

Titolo

FAST-1: Evaluation of the Fukushima Accident Source Term through the fast running code RASCAL 4.2: Methods & Results

Descrittori

Tipologia del documento:

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Sommario

This report presents the methodology and the hypotheses used to estimate the Source Term of the Fukushima accident using the RASCAL 4.2 code. The results obtained are compared with other available numerical evaluations as well as with some experimental data.

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1. Introduction

The Fukushima accident prompted and triggered in all countries a series of activities targeted at the building and/or strengthening of those capacities needed for the rapid assessment of off-site consequences following a severe nuclear accident. The worldwide general framework is that of Emergency Preparedness and Response. At ENEA this is conjugated presently in the creation or strengthening of the capabilities devoted to the:

- rapid direct estimation of Source Terms;
- rapid evaluation of direct atmospheric transport;
- rapid evaluation of dose to the population;
- inverse modeling.

The development of dedicated tools to deal with the foreign NPPs which are at less than 200 km from the Italian national border is also a priority for ENEA. In this regard, the use of the Fukushima accident as a benchmark for assessing the presently available capabilities is extremely useful; while it is on one side true that still much is to be done to know with high accuracy the Source Term (ST) of the Fukushima accident, on the other side it must be recognized that at least the order of magnitude of the releases to the atmosphere can be taken as known.

This Report presents the results of the application of the fast-running US-NRC direct code RASCAL 4.2 to the estimation of the Fukushima Source Term. By direct it is intended here an approach which requires no knowledge of the experimentally measured values of gamma dose rates to be used to make a backward calculation of the emission source. Inverse methods instead rely on measured data and inverse atmospheric transport to estimate the Source Term and don't rely too much on the knowledge of the accident sequence.

The complete methodology to carry out such direct calculations with RASCAL 4.2 is presented together with the hypotheses introduced; the results obtained and the comparison of these with the other publicly available Source Terms are included. The results are in agreement, thus suggesting the correctness of the hypotheses on the accident sequence (nature of material emitted during venting).

In November 2013 RASCAL version 4.3 was released with many new and important features derived from the experience gained from the Fukushima accident. For example RASCAL 4.2 can follow an accident progression, with emissions and atmospheric transport, only for 48 hours since the first core uncover; RASCAL 4.3 has instead been extended to 96 hours. RASCAL 4.2 has a pre-built standard inventory formula for core inventories while RASCAL 4.3 can be provided with a user-calculated inventory. RASCAL 4.3 has also been provided with improved Source Term modeling for situations like the prolonged Complete Station Blackout and other recent US-NRC SOARCA-derived Source Term phenomena. Multi-unit treatment has been included, as well as the possibility to import Source Terms calculated via other routes (i.e. best-estimate codes) and later used only for atmospheric transport and dose estimate. In this regard it is envisaged to prepare in the near future a Report, similar to the present one, containing the results which can be obtained by RASCAL 4.3, in order to compare the two versions of the code.

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2. Methods used in RASCAL 4.2

The RASCAL 4.2 code has been developed by Athey Consulting for the US-NRC as the reference tool for the fast-running calculations of radiological consequences of severe accidents to any plant related to civilian uses of nuclear energy [1,2]. Version 4.2 was distributed in March 2012 and incorporated some partial feedback from the Fukushima accident.

The most important module of RASCAL 4.2 concerning the estimation of a time-dependent source term is the **Source Term to Dose** module, which is largely based on the accumulated US experience on PWRs and BWRs as described in NUREG-1228 [3] and NUREG-1465 [4] and in the literature therein quoted.

The module requires the specification of some parameters in order to calculate the dose to the population; these parameters are input to the following sub-modules:

- **Event Type**
- **Event Location**
- **Source Term**
- **Release Path**
- **Meteorology**

If the interest is on the ST alone, certain parameters for the **Event Location** sub-module and all the parameters needed for the **Meteorology** sub-module are not influential.

The **Event Type** sub-module requires the choice between the following types of plants:

- NPP
- Spent Fuel
- Fuel Cycle/UF6/Criticality Event
- Other types of Releases

In the case of the Fukushima accident the choice was NPP. The **Event Location** sub-module is used to locate in space the ST for the subsequent dose calculations (this being out of the scope of the present Report) and to load all the necessary plant data in order to evaluate Activity Inventory and ST. All the relevant data for US LWRs are already present in an internal database, however it is also possible to define a generic NPP if the relevant data are available. As will be discussed later, our procedure to estimate the Fukushima ST involves the use of *s.c. surrogate NPPs* and there is no need to input by hand all the data of the Daiichi plants; basically, only the reactor power and the average core burnup are given for the real Daiichi plants. The **Source Term** sub-module is used to define the calculation route for the ST. Several possibilities are offered:

- Time to core uncover
- Specified core damage endpoint
- Containment radiation monitor
- Coolant sample
- Containment-air sample
- Effluent release rates
- Effluent release concentrations
- Effluent release by mixtures

The choice is dictated by what is known of the accident itself. For Fukushima it was selected the Time to core uncover option. This basically assumes that no significant releases are produced until the

core starts to get uncovered. For PWRs this means the uncovering of the TAF (Top of Active Fuel), for BWRs, given their typical two-phase flow nature, this means the uncovering of about 1/3 of the active fuel (i.e. up to about 2/3 of TAF level). The only information that the user needs to input is the SCRAM date and time, and the estimated date and time for uncovering. In the case that the water level returns to the original value, after the core has been partially uncovered, the user is prompted to input the estimated date and time of re-covering. These temporal data can be derived from the user knowledge of severe accidents, from the user's own experience, from real-time calculations of an accident progression or by other ways. The re-covering date and time are needed to improve the calculation and to produce better and more realistic results; however for conservative estimates one can assume that the core is not recovered at all. It should also be noted that core re-covering has an important effect in lowering the ST only if it occurs within the first 3-5 hours, otherwise the radiologically important nuclides of the ST may be considered released as if there were no re-covering at all. It must also be kept in mind that the Time to core uncovering option tends to slightly overestimate the ST [1], and this even more for BWRs, given the fact that secondary buildings (certainly not of containment type, but nonetheless still contributing to a lesser extent to release reductions) of these plants have not been modeled at all in the RASCAL 4.2 version.

The **Release Path** sub-module is most important. It is used to define the pathways through which activity is released to the atmosphere. It is of course plant-dependent (i.e. different between PWRs and BWRs). This sub-module requires also the input of the accident sequence, naturally limited to the date and time of those events directly related to releases (ventings, explosions etc.). These events are declared in advance with time, whenever possible, during the accident management (ventings), or are known in real-time (explosions). It must be said that a very precise knowledge of the most important release pathway is required; this of course may differ from the official and intended release pathway because of the plant status and condition. As it will be discussed later, it is plausible that the ventings that TEPCO announced during the accident as being conducted from the wetwell were, as a matter of fact and because of the degraded conditions of the plants, conducted actually from the drywell.

It should also be understood that more than one release pathway is available during accidents, however RASCAL 4.2 requires the selection of the most important one with respect to the amount of activity released. For example, if both drywell and wetwell paths are simultaneously available, then only drywell should be used, given the fact that the wetwell properties imply releases which can be several orders of magnitude lower than those from the drywell; the wetwell pathway simply becomes negligible in comparison.

Limiting the discussion to Mark-1 BWRs, the release pathway is splitted in two parts, namely the path from primary containment to secondary containment, and the path from secondary to the atmosphere. Choices for the first term of the release pathway are:

- Through wetwell
- Through drywell
- Bypass of reactor building (direct to the atmosphere)

Choices for the second term of the release pathway are:

- Through the Stand-by Gas Treatment System (SBGTS) and Stack (filtered path)
- Direct from reactor building or other rapid non-filtered releases (typical for unfiltered venting or explosions).

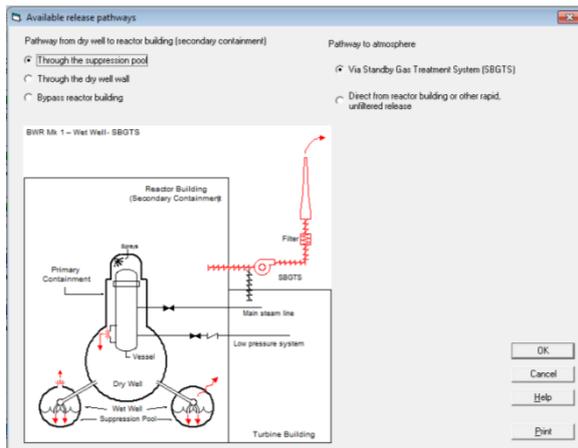


Fig. 1. Wetwell-SBGTS release pathway for a Mark-1 BWR.

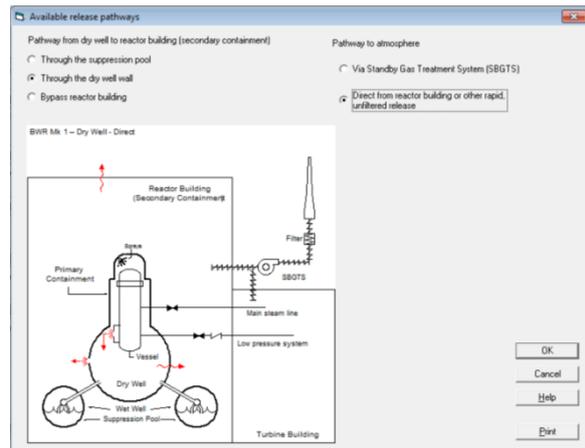


Fig. 2. Drywell-Direct release pathway for a Mark-1 BWR.

In Fig. 1 is shown in red the Wetwell-SBGTS release pathway for a Mark-1 BWR, while in Fig. 2 is shown, again in red, the Drywell-Direct release pathway for the same plant.

All these data are used to evaluate rapidly the total core and coolant inventories, their time decay, and their time-release to the atmosphere (i.e. ST) using predefined average coefficients or release fractions as explained in [1,2,3,4].

3. General remarks on Surrogate Plants and Release Pathways for the Fukushima ST

The first step in the calculation of an ST is the choice of a surrogate plant. We call a surrogate plant, a plant already available in the RASCAL 4.2 database of US plants which differs from the real plant only as regards actual power and actual core average burnup. In practice this means to find among the US fleet some BWR Mark-1 plants which can be used to mock-up the Fukushima plants [1].

We choose Duane Arnold as surrogate for Daiichi 1, and Cooper as surrogate for Daiichi 2 and 3. Data pertaining to Duane Arnold and Cooper are given in Table 1.

The second step is to prepare a reliable sequence of emission events; these will be discussed in depth in the next sections for each Daiichi unit. This however involves also the choice of the most realistic release path. For the Fukushima accident analysis we make recourse to an a posteriori comparison with the ST estimates already available; in a real emergency case, experience (i.e. expert judgement) is the main guidance.

In Table 2 are reported rough calculations for two isotopes (I-131 and Cs-137) for 4 different combinations of release pathways; by comparing the results with an already available plausible estimate, it can clearly be seen that the most probable path is the combination of Drywell+Direct option. The Direct option is self-evident, because the largest part of the emissions were through ventings and/or explosions; the adoption of the Drywell option is not so immediate, as will be discussed later. In the calculations the user has to choose values for the release rates for ventings and explosions, and their durations. In practice explosions are assimilated to and mocked-up with controlled leakages through the primary containment; this is coherent with the fact that secondary buildings for BWRs are not modeled in RASCAL 4.2. This of course doesn't mean that the primary building actually exploded, but simply that the explosions of the secondary buildings have the same net effect in terms of emissions of a large direct leakage from the primary. This has of course an impact on the effective inventory release ratios.

Typical average values, used throughout this report unless otherwise explicitly stated, are:

- Ventings: 25%/h for 1 hour;
- Explosions: 50%/h for 1 hour;

Until one of these two events happens, one must also specify a release rate for leakages through the reactor buildings. For an intact building, typical values are about 0.5%/d for an intact core, and about 1%/h at the beginning of core damage, and especially for those NPPs which may have been structurally damaged by a BDB earthquake. After ventings or explosions, the building leak rate is reset to 1%/h.

These values clearly define the time derivative of the emissions.

	Cooper	Duane Arnold
Reactor Type	BWR	BWR
Licensed thermal power limit (MWt)	2419	1912
Ultimate heat sink -		
supply to main turbine condenser	Missouri River	Cedar River / Cooling Towers
supply to ECCS service water system	Missouri River	Cedar River / Cooling Towers
NSSS vendor	General Electric	General Electric
Turbine generator	Westinghouse	General Electric
Architect - Engineer	Burns & Roe, Inc.	General Electric
Constructor	Burns & Roe, Inc.	Bechtel
Operating license issued	1/18/1974	2/22/1974
Commercial operation	7/1/1974	2/1/1975
Containment		
Containment type	Mark-1	Mark-1
Design pressure	56 psig	56 psig
Fuel		
Number of fuel assemblies	548	368
Number of fuel rods per assembly	8 x 8	8 x 8

Table 1. Cooper and Duane Arnold BWRs data.

Emission (Bq)							
RASCAL 4.2	Wetwell	Drywell	Bypass	SGTS	Direct	I-131	Cs-137
	X			X		3.2E11	3.9E10
	X				X	3.2E13	3.9E12
		X			X	3.2E15	3.9E14
			X			1.4E18	1.5E17
Comparison value	-					1.3E17	1.1E16

Table 2. Selection of a release pathway.

As shown later, the true venting path, i.e. from Drywell instead of from Wetwell, is an extremely important issue. In Figs 3 and 4 are reported the schemes of the vent lines of Daiichi units 1 and 2-3 respectively. The two schemes differ basically only as regards the labeling of the components. As

already mentioned, two potential starting points are possible: drywell and/or wetwell. Both lines end in a pressure-driven rupture disk before reaching the stack or other buildings; the pressure needed to break the rupture disk is quite high [5], about 448 kPa for Unit 1 and 427 kPa for Units 2 and 3 [6], and in several instances when TEPCO tried to operate venting, in order to release pressure outside the building through the stack, it proved impossible to break it, resulting in the impossibility to vent to the atmosphere. Moreover it has not yet been possible to identify the real present status of these disks, if broken or not. Between the paths' starting points and the rupture disk several valves are present which should be opened to allow the releases.

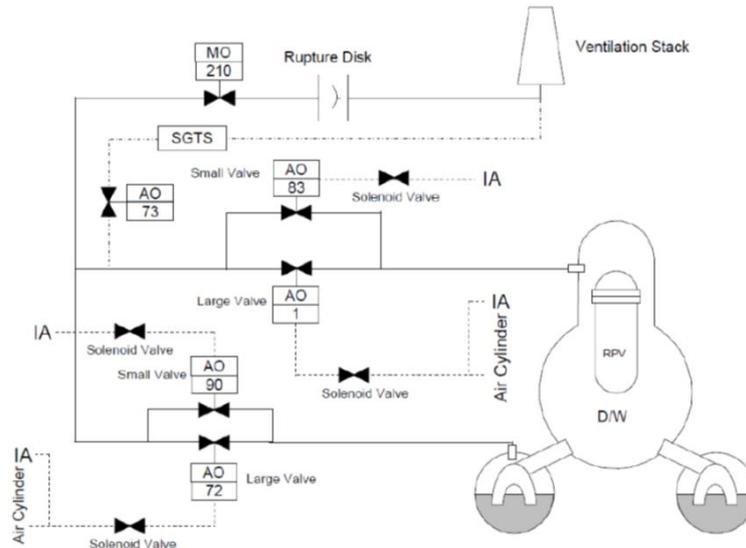


Fig. 3. Fukushima Daiichi-1 vent lines [5].

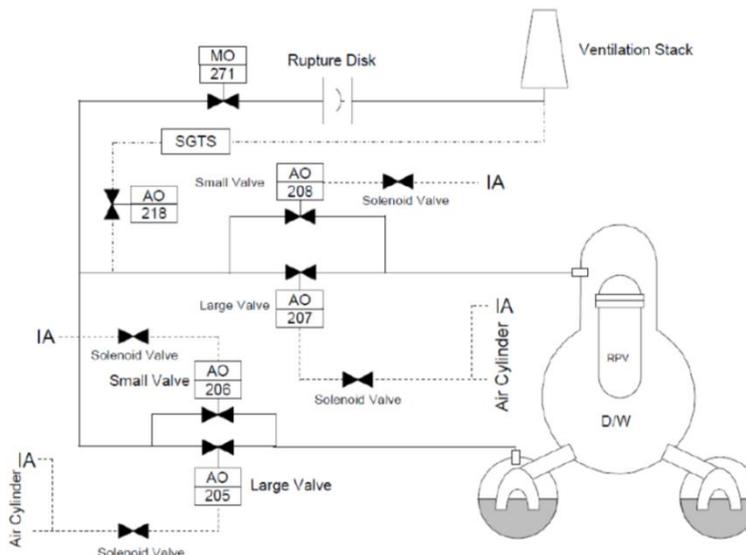


Fig. 4. Fukushima Daiichi-2 and Daiichi-3 vent lines [5].

In the case of CSBO, it was extremely difficult for TEPCO to operate these valves, just because of lack of any AC or DC power and because of the low amount of pressurized air to drive the air-operated valves. It is well known that TEPCO had to make recourse to car batteries and/or transportable air compressors if and when available. Both venting paths have a bypass line: the main pathway is

through a large air-operated valve (W/W: AO-72 for Unit 1 and AO-205 for Units 2 and 3; D/W: AO-1 for Unit 1 and AO-207 for Units 2 and 3); the bypass pathway, used when the main path is impossible to open, is through a smaller air-operated valve (W/W: AO-90 for Unit 1 and AO-206 for Units 2 and 3; D/W: AO-83 for Unit 1 and AO-208 for Units 2 and 3). A common motor-operated valve (MO-210 for Unit 1 and MO-271 for Units 2 and 3) is placed just before the rupture disk. Air-operated valves are connected to the Instrument-Air System (IA). Therefore to properly operate a venting, there should be the simultaneous combination (logical AND) of several conditions: large or small air-operated valve OPEN + motor-operated valve OPEN + pressure HIGH enough to break the rupture disk. TEPCO attempted several times to accomplish all this, but in several cases they didn't properly succeed, because of the well-known dramatic situation, delaying in time an accident management sequence which in principle was correct if timely actuated. The attempts made by TEPCO were mainly to discharge from the W/W vent lines, just because lower releases to the atmosphere are attained through them. However there are many indications that probably the radioactive material escaped from the drywell; this may have occurred without TEPCO's immediate knowledge and because of several factors; for example: structural damages to the pipings connecting drywell to torus room (vent piping bellows), due either to the earthquake, and/or to the too violent pressure and temperature increase in the D/W; leakages through the top head manhole, the top head flange, the piping penetrations, the electrical wiring penetrations, the personal airlocks, the S/C manholes, the machine hatches, etc. [6].

4. Unit 1 Sequence and Modeling

Date	Fukushima Local Time	Event	ΔT since SCRAM		ΔT since SCRAM	Notes	ENEA Hypotheses	ENEA References
			[h]	[min]	[h]			
11/3/2011	14:46:00	SCRAM						
11/3/2011	17:16:00	Core uncover	2	30	2.50	[7]	Core uncover (2/3 TAF) @ 4.5 h	[1,8]
12/3/2011	14:30:00	Start Venting, opening valve AO-72, disk ruptured	23	44	23.73	[5]	Drywell Venting	[1,6]
12/3/2011	15:36:00	Explosion	24	50	24.83	[5]		

Table 3: Unit 1 sequence and notes.

In Table 3 is shown the sequence of RASCAL-relevant events for Unit 1. The first two columns provide date and time, the third column describes the officially-announced event, the following columns provide time lapse since SCRAM using the results provided in the references given in the "Notes" column, the next column shows instead the hypotheses made by ENEA for the calculations, and the last column gives the references used for the ENEA hypotheses.

It is here worth mentioning that Unit 1 had no core re-recovery. Moreover, even if the vent valve AO-72 was opened by TEPCO in order to do a W/W venting, venting is assumed to be instead from D/W, as discussed in the previous paragraph. The rupture disk was indeed broken.

5. Unit 2 Sequence and Modeling

In Table 4 is shown the sequence of RASCAL-relevant events for Unit 2. There is some evidence, see the "ENEA References" column, that the rupture disk was indeed broken, even if this has not yet been officially and unambiguously confirmed by TEPCO.

Also the situation about venting is not crystal clear; TEPCO attempted a venting through valve AO-208, i.e. the small D/W path valve. They attempted a D/W venting just because of the previous negative experience with W/W venting at Unit 1 (in case of failure to open the AO W/W valve, a W/W venting might still be tried through the opening of a manual valve inside the torus room, but

this space was found in Unit 1 to be too dangerous in terms of radioactivity levels). Officially the AO-208 valve remained open for only a very limited amount of time, just because of loss of air pressure, and again officially, nothing is said about the status of the rupture disk which is assumed to be BROKEN for the ENEA analysis.

Date	Fukushima Local Time	Event	ΔT since SCRAM		ΔT since SCRAM	Notes	ENEA Hypotheses	ENEA References
			[h]	[min]	[h]			
11/3/2011	14:46:00	SCRAM						
14/3/2011	16:16:00	Core uncover	73	30	73.50	[1,7]		
15/3/2011	0:02:00	Start Venting, opening of valve AO-208, disk status unknown	81	16	81.27	[7]	Disk ruptured	[1, 9 pg. 39, 10 pg. 68, 6 part 2 pg. 61]
15/3/2011	0:12:00	End Venting, closure of valve AO-208	81	26	81.43	[7]		

Table 4: Unit 2 sequence and notes.

It is highly probable that the core of Unit 2 was recovered. In [8] several timing schemes have been reported concerning the core reflooding; three cases were advanced, with uncover durations of 1.5, 3.0 and 4.5 hours respectively (water reference level: 2/3 TAF). In the RASCAL calculations it was decided to adopt 4.5 hours.

6. Unit 3 Sequence and Modeling

The modeling of Unit 3 is the most challenging one, because of the complicated sequence of trials of ventings, not always successful and not always precisely known, and because of the not so clear status of the rupture disk.

Date	Fukushima Local Time	Event	ΔT since SCRAM		ΔT since SCRAM	Notes	ENEA Hypotheses	ENEA References
			[h]	[min]	[h]			
11/3/2011	14:46:00	SCRAM						
13/3/2011	8:46:00	Core uncover	42	0	42.00	[7], [1] @ $\Delta T= 37.50$	[7]	
13/3/2011	9:20:00	Start Venting, opening valve AO-205, disk ruptured/removed	42	34	42.57	[5]	Drywell Venting	[1,6]
13/3/2011	11:17:00	End Venting, closure AO-205	44	27	44.45	[5]		
13/3/2011	12:30:00	Start Venting, opening valve AO-205	45	40	45.67	[5]	Neglected	
13/3/2011	15:00:00	End Venting, closure AO-205	48	10	48.17	[5]		
13/3/2011	20:10:00	Start Venting, opening valve AO-205	53	50	53.83	[5]	Neglected	
14/3/2011	1:00:00	End Venting, closure AO-205	58	40	58.67	[7]		
14/3/2011	6:10:00	Start Venting, opening valve AO-206	63	50	63.83	[5]	Neglected	
14/3/2011	6:10:00	End Venting, closure AO-206	63	50	63.83	[5]		
14/3/2011	11:01:00	Explosion	68	41	68.68	[5]		

Table 5: Unit 3 sequence and notes.

In Table 5 is reported the sequence of RASCAL-relevant events for Unit 3.

Three possible venting events were neglected because it is not clear if really the line valves were successfully opened or not.

The core of Unit 3 was most probably recovered; [8] gives two possible uncover durations (2/3 TAF), namely 2.5 hours and 6.0 hours. In the RASCAL calculations it was decided to adopt 2.5 hours.

7. Results and Discussion

As an example of the results which were obtained with RASCAL 4.2, the time evolutions of the total (i.e. 3 Units) cumulative activities of I-131 and Cs-137 released to the atmosphere are shown in Figs 5 to 7. Semilog scale is used in Fig. 5 for comparison purposes; linear scale is used in Figs 6 and 7 to better catch the absolute values of each single emission. The arrows indicate for better reference the

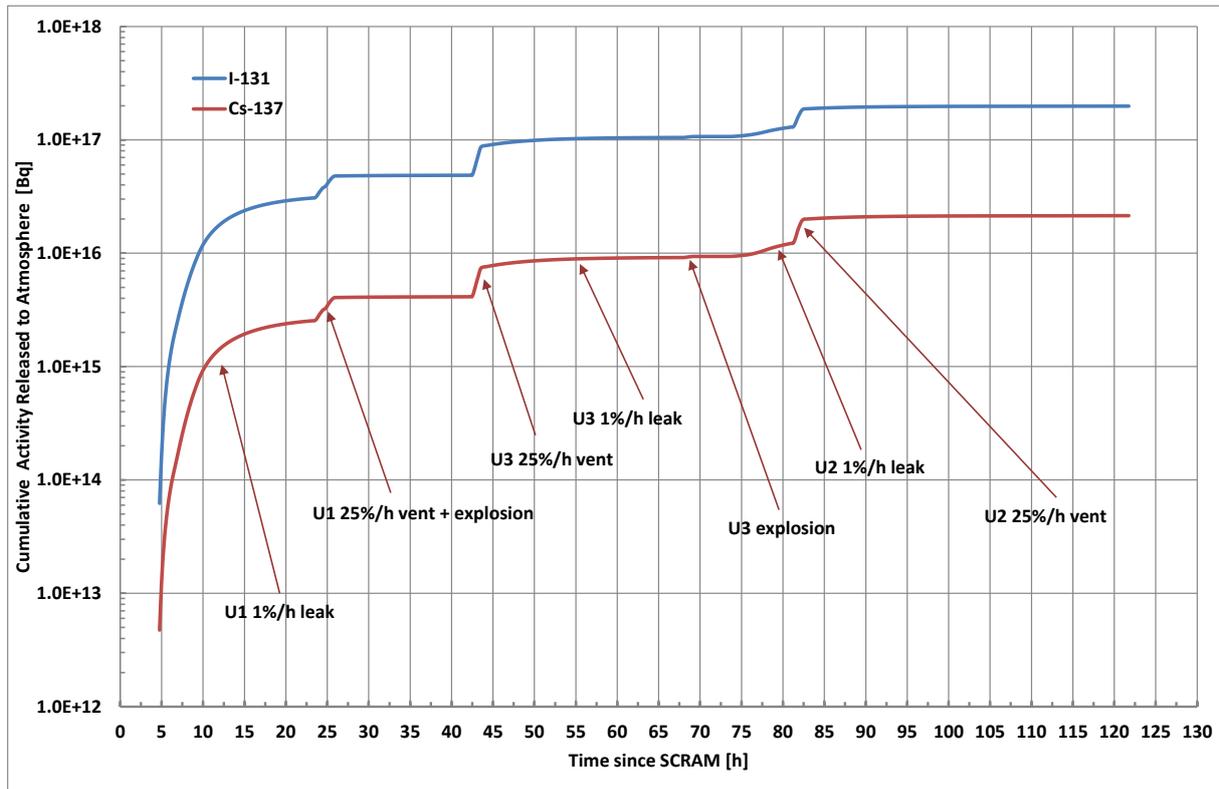


Fig. 5. Total cumulative activity released to the atmosphere of I-131 and Cs-137.

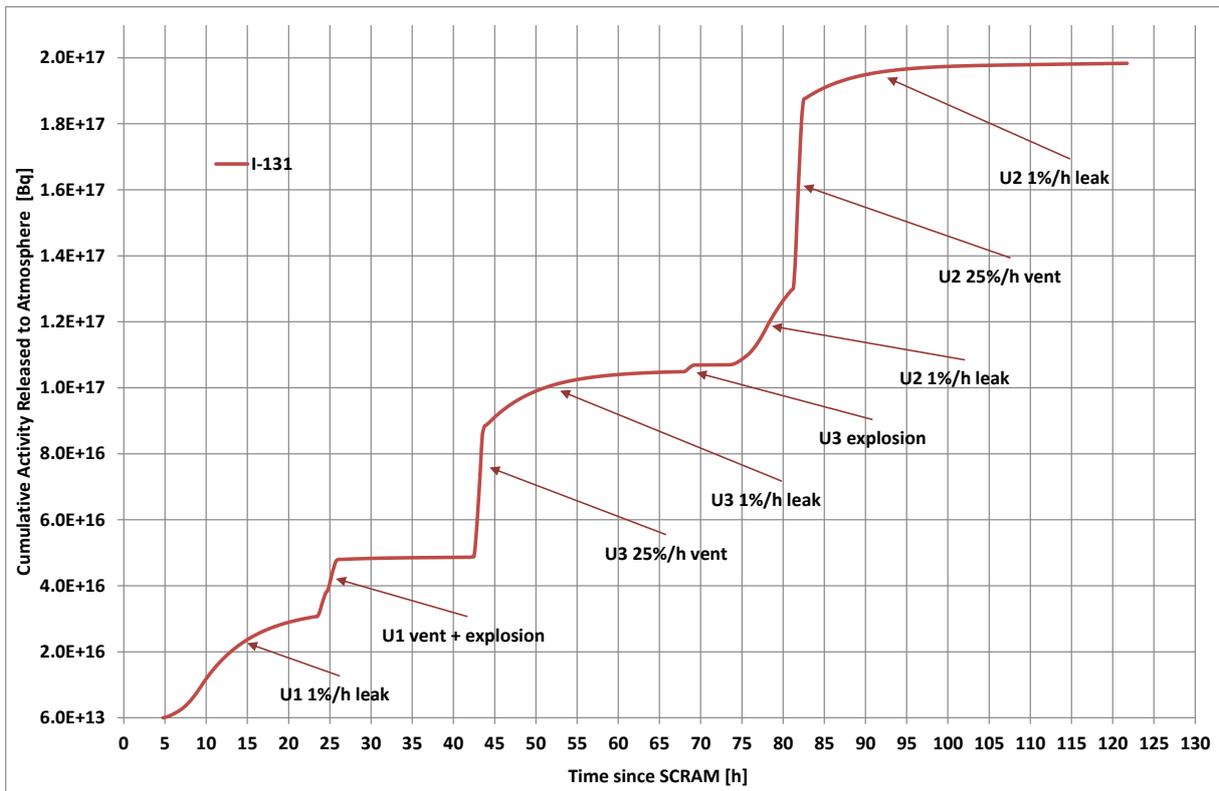


Fig. 6. Total cumulative activity released to the atmosphere of I-131.

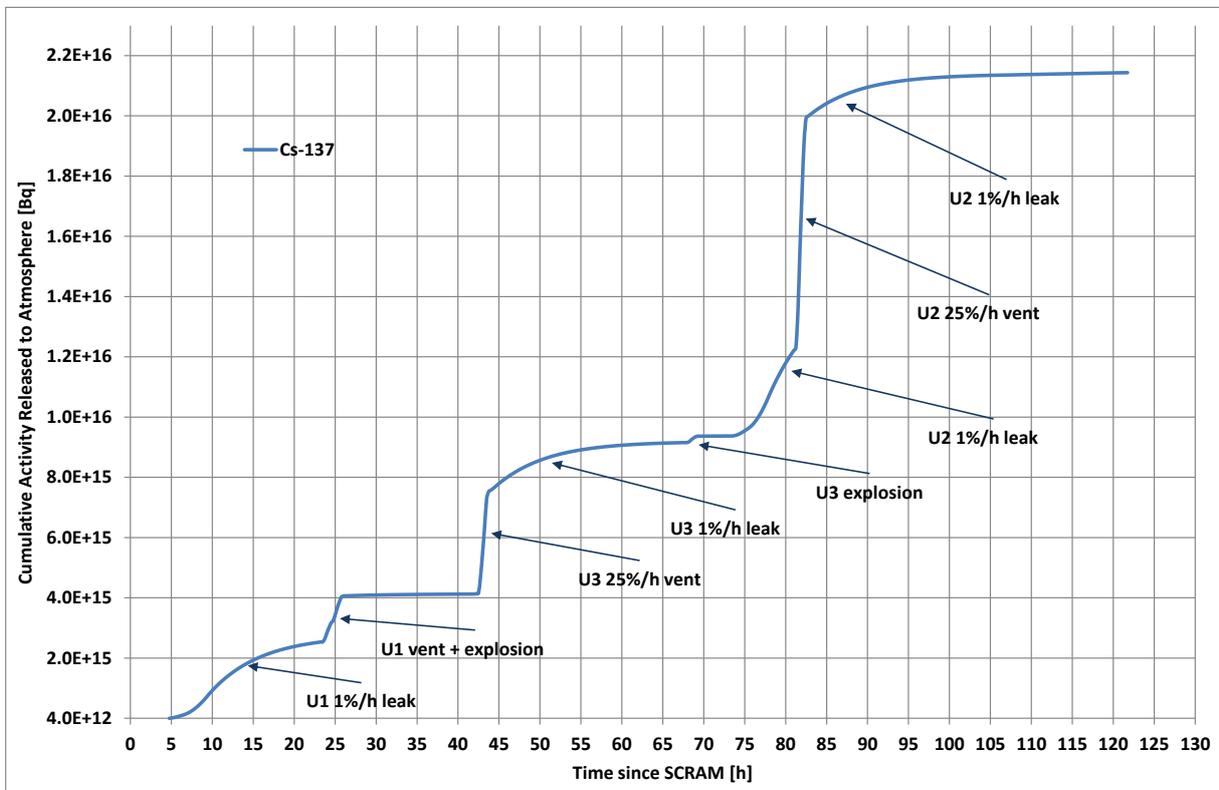


Fig. 7. Total cumulative activity released to the atmosphere of Cs-137.

events that occurred to each unit in their proper chronological order.

In Table 6 are shown the core inventories (Bq) for Units 1 to 3 and total, evaluated as per US-NRC procedure [1].

In Table 7 and Table 8 are shown several already published estimates of the total cumulative STs (Bq) for various isotopes. In Table 7, the “ENEA” columns refer to the present calculations, splitted per unit (1 to 3) and total; the next column shows two values published by the RASCAL developers; then the IRSN direct-method estimate of the ST follows; a subsequent column gives the ratio of the ENEA total to IRSN direct-method values; finally other three columns with IRSN inverse-method estimate, TEPCO direct-method estimate and TEPCO inverse-method estimate are given. In Table 8 are shown the values available for I-131 and Cs-137 as produced by several authors. The column with nuclides has been coloured as follows:

- Red: nuclides with half-lives > 10 days;
- Green: nuclides with half-lives < 2 days;
- White: intermediate half-lives.

It can clearly be seen from the column with ENEA/IRSN ratios that the agreement for nuclides with red colour is very good; ENEA slightly overestimates these nuclides; this overestimate, clearly of systematic nature, of the order of about a factor 2, is probably due to two main reasons:

- RASCAL 4.2 doesn't take into account the effect of secondary building for BWRs [1], which is probably the main contribution to the overestimate;
- RASCAL 4.2 Source Term to Dose module always overestimates the ST [1].

The ENEA/IRSN ratios for the green nuclides is instead much higher than 2; there is an evident reason for this, since these nuclides have half-lives which are lower than the accident progression time-scale. This means that if IRSN considered for these nuclides a release rate to the atmosphere prior to the first venting which was much lower than that used by ENEA (1%/h), then these nuclides in the IRSN simulations are retained within the containment and decay there before any possible release event. The value of 1%/h was chosen by ENEA because of the possible highly damaged conditions of the Fukushima NPPs due to the BDB earthquake. This interpretation may be verified by other RASCAL 4.2 calculations in which the release rate before the first venting is lower than 1%/h.

Nuclide	Unit 1	Unit 2	Unit 3	Total	Nuclide	Unit 1	Unit 2	Unit 3	Total
Kr-83m	1.56E+18	2.69E+18	2.69E+18	6.93E+18	Ba-140	2.43E+18	4.19E+18	4.19E+18	1.08E+19
Kr-85	8.72E+15	1.90E+16	1.90E+16	4.68E+16	Ce-141	2.24E+18	3.87E+18	3.87E+18	9.98E+18
Kr-85m	3.15E+17	5.44E+17	5.44E+17	1.40E+18	Ce-143	2.04E+18	3.52E+18	3.52E+18	9.09E+18
Kr-87	6.28E+17	1.08E+18	1.08E+18	2.80E+18	Ce-144	1.81E+18	3.12E+18	3.12E+18	8.04E+18
Kr-88	8.68E+17	1.50E+18	1.50E+18	3.86E+18	Cm-242	5.72E+16	9.87E+16	9.87E+16	2.55E+17
Xe-131m	1.86E+16	3.22E+16	3.22E+16	8.29E+16	Mo-99	2.71E+18	4.67E+18	4.67E+18	1.20E+19
Xe-133	2.77E+18	4.78E+18	4.78E+18	1.23E+19	Nd-147	8.94E+17	1.54E+18	1.54E+18	3.98E+18
Xe-133m	8.78E+16	1.52E+17	1.52E+17	3.91E+17	Np-239	2.91E+19	5.01E+19	5.01E+19	1.29E+20
Xe-135	7.25E+17	1.25E+18	1.25E+18	3.23E+18	Pr-143	2.02E+18	3.49E+18	3.49E+18	9.00E+18
Xe-135m	5.87E+17	1.01E+18	1.01E+18	2.61E+18	Pu-241	2.18E+17	3.75E+17	3.75E+17	9.68E+17
Xe-138	2.33E+18	4.02E+18	4.02E+18	1.04E+19	Ru-103	2.22E+18	3.82E+18	3.82E+18	9.86E+18
I-131	1.36E+18	2.35E+18	2.35E+18	6.07E+18	Ru-105	1.56E+18	2.70E+18	2.70E+18	6.95E+18
I-132	1.98E+18	3.42E+18	3.42E+18	8.82E+18	Ru-106	4.86E+17	1.06E+18	1.06E+18	2.61E+18
I-133	2.77E+18	4.77E+18	4.77E+18	1.23E+19	Sr-89	1.23E+18	2.12E+18	2.12E+18	5.48E+18
I-134	3.05E+18	5.27E+18	5.27E+18	1.36E+19	Sr-90	7.50E+16	1.64E+17	1.64E+17	4.02E+17
I-135	2.64E+18	4.56E+18	4.56E+18	1.18E+19	Sr-91	1.54E+18	2.65E+18	2.65E+18	6.84E+18
Cs-134	1.47E+17	3.22E+17	3.22E+17	7.91E+17	Sr-92	1.65E+18	2.85E+18	2.85E+18	7.36E+18
Cs-136	7.61E+16	1.31E+17	1.31E+17	3.39E+17	Y-90	1.25E+17	2.16E+17	2.16E+17	5.57E+17
Cs-137	1.02E+17	2.23E+17	2.23E+17	5.47E+17	Y-91	1.62E+18	2.79E+18	2.79E+18	7.20E+18
Rb-86	2.70E+15	4.66E+15	4.66E+15	1.20E+16	Y-92	1.66E+18	2.87E+18	2.87E+18	7.41E+18
Sb-127	1.22E+17	2.11E+17	2.11E+17	5.43E+17	Y-93	1.29E+18	2.22E+18	2.22E+18	5.73E+18
Sb-129	4.43E+17	7.65E+17	7.65E+17	1.97E+18	Zr-95	2.27E+18	3.91E+18	3.91E+18	1.01E+19

Te-127	1.21E+17	2.08E+17	2.08E+17	5.36E+17	Zr-97	2.16E+18	3.73E+18	3.73E+18	9.61E+18
Te-127m	2.03E+16	3.50E+16	3.50E+16	9.02E+16	La-140	2.51E+18	4.33E+18	4.33E+18	1.12E+19
Te-129	4.22E+17	7.28E+17	7.28E+17	1.88E+18	La-141	2.21E+18	3.81E+18	3.81E+18	9.84E+18
Te-129m	8.58E+16	1.48E+17	1.48E+17	3.82E+17	La-142	2.15E+18	3.71E+18	3.71E+18	9.57E+18
Te-131m	2.76E+17	4.77E+17	4.77E+17	1.23E+18	Nb-95	2.30E+18	3.96E+18	3.96E+18	1.02E+19
Te-132	1.95E+18	3.36E+18	3.36E+18	8.66E+18	Rh-105	1.43E+18	2.48E+18	2.48E+18	6.39E+18
Ba-139	2.42E+18	4.18E+18	4.18E+18	1.08E+19	Tc-99m	2.23E+18	3.85E+18	3.85E+18	9.93E+18

Table 6. Core inventories (Bq) for Units 1-3 and total.

These calculations have been indeed performed, and the results are reported in Table 9 for four different initial leak rates:

- 1%/h, used in the reference ENEA calculations;
- 0.5%/d, US-NRC reference value for design (i.e. intact plant) leakages;
- 0.25%/d;
- 0.0%/d i.e. no leakage.

As it can be seen, the lower the leak rate, the nearer to 1 the ratio (with very few exceptions). The no-leakage condition worsens the ratios to values lower than 1; this means that the calculations performed at IRSN have an equivalent leak rate higher than 0.0%/d, most probably between 0.0%/d and 0.25%/d.

Reference		ENEA	ENEA	ENEA	ENEA	[1]	[11]		[12]	[13]	[14]
T1/2 [d]	Nuclide	RASCAL 4.2 U1	RASCAL 4.2 U2	RASCAL 4.2 U3	RASCAL 4.2 total	RASCAL 4.2 total	IRSN d	ENEA / IRSN d Ratio	IRSN i	TEPCO MELCOR 2011	TEPCO i 2012
7.63E-02	Kr-83m	5.24E+14		3.04E+09	5.24E+14		1.0E+13	52.4			
3.93E+03	Kr-85	6.20E+15	9.76E+15	1.21E+16	2.81E+16		2.0E+16	1.4			
1.87E-01	Kr-85m	1.12E+16	6.75E+11	1.20E+14	1.13E+16		1.0E+14	113.0			
5.30E-02	Kr-87	6.01E+14		9.08E+06	6.01E+14		7.0E+11	858.4			
1.18E-01	Kr-88	9.93E+15	1.31E+09	5.84E+12	9.94E+15		5.0E+13	198.7			
1.19E+01	Xe-131m	1.25E+16	1.33E+16	1.86E+16	4.43E+16		2.0E+16	2.2			
5.25E+00	Xe-133	1.74E+18	1.52E+18	2.31E+18	5.56E+18		2.0E+18	2.8	1.21E+19	1.10E+19	
2.19E+00	Xe-133m	4.58E+16	2.40E+16	4.57E+16	1.16E+17		2.0E+16	5.8			
3.81E-01	Xe-135	3.83E+17	7.78E+15	8.07E+16	4.71E+17		2.0E+16	23.6			
1.06E-02	Xe-135m	2.26E+16	1.32E+13	8.06E+14	2.35E+16		6.0E+14	39.1			
9.78E-03	Xe-138	2.18E+09			2.18E+09		9.0E+01	2.4E+07			
1.74E-02	I-128						4.0E+04				
5.73E+09	I-129						2.0E+09	0.0			
5.15E-01	I-130						5.0E+13				
8.02E+00	I-131	4.89E+16	9.12E+16	5.84E+16	1.99E+17	2.00E+17	9.0E+16	2.2	1.03E+17	1.60E+17	4.70E+17
9.54E-02	I-132	5.28E+16	6.78E+16	5.56E+16	1.76E+17		7.0E+16	2.5	3.55E+16	4.70E+14	
5.79E-02	I-132m						2.0E+10				
8.67E-01	I-133	6.12E+16	1.60E+16	3.01E+16	1.07E+17		2.0E+16	5.4		6.80E+14	
3.65E-02	I-134	9.79E+13			9.79E+13		4.0E+11	244.6			
2.74E-01	I-135	2.21E+16	4.62E+13	1.20E+15	2.34E+16		2.0E+15	11.7		6.30E+14	
7.54E+02	Cs-134	6.00E+15	1.74E+16	7.59E+15	3.10E+16		1.0E+16	3.1		1.80E+16	1.20E+16
1.21E-01	Cs-134m						1.0E+12				
1.32E+01	Cs-136	2.98E+15	5.93E+15	2.94E+15	1.19E+16		6.0E+15	2.0	3.72E+15		
1.10E+04	Cs-137	4.15E+15	1.20E+16	5.25E+15	2.14E+16	2.17E+16	1.0E+16	2.1	1.55E+16	1.50E+16	9.00E+15
2.32E-02	Cs-138	2.58E+12			2.58E+12		3.0E+09	859.1			
1.00E-01	Br-83						2.0E+12				
2.21E-02	Br-84						7.0E+07				
1.86E+01	Rb-86	1.07E+14	2.22E+14	1.08E+14	4.37E+14						
1.23E-02	Rb-88	8.60E+15	1.22E+09	2.84E+12	8.60E+15		5.0E+13	172.1			
1.05E-02	Rb-89						3.0E+02				
1.01E+03	Sb-125						6.0E+14				
3.85E+00	Sb-127	2.71E+15	3.78E+15	1.03E+15	7.52E+15		4.0E+15	1.9		6.40E+15	
3.75E-01	Sb-128						1.0E+10				
6.97E-03	Sb-128m						1.0E+13				

1.83E-01	Sb-129	1.18E+15	5.61E+10	3.55E+12	1.18E+15		4.0E+13	29.5		1.60E+14	
2.74E-02	Sb-130						1.0E-15				
1.60E-02	Sb-131						8.0E+05				
3.99E-02	Te-125m						1.0E+14				
6.49E-03	Te-127	3.24E+15	5.32E+15	1.36E+15	9.93E+15		5.0E+15	2.0			
1.09E+02	Te-127m	5.13E+14	1.16E+15	2.43E+14	1.91E+15		1.0E+15	1.9		1.10E+15	
4.83E-02	Te-129	1.42E+15	3.00E+15	6.49E+14	5.07E+15						
3.36E+01	Te-129m	2.15E+15	4.62E+15	9.97E+14	7.76E+15		7.0E+15	1.1		3.30E+15	
1.74E-02	Te-131	1.06E+15	5.38E+14	2.50E+14	1.85E+15		5.0E+14	3.7			
1.25E+00	Te-131m	4.72E+15	2.39E+15	1.11E+15	8.22E+15		2.0E+15	4.1		9.70E+13	
3.20E+00	Te-132	4.22E+16	5.39E+16	1.53E+16	1.11E+17		6.0E+16	1.9		7.60E+14	
8.68E-03	Te-133						7.0E+09				
3.85E-02	Te-133m						4.0E+10				
2.90E-02	Te-134						6.0E+09				
5.77E-02	Ba-139	1.00E+14		2.45E+06	1.00E+14						
1.28E+01	Ba-140	2.36E+16	4.62E+16	1.05E+16	8.04E+16					3.20E+15	
3.25E+01	Ce-141	1.08E+15	2.31E+15	3.58E+14	3.75E+15					1.80E+13	
1.38E+00	Ce-143	6.98E+14	4.06E+14	1.23E+14	1.23E+15						
2.85E+02	Ce-144	8.82E+14	1.98E+15	2.99E+14	3.16E+15					1.10E+13	
1.63E+02	Cm-242	2.73E+13	6.07E+13	6.89E+12	9.48E+13					1.00E+11	
2.75E+00	Mo-99	6.83E+14	8.00E+14	6.50E+14	2.13E+15					8.80E+07	
1.10E+01	Nd-147	4.08E+14	7.75E+14	9.51E+13	1.28E+15					1.60E+12	
2.36E+00	Np-239	1.15E+16	1.17E+16	2.65E+15	2.59E+16					7.60E+13	
1.36E+01	Pr-143	9.60E+14	1.99E+15	2.36E+14	3.19E+15					4.10E+12	
3.20E+04	Pu-238	4.39E+08	4.62E+09	4.15E+08	5.48E+09					1.90E+10	
8.80E+06	Pu-239	7.28E+08	5.45E+09	5.85E+08	6.76E+09					3.20E+09	
2.39E+03	Pu-240									3.20E+09	
5.22E+03	Pu-241	6.52E+13	1.86E+14	2.67E+13	2.78E+14					1.20E+12	
3.93E+01	Ru-103	7.07E+14	1.56E+15	8.96E+14	3.17E+15					7.50E+09	
1.85E-01	Ru-105	6.23E+13	3.85E+09	6.26E+11	6.29E+13						
3.74E+02	Ru-106	1.57E+14	4.59E+14	2.43E+14	8.58E+14					2.10E+09	
5.06E+01	Sr-89	1.23E+16	2.69E+16	5.76E+15	4.50E+16					2.00E+15	
1.05E+04	Sr-90	7.59E+14	2.18E+15	4.34E+14	3.37E+15					1.40E+14	
4.01E-01	Sr-91	4.93E+15	9.21E+13	2.49E+14	5.27E+15						
1.11E-01	Sr-92	6.43E+14	4.03E+07	8.03E+10	6.43E+14						
2.67E+00	Y-90	1.27E+14	2.34E+14	3.29E+13	3.94E+14						
5.85E+01	Y-91	8.08E+14	1.68E+15	1.94E+14	2.69E+15					3.40E+12	
1.48E-01	Y-92	5.04E+14	1.03E+09	9.54E+10	5.04E+14						
4.24E-01	Y-93	2.05E+14	5.12E+12	5.49E+12	2.15E+14						
6.40E+01	Zr-95	1.07E+15	2.35E+15	2.69E+14	3.69E+15					1.70E+13	
6.98E-01	Zr-97	5.21E+14	8.11E+13	3.45E+13	6.37E+14						
1.68E+00	La-140	4.53E+15	6.29E+15	9.91E+14	1.18E+16						
1.63E-01	La-141	8.96E+13	1.46E+09	6.51E+10	8.96E+13						
6.33E-02	La-142	5.90E+12		3.06E+05	5.90E+12						
3.50E+01	Nb-95	1.10E+15	2.47E+15	2.79E+14	3.84E+15						
1.47E+00	Rh-105	3.79E+14	2.50E+14	2.82E+14	9.11E+14						
2.50E-01	Tc-99m	6.46E+14	7.71E+14	6.25E+14	2.04E+15						

Table 7. Cumulative Source Term (Bq) – Part 1 (see Table 8).

Reference	ENEA	[1]	[1]	[15]	[16]	[17]	[18]	[19]	[20]	[21]	
T1/2 [d]	Nuclide	RASCAL 4.2 total	NSC 22/08/2011	JNES	NISA 16/2/2012	Terada as per IRSN	Terada (manual sum)	Winiarek	Chino	NSC	Morino
8.02E+00	I-131	1.99E+17	1.30E+17	1.30E+17	1.50E+17	1.5E+17	1.24E+17	1.90E+17	1.50E+17	1.50E+17	1.42E+17
1.10E+04	Cs-137	2.14E+16	1.10E+16	6.10E+15	8.20E+15	1.3E+16	8.83E+15	1.20E+16	1.30E+16	1.20E+16	9.94E+15

Table 8. Cumulative Source Term (Bq) – Part 2 (see Table 7).

T1/2 [d]	Nuclide	1 %/h			0.5 %/d		0.25 %/d		0.0 %/d	
		IRSN d [Bq]	ENEA total [Bq]	Ratio [-]						
7.63E-02	Kr-83m	1.0E+13	5.2E+14	52.4	1.9E+13	1.9	1.3E+13	1.3	7.7E+12	0.8
3.93E+03	Kr-85	2.0E+16	2.8E+16	1.4	2.1E+16	1.4	2.7E+16	1.4	1.8E+16	0.9

1.87E-01	Kr-85m	1.0E+14	1.1E+16	113.0	4.2E+15	42.5	4.2E+15	41.7	4.1E+15	40.9
5.30E-02	Kr-87	7.0E+11	6.0E+14	858.4	1.3E+13	18.9	6.9E+12	9.8	5.2E+11	0.7
1.18E-01	Kr-88	5.0E+13	9.9E+15	198.7	1.4E+15	27.1	1.3E+15	25.3	1.2E+15	23.4
1.19E+01	Xe-131m	2.0E+16	4.4E+16	2.2	3.4E+16	2.1	4.2E+16	2.1	3.0E+16	1.5
5.25E+00	Xe-133	2.0E+18	5.6E+18	2.8	4.3E+18	2.6	5.3E+18	2.6	3.9E+18	1.9
2.19E+00	Xe-133m	2.0E+16	1.2E+17	5.8	9.0E+16	5.4	1.1E+17	5.4	8.6E+16	4.3
3.81E-01	Xe-135	2.0E+16	4.7E+17	23.6	3.5E+17	18.5	3.7E+17	18.5	3.6E+17	18.1
1.06E-02	Xe-135m	6.0E+14	2.3E+16	39.1	1.1E+16	19.0	1.1E+16	18.8	1.1E+16	18.6
9.78E-03	Xe-138	9.0E+01	2.2E+09	2.4E+07	4.6E+07	5.1E+05	2.3E+07	2.5E+05	0.0	0.0
1.74E-02	I-128	4.0E+04								
5.73E+09	I-129	2.0E+09								
5.15E-01	I-130	5.0E+13								
8.02E+00	I-131	9.0E+16	2.0E+17	2.2	1.2E+17	1.7	1.5E+17	1.7	7.9E+16	0.9
9.54E-02	I-132	7.0E+16	1.8E+17	2.5	1.1E+17	1.8	1.2E+17	1.8	7.4E+16	1.1
5.79E-02	I-132m	2.0E+10								
8.67E-01	I-133	2.0E+16	1.1E+17	5.4	5.4E+16	3.2	6.3E+16	3.1	5.0E+16	2.5
3.65E-02	I-134	4.0E+11	9.8E+13	244.6	2.1E+12	5.1	2.1E+12	5.1	1.8E+08	0.0
2.74E-01	I-135	2.0E+15	2.3E+16	11.7	4.6E+15	2.4	4.7E+15	2.3	4.4E+15	2.2
7.54E+02	Cs-134	1.0E+16	3.1E+16	3.1	2.0E+16	2.4	2.4E+16	2.4	1.0E+16	1.0
1.21E-01	Cs-134m	1.0E+12		0.0		0.0				0.0
1.32E+01	Cs-136	6.0E+15	1.2E+16	2.0	7.4E+15	1.5	9.0E+15	1.5	4.3E+15	0.7
1.10E+04	Cs-137	1.0E+16	2.1E+16	2.1	1.4E+16	1.7	1.7E+16	1.7	7.1E+15	0.7
2.32E-02	Cs-138	3.0E+09	2.6E+12	859.1	5.4E+10	18.0	2.7E+10	9.0		0.0
1.00E-01	Br-83	2.0E+12								
2.21E-02	Br-84	7.0E+07								
1.86E+01	Rb-86		4.4E+14		2.7E+14		3.3E+14		1.5E+14	
1.23E-02	Rb-88	5.0E+13	8.6E+15	172.1	1.3E+15	27.0	1.3E+15	25.4	1.2E+15	23.8
1.05E-02	Rb-89	3.0E+02								
1.01E+03	Sb-125	6.0E+14								
3.85E+00	Sb-127	4.0E+15	7.5E+15	1.9	4.4E+15	1.3	5.3E+15	1.3	2.2E+15	0.6
3.75E-01	Sb-128	1.0E+10								
6.97E-03	Sb-128m	1.0E+13								
1.83E-01	Sb-129	4.0E+13	1.2E+15	29.5	1.2E+14	3.1	1.1E+14	2.8	1.0E+14	2.5
2.74E-02	Sb-130	1.0E+15								
1.60E-02	Sb-131	8.0E+05								
3.99E-02	Te-125m	1.0E+14								
6.49E-03	Te-127	5.0E+15	9.9E+15	2.0	5.9E+15	1.4	7.2E+15	1.4	2.9E+15	0.6
1.09E+02	Te-127m	1.0E+15	1.9E+15	1.9	1.2E+15	1.4	1.4E+15	1.4	4.8E+14	0.5
4.83E-02	Te-129		5.1E+15		3.1E+15		3.7E+15		1.3E+15	
3.36E+01	Te-129m	7.0E+15	7.8E+15	1.1	4.7E+15	0.8	5.7E+15	0.8	2.0E+15	0.3
1.74E-02	Te-131	5.0E+14	1.9E+15	3.7	9.3E+14	2.2	1.1E+15	2.2	6.6E+14	1.3
1.25E+00	Te-131m	2.0E+15	8.2E+15	4.1	4.1E+15	2.5	4.9E+15	2.4	2.9E+15	1.5
3.20E+00	Te-132	6.0E+16	1.1E+17	1.9	6.4E+16	1.3	7.7E+16	1.3	3.4E+16	0.6
8.68E-03	Te-133	7.0E+09								
3.85E-02	Te-133m	4.0E+10								
2.90E-02	Te-134	6.0E+09								
5.77E-02	Ba-139		1.0E+14		2.2E+12		1.1E+12		4.7E+10	
1.28E+01	Ba-140		8.0E+16		4.8E+16		5.9E+16		2.1E+16	
3.25E+01	Ce-141		3.8E+15		2.3E+15		2.8E+15		8.7E+14	
1.38E+00	Ce-143		1.2E+15		6.2E+14		7.3E+14		4.0E+14	
2.85E+02	Ce-144		3.2E+15		1.9E+15		2.3E+15		7.1E+14	
1.63E+02	Cm-242		9.5E+13		5.7E+13		6.9E+13		2.0E+13	
2.75E+00	Mo-99		2.1E+15		1.3E+15		1.6E+15		9.3E+14	
1.10E+01	Nd-147		1.3E+15		7.6E+14		9.2E+14		2.8E+14	
2.36E+00	Np-239		2.6E+16		1.4E+16		1.7E+16		7.6E+15	
1.36E+01	Pr-143		3.2E+15		1.9E+15		2.3E+15		6.9E+14	
3.20E+04	Pu-238		5.5E+09		3.6E+09		4.5E+09		7.1E+08	
8.80E+06	Pu-239		6.8E+09		4.5E+09		5.5E+09		1.1E+09	
2.39E+03	Pu-240									
5.22E+03	Pu-241		2.8E+14		1.7E+14		2.1E+14		5.7E+13	
3.93E+01	Ru-103		3.2E+15		2.0E+15		2.4E+15		1.2E+15	
1.85E-01	Ru-105		6.3E+13		6.4E+12		5.9E+12		5.3E+12	
3.74E+02	Ru-106		8.6E+14		5.4E+14		6.7E+14		3.1E+14	
5.06E+01	Sr-89		4.5E+16		2.7E+16		3.3E+16		1.1E+16	
1.05E+04	Sr-90		3.4E+15		2.1E+15		2.6E+15		7.9E+14	

4.01E-01	Sr-91		5.3E+15		1.5E+15		1.5E+15		1.4E+15	
1.11E-01	Sr-92		6.4E+14		2.7E+13		2.1E+13		1.4E+13	
2.67E+00	Y-90		3.9E+14		2.6E+14		3.2E+14		1.2E+14	
5.85E+01	Y-91		2.7E+15		1.6E+15		2.0E+15		5.9E+14	
1.48E-01	Y-92		5.0E+14		9.5E+13		9.0E+13		8.6E+13	
4.24E-01	Y-93		2.2E+14		6.1E+13		6.3E+13		5.8E+13	
6.40E+01	Zr-95		3.7E+15		2.2E+15		2.7E+15		7.8E+14	
6.98E-01	Zr-97		6.4E+14		2.5E+14		2.8E+14		2.1E+14	
1.68E+00	La-140		1.2E+16		8.0E+15		9.9E+15		4.2E+15	
1.63E-01	La-141		9.0E+13		8.1E+12		7.3E+12		6.4E+12	
6.33E-02	La-142		5.9E+12		1.3E+11		7.0E+10		7.3E+09	
3.50E+01	Nb-95		3.8E+15		2.3E+15		2.8E+15		8.0E+14	
1.47E+00	Rh-105		9.1E+14		5.2E+14		6.2E+14		4.3E+14	
2.50E-01	Tc-99m		2.0E+15		1.2E+15		1.5E+15		8.9E+14	

Table 9. ENEA-IRSN comparison with different RASCAL 4.2 initial leak rates.

As far as release fractions to the atmosphere, it must be said that these can be easily calculated for each unit as ratios of ST (Table 7) to inventory (Table 6); they are reported for ease of reference in Table 10. However it must be kept in mind that these fractions are affected by the peculiar modeling adopted for the explosions, i.e. through controlled equivalent openings of the Mark-1 buildings. Also it must be remembered that at present RASCAL 4.2 calculates emissions only in the first 48 hours after the beginning of core uncovering and that the core inventories used by RASCAL 4.2 are based on a calculation scheme that is valid mainly for PWRs [1]. In the same Table 10 are also given in the last two columns the release fractions to the atmosphere for Unit 1 and Unit 3 in the case of a “total failure” of the Mark-1 containments following the secondary building explosions; it can be seen that in this case the release fractions for long-lived noble gases approach unity. The equivalent average ¹³³Xe release fraction for all three units has been estimated in [25] as about 60%, with an initial total inventory of 1.27E19 Bq, which is identical to that evaluated with RASCAL 4.2 (see Table 6; it should be remembered that the ¹³³Xe activity essentially depends on specific power and not on burnup). This fraction is somehow different, but not exceedingly much, from those reported in Table 10 and which range between 32% and 63%.

T1/2	Nuclide	Fractions U1	Fractions U2	Fractions U3	Fractions U1 Total Mark-1 Failure	Fractions U3 Total Mark-1 Failure
[d]	[-]	[-]	[-]	[-]	[-]	[-]
7.63E-02	Kr-83m	3.4E-04	0.0E+00	1.1E-09	3.4E-04	1.1E-09
3.93E+03	Kr-85	7.1E-01	5.1E-01	6.4E-01	1.0E+00	9.5E-01
1.87E-01	Kr-85m	3.5E-02	1.2E-06	2.2E-04	4.1E-02	2.3E-04
5.30E-02	Kr-87	9.6E-04		8.4E-12	9.6E-04	8.4E-12
1.18E-01	Kr-88	1.1E-02	8.8E-10	3.9E-06	1.2E-02	3.9E-06
1.19E+01	Xe-131m	6.7E-01	4.1E-01	5.8E-01	9.5E-01	8.6E-01
5.25E+00	Xe-133	6.3E-01	3.2E-01	4.8E-01	8.8E-01	7.1E-01
2.19E+00	Xe-133m	5.2E-01	1.6E-01	3.0E-01	7.3E-01	4.4E-01
3.81E-01	Xe-135	5.3E-01	6.2E-03	6.4E-02	7.8E-01	7.7E-02
1.06E-02	Xe-135m	3.9E-02	1.3E-05	8.0E-04	1.7E-01	8.0E-04
9.78E-03	Xe-138	9.4E-10			9.4E-10	0.0E+00
8.02E+00	I-131	3.6E-02	3.9E-02	2.5E-02	4.3E-02	2.6E-02
9.54E-02	I-132	2.7E-02	2.0E-02	1.6E-02	3.1E-02	1.7E-02
8.67E-01	I-133	2.2E-02	3.3E-03	6.3E-03	2.5E-02	0.0E+00
3.65E-02	I-134	3.2E-05			3.2E-05	5.8E-03
2.74E-01	I-135	8.4E-03	1.0E-05	2.6E-04	8.9E-03	2.6E-04
7.54E+02	Cs-134	4.1E-02	5.4E-02	2.4E-02	4.9E-02	2.5E-02
1.32E+01	Cs-136	3.9E-02	4.5E-02	2.2E-02	4.7E-02	2.3E-02
1.10E+04	Cs-137	4.1E-02	5.4E-02	2.4E-02	4.9E-02	2.5E-02
2.32E-02	Cs-138					
1.86E+01	Rb-86	4.0E-02	4.8E-02	2.3E-02	4.8E-02	2.4E-02
1.23E-02	Rb-88					
3.85E+00	Sb-127	2.2E-02	1.8E-02	4.9E-03	2.7E-02	5.1E-03

1.83E-01	Sb-129	2.7E-03	7.3E-08	4.6E-06	2.7E-03	4.7E-06
6.49E-03	Te-127	2.7E-02	2.6E-02	6.6E-03	3.3E-02	6.9E-03
1.09E+02	Te-127m	2.5E-02	3.3E-02	7.0E-03	3.1E-02	7.4E-03
4.83E-02	Te-129	3.4E-03	4.1E-03	8.9E-04	4.1E-03	9.5E-04
3.36E+01	Te-129m	2.5E-02	3.1E-02	6.7E-03	3.0E-02	7.2E-03
1.74E-02	Te-131					
1.25E+00	Te-131m	1.7E-02	5.0E-03	2.3E-03	2.0E-02	2.4E-03
3.20E+00	Te-132	2.2E-02	1.6E-02	4.6E-03	2.6E-02	4.8E-03
5.77E-02	Ba-139	4.1E-05		5.9E-13	4.1E-05	5.9E-13
1.28E+01	Ba-140	9.7E-03	1.1E-02	2.5E-03	1.2E-02	2.7E-03
3.25E+01	Ce-141	4.8E-04	6.0E-04	9.3E-05	5.9E-04	1.0E-04
1.38E+00	Ce-143	3.4E-04	1.2E-04	3.5E-05	4.0E-04	3.7E-05
2.85E+02	Ce-144	4.9E-04	6.3E-04	9.6E-05	5.9E-04	1.0E-04
1.63E+02	Cm-242	4.8E-04	6.2E-04	7.0E-05	5.8E-04	7.7E-05
2.75E+00	Mo-99	2.5E-04	1.7E-04	1.4E-04	3.0E-04	1.4E-04
1.10E+01	Nd-147	4.6E-04	5.0E-04	6.2E-05	5.5E-04	6.7E-05
2.36E+00	Np-239	4.0E-04	2.3E-04	5.3E-05	4.7E-04	5.6E-05
1.36E+01	Pr-143	4.7E-04	5.7E-04	6.8E-05	5.8E-04	7.4E-05
3.20E+04	Pu-238					
8.80E+06	Pu-239					
5.22E+03	Pu-241	3.0E-04	5.0E-04	7.1E-05	3.7E-04	7.7E-05
3.93E+01	Ru-103	3.2E-04	4.1E-04	2.3E-04	3.8E-04	2.4E-04
1.85E-01	Ru-105	4.0E-05	1.4E-09	2.3E-07	4.1E-05	2.3E-07
3.74E+02	Ru-106	3.2E-04	4.3E-04	2.3E-04	3.9E-04	2.4E-04
5.06E+01	Sr-89	1.0E-02	1.3E-02	2.7E-03	1.2E-02	2.9E-03
1.05E+04	Sr-90	1.0E-02	1.3E-02	2.6E-03	1.2E-02	2.8E-03
4.01E-01	Sr-91	3.2E-03	3.5E-05	9.4E-05	3.5E-03	9.5E-05
1.11E-01	Sr-92	3.9E-04	1.4E-11	2.8E-08	3.9E-04	2.8E-08
2.67E+00	Y-90	1.0E-03	1.1E-03	1.5E-04	1.3E-03	1.9E-04
5.85E+01	Y-91	5.0E-04	6.0E-04	6.9E-05	6.1E-04	7.6E-05
1.48E-01	Y-92	3.0E-04	3.6E-10	3.3E-08	3.2E-04	3.3E-08
4.24E-01	Y-93	1.6E-04	2.3E-06	2.5E-06	1.8E-04	2.5E-06
6.40E+01	Zr-95	4.7E-04	6.0E-04	6.9E-05	5.8E-04	7.5E-05
6.98E-01	Zr-97	2.4E-04	2.2E-05	9.3E-06	2.8E-04	9.7E-06
1.68E+00	La-140	1.8E-03	1.5E-03	2.3E-04	2.4E-03	2.9E-04
1.63E-01	La-141	4.1E-05	3.8E-10	1.7E-08	4.2E-05	1.7E-08
6.33E-02	La-142	2.7E-06		8.3E-14	2.7E-06	8.3E-14
3.50E+01	Nb-95	4.8E-04	6.2E-04	7.0E-05	5.8E-04	7.7E-05
1.47E+00	Rh-105	2.6E-04	1.0E-04	1.1E-04	3.1E-04	1.2E-04
2.50E-01	Tc-99m	2.9E-04	2.0E-04	1.6E-04	3.4E-04	1.7E-04

Table 10. Release fractions for Units 1 to 3.

To overcome the problem of different initial core inventories, one can resort to ratios of ratios, the most widely used being the following Fr factor [25]:

$$Fr = \frac{\frac{Isotope(t)}{137Cs(t)}}{\frac{Isotope(0)}{137Cs(0)}}$$

where Isotope(t) is the measured activity of a generic isotope at time t while 137Cs(t) is the measured activity of 137Cs at the same time; the values at time t=0 being the inventory ones.

In Table 11 are given, as an example, the ENEA FR factors for some FPs. They are compared with the FR factors evaluated by CEA in [25] using measured activity concentrations at the JPP38 CTBTO station on the 22nd of March 2011. The last column of the table gives the relative difference between the two factors. The agreement is very good for 86Rb, 134Cs and 136Cs. The agreement should be good also for other long-lived (long in comparison to the accident progression time-scale and to the time separation between accident and measurement) and volatile isotopes.

Radionuclide	T 1/2	Type	Fr ENEA @ 48 hours since SCRAM	Fr CEA [25]	Var
[-]	[d]	[-]	[-]	[-]	[%]
86Rb	1.86E+01	Volatile	9.31E-01	9.15E-01	1.7
134Cs	7.54E+02	Volatile	1.00E+00	1.03E+00	-3.0
136Cs	1.32E+01	Volatile	8.97E-01	8.03E-01	11.8
127mTe	1.09E+02	Volatile	5.41E-01	1.57E+00	-65.5
132Te	3.20E+00	Volatile	3.28E-01	2.44E+00	-86.6
99Mo	2.75E+00	Semi-volatile	4.54E-03	9.06E-02	-95.0
131I	8.02E+00	Volatile	8.38E-01	1.73E+01	-95.2
129mTe	3.36E+01	Volatile	3.62E-02	2.24E+00	-98.4
95Nb	3.50E+01	Low volatile	9.62E-03	5.48E-04	1655.0
103Ru	3.93E+01	Low volatile	8.22E-03	1.31E-04	6158.9

Table 11. Examples of some FR factors for a few isotopes.

If one considers the volatile, long-lived isotopes reported in Table 11, a noteworthy disagreement is represented by the Te isotopes. The discrepancies between the ENEA and the CEA evaluations for these isotopes might be explained if one considers that NUREG 1465, upon which RASCAL 4.2 is based, underestimates Tellurium release fractions, as pointed out in [26,27], even by a factor 2. This is one of the reasons that prompted the establishing of the (French) VERCORS and of the (international) PHEBUS large experimental programs devoted to a better understanding of severe accident STs. It must also be said that 132Te has the additional problem of possessing a relatively short half-life.

Other data from CTBTO measurements may be used; for example the 136Cs/137Cs activity ratio measured at Sand Point and Vancouver [25], once corrected for time decay, is found to be on average 0.26, while the same ratio using the ENEA ST is about 0.55; the agreement is not perfect, but still the evaluation may be considered acceptable.

The 131I/137Cs ratio around the Fukushima site, once corrected for decay, was found to be fluctuating between 10 and 20 from 18 March 2011 and 27 May 2011 [25]; the ENEA ST gives a ratio of about 9.3, which is not much different from the measured one.

Finally, as regards the time performances of the RASCAL 4.2 code, it can be reported that the computation time required for each unit is about 30 seconds on a standard desktop.

The time required to the user to setup each calculation, once the sequence has been established, is less than 2.5 minutes.

8. Conclusions

The results shown in this report clearly indicate that RASCAL 4.2 can lead to good estimates of the ST in the case of severe accidents like the CSBOs at the Fukushima site. Some hypotheses need nonetheless to be made, and in the case of a real-time emergency the experience of the user is of fundamental importance. In particular a sound (i.e. realistic) time sequence of release events is absolutely necessary.

Moreover the RASCAL analysis may also be seen as an *a posteriori* validation of the hypothesis that the material ejected during the ventings and/or explosions was of D/W nature or origin, and not of W/W (i.e. after pool scrubbing) origin (see Appendix).

As a follow-up to this study, a second analysis is foreseen for the next future, using the recently released version 4.3 of RASCAL which incorporates many new and advanced improvements derived from the accumulated Fukushima experience.

Appendix 1 – Pool Scrubbing

In Fig. 8 are shown the details of a Mark-1 suppression chamber; several weak points, where damage, whatever its origin, may lead to failure of the scrubbing mechanism, can clearly be imagined [22].

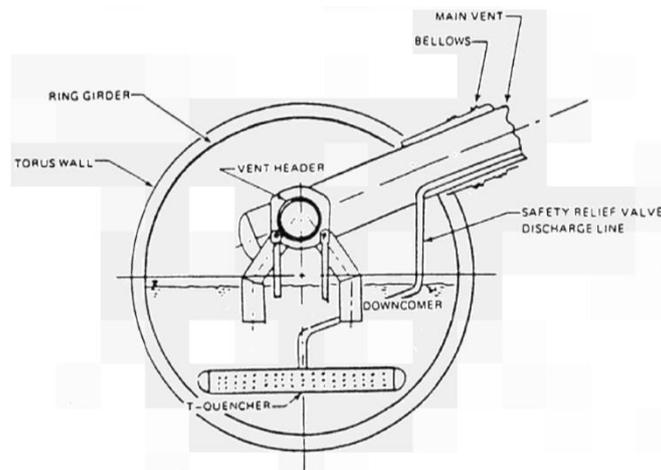


Fig. 8. Scheme of a Mark-1 S/C [22].

The US-NRC stated in 1988 [24] that *“The decontamination factor... associated with suppression pool scrubbing can range anywhere from 1 (no scrubbing) to well over 1000 (99.9% effective). This wide band is a function of the accident scenario and composition of the fission products, the pathway to the suppression pool (through spargers, downcomers, etc.), and the conditions in the suppression pool itself. Conservative decontamination factor values of five [80% removal] for scrubbing in Mark-1 suppression pools, and 10 [90% removal] for Mark-2... suppression pools, have recently been proposed for licensing review purposes.”*

This statement is in further support of the fact that the real accidental conditions of the S/C should be considered when evaluating the “effective” release path of a venting. In view of this, several new strategies are presently being considered worldwide for older NPPs to lower the amount of activity released during a severe accident; see for instance [23].

Appendix 2 – Uncertainty Analysis

The numerical results of the RASCAL ST evaluation are inevitably affected by uncertainties related to the values of some parameters which must be specified in the accident sequences. These parameters are those that appear in the Source Term and Release Path RASCAL sub-modules for which, because it's impossible provide an exact value, it is necessary to propose a realistic estimate based on the type of accident pathway. It is therefore very useful and recommended to perform a sensitivity analysis to evaluate how the uncertainties of the most relevant model parameters contribute to the global uncertainty of the ST. Given the very complex and demanding nature of such a task, it is decided to restrict it to only the total amounts of Cs-137 and I-131 released by the three Units together. This represents an effective way, albeit simple, to assess the global robustness of the method employed.

The first step of this analysis consists in the identification of the parameters to be perturbed. Those, which on one side affect the final outcome of the global source term and on the other are affected by the largest uncertainties, are the following:

- P_c , accident containment leak rate [%/h];
- P_v , the venting leak rate [%/h];
- P_e , the explosion leak rate [%/h];
- T_l , duration of releases (both ventings and explosions) [h];
- T_u , time to core dewatering [h];
- Venting events (number and time of venting releases) [-].

After careful analysis of the TEPCO official information and of the available literature, the venting parameter can be considered unaffected by uncertainties; it's in fact recognized that the sequences of Unit 1 and 2 presented in this Report are basically correct; the second venting of the Unit 3 sequence, which has been neglected in the reference (i.e. unperturbed) calculation, is ignored on the basis of the available data concerning the extremely low probability of opening of the air-operated valve.

To assess the propagation of the model parameters uncertainties to the global results, it is then necessary to define plausible ranges of variations for these very same parameters. The choice of the range of variability of the parameters is based on physical considerations on the plant type and on the comparison between different literature values. The ranges chosen are reported, for each Unit, in Tab. 12.

Unit	Parameters variabilities				
	ΔP_c	ΔP_v	ΔP_e	ΔT_l	ΔT_u
[-]	[%/h]	[%/h]	[%/h]	[min]	[h]
1	±0.5	±5	±10	±15	±2.0
2	±0.5	±5	-	±15	±6.5
3	±0.5	±5	±10	±15	±5.0

Table 12. Parameters ranges for this uncertainty study.

The analysis is carried out in two further steps.

In the first step a series of calculations of the source term modifying one parameter at a time is performed using the two extreme values of the ranges reported in the previous table; this resulted in 28 new RASCAL calculations (Unit 2 didn't explode [$\Delta P_e=0$]) with which a first-order partial derivatives expansion of the ST can be derived around the unperturbed state (hypothesis of small variations). The single variations are then evaluated by simple multiplication of the derivatives times the corresponding variations.

The results of the total source term variation (ΔST) for each parameters is shown in Tab. 13.

The second step was to combine all the previous parameter-dependent source term variations to quantify the impact of the total source term variation to the global ST. The mathematical relationship to combine the source term variations (Tab. 13) into a single number is a function of the nature of the uncertainties (deterministic or stochastic); in particular the mathematical expressions for the two cases are:

$$\Delta S_{tot} = \sum_i \Delta S_i \quad (\text{all deterministic uncertainties})$$

$$\Delta S_{tot} = \sqrt{\sum_i \Delta S_i^2} \quad (\text{all stochastic uncertainties})$$

Parameters	All Units	
	Δ ST I-131	Δ ST Cs-137
[-]	[Bq]	[Bq]
P_c	3.67E+16	3.85E+16
T_l	1.13E+16	1.30E+15
T_u	9.63E+15	8.85E+14
P_v	8.84E+15	9.91E+14
P_e	1.70E+15	1.55E+14

Table 13. Source Term variation for each parameter.

It is also assumed that all parameter values of the purely stochastic case are equally likely. It is self-evident that purely deterministic parameter uncertainties lead to higher result uncertainties. The result obtained by this analysis in terms of percentage change of the global ST is shown in Tab.14.

	Nuclide	Unperturbed ST	Uncertainty
Nature of uncertainties	[-]	[Bq]	[% of unperturbed value]
Deterministic	Cs-137	2.14E+16	± 34
	I-131	1.98E+17	± 34
Stochastic	Cs-137	2.14E+16	± 21
	I-131	1.98E+17	± 20

Table 14. Global uncertainty values.

The parameters values adopted for the unperturbed (or reference) ENEA sequences for the three units are based on assessments made both on the available data and on the information given in the reference literature. Both these two information sources provide parameters values of inherent statistical nature; nevertheless the uncertainty resulting assuming a purely deterministic nature for the parameters uncertainties is considered numerically more appropriate because of the need of a conservative approach for the problems of ST estimation. Moreover the simplified analysis carried on in this Appendix provides evaluations of the result uncertainties which are somehow lower than those which could be obtained through a more comprehensive and detailed approach, f.i. varying all the possible model parameters of the code over wider ranges. Therefore the uncertainties deriving from the assumption of the purely deterministic case are considered better representative of the real uncertainties.

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List of Acronyms

AO	Air Operated
BDB	Beyond Design Basis
CSBO	Complete Station Black-Out
d	day
D/W	Drywell
h	hour
IA	Instrument Air
MO	Motor Operated
S/C	Suppression Chamber
ST	Source Term
SBGTS	Stand-by Gas Treatment System
TAF	Top of Active Fuel
W/W	Wetwell