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Derivation of the Source Term and Analysis of the Radiological Consequences of Research Reactor Accidents

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DERIVATION OF THE SOURCE TERM AND ANALYSIS OF THE RADIOLOGICAL CONSEQUENCES OF RESEARCH REACTOR ACCIDENTS

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FOREWORD

This report was developed within the framework of the IAEA's activities in the area of research reactor safety. The objective is to provide a set of suggested methods and practices, based on current good practices around the world, for deriving the source term and analysing the radiological consequences of research reactor accidents.

The report is intended to provide safety analysts, reactor facility management and operations staff, and regulators with the necessary calculation methods and techniques. The examples given are intended to meet the requirements and recommendations for performing safety analyses of research reactor facilities, particularly for preparing the safety analysis report. The information in this Safety Report provides guidance and specific examples for all steps of the calculations required for deriving the source term, including all factors relevant to the formation of radioactive releases, and for estimating on-site and off-site radiological consequences.

Specialists from 13 Member States contributed to the present report. The IAEA officer responsible for this publication was C. Ciuculescu of the Division of Nuclear Installation Safety.

EDITORIAL NOTE

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CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

Research reactors are used primarily for research, training, radioisotope production, neutron radiography and materials testing, and thus they have unique design features and operational regimes that differ from those of power reactors. Being smaller in size and generally generating much less nuclear energy than power reactors, research reactors demonstrate a broader range of designs, nuclear fuel compositions, modes of operation and safety arrangements. The need for greater flexibility in their use requires a comprehensive approach to safety analysis.

Safety analysis is mainly used to enable the operator to understand the basis for safe operation of the reactor and to demonstrate to the regulatory body how the design of the facility and the related operational procedures contribute to the prevention and mitigation of accidents. Safety analysis must also consider experimental devices and programmes with respect to both their safety and their effect on reactor safety.

In meeting the requirements for research reactor safety, one of the initial steps is to determine the postulated initiating events (PIEs). The PIEs define the scope of the accidents to be used in the safety analysis. They establish the scenarios to be analysed in order to predict the consequences of accidents. The analytical and computational resources needed for the safety analysis of a reactor follow from an assessment of the PIEs. For each PIE, qualitative and quantitative information needs to be given on the source term derivation and the analysis of radiological consequences.

The source term is defined as the magnitude, composition, form (physical and chemical) and mode of release (puff, intermittent or continuous) of radioactive elements (fission and/or activation products) released during a reactor accident. The mechanism, time and location of the release must also be identified. To facilitate their assessment, the radiological consequences can be grouped into the following categories:

(a) Consequences inside the reactor building with doses to operating staff or personnel within the building;
(b) On-site consequences (outside the reactor building) from:
    (i) Direct radiation doses from the containment;
    (ii) Inhalation and ingestion doses from active material released from the containment;

(c) Off-site consequences (to members of the public) from:
   (i) Direct radiation doses from the containment;
   (ii) Inhalation and ingestion doses from active gaseous or liquid releases from the containment to the environment.

This report is part of the set of publications developed within the framework of the IAEA's activities in the area of research reactor safety. Reference [1] provides safety requirements for the design and operation of research reactors. It covers the large variety of designs, the wide range of power levels, the different modes of operation and purposes of utilization, the particularities of siting and the differences among organizations operating research reactors, in particular concerning their resources. The wide variety of these characteristics demands flexibility in the setting and fulfilment of basic requirements when dealing with certain specific topics. IAEA Safety Guides [2–4] provide guidance on fulfilling these requirements.

This report offers guidance on and examples of methods of analysis that may be useful for meeting the general requirements related to the derivation of the source term and the evaluation of the radiological consequences of research reactor accidents. It should be used in conjunction with the IAEA Safety Requirements and Safety Guides for research reactors [1–4].

1.2. OBJECTIVE

The objective of this report is to provide a set of suggested methods and practices, based on current good practices around the world, for deriving the source term and analysing the radiological consequences of research reactor accidents. The report covers all steps involved in performing analyses, that is, the selection of initiating events and the analysis of the core damage mechanisms and progression, radionuclide inventory releases, and radiological consequences inside the reactor building, outside the reactor building and off the site. It also presents practical examples applied to existing research reactors.

The aim is to provide safety analysts and reviewers with methods and techniques for, and practical examples related to, the derivation of the source term and the evaluation of the radiological consequences of postulated research reactor accidents having the potential to lead to radioactive releases. These practical examples will also assist in meeting the requirements and recommendations related to carrying out safety analyses and preparing the safety analysis report (SAR). In particular, the present publication provides guidance on the requirements established in paras 6.72–6.78 of Ref. [1] and

on the recommendations given in section A.16 (in particular, paras A.1626–A.1645) of Ref. [2], including detailed discussions and examples of related topics.

1.3. SCOPE

The report will be particularly useful to safety analysts and reviewers in fulfilling the requirements and recommendations related to carrying out safety analyses and preparing SARs. In addition, it will help regulators to conduct safety reviews and assessments of the topics covered. It will be also useful to research reactor designers and operators.

The report may be applied, to varying degrees, to all research reactors. Sophisticated calculations and methods such as those described herein may be used as deemed necessary for the preparation of safety analyses of newly designed or reconstructed research reactors. These methods may also be used for updating or reassessing previous safety analyses of operating research reactors when one or more of the following circumstances occur:

— The results of the application of conservative assumptions and simplified methods do not meet the applicable acceptance criteria.
— The review of the safety analysis of an operating research reactor reveals deficiencies in previous analyses.
— New accident scenarios are envisaged or postulated during the operation of the research reactor.

1.4. STRUCTURE

Sections 2–4 of this publication provide information and background on various aspects of source term derivation, and considerations to be taken into account when evaluating radiological consequences. Section 2 deals with the general aspects of the analytical approach to source term derivation (deterministic or probabilistic); application of the graded approach to analysis of research reactors according to reactor type, size or power level, utilization and fuel characteristics; identification of PIEs and scenarios that may lead to core damage; and selection of the scenarios. Section 3 discusses the derivation of the source term. This includes: determination of inventories of fission products, transuranic elements and activation products; analysis of the core damage mechanisms and progression; and evaluation of radionuclide releases from different kinds of fuel and from experimental devices, taking into account

retention in systems and components. It also discusses releases from the reactor containment. Section 4 addresses the analysis of radiological consequences, grouped into three categories: those on the site inside the reactor building, those on the site outside the reactor building and those off the site. External and internal doses are analysed, taking into account the various pathways for gaseous and liquid releases. Section 5 provides an overview of an integrated approach to the derivation of the source term and related radiological consequences. Appendices I–VII complement the technical information provided in Sections 2–4, and Annexes I–III present practical examples from specific research reactors.

2. GENERAL CONCEPTS AND METHODS FOR DERIVATION OF THE SOURCE TERM

2.1. GENERAL ASPECTS

Several different approaches can be followed for source term derivation, ranging from a purely deterministic approach through a combination of deterministic and probabilistic approaches to a fully risk based approach.

2.1.1. Safety analysis of research reactors: Evaluation approaches and methods

The evaluation of reactor safety includes analysis of the reactor response to a range of PIEs that could lead either to anticipated operational occurrences or to accident conditions. These PIEs include malfunctions, failures and human errors, and external events. Selected PIEs for consideration when preparing the PIE list for the safety analysis of a particular reactor are presented in the appendix to Ref. [2]. Ideally, this list will also include PIEs related to experimental devices. Relevant guidance on conducting the analyses and presenting them in the SAR is given in section A.16 (Safety Analysis) of Ref. [2]. The analysis should include consideration of the safety aspects of specific systems (e.g. erroneous handling and failures) and their effect on the reactor. Guidance on the analysis of PIEs related to experimental devices is given in Ref. [3].

The range of PIEs must cover all credible accidents that could have an influence on the safety of the reactor. The methodology applied in this analysis

predicts the risk and the safety margin associated with the operation of the reactor, and must demonstrate that both are acceptable to the regulatory body. The analysis must cover all accidents identified as design basis accidents (DBAs). In addition, the safety analysis of the reactor must cover beyond design basis accidents (BDBAs) for the purposes of emergency planning and accident management. A BDBA is an event whose probability of occurrence is considered to be very low but that is still theoretically possible; the consequences of a BDBA, should it occur, may be quite important. Some DBAs and BDBAs could lead to the release of radionuclides to the environment (the source term).

One approach commonly used in the conduct of safety analyses of research reactors is to assume a hypothetical accident that results in a limiting source term, that is, one producing the most severe consequences. An alternative approach is to perform a detailed assessment of accident progression for a number of accident scenarios to derive several different source terms. Irrespective of the approach used, the radiological consequences of any releases to the environment [2] must be evaluated in terms of the estimated radiological doses in the three categories defined in Section 1.1 above.

Various approaches and methods are also available for evaluating factors necessary to determine the hazard associated with the accident scenarios. Simple manual calculations based on fission product yield inventory tables, conservative fission product fractional release data and site specific atmospheric dispersion data may be adequate in many cases. However, a more detailed evaluation of the above factors using more sophisticated methods will often result in more realistic and substantially lower dose estimates.

Annex I to Ref. [2] provides detailed guidance on the safety analysis approach and the methods used to carry out the safety analysis. It deals mainly with deterministic techniques, which are associated with the deterministic approach, but also provides guidance concerning the use of techniques related to the probabilistic approach.

2.1.1.1. Deterministic approach

Deterministic techniques are characterized by conservatism and are based on defined sets of rules for event selection, analytical methods, parameter specifications and acceptance criteria. These techniques provide reasonable assurance that the ultimate objective of determining a limiting source term can be achieved without performing complex calculations, because the methods used tend to overestimate the amount of radioactive release. The

most severe releases (arising from either a DBA or a BDBA) are taken into account in the selection of a site or in setting design requirements for the engineered safety features (ESFs) of the reactor. These releases may also be used for the purpose of emergency preparedness.

In this approach, the choice of accidents to be considered is based on experience and engineering judgement, without taking into account the probabilities associated with the event sequences, which are necessary for defining the concept of risk associated with the operation of a particular reactor. As a consequence, it is generally recognized that the deterministic approach has limitations with regard to effectively treating system interdependencies and common cause failures.

The safety analysis approach and methods must cover a wide spectrum of research reactors. Because of the limitations inherent to deterministic techniques, Ref. [2] points out that probabilistic techniques may be used to supplement the deterministic evaluations. The application of probabilistic techniques could lead to significant improvements in the understanding of the accident sequences and consequences.

2.1.1.2. Probabilistic approach

The probabilistic approach assumes that all reactor accidents are possible and that any number of simultaneous failures may occur, although their probabilities of occurrence may be very low. It also takes into account that some accidents or accident combinations may have less serious consequences than those used in the deterministic approach, but, when weighted by their likelihood, may represent a real risk and may impose different demands on the reactor design.

The probabilistic approach uses the techniques of probabilistic safety assessment (PSA), which:

— Provide methods to identify accident sequences that may be derived from a broad range of PIEs;
— Lead to significant improvements in the understanding of system behaviour and interactions, and of the role of operators under accident conditions;
— Quantify the risk of reactor operation to the environment, to the public and to site personnel.

Specific guidance on using this technique in the safety analysis of research reactors is given in Refs [5, 6].

2.1.2. Design and operational considerations

In contrast to power reactors, research reactors are used for a wide variety of purposes, resulting in many different design features and operational regimes. Design and operational characteristics may also vary significantly due to experimental devices, which can impact performance and safety. These differences, together with the need for great flexibility in the use of research reactors, demand a particular approach to achieving or managing safety at a research reactor.

Most research reactors have relatively small source terms, and thus a smaller potential for hazard to the public compared with power reactors. However, they may pose a greater potential hazard to operators and on-site workers.

The nature and magnitude of a possible radioactive release from a research reactor facility depend on the accident scenario, that is, the initiating event and the event sequence that caused the release. Different accident scenarios generally have characteristic developments or progressions that lead to different 'typical' release patterns.

PIEs and accident sequences should be selected using a graded approach, as described in Ref. [1]. The following subsections discuss a number of design and operational characteristics important for derivation of the source term and analysis of radiological consequences.

2.1.2.1. Reactor type

While loss of coolant accidents (LOCAs) and power excursions induced by rapid insertion of excess reactivity can be postulated for many reactor types, there are numerous other PIEs that are more or less probable depending on the type of reactor in question. Moreover, the consequences of similar PIEs may be quite different for different facilities.

Open pool reactors, for example, offer free access to the core and are quite flexible with regard to allowing changes to core geometry. Because of this flexibility, various incidents associated with the insertion, removal or relocation of fuel elements, experimental devices and other core components become possible.

In sodium cooled reactors, a serious accident scenario can involve sodium fires and violent sodium–water reactions.

2.1.2.2. *Reactor size or power level*

Whether or not an important group of accident scenarios can develop in a particular reactor is determined by the power level of the reactor. Many research reactors having a power level of less than 1–2 MW cannot reach melting temperatures for their fuels, even under conditions of a complete LOCA. One can envisage a meltdown incident involving one or more of the fuel elements in such reactors only under nearly adiabatic conditions where almost no heat transfer from the fuel to the environment takes place.

A loss of flow accident (LOFA) in a small or medium-sized (having a power level of a few megawatts) research reactor is unlikely to cause serious fuel damage, provided the reactor is shut down in time and the fuel remains immersed in water and is geometrically unobstructed such that natural convection can take place.

Small and very low power reactors are not likely to suffer serious damage from a LOCA, since natural airflow around the core normally suffices to remove the residual heat from the shutdown reactor. However, power excursions are technically possible in these reactors. For example, in SLOWPOKE type reactors, it periodically is necessary to add reflector shims to the core to compensate for the reactivity reduction caused by fuel burnup. This is an intentional reactivity insertion operation and must be carried out taking appropriate care to avoid excessive addition of reactivity.

Finally, it should be pointed out that the term 'power' as used above refers to the continuous mode of reactor operation ('steady state'). In the case of pulse power reactors (e.g. a TRIGA type reactor), very high peak power levels (in the range of several GW) can be achieved without any core melting and/or release of fission products.

2.1.2.3. *Reactor utilization*

Many research reactors follow a duty cycle consisting of periods of operation and shutdown as a normal routine practice. This practice affects the fission product inventory and has some influence on the stored energy of the reactor's core. In this sense, the reactor duty cycle affects the potential release from the reactor.

The term 'utilization' refers to the types of research or other purposes for which the reactor is operated. Loops for materials testing, cold and/or hot neutron sources and other experimental devices can leak, undergo an overpressure transient, cause fuel melting or give rise to some other sequence of occurrences that may result in an incident or serious accident.

2.1.2.4. Fuel characteristics

The design objectives of research reactor fuel differ from those of power reactor fuel owing to the fuel's required thermal performance. Research reactors have much higher demands on fuel specific powers (i.e. power generated per unit mass of fuel). Consequently, it is important to be able to control the power densities to prevent the fuel temperature from approaching the melting point at any point within the core. It is also important to note that fuel temperature limits are sensitive to irradiation time and tend to decrease with increasing fuel burnup.

Other fuel temperature limitations exist. For example, for metallic fuels the fuel temperature must be kept below the phase transition point. Low thermal conductivity, which is evident in cermet type fuel, leads to large temperature gradients, which can cause fuel cracking and swelling. Fuel elements subjected to temperatures high enough to induce centre line melting will tend to experience a significant increase in the probability of failure.

The fuel element cladding design is also very important with regard to fission product release potential. The cladding is subjected to stresses from fuel irradiation and the thermal history (e.g. fuel temperature variation due to power level changes). Blistering of the cladding could occur as a result of these stresses at temperatures well below the cladding's melting temperature. Significant quantities of fission product gases could be released at the onset of blistering of the cladding.

Another limitation is frequently placed on the peak heat flux in water cooled reactors that can be transferred from the cladding to the coolant. Above certain critical heat flux (CHF) values, the heat transfer will become unstable, resulting in the formation of a vapour film on the cladding surface, which reduces the heat transfer coefficient and causes the cladding surface temperature to increase drastically. This shift from a stable to an unstable heat transfer process can lead to cladding failure and the subsequent release of fission products if the cladding's melting temperature is approached or exceeded. Thermohydraulic correlations are available to predict CHF under typical research reactor operating conditions.

Manufacturing defects are another fuel design concern. Examples are: inadequate bonding between the fuel meat and the cladding; excessive local inhomogeneities in the fuel meat loading, leading to hot spots; and excessive local variations in the fuel meat thickness, again leading to hot spots.

2.1.2.5. Specific design features

The design feature that most obviously affects the potential release of radiation from a given reactor is the performance of the reactor's means of confinement. For most research reactors, overpressurization of the containment or building is not a serious problem. For research reactors having the potential for accidents resulting in high reactor building pressure, a robust, leaktight containment may prevent any significant radioactive releases to the environment.

Also important are the ESFs employed. Emergency core cooling systems, for example, can overcome certain LOCA scenarios, which narrows down the range of PIEs that must be considered as 'source term' initiators.

Other design features that influence the range of possible incident scenarios are the core design, the fuel design and the reactor's thermohydraulic design. Core designs that feature strong negative reactivity coefficients from temperature and coolant voids tend to be self-limiting in a wide range of reactivity insertion incidents, preventing core damage in these cases. Similarly, core designs that limit the excess reactivity that may be inserted as a result of inappropriate fuel handling (in the core, or in an experimental loop) or the removal of an experimental device are less prone to developing serious damage from operational mistakes of this nature. The SLOWPOKE reactor, for instance, is not accessible to the operator. The reactor's thermohydraulic design determines whether the core can be safely cooled by natural circulation when forced circulation is lost (e.g. pump failure, electrical power loss). The nature, location and response characteristics of the instrumentation installed in the reactor determine how fast and how reliably a fault condition can be detected and mitigated.

2.1.2.6. Maintenance and periodic testing

A rigorous programme for maintenance, periodic testing and inspection is established, among other reasons, to ensure that all safety related systems remain functional in accordance with their original purpose and specifications without the presence of the design authority.

The source term depends on both active and passive components of the containment and reactor design. It should be emphasized that active or passive components for which credit is taken in the safety analysis also need to include safety related parameters such as the reactor building leakage rate, the emergency ventilation rate and the iodine filter decontamination factor. Credit for the proper functioning of any active or passive components may only be taken in the source term evaluation if the components are under a

strict programme of maintenance and periodic testing. This is particularly important for the emergency ventilation, since its efficiency may be drastically affected by degraded or failed seals in doors and locks, by loss of liquids in traps or by failure of seals in building penetrations. It is also of high significance for any installed high efficiency particulate air (HEPA) and iodine filters in the exhaust duct, and to a lesser degree for any recirculation ventilation system.

A detailed description of the requirements for and performance of maintenance and periodic testing is given in Ref. [4].

2.1.3. Actual source terms experienced in historical nuclear accidents

Appendix I presents a summary of past research reactor accidents involving fuel failure, including the amount of radioactive material released and the radiological consequences. For many of these accidents, the radiological consequences were minimal. The most significant groups of accidents from the point of view of the radiological consequences were reactivity additions involving the manual manipulation of control rods or other operational errors, local losses of flow due to blockage of a few flow channels, and human errors involving experimental facilities.

Appendix II presents the results of destructive reactor experiments: BORAX, SPERT and SNAPTRAN. For these experiments, the overall fractions of activity released from the reactor building ranged between 0.4 and 21%.

2.2. POSTULATED INITIATING EVENTS AND ACCIDENT SCENARIOS POTENTIALLY LEADING TO CORE DAMAGE

Initiating events originate from component failures, system malfunctions, human errors, external events or a combination of these, either in the reactor itself or in one of its experimental devices. PIEs should be grouped according to both their impact on the integrity of the reactor core or other components and the protective actions designed to deal with an occurrence of the events. The main reason for grouping PIEs is to analyse quantitatively only the limiting cases of each group. The following is an appropriate grouping or categorization that takes into account the categorization used for PIEs recommended in the appendix to Ref. [1]:

(1) Loss of electrical power supplies;
(2) Insertion of excess reactivity;

(3) Loss of flow;
(4) Loss of coolant;
(5) Erroneous handling or malfunction of equipment or components;
(6) Special internal events;
(7) External events;
(8) Events involving experimental devices.

It should be emphasized that not every group of PIEs is applicable to every type of reactor. On the contrary, some of the above groups may not be applicable to many types of research reactor; for example, excess reactivity insertions (group 2) may not be applicable to reactors having very large negative temperature coefficients of reactivity, and unbalanced heat removal (groups 3 and 4) may not be applicable to low power reactors.

Another major cause of system or component malfunction is the mishandling or improper re-installation of systems or components during maintenance or repair work. This is not explicitly mentioned among the PIEs, since it may be the cause of any of the PIEs listed.

2.2.1. Selection of postulated initiating events and accident sequences

An approach to the selection of PIEs for credible accidents is outlined in annex I to Ref. [2]. A more comprehensive review of all possible PIEs can be ensured by performing a Level 1 PSA. By establishing event trees for all possible accident sequences originating from those initiating events, one can determine the accident sequences that lead to a source term. The likelihood of these accident sequences can be evaluated by probabilistic methods. By quantifying the corresponding event trees, one can rank these sequences according to their frequency of occurrence. Guidance on the application of PSA to research reactors can be found in Refs [5, 6].

A cut-off frequency of occurrence should be defined in order to limit further analysis efforts to only those sequences that make a significant contribution to the overall risk. A more detailed discussion of the most important PIEs is given in the subsections below.

2.2.1.1. Power excursion due to insertion of excess reactivity

Insertion of excess reactivity can lead to a source term because of its potential to deposit significant amounts of thermal energy into the fuel at a rapid rate. This initiating event is postulated to determine a limiting specific insertion of reactivity, covering accidents that may happen because of failures during: (i) a critical experiment; (ii) reactor startup; (iii) fuel element loading or

unloading; (iv) manipulation or operation of equipment close to the reactor core or of other components including experimental devices; or (v) removal of large absorbers from the core. This type of PIE is usually analysed in the SAR of a research reactor because of the frequent changes of core configurations or experimental devices that take place during the reactor lifetime.

The accident scenarios associated with this PIE usually lead to the meltdown of a fraction of the reactor core if no credit is given to the actuation of the protection or shutdown systems. The meltdown fraction may range from one or a few fuel plates or rods to a significant fraction of the core, depending on the characteristics of the reactor and accident sequence. These accidents may be analysed with system codes such as PARET [7] and RELAP5 [8], which include different degrees of complexity.

The amount of fuel that may melt can be estimated on the basis of experimental research or through conservative, simple manual calculations. However, on certain occasions, more sophisticated models and codes are used to estimate the source term, particularly for higher power research reactors. Among these are codes used in the source term analysis of severe accidents in power reactors such as MELCOR [9], SCDAP [10] and MAAP4 [11].

In any case, a precise estimate of the fraction of the fuel meltdown requires thermohydraulic calculations, which tend to be very sensitive to input parameters. In particular, some specific fuel element features, such as emissivity of the fuel plate surface, will greatly influence the temperature distribution and the maximum fuel temperature of the reactor after the excursion.

2.2.1.2. Loss of flow accident

Reduction of the coolant flow as a result of various initiating events such as pump or valve failure, channel blockage or flow redistribution may lead to cladding failure due to overheating. The probability of such an accident depends on the design of the reactor core and the type of forced flow (upward or downward). However, most SARs of research reactors include one or more of these accidents leading to fuel failure and the need for source term evaluation. For some research reactors, these accidents are considered as DBAs, and some ESFs are built in to cope with their consequences.

The accident scenarios range from the damage of one or several fuel plates (cladding failure) due to overheating to the meltdown of one or several fuel plates or even fuel elements in research reactors using MTR type fuel. The event sequence depends on the specific scenario. This reduced flow condition eventually undergoes a transition to natural convection cooling, in the case of

pump coastdown, or to stagnant coolant followed by rapid coolant voiding (e.g. steam explosion), in the case of core blockage.

2.2.1.3. Loss of coolant accident

A LOCA may lead to fuel damage resulting in a source term in the core of a research reactor above a certain power level. This level varies according to the fuel design and is typically around 2 MW for research reactors using plate type fuel. Higher power reactors usually incorporate appropriate ESFs to avoid such an accident. However, the consequences of a LOCA need to be examined for all reactors, including those at the lower end of the range, to determine the likelihood of fuel damage. Even if the likelihood is found to be very low, the potential for direct irradiation of operating staff or personnel due to loss of shielding must also be examined.

A LOCA leading to partial uncovering of the core should be analysed to determine its potential for fuel damage. In some cases, it could result in boiling of the water adjacent to the immersed part of the fuel. Although in this case conduction and steam cooling may be adequate to prevent melting, a cladding failure could take place.

Most research reactors with power levels above 2 MW have incorporated adequate ESFs such as antisiphon devices, elevation of primary pool pipework above the reactor core, use of pool liner throughout beam tubes, automatic operation of the beam tube shutter and redundant water storage tanks plus core spray systems or other emergency core cooling systems (ECCSs). However, the SARs of some of these reactors include small LOCAs with the meltdown of a fraction of the core as a DBA. In these cases, it is advisable to conduct a careful study of the most limiting LOCA sequence that could lead to fuel damage. This study should take into account the action (or failure) of the incorporated ESFs or of any other components that are vital to ensuring core cooling. The frequency of this accident sequence should be shown to be low enough to demonstrate that the overall risk associated with the meltdown of large portions of the core is acceptable.

2.2.1.4. Fuel handling accident

Fuel handling accidents include fuel dropping, dropping of the transfer cask on fuel elements and fuel uncovering. The potential hazards are mechanical damage to the fuel, insufficient cooling leading to melting and the possibility of a reactivity accident.

2.2.1.5. Accident involving experimental devices

Most research reactors have a range of experimental facilities associated with them, from relatively simple material irradiation systems to high pressure fluid loops carrying out fuel irradiation in support of nuclear power programmes or containing cold/hot neutron sources with neutron moderators having the potential for reactions. Some of the experiments have the potential to release radionuclides to the environment and to damage the reactor core and its associated systems. Therefore, such experimental facilities have, according to their hazard potential, their own safety functions and safety systems to protect not only the experiment but also the reactor. The influence of all such experiments, including their own safety features, must be fully considered in the context of overall reactor safety. Further guidance on the subject can be found in Ref. [3].

2.3. SELECTION OF ACCIDENT EVENTS AND SCENARIOS FOR SOURCE TERM ASSESSMENTS

This section describes important features concerning the selection of accident events and scenarios for detailed studies related to the derivation of source terms.

2.3.1. Influence of accident frequency

For source term assessments, low frequency BDBAs with core damage are usually investigated. In principle, various combinations of failures leading to the onset of core damage (and fission product release) can be postulated. However, a cut-off occurrence frequency is very useful and ideally will be established along with a quantification of the associated uncertainties. This permits a focus on risk-dominant scenarios without the use of bounding conservative prescriptions (e.g. as in Ref. [12], which prescribes the use of 100% of the noble gases, 50% of the iodine inventory and 1% of the rest of the core fission product inventory) that may unduly overstate the risk profile, especially for on-site personnel and experimenters.

Reference [2] provides guidance on the safety analysis of research reactors. It recommends that States develop their own acceptance criteria and encourages them to apply probabilistic approaches. Hence, acceptable risk to on-site personnel and off-site residents is also established on a case-by-case basis. The selection of an appropriate cut-off frequency for various events is tied to the acceptable risk.

The selection of a cut-off frequency is also closely tied to what is perceived to be 'physically' reasonable or credible. Recent safety analyses of power reactors suggest the use of values of ~10^{-6} to 10^{-7} per year as a cut-off point below which numerical description of probability is unreasonable. Use of these cut-off frequency values may exclude consideration of a few to several initiating events. Guidance on reducing the number of scenarios is also provided in Ref. [2], and detailed examples of the application of PSAs to research reactors are given in Refs [5, 6].

Reference [13] provides an example of results of a Level 1 PSA study for the Advanced Neutron Source Reactor (ANSR) project. The rank-ordered list of leading core damage accidents indicates that events resulting from core flow blockage would dominate. Indeed, with a cut-off frequency of ~10^{-6} per year, detailed source terms would be evaluated only for a flow blockage and for LOCA events. Other events, despite their potential to lead to core damage, would be considered not to be physically credible and would not be included in source term assessments.

2.3.2. Other factors

Selection of accident scenarios for source term assessments depends on the specific characteristics of the research reactor in question (e.g. type of fuel, power, utilization). In this context, we can distinguish among:

— Reactors using U–Al and U–Si fuel, with aluminium cladding;
— Reactors using UZrH (TRIGA reactors), uranium metal or uranium oxide fuel with Zircaloy, Inconel or stainless steel cladding;
— Reactors having fuel irradiation pressurized loops, where the partial meltdown of the irradiated fuel is a normal occurrence in the experimental programme.

For reactors having sufficiently low power levels, the PIEs corresponding to loss of flow or loss of coolant may not be important. However, in the case of a LOCA, the uncovering of the core must be assessed, mainly because of the risk of direct irradiation.

For reactors using U–Al fuel and having a sufficient amount of excess reactivity in the core (>1%($\Delta k/k$)), the risk of meltdown of the fuel due to reactivity insertion accidents must be assessed. In several States, a 'BORAX type accident' [14] leading to the total melting of the core is taken into account as a DBA.

For TRIGA reactors with a pulsed mode capability and strong prompt negative feedback, reactivity insertion need not be considered as a PIE, provided

that this reactivity does not exceed the limiting design reactivity producing the power pulses. For source term assessment of TRIGA reactors, the SAR considers a cladding rupture during the handling of an irradiated fuel rod.

In the case of reactors with a pressurized fuel irradiation loop, an accident corresponding to a molten fuel–coolant interaction should be considered and the associated risk of core damage should be assessed. In the case of reactors with cold or hot neutron sources, accident situations are to be evaluated on the basis of their impact on the reactor core and reactor safety systems. Finally, for reactors using heavy water as a reflector or as a coolant, or having cold neutron sources or any other experimental device utilizing deuterium, special consideration should be given to the evaluation of the tritium source term and to doses associated with tritium releases.

3. DERIVATION OF THE SOURCE TERM

3.1. FACTORS INFLUENCING THE SOURCE TERM

The release of radioactive substances from a research reactor to the environment (the source term) depends on the following factors:

- The inventory of fission products and other radionuclides in the core (or the inventory in experimental devices or other locations such as the spent fuel pool or isotope production facilities);
- The progression of core damage (or failure of experimental devices or isotope production facilities);
- The fraction of radionuclides released from the fuel (or from experimental devices or other locations), and the physical and chemical forms of released radioactive materials;
- The retention of radionuclides in the primary cooling system;
- The performance of means of confinement (e.g. emergency ventilation rate, filter efficiency, leak rate, liquid effluent release rate, radioactive decay due to time delay of release, deposition on surfaces and resuspension).

In addition, the doses associated with the source term depend on the release mode (single puff, intermittent, continuous) and the release point (stack, ground level, confinement bypass). The above factors are taken into

consideration by the methods presented in detail in this section; first, however, it is pertinent to discuss the following general considerations.

3.1.1. General considerations for source term calculations

Over the past few decades, considerable work has been undertaken to realistically evaluate source terms for core meltdown accident sequences for power reactors and, in some cases, for research reactors. Several important conclusions have been drawn from these research efforts [15]:

- Fuel inventory source terms can be calculated reasonably well using analytical methods.
- Generally, source terms have tended to be grossly overestimated.
- Source terms are facility specific; generalization of source term values from one reactor to another is not possible; each research reactor must be considered individually according to its design and mitigatory features.
- The magnitude of releases depends on the research reactor performance characteristics, in particular the performance of the means of confinement.
- Uncertainties may be significant in predicting building release source terms, and the significance of many of these uncertainties varies with the timing and mode of confinement failure.
- Realistic assumptions will usually result in lower source terms and less severe consequences.

The level of detail to which the safety analysis, including the derivation of the source term, has to be carried out will depend on the potential hazard presented by the research reactor, which to some extent depends on the operating power, the fission product inventories, the engineered and inherent (physical) safety features, and the nature of the accident sequences that are considered credible. The sophistication of the calculation methods employed to determine source terms and doses will be influenced by the magnitude of the potential consequences, the complexity of the techniques, the availability of data and codes, and the regulatory requirements. Where realistic assumptions or data are not readily available, conservative assumptions are to be used.

Conservative assumptions will greatly simplify the calculation effort, but often can lead to predicted consequences that are more severe than can realistically be expected. Realistic assumptions, on the other hand, will usually result in source terms and consequences that are less severe. However, in some instances the use of realistic data and assumptions will be complex and may involve substantial effort. This is particularly true for determining fission product

releases within the reactor, the primary cooling system and the reactor building. In addition, the results of such calculations are highly specific to the individual reactor/facility design and to particular accident sequences. To avoid sophisticated calculations (unless they are considered necessary to determine a realistic source term for a particular reactor, mainly owing to potentially severe consequences), the use of conservative release fractions may be preferable. For the same reason, the use of various conservative assumptions often is a regulatory requirement. In other areas, however, realistic calculation methods are normally used, and validated computer codes are available. Examples include the determination of realistic fission product inventories, atmospheric dispersion modelling and dose calculations.

It is also usual to assess reactor specific building/containment designs and associated ESFs on a case-by-case basis. Performance of confinement and air cleaning features can be crucial to mitigating releases, even where the building is not designed for pressure loads as a tight containment system. This point is apparent from the limited releases associated with the severe reactivity excursion accidents at the NRX reactor at Chalk River [16] and the SL-1 reactor at Idaho Falls [17], and others discussed in Ref. [18]. Confinement performance is specific to the facility design and the nature of the accident sequence. Event tree analysis can be used to assess the structural and system response and performance to ultimately characterize release mechanisms.

As research reactors usually have much lower fission product inventories than power reactors, as well as less stringent requirements for post-shutdown heat removal, many of the more complex analytical techniques have not generally been applied to research reactor analyses, and the methods currently applied often are relatively simple. The following factors should be considered in determining the need for an extensive analysis:

— Reactor type, associated irradiation and experimental facilities, and specific hazards posed by them;
— Reactor design features, including fuel type, coolant/moderator systems and other features influencing releases to the building;
— Reactor operating history and modifications made, if any;
— Type of accident sequence and factors that would influence releases (fire, pressure pulse, loss of pool water, structural integrity of the confinement building, operator actions, etc.);
— Availability of validated modelling methods and empirical data for realistic release assumptions.

Fuel melting as a result of LOCAs or LOFAs is not a concern for low power research reactors. However, the extent of the analysis required for

higher power research reactors will depend critically on the reactor design and the limiting DBA. Design features such as routing of coolant piping and beam tube locations will influence the fraction of the core exposed to air during a LOCA, and thus the fraction of the core that might degrade. This complexity arises because it is more usual to assume overly large, conservative release fractions and to show that the resulting consequences are acceptable. However, the use of realistic assumptions, supported by experimental or accident data, may substantially reduce the source term.

Some estimates of fission product releases may be obtained from experimental or accident data. However, this type of information may be highly specific to the reactor type and to the experimental or accident sequence conditions. Thus, extrapolations are to be made with caution. Some examples of radioactive releases from materials testing reactors (MTRs) are given in Appendix II.

Factors important in determining the source term — namely, fission product inventory, fuel damage progression, radionuclides released from the fuel, retention of fission products in the primary coolant and confinement performance — are discussed in detail below.

3.1.2. Atmospheric dispersion modelling

The dispersion and deposition of material released to the atmosphere are typically modelled as a plume. One such model is based on a Gaussian plume [19] with Pasquill–Gifford dispersion parameters. Simple plume models can simulate phenomena such as buoyant plume rise, wake effects on plume dispersion caused by obstructions such as buildings, and wet and dry deposition. Time dependent radioactive buildup and decay in the plume can also be calculated. Some newly developed codes on accidental dispersion, transportation and deposition of radioactive material are described in Refs [20, 21].

3.2. FISSION PRODUCT INVENTORY

3.2.1. Effect of reactor design and duty cycle

The inventory of fission products and other radionuclides in the reactor fuel and core depends on a number of factors such as:

— Quantity of fissile material, reactor power and burnup;
— Neutron flux distribution in the core, operating history (including transients) and fuel management;
— Load of irradiation targets for radioisotope production, materials testing, etc.

The power level and operational performance assumed in the calculation of the source term are often taken as the maximum values for which the licence is issued.

Computer programs such as ORIGEN-2 [22] and FISPIN [23] are often used to calculate fission product inventories for various operating histories (i.e. operating power sequences over time) as a function of operating or decay time. Fission product inventories are strongly dependent on the power cycling; thus the use of codes to accurately account for operating history may result in more realistic inventories. It must be kept in mind, however, that these calculations are only to be used if it can be shown that bounding inventories have been attained.

3.2.2. Fission product characteristics

The large number of fission and activation products that are formed during the fission process can be grouped into a small set of categories of elements with similar physical or chemical behaviours. The radionuclide classification scheme used in the Reactor Safety Study (Ref. [24]) is given in Table 1.

The nuclides of interest in source term calculations are gaseous, volatile and semi-volatile nuclides, since these are the most likely to be released from overheated fuel elements. The gaseous elements are the noble gas isotopes of

TABLE 1. RADIONUCLIDE CLASSIFICATION SCHEME USED IN THE REACTOR SAFETY STUDY [24]

Class	Relevant radionuclides
Noble gases	Xe, Kr
Halogens	I, Br
Alkali metals	Cs, Rb
Tellurium group	Te, Se, Sb
Alkaline earths	Sr, Ba
Transition metals	Ru, Mo, Pd, Rh, Tc
Rare earths/lanthanides and actinides	La, Nd, Eu, Y, Ce, Pr, Pm, Sm, Np, Pu, Zr, Nb

krypton and xenon, and the volatile elements are iodine, caesium and the tellurium group, except antimony. The semi-volatile elements, roughly in order of decreasing volatility, are: ruthenium, antimony, barium, strontium, cerium and lanthanum, among others. The rare earths and actinides have much higher boiling points and usually remain dissolved in the fuel [25].

Precursor sources of radionuclides of interest such as iodine can be determined from their decay chains and yields. Precursor sources are frequently neglected, but can be important under some circumstances. For example, the post-shutdown production of ^{131}I from ^{131}Te and the production of ^{135}Xe from ^{135}I are of importance and should be considered. On the other hand, tellurium reacts strongly with some core materials such as zirconium, delaying its release. Thus, each reactor and possible accident sequence type must be considered on an individual basis. Frequently, the iodine fractions are increased by some conservative factor to allow for precursor production.

A further consideration is desirable when selecting which radionuclides contribute significantly to the dose. With medium sized power reactors, it usually suffices to consider the following set of radionuclides:

— Whole body: noble gases (particularly ^{88}Kr, ^{135}Xe and ^{133}Xe);
— Thyroid: iodines (particularly ^{131}I, ^{133}I);
— Lung/internal: volatile nuclides (e.g. ^{131}I, ^{132}Te, ^{106}Ru, ^{134}Cs, ^{137}Cs) and, for scenarios of high core temperatures (>1000°C), ^{90}Sr.

Although some radionuclides deliver a skin dose, they are not major contributors to the limiting dose, and it is usual to neglect the skin dose.

Tables of radionuclide and fission product inventories for a generic 10 MW research reactor are contained in IAEA publications (e.g. Ref. [5] and appendix D-1 to Ref. [26]). End of cycle inventories for the 20 MW SAFARI reactor and the 35 MW SILOE reactor are provided in Annex II to this publication.

3.3. OTHER RADIONUCLIDE INVENTORIES

For research reactors, the amount of activity of other radionuclides (e.g. activation products and transmutation isotopes) usually is significantly less than that of the fission products; therefore, other radionuclides do not contribute significantly to the source term and accident consequences. Thus, the source term for these other radionuclides is of significantly less importance and in many cases may be omitted from the source term evaluation and consequence analysis or only included as an approximate estimate. Depending

on the initial enrichment of the fuel, however, the production of transuranic radionuclides by activation of uranium may have to be considered in the source term evaluation.

Also, with specific reactor designs, special applications or experimental facilities, large inventories of other radionuclides may be present or activated under special circumstances, and a separate source term evaluation for these radionuclides may be required. Experience from research reactors shows that failure and malfunctioning of irradiation devices and experimental facilities, and the resulting releases to the building, are more probable than are releases due to core damage. Therefore, a source term and consequence evaluation must also be performed for radionuclide inventories other than the fission product inventory, in particular for irradiation devices, isotope production facilities and experimental facilities, especially with regard to consequences inside the reactor building. This may be of particular importance for reactors with special safety designs to protect the core so that the potential source term for the fission product inventory is very low or essentially zero. In such cases, the possible release of radionuclides from experimental facilities may pose the most significant source term potential.

3.3.1. Transuranic elements

Transuranic elements will inevitably be produced in the core, but the production rate will be inversely dependent on the initial enrichment of the research reactor fuel and its burnup.

Highly enriched uranium (HEU) fuel may have a greater than 20% concentration of ^{235}U. In the case of 20% enrichment, the plutonium inventory of a HEU core would not be much different from that of a low enriched uranium (LEU) core of any enrichment level. In the case of HEU of greater than 50% enrichment, the plutonium inventory would be very different from that in the LEU case.

By definition, LEU fuel has an initial concentration of ^{235}U of less than 20% (i.e. more than 80% ^{238}U) and, therefore, a greater potential for plutonium buildup with increasing burnup. In research reactors operating with natural uranium (0.71% ^{235}U and 99.29% ^{238}U), plutonium buildup is important and must be included in the source term evaluation if the reactor is operated at higher power levels. Depletion codes such as ORIGEN [22] provide good estimates of the inventory of actinides for specified irradiation histories. An irradiation history for a complex variation of power versus time can always be simplified to an equivalent sequence of constant power irradiations.

It has been demonstrated, however, that, even with LEU fuel of 20% enrichment, the inhalation dose resulting from an accident would not increase

significantly compared with that from an accident involving HEU fuel [26]. Therefore, compared with HEU fuel, plutonium buildup in the LEU fuel does not significantly increase the radiological consequences. For these reactors, the contribution of plutonium to the radiological consequences may be neglected in the evaluation without significantly altering the result. The conversion of a research reactor from the use of HEU to LEU fuel, such as is demonstrated in Ref. [26], may not require a new licensing procedure with regard to the accident source term and consequence analysis.

3.3.2. Activation products

With certain irradiation devices, significant amounts of radionuclides may be formed. Some of these radionuclides may be in a gaseous or liquid form, for which the probability of a release may be more significant than with solid activation products. This is particularly true for ^{41}Ar buildup in air filled containers and loops.

The accidental release of radioactive material contained in such irradiation devices or experimental loops may pose a significant hazard to on-site personnel, in particular to the personnel working with or close to irradiation devices, and must be included in a separate source term and consequence evaluation.

With activation products in irradiation devices, particular attention should be given to the fact that the possible accidental release of radioactive material may occur to an area outside the confinement building, with different release configurations and retention parameters.

3.3.2.1. Activation of core materials

With special reactor designs incorporating non-water coolants such as liquid metal coolants, the activity inventory (^{24}Na, ^{42}K, etc.) in the coolants may be significant. A special source term evaluation may be required for the case of a break in the coolant piping or other possible leakages. This may be also the case with D_2O coolants and the tritium contained therein. As in the case of activation products in irradiation facilities, attention must be given to the possibility of leakages from the coolant pipes occurring in areas outside the confinement building, incorporating retention conditions (e.g. no filters) and release configurations (ground release rather than stack release) different from those with a release inside the confinement building.

Reactors employing heavy water as a coolant or neutron reflector can generate substantial quantities of tritiated heavy water with potentially significant dose consequences owing to the high inventory of tritium. Typical

values of tritium inventory buildup amount to ~3.7×10^9 Bq·L^{-1}·MW^{-1} per year of operation. The actual buildup will depend on the operational strategies employed (i.e. maximum buildup allowed prior to detritiation by dilution with less tritiated heavy water).

It should be emphasized that, with light water coolant loops, no significant long lived activity inventory is produced that would require a separate source term calculation.

3.3.2.2. Experimental devices

In experimental facilities, experimental loops and isotope production loops, special activity inventories may be produced by planned utilization or under accident conditions, which may lead to accidental exposure of persons on the site or, possibly, off the site. Some examples are the unforeseen simultaneous production of short lived radionuclides of high activity in an irradiation device, the accidental rupture of piping and the accidental connection of an argon bottle instead of a pressurized air bottle to irradiation devices such as rabbit systems.

Since these experimental facilities may incorporate ducts for the activated nuclides to areas outside the confinement, special attention must be given to possible leakages or radioactive releases outside the confinement building or outside the designated area resulting in possible irradiation of on-site personnel. A more detailed discussion of the source term calculation requirements for such devices is given in Section 3.5.2.

Possible impacts of experimental devices on core integrity and fission product source terms are considered in the assessment of PIEs and accident scenarios in Section 2.2.

3.4. CORE DAMAGE MECHANISMS AND PROGRESSION

Knowledge and characterization of core damage progression and fuel relocation are central to many aspects of BDBA assessments. The areas where such characterization becomes useful include:

— Fission product release and transport;
— Recriticality (if configurations are reached that enhance criticality upon molten fuel relocation, leading to a burst of reactivity or a pulsating mode and loads to the confinement system);
— Accident management/mitigation activities;

— Evaluation of challenges to the confinement system [27, 28];
— Cleanup activities (through knowledge of the extent of fuel debris relocation in the system).

In general, fuel meltdown progression can occur explosively (with fuel material dispersion) or in a quasi-static mode. Some of the key issues to be addressed during evaluation of fuel meltdown progression are presented in the sections that follow. The possibility, degree and extent of fuel damage progression will vary with the reactor type, power level and accident sequences (including operator actions).

Adequate front end analyses to determine the onset of fuel damage progression are also assumed. Such front end analyses may range from simple assumptions and postulates to relatively sophisticated coupled neutronic–thermohydraulic assessments. Simple, albeit bounding, assumptions (e.g. postulating a large core melt fraction) may prove adequate for certain classes of reactor systems (i.e. if the resulting source terms and consequences are proven acceptable). An example of the use of such assumptions is provided in Ref. [27]. For situations where such bounding assumptions prove unacceptable, more refined and realistic evaluations can contribute to significant source term reduction. References [28, 29] provide examples of such mechanistic assessments for front end scenarios.

A review of past research reactor fuel damage events (see Appendix I) indicates that fuel meltdown is more likely for reactors employing U–Al type fuels. Therefore, the features on fuel meltdown progression presented in the remainder of this section are in relation to reactors employing U–Al type fuel designs. As a cautionary note, this discussion is not to be construed as drawing safety conclusions related to the potential for fission product releases from TRIGA fuels versus U–Al fuels. The important subject of fission product releases from various fuel types is treated separately later in the report.

To set the basis for subsequent discussions, various typical phenomena to be considered (for capturing the physics of fuel meltdown progression in U–Al reactors) are depicted pictorially in Fig. 1. Various aspects of core damage progression are described in more detail in subsequent sections of this report.

3.4.1. Cladding failure

During normal operation, cladding failure may be induced by:

— Manufacturing defects;
— Corrosion phenomena;

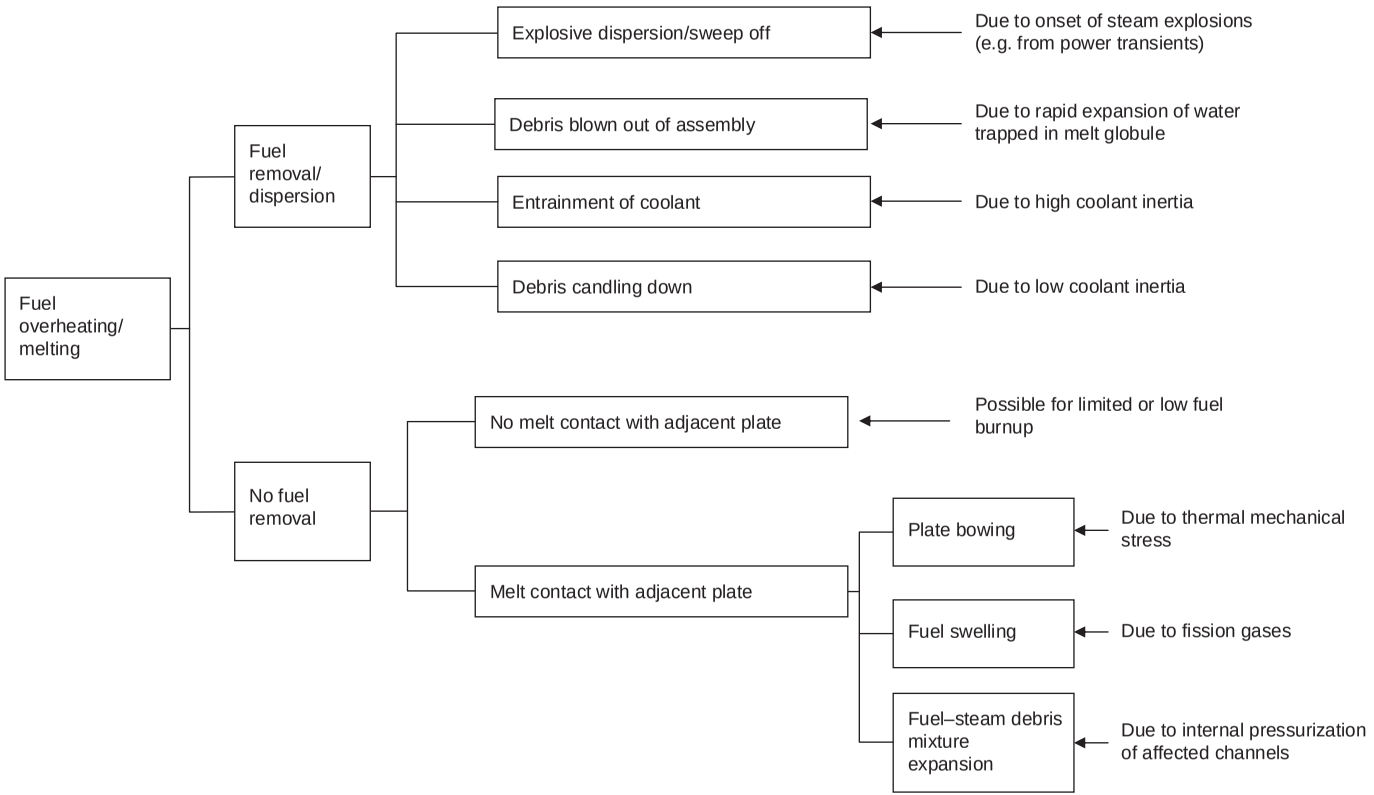


FIG. 1. Core damage propagation and relocation modes.

— In-core damage due to vibration;
— Damage due to improper handling;
— Power ramping.

Under accident conditions, cladding failure may occur due to:

— Overheating and melting because of insufficient cooling;
— Excessive cladding oxidation;
— Interaction between fuel and cladding.

Tubular and plate type aluminium cladding fuels are the most commonly used research reactor fuels. For such fuels and for heat-up rates of less than 100°C/s, experimental results [30–32] have shown three stages of fuel failure. First, blistering of the aluminium cladding is observed at temperatures about 100°C below the melting point of aluminium. This blistering occurs when the internal gas pressure at the fuel–cladding interface and the cladding temperature allow the cladding to locally debond from the fuel. The onset of blistering is not a strong function of burnup. The second and third stages of fuel failure are cladding cracking and cladding melting, respectively.

However, some aluminium alloys have a melting temperature significantly below the melting point of aluminium (e.g. the solidus temperature of Al-6061 is 582°C). The release of fission gases starts when the cladding blisters. In the case of alloy fuels, most of the noble gases will be released at about 650°C. Details of fission product release from U–Al fuels are provided in Section 3.5.1.2.

3.4.2. Reactivity transients

Reactivity transients have the potential to introduce significant amounts of thermal energy into the fuel at a rapid rate. If the magnitude and rate of the energy deposition are high enough, various levels of fuel and cladding melting can occur, with the potential for superheating all the way to aluminium vaporization. Key phenomena affecting the progression of fuel melting relate to onset of structural deformation, loss of rigidity (leading to melt relocation) and onset of fuel–coolant interactions (FCIs) (such as steam explosions). Fuel–coolant interactions can result in considerable fuel melting fragmentation, shock pressures and gross system energetics (e.g. as in the incident at the SL-1 reactor, in which the reactor vessel was displaced upwards by about 3 m).

Fortunately, recent analyses [33, 34] of data from the NSRR and TREAT facilities for plate type U–Al fuels indicate that initiating a steam explosion in

a plate type configuration requires significant melt superheat. These experiments in the NSRR and TREAT facilities were conducted with small coupon samples of fuel plates. Details of the assessments are provided in Ref. [33]. The assessments reveal that the onset of steam explosions for U–Al fuel plates, which are submerged in water and subjected to bursts of reactivity insertion, coincides with a sharp inflection in the strain rate imposed on the material (as is also known to occur in situations involving spallation of materials during high velocity impact dynamics). Further, the inflection in the induced material strain rate was found to coincide with either (i) plate surface temperatures exceeding the melting temperature of aluminium oxide or (ii) the onset of aluminium vaporization at any location within the fuel meat section of the U–Al fuel plate.

It should be kept in mind that the test data obtained from the NSRR and TREAT facilities largely involved the use of fresh (i.e. unirradiated) fuel plates. Only one test was conducted in the TREAT facility involving the use of a fuel plate with a burnup of approximately 1 wt%. This test resulted in steam explosions and fuel dispersion. However, the rate and magnitude of the energy deposition were high enough to have caused dispersion even in fresh fuel. Thus, data are lacking for provision of firm guidance on fuel dispersion thresholds at various burnup levels. Overall guidance on quantification of the effect of fission gases on possible reductions in the required energy deposition threshold is given in Ref. [34].

Fuel meltdown progression can also be affected by the onset of structural changes (such as excessive bowing and blistering). Results of tests conducted in the NSRR facility with various U–Al fuelled miniplates provide useful guidance on energy deposition thresholds. Analyses of these tests are reported in Ref. [35].

3.4.3. Loss of flow accidents

Loss of flow accidents under full power or decay heat conditions can lead to a significant reduction of heat removal capability. Loss of flow conditions can result from several causes, including:

— Loss of pumping power;
— Core inlet flow blockage;
— Loss of pressure in the cooling system.

A principal mechanism contributing to damage propagation is the onset of flow excursions induced by parallel channel (i.e. Ledinegg type) instability. This can be initiated by the onset of significant voiding (in the subcooled

boiling flow regime) in any of the parallel flow channels. The well-known Saha–Zuber correlation that predicts the point of net vapour generation may be used for this purpose. Voiding can be initiated in flow channels as a result of system depressurization, flow blockages from debris or manufacturing defects, or loss of circuit flow. Flow starvation in several channels would require that the nuclear heat be transferred to the coolant in neighbouring unaffected channels. If the heat generated in affected channels cannot be dissipated, then fuel heat-up to melting may occur. Upon melting, fuel foaming (with the amount being dependent on burnup) can lead to plate contact with neighbouring plates, and hence to propagation of damage in the fuel assembly. Various other mechanisms for damage propagation are identified in Fig. 1. Reference [35] provides further details.

Another important feature of many reactors is that liquid coolant flows downward through the core. This feature is undesirable from the view of damage initiation and propagation, especially for high power reactors. Any debris lodged at the entrance to the core will tend to stay entrapped. Also, during loss of pumping accidents, coolant flow reversal from forced to natural convection modes will lead to periods of stagnation and possibly to the onset of core damage.

3.4.4. Loss of coolant accidents

Although parallels may be drawn between LOCAs and LOFAs, flow degradation under LOCA conditions is assumed to occur when the water level in the reactor core has dropped significantly (e.g. due to a ruptured beam tube), such that fuel plates can transfer energy only to a gaseous environment. Under these conditions, if a sufficient heat sink does not exist, then core melting and subsequent melting propagation could occur, starting from the regions of highest power densities. Once again, fuel foaming needs to be considered under high burnup conditions. Fuel melting and relocation downward can be expected to occur in a candling type mode, or as rivulets. One aspect to be concerned about is associated with the formation of a molten aluminium pool at the lower core support plate. Based on experiences in the aluminium industry and on several large scale (e.g. >5–10 kg of melted aluminium) tests, the possibility of steam explosions and the resulting consequences need to be considered. Reference [28] provides an example of such an assessment for research reactors.

3.4.5. Chemical reactions

The energy source customarily utilized for fuel melting considerations is the nuclear decay heat (upon successful reactor shutdown) or the nuclear fission energy for reactivity excursions. While these may be reasonable for many situations, two additional energy sources based on chemical kinetics may play a predominant role under certain thermohydraulic conditions and for certain material compositions. Chemical energy sources may result from the onset of thermite type reactions (e.g. between U_3Si_2 and aluminium), or from reactions between the fuel and cladding materials (such as uranium metal, aluminium or Zircaloy) and water. The thermal energy released from such exothermic or melt–water reactions can be very significant (e.g. up to about 1 MJ/kg of mixture for reactions between U_30_8 and aluminium, about 0.4 MJ/kg of U_3Si_2–Al for typical LEU fuels and about 17 MJ/kg for reactions between aluminium and water), as it takes about one megajoule to heat up and melt a kilogram of aluminium.

Under certain circumstances, these chemical reactions can proceed explosively (i.e. within milliseconds), generating several gigawatts of power. Therefore, they have the potential to significantly alter the course of meltdown progression and source term generation.

Finally, and not directly connected with fuel reactions, some experimental devices like cold neutron sources using H_2 or D_2 as the cold neutron moderator have the potential to produce explosive H–O chemical reactions, with a consequent impact on the mechanical integrity of the reactor core and associated beam tubes, eventually triggering damage to fuel elements.

3.4.6. Relocation of fuel

To obtain an accurate source term, it is necessary to assess the amount of fuel that may melt in a particular accident sequence. For high power density reactors, it is possible that the fuel may relocate, leading to the need to consider steam explosions [28] or recriticality (especially in cases where HEU fuel is used) [36].

Fuel relocation can follow different paths (see Fig. 1), depending on whether the molten fuel remains in place or is swept away by coolant inertia or the sudden expansion of water trapped in the melt. If the molten fuel is removed from the channel, fuel damage propagation will only occur by the transfer of heat radiation. Relocated fuel may attain a stable, coolable configuration and solidify, thus limiting further release of radionuclides. It may also form a molten metal pool with the potential for an explosive interaction with

water. For highly enriched fuels, potential recriticality needs to be addressed. Further details concerning these issues are given in Ref. [36].

3.4.7. Fuel handling accidents

A fuel handling accident in which a fuel element is dropped could lead to mechanical damage to the fuel or heat-up of the fuel element to melting if, for example, channel flow reduction occurs. The release of fission products in this case, however, is limited and comparable with that resulting from a fuel channel blockage (Section 3.4.3). If the fuel element has an extensive cooling period, no melting of fuel plates will occur; however, for accident analysis and source term derivation, limiting conditions are to be assumed and no credit is to be taken for the cooling period.

For metallic fuels used in research reactors, fission products will remain trapped within the fuel as long as the fuel temperature remains below the level at which blistering occurs.

It is important that possible spent fuel melting or failure also be included in the source term evaluation for fuel handling events in the spent fuel bay. This is particularly important for spent fuel that is stored within the containment structure.

For separate spent fuel storage facilities, the possibility of a reactivity accident due to organizational errors or erroneous handling of spent fuel may need to be assessed as possible input for a source term evaluation.

3.5. RADIONUCLIDE RELEASES FROM FUEL AND EXPERIMENTAL DEVICES

3.5.1. Fission product releases from fuel

3.5.1.1. Factors influencing releases

It is well-known that radioactive fission and activation products generated in nuclear fuel constitute the principal source of hazard to on-site personnel and to the public off the site, especially following a BDBA. As a consequence, considerable experimental [37–41] and analytical [42] work has been undertaken over the past few decades to evaluate fission product releases from various types of research reactor fuel and to establish models for inclusion in computer codes for BDBA assessments.

As may be expected, fission product releases from research reactor fuels depend on various factors such as:

— Accident sequence;
— Fraction of core affected;
— Fuel temperature;
— Fuel environment (e.g. oxidizing or reducing);
— Fuel burnup;
— Rate of heat-up;
— Fuel type;
— Fission product chemistry.

A mechanistic approach to evaluating releases with sufficient accuracy is at best a complex and daunting task. Fortunately, for most applications of practical interest, non-mechanistic, empirically based approaches (correlations) are adequate (as is the case for similar assessments for BDBA analyses for power reactors). Such approaches may be employed directly via manual calculations or, as necessary, implemented in system codes such as MELCOR [9] for evaluation of the source term and releases to the environment.

Most of the research reactor fuels currently in use can be broadly classified into the following types:

— U–Al fuel (either rod or plate geometry);
— UZrH TRIGA fuel;
— Uranium metal fuel;
— UO_2 fuel with zirconium, steel or aluminium cladding.

For fuel types similar to power reactor fuel rods, well developed models are available, ranging from simple models such as CORSOR-M [18] to fully mechanistic codes such as FRAPTRAN [43]. In subsequent sections of this report, various aspects associated with each fuel type are described and correlations for predicting various volatile fission product species are given.

3.5.1.2. Fission product releases from U–Al fuels

A vast number of materials testing, isotope production and beam research reactors use U–Al fuels. The fuel forms generally fall into the following categories:

— U–Al alloy fuel cladding with aluminium;
— Dispersed UAl_x fuel cladding with aluminium;
— U_3O_8 cermet type fuel cladding with aluminium;
— U_3Si_2 or U_xSi_y fuel particulates in cladding with aluminium;
— LEU UMo fuel (under development).

A concise summary and assessment of the published data is provided in Ref. [42]. A few important points for practical use are discussed below.

For all U–Al fuels, the release of volatile fission products is negligible (<0.1%) below 773 K. At higher temperatures, noble gas release occurs in three distinct stages for all fuel types (except silicide fuels). The first stage corresponds to the onset of fuel plate blistering at around 833 K. The next coincides with the solidus temperature of aluminium of around 855 K. The last stage corresponds to the eutectic temperature of around 923 K, at which point almost all the noble gases have been released. Negligible quantities (<0.1%) of the other volatile species are released up to this point. Silicide fuel types exhibit significant retention capability for most volatile fission product species, including the noble gases. However, this retention capability depends considerably on the degree of fuel burnup.

As a cautionary note, the term 'negligible' as used above supposes application to BDBA analysis. What is 'negligible' for such a purpose may be unacceptably large for other situations such as accidents involving fuel handling.

For volatile species other than the noble gases (e.g. iodine, caesium, tellurium and ruthenium), it generally has been found that:

— Oxidizing environments enhance releases of caesium, iodine and tellurium;
— The amounts of radioactive material released increase with burnup;
— The rate of release varies with time and temperature;
— Smaller amounts are released from dispersion fuels.

A library of empirical correlations for release from various U–Al fuels for each individual fission product class (i.e. noble gases, iodine, caesium, tellurium and ruthenium) has been developed. The general correlation for predicting releases can be represented by:

$$\mathcal{R}(t, T) = \mathcal{R}(120, T)t/120 \quad (\text{for } t < 120 \text{ s}) \text{ and} \tag{1}$$

$$\mathcal{R}(t, T) = \mathcal{R}(120, T) + \mathcal{R}(120, T)[\mathrm{K}'(\mathrm{T}) - 1](t - 120)/3480 \quad (\text{for } t > 120 \text{ s}) \tag{2}$$

where

$\mathcal{R}(120, T)$ is the amount (%) released over 120 s of heating time (accounting for effects of environment and burnup);
t is time;
T is the fuel temperature (K);

K′(T) is a time dependent parameter obtained from heat-up tests over 120 and 3600 s.

Additional details can be found in Ref. [42].

The correlation $\mathcal{R}$ has been developed in three forms: the well-known CORSOR form, the CORSOR-M form and the trend line (i.e. polynomial). The CORSOR form is given in an exponential form as $A\exp(BT)$, and the CORSOR-M form is given in the Arrhenius form as $A\exp(-Q/RT)$, where A is a dimensionless constant, B is a constant for a specified temperature range, Q is the activation energy (kCal/mol), R is the universal gas constant ($= 0.00199$ kCal·mol^{-1}·K^{-1}) and T is the temperature (K).

An extensive set of constants for each of the three forms of the correlation is provided in Tables 8–10 in Appendix III. The reader is referred to this appendix when evaluating release fractions for particular fuel types and for specific fission product species of interest. It should be noted that no single form provides universal accuracy for all fuel types and for all species. Therefore, Table 11 in Appendix III provides a correlation form giving the best accuracy. Further details on possible adjustments for burnup dependence, etc., are available in Ref. [42].

3.5.1.3. Fission product releases from UZrH (TRIGA type) fuels

Reference [41] provides details of experiments and derived data for releases of volatile fission products from UZrH fuels. Both high uranium loaded (45 wt%) and low uranium loaded (8.5 wt%) fuel samples were tested, and a single correlation conservatively bounding the release fractions of noble gases (^{85m}Kr, ^{87}Kr, ^{88}Kr and ^{89}Kr) and iodine species (^{133}I and ^{135}I) was obtained. The correlation takes the form:

$$\mathcal{R} = 1.5 \times 10^{-5} + (3.6 \times 10^{3}) \left[\exp\left(-1.34 \times 10^{4}/T\right)\right] \quad (3)$$

where T is the fuel temperature (K).

For temperatures below 400ºC, recoil releases are predominant and the first term of the equation is controlling. For higher temperatures (>400ºC), releases are mainly governed by a diffusion like process and the second term becomes the dominant contributor.

It is useful to keep in mind that release fractions given by this equation assume failed or ruptured cladding. Unless cladding is breached from other stresses or mechanisms, fuel and cladding temperatures need to be at least 960ºC for internal pressures to be high enough to cause cladding rupture in TRIGA fuels [41].

3.5.1.4. Fission product releases from other fuel types

While the majority of research reactors employ U–Al or TRIGA type fuels, there are cases where other fuel types are utilized, the most common being uranium metal fuel, followed by UO_2 fuel. Data for fission product releases from irradiated uranium metal plates in steam–helium and steam–air mixtures were obtained by Parker et al. [38]. Notably, Parker et al. found that data for the steam–air (i.e. the most probable) environment compared well with earlier data obtained for air. For completeness, selected data are summarized in Table 2.

For UO_2 fuel, the well-known correlations (e.g. those used in CORSOR-M) or mechanistic models (e.g. FRAPTRAN) developed for power reactors generally can be applied.

3.5.1.5. Releases from fuel during rapid temperature ramps and explosive events

The aforementioned correlations for releases from U–Al reactor fuels were developed for the evaluation of releases of volatile fission products at relatively low (<10 K/s) heat-up rates. However, rapid temperature excursions leading to fuel and cladding melting may significantly change the internal driving gradients, and therefore the magnitude and rates of release. Very few well characterized data are available in the literature to provide firm guidance for such cases. Data from destructive tests in facilities like SPERT indicate that, upon the onset of steam explosions, most of the volatile fission product inventory of the melted fuel is released within milliseconds. There is also evidence of such behaviour from testing with pre-irradiated U–Al fuel samples in the TREAT facility.

Until systematic data become available, the evidence in the interim suggests that, for non-explosive conditions, the correlations presented in the preceding sections can also be used for predicting release magnitudes. For conditions where a steam explosion is predicted to occur, the assumption of

TABLE 2. SELECTED STEAM–AIR ENVIRONMENT DATA

Temperature (°C)	Released fraction (%)				
	Noble gases	Iodine	Caesium	Ruthenium	Tellurium
800	100	48	0.06	73	2.9
1000	100	89	0.4	77	80
1200	100	99	16	85	96

100% volatile fission product release from molten fuel is 'reasonably' conservative. As a point of clarification, this is related to the magnitude released to the medium surrounding the molten fuel, but not necessarily to the source term released as a gas plume to the environment (since removal mechanisms such as water dissolution and plate-out may considerably reduce the amount of fission products actually in the plume).

3.5.2. Radionuclide releases from experimental devices

Usually, the radionuclide inventory in experimental devices is much lower than the activity inventory of the core. Thus, the hazard associated with failures of experimental devices — in particular the off-site consequences — will be considerably lower than the hazard associated with fuel failures in the reactor core. In many cases, there will be no off-site hazard at all. However, experimental devices may pose special hazards to on-site staff, particularly to experimenters and personnel operating the reactor, for various reasons, including the following:

— Experiments often change and quite often vary in their set-up.
— Experiments and irradiation facilities often involve rapidly changing levels of activity in the inventories of radioactive materials, or the levels of activity are not well-known or estimated in advance.
— Owing to their ad hoc character, experiments and irradiation facilities usually have a higher probability of failure than does the reactor installation itself.
— Depending on the nature of their use, experimental devices are usually located close to operating personnel.

In particular, releases may occur from incidents relating to any of the following situations:

— Irradiation of a nuclear material (e.g. ^{99}Mo production from fissile targets, fuel rod testing under transient conditions up to or beyond failure);
— Frequent handling operations (e.g. loading and unloading of targets, sampling at glove boxes);
— Transfer of radioactive material from the reactor core to some out-of-pile equipment (e.g. on-line analysis of fission gas releases during destructive fuel testing, irradiation of targets in a rabbit system);
— Excessive ^{41}Ar production within air loops close to or inside the core;
— Direct radiation risk at neutron beam ports;

— Production of certain radionuclides (e.g. H_2/D_2 cold neutron sources or in-core ^3He circuits with out-of-pile control may transport an appreciable amount of tritium).

3.6. RETENTION OF RADIONUCLIDES IN THE PRIMARY COOLING SYSTEM

3.6.1. Retention of fission products

The transport of fission products is dependent on their chemical and physical forms. The chemical form is determined by the chemical properties of the species; the presence of other fission products as well as structural materials and impurities; and the environmental conditions. The equilibrium chemical composition of a mixture of fission products can be readily calculated, for example, using the computer code SOLGASMIX [44]. For temperatures above about 1000 K, chemical equilibrium is achieved very quickly and the kinetics of the reactions can usually be ignored. At lower temperatures, the equilibrium takes longer to become established and the effect of chemical kinetics may need to be considered.

Accident sequences where radionuclides are released into water will be characterized by a substantially reduced or delayed release of the fission product to the containment structure.

For noble gases, it is usually assumed that 100% of the fraction released from the fuel to the primary coolant system is released into the containment structure; however, experimental data [45], data from an accident with core degradation under water [46] and results of destructive reactor tests [14] indicate that not all of the noble gas content in water is released to the atmosphere. This may be due to bubble entrainment in the primary cooling system and dissolution of non-condensables within the coolant according to Henry's law. In the case of dissolution within the primary coolant, the dissolved gases may come out of solution and be transferred to the containment volume at a later time.

In a tank type reactor, fission products may be deposited in the piping of the primary cooling system. Volatile fission products in the form of vapours or aerosols, excluding the noble gases, will deposit on the cooling system surfaces. The predominant mechanisms are vapour condensation and vapour chemisorption, and aerosol deposition by sedimentation, impact, thermophoresis, turbulent motion and Brownian diffusion. Resuspension can also occur and is influenced principally by temperature, flow, gas concentration, etc. The transport of fission products through the primary cooling system can be

analysed by modelling the transport and deposition of aerosols, as is done by computer codes such as TRAP-MELT [47].

In pool type reactors, a significant reduction of the source term can be achieved for fission product aerosols through the process of pool scrubbing. The decontamination factor due to pool scrubbing is dependent on various factors such as bubble size, aerosol particle size, steam fraction, subcooling and water depth. The scrubbing efficiency of pools also depends on the mode of gas/vapour injection, and the presence and type of non-condensable gases. Aerosol retention in subcooled water pools has been studied extensively for the following conditions: a low carrier gas flow rate, a single orifice type (~1 cm diameter) injector and a two-phase bubbly flow regime. Computer codes such as SPARC and BUSCA predict decontamination factors (DFs) under such situations quite well. In general, the DF strongly depends on the steam content of the carrier gas. Steam condensation promotes particle removal by diffusiophoresis through the bubble–water interface. Thus, the DF increases with the mass fraction of the steam. The DF typically shows a minimum particle size in the range of 10^{-1} to 1 μm. Larger particles (>1 μm) are removed mainly by centrifugal deposition, whereas smaller particles (<10^{-1} μm) are retained by diffusion. An example of a calculation of radionuclide behaviour in water pools is given in Ref. [48].

The chemistry of iodine under accident conditions is complex and is still under investigation in power reactor source term studies. The iodine released from the fuel is most likely to be in the form of elemental iodine. Upon its release into gaseous media, iodine is likely to react with caesium, forming CsI, or with hydrogen, forming HI, depending on the relative timings of the releases of iodine and caesium, the temperature of the fission products and the length of time they have to react. In either case, once in the water, CsI or HI will decompose to form I^-.

Under accident conditions, iodine is known to exist in the atmosphere in several volatile chemical forms, including I_2 and organic iodine (e.g. CH_3I). Several chemical processes may be responsible [49]. The airborne organic compounds of iodine are less likely to be affected by removal processes that are effective for aerosols or elemental iodine (e.g. plate-out or spray system removal). The process of producing volatile iodine is significantly affected by radiolysis and involves the interaction of iodine with various organic compounds (such as paints and coolant impurities) to form volatile organic species. Both these processes are enhanced at a pH of less than about 8. A number of complex models exist that allow modelling of the complex iodine chemistry in the coolant and atmosphere, which can be used for realistic best estimate assessment of iodine chemistry for research reactor accidents.

As has been mentioned, one major factor that affects the rate of volatile iodine production in the aqueous phase is the pH. Another important factor is

the rate of mass transfer between the aqueous and gaseous phases, especially for situations where pool water boiling takes place. An upper bound value can be obtained by assuming equilibrium between iodine in the steam and iodine in the liquid phase. An effective rate constant for iodine removal from the water is given by:

$$P_{evap} = (steam/V_{liq} \times pc) \quad (4)$$

where

steam is the boiling rate (m^3/s);
V_{liq} is the volume (m^3);
pc is the partition coefficient.

The partition coefficient is used as a measure of volatility and is defined as the ratio of equilibrium concentration in aqueous solution to equilibrium concentration in gas.

It follows, therefore, that small partition coefficient values are correlated with high volatility, and vice versa. Reference [50] provides data on iodine partition coefficients in water for temperatures up to about 185°C (the boiling point of iodine), which should cover most conditions of practical interest for research reactor situations. For more updated results, Ref. [51] gives an assessment of the partition coefficient for trace and high concentration solutions at high pressures.

3.6.2. Retention of other radionuclides

In general, the transuranic elements are retained in the fuel up to very high temperatures and are not usually released to the coolant (except under steam explosion type conditions).

For heavy water and liquid metal reactors, the coolant materials can be activated, forming tritiated heavy water (DTO), and ^{24}Na and ^{42}K, respectively. These remain dissolved in the coolant but can be released into the containment atmosphere with the evaporating coolant. Activated corrosion products from reactor piping and core structures are entrained and remain in the coolant.

Irradiated materials from experimental facilities can contain fission products, in the case of a fuel test, or neutron activation products from other experimental facilities. (See Section 3.6.1 for a discussion of the retention of fission products in the coolant.) The retention of activated material from experimental facilities within the primary cooling system is case dependent. Radioactive liquids, which are miscible in water, are generally retained within

the cooling system with some release due to evaporation. Activated gases, such as ^{41}Ar, which is produced by the irradiation of air contained within experimental facilities, can be treated in a manner similar to that used for noble gases, as described in Section 3.6.1.

3.7. CONFINEMENT PERFORMANCE AND RELEASES

Retention of radioactive material in the containment structure may be a very important factor in reducing the source term. The efficiency of source term reduction predominantly depends on the following factors:

- The emergency ventilation rate;
- The containment leakage rate;
- The radionuclide hold-up time influencing decay and precursor production;
- Surface deposition and resuspension;
- Removal by filters incorporated in the emergency ventilation system;
- Additional cleaning by filtered recirculation systems, and liquid and gaseous hold-up systems;
- The liquid effluent release rate;
- The capacity of containment structures to withstand potential challenges from steaming and explosive loads (e.g. missiles from internal steam explosions).

With regard to the impact on the environment and the radiological consequences, the release point from the containment, the release mode (single puff, intermittent or continuous release) and the energy content of the release are also very important.

If an emergency ventilation system is available, its exhaust rate is considered to be an important factor for reducing the source term, since this rate, in combination with the containment volume, determines the extent of radioactive decay before release to the environment. It also has a significant impact on the residence time of the air in the containment and therefore has an impact on the amount of radionuclides deposited on the surfaces inside the containment. The emergency ventilation rate may also affect the filter retention factor of charcoal filters for iodine removal, in particular for accident scenarios with high humidity in the exhaust air.

High efficiency particulate air filters have efficiencies of >99.97% for particulates, depending on the flow rate. If the emergency ventilation rate is sufficiently low (<0.2 m^3/s), a typical 5 cm thick bed of activated charcoal can be

credited to remove up to 99.9% of iodine, depending on the chemical form of the iodine, the flow rate, water vapour, etc. [52].

If liquid effluents are released to an area outside the containment structure, the assessment needs to evaluate the possibility and extent of releases of volatile radioactive species from the liquid effluents into the atmosphere.

3.7.1. Basics of confinement performance

The rate of release to the confinement building atmosphere will depend on the accident sequence and release mechanisms. Usually, the release will take place over some time period. However, since the release rate will not significantly change the exposure outside the confinement, an instantaneous release is assumed in the calculations given below.

If the conservative assumption of such a rapid release to the containment is made, the time dependent change in airborne activity following the release may be expressed as follows:

$$\frac{\mathrm{d}q(t)}{\mathrm{d}t} = -\lambda q(t) - \frac{F}{V} q(t) - Y \frac{S}{V} q(t) + Y \cdot r \frac{S}{V} q_s(t) - \frac{f}{V} \eta \cdot q(t) + P(t) \tag{5}$$

where

$q(t)$ is the airborne concentration of a given radionuclide at time t after the release to the containment volume (Bq/m^3);
$q_s(t)$ is the surface concentration of the radionuclide (Bq/m^2);
$\lambda q(t)$ is the radioactive decay rate (Bq·m^{-3}·s^{-1});
$(F/V)q(t)$ is the removal rate due to ventilation and leakage (Bq·m^{-3}·s^{-1});
F is the building exhaust rate (normal and emergency ventilation and leak rates) (m^3/s);
V is the building volume (m^3);
$Y(S/V)q(t)$ is the surface deposition rate (Bq/s);
Y is the deposition velocity (m/s);
S is the affected surface area (m^2);
$Y \cdot r(S/V)q_s(t)$ is the surface to volume resuspension rate, where r is the resuspension rate (s^{-1});
$(f/V)\eta\, q(t)$ is the recirculation/cleaning removal rate, where f is the recirculation flow rate (m^3/s);
η is the filter efficiency factor (dimensionless);
$P(t)$ is the precursor production (Bq·m^{-3}·s^{-1}).

This expression may be simplified if the precursor term is neglected and is instead included with the initial release using a conservative factor. If the surface to volume resuspension is accounted for by a conservative decrease in the surface deposition rate, the expression can then be readily integrated to yield the airborne activity concentration over time since the release to the containment:

$$q(t) = q_0 \exp\left(-(\lambda + (F/V) + \nu(S/V) + (f/V)\eta)t\right) \tag{6}$$

where

q_0 is the initial airborne activity, following release to the confinement (Bq/m^3);
ν is the deposition velocity factor Y reduced to allow for resuspension (m/s), with $Y[1 - r\, q_s(t)/q(t)]$ as a constant to simplify the integration.

Only the first two terms of the exponent are applicable to the noble gases, which are assumed neither to deposit on surfaces nor to be affected by filtered air cleaning. All four terms are important for iodine and other radionuclides.

The source term for releases from the containment can now be determined. The total activity of a radionuclide in a containment of volume V is $Vq(t)$. The rate of radionuclide release from the containment to the environment is:

$$\frac{F}{V}(1-\eta_v)Vq(t) = F\ (1-\eta_v)\ q(t) \tag{7}$$

where

F is the exhaust air rate (m/s) for the emergency ventilation system;
η_v is the efficiency of the emergency ventilation system particulate and charcoal filters.

For the normal ventilation system and building leakage, η_v should be assumed to be zero. Thus the total activity of radionuclide i released into the environment at time t is given by:

$$q_i(t) = \int_0^t \frac{F}{V}(1-\eta_v)Vq_i(t)dt \tag{8}$$

or

$$q_i(t) = q_0 \frac{F(1-\eta_v)}{\xi}(1-e^{-\xi t}) \tag{9}$$

where

$$\xi = \left(\lambda + \frac{F}{V} + v\frac{S}{V} + \frac{f}{V}\eta\right)$$

For low power reactors, the source term $q_i(t)$ can be further simplified if conservative release fractions that take no account of deposition are used. Allowance should be made, however, for radioactive decay and for those ESFs from the confinement system that may be credited in the analysis (e.g. air cleaning or filtered exhausts).

The application of detailed methods may not be necessary if the safety analysis requirements for a particular research reactor can be met by using simple, conservative techniques. Various conservative assumptions may be used to estimate the release of radioactive material from the fuel and from the containment structure. It is quite common in research reactor safety assessments, particularly for low power reactors, to assume an arbitrary source term that is larger than that expected for probable accident sequences. This is done to account for uncertainties in the analysis and thus to avoid extended calculations or evaluations. With research reactors of higher power levels, it may be necessary to consider additional factors affecting the release — such as deposition on surfaces in the containment or retention in hold-up systems — in order to meet the site specific dose limits as defined in the facility licence.

3.7.2. Noble gas release

With certain accident scenarios (e.g. release of fission products into or under water), only the release of noble gases may be of importance, because all other radionuclides will be effectively retained in the water pool or containment. Thus, usually the radiological consequences during the initial phase of an accident at a research reactor will be predominantly influenced by the release of noble gases. (For some scenarios, however, other volatile radionuclides, such as iodines, will need to be considered.) Therefore, the reduction of the source term for fission noble gases is of primary interest for meeting site specific limitations for many accident scenarios.

After the release of noble gases from the fuel, the source term is basically controlled only by radioactive decay within the confinement and, possibly, hold-up systems. Thus the primary factor determining the source term is the ventilation exhaust rate or the leakage rate from the building or containment. Therefore, for a research reactor of a higher power level, it may be desirable, in the case of an accident, to reduce the normal ventilation considerably to an 'emergency ventilation rate'. Depending on the power level and accident scenarios, the emergency ventilation rate may be one to two orders of magnitude lower than the normal ventilation rate. However, the ventilation rate should remain sufficiently high to maintain the containment pressure below atmospheric pressure to ensure that no uncontrolled releases occur.

Improvements with regard to the containment leakage rate (and, consequently, the source term and radiological consequences) may be obtained by carefully checking all possible leakage points of the containment. Special attention has to be given to seals of doors and locks, to seals and ducts for electrical wiring and heating, to traps in liquid ducts and to isolation of any air ducts penetrating the containment surface. This also applies to isolation devices closing the normal ventilation ducts.

3.7.3. Retention of particulates and iodine

As Section 3.7.1 makes apparent, for radionuclides other than noble gases, several removal mechanisms in addition to radioactive decay further reduce the activity concentration within the containment prior to release. Since the important particulate and iodine radionuclides have comparatively long half-lives (and thus radioactive decay does not result in their significant reduction even at rather low exhaust rates), these effects are even more important in the reduction of the source term.

The effectiveness of these removal mechanisms depends on several factors:

— The surface deposition of particulates and iodine isotopes depends to a large extent on the residence time within the containment. Therefore, low ventilation and leakage rates resulting in an increase of the time the particulates and isotopes are retained in the containment are essential for effective iodine and particulate deposition on containment surfaces.
— The removal of iodine and particulates by aerosol and charcoal filters during confinement air recirculation also depends greatly on a low emergency ventilation rate. A sufficiently low ventilation rate and a high recirculation rate will allow significant reductions of both particulates and iodine isotopes.

— The retention of iodine by charcoal filters also depends on the residence time of iodine compounds in the filter and thus, with a given filter volume, on the flow rate through the filter. This is particularly relevant with high humidity in the air passing through the filter and organic iodine components. Since high humidity is expected for some of the possible accident sequences, low emergency ventilation rates are important for low iodine release. If low ventilation rates cannot be achieved, large charcoal volumes are required to ensure high retention efficiency.
— The retention of particulates by HEPA filters in the emergency ventilation system does not depend on the ventilation rate. Because of the very high retention capacity, such filters are an extremely efficient way to remove radioactive materials and reduce the aerosol activity before release. However, it is important to ensure, through an appropriate maintenance and testing programme, that there are no significant bypass flows around the HEPA filters.

The deposition of iodine takes place on practically every surface inside the confinement building. Therefore, an adequate estimate of the effect of iodine deposition on the source term would require an assessment of the deposition on all surfaces in the building, taking into account their size and the deposition velocity. Since the deposition velocity on bare concrete and metal surfaces is about one order of magnitude higher than that on painted surfaces, it is particularly important to account for all unpainted surfaces. On the other hand, painted surfaces and PVC can contribute significantly to the formation of volatile organic iodine. The surfaces of mobile systems, although possibly contributing significantly to iodine removal owing to the large area of additional surfaces, are not to be considered in the source term assessment, as they may not be present at the time of the accident.

3.7.4. Confinement challenges for consideration during beyond design basis accidents

In the assessment of the ability of a research reactor containment structure to retain radionuclides, various potential challenges, failure modes and mechanisms arising in the case of BDBAs must be considered.

3.7.4.1. Failure from explosive fuel–coolant interactions that lead to dynamic loads and missiles

In large power research reactors, explosive FCIs, such as steam explosions, may occur under certain conditions in which molten fuel comes into

contact with water. Such interactions may lead to shock waves accompanied by energetic missiles, which may pose a threat to the integrity of the containment.

Most of the severe accidents in research reactors where steam explosions were observed (see Appendix I) were initiated by a rapid reactivity insertion. However, other scenarios, such as flow blockages at high power conditions for high power density research reactors, could also result in a fuel–meltwater interaction that would be similar to a reactivity initiated steam explosion. For example, an extensive study [35] was conducted along these lines for the ORNL 100 MW High-Flux Isotope Reactor (HFIR). This study included modelling and analysis of core melt initiation, melt propagation, steam explosion energy, vessel failure from fractures, bolts failure, and, finally, missile evolution and transport. Aluminium ignition was neglected. Evaluations indicated that a thermally driven steam explosion with more than 65 MJ of energy deposition in the core region over several milliseconds would be needed to cause an energetic missile with the capacity to cause early containment failure. This amounted to ~70% of the HFIR core mass melting and participating in the steam explosion. However, conservative assessments indicated that only up to 24% of the core could melt during any credible large flow blockage event. Therefore, it was concluded that the HFIR vessel and top head structure would be able to withstand loads from steam explosions initiated by flow blockages. Further details are provided in Annex III to this report.

Another aspect to be considered for research reactors with beam tubes is that in-core steam explosions may generate loads large enough to cause rupture of the beam tubes. Such ruptures can open up pathways for in-core fission products to escape into the containment without retention in the water pool, with the potential exposure of reactor staff and experimenters.

3.7.4.2. Failure from static overpressurization caused by steam loads

In the case of an accident, adequate cooling is required to absorb the decay power of fission products or the fission energy generated by pulsating recriticality of fuel debris (if this can occur). As can be inferred, this confinement failure mode refers to situations where loss of primary and secondary heat sinks may result in heating and steaming of the water pool. The containment should be able to withstand overpressurization during credible events with loss of heat sinks. Typically, such a confinement challenge may be expected only in high power research reactors employing highly enriched fuel cores. For example, fuel melt relocation in high power research reactors may lead to end states where the new fuel geometry is conducive to the onset of pulsating criticality. Pulsating criticality could conceivably lead to far greater steam loads than would arise from the fission product decay heat alone, and for

a much longer duration (i.e. as long as sufficient water and fuel are available to attain criticality).

3.7.4.3. Direct bypass and failure to isolate

Owing to their research oriented design characteristics, certain research reactors may encompass potential confinement bypass pathways (beam tubes, experimental loops, filtration devices, etc.) that may permit radionuclides to escape, with direct exposure of the operating personnel and experimenters. This potential challenge to the confinement is reactor design specific and should be considered on a case-by-case basis.

3.7.4.4. Miscellaneous challenges

Other potential challenges that may apply only to very high power research reactors, in addition to power reactors, include: direct melt-through of confinement barriers, loads from molten core–concrete interactions, combustible gas detonation and containment pressurization from high pressure melt ejection. Such challenges certainly will not pose a threat to the majority of operating research reactors, but for high power research reactors they should be systematically considered and excluded through judicious design or other means. An example of such treatment is provided in Ref. [13].

Another set of challenges that may be unique to research reactors involves cold neutron sources. These devices usually employ significant amounts of cryogenic liquids, such as liquid hydrogen or deuterium, as neutron moderators. The relevant safety related issue associated with these materials is their potential for an energetic reaction when mixed with oxygen (fire or detonation, depending on the H_2–O_2 concentration).

The consequences of such a reaction can be:

— Direct radiological hazards resulting from the release of activated gases and impurities, and the spread of activated materials, which may result in a distribution of radioactivity in the reactor building.
— Indirect damage (from pressure waves or missiles) to the reactor core or the reactor's safety systems that may result in a reactor accident scenario. Because of their proximity to the reactor core, the impact on the beam tubes, with a consequent LOCA scenario, must also be considered.

The specific modes by which these interactions could take place must be carefully investigated, and appropriate protective measures must be designed, since a cold source accident scenario may ultimately result in core disruption

and melting. Alternatively, an explosive FCI close to the cold source (e.g. due to fuel melting) may also result in a cold source rupture. If the cold source utilizes liquid deuterium, the possible release of tritium to the containment areas accessible by personnel or experimenters must also be considered and evaluated.

3.7.5. Behaviour of materials released from experimental devices

Usually, the activity inventory of experimental devices is much lower than that of the core. As discussed in Section 3.5.2, a variety of situations can result in the release of some amounts of radionuclides from experimental devices. Generally, it is expected that the behaviour of these radionuclides inside the containment will be similar to that of fission products and can be described by the model presented in Section 3.7.1. However, some special considerations must be kept in mind, such as the possibility of confinement bypass, the presence of cryogenic liquids and the absence of retention in the emergency ventilation filters (e.g. if the emergency ventilation system is not started in the case of an experimental device failure).

4. RADIOLOGICAL CONSEQUENCES

4.1. FACTORS INFLUENCING RADIOLOGICAL CONSEQUENCES

In general, depending on the reactor/facility type, the power level and operating history, and the particular accident scenario, the radiological consequences associated with the source term may range from accidents with practically no consequences to those with releases leading to on-site and off-site consequences. To facilitate their assessment, the radiological consequences may be grouped into the following categories:

— On-site consequences inside the reactor building with doses to operating staff or personnel within the building;
— On-site consequences outside the reactor building from:
  • Direct radiation from the containment;
  • Gaseous or liquid release of radioactive material from the containment;
— Off-site consequences (to members of the public) from:
  • Direct radiation from the containment;

- Gaseous or liquid release of radioactive material from the containment to the environment.

Not all of these consequence categories will be applicable to all reactor types and sites. In low power reactors with limited excess reactivity, radiological consequences are expected to be significant only for overexposure of operating staff inside the building as a result of direct radiation from the undamaged fuel.

In most accident sequences, the most intensive radiation levels are expected inside the reactor building. However, the exposure time in these cases may be kept sufficiently low by evacuating the building or by relying on effective building filters, so that only acceptable doses occur.

4.1.1. Special considerations for research reactors

The approaches used for evaluation of the radiological consequences of research reactor and power reactor accidents are very similar. However, owing to the unique utilization and built-in design features of research reactors, special considerations for evaluation of radiological consequences at these facilities must be taken into account. The following three main features are worth consideration:

— *Usage related routine presence of experimenters and operators at the reactor building*. Unlike power reactor containments, research reactor buildings are usually designed to provide adequate space for experimenters and operators (in some cases, research reactor control rooms are located inside the reactor building). Consequently, radionuclides released to the building atmosphere during a BDBA may directly affect experimenters and control room operators. Since the risk profile may be dominated by in-containment fission product exposures, radionuclide transport pathways leading to the reactor containment require careful characterization for evaluation of radiological consequences. Such a situation would be unusual at a power reactor plant, for which similar situations generally need not be considered.
— *Potential for exposure from usage related confinement bypass pathways*. Certain research reactors include experimental devices such as beam or rabbit tubes leading directly into or located in very close proximity to the reactor core. Some beam tubes may also contain cryogenic materials (e.g. liquid hydrogen cold sources), which for mission related optimization are built with the minimum structural materials necessary (to reduce parasitic capture). Embrittlement fracture, line ruptures and beam tube perforation (e.g. during in-core steam explosion events) may result in

bypass pathways for radionuclides causing direct exposure to experimenters and/or the environment.

— *Research reactor siting*. Unlike power reactors, which have large potential inventories of fission products and are sited far from areas of high population density, many research reactors are located on university campuses or are otherwise surrounded by large populations immediately outside the containment. While this may be perfectly acceptable for facilities with small source terms, the concept of 'on-site and off-site exposures' used to calculate radiological consequences has a different meaning at such facilities than at power reactors. The risk profile to the off-site public under such circumstances may be dominant from the standpoint of distance, even with a drastically reduced source term (compared with power reactor situations). Research reactor siting in a populated area increases the potential for sabotage with possible radiological consequences, and therefore adequate security means must be in place.

4.2. ON-SITE CONSEQUENCES INSIDE THE REACTOR BUILDING

4.2.1. Exposure pathways

A distinguishing feature of research reactors, compared with power reactor facilities, is the presence of large numbers of experimenters within the reactor building. In some research reactors the main control room is located within the reactor building. The exposure pathways to be considered may include:

— Exposure resulting from direct expulsion of reactor coolant–fission product mixtures into the experiment/beam rooms (which may take place as a consequence of beam tube rupture);
— Exposure to burning radioactive gases (such as may take place during cold source deuterium/hydrogen–oxygen reactions);
— Direct exposure to fission product gases and aerosols within the containment (especially of control room operators);
— Exposure to liquid effluents (containing fission products) that bypass containment;
— Exposure to radioactive materials trapped in aerosol and charcoal filter banks (especially if these are located within the containment amid the operations and experiment staff).

The above exposure pathways are somewhat unique to research reactors. They may require specialized methodologies for evaluation of dose and other consequences. For power reactors, in general, such pathways need not be considered.

Radiological consequences inside the reactor building might arise from:

— External irradiation from:
  • Direct radiation fields;
  • Airborne radioactive material;
  • Deposited radioactive material;
— Internal exposure from:
  • Airborne radioactive material;
  • Resuspension of deposited radioactive material.

4.2.2. External exposure due to direct exposure to radiation source

Once the radiation source is defined, doses at relevant points inside the containment are calculated. Different rooms and/or fixed shields need to be represented in as much detail as possible. Fixed shields are understood to be shields that cannot be disassembled and that are included at the design or backfitting stage of the facility. In some cases, shielding will prove to be an adequate and significant dose reduction factor in the accident analysis; before credit can be given to such shielding, it should be guaranteed by physical or administrative procedures that the shields cannot be easily removed.

For simple source/shielding geometry, these calculations can be made manually using an analytical solution of point kernel integrals and proper buildup factors. For more complicated geometries, it is advisable to use more sophisticated methods, such as any available numerical point kernel integration or radiation transport computer code capable of handling the required geometric detail. The use of computer codes is encouraged to avoid tedious repetitive calculations, taking into account the large number of positions that usually need to be evaluated.

Doses to operating personnel and experimenters can be reduced by protective measures. If credit is taken in the accident analysis for the use of any active or passive protective or mitigating measures to obtain acceptable dose values for the operating staff, maintenance and periodic testing must be undertaken to ensure the appropriate performance and availability of these measures.

External radiation exposure of personnel inside the reactor building can occur by direct or indirect (scattered) irradiation from the source. Changes in

shielding set-ups or power excursions with insufficient shielding provisions may lead to considerable doses to operating personnel or experimenters.

The evaluation of doses or dose rates inside the reactor building due to direct or indirect radiation often requires the consideration of special geometrical arrangements between the source and the exposed body; some of these arrangements may be readily taken from references, while others may have to be calculated using more or less approximate methods. In some cases, the dose evaluation requires the calculation of radiation fields in the vicinity of large sources (e.g. reactor pool or tank with little shielding, radioactive substances accidentally released to containers or vessels in the reactor building).

For research reactors having a power level below about 2 MW, under LOCA conditions the most important radiological hazard usually is the intense direct gamma radiation due to radionuclides in the core (if there is loss of shielding) or to radionuclides released from the core into the confinement.

The dose evaluation will involve specific shielding calculations for a source, which may be modelled as a point source, a line source, a surface source or a volume source, depending on how the radioactive source is distributed in space. Since radiation from a series of independent sources is cumulative, a non-uniform line source can be modelled as a series of line segments of different strengths. The same is true for surface or volume sources. The source will be accident specific, depending on the nature of the release, the types of radionuclide released, the resulting radionuclide distribution within the facility, and the geometry and materials of the building structure.

Given an isotopic source whose size is negligible relative to the distance between the source and the receptor point (with only air between), the gamma dose rate calculation can be performed by accounting for geometrical attenuation alone from the formula:

$$D = K_{\gamma} A/r^2 \tag{10}$$

where

D is the dose rate (μGy/h);
K_{γ} is the dose conversion factor (μGy/m^2 per GBq/h);
A is the activity (Bq);
r is the distance (in air) from the source to the receptor (m).

The radionuclide specific dose conversion factors (K_{γ}) are summarized in Ref. [53].

In cases where shielding materials lie between the source and the dose point (e.g. by a reactor wall), the attenuation from gamma absorption and scattering must be taken into account. In this case:

$$D' = De^{-\mu d} \tag{11}$$

where

D' is the modified dose rate behind the shielding (μGy/h);
μ is the attenuation coefficient (1/m);
d is the width of the shielding (m).

The nuclide specific attenuation factors are summarized in Ref. [53].

For a complex geometry — for example, a volume gamma source with shielding — a sophisticated computer code such as MicroShield [53], MERCURE [54] or QAD [55] can be used. Any of these codes will also account for changes in the effective attenuation coefficient by computing a build-up factor as a function of the optical thickness. If any walls or dedicated shielding inside the reactor building are present that can be credited to reduce the dose at specific points, they may be also taken into account. This applies, in particular, to accident scenarios where the released fission products are expected to concentrate in sections of the building where they are better shielded, such as in the decay tank or the pump room.

In the evaluation of direct radiation fields, attention ideally will be given to possible relocation of radioactive material. The most important aspect is the possible relocation of particulate radionuclides and iodine isotopes to the off-gas or to recirculation filters in the case of releases to the containment. If for some reason these filters are located outside the reactor building and incorporate little shielding, they may cause a dose due to direct radiation in the vicinity of the reactor building that is significantly higher than that derived from a calculation assuming that these radionuclides are contained within the containment.

4.2.3. External exposure from airborne radioactive material

If there are releases of radioactive materials inside the reactor building, the external radiation due to immersion in a radioactive cloud of limited volume (e.g. standing in the reactor building) has to be considered. To simplify the calculations and use a conservative approach, the cloud volume may be taken as semi-infinite.

To perform a dose assessment inside the reactor building, it is necessary to have detailed knowledge of the source term, not only its isotopic composition but also its spatial distribution within the reactor containment. If such information is not already available, some conservative assumptions can be made, such as considering homogeneous sources distributed according to the containment layout. In that case, such assumptions will be properly justified, and sensitivity analyses must be carried out to define the escape routes. In addition, it must be recalled that the source term (and therefore the radiation source) is time dependent, a fact that must be taken into account during dose calculations.

The effective dose rate from immersion in a radioactive cloud (for each radionuclide) of limited volume (E_{im} (Sv/h)) is given by:

$$E_{im} = C_A DF_{im} \tag{12}$$

where

C_A is the specific activity of the radionuclide in air (Bq/m^3);
DF_{im} is the effective dose coefficient for immersion (Sv/h per Bq/m^3).

Taking into account the actual geometry, the dose rate and the effective dose coefficient can be calculated using codes such as MicroShield [53] or MARMER [56], or using the point kernel integration method. For conservative estimation, the values in Ref. [57] can be used.

4.2.4. External exposure from deposited radioactive material

The effective dose rate from the ground deposition (for each radionuclide) (E_{gr} (Sv/h)) is given by:

$$E_{gr} = C_{gr} DF_{gr} \tag{13}$$

where

C_{gr} is the surface specific activity of the radionuclide on the ground (Bq/m^2);
DF_{gr} is the effective dose coefficient for ground deposition (Sv/h per Bq/m^2).

Taking into account the actual geometry, the dose rate and the effective dose coefficient can be calculated using codes such as MicroShield [53] or MARMER [56], or using the numerical point kernel integration method

(MERCURE [54], QAD [55]). For conservative estimation, the values in Ref. [57] can be used.

4.2.5. Internal exposure from airborne radioactive material

Doses to operating personnel and experimenters can be reduced by mitigation measures. Breathing filters are an inexpensive and efficient way to reduce the inhalation of radionuclides and ideally will be used in any accident conditions involving a release of radioactive material. Simple aerosol filters, possibly in combination with iodine filters, are usually employed to significantly reduce inhalation doses. If credit is taken in the accident analysis for the use of such filters to obtain acceptable dose values for the operating staff, then maintenance and periodic testing are needed to ensure their appropriate performance and availability.

Internal doses arising from the inhalation of radionuclides are a function of the time integrated activity concentration at the location of interest:

$$H_{in} = CBD_{in} \tag{14}$$

where

H_{in} is the dose to the specified organ (Sv);
C is the integrated activity concentration (Bq·s·m^{-3});
B is the breathing rate (m^3/s);
D_{in} is the inhalation dose conversion factor (Sv/Bq).

Generic procedures (Ref. [58]) suggest the use of C_A, as introduced above, for external irradiation. The value of C is equal to $C_A T_e$, where T_e is the time of exposure and C_A is the specific activity of the nuclide in air (Bq/m^3). The inhalation dose conversion factors are summarized in Refs [57, 59, 60]. The breathing rate for an adult can be assumed to be 3.30×10^{-4} m^3/s [19].

Inhalation doses from resuspension of radionuclides deposited on surfaces may be important, in particular when the airborne nuclides are removed from the contaminated air. The relevant resuspension factors are usually much less than one, but doses obtained during a long term presence in the contaminated area are not negligible.

4.3. ON-SITE CONSEQUENCES OUTSIDE THE REACTOR BUILDING

In the event of radioactive releases to the reactor building interior, atmospheric releases to the exterior of the reactor building, apart from external direct radiation to the site, may occur. In particular, such releases may occur when the reactor building is not designed as a means of confinement. In this case, contamination in the vicinity of the building due to leakage through windows and doors may have to be considered in the consequence analysis.

Radioactive liquid spills may cause contamination in the vicinity of the building if no systems for draining liquid releases to liquid waste collection tanks are included in the design. It is important that consideration be given to the possibility of liquid radioactive material bypassing the designed barriers, or the possible relocation of radioactive material to areas with no retention or less retention than foreseen in the designed safety features. For screening purposes, the total critical group dose due to a particular accident scenario (or source term) is taken as the sum over the doses from all pathways and all radionuclides included in the assumed source term. One of these contributions is the dose due to external direct radiation fields.

It is clear that the relative importance of this contribution will depend on many factors; usually, it will be most important at points inside the reactor containment or outside but close to the reactor building, with decreasing importance as the evaluation point is moved farther from the facility.

It is advisable that, at this point, a detailed description of the source term be available. This will define the radiation source for the radiation transport calculations to be done.

On-site consequences may include release of radionuclides in gaseous, liquid or, in some cases, particulate form from the reactor building, and direct radiation from the containment. One of the on-site radiological hazards is intense direct gamma radiation from the reactor building/containment following a release to the building of even a relatively small fraction of core inventory. Dose determination will involve shielding calculations for a volume source and is both reactor and accident specific, depending on the nature of the release, the type of radionuclide released, the radionuclide distribution within the facility, and both the geometry and the materials of the building structure.

Liquid releases from the reactor building may have on-site consequences but usually do not pose a hazard comparable with that posed by gaseous releases. They need to be considered, however, if one of the following hazards cannot be excluded:

— Direct radiation from the released liquid;
— Evaporation of substantial amounts of radionuclides from the released liquid;
— Contamination of aquifers.

Usually, liquid releases occur to some type of reservoir where the liquid is collected before release. This reduces the risk of off-site contamination and leaves more time for enacting countermeasures. However, it should be considered that liquid releases of fractions of the core may produce significant radiation levels in areas not well protected or shielded. This may pose great problems concerning treatment of the radioactive liquid. Therefore, sufficient storage capacity with adequate shielding needs to be considered in the accident analysis.

4.3.1. On-site exposure resulting from containment release

Doses due to external irradiation from the containment on the site or in cases where the reactor building is only a short distance from the site boundary can be calculated using the shielding methodology discussed for consequences inside the building. In this case, the geometric modelling is not so important, and it is usually acceptable to use simplified geometric models.

For distances greater than 100 m, a simplified approach that treats the radioactivity inside the building as a point source is sufficiently accurate for the purpose of dose estimates. Furthermore, taking into account the accuracy of the evaluations of radioactive releases to the containment, the error caused by assuming a point source instead of a volume source may be acceptable at shorter distances as well. At points very close to the reactor building (i.e. for areas just outside the building, if this is a realistic exposure possibility to persons on the site), calculations ideally will take into account the actual volume source.

4.3.2. External exposure from airborne radioactive material

Because of the complexity of airflow near a building, there is no single model that is applicable to all situations. The models discussed in Refs [57, 59] are applicable in the case of a source on or just above the building roof, or within the recirculation wake.

The effective dose rate from immersion in a radioactive cloud (for each radionuclide) of limited volume (E_{im} (Sv/h)) is given by:

$$E_{im} = C_A DF_{im} \tag{15}$$

where

C_A is the specific activity of the radionuclide in air (Bq/m^3);
DF_{im} is the effective dose coefficient for immersion (Sv/h per Bq/m^3).

The effective dose coefficients for immersion (for gamma and beta radiation) are summarized in Ref. [57]. To calculate the effective dose, the dose rate E_{im} must be integrated for the time period of exposure.

4.3.3. External exposure from deposited radioactive material

The effective dose rate from the ground deposition (for each radionuclide) (E_{gr} (Sv/h)) is given by:

$$E_{\mathrm{gr}} = C_{\mathrm{gr}} DF_{\mathrm{gr}} \tag{16}$$

where

C_{gr} is the surface specific activity of the radionuclide on the ground (Bq/m^2);
DF_{gr} is the effective dose coefficient for ground deposition (Sv/h per Bq/m^2).

The effective dose coefficients for ground deposition are summarized in Ref. [57]. To calculate the effective dose, the dose rate E_{gr} must be integrated for the time period of exposure.

4.3.4. Internal exposure from airborne radioactive material

Internal doses arising from the inhalation of radionuclides are a function of the time integrated activity concentration of the air at the location of interest:

$$H_{in} = CBD_{in} \tag{17}$$

where

H_{in} is the dose in the specified organ (Sv);
C is the integrated activity concentration (Bq·s·m^{-3});
B is the breathing rate (m^3/s);
D_{in} is the inhalation dose conversion factor (Sv/Bq).

The breathing rate for an adult can be assumed to be 3.30×10^{-4} m^3/s [19]. The inhalation dose conversion factors are summarized in Refs [57, 59, 60]. Exposure to dose conversion factors are available for a wide variety of tissues and organs, such as bone marrow, bone surface, breast, gonad, lung, skin, spleen, stomach, thyroid and uterus.

4.3.5. External exposure due to liquid releases

External doses due to liquid releases from the reactor building usually do not pose a hazard comparable with that posed by gaseous releases. They need to be considered, however, if any one of the following hazards cannot be excluded:

— Direct radiation from the released fluid;
— Evaporation of substantial amounts of radionuclides from the released fluid;
— Contamination of aquifers (on the site or off the site).

Usually, liquid releases occur to some type of reservoir where the liquid is collected before release. This reduces the risk of off-site contamination and increases the length of time for enacting countermeasures. However, it must be considered that liquid releases of fractions of the core may produce significant radiation levels in areas that are not well protected or shielded. This may pose great problems during treatment of the radioactive liquid. Therefore, it is important that sufficient storage capacity with adequate shielding be considered in the accident analysis.

Liquid releases into the reactor building are usually collected at special dedicated and shielded reservoirs designed to contain a specified activity. For a given accident scenario, the shielding provided must be verified. If such reservoirs are not available at the facility, doses due to liquid releases into the reactor building must also be assessed. The activity concentration of the liquid release must be determined; a simplified analytical point kernel buildup factor method with the estimated equivalent surface source can be used to evaluate the dose rate.

When evaporation releases substantial amounts of radionuclides to the atmosphere, the radiation source can be determined by using the appropriate atmospheric dispersion models [59].

4.4. OFF-SITE CONSEQUENCES

An atmospheric release occurs when radioactive material from the primary coolant system is released into the reactor building as a gaseous discharge and eventually escapes to the outside. The source term for atmospheric release is assumed to be known from calculations carried out following the guidance and examples given in Section 3 of this publication. The radioactive material will mainly be held within the containment, with releases to on-site or off-site locations in proportion to the leak rate of the reactor building. In the case of small reactors located in urban areas, the site boundary may be the containment building itself.

Usually, off-site consequences of research reactor accidents have to be expected only if significant amounts of radioactive substances are released from the reactor building. In the case of reactors located at large sites, because of the distance and shielding by air, the direct radiation from the reactor building will be reduced to values that do not contribute significantly to the total dose, which mainly will be due to atmospheric releases. Exceptions are possible, however, at small sites (short distances to the fence), special sites such as university campuses with close access to the building, and facilities with low shielding (zero power facilities) or special beam tube experiments. Thus, the safety assessment must be performed on a reactor specific basis.

The most important off-site consequences are expected from atmospheric releases. Accidents involving such releases are likely to occur only in research reactors above a certain power and in certain accident scenarios. The radiological consequences will depend on factors such as the source term, the mode and point of release (stack or ground), the duration of the release and the meteorological conditions prevalent at the time of release.

An example of a calculation of the off-site consequences of a 20 MW research reactor BDBA and a U–Mo rig is presented in Appendix VI.

Off-site consequences due to liquid releases are similar to the on-site consequences of such releases. Hazards due to direct radiation can be minimized by dilution and runoff. However, contamination may be of greater significance. This can be minimized by providing sufficient on-site storage capacity in advance. The safety assessment should therefore identify possible liquid releases under accident conditions, and sufficient storage capacity for liquid wastes should be provided.

Seven different dose exposure pathways are conventionally considered:

— Direct exposure to the passing plume (also called cloud shine);
— Direct exposure to materials deposited on the ground (also called ground shine);

— Skin exposure and absorption contribution from materials deposited on the skin;
— Inhalation of materials directly from the passing plume;
— Inhalation of materials resuspended from the ground by natural and mechanical processes;
— Ingestion of contaminated foodstuffs;
— Ingestion of contaminated water.

Long term doses result from exposure pathways connected with ground shine, inhalation of resuspended materials, and ingestion of contaminated foodstuffs and water. Short term doses are dominated by exposure to the plume, materials directly deposited on the skin and inhalation of materials from the plume. For each exposure pathway, modelling of the radiological burden must take into account reductions due to actions taken to mitigate that pathway dose (e.g. evacuation, sheltering). In general, ingestion doses do not contribute to the doses calculated for the emergency phase of the accident.

4.4.1. External exposure from airborne radioactive material

The effective dose rate from immersion in the atmospheric discharge plume (for each radionuclide) of a semi-infinite cloud (E_{im}(Sv/h)) is given by:

$$E_{im} = C_A DF_{im} \tag{18}$$

where

C_A is the specific activity of the radionuclide in air (Bq/m^3);
DF_{im} is the effective dose coefficient for immersion (Sv/h per Bq/m^3).

The effective dose coefficients for immersion are summarized in Refs [57, 61]. To calculate the effective dose, the dose rate E_{im} must be integrated for the time period of exposure.

4.4.2. External exposure from materials deposited on skin

Some radionuclides give rise to beta radiation, which can lead to exposure of skin. The equivalent dose to skin can be calculated from the equation below:

$$E_{im,s} = C_A DF_S \tag{19}$$

where

$E_{im,s}$ is the skin dose rate from beta irradiation (Sv/h);
C_A is the specific activity of the radionuclide in air (Bq/m^3);
DF_S is the skin dose due to beta irradiation per unit activity in air (Sv/h per Bq/m^3).

To calculate the skin dose, the dose rate $E_{im,s}$ must be integrated for the time period of exposure.

The effective dose from skin exposure can be obtained by multiplying the skin dose by the weighting factor 0.01. The skin dose coefficients for immersion in a semi-infinite radioactive cloud are summarized in Ref. [59].

4.4.3. External exposure from deposited radioactive material

The effective dose rate from the ground deposition (for each radionuclide) (E_{gr}(Sv/h)) is given by:

$$E_{gr} = C_{gr} DF_{gr} \tag{20}$$

where

C_{gr} is the surface specific activity of the radionuclide on the ground (Bq/m^2);
DF_{gr} is the effective dose coefficient for ground deposition (Sv/h per Bq/m^2).

The effective dose coefficients for ground deposition are summarized in Ref. [57]. To calculate the effective dose, the dose rate E_{gr} must be integrated for the time period of exposure.

4.4.4. Internal exposure from airborne radioactive material

Internal doses arising from the inhalation of radionuclides are a function of the time integrated activity concentration at the location of interest:

$$H_{in} = CBD_{in} \tag{21}$$

where

H_{in} is the dose in the specified organ (Sv);
C is the integrated activity concentration (Bq·s·m^{-3});

B is the breathing rate (adult: 3.30×10^{-4} m^3/s; 10 year old child: 2.2×10^{-4} m^3/s; 1 year old child: 0.69×10^{-4} m^3/s) [19];
D_{in} is the inhalation dose conversion factor (Sv/Bq).

The inhalation dose conversion factors are summarized in Refs [57, 59, 60]. The effective dose from inhalation exposure may be obtained by multiplying the thyroid dose by the weighting factor 0.05.

4.4.5. Internal exposure from contaminated food

The transfer of radionuclides through the terrestrial environment to foodstuffs is a complex process. Recently, a number of 'dynamic' food chain models have been developed that enable the important time dependent processes to be evaluated, where appropriate [57, 59, 62, 63].

The ingestion doses for infants and adults are then calculated using the following general equation:

$$E_{in,p,i} = C_{p,i} H_{p,i} DF_{in,i} \quad (22)$$

where

$E_{in,p,i}$ is the annual effective dose from consumption of radionuclide i in foodstuff p (Sv/a);
$C_{p,i}$ is the concentration of radionuclide i in foodstuff p at the time of consumption (Bq/kg);
$H_{p,i}$ is the consumption rate for foodstuff p (kg/a) containing radionuclide i;
$DF_{in,i}$ is the dose coefficient for ingestion of radionuclide i (Sv/Bq).

Equation (22) can be used to estimate the doses from drinking water, in which case the $H_{p,i}$ would be the drinking water intake rate and $C_{p,i}$ would be the concentration of radionuclide i in the drinking water.

The default intake rates and dose coefficients can be taken from Refs [57, 59].

4.4.6. Computer codes for consequence analysis

Various computer codes [62, 63] have been developed to determine atmospheric dispersion, and individual and collective dose. The input parameters used in these codes, or in any other kind of calculation, need to be chosen carefully. Thus, the meteorological parameters used for the calculations must include at least the most common meteorological conditions at the facility

site and one adverse meteorological condition of rather low probability to cover the whole range of consequences of various release categories. It is common practice to use constant meteorological parameters, especially with regard to wind speed and wind direction, in these calculations, although it is expected that these would change within the release time, particularly with long term releases, thus reducing the maximum doses considerably.

The European Commission's MARIA (Methods for Assessing the Radiological Impact of Accidents) programme was initiated in 1982 to review and build on the nuclear accident consequence assessment (ACA) methods in use within the European Union. One of the objectives of the MARIA programme was the development of a computer program system for assessing the off-site consequences of accidental releases of radioactive material to the atmosphere. A new program system, COSYMA (Code System for MARIA) [63], was therefore developed jointly by different European countries. The first version was released in 1993, the second in 1996.

COSYMA is a probabilistic ACA system for use in calculating the risk posed by potential nuclear accidents involving a release to the atmosphere, taking into account the range of conditions that may prevail at the time of the accident. It can be used for deterministic or probabilistic assessments. Deterministic assessments give detailed results for a single set of atmospheric conditions. Probabilistic assessments give results taking into account the full range of atmospheric conditions and their respective frequencies of occurrence.

The end points of the system are:

— Air concentration and deposition at particular points;
— Numbers of people and areas affected by countermeasures;
— Types and amounts of food banned;
— Doses received in selected time periods;
— The number of near and long term fatal and non-fatal health effects;
— The economic costs of countermeasures and health effects.

The MACCS2 code [62] was developed under the sponsorship of the United States Nuclear Regulatory Commission to provide the nuclear safety community with a state of the art framework for comprehensively assessing radiological risks from nuclear facility source terms.

MACCS2 was developed from the well-known CRAC code [64] series and has been utilized extensively at various levels for nuclear reactor related consequence assessment studies. It has been used for assessing radiological consequences for a relatively high powered research reactor under hypothetical BDBA conditions [65].

MACCS2 consists of a sequence of mathematical and statistical models representing radionuclides immediately after release from the containment, movement of the material as it disperses downwind of the plant, deposition of the radioactive material, and the effects of the airborne and deposited material on humans and the environment. MACCS2 estimates the near term health effects, chronic health effects and economic consequences. Seven different exposure pathways are included in MACCS2, along with accounting for emergency response actions (both near and long term).

5. INTEGRATED ASSESSMENTS AND PRESENTATION OF RESULTS

Sections 3 and 4 provide information and background on various aspects of source term derivation, and considerations to be taken into account when evaluating radiological consequences. Examples and methodologies are presented that can help in conducting simple conservative assessments. This section provides an overview of an integrated approach to the derivation of source terms and related radiological consequences.

The assessment of accident source terms and radiological consequences for research reactors should start with the identification of PIEs — such as DBAs and BDBAs — to be considered in the SAR. In some States, the regulatory body may prescribe the set of events to be analysed. In the absence of clear regulatory prescriptions, PIEs and accident sequence scenarios can be identified by applying engineering judgement, using experience gained at similar reactors or conducting a PSA to identify credible events. Table 1 in Ref. [2] presents a recommended list of PIEs.

Once the PIEs and accidents to be considered in the SAR have been selected, plant damage states can be assessed. Initial and boundary conditions, analysis assumptions, correlation models and methods need to be judiciously selected to balance the required assessment effort and the desired accuracy of the predictions.

Depending on the specific application, various approaches may be utilized for conducting assessments of accident consequences and presenting them in an integrated manner. For many low power research reactors and accidents characterized by a small source term, it may be adequate to use bounding assumptions for key parameters (e.g. radionuclide release fractions, retention in water and containment, energy and height of release to the

atmosphere, wind speed, rainfall, atmospheric stability) in order to demonstrate that maximum possible doses to individuals on the site and off the site are well within acceptable limits for the site in question. If the accident consequences are shown to be below the relevant limits for various exposed groups (i.e. personnel, members of the public), the safety analysis is finalized.

For certain situations, however, this bounding approach may lead to the prediction of unacceptable consequences, and a more systematic and detailed approach to the assessment of accident consequences may be advisable or necessary. Options include the use of more sophisticated and accurate analysis methods, relaxation of restrictive analysis assumptions (with adequate justifications backed by experimental evidence), implementation of design changes to prevent occurrence of specific failures and accidents, and development of accident mitigation/management provisions. This loop is repeated until regulatory and other acceptance criteria are met. As an example, for an operating facility, modification of the operational limits may be needed; for a new facility, design modifications are to be considered.

Extensive research conducted over the past few decades has led to the development of sophisticated computer codes, which are readily available for conducting detailed integral assessments of research reactor source terms and radiological consequences. Despite the fact that most such codes were originally developed for use in safety assessments of power reactors, many of them have been validated, accepted by relevant regulatory bodies and successfully used in safety assessments of research reactors.

The ORIGEN 2 code [22] is widely used for the calculation of fission product inventories and associated decay heat for any operating history. FISPIN [23] is another code for calculating radionuclide inventories.

Source term evaluations for research reactor facilities can be conducted using system codes such as MELCOR, SCDAP or MAAP4 [9–11]. MELCOR is a fully integrated computer code for severe accident analyses that includes specific modules for (non-explosive) core melt progression, fission product release and transport within multivolume interconnected systems.

Codes such as SPARC, BUSCA, SOPHAEROS, CONTAIN, GOTHIC and many others allow detailed assessments to be conducted of radionuclide transport and retention in various reactor systems, such as the pool, the primary heat transport system and the containment structure [66–70]. Codes such as MicroShield, MARMER, MERCURE and QAD [53–56] are appropriate for evaluating direct dose rates in a wide variety of source/shielding configurations. The MACCS2 [62] and COSYMA codes [63] provide a framework for a comprehensive assessment of radiological risks from nuclear facility source terms.

Recently, the application of statistical methods permitting the derivation of realistic, best estimate assessments with the associated uncertainties has become possible. The principal advantage of such techniques is that they can provide a systematic, balanced perspective on various factors influencing risks. Results can be presented as best estimates (mean values) with associated uncertainties at different confidence levels. It is expected that such an approach will result in the prediction of much smaller radiological consequences. The desired degree of conservatism can then be obtained by applying high confidence limits.

Whatever the specific methods used for the derivation of source terms and radiological consequences, it is important that the results be included in the facility SAR in order to:

— Confirm that the reactor and its safety systems have been designed and integrated correctly, and allow the required accident mitigation and management capabilities;
— Identify credible accident sequences for emergency preparedness;
— Establish operational limits and conditions to ensure adequate safety margins;
— Assist reactor operators in understanding the consequences of various failures, events and accidents.

As the SAR is the most, and sometimes the only, comprehensive document providing a complete assessment of various safety issues, it is important that an adequate description and justification of the methods also be provided. Ideally this will include:

— Input parameters, in particular, initial and boundary conditions;
— A description of the correlations and models;
— Simplifying or bounding assumptions;
— Operating limits or conditions used as assumptions in the analysis.

The remainder of this report is structured as follows: Appendix I gives the historical background of different incidents having safety significance that have occurred at research reactors, classified according to the initiating event. Appendix II describes radioactive releases from, and related information concerning, research reactor fuels under simulated severe accident conditions. Appendix III presents coefficients for CORSOR-M correlations for predicting the release of fuel fission products.

Appendix IV presents a summary of the radiological consequences of the SILOE reactor DBA. Appendix V presents the source term evaluation and the

radiological consequences for the 10 MW ASTRA research reactor. Appendix VI presents two examples of BDBAs, one occurring under water, with partial blockage of cooling channels in a fuel assembly, and the other occurring in air, with an erroneous early removal of a U–Mo rig into the hot cells. Appendix VII provides a methodology for source term evaluation and a practical example of its application for a hypothetical accident scenario at the 14 MW TRIGA research reactor at INR Pitești.

Annexes I–III provide a brief summary of an integrated approach to the derivation of the source term and the assessment of the radiological consequences, including examples of typical recommendations for estimating source terms, typical fission product inventories for research reactors and an assessment of confinement responses to challenges from explosive FCIs.

Appendix I

PAST RESEARCH REACTOR ACCIDENTS INVOLVING SOURCE TERMS

Evaluation of the safety of a reactor or experimental facility includes analysis of its response to a given range of PIEs in order to demonstrate that the risk and the safety margin associated with its operation are acceptable.

As described in previous sections of this report, it is clear that the source term is highly dependent on, among other factors, the facility design and the confinement performance. This makes unavoidable the need to make specific assumptions based on experimental or specific measured data. It is advisable that these assumptions always be conservative, and that their implications be taken into account, not only in the source term derivation, but also in the other stages of the analysis.

To put into perspective the degree of conservatism to be used in the source term evaluation, this appendix gives the historical background of incidents having safety significance that have occurred at different research reactors. Although the available information is generally insufficient to define a source term as understood in the present publication, it permits an estimate to be made of the magnitude of the source terms associated with incidents that have actually occurred.

The information given in Tables 3–6, which was prepared from data included in Ref. [71], covers the time since the first experimental reactor began operation (1942) — i.e. about 11 000 reactor years of operational experience. This experience base provides compelling evidence of the risk profile associated with such facilities. The information is not entirely comprehensive, since only those accidents having some quantified activity release have been included.

For consistency with the structure of the present publication, the incidents have been classified according to the following four major groups of initiating events:

— Group 1: Insertion of excess reactivity;
— Group 2: Loss of flow;
— Group 3: Loss of coolant;
— Group 4: Human error and/or equipment or component failure.

Based exclusively on the operational experience presented, it is clear that uncontrolled reactivity changes and coolant channel blockages are the most serious initiating events to be considered from the viewpoint of the impact on fuel integrity, and therefore on the magnitude of the source term.

TABLE 3. PAST RESEARCH REACTOR ACCIDENTS, GROUP 1: INSERTION OF EXCESS REACTIVITY

Reactor	Date	Reactor type	Event description	Yield energy release[a]	Fuel damage	Activity released	Dose to personnel	Reactivity overcome by
Water boiler (USA)	Dec/49	Homogeneous uranyl nitrate/ graphite reflected	During reactor upgrade, manual withdrawal of control rods during drop time test; reactor control panel had been deactivated	3–4 × 10^{16} fissions,or 1–1.3 MJ	No	None	0.25 mSv	Negative temperature coefficient
JEMIMA (USA)	Apr/52	Cylindrical, unreflected, 93% enriched ^{235}U metal assembly	Power excursion due to computational error and violation of operational procedure	1.5 × 10^{16} fissions, or 0.5 MJ	No	None	No	Automatic scram system
NRX (Canada)	Dec/52	42 MW, HW moderated, LW cooled, natural U fuel reactor	Core destruction due to erroneous control rod withdrawal and H explosion	2000 MJ	Several FE ruptured (overheating plus steam pressure)	Approximately 10^4 Ci of long lived fission products carried to basement	None during accident; 40 mSv per worker during cleanup	Delayed manual dumping of moderator
EBR-1 (USA)	Nov/55	2.4 MW(th), 0.2 MW(e) fast breeder NaK cooled reactor	Power excursion and core meltdown due to operator error and poor experiment planning	16 MJ	Approximately 40% of core melted	N/A[b]	Low	Automatic shutdown
VINČA Institute (Yugoslavia)	Oct/58	Unreflected D_2O moderated natural U critical assembly, unshielded	Power excursion due to design deficiency and operational errors	2.6 × 10^{18} fissions, or 81 MJ	No	None	Ranging between 2.0 and 4.3 Sv	Operator action

TABLE 3. PAST RESEARCH REACTOR ACCIDENTS, GROUP 1: INSERTION OF EXCESS REACTIVITY (cont.)

Reactor	Date	Reactor type	Event description	Yield energy release[a]	Fuel damage	Activity released	Dose to personnel	Reactivity overcome by
ALIZE Saclay (France)	Mar/60	Critical assembly, UO_2 rods, 1.5% enrichment, water reflected and moderated	Power excursion due to operational error and poor safety system design	3×10^{18} fissions, or 97 MJ	No	None	Low	Doppler effect; failure of safety system
HTRE-3, NRTS, Idaho (USA)	Nov/58	Air cooled experimental facility for testing high temperature reactor cores	Power excursion due to instrumentation malfunction during planned test with zirconium hydride moderated, enriched U fuel	N/A	Partial meltdown of core	N/A; only high contamination reported	N/A	Manual scram
SL-1, NRTS, Idaho (USA)	Jan/61	Experimental natural circulation BWR reactor, 3 MW(th), 70 kW(e)	Power excursion due to manual withdrawal of the central control rod; a high pressure wave developed, moving the reactor vessel approximately 3 m above normal position	133 MJ	Partial meltdown of core	10% of fission product inventory escaped from pressure vessel; only 0.01% escaped from reactor building	Three worker fatalities; rescue team members: 25–75 mSv	Steam void and core vaporization (estimated)
VENUS (Belgium)	Dec/65	Critical facility, 7% enriched UO_2 rods in H_2O-D_2O, tank type	Power excursion due to operator error and operating procedures violation	4.3×10^{17} fissions, or 13 MJ	No	None	One worker: 5 Sv	Human action

TABLE 3. PAST RESEARCH REACTOR ACCIDENTS, GROUP 1: INSERTION OF EXCESS REACTIVITY (cont.)

Reactor	Date	Reactor type	Event description	Yield energy release[a]	Fuel damage	Activity released	Dose to personnel	Reactivity overcome by
RA-2 (Argentina)	Sept/83	Experimental facility, 0.1 W	Criticality excursion due to operator error	10 MJ	No	None	One operator: 43 Gy (fatality); 13 workers: 0.006–0.25 Gy	Manual action
KRR-2 (Republic of Korea)	Nov/94	TRIGA Mark-III 2 MW natural convection cooled reactor	Power transient due to power signal disconnection and poorly designed features		Cladding rupture of an instrumented FE	Small amount of gaseous fission products (gap activity)	None detected	Normal shutdown
TRIGA-IRNE Pitești (Romania)	Nov/89	14 MW TRIGA reactor	Power excursion due to operator error. During shuffling operations, two fuel assemblies were mistakenly lifted from the core. To prevent the danger of one of them falling, the operator placed them at two free positions in the reactor grid beyond the reach of control rods.	Maximum power reached (21.5 MW)	No	None	Not reported	Operator action

[a] 1 MJ $\approx 3.1 \times 10^{16}$ fissions.
[b] N/A: not applicable.

TABLE 4. PAST RESEARCH REACTOR ACCIDENTS, GROUP 2: LOSS OF FLOW

Reactor	Date	Reactor type	Event description	Fuel damage	Activity released	Dose to personnel
SRE (USA)	Jul/59	Graphite moderated and sodium cooled	Decomposition elements in water coated the FE, preventing proper cooling function. Some coolant channels were also blocked.	10 of 43 FEs severely damaged	Some activity released through stack	None
WTR (USA)	Apr/60	Light water cooled and moderated	Due to faulty design, the cladding separated from the fuel meat, blocking the coolant flow.	1 FE melted	Some activity released	None
ETR (USA)	Dec/61	MTR type reactor, operating at 90 MW	A plastic 'sight box' forgotten during maintenance sank down onto the core, blocking the water flow.	Portions of 6 FEs melted	High contamination levels; fission products and particulates detected	None
ORR (USA)	Jul/63	30 MW MTR type, light water cooled and moderated, downward cooling flow	Coolant flow was blocked by a foreign object.	1 FE plate melted	1000 Ci of fission products released into water	0.20 Gy/h inside, 0.02 Gy/h outside building.
SILOE (France)	Nov/67	30 MW MTR type pool reactor	An unidentified object blocked several flow channels.	6 fuel plates melted, low burnup (4%)	55 kCi released to reactor water, 2 kCi released through stack	Very limited

TABLE 4. PAST RESEARCH REACTOR ACCIDENTS, GROUP 2: LOSS OF FLOW (cont.)

Reactor	Date	Reactor type	Event description	Fuel damage	Activity released	Dose to personnel
Dhruva (India)	Nov/94	100 MW HWRR, metallic natural uranium fuel, upward coolant flow	Due to an open interconnecting valve, the coolant flow to the core dropped below the safety limit.	No	No	None
CIRUS IN-2 (India)	Jan/93	40 MW HW moderated, LW cooled, metallic natural uranium fuel	A closed outlet valve at the bottom of one of the fuel channels prevented coolant flow over the FE rod for 18 h.	No	No	None
CIRUS IN-2 (India)	Apr/94	40 MW HW moderated, LW cooled, metallic natural uranium fuel	A fuel rod remained in the transfer flask for 15 min without cooling.	No	No	None

TABLE 5. PAST RESEARCH REACTOR ACCIDENTS, GROUP 3: LOSS OF COOLANT

Reactor	Date	Reactor type	Event description	Fuel damage	Activity released	Dose to personnel
NRU, Chalk River (Canada)	May/58	135 MW research reactor, heavy water cooled and moderated, natural uranium	Fuel element rupture and loss of coolant flow in fuel road removal flask led to uranium burning.	One FE burst, other two failed	N/A[a]; building contaminated	Max.: 0.19 Sv
University of Michigan (USA)	Aug/63		A horizontal beam thimble was ruptured due to the insertion of a long shielding plug.	No	No	None
University of Virginia (USA)	Oct/63		A pipe break occurred in the demineralized room.	No	No	None
TAMCR Texas (USA)	Nov/63		A gasket in the demineralizer tank access hole failed and water flowed to the hot sump.	No	Reported pool water activity: 43 μCi/mL	None
SILOE (France)	Dec/86	30 MW MTR type pool reactor	Slight degradation of leaktightness of the bottom of the pool caused water losses.	No	Contamination of underground water (max. measured activity = 170 kBq/L of tritium)	None
University of North Carolina (USA)	Feb/88	Pool reactor	Leak of 378 L/d (estimated) from the piping of the primary cooling system.	No	No	None

TABLE 5. PAST RESEARCH REACTOR ACCIDENTS, GROUP 3: LOSS OF COOLANT (cont.)

Reactor	Date	Reactor type	Event description	Fuel damage	Activity released	Dose to personnel
UI-TRIGA (USA)	Nov/94	TRIGA Mark-II reactor	Due to erroneous valve positioning, the delay tank was subjected to excessive compressive stress, which caused a 10 cm fracture in its circumferential weld.	No	No	None

[a] N/A: not applicable.

TABLE 6. PAST RESEARCH REACTOR ACCIDENTS, GROUP 4: HUMAN ERROR AND/OR EQUIPMENT OR COMPONENT FAILURE

Reactor	Date	Reactor type	Event description	Fuel damage	Activity released	Dose to personnel
Idaho MTR (USA)	Jul/56	MTR type reactor	A highly radioactive reactor component was placed in position without appropriate shielding due to a lowered water level in the reactor tank.	No	No	8 workers: 25–215 mSv
HFR (The Netherlands)	May/68	30 MW high flux reactor	After startup for a fuel irradiation experiment (85 g of UO_2, 4.1 g of ^{235}U), reactor was inadvertently left in a high flux zone with its instrumentation and cooling systems in a non-operative state.	No FEs of the core; fuel at irradiation capsule melted	2.8 TBq of noble gases (estimated)	2 persons: maximum of 3 mSv gamma and 8 mSv beta
BR2 (Belgium)	Feb/71		TeO_2 capsule burned in a thimble tube.	No	10 Ci of ^{131}I liberated to thimble tube; estimated 0.5 mCi of ^{131}I released to environment	2 operators: accumulated doses in thyroid of 0.8 mSv and 0.5 mSv, respectively
BR2 (Belgium)	Jul/72		During the transfer of the MFBS-6 experiment (Na loop for irradiation of fossil rods).	No	Small	None
BR2 (Belgium)	May/73		Fuel element cladding failure due to defective fabrication process.	Yes	Small	None

TABLE 6. PAST RESEARCH REACTOR ACCIDENTS, GROUP 4: HUMAN ERROR AND/OR EQUIPMENT OR COMPONENT FAILURE (cont.)

Reactor	Date	Reactor type	Event description	Fuel damage	Activity released	Dose to personnel
KRR2 (Republic of Korea)	Nov/94	TRIGA Mark-III	Due to a leak at the heat exchanger, the reactor coolant was gradually degraded, which caused corrosion of the stainless steel cladding of the fuel elements.	Leakage in 33 of 114 FEs	FP detected in coolant water and reactor hall air	None
EWA (Poland)	Feb/91	WWRS-M 10 MW	Due to excessive initial loading, a container with Xe was overpressurized during reactor operation and lost its leak-tightness, releasing radioactive Xe and I.	No	5 MBq of ^{125}I released to the environment	Less than 0.02 mSv to operator; estimated 0.014 μSv to public
CIRUS IN-2 (India)	Dec/91	Heavy water research reactor 40 MW	Corroded bottom of the plastered brick wall of the inspection chamber caused loss of activated water.	No	Soil contamination with 1.7×10^{12} Bq of ^{137}Cs (estimated)	Dose rate at site: 0.5–2.0 Sv/h

Appendix II

RADIOACTIVE RELEASES FROM RESEARCH REACTOR FUELS UNDER SIMULATED SEVERE ACCIDENT CONDITIONS

The accident sequence groups that have the greatest potential for fuel damage are those involving excess reactivity insertions, loss of coolant and loss of flow. Some experimental data for reactivity/power excursions are available from the BORAX, SPERT and SNAPTRAN destructive reactor tests [72–74]. The results show that large step insertions of reactivity (greater than 2% $\Delta k/k$) and short time periods (less than 5 ms) were required before significant core damage, core disruption and violent pressure pulsing occurred (all reactors had negative reactivity coefficients).

All but the SNAPTRAN-2 tests were carried out in a water medium and showed that, when fuel immersed in water was damaged, essentially all the radioiodine and most (≥90%) of the noble gases were retained in the water (however, a small fraction of the water, in vapour form, drifted downwind). Damage to the U–Al plate fuel for the BORAX and SPERT tests was due to melting; most of the BORAX plates melted, and about 8% of the SPERT plates melted completely, along with partial melting of 35% of the core (under water). The SNAPTRAN fuel was UZrH with Hastelloy-N cladding; the fuel damage mechanism was fragmentation resulting from hydrogen liberation. In all three test programmes, the fuel was essentially recovered (within about 100 m of the reactor for the BORAX test, which produced the most energy). The energy releases in the terminal BORAX test and the accompanying pressure pulse caused by contact between the molten fuel and the pool water broke the reactor tank. However, calculations carried out for American Machine and Foundry (AMF) reactor types indicate that the maximum pressure rise for the BORAX terminal excursion in a containment column of 104 m^3 would be about 3 kPa.

A summary of some of the data obtained from these tests, including fission product release fractions, is given in Table 7 [14]. It should be noted that the time frame of the excursions was very short, and that the relative abundance of radionuclides associated with long term operation at full power would be very different. The fission product nuclides were produced during the transient phase, and those that dominate the activity would tend to have shorter half-lives. The iodine in particular would not have reached the higher concentrations characteristic of equilibrium production. Higher iodine concentrations and longer accident sequence duration could influence the quality of experimental data, the iodine chemistry and the water partition coefficients.

TABLE 7. RELEVANT DATA FROM DESTRUCTIVE REACTOR TESTS

Test	Date	Reactivity ($)	Period (ms)	Peak power (MW)	Cumulative power (MW-s)	Medium	Iodine release fraction (%)[a]	Noble gases release fraction (%)[a]	Overall release fraction (%)[a]
BORAX-I	22 July 1954	4	2.6	18 000	135	Water	N/A[b]	N/A	N/A
SPERT-1	06 November 1962	3.5	3.2	2 300	31	Water	0	7	0.4
SNAPTRAN 2/10A-3	01 April 1964	3.8	0.64	18 000	45	Water	0	3–4	N/A
SNAPTRAN-2	11 January 1966	5.1	0.2	74 000	54	Air	70	75	21

[a] Estimates of release fraction out of reactor building.
[b] N/A: not applicable.

Some of these test programmes also included ramp insertions, at various rates and from various initial power levels, which did not result in mechanical damage. In addition, various tests were carried out with large step reactivity insertions without fuel damage. The negative temperature coefficient of reactivity, and particularly the large, negative void coefficient, quenched the excursion abruptly and reduced the power to relatively stable and much lower levels. In most cases, however, the tests were terminated with a programmed shutdown shortly after the initial pulse. Effects such as flow instabilities and power oscillations were not systematically investigated. Some information on flow instability for overpower conditions is available from tests on the plate type CABRI reactor [46]. These tests showed that the flow instability regime is preceded by visible power oscillations due to local boiling and the void effect. Some fuel melting occurred during the flow instability tests; however, as the tests are reactor specific, they may not be applicable to all reactor types.

Appendix III

COEFFICIENTS FOR CORSOR-M CORRELATIONS FOR PREDICTING THE RELEASE OF FUEL FISSION PRODUCTS

Tables 8–10, from Ref. [42], present coefficients for three forms of the CORSOR-M correlation. Table 8 provides a summary of coefficients for the exponential form; Table 9 gives coefficients for the Arrhenius form; and Table 10 presents a summary of coefficients for the polynomial form. As no single form provides universal accuracy for all fuel types and for all species, the correlation form giving the greatest accuracy is provided in Table 11.

TABLE 8. SUMMARY OF EXPONENTIAL FORM CORRELATION COEFFICIENTS

Institution (researchers)	Fuel	Burnup, %	Species	Ambient	Range variation: Number	Range variation: Temperature range, K	A	B	Coefficient of correlation
ORNL									
(Parker et al.)[3]	UAl_4 alloy	24.00	Cs	Steam	1	>873	7.63×10^{-4}	7.65×10^{-3}	0.97
				Air	1	>873	6.05×10^{-2}	3.89×10^{-3}	0.99
			I	Steam–air		>1050	1.17×10^{-6}	1.70×10^{-2}	0.87
					2	>1050	3.73×10^{1}	7.30×10^{-4}	0.83
			Te	Steam	1	>873	2.94×10^{-6}	1.24×10^{-2}	1.00
				Air	1	>873	1.16×10^{-7}	1.43×10^{-2}	0.99
HEDL									
(Woodley et al.)[4]	UAl_4 alloy	52.00	Cs	Steam	1	>873	1.27×10^{0}	3.20×10^{-3}	0.63
				Air	1	>873	3.56×10^{-1}	4.08×10^{-3}	0.72
			Te	Steam	1	>873	4.86×10^{-10}	2.30×10^{-2}	1.00
				Air	1	>873	2.59×10^{-6}	1.40×10^{-2}	0.96
			I	Steam	1	>873	5.18×10^{-1}	4.24×10^{-3}	0.83
					2	<1100	3.14×10^{-4}	1.12×10^{-2}	0.91
						>1100	3.83×10^{1}	7.21×10^{-4}	0.80
				Air	1	>873	2.45×10^{-1}	4.79×10^{-3}	0.50
					2	<950	1.00×10^{-13}	3.52×10^{-2}	1.00
						>950	4.33×10^{1}	5.94×10^{-4}	0.49
	U_3O_8–Al		Cs	Steam	1	>873	4.81×10^{-1}	3.45×10^{-3}	0.19
				Air	1	>873	1.21×10^{-4}	1.01×10^{-2}	0.60
			Te	Steam	1	>873	5.12×10^{-5}	1.06×10^{-2}	1.00
				Air	1	>873	1.69×10^{-4}	1.02×10^{-2}	0.72
			I	Steam	1	>873	1.26×10^{-2}	7.92×10^{-3}	0.32
					2	<950	7.71×10^{-17}	4.31×10^{-2}	0.65
						>950	4.27×10^{1}	5.64×10^{-4}	0.12
				Air	1	>873	1.08×10^{0}	3.34×10^{-3}	0.65
					2	<1050	3.94×10^{-10}	2.42×10^{-2}	0.97
						>1050	1.97×10^{0}	2.88×10^{-3}	0.75
JAERI									
(Saito et al.)[5]	U_3Si_2–Al	23.00	Cs	Air	1	>873	2.28×10^{-2}	5.94×10^{-3}	0.92
			I	Air	2	<1100	7.09×10^{-6}	1.47×10^{-2}	1.00
						>1100	1.97×10^{1}	1.09×10^{-3}	0.98
			Te	Air	1	>873	4.04×10^{-8}	1.51×10^{-2}	1.00
			Noble gases	Air	1	>850	4.27×10^{-3}	8.17×10^{-3}	0.46
					2	850 to 873	1.69×10^{-42}	1.14×10^{-1}	1.00
						>873	3.30×10^{1}	8.53×10^{-4}	0.81
	UAl–dispersed	23.00	Cs	Air	1	>873	1.02×10^{-6}	1.33×10^{-2}	0.65
					2	<1175	2.08×10^{-20}	4.17×10^{-2}	1.00
						>1175	7.46×10^{0}	1.24×10^{-3}	0.99
			I	Air	1	>873	1.31×10^{0}	2.79×10^{-3}	0.98
			Te	Air	1	>873	7.70×10^{-2}	2.88×10^{-3}	1.00
	U_xSi_y–Al	23.00	Cs	Air	1	>873	1.47×10^{-3}	6.93×10^{-3}	1.00
			I	Air	1	>873	3.39×10^{-2}	5.83×10^{-3}	0.96
					2	<1150	3.92×10^{-3}	8.06×10^{-3}	0.96
					2	>1150	2.86×10^{-1}	4.15×10^{-3}	0.97

TABLE 8. SUMMARY OF EXPONENTIAL FORM CORRELATION COEFFICIENTS (cont.)

Institution (researchers)	Fuel	Burnup, %	Species	Ambient	Range variation: Number	Range variation: Temperature range, K	A	B	Coefficient of correlation
			Te	Air	1	>853	4.78×10^{-12}	2.08×10^{-2}	1.00
			Noble gases	Air	1	>850	7.44×10^{-3}	7.74×10^{-3}	0.39
					2	850 to 873	3.76×10^{-46}	1.24×10^{-1}	1.00
						>873	3.79×10^{1}	7.46×10^{-4}	1.00
JAERI (Saito et al.)[5]	U_3Si_2–Al	65.00	Cs	Air	1	>873	1.42×10^{-2}	6.79×10^{-3}	0.89
					2	<1073	6.96×10^{-5}	1.23×10^{-2}	1.00
						>1073	3.03×10^{0}	2.55×10^{-3}	1.00
			I	Air	2	850 to 1050	4.64×10^{-3}	9.27×10^{-3}	0.92
						>1050	4.48×10^{1}	5.91×10^{-4}	0.94
					1	>850	1.24×10^{0}	3.46×10^{-3}	0.67
			Te	Air	1	>1050	4.81×10^{-11}	2.10×10^{-2}	0.87
			Ru	Air	1	>1100	6.61×10^{-7}	1.30×10^{-2}	0.88
					2	<1173	9.03×10^{-14}	2.72×10^{-2}	1.00
						>1173	1.72×10^{-3}	6.98×10^{-3}	1.00
			Noble gases	Air	1	>850	1.28×10^{-2}	7.33×10^{-3}	0.34
					2	850 to 873	4.75×10^{-49}	1.32×10^{-1}	1.00
						>873	7.23×10^{1}	2.51×10^{-4}	0.75
	UAl–dispersed	65.00	Cs	Air	1	>873	4.65×10^{-2}	5.92×10^{-3}	0.89
			I	Air	1	>873	1.10×10^{-5}	1.45×10^{-2}	0.96
					2	873 to 975	8.38×10^{-8}	1.99×10^{-2}	1.00
						>975	3.31×10^{9}	8.13×10^{-4}	0.96
			Te	Air	1	>873	1.88×10^{-18}	3.27×10^{-2}	1.00
			Ru	Air	1	>873	1.06×10^{-7}	1.23×10^{-2}	0.92
	U_xSi_y–Al	65.00	Cs	Air	1	>873	3.20×10^{-4}	9.54×10^{-3}	0.93
					2	873 to 1073	1.16×10^{-6}	1.53×10^{-2}	0.99
						>1073	2.45×10^{-1}	4.31×10^{-3}	1.00
			I	Air	1	>850	2.69×10^{-1}	4.67×10^{-3}	0.70
					2	850 to 1050	1.94×10^{-4}	1.22×10^{-2}	0.96
						>1150	3.88×10^{-1}	7.02×10^{-4}	0.90
			Te	Air	1	>873	1.80×10^{-8}	1.62×10^{-2}	0.93
					2	873 to 1173	7.40×10^{-6}	1.00×10^{-2}	0.96
						>1173	4.83×10^{-4}	8.92×10^{-3}	1.00
			Ru	Air	1	>873	3.64×10^{-8}	1.50×10^{-2}	0.96
						873 to 1100	1.07×10^{-5}	9.16×10^{-3}	1.00
						>1100	3.19×10^{-5}	9.80×10^{-3}	1.00
			Noble gases	Air	1	>850	1.06×10^{-2}	7.46×10^{-3}	0.36
					2	850 to 873	5.01×10^{-48}	1.30×10^{1}	1.00
						>873	5.58×10^{1}	4.45×10^{-4}	0.85
ORNL (Shibata et al.)[2]	UAl_x–dispersed	62.00	Noble gases	Helium	1	>850	9.89×10^{-5}	1.52×10^{-2}	0.62

TABLE 9. SUMMARY OF ARRHENIUS FORM CORRELATION COEFFICIENTS

Institution (researchers)	Fuel	Burnup, %	Species	Ambient	Range number	Variation range	ln(A)	Q, kcal/mol	Coefficient of correlation
ORNL									
(Parker et al.)[3]	UAl_4 alloy	24.00	Cs	Steam	1	>873	12.17	24.26	0.95
				Air	1	>873	6.71	11.45	1.00
			I	Steam–air	2	933 to 990	69.25	128.55	1.00
						>990	6.94	5.98	0.75
			Te	Steam	1	>873	18.79	39.76	0.99
				Air	1	>873	20.94	47.22	1.00
HEDL									
(Woodley et al.)[4]	UAl_4 alloy	52.00	Cs	Steam	1	>873	7.48	8.00	0.68
				Air	1	>873	8.42	10.75	0.75
			Te	Steam	1	>873	25.90	48.10	1.00
				Air	1	>873	17.37	32.06	0.98
			I	Steam	1	>873	8.92	10.66	0.89
					2	873 to 970	45.43	80.73	1.00
						>970	5.41	2.13	0.83
				Air	1	>873	9.64	12.39	0.59
					2	873 to 950	35.81	60.77	1.00
						>950	5.20	1.69	0.59
	U_3O_8–Al		Cs	Steam	1	>873	7.15	8.91	0.19
				Air	1	>873	8.26	11.44	0.48
			Te	Steam	1	>873	13.09	24.32	1.00
				Air	1	>873	14.56	26.05	0.80
			I	Steam	1	>873	13.24	19.09	0.40
					2	873 to 975	42.68	73.33	0.65
						>975	5.10	1.57	0.17
				Air	1	>873	7.92	9.06	0.64
					2	873 to 1000	25.17	44.74	0.97
						>1000	7.92	9.06	0.64
JAERI									
(Saito et al.)[5]	U_3Si_2–Al	23.00	Cs	Air	1	>873	9.55	14.55	0.97
					2	873 to 1100	11.10	17.51	0.97
						>1100	6.63	7.21	0.99
			I	Air	1	>873	11.25	17.14	0.90
					2	873 to 1100	16.62	27.30	1.00
						>1100	5.76	3.51	0.99
			Te	Air	1	>873	19.73	44.10	0.99
			Noble gases	Air	2	850 to 873	97.29	163.10	1.00
						873 to 1373	6.73	5.27	0.79
	UAl-dispersed		Cs	Air	1	>873	19.47	40.91	0.71
					2	873 to 1175	48.09	104.08	1.00
						>1175	5.16	3.96	1.00
			I	Air	1	>873	10.03	15.33	0.80
					2	873 to 950	25.61	43.78	1.00
						>950	6.78	7.42	0.98

TABLE 9. SUMMARY OF ARRHENIUS FORM CORRELATION COEFFICIENTS (cont.)

Institution (researchers)	Fuel	Burnup, %	Species	Ambient	Range number	Variation range	ln(A)	Q, kcal/mol	Coefficient of correlation
			Te	Air	1	>873	5.05	10.01	1.00
	U_xSi_y-Al		Cs	Air	1	>873	20.08	45.81	0.98
			I	Air	1	>873	9.61	14.10	0.99
			Te	Air	1	>873	28.96	72.32	1.00
			Noble gases	Air	2	850 to 873 873 to 1373	106.20 5.89	177.71 3.23	1.00 0.88
JAERI (Saito et al.)[3]	U_3Si_2-Al	65.00	Cs	Air	1	>873	11.07	16.81	0.96
			I	Air	1 2	>850 850 to 973 >973	8.19 16.05 5.21	8.91 23.43 1.75	0.78 1.00 0.97
			Te	Air	1	>1100	30.06	68.34	0.90
			Ru	Air	1 2	>1075 1075 to 1173 >1173	17.78 30.97 11.34	38.91 68.03 22.27	0.92 1.00 1.00
			Noble gases	Air	2	850 to 873 873 to 1373	113.28 5.13	189.30 1.31	1.00 0.83
	UAl-dispersed	65.00	Cs	Air	1	>873	10.30	14.67	0.96
			I	Air	1 2	>873 873 to 1073 >1073	11.06 16.96 5.57	16.21 27.36 2.62	0.87 0.98 0.97
			Te	Air	1	>873	30.05	72.08	0.91
			Ru	Air	1	>873	10.78	28.43	0.86
	U_xSi_y-Al	65.00	Cs	Air	1	>873	13.39	23.41	0.98
			I	Air	1 2	>850 873 to 1073 >1073	9.43 15.21 5.09	11.99 22.89 1.33	0.81 0.98 0.97
			Te	Air	1 2	>873 873 to 1173 >1173	17.44 8.38 15.96	37.30 19.98 31.01	0.87 0.92 1.00
			Ru	Air	1 2	>873 873 to 1073 >1073	17.81 7.31 14.55	39.93 19.04 31.36	0.97 1.00 0.99
			Noble gases	Air	2	850 to 873 873 to 1373	110.79 5.39	185.21 1.97	1.00 0.88
ORNL (Shibata et al.)[2]	UAl_x-dispersed		Noble gases	Helium	1	>850	17.85	23.97	0.65

TABLE 10. SUMMARY OF POLYNOMIAL FORM CORRELATIONS

Institution (researchers)	Release time, s	Burnup, %	Fuel	Species	Ambient	Range variation		Release formula = f[Temperature (K)] %
						Number	Range	
ORNL (Parker et al.)[3]	120.00	24.00	UAl_4 alloy	I	Steam–air	2	875 to 1125 >1125	0.34 – 297.5 0.075T + 0.62
HEDL (Woodley et al.)[4]	120.00	53.00	UAl_4 alloy	I	Steam	2	875 to 1125 >1125	0.34T – 297.5 0.075T + 0.62
					Air	2	875 to 975 >975	0.8T – 700 0.05T + 31.25
			U_3O_8–Al	I	Steam	2	875 to 1125 >1125	0.34T – 297.5 0.075T + 10
					Air	1	>875	0.222T – 194.4
JAERI (Saito et al.)[5]	3600.00	23.00	U_3Si_2–Al	Cs	Air	1	>883	–96.29 + 0.109T
				Noble gases	Air	1	825 to 1373	$-2138.8 + 5.31T - 4.21 \times 10^{-3}T^2 + 1.115 \times 10^{-6}T^3$
			UAl–dispersed	Cs	Air	2	1075 to 1175 >1175	0.3T – 322 0.0495T – 26
				I		1	>873	–90.05 + 0.1085T
			U_xSi_y–Al	Cs		1	>1175	0.105T – 123
				I		2	825 to 1100 >1100	0.0909T – 75 0.214T – 211
				Te	Air	1	>1250	0.107T – 133
				Noble gases	Air	2	823 to 873 873 to 1373	T – 823 $-839.2 + 2.01T - 1.438 \times 10^{-3}T^2 + 3.426 \times 10^{-7}T^3$
		65.00	U_3Si_2–Al	Cs	Air	2	840 to 973 >973	0.09T – 75.57 0.22T – 202.06
				I	Air	2	850 to 1050 >1050	–291.22 + 0.35T 26.46 + 0.054T
				Ru	Air	1	>1086	–86.412 + 0.0796T
				Noble gases	Air	2	823 to 875 875 to 1373	–1234.5 + 1.5T $-666.25 + 1.7905T - 1.393 \times 10^{-3}T^2 + 3.611 \times 10^{-7}T^3$
			UAl-dispersed	Cs	Air	2	827 to 1000 >1000	0.11T – 91 0.264T – 243.3
				I	Air	2	857 to 1100 >1100	–162.87 + 0.19T –2.1417 + 0.075T
					Air	1	>873	–226.31 + 0.26T
				Ru	Air	1	1153 to 1373	0.01T – 11.53
			U_xSi_y–Al	Cs	Air	3	870 to 973 973 to 1073 >1073	0.33T – 287.4 –103 + 0.11T –253 + 0.25T
				I	Air	2	850 to 1100 >1100	–311.6 + 0.365T 14.701 + 0.063T
				Te	Air	2	1030 to 1178 >1178	0.0077T – 7.932 –671.4 + 0.57T
				Noble gases	Air	2	823 to 873 873 to 1373	1.3T – 1069.9 $-830.05 + 2.165T - 1.694 \times 10^{-3}T^2 + 4.44 \times 10^{-7}T^3$
ORNL (Shibata et al.)[2]	1800.00	62.00	UAl_4-dispersed	Noble gases	Helium	1	823 to 923	$R(120,T) = -10216 + 22.57T - 1.23 \times 10^{-2}T^2$

TABLE 11. SUMMARY OF SUGGESTED CORRELATION FORMS

Institution (researchers)	Burnup (%)	Fuel	Species	Ambient	No. of temp. ranges	Suggested formulation
ORNL (Parker et al.)	24	U–Al_4 alloy	Cs	Steam	1	Exponential (see Table 8)
				Air	1	Exponential (see Table 8)
			I	Steam	2	Polynomial (see Table 10)
				Air	2	Polynomial (see Table 10)
			Te	Steam	1	Exponential (see Table 8)
				Air	1	Exponential (see Table 8)
HEDL (Woodley et al.)	52	U–Al_4 alloy	Cs	Steam	1	Arrhenius (see Table 9)
				Air	1	Arrhenius (see Table 9)
			Te	Steam	1	Arrhenius (see Table 9)
				Air	1	Arrhenius (see Table 9)
			I	Steam	2	Polynomial (see Table 10)
				Air	2	Polynomial (see Table 10)
	52	U_3O_8–Al	Cs	Steam	1	Arrhenius (see Table 9)
				Air	1	Arrhenius (see Table 9)
			Te	Steam	1	Arrhenius (see Table 9)
				Air	1	Arrhenius (see Table 9)
			I	Steam	2	Arrhenius (see Table 9)
				Air	2	Polynomial (see Table 10)
				Air	2	Polynomial (see Table 10)
JAERI (Saito et al.)	23	U_3Si_2–Al	Cs	Air	1	Polynomial (see Table 10)
			I	Air	2	Polynomial (see Table 10)
			Te	Air	1	Polynomial (see Table 10)
			Noble gases	Air	1	Polynomial (see Table 10)
	23	U-Al–Disp.	Cs	Air	2	Arrhenius (see Table 9)
			I	Air	1	Polynomial (see Table 10)
			Te	Air	1	Arrhenius (see Table 9)
			Noble gases	Air	1	Polynomial (see Table 10)
	23	U_xSi_y–Al	Cs	Air	1	Polynomial (see Table 10)
			I	Air	2	Polynomial (see Table 10)
			Te	Air	1	Arrhenius (see Table 9)
			Noble gases	Air	2	Polynomial (see Table 10)
	65	U_3Si_2–Al	Cs	Air	1	Polynomial (see Table 10)
			I	Air	2	Polynomial (see Table 10)
			Te	Air	1	Arrhenius (see Table 9)
			Ru	Air	2	Exponential (see Table 8)
			Noble gases	Air	2	Polynomial (see Table 10)

TABLE 11. SUMMARY OF SUGGESTED CORRELATION FORMS (cont.)

Institution (researchers)	Burnup (%)	Fuel	Species	Ambient	No. of temp. ranges	Suggested formulation
	65	U–Al–Disp.	Cs	Air	1	Polynomial (see Table 10)
			I	Air	2	Arrhenius (see Table 9)
			Te	Air	1	Arrhenius (see Table 9)
			Ru	Air	2	Polynomial (see Table 10)
			Noble gases	Air	1	Polynomial (see Table 10)
	65	U_xSi_y–Al	Cs	Air	1	Polynomial (see Table 10)
			I	Air	2	Arrhenius (see Table 9)
			Te	Air	2	Arrhenius (see Table 9)
			Ru	Air	2	Arrhenius (see Table 9)
			Noble gases	Air	2	Polynomial (see Table 10)
ORNL (Shibata et al.)	62	UAl_x–Disp.	Noble gases	Air	1	Polynomial (see Table 10)

Appendix IV

CALCULATION OF THE RADIOLOGICAL CONSEQUENCES OF THE SILOE REACTOR DESIGN BASIS ACCIDENT

IV.1. THE SILOE REACTOR

SILOE is a pool type research reactor operating at a rated power of 35 MW. The standard configuration of the core comprises 33 MTR type fuel elements (640 flat U–Al plates) with aluminium cladding. The uranium is enriched to 94%.

The containment building, which has a volume of 14 000 m^3, is equipped with a normal ventilation system having particulate filters, and an emergency ventilation system having particulate filters and iodine traps (two independent circuits). In the case of a radioactive release exceeding fixed limits inside the containment, the normal ventilation is stopped and isolated automatically. The emergency ventilation with reduced airflow is then started automatically.

The SILOE reactor operates in continuous cycles of 21 d each. Successive cycles are separated by a shutdown period of 7 d for fuel loading and maintenance work.

IV.2. DESIGN BASIS ACCIDENT

The DBA taken into account for SILOE, and for other research reactors using U–Al fuel in operation in France, consists of an explosive reactivity accident called a 'BORAX type accident'.

The assumed accident scenario:

— Leads to the total melting of the fuel, with a total thermal energy release of 135 MJ (9% of this energy is converted into mechanical energy);
— Creates an overpressure inside the containment (calculated value of 58 mbar) due to the transfer of thermal energy to the air.

The existing technical provisions prevent water leakage from the pool or through the neutron beam tubes. The containment building withstands the internal pressure during and after the accident. The safety analysis performed by the operator and reviewed by the regulatory body confirms the absence of internal missiles affecting the integrity of the containment during the accident.

It also confirms that the water column expelled outside the pool does not reach the upper part of the containment building.

IV.2.1. Source term derivation

The source term was calculated using the conservative assumption that the accident occurs at the end of the fifth cycle of operation (see the inventory of fission products presented in Table II–2 in Annex II).

The release and transfer factors of fission products from the molten fuel to the pool water and then to the containment were defined in a prescriptive manner. The adopted values are listed in Table 12.

The efficiency of the filtration system is checked each year. Air heaters are installed in front of iodine traps to avoid a possible loss of efficiency due to the high level of humidity inside the containment during and after the accident. These provisions are intended to guarantee the value used for the source term calculations (10^{-3}).

IV.2.2. Ventilation and filtration

Three different releases to the environment were considered:

— Instantaneous release of 2.5% of the air (non-filtered release due to leakage from the containment);
— Release through the stacks of 2.9% of the air due to an initial overpressure inside the containment;
— Continuous release by the emergency ventilation (max.: 400 m^3/h)

The height of the stack is 35 m.

TABLE 12. RELEASE AND TRANSFER FACTORS OF FISSION PRODUCTS

Transfer	I, Te, Br and Cs	Noble gases	Other fission products
From fuel to water	50%	100%	1%
From water to air, immediate	5×10^{-4}	5×10^{-2}	$<5 \times 10^{-4}$
From water to air, per day	1.3×10^{-4}	0.5	—
Decontamination factors	10^{-3}	1	10^{-3}

IV.2.3. Dose calculation

Calculated doses are related to the different zones inside the containment, the control room of the reactor and the environment (on the site and off the site). The evaluation of radiological consequences takes into account the most unfavourable weather conditions on the site. The integrated equivalent doses to an individual 500 m from the reactor (approximately corresponding to the distance to the nearest habitation) are presented in Table 13.

TABLE 13. INTEGRATED EQUIVALENT DOSES AT 500 m FROM SITE, 1 h POST-RELEASE

Mode of exposure	Dose (mSv)
Dose due to direct irradiation by fission products enclosed in the containment	0.9
Dose due to ground shine	0.13
Dose due to fission products released in the plume	1.9
Dose to thyroid	4.3

Appendix V

SOURCE TERM EVALUATION FOR THE 10 MW ASTRA RESEARCH REACTOR

V.1. INTRODUCTION

The ASTRA reactor is a multipurpose pool type research reactor with MTR fuel elements and a maximum power of 10 MW. In the context of the SAR [75, 76], a source term evaluation for accidents with significant releases to the environment was performed. For that purpose, possible initiating events leading to accident sequences were evaluated. An initiating event spectrum was established, and a PSA study was used to estimate the accident scenarios and their probability of occurrence [77]. Only those event tree branches with releases to the environment were considered further.

For these accident sequences, the radioactive release was estimated. The factors and parameters that influence the type and magnitude of release (the source term) are described in the sections that follow. It should be emphasized that the accident scenarios, as well as the parameters and factors influencing the source term, are specific to the ASTRA reactor and may not be generalized to other reactors without careful consideration of the system configuration of each reactor type. The fraction of release and the various retention factors as given in Section V.5 are derived using these parameters.

The actual source term for events with a probability of occurrence of less than 10^{-5} per year and the resulting dose for these accident event trees are calculated. No source term calculations for event trees of lower probability are performed, as the occurrence of these events is considered to be too rare to justify the laborious calculation effort. The source term and resulting dose may, however, be calculated in the same manner by use of the appropriate retention factors (see Table 16 in Section V.6).

V.2. DESCRIPTION OF REACTOR

The ASTRA reactor is a 10 MW pool type reactor. The core is composed of MTR type fuel elements of 20% enrichment. The fuel is in the form of U_xSi_y. Each of the approximately 18 standard elements is made up of 23 fuel plates containing 390 g of ^{235}U. The cladding material is aluminium. The majority of the other components of the pool are also made of aluminium; hafnium and stainless steel make up a small fraction of the surface material. Typically, four

control elements contain 17 plates and hold a correspondingly smaller amount of U_xSi_y [75].

Power is controlled by a minimum number of four fork type control rods, using hafnium as the absorber material. The reactivity worth of a control rod is between 2.25 and 5.5% $\Delta k/k$, depending on its position in the core.

The average burnup of the fuel is near 40%, and the maximum burnup is 62 ± 5%. The higher value holds for equilibrium cores, the lower value for transition cores (e.g. after changes in core configuration, fuel enrichment).

V.3. RADIOACTIVE RELEASE FROM CORE

V.3.1. Activity inventory in core

The activity inventory in the core may vary by orders of magnitude, depending on the power level and the operating history. Although the ASTRA reactor is currently operated on a power schedule of approximately 110 h of continuous operation for 1 week and daytime operation during the subsequent 1.5 weeks, to assess the accident consequences it was assumed that the reactor remains in continuous operation for 180 d. This is the maximum operating period with a core loading within the limits of maximum reactivity for a fresh fuel loading.

Under this assumption, no restrictions on reactor operation with regard to operating schedule are required as part of the operating licence. It should be kept in mind, however, that, under this assumption, the release expected in the case of an actual accident is significantly overestimated.

Although an accident can occur at any time — not just at the end of a long operating history, when long lived radionuclides have had sufficient time to build up, but even at the beginning of the reactor operation (startup accident), when there is a rather low activity inventory in the core — for the source term evaluation, an operating history with continuous operation of the reactor at the maximum power level for the maximum possible operating time with one core was assumed. This represents a worst case scenario with regard to the core inventory, and therefore the activity released.

The following assumption was used: reactor operation for 180 d at a continuous power level of 10 MW (1800 MW-d) (maximum operable time with one equilibrium core). On the basis of this assumption, the activity inventory can be calculated from the following:

$$C_i = 3.121\times10^{10}\frac{Pf_i\lambda_i}{\lambda_i+\sigma_a^i\phi}\left(1-e^{-(\lambda_i+\sigma_a^i\phi)t}\right) \quad (23)$$

where

C_i is the activity of fission product i (Bq);
P is the reactor power (W);
f_i is the fission yield (%);
λ_i is the decay constant (s^{-1});
σ_a^i is the absorption cross-section (cm^2);
ϕ is the neutron flux ($cm^{-2} \cdot s^{-1}$);
t is the reactor operation time (s).

Alternatively, it can be calculated by codes [22] (see Table 17 in Section V.6).

V.3.2. Fraction of core damaged

This section describes the fraction of core damage assumed for each accident scenario. The term 'core damage' as used hereinafter refers only to damage with significant release of radioactive material from the damaged region, not to damage such as bending or distortion of fuel plates or cladding effects that do not lead to any significant releases.

V.3.2.1. Startup accident

The startup accident is characterized by a continuous introduction of reactivity, for example, by a simultaneous, continuous withdrawal of control rods. If no counteractions (e.g. shutdown) are taken, such a withdrawal will eventually result in a partial meltdown of the core if the reactivity introduced is sufficiently large to cause short reactor periods, which cannot be coped with by the inherent safety features (temperature and Doppler coefficient, evaporation and dryout of cooling channels).

At several test facilities (BORAX, SPERT, SNAPTRAN), experiments were performed to yield data concerning the minimum reactivity insertion required for core damage to result and the maximum radioactive releases to be expected in such a case [14, 78, 79]. It was shown that a reactor period of less than 5 ms, which would result in partial destruction of the core, was achieved only when the insertion was sufficiently rapid and the burnup level of the core was low.

The high average burnup levels of equilibrium cores typical of the ASTRA reactor and the high neutron background levels due to the beryllium reflected core make highly unlikely a reactivity insertion by extreme accidental or deliberate control rod movements that would result in a reactor period of less than 5 ms [75]. Nonetheless, the assumptions were based on such a

hypothetical reactivity insertion resulting in a reactor period sufficiently short to cause core damage with a release of radioactive material. A maximum of 10% of the core was assumed to melt in this type of accident (the melted fractions observed in previous, actual excursions were lower: 7% in the SPERT test, 3–4% in the SNAPTRAN test and 5% in the SL accident [14]).

With this type of accident, the release occurs under water. Since the reactivity increase is continuous (owing to the continuous control rod withdrawal), the power excursions (oscillations) become larger and larger, with an increasingly shorter reactor period from one oscillation to the next. When the period falls below 5 ms, destruction of the centre part of the core terminates the criticality. The total power production is not large enough to cause an extensive expulsion of water from the pool. Thus, the release is into water, with enough water remaining above the damaged core to significantly reduce the release to the containment atmosphere (see Section V.4).

V.3.2.2. Loading accident

A loading accident is similar to a startup accident, except that the reactivity increase may be a very steep ramp function, or even close to a step form if the error occurs under extreme conditions. Such accidents are virtually impossible by design (inadvertent dropping of fuel elements into an open position of a critical core is prevented by operating rules and is almost impossible owing to the reactor design). Nonetheless, this type of extremely unlikely loading error was assumed, by which a reactor period of less than 5 ms would be feasible, resulting in core damage similar to that in the startup accident, but with the expulsion of a significant amount of the pool water into the confinement.

As with the startup accident, the probability of such short reactor periods becomes very low (if not zero) with higher burnup levels and with the high neutron background level due to the beryllium reflector. This fact is ignored here to maintain a conservative approach. With low burnup levels, the activity inventory — in particular, with regard to long lived radionuclides — is much lower. This fact is also neglected in the source term calculations.

As with the startup accident, a maximum fraction of 10% of the core is assumed to melt. The release occurs into water, with the expulsion of water from the pool resulting in lower retention factors for noble gases in the water, but with retention factors for the volatiles and aerosols that are only slightly lower than in the case of complete coverage by water [76].

V.3.2.3. Common mode failures

Even under the most adverse conditions, severe common mode failures will not lead to a serious reactivity excursion at the ASTRA reactor [76]. Under extreme conditions, however, they may result in a LOCA with the same core damage fraction as a LOCA due to other reasons (see below).

V.3.2.4. Fuel channel blockage

The special construction of the fuel elements provides a high level of protection against blockage by pieces dropped on the core; thus, a blockage of the fuel element channel by particles falling on the core is feasible only for specific types of small particles. Owing to the small size of the possible blocking particles, the damage to the core in this case is expected to be rather low (in the range of less than 1%). The release occurs under water.

V.3.2.5. Loss of forced cooling

With a 10 MW MTR reactor, a loss of forced cooling accident does not lead to any destruction of the core, not even under the most adverse conditions [75]. Therefore, it is not considered further.

V.3.2.6. Large beam tube rupture

Beam tubes at the ASTRA reactor are horizontal. They are designed as double tubes, with an additional plate on the outside of the concrete shield to prevent leakage in case of a break of the inner tubes. A tube rupture leading to a loss of coolant from the pool is thus extremely unlikely. Nevertheless, an accident was assumed in which a double fracture of the tubes and the cover plate occurs, which could eventually lead to a LOCA. In this case, the core spray system would start operating and cool the core. In the case of a failure of this system, a further emergency core cooling system, which may be supplied by water from various sources (tap water, fire pond water), becomes operational. With this multiple, diversified protection system, the probability of occurrence of this type of accident leading to a core melt event is less than 10^{-5} per year [77]. Therefore, as described in Section 1 of this report, this accident scenario was not evaluated further with regard to its source term. However, the parameters and assumptions that would have been used in this accident scenario are given in the annexes to this report to demonstrate the procedure for research reactors where the possibility of a core melt event is greater than 10^{-5} per year.

To be sufficiently conservative, it is assumed that 100% of the core melts, although:

— The fraction of the core that would actually melt is smaller owing to significant cooling at the core surface (outer region of fuel elements), as demonstrated by Bartzis [80].
— The beam tubes are positioned in the middle or slightly above or below the centre line of the core. Thus, a beam tube rupture would not cause complete exposure of the core to air; a fraction of more than 20% of the core would remain submersed in the water. Therefore, a fraction of less than 80% — and more realistically, of less than 50% — may be expected to melt [76].
— The rough surface of older fuel elements yields a higher heat transfer to the convecting air, resulting in better cooling of the core in air [75].

V.3.3. Radioactive release from degraded core fraction

In accordance with Barton et al. [81] and Shibata et al. [37], the following conservative assumptions regarding the radioactive release from the molten fraction of the core were made:

— Noble gases: 100%
— I, Te, Cs: 27%
— Ba, Sr, Ru: 3%
— Other fission products: 0.1%

This is a conservative approach, because, for example, the noble gases in the intersection of the molten and non-molten regions of the core certainly would not be completely released.

V.4. RADIOACTIVE RELEASE INTO CONTAINMENT ATMOSPHERE

The fission products from the molten core fraction are released into the pool water, from which a certain fraction is released into the confinement atmosphere. Three cases can be distinguished. In the first case, the release occurs under water, and the water is not expelled from the pool (startup accident or fuel element channel blockage). In this case, very high retention of noble gases and to an even higher degree of iodine and other aerosols is obtained. The release of noble gases to the confinement air in actual accidents

or tests respectively was 1.5% [82] and 0.5% [45]. Conservatively, a value of 2.0% was assumed. Release factors for iodine were determined in several experiments, resulting in values of 4×10^{-6} to 5×10^{-5} [45] and 1×10^{-4} [83]. In the source term calculation, a value of 1×10^{-4} is used. For aerosols (solids), a release factor of less than 10^{-6} was assumed.

The second case involves a power transient and a significant energy release that causes the water surrounding the core to be expelled from the pool (loading accident). For this accident scenario, larger release parameters have to be assumed. In accordance with the values observed during the SPERT and SNAPTRAN experiments [79], a water release fraction of 40% for noble gases, 5×10^{-3} for iodine and 10^{-6} for aerosols was assumed.

The third case is a LOCA (beam tube rupture). The core is uncovered, and the release occurs directly into the confinement air with no retention in pool water.

It is assumed that there is no radioactive decay from the time of release from the fuel until the release to the confinement atmosphere. It is also assumed that there are no cleaning systems in the primary water such as filters or ion exchangers. Such systems are quite efficient at capturing the iodine and the aerosol fractions that are predominantly released in the case of fuel channel blockage.

V.5. INFLUENCES OF CONTAINMENT ON RELEASE FRACTION

The activity concentration in the containment atmosphere is given by:

$$\frac{dq_a}{dt} = -\lambda q_a - \frac{L_r}{V} q_a - v_d \frac{S}{V} q_a + r \frac{S}{V} q_s - \frac{fR_r}{V} q_a + Q \tag{24}$$

and

$$\frac{dq_s}{dt} = v_d q_a - r q_s \tag{25}$$

where

q_a is the activity concentration in the confinement air (Bq/m^3);
q_s is the activity concentration on surfaces in the confinement (Bq/m^2);
L_r is the leakage rate from the confinement (m^3/s);

v_d is the deposition rate on surfaces (m/s);
S is the free surface area (m^2);
V is the confinement volume (m^3);
r is the resuspension rate from surfaces (s^{-1});
R_r is the recirculation rate (m^3/s);
f is the filter retention factor of the recirculation filter;
Q is the production term from precursor nuclides ($m^{-3} \cdot s^{-1}$);
λ is the radioisotope decay constant (s^{-1}).

For the evaluation, the following assumptions and parameters were used:

— Instantaneous mixture of released radionuclides with the containment air (no decay during the movement from the release point to the exhaust point).
— Containment volume: 10 622 m^3.
— Total surface area (see Table 14): 5280 m^2.
— Emergency ventilation to achieve an underpressure of 0.5 mbar in the containment: 60 m^3/h.
— Retention of the filters in the exhaust stack [84]:
  • Rare gases: 0;
  • Iodine: 99.99% for I_2, 97.5% for CH_3I;
  • Aerosols: 99.97%.
— Air cleaning by recirculation of containment air through HEPA filters at a rate of 33 000 m^3/h. No coagulation or deposition of aerosols on containment surfaces was taken into account, which would yield an additional reduction in aerosol release as demonstrated in Ref. [85].

Each radionuclide and surface material has a typical decay constant λ, deposition rate and resuspension rate. Therefore, an evaluation using the above formula has to be performed for each relevant radionuclide and surface material. For noble gases (no deposition on the surfaces), v_d and r are zero.

Table 14 describes in detail the calculation of deposition and resuspension of iodine in the confinement according to Ref. [86]. A detailed description of the effect of confinement on the retention and decay of other radionuclides (noble gases, aerosols) is given in Ref. [76].

Not included in the evaluation are the surfaces of mobile devices such as experimental set-ups and shielding material, since they may be removed from the reactor building; however, deposition on these surfaces can contribute significantly to the reduction of airborne activities in the containment atmosphere.

TABLE 14. PARAMETERS RELEVANT TO DEPOSITION AND RESUSPENSION OF IODINE IN THE CONFINEMENT OF THE ASTRA REACTOR

Type of surface	Surface area, A (m^2)	Deposition velocity, v_d (cm/s)	$v_d \frac{A}{V}$	Resuspension rate, r (s^{-1})	Fraction of methyl iodide
Painted concrete	3320.4	0.3	9.38×10^{-4}	5×10^{-7}	0.074
Painted metal	535.6	0.18	9.08×10^{-5}	5×10^{-7}	0.059
PVC floor covering	1095.8	5×10^{-3}	3.09×10^{-6}	7.5×10^{-6}	0.074
Aluminium, dry	91.6	0.1	8.63×10^{-6}	5×10^{-7}	0.050
Aluminium, wet	120.4	0.29	3.29×10^{-5}	1.6×10^{-5}	0.083
Iron	80.1	0.24	1.73×10^{-5}	1.3×10^{-5}	0.055
Glass	35.4	7.8×10^{-4}	2.60×10^{-8}	—	—

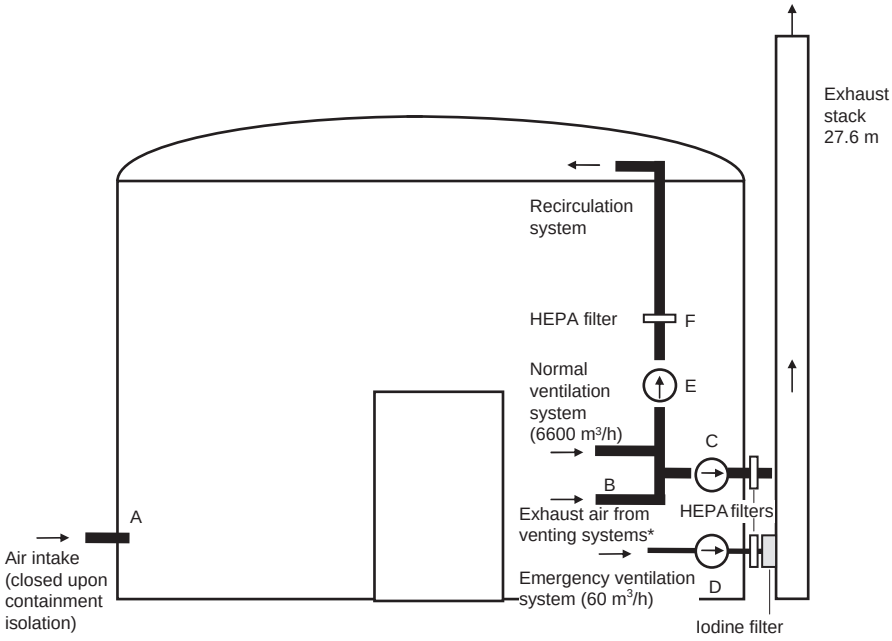
Figure 2 shows the ventilation system of the ASTRA reactor. Under normal operating conditions, the normal ventilation system draws air from the building at a rate of 6600 m^3/h. In an emergency situation, the intake pipe (A) and the various exhaust ducts (B) from experimental and other systems are closed, the normal ventilation system (C) is turned off, and the emergency ventilation system (D) begins operating. In the recirculation system, whose pump (E) is always operating, a HEPA filter (F) is inserted, which cleans the recirculated air.

A possible accident sequence discussed in the PSA study is the failure of the confinement isolation. This would occur if the normal ventilation system were not shut down, the air ducts for the inlet and outlet were not closed, and the emergency ventilation system were not started. Only in the case of fuel channel blockage is the probability of the 'confinement isolation failure' sequence greater than 10^{-9} per year, and this scenario was considered in the source term analysis.

In this case, four conditions with regard to the source term are possible:

(a) *Normal ventilation system on; air ducts not completely closed*. This case resembles that of a failure to switch from normal to emergency ventilation. The ventilation rate is 110 m^3/min (the normal ventilation rate), with no operational iodine filter in the exhaust system.

(b) *Normal ventilation system off; air ducts not completely closed; emergency ventilation system on*. In this case, the switch from normal to emergency ventilation is performed correctly, but the valves are not completely



** Venting systems for pool surface, beam tubes, gamma cell, thermal column and hot cells.*

FIG. 2. Reactor building (containment), and normal and emergency ventilation systems of the ASTRA reactor.

closed. Depending on the amount of leakage from the valve, the ventilation rate is somewhere between 1 and 20 m³/min, with an operational iodine filter in the exhaust system.

(c) *Normal ventilation system off; air ducts not completely closed; emergency ventilation system off.* In this case, the ventilation rate is not defined, but is probably lower than in the case of effective confinement isolation, with no operational iodine filter in the exhaust system.

(d) *Normal ventilation system off; air ducts completely closed; emergency ventilation system off.* In this case, the ventilation rate again is not defined, but due to closure of the valves it is probably lower than in case (c), with no operational iodine filter in the exhaust system.

Case (a) represents the worst case scenario for a loss of confinement, with a normal ventilation rate (little decay and deposition in the containment) and no operational iodine filter in the exhaust system. It may be caused by either failure to automatically switch on the emergency ventilation system in the case of high activity levels or failure of the system to respond. In case (b), the system

is switched on correctly, but the valves fail to close properly, resulting in a significantly lower release due to the lower ventilation rate and the fact that the air passes through an iodine filter. Cases (c) and (d) show significantly reduced radioactive releases, since neither the normal nor the emergency ventilation system is operating. Because the exhaust air is not forced through the emergency filters, there is no filtering of the air released to the environment. In this case, it is expected that the air recirculation system is not operational.

For simplification, only three confinement states are considered with regard to source term evaluation: normal functioning as installed and tested (effective confinement isolation, air recirculation and operational iodine filter); normal ventilation turned off and emergency ventilation not operational (no iodine removal and no air recirculation); and no confinement isolation (worst case scenario), with a ventilation rate of 110 m³/min and no effective iodine filter. Table 15 shows the confinement retention factors for these scenarios.

TABLE 15. CONFINEMENT RETENTION FACTORS FOR RELEVANT RADIONUCLIDES FOR DIFFERENT FUNCTIONAL STATES OF THE CONFINEMENT SYSTEM

Radionuclide	Effective confinement isolation, L_r = 1 m³/h		No confinement isolation, L_r = 110 m³/h, no iodine filter
	Iodine removal and air recirculation	No iodine removal and no air recirculation	
^{87}Kr	0.0103	0.0103	0.533
^{88}Kr	0.0223	0.0223	0.715
^{133}Xe	0.5085	0.5085	0.991
^{135}Xe	0.0695	0.0695	0.892
^{138}Xe	0.0019	0.0019	0.174
^{131}I	1.0×10^{-5}	7.4×10^{-3}	0.718
^{133}I	5.0×10^{-6}	4.2×10^{-3}	0.401
^{132}Te	5.5×10^{-9}	1.2×10^{-6}	3.0×10^{-4}
^{137}Cs[a]	5.5×10^{-9}	3.0×10^{-6}	3.0×10^{-4}

Note: L_r: Leakage rate from the confinement.

[a] Valid for all aerosols with a half-life of more than 30 d.

V.6. RELEASE CATEGORIES AND SOURCE TERMS

Table 16 shows the core damage fraction and the release fractions from water to containment air and to the environment for each accident scenario. The source terms for each accident scenario can be calculated from Table 16 and the activity inventory in the core after 180 d of continuous power operation at 10 MW. The results are given in Table 17 for event sequences with a probability of occurrence of greater than 10^{-9} per year.

V.7. CALCULATION OF DOSE IN ENVIRONMENT

From the fraction of the radioactive inventory released to the environment that was obtained using the above assumptions, the dose to be expected in the environment was calculated by the FONTA code [87] using the following assumptions:

- Dose conversion factors as used in WASH-1400 [24].
- Release height: 27.6 m (stack release), 0 m (ground release):
  - Cross-section of building: 560 m²;
  - Deposition velocity: 10^{-2} m/s;
  - Washout factor: 10^{-4} per second.
- Calculations were performed for three meteorological diffusion characteristics:
  - Diffusion class 4 (neutral), wind speed 3.5 m/s (most frequent class);
  - Diffusion class 4 (neutral), wind speed 1.5 m/s (most frequent wind speed);
  - Diffusion class 6 (stable), wind speed 1.0 m/s (highly adverse parameters).
- The diffusion characteristics (diffusion class, speed and direction of wind) were assumed to remain constant throughout the release. This is a very conservative assumption, since the release may last for days or weeks, during which a change in the wind direction or the other diffusion parameters is very likely. A change in these parameters would lead to a significant reduction of the expected maximum doses.
- Iodine released into the environment in the form of methyl iodine is treated as being in the I_2 stage (a conservative approach, as the fallout and washout rates of I_2 are higher by a factor of 1000 than those of CH_3I).

The doses resulting from the calculated source terms and the above meteorological assumptions and parameters are given in Table 17.

TABLE 16. RELEASE FRACTIONS AND RETENTION FACTORS FOR RELEVANT RADIONUCLIDES

Accident scenario	Core damage	Water release fraction			Confinement retention					
		Noble gases	Iodine	Solids	^{88}Kr	^{133}Xe	^{135}Xe	^{138}Xe	^{131}I [a]	Solids[b]
Startup accident	10	0.02	1×10^{-4}	10^{-6}	0.022	0.509	0.070	0.002	0.0074	6×10^{-9}
Loading accident	10	0.4	5×10^{-3}	10^{-4}	0.022	0.509	0.070	0.002	0.0074	6×10^{-9}
Fuel channel blockage, confinement isolation	1	0.02	1×10^{-4}	10^{-6}	0.022	0.509	0.070	0.002	0.0074	6×10^{-9}
Fuel channel blockage, confinement isolation, no iodine removal	1	0.02	1×10^{-4}	10^{-6}	0.022	0.509	0.070	0.002	0.0074	3×10^{-6}
Fuel channel blockage, confinement failure	1	0.02	1×10^{-4}	10^{-6}	0.715	0.991	0.892	0.174	0.718	3×10^{-4}
Beam tube rupture	100	1	1	1	0.022	0.509	0.070	0.002	0.0074	6×10^{-9}

[a] Not including iodine removal in emergency exhaust filter.
[b] Including HEPA filter in exhaust duct (effective under both normal and emergency ventilation).

TABLE 17. SOURCE TERM FOR RELEVANT RADIONUCLIDES AND RESULTING DOSE IN THE VICINITY (TBq)

Radionuclide	Activity in core	Activity released				
				Fuel channel blockage		
		Startup accident	Loading accident	Effective confinement isolation, iodine removal and air recirculation	Effective confinement isolation, no iodine removal or air recirculation	Confinement failure, iodine removal and air recirculation
^{87}Kr	8.88×10^{3}	0.183	3.66	0.0183	0.0183	0.947
^{88}Kr	13.56×10^{3}	0.605	12.10	0.0605	0.0605	1.939
^{133}Xe	25.20×10^{3}	25.63	512.6	2.563	2.563	4.995
^{135}Xe	3.38×10^{3}	0.470	9.40	0.047	0.0470	0.603
^{138}Xe	24.23×10^{3}	0.092	1.84	0.0092	0.0092	0.843
^{131}I	10.56×10^{3}	2.9×10^{-7}	1.4×10^{-5}	2.9×10^{-8}	2.1×10^{-5}	2.0×10^{-3}
^{133}I	25.28×10^{3}	2.8×10^{-7}	1.4×10^{-5}	2.8×10^{-8}	2.9×10^{-5}	2.7×10^{-3}
^{129m}Te	1.28×10^{3}	1.9×10^{-13}	1.9×10^{-11}	1.9×10^{-14}	1.0×10^{-11}	1.0×10^{-9}
^{132}Te	15.73×10^{3}	2.3×10^{-12}	2.3×10^{-10}	2.3×10^{-13}	5.1×10^{-11}	1.3×10^{-8}
^{137}Cs	0.26×10^{3}	3.9×10^{-14}	3.9×10^{-12}	3.9×10^{-15}	2.1×10^{-12}	2.1×10^{-10}
^{89}Sr	16.46×10^{3}	2.7×10^{-13}	2.7×10^{-11}	2.7×10^{-14}	1.5×10^{-11}	1.5×10^{-9}
^{90}Sr	0.26×10^{3}	4.3×10^{-15}	4.3×10^{-13}	4.3×10^{-16}	2.3×10^{-13}	2.3×10^{-11}
^{91}Sr	21.95×10^{3}	3.6×10^{-13}	3.6×10^{-11}	3.6×10^{-14}	7.9×10^{-12}	2.0×10^{-9}
^{103}Ru	11.19×10^{3}	1.9×10^{-13}	1.9×10^{-11}	1.9×10^{-14}	1.0×10^{-11}	1.0×10^{-9}
^{106}Ru	0.42×10^{3}	6.9×10^{-13}	6.9×10^{-13}	6.9×10^{-16}	3.8×10^{-13}	3.8×10^{-11}

TABLE 17. SOURCE TERM FOR RELEVANT RADIONUCLIDES AND RESULTING DOSE IN THE VICINITY (TBq) (cont.)

Radionuclide	Activity in core	Activity released				
				Fuel channel blockage		
		Startup accident	Loading accident	Effective confinement isolation, iodine removal and air recirculation	Effective confinement isolation, no iodine removal or air recirculation	Confinement failure, iodine removal and air recirculation
^{140}Ba	23.74×10^{3}	3.9×10^{-11}	3.9×10^{-11}	3.9×10^{-14}	8.5×10^{-12}	2.1×10^{-9}
Other	$0.1–30 \times 10^{3}$	2×10^{-12}	2×10^{-12}	2×10^{-15}	3×10^{-13}	8×10^{-11}
Dose in vicinity (mSv)[a]	—	(1) 0.006 (2) 0.014 (3) 0.010	(1) 0.18 (2) 0.35 (3) 0.24	(1) 0.0006 (2) 0.0014 (3) 0.0010	(1) 0.0006 (2) 0.0014 (3) 0.0010	(1) 0.050 (2) 0.011 (3) 0.010
Probability of occurrence	—	2×10^{-8}	4×10^{-9}	4×10^{-4}	3×10^{-6}	5×10^{-6}

[a] At point of maximum exposure 200 m downwind from stack: (1) diffusion class 4, wind speed 3.5 m/s; (2) diffusion class 4, wind speed 1.5 m/s; (3) diffusion class 6, wind speed 1.0 m/s.

V.8. CONCLUSION

Owing to the installed cleaning features and the low emergency ventilation rate resulting from significant efforts to improve the leaktightness of the ASTRA reactor confinement building, those source terms derived for accident sequences having a probability of greater than 10^{-9} per year showed relatively low release values compared with the activity inventory in the core, and therefore less serious consequences to members of the public and the environment in the vicinity of the reactor in the case of an accident. For the startup accident, with an estimated probability of occurrence of 2×10^{-8} per year, a release of 0.6 TBq of ^{88}Kr, 25.6 TBq of ^{133}Xe and 0.3 MBq of ^{131}I was estimated, which would give a dose of about 0.01 mSv external exposure and a less than 0.8 μSv thyroid dose (0.02 μSv effective dose) at the point of highest exposure in the vicinity of the reactor for a person remaining outdoors for an unlimited period of time.

For the loading accident, with an estimated probability of occurrence of 4×10^{-9} per year, a higher release of about 12 TBq of ^{88}Kr, 512 TBq of ^{133}Xe and 14 MBq of ^{131}I was evaluated. This would result in a dose of 0.2 mSv external exposure and a thyroid dose of about 0.02 mSv (0.0005 mSv effective dose) at the point of highest exposure in the vicinity of the reactor for a person remaining outdoors for an unlimited period of time. Thus, even the loading accident would result in only negligible consequences to members of the public and the environment in the vicinity of the reactor.

The fuel channel blockage accident, which, as a consequence of the downward flow of the coolant through the core, has a comparatively high probability of occurrence (4×10^{-4} per year), results in a negligible release to the environment. This is due to both the small fraction of the core that is degraded and the high retention of released radionuclides in the primary water. The derived values were 0.06 TBq of ^{88}Kr, 2.56 TBq of ^{133}Xe and 0.03 MBq of ^{131}I. The corresponding exposure would be 0.001 mSv external exposure and a less than 0.08 μSv thyroid dose.

All possible accident sequences with a probability of greater than 10^{-9} per year show source terms that would result in acceptable dose levels in the vicinity of the reactor, even under the most adverse meteorological conditions.

Appendix VI

CALCULATION OF RADIOLOGICAL CONSEQUENCES IN THE CASE OF A REPLACEMENT RESEARCH REACTOR BDBA

VI.1. THE REPLACEMENT RESEARCH REACTOR

The replacement research reactor (RRR) used as the example in this assessment is a pool type research reactor operating at a rated power of 20 MW. The standard configuration of the core comprises 16 MTR type fuel elements, with U–Si plates and aluminium cladding (19.7% enriched uranium). The operation cycle duration is 33 d, followed by a 2 d period for refuelling.

A reflector tank containing heavy water surrounds the reactor core. This tank has, among other irradiation and experimental facilities, 12 positions for U–Mo rigs. The irradiation cycle for the U–Mo rigs has a duration of 7 d. After a decay period of about six hours, the U–Mo rigs are ready to be safely moved to the hot cell for post-irradiation processing.

The containment of the reactor is provided with an air supply system that brings outside air into the building and an exhaust system with fans and absolute filters that releases inside air through the stack, where activity is monitored. In the event of high activity in the released air, the reactor protection system initiates the containment isolation to prevent further radioactive material releases and begins recirculation of the containment air through absolute and charcoal filters.

VI.2. BEYOND DESIGN BASIS ACCIDENTS

On the basis of the analyses performed, a number of BDBA sequences with the potential to lead to damage to the core or the irradiation rigs have been identified. The purpose of this appendix is to analyse these sequences further, with a view to defining an accident to be used for emergency planning purposes.

Two BDBAs were chosen for illustration, one occurring under water and the other in air, namely:

— Partial blockage of cooling channels in a fuel assembly;
— Erroneous early removal of a U–Mo rig to the hot cells.

In the case of a partial blockage of cooling channels, it is assumed that three of the plates melt. This assumption is highly conservative. Furthermore, no credit is given for the probability of flux perturbations causing a reactor trip following the onset of nucleate boiling, which would occur long before melting.

The erroneous early removal of a U–Mo rig is assumed to result in the melting of the U–Mo targets when they leave the water and enter the hot cell. The heat dissipated to the surrounding air from the targets as a result of natural circulation is assumed to be insufficient to prevent their melting.

VI.3. BLOCKAGE OF COOLING CHANNELS IN A FUEL ASSEMBLY

It is postulated that a small object may enter the pool cooling system, bypass the different filters and block two fuel channels. Even though it is not credible to postulate the presence inside the pool cooling system of an object large enough to totally block two channels, total blockage is assumed and no credit is given to the coolant flow that is in contact with the outermost faces of the two outer plates. All three fuel plates are assumed to melt and release their inventory into the reactor pool. The following is assumed for the accident:

— Initial release is via the stack at a height of 45 m.
— Containment isolation is not initiated until 2 min after the detection of activity in the stack. This is a conservative assumption that accounts for the integration time of the stack activity monitors and the delay of the associated electronics.
— Following containment isolation, fission product release occurs at ground level. During the first day, 3% of the volume of the containment is released. Of this release, one third (i.e. 1% of the containment volume) is released during the initial pressure transient following containment isolation. The remaining 2% is released due to the variation in barometric pressure outside the containment. Thereafter, a 2% release per day is assumed based on the barometric pressure variation. This assumes that the worst historical barometric pressure variation recorded at the reactor site is maintained during the 100 d following the accident.
— The analysed sequence has been divided into five periods:
  • Prompt period: Includes the first 2 min of the release of fission products through the stack. The venting rate during this period is one reactor building volume per hour, and the release occurs 45 m after having been filtered. The filters are assumed not to retain noble gases.

- Period 1: Lasts from the end of the Prompt period (120 s) to 12 h (42 300 s) post-accident. Owing to the change in the release flow rate, period 1 has been divided into three subperiods: P1A, P1B and P1C.
- Period 2: Starts 12 h after the beginning of the sequence and lasts 12 h. The only change during this period is in the atmospheric conditions, as shown in Table 18.
- Period 3: Lasts 12 h and represents a return to the meteorological conditions used in the Prompt period and period 1 (see Table 18).
- Period 4: Lasts 98.5 d. Introduces a change in the meteorological conditions, as shown in Table 18.

— The environmental release is assumed to begin at the start of the least dispersive weather conditions, as this maintains a concentrated airborne plume, maximizing the estimate of individual dose. These conditions are typical of night-time inversion conditions with Pasquill atmospheric stability category F and a low wind speed of 1 m/s. It is assumed that these conditions last for 12 h periods over two consecutive nights. During the

TABLE 18. SUMMARY OF CONDITIONS DURING BDBA SEQUENCES WITH FAILURE OF FUEL PLATES UNDER WATER

Period	Time (s)	Release status		Meteorological conditions	Wind speed (m/s)
Prompt	0–120	Normal (exhaust)	2400%/d	F[a] (winter)	1
P1A	120–10^3	Isolated containment CERS[b]	1% + (1/12)% is released during the first hour	F (winter)	1
P1B	10^3–10^4	Isolated containment CERS	2%/d	F (winter)	1
P1C	10^4–4.32×10^4	Isolated containment CERS	2%/d	F (winter)	1
2	4.32×10^4–8.64×10^4	Isolated containment CERS	2%/d	D (winter)	3
3	8.64×10^4–1.296×10^5	Isolated containment CERS	2%/d	F (winter)	1
4	1.296×10^5–8.64×10^6	Isolated containment CERS	2%/d	D (winter)	3

[a] Pasquill atmospheric stability category.
[b] CERS: containment energy release system.

alternate 12 h daytime period, Pasquill atmospheric stability category D with a wind speed of 3 m/s is assumed. This is also assumed to be the average condition for the final release period up to 100 d. Furthermore, it is also assumed that the wind constantly remains in the direction of the most populated sector for the Prompt period and release periods 1 and 3, but changes direction to the adjacent sector during periods 2 and 4.

— Starting in period 1, the deposition/plate-out effects and nuclide radioactive decay produce a variation in the inventory inside the reactor building.

The release of fission products occurs under water; therefore, partition factors that represent the transfer up to the reactor building atmosphere are applied. The absence of coolant flow is assumed to result in the formation of steam surrounding the overheated rig. Bubbles breaking out from this blanket entrain fission products released from the damaged rig. The bubbles are small, of the order of a few centimeters in diameter [88]. The pool water above the irradiation rigs is significantly subcooled owing to the large mass of water inside the pool. Thus, the bubbles condense over a few centimeters and do not transport fission products to the pool top.

The degree of fission product retention in pool water has been studied experimentally [89, 90]. Table 19 presents a summary of the partition fractions adopted in the calculations. No delay has been considered between the underwater release and the release into the reactor hall atmosphere. This is a conservative approach, as it neglects radioactive decay and the transit time in the pool circuits.

TABLE 19. PARTITION FRACTIONS IN RADIONUCLIDE TRANSPORTATION

Radionuclide	Release from fuel type source, F_{cp} (%)	Release from pool water to containment, F_{pb} (%)
Noble gases (xenon and krypton)	100	100
Iodine	30	0.5
Caesium	30	0.01
Rubidium	30	0.01
Tellurium	1	0.01
Ruthenium	1	0.01

Over time, deposition and plate-out within the containment reduce the amount of airborne fission products in the containment. Exponential decay has been assumed, with an associated decay constant λ_d. Table 20 shows the values of λ_d for different radionuclides.

Leakage of air from the reactor building leads to a reduction in the source strength that can be represented by an exponential decay with a decay constant λ_l, shown in Table 21 for different leakage rates.

Throughout the sequence, the iodine fraction of fixed to organic substances is taken to be 5%.

The activity inventory for three fuel plates is presented in Table 22.

Calculations are performed with PC-COSYMA [63] to determine the dose to an average person 1.6 km from the release point. It is assumed that the Prompt release occurs at a height of 45 m and that all subsequent releases occur at ground level. The results are shown in Table 23.

TABLE 20. DEPOSITION AND REMOVAL DECAY CONSTANT (λ_d)

Radionuclide	λ_d (s^{-1})
Noble gases (xenon and krypton)	0
Iodine	
Inorganic	3.85×10^{-5}
Organic	0
Caesium	3.85×10^{-6}
Rubidium	3.85×10^{-6}
Tellurium	3.85×10^{-6}
Ruthenium	3.85×10^{-6}

TABLE 21. LEAKAGE DECAY CONSTANT (λ_l) FOR DIFFERENT LEAKAGE RATES (L/t_l)

L/t_l	λ_l (s^{-1})	Comment
100%/h	2.78×10^{-4}	Applicable to a ventilation exhaust flow of 10^4 m^3/h
3%/d	3.47×10^{-7}	
5%/d	5.79×10^{-7}	
10%/d	1.16×10^{-6}	
30%/d	3.47×10^{-6}	

TABLE 22. ACTIVITY INVENTORY FOR THREE FUEL PLATES (Bq)

Isotope	Prompt	Period 1	Period 2	Period 3	Period 4
Xe-131m	6.12E+10	2.64E+10	2.53E+10	1.62E+10	4.06E+11
Xe-133m	3.55E+11	1.43E+11	1.21E+11	6.80E+10	3.72E+11
Xe-133	1.17E+13	4.94E+12	4.57E+12	2.82E+12	3.56E+13
Xe-135m	1.99E+12	2.58E+10	0.00E+00	0.00E+00	0.00E+00
Xe-135	9.11E+11	2.62E+11	1.04E+11	2.76E+10	1.83E+10
Xe-138	1.03E+13	1.22E+11	0.00E+00	0.00E+00	0.00E+00
Kr-83m	9.03E+11	8.61E+10	9.12E+08	6.37E+06	6.76E+04
Kr-85m	2.10E+12	4.19E+11	6.48E+10	6.66E+09	1.22E+09
Kr-85	1.54E+10	6.73E+09	6.64E+09	4.37E+09	3.72E+11
Kr-87	4.19E+12	2.79E+11	4.04E+08	3.83E+05	5.49E+02
Kr-88	5.94E+12	8.41E+11	4.47E+10	1.57E+09	8.78E+07
I-130	1.72E+08	3.04E+07	4.04E+06	6.36E+05	4.96E+05
I-131	8.15E+09	1.80E+09	4.58E+08	1.39E+08	1.92E+09
I-132	1.18E+10	9.86E+08	6.35E+06	4.58E+04	8.97E+02
I-133	1.78E+10	3.46E+09	6.10E+08	1.27E+08	1.91E+08
I-134	1.98E+10	7.79E+08	1.42E+04	2.75E-01	0.00E+00
I-135	1.66E+10	2.44E+09	1.76E+08	1.49E+07	4.38E+06
Te-125m	1.42E+03	5.70E+02	4.74E+02	2.63E+02	1.31E+03
Te-127m	3.26E+04	1.31E+04	1.09E+04	6.07E+03	3.09E+04
Te-127	3.51E+05	9.49E+04	3.27E+04	7.48E+03	3.93E+03
Te-129m	2.03E+05	8.13E+04	6.73E+04	3.72E+04	1.81E+05
Te-129	1.35E+06	8.01E+04	5.24E+01	2.24E-02	1.45E-05
Te-131m	6.96E+05	2.46E+05	1.56E+05	6.59E+04	1.15E+05
Te-131	4.61E+06	9.83E+04	0.00E+00	0.00E+00	0.00E+00
Te-132	7.83E+06	3.00E+06	2.26E+06	1.13E+06	3.46E+06
Te-133m	4.93E+06	2.34E+05	2.45E+01	0.00E+00	0.00E+00
Te-133	6.56E+06	6.83E+04	0.00E+00	0.00E+00	0.00E+00
Te-134	1.13E+07	4.04E+05	2.27E+00	0.00E+00	0.00E+00
Cs-134m	6.48E+06	8.93E+05	4.31E+04	1.38E+03	6.96E+01
Cs-134	4.48E+06	1.80E+06	1.51E+06	8.41E+05	4.35E+06
Cs-136	2.42E+06	9.62E+05	7.84E+05	4.26E+05	1.90E+06

TABLE 22. ACTIVITY INVENTORY FOR THREE FUEL PLATES (Bq) (cont.)

Isotope	Prompt	Period 1	Period 2	Period 3	Period 4
Cs-137	3.88E+06	1.56E+06	1.31E+06	7.29E+05	3.78E+06
Cs-138	3.44E+08	9.47E+06	1.53E+00	0.00E+00	0.00E+00
Rb-86	2.45E+05	9.77E+04	8.03E+04	4.39E+04	2.04E+05
Rb-88	1.74E+08	2.62E+06	0.00E+00	0.00E+00	0.00E+00
Rb-89	2.25E+08	2.87E+06	0.00E+00	0.00E+00	0.00E+00
Ru-103	1.83E+08	7.34E+07	6.08E+07	3.36E+07	1.65E+08
Ru-105	8.18E+07	1.53E+07	1.97E+06	1.69E+05	2.49E+04
Ru-106	1.04E+07	4.21E+06	3.52E+06	1.96E+06	1.01E+07

TABLE 23. CALCULATED DOSE TO AVERAGE PERSON 1.6 km FROM RELEASE POINT (μSv)

Distance	Prompt	Period 1	Period 2	Period 3	Period 4	Total
1600 m	5.5	1.85	0.04	0.08	0.15	7.62

The collective effective dose for this scenario, calculated for the population within a radius of 22.5 km from the reactor, is 0.11 person-Sv, well below the 200 person-Sv limit in the nuclear authority regulations. Therefore, a cooling channel blockage that could lead to the melting of three fuel plates would result in a dose to the public that is well below nuclear authority limits and would require no emergency interventions or countermeasures such as evacuation and supply of iodine tablets.

VI.4. ERRONEOUS EARLY REMOVAL OF IRRADIATED U–Mo TARGETS TO A HOT CELL

It is postulated that three U–Mo targets are erroneously removed from the decay rack in the service pool before having undergone adequate cooling, and are placed in the elevator and transported to the hot cell. A further assumption is the failure of the interlock that inhibits the transport from the pool to the hot cell in the event of high activity. The occurrence of this event

requires sequential failures to adhere to operating procedures and the presence of an undisclosed mechanical failure. This is an unlikely sequence of events.

This event is assumed to result in the melting of the U–Mo targets when they leave the water and enter the cell. The heat dissipated from the targets by natural circulation of the surrounding air is assumed to be insufficient to prevent their melting. Taking a very conservative approach, no credit is given to refreezing of the targets when they come into contact with colder surfaces, such as those in the hot cell.

The air inside the hot cells is circulated by means of a dedicated ventilation system. Five per cent of the ventilation flow rate is sent to the stack to compensate for air leakage from the containment to the hot cell. This results in a release to the atmosphere of 1440% of the volume of the cell (50 m^3) per day. Both the recirculated and the vented air is filtered through absolute filters (to retain aerosols) and activated charcoal filters (to retain iodine). Subsequent to the release of fission products from the targets, 100% of the noble gases is assumed to be released and not retained by the filters. It is further assumed that the activated charcoal filters have a total degraded efficiency of 90% and that the absolute filters have a total degraded efficiency of 99.99%. The containment is isolated after 2 min, and the cell ventilation system continues to recirculate the air in the hot cell. The negative pressure of the hot cell relative to the containment is lost after containment isolation. The conservative assumption is made that all the noble gases that remain within the cell after the Prompt discharge (2 min) are released into the containment. After the noble gases are transferred to the containment, they are released to the atmosphere following the same pattern as the release for the accident inside the reactor pools. Thus, during the first hour, the transient of the containment conditions leads to a release of 1% of the containment volume. In addition, after the containment isolation, a 2% volume per day release due to variations in barometric pressure is assumed. Table 24 provides a summary of the assumptions adopted for the analysis of this accident.

In addition, it is assumed that the filters, the plate-out inside the cell and the containment isolation with recirculation remove all the iodine and particulates. Therefore, iodine and particulates are released only during the Prompt period and consequently do not make a significant contribution to the dose once the containment has been isolated, as can be seen in Table 25.

Calculations are performed with PC-COSYMA [63] to determine the dose to an average person 1.6 km from the release point. It is assumed that the Prompt release occurs at a height of 45 m, and that all subsequent releases occur at ground level. The results are shown in Table 26.

The collective effective dose for this scenario, calculated for the population within a radius of 22.5 km from the reactor, is 0.18 person-Sv, well

TABLE 24. SUMMARY OF CONDITIONS DURING BDBA SEQUENCES WITH FAILURE OF U–Mo TARGETS IN AIR INSIDE HOT CELL

Period	Time (s)	Release status		Meteorological condition	Wind speed (m/s)
Prompt	0–120	Normal (exhaust)	1440% hot cell volume/d	F[a] (winter)	1
P1A	120–10^3	Isolated containment CERS[b]	1%[c] + (1/12)%[c] is released during first hour	F (winter)	1
P1B	10^3–10^4	Isolated containment CERS	2%/d	F (winter)	1
P1C	10^4–4.32×10^4	Isolated containment CERS	2%/d	F (winter)	1
2	4.32×10^4–8.64×10^4	Isolated containment CERS	2%/d	D (winter)	3
3	8.64×10^4–1.296×10^5	Isolated containment CERS	2%/d	F (winter)	1
4	1.296×10^5–8.64×10^6	Isolated containment CERS	2%/d	D (winter)	3

[a] Pasquill atmospheric stability category.
[b] CERS: containment energy release system.
[c] Corresponds to percentage of volume of the containment.

below the 200 person-Sv limit in the nuclear authority regulations. Thus, the removal of three U–Mo targets to the hot cell before the predetermined decay time would result in a dose to the public that is well below nuclear authority limits and would require no emergency interventions or countermeasures such as evacuation and supply of iodine tablets.

TABLE 25. ACTIVITY INVENTORY FOR FAILURE OF U–Mo TARGET IN AIR INSIDE HOT CELL (Bq)

Isotope	Prompt	Period 1	Period 2	Period 3	Period 4
Xe-131m	4.00E+09	1.97E+11	1.46E+08	1.06E+05	7.67E+01
Xe-133m	1.501E+11	7.29E+12	4.74E+09	3.02E+06	1.92E+03
Xe-133	3.75E+12	1.84E+14	1.31E+11	9.16E+07	6.40E+04
Xe-135m	9.05E+11	7.73E+12	0.00E+00	0.00E+00	0.00E+00
Xe-135	3.22E+11	1.41E+13	4.34E+09	1.30E+06	3.91E+02
Xe-138	4.61E+12	3.67E+13	0.00E+00	0.00E+00	0.00E+00
Kr-83m	4.37E+11	1.32E+13	1.08E+08	8.54E+02	0.00E+00
Kr-85m	1.05E+12	4.11E+13	4.91E+09	5.72E+05	6.67E+01
Kr-85	4.35E+08	2.15E+10	1.64E+07	1.22E+04	9.13E+00
Kr-87	2.07E+12	5.31E+13	5.95E+07	6.40E+01	0.00E+00
Kr-88	2.84E+12	9.94E+13	4.08E+09	1.63E+05	6.49E+00
I-130	1.12E+08	2.14E+08	0.00E+00	0.00E+00	0.00E+00
I-131	1.13E+11	2.15E+11	0.00E+00	0.00E+00	0.00E+00
I-132	2.58E+11	4.80E+11	0.00E+00	0.00E+00	0.00E+00
I-133	4.52E+11	8.63E+11	0.00E+00	0.00E+00	0.00E+00
I-134	5.19E+11	9.22E+11	0.00E+00	0.00E+00	0.00E+00
I-135	4.21E+11	7.97E+11	0.00E+00	0.00E+00	0.00E+00
Te-125m	1.44E+02	2.76E+02	0.00E+00	0.00E+00	0.00E+00
Te-127m	5.09E+04	9.74E+04	0.00E+00	0.00E+00	0.00E+00
Te-127	5.88E+06	1.12E+07	0.00E+00	0.00E+00	0.00E+00
Te-129m	1.70E+06	3.25E+06	0.00E+00	0.00E+00	0.00E+00
Te-129	4.21E+07	7.61E+07	0.00E+00	0.00E+00	0.00E+00
Te-131m	2.74E+07	5.24E+07	0.00E+00	0.00E+00	0.00E+00
Te-131	1.68E+08	2.76E+08	0.00E+00	0.00E+00	0.00E+00
Te-132	2.59E+08	4.95E+08	0.00E+00	0.00E+00	0.00E+00
Te-133m	2.15+08	3.83E+08	0.00E+00	0.00E+00	0.00E+00
Te-133	2.39E+08	3.41E+08	0.00E+00	0.00E+00	0.00E+00
Te-134	4.53E+08	7.89E+08	0.00E+00	0.00E+00	0.00E+00
Cs-134m	5.25E+04	9.81E+04	0.00E+00	0.00E+00	0.00E+00
Cs-134	1.79E+03	3.43E+03	0.00E+00	0.00E+00	0.00E+00
Cs-136	1.58E+05	3.03E+05	0.00E+00	0.00E+00	0.00E+00

TABLE 25. ACTIVITY INVENTORY FOR FAILURE OF U–Mo TARGET IN AIR INSIDE HOT CELL (Bq) (cont.)

Isotope	Prompt	Period 1	Period 2	Period 3	Period 4
Cs-137	2.71E+05	5.18E+05	0.00E+00	0.00E+00	0.00E+00
Cs-138	4.38E+08	7.42E+08	0.00E+00	0.00E+00	0.00E+00
Rb-86	8.02E+02	1.53E+03	0.00E+00	0.00E+00	0.00E+00
Rb-88	2.29E+08	3.55E+08	0.00E+00	0.00E+00	0.00E+00
Rb-89	3.03E+08	4.53E+08	0.00E+00	0.00E+00	0.00E+00
Ru-103	3.40E+07	6.50E+07	0.00E+00	0.00E+00	0.00E+00
Ru-105	6.540E+07	1.23E+08	0.00E+00	0.00E+00	0.00E+00
Ru-106	5.17E+05	9.90E+05	0.00E+00	0.00E+00	0.00E+00
Ba-139	4.54E+07	8.28E+07	0.00E+00	0.00E+00	0.00E+00
Ba-140	1.43E+04	2.74E+04	0.00E+00	0.00E+00	0.00E+00
Sr-90	6.98E+01	1.33E+02	0.00E+00	0.00E+00	0.00E+00
Sr-91	3.45E+05	6.56E+05	0.00E+00	0.00E+00	0.00E+00
Sr-92	1.78E+01	3.33E+01	0.00E+00	0.00E+00	0.00E+00
Sr-93	1.20E+03	2.39E+03	0.00E+00	0.00E+00	0.00E+00
La-141	9.66E+04	1.82E+05	0.00E+00	0.00E+00	0.00E+00

TABLE 26. CALCULATED DOSE TO AVERAGE PERSON 1.6 km FROM RELEASE POINT (μSv)

Distance	Prompt	Period 1	Period 2	Period 3	Period 4	Total
1600 m	11.8	0.8	0.012	0.031	0.059	12.7

Appendix VII

SOURCE TERM AND RADIOLOGICAL CONSEQUENCE ANALYSIS

VII.1. METHODOLOGY

Source term and radiological consequence analysis of research reactor accidents follows the sequence below:

(1) Define the accident scenarios. Determine which scenarios are DBAs and which are BDBAs. Select scenarios that are credible for analysis.
(2) Define the radioisotope content of the reactor core for burnup conditions that match and bound the scenarios selected.
(3) For each scenario selected, calculate the reactor time–evolution history (power and energy released versus time, peak fuel, cladding, coolant temperatures attained, and any other necessary safety parameters such as coolant pressure peak, safety system response, operator response).
(4) Determine the kind and extent of fuel damage to permit bounding of the amount of fission products, actinides and other radioisotopes released to the environment.
(5) If there is fuel damage and radiation is released from the fuel, determine the release pathways and amounts to various key points. For example:
    (a) What fraction of the radioisotopes is released from the fuel?
    (b) What fraction of the radioisotopes remains in the coolant?
    (c) What fraction of the radioisotopes emerges from the coolant system into the reactor building air?
    (d) What are the radiation dose rates to operations staff and experimenters in the reactor building?
    (e) What are the release pathways and rates from the reactor building?
    (f) Is radiation released from the reactor building as a puff, as a series of puffs or continuously?
    (g) For several wind conditions, determine the radiation dose rates downwind at the site boundary or at any other locations required by the licensing authority.
    (h) Using personnel occupancy estimates, determine doses to reactor building occupants and to the general public.
(6) Review the consequence analysis for all scenarios. Rank the scenarios in order of seriousness. Verify that those having the most serious consequences are both credible and well defined. Refine the scenarios if necessary.

VII.2. SOFTWARE

Computer codes are necessary to determine the isotopic content of the fuel with acceptable precision. Although there may be hundreds of isotopes present in irradiated fuel, only a relatively few will turn out to be responsible for the bulk of the consequences for a given scenario.

For example, given the fuel composition, power versus time history and nominal neutron spectrum, the ORIGEN code [22] can determine the radioisotope inventory. ORIGEN breaks the photon (gamma) source down by origin — from actinides, fission products, or other structural or target materials. ORIGEN assumes a zero dimensional model for a lump of fuel irradiated uniformly by a selected neutron spectrum, with the neutron flux level or power prescribed. The irradiation time history of the fuel lump is assumed to be represented by a histogram of constant power (or flux) levels. The isotopic content for a given burnup depends on the past history, but not strongly. Consequently, it is relatively simple to bound the content for each scenario. ORIGEN also provides information on the neutron source from the (gamma, n) reaction.

Predicting the accident evolution requires thermohydraulic analysis of both transient and steady state conditions. Codes such as PARET [91] and PLTEMP/ANL [92] can be used for these conditions, respectively. The RELAP5/Mod3 code [10] can also be used. PARET represents a reactor as a small number of radial ring volumes and effective coolant channels. Axial segmentation is also featured. Reactivity insertion events can be defined as functions of time. Reactivity feedback effects from fuel, Doppler and coolant voiding are accounted for. Power and energy release versus time are computed, along with peak fuel centre line and cladding temperatures. PLTEMP represents the hottest fuel assembly as a series of fuel plates and coolant channels. The fuel plate geometry can be either flat or tubular. PLTEMP provides information on the steady state temperature profile, both axially and across each fuel plate and coolant channel. It also provides safety margins to such key events as onset of nucleate boiling, critical heat flux and onset of flow instability. Both PARET and PLTEMP can be used for light and heavy water coolants.

The MACCS2 code [62] can be used to estimate the movement, isotopic transmutation and dissipation of a radioactive cloud formed by the release of a quantity of radioisotopes as a gaseous discharge. It provides estimates of radiation dose rates and integrated doses at various off-site distances from the source point, given certain assumptions about atmospheric conditions. Using the radioactive cloud content (radioisotope activity of alpha, beta and/or gamma emissions) provided by MACCS2, manual calculations can also be used

to estimate (and bound) the dose rates and committed doses from submersion, inhalation and ingestion.

The MicroShield code [53] can be used to estimate the gamma radiation dose rates to operations staff and experimenters on the site. As a given scenario is analysed, certain geometrical arrangements will become clear that address the issue of direct and indirect exposure of personnel in the reactor building. It is important to realize that a single calculation can be made by a code such as MicroShield that includes all of the gamma rays from a source containing many radioisotopes and kernel numerical integration codes.

Generally, the (gamma, n) neutron source is small in any reactor fuel. Simple manual calculations can be used to estimate the dose rates from neutrons at key locations, in the event that heavy actinides mixed with strong gamma sources are released from the core as solid lumps. Should the dose rates from neutrons become an issue, then fixed-source calculations will be necessary to determine the neutron flux and energy spectrum (and dose rate) at points of interest. For this type of calculation, radiation transport codes will be needed. Both multigroup (diffusion theory, Sn transport) and continuous energy (Monte Carlo) methods can be used to determine the dose rate at specific locations. Radiation transport analysis by these methods is quite complex.

VII.3. SOURCE TERM AND RADIOLOGICAL CONSEQUENCE EVALUATION FOR A HYPOTHETICAL ACCIDENT AT THE TRIGA RESEARCH REACTOR AT INR PITEȘTI

The TRIGA research reactor at INR Pitești is basically a pool type reactor with a special design to fulfill the requirements for materials testing, fuel testing and nuclear safety studies. The dual core concept involves the operation of a high flux, steady state reactor (TRIGA SSR) at one end of a large pool and the independent operation of an annular core pulse reactor (TRIGA ACPR) at the other end of the pool (Fig. 3). The SSR is used for long term testing of power reactor fuel components (pellets, pins, subassemblies and fuel assemblies), and the ACPR is used for transient testing of power reactor fuel specimens.

Both reactors have beam tubes for experimental purposes and may be operated separately or at the same time. The two reactors are completely independent of each other, with two exceptions: they share a reactor pool and a cooling and water purification system.

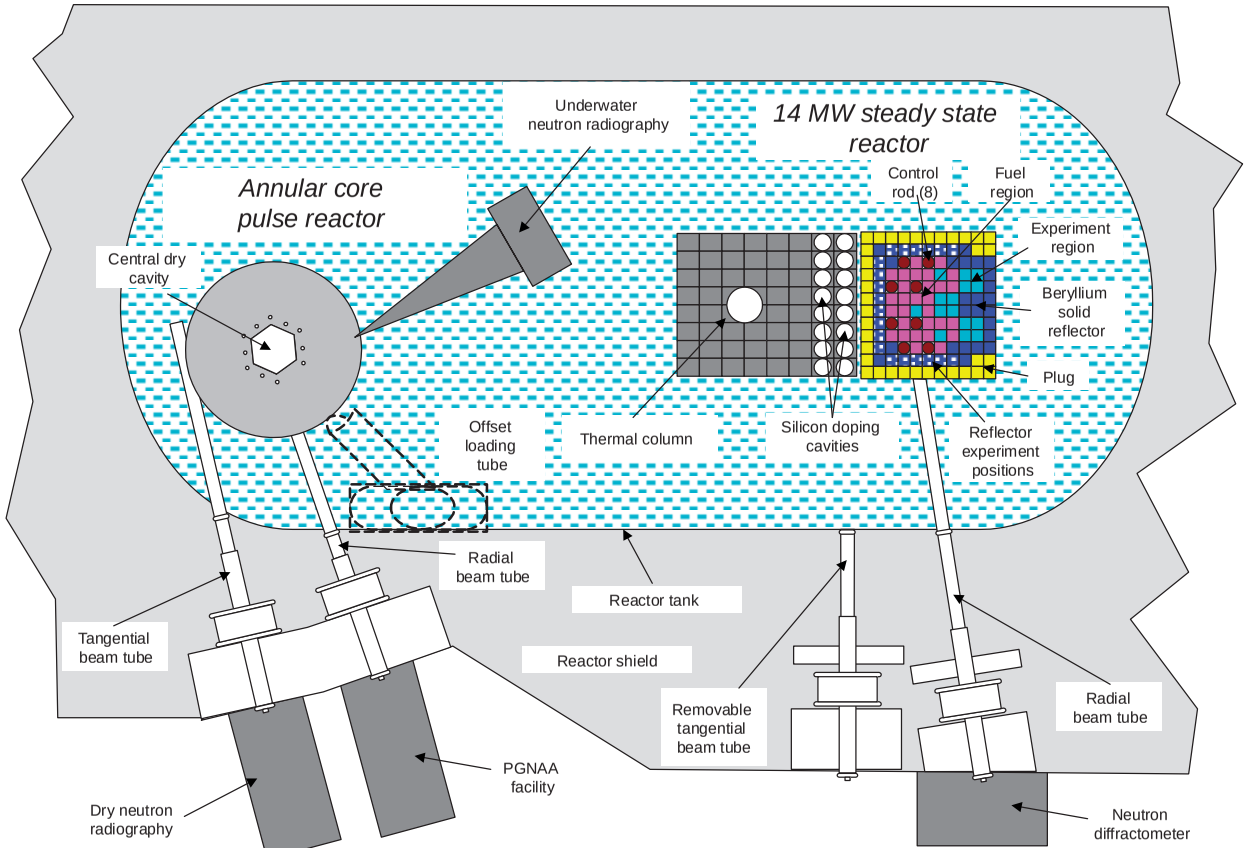


FIG. 3. Layout of the TRIGA research reactor at INR Pitești.

To illustrate the use of the methodology for source term and radiological consequence evaluation, a hypothetical severe accident scenario involving the 14 MW TRIGA research reactor at INR Pitești is presented:

Step 1: Define the accident scenarios. Determine which scenarios are DBAs and which are BDBAs. Select scenarios that are credible for analysis.

The safety evaluation of the TRIGA research reactor at INR Pitești involves several DBAs [93]: single pin cladding failure in water, 25 pin fuel bundle failure in water and in air, accidental reactivity insertions, loss of flow from main coolant pump accidents, and interaction between the two cores within the tank.

In this hypothetical severe accident scenario, it is assumed that a large part of the reactor hall roof falls or that a heavy object is dropped from the crane hook onto the 14 MW TRIGA SSR core, resulting in mechanical damage to it. It is also assumed that no melting of the core occurs, with only fuel cladding rupture being involved for several 25 pin fuel bundles. In fact, this is an extension of a DBA. This particular scenario is unlikely, having an occurrence probability of less than 10^{-5}, and was developed for training purposes only.

Step 2: Define the radioisotope content of the reactor core for burnup conditions that match and bound the scenarios selected.

Step 3: For each scenario selected, calculate the reactor time–evolution history.

The core has operated discontinuously for a total of 1780 MW·d. On the basis of the operating history and the composition of TRIGA fuel, we can construct input for the ORIGEN computer code to evaluate the core inventory [94].

Step 4: Determine the kind and extent of fuel damage to permit bounding of the amount of fission products, actinides and other radioisotopes released to the environment.

It is assumed that no core melting occurs, with only fuel cladding rupture being involved for several 25 pin fuel bundles. The affected fraction of the core is 45%. The isotopic mixture of the released effluents during a reactor accident will strongly depend on the mechanism involved in fuel damage, on the status of the safety barriers and on the dynamic of the accident. The released fraction is a function of the isotope volatibility and the temperature reached by the core

during the accident. If the fuel cladding rupture occurs during normal reactor operation, the following are released into the main coolant loop: noble gases (Xe and Kr), extremely volatile fission products (iodine) and volatile fission products (Cs, Te, Ru). As the temperature rises, other isotopes could also be released into the main coolant loop.

In this scenario, the fuel temperature during the accident is far below the melting point. For the release, an experimentally determined fraction of volatile products released from the fuel material (about 6.3×10^{-4}) is considered [95]. It is also assumed that there is no retention of volatile fission products in the fuel moderator material. From the damaged fraction of the core, 100% of the noble gases in the fuel–cladding gap are released from the fuel bundles and are subsequently transferred directly to the reactor hall. Also, 25% of the halogens are released from the fuel bundles (with the remainder assumed to plate-out on the relatively cool cladding). Regarding the halogens that escape, 10% are assumed to form organic compounds that escape into the pool water. Only 1% of the balance is not dissolved in the pool water and appears in the reactor hall air. The net halogen release to the reactor room and potentially outside is 2.725%. All other fission products remain in the pool or are otherwise unable to escape from the reactor room because of plate-out on cool surfaces [94–96].

Step 5: If there is fuel damage and radiation is released from the fuel, determine the release pathways and amounts to various key points.

In this accident scenario, the quantities of fission product gases released are great enough to be detected in the reactor hall air, but the emergency ventilation system fails, and therefore the normal ventilation system remains operative. Although the removal of halogens by activated charcoal filters can be easily accomplished with efficiencies greater than 99.0%, here it is assumed that no noble gases are removed by the filters.

It is assumed that a fraction of isotope *i* from this inventory is released to the reactor hall instantaneously. This fraction w_i is:

$$w_i = \left(\frac{p}{N}\right) e_i f_i g_i \tag{26}$$

where

(p/N) is the relative power density in the failed bundle (= 2/29);
e_i is the fraction of isotope *i* released to the fuel–cladding gap;

f_i is the fraction of isotope *i* released to the pool;
g_i is the fraction of isotope *i* released to the reactor room.

For the anticipated release, the value of e_i is 6.3E-04, whereas for the design basis release it is assumed to be equal to 1. For release in water, the values for f_i and g_i are shown in Table 27.

The value of g_i for the halogens is based on the assumption that 10% of the halogens form organic compounds that are insoluble in water, and 90% of the halogens are in elemental or particulate form, of which all but 1% are retained in the water.

Thus, the release conditions can be summarized as follows:

— Fraction of core involved: 45%;
— Fraction of fission products available for release to the fuel–cladding gap: 6.3×10^{-4};
— Fraction of available fission products released to the pool:
  • Noble gases: 100%;
  • Halogens: 25%;
— Fraction of fission products released from the pool water:
  • Noble gases: 100%;
  • Organic halogens: 25%;
  • Elemental and particulate halogens (90% of total): 1%;
— Condition of ventilation system: normal;
— Exhaust rate from stack: 24 360 m^3/h;
— Release height: 60 m.

To evaluate the radiological consequences of this accident to both the general public and the environment, an accident consequence assessment (ACA) computer code must be used. The COSYMA computer code was used here to evaluate the radiological consequences; the MACCS computer code can also be used. To prepare the input for COSYMA [96, 97], some site specific data must be supplied and some assumptions have to be made related to specific conditions for each site.

TABLE 27. VALUES FOR f_i AND g_i

Fission product	f_i	g_i
Noble gases	1.00	1.00
Halogens	0.25	$0.109 = 0.1 + (0.1 \times 0.9)$
Other	0.00	0.00

From a meteorological point of view, a site specific meteo data file must be provided, but calculations for a single Pasquill atmospheric stability category, wind speed and rainfall rate can be also performed.

From an emergency response point of view, the calculations can be performed for both the case with no countermeasures and that with countermeasures implemented. In the latter case, site specific pieces of information about the emergency response must be provided (e.g. threshold values of doses and/or concentrations for protective actions, options for decontamination, options for relocation) [98, 99]. If the ingestion pathway is used in the calculation, a decision about the use of ECOSYS or FARMLAND libraries must be taken. Also, site specific values for food consumption rates, and agricultural and animal production, for example, must be provided. Food consumption rates for this working example [100] are presented in Table 28.

Step 6: Review the consequence analysis for all scenarios.

The evaluation of the radiological consequences takes into account both near and long term consequences. For the near term consequences, a deterministic analysis was performed. Table 29 presents the mean concentration of radionuclides 2.250 km from the release point, with the limit representing the approximate boundary with the residential area of Mioveni (with approximately 30 000 inhabitants). Figure 4 shows the mean individual 1 d dose for

TABLE 28. ANNUAL FOOD CONSUMPTION RATES

Type of food	Consumption rate (kg/a)
Milk (fresh or products)	124
Meat	64
Beef	23
Pork	37
Lamb (sheep)	4
Fruit and vegetables	122
Fruit	60
Leafy vegetables	38
Root vegetables	24
Potatoes	81
Cereals	84

TABLE 29. MEAN CONCENTRATION OF RADIONUCLIDES 2.250 km FROM RELEASE POINT

Radionuclide	Concentration	
	Ground (Bq/m^2)	Air (Bq/m^3)
Kr-87	0.00E+00	4.90E+07
Kr-88	0.00E+00	9.72E+07
I-131	9.51E+05	9.60E+07
I-132	1.02E+06	1.03E+08
I-133	2.22E+06	2.24E+08
I-134	1.07E+06	1.08E+08
I-135	1.92E+06	1.94E+08
Xe-133	0.00E+00	2.38E+08
Xe-135	0.00E+00	9.37E+07
Xe-138	0.00E+00	7.38E+06
Cs-134	2.51E+00	2.51E+03
Cs-137	7.99E+00	7.99E+03

different organs. Tables 30 and 31 present the results of the evaluation of long term individual risk (fatal) and the mean long term individual dose at 50 years, respectively, using a probabilistic approach, without the implementation of countermeasures. The consequences with countermeasures implemented according to Refs [98, 101] were also calculated, and the results for affected areas are presented in Table 32.

In the example provided here, all calculations were made for the TRIGA HEU fuel. Following the general trend of using LEU fuel to the greatest extent possible, the TRIGA LEU fuel can also be considered. The ^{235}U enrichment for TRIGA HEU fuel is 93.15 wt% and that for TRIGA LEU fuel is 20%. The calculations show that the differences between the two fuel types will result in a higher core inventory for HEU, especially for heavy elements (i.e. actinides and transuranic elements), while modifications for noble gases, halogens and other volatile fission products are not so important. Taking into account the special design of the INR Piteşti TRIGA research reactor (with the core located in a large, deep pool), the influence of the core conversion on the source term will be limited to the extremely volatile and volatile fission

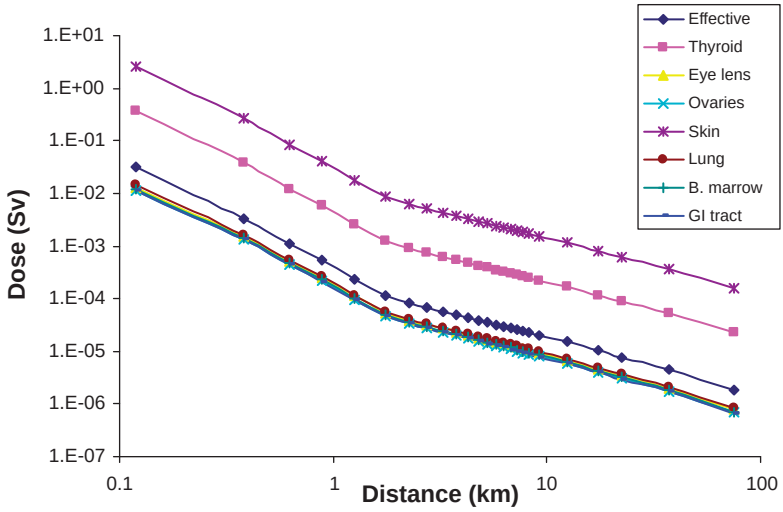


FIG. 4. Mean individual 1 d dose: deterministic approach.

products, that is, the noble gases and halogens. This will result in basically the same radiological consequences for the general public and the environment. For other types of research reactor, with different fuel types and different reactor designs, the differences between the radiological consequences generated by the two source terms for HEU and LEU type fuels may be important.

VII.4. MANUAL DOSE CALCULATION

VII.4.1. Description of the method used for calculation

While manual calculations have been replaced by computerized solutions, preparation of input to run the code can be a time consuming, laborious process. Thus, in some instances, manual calculations based on nomograms can still play an important role. One example is an emergency response to an accidental release of radioactive contaminants when the health of persons close to the accident site might be at risk. In this case, results from computerized ACA models may be delayed owing to equipment malfunction or the time required to develop minimal input files and perform the calculations (typically longer than 5 min). A simple nomogram (developed using computerized dispersion model calculations) can provide dispersion and dose estimates within a minute.

TABLE 30. LONG TERM INDIVIDUAL RISK (FATAL): TOTAL, PROBABILISTIC APPROACH, WITHOUT COUNTERMEASURES

Distance (km)	Individual risk				
	Mean	Max.	99%	95%	90%
0.12	6.67E-02	9.50E-01	9.50E-01	4.68E-01	2.19E-01
1.25	1.99E-03	9.92E-02	3.24E-02	1.38E-02	6.46E-03
2.75	7.94E-04	4.70E-02	1.35E-02	5.75E-03	2.34E-03
5.25	3.69E-04	1.57E-02	6.31E-03	2.88E-03	9.33E-04
7.75	2.29E-04	1.09E-02	4.07E-03	1.66E-03	5.75E-04
12.5	1.14E-04	8.88E-03	2.14E-03	8.13E-04	2.14E-04

TABLE 31. MEAN LONG TERM INDIVIDUAL DOSE AT 50 YEARS: PROBABILISTIC APPROACH, WITHOUT COUNTERMEASURES

Distance (km)	Dose (Sv)					
	Effective	Skin	Bone marrow	Lung	Stomach	Colon
0.12	2.03E+00	1.28E+00	5.81E-02	6.27E-02	7.77E-02	5.97E-02
0.62	1.55E-01	9.02E-02	4.86E-03	5.23E-03	6.34E-03	4.97E-03
1.25	5.53E-02	3.06E-02	1.80E-03	1.93E-03	2.32E-03	1.84E-03
2.25	2.76E-02	1.48E-02	9.09E-04	9.73E-04	1.17E-03	9.26E-04
5.25	1.03E-02	5.37E-03	3.31E-04	3.54E-04	4.26E-04	3.38E-04
7.25	6.98E-03	3.62E-03	2.21E-04	2.37E-04	2.86E-04	2.27E-04
8.25	5.87E-03	3.07E-03	1.85E-04	1.98E-04	2.39E-04	1.90E-04
9.25	4.97E-03	2.63E-03	1.56E-04	1.67E-04	2.02E-04	1.60E-04
12.5	3.17E-03	1.74E-03	9.82E-05	1.05E-04	1.28E-04	1.01E-04
17.5	2.11E-03	1.16E-03	6.42E-05	6.86E-05	8.36E-05	6.58E-05
22.5	1.59E-03	8.46E-04	4.68E-05	5.00E-05	6.14E-05	4.82E-05

TABLE 32. EVALUATION OF AFFECTED AREAS, WITH COUNTERMEASURES

	Area (km^2)				
	Area A	Area B	A + B	Sheltering	Iodine
Maximum	2.36E+01	9.28E+00	3.28E+01	4.86E+02	9.03E+01
Mean	2.36E+01	1.56E-01	2.37E+01	8.02E+01	1.81E+01
Probability <1.0E-04	0.00E+00	9.51E-01	0.00E+00	0.00E+00	0.00E+00
99th percentile	2.36E+01	3.39E+00	2.75E+01	4.86E+02	4.17E+01

The objective of this method is to use some simple nomograms, tables and manual calculations in combination with the meteorological conditions and accident scenario assumptions to estimate the dose to the general public at a certain distance near the site. To this end, a set of nomograms and tables based on the formulas presented below has been developed for rapid dose calculation.

Any calculation of the consequences of an accidental radioactive release into the atmosphere starts with the identification of an accident scenario and release path and characteristics of the source term. Here, the status of the fuel must be established (e.g. possible damage to the fuel), as well as the core inventory at the start of the accident and the fraction of the inventory released to the atmosphere. The amount of the core inventory released to the reactor pool, the fraction escaped from the pool into the reactor room and the amount of radioactivity released into the atmosphere directly from the reactor room or through the stack can be calculated on the basis of the accident scenario and the safety design of the nuclear reactor. If there is a stack release, the efflux velocity of the gaseous releases can be expressed by:

$$w_0 = 1.274 \frac{R_{\text{efflux}}}{D^2} \tag{27}$$

where R_{efflux} is the release rate (rate of exhaust from the stack, based on the design of the ventilation system) and D is the internal stack diameter. This efflux velocity will be used later to calculate the buoyant plume rise and the effective release height.

The purpose of the atmospheric transport and diffusion calculation is to provide estimates of the concentration and surface deposition from routine and

accidental releases of pollutants to the atmosphere. These calculations provide the link between emissions to the atmosphere and direct or indirect pathways to humans, which are important for dose calculation.

The most commonly used atmospheric concentration calculation method is the Gaussian plume equation. This is an empirical formula that is based on an analytical solution to the diffusion equation under the assumption of constant wind speed, no wind shear, flat topography and Fickian diffusion. The equation for a continuous source point is:

$$\chi(x,y,z) = \frac{Q}{2\pi \cdot \bar{u}\sigma_y\sigma_z} \exp\left(-\frac{y^2}{2\sigma_y^2}\right)\left[\exp\left(-\frac{z-H}{2\sigma_z^2}\right) + \exp\left(-\frac{z+H}{2\sigma_z^2}\right)\right] \tag{28}$$

where

$\chi(x,y,z)$ is the time integrated concentration (Bq·s·m^{-3}) at point (x, y, z); the time period for integration is greater than or equal to the release time;

Q is the the total activity released into the atmosphere (Bq), corrected for buildup, deposition and disintegration;

σ_y, σ_z are the standard deviations (m) of the normal crosswind and the vertical concentration distributions of plume materials, respectively;

$\bar{u}$ is the mean wind speed (m/s) at effective release height H;

H is the effective release height (m).

For $y = 0$ (i.e. plume centre line calculations), the above formula becomes:

$$\chi(x,0,z) = \frac{Q}{2\pi \cdot \bar{u}\sigma_y\sigma_z}\left[\exp\left(-\frac{z-H}{2\sigma_z^2}\right) + \exp\left(-\frac{z+H}{2\sigma_z^2}\right)\right] \tag{29}$$

The standard deviations of the normal crosswind and the vertical concentration distributions of plume materials are respectively [102, 103]:

$$\sigma_{yi} = a_i \cdot x^{b_i} \quad \text{and} \quad \sigma_{zi} = c_i \cdot x^{d_i} \tag{30}$$

with coefficients a_i, b_i, c_i and d_i from Table 33.

Taking into account the intention of finding a simplifying assumption in order to create nomograms for rapid calculations, a non-depositing plume is considered, and instead of the time integrated concentration, the normalized surface concentration ($\chi u/Q$) is used, which is a particular solution of the Gaussian diffusion equation [104] and can be expressed by:

TABLE 33. VALUES FOR CONSTANTS FOR σ_{yi} AND σ_{zi}, EQUATION (30)

Stability category		Constant			
P-G	i	a_i	b_i	c_i	d_i
A	1	0.3658	0.9031	0.00025	2.125
B	2	0.2751	0.9031	0.0019	1.6021
C	3	0.2089	0.9031	0.2	0.8543
D	4	0.1474	0.9031	0.3	0.6532
E	5	0.1046	0.9031	0.4	0.6021
F	6	0.0722	0.9031	0.2	0.6020

$$(\chi u/Q) = \frac{\chi(x,0,z)u}{Q} = \frac{1}{\pi\sigma_y\sigma_z}\exp\left(-\frac{H^2}{2\sigma_z^2}\right) \text{ (m}^{-2}\text{)} \quad (31)$$

The graphical solutions of the above equation are presented in Figs 5–10 for each atmospheric stability category and each release height.

To evaluate the expression $(\chi u/Q)$ and obtain the nomograms from Figs 5–10, the atmospheric stability category and then the effective release height must be evaluated.

The most widely known method for determining atmospheric stability categories is that originally proposed by Pasquill [101] in association with his first tentative σ_y and σ_z curves. This general classification scheme, based on insolation, cloud cover and wind speed, was adopted by Turner [105] for use with standard US National Weather Service observations at airports. The latter computational scheme evolved into what is known as the STAR computer program. Slade [106] found that σ_θ (the standard deviation of the horizontal wind direction) stratified the various data from different diffusion experiments. This method was also simplified by taking the range of wind direction from an analogue trace over a half hour period and dividing by six to obtain an approximate value for σ_θ. US NRC Regulatory Guide 1.23 [107] also recommends the use of a temperature gradient, with one temperature being measured at 10 m above the ground and the other temperature being measured at 40 m above the ground (or at the top of the stack) to determine the atmospheric stability categories. Golder [108] developed relations between the Pasquill atmospheric stability categories and the Richardson number. Smith et al. [109] used the measurements of σ_θ directly in the equations for σ_y and σ_z as

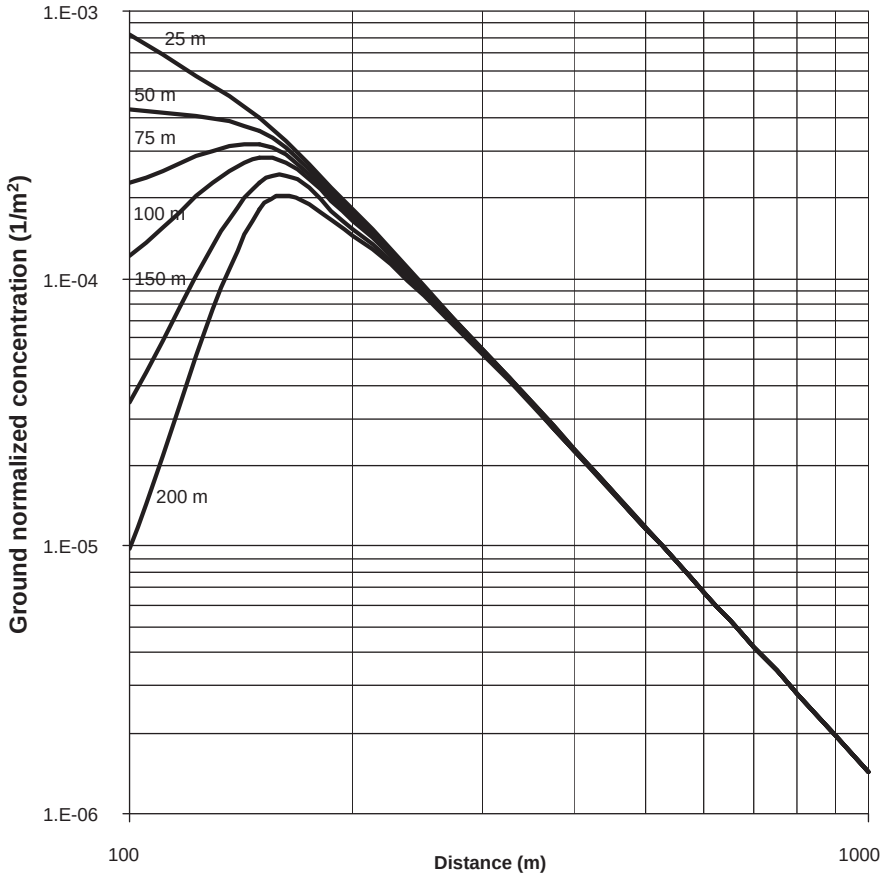


FIG. 5. Normalized surface concentration for several release heights for atmospheric stability category A.

a function of distance. These various classification methods and the relationships between them are summarized in Tables 34 and 35.

Plume segments that are hot (i.e. that contain appreciable sensible heat), and thus are buoyant, may rise to heights much greater than their initial release height. This is why the effective height of the release accounts for the initial release height (stack height), the buoyant plume rise (due to the heat content of the plume and the efflux velocity of the airborne contaminants) and the gravitational settling of each isotope in the plume. Plume rise can be calculated using equations recommended by Briggs [110, 111]. Plume rise is inhibited whenever the prevailing wind speed at the time of release exceeds a critical wind speed (lift-off criterion). Plume rise is also limited by the mixing height (height of any capping inversion layer).

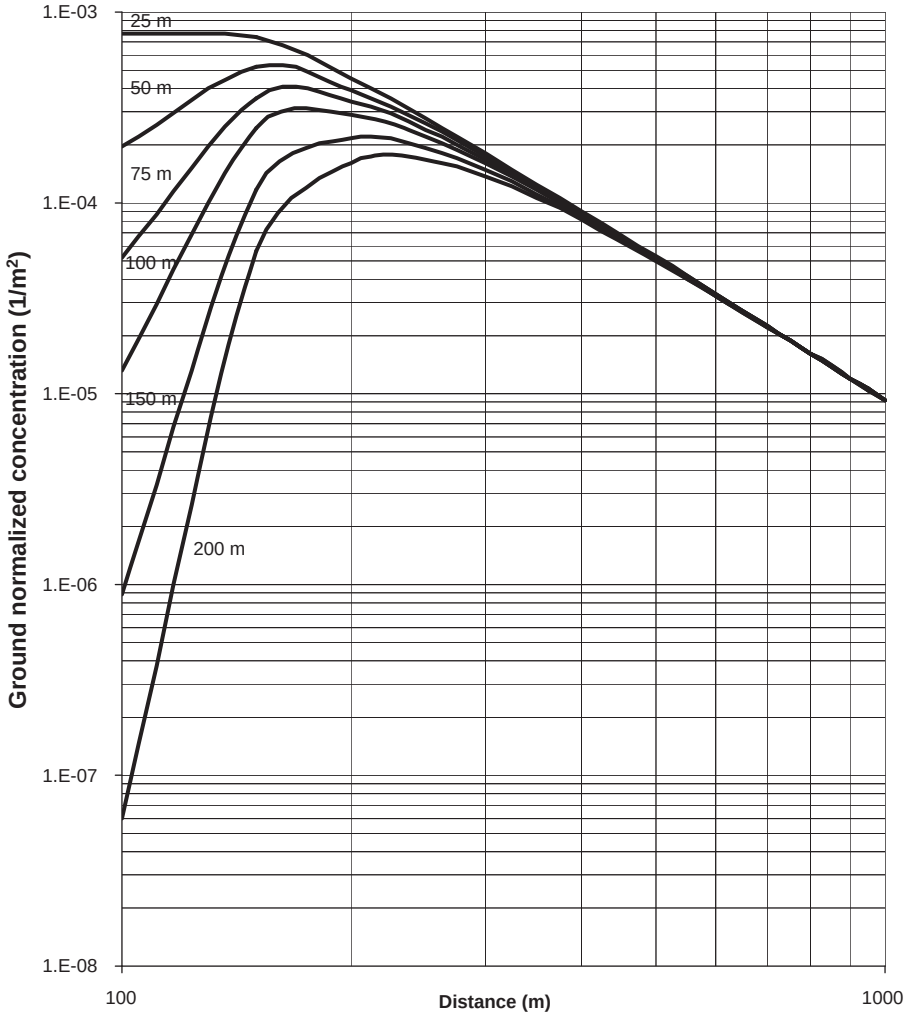


FIG. 6. Normalized surface concentration for several release heights for atmospheric stability category B.

The effective release height can be expressed as:

$$H = h + \Delta h_d \tag{32}$$

where

h is the stack height (m);
Δh_d is the buoyant plume rise (m).

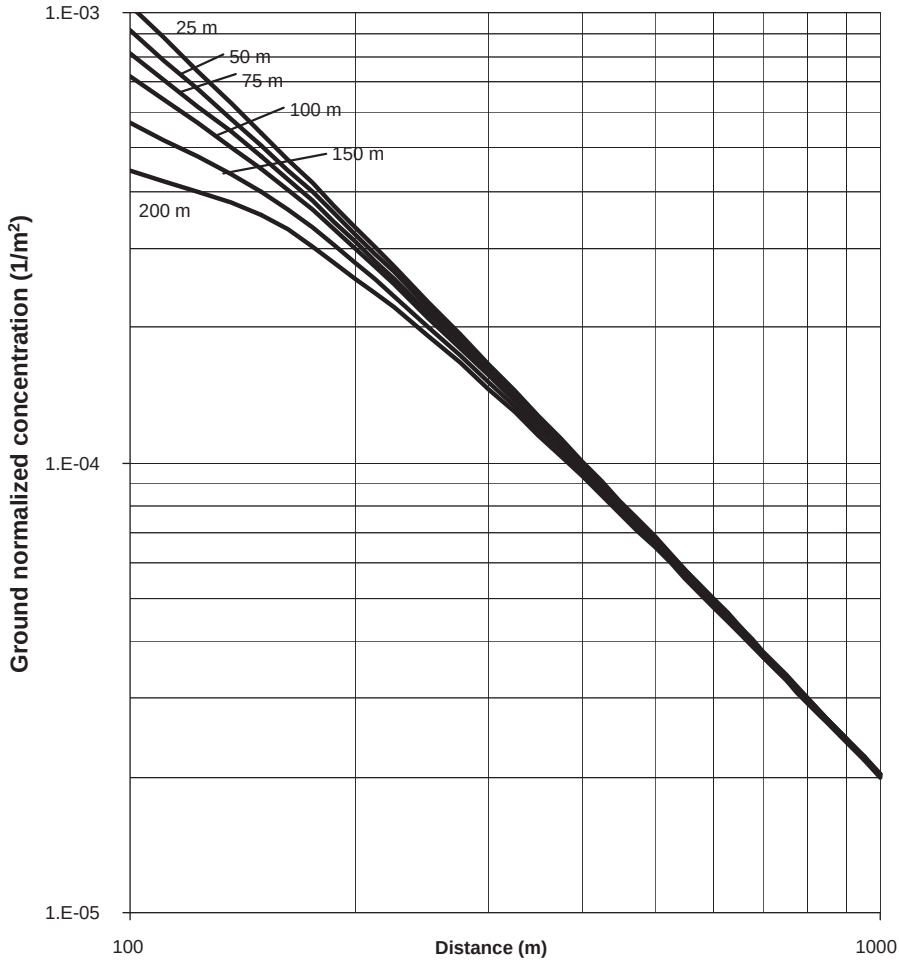


FIG. 7. Normalized surface concentration for several release heights for atmospheric stability category C.

The buoyant plume rise can be calculated by two methods [110–112]:

(1) According to the Briggs formula for neutral conditions:

$$\Delta h_d = 3 w_0 D (u)^{-1} \tag{33}$$

where w_0 is the efflux velocity (m/s), u is the wind speed (m/s) and D is the internal stack diameter (m).

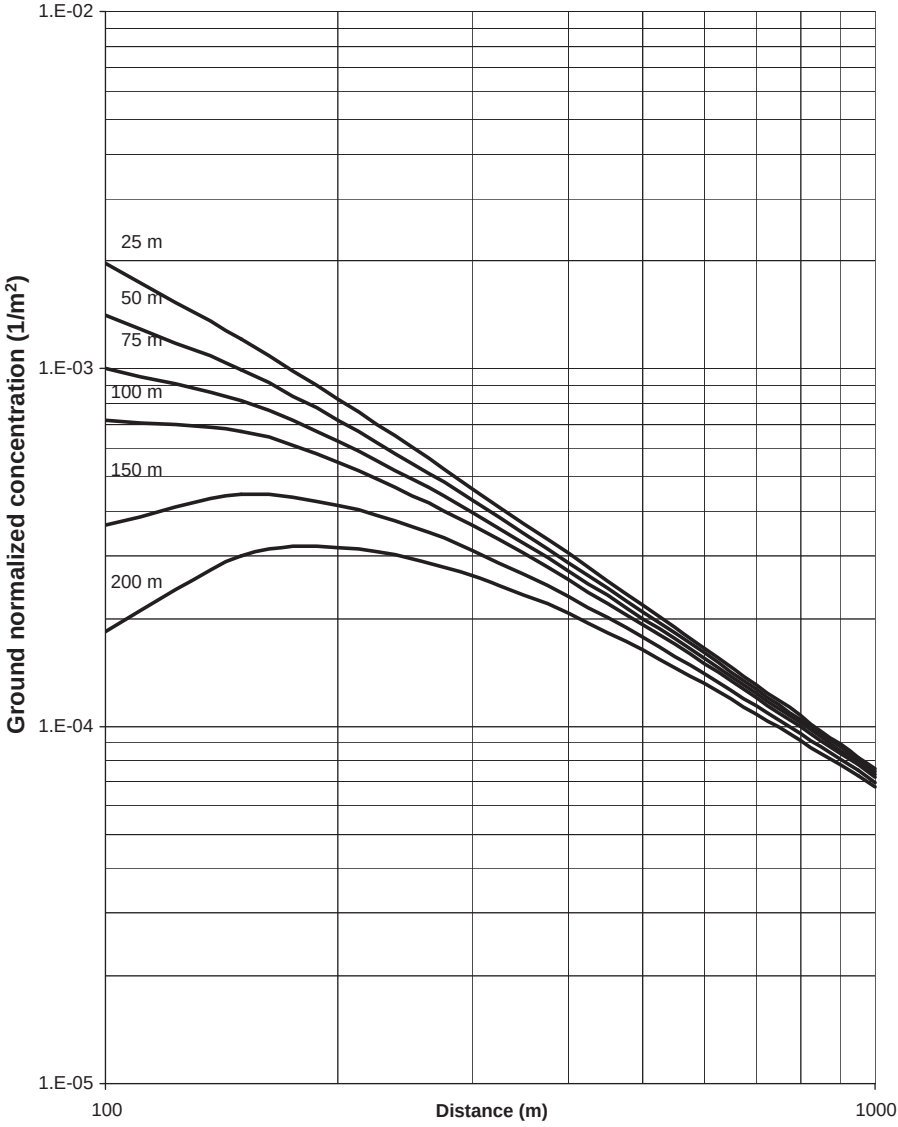


FIG. 8. Normalized surface concentration for several release heights for atmospheric stability category D.

Using Eq. (27) for efflux velocity, this becomes:

$$\Delta h_d = 3.822(R_{\text{efflux}}/Du) \tag{34}$$

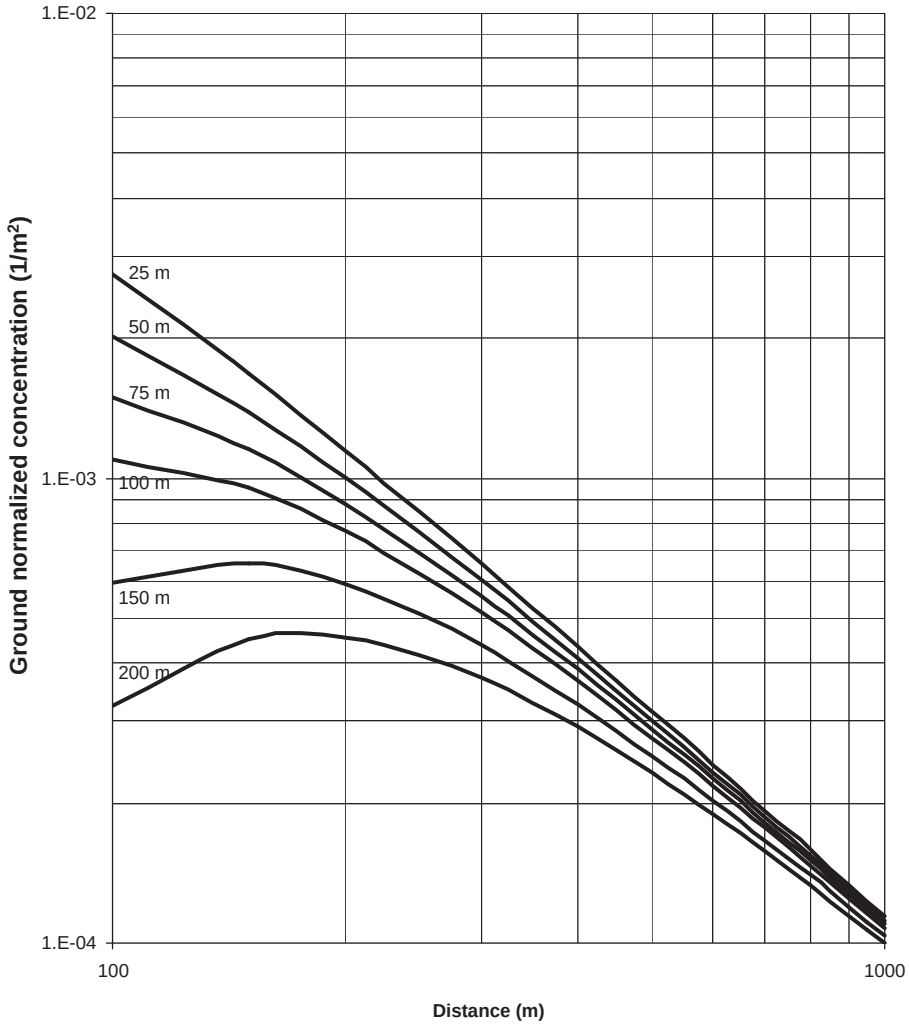


FIG. 9. Normalized surface concentration for several release heights for atmospheric stability category E.

(2) For neutral conditions corresponding to:

(a) Atmospheric stability categories A, B, C and D:

$$\Delta h_d = 3w_0 D(u)^{-1} \tag{35}$$

Using Eq. (27) for efflux velocity, this becomes:

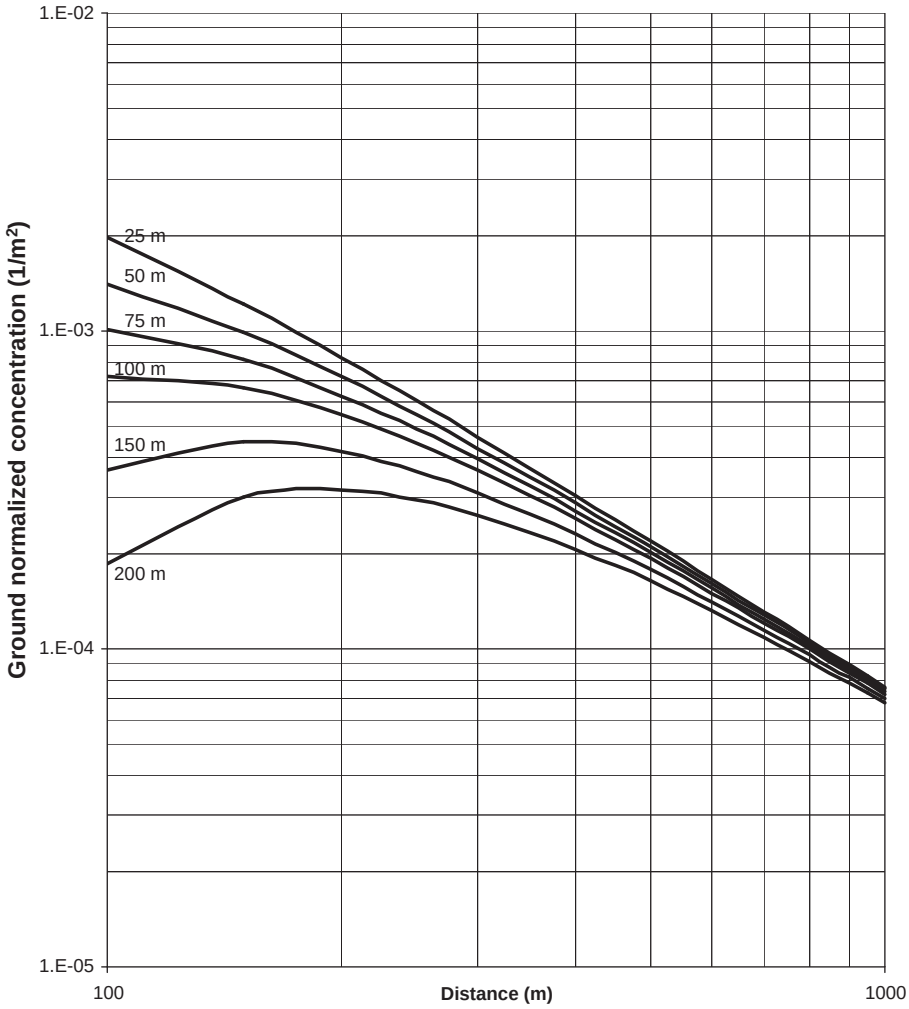


FIG. 10. Normalized surface concentration for several release heights for atmospheric stability category F.

$$\Delta h_d = 3.822(R_{\text{efflux}}/Du) \tag{36}$$

$$\Delta h_d = 1.44D[w_0(u)^{-1}]^{2/3}(x/D)^{1/3} - C \tag{37}$$

where x is the downwind (radial) distance (m) and C is the correction for plume washout for $w_0 < 1.5u$:

TABLE 34. DEFINITION OF PASQUILL ATMOSPHERIC STABILITY CATEGORIES [101]

Surface wind speed (at 10 m) (m/s)	Daytime insolation			Night-time conditions	
	Strong	Moderate	Slight	>3/8 cloudiness[a]	≤3/8 cloudiness
< 2	A	A–B	B		
2–3	A–B	B	C	E	F
3–5	B	B–C	C	D	E
5–6	C	C–D	D	D	D
>6	C	D	D	D	D

Note: Stability category characteristics are as follows: A: extremely unstable; B: moderately unstable; C: slightly unstable; D: neutral[b]; E: slightly stable; F: moderately stable.

[a] The degree of cloudiness is defined as that fraction of the sky above the local apparent horizon that is covered by clouds.

[b] Applicable to heavily overcast day or night conditions.

TABLE 35. DIFFERENT METHODS OF DEFINING ATMOSPHERIC STABILITY CATEGORIES

Pasquill	Turner (STAR code) [105]	σ_θ (deg) [106]	$\Delta T/\Delta z$ (°C/100 m) [107]	Golder (Ri) [108]
A	1	25	<–1.9	<–3.5
B	2	20	–1.9 to –1.7	–3.5 to –0.75
C	3	15	–1.7 to –1.5	–0.75 to –0.1
D	4	10	–1.5 to –0.5	–0.1 to 0.15
E	5	5	–0.5 to 1.5	0.15 to 0.75
F	6	2.5	1.5 to 4.0	0.75 to 3.5
G[a]	7	1.7	>4.0	>3.5

[a] The original Pasquill category F was split into two categories to allow for extremely stable conditions.

$$C = 3[1.5 - w_0(u)^{-1}]\, D_E$$

where D_E is the external stack diameter (m).
The lowest value obtained from Eqs (33)–(37) is used as the most probable for buoyant plume rise for neutral conditions.

(b) Atmospheric stability categories E and F (and G):

$$\Delta h_d = 1.44D\,[w_0(u)^{-1}]^{2/3}(x/D)^{1/3} - C \quad (38)$$

$$\Delta h_d = 4(F_m/S)^{1/4} \quad (39)$$

$$\Delta h_d = 1.5(S)^{-1/6}[F_m(u)^{-1}]^{1/3} \quad (40)$$

where the flux parameter is:

$$F_m = (w_0 D/2)^2 = (0.637 R_{\text{efflux}}/D)^2$$

and the stability parameter is:

$$S = \frac{g}{T_0} 9.8 \times 10^3 \left(\frac{\partial T}{\partial z}\right) = 8.7 \times 10^{-4} \text{ for atmospheric stability category E}$$

$$= 1.8 \times 10^{-3} \text{ for atmospheric stability category F.}$$

In this case:

$$\Delta h_d = 4.85[F_m(u)^{-1}]^{1/3} \text{ for atmospheric stability category E} \quad (41)$$

$$\Delta h_d = 4.32[F_m(u)^{-1}]^{1/3} \text{ for atmospheric stability category F} \quad (42)$$

The lowest value obtained from Eqs (38)–(42) is used as the most probable for buoyant plume rise for neutral conditions.

The buoyant plume rise is terminated when either of the following conditions occurs:

— When $H = L$, where H is the height of the plume centre line and L is the mixing height (height of the capping inversion layer);
— When 1 h has elapsed since the release of the plume segment began.

The height of the mixing layer can be calculated using complicated formulas and depends on the atmospheric stability category. Alternatively, the values suggested in Ref. [94] can be used (see Table 36).

After evaluation of the normalized surface concentration ($\chi u/Q$), the next step is to determine the activity Q released into the atmosphere. This value must be corrected for disintegration, buildup and deposition. In this case:

$$Q = Q_0 e^{-\lambda t}(DF) \tag{43}$$

where Q_0 is the total initial released activity, λ is the decay constant, t is the time elapsed since the release of contaminants and DF is the depletion factor accounting for the ground deposition of contaminants. The values of the decay constant and decay correction $e^{-\lambda t}$ for some of the most common isotopes are presented in Table 37.

The correction for deposition must be calculated separately for dry and wet deposition [96]. For dry deposition:

$$(DF)_d = \exp\left[-\left(\frac{2}{\pi}\right)^{1/2} \frac{v_{dL}}{u} \int_0^x \frac{dx'}{\sigma_z(x') \exp\left(\frac{H^2}{2\sigma_z^2(x')}\right)}\right] \tag{44}$$

where v_{dL} is the average deposition velocity during the plume passage.

TABLE 36. HEIGHT OF MIXING LAYER (L) FOR DIFFERENT ATMOSPHERIC STABILITY CATEGORIES

Stability category	L (m)
A	1500
B	1500
C	1000
D	500
E	200
F	200

TABLE 37. DECAY CONSTANT AND DECAY CORRECTION VALUES FOR SELECTED ISOTOPES

Isotope	Decay constant λ (s^{-1})	Decay correction $e^{-\lambda t}$ at different times (h) after release						
		0.5	1	2	3	6	12	24
I-131	9.94E-07	9.98E-01	9.96E-01	9.93E-01	9.89E-01	9.79E-01	9.58E-01	9.18E-01
I-132	8.53E-05	8.58E-01	7.36E-01	5.41E-01	3.98E-01	1.58E-01	2.51E-02	6.30E-04
I-133	9.28E-06	9.83E-01	9.67E-01	9.35E-01	9.05E-01	8.18E-01	6.70E-01	4.49E-01
I-134	2.20E-04	6.73E-01	4.53E-01	2.05E-01	9.29E-02	8.63E-03	7.46E-05	5.56E-09
I-135	2.89E-05	9.49E-01	9.01E-01	8.12E-01	7.32E-01	5.36E-01	2.87E-01	8.23E-02
Kr-83m	1.02E-04	8.32E-01	6.93E-01	4.80E-01	3.32E-01	1.10E-01	1.22E-02	1.49E-04
Kr-85m	4.42E-05	9.24E-01	8.53E-01	7.27E-01	6.20E-01	3.85E-01	1.48E-01	2.20E-02
Kr-85	2.13E-09	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00	1.00E+00
Kr-87	1.49E-04	7.65E-01	5.85E-01	3.42E-01	2.00E-01	4.00E-02	1.60E-03	2.56E-06
Kr-88	6.94E-05	8.83E-01	7.79E-01	6.07E-01	4.73E-01	2.23E-01	4.99E-02	2.49E-03
Xe-131m	6.69E-07	9.99E-01	9.98E-01	9.95E-01	9.93E-01	9.86E-01	9.72E-01	9.44E-01
Xe-133m	3.50E-06	9.94E-01	9.87E-01	9.75E-01	9.63E-01	9.27E-01	8.60E-01	7.39E-01
Xe-133	1.53E-06	9.97E-01	9.95E-01	9.89E-01	9.84E-01	9.67E-01	9.36E-01	8.76E-01
Xe-135m	7.42E-04	2.63E-01	6.92E-02	4.78E-03	3.31E-04	1.10E-07	1.20E-14	1.44E-28
Xe-135	2.12E-05	9.63E-01	9.27E-01	8.58E-01	7.95E-01	6.33E-01	4.00E-01	1.60E-01
Xe-138	6.81E-04	2.94E-01	8.62E-02	7.42E-03	6.40E-04	4.09E-07	1.67E-13	2.80E-26

For wet deposition:

$$(DF)_w = \exp(-\Lambda_L t_1) \quad (45)$$

where t_1 is the time for the cloud scavenging effect (s) and Λ_L is the scavenging coefficient (s^{-1}).

Values for v_{dL} and Λ_L are presented in Tables 38 and 39, respectively.

The values of the dry depletion factor for each atmospheric stability category and for several release heights and downwind distances from the release points can be approximated to 0.99 for the following conditions:

— Deposition velocity: 0.07×10^{-2}, surface type: soil;
— Deposition velocity: 0.06×10^{-2}, surface type: soil;
— Deposition velocity: 0.03×10^{-2}, surface type: soil.

TABLE 38. VALUES OF DRY DEPOSITION VELOCITY FOR ALL ATMOSPHERIC STABILITY CONDITIONS

Element	Surface type				
	Water	Soil	Snow	Grass	Forest
Iodine					
v_{dL} [a]	0.2E-02	0.07E-02	0.07E-02	0.2E-02	1.0E-02
v_{dH} [b]	2.0E-02	1.00E-02	0.70E-02	3.0E-02	10.0E-02
Ruthenium					
v_{dL}	0.2E-02	0.06E-02	0.2E-02	0.1E-02	0.5E-02
v_{dH}	3.0E-02	0.30E-02	1.0E-02	1.0E-02	5.0E-02
Caesium					
v_{dL}	0.1E-02	0.03E-02	0.1E-02	0.07E-02	0.4E-02
v_{dH}	1.0E-02	0.10E-02	0.3E-02	0.30E-02	2.0E-02
Other					
v_{dL}	0.2E-02	0.2E-02	0.2E-02	0.2E-02	1.0E-02
v_{dH}	3.0E-02	3.0E-02	3.0E-02	3.0E-02	10.0E-02

[a] v_{dL}: minimum dry deposition velocity (used for estimation of plume depletion).
[b] v_{dH}: maximum dry deposition velocity (used for estimation of radioactive deposition on soil).

Because this is intended to be only a simplified calculation, the contribution of any daughter from other isotopes is not considered; moreover, if calculations are performed for long lived radionuclides, the exponential factors in Eqs (41) and (42) are not considered.

Once the normalized surface concentration has been evaluated, and the total activity released has been evaluated and corrected for disintegration and buildup, the doses for each organ of interest and for each pathway can be calculated.

Any person immersed in a radioactive cloud will receive a certain cloud shine external dose. The dose to organ *o* received by an individual immersed in the plume can be calculated by [96]:

$$D_{cld,o} = F_{Wext} \sum \chi_k \left(DCF\right)_{cld,o,k} \tag{46}$$

where

$D_{cld,o}$ is the gamma dose to organ *o*, due to immersion in the cloud (Sv);
F_{Wext} is the shielding factor accounting for time spent inside the buildings;
χ_k is the time integrated concentration for isotope *k* (Bq·s·m^{-3});
$(DCF)_{cld,o,k}$ is the dose conversion factor for isotope *k* and organ *o* for immersion in the cloud (Sv/(Bq·s·m^{-3})).

TABLE 39. VALUES OF SCAVENGING COEFFICIENTS FOR WET DEPOSITION

Element	Rain (mm/h)				Snow (mm/h)			
^{3}H and I	0.5	1	3	5	0.5	1	3	5
Λ_L[a]	5E-06	1E-05	2E-05	3E-05	≤1E-07	1E-07	2E-07	3E-07
Λ_H[b]	1E-04	2E-04	4E-04	6E-04	2E-07	4E-07	8E-07	1E-06
Other								
Λ_L	1E-05	2E-05	3E-05	5E-05	3E-04	5E-04	8E-04	1E-03
Λ_H	2E-04	3E-04	7E-04	1E-03	1E-02	2E-02	4E-02	5E-02

[a] Λ_L: minimum scavenging coefficient (used for estimation of plume depletion).
[b] Λ_H: maximum scavenging coefficient (used for estimation of radioactive deposition on soil).

Considering a single isotope, taking into account the definition of the normalized surface concentration, the corrected amount of radioactive material released and a shielding factor equal to 1, results in the following:

$$D_{cld,o} = (\chi u/Q)(Q/u)(DF)\exp(-\lambda t)(DCF)_{cld,o,k} = (\chi u/Q)(Q/u)(DCF)_{cld,o,k} \quad (47)$$

The values of the dose conversion factor for several isotopes and organs are shown in Table 40.

The dose due to inhalation of radioactive materials can be expressed as:

$$D_{Inh,o} = BF_{F,Inh}\sum_{k}\chi_k(DCF)_{Inh,o,k} \quad (48)$$

where

$D_{Inh,o}$ is the gamma dose to organ o due to inhalation of radioactive material (Sv);

$F_{F,Inh}$ is the shielding factor accounting for time spent inside the buildings;

TABLE 40. DOSE CONVERSION FACTORS FOR IMMERSION IN CLOUD (Sv/(Bq·s^{-1}·m^{-3}))

Isotope	Thyroid	Effective dose
Kr-88	1.10^{-13}	1.10^{-13}
Ru-103	2.10^{-14}	2.10^{-14}
Ru-106	1.10^{-14}	9.10^{-15}
Te-132	1.10^{-14}	9.10^{-15}
I-131	2.10^{-14}	1.10^{-14}
I-135	9.10^{-14}	7.10^{-14}
Xe-133	1.10^{-14}	1.10^{-15}
Xe-135	1.10^{-14}	1.10^{-14}
Cs-134	8.10^{-14}	7.10^{-14}
Cs-135	2.10^{-14}	2.10^{-14}
Cs-137	2.10^{-14}	2.10^{-14}

B is the breathing rate (m^3/s);

χ_k is the time integrated concentration for isotope k ($Bq \cdot s \cdot m^{-3}$);

$(DCF)_{Inh,o,k}$ is the dose conversion factor for isotope k and organ o for inhalation.

Considering a single isotope, taking into account the definition of the normalized surface concentration, the corrected activity released and a shielding factor equal to 1, results in the following:

$$D_{Inh,o} = B(\chi u/Q)(Q/u)(DF)\exp(-\lambda t)(DCF)_{Inh,o,k}$$

$$= B(\chi u/Q)(Q/u)(DCF)_{Inh,o,k} \quad (49)$$

The values of the dose conversion factor for several isotopes and organs are shown in Table 41.

VII.4.2. Calculation method based on step-by-step methodology

To provide an example of how the above formulas, nomograms and tables can be used for a rapid dose calculation based on the step-by-step methodology presented in Section VII.1, the scenario considered above (computerized calculation of consequences) is used, and the same steps are followed as in the

TABLE 41. DOSE CONVERSION FACTORS FOR INHALATION OF RADIOACTIVE MATERIAL (Bq/Sv): ADULTS [102]

Isotope	Thyroid	Effective dose
Sr-90	2.10^{-9}	4.10^{-8}
Ru-103	2.10^{-10}	1.10^{-9}
Ru-106	4.10^{-9}	3.10^{-8}
Te-132	5.10^{-8}	3.10^{-9}
I-131	2.10^{-7}	1.10^{-8}
I-135	2.10^{-9}	4.10^{-10}
Cs-134	1.10^{-8}	1.10^{-8}
Cs-135	1.10^{-8}	4.10^{-7}
Cs-137	8.10^{-9}	8.10^{-9}

methodology used for consequence evaluation. The evaluation is performed for the ^{135}I isotope only.

Step 1: Define the accident scenarios. Determine which scenarios are DBAs and which are BDBAs. Select scenarios that are credible for analysis.

The accident is assumed to occur during the daytime with slight insolation and a wind speed of 5 m/s. From Table 34 the atmospheric stability category is category D, and from Table 36 the height of the mixing layer (L) is 500 m.

In this hypothetical severe accident scenario, it is assumed that a large part of the reactor hall roof falls or that a heavy object is dropped from the crane hook onto the 14 MW TRIGA SSR core, resulting in mechanical damage to it. It is also assumed that no core melting occurs, with only fuel cladding rupture being involved for several 25 pin fuel bundles.

Based on this information, the initial conditions can be summarized as in Table 42.

Step 2: Define the radioisotope content of the reactor core for burnup conditions that match and bound the scenarios selected.

Step 3: For each scenario selected, calculate the reactor time–evolution history.

The core has operated discontinuously for a total of 1780 MW-d. Based on the power operation history and the composition of the TRIGA fuel, input for the ORIGEN computer code is constructed to evaluate the core inventory.

The result of the ORIGEN calculation of the core inventory of ^{135}I is:

$$Q_{\text{core}} = 2.77\text{E+16 Bq } (= 7.51\text{E+05 Ci}) \quad (50)$$

TABLE 42. SUMMARY OF INITIAL CONDITIONS

	Value	Determined by
Wind speed	u = 5 m/s	Assumption
Daytime insolation	Slight	Assumption
Atmospheric stability category	D	Table 8
Height of the mixing layer	L = 500 m	Table 10

Step 4: Determine the kind and extent of fuel damage to permit bounding of the amount of fission products, actinides and other radioisotopes released to the environment.

It is assumed that no core melting occurs, with only fuel cladding rupture being involved for several 25 pin fuel bundles. The affected fraction of the core is 45%. In this case, the available ^{135}I concentration for release is:

$$Q'_{\text{core}} = 1.25\text{E+16 Bq } (= 3.38\text{E+05 Ci}) \tag{51}$$

Step 5: If there is fuel damage and radiation is released from the fuel, determine the release pathways and amounts to various key points.

For this accident scenario, the quantities of fission product gases released are great enough to be detected in the reactor hall air; however, the emergency ventilation system fails, and thus the normal ventilation system remains operative. The removal of halogens by activated charcoal traps can be easily accomplished with efficiencies greater than 99.0%, but in this case it is assumed that no noble gases are removed by the traps.

A 1 h stack release and a point 250 m from the stack are assumed for the calculations. The stack height is 60 m, and the internal stack diameter is 6 m.

Some parameters important to atmospheric diffusion can now be calculated.

Based on Table 42, and Eq. (34), the buoyant plume rise is:

$$\Delta h_d = 3.822(\mathrm{R}_{\text{efflux}}/Du) = 3.822\left(\frac{6.7667\left(\frac{\mathrm{m}^3}{\mathrm{s}}\right)}{6\ \mathrm{m}\times 5\left(\frac{\mathrm{m}}{\mathrm{s}}\right)}\right) = 0.86\ \mathrm{m}$$

Using the stack height, the effective release height can now be approximated; based on Eq. (32), H = 60 m.

The calculated values are summarized in Table 43.

With these values calculated, only the values of the normalized surface concentration, the total activity released corrected for deposition, and the buildup and dose conversion factor for ^{135}I are needed to calculate the dose to the public.

TABLE 43. SUMMARY OF CALCULATED VALUES

	Value	Determined by
Stack height	H = 5 m	Assumption (reactor design)
Internal stack diameter	D = 6 m	Assumption (reactor design)
Downwind distance from stack	X = 250 m	Assumption
Ventilation rate (exhaust rate from stack)	24 360 m^3/h = 6.7667 m^3/s	Assumption (reactor design and accident scenario)
Exposure time	1 h	Assumption
Breathing rate	B = 3.30 E-04 m^3/s	See Section 4 of main text
Effective release height	H = 60 m	Calculation formulas (31) and (33), and Table 42

From Fig. 8, the normalized surface concentration for atmospheric stability category D and a release height of 60 m at a downwind distance 250 m from the stack is:

$$(\chi u/Q) = 5.0\text{E-}04\ (\text{m}^{-2}) \tag{52}$$

Based on Eq. (27) and Table 27, the concentration of the ^{135}I isotope released to the atmosphere is:

$$\begin{aligned} Q_0 &= Q'_{\text{core}} w_i = Q'_{\text{core}} \left(\frac{p}{N}\right) e_i f_i g_i \\ &= 1.2504\text{E+}16\ \text{Bq} \times (2/29) \times 0.25 \times 0.109 \\ &= 2.35\text{E+}13\ \text{Bq} \end{aligned} \tag{53}$$

Based on Eq. (43), on Table 37 for decay and buildup correction and on Table 40 for plume dry depletion factor for atmospheric stability category D, the total released activity Q corrected for disintegration, buildup and deposition is:

$$Q = Q_0 e^{-\lambda t}(DF) = (2.35\text{E+}13\ \text{Bq}) \times (9.01\text{E-}01) \times 0.993 = 2.1025\text{E+}13\ \text{Bq}$$

$$= 2.10\text{E+}13\ \text{Bq} \tag{54}$$

Finally, the dose to a person 250 m from the stack, 1 h after immersion in the cloud can be calculated based on Eq. (47) and the value of the dose conversion factor from Table 42.

$$D_{cld,eff} = (\chi u/Q)(Q/u)(DCF)^{135}_{\text{cld},eff,I}$$

$$= (5.0\text{E-}04\ (\text{m}^{-2})) \times (2.1025\text{E+}13\ (\text{Bq})/5\ (\text{m/s})) \times 7.908\text{E-}14\ [\text{Sv}/(\text{Bq}\cdot\text{s}\cdot\text{m}^{-3})]$$

$$= 1.66\text{E-}04\ \text{Sv} = 0.166\ \text{mSv} \quad (55)$$

The dose resulting from inhalation of radioactive materials based on Eq. (48) and the value of the dose conversion factor from Table 43 is:

$$D_{Inh,eff} = B(\chi u/Q)(Q/u)(DCF)^{135}_{Inh,eff,I}$$

$$= 3.30\text{E-}04\ (\text{m}^3/\text{s}) \times (5.0\text{E-}04\ (\text{m}^{-2})) \times (2.1025\text{E+}13\ (\text{Bq})/5\ (\text{m/s})) \times 4.6\text{E-}10\ (\text{Bq/Sv})$$

$$= 3.19\text{E-}04\ \text{Sv} = 0.3192\ \text{mSv} \quad (56)$$

If we consider only cloud immersion and inhalation as pathways for early exposure, the total dose received by a person 250 m downwind of the stack 1 h post-exposure is:

$$D_{eff} = D_{cld,eff} + D_{Inh,eff}$$

$$= 0.4852\ \text{mSv} \quad (57)$$

Of course, this dose calculated manually will differ from the dose calculated by computer code. Manual calculation of the dose received by a person in a certain location following a radioactive release to the air is a rapid calculation method, but the result obtained is only an estimated value of the dose. Plume travel can be described in several ways, depending on the degree of complexity of the model used. Plume rise and plume travel (including reflections on the mixing layer) are complex phenomena and are not presented here. Also taken into account is the fact that the sigma dispersion parameters, and hence the dilution factor and time integrated concentration, strongly depend on the site location; here, those suggested in Refs [10] and [92] are used. Also, some shielding factors were not considered (or were considered to be equal to one). The purpose of this calculation is to provide an example of how basic dose calculations can be performed manually.

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Annex I

TYPICAL RECOMMENDATIONS FOR ESTIMATING SOURCE TERMS

Typical recommendations for estimating source terms are given below for a release from the fuel directly to the air.

One hundred per cent of the equilibrium radioactive noble gas inventory developed from maximum or full power operation is immediately available for leakage to the reactor building in direct proportion to the percentage of failure of the reactor core, a fuel element or a radioactive experiment.

Twenty-five per cent of the equilibrium radioactive iodine inventory of a fuel element in a radioactive experiment, developed from maximum or full power operation of the core, is immediately available for leakage to the reactor building in direct proportion to the percentage of failure. Ninety-one per cent of this 25% is in the form of elemental iodine, 5% is in the form of particulate iodine and 4% is in the form of organic iodine. The effect of radioactive decay during hold-up in the building or containment can be taken into account.

The reduction of the amount of radioactive material available for release to the environment by engineered safety features may be taken into account, but is evaluated on a case-by-case basis for each facility.

Event trees can be used to assess the performance of confinement engineered safety features for all core damage scenarios identified in the accident analysis. Typical headings include: containment sealing (including shutdown of normal ventilation), emergency ventilation and recirculation air cleaning. Success or failure states of these systems at nodes in the trees will lead to a set of final release categories, representing stack or ground releases, effectiveness of iodine removal, etc.

For each of these categories, the corresponding radionuclide releases may be determined and grouped as noble gases, iodines and volatiles, semi-volatiles or particulates, where appropriate.

Annex II

TYPICAL FISSION PRODUCT INVENTORIES FOR RESEARCH REACTORS

In this annex, the equilibrium fission product inventories are presented for the 20 MW SAFARI and the 35 MW SILOE research reactors. Activity values for these reactors are given in Tables II–1 and II–2, respectively.

For the SAFARI reactor, the fission product inventory was obtained from ORIGEN, with the fuel elements burning at an average thermal flux of 9.95×10^{13} $n \cdot cm^{-2} \cdot s^{-1}$ (corresponding to a reactor power of 20 MW).

Each fuel element is burned for a total of 11 cycles of 21.7 d each. The total nuclide inventory is then determined (assuming that three new fuel elements are loaded at each cycle) in the following manner:

— Three burned for 1 cycle;
— Three burned for 2 cycles;
— Three burned for 3 cycles;
— One burned for 11 cycles.

The control follower fuel is treated in a similar manner. The 28 fuel and 6 control follower fuel element inventories are then added to give a total inventory at the end of the equilibrium cycle.

For the SILOE reactor, the equilibrium fission product activities correspond to five operating cycles of 21 d each. The successive operating cycles are separated by a shutdown period of 7 d.

TABLE II–1. FISSION PRODUCT INVENTORY FOR THE SAFARI REACTOR (20 MW)

Isotope	Activity (Bq)
Kr-83m	3.29E+15
Kr-85	5.72E+13
Kr-85m	8.14E+15
Kr-87	1.58E+16
Kr-88	2.23E+16
Kr-89	2.90E+16
Kr-90	2.94E+16
Xe-131m	2.05E+14
Xe-133m	1.25E+15
Xe-133	4.32E+16
Xe-135m	7.13E+15
Xe-135	3.34E+15
Xe-138	3.83E+16
Xe-138	3.88E+16
Xe-139	3.20E+16
Br-83	3.30E+15
Br-84	5.99E+15
Br-85	8.02E+15
Br-87	1.36E+16
I-129	9.10E+07
I-131	1.83E+16
I-132	2.75E+16
I-133	4.20E+16
I-134	4.72E+17
I-135	3.95E+16
I-136	1.88E+16
Cs-134	3.60E+14
Cs-136	1.15E+14
Cs-137	5.63E+14
Te-125m	3.10E+12
Te-127m	7.65E+13
Te-129m	5.78E+14
Te-131m	2.28E+15
Te-131	1.58E+16

TABLE II–1. FISSION PRODUCT INVENTORY FOR THE SAFARI REACTOR (20 MW) (cont.)

Isotope	Activity (Bq)
Te-132	2.70E+16
Te-133m	1.90E+16
Te-134	4.19E+16
Te-135	2.03E+16
Ru-103	1.88E+16
Ru-106	7.41E+16
Tc-99	6.20E+10
Tc-99m	3.57E+16
Mo-99	3.89E+16
Sr-89	2.74E+16
Sr-90	4.53E+14
Sr-91	3.68E+16
Ba-140	4.08E+16
Sb-125	3.88E+13
Sm-151	5.13E+12
Sm-153	3.69E+15
Sm-156	8.74E+13
Pm-147	9.73E+14
Pm-149	1.17E+16
Pr-143	3.81E+16
Pr-145	2.40E+16
Y-90	4.77E+14
Y-91	3.25E+16
Y-92	3.70E+16
Nd-147	1.48E+16
La-140	4.14E+16
Ce-141	3.58E+16
Ce-143	3.74E+16
Ce-144	1.24E+16
Zr-95	3.42E+16
Nb-95m	2.38E+14
Nb-95	2.81E+16

TABLE II–2. FISSION PRODUCT INVENTORY IN THE CORE OF THE SILOE REACTOR (35 MW)

Isotope	Activity (Bq)
Br-83	6.08E+15
Kr-83m	5.73E+15
Se-84	1.08E+16
Br-84m	3.84E+14
Br-84	1.09E+16
Br-85	1.47E+16
Kr-85m	1.40E+16
Kr-85	3.81E+13
Br-86	2.12E+16
Br-87	2.42E+16
Kr-87	2.82E+16
Br-88	2.22E+16
Kr-88	3.64E+16
Kr-89	4.75E+16
Kr-90	5.14E+16
Te-125	1.41E+12
Te-127m	9.76E+10
Te-127	1.28E+15
Te-129m	9.66E+14
Te-129	6.56E+15
Xe-129m	8.31E+7
Te-131m	3.56E+15
Te-131	2.96E+16
I-131	2.94E+16
Xe-131m	1.74E+14
Te-132	5.74E+16
I-132m	3.08E+14
I-132	4.80E+16
Sb-133	2.58E+16
Te-133m	3.84E+16
Te-135	3.62E+16

TABLE II–2. FISSION PRODUCT INVENTORY IN THE CORE OF THE SILOE REACTOR (35 MW) (cont.)

Isotope	Activity (Bq)
I-133m	2.06E+15
I-133	1.52E+16
Xe-133m	2.16E+15
Xe-133	6.73E+16
Te-134	7.12E+16
I-134m	6.56E+15
I-134	8.64E+16
Xe-134m	7.30E+14
Cs-134m	7.64E+13
Cs-134	3.28E+13
I-135	6.80E+16
Xe-135m	1.39E+16
Xe-135	6.75E+15
Cs-135	4.72E+8
Te-136	2.12E+16
I-136m	1.95E+16
I-136	3.18E+16
Cs-136m	1.55E+14
Cs-136	2.40E+14
Xe-137	6.39E+16
Cs-137	3.08E+14
Xe-138	6.42E+16
Cs-138m	1.13E+15
Cs-138	6.92E+16
Cs-139	6.92E+16
Cs-140	6.28E+16

Annex III

ASSESSMENT OF CONFINEMENT RESPONSES TO CHALLENGES FROM EXPLOSIVE FUEL–COOLANT INTERACTIONS

III–1. GENERAL

Steam explosion events due to explosive fuel–coolant interactions (FCIs) have occurred at several research reactors. Well-known examples include those at the SL-1, SPERT and BORAX reactors. Most of these reactors employ metallic fuels in which aluminium is a principal component. It is also well-known that molten aluminium mixed with water can result in energetic steam explosions (as in aluminium cast houses). These explosions may be accompanied by chemical reactions between aluminium and water on an explosive timescale (also referred to as ignition), a phenomenon that can increase the resulting energy by close to a factor of 20. Evaluating the onset, propagation, expansion and possible missile formation stages of a steam explosion event (and subsequent challenges to the containment barriers) can be expected to be a complex undertaking. Such analysis can range in sophistication from making bounding estimates to conducting mechanistic evaluations of various stages of the transient. This annex summarizes salient aspects of a comprehensive study [III–1] conducted for the High Flux Isotope Reactor (HFIR), an operating high power density research reactor at Oak Ridge National Laboratory (ORNL) in the United States of America.

A schematic diagram of the HFIR is shown in Fig. III–1. The HFIR is a 100 MW research reactor employing U_3O_8–Al fuel with aluminium cladding in involute plate form [III–1]. The entire core is composed of only two fuel elements with a total of 540 plates (with a plate thickness and coolant gap of ~1.27 mm each) and a total core fuel mass of 100 kg and a volume of ~50 L. As seen in Fig. III–1, the core is enveloped in a beryllium reflector and contained within a steel pressure vessel (2.6 m in diameter). The pressure vessel is located at the bottom of a 4.3 m deep pool. A Level I PRA study conducted for the HFIR concluded that core damage frequency from internal events is dominated by flow blockage events. A large enough flow blockage may cause rapid (<1 s) fuel melting under full power conditions, and therefore a steam explosion could result.

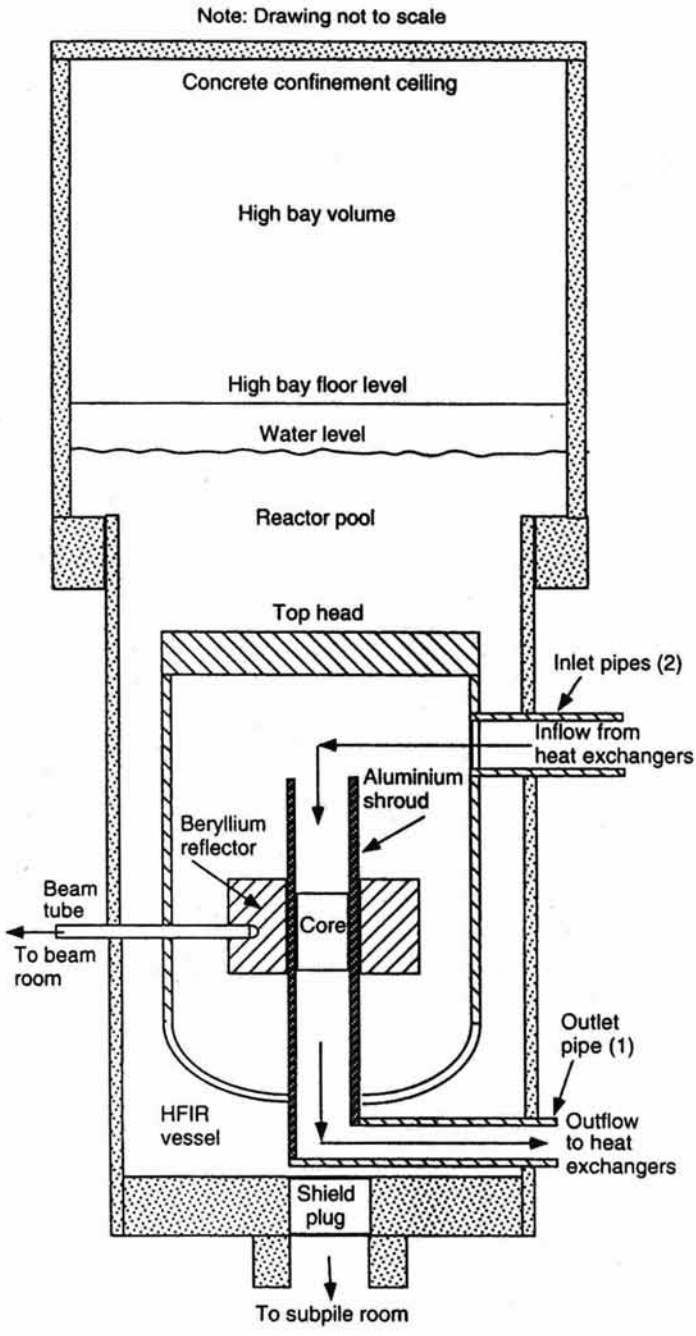

FIG. III–1. Schematic diagram of the HFIR.

III–2. OVERALL MODELLING APPROACH

The overall modelling approach followed is illustrated in Fig. III–2. The methodology follows a stepwise approach. First, information is derived as to what fraction of the HFIR core could realistically melt from possible small and large flow blockages. Thereafter, bounding evaluations are conducted for explosion energetics using thermodynamic models [III–2]. If the resulting loads are acceptable, no more needs to be done. Otherwise, modelling is done to capture the aspect of melt fragmentation and time history of thermal to mechanical energy conversion in the core region. This information is then used to mechanistically model the generation of explosive loads, shock wave generation/propagation and interaction within the HFIR pressure vessel. Dynamic load histories are then coupled with failure envelopes generated from structural dynamic calculations. From these assessments, estimates are generated for the maximum tolerable core melt fractions for which resulting missiles do not breach the confinement walls.

III–3. CORE THERMAL HYDRAULICS AND MELT MASS ESTIMATION

The RELAP5 code [III–3] was used to model HFIR core thermal hydraulics and to estimate the maximum fraction of core melting prior to a scram. This work resulted in an estimate of ~14% of the fuel plates that could melt from flow blockages. A more conservative approach provided estimates of up to 24%.

III–4. VESSEL AND BOLTS FAILURE ANALYSIS

Estimates were derived for generating failure onset figures of merit for the HFIR vessel and top head bolts. These two are described separately.

III–4.1. Vessel failure analysis

Elastic dynamic assessments were conducted using the comprehensive finite element code ADINA [III–4]. ADINA was used to develop a model for the HFIR pressure vessel, after which failure envelope data were developed, as

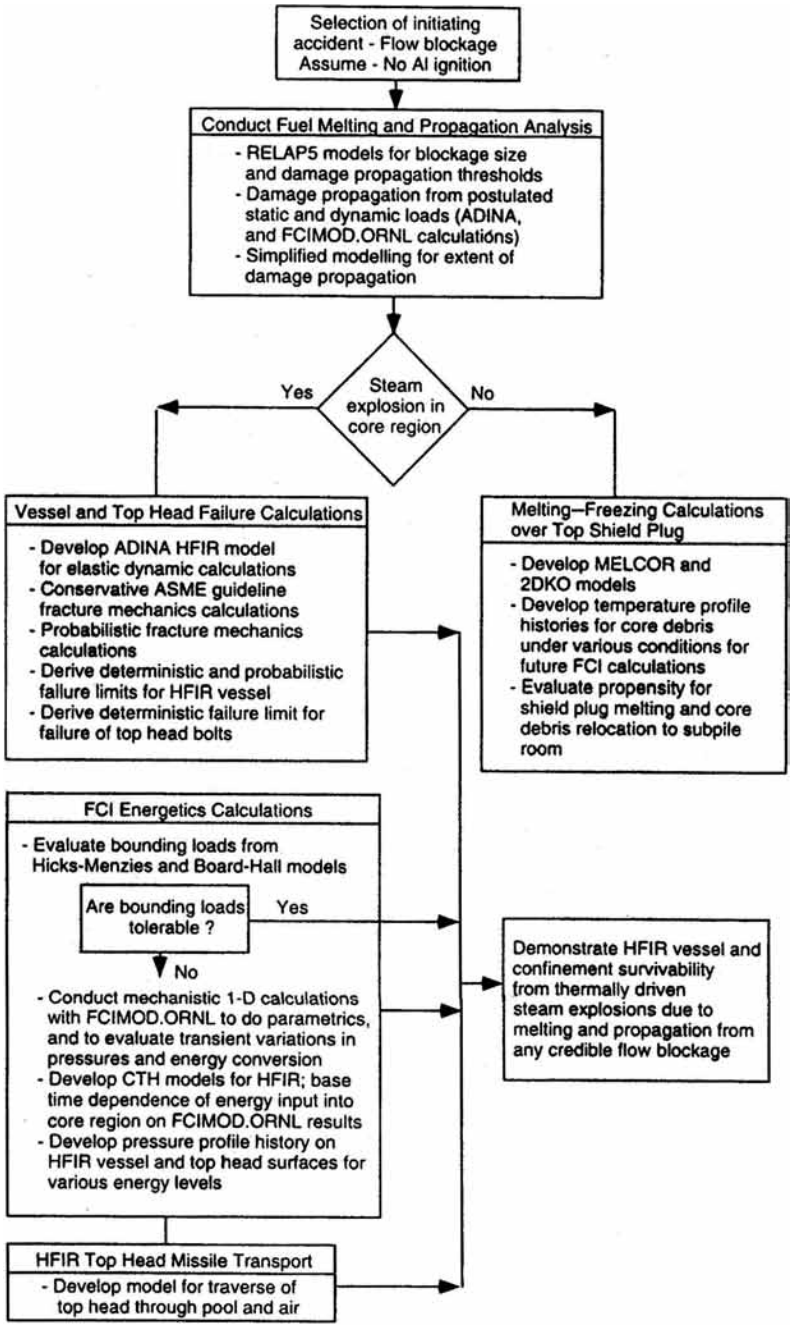

FIG. III–2. Fuel–coolant interaction work elements and framework for the HFIR safety analysis report.

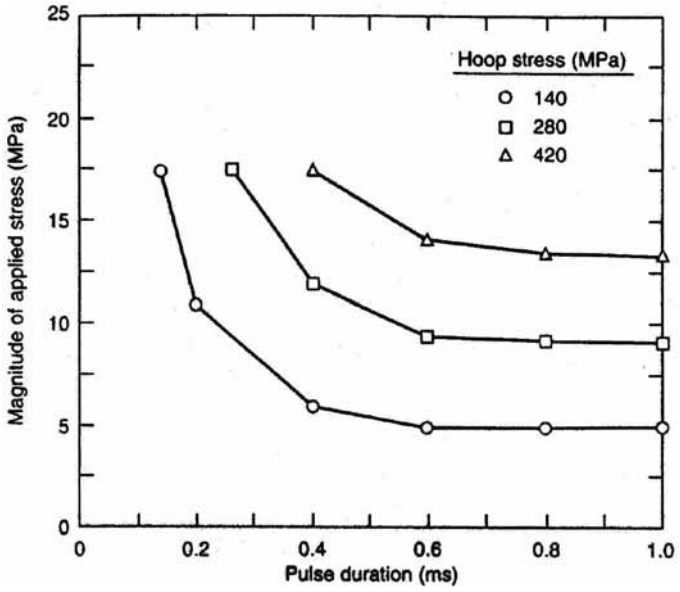

FIG. III–3. Failure envelopes for HFIR pressure vessel wall at midplane.

shown in Fig. III–3. The model provides values of induced hoop stress for various combinations of pressure pulse magnitudes and pulse durations. In addition, a probabilistic fracture mechanics approach was used to evaluate a failure curve, as shown in Fig. III–4.

III–4.2. Top head bolt failure analysis

An analysis was then conducted to estimate the level of loads that would be sufficient to cause the bolts holding the top head of the vessel to fail. Combining results of previous studies and knowledge of the ultimate strength of HFIR bolts (~840 MPa), along with stress amplifications (i.e. from head surface to bolts), it was estimated that an imposed fluid side pressure of above 26 MPa lasting for more than 0.8 ms would be sufficient to cause bolt failure.

III–5. STEAM EXPLOSION ENERGETICS

As mentioned previously, steam explosion energetics calculations were first initiated using conservative thermodynamic models. However, this approach resulted in peak pressurization ranging from 60 to several hundred MPa

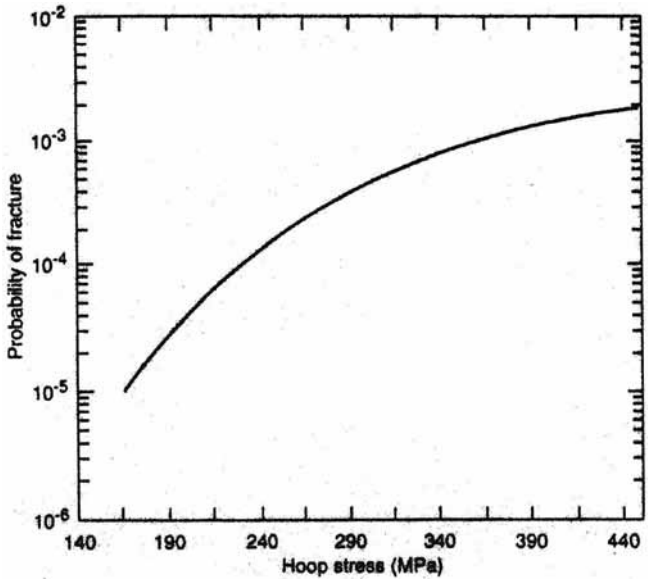

FIG. III–4. Failure envelopes for variation of HFIR vessel failure probability versus hoop stress (0.007 crack/ft^2).

and conversion factors as high as 45%. These estimates were unacceptably conservative. Therefore, mechanistic evaluations were made to estimate the transient variations of fuel fragmentation induced thermal energy deposition into the reactor coolant during steam explosions. This gave rise to energy conversion estimates of ~8%, as shown in Fig. III–5.

This information was then used to initialize the explosion dynamics model using the sophisticated, multiple-material motion and shock physics code CTH [III–5]. A model was built of the HFIR core and pressure vessel, after which multidimensional evaluations were conducted to derive transient pressure field histories at key locations. Sample results are shown in Fig. III–6.

Results of CTH calculations were then utilized to derive information on what core melt fraction induced steam explosions would be sufficiently robust to cause bolt failure and create a containment threatening missile (namely, the top head cover). Results indicated that ~65% of the core would need to melt to cause a containment threatening missile. Since this value is considerably above the maximum value of up to 24% that could possibly melt during flow blockages, it was concluded that the HFIR pressure vessel and top head structure would be able to withstand loads generated by 'thermally' driven (i.e. no Al ignition permitted) steam explosions initiated by any credible flow blockage.

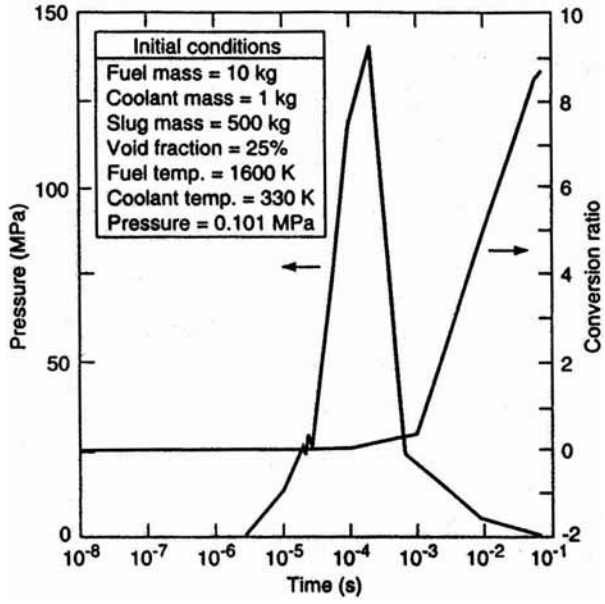

FIG. III–5. Typical variation of pressure and conversion ratio over time.

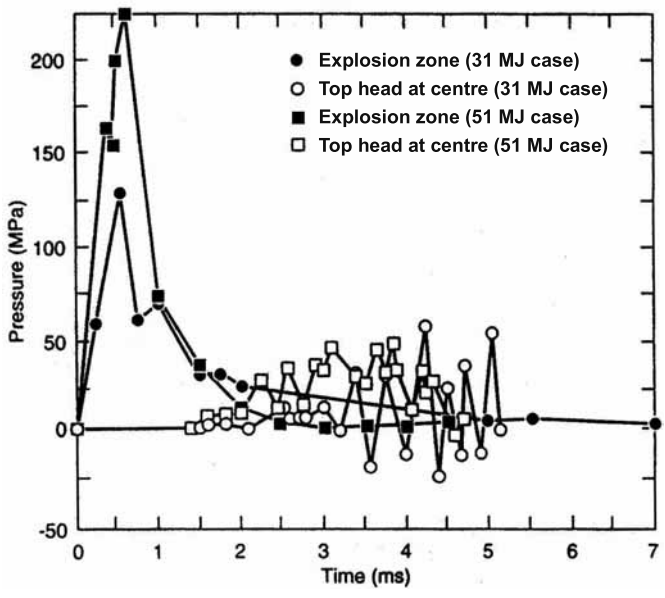

FIG. III–6. Variation of pressure in explosion zone and top head centre lower surface for 31 and 51 MJ energy deposition cases.

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