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Replacement Research Reactor Project

SAR CHAPTER 16
SAFETY ANALYSIS

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16 SAFETY ANALYSIS

16.1 INTRODUCTION

This Chapter presents the accident analysis aspects of the safety analysis of the Replacement Research Reactor based on the design features presented in other chapters. The objective of this analysis is to demonstrate that the design meets safety and licensing requirements and the safety design criteria set out in Chapter 2.

Following the ARPANSA Safety Assessment Principles for Controlled Facilities, a deterministic analysis of the behaviour of the reactor and associated systems following a Design Basis Initiating Event has been performed. The quantitative analyses have been performed with computer codes. Several parameters in the reactor core have been studied, such as temperature of the fuel cladding and the coolant, flow rate through the core and the flap valves, temperature of the pool, reactor power, etc. All assumptions in these analyses are conservative. In a number of cases complete failure of a safety system has also been assumed. In addition, reactor trip by the second acting signal have been assumed wherever relevant. The numerical calculations show that the reactor goes through a series of safe states following the occurrence of a Design Basis Initiating Event. The description and analysis of each Design Basis Initiating Event and event sequence is presented in Sections 16.7 to 16.18.

Three representative Beyond Design Basis accident sequences involving fission product release have been analysed to determine the impact on the public of that release. These were:

a) The failure underwater of 12 uranium-molybdenum rigs.

b) The melting of 3 Fuel Assembly fuel plates.

c) The melting of a uranium metal rig in the hot cell.

For all these accidents, unfiltered release of fission products through the stack was assumed together with very conservative behaviour of the containment after containment isolation (section 16.19). The analysis showed no need would exist for any counter-measures for people living beyond the 1.6 km buffer zone around the Reactor Facility.

In addition, compliance with the ARPANSA dose-frequency curve has been shown with the aid of a Level 1+ Probabilistic Safety Analysis. The Probabilistic Safety Analysis shows that the likelihood of significant core damage to the facility is well within ARPANSA requirements. A summary of the methodology and the results of the Probabilistic Safety Analysis are presented in Section 16.20.

In summary, the deterministic safety analysis of the Reactor Facility shows no damage to the core following any design basis accident. The Probabilistic Safety Analysis shows that the design complies with the dose-frequency curve established by ARPANSA. The analysis of beyond design basis accident sequences show that no evacuation or other emergency counter-measures are necessary for the population in the vicinity of the reactor.

End of Section

16.2 DEFENCE IN DEPTH

Fundamental to the safety of the Reactor Facility, is the adoption of the strategy of Defence in Depth. The strategy of Defence in Depth is twofold: first, to prevent accidents and second, in the unlikely event that prevention fails, to limit the potential consequences of accidents and prevent their evolution to more serious conditions.

Defence in depth is structured in five levels. The objectives of each level of protection and the essential means of achieving them are shown below. If one level were to fail, the subsequent level comes into play, and so on.

Level	Objective	Essential means
Level 1	Prevention of abnormal operation and failures	Conservative design and high quality in construction and operation
Level 2	Control of abnormal operation and detection of failures	Control, limiting and other surveillance features
Level 3	Control of accidents within the design basis	Engineering safety features and accident procedures
Level 4	Control of severe plant conditions, including prevention of accident progression and mitigation of the consequences of severe accidents	Complementary measures and accident management
Level 5	Mitigation of radiological consequences of significant releases of radioactive materials	Siting: 1.6 km Buffer Zone Off-site emergency response

Substantial Defence-in-Depth measures and characteristics have been included in the design of the Reactor Facility to:

a) Compensate for potential human and component failures.

b) Maintain the effectiveness of the barriers to fission product release by avoiding damage to the plant and to the barriers themselves.

c) Protect the public and the environment from harm in the event that these barriers are not fully effective.

All postulated initiating events are analysed with a summary of relevant aspects of the first two levels of defence in depth. For selected design basis initiating events, information is presented on the relevant aspects of the third level of defence.

End of Section

16.3 ANALYSIS METHODOLOGY

16.3.1 Identification of Initiating Events

Postulated Initiating Events (PIEs) are events that have the potential to challenge the safety limits of the plant. They are the initiators of fault sequences. The primary causes of Postulated Initiating Events may be equipment failure and operator errors (both within and external to the reactor facility) and man-induced or natural events.

Following the guidance of the IAEA, a set of Postulated Initiating Events was assembled for assessment against the design of the Replacement Research Reactor. This list covered all aspects of the design, operation and utilisation of research reactors. The Postulated Initiating Events presented here have been obtained by means of a systematic comparison of the IAEA list of initiating events (Safety Series 35-G1) with the design of the facility and the application of engineering judgement. The assessment of the specific design resulted in the identification of other initiating events that are specific to the Replacement Research Reactor. These include events such as those involving the Cold Neutron Source. In parallel to the identification of the Postulated Initiating Events for this chapter, a Probabilistic Safety Analysis was carried out. The results of the evaluation of initiating events have been compared with the Postulated Initiating Events considered in the Probabilistic Safety Analysis. The Postulated Initiating Events presented in this deterministic analysis agree with those identified in the Probabilistic Safety Analysis.

Pool-type experimental reactors have been in operation around the world for over 30 years. The proposed design has been evaluated against the information available on operational events and incidents gathered by IAEA. The results of the evaluation are presented in Section 16.21. The list has been reviewed in the light of the Reactor Facility design and the design provisions that render these events inapplicable have been highlighted.

Using all the above methods, the following fault-schedule has been prepared for the Reactor Facility:

1. Loss of electric power supplies.
   a) loss of Normal Power
2. Insertion of excess reactivity.
   a) accidental drop of a Fuel Assembly
   b) inadvertent fast insertion of irradiation fissile material
   c) start-up accident
   d) inadvertent Control Rod (CR) withdrawal during operation
   e) Control Rod Drive (CRD) or system failure
   f) inadvertent CR bank extraction
   g) inadvertent extraction of a fixed absorbing irradiation material
   h) inadvertent extraction of a pneumatic can with excess irradiation material
   i) cold water insertion
   j) inadvertent refill of the Reflector Vessel
3. Loss of flow.
   a) Primary pump failure.
   a) Primary coolant flow reduction (e.g. valves failure, blockage in piping or heat exchanger).
   b) Influence of reactor utilisation failure or mishandling.

c) Emergency Make-up Water System spurious trip.
d) Fuel channel blockage.
e) Improper power distribution due, for example, to unbalanced rod positions, in-core experiments, or fuel loading (power-flow mismatch).
f) Coolant reduction due to core bypass.
g) Malfunction of reactor power control.
h) System pressure deviation from specified limits.

4. Loss of heat sink (e.g. valve or pump failure, system rupture).

5. Loss of coolant in the Primary Cooling System (PCS)

   a) primary coolant boundary rupture

6. Loss of coolant in the Reactor and Service Pools Cooling System (RSPCS)

   a) damaged pool
   b) pump-down of pool
   c) failure of beam tubes or other penetrations

7. Loss of heavy water

8. Erroneous handling or failure of equipment or components.

   a) fuel plate cladding failure
   b) mechanical damage to core or fuel (e.g. fuel handling, dropping or transferring flask on fuel)
   c) criticality in fuel storage
   d) containment system or ventilation system failure
   e) loss of coolant to fuel in transfer or storage
   f) loss or reduction of proper shielding

9. Special internal events.

   a) internal fire or explosion
   b) internal flooding
   c) loss of supporting systems
   d) security incidents
   e) improper access to restricted areas

10. Reactor utilisation malfunctions.

   a) bulk production irradiation facilities

      (i) early removal of a U-Mo target to the transfer hot cell
      (ii) excessive power
      (iii) failure of the cooling system
      (iv) rigs exchange
      (v) staff irradiation due to inappropriate handling

   b) pneumatic transfer systems and neutron activation analysis

      (i) excessive target activity
      (ii) excessive target heating power
      (iii) interruption of cooling
      (iv) stuck sample
      (v) can breakage inside the pneumatic system piping
      (vi) can breakage inside a hot cell
      (vii) failure of the electrical system

   c) transfer, loading and pneumatic cells

(i) failure of the ventilation system
(ii) fire/short circuit
(iii) failure in the electrical system

d) large volume irradiation facilities

(i) fall during manipulation
(ii) Inter-building Pneumatic Transport System (IPTS)
(iii) damage to transport cask

e) Cold Neutron Source

(i) Leak in the D_2 Pipe/Moderator loop
(ii) Failure of the He cooling system
(iii) Explosion due to explosive mixture

f) Neutron Beam Facilities

(i) unauthorised access to the Neutron Guide Bunker
(ii) primary shutter opened without warning
(iii) failure in the electrical system
(iv) loss of light water
(v) loss of heavy water
(vi) loss of coolant to the Neutron Guides front section

11. Spurious trigger of Safety System components.

a) spurious triggering of the First Shutdown System (FSS)
b) spurious triggering of the Second Shutdown System (SSS)
c) spurious containment isolation
d) spurious start up of a Diesel Generator

12. External events.

a) aircraft impact
b) bushfires
c) industrial activities
d) military activities
e) on-site activities outside the facility
f) transportation accidents
g) extreme wind
h) earthquake
i) sabotage

13. Human factor.

16.3.2 Identification of Design Basis Initiating Events

Each Postulated Initiating Event was then assessed to determine whether or not it is relevant to the design of the Reactor Facility. If it was deemed relevant, then it was identified as falling within the design basis and considered a Design Basis Initiating Event or ‘DBIE’. The philosophy utilised was that all Postulated Initiating Events were applicable to the design of the RRR unless they could be screened out. The screening criteria were;

a) inapplicability to the design,

b) elimination by inherent design provisions,

16.3.3 Methods of Analysis

The set of DBIE identified covers all credible accidents that have the potential to influence reactor safety. The various DBIEs identified were reviewed to identify those DBIEs that had consequences that bounded other DBIEs. The identification of such bounding DBIEs reduced the amount of necessary analysis. Conservative assumptions were made in the determination of the bounding DBIEs.

Having identified the bounding DBIEs requiring further analysis, particular transients representative of the DBIEs were defined for detailed analysis. The response of the reactor to the transients was analysed and evaluated to demonstrate that the design met safety objectives and was acceptable to ANSTO and the regulatory body. Analyses of reactor response to the transients are provided. In some cases it was not necessary to perform detailed assessment of the sequence to ascertain its minor consequences. Where this occurred, it is explicitly noted.

Although unlikely, failure of one safety system, the FSS, is considered within the Design Basis. Therefore, for some transients, failure of the FSS with actuation of the SSS is postulated. For those DBIEs deemed to have a very low likelihood of occurrence, failure of the FSS is considered to render the combined event sequence beyond the design basis and it is not considered further. Failure of two safety systems is considered extremely unlikely and is not considered to lie within the design basis. In particular, given the redundancy, independence and fail-safe characteristics of the Shutdown Systems, failure to shutdown the reactor is not considered credible and it is not postulated in the analysis.

The transients were analysed by calculating the evolution of the main reactor parameters using appropriate computer programs. Conservative modelling assumptions were made regarding the response of the reactor and the actuation of the safety systems. These assumptions included neglecting the negative reactivity inserted by the First Shutdown System when its failure was postulated (failure of the FSS entails the failure to reach the end of stroke for two or more control plates within the necessary time. The analysis assumed that no control plates were inserted for the duration of the transient). Moreover, even though the SRPS trip of the SSS sends an actuation signal to the FSS FAL (directly, without going through the FRPS), no credit has been given to the insertion of plates. In addition, no credit was given to the actuation of Safety Category 2 systems, such as the RCMS. These are very conservative assumptions.

The design philosophy is that no significant damage to either the core or the rigs shall occur for any design basis accident. The intent is that such damage be restricted to accidents having a low likelihood of occurrence. Minor damage to material is tolerable, e.g., mechanical damage to a fuel assembly causing a crack in the cladding, as the consequences would be minor. The aim of the analysis is to show that the core and rigs are safely brought to a shutdown state with the core and rigs being cooled by natural convection. As part of the conservatism in demonstrating this, the safety systems called upon are assumed to work at their minimum design values.

Beyond Design Basis Accidents are either initiated by a very unlikely event or include the simultaneous failure of two or more Safety Systems. As shown in the Probabilistic Safety Analysis, the redundancy, diversity and independence of the Safety Systems results in a low probability of occurrence of simultaneous failure.

Beyond Design Basis Accidents are considered for the purposes of emergency planning and accident management.

16.3.3.1 Event Sequence Analysis

Once the DBIE is identified, the sequence of events is outlined. The sequence of events includes the actuation of the Safety Category 1 systems that control the process initiated by the DBIE. Where prompt reliable action is required to deal with DBIE, the reactor design includes the means to automatically initiate the operation of the necessary safety systems. This ensures that the three main safety functions, namely: reactor shutdown, core cooling, and radionuclides confinement are carried out with a high degree of reliability. In some cases, in presence of a DBIE, it will be useful for the operator to take further action to bring the reactor to a stable long-term state. The design reduces demands on the operator as far as feasible, particularly for the period during and following an accident condition (within 30 minutes).

Single Failure within the safety systems is assumed in the calculations. No reliance is placed on Safety Category 2 systems to mitigate a DBIE. Thus successful operation of the RCMS is not considered in this conservative analysis although it would be expected in reality.

16.3.3.2 Safety Systems Settings

The objective of the Safety Systems trip set points is to trigger an automatic protective action before a Safety Limit is exceeded. For the RRR the Safety Limit is the fuel meat temperature. This Safety Limit guarantees the integrity of the FA, and thus undue radioactive releases are prevented. The fuel meat temperature is not directly measured by the reactor instrumentation, and the phenomena governing this temperature are non-linear. This requires that the Safety Limit be expressed in terms of other limits, more readily related to actual process variables.

The design criterion adopted to avoid exceeding the Safety Limit has been to avoid the occurrence of critical phenomena in the reactor core. The relevant phenomena for the RRR are Redistribution and Critical Heat Flux in the Power state, and boiling and burn-out in the Physics Test state. Thus, limits on these phenomena are imposed and evaluated for reactor operation and accident conditions. These limits, expressed as ratios of limiting conditions to nominal or maximum allowed conditions, are presented in Chapter 5, Section 5.8.

The margins to critical phenomena existent in the reactor core for given conditions of power and cooling are also not directly measured by the instrumentation, so calculations are carried out to relate the Safety Limits to process variables that can be measured, so that limits can be placed on these process variables. The limits thus determined are called within this work "analytical limits". The analytical limits are then values of process variables that, if not exceeded, ensure that critical phenomena will not occur, which in turn ensure that the Safety Limit will not be exceeded.

The analytical limits are developed from event analyses models that consider parameters such as process delays, rod insertion times, reactivity changes, instrument response times, etc. The trip set points are the values of measured process variables at which the final set point device is set to actuate. The trip set points are determined from the analytical limits, corrected for defined process, calibration, and instrument errors. This correction ensures that, in practice, the protective action will be initiated before the process variable exceeds its analytical limit.

Therefore, a trip set point established at the analytical limit is a conservative assumption. The actual set point will be lower or higher (depending on the nature of the parameter), towards a more conservative number. All the analyses of Design Basis Accidents have been performed considering the analytical value for the trip set point.

16.3.3.3 Transient Analysis

Modelling and simulation of transients represent a valuable tool in the assessment of the behaviour of the reactor and its systems during the DBIE sequences.

The models implemented in the computational tools are applicable over the expected range of operational parameters, except where explicit limitations are outlined.

Results from the computer modelling yield conservative predictions, mainly due to the safety factors, modelling assumptions and data implemented. An uncertainty of 20% is considered in the reactivity worth of all insertions postulated in the reactivity transients. Uncertainties adopted for the temperature, neutronic parameters, flow rate, and other thermal hydraulics and core parameters are presented in Chapter 5.

16.3.3.3.1 Computational Programmes

16.3.3.3.1.1 PARET-PC

This thermal-hydraulic computational code has been designed to calculate transients and accidents initiated by reactivity or power changes. It solves thermal balance equations in the coolant and fuel. The coolant may be in either single or two-phase state. The current version of the programme has been developed by ANL-USA .

The code includes a selection of flow instability, Departure from Nucleate Boiling (DNB), single and two-phase heat transfer correlations, and a properties library applicable to the low pressures, temperatures, and flow rates encountered in research reactors. The PARET code provides a coupled thermal/hydraulic and point kinetics capability with continuous reactivity feedback, and a voiding model which estimates the voiding produced by sub-cooled nucleate boiling.

Fission power is calculated with a point-kinetics model, while feedback terms are included for coolant temperature, moderator density and fuel expansion effects.

The programme can be interrupted at any given point to plot the state of certain variables and it can be resumed afterwards.

The set of heat transfer correlations selected, suggested by ANL as those that best reproducing the SPERT experiences[2] are:

a) single phase heat transfer:	Dittus-Boelter
b) two phase heat transfer:	McAdams
c) Transient two-phase scheme	Transition Model
d) Subcooled region void fraction	Zuber
e) critical heat flux forced convection:	Tong
f) Heat transfer correlation in natural convection:	PARET [1]
g) CHF natural convection	Fabrega

These correlations are all applicable to the operating conditions and transients of the Reactor Facility.

The program has been extensively validated against results of the SPERT and CABRI series of experiments [2]

[1] PARET – A program for the analysis of reactor transients, C.F. Obenchaim. IDO-17282 (1969)

16.3.3.3.1.2 RELAP-5-mod 3.2

This is a code for transient thermal hydraulic analysis of complex fluid systems, developed at the Idaho National Engineering Laboratory (INEL) for the U.S. Nuclear Regulatory Commission (NRC).

The code is used in the analysis of all thermal hydraulic transients as well as for some reactivity transients. Even though RELAP has been developed for Nuclear Power Plants, it is widely used for analysis of research reactor transients, at low power and low temperature. The equations and correlations are valid for single phase flow at low power and low temperature. In transients where nucleate boiling can occur, the code can predict the Onset of Nucleate Boiling and it can accurately handle two-phase flow heat transfer for the operating and design bases accident conditions in the RRR.

For a description of RELAP and its features, see Chapter 5, Section 5.10.3.2.2.

16.3.3.3.2 Data Input and Modelling Assumptions

16.3.3.3.2.1 General

The calculations were performed for the reference core with the most conservative parameters for each transient. Parameters for the reference cores are presented in Chapter 5, Section 5.7. For the safety analysis calculations carried out using PARET, an uncertainty of 15% was applied to the values of the reactivity feedback coefficients. RELAP assumed no reactivity feedback. A six-group point kinetics model has been programmed in both PARET and RELAP-5. In the calculations presented in this SAR, photo neutrons from the heavy water reflector have been not considered.

It is assumed in all calculations that the nominal power generated by the core during normal operation is 20 MW. This value exceeds the actual core power (18.8 MW) and the assumption is conservative. The reactor trip 1 setting for high power (high neutron flux) is set at the analytical limit for reactor power, that is 26MW, for both RPS (higher than the 21.6 MW (Trip 1) and 22.6MW (Trip 2) actual trip settings, again a conservative assumption).

All the simulations assume single failure for the FSS. This means that successful actuation of the FSS implies the insertion of four out of five control plates. Failure of the FSS implies that no end-of-stroke signal has been received by the Reactor Protection Systems from two or more control plates within one second following the actuation of the FSS. In all numerical simulations, it has been assumed that no control plates are inserted when the FSS fails and the SSS is tripped, i.e. all 5 plates remain in position. This is equivalent to assuming failure of the FRPS to request actuation of the FSS. This very conservative assumption has been adopted in order to simplify the modelling. It bounds the behaviour of the reactor in the more realistic case when three control plates are inserted and the SSS is tripped after considerable negative reactivity has been inserted. The actuation time of the FSS has been taken as longer than that expected in practice (Section 5.7). This is another conservative assumption, adopted to allow margin in the insertion time.

The reactivity worth of the SSS as a function of time has been calculated from the experimental results obtained in a mock up of the SSS (described in Section 5.5) and calculations of the reactivity worth as a function of heavy water height. The mock up, and hence the reactivity insertion as a function of time, considers single failures. It has been

[2] W. Woodruff, “The PARET Code and the analysis of the SPERT I transients”, ANL/RERTR/TM-4, Conf-821155, September 1982

conservatively assumed that no reactivity insertion occurs prior to the full opening of the valves. The SRPS trips the FSS FAL simultaneously with the actuation of the SSS. This modification was implemented to ensure that the Control Rods stop during a withdrawal transient, and not because the SSS is not capable of stopping all postulated DBIE.

A cosine shaped power distribution has been adopted, as described in Section 5.7. The total Power Peaking Factor for the hot channel is 3 while that for the average channel is 1.311.

From the safety point of view the following conditions have been considered for the calculations:

a) A shutdown margin of –3000 pcm in subcritical condition with all CRs inserted

b) A shutdown margin of –1000 pcm in subcritical condition considering single failure

c) A safety margin of 20% in the reactivity insertion calculations

d) Failure of a CR to fall into the core (compliance with Single Failure Criterion)

16.3.3.3.2.2 Safety Systems Trip Parameters

Trip signals are generated in the protection systems for all the transients analysed in this chapter. Wherever possible, the second acting signal has been considered. This is a conservative assumption

16.3.3.3.2.3 PARET-PC

For the reactivity insertion transients analysed with PARET, 20% uncertainty has been assumed in the reactivity worth of the control plates for positive reactivity insertions caused by the extraction of a control plate and for the worth of the control plates as part of the FSS.

The CR fall speed is not constant. Since the input for PARET requires a constant speed, an average velocity has been adopted. The variation in speed is introduced by relating the reactivity insertion and rod position according to a real time dependent insertion speed.

16.3.3.3.2.4 RELAP-5

The RELAP code has been used in this SAR to analyse the subsequent transient following a DBIE involving the cooling systems. A computational model of the PCS has been constructed. The piping of the PCS and the pool volume were divided into nodes with similar dynamic characteristics. The core was divided into channels with axial nodes. The hot channel represents the hottest cooling channel inside the core. This corresponds to a cooling channel with maximum heat flux, while the average channel represents the rest of the cooling channels. The flap valves in the PCS were included. The flap valves open according to the flow rate in the PCS piping. The decay tank, PCS pumps and heat exchangers were modelled together with the pump recirculation and the interconnection lines (see Chapter 6 for a description of the PCS). The nuclear behaviour of the core was modelled through six group point kinetics. No credit was given to the fuel and coolant temperature negative feedback coefficients or the negative void feedback coefficient (see Chapter 5) for the analyses reported here. This assumption is very conservative, since the negative feedback coefficients insert negative reactivity during the evolution of the transient when the temperature increases.

This system is simpler than the PCS. To ensure consistency in the data input to the different transient simulations, a single data entry spreadsheet was used. This data sheet includes the geometrical information of the nodalisation as well as the decay heat curve for the reactor after shutdown, actuation time for the safety systems, friction factors for the different components and liquid properties at the operating conditions. Uncertainty values were determined for the input parameters. The input and the uncertainties are detailed in the specific sections presenting the analysis of each case.

16.3.3.4 Radiological Consequence Analysis

The design philosophy is that there shall be no core damage for design basis accidents, although fission product release may occur for beyond design basis accidents. In these cases, the radiological impact has been calculated using ORIGEN and PC-COSYMA[3]. ORIGEN[4] calculates the source term as a function of the radionuclides contained in an active element. PC COSYMA is used to calculate the dose to a given population given the source term and prevailing atmospheric conditions. Release fractions and retention factors in water (for those cases involving underwater release) of fission products have been obtained from open literature.

End of Section

[3] PC COSYMA Version 2.0 USER GUIDE National Radiological Protection Board Forschungszentrum Karlsruhe EUR 16240 EN (NRPB-SR280) ISBN no 92-87-4480-9 NOVEMBER 1995

[4] ORIGEN 2.0: Isotope generation and depletion code Matrix exponential method, ORNL-RSIC-CCC 371, 1980

16.4 REACTOR CHARACTERISTICS

16.4.1 Core Parameters

General core parameters are presented in Chapter 5, Sections 5.3, 5.7 and 5.8. Specific aspects relevant to the sequences associated with each DBIE are discussed in the relevant section below.

16.4.2 Functions of the Reactor Protection System

The Reactor Protection Systems (RPS) are capable of automatically triggering the required protective actions for the full range of DBIEs to terminate the event safely. This capability takes into account the possible malfunction of parts of the systems (single failure).

There are two independent RPS for the management of safety systems:

a) The First Reactor Protection System (FRPS), initiates the protective action of the FSS and the Containment system Isolation System, and

b) The Second Reactor Protection System (SRPS), which handles the trip signals for the SSS.

Chapter 8 describes the RPS and safety system settings are discussed in Chapter 17.

16.4.3 Natural Circulation Cooling of the Core and Irradiation Rigs

The Reactor Facility is designed such that, on cessation of pumped flow, flap valves open in the PCS and RSPCS and permit continued core cooling via natural circulation. The heat from the core and irradiation rigs is transferred to the pool water. Under normal operation, the heat deposited in the pool is transferred to the SCS via the RSPCS. The RSPCS can cool the pool either in cooling mode or in long term pool cooling mode. In case the SCS is not available, the large amount of water in the pool acts as a heat sink with heat being lost to the atmosphere via evaporation. There is sufficient water in the pool to permit core cooling for 10 days before the pool water reaches the actuation level for the EMWS.

End of Section

16.5 SAFETY CHARACTERISTICS OF THE DESIGN, OPERATION AND PLANNING ISSUES

A wide range of conservative provisions are in place within the framework of the defence in-depth philosophy. These provisions are aimed at minimising deviations from normal operating conditions, including transient conditions, and ensuring the continued confinement of radioactive material. Safety considerations are implemented through site selection, design, manufacturing, construction, commissioning, operating, and maintenance requirements such as:

a) Adoption of inherently safe features in the design.

b) Proven design.

c) Fail safe design.

d) Redundancy, independence and diversity in safety systems.

e) The clear definition of normal and off normal operating conditions.

f) Use of appropriate margins in the design of systems and plant components, including robustness and resistance to accident conditions.

g) Provision of appropriate time for operators to respond to events and adequate human-machine interfaces, including operator aids, to reduce the burden on the operators.

h) Careful selection of materials and use of qualified manufacturing processes and proven technology, together with extensive testing.

i) Comprehensive training of adequately selected operating personnel whose behaviour is consistent with a sound safety culture.

j) Precise operating instructions and reliable monitoring of plant status and operating conditions.

k) Recording, evaluation, and use of operating experience.

l) Comprehensive preventive maintenance prioritised in accordance with the safety significance and reliability requirements of the systems.

m) Comprehensive Quality Assurance (QA) programme (see Chapter 18).

End of Section

16.6 CLASSIFICATION OF INITIATING EVENTS

Of Postulated Initiating Events in section 16.3.1, the following DBA groupings are discussed:

Loss of Normal Power

Reactivity Insertion Accidents

Loss of Flow Accidents

Loss of Heat Sink

Loss of Coolant Accidents

Loss of Heavy Water

Erroneous Handling or Failure of Equipment or Components

Events Arising from the Reactor Utilisation

In addition, the following postulated events are evaluated for completeness:

Internal Events

Events Due to the Spurious Trigger of the Safety Systems

External Events

Human Factors

Each of these initiating events groups have been assessed to evaluate the defence in depth barriers provided by the facility's design to cope with them.

End of Section

16.7 ANALYSIS OF THE LOSS OF ELECTRIC POWER

16.7.1 Introduction

This Section examines the loss of electric power and its consequences for the facility. Chapter 9 describes the Normal Power System and the Standby Power System which provide electric power to the RRR Facility. These systems provide diverse reliable sources of power that are physically and electrically isolated, so that any single failure affects only one source of supply and does not propagate to other sources. Power is provided from the following sources:

a) Normal Power: Two separated off-site supplies.

b) Standby Power: Two Safety Category 1 Diesel Generators.

c) Uninterruptible Power Supply: Battery supply units within the Standby Power System.

All Engineered Safety Features' instrumentation and electrical equipment is connected to the Standby Power System. The Containment Energy Removal Systems are by far the largest loads of all the Engineered Safety Features, other Engineered Safety Features are supplied through Uninterruptible Power Supply units. However, they are only called upon to supply until the Standby Power System Diesel Generators are available. The Long Term Pool Cooling pumps of the RSPCS and SCS are supplied from the Standby Power System; all other cooling circuit pumps are supplied by the Normal Power System.

Loss of Normal Power Supply results in the de-energising of the CR electromagnets and uncoupling of the CRD motors. In addition, the FSS compressed air valves also fail open on loss of Normal Power Supply. These two events allow the control plates to fall into the core. Nevertheless, loss of Normal Power Supply initiates a reactor trip by the FRPS, enabling the SRPS trip. If the SRPS then detects a failure of two or more CRs to fully insert within a preset time following initiation of the FSS, actuation of the SSS occurs.

16.7.2 Loss of Normal Power Supply

On the de-coupling of the CRD motors or de-energising of the CR electromagnets, the CRs fall into the core with compressed air assistance provided by fail-open valves.

Loss of Normal Power Supply also initiates a reactor trip by the FRPS. If the SRPS detects a failure of two or more CRs to fully insert within a preset time following initiation of the FSS, actuation of the SSS occurs.

Loss of Normal Power is an anticipated operational occurrence rather than a Postulated Initiating Event. It is included for completeness. It has no consequence due to the inherent design provisions and will not be considered further.

16.7.2.1 Defence in Depth Barriers

The following table summarises the first two levels of protection related to this PIE consistent with the defence in-depth philosophy:

Level	Main Characteristics	Safety Features
1	Conservative design	Passive natural convection cooling systems

Level	Main Characteristics	Safety Features
	and Inherent safety features	Design of the electric system according to IEEE and AS standards, and Energy Australia Service and Installation Rules and General Supply Conditions.
		Equipment separation to avoid fire and other damage to several items from one event.
		Separated earthing systems. The power and signal earthing systems are installed as separate systems. The lightning protection system is separate from the other earthing systems.
		Fire resistant and low smoke/Halogen-free electric cables.
		Special electric cables for radioactive areas.
		Appropriate electrical system maintenance program
		Safety Category 1 UPS serving the RPS and the PAM System designed according to the criteria for Class 1E systems as defined in IEEE standards.
		Safety Category 1 Standby Power System designed according to criteria for Class 1E systems as defined in IEEE standards.
2	Operation control and response to irregular operation	RCMS initiated power reduction
		Loss of electric power directly causes FSS actuation.

16.7.3 Loss of all Normal Power and Diesel Generators

On loss of the Normal Power System, the diesels of the Standby Power System will become available. The diesels are Safety Category 1. Only one of the two is needed. The Uninterruptible Power Supply supplies power to all necessary equipment for a period after the loss of Normal Power. This is considered within the design basis. The likelihood of a loss of Normal Power Supply together with the inability to start and run both diesels is considered so low as to render it beyond the design basis.

16.7.3.1 Defence in Depth Barriers

The following table summarises the first three levels of protection related to this PIE consistent with the defence in-depth philosophy:

Level	Main Characteristics	Safety Features
1	Conservative design and Inherent safety features	Passive Natural Convection Cooling Systems
		Design of the electric system according to IEEE and AS standards, and Energy Australia Service and Installation Rules and General Supply Conditions.
		Safety Category 1 Standby Power System design according to criteria for Class 1E systems as defined in IEEE standards.
		Equipment separation to avoid fire and other damage to several items from one event.
		Separated earthing systems. The power and signal earthing systems are installed as separate systems. The lightning protection system is separate from the other earthing systems.
		Fire resistant and low smoke/Halogen-free electric cables.

Level	Main Characteristics	Safety Features
		Special electric cables for radioactive areas.
		Adequate electrical system maintenance program
		Electrical and electronic equipment such as Diesel Generators' Control and UPS units radiation and seismically qualified.
2	Operation control and response to irregular operation	RCMS initiated power reduction
		Loss of electric power directly causes FSS actuation.
		Fail safe characteristics of FSS and SSS
3	Control of accidents within the design basis	FRPS trips the reactor on: a) Loss of Normal Power b) Low PCS flow c) Low RSPCS flow SRPS trips the reactor on: a) Core pressure difference low b) Failure of the FSS

16.7.4 Loss of all Normal Power and all Standby Power for a period up to 10 days

As indicated above, on loss of the Normal Power System, the diesels of the Standby Power System will available within about 60 seconds. The diesels are Safety Category 1. Only one of the two is needed. The likelihood of a loss of Normal Power Supply together with the inability to start and run both diesels is considered so low as to render it beyond the design basis.

Failure to restore power supply would result in the unavailability of the RCMS; RPS and PAM. Nevertheless, the reactor would be in a safe shutdown mode and cooling by natural circulation will remove the decay heat from the core.

The likelihood of failing to restore power for 10 days is even more remote and even further outside the design basis. It is discussed here, however, as the sequence will be analysed as a Beyond Design Basis Accident in Section 16.19 as part of demonstrating the robust nature of the Reactor Facility and its ability to cope with an extended period of natural circulation.

16.7.5 Design Basis Postulated Initiating Events

From the discussion in the previous sections of this document, some of the events do not lead to accidental conditions and as such, need not be considered further as Initiating Events. Only those that merit further analysis are considered.

A summary of previous considerations and the design-basis PIE are presented below:

PIE	Not applicable to the design	Eliminated by inherent design provisions	Sufficiently unlikely to occur (BDB)		Design Basis Initiating Events (DBIEs)		
					To be considered in another DBIE group	Bounded by another DBIE	Further Analysis
Loss of Normal Power Supply		X					
Loss of Normal Power and diesels not started							X
Loss of all Normal and all Standby Power for up to 10 days			X				

On the basis of the above, one DBIEs is identified for analysis in this section:

Loss of Normal Power and diesels not started.

This event is conservatively assumed to occur with the reactor at full power.

16.7.5.1 Detection of the Initiating Event

Loss of Normal Power initiates a reactor shutdown by the FSS. This protective action is not dependant on the operation of the FRPS since, on de-coupling of the CRD motors or de-energising of the CR electromagnets, the CRs fall into the core. In addition, the FRPS sends a signal to request the actuation of the FSS. Furthermore, if the SRPS detects a failure of two or more CRs to fully insert within a preset time after initiation of the FSS, actuation of the SSS occurs.

16.7.5.2 Design Basis Fault Sequence

A description follows of the events that are an immediate and direct effect of the initiating event.

a) Interruption of Normal Power.

b) UPS units provide electric supply from batteries to the RCMS, the RPS, and the Post Accident Monitoring system.

c) RPS produce alarms and warnings in the Main Control Room.

d) Loss of Normal Power Supply to the FSS will de-energise the electromagnets and the motors, the pistons will de-couple from the control rod drive mechanism, and the CRs will fall into the core by gravity forces.

e) The FRPS sends a signal to actuate the FSS and thus enable the SRPS's waiting period for the end-of-run signals from the CRs.

f) If the SRPS detects a failure of two or more CRs to fully insert within a preset time after initiation of the FSS actuation of the SSS is initiated

g) The PCS and RSPCS pumps coast down.

h) PCS and RSPCS flap valves open and natural circulation is established.

i) Decay heat from the core and the irradiation rigs is removed by natural circulation of the water contained in the Reactor Pool.

j) At least one Diesel Generator is started.

In addition, the RPS would produce alarms and warnings in the MCR. The Loss of Normal Power Supply to the FSS would de-energise the electromagnets and the motors, the pistons would de-couple from the control rod drive mechanism, and the CRs would fall into the core by gravity. These are expected occurrences but are not credited.

16.7.5.3 Numerical Analysis

The behaviour of the PCS and the RSPCS has been analysed after a loss of Normal Power Supply. Calculations have been made with the RELAP 5 code. The nodalisation used for the PCS and the RSPCS is presented in Section 16.3.

16.7.5.3.1 Modelling Assumptions

To calculate the increase in the pool water temperature, only the volume comprised between the top of the chimney and the bottom of the transfer channel has been considered. This volume is relatively small compared to the total volume of water in the pools. Thus, it represents a conservative assumption.

For the PCS calculations, the temperature of the coolant corresponds to the uppermost node in the hot channel. The cladding temperatures correspond to the fuel node in that same channel.

The loss of power results in a loss of heat removal from the PCS via the heat exchangers (due to the shutdown of the SCS pumps) and the loss of flow in the PCS and RSPCS. The RSPCS and the PCS pumps stop and the flow coasts down according to the dynamics of the pumps.

The loss of power has been simulated numerically with successful actuation of the FSS and with failure of the FSS and actuation of the SSS. The reactor trip is triggered by a low flow signal in the PCS or RSPCS. This is a very conservative assumption, since the fail safe characteristics of the FSS are being ignored to introduce the delay. In reality, the actuation of the FSS is instantaneous and immediately follows the loss of Normal Power.

The RSPCS and the PCS have been treated separately, with connections through the pool water and the interconnection line flow. For the RSPCS calculations, the pool temperature and the interconnection flow rate come from the PCS calculations results.

16.7.5.3.2 Primary Cooling System

Variation of reactor power with time occurs following the loss of Normal Power Supply. On loss of Normal Power Supply the speed of the pumps decreases in accordance with their dynamics. The switch between states occurs simultaneously with the loss of power.

The flow rate through the core diminishes until the PCS flap valves open and natural circulation is established. From this instant, there is a slow reduction in the flow rate caused by the decreasing decay heat being generated in the core.

The temperature of the coolant at the exit from the core undergoes first an increase until the reactor is shutdown by the FSS. After reactor shutdown, it decreases rapidly and temporarily because the decrease of the heat dissipated in the core is faster than the increase of the inlet temperature due to the loss of heat exchangers. After this decrease, the reduction of flow rate through the core causes an increase in the temperature at the core outlet. Then, the core inlet and outlet temperature increases slowly following the increase of pool temperature.

Once the flap valves open, the natural convection flow rate that is established adequately cools the core. The temperature of the cladding increases sharply due to the interruption of coolant flow through the core. The cladding temperature reaches a maximum value almost simultaneously with the reactor trip. This increase is followed by a decrease of these temperatures caused by the decay in heat dissipated at the core. The temperatures start to increase due the inlet of hot water to the core. The opening of the flap valves leads to an increase in the flow rate through the core and a decrease in the temperatures. In the long term, the temperature evolution follows the variation of temperature of the pool water.

There is no need for the operator to initiate the Long Term Pool Cooling mode of the RSPCS in the first 30 minutes after the loss of power, since the temperature of the pool is still below the initial temperature of the hot water layer.

The transient has been analysed assuming failure of the FSS and reactor shutdown by the SSS. The evolution of the transient is similar to that considered previously. A delay in temperature decrease is observed due to the later actuation of the SSS. There is a slower decrease in the temperature after reactor trip due to the slower decrease in reactor power. There is a higher maximum temperature for the coolant at the core outlet due to the delay in actuation of the SSS. Similarly, the temperature peak for the cladding in the hot channel is above the maximum temperature with actuation of the FSS.

16.7.5.3.3 Reactor and Service Pools Cooling System

The effect of the loss of Normal power Supply has been analysed with actuation of the FSS and with failure of the FSS and actuation of the SSS. Flow reversal from downwards forced convection to upwards natural convection occurs after the pumps coastdown ceases. Soon afterwards, the flow stabilises. For the first few seconds after the loss of power, the pool temperature remains steady. After the reactor trip, the reduction in reactor power is very fast and results in a drop in the temperature of the suction plenum. The change in behaviour corresponds to onset of natural circulation. The temperature of the pool water rises very slowly, whereas the heat generated by the drops abruptly. The temperature at the pool outlet also decreases rapidly. In the long term this temperature increases, following the rise in the pool water temperature. The temperature sharply increases following flow interruption due to the delay in FSS trip introduced by ignoring the FSS fail safe characteristics under loss of power and the loss of power trip signal. Then the temperature decreases abruptly after the reactor trip and it increases again after the opening of the flap valve and the flow reverses. In the long term, the natural circulation ensures cooling. The behaviour of the RSPCS after loss of power with failure of the FSS and successful actuation of the SSS has also been analysed. The FSS is tripped by the loss of power signal and the SSS trips on failure of the FSS. Radiological Impact Analysis

From the results of the numerical analyses, no damage is expected to either the core or the rigs. Therefore, no radiological impact is expected arising from the loss of power.

16.7.6 Conclusions

Loss of Normal Power Supply is an Anticipated Operational Occurrence. The bounding event of this DBA grouping involves a prolonged loss of electric power concurrent with a failure of both standby diesel generators. Either RPS is capable of shutting down the reactor. The availability of UPS power is not relevant to this bounding DBA. Heat removal is adequate. It is concluded that nuclear safety is guaranteed for all credible events involving a Loss of Electric Power. The effect of an extended loss of all electric power is considered in the section dealing with Beyond Design Basis Accidents, Section 16.19.

End of Section

16.8 ANALYSIS OF EXCESS REACTIVITY INSERTION EVENTS

16.8.1 Introduction

This Section analyses the group of accidents caused by uncontrolled reactivity insertions into the reactor core.

Uncontrolled reactivity insertions would lead to a sequence of alarms, warning interlocks, and control actions, with the RPS triggering one or both of the two independent shutdown systems on either high neutron flux rate or high neutron flux. For slow reactivity insertions, high core outlet temperature and high core temperature difference are also able to trigger alarms and produce reactor shutdown.

The initiating events for reactivity insertion incidents have been grouped according to their cause. Each initiating event will be analysed in the context of defence in depth barriers. The response of the facility to the postulated DBIE will then be quantified.

16.8.2 Insertion of Fissile Material

16.8.2.1 Accidental Drop of a Fuel Assembly

This initiating event refers to:

a) The mishandling and drop of a fuel assembly during core loading operations.

   No fuel loading will be performed with the reactor in operation. All core configurations during core operation (initial, intermediate and final configurations) with all the CRs fully inserted are subcritical. Considering an extreme bounding case, i.e. the fall of a fuel assembly without burnable poison (an extremely conservative assumption since this type of fuel assembly will not be used in the reactor), the reactivity insertion would be a small fraction of the shutdown margin.

   Notwithstanding the above arguments, the dropping of a fuel assembly is considered within the design basis. The reactivity insertion is bounded by the inadvertent withdrawal of a pneumatic can with excess absorbing material (Section 16.8.3.6). The potential for physical damage to the dropped fuel assembly is considered in Section 16.13.

b) The mishandling and drop of a spent fuel assembly stored in the Reactor Pool during transfer to the Service Pool.

   The protective grille on top of the chimney will withstand the impact of a can containing silicon ingots. It will prevent a dropped fuel assembly from affecting the core. The Control Rod Guide Box acts as an additional barrier.

   The same arguments as presented above apply. The dropping of a fuel assembly is considered within the design basis. The reactivity insertion is bounded by the inadvertent withdrawal of a pneumatic can with excess absorbing material (Section 16.8.3.6). The physical damage to the dropped fuel assembly is considered in Section 16.13.

c) Inadvertent ejection and reinsertion of a fuel assembly in the core grid during reactor operation.

   Fuel assembly ejection is considered extremely unlikely. The clamp is designed to withstand an upward force much greater than the drag force exerted by the maximum PCS flow under low temperature conditions. There are no abnormal situations where the fuel assembly could be submitted to a greater force. It will not be considered further.

16.8.2.1.1 Defence in Depth Barriers

Level	Main Characteristics	Safety Feature
1	Conservative design and Inherent safety features	Proven design of tools and clamps
		Large design margin in mechanical components
		Operational experience
		Refuelling with FSS at maximum negative reactivity worth
		Large shutdown margins (FSS and SSS)
		Adequate fuel management programme: knowledge of core reactivity
2	Operation control and Response to abnormal operation	Alarm on : high neutron flux

16.8.2.2 Inadvertent Fast Insertion of Irradiation Fissile Material

This event refers to the inadvertent fast insertion of a uranium-molybdenum (U-Mo) irradiation rig, with the reactor at full power. Chapter 11 provides a description of the U-Mo irradiation rigs and their handling.

Operating procedures will ensure adequate handling of U-Mo rigs. The QA procedures in the manufacturing of the U-Mo rigs will ensure that the correct amount of fissile material is placed in the irradiation rigs.

Given the reactivity worth of these rigs, simulations will be carried out of an inadvertent insertion. Section 16.8.7 presents the assumptions and the results of the analysis.

A further postulated event involving U-Mo rigs is the insertion of this fissile material into an irradiation position intended for iridium irradiation. Uranium is a fissile material, as opposed to iridium, a neutron absorber. Thus, inadvertently placing a U-Mo rig in an iridium irradiation position would insert significant positive reactivity to the core. To avoid this type of occurrence, the irradiation positions are designed with a different geometry that will not allow interchange between the two different types of rigs (see Chapter 11). This event has therefore been designed out.

Inadvertent fast insertion of irradiation fissile material into its irradiation position with the reactor at full power is considered to be within the design basis and will be analysed further. The reactivity worth of this insertion will be based on the proposed design of the U-Mo irradiation target. The ramp is calculated considering a fast manual insertion of the rig.

16.8.2.2.1 Defence in Depth Barriers

Level	Main Characteristics	Safety Feature
1	Conservative design and Inherent safety features	Limit on maximum reactivity worth allowed for movable uses.
		Operational experience
		Adequate rigs management programme

Level	Main Characteristics	Safety Feature
		Different geometry of U-Mo and iridium irradiation rigs with physical interlocks.
		QA system for the manufacturing and loading of targets.
2	Operation control and Response to abnormal operation	Alarm on: a) high neutron flux b) low reactor period
3	Control of accidents within the design basis	FRPS reactor trip on: a) High neutron flux rate b) High neutron flux SRPS reactor trip on: a) High neutron flux rate b) High neutron flux c) Failure of the FSS

16.8.3 Absorber Withdrawal

16.8.3.1 Start-up Accident

This event considers the continuous withdrawal of a single Control Rod due to motor or controller failure during reactor start-up, at low power and with full flow rate and at low power and natural circulation. It is postulated that the reactor is almost critical and one of the CRs is withdrawn continuously.

The withdrawal of a CR inserts reactivity. The extraction of a CR at a high speed is prevented by the design of the Control Rod Movement Protection Interlock (CRMPI, Safety Category 1). The design of the CR and the CRD motor is based on the design of the ETRR-2. Nevertheless, a prototype has been built and the performance of the CR and the CRD motor has been evaluated. The design and testing of the CRD is directed towards ensuring its reliable performance and minimising the potential for its failure.

Inadvertent continuous withdrawal of a CR at the design withdrawal velocity during start up is considered to be within the design basis and will be considered further. The event will be simulated with actuation of the FSS, and with failure of the FSS and actuation of the SSS. This is the bounding reactivity insertion DBIE. The calculation assuming actuation of the SSS acts as the verification of the design actuation time for the SSS. . Defence In Depth Barriers

Level	Main Characteristics	Safety Feature
1	Conservative design and Inherent safety features	Limitation in the withdrawal velocity of control rods (CRMPI)
		Operational experience with the CRD
		Neutron flux measurement from the source level up to power level
		Adequate start-up procedures

		CR position is not used as a variable for reactor control
		Reactivity compensation by addition of burnable poisons to fresh fuel.
		Automatic inhibition on rod withdrawal
2	Operation control and Response to abnormal operation	Alarm on: a) high neutron flux rate (low reactor period) b) high neutron flux (high power)
3	Control of accidents within the design basis	Safety functions independent of control function
		FRPS trips the reactor on: a) High neutron flux rate (low reactor period) b) high neutron flux (high power) SRPS trips the reactor on: a) high neutron flux rate b) high neutron flux c) Failure of the FSS

16.8.3.2 Inadvertent Control Rod Withdrawal during operation

This event refers to the inadvertent continuous withdrawal of a high reactivity worth CR with the reactor at full power and full flow rate.

All plates move inside the core at Beginning Of Cycle for a short period, to compensate for xenon build-up or otherwise on transient demand. For the rest of the cycle, only the central plate (of lowest reactivity worth) controls core reactivity, and the remaining four plates are almost fully out. The position of the control plates during operation implies that the insertion of positive reactivity produced by a continuous withdrawal of a plate will be limited. The movement of the CRDs is sequential, not simultaneous. Simultaneous withdrawal of all rods is not an RCMS action and is prevented by the CRMPI.

The design provisions that limit the velocity of extraction of the CRs has been described above.

The erroneous continuous withdrawal of a CR at the maximum allowed velocity is considered to be within the design basis and will be analysed further. The withdrawal could be caused by a malfunction of a CRD motor or Controller Unit. This DBIE will be analysed with actuation of the FSS, and with failure of the FSS and actuation of the SSS. Defence in Depth Barriers

Level	Main Characteristics	Safety Feature
1	Conservative design and Inherent safety features	Only the central plate moves during the entire operation cycle. The remaining four plates are required for a short period after start-up and are withdrawn for most of the operation cycle, ready to shut down the reactor

		Reactivity compensation through burnable poison in the fuel assembly.
		Proven design of CRD system and devices
		Limit in control rod withdrawal velocity (CRMPI)
		CR position is not used for power measurement
		Adequate QA inspection and maintenance programme
2	Operation control and Response to irregular operation	Alarm on: a) high neutron flux rate (low reactor period) b) high neutron flux . (high power)
		Automatic reactivity control system
3	Control of accidents within the design basis	Safety functions independent of control function
		FRPS trips the reactor on: a) high neutron flux rate (low reactor period) b) high neutron flux (high power) SRPS trips the reactor on: a) high neutron flux rate b) high neutron flux c) Failure of the FSS

16.8.3.3 Control Rod Drive or System Failure

This event refers to the potential for uncoupling and upward dragging by the flow of a control plate. The uncoupling could arise as a result of failure of the joint between the plate and the CRD stem or failure of the coupling between the stainless steel lower part of the rod.

The CRs move within their respective guide boxes. Control Rod Guide Boxes share a common inlet plenum with fuel assemblies and withstand the same core pressure drop. As a consequence of the hydraulic design of the guide boxes, almost all the pressure drop takes place at the guide box outlet and not over the plate. Thus, the force exerted on the control plate by the flow is minimised. Should one of the control plates mechanically uncouple from its supporting drive rod, it will not be dragged upwards.

The CRD assembly (Chapter 5) has been tested successfully for physical integrity when subjected to the upward drag force exerted by the PCS flow. A prototype will be built to test for the performance of the assembly during reactor lifetime, as described in the previous section.

The upward dragging of a control plate following its uncoupling from the drive rod is considered not possible as a result of the inherent characteristics of the design of the control plates, followers, drives and guide boxes. It will not be considered further.

In case an electromagnet loses electric power supply, the corresponding plate will fall into the core by gravity forces and it will not be dragged upwards. The RCMS will trigger an alarm in the Main Control Room, showing that an unexpected power drop has occurred. The fall of a control plate into the core implies a negative reactivity insertion;

no damage is caused to the core due to the drop in power. This event will not be considered further.

16.8.3.3.1 Defence in Depth Barriers

Level	Main Characteristics	Safety Feature
1	Conservative design and Inherent safety features	Proven design of CRD system and devices
		Control Rods drop due to gravity force
		Hydraulic design avoids upward drag forces on the control rod plates
		Adequate QA inspection and maintenance programme
2	Operation control and Response to irregular operation	Alarm on a) high neutron flux b) high neutron flux rate
		Automatic reactor control system

16.8.3.4 Inadvertent Control Rod Bank Extraction

There is no bank extraction mode in the RCMS logic. The CRD control logic protects against continuous withdrawal of more than one CR during reactor operation. This protection acts at two levels. A malfunction of the RCMS simultaneously with CRMPI failure that could lead to the extraction of more than one CR is sufficiently unlikely as to render it beyond the design basis. (Note that the RCMS has a bank insertion mode and the hardwired watchdogs do not prevent bank insertion. This event has been analysed at ARPANSA's request. The results are presented in Section 16.19.

16.8.3.5 Inadvertent Extraction of a Fixed Absorbing Irradiation Material

This event refers to the inadvertent withdrawal by an operator of an irradiation rig classified as fixed (i.e., loading and unloading are performed during reactor shutdown) while the reactor is at full power and full flow rate.

The extraction of absorbing irradiation material has two effects:

a) Insertion of positive reactivity.

b) Limited redistribution of the cooling flow through the irradiation rigs. The irradiation positions are designed to concentrate the pressure drop in a flow restriction at the exit of the cooling channel. Thus, removal of one or several of the irradiation rigs will cause a limited flow redistribution but no damage to other irradiation rigs arising from lack of cooling.

This event is considered unlikely. Handling of fixed irradiation rigs is neither necessary nor authorised during reactor operation. All handling of fixed rigs is performed during reactor shut down. Operating procedures will be in place to avoid this event. In addition, to further inhibit extraction of fixed irradiation rigs, locks are placed in each irradiation position, at the top of the Reflector Vessel. Tools to unlock and remove fixed irradiation rigs are kept under lock and the key is under the shift supervisor's control.

Even though the likelihood of this event is considered low, it will be assumed to lie within the design basis and will be analysed. The size of the rig is such that it is not possible to extract the rig quickly. Consequently, the rate of reactivity addition is bounded by that arising from inadvertent withdrawal of a pneumatic can. A very conservative assumption has been made regarding the worth of the reactivity insertion caused by the withdrawal of a fixed Iridium rig. This represents almost four times the design inserted reactivity worth. Defence in Depth Barriers

Level	Main Characteristics	Safety Feature
1	Conservative design and Inherent safety features	Administrative requirements on fixed experiments
		Locks in place at the top of the Reflector Vessel for fixed irradiation rigs
2	Operation control and Response to irregular operation	Alarm on: a) high neutron flux rate (low reactor period) b) high neutron (high power)
		Automatic reactor control system
3	Control of accidents within the design basis	FRPS trips the reactor on: a) high neutron flux rate (low reactor period) b) high neutron flux (high power) SRPS trips the reactor on: a) high neutron flux rate b) high neutron flux c) Failure of the FSS

16.8.3.6 Inadvertent Withdrawal of a Pneumatic Can with Excessive Absorbing Material

This event considers the inadvertent extraction of a can in a pneumatic facility containing excess absorbing material, with the reactor at full power and full flow rate. Operating procedures will be in place to ensure that the likelihood of operator error in inadvertently withdrawing a pneumatic can is minimised.

Inadvertent withdrawal of a pneumatic can is considered to be within the design basis and will be analysed further. The worth of this reactivity insertion is small. The rate of insertion is high due to the speed of movement in the pneumatic channels. The analysis considers a QA violation in the preparation of the can such that it results in an excess of absorbing material inside the can. Therefore, the extraction of the can will lead to a reactivity insertion higher than the reactivity inserted due to the normal operation of the pneumatic conveyor system.

Inadvertent simultaneous extraction of two or more irradiation cans is inhibited by a hardwired mechanism. Thus, only one tube is enabled to insert o remove cans at a time. This limiting system has a built in time delay that separates the ejection and insertion of cans to prevent the fast successive extraction or insertion of irradiation cans.

The event has been simulated with the successful actuation of the FSS and with failure of the FSS and actuation of the SSS. Defence in Depth Barriers

Level	Main Characteristics	Safety Feature
1	Conservative design and	Limit of reactivity worth for movable irradiation targets

Level	Main Characteristics	Safety Feature
	Inherent safety features	Operation procedures for handling of irradiation material and preparation of targets
		Targets loads prepared in accordance with QA system
2	Operation control and Response to irregular operation	Alarm on: a) high neutron flux rate (low reactor period) b) high neutron (high power)
		Automatic reactor control system
3	Control of accidents within the design basis	FRPS trips the reactor on: a) high neutron flux rate (low reactor period) b) high neutron flux (high power) SRPS trips the reactor on: a) high neutron flux rate b) high neutron flux c) Failure of the FSS

16.8.4 Increase in Moderation Effect

16.8.4.1 Cold Water Injection

This event refers to cold water injection into the core due to:

a) The spurious initiation of the Emergency Make-up Water System at zero power and zero flow rate.

   The design of the Emergency Make-up Water System makes spurious initiation unlikely. The opening of float valves activates the system. These float valves open when the level of the pool water reaches the top of the chimney. No triggering signal is involved.

   The float valves and piping are encased in a protective structure to avoid erroneous manipulation of the system and accidental actuation. The operator can actuate the float valve to test the system but cannot inadvertently initiate the system while manipulating tools or rigs inside the pool. During normal operation the Emergency Make-up Water System cannot inject water into the core as the pressure inside the PCS piping is higher than the pressure in the Emergency Make-up Water System. Thus, no injection is possible with the reactor at full flow.

   The spurious initiation of the Emergency Make-up Water System at zero power and zero flow rate is considered to lie within the design basis. Its consequences are minor, bounded by the inadvertent withdrawal of a CR at start-up.

b) The insertion of cold water due to start up of the PCS pumps during Physics Test operation.

   A protection interlock prevents the start up of the PCS pumps when the reactor is in Physics Test State. The insertion of a cold slug is therefore unlikely. This event is considered to be beyond the design basis.

d) The insertion of cold water through the PCS caused by variation in the temperature of the heat sink.

The inlet temperature to the core is monitored. A variation of conditions in the heat sink will start control actions to ensure a constant temperature at the inlet to the core. It is highly unlikely that the variation in the temperature of the heat sink could be sufficiently rapid or large enough to significantly affect reactivity. Variations in the temperature of the heat sink are considered to be within the design basis. The reactivity insertion arising from the change in temperature of the heat sink flow is bounded by the CR withdrawal start-up accident

16.8.5 Increase of Reflector Effect

16.8.5.1 Inadvertent Refill of Reflector Tank

This event considers the inadvertent refill of the Reflector Vessel due to a start up logic failure with the reactor at decay power and zero flow rate. The speed of refilling of the moderator tank is limited by the capacity of the pumps of the Reflector Cooling & Purification System. Reactor Protection System interlocks are provided on the reflector refill system, so that start up of these pumps is only possible under controlled conditions during reactor start up. Therefore the insertion of reactivity is very slow. After actuation of the SSS, the Reflector Vessel can only be refilled with all CRs inserted to absorb the reactivity worth of the SSS, the RPS functioning and all necessary nuclear instrumentation in operation. The multiple failures required in highly reliable systems render the likelihood of this event sufficiently low as to place it beyond the design basis. It will not be considered further.

16.8.6 Fast Reactivity Insertion Accidents

A fast reactivity insertion has the potential to lead to the release of mechanical energy into the water and the structures within the pool[1].

The best known and most analysed fast reactivity transients are the BORAX accident[2] and the SPERT and CABRI experiments[3] [4]. Recent reports conclude that the event sequence in the BORAX accident is incompatible with modern pool type reactor design[5].

The BORAX ‘accident’ was an experiment specifically designed to insert large amounts of reactivity in a very short time span. To accomplish this objective one of the absorber plates was specially modified to be withdrawn at high velocity. The withdrawn position of this plate was below the core. The plate fell by gravity forces aided by a large spring.

[1] Abou Yehia, H. , Berry, J.L. and Sinda, T., “Prise en Compte d’un Accident de Réactivité dans le Dimensionnement des Réacteurs de Recherche”, Colloque International sur la Sûreté, l’Exploitation et la Modification des Réacteurs de Recherche”, IAEA-SM-310/107, Chalk River, Ontario, Canada, 23-27 Octobre 1989.

[2] Thompson, T.J. and Beckerley, J.G. Eds., “The Technology of Nuclear Reactor Safety”, MIT Press, Massachusetts, U.S.A., 1964.

[3] W. Woodruff, “The PARET and the Analysis of the SPERT I Transients”, ANL/RERTR/TM-4, Cnf-821155, September 1982.

[4] F. Merchie, “Presentation bibliographique des résultats obtenus à CABRI dans le domaine de la sécurité des réacteur à eau légère”, CEA-CENG, Pi®710-87/67.

[5] C. Hickman, J.L. Minguet and F: Arnould, “Compilation of the Replies Given to the Technicatome Questionnaire ‘Comparison of Regulations for Research Reactors’ sent out to IGORR Members in 1997”, Technicatome, Établissement de Saclay: Centre d’Études de Saclay, France.

The absorber was configured to be ejected from the core in 0.2 s. This resulted in an insertion of 4,000 pcm with a ramp of 20,000 $pcms^{-1}$.

There is no physical mechanism to allow such a fast reactivity insertion in the core of the RRR. The extraction speed limit of the CRD motor will result in the removal of a CR in 200 s. This speed is well below the speed necessary to insert into the core the whole worth of positive reactivity associated with the withdrawal of a control plate. The design requirement on the velocity of the coolant inside the guide box is a maximum value of about 60% of the design margin to avoid the dragging upwards of the control plate. At this maximum speed, even if the connection between the plate and the driving rod were severed, the control plate would not be dragged upwards and it would remain inside the guide box, 75% of its length inserted inside the core. The design of the guide box prevents the sudden ejection of the control plate.

Inadvertent manual withdrawal is not credible because a specific tool is required to lift an absorber plate, and this tool must be fitted down into the guide box. Furthermore, the operator would have to lift the whole piston and cylinder assembly and rotate the stepper motor. Maintenance tasks requiring the removal of the absorber plates are not common and by administrative procedures require removal of all the fuel in the core to the Reactor Pool storage racks. To carry out the plate removal, an operator must enter the CRD Room and de-couple the absorber plate. Special permission is required to enter this room.

There is no potential for explosions inside the CRD Room that could generate pressure waves and cause the control absorber plates to be lifted up rapidly. In case of an increase of pressure of the compressed air for the pneumatic system in the FSS, the hoses in the pneumatic system will rupture and there will be no build up of energy that could be suddenly released.

There is no bank extraction mode in the RCMS. The extraction of CRs, and subsequent insertion of reactivity, is sequential and not simultaneous. In addition, a hardware system, the CRMPI, inhibits movement of more than a CR at a time. This impediment for bank extraction removes another mechanism for a large insertion of reactivity into the core.

This event is not considered applicable to the design of the RRR Facility. It will therefore not be considered as a DBIE and will not be analysed further.

16.8.7 Design Basis Postulated Initiating Events

A summary of previous considerations and the identification of the design basis initiating events are presented in the following table:

PIE	Not applicable to the design	Eliminated by inherent design provisions	Sufficiently unlikely to occur (BDB)		Design Basis Initiating Events (DBIEs)		
					To be considered in other DBIE group	Bounded by another DBIE	Further Analysis

PIE	Not applicable to the design	Eliminated by inherent design provisions	Sufficiently unlikely to occur (BDB)	Design Basis Initiating Events (DBIEs)		
				To be considered in other DBIE group	Bounded by another DBIE	Further Analysis
Reactivity insertion following accidental drop of a fuel assembly during core loading					X (Inadvertent extraction from pneumatic channel)	
Reactivity insertion following accidental drop of fuel assembly during transfer from Reactor to Service Pool					X (Inadvertent extraction from pneumatic channel)	
Fuel assembly ejection and reinsertion			X			
Inadvertent Insertion of irradiation fissile material						X
Start-up accident						X
Inadvertent CR withdrawal at power						X
CR or system failure (CR ejection)		X				
Inadvertent control bank extraction			X			
Inadvertent extraction of fixed irradiation target						X
Inadvertent extraction of a pneumatic channel can with excess absorbent material						X
Spurious initiation of the EMWS			X			

PIE	Not applicable to the design	Eliminated by inherent design provisions	Sufficiently unlikely to occur (BDB)	Design Basis Initiating Events (DBIEs)		
				To be considered in other DBIE group	Bounded by another DBIE	Further Analysis
Start up of PCS in Physics Test operation			X			
Variation on PCS temperature due to fluctuations in temperature of the heat sink					X (start up accident)	
Inadvertent refill of the Reflector Vessel			X			
Fast large reactivity insertion	X					

Design Basis PIE:

Two events will be analysed in relation to inadvertent CR withdrawal:

a) Inadvertent withdrawal during core start-up. The core is critical, in cold state and one of the safety absorbers is withdrawn at its maximum velocity .

b) Inadvertent withdrawal with core at full power operation. It is assumed that the most effective control rod is withdrawn at its maximum velocity .

In both events, a safety factor of 20% will be adopted for the reactivity worth of the rod withdrawn.

Two events will be analysed in relation to inadvertent removal or insertion of irradiation targets:

a) Inadvertent insertion of a U-Mo irradiation target.

b) Inadvertent removal of a can with excess irradiation material from a pneumatic irradiation channel

c) Inadvertent withdrawal of a fixed Iridium rig during reactor operation in Physics Test

16.8.7.1 Detection of the Initiating Event

The RCMS responds to the uncontrolled reactivity insertion with alarms. The FRPS triggers the FSS on high neutron flux rate or high neutron flux. If the FRPS fails to request shutdown, the SRPS will trigger the SSS due to high neutron flux rate, high neutron flux or failure of the FSS.

16.8.7.2 Design Basis Fault Sequence

16.8.7.2.1 Inadvertent Withdrawal of a Control Plate during Start-up

a) an initially fully inserted control plate is withdrawn

b) neutron flux rate and neutron flux increase and, consequently, core power increases

c) control plates withdrawal inhibited by CR interlocks

d) FRPS requests reactor trip

e) reactor is shut down

f) PCS and RSPCS pumps are manually shut down

g) operator starts up Long Term Pool Cooling mode of RSPCS and SCS

16.8.7.2.2 Inadvertent Withdrawal of an Absorber Plate with Core at Full Power Operation

a) A partially inserted CR is withdrawn.

b) Neutron flux rate, neutron flux and core power increase.

c) RCMS alarm on high neutron flux.

d) Bank insertion of control rods.

e) In case the RCMS fails to stop the sequence, the FRPS triggers the actuation of the FSS due to high neutron flux or low period.

f) Reactor is shutdown.

g) PCS and RSPCS pumps are manually stopped.

h) Operator starts up Long Term Pool Cooling mode of RSPCS and SCS.

16.8.7.2.3 Inadvertent Fast Insertion of a U-Mo Irradiation Target

a) One U-Mo containing three targets is manually loaded rapidly into a U-Mo irradiation position while the reactor is at full power.

b) Neutron flux rate increases and, consequently, core power increases.

c) RCMS alarm on high neutron flux.

d) Bank insertion of CRs.

e) In case the RCMS fails to stop the sequence, the FRPS triggers the actuation of the FSS due to high neutron flux or low period.

f) Reactor is shutdown.

g) PCS and RSPCS pumps are manually stopped.

h) Operator starts up Long Term Pool Cooling mode of RSPCS and SCS.

16.8.7.2.4 Inadvertent Removal of a Fixed Irradiation Rig

a) Removal of fixed irradiation from one of the irradiation positions.

b) Neutron flux rate increases and, consequently, core power increases.

c) In case limiting values in core variables are exceeded, the reactor is shutdown by the FSS triggered by high neutron flux rate or high neutron flux.

16.8.7.2.5 Inadvertent Removal of a Can with Excess Irradiation Material from a Pneumatic Channel

a) Removal of a can with excess absorbent material from one of the pneumatic irradiation positions.

b) Neutron flux rate increases and, consequently, core power increases.

c) In case limiting values in core variables are exceeded, the reactor is shutdown by the FSS triggered by high neutron flux or high neutron flux rate.

16.8.7.3 Numerical Analysis

16.8.7.3.1 Modelling Assumptions

Modelling assumptions are presented in Section 16.3.

16.8.7.3.2 Calculation Methodology

The transients have been simulated with the PARET code. The PARET code is described in Section 16.3.

16.8.7.3.3 Hypotheses

16.8.7.3.3.1 Events condition

a) In the transients with withdrawal of a control plate, the control plate withdrawn is the one with the highest reactivity worth.

d) Inadvertent control plate withdrawal starts with the plate at 20 % withdrawal. This corresponds to the maximum reactivity insertion ramp.

e) The total reactivity worth of the FSS includes a 7 % reduction to allow for depletion of the control absorber in all cases.

f) Single failure criterion is adopted for the FSS (four out of five control plates fall into the core, the control plate with the largest extinction worth remains out) and SSS.

g) In PARET, trips are triggered by overpower. Therefore, a power equivalent to the neutron flux trip level for the FRPS has been modelled. PARET allows also modelling of a period trip (neutron flux rate). In all cases, either the period trip will occur before the overpower trip, or the difference in time of the two trips will be insignificant. In all transients, adopting a FRPS trip by overpower is conservative.

h) In case of FSS failure the trip for the actuation of SSS is considered to be overpower, i.e., the period trip in the SRPS is ignored.

16.8.7.3.3.2 Safety margins and conservative assumptions

a) PARET is a “best estimate” code. Uncertainties are not explicitly considered, rather conservative data are adopted for the calculations. This is in accordance with international accepted practice for nuclear power plants[6] .

b) For all transients, the total power generated heats the fuel plates and core coolant (20MW). However, in the RRR part of this heat is dissipated by the Reactor and Service Pool Cooling System. The design value for the power generated by the

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[6] IAEA Safety Report “Accident Analysis of Nuclear Power Plants”, Draft, June 2000.

core is 18.8MW, (Chapter5, Section 5.8), therefore a safety margin of 6.4% is applied.

c) Given the reliability of the FSS, its failure is considered unlikely. In a number of cases, this low likelihood of failure, together with the low frequency of occurrence of the initiating event, is sufficient to render the subsequent fault sequence beyond the design basis (e.g., start up accident). However, in order to demonstrate the robustness of the design, some selected cases have been analysed assuming the failure of the FSS

d) In this analysis, failure of the FSS signifies that no plate falls into the core. This is a very conservative assumption, since this implies either all five plates stuck in a position above the core or failure of the all trip signals to the FSS associated to the PIE. All PIEs are covered by at least two signals.

e) The Second Reactor Protection System trip signal to the SSS also disconnects the motor of the Control Rod Drives (CRDs). This results in the simultaneous insertion of the plates and emptying of the reflector vessel. To demonstrate the independence of the Shutdown Systems, this simultaneous insertion of the control plates with SSS trip has been ignored in all cases, rather the plates have been kept in the position they held prior to the SSS trip.

f) Photoneutrons are not included when considering the delayed neutron fractions. This is a conservative assumption.

g) Positive reactivity insertions are considered with a +20 % safety margin. This margin covers calculation errors.

h) The Ir rig reactivity worth considers approximately four times the maximum calculated reactivity worth (design value) plus a 20% uncertainty margin.

i) Feedback coefficients are considered with –15 % margin. This margin is appropriate and in line with international benchmarks (ANL benchmark for silicide plates)[7].

j) The value of FSS insertion time adopted in the analysis corresponds to the maximum design value. Measurements in the FSS prototype indicate that the insertion time is significantly less.

k) A hot channel peaking factor of approximately 25 % over the best estimate value is considered.

l) During reactivity transients, with a significant rise in thermal power, the Departure from Nucleate Boiling or Critical Heat Flux precedes flow instabilities phenomenology. Consequently redistribution is no longer considered the limiting phenomenon as is the case in steady state operation. In all cases, a limit of 1.5 has been adopted for the Critical Heat Flux Ratio (CHFR).[8]. Values of CHFR above this limit indicate within a high level of confidence that no damage to the plates occurs.

m) A steady state CHF correlation has been used. Tong correlation has been adopted. Use of a steady state correlation for transient calculations is conservative.

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[7] Research reactor core conversion from the use of higly enriched uranium to the use of low enriched uranium fuels guid book. IAEA-TECDOC-233. Vienna 1980

[8] Sudo, Y. And Kaminaga, M., “A new CHF correlation scheme proposed for vertical rectangular channels heated from both sides in nuclear research reactors”, Journal of Heat Transfer, Vol 115, pp426-434, May 1993.

n) The overall SSS drainage velocity has been reduced by a uniform margin of 10%.

o) A -10 % margin was considered for SSS reactivity calculations. These calculations were performed using the MCNP-4C Code from the RSICC Computer Code Collection (CCC-700) of the Oak Ridge National Laboratory, and its associated nuclear data collection (DLC-200 MCNPDATA) which is based on Monte Carlo methods. This code has been widely and internationally validated.

p) The thermal conductivity of the cladding was taken from ANL [9] and is conservative with respect to the design value.[10]

16.8.7.3.4 Inadvertent Withdrawal of a Control Plate during Start-up

The inadvertent continuous withdrawal of the control plate with maximum reactivity worth at nominal speed is analysed during start-up.

16.8.7.3.4.1 Start up to High Power Operation

For the DBIE of control plate withdrawal at start-up with FSS actuation a narrow peak in power is reached at 19.1 s, triggering the FSS due to high neutron flux.

Coolant and fuel plate cladding temperatures follow power evolution The hot channel cladding temperature increases and the coolant temperature reaches a maximum temperature well below saturation temperature. The evolution of the Critical Heat Flux Ratio ($q''_{CHF}/q''_{max\ hot\ channel}$) fulfils the acceptance criterion (CHFR>1.5) during the whole transient.

For the DBIE of control plate withdrawal at start-up with SSS actuation a power peak is reached at 25.3 s. The power peak lasts longer, as expected given the delay in actuation of the SSS. This delay (Section 16.3.3.3.2) is the sum of the delay in the electronic of the FRPS and SRPS plus the actuation of the SSS valves.

Coolant and cladding temperatures of the fuel plate follow power evolution. The hot channel cladding temperature increases. The coolant temperature reaches a maximum temperature still below saturation temperature. The evolution of the Critical Heat Flux Ratio fulfils the acceptance criterion (CHFR>1.5) during the whole transient.

A similar behaviour is observed in the hot (U-Mo) rig. This calculation has been performed with RELAP, with the power evolution obtained with PARET as input. The cladding temperature will remain safe while the bulk coolant remains well below saturation temperature.

Start-up to Physics Test Reactor State Operation

The same insertion has been analysed for start up to Physics Test operation. A narrow peak power is reached at 14.8 seconds. The trip is triggered at 14.7 s.

Coolant and clad temperatures follow power evolution. The maximum coolant temperature for the hot channel is about 45.9 ^{0}C and the maximum clad temperature is safe. The maximum heat flux for the hot channel is well below the Critical Heat Flux obtained with Fabrega's[11] correlation.

[9] The whole core LEU U_3Si_2-Al fuel demonstration in the 30 MW Oak Ridge Research Reactor. ANL/REWRTR/TM-14. July 1991.

[10] Properties and selection non ferrous alloys and special materials. ANS Handbook, Vol 2.

[11] "Le calcul thermique des Réacteurs de Recherche refroidis par Eau", S. Fabrega, Commissariat a l'énergie atomique, CEA-R-4114.

16.8.7.3.5 Inadvertent Withdrawal of a Control Plate at Full Power Operation

This Section presents the numerical analysis of the inadvertent continuous withdrawal at nominal speed of the central control plate during normal operation; i.e. at full power and flow rate.

The accident is analysed with actuation of the FSS. The SSS can cope with the extraction of a CR at nominal speed during reactor start up, as shown in Section 16.8.3.1, and the start-up accident bounds all other reactivity insertions. Therefore, it can be concluded that the SSS will cope with all other reactivity insertions, including the one presented in this Section.

This scenario is represented by means of a reactivity insertion. At the beginning of the transient the reactor is in high power operation, i.e. full power and full flow rate.

During the transient the peak power is reached approximately at 7.0 seconds. The FSS trips at 6.8 seconds.

Channel temperatures follow the power evolution with a small increase. The temperature of the cladding reaches a safe maximum for the hot channel, and the coolant maximum temperature is 66º C, well below the saturation temperature at core pressure.

16.8.7.3.6 Inadvertent Insertion of a U-Mo Irradiation Target

This transient simulates the fast insertion of a U-Mo target, compared to normal operating insertion rates,, during operation at 20 MW and full flow rate. The transient is analysed considering the successful actuation of the FSS on trip by overpower and the failure of the FSS with successful actuation of the SSS (overpower). The analysis with actuation of the SSS has been done to show the behaviour of this system after a reactivity insertion ramp steeper than the ramps analysed in the previous cases.

Analysis of a DBIE involving inadvertent insertion of a U-Mo rig at full power shows power and reactivity evolution for this transient with successful actuation of the FSS. The trip takes place at 17.31s and the peak power occurs 160 ms later.

Coolant and fuel plate cladding temperatures follow power evolution with a low increase. The cladding temperature in the hot channel increases to a safe maximum value. The maximum coolant temperature is 66º C, far below saturation temperature.

Another analysis has been performed of a DBIE involving inadvertent insertion of a U-Mo rig at full power representing power and reactivity evolution for the transient with failure of the FSS and successful actuation of the SSS on overpower. The trip occurs at 17.4s. and peak power occurs at 18.5 s. Coolant and fuel plate cladding temperatures follow power evolution with a small increase. The peak cladding temperature is safe and the coolant temperature reaches a maximum value of 66.1 ^{o}C, far below saturation conditions.

16.8.7.3.7 Inadvertent Removal of a Can with Excess Absorbent Material from a Pneumatic Irradiation Target

This event refers to the inadvertent withdrawal of an absorber material canned in cadmium cylinders being irradiated in a pneumatic channel during normal operation (i.e., full power and flow rate). This analysis assumes a QA procedure violation in the preparation of the target, resulting in a higher than normal reactivity worth inserted in this accident. The reactivity insertion ramp could still be significant due to the high withdrawal speed of the pneumatic channel. Assuming the target could be withdrawn rapidly from the irradiation position, the reactivity ramp is steep. The transient is analysed with actuation of FSS and with failure of the FSS and actuation of the SSS.

The analysis shows maximum power is reached at 0.14 s. The FSS is triggered at 0.0663 s due to overpower. The cladding temperature in the hot channel remains safe while the coolant temperature increases to 68ºC which is less than the saturation temperature. These peaks are very short in duration and soon after reactor trip, the temperatures go down to shutdown state values. The system remains within the acceptance criterion during the evolution of the transient.

An analysis of a DBIE involving inadvertent removal of a pneumatic irradiation target with failure of the FSS and actuation of the SSS has been undertaken. The SSS trips at 0.066s due to overpower. The maximum power is at 6.32 s. The cladding reaches a safe maximum temperature while the coolant reaches a maximum temperature of 75ºC, far from saturation. The system remains within the acceptance criterion for Critical Heat Flux Conditions during the duration of the transient.

16.8.7.3.8 Inadvertent Removal of a Fixed Irradiation Target

This case assumes that a fixed Iridium irradiation rig is removed during full power and Physics Test operation.

This transient is analysed with successful actuation of the FSS for Power and Physics Test states.

16.8.7.3.8.1 Power State

A DBIE involving inadvertent removal of a fixed iridium rig with actuation of the FSS at high power operation has been analysed. The maximum power is at about 0.37 seconds. The FSS trip occurs trip occurs at 0.15 seconds due to high neutron flux The maximum coolant temperature is 82.2 ^{o}C and the maximum clad temperature is safe.

Physics Test Operation

A DBIE involving inadvertent removal of a fixed iridium rig with actuation of the FSS in physics test operation has been analysed. The FSS trips due to high neutron flux at 0.2 s. The maximum power occurs at about 0.51 seconds. The maximum clad temperature is safe and the maximum coolant temperature is 59ºC. The maximum heat flux in the hot channel is fully acceptable as a normal operation hot channel heat flux for natural convection and is well below the burn-out heat flux for low flow rates and heat fluxes determined by Fabrega's correlation.

16.8.7.3.9 Comparison with Other Cores

As compared with the Cores of the CABRI[12] and SPERT[13] experiments, the RRR core has low-enriched uranium fuel. Consequently, it presents an additional negative reactivity feedback contribution due to fuel Doppler effect. Another characteristic of this core is the larger neutron generation time due to the heavy water reflector. These characteristics result in later and lower power peaks. Both characteristics improve the intrinsic safety of the Core.

16.8.7.4 Radiological Impact Analysis

No damage to the core is predicted for these transients, therefore there will be no release of fission products. There will be a peak in radiation at the Reactor Pool top. This peak will be due to the over power, but since the power peaks are short in time, the total

[12] Experimental reactor for fast reactivity transients experiments (See [3, 4])

dose to the operator will be within acceptable limits for abnormal operation. During the start up accident, the dose rate will be double the dose rate during normal operation. The period of the increased radiation is a few seconds and the total dose will therefore remain very low.

16.8.8 Conclusions

Analyses of Excess Reactivity Insertions show that the bounding event of this DBA grouping involves an insertion of a significant amount of positive reactivity and that either RPS is capable of shutting down the reactor. Heat removal from the core and rigs is adequate in all cases. It is concluded that nuclear safety is guaranteed for all credible events involving Excess Reactivity Insertion. The effect of bank rod withdrawal is considered in the section dealing with Beyond Design Basis Accidents, Section 16.19.

End of Section