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Nuclear criticality safety — Estimation of the number of fissions of a postulated criticality accident

Sécurité de criticité nucléaire — Évaluation du nombre de fissions en cas d'un hypothétique accident de criticité

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Foreword

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The committee responsible for this document is ISO/TC 85, *Nuclear energy, nuclear technologies, and radiological protection*, Subcommittee SC 5, *Nuclear fuel cycle*.

Introduction

In activities involving fissile materials, the potential for a criticality accident occurrence cannot be totally excluded. Therefore, in order to prepare emergency responses in case of such an occurrence, ISO 27467 specifies areas to be studied (Annex A) to perform the analysis of potential consequences whenever a credible criticality accident may occur. This International Standard deals with one of these areas and is devoted to the estimate of number of fissions (also commonly named “fission yield”) for a postulated criticality accident. This topic is essential because most of the other issues of the criticality accident analysis depend on a suitable estimate of this number of fissions.

Nuclear criticality safety — Estimation of the number of fissions of a postulated criticality accident

1 Scope

This International Standard provides a methodology to estimate a reasonably maximal value of the number of fissions of a postulated criticality accident.

The fission number estimate, associated with its postulated criticality accident, impacts the accident emergency planning and response because it is used for the estimation of radiation doses and of radioactive materials release.

This International Standard does not provide a methodology and guidance to determine bounding accident scenarios.

This International Standard does not cover criticality accident detection which is dealt with by ISO 7753.

This International Standard does apply to nuclear facilities, plants, laboratories, storage, and transportation of fissile material (but not to nuclear power reactor cores) where a credible criticality accident may occur.

2 Terms and definitions

For the purposes of this document, the following terms and definitions apply.

2.1
postulated criticality accident
postulated association of one accident scenario and one accident evolution

Note 1 to entry: One postulated criticality accident is associated with one estimated number of fissions.

2.2
accident scenario
set of credible, postulated conditions under which a fissile material-containing facility/process develops one or more fault conditions such that it is likely to exceed the critical state and thus to result in a criticality accident

Note 1 to entry: This definition is drawn from ISO 27467.

2.3
accident evolution
progress of the criticality accident (after the critical state is exceeded), taking into account physical phenomena (for example, temperature and void effects) and possible human interventions to stop it

2.4
area of applicability
set of parameters (for example, environment, geometrical characteristics, fissile material, accident duration) within which a tool/model is intended to be used

Note 1 to entry: In Annex D, the last columns of the tables summarize the area of applicability of some simplified formulae.

3 General principles

PREREQUISITES Once the objectives of the criticality accident analysis (analysis based, for example, on ISO 27467) are defined, one or several criticality accident(s) may be postulated. The assumptions of the postulated criticality accident, and therefore the potential consequences, are to be related with the objectives of the criticality accident analysis (for example, design of evacuation routes, dose mapping, assembly station(s) choice).

EXAMPLE 1 Because bounding assumptions may be different for radiation dose estimates and for radioactive materials release estimates, it is possible to choose a set of assumptions adapted for each estimate.

EXAMPLE 2 The design of evacuation routes may be performed with an arbitrary number of fissions; the goal is to optimize the operators' evacuation routes, whatever the value of the dose is. In this case, the location of the postulated criticality accident is the most important parameter.

3.1 For the estimation of the number of fissions, the following assumptions, as well as their variations, should be considered:

— description of the equipment (geometric configuration, reflector, etc.);

— degree of confinement and environment (vessel open or closed, pressure, cooling, etc.);

— fissile material (quantity, enrichment, media, physical shape, chemical form, etc.);

— total reactivity addition;

— rate of reactivity addition;

— time delay before the first persistent chain reaction (function of the initial neutron source, i.e. spontaneous fissions, (alpha, n)-reactions, etc.);

— duration of the criticality accident (calculated/estimated with and without intervention, where applicable).

3.1.1 The determination of these assumptions should be drawn from the accident scenario and the accident evolution of the postulated criticality accident.

3.1.2 The chosen assumptions shall be within the domain physically possible according to the characteristics of the considered activity (characteristics of the facility, of the transportation, etc.).

WARNING — The estimation of the number of fissions is only the first part of the determination of the consequences of the postulated criticality accident (see, for example, the flow diagram from ISO 27467 in Annex A). The overall estimation of the consequences shall take into account all the aspects of the criticality accident and iterations between estimation of the number of fissions and subsequent actions (for example, doses estimation) should be performed. For example, in case of different possible locations for a criticality accident, the postulated criticality accident leading to the highest number of fissions may not necessarily lead to the maximum doses for workers and the public because of its location. Other assumptions affecting the consequences of the postulated criticality accident should then be considered, such as:

— location of the equipment, place of the criticality accident;

— building description;

— location of people;

— criticality accident alarm system presence/absence.

3.2 Each fissions number estimate shall be associated with an approximated duration. Account should be made of any anticipated human interventions in the accident evolution.

3.3 Number of fissions shall be determined by using simplified models (4.3 and notably 4.3.2) or calculation tools (4.4) or both.

4 Fissions number estimate

4.1 General

4.1.1 For the estimate of the number of fissions, the use of the simplified models route (4.3) should be firstly considered.

4.1.2 The use of the calculation tools route (4.4) may then be considered, according to the objectives of the criticality accident analysis (for example, design of evacuation routes, dose mapping, and assembly station(s) choice). This route requires:

— the availability of a calculation tool able to simulate the criticality accident, and

— the determination of all input data needed for the calculation tool.

4.1.3 In the case where the two routes of estimate are used, the origin of a different order of magnitude between the two results should be understood and documented.

4.2 Input data

4.2.1 The input data needed for the simplified models or the calculation tools (geometry, external environment, media characteristics, etc.) shall be taken from assumptions considered for the accident scenario and the accident evolution. When the accident scenario and the accident evolution do not set necessary input data, these should be measured or calculated or estimated from the international literature.

NOTE Depending on the way estimates are made, the type and the number of input data needed may vary.

4.2.2 The selected input data sensitivities (linked to uncertainties and possible variations pointed out in 3.1) should be studied for the chosen route(s) of estimate (4.3 and/or 4.4). This study will provide a better understanding of the uncertainties associated with the estimated number of fissions. This study may be one possible basis for the nuclear criticality safety specialist to appropriately select a maximal estimate. Otherwise, further justifications should be made as to the applicability of the result.

4.2.3 Account shall be made for parameters that could vary significantly for the criticality accident duration.

4.3 Use of simplified models

4.3.1 The estimate of number of fissions should be based on simplified options providing "order-of-magnitude" values.

4.3.2 This estimate should rely on the collective experiences from past criticality accidents (Annex B) and criticality experiment results (Annex C) and the possible use of simplified formulae (Annex D).

4.3.3 When a simplified model is used, the consistency of its area of applicability with the chosen assumptions of the postulated criticality accident shall be justified and documented.

NOTE The duration of the criticality accident has a significant impact on the evaluation. Actually, simplified models are mainly based on criticality experiments and past criticality accidents stopped after human intervention.

4.3.4 To estimate the number of fissions, the simplified models results should be associated with the sensitivity study performed (4.2.2).

4.4 Use of calculation tools

WARNING — Care should be taken when using the criticality accident calculation tool results for the estimation of the number of fissions. In particular, a complete validation of a criticality accident calculation tool is presently difficult, mainly due to the complexity of models and paucity of criticality experiment and precise information from past criticality accidents.

4.4.1 The calculation tool used shall be documented, including the verification of the adequate implementation of the different models (for example, neutron physics, thermal transfer, bubbles behaviour).

4.4.2 When it is possible, comparison between the calculation tool results and experiments/accidents close to the chosen assumptions of the postulated criticality accident should be documented.

4.4.3 When a calculation tool is used, the consistency of its area of applicability with the chosen assumptions of the postulated criticality accident shall be justified and documented. In case of inconsistency, the calculation tools may still be used; however, justification for its use shall be documented.

4.4.4 Free evolution of the system during the accident duration shall be accounted for. Resulting assumptions used in the calculation should lead to a maximal evaluation of the number of fissions.

4.4.5 To estimate the number of fissions, the calculation tool results should be associated with the sensitivity study performed (4.2.2) and with other available elements (for example, results obtained from comparison with experiments, complexity of models, possible penalizing hypothesis in the models).

Annex A
(informative)

Flow diagram of a criticality accident analysis(from ISO 27467:2009)

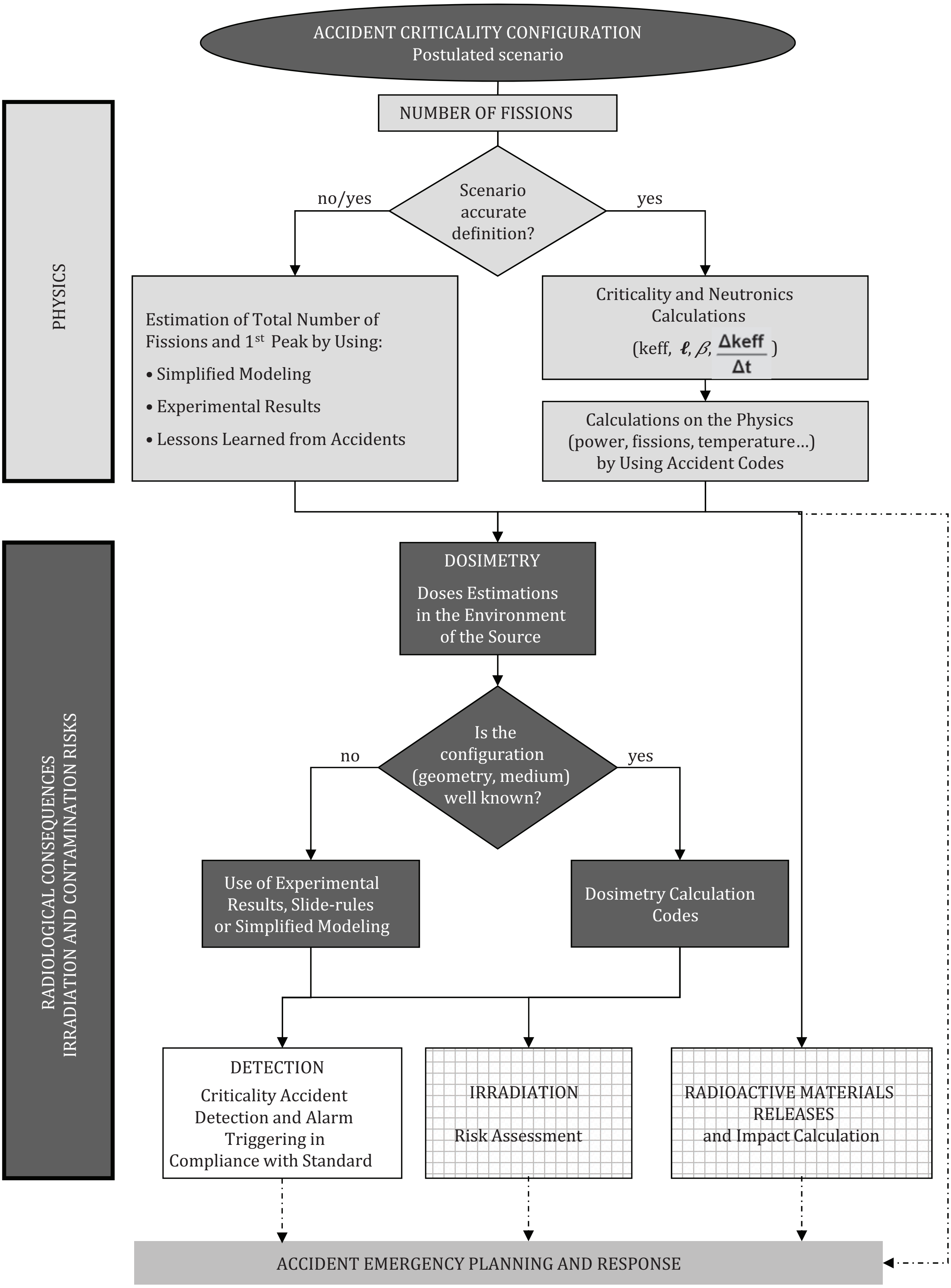

Figure A.1 — Flow diagram of a criticality accident analysis

Annex B
(informative)

Characteristics of criticality accidents that occurred during process operation

The following information is mainly taken from[4]: 22 known criticality accidents occurred during process operations, 21 occurred with fissile material in solutions or slurries, 1 occurred with metal ingots.

So far, no process accident occurred in dry powder, with rods in water, or with fissile materials in storage or being transported.

The characteristics of the fissile media involved in past criticality accidents are various by element (U, Pu) and enrichment. For solution, the fuel volume goes from 19 l to 800 l.

For these criticality accidents, the number of fissions of the first power spike, when known, was less than $2,0 \times 10^{17}$ fissions. The total number of fissions for criticality accidents goes from approximately 10^{15} fissions to $4,0 \times 10^{19}$ fissions. These estimations of number of fissions are nevertheless very rough for some of them and should be taken with care.

Accident experience shows that these events might be only a single brief pulse, multiple pulses, or they can be a quasi-steady-state excursion that continues for a very long time. The accident duration ranges from a few seconds to about 40 h. Without intervention, some of the process accidents might have continued for much longer.

The analysis of past criticality accidents shows that scenarios leading to the criticality excursion were due to several failures (more than two). Many of them occurred during non-routine operations.

A summary of criticality accidents in nuclear fuel processing plants is presented in Table B.1. To increase the amount of data concerning the estimation of the number of fissions based on the past criticality accidents, a summary of criticality accidents in reactor and critical experiments is also presented in Table B.2. Even if these accidents occurred with configurations different from process operations (for example, important reactivity insertion due to configuration with critical or slightly subcritical experiments, detection system, safeguard system, etc.), they could give some information about fissile media that have had no process criticality accidents (for example, metal, rods in water), as well as providing more information about fissile media that were involved in process criticality accidents.

Table B.1 — Summary of criticality accidents in nuclear fuel processing plants

No.[1]	Site[2]	Date	Fuel type	Fissile media[3]	Geometry	Fuel volume (l)	Vessel volume (l)	Fissions in initial burst (fiss)	Total fissions (fiss) [4]	Duration [4]
1	**Mayak**	1953/03/15	Solution	Pu	Vertical cylinder	31	40	unknown	$2,0 \times 10^{17}$	< 1 min[5]
2	**Mayak**	1957/04/21	Slurry	U(90)	Horizontal cylinder	30	100	unknown	$1,0 \times 10^{17}$	10 min
3	**Mayak**	1958/01/02	Solution	U(90)	Vertical cylinder	58,4	442	$2,0 \times 10^{17}$	$2,0 \times 10^{17}$	< 1 min[5]
4	**Y-12**	1958/06/16	Solution	U(93)	Vertical cylinder	56	208	$1,0 \times 10^{16}$	$1,3 \times 10^{18}$	20 min
5	**LASL**	1958/12/30	Solution (Org.)	Pu	Vertical cylinder	160	982	$1,5 \times 10^{17}$	$1,5 \times 10^{17}$	< 1 min[5]
6	**ICPP**	1959/10/16	Solution	U(91)	Horizontal cylinder	800	18900	$1,0 \times 10^{17}$	$4,0 \times 10^{19}$	20 min
7	**Mayak**	1960/12/05	Solution	Pu	Vertical cylinder	19	40	unknown	$2,5 \times 10^{17}$	1 h 50 min
8	**ICPP**	1961/01/25	Solution	U(90)	Vertical cylinder	40	461	$6,0 \times 10^{16}$	$6,0 \times 10^{17}$	< 3 min[6]
9	**Tomsk**	1961/07/14	Solution (Org.)	U(22,6)	Vertical annular cylinder	42,9	65	none	$1,2 \times 10^{15}$	< 1 min[5]
10	**Hanford**	1962/04/07	Solution	Pu	Vertical cylinder	45	69	$1,0 \times 10^{16}$	$8,0 \times 10^{17}$	37 h 30 min
11	**Mayak**	1962/09/07	Solution	Pu	Vertical cylinder	80	100	none	$2,0 \times 10^{17}$	1 h 40 min
12	**Tomsk**	1963/01/30	Solution	U(90)	Vertical cylinder	35,5	49,9	unknown	$7,9 \times 10^{17}$	10 h 20 min
13	**Tomsk**	1963/12/02	Solution (Org.)	U(90)	Vertical cylinder	64,8	100	none	$1,6 \times 10^{16}$	16 h
14	**Wood River**	1964/07/24	Solution	U(93)	Vertical cylinder	51	103,7	$1,0 \times 10^{17}$	$1,3 \times 10^{17}$	1 h 30 min
15	**Electrostal**	1965/11/03	Slurry	U(6,5)	Vertical cylinder	100	300	none	$1,0 \times 10^{16}$	< 1 min[5]
16	**Mayak**	1965/12/16	Solution	U(90)	Vertical cylinder	28,6	100	none	$5,5 \times 10^{17}$	7 h
17	**Mayak**	1968/12/10	Solution (Org.)	Pu	Vertical cylinder	28,8	62,1	$3,0 \times 10^{16}$	$1,3 \times 10^{17}$	> 15 min
18	**Windscale**	1970/08/24	Solution (Org.)	Pu	Vertical cylinder	40	156	none	$1,0\times10^{15}$	10 s
19	**ICPP**	1978/10/17	Solution	U(82)	Vertical cylinder	315,5	315,5	unknown	$2,7 \times 10^{18}$	~2 h
20	**Tomsk**	1978/12/13	Metal	Pu	Vertical cylinder	0,54	3,2	$3,0 \times 10^{15}$	$3,0 \times 10^{15}$	< 1 min[5]
21	**Novosibirsk**	1997/05/15	Slurry	U(70)	Two vertical parallel vessels	unknown	2 x 700	none	$5,5 \times 10^{15}$	27 h 5 min
22	**Tokai-mura**	1999/09/30	Solution	U(18,8)	Vertical cylinder	45	100	$5,0 \times 10^{16}$	$2,5 \times 10^{18}$	19 h 40 min

Table B.2 — Summary of selected criticality accidents in reactor and critical experiments

No.[1]	Site[2]	Date	Fuel type	Fissile media[3]	Geometry	Fuel mass (kg) or volume (l)	Total fissions (fiss) [4]	Duration [4]
A-1	**LASL**	1949/12	Solution	U(14)	Sphere, graphite reflected	13,6 l	~3×10^{16}	< 1 min[5]
A-2	**Hanford**	1951/11/16	Solution	Pu	Bare sphere	63,8 l	8×10^{16}	< 1 min[5]
A-3	**ORNL**	1954/05/26	Solution	U(93)	Cylindrical annulus, bare	55,4 l	1×10^{17}	< 1 min[5]
A-4	**ORNL**	1956/02/01	Solution	U(93,2)	Cylindrical bare	58,9 l	$1{,}6 \times 10^{17}$	< 1 min[5]
A-5	**ORNL**	1968/01/30	Solution	^{233}U(98)	Sphere, water reflected	5,84 l	$1{,}1 \times 10^{16}$	< 1 min[5]
B-1	**LASL**	1945/08/21	Metal	Pu	Sphere with WC reflector	6,2 kg	~1×10^{16}	< 1 min[5]
B-2	**LASL**	1946/05/21	Metal	Pu	Sphere with Be reflector	6,2 kg	~3×10^{15}	< 1 min[5]
B-3	**LASL**	1951/02/01	Metal	U(93,5)	Cylinder and annulus in water	62,9 kg	~1×10^{17}	< 1 min[5]
B-4	**LASL**	1952/04/18	Metal	U(93)	Cylinder, unreflected	92,4 kg	$1{,}5 \times 10^{16}$	< 1 min[5]
B-5	**Sarov**	1953/04/09	Metal	Pu	Sphere with natural U reflector	~8 kg	~1×10^{16}	< 1 min[5]
B-6	**LASL**	1954/02/03	Metal	U(93)	Sphere, unreflected	53 kg	$5{,}6 \times 10^{16}$	< 1 min[5]
B-7	**LASL**	1957/02/12	Metal	U(93,7)	Sphere, unreflected	54 kg	$1{,}2 \times 10^{17}$	< 1 min[5]
B-8	**LASL**	1960/06/17	Metal	U(93)	Cylinder with C reflector	48 kg	6×10^{16}	< 1 min[5]
B-9	**ORNL**	1961/11/10	Metal	U(93)	Paraffin reflected	75 kg	~1×10^{16}	< 1 min[5]
B-10	**Sarov**	1963/03/11	Metal	Pu	Sphere with LiD reflector	~17,35 kg	~5×10^{15}	< 1 min[5]
B-11	**Livermore**	1963/03/26	Metal	U(93)	Cylinder with Be reflector	47 kg	$3{,}7 \times 10^{17}$	< 1 min[5]
B-12	**WSMR**	1965/05/28	Metal	U(93) + Mo	Cylinder, unreflected	96 kg	$1{,}5 \times 10^{17}$	< 1 min[5]
B-13	**Chelyabinsk-70**	1968/04/05	Metal	U(90)	Sphere with natural U reflector	47,7 kg	6×10^{16}	< 1 min[5]
B-14	**Aberdeen**	1968/09/06	Metal	U(93) + Mo	Cylinder, unreflected	123 kg	$6{,}09 \times 10^{17}$	< 1 min[5]
B-15	**Sarov**	1997/06/17	Metal	U(90)	Sphere with Cu reflector	~44 kg	~1×10^{19}	6 d 13 h 55 min

Table B.2 — *(continued)*

No.[1]	Site[2]	Date	Fuel type	Fissile media [3]	Geometry	Fuel mass (kg) or number of rods	Total fissions (fiss)[4]	Duration[4]
C-1	**LASL**	1945/06/06	Metal	U(79,2)	Water reflected pseudosphere	35,4 kg	$\sim 4 \times 10^{16}$	< 1 min[5]
C-3	**ANL**	1952/06/02	UO_2 particles in plastic	U(93)	Fuel elements in water	324 fuel elements	$1,22 \times 10^{17}$	< 1 min[5]
C-4	**Chalk River**	1952/12/12	Natural uranium rods	U(0,71)	Heavy water moderated reactor	~190 fuel rods	$1,2 \times 10^{20}$	< 1 min[5]
C-6	**Vinca**	1958/10/15	Natural uranium rods	U(0,71)	Fuel rods in heavy water	3995 kg (U mass)	$\sim 2,6 \times 10^{18}$	~10 min
C-7	**Saclay**	1960/03/15	Oxide	U(1,5)	Fuel rods in water	2200 kg	3×10^{18}	< 3 min[6]
C-8	**IRTA**	1961/01/03	U fuel	U(93)	Fuel rods in water	40 elements	$4,4 \times 10^{18}$	< 1 min[5]
C-10	**Mol**	1965/12/30	Oxide	U(7)	Rods in water/ heavy water	1200 kg	$\sim 4 \times 10^{17}$	< 1 min[5]
C-11	**Kurchatov**	1971/02/15	UO_2 fuel rods	U(20)	Fuel rods Be reflected	349 rods	2×10^{19}	15 min
C-12	**Kurchatov**	1971/05/26	UO_2 fuel rods	U(90)	Fuel rods, water reflected	1790 rods	5×10^{18}	< 1 min[5]
C-13	**Buenos Aires**	1983/09/23	MTR type fuel elements	U(90)	Pool-type reactor	20 fuel elements	$\sim 4 \times 10^{17}$	< 1 min[5]
D-1	**LASL**	1945/02/11	UH_3 pressed in styrex	unknown	Dragon assembly	5,4 kg	$\sim 6 \times 10^{15}$	< 1 min[5]
D-2	**NRTS**	1955/11/29	Uranium in NaK	U(93,2)	EBR-1	52 kg (U mass)	$\sim 4 \times 10^{17}$	< 1 min[5]
D-3	**LASL**	1956/07/03	U metal foils moderated with graphite	U(93)	Honeycomb	58 kg (U mass)	$3,2 \times 10^{16}$	< 1 min[5]
D-4	**NRTS**	1958/11/18	Uranium oxide in nickel-chromium	U(93,2)	HTRE	~220 kg (UO_2 mass)	$2,5 \times 10^{19}$	< 1 min[5]
D-5	**LASL**	1962/12/11	U foils moderated with graphite	U(93)	Cylinder plus annular reflector	unknown	$\sim 3 \times 10^{16}$	< 1 min[5]

1) The numbering in the tables in Annex B corresponds to the numbering used in.[4]

2) **Mayak**: Mayak Production Association (Russia), **Y-12**: Oak Ridge Y-12 Plant (USA), **LASL**: Los Alamos Science Laboratory (USA), **ICPP**: Idaho Chemical Processing Plant (USA), **Tomsk**: Siberian Chemical Combine (Russia), **Hanford**: Hanford Works (USA), **Wood River**: United Nuclear Fuel Recovery Plant (USA), **Electrostal**: Electrostal Machine Building Plant (Russia), **Windscale**: Windscale Works (UK), **Novosibirsk**: Novosibirsk Chemical Concentration Plant (Russia), **Tokai-mura**: JCO Fuel Fabrication Plant (Japan), **ORNL**: Oak Ridge National Laboratory (USA), **Sarov**: Sarov (Arzamas-16) (Russia), **Livermore**: Lawrence Livermore Laboratory (USA), **WSMR**: White Sands Missile Range (USA), **Chelyabinsk-70**: Chelyabinsk-70 (Russia), **Aberdeen**: Aberdeen Proving Ground (USA), **ANL**: Argonne National Laboratory (USA), **Chalk River**: Chalk River Laboratory (Canada), **Vinca**: Vinca (ex-Yugoslavia), **Saclay**: Centre d'Etudes Nucléaires de Saclay (France), **IRTA**: Idaho Reactor Testing Area (USA), **Mol**: Mol (Belgium), **Kurchatov**: Kurchatov Institute (Russia), **Buenos Aires**: Buenos Aires (Argentina), **NRTS**: National Reactor Testing Station (USA).

3) Number in parentheses shows enrichment of ^{235}U (except for accident A-5, Table B.2 where it is enrichment of ^{233}U).

4) For some accidents, there is a significant uncertainty about the number of fissions estimate and the duration.

5) Set < 1 min for "a short time" or "single excursion".

6) Set < 3 min for "a few minutes".

NOTE 1 There is no estimation of the total number of fissions for accident C-2.

NOTE 2 Accidents C-5 and C-9 were experiments performed in BORAX and SPERT reactors and are presented in Annex C.

Annex C
(informative)

Experimental results

Many experiments have been carried out in the world to study criticality accident features, particularly the number of fissions. The main medium studied is solution. The data obtained for it cover a wide range of key parameters such as the volume of solution, the reactivity insertion rate, and the solution concentration. Some of these experimental programs allowed the elaboration of simplified formulae (Annex D). Experiments with metal and heterogeneous (assemblies in water) media are also presented even if the main purpose of these experiments was not the study of criticality accidents.

Experimental results could be used for the determination of the number of fissions in the initial spike and the total number of fissions. Tables below present the main features of the experimental facilities for solution (Table C.1), metal (Table C.2), and heterogeneous (Table C.3) media. Detailed features of these facilities are provided in the cited references. To use experimental facilities results, justification of the representativeness of the experiment features in comparison with the accident scenario must be provided.

It must be taken into account that, for experiments, transient duration can be controlled by the experimenter, using control systems, which is not the case for real criticality accident.

NOTE 1 The uncertainties on the value of the number of fissions are not always given and it should be kept in mind that the experimental evaluation depends on the calibration of the experimental facility.

NOTE 2 General information and literature search about experimental facilities are listed in[27] and[31] to.[39] These references might also be useful to get additional information about some experimental facilities.

NOTE 3 In the following tables and for each experimental facility, maximal (or minimal) parameters do not necessarily come from a single experiment.

Table C.1 — Main features for experimental solution facilities

Facility	CRAC	SILENE	TRACY	SHEBA II
Country	France	France	Japan	USA
Working date	1968-1972	1974-2010	1996-2011	1993-2004
References	[5] to[9]	[6] to[11]	[6],[11] to[17],[130]	[22] to[26],[50]
Geometry	Bare cylinder	Bare annular cylinder	Bare annular cylinder	Bare annular cylinder
External diameter of the vessel (cm)	30 or 80	36,8	52	56,9 or 50,8
Fissile media	Uranyl nitrate	Uranyl nitrate	Uranyl nitrate	Uranyl fluoride
^{235}U enrichment	93 %	93 %	10 %	5 %
Fuel volume (l)	20 to 259	23 to 54	94 to 125	77 to ~90
C(U) (g/l)	21 to 383	49 to 221	375 to 426	~1000
[H$^+$]	0,91 to 2,87	1,87 to 2,84	0,56 to 0,85	< 0,5
Potential reactivity insertion	< 27 $	< 7 $	< 3 $	< 0,79 $
Rate of reactivity insertion	Solution feed < 31 l/min	Step < 3,28 $ Ramp < 2 $/s	Step < 3 $ Ramp < 0,8 $/s Solution feed < 60 l/min	Step < 0,79 $
Reactor period	> 1 ms	> 2,1 ms	> 3 ms	> 1 s
Number of fissions in the initial spike	< 8 × 10^{17} fiss	< 1,9 × 10^{17} fiss	< 3 × 10^{17} fiss	-
	< 3,9 × 10^{15} fiss/l	< 4,9 × 10^{15} fiss/l	< 2,5 × 10^{15} fiss/l	-
Total number of fissions	< 5 × 10^{18} fiss	< 8,7 × 10^{17} fiss	< 8 × 10^{17} fiss	< 4,3 × 10^{17} fiss
	< 1,4 × 10^{16} fiss/l	< 1,8 × 10^{16} fiss/l	< 7 × 10^{15} fiss/l	< 5,1 × 10^{15} fiss/l
Commentary	Two-phase flow experiments were also performed	Experiments with reflectors were also performed	Experiments with reflectors were also performed	-

Table C.1 — *(continued)*

Facility	KEWB		IGRIK	YaGUAR	VIR family
Country	USA		Russia	Russia	Russia
Working date	1956-1959	1960-1966	1976-present	1990-present	1964-present
References	[18] to[21]	[19]	[27],[28],[29]	[27],[30]	[27],[101],[102]
Geometry	Reflected sphere	Cylinder (bare or reflected)	Annular cylinder	Annular cylinder	“Cylindrical form”
External diameter of the vessel (cm)	~32,4	~30,5	61	55	~40 or ~55
Fissile media	Uranyl sulfate		Uranyl sulfate	Uranyl sulfate + cadmium sulfate (5 g/l)	Uranyl sulfate
^{235}U enrichment	93,2 %		90 %	90 %	90 %
Fuel volume (l)	11,45 or 13,65	18, 24 or 26	~60	~40	31,6 to 147
C(U) (g/l)	178 or 114	61, 100 or 217	116	170 or 465	54 to 158
Potential reactivity insertion	< 6,25$	< 5,8$	6$	4,7$	<9,3$
Rate of reactivity insertion	Step < 3,75$ Ramp < 0,16$/s	Step < 5,8$	Unknown	Unknown	Unknown
Reactor period	> 2 ms	> 0,56 ms	Unknown	Unknown	Unknown
Total number of fissions	< 1,9 × 10^{17} fiss	< 3,3 × 10^{17} fiss	< 2 × 10^{18} fiss	< 1,1 × 10^{18} fiss	< 2,7 × 10^{18} fiss
	< 1,4 × 10^{16} fiss/l	< 1,8 × 10^{16} fiss/l	< 3,3 × 10^{16} fiss/l	< 2,75 × 10^{16} fiss/l	< 2,4 × 10^{16} fiss/l

Table C.2 — Main features for experimental metal facilities

Equipment	Country	Working date	Fissile media	Enrichment	Fuel mass (kg)	Geometry	Reactivity insertion ($)	Reactor period	Total number of fissions (fission)	Reference
CALIBAN	France	1971-present	U + 10 % Mo	93,5 %	113	Vertical cylinder	< 1,1	> 18 µs	< 6 × 10^{16}	[45],[77] to[81]
GODIVA I	USA	1951-1957	U	93,7 %	53	Bare sphere	< 1,1	> 11,6 µs	< 2 × 10^{16}	[40] to[43]
GODIVA II	USA	1957-1960	U	93,2 %	57,7	Cylinder with a spherically shaped top	~1,05	> 11,6 µs	< 2,7 × 10^{16}	[49]
GODIVA IV	USA	Early 1960s-2004	U + 1,5 % Mo	93 %	66	Cylinder	< 1,15	> 8,4 µs	< 10^{17}	[44],[45],[50]
KUKLA	USA	1961-1964	U	93,2 %	60,13	Bare spherical assembly	< 1,08	> 11,1 µs	< 2 × 10^{16}	[84],[85]
JEZEBEL	USA	1954-1977	Pu (delta-phase)	4,5 % ^{240}Pu	16,745	Bare sphere	< 0,6	> 4 s	-	[89]
FRAN	USA	1962-1965	U	93,5 %	63,2	Unreflected, unmoderated, cylinder	< 1,2	11 µs	< 5,6 × 10^{16}	[51],[52],[53]
HPRR	USA	1962-1987	U + 10 % Mo	93,2 %	116	Unreflected, unmoderated, cylinder	< 1,1	11,6 µs	< 1,8 × 10^{17}	[59] to[67],[105]
Molly-G /WSMR (FBR)	USA	Late 1950s - present	U + 10 % Mo	93,2 %	97	Unreflected, unmoderated, cylinder	< 1,11	> 11,1 µs	< 1,2 × 10^{17}	[86],[87],[88]
Super Kukla	USA	1963-1974	U + 10 % Mo	20 %	~5000	Annular cylinder	< 1,3	> 202 µs	< 4 × 10^{18}	[82],[83]
SPR	USA	1961-1967	U	93,2 %	57,2	Right circular cylinder with a domed cap	< 1,07	-	< 2 × 10^{16}	[90],[91],[92]

Table C.2 — *(continued)*

Equipment	Country	Working date	Fissile media	Enrichment	Fuel mass (kg)	Geometry	Reactivity insertion ($)	Reactor period	Total number of fissions (fission)	Reference
SPR-II	USA	1967-2006	U + 10 % Mo	93 %	105	Cylinder	< 1,12	> 11,8 µs	< 1,6 × 10^{17}	[46],[90],[93] to[98]
SPR-III	USA	1975-2006	U + 10 % Mo	93,2 %	252	Reflected annular cylinder	< 1,12	23 µs	< 4,3 × 10^{17}	[90],[99],[100]
APRFR	USA	1966-2003	U+ 10 % Mo	93,2 %	< 125	Unreflected or reflected unmoderated cylinder	< 1,14	> 17,9 µs	< 3,7 × 10^{17}	[48],[64],[68] to[74]
VIPER	England	1967-2008	U + 1,25 % Mo	37,5 %	< 312 of U	Core made of cylindrical rods (max 744)	< 1,22	> 100 µs	< 3,63 × 10^{17}	[47],[54] to[58]
SKUA	USA	1978-1996	U + 1,5 % Mo	93 %	175	Reflected annular cylinder	< 0,95	-	< 2 × 10^{17}	[75],[76],[104]
BIR-2M	Russia	1965-unknown	U + 6 % Mo	~85 %	121	Cylinder	< 1,08	-	< 9,9 × 10^{16}	[27],[101],[102]
TIBR	Russia	1970-unknown	U + 9 % Mo, $ZrH_{1,9}$ combined	~90 %	124	Sphere	< 1,42	-	< 2,3 × 10^{17}	[27],[101],[102]
BIGR	Russia	1977-unknown	UO_2+C ceramics	~90 %	833	Cylinder	< 1,10	-	< 1 × 10^{19}	[27],[101],[102],[103]
BR-1	Russia	1978-unknown	U + 9 % Mo	~90 %	176	Cylinder	-	-	< 3,6 × 10^{17}	[27],[101],[102]
RIR	Russia	1981-unknown	U	~90 %	~25	Sphere	-	-	< 1,5 × 10^{19}	
GIR2	Russia	1993-unknown	U + 9 % Mo	~90 %	178	Sphere	< 1,26	-	< 2,3 × 10^{17}	
BR-K1	Russia	1995-unknown	U + 9 % Mo	36 %	1511	Cylinder	-	-	< 9,9 × 10^{17}	

Table C.3 — Main features for experimental heterogeneous facilities

Equipment	SPERT I		BORAX I	TRIGA[a]	PULSTAR
Country	USA		USA	-	USA
Working date	1954-1962		1953-1954	1958-present	1961-
References	[111] to[117],[125]	[118],[123],[124]	[119],[120]	[106] to[109]	[110],[121],[122]
Core	4 ft diameter 10 ft high		4 ft diameter 10 ft high	28 inch diameter 22 inch high	-
Core	Non-pressurized, light water-moderated, and reflected reactors			Light water, 1-ft-thick graphite reflector reactor	Non-pressurized, light water-moderated, and reflected reactors
Geometry	Plate	Rod	Plate	Rod	Rod
Fissile media	U-Al fuel	UO_2 fuel	U-Al fuel	U-ZrH_x	UO_2 fuel
Clad	Al or stainless steel	Stainless steel	Al	Stainless steel	Zr
^{235}U enrichment	93,5 %	4 %	Highly enriched uranium	20 %	6 %
Potential reactivity insertion	< 3,6$	< 2,7$	< 3,1$	< 4,6$	< 2,5$
Ramp	< 1$/s	-	-	-	-
Number of assemblies	19-64	592 rods	26-30	87-123	20
Number of plate per assemblies	12-24	x	10-18	x	x
Water/Fuel ratio	0,88-3,3	1,57	0,4-0,6	0,5	unknown
Total number of fissions (fission)	$< 1,4 \times 10^{18}$	$< 5 \times 10^{18}$	$< 4,5 \times 10^{18}$	$< 1,5 \times 10^{18}$	$< 1,6 \times 10^{18}$
Reactor period	> 3,2 msec	> 2,2 msec	> 2,6 msec	> 1,68 msec	2,8 msec

[a] Many (~70) TRIGA reactors exist in the world. Some features may vary.

Annex D
(informative)

Simplified formulae

Simplified formulae were created to give a bounding number of fissions without precise knowledge of the events that led to supercriticality. Some of these formulae were derived from experiments and others were based on theoretical considerations from the one-point reactor kinetic equation and/or the thermal theory.

Some simplified formulae are presented below. Note that other formulae could be found and be established to better take into account specific models and configurations (powder, metal, rods). Some of these formulae are listed in,[129],[144] and.[145]

Before using a simplified formula, one should well understand the basis and applicable conditions of this formula. A review of the original publication may help in understanding them.

NOTE 1 The bibliography does not include references suggesting recommended fixed numbers of fissions.

The following symbols and abbreviated terms are necessary for the understanding of the tables of Annex D:

A is the numerical value of the solution feed rate, expressed in litre per second;

a is the numerical value of the ramp rate of reactivity insertion, expressed in dollar ($);

α is the numerical value of the temperature feedback, expressed in per cent mille per degree (pcm/°C);

b is the numerical value of the quenching constant, expressed in per cent mille per fission (pcm/fission);

β is the numerical value of the delayed neutron fraction, expressed in per cent mille (10^{-5}) (pcm);

D is the numerical value of the tank diameter, expressed in centimetre (cm);

d_{sol} is the numerical value of the total solution density, expressed in kilogram per litre (kg/l);

d_{H_2O} is the numerical value of the water density, expressed in kilogram per litre (kg/l);

$E_2(x)$ is the error function of the value x;

φ is the numerical value of the void volume feedback, expressed in per cent mille per litre (pcm/litre);

Γ_2 is the numerical value of the Diven's parameter;

NOTE 2 Γ_2 is generally equal to 0,8 (see[142]).

H is the numerical value of the solution height, expressed in centimetre (cm);

h is the numerical value of the average convection heat transfer coefficient, expressed in watt per square meter per degree [W/m^2/°C];

K is the numerical value of the reciprocal heat capacity, expressed in degree per fission (°C/fission);

κ is the numerical value of the constant depending on the geometry;

NOTE 3 κ is equal to $(4.\pi)^{1/3}.3^{2/3} \approx 4{,}836$ for a sphere and is equal to 6 for a cube (see[133]).

m_{sol} is the numerical value of the total solution mass, expressed in kilogram (kg);

$\check{m}_c(\Phi)$ is the numerical value of the minimum critical mass of solution for the considered geometry, expressed in kilogram (kg);

N_{B} is the numerical value of the number of fissions in the first power spike;

N_{P} is the numerical value of the number of fissions of the plateau;

N_{f} is the numerical value of the total number of fissions;

v is the numerical value of a coefficient, expressed in litre per square fission [litre/(fission)^2];

$\bar{v}$ is the numerical value of the average value of neutrons emitted per fission;

ρ_0 is the numerical value of the step reactivity input, expressed in per cent mille (10^{-5}) (pcm);

t is the numerical value of duration of the criticality accident, expressed in second (s);

t_{p} is the numerical value of the first peak time, expressed in second (s);

t_1 is the numerical value of the delay after reaching a critical mass before the first persistent chain reaction occurs, expressed in second (s);

τ is the numerical value of the neutron lifetime, expressed in second (s);

V is the numerical value of the solution volume, expressed in litre (l);

V_{B} is the numerical value of the solution volume at the time of the burst, expressed in litre (l);

$\check{V}_c(\Phi)$ is the numerical value of the minimum critical volume of solution for the considered geometry, expressed in litre (l);

V_{T} is the numerical value of the total solution volume, expressed in litre (l).

Table D.1 — Simplified formulae for the determination of total number of fissions for solutions

Formulae		Area of applicability
Barbry[a] (1982) [128],[131],[139]	$N_f = \frac{t}{3{,}55\times10^{-15} + 6{,}38\times10^{-17}.t}.V$	— uranyl nitrate solutions in a homogeneous medium with highly enriched uranium $^{235}U/U$ = 93 % — fuel concentration: 20~360 gU/l — tank: cylindrical shape with diameters of 30,80 cm (CRAC) and 36 cm (SILENE) — volume between 20 l and 260 l — the model could be applicable to Pu solutions — no boiling of the solution — t < 600 s (without delayed triggering of the reaction, i.e. with an appreciable neutron source) — not a “step” criticality accident
Olsen (1974) [127],[131]	$N_B = 2{,}95\times10^{15}.V_B^{0{,}82}$ $N_P = 3{,}2\times10^{18}.(1-t^{-0{,}15})$ $N_f = N_B + N_P$	— solution with highly enriched uranium, slightly enriched uranium, or plutonium system — tank diameter between 30 cm and 80 cm — solution feed rate between 97 l/h and 1872 l/h
Nomura (1995) [129],[131]	$N_f = 2{,}6\times10^{16}.V_T$ (without boiling) $N_f = 6\times10^{16}.V_T$ (with boiling)	— density of solution lower than 1,85 — evaporation of less than 25 % of the solution during boiling — no forced cooling — no condensation of the solution during boiling

a The Barbry formula is also devoted to the estimate of the number of fissions in the first power spike for solutions.

Table D.1 *(continued)*

Formulae		Area of applicability
Tuck (1974) [126],[131]	$N_f = 10^{17}.V_T$	— density of solution lower than 1,2 — no forced cooling — no condensation of the solution during boiling
Knemp - Duluc (2008) [132],[133]	without boiling: $N_f = 1{,}3\times10^{16}.V_T.d_{sol}.\left(1+\frac{\kappa.h.t}{4{,}184\times10^5.d_{sol}.(V_T)^{1/3}}\right)$ or $N_f = 1{,}3\times10^{16}.m_{sol}.\left(1+\frac{\kappa.h.t}{4{,}184\times10^5.(d_{sol})^{2/3}.(m_{sol})^{1/3}}\right)$ with boiling: $N_f = 1{,}3\times10^{16}.V_T.d_{sol}+8\times10^{16}.\left[V_T-\overset{\vee}{V}_c(\Phi)\right].d_{H2O}$ or $N_f = 1{,}3\times10^{16}.m_{sol}+8\times10^{16}.\left[m_{sol}-\overset{\vee}{m}_c(\Phi)\right]$	— homogeneous medium — constant pressure — homogeneous medium — constant pressure — no forced cooling — no condensation of the solution during boiling
a The Barbry formula is also devoted to the estimate of the number of fissions in the first power spike for solutions.		

Table D.2 — Simplified formulae for number of fissions in the first power spike for solutions

Formulae		Area of applicability
Tuck (1974) [126],[131]	For Uranium system (accuracy +70 %, −90 %) $N_B = 2{,}4 \times 10^{15}.V_T$ For Plutonium system (accuracy +100 %, −70 %) $N_B = 4{,}6 \times 10^{16}.A^{1/4} \exp\left(0{,}0177.D - \frac{(150 - H).0{,}8.A}{D}\right)$ if H>150, then use 150.	— maximum fissions in the initial burst (maximum fissions during a five-second interval) — solution feed rate is the maximum design capacity of the system (between 0,47 l/s and 0,006 l/s) — tank volume refers to the largest tank in the system. The diameter between 28 cm and 152 cm — fissile solution volume is the maximum one for normal operations plus that which could credibly be inadvertently added — the tank bottom is 30 cm or more above any reflecting material such as concrete — the fuel concentration is that which will produce the worst accident for the conditions involved — after a criticality alarm, any operating equipment, such as pumps or valves, is not shut off by the operator — the heat transfer from the tank to the environment must be not greater than a bare tank cooled by room air. Little venting steam will condense in the vent lines and drain back into the tank
Olsen (1974) [127],[131]	$N_B = 2{,}95 \times 10^{15}.V_B^{0{,}82}$	— solution with highly enriched uranium, slightly enriched uranium, or plutonium system — tank diameter must be between 30 cm and 80 cm — solution feed rate must be between 97 l/h and 1872 l/h

Table D.3 — Simplified formulae based on point reactor model (for number of fissions in the first power spike)

Formulae		Area of applicability
Nordheim — Fuchs [140] to[143]	$N_B = \frac{2.\left(\rho_0 - \beta\right)}{\alpha.K}$ for $\rho_0 > \beta$ [a] $N_B = \frac{2.\rho_0}{\alpha.K}$ for $0 < \rho_0 < \beta$ [b]	— extraneous source neutron may be neglected entirely — step reactivity — adiabatic model — reactivity feedback only due to temperature
Hetrick – Gamble [c] [140]	$N\left(t_p\right) = \frac{\alpha.K}{\varphi.\nu}.\left[-1 + \sqrt{1 + \frac{2.\varphi.\nu.\left(\rho_0 - \beta\right)}{\left(\alpha.K\right)^2}}\right]$	— delayed neutrons and extraneous source neutron may be neglected entirely — step reactivity $\rho_0 > \beta$ — adiabatic model — reactivity feedback due to temperature and radio-lytic gas
Hansen – Hankins [134] to[138],[143]	$b.N\left(t_1\right) \approx \sqrt{a^2.t_1{}^2 + 2.a.\tau.\ln\left[2.a.\tau.W\left(t_1\right) / b\right]}$ with $W\left(t\right) = \frac{\sqrt{\left(8.a.\tau\right) / \pi.\bar{\nu}^2.\Gamma_2^2}}{1 - E_2\left(\sqrt{a / 2.\tau}.t\right)}.\exp\left(-a.t^2 / 2.\tau\right)$	— reactivity feedback proportional to the energy release — ramp insertion — adiabatic model — time delay of first per-sistent chain taken into account

a Delayed neutrons neglected.

b One group delayed neutron considered.

c This formula estimates the number of fissions until the first peak time and not in the first power spike.

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