists of the capacity to withstand the accident loads and keep the fission products leaking to the environment as lower as possible. The phenomena considered to cause containment failure is the internal overpressure from long blackout scenarios.

Some accident sequences that could threaten the containment, such as those involving steam explosion of the Reactor Coolant System are considered to have a probability so low that their contribution to risk can be neglected (LIBMANN, 1996). The overpressure of containment by hydrogen explosion can be an important contributor to containment failure. The hydrogen can be generated from oxidation of fuel cladding, especially if the cladding material is zircalloy. However, the material of fuel cladding is a design specific choose and hydrogen explosion can be drastically reduced according selected material. The radiolysis of coolant can also release hydrogen, but in a smaller amount. Chemical recombinators to absorb hydrogen can be installed all along the containment, and since they are independent of electrical energy, they constitute a relevant solution to drastically reduce the risk of hydrogen explosion. Giving the dependance of fuel material jointly with the passive recombinators solution, the risk of hydrogen can be manageable, and therefore not considered in this work.

The containment overpressure is mainly caused from steam released by the Pressurizer Relief Valve actuation to control RCS pressure in absence of electrical energy. The relief valve of the pressurizer actuates intermittently, transferring coolant to the pressurizer relief tank. When the electric power is not recovered in long term, the relief will continue up to the pressurizer relief tank pressure limit and break of the rupture disk, leaking the coolant directly to the containment atmosphere.

Two containment failure modes can take place from containment internal pressurization. After a certain pressure, systems crossing networks, like the containment penetrations of cables and pipes can fail, leading to a bypass leakage. If the pressurization continues, a total containment failure can take place, due to structure collapse or fracture. It is considered that all containment atmosphere can be released to the environment in case of containment complete failure. If the containment does not fail, a smaller release of the containment atmosphere is considered. The design leakage rate takes place independently of containment pressurization. It is the amount of containment atmosphere released to the atmosphere without containment bypass or failure.

A maximum containment pressure created by the blackout accident must be established for the structural failure probability. To calculate the accident pressure in containment, (PETRANGELI, 2006) proposed a method to model the containment initial overpressure in LOCA. Heat dissipation to the environment has been included in the algorithm to extend it for the blackout scenario. Details of the algorithm used for containment pressure calculation can be obtained in the code of Annex A.

The containment structure can fail by collapse when the internal pressure reaches the rupture stress of the structure, or by fracture, mainly caused by the reaching of maximum stress intensity of a crack. The fracture mechanics approach contrasts with the traditional approach to structural design and material selection.

In the traditional structural reliability approach, the anticipated design stress is compared to the flow properties of candidate materials. A material is assumed to be adequate if its strength is greater than the expected applied stress in a ratio according to a safety factor to cover some uncertainties (ANDERSON, 2005). Such an approach may attempt to guard against brittle fracture by imposing a safety factor on stress, combined with minimum tensile elongation requirements on the material.

The fracture mechanics approach has three important variables. The additional structural variable is flaw size, and fracture toughness replaces strength as the relevant material property. Fracture mechanics quantifies the critical combinations of these three variables (ANDERSON, 2005). The stress applied to structure continues as on the traditional approach. Therefore, the cracks in the structure due to the manufacturing process are taken into account, leading to a higher probability of failure when compared with the traditional structural reliability approach. There are two alternative approaches to fracture analysis: the energy criterion and the stress intensity approach. The stress intensity approach states that crack extension increase (i.e., fracture) occurs when the stress available for crack growth is sufficient to overcome the fracture toughness of the material. This study applied the stress intensity methodology to a generic case study, as shown in chapter 4.

Other containment design data must also be available, like the material, dimensions and thickness, etc. to calculate the stress on the structure.

3.2.4 Frequency of transients leading to SCRAM

SCRAM, the reactor trip, is an unplanned event and it leads to plant shutdown to achieve a stable safe state. The adopted frequency of this event is usually defined by historical data. Since there is no marine reactors event reports reference, the frequency of occurrence of the initiating events leading to SCRAM (the transients and design base accidents) are obtained from (NUREG/CR-5750, 1999), which is a compendium of stationary U.S. NPP initiating event data. Except for some transients not applicable to a marine NPP, the frequency of occurrence of most transients is a reasonable reference. All these frequencies of transients are accounted for the total frequency of events leading to reactor shutdown. The LOCA events are not accounted for the frequency of the initiating event because such a scenario is not credible according to (CARVALHO, 2004). Chapter 4 presents the results of occurrence frequency adopted in the generic case study.

3.2.5 Event tree probabilities

On a marine NPP any transient leading to Reactor SCRAM is an important contributor to the development of blackout events. After SCRAM on mobile reactors, due to normal power loss, only emergency power sources are available when the ship is at sea, for example the Diesel-Generators and the batteries. In the case of complete failure of these emergency power sources, no electric power is available to ensure safety functions. Therefore, all plant transients leading to SCRAM are considered as initiating events of blackout severe accidents.

Even if the control of the reaction is independent of electric power (through SCRAM), there is a probability around 10^{-5} that the protection system does not generate the trip signals or safety bars do not ensure total SCRAM (LIBMANN, 1996). In the case of an unsuccessful SCRAM, the reactor keeps critical, and the loss of cooling (due to blackout) will lead to core damage in a short time. This family of unsuccessful SCRAM accidents after transients is called ATWS.

After the loss of normal power generation by the reactor, the emergency power sources must supply the ship. A ship would be able to maintain a very limited propulsion capability, due to the lack of power for propulsion. The ship must receive support from the nearest base or from rescue ships able to supply power. It is postulated that the ship will take a certain period to reach the necessary support when the normal power is not recovered by the self-means of the ship. This period depends on the position of the ship and of the bases or rescue ship closer. A conservative period of two weeks is postulated to acquire support for the ship.

The initiating events can also take place when the ship is in the harbor. Such a scenario has not been explored in this work, mainly because the ship alongside counts with electric power shore connections, increasing the redundancy of sources, higher reparability capacity and more possibilities for accident management. These reasons enable to postulate that the blackout probability is decreased considerably. In addition, the reactor is shutdown alongside and the decay heat is low. The maintenance, testing and inspection, foreseen in technical specification when reactor is in shutdown, can increase the blackout probability at harbor. However, it is expected that the ship is submitted to maintenance on safety systems in dedicated infrastructures, under controlled conditions to manage the risk

The consequences to the public can vary according to the local of the release of the source term. From the point of view of the consequence to the public, a release at the harbor is more severe than a release far from the coast. This work measures the dose level according to the distance and exposure time. The accounting of the number of individuals of society impacted by the release is not considered, since it is a location dependent analysis.

The recovery of the normal power is mainly dependent on the initiating event. It is postulated that transients of category 2 can be recovered after reactor dead time. The category 3 transients do not enable the safe restart of the reactor at sea, and longtime of the shutdown is expected. Therefore, an important difference in accidental scenario development is that after offsite power loss, the stationary NPP must ensure onsite emergency power sources up to offsite power recovery. In the case of marine NPP, it accounts on the recovery of normal power generation while the emergency and alternative power sources must supply power up to external support arrival. This scenario on marine NPP may last 14 days as an assumption.

In case of a blackout taking place before the ship obtains external support for power supply, the lack of cooling can lead to core damage according to the remaining decay heat at blackout time. This work assumes the core melts if the top of the core is uncovered by the coolant. After the blackout, the coolant inventory will decrease by the relief of the pressurizer relief valve, up to uncover the core during long time blackout.

Figure 4 shows an example of an event tree in the case of marine NPP blackout.

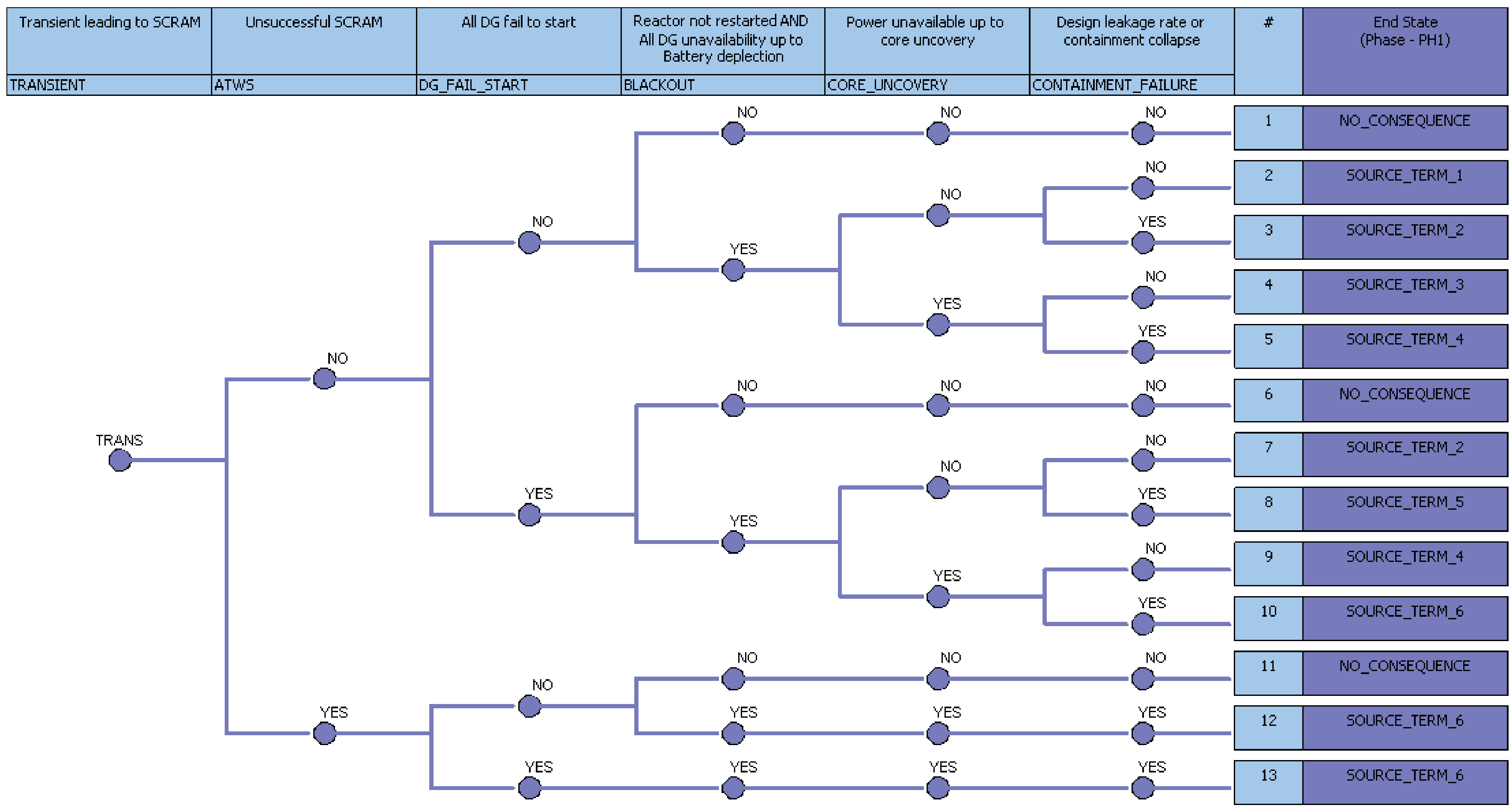

Figure 4: Generic event tree proposed to analyze blackout accident.

3.3 SOURCE-TERM QUANTIFICATION

The amount of fission products that can be released from a nuclear power plant in an accident is the fundamental parameter to estimate the consequences of the accident.

This source term is defined as the activity of each radionuclide released from the containment atmosphere to the environment by unit of time. Concerning radionuclides release to the environment, an important issue needs to be addressed, the performance of the containment to withstand the accident loads. The complete failure or just leakage of containment strongly impacts the resultant amount of fission products released to the environment. Even when no containment failure takes place by collapse or fracture, it is considered that an amount of radionuclides is released. (NUREG-1940, 2012) defines as design leakage rate, which is by default considered on the reference as 0.1% of the containment atmosphere volume per day. Considering that the radionuclides are homogeneously dispersed on containment, the activity by radionuclides released to the environment by unit of time can be calculated.

Containment bypass accidents also release radioactivity to the atmosphere. There are systems with pipes connected to RCS and/or crossing the containment. A release of radionuclides by equipment of these systems located outside containment, is called bypass. Moreover, the release can take place right on the containment penetration by pipes or cables. The SG tube rupture is also a pathway of fission products of RCS leakage to the environment. In SG tube rupture, the radionuclides present on the reactor coolant are leaked to the secondary system, resulting in a pressure increase on the secondary loop steam system, obliging the relief of these radionuclides direct to the atmosphere.

The determination of the source term requires knowledge of the thermal-hydraulic progression that leads to the coolant boiling off, and the subsequent core heat-up, melting and degradation. Computer codes that model the core and primary circuit as finite elements are usually adopted. It leads to a plant specific study that can be done when the design of the RCS is mature. An option to calculate source term in early design phases is to postulate a source term for light water reactors accidents. Such a solution does not model a wide variety of accident progressions. Therefore, it has been chosen an intermediate solution, a simplified model to calculate source terms according to a time dependent core level of degradation.

Firstly, it is necessary to calculate the time from the initiating event (in this work the transients leading to SCRAM, however it can also be a LOCA on the proposed method) up to core uncover, and the remaining coolant condition, i.e. the remaining coolant mass, the title and pressure, at the same time instant. Such calculation has been performed based on the first thermodynamic law, neglecting complex events, but obtaining a reference of *“Time to core uncover”* for the quantification of the source term. Based on the coolant mass to be vaporized for core uncover, and the latent heat at primary circuit pressure, it is possible to know the necessary heat to be transferred from the core to coolant by decay heat. Adopting the generic decay heat curve of (KNIEF, 1992), it is possible to calculate the necessary time to transfer the heat required to vaporize coolant up to core uncover, and therefore the time to start core uncover. The code to perform such calculation is presented in Annex A. The code also calculates the *“Time to core uncover”* for some LOCA phenomena, but it is not discussed in this work.

When the core uncovery takes place, the (IAEA-TECDOC-1127, 1999) proposes a simple, but robust algorithm that can model the essence of the thermal-hydraulic behavior, without having to resort to complex codes. The algorithm proposed by the (IAEA-TECDOC-1127, 1999) has been applied to this work, with the following improvements:

- Inclusion of the generic heat decay curve of (KNIEF, 1992) on adiabatic heat up;
- Calculate an infinitesimal time depended fission products release fraction;
- Considering the radionuclides concentration reduction mechanisms on containment (as sprays or gravitational settling); and
- Adopt a leakage rate from containment to calculate the time dependent source term.

The developed code is presented in Annex A. The source term outputted from the developed code permits to build a CSV file on the format accepted by the module Source Term to Dose calculator of RASCAL, to obtain the dose for an individual of society. The RASCAL source term was based on the core damage progression timing outlined in NUREG-1465, "Accident Source Terms for Light-Water Nuclear Power Plants,". RASCAL has a ready method for calculating reactor source terms for LTSBO events, but it is based on prescribed release fractions of radionuclides, and not a time dependent release fraction calculated based on remaining residual heat and resultant core degradation.

3.3.1 Simplified thermo-hydraulic transient analysis

After blackout, the time at which the core uncover starts depend on the decay heat and the coolant mass to be boiled off. The time from reactor SCRAM up to core uncover has been calculated based on the latent heat transferred by fission products decay,

$$\int_{T_{blackout}}^{T_U} \boldsymbol{Q(t)dt} = \boldsymbol{\omega \Delta H_{fg}} \tag{7}$$

Where

$T_{blackout}$ is the time when blackout takes place, accounted from the instant of the transient leading to SCRAM;

T_U is the additional time to reach the core uncover;

$Q(t)$ is the decay heat curve starting at the beginning of the blackout and ending at T_U;

ω is the guaranteed water available to the RCS; and

ΔH_{fg} is the latent heat of vaporization of water at the primary circuit condition pressure.

In the case of blackout, the RCS may not be depressurized, so the ΔH_{fg} is the latent heat of vaporization of water at the normal RCS condition. A significant parameter for the time of core uncover is ω, the residual water available within the RCS. Q(t) depends on the thermal power of the core.

When core uncovering takes place, the thermal pattern for a fuel bundle proposed by (IAEA-TECDOC-1127, 1999) is shown in the schematic diagram of Figure 5.

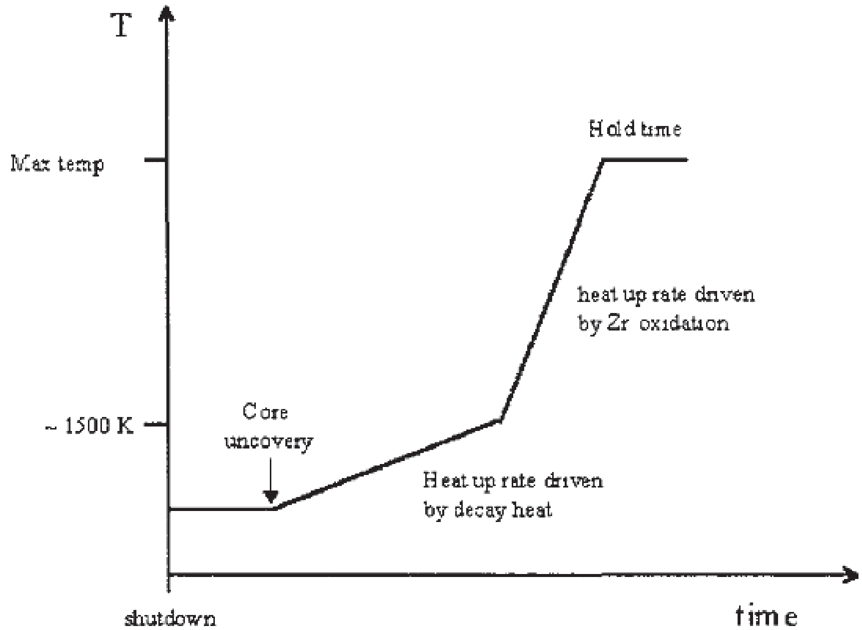

Figure 5: Thermal pattern for a fuel bundle (IAEA-TECDOC-1127, 1999).

Complicated features of core degradation behavior are not considered, such as the mechanics of fuel melting, relocation and blockage. Nevertheless, it is considered that the resulting source term is representative of radiological consequence analysis.

To calculate the generalized thermal transient, the following input data is required by the algorithm:

- The sensible heat to increase fuel/clad temperature;
- The Zr oxidation runaway heat-up rate;
- The maximum temperature reached in the transient; and
- The hold time at the maximum temperature.

The input data above enable plant specific characteristics, providing representative source terms for generic cases, and enable plant specific data.

3.3.2.1 Heat up driven by decay heat

The initial heat-up rate is determined as a balance between the rate of decay heat generation, the sensible heat required to increase the fuel/clad temperature and the amount of heat lost to the steam. Very small quantities of fission products are released during this phase, the major effect of this heat-up rate is on the source term release start.

3.3.2.2 Heat-up rate driven by Z_R oxidation

Establishing the associated rate of heat-up that is rooted in the Zr-steam oxidation behavior. The Zr-steam oxidation kinetics that is used in most severe accident codes are discussed in (WOOTON; AVCI, 1980). An adiabatic heat-up rate characteristic of the runaway Zr oxidation phase would be around 18 K/s and 25 K/s at the beginning and end of this phase respectively. It is recommended that an average value of 21 K/s be used for the generalized thermal transient. This rate is consistent with experimentally observed rates and rates calculated by several core degradation computer codes, according to (IAEA-TECDOC-1127, 1999).

3.3.2.3 Hold at melt

The melted core reaches the maximum temperature and thus the maximum release of fission products from the core. There is a short time duration from the end of the previous phase to the maximum release of this phase.

3.3.3 Relative volatile model of fission products release

Once the thermal transient and the timing for whole core involvement have been determined by the previous steps, an appropriate fission product release model can be used as long as it can incorporate temperature transients and burn up.

Some possible choices are the models in FASTGRASS, SCDAP, VICTORIA, MAAP, MELCOR and KESS. An alternative is to use a simplified model, such as the Relative Volatile, which is based on temperature and burn-up level.

The RASCAL is capable to calculate the source term of Long Term Station Blackout. RASCAL uses the core inventory release fraction specified on (NUREG-1465, 1995). Such reference presents the fission products release fraction for three phases of core degradation: gap release phase (cladding failure), core melt phase and post vessel melt-through phase, each one considered to take place on the specific time of 30 minutes, 1.3 hours and 2 hours, respectively. The core uncover is fixed as 8 hours after reactor shutdown. Afterward, if the power is not recovered, the three phases of core degradation will take place at the specific times described, when the releases fraction of (NUREG-1465, 1995) is adopted.

The RASCAL model is not flexible enough to include the several decay heat levels of the blackout accidental scenarios. Such limitation imposes the need to adopt the relative volatile model, in order to calculate the different release fractions according to the temperature pattern of the core.

The determination of the chemistry associated with the release of fission products from fuel under accident conditions is complex, and the complexity is increased by reactions between the core and the point of release to the containment. However, except the noble gases, all the fission products should have condensed before entering the atmosphere of the containment. Thus, to derive the in-containment source term, all the non-noble gas fission products in the atmosphere of the primary containment can be considered aerosols.

According to (IAEA-TECDOC-1127, 1999) the relative volatile fission products release model is much more accurate in predicting data for fission product release than CORSOR codes and is just as easy to implement.

(KRESS, 1987) have proposed such a model in which the fission product release would be expressed by a spherical diffusion equation

$$\frac{\partial C}{\partial t} = \frac{D}{r}\frac{\partial^2 (rC)}{\partial r^2} \tag{8}$$

Where,

$D = D_0 e^{-\frac{Q}{RT}}$;

T is the temperature; and

D_0 and Q are Arrhenius-like correlation parameters.

D is a sort of effective diffusion parameter to be used along with an effective spherical distance parameter, r_0 = a. (BOOTH, 1958) proposed an approximate solution for the spherical diffusion equation

$$f = \begin{cases} 6\sqrt{\frac{Dt}{\pi a^2}} - 3\frac{Dt}{a^2} & for\ \frac{Dt}{a^2} \leq 0.1 \\ 1 - \frac{6}{\pi^2} e^{-\frac{\pi^2 Dt}{a^2}} & for\ \frac{Dt}{a^2} > 0.1 \end{cases} \tag{9}$$

Where

f is the cumulative fraction of the original quantity of the fission products present at time t = 0, i.e. a percentage of the total amount of the fission product released from the core.

The thermal transient algorithm outputs the core temperature as function of time. Using it as input on the solution for the spherical diffusion equation, it is possible to obtain the percentage of fission products activity released as a function of time, which has been largely explored in this work to calculate a time dependent source term.

To determine best-fit values for D_O and Q, for each fission product, it is performed a correlation using experimental data. Data for the release of Cs in the ORNL fission product release program were best-fit on the Relative Volatile approach at a distance a = 6μm, obtaining the results (10) and (11)

$$\boldsymbol{D_O = 2.6833 \times 10^5 e^{-6.052x10^{-4}}(BU)} \qquad (10)$$

$$\boldsymbol{Q = 2.065 \times 10^5 - 3.629(BU)} \qquad (11)$$

Where

BU is the burnup in MW.day/ton;

D is in cm^2/s; and

Q is in cal/mole.

Figure 6 shows the accuracy of this correlation in fitting the ORNL VI-3 data.

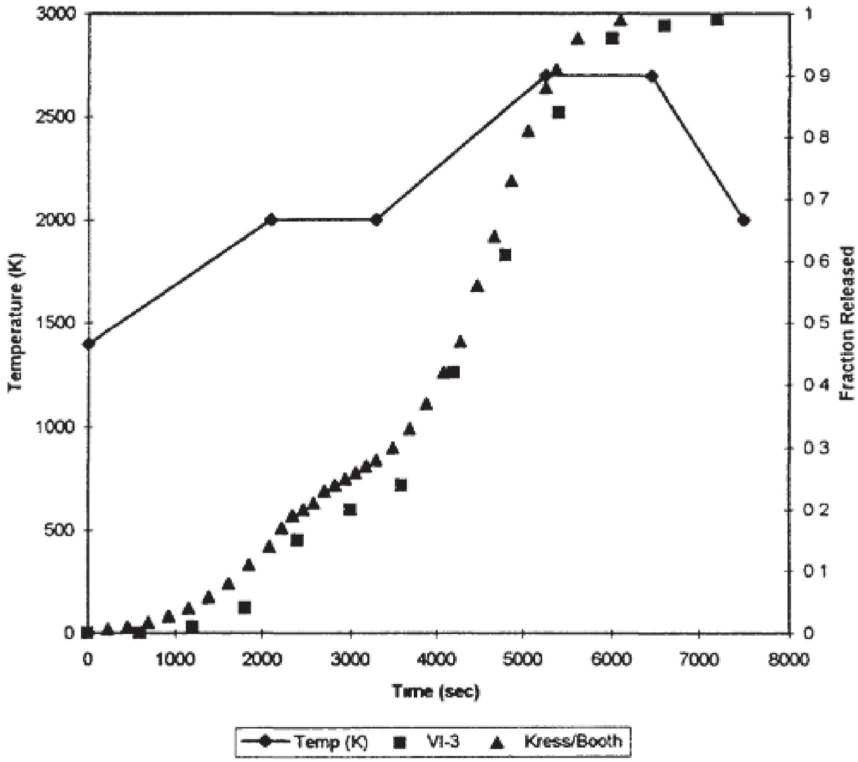

Figure 6: Comparison of Kress/Booth Cs release model with ORNL VI-3 test data (IAEA-TECDOC-1127, 1999).

The same correlation has been implemented to Sb. According to (IAEA-TECDOC-1127, 1999) the results of the correlation of ORNL VI-3 data by the diffusive spherical model are shown on (12) and (13)

$$\boldsymbol{D_O = 3.4608 \times 10^6 e^{-6.052x10^{-4}}(BU)} \quad (12)$$
$$\boldsymbol{Q = 2.494 \times 10^5 - 3.629(BU)} \quad (13)$$

Where

BU is the burnup in MW.day/ton;

D is in cm^2/s; and

Q is in cal/mole.

The release of all other species can be interpolated using the relative volatility scale. This can be done by using the Kress/Booth release model to interpolate D_0 and Q for other radionuclides. Based on Cs and Sb a relative volatility scale of Table 2 is used to establish the fractional release of other species.

Table 2: Relative volatile values for other specifies (IAEA, TECDOC-1127, 1999).

Species group	RV
NG	1.1
Te	1.07*
I	1.03
Cs	1.00
Sb	0.68
Ba	0.42
Sr	0.34
Ru	0.25
La	0.14
Ce	0.085
Structure materials	
Mn	0.69
Sb	0.68
Sn	0.63
Cr	0.47
Fe	0.44
Co	0.41
UO_2	0.17

The obtained release fraction of radionuclides, as a function of time, represents the quantity of radionuclides present on the coolant. A conservative hypothesis that these radionuclides reach the containment atmosphere has been considered. In the case of LOCA, these radionuclides reach the containment due to RCS break flow. In case of a long blackout, the reliefs of RCS up to pressurizer relief tank disk rupture leak the radionuclides to containment. Therefore, the relative volatile models calculate de in-containment source term.

The in-containment source-term will be leaked to the environment due to leakage on containment, or its failure. Except in the case of noble gases, the other radionuclides present in the containment atmosphere in the form of aerosols, suffer from gravitational deposits, decreasing the radionuclides available to be leaked to the environment. The reduction mechanism applied to the algorithm is the exponential model recommended by (NUREG-1940, 2012). The reduction mechanism effect can be increased by containment sprays, but such a possibility is not present in blackout accidents due to the lack of energy for the spray pumps.

The source term to the environment depends on the amount of leakage of the containment. In case of failure, the in-containment source term leaks to the environment, with exception of radionuclides deposited on the containment surface. In the case that the containment resists to accidental pressure loads, a little quantity of radionuclides are leaked to the environment, since a design leakage rate of 0.1% of containment volume per day is considered.

3.4 DOSE TO PUBLIC INDIVIDUAL

3.3.4 RASCAL source term to dose calculation

The source term, i.e., the amount of radionuclides activity released by time unit must be translated into dose for an individual of the public, to obtain an appropriate consequence measure. The radionuclides released to the environment will disperse on the atmosphere, according to meteorological conditions. An individual would be irradiated by gamma radiation and by inhalation. The food chain dispersion of radionuclides is not considered, since in front of an accident, the local food and water are not recommended for human consumption.

The RASCAL software is used to perform such calculations, based on the input data provided by the relative volatile model of fission products release and the containment performance of attenuation of the release of the radionuclides. This work will not discuss the several methods for dose calculation (RASCAL was explored for such an objective), as well as the biological effects of radiation because it was not possible to be exhaustive on such issue herein.

The demographic aspect would not be treated on a mobile plant, therefore the consequence measure would be limited on dose, not the number of deaths or latent cancer. On cruise at sea, a ship is far from inhabited places, reducing drastically the consequence to the public. At port or base, shore power supply and accident management can reduce the probability of blackout.

3.5 RISK ASSESSMENT

With the results of probability calculation and the results of consequence, the risk of a marine NPP is raised. In the lack of a normative base for mobile NPP, the risk level raised can be compared to operating stationary NPP, in order to be sure that the level of risk is acceptable.

In the case of an unacceptable level of blackout risk, several options can be explored, because it has been identified in early design phases. For instance, architectural alternatives for the electrical plant, increase equipment reliability or increase the repair capacity.

3.4.1 Treatment of Uncertainties

Firstly, it is important to differentiate sensitivity and uncertainty analysis.

Sensitivity analysis measures the change in the results when a variable changes on the input of the model. It estimates the risk on different values of a given variable. They determine which parameter drives the results of accident progression and source terms.

The uncertainty analysis assesses an interval of confidence in the obtained results. A parametric analysis must be performed in order to define and risk probability density function. Therefore, uncertainty analysis assigns probabilities to the risk results based on individual uncertain events and combines them to evaluate their combined effect on the results.

The probability calculations proposed in chapter 3.2 leads to random behaviors of the uncertainties. Such types of uncertainty are treated by traditional data analysis.

There are epistemic uncertainties on the source term quantification of chapter 3.3. The proposed deterministic model brings uncertainties due to the hypothesis, no precise input data and lack of modeling knowledge. Epistemic uncertainties can be reduced on time by model review based on feedback, the precision of input data, etc.

To perform the uncertainties treatment in this work, it is proposed to start by a sensitivity analysis to identify relevant input data. Afterward, classify the input data according to the generated uncertainty type (random or epistemic). A probabilistic analysis is important for uncertain random data with high sensitivity to results. The probabilistic analysis must input both the upper and lower bounds of data confidence intervals and calculate the risks of the results. For epistemic uncertainties of hypothesis, models and input data, the blackout risk must be calculated for the maximum and minimum range of values of the uncertainty.

4 BLACKOUT RISK ASSESSMENT OF A HYPOTHETICAL MARINE BASED NUCLEAR REACTOR

In order to consolidate the methodology presented in the previous chapter, data for a hypothetical marine NPP is obtained from a mix of sources, in order to exemplify how to use such data in the methodology. In despite of the results are not applicable to a specific plant, they demonstrate the methodology and how it can be useful in early stages of design.

Even though the methodology presented herein does not demand detailed data, an effort has been made to obtain coherent values. There is no marine NPP reference with complete data. Each time a data is defined, explanations are provided as far as possible to justify its applicability.

4.1 DATA ACQUISITION

The following chapters define the data used in the case study of blackout risk analysis of a generic marine NPP. The most difficult is to define coherent data for a generic plant. The hypothesis and considerations are explained hereinafter.

4.1.1 Data of reactor

A PWR is the technology adopted in the study to marine nuclear plants and adopted in this case study. Thermal power of 100 MWt on a low enriched nuclear fuel is assumed, since such power is coherent with some reactors of merchant ships already built, as N.S. Savannah (74MWt) and N.S. Lenin (90MWt). The average burn-up adopted for the study is around 30000 megawatt days per metric ton of uranium (MWd/MTU), such burn-up is coherent at end of the cycle of a stationary NPP.

The coolant mass in the reactor vessel can be obtained by extrapolation of reference values presented on (ANSI/ANS 18.1, 1999). It has been adopted as a mass for the complete RCS of 20 Tons. In addition, from this total mass, it is necessary to define the coolant mass surrounding the core, and the coolant mass below the core. Such values are specific to each reactor design. A hypothetical value of 3 Tons of water mass surrounding the core and 2 Tons of residual water below the core have been specified to enable the analysis of this work. Plant-specific values are more accurate on time to core uncover calculation, but the order of magnitude will be kept even with extrapolation. The dose associated with releases from RCS coolant only is much smaller compared with doses associated with releases of core damage, therefore the coolant inventory is more important on time to core uncover.

The curve of heat produced by the decay of the fission products is dependent on the operation history and the operation power of the reactor at the time right before the SCRAM. Moreover, no nuclear fuel specific data are available on a generic case. Therefore, in this study, decay heat cannot be calculated by methods specified in (ANSI/ANS-5.1, 2005). The generic decay heat curve formula of (KNIEF, 1992) was used.

The core inventory of fission products is extremely important to calculate the accident consequence. In a generic case, the core inventory proposed by (NUREG-1940, 2012) has been adopted with the necessary corrections according to burn-up level, described by formula (1).

The Bateman's solution used to calculate the iodine and xenon concentrations as a function of time gives the graph in Figure 7 for a Flux of $5x10^{14}$ [n/(cm^2.sec)].

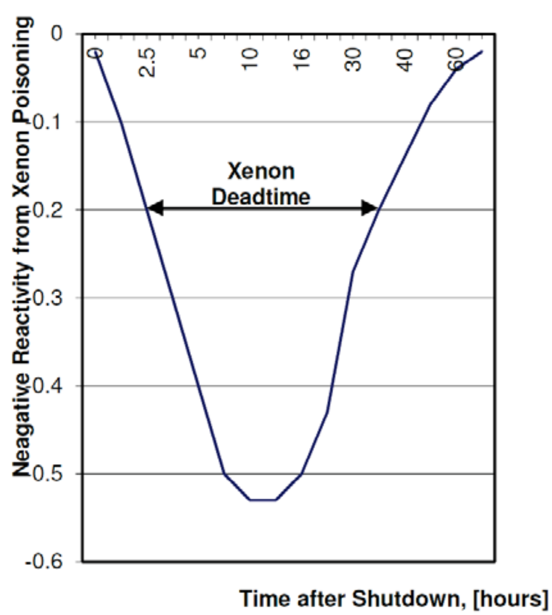

Figure 7: Negative reactivity due to xenon poisoning. Flux = 5x1014 [n/(cm2.sec)] (RAGHEB, 2011).

If at any time after shutdown, the positive reactivity available by removing all the control rods is less than the negative reactivity caused by xenon, the reactor cannot be restarted until the xenon has decayed. Assuming a reactivity reserve of 20%, the reactor cannot be restarted for up to 35 hours. Such reactor dead time has been considered in this generic case study.

4.1.2 Dimensioning and architecture of the electric power circuit

The electric power circuit adopted as a case study of a marine NPP is the one present in Figure 1. The circuit is composed of two sides for power distribution, each one powered by a DG and battery able to fully supply safety-related systems in all credible scenarios of reactor shutdown. The safety-related systems are connected on each side, configuring so a fully redundant power supply to loads. Therefore, the electric power distribution system enables the safety-related system power supply by redundancy level 2.

The focus on blackout probability evaluation is the emergency power sources. The DG sets are not the only equipment to be involved in reliability modeling, since it is dependent on its ancillaries. A DG set usually comprises five external ancillary systems:

- the diesel engines and generators control, monitor and protect installations;
- the diesel fuel oil installations;
- the lubricating oil system of the diesel engines;
- the cooling circuit of the diesel engines installation; and
- the starting and control Air system.

Figure 8 shows the DG set including external ancillaries systems. Each DG has its dedicated ancillaries.

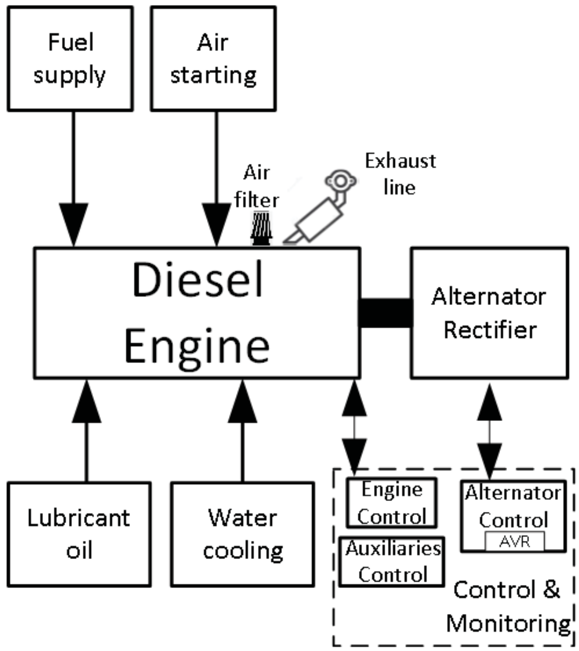

Figure 8: Diesel engine and generator including external ancillaries systems.

Figure 9, Figure 10, Figure 11 and Figure 12 show DG ancillaries systems generic architectures based on (EPRI NP-5924, 1988).

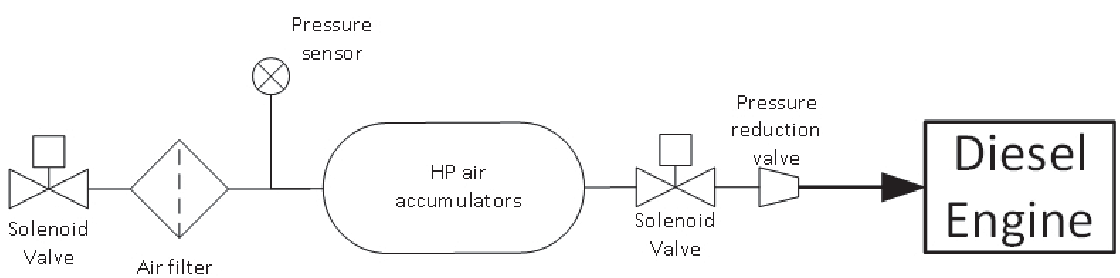

Figure 9: Air starting sub-system (EPRI NP-5924, 1988).

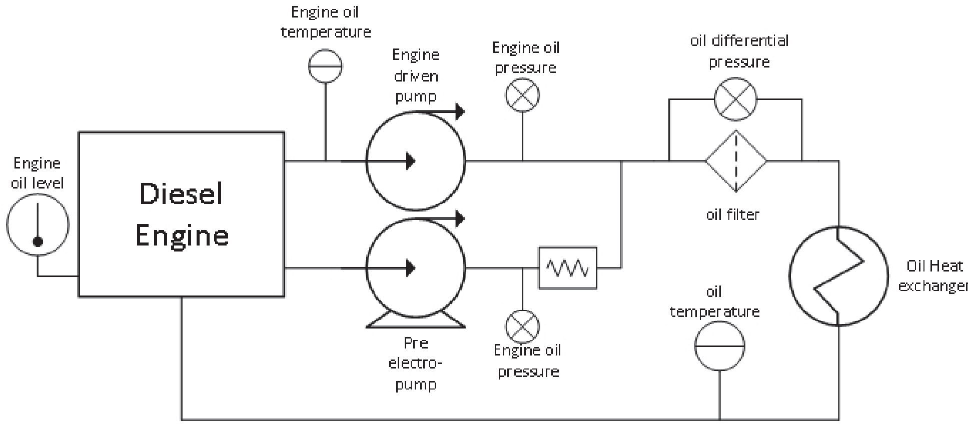

Figure 10: Lubricant oil sub-system (EPRI NP-5924, 1988).

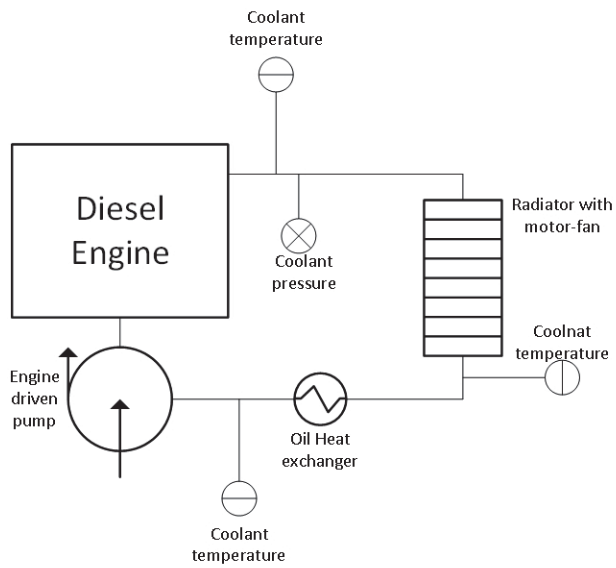

Figure 11: Water cooling system (EPRI NP-5924, 1988).

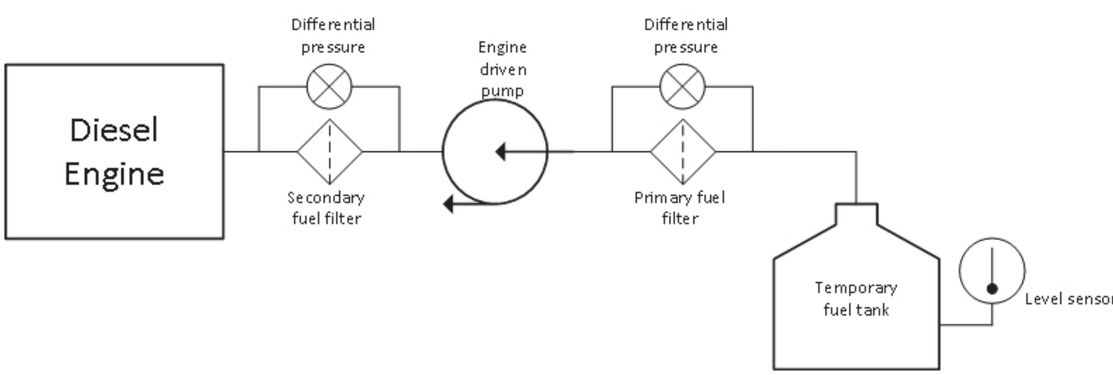

Figure 12: Fuel supply sub-system (EPRI NP-5924, 1988).

The Control & Monitoring system is composed of the LCP for ancillaries, diesel engine and a generator LCP, as can be seen in Figure 8.

4.1.2.1 Battery autonomy

The battery autonomy is a plant specific data, depending on the battery bank's capacity and the worst case of power consumption in the accidental scenario. It is postulated that the battery autonomy would be coherent with the necessary time to perform some short repairs on DG sets, decreasing the blackout probability. To illustrate the methodology in a generic case study, it has been considered that batteries can provide power to safety-related systems for around 12 hours in the worst case of power consumption in accident scenarios initiated by transients. This value is coherent with the DG MTTR for failures repairable on the accidental scenario during a ship cruise.

4.1.3 Data of containment

A cylindrical containment has been considered for the structural reliability analysis. A half sphere as stationary NPP containment presents volume constraint to be integrated on board a ship. SNN-688 Los Angeles submarine class containment data can be found on open sources. Such naval NPP has a power considered close in order of magnitude to the power adopted in this case study. Therefore, the reactor compartment adopted has a diameter of 10m and 12.8m long. The hull of SNN-688 is manufactured by HY-80 steel alloys, and plates of 19mm of thickness are considered. The adopted containment material is a generic high-strength steel, different from the submarine hull. The fracture toughness of high-strength steel can vary from 50 to 140 [Mpa x $m^{½}$]. The lower value of the stress intensity factor has been adopted for the structural reliability calculation.

4.2 PROBABILITY QUANTIFICATION

Based on the methods proposed in Chapter 3, and the electrical system architecture defined previously, a probability calculation of the electrical system reliability is performed. The following chapter explains how to acquire the reliability data, the model applied and the probability results of the several accidental scenarios on the event tree.

4.2.1 Reliability data acquisition

To acquire a repair rate of the DG set, a detailed analysis of components repair must be performed on the equipment. The generic breakdown of a diesel engine and an electric generator has been obtained from (OREDA, 2002), as well as their contribution to the equipment failure rate. For each component, it is defined the feasibility of on board repair by the crew during the accidental scenario development. Therefore, based on each component failure rate contribution, and its feasibility of repair or not, a repair rate factor of the equipment is calculated, from the ratio between repairable components failure rate, and the global failure rate. The Table 3 shows the repair rate factor resulting from such analysis for the diesel engine and the related electric generator.

To calculate the repair rate factor of the diesel engine and electric generator set, it must be sum the repair rate of both equipment, considering a proportional factor of the failure rate contribution of each equipment to the DG set. It has been performed to obtain the Diesel-Generator Repair Rate on the last line of Table 3.

Table 3: DG Generic Breakdown with failure rate contribution (OREDA, 2002).

Diesel engine Failure Rate [%]		57.4%	Electric Generator Failure Rate [%]		42.6%
Item	**Failure Rate [%] pg246**	**Repairable**	**Item**	**Failure Rate [%] pg303**	**Repairable**
Actuating Device	2.55	X	Actuating Device	13.81	X
Air Inlet	2.08	X	Anti cond. heater	1.05	
Cabling	0.46	X	AVR	2.1	X
Control unit	3.82	X	Cabling	1.4	X
Cooler(s)	0.69		Control unit	7.69	X
Cylinders	1.39		Excitation	1.05	X
Exhaust	3.01		Fan w/ motor	2.1	X
Fan w/ motor	0.69	X	Heat exchanger	0.35	
Filter(s)	3.47	X	Instrument	18.89	X
Fuel Filter	1.97	X	Power supply	11.71	X
Fuel Pump	1.39	X	Monitoring	6.64	X
Heat exchanger	2.78		Other	16.08	
Heater	1.85	X	Piping	0.7	
Hood	1.16		Radial bearing	0.35	
Injections	4.4		Stator	0.35	
Instrument	23.37	X	Subunit	4.9	X
Power supply	0.23	X	Unknown	10.49	
Monitoring	1.16	X	Valves	0.35	X

Diesel engine Failure Rate [%]		57.4%	Electric Generator Failure Rate [%]		42.6%
Item	**Failure Rate [%] pg246**	**Repairable**	**Item**	**Failure Rate [%] pg303**	**Repairable**
Oil	0.69				
Other	1.62				
Piping	8.45				
Piston	0.12				
Pump	1.39	X			
Pump w/motor	0.93	X			
Radial bearing	0.23				
Seals	3.24				
Shaft	0.12				
Start control	0.93	X			
Start energy	6.48				
Starting unit	2.08	X			
Subunit	3.47	X			
super charger	1.16				
Timing chain/belt	0.23				
Unknown	7.06				
Valves	5.32	X			
Σ repairable items		**57%**	**Σ repairable items**		**71%**
Diesel-Generator Repair Rate					**62.9%**

The failure rate of the DG set has been obtained from open databases like (EIREDA,1998) and (OREDA, 2002). The acquired data has been summarized in Table 4. Each line of Table 4 corresponds to an equipment of Figure 8, Figure 9, Figure 10, Figure 11 or Figure 12. Afterwards, the reliability data is summed up at DG set level.

Table 4: Reliability data adopted on the generic case study.

Equipment	Lambda	MTBF	MTTR	Repair rate	Failure modes	Source
Diesel Generator & Ancillaries	**4.11E-04**	**2,432**	**22**	**64%**		
Diesel Engine	3.93E-04	2,545	22	63%	Loss of performance Fail to start	(EIReDA,1998) pg 187
Air starter system	2.31E-06	432,900	15	94%		
Solenoid valves	5.00E-07	2,000,000	5	100%	Wont open Leakage	(OREDA,2002)
HP Air accumulators	1.30E-07	7,692,308	88	0%	Leakage	(OREDA,2002)
Pressure reduction valves	3.80E-07	2,631,579	12	100%	Clogged Leakage	(OREDA,2002)
Air filter	5.40E-07	1,851,852	12	100%	No/low flow	(OREDA,2002)
Instrumentation	7.60E-07	1,315,789	12	100%	No/erratic output	(OREDA,2002)
Lubrication system	4.13E-06	242,131	14	90%		
Engine driven pump	2.20E-07	4,545,455	9	100%	Loss of performance Leakage	(OREDA,2002)
Heat exchanger	4.30E-07	2,325,581	36	0%	Leakage Flow restriction	(OREDA,2002)
Oil Filter	5.40E-07	1,851,852	8	100%	No/low flow	(OREDA,2002)
Insturmentation	2.50E-06	400,000	12	100%	No/erratic output	(OREDA,2002)
Pre electro-pump	1.50E-07	6,666,667	9	100%	Loss of performance Leakage Fail to start	(OREDA,2002)
Pre heater	2.90E-07	3,448,276	17	100%	No operation	(OREDA,2002)
Cooling/Pre-heating system	3.96E-06	252,525	6	89%		
Engine driven pump	2.20E-07	4,545,455	9	100%	Loss of performance Leakage	(OREDA,2002)
Radiator	4.30E-07	2,325,581	7	0%	Leakage Flow restriction	(OREDA,2002)
Fan w/ motor	1.10E-07	9,090,909	27	100%	Loss of performance Fail to start	(OREDA,2002)
Instrumentation	3.20E-06	312,500	5	100%	No/erratic output	(OREDA,2002)
Fuel System	2.13E-06	469,484	12	95%		
Tank	1.00E-07	10,000,00 0	12	0%	Leakage	(OREDA,2002)
Engine driven pump	2.20E-07	4,545,455	8	100%	Loss of performance Leakage	(OREDA,2002)
Filter	3.10E-07	3,225,806	13	100%	No/low flow	(OREDA,2002)
Instrumentation	1.50E-06	666,667	12	100%	No/erratic output	(OREDA,2002)
Control & Monitoring	5.62E-06	178,063	10	100%		
Engine Control unit	6.00E-07	1,666,667	15	100%	No control	(OREDA,2002)

Equipment	Lambda	MTBF	MTTR	Repair rate	Failure modes	Source
Generator Control Unit	5.60E-07	1,785,714	15	100%	No control	(OREDA,2002)
Auxiliaries Control Unit	6.00E-07	1,666,667	15	100%	No control	(OREDA,2002)
Instrumentation/ Cabling/ Junction Boxes	1.70E-07	5,882,353	9	100%	No/erratic output	(OREDA,2002)
AVR	1.50E-07	6,666,667	11	100%	No/erratic output	(OREDA,2002)
UPS	2E-06	491,159	8	100%	No power	(OREDA,2002)
Circuit Breaker	1.50E-06	666,667	8	100%	Wont open/close	(OREDA,2002)

Regarding the maintainability data, the MTTR for DG set applies to all kinds of failure, including failures considered as repairable and non-repairable in Table 3. An MTTR of 12 hours has been considered coherent for repairable failures. Such value must be well demonstrated based on the ship layout of the DG and its ancillaries, onboard spare parts stock and tools, and crew members' maintenance capacity.

4.2.2 Frequency of transients leading to SCRAM

Analysis of initiating event rates is important because it indicates performance among plants and provides the frequency of emergency power sources' demand. (NUREG/CR-5750, 1999) presents an analysis of initiating event frequencies at U.S. commercial nuclear power plants. The evaluation is based on the operating experience from 1988 through 2013.

Table 5 shows the events frequency summary. LOCA and SG tube rupture is not considered, since such events cumulated with blackout is not credible accident (CARVALHO, 2004). Also, Table 5 shows a comparison of events frequency to other references, like (NUREG/CR-3862, 1999) and (NUREG-1150, 1999).

Table 5: Initiating Event frequencies based on operating experience compared to NUREG/CR-3862 and NUREG-1150 (NUREG/CR-5750, 1999).

Description	Mean Frequency (per critical year)[b] Operating Experience[a]	NUREG/CR-3862[c]	NUREG-1150[c]
Small Pipe Break LOCA (G3)	5E-4[d]	—	1.3E-3[d]
Steam Generator Tube Rupture (F)	7.0E-3	—	1.0E-2
Loss of Offsite Power (B)—PWR	4.6E-2[d]	1.9E-1	1.9E-1
Loss of Offsite Power (B)—BWR	4.6E-2[d]	1.1E-1	1 1E-1
Total Loss of Condenser Heat Sink (L)—PWR	1.2E-1	2.4E-1	2.4E-1
Total Loss of Condenser Heat Sink (L)—BWR	2.9E-1	9.1E-1	9.1E-1
Total Loss of Feedwater Flow (P) —PWR	8.5E-2[d]	2.2E-1	2.2E-1
Total Loss of Feedwater Flow (P) —BWR	8.5E-2[d]	9.3E-2	9.3E-2
General Transient—PWR (Q)	1.2E+0	1.0E+1	1.0E+1
General Transients—BWR (Q)	1.5E+0	8.6E+0	8.6E+0
Total of all events—PWR	1.4E+0	1.1E+1[e]	1.1E+1[e]
Total of all events—BWR	1.8E+0	9.7E+0[e]	9.9E+0[e]

a 1987–1995 experience except for Small Pipe Break LOCA category which included total U S operating experience (1969–1997)

b Units are in per critical year One critical year equals 8,760 hours of reactor criticality

c The units stated in the report are per reactor year (i e , numbers of years from start of commercial operation) For comparison purposes, the per reactor year was converted to critical year One critical year equals one calendar year divided by the fraction of time the reactor was critical; 75% criticality factor was used based on the results of this study Therefore the rate per critical year equals the rate per calendar year divided by 0.75

d The estimate did not differentiate with respect to plant type (i e , PWR and BWR); therefore the value is same for either plant type

e. This total represents the sum of all frequencies presented in the referenced report.

The frequency of transients leading to SCRAM adopted in the study is 1.4 transients per reactor.years. The transients of category 2 are minor incidents considered frequent. Even if they lead to reactor SCRAM, it can be postulated that the reactor would be able to be restarted after the dead time. The transients of category 3 are unlikely incidents, which cause the impossibility of safely restarting the reactor. Since the category 2 and 3 transients have an annual frequency ratio of around 100, it can be postulated that after reactor SCRAM, there is a probability of around 99% to restart the reactor after dead time, since it is expected that only 1% of transients are from 3rd category.

(NUREG/CR-5750, 1999) also analyzed the frequency of Anticipated Transients without SCRAM. Such an accident is severe since some control rods refuse to drop and the reactor cannot be shutdown, staying in power and demanding a cooling that safety systems are not dimensioned for. The ATWS leads to core damage in a short time. Table 6 shows the frequency of ATWS considered in this case study.

Table 6: ATWS frequencies based on operating experience compared to SECY-83-293 (NUREG/CR-5750, 1999).

	PWR	BWR
1987–1995 experience	8.4E-6	3.3E-6
SECY-83-293	2.4E-5	1.2E-5

A frequency of around 10^{-5} by demand has been considered for ATWS.

4.2.3 Dynamic reliability analysis of emergency power sources

A reliability modeling of the electric power circuit has its scope reduced to diesel generators, the external auxiliary systems and the batteries because these equipment are the main failure contributors. The remaining equipment in the electric power distribution system is non-sensible and/or passive equipment, which usually are low risk contributors and not considered in probabilistic analysis. The reduction of scope to main failure contributors still allows demonstration of the methodology, the main goal of this chapter.

The analysis performed illustrates the state-of-art on dynamic reliability analysis, including the Partially Repairable System (PRS) proposed by (SOUZA; GABE, 2017). This reference proposes to consider repair on reliability since short time repairs are feasible for many safety and emergency scenarios. Critical failures are non-repairable failures. The reliability model of Figure 13 includes characteristics not modeled in Reliability Block Diagrams and Fault Trees. It includes a stand-by partially repairable model, a single maintenance team to perform repairs and the batteries contributing to reliability during its autonomy (time that permits the performance of repairs on DG).

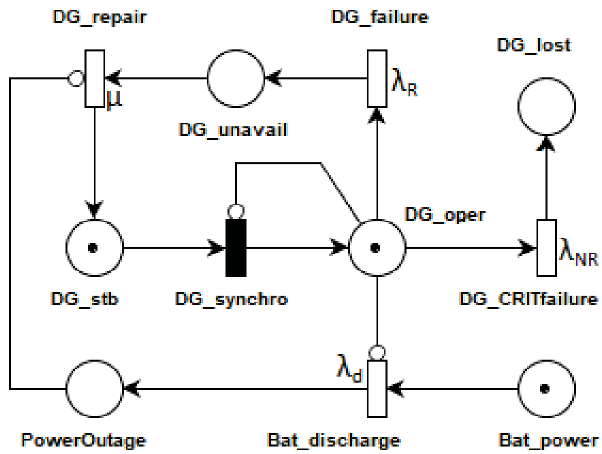

Figure 13: Stand-by redundant of partial repairable equipment and battery contribution on reliability model.

The Generalized Stochastic Petri net has an initial condition, marked by tokens, with a DG operating, while another is on standby. Each DG when operating can fail by two distinct failure modes:

- Critical: such failure mode happens with an average time equal to MTBCF. Such failure mode does not permit the repairs by the crew, therefore the DG is considered lost.
- Repairable: such failure mode happens with an average time equal to MTBF. Such failure mode permits the repair, and DG is considered unavailable temporarily. The repairs start with crew members taking an average time MTTR.

In the case of a DG experiencing any kind of failure mode, the other DG on stand-by place suffers a transition to operating place. When both DG are unavailable or lost simultaneously, the token on place Bat_power is enabled to be fired to PowerOutage in a time after battery autonomy. If an unavailable DG is repaired before the Bat_discharge firing, it goes to DG_oper place, and no blackout takes place (token at PowerOutage place).

The boundless of PN of Figure 13 is ensured. Since the PN models reliability, the system is composed of some absorbing states and liveness is not reached. In this case, only transient analysis is possible. The place invariants of Figure 13 PN emphasize the model correctness,

$$\begin{cases} M(Bat\_power) + M(PowerOutage) = 1 \\ M(DG\_stb) + M(DG\_lost) + M(DG\_oper) + M(DG\_unavail) = 2 \end{cases}$$

Through invariants analysis it is possible to verify that no power outage takes place if batteries have power. There are several Petri Nets software to assist the analysis described. The Platform Independent Petri Net Editor have been used to obtain the reachability graph. The reachability graph of Figure 13 is shown in Figure 14. It is necessary to remove the vanish states (represent in blue), resulting in the reduced reachability graph, which is isomorph to the Markov Chain of the stochastic process of Figure 4 PN.

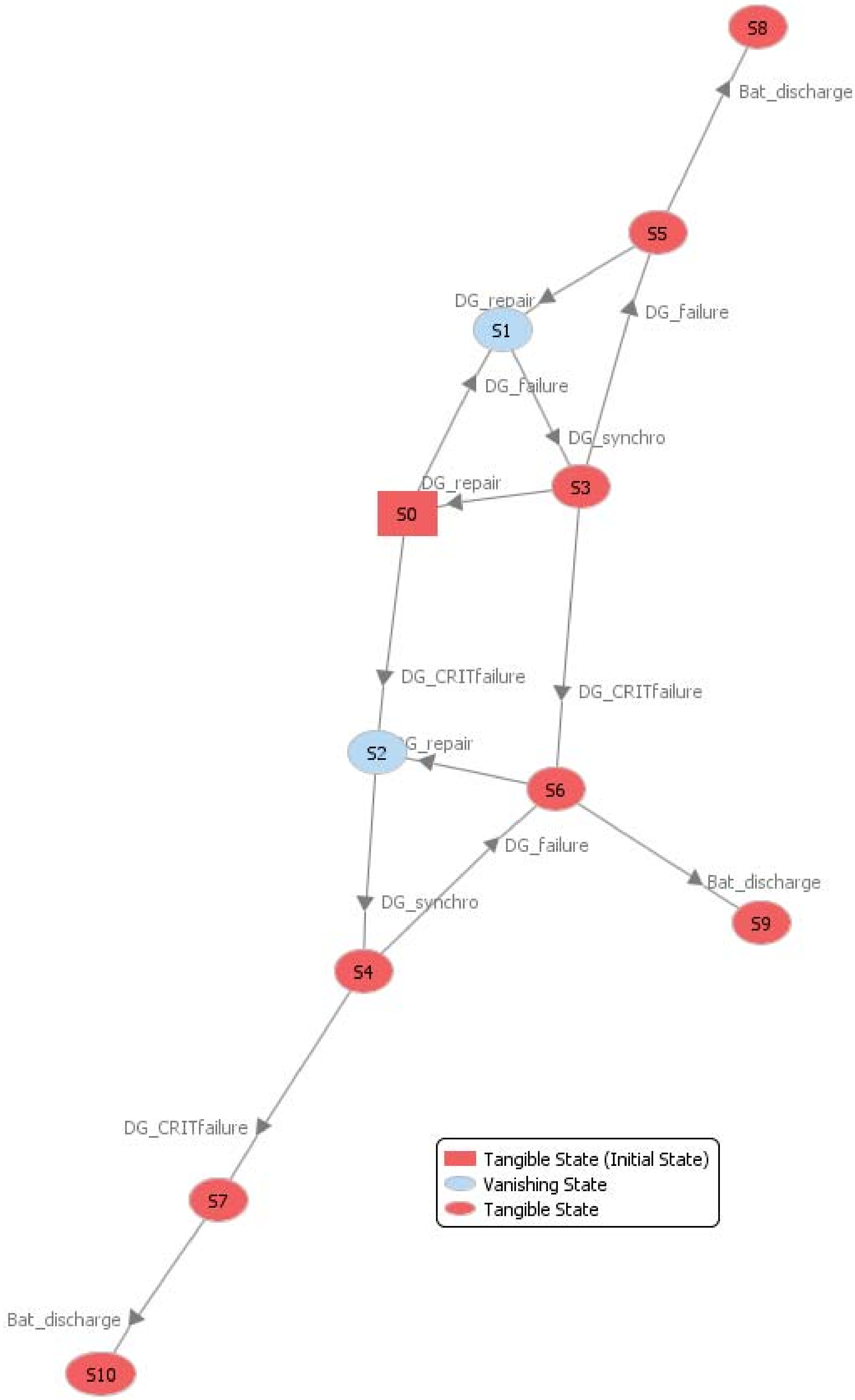

Figure 14: Reachability graph of the GSPN of Figure 13.

From the Markov Chain isomoph to the reduced reachability graph, the system state equations can be obtained, enabling even the analytical solution. However, for a 9 states system, the analytical soluction is cumbersome. Therefore, for the case study of the work, the numerical solution had been explored, bringing relative results precision. For numerical solution, the first step is to building the Discrete Markovian Transition Matrix Q, obtained from the system states equation as per

$$
Q = \begin{array}{c}
\begin{array}{ccccccccc} S0 & S3 & S4 & S5 & S6 & S7 & S8 & S9 & S10 \end{array} \\
\left|\begin{array}{ccccccccc}
1-(\lambda_R+\lambda_{NR})*\Delta T & \mu*\Delta T & 0 & 0 & 0 & 0 & 0 & 0 & 0 \\
\lambda_R*\Delta T & 1-(\lambda_R+\lambda_{NR}+\lambda_d)*\Delta T & 0 & \mu*\Delta T & 0 & 0 & 0 & 0 & 0 \\
\lambda_{NR}*\Delta T & 0 & 1-(\lambda_R+\lambda_{NR})*\Delta T & 0 & \mu*\Delta T & 0 & 0 & 0 & 0 \\
0 & \lambda_R*\Delta T & 0 & 1-(\mu+\lambda_d)*\Delta T & 0 & 0 & 0 & 0 & 0 \\
0 & \lambda_{NR}*\Delta T & \lambda_R*\Delta T & 0 & 1-(\mu+\lambda_d)*\Delta T & 0 & 0 & 0 & 0 \\
0 & 0 & \lambda_{NR}*\Delta T & 0 & 0 & 1-\lambda_d*\Delta T & 0 & 0 & 0 \\
0 & 0 & 0 & \lambda_d*\Delta T & 0 & 0 & 1 & 0 & 0 \\
0 & 0 & 0 & 0 & \lambda_d*\Delta T & 0 & 0 & 1 & 0 \\
0 & 0 & 0 & 0 & 0 & \lambda_d*\Delta T & 0 & 0 & 1
\end{array}\right|
\end{array}
\begin{array}{c} \\ S0 \\ S3 \\ S4 \\ S5 \\ S6 \\ S7 \\ S8 \\ S9 \\ S10 \end{array}
$$

The Kolmogorov differential equation (6) has been discretized, assuming the following form

$$\boldsymbol{\pi}(\boldsymbol{t}+\Delta \boldsymbol{t}) = \boldsymbol{\pi}(\boldsymbol{t})\boldsymbol{Q}$$
$$\boldsymbol{\pi}(\boldsymbol{n}\Delta \boldsymbol{t}) = \boldsymbol{\pi}((\boldsymbol{n}-\mathbf{1})\Delta \boldsymbol{t})\boldsymbol{Q}$$

Where

$\boldsymbol{\pi}(\boldsymbol{n}\Delta\boldsymbol{t})$ is the probability vector of n step

$\Delta\boldsymbol{t}$ is the time step

Q is the transition matrix

The time dependent probability vectors $\boldsymbol{\pi}(\boldsymbol{n}\Delta\boldsymbol{t})$ numerical solution is obtained through the iterative resolution of Kolmogorov differential equation, starting by the system in a fully available state S0, by the initial condition vector

$$\boldsymbol{\pi}(\mathbf{0}) = \begin{bmatrix} \mathbf{1} \\ \mathbf{0} \\ \vdots \\ \mathbf{0} \end{bmatrix}$$

The unreliability has been obtained by the probability of reaching the states S_8, S_9 and S_{10}. It resulted in a probability to fall into the absorbing states that represent power blackout equal to 2 * 10^{-3} in 2 weeks.

The blackout probability is greater on the last day of this accidental scenario. However, during such an accidental scenario, the decay heat is reduced and the time to core uncover increases. Increasing the time to core uncover, the probability of repairing at least one DG increases.

The time to core uncover is calculated by the simplified thermo-hydraulic transient algorithm, explored in chapter 4.3.1.

Figure 15 shows the graph of blackout probabilities calculated by the model in Figure 13. The worst case calculated is at 300 hours, since after that it is considered that the port or external support is reached before the core uncover. The accidental scenario duration must be defined case by case, depending of the ship route.

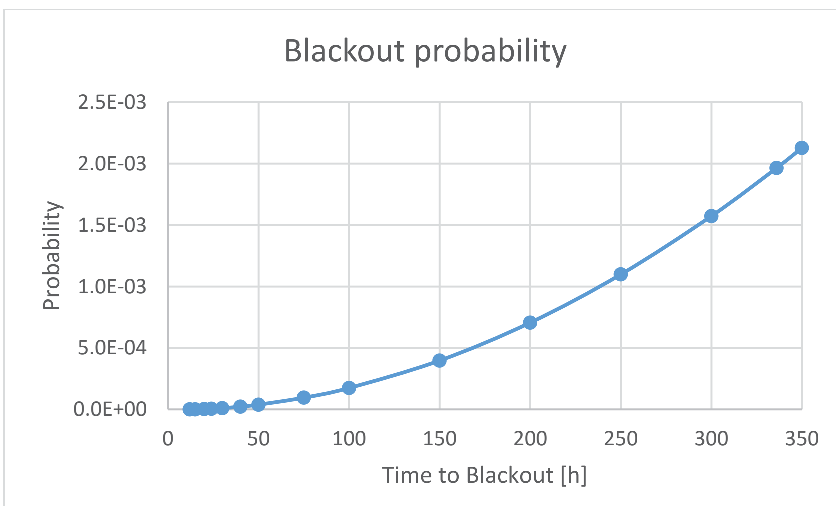

Figure 15: Probability of blackout and time to core uncover.

4.2.4 Containment Structural Reliability analysis

The containment structural reliability is analyzed by the probability of containment fracture failure mode. The probability of collapse, when the stress overcome the Tensile yield strength, is usually much lower compared to fracture probability.

For the infinite plate illustrated in Figure 16, according to (ANDERSON, 2005) the stress-intensity factor is given by formula (14)

$$\boldsymbol{K_I = \sigma\sqrt{\pi a}} \tag{14}$$

Where

$\boldsymbol{K_I}$ is the stress-intensity factor caused by applied stress;

$\boldsymbol{\sigma}$ is the stress; and

$\boldsymbol{a}$ is the flaw size.

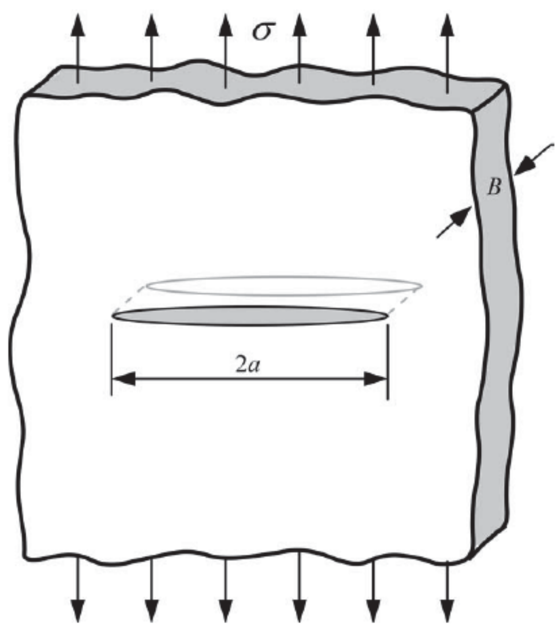

Figure 16: Through thickness crack on an infinite plate subject to tensile stress (ANDERSON, 2005)

The crack size is defined as a crack detectable size during inspections of the structure. A value of 500 μm has been initially adopted.

The stress, caused by internal pressure, on a cylinder with a radius much greater than its thickness, is given by the formula (15)

$$\boldsymbol{\sigma = \frac{Pr}{t}} \tag{15}$$

Where

P is the internal pressure;

r is the radius of the cylinder; and

t is the thickness of the cylinder.

A state limit equation Z of (16) can be defined as the material maximum stress intensity reduced by the stress provoked on the crack by the internal pressure on the cylinder,

$$\boldsymbol{Z = K_{IC} - \frac{Pr}{t}\sqrt{\pi a}} \qquad (16)$$

Where

$\boldsymbol{K_{IC}}$ is the material maximum stress-intensity factor.

To avoid the fracture, the Z must be greater than zero. However, such equation can have several random variables on stress and even the material characteristics. The statistic solution method consists of considering a probability density function for each random variable and then calculating the probability density function for the limit state equation. The fracture failure probability can be obtained by calculation of the probability that the limit state equation assumes the value zero or less.

Assuming that the random variables are normally distributed, a linear limit state equation will also have a probability density function normally distributed. To calculate the mean and standard deviation of the limit state equation, basic concepts of statistics are used when the limit state function is linear. For a non-linear equation, a solution is expanding the function on the Taylor series, on the point of mean of each variable, and considering only the first order terms.

For the calculation of structural reliability, a reliability index β is used. Such reliability index relates the mean and standard deviation as follows on (17),

$$\boldsymbol{\beta = \frac{\mu_Z}{\sigma_Z}} \qquad (17)$$

The failure probability can be calculated by,

$$P_f = 1 - \emptyset(\beta) \quad (18)$$

This method presents precise results only when Z is linear and all random variables are normally distributed. To calculate the non-linear case of equation (16), the algorithm of the advanced conditional probabilistic method proposed by (SOUZA, 2001) has been applied. The method of (SOUZA, 2001) enables to calculate the probability of non-linear limit state equation with linearization points outside random variable mean values.

4.2.4.1 Containment over pressurization

Regarding the containment pressurization by the release of coolant through the relief RCS valve, it is necessary to identify the maximum pressure that can be reached in each accidental scenario. Such pressure is used to obtain the structure stress and subsequently the failure probability.

To calculate the containment overpressure, it has been considered the worst case in which the total coolant inventory is vaporized and released to the containment by the operation of PZR relief valve. The calculation also considers the dissipated heat from containment to the environment. An impacting data for primary circuit pressurization is the containment free volume for coolant expansion. It is considered herein that half of the containment volume is available for coolant expansion. Half occupation of a ship compartment is a crowded compartment. Design specific data can be used to reduce conservatism of this hypothesis.

Considering that the containment has a heat transfer coefficient of around 45 kW/°C.m^2, and that the blackout takes place right after the transient, where the decay heat is the maximum, it has been simulated a peak pressure of around 20 bar. When the decay heat is high, the coolant is quickly vaporized, and low heat is dissipated to the environment. On other hand, the blackout scenario taking place 300 hours after the SCRAM will result in a containment pressure of around 4.3 bar as peak pressure.

At 20 bar pressure, the failure taken into account is the containment fracture. This failure mode leads to the worst consequence, because the containment fails, leaking all internal atmosphere to the environment. The containment penetrations of cables and pipes are the most fragile components, also susceptible to leak containment atmosphere. They are considered to fail on the lower 4.3 bar pressure.

Table 8 shows the input data adopted on the calculation of fracture failure probability of the containment.

Table 7: Input data for calculation of fracture failure probability.

Variable	μ	σ	Unit	Distribution
Stress intensity fator (Kic)	50	5	Mpa*m$^{1/2}$	NORMAL
Pressure (P)	2	0.2	Mpa	NORMAL
Thickness (t)	0.019	0.0019	m	NORMAL
Radius (r)	10	0.05	m	NORMAL
Flaw size (a)	0.0005	-	m	Cte

The result obtained by application of the advanced conditional probabilistic method for fracture failure using the algorithm proposed by (SOUZA, 2001) is a failure probability of containment fracture of 1.5 x 10^{-1}, based on the data in Table 8.

The blackout scenario with a higher probability takes place after 300 hours of the SCRAM, according to Table 7. In such case, with a peak pressure of 4.3 bar, it has been postulated failure on containment penetrations at the order of probability of 10^{-2}. This containment by-pass failure mode results in a lower leakage rate when compared to containment fracture failure mode. A reference value of 5% of containment volume per hour has been applied.

4.2.5 Event tree probabilities

In the previous chapters, the basic events that build up the event tree of Figure 4 have their probability calculated. Figure 17 shows the probability results of each accidental scenario.

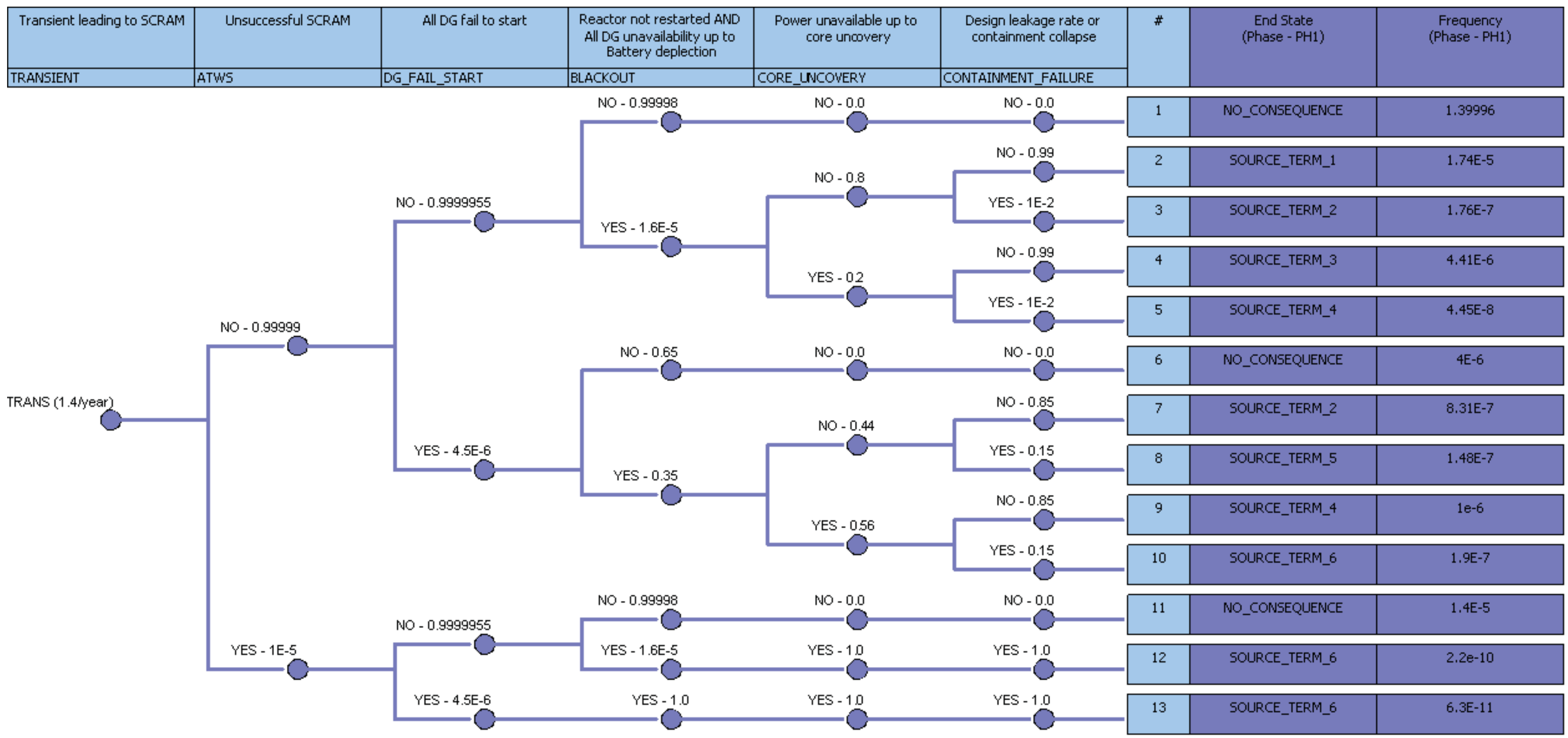

Figure 17: Results of probability for the accidental pathways of blackout.

Hereafter short explanation of each accidental scenario of the event tree in Figure 4:

- **#1:** After the successful SCRAM, the emergency power sources do not fail up to the reactor being restarted or to reach external support. Therefore, no accident or consequence is expected.
- **#2:** The reactor is not restarted, and both DG are unavailable up to battery depletion. The blackout takes place when the decay heat is low. However, at least one DG is recovered by the repairs performed on board by the crew. The core is not uncovered, but some consequences can be expected since the coolant release to the containment atmosphere has fission products according to the inventory specified on (NUREG-1940, 2012). The containment does not suffer pressurization because the decay heat is low. No containment bypass takes place, but the containment design leakage rate is applied.
- **#3:** Similar scenario to #2, however the containment penetrations fail.
- **#4:** No power is recovered after the blackout, therefore even at low decay heat, the coolant is vaporized up to the top of the core uncover. The core is damaged and a great amount of fission products are released to the containment atmosphere. The containment does not fail and only a small part of them is leaked to the environment by the design leakage rate.
- **#5:** Similar to #4, however, the containment penetrations fail, releasing radionuclides to the environment.
- **#6:** Even if both DG fails to start, the batteries provide the power to safety-related systems keeping the cooling of the core. During battery supply, power is recovered, and the blackout does not take place.
- **#7:** Power is not recovered in due time to avoid a blackout, but the core is not damaged probably by at least one DG recovered. Only radionuclides present on coolant are released at containment. In despite of the blackout taking place at high decay heat, the containment does not fail by fracture, but their penetrations fail causing bypass containment failure mode. A low inventory of coolant radionuclides is leaked to the environment.
- **#8:** Similar to #7, however, the peak pressure of containment causes a fracture, releasing the coolant radionuclides to the environment.

- **#9:** DG is not repaired in due time to avoid core uncover. A great amount of radionuclides are released at containment. Even if the blackout takes place when decay heat is high, the containment does not fail by fracture and radionuclides are leaked by containment penetrations bypass.
- **#10:** Similar to #9, however, the peak pressure of containment leads to fracture, releasing the radionuclides from the containment to the environment. Such event constitutes the worst credible accident.
- **#11:** The scenario of ATWS, but without blackout, no consequence takes place.
- **#12:** The scenario of ATWS cumulated with blackout leads to core uncover in a very short time. This accident is not mitigated by the containment, because the containment reaches a peak pressure in which the fracture can be postulated.
- **#13:** Similar to #12, but the blackout takes place right after the SCRAM.

The consequence to an individual of the public of the different source term categories is analyzed in the next chapters. The event from #12 and #13 has been neglected due to their low level of probability, and a consequence level equal to #10.

4.3 SOURCE-TERM QUANTIFICATION

The first step for source term quantification is the definition of the *“Time to core uncover”* accounted from the initiating event. The *“Time to core uncover”* does not affect the amount of fission products release, but the time when it starts. When the core uncover takes place, the (IAEA-TECDOC-1127, 1999) proposes a simple, but robust algorithm that can model the essence of the thermal-hydraulic behavior after it, without having to resort to complex codes. Capabilities are included in the algorithm, for instance a time dependent source term, containment fission products reductions mechanisms and different containment leakage rates. A computational code on Excel Visual Basic has been developed to perform the modeling of the simplified source term, and export it to *"CSV"* structured files for importation on RASCAL software. The RASCAL software is used to calculate the source term for the radiological dose of an individual of public. Details of the algorithm used for source term quantification can be obtained in the code of Annex A.

4.3.1 Simplified thermo-hydraulic transient analysis

A model based on the first thermodynamic law has been implemented on annex A, based on the coolant mass to be vaporized for core uncover, and the latent heat at primary circuit pressure, it is possible to know the necessary heat to be transferred from the core to coolant by decay heat transfer. From the generic decay heat curve of (KNIEF, 1992), the spent time to produce the amount of heat to core uncover is calculated.

The necessary input data of the plant to perform the thermal transient of the core can be seen in Table 9. Such data is important for both, the thermal transient of the reactor and the fission products release factor. The input data of the case study of the hypothetical marine nuclear power plant is summarized in Table 9.

Table 8: Input data of the hypothetical marine nuclear power plant for General Thermal Transient calculation.

Plant data	VALUE	UNIT
Reactor Normal Operating Power	100	MWth
Average Burn-Up	30,000	MWD/Ton
Coolant Mass	20,000	Kg
Water Mass Covering Core	3,000	Kg
Residual Water Below Core	2,000	Kg
Primary circuit pressure	130	bar
High Safety Injection Injentory	0	Kg
Fuel/clad sensible heat transfer	0.9	MJ/K
Runaway ZR Oxidation Heat Up Rate	21	K/s

Afterward, it is necessary to provide the accident to be modeled. The successful reactor shutdown or not is the most impacting data. In the case of unsuccessful shutdown (ATWS), the *"Time to core uncover"* is calculated from the normal operating power of the reactor, otherwise the decay heat is applied. At the instant that a blackout takes place, it is important to know the decay heat intensity. For instance, if a blackout takes place one week after shutdown, the reactor takes a longer time to vaporize coolant up to core uncover, because the decay heat is low.

Based on the provided plant and accident input data, the time events of the thermal transient are calculated. The most important time events calculated are:

- Time to Core Uncover;
- Start of Fission Products Release;

- Time to Adiabatic Heat Up;
- Time to Zr Oxidation; and
- Time to Melt.

The time to core uncover as a function of the time to blackout after the SCRAM is plotted in Figure 18.

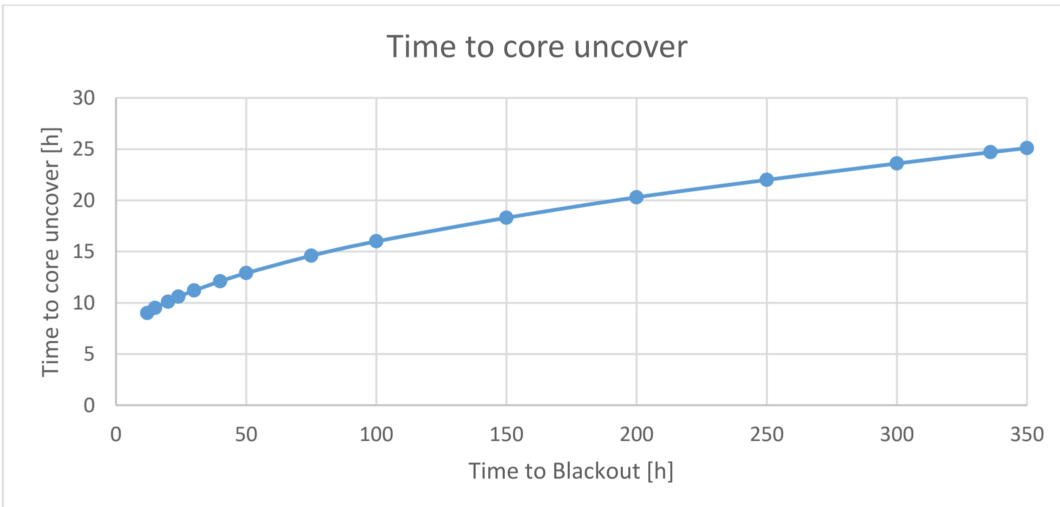

Figure 18: Results of time to core uncover according to a different time to blackout.

4.3.2 Relative volatile model of fission products release

The source terms resulting from the different accidental pathways in Figure 4 are analyzed on this chapter. The source terms 1 and 2 do not involve core damage, since the core is not uncovered. The only fission products released on these source terms are the ones present on the coolant, according to the inventory recommended by (NUREG-1940, 2012). On other hand, the source terms 3 and 4 involve core damage. On source terms 1 and 3, the fission products are contained and just released by the design leakage rate, which represents a small value of 0.1% of the containment atmosphere by day, according to the recommendation of (NUREG-1940, 2012). On source terms 2 and 4, the containment fails on the cables and pipes penetrations, resulting in bypass failure mode. The leakage rate attributed to such event is 5% of the containment volume by hour.

The source term 5 does not involve core damage, however the containment fails by fracture, releasing 100% of fission products in one hour. The most severe source term is the 6 since the core is damaged and the fission products on containment are 100% released to the atmosphere in one hour.

The generalized thermal transient algorithm provides the core temperature as a function of time. The time dependent temperature is applied to the solution for the spherical diffusion equation to calculate the amount of fission products released from the core as a function of time. The fission products activity released from the core is considered to spread homogeneously on the RCS. The radioactivity on RCS can be entirely leaked to the containment atmosphere in the case of LOCA. A complete release of RCS radioactivity to containment is considered in the case of long term blackout because the pressurized relief tank will suffer disk rupture. Table 10 shows the input data of containment used to measure the containment performance in front of the accidental scenarios.

Table 9: Containment input data to calculate the source term to the environment.

Containment data	VALUE	UNIT
Containment Free Volume	471	m^3
Heat Tranfer coefficient (Cont2Env)	45	kW/°C.h.m^2
Leakage Rate	5%	%vol/h
Concentration reduction rate 1	0.64	
Concentration reduction rate 2	0.15	
Reduction rate changing time	1.75	h

The concentration reduction mechanisms have been considered, according to recommendations of (NUREG-1940, 2012). These mechanisms as gravitational settling have an impact on the concentration of non-gaseous fission products, especially when the containment does not experience failure. Water mist sprays to reduce activity concentration in the containment atmosphere are not available in case of a blackout.

Table 11 shows the results of release fraction to the environment by radionuclide. The sources term involves core damage, and are calculated for 2 hours and 24 hours accounted from the start of core release.

Table 10: Release fraction to the environment of the source terms.

Radionuclide	Source Term 3		Source Term 4		Source Term 6	
	2hours	24 hours	2hours	24 hours	2hours	24 hours
NG	0%	0.07968%	0.01394%	95.40%	1.761%	100%
Te	0%	0.02077%	0.00554%	24.91%	0.888%	63%
I	0%	0.01583%	0.00242%	18.99%	0.459%	48%
Cs	0%	0.01291%	0.00130%	15.49%	0.280%	39%
Sb	0%	0.00147%	0%	1.77%	0.001%	4%
Ba	0%	0.00025%	0%	0.30%	0.00002%	1%
Sr	0%	0.00015%	0%	0.18%	0.00001%	0.4%
Ru	0%	0.00008%	0%	0.10%	0%	0.2%
La	0%	0.00004%	0%	0.05%	0%	0.1%
Ce	0%	0.00003%	0%	0.03%	0%	0.1%

The results are fractions of the total amount of fission products contained in the nuclear core before the accident. For instance, the values of Table 11 must be multiplied by the core inventory according to the radionuclides group, to obtain the activity released by each radionuclide. For instance, the source term 1 and 3 have the same release fraction but are multiplied by different radionuclides inventory, the source term 1 by the radionuclides present on coolant, and the source term 3 by the radionuclides in the core.

Based on the radionuclides inventories of core and coolant proposed by (NUREG-1940, 2012) for generic nuclear fuel, Table 12 present the total activity released for each source term.

Table 11: Total activity released by the source terms.

Source Term	Activity [Bequerel]	
	2hours	24 hours
1	1.2×10^{8}	1.5×10^{9}
2	1.5×10^{11}	1.5×10^{12}
3	7.7×10^{10}	5.3×10^{14}
4	9.2×10^{13}	6.3×10^{17}
5	1.5×10^{12}	1.5×10^{12}
6	1.3×10^{16}	9.1×10^{17}

4.3 DOSE TO PUBLIC INDIVIDUAL

4.3.4 RASCAL source term to dose calculation

Based on the source term defined previously, the RASCAL software performs the radionuclides dispersion on the environment and it calculates the dose for individuals of public. A generic meteorological data has been adopted as follows:

- Wind speed and direction of 4mph
- Atmospheric stability Pasquill classes D
- No precipitation
- Air temperature of 70 F
- Relativity humidity of 50%

The calculations on RASCAL software are performed on a time frame of 2 and another time frame of 24 hours, both accounted from the start of fission products release from core. These times are useful to compare results with the siting criteria of the Brazilian standard (CNEN, 09/69), in order to confirm the exclusion area and low population area in front of the different blackout accidental pathways. Moreover, 24 hours is usually adopted as a reference for contingency plan of population evacuation, measuring the dose individuals from public will receive according to their relative distance from the release position.

Table 13 shows the results of Total Effective Dose Equivalent in Sieverts, according to the distance from the release, for the different source terms identified by the event tree of Figure 4. Figure 19 shows the footprint of the doses in Table 13. The source term 1 and 2 have negligible doses to be considered as accident consequences.

Table 12: Total Effective Dose Equivalent per distance for the several source terms.

Total Effective Dose Equivalent [Sv]								
Distance [Km]	**0.16**	**0.32**	**0.48**	**0.8**	**1.13**	**1.61**	**2.41**	**3.22**
Source Term 1 (2 hours)	0	0	0	0	0	0	0	0
Source Term 1 (24 hours)	0	0	0	0	0	0	0	0
Source Term 2 (2 hours)	$2x10^{-5}$	0	0	0	0	0	0	0
Source Term 2 (24 hours)	$2.3x10^{-4}$	$7.4x10^{-5}$	$4x10^{-5}$	$1.8x10^{-5}$	$1.1x10^{-5}$	0	0	0
Source Term 3 (2 hours)	0	0	0	0	0	0	0	0
Source Term 3 (24 hours)	$2.4x10^{-2}$	$7.7x10^{-3}$	$4.1x10^{-3}$	$1.9x10^{-3}$	$1.1x10^{-3}$	$6.3x10^{-4}$	$3.7x10^{-4}$	$2.8x10^{-4}$
Source Term 4 (2 hours)	$4x10^{-2}$	$1.3x10^{-2}$	$6.8x10^{-3}$	$3.1x10^{-3}$	$1.9x10^{-3}$	$1x10^{-3}$	$6.1x10^{-4}$	$4.7x10^{-4}$
Source Term 4 (24 hours)	$2.9x10^{1}$	9.3	5.0	2.3	1.4	0.77	0.45	0.34
Source Term 5 (2 hours)	$3.9x10^{-4}$	$1.2x10^{-4}$	$6.6x10^{-5}$	$3x10^{-5}$	$1.8x10^{-5}$	0	0	0
Source Term 5 (24 hours)	4.7x10-3	$1.5x10^{-3}$	$7.9x10^{-4}$	$3.6x10^{-4}$	$2.1x10^{-4}$	$1.2x10^{-4}$	$7x10^{-5}$	$5.4x10^{-5}$
Source Term 6 (2 hours)	$7.7x10^{-1}$	$2.5x10^{-1}$	$1.3x10^{-1}$	$6x10^{-2}$	$3.6x10^{-2}$	$2x10^{-2}$	$1.2x10^{-2}$	$9x10^{-3}$
Source Term 6 (24 hours)	$7.3x10^{1}$	$2.3x10^{1}$	$1.2x10^{1}$	5.60	3.40	1.90	1.10	0.84

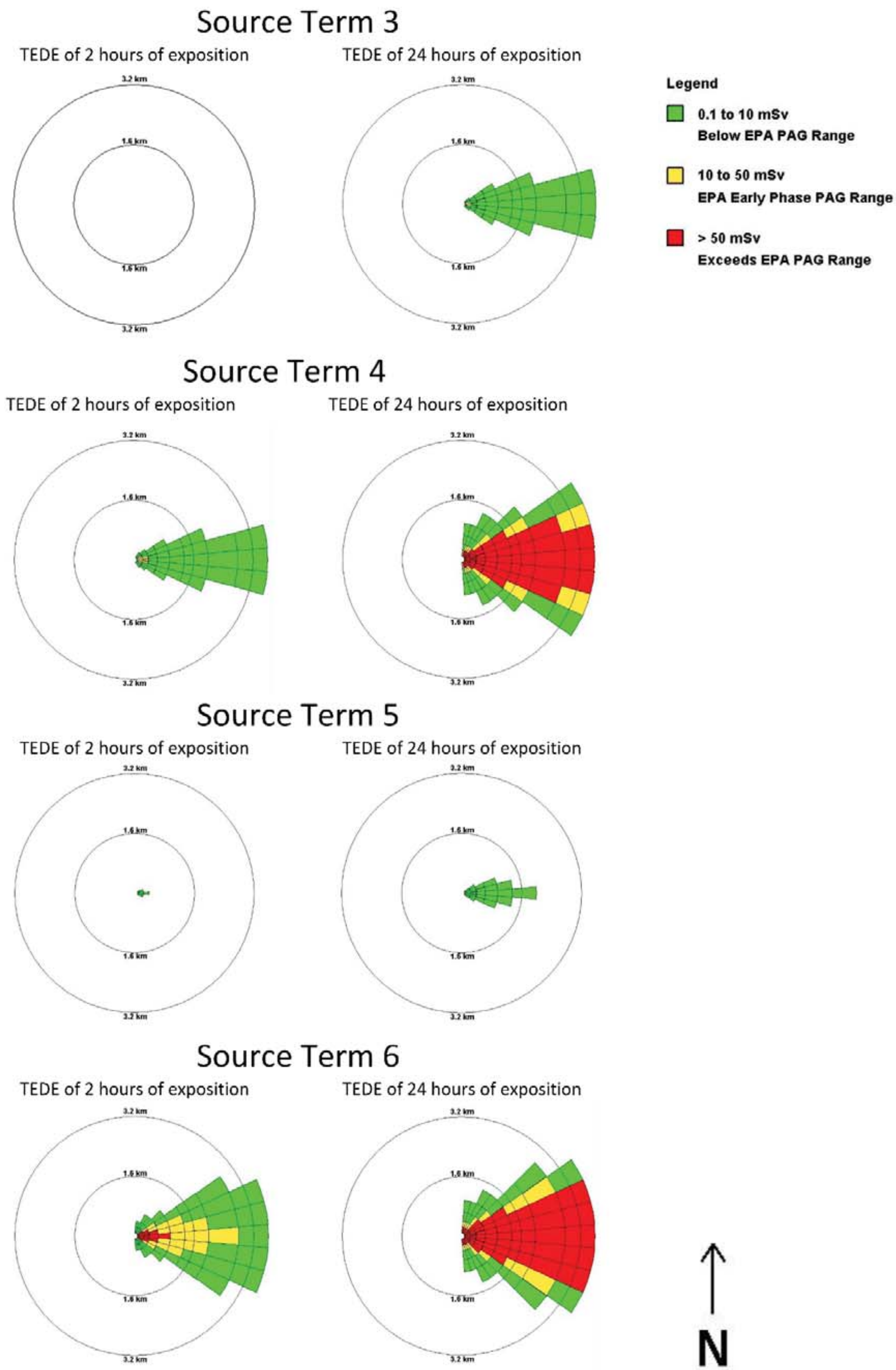

Figure 19: Total Effective Dose Equivalent graph for the several source terms.

The TEDE of source terms 3, 4 and 6, the ones involving core damage, are calculated on RASCAL for the amount of 24 hours of individual exposition. The objective is to compare the result of the developed code, with results that can be rapidly obtained in case of adopting the release fractions specified on (NUREG-1945, 1995). The results already results can be seen in Table 14.

Table 13: TEDE obtained through the source terms defined by RASCAL method.

Total Effective Dose Equivalent [Sv] in 24 hours of exposition								
Distance [Km]	**0.16**	**0.32**	**0.48**	**0.8**	**1.13**	**1.61**	**2.41**	**3.22**
Source Term 3	4.1×10^{-2}	1.3×10^{-2}	6.9×10^{-3}	3.1×10^{-3}	1.9×10^{-3}	1×10^{-3}	6.1×10^{-4}	4.7×10^{-4}
Source Term 4	3.8×10^{1}	12	6.5	2.9	1.7	0.97	0.57	0.44
Source Term 6	1.4×10^{2}	4.5×10^{1}	2.4×10^{1}	11	6.5	3.6	2.1	1.6

Comparing the results of Table 14 with the results of Table 13, the RASCAL method based on (NUREG-1945, 1995), shows more conservative results than the relative volatile calculation implemented on the developed algorithm of Annex A. Among the reasons for the less conservative results of the developed code, it can be pointed out:

- Consideration of the transient durations, according to different decay heat levels; and
- Release fractions are calculated at each temperature step, not defined by a phase of core degradation, as specified on (NUREG-1945, 1995).

4.4 RISK ASSESSMENT

The results of probability and dose to an individual for each accidental scenario with consequences are now joined in Table 15 in order to obtain a risk level of the blackout event on the Generic NPP.

Table 14: Frequency x consequence of the several blackout accidental scenarios on a generic marine NPP.

Event	Frequency [reactor.year]	Consequence [TEDE in Sv at 3.2 Km and 24 hours]	TEDE Risk [Sv.year] at 3.2Km and 24 hours
2	1.7×10^{-5}	0	0
3	1.8×10^{-7}	0	0
4	4.4×10^{-6}	2.8×10^{-4}	1.23×10^{-9}
5	$\mathbf{4.5 \times 10^{-8}}$	**0.34**	$\mathbf{1.53 \times 10^{-8}}$

Event	Frequency [reactor.year]	Consequence [TEDE in Sv at 3.2 Km and 24 hours]	TEDE Risk [Sv.year] at 3.2Km and 24 hours
7	$8.3x10^{-7}$	0	0
8	$1.5x10^{-7}$	$5.4x10^{-5}$	$8.10x10^{-12}$
9	**$1.1x10^{-6}$**	**0.34**	**$3.74x10^{-7}$**
10	**$1.9x10^{-7}$**	**0.84**	**$1.6x10^{-7}$**
12	$2.2x10^{-10}$	0.84	$1.85x10^{-10}$
13	$6.3x10^{-11}$	0.84	$5.29x10^{-11}$

Table 15 is plotted in Figure 20, in order to support also a graphical analysis. A tendency line is extrapolated with the risk results.

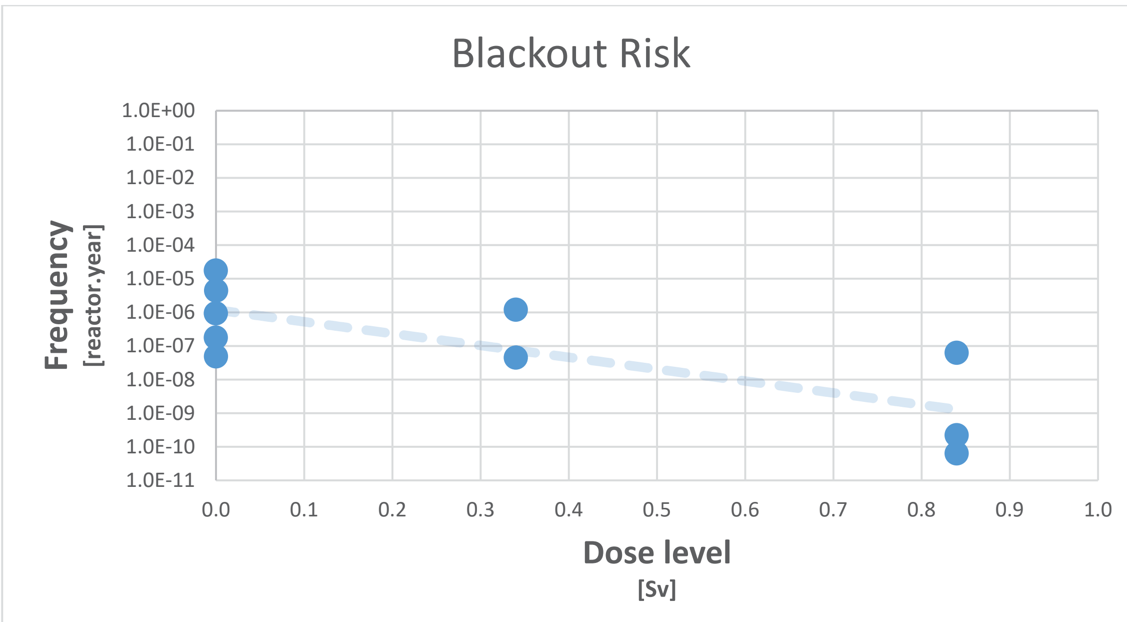

Figure 20: Risk curve of the blackout on a generic marine NPP.

There are events with considerable occurrence frequency, however with doses negligible or below the EPA recommended level of 10 mSv. These events do not involve core damage, or in case of core damage, the containment does not experience containment failure through bypass or fracture. This means that the risk assessment must focus on the event involving core damage and containment failure. Another relevant point is that the ATWS event has a really low frequency of occurrence, leading to low risk, even though this event leads to core damage and containment fracture.

From Table 15, it can be highlighted that all events are under the maximum probability and whole body dose specified on siting criteria of the Brazilian standard (CNEN, 09/69).

The results of blackout risk are compared to stationary NPP in operation. The (NUREG-1150,1990) performed a risk assessment of some U.S. NPP, from where the results of short and long term station blackout probability are extracted. The PWR plants of this study are Sequoyah Unit 1, a PWR 3455 MWt of 4 loops, and Surry Unit 1 a PWR 2441 MWt with 3 loops. The accident consequence estimated on (NUREG-1150,1990) are early fatalities and latent cancer fatalities because the plant has a stationary position and demographic studies were performed. The results of the radiological dose are not presented. Therefore, to enable a coherent risk comparison, the dose levels of these plants are also calculated by the Relative Volatile algorithm.

The comparison of blackout risk between stationary and marine NPP is performed based on the following considerations:

- The short term blackout will generate consequences at high heat decay, i.e. at blackout right after the reactor SCRAM, otherwise, the core will not be uncovered. Therefore, such events include DG failing to start. The probability considered for comparison with stationary NPP short SBO is the transient frequency cumulated with the probability of DG failing to start. Batteries' autonomy is not considered, because the SBO definition of stationary NPP considers only AC bus bars.
- The source term adopted on short SBO is source term 6 since short SBO is the only one to cause containment fracture.
- The long term blackout of stationary NPP was compared to the DG probability of failure during operation, a scenario of low heat decay in which a long period is necessary to reach the core uncover. The probability considered for comparison with stationary NPP long SBO is the transient frequency cumulated with the probability of DG failing during operation.
- The source term adopted on long SBO is source term 4 since the containment pressure on such a scenario is not enough to cause containment fracture, only bypass failure.

Table 16 shows the blackout risk of stationary and marine NPP, based on the comparison criteria described above.

Table 15: Blackout risk assessment of hypothetical marine NPP with some stationary NPP.

Plant	Generic Marine NPP	Sequoyah Unit 1 (NUREG-1150,1990)	Surry Unit 1 (NUREG-1150,1990)
Power	100MWt	3455MWt	2441MWt
Emergency Power	10-hour batteries 2 emergency DG	2-hour station batteries 2 emergency DG	2-hour station batteries 1 emergency DG 1 swing DG
Frequency of Core Damage by Short SBO	6.3×10^{-6}	3.8×10^{-6}	1.7×10^{-6}
Source term 6 TEDE at 3.2 Km (24 hours) [Sv]	0.84	29	22
Short SBO TEDE Risk [Sv.year]	5.3×10^{-6}	1.1×10^{-4}	3.74×10^{-5}
Frequency of Core Damage by Long SBO	2.24×10^{-5}	1.4×10^{-6}	8.2×10^{-6}
Source term 4 TEDE at 3.2 Km (24 hours) [Sv]	0.34	12	8.9
Long SBO TEDE Risk [Sv.year]	7.6×10^{-6}	1.68×10^{-5}	7.3×10^{-5}

According to the results of blackout risk shown in Table 16, the marine NPP presents a lower risk when compared to some stationary NPP. The blackout has a higher frequency occurrence on marine NPP, however the risk is balanced by lower TEDE for individuals of the public, mainly due to minor fission products inventory. It must be highlighted that the comparison was not direct, due to the difference in blackout risk analysis. The adopted figures for comparison seem the most indicated, however uncertainties can be raised from distinct risk analysis methods.

4.4 INSIGHTS OF RISK MANAGEMENT

Even the marine NPP blackout has a lower risk than some stationary plants, based on the ALARP policy, exhaustiveness to reduce risk must be demonstrated. Analyzing Table 15, the accidental pathways 5, 9 and 10 must be studied to reduce the risk up to feasibility boundaries.

The proposals presented herein to risk management are not part of the methodology for blackout risk assessment on marine NPP, since such solutions do not apply to any plant. Table 15 presents the results of the case study of the hypothetical marine NPP. The accidental pathways subject to risk in ALARP studies can vary from plant to plant. However, insights of risk management presented herein can be extent to specific plants.

Either the increase in emergency power sources reliability or higher redundancy levels are options to decrease the blackout risk. However, the reliability of combustion engine technology is beyond the state of the art, therefore not trivial to increase it. Moreover, small increases in reliability would be associated with considerable costs. Increasing the level of redundancy also leads to a cost increase. Therefore, this work proposes to reduce the blackout risk by improving maintainability and maintenance engineering aspects. Maintainability and maintenance requirements are verifiable during plant commissioning by demonstrations and tests. Equipment reliability must be verified by analysis of reliability studies because reliability proof tests could cause equipment degradation. In addition, the maintainability and maintenance requirements as blackout risk management are easier implemented on design, but not restricted to it, being possible to apply it on plants in operation. Increasing the reliability of equipment is not easily implemented in operating plants without an equipment modernization program, and also costly.

According to the definition of (SOUZA; GABE, 2017), the equipment repair rate is the figure that most increase reliability in redundant partially repairable systems. Therefore, the first risk management proposal is to increase the repair rate of the DG. A target objective of 85% of the DG repair rate is assumed. Such a solution imposes high equipment maintainability, demanding equipment suppliers' strong engagement on such objectives, and the provisioning of onboard maintenance resources. The crew must be well trained in equipment maintenance as well. The MTTR objective of repairable failure is also decreased to 8 hours, in order to jointly with the repair rate increase, and obtain a considerable functional reliability increase.

Despite the proposed solutions impact strongly the system reliability in operation, the gain of reliability on demand does not increase in the same proportion, i.e. the probability of DG failure to start does not decrease considerably by the proposed reparability solutions. Meaning that this solution reduces the risk of accidental pathway 5, but does not reduce the risk of pathways 9 and 10 in the same proportion.

The equipment average probability of failure on demand is approximated by the formula (19), valid when $\boldsymbol{\lambda} \ll \boldsymbol{\theta}$,

$$\boldsymbol{F_{demand}} = \frac{\boldsymbol{\lambda\theta}}{\boldsymbol{2}} \tag{19}$$

Where

F_{demand} is the average probability of failure on demand;

λ is the equipment or system stand-by failure rate (or calendar failure rate); and

θ is the proof test periodicity.

A solution to decrease the blackout risk of accidental pathways 9 and 10 is related to the proof test periodicity. The probability of failure on demand of a DG and their auxiliaries can be reduced by half, by double the frequency of the proof test. In addition, the DG and auxiliaries are fully redundant, so the electric power sources' average probability of failure by demand can be reduced by 4 times. To reduce the periodicity of the DG proof test at half is the risk management proposal for accidental pathways 9 and 10.

The last proposal to decrease the probability of the most severe accident, the accidental pathway 10, is to specify some requirements of quality to containment structure. The stress intensity factor of the containment structure can be increased, however such a solution may demand the adoption of different materials and welding technologies. Decreasing the flaw size of metal plates can be a solution just involving the manufacturing quality process. In Table 8, a detectable crack size of 500um has been adopted, if a maximum acceptable crack size of 400um is specified in the manufacturing process, the probability of fracture can be reduced considerably.

Table 17 shows the blackout risk after applying the following management proposals:

- Increase the DG repair rate to 85%;
- Specify an MTTR objective for DG of 8 hours;
- Decrease the DG proof test periodicity; and

- Impose manufacturing quality requirements to containment plates.

Table 16: Blackout Risk after management proposals for the hypothetical marine NPP.

Event	Frequency [reactor.year]	Consequence [TEDE in Sv at 3.2 Km and 24 hours]	TEDE Risk [Sv.year] at 3.2Km and 24 hours
2	$8.3x10^{-6}$	0	0
3	$8.4x10^{-8}$	0	0
4	$6.5x10^{-7}$	$2.8x10^{-4}$	$1.8x10^{-10}$
5	**$6.5x10^{-9}$**	**0.34**	**$2.2x10^{-9}$**
7	$3.1x10^{-8}$	0	0
8	$1.6x10^{-9}$	$5.4x10^{-5}$	$8.8x10^{-14}$
9	**$1.4x10^{-7}$**	**0.34**	**$4.8x10^{-8}$**
10	**$7.3x10^{-9}$**	**0.84**	**$6.2x10^{-9}$**
12	$9x10^{-11}$	0.84	$7.6x10^{-11}$
13	$6.3x10^{-11}$	0.84	$5.3x10^{-11}$

Table 17 is plotted in Figure 21, in order to support also a graphical analysis.

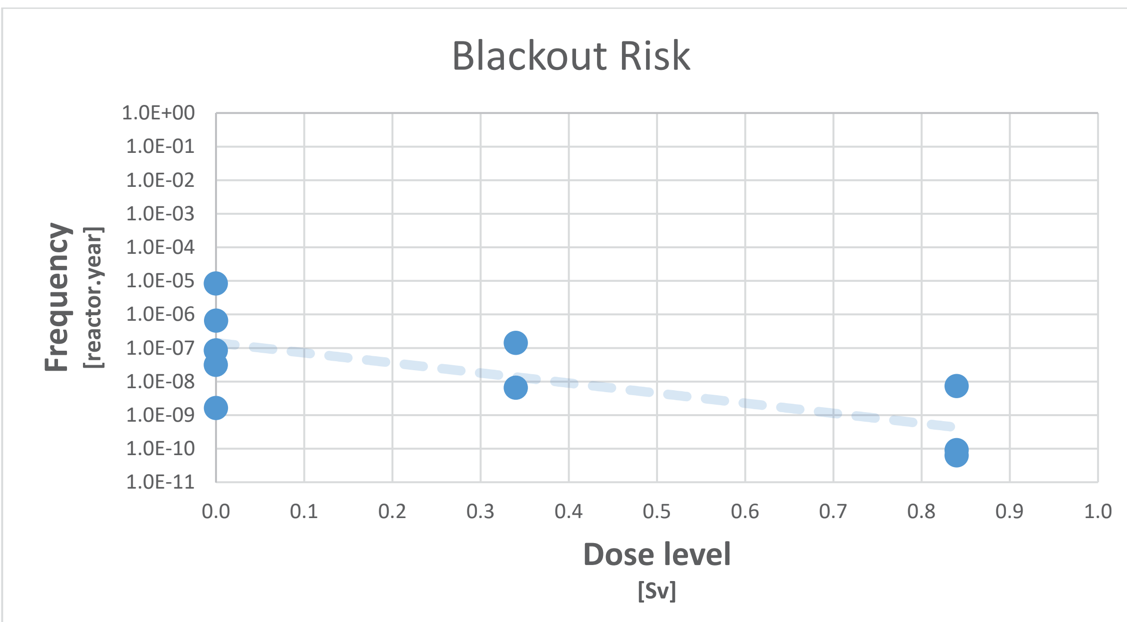

Figure 21: Risk curve of the blackout after management proposals.

These results reveal a reduction of the blackout risk of around 7 times, based only on maintainability and maintenance aspects. The most severe accidental pathway reveals a risk reduction of 25 times after management. In addition, no risk point is far from the tendency line.

Comparing the new blackout risk to the stationary NPP of Table 16, based on the same considerations described in the previous comparison, the short SBO TEDE risk is 1.3 x 10^{-6} [Sv.year], and the long SBO TEDE risk of 3.1 x 10^{-6} [Sv.year]. Such results demonstrated a considerable difference between the risk of the hypothetical marine NPP and the stationary NPP of Table 16. The risk of long SBO of the marine NPP is more than 5 times below the Sequoyah Unit 1. In the other cases, the blackout risk of hypothetical marine NPP is more than 20 times below stationary NPP.

4.4 BLACKOUT AS REFERENCE ACCIDENT FOR EMERGENCY PLANNING ZONES DEFINITION

The Brazilian standard (CNEN, 09/69) establishes dose limits for the exclusion area as the total dose of whole body radiation must not exceed 250 mSv, and the total dose of iodine-131 inhalation radiation in the thyroid cannot exceed 3 Sv. Both for an individual within the limits of the property area under the control of a port facility. These values are calculated for an irradiation time of two hours, counted from the beginning of the accident. For the Low Population Zone, the same dose levels are expected, but the time for dose calculation extends over the entire period of passage of the radioactive cloud resulting from the release of fission products due to the accident. These dose levels must be calculated for the worst credible accident.

Based on the accidental pathways of Figure 17 and their probabilities, events 9 and 10 can be considered candidates for the most credible accident. These accidental scenarios release the source term 4 and 6 respectively. Table 18 shows the TEDE and Thyroid EDE of source terms 4 and 6 for distance resulting in doses matching the maximum values established on (CNEN, 09/69).

Table 17: TEDE and Thyroid EDE of source terms 4 and 6 for siting criteria assessment.

	Source Term 4		Source Term 6	
	Exclusion Area (2 hours)	Low Population Zone (96 hours)	Exclusion Area (2 hours)	Low Population Zone (96 hours)
Distance	320 m	4.8 Km	320 m	16 Km
TEDE	13 mSv	250 mSv	250 mSv	230 mSv
Thyroid EDE	110 mSv	2.3 Sv	2.2 Sv	2.2 Sv

The accidental pathway 10 has an occurrence frequency of 7.3 x 10^{-9} reactor.year, which is an extremely low occurrence frequency to be considered a credible accident. On other hand, accidental pathway 9 has also a low occurrence frequency of 1.4 x 10^{-7} reactor.year, which is a candidate to be considered a credible accident. Therefore, the credible accident adopted as a reference includes the core damage by blackout and containment bypass. Based on Table 18, a possible exclusion zone is 320 m and a low population zone can be 4.8 Km. These siting criteria are useful for a port installation analysis about its capacity to accept or not a nuclear ship.

4.4 TREATMENT OF UNCERTAINTIES

The higher blackout risk comes from accidental pathways 5, 9 and 10. A sensitivity analysis of risk results according to changes in input data has been performed, demonstrating that the input data below are the most impacting ones:

- Reactor power;
- Leakage rate;
- Decay heat curve;
- DG repair rate factor.

The reactor thermal power is the most important figure to define the amount of fission products inventory that can be released through a source term and subsequently define the dose level to individuals of society. The reactor thermal power also influences directly the amount of decay heat, defining the thermal transient duration. Despite the sensitivity of such a figure, the uncertainty associated with it is not considered important, because such a figure is one of the most important ones in the NPP design. The plant will be designed to reach a power objective. It is not considered that uncertainty treatment must be carried out for reactor thermal power.

The decay heat curve creates epistemic uncertainty that can be decreased by the adoption of precise models such as (ANSI/ANS-5.1, 2005). In the case of a hypothetical marine NPP, without specific nuclear fuel data, the adoption of a generic decay heat curve such as (KNIEF, 1992) increases the epistemic uncertainty of the results. The uncertainty of risk according to the decay heat curve is not evident, because it is not a numerical uncertain value, but a curve describing long term behavior. The decay heat curve uncertainty must be dealt with through the adoption of specific data of the nuclear fuel, being not possible to implement such a solution on a generic marine NPP.

The leakage rate of source term according to blackout containment pressurization generates epistemic uncertainty. There is no certitude of the leakage rate that an overpressure on containment could generate. Moreover, the risk shows high sensitivity to such input data. An uncertainty margin of 10% around the adopted leakage rate is applied, and the tolerance around the risk level is obtained. The results of TEDE variation according the 10% of uncertainty in leakage rate are presented on Table 19. The risk sensitivity of leakage rate uncertainties on accidental pathways 5 is negligible. For the accidental pathway 10, the leakage rate uncertainties reflect only of short time effects, in a long term, all the containment atmosphere would be released to the environment. This short term effect could be important for siting criteria, therefore the same uncertainty margin of 10% is applied on accidental pathway 10 for the 2 hours consequence analysis.

Table 18: TEDE sensitivity according to minimum to maximum uncertain leakage rate.

Consequence	**TEDE [Sv]**		
	-5% of nominal Leakage Rate	**Nominal Leakage Rate**	**+5% of nominal Leakage Rate**
Source Term 4 (2 hours) TEDE	0.44 mSv	0.47 mSv	0.49 mSv
Source Term 4 (24 hours) TEDE	330 mSv	340 mSv	360 mSv
Source Term 6 (2 hours) TEDE	8.6 mSv	9 mSv	9.4 mSv

The range of risk uncertainty demonstrates a tolerance of around 9%, for the consequences of pathways 9 and 10. The risk of these accident sequences has an uncertainty attributed to results of around 9% considering the consequence analysis.

The DG repair rate factor is a random variable calculated from two other random variables. According to the definition of (SOUZA; GABE, 2017), the repair rate factor is

$$\boldsymbol{F_r} = \frac{\lambda_{repairable}}{\lambda_{global}} \tag{20}$$

Where

Fr is the repair rate factor;

$\lambda_{repairable}$ is the failure rate of repairable failures; and

λ_{global} is the global failure rate.

The uncertainty analysis has been performed considering $\lambda_{repairable}$ and λ_{global} as the random variables, to calculate the confidence interval of the risk results. The uncertainty analysis is performed on the Petri model of Figure 13, attributing the maximum and minimum values of the failure rates, for a confidence interval of 90%, and calculating the probability of occurrence bounds, to measure the level of uncertainty on the risk results. Therefore, only the equipment reliability is considered uncertain, and no uncertainty has been attributed to the repair rate factor and MTTR. Table 20 shows the low and high bounds of probability for events of the event tree of Figure 4, according the interval of DG reliability

Table 19: Low and high bound of relevant event probabilities for the blackout.

Event	Probabilities		
	Lower bound	Average value	Upper bound
DG Fail to Start	$5.1\text{x}10^{-7}$	$1.1\text{x}10^{-6}$	$2\text{x}10^{-6}$
DG Fail on operation	$3.3\text{x}10^{-6}$	$6.5\text{x}10^{-6}$	$1.3\text{x}10^{-5}$
Accidental pathway 5	$3.4\text{x}10^{-9}$	$6.5\text{x}10^{-9}$	$1.3\text{x}10^{-8}$
Accidental pathway 9	$6.4\text{x}10^{-8}$	$1.4\text{x}10^{-7}$	$2.5\text{x}10^{-7}$
Accidental pathway 10	$3.3\text{x}10^{-9}$	$7.3\text{x}10^{-9}$	$1.3\text{x}10^{-8}$

The lower and upper bound of Table 19 and Table 20 were joined to calculate the blackout risk bounds of relevant accidental pathways. Table 21 shows the obtained risk bounds for the proposed uncertainty analysis.

Table 20: Low and high bound of blackout risk for main accidental pathways.

Accidental Pathways	Blackout Risk		
	Lower bound	Average value	Upper bound
Accidental Pathways 5	$1.1\text{x}10^{-9}$	$2.2\text{x}10^{-9}$	$4.6\text{x}10^{-9}$
Accidental Pathways 9	$2.1\text{x}10^{-8}$	$4.8\text{x}10^{-8}$	$8.9\text{x}10^{-8}$
Accidental Pathways 10	$2.8\text{x}10^{-9}$	$6.2\text{x}10^{-9}$	$1.1\text{x}10^{-8}$

The tolerance on the relevant accidental pathways risk reveals a large range around the average value. A conservative approach to overcome epistemic uncertainties and random variables uncertainties is to adopt the high bound of blackout risk. Even in this conservative approach, the risk figures continue below the stationary NPP shown in Table 16. Therefore, the methodology presented in Figure 3 resulted in the blackout risk level of a hypothetical marine NPP lower than stationary NPP, considering the worst case of uncertainties involved in the analysis.

5 CONCLUSION

The contribution that nuclear power can provide to maritime transport has been discussed in this work. The technical feasibility of such an application has been demonstrated through the cargo ships designed, constructed and operated by different countries. The economical solutions for this application are under study. This work contributed with a methodology to investigate the feasibility of this application from a nuclear safety standpoint. The methodology is based on the risk assessment of a severe accident with specific characteristics due to maritime operational profile.

The proposed methodology enables the assessment of blackout accident risk during early design phases. The input data for the methodology is based on the main figures available in the conceptual design phase. The methodology was attested by a case study based on a hypothetical marine NPP. Despite the case study not presenting results for a specific plant or for all marine NPP, it illustrates the use of the methodology, providing guidelines for applying the methodology for existing and future reactor designs.

The contributions of this work do not limit to the methodology, several complementary contributions to nuclear safety are also raised. The compendium of data, studies, models and considerations presented herein constitutes a relevant contribution. The blackout accident on mobile NPP has been defined and the accident pathways were modeled, based on an electric power system applicable to nuclear-powered ships. From the probabilistic analysis point of view, the state-of-art of dynamic reliability modeling has been explored in the calculation of emergency power sources reliability. Generalized Stochastic Petri Nets have been used, involving several complex aspects not possible to be modeled on combinatorial models as Fault Tree diagrams. Structural reliability analysis has also been performed on the containment, to define a probability for the fracture failure mode.

From the consequence analysis point of view, another relevant contribution of this work was the developed code to apply a simplified model to calculate the time dependent source term of the accident. The simplified model recommended by IAEA was improved to build a flexible algorithm for several accident progressions. The algorithm calculates the time depended source term, the containment overpressure and also exports the source term to RASCAL software. The doses to individuals of society are calculated by RASCAL software.

The results of the case study do not indicate the risk of a specific nuclear power plant. Nevertheless, their results can be considered as indications of the risk of marine plants. The results of the methodology applied to a hypothetical marine NPP indicate that the blackout accident occurrence frequency is higher than stationary NPP, however the risk of the accident is balanced by lower radiological consequences to individuals of society. It has proposed risk reductions through maintenance and maintainability solutions. These risk reduction proposals present low cost and effectiveness and can be readily demonstrated to regulatory bodies. The limitation of the results and the uncertainties associated with the methodology have been quantified, in order to increase the confidence of the results. The high risk bound of marine NPP still lower than the blackout risk of the stationary NPP compared herein.

Regulatory bodies can apply the methodology in order to support the decision to accept nuclear ships in country ports, including the warships like submarines and aircraft carriers. Through the proposed methodology, siting criteria for exclusion area and low population zone according to dose levels on worst credible accident can be obtained and compared to base or port areas. Moreover, the methodology proposed can serve as a subsidy for a *"Risk-Informed"* approach since the conceptual design, in order to obtain design licensing, instead of the prescriptive requirements of safety-based normative. For a new design, several requirements can be defined early on design, in order to reach an objective risk level.

Concluding, the results encourage complementing the systems design approach of "redundancy, segregation and independency" with "maintainability, maintenance and operational procedures" approach, reducing capital costs and keeping safety. It is believed that the contribution to safety analysis promotes the deployment of nuclear-powered commercial shipping. However, it is recognized that the demonstration of method success demands the application of a complete PSA of a specific design, to confirm the insights and contributions highlighted.

There are several areas and disciplines applied herein that can be candidates for future works. Since the methodology is applied for early design phases, notably the methods applied on source term quantification and containment failure analysis, application of precise models would certainly contribute to improve this work. The more recommended improvements for this work are listed hereinafter:

- Detailing the accident pathways if blackout according the increase in design detail level;
- Increase the quantity of equipment in the reliability model;
- Use of a computer code to model the severe accident progression, instead of the simplified thermo-hydraulic algorithm applied;
- Use of a computer code to define the source term release to environment
- To analyze the offsite consequence, instead of RASCAL software, more recent codes, like MAACS from Sandia National Laboratories, can also be an important contribution to improve this work.

Naturally, the positive feedback of nuclear energy to naval warships would lead the application of such energy to maritime trade, considering the perfect match between such a source with the newer demands for affordable green energy. Any works exploring the safety issues of such application will contribute to such future become closer to reality, which the international confidence that the protection of people and environment is kept in the higher degree of importance.

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Appendix A - Simplified Source Term Algorithm

```
Sub LaunchTransCalc()

    'Case study input data
    'NPP data
    'Reactor Normal Operating Power
    Pth = Range("Pth").Value
    'Containment free volume
    Vcont = Range("Vcont").Value
    'Average Burn-Up
    BU = Range("BU").Value
    'Coolant Mass
    WRCS = Range("WRCS").Value
    'Water Mass Around Core
    WU = Range("WU").Value
    'Residual Water Below Core
    WR = Range("WR").Value
    'Primary circuit pressure
    p_1 = Range("p_1").Value
    'High Safety Injection Injentory
    WSI = Range("WSI").Value
    'Heat transfer coefficient of Containment to environment [kW/°C s]
    hc = Range("hc").Value / 3600
    'Total Core Fuel/Clad MCP
    MCP = Range("MCP").Value
    'RunAway ZR Oxidation HeatUp Rate
    ZRC = Range("ZRC").Value
    'Accident data
    'ATWS
    ATWS = Range("ATWS").Value
    'Accident parameter
    AccParam = Range("AccParam").Value
    'Scenarion duration
    TimeMAX = Range("TimeMAX").Value * 3600
    'Internal Energy Maximum Error
    Uerror = 0.01
    'Time initialization
    T = 0
    Tuncover = 0

    'Isolated LOCA + ECCS failure (lost coolant mass)
    If Range("AccType").Value = _
        "Isolated LOCA + no injection (coolant mass instantaneous loss)" Then
        'Primary circuit pressure initial condition
        Pprim = p_1
        'Primary circuit title initial condition
        Xprim = 0
        'Remaining coolant Internal Energy before LOCA [kJ]
```

```
'U = h * M
U_1 = hL_p(p_1) * (WRCS - AccParam)
'Remaining coolant Internal Energy after LOCA initial condition [kJ]
'h_px -> Entalpy as a function of pressure and vapour fraction
'vL_p -> Saturated liquid specific volume as a function of pressure
'U = h * M - p*V
U_2 = h_px(Pprim, Xprim) * (WRCS - AccParam) - Pprim * vL_p(p_1) * WRCS
'Loop to calculate Pprim and Xprim for U_1 = U_2
While Abs(U_1 - U_2) > Uerror
   'Pprim value update
   Pprim = Pprim * (1 - (U_2 - U_1) / U_2)
   'Xprim value update based on specific volumes as function of pressure
   'vL_p -> Saturated liquid specific volume as a function of pressure
   'vV_p -> Saturated vapour specific volume as a function of pressure
   Xprim = (vL_p(p_1) * WRCS / (WRCS - AccParam) - vL_p(Pprim)) / _
   (vV_p(Pprim) - vL_p(Pprim))
   'Control of Xprim  convergence error
   If Xprim > 1 Then
      Xprim = 1
   End If
   'Remaining coolant Internal Energy after LOCA update [kJ]
   U_2 = h_px(Pprim, Xprim) * (WRCS - AccParam) - Pprim * vL_p(p_1) * WRCS
Wend
'Containment pressure initial condition
Pcont = 0.1
'Released coolant title initial condition
Xcont = 1
'Released coolant Internal Energy before LOCA [kJ]
'hL_p -> Saturated liquid enthalpy as a function of pressure
'U = h * M
U_1 = hL_p(p_1) * AccParam
'Released coolant Internal Energy after LOCA [kJ]
'h_px -> Entalpy as a function of pressure and vapour fraction
'U = h * M - p*V
U_2 = h_px(Pcont, Xcont) * AccParam - Pcont * Vcont
'Loop to calculate Pcont and Xcont for U_1 = U_2
While Abs(U_2 - U_1) > Uerror
   'Pcont value update
   Pcont = Pcont * (1 + (U_1 - U_2) / U_2)
   'Xcont value update based on specific volume as function of pressure
   'vL_p -> Saturated liquid specific volume as a function of pressure
   'vV_p -> Saturated vapour specific volume as a function of pressure
   Xcont = (Vcont / AccParam - vL_p(Pcont)) / (vV_p(Pcont) - vL_p(Pcont))
   'Control of Xcont  convergence error
   If Xcont > 1 Then
      Xcont = 1
   End If
```

```
        'Released coolant Internal Energy after LOCA update [kJ]
        U_2 = h_px(Pcont, Xcont) * AccParam - Pcont * Vcont
    Wend
    'Necessary Energy to vaporize the remaining coolant up to core uncover[in MW.s = MJ]
    R = ((WRCS - AccParam) * (1 - Xprim) - (WU + WR) + WSI) * _
    (hV_p(Pprim) - hL_p(Pprim)) / 1000
    If R <= 0 Then
        'Core uncover right after LOCA
        T = 1
        Tuncover = 0
    Else
        'Remaining coolant enough to avoid core uncover
        Eth = 0#
        'Thermal energy provided by core
        If ATWS Then
            While ((Eth < R) And (T < (TimeMAX)))
                T = T + 1
                'In case of ATWS, the core continue at operating power
                Eth = Eth + Pth
            Wend
        Else
            While ((Eth < R) And (T < (TimeMAX)))
                T = T + 1
                'Decay energy given by (KNIEF, 1992)
                Eth = Eth + Pth * 0.066 * (T ^ -0.2 - (T + 3600 * 8760#) ^ -0.2)
            Wend
        End If
    End If
    'Core Uncover time
    Tuncover = T
End If

'LOCA + ECCS failure (average flow rate)
If Range("AccType").Value = "LOCA + ECCS failure (average coolant mass loss)" Then
    'Core Uncover time
    Tuncover = (WRCS - (WU + WR) + WSI) / AccParam
    'Containment pressure initial condition
    Pcont = 1
    'Released coolant inventory initial condition
    Xcont = 1
    'Decay energy to coolant before complete coolant release [MW.s]
    Qdecay = 0
    If ATWS Then
        While T < (WRCS + WSI) / AccParam
            T = T + 1
            'In case of ATWS, the core continue at operating power
            Qdecay = Qdecay + Pth
```

```
        Wend
    Else
        While T < (WRCS + WSI) / AccParam
            T = T + 1
            'Decay energy given by (KNIEF, 1992)
            Qdecay = Qdecay + Pth * 0.066 * (T ^ -0.2 - (T + 3600 * 8760#) ^ -0.2)
        Wend
    End If
    'Released coolant Internal Energy before LOCA and Decay energy received [kJ]
    'U = h * M + Qdecay
    U_1 = hL_p(p_1) * (WRCS + WSI) + 1000 * Qdecay
    'Containment temperature [°C]
    'T_ph -> Temperture as a function of pressure and enthalpy
    'h_px -> Entalpy as a function of pressure and vapour fraction
    Tcont = T_ph(Pcont, h_px(Pcont, Xcont))
    'Thermal energy dissipated from containment to environment [kJ]
    'Containment area considered near to containment free volume
    'Environment temperature equal to 32°C
    'Average temperature diferrence between containment and
    'environment = (Tcont_final - Tenv) / 2
    'T_ph -> Temperture as a function of pressure and enthalpy
    Qdissip = hc * Vcont * (WRCS + WSI) / AccParam * (Tcont - 25)
    'Released coolant Internal Energy after LOCA initial condition [kJ]
    'h_px -> Entalpy as a function of pressure and vapour fraction
    'U = h * M - p*V + Qdissip
    U_2 = h_px(Pcont, Xcont) * (WRCS + WSI) - Pcont * Vcont + Qdissip
    'Loop to calculate Pcont and Xcont for U_1 = U_2
    While Abs(U_2 - U_1) > Uerror
        'Pcont value update
        Pcont = Pcont * (1 + (U_1 - U_2) / U_1)
        'Xcont value update based on specific volume as function of pressure
        'vL_p -> Saturated liquid specific volume as a function of pressure
        'vV_p -> Saturated vapour specific volume as a function of pressure
        Xcont = (Vcont / (WRCS + WSI) - vL_p(Pcont)) / (vV_p(Pcont) - vL_p(Pcont))
        'Control of Xcont  convergence error
        If Xcont > 1 Then
            Xcont = 1
        End If
        'Containment temperature [°C]
        Tcont = T_ph(Pcont, h_px(Pcont, Xcont))
        'Thermal energy dissipated updated [kJ]
        Qdissip = hc * Vcont * (WRCS + WSI) / AccParam * (Tcont - 25)
        'Released coolant Internal Energy after LOCA update [kJ]
        U_2 = h_px(Pcont, Xcont) * (WRCS + WSI) - Pcont * Vcont + Qdissip
    Wend
End If
```

```
    'Transient + Blackout (time after transient)
    If Range("AccType").Value = "SCRAM + Blackout (time after SCRAM)" Then
        'Time of blackout  (ECCS stop)
        T = AccParam * 3600
        'Necessary Energy to vaporize the coolant inventory [MW.s]
'        R = (WRCS - (WU + WR) + WSI) * (hV_p(p_1) - hL_p(p_1)) / 1000
        R = (WRCS + WSI) * (hV_p(p_1) - hL_p(p_1)) / 1000
        'Thermal energy provided by core
        Eth = 0#
        If ATWS Then
            While ((Eth < R) And (T < (TimeMAX)))
                T = T + 1
                'In case of ATWS, the core continue at operating power
                Eth = Eth + Pth
            Wend
        Else
            While ((Eth < R) And (T < (TimeMAX)))
                T = T + 1
                'Decay energy given by (KNIEF, 1992)
                Eth = Eth + Pth * 0.066 * (T ^ -0.2 - (T + 3600 * 8760#) ^ -0.2)
            Wend
        End If
        'Time to Vaporize the complete coolant inventory
        T = T - AccParam * 3600
        'Containment pressure initial condition
        Pcont = 1
        'Coolant released title initial condition
        Xcont = 0
        'Released coolant Internal Energy
        'U = h * M
'        U_1 = hV_p(p_1) * (WRCS - (WU + WR) + WSI)
        U_1 = hV_p(p_1) * (WRCS + WSI)
        'Containment temperature
        Tcont = T_ph(Pcont, h_px(Pcont, Xcont))
        'Thermal energy dissipated from containment to environment
        'Containment area considered near to containment free volume
        'Environment temperature equal to 32°C
        'Average temperature diferrence between containment and
        'environment = (Tcont_final - Tenv) / 2
        'T_ph -> Temperture as a function of pressure and enthalpy
        Qdissip = hc * Vcont * T * (Tcont - 25)
        'Released coolant Internal Energy Initial Condition
        'U = h * M - p*V + Qdissip
'        U_2 = h_px(Pcont, Xcont) * (WRCS - (WU + WR) + WSI) - Pcont * Vcont + Qdissip
        U_2 = h_px(Pcont, Xcont) * (WRCS + WSI) - Pcont * Vcont + Qdissip
        'Loop to calculate Pcont and Xcont
        While Abs(U_2 - U_1) > Uerror
```

```
            'Pcont value update
            Pcont = Pcont * (1 + (U_1 - U_2) / U_2)
            'Xcont value update based on specific volume as function of pressure
            'vL_p -> Saturated liquid specific volume as a function of pressure
            'vV_p -> Saturated vapour specific volume as a function of pressure
'           Xcont = (Vcont / (WRCS - (WU + WR) + WSI) - vL_p(Pcont)) / (vV_p(Pcont) - vL_p(Pcont))
            Xcont = (Vcont / (WRCS + WSI) - vL_p(Pcont)) / (vV_p(Pcont) - vL_p(Pcont))
            'Control of Xcont convergence error
            If Xcont > 1 Then
                Xcont = 1
            End If
            'Containment temperature
            Tcont = T_ph(Pcont, h_px(Pcont, Xcont))
            'Thermal energy dissipated updated
            Qdissip = hc * Vcont * T * (Tcont - 25)
            'Released coolant Internal Energy update
'            U_2 = h_px(Pcont, Xcont) * (WRCS - (WU + WR) + WSI) - Pcont * Vcont + Qdissip
            U_2 = h_px(Pcont, Xcont) * (WRCS + WSI) - Pcont * Vcont + Qdissip
        Wend
        'Time to Vaporize coolant up to uncover
        T = AccParam * 3600
        'Necessary Energy to vaporize the coolant up to core uncover [MW.s]
        R = (WRCS - (WU + WR) + WSI) * (hV_p(p_1) - hL_p(p_1)) / 1000
        'Thermal energy provided by core
        Eth = 0#
        If ATWS Then
            While ((Eth < R) And (T < (TimeMAX)))
                T = T + 1
                'In case of ATWS, the core continue at operating power
                Eth = Eth + Pth
            Wend
        Else
            While ((Eth < R) And (T < (TimeMAX)))
                T = T + 1
                'Decay energy given by (KNIEF, 1992)
                Eth = Eth + Pth * 0.066 * (T ^ -0.2 - (T + 3600 * 8760#) ^ -0.2)
            Wend
        End If
        'Core Uncover time
        Tuncover = T - AccParam * 3600
    End If

    'Transient calculation based on GENERALIZED THERMAL TRANSIENT of IAEA-TECDOC-1127
    'Necessary Energy to vaporize the coolant around core [MW.s]
    R = WU * (hV_p(p_1) - hL_p(p_1)) / 1000
   'Energy to adiabatic core heat up
   '1800 K is the Runaway ZR Oxidantion temperature'
```

```
 '600 K is the assumed core temperature at uncover time'
  Uadiab = (1800 - 600) * MCP
  'Thermal energy provided by core
  Eth = 0#
  If ATWS Then
     While ((Eth < R + Uadiab) And (T < (TimeMAX)))
        T = T + 1
        'In case of ATWS, the core continue at operating power
        Eth = Eth + Pth
     Wend
  Else
     While ((Eth < R + Uadiab) And (T < (TimeMAX)))
        T = T + 1
        'Decay energy given by (KNIEF, 1992)
        Eth = Eth + Pth * 0.066 * (T ^ -0.2 - (T + 3600 * 8760#) ^ -0.2)
     Wend
  End If
  'Core Uncover time
  Tadiab = T - Tuncover
  'Transient + Blackout (time after transient)
  If Range("AccType").Value = "SCRAM + Blackout (time after SCRAM)" Then
     Tadiab = Tadiab - AccParam * 3600
  End If
  '1200 K is the assumend clad failure temperature, hence start of fission
  'products release
  'Average decay heat = Eth / Tadiab
  Tstart = (1200 - 600) / (Eth / Tadiab / MCP)
  '2960 K is the assumend melt temperature
  Tzrc = (2960 - 1800) / ZRC
  '827 K was developed as equivalent temperature change that leads to melt
  Tmelt = 827 / ZRC

  'Time to Core Uncover
  Range("Tuncover").Value = Tuncover
  'Start of Fission Products  Release
  Range("Tstart").Value = Tuncover + Tstart
  'Time to Adiabatic HeatUp
  Range("Tadiab").Value = Tuncover + Tadiab
  'Time to Zr Oxidation
  Range("Tzrc").Value = Tuncover + Tadiab + Tzrc
  'Time to Melt
  Range("Tmelt").Value = Tuncover + Tadiab + Tzrc + Tmelt
  'Containment Pressure
  Range("Pcont").Value = Pcont

End Sub
```

```
Sub LaunchSourceTermCalc()

    'Case study input data
    'NPP data
    'Average Burn-Up
    BU = Range("BU").Value
    'Scenarion duration
    TimeMAX = Range("TimeMAX").Value * 3600
    'Transient output
    'Time to Core Uncover
    Tuncover = Range("Tuncover").Value
    'Start of Fission Products  Release
    Tstart = Range("Tstart").Value
    'Duration of Adiabatic HeatUp
    Tadiab = Range("Tadiab").Value - Tuncover
    'Duration of Zr Oxidation
    Tzrc = Range("Tzrc").Value - Tuncover - Tadiab
    'Duration to Melt start
    Tmelt = Range("Tmelt").Value - Tuncover - Tadiab - Tzrc
    'Containment data
    'Factor 1 of gravitational settling according Appendix B of NUREG-1150
    Lambda1 = Range("Rate1")
    'Factor 2 of gravitational settling according Appendix B of NUREG-1150
    Lambda2 = Range("Rate2")
    'Time to change Factors of gravitational settling according Appendix B of NUREG-1150
    LambdaTime = Range("RateTime")
    'Containment Leakage rate [percentage of containment volume per hours]
    'Hypothesis: the FP in aerosols (not settled) are homogeneously distributed
    'on containment athmosphere
    LeakageRate = Range("LeakageRate")

    'RelVol FP release calculation algorithm of IAEA-TECDOC-1127
    'Relative Volatile factors of FP
    Dim f(), RV(8) As Single
    'NG
    RV(0) = 1.1
    'Te
    RV(1) = 1.07
    'I
    RV(2) = 1.03
    'Ba
    RV(3) = 0.42
    'Sr
    RV(4) = 0.34
    'Ru
    RV(5) = 0.25
    'La
```

```
RV(6) = 0.14
'Ce
RV(7) = 0.085

'Q Cs arrhenius correlation parameter
Q_Cs = (2.065 * 10 ^ 5 - 3.629 * BU) / 1.99
'Do Cs arrhenius correlation parameter
Do_Cs = (2.6833 * 10 ^ 5) * Exp(-(6.052 * 10 ^ -4) * BU)
'Q Sb arrhenius correlation parameter
Q_Sb = (2.494 * 10 ^ 5 - 3.629 * BU) / 1.99
'Do Sb arrhenius correlation parameter
Do_Sb = (3.4608 * 10 ^ 6) * Exp(-(6.052 * 10 ^ -4) * BU)
'Temperature ar start of FP release
Ts = 1200
'Temperature step initial condition
DELT = 10
'Temperature at core initial condition
TBAR = (Ts + DELT / 2)
'Time Step initial condition
DTM = (Tuncover + Tadiab - Tstart) / (600 / DELT)
'Cs diffusion parameter initial condition
DTBAR_Cs = 0
'Sb diffusion parameter initial condition
DTBAR_Sb = 0
'Time initial condition
T = Tstart
i = 0
ReDim Preserve f(11, i)
While T <= (Tuncover + Tadiab + Tzrc + Tmelt) * 1.00001
  'Cs
  'Cs Diffusion parameter as a function of temperature
  DBAR = Do_Cs * Exp((-Q_Cs / TBAR))
  'Sum of Diffusion parameters
  DTBAR_Cs = (DBAR * DTM) + DTBAR_Cs
  'FP fraction released based on (Kress, Booth) solution for spherical
  'concentration differential equation
  'as a function of diffusion parameter sum
  If (DTBAR_Cs / 0.00000036) > 0.1 Then
    FF = 1 - (6 / 3.142 ^ 2) * Exp((-(3.142) ^ 2 * DTBAR_Cs) / 0.00000036)
  Else: FF = 6 * ((DTBAR_Cs / (3.142 * 0.00000036)) ^ 0.5) - 3 * DTBAR_Cs / (0.00000036)
  End If
  'Cs fraction released from fuel pallets
  f(2, i) = FF

  'Sb
  'Sb Diffusion parameter as a function of temperature
  DBAR = Do_Sb * Exp((-Q_Sb / TBAR))
```

```
'Sum of Diffusion parameters
DTBAR_Sb = (DBAR * DTM) + DTBAR_Sb
'FP fraction released based on (Kress, Booth) solution for spherical concentration
'differential equation
'as a function of diffusion parameter sum
If (DTBAR_Sb) / (0.00000036) > 0.1 Then
   FF = 1 - (6 / 3.142 ^ 2) * Exp((-(3.142) ^ 2 * DTBAR_Sb) / 0.00000036)
Else: FF = 6 * ((DTBAR_Sb / (3.142 * 0.00000036)) ^ 0.5) - 3 * DTBAR_Sb / (0.00000036)
End If
'Sb fraction released from fuel pallets
f(3, i) = FF

'Others FP fraction released from fuel pallets based on Relative Volatile factors
For j = 4 To 11
   f(j, i) = f(2, i) * ((f(2, i) / f(3, i)) ^ ((RV(j - 4) - 1) / 0.32))
   If f(j, i) > 1 Then f(j, i) = 1
Next j

'Time
f(0, i) = T
'Temperature
f(1, i) = Ts

'Adiabatic heat up phase
If Ts >= 1200 And Ts < 1800 Then
   'Time Step during adiabatic heat up phase
   DTM = (Tuncover + Tadiab - Tstart) / (600 / DELT)
   'Time during adiabatic heat up phase
   T = T + DTM
   'Temperature during adiabatic heat up phase
   Ts = Ts + DELT
   'Temperature at core
   TBAR = Ts + DELT / 2
Else
   'ZR runaway phase
   If Ts >= 1800 And Ts < 3000 Then
      'Time Step during ZR runaway phase
      DTM = Tzrc / (1200 / DELT)
      'Time during ZR runaway phase
      T = T + DTM
      'Temperature during ZR runaway phase
      Ts = Ts + DELT
      'Temperature at core
      TBAR = Ts + DELT / 2
   Else
      'Melt phase
      If Ts >= 3000 Then
```

```
            'Time Step during melt phase
            DTM = Tmelt
            'Time during melt phase
            T = T + DTM
            'Temperature at core
            TBAR = 3000
        End If
    End If
  End If

  i = i + 1
  'Dynamic Allocation of FP fraction release matrix
  ReDim Preserve f(11, i)
Wend

i = i - 1
Tstep = 225
'Dynamic Allocation of FP fraction release matrix
ReDim Preserve f(11, i + ((TimeMAX - f(0, i)) / Tstep))
k = i + 1
'Loop to repeat the melt fission products release fraction up to end of scenario
Do While (f(0, k) < TimeMAX)
  For l = 2 To 11
     'FP release fraction kepts the same up to end of scenario
     f(l, k) = f(l, i)
  Next l
  'Time increment
  f(0, k) = f(0, k - 1) + Tstep
  'Core temperature kepts in 3000 K
  f(1, k) = f(1, i)
  k = k + 1
  If k > i + ((TimeMAX - f(0, i)) / Tstep) Then
    Exit Do
  End If
Loop

'Matrix of FP core fraction on containment air (not gravitational settling)
Dim Fcont() As Single
'Containment FP core fraction matrix dimensioning
ReDim Fcont(9, i + (TimeMAX - f(0, i)) / Tstep)
'FP core fraction released to encironment matrix
Dim Fenv() As Single
'FP core fraction released to encironment matrix dimensioning
ReDim Fenv(9, (i + ((TimeMAX - f(0, i)) / Tstep)))

'FP fraction release at Tstart along all scenario
For k = 0 To (i + (TimeMAX - f(0, i)) / Tstep)
```

```
    If (f(0, k) - f(0, 0)) / 3600 <= LambdaTime Then
      'Array of reduction mechanism Factors 1
      Lambda = Array(Lambda1, Lambda1, 0, Lambda1, Lambda1, Lambda1, Lambda1, _
      Lambda1, Lambda1, Lambda1)
    Else
      'Array of reduction mechanism Factors 2
      Lambda = Array(Lambda2, Lambda2, 0, Lambda2, Lambda2, Lambda2, Lambda2, _
      Lambda2, Lambda2, Lambda2)
    End If
    For l = 0 To 9
      If LeakageRate * (f(0, k) - f(0, 0)) / 3600 <= 1.001 Then
        'FP fraction release to containment at Tstart along all scenario
        Fcont(l, k) = f(l + 2, 0) * Exp(-Lambda(l) * (f(0, k) - f(0, 0)) / 3600) * _
        (1 - LeakageRate * (f(0, k) - f(0, 0)) / 3600)
        If k > 0 Then
          'FP fraction release to environment at Tstart along all scenario
          Fenv(l, k) = f(l + 2, 0) * Exp(-Lambda(l) * (f(0, k) - f(0, 0)) / 3600) * _
          LeakageRate * (f(0, k) - f(0, k - 1)) / 3600
        End If
      End If
    Next l
  Next k

  'FP fraction release along all scenario
  For j = 1 To (i + (TimeMAX - f(0, i)) / Tstep)
    For k = j To (i + (TimeMAX - f(0, i)) / Tstep)
      If (f(0, k) - f(0, k - j + 1)) / 3600 <= LambdaTime Then
        'Array of reduction mechanism Factors 1
        Lambda = Array(Lambda1, Lambda1, 0, Lambda1, Lambda1, Lambda1, Lambda1, _
        Lambda1, Lambda1, Lambda1)
      Else
        'Array of reduction mechanism Factors 2
        Lambda = Array(Lambda2, Lambda2, 0, Lambda2, Lambda2, Lambda2, Lambda2, _
        Lambda2, Lambda2, Lambda2)
      End If
      For l = 0 To 9
        'In the case the FP of containment have not completelly released to environment
        'calculus of the FP concentration on containment and released to environment
        If LeakageRate * (f(0, k) - f(0, k - j + 1)) / 3600 <= 1.001 Then
          'FP fraction release to containment along all scenario
          Fcont(l, k) = Fcont(l, k) + (f(l + 2, k - j + 1) - f(l + 2, k - j)) * _
          Exp(-Lambda(l) * (f(0, k) - f(0, k - j + 1)) / 3600) * (1 - LeakageRate * _
          (f(0, k) - f(0, k - j + 1)) / 3600)
          'FP fraction release to environment along all scenario
          Fenv(l, k) = Fenv(l, k) + (f(l + 2, k - j + 1) - f(l + 2, k - j)) * _
          Exp(-Lambda(l) * (f(0, k) - f(0, k - j + 1)) / 3600) * LeakageRate * _
          (f(0, k) - f(0, k - 1)) / 3600
```

```
            End If
        Next l
    Next k
Next j

k = 1
M = 0

'Matrix source term with time step adapted to RASCAL (15 minutes)
Dim st() As Single
'Matrix source dimensioned according scenario duration
ReDim st(9, (TimeMAX - f(0, 0)) / 900)

'Disable screen update
Application.ScreenUpdating = False
'Enable calculation
Application.Calculation = xlCalculationManual

' Clean old source term data
Range("STstart", Range("STstart").Cells(10, 10000)).ClearContents

For j = 0 To Fix((TimeMAX - f(0, 0)) / 900)
   Do While ((f(0, k) - f(0, M) < 900) And k <= Fix(i + ((TimeMAX - f(0, i)) / Tstep)))
        For l = 0 To 9
           'Sum of fraction releases to adapt time step
           st(l, j) = st(l, j) + Fenv(l, k)
        Next l
        k = k + 1
        If k >= Fix(i + ((TimeMAX - f(0, i)) / Tstep)) Then
            Exit Do
        End If
   Loop
   M = k
   'NG
   Range("STstart").Cells(1, j + 1) = st(2, j)
   'Te
   Range("STstart").Cells(2, j + 1) = st(3, j)
   'I
   Range("STstart").Cells(3, j + 1) = st(4, j)
   'Cs
   Range("STstart").Cells(4, j + 1) = st(0, j)
   'Sb
   Range("STstart").Cells(5, j + 1) = st(1, j)
   'Ba
   Range("STstart").Cells(6, j + 1) = st(5, j)
   'Sr
   Range("STstart").Cells(7, j + 1) = st(6, j)
```

```
    'Ru
    Range("STstart").Cells(8, j + 1) = st(7, j)
    'La
    Range("STstart").Cells(9, j + 1) = st(8, j)
    'Ce
    Range("STstart").Cells(10, j + 1) = st(9, j)
  Next j

  'Enable screen update
  Application.ScreenUpdating = True
  'Enable calculation
  Application.Calculation = xlCalculationAutomatic

End Sub
```