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Determining the source term for a hypothetical nuclear submarine using “ORIGEN” and “ACCIDENT” codes

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Emneord

Kildeterm for atomberedskap, Ubåter, Analytisk analyse for atomulykker

Resymé

Dokumentet gir en kort beskrivelse av alvorlige atomulykker analyser i ubåter. Radionuklide kildeterm i reaktorkjerne og dens utslipp til miljøet som følge av atomulykke er analysert ved bruk av to metoder «ACCIDENT» og «ORIGEN». Resultatene fra to metoder er analysert og presentert i dokumentet.

Reference

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Key words

Radionuclide reactor core inventory, Radionuclide source term for emergency preparedness, nuclear submarine, analytical analysis for severe nuclear accidents

Abstract

The severe nuclear accident methodology is briefly described. The determination of radionuclide inventory in a reactor core and its release to the environment because of severe accident has been described for ACCIDENT and ORIGEN codes. The codes were applied to a hypothetical submarine accident and results were compared.

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Determining the source term for a hypothetical nuclear submarine using “ORIGEN” and “ACCIDENT” codes

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fra

Direktoratet for strålevern
og atomsikkerhet (DSA)

Østerås, 2024,
Norway

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LIST OF ABBREVIATIONS:

ACCIDENT - Nuclear Accident Consequence Assessment Code
ARPANSA – Australian Radiation Protection and Nuclear Safety Agency
BWR – Boiling Water Reactor
ENDF - Evaluated Nuclear Data File
FP – Fission Products
LBE – Lead Bismuth Eutectic
LOCA – Loss of Coolant Accident
LWR – Light Water Reactor
MCCI - Molten Corium Concrete Interactions
NPW – Nuclear Powered Warship
NRC – United States Nuclear Regulatory Commission
ORIGEN - The ORNL Isotope Generation and Depletion Code
ORNL – Oak Ridge National Lab
RCS – Reactor coolant system
SFR – Sodium Fast Reactor
VPD – Volume per day

1 Introduction

This document is a description of source term calculation for a nuclear-powered warship (NPW). Nuclide source term refers to the magnitude and mixture of the radionuclides released from the fuel, expressed as fractions of the fission product inventory in the fuel, as well as their physical and chemical form, and the timing of their release. The source term defines the release parameters (radionuclides type, quantity, location, energy, and timing) relevant for the dispersion calculation and consequence assessment. Fission products are formed in the nuclear fuel mainly through nuclear fission. Other radionuclides are also formed in the fuel due to other nuclear reactions like neutron pick-up reaction. The release of these radionuclides from the fuel and reactor to the environment, may only occur on the deterioration of physical barriers of the nuclear reactor: fuel matrix; cladding; boundary of reactor coolant system; and reactor containment. To determine the radionuclide inventory of the nuclear fuel, two independent calculation methods has been used. The two methods ACCIDENT (ARPANSA code to model the consequences of 2000 Reference Accident) [1] and ORIGEN (Oak Ridge Isotope Generation code) [2] codes are developed by ARPANSA and Oak Ridge National Labs (ORNL). The ACCIDENT code [1] is a Microsoft Excel based code, developed by ARPANSA to find the source term for a NPW reference accident scenario [3]. ORIGEN [2] on the other hand is a high-fidelity code, developed for calculating the build-up, decay, and processing of radioactive materials. For the determination of the in-containment and atmospheric releases from a submarine, the ACCIDENT code methodology is used. The purpose of this document is to give a qualitative description of the different phases of a nuclear accident progression and briefly describe the theoretical background and main assumptions made in both the methods for its usage in emergency preparedness and response. Additionally, a case study of applying the methods on a hypothetical nuclear submarine with hypothetical reactor parameters to determine the source term in a reference accident scenario.

It is important to note that estimation of radiological releases is performed through a simplified approach. Several reactor parametric assumptions have been made in the analysis that may incur large uncertainties. However, no quantitative uncertainty analysis has been conducted in this work.

2 Overview of the Severe accident consequence assessment

An overview of the potential release of radionuclides from a nuclear reactor as a result of “severe nuclear accident”, is briefly discussed below.

The assessment of the environmental source term – release of radionuclides into the environment, following a severe accident is performed using following steps:

A. Estimation of the inventory of fission products in the core.

B. Estimation of the fraction of fission product released from the core.

C. Estimation of fraction of the source term into the containment.

D. Estimation of the in-containment source term by identifying the impact of the retention mechanisms in the containment.

E. 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 light water reactors (LWRs) and discuss possible relevance to the NPWs.

2.1 Fission product inventory

The production of radionuclides and the variation in their inventory during and after reactor irradiation are governed by the Bateman’s equations (see eq. (1)) [4]. This system of equations is typically solved numerically by using specific computer codes, such as ORIGEN [2], developed by the ORNL.

The inventory of fission products in the reactor core, as per US Nuclear Regulatory Commission (NRC) regulations [5], should be based on the maximum full-power operation of the core. However, the NPWs rarely operate at full-power capacity in contrast to commercial power reactors.

Therefore, an assumption of NPW reactor operating at around one-third of its rated power is made in this report. 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.

In this report, the nuclide inventory inside the reactor was calculated using ACCIDENT code method [1] and verified using the ORIGEN code [2].

The neutron balance inside the reactor core is represented by the differential equation (1) that describes nuclear generation, depletion, and decay:

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

Where:

→ N_i is the amount of nuclide i (atoms)

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

→ σ_f^n is the microscopic fission cross section of each heavy nucleus.

→ y_n^i is the fission yield of each heavy nucleus for the FPi,

- → 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 Pj and Pk to the FPi,
- → N_{Pj} and N_{Pk} are the concentrations of the parents Pj and Pk in the fuel,
- → λ_{Pj} is the radioactive decay constant of the parent Pj,
- → σ_c^{Pk} is the microscopic capture cross section of the parent Pk,
- → σ_a^j is the microscopic capture cross section of the FPi,
- → λ_i is the radioactive decay constant of the FPi.

In equation (1), the first term represent generation from heavy nuclei by fission (which can be represented by another term, $\vec{S}$), the second term represents generation from one (or several) parent nuclide by radioactive decay or by neutronic capture and the third term represents removal by radioactive decay and/or by neutronic capture.

2.1.1 ORIGEN Methodology

The ORIGEN sequence of SCALE 6.2.4 [6] is capable of performing the reactor core depletion[1], neutron activation, actinide transmutation, fission product generation and radiation source term calculations.

Considering only the time dependence Eq. 1 can be written in matrix form as

$$\frac{d\vec{N}}{dt} = \boldsymbol{A_n}\vec{N}(t) + \overrightarrow{S_n}, \tag{2}$$

$$\text{Over time step } t_{n-1} \leq t \leq t_n,$$

where $\boldsymbol{A_n}$ is referred to as the "transition matrix", $\boldsymbol{A_n} = \boldsymbol{A_{\sigma,n}\phi} + \boldsymbol{A_\lambda}$. Here $\boldsymbol{A_{\sigma,n}}$ contains reaction terms and $\boldsymbol{A_\lambda}$ contains decay terms. Problem- dependent transition matrix is interpolated for ORIGEN calculations. ORIGEN libraries contain both problem-independent data (decay constants, molar masses, recoverable energy from decay, radiotoxicity factors) and problem-dependent data (transition coefficient matrix, cross sections, neutron yield and energy released during fission and neutron capture reactions). In eq. 2 $\boldsymbol{S_n}$ refers to time dependent continuous feed or neutron source. For nuclear fuel depletion and radiative decay studies, ORIGEN sequence also includes solving the depletion/decay equations of Eq. 2 using hybrid matrix exponential/linear chain method.

For decay / depletion external feed, $\overrightarrow{S_n}$= 0, the solution to Eq. 2 can be written as:

$$\vec{N}(t) = \exp(\boldsymbol{A\,t})\,\vec{N}(0), \tag{3}$$

where $\exp(\boldsymbol{A\,t})$ can be expanded as $\boldsymbol{I} + \boldsymbol{A}t + \frac{(\boldsymbol{A}t)^2}{2} + \cdots = \Sigma_{k=0}^{\infty}\,((At)^k/k!)$. Here, $\boldsymbol{I}$, is the identity matrix. Thus, the solution as presented in Eq. 4, is applicable to fuel burnup, activation and decay calculations:

$$\vec{N}(t) = \Sigma_{k=0}^{\infty}\,((At)^k/k!)\,\vec{N}(0) \tag{4}$$

For systems where, external feed term, $\overrightarrow{S_n} \neq 0$, the solution to non-homogenous equation for a fixed feed or removal rate $\vec{S}$ will take the form as under:

[1] During the reactor operation, the change in the fissile content of the fuel is referred to fuel depletion.

$$\vec{N}(t) = \sum_{k=0}^{\infty} \frac{(At)^k}{k!} \vec{N}(0) + \sum_{k=0}^{\infty} \frac{(At)^k}{(k+1)!} \vec{S}t \quad (5)$$

2.1.2 ACCIDENT methodology

In the Accident methodology [3], it is assumed that the main term expressing fission products (FPs) production comes from the fission of heavy nuclei via the associated fission yield. Similarly, the main term for the removal of radioactive FPs comes from their radioactive decay. Therefore, the Eq. (1) above can be simplified as follows:

$$\frac{dN_i}{dt} = y_{eq}^i F - \lambda_i N_i \quad (6)$$

where y_{eq}^i is the equivalent fission yield for all of the heavy nuclei (this information can be obtained from IAEA Nuclear Data Library), and $F = \phi \sum_{hn} \sigma_f^n N_{hn}$ represents the number of fissions/second and 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 $\sim 3.2 * 10^{-11}$ (W s)).

Considering only the time dependence, the solution of Eq. 6 will be as under:

$$N_i = y_{eq}^i * \frac{F}{\lambda_i}(1 - e^{-\lambda_i t}) \quad (7)$$

Since,

$$A_i = \lambda_i N_i = y_{eq}^i F(1 - e^{-\lambda_i t}) \quad (8)$$

Thus,

$$A_i = y_{eq}^i P_{th} E_f^{-1}(1 - e^{-\lambda_i t}) \quad (9)$$

where:

- A_i – activity of nuclide i.
- λ_i is the radioactive decay constant of the FPi,
- y_{eq}^i is the equivalent fission yield for all of the heavy nuclei,
- P_{th} - thermal power,
- E_f - mean fission energy equivalent to about ~200 MeV (or $\sim 3.2 * 10^{-11}$ (W s)).

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:

$$N_i = y_{eq}^i P_{th} E_f^{-1} t \quad (10)$$

- 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:

$$N_i = y_{eq}^i P_{th} E_f^{-1} / \lambda_i \quad (11)$$

Note that the ACCIDENT code nuclide inventory is modelled based on equation (9). See eq. 1 in appendix A.

2.2 Estimation of the fission product release from reactor core

In an event of severe accident[2], which involve the reactor core melt-down, gases, vapors, and airborne particles (aerosols) are formed. A part of these substances are the radioactive fission products (FPs), representing the source terms.

The FP inventory into the core is mainly driven by the fuel burnup[3]. However, other factors like fuel composition, material etc. also play important role in the FP release kinetics. The release of FPs from the fuel is classified into following categories based on its degree of volatility [7]:

- Fission gasses (Xe, Kr) and volatile FPs (I, Cs, Ru, Te, Sb)
- Semi-volatile FPs (Mo, Ru, Pa, Tc)
- Low volatile FPs (Sr, Y, Ni, Ru, La, Ce, Eu)
- Non-volatile FPs (Zr)
- Actinides

2.3 Estimation of the source term into the containment

During a radiological release following a severe accident, radionuclide inventory inside the reactor containment in terms of composition and magnitude is called the “in-containment source term”. A reference accident scenario of the relevant release phases of FPs to the containment is described in ref [7], and is referred to as containment source term:

- → Gap release (starts at 30 mins and continues till 80 min, provided no mitigation measures are taken)
  - Gap release starts at clad failure during fuel heat-up to 800-900°C, allowing release of the more volatile radionuclides such as noble gases, iodine, and cesium, which have been released from the fuel during normal operation, and accumulated in the gap between the fuel and the cladding. The duration of the release is typically about 30 – 50 minutes.
  - 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 release (typically starts 80 min and ends around ca 120 min, provided no mitigation measures are taken)
  - In this 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. This phase ends with the reactor pressure vessel melt-through.
- → Ex-vessel release (starts 120 min and ends around ca 600 min, provided no mitigation measures are taken).
  - This release phase is related to large amounts of radioactive release on loss of integrity of containment. Even though molten corium concrete interactions (MCCI) are important phenomena for marine reactors, following ex-vessel phenomena should be considered:
    - Loss of containment due to high-pressure failure of reactor vessel lower head.
    - Steam explosion in containment due to low-pressure failure of reactor vessel lower head.
    - Containment melt-through due to low-pressure failure vessel lower head.

[2] A nuclear accident is considered to be “severe” when the reactor core is severely degraded as a consequence of a postulated accident condition.
[3] The fuel burnup is simply the amount of cumulative energy generated per weight of fuel.

2.4 Retention mechanisms in the reactor containment

The containment source term is defined by the release from the fuel as well as by retention of different fission products in the Reactor Coolant System (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. Several important physicochemical 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). The physical effects taking place involve primarily aerosol physics and dynamics [4].

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 impaction (due to turbulence or flow geometry changes); etc.

For NPW accidents, detailed modelling of above-mentioned phenomena is very difficult. This is mainly due to the lack of detailed data of NPW designs. Therefore, it would be more adequate to use estimates of retention factors, in case of severe accident initiated by unmitigated LOCA, as described in NUREG-1465 [7], for different FPs in the RCS and/or release fractions for the containment source term.

2.4.1 Fuel- release fraction

The fuel release fractions are calculated for commercial PWR and BWR reactors in NUREG 1150 for number of accident sequences. In Table-1 is given the mean estimated core fractions released into containment for unmitigated LOCA [7] for PWRs. For metallic uranium fuel Table 2 core release fractions are used as given in ref. [8].

Table 1: Reactor core inventory fractions released to the reactor containment in case of unmitigated LOCA in PWRs [7].

	Gap Release	In-vessel	Ex-vessel
Duration (Hours)	0.5	1.5	3
Noble Gases (Xe, Kr)	0.05	0.95	0
Halogens (I, Br)	0.05	0.25	0.3
Alkali Metals (Cs, Rb)	0.05	0.2	0.35
Tellurium Group (Te, Sb)	0	0.05	0.25
Barium, Strontium (Ba, Sr)	0	0.02	0.1
Noble Metals (Ru)	0	0.0025	0.0025
Cerium group (Ce)	0	0.005	0.005
Lanthanides	0	0.002	0.005

Table 2: Core release fraction estimates for metallic uranium fuel [8].

RN Group		Normal Operation ~500°C	Eutectic Formation ~700 °C	Fuel Melting ~1100 °C.	High Temperatures Over 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

With reference to Table 2, it is important to note that a 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 severe accident progression in metal-cooled reactors (e.g., fuel reactions with steam, etc.).

2.4.2 Fission products retention in the containment

Fission products released to the containment are subject to the same retention/remobilization mechanisms as described before for FPs transport in the RCS. These were: (i) chemical reactions of gases and vapors with aerosols and structural surfaces; (ii) nucleation of vapors and formation of aerosols; (iii) agglomeration of aerosols (Brownian diffusion, sedimentation, turbulence); (iv) deposition of aerosols by diffusion and thermophoresis etc.

Due to lack of detailed design parameters of reactor and containment of NPW, in ACCIDENT a few retention phenomena are modelled namely: decay of radionuclides; agglomeration of aerosols and gravitational deposition of radionuclides.

2.5 Releases of radioactive substances into the environment

The radiological release to the environment is the radioactive material released from the containment and is obtained from the containment leakage rate. The FPs release into the atmosphere depends on the quantity of airborne radionuclides inside the containment and release paths.

2.5.1 Fission products retention in the containment

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, as shown in Figure 1.

For the NPW reference severe accident, following release paths are considered relevant:

- Containment By-pass

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

- Containment rupture

  Containment rupture due to over pressurization, melt-through, hydrogen combustion, and ex-vessel steam explosion.

- Diffused leakage

  In cases where containment integrity is maintained, diffuse leakage may occur, since containment cannot be assumed as leak tight.

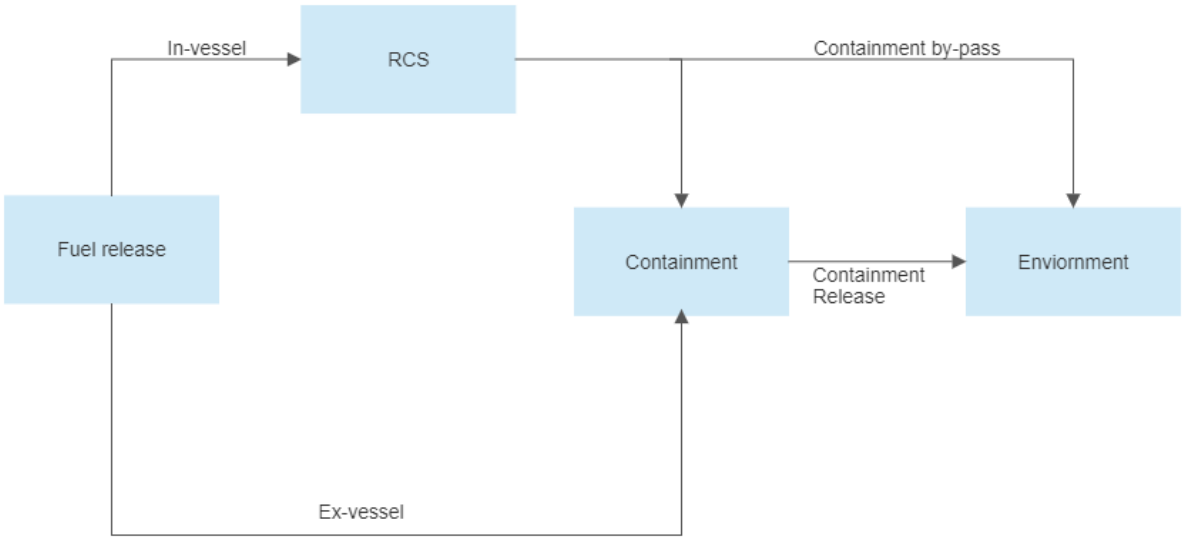

Figure 1: Radionuclide release path in case of a postulated unmitigated LOCA accident

2.5.2 Environmental Source term

The quantity of FPs initially airborne, inside the containment is depleted with time after the accident, due to the processes of deposition onto containment surfaces, radioactive decay, and leakage from the containment. The deposition rate of each radionuclide is determined from the calculated quantity of that nuclide airborne inside the containment, the containment dimensions (surface area and volume), and the

rate at which the nuclide settles onto containment surfaces (specified by a parameter termed the containment deposition velocity).

Estimation of environmental source term is usually performed using integrated plant response codes (like MELCORE, MAAP etc.). These codes evaluate the coupled effects (accident scenario, phenomena, scenarios) of different aspects of severe accident progression on the source term. Experience and literature show [9] that for unmitigated severe accidents, the environmental release starts around ca 2 hours after the initiation of accident. In the ACCIDENT code methodology, the environmental release starts imminently on the initiation of accident.

Non-gaseous fission products trapped inside the reactor containment are deposited onto surfaces within the containment, reducing the amount of activity available for release to the atmosphere. The dominant deposition processes as discussed before are agglomeration / coagulation of particles, and gravitational settling [3, 10].

In the ACCIDENT code, the containment deposition of fission products is modeled. The deposition behavior is characterized by deposition half-life of aerosols inside the reactor containment. From the results of reference [11], deposition half-life of aerosols is ca 2.5 hours. For Cs and I, deposition half-live is of the order of 40 minutes. To include the conservatism, deposition half-life of aerosols is assumed to be 5 hours for all FPs except noble gases. The containment deposition velocity is related to the deposition half-life through the surface area to volume ratio of the containment. The external surface area of the reactor compartment of NPW has been assumed to be 470 m^2, while the free volume of the reactor compartment is estimated to be 390 m^3 [8], giving a surface area to volume ratio of 1.2 m^{-1}. Thus, the deposition velocity corresponding to a deposition half-life of 5 hours is 30μms^{-1}.

Is it assumed that containment system of NPW is dual containment system, primary and secondary. The leakage rate from the NPW primary containment is estimated to be no more than 1% of the contained volume per day (%VPD) at pressure 2MPa [12]. In ACCIDENT code methodology, the design overpressure is assumed to be maintained for 24 hrs. Thus 1 % VPD primary containment leak rate over 24 hours is assumed in ACCIDENT code. The FPs from primary containment into the secondary containment (bulkhead) can further be leaked into the atmosphere. For situations where containment systems remain intact and only diffused leakage occur from the penetrations, leakage from secondary would not be more than 10 % VPD [3]. The effective overall containment leak rate will be 0.1 %VPD.

3 Case Study - Application of the method on a hypothetical submarine

Consider an accident, LOCA with diffused leakage from containment occurs on a submarine having following characteristics:

- → Hypothetical Los Angeles class submarine
- → Maximum/rated reactor power ~ 160 MW(t)
- → The average reactor operating power ~ 40 MW(t)
- → Avg. fraction of core replaced at refueling - 1
- → Fuel discharge: 10 years
- → Reactor power: 40 MW;160MW (last 10days)
- → Last fuel cycle – 160 MW for 10 days

The core inventory calculated by the ACCIDENT code is given in Table 3:

Table 3: Reactor core Inventory estimated for a hypothetical submarine using ACCIDENT code [1].

Nuclide	Conc. (Bq)	Nuclide	Conc. (Bq)	Nuclide Group	Conc. (Bq)
Kr85m	6,50E+16	Sb127	6,00E+15	Total Xe,Kr	1,08E+18
Kr87	1,25E+17	Sb129	5,00E+16	Total I,Br	1,35E+18
Kr88	1,78E+17			Total Cs,Rb	1,03E+18
Xe131m	6,39E+14	Ba140	1,78E+17	Total Te,Sb	8,07E+17
Xe133	2,68E+17	Sr89	8,24E+16	Total Ba,Sr	1,16E+18
Xe133m	8,23E+15	Sr90	1,56E+16	Total Ru	1,06E+17
Xe135	3,15E+17			Total Mo,Tc,Rh	6,05E+17
Xe135m	9,00E+16			Total Ce,La,Pr,Y	2,44E+18
I131	9,94E+16	Ru103	5,57E+16	Total Nb,Zr	8,29E+17
I132	2,17E+17	Ru106	5,14E+15	Total All Nuclides	9,16E+18
I133	3,34E+17				
I134	3,90E+17	Mo99	2,89E+17		
I135	3,08E+17	Tc99m	2,71E+17		
Br82	1,99E+12				
Cs134	8,32E+16				
Cs136	1,80E+17	Ce143	2,94E+17		
Cs137	1,62E+16	Ce144	7,25E+16		
Cs138	3,35E+17				
Rb88	1,79E+17				
		Y90	2,75E+17		
Te127	6,50E+15				
Te129	4,50E+16				
Te129m	6,82E+15				
Te131m	2,19E+16	Nb97	3,10E+17		
Te132	1,97E+17				
		Zr97	2,95E+17		

Nuclide inventory is also calculated using ORIGEN methodology with similar assumptions as for the ACCIDENT code. Other reactor parameters necessary for reactor simulations are assumed as under:

- → Fuel – UO_2 in aluminum cladding
- → Fuel enrichment ~ 90 % by weight

- → Fuel basis – 1MTU (Mega ton uranium)
- → Average moderator (H_2O) density ~0.75 g/cm^3
- → Cycle specific power 40 MW/MTU
- → Reactor availability during a year ~ 85 %
- → Reactor specific power from day 3640 to 3650 ~165 MW/MTU

Nuclide inventory of the core at the time of accident using ORIGEN is shown in Table 4.

Table 4: Radionuclide inventory of a hypothetical submarine core determined through ORIGEN [2].

Nuclide	Conc (Bq)	Nuclide	Conc. (Bq)
Kr85m	6.615e+16	Te127	6.104e+15
Kr87	1.301e+17	Te129	2.458e+16
Kr88	1.804e+17	Te129m	1.781e+15
Xe131m	6.831e+14	Te131m	2.106e+16
Xe133	2.570e+17	Te132	2.002e+17
Xe133m	9.302e+15	Sb127	7.017e+15
Xe135	2.016e+17	Sb129	2.741e+16
Xe135m	6.060e+16	Ba140	1.786e+17
I131	9.897e+16	Sr89	8.302e+16
I132	2.061e+17	Sr90	1.354e+16
I133	3.369e+17	Mo99	2.924e+17
I134	3.988e+17	Tc99m	2.558e+17
I135	3.206e+17	Ce143	3.014e+17
Br82	4.062e+13	Ce144	6.812e+16
Cs134	1.735e+15	Y90	1.357e+16
Cs136	1.042e+15	Nb97	3.059e+17
Cs137	1.396e+16	Zr97	3.047e+17
Cs138	3.421e+17	Total	4.913e+18
Rb88	1.817e+17		

The nuclide determination using ORIGEN method and ACCIDENT method are compared in figure 2. It is evident from the fig. 2 that ACCIDENT code gives higher concentrations of ^{136}Cs, ^{134}Cs, ^{90}Y, ^{135}Xe, and ^{135m}Xe, at the time of accident. The largest differences in terms of overestimated nuclide concentrations in ACCIDENT code are: ^{136}Cs (173 times), ^{134}Cs (48 times), ^{90}Y (20 times), ^{129m}Te (3.83 times), ^{129}Te (1.83 times), ^{135}Xe (1.56 times), and ^{135m}Xe (1.49). However, ACCIDENT code underestimated ^{82}Br (20.4 times) from ORIGEN code. For other important nuclides the difference was less than 16%. The overall concentration of radionuclides from ORIGEN and ACCIDENT codes are 4.913E+12 MBq and 5.61E+12 MBq.

It is notable that average reactor core operating parameters were used in both these methods to determining the radionuclide inventory of the core. The actual reactor operating parameters, including detailed reactor power history and fuel burn-up, would give the more correct picture of the radionuclide inventory.

In ACCIDENT code the post shutdown precursor sources were neglected like the production ^{131}I from ^{131}Te and ^{135}Xe from ^{135}I were neglected [3]. Additionally, the FP yields of ^{133}Cs and ^{136}Xe were used in ACCIDENT code to estimate the concentrations of ^{134}Cs and ^{136}Cs, respectively [3]. However, the ORIGEN methodology is based on solving the Bateman's equation (equation 1) [3] using interpolated ENDF nuclear

cross-section data for the given reactor conditions. Therefore, one may expect these differences in results as shown above. The theoretical basis of ACCIDENT code is given in Appendix A.

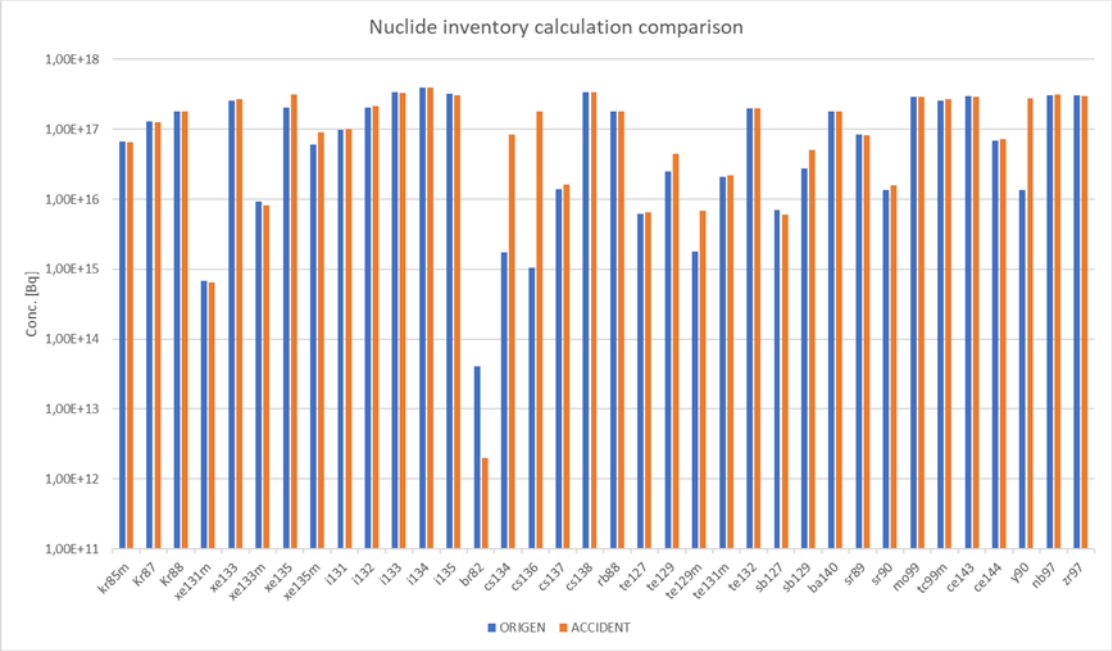

Figure 2: Comparison between nuclide inventory determination using ORIGEN method and ACCIDENT method.

The cumulative core release fractions of Table 1 were used to determine the release of radionuclides from the fuel.

To determine the release from the containment, the release duration of ca 12 hours was used. The containment dimensions were assumed to be, internal surface area ~ 470 m2 and internal volume ~ 392.5 m^3. Moreover, the containment deposition velocities for all radionuclides were assumed to be 3 x 10^{-5} m/s. Regarding Iodine chemistry in the containment, only 2 % of iodine was assumed to be of organic form.

The containment leak rate was assumed to be 0.1 % VPD.

Table 5: Total released activity in Becquerels from containment into the environment after 12 hrs, using ACCIDENT method.

Nuclide	Conc [Bq]	Nuclide	Conc [Bq]
Kr85m	1,48E+13	Cs138	5,90E+12
Kr87	9,53E+12	Rb88	1,82E+12
Kr88	2,88E+13	Te127	3,65E+11
Xe131m	3,15E+11	Te129	7,75E+11
Xe133	1,29E+14	Te129m	5,18E+11
Xe133m	3,81E+12	Te131m	1,51E+12
Xe135	1,03E+14	Te132	1.45E+13
Xe135m	1,35E+12	Sb127	4.43E+11
I131	1,52E+13	Ba140	5.37E+12
I132	1,26E+13	Sr89	2,51E+12
I133	4,48E+13	Sr90	4,77E+11
I134	1,06E+13	Mo99	7,02E+11
I135	3,14E+13	Tc99m	4,38E+11
Br82	2,82E+08	Ce143	6.82E+11
Cs134	1,27E+13	Ce144	1.84E+11
Cs136	2,72E+13	Y90	6.67E+11
Cs137	2,47E+12	Nb97	1.83E+11
		Zr97	6.29E+11

Nuclide Group	Conc [Bq]
Total Xe,Kr	2.95E+14
Total I,Br	1.15E+14
Total Cs,Rb	5.22E+13
Total Te,Sb	2.35E+13
Total Ba,Sr	2.09E+13
Total Ru	1.09E+11
Total Mo,Tc,Rh	1.24E+12
Total Ce,La,Pr,Y	4.91E+12
Total Nb,Zr	1.38E+12
Total All Nuclides	5.14E+14

4 Summary and Conclusion

In this document, severe accident phenomenology for submarine reactors has been briefly described. The radionuclide inventory of a hypothetical submarine reactor core has been estimated using average reactor parameters for the purpose of emergency preparedness. To achieve this, two methods were compared, ACCIDENT code and ORIGEN code.

The methods were applied on a hypothetical submarine. The cumulative total reactor core inventory for the similar reactor operational conditions using the two methods is comparable. However, regarding the individual radionuclides it is evident that the ACCIDENT code gave higher estimates of the radionuclides of cesium, ytterbium, tellurium, and xenon (^{135}Xe, and ^{135m}Xe) in the reactor core inventory. The biggest difference was found in ^{136}Cs (~173 times) and ^{134}Cs (48 times). For ^{82}Br, it is also observed that the ACCIDENT code estimated nuclide concentration ~20 times lower than the ORIGEN estimates. However, the difference for most of other important radionuclides lies within 16 %.

These overestimations by the ACCIDENT code have already been noted down by ARPANSA in ref. [3], which states that the ACCIDENT code does not take into account of the activation yield of stable nuclides ^{133}Cs and ^{136}Xe while estimating ^{134}Cs and ^{136}Cs concentrations. Instead, the fission product yields of ^{136}Xe and ^{133}Cs were used for ^{136}Cs and ^{134}Cs respectively, for the sake of conservatism.

References

1. A.J. Frikken, “ACCIDENT” Nuclear Accident Consequence Assessment Code, Descriptive Manual, NSB12/1997, 1997.
2. M. J. Bell, ORIGEN B—The ORNL Isotope Generation and Depletion Code, ORNL-4628 (CCC-217), Union Carbide Corporation (Nuclear Division), Oak Ridge National Laboratory, May 1973.
3. ARPANSA, “The 2000 Reference Accident Used to Assess the Suitability of Australian Ports for Visits by Nuclear Powered Warships, RB–NPW–66/00”, ARPANSA, 2000
4. H. Bateman, “The solution of a system of differential equations occurring in the theory of radioactive transformations”, Proc. Cambridge Philos. Soc. 15 (1910) 423–427.
5. Regulatory Guide 1.183, Alternative Radiological Source Terms For Evaluating Design Basis Accidents At Nuclear Power Reactors, U.S. NRC., 2000.
6. W. A. Wieselquist, R. A. Lefebvre, Eds., SCALE 6.3.1 User Manual, ORNL/TM-SCALE-6.3.1, UT-Battelle, LLC, Oak Ridge National Laboratory, Oak Ridge, TN (2023)
7. L Soffer, S. B. Burson, C. M. Ferrell, R. Y. Lee, J. N. Ridgely, Accident Source Terms for Light-Water Nuclear Power Plants, NUREG-1465, US. NRC, 1995.
8. D. Grabaskas, M. Bucknor, J. Jerden, Regulatory Technology Development Plan Sodium Fast Reactor, Mechanistic Source Term – Metal Fuel Radionuclide Release, ANL-ART-38, 2016.
9. NEA, State-of-the-art Report on Innovative Fuels for Advanced Nuclear Systems, NEA OECD Report 6895, 2014.
10. J J M R Hugron, Consequence Analysis of a Nuclear Submarine Reactor Accident, Thesis submitted to the Dept. of Chemistry and Chemical Engineering, Royal Military College of Canada, Kingston, Ontario, April 1996.
11. Committee on the Safety of Nuclear Installations, OECD Nuclear Energy Agency, Workshop on Aerosol Behaviour and Thermal-Hydraulics in the Containment: Proceedings, CSNI Report No. 176, November 1990.
12. D. A. Powers and S. B. Burson, A Simplified Model of Aerosol Removal by Containment Sprays, NUREG/CR-5966, SAND92-2689, Sandia National Laboratories, Albuquerque, NM, June 1993.

Appendix A - Equations of ACCIDENT code [1]

Core inventory equation:

$$I_n = FY_n(P_{av}\exp(t_{max}+t_{sd})(1-\exp(-\lambda_{d,n}t_{irr})) + P_{max}\exp(-\lambda_{d,n}t_{sd})(1-\exp(\lambda_{d,n}\,t_{max})) \quad (1)$$

Here, $t_{irr} = \frac{t_f}{2}\left(1+\frac{1}{f}\right)$

here,

I_n – Core inventory in (Bq)

F – Fission rate per unit thermal reactor power ~3.125 x 1016 (fissions s-1 MW-1)

P_{av} – Avg reactor thermal operating power (MW)

P_{max} – Max reactor thermal power (MW)

Y_n – Fission yield of nuclide n from fission of uranium 235U

$\lambda_{d,n}$ – Radioactive decay time constant for nuclide n (s-1)

t_{irr} – Max average irradiation time of fuel in core (s)

t_f – operating time between refueling (s)

f – fraction of core replaced at refueling

t_{max} – time for which reactor was operated at max power prior to refueling (s)

$t_{s,d}$ – time reactor was shut down prior to accident (s).

Release from containment in a nuclear-powered vessel (NPW):
Containment depletion time constant, $\lambda_{p,n}$:

$$\lambda_{p,n} = \frac{v_{p,n}S}{V}\ [s^{-1}], \text{ where} \quad (2)$$

$v_{p,n}$ is the containment deposition velocity [m/s],

S = Internal Surface Area of Containment [m2]

V = Internal Volume of containment

Leakage time constant, λ_l^i:

$$\lambda_l^i = -\ln\left[\frac{1}{100}\left(100-\frac{l}{86400}\right)\right]\ [s^{-1}], \text{ where} \quad (3)$$

l = leakage rate of containment [%volume/day]

Containment Activity Depletion time constant, λ_{en}^i:

$$\lambda_{en}^i = \lambda_{d,n}^i + \lambda_{p,n} + \lambda_l^i, \text{ were} \quad (4)$$

$\lambda_{d,n}^i = \frac{ln2}{T_{\frac{1}{2}}^i}$ is radioactive decay constant

Initial airborne Activity in containment, $Q_{0,n}^i$:

$$Q_{0,n}^{i} = \begin{cases} I_{n,f} & for\ i = 1 \\ Q_{0,n}^{i-1} \exp\left(-\lambda_{e,n}^{i-1} T\right) & for\ i > 1 \end{cases}$$, where (5)

$I_{n,f}$ is the nuclide inventory given in Table 3,

T is the duration of release intervals

Airborne activity in the containment, $Q_n^i(t)$ at time t during release interval:

$$Q_n^i(t) = Q_{0,n}^i \exp(-\lambda_{e,n} t) \quad (6)$$

Assuming a constant containment leak rate during release interval, instantaneous activity leak rate $l_n^i(t)$ of nuclide from containment at time t during release interval i [Bq]

$$l_n^i(t) = \lambda_l^i Q_n^i(t) = Q_{0,n}^i \lambda_l^i \exp(-\lambda_{e,n}^i t), \quad (7)$$

Time integrated activity leak rate $L_n^i(t)$ of nuclide n from containment after time t during release interval i [Bq]:

$$L_n^i(t) = \int_0^t l_n^i(t) dt = \frac{Q_{0,n}^i \lambda_l^i}{\lambda_{e,n}^i} \exp(1 - \lambda_{e,n}^i t) \quad (8)$$

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