SOURCE TERM ANALYSIS FOR A NUCLEAR SUBMARINE ACCIDENT

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ABSTRACT

A source term analysis has been conducted to determine the activity release into the environment as a result of a large-break loss-of-coolant accident aboard a visiting nuclear-powered submarine to a Canadian port. This best-estimate analysis considers the fractional release from the core, and fission product transport in the primary heat transport system, primary containment (i.e., reactor compartment) and submarine hull. Physical removal mechanisms such as vapour and aerosol deposition are treated in the calculation. Since a thermalhydraulic analysis indicated that the integrity of the reactor compartment is maintained, release from the reactor compartment will only occur by leakage; however, it is conservatively assumed that the secondary containment is not isolated for a 24-h period where release occurs through an open hatch in the submarine hull. Consequently, during this period, the activity release into the atmosphere is estimated as 4.6 TBq, leading to a maximum individual dose equivalent of 0.5 mSv at 800 metres from the berthing location. This activity release is comparable to that obtained in the BEREX TSA study (for a similar accident scenario) but is four orders of magnitude less than that reported in the earlier Davis study where, unrealistically, no credit had been taken for the containment system or for any physical removal processes.

1. INTRODUCTION

A few accidents have occurred aboard nuclear powered vessels (NPVs) from various countries as shown in Table 1.[1] In light of the possibility of an accident, some public concern has been expressed over visits of foreign NPVs to Canadian ports, and the passage of NPVs through Canadian waters. Consequently, the possible consequences of an accident aboard an NPV have been the subject of several studies. Davis considered a hypothetical reactor accident aboard a submarine berthed at Esquimalt; however, no credit was taken for the containment system or for any physical removal processes inside the vessel in the radiological analysis.[2] The Department of National Defence (DND) conducted a comprehensive Technical Safety Assessment (TSA) as part of a berthing reexamination (BEREX).[3] The objective of the TSA was to define and study postulated accidents, determine their frequency of occurrence and the amount of radiation released. In contrast, the TSA showed that the large majority of serious accidents (involving core melting) would lead to very small dose equivalents outside the submarine. Even extreme postulated accidents would have a very low probability of occurrence, where the resulting radiation dose equivalents would only slightly exceed the current occupational dose limits.

As an independent analysis, a source term assessment is considered in this work, which takes into account various physical removal mechanisms, in order to determine a best estimate of the activity release into the environment as a result of a large-break loss-of-coolant accident (LOCA) aboard a visiting nuclear-powered submarine to a Canadian port.

2. ACCIDENT ANALYSIS

2.1 Submarine Design Criteria

The NPV design for the visiting submarine is based on open literature information on US and UK attack submarines.[4] The Los Angeles class of attack submarines, which has the largest hull and the largest power source, has been the focus of the reference study. The nuclear reactor design is mainly based on that used for the commercial Savannah NPV, which followed from the design of the first nuclear powered submarine, the USS Nautilus. Since there is no publicly-available information on the nuclear power plant of a modern NPV, the assumptions used in the reference design may vary in accuracy.

The reference NPV reactor is assumed to be a 200 MW (thermal) pressurized water reactor (PWR) fuelled with a highly enriched (97 wt%) ^{235}U fuel assembly. The fuel is assumed to be an alloy of uranium and zirconium, surrounded by Zircaloy-2 cladding. The total mass of uranium in the core at the beginning of its life is 400 kg (i.e., 388 kg of ^{235}U). The fuel composition is estimated to be roughly 26 wt% uranium in the U-Zr alloy. Thirty-seven fuel assemblies are assumed to be in the reactor core, with 17 in the inner core and 20 in the outer core. The structural material in the fuel assemblies is assumed to be stainless steel. The fuel is in the form of plates where the uranium fuel meat in each plate is sandwiched between the Zircaloy cladding.

2.2 Accident Scenario

For this analysis, the reactor core life is expected to be approximately 12 y, with the power reactor operating at an average power of 25% of full power. Thus, to maximize the long and short-lived radionuclide inventory in the core, it is assumed that the reactor is near its end-of-life prior to the accident and is operated at full power for 100 h, after which it is reduced to half power for transit into the harbour. An accident is assumed to happen two hours after the reactor has been operating at half power, while it is entering into the harbour. The postulated accident scenario is a large-break LOCA, with a large instantaneous guillotine break at the inlet pipe (cold leg) between the reactor vessel and the isolation valve. This scenario has been chosen because this break cannot be isolated, and it gives the largest discharge rate into containment and the fastest core heat up rate. No coolant make-up is assumed throughout the accident. It is further assumed that an escape hatch is left open a machinery compartment of the submarine (see Fig. 1) so that a pathway exists for release into the atmosphere.

A thermohydraulic analysis was performed as part of the BEREX TSA.[5] The CATHENA code was used to calculate the coolant mass and energy discharge rates from the primary system into the containment vessel, the coolant pressures, temperatures and flow rates in the heat transport system, the fuel and cladding temperatures, and the heat and hydrogen generation from the zirconium-steam reaction. For this analysis, the fuel core was divided into two regions: (i) an inner, and (ii) an outer core region, which were further subdivided into seven axial sections. The timing of major events during the accident is given in Table 2.[6] The volume-average fuel temperature transients arising from the given accident scenario for the two core regions are shown in Figs. 2 (a) and (b). The rate of the temperature rise becomes significant only after the fuel is uncovered. For most of the fuel in the core, the increase in temperature from 750°C to its melting point (~1650°C) occurs in only a few seconds. The increased fuel cladding temperatures cause the Zircaloy cladding to react chemically with steam to produce additional heat and hydrogen.

2.3 Source Term Analysis

For the source term analysis, it is necessary to determine the fission product pathway through the submarine (see Fig. 1). This analysis requires an estimate of the inventory of fission products (FPs) in the fuel core (Section 2.3.1) and their subsequent release fraction (Section 2.3.2), as well as their transport and release from the reactor vessel and primary heat transport system (PHTS) (Section 2.3.3), containment structure (Sections 2.3.4 and 2.3.5) and submarine hull (Sections 2.3.6).

2.3.1 *Inventory in the Fuel*

The ORIGEN2 computer code was used to calculate the core inventory for the given irradiation history.[7] As detailed in Ref. 8, the inventory of FPs in the fuel was calculated at the time that each part of the core is expected to melt (based on the thermohydraulics analysis in Section 2.2). The calculations were performed using the cross-section libraries developed for a commercial PWR. Even though there are major differences between the NPV and commercial reactor, these differences will not significantly alter the FP inventory, which is mainly dependent on the initial quantity of ^{235}U in the fuel and the reactor power history.

Table 3 presents the inventory of 74 isotopes (FPs and actinides) that contribute to the majority of activity during the accident, along with some radiological-important isotopes. For the given irradiation history, the total activity is 21 EBq. The present analysis also takes into account decay following reactor shutdown.

2.3.2 *Fuel Release Fraction*

For the high fuel temperatures experienced during the accident, cladding integrity is not maintained and significant volatile FP release is expected from the fuel. The literature has been reviewed to estimate the release fraction from the fuel. Unfortunately, data on the behaviour of U-Zr alloy fuel is limited, where only one study, conducted at the Oak Ridge National Laboratory (ORNL), was reported in which the fuel was exposed to a steam environment at 1750°C for 30 s.[9] Therefore, technical judgement has been used to estimate the release fraction from available information on U-Zr alloy fuel, as well as for other fuels including U-Al alloy,[9,10] uranium metal[11] and uranium dioxide (UO_2) (see Table 4).[12-21]

Figure 2 shows that the heat-up of the fuel is quite rapid. Since insufficient data are available to calculate the transient release behaviour, it is therefore conservatively assumed in the present analysis that the FPs are instantaneously released when the fuel melting temperature of ~1650°C is reached. For this analysis, elements with similar chemical properties were also classified into twelve groupings, as proposed for the MELCOR code package, in which the same fractional release value has been applied to the various elements in a given group.[12] For conservatism, the release fractions assumed in this study correspond to the maximum values observed in the U-Zr alloy tests. For missing FP groupings, data on the other fuel types have been considered (see Table 4), although considerable uncertainty can arise due to the different melting points of the fuel (e.g., 2840°C for UO_2 fuel versus 1650°C for the U-Zr alloy). The effect of fission-product trapping (e.g., for Te and Sb) in the unoxidized Zircaloy cladding is conservatively neglected in this analysis.[17]

Using the radioisotope inventories in Table 3, with the release fractions in Table 4 and the fuel temperature curves in Fig. 2, the FP release is evaluated as a function of time as a given fuel node melts.

2.3.3 *Retention Within the PHTS*

Before failure of the reactor vessel, FPs released from the fuel will travel inside the reactor coolant system (RCS) before reaching the containment atmosphere through the break in the PHTS. Some FPs will be retained in the RCS due to deposition or condensation onto relatively colder internal surfaces. In addition, the FPs may condense onto aerosols that would also settle and deposit within the RCS. Such retention in the RCS was investigated using the VICTORIA computer code, which determines the physical (i.e., vapour or aerosol) and chemical form of the FPs.[22] As detailed in Ref. 8, the thermohydraulic analysis from the BEREX TSA,[5,6] and the FP release rates from Section 2.3.2, were used as input into the VICTORIA code.

This calculation suggests that at the time of vessel failure, ~10% of the FPs will deposit on structural materials within the RCS, while another 10% are still in the reactor vessel in the form of either a vapour or aerosol. However, for conservatism in this work, no credit is taken for deposition in the RCS in light of the uncertainty in the various input parameters. The FPs are therefore assumed to be transported directly to the containment atmosphere. This analysis also shows that, except for the noble gases, all FP elements enter the containment vessel almost exclusively in the form of an aerosol, e.g., more than 99% of iodine is in form of a cesium-iodide particulate.

2.3.4 *FP Behaviour in the Primary Containment Atmosphere*

Initially, the reactor compartment is assumed to be at a temperature of 50°C and pressure of 101.3 kPa, with a relative humidity of 50%.[23] Shortly after the rupture, the containment atmosphere would consist of a mixture of air and saturated steam. Heat would be transferred rapidly by mass transfer of water vapour to the cooler wall surfaces. The steam condensate would drain to the bottom of the vessel and accumulate in a water pool, and the film would be constantly replenished until the temperature of the wall reaches equilibrium with the atmosphere.

Approximately five minutes after the break, the FPs would start to be released from the PHTS and enter into the primary containment. Only the inert noble gases will exist in a vapour form in the primary containment, where losses will occur through radioactive decay and leakage. Since aerosol particles are expected to settle quickly, it is therefore more conservative to assume that iodine exists as a vapour. Section 2.3.4.1 treats the removal of iodine vapour from the containment atmosphere and Section 2.3.4.2 considers the removal of aerosol particles.

2.3.4.1 *Iodine Vapour Removal*

The removal of molecular iodine vapour from the primary containment atmosphere by plate-out due to a condensing steam atmosphere has been evaluated with the Containment System Experiment (CSE) scale model.[24] Conservatively, chemical reactions have been neglected and only natural removal processes are considered in this analysis where no credit is taken for engineered removal systems.

To estimate heat losses from the primary containment to the surroundings, a simple model of heat transfer to a vertical wall composed of paint and steel is considered. The effective heat transfer coefficient, h_w, to the wall is given by:

$$h_w = \left(\frac{t_p}{k_p} + \frac{t_h}{k_h} \right)^{-1} \quad \textbf{(1)}$$

where the paint has a thickness $t_p = 0.0635$ cm and a thermal conductivity $k_p = 0.173$ W m^{-1} °C^{-1}. The submarine hull is made of a high-strength low-alloy steel, and the corresponding parameters for this steel layer are $t_h = 5$ cm[5] and $k_h = 34$ W m^{-1} °C^{-1}. Equation (1) therefore yields $h_w = 200$ W m^{-2} °C^{-1}. Using this value, along with a bulk vapour temperature of $T_b = 120$°C, surrounding air temperature of $T_a = 27$°C and containment pressure of $P = 335$ kPa, a temperature difference $T_b - T_{si} = 5$°C is evaluated, where T_{si} is the inside surface temperature.[25]

The removal rate of iodine occurs from the sweep effect of condensing steam, and the diffusion of molecular iodine across the gas-film boundary. Initially, the iodine plate-out rate, D_{po} (s^{-1}), can be expressed by:[24]

$$D_{po} = \frac{k_d A_T + k_{sw} A_w}{V} \quad \textbf{(2)}$$

where k_{sw} is a mass transfer coefficient due to the sweep effect of steam condensation, k_d is a mass transfer coefficient due to diffusion at the wall, A_T is the total surface area on which material is deposited by diffusion, A_W is the area on which steam is condensing (= 470 m^2), and V is the containment volume (= 393 m^3). Since the internal surface area is not well known, it can be assumed that $A_T \approx Aw$, which conservatively leads to less plate-out. As shown in Ref. 8, for the given bulk temperature and using the analysis in Ref. 24, $k_d \approx 4.0$ m h^{-1} (for turbulent gas-film boundary conditions in a closed vessel). Also, for a 5°C temperature difference, the analysis in Ref. 25 gives $k_{sw} = 0.82$ m h^{-1}. Hence, using Eq. (2), the plate-out rate for iodine is evaluated as 0.0016 s^{-1}, which corresponds to a removal half-life for plate-out of 430 s.

A rapid removal ocurs until the concentration decreases to about 0.5% of the initial value, when an equilibrium between the gas phase and the liquid phase is attained.[24] Thus, it is conservatively assumed in this study that the removal of iodine by plate-out stops when the concentration in the gas phase decreases to 1% of the initial value. In addition, it is assumed that 2% of the released iodine is in an organic form (methyl iodide) where plate-out does not occur. This latter quantity will be similarly removed (as for a noble gas) by radioactive decay and leakage to the hull.

2.3.4.2 *Aerosol Removal*

Aerosol particles grow rapidly in size by agglomeration. Although several mechanisms occur which remove aerosols from the atmosphere (i.e., gravitational settling, diffusiophoresis, thermophoresis and electrical attraction to surfaces), gravitational settling is normally the predominant mechanism and is only considered in this calculation. It is further assumed that the aerosol particles remain at the size that they were formed in the PHTS, i.e., this assumption is conservative since agglomeration will lead to larger particles that deposit more quickly.

From the VICTORIA analysis in Section 2.3.3, the particles have a weighted average radius of 2.9 μm and a weighted average deposition velocity of $v_t = 2.7$ mm s^{-1}. Using this deposition velocity, the removal rate constant for aerosols due to gravitational settling is calculated from:[26]

$$D_{GS} = \frac{v_t A_f}{V}. \quad \textbf{(3)}$$

Given that the surface area of the floor in the reactor compartment is $A_f = 100$ m^2 and $V = 393$ m^3,[23] Eq. (3) yields a removal rate constant for aerosol deposition of 0.00070 s^{-1}, or equivalently a removal half-life of 990 s. This half-life is adopted for all other FP species (except for iodine as discussed in Section 2.3.4.1 and the chemically-inert noble gas species). Consequently, only 8% of the aerosols remain in the containment atmosphere one hour after release from the PHTS.

Since larger particles deposit faster than smaller ones, the average size of the remaining particles in the primary containment atmosphere will become smaller. Thus, the average terminal velocity will decrease, which will also result in a smaller removal rate constant. To account for this effect, it is conservatively assumed that aerosol deposition ceases when the concentration of particles in the primary containment atmosphere is 1% of the initial value.

2.3.5 *Release From Primary Containment*

Fission products are therefore removed from the primary containment atmosphere by either vapour plate-out or aerosol deposition (see Section 2.3.4). Losses will also occur by radioactive decay and leakage to adjacent compartments.

Thermohydraulic calculations show that the pressure in the reactor compartment is not expected to increase over its design pressure of 2 MPa,[23] and therefore this compartment will maintain its integrity. The leakage rate will therefore be a function of the differential pressure between the reactor compartment and the submarine hull. The fraction of FP activity released from the reactor compartment can be calculated assuming a volumetric leakage rate of 1% per day (for a pressure differential of 2 MPa).[23] A typical leakage rate from the reactor containment system for land-based commercial reactors is ~0.1% per day under the design pressure.[26,27] Since the same containment structure (i.e., large concrete structure) cannot be afforded for the submarine, and considering that numerous pipes also penetrate the containment structure, a leakage rate which is 10 times higher seems reasonable. Since a pressure differential of 2 MPa is never attained, the leakage rate L_P (s^{-1}) will always be smaller than the previous stated value, which can be calculated from:[23,26]

$$L_P = \frac{A_0}{V}\sqrt{\frac{2\Delta P}{\rho k_f}} \qquad (4)$$

where A_0 is the leakage flow area (m^2), V is the reactor compartment volume (m^3), ΔP is the pressure difference between the primary and secondary containment (Pa), ρ is the density of the steam-air atmosphere (kg m^{-3}), and k_f is the friction loss coefficient. The density in Eq. (4) is evaluated from the ideal gas law:

$$\rho = \frac{PM}{RT} \qquad (5)$$

where P is the absolute pressure (Pa), M is the molecular weight of air (= 28.97 g $mole^{-1}$), R is the ideal gas constant (= 8.314 J mol^{-1} K^{-1}), and T is the absolute temperature (K). For a volumetric leakage rate of 1% per day at a design pressure of 2 MPa(g), and using a friction loss coefficient of $k_f = 2.85$ and $V = 393$ m^3 in Eq. (4), A_o is estimated as 1.79×10^{-7} m^2.[23] Since the secondary containment is at atmospheric pressure, the pressure differential (ΔP) is simply the gauge pressure of the reactor compartment. Thus, using the saturated temperature at a given pressure, the time-dependent leakage rate $L_p(t)$, can be evaluated as shown in Fig. 3.

Consequently, FPs will be removed from the primary containment atmosphere by: (i) aerosol deposition or vapour plate-out, (ii) radioactive decay, and (iii) leakage to adjacent compartments. If the FPs enter the primary containment at time $t = 0$, which corresponds to the time of fuel melting (of a given section), the FP inventory in the reactor compartment atmosphere, N_P (Bq), can be determined from the mass-balance equation:

$$\frac{dN_P}{dt} = -(D_R + \lambda + L_P)N_P \qquad (6)$$

where D_R is the deposition rate due to plate-out for iodine (see Section 2.3.4.1) or gravitational settling for aerosol particulates (see Section 2.3.4.2) (s^{-1}) and λ is the radioactive decay constant of the given isotope (s^{-1}). Since each isotope has a different radioactive half-life and deposition rate, this equation has to be solved separately for each isotope in Table 3. For a variable leakage rate $L_p(t)$ (see Fig. 3), the solution of Eq. (6) is:

$$N_p(t) = N_{po}\exp\left\{-(D_R+\lambda)t - \int_0^t L_p(t')dt'\right\} \qquad (7)$$

where N_{Po} is the initial inventory of FPs.

The release rate of FPs from the primary containment is therefore given by the first-order relation,

$$R_p(t) = L_p(t)\, N_p(t). \tag{8}$$

The cumulative amount of FPs released from the primary containment over a given period of time is evaluated from:

$$S_s(t) = \int_0^t R_p(t')dt'. \tag{9}$$

The activity release into the secondary containment (i.e., submarine hull) for the most important radioisotopes are shown in Table 3. A total of 1.1 PBq is released in 24 hours.

2.3.6 *Release From Submarine Hull*

In the BEREX TSA analysis, it is assumed that a hatch is left open in the machinery compartment, so that FPs which are transported down the submarine hull can be released into the atmosphere (see Fig. 1).[3] Fission-product transport in the hull may result from both a diffusive (see Section 2.3.6.1) and convective (see Section 2.3.6.2) process.

2.3.6.1 *Diffusion in the Submarine Hull*

The binary diffusion coefficient (in m^2 s^{-1}) of FP vapours (species A) in an air medium (species B), can be evaluated from the Chapman-Enskog kinetic theory:[28]

$$D_{AB} = 1.8583 \times 10^{-7} \frac{\sqrt{T^3\left(\frac{1}{M_A} + \frac{1}{M_B}\right)}}{P\, \sigma_{AB}^2\, \Omega_{D,AB}} \tag{10}$$

where T is the temperature (= 298 K), M is the molecular weight (g mol^{-1}), P is the pressure (= 1 atm), $\sigma_{AB} = ½(\sigma_A + \sigma_B)$ is combined the Lennard-Jones collision diameter (Å), and $\sigma_{D,AB}$ is the collision integral. For a "typical" FP it can be assumed that $M_A \approx 100$ g mol^{-1} and $\sigma_A \approx 3.2$ Å, while for air $M_B \approx 28.97$ g mol^{-1} and $\sigma_B \approx 3.62$ Å.[8] Thus, with $\sigma_{D,AB} \approx 1$, a nominal diffusion coefficient of 1.7 x 10^{-5} m^2 s^{-1} is evaluated.

For the FP aerosols, Brownian diffusion theory can be employed:[22]

$$D_p = k\, T\, B \tag{11}$$

where k is the Boltzmann constant (= 1.38 × 10^{-23} J K^{-1}) and T is the temperature (K). The mobility B (s kg^{-1}) is defined as:

$$B = \frac{C_n}{6\pi\chi\mu r_p} \tag{12}$$

where μ is the atmospheric viscosity (kg m^{-1} s^{-1}), χ is the dynamic shape factor (= 1 for wet aerosols), r_p is the particle radius (= 2.9 ×10^{-6} m from the VICTORIA analysis) and C_n is the Cunningham slip correlation:[22]

$$C_n = 1 + 1.25\, K_n + 0.41\, K_n e^{-0.88/K_n} \tag{13}$$

The particle Knudsen number, $K_n = l_p/r_p$, where the mean free path length is determined from:[22]

$$l_p = 1.5\mu\sqrt{\frac{\pi}{2P\rho}}, \tag{14}$$

and P is the pressure (= 1.01 × 10^5 Pa) and ρ is the air density (= 1.3 kg m^{-3}). The viscosity for air can also be evaluated from kinetic theory as:[22]

$$\mu = 2.6693 \times 10^{-6} \frac{\sqrt{M\,T}}{\sigma^2\,\Omega} \tag{15}$$

where Ω is the viscosity collision integral (= 0.9428). As detailed in Ref. 8, the diffusion coefficient in Eq. (11) is estimated as 4.0 x 10^{-12} $m^2 s^{-1}$.

Finally, the transport of FPs in the submarine hull can be determined from the radioactive diffusive equation:[29]

$$\frac{\partial C_s}{\partial t} = D \frac{\partial^2 C_s}{\partial x^2} - \lambda C_s \tag{16}$$

where C_s is the FP concentration, D is the diffusion coefficient for the FP vapour (Eq. (10)) or aerosol (Eq. (11)), t is the time and x is the axial distance along the submarine hull. The initial condition for Eq. (16) simply follows where no FPs are initially present (at $t = 0$) in the hull:

$$C_s = 0, \quad t = 0, \quad x > 0. \tag{17}$$

The two boundary conditions account for: (i) a source condition due to leakage from the reactor compartment bulkhead (i.e., at $x = 0$) where, in accordance with the Fick's law of diffusion,

$$\frac{R_p}{A_{sub}} = - D \left. \frac{\partial C_s}{\partial x} \right|_{x=0,} \quad t > 0 \tag{18}$$

and; (ii) a zero concentration at the location of the open hatch (i.e., at $x = l$), where the atmosphere is essentially an infinite sink:

$$C_s = 0, \; x = l, \; t > 0. \tag{19}$$

In Eq. (18), R_p is the average release rate from the reactor vessel as determined from Eq. (8), A_{sub} is the cross-section area of the submarine (the submarine diameter ~10 m), and l (= 5 m) is the path length between the reactor vessel and the open hatch (see Fig. 1).

The subsequent release rate through the open hatch, R_s (atom s^{-1}), can be evaluated from the Fick's law of diffusion (where conservatively using the cross-sectional area of the hull for the one-dimensional model):

$$R_s = - A_{sub} D \left. \frac{\partial C_s}{\partial x} \right|_{x=l}. \tag{20}$$

Thus, solving Eq. (16) subject to the given initial and boundary conditions, and using this solution in Eq. (20), the release rate as derived from Laplace transform methods into the atmosphere is:[8]

$$R_s = R_p \left[\frac{1}{\cosh(l\sqrt{\lambda/D})} + \frac{\pi D}{l^2} \sum_{n=1}^{\infty} \frac{(-1)^n (2n-1)}{\phi_n} e^{-\phi_n t} \right], \tag{21}$$

where:

$$\phi_n = \frac{(2n-1)^2 \pi^2 D}{4 l^2} + \lambda. \tag{22}$$

Similarly, the cumulative FP release into the atmosphere over a given period of time t, is given by:

$$S_{atm}(t) = \int_0^t R_s(t')dt'. \tag{23}$$

The transport equation of Eq. (16) does not consider any deposition in the hull, which again is conservative. However, temperature gradients in the hull may lead to further migration by a natural convection process (see Section 2.3.6.2).

2.3.6.2 *Convection in the Submarine Hull*

Transport of FPs in the secondary containment (hull) can also occur from forced or free (natural) convection. If no ventilating fan is operating in the submarine, forced convection can only result from a pressure build-up in the hull (due to leakage from the primary containment). Natural convection, on the other hand, can arise as a result of air-temperature gradients.

Using Eq. (9), the total number of moles released to the secondary containment in a period Δt of 24 h is S_s = 58 moles. Using the relation:

$$Q = \frac{S_s RT}{\Delta t\, P} \tag{24}$$

the corresponding average volumetric flow rate through the submarine hull (at 298 K and 1.01×10^5 Pa) is $Q = 1.6 \times 10^{-5}$ m^3 s^{-1}.[8] Assuming that air is incompressible, the corresponding linear flow velocity (in which flow occurs through a one-metre diameter hatch) is 2.09×10^{-5} m s^{-1}.[8] Hence, it would take about 61 h for FPs to travel along the 5-metre path length of the hull and therefore forced convection can be neglected.

Free convection will result from a temperature difference in the fluid medium (air). Since the temperature in the primary containment rises to approximately 120°C, the bulkheads separating the primary and secondary containment (see Fig. 1) will become warmer than the bulk atmosphere in the hull. The air near these bulkheads will therefore heat up, leading to a circular air-flow current. However, because the water surrounding most of the submarine is a large heat sink, and the steel hull has a high thermal conductivity of 34 W m^{-1} °C^{-1}, the temperature difference is expected to be small (5 to 10°C). Assuming a characteristic length L_c = 35 m and a temperature difference ΔT = 10°C, a large Rayleigh number of Ra = 4.0×10^{13} is estimated for the submarine hull.[8] A large Rayleigh number indicates the development of localized convection cells.[30] It is therefore expected that any free convection would be relatively localized, so that diffusion would still be the rate-limiting step. However, an enhanced diffusivity would result from these localized currents, and therefore the diffusion coefficient in Section 2.3.6.1 must be suitably increased (i.e., by a factor of two) to account for this effect.[8,29]

The isotopic activity release into the atmosphere during a 24-hour period is shown in Table 3, with a total release of 4.6 TBq. However, the probability that the secondary containment is not isolated within a few hours is less than 2×10^{-9}, and the releases will therefore be much less than that quoted in Table 3.[3] Using this calculation for the release into the atmosphere, a subsequent dispersion analysis can be performed to determine the radioactive dose equivalents for the general public (see Section 2.4).

2.4 Atmospheric Dispersion and Dose Equivalent Calculation

The dose equivalent resulting from the release from the submarine has been calculated using the RASCAL computer code.[8,31] A moderately stable atmosphere (i.e., Pasquill condition F) and calm wind speed (e.g., 1 m s^{-1}) is chosen to maximize the dose equivalent in the atmospheric dispersion calculation.

Table 5 provides the dose equivalent for the given source term calculated in Section 2.3. In general, the quantity of release to the atmosphere is typically small, and hence the calculated dose equivalent (~0.5 mSv) is below the annual regulatory limit for the general public (i.e., 5 mSv). The uncertainty associated with the Gaussian plume model in the RASCAL code is ~20% for a ground level release.[32]

3. COMPARISON WITH OTHER WORK

For the inventory calculation in the fuel (see Section 2.3.1), Davis assumed that the NPV was powered by a 100-MW (thermal) reactor, which was similarly operated at an average of 25% of full power for 8 y (compared to 12 y in the present work).[2] Davis scaled a previous ORIGEN calculation for a 1 MW research reactor fuelled with 93% enriched uranium that had operated at full power for one year. To account for the different fuel burnup histories, it was assumed that radionuclides with half-lives less than a year would reach equilibrium in a few weeks, while those isotopes with half-lives longer than a year would grow linearly with time. The assumption for the short half-life isotopes is reasonable and, as expected, the values are in agreement with the present study (accounting for the factor of two difference in the assumed reactor powers). However, the longer-lived radionuclides, which are affected by neutron absorption effects, cannot be easily scaled (i.e., the ^{135}Cs and ^{136}Cs activities are significantly underpredicted while the ^{135}Xe activity is overpredicted by a factor of about two). However, despite these discrepancies, the total inventory calculated by Davis is in general agreement with the present study.

The release fractions in Section 2.3.2 are similar to those used by the BEREX TSA (see Table 4). For the elements contributing to the largest dose equivalent (i.e., iodine and noble gases), the same values have been used. Although the release fractions used by Davis are generally smaller (except for the noble gases), these do not specifically correspond to the fuel itself. Rather, these quantities are composite values for release directly to the atmosphere since Davis does not take credit for any physical holdup processes in the RCS, reactor compartment or submarine hull. These holdup processes have been treated more physically in the BEREX TSA and present study.

Due to uncertainties in the physical processes that may be responsible for FP retention on internal surfaces, and the lack of proper analysis tools, the BEREX TSA study neglected the effect of deposition in the RCS. A sophisticated analysis in this study, performed with VICTORIA, supported this assumption where only ~10% of the FPs were deposited onto RCS surfaces (see Section 2.3.3).

The FISSCON computer code was used to calculate plate-out in the containment atmosphere for the BEREX TSA study.[23] This code was verified against data from the CSE test facility, and since the present calculations also use the CSE model (see Section 2.3.4.1), the results should be similar. However, the half-life calculated for iodine plate-out are somewhat different (i.e., 700 versus 430 s). This discrepancy is believed to result from the use of a smaller surface area in Eq. (2) (i.e., 370 m^2 versus 470 m^2). A smaller half-life in this study will lead to a faster removal of iodine from the containment atmosphere. However, after two hours, the calculated release rates are similar since both studies assume that plate-out stops when the concentration of iodine reaches 1%. Both studies also assume that 2% of the iodine does not plate-out. Aerosol losses due to gravitational settling have been neglected in the BEREX TSA analysis as well (see Section 2.3.4.2). Thus, the estimated quantity of FPs in the primary containment atmosphere will diminish more slowly in the BEREX TSA study than in the present one, where removal will only occur by radioactive decay and leakage.

The PRESCON code was used in the BEREX TSA study to calculate the release rates from the primary containment. However, it is unclear why the leakage rate in this study (see Section 2.3.5) is slightly higher than the PRESCON estimate.

The activity release into the atmosphere varies among the given studies. In particular, Davis took no credit for any containment and therefore over-predicted the atmospheric release by up to four orders of magnitude, compared to the other studies. A more realistic approach was used in the BEREX TSA study, except for the fact that aerosol deposition was not taken into account. In fact, most assumptions are similar in both the BEREX TSA and present work. However, the methodology for FP transport in the submarine hull diverge in these two studies (see Section 2.3.6). The BEREX study does not consider physical transport in the hull. Rather, an instantaneous volumetric release rate of 1% per day was assumed to leak from the submarine hull into the atmosphere. Nevertheless, the total amount of activity release in a 24-hour period is similar to the diffusion estimate of Section 2.3.6, although there is an important difference in the timing of this release. With the diffusional transport model, little release occurs in the early stages of the accident due to the requirement of a physical transport time for migration down the hull; on the other hand, an instantaneous release rate of 1% per day leads to an immediate release of FPs to the environment, which is clearly over-conservative and has no physical basis.

As expected, the predicted dose equivalents are proportional to the atmospheric activity release. Hence, the Davis study predicts the highest dose equivalents after 24 hours (see the comparison in Table 5). The values shown for the BEREX TSA study employ the BEREX TSA source term with a RASCAL dispersion analysis. These calculations are comparable to the present estimates.

4. SUMMARY

A best-estimate analysis has been conducted to determine the consequences of fission product release to the environment from a large-break loss-of-coolant accident aboard a (reference design) nuclear-powered submarine. Based on a reactor irradiation history that maximizes the inventory of both the short and long half-life radioisotopes, a core inventory of 21 EBq is calculated with the ORIGEN2 code. Using available release fraction data for U-Zr alloy, U-Al alloy, uranium metal and uranium dioxide fuel, an activity of 2.2 EBq is estimated for release from the fuel core (of which the iodine and noble gas species constitute ~80% of this source activity). A VICTORIA code analysis indicated that only ~10% of the FPs are retained in the RCS; however, for conservatism, retention in the PHTS was not taken into account in the current estimates. Iodine removal in the reactor compartment atmosphere was determined with a CSA scale model for plate-out (yielding a deposition rate of $0.0016 \, s^{-1}$), where it was assumed that only 98% of iodine is in a volatile form. Gravitational settling of aerosols was also considered for FP removal in the primary containment atmosphere. Using the particle diameter as calculated in a VICTORIA analysis, the aerosol deposition rate is estimated to be $0.00070 \, s^{-1}$. Previous thermalhydraulic calculations indicated that the integrity of the reactor compartment is maintained, and hence release will only occur by leakage (i.e., at a volumetric leakage rate of 1% per day for a design pressure of 2 MPa). The total activity release into the secondary containment in 24 hours is estimated to be 1.1 PBq. Subsequent transport in the submarine hull was also considered, where localized convective currents resulted in an enhanced vapour diffusivity. The total activity release into the atmosphere during a 24-hour period is estimated at 4.6 TBq. This value is comparable to the source term obtained in the BEREX TSA study (8.4 TBq), but is four orders of magnitude less than that obtained in the Davis study, because no credit was taken for containment in the latter study.

Atmospheric dispersion and dose equivalent estimation was performed with the RASCAL code. The maximum (individual) dose equivalent at 800 metres from the submarine berthing location is 0.5 mSv, which is well below the annual regulatory limit for the general public. This dose equivalent has been calculated for a 24-hour release period in which the secondary containment is not isolated due to an open hatch; however, the probability of such an event is extremely low (2.4×10^{-9} per year) and limited release is therefore expected.

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Table 1. Summary of Submarine Reactor Core Accidents[a]

Country	Date	Submarine	Location Sunk (Not Recovered)
USA	10 April 1963	SSN-593 Thresher	Atlantic Ocean (100 miles E Cape Cod, 2590 m depth); 1.15×10^6 GBq total radioactivity
	21 May 1968	SSN-589 Scorpion	Atlantic Ocean (400 miles SW Azores, <3000 m depth); 1.30×10^6 GBq total radioactivity
USSR	11 April 1968	Submarine[b]	Pacific Ocean (750 miles NW Oahu Island)
	1958	Submarine[b]	NE Atlantic Ocean
	12 April 1979	Submarine[b]	NE Atlantic Ocean
	June 1983	Submarine[b]	NW Pacific Ocean (off Kamchatka Penninsula)
	4 October 1986	Submarine	Atlantic Ocean (Bermudas, 6000 m depth)
	7 April 1989	Kosmsomolets	Spitzbergen Medvezhy Island (180 km SW Bear Island, 1500 m depth)

(a) Taken from Ref. 1.
(b) Unconfirmed.

Table 2. Main Event Sequences During the Large-Break LOCA[a]

Time (s)	Events
0	Instantaneous guillotine break at the inlet pipe between the reactor vessel and the loop isolation valves. Mass and energy discharges into containment starts.
< 0.1	Core flow reverses and primary system pressure decreases to the saturation pressure.
0.5	Reactor is shutdown due to low primary system pressure.
6	Pressurizer becomes empty.
10	Primary pressure reaches equilibrium with the containment pressure. Primary loop and core coolant inventory becomes very small (average void fraction >90%). Some fuel rod temperatures start to rise due to fuel clad dryout.
10-100	Fuel and clad temperatures continue to rise. Coolant in some parts of the core becomes superheated.
260	At several core locations, the rate of the fuel temperature increase starts to accelerate due to a lower convective heat transfer and heat production from zirconium-steam reaction.
276	Some fuel and clad melting occurs and falls to the bottom of the pressure vessel. Fission products start to be released from the molten fuel. Temperature of bottom part of pressure vessel starts to rise following contact with the corium.
1064	Pressure vessel failure is predicted due to contact with the corium (corium accumulated mass is 2900 kg). The corium begins to contact the shield tank, and the shield tank wall temperature begins to rise.
20000	Stored heat and fission product decay heat generated in the corium are completely removed by: (i) radiation from the top surface of the corium to the surrounding material; (ii) shield tank water heat exchanger; (iii) heat transfer to the sea water via conduction through the hull. Shield tank integrity is preserved.

(a) Taken from Ref. 6.

Table 3. Core Inventory and Release to Secondary Containment and Atmosphere

Isotope	Core Inventory ($\times 10^{16}$ Bq)	Cumulative Fission Product Release (after 24 h)	
		Release to Hull ($\times 10^{11}$ Bq)	Release to Atmosphere (Bq)
Se81	0.669	1.50	1.89×10^{2}
Se83	0.6334	1.57	1.32×10^{3}
Br83	2.626	10.3	4.74×10^{8}
Kr83m	3.092	114	1.09×10^{9}
Se84	3.068	0.592	< 3.7
Br84	3.273	7.73	1.14×10^{5}
Br85	3.889	0.459	< 3.7
Kr85	0.286	73.3	9.77×10^{10}
Kr85m	6.827	566	7.77×10^{10}
Rb86	0.02907	0.0699	1.41×10^{8}
Kr87	10.61	238	3.89×10^{8}
Kr88	18.10	951	4.07×10^{10}
Rb88	19.03	18.6	1.47×10^{3}
Kr89	14.24	2.94	< 3.7
Rb89	14.02	8.99	1.49×10^{2}
Sr89	8.621	3.96	8.10×10^{9}
Rb90	14.66	0.599	< 3.7
Sr90	2.207	1.01	2.08×10^{9}
Sr91	33.75	12.2	8.62×10^{9}
Sr92	29.59	8.36	5.62×10^{8}
Sr93	19.63	0.770	< 3.7
Nb97	35.45	3.77	1.10×10^{7}
Nb97m	32.87	0.263	< 3.7
Mo99	27.43	4.85	8.47×10^{9}
Tc99m	23.48	3.25	1.27×10^{9}
Tc101	16.17	0.778	< 3.7
Ag109m	0.4699	0.132	< 3.7
Ru103	6.020	0.544	1.11×10^{9}
Ru105	5.776	0.366	8.36×10^{7}
Rh105	5.365	0.455	6.96×10^{8}
Sn127	0.4814	2.57	7.77×10^{7}
Sb127	0.6386	5.70	1.05×10^{10}
Te127	0.5535	4.11	2.86×10^{9}
Sn128	1.429	5.99	5.59×10^{6}
Sb128m	1.528	3.40	< 3.7
Sb129	3.695	23.3	5.22×10^{9}
Te129	3.694	1.91	4.22×10^{6}
Te129m	0.1956	1.79	3.64×10^{9}
Sb130	0.7618	2.66	1.58×10^{5}
Sb130m	3.608	3.32	< 3.7
Sn130	2.926	0.710	< 3.7
I130	0.8569	4.66	1.36×10^{8}
Sb131	8.125	20.8	2.34×10^{4}
Te131	9.302	31.7	7.77×10^{4}
Te131m	2.051	17.4	2.58×10^{10}
I131	8.303	68.8	5.11×10^{9}
Sb132	5.169	1.03	< 3.7
Sb132m	3.308	0.751	< 3.7
Te132	18.21	162	2.90×10^{11}
I132	18.24	50.0	2.49×10^{7}
Sb133	6.945	0.662	< 3.7
Te133	12.08	21.1	2.45×10^{1}
Te133m	11.06	45.5	2.90×10^{7}
I133	39.33	253	1.10×10^{10}
Xe133	20.25	6110	3.53×10^{12}
Xe133m	0.8484	234	1.20×10^{11}
Te134	23.58	85.5	9.47×10^{6}
I134	33.96	63.6	1.15×10^{5}
Cs134	2.505	6.96	1.43×10^{10}
Cs134m	1.403	2.42	2.00×10^{8}
I135	35.16	147	1.80×10^{9}
Xe135	7.981	1300	2.41×10^{11}
Xe135m	6.046	33.8	7.07
Cs136	1.039	2.85	5.66×10^{9}
Xe137	18.56	52.2	2.22×10^{5}
Cs137	2.302	6.36	1.31×10^{10}
Xe138	19.35	57.4	3.77
Cs138	23.50	25.8	5.48×10^{5}
Cs139	19.51	6.51	< 3.7
Ba139	27.95	6.96	3.85×10^{7}
Ba140	15.41	7.03	1.40×10^{10}
Ba141	18.27	1.95	2.20×10^{2}
Ba142	17.86	1.10	< 3.7
Ce143	32.74	0.269	4.03×10^{8}
Total	2067	11000	4.55×10^{12}

Table 4. Comparison of Release Fractions

Group Name	Chemical Elemnts	Fuel Type	Measured FP/Actinide	Measured Release Fraction (%)	Assumed Release Fraction (%)		
					This Study	BEREX TSA	Davis
Noble Gases	Kr, Xe	U-Zr alloy	Kr, Xe	100	100	100	100
Halogens	Br, I	U-Zr alloy	I	up to 66	70	70	10
Alkali Metals	Rb, Cs	U-Zr alloy	Cs	up to 24	30	40	10
Chalcogens	Te, Se	U-Al alloy U Metal UO_2	$Te^{(a)}$	up to 68 up to 79 $99^{(b)}$	100	5	10
Alkaline Earths	Sr, Ba	U-Zr alloy	Sr, Ba	up to 1 (Ba) up to 5 (Sr)	5	3	1
Platinoids	Ru, Rh, Pd	U-Al alloy U Metal UO_2	Ru	<1 0.01 $5^{(b)}$	1	1	10
Early Transition Elements	Mo, Tc Nb	UO_2	Mo	$\leq 2^{(b)}$ $86^{(c)}$	2 2	1 0.1	1 1
Tetravalents	Zr, Ce, Np, Pu	U-Zr alloy	Ce	≤0.05	0.1	0.1	1
Trivalents	Y, La, Pr, Nd, Pm, Sm, Eu, Am	UO_2	Eu	$<0.1^{(d)}$ $<0.6^{(e)}$	0.1	0.1	1
Uranium	U	UO_2	U	up to $0.7^{(b)}$	1	0.1	1
More Volatile Main Group	Sb, As Cd, Zn	UO_2	$Sb^{(a)}$	up to 99 [illegible]	100 100 100	5 5 100	1 n/a n/a
Less Volatile Main Group	Ga, In Sn, Ge,Ag	UO_2	$Ag^{(a)}$ $Sn^{(a)}$	up to 84 [illegible] up to 94 [illegible]	100 100	100 1	n/a n/a

(a) Affected by fuel oxidation state.
(b) Measured in VI-3 steam test at 2431°C.
(c) Measured in VI-2 steam test at 2029°C.
(d) Measured in HI-1 steam test at 1402°C.
(e) Measured in HI-4 steam test at 1927°C.
(f) Measured in VI-1 steam test at 2027°C.

Table 5. Comparison of Individual Dose Equivalents at 800 Metres

Dose Equivalent Contibution	Davis (mSv)	BEREX TSA[(a)] (mSv)	Present Study (mSv)
Thyroid		2.4	3.2
Inhalation	380	1.5	0.30
Cloudshine	22	< 0.01	< 0.01
Groudshine	600	0.12	0.16
Total	6400	1.6	0.46

(a) Dose equivalents calculated with the RASCAL computer code using the BEREX TSA source term after 24 hours.

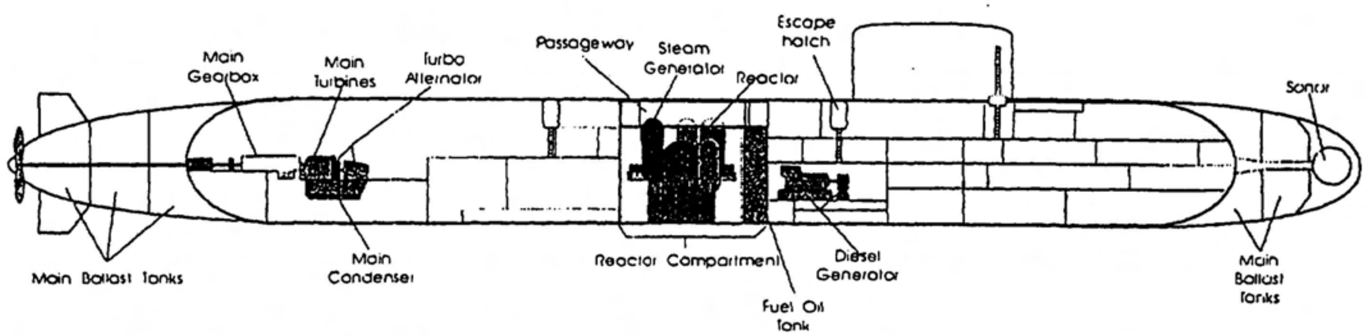

Figure 1. Schematic of a typical nuclear-powered submarine.

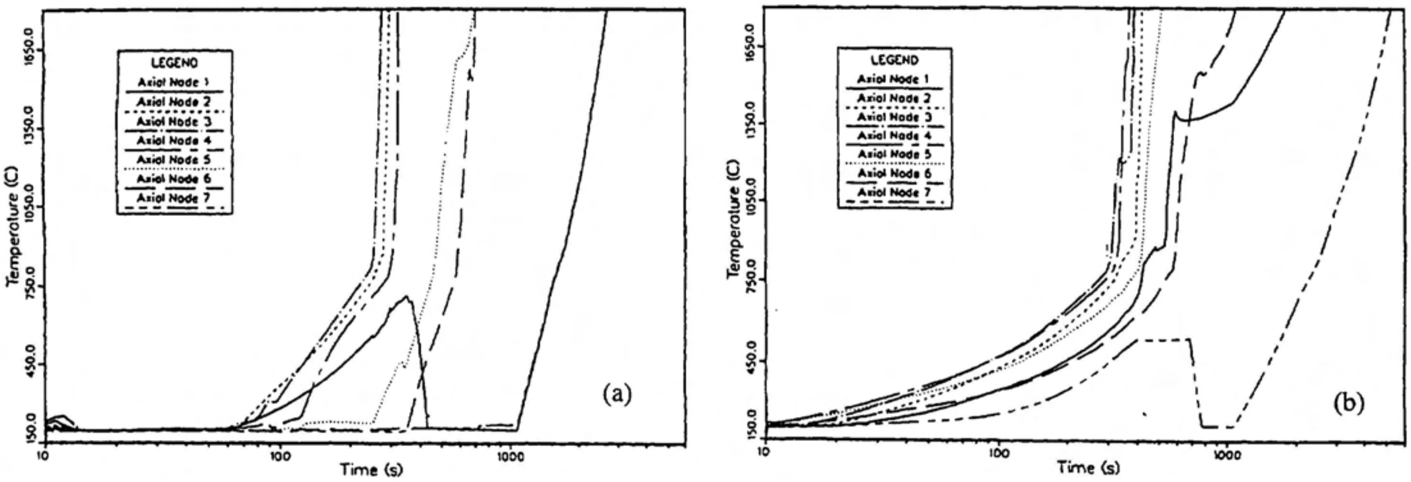

Figure 2. Fuel temperature transient for a guillotine inlet (cold leg) break. Node 1 = bottom, Node 7 = top.

(a) Inner core temperature

(b) Outer core temperature.

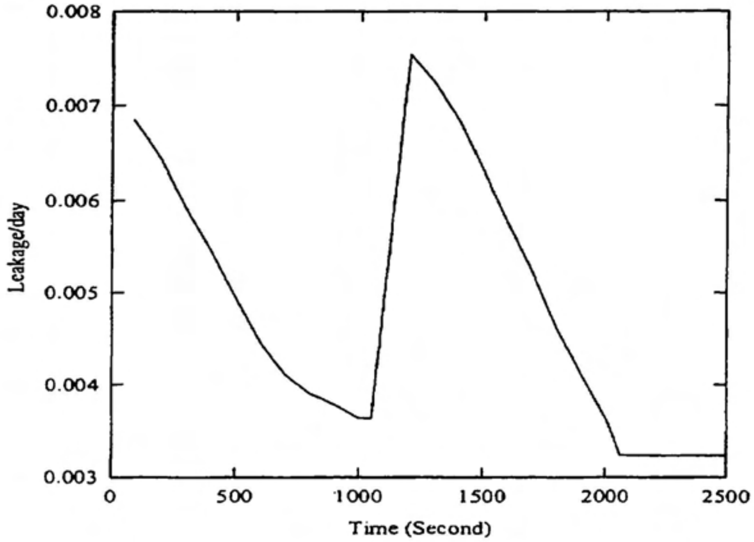

Figure 3. Leakage rate from the reactor compartment.