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Replacement Research Reactor Project

PROBABILISTIC SAFETY ASSESSMENT (PSA) SUMMARY FOR PUBLIC RELEASE

Prepared By

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PREFACE

This document has been specifically prepared for public release and accordingly, does not contain any in-confidence material. It is based on the complete version of classified document that has been formally submitted to ARPANSA. A specific public release version has been prepared because a review of the base document indicated that simply deleting the in-confidence material would make them unreadable or unintelligible. In addition, the language and terminology used may be modified as appropriate to make it more readily understandable to a non-specialist reader. However, it should be noted that there are no technical differences between this public release version and the complete document upon which it is based.

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1 INTRODUCTION

This report is a summary of the Probabilistic Safety Assessment (PSA) study of the Open Pool Australian Light-water research reactor (OPAL) built at Lucas Heights, NSW.

The methodology adopted for this PSA basically follows the NUREG/CR-2300 [1] recommendations, except in the treatment of common cause failures which are discussed in section 1.6. The approach for this treatment is considered more appropriate for a new reactor which does not have operational experience.

1.1 BACKGROUND

1.1.1 Background to project

The OPAL Reactor is an open pool type reactor, that is, a reactor where the core sits in a deep pool of water that provides cooling of the core, and protection against the effects of radiation. The metallic pool is inserted in a high integrity reinforced concrete block.

The OPAL Reactor provides facilities for irradiating targets for the production of radioisotopes, for irradiating silicon and for experiments; as well as providing high quality neutron beams for specialised research.

Details are given in the Safety Analysis Report (SAR).

1.1.2 How does the PSA fit with the Safety Analysis?

The safety analysis in Chapter 16 of the SAR considers the comprehensive range of postulated initiating events (PIEs). It then examines each of these PIEs and determines which can be eliminated from further consideration because their likelihood is very low, or because of plant design features which mean the event cannot credibly occur, or because the event consequences are bounded by other initiating events. The events that remain are considered Design Basis events and are analysed using thermal-hydraulic and neutronics computer codes with conservative assumptions, including an assumption that at least one item in each safety system fails. The analyses show that the many safety features included in the design are effective in preventing accidents, protecting against accidents and mitigating against their consequences (were an accident to occur), even if one part of a safety system were to fail. The reactor is demonstrated to behave within the safety limits in all design basis PIEs and the safety features and design characteristics are shown to cope with these scenarios.

In contrast to the Safety Analysis, the PSA attempts to determine all the possible combinations of how the plant could respond to an initiating event and obtain conservative estimates of the frequency and bounding estimates of the consequences. The frequency and consequence constitute the risk, and can be compared against the safety objectives set out in ARPANSA's regulatory assessment principles [2].

In addition to examining beyond design sequences, the PSA also considers some additional beyond design initiators, such as undesired closure of one of the primary cooling system isolation valves, when the reactor is operating at full power.

1.1.3 Meaning of risk, frequency and consequence

"Risk is the chance of something happening that will impact upon objectives. It is measured in terms of consequence and likelihood" [3].

Likelihood can be expressed in terms of the probability of occurrence in a given time. If that unit time is a standard unit (eg year) then this is equivalent to frequency providing the probability is small. For rare events we often talk about their frequencies in very small fractions per year.

Consequence is measured in some unit that represents the kind of outcomes. For financial risk, the consequence might be measured in dollars; for road traffic risks, the consequences might be measured in number of fatalities. In this PSA, consequence is measured by effective radiation dose to members of the public.

Different kinds of radiation have different effects on living tissue. Because of this, it is convenient in nuclear plant PSA to use effective radiation dose as the measure of consequence, and this is typically estimated for the "worst placed" individual member of the public, making conservative assumptions about the weather and that individual's behaviour. For example, it was assumed the individual stays outdoors for the full duration of the postulated accident.

Effective (whole body) radiation dose is measured in sieverts (Sv). A sievert is a large dose and so for occupational and accidental exposures, doses are usually measured in millisieverts (mSv or one thousandth of a sievert). Background radiation is also measured in mSv. Exposures to the public due to normal operation of reactors or nuclear facilities are typically measured in micro-sieverts (μSv or one millionth of a sievert).

The annual average effective dose (per person) received from natural sources in a typical Australian lifestyle is about 2.4 mSv. For the area of the proposed reactor it is 1.8 mSv [4]. From man-made sources, the dose received is very lifestyle specific. For example, a frequent flyer from UK to Australia might receive double this in total "background" radiation due to the additional exposure to cosmic rays at high altitudes (that is, 1-5 micro-sieverts per hour at a typical cruising altitude of 8000 to 11000 metres [4]). Typical total background radiation doses are about 3.2 mSv including medical radiation exposures such as X-rays.

For members of the public, the annual dose limit for radiation exposure caused by industry is 1 mSv. For members of an occupationally exposed group, the limit is 20 mSv per year, averaged over five consecutive calendar years, with no more than 50 mSv in one year. ANSTO has adopted a dose constraint of 15 mSv for its radiation workers. These limits are set for the normal operation of a plant.

In the case of accidents, ARPANSA has set a risk-based limit, which takes into account both the frequency and the dose on the most exposed individual of the public. These criteria are indicated in Table 1, both as safety limits (mandatory) and safety objectives (desirable to comply with). The safety limits set the maximum acceptable risk of accidents, grouped by the range of individual doses a member of the public could receive at the site boundary. For each dose range, the risk of accidents should be as low as reasonably achievable (ALARA) below the safety limit. If the risk is less than the safety objective specified for each dose range, it is not necessary to show that the risk is as low as reasonably achievable.

1.2 OBJECTIVES

The basic objective of this PSA is the quantitative evaluation of the risks associated with the operation of the OPAL Reactor. The PSA was also used to compare risks with the regulatory objectives [2].

Since the preliminary PSA was developed in parallel with the basic engineering phase of the OPAL Reactor, the preliminary results were used as input to the design process,

thus permitting improvements to be made to the design. These improvements are already included in the present PSA.

1.3 SCOPE

The present PSA was developed for the OPAL Reactor constructed in Lucas Heights, NSW. For the purposes of the PSA, the plant consists of the reactor itself, its irradiation and beam facilities, and its fuel handling facilities, their corresponding buildings, structures, systems and components, including human actions by plant personnel.

The risk evaluation was performed, taking into account internal events, internal hazards (eg fire) and external hazards relevant to the plant. The exclusion of certain events is discussed in the PSA. Sabotage and security related events are explicitly excluded from the present PSA.

The different PSA scopes are usually classified in three levels:

a) Level I: considers the development of accident sequence models to the point needed to define whether damage to the elements containing radioactive material (eg nuclear fuel) is expected to occur, and to determine the corresponding expected frequency. The prime measure is the CDF.

b) Level II: goes beyond Level I by modelling the release, transport and deposition of radioactive materials from their original location (eg fuel) through the different barriers that contain it (eg fuel matrix, cladding, primary system, containment) through to the environment.

c) Level III: goes beyond level II by modelling the dispersion of the radioactive material outside the plant through the environment and the impact on the public property and the environment.

This PSA is Level I with certain Level III considerations including the selection of accidents which bound the risk and make a significant contribution to the frequency. “Significant” in this context means having a frequency greater than the most stringent frequency set in the safety objective of the ARPANSA regulation [2]. This means a frequency greater than 10^{-6} per year.

1.4 METHOD FOR IDENTIFICATION AND SELECTION OF INITIATING EVENTS

Source and Event Analysis (SEA) was used for the identification and selection of initiating events and it is a bottom-up approach that consists of:

a) Identification of the relevant radiation sources in the plant (eg core, fuel in the spent fuel pool, fuel in the shipping cask, etc).

b) Identification of the barriers that separate the radiation sources from the public and/or plant personnel.

c) Identification of the primary failure mechanisms of these barriers.

d) Identification of the initiating events that may cause the identified failure mechanisms.

e) Systematic selection and grouping in a set of representative initiating events.

A screening process is performed at each step, in order to eliminate those events which make a negligible contribution to the overall risk.

External events were identified and screened according to the IAEA guidelines [5] and later compared with those based on the experience of the HIFAR PSA [6].

A review of incidents in research reactors was prepared independently in Chapter 16 of the SAR and no additional initiating events were identified. A Master Logic Diagram was prepared separately and again, no additional initiating events were identified.

1.5 ANALYSIS METHODS

The PSA followed the following main analysis steps:

a) Initiating events were identified, screened and grouped.

b) The safety systems and functions required for each sequence were identified for each initiating event.

c) Fault Trees were developed for each safety system to model the probability of failure of that system.

d) Event Trees were developed to model the possible sequences following the initiating event.

e) The models were programmed into an appropriate software package and quantified using component failure data from appropriate source such as the IAEA databases [7,8] and the RAC Database [9].

f) The Level I PSA results such as the CDF were quantified and importance measures were determined. The importance measures give an indication of how significant each basic event is to the CDF.

g) The uncertainty and upper bounds on the key Level 1 results were assessed.

h) The plant damage states were grouped into bounding sets, and the plant behaviour beyond the scope of the Level I PSA was modelled using conservative assumptions (rather than full probabilistic models).

i) The public doses were estimated for a worst placed member of the public, and the results are then compared against the safety objectives.

In this PSA, the SAPHIRE software package [10] (approved by the US NRC) was used in this study for the quantification of plant damage frequencies specifically the CDF. For some plant damage states, the frequencies could be estimated by direct calculation without use of PSA software.

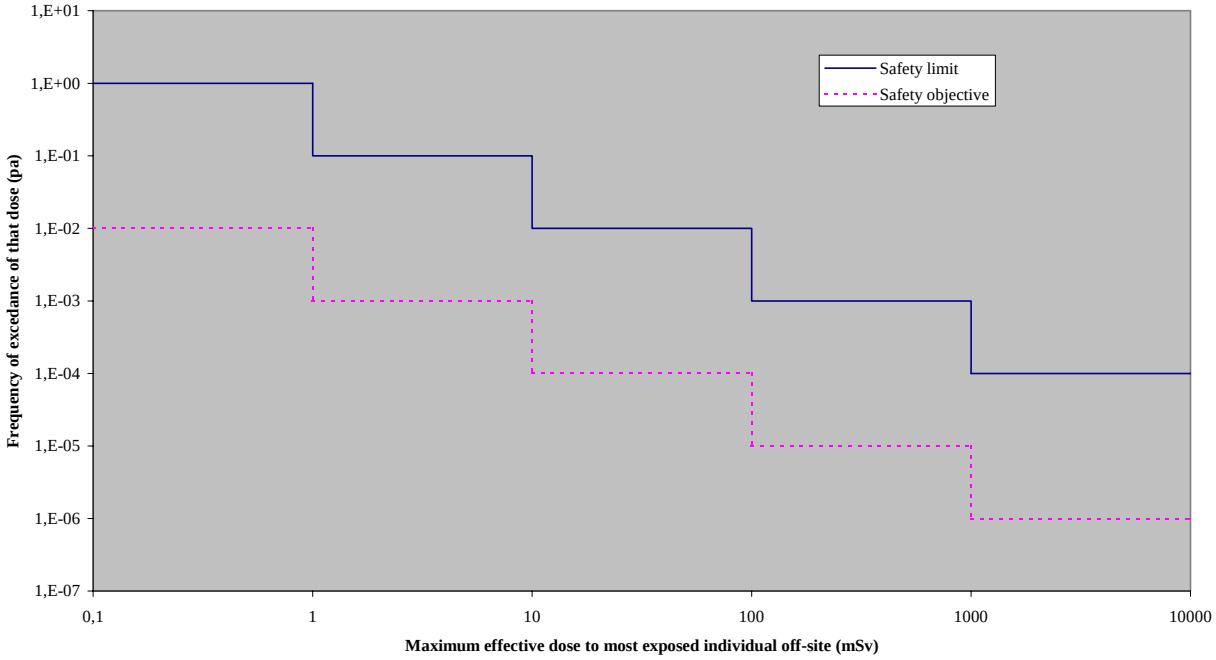
1.6 SAFETY OBJECTIVES

The safety objectives of the risk analysis are those indicated in the Regulatory Assessment Principles of the ARPANSA [2]. It is stated that the acceptability of the PSA is judged by comparison with the Safety Objective and the Safety Limit for the total predicted frequency of accident which result in doses to the public. These are shown in Table 1, and graphically interpreted in Figure 1.

Table 1 Safety Limits and Objectives

Maximum Effective dose to most Exposed individual off-site (mSv)	Total frequency, per controlled facility year	
	Safety limit	Safety objective
0.1 – 1	1	10^{-2}
1 – 10	10^{-1}	10^{-3}
10 – 100	10^{-2}	10^{-4}
100 – 1000	10^{-3}	10^{-5}
> 1000	10^{-4}	10^{-6}

Figure 1 Safety Limits and Objectives



Other regulatory expectations are:

a) The likelihood of failure of a reactor shutdown system does not exceed 10^{-3} per demand (RAP38 [2]), and

b) The frequency of significant damage to core is as low as reasonably achievable and is less than 10^{-4} per year (RAP32 [2]).

The expectation on the frequency of significant core damage is consistent with the most stringent safety limit (*ie* less than 10^{-4} per year), as indicated in Table 1. It is understood that ALARA optimisation is not expected if the CDF proves to be less than the most stringent safety objective (*ie* less than 10^{-6} per year).

2 DEPENDENT FAILURES ANALYSIS

Operating experience with complex plants demonstrates that, although the likelihood of a series of failures is quite small, it is numerically higher than would be estimated solely from a chain of independent failures. This is because physical, human and

environmental interactions result in dependent failures that increase the conditional probability of each successive failure in the chain.

Since essentially all important accident sequences that can be postulated for nuclear reactor systems involve the failure of multiple components, systems, and containment barriers, dependent-failure analysis is an extremely important aspect of PSA.

There are three main types of dependent failures:

a) Functional dependencies between systems,
b) Dependencies or common causes between basic events, and
c) Dynamic human interactions.

These different types of dependencies were treated differently in the PSA.

2.1.1 Functional dependencies between systems

Functional dependencies between systems are where the success of one system is dependent on the success of another, for example due to reliance on a support system or where there is a shared component or subsystem in two systems. In this PSA, functional dependencies were treated explicitly in the event trees.

2.1.2 Dependencies or common causes between basic events

Traditionally in PSAs of existing plants, common cause dependencies (the second type of dependencies) are modelled as a numeric fraction of the basic event failure probability that occur dependently (*eg.* the beta factor or multiple Greek letter methods). Parameters for these models make use of generic data, where available, which are then updated with plant-specific data. The lack of plant-specific data motivated a different approach using human error probabilities to estimate the probability of a latent error remaining in a design; the probability of a maintenance error and of testing errors.

2.1.3 Dynamic human interactions

In the OPAL Reactor the actuation of the safety systems is automatic, and in the PSA, no credit is taken for manual actions. In addition, the operator can not prevent these automatic actions. Therefore, dependent failures which correspond to dynamic human interactions are not relevant because they do not contribute to the failure of safety systems. Where credible operator actions that could jeopardise safety system functions were identified, they were included as basic events in the fault tree models. Furthermore, conservative assumptions were made regarding plant operations where appropriate.

2.1.4 Summary of Common Cause Basic Events

To estimate the human error probabilities in design, maintenance and testing, the THERP (Technique for Human Error Rate Prediction) [11] method was applied and data derived from that same source.

Table 2 shows the summary of common cause basic events data used in this PSA.

Table 2 Common Cause Basic Events Summary

Type	Description	HEP	EF
D	Type D common cause failure represents a hidden design flaw in a set of identical components that fulfil a redundant function.	6.5×10^{-6}	5
M	Type M common cause failure represents an unrecognised failure caused during a maintenance operation on a safety component, or by a restoration operation after maintenance of a safety component.	1.1×10^{-3}	5
T	Type T common cause failure represents an unrecognised failure caused during test or calibration procedure on a safety component, or by a restoration operation after testing of a safety component.	1.1×10^{-3}	5

HEP is the Human Error Probability and EF is the error factor (that is, the 95th percentile divided by the median).

3 INITIATING EVENTS

Having identified initiating events and conducted the screening process, the initiating events were grouped and then quantified, either by using appropriate data from the sources already described [6, 7, 8] or, for more complex events, by developing fault trees. Table 3 shows the initiating events grouped and quantified.

Table 3 Initiating Events

Group	ID	Description	Initiating Event Frequency (per year)		
			5th percentile	Mean	95th percentile
A		**Reactivity transients**			
	A1	Erroneous withdrawal of a control rod during start-up	4.4×10^{-5}	9.8×10^{-5}	1.9×10^{-4}
	A2	Erroneous withdrawal of a control rod during normal operation	6.9×10^{-4}	1.5×10^{-3}	2.9×10^{-3}
B		**Loss of flow**			
	B1	Core Bypass	2.9×10^{-7}	7.4×10^{-5}	2.6×10^{-4}
	B2	Loss of main power supply	2.0	3.0	4.2
	B3	Primary pump failure	4.6×10^{-2}	7.0×10^{-2}	9.9×10^{-2}
	B4	Primary isolation valve undesired closure	3.4×10^{-5}	2.8×10^{-4}	8.5×10^{-4}
	B5	Fuel channel local blockage		1.3×10^{-6}	
C		**Loss of coolant**			
	C1	Primary LOCA caused by a rupture upstream of the primary pump	2.3×10^{-7}	5.9×10^{-5}	2.1×10^{-4}
	C2	Primary LOCA caused by a rupture downstream of the primary pump	4.7×10^{-7}	1.2×10^{-4}	4.2×10^{-4}
	C3	Pool cooling system LOCA	5.3×10^{-7}	1.3×10^{-4}	4.7×10^{-4}
D		**Loss of heat sink**			
	D	Loss of heat sink	1.2×10^{-2}	4.4×10^{-2}	1.4×10^{-1}
E		**Mechanical damage to fuel assemblies**			
	E1	Fuel assembly mechanical damage in the irradiated fuel assemblies pool*			
	E2	Fuel assembly mechanical damage in the spent fuel storage racks in the reactor pool*			
	E3	Fuel assembly mechanical damage while in transit	3×10^{-4}	3×10^{-3}	3×10^{-2}
F		**Heavy water leak**			
	F	Heavy water spill outside reactor pool*			
G		**Events related to reactor utilisation**			
	G1	Loss of local RSPCS cooling flow through a rig#	1.2×10^{-6}	1.2×10^{-5}	1.2×10^{-4}
	G2	Loss of global RSPCS cooling flow through rigs#	3.6×10^{-6}	7.2×10^{-5}	3.6×10^{-4}

Group	ID	Description	Initiating Event Frequency (per year)		
			5th percentile	Mean	95th percentile
	G3	Loss of coolant during transit of a uranium metal target rig	6.9×10^{-6}	6.9×10^{-5}	6.9×10^{-4}
S		**Seismic events**			
	S	Seismic Event bin range acceleration level (g)			
		0.060 - 0.120	3.91×10^{-4}	9.00×10^{-3}	8.04×10^{-2}
		0.120 - 0.230	1.24×10^{-4}	7.36×10^{-4}	7.91×10^{-3}
		0.230 - 0.300	1.55×10^{-5}	1.11×10^{-4}	6.03×10^{-4}
		0.300 - 0.500	9.84×10^{-6}	1.10×10^{-4}	3.48×10^{-4}
		0.500 - 0.550	6.68×10^{-7}	1.01×10^{-5}	2.12×10^{-5}
		0.550 - 0.600	4.62×10^{-7}	7.14×10^{-6}	1.43×10^{-5}
		0.600 - 0.650	3.30×10^{-7}	5.21×10^{-6}	9.94×10^{-6}
		0.650 - 0.700	2.42×10^{-7}	3.90×10^{-6}	7.12×10^{-6}
		0.700 - 0.750	1.81×10^{-7}	2.98×10^{-6}	5.23×10^{-6}
		0.750 - 0.800	1.38×10^{-7}	2.31×10^{-6}	3.91×10^{-6}
		0.800 - 0.850	1.08×10^{-7}	1.83×10^{-6}	2.99×10^{-6}
		Other external events			
		Aircraft crash	3.7×10^{-9}	1.1×10^{-8}	4.8×10^{-8}
		Military		1.0×10^{-8}	
		Fire†			
		Wind†			

* These events were later screened out due to low frequency and insignificant consequences.
# The frequency of these initiators takes into account specific system failures which are part of the sequence leading to the plant damage state.
† These are discussed in Chapter 6 of this PSA.
The seismic event bin frequencies derive from the seismic hazard analysis performed by external geoscience consultants [12 and 13].

The seismic event S is a special case of initiating event because it can occur with various levels of ground acceleration, each with its own frequency. The seismic event is assumed to cause also the loss of main power supply. Mains water supply could also be expected to be disrupted and for extreme events, the LHSTC water tower might also be damaged, but the OPAL Reactor does not rely on mains water in the short term. The Emergency Makeup Water System would be expected to be full at the start of any seismic event.

4 EVENT TREE HEADINGS

The Event Tree Headings used in the Event Trees represent the systems that protect against accidents and/or mitigate the consequences of accidents and include:

- First and Second Reactor Protection Systems (FRPS and SRPS)
- First Shutdown System (FSS)
- Second Shutdown System (SSS)
- Flap Valves and Siphon Breakers – which includes five headings:
  - Siphon Breakers (SEB)
  - Suction & Impulsion Siphon Breakers (S&I-SEB)
  - Flap Valves at Level +5.80 (FVL6000)
  - Flap Valves at Level +7.20 (FVL7000)
  - Flap Valves at Level +5.80 and +7.20 (FV6&7)
- Emergency Makeup Water System
- UPS – Standby Power Supply
- Containment Isolation System (which consists of the containment isolation and the Containment Energy Removal System)
- Diesel Generators - Standby Power Supply (SPS)

Each of these systems has a corresponding fault tree that details the combinations of events that can cause failure of that system. The model development begins with the main function of the system, its description, and its success criteria. Seismic considerations were also developed so as to correctly model the systems under the seismic event taking account of those components which are assumed vulnerable to seismic loads.

The modelling also took into account the Failure Modes and Effects Analysis (FMEA) of each system.

5 EVENT TREES

Event Trees were derived based on the initiating events described in Chapter 3 of this PSA and the Accident Analyses included in Chapter 16 of the SAR.

In the development of the Event Trees, the following conservative assumptions were made:

a) No credit was taken for any action of the Reactor Control and Monitoring System (RCMS) that could interrupt an accident sequence before any safety system is

triggered, although it is assumed that the RCMS can initiate fault sequences *eg* inadvertent control rod withdrawal.

b) No credit was taken for any possible human action to activate a Safety System on alarms during an accident sequence, nor for any human recovery action that may rectify or activate a Safety System after its failure, nor for any other human recovery actions (such as the manual closure of the Primary Coolant System isolation valves during LOCA sequences).

c) In LOCA sequences, the Primary Cooling System (PCS) Pumps were assumed to continue operating despite any trip indicated by the RCMS. This implies, for the LOCA sequences, that the pumps will run until cavitation failure.

d) For the LOCA sequences, those terminated with successful action of natural circulation (at either +7.20 or +5.80 levels) are considered successful and therefore they do not result in core damage.

e) Provided the reactor is successfully shut down, and the core remains covered with water, natural circulation is sufficient to prevent core damage, irrespective of the state of the flap valves (see the SAR).

5.1 DEVELOPMENT AND QUANTIFICATION OF EVENT TREES

The event trees for the postulated initiating events are developed in the following sections. These event trees are then quantified with SAPHIRE using the initiating event frequency data derived in chapter 3, and the failure probability data for the event tree headings, derived in chapter 4.

6 FIRE PSA AND EXTREME WIND PSA

6.1 FIRE PSA

A Fire PSA is the probabilistic analysis of fire events and their potential impact on the nuclear safety of a plant. Using probabilistic models, the fire PSA takes into account the possibility of a fire at specific plant locations; the propagation, detection and suppression of the fire; the effect of the fire on safety related cables and equipment; the possibility of damage to these cables and equipment, and in severe fires, the structural integrity of the walls, columns, roof beams, etc.; and the assessment of the impact on plant safety. Since the physical separation between redundant safety trains can limit the extent of fire damage, quantification of the damage frequency calculations generally includes those equipment failure probabilities that are not affected by the fire, *eg* random failure probabilities, and the likelihood of a maintenance outage.

The fire risk assessment methods introduce the likelihood of a fire in each plant location, the effect of the fire on equipment and cables, and the impact of equipment failures and human actions coincident with the fire.

The Fire PSA relies on the plant response model developed for the internal initiating events. The availability of the internal PSA that logically examines the contributions to core damage, plant damage, etc. is a prerequisite for the fire PSA.

The fire PSA approach is based on systematic examination of all plant locations. To facilitate this examination, the plant is subdivided into distinct fire locations, which are then scrutinised individually.

In keeping with good practice, the screening process was carried out in stages, starting with relatively simple, conservative models and progressing to more realistic representation of the fire scenarios at subsequent stages.

6.1.1 Main Assumptions

The fire PSA discussed in this report is intended to reflect the status of plant design using a best estimate assessment.

Only a single, independent fire is assumed to occur in any plant location. The spread of this fire throughout the fire compartment is assumed, but spreading to adjacent fire compartments is not taken into account. This is based on the deterministic Fire Hazard Analysis, where very low level of combustible materials and finishes were found, and the provision of appropriate fire segregation between fire compartments in accordance with the BCA.

The most severe natural phenomena, *eg* tornadoes, flooding or earthquakes, are not assumed to occur concurrently with a fire. Internal initiating events (*eg* LOCA) are also not considered to be concurrent with a fire, unless they are a consequence of that fire.

Fires induced by other initiating events (*eg* earthquakes, sabotage) are not considered to be within the scope of this fire PSA.

6.1.2 Method

The fire PSA methodology used in this report is based on Safety reports series N° 10, IAEA “Treatment of internal fires in probabilistic safety assessment for nuclear power plants” [14]. The major steps are:

Data collection and assessment

Definition of fire compartments and cells divides all plant buildings and structures into distinct fire compartments and cells, which are examined individually. Based on the internal events PSA, all compartments that are involved in internal events were considered. Compartments not involved in internal events were screened out.

Familiarisation with the internal events establishes a link between the existing internal events PSA models and the fire related models. It starts with examination of the internal events logic models (*eg* fault trees and events trees), and their applicability to fire risk modelling. This task also identifies those plant systems and equipment, and all the related elements of the model that are important to fire PSA. Identification of all the related cables and circuits is an integral part of this examination.

Inventory of equipment and cables is developed and a list of PSA related items is prepared for each fire compartment and cell defined above.

Screening by impact is used to screen out non-critical fire scenarios on the basis of impact oriented criteria. It starts with definition of the critical fire locations, followed by definition of the possible single and multicompartment fire scenarios. Only a single, independent fire is assumed to occur in any plant location based on the deterministic Fire Hazard Analysis where very low level of combustible materials and finishes were found, and the provision of appropriate fire segregation between fire compartments in accordance with the BCA.. The result of this step is a list of fire scenarios that can be significant contributors to risk.

Screening by frequency: is aimed at the further elimination of those fire scenarios that are retained after the first stage of screening. Screening is performed on the basis of a simple, conservative estimate of the damage frequency. The conditional unavailability of

the required safety functions because of a fire is calculated from the existing internal events PSA model. For each of the remaining fire scenarios, a quantitative PSA model is developed for further analysis.

6.1.3 Internal Events

Each initiating event was reviewed in order to determine whether a fire could induce it. This could happen for example, by fire induced failure of actuation signals for equipment operation from the control room and control relay cabinets.

The following fire induced internal events were identified as important to the fire PSA:

- A1: Erroneous withdrawal of a control rod during start-up.
- A2: Erroneous withdrawal of a control rod during normal operation
- B3: Primary Pump Failure
- D: Loss of heat sink

The other internal events were screened out because they cannot credibly be caused by fire. In addition, no “new” internal events that could be induced by a fire have been identified.

Inspection of the event trees for these internal events shows which systems would be required to respond in the case of a fire initiated event. Of these systems, only the First Reactor Protection System and the Second Reactor Protection System could be credibly degraded by fire, the mechanism being short circuit induced by fire damage to cabling.

6.1.4 Fire Events Summary Results

By examination of the four events discussed above and assessment of the fire compartments and equipment therein, nine fire scenarios were identified that could lead to core damage. The CDF obtained by the summation over all the frequencies of fire event sequences that may lead to core damage is 1.28×10^{-8}/year.

This value has been obtained under very conservative assumptions, and even so the fire CDF results are well below the most stringent regulatory safety objective. Therefore, the internal fire modelling is not developed in further detail, which would certainly produce even lower CDF estimations.

6.2 EXTREME WINDS AND TORNADOES HAZARDS

The frequency of winds of various velocities at the site is obtained for the Wind Hazard Analysis performed to evaluate the risk to the HIFAR facilities from wind loading and from missiles generated by tornadoes (HIFAR PSA Section 6.3) [6]. This hazard is expressed in terms of the frequency of exceedance of the fastest-mile wind speed for the site.

The only credible consequence of extreme winds are (i) a loss of offsite power due to its impact on electric lines that connect to the OPAL Reactor, (ii) degradation of secondary cooling and (iii) damage to the stack (although the stack is not needed to prevent any core damage).

The conditional probability of loss of offsite power, given extreme wind occurrence, was conservatively assumed at unity. However, loss of offsite power (not necessarily induced by extreme winds or tornadoes) has a relatively high frequency so the addition of the frequency of loss of offsite power due to such winds would be insignificant.

Provided that the core damage frequency of loss of offsite power is acceptable from the frequency point of view, and that the extreme wind frequency for the site is rather low (of the order of 10^{-6}/year), the overall effect on CDF due to extreme winds is screened out for the PSA.

7 LEVEL 1 PSA RESULTS

The quantification of the PSA models provides estimates of the frequencies of plant damage states, notably the CDF.

The Level I PSA results represent a quantitative evaluation of the Core Damage Frequency (CDF), its contributors, their relative contribution, and its uncertainty range.

Those events that do not contribute to the CDF were not developed in Event Trees and are discussed in chapter 8.

7.1 INTERNAL EVENTS CONTRIBUTION TO CORE DAMAGE FREQUENCY

7.1.1 Internal events summary results

The CDF obtained by the summation over all the frequencies of internal event sequences that may lead to core damage is

Mean CDF: **1.43×10^{-7}/year**

5% Percentile CDF: **2.28×10^{-8}/year**

95% Percentile CDF: **3.37×10^{-7}/year**

These values, when compared with the Safety Limits and Objectives of Table 1 and Figure 1, indicate, with a 95% confidence, that the CDF induced by internal events satisfies the most stringent frequency set out in the Safety Objective (10^{-6}/year). Therefore, it can be stated that those accidents with the potential to cause a significant damage to the core, pose a negligible risk to the public in the vicinity of LHSTC.

7.2 SEISMIC EVENTS CONTRIBUTION TO CORE DAMAGE FREQUENCY

The seismic contribution to the Core Damage Frequency was analysed in two different scenarios.

The first scenario considers the contribution to the CDF for those seismic events whose frequency is up to that stated for the SL2 earthquake, that is, with a frequency of 10^{-4}/year or greater. This scenario is considered appropriate from the standpoint that the critical systems in the OPAL Reactor have been designed to withstand this seismic event, in accordance with IAEA recommendations.

The second scenario considers the contribution to the CDF for all the seismic events indicated by the Hazard Curve. This scenario is not considered credible, but it was analysed in order to have an estimation of the seismic events beyond SL2.

The resulting contribution to the CDF of the Seismic Events (full hazard curve) is given here.

The CDF obtained by the summation over all the frequencies of seismic event sequences (for the full hazard curve) that may lead to core damage is:

Mean CDF: **3.75×10^{-7}/year**

5% Percentile CDF:	**2.24×10^{-8}/year**
95% Percentile CDF:	**6.34×10^{-7}/year**

These values, when compared with the contribution of internally initiated events, indicate that the extreme seismic events have a contribution of the same order of magnitude. Given the simplistic conservative assumptions for the calculations in these extreme cases, and the fact that even for these cases the ARPANSA criterion is fulfilled, it can be concluded that the OPAL Reactor design is very robust for these extremely rare events.

7.3 TOTAL CONTRIBUTION TO CORE DAMAGE FREQUENCY

The overall core damage frequency, taking into account the internal initiators, the full seismic hazard, and all identified external initiators is given below.

Overall CDF with FULL seismic hazard

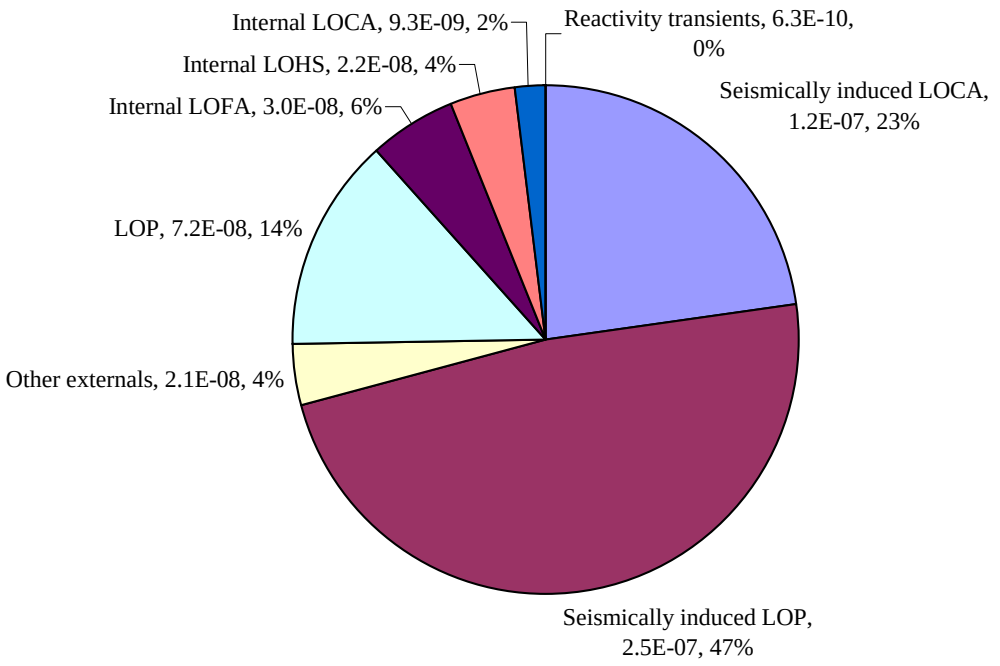
Mean Core Damage Frequency:	**5.4×10^{-7}/year**
5th Percentile Core Damage Frequency	**3.0×10^{-7}/year**
95th Percentile Core Damage Frequency	**9.1×10^{-7}/year**

The regulatory requirement is that the Core Damage Frequency should be less than 10^{-4}/year.

It may thus be concluded that those accidents involving core damage in the OPAL Reactor have a very low likelihood of occurrence which is well below the ARPANSA criterion.

The relative contribution to the mean total CDF due to each of the main classes of initiating events is indicated in Figure 1. It can be seen that the seismically induced loss of offsite power is the greatest contributor to the CDF with (about 48% of CDF) and seismically induced LOCAs the next most significant cause. Loss of offsite power (non-seismic) and internal (or randomly occurring) LOCAs are the next two most significant classes of initiators.

Figure 2 Initiator Contributions to the Core Damage Frequency (LOCA – Loss of coolant accident; LOHS – Loss of heat sink; LOP – Loss of offsite power; LOFA – Loss of flow accident)



8 LEVEL III PSA CONSIDERATIONS AND COMPARISON WITH OBJECTIVES

8.1 INTRODUCTION

This section of the PSA analyses those accidents that have potential for offsite consequences.

It is usual that the accident scenarios that contribute the most to the risk of a nuclear reactor are those that involve substantial damage to the reactor core. However, the very robust design of the OPAL Reactor, with a high degree of redundancy and independence of its safety functions, makes these sequences to be of such low probability that they are not considered as credible.

This was demonstrated in chapter 7 of the present PSA, where the low values for core damage frequency (CDF) were determined from all identified initiators. These very low values comply with the most stringent ARPANSA regulatory requirements of frequency for any dose level. One of the reasons for this is that the PSA was developed at the same time as the basic engineering was developed, and the weak points identified

during the system analyses for the PSA, were fed back to the designers, who in turn improved those system designs.

It is important to mention that the uncertainty effects were also considered in the CDF evaluations, and the upper bound (95%) confidence for the CDF is also within the most stringent safety objectives of ARPANSA. Therefore, core damage is not a scenario warranting consequence assessment for the purposes of this PSA and the representative risk scenarios do not include significant core damage.

However, in addition to accidents with a credible potential for damage to the core, there are other potential sources of radioactive release in the OPAL Reactor.

These types of accidents are related to fuel management operations, experimental irradiations, etc. The analysis of these events has been done by making extremely conservative assumptions in order to obtain a set of risk-representative scenarios where a source term and a yearly frequency can be derived. Hence, the use of these assumptions produces results which are overestimations of the risk.

The accident scenarios B5, E3, G1, G2, and G3 were analysed and the analysis demonstrates that even with these extremely conservative assumptions, the regulatory requirements are fulfilled. For these accidents, their release fractions were derived, based on conservative assumptions, and the containment response was analysed, in order to obtain representative source terms for the installation. The dose expected for these source terms on the public was also calculated, taking into account conservative weather conditions.

8.2 CONTAINMENT RESPONSE

The containment has been designed to limit any radioactive release to the environment. To estimate doses to the public, conservative assumptions were made regarding the containment behaviour. These assumptions include:

a) The “prompt” release would be through the stack at 45 m.

b) Containment closure would occur two minutes after the accident.

c) Subsequent release would be by leakage of the containment and this would be assumed to occur at ground level (0m).

d) There is no filter retention of any kind.

e) The containment leak rate of would be 3% of the volume per day for the first day following closure of the containment (while pressure in the containment could be elevated) and 2% per day thenceforth, for up to 99 additional days.

f) The release continues for up to 100 days.

For the case of the targets melting in the hot cell, although the stack filter retention was again assumed to be zero, the cell recirculation filtering was assumed to be degraded. In the recirculation there are two charcoal filters and three absolute filters, while in exhaust there is one charcoal and two absolute filters. The individual efficiencies were considered degraded (99% for each absolute filter and 90% for each charcoal filter).

These assumptions are realistic but conservative. For example, the containment would be expected to close within one minute of an accident.

Dose estimations were made using PC-Cosyma. To estimate individual doses at the buffer zone boundary (1600m) the following additional conservative assumptions were made in addition to those already mentioned in section 8.2:

a) for the fuel and fuel plates events, the inventory was calculated for equilibrium (maximum) core and increased by 10% to take into account uncertainties

b) conservative retention factors for the water were assumed

c) radioactive decay was neglected during water transport

d) the worst atmospheric conditions (Pasquill F with 1ms^{-1} wind speed, winter season) were assumed during the first 12 hours followed by 12 hours of Pasquill at 3 ms^{-1}, another 12 hours of Pasquill F at 1 ms^{-1} and the final 98.5 days, Pasquill D at 3 ms^{-1}.

The results for these five scenarios are given below.

Release category RC-B5:

- Annual frequency: 1.3x10^{-6}/year.
- Inventory: 0.9% of the core inventory (melting).
- Maximum individual effective dose: 7.67 μSv.

Release category RC-E3:

- Annual frequency: 3x10^{-3}/year.
- Inventory: one assembly mechanically damaged, conservatively bounded by the equivalent of 6.25x10^{-4} of the core fission product inventory.
- Maximum individual effective dose: 0.533 μSv.

Release category RC-G1:

- Annual frequency: 1.2x10^{-5}/year.
- Inventory: 100% of one uranium metal rig inventory (melting).
- Maximum individual effective dose: 5.64 μSv.

Release category RC-G2-FRPS:

- Annual frequency: 7.2x10^{-5}/year.
- Inventory: 1200% of one uranium metal rig inventory (melting)
- Maximum individual effective dose: 67.7 μSv.

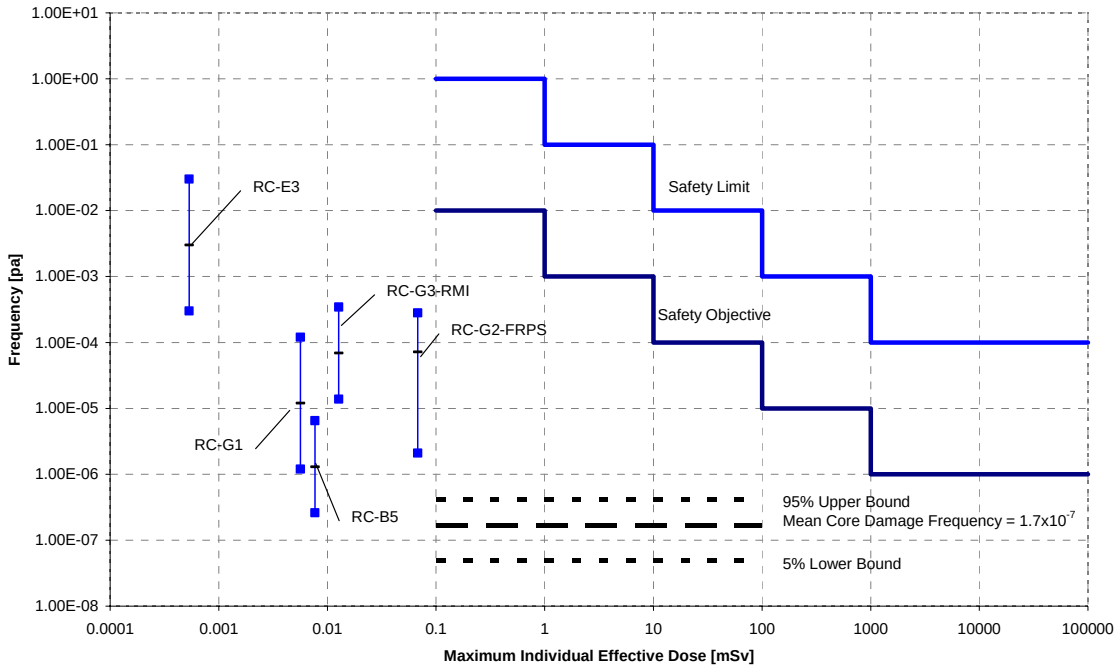
Release category RC-G3-RMI:

- Annual frequency: 6.9x10^{-5}/year.
- Inventory: 100% of one uranium metal rig inventory (melting in air).
- Maximum individual effective dose: 12.7 μSv.

8.3 RISK RESULTS AND COMPARISON WITH REGULATORY ASSESSMENT PRINCIPLES

The estimates for the five release categories are indicated in the following Figure 3, and compared to the regulatory requirements. For clarity, the sum of the frequencies in each dose band is not shown. There are at most, only two potential release categories in each band, so it can be seen that all of them fall below the regulatory safety objectives.

Figure 3 Risk Results and Comparison with Regulatory Assessment Principles



The PSA demonstrates that the FSS (with unavailability 5.7×10^{-5}) and the SSS (with unavailability 3.2×10^{-4}) individually meet the regulatory assessment principle on the likelihood of failure of shutdown systems (RAP38 [2]). Furthermore, these two systems are independent and diverse, and so failure of both is not credible.

The PSA also demonstrates that the OPAL Reactor meets the regulatory assessment principle on the frequency of significant core damage (RAP32 [2]). It is understood that ALARA optimisation is not expected because the core damage frequency is one order of magnitude below the most stringent safety objective (*ie* less than 10^{-6} per year). The regulatory requirement is that the Core Damage Frequency should be less than 10^{-4}/year.

Therefore, the OPAL Reactor is capable of meeting the Regulatory Assessment Principles with respect to doses to the public, core damage frequency and reliability of shutdown systems.

The PSA includes consideration of all envisaged operation modes of the OPAL Reactor and all expected radiation sources that may exist in the plant. However, as the specific operating and maintenance manuals are not yet complete, several conservative hypotheses were made. If more realistic assumptions are made, it is expected that there will be a reduction in the risk estimations from the present PSA. In this sense, the present results are a bounding estimation of the risk. They demonstrate that there is no design basis accident or credible beyond design basis accident that can cause a high dose in the worst placed member of the public, thus demonstrating that no off-site emergency countermeasures would be required.

The calculated doses are well below the minimum intervention level for sheltering, which is set at 5 mSv [15].

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