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Nuclear Powered Warship Reference Accident Review

Modelling, Vessel Design, Scenarios and Assumptions

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Summary

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Glossary/abbreviations

ACCIDENT	ARPANSA Excel-based tool for assessment of reactor accidents
ARTIST	Experimental program to investigate aerosol retention in a steam generator during SGTR
ASTEC	Accident Source Term Evaluation Code
B&W	Babcock & Wilcox
BWR	Boiling Water Reactor
CDF	Core Damage Frequency
CEA	Commisariat à l'Energie Atomique et aux Energies Alternatives
CONTAIN	Analysis tool developed by SNL for predicting the physical, chemical, and radiological conditions inside the containment and connected buildings of a nuclear reactor in the event of an accident
CORSOR	The CORSOR code simulates the release of fission products and structural materials from a reactor core during the in-vessel period of a severe accident in a light water reactor
CSE	Containment System Experiments
DARWIN-PEPIN	Computer code system for calculating the buildup, decay, and processing of radioactive materials
DCH	Direct Containment Heating
ECC	Emergency Core Cooling
EPR	Emergency Preparedness & Response
FP	Fission Product
HEU	High-Enriched Uranium
HX	Heat Exchanger
HXR	Heat Exchanger Rupture
IAEA	International Atomic Energy Agency
INSAG	International Nuclear Safety Group
IS-LOCA	Interfacing System – Loss Of Coolant Accident
IVR	In-Vessel Retention
KLT	PWR design developed for FNPP applications.
LBE	Lead Bismuth Eutectic
LB-LOCA	Large Break - Loss of Coolant Accident
LEU	Low-Enriched Uranium
LOCA	Loss Of Coolant Accident
LR	Lloyd's Register
LRF	Large Release Frequency
LTSBO	Long Term Station Blackout
LWR	Light Water Reactor
MAAP	Modular Accident Analysis Program
MCCI	Molten Core-Concrete Interaction
MELCOR	Methods for Estimation of Leakages and Consequences Of Releases

MELPROG	Integrated model for in-vessel melt progression analysis developed by SNL
MN	Marine Nationale
MOX	Mixed Oxide
NPW	Nuclear-Powered Warship
NRC	Nuclear Regulatory Commission
NUREG	Nuclear Regulatory Guide
ORIGEN	Computer code system for calculating the build-up, decay, and processing of radioactive materials
ORNL	Oak Ridge National Laboratory
PORV	Pilot Operated Relief Valve
PSA	Probabilistic Safety Assessment
PWR	Pressurized Water Reactor
RCS	Reactor Coolant System
RHR	Residual Heat Removal
RHR HX	Residual Heat Removal Heat Exchanger
RN	Royal Navy
RV	Relief Valve
SAMG	Severe Accident Management Guidelines
SFR	Sodium Fast Reactor
SGTR	Steam Generator Tube Rupture
STSBO	Short-Term Station Blackout
TISGTR	Thermally Induced Steam Generator Tube Rupture
TSA	Technical Safety Assessment
USN	United States Navy

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Executive summary

The safety aspect of nuclear-powered warships (NPWs) visiting Australian and Norwegian ports have been subject to several investigations over the years. ARPANSA uses, for their emergency preparedness and response activities, an NPW reference accident scenario and a Microsoft Excel tool, called the ACCIDENT code to assist in decision-making [1][2]. As the last major assessment on the reference scenario and ACCIDENT code was made in the year 2000, and considerable development has taken place since then, ARPANSA in collaboration with their Norwegian equivalent, DSA, has tasked Lloyd's Register (LR) with performing a review and assessment of the current reference accident scenario used for visiting NPWs in in Australian and Norwegian ports and territorial waters.

The review and assessment of the current reference accident scenario covers the following areas:

- Brief review of the current state of the art in assessment of consequences of severe accidents in LWRs.
- Review of nuclear-powered warship reactor designs and reference accident scenario applicability
- Review of assumptions and limitations of the ACCIDENT code
- Review of assumptions and limitations of the reference accident scenario

The review has highlighted several ways the current methodology and reference accident scenario can be improved with respect to its scope and quality.

For instance, the current methodology considers only LOCA (Loss Of Coolant Accident) as the most severe accident that can occur on an NPW, however there are other accident scenarios, such as SGTRs (Steam Generator Tube Rupture), IS-LOCAs (Interfacing System LOCAs), pipe breaks in the heat-exchangers for the shutdown state – where, in case of an accident involving core melt, fission products can bypass the reactor compartment to the secondary systems and/or directly to the environment.

The current reference accident scenario considers only particulate and organic iodine forms while the gaseous iodine form also has quite a significant contribution to immediate off-site consequences and should be considered in the analysis. Furthermore, this review has identified a set of parameters, such as release fractions for specific radionuclide groups, decontamination factors (DF), burn-up etc. that can be revised based on the values and insights presented in this report as well as considered as candidates for sensitivity analysis and uncertainty quantification.

Finally, since the detailed knowledge on NPW reactor safety design is quite limited, and the current state-of-the-art knowledge regarding many aspects of severe accident phenomena also involves significant amounts of uncertainties (both epistemic and aleatory), it is recommended to perform an iterative sensitivity analysis and uncertainty quantification. Such sensitivity analysis can assist in identifying the ACCIDENT code modelling parameters that have significant impact on the results (in terms of emergency preparedness and response, EPR), while uncertainty quantification will show the effect of the variability in these parameters on the model response (in terms of EPR) and the effect of risk acceptance criteria. The results of such analyses can be used to support decision-making, provide valuable safety insights into possible modifications to the requirements for ports visits, as well as insights regarding the knowledge gaps where reduction of uncertainties will be the most valuable.

Overall, the inherent uncertainties in this field point out the importance of having an emergency response planning that is relatively independent of the scenario details, and considering including requirements on stating limiting values on reactor burnup or time since refuelling in the conditions for entry.

1 Introduction

The safety aspect of nuclear-powered warships (NPWs) visiting Australian and Norwegian ports have been subject to several investigations over the years. ARPANSA uses, for their emergency preparedness and response activities, an NPW reference accident scenario and a Microsoft Excel tool, called the ACCIDENT code to assist in decision-making [1][2]. As the last major assessment on the reference scenario and ACCIDENT code was made in the year 2000, and considerable development has taken place since then, ARPANSA in collaboration with Norwegian DSA has tasked Lloyd's Register (LR) with performing a review and assessment of the current reference accident scenario used for visiting NPWs in in Australian and Norwegian ports and territorial waters.

The work is limited to western designs, specifically to ships and reactors belonging to the US Navy (USN), the Royal Navy (RN) and the Marine Nationale (MN), although some sections will be applicable also to NPW reactors of other countries.

As detailed design data on these ships and systems are classified, investigations will always have to rely on extrapolation from:

- Known data for commercial designs, both for land-based reactors as well as reactors for nuclear-powered civilian ships.
- Known approximate design data for NPW reactors.
- Known detailed data for specific accidents in NPW reactors.

This work is no different from earlier efforts in terms of access to classified information. The key elements where it strives to bring new knowledge to the NPW emergency preparedness table relate to:

- A detailed assessment of the state of the art in severe accident modelling.
- A reverse-engineering approach to possible reactor designs, allowing some quantification of uncertainties in crucial parameters.
- A detailed assessment of existing assumptions and limitations from a modern nuclear safety analysis perspective.
- An updated literature review on relevant topics.

Owing to the multidisciplinary profile of LR, the work has been assisted and reviewed (in relevant parts) by a senior expert in marine technology, with experience both of naval regulations and operation of nuclear submarines in the Royal Navy. What the senior expert has been able to reveal is necessarily limited by the UK Official Secrets Act.

2 Background

A few points on current reactor safety records can be enlightening. Today, it is estimated that commercial nuclear power has accumulated around 17 000 reactor years globally [3] while naval reactor operation for the US, the UK and France together can be estimated to have accumulated around 10 000 reactor years. While five major severe accidents in commercial western design power reactors have occurred, Three Mile Island (1 reactor) and Fukushima Daichi (4 reactors), no severe accidents in western-type NPW reactors are known to date. This discrepancy in numbers can be attributed to one or several of the following possible reasons, in no particular order:

- Fewer reactor years accumulated on naval reactors
- Inherently safer design or operation (e.g. larger safety margins) of naval reactors
- Statistical fluctuation
- Cover-up (less likely)

Since the Fukushima accident in 2011, which in addition to severe core damage led to large releases of radioactive substances, the commercial nuclear industry has experienced a continued increase in focus on beyond design basis accidents, stress tests, passive safety functions as well as procedures and tools for nuclear emergency preparedness and response (EP&R). It is reasonable to assume, but not guaranteed, that a similar development has occurred within nuclear navies around the world.

This work and report are organized according to the following four areas:

- Section 3 – A.1 – State of the art in severe accident consequence assessment
- Section 4 – A.2 – Nuclear-powered warship reactor designs and reference accident scenario applicability
- Section 5 – A.3 – Assumptions and limitations of the ACCIDENT code
- Section 6 – A.4 – Assumptions and limitations of the reference accident scenario

3 A.1 – State of the art in severe accident consequence assessment

This section gives a brief overview of the current state of the art in assessing consequences of severe accidents, with a main focus on the source term released to the environment.

Assessment of the environmental source term is a very complex problem that involves a great number of different sources of uncertainty that can affect the magnitude and the timing of fission products released to the environment.

Typically, estimation of the severe accident source term is performed using the following steps:

1. Estimate the inventory of fission products in the core.
2. Estimate the amount of fission products released from the core.
3. Estimate the source term into the containment.
4. Estimate the in-containment source term.
   - Identify and estimate the impact of the retention mechanisms in the containment.
5. Estimate the releases of radioactive substances into the environment (severe accident source term).

The review of the severe accident phenomenology will cover typical accident scenarios and phenomena in commercial LWRs and discuss possible relevance to the NPWs.

3.1 Fission product inventory

The production of radionuclides and the variation in their inventory during and after reactor irradiation are governed by the Bateman equations (see eq. (1) and [4]). This system of equations is typically solved numerically by using specific computer codes, such as ORIGEN2 [5] developed in the United States by ORNL and widely used in the international scientific community, or DARWIN-PEPIN [6] developed in France by CEA.

Note that based on the NRC regulations (e.g. [7]), the inventory of fission products in the reactor core, and available for release to the containment, should be based on the maximum full-power operation of the core. This means applying the following as a minimum:

- Currently licensed values for fuel enrichment and fuel burnup.
- An assumed core power equal to the currently licensed rated thermal power, multiplied with the emergency core cooling (ECC) system evaluation uncertainty.

These parameters should be examined to maximize the fission product inventory. On the other hand, the main difference between the operating history of commercial LWRs and NPW reactors is that the NPW reactors rarely operate at full power. Therefore the assumptions made in the reference accident scenario and ACCIDENT code (see [1] and [2]) can be considered more realistic for NPWs.

The period of irradiation should be of sufficient duration to allow the activity of dose-significant radionuclides to reach equilibrium or to reach maximum values. The core inventory should be determined using an appropriate isotope generation and depletion computer code, such as ORIGEN2.

Note that simplified approaches can be used for assessment of core inventories. Consider, e.g., the equations presented below. There are five routes of destruction and production of nuclide *i:*

- decay of the nuclide
- destruction of the nuclide by neutron reactions
- production of the nuclide by decay of precursors

- production by neutron reactions with other nuclides present
- production from fission of fissile materials present.

These can be accounted for by the three terms in the equation below. The first term represents generation from heavy nuclei by fission, the second term models generation from one (or several) parent nuclei by radioactive decay or by neutronic capture from one (or several) parents, and the third term represents disappearance by radioactive decay and/or by neutronic capture,

$$\frac{dN_i}{dt} = \phi \sum_{hn} \sigma_f^n y_n^i N_{hn} + \phi \sum_{j,k} (\alpha_i^j \lambda_{Pj} N_{Pj} + \beta_i^j \sigma_c^{Pk} \phi N_{Pk}) - (\sigma_a^j \phi + \lambda_i) N_i, \quad (1)$$

where

ϕ is the neutron flux ($n * cm^{-2}s^{-1}$),

σ_f^n is the microscopic fission cross section for each heavy nucleus,

y_n^i is the fission yield of each heavy nucleus for the fission product FP_i,

N_{hn} is the concentration of each heavy nucleus in the fuel,

α_i^j and β_i^j are the branching ratios (generally equal to 1) of the parents P_j and P_k to FP_i,

N_{Pj} and N_{Pk} are the concentrations of the parents P_j and P_k in the fuel,

λ_{Pj} is the radioactive decay constant of the parent P_j,

σ_c^{Pk} is the microscopic capture cross section of the parent P_k,

σ_a^j is the microscopic capture cross section of FP_i

λ_i is the radioactive decay constant of FP_i.

For most fission products (FP), the main term expressing their production comes from the fission of heavy nuclei via the associated fission yield, whereas the main term for the disappearance of radioactive FPs comes from their radioactive decay. Thus, the equation (1) above can therefore be simplified to

$$\frac{dN_i}{dt} = y_{eq}^i F - \lambda_i N_i, \quad (2)$$

where y_{eq}^i is the equivalent fission yield for all of the heavy nuclei (this information can be obtained from e.g. IAEA Nuclear Data Library), and $F = \phi \sum_{hn} \sigma_f^n N_{hn}$ represents the number of fissions/second, which can be simplified by the ratio P_{th}/E_f of the irradiation or thermal power (P_{th}) over the mean fission energy (E_f), equivalent to about ~200 MeV (or ~$3.2 \cdot 10^{-11}$ Ws).

Integration of equation (2) over the irradiation time t yields:

$$N_i = y_{eq}^i * \frac{F}{\lambda_i}(1 - e^{-\lambda_i t}) \quad (3)$$

Since

$$A_i = \lambda_i N_i = y_{eq}^i F (1 - e^{-\lambda_i t}), \quad (4)$$

we get

$$A_i = y_{eq}^i P_{th} E_f^{-1} (1 - e^{-\lambda_i t}), \quad (5)$$

where

A_i is the activity of nuclide i,

λ_i is the radioactive decay constant of FP_i,

y_{eq}^i is the equivalent fission yield for all heavy nuclei,

P_{th} is the thermal power,

E_f is the mean fission energy, equivalent to about ~200 MeV (or ~$3.2 \cdot 10^{-11}$ Ws).

Furthermore, two boundary conditions can be deduced from the simplified equation:

- For FPs with long half-lives in terms of irradiation time ($\lambda_i \to 0$), the inventory increases linearly as a function of time:

$$N_i = y_{eq}^i P_{th} E_f^{-1} t \tag{6}$$

- For FPs with short half-lives in terms of the irradiation time ($\lambda_i \to \infty$) the inventory will be limited by the saturation value and therefore stabilizes at this value when the irradiation time exceeds the FP half-life by approximately a factor of 5:

$$N_i = y_{eq}^i P_{th} E_f^{-1} / \lambda_i \tag{7}$$

Note that equation (5) reduces to the same form as in the ACCIDENT code (see eq. A.1.1.1 in [1]), if we assume a similar reactor core operating history as in Figure 3.3 in [1].

3.2 In-vessel fission products release

During severe accidents involving core melt, gases, vapors and airborne particles (aerosols) are formed. A small part of these substances are the radioactive fission products (FPs), representing the source terms.

Although there are some major differences between commercial LWRs and NPWs, these differences should not affect the fission products inventory in the core, which is mainly driven by fuel burnup. When it comes to FP release kinetics, other factors such as fuel composition, cladding material, etc. start to play a significant role. In the following we discuss the main aspects of FP release from the fuel (oxidic fuel with zircalloy cladding used in commercial western LWRs). In section 3.2.1.1 we give some insights into the behavior of metallic fuels under severe accident conditions.

The release of fission products from the fuel depends on the degree of volatility, which is typically classified into the following categories [9][10]:

- **Fission gases** (xenon and krypton) and **volatile FPs** (iodine, caesium, rubidium, tellurium, antimony, etc.)

  The major part of these products is released before reaching molten pool conditions. The release kinetics of these elements are accelerated under oxidizing conditions.

- **Semi-volatile FPs** (molybdenum, rhodium, barium, palladium, and technetium)

  These are characterized by high levels of release, which are sometimes equivalent to those of volatile FPs. Release of semi-volatile FPs is highly sensitive to the oxidizing-reducing conditions and resulting in significant retention in the upper core structures.

- **Low-volatile FPs** (strontium, yttrium, niobium, ruthenium, lanthanum, cerium, and europium)

  These are characterized by low but significant levels of release, ranging from a few percent to 10 % of the initial inventory during the fuel rod degradation phase (prior to loss of core geometry). Nevertheless, these releases can reach much higher levels for oxide fuels with very high burnups (15 to 30 % releases were measured for Nb, Ru, and Ce in UO_2 fuel at 70 GWd/t), or under specific conditions. For example, the release of Ru can be almost total in air, whereas reducing conditions will favor the release of Sr, La, Ce, and Eu. Significant retention of these elements is nevertheless expected in the upper core structures.

- Non-volatile FPs (zirconium, etc.)

  To date, no significant release of these three elements has been demonstrated experimentally. These are the most refractory FPs.

- Actinides

  These can be subdivided into two categories:

  - The first category includes uranium and neptunium whose releases can reach 10 % before the fuel melts. This behavior is similar to that of low-volatile FPs. Uranium releases are higher in oxidizing conditions, whereas neptunium releases are favoured by reducing conditions.
  - The second category includes plutonium whose releases are always very low; typically below 1 % which means that they tend to behave more like non-volatile FPs. Plutonium releases can be higher under reducing conditions.

The typical release phases of fission products to the containment (the containment source term) during an accident are summarized in [9] and presented below:

- Coolant activity release

  NUREG-1465 [9] assumes an accident initiated by a large-break LOCA, therefore the first activity release to the containment atmosphere will be the activity dissolved in the coolant during normal operation of the reactor. It has been evaluated that this phase can go on for 10-20 seconds for commercial LWRs [9]. It should be noted that the coolant activity release phase is outside the scope of this project.

- Gap release

  Gap release starts at cladding failure during fuel heat-up to 800-900 °C (e.g., in the MELCOR code, there is cladding failure and gap release when the cladding temperature exceeds 1173 K [11]), allowing release of the more volatile radionuclides such as noble gases (xenon and krypton), iodine, and caesium, which have been released from the fuel during normal operation, and accumulated in the gap between the fuel and the cladding. Typically, this is a small percentage of the fuel inventory of these elements, and the duration of the release is typically about half an hour.

  In this phase, most of the FPs are still retained in the fuel matrix itself. This phase ends when the temperature reaches such a level that significant amounts of FPs can no longer be retained within the fuel.

- In-vessel (RPV) release

  In the in-vessel phase, as the temperatures of fuel and structural materials reach melting temperatures, all gaseous and volatile products are mostly released, as well as a part of the less volatile species. The in-vessel phase ends with the vessel lower head failure and melt/debris ejection to the space under the reactor pressure vessel.

  - Note that for some reactor designs, vessel lower head failure can be avoided by so called In-Vessel Melt Retention (IVMR) – which is a part of the Severe Accident Management Guidelines (SAMG) for some PWR) designs (e.g. Loviisa VVER-440 in Finland, AP-600 and AP-1000, Korean APR-1400, etc.) [10].

IVMR may be relevant for NPWs, since these typically have relatively small thermal reactor power, since PWR designs typically have in-core instrumentation and control systems inserted from the above (thus no or few vessel lower head penetrations), and since there is availability of an ultimate heat sink.

Some important aspects of the in-vessel phase of severe accident progression and fission products release are the following [10].

- The fuel temperature is one of the main parameters that drives the process of fission product release from the fuel, at least until loss of the core geometry.
- The oxidizing-reducing conditions have a significant impact on the fuel. The release kinetics of volatile FPs are particularly accelerated under oxidizing conditions. Furthermore, the overall release of certain FPs is very sensitive to the oxygen potential. For example, the release of Mo increases in steam, whereas that of Ru can be very high in air. Conversely, the release of Ba (as for Sr, Rh, La, Ce, Eu, and Np) increases under reducing conditions.
- The interactions with the cladding and/or the structural elements can play a major role.
- The burnup accentuates releases, in terms of both the kinetics of volatile FPs and the release amplitude of low-volatile species such as Nb, Ru, La, Ce, and Np.
- The fuel type also seems to have a significant impact: MOX releases tend to be higher than those of UO_2. This phenomenon is probably related to its heterogeneous microstructure, with the presence of plutonium-rich agglomerates where the local burnup can be very high.
- The state of the fuel during its in-vessel degradation has a significant influence: The transition from a “degraded rod” geometry to a “debris bed” geometry also involves an increase in releases via the increase in the surface-to-volume ratio. Conversely, the transition from a debris bed to a molten pool slows down the release of FPs as a solid crust forms on the surface of the molten pool.
- Rapid fuel or debris cooling, or heating (e.g. due to cladding oxidation, or debris quenching during reflooding or when the core debris collapses to the lower plenum due to failure of the support structures) can also significantly increase the release of fission products from the fuel, due to microcracking which can accelerate the release of fission products trapped in the grain boundaries.

- Ex-vessel release

  One of the major concerns of ex-vessel severe accident progression in commercial LWRs from the containment integrity and source term perspective is the molten corium concrete interactions (MCCI), which can result in generation of significant amounts of non-condensable gases, hydrogen, and large amounts of aerosols. MCCI is not expected to be an issue for NPWs, since the containment hull is made of steel.

  - Note that even though the MCCI has a negative effect on the containment pressurization rate, it has a positive effect on the rate of aerosol deposition in the containment (due to increased concentration of non-radioactive aerosols in the containment atmosphere) [10].

  However, there are other ex-vessel phenomena that can significantly affect the containment source term, such as the mode of containment failure and the magnitude of fission product release to the environment. In particular, these phenomena may be

  - direct containment heating in case of high-pressure failure of the reactor vessel lower head

- ex-vessel steam explosion in case of low-pressure failure of the vessel lower head
- ex-vessel debris coolability and containment melt-through in case of low-pressure failure of the vessel lower head.

3.2.1 Fuel release fractions

In accident simulations, fuel release rates (and thus fuel release fractions) are typically calculated using specific models that are generally based on experimental data (empirical correlations).

For example, the MAAP and MELCOR codes employ the CORSOR, CORSOR-M or CORSOR-BOOTH models to calculate release rates from the fuel for different fission products (for details, see [11]).

On the other hand, different sources provide estimates of fuel release fractions for different accident scenarios. For example NUREG-5747 (Estimate of Radionuclide Release Characteristics Into Containment Under Severe Accident Conditions) [12] provides estimates (see Table 1) of fuel release fractions to the reactor coolant system (RCS) prior to vessel lower head failure. Note that NUREG-5747 is majorly based on the calculations performed to support the development of NUREG-1150 (using the Source Term Code Package, that includes the CONTAIN, MELCOR and MELPROG codes), as well as conclusions from the expert panel elicitation presented in the Appendix B of NUREG-1150 and NUREG-4551.

Table 1. Mean and median values for fission product releases from the core into the RCS for PWRs [12].

	Core Release Fractions for PWRs 1)	
	High Zr oxidation	Low Zr oxidation
NG	0.92 (0.83)	0.9 (0.8)
I	0.75 (0.71)	0.69 (0.6)
Cs	0.62 (0.61)	0.58 (0.55)
Te	0.33 (0.36)	0.19 (0.3)
Sr	0.006 (0.07)	0.004 (0.07)
Ba	0.009 (0.08)	0.006 (0.08)
Ru	0.005 (0.02)	0.002 (0.01)
La	0.0001 (0.004)	0.0001 (0.004)
Ce	0.00015 (0.02)	0.00015 (0.02)

1) Mean values are shown within parentheses.

These results indicate that the core release fractions are typically higher for scenarios with significant Zr oxidation. Furthermore, the results (and distributions presented in [12]) indicate that there are significant uncertainties in core release fractions.

3.2.1.1 Behavior of metallic fuels under severe accident conditions

Note that the release fractions presented above and in the following sections are based on the analysis performed for the conventional western LWRs with ceramic fuels. US and UK NPW reactor designs however typically employ metallic fuels (e.g. U-Zr based fuels) [13][15][16][17][18]. It is pointed out in [19] and [20] that one of the major issues with metal alloy fuels is that they may undergo a large amount of swelling from low burnup. This typical behaviour is primarily due to fission gas, neutron flux irradiation damage, fuel composition and high operating temperatures. Fuel swelling was deemed to have a high regulatory concern [21]. Most swelling occurs during the first few percentages of burnup (~2-3 %) at which point the voids causing the swelling interconnect and the subsequently generated fission gas is released to the fission gas plenum [19][20][21].

Work performed in [22] gives an overview of the experimental programs related to metallic fuel behaviour under different conditions, major knowledge gaps and sources of uncertainty, as well as bounding assessment of the fuel release fractions. The main results of this work are summarized in Table 2.

Table 2. Fuel release fraction estimates [22].

Radionuclide group	Characteristics	Normal operation ~500 °C	Eutectic formation ~700 °C	Fuel melting ~1100 °C	High temperatures ≥ 1300 °C
Noble gases (Xe, Kr)	Release Percentage	≤ 85 %	≤ 100 %	~100 %	~100 %
	Dependencies	Burnup	Burnup	Burnup	None
	Uncertainty Level	Low	Medium	Low	Low
Halogens (I)	Release Percentage	≤ 15 %	≤ 20 %	≤ 30 %	≤ 100 %
	Dependencies	Burnup	Burnup	Burnup	Time, Temp.
	Uncertainty Level	Medium	Medium	Low	Low
Alkali metals (Cs)	Release Percentage	≤ 55 %	≤ 60 %	≤ 100 %	≤ 100 %
	Dependencies	Burnup, Composition	Burnup, Composition	Burnup, Time, Composition	Time
	Uncertainty Level	Low	Medium	Medium	Low
Tellurium group (Te)	Release Percentage	≤ 1 %	≤ 1 %	≤ 5 %	No data
	Dependencies	Composition	Composition	Composition	-
	Uncertainty Level	Medium	Medium	High	-
Barium Strontium group (Ba)	Release Percentage	≤ 5 %	≤ 10 %	≤ 15 %	≤ 20 %
	Dependencies	Unknown	Unknown	Unknown	Unknown
	Uncertainty Level	Medium	Medium	High	Medium
Barium Strontium group (Sr)	Release Percentage	≤ 0.1 %	≤ 5 %	≤ 20 %	≤ 20 %
	Dependencies	Unknown	Unknown	Unknown	Unknown
	Uncertainty Level	Medium	Medium	High	High
Noble metals (Ru)	Release Percentage	≤ 0.1 %	≤ 1 %	≤ 5 %	≤ 5 %
	Dependencies	None	None	None	None
	Uncertainty Level	Low	Medium	Medium	Medium
Lanthanides	Release Percentage	≤ 0.1 %	≤ 1 %	≤ 30 %	≤ 30 %
	Dependencies	Unknown	Unknown	Unknown	Unknown
	Uncertainty Level	Medium	High	High	High
Cerium group (Ce)	Release Percentage	≤ 0.1 %	≤ 1 %	≤ 30 %	≤ 30 %
	Dependencies	Burnup	Burnup	Burnup	Burnup
	Uncertainty Level	Medium	High	High	High

Note that the results presented in Table 2 and Table 1 are not directly comparable, since Table 2 shows estimates of the upper boundaries, while Table 1 shows mean and median values.

Furthermore, it is important to note that the major part of the experimental and analytical work related to the behavior of metal fuels is performed for Sodium Fast Reactors (SFR), or for Lead or Led-Bismuth Eutectic (LBE) Reactors. Therefore, some aspects that might be relevant to LWR accident progression can be irrelevant for early phases of SA progression in metal-cooled reactors (e.g. fuel reactions with steam, etc.).

3.3 Containment source term and possible release paths to the environment

The containment source term is defined by the release from the fuel as well as by retention of different fission products in the RCS. During a severe accident in a nuclear power plant, fission products and structural materials are released as gases or vapors from the degrading core into the reactor coolant system. These are then swept, in general, by the gas mixture of steam and hydrogen down the RCS towards an opening/breach location. A number of important physico-chemical processes occur between the point of release from the core and release via the breach of still-suspended materials into the containment (or into the auxiliary building in the case of a containment-bypass sequence). The physical effects taking place involve primarily aerosol physics and dynamics [10].

The main mechanisms that can affect retention of fission products in the RCS are (i) chemical reactions of gases and vapors with other gases and vapors, aerosols and structural surfaces; (ii) homogeneous and heterogeneous nucleation of vapors and formation of aerosols; (iii) agglomeration of aerosols (Brownian diffusion, sedimentation, turbulence); (iv) deposition of aerosols by diffusion, thermophoresis (due to temperature gradient), inertial impactation (due to turbulence or flow geometry changes); etc. All these mechanisms, and many others, are modelled in computer codes such as ASTEC, MAAP or MELCOR, which are used to predict the consequences of severe accidents in LWRs. More information about these phenomena and the respective modelling approaches can be found in [10], [11] and [23].

For the purposes of NPW accident analysis it is a very daunting task to implement such approaches in the ACCIDENT code due to the complexity of these phenomena (e.g. thermal-hydraulic conditions), lack of detailed data regarding NPW designs, as well as significant uncertainties in existing models.

Therefore, it would be more adequate to use estimates of retention factors for different FPs in the RCS and/or release fractions for the containment source term. One of the most common references for the fractions of total core inventory released to the containment is the NUREG-1465 [9], where Table 3 shows the fractions of total core inventory released to the containment in case of a severe accident initiated by an unmitigated LOCA.

Table 3. Core inventory fractions released to the containment in case of unmitigated LOCA in PWRs [9].

	Gap release	Early in-vessel	Ex-vessel	Late in-vessel
Duration (hours)	0.5	1.5	3	10
Noble gases[1]	0.05	0.95	0	0
Halogens	0.05	0.25	0.3	0.01
Alkali metals	0.05	0.2	0.35	0.01
Tellurium group	0	0.05	0.25	0.005
Barium, strontium	0	0.02	0.1	0
Noble metals	0	0.0025	0.0025	0
Cerium group	0	0.005	0.005	0
Lanthanides	0	0.002	0.005	0

[1] The gap release is 3 % if long-term fuel cooling is maintained.

Note that the results presented in NUREG-1465 were derived from the simplification of the results presented in NUREG-1150 [25] and NUREG-5747 [12].

For example, Table 4 illustrates the fractions of the core inventory released to the containment up to the vessel breach in PWRs.

Table 4. Mean and median values of in-vessel releases into containment up to vessel breach for PWRs [12].

Conditions		Release fractions[1]								
RCS Pressure[2]	Zr oxidation[3]	NG	I	Cs	Te	Sr	Ba	Ru	La	Ce
SP	H	0.92 (0.83)	0.05 (0.14)	0.02 (0.11)	0.007 (0.05)	1.5E-4 (0.01)	3.E-4 (0.015)	2.E-5 (3.E-3)	3.E-6 (9.E-4)	3.E-6 (4.E-3)
SP	L	0.9 (0.79)	0.04 (0.12)	0.02 (0.1)	0.005 (0.04)	8.E-5 (0.01)	1.E-4 (0.01)	1.E-5 (2.E-3)	2.E-6 (9.E-4)	3.E-6 (4.E-3)

Conditions		Release fractions[1]								
RCS Pressure[2]	Zr oxidation[3]	NG	I	Cs	Te	Sr	Ba	Ru	La	Ce
H&I	H	0.92 (0.83)	0.26 (0.29)	0.16 (0.22)	0.06 (0.11)	1.E-3 (0.025)	2.E-3 (0.03)	6.E-4 (6.E-3)	2.E-5 (1.5E-3)	3.E-5 (8.E-3)
H&I	L	0.9 (0.79)	0.18 (0.24)	0.12 (0.2)	0.04 (0.09)	7.E-4 (0.02)	1.E-3 (0.03)	3.E-4 (4.E-3)	2.E-5 (1.5E-3)	3.E-5 (8.E-3)
L	H	0.92 (0.83)	0.34 (0.39)	0.21 (0.3)	0.08 (0.15)	2.E-3 (0.03)	3.E-3 (0.04)	1.E-3 (8.E-3)	3.E-5 (2.E-3)	6.E-5 (0.01)
L	L	0.9 (0.79)	0.26 (0.26)	0.17 (0.26)	0.06 (0.12)	1.E-3 (0.03)	2.E-3 (0.04)	4.5E-4 (6.E-3)	3.E-5 (2.E-3)	5.E-5 (0.01)

[1]The mean values are presented in parentheses.

[2] SP refer to Set-Point pressure (172 bar, release through cycling PORVs), H&I - High (41-138 bar, release via small break or pump seal LOCA) and Intermediate pressure (13 to 41 bar, release through ~2" break (5.08 cm)), and L - Low pressure (below 13 Bar – large break LOCA) RCS pressure, respectively

[3]H and L refer to High and Low In-vessel Zr oxidation

The estimated fractional releases depend strongly on the volatility of the fission products, as might be expected. The volatile fission products iodine and caesium have similar releases. The difference between Sr and Ba is small. The low volatile fission products Ce and La also have similar releases. Low-pressure sequences are characterized by rapid blowdown (small residence time) of the RCS and with little gravitational settling (one of the dominant mechanisms for aerosol deposition in the reactor coolant system). On the other hand, for high pressure sequences the fission products released from the fuel (except for noble gases) are retained in the reactor coolant system with higher efficiency.

Furthermore, NUREG-5747 [12] provides bounding values for RN releases into the containment under severe accident conditions for PWRs. These bounding values (except the source term due to MCCI) are presented in Table 5.

Table 5. Bounding values of radionuclide releases into the containment under severe accident conditions for PWRs [12].

	In-vessel release		DCH[1] (high pressure melt ejection)	Re-volatilization	
RN Groups	High RCS Pressure	Low RCS Pressure	High RCS Pressure	High RCS Pressure	Low RCS Pressure
NG	1.0	1.0	0	0	0
I	0.3	0.75	0.1	0.05	0.02
Cs	0.3	0.75	0.1	0.02	0.02
Te	0.2	0.5	0.05	0.02	0.01
Sr-Ba	0.003	0.01	0.01	-	-
Ru	0.003	0.01	0.05	-	-
La-Ce	5.E-5	1.5E-4	0.01	-	-
Release duration	40 minutes		-	10 hours	

[1]Direct Containment Heating

3.3.1 Fission products retention in the containment

Fission products released to the containment are subject to the same retention/remobilization mechanisms as during the transport in the RCS. The work performed in NUREG-6189 [24] presents a set of simplified models for assessment of aerosols retention by natural processes in reactor containments.

Table 6 to Table 9 show the estimates of decontamination factors (decontamination factors and effective decontamination factors per hour) for different reactor powers (1000 and 2000 MWth) and containment residence times.

Table 6. Median (50 percentile) decontamination factors for pressurized water reactors, 1000 MWth.

Time (sec)	Gap release (DF)	In-vessel release (DF)
0	1	1
1800	1.0139 – 1.0141	1
6480	1.0865 – 1.0921	1.0379 – 1.0396
13680	1.722 – 1.764	1.65 – 1.665
42480	41.864 – 46.577	40.054 – 44.162
80000	190.68 – 225.78	176.58 – 214.29
100000	302.30 – 364.41	219.24 – 344.76
120000	440.65 – 502.47	416.65 – 473.37

Table 7. Median (50 percentile) decontamination factors for pressurized water reactors, 2000 MWth.

Time (sec)	Gap release (DF)	In-vessel release (DF)
0	1	1
1800	1.0166 – 1.0170	1
6480	1.106 – 1.109	1.0449 – 1.0458
13680	1.536 – 1.562	1.448 – 1.478
42480	23.579 – 25.203	22.110 – 23.811
80000	116.14 – 128.28	110.98 – 121.36
100000	183.43 – 206.29	167.54 – 195.85
120000	239.19 – 296.71	225.36 – 281.06

Table 8. Effective median (50 percentile) decontamination factors for pressurized water reactors, 1000 MWth.

Time (sec)	Gap release (DF) (1/hour)	In-vessel release (DF) (1/hour)
0 - 1800	0.0276 – 0.0280	0
1800 - 6480	0.0532 – 0.0558	0.0286 – 0.0299
6480- 13680	0.233 – 0.237	0.233 – 0.237
13680 - 42480	0.406 – 0.411	0.406 – 0.411
42480 - 80000	0.134 – 0.147	0.134 – 0.147
80000 - 100000	0.0832 – 0.0849	0.0832 – 0.0849
100000- 120000	0.0658 – 0.0682	0.0658 – 0.0682

Table 9. Effective median (50 percentile) decontamination factors for pressurized water reactors, 2000 MWth.

Time (sec)	Gap release (DF) (1/hour)	In-vessel release (DF) (1/hour)
0 - 1800	0.0329 – 0.0337	0.0
1800 - 6480	0.0653 – 0.0673	0.0338 – 0.0334
6480- 13680	0.164 – 0.173	0.164 – 0.173
13680 - 42480	0.338 – 0.348	0.338 – 0.348
42480 - 80000	0.144 – 0.152	0.144 – 0.152
80000 - 100000	0.0835 – 0.0843	0.0835 – 0.0843
100000- 120000	0.0669 – 0.0677	0.0669 – 0.0677

Note that the decontamination factors increase with increase of the thermal power of the reactor. This behavior can be explained by the difference in the aerosol concentration in the containment atmosphere. Thus, the decontamination factors for NPW reactors can be somewhat smaller compared to the results presented in the tables above.

Furthermore, engineered safety systems such as containment sprays can significantly increase the suspended activity deposition and decrease the containment source term. According to [9], for typical PWRs, containment spray systems are capable of rapidly reducing the concentration of airborne activity. Once the bulk of the activity has been removed, however, the spray becomes significantly less effective in reducing the remaining fission products. Based on the parametric model developed in [26], decontamination factors of 10 can be achieved within 30 minutes of activity of containment sprays and factors of 100 within approximately 2 hours in typical PWRs.

3.3.2 Release paths to the environment

The paths taken by radionuclides during a severe accident with releases to the environment can be schematically visualized in a so-called release path diagram. The basic paths and retention volumes for light-water reactors are shown in Figure 1.

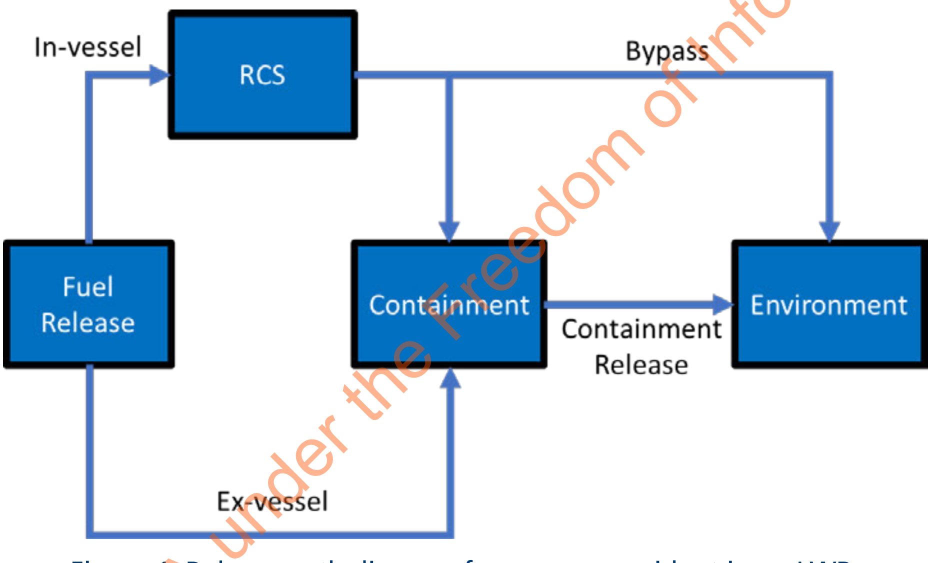

Figure 1 Release path diagram for a severe accident in an LWR.

The release paths can be classified into the following categories, which are also relevant for NPWs:

- Containment bypass.

  Direct release of radioactive material to the environment or the structures surrounding the containment (e.g. reactor buildings) which may not retain the FPs effectively, by which the retention capability of the containment is bypassed.

  Typical examples of containment bypass are:

  - Steam generator tube ruptures (SGTRs) in PWRs, which allow release of radioactive materials through the secondary side of the steam generators.
    - Note that SGTR sequences can a priori be relevant for NPWs, since all NPWs studied in this project employ pressurized water reactor design.
    - Interfacing system LOCAs (IS-LOCA), which allow radioactive materials to be released through a breach to a system outside the containment that interfaces with the RCS. Note that in the case of an NPW, the reactor compartment, which act as a containment, is surrounded by a watertight hull, which can act as a secondary containment with significantly higher design pressure compared to similar structures in land based LWRs.

- It should be noted that the regulatory authorities in many countries consider spontaneous SGTR as design basis accidents for western PWRs. Plants are designed to cope with such accidents, and no major consequences should be expected. However, a particular safety challenge arises from SGTR in combination with other failures. For example, if the safety valves of the affected steam generator are stuck open, the result will be a loss of coolant that eventually will lead to core degradation and meltdown if operator actions to prevent/mitigate the consequences are ineffective. Under these conditions, FPs released from the reactor would bypass the containment. Such scenarios are very unlikely, but given the severity of potential consequences of a direct path for FPs from the primary coolant system to the environment, they are estimated to be important risk contributors [28]. For the case of visiting NPWs in port, it is expected that the electric power will be supplied from the shore side and that the reactor will be in shutdown. Thus, there is a possibility that the primary pressure can be decreased (limited by the saturation pressure in the primary side of the RCS) to reduce stress on the RCS components (piping, valves, seals, etc.).
  - In addition to spontaneous SGTR, steam generator tube integrity may be challenged by high-temperature and high-pressure conditions during severe accidents. Consequently, they may have a potential to fail due to creep rupture or other flaws. These sequences are called severe accident-induced SGTRs.
  - The potential retention within the secondary side of a failed steam generator during a SGTR severe accident sequence was seen as one of the largest uncertainties in the analyses reported in NUREG-1150 [28]. An expert elicitation panel considered that little retention of radionuclides would occur in the reactor coolant piping and the failed steam generator. They estimated the overall transmission factor from the reactor to the environment to be higher than 75 % for all radionuclides considered. Consequently, attenuation of this magnitude was attributed to retention in the primary coolant piping.
  - It was estimated from ARTIST tests that the steam generator aerosol decontamination factor in a full-scale steam generator would be between 4.7 and 9 [27][10].

- Containment rupture.

  Containment rupture due to over-pressurization.

  - A severe accident and core melt can cause pressure build-up in the containment that may eventually lead to containment failure due to over-pressurization. There are several factors that influence containment pressurization, such as: (i) initial free volume in the containment; (ii) reactor thermal power; (iii) availability of pressure suppression systems; (iv) phenomena (DCH, hydrogen deflagrations and detonations, production of non-condensable gases). Typically, containment pressurization is a relatively slow process, however in some accident sequences containment overpressure can occur very early, e.g. for large break LOCA in BWRs with inadequate pressure suppression.
  - A paper by Lewis B.J. [29] suggests that the design pressure of NPW containments can reach 2 MPa, and that the pressure in the reactor compartment is not expected to increase over that value in the event of a LOCA. Given that the estimated design pressure of the NPW containment is then ~4 times the design pressure for typical western LWR containments, the issue of containment over-pressurization can be considered as relevant but unlikely. Furthermore, design parameters of vessel hulls are such that it should withstand severe loads due to external conditions, especially for submarines. Therefore, the vessel hull should provide an additional safety barrier between the fission products and the environment.

  Containment melt-through.

- In case of a severe accident and core melt, the molten corium can fail the reactor pressure vessel and relocate to the containment floor where it can come in contact with the containment floor and result in eventual failure. The main factors that can affect the likelihood of containment melt-through are: (i) reactor thermal power; (ii) thickness of the reactor pressure vessel and the containment walls; (iii) amount of water in the vessel and the containment (IVMR, melt fragmentation and quench). Furthermore, the containment melt-through should be considered in the context that on the other side of the vessel hull there is an infinite supply of relatively cold water. Containment melt-through can therefore be a relevant failure mode, however the release to the environment can be limited due to the additional safety barrier provided by the vessel hull.

Containment rupture due to phenomena.

- Hydrogen combustion.
    - Hydrogen is a burnable gas, which means that it reacts chemically with oxygen to form water: $2H_2 + O_2 \rightarrow 2H_2O$ + 120 MJ/kg.
    - Hydrogen combustion can occur in the containment atmosphere during a severe accident if the following conditions are satisfied: (i) flammable gas concentration (can be inferred from the well-known Shapiro diagram, see Figure 2); (ii) presence of an ignition source.
    - Depending on the gas mixture, several combustion regimes are possible: (i) hydrogen deflagration – characterized by a subsonic flame propagation with pressure spikes of a few bars magnitude; (ii) detonation – characterized by supersonic flame propagation, with large pressure spikes of typically 15-20 bar, but in some cases reaching 40 bar due to reflections and super-positioning of shock waves.
    - Hydrogen combustion can pose a credible threat to containment integrity in case of air-filled containments. Therefore, hydrogen combustion risks should be considered in the analysis of potential consequences of severe accidents in NPWs.

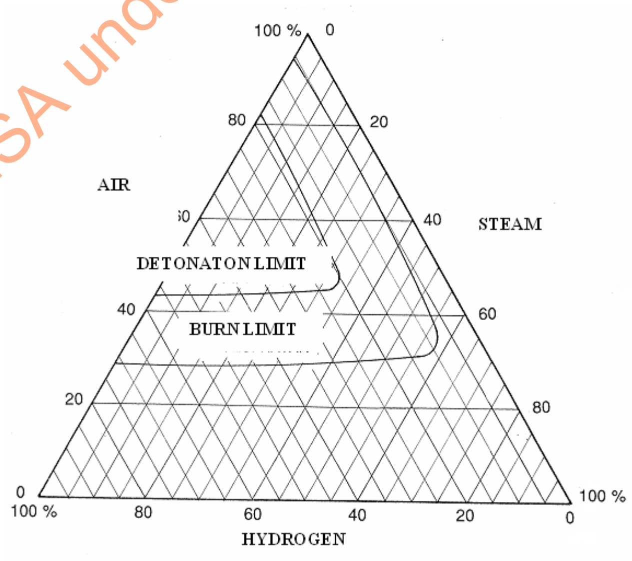

Figure 2. Shapiro diagram. Hydrogen flammability limits.

- Ex-vessel steam explosion.
    - After vessel melt-through, the molten corium slumps into the space under the reactor pressure vessel. Due to the (commercial LWR) accident management strategy to flood the cavity under the RPV before failure for quenching the melt, or

because it is very likely that water will collect there during the earlier phases of the accident, ex-vessel steam explosion has to be considered as a potential risk for containment integrity. The main contributing factors to ex-vessel steam explosion energetics are: (i) the amount of corium available for interactions with fluid; (ii) the amount and temperature of water in the cavity; (iii) the corium superheat; (iv) the corium composition.

- The issue of ex-vessel steam explosion (even in shallow pools) can pose a credible threat to the containment and should be considered in the analysis of potential consequences of severe accidents in NPWs.

Diffuse leakage (intact containment)

- Even if containment integrity is maintained during a severe accident, it cannot be assumed to be absolutely leak-tight. A so-called diffuse leakage occurs, which can amount to several tenths of a percent by volume per day. Furthermore, accident phenomena can negatively impact the leak-tightness of the containment.
- Large-scale experiments were performed in Japan by NUPEC on actual containment penetrations of a BWR plant using dry CsI aerosol particles, indicating decontamination factors between 10 and 1000 [10].
- The Containment System Experiments (CSE) program results indicated aerosol decontamination factors from 10 to 100 under dry conditions (15 for iodine and 100 for caesium), and complete retention under wet conditions [30].
- Note that in case of an NPW, the vessel hull can act as an additional barrier and limit releases of radioactive materials to the environment.

Another important aspect that can significantly affect the accident progression and the magnitude of the release is the reactor vessel lower head failure mode, which depends on design, accident scenario and phenomena.

3.3.3 Iodine chemistry

In a core meltdown accident, iodine with its high volatility can be almost completely released from the degraded fuel and be subsequently transported through the RCS to the containment

Based on RG 1.195 [32], in case of a DBA LOCA, 5 % of the radioiodine released from the RCS to the containment in a postulated accident should be assumed to be particulate iodine, 91 % to be elemental iodine, and 4 % to be organic iodide. In case of a SGTR, iodine releases from the steam generators to the environment should be assumed to be 97 % elemental and 3 % organic [32]. On the other hand, RG 1.183 [7] suggests that the iodine will be released in form of CsI (95 %), elemental iodine (4.85 %) and organic form (0.15 %). Note that RG1.183 requires licensees to calculate the total effective dose equivalent, while RG1.195 is used for calculation of whole-body and thyroid dose (see [7] and [32] for details)

While it was historically considered that iodine entering the containment was principally CsI, results from the Phébus FP series showed that the situation was more complex. Firstly, with CsI not being the dominant form, and secondly because the fraction of iodine aerosol and gas/vapour depend on the fission product release kinetics and probably on the nature of the control rods. Depending on the test conditions, the early gaseous iodine fraction measured in the containment can range from 1 to 2 % (FPT1, FPT2, with Ag/In/Cd control rods – which is typical for PWRs) to 97 % (FPT3, B_4C control rods – which is typical for BWRs) [31].

Furthermore, organic materials are present in the containment due to different sources such as paints and cables. Their radiolysis leads to the formation of organic radicals that will react with iodine (I_2) to form organic iodides (CH_3I, methyl-iodide, is one the most volatile forms – other forms are generally not considered in the safety analysis). Several studies (e.g. BIP, BIP2, STEM Phase 1) showed that that I_2 can be much more efficiently trapped by painted surfaces than by stainless steel ones. The paints have a twofold action: They act not only as a sink for volatile inorganic iodine I_2, but also as a source of volatile organic iodides. Radiation plays a strong enhancing role in the latter process and has much more influence than temperature [10].

IRSN performed a study [33] to calculate a "reasonably" conservative source term for the French 900 and 1300 MWe PWRs in case of a LB-LOCA (large break LOCA) accident with a total failure of the safety systems (safety injection and containment heat removal system) and a late containment failure by over-pressurization and base-mat penetration. Table 10 shows the resulting release fractions (of initial core inventory) of different chemical forms of iodine released to the environment.

Table 10. Release fractions of different forms of iodine obtained by IRSN for the S3 source term evaluation in 2000 [33].

Reactor power	Particulate iodine	Gaseous iodine (I_2)	Non-organic iodine (total)	Organic iodine
900 MWe PWR	4.2E-5	2.5E-7	4.5E-5	4.2E-3
1300 MWe PWR	4.5E-5	2.2E-3	2.2E-3	2.2E-2

The results of this study showed

- a significant decrease of aerosols due to updated modelling of in-containment aerosol deposition and of filtered containment venting system filtration efficiency;
- higher importance of the organic iodine contribution, due to the iodine interaction with containment paints;
- higher contribution for gaseous iodine for the 1300 MWe series, which was due to a lower quantity of silver in the control rods for these reactors [33].

Another study performed by IRSN in 2010 [33] showed a reduction of the organic iodine source term by two orders of magnitude, due to refined models for radiolitic destruction of the organic iodine in the containment. Furthermore, this study showed a significant reduction of the gaseous iodine contribution and an increase of the particulate iodine contribution to the source term released to the environment (see Table 11).

Table 11. Comparison of iodine forms in source term calculations: L2 PSA 2010 / S3 (2000) [33].

Iodine form	Ratio, L2 PSA (2010) / S3 (2000) studies	IRSN L2 PSA 2010 explanation of differences
Particulate iodine (including iodine oxides)	1.8	Higher leak flow rate from the internal containment. No ventilation of the secondary containment. The new modelling of iodine oxide production in L2 PSA increases the quantity of iodine in particulate form.
Gaseous iodine	0.02	Gaseous iodine oxidation (production of iodine oxide in the L2 PSA modelling that was not considered before)

Iodine form	Ratio, L2 PSA (2010) / S3 (2000) studies	IRSN L2 PSA 2010 explanation of differences
Organic iodine	0.01	Radiolytic destruction of organic iodine in the new L2 PSA modelling

Note that the issue of iodine chemistry and behavior under severe accident conditions, and the effect on the source term still involve large uncertainties in the quantification of these processes, and efforts are underway for reducing them [31].

3.3.4 Environmental source term

Estimation of the environmental source term is typically performed using integrated plant response codes, such as MAAP or MELCOR, that can evaluate coupled effects of different aspects of severe accident progression on the source term (accident scenario, phenomena, thermal-hydraulic conditions, etc.). Such analyses are typically performed for a limited set of accident scenarios and subject to uncertainty, due to phenomena, scenarios, and their mutual interactions.

Table 12 presents the overview of the SOARCA study results for Surry PWR [41]. For comparison purposes, a consequence analysis using the old SST1 source term is presented in the table. This allows a direct comparison, using the same modelling options and result metrics, of the SST1 source term and the SOARCA best-estimate source terms.

Table 12. Brief Source-Term Description for Unmitigated Surry Accident Scenarios and the SST1 Source Term from the 1982 Siting Study [41].

	CDF (1/yr)	Xe	Cs	Ba	I	Te	Ru	Mo	Ce	La	Start (hr)	End (hr)
Surry STSBO	2.E-6	0.518	0.001	0	0.006	0.006	0	0	0	0	25.5	48
Surry STSBO w/ TISGTR	4.E-7	0.592	0.004	0	0.009	0.007	0	0.001	0	0	3.6	48
Surry mitigated STSBO w/ TISGTR	4.E-7	0.085	0.004	0	0.005	0.004	0	0.001	0	0	3.6	48
Surry LTSBO	2.E-5	0.537	0	0	0.003	0.006	0	0	0	0	45.3	72
Surry IS-LOCA	3.E-8	0.983	0.02	0	0.154	0.132	0	0.003	0	0	12.8	48
SST1	1.E-5	1	0.67	0.07	0.45	0.64	0.05	0.05	0.009	0.009	1.5	3.5

These results indicate that the unmitigated IS-LOCA accident is the largest in terms of release magnitude, but the release begins at 12.8 hours after initiation of the accident. The release begins at its earliest in the two short-term station blackout (STSBO) scenarios with thermally induced steam generator tube rupture scenarios (TISGTR), only 3.6 hours after the accident initiation, but the magnitudes are very small. The unmitigated STSBO and LTSBO (long term station blackout) scenario releases begin very late in time and have very small release magnitudes [41].

In comparison, the SST1 source term is significantly larger in magnitude, especially for the caesium class, than any of the Surry source terms. Moreover, it begins only 1.5 hours after accident initiation, about 2 hours earlier than the fastest release of the set of Surry source terms. The current understanding of accident progression has led to a very different characterization of release signatures than what was assumed for the 1982 Siting Study [41].

The values presented in this section are not directly applicable to the NPWs, however can be used to provide some valuable insights into the effect of different accident scenarios and mitigative actions on the magnitude of the fission products release to the environment.

4 A.2 – Nuclear-powered warship reactor designs and reference accident scenario applicability

In comparison to commercial power reactors, naval reactor systems are subject to different design criteria [13] regarding

- Size
- Reliability and manoeuvrability
- Mechanical movements and shock
- Xenon-poisoned dead-time of the reactor
- Refuelling period
- Radiation shielding
- Acoustic signature

At the same time, the design of both commercial and NPW pressurized water reactors in some respect trace back to the early reactors developed by Bettis Atomic Power Laboratory and Westinghouse in the 1950s [14], and knowledge on these systems have been passed on both to the UK [15] and France [16]. A guiding hypothesis used in this work is therefore that naval reactors do not use some entirely unknown technology or design principles, but rather that relevant parameters can be estimated or "reverse-engineered" by sound reasoning and available knowledge on commercial reactors, known marine reactors and land-based research reactors.

4.1 Reactor and safety systems design

The known design differences between commercial and naval PWRs (except regarding size) relate to:

- Fuel type (including e.g. enrichment level and chemical composition)
- Integrated design (i.e. combining reactor pressure vessel and steam generator into one vessel) (MN)
- Containment flooding possibilities
- Containment design pressure

Furthermore, safety systems of all nuclear reactors are typically subject to requirements on redundancy and diversification as well as on radiation shielding, basically leading to a relatively space-consuming plant. It is plausible that the limited space aboard a submarine will bring about a different approach in this respect, i.e. radiation shielding for crew protection might have to be prioritized over system redundancy for reactor protection. It is hard to quantify this effect but a naval reactor aboard an aircraft carrier would probably not be equally restricted.

It has been stated that naval reactors in normal operation can be cooled by natural circulation [34]. It is unclear whether this statement concerned the entire system, or the primary circuit only. In any case, also commercial PWRs use natural circulation in the primary circuit at shutdown [36]. As events with failing reactor shutdown can be expected to have very low probability, natural circulation in the primary circuit would therefore in itself not give naval reactors a safety advantage over commercial reactors. (The ability to run a submarine reactor with natural circulation also at significant power is of course advantageous from an operational point of view due to the reduced acoustic signature.)

Some uncertainty is present in the documentation [1][34] regarding whether naval reactors are equipped with an emergency core cooling (ECC) system or not. The answer to this question probably lies more in the definition of such a system than in the actual presence of it. Typically, the role of such a system is to provide additional water to the core in the short term, following a LOCA accident, and this can be achieved in several different ways. It is judged unlikely that naval reactors would have no such capability.

In the long run, also for accident conditions, an equally important function is assured by the residual heat removal (RHR) systems. For commercial plants, at normal operation, heat is removed through the steam generators, turbine and condenser, while at shutdown, a dedicated RHR heat exchanger is typically activated. Being able to run an ECC system to provide the reactor with additional coolant in case of a LOCA is of little use in the long run if the residual heat cannot be removed from the reactor compartment. Since the submarine is immersed in its own heat sink, it is probable that the ship is equipped with one or several RHR systems to be used when the secondary circuit and turbine is not in use, as indicated by [37]. The existence and use of such systems, including to what extent they may function passively, are important to the understanding of risks for an NPW in port.

In addition to the supposed improved RHR capability, an important difference-in-principle between naval and commercial land-based reactors from a safety design perspective would be the containment design pressure, which is substantially higher for naval reactors. Design pressures of around 20 bar have been mentioned [34]. If this is true, it represents a significant safety advantage in terms of the risk of large releases, however not in terms of the risk of core damage.

An important part of reactor safety is operator training and man-machine interaction. There is some evidence in the literature for an assumption that naval reactor operation follows a very strict organizational scheme and that operators have a simpler system to operate compared to their counterparts in a commercial plant [35]. As human error is typically an important cause in many known accidents, this fact might have a significant impact on NPW reactor safety.

4.2 Reactor power, fuel type, core inventory

4.2.1 Reactor power

An important input value when estimating the potential radioactive releases from an NPW reactor is the reactor power. Since NPW reactors are known to run well below full power for a relatively large part of the time and the navy has interest to conceal the true maximum speeds of these vessels, there is (a priori) a considerable uncertainty in the stated reactor power values of these ships. In order to evaluate the extent of this uncertainty, reactor power and displacement for each submarine class, based on open sources [45][37], is studied. This exercise shows that there are submarine classes that have both relatively large and relatively small reactor powers compared to their mass, c.f. Table 13 and Figure 3.

Ship classes and reactors studied in this report are listed in Table 13 together with stated numbers on reactor power and displacement, based on [45] and [37].

Table 13. Studied classes, reactors and stated numbers on reactor power and displacement.

Class	Reactor	Reactor power [MW]	Displacement [tons] (submerged for submarines)
Los Angeles	S6G	165	6927
Ohio	S8G	220	18750
Seawolf	S6W	200	9138
Virginia	S9G	210	7900
Trafalgar		78	5250
Vanguard	PWR2	145	15900

Class	Reactor	Reactor power [MW]	Displacement [tons] (submerged for submarines)
Astute	PWR2	145	7600
Rubis	K48	48	2600
Triomphant	K15	150	14335
Barracuda	K15	150	5300
Nimitz	2 x A4W	2 x 550	~100 000
Gerald Ford	2 x A1B	2 x 700	~100 000

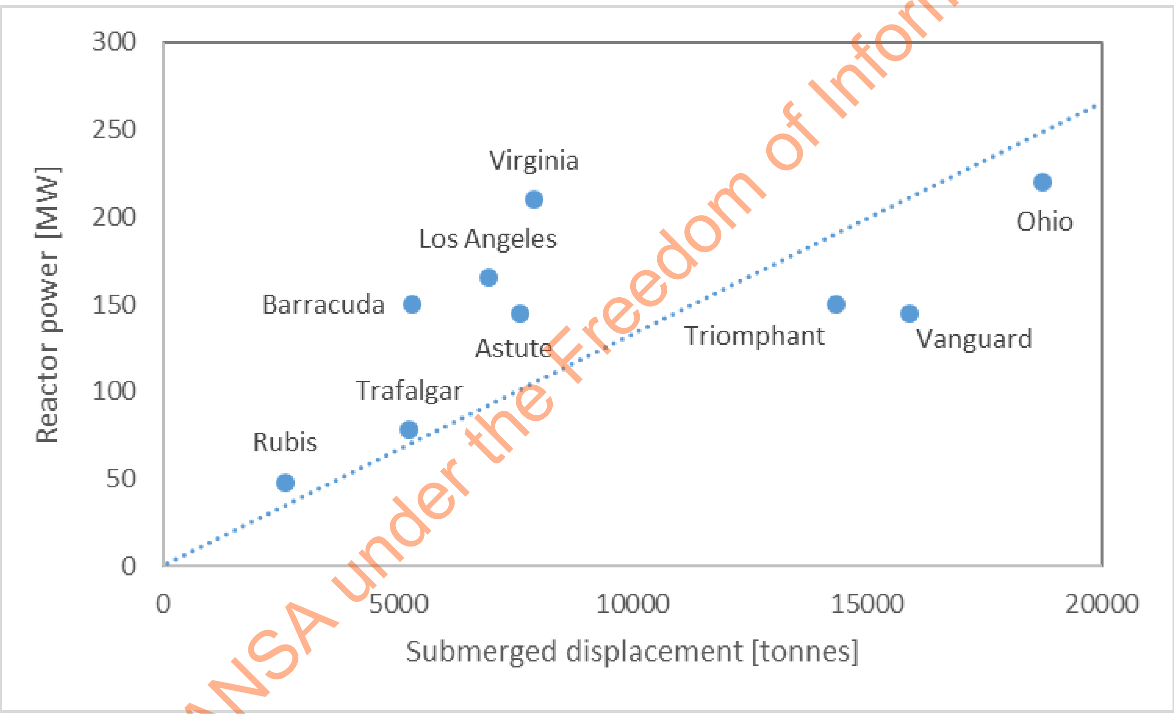

Figure 3 Reactor power versus submerged displacement for studied submarine classes.

An alternative assumption for the evaluation of stated reactor power values is that all submarines have similar maximum speed. For an object with cross-section area A and drag coefficient C_d which moves with velocity v in a stationary fluid of density ρ, the power P_d needed to overcome the drag force is given by

$$P_d = \frac{1}{2}\rho v^3 A C_d \tag{8}$$

By rearranging and, for simplicity, taking the cross-section area proportional to the square of the beam width D it is possible to define a constant K_{vmax}, here denoted the *maximum speed multiplier*, as

$$v \sim \sqrt[3]{\frac{P_d}{D^2}} = K_{vmax} \tag{9}$$

By calculating this number for the submarine classes and normalizing to the average result, a value with lower spread and without trend with respect to displacement is indeed found, c.f. Figure 4. Calculated values of the maximum speed multiplier are given in Appendix A.

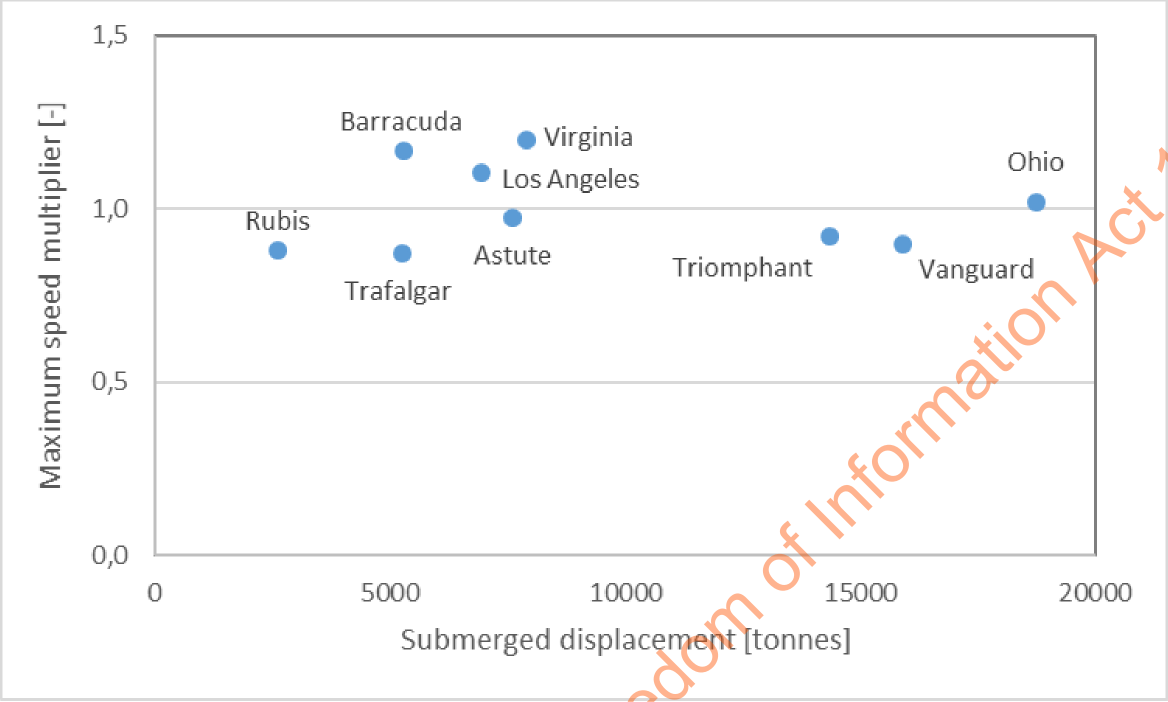

Figure 4. Maximum speed multiplier for studied submarine classes.

If the full spread seen in Figure 4 (about ±20 %) would be only due to uncertainty in reactor power, from equation (9) it follows that the corresponding relative reactor power range would be about (0.8^3 ; 1.2^3), i.e. a spread of -50 %, +70 % in reactor power, which is a considerable uncertainty. However, as the true maximum speed of these ships will depend on other properties that are more complicated to estimate, and thereby easier to conceal, such as the drag coefficient, the need for onboard power, turbine plant and propeller efficiency, acoustic properties etc., it is reasonable to assume that these properties account for a large fraction of the spread and that the official reactor power values are more or less the true values.

Based on this reasoning, we would suggest that a reasonable estimate on the uncertainty in reactor power for these ships would be ±20 %, with Figure 4 providing a reasonable guess of the direction of uncertainty for some of the submarine classes.

4.2.2 Fuel and core design - typical burnup

One of the main reasons for using nuclear power for propulsion in a submarine is the ability to run the ship for very long times without refuelling. To this end, the US and Royal Navies use high enriched uranium (HEU, >90 % U-235) in the fuel [13][15], opting for very long refuelling intervals. The French navy has instead opted for reactor designs using low-enriched uranium (LEU, <20 % U-235), with shorter refuelling intervals and ease of refuelling taken into account in the ship design [16].

To assess the source term of a nuclear reactor, the core inventory of radionuclides needs to be estimated, for example along the lines of the analysis presented in section 3.1. This inventory can be primarily expected to correlate with reactor power and the so-called burnup level. The absolute burnup level is basically a measure of the number of split heavy atoms in the core from the operational history, typically measured in MWd. In view of the large uncertainties in reactor design and long refuelling intervals, the burnup level might be one of few possible benchmark numbers for core inventory estimation which can allow some uncertainty reduction for a given ship, e.g. regarding possible releases of long-lived fission products such as Cs-137 and Sr-90.

It is possible to make estimates of burnup based on reactor power, typical use and load factors as well as the total mass of uranium in the reactor. To do this, a stepwise benchmarking exercise was undertaken, based on scaling and evaluation of critical core design parameters. This benchmarking was performed according to the steps described below:

Step 1

Data for a typical Westinghouse commercial PWR were assembled, based on [36] and [37], and used to deduce:

- Ratio of core to primary vessel volumes
- Ratio of fuel to core volumes
- Core power density (benchmarked against [36])
- In-fuel power density
- Thermal neutron flux (value from [39])

Step 2

Data for the Russian KLT-40S reactor were assembled, based on [38], and used to deduce:

- Ratio of core to primary vessel volumes
- Fuel mass (benchmarked against [38])
- Core power density (benchmarked against [38])
- In-fuel power density
- Thermal neutron flux (scaled from commercial PWR based on the in-fuel power density)

For this reactor, the ratio of the fuel to core volume was not known but its' value could be tuned to benchmark the fuel mass value.

This step is important as it allows a first comparison between a commercial and a marine PWR. It seems that the smaller marine reactors will need to accommodate a larger thermal neutron flux and in-fuel power density, but that (at least the KLT-40S) has about the same core power density as a commercial PWR. The resulting thermal neutron flux is not extreme, e.g. compared to research reactors [46].

Step 3

Data for the KLT-40 reactor as used with HEU fuel in the Sevmorput vessel were assembled, based on [40], and used to deduce:

- Fuel mass (benchmarked against [40])
- Core power density (benchmarked against KLT-40S and commercial PWR)
- In-fuel power density (benchmarked against KLT-40S)
- Thermal neutron flux (benchmarked against KLT-40S)

For this reactor, the fuel density was not known, but an estimated value for HEU fuel of 1300 kg/m^3 based on sources such as [17] and [18] was used.

Step 4

Finally, data for the naval reactors studied for ARPANSA were assembled according to the following:

- Reactor power values as stated in open sources and assessed in Section 4.2.1 were used.
- Typical values for fuel density and enrichment of HEU fuel (USN and RN) and LEU fuel (MN) were estimated based on [13], [16], [17] and [18].

- Reactor geometry data was taken similar to the KLT-40S and KLT-40 reactors, slightly tuned to avoid too high deviations in fuel mass, thermal neutron flux, in-fuel power density and core power density compared to the more well-known reactors of Steps 1-3.

This exercise leads to a tool that can be used to estimate typical values and uncertainty in burnups of NPW reactors, given that the time of refuelling is known for each ship. Snapshots of the tool are given in Appendices B and C. The tool shows that the absolute burnup level varies considerably, more than two orders of magnitude, between newer ships with smaller reactors and older ships with larger reactors. To illustrate this, a Monte-Carlo uncertainty simulation of possible naval reactor designs and resulting burnup can be performed. An example for the largest reactor, the S8G, used in the Ohio class submarine is given in Figure 5. From this it can be deduced that a typical burnup uncertainty based on design uncertainty for a vessel with *known* time of refuelling is on the order of ±50 %.

Table 14. Input to Monte-Carlo simulation of burnup levels in the S8G reactor.

Parameter	Mean	Standard deviation
Thermal power [MW]	220	10 %
Use factor [-]	0,5	20 %
Load factor [-]	0,25	20 %

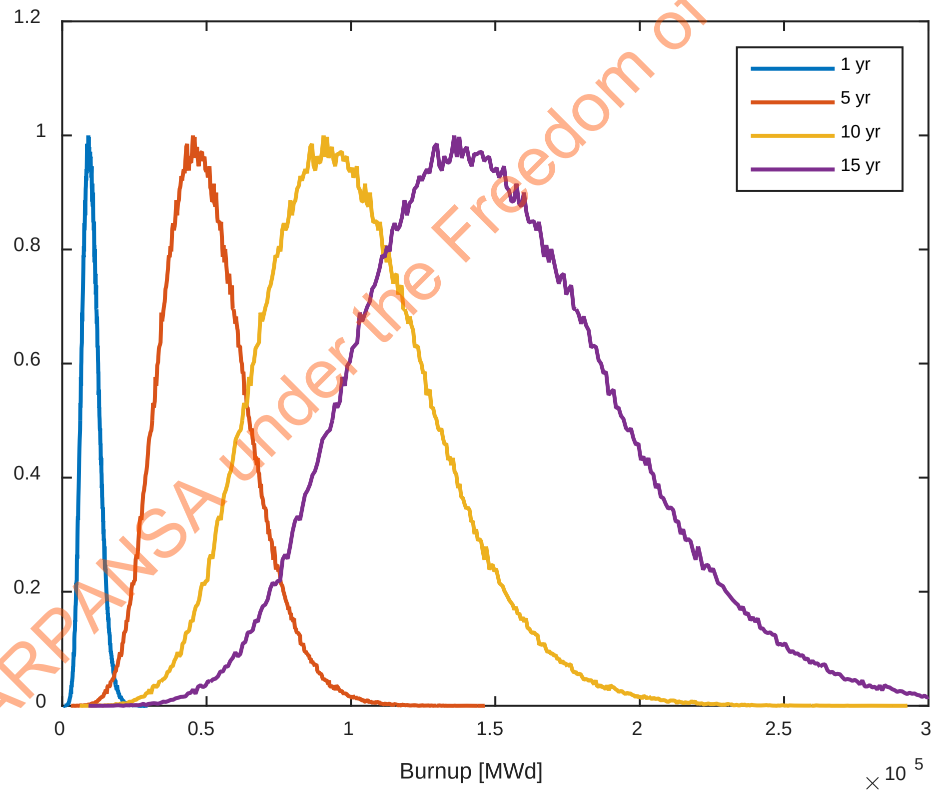

Figure 5. Monte-Carlo-simulated burnup levels for the S8G reactor.

4.2.3 Core inventory

Exact core inventories cannot be calculated without information on the detailed design and operation history. Attempts to estimate inventories of naval reactors using relatively advanced tools have been made [40]. It is perhaps possible to scale such inventories, using the tool described in the previous section. The result will, naturally, not be an exact number for the vessel in question, but it will give a more nuanced picture compared to an approach where the same inventory is used for all reactors of approximately the same size.

4.3 Initiating events, sequences, consequences

A probabilistic safety analysis (PSA) of a nuclear plant aspires to give a comprehensive picture of all event combinations that can lead to core damage. A typical PSA model does this by structuring the set of event combinations into *sequences* which start with an *initiating event*, goes through a number of failed or successful *function events* and finally leads to a specified *consequence*. Both the evaluation of function events into failed or successful, as well as the attribution of consequences, will be supported by detailed deterministic simulations of the entire reactor system, using integrated codes such as e.g. MAAP or MELCOR.

A level 1 PSA is an analysis where the consequences refer to the core status, categorized by so-called *plant damage states*, whereas a level 2 PSA continues from this point and goes on to consequences referring to radioactive release status. Such radioactive release consequences are typically grouped into so-called *release categories*. Each release category will typically be the end point of several, but phenomenologically similar, sequences and can be associated with one or several typical source terms. The typical source term can be selected in several ways, for example by simulating the sequence with the highest frequency or the sequence with the most severe consequences within the release category.

As has been pointed out in other works, it is evident that a PSA cannot be performed for a system where most of the detailed design is unknown. However, also on this topic it is instructive to make comparisons to known results of commercial reactors of the elements that are known or estimated in the literature. Such a comparison naturally also must take into account the known differences between the systems.

The set of relevant initiating events and safety functions can, a priori, be expected to be more or less the same as in a commercial PWR. This implies that the radioactive material is protected by two or three barriers; the fuel cladding (if present), the primary system and the containment. In addition to these, the submarine hull itself can be credited for some retention and filtering of fission products, with the possible exception of sequences activating atmospheric venting paths from the containment or secondary system.

After an initiating event with successful reactor shutdown (the normal shutdown procedure is regarded as an initiating event in itself), the next functions typically called upon in a PWR are pressure and level control. If pressure or level control fails, due to the initiating event being a transient or a LOCA, the next function called upon will be some kind of high- or low-pressure safety injection with the aim of supplying emergency core cooling in the short-term. In addition to this, in order to ensure long-term cooling, some kind of residual heat removal (RHR) circulation needs to be established. For transients and smaller LOCAs, RHR through the normal steam generators or dedicated RHR heat exchangers are typically used, calling upon secondary system functions. For larger LOCAs, recirculation from the containment sump can be used. In this case, cooling of the sump needs to be achieved. It is possible that naval reactors in general can achieve RHR easier than commercial reactors, simply by cooling certain circuits or compartments directly with seawater, possibly using natural circulation. Furthermore, it is plausible that the reactor, at least on a submarine, would be shut down while in port, which means that some smaller RHR system will be used instead of steam generators.

If core cooling fails during a sufficiently long time, the reactor will experience some level of core damage. To create a major release of radioactive materials, both the primary system and the containment will need to fail or be bypassed.

A high containment design pressure does not in itself reduce the risk of large releases, due to the existence of so-called bypass sequences. These are sequences including leaks or breaks in pipework connected to the primary system which penetrates the containment boundary. Typical examples for commercial PWRs include SGTR and IS-LOCA. For naval reactors, leaks or breaks in the RHR heat exchanging system would also be a possibility. It is reasonable to assume that such a system is cooled by seawater and that such events would then create a direct bypass to the sea.

In case of a LOCA, the primary system is by definition failed already on the initiating event. If the initiating event is not a LOCA, then failure or opening of the primary circuit can happen only by opening of depressurization valves, or by later induced LOCAs (including reactor vessel melt-through).

It is also possible that a submarine would be able to flood the containment, thereby ensuring cooling and protection of the primary system. However, western submarines are known to have a relatively small buoyancy margin which means that reactor containment flooding would probably not be easily achievable through a dedicated system due to the risks associated with inadvertent activation when submerged.

It is also possible that the containment can withstand the overpressure more or less irrespective of the primary system status.

Initiating events, functions and consequences in PSA can be represented at a high level using so-called block diagrams. An attempt to visualize the possible differences between commercial and naval PWRs from a PSA perspective is shown in Figure 6. For sequences with core damage, with the core still inside the reactor pressure vessel (RPV) and with an intact containment, the release category is denoted "DIFFUSE LEAKAGE 1. If in-vessel retention (IVR) fails, the release category is denoted DIFFUSE LEAKAGE 2.

Sequences ending with a deliberately vented containment due to a need for depressurization lead to a release category denoted VENTING. If this fails, the result will be a release category denoted CONTAINMENT RUPTURE.

Release categories for sequences ending with a bypassed containment due to RHR heat exchanger (HX) rupture, IS-LOCA or SGTR are denoted BYP-RHR-HXR, BYP-ISLOCA and BYP-SGTR respectively.

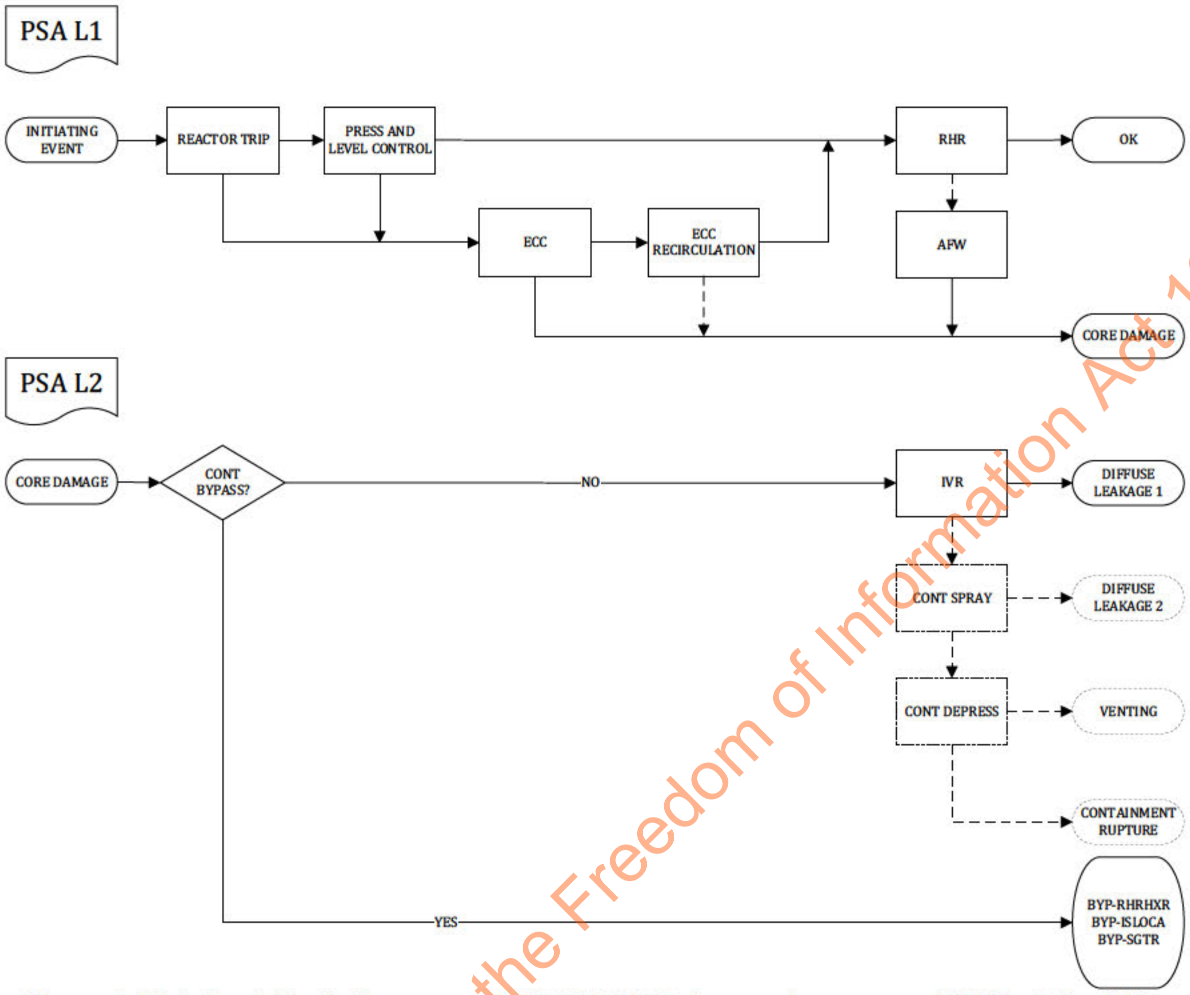

Figure 6. High-level block diagram of PWR PSA L1&2. Areas where a naval PWR might differ significantly from a commercial PWR are marked with dashed lines.

When quantifying results of a PSA, initiating events are given frequencies, and the success/failure of functions are given probabilities. The result is typically expressed in frequencies of certain groups of consequences, the two most common ones being the core damage frequency (CDF) in PSA L1 and the large release frequency (LRF) in PSA L2. Not much is known about these numbers for naval reactors. The British government has stated that the frequency of a contained meltdown in a naval reactor is "no greater than" 10^{-4} per year and that the frequency of an uncontained accident is "no greater than" 10^{-6} per year. These numbers are however no more enlightening than the fact that they correspond to fulfilment of standard safety goals. For calculated CDF in commercial reactor designs, the International Nuclear Safety Group (INSAG) proposed (in 1999) that the safety goal should be less than 10^{-4} per reactor year for existing plants and less than 10^{-5} for future plants [42]. For calculated LRF in existing plants, INSAG proposed that this should be a factor 0.1 smaller than the core damage frequency.

Some observations from modern power and low power operation PSA of commercial PWRs:

- Sequences that start with fires and transients (excluding primary circuit LOCAs) account for an important part of the CDF.
- Sequences that start with primary circuit LOCAs account for a smaller part of the CDF.
- Typical LOCA initiating event frequencies lie around 10^{-4}–10^{-3} [43].
- Sequences with SGTR represent an important part of the LRF.

Aboard a submarine, fire extinction will always have a very high priority, with requirements to be able put out fires within a few minutes. In addition to this, if NPW reactor systems are indeed simpler to operate and can achieve RHR in a less complex way, there is rationale to assume that sequences starting with LOCA are a more important part of the core damage frequency for naval reactors, at least on submarines. The case might be different for aircraft carriers.

We can now broadly sum up our knowledge and judgement of risks in NPW reactor systems in port, relative to commercial reactor systems. We will qualify these differences in terms of the risk of a large radioactive release, as defined by impact on frequency and consequence.

Table 15. Relative risk judgement for visiting NPWs compared to commercial plants.

	Impact on		
Item	Frequency	Consequence	Risk
Naval reactors have lower power	No change	Decreased	Decreased
Visiting naval reactor systems may be sited closer to populated areas	No change	Increased	Increased
Naval reactors (US and UK) have different fuel	No change	Uncertain	No change
Naval reactor systems are operated differently	Increased	Uncertain	Increased
Naval reactor containments, at least on submarines, have substantially higher design pressure	Decreased	No change	Decreased
Naval reactor systems on submarines can probably achieve RHR easily	Decreased	No change	Decreased
Possible presence of seawater-cooled RHR system for use in port	None	Increased for release to water	Increased
Naval reactor systems are only present at visits	Decreased	No change	Decreased

Disregarding the obvious conclusion that sporadic visits of NPWs represent a lower risk compared to a constantly present commercial reactor, and the fact that the relative importance of the items in Table 13 is difficult to quantify, the remaining conclusions are:

- For a visit close to a populated area, the lower reactor power will be partly or fully offset by the decreased siting distance.
- The fuel types of naval reactors are not judged to have a major impact on risks.
- The operational uncertainty and large dependency of the source term on burnup creates a major uncertainty with regards to emergency preparedness and response, especially for carriers.
- Submarine reactors are judged to be somewhat more robust than commercial reactors in terms of RHR capability and containment design pressure.
- The possible presence of a seawater-cooled RHR system for use in port introduces a unique risk for releases to water.

5 A.3 – Assumptions and limitations of the ACCIDENT code

The review of the models and modelling approaches used for estimation of the source term in the ACCIDENT code has been performed. The review comments are presented in Table 16.

Table 16. Review summary of the source term models in the ACCIDENT code.

Parameter/ Equation	ACCIDENT Reference	ACCIDENT Assumption	Reference in this text	Comment/Recommendation	Suitable/recommendation
-	2.2.	-	-	Chapter 2.2 should be revised.	Description of initiating events and categorization should be revised, based on internationally accepted standards, e.g. "cooling water pump failure (loss of flow)" is not considered as LOCA. For more details and examples, see IAEA-TECDOC-719: Defining initiating events for purpose of probabilistic safety assessment.
Fission product inventory – equation	3.1.1 Eq. A.1.1.1	Fission product inventory is based on many factors, including yield of U-235 fission, operating power, time prior to shutdown, irradiation time etc.	3.1 Appendix	-	Suitable. The fuel burn-up for the vessel could be considered to replace the 'average' operating time in order to provide the activity of fission products. Consider confirmatory calculations with high fidelity models (such as ORIGEN2) for limiting conditions.

Parameter/ Equation	ACCIDENT Reference	ACCIDENT Assumption	Reference in this text	Comment/Recommendation	Suitable/recommendation
Fission product inventory – fission yield	Table B.1	The cumulative yield used is from [6] in [1].		-	The yield should be reviewed in comparison to updated yield databases such as ENDF etc.
Fission product inventory – fission rate per thermal power	Eq. A.1.1.1	Value of 3.125 x 10^16 (fissions s^{-1} MW^{-1}) is given	3.1	-	Value should be updated based on the mean fission energy for U-235 (202.5 MeV).
Nuclide groupings	3.1.1	ACCIDENT considers 58 nuclides. Nuclides have been placed into 9 groups based on their relative volatility.	3.2.1 Table 2	The nuclide grouping for core release fraction were reviewed.	Suitable.
Containment deposition	Eq. A.1.2.2	The containment time constant is based on the nuclide specific deposition velocity and the area of containment relative to the volume.	3.3	-	Suitable. The model for aerosols deposition in the containment can be updated based on the approaches proposed in the (time dependent modelling) NUREG-6189 [24].
Atmospheric release	Eq A1.2.4.	No modelling of aerosols deposition in penetrations, fittings, seals, etc.	3.3.2	Aerosols deposition in case of diffuse leakage from the containment.	Current modelling is suitable (conservative). The model can be extended to consider deposition of aerosol FPs in the penetrations, fittings, seals, etc. in case of diffuse leakage from the containment (with intact containment).

Parameter/ Equation	ACCIDENT Reference	ACCIDENT Assumption	Reference in this text	Comment/Recommendation	Suitable/recommendation
Core release timing	Eq. A.1.2.5	Fission products are effectively released from the core instantly, with the magnitude based on the core activity and the release fraction	3.2	-	Suitable (conservative). Release kinetics are hard to estimate without detailed knowledge about thermal-hydraulic response of the system. Note that release kinetics can be somewhat different for metallic fuels, although cumulative release fractions are expected to be comparable with those of oxidic fuels. Consideration of timing during different phases of FPs release to the containment can be considered as a sensitivity analysis.
Iodine chemistry	3.1.2.2	Only two forms of iodine are considered: organic and aerosol	3.3.2	Iodine is generally released from the core in three chemical forms: organic, gaseous and aerosol.	Gaseous iodine should also be included. The effect of variability of release fractions, as well as deposition velocities for different iodine forms can be considered in sensitivity and uncertainty analysis.

6 A.4 – Assumptions and limitations of the reference accident scenario

The review of the assumptions and limitations made in 2000 NPW Reference Accident Assessment has been performed. The review comments are presented in Table 17.

Table 17. Review summary of the source term estimation in the 2000 NPW Reference Assessment.

Parameter/Equation/Assumption	Reference accident ref	Reference accident assumption	Reference in this text	Comment	Suitable/recommendation
Accident sequence	2.1	Contained LOCA	4.3	Possible initiating events and accident scenarios: Bypass cases (SGTR, IS-LOCA or RHR heat exchange tube rupture) are not considered.	Consider other accident sequences presented in this work. There may be sequences which bypass the containment or result in releases to the aquatic environment.
Maximum thermal power	3.1.1	Carriers – 600 MWt Submarines – 160 MWt	4.2.1 Appendix		Consider the values presented by this work.
Time at maximum power	3.1.3	4 days prior to accident	3.1	-	Time for which reactor was operated at maximum power prior to the accident can be revised based on the saturation time for the FPs with short half-lives in terms of the irradiation time. Furthermore, sensitivity and uncertainty analysis is suggested to assess the effect of the time reactor has been shut down prior to the accident.

Parameter/Equation/Assumption	Reference accident ref	Reference accident assumption	Reference in this text	Comment	Suitable/recommendation
Time at average power	3.1.1	15 years. Linked to the refueling cycle.		The uncertainty analysis examples presented in this work suggest that the reactor design uncertainty in itself might yield a burnup equivalent to an additional 10-15 years at average power.	Consider how much design uncertainty conservatism to take into account, if any.
Time since refueling	3.1.2	15 years. Full core replacement	4.2.2 Appendix	The burn-up time could be use used to establish the fission product inventory in the time before the NPW reaches the port.	Consider requesting the burn-up of the reactor, or the time since refueling, or a limiting value as condition for entry.
Containment release fractions	Table 3.2.2	Based on studied from NZ and Canada	3.2.1	Release fractions presented in Table 3.2-2 [2] are in agreement with respective mean and median values suggested in the NUREG-5747 (see Low pressure scenario with High and Low Zr oxidation in Table 4), on the other hand these values are below the bounding estimates for I, Cs, Te groups presented in Table 5 (see the values for low pressure scenario).	Suitable. Consider bounding assessment and analysis of uncertainty.
Iodine chemistry	3.2	98 % is aerosol and 2 % is organic	3.3.3	Recent studies showed that the particulate and gaseous iodine fractions released to the containment strongly depend on the type and materials used	Gaseous iodine form should be included. Furthermore, since the issue of iodine chemistry and behavior

Parameter/Equation/Assumption	Reference accident ref	Reference accident assumption	Reference in this text	Comment	Suitable/recommendation
				in the control rods; Formation of organic iodine forms depend strongly depend on the presence of organic materials (paints, cables) in the containment, as well as radiation.	under severe accident conditions is still involve significant uncertainties, it is suggested to complement current study by additional sensitivity and uncertainty analysis. The values presented in Section 3.3.3 and in [32] and [33] can be used as indicative to support selection of the ranges of uncertain parameters. Parametrization can be made based on the fractions for organic/inorganic forms, where inorganic forms can be further subdivided into particulate/gaseous forms.
Containment surface to volume ratio	Section 3.3.3	1.2 (m^{-1})	-	Containment surface to volume ratio can be a very important parameter for the aerosols' deposition on the containment surfaces. Since it is very hard to estimate this parameter without detailed knowledge of the system, it is suggested to estimate the effect of variability of this parameter on the outcome via sensitivity and uncertainty analyses.	Suitable. Complement by sensitivity and uncertainty analysis.
-	Section 3.3.	-	3.3.2	Based on the, some additional aerosols retention in the	Current modelling is suitable (conservative).

Parameter/Equation/Assumption	Reference accident ref	Reference accident assumption	Reference in this text	Comment	Suitable/recommendation
				cracks/openings/pipe fittings, etc. (especially under wet conditions) is possible. Experimental programs for concrete containments suggest the values of DF10-100 [10][30]	Additional decontamination factor (DF) can be implemented as a sensitivity coefficient in the ACCIDENT code and used for sensitivity analysis and quantification of uncertainty.
Leak rate – primary containment	3.3.1	1 % per day	-	-	Suitable (the value is 10 times larger the typical value for land based commercial NPPs under design pressure). Note that the value is not applicable for the containment bypass scenarios.
Leak rate – secondary containment (vessel hull)	3.3.1	10 % per day	-	-	Suitable (conservative, given the values provided by ref [39] in [2].
-	-	-	Section 4	Burnup uncertainty is large, especially for carriers. It is suggested to perform uncertainty analysis to evaluate the effect of burnup uncertainty.	Uncertainty analysis
-	Chapter 2.1.	-	-	The Canadian technical safety assessment (TSA) referred in the reference accident description states that “the total core damage frequency is dominated by primary system failures giving rise to LOCAs”	Based on the simplified block diagrams presented in this report, a credible reason for the assumption that LOCAs dominate the CDF of naval reactors, would be that RHR can be achieved easier for a system that is

Parameter/Equation/Assumption	Reference accident ref	Reference accident assumption	Reference in this text	Comment	Suitable/recommendation
				[1]. This corresponds well to earlier views on reactor safety [44].	immerged in its heat sink, possibly crediting natural circulation. Using a sequence starting with LOCA as the reference accident scenario for EPR with diffuse leakage is justified from a timing perspective since it gives an early breach of the primary circuit.
	Chapter 2.1.			"*radioactive fission products are assumed to be released from the reactor core into the containment surrounding the reactor under the driving force of the high primary circuit pressure*" [1]. It is also slightly confusing to refer to a driving force of high primary circuit pressure after a LOCA (which in itself will depressurize the primary system).	Revise text.
			Section 3	Although not entirely clear, the Australian reference accident scenario seems to consider the in-vessel melt retention case (IVMR).	If no recovery of ECC is credited and it since is not certain that flooding of the reactor containment would be used, a clear motivation as to why a case with the core still inside of the primary vessel is used should be

Parameter/Equation/Assumption	Reference accident ref	Reference accident assumption	Reference in this text	Comment	Suitable/recommendation
				Ex-vessel phenomena can contribute to the containment source term as well as release paths to the environment (e.g. containment leakage rate or failure of the containment) and environmental source term.	given. Alternatively, change the reference scenario to one with reactor vessel melt-through and respective ex-vessel consequences. Furthermore, the implications of IVMR assumption on the need for reactor containment flooding and thereby on free gas volume and deposition inside should be stated.
-	-	-	Section 3	No consideration of ex-vessel phenomena.	Clear motivation why ex-vessel phenomena are not considered in the analysis should be given, alternatively, ex-vessel phenomena can be considered
-	Section 6.2.2. Page 48	-		According to ARPANSA, the vessel removal time will be defined on a port-specific basis and its starting point will probably be set by the first indication of an anomaly at the ship. There is a considerable uncertainty in what this first indication will be. At the same time, different accident sequences can have very different evolutions in time. This means that the vessel removal time probably cannot be aligned to the accident	Vessel removal planning should, as much as possible, be made independent of release characteristics.

Parameter/Equation/Assumption	Reference accident ref	Reference accident assumption	Reference in this text	Comment	Suitable/recommendation
				progression and that vessel removal planning therefore should, as much as possible, be made independent of release characteristics.	

7 Conclusions

The present report gives a brief overview of the current state of the art in assessment of the source term released to the environment, NPW reactor designs and the reference accident applicability, as well as an overview of the ACCIDENT code and the NPW reference accident scenario.

Based on the review the following main recommendations can be made:

- Clarify in-vessel / ex-vessel melt scenario as well as the primary system pressure status and use applicable assumptions in the ACCIDENT code.
- Accident frequency consideration – Visit frequency. As frequencies for uncontained accidents in NPWs have been estimated to be on the order of 10^{-7} to 10^{-6}, which represent the typical safety goals for LRF of INSAG, this estimate is in itself not a strong argument for not considering uncontained accidents in EPR for NPWs. There might however be other arguments for not doing so, based on the magnitude of the possible source terms and the (risk) exposure time, since the vessels have small reactors and are visiting for a short time only.
- Accident consequence consideration – large uncertainty in burnup. It is suggested to perform uncertainty analysis, using e.g. Excel tool for burnup calculations, or, alternatively, reduce burnup uncertainty by requesting e.g. a limiting value in MWd as condition for entry.
- Containment bypass scenarios, such as SGTR or passive RHR HX rupture. No rationale for ruling out the importance of bypass sequences for naval reactors has been found. Therefore it is recommended to include such sequences as an alternative scenario for EPR. Such sequences might also more often result in releases to water.
- Given the current state of modelling in the field of iodine behaviour in the containment under severe accident conditions, it is recommended to perform sensitivity and uncertainty analysis of the effect of the fraction of organic/elemental iodine on the EPR.

The recommendations listed above are based on the review comments presented in the respective sections of the report. The complete list of comments and suggestions can be found in sections 5 and 6.

Furthermore, the following general recommendation can be made to enhance the methodological aspects of the NPW reference accident scenario. It is recommended to perform iterative sensitivity analysis and uncertainty quantification. Sensitivity analysis helps to identify the ACCIDENT code modelling parameters that have significant impact on the results (in terms of EPR), while uncertainty quantification will show the effect of the variability in these parameters on the code response (in terms of EPR). This information can be used to support decision making, as well as give valuable insights regarding the knowledge gaps, and highlight the models and modelling parameters where reduction of uncertainties will be the most valuable (see [48][49][50] for more details).

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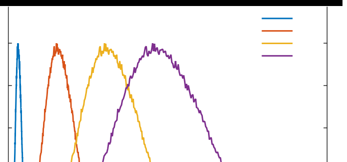

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