16.11 ANALYSIS OF LOSS OF COOLANT EVENTS

16.11.1 Introduction

This Section describes the analysis of accidents arising from losses of reactor coolant from the PCS, RSPCS and connected systems.

The analysis applies to the behaviour of the core and irradiation rigs.

The following main components of the circuits are considered: pipes, fittings, casings and seals of pumps, valves and connections to equipment.

The analysis is divided into two main groups:

a) Loss of Coolant Accidents (LOCAs) in the PCS.

b) LOCAs in the Reactor and Service Pools, the RSPCS and connected systems.

In this analysis, small leaks are not considered initiating events as they are controlled by make-up through the Reactor Pool Hot Water Layer System. These leaks are treated as anticipated operational occurrences. A description of the Reactor Pool Hot Water Layer System is given in Chapter 6.

The occurrence of a LOCA would lead to a sequence of alarms, warnings and, finally, triggering of the FSS and SSS. The FSS would be triggered by any of the following;

a) Low water level in the Reactor Pool

b) High radiation level in the open end of the Reactor Pool

c) Low flow rate in the PCS

d) Low core pressure drop

e) Low RSPCS flow

The SSS would be triggered by either

a) Very low Reactor Pool water level

b) Low pressure drop across the core, or

c) Failure of the FSS

Events involving the loss of Heavy Water from the Reflector Cooling and Purification System are considered in Section 16.12.

The initiating events for LOCAs have been grouped according to type of accident. Each initiating event is assessed in the context of defence in depth barriers. On the basis of the arguments presented, DBIEs are identified for further analysis. The final sections detail the event sequences and their numerical analysis.

16.11.2 LOCAs in the PCS

16.11.2.1 PCS Pipe Break or Valve Failure

This initiating event considers a failure in one of the PCS pipes, valves or connected systems.

Large safety margins are present in the PCS piping specifications. These design values are set in order to define the conditions under which the mechanical design of the piping is performed. This does not mean that the pipe would fail when submitted to a higher pressure or temperature. The failure pressure for this piping is one order of magnitude above the design pressure. The mechanical design and stress analysis of the PCS piping ensures that it withstands the SL-2 seismic event. Furthermore, the piping is properly supported.

Although all the PCS piping penetrations into the Reactor Pool are far above the top of the core, some of the pipes do run below the core level once they leave the reactor block. For this reason, the design provides redundant siphon breakers on the inlet and outlet lines of the PCS to stop the loss of coolant from the Reactor Pool and prevent uncovering of the core. Configuration details of the siphon breakers are presented in Chapter 6. The siphon breakers are protected against blockage by a mesh.

The PCS pumps are firmly fixed to their supports, preventing the imposition of excessive stresses on the connected piping. Vibration of the pumps capable of inducing vibration and stress in the PCS piping is prevented by tripping of the pump. This trip sends a signal to the RCMS.

Operator maintenance errors are minimised by operator training and clear maintenance and operation procedures. Walkthroughs prior to start-up minimise the likelihood of a drainage valve left inadvertently open or a flange not properly secured.

Another cause for a LOCA is the failure of the seal of one of the valves in the PCS. There are three types of valves in the PCS: diaphragm valves, check valves, and butterfly valves:

a) Diaphragm valves are used on discharge and venting lines. The valve body is made of stainless steel and has welded ends.

b) Check valves are used to prevent backward flow in case of pump shutdown. Valves are wafer type, with the body made of cast iron and seats of ethylene propylene rubber. The latter has been shown to be thoroughly resistant to the environmental conditions expected.

c) Butterfly valves are used to isolate portions of the PCS. There is one valve located on each side of the main components (i.e., pumps). There are two valves located on the PCS pipes that penetrate the reactor block with the purpose of isolating the Reactor Pool to prevent the core from uncovering. The valve is of the lug type, its body made of cast iron and seats of ethylene propylene rubber.

There are four pipes along the PCS that could initiate a large LOCA:

Overall, the occurrence of a LOCA through failure of the pipework and valves is considered very unlikely. Seal failure is similarly unlikely due to the construction of the valves and mild operating conditions. The high quality of the water further minimises the risk of damage due to corrosion. Nevertheless, failures of the PCS piping outside the reactor block are considered to lie within the design basis. The effect on the core and irradiation rigs of a PCS LOCA is assessed.

There is an RCMS signal that stops the PCS pumps on low pool water level. Depending on the location of the failure, this signal would significantly diminish the flow rate through the rupture simultaneously with the reactor shutdown initiated by the FRPS. However, no credit is given to the RCMS PCS pump trip signal in this analysis. The cessation of the leak flow is due to the action of the passive siphon breakers on the PCS pipe.

16.11.2.1.1 Defence in Depth Barriers

Level	Main Characteristics	Safety Feature
1	Conservative design and inherent safety features	Large safety margins in piping and valves specification

Level	Main Characteristics	Safety Feature
		Piping built of stainless steel following strict application of design codes.
		Piping properly supported
		Piping designed to withstand SL-2 earthquake
		High water quality to avoid corrosion
		Appropriate maintenance programme
		Appropriate system inspection and tests during installation and on service inspection
		Leak detectors
		Siphon Breakers
		PCS pumps adequately fixed to support basis that permit to prevent stress on the connected piping.
2	Operation control and response to abnormal operation	RCMS pump trip signal on low pool water level
		High vibration and high temperature indication on motors and pumps
		RCMS pump trip signal on very high vibration in pump bearings.
		Alarms on: a) high water level in leak detectors b) high water level in LOCA sumps c) low water level in the Reactor Pool d) low core pressure drop and low flow (depending on the location of the break) e) high radiation level
3	Control of accidents within the design basis	Passive redundant siphon breakers
		Two independent and diverse shutdown systems
		FRPS reactor trip signal on: a) Low water level in the Reactor Pool b) Low core pressure drop c) Low PCS flow (depending on the location of the break) d) Core temperature difference high SRPS reactor trip signal on a) Low water level in the Reactor Pool b) Low core pressure drop c) Core outlet temperature high d) Failure of the FSS

16.11.2.2 Decay Tank Break

This initiating event considers partial failure of the Decay Tank.

The tank design pressure and temperature, and the normal operating values for these parameters, are the same as those for the PCS piping. Rupture of the Decay Tank due to corrosion or overpressure is considered very unlikely. In addition, there is no source of potential missiles inside the Decay Tank Room that might initiate a catastrophic failure. The same design and working values extend to the drainage piping. The controlled water quality of the PCS and operating conditions ensure that corrosion and overpressure can be disregarded as a cause for a loss of coolant event. A gate valve isolates the decay tank drainage pipe and the pipe end is also closed with a cap. The design of the gate valve minimises the potential for leaks. The cap at the end of the pipe provides redundant means of isolation as it has been conservatively designed for full tank pressure.

The Decay Tank Room is provided with a leak detector that provides an alarm to the Main Control Room in the event of a leak.

Failure of the Decay Tank is considered to lie within the design basis. Its consequences are bounded by failure of the PCS piping.

16.11.2.2.1 Defence in Depth Barriers

Level	Main Characteristics	Safety Feature
1	Conservative design and inherent safety features	High quality tank and drainage pipe materials
		Large safety margins in tank specification
		High water quality to avoid corrosion
		Low water pressure and temperature
		Appropriate maintenance programme
		Appropriate system inspection and test programme
		Piping designed to withstand the SL-2 earthquake
2	Operation control and response to abnormal operation	Leak detector
		Leak discharge to the Radioactive Liquid Waste Management System
		Alarm on: a) high level on leak detector b) high water level in LOCA sumps c) low pool water level d) high radiation level
3	Control of accidents within the design basis	Passive redundant siphon breakers
		Two independent and diverse shutdown systems
		FRPS reactor trip signal on: a) Low water level in the Reactor Pool b) Low core pressure drop c) Low PCS flow (depending on the location of the

Level	Main Characteristics	Safety Feature
		break) d) Core temperature difference high SRPS reactor trip signal on a) Low water level in the Reactor Pool b) Low core pressure drop c) Core outlet temperature high d) Failure of the FSS

16.11.2.3 Primary Cooling System Pump Break

This event considers failure of one of the PCS pump casings or pump shaft seals.

The most realistic event is considered to be partial pump shaft seal failure leading to a LOCA. PCS pumps use metallic shaft seals. These metallic seals are conservatively designed to withstand the system design pressure and temperature. Total failure of the shaft seal is considered highly unlikely since pump casing disassembly is required for complete seal removal.

Each pump is located in a separate room. Leak detectors are located inside each pump room to provide warning in case of a loss of coolant. Shaft seals also have leak detectors. A discharge path to the Radioactive Liquid Waste System capable of coping with the leakage arising from total rupture of the pump outlet pipe is also provided.

A LOCA arising from failure of a pump is considered within the design basis. Its consequences are bounded by those of PCS piping failure.

16.11.2.3.1 Defence in Depth Barriers

Level	Main Characteristics	Safety Feature
1	Conservative design and inherent safety features	High quality pumps
		High quality water to avoid corrosion
		Appropriate inspection and test programme
		Appropriate maintenance programme
		Low system pressure and temperature
		Temperature indication on motor and pump
		Designed to withstand the SL-2 earthquake
2	Operation control and response to abnormal operation	Vibration indication on motor and pump
		Leak detection in shaft seals
		Alarm on: a) high vibration of rotating machinery b) high level in leak detector c) high water level in LOCA sumps d) low water level in the Reactor Pool e) low core pressure drop f) high radiation level

Level	Main Characteristics	Safety Feature
3	Control of accidents within the design basis	Passive redundant siphon breakers
		Two independent and diverse shutdown systems
		FRPS reactor trip signal on: a) Low water level in the Reactor Pool b) Low core pressure drop c) Low PCS flow (depending on the location of the break) d) Core temperature difference high SRPS reactor trip signal on a) Low water level in the Reactor Pool b) Low core pressure drop c) Core outlet temperature high d) Failure of the FSS

16.11.2.4 Heat Exchanger Breakage

Heat exchangers are composed of several plates that are sandwiched by frame plates at the ends. Tightening bolts join both frame plates and compress the heat exchanger plates and its gaskets.

In the event of gasket failure, only small leaks are expected to occur as the seal cannot be removed completely by water pressure due to the compression exerted by the plates. These leaks can be compensated for by the make-up from the Reactor Pool Hot Water Layer System. The gasket design ensures leaks flow out from the heat exchanger, preventing mixing of primary and secondary waters.

Bolts are designed with large safety margins in relation to the operating pressure of the system. SL - 2 seismic forces are also taken into account.

The entire component has been designed for SL-2 seismic forces. Leak detection is provided to generate an alarm in case of leakage of coolant.

Catastrophic structural failure of the heat exchangers caused by corrosion and degradation of material is not considered credible given the mild operating conditions of both the PCS and the Secondary Cooling System.

An activity monitor in the Secondary Cooling System would provide early warning in the event a heat exchanger leaked PCS coolant into the Secondary Cooling System.

On the basis of the above discussion, failure of a heat exchanger is considered to lie within the design basis. The consequences on the core and irradiation rigs are considered bounded by those of a failure of the PCS piping. The consequences of a leak of coolant into the Secondary Cooling System are considered minor and are bounded by the Loss of Heat Sink (Section 16.10).

16.11.2.4.1 Defence in Depth Barriers

Level	Main Characteristics	Safety Feature
1	Conservative design and	High quality heat exchanger materials

	inherent safety features	High water quality to avoid corrosion and seal degradation
		Appropriate inspection and test programme
		Appropriate maintenance programme
		Low system pressure and temperature
		Designed to withstand SL-2 earthquake
2	Operation control and response to abnormal operation	Alarms on: a) high water level in leak detectors b) high water level in LOCA sumps c) low water level in the reactor pool d) low core pressure drop e) high radiation level
3	Control of accidents within the design basis	Passive redundant siphon breakers
		Two independent and diverse shutdown systems
		FRPS reactor trip signal on: a) Low water level in the Reactor Pool b) Low core pressure drop c) Low PCS flow (depending on the location of the break) d) Core temperature difference high SRPS reactor trip signal on a) Low water level in the Reactor Pool b) Low core pressure drop c) Core outlet temperature high d) Failure of the FSS

16.11.3 Loss of Coolant Accidents in the Reactor and Service Pool Cooling System

16.11.3.1 Damaged Pool

In the unlikely event that the Reactor or Service Pool were damaged, for example due to corrosion or impact of missiles, the subsequent leak flow would be small because the pools are embedded in a massive concrete structure that serves as a robust, leak-proof, second barrier. Any small leaks arising would be compensated for by make-up from the Reactor Pool Hot Water Layer System.

Penetrations to the pools have continuous leak monitoring. Any leaks would raise alarms in the Main Control Room via the RCMS. Penetrations are thoroughly tested for leak tightness during construction and installation. All process penetrations are located above core level, except for the Second Shutdown System’s discharge pipe from the Reflector Tank. A break in this pipe is considered very unlikely. Heavy water quality is strictly controlled.

The openings at the base of the Reactor Pool for the Control Rods are connected to the Control Rod Drive Room below. These openings have double poly-pack seals (see Chapter 5) that inhibit water leakage into the room. Should a failure of any of these seals occur, water would drain into the Control Rod Drive Room. This room is small in volume and pool water level would not fall to a dangerous level, even with the transfer channel gate in place. This is the worst case scenario, since the Reactor Pool water level would be higher if the Reactor and Service Pools are connected. The largest penetrations to the pool are the beam tubes. The behaviour of the beam tubes is assessed in Section 16.11.3.6.

On the basis of the above, failure of the Reactor or Service pool boundaries (excluding the beam tube penetrations) resulting in a leak of coolant from the pools is considered to lie within the design basis. The consequences of such a leakage are minor and are not analysed further.

16.11.3.2 Reactor and Service Pools Cooling System Pipe and Valve Failure

This event refers to the breakage of one of the pipes or valves and the failure to close a valve in one of the drainage lines of the RSPCS, including those branches of the system associated with the Long Term Pool Cooling mode of operation. There are two pipes along the RSPCS that could contribute to a large LOCA:

a) the outlet RSPCS line
b) inlet RSPCS line

The RSPCS piping and valves have the same mechanical design values of pressure and temperature as the PCS pipes. The mechanical design of the RSPCS ensures that it withstands the SL-2 seismic event. As in the case of the PCS piping, all the penetrations to the pools are well above the top edge of the core. Some of these pipes do run below the level of the core once they leave the reactor block. For this reason, the design provides siphon breakers on the outlet and inlet pipes of the RSCPS to stop the leak and prevent uncovering of the core.

There are three types of valves used for RSPCS: diaphragm valves, check valves and butterfly valves.

a) Diaphragm valves are used on discharge and venting lines. The valve body is made of stainless steel and has welded ends.

b) Check valves are used to prevent reverse flow in the event of pump shutdown. The valves are of the wafer type, with the body, disk and stem made of AISI 316, and seats and O-ring of ethylene propylene rubber. The latter has been shown to be thoroughly resistant to the environmental conditions expected.

c) Butterfly valves are used to isolate portions of the RSPCS. There is one located on each side of the main components. In addition, there are valves located on the RSPCS pipes that penetrate the reactor block with the purpose of isolating the Reactor Pool. The valves are of the lug type. The potential for failure due to missile impact is equivalent to that in the PCS. Provisions are in place regarding the RSPCS pump testing and inspection during manufacturing installation and operation. In addition, instrumentation is in place to alert the operator to vibrations that could damage the unit. The pump would trip on very high vibration. The pump is bolted to a base to ensure that vibration does not cause stress on the connected piping.

The diameter of the piping of the RSPCS is smaller than the diameter of the PCS piping. Thus, the consequences of any loss of coolant event involving RSPCS piping would be bounded by those of the PCS piping.

Breaks of piping would be detected by means of low Reactor Pool water level, low RSPCS flow rate, leak detectors, high radiation level and high level in LOCA sumps.

On the basis of the above discussion, failures of the RSPCS piping and valves are considered to lie within the design basis. Their consequences on the core are considered bounded by the failure of the PCS piping. However, since a loss of coolant in the RSPCS has impact on the cooling capability of the irradiation targets, its effects are analysed further.

16.11.3.2.1 Defence in Depth Barriers

Level	Main Characteristics	Safety Feature
1	Conservative design and inherent safety features	High quality piping materials and valves
		QA procedures for procurement and installation
		Periodic inspection and testing
		Large safety margins in piping specification
		Seismically designed
		High water quality to avoid corrosion
		Appropriate maintenance programme
		Appropriate system inspection and tests
2	Operation control and response to abnormal operation	Alarms on: a) high level in leak detectors b) high water level in LOCA sumps c) low water level in the reactor pool d) low flow rate of the RSPCS (depending on the location of the break) e) high radiation level
3	Control of accidents within the design basis	Passive redundant siphon breakers
		Two independent and diverse shutdown systems
		FRPS reactor trip signal on: a) Low water level in the Reactor Pool b) Low RSPCS flow (depending on the location of the break) SRPS reactor trip signal on a) Very Low water level in the Reactor Pool b) Failure of the FSS

16.11.3.3 Reactor and Service Pools Cooling System Decay Tank Break

This initiating event considers failure of the RSPCS Decay Tank and its drainage pipe.

The tank contains demineralised water. Tank failure due to corrosion or overpressure is considered highly unlikely. In addition, there is no source of potential missiles identified inside the Decay Tank Room. The tank and its supports have been designed to withstand the SL-2 earthquake.

The discharge pipe only conducts demineralised water. Non-dissolved solids falling into the pool in the Reactor Hall have the potential to cause corrosion although the water of the RSPCS is isolated from the Reactor Hall by the hot water layer. A fraction of the RSPCS flow rate is circulated through a particulate filter and a resin bed. The total volume of the RSPCS is filtered in a 24 h interval. Rupture of the piping due to corrosion or overpressure is considered highly unlikely.

A gate valve isolates the decay tank drainage pipe and the end of the pipe is capped. The design of the gate valve minimises leaks. The cap at the end of the pipe provides a redundant means of isolation.

A leak detector in the decay tank room provides an alarm to the monitoring control room in case of a leak.

On the basis of the above arguments, failure of the Decay Tank is considered to lie within the design basis. The consequences of its failure are bounded by failure of the PCS piping.

16.11.3.3.1 Defence in Depth Barriers

Level	Main Characteristics	Safety Feature
1	Conservative design and inherent safety features	High quality tank and drainage pipe materials
		Large safety margins in tank specification
		High water quality to avoid corrosion
		RSPCS water purification system
		Low water pressure and temperature
		Appropriate maintenance programme
		Appropriate system inspection and test programme
		Designed to withstand the SL-2 earthquake
2	Operation control and response to abnormal operation	Leak discharge to the Radioactive Liquid Waste Management System
		Alarm on: a) high level on leak detector b) low pool water level
3	Control of accidents within the design basis	Passive redundant siphon breakers
		Two independent and diverse shutdown systems
		FRPS reactor trip signal on: a) Low water level in the Reactor Pool b) Low RSPCS flow (depending on the location of the break) SRPS reactor trip signal on a) Very Low water level in the Reactor Pool b) Failure of the FSS

16.11.3.4 Reactor and Service Pools Cooling System Pump Break

This event considers breakage of one of the RSPCS pumps or pump shaft seals.

The RSPCS pump shaft seals of the pumps are identical to the seals in the PCS pumps. They are equipped with leak detection.

A LOCA could occur due to failure of a seal or a union. Seals are manufactured out of high quality material and operating conditions are mild. Flanged unions are inspected according to surveillance requirements. A leak through a union would be detected during inspection and/or walk down and would be corrected. Nevertheless, leak through a faulty seal or union is considered within the design basis.

On the basis of the above, LOCA due to failure of the pump is considered within the design basis. The consequences of the failure are bounded by failure of the RSPCS piping and therefore not further analysed.

16.11.3.4.1 Defence in Depth Barriers

Level	Main Characteristics	Safety Feature
1	Conservative design and inherent safety features	High quality pumps
		High quality water to avoid corrosion
		Appropriate inspection and test programme
		Appropriate maintenance programme
		Low system pressure and temperature
		Designed to withstand SL-2 earthquake
2	Operation control and response to abnormal operation	Leak detection in the shaft seals
		Vibration indication on motor and pump
		Temperature indication on motor and pump
		Alarm on: a) high vibration on rotating machinery b) high temperature on rotating machinery c) high level in leak detector d) high water level in LOCA sumps e) low water level in the reactor tank f) low flow rate of RSPCS
		RCMS pump trip signal on very high vibration
3	Control of accidents within the design basis	Passive redundant siphon breakers
		Two independent and diverse shutdown systems
		FRPS reactor trip signal on: a) Low water level in the Reactor Pool b) Low RSPCS flow (depending on the location of the break) SRPS reactor trip signal on a) Very Low water level in the Reactor Pool

Level	Main Characteristics	Safety Feature
		b) Failure of the FSS

16.11.3.5 Reactor and Service Pools Cooling System Heat Exchanger Break

The heat exchangers are composed of several plates that are packed by frame plates at the ends. Tightening bolts join both frame plates and compress the heat exchanger plates and its gaskets.

In case of gasket failure only small leaks are expected because the seal cannot be removed completely by water pressure due to the compression between plates. Gasket design ensures that any leaks flow out from the heat exchanger, preventing mixing of primary and secondary waters.

Bolts are designed with large safety margins in relation to the design pressure and temperature of the system and the strains and accelerations that arise from the SL-2 earthquake. Leak detectors are also provided to generate an alarm in case of coolant leakage.

Missile impacts that could result in large deformation of the heat exchanger structure and, as a consequence, the loss of seal due to compression between plates can be originated from missile sourcesFailure of the heat exchanger is considered to lie within the design basis. Its consequences are bounded by the effects of failure of the RSPCS piping leading to a LOCA.

16.11.3.5.1 Defence in Depth Barriers

Level	Main Characteristics	Safety Feature
1	Conservative design and inherent safety features	High quality heat exchanger materials
		High water quality to avoid corrosion and seal degradation
		Appropriate inspection and test programme
		Appropriate maintenance programme
		Low system pressure and temperature
		Designed to withstand SL-2 earthquake
2	Operation control and response to abnormal operation	Alarms on: a) high level in leak detectors b) high water level in LOCA sumps c) low water level in the reactor tank d) low flow rate in the RSPCS
3	Control of accidents within the design basis	Passive redundant siphon breakers
		Two independent and diverse shutdown systems

		FRPS reactor trip signal on: a) Low water level in the Reactor Pool b) Low RSPCS flow (depending on the location of the break) SRPS reactor trip signal on a) Very Low water level in the Reactor Pool b) Failure of the FSS

16.11.3.6 Failure of Beam Tubes

Consistent with ARPANSA Regulatory Guideline RG-5 “Criteria for the Design of Nuclear Installations” and international best safety practices, the beam penetrations to the Reactor Pool have been provided with two static barriers to prevent a loss of coolant. These barriers have been designed to withstand the hydraulic, seismic and other relevant loads that may eventually act on them. Both barriers need to fail before a loss of coolant is possible. A description of the barriers and an analysis of their behaviour under load is provided in Section 4.5.

The first barrier is the beam tube enclosure that penetrates the reactor block, crosses the Reactor Pool liner and enters into the Reflector Tank. The material selected and its fabrication ensure that no damage will appear as a result of irradiation. The mild operating conditions (moderate temperature and pressure, good water quality) and the high quality of the materials selected preclude corrosion as a source of damage and failure.

The second barrier consists of a closure plate at the reactor block face. This plate covers the beam tube and shutters. It has a static seal and it is bolted to the reactor block. On the plate there are small rectangular openings. A plate, designed to withstand the water column pressure of the Reactor Pool, covers each of these openings.

The beam tubes are fitted with double bellows to provide flexibility in the zone between the Reactor Pool liner and the reflector tank. These bellows form redundant barriers. Furthermore, to minimise the probability of rupture of the beam tube wall by the impact of falling objects, the beam tubes are protected against missiles that could fall into the Reactor Pool by a metallic mesh. The mesh is designed to withstand the fall of a silicon ingot. The beam tubes are designed to withstand a SL-2 earthquake. In the extremely unlikely case of a beam rupture, the metallic plates on the face of the reactor block remain as additional barriers. These components are designed to withstand the load of the static pressure exerted by the water column inside the pool. This load is larger than the load the plate would experience when loaded with an acceleration of 1g. A stopper prevents the beam shutters from impacting on the plate and windows in the unlikely event they become dislodged during a seismic event.

A leak detector system inside the beam tubes would trigger the corresponding alarms should pool water leak into the beam tubes.

Design provisions, multiple barriers, material quality, QA manufacturing and installation procedures are all in place to provide an extremely robust beam tube boundary. LOCAs through the beam tubes are therefore considered to be so unlikely as to render them beyond the design basis.

16.11.3.6.1 Defence in Depth Barriers

Level	Main Characteristics	Safety Feature
1	Conservative design and inherent safety features	High quality beam materials
		Low pressure system
		Design providing two independent static barriers to prevent leaks
		Short section of the tubes between the reflector tank and the Reactor Pool
		Tube protection against impact by falling objects
		Cover withstanding the pool water column pressure
		Appropriate beam tube inspection and test programme
		Appropriate maintenance programme
		Stoppers to avoid impact of shutters on beam plates
		Designed to withstand the SL-2 earthquake and beyond
2	Operation control and response to abnormal operation	Detectors and alarms on: a) leak detection inside the beam tube b) low water level in the reactor tank

16.11.3.7 Failure of the Seals for Control Rods

Seals for CRs are formed by rubber ‘O’ rings and dynamic poly-packs. The seals are designed to withstand the pressure due to the column of Reactor Pool water and represent a multiple safety barrier against water pool leakage. Failure of this seal assembly allowing significant quantities of coolant to escape from the Reactor Pool requires either the multiple failure of ‘O’ rings or poly-packs. This is considered highly unlikely.

The seals have been tested by irradiation with doses equivalent to 10 years’ full power operation. Results of these tests indicated very good seal performances. Replacement of the seals would be carried out with the FA removed from the Core and the opening at the bottom of the pool tightly closed by a specially designed tool.

Failure of both sets of seals in a CR would result in leakage. In case of failure of one of the seals, leakage detectors are provided inside the CRD Room to detect any abnormal leakage of pool water.

A watertight door closes the CRD room. The size of the CRD room is such that, in case of it being totally flooded by a leak from the Reactor Pool, the water in the Reactor Pool (with the gate in the transfer channel in place) does not drop to a dangerous level. Enough water inside the Reactor Pool is maintained to establish core cooling by natural convection and appropriate shielding for any operations at the Reactor Pool top.

On the basis of the above arguments, failure of the CR seals is considered to lie within the design basis. The effect of the failure of the seals is bounded by the LOCA arising from failure of the RSPCS piping.

16.11.3.7.1 Defence in Depth Barriers

Level	Main Characteristics	Safety Feature
1	Conservative design and inherent safety features	Proven design of seal for CRs
		Low system pressure
		Appropriate inspection and test programme
		Appropriate maintenance programme
		Small size of CRD Room
		Watertight protection of the CRD Room and CRD Room door
2	Operation control and response to abnormal operation	Alarm on: a) high level in leak detector in the CRD room b) low Reactor Pool water level
3	Control of accidents within the design basis	Passive redundant siphon breakers
		Two independent and diverse shutdown systems
		FRPS reactor trip signal on: a) Low water level in the Reactor Pool b) High radiation level SRPS reactor trip signal on a) Low water level in the Reactor Pool b) Failure of the FSS

16.11.4 Design Basis Postulated Initiating Events

A summary of previous considerations and the design basis PIE 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
PCS pipe or valve break or failure							X
Decay Tank Break						X (PCS piping failure)	
PCS pump break						X (PCS piping failure)	

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
PCS heat exchanger break					X (PCS piping failure)	
Damaged pool				Minor consequences. Not analysed further.		
RSPCS pipe or valve break or failure						X
RSPCS Decay Tank Break					X (RSPCS piping failure)	
RSPCS Pump Break					X (RSPCS piping failure)	
RSPCS Heat Exchanger Break					X (RSPCS piping failure)	
Failure of Beam Tubes			X			
Failure of Seals for CRD					X (RSPCS piping failure)	

On the basis of the above, two DBIEs are identified for further analysis:

a) Failure of the PCS piping, and

b) Failure of the RSPCS piping.

The postulated failure can occur at any point around the circuit and be of any size. In order to obtain an insight into the effect of location and size of the failures, a number of locations and leak sizes are analysed. They are;

a) break at the PCS pump discharge line

b) break at the interconnection line

c) break at the PCS pump discharge line

d) failure of the line connecting the main RSPCS pump with the standby pump

e) full break of the RSPCS pool inlet pipe

In addition, a parametric analysis has been carried out to examine the variation in core parameter values with increase in PCS leak size.

16.11.4.1 Detection of the Initiating Event

Following the onset of the LOCA, low flow, water in LOCA sumps and low pool water alarms are triggered. The FRPS and SRPS shut down the reactor. A list of triggering signals is given below:

a) For the FSS:

i) Low core pressure drop. (PCS LOCA with rupture in the discharge line).

ii) High core temperature difference

iii) Low water level in Reactor Pool (this variable covers both DBIEs).

iv) Low PCS flow rate (PCS LOCA with rupture in pump discharge line).

v) Low RSPCS flow rate (RSPCS LOCA with rupture in the discharge line).

vi) High radiation levels

b) For the SSS:

i) Low core pressure drop (PCS LOCA) in conjunction with no end-of-stroke signal from 2 or more CRs.

ii) High core outlet temperature

iii) Very low water level in Reactor Pool (set point is lower than the FSS set point).

iv) Failure of FSS.

16.11.4.2 Design Basis Fault Sequences

These sections consider the most significant events occurring during the PCS LOCA and RSPCS LOCA sequences.

16.11.4.2.1 Loss of Coolant Accident in the Primary Cooling System

a) Failure of PCS pipe.

b) Reactor shutdown

c) RCMS trips the PCS and RSCPS pumps on low pool water level signal.

d) Pool water level drops until it reaches the level of the siphon effect breakers and flap valves

e) In the unlikely event of the RCMS pumps trip signal failing, the PCS and RSCPS pumps lose suction at the siphon breaker level and the loss of pool coolant inventory ceases.

f) Flap valves open and natural circulation is established.

A number of operator actions would be expected to be carried out by procedures. They have not been considered in the analysis of the transients. The operator actions are;

a) Identify LOCA area by alarm in leak detector or LOCA sumps.

b) Shutdown PCS pumps in the event that the RCMS pump trip signal failed and the pool water surface has not reached siphon breaker level.

c) Shutdown the RSPCS pumps.

d) If possible, initiate the Long Term Pool Cooling mode of the RSPCS and SCS.

16.11.4.2.2 Loss of Coolant Accidents in the Reactor and Service Pools Cooling System

a) Breakage of RSPCS pipe.

b) Reactor shutdown by:

(i) the FRPS on low RSCPS flow, low water level in Reactor Pool

(ii) the SRPS on very low water level in Reactor Pool (set point is lower than the FRPS set point), failure of the FSS.

c) RCMS stops the PCS and RSPCS pumps due to low pool water level.

d) Pool water level drops until it reaches the siphon breaker level

e) In the event of the RCMS pumps trip signal failing, RSPCS and PCS pumps lose suction head and the loss of pool coolant inventory ceases.

f) Flap valves open and natural circulation is established.

A number of operator actions would be expected to be carried out by procedures. They have not been considered in the analysis of the transients. The operator actions are;

a) Identify LOCA area by alarm in leak detector or LOCA sumps

b) Shutdown PCS pumps in the event that the RCMS pump trip signal failed and the pool water surface has not reached siphon breaker level.

c) Shutdown RSPCS pumps

d) Where possible, initiate the Long Term Pool Cooling mode of the RSPCS

16.11.4.3 Numerical Analysis

16.11.4.3.1 Primary Cooling System

The numerical analysis has been performed with RELAP 5. The results of the transient following a PCS LOCA are presented in this section.

16.11.4.3.1.1 Modelling Assumptions

The nodalisation of the RRR presented in Section 16.3 was used as the basis for this numerical analysis. A time dependent volume was introduced as a boundary condition to simulate the discharge through the break. This volume was connected to the PCS by a junction, with a cross-sectional area corresponding to the size of the simulated break. The junction includes a valve to simulate the onset of the LOCA.

The core geometry, the properties of the fuel and coolant and the shutdown systems are presented in Chapter 5.

The reactor is stable at full power prior to the occurrence of the initiating event. Operating core power is conservatively taken as 20 MW. Opening the valve at the junction simulates the break occurrence.

The pool coolant level reduction is calculated by integrating the volumetric flow per pool cross sectional area over time. It is conservatively assumed that the FSS is not actuated on a low PCS flow signal. Reactor trip is triggered by a low pool water level signal to the

FRPS and by the SRPS on failure of the FSS. The trip set point for the FRPS is set at the analytical limit for low pool water level. This is lower than the actual trip set point and it is a conservative assumption.

Cessation of power to the pump motors is simulated by a drop in the torque from its nominal value to zero in 1 second.

16.11.4.3.1.2 Safety System Trip Parameters

The FRPS triggers the FSS on low level of the pool water. The SRPS triggers the SSS on failure of the FSS.

16.11.4.3.1.3 PCS Pump Discharge Line

The sequence starts with break at the PCS pump discharge line. When liquid level reaches the set point value, the FRPS triggers the FSS. The RCMS would trip the PCS and RSCPS pumps but, as this is a Safety Category 2 system, no credit is given to its actuation. In a realistic scenario (as opposed to the conservative scenario presented here), the pumps would be shut down simultaneously with the reactor, and the rate of fall of the pool water level would be much slower. The core flow decreases from the steady state value before the reactor trip. This decrease in flow rate would result in a reactor trip on low flow, which would act sooner than the low pool water level signal. However, no credit is given to this signal. The initial temperature rise in the core outlet coolant occurs as a result of the drop in coolant flow through the core. The core average outlet temperature peaks while the inlet remains relatively steady. In the long term, the core inlet temperature decreases smoothly.

After the reactor is shutdown, the core outlet temperature falls rapidly to a value that depends on the balance between the power generated and the removal capacity of the coolant based on the flow evolution. From this point onwards the temperature difference over the core is very low as it is generating only decay heat and the primary pumps are on.

When the cold slug from core outlet following FSS trip reaches the heat exchanger the inlet core temperature decreases. The flap valves do not open in this transient as both pumps are on and the flow reduction through the core is insufficient to permit their opening.

Following the occurrence of the break, a rapid rise in coolant temperature is observed in the hot channel, with a maximum value at the hot channel outlet. A similar rise occurs for the cladding, where the maximum temperature is at the central node. An abrupt drop occurs in both temperatures after reactor shutdown. A further temperature reduction occurs following the arrival of the cold slug. Thereafter the temperatures remains nearly constant. Towards the end of the analysed period, the temperatures continue falling more slowly, following the decay power rate. In the long term however, a slow pool warming occurs.

The sequence was re-analysed using the same assumptions but with the failure of the FSS. The SSS actuates after the onset of the break. Its decay power curve is slightly different for the first few seconds, tending to the same power for large times. This difference causes nearly no difference in the results for temperatures and flows from those obtained with actuation of the FSS. On this basis, therefore, from this point onwards only sequences assuming success of the FSS are analysed.

The LOCA in the PCS has no impact on the RSCPS. When no credit is given to the RCMS pump trip signal, the RSPCS continues running and cooling of the irradiation rigs is not affected. There is low interconnection flow, but this has no effect on the cooling of the rigs. In the more realistic case where the RCMS trip signal does stop the PCS and RSPCS pumps, a loss of flow occurs in the RSPCS. The effect of this transient is equivalent to that of the Loss of Electric Power transient analysed in Section 16.7.

16.11.4.3.1.4 Break at the Pipe in the Interconnection Line

This sequence is initiated with an equivalent break diameter at the pipe in the interconnection line. This sequence, together with the previous one, represent the upper and lower bounds for the total break of the interconnection line. The reactor is tripped by a low pool water level signal to the FRPS. The flow through the break represents some 60% of the leak flow in the previous sequence. Because of the location of the break, the core flow is only slightly reduced. The FSS would not be tripped on low core flow in this situation. After the reactor is shutdown by the FSS, the outlet core temperature falls rapidly. From this point onwards, the temperature difference across the core is very low, as it is generating only decay heat and the primary pumps are still on.

After the onset of the accident, the cold slug coming from core outlet via the heat exchanger causes the inlet core temperature to decrease. The flap valves do not open during the interval analysed in this transient as both PCS pumps are on and the flow reduction through the core is insufficient to cause their opening.

The temperature evolution is determined in the average and hot core channels throughout the period of analysis. As the reduction in core flow is very small, following the occurrence of the break, the temperatures change only slightly compared to the steady state situation. After several minutes the temperature remains nearly constant for the coolant in the average and hot channels.

16.11.4.3.1.5 Break at Primary Cooling System Pump Discharge Line

This sequence starts with a break in the drainage line at the PCS pump discharge line. As in the previous case, the core flow reduction is very low but in this case the break flow is also very low. Therefore, the behaviour is similar, but very extended in time.

16.11.4.3.1.6 Primary Cooling System Pump Outlet Break With Pumps On. Parametric Analysis

Although they are considered very unlikely, the behaviour of the reactor after larger size breaks has been analysed to show that the reactor can continue to be cooled by natural circulation at the time the pool coolant level reaches the siphon breakers. Several break sizes at PCS pump outlet have been analysed.

Results for mass flow, pressure drop, time and remaining core power for all break sizes, have been both calculated and extrapolated. For full breakage of the line and FSS reactor trip on low pool water level, the decay heat being generated by the core when the pool water level reaches that of the siphon breaker is sufficiently low that no damage is caused to the Fuel Assemblies. The analysis shows that the reactor can withstand the consequences of the full break of the PCS pump discharge line. This is the point with the highest pressure along the circuit and it bounds breaks at any other point in the PCS.

16.11.4.3.2 Reactor and Service Pools Cooling System

Two break sizes at different locations in the RSPCS were analysed. The first transient analysed was caused by the full break of the diameter pipe that connects the RSPCS pump with the standby pump. This pipe is located on the pump outlet pipe. The other transient analysed is a full break in the pool inlet pipe. Both transients were modelled with RELAP 5.

16.11.4.3.2.1 Break in the Pipe Connecting the Main RSPCS Pump with the Standby Pump

The sequence begins with a break at the RSPCS pump outlet pipe. The flow rate through the break increases quickly but is lower than that in the case of a break in the PCS because the pump outlet pressure is lower for the RSPCS circuit. When the pool water level reaches the trip set point, the FSS is triggered. The time evolution of power dissipated by the irradiation rigs has been analysed. After the reactor is shutdown, the rigs' temperature falls rapidly to a value that depends on the balance between the heat being generated and the capacity of the coolant to remove it. From this point onwards, the temperature difference across the rigs is very low because the rig is generating decay heat and the pump is on. The initial difference between centre and surface temperature is greater for the generic rig as a consequence of its size, although the surface temperature is higher for a hot rig.

Analysis shows that the evolution of the temperature of the coolant at the inlet and outlet of the rigs follows the evolution of the cladding and the coolant inside the channel. The flap valves do not open while the pump is on, because the reduction in RSPCS flow is insufficient to permit the flap valves to open.

As a consequence of the break, there is a reduction in the pump outlet pressure. Following the pump curve, this pressure reduction causes an increase in flow, which in turn causes the pump to speed up. After the FSS trip, the heat dissipated by the rigs drops rapidly. As the pump continues running, the temperature difference between the hot and cold branches is very small. The reduction in overall temperature causes an increase in the coolant density, which in turn reduces the pump speed.

16.11.4.3.2.2 Break at the Pool Inlet Pipe

The sequence starts with a full break of equivalent diameter at the RSPCS pool inlet pipe. Analysis shows that the flow through the break rises quickly. The flow rate is much greater than the one in the previous transient because, although the pressure is lower, the break diameter is much larger. At the time the water level reaches the siphon breaker the rigs are dissipating very little heat. This level of heat can be removed by natural circulation. Analysis has shown the evolution of the temperatures of the average and hot rigs, respectively. All the temperatures remain well below safety limits.

16.11.4.4 Radiological Impact Analysis

No damage to either the core or the rigs occurs as a consequence of a loss of coolant accident. The only variation with respect to normal operation is the presence of pool water in the vicinity of the leak. In the DBIEs postulated, the pool water would spill into one of the pump rooms where it would be collected in the LOCA sump and drained into the LOCA pool. The LOCA drainage system is sized to accommodate the flow rate through a break at the pump discharge line with pumps running. The analysis gives no credit to the RCMS signal that would stop the pumps on low pool water level. On the basis of the results of the calculations, for the worst design basis PCS LOCA, no flooding is expected in the pump room and all the water remains inside the Containment. Therefore the dose to the public would be negligible. There would be no need to access the pump room while the transient is occurring. Once the pumps have stopped and the flow through the break has been interrupted, the operators would have to access the room for repair activities.

16.11.4.5 Natural Circulation after LOCA

A numerical analysis has been performed of the cooling of the core under natural circulation following a LOCA. The analysis considers the limiting case arising from a LOCA where the water is drained down to the level, uncovering the Flap Valves and preventing their use in natural circulation cooling of the core. The natural circulation loop analysed utilises one of the Flap Valves. The analysis is taken to the point at which evaporation from the pool would lead to the uncovering of the Flap Valves and the cessation of natural circulation

To analyse the natural circulation after LOCA, the postulated event under analysis is the break of the inlet pipe of the RSPCS as it gives the fastest time for pool drainage (Section 16.11.4.3.2.2), thus providing a conservative scenario that bounds other transients.

16.11.4.5.1 Transient

In this section a qualitative description of events is presented to explain the phenomena involved.

Comments and analysis are referred to the hot channel as it summarises the most limiting conditions.

The transient analysis is split in two stages,

a) corresponding to a quick succession of events lasting a few minutes, during which the pipe breaks, the reactor trips, the PCS pumps stop and the pool drains and

b) a longer stage lasting hours in which natural convection is established through the core and the pool water heats up.

The transient description that follows is based on the calculations and analysis included in section 16.11.3.

For the sake of understanding the reactor pool is divided in different regions with the following characteristics:

a) <u>Region A</u> corresponds to the region between the open end of the pool and the Siphon Breakers .

b) <u>Region B</u> is the region between the flap valves.

c) <u>Region C</u> is the region below the flap valves and the upper edge of the chimney.

d) <u>Region D</u> considers the rest of the pool, i.e. all the water volume below the upper edge of the chimney level.

16.11.4.5.1.1 First Stage

Once that the RSPCS inlet pipe breaks, the reactor shuts down and the PCS pumps stop. The core flow decreases smoothly due to the inertia flywheels ensuring the forced convection cooling of the core until the flap valves open. Due to the sudden pool drainage it can be considered that the hot water layer remains intact so, the water temperature entering the flap valves is that of the hot water layer.

Since the time scale for natural circulation in the loop is much smaller than those associated with the decay of core power and changes to bulk coolant conditions, core heat removal by natural convection can be viewed as a quasi-stationary process.

Steady state calculations have been performed for the decay power at the moment the flap valves open and show that natural circulation is established through the core. During the transient, the irradiation facilities are cooled by the forced convection provided by the pump inertia until the flap valves in the open. When the SB level is reached the pool water drainage stops, it means that region A disappeared.

16.11.4.5.1.2 Second Stage

During the first part of this stage the decay heat, both in the core and in the irradiation facilities, is removed by several mechanisms:

a) Nucleate boiling mode in the hot channel (which would also enhance natural circulation)

b) Coolant flowing by natural circulation through the core and through the irradiation facilities

c) Conduction across the structures

d) Natural convection in region C as a consequence of the temperature difference between the pool water in region C and the reactor chimney structure and the Primary Cooling System (PCS) pipes

e) Heating of the water mass in region B due to the core natural circulation flow

f) Evaporation at the top of the pool

At the beginning of this stage single-phase natural circulation is established.

The natural circulation is driven by the cooler (relative to the core outlet) reactor pool water entering through the flap valves and being drawn downwards through the primary cooling system pipes towards the core inlet plenum, and by the hotter water leaving the core and filling the chimney extending above the top of the core.

As the transient proceeds and the decay power decreases, the water temperature in Region B increases with the system remaining under single-phase natural circulation. Three hours after reactor shutdown, the temperature in the region B reaches the saturation temperature condition.

At this time, the decay heat is low enough for the core to be cooled by a natural circulation regime in single phase. The core inlet temperature is limited to saturation temperature at atmospheric pressure; the outlet temperature at the top of the core is lower than saturation pressure considering the hydrostatic column. As the coolant rises in the chimney, bubbles of steam are generated as a result of the change in hydrostatic pressure. (Note that these bubbles would reduce the average density of the rising fluid, serving to further enhance the rate of natural circulation.) The steam bubbles would reach the surface of the pool and leave it.

The evolutions of the pool temperature in Region B has been assessed and the maximum heat flux in the hot channel determined. This heat flux is compared with the estimated heat flux (for the same core flow) where significant void fraction begins to appear (Onset of Void Departure, OVD[1]). (Note that, with the adoption of the Saha-Zuber correlation, void departure is equivalent to void survival following departure. Bubble departure can be expected prior to this. The neglecting of this is a further conservatism in the analysis.)

[1] Saha, P. And Zuber, N., "Point of Net Vapor Generation and Vapor Void Fraction in Subcooled Boiling", Proc. Fifth Int. Heat Transfer Conf., Vol IV (1974).

The results show that only nucleate boiling occurs in the channels, and that the core flow is sufficient to preclude the onset of significant voidage

A simple check of the results can be performed for when the hot layer of the pool (Region B) reaches the saturation value, based on experimental results for the low flow condition. Making the conservative assumption that the measured pressure drop by friction in a Fuel Assembly is applicable during the natural circulation cooling process and balanced by the buoyancy force, it is possible to calculate the core outlet temperature producing that buoyancy force. An energy balance then allows an estimate to be made of the decay heat being removed from the core.

The total core flow is calculated using the core power.

The average power per FA when the pool temperature reaches 100°C has been calculated.

It has been shown that saturation temperature is not reached in the core.

A similar behaviour takes place at the irradiation facilities. Although the temperature in region C increases, the water coming from the irradiation positions due to the density differences is warmer and creates convective streams enhancing the heat removal along this region.

In the long term, the water mass in region B evaporates, the water level decreases and, without further water addition, the natural circulation loop is broken. At this time into the transient, although region B disappeared, there is still a water layer above the top of the chimney that continues evaporating and removing the decay heat.

Once the pool water level reaches the top of the chimney, 8 days after reactor shutdown, the Emergency Make-up Water System (EMWS) injects a water flow to compensate water evaporation and leakage from the chimney.

16.11.4.5.1.3 Conclusions for Natural Circulation after LOCA

The purpose of the work reported here was to extend the analysis of natural circulation cooling while shutdown beyond that reported in the SAR. The bounding transient analysed was that resulting from a LOCA in the RSPCS outside the Reactor Pool that drains the water and uncovers the Flap Valves.

The analysis has shown that, even in the extreme case of inlet water temperature, natural circulation is effective in removing the total core decay heat without the occurrence of burnout. The hydrostatic pressure due to the head of pool water provides a sufficient degree of subcooling to the core inlet flow to prevent the onset of significant voidage within the core. The high chimney also makes a significant contribution to the core cooling flow.

Evaporation from the top of the Reactor Pool would eventually lead to uncover the Flap Valves and the interruption of natural circulation. The water layer above the top of the chimney continues evaporating and removing the decay heat

In the long term, 8 days after reactor shutdown when the upper edge of the chimney is reached, the EMWS injects water to compensate coolant evaporation and leakage.

16.11.4.6 Conclusions

The bounding events for this DBA grouping involve failures of PCS and RSPCS pipework. Operation of only a single flap valve is assumed. Either RPS is capable of shutting down the reactor. Heat removal from the core and rigs is adequate. It is concluded that nuclear safety is guaranteed for all credible events involving Loss of Coolant.

End of Section