Replacement Research Reactor Project

OPERATIONAL LIMITS AND CONDITIONS FOR THE OPAL REACTOR – SUMMARY FOR PUBLIC RELEASE

Prepared By

Australian Nuclear Science and Technology Organisation

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PREFACE

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

TABLE OF CONTENTS – OLC

1.0 USE AND APPLICATION

1.1 Definitions

--NOTE--
The defined terms of this section appear in capitalised type and are applicable throughout these Operational Limits and Conditions (OLC) and Bases.

Term	Definition
CHANNEL CALIBRATION	The adjustment, as necessary, of the channel output such that it responds within the necessary range and accuracy to known values of the parameter that the channel monitors. The CHANNEL CALIBRATION shall encompass all devices in the channel required for channel OPERABILITY and the CHANNEL FUNCTIONAL TEST. Calibration of instrument channels with resistance temperature detector (RTD) or thermocouple sensors may consist of an in-place qualitative assessment of sensor behaviour and normal calibration of the remaining adjustable devices in the channel. The CHANNEL CALIBRATION may be performed by means of any series of sequential, overlapping, or total channel steps.
CHANNEL CHECK	The qualitative assessment, by observation, of channel behaviour during operation. This determination shall include, where possible, comparison of the channel indication and status to other indications or status derived from independent instrument channels measuring the same parameter.
CHANNEL FUNCTIONAL TEST	The injection of a simulated or actual signal into the channel as close to the sensor as practicable to verify OPERABILITY of all devices in the channel required for channel OPERABILITY. The CHANNEL FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total channel steps.
CORE ALTERATION	The movement of any fuel or reactivity control components within the reactor core with fuel in the core. Normal control rod movement during the POWER or PHYSICS TEST STATE is an exception and is not considered to be a CORE ALTERATION. Suspension of CORE ALTERATIONS shall not preclude completion of movement of a component to a safe position.
ENSURE	Perform a check to determine compliance with the requirement, bring to compliance if in non-compliance record and report the result.

1.1 Definitions (continued)

FIRST REACTOR PROTECTION SYSTEM (FRPS) RESPONSE TIME	The FRPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its FRPS trip setpoint at the channel sensor until de-energisation of the control rod drive electromagnets. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
INOPERABLE	A system, subsystem, component, or device shall be INOPERABLE when it is not capable of performing its specified safety function(s). This includes when any of the necessary supporting services that are required for the system, subsystem, component, or device to perform its specified safety function(s) are not capable of performing their related support function(s) or when the system, subsystem, component or device setting is not within its allowable value.
ISOLATION SYSTEM RESPONSE TIME	The ISOLATION SYSTEM RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its isolation initiation setpoint at the channel sensor until the isolation valves travel to their required positions. Times shall include diesel generator starting, where applicable. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
LOGIC SYSTEM FUNCTIONAL TEST	A LOGIC SYSTEM FUNCTIONAL TEST shall be a test of all logic components required for OPERABILITY of a logic circuit, from as close to the sensor as practicable up to, but not including, the actuated device, to verify OPERABILITY. The LOGIC SYSTEM FUNCTIONAL TEST may be performed by means of any series of sequential, overlapping, or total system steps so that the entire logic system is tested.
OPERABLE – OPERABILITY	A system, subsystem, component, or device shall be OPERABLE or have OPERABILITY when it is capable of performing its specified safety function(s) and when all necessary attendant instrumentation, controls, normal or emergency electrical power, cooling and seal water, lubrication, and other auxiliary equipment that are required for the system, subsystem, component, or device to perform its specified safety function(s) are also capable of performing their related support function(s). This includes the system, subsystem, component or device setting being within its allowable value where appropriate.
POWER PEAKING FACTOR (PPF)	The PPF shall be the ratio between the maximum heat flux at any point in the reactor core and the average heat flux over the entire reactor core.

1.1 Definitions (continued)

RECENTLY IRRADIATED	In the case of Bulk Irradiation Facility (BIF) rigs, within [6] hours of reactor shutdown or removal from Reactor Core. In the case of Fuel Assemblies, within [15] days of reactor shutdown.
REFLECTOR ALTERATIONS	The movement of any Bulk Irradiation Facility rig within the REFLECTOR VESSEL with fuel in the core. Normal rig movement during the POWER or PHYSICS TEST STATE is an exception and is not considered to be a REFLECTOR ALTERATION. Suspension of REFLECTOR ALTERATIONS shall not preclude completion of movement of a component to a safe position.
SECOND REACTOR PROTECTION SYSTEM (SRPS) RESPONSE TIME	The SRPS RESPONSE TIME shall be that time interval from when the monitored parameter exceeds its SRPS initiation setpoint at the channel sensor until the Second Shutdown System discharge valves travel to their required positions. The response time may be measured by means of any series of sequential, overlapping, or total steps so that the entire response time is measured.
SHUTDOWN MARGIN (SDM-1)	The amount of negative reactivity that is or would be provided in addition to that negative reactivity necessary to maintain the reactor in a subcritical condition without time limit assuming the reactor is cold (20° C) and xenon free, all irradiation rigs and targets are in their most reactive state and assuming the most reactive control rod is fully withdrawn.
SHUTDOWN REACTOR	Shutdown reactor by Bank Insertion of Control Rods (or by Trip 1 if bank insertion is not available, or Trip 2 if both these are not available), and de-energise Control Rod Drives by selecting Disable Reactor Operation on RPS Configuration panel.
STATE	A STATE shall correspond to any one inclusive combination of CORE ALTERATION state, control rod drive status, k_{eff}, and core cooling condition, specified in Table 1.1-1 with fuel in the reactor core.
VERIFY	Perform a check to determine compliance with the requirement, record the result and report non-compliance.

Table 1.1-1 (page 1 of 1)
STATES

STATE	K_{eff}	CONTROL ROD DRIVE ELECTROMAGNET STATUS	CORE COOLING CONDITION
POWER[(a)]	< 1.0 or ≥ 1.0	Energised	Forced Circulation
PHYSICS TEST[(a)]	< 1.0 or ≥ 1.0	Energised	Natural Circulation
SHUTDOWN[(a)]	< 1.0	De-energised	Forced or Natural Circulation
REFUELLING	< 1.0	De-energised	Natural Circulation

(a) CORE ALTERATIONS permitted in REFUELLING STATE only.

1.0 USE AND APPLICATION

1.2 Logical Connectors

PURPOSE

The purpose of this section is to explain the meaning of logical connectors.

Logical connectors are used in Operational Limits and Conditions (OLCs) to discriminate between, and yet connect, discrete Conditions, Required Actions, Completion Times, Surveillances, and Frequencies. The only logical connectors that appear in OLCs are AND and OR. The physical arrangement of these connectors constitutes logical conventions with specific meanings.

BACKGROUND

Several levels of logic may be used to state Required Actions. These levels are identified by the placement (or nesting) of the logical connectors and by the number assigned to each Required Action. The first level of logic is identified by the first digit of the number assigned to a Required Action and the placement of the logical connector in the first level of nesting (i.e., left justified with the number of the Required Action). The successive levels of logic are identified by additional digits of the Required Action number and by successive indentions of the logical connectors.

When logical connectors are used to state a Condition, Completion Time, Surveillance, or Frequency, only the first level of logic is used, and the logical connector is left justified with the statement of the Condition, Completion Time, Surveillance, or Frequency.

EXAMPLES

The following examples illustrate the use of logical connectors.

EXAMPLE 1.2-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LC not met.	A.1 Verify . . . AND A.2 Restore . . .	

In this example the logical connector AND is used to indicate that when in Condition A, both Required Actions A.1 and A.2 must be completed.

1.2 Logical Connectors (continued)

EXAMPLES (continued)

EXAMPLE 1.2-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. LC not met.	A.1 Trip . . . OR A.2.1 Verify . . . AND A.2.2.1 Reduce . . . OR A.2.2.2 Perform . . . OR A.3 Align . . .	

This example represents a more complicated use of logical connectors. Required Actions A.1, A.2, and A.3 are alternative choices, only one of which must be performed as indicated by the use of the logical connector OR and the left justified placement. Any one of these three Actions may be chosen. If A.2 is chosen, then both A.2.1 and A.2.2 must be performed as indicated by the logical connector AND. Required Action A.2.2 is met by performing A.2.2.1 or A.2.2.2. The indented position of the logical connector OR indicates that A.2.2.1 and A.2.2.2 are alternative choices, only one of which must be performed.

1.0 USE AND APPLICATION

1.3 Completion Times

PURPOSE	The purpose of this section is to establish the Completion Time convention and to provide guidance for its use.
BACKGROUND	Limiting Conditions (LCs) specify minimum requirements for ensuring safe operation of the unit. The ACTIONS associated with an LC state Conditions that typically describe the ways in which the requirements of the LC can fail to be met. Specified with each stated Condition are Required Action(s) and Completion Times.
DESCRIPTION	The Completion Time is the amount of time allowed for completing a Required Action. It is referenced to the time of discovery of a situation (e.g., inoperable equipment or variable not within limits) that requires entering an ACTIONS Condition unless otherwise specified, providing the unit is in a STATE or specified condition stated in the Applicability for the LC. Required Actions must be completed prior to the expiration of the specified Completion Time. An ACTIONS Condition remains in effect and the Required Actions apply until the Condition no longer exists or the reactor is not within the LC Applicability. If situations are discovered that require entry into more than one Condition at a time within a single Operational Limits and Condition (OLC) (multiple Conditions), the Required Actions for each Condition must be performed within the associated Completion Time. When in multiple Conditions, separate Completion Times are tracked for each Condition starting from the time of discovery of the situation that required entry into the Condition. Once a Condition has been entered, subsequent divisions, subsystems, components, or variables expressed in the Condition, discovered to be inoperable or not within limits, will <u>not</u> result in separate entry into the Condition, unless specifically stated. The Required Actions of the Condition continue to apply to each additional failure, with Completion Times based on initial entry into the Condition.
EXAMPLES	The following examples illustrate the use of Completion Times with different types of Conditions and changing Conditions.

1.3 Completion Times (continued)

EXAMPLES (continued)

EXAMPLE 1.3-1

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One channel inoperable.	A.1 Implement alternate means of monitoring.	24 hours
	AND	
	A.2 Restore channel to OPERABLE status.	30 days

Condition A has two Required Actions. Each Required Action has its own separate Completion Time. Each Completion Time is referenced to the time that Condition A is entered.

The Required Actions of Condition A are to implement alternate means of monitoring within 24 hours and restore the channel to OPERABLE status within 30 days. A total of 24 hours is allowed to implement the alternate means of monitoring and a total of 30 days (not 31 days, i.e., 30 days + 24 hours) is allowed for restoring the channel to OPERABLE status from the time Condition A is entered. If the alternate means of monitoring is established within 12 hours, the time allowed for restoring the channel to OPERABLE status is 29 days and 12 hours because the total time for restoring the channel to OPERABLE status is 30 days.

1.3 Completion Times (continued)

EXAMPLES (continued)

EXAMPLE 1.3-2

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One pump inoperable.	A.1 Restore pump to OPERABLE status.	7 days
B. Required Action and associated Completion Time not met.	B.1 Be in SHUTDOWN STATE.	1 hour

When a pump is declared inoperable, Condition A is entered. If the pump is not restored to OPERABLE status within 7 days, Condition B is also entered and the Completion Time clocks for Required Action B.1 start. If the inoperable pump is restored to OPERABLE status after Condition B is entered, Conditions A and B are exited, and therefore, the Required Actions of Condition B may be terminated.

When a second pump is declared inoperable while the first pump is still inoperable, Condition A is not re-entered for the second pump. LC 3.0.3 is entered, because the ACTIONS do not include a Condition for more than one inoperable pump. The Completion Time clock for Condition A does not stop after LC 3.0.3 is entered, but continues to be tracked from the time Condition A was initially entered.

While in LC 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has not expired, LC 3.0.3 may be exited and operation continued in accordance with Condition A.

While in LC 3.0.3, if one of the inoperable pumps is restored to OPERABLE status and the Completion Time for Condition A has expired, LC 3.0.3 may be exited and operation continued in accordance with Condition B. The Completion Time for Condition B is tracked from the time the Condition A Completion Time expired.

On restoring one of the pumps to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first pump was declared inoperable.

1.3 Completion Times (continued)

EXAMPLES (continued)

EXAMPLE 1.3-3

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve(s) to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in SHUTDOWN STATE.	1 hour

A single Completion Time is used for any number of valves inoperable at the same time. The Completion Time associated with Condition A is based on the initial entry into Condition A and is not tracked on a per valve basis. Declaring subsequent valves inoperable, while Condition A is still in effect, does not trigger the tracking of separate Completion Times.

Once one of the valves has been restored to OPERABLE status, the Condition A Completion Time is not reset, but continues from the time the first valve was declared inoperable.

If the Completion Time of 4 hours expires while one or more valves are still inoperable, Condition B is entered.

1.3 Completion Times (continued)

EXAMPLES (continued)

EXAMPLE 1.3-4

ACTIONS

---NOTE---
Separate Condition entry is allowed for each inoperable valve.

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One or more valves inoperable.	A.1 Restore valve to OPERABLE status.	4 hours
B. Required Action and associated Completion Time not met.	B.1 Be in SHUTDOWN STATE.	1 hour

The Note above the ACTIONS Table is a method of modifying how the Completion Time is tracked. If this method of modifying how the Completion Time is tracked was applicable only to a specific Condition, the Note would appear in that Condition rather than at the top of the ACTIONS Table.

The Note allows Condition A to be entered separately for each inoperable valve, and Completion Times tracked on a per valve basis. When a valve is declared inoperable, Condition A is entered and its Completion Time starts. If subsequent valves are declared inoperable, Condition A is entered for each valve and separate Completion Times start and are tracked for each valve.

If the Completion Time associated with a valve in Condition A expires, Condition B is entered for that valve. If the Completion Times associated with subsequent valves in Condition A expire, Condition B is entered separately for each valve and separate Completion Times start and are tracked for each valve. If a valve that caused entry into Condition B is restored to OPERABLE status, Condition B is exited for that valve.

1.3 Completion Times (continued)

EXAMPLES (continued)

EXAMPLE 1.3-5

ACTIONS

CONDITION	REQUIRED ACTION	COMPLETION TIME
A. One subsystem inoperable.	A.1 Verify affected subsystem isolated.	1 hour AND Once per 8 hours thereafter
	AND A.2 Restore subsystem to OPERABLE status.	24 hours
B. Required Action and associated Completion Time not met.	B.1 Be in SHUTDOWN STATE.	1 hour

Required Action A.1 has two Completion Times. The 1 hour Completion Time begins at the time the Condition is entered and each "Once per 8 hours thereafter" interval begins upon performance of Required Action A.1.

If after Condition A is entered, Required Action A.1 is not met within either the initial 1 hour or any subsequent 8 hour interval from the previous performance (plus the extension allowed by SR 3.0.2), Condition B is entered. The Completion Time clock for Condition A does not stop after Condition B is entered, but continues from the time Condition A was initially entered. If Required Action A.1 is met after Condition B is entered, Condition B is exited and operation may continue in accordance with Condition A, provided the Completion Time for Required Action A.2 has not expired.

IMMEDIATE COMPLETION TIME

When “Immediately” is used as a Completion Time, the Required Action should be pursued without delay and in a controlled manner.

1.0 USE AND APPLICATION

1.4 Frequency

PURPOSE	The purpose of this section is to define the proper use and application of Frequency requirements.
DESCRIPTION	Each Surveillance Requirement (SR) has a specified Frequency in which the Surveillance must be addressed in order to meet the associated LC. An understanding of the correct application of the specified Frequency is necessary for compliance with the SR.

The "specified Frequency" is referred to throughout this section and each of the Specifications of Section 3.0.2, Surveillance Requirement (SR) Applicability. The "specified Frequency" consists of the requirements of the Frequency column of each SR as well as certain Notes in the Surveillance column that modify performance requirements.

Sometimes special situations dictate when the requirements of a Surveillance are to be met. They are "otherwise stated" conditions allowed by SR 3.0.1. They may be stated as clarifying Notes in the Surveillance, as part of the Surveillance, or both.

Situations where a Surveillance could be required (i.e., its Frequency could expire), but where it is not possible or not desired that it be performed until sometime after the Applicability for the associated LC is entered, represent potential SR 3.0.4 conflicts. To avoid these conflicts, the SR (i.e., the Surveillance or the Frequency) is stated such that it is only "required" when it can be and should be performed. With an SR satisfied, SR 3.0.4 imposes no restriction.

The use of "met" or "performed" in these instances conveys specific meanings. A Surveillance is "met" only when the acceptance criteria are satisfied. Known failure of the requirements of a Surveillance, even without a Surveillance specifically being "performed," constitutes a Surveillance not "met." "Performance" refers only to the requirement to specifically determine the ability to meet the acceptance criteria.

Some Surveillances contain notes that modify the Frequency of performance or the conditions during which the acceptance criteria must be satisfied. For these Surveillances, the STATE-entry restrictions of SR 3.0.4 may not apply. Such a Surveillance is not required to be performed prior to entering a STATE or other specified condition in the Applicability for the associated LC if any of the following three conditions are satisfied:

1.4 Frequency (continued)

DESCRIPTION (continued)

a. The Surveillance is not required to be met in the STATE or other specified condition to be entered; or

b. The Surveillance is required to be met in the STATE or other specified condition to be entered, but has been performed within the specified Frequency (i.e., it is current) and is known not to be failed; or

c. The Surveillance is required to be met, but not performed, in the STATE or other specified condition to be entered, and is known not to be failed.

Examples 1.4-3, 1.4-4, and 1.4-5 discusses these special situations.

EXAMPLES

The following examples illustrate the various ways that Frequencies are specified. In these examples, the Applicability for the LC (LC not shown) is POWER, PHYSICS TEST, and SHUTDOWN STATES.

EXAMPLE 1.4-1

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Perform CHANNEL CHECK.	24 hours

Example 1.4-1 contains the type of SR most often encountered in the Operational Limits and Conditions (OLCs). The Frequency specifies an interval (24 hours) during which the associated Surveillance must be performed at least one time. Performance of the Surveillance initiates the subsequent interval. Although the Frequency is stated as 24 hours, an extension of the time interval to 1.25 times the interval specified in the Frequency is allowed by SR 3.0.2 for operational flexibility. The measurement of this interval continues at all times, even when the SR is not required to be met per SR 3.0.1 (such as when the equipment is inoperable, a variable is outside specified limits, or the reactor is outside the Applicability for the LC). If the interval specified by SR 3.0.2 is exceeded while the reactor is in a STATE or other specified condition in the Applicability for the LC, and the performance of the Surveillance is not otherwise modified (refer to Example 1.4-3), then SR 3.0.3 becomes applicable.

1.4 Frequency (continued)

EXAMPLES (continued)

If the interval as specified by SR 3.0.2 is exceeded while the reactor is not in a STATE or other specified condition in the Applicability for the LC for which performance of the SR is required, the Surveillance must be performed within the Frequency requirements of SR 3.0.2 prior to entry into the STATE or other specified condition. Failure to do so would result in a violation of SR 3.0.4.

EXAMPLE 1.4-2

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
Verify flow is within limits.	Once within 12 hours after reactor power ≥ 400 kW AND 24 hours thereafter

Example 1.4-2 has two Frequencies. The first is a one time performance Frequency, and the second is of the type shown in Example 1.4-1. The logical connector "AND" indicates that both Frequency requirements must be met. Each time reactor power is increased from a power level < 400 kW to ≥ 400 kW, the Surveillance must be performed within 12 hours.

The use of "once" indicates a single performance will satisfy the specified Frequency (assuming no other Frequencies are connected by "AND"). This type of Frequency does not qualify for the 25% extension allowed by SR 3.0.2. "Thereafter" indicates future performances must be established per SR 3.0.2, but only after a specified condition is first met (i.e., the "once" performance in this example). If reactor power decreases to < 400 kW, the measurement of both intervals stops. New intervals start upon reactor power reaching 400 kW.

1.4 Frequency (continued)

EXAMPLES (continued)

EXAMPLE 1.4-3

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
---------------------------NOTE--------------------------- Only required to be met in POWER STATE. -- Verify leakage rates are within limits.	24 hours

Example 1.4-3 specifies that the requirements of this Surveillance do not have to be met until the reactor is in the POWER STATE. The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), but the reactor was not in the POWER STATE, there would be neither failure of the SR nor failure to meet the LC. Therefore, no violation of SR 3.0.4 occurs when changing STATES, even with the 24 hour Frequency exceeded, provided the STATE change was not made into the POWER STATE. Prior to entering the POWER STATE (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

EXAMPLE 1.4-4

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
---------------------------NOTE--------------------------- Only required to be performed in POWER STATE. -- Perform complete cycle of the valve.	7 days

1.4 Frequency (continued)

EXAMPLES (continued)

The interval continues, whether or not the reactor operation is in the POWER STATE, PHYSICS TEST STATE, or SHUTDOWN STATE (the assumed Applicability for the associated LC) between performances.

As the Note modifies the required performance of the Surveillance, the Note is construed to be part of the "specified Frequency." Should the 7 day interval be exceeded while operation is not in the POWER STATE, this Note allows entry into and operation in the PHYSICS TEST and SHUTDOWN STATES to perform the Surveillance. The Surveillance is still considered to be performed within the "specified Frequency" if completed prior to entering the POWER STATE. Therefore, if the Surveillance were not performed within the 7 day (plus the extension allowed by SR 3.0.2) interval, but operation was not in the POWER STATE, it would not constitute a failure of the SR or failure to meet the LC. Also, no violation of SR 3.0.4 occurs when changing STATES, even with the 7 day Frequency not met, provided operation does not result in entry into the POWER STATE.

Once the reactor reaches the POWER STATE, the requirement for the Surveillance to be performed within its specified Frequency applies and would require that the Surveillance had been performed. If the Surveillance were not performed prior to entering the POWER STATE, there would then be a failure to perform a Surveillance within the specified Frequency, and the provisions of SR 3.0.3 would apply.

EXAMPLE 1.4-5

SURVEILLANCE REQUIREMENTS

SURVEILLANCE	FREQUENCY
----------------------------NOTE---------------------------- Not required to be met in SHUTDOWN STATE. --- Verify parameter is within limits.	24 hours

1.4 Frequency (continued)

EXAMPLES (continued)

Example 1.4-5 specifies that the requirements of this Surveillance do not have to be met while the reactor is in the SHUTDOWN STATE (the assumed Applicability for the associated LC is the POWER, PHYSICS TEST, and SHUTDOWN STATES). The interval measurement for the Frequency of this Surveillance continues at all times, as described in Example 1.4-1. However, the Note constitutes an "otherwise stated" exception to the Applicability of this Surveillance. Therefore, if the Surveillance were not performed within the 24 hour interval (plus the extension allowed by SR 3.0.2), and the reactor was in the SHUTDOWN STATE, there would be neither failure of the SR nor failure to meet the LC. Therefore, no violation of SR 3.0.4 occurs when changing STATES to enter the SHUTDOWN STATE, even with the 24 hour Frequency exceeded, provided the STATE change does not result in entry into the POWER or TEST STATE (assuming again that the 24 hour Frequency were not met), SR 3.0.4 would require satisfying the SR.

2.0 SAFETY LIMITS (SL)

2.1 Reactor Core Safety Limits

2.1.1 Reactor Power

2.1.1.1 During POWER STATE, reactor power and reactor core flow shall be maintained below and to the right of the Safety Limit shown in Figure 1.1-1.

2.1.1.2 During PHYSICS TEST STATE, the reactor power shall be maintained less than 1400 kW.

2.1.2 Reactor Pool Water Level

During PHYSICS TEST, SHUTDOWN, and REFUELLING STATES, the Reactor Pool water level shall be ≥ + 5.8 m.

2.2 Safety Limit Violations

With any Safety Limit violation, the following actions shall be taken.

2.2.1 In the event of non-compliance with Reactor Core Safety Limit 2.1.1, immediately shutdown the reactor by manually initiating Trip 1 (or Trip 2 if Trip 1 unavailable).

2.2.2 In the event of non-compliance with Reactor Core Safety Limit 2.1.2, immediately initiate action to restore compliance with the Safety Limit and continue until compliance with the Safety Limit is achieved.

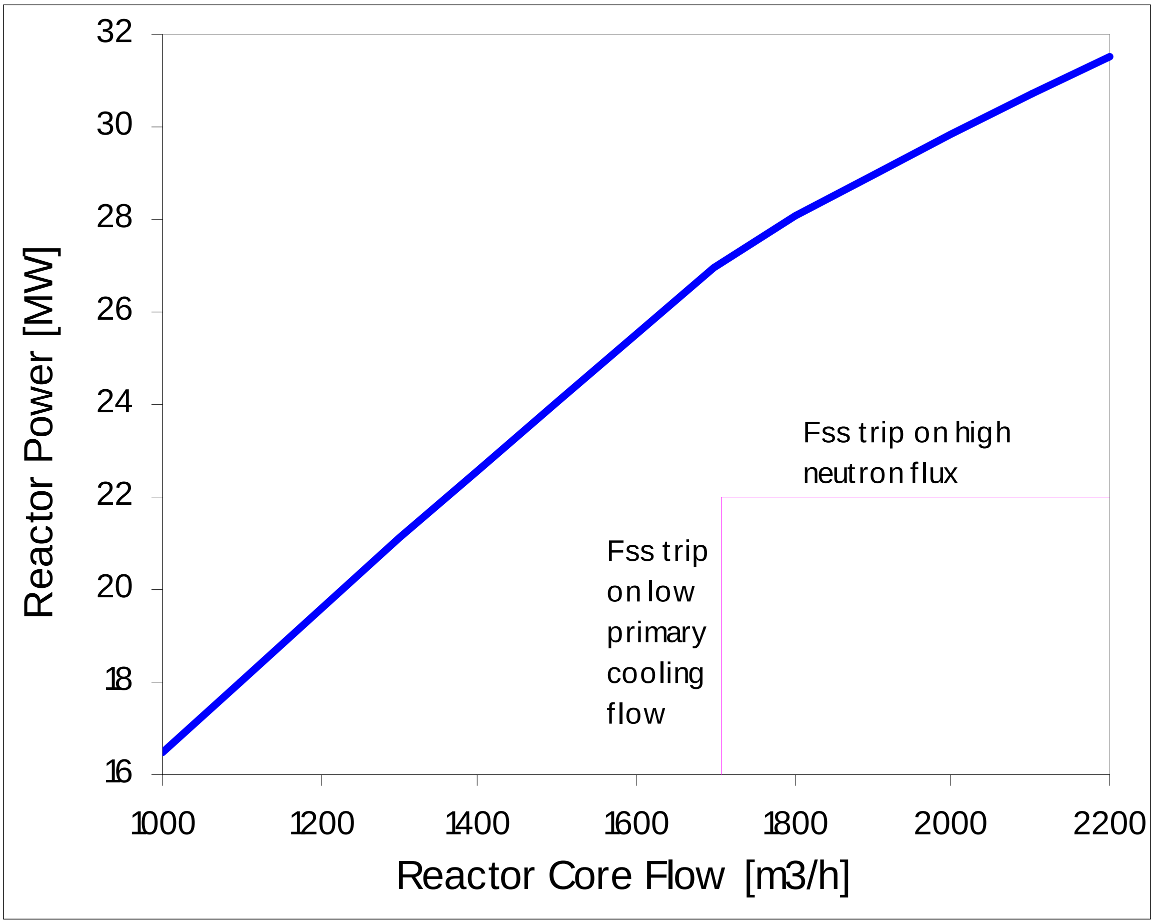

Figure 2.1-1 (page 1 of 1)
POWER STATE Reactor Core Safety Limit

3.0 LIMITING CONDITIONS OF OPERATION (LC)

LC 3.0.1	LCs shall be met during the STATES or other specified conditions in the Applicability, except as provided in LC 3.0.2.
LC 3.0.2	Upon discovery of a failure to meet an LC, the Required Actions of the associated Conditions shall be met, except as provided in LC 3.0.3. If the LC is met or is no longer applicable prior to expiration of the specified Completion Time(s), completion of the Required Action(s) is not required unless otherwise stated.
LC 3.0.3	Equipment removed from service or declared inoperable to comply with ACTIONS may be returned to service under administrative control solely to perform testing required to demonstrate its OPERABILITY or the OPERABILITY of other equipment. This is an exception to LC 3.0.2 for the system returned to service under administrative control to perform the testing required to demonstrate OPERABILITY.
LC 3.0.4	When an LC is not met, entry into a STATE or other specified condition in the Applicability shall not be made except when the associated ACTIONS to be entered permit continued operation in the STATE or other specified condition in the Applicability for an period of time equal to or greater than 7 days. LC 3.0.4 shall not prevent changes in STATES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the reactor.
LC 3.0.5	When a supported system LC is not met solely due to a support system LC not being met, the Conditions and Required Actions associated with this supported system are not required to be entered. Only the support system ACTIONS are required to be entered. This is an exception to LC 3.0.2 for the supported system. If a loss of safety function is determined to exist by OLC 5.2.6, "Safety Function Determination Program (SFDP)", the appropriate Conditions and Required Actions for the LC in which the loss of safety function exists are required to be entered.

3.0 SURVEILLANCE REQUIREMENT (SR) APPLICABILITY

SR 3.0.1	SRs shall be met during the STATES or other specified conditions in the Applicability for individual LCs, unless otherwise stated in the SR. Failure to meet a Surveillance, whether such failure is experienced during the performance of the Surveillance or between performances of the Surveillance, shall be failure to meet the LC. Failure to perform a Surveillance within the specified Frequency shall be failure to meet the LC except as provided in SR 3.0.3. Surveillances do not have to be performed on inoperable equipment or variables outside specified limits.
SR 3.0.2	The specified Frequency for each SR is met if the Surveillance is performed within 1.25 times the interval specified in the Frequency, as measured from the previous performance or as measured from the time a specified condition of the Frequency is met. For Frequencies specified as "once", the above interval extension does not apply. If a Completion Time requires periodic performance on a "once per . . ." basis, the above Frequency extension applies to each performance after the initial performance. Exceptions to SR 3.0.2 are stated in the individual OLCs.
SR 3.0.3	If it is discovered that a Surveillance was not performed within its specified Frequency, then compliance with the requirement to declare the LC not met may be delayed, from the time of discovery, up to 24 hours or up to the limit of the specified Frequency, whichever is greater. This delay period is permitted to allow performance of the Surveillance. A risk evaluation shall be performed for any Surveillance delayed greater than 24 hours and this risk impact shall be managed. If the Surveillance is not performed within the delay period, the LC must immediately be declared not met, and the applicable Condition(s) must be entered. When the Surveillance is performed within the delay period and the Surveillance is not met, the LC must immediately be declared not met, and the applicable Condition(s) must be entered.
SR 3.0.4	Entry into a STATE or other specified condition in the Applicability for an LC shall not be made unless the LC's Surveillances have been met within their specified Frequency. This provision shall not prevent entry into STATES or other specified conditions in the Applicability that are required to comply with ACTIONS or that are part of a shutdown of the reactor. SR 3.0.4 is only applicable for entry into a STATE or other specified condition in the Applicability in the POWER, PHYSICS TEST, and SHUTDOWN STATES.

3.1 REACTIVITY CONTROL SYSTEMS

LC 3.1.1 Shutdown Margin (SDM-1)

This OLC ensures that the SDM-1 limit is consistent with the reactivity design bases provided in the SAR. The limit reflects the amount of negative reactivity required to be available in addition to that amount required to maintain the reactor in a subcritical state assuming the reactor core is cold and xenon free, all irradiation targets are in their most reactive state, and the most reactive control rod is fully withdrawn.

Shutdown Margin requirements are specified to ensure:

a. The reactor can be made subcritical from all operating conditions and transients and Design Basis Accidents;

b. The reactivity transients associated with postulated accident conditions are controllable within acceptable limits; and

c. The reactor will be maintained sufficiently subcritical to preclude inadvertent criticality in the shutdown condition.

These requirements can be satisfied by the First Shutdown System (FSS) or the Second Shutdown System (SSS). Either one of the two systems can compensate for the reactivity effects of the fuel and water temperature changes experienced during all operating conditions.

LC 3.1.2 Reactivity Balance

This OLC ensures that reactor operation is maintained within the assumptions of the safety analyses. Large differences between monitored and predicted core reactivity may indicate that the assumptions of the Design Basis Accident and transient analyses are no longer valid, or that the uncertainties in the numerical methodology are larger than expected. A limit on the reactivity of ± 1000 pcm has been established based on engineering judgment and operational experience. A 1000 pcm deviation in measured excess reactivity from that predicted is larger than is acceptable for normal operation and should therefore be investigated prior to continued operation.

LC 3.1.3 First Shutdown System (FSS)

This OLC ensures that the availability and performance of the FSS in the event of a Design Basis Accident or transient meets the assumptions used in the safety analyses in Chapter 16 of the SAR.

The FSS is the primary means to shut down the reactor. This is accomplished by rapidly inserting the five control plates into the reactor core in response to a demand from the First Reactor Protection System (FRPS). The plates are fixed to rods which in turn are connected to the control drives via electromagnets. The FSS initiates a trip by de-energising the electromagnets and actuating the FSS Pneumatic System.

The plates and rods will fall into the fully inserted position by gravity and meet safety analysis assumptions for control rod insertion times. The FSS pneumatic system, provided for defence-in-depth so as to ensure that the plates and rods will fall into the fully inserted position under all design basis conditions, accelerates the control rods in their fall by injecting compressed air. Each line serving a control rod drive is provided with redundant trip valves that are in a parallel configuration. The trip valves normal position is open and the trip valve solenoids need to be electrically energised to allow the valves to close and permit the control rods to be raised from the fully inserted position. The absorber plates and control rod stems are guided and protected from flow induced forces and seismic induced strains.

The FSS complies with the single failure criterion and as such the insertion of any four of the five control rods is sufficient to bring the reactor to a safe shutdown condition. In the unlikely event of a total failure of the FSS, the Second Shutdown System (SSS) is provided as an alternate, independent and diverse method to shut down the reactor.

LC 3.1.4 Second Shutdown System (SSS)

This OLC ensures that the availability and performance of the SSS in the event of a Design Basis Accident or transient meets the assumptions used in the safety analyses of Chapter 2 of the SAR.

The SSS is the backup means to shut down the reactor. This is accomplished by automatically dumping the heavy water stored in the reflector vessel. This system provides an alternate shutdown system, completely independent to the First Shutdown System (FSS).

The SSS is connected to the reflector vessel through a pipeline that conducts heavy water to a distribution header. The distribution header diverts heavy water through six air-actuated ball valves in a parallel configuration. A header discharges into the heavy water storage (dump) tank belonging to the Reflector Cooling and Purification System (RCPS). Pressure equalisation lines for helium connect the reflector vessel with the heavy water storage tank through the RCPS. Compressed air supply to the air actuated ball valves (SSS discharge valves) is provided from a dedicated compressed air storage tank. The supply of compressed air to the SSS discharge valves is controlled by solenoid valves and loss of electrical supply causes the valves to open, thereby initiating a heavy water dump.

The SSS shuts down the reactor by partially draining the reflector vessel water and leaving the reactor core in a subcritical condition. The absence of moderator (heavy water) in the reflector vessel causes a large decrease in neutron reflection to the reactor core, which results in a subcritical condition.

The SSS complies with the single failure criterion and as such the opening of any five of the six SSS discharge valves ensures an adequate safety action sufficient to bring the core to a safe shutdown condition.

3.2 POWER LIMITS

LC 3.2.1 Power Limit – Power State

This OLC ensures that in the POWER STATE, the reactor THERMAL POWER limited to 21.2 MW, thus ensuring that the results of the Design Basis Initiating Event and transients analyses contained in the SAR will remain valid.

Reactor THERMAL POWER is limited during normal operations to preserve the initial conditions assumed in the analyses for Design Basis Initiating Events and transients. The reactor is designed to be operated in one of two STATES when the reactor is critical. These two STATES are the POWER STATE and the PHYSICS TEST STATE. In the POWER STATE, the reactor core is cooled by forced circulation using two of the Primary Cooling System (PCS) pumps. In the PHYSICS TEST STATE, the reactor core is cooled by natural circulation with the PCS pumps off. This OLC addresses reactor THERMAL POWER requirements for the POWER STATE. Reactor THERMAL POWER requirements for the PHYSICS TEST STATE are addressed in OLC 3.2.2, "Power Limit – Physics Test State."

LC 3.2.2 Power Limit – Physics Test State

This OLC ensures that in the PHYSICS TEST STATE, when the PCS pumps are off and the core is cooled by natural circulation, the reactor THERMAL POWER is limited to 400 kW such that natural circulation is capable of providing adequate reactor core cooling.

LC 3.2.3 Power Peaking Factor (PPF)

This OLC limits reactor operation with the PPF $\leq$ 3.0 so as to ensure that fuel failures due to inadequate cooling do not occur during normal operation or during Design Basis Accidents or transients.

The thermal-hydraulic design is based on the analysis of a “hot channel” (the fuel cooling channel subject to the maximum heat load) and an “average channel” (a fuel cooling channel subject to an average heat load) rather than on a specifically measured or calculated power density distribution. PPF is the ratio between the maximum heat flux at any point in the reactor core and the average heat flux over the entire reactor core. The reactor core power distribution is a function of the reactor core configuration design, reactor core loading, control rod pattern and coolant density. Considering the rather small reactor core size, it exhibits a low PPF, which leads to large safety margins from a thermal-hydraulic point of view. The low PPF is a consequence of two factors that tend to flatten the flux within the reactor core; the use of burnable poison that is distributed in a non-homogeneous way in the reactor core (being placed as wires on the sides of the fuel plates) and heavy water that acts as a neutron reflector.

3.3 INSTRUMENTATION

LC 3.3.1 First Reactor Protection System (FRPS) Trip 1 Instrumentation

The purpose of this OLC is to ensure that all FRPS CHANNELS are capable of providing proper input to their FRPS Safety Functions such that they can initiate signals to shut down the reactor to prevent reactor fuel damage and prevent the release of radioactive material from the reactor pool.

The Reactor Protection Systems (RPS) consists of two functionally independent and diverse protection systems that initiate, among other actions, automatic reactor shutdown. The FRPS uses computer technology qualified for use in safety shutdown applications. This system initiates the fast insertion of control rods (called a Trip 1) via the First Shutdown System (FSS) whenever FRPS monitored parameters exceed pre-established limits. The Second Reactor Protection System (SRPS) (a hard-wired system) initiates the partial draining of heavy water from the reflector vessel (called a Trip 2) via the Second Shutdown System (SSS) whenever SRPS monitored parameters exceed pre-established limits. These actions are aimed at avoiding fuel damage and preventing the release of radioactive material from the reactor pool. These actions can be accomplished either automatically or manually.

The protection and monitoring functions of the FRPS have been designed to ensure safe operation of the reactor. This is achieved by specifying Limiting Safety System Settings (LSSS) in terms of parameters directly monitored by the FRPS, as well as Limiting Conditions (LC) on other reactor system parameters and equipment performance. The LSSS are defined in this OLC as the Allowable Values, which, in conjunction with the Limiting Conditions, establish the threshold for protective system action to prevent exceeding acceptable limits including Safety Limits during Design Basis Accidents or transients.

The Safety Parameters for FRPS Trip 1 Instrumentation are:

1. Start-up Range Log Neutron Flux
2. Wide Range Log Neutron Flux
3. Wide Range Log Neutron Flux Rate
4. Core Temperature Difference
5. Core Inlet Temperature
6. Core Pressure Difference
7. Primary Cooling System Flow
8. Reflector Primary Cooling System Flow
9. RCPS Expansion Tank Level
10. RSPCS Rigs Cooling Flow
11. RSPCS Flap Valves Position
12. Reactor Pool Water Level
13. Pool Top Dose Rate
14. RCMS
15. Normal Power System
16. Seismic Level
17. Cold Neutron Source
18. Manual Trip 1
19. RPS Selected Configuration POWER

20. RPS Selected Configuration PHYSICS TEST

The Safety Parameters for FRPS Interlock Instrumentation are:

1. FSS Pneumatic Tank Pressure
2. SSS Block Isolation Valve Position

Each FRPS Safety Parameter is configured with three independent, redundant measurement CHANNELS. Each of these CHANNELS is composed of a sensor and a conditioning unit if required. The signals are then input to the TRICON system for each Train where a comparison is done to the Safety Function Safety Settings. Using fiber optic communication links, each Train receives the CHANNELS of the other two Trains and hence two-out-of-three voting is done for the same Safety Function on each Train. If a Train determines that two or more CHANNELS have surpassed their Safety Setting for the same Safety Function, a Protection Action is output from that Train. This signal is then combined with the other Trains in a final stage of two-out-of-three voting logic in the First Final Actuation Logic (FFAL) unit which produces initiation of the FSS. The FRPS fails safe on loss of electrical power.

The FRPS also initiates containment isolation and controls the CERS under accident conditions. The protection and monitoring functions associated with these actions are addressed separately in LC 3.3.2.

LC 3.3.2 First Reactor Protection System (FRPS) Containment Instrumentation

As stated under LC 3.3.1 above, the FRPS also performs a function equivalent to an Engineered Safety Features Actuation System in that it actuates containment isolation upon detection of high activity in the ventilation discharge stack and performs various functions associated with the CERS (ie transfer between the duty and standby CERS trains, maximising cooling and tripping ventilation heaters). This LC addresses this function of the FRPS.

The Safety Parameters for FRPS Containment Instrumentation – CERS are:

1. Supply Air Temperature
2. Return Air Temperature
3. Supply Air Pressure Difference
4. Manual Transfer of CERS
5. CERS Select UNIT 1
6. CERS Select UNIT 2
7. CERS Standby Unit ENABLE
8. CERS Control FRPS
9. CERS Control RCMS

The Safety Parameters for FRPS Containment Instrumentation – Containment Isolation are:

1. Stack Particulate Activity
2. Stack Iodine Activity
3. Stack Noble Gas Activity
4. Stack Particulate Activity Rate
5. Stack Iodine Activity Rate
6. Stack Noble Gas Activity Rate
7. Manual Group 1 CLOSE

8. Manual Group 2 CLOSE
9. CIS Group 1 OPEN
10. CIS Group 2 OPEN
11. CIS Group 4 Fire OPEN
12. CIS Group 4 Fire CLOSE
13. CIS Group 4 Vacuum OPEN
14. CIS Group 4 Vacuum CLOSE
15. CIS Group 4 Pressure INTERNAL OPEN
16. CIS Group 4 Pressure INTERNAL CLOSE
17. CIS Group 4 Pressure EXTERNAL OPEN

Each FRPS Safety Parameter is configured with three independent, redundant measurement CHANNELS in the same way as described for LC 3.3.1 above.

LC 3.3.3 Second Reactor Protection System (SRPS) Instrumentation

The purpose of this LC is to ensure that all SRPS CHANNELS are capable of providing proper input to their SRPS Safety Functions such that they can initiate signals to shut down the reactor to prevent reactor fuel damage and prevent the release of radioactive material from the reactor pool.

The protection and monitoring functions of the SRPS have been designed to ensure safe operation of the reactor. This is achieved by specifying Limiting Safety System Settings (LSSS) in terms of parameters directly monitored by the SRPS, as well as Limiting Conditions (LC) on other reactor system parameters and equipment performance. The LSSS are defined in this OLC as the Allowable Values, which, in conjunction with the Limiting Conditions, establish the threshold for protective system action to prevent exceeding acceptable limits including Safety Limits during Design Basis Accidents or transients.

The Safety Parameters for SRPS Trip 2 instrumentation are:

1. Power Range Log Neutron Flux
2. Power Range Log Neutron Flux Rate
3. Power Range Linear Neutron Flux
4. Core Outlet Temperature
5. Core Pressure Difference
6. Reactor Pool Water Level
7. Reflector Tank Temperature
8. Seismic Level
9. Manual Trip 2
10. FSS Failure
11. Control Rod Position
12. RPS Selected Configuration POWER
13. RPS Selected Configuration PHYSICS TEST

Each SRPS Safety Parameter is configured with three independent, redundant measurement CHANNELS. Each of these CHANNELS is composed of a sensor, transmitter, Conditioning Unit, current-to-voltage converter module, Trip Comparator Module (TCM) and Voting and Protective Logic (VPL) module. The field signals are processed by the Conditioning Units. The outputs of the Conditioning Units are sent to the TCM to determine if a CHANNEL is outside any associated Safety Function Safety

Setting. The outputs of the TCM are sent to the VPL of all three Trains and hence the VPL of each Train performs the same two-out-of-three (2oo3) voting for each Safety Function. If a Train determines that two or more CHANNELS have surpassed their Safety Setting for the same Safety Function, a Protection Action signal is output from that Train. This signal is then combined with the other Trains in a final stage of two-out-of-three voting logic in the Second Final Actuation Logic (SFAL) unit which produces initiation of the Second Shutdown System (SSS). The SRPS fails safe on loss of electrical power, however the SSS Trigger Valves are backed up by an Uninterruptible Power Supply (UPS) to prevent spurious activation.

LC 3.3.4 Post Accident Monitoring (PAM) Instrumentation

The purpose of this OLC is to ensure that there is sufficient information available on selected PAM parameters to monitor and assess reactor status and behavior during and following an accident by requiring the OPERABILITY of PAM Instrumentation.

PAM Instrumentation provides the necessary information needed for operators to monitor conditions after an accident. In addition, PAM provides information to indicate whether Safety Functions are being accomplished and is an important tool for implementing manual recovery actions. PAM Instrumentation comprises the electrical devices and circuitry involved in generating the PAM signals for display in the Main Control Room and the Emergency Control Centre.

The PAM instruments are considered Engineered Safety Features. Some primary sensors are shared with the First Reactor Protection System (FRPS) and some with the Second Reactor Protection System (SRPS). Isolated PAM signals are used by the Reactor Control and Monitoring System to display the PAM parameters.

LC 3.3.5 Loss of Power (LOP) Instrumentation

The purpose of this OLC is to ensure that the performance and availability of instrumentation used to sense loss of voltage or degraded voltage of the normal power supply and transfer Engineered Safety Features (ESF) loads to the diesel generator (DG) is consistent with the assumptions of Design Basis Accidents and transient analyses.

Successful operation of the required safety functions of some ESF equipment is dependent upon the availability of adequate power sources for energising the associated control components. The LOP instrumentation monitors the 415 V standby distribution switchboards of the Standby Power System (SPS). Power from the Normal Power System (NPS) is the preferred source of power for the 415 V standby distribution switchboards. If the LOP instrumentation determines that insufficient voltage is available, the associated switchboards are disconnected from the NPS and connected to the (DG) power sources of the SPS.

Each 415 V standby distribution switchboard has its own independent LOP instrumentation and associated transfer logic. The voltage for each standby distribution switchboard is monitored for Degraded Voltage.

LC 3.3.6 Neutron Flux Monitoring Instrumentation

This OLC specifies OPERABILITY requirements only for the monitoring and indication functions of the neutron flux monitoring instrumentation channels to ensure that neutron flux measurement information is available to the operator during CORE ALTERATIONS and movement of irradiation targets.

The neutron flux instrumentation (i.e, start-up neutron flux measurement channels for unirradiated or irradiated reactor cores or wide range logarithmic measurement channels for irradiated cores only) provides the operator with information relative to the neutron flux level at very low flux levels in the core. During refuelling, shutdown, and low power operations, the primary indication of neutron flux levels is provided by the neutron flux monitoring instrumentation. The neutron flux monitoring instrumentation provides monitoring of reactivity changes during fuel, safety rod, or irradiation rig or target movement and gives the control room operator early indication of unexpected subcritical multiplication that could be indicative of an approach to criticality.

The start-up neutron flux channels are used to monitor the core neutron flux from the earliest stages of reactor operation (neutron source level), up to 5 measurement decades. The start-up neutron flux measurement channels are part of the First Reactor Protection System (FRPS) and consist of three identical channels, one in each of the FRPS trains. Each channel consists of a detector, separated cable trays for each train, and electronic processing modules, all located in independent instrumentation rooms for each train. For a given detector position, measurements are performed from source level (3 to 100 cps), up to approximately 50,000 cps. Pulse fission chambers are used as detectors in the start-up channels. A pulse transformer, near to the fission chamber, converts the common state signal to a differential state signal to a clean detector signal, free of noise, even in case of long distances between detector and amplifier. The pulse transformer is connected to a start-up channel amplifier module located in the electronics cabinet. The electronics cabinet contains electronic modules that perform the interface between the detector and the analogue or digital instrumentation system. The start-up channel amplifier module generates an output signal (pulse train) that is applied to a pulse integrator. This is a pulse-rate to voltage converter, which produces an analogue signal proportional to the average count rate expressed in counts per seconds. The output of the pulse integrator is fed to the linear output and to the input of a logarithmic amplifier. The output from the logarithmic amplifier is fed to the logarithmic output and to the input of a differentiating amplifier, which provides a flux rate output. Each channel provides indication, alarm and FRPS trip signals.

The wide range logarithmic neutron flux channels are used to monitor the reactor core neutron flux over more than 10 decades combining the outputs of two separate pulse and campbell processing modules that are connected to a wide range fission chamber. The wide range logarithmic neutron flux channels measure the reactor's neutron flux from 10 decades below full power to 125% of the full power level and is fed into the FRPS. There are three wide range logarithmic neutron flux channels, one in each of the FRPS trains. The channels consist of independent detectors, separated cable trays for each train, and electronic processing modules all located in independent instrumentation rooms for each train. A fission chamber suitable for operation in both pulse and campbell states is used in the wide range logarithmic neutron flux channels. The detector signal is sent to the preamplifier module. The amplified detector signal is sent to the pulse and campbell processor modules located in the electronics cabinet. The campbell processor module provides a logarithmic campbell flux signal with variable gain and provides a flux rate output signal. The pulse processor module provides input attenuation and pulse-shaping amplification to bring the pulses from the preamplifier to an amplitude suitable to trigger a fixed level discriminator, provides a standard pulse of defined amplitude and width for all input pulses exceeding the discriminator level, and converts pulse-rate to a direct current signal. These modules generate three types of output signals, which are sent to the FRPS trip units. In addition, the outputs of two channels, electrically isolated, are sent to the Post Accident Monitoring System. The wide range logarithmic neutron flux channels have a logarithmic and logarithmic rate (inverse of period) output over their entire operating range. Each channel provides indication, alarm, and FRPS trip signals.

3.4 REACTOR POOL COOLING SYSTEMS

LC 3.4.1 Reactor Pool Water Level

This LC specifies the water level in the Reactor Pool that is sufficient to ensure that there is adequate cooling of the core, spent fuel stored in the Reactor Pool, and irradiated targets.

The Reactor Pool is the ultimate heat sink for both core cooling and for rigs cooling, by Natural Circulation. This is ensured by the opening of Flap Valves of the Primary Cooling System (PCS) and Reactor and Service Pool Cooling System (RSPCS), and by the availability of a sufficient water inventory. The Reactor Pool water inventory provides for cooling of the reactor core, spent fuel stored in the Reactor Pool, and irradiated targets. The Reactor Pool water inventory also provides radiation shielding. The Reactor Pool is an open cylindrical pool embedded in the concrete of the reactor block that extends from Level 0.00 up to Level +14.10. It normally contains approximately 186 m^3 of water, maintained during normal operation at Level +12.60 by the level control of the Hot Water Layer System.

LC 3.4.2 Primary Cooling System (PCS)

The purpose of this OLC is to ensure adequate core cooling flow for heat removal to prevent fuel damage. The LC requires reactor core heat removal by forced circulation during the POWER STATE and reactor core heat removal by natural circulation in all other STATES.

The function of the Primary Cooling System (PCS) is to remove heat from the reactor core in all operational and accidents situations to maintain the reactor core in a safe condition. Heat is extracted by the flow of cooling water through the reactor core either by forced or natural circulation. During forced circulation, the heat extracted from the reactor core is transferred to the Secondary Cooling System through the heat exchangers of the PCS, or indirectly through the Reactor and Service Pools Cooling System (RSPCS). Under abnormal conditions, or if the RSPCS is not available, the heat is transferred to the water of the Reactor and Service Pools.

The PCS has a circuit open to the atmosphere above the Reactor Pool. The circuit is completely within the containment. After leaving the reactor core in its upward flow, PCS water is collected in the upper chimney above the reactor core and is drawn through a lateral pipe towards the Reactor Pool boundary. This pipe is provided with a siphon breaker at its horizontal stretch before it leaves the Reactor Pool and goes down through the concrete of the reactor block to the decay tank. The main coolant line leaves the decay tank and the concrete shielding, and enters the PCS Pump room area, where it splits into three branches with a centrifugal pump and one plate-type heat exchanger in each branch. Two branches in operation will provide adequate cooling during the POWER STATE of operation. The three branches of the primary side of the main heat exchangers merge into the main pump discharge line. This main line traverses the reactor concrete block and then splits into two branches that enter the Reactor Pool and discharge the cooling water in the reactor core lower plenum. Water diffuses in the plenum before it flows into the reactor fuel element inlet nozzles and enters the reactor core cooling channels.

In order to allow the establishment of natural circulation when the PCS Pumps are not in operation, two Flap Valves are located in each of the two lines returning to the pool. One set of valves is located at Level +7.00 and the other set at Level +5.80. In PCS forced circulation mode, the four PCS Flap Valves are maintained in the closed position by the pressure of the PCS Pumps. They open by gravity when the pumps stop due to failure or are shutdown. The open position of these valves establishes a path within the Reactor Pool, to remove the decay heat by natural circulation using the pool water after reactor shutdown. The flywheels on the PCS Pumps are designed to provide a gradual coasting down of the pumps to continue the decay heat removal in the short–term, before the onset of natural circulation.

In the event of a loss of primary coolant causing a drop of Reactor Pool water level, the upper Flap Valves will act as siphon breakers and will prevent the water level from dropping below Level +7.00. Upon opening of the Flap Valves natural circulation of Reactor Pool water is established to extract reactor core decay heat as follows: hot water from the reactor core flows upward through the reactor core into the chimney and cold water enters the two PCS Pump discharge lines through the open Flap Valves and flows downward towards the reactor core inlet plenum. The reactor core heat is transferred to the Reactor Pool. The large amount of water in the pool acts as a heat sink with heat being lost to the atmosphere by evaporation. There is sufficient water in the pools to remove core decay heat for at least 10 days in the absence of any other means of heat removal before the pool water reaches the actuation level for the Emergency Make-up Water System (EMWS).

LC 3.4.3 Reactor and Service Pools Cooling System (RSPCS)

The purpose of this OLC is to ensure the availability of the Reactor and Service Pool Cooling System (RSPCS) to provide irradiation rigs cooling by forced downward circulation of Reactor Pool water during the POWER STATE and to provide irradiation rigs cooling by natural circulation of Reactor Pool water under abnormal conditions, the PHYSICS TEST, SHUTDOWN and the REFUELLING STATES. The RSPCS also provides long term pool cooling and Service Pool cooling.

To achieve these functions, the flow through the irradiation rig channels is downward during forced circulation and upward during natural circulation operation and there is a flow reversal during the transition between the two. The heat removed by the RSPCS is transferred to the Secondary Cooling System through the RSPCS heat exchanger.

In the irradiation Rigs Cooling Mode by forced circulation, the flow of water from the RSPCS Main Pump and heat exchanger (through the irradiation rigs cooling branch) is downward through the irradiation facilities. In this cooling arrangement, which includes cooling of the Service Pool, one of the RSPCS Main Pumps is in operation with flow through the RSPCS heat exchanger and the other pump is on standby. One RSPCS Main Pump in operation provides adequate irradiation rigs cooling in the POWER STATE. The irradiation rigs cooling branch has two Flap Valves and a siphon breaker in the horizontal stretch of the pipe. In the event of a loss of coolant accident of the RSPCS, the Reactor Pool is protected from draining completely by siphon breakers in the suction and discharge lines of the RSPCS. The RSPCS Flap Valves are maintained in their closed position by the suction action of the pump causing the pressure differential between the circuit and the Reactor Pool. The rigs cooling branch delivers water to the RSPCS decay tank. The RSPCS line then leaves the decay tank and is connected to the suction lines of the RSPCS Main Pumps and Long Term Pool Cooling pumps (only the RSPCS Main Pumps are used in the Rigs Cooling Mode).

Under abnormal conditions where the RSPCS Main Pump in service fails, a low flow signal from the RSPCS triggers the FSS and the reactor is shutdown. Once the pump is stopped, the differential pressure between the RSPCS and the Reactor Pool decreases gradually according to the dynamics of the inertia flywheel and the Flap Valves of the RSPCS open; establishing natural circulation through the rigs cooling branch ensuring cooling of the rigs. Opening of only one Flap Valve is enough to guarantee enough water flow for adequate cooling of the rigs.

LC 3.4.4 Emergency Makeup Water System (EMWS)

The EMWS is provided to ensure that the reactor core is covered with water in the event of a beyond design basis Loss of Coolant Accident (LOCA), which involves a drop in the water level of the Reactor Pool to below the edge of the upper chimney. This OLC ensures that the performance of the EMWS, in the unlikely event of a beyond design basis LOCA, meets the safety design basis provided in Chapter 16 of the SAR.

The EMWS consists of two storage tanks, two manual isolation ball valves, piping, an orifice plate and two float valves. Total tank volume is approximately 14 m^3. The EMWS injects water under gravity into the two legs of the PCS pool inlet pipelines, below the PCS flap valves. The water is maintained in the reactor chimney; hence the reactor core is kept covered.

EMWS is a passive system. Water injection is initiated by the automatic opening of at least one out of two float valves when Reactor Pool water level drops to the upper edge of the reactor chimney. If one of the valves fails to open, the system can still fulfil its function via the remaining valve. The EMWS is designed to keep the reactor chimney full of water for 24 hours, compensating for primary coolant evaporation losses caused by the reactor core decay heat. This is achieved with a water inventory of 13.2 m^3 in the water storage tanks and an average injection flow rate of 0.55 m^3/h.

It is possible to maintain the system in operation for more than 24 hours by refilling the EMWS storage tanks with water from the Demineralised Water Supply System, Refilling Pool of the Radioactive Liquid Waste Management System or from another external water source such as a tanker. Refilling of water from the Demineralised Water Supply System is automatic by opening of a float valve in one of the EMWS storage tanks.

3.5 CONTAINMENT SYSTEMS

LC 3.5.1 Containment

The purpose of this OLC is to ensure that the performance of the containment, in the event of a beyond design basis accident, meets the objectives of the analyses as contained in the SAR.

The containment represents a third barrier to prevent the release of significant amounts of fission products to the environment. The two other barriers are the fuel matrix and cladding and the pool coolant boundary. The physical barrier of the containment consists of the walls, floors and ceilings that form the external boundary of the containment, the access ways through that boundary, and the service penetrations of that boundary including hatches. The containment surrounds the main part of the reactor and includes a number of interconnected rooms listed in Chapter 7 of the SAR. The containment provides a barrier against an uncontrolled release of radioactive material to the environment. Containment integrity is provided through the following components:

a. walls, floors and ceilings that form the external boundary of the containment;
b. windows;
c. access ways through that boundary including doors and floor hatches; and
d. penetrations for electric power cables, instrumentation and control cables, communication cables, piping (water, gases, waste), ventilation conduits, and radioisotope transport tubes and elevators.

The isolation devices for the penetrations in the containment boundary are a part of the containment barrier.

Containment OPERABILITY is maintained by limiting leakage from the containment to the environment. Detailed requirements and testing methodology account for the unique design of the containment and are provided in the Containment Performance Testing Program. Compliance with this LC will ensure a containment configuration that is structurally sound and that will limit leakage to that assumed in the analyses in the SAR.

LC 3.5.2 Containment Isolation Provisions

This LC ensures that each containment isolation valve and interlocked door and valve is OPERABLE.

The function of the Containment Isolation Provisions, in combination with other mitigation systems, is to limit fission product release during and following postulated beyond design basis events. Isolation of penetration flow paths ensures that the release of radioactive material to the environment will be consistent with the analysis for postulated beyond design basis events.

Containment Isolation provisions for penetration flow paths consist of various means of facilitating isolation. These include Containment Isolation Valves (CIV) and interlocked double doors or valves. Although removable access ways through the containment boundary represent a penetration flow path, the removal of such an access way does not provide any redundant means of containment isolation within the flow path and represents inoperability of the containment boundary. Therefore the removal of a floor hatch is covered by LC3.5.1 Containment.

The OPERABILITY requirements for CIVs help ensure that an adequate containment boundary is maintained during and after a postulated beyond design basis event by minimising potential paths to the environment. Therefore, the OPERABILITY requirements provide assurance that containment function specified in the analyses will be maintained. These isolation devices are either passive or active (automatic). Manual valves, de-activated automatic valves secured in their closed position (including check valves with flow through the valve secured), blind flanges, and closed systems are considered passive devices. Check valves, or other automatic valves designed to close without operator action following a postulated beyond design basis event, are considered active devices. Two barriers in series are provided for each penetration, except for penetrations associated with systems with re-entrant lines not directly connected to the containment environment, so that no single credible failure or malfunction of an active component can result in a loss of isolation or leakage that exceeds limits assumed in the analyses. One of these barriers may be a closed system. Systems with re-entrant lines not directly connected with the containment environment are provided with one CIV at the intake point and one CIV at the exit point.

Power operated, automatic isolation valves are not required to have isolation times. This is due to the overly conservative estimate for containment isolation initiation of 2 minutes after the detection of activity in the stack.

Double-door air locks are built into the containment building to provide personnel access to and from the reactor building. During normal operations, at least one door in each airlock remains closed in order for the normal building ventilation to perform as designed. The air locks provide isolation during the process of personnel entering and exiting the building. As part of the containment, the containment air locks help to limit the release of radioactive material in the event of beyond design basis accidents. As part of the containment boundary, the safety function of the containment isolation valves and interlocked doors and valves is related to control of containment leakage following a beyond design basis event. Thus, the structural integrity and tightness of each containment isolation valve and interlocked door and valve are essential to the successful mitigation of such an event.

LC 3.5.3 Containment Energy Removal System (CERS)

The purpose of this OLC is to ensure that each CERS train is capable of removing the required heat load from the containment during and following a beyond design basis event.

The safety function of CERS is to provide an assured heat sink for all thermal loads inside the containment whenever the containment is isolated, thus eliminating the possibility of any significant increase in pressure. By controlling the pressure, the CERS minimises the amount of the containment atmosphere released to the environment.

During normal operations, the reactor building ventilation system is in service. The reactor building encloses the reactor and those sources that may release radioactivity. The normal ventilation maintains building pressure slightly negative with respect to outside conditions. The normal reactor building ventilation serves to contain, dilute, filter, and monitor radioactivity that may result from facility operations and Design Basis Accidents.

The CERS consists of two independent 100% capacity trains, each composed of a chiller unit, a chilled-water circuit with a centrifugal pump, a cooling-coil, and a fan that circulates the air within the containment. The CERS is supplied from the Standby Power System. As such, the CERS is capable of fulfilling its function even with the failure of one train and a loss of normal power.

Each CERS chiller unit has a freon circuit with a compressor, a condenser (air cooled), an expansion valve and an evaporator and provides approximately 400 kW of cooling power. A CERS chilled-water circuit is a closed circuit with its corresponding expansion tank, buffer tank, 3-way valve and centrifugal pumps and provides enough water flow to remove the required amount of heat from the cooling-coils. The fans of the CERS establish the necessary air flow inside the containment to ensure appropriate heat removal from the containment air. Treated air is distributed and returned by standard galvanised metal sheet ducts and supply grilles.

When the containment is not isolated, the CERS supplements the cooling capacity of the non-Engineered Safety Feature Containment HVAC System; and when the containment is isolated, the CERS provides an adequate heat sink to manage all thermal loads in the containment. If the normal power supply is not available, the CERS is supplied from the Standby Power System.

LC 3.5.4 Hot Cells Exhaust System

The purpose of this OLC is to ensure the performance of the Hot Cells Exhaust System is maintained consistent with the beyond design basis event analysis.

The function of the Hot Cells Exhaust System is to control and treat airborne releases from the hot cells to the environment in the event of a beyond design basis event.

The Hot Cells Exhaust System consists of two trains. Each train consists of a filter bank (with two absolute filters and one charcoal filter), a centrifugal fan, ductwork and dampers. Ductwork from the hot cells to the filter banks, from the filter banks to the centrifugal fans, and from the centrifugal fans back to the hot cells is common to both trains. This is acceptable because ductwork is a passive component and not assumed to fail.

The Hot Cells Exhaust System provides the above pool hot cell complex (pneumatic hot cells A and B, and the transfer hot cell) with a dedicated ventilation system. Supply air to the hot cells is filtered through charcoal and absolute filters on the common ingress line. The exhausts from each of the hot cells are collected in a common duct and pass through a one of two banks of absolute and charcoal filters (one filter bank in stand-by). The filtered air is handled by two 100% capacity centrifugal fans (one fan in stand-by) and recirculated back to the hot cells. The hot cells are kept at a slight negative pressure relative to the reactor hall and the above pool hot cell complex area. The pressure is controlled by drawing out a small flow from the air being recirculated and diverting it into the discharge duct of the Reactor Air Exhaust System. In the containment isolation mode, an isolation valve closes the connection to the exhaust discharge duct and the injection valve opens fully so that all the filtered air is recirculated through the hot cells.

3.6 PLANT SYSTEMS

LC 3.6.1 Service Pool Water Level

This LC specifies the minimum service pool water level so as to ensure that there is adequate cooling of the stored spent fuel and irradiated targets.

The Service Pool provides shielding, cooling water, and working areas connected with the Reactor Pool by means of the transfer canal. The Service Pool contains storage space for spent Fuel Assemblies. It is also an area for handling irradiation rigs and allows communication with the Transfer Hot Cell by means of the Service Pool Elevator. The Service Pool acts as a large heat sink and provides a medium for retention of radioactive material in the case of accidents involving breaches of fuel cladding. The Service Pool water inventory provides for cooling of stored spent fuel and irradiation targets by receiving the decay heat extracted by natural circulation of pool water. The Service Pool water inventory also provides radiation shielding. The Service Pool normally contains approximately 182 m^3 of water, maintained during normal operations at Level +12.60 m by the level control of the Hot Water Layer System.

LC 3.6.2 Reflector Explosive Gas Concentration

The purpose of this OLC is to ensure the concentration of deuterium and oxygen in the reflector is maintained below explosive levels.

During reactor operation, a decomposition reaction (radiolysis) takes place in the Reflector Vessel due to the irradiation of heavy water. As a consequence, deuterium and oxygen gases are produced and accumulated inside the heavy water expansion tank, mixing with the helium cover gas. With the reflector explosive gas concentration $\leq$ 4 volume percent deuterium (% D2) and $\leq$ 5 volume percent oxygen (% O2), an explosive mixture cannot be present in the reflector or heavy water expansion tank. The Deuterium Recombination System keeps the concentration of these gases below explosive levels by recombining deuterium and oxygen in a catalyst bed, producing heavy water.

LC 3.6.3 Irradiation Rig and Target Requirements

The purpose of this OLC is to ensure that restrictions on the use of irradiation facilities are implemented in order that irradiation target reactivity is consistent with the safety analyses and that adequate irradiation target heat removal is maintained.

One of the main purposes of the reactor facility is the production of radioisotopes, the neutron transmutation doping of Silicon, and the irradiation of samples for elemental and isotopic analysis. For this purpose, irradiation facilities of different design are provided in the reflector vessel. The Irradiation facilities can be grouped as:

a. Bulk Production Irradiation Facilities (BIF)

b. Long Residence Time Irradiation Facilities (LRT)

c. Short Residence Time Irradiation Facilities (SRT)

d. Large Volume Irradiation Facilities (LVF)

The Bulk Production Irradiation Facilities (BIF) enable the irradiation of targets contained in removable rigs, located inside irradiation tubes provided within the Reflector Vessel. The LRT and SRT are pneumatic facilities capable of irradiating targets in sealed cans, provided the reactivity worth does not exceed the limit specified. The Large Volume Facilities (LVF) are utilised for the irradiation of NTD Silicon in unsealed cans, and are capable of being moved during reactor full power operation.

The reactivity worth of rigs and targets are controlled such that subcriticality is maintained under cold conditions and acceptable fuel design limits are not exceeded during normal operation and transients. Restrictions on the reactivity worth of “fixed” BIF irradiation rigs are specified. This limit is provided to support the SDM requirements of OLCs 3.1.1, “Shutdown Margin (SDM)-1,” in assuring the reactor can be brought safely to, and maintained in, cold, subcritical conditions under all conditions. Restrictions on the movement of irradiation targets are provided to ensure that acceptable fuel design limits are not exceeded due to reactivity perturbations.

Materials in the irradiation facilities produce heat during irradiation. The thermal-hydraulic design basis of the irradiation facilities is that, during the irradiation process, the heat generated in the material must be removed to avoid any thermally induced damage. Therefore, the heat produced by irradiation targets must be restricted to within the heat removal capabilities of the Reactor and Service Pool Cooling System (RSPCS) and Pneumatic Target Cooling System.

LC 3.6.4 Bulk Irradiation Rig Loading

The purpose of this OLC is to ensure that restrictions on the use of irradiation facilities are implemented in order that irradiation target reactivity changes due to irradiation target movement are maintained consistent with the safety analyses.

The Bulk Production Irradiation Facilities (BIF) enable the irradiation of targets contained in removable rigs, located inside irradiation tubes provided within the Reflector Vessel. Depending on their reactivity worth, rigs are designated as “fixed” or “non-fixed” (“movable”) from their irradiation position. All BIF rigs with reactivity worth exceeding the limit specified in 3.6.3.3 (2) are designated as “fixed” and shall only be moved with the reactor in shutdown mode. A “non-fixed” irradiation rig may be moved while the reactor is at power. These movements introduce changes in the reactivity of the reactor core. A limit on the reactivity worth is imposed to ensure that the RCMS could cope with the rate of change of reactivity when rigs are moved from/to the reflector vessel. The LRT and SRT are pneumatic facilities capable of irradiating targets in sealed cans, provided the reactivity worth does not exceed the limit specified in 3.6.3.3 (3). The Large Volume Facilities (LVF) are utilised for the irradiation of NTD Silicon in unsealed cans, and are capable of being moved during reactor full power operation.

Irradiation rigs installed in the Bulk Production Irradiation Facilities and the Large Volume Irradiation Facilities are cooled by the Reactor and Service Pool Cooling System (RSPCS). A thermal-hydraulic analysis of irradiation rig cooling for the POWER STATE was performed to ensure adequate cooling is provided to each of the irradiation rigs installed in the Bulk Production Irradiation Facilities and the Large Volume Irradiation Facilities. The analysis considered that operational removal of irradiation rigs may result in inadequate cooling to remaining rigs installed in the reflector vessel. Therefore, hydraulic restrictions have been designed for the Bulk Production Irradiation Facilities and the Large Volume Irradiation Facilities. The design allows for a limited number of empty irradiation facilities positions without impacting the ability to provide adequate cooling to the remaining irradiation rigs installed in the Bulk Production Irradiation Facilities and the Large Volume Irradiation Facilities of the reflector vessel. The LC 3.6.4(b) is conservative in view of results of the analysis which demonstrates that up to two "Low Flux" BIF position and one "Medium Flux" BIF position can be empty during POWER STATE (Reference 1).

LC 3.6.5 Emergency Control Centre (ECC) Ventilation and Pressurisation System

The purpose of this OLC is to ensure that each ECC Ventilation and Pressurisation System train is capable of ensuring the habitability of the ECC during and following a design basis event when the Main Control Room is uninhabitable or unavailable. It provides fresh, filtered air to the ECC and a protected environment where the occupants are safeguarded against hazards that could jeopardise necessary operator action.

The ECC Ventilation and Pressurisation System is an Engineered Safety Feature (ESF) of the Reactor Facility and consists of two independent trains each capable of independently fulfilling the full safety function of the system.

Each train consists of a pre-filter to remove large particulate mater, an activated charcoal adsorber to provide a hold-up period for gaseous iodine to decay and a high efficiency particulate air (HEPA) filter to remove carbon fines and any other fine particulate matter that may be radioactive. Additionally each train has a centrifugal fan, ductwork and grilles. The fan of each train draws air from a different location providing alternate supplies of filtered air from both inside and outside the building. Fan A draws air from outside through the roof of the building while Fan B draws air from within the Neutron Guide Hall.

Each train of the ECC Ventilation and Pressurisation System is also provided with an independent air conditioning unit powered from the Normal Power System for optimum environmental control. These units are not required for the system to fulfil its designed safety function and thus do not form part of system OPERABILITY.

3.7 ELECTRICAL POWER SYSTEMS

LC 3.7.1 AC Sources

The purpose of this OLC is to ensure the availability of adequate AC power to required ESF loads is maintained under all design basis conditions.

The Electrical Power Supply System is made up of the Normal Power System (NPS) and the Standby Power System (SPS). The NPS includes the high voltage distribution switchboards in the Reactor Facility Substation Auxiliary Building, and downstream equipment (with the exception of the SPS), finishing at motor control centres and distribution switchboards for lighting, general purpose outlets and supplies to packaged plant such as the Heating Ventilation and Air Conditioning Systems. The SPS is an Engineered Safety Feature (ESF). The function of the SPS is to provide electric power to essential loads when the NPS is not available. The SPS contains standby diesel generators (DGs) to provide power during a loss of normal power. It also includes Uninterruptible Power Supply (UPS) units to ensure continuity of power to selected loads, during the short delay between loss of normal power and availability of power from the DGs. Normally, all interruptible electric loads are fed by NPS, including the input of the Uninterruptible Power Supply (UPS) units. ESF loads are supplied via switchboards of the SPS.

When normal power supply is not available, ESF loads receive power from the standby DGs. Supply to loads is maintained by UPS units without interruption. Other loads are supplied by the DGs following DG start (typically less than a minute). The DGs are able to supply the total maximum demand of the ESF loads, including in rush currents of UPS units and motors. The arrangement of the SPS, comprising two independent DGs, switchboards and UPS units, ensures that a single failure cannot prevent supply of power to ESF loads in the event of a loss of normal power. The two independent trains of the SPS cannot be connected. One low voltage standby distribution switchboard, SD-C, can be fed from either train using a manual changeover switching arrangement, but not to both trains simultaneously. When normal power is available, each train is fed from the associated normal supply train. When standby power (i.e., DG power) is required, the associated standby distribution (SD) switchboard is automatically disconnected from the normal power supply and connected to the DG by means of an automatic transfer switch. Returning to the supply to normal power is a manual operation.

The standby power supply for the SPS consists of two DGs. Either one of these two DGs can supply the necessary ESF loads in the event of a loss of normal power. DG-A and DG-B are dedicated to low voltage standby distribution switchboards SD-A and SD-B, respectively. In addition to DG-A and DG-B, a spare DG that can be used to replace either DG-A or DG-B during an extended maintenance outage is provided. The spare DG will be manually connected after the out-of-service DG is disconnected, ensuring that separation of the two standby power trains is maintained. A DG starts automatically on a switchboard degraded voltage signal (refer to OLC 3.3.6, “Loss of Power (LOP) Instrumentation”). After the DG has started, it automatically ties to its respective low voltage standby distribution switchboard after offsite power is tripped as a consequence of [switchboard undervoltage or degraded voltage. When the DG is tied to its associated low voltage standby distribution switchboard, ESF loads are connected to their respective supplies.

LC 3.7.2 Reactor Protection System (RPS) Uninterruptible Power Supplies (UPS)

The purpose of this OLC is to ensure the availability of adequate power to instrumentation and control systems that are required to operate and maintain essential reactor facility equipment in the event of a loss of normal power, following a design basis event.

The Electrical Power Supply System is made up of the Normal Power System (NPS) and the Standby Power System (SPS). The SPS is an Engineered Safety Feature (ESF). The function of the SPS is to provide electric power to ESF loads when the NPS (i.e., normal power) is not available. The SPS contains standby diesel generators (DGs) to provide power during a loss of normal power. It also includes Uninterruptible Power Supply (UPS) units to ensure continuity of power to selected loads, during the short delay between loss of normal power and availability of power from the DGs. The required UPS units supported by the SPS are as follows.

1. UPS-RPS1
2. UPS-RPS2
3. UPS-RPS3

The function of these UPS units is to provide reliable 240 VAC uninterruptible power to loads required for continuity of equipment operation in the event of loss of normal power. Each UPS unit consists of a battery-charger rectifier, batteries, DC/AC converter, static by-pass switch, maintenance by-pass switch and isolation transformers. On a loss of normal power, and until power from the associated DGs becomes available, long-life batteries provide power to the associated UPS. Each battery is separately housed in a ventilated room to avoid the concentration of explosive fumes, and is separate from its charger. All the batteries are sized so that the designed loads will not exceed warranted capacity at the end-off-installed-life with 100% design demand, in accordance with IEEE 485 (Ref. 1). The batteries have sufficient stored energy to operate connected loads continuously or intermittently, as required, for at least 30 minutes without recharging. When power is restored to the standby distribution switchboards, the battery charger of the UPS units will recharge the batteries while simultaneously supplying the inverter. The capacity of each UPS unit is based on the largest combined demand of the various continuous loads, plus the largest combination of non-continuous loads that are likely to be connected to the power supply simultaneously.

The safety system and logic control for the First Reactor Protection System (FRPS), the Second Reactor Protection System (SRPS), and the Post Accident Monitoring (PAM) System derive their power from three independent UPS units (UPS-RPS1, UPS-RPS2, and UPS-RPS3) fed from different trains, the third system being fed from low voltage standby distribution switchboard SD-C. This arrangement provides redundant, reliable power of acceptable quality to support the safety logic and control functions during normal, abnormal and post-accident conditions. The availability of these three UPS units associated with the RPS is addressed by this OLC.

LC 3.7.3 Standby Power Supply Distribution System

The purpose of this OLC is to ensure the availability of adequate power to ESF systems that are required to operate and maintain essential reactor facility equipment in the event of a Design Basis Accident or transient.

The Electrical Power Supply System is made up of the Normal Power System (NPS) and the Standby Power System (SPS). The SPS is an Engineered Safety Feature (ESF). The function of the SPS is to provide electric power to essential loads when the NPS is not available. The Standby Power Supply Distribution System portion of the SPS is divided into redundant and independent AC electrical power distribution trains (Trains A and B). The two independent trains of the Standby Power Supply Distribution System cannot be connected. One low voltage standby distribution switchboard, SD-C, can be fed from either train using a manual changeover switching arrangement, but not from both trains simultaneously.

When normal power is available, each standby power supply distribution train is fed from the associated normal supply train. When standby power (i.e., DG power) is required, the associated low voltage standby distribution switchboard is automatically disconnected from the normal power supply and connected to the associated DG by means of an automatic transfer switch.

The SPS supplies Engineered Safety Feature loads such as PAM instrumentation, ECC ventilation, RPS UPS batteries and some containment systems such as the CERS.

3.8 CORE ALTERATIONS

LC 3.8.1 Control Rod Position During Refuelling

The purpose of this OLC is to ensure all control rods are inserted to prevent the reactor from inadvertently achieving criticality during refuelling operations.

Control rods provide the capability to maintain the reactor subcritical under all conditions. During the REFUELLING STATE, the reactor is maintained subcritical at all times with the five control rods in the fully inserted position, as described in the SAR.

LC 3.8.2 Control Rod Maintenance

The purpose of this OLC is to ensure the plant is in the appropriate state with the reflector vessel partially drained during control rods maintenance so as to prevent the reactor from inadvertently achieving criticality during such maintenance operations.

Control rods provide the capability to maintain the reactor subcritical under all conditions. The reactor is maintained subcritical at all times with the five control rods in the fully inserted position, as described in the SAR. However, during maintenance activities where the negative reactivity worth of the control rods is altered (including control rod replacement activities), insertion of control rods may no longer be sufficient to prevent inadvertent criticality. For these particular maintenance activities, the reflector water level shall be drained to compensate for the potential loss of negative reactivity associated with the control rod maintenance. The absence of heavy water in the upper portion of the reflector causes a large decrease in neutron reflection to the reactor core, which results in a subcritical condition. This ensures a sufficiently large shutdown margin is maintained to prevent inadvertent criticality during the control rod maintenance.

4.0 DESIGN FEATURES

4.1 Reactor Core

4.1.1 Core Power and Flow

The analyses presented in the SAR demonstrate that the reactor is capable of operating at a total steady-state power of 21.2 MW, which corresponds to a heat transfer rate to the PCS of 20.0 MW, with a steady-state flow of not less than 1900 m^3/h.

4.1.2 Fuel Assemblies

The reactor shall contain 16 Fuel Assemblies during operation. Each assembly shall consist of 19 inner fuel plates and 2 outer fuel plates. The fuel plates contain a fuel meat sealed by aluminium cladding.

Fuel Assemblies shall be limited to those fuel designs that have been analysed with appropriate codes and methods and have been shown by tests or analyses to comply with the safety design basis. The currently approved fuel, as described in the SAR, is described in the following table (nominal values shown).

Fuel Type	Dispersed silicide; U_3Si_2
U Density	4.8 g.cm^{-3}
Enrichment	≤ 20 weight percent
Maximum U-235 mass (per FA)	489.3g

4.1.3 Control Rods

The reactor core shall contain 5 control rods. The control rod absorber material shall be a hafnium alloy. The design and configuration of the Control Rods shall be as described in the SAR.

4.2 Fuel Storage

4.2.1 Criticality

4.2.1.1 The Service Pool and Reactor Pool spent fuel storage racks are designed and maintained for:

a. Fuel Assemblies as described in, 4.1.2 above,

b. k_{eff} ≤ 0.9 under normal and accident conditions as described in the SAR, and

c. A nominal 92 mm centre-to-centre distance between fuel assemblies placed in storage racks with neutron absorber as described in the SAR.

4.0 DESIGN FEATURES (continued)

4.2 Fuel Storage (continued)

4.2.1.2 The fresh fuel storage racks are designed and maintained for:

a. Fuel Assemblies as described in, 4.1.2 above,

b. $k_{eff} \leq 0.9$ under normal and accident conditions as described in the SAR, and

c. A nominal 170 mm centre-to-centre distance between Fuel Assemblies placed in storage racks as described in the SAR.

4.2.2 Drainage

The Reactor Pool spent fuel storage area and Service Pool spent fuel storage area are designed and shall be maintained to prevent inadvertent draining of the pools below elevation +7.60 m.

4.2.3 Capacity

The spent fuel storage racks are designed and maintained with a storage capacity limited to:

a. 24 Fuel Assemblies in the Reactor Pool spent fuel storage racks and

b. 336 Fuel Assemblies in the Service Pool spent fuel storage racks.

5.0 ADMINISTRATIVE CONTROLS

5.1 Organisation

5.1.1 The organisation shall be established for safe operation and maintenance of the facility. It shall include the roles for activities affecting nuclear safety.

a. Lines of authority, responsibility and communication shall be defined and established. These relationships shall be documented for key roles.

b. The reactor manager shall be responsible for overall safe operations and shall have control over those activities necessary for safe operation and maintenance.

c. A specified ANSTO officer shall have responsibility for overall nuclear safety of the facility and shall take any measures needed to ensure acceptable performance of the staff in operating, maintaining, and providing technical support.

5.1.2 The facility staff organisation shall include an authorised Reactor Operator in the Main Control Room during all reactor STATES. In addition, a minimum of three personnel shall be within the facility during the POWER and PHYSICS TEST STATES and a minimum of two personnel during the SHUTDOWN and REFUELLING STATES.

Engineering expertise shall be available during POWER and PHYSICS TEST STATES. This expertise may be provided by one of the authorised Reactor Operators. In addition, expertise in radiation protection shall be available during all reactor STATES.

5.0 ADMINISTRATIVE CONTROLS

5.2 Programs

The following processes shall be established, implemented, and maintained.

5.2.1 POWER PEAKING FACTOR (PPF) Verification Program

A program to implement the verification of PPF shall be established. The program shall include the criteria and frequency to be used for verifying PPF is within the limits of OLC 3.2.3, “POWER PEAKING FACTOR (PPF).”

The verification is performed by calculations which are required to be performed prior to and during the first startup after completion of fuel movement or control rod replacement. The calculations shall be performed using appropriate computer codes and methods consistent with the code verification and validation process.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the PPF Verification Program surveillance frequency.

Changes to the PPF Verification Program shall be made under appropriate administrative controls and review. ANSTO may make changes to the PPF Verification Program without prior ARPANSA approval provided the changes do not alter this OLC and do not involve a significant implication for safety.

5.2.2 Containment Performance Testing Program

A program shall establish the performance testing of the containment.

a. The maximum allowable beyond design basis containment leakage rate, L_{BDB}, shall be 3 % of containment air volume per day.

b. The containment performance acceptance criterion is $\leq L_{BDB}$.

c. The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Containment Performance Testing Program.

Changes to the Containment Performance Testing Program shall be made under appropriate administrative controls and review. ANSTO may make changes to the Containment Performance Testing Program without prior ARPANSA approval provided the changes do not alter this OLC and do not involve a significant implication for safety.

5.2.3 Irradiation Requirements Program

A program under the Quality System to implement the requirements of LC 3.6.3 and 3.6.4 for irradiations undertaken in the Reflector Vessel facilities shall be established. The program shall include the following elements to ensure the safe conduct of irradiations:

- development of specifications for irradiation target materials and canning, including consideration of:
  - irradiation conditions
  - materials and limitation on material mass or activity as appropriate
  - means of encapsulation

  - o pressure, heat generation and temperature
  - o radiological aspects
- review and approval of specifications prior to irradiation
- irradiation rig design review and approval
- scheduling of irradiation target loading and unloading to ensure compliance with rig loading requirements
- monitoring of irradiation rig and target reactivity worth

Changes to the Irradiation Requirements Program shall be made under appropriate administrative controls and review. ANSTO may make changes to the Irradiation Requirements Program without prior ARPANSA approval provided the changes do not alter this OLC and do not involve a significant implication for safety.

5.2.4 Diesel Fuel Oil Testing Program

A diesel fuel oil testing program to implement required testing of both new fuel oil and stored fuel oil shall be established. The program shall include sampling and testing requirements, and acceptance criteria, in accordance with applicable standards. The purpose of the program is to establish the acceptability of new fuel oil for use prior to addition to storage tanks. In addition, the program provides for periodic testing of stored fuel oil.

The provisions of SR 3.0.2 and SR 3.0.3 are applicable to the Diesel Fuel Oil Testing Program test frequencies.

Changes to the Diesel Fuel Oil Testing Program shall be made under appropriate administrative controls and review. ANSTO may make changes to the Diesel Fuel Oil Testing Program without prior ARPANSA approval provided the changes do not alter this OLC and do not involve a significant implication for safety.

5.2.5 Operational Limits and Conditions (OLCs) Bases Control Program

This program provides a means for processing changes to the Bases of these OLCs.

a. Changes to the Bases of the OLCs shall be made under appropriate administrative controls and review.

b. ANSTO may make changes to Bases without prior ARPANSA approval provided the changes do not require either of the following:

   1. A change in the OLCs incorporated in the licence or
   2. A change to the SAR or Bases that involves a significant implication for safety.

c. The Bases shall be maintained consistent with the SAR.

d. Proposed changes that meet the criteria of OLC 5.2.5.b.1 or OLC 5.2.5.b.2 above shall be reviewed and approved by ARPANSA prior to implementation. Changes to the Bases implemented without prior approval shall be provided to ARPANSA.

5.2 Programs (continued)

Bases for the OLCs are not considered part of the OLCs. Therefore, the OLC change process does not apply to the Bases for the OLCs.

5.2.6 Safety Function Determination Program (SFDP)

This program ensures loss of safety function is detected and appropriate actions taken. The SFDP shall contain the following:

a. Provisions for cross train checks to ensure a loss of the capability to perform the safety function assumed in the accident analysis does not go undetected,

b. Provisions for ensuring the facility is maintained in a safe condition if a loss of function condition exists,

c. Other appropriate limitations and remedial or compensatory actions.

A loss of safety function exists when, assuming no concurrent single failure, no concurrent loss of offsite power, or no concurrent loss of onsite diesel generator(s), a safety function assumed in the accident analysis cannot be performed. Safety functions include the ability to shutdown the reactor and maintain it shutdown, remove adequate heat from the core, and prevent or mitigate the consequences of events that could result in radioactive releases. For the purpose of this program, a loss of safety function may exist when a support system is inoperable, and one of the following exists:

a. A required system redundant to the system(s) supported by the inoperable support system is also inoperable,

b. A required system redundant to the system(s) in turn supported by the inoperable supported system is also inoperable, or

c. A required system redundant to the support system(s) for the supported systems (a) and (b) above is also inoperable.

The SFDP identifies where a loss of safety function exists. If a loss of safety function is determined to exist by this program, the appropriate Conditions and Required Actions of the OLC in which the loss of safety function exists are required to be entered. When a loss of safety function is caused by the inoperability of a single OLC support system, the appropriate Conditions and Required Actions to enter are those of the support system.

Changes to the SFDP shall be made under appropriate administrative controls and review. ANSTO may make changes to the SFDP without prior ARPANSA approval provided the changes do not alter this OLC and do not involve a significant implication for safety.

5.0 ADMINISTRATIVE CONTROLS

5.3 Reporting Requirements

5.3.1 Special Reports

When a report is required by Condition B of OLC 3.3.4, “Post Accident Monitoring (PAM) Instrumentation” or Condition B of OLC 3.6.5 “ECC Ventilation and Pressurisation System”, a report shall be submitted to ARPANSA within 14 days. The report shall outline the cause of the inoperability, the pre-planned alternate method of fulfilling the safety function, and the plans and schedule for restoring the systems to an OPERABLE status.

5.3.2 CORE OPERATING LIMITS REPORT (COLR)

The COLR is the document that provides specific parameter limits for the operating program. These limits shall be determined for each operating program.

a. Core operating limits shall be established prior to startup for each operating program and documented in the COLR for the following.

b. The analytical methods used to determine the core operating limits shall be identified.

c. The core operating limits shall be determined such that all applicable limits (e.g., fuel thermal limits, core hydraulic limits, and accident analysis limits) of the safety analyses are met.

d. Changes to the COLR shall be made under appropriate administrative controls and review. ANSTO may make changes to the COLR without prior ARPANSA approval provided the changes do not alter this OLC and do not involve a significant implication for safety.

e. The COLR shall be provided upon issuance to ARPANSA.