XA9953383

IAEA-TECDOC-1127

A simplified approach to estimating reference source terms for LWR designs

INTERNATIONAL ATOMIC ENERGY AGENCY IAEA

December 1999

30 - 50

The originating Section of this publication in the IAEA was

Engineering Safety Section
International Atomic Energy Agency
Wagramer Strasse 5
P.O Box 100
A-1400 Vienna, Austria

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A SIMPLIFIED APPROACH TO ESTIMATING
REFERENCE SOURCE TERMS FOR LWR DESIGNS
IAEA, VIENNA, 1999
IAEA-TECDOC-1127
ISSN 1011–4289

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Printed by the IAEA in Austria
December 1999

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FOREWORD

The IAEA has initiated an extensive programme to review and bring up to date the complete set of the Standards (Codes and Guides) of the Nuclear Safety Series (NUSS). The main reason behind this is to make all the Standards consistent with the Safety Fundamentals Safety Series No. 110 "The Safety of Nuclear Installations", which represents the top level publication in the hierarchy of the IAEA Safety Series. The work has begun and is expected to be concluded by the year 2000.

The revision of the standards which address the design process is of particular interest because of its impact on the design of the next generation of reactors. This programme was initiated at the IAEA in 1991 following the recommendations of the General Conference. The first task, preparing safety objectives and principles for future reactors, has been completed and its results published in IAEA-TECDOC-801 "Development of Safety Principles for the Design of Future Nuclear Power Plants".

The substantial innovation proposed in TECDOC-801 is that severe accidents must be explicitly considered in the design of future NPPs to ensure that the impact on individuals and the environment beyond the site fence is limited to an acceptably low level.

Since TECDOC-801 reflects the quite general consensus that severe accidents must be considered explicitly in the design of future plants, the IAEA has decided to devote a significant amount of resources to the identification of safety issues associated with severe accidents which should be addressed in the design of future NPPs.

As limiting the need for off-site countermeasures is an objective in many countries when considering the designs of future NPPs, work has been initiated on the evaluation of the quantity of fission products that is available for leakage from the containment in the case of severe accident sequences that should be used for design purposes. This was considered a necessary input for the design of the containment and its associated systems.

The publication of this IAEA technical document represents the conclusion of a task, initiated in 1996, devoted to the estimation of the radioactive source term in nuclear reactors. It focuses mainly on light water reactors (LWRs).

The IAEA is grateful to the experts who contributed to this publication. The major contributors were M.C. Dutton (NNC Limited, UK), T. Kress (Kress Associates, USA), J. Eyink (Nuclear Power International) and C. Benson (AEA Technology, UK) who provided a comprehensive review. The officer of the IAEA responsible for the TECDOC was M. Gasparini of the Division of Nuclear Installation Safety.

EDITORIAL NOTE

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CONTENTS

1. INTRODUCTION

1.1. BACKGROUND

The amount of fission products that can be released from a nuclear power plant in an accident is a fundamental parameter to estimate the consequences of the accident on individuals and environment.

The regulatory environment and industrial expectations vary in different countries. There is, however, a clear convergence on the need to define a reference quantity, called either "source term" or "in-containment source term".

This source term, which is defined as the timing, fraction, speciation of fission products released to the containment during a severe accident and their behaviour when they are suspended in the containment atmosphere, is intended to be used for:

— assessing the robustness of the containment to contain fission products,

— assessing external consequences to demonstrate compliance with deterministic criteria,

— designing some containment systems,

— defining conditions for the environmental qualification and assessing the survivability of key components,

— providing adequate protection from direct radiation for the operating staff in the control room and in other areas where access is required.

This source term must be such that the regulators and the industry have a high confidence that the containment systems and safety related structures and components, which are designed using this source term, will result in an acceptable level of safety. Thus, it is required to be reasonably representative for all the risk dominant sequences that involve a partial or total melting of the core.

The work that the IAEA is carrying out on source terms is considered to make a valuable contribution to the subject by promoting the exchange of information and the harmonization of views. It is considered that it will significantly contribute to the revision of the IAEA NUSS standards.

1.2. OBJECTIVE

This report focuses on the source term for light water reactors.

For the purposes of this report, the expression 'source term' is defined as follows:

(a) *Source term to the containment* is the magnitude, physical and chemical form and timing of the release of fission products and other aerosols from core materials and concrete to the primary containment atmosphere or to the suppression pool from both in- and ex-vessel sources.

(b) *In-containment source term* is the airborne radioactivity and its physical and chemical form in the atmosphere of the primary containment as a function of time. Thus, the in-containment source term is the radioactivity that is available to be released from the primary containment.

The first of these definitions represents the material that is released into the containment from both the reactor coolant circuit and ex-vessel sources, while the second definition represents the behaviour of that material in the containment.

The most important barrier to the release of fission products to the environment is the containment. To mitigate the consequences of severe accidents, it is important that the containment maintain its capability as an effective barrier and that the combination of the fission product activity in the containment that is available for release (the airborne activity or in-containment source term) and the performance of the containment systems result in acceptable radiological consequences as defined in IAEA-TECDOC-801 for new generation plants.

With respect to releases to the environment, three important issues need to be addressed:

- the source term to containment and the in-containment source term as defined above,
- the performance of the containment in withstanding the severe accident loads and the resultant leakage of fission products,
- the pathway for fission-products to escape from the containment and the associated depletion due to the secondary containment, the filters and the stack.

Only the first is the subject of the present report. The other two (and possibly a fourth which may address the demonstration of compliance with regulatory acceptance criteria and thus deals with dose calculations) will be the subject of future work.

The third aspect is important because European NPPs differ not only in their containment design but also in the extent to which the primary containment is enclosed by adjacent buildings such as a secondary containment and the auxiliary buildings. If the containment fails or leaks, the release of fission products will not be directly into the atmosphere but into the adjacent buildings. Because these buildings are relatively cold and have large surface areas, a significant additional depletion of aerosols and gaseous iodine can be expected, depending on the residence time and, in general, any extract to the environment will be via filters. It is therefore necessary to consider carefully the release paths in order to quantify the release and to assess the accident management procedures, such as activating fans and filtered release paths.

The objectives of the present report are:

(1) To identify the main aspects associated with new generation plant designs which could significantly affect the source term to the containment.

(2) To identify the worldwide methodologies (codes and models) for evaluating the effect of these design aspects on the source term to the containment and the airborne activity in the containment atmosphere.

(3) To identify the major uncertainties which affect the source term to the containment and the airborne activity in the containment atmosphere.

(4) To identify the R&D requirements needed to address these uncertainties for new generation NPPs.

(5) To determine the usefulness of providing guidance on how a reference source term to the containment could be evaluated parametrically based on simple algorithms and, if useful, describe such algorithms as far as practicable.

1.3. SCOPE

The scope of this report is to provide regulatory bodies, utilities and vendors with a simple means for evaluating a source term, both to and in the containment, which provides a reasonably bounding envelope for the risk-dominant sequences that are applicable to the design being considered. This is relevant to both existing and future reactor designs.

Events at shutdown may contribute significantly to the overall core melt frequency. With respect to the release into the containment, they are bounded by the events at power because, in both cases, the temperature excursion is the same, driven by the Zr oxidation rate and not by the decay heat.

This report does not address the release of radioactivity via paths that by-pass the primary containment structure since the consideration of by-pass paths does not affect the design of the primary containment systems. They will, however, affect the design of local containment systems outside the primary and secondary containments and will need to be addressed separately for this purpose. It is anticipated that the designs of future plants will reduce the frequency of by-pass faults involving a core melt to low levels.

It is considered that specifying a single source term for all light water reactors is not appropriate, due to the wide variety of design features, fuels and cores in future designs. However the following are proposed for any individual plant:

- A reference sequence which has the necessary attributes to derive a numerical reference source term;
- the identification and a description of the relevant core degradation mechanisms for reference sequence do not take credit for accident management other than depressurizing the primary circuit before the vessel fails;
- simple algorithms for the evaluation of the melt progression, the quantification of releases to the containment; and
- their speciation and behaviour in the containment for the reference sequence.

As engineering judgement must be exercised due to the complexity of phenomena and all the associated uncertainties, guidance is also provided on the following:

- the remaining uncertainties and sensitivity analyses that are needed to assess the results; and
- the computer codes available which can also be used to predict the in-containment source term for those situations for which simple algorithms are inadequate.

In developing the source term, the following considerations are taken into account:

- the loss of all permanent sources of water to the reactor coolant system (RCS) apart from pressurized vessels, such as the accumulators, is assumed, as otherwise there would not be a core melt;
- future designs are expected to provide a robust and reliable means for depressurizing the reactor coolant system;
- the application of the proposed algorithms should be straightforward.

The proposed reference sequence for deriving the reference source term is a low pressure core melt sequence in which the reactor coolant system is depressurized well before the reactor pressure vessel fails. This approach is consistent with the risk dominant sequences that have been identified in plant assessments [1–5].

The evaluation of the proposed reference source term contains simplifying assumptions. However, these should be modified on the basis of engineering judgement in cases where they would otherwise lead to a less robust design. For example, the derived timing of the blowdown should not be used to justify less reliable or slower reacting containment isolation valves.

2. SOURCE TERM TO THE CONTAINMENT

2.1. FUEL DESIGN AND CORE INVENTORY

Although the quantification of the source term is generally expressed in terms of the fractions of all the relevant radionuclides, the utilization of these fractions for various purposes requires that they be converted into actual quantities using the fission-product and actinide inventories of the core.

These data are not expected to be the same for all plants, even if they have the same design. Design parameters such as the core thermal output, the maximum burnup of the fuel, the type of fuel (e.g. UO_2 or MOX), may have a significant impact on the inventory of all the relevant radionuclides. Thus, in the analysis of accidents for the purposes of designing the containment, a bounding inventory should be evaluated which anticipates the likely changes to the core and fuel design or management over the lifetime of the plant, which may be up to 60 years.

2.2. METHODOLOGY AND CODES FOR DETERMINATION OF THE FISSION PRODUCT AND ACTINIDE INVENTORY

The evaluation of the core inventory is a complex process, which has at least two objectives that are relevant to this report, namely:

- to provide the necessary data for quantifying the source term by using simple algorithms or computer codes (examples of these data are the decay heat or the power distribution in the core);
- to provide the absolute magnitude of the maximum radioactivity which is required that is available in the core to be released to the containment in order to demonstrate compliance with the defined acceptance criteria, for equipment qualification and for demonstrating equipment survivability.

To evaluate these quantities, the assumption is made that the accident occurs at the end of an equilibrium fuel cycle.

Systematic studies are required to define the fuel loading patterns that are compatible with the fuel design limits and, based on these to derive the maximum inventories of the relevant fission products and actinides (i.e. those that will contribute significantly to the decay heat or the environmental consequences). This process is similar to that performed by a fuel designer when optimizing his product.

Although simple algorithms and approximate methods have been used in the past for such evaluations, it is recommended that the inventory be evaluated using a well validated computer code together with a critically assessed and internationally accepted database.

Examples of such codes, which have typically been validated for the core and fuel designs and fuel management strategies are ORIGEN (USA), APOLLO (France) and FISPIN (UK).

2.3. UNCERTAINTIES

In addition to the uncertainties associated with the conditions for which the codes have been validated, there are uncertainties associated with (a) the potential evolution of the design of the core and fuel and the fuel management strategy, and (b) the limited database against which the codes are validated. However, these uncertainties in predicting the fission product inventory are small compared to the uncertainties in estimating the fission product release from the primary circuit and are negligible compared to those associated with evaluating the fission product distribution within the containment.

Some middle term to long term developments in the design of fuel as described in Section 8.3 have the potential for significantly affecting the quantification of the inventory.

Examples of possible developments are:

- increased burnup,
- extended use of MOX fuel,
- extended use of burnable poisons and new burnable poisons,
- new control rod designs,
- new types of fuel, and
- changes to the dimensions and material of the cladding, spacers, etc.

3. THERMAL HYDRAULICS AND CORE DEGRADATION CONSIDERATIONS

Generally, the determination of the source term requires a knowledge of the thermal-hydraulic progression that leads to the coolant boiling off, and the subsequent core heat-up, melting and degradation. For such information, use has traditionally been made of computer codes that model the core and primary circuit as finite elements and include such phenomena as steam/Zr chemical reactions, ZrO_2–uranium dissolution and fission product/aerosol release. These codes have been partially validated using experiments carried out under the relevant conditions, but there are still significant uncertainties.

It is clear that plant and sequence specific source terms could be developed for future plants after their designs have been specified by using such codes to analyse a range of relevant accident sequences. The selection of a bounding case among these could be an appropriate way to proceed to develop design specific source terms provided that sufficient guidance is available on what constitutes a "valid" code along with guidance on how to make the key assumptions that are required for the application of such codes, which sequences need evaluating for the purposes of designing the containment systems, how the results should be interpreted and how sensitivity/uncertainty analyses should be utilized.

While such a procedure should be acceptable, it amounts to a case-by-case determination that is done after the design of the RCS is complete. Guidance would still be required on which sequences are appropriate for use in comparing with regulatory dose criteria, which codes can be certified for such use, and how to quantify the uncertainties and interpret the results. This procedure will give the designer or the regulator little prior guidance as to what is likely to be an acceptable design basis source term.

One of the objectives of this report is to explore the feasibility of developing simple, but robust, algorithms (or an algorithmic methodology) that can capture the essence of the thermal-hydraulic behaviour of future plants related to their important (to source terms) design differences without having to resort to such large complex codes. Such algorithms should be useful to both regulators and designers as they easily accommodate readjustments and different design options. They can make clear the extent of the effects that different design options have on the source term.

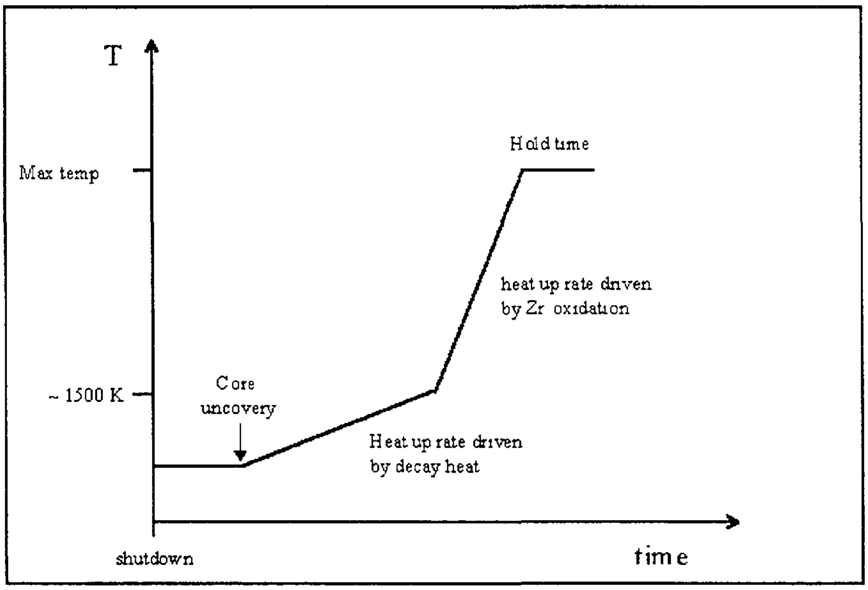
The purpose of developing source term algorithms, as advocated here, is to have an easily applied, consistent and transparent means to give sufficient guidance (quantity and timing of fission product releases) to designers and regulators as to whether or not a specific design is likely to meet regulatory requirements that involve the use of a design basis source term.

A primary requirement for such a design basis source term is that it should be able to result in a containment design that would provide "defence in depth" to compensate for the lack of knowledge on calculating risk or lack of confidence in the ability to terminate accidents before they lead to extensive core damage. Almost by definition, a reference source term must be based on a low probability sequence involving failure of the ECCS (whether active or passive) to terminate the accident sequence and which would allow a major degree of core melt. Thus, the choice of the type of sequence (or sequences) to use to develop representative source terms for design basis is largely a matter of judgement and experience as to what would result in a sufficiently robust containment design.

Experience has shown that low pressure LOCA sequences generally develop faster than high-pressure sequences and will release equally significant amounts of fission products. Since it is expected that future LWRs will utilize a strategy of depressurization for all accident sequences in order to avoid high pressure melt ejection events, it is judged that an appropriate choice for developing a representative design basis source term is a depressurized sequence (to give the fastest timing), coupled with sufficient conservatism in the heat-up transient to give fission product releases that are appropriately large for design basis use. The "appropriateness" would be related to both the probabilities and source term quantities associated with the spectrum of severe accidents sequences. The overall objective, of course, is to end up with a design that leads to an acceptable risk but which also provides a sufficient degree of confidence that any fission products released from the fuel in the unlikely event of an accident will be adequately contained.

3.1. GENERALIZED THERMAL HYDRAULIC ALGORITHM

For the purposes of developing a representative design specific source term algorithm that can capture the effects of important design differences for future plants, it is instructive to postulate a general heat-up pattern for core regions experiencing a typical low pressure boil-off and core uncovery. Experience with the application of core-melt thermal hydraulic codes and with limited in-pile experimental data leads us to postulate the general thermal pattern for a fuel bundle which is shown in the schematic diagram below.



This, of course, is a "stylized" representation that does not recognize several complicated features of core degradation behaviour such as the mechanics of fuel dissolution, relocation and blockage. Nevertheless, it is considered that such a representation provides a simple vehicle that, when applied with the appropriate fission product release modelling, will result in a representative source term that can readily account for the important design differences that might exist in future plants. Such a generalized pattern can be extended to whole core behaviour if a specification is also made for the fraction of the core that enters this transient as a function of time.

Whilst the temperature profile of the core will not be uniform, a simplified approach is to assume that the whole core will behave in the same way as the fuel pin. This can affect the timing of the release, as well as the magnitude of the overall source term to the containment. However, treating all parts of the core as behaving in this way simultaneously is conservative with respect to the timing and magnitude of the source term. Therefore additional conservatism can be incorporated into the algorithm by choosing a sufficiently high value for the "maximum temperature before relocation" (i.e. the maximum temperature applied to the whole core, and a sufficiently long "hold time" (i.e. the length of time for which the core is at this temperature). Slower heat-up transients are known to release more fission products than do rapid transients. Therefore, the selection of the "maximum temperature" and the "hold time" needs to be such that the resulting fission product release is also representative of the slower sequences. The appropriate selection of these parameters is largely a matter of judgement. The "test" for sufficiency in the conservatism provided by the recommended values in this section is to compare the magnitude of the resulting source term with one that has been developed from the full range of sequences using an equivalent release model. This was performed in the Appendix to this report by comparing the NUREG-1465 [6] PWR values with the results of the algorithms in the last two columns of Table A.4. This comparison indicates that the values selected for a maximum temperature of 2960 K and zero hold time are sufficiently conservative for the volatile elements, which are the most radiologically significant.

Thus, simple but robust algorithms are proposed in which these parameters can be incorporated to envelope their effects on quantifying the generalized thermal transient pattern and how that can be converted directly into release timing and quantities. To convert the generalized thermal transient into fission-product release, the following input is required:

(1) The Zr oxidation runaway heat-up rate;

(2) The maximum temperature reached in the transient;

(3) The hold time at the maximum temperature. Using these data, the algorithms need to quantify the following elements associated with the generalized thermal transient;

(4) The fraction of the core entering the generalized thermal transient as a function of time;

(5) The time to uncovery of the core;

(6) The initial heat-up rate and associated timing;

(7) The fission product/aerosol release rates throughout the transient.

3.1.1. Time to start of core uncovery

Core uncovery during a depressurization accident occurs after the choked-flow blowdown of the coolant through either the break and/or the depressurisation system followed by both drainage of the primary system coolant out of the break and boiling away of the residual coolant.

(a) Blowdown time

Because leak before break can be justified for the pipework of the reactor coolant circuit in future plants, we recommend that the blowdown time be estimated (with any appropriate two-phase choked flow model) as being controlled by the particular design of the depressurization system (valve opening time and sequence), orifice and line sizes. and associated blowdown dynamics). This should be an easy hand calculation giving a blowdown time that can be input into the algorithm.

(b) Time to start of core uncovery after blowdown and drainage

After blowdown and drainage, the time at which core uncovery starts clearly depends on the decay heat level (determined by the design specifications for power and burnup) and the quantity of water that can be guaranteed (minimum) to be available to the Reactor Coolant Circuit (RCS) for the design-basis LOCA. The proposed algorithm to determine the additional time for the start of core uncovery after the blowdown period is:

$$\int_0^{T_u} Q(t)dt = \omega \Delta H_{fg}$$

where

0 is the additional time at the end of blowdown plus any time allotted for drainage,

T_u is the additional time to reach the start of core uncovery,

$Q(t)$ is the decay heat curve starting at the beginning of the boil off period and ending at T_u,

ω is the guaranteed water available to the RCS, and

ΔH_{fg} is the latent heat of vaporization of water at the depressurized condition.

Clearly, a significant parameter for the time of core uncovery is ω, the residual water available within the RCS. This parameter is considered to be a design feature that is open to selection by the designer as is Q(t), which is determined by the design selection of the operating power level.

In order to achieve a core melt, it must be postulated that the water sources of the Emergency Core Cooling System (ECCS) that are included in the design fail to reach the core and therefore fail to terminate the sequence. Therefore, it is recommended here that ω should include only the residual RCS and accumulator water that remains after the blowdown and drainage out of a leak path located in the RCS. The guaranteed water must be available at a very high probability and should not be subject to loss from any postulated LOCA paths. Any ECCS water, whether passive or active apart from the accumulator contents, should not be included in this because design basis representative source terms inherently imply failure of the ECCS system to terminate accident sequences.

3.1.2. The fraction of core involvement as a function of time

It is proposed that the fraction of core involvement (with the generalized thermal transient) can be sufficiently approximated (for reference source term purposes) by the rate of uncovery of the rest of the core (neglecting power profile effects). Thus, the total time for core uncovery ($T_e - T_u$), is given by:

$$Q(T_u).(T_e - T_u) = \omega_c \Delta H_{fg} ,$$

where

$Q(T_u)$ is the decay heat level at the start of core uncovery,

T_e is the time of uncovery of the bottom of the active core, and

ω_c is the quantity of water required to fill the void volume within the active core and downcomer.

The desired rate of core uncovery is then

$$\frac{dl}{dt} = \frac{L}{(T_e - T_u)},$$

where L is the length of the active core.

The fraction of the core entering the generalised thermal transient as a function of time is given directly by

$$(t - T_u)/(T_e - T_u)$$

A conservative approach which is taken in the Appendix is to assume that the whole core undergoes the generalized thermal transient simultaneously. This has the effect of producing essentially the same fission product release, but on a slightly shorter time scale.

3.1.3. The initial heat-up rate

In mechanistic codes, the initial heat-up rate is generally determined as a balance between the rate of internal heat generation, the sensible heat required to increase the fuel/clad temperature and the amount of heat lost to the steam by radiative and convective cooling. In the interest of developing simplified algorithms for reference source terms, it is proposed that it is sufficient to use the adiabatic heat-up rate for the clad-fuel with the internal heat source being the decay heat level at the time of start of core uncovery.

As experience has shown that only very small quantities of fission products are released during this phase of the generalized transient, the major effect of this heat-up rate is on the source term timing. The above recommendation minimizes this period of time.

3.1.4. Heat-up rate due to runaway Zr oxidation

Experimental data and code calculations indicate that there is a relatively narrow range of Zr temperatures that is associated with the start of the rapid heat-up transient. The minimum value of 1500 K is taken because this choice will not significantly affect the overall amount of fission products that are released but will shorten the overall generalized sequence time by a very small amount.

In order to exercise an appropriate fission product/aerosol release model to determine the release associated with the runaway Zr-oxidation phase of the generalized thermal transient, it is proposed to use a ramp rate that would be representative of a low pressure scenario (which rarely goes into a steam-starved condition). This ramp rate, however, appears to be mass-transfer limited due to diffusion of steam and hydrogen through a layer of ZrO_2 product to the reaction site at the surface of the metal layer, rather than being limited by the chemical reaction kinetics [7]. The determination of this rate, therefore, requires appropriate mass transport considerations. An expression for a limiting value for this rate resides in the Source-Term Code Package (STCP) and is reproduced in the Appendix to this report. This expression was evaluated in the Appendix for the temperatures at each end of this heat-up phase to give 18 K/s and 25 K/s respectively. It is recommended that an average value of 21 K/s be used for the generalized thermal transient. This value is consistent with the maximum rates that have been observed in appropriate experiments and in calculations using various mechanistic codes. Clearly, if the heat-up rate for this period were to be associated with "steam-starved" conditions, it could be considerably lower (and, consequently, would produce greater fission product release under such conditions). To use the larger heat-up rate and still maintain sufficiently "representative" fission product release, an appropriately high value can be selected for the "maximum temperature" for the generalized thermal transient. It is evident that the recommended value for this is indeed representative from the comparisons made in Table A.4 of the Appendix.

3.2. UNCERTAINTIES

The major uncertainties associated with the quantification of the generalized thermal transient are the heat-up rate for the runaway Zr-oxidation phase, the maximum temperature to which this phase is specified to reach, and any hold time selected for use at that temperature. (Note that the hold time has the equivalent effect, as the maximum temperature on the quantities of fission products released so that both do not have to be specified. The inclusion of the hold time here in the algorithm is to provide an apparently better representation of reality and to give an independent means to somewhat vary the timing.)

Although these uncertainties exist, the effect on the source term of the recommended values for these parameters are shown to be sufficiently representative by the comparisons in Table A.4 of the Appendix.

Another uncertainty is associated with the algorithm for the fraction of the core which enters the generalized thermal transient as a function of time. The algorithm in the Appendix treats the core homogeneously and does not recognize radial or axial power profiles. A detailed analysis would lead to a less conservative release.

The algorithms do not deal with such traditional source term concepts as "gap release". Consequently, the algorithms should not be used for the purpose of making decisions on isolation valve closure times.

4. IN-VESSEL SOURCES

4.1. FISSION PRODUCT/AEROSOL RELEASE

4.1.1. Introduction

Once the thermal transient and the timing for whole core involvement have been determined by the previous steps, an appropriate fission product release model can be used as long as it can incorporate temperature transients and burnup. No particular release model is recommended. Some possible choices are the models in FASTGRASS, SCDAP, VICTORIA, MAAP, MELCOR, ICARE, ELSA and KESS or a simplified "Booth-type" model, but note that not all of these include correlations for the effects of burnup. An alternative is to use a simplified model, such as RelVol, which is based on a Booth type kinetics and described in the Appendix.

4.1.2. Chemical forms

The chemical forms of the fission products, which could be released from the cores of future LWRs, should be similar to those that are applicable to current LWRs. The determination of the chemistry associated with the release of fission products from fuel under accident conditions is complex, and the complexity is increased by reactions in the reactor coolant circuit between the core and the point of release to the containment. However, with the exception of the noble gases and the possible exception of a small fraction of iodine, all the fission products should have condensed prior to entering the atmosphere of the containment and thus, for the purposes of deriving the in-containment source term, all the non-noble gas fission products in the atmosphere of the primary containment can be taken to be present as aerosol.

A small fraction of iodine can be released from the fuel in gaseous forms and can leave the reactor coolant circuit in the forms of I_2 or HI. However, the reactive nature of the latter means that it is more likely to be associated with metallic or metal-oxide systems (i.e. surfaces or aerosol). The presence of a small percentage of iodine in the form of I_2 has been taken into account in some estimations of the source term for design basis accidents for many years [8]. Preliminary confirmation of its presence in the release from degraded fuel has been confirmed by the Phebus test FPT-0, which was conducted with trace-irradiated fuel. However, the results of Phebus test FPT-1 (conducted with high burnup fuel) were less clear. Based on the results of these tests and of chemical kinetic calculations under severe accident conditions, it appears that a level of about 5% would be a conservative estimate of the percentage of iodine that enters the containment in gaseous form.

In a number of models there is an option that fission products Te and Sb (as well as the Sn content in Zr) are retained in unoxidized Zr and subsequently released only after about 90% to 95% of the (fuel region) Zr has been oxidized. In a number of studies [9, 10], it has been shown that the activity of tin, (with which Te and Sb are associated) is only significant when the activity of zirconium is relatively low (i.e. >90% oxidized). Alexander and Ogden [10] derived an expression for the activity coefficient of tin in Zircaloy (γ) which was shown to be

$$\ln\gamma = \frac{(2700 \pm 670)}{T(K)} - (5.57 \pm 0.41),$$

where T is the absolute temperature (K). The activity of tellurium or antimony trapped in the Zircaloy as the relevant tin compounds is therefore only considered to be significant later in the accident sequence, when the zirconium is almost completely oxidized. It is therefore unlikely that these fission products will be released at a relatively constant rate, but rather as a burst later in the accident sequence. One approach could be to neglect this effect, such that the release is similar to that of other volatile fission products. In this method, the inclusive approach for Te and Sb utilized in more detailed models would not be used, and the nuclides would be modelled as if they are being released at constant rates, together with the other fission products. However, it should be recognized that this approach will not adequately describe the late-phase behaviour of tellurium and antimony, which could ultimately compromise the accuracy of the magnitude of the overall source term.

In BWRs, B_4C from the control blades may react with steam to produce methane and other organics which may enhance the production of organic forms of iodine in the containment as discussed in Section 7.3. However, B_4C will react with both the Zircaloy cladding and the steel sheath of the control rods, and thus it is unlikely that the enhancement will be significant

One of the forms of Te is Te_2 and, like all Te, it will decay to I However the iodine which is produced in this way will be highly reactive and will rapidly be associated with the aerosol or the contents of the sump water

4.1.3. Control rod and structural release

The release of material from the control rods will be the dominant source of aerosol in the atmosphere of the containment at certain times after the core has melted and, for these periods, it will determine the rate at which the fission products are removed from the atmosphere of the containment In the case of Ag-In-Cd control rods used in PWRs, the volatile cadmium will vaporize initially, followed by silver which is less volatile [11–13] The vaporization of indium will largely depend on the steam concentration, with the formation of volatile InOH favoured at high temperatures in the presence of steam [14]

$$In + 2H_2O \rightarrow 2InOH + H_2$$

However, the high volatile InOH is only stable at high temperatures, and this will decompose to form low volatile In_2O_3 at lower temperatures It is also possible that cadmium release and aerosol formation could be influenced by the presence of steam, with the formation of CdO or $Cd(OH)_2$

$$Cd + H_2O \rightarrow CdO + H_2,$$

$$Cd + 2H_2O \rightarrow Cd(OH)_2 + H_2$$

The different volatilities of these compounds will not only have an effect on the timings of their release, but also on the morphology of the particles Low volatile species that condense rapidly, such as silver and In_2O_3, will tend to form the smallest particles These will remain suspended for the longest time and will have a greater probability of reaching the containment However, it is more likely that cadmium will be oxidized once the aerosol particles have been formed, which will result in the formation of relatively large particles Transport of these compounds to the containment could also have an effect on the aqueous iodine chemistry in the containment (see also Section 7 3 3)

In order to account of these effects on the natural attenuation of the airborne concentration of the fission-product aerosol in the containment, it is necessary to know the relative timings of the releases of the fission products and the control rod materials This is a significant uncertainty and, in some sequences, where much of the fission products are deposited in the reactor coolant circuit early in the sequence and revaporized later, the control rod material may have largely been removed from the atmosphere of the containment before a significant fraction of the fission product revaporization from the coolant circuit has occurred As stated above, the initial release will be of relatively high-volatile cadmium This will take place over a relatively short period of time However, the bulk of the control rod alloy is comprised of silver (80%), which will vaporize much more slowly, providing a constant source of aerosol during the course of the accident Whilst cadmium will have some influence on the chemistry of fission products in the primary circuit, and indium could provide a significant source of small aerosol particles, it is considered that the uncertainties associated with the aerosol behaviour of these materials are too great to be included in the specification of the reference source term However, it should be possible to include silver This could be accomplished at a fairly simple level by using the vapour pressure equation coupled with the total mass of silver present to describe vaporization Condensation could then be calculated along the relevant temperature gradient. A good knowledge of silver transport is also important in order that the iodine behaviour in the containment can be accurately quantified. This is dealt with in more detail in Section 7.3.

Boric acid is present in the primary coolant of a PWR as a soluble moderator. The concentration of this material will depend on the point in the fuel cycle [15]. It will be at its highest with fresh fuel (typically ~1200 ppm) and is systematically reduced as the cycle proceeds (typically to a minimum of ~50 ppm). In a BWR, the B_4C control blades can react with steam to form boric acid and methane [16] under accident conditions. Whilst boric acid will react with metallic surfaces, typically to form borates [17], this will not act as a significant attenuation mechanism. The influence of boric acid is primarily on the formation of caesium species. Both caesium iodide and caesium hydroxide react with boric acid to form caesium borate:

$$CsOH + HBO_2 \rightarrow CsBO_2 + H_2O\,,$$

$$CsI + HBO_2 \rightarrow CsBO_2 + HI\,.$$

In the case of the reaction with CsI, volatile hydrogen iodide will be formed, thus influencing the behaviour of iodine as well as caesium. Boric acid can react with CsI or CsOH either in the vapour phase or as a reaction between the fission product vapours and boric acid aerosol, formed by flashing or condensation. The uncertainties associated with the latter are too great to be included in the reference source term, but it may be possible to include the vapour-vapour reaction. Vaporization of boric acid is most significant in the presence of steam. Studies at ORNL have shown [18, 19] that voltilization occurs up to core melt, after which it ceases due to its affinity for metal oxides. The $CsBO_2$ formed by the reactions with caesium species is far less volatile than either CsI or CsOH, and thus it may deposit earlier than would be the case for the precursors. However, because of its low volatility, $CsBO_2$ is likely to condense to form small particles that will remain suspended for longer, and thus have a greater chance of reaching the containment. Once the boric acid aerosol reaches the containment, it will deposit and ultimately influence the pH of the sump. Volatilization of iodine from the sump is strongly dependent on pH. Therefore boric acid transport to the containment could influence accident management strategies which depend on an alkaline sump. It is also possible that $CsBO_2$ could have an effect on the aqueous iodine chemistry in the containment. These phenomena are discussed in more detail in Section 7.3.3.

Similarly, because of the large uncertainties in the release models for structural aerosols, we recommend these not be included in the representative source term specification. On the other hand, if particular containment design features are proposed for which a large quantity of structural aerosol would interfere with their proper functioning, there is a need to estimate the structural aerosol release. This could be done by applying structural release models to the structural materials in the core assuming that they undergo the same generalized thermal transient as those which prevailed for the core.

4.1.4. In-vessel melt retention strategies

A feature for some future NPPs may be external cooling of the rector vessel in order to retain the melt within the vessel. If a future plant proposes this strategy, it is recommended that no additional in-vessel fission product or aerosol release (over than that calculated for the heat-up and melt period) be included in the source term. The reason for this recommendation is that it is considered that the driving forces for the release of fission products from unsparged pools and for driving fission products and aerosols out of the primary vessel to the leakage paths will be small, while the melt is in the bottom head, and that little will escape from the RCS into the containment.

4.1.5. Reflood

Experiments that have included reflooding of the degrading core have shown "spikes" in the production of hydrogen and the release of fission products. However, the probability of the reflood occurring just at the time when the fission product release has neared its maximum value such that any additional release can be considered as an enhancement is extremely small. For this reason, it is not considered to be appropriate to include an enhancement of the source term due to reflooding the core. There may also be accident management considerations resulting from the long term effects of a core immersed in a water pool over a long period of time.

4.2. UNCERTAINTIES

Increasing the average burnup to values of greater than 40 000 MW·d/t U may have a significant effect on the release of fission products and other materials.

The effect of burnup on fission product release is included in the RelVol model discussed and used in the Appendix. However, the database underlying this model is sparse and there are significant uncertainties to be associated with extrapolating the model to (for example) 60 000 MW·d/t U.

There is a potential for air-ingress to influence the late stages of the source term with respect to its potential for adding significant Ru, Mo and, perhaps, Pu into the source term. However, the probability of this occurring is considered to be low enough that it need not be accounted for in the reference source term.

5. EFFECTS OF THE RCS ON FISSION PRODUCT BEHAVIOUR

The effects of the RCS on attenuating the release of fission products and aerosols, changing the timing via revaporization or resuspension, and affecting the chemical forms are undoubtedly important for many severe accident sequences. It is noted, however, that these effects are minimized in low pressure sequences in which the fission product residence time in the RCS is small. It is very likely that future LWRs will employ a strategy of depressurization. Therefore, in the interest of simplifying the regulatory expectations and in developing a representative reference source term, it is proposed that no credit be given for the effects of the RCS. This greatly simplifies the application of the algorithmic methodology and is not considered to introduce any unacceptable distortion into the design basis "representative" source term. However, it should be remembered that exclusion of all primary circuit effects from the definition of the "representative source term" will not necessarily give a maximum value. Whilst this may be the case for fission products, other materials such as control rod alloy and boric acid could have an effect on the behaviour of iodine in the containment in terms of both mitigating or exacerbating the consequences of an accident.

6. EX-VESSEL SOURCES

Despite the many phenomena that are important for evaluating melt behaviour within the containment, the associated fission product source term is generally low compared to the source term from heating-up and melting of the core within the vessel. This is mainly because of the geometrical conditions and reduced temperatures. In order to describe the ex-vessel source term properly, one has to distinguish between existing plants, where there is no provision against molten core–concrete interaction (MCCI) and those future plants, which will be designed to avoid penetration of the basemat.

In the case of existing plants, a significant release of fission products and especially non-active aerosols is predicted to occur as a result of MCCI after the corium has melted through the reactor pressure vessel (RPV) and fallen into the reactor cavity. The release is mainly driven by gas and steam, which results from the concrete decomposition. Other ex-vessel sources are related to transient phenomena, which occur at about the time that the vessel is breached, or to long-lasting but small effects, namely:

- direct containment heating (DCH) by the corium following high pressure melt ejection,
- fuel–coolant interactions (FCI) in the reactor cavity or spreading area,
- resuspension of fission products from boiling water pools or from surfaces in the containment or the RCS due to mechanical or thermal forces,
- the formation of gaseous forms of iodine from involatile forms in the aqueous aerosol, water pools and films (see Section 7).

For future reactor designs, the requirement to mitigate the consequences of severe accidents has to be taken into account. As a result, measures are expected to be introduced that might reduce the probability of vessel failure and/or will stabilize the corium in the containment to avoid penetration of the basemat. In addition, measures to depressurize the reactor coolant system deliberately will significantly reduce the probability of a high pressure melt ejection and the early breach of the containment as a result of a missile. However, where core retention measures are introduced, there is a need to consider the potential of a release from the molten corium if there is a dry phase and, where water is introduced to cool the corium, there is the need to consider the release of fission products as a result of the interaction of the corium with the water.

These phenomena are complex and, in many cases, are plant specific, so that it is not possible to recommend generic numerical source terms. However, methodologies for deriving the associated source terms to the containment are given in the following sections. As mentioned above, the additional ex-vessel source term is generally small.

6.1. MOLTEN CORE–CONCRETE INTERACTIONS (MCCI)

As noted above, MCCI may occur at operating plants but its probability is expected to be significantly reduced by the design measures that are expected to be incorporated into the designs of future plants. A similar type of release may occur if core retention devices involve some sacrificial materials that also provoke a sparging process.

The reaction between molten corium and concrete without water is sufficiently well understood and can be calculated using codes such as CORCON (NRC) or WECHSL (FZK and IPSN), which have been validated using the SURC (SNL), BETA (FZK) and ACE experimental programmes. In spite of the Mace experimental programme, which addressed the late addition of water, uncertainties in the source term to the containment still exist as to when MCCI stops. Codes to calculate fission-product and aerosol release associated with MCCI, such as VANESA or CHEMSAGE have been validated by the ACE experiments. If MCCI were to occur, it is expected that all the noble gases, iodine, caesium and tellurium that remain in the corium will be rapidly released. In addition, the release of moderately volatile fission products such as ruthenium, barium and strontium, and the isotopes of plutonium need to be evaluated, especially if oxidizing conditions occur.

The corium spreading and freezing process in a large reactor cavity or spreading area is the subject of experimental programmes, which have been mainly carried out in France and Germany (FZK KATS, Siempelkamp CORESA, CEA CORINE, VULCANO, JRC FARO [20, 21]) and these experiments will provide the necessary data to validate the codes CORFLOW (Siemens), MELTSPREAD (EPRI) and CROCO (IPSN). However, the consensus of expert opinion is that if the corium can spread over a large enough area and if it is quenched with water, there will be no significant enhancement of the source term to containment as a result of MCCI.

6.2 RELEASE FROM MOLTEN POOLS

Fission product release from molten pools is generally taken to be an ex-vessel phenomena. However, it is also relevant to in-vessel processes where fuel liquefaction occurs prior to vessel failure. The general assumption with respect to molten pool behaviour is that, where a core retention device retains the corium in the form of a molten pool and no sparging occurs, there will be no significant addition to the source term to the containment. In the case of the noble gases, iodine and caesium, around 50% of the inventory is expected to be released during the in-vessel phase, as outlined in Section 4. For the other nuclides, the low equilibrium activity coefficients and the potential for the formation of a crust lead to the judgement that the release will not be significant. A programme comprising experiments to examine the release of fission products from metallic and oxide melts, associated code development and testing, and plant calculations has been conducted by AEA Technology, NES/Ruhr University of Bochum, IPSN, Siemens/KWU and Leningrad Special Kombat Institute “Radon” (LSK, St. Petersburg) as part of the Commission of the European Communities 4th Framework Programme. The experiments have addressed such issues as sparging, and crust formation, whilst the code testing and development has been concerned with development of the RELOS/CHEMSAGE code package. Plant calculations have been mainly concerned with issues associated with the EPR. Preliminary results from the experimental programme [22, 23] have shown that there was significant release of fission products of intermediate volatility (Ba, Sr, etc.), even in the absence of sparging. Ruthenium will only be released in oxidizing atmospheres due to the formation of volatile RuO_3. The experimental results for this programme need to be assessed further with respect to the behaviour of ruthenium and that of the lanthanides and actinides.

6.3 FUEL–COOLANT INTERACTION (FCI) AND DIRECT CONTAINMENT HEATING (DCH)

In the context of the source term to the containment, the phenomena of FCI and DCH have the potential to fragment the molten fuel thermally or mechanically, provoke chemical reactions and change the chemical nature of the fission products into more volatile forms. The most important chemical reactions are those that lead to the formation of ruthenium oxides which are both volatile and radiologically significant.

The subject of energetic fuel-coolant interactions or steam explosions has been extensively considered from the point of view of creating missiles which would threaten the integrity of the containment. The work prior to 1985 was considered by the first Steam Explosion Review Group in the USA, which concluded that the probability of such an event is low enough to be considered negligible. More recent work presented at the CSNI Specialists Meeting in Santa Barbara in 1993 supported the above conclusion. For several years, considerable research, in the form of the QUEOS, PREMIX and BERDA experimental programmes and the development of the IVA-KA code, has been under way at FZK to demonstrate that a steam explosion with the consequence of a containment failure is practically excluded. Nevertheless, 'minor' steam explosions, which are limited to a small fraction of the core, may have an influence on the source term. Because of the difficulty of calculating the corresponding fission product release, it is recommended that the release of all low and medium volatile species, including Ru, which are contained in this fraction of the core, is assumed. As this fraction of the core is small, the reference source term is still considered to be adequately conservative to have included this contribution. For both present reactor designs, as a result of accident management, and future designs, as a result of design measures, DCH will be practically avoided or limited by depressurizing the reactor coolant circuit sufficiently early, in order to avoid vessel failure above a threshold pressure level. Experiments will be performed by FZK to define this threshold.

6.4. ADDITIONAL EX-VESSEL PHENOMENA

In addition to the production of gaseous forms of iodine, which is addressed in Section 7, the other phenomena which have the potential to affect the source term are:

- the revolatilization of fission products which are deposited on the surfaces of the reactor coolant system and the containment,
- the resuspension of radionuclides from boiling water pools,
- the effects of hydrogen burns and the use of recombiners,
- the use of spray with chemicals added to the sump or spray water.

It is the consensus of expert opinion that none of these phenomena are really important in deriving the cumulative source term from the containment for the sequence that is relevant to the present report, where the integrity of the containment is maintained and the source term is to be used for design purposes.

Revolatilization may affect the timing of the release to the containment, but neglecting retention in the reactor coolant circuit, as discussed in Section 4, will lead to a reasonably robust source term. Revolatilization will, however, be an important phenomenon in risk analyses which consider sequences where the containment fails or is vented.

The resuspension phenomenon in pipes has been studied in various national and international experimental programmes and continues to be the subject of the STORM programme, but again, neglecting retention in the primary circuit leads to a reasonably robust source term and avoids the uncertainties associated with predicting the thermal-hydraulic behaviour of the reactor coolant circuit.

The judgement that resuspension from the boiling sump will not significantly affect the source term is expected to be confirmed by the ongoing experimental programme at FZK (KAREX) and related theoretical work.

The following effects are associated with hydrogen mitigation:

- hydrogen burns: mechanical resuspension, effect on iodine chemistry (short, but high temperature peak);
- recombiners: superheated atmosphere and thus less condensation of steam on aerosols, smaller aerosol size distribution and lower removal rate, effect on iodine chemistry (see Section 7).

The magnitude of these effects is negligible in the context of the conservative procedure for calculating the removal of airborne aerosols, as proposed in Section 7.

7. IN-CONTAINMENT BEHAVIOUR OF FISSION PRODUCTS

7.1. INTRODUCTION

The release of radioactivity from the containment depends on the magnitude and chemical and physical form of the radioactivity that is airborne in the containment as a function of time. Thus, once the source term to the containment is established, it is the subsequent behaviour of the released material within the containment that determines the in-containment source term.

The regulatory requirements associated with the use of design-basis source terms and the subsequent behaviour of the fission products in the containment sometimes take the form of an acceptable dose criteria at the site boundary given an additional design basis specification that establishes a pressure source in the containment to be coupled to a design leak rate. The pressure "source term" specification in some countries is to use the maximum pressure resulting from the containment response to a set of design basis accidents. This pressure is specified to be held constant for a relatively long time (e.g. 24 hours). The pressure source also includes a contribution from the burning of some specified fraction (of the total possible) of the hydrogen that is expected to be generated by severe accidents. In developing the containment response, credit has been allowed for engineered safety features such as the use of sprays for containment cooling and the removal of gaseous iodine, the use of suppression pools if they are not saturated, and the use of coolers.

It was never intended that design-basis source terms be used with any particular accident sequence to determine the dose at the site boundary. Instead, the intent was that the combination of the fission product source term, the pressure source term, and the containment design leakage rate results in an acceptable containment design with respect to overall risk resulting from all accident sequences. Although it would seem consistent to develop the in-containment behaviour by using an actual depressurized (large break LOCA) sequence, this should not be done because the "representative" source term presented here is a surrogate for the spectrum of credible severe accident sequence source terms. It is not appropriate to try to identify it with any one sequence.

It is clear that the evaluation of the thermal-hydraulic behaviour of the containment with respect to any particular accident sequence is only appropriate for general guidance and is not to be used to evaluate the in-containment behaviour of the reference source term aerosols. The question then is how to deal with containment thermal-hydraulic effects in evaluating the in-containment behaviour of the design-basis representative source term. Clearly, since there is no technically defendable basis for coupling the source term to any specific timing associated with thermal-hydraulic phenomena and to parameters such as diffusiophoresis, thermophoresis, and relative humidity, then it is inappropriate to take these into account.

The aerosol behaviour phenomena, in which high confidence can always be placed on the expectation that the effect will always be present for all accident sequences at all times, are agglomeration (gravitational and Brownian), gravitational settling and diffusional plateout. In addition, in order to encourage the innovative use of engineered safety features (e.g. containment sprays and suppression pools) for source term mitigation, credit should be allowed for these only in proportion to the confidence level that can be placed on their availability during severe accidents. Thus, it is proposed that only the above phenomena and reliable ESF systems be used in evaluating the in-containment behaviour of fission product aerosols.

The reference pressure "source" to be used along with the in-containment aerosol behaviour is a separate issue. An option would be to stay with the above-mentioned approach of design basis accidents and utilize the maximum pressure arbitrarily held for some specified period of time. While this appears to be an appropriate regulatory approach, there is still an issue of what credit, if any, should be allowed for the use of ESFs in reducing the design basis pressure. In making any decision on this issue, appropriate consideration should be given as to whether or not an acceptable level of defence in depth is achieved in view of the reliabilities of the proposed ESFs.

Aerosol agglomeration, gravitational settling and the effect of ESF actions on the transient aerosol concentrations in the containment are generally included within detailed mechanistic aerosol codes such as CONTAIN, ECART, FIPLOC, FUMO, GOTHIC, JERICHO, MAAP, and MELCOR. One possible acceptable approach to account for the in-containment aerosol behaviour would be to use such codes to determine the transient aerosol behaviour but, in deriving the reference in-containment source term, it will be necessary to "turn off" all associated thermal hydraulics along with the models for diffusiophoresis, thermophoresis, condensation and hygroscopic effects.

In keeping with the spirit of this report, simple algorithms are proposed to capture the essence of these mechanistic models for the effects of:

- Brownian and gravitational agglomeration,
- gravitational settling,
- spray removal,
- decontamination by water pools.

7.2. IN-CONTAINMENT AEROSOL BEHAVIOUR ALGORITHMS

7.2.1. Agglomeration and gravitational settling

The severe accident literature contains a number of simplified algorithms that have been developed to capture the complex physics associated with Brownian and gravitational agglomeration and gravitational settling. [Note: diffusional plateout can also be included in these but, in essentially all containments, the surface area to volume ratio is small enough to neglect this effect]. One of the most complete of such algorithms, that has also been demonstrated to be of acceptable accuracy by comparison with both experimental data and with the results of the detailed aerosol codes, is the model in the MAAP code developed by Fauske and Associates and further refined by EPRI [24]. It is recommended that the EPRI version of this algorithm be used as presented below.

This correlation is intended to reflect the transient variation in the removal rates. Therefore, the removal rate coefficient, λ, in the general sedimentation relationship,

$$\frac{dC}{dt} = -\lambda C,$$

will be a function of time. Consequently, this relationship cannot be explicitly integrated. Therefore, the algorithm must be applied on a finite difference basis marching forward in time using appropriately sized time steps.

Because the nature of the removal rate coefficient depends on the strength of the source of the aerosols relative to the concentration level, the correlation is in two parts called the "steady state" part (λ^{ss}) corresponding to strong and continuing sources, and the "decay" part λ^{D} corresponding to a weak or zero source. A criterion is included to decide which of these to use at any time step. They are never used simultaneously and, generally, the "decay" part is used after the source has ceased.

The calculational sequence at each time step is as follows:

1. Determine the coefficient for the fractional rates of change of suspended concentration due to the source, λ_s, and due to outflow, λ_o

$$\lambda_s = \left(\dot{S} / V / C\right)$$

where

$\dot{S}$ = source mass rate

V = volume of the containment

C = suspended mass concentration

λ_o is input from the design leak rate as λ_o = (leakage rate) / V

2. Calculate the optimal coefficient, λ^{ss}, from

$$\lambda^{ss} = \left(\frac{\alpha K_0 g \rho_p}{\gamma \chi^2 \mu_g h^2 \varepsilon_0} \right)^{1/2} \Lambda^{ss}$$

where

α = "density" morphology correction factor (set = 1 for practical reasons)

K_0 = Brownian agglomeration rate coefficient

= $4\, k_B T / 3\, \mu_g$

μ_g = gas kinetic viscosity

g = acceleration due to gravity

ρ_p = particle material density

ε_o = collision efficiency scale factor

= 1/3 for Pruppacher-Klett formulation

γ = collision morphology correction factor (set = 1 for conservatism)

h = settling height (this is the ratio of the containment free volume to the horizontal surface area),

and

$$\Lambda^{SS} = 0.226 M^{0.282} \left(1 + 0.189 M^{0.8}\right)^{0.695}$$

where

$$M = \left(\frac{\varepsilon_0^5 \gamma^9 g h^4}{\alpha^3 K_0 \mu_g \rho_p^3} \right)^{1/4} \cdot C$$

Note that, since the concentration, C, is part of the correlation,

it is recommended that the value at the start of each time step be used.

3. Decide whether or not to use the “steady-state” or the “decay” correlation from the criteria below.

$$\text{If} \qquad 3\dot{S}/\lambda^{``}\begin{cases} \le C & use''\,steady-state'' \\ > C & use''\,decay'' \end{cases}$$

4. If decay correlation is selected from the criteria, calculate the coefficient for the “decay” option from

$$\lambda^{D} = \lambda^{SS}\Lambda^{D}\big/\Lambda^{SS}$$

where

$\Lambda^{D} = 0.528M^{0.235}\,(1 + 0.473M^{0\,754})^{0.786}$

5. Either total fractional rate of increase of suspended mass for the time step using the appropriate coefficient (1) or (2) below as indicated by the criterion in step 3:

$$\text{(1)} \quad \text{“Steady state”: } \lambda = \left(\frac{\lambda_{SED}}{\lambda^{SS}}\right)\lambda^{SS} + \lambda_{S}$$

where

$$\lambda_{SED}\big/\lambda^{SS} = \left(1+\frac{1}{3}R_{L}^{SS}\right)\left[1+\left(\frac{R_{L}^{ss}}{1+R_{L}^{ss}/3}\right)^{4\,5}\right]^{0\,222}$$

$$R_{L}^{SS} = \lambda_{0}\big/\lambda^{SS}$$

or

$$\text{(2) “Decay”: } \lambda = -\left(\frac{\lambda_{SED}}{\lambda^{D}}\right)\lambda^{D} + \lambda_{S}$$

$$\lambda_{SED}\big/\lambda^{D} = 1+R_{L}^{D}$$

$$R_{L}^{D} = \lambda_{0}\big/\lambda^{D}$$

Note that, if the leakage rate is zero or very small, then the above correlation options reduce to

(1) “steady state”: $\lambda = -\lambda^{SS} + \lambda_{S}$ and

(2) “decay”: $\lambda = -\lambda^{D} + \lambda_{S}$

It should be noted that this algorithm implicitly assumes the source aerosol size to be small and log-normally distributed and, therefore, does not require an input specification for either the mean size or the variance.

7.2.2. Spray removal of aerosols

The basic equation used for spray removal of aerosols is,

$$\frac{dm_f}{dt} = -\lambda m_f , \tag{1}$$

in which m_f is the mass fraction $[M(t)/M_o]$ of aerosol remaining airborne in the containment as a function of time, t

Generally, the removal rate coefficient, e, has been found to be a function of the spray flow rate (on a per nozzle area coverage basis), Q (cm^3/cm^2 s) and the spray droplet fall height, H(cm) as well as the spray droplet diameter, the aerosol size, and the mass concentration of aerosols airborne, M(t)

Powers [25], however, noted that the distributed range of droplet sizes from standard nozzles currently in use is sufficiently narrow that it can be treated as an uncertainty parameter rather than as an independent variable Similarly, from the results of severe accident calculations that used the Source Term Code Package (STCP), he noted that the distributed aerosol size as released into containment from the RCS or from dry MCCI can be well represented by a log-normal distribution with the range in the mean size of 1 5 to 5 5 ìm (assumed uniformly distributed in his uncertainty analysis) and a geometric standard deviation range of 1 6 to 3 7 (also assumed uniformly distributed) and can, therefore, also be treated as an uncertainty parameter

Consequently, these two parameters, along with other important influencing parameters, were exercised in an uncertainty assessment [Monte Carlo sampling] from appropriately assigned distributions by Powers using a detailed mechanistic model for spray removal This mechanistic model purposely did not include condensation (or diffusiophoresis) and thermophoresis effects

Powers correlated the results for the calculated distribution of the rate coefficient for the 10%, the 90%, and median values in terms of the important independent variables, Q and H, and the important dependent variable, m_f These correlations were developed in two parts

(a) $\lambda(m_f) = 0\ 9$ → presented as a function of Q and H, and

(b) $\lambda(m_f) / \lambda(m_f = 0\ 9)$ → presented as a function of Q and m_f,

to give an overall $\lambda(m_f)$ as a function of Q, H, and m_f

The basic relationship, then, becomes

$$\frac{dm_f}{dt} = -\lambda(Q,H,m_f)m_f \tag{2}$$

Power's correlation for the median result of the uncertainty analysis (which is recommended here) is

$$\ln \lambda(m_f = 0\ 9) = 6\ 83707 + 1\ 0074 \ln Q - (4{,}1731 \times 10^{-3})Q^2H - (1\ 2478)Q$$
$$- (2\ 4045 \times 10^{5})H + (9\ 006 \times 10^{-8})QH^2$$

where

λ has units of h^{-1},
Q is in cm^3/cm^2 s, and
H is in cm

This is to be combined with

$$\lambda(m_f)/\lambda(m_f{=}0.9) = (0.1815 - 0.01183\ \log_{10}Q)\cdot[1 - (m_f/0.9)^{0.5843}] + (m_f/0.9)^{0.5843}.$$

Since λ is a function of m_f, the procedure recommended here for the simplified algorithm is as follows:

(1) Input into the correlations the known values for Q and H to get λ as a function of m_f alone.

(2) Substitute the result of (1) into Equation (2) and numerically solve for m_f as a function of time.

If there are fractional volumes of the containment that are unsprayed, [α = V (unsprayed)/V(sprayed)], then the effective λ would be

$$\lambda(eff) = \lambda(Q, H, m_f) / (1+\alpha).$$

7.2.3. Pool scrubbing decontamination factors

Similar to the above treatment to develop a simplified correlation for spray removal of aerosols, Powers [26] also exercised a detailed mechanistic model in an uncertainty manner over the ranges of important parameters to develop a simplified correlation for decontamination by suppression pools.

Along the lines of his spray analysis, Powers treated the assumed log-normal aerosol size distribution characteristic of the in-vessel release as being an uncertainty parameter (both μ and σ). He also treated the carrier flow (steam and hydrogen) as an uncertainty parameter along with the uncertainties in the phenomenological relationships. This approach left only the depth of submergence of the quenchers as being the important independent variable. The correlation developed by Powers applies to both "T" and "X" quencher designs.

The parameter of interest, of course, is the decontamination factor, DF, defined as the ratio of the total mass of aerosol entering the pool to the total mass escaping after passage through the quencher submergence depth, H.

The median correlation (considered here to be the best estimate value) developed by Powers is simply:

$$\log_{10}DF = 1.791 + (2.477 \times 10^{-3})H(\text{cm})$$

A measure of the spread about the median can also be obtained with Power's correlations for the 10 and the 90 percentile values which can be viewed as lower and upper bounds on the DF:

$$\text{and}: \begin{array}{l} (10\%)\log_{10}DF = 1.034 + (0.875 \times 10^{-3})H(\text{cm}) \\ (90\%)\log_{10}DF = 3.964 + (6.028 \times 10^{-3})H(\text{cm}) \end{array}$$

Note in evaluating the reduction of the reference source term into containment as a result of its passage through a suppression pool, it is important to make an assessment of the bypass and exclude that portion from the removal by the suppression pool It should also be noted that the description of Powers does not apply to the churn-turbulent flow regime, but is only applicable to bubbly flow conditions It is possible to envisage scenarios in which churn-turbulent flow will be important, and this effect is not currently modelled by pool-scrubbing codes such as BUSCA and SPARC [27, 28]

7.2.4. Containment coolers

In some designs, fan coolers are provided to reduce the containment temperature following a severe accident Fan coolers will also remove aerosols from the flow stream but the removal process is diffusiophoresis via steam condensation As discussed above, diffusiophoresis processes are specific to individual fault sequences and have therefore not been included in the derivation of the representative in-containment source term

7.2.5. Major uncertainties and R&D requirements in aerosol behaviour

The basic phenomena associated with aerosol physics including particle agglomeration and gravitational settling are well understood There are, however, uncertainties associated with the specification of the initial aerosol size distribution and with the so-called shape factors. The uncertainty in size can be compensated for by specifying a small size as recommended above The uncertainties associated with shape factors can partially be dealt with by conservatively setting the agglomeration shape factor to 1 Setting the dynamic shape factor (settling velocity effect) to a value of 1 will not be conservative However, in view of the fact that most severe accident sequences occur at high humidity and that the resulting aerosols are hygroscopic, using a value of 1 for this shape factor is probably appropriate

7 3 IODINE BEHAVIOUR INCLUDING pH CONTROL

7.3.1. Introduction

For PWR designs where the pH of the sump water is controlled at 8 or greater for the duration of the accident and where the materials in the containment and those released from the reactor coolant circuit are typical of current designs, the simplification of assuming that all the iodine remains in the form of CsI will be appropriate, providing that it can be demonstrated that the formation of acids does not significantly reduce the pH of the aqueous aerosol (airborne) It should also be noted that boric acid from the primary coolant in a PWR or the B_4C control blades in a BWR will also have the effect of reducing the pH of the sump in an accident (see also Section 4 1.3).

However, if the pH is not controlled, there is the potential for it to fall significantly below 8 for the reasons outlined in Section 7 3 3 below In this case, there will be a considerable increase in the production of gaseous iodine, such that the total release from the containment is measured by a factor of 5 compared to that which is predicted if all the iodine is present as CsI Currently data are not sufficient to predict with confidence that molecular iodine will be trapped by materials such as silver from the control rods (if present, see Section 4 1 3), and thus a full account of iodine chemistry must be made when evaluating this potential contribution to the source term This evaluation must include the production of volatile forms by radiolysis in the aqueous aerosol and in the sump or suppression pool water as well as the subsequent production of organic iodine

7.3.2. Phenomena

The dominant form of iodine released from the core in most accident sequences is expected to be I^-, which will be converted into CsI vapour within the RCS. In the RCS or at least on entering the containment, the CsI vapour will condense to form an aerosol but, because of its hygroscopic nature, CsI will either dissolve in airborne water droplets or, in the steam-rich atmosphere, it will form nuclei for condensation droplets and form Cs^+ and I^-. Thus, the majority of the iodine will initially be in the form of I^- in the aqueous phase of the primary containment atmosphere.

The aqueous aerosol will be transported to the sump or to other surfaces within the containment by the natural processes described in the previous section, or by the containment sprays. However, I^- in the aqueous aerosol or the sump, can be converted to the volatile forms I_2 or HOI by radiolysis, due to the high radiation fields in the containment. The same will occur in the water of the suppression pool in a BWR. These volatile forms can be transferred to the gas phase by mass transfer and, since they will be removed from the containment atmosphere at a slower rate than the aerosol, they have the potential to significantly enhance the quantity of iodine that is airborne in the containment atmosphere and thus the source term to the environment.

In turn, gas phase iodine can undergo mass transfer to the aqueous phases of the aerosol, the sump or suppression pool or the droplets of the containment spray system. Another potentially important mechanism for removing iodine from the containment atmosphere is diffusiophoresis, whereby both gaseous iodine and the aerosol are transported to surfaces where condensation is occurring and iodine in the aqueous phase can drain to the sump. In addition, I_2 and HOI can further oxidise to IO_3, which is involatile, or it can be formed by I_2 reacting with ozone in the atmosphere of the containment, although this behaviour can be considered to be atypical.

In addition to radiolysis, there are several other chemical reactions which have the potential to affect the behaviour of iodine in the containment significantly. In particular, volatile organic iodine compounds can be formed by reactions of iodine in both the gaseous and aqueous phases. The hydrocarbons needed for such reactions may be present in the containment before the onset of the accident, or they may be produced during the course of the accident. Potential sources of hydrocarbons are, for example, paints, coatings, oils, lubricants, and cable insulation material in the containment and boron carbide in the core (see Section 4.1.3). The release of hydrocarbons may be due to radiolysis, heating, or chemical reactions. Organic iodine is not as reactive as molecular iodine and it is not readily removed by spray droplets. Thus, it has the potential to remain airborne in the containment atmosphere.

Other potentially important reactions include the reactions of volatile iodine with silver in the aqueous phase and with ozone in the gaseous phase. In some operating PWRs, silver is present in the control rods and a much larger mass of silver is released to the primary containment during a core melt than of iodine (see Section 4.1.3). The presence of silver leads to the potential for the formation of AgI, which would reduce the amount of volatile iodine that is present in the containment atmosphere. However, this potential mechanism for trapping iodine will not occur in those new generation reactors where silver is not used as a rod material. In the absence of a radiation field, molecular iodine has been measured to react with ozone in the air to produce IO_3^-, which will condense on the aerosol. This may also be an important mechanism in wet atmospheres where the radiation levels are high.

A major factor which affects the rate at which volatile iodine is produced in the aqueous form is the pH of the water as discussed in Section 7.3.2. A second important factor is the rate at which mass transfer occurs between the aqueous and gas phases. In the case of a PWR sump, the water can be boiling when it comes into contact with corium released from the vessel. Currently there are few experimental data for the mass transfer between the sump water and the gas phase for a boiling sump.

An upper bound value is obtained by assuming instantaneous equilibrium between I_2 in the steam and I_2 in the sump, which is currently considered to be the best estimate approach. An effective rate constant for pool boiling is given by:

$$k_{evap} = B / (V_{liq} \cdot pc)$$

where:

B is the sump boiling rate ($m^3.s^{-1}$),

V_{liq} is the sump volume (m^3) and

pc is the I_2 volume partition coefficient.

The overall rate constant is, therefore:

$$k_{total} = 1/V_{liq} \cdot (B/pc + V_{mt} \cdot A_{lg})$$

where:

V_{mt} is the mass transfer rate ($m.s^{-1}$) and

A_{lg} is the liquid/gas interfacial area (m^2).

The overall rate constant will be dominated by the sump boiling effect. The boiling of the sump is expected to result in the above behaviour for the volatile reactants, I_2, HOI, H_2 and O_2.

In the non-equilibrium case, the reverse partitioning rate constant is that which would apply to a non-boiling sump with a forward (gas to liquid) mass transfer coefficient in the order of 3.10^{-6} m s^{-1}. The actual behaviour is expected to lie between these extremes.

In order to evaluate the in-containment source term, all the above processes need to be considered and, as a result, computer models have been developed to predict iodine behaviour. These are briefly described in Section 7.3.4. However, the relative importance of these processes critically depends on the pH of the aqueous phase, as discussed immediately below.

7.3.3. pH control

Controlling the pH is a major factor which affects the distribution between volatile and non-volatile iodine species in the containment. If, during the whole duration of the accident sequence, the pH of the aqueous phase is kept alkaline, at a value greater than 8, a very limited fraction of iodine will be converted into volatile species (elemental iodine and organic iodide) in the sump or suppression pool water. If, on the contrary, during the course of the accident the pH drops into the neutral or acidic range, the distribution of iodine species can be considerably different, with the volatile forms becoming dominant inside the containment.

In some NPP designs, pH control is practised either by storing chemicals in the containment at a place where they are inherently flooded and dissolve in the sump water or by pumping them into the containment during an accident using the containment spray system. The disadvantage of the former is that in some plant designs, water in the containment may exist in the form of separate pools, and in order to inhibit the production of elemental iodine, the engineering design should ensure that the pH of each pool is adequately controlled. The disadvantages of the latter are that

- highly alkaline water is introduced into the containment if the spray system is activated inadvertently and
- pH control depends on the successful operation of the spray system.

If the design does not include an engineered system for controlling the pH, the pH will need to be calculated in order to evaluate the iodine source term. The assessment of the pH of the containment water pools must take into account the following materials and phenomena:

- boric acid and boron oxides (acidic);
- caesium hydroxide and caesium borates (basic);
- other basic compounds which may be formed such as metal oxides and hydroxides (e.g. $Cd(OH)_2$);
- hydrogen iodide (acidic);
- pH additives such as lithium hydroxide, sodium hydroxide and tri-sodium phosphate (all basic);
- absorption of atmospheric CO_2 and CO_2 generated by MCCI by water to form carbonic acid;
- irradiation of water and air to form nitric acid;
- core-concrete interactions producing basic oxides (K_2O, Na_2O and CaO);
- pyrolysis and radiolysis of organo-chlorine and -sulphur materials producing hydrochloric and sulphuric acids;
- acids formed due to dissolution of paints.

Some sources of chemicals are difficult to quantify, for example radiolysis and pyrolysis products. Since exact data on release rates may not be available, the pH must be calculated using estimates for the amounts of acidic and basic materials that are conservative. This applies to the accident sequences both with and without the pH control systems.

7.3.4. Computer codes

In order to predict the behaviour of iodine in the containment, several computer codes have been developed. Five codes which model the behaviour of iodine are INSPECT [29, 30], IODE [31, 32], IMPAIR [33, 34], the LIRIC database [35] and TRENDS [36]. All of these consider many chemical processes that occur in the course of the accident. The modelling concepts and underlying databases of the mechanistic INSPECT and empirical IODE and IMPAIR codes are discussed in detail in Ref. [37]. This comparison identifies several important differences. In particular, INSPECT is the only code which models radiolysis in the aqueous aerosol which is potentially the greatest source of gaseous iodine in the containment atmosphere. On the other hand, it does not model the production of organic iodine, which is potentially another important source of gaseous iodine and which is modelled empirically in the other codes. However, a general requirement is that a good overall model is required for this process. As illustrated in Ref. [38], there are other important differences in terms of the modelling of mass transfer from the boiling sump water, condensation, the transfer of condensed water from the containment surfaces to the sump and the reaction of molecular iodine with ozone to produce iodinate. Thus, for a complete description of iodine behaviour in the containment, more than one of the existing codes is required. However, from analyses such as those described in Ref [38], some general conclusions can be drawn which have been summarised in Section 7 1 1 above

7.3.5. Uncertainties and R&D requirements

The major uncertainties which affect the production of the in-containment iodine source term are as follows

(1) Although there is considerable evidence to support the view that the major release of iodine into the containment will be in the form of CsI, the completed Phebus FP tests show that a significant proportion of the iodine release was in gaseous form In addition, there is an uncertainty associated with the chemical forms of the constituents of the release from the reactor coolant circuit which affects the quantity of chemicals that is required to achieve a pH in excess of 8

(2) The effect of acid formation and other processes on the pH of the aerosol and of the sump water in sequences where the pH control is not effective should be determined for releases into the reactor building, taking into account the nature of the non-aqueous aerosol

(3) For reactor designs where low pH sequences have a high frequency, validated models are required for reverse partitioning and the return flow of iodine to the boiling sump (either by condensation or “wash-out”) and the pH dependency of iodine behaviour at these low pH values

(4) The extent to which the iodine transferred to water pools will dissolve should be determined for releases to suppression pools If significant pool scrubbing can be established, the following code developments should be addressed

    (a) the effect of the presence of silver on slowing the radiolytic oxidation of iodine,
    (b) the effect of the formation of nitric acid, and competing chemical reactions, on the solubility of the silver aerosol This could greatly increase the formation rate of AgI

(5) Data are required in severe accident conditions to determine the importance of iodinate formation as a result of iodine reactions with ozone The data need to address both the formation of ozone and its reactions with iodine

(6) In plants which use boron carbide as an absorber material, when the core is heating up, boron carbide can form an eutectic melt with the steel cladding surrounding it After cladding failure, direct reactions between boron carbide and the steam present in the surrounding atmosphere have the potential to generate methane This phenomenon could lead to the production of organic iodine Therefore, there is requirement to determine whether methane will be formed by this process and if it will react with iodine

(7) Another area of uncertainty is related to the hydrogen production inside the containment during a severe accident Experiments have shown that hydrogen burns due to the actuation of igniters, can convert a significant amount of iodine into volatile species Further data are required to remove the existing uncertainties including the effects of blank recombiners

(8) The reaction schemes between iodine and the containment surfaces, both in the aqueous and the gaseous phase, need to be better understood in order to determine which kinds of surface have the potential to irreversibly retain volatile iodine

(9) More information is required on the reactions of iodine in the gas phase

8. SPECIFICS OF NEW GENERATION DESIGNS

8.1. GENERAL CONCEPTS OF NEW GENERATION PLANTS

It is expected that the designs of new generation nuclear power plants will build on the design and operating experience of existing plants. It is anticipated that some of these developments will address severe accidents by design, as outlined in the EPRI Utility Requirements Document (EPRI URD) and in the European Utility Requirements Document (EUR).

As economics, in general, and investment protection, in particular, are major concerns for the utilities, in addition to safety, their major priorities include:

- improving the system reliability;
- making the plant more forgiving in terms of the effect of incidents and accidents on both economic penalties and risk;
- decreasing the major contributors to the generation cost, i.e.
  - the investment cost, through simplification of the design,
  - the operational and maintenance costs, and
  - the fuel cycle cost.

The requirement to control severe accidents mainly affects:

- the reactor coolant system (melt retention, depressurization, passive heat removal);
- containment devices (for melt retention and hydrogen mitigation);
- containment design (heat removal, leak rate, fission product pathways).

In pursuing these objectives, some potential design changes will affect the source term, and an integral part of the design process is to ensure that developments do not lead to an unacceptable source term.

For the sake of clarity, a distinction is hereby made between two aspects of the design of new generation nuclear power plants, namely:

- aspects which are not expected to change during the life of the plant, or whose modification would result in heavy costs for the utility (e.g. the RCS water inventory and means for melt retention);
- those modifications which can be introduced during the life of the plant, depending on the implications on cost or regulatory acceptability, e.g. new fuel designs: increasing the burnup, new control rod materials, burnable poisons, new cladding material, and means for hydrogen control such as: recombiners, igniters and post-inertization devices.

The former will be illustrated in Section 8.2 which discusses some examples of specific design features of the AP 600, the EPR, a simplified BWR and the WWER-640 designs, which have the potential to influence the source term. Some aspects of fuel development will be addressed in Section 8.3.

8.2. SPECIFIC CHARACTERISTICS OF NEW GENERATION PLANTS (AP600, EPR, SIMPLIFIED BWR, AND WWER-640) WHICH AFFECT THE SOURCE TERM

8.2.1. Specific characteristics of the AP600

The AP600 design incorporates some specific design features which will affect the source term, including the following:

- The thermal output of the core is less than for most current plants. This is not expected to result in major differences with respect to evaluating the source term. However, the lower core inventory could result in less constraints on the design of some systems due to increased grace periods.
- Safety injection system design:
  - *High and medium pressure components of the system.* These are essentially the core make-up tanks (CMTs) and the accumulators. These are storage tanks of borated water whose function is to provide water to the RCS in the case of a depletion of the inventory in the short term after the accident. As they cannot prevent a core melt and since system actuation is independent of those that are used for long term cooling, they can reasonably be considered to be available during fission product release. Their effect is to provide additional water to the RCS and, thus, to delay core degradation. Therefore, they influence the timing of events but they are not expected to have a major effect on the source term.
  - *Low pressure part of the system.* The IRWST is the permanent source of water to the RCS for the long term removal of decay heat. It therefore must be unavailable for a core melt to occur.
- Flooding the cavity to provide ex-vessel cooling is possible through dedicated lines. If this design provision is shown to prevent penetration of the vessel by the molten core in most or all low pressure sequences reliably, the contribution of MCCI to the source term need not be considered.
- Automatic depressurization system: this engineered safety system is provided for rapid RCS depressurization after a LOCA. Its influence on the source terms defined in this document, is accommodated in the specification of the type of LOCA which should be considered to evaluate the source term. However, the design of the system is such that it provides confidence that high pressure sequences need not be considered in the proposed approach.
- Passive containment cooling system: water supplied from an external storage tank for the first 72 hours, or air in the longer term if the tank is not replenished, provides for decay heat removal from the containment. This heat sink promotes turbulence inside the containment atmosphere and facilitates aerosol growth and gravitational settling.
- pH is maintained higher than 8 in containment water pools by a pH control system. This limits the formation of gaseous iodine, either elemental or organic.
- Igniters and passive autocatalytic recombiners limit H_2 buildup inside the containment and can influence aerosol chemistry, in particular, iodine species.

8.2.2. Specific characteristics of the EPR

The EPR is being designed for compliance with regulatory recommendations, as specified in the GPR-RSK document, and with EUR. The most important features or requirements which have the potential for influencing the source term are:

- High thermal output (about 4900 MW) together with the target of a high burnup (envisaged as 60 000 MW·d/t U) and a large flexibility in the use of MOX fuel (The design target is 50% MOX fuel). This obviously influences the decay heat level, and thus the timing of events, the fission product inventory (long-lived fission products are proportional to the burnup) and the release fraction. The influence of MOX fuel on the composition of the fission products is of minor importance.
- Reliable depressurization system (normal bleed system for design-base accidents and a dedicated bleed system, with the same capacity, for severe accident conditions) provided a high confidence that low pressure sequences will be the only relevant ones for evaluating the source term.
- Ex-vessel melt retention to avoid basemat ablation. This is done in three steps involving the reactor pit and a dedicated spreading area:
    - Retention of the melt in the dry reactor pit to avoid fuel coolant interaction, to collect the whole corium melt, and thus avoiding any late discharge of melt in the presence of water. Special material is foreseen to be dissolved by the melt, oxidizing the remaining Zr, lowering the density of the oxidic phase.

      Major effect on the source term: enclosure of fission products in a glass-like structure.
    - After the dissolution of the sacrificial material, the melt will destroy a thin iron gate and spread — in one event — over a large dry surface (in the order of 170 m^2) which is foreseen to be in a dedicated compartment. Two layers of sacrificial material (one oxidic layer to guarantee that the oxidic phase has a lower density than the metallic phase and one metallic layer to guarantee that in any case, a metallic layer is in contact with the protective layer) and a layer of protective material (based on ZrO_2 to limit the heat transfer to the basemat) are foreseen.

      Major effect on the source term: lowering the temperature.
    - After a significant fraction of the sacrificial material of the spreading compartment has dissolved, with the oxidic melt at the top, the melt will be flooded automatically at a limited rate using water from the IRWST. The subsequent removal of the heat is accomplished first by evaporating this water, which condenses on the walls of the containment and later by activation of the spray system. Fission products which leach out of the melt have to pass through the water layer where they are effectively scrubbed.

      Major effect on the source term: water layer above the melt.
- Recirculation spray using water from the IRWST for fast depressurization of the containment. As the spray system is designed to be operated late in the accident when natural removal mechanisms have already reduced the airborne activity significantly, the spray system will limit the long term activity concentration, which results from resuspension processes.

Passive autocatalytic recombiners and igniters for continuous hydrogen removal from the containment atmosphere. Major effect on source term: superheated atmosphere.

Another feature which is worth noting, although it does not have a direct consequence on the in-containment source term of the evaluation, is the double containment, which consists of pre-stressed concrete and reinforced concrete shells, with a low leak rate for the inner containment (1 vol% per day at a design pressure of 6.5 bar), with sub-pressure in the annulus, which is maintained by an air ventilation system and exhausted to the stack through a filter system. This avoids a ground release, as well as contamination of the surrounding buildings.

For a LOCA, natural processes lead a reduction in the airborne aerosol concentration in the containment by more than three orders of magnitude within 12 h, which is before the spray system is activated. Additional depletion can be gained in the annulus between the inner and outer containments (by between one and two orders of magnitude) and by the filters (by a factor of 100).

8.2.3. Specific characteristics of a simplified BWR

The development of the future small BWR is in the direction of using more passive features for core melt prevention and mitigation. This leads, on the one hand, to a very low core melt frequency and on the other hand, to a high probability that the containment will retain its function after a core melt and thus limit the consequences to the environment, as for future PWRs. For the in-containment source term, the following features are important:

- limited core thermal output (see the AP600) together with a large water inventory, with consequences for the grace periods and timing of events;
- passive emergency core cooling (potential for water injection on the hot core) and dry well flooding for outside cooling of the RPV;
- passive containment heat removal to a water pool above the containment;
- inerted containment to avoid the risk of hydrogen detonation (but the pressure load as well as the effect on heat transfer from the existing hydrogen remains).

8.2.4. Specific characteristics of the WWER-640

Specific characteristics or features of the WWER-640 that could influence the source term evaluation are:

- the limited core thermal output (see AP600);
- the large water inventory in the RCS (and the secondary system) (see AP600 and EPR);
- passive emergency core cooling system with the additional possibility of cavity flooding (see AP600);
- large water inventory inside containment (see AP600 and EPR);
- passive containment cooling system (see AP600);
- double containment system (see EPR).

Table I summarizes the major characteristics of future plants that influence the source term.

TABLE I. CHARACTERISTICS OF FUTURE PLANTS THAT INFLUENCE THE SOURCE TERM

Feature	AP600	EPR	Simplified BWR	WWER-640
Fuel	There is a general tendency towards higher burn-up use of MOX, Hf control rods			
Core thermal output	low	high	low	low
Passive primary systems	core cooling	accumulators	core cooling (emergency condensers)	accumulators, flooding tanks
Melt retention	RPV cooling	Spreading	RPV cooling	RPV cooling
Containment	steel	pre-stressed concrete reinforced concrete	reinforced concrete	steel
Heat removal from the containment	passive (ext cont shell)	spraying	passive (condenser)	passive (ext cont shell)
Hydrogen mitigation	recombiners+ igniters	recombiners+ igniters	inerted	recombiners (igniters?)
Venting	no	no (Annulus ventilation)	yes	no
Suppression pool	IRWST	IRWST	suppression pool	no

8.3. EVOLUTIONS IN FUEL AND CORE DESIGNS

In addition to plant design, the evolution of fuel and core design may have an impact on the in-containment source term. According to present trends, new plants will be introduced very slowly in the next century for economic reasons but new fuel development may be quite extensive because of the large number of existing plants. It is therefore worthwhile to consider the potential for innovation in the light of its impact on the source term.

A change that is desirable for future fuel designs is that the fuel cycle will be considered as a whole entity and not as many separate parts. All parts of the fuel cycle have an effect on each other and will not be considered in isolation. This holistic approach means that fuels will be designed so that the nuclear utility and its customers will receive the largest possible economic benefit by virtue of a framework of a fuel cycle which is optimized in terms of the total cost. It may be expected that future fuels will be easier and cheaper to manufacture, as well as to reprocess or to store, and will provide an overall environmental impact that is as low as reasonably achievable (ALARA). Nevertheless, it is expected that advanced fuel cycles should make maximum use of fissile materials. At present, thermal reactors use only about 3% of the original fissile material in their fuel.

This philosophy provides a driver for the definition of source terms for future reactor systems. It is expected that the following fuel design issues will impact the source term:

- fuel discharge burnup targets will be set at optimum levels, up to 60 000 (MW·d/t U) which increases the fission product inventory (proportional for the long lived nuclides) and increases the release rates (see Section 4);
- MOX fuel cycles will be more widespread with potential consequences on the Pu source term;
- extended use of more burnable absorbers (limited to 2–5% of a PWR and to 50% of a BWR core) with a potential consequence on release rate;
- new rod control cluster assembly (RCCA) designs;
- more RCCAs will be required (for MOX cycles especially). The choice of control rod material has two effects on the source term,
  - provides a source of inactive aerosols (Cd, Ag, H_3BO_3, etc.),
  - influences the liquefaction temperature and thus the relevant melting temperature (B_4C probably leading to the lowest, and Ha to highest temperature to be considered for fission product release),
  - cladding changes based on fuel rod performance considerations. Stainless steel cladding may be considered in this context with some advanced core designs (high conversion reactor). Whereas the hydrogen production (by steam) is quite similar to reactions with Zircaloy cladding, the amount of associated heat. and therefore also the heat-up rate of the core, is much smaller.

9. CONCLUSIONS

The important phenomena which determine both the source term to the containment and the in-containment source term have been addressed. A deterministic approach has been adopted so that guidance has been provided for the relevant phenomena. However, it has been recognized that future designs and accident management measures will reduce to very low values the probability that the vessel will fail at a high reactor coolant circuit pressure and thus, no details have been provided for phenomena that are specifically associated with these sequences.

In the case of both the release to the containment and the behaviour of fission products in the containment, a reasonably bounding approach has been proposed. Numerical values are not given, but a parametric approach has been adopted which will enable to evaluate the source term to the containment. The proposed approach is equally applicable to existing and future generation reactors, but the features that are specific to some future generation designs have been identified.

The major uncertainties which affect the evaluation of the source term to the containment and the in-containment source term have been identified together with the associated R&D requirements.

Therefore, all the objectives that are identified in Section 1 have been met.

9.1. FUEL DESIGN AND CORE INVENTORY

It is recommended that the core inventory of fission products and actinides be evaluated using one of the recognized core inventory computer codes. These codes and their associated databases have been validated for current core and fuel designs and fuel management strategies.

However, it is anticipated that future developments will extend beyond the existing range of validation. Thus, additional validation will be required or engineering judgement will need to be applied which reasonably bounds the associated uncertainties.

9.2. THERMAL HYDRAULICS AND CORE DEGRADATION CONSIDERATIONS

(1) Design differences between current and future plants that may be important for the thermal hydraulics of core degradation are:

- increased water inventory;
- lower power levels;
- higher discharge burnups;
- depressurization of the primary circuit;
- heat removal from the containment;
- passive flooding of the core or the basement of the containment.

(2) A LOCA with no intervention of the ECCS seems to be an appropriate representative sequence to evaluate for determining a reference source term for future reactors.

(3) It is useful to develop simple algorithms for evaluating the effects of different design features on the thermal-hydraulic behaviour important to source term evaluation.

9.3. IN-VESSEL SOURCES

(1) The simple thermal-hydraulic algorithms developed in Section 3 need to be used along with a stand-alone fission product/aerosol model to predict the release from the core. Specific models may be required to describe the late-phase release of tellurium and antimony retained in Zircaloy up to 90% oxidation.

(2) The chemical forms of the fission products that may be released from the RCS for future core designs are likely to be the same as for current core designs.

(3) There are large uncertainties associated with release of materials from control rods and structures and there is a justification for excluding one or both of these from the source term specification. The effect of neglecting the latter is expected to be generally conservative with respect to the robustness of the containment design. Whilst it may be possible to exclude cadmium because of its release early in the accident, the low volatility of silver means that there will be a release over a longer period of time. This will have a significant impact on the containment aerosol behaviour, as well as on the aqueous containment iodine behaviour. Therefore additional considerations of these phenomena may be needed, particularly if they impact the function of important systems.

(4) The role of boric acid from the primary coolant (PWR) or from the B_4C control blades (BWR) should also be considered. This has the potential of influencing the caesium and iodine species formed, and may also affect the containment sump pH.

(5) Whilst the largest release of radioactive products from a molten pool is likely to take place when it is sparged, there is experimental evidence that release will take place where there is no sparging. These data therefore require further assessment.

(6) There is not a unique sequence which results in both a bounding fission product release and the highest concentration of hydrogen in the containment. The intention of the present report is to provide a reasonably bounding value for the latter.

9.4. EFFECTS OF THE RCS ON FISSION PRODUCT BEHAVIOUR

For a low pressure sequence, it is considered reasonable to neglect the effects of the RCS on the reference source term.

9.5. EX-VESSEL SOURCES

(1) The strategy of depressurizing the reactor coolant circuit before the vessel leads to a very low probability of high pressure melt ejection and direct containment heating. Their contribution to the source term for design purposes can be neglected.

(2) If the probability of molten core–concrete interactions is not very low or if concrete is used as a sacrificial material in a core retention device which also will result in sparging, the molten core–concrete interactions must be addressed. The phenomenon is well understood and several codes which have been validated by representative experimental programmes are recommended. These may require significant modification for application to any catcher material other than concrete and which also results in sparging.

9.6. IN-CONTAINMENT BEHAVIOUR

9.6.1. Important phenomena

(1) The most important aspects which affect the in-containment source term are the rate of aerosol deposition, the behaviour of gaseous iodine, the effect of the spray system and, in the case of accidents where the release to the containment is via a water pool, the decontamination factor due to pool scrubbing.

(2) The rate of aerosol deposition with respect to agglomeration and the subsequent gravitational settling can always be assured and credit is taken for these phenomena in the reference source term. They are, however sensitive to the particle size distribution. Because the thermal hydraulic behaviour in the containment is sequence specific, no

credit is taken for the rate of particle growth due to condensation on the aerosol. The latter is also sensitive to the solubility of the initial aerosol.

(3) The production of gaseous iodine is sensitive to the pH of both the aqueous phases namely the aqueous aerosol and the water pools, and will be minimized if the pH of the sump water is controlled to be 8 or greater.

(4) The decontamination factor associated with pool scrubbing is sensitive to the velocity, steam content and size distribution of the aerosol, the flow regime within the water pool and whether or not the pool is boiling.

(5) A spray system can be a very effective means of removing fission products from the in-containment source term. Its effectiveness is sensitive to the size distribution of both the spray droplets and the aerosol. The effectiveness increases as the size of the spray droplets decreases and has a minimum for an aerosol size of approximately 1 micron.

9.6.2. Major uncertainties

(1) If the sump pH is controlled at or greater than 8, the consensus of expert opinion is that the iodine can be modelled as if it behaves as an aerosol. The major uncertainty associated with this approach is the effect of acid formation in the containment atmosphere on the pH of the aqueous aerosol. There is a requirement to determine and model the importance of this effect and the potential for the formation of AgI to retain iodine in the aqueous phases.

(2) If the sump pH is not controlled, there is a requirement to perform a reasonably conservative evaluation of the pH of the aqueous aerosol and water pools and to evaluate the behaviour of the volatile forms of iodine and their contribution to the release from the containment. The main uncertainties are the constituents of the aqueous phases as a function of time, which govern the pH, and the ability of reactions with silver to retain iodine in an involatile form. For designs where the pH is not controlled, there is a requirement to develop modelling for both these effects.

(3) There is a good database for pool scrubbing in bubbly conditions which are applicable to many BWR conditions but there are less data for churn turbulent conditions. Thus, the major uncertainties associated with pool scrubbing are the size distribution of the aerosol, decontamination factors in churn-turbulent regimes and the effect of saturated and boiling pools.

9.7. THE DESIGN OF NEW GENERATION PLANTS AND FUELS

Some aspects of the designs of both the designs of new generation plants and their fuels are expected to have a significant impact on the source term. Because it is a requirement to address severe accidents in the design of future reactors, the overall effect of changes in the plant design is expected to reduce the source term. Some of the anticipated changes to the design of the fuel and its management are expected to increase the fission product and actinide inventories, but source term considerations are not expected to rule out any developments that are currently envisaged.

Examples of potential changes with respect to the design are:

— an increased water inventory in the RCS which will delay the time at which the release from the fuel occurs and

— quenching of the molten corium if it penetrates the reactor pressure vessel which will greatly reduce the probability of any ex-vessel contribution to the source term.

Examples of potential changes with respect to the fuel are

- higher burnups than are utilized today and
- changes to the designs of the control rods and burnable poisons and the introduction of new cladding designs

Some of the developments in fuel design may extend beyond the existing range of validation of the codes and data that are used to evaluate the core inventory Thus, an extension to the validated database will be required

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Appendix

APPLICATION EXAMPLE FOR THE SIMPLIFIED SOURCE TERM METHODOLOGY

A.1. INTRODUCTION

The TECDOC to which this appends proposes a simplified source term methodology as being a viable approach for developing plant specific design source terms. The proposed methodology uses simple algorithms to quantify an associated reference sequence "generalised thermal transient" that is to be used along with an appropriate fission product release model to develop the source term. The method is intended to accommodate different design options for future nuclear power plants (NPPs).

The proposal spelled out in the TECDOC is an innovative approach not known to have been used previously. Consequently, this appendix was developed to provide an example application of the methodology — hopefully to demonstrate its validity and usefulness.

The approach taken in this appendix for the application of the example was to develop a PC computer model for quantification of the proposed "generalized thermal transient" of Section 3 using the simple algorithms suggested. The computer model developed [Simplified Source Term (SST)] required a fission product release model to "translate" the generalized thermal transient into the desired source term. Two optional models for fission product release were included in SST: (1) the CORSOR-M model currently in MELCOR, and (2) a new model called the Kress/Booth RelVol model.

The CORSOR-M model was included in SST to allow benchmarking the derived source term because the CORSOR-like models were used [along with the Source Term Code Package (STCP) and along with MELCOR] to "initialize" best-estimate values for the expert elicitation process of NUREG-1150 that went into quantifying the STs and uncertainties on which the NUREG-1465 source terms were based.

The new RelVol model was included because of the now widely known tendency for the various CORSOR models to significantly overestimate the releases of volatile fission products compared with experimental data and with the calculated results of more sophisticated fission product release models such as FASTGRASS. In addition, the RelVol model includes a correlation for the effects of burnup which will be required if SST is to be used for estimating source terms for future NPPs (CORSOR does not have such a correlation). The RelVol fission product release model is much more accurate in predicting experimental data for fission product release than is CORSOR and is just as easy to implement.

The Kress/Booth aspect of RelVol for Cs release was developed as a stand-alone model (called ASSTCs) that will accept arbitrary thermal transient input (as a series of linear functions) for purposes of comparison with experimental data. This ASSTCs Kress/Booth model was then compared with the Cs release results from the ORNL VI-3 experiment to illustrate the validity and accuracy of the model.

The SST code, with the complete Kress/Booth RelVol and the CORSOR-M models, was used to develop the source term for a large-break loss of coolant accident (LBLOCA) sequence for the SURRY PWR plant. SURRY was selected for this example application because it was one of the reference plants used in NUREG-1150 on which the PWR source term in NUREG-1465 was based. The resulting calculated source terms using both RelVol and CORSOR-M were then compared with the NUREG-1465 values to illustrate the efficacy of the simplified source term methodology.

A 2 SIMPLIFIED SOURCE TERM COMPUTER CODE DESCRIPTION

The SST code is written in the QBASIC language because that happens to be convenient for use on most PCs with WINDOWS applications A listing of the SST code and the stand-alone ASSTCs code is included at the end of this appendix

The SST code models the simple algorithms proposed in Section 3 of the TECDOC to which this appends to quantify the generalized thermal transient The various parts of the calculational process for the generalized thermal transient and the resulting source terms are described in this section

A.2.1. Time to start of core uncovery

The time to start of core uncovery (when the water level has reached to top of the active core) includes the blowdown time (following a postulated break or on automatic depressurization) and the time required to boil off the residual RCS water after blowdown

For a LBLOCA (double ended break in the largest RCS pipe), the blowdown is fast and is generally insignificant compared with the subsequent boil-off time It may generally be estimated at a typical value of seconds (e g about 20 s) It is included in SST just for completeness and was input for the example case at 20 seconds For an advanced plant design with automatic depressurization, this can be design specific and will require estimating It is treated as an input value in SST

The subsequent boil-off time increment, T_u, is calculated from the algorithm suggested in Section 3 1 1

$$\int_{Tbd}^{Tu} Q(t)dt = \omega\Delta H_{fg} \qquad \text{(A1)}$$

The quantification of this algorithm for T_u clearly requires a correlation for the decay heat as a function of time The decay heat would, to some extent, depend on the core design as well as burnup

For SST, use was made of a correlation developed by Sandia National Laboratory (SNL) for the MELCOR code SNL used the "Sandia-ORIGEN" code to calculate the decay-heat power at the end of a three-batch PWR equilibrium cycle [A 1] For use in SST, the results of this calculation were converted to fraction of the original power and the calculated curve for the decay heat fraction (a function of time after scram) was "fit" by the following correlation

For $0 < T \leq 10^4$ s, $f_{dh} = [0\ 019 - (0\ 00255 / 2\ 3)\ln T]$
For $T > 10^4$ s, $f_{dh} = [0\ 06 - (0\ 0128 / 2\ 3) \ln T]$

where T is in seconds and f_{dh} is the decay heat as a fraction of the nominal full power

As the above decay heat correlation is of the form Q(T) = A-BlnT, then Eq (A1) was first integrated to give

$$\int_{Tbd}^{Tu} (A - B\ln T)dT = A(T_u - T_{bd}) - B[T_u \ln T_u - T_u] + B[T_{bd} \ln T_{bd} - T_{bd}]$$
$$= \omega\Delta H_{fg} / K \qquad \text{(A2)}$$

where K is needed for the conversion of units and includes the input nominal core power. The quantity of residual RCS water, ω, is a code input and the ΔH_{fg} is assumed to be a value for saturated water at 600°F[1] — a typical end-of-blowdown RCS water temperature.

Equation (A2) was implemented in SST with a convergence routine to determine T_u.

A.2.2. Adiabatic heat-up rate averaged for whole core

As the core uncovers, the fraction of the total core afterheat that goes into boiling of additional water decreases. Consequently, developing this time increment is more complicated than for the previous boil-off period. In addition, the first region of the core that uncovers enters its "adiabatic" heat-up phase first. After full uncovery, the bottom portion of the core then enters its "adiabatic" heat-up phase at a slightly later time and at a lower decay heat level than the top layer.

Because the time for complete core uncovery is generally short compared with the subsequent adiabatic heat-up time, a representative whole-core averaged adiabatic heat-up rate was determined that includes the decrease in decay heat level as the core uncovers and as various portions of the core heat up to the runaway Zr oxidation temperature. The averaging process used in SST for this is as follows:

(a) The decay heat level at the time the RCS water reaches the top of the core is first used to calculate the time (ΔT) to heat the core adiabatically to the temperature assumed for start of runaway Zr oxidation.
(b) A new decay heat level is then calculated at $T_u + \Delta T$ and linearly averaged with the decay heat level in step (a).
(c) Steps (a) and (b) are iterated on until convergence. The "average" decay heat level is then considered applicable for the *top* portion of the core.
(d) The averaged decay heat level determined in (c) is then used to calculate a core uncovery time, Δt_u.
(e) An "initial" decay heat value for the bottom of the core is determined for time $T_u + \Delta t_u$. Steps (a)–(c) are repeated to converge upon an "average" decay heat representative of the adiabatic heat-up of the *bottom* of the core.
(f) The top-of-core averaged decay heat and the bottom-of-core averaged decay heat are further linearly averaged together to calculate the average whole core adiabatic heat-up rate for the generalized thermal transient quantification. Essentially, then, the core uncovery time is considered to be captured in the averaged heat-up time.

A.2.3. Heat-up rate for runaway Zr oxidation and hold time at melt

Experience with fission product release models has shown that the release of the volatiles is dominated by this phase of the generalized thermal transient. Consequently, there needs to be a technical basis for establishing the associated rate of heat-up that is rooted in the Zr-steam oxidation behaviour and kinetics.

The Zr-steam oxidation kinetics that resides in the Source Term Code Package (STCP) is still the standard that is used in most severe accident codes. These kinetics are discussed in Ref. [A.2].

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[1] i.e., 316°C. In fact: Tc = (5/9) × (Tf-32) and Tf = (9/5) × Tc+32, where Tc = temp. in degrees Celsius and Tf = temp. in degrees Fahrenheit.

One of the options in the STCP represents a limiting rate of heat generation given by

$$Q = (8.742 \times 10^{6})(1.849 \times 10^{-7})T^{0.68}/(2R^{2}) \quad [\text{BTU/min.ft}^{3}] \tag{A3}$$

where Q is the volumetric heat generation rate, T is the temperature, and R is the fuel pin radius.

This correlation is based on the rate limiting step being diffusion in the solid state along with "Cathcart-Powel" chemical kinetics, no hydrogen blanketing, and no oxide layer build-up.

With this correlation for the heat generation rate, an adiabatic heat-up rate characteristic of the runaway Zr oxidation phase would be

$$\text{rate} = Q/\rho C_p = (8.742 \times 10^{6})(1.849 \times 10^{-7})T^{0.68} / (2R^{2}\rho C_p)(60) \ (^\circ\text{F/s}) \tag{A4}$$

Using, for UO_2, $\rho = 10150$ kg/m^3, $C_p = 500$ joules/kg.K, and R = 0.422/12 ft, Equation (A4) gives a heat-up rate at the temperature of start of the runaway Zr oxidation phase (2780°F = 1527°C) of

rate = 32°F/s (17.8°C/s)

At the temperature assumed for the end of runaway Zr oxidation (4868°F), Equation (A4) gives

rate = 46°F/s = 25.5°C/s.

The SST code allows the rate of heat-up during the runaway Zr phase to be input. It is recommended that this input be the average of the above two rates (i.e. 39°F/s = 21.7°C/s). This rate is consistent with experimentally observed rates and rates calculated by several core degradation computer codes.

In SST, the hold time at melt simply assumes the same heat generation rate as represented by the input runaway Zr heat-up rate and holds the fuel together until fully molten — that is until the adiabatic addition of this heat is sufficient to equal the heat required for melting the fuel. This quantity of heat is equivalent to an amount required to increase the UO_2 /clad temperature by 1029°F (553.9°C). Consequently, SST calculates the hold time as being 1029 divided by the runaway Zr-oxidation heat-up rate that is input into the code.

A.2.4. Kress/Booth RelVol fission product release model

For the calculation of source terms associated with severe accidents, it is necessary to model the release of fission products from fuel as it heats and melts. Some exist relatively sophisticated models (e.g. FASTGRASS [A.3]) deal with: diffusive transfer of fission product species within grains to internal fission gas bubbles and to grain boundaries; diffusive transfer of bubbles to grain boundaries; diffusion of fission product species along grain boundaries and through interlinked cracks; growth of the grain size with irradiation and with time at temperature, and the effects of such grain growth on the transport.

The experimental evidence is persuasive that the above are the appropriate phenomena to be modelled as important in the release process. Nevertheless, it has been found convenient to have much simpler models which attempt, in some simple empirical manner, to capture the essence of these phenomena (which can be described as "diffusive-like" phenomena acting in series). Consequently, it is believed that a correlation (of experimental data) which has a diffusive-like character would be an appropriate choice for a simple empirical fission product release model.

Kress and co-workers at ORNL [A.4] have proposed such a model in which the fission product release would be expressed by a "spherical" diffusion equation

$$\partial C/\partial t = (D/r)[\ \partial^2(rC)/\partial r^2\ ] \qquad \text{(A5)}$$

where $D = D_0\exp(-Q/RT)$

T is the temperature, and D_0 and Q are arrhenius-like correlation parameters.

D is a sort of "effective" diffusion parameter to be used along with an "effective" spherical distance parameter, $r_0 = a$, which together capture the operative multiple series diffusion mechanisms discussed previously.

Such an empirical correlation equation can be solved in a computer code by simple numerical time integration methods for the various fission product species if the temperature transient is specified and if values are known for D_0, Q, and a. Fortunately, it is not necessary to resort to numerical integration because expression (A5) has been prominent in the engineering literature for many years and approximate explicit solutions exist for constant D. Booth and Rymer [A.5] proposed an approximate solution

$$\text{for } Dt/a^2 \le 0.1,\quad f = 6(Dt/\pi a^2)^{1/2} - 3Dt/a^2 \qquad \text{(A6)}$$

$$\text{for } Dt/a^2 > 0.1,\quad f = 1 - (6/\pi^2)\exp(-\pi^2 Dt/a^2)$$

where f is the cumulative fraction of the original quantity of the species present at time = 0 that has been released from the fuel up to time t in seconds, a is the effective spherical parameter in centimetres, and D is in cm^2/s.

Generally, severe accident conditions have the fuel temperature varying with time (and thus also D). An effective approach for dealing with the transient nature of D is to break the transient into very small time increments, Δt, and use and effective Dt in (A6) that is a summation

$$Dt = \sum D(t)\Delta t$$

in which D is a function of time because temperature is a function of time. This approach can actually be considered as part of the empirical correlation if it is used to determine best-fit values for D_0 and Q from transient experimental data.

Using the above approach, data for the release of Cs in the ORNL fission product release program [A.6] were "best-fit" to obtain

$$D_0 = (2.6833 \times 10^5)\exp[-6.052 \times 10^{-4}(BU)] \qquad \text{(A7)}$$

$$Q = (2.065 \times 10^5) - (3.629)(BU)$$

and

$$a = 6.0\ \mu m,$$

in which BU is the burnup in MW·d/t, D_0 is in cm^2/s and Q is in cal/mole.

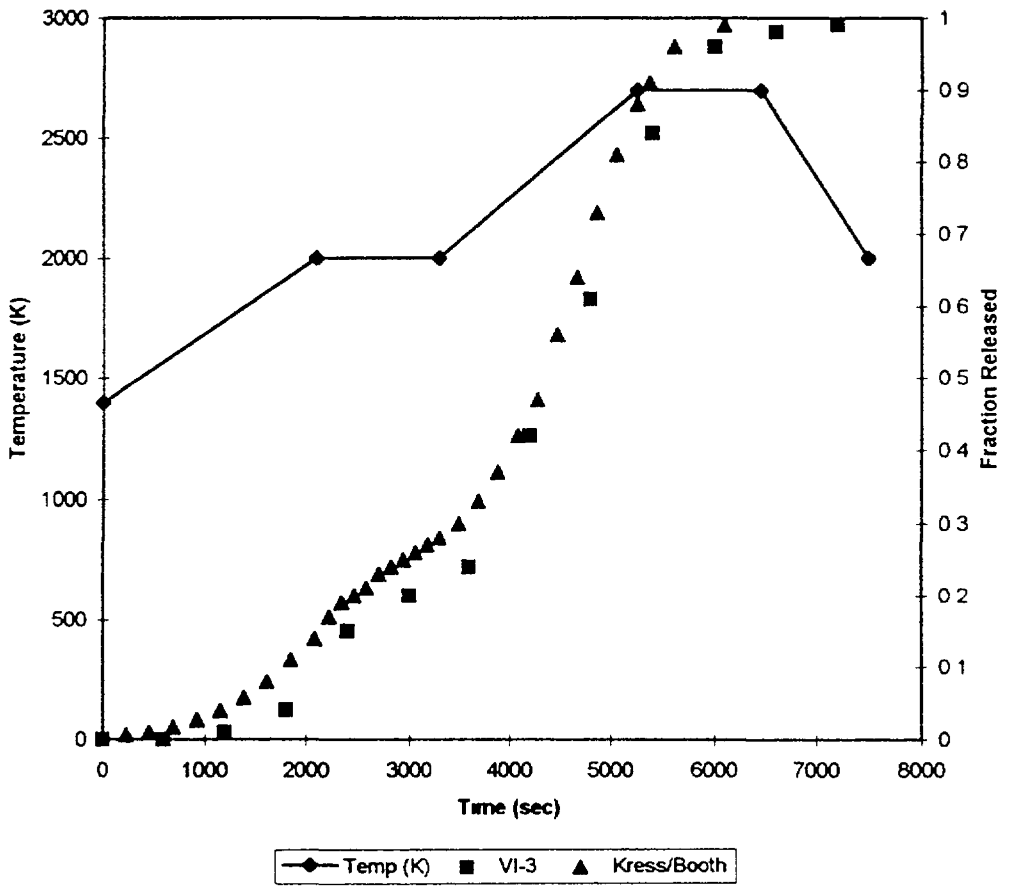

FIG A 1 Comparison of Kress/Booth Cs release model with ORNL VI-3 test data

To give an indication of the accuracy of this correlation in fitting the ORNL data, the ASSTCs stand-alone code was developed which uses this correlation with any arbitrary set of input linear temperature transient phases The results of applying ASSTCs to the ORNL VI-3 test [A 6]) are illustrated in Fig A 1 and show that the correlation agrees well with these test data (this is equally true for the other tests in the series) and tracks the time-trends extremely well

Because the (A5) correlation makes use of the Booth-Rymer approximate solution to the diffusion equation, this part of the new fission product release model (which includes the empirical correlations for "D_0 ", "Q", and "a") is called the Kress/Booth model If sufficient empirical data were available to fit appropriate values for "D_0 ", "Q", and "a" as functions of temperature and burnup for all of the fission product species of interest (mostly the volatile ones), this could serve as a viable empirical fission product release model Unfortunately, however, such an extensive database does not exist

The database that does exist for fission product release, however, is substantial and has been widely reported in the literature in terms of the fraction (of the original content) released over a given amount of time for experiments that may involve various combinations of temperature transients It would be of great utility if this database could some way be captured for use along with the Kress/Booth empirical diffusion-like correlation Kress [A 4] has proposed a way to do this based on a remarkable observation related to the bulk of the experimental database Lorenz [A 7] observed that there is a consistency in the fission product and structural release data when reported in terms of the above "release rate coefficient" defined as $\Delta M/M_0 \Delta t$ (fraction released/time increment) This is the integral release fraction for fuel specimen heated to very high temperatures divided by the total time of the experiment. Figures A.2 and A.3 of [A.7] illustrate the observed consistency. If the results of any particular tests for the measured release rate coefficient are placed on a logarithmic scale ordinate in such a way that they form a straight line, then the resulting positions of the constituents on the linear-scale abscissa defines what we term here as a "relative volatility" for each. Lorenz observed that, if one preserves the same relative location for the various species on the abscissa, then the observed release rate coefficient data from essentially all experiments also fall on straight lines (albeit at different slopes) regardless of the nature of the transient.

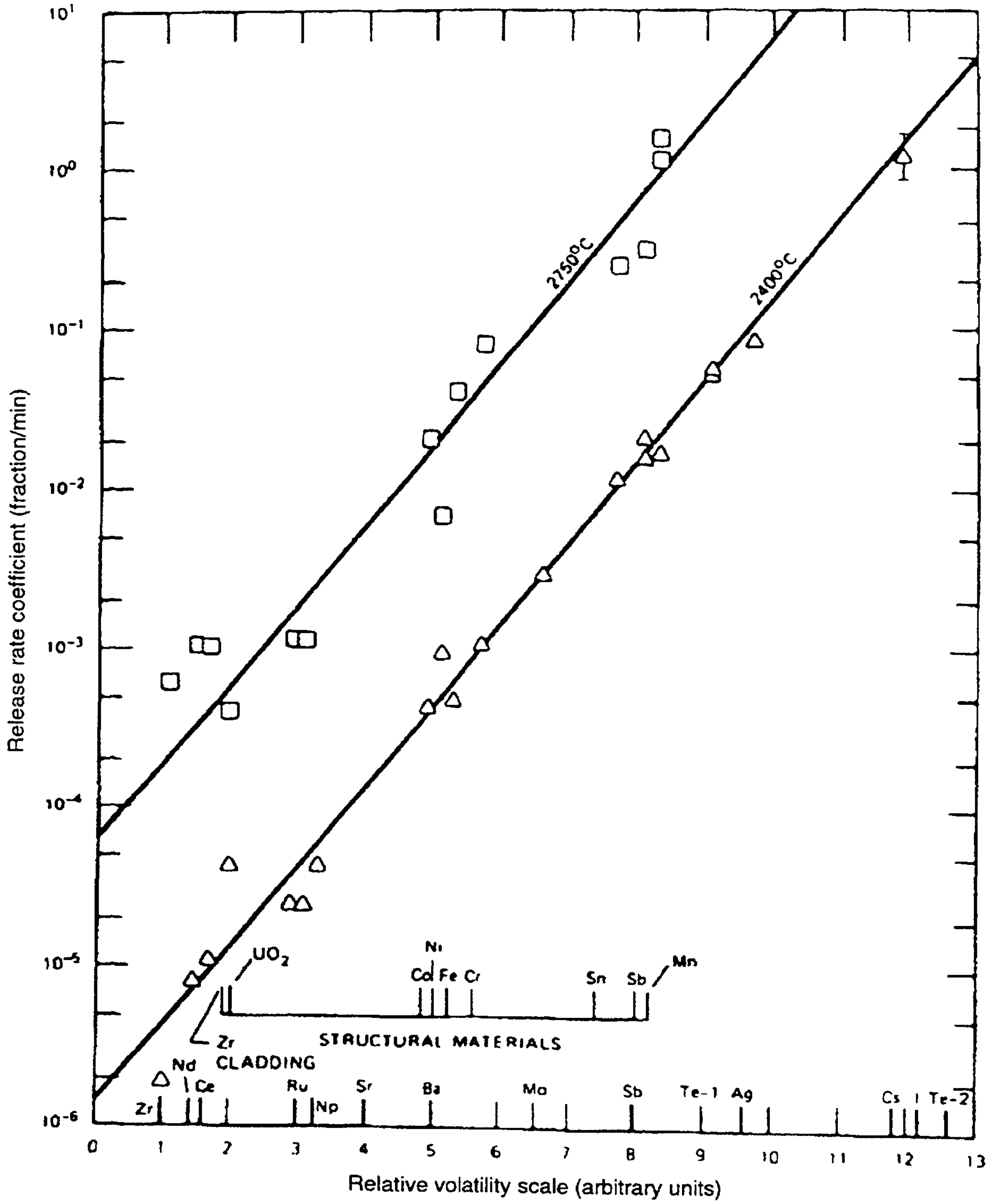

FIG. A.2. Fission product/structure release rate coefficient data measured in Sascha tests in air at 2400°C and 2750°C illustrating the 'relative volatility' scale.

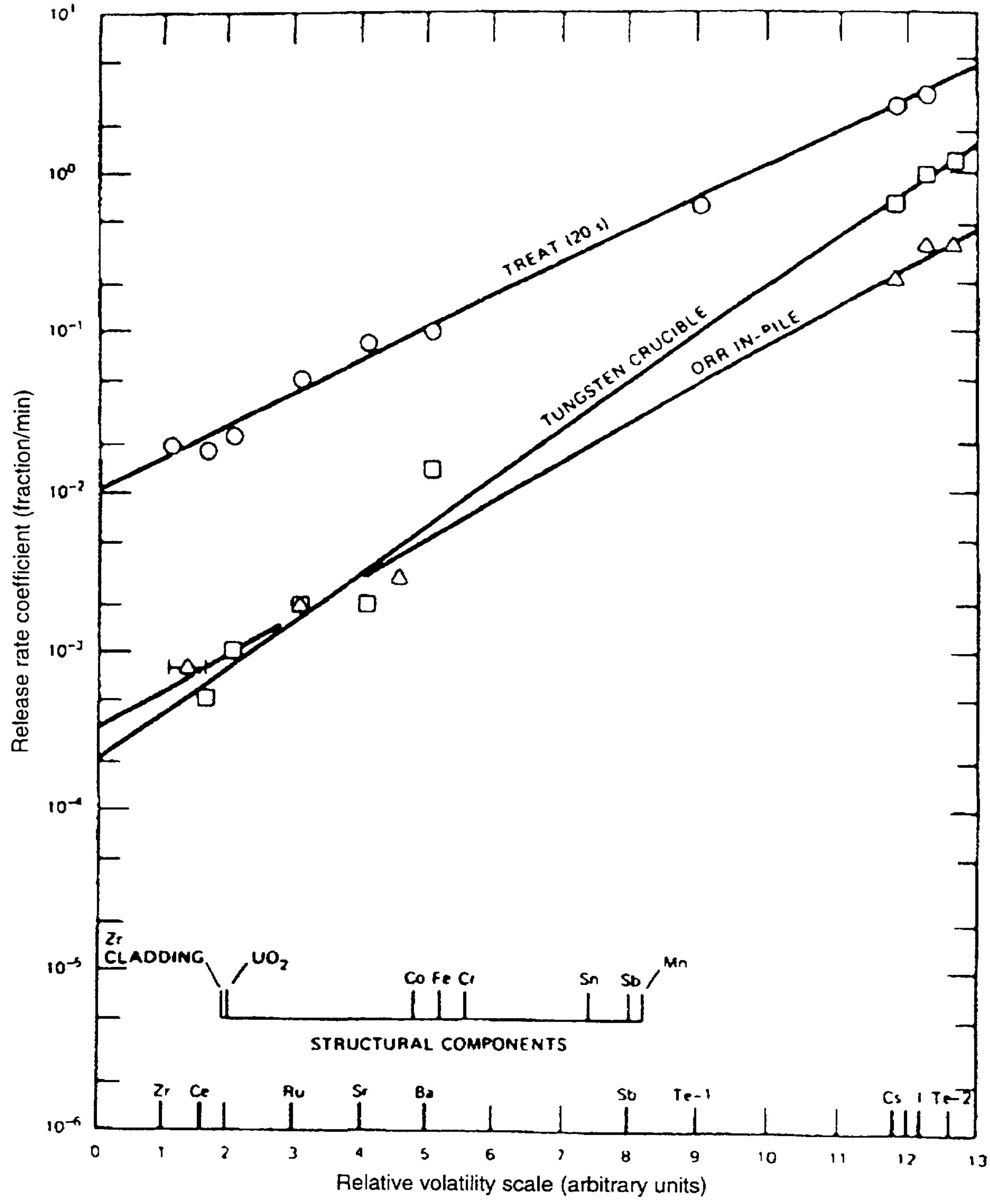

FIG. A.3. Fission product/structure release rate coefficient data from a variety of tests illustrating the consistency of the 'relative volatility' concept.

If use could be made of this somewhat universal behaviour, it would present us with a powerful tool to basically incorporate the bulk of the fission product release data into an empirical correlation. Kress observed that all that is needed to do this is to establish the release of one species and the slope of the line for any particular temperature transient. The release of all other species can be interpolated using the relative volatility scale. This can be done by using the Kress/Booth release model if there were an empirical correlation for "D_0 ", "Q", and "a" for one other species to go with the Cs correlation reported above (two points establish the slope of a straight line; we only need to know the release values for two elements — we can then interpolate using the relative volatility scale to find the release values for all the other constituents).

In the ORNL fission product release experiments, the experimental on-line time release behaviour of Cs was found by continuous reading of the downstream of the test chamber. As the real fuel specimens used in the test had aged considerably since their original irradiation, the activity measurements were dominated by Kr and Cs, so these were the only two elements for which continuous measurements were made. The "relative volatility" of Kr lies much too close to that of Cs for it to be a good second standard for the Kress/Booth model. Fortunately, the ORNL tests obtained limited time-correlated release data for other species by using three separate independent collection trains operated at different times during the tests. While these data are limited, they were sufficient to develop a best-fit Kress/Booth correlation for antimony as shown below.

For Sb:

$$a = 6\ \mu\text{m}$$

$$D_0 = (3.4608\times10^6)\exp[(-6.052\times10^{-4}(BU)]\quad (\text{cm}^2/\text{s}) \qquad \text{(A8)}$$

$$Q = (2.494\times10^5) - 3.629(BU) \quad (\text{cal/mole})$$

With the two standards (Cs and Sb) for the Kress/Booth model and the quantification of the generalized thermal transient, SST uses the "relative volatility" scale of Table A.1 to establish the fractional release of all species (hence the designation of the full fission product release model as being the Kress/Booth RelVol model).

TABLE A.1. RELATIVE VOLATILITY (RV) VALUES

Species group	RV
NG	1.1
Te	1.07*
I	1.03
Cs	1.00
Sb	0.68
Ba	0.42
Sr	0.34
Ru	0.25
La	0.14
Ce	0.085
Structure materials	
Mn	0.69
Sb	0.68
Sn	0.63
Cr	0.47
Fe	0.44
Co	0.41
UO_2	0.17

*For oxidized clad.

Recognizing that the release rate coefficients (RRC) lying on the straight lines of Figs A 2 and A 3 are related to the fractional release by definition

$$\mathrm{RRC} = f/\Delta t,$$

and knowing the calculated fractional releases and relative volatilities for Cs and Sb, a straightforward interpolation can be used to determine the releases for the other fission product groups as follows

$$f(i) = f(\mathrm{Cs})[f(\mathrm{Cs})/f(\mathrm{Sb})]^{(\mathrm{RV}(i)\ \mathrm{RV}(\mathrm{Cs}))/(\mathrm{RV}(\mathrm{Cs})\ \mathrm{RV}(\mathrm{Sb}))} \tag{A9}$$

Relationship (A9), along with (A8), (A7), (A6) and Table A 1, constitute the full Kress/Booth RelVol fission product release model It is noted that SST only deals with fission products The RelVol model, however, can easily be expanded to include the release of structural materials as their "relative volatilities" are included in Table A 1

A.2.5. CORSOR-M fission product release model

For comparison purposes, SST also includes the CORSOR-M fission product release model which currently resides in the MELCOR code.

CORSOR uses an empirical correlation of fission product release data in the form

$$dM(i)/dt = -K(i)M(i)$$

in which the release coefficients, K(i), for the various fission product species are correlated (from experimental data) as functions of temperature

$$K(i) = K_0(i)\exp[-Q(i)/RT]$$

CORSOR-M uses the following values for the various fission product species groups

Examination of Table A 2 reveals that CORSOR-M will predict equal fractional releases (source terms) for NG, Cs, I, and Te

TABLE A 2 CORSOR-M VALUES FOR RELEASE RATE COEFFICIENT PARAMETERS

Class identifier	Class	K_0 (min^{-1})	Q(kcal/mole)
1	NG	2 00E5	63 8
2	Cs	2 00E5	63 8
3	Ba, Sr	2 95E5	100 2
4	1	2 00E5	63 8
5	Te	2 00E5	63 8
6	Ru	1 62E6	152 8
7	Mo	–	–
8	Ce	2 67E8	188 2
9	La	–	–
10	U	1 46E7	143 1
11	Cd, Sb	–	–
12	Sn	5 95E3	70 8

The implementation of CORSOR-M in SST is similar to that of the Kress/Booth model in that a simple trapezoidal numerical integration in time is made for the classes of interest above (Group 1 = NG, I, Cs, & Te; Group 2 = Ba, Sr; Group 3 = Ru; and Group 4 = Ce).

A.3. APPLICATION OF SST TO SURRY LBLOCA FOR COMPARISON WITH THE NUREG-1465 SOURCE TERMS

One purpose of developing SST was to conduct an application of the simplified source term methodology to demonstrate its usefulness. For this example application, a large break loss of coolant accident (LBLOCA) sequence for the SURRY PWR plant was selected because SURRY was one of the "reference" plants used in the NUREG-1150 risk analysis on which the NUREG-1465 source terms were based. A LBLOCA was selected because LBLOCAs result in shorter time durations and larger source terms than do small break LOCAs and, consequently, were given more weight by the developers of the NUREG-1465 source terms.

The required input for SST for this example application was obtained primarily from BMI-2104 [A.8] and consists of the following:

TABLE A.3. SST INPUT FOR SURRY LBLOCA

Burnup	30 000 (MW·d/t)
Normal operating power	2441 (MWth)
Blowdown time	20 (s)
Core fuel/clad MC_p	25 428 (BTU/F)
Residual RCS water to top of core	348 200 (lb)
Residual water from top of core to bottom of core	25 000 (lb)
Value for runaway Zr oxidation heat-up rate	38 (F/s)

The printed output of SST for this case is attached to this appendix for both the RelVol and CORSOR-M results. These are compared in Table A.4 with the NUREG-1465 PWR Source Term.

A.4. OBSERVATIONS AND RECOMMENDATIONS

From the printed outputs for the sample application, it is clear that CORSOR-M predicts higher releases than RelVol and does not differentiate among NG, I, Cs, & Te (the NG result in CORSOR-M is for Kr whereas the NG result in RelVol is for Xe).

It is also clear that insignificant release occurs up to the point of runaway Zr oxidation and there is a significant increase in the release (for both CORSOR-M and RelVol) for the period we have called "hold time at melt". This author currently believes that neither the CORSOR-M model nor the RelVol model is appropriate for that phase and, in fact, that fission product release will virtually "shut off" from molten fuel (this is because of the change in both diffusion path length and diffusion coefficient). The recommendation at this time is to use the release only up to the end of runaway Zr and not include the release from the hold-time at melt. Consequently, the full release will be dominated by the runaway Zr-oxidation period. The value for the rate of heat-up (39 F/s) recommended in this Appendix is believed to be an appropriate choice. However, it should be realized that use of this rate in SST also

TABLE A.4 COMPARISON OF PWR TO-CONTAINMENT SOURCE TERMS

	NUREG-1465 (early in vessel)	NUREG-1465 (late in vessel)	NUREG-1465 1+2	SST* RelVol with melt phase	SST* RelVol with melt phase	SST** C-M w/o melt phase	SST*C M with melt phase
Duration (h)	1.8	10.0	1.8	1.8	~1.8	1.8	~1.8
NG	0.95	0	0.95	0.47	0.87	0.51	0.92
I	0.35	0.1	0.45	0.27	0.53	0.51	0.92
Cs	0.25	0.1	0.35	0.22	0.43	0.51	0.92
Te	0.05	0.005	0.055	0.37	0.71	0.51	0.92
Ba, Sr	0.02	0	0.02	0.002	0.007	0.002	0.007
Ru	0.0025	0	0.0025	0.007	0.002	7×10^{-7}	5×10^{-6}
Ce	0.0005	0	0.0005	0.0002	0.0007	2×10^{-7}	2×10^{6}
La	0.0002	0	0.0002	0.0003	0.001	–	–

* All SST calculations were done for a LBLOCA in SURRY with avg. burnup of 30 000 MW·d/t.

** This is the recommended model for the determining advanced reactor source terms

*** This is believed to be the basis for NUREG-1465.

assumes adiabatic heat up. If there are significant heat losses to the steam, then the heat-up rate could be slower. Slower heat-up rates result in greater fission product release because longer times are spent at any given temperature during the transient. In addition, the results of recent experimental programmes need to be assessed before release from molten systems without sparging can be discounted.

Overall, the results of SST using the generalized thermal transient and the Kress/Booth RelVol fission product release model appear to be reasonable and proper for use to develop plant specific design basis source terms for future nuclear power plants.

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[A.8] GIESEKE, J.A., CYBULSKIS, P., DENNING, R.S., KUHLMAN, M.R., LEE, K.W., Radionuclide Release under Specific LWR Accident Conditions, Rep. BMI-2104, Vol. 5, Battelle Columbus Labs, OH (1984).

Annex

CODE LISTINGS AND CALCULATION RESULTS

```
LPRINT "                              SST"
LPRINT
LPRINT
LPRINT
CSE = 1
DIM N(30), RV(10), FCS(5), FSB(5), F(5, 10), FA(5, 10), FB(5, 10), TOTAL1(5)
DIM TOTAL2(5), Q(5), K(5), BAR(5, 10), TERM2(5), NN(50)
DIM F1(5), F2(5), F3(5)
10 INPUT "WHAT IS THE NUMBER OF TIME INCREMENTS FOR FP RELEASE"; NT
INPUT "WHAT IS THE AVG. CORE BURNUP (MWD/T)?"; BU
INPUT "WHAT IS THE REACTOR NORMAL OPERATING POWER (MWth)?"; P
INPUT "WHAT IS THE BLOWDOWN TIME (S)?"; TBD
INPUT "WHAT IS THE TOTAL CORE FUEL/CLAD MCP (BTU/F)?"; MCP
INPUT "WHAT IS THE RESIDUAL WATER MASS TO THE TOP OF THE CORE (LB)?"; WTC
INPUT "WHAT IS THE RESIDUAL WATER TO CORE UNCOVERY (LB)?"; WU
INPUT "WHAT FRACTION OF THE TOTAL CORE IS ASSUMED TO MELT?"; FCM
INPUT "WHAT IS THE RUNAWAY ZR OXIDATION HEATUP RATE (F/S)?"; ZRC
INPUT "TO USE REL-VOL FP REL., INPUT 1. TO USE CORSOR-M, INPUT 2"; REL

LPRINT "THIS IS CASE"; CSE
LPRINT
LPRINT "BURNUP (MWD/T) ="; BU
LPRINT "NORMAL OPERATING POWER (MW) ="; P
LPRINT "INPUT BLOWDOWN TIME (S) ="; TBD
LPRINT "CORE FUEL/CLAD MCP (BTU/F) ="; MCP
LPRINT "RESIDUAL RCS WATER TO TOP OF CORE (LB) ="; WTC
LPRINT "RESIDUAL WATER FROM TOP OF CORE TO COMPLETE CORE UNCOVERY (LB)="; WU
LPRINT "FRACTION OF CORE ASSUMED TO PARTICIPATE IN MELT ="; FCM
LPRINT "VALUE ASSUMED FOR RUNAWAY ZR OX. HEATUP RATE (F/S) ="; ZRC
LPRINT
'CALCULATION TO DETERMINE GENERALIZED THERMAL TRANSIENT FOLLOWS'

'BOILDOWN TIME (TBOIL) (S)'

C2 = P * 3414.4 * 1000! / 3600!
R = WTC * 548.5 / C2
LL = 0
RR = 0
TBOILT = TBD + 100!
20 IF TBOILT > 10000! THEN GOTO 21
L1 = (.06 + .0128 / 2.3) * (TBOILT - TBD)
L2 = (.0128 / 2.3) * TBOILT * LOG(TBOILT)
L3 = (.0128 / 2.3) * TBD * LOG(TBD)
L = L1 - L2 + L3
IF (R - L) < 0 THEN GOTO 22
TL = TBOILT
TBOILT = TBOILT + 100!
LL = LL + 1
IF (LL > 0) AND (RR > 0) THEN GOTO 23
GOTO 20
21 L1 = (.06 + .0128 / 2.3) * (10000! - TBD)
L2 = (.0128 / 2.3) * (10000!) * LOG(10000!)
L3 = (.0128 / 2.3) * (TBD) * LOG(TBD)
L4 = (.019 + .00255 / 2.3) * (TBOILT - 10000!)
L5 = (.00255 / 2.3) * (TBOILT) * LOG(TBOILT)
L6 = (.00255 / 2.3) * (10000!) * LOG(10000!)
L = L1 - L2 + L3 + L4 - L5 + L6
IF (R - L) < 0 THEN GOTO 22
TL = TBOILT
TBOILT = TBOILT + 100!
LL = LL + 1
IF (LL > 0) AND (RR > 0) THEN GOTO 23
GOTO 20
22 TR = TBOILT
RR = RR + 1
23 TBOILT = (TR + TL) / 2!
IF TBOILT > 10000! THEN GOTO 25
L1 = (.06 + .0128 / 2.3) * (TBOILT - TBD)
L2 = (.0128 / 2.3) * TBOILT * LOG(TBOILT)
L3 = (.0128 / 2.3) * TBD * LOG(TBD)
L = L1 - L2 + L3
IF ABS((R - L) / R) < .001 THEN GOTO 40
IF (R - L) < 0 THEN GOTO 24
```

```
TL = TBOILT
GOTO 23
24 TR = TBOILT
GOTO 23
25 L1 = (.06 + .0128 / 2.3) * (10000! - TBD)
L2 = (.0128 / 2.3) * (10000!) * LOG(10000!)
L3 = (.0128 / 2.3) * (TBD) * LOG(TBD)
L4 = (.019 + .00255 / 2.3) * (TBOILT - 10000!)
L5 = (.00255 / 2.3) * TBOILT * LOG(TBOILT)
L6 = (.00255 / 2.3) * (10000!) * LOG(10000!)
L = L1 - L2 + L3 + L4 - L5 + L6
IF ABS((R - L) / R) < .001 THEN GOTO 40
IF (R - L) < 0 THEN GOTO 26
TL = TBOILT
GOTO 23
26 TR = TBOILT
GOTO 23
40 TBOIL = TBOILT

'ADIABATIC HEAT-UP RATE AVERAGED FOR WHOLE CORE (B1) (F/S)'

IF TBOIL > 10000! THEN GOTO 41
A = .06: C1 = (.0128 / 2.3)
GOTO 42
41 A = .019: C1 = (.00255 / 2.3)
42 DH1 = (A - C1 * LOG(TBOIL)) * P * 3414.4 * 1000! / 3600!
B1 = DH1 / MCP
DELTM1 = (2780! - 600!) / B1

'2780.0 IS RUNAWAY ZR OXIDATION TEMP. IN F'
'600.0 IS THE ASSUMED STARTING TEMP. AFTER BOILOFF (F)'

TM1E = TBOIL + DELTM1
50 IF TM1E > 10000! THEN GOTO 54
A = .06: C1 = (.0128 / 2.3)
GOTO 55
54 A = .019: C1 = (.00255 / 2.3)
55 DHEND1 = (A - C1 * LOG(TM1E)) * P * 3414.4 * 1000! / 3600!
DHAVG1 = (DH1 + DHEND1) / 2!
B = (DHAVG1 / MCP)
IF ABS((B1 - B) / B) < .01 THEN GOTO 60
B1 = B
DELTM = (2780! - 600!) / B1
TM1E = TBOIL + DELTM
GOTO 50
60 BB1 = DH1 / MCP
DTUN1 = (WU) * (548.5) / DH1
70 TUN1 = TBOIL + DTUN1
IF TUN1 > 10000! THEN GOTO 74
A = .06: C1 = (.0128 / 2.3)
GOTO 75
74 A = .019: C1 = (.00255 / 2.3)
75 DHEND2 = (A - C1 * LOG(TUN1)) * (P * 3414.4 * 1000! / 3600!)
DH2AVG = (DH1 + DHEND2) / 2!
BB = DH2AVG / MCP
IF ABS((BB1 - BB) / BB) < .01 THEN GOTO 80
BB1 = BB
DTUN1 = (WU * 545.5) / DH2AVG
GOTO 70
80 DHAVG = (DHAVG1 + DH2AVG) / 2!
B1 = DHAVG / MCP
B2 = ZRC
DELTMFP = (1700! - 600!) / B1

'1700 (F) IS THE ASSUMED CLAD FAILURE TEMP. &, HENCE, START OF F.P. RELEASE'

TSTART = TBOIL + DELTMFP
DELTM1 = (2780! - 600!) / B1
DELTM2 = (4868! - 2780!) / B2

'4868 (F) IS THE ASSUMED MELTING TEMP. OF FUEL (ZRO2)'

DELTM3 = (1029!) / B2
```

```
'1029 (F) WAS DEVELOPED AS AN EQUIV. CHANGE IN TEMP. THAT WOULD MELT'
'THE FUEL IF IT IS ASSUMED THAT THE RATE OF HEAT INPUT IS THE SAME AS IN '
'PHASE 2'

LPRINT
LPRINT "THE FOLLOWING ARE THE GENERALIZED THERMAL TRANSIENT PARAMETERS"
LPRINT
LPRINT "TIME OF START OF TRANSIENT (S) = 0"
LPRINT "TIME AT BLOWDOWN END (S) ="; TBD
LPRINT "TIME DURATION FOR BOILOFF TO TOP OF CORE (S) ="; (TBOIL - TBD)
LPRINT "TIME DURATION FOR ADIABATIC HEATUP (S) ="; DELTM1
LPRINT "TIME DURATION FOR RUNAWAY ZR OXIDATION PERIOD (S)="; DELTM2
LPRINT "HOLD TIME AT MELT (S)="; DELTM3
LPRINT "B1 (F/S)="; B1; "B2 (F/S)="; B2
LPRINT "TIME FOR START OF F.P. RELEASE (S) ="; TSTART

'FISSION PRODUCT RELEASE CALCULATION FOLLOWS'

IF REL = 2 THEN GOTO 280
LPRINT
LPRINT "YOU HAVE        CHOSEN THE KRESS/BOOTH RELATIVE VOLALTILITY MODEL"
LPRINT "(RelVol)                            "

'RelVol FP RELEASE CACULATIONAL ALGORITHM FOLLOWS'

'FIRST CALCULATES RELEASE FOR ADIABATIC PERIOD'

IT = 0: JT = 1
Q = (2.065 * 10 ^ 5 - 3.629 * BU) / 1.99
C = (2.6833 * 10 ^ 5) * EXP(-(6.052 * 10 ^ -4) * BU)
100 IT = IT + 1
IF IT = 1 THEN GOTO 110
IF IT = 2 THEN GOTO 120
IF IT = 3 THEN GOTO 130
110 DTM = (TBOIL + DELTM1 - TSTART) / NT
TS = 1200!
DELT = 600! / NT
OLDSUM = 0!
GOTO 140
120 DTM = (DELTM2) / NT
TS = 1800!
DELT = 1160! / NT
GOTO 140
130 D3 = C * EXP(-Q / 2960!)
NEWSUM = (D3 * DELTM3) + OLDSUM
GOTO 160
140 N = 0
150 N = N + 1
TBAR = TS + DELT / 2!
DBAR = C * EXP((-Q / TBAR))
NEWSUM = (DBAR * DTM) + OLDSUM
OLDSUM = NEWSUM
IF N = NT THEN GOTO 160
TS = TS + DELT
GOTO 150
160 DTBAR = NEWSUM
170 IF (DTBAR) / (.00000036#) > .1 THEN GOTO 180
F = 6! * ((DTBAR / (3.142 * .00000036#)) ^ .5) - 3! * DTBAR / (.00000036#)
GOTO 190
180 F = 1! - (6! / 3.142 ^ 2!) * EXP((-(3.142) ^ 2 * DTBAR) / .00000036#)
190 IF JT = 2 THEN GOTO 225
IF IT = 1 THEN GOTO 200
IF IT = 2 THEN GOTO 210
IF IT = 3 THEN GOTO 220
200 FCS(1) = F
GOTO 100
210 FCS(2) = F
GOTO 100
220 FCS(3) = F
JT = JT + 1
Q = ((2.494 * 10 ^ 5) - 3.629 * BU) / 1.99
C = (3.4608 * 10 ^ 6) * EXP(-(6.052 * 10 ^ -4) * BU)
IT = 0
```

```
GOTO 100
225 IF IT = 1 THEN GOTO 226
IF IT = 2 THEN GOTO 227
IF IT = 3 THEN GOTO 230
226 FSB(1) = F
GOTO 100
227 FSB(2) = F
GOTO 100
230 FSB(3) = F

'NOW WE USE THE RELATIVE VOLATILITY FORMULA FOR OTHER SPECIES'

RV(1) = 1.1: RV(2) = 1.07: RV(3) = 1.03: RV(4) = .42
RV(5) = .34: RV(6) = .25: RV(7) = .14: RV(8) = .085
FOR J = 1 TO 3
FOR I = 1 TO 8
F(J, I) = FCS(J) * ((FCS(J) / FSB(J)) ^ ((RV(I) - 1!) / .32))
IF F(J, I) < 1! THEN GOTO 240
F(J, I) = 1!
240 NEXT I
NEXT J
LPRINT
LPRINT "YOU HAVE CHOSEN TO ASSUME ONLY A FRACTION OF THE CORE"
LPRINT "PARTICIPATES IN THE RELEASE"
LPRINT
LPRINT "FRACTION OF CORE PARTICIPATING ="; FCM
LPRINT
LPRINT "THE FOLLOWING ARE THE VARIOUS SOURCE TERMS AT THE END OF"
LPRINT "THE THREE PHASES OF THE GENERALIZED THERMAL TRANSIENT"
LPRINT
LPRINT "AT THE END OF ADIABATIC HEATUP, FRACTIONAL RELEASES ARE AS FOLLOWS"
LPRINT
LPRINT "        NG ="; F(1, 1) * FCM
LPRINT "        Te ="; F(1, 2) * FCM
LPRINT "         I ="; F(1, 3) * FCM
LPRINT "        Cs ="; FCS(1) * FCM
LPRINT "        Sb ="; FSB(1) * FCM
LPRINT "        Ba ="; F(1, 4) * FCM
LPRINT "        Sr ="; F(1, 5) * FCM
LPRINT "        Ru ="; F(1, 6) * FCM
LPRINT "        La ="; F(1, 7) * FCM
LPRINT "        Ce ="; F(1, 8) * FCM
LPRINT
LPRINT "AT THE END OF RUNAWAY ZR OXIDATION PERIOD, THE FRACTIONAL RELEASES"
LPRINT "ARE AS FOLLOWS"
LPRINT
LPRINT "        NG ="; F(2, 1) * FCM
LPRINT "        Te ="; F(2, 2) * FCM
LPRINT "         I ="; F(2, 3) * FCM
LPRINT "        Cs ="; FCS(2) * FCM
LPRINT "        Sb ="; FSB(2) * FCM
LPRINT "        Ba ="; F(2, 4) * FCM
LPRINT "        Sr ="; F(2, 5) * FCM
LPRINT "        Ru ="; F(2, 6) * FCM
LPRINT "        La ="; F(2, 7) * FCM
LPRINT "        Ce ="; F(2, 8) * FCM
LPRINT
LPRINT "AT THE END OF THE HOLD TIME AT MELT, THE FRACTIONAL RELEASES"
LPRINT "ARE AS FOLLOWS"
LPRINT
LPRINT "        NG ="; F(3, 1) * FCM
LPRINT "        Te ="; F(3, 2) * FCM
LPRINT "         I ="; F(3, 3) * FCM
LPRINT "        Cs ="; FCS(3) * FCM
LPRINT "        Sb ="; FSB(3) * FCM
LPRINT "        Ba ="; F(3, 4) * FCM
LPRINT "        Sr ="; F(3, 5) * FCM
LPRINT "        Ru ="; F(3, 6) * FCM
LPRINT "        La ="; F(3, 7) * FCM
LPRINT "        Ce ="; F(3, 8) * FCM
GOTO 380
280 LPRINT
LPRINT "            YOU CHOSE TO USE THE CORSOR-M MODEL RATHER THAN THE"
```

```
LPRINT "                                              ' KRESS/BOOTH RelVol MODEL
K(1) = 2! * 10 ^ 5: K(2) = 2.95 * 10 ^ 5: K(3) = 1.62 * 10 ^ 6
K(4) = 2.67 * 10 ^ 8
Q(1) = 63.8 / (1.987 * 10 ^ -3): Q(2) = 100.2 / (1.987 * 10 ^ -3)
Q(3) = 152.8 / (1.987 * 10 ^ -3): Q(4) = 188.2 / (1.987 * 10 ^ -3)
IT = 0
290 IT = IT + 1
IF IT = 1 THEN GOTO 300
IF IT = 2 THEN GOTO 310
IF IT = 3 THEN GOTO 350
300 DTM = (TBOIL + DELTM1 - TSTART) / NT
TS = 1200!
DELT = 600! / NT
FOR I = 1 TO 4
OLDSUM(I) = 0!
NEXT I
GOTO 330
310 DTM = DELTM2 / NT
TS = 1800!
DELT = 1160! / NT
330 N = 0
340 N = N + 1
TBAR = TS + DELT / 2!
FOR I = 1 TO 4
RBAR(IT, I) = K(I) * EXP(-Q(I) / TBAR)
NEWSUM(IT, I) = OLDSUM(I) + RBAR(IT, I) * DTM / 60!
OLDSUM(I) = NEWSUM(IT, I)
NEXT I
IF N = NT THEN GOTO 360
TS = TS + DELT
GOTO 340
350 FOR I = 1 TO 4
RBAR(IT, I) = K(I) * EXP(-Q(I) / 2960!)
NEWSUM(IT, I) = OLDSUM(I) + RBAR(IT, I) * DELTM3 / 60!
NEXT I
360 FOR I = 1 TO 4
F(IT, I) = 1! - EXP(-NEWSUM(IT, I))
NEXT I
IF IT = 3 THEN GOTO 370
GOTO 290
370 LPRINT
LPRINT "THE FOLLOWING ARE THE VARIOUS SOURCE TERMS AT THE END OF EACH"
LPRINT "OF THE THREE PHASES OF THE GENERALIZED THERMAL TRANSIENT"
LPRINT
LPRINT "YOU HAVE CHOSEN TO ASSUME ONLY A FRACTION OF THE CORE"
LPRINT "PARTICIPATES IN THE RELEASE"
LPRINT
LPRINT "FRACTION OF CORE PARTICIPATING ="; FCM
LPRINT
LPRINT "AT THE END OF ADIABATIC HEATUP, FRACTIONAL RELEASES ARE"
LPRINT
LPRINT "NG, I, Cs, & Te ="; F(1, 1) * FCM
LPRINT "        Ba & Sr ="; F(1, 2) * FCM
LPRINT "             Ru ="; F(1, 3) * FCM
LPRINT "             Ce ="; F(1, 4) * FCM
LPRINT
LPRINT "AT THE END OF THE RUNAWAY ZR OXIDATION PHASE, THE FRACTIONAL"
LPRINT "RELEASES ARE AS FOLLOWS"
LPRINT
LPRINT "NG, I, Cs, & Te ="; F(2, 1) * FCM
LPRINT "        Ba & Sr ="; F(2, 2) * FCM
LPRINT "             Ru ="; F(2, 3) * FCM
LPRINT "             Ce ="; F(2, 4) * FCM
LPRINT
LPRINT "AT THE END OF THE HOLD TIME AT MELT, THE FRACTIONAL"
LPRINT "RELEASES ARE AS FOLLOWS"
LPRINT
LPRINT "NG, I, Cs, & Te ="; F(3, 1) * FCM
LPRINT "        Ba & Sr ="; F(3, 2) * FCM
LPRINT "             Ru ="; F(3, 3) * FCM
LPRINT "             Ce ="; F(3, 4) * FCM
LPRINT
380 INPUT "TO RUN ANOTHER CASE, INPUT 1. TO END, INPUT 2"; DECD
```

```
IF DECD = 2 THEN GOTO 390
CSE = CSE + 1
GOTO 10
390 END



LPRINT "                ASSTCs"
LPRINT
LPRINT
LPRINT
10 INPUT "HOW MANY LINEAR TEMPERATURE PHASES"; NQ
INPUT "HOW MANY INCREMENTS PER PHASE "; NIP
INPUT "HOW MANY PRINT INCREMENTS PER PHASE "; NP
INPUT "WHAT IS THE BURNUP (MWD/T)"; BU
FOR I = 1 TO NQ
PRINT "WHAT IS THE STARTING TEMP (K) FOR PHASE "; I
INPUT ; TST(I)
PRINT
PRINT "WHAT IS THE ENDING TEMP (K) FOR PHASE "; I
INPUT ; TEND(I)
PRINT
PRINT "WHAT IS THE RATE OF TEMP CHANGE (K/S) FOR PHASE"; I
INPUT ; B(I)
PRINT
IF B(I) > 0 GOTO 20
PRINT "WHAT IS THE TIME DURATION (S) FOR CONSTANT TEMP PHASE"; I
PRINT
INPUT ; DCTM(I)
20 NEXT I
LPRINT "KEY INPUT ARE AS FOLLOWS"
LPRINT "BURNUP ="; BU
LPRINT
FOR I = 1 TO NQ
LPRINT "STARTING TEMP (K) FOR PHASE "; I; "IS"; TST(I)
LPRINT "ENDING TEMP (K) FOR PHASE "; I; "IS"; TEND(I)
LPRINT "THE RATE OF TEMP CHANGE (K/S) FOR PHASE"; I; "IS"; B(I)
LPRINT
IF B(I) > 0 THEN GOTO 25
LPRINT "THE TIME DURATION (S) FOR PHASE"; I; "IS"; DCTM(I)
LPRINT
25 NEXT I
LPRINT
LPRINT
LPRINT "        TIME (S)         FRACTIONAL RELEASE"
LPRINT
Q = (2.065 * 10 ^ 5 - 3.629 * BU) / 1.99
C = (2.6833 * 10 ^ 5) * EXP(-(6.025 * 10 ^ -4) * BU)
T = 0!:  OLDSUM = 0!
30 FOR I = 1 TO NQ
TS = TST(I)
TPRINT = 0!
IF B(I) = 0 THEN GOTO 60
DTM = (TEND(I) - TST(I)) / (B(I) * NIP)
40 T = T + DTM
TPRINT = TPRINT + DTM
DELT = B(I) * DTM
TBAR = TS + DELT / 2
TS = TS + DELT
DBAR = C * EXP(-Q / TBAR)
NEWSUM = OLDSUM + (DBAR * DTM)
OLDSUM = NEWSUM
DTBAR = NEWSUM
IF TPRINT >= (DTM * NIP / NP) THEN GOTO 110
45 IF TS >= TEND(I) THEN GOTO 50
GOTO 40
50 NEXT I
GOTO 120
'THIS PART IS FOR CONSTANT TEMP PHASES'
60 TDUR = 0!
80 TDUR = TDUR + DCTM(I) / NP
90 T = T + DCTM(I) / NP
DBAR = C * EXP((-Q / TEND(I)))
```

```
NEWSUM = OLDSUM + (DBAR * DCTM(I) / NP)
OLDSUM = NEWSUM
DTBAR = NEWSUM
IF DTBAR / (.00000036#) > .1 THEN GOTO 100
F = 6! * ((DTBAR / (3.142 * .00000036#)) ^ .5) - 3! * DTBAR / (.00000036#)
LPRINT "      "; T; "      "; F
IF TDUR >= DCTM(I) THEN GOTO 50
GOTO 80
100 F = 1! - (6! / 3.142 ^ 2) * EXP((-(3.142) ^ 2 * DTBAR) / .00000036#)
LPRINT "      "; T; "      "; F
IF TDUR >= DCTM(I) THEN GOTO 50
GOTO 80
110 TPRINT = 0!
IF DTBAR / (.00000036#) > .1 THEN GOTO 115
F = 6! * ((DTBAR / (3.142 * .00000036#)) ^ .5) - 3! * DTBAR / (.00000036#)
LPRINT "      "; T; "      "; F
GOTO 45
115 F = 1! - (6! / 3.142 ^ 2!) * EXP(-((3.142 ^ 2 * DTBAR) / (.00000036#)))
LPRINT "      "; T; "      "; F
GOTO 45
120 INPUT "TO RUN ANOTHER CASE, INPUT 1. TO END, INPUT 2"; DECD
IF DECD = 2 THEN GOTO 130
GOTO 10
130 END
```

ASSTCs

This case is for comparison with ORNL VI-3 Test data

```
KEY INPUT ARE AS FOLLOWS
BURNUP = 40000

STARTING TEMP (K) FOR PHASE  1 IS 1400
ENDING TEMP (K) FOR PHASE  1 IS 2000
THE RATE OF TEMP CHANGE (K/S) FOR PHASE 1 IS .2858

STARTING TEMP (K) FOR PHASE  2 IS 2000
ENDING TEMP (K) FOR PHASE  2 IS 2000
THE RATE OF TEMP CHANGE (K/S) FOR PHASE 2 IS 0

THE TIME DURATION (S) FOR PHASE 2 IS 1200

STARTING TEMP (K) FOR PHASE  3 IS 2000
ENDING TEMP (K) FOR PHASE  3 IS 2700
THE RATE OF TEMP CHANGE (K/S) FOR PHASE 3 IS .35897

STARTING TEMP (K) FOR PHASE  4 IS 2700
ENDING TEMP (K) FOR PHASE  4 IS 2700
THE RATE OF TEMP CHANGE (K/S) FOR PHASE 4 IS 0

THE TIME DURATION (S) FOR PHASE 4 IS 1200


        TIME (S)          FRACTIONAL RELEASE

       230.9307         5.631159E-03
       461.8615         1.061468E-02
       692.7924         1.742244E-02
       923.7232         2.688195E-02
       1154.654         3.982418E-02
       1385.584         5.713573E-02
       1616.514         .0797488
       1847.444         .1086072
       2078.375         .1446144
       2219.368         .1678723
       2339.368         .1851277
       2459.368         .2006709
       2579.368         .2148933
       2699.368         .2280567
```

```
     TIME (S)         FRACTIONAL RELEASE

    2819.368           .2403462
    2939.368           .2518986
    3059.368           .2628185
    3179.368           .2731876
    3299.368           .2830716
    3494.371           .3028198
    3689.373           .3321172
    3884.375           .3726572
    4079.378           .4251665
    4274.378           .4891995
    4469.378           .5631105
    4664.378           .644051
    4859.378           .7278975
    5054.378           .8107307
    5249.378           .881249
    5369.378           .9148152
    5489.378           .9388937
    5609.378           .956166
    5729.378           .9685562
    5849.378           .9774442
    5969.378           .9838198
    6089.378           .9883933
    6209.378           .9916741
    6329.378           .9940275
    6449.378           .9957157
```

```
                         SST

CASE SST 1 - SURRY LBLOCA WITH KRESS/BOOTH RelVol
THIS IS CASE 1

BURNUP (MWD/T) = 30000
NORMAL OPERATING POWER (MW) = 2441
INPUT BLOWDOWN TIME (S) = 20
CORE FUEL/CLAD MCP (BTU/F) = 25428
RESIDUAL RCS WATER TO TOP OF CORE (LB) = 348200
RESIDUAL WATER FROM TOP OF CORE TO COMPLETE CORE UNCOVERY (LB)= 25000
FRACTION OF CORE ASSUMED TO PARTICIPATE IN MELT = 1
VALUE ASSUMED FOR RUNAWAY ZR OX. HEATUP RATE (F/S) = 38


THE FOLLOWING ARE THE GENERALIZED THERMAL TRANSIENT PARAMETERS

TIME OF START OF TRANSIENT (S) = 0
TIME AT BLOWDOWN END (S) = 20
TIME DURATION FOR BOILOFF TO TOP OF CORE (S) = 4412.5
TIME DURATION FOR ADIABATIC HEATUP (S) = 1895.888
TIME DURATION FOR RUNAWAY ZR OXIDATION PERIOD (S)= 54.94737
HOLD TIME AT MELT (S)= 27.07895
B1 (F/S)= 1.149857 B2 (F/S)= 38
TIME FOR START OF F.P. RELEASE (S) = 5389.141

YOU HAVE WISELY CHOSEN THE KRESS/BOOTH RELATIVE VOLALTILITY MODEL
(RelVol) [PRONOUNCED - REAL VOL ]

YOU HAVE CHOSEN TO ASSUME ONLY A FRACTION OF THE CORE
PARTICIPATES IN THE RELEASE

FRACTION OF CORE PARTICIPATING = 1

THE FOLLOWING ARE THE VARIOUS SOURCE TERMS AT THE END OF
THE THREE PHASES OF THE GENERALIZED THERMAL TRANSIENT
```

```
AT THE END OF ADIABATIC HEATUP, FRACTIONAL RELEASES ARE AS FOLLOWS

       NG = 1.815236E-02
       Te = 1.148687E-02
        I = 6.240591E-03
       Cs = 3.949066E-03
       Sb = 2.997032E-05
       Ba = 5.680037E-07
       Sr = 1.676488E-07
       Ru = 4.248209E-08
       La = 7.934573E-09
       Ce = 3.429118E-09

AT THE END OF RUNAWAY ZR OXIDATION PERIOD, THE FRACTIONAL RELEASES
ARE AS FOLLOWS

       NG = .4705656
       Te = .3732178
        I = .2740006
       Cs = .217317
       Sb = 1.834042E-02
       Ba = 2.460598E-03
       Sr = 1.326233E-03
       Ru = 6.616769E-04
       La = 2.82857E-04
       Ce = 1.849385E-04

AT THE END OF THE HOLD TIME AT MELT, THE FRACTIONAL RELEASES
ARE AS FOLLOWS

       NG = .8744296
       Te = .7069204
        I = .5323913
       Cs = .4304043
       Sb = 4.454124E-02
       Ba = 7.052723E-03
       Sr = 4.000168E-03
       Ru = 2.113561E-03
       La = 9.691295E-04
       Ce = 6.562439E-04
```

SST

```
CASE SST 2 - SURRY LBLOCA WITH MELCOR-M
THIS IS CASE 2

BURNUP (MWD/T) = 30000
NORMAL OPERATING POWER (MW) = 2441
INPUT BLOWDOWN TIME (S) = 20
CORE FUEL/CLAD MCP (BTU/F) = 25428
RESIDUAL RCS WATER TO TOP OF CORE (LB) = 348200
RESIDUAL WATER FROM TOP OF CORE TO COMPLETE CORE UNCOVERY (LB)= 25000
FRACTION OF CORE ASSUMED TO PARTICIPATE IN MELT = 1
VALUE ASSUMED FOR RUNAWAY ZR OX. HEATUP RATE (F/S) = 38


THE FOLLOWING ARE THE GENERALIZED THERMAL TRANSIENT PARAMETERS

TIME OF START OF TRANSIENT (S) = 0
TIME AT BLOWDOWN END (S) = 20
TIME DURATION FOR BOILOFF TO TOP OF CORE (S) = 4412.5
TIME DURATION FOR ADIABATIC HEATUP (S) = 1895.888
TIME DURATION FOR RUNAWAY ZR OXIDATION PERIOD (S)= 54.94737
HOLD TIME AT MELT (S)= 27.07895
B1 (F/S)= 1.149857 B2 (F/S)= 38
TIME FOR START OF F.P. RELEASE (S) = 5389.141
```

```
YOU CHOSE TO USE THE CORSOR-M MODEL RATHER THAN THE KRESS/BOOTH RelVol MODEL

THE FOLLOWING ARE THE VARIOUS SOURCE TERMS AT THE END OF EACH
OF THE THREE PHASES OF THE GENERALIZED THERMAL TRANSIENT

YOU HAVE CHOSEN TO ASSUME ONLY A FRACTION OF THE CORE
PARTICIPATES IN THE RELEASE

FRACTION OF CORE PARTICIPATING = 1

AT THE END OF ADIABATIC HEATUP, FRACTIONAL RELEASES ARE

NG, I, Cs, & Te = 8.479058E-03
        Ba & Sr = 3.147893E-07
             Ru = 4.751012E-13
             Ce = 3.222032E-15

AT THE END OF THE RUNAWAY ZR OXIDATION PHASE, THE FRACTIONAL
RELEASES ARE AS FOLLOWS

NG, I, Cs, & Te = .5149707
        Ba & Sr = 1.451947E-03
             Ru = 7.068666E-07
             Ce = 2.329908E-07

AT THE END OF THE HOLD TIME AT MELT, THE FRACTIONAL
RELEASES ARE AS FOLLOWS

NG, I, Cs, & Te = .9162055
        Ba & Sr = 6.744899E-03
             Ru = 4.518954E-06
             Ce = 1.761283E-06
```