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**HIFAR SAFETY ANALYSIS:
FREQUENCY AND OFFSITE CONSEQUENCES OF
FAULT SEQUENCES INITIATED BY WITHIN-PLANT FAILURES**

Report of the
HIFAR Safety Analysis Working Party Task (b) Group

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Research Establishment

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by

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ABSTRACT

HIFAR fault sequences, initiated by failures of within-plant equipment and operational procedures, are analysed using probabilistic methods, and their frequencies estimated. Sequence consequences are estimated in terms of potential radiation doses to an individual at 1.6 km radius from the reactor, expressed in terms of emergency reference levels recommended by the National Health and Medical Research Council for consideration of limited evacuation. The results show that the public risk from all such sequences is extremely low.

FOREWORD

This document presents the findings of a safety study of the research reactor HIFAR, carried out by a Technical Working Group. The study is complementary to existing HIFAR safety analyses which are based on deterministic approaches. It is expected that this material will, in whole or part, be incorporated in a future updating of the formal Safety Case for HIFAR operation.

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EXECUTIVE SUMMARY

PREFACE

The cooperation and contributions of many persons from diverse disciplines are indispensable to the successful conclusion of any study of the type now being reported. Such assistance was freely given by many members of AAEC staff throughout the period of the study.

Executive responsibility for the study was through a working party structure designated the 'HIFAR Safety Analysis Working Party (HSAWP) Task (b) Group'. The members were

D.B. McCulloch	(Chairman)
R.W. Innes	(Secretary)
E.R. Corran	
F.D. Nicholson	
M.C.E. Petersen	

The study extended over more than two years, and E.R. Corran acted as Chairman for a period of several months during 1984, while the appointed Chairman was assigned to other duties.

Other major contributors to the study, who were not members of the Task Group, include

- | | |
|---------------|---|
| A.G. Chapman | - Thermodynamics of post-accident cooling systems |
| G.H. Clark | - Meteorological data and dispersion modelling |
| J.W. Connolly | - Reactivity accidents analysis and LOCA modelling |
| F.G. May | - Fission product source terms |
| R. Miller | - Calculations of fission product decay, release, and doses |
| J. Marshall | - Post-LOCA RCB thermodynamics |
| J.T. Rodd | |
| T.J. Moss | |
| H.H. Witt | - Reliability and plant performance analyses. |

Their contributions, and those of many others not named, are gratefully acknowledged.

D.B. McCulloch
Chairman, HSAWP Task (b) Group

EXECUTIVE SUMMARY

- (i) The Task Group was established by the Director, AAEC Research Establishment. Its task was to identify and examine, using probabilistic risk analysis (PRA) techniques, all HIFAR accident sequences with potential for significant exposure of members of the general public to radiation.
- (ii) The study presently reported considers only accident sequences initiated by within-plant equipment failures or operational errors. It is intended that sequences initiated by external events (e.g. seismic activity) will be examined in a later extension of the study. Deterministic assessments of such events are, of course, included in existing HIFAR safety documentation.
- (iii) HIFAR safety philosophy is based on 'defence in depth' principles, in which multiple protective devices or barriers are provided to terminate or mitigate the consequences of event sequences which might otherwise follow some initiating plant malfunction or human error. No significant release of radioactive material to the environment can occur as a result of such internally initiated event sequences unless there is also concurrent failure of one or more protective devices.
- (iv) The purpose of the study was to quantify, for a full spectrum of postulated initiating events, the likelihood and degree of any significant public exposure to radiation which might then ensue from any combination of failures in the relevant protective devices. A particular requirement was to establish the likelihood of all event sequences for which the consequences might be sufficient that some evacuation of the local population, beyond the 1.6 km radius buffer zone in which HIFAR operates, would have to be considered.
- (v) The National Health and Medical Research Council (NH&MRC) currently recommends that limited evacuation of the local population should be considered by Competent Authorities following releases of radioactivity in which exposure of individual members of the public has potential to exceed

Whole Body	300 mSv
Child Thyroid	1000 mSv
(Adult Thyroid	3000 mSv)
Foetus	150 mSv

These are designated Emergency Reference Levels (ERLs). In the present study, the child thyroid dose limit was found to be the most restrictive in all cases.

- (vi) Event sequences originating from failures of HIFAR plant and equipment, including procedural errors, which have potential (in the absence or failure of protective systems) for exceeding the ERL at 1.6 km radius, have been identified and analysed.

Only sequences proceeding at least to significant melting of irradiated fuel have potential to exceed the ERL; the fuel may be in the reactor core, so involving the reactor *per se*, or it may be outside the core, during handling and transport operations or in storage, following its removal from the reactor.

- (vii) For fuel in the reactor core, either

- (a) a loss of coolant from a substantial leak in the primary coolant circuit coupled with failure of the emergency core cooling system, or
- (b) a reactivity insertion accident of sufficient severity not only to melt fuel, but to generate pressures capable of breaching the primary coolant circuit,

is a necessary prerequisite for potential to exceed the ERLs defined above. The combined frequency of occurrence of all such sequences is estimated to be less than 7×10^{-6} per year, based on conservative analyses of basic plant reliability data.

The public radiation exposure potential for a single composite model, the 'Bounding Case', was analysed in detail. This model

- (a) envelops adverse outcomes of the various complex components of the chains of physical processes making up the actual event sequences studied;
- (b) embraces a variety of protective barrier malfunctions or degraded performance and assumes '90% confidence level' data, wherever feasible and appropriate, for their reliability and performance characteristics; and
- (c) uses '90% non-exceedence' estimates in the conservative direction of the values of key physical and performance parameters.

The consequences of this Bounding Case embrace almost 90% of the above 7×10^{-6} sequences per year, and were found to be less than 1/30th of the child thyroid ERL. The frequency of event sequences which could result in more severe consequences is estimated to be less

than 9×10^{-7} per year. For more than 80% of such sequences, the consequences would still be less than 1/20th of the ERL. Only for a small residual class of sequences, whose total frequency is estimated to be less than 2×10^{-7} per year, might the consequences in some cases be sufficient that limited evacuation of the surrounding population would have to be considered.

Analysis of some extensions to the Bounding Case, shows it to be extremely unlikely that any in-core fuel melting accident could give rise to radiation doses of more than a few times the ERL, at or beyond 1.6 km radius from the reactor.

(viii) Accidents in the handling, storage and transport of irradiated fuel have been studied in two classes, depending on whether fuel melting occurs (a) within the outer boundary of the Reactor Containment Building (RCB), or (b) outside the RCB boundary.

(a) The estimated frequency of all sequences in which some overheating of fuel within the RCB could occur is less than 5×10^{-4} per year. By far the majority of these sequences arise from Load/Unload (L/U) flask operations which involve a single fuel element only. The estimated consequences at 1.6 km radius, on a similarly conservative basis to that adopted in (vii), are less than 1/200th of the child thyroid ERL. The associated frequency is estimated to be less than 4×10^{-4} per year.

Most remaining sequences (estimated frequency less than 10^{-4} per year) are associated with coolant failure in the spent-fuel storage block, SB-1. In normal circumstances, it is very unlikely that the SB-1 inventory would exceed 20 elements, of which no more than 8 would be from the most recent operating program; the usual inventory is considerably less. The consequences, at 1.6 km radius, of full or partial melting of any such inventory, are estimated to be less than 1/80th of the child thyroid ERL.

Another small class of sequences (estimated frequency less than 10^{-5} per year), involving the Shear and Transport (ST) flask which holds a maximum of 4 fuel elements, has similar bounding consequences unless complete separation of the flask main components occurs.

In emergency or specially scheduled situations the SB-1 inventory could be greater than normal. The estimated frequency for coolant failure in such circumstances is less than 4×10^{-6}

per year. The bounding case is considered to be melting of 25 elements cooled for an average of 2 days from reactor shutdown. The estimated consequences at 1.6 km radius are still less than 1/25th of the child thyroid ERL. These consequences also embrace those for ST flask events (estimated frequency less than 2×10^{-6} per year) in which complete separation of the flask main components is assumed to occur.

- (b) The only events in which fuel overheating could occur outside the RCB boundary are those involving toppling of the ST flask; this can occur only in handling operations in Building 23 (B23), the storage pond building, or by transport accidents on the very short (~ 50 m) road transit route between the RCB and B23.

The frequency of all such sequences is estimated to be less than 10^{-4} per year and, assuming that the 4 fuel elements concerned can melt, the bounding consequences at 1.6 km radius, evaluated for appropriate fuel histories and atmospheric conditions, are found for more than 90 per cent of those sequences to be less than 1/20th of the child thyroid ERL.

For almost all the remaining sequences, whose estimated frequency is only about 10^{-5} per year, the bounding dose at 1.6 km is less than 2/3rds of the child thyroid ERL. Only in an extremely small class of events of estimated frequency less than 2×10^{-8} per year, in which a fuel loading error as well as severe flask damage is involved, are greater doses found; the bounding level is still less than $2\frac{1}{2}$ times the child thyroid ERL.

(ix) Conclusions

The principal results of the study are summarised in Table ES-1 and Figure ES-1.

The combined frequency of all sequences initiated by within-plant failures, including allowance for human procedural errors, as a result of which potential radiation doses to some members of the general public might be more than a very small fraction of the ERL, is conservatively estimated to be less than 6×10^{-4} per year. In the great majority of these sequences, the consequences for an individual at 1.6 km radius from HIFAR would amount to at most a few per cent of the most restrictive (child thyroid) ERL.

Based on the probably over-cautious assumption that melting of the contained fuel ensues, accidents involving the ST flask present the greatest overall risk. However, the total frequency of such occurrences is estimated to be less than 10^{-4} per year and the consequences for more than 90 per cent of them are less than 1/20th of the child thyroid ERL. For almost all the residual, more severe occurrences, whose total frequency is estimated to be only about 10^{-5} per year, the bounding consequences are less than 2/3rds of the child thyroid ERL. Only an extremely small residual class of ST events, whose estimated frequency is less than 2×10^{-8} per year, was found to result in greater doses, ranging up to about 2½ times the child thyroid ERL.

Only as a result of these rare ST events, and a few slightly less unlikely but still extremely infrequent sequences associated with reactor loss of coolant accidents, could the ERL ever be exceeded, and hence the need to consider limited evacuation of local residents arise. The total frequency is conservatively estimated to be less than 2×10^{-7} per year, and it is very unlikely that the consequences could ever be greater than a few times the ERL.

Since all estimates of potential doses are based substantially on '90% conservative bound' rather than 'most probable' data, it seems extremely improbable that the ERL could ever be reached in practice.

TABLE ES-1 FREQUENCY AND CONSEQUENCES OF ACCIDENTS*

Class of Occurrence	Estimated Maximum Frequency of Class (y^{-1})	Sub-class Maximum Frequency (y^{-1})	Sub-class Maximum Consequences (Child Thyroid ERLs)	Notes	Figure ES-1 Data Point
1. In-core Fuel Melting (LOCAs with ECCS failure plus reactivity additions causing PCC breach).	7×10^{-6}	6×10^{-6}	1/30	'Bounding Case'	A1
		7×10^{-7}	1/20	Bounding Case plus high RCB leak-rate	A2
		2×10^{-7}	2 - 3	All others in class	A3
2. Fuel Storage and Handling	5×10^{-4}	4×10^{-4}	1/200	L/U flask/Single fuel element	B1
		1×10^{-4}	1/80	Coolant failure SB-1 (normal situations) and ST flask events (no or partial separation)	B2
		6×10^{-6}	1/25	Coolant failure SB-1 (special or emergency situations) and ST flask events (complete separation)	B3
2.2 Fuel melts outside the RCB	1×10^{-4}	9×10^{-5}	1/20	ST flask - fuel correct, any degree of flask damage, pre-1800 hours release	C1
		2×10^{-8}	2.4	ST flask - fuel incorrect, partial or complete separation of main flask components and 1800 - 2100 hours release	C2
		1×10^{-5}	2/3	Balance of combinations of fuel contents, flask damage and time of release.	C3

* For explanation of acronyms, see glossary, page 35.

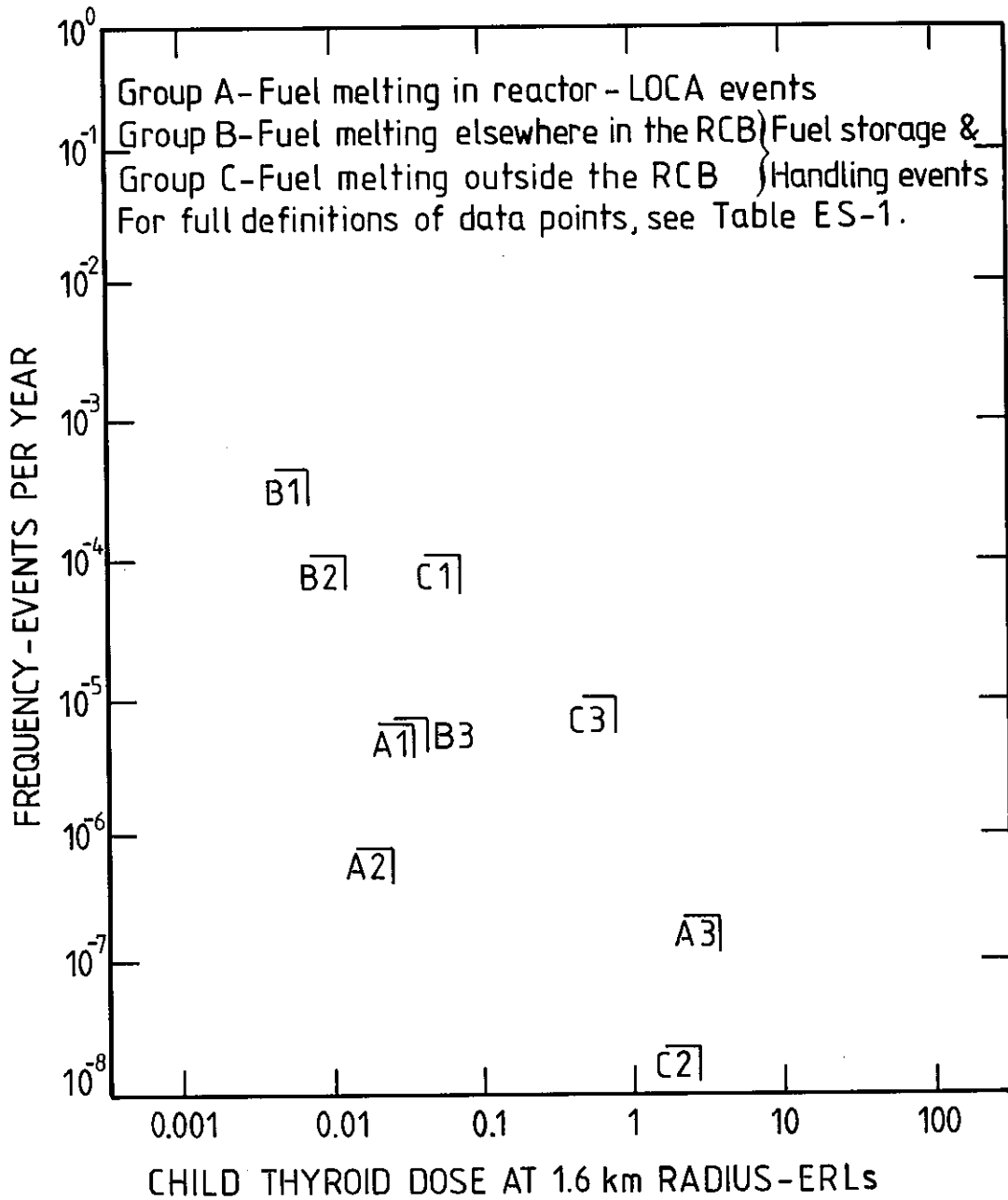


Figure ES-1 Frequency and consequences of accidents

REPORT

1. INTRODUCTION

The HS&WP Task (b) Group was established by the Director, AAEC Research Establishment. Its task required, and was restricted to, a re-examination, using Probabilistic Risk Analysis (PRA) methods and updated data and techniques of analysis, of all HIFAR accident sequences with potential for significant exposure of members of the general public to radiation. Its purpose was to quantify the public hazard from HIFAR operation; this had been demonstrated by deterministic analyses in the 1972 HIFAR Safety Document (HSD) and its 1983 Supplement (HSD-S) to be very low. Those analyses formed a part of the evidence on which the Commission's decision of February 1983, that retention of the 1.6 - 4.8 km radius land-use restrictions was no longer necessary, was based.

At present, only fault sequences initiated by within-plant failures (including human procedural errors) have been considered, and it is intended that PRA re-evaluations of the deterministically based HSD and HSD-S treatments of accidents initiated by external causes will be made as a later extension of the study.

2. BACKGROUND

The HSD described the reactor and its associated plant in detail, and by examination of possible, but very unlikely plant malfunctions and the performance of barriers provided for protection and for mitigation of consequences, demonstrated that operation of the reactor presented insignificant hazard to the general public.

In developing the HSD, a number of engineering features which could be modified or improved with advantage to overall safety were identified. Over a period, these changes were gradually incorporated either individually, or by being absorbed into more comprehensive plant improvement projects. Among the latter, perhaps those of greatest significance were upgrading of the reactor containment, and installation of an Emergency Core Cooling System (ECCS) as an additional barrier to Loss of Coolant Accidents (LOCAs). A LOCA was identified by the HSD analysis as the 'Maximum Credible Accident' (MCA).

In January 1983, the HSD-S was published. This supplement describes the above engineering changes, and reassesses the LOCA/MCA in the context of those changes, taking account also of more complex analyses of many aspects of reactor behaviour which have become feasible and been completed since the HSD.

The HSD-S, using essentially identical methods and data to those used in the original HSD MCA analysis, demonstrated that the modified reactor

substantially complied with all aspects of the AAEC's Interim Siting Criteria (1969).

Consequently, the Commission decided that maintenance of the existing land-use restrictions in the zone between 1.6 and 4.8 km radius from HIFAR was no longer justified, and recommended in February 1983 that they could safely be relaxed.

Regulatory practice world-wide, however, has been tending away from dependence solely on the MCA approach as used in the HSD and HSD-S, towards the use, where appropriate, of probabilistic treatments which take into account not only the consequences of plant failures, but also the probability or frequency of their occurrence. Unfortunately, a complete reactor study of this type is very costly both in time and in use of skilled resources.

The present study uses the probabilistic approach, and can be regarded as a subset of a full probabilistic risk analysis. It was intended to provide, on the shortest reasonable timescale, a quantitative PRA-based assessment of the more serious potential accident sequences for HIFAR. Hence the principal objectives of the study were

- (i) to identify fault sequences arising within the reactor plant which may have potential to produce radiation doses to the individual at 1.6 km radius in excess of NH&MRC [1973] recommended levels (ERL - Table 1) for which limited evacuation of some local population should come into consideration;
- (ii) to make use of reliability and performance data for the plant items and protective barriers in the relevant fault sequences to estimate the frequency with which the ERL can be expected to be reached or exceeded; and
- (iii) to make estimates, more realistic than those of the simplified and excessively conservative analysis of the HSD supplement, of the potential radiation doses at 1.6 km in such sequences, or in suitable artificial but bounding sequences.

The models analysed have been kept as simple as considered reasonable, whilst still remaining demonstrably conservative in relation to the more realistic qualitative descriptions provided of the complex physical processes occurring in actual accident situations. Furthermore, the safety philosophy of HIFAR, in common with normal reactor practice world-wide, is based on defence in depth; there are consequently multiple devices and procedures whose effect in practice would be to aid in mitigation of the consequences of LOCA events. Credit for some of these has been fully

or partially omitted in the analysis. It can therefore be anticipated that the results of the present study would still prove very conservative in relation to those which would derive from a more comprehensive and detailed PRA study.

3. REVIEW OF ACCIDENT SEQUENCES

The HSD (Section 7) discussed a very broad spectrum of potential accident sequences for the reactor. The HSD-S reviewed this spectrum and, in terms of 'credible' single and double fault concepts, concluded that the LOCA remained the maximum credible accident.

Only accidents in which a significant release of fission products from irradiated fuel could occur have potential for significant hazard to the general public. If the fuel from which the release occurs is in the reactor core, a concurrent or resultant breach of the Primary Coolant Circuit (PCC) is an additional necessary condition for potential public hazard. In this context, the present study has reconsidered the HSD potential accident spectrum, and has concluded that only three categories of sequences initiated by within-plant failures need be considered. They arise from

- (i) unscheduled reactivity additions to the reactor;
- (ii) loss of primary coolant circuit integrity; or
- (iii) events leading to coolant failure in handling, transport or storage of irradiated fuel.

The HSD (Section 7.4.3) considered that 'failure' of the PCC and later fuel melting might arise from a particular loss of primary coolant flow accident (LOFA), viz., one caused by a loss of all reactor electrical power supplies for a minimum of 3 hours. The 'failure' mode perceived, however, was rupture of pressure relief bursting discs, which would not result in coolant drainage. Although it is now considered that automatic venting via the bursting discs before the PCC design pressure is reached is not necessarily assured in this situation, the time available and the nature of the LOFA event are such that PCC damage can be readily prevented by valve operation, allowing venting to the active ventilation system. Coolant can therefore be lost from the circuit only by boiling, and at least 20 hours (adiabatic conditions) would have to elapse from the beginning of the LOFA before fuel could start to be exposed. Abundant time is therefore available to connect the emergency coolant make-up system (no electrical power required) which would prevent fuel melting. LOFAs are therefore not considered a potential public hazard.

The classes of event sequences analysed in the study are reviewed briefly in Sections 3.1 to 3.4, and their relationship in the study structure is shown schematically in Figure 1.

3.1 Reactivity Additions

No credible reactivity addition accident has been identified which could give rise to public hazard unless followed by multiple unrelated failures of the reactor protective systems and barriers. The probability of public hazard from any reactivity addition accident is therefore extremely low.

The HSD-S review of reactivity accidents identified two bounding cases (loss of a single Coarse Control Arm (CCA), and uncontrolled withdrawal of the CCA bank) and demonstrated conclusively that the loss of a single CCA, even when followed by failure of the protective system, could lead, at worst, to some fuel melting and damage to the PCC. The consequences would therefore be encompassed by those evaluated for LOCAs. Similarly firm conclusions regarding the consequences of uncontrolled CCA withdrawal could not be drawn, and the possibility of damage to both the PCC and the Reactor Containment Building (RCB) was conceded in this case. The necessary sequence of failures was, however, so unlikely that it was judged to be incredible.

Since the HSD-S, analysis of the latter event has progressed further. Although some work still remains to be done, an interim report [Connolly 1985] shows the resulting transient to be rather slow, and that it might result in melting of about the same amount of fuel as a $\sim 10 \text{ L s}^{-1}$ LOCA accompanied by ECCS failure. Generation of disruptive mechanical energy, capable by any means of breaching the RCB, can consequently be discounted [Clancy and Connolly 1985, Jostsons 1985]. The maximum consequences are therefore also similar to those of LOCAs accompanied by ECCS failure.

Connolly [1985] shows that most other credible reactivity accidents can result in no worse consequences than some melting of fuel, with the fission products so released being maintained within the PCC, and hence representing no public hazard. Only for a few very unlikely events could the possibility of consequent PCC damage not be entirely ruled out; for these, the maximum potential consequences are again equivalent to those of a LOCA accompanied by ECCS failure.

Corran and Moss [1985] have estimated frequencies for event sequences initiated by accidental reactivity additions accompanied by failures of the relevant protection systems. For simplicity, they very conservatively assumed that the PCC would be breached in all sequences in which any fuel melting might occur, and therefore would result in consequences equivalent to those of LOCAs accompanied by ECCS failure. Even so, the total frequency was estimated to be less than 5×10^{-7} per year. The estimated

frequency of reactivity initiated sequences which might conceivably result in worse consequences is much lower still, and detailed evaluation of such sequences is not justified. Reactivity initiated sequences have therefore been incorporated in the study by appropriately augmenting the frequency of 'LOCA with ECCS failure' sequences.

3.2 Loss of Primary Coolant Circuit Integrity (LOCA)

No LOCA can result in public hazard unless accompanied by failures of multiple reactor protective systems and barriers. LOCAs accompanied by such failures constitute the MCA analysed in detail in the HSD. Those analyses were updated in the HSD-S to take account of plant changes, particularly the provision of upgraded emergency core cooling arrangements. However, the very conservative simplifying assumptions of the HSD in relation to details of physical processes in the LOCA itself, the subsequent heating and pressurisation of the containment building, the physico-chemical behaviour of released fission products and the prevailing meteorological conditions were retained.

In the present study, assumptions relating to the above factors, although still remaining demonstrably conservative, have been refined, and probabilities assigned to all contributing events and failures. Analysis of LOCAs constitutes a major part of the study, and the principal contributing aspects are treated in Section 4.

3.3 Irradiated Fuel Storage and Handling within the RCB Boundary

The quantity of fission products that could be released in an accident involving irradiated fuel element transfers within the RCB or loss of cooling to the irradiated fuel element storage block SB-1 would be substantially less than for the whole core meltdown assumed in the Bounding Case LOCA analysis. Furthermore, no over-pressure would be generated in the RCB, and release of fission products to the external environment would be slower, being determined by diffusion and barometric pressure changes.

On the other hand, attenuation mechanisms for fission products might be less for these events than for LOCAs, where the fission products are released into the PCC and the Heavy Water (D₂O) Plant Room (DPR). Consequently, it was deemed necessary to analyse [Corran 1985(a) and (b)] the events discussed in the HSD and HSD-S, and by Nicholson [1983]. Accidents involving drainage of the coolant from fuel stored in SB-1 are of greatest significance in this class, which is fully discussed in Section 6.1.

3.4 Irradiated Fuel Transport and Handling Outside the RCB Boundary

Incidents in which there is potential for overheating of irradiated fuel to occur outside the boundary of the RCB involve only the ST flask,

which holds a maximum of 4 fuel elements. The pertinent operations are the transport by road of the flask on its purpose-built trailer from the RCB to B23, a distance of about 50 m, and a sequence of crane lifts, terminating in separation of the sections of the ST flask in the B23 storage pond. Subsequent movements of the fuel to the storage racks take place entirely under water in the B23 pond, and cannot lead to fuel overheating.

The ST flask is filled with water in which the fuel elements are immersed. The water acts as a heat transfer medium for fuel decay heat to the massive body of the flask and thence to atmosphere. The fuel transferred has normally cooled for a minimum of 37 days after shutdown of the reactor, and it is probable that melting could not occur even if the water were lost. Thermal analyses to establish this could not however be completed for the present study, and it is consequently assumed that fuel in the ST flask will melt if coolant is lost in circumstances such that its restoration cannot be guaranteed within a short time.

Corran [1985(b)] has studied ST flask operations, and estimates that the frequency of all sequences in which fuel melting was assumed possible is less than 9×10^{-5} per year. The consequences are considered in several categories, depending on the cooling time of the fuel, the time of day at which the flask movements take place and the severity of damage to the flask. A full discussion is given in Section 6.3.

4. THE BOUNDING CASE LOCA

The Bounding Case LOCA was intended to embrace the consequences, for all but a very small class of extreme situations, of all LOCA sequences from which some public hazard might result. Realistic models were adopted to the fullest extent possible while remaining demonstrably conservative. The latter requirement led to the use of 90% non-exceedence probability data (in the conservative direction) throughout, and necessitated adoption of mutually exclusive 'worst outcome' contributing components in some areas of the complex fuel melt/fission product transport scenario. The overall degree of conservatism introduced by the combination of these assumptions is difficult to quantify. The sensitivity of the results to some important individual assumptions has, however, been indicated by studying a range of variations around the bounding case data.

The major components of the Bounding Case LOCA study are briefly summarised in Table 2 and discussed in more detail in Sections 4.1 to 4.8. Because of the complexity of the topics covered and their interactions, it is strongly advised that the referenced supporting documents be consulted if more detailed and precise information is sought. It should be noted

that most aspects of Sections 4.3 - 4.8 are, except in respect of some specific data values, applicable also to the evaluation of the consequences of fuel storage transport and handling incidents described in Section 6.

4.1 The Melting Transient

Nicholson [1981] has exhaustively examined the failure probability of primary coolant circuit components and derived frequencies for 'small' ($\leq 12 \text{ L s}^{-1}$) and 'large' ($> 12 \text{ L s}^{-1}$) LOCAs. He argued that leak-rates $< 0.75 \text{ L s}^{-1}$ need not be considered as LOCAs, because of the long time available ($\sim 3/4$ hour minimum) after their inevitable detection, for action to prevent uncovering and subsequent melting of fuel even if the ECCS scavenge pumps failed to operate. Leaks smaller than 0.75 L s^{-1} would therefore be an operational inconvenience, rather than a source of public hazard, and require no further consideration in the LOCA analysis.

Nicholson [1982] reviewed the performance of the ECCS for the range of potential PCC leak-rates, and concluded that if both scavenge pumps operate, fuel melting will not occur for any size leak. If one scavenge pump fails, however, it is conceivable, although unlikely, that the 'Wolters' cooling' system may be only partially effective as a result of perturbations created by free fall return of the scavenged D_2O to the surface of the D_2O in the Reactor Aluminium Tank (RAT), and some fraction of the fuel may be at risk. This has been allowed for in the event tree (Figure 2), and it is conservatively assumed that 25% of the fuel (representing the tube for which Wolters' cooling is least effective in each fuel element) melts [Connolly 1982(b)].

For the Bounding Case, the pertinent situation is the very low probability one of a LOCA followed by ECCS failure. In these circumstances, the reactor top D_2O reflector and coolant annuli will drain at a rate determined by the size and location of the PCC leak, and will reach a level determined by the breach location. The bulk D_2O within the RAT but outside the fuel element shroud tubes will fall to the level of the Wolters' cooling holes, about 50 mm below the tops of the downcomers; thence the level can fall further only at a rate determined first by any leakage past the fuel element nozzle seats, and subsequently by boiling caused by fission product decay heat from fuel still remaining in the RAT.

Clancy and Connolly [1985] have examined the timing of events in LOCAs of different size when the ECCS scavenge pumps fail to operate. Shut-down of the reactor is inevitable even in the very unlikely event of failure of the protection system to trip on falling D_2O level, because the LOCA, by definition, involves removal of the D_2O top reflector, and renders the reactor subcritical.

Because the power of different fuel elements in the normally operating reactor varies widely, the maximum to average ratio typically being ~ 1.5 or more, coherent melting of a large number of fuel elements is impossible. Furthermore, within a given fuel element, individual tubes melt in sequence after the coolant is drained, so producing a further spread in time of the melting process. Depending on the size of LOCA, and hence on the decay power at the time the fuel elements drain, this first phase of the melting process is spread over periods from a few minutes to tens of minutes. The sudden fall of a large coherent mass (several kg) of molten fuel into water is hence impossible. Clancy and Connolly consequently considered it extremely unlikely that conditions for a significant chemical metal/water reaction could ever be achieved as a result of any LOCA. This conclusion is endorsed by Jostsons [1985].

Clancy and Connolly also showed that the outer fuel tube of each element cannot melt as long as the shroud-tube is surrounded by D_2O , and helium remains in the element coolant annuli. Provided that not less than 8 m^3 helium is in the gas-holder (the minimum permitted by HIFAR operational instructions), helium can be replaced by air only as a result of back-diffusion via the LOCA breach and is hence a relatively slow process. A second phase of melting may then occur, depending on whether D_2O is still in the RAT, on the time delay and on the original operating power of the particular fuel element. For completion of first-phase melting only, the calculations show a range from about 2/3 of the core inventory melted at about 8 minutes into the LOCA for a (highly improbable) leak-rate of 375 L s^{-1} down to about 1/5 of the core at about 50 minutes for 1 L s^{-1} .

Because molten uranium-aluminium alloy and molten aluminium are extremely mobile (viscosity similar to that of water and surface tension to that of mercury), any molten material will immediately fall from the fuel elements into the bottom of the RAT and thence to the risers and heat exchangers. There it will re-solidify quickly, giving up its latent heat of fusion to the PCC pipework and heat exchangers, and, depending on the location of the PCC breach, to the light water only or to both light and heavy water in the heat exchangers. A very large and readily accessible sink for the subsequent fission product decay heat is therefore available, even if the secondary coolant circulation has temporarily failed.

In the extremely unlikely event of a near double-ended 'guillotine' break in one of the PCC risers immediately below the bellows/gland assemblies, the geometry is such that up to a maximum of 40% of any fuel melted could conceivably fall direct to the DPR floor rather than reach the PCC

heat exchangers. However, this would involve so large a displacement of the broken ends of very large pipes in very short runs as to be inconceivable, and it is regarded as very conservative to assume that up to 20% of any molten fuel could reach the floor. There it would be associated with at least 2.85 t of D₂O, which is the content of the circuit above the bottom of the bellows units, and is the minimum quantity that must drain following a large LOCA.

Following first-phase melting, the fraction of fuel remaining unmelted in the RAT is assumed to give up all its decay heat to the D₂O (3.0 t), remaining in the RAT, which is slowly raised to boiling point. Taking account of typical, but conservative, loss-rates for the RAT D₂O by leakage past the fuel element nozzles, Petersen [1985(a)] shows that the minimum time to boiling for the least favourable situation is about 80 minutes. This is additionally conservative because of the adiabatic conditions assumed. The least favourable initial steam evolution rate at commencement of boiling would be about $0.07 \text{ m}^3 \text{ s}^{-1}$. Similar adiabatic calculations, for bounding cases of fuel reaching the heat exchangers and the D₂O plant room floor, show that no significant steam evolution could arise from either of these sources until some 3 hours (no secondary coolant circulating) or 9 hours (normal secondary coolant circulation) into the LOCA.

4.2 Heat Removal from the RCB

In considering the melting transients above, it has been assumed that the ECCS scavenge pumps have failed in addition to the breach of the primary coolant circuit which originally caused the LOCA. Although the probability that a further plant item would fail seems very remote, a fraction of the total unavailability of the ECCS scavenge pumps is attributable to loss of electrical supplies, and a fraction of this, due to loss of electricity feeders to the site, could in principle mean that secondary coolant circulation was also not available. Although diesel generators are provided for back-up electricity supplies to both the ECCS and the secondary coolant shutdown pumps, it is not mandatory that the diesel generator for the latter be available at all times when the reactor is operating. Accordingly, it has been assumed that secondary coolant circulation is lost concurrently with the ECCS, and restored 1 hour into the LOCA, by restart of the shutdown pumps. This is a very conservative assumption, since the loss of power supply feeders is independent of the initiating PCC breach, and power outages of longer than a few minutes' duration are very infrequent.

The heat exchangers are located in the DPR. Their external surfaces are fully accessible to the DPR atmosphere and provide an effective means for removal of heat and condensation of steam in the post-LOCA period. This is so, even with no secondary coolant circulation, because of the very large heat sink represented by the mass of water they contain. Heat transfer to the heat exchangers from the DPR atmosphere under an appropriate range of conditions has been studied by Chapman [1985] for no secondary coolant flow, and for circulation by the shutdown pumps. The details have been incorporated in the ZOCO model [Marshall and Rodd 1984] for RCB post-LOCA thermodynamics (Section 4.5).

Three space-conditioner subsystems (each of 2 units) are also available for removal of decay heat and control of overpressure in the RSB following a LOCA, as described in Section 3.39 of the HSD. The space-conditioners and their associated control systems, which are of commercial, rather than reactor engineering standard, have recently been modified and upgraded. The performance and reliability of the revised systems have been analysed by Allen [1983]. They were shown to be of only modest reliability, but to be effective, if available, in mitigating the consequences of a LOCA by controlling temperature and overpressure in the RCB, and removing aerosol particulates. Nevertheless, their classification as an 'engineered safety system' may be difficult to sustain formally, and they have conservatively been assumed unavailable for evaluating the Bounding Case LOCA consequences.

4.3 Leak-rate of the RCB

The specified operational target for RCB leakage is less than 1 vol.% per day at a test pressure of 10.3 kPa gauge. In practice, the regular test programs disclose from time to time that the leak-rate exceeds 1% per day; the source of enhanced leakage must then be identified and remedied. A level of 3% per day at 10.3 kPa gauge is defined in the HIFAR Operating Manual for mandatory notification to the Regulatory Bureau, and was adopted in the HSD supplement for evaluation of the radiological consequence of LOCAs.

Corran, Witt and Barbagallo [1985] have reviewed all available leak test data from HIFAR's operational records. From their analysis, they deduced the fractions of time for which it can be expected that RCB leak-rates of 1, 3 and 10 vol.% per day might be exceeded.

Corran, Witt and Barbagallo also showed that the $L \propto \sqrt{\Delta P}$ relationship between leak-rate L and pressure differential ΔP , traditionally used as a believed conservative means of deriving the RCB leak-rate at pressures

prevailing in accident situations from that measured in tests at the proof overpressure of 10.3 kPa, is of doubtful validity. They recommended a more conservative treatment, where the 10.3 kPa leak-rate is assumed constant with decreasing ΔP down to 0.5 kPa. Below 0.5 kPa, Petersen [1985(b)] recommended an error function transition to zero at 0 kPa. These recommendations have been used throughout, except for leakage due only to barometric pressure fluctuations, which was calculated using the model of Miller [1984].

4.4 The Fission Product Source Term

In the regulatory context, particularly for reactor siting judgments, it has been customary in considering the consequences of fuel melting accidents to assume that a large, arbitrary fraction of the fission product inventory is freed in a form in which it is subsequently available for release to the external environment [e.g. USNRC 1974].

Additional information, from laboratory experiments and from a number of fuel-melting accidents in research and power reactors, including the much publicised 'Three Mile Island' incident in 1979, has consistently indicated that this approach has been ultraconservative, particularly in relation to radio-iodines and volatile metallic fission products. The evidence is strong that the complex physico-chemical conditions of an accident/fuel melt situation drastically curtail the fraction of the fission product inventory which is available in forms conducive to subsequent release from the containment.

In the light of the data now available, regulatory bodies world-wide are considering the adoption of more realistic fission product release fractions for reactor safety assessment. Although eventual favourable response seems inevitable, the implementation of new criteria in the regulatory process is likely to be slow. This may be quite appropriate within the regulatory framework and the context of the deterministic 'maximum credible accident' approach, which still almost universally dominates legal aspects of regulatory processes.

The present HIFAR study was not intended to furnish a formal regulatory submission. Rather, it was intended to provide realistic, but conservative estimates of how often events in which the local population could be accidentally exposed to radiation could be expected to occur, and how large those radiation doses could be expected to be. Realistic, rather than arbitrarily conservative data are the only rational basis for such estimates.

Using the conditions of the Bounding Case LOCA melting transient as the base, May [1985] has assessed 'most probable' values for the key parameters relating to release of fission products from fuel and their subsequent behaviour, which govern their availability for eventual release outside the RCB. On the basis of the uncertainties ascribed to the most probable values, May derived '90% conservative bound' estimates which were adopted throughout for evaluation of accident consequences. Very simple, conservative models (Section 4.6) were adopted in the studies to take account of transport from the PCC to the DPR and thence to the main containment volume of the RCB, prior to atmospheric release.

Similar assessments were made by May for the conditions appropriate to fuel melting in the SB-1 storage block, in the L/U and ST fuel handling flasks, on the reactor top and in open air (Section 6).

4.5 Post-LOCA RCB Thermodynamics

The principal features of post-accident heat removal processes available to control and reduce subsequent temperature and pressure in the RCB have been discussed in Section 4.2. However, the inherent thermal capacity of the RCB structure and contained equipment, the building leak-rate, and changes in external atmospheric conditions are also important factors in the overall thermodynamics of the post-accident period.

The internationally used code ZOCO V was developed for thermodynamic analysis of containment building pressure/temperature transients resulting from LOCAs and PCC blowdowns in pressurised water power reactors. This code was suitably adapted by Marshall and Rodd [1984] for analysis of HIFAR RCB pressure transients using a two-compartment (DPR and rest of RCB) model. Time profiles of pressure and temperature were generated for the RCB for a variety of initiating events and combinations of heat removal mechanisms and leak-rates.

4.6 Fission Product Transport within the RCB

The RCB is a multi-compartment system in respect of the transport of fission products, which first must be transferred from the molten fuel to the primary coolant circuit, and thence via the DPR to the main containment volume. In practice, some hold-up and removal of fission products by decay and/or plate-out would occur at each stage, but to simplify the modelling requirements, transfer of the appropriate fission product fraction (Section 4.4) to the succeeding compartment was, with the exception that the release of iodine from contaminated water in the DPR was specifically modelled, assumed to be instantaneous. The result is that the activity released and remaining airborne after the first and second stage fuel melting phases is transferred unattenuated to the main RCB volume.

Delay, decay and deposition of fission products in the RCB have been treated using the AIRBORNE code [Miller 1985(a)]. The fission product physico-chemical parameters appropriate to the temperature, humidity and pressure calculated in ZOCO (Section 4.5) were obtained from May [1985]. Radioactive decay data for fission product nuclides were taken from ICON [Faircloth and Hopper 1970]. The AIRBORNE model allowed for appropriate changes of species (e.g. from gas to particulate) with the decay process for the calculation of subsequent migration. For each RCB leak-rate, AIRBORNE calculated the quantity of each nuclide released from the RCB as a function of time.

4.7 Meteorology and Dispersion of Released Radionuclides

Records of meteorological conditions for the Lucas Heights region extending over eight years have been analysed by Petersen and Clark [1985]. They derived summary statistics for the occurrence and persistence properties of the wind direction time series and for the correlated wind speed and dispersion parameter time series in terms of daily and seasonal variations.

Assuming releases to be at ground level, and adopting 90% non-exceedence values in the conservative direction for the relevant meteorological parameters, Petersen and Clark derived dilution factors at 1.6 km suitable for making conservative assessments of the consequences of episodic releases of radioactivity from the RCB.

Because of the nature of the Lucas Heights terrain, it has been traditionally assumed that the Woronora Valley would be likely preferentially to receive a high proportion of any radioactive nuclides escaping the HIFAR RCB, through 'funneling' via natural drainage contours, and that there would be little further dilution of airborne radioactivity because of the constraining walls of the main river valley. In consequence, the same radioactivity concentrations were expected at greater distances from HIFAR along the Woronora Valley than would be found in other directions.

Accumulated experience world-wide has indicated that such deductions based on qualitative general arguments are not always valid. Rather, the interactions of contributing phenomena in particular situations must be taken into account. The influence of the Woronora Valley on the transport and dispersion of airborne radioactivity is specifically considered by Petersen and Clark [1985], using simple physical models. They demonstrate that, for corresponding meteorological conditions, radiation doses in the Woronora Valley would not be significantly greater than those at the same distance from HIFAR along any other radius vector.

On the basis of Petersen and Clark's meteorological models and data, the dispersion of fission products released from the RCB and the resulting potential radiation doses to an individual at 1.6 km radius (see Section 4.8) were calculated using the DISPDOSE code [Miller 1985(c)].

4.8 Dosimetry

Petersen [1985(c)] has reviewed the various pathways by which an individual can receive radiation doses (beta and gamma) to the whole body or particular organs as a result of airborne radioactivity. Broadly, there are four main pathways

- direct external doses from the radioactive plume;
- committed internal doses from inhalation of activity from the plume;
- direct external doses from activity deposited on the ground; and
- committed internal radiation doses from ingestion of contaminated food or water.

The first two pathways contribute the major doses to an individual, and have been analysed assuming unshielded, out-of-doors exposure throughout the passage of the activity.

Both of the last two pathways stem from deposition of activity onto the ground. Dry deposition during atmospheric transport is not expected to be significant for HIFAR accidents as only submicrometre aerosols are likely to be created [May 1985]. Rain can enhance the deposition processes, but activity deposited on foodstuffs or the ground is then likely to be washed away. Furthermore, simple countermeasures, such as adequately washing or discarding domestic agricultural products will effectively eliminate potential doses via the ingestion pathway; rainwater from domestic tanks can be discarded. Only the direct external dose contribution arising from β and γ emitters deposited on the ground has therefore been considered.

Individuals exposed to a radioactive plume will receive differing direct radiation doses to specific organs and to the whole body. The gamma radiation emitted will give doses to all organs and to the whole body, whereas the beta radiation will give a significant dose only to the skin. There are several sets of dose conversion factors which give the dose to a specific organ when an individual is immersed in a semi-infinite cloud of radioactivity. The most recent set [Kocher 1980] has been used in this study. Conservatively, no finite cloud correction factor has been applied for the overestimate of doses given by the semi-infinite cloud approximation at distances within a few kilometres of the source.

Individuals immersed in a passing radioactive plume will also receive differing radiation doses to specific organs by inhalation. These doses were evaluated using the inhalation committed dose conversion factors published by the International Committee on Radiological Protection [ICRP 1979, 1981, 1982(a) and 1982(b)]; breathing rate data based on 'Reference Man' [ICRP 1975]; and child thyroid dose enhancement factors from the US Nuclear Regulatory Commission's Reactor Safety Study [USNRC 1975]. Twenty-four hour per day exposures were assumed throughout.

For the range of events studied, calculated doses at 1.6 km radius from HIFAR were compared with NH&MRC [1973] ERLs for consideration of limited evacuation of the local population. These ERLs are summarised in Table 1. Also given in Table 1, for comparison, are more recent ERLs of broadly similar intent, proposed by the UK National Radiological Protection Board (NRPB) and the ICRP. These data were taken from a recent review by Petersen [1985(d)] which concludes that the 1973 NH&MRC recommendations are consistent with the more recent approaches for all practical purposes.

5. REACTIVITY ACCIDENTS

Strict administrative controls are applied to limit the probability of occurrence and resulting consequences of any inadvertent addition of reactivity to HIFAR; they cover, for example, procedures for changing fuel or experimental rigs, operation of particular plant items, and mandatory limitation of the reactivity which may be invested in individual devices.

As a high flux research reactor, however, it is essential for practical operational requirements that HIFAR carry a rather large core excess reactivity margin, held down by fixed or movable neutron absorbers. Potential for significant, inadvertent additions of reactivity therefore cannot be ruled out, even though the likelihood of such occurrences is extremely small.

The first line of defence against power transients arising from any unscheduled reactivity additions is provided by the reactor Safety Shutdown System (SSS). This system uses diverse and redundant detectors and logic elements to monitor key operational parameters, particularly the neutron flux. If any of these parameters (or, in some cases, the rate of change of the parameter) is established to be outside predetermined limits, the SSS responds by allowing strong neutron absorbers (the CCAs and, in some cases, the safety rods as well) to fall into the core under gravity, to terminate the reactivity excursion. It has been demonstrated, using conservative data for system response times and CCA reactivity insertion characteristics, that no credible accidental reactivity addition will result in core

damage, and hence no possibility of any public hazard can arise, provided the SSS operates as designed [Connolly 1985]. The reliability of the SSS in detecting and responding correctly to transients arising from reactivity additions has been analysed by Corran, Witt and Moss [1985], and Moss and Corran [1985] who concluded that the probability of failure is less than 10^{-4} per demand.

In the unlikely event that this very reliable engineered defence against accidental reactivity additions should fail, the inherent physical response characteristics of the reactor itself provide a further and fundamental line of defence. Temperature changes in fuel and moderator, and in more severe cases, the generation of moderator voids by boiling, all constitute powerful negative reactivity feedback mechanisms which act to limit the severity of the power transient that would otherwise follow the initiating accidental reactivity addition.

Clancy, Connolly and Harrington [1975 and 1976] developed models to represent the response of water moderated research reactor cores to a wide range of reactivity transients. They were incorporated in a computer code ZAPP [Clancy 1983] and applied to a representative range of reactor cores from the SPERT series of experiments, including one closely resembling HIFAR [Connolly 1982(a)]. The results were in excellent agreement with the measured burst characteristics (energy, peak power, peak temperatures) for all transients up to those in which substantial damage to the core resulted. Self-limiting transients for HIFAR were studied using these methods by Connolly and Ferguson [1978].

Connolly [1985] has recently extended these models and applied them to determine the response of HIFAR to a generalised broad range of ramp and step additions of reactivity. In addition to reliance only on inherent reactivity feedback mechanisms, the extended models allow the effects of normal and some notionally degraded responses of the SSS to be assessed. The response fields defined by this generalised approach were then used to assess the probable consequences of particular accidental reactivity additions in HIFAR.

Since the present study is concerned with potential for public hazard, it is necessary to discuss in detail only reactivity additions sufficiently large that both melting of fuel in the core and breach of the PCC might occur.

Reactivity accidents considered capable of causing significant power excursions were extensively reviewed in the HSD and HSD-S. Only a few categories were regarded as credible, viz. maloperation of secondary cool-

ant pumps or valves; flooding of in-core voids; rig withdrawal; single CCA failure; and continuous withdrawal of the CCA bank. Re-examinations of these categories [Connolly 1982(a) and 1985] showed that only the last two have potential for public hazard.

5.1 Failure of a CCA

The HIFAR CCAs are of the semaphore type, operating between horizontal (maximum core reactivity) and 56° below horizontal (minimum core reactivity). Failure of a CCA mechanical linkage could allow the arm to swing past the minimum core reactivity position, to hang vertically just outside the core, where it controls very little reactivity. The consequence is the rather rapid addition of the reactivity worth of the failed arm to the core. The resulting power transient must be terminated by the release of the remaining 5 CCAs to fall to the minimum core reactivity position.

In practice, if such a failure were to occur at all, it would almost certainly occur when the stress in the linkage was greatest, namely during the arrest of the CCA bank at the minimum core reactivity position. The reactor would then be subcritical by the worth of the bank, and since no individual CCA can contribute more than at most 1/4th of the total bank reactivity, no power transient would occur, and the reactor would remain safely in the shutdown state.

Nevertheless, there is a small but finite probability of linkage failure at other times. A number of analyses, of which the most recent and comprehensive treatment is that of Connolly [1982(a)], have over the years examined the failure of the most reactive CCA under least favourable conditions. The consequences can be limited by the SSS which provides halving time, doubling time and high flux trip protection against the event. If these fail, the inherent reactor reactivity feedback mechanisms operate to restrict the energy release in the first transient power spike, and provide additional time for insertion of the remainder of the CCA bank to terminate the excursion.

All analyses have shown conclusively that, even under conservative assumptions, operation of the halving time trips in accord with their performance characteristics, as measured by Corran and Witt [1978], will terminate any CCA failure transient without damage to the core. Connolly [1982(a)], using the models validated by comparison with the SPERT experiments, has further shown that in the event of failure of the halving time trips, the high flux trips by a large margin and the doubling time trips by a lesser one have sufficiently rapid response to terminate the transient without core damage. Therefore, only in the extremely unlikely situation

of CCA failure followed by failure of the SSS to trip the reactor, does the possibility of any fuel melting arise. Corran and Moss [1985] estimated the associated frequency of occurrence to be less than 1.1×10^{-7} per year.

Connolly's analysis of this latter situation, again using conservative data and assumptions, leads to the conclusion that although some fuel melting cannot be ruled out, it is extremely unlikely that pressures capable of damaging the PCC by any means could result. The possibility of generating missiles capable of penetrating the RCB can be totally discounted. Because it is difficult to define with precision the threshold at which breach of the PCC could occur, it is conservatively assumed for this study that it occurs in all events involving some fuel melting; the consequences therefore are similar to those of LOCAs followed by ECCS failure.

5.2 CCA Bank Withdrawal due to Control System Failure

Administrative controls set a minimum critical CCA angle of 10° for HIFAR; this corresponds under least favourable core configurations to about 10 per cent excess reactivity invested in the CCA bank. Clearly, the addition of such a large quantity of excess reactivity to any reactor could lead to severe consequences, and the CCA control system is provided with redundant features and safeguards to render inadvertent withdrawal of the CCAs very improbable.

Corran and Moss [1985] analysed the CCA control system, including its human operator interface, for failures which could result in unscheduled continuous operation of the CCA withdrawal mechanisms. They conservatively estimated the frequency of occurrence from all initiating causes to be less than 1.3×10^{-2} per year.

Connolly's [1985] review of generalised ramp reactivity additions to HIFAR embraced those due to unscheduled CCA withdrawal. Although the height of the CCA withdrawal ramp may be very large, the ramp-rate is modest, and Connolly's results show conclusively that even with very conservative data and assumptions no core damage can occur provided that the SSS operates as designed.

Complete CCA withdrawal, with concurrent failure of the SSS, is extremely unlikely. Corran and Moss conservatively estimated the frequency of occurrence to be less than 3.5×10^{-7} per year. Connolly, by induction from the observed consequences of accidents caused in several reactors by large reactivity additions at comparable rates, concluded that the resulting transient would be terminated by void formation and molten fuel transport. Core damage would be extensive, but generation of significant disruptive mechanical energy can be discounted. Breach of the PCC would be

unlikely to result, but is again conservatively assumed to occur in the present study. The consequences of unscheduled CCA withdrawal accompanied by SSS failure are therefore again similar to those evaluated for LOCAs accompanied by ECCS failure.

6. IRRADIATED FUEL STORAGE AND HANDLING ACCIDENTS

As part of each HIFAR 28-day routine operating fuel cycle, a number of irradiated fuel elements reach the end of their useful in-core life and must be removed for transfer to the spent fuel storage pond in Building 23. These elements may be directly replaced by new, unirradiated fuel elements at the same core positions or, more usually, further transfers of partially burnt-up fuel elements between core positions or from the SB-1 storage block may be required. This involves additional movements of highly radioactive fuel elements outside the reactor core. It is mandatory that no fuel element be moved until it has cooled for a minimum of 48 hours after reactor shutdown unless specific written approval is given by the Leader, HIFAR Operations Section.

The fuel elements involved in these operations are intensively radioactive and generate substantial self-heating. As well as providing adequate radiation shielding, the transfer/transport containers must therefore make adequate provision for heat removal to ensure that the contained fuel cannot overheat. Such overheating, if it occurs, may lead to loss of integrity of the fuel cladding, or fuel melting, and subsequent release of fission products to the environment.

Transport and storage procedures for irradiated HIFAR fuel were examined in detail by Corran [1985(a) and (b)]. He concluded that the potential for public hazard due to accidental overheating of irradiated fuel outside the reactor can logically be analysed in three categories:

- (i) storage block SB-1 coolant failures;
- (ii) transfer operations within the RCB boundary (single element L/U flask and four-element ST flask);
- (iii) transfer operations outside the RCB (four-element ST flask).

6.1 Storage Block, SB-1

SB-1 contains two tanks, water filled and cooled. The main tank can hold up to 48 fuel elements, and the subsidiary tank a maximum of 4. The latter is intended only for fuel with defective sheaths, and is only very infrequently used. It is even less likely that it would contain more than one element at any given time, and only the main tank is considered in the following analysis.

In addition to spent elements discarded from the HIFAR operating program immediately completed, and awaiting transfer to the B23 pond, SB-1 may contain partially burnt-up elements discharged on completion of earlier operating programs, and awaiting return to the core.

The integrity of SB-1 and the reliability of its cooling systems were analysed by Corran [1985(a)], who estimated the frequency of failures which could result in fuel overheating to be less than 10^{-4} per year. Since Corran's analysis, the SB-1 tank has been replaced by a new one, manufactured to a higher integrity design, and to modern quality assurance standards. Although the theoretical reliability of the new tank should be better than the old, it is not considered prudent to claim credit for such improvement until reasonable operating experience with the new system has been accumulated. Data for the old SB-1 tank have been used for this study, with the implied assumption that the reliability of the new tank will at least not be inferior to the old.

Only infrequently would more than 20 elements be present in the block [Bezimienny 1984] and the actual inventory is usually substantially less. For 20 elements, not more than 8 would be from the immediately preceding program with a minimum of 2 days' cooling from reactor shutdown, with the remainder having cooled for a minimum of 30 days since removal from the reactor at previous shutdowns. Although it is entirely possible that there would be insufficient self-heating to melt any of this fuel in the event of loss of coolant from the block, melting has been assumed to occur, and the resulting consequences evaluated.

The 'infrequent' occasions on which the contents of SB-1 might significantly exceed those just considered may be specially scheduled, as for example, in connection with a major reactor maintenance shutdown, or may be consequent upon an emergency, such as a reactor LOCA. In either case, it may be necessary to transfer a full core of 25 elements to the block.

The estimated frequency of coolant failure in such scheduled circumstances is less than 4×10^{-6} per year and, because of the pre-planned nature of the event, it is likely that no other fuel would be in the block. Normal restrictions on the maximum decay power of fuel elements being transferred would apply.

In the emergency situation (estimated frequency of SB-1 coolant failure in such circumstances is less than 10^{-7} per year), it must be assumed that some or all of the fuel elements in the core would be transferred to SB-1 as soon as possible. Although it would be very difficult to transfer all 25 elements from the core in less than 2 days, some fuel with lesser

cooling times could be in SB-1 for a short period. In addition, up to 20 other fuel elements, in accord with standard practice, could already be present in the storage block. Of these, however, up to 8 must have cooled for at least 5 days more, and the remainder at least 33 days more than the elements newly transferred. The prevailing physico-chemical conditions determine that the highest external radiation doses will result if a newly transferred full core of elements are alone considered to melt [May 1985]. Overall it was considered adequately conservative to take melting of 25 2-day cooled elements as providing the upper bound estimate of consequences for all SB-1 events included in the 'infrequent' category.

Compared with the reactor LOCA fuel melting considered earlier, the consequence evaluations require changes in the fission product source term both because of the additional fuel cooling time, and the different physico-chemical conditions prevailing in the melt locale. Perhaps of even greater significance is the absence, in SB-1 events, of a significant heat source to pressurise the RCB and drive the fission product leakage, which is therefore mainly governed by 'breathing' due to barometric pressure changes.

6.2 Fuel Transfer Operations within the RCB

Most irradiated fuel transfer operations within the RCB involve only single fuel elements and the L/U flask. The sole significant exception is the ST flask, in which up to 4 elements could be involved if a coolant failure occurred after its loading but prior to its leaving the RCB for transfer to B23, or if it were returned to the Vehicle Air Lock (VAL) following development of a leak outside the RCB.

It is by no means certain that equipment malfunctions or maloperations leading to cooling impairment during these operations would necessarily cause sufficient overheating for significant fuel damage to result. For most such occurrences, the pace of events following the failure would allow reasonable time for the implementation of action to prevent fuel melting. Some credit for such action has been assumed in the analysis.

Fuel transfer operations have been reviewed in detail by Corran [1985(a)], who has made numerical estimates of the likelihood of equipment faults and human errors. Corran considered that fuel element damage caused by jamming in the L/U flask door due to procedural errors dominated the probability of occurrence of fuel overheating. He recommended that a lapsed proposal to install a door interlock to reduce this probability be revived. This recommendation will have been implemented [Parsons 1985] before this report is formally issued. Corran [1985(d)] has revised his

[1985(a)] results to give credit for this modification; the revised data are used in the present analysis.

Assuming, as in the case for SB-1 coolant failures, that fuel melting is the eventual result of L/U flask failure sequences, it is now estimated that the frequency of this class of accidents is less than 5×10^{-4} per year. The circumstances determining the release of fission products are then in most respects rather similar to those for fuel melting in SB-1; the actual consequences are, however, lower because of the smaller quantity of fuel involved.

6.3 Fuel Transfer Operations Outside the RCB

Fuel transfer operations outside the RCB, in which fuel overheating could conceivably occur, are restricted to those involving the ST flask. They can occur only in handling operations in B23, or by transport accidents during the very short (~ 50 m) road transit between the RCB vehicle air lock and B23.

The ST flask holds a maximum of 4 fuel elements, and is water-filled. The water acts as a heat transfer fluid, and any damage leading to its loss can be expected to lead to higher than normal temperatures of the contained fuel elements.

Except for rare, emergency situations where, subject to specific approvals, earlier transfer might be deemed advantageous, fuel element transfer operations involving the ST flask do not commence earlier than the 9th day after reactor shut-down [Bezimienny 1985]; this is therefore the minimum fuel cooling period which need be considered in evaluating ST flask events. However, standing instructions require that fuel to be moved in the ST flask must have cooled for a further full operating period of 28 days [Gleed 1986]. Unless a loading error has occurred, fuel elements being transferred in the ST flask have normally cooled for a minimum of 37 days since the end of their last power cycle in the reactor.

It is considered unlikely that even 9-day cooled fuel could reach melting temperature in the event of loss of ST flask cooling water; for 37-day cooled fuel, such temperatures are even less likely to be attainable. Nevertheless, it has not been possible to demonstrate this conclusively to date and it has been assumed that fuel melting will follow loss of ST flask water in all except the minor damage situation discussed below.

Corran [1985(b)] has made a detailed assessment of the failure modes and transport accidents which might result in the loss of flask coolant. Events are considered in two main classes, depending on whether the flask remains upright, sustaining only minor damage, or topples and falls, so

that more significant damage, in some cases with separation of flask components, must be assumed to occur.

For events in the first class, there would be no local radiation hazard, and it is considered that the time available would be sufficient to enable temporary plugging of leaks and refilling of the flask with water to be effected and so prevent fuel melting. There would consequently be no associated public hazard.

For events in the second class, whose total frequency is estimated to be less than 9×10^{-5} per year, it is assumed that fuel melting ensues. In a fraction (conservatively estimated to be not more than 20 per cent) of these, some separation of flask components has been assumed possible, allowing ('STF halves ajar') either a fairly short and direct escape path to atmosphere for the fission products released, or worse ('STF halves apart'), direct exposure of the fuel elements.

ST flask movements are scheduled for completion during normal day shift hours [Bezimienny 1985]. Quite favourable meteorological dispersion conditions would therefore apply to the fission product releases resulting from the ST flask events in which fuel melting is assumed to occur. However, less favourable (evening) dispersion conditions are assumed to apply in a fraction (1/10th) of the events considered. This allows for the occasional unforeseen operational delay, or for a slow melt/release sequence occurring late in the day shift.

7. FREQUENCY AND CONSEQUENCES OF IN-CORE FUEL MELTING ACCIDENTS

The analysis of these accidents includes both those for which the initiating event is a direct failure of the PCC, and those for which such failure is assumed to occur as a consequence of a sufficiently severe inadvertent reactivity addition.

The analysis is carried out in terms of the contributing parameters for the 'Bounding Case LOCA' discussed in Section 4 and includes appropriate contributions in the frequency domain from the relevant reactivity addition events discussed in Section 5. The event tree is shown in Figure 2, and details of the associated discrete fault sequences are given in Table 2.

7.1 Frequency

Should a LOCA occur, its consequences would be influenced by the leak size; the number of ECCS pumps available to return the D₂O to the RAT; the availability of adequate secondary coolant; the availability of space conditioners to cool the RCB; the leak-rate of the RCB; and the meteorological conditions prevailing after the accident.

LOCA fault sequences were developed by Corran [1985(c)] and are summarised in Figure 2. The consequences can be divided into three classes. Sequences (the great majority) which pass from LOCA initiation (point 0) to point 6 are those in which the fuel does not melt, no fission products are released, and there is no public hazard. Sequences which pass from point 0 to point 13 (the Bounding Case LOCA) or point 14 are those in which the fuel melts to some extent and fission products are released to the environment, but the consequences are in no case more than a small fraction of the child thyroid ERL. The remaining sequences are those which pass from point 0 to point 15. Such sequences include those of the greatest severity considered credible. There may be prolonged failure of the cooling systems and the RCB may have an abnormally high leak-rate. As a result, the fission product source term may be sufficiently high in some of the sequences to necessitate considering some public evacuation. However, the development of these sequences is relatively slow and there is no likelihood of instantaneous release of large quantities of fission products.

The estimated probabilities and frequencies of the various sequences were derived from a number of sources, and have been [discussed in detail by Corran [1985(c)]]. They are shown in Figure 2 and Table 3; the additional small contributions from reactivity insertion events are also indicated.

Over 99 per cent of all LOCAs that occur will be terminated without any fission products being released from the fuel. The frequency of the whole class of sequences in which some fission products are released from the fuel and from which some public radiation exposure may result is estimated to be less than 7×10^{-6} per year. The frequency of the class of sequences whose consequences might in some cases require that limited evacuation of the local population be considered is estimated to be less than 2×10^{-7} per year.

7.2 Consequences

Each major factor contributing to the Bounding Case LOCA fission product release, and eventual public radiation exposure at 1.6 km radius from HIFAR has been discussed in some detail in Section 4 and the relevant references. The models adopted and the conservative assumptions associated with them require no further description or justification in this section.

The Bounding Case LOCA dose estimates have been calculated by Miller [1985(b)] using a suite of computer codes which model the processes described in Sections 4.6 - 4.8. Miller also calculated the necessary fuel fission product inventories using the ICON code [Faircloth and Hopper 1970]. Case data given by Miller [1985(b)] include the appropriate fuel

history, the melt model (Section 4.1), the appropriate release fractions from molten fuel (Section 4.4), RCB time-dependent thermodynamic conditions (Section 4.5), atmospheric transport and dispersion data (Section 4.7) and dosimetry data (Section 4.8).

In addition, a range of cases was calculated in which individual features of the Bounding Case LOCA model and data were changed; these cases provide some guidance to the effect of the various conservative assumptions of the model, and to the limiting consequences for the small group of sequences whose consequences are not embraced by the Bounding Case LOCA.

The cases considered are detailed in Table 4(a), and the corresponding estimates of doses at 1.6 km radius from HIFAR are given in Table 5(a). To facilitate comparison, if required, with other fission product release treatments (cf Sections 4.4, 4.6), radioactivity inventories initially present and the amounts released for some important fission product categories are given for each case in Table 6(a).

Comparison of Table 5 results with the ERLs [NH&MRC 1973] summarised in Table 1 shows that in all cases the child thyroid dose is the most critical parameter. For the Bounding Case LOCA, Case 4, the child thyroid dose is only 32 mSv, less than 1/30th of the ERL of 1000 mSv. The estimated frequency (Figure 2, point 13), is less than 6.3×10^{-6} per year. Even if the 99% bound of 10 vol.% per day is assumed for the RCB leak-rate (Case 19), the consequences are still less than 1/20th of the ERL. The estimated frequency of these additional sequences (Figure 2, point 15) is less than 7×10^{-7} per year. It follows that all event sequences which could result in worse consequences are included in the small residual class (Figure 2, point 15) for which the estimated frequency of occurrence is less than 2×10^{-7} per year.

Case 20 of Table 5(a) further extends the plant failures encompassed by the Bounding Case LOCA, by assuming that the RCB fails to seal, so allowing the one change of air per hour typical of an ordinary building. Even then, the resulting child thyroid dose (extrapolating from 1.9 ERL at 10 days after the LOCA) reaches only 2-3 times the ERL. This can be regarded with some confidence as being the practicable upper bound of consequences for the full spectrum of in-core fuel melting accidents.

The remaining cases of Tables 4(a) and 5(a) can be used to estimate the sensitivity of the calculated doses to specific conservative assumptions in different parts of the overall model. Case 16 is of particular importance. It differs from Case 4 by using most probable or median data values throughout, rather than 90% non-exceedence probability

values, and hence by comparison shows reasonably directly the influence of compounded conservatism on the bounding case doses. The doses, even though the full core melting scenario is retained, are less than 1/10th of those for Case 4; hence by transposition to Case 20, it appears so unlikely as to be in practical terms incredible that evacuation would ever need to be considered as a result of any in-core fuel melting accident.

8. FREQUENCY AND CONSEQUENCES OF OUT-OF-CORE FUEL MELTING ACCIDENTS

As mentioned in the Section 6 review of these accidents, it is very probable that for most, if not all, the events included in the category, natural heat-loss mechanisms would in practice be sufficient to ensure that self-heating cannot lead to fuel melting and the release of fission products.

In the absence of definitive proof to the contrary, however, it has been assumed that fuel melting and fission product release will follow the failure of designed cooling provisions. The only exceptions are some cases of minor damage to SB-1 and the ST flask, where it is considered that temporary repairs could be effected to prevent fuel melting. The derived frequency/consequences spectrum is therefore very conservative. Heat transfer studies now in progress may, in due course, enable most of the sequences, which in the present work are found to result in some public exposure to radiation, to be eliminated from that category.

Assessment of the consequences of out-of-core fuel melting accidents is less complex than for LOCAs, particularly in regard to post-accident thermodynamics. Fission product releases from the melt appropriate to the physico-chemical conditions prevailing must be established, and May [1985] has treated the necessary situations. If melting occurs outside the RCB, only atmospheric dispersion modelling appropriate to the time and period of the release [Petersen and Clark 1985] is then necessary to estimate dose-rates. A further step must be interposed to allow for attenuation of the release if melting occurs within the RCB. In contrast with the LOCA situation, however, RCB leakage is then not governed by temperature induced pressurisation, but by RCB 'breathing' due to barometric pressure changes [Miller 1984]. Consideration of RCB heat removal systems is therefore not required.

For these reasons, it is logical and convenient to treat out-of-core fuel melting accidents under two main categories only, depending on whether the resultant assumed fuel melting occurs within or outside the RCB boundary. Event and fault trees for all fuel storage and handling accidents have accordingly been developed in a manner consistent with this categorisation.

Corran [1985(a) and (b)] has examined fuel storage and handling accidents in detail, and developed appropriate event and fault trees, which are given in Figures 3 - 6. Corran's analysis is the basis of all ensuing frequency-domain results. It should be noted that his SB-1 data relate to the tank which has now been replaced; an assumption that the reliability of the new tank is not inferior to that of the old is therefore inherent in the analysis, as discussed in Section 6.1. Credit, by appropriate reduction in frequency, has however been taken in the analysis for the L/U flask door interlock discussed in Section 6.2.

A set of 'bounding consequences', corresponding to the set of 'Top Events' from the above event and fault tree analyses, is included in Table 5. Except for the omission of RCB heating/cooling processes as discussed above, the procedures closely followed those for LOCAs. Fission product inventories were appropriate to the fuel quantity and cooling period. The fractions and forms released on melting were appropriate to the location and physico-chemical conditions [May 1985]. The resulting doses, which are discussed in Sections 8.1.2 and 8.2.2, were derived using atmospheric dispersion models [Petersen and Clark 1985] appropriate to the time and duration of release for each event considered. 90% non-exceedence conservative values of participating parameters were again assumed throughout.

8.1 Fuel Melting Occurring Within the RCB Boundary

Initiating events which could result in fuel melting within the RCB boundary are:

- (i) SB-1 storage block coolant failure (Section 6.1).
- (ii) Coolant failure or procedural faults (including crane failures) in single fuel element (L/U flask) transfer operations (Section 6.2).
- (iii) Damage to the ST flask (containing up to 4 fuel elements), causing drainage of the contained water. The only events included in the present category are those in which damage is sustained before the flask leaves the RCB, or if sustained outside, is not of a type which would prevent the unhampered return of the flask and trailer to the VAL (Section 6.3). Other ST flask events are considered in Section 8.2.

8.1.1 Frequency

- (i) From his detailed review of SB-1 storage block events, Corran [1985(a)] concluded that only leaks of greater than 0.04 L s^{-1} accompanied by other plant or procedural failures could result in

significant overheating of fuel. On the basis of operating experience with DIDO type reactors, but taking account of data for failure of tanks of all kinds, he adopted an occurrence rate for all leaks in the tank and associated pipework of 5×10^{-3} per year. From assessment of the probable size and location distribution for the leaks which might occur, an appropriate event tree was developed (Figure 3). The estimated frequencies for events with bounding SB-1 inventories corresponding to the normal, infrequent and emergency situations discussed in Section 6.1 are less than 10^{-4} , 4×10^{-6} and 10^{-7} per year, respectively.

- (ii) Relevant events involving the L/U flask can involve only one fuel element, which has cooled for a minimum of two days following reactor shutdown. Corran's detailed review of L/U flask events showed procedural errors and crane failure to be the most likely causes of fuel overheating. He assessed the overwhelmingly dominant contribution ($\sim 10^{-2}$ per year) as likely to arise from maloperation of the flask door, leading to jamming of the fuel. In comparison, all other contributing causes were together estimated to have a frequency of occurrence of less than 3×10^{-4} per year.

As a result of the proposed installation of an interlock (see Section 6.2) Corran [1985(d)] reassessed the contribution arising from flask door maloperation, and estimated it to be less than 10^{-4} per year. The frequency of all relevant L/U flask events is therefore now estimated to be less than 4×10^{-4} per year (Figure 4).

- (iii) The remaining contribution to accidents in which fuel melting occurs within the RCB boundary arises from events involving the ST flask, containing up to 4 fuel elements, normally having a minimum of 37 days' post-shutdown cooling.

The relevant initiating events are those in which flask damage and drainage of coolant water occurs either before the flask on its trailer leaves the VAL, or outside the RCB in circumstances which do not damage the trailer significantly, so that return of the flask to the VAL is possible with only minor delay. The only sequences in which it is considered that the flask might be sufficiently damaged for fuel melting not to be preventable are those involving crane operations. The estimated frequency is less than 10^{-5} per year (Figure 5). Separation of

the flask main components is assumed to occur in up to 20 per cent of these events.

8.1.2 Consequences

It was not evident, a priori, whether the melting of a large quantity of fuel in SB-1 was likely to have worse or lesser consequences at 1.6 km radius than the melting of a much smaller quantity in less favourable conditions elsewhere in the RCB. Furthermore, physico-chemical considerations introduce non-linearities which may make simple scaling in terms of fuel quantity invalid in some situations. Accordingly, it was necessary to evaluate consequences for a greater variety of cases than might be expected from simple considerations. Some reduction in the number of cases was obtained by conservatively assuming all ST flask events within the RCB to involve 9-day rather than 37-day cooled fuel. The cases studied, with their corresponding consequences, ex Miller [1985(b)], are given in Tables 4(b) and 5(b). The corresponding fission product inventory and release data are given in Table 6(b).

Cases 24 and 25 of Table 5(b) show that the consequences of any L/U flask accident are less than 1/200th of the child thyroid ERL. For ST flask events in which less than complete separation of the flask main components occurs (Cases 31 and 36) or the 'standard' SB-1 event (Cases 27 to 30), the bounding consequences are still less than 1/80th of the child thyroid ERL. Comparison of Cases 27 and 29 shows that stable iodines from melted, older fuel can, by taking up their portion of available organic radicals, actually reduce the radioactive iodine released to atmosphere from melting of the newest fuel.

For ST flask events in which the main flask components are fully separated, or for the infrequent 'specially scheduled' or 'emergency' SB-1 events in which greater than normal fuel fission product inventories are involved, the bounding consequences, estimated by a very conservative scaling approximation from Case 29, are still less than 1/25th of the child thyroid ERL. This remains true even if the ST flask is assumed to be incorrectly loaded with fuel elements all cooled for only 9 instead of 37 days.

It follows from these very conservatively based results that the total public risk from fuel handling and storage accidents within the RCB is very small. The probability that any consideration of evacuation would arise is entirely negligible.

8.2 Fuel Melting Occurring Outside the RCB

In Section 6.3, it was shown that fuel melting outside the RCB could result only from accidents involving toppling of the ST flask. Corran's analysis [1985(b)] showed trailer accidents and crane failures to be the only significant initiating events. In most of these, flask damage will be modest, the fuel will remain in situ, and fission product escape will occur via restricted paths. In the remainder, partial or complete separation of the main flask components, with some degree of direct fuel exposure cannot be ruled out.

Because there is no containment building to attenuate the fission product release, the post-accident disposition of the fuel relative to the flask assumes importance. Three cases with differing physico-chemical conditions for fission product release were studied; fuel fully exposed ('STF halves apart' - 1 h release) and fuel remaining in the flask with either rather direct ('STF halves ajar' - 1 h release) or restricted ('STF coolant lost' - 3 h release) fission product escape routes. These cases were evaluated for two atmospheric dispersion conditions; the first, applicable to the majority (assumed 90%) of events assumed that the fission product release is complete by 1800 hours, and correlates with normal flask movement times and/or a fairly rapid fission product release; the second assumes a late event and/or a slow fission product evolution extending the release into the 1800-2100 hours period. Evaluations were made for both 37-day cooled and 9-day cooled fuel elements to enable the consequences of flask loading errors to be estimated. Events in B23 were assumed to have the same consequences as if they had occurred in the open, since credit for attenuation of fission products by that building structure is difficult to quantify.

8.2.1 Frequency

Corran estimated the frequency of all sequences (Figure 6) in which fuel could melt outside the RCB to be less than 9×10^{-5} per year. Of this, 6×10^{-5} was due to road transport occurrences, and the remainder to events in B23. It was considered that partial or complete separation of the main flask components, with the possibility of some direct exposure of fuel, could not be entirely discounted for a fraction, conservatively estimated to be less than 20 per cent, of all ST flask fuel melting events.

Accidental selection of one 9-day cooled element as part of the intended ST flask complement of four 37-day cooled elements is considered to be the only significant, plausible flask loading error which need be considered for the present purpose. About 1500 HIFAR elements have been

transferred using the ST flask over a period of about 25 years; no known instance of incorrect selection has occurred [McCulloch 1986]. The likelihood of $< 10^{-2}$ per year assumed for incorrect ST flask loading is therefore considered to be very conservative.

8.2.2 Consequences

Details of the selected conditions applicable to each component case of the sets corresponding to fuel cooling time, flask damage and atmospheric dispersion (time of release) combinations discussed in Sections 6.3 and 8.2 are given in Table 4(c), the corresponding consequences in Table 5(c), and the fission product inventory and release data in Table 6(c). Consequences for the 'ST flask loading error' situation discussed in Section 8.2.1 have been derived with adequate accuracy for practical purposes by linear interpolation between corresponding 37-day cooled and 9-day cooled fuel cases.

For 90% of the sequences considered, the consequences are less than 1/20th of the child thyroid ERL. For almost all the remainder, the consequences are less than 2/3rds of the ERL. Only for an extremely small class of sequences in which there is a flask loading error, separation of the flask components occurs, and the release extends into the evening hours, might the ERL sometimes be exceeded, with a bounding dose of 2.4 times the ERL.

9. SUMMARY

An extended summary of findings, with basic explanatory material, is provided in the Executive Summary preceding the main body of this report.

For convenience, the frequencies and bounding consequences of the principal event categories studied are given in Table 7, and presented graphically in Figure 7.

10. CONCLUSIONS

Important aspects of HIFAR's overall safety assessment have been reviewed. Equipment failures and procedural errors within the plant, as a result of which some exposure of the public to significant abnormal radiation levels might conceivably occur, have been quantitatively studied, using a simplified probabilistic risk analysis approach. The consequences have been expressed in terms of radiation doses to an individual at 1.6 km radius compared with the ERLs for which the NH&MRC recommends that consideration of limited evacuation of the surrounding population should occur. In all cases studied, the child thyroid ERL was found to be the most restrictive condition.

The evaluations have been based on data and assumptions which are considered realistic but demonstrably conservative. Where possible, 90%

non-exceedence values (in the conservative direction) have been used for key parameters. On this basis, the studies show it to be extremely unlikely that any conceivable within-plant accident could result in consequences at 1.6 km radius exceeding the most critical (child thyroid) ERL by more than a factor of between 2 and 3. The total frequency of all sequences identified in which more than trival off-site radiation consequences could arise is conservatively estimated to be less than 6×10^{-4} per year. In over 85 per cent of these sequences, the consequences would not exceed a few per cent of the child thyroid ERL.

Almost all the remaining sequences involve damage to the ST flask by toppling during fuel transfer to the B23 storage pond. The frequency of such sequences is estimated to be less than 9×10^{-5} per year. Assuming that fuel melting can result, which is by no means certain, the upper bound of consequences for most of these sequences is still less than 1/20th of the child thyroid ERL. For almost all the residual sequences, whose estimated frequency is only about 10^{-5} per year, the consequences are less than 2/3rds of the child thyroid ERL. Only if a fuel loading error has occurred, the flask is sufficiently damaged for the main components to separate, and any resulting release takes place at an unfavourable, less likely time of day, could greater dose levels arise. The frequency of such sequences is estimated to be less than 2×10^{-8} per year, and the bounding dose less than 2.4 times the child thyroid ERL.

Using a 'bounding case' approach, only one other small class of sequences has been found for some of which the consequences might equal or exceed the most restrictive component of the ERL. This class comprises a few reactivity addition sequences and reactor LOCAs associated with ECCS failure, accompanied in all cases by very impaired performance of the RCB and other plant items whose correct operation would reduce the potential consequences. The frequency of this class of sequences is conservatively estimated to be less than 2×10^{-7} per year.

The likelihood that any evacuation of the population around HIFAR might have to be considered as a result of within-plant equipment failures, malfunctions or procedural errors is therefore estimated to be less than 2×10^{-7} per year of operation.

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Note: * denotes reference material available in draft form only, at the
time this report went to press.

GLOSSARY OF ACRONYMS

- . AAEC - Australian Atomic Energy Commission
- . CCA - Coarse Control Arms
- . D₂O - Heavy Water
- . DPR - D₂O Plant Room
- . ECCS - Emergency Core Cooling System
- . ERL - Emergency Reference Level
- . FPs - Fission Products
- . HE - Heat Exchangers
- . HIFAR - High Flux Australian Reactor
- . HOS - HIFAR Operations Section
- . HSAWP - HIFAR Safety Analysis Working Party
- . HSD - HIFAR Safety Document
- . HSD-S - HIFAR Safety Document Supplement
- . IAEA - International Atomic Energy Agency
- . ICRP - International Commission on Radiological Protection
- . LH - Lucas Heights
- . LHRL - Lucas Heights Research Laboratories
- . LOCA - Loss of Coolant Accident
- . LOFA - Loss of Flow Accident
- . L/U - Load Unload Flask
- . MCA - Maximum Credible Accident
- . NH&MRC - National Health and Medical Research Council
- . NRPB - National Radiological Protection Board
- . PCC - Primary Coolant Circuit
- . PRA - Probabilistic Risk Analysis
- . RAT - Reactor Aluminium Tank
- . RCB - Reactor Containment Building
- . RD - Reactors Department
- . SB-1 - Storage Block No. 1
- . SCS - Space Conditioner System
- . SCs - Space Conditioners
- . SSS - Safety Shutdown System
- . ST - Shear and Transport Flask
- . STF - Shear and Transport Flask
- . UKAEA - United Kingdom Atomic Energy Authority
- . USNRC - United States Nuclear Regulatory Commission
- . VAL - Vehicle Air Lock

TABLE 1

**A COMPARISON OF THE NH&MRC EMERGENCY REFERENCE LEVELS
OF DOSE WITH INTERVENTION (EVACUATION COUNTER-MEASURE)
LEVELS OF DOSE FOR THE NRPB AND ICRP
(After Petersen [1985(d)])**

	Organ Dose (mSv)					
	Whole Body			Thyroid ⁽³⁾		
	NRPB ⁽¹⁾	NH&MRC ⁽²⁾	ICRP	NRPB ⁽¹⁾	NH&MRC ⁽²⁾	ICRP
Lower Inter- vention Level ⁽⁴⁾	100		50	300		500
ERL		300			1000 ⁽³⁾	
Upper Inter- vention Level ⁽⁵⁾	500		500	1500		5000

- Notes:**
- (1) Also applies to any other single organ except that the NRPB recommend 1000 and 5000 mSv for the skin.
 - (2) NH&MRC: 'consider evacuation at these levels to avoid airborne exposure'.
 - (3) NH&MRC specifies 'child thyroid' and 3000 mSv for adult thyroid.
 - (4) Lower Intervention Level: 'Introduction of counter-measure not warranted', ICRP, 1984.
 - (5) Upper Intervention Level: 'Introduction of counter-measure almost certain', ICRP, 1984.

TABLE 2

SUMMARY OF 'BOUNDING CASE LOCA' CONDITIONS

- (1) Initiating Event: LOCA - Primary coolant circuit failure accompanied by ECCS failure
- OR
- Large reactivity insertion (by CCA failure or unlimited CCA bank withdrawal) with power excursion resulting in both fuel melting and PCC breach.
- (2) Fuel Damage: Full core fuel inventory melted. Large fraction of inner fuel tubes melt early. D₂O remaining in RAT boiled off by decay heat from rest of fuel tubes, which then melt. Fission products available at each melt stage assumed transported without delay to DPR.
- (3) Fission Product Releases: Based on realistic but conservative models and data (90% conservative direction non-exceedence estimates wherever possible) for release fractions, chemical form partition fractions, deposition constants, etc.
- (4) Plant Conditions: Space Conditioner Subsystem unavailable. Secondary coolant circulation unavailable for 1 hour, and then only by standby pump. RCB leak-rate 3 vol.% per day.
- (5) Atmospheric Dispersion: Wind-speed and change-of-direction relations selected so that the estimated probability of worse dispersion characteristics over the duration of the release than those adopted is less than 10%.

TABLE 3
DISCRETE FAULT SEQUENCES ASSOCIATED WITH BREACHES OF THE FCC

TABLE 3(a): INITIATED BY CIRCUIT FAILURE (LOCAS)

Node No.	Description	Node probability given reactivity addition	Node frequency (y^{-1})
0	--		
1	--		
2	--		
3	--		
4	--		
5	FCC breach following reactivity accident	1.0000	4.6E-7
6	--		
7	Melt, secondary cooling after first hour	1.0E-1	4.6E-7
8	Melt, no secondary cooling	1.0E-4	4.6E-11
9	Melt, secondary cooling, SCs	9.9E-1	4.6E-7
10	Melt, secondary cooling, no SCs	6.9E-3	3.2E-9
11	Melt, no secondary cooling, SCs	9.7E-5	4.4E-11
12	Melt, no secondary cooling, no SCs	3.3E-6	1.5E-12
13	'Bounding Case' (Node 7 with RCB leak-rate \leq 3% per day)	9.0E-1	4.1E-7
14	RCB leak-rate greater than 'Bounding Case' (3% \leq leak-rate \leq 10%)	9.0E-2	4.1E-8
15	All other sequences potentially more severe than 'Bounding Case'	1.0E-2	4.6E-9

TABLE 3(b): INITIATED BY MAJOR REACTIVITY ADDITIONS

Node No.	Description	Node probability given LOCA	Node frequency (y^{-1})
0	Circuit failure (LOCA)	1.000	1.0E-3
1	Full ECCS function	9.0E-1	9.0E-4
2	Partial ECCS function	9.6E-2	9.6E-5
3	No ECCS function	5.6E-3	5.6E-6
4	No early melt	9.9E-1	9.9E-4
5	Early melt	6.6E-3	6.6E-6
6	No melt	9.9E-1	9.9E-4
7	Melt, secondary cooling after first hour	6.6E-3	6.6E-6
8	Melt, no secondary cooling	1.0E-4	1.0E-7
9	Melt, secondary cooling, SCs	6.5E-3	6.5E-6
10	Melt, secondary cooling, no SCs	4.5E-5	4.5E-8
11	Melt, no secondary cooling, SCs	9.7E-5	9.7E-8
12	Melt, no secondary cooling, no SCs	3.3E-6	3.3E-9
13	'Bounding Case' (Node 7 with RCB leak-rate \leq 3% per day)	5.9E-3	5.9E-6
14	RCB leak-rate greater than 'Bounding Case' (3% \leq leak-rate \leq 10%)	5.9E-4	5.9E-7
15	All other sequences potentially more severe than 'Bounding Case'	1.7E-4	1.7E-7

NOTE: Refer to Figure 2 for development of these sequences

TABLE 4
MAIN PARAMETERS OF CASES SELECTED FOR EVALUATION OF DOSES AT 1.6 km RADIUS

4(a) In-core Cases

Case No.	Description	Comments
1	LOCA; full core inventory; SCS and HE operative; 90% bounding source term, RCB leak-rate and dispersion data; release commences 2100 hours.	
2	As for Case 1, but \overline{HE} .	
3	As for Case 1, but SCS inoperative.	
4*	LOCA; full core inventory; inoperative SCS, \overline{HE} ; 90% bounding source term, RCB leak-rate and atmospheric dispersion data; release commences 2100 hours.	Reference 'Bounding Case'. (Fig. 7, A1)
5	As for Case 4, but median atmospheric dispersion data.	
6	As for Case 1, but with SCS and HE operative and median RCB leak-rate data.	
7	As for Case 1, but release commences 0900 hours.	
8	As for Case 5, but release commencing 0900 hours.	
9	As for Case 1, but 90% bounding atmospheric dispersion conditions occurring 60 h into release commencing 2100 hours.	
10	As for Case 4, but 90% bounding atmospheric dispersion conditions occurring 60 h into release commencing 2100 hours.	
11	As for Case 1, but with low fuel element nozzle leak-rate and 20 h D ₂ O boil off.	
12	As for Case 1, but most probable source term data.	
13	As for Case 4, but with release commencing 0900 hours.	
14	As for Case 4, but 90% bounding atmospheric dispersion conditions occurring 60 h into release commencing at 2100 hours.	
15	As for Case 16, but SCS.	
16*	LOCA; full core inventory; inoperative SCS; \overline{HE} ; most probable source term data; median RCB leak-rate and atmospheric dispersion data; release commences 2100 hours.	Case 4, but using most probable/median instead of '90% bounding' data values.
17	As for Case No. 1; but SCS inoperative, \overline{HE} , but with median RCB leak-rate data.	
18	As for Case No. 1, but with 99% bounding RCB leak-rate data.	
19*	As for Case 1, but SCS inoperative, \overline{HE} and 99% bounding RCB leak-rate data (10% volume per day).	Fig. 7, A2.
20*	LOCA; full core inventory; inoperative SCS; \overline{HE} ; 90% bounding source term and atmospheric dispersion data; RCB open; release commences 2100 hours.	Extremely unlikely coincident events. (Fig. 7, A3)
21	As for Case 20, but median atmospheric dispersion data.	
22	As for Case 20, but most probable source term and median data.	
23	As for Case 20, but most probable source term data.	

Notes: * Denotes cases considered most important to the study. Other cases are intended mostly for use as indicators of the sensitivity of principal results to various input data items.

SCS denotes Space Conditioner Subsystem.

\overline{HE} denotes no secondary coolant circulation during first hour after reactor shutdown.

HE denotes no loss of secondary coolant circulation.

TABLE 4 (continued)

4(b) Out-of-core Cases - Fuel Melts Within the RCB

Case No.	Description	Comments
24*	Single fuel element cooled 2 days; 90% bounding source term, RCB leak-rate and atmospheric dispersion data; release commences 2100 hours.	Reference case, L/U flask events. (Fig. 7, B1)
25	As for Case 24, but release commences 0900 hours.	
26	As for Case 24, but RCB with 10 mm diameter orifice in wall.	Simulates L/U flask reference case occurring while RCB being modified.
27	Storage Block 1 (SB-1); 20 fuel elements, 8 cooled 2 days and 12 cooled 30 days; 90% bounding source term, RCB leak-rate and atmospheric dispersion data; release commences 2100 hours.	Reference case, SB-1 (Fig. 7, B2).
28	As for Case 27, but release commences 0900 hours.	
29*	As for Case 27, but only 8 fuel elements cooled 2 days.	Reference case (scaled to 25 elements) for SB-1 'special or emergency' situations. (Fig. 7, B3).
30	As for Case 27, but only 12 fuel elements cooled 30 days.	
31*	Shear and Transport flask (STF); 4 fuel elements cooled 9 days; 90% bounding source term, RCB leak-rate and atmospheric dispersion data; release commencing 2100 hours.	Reference case, STF coolant loss inside the RCB. (Included in Fig. 7, B2.)
36	As for Case 31, but STF halves ajar.	
41	As for Case 31, but STF halves apart and 4 elements exposed.	Reference case, STF severely damaged inside the RCB. (Included in Fig. 7, B3.)

Note: * Denotes cases considered most important to the study. Other cases are intended mostly for use as indicators of the sensitivity of principal results to various input data items.

TABLE 4 (continued)

4(c) Out-of-core Cases - Fuel Melts Outside the RCB

Case No.	Description	Comments
32	STF coolant lost; 4 elements cooled 9 days; 90% bounding source term and dispersion data; 3 hr release commencing 1700 hours.	
33	As for Case 32, but 3 h release between 1800 and 2100 hours.	
34	As for Case 33, but fuel elements cooled 37 days.	
35	As for Case 32, but fuel elements cooled 37 days.	Corresponds to normal STF loading and movement time.
37	STF halves ajar; 4 elements cooled 9 days; 90% bounding source term and dispersion data; 1 h release between 1800 and 2100 hours.	
38	As for Case 37, but 1 h release between 1500 and 1700 hours.	
39	As for Case 38, but fuel elements cooled 37 days.	Corresponds to normal STF loading and movement time, with quite severe flask damage.
40	As for Case 37, but fuel elements cooled 37 days.	
42*	STF halves apart and elements exposed; 4 elements cooled 9 days; 90% bounding source term and dispersion data; 1 h release between 1800 and 2100 hours.	Scales with Case 45 for 'worst STF consequences' (Fig. 7, C2).
43	As for Case 42, but 1 h release between 1500 and 1700 hours.	
44*	As for Case 43, but fuel elements cooled 37 days.	Reference case, correct fuel, any STF damage, pre-1800 hours release (Fig. 7, C1).
45*	As for Case 42, but fuel elements cooled 37 days.	Reference case, sets bounding consequences for Fig. 7, C3. Also scales with Case 42 for 'worst STF consequences' (Fig. 7, C2).

Note: * Denotes cases considered most important to the study. Other cases are intended mostly for use as indicators of the sensitivity of principal results to various input data items.

TABLE 5
CALCULATED DOSES AT 1.6 km RADIUS

5(a) In-core Cases

Case	Dose (mSv)					
	Child Thyroid	Whole Body	Uterus	Skin	Marrow	Lung
1	13	0.45	0.30	1.6	0.53	0.53
2	15	0.46	0.31	1.8	0.56	0.54
3	29	0.48	0.31	1.9	0.59	0.59
4*	33	0.50	0.32	2.0	0.61	0.61
5	16	0.17	0.10	0.79	0.23	0.21
6	8.7	0.24	0.16	0.97	0.30	0.28
7	12	0.16	$8.6 \cdot 10^{-2}$	0.78	0.22	0.17
8	34	0.21	0.11	1.2	0.32	0.27
9	6.5	0.13	$7.8 \cdot 10^{-2}$	0.58	0.17	0.15
10	29	0.19	0.10	1.1	0.28	0.24
11	13	0.45	0.30	1.6	0.53	0.53
12	4.3	0.44	0.29	1.6	0.52	0.49
13	11	0.48	0.31	0.20	0.60	0.54
14	4.9	0.17	$9.9 \cdot 10^{-2}$	0.78	0.22	0.18
15	1.0	$8.0 \cdot 10^{-2}$	$4.8 \cdot 10^{-2}$	0.37	0.11	$8.5 \cdot 10^{-2}$
16*	3.1	0.10	$4.5 \cdot 10^{-2}$	0.53	0.14	0.10
17	19	0.27	0.17	1.1	0.34	0.32
18	17	1.1	0.76	3.4	1.2	1.3
19*	46	1.4	0.96	4.8	1.6	1.7
20*	1900	68	49	190	74	77
21	1000	27	20	76	29	31
22	720	66	48	190	72	74
23	360	26	190	75	29	29
NH&MRC ERLs	1000	300	150 ⁽¹⁾	NS ⁽²⁾	NS ⁽²⁾	NS ⁽²⁾

Notes: * Denotes cases considered most important to the study. Other cases are intended mostly for use as indicators of the sensitivity of principal results to various input data items.

(1) The dose to the uterus is estimated as a surrogate for the foetal dose.

(2) NS = not specified; however, the geometric means of the upper and lower NRPB levels for the skin, marrow and lung are 2240, 670 and 670 mSv respectively.

TABLE 5 (continued)

Table 5(b) Out-of-core Cases - Fuel Melts Within the RCB

Case	Dose (mSv)					
	Child Thyroid	Whole Body	Uterus	Skin	Marrow	Lung
24*	4.7	$2.6 \cdot 10^{-3}$	$1.3 \cdot 10^{-3}$	$1.3 \cdot 10^{-2}$	$3.9 \cdot 10^{-3}$	$2.7 \cdot 10^{-3}$
25	4.0	$1.7 \cdot 10^{-3}$	$7.5 \cdot 10^{-4}$	$1.1 \cdot 10^{-2}$	$2.8 \cdot 10^{-3}$	$1.5 \cdot 10^{-3}$
26	45	$2.0 \cdot 10^{-2}$	$8.7 \cdot 10^{-3}$	0.12	$3.3 \cdot 10^{-2}$	$1.8 \cdot 10^{-2}$
27	11	$1.5 \cdot 10^{-2}$	$6.1 \cdot 10^{-3}$	$9.4 \cdot 10^{-2}$	$2.4 \cdot 10^{-2}$	$3.7 \cdot 10^{-2}$
28	7.0	$1.0 \cdot 10^{-2}$	$3.4 \cdot 10^{-3}$	$7.9 \cdot 10^{-2}$	$1.8 \cdot 10^{-2}$	$9.2 \cdot 10^{-3}$
29*	12	$1.4 \cdot 10^{-2}$	$6.1 \cdot 10^{-3}$	$9.0 \cdot 10^{-2}$	$2.3 \cdot 10^{-2}$	$1.4 \cdot 10^{-2}$
30	1.3	$6.4 \cdot 10^{-4}$	$2.5 \cdot 10^{-4}$	$5.3 \cdot 10^{-3}$	$1.3 \cdot 10^{-3}$	$1.1 \cdot 10^{-3}$
31*	4.4	$2.7 \cdot 10^{-3}$	$1.0 \cdot 10^{-3}$	$1.9 \cdot 10^{-2}$	$4.6 \cdot 10^{-3}$	$2.5 \cdot 10^{-3}$
36	8.0	$3.3 \cdot 10^{-3}$	$1.4 \cdot 10^{-3}$	$2.0 \cdot 10^{-2}$	$6.1 \cdot 10^{-3}$	$4.8 \cdot 10^{-3}$
41	21	$5.3 \cdot 10^{-3}$	$2.7 \cdot 10^{-3}$	$2.5 \cdot 10^{-2}$	$9.2 \cdot 10^{-3}$	$8.2 \cdot 10^{-3}$
NH&MRC ERLs	1000	300	150 ⁽¹⁾ mSv	NS ⁽²⁾	NS ⁽²⁾	NS ⁽²⁾

Notes: * Denotes cases considered most important to the study. Other cases are intended mostly for use as indicators of the sensitivity of principal results to various input data items.

(1) The dose to the uterus is estimated as a surrogate for the foetal dose.

(2) NS = not specified; however, the geometric means of the upper and lower NRPB levels for the skin, marrow and lung are 2240, 670 and 670 mSv respectively.

TABLE 5 (continued)

5(c) Out-of-core Cases - Fuel Melts Outside the RCB

Case	Dose (mSv)					
	Child Thyroid	Whole Body	Uterus	Skin	Marrow	Lung
32	140	0.14	$6.4 \cdot 10^{-2}$	0.80	0.22	0.15
33	1300	1.4	6.3	7.7	2.1	0.15
34	120	$4.1 \cdot 10^{-2}$	$1.7 \cdot 10^{-2}$	0.27	$7.3 \cdot 10^{-2}$	$5.2 \cdot 10^{-2}$
35	13	$4.3 \cdot 10^{-3}$	$1.8 \cdot 10^{-3}$	$2.9 \cdot 10^{-2}$	$7.8 \cdot 10^{-3}$	$5.5 \cdot 10^{-3}$
37	2800	2.2	1.1	10	3.2	2.7
38	180	0.14	$7.3 \cdot 10^{-2}$	0.64	0.21	0.17
39	16	$4.0 \cdot 10^{-3}$	$1.9 \cdot 10^{-3}$	$2.2 \cdot 10^{-2}$	$9.9 \cdot 10^{-3}$	$1.2 \cdot 10^{-2}$
40	250	$6.2 \cdot 10^{-2}$	$3.0 \cdot 10^{-2}$	0.35	0.15	0.19
42*	7400	4.3	2.6	14	5.8	5.9
43	480	0.28	0.17	0.93	0.37	0.38
44*	43	$7.5 \cdot 10^{-3}$	$4.2 \cdot 10^{-3}$	$3.0 \cdot 10^{-2}$	$1.7 \cdot 10^{-2}$	$2.2 \cdot 10^{-2}$
45*	670	0.12	$6.6 \cdot 10^{-2}$	0.47	0.26	0.34
NH&MRC ERLs	1000	300	150 ⁽¹⁾ mSv	NS ⁽²⁾	NS ⁽²⁾	NS ⁽²⁾

Notes: * Denotes cases considered most important to the study. Other cases are intended mostly for use as indicators of the sensitivity of principal results to various input data items.

(1) The dose to the uterus is estimated as a surrogate for the foetal dose.

(2) NS = not specified; however, the geometric means of the upper and lower NRPB levels for the skin, marrow and lung are 2240, 670 and 670 mSv respectively.

TABLE 6

RADIONUCLIDES INITIALLY IN FUEL AND RELEASED TO ATMOSPHERE

Table 6(a) In-core Cases

Case No.	Activity Initially in Fuel (Bq)					Cumulative Activity Released (Bq)				
	^{131}I	Total Halogens	^{133}Xe	Total Noble Gases	Others	^{131}I	Total Halogens	^{133}Xe	Total Noble Gases	Others
1	1.0×10^{16}	9.8×10^{16}	2.2×10^{16}	6.7×10^{16}	7.3×10^{17}	2.4×10^{12}	6.5×10^{12}	1.4×10^{15}	1.6×10^{15}	3.8×10^{13}
2	"	"	"	"	"	2.6×10^{12}	7.1×10^{12}	1.7×10^{15}	1.9×10^{15}	3.8×10^{13}
3	"	"	"	"	"	6.5×10^{12}	1.4×10^{13}	2.2×10^{15}	2.4×10^{15}	3.9×10^{13}
4*	"	"	"	"	"	8.5×10^{12}	1.7×10^{13}	2.8×10^{15}	3.0×10^{15}	3.9×10^{13}
5	"	"	"	"	"	8.5×10^{12}	1.7×10^{13}	2.8×10^{15}	3.0×10^{15}	3.9×10^{13}
6	"	"	"	"	"	1.9×10^{12}	4.4×10^{12}	1.3×10^{15}	1.4×10^{15}	1.9×10^{13}
7	"	"	"	"	"	2.4×10^{12}	6.5×10^{12}	1.4×10^{15}	1.6×10^{15}	3.8×10^{13}
8	"	"	"	"	"	8.5×10^{12}	1.7×10^{13}	2.8×10^{15}	3.0×10^{15}	3.9×10^{13}
9	"	"	"	"	"	2.4×10^{12}	6.5×10^{12}	1.4×10^{15}	1.6×10^{15}	3.8×10^{13}
10	"	"	"	"	"	8.5×10^{12}	1.7×10^{13}	2.8×10^{15}	3.0×10^{15}	3.9×10^{13}
11	"	"	"	"	"	2.4×10^{12}	6.5×10^{12}	1.4×10^{15}	1.6×10^{15}	3.8×10^{13}
12	"	"	"	"	"	6.9×10^{11}	2.0×10^{12}	1.4×10^{15}	1.6×10^{15}	3.7×10^{13}
13	"	"	"	"	"	2.7×10^{12}	5.5×10^{12}	2.8×10^{15}	3.0×10^{15}	3.9×10^{13}
14	"	"	"	"	"	2.7×10^{12}	5.5×10^{12}	2.8×10^{15}	3.0×10^{15}	3.9×10^{13}
15	"	"	"	"	"	5.4×10^{11}	1.3×10^{12}	1.3×10^{15}	1.4×10^{15}	1.9×10^{13}
16*	"	"	"	"	"	1.7×10^{12}	3.2×10^{12}	2.2×10^{15}	2.2×10^{15}	1.9×10^{13}
17	"	"	"	"	"	5.4×10^{12}	9.8×10^{12}	2.2×10^{15}	2.2×10^{15}	2.0×10^{13}
18	"	"	"	"	"	2.2×10^{12}	7.5×10^{12}	1.3×10^{15}	1.6×10^{15}	9.2×10^{13}
19*	"	"	"	"	"	7.9×10^{12}	2.3×10^{13}	2.6×10^{15}	3.1×10^{15}	1.3×10^{14}
20*	"	"	"	"	"	4.6×10^{14}	1.2×10^{15}	1.6×10^{16}	3.5×10^{16}	4.2×10^{15}
21	"	"	"	"	"	4.6×10^{14}	1.2×10^{15}	1.6×10^{16}	3.5×10^{16}	4.2×10^{15}
22	"	"	"	"	"	1.5×10^{14}	4.1×10^{14}	1.6×10^{16}	3.5×10^{16}	4.2×10^{15}
23	"	"	"	"	"	1.5×10^{14}	4.1×10^{14}	1.6×10^{16}	3.5×10^{16}	4.2×10^{15}

Note: * Denotes cases considered most important to the study. Other cases are intended mostly for use as indicators of the sensitivity of principal results to various input data items.

TABLE 6 (continued)

6(b) Out-of-core Cases - Fuel Melts Within the RCB

Case No.	Activity Initially in Fuel (Bq)					Cumulative Activity Released (Bq)				
	^{131}I	Total Halogens	^{133}Xe	Total Noble Gases	Others	^{131}I	Total Halogens	^{133}Xe	Total Noble Gases	Others
24*	3.5×10^{14}	9.6×10^{14}	7.6×10^{14}	8.4×10^{14}	8.8×10^{15}	1.4×10^{12}	1.7×10^{12}	3.3×10^{13}	3.4×10^{13}	1.5×10^9
25	"	"	"	"	"	1.4×10^{12}	1.7×10^{12}	3.3×10^{13}	3.4×10^{13}	1.5×10^9
26	"	"	"	"	"	1.4×10^{13}	1.7×10^{13}	3.5×10^{14}	3.6×10^{14}	1.9×10^{10}
27	3.2×10^{15}	8.0×10^{15}	6.5×10^{15}	6.9×10^{15}	1.1×10^{17}	1.6×10^{12}	2.1×10^{12}	2.8×10^{14}	2.9×10^{14}	8.4×10^9
28	"	"	"	"	"	1.6×10^{12}	2.1×10^{12}	2.8×10^{14}	2.9×10^{14}	8.4×10^9
29*	2.8×10^{15}	7.7×10^{15}	6.1×10^{15}	6.7×10^{15}	7.0×10^{16}	2.1×10^{12}	2.8×10^{12}	2.6×10^{14}	2.6×10^{14}	3.6×10^9
30	3.8×10^{14}	3.9×10^{14}	3.1×10^{14}	3.3×10^{14}	4.0×10^{16}	3.0×10^{11}	3.0×10^{11}	1.3×10^{13}	1.7×10^{13}	4.6×10^9
31*	7.7×10^{14}	1.1×10^{15}	1.4×10^{15}	1.4×10^{15}	2.2×10^{16}	1.6×10^{12}	1.6×10^{12}	5.9×10^{13}	6.0×10^{13}	8.4×10^9
36	7.7×10^{14}	1.1×10^{15}	1.4×10^{15}	1.4×10^{15}	2.2×10^{16}	2.9×10^{12}	2.9×10^{12}	5.9×10^{13}	6.0×10^{13}	5.5×10^{10}
41	7.7×10^{14}	1.1×10^{15}	1.4×10^{15}	1.4×10^{15}	2.2×10^{16}	7.6×10^{12}	7.7×10^{12}	5.9×10^{13}	6.0×10^{13}	9.1×10^{10}

Note: * Denotes cases considered most important to the study. Other cases are intended mostly for use as indicators of the sensitivity of principal results to various input data items.

TABLE 6 (continued)

6(c) Out-of-core Cases - Fuel Melts Outside the RCB

Case No.	Activity Initially in Fuel (Bq)					Cumulative Activity Released (Bq)				
	^{131}I	Total Halogens	^{133}Xe	Total Noble Gases	Others	^{131}I	Total Halogens	^{133}Xe	Total Noble Gases	Others
32	"	"	"	"	"	2.7×10^{13}	3.9×10^{13}	1.4×10^{15}	1.4×10^{15}	1.6×10^{10}
33	"	"	"	"	"	2.7×10^{13}	3.9×10^{13}	1.4×10^{15}	1.4×10^{15}	1.5×10^{10}
34	6.9×10^{13}	7.0×10^{13}	4.3×10^{13}	5.0×10^{13}	1.2×10^{16}	2.4×10^{12}	2.4×10^{12}	4.4×10^{13}	4.9×10^{13}	1.3×10^{10}
35	"	"	"	"	"	2.4×10^{12}	2.4×10^{12}	4.4×10^{13}	4.9×10^{13}	1.2×10^{10}
37	"	"	"	"	"	4.9×10^{13}	7.1×10^{13}	1.4×10^{15}	1.4×10^{15}	1.1×10^{11}
38	"	"	"	"	"	4.9×10^{13}	7.2×10^{13}	1.4×10^{15}	1.4×10^{15}	9.6×10^{10}
39	6.9×10^{13}	7.0×10^{13}	4.3×10^{13}	5.0×10^{13}	1.2×10^{16}	4.4×10^{12}	4.4×10^{12}	4.4×10^{13}	4.9×10^{13}	8.8×10^{10}
40	"	"	"	"	"	4.4×10^{12}	4.4×10^{12}	4.4×10^{13}	4.9×10^{13}	7.8×10^{10}
42*	"	"	"	"	"	1.3×10^{14}	1.9×10^{14}	1.4×10^{15}	1.4×10^{15}	1.8×10^{11}
43	"	"	"	"	"	1.3×10^{14}	1.9×10^{14}	1.4×10^{15}	1.4×10^{15}	1.6×10^{11}
44*	6.9×10^{13}	7.0×10^{13}	4.3×10^{13}	5.0×10^{13}	1.2×10^{16}	1.2×10^{13}	1.2×10^{13}	4.4×10^{13}	4.9×10^{13}	1.5×10^{11}
45*	"	"	"	"	"	1.2×10^{13}	1.2×10^{13}	4.4×10^{13}	4.9×10^{13}	1.3×10^{11}

Note: * Denotes cases considered most important to the study. Other cases are intended mostly for use as indicators of the sensitivity of principal results to various input data items.

TABLE 7
FREQUENCY AND CONSEQUENCES OF ACCIDENTS

Class of Occurrence	Estimated Maximum Frequency of Class (y^{-1})	Sub-class Maximum Frequency (y^{-1})	Sub-class Maximum Consequences (Child Thyroid ERLs)	Notes	Fig. 7 Data Point
1. In-core Fuel Melting (LOCAs with ECCS failure plus reactivity additions causing PCC breach)	7×10^{-6}	6×10^{-6}	1/30	'Bounding Case'	A1
		7×10^{-7}	1/20	Bounding Case plus high RCB leak-rate	A2
		2×10^{-7}	2 - 3	All others in class	A3
2. Fuel Storage and Handling	5×10^{-4}	4×10^{-4}	1/200	L/U flask/Single fuel element	B1
			1/80	Coolant failure SB-1 (normal situations) and ST flask events (no or partial separation)	B2
		6×10^{-6}	1/25	Coolant failure SB-1 (special or emergency situations) and ST flask events (complete separation)	B3
			1/20	ST flask - fuel correct, any degree of flask damage, pre-1800 hours release	C1
		2×10^{-8}	2.4	ST flask - fuel incorrect, partial or complete separation of main flask components and 1800 - 2100 hours release	C2
			2/3	Balance of combinations of fuel contents, flask damage and time of release.	C3
2.2 Fuel melts outside the RCB	1×10^{-4}				

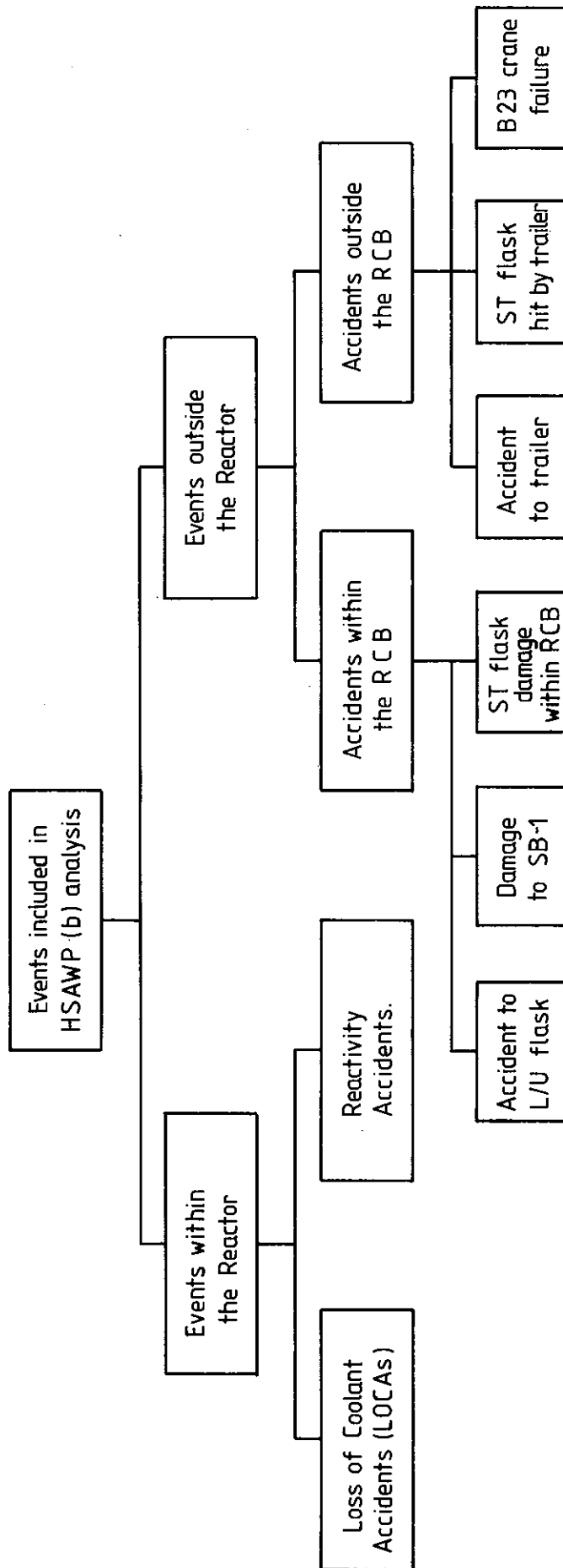


Figure 1 Event sequence classes in the study structure

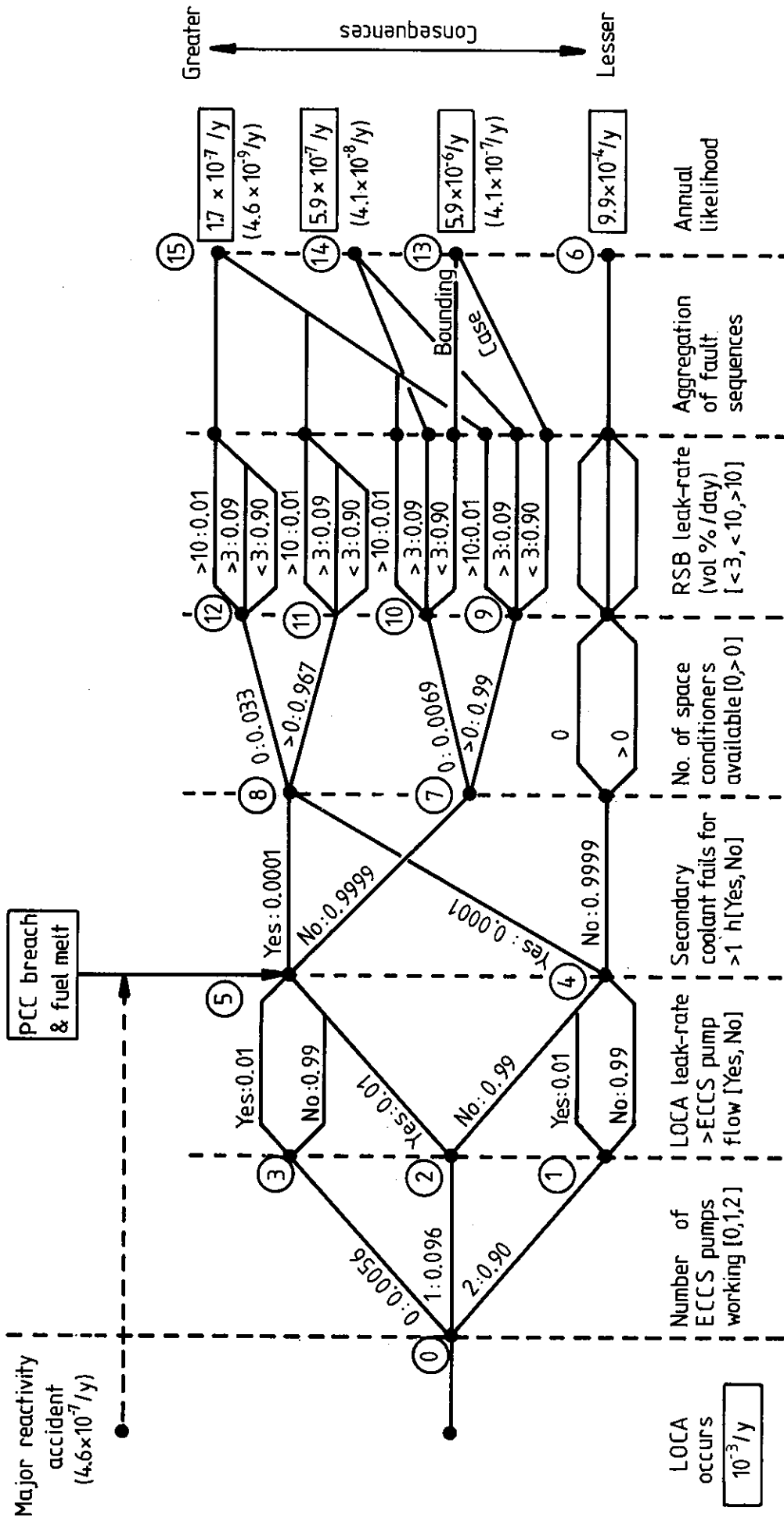


Figure 2 Event tree for LOCAs and for PCC breaches due to reactivity accidents

Notes:

- (1) The estimated number of events per year is shown at each input and output. Estimates in boxes are contributions due to LOCAs, those in brackets due to reactivity accidents.
- (2) Information on branches is condition and probability of condition given the circumstances of the previous node.
- (3) For further information see Table 2.

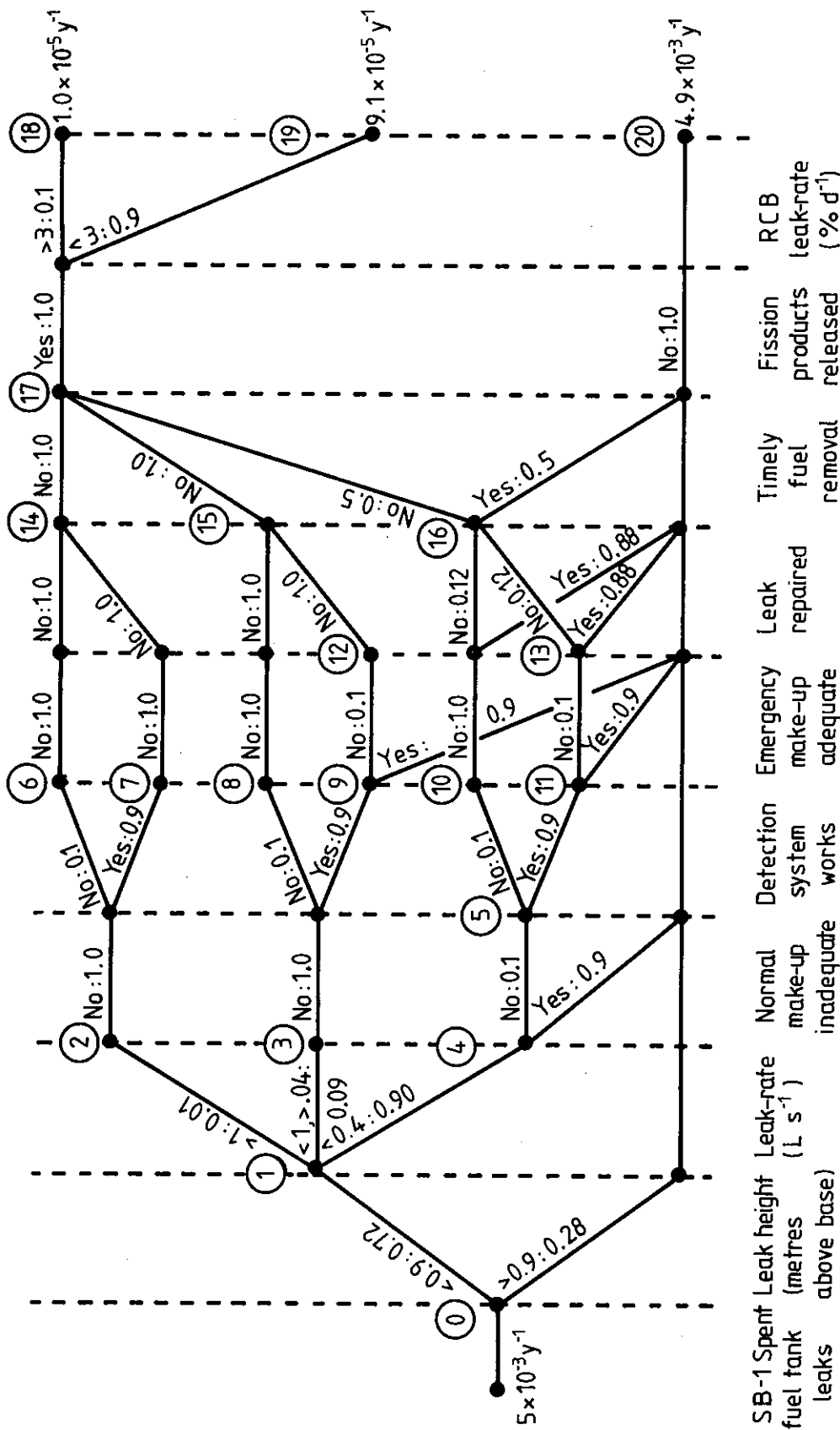


Figure 3 Event tree for SB-1 spent fuel tank leaks

Notes:

- (1) The estimate number of events per year is shown at the input and outputs.
- (2) Information on branches is condition and probability of condition given the circumstances of the previous node.

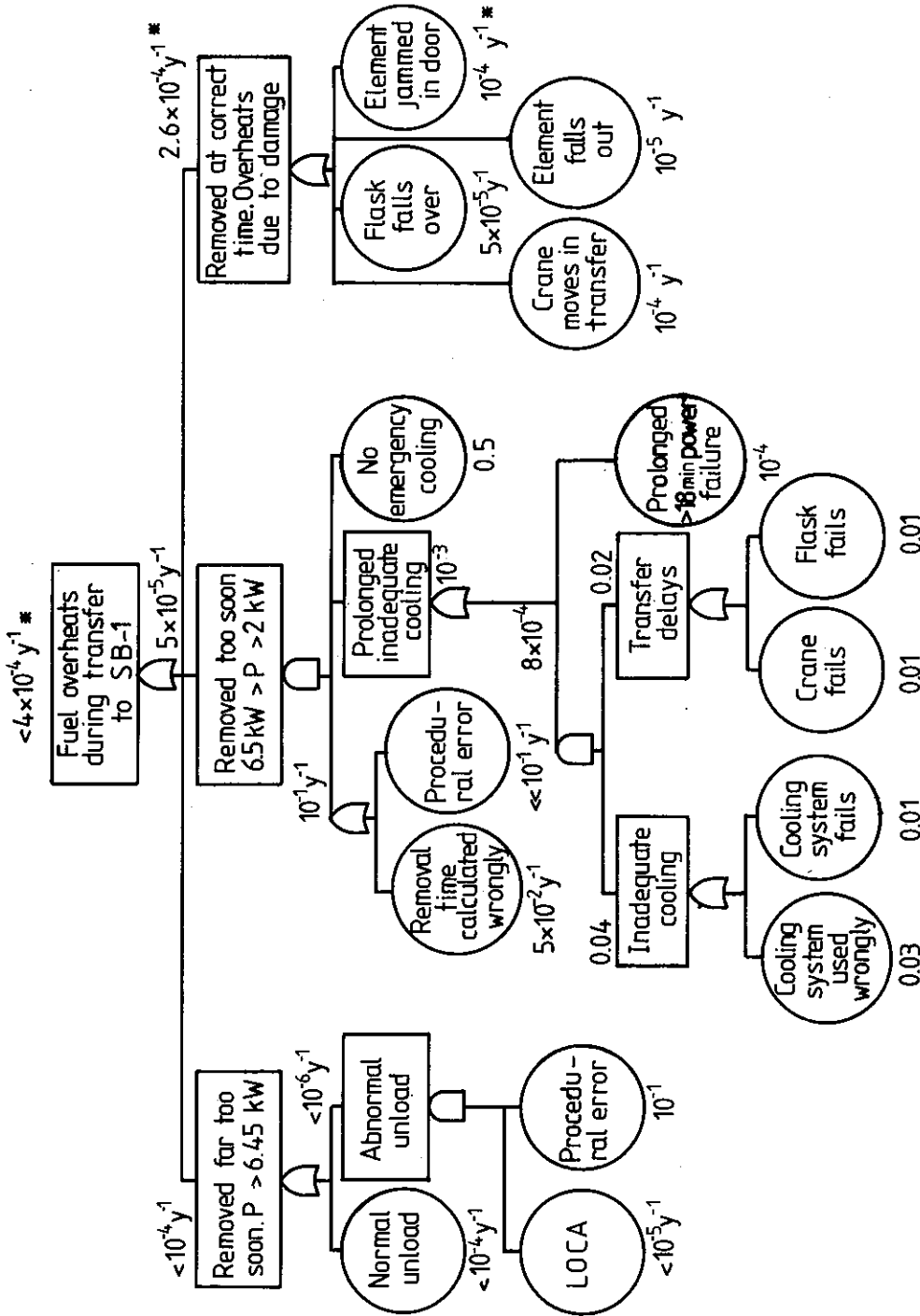


Figure 4 Fault tree for fuel overheating during transfer to SB-1

Notes:

- (1) Estimates with units y^{-1} are numbers of events per year, others are probabilities.
- (2) Estimates marked thus (*) are conditional on installation of flask door interlock.

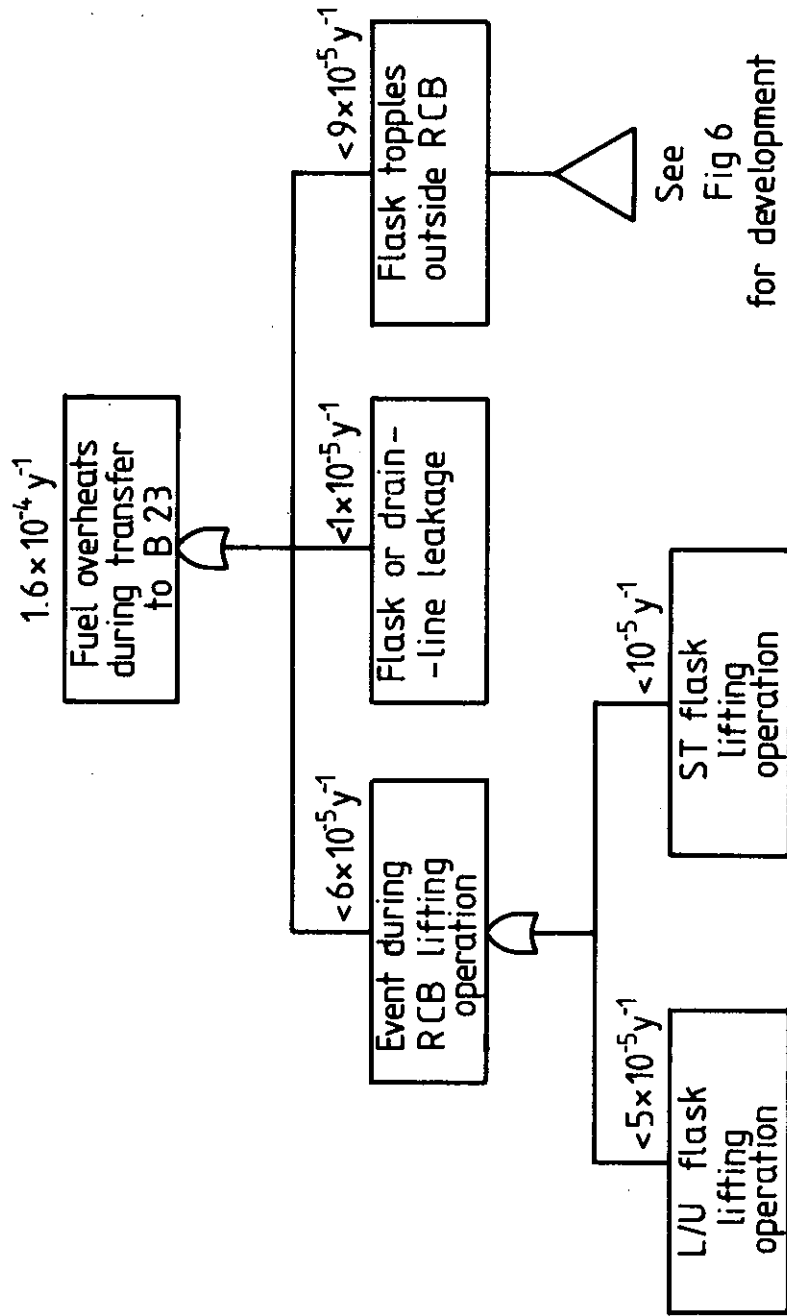


Figure 5 Fault tree for fuel overheating during transfer to B23

Note: Estimates shown are annual numbers of events.

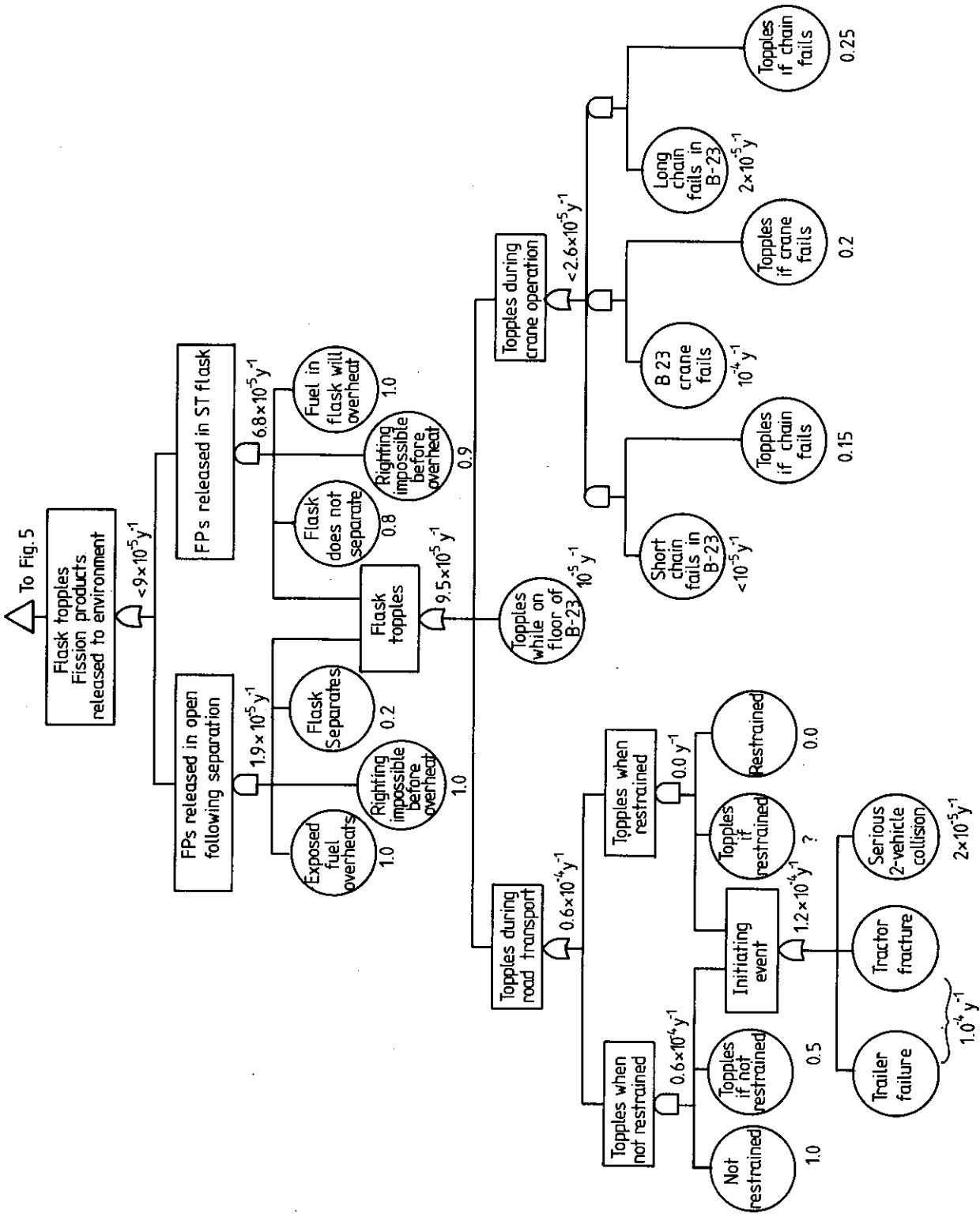


Figure 6 Fault tree for flask toppling outside RCB

Note: Estimates with units y^{-1} are annual number of events, others are probabilities.

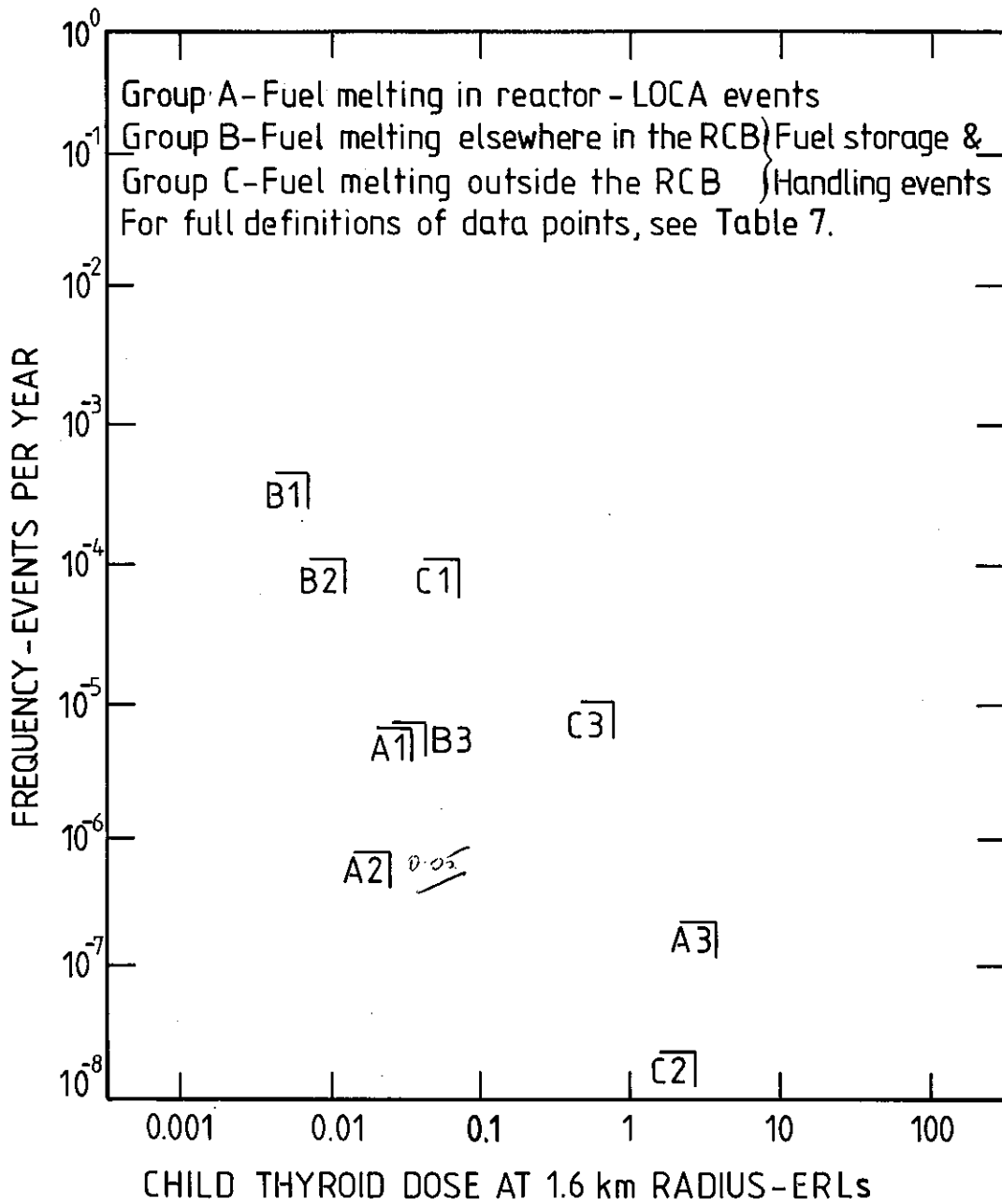


Figure 7 Frequency and consequences of accidents

