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Paper

SEVERE ACCIDENT CONSIDERATIONS
IN CANADIAN NUCLEAR POWER REACTORS

A paper presented by A.M. Omar and M.P. Measures (Atomic Energy Control Board), C.K. Scott (Atlantic Nuclear Services Ltd.) and M.J. Lewis (Electrowatt Consulting Engineers and Scientists Ltd.) at the CEC Seminar on Methods and Codes for Assessing Off-Site Consequences of Nuclear Accidents, Athens, May 7-11, 1990.

ABSTRACT

This paper describes a current study on severe accidents being sponsored by the Atomic Energy Control Board (AECB) and provides background on other related Canadian work.

Scoping calculations are performed in Phase I of the AECB study to establish the relative consequences of several permutations resulting from six postulated initiating events, nine containment states, and a selection of meteorological conditions and health effects mitigating criteria. In Phase II of the study, selected accidents sequences would be analyzed in detail using models suitable for the design features of the Canadian nuclear power reactors.

RÉSUMÉ

Le présent rapport décrit une étude en cours financée par la Commission de contrôle de l'énergie atomique (CCEA) sur les accidents graves dans les centrales nucléaires et fournit des renseignements sur d'autres travaux canadiens connexes.

Dans la première phase de l'étude de la CCEA, on a effectué des calculs de portée pour établir les incidences relatives de plusieurs permutations découlant de six événements déclencheurs hypothétiques, de neuf états de confinement, ainsi que d'une sélection de conditions météorologiques et de critères d'atténuation des effets sur la santé. Dans la deuxième phase de l'étude, on se propose d'analyser en détail plusieurs séquences choisies d'accidents à l'aide de modèles compatibles avec les caractéristiques nominales des réacteurs nucléaires canadiens.

1. INTRODUCTION

The Three Mile Island/Unit-2 (TMI-2) accident in 1979 prompted renewed interest in re-evaluating existing computer codes and developing new models used to estimate the source term[1]. The accident was contained and managed within the containment[2]. The studies that followed the TMI-2 accident have been focussed on understanding and simulating the possible sequence of events which occurred. Significant progress in the fields of core damage progression and core debris coolability has been achieved from the data collected by examining the TMI-2 damaged core.

The accident at Chernobyl in 1986 emphasized another aspect of what are currently known as severe accidents[3] ; that is the detrimental health and socioeconomic consequences of such accidents at both the national and international scales. It was emphasized after the Chernobyl accident that a severe accident anywhere can result in consequences with worldwide impacts (Ref. 2). Also, examination of existing nuclear power stations was performed by several utilities and vendors to identify any potential vulnerability of the containment under severe accidents, and to take appropriate actions for the prevention, mitigation, and management of such accidents (Ref. 3).

Since 1986, interest in the Chernobyl accident and its consequences has dominated nuclear fora. Recent information (Ref. 4) suggests that the number of fatalities that have been directly attributed to the Chernobyl accident is considerably lower than what might have been anticipated for an accident of that size.

In the second half of the 1980's, but particularly after the Chernobyl accident, the public worldwide showed interest in obtaining re-assurance of the safety of nuclear power installations. This is indicated by polls conducted in several countries (Ref. 5). Numerous countries have either initiated new research/design programs or enhanced existing ones to investigate measures for mitigating and managing severe accidents. Evidence of this worldwide interest is manifested in the number of international and local technical gatherings that have dealt with topics such as severe event sequences, the source terms, core melt progression, containment design, and fission product dispersion. In addition, design assessments of various types of containment have been performed in respect to their abilities to confine severe accidents. For some Pressurized Water Reactors (PWRs) several measures have been taken to enhance the capability of the containment system to minimize the release of radioactive materials to the environment under severe accident conditions (Ref. 1). A recent conceptual PWR design alluded to a containment system which could withstand the consequences of some of the most unlikely events such as

[1] For the purpose of this paper source term means the fraction, composition and timing of fission products released from damaged fuel to containment and then to the environment.

[2] As defined in Ref. 1, containment is "a structural envelope which completely surrounds the reactor system and is designed to hold the release from design basis accidents with little or no release to the environment". The term in this paper is used in a broader sense to include associated buildings for single- or multi-unit plants which would contain the release of severe accidents.

[3] Severe accidents denote those accidents that are more severe than those currently considered as design-basis accidents.

steam explosion within the pressure vessel and hydrogen detonation within the containment (Ref. 6).

2. SEVERE ACCIDENT ACTIVITIES IN CANADA

A few event sequences which might be categorized as severe accidents have been analyzed as design-basis accidents (Ref. 7) within and according to the requirements of the Canadian licensing philosophy (Ref. 8). This approach assesses the radiological consequences of single and dual failure[4] accidents without consideration to their expected frequency. The predicted dose levels must be within the guidelines established by the Atomic Energy Control Board. This deterministic approach has been supplemented by probabilistic design evaluations to assess event sequences that would result in significant fuel damage. A probabilistic evaluation study (Ref. 9 and 10) done by the Canadian nuclear industry concluded that the frequency of an accident resulting in core damage is on the order of $5x10^{-6}$ per year, and the frequency of a power excursion coupled with failure to shut down is on the order of $3x10^{-8}$ per year. These low probabilities have been attributed to the independence of the two shutdown systems and the redundancy of their components, and to the numerous heat sinks which could mitigate core damage in the CANDU designs. This probabilistic study, however, stopped short of estimating the consequences of such events.

In 1986 the province of Ontario established the Ontario Nuclear Safety Review (ONSR) Commission to review the safety of Ontario's nuclear power reactors. The final report of this commission was published in February 1988 (Ref. 11). Several submissions presented by the nuclear industry and consultants alluded to severe accident considerations. Among them only one, submitted by Lonergan et al, attempted to estimate the potential health and economic consequences of severe accidents. Lonergan et al (Selected Consultants Reports, Ref. 11) used the code MACCS[5], with demographic, economic and land use inputs drawn, when possible, from Canadian data bases. They studied the potential health and economic consequences for a source term based on the release fraction characteristics of the "PWR-2" source term in the Rasmussen report (WASH-1400). Five weather conditions and three wind directions were considered. Their results suggest that, in the absence of effective emergency response, the health impact could range up to 10's to 100's of early fatalities and 100's to 1000's of early injuries, and that economic impact could vary from 100's of millions of dollars to tens of billions of dollars. This broad range of consequence estimates was attributed to differences in population densities and land values in different directions from the nuclear power station and in the wide range of meteorological conditions which could disperse the radioactive materials. It is to be mentioned, however, that Lonergan et al did not consider in their study several CANDU design characteristics which might lower fission product release fractions and prolong

[4] Please refer to References 8 and 9 for additional information on definitions of single and dual failures event sequences.

[5] Melcor Accident Consequence Code System, designed for Light Water Reactor (LWR) assessment.

the duration of releases compared to the light water reactors.

In response to the recommendations of the ONSR Commission, Ontario Hydro initiated a study on severe accident consequences in 1987/88 (Ref. 11). The Light Water Reactor (LWR) Modular Accident Analysis Program (MAAP) computer code was acquired in 1988 and modified to reflect CANDU plant features (such as the horizontal pressure tube reactor with consecutive calandria vessel and shield tank heat sinks, the compartmentalized multi-unit containment, etc.). The fully integrated MAAP-CANDU computer code became operational in late 1989 (Ref. 13). The qualification, testing and quality assurance review of this code are expected to be completed by the summer of 1990, with the first results expected by the end of the year.

3. THE AECB SEVERE ACCIDENT STUDY

3.1 Background

The AECB assessed the Chernobyl accident in 1987 (Ref. 14) with respect to its implications for CANDU safety. In this assessment, it was concluded at that time that no important new information had evolved that would have a major effect on the licensing requirements for CANDU reactors. It was recommended, however, that the AECB and the nuclear industry re-examine the relevant safety aspects of the Canadian power reactors, and the safety of one of the nuclear power stations in Canada with respect to accidents involving failure of the control systems, loss of coolant accidents and unavailability of the shutdown systems.

From 1981 till 1984 the AECB sponsored a study, by J.T. Rogers, on CANDU core behaviour under severe accident conditions (Ref. 15). Two computer subroutines unique to CANDU have been produced: MODBOIL, for moderator boil-off calculation, and DEBRIS, for determining the level of coolability of the core debris within the calandria. These two subroutines are currently used by the Canadian nuclear community in several severe accident studies (e.g., see Refs. 9, 10, 13).

The current AECB study on severe accidents was initiated in 1989. This study is described in the following sections.

3.2 Objective and Scope

The overall long-term aim of the AECB study is to obtain an independent evaluation of the consequences to the public and the environment of postulated severe accidents at a Canadian nuclear power station. The information generated would be used to: a) assess the results of licensee calculations, b) afford the AECB an opportunity to evaluate the degree of conservatism in its own safety criteria, and c) develop advice or make decisions regarding nuclear insurance or siting policy for large nuclear power stations.

The study is divided into two phases. Phase I, which is near completion, is a scoping study to establish the relative consequences of a number of postulated event sequences to determine which event sequence(s) would lead to the maximal health and economic consequences, and/or pose the most severe tests for computer codes and calculational models. Phase I also includes a review of models and computer codes which could be used later for detailed calculations. In Phase II detailed calculations are to be performed on selected event sequence(s) using codes and methods appropriate to CANDU design.

It must be emphasized here that the scope of Phase I did not include any estimation of the frequency of occurrence of the accidents postulated although the frequencies are extremely low and below values considered in the design process.

4. PHASE I OF THE AECB SEVERE ACCIDENT STUDY

The event sequences selected in Phase I would represent three variations of energy and fission product release. These are: 1) short term and high rate of release, 2) long term and low rate of release, and 3) a combination of high rate / short term release initially, which develops into a long term / low rate release. These three variations of release are identified and then used to compare the consequences for each of the nuclear power stations which are the subject of this study.

4.1 Factors Used For Screening Event Sequences

The event sequences considered for Phase I of the study are based on a combination of four concurrent factors. These are: 1) initiating event; 2) containment state; 3) metreological conditions; and 4) health effects mitigating criteria. Table 1 lists the elements that comprise each factor. The following discussions give details of these four factors and the scoping approach.

4.1.1 Initiating events

Six initiating events were selected as shown in Table 1. The massive failure of the steam generator's secondary side is an event which could induce a loss of coolant accident (LOCA) and potentially cause consequential damage due to projectiles. The large civil aircraft (e.g. Boeing 747) crash is assumed to represent an ex-containment event which could possibly cause perforation of the containment and consequential damage to the core. The containment bypass is postulated to represent an event in which fission products would be released to the environment while the containment remains intact. Loss of coolant accident, loss of power regulation (LOR), and loss of all pumped heat sinks are, by themselves, events that are normally analyzed as design-basis accidents within the Canadian safety approach. Here, the LOCA and LOR cases would be analyzed in association with an assumed containment state, as well as assumed failure of the shut down systems. The loss of all pumped heat sinks case is extended to a failure of backup heat removal systems following the automatic shutdown.

4.1.2 Containment state

As shown in Table 1, nine containment states were selected ranging from no failure to a state in which all airlocks are assumed to be open, concurrent with the Reactor Building Pressure Relief Valves (RB-PRV) being impaired. It should be emphasized here that each of these containment states is postulated to be concurrent with the initiating events. Consequential damage to the containment due to an initiating event would modify the assumed state of containment.

4.1.3 Meteorological conditions

Weather conditions with probabilities of a few percent or more are considered. The impact of radionuclide dispersion under these various conditions is evaluated using a plume model with uniform horizontal dispersion over a 22.5 degree sector and Gaussian vertical dispersion. Standard Pasquill weather categories are assumed. Early examination of the available demographic data suggests that the worst case wind direction is from the North-East, aligned along the northern shore of Lake Ontario (generally corresponding, though with some adjustments, to Sector D in the study of Lonergan et al. (Ref. 11)).

4.1.4 Health effects mitigating criteria

Intervention is postulated to commence not earlier than four hours after the initiation of the accident. To return people back to the decontaminated areas and/or to initiate an intervention 50 mSv is used as a threshold for average life-time dose.

4.2 Consequences

During Phase I the event sequences resulting from permutations of the factors described in 4.1 above are screened with the aim of determining which event sequence(s) would yield the maximal health and socioeconomic effects, in order to provide the most severe test of the methodology. Health impact considerations would include direct effects such as early mortality and morbidity, long-term stochastic effects, and indirect effects such as long-term psychological impacts. The consideration of socioeconomic consequences include direct costs such as those associated with relocation, decontamination, farm produce disposal, etc.; and indirect costs associated with litigation, liability, public acceptance and perception, etc.

4.3 Methodology

The approach adopted for Phase I is based on calculations utilizing simplified correlations, an ad-hoc variety of existing and special purpose computer programs, and, often, engineering judgment. Figure 1 shows a module flow chart for the approach to Phase I of the study. It is to be emphasized that because of the numerous simplifications inherent in this approach, the outcome of the calculations are not representative of those which might be produced in Phase II, and will be used only for scoping and ranking purposes.

4.3.1 Source term

4.3.1.1 Heat transport system and fuel response

Simple calculational models are devised to represent the CANDU reactor system geometry. In some cases models from fundamental equations are developed, and in other cases models are developed on the basis of modified expressions found in established computer codes. Coolant inventories (primary, secondary, moderator and shield tank) are traced for each event sequence since the degree of fuel cooling determined the rate of release of fission products from the fuel. Figure 2 shows some of the methods and models used in the calculations of fuel cooling. For example, HOTMOD models the depletion of moderator inventory and calculates the progressive uncovering of rows of pressure tubes. Once uncovered, adiabatic heating of the fuel is assumed. Zircaloy oxidation rates are monitored as temperatures increase prior to the potential for pressure tube failure at high temperatures. DEBCHAN calculates the temperature distribution of fuel inside a fuel channel by a one-dimensional conduction model. This is derived from DEBRIS (Ref. 15) which deals with fuel debris coolability.

A containment thermal hydraulics computer code, CREM, is being developed to model the negative-pressure vacuum building, unique to the multi-unit CANDU stations. This model calculates the pressures, temperatures and mass flows within and from the containment envelope following a discharge from the Heat Transport Systems (HTS). This model is used to assess the potential for containment failure by overpressure , and also to provide the boundary conditions for the analysis of fission product behaviour outside of the HTS.

4.3.1.2 Fission product behavior within containment

An overview of the calculational routes for fission product behaviour within the containment is also shown in Figure 2. Many of these models are derivatives of the USNRC's Source Term Code Package (STCP). The models treat the behaviour of the fuel and fission products outside of the confines of the HTS and calandria. They deal with issues such as core-concrete interaction and aerosol depletion inside the containment. The inventories of a range of nuclides as they are released into and from the containment atmosphere are estimated. The result is a ranking of the source terms to the environment for postulated initiating events and assumed containment states.

4.3.2 Dispersion and direct health and economic impacts

Dispersion is modeled using the methods recommended by the U.K. Working Group on Atmospheric Dispersion. Assessment is performed of the radiological consequences (Ref. 16) of the postulated event sequences, then the relative health effects are examined using relevant demographic data and making judgments as to the intervention measures likely to be employed.

Existing socioeconomic models (Ref. 17) for the costs associated with mitigating the direct health effects of the accident are considered. These models include measures for protective actions and means to estimate health effects.

4.3.3 Indirect consequences

The potential indirect health and socioeconomic effects (Ref. 18) are considered for inclusion in the second phase of the study.

5. PHASE II of the AECB STUDY

5.1 Objective and Scope

The objective of Phase II of the AECB severe accident study would be to calculate in detail the consequences of specific event sequences for a specific site(s). Existing or modified computing tools and methods suitable for the design features of CANDU reactors would be used.

5.2 Possible Options

To perform Phase II of the study the AECB will have to take a decision as to which route should be taken to achieve its objective. The number of severe accident computer codes which could be used for the CANDU design is limited. Attempts are continuing to devise a severe accident package for CANDU reactors (Refs. 10 and 13).

If the AECB, as the Canadian nuclear regulatory agency, decides to acquire the severe accident technology and related computational tools then one of the following options could be selected:

a) Acquire LWR severe accident computer codes and modify them to suit the CANDU design;

b) Acquire computer code packages that are being developed by the Canadian industry and modify them to suit the AECB objectives;

c) Wait until a comprehensive package is developed by the Canadian industry , acquire the package and use it for the AECB purposes; or

d) Participate with the Canadian industry in developing a severe accident computer code package.

Each of the above options has its advantages and drawbacks. It is worth mentioning here that while technology developed by Canadian industry is more directly relevant to the analysis of CANDU reactors, participation in developing such technology might jeopardize the independence of the AECB study.

6. SUMMARY

This paper presents an overview of some of the severe accident activities in Canada. The paper then describes an AECB multi-phase deterministic study in which the short-term and long-term, direct and indirect, health and socioeconomic consequences of a postulated severe accident will be evaluated.

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Table 1: Factors Considered for Screening and Ranking Event Sequences

I. Initiating Event

1. Massive failure of steam generator.
2. Large civil aircraft (e.g, Boeing 747) crash.
3. Containment bypass.
4. Loss of Coolant (LOC) plus failure to shut down.
5. Loss of regulation at different rates and from different initial operating states, and failure to shut down.
6. Loss of all (pumped) heat sinks.

II. Containment State

1. Zero failure.
2. One large airlock open (e.g., maintainance airlock).
3. All airlocks open.
4. Reactor Building Pressure relief valves (RB-PRV) do not operate.
5. No vacuum.
6. No dousing.
7. No venting.
8. States 2 and 4.
9. States 3 and 4.

III. Meteorological Conditions

1. Weather conditions with probabilities of a few percent or more are considered.
2. Gaussian dispersion is assumed.

IV. Health Effects Mitigating Criteria

1. Intervention is assumed to commence four hours after the onset of the accidents.
2. 50 mSv is to be used to initiate intervention and to return people back to decontaminated area.

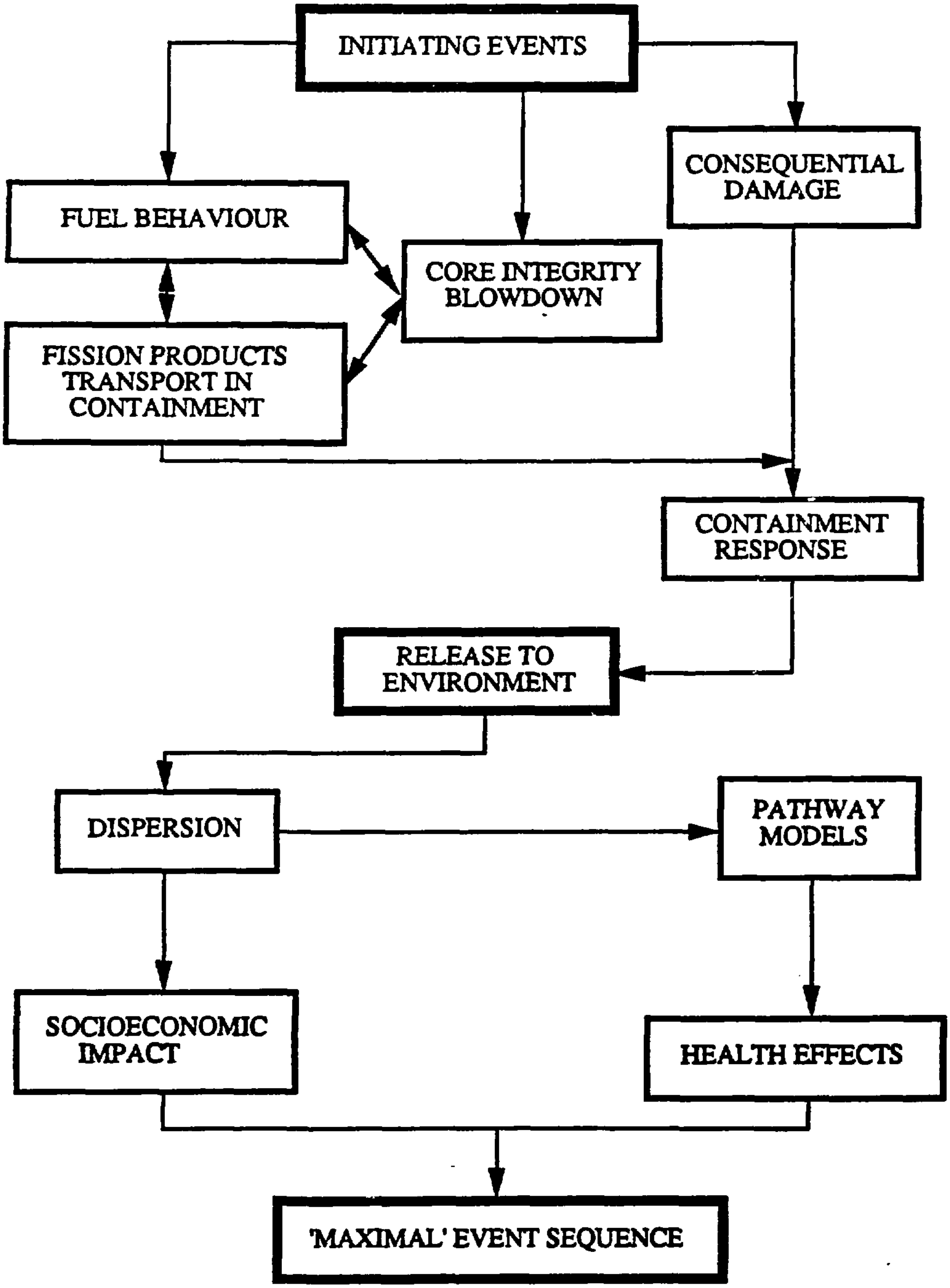

Figure 1: Approach to Phase I

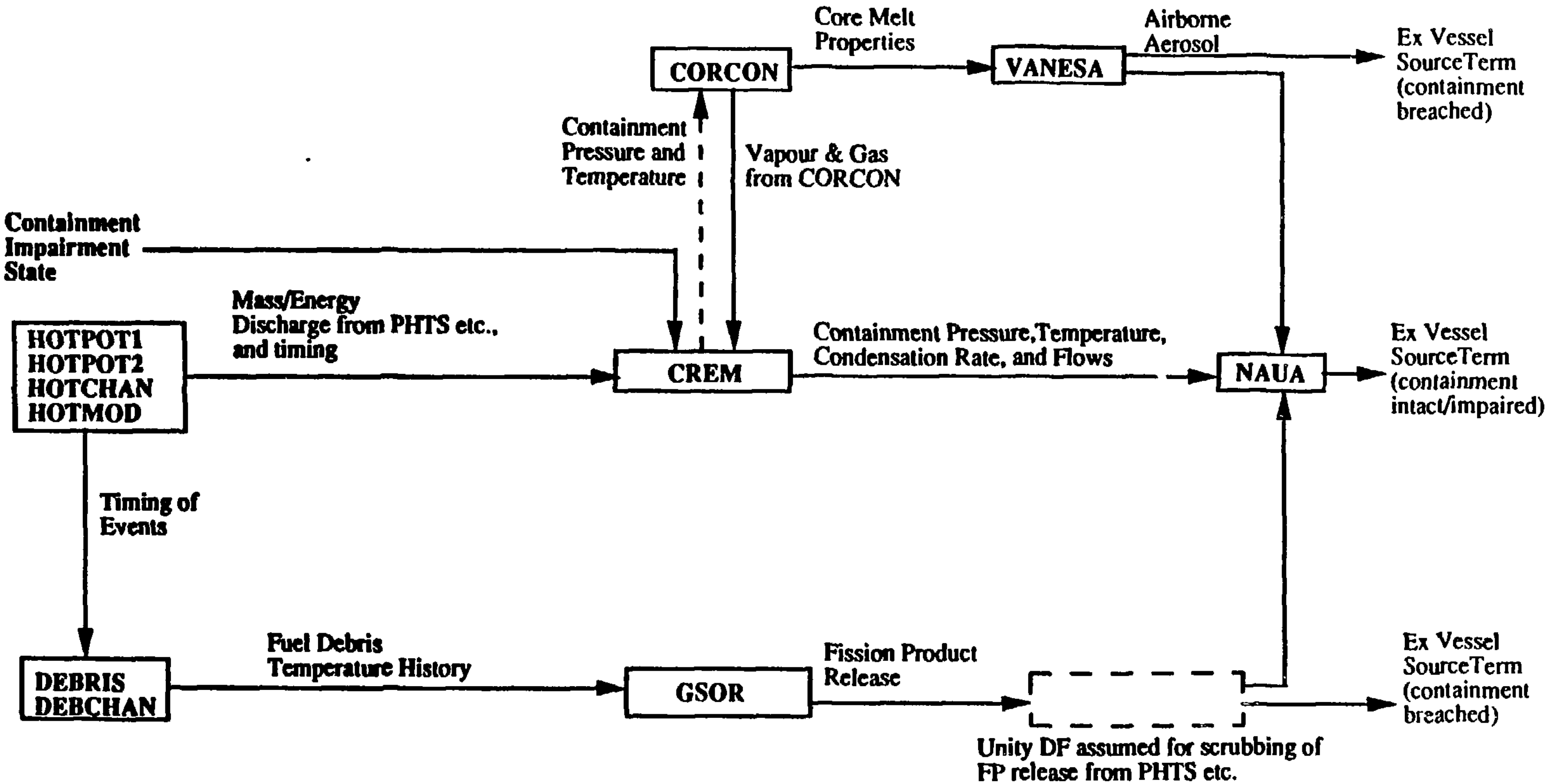

Figure 2: Calculational Routes for Fission Product Behaviour within Containemnt