Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
5.61	5.1 Summary Description	Low Power Mode: Reactor power limited to 400kW, and is cooled by Natural Circulation (NC), with the Primary Cooling System (PCS) pumps off.	The low power operation mode appears to be before power rise and with a fresh core load. The reactor trips associated with PCS flow would not be part of the FRPS. Has Ch.16 and the PSA given any consideration to accident sequences for this mode of operation?
			Response: In fact low power operation mode is not limited to operation with a fresh core. Relevant trips in low power mode include the high power and low reactor period trips. Accident sequences in low power mode are considered at Chapter 16, Sections 16.8.3.1, and 16.8.4.1. The probabilities of accident sequences in low power mode have not been evaluated in the PSA. Some discussion is included in the PSA Section 3.1.2.
5.62	5.2.4 Description of Reactor Structures	Unions and joints of the reactor structures allow easy assembly and disassembly.	The ease of assembly /disassembly may compromise the vibration or loose part defences of structures. What standards are in place for locking structures, and components within the pools.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
			Response: Easy assembly/disassembly does not mean poorly locked structures. It is related to appropriate component layout and simplicity of operations (e.g. minimise the number of components when there is difficult access to a system, simple handling tools, etc). Under no circumstances will the strength of the assembly and prevention of vibrations be compromised by these concepts. All nuts are selected according to the relevant Australian Standard, e.g. AS 1110x, AS 1112.x etc. For the case of fixation of structures inside the pools, removable components are bolted with operation nuts. An operation nut is a standard nut to which a special cap is attached to enable the nut to be locked or unlocked using a special tool from the poolside. Non-removable components are bolted using hexagonal nuts with washers. Such components will not be required to be unfastened during normal operation.
5.63	5.2.4.2 Core inlet Plenum	In order to avoid excessive vibrations all of the components located inside the plenum are embedded in cylindrical tubes to protect them from the PCS flow.	Will there be a flow visualisation model of the plenum, since it is too late to fix a vibration problem that emerges after manufacture and operation?
			Response: Individual components to be installed inside the inlet plenum will be tested using mock-up facilities. In addition, numerical models of the inlet plenum will be constructed in order to obtain estimates of the flow and to assess effects on the components inside the plenum (including flow-induced vibrations).

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
5.64	5.2.4.3 Core grid	The lower part is made from stainless steel and the upper part from aluminium. It is kept in position by four fastener shafts.	The core grid is an important component. What is the protection against fastener breakage and loosening. Also can there be galvanic action between the aluminium and stainless parts if directly in contact.
			Response: Core Grid fasteners are designed with an adequate safety margin and the operating conditions are very mild (low temperature and high quality water). The materials used for this component (Al 6061 and stainless steel) have very good corrosion resistance. The behaviour of these materials for the conditions to be present in the RRR is analysed in Section 5.9. Note that galvanic pair formation depends on the environment properties. In the case of the RRR the medium involved is demineralised water. Water quality will be strictly maintained in the prescribed range (conductivity < 1 µS/cm). In a medium like this, a protective oxide layer will develop at the interface making the material more passive as the oxide layer is more electropositive than the base alloy. Please also refer to: 1. Aluminium and aluminium alloys, ASM Specialty Handbook, 1993 2. Chawla and Gupta, Corrosion Control, ASM International, 1993. Also note that in the very unlikely event of failure of one out of the four core grid fasteners, the other three are sufficient to keep the grid safely fixed.
5.65	5.2.4.4 Control Rod Guide Box (CRGB)	The CRGB is a manufactured from zircaloy 4, and fixed to the core chimney by fasteners.	What locks the fasteners to prevent loosening or vibration under the fluid flow through the box.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
			Response: The CRGB lock will be bolted to the top of the upper chimney (riser). Note that the bolting is not influenced by the main PCS flow.
5.66	5.2.4.6 Reflector Vessel	The Reflector Vessel is manufactured from Zircaloy 4 and is generally a welded structure, but with some bolted flanges for the Cold and Hot Source penetrations. It serves many roles , including the reservoir for the heavy water dump used as the Second Shutdown System (SSS).	The Reflector Vessel is not considered a Safety Class 1 component, although it is Seismic Category 1 and QA class A. In view of its safety role and its important role in the integrity of the reactor, consideration should be give to it as a Safety Class 1 component, that is similar to the reactor tank.
			Response: The various safety functions of the Reflector Vessel are adequately covered by the definition of Safety Function O (Chapter 2, Table 2.5/1). The vessel itself does not perform a primary role in achieving safety. Shutdown of the reactor is performed through the SRPS and SSS should the FRPS or FSS fail to function as designed.
5.67	5.2.4.7 Core Chimney	The core chimney separates the light water coolant from the heavy water in the reflector vessel. The upper part is manufactured from stainless steel and the lower part from zircaloy.	This item is important to safety, yet is considered as Safety Class 2 and QA class B. The bases for this lower categorisation needs to be justified.
			Response: According to Table 2.5/2 in Chapter 2, the riser part of the chimney is Safety Category 1, Seismic Class 1, QA Level A. The chimney adjacent to the core is part of the reflector vessel. See also the response to Question 5.66.

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Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
5.68	5.2.4.9 Irradiation Tubes	There are irradiation tubes and neutron sources within the reflector vessel that are cooled by the reflector cooling system.	It is not clear how the reflector dump affects the heat removal capacity of the reflector vessel. There will be no flow, and thus the neutron source vessels and irradiation tubes will heat up. The thermal stresses in the components within the RV and in the RV itself need to be estimated.
			Response: Outcomes from the analysis are presented in Section 5.2.6.5. The maximum temperature is estimated to be approx 70°C and the induced stresses are acceptable.
5.69	5.2.4.10 Instrumentation support structures	Core instruments are fixed either to the Reflector Vessel or to Reactor Tank.	From Fig 5.2.8 it is clear that the nucleonics instrumentation cannot be shadowed by the irradiation rigs.
			Response: Comment noted.
5.70	5.2.4.11 Provisions for Access, Operation and Maintenance	The design of the core and associated structures provides for easy and fast access for maintenance operations.	The standards used to lock fasteners to prevent loosening or vibration are important. There could be a conflict between ease of maintenance and locking of fasteners. More information is needed.
			Response: Please see response to Question 5.62. As an example, the fixations of core-associated components such as Control Rod Drives, Fuel Clamps, and Core Grid are easily accessed in the CRD Room.
5.71	5.2.5.1 Manufacturing and Installation	Tests are identified for the inspection and testing of reactor structures, including hydraulic tests.	There is no mention of what is intended for weld inspections, radiograph or ultra-sonic for the reactor components. Are these specified in the appropriate codes and, what influence has the structure/component safety and quality class on such inspections.
			Response: Welds will be tested using Non Destructive Testing methods (e.g. die penetration, ultrasonic, radiograph, etc) depending on the design code applicable to each component.
5.72	5.2.6.1 Core Inlet Plenum	The design takes into account operational and seismic loads.	More information is needed on the load combinations and to what standard.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
			Response: It is described in the next paragraph.
5.73	5.2.6.4 Fuel Clamps	Failure to clamp effectively will result in a Fuel Assembly being dragged out of the core by the PCS flow, a negative reactivity insertion of at least 2150 pcm. The fall in core pressure drop leads to a reactor trip (FSS and SSS).	Has this scenario been examined in more detail, with respect to impact of the bypass flow on the remaining elements, the fuel element dropping back into the core, and halving time trip to detect the FA removal from the core.
			Response: This event has been analysed in Chapter 16, Section 16.8.2.1 where the failure of this component is considered as very unlikely (please refer to the response to Question 5.7 where the safety margins that apply to the Fuel Clamps are indicated). In this sense, the event of the dragging of a FA due to this failure is considered as beyond the design basis. See also the response to Question 16.58.
5.74	5.2.6.5 Reflector Vessel and Core Chimney (5.2.6.6)	The most demanding load condition on the RV is associated with pulling a vacuum on the vessel for drainage of the heavy water (every 10 years).	The need to use a vacuum to remove heavy water is questioned. Alternative means of drying or draining the tank should be investigated. It is likely that the SSS will cause an RV dump during an earthquake. The load combinations should include, seismic, hydraulic head, and helium gas expansion loads associated with the dump.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
			Response: Other alternatives have been analysed but it was concluded that the proposed method is the most appropriate. Other methods imply the use of fluids or substances to clean the Reflector Vessel that would generate additional radioactive waste (tritium). It is noted that this operation is anticipated to be necessary once every 10 years. Note that the Reflector Vessel has been designed to withstand the loads arising from this operation plus other loads that can be present at the time of the operation (e.g. the effect of irradiation induced growth). The condition of the Reflector Vessel material will be assessed through the Materials Surveillance Program that will be implemented from the commencement of the reactor operation. Analysis of the load combinations and resulting stresses due to actuation of the SSS during a seismic event will be carried out during the Detail Engineering phase.
5.75	5.2.6.5 Reflector Vessel and Core Chimney (5.2.6.6)	A finite element analyses using MSC/NASTRAN has been used to look at the load cases.	There is no mention of any fatigue analysis for the loading cases. Such an analysis should be undertaken including all operational cycles and spurious trips.
			Response: During Detail Engineering, an identification of items where fatigue could be present will be carried out and appropriate analysis will be included for those components. For an earthquake, the number of peak amplitudes for each seismic event is taken as 10, in accordance with IAEA Safety Guide 50-SG-D-15. For this low number of cycles a fatigue analysis is not required.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
5.76	5.2.7 Seismic Verification of Reactor Structures	The reactor core structures are estimated to have fundamental oscillation frequencies around 30.7 Hz. Such high frequencies are not affected by seismic loading, since they approach the zero peak ground acceleration.	A table should be provide of the frequency of oscillation of all the components and structures within the reactor tank
			Response: The oscillation modes of reactor pool internals will be determined during Detail Engineering and will be reported in the FSAR.
5.77	5.3.1 Fuel assemblies	The general description shows that cadmium wire is used as a burnable poison. The method of securing the cadmium wires in slots is shown in Fig 5.3/3.	The cadmium wire would appear to be in direct contact with the water, so there is a need to consider corrosion and activation of cadmium crud that goes through the core. The cadmium has a relatively low melting point (325 C), so it could be lost in a transient that heats up fuel, with an associated positive reactivity addition. Has either of these matters been considered.
			Response: See response to Question 5.9 relating to corrosion of cadmium wire. Please see response to Question 5.14 on Cd wire temperature under transient conditions.
5.78	5.3.1 Fuel assemblies	The reactor core is loaded with plate type fuel assemblies containing uranium silicide. The fuel type has been successfully operated in many research reactors.	The question of re-processing is an issue for silicide fuel, and this should be addressed in this section of the PSAR.
			Response: The PSAR deals with the safety of the reactor for its operation. The question of reprocessing spent fuel is a separate issue covered in Appendix 3 of the Application.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
5.79	5.3.2.7 Chemical Environmental Effects	Corrosion of aluminium and its alloys is one of the limiting factors on fuel lifetime. The water chemistry specified for the reactor pool and core prevent uniform corrosion from being a hazard.	Galvanic corrosion is possibly an issue for the cadmium wire embedded in the fuel, and exposed to water. Also there are other sources of potential galvanic actions in the reactor structure (such as aluminium in contact with stainless steel).
			Response: See responses to Question 5.9 and 5.77 for corrosion of cadmium. Dissimilar metal joints are discussed in Sections 5.9.2.4 and 5.9.6.5.
5.80	5.3.4.3 Structural Stability of the Fuel Plates	Resistance to buckling of the fuel plates due to thermal expansion against the side plates is considered.	It is not clear why the estimate of the compressive stress from lateral expansion ($8*10^{-5}$ M Pa) is compared with the 0.15 M Pa from the pressure differences. These stresses are at right angles to each other.
			Response: The two stresses are compressive and act along the plane of the fuel plates. Therefore they are parallel and cumulative. The compressive stress generated by thermal expansion of the fuel plates, constrained by the side plates is negligible. It is compared with the compressive stress generated by pressure difference to illustrate this point.
5.81	5.5.2 Reactivity Regulation	The Control Rods can be controlled automatically by the RCMS and manually from the Main Control Room (MCR).	There needs to be more information on manual control; and when it is permitted. For example is the reactor under manual control during start-up and low power operation. Are the design defences against the simultaneous movement of more than one CR affected by manual control?

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Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
			Response: Manual control is always allowed. In fact, it has priority over automatic control. There are soft and hard protections against the simultaneous extraction of more than one CR. The manual control allows the operator to select one and only one out of the five CRs to move. Hard interlocks are provided in the CRDs that prevent the simultaneous extraction of more than one CR.
5.82	5.5.2.2 Reactor Regulation - Design Bases	Design bases (a) to (g) are specified and cover speed of extraction/insertion, independence between safety (FSS) and control (RCMS), and design for low maintenance, easy access and simple assembly, disassembly.	Maintenance of a CR requires a CR to be dropped out of the core. It is not clear if this is done with the fuel removed and the SSS dump in place. Ease of maintenance should not be at the expense of increased likelihood of a failure and a reactivity insertion.
			Response: Procedures and details for inspection and testing will be developed during Detail Engineering and will be reported in the FSAR (see Section 5.5.2.7.2). Maintenance does not necessarily mean that the CR will be fully dropped out of the core. It is envisaged that most maintenance can be performed by removing the shock absorber lock that lowers the CR by 80 mm (corrected from 125 mm in an erratum) (see Section 5.5.2.5 page 5.5-4). The need for dumping the reflector during CR maintenance operations will also be evaluated during the Detail Engineering phase. Procedures for removal of a CR will be developed during Detail Engineering and will be reported in the FSAR. Please also refer to Section 5.7.5.6.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
5.83	5.5.2.4 Reactivity Regulation - Description	The CRs are shown in Figs 5.5/2 to 5.5/4. . The CRs are connected with the CRD room through penetrations through the bottom plate of the reactor pool.	An area of concern is the demonstration of the seismic fragility of the anchorages connecting the CRs to the table and the table to the floor. The detail calculations are required since an anchorage failure could prevent the CRs dropping into the core.
			Response: A detailed analysis of the CRD will be carried out during the Detail Engineering phase. The report will include modal shapes for the CRs and the whole assembly with the structure. Stresses will be calculated and reactions at anchoring positions with bolt verifications will be carried out.
5.84	5.5.2.5 Reactivity Regulation - Operation	The logic of the RCMS allows only one CR to move at a time. Hardwired watchdogs prevent more than on CR being moved at the same time. Unique pulse trains are used to maintain a CR at a set position, or move in or out of the core.	There is a reliance on the RCMS to ensure the CRs are moved individually and at correct speed. Will the RCMS be effective under manual control in the same way as it is under auto-control. An FMEA is probably necessary to examine the range of failure modes and effects under both manual and automatic control.
			Response: Engineered design limits the speed of CR withdrawal to 3.5mm/s. The hard-wired watchdog unit performs in the same way under both automatic and manual control. An FMEA for the CRDM has been performed. It was considered in the FMEA that analysing different modes of operation for the CRDM was inappropriate.
5.85	5.5.2.5 Reactivity Regulation - Operation	The effective drop of the CRs into the core is monitored by the FRPS, and success of the FSS is signalled if 4 out of 5 drop. If only 3 oo5 drop in the SSS is actuated.	Why does the SSS not actuate if there is failure of any one of the CRs to drop into the core on demand. The reactor should not be permitted to return to power on a CR failure to drop without a major investigation, so actuation of the SSS is not a "poison out " issue.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
			Response: The design limits are such that the FSS will shutdown the reactor safely with the required margin (> 1000 pcm) should one CR fail to drop within the specified time. Consequently there is no need to actuate the SSS in this circumstance. Procedures for investigation and reporting of abnormal operating occurrences will be detailed in the operating licence submission and would not permit a return to power on a CR failure to drop without an appropriate investigation. Please also refer to Section 5.5.2.5 for the conditions monitored by the FRPS to allow the CRs to be recovered for operation (conditions to switch the electromagnets on).
5.86	5.5.2.5 Reactivity Regulation - Operation	During maintenance a CR can be released from the CRD by manually operating an incorporated locking rod. The shock absorber lock can be removed allowing the CR to be lowered for maintenance.	Each CRD has its own locking rod, so it is only administrative control that prevents maintenance removal of more than 1 CR. Such maintenance should only be done with the fuel removed from the core.
			Response: The removal of the shock absorber locking rod drops the CR by 80 mm. The effect on reactivity is estimated to be negligible when this occurs. Please also refer to Section 5.7.5.6 and the response to Question 5.82.
5.87	5.5.2.7 Inspection and Testing of CRs	A full scale proto-type of a CRD has been constructed to confirm a range of operational and life cycling characteristics.	Will a proto-type be provided to ANSTO for ongoing test and maintenance planning purposes. There is also a need to consider full scale seismic testing of a CRD.
			Response: A prototype will be constructed by INVAP and will be tested in INVAP's workshops. The necessary seismic tests will be carried out on the prototype.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
5.88	5.5.2.8 Design Evaluation	The SRPS initiates the SSS on the basis of a timer that looks at successful drop of 4 oo 5 of the CRs.	The engineering of the SRPS into the FSS function has the potential to reduce separation and independence. It also appears that the SSS initiation signal is not from a direct measurement of CR position, but indirect via a timer. What controls this timer, the FRPS, SRPS or the RCMS?
			Response: The SRPS is not engineered into the FSS function. The SRPS monitors FSS parameters and initiates the SSS should those parameters exceed trip values. (Please see Chapter 8, Section 8.2.3.4.2.1). The SRPS controls the timer. Chapter 8 of the PSAR discusses the various functions of the FRPS, SRPS and RCMS in more detail.
5.89	5.5.3.6 Instrumentation	A table shows the CR proximity switches, some of which are on the RCMS and one set (shock absorber) on the SRPS and PAM. This SRPS proximity switch is 2oo3 and provides an alarm and triggering of the SSS following a demand on the FSS.	The logic of the demand signal and proximity switch appears to also involve a timer in the SRPS. More details are required on how the system works and how separation is maintained between SRPS, FRPS and PAM.
			Response: This is discussed in Chapter 8 of the PSAR.
5.90	5.5.3.9 Seismic Evaluation of CRDs	A preliminary analysis of the CRDs has been carried out in order to evaluate their response under the SL-2 seismic event. The deformed model is shown in fig 5.5/6 and the natural frequency is estimated at 34 Hz.	The failure of the anchorages connecting the CRDs to the table, and the table to the floor should be investigated, and should be capable beyond SL-2. The estimate of the natural frequency should be confirmed, but it does appear to be in the range where the Zero Period ground acceleration is relevant.
			Response: See response to Question 5.83.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
5.91	5.5.4 Second Shutdown System	The SSS fulfils eleven design bases (a) to (k) in order to provide a diverse, independent back-up to the FSS. The performance requirement is 3000 pcm in 15 seconds and that eventually a shutdown margin of 1000 pcm is attained.	It is noted that, unlike the FSS, the SSS is not claimed to be fail-safe on loss of power, although it may be on loss of air. Thus, for frequent initiators, such as loss of power when diversity is most important, the SSS will not operate unless the UPS is lost, or it is triggered from SSS trip parameters.
			Response: The FSS is sufficient to cope with the postulated event (loss of main electrical power) as stated in Section 5.5.5.2 and demonstrated in Chapter 16, Section 16.7. In addition the SRPS continues the monitoring of its safety system settings during this event and is capable of triggering the SSS if one of them is exceeded. The SSS is failsafe on loss of power, in that if both normal supply and UPS power is lost to the SSS, the SSS will act to shutdown the reactor. However, the SSS will not trip on loss of normal supply to the facility.
5.92	5.5.4 Second Shutdown System	When the SSS actuation is triggered by the SRPS the heavy water in the RV and the RC&PS expansion tank is drained through a single pipe to the heavy water storage tank.	The drain down time is long (>15s) and there is no drop in RV level for about 5s, until the expansion vessel is drained. There should be investigations on design improvements to decrease the drop time of the SSS.
			Response: The requirements on the SSS are for shutdown margin and mission time and are based mainly on the requirements of the safety analysis (including accident conditions). The adequacy of the performance of the SSS is demonstrated in Chapter 16 where it is shown that the SSS adequately protects the reactor from the postulated design basis accidents. Please also see the response to Question 5.29.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
5.93	5.5.4 Second Shutdown System	The position switch on each of the 6 SSS actuation valves and the pressure switches on the SSS compressed air are on the RCMS.	Why have this instrumentation on the RCMS – it should be part of the SRPS.
			Response: The pressure switches do not belong to the SRPS as the compressed air is not necessary for the SSS to perform its safety function. Table 5.5.4.7.2 indicates that these switches belong to the SRPS, however, this is incorrect, as they belong to the RCMS. This will be amended in the next revision of the PSAR. The position switches on the 6 trigger valves are not required to be part of the SRPS as the verification of the actuation of the SSS is indicated by the neutron flux level indications and the water level indication in the heavy water storage tank.
5.94	5.5.4 Second Shutdown System	The discharge valves will be cycled once per normal shutdown and the system actuated once per year or major shutdown.	The in service test proposals suggests there will be very few checks on effective dumping. In view of the SSS importance it is not clear there is not a dump test during each shutdown to measure the SSS operational effectiveness.
			Response: The procedures for testing the SSS will be developed during Detail Engineering and will be reported in the FSAR.
5.95	5.5.4.10 Seismic Evaluation of the SSS	Seismic evaluation of the SSS components will be performed at the detail design stage.	A seismic evaluation has been carried for the FSS, but not for the SSS. The seismic evaluation of the SSS should be available for this PSAR, and should cover all systems, structures, components and load combinations.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
			Response: A seismic evaluation has been performed for the Reflector Vessel. See Section 5.2.7. A complete analysis of the rest of the SSS components will be developed during the Detail Engineering phase and will include all SSS components and their anchorages. It will also include a detailed analysis of the piping inside the reactor pool, joined to the reflector vessel.
5.96	5.7.2 Nuclear design – Definitions	The default state is the hot, power and beginning of cycle state (BOC).	The default state may not be the most severe for accident analysis. The burn up of the CRs should be included in all the transients considered in the PSAR Ch.16.
			Response: ANSTO agrees. The most severe state is always considered for accident analysis. The default state defined in Section 5.7 is used for nuclear design description purposes only.
5.97	5.7.4.2 Reactivity design Bases	All temperature and void coefficients associated with the fuel and core are negative for all operating states and accident conditions, unless it is shown that the coefficient has an insignificant effect.	Need to check if there are any circumstances where there could be positive feed back even if insignificant. Some consideration needs to be given in the accident analysis (Ch.16) of loss of Xenon from the core if there is clad damage. Can there be re-criticality?
			Response: All circumstances in which reactivity feedback is expected to be important have been considered. The core will always be sub-critical with a shutdown margin of at least 1000 pcm (in the case of single failure of the FSS or actuation of the SSS with the core in its cold, Xenon free state).

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Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
5.98	5.7.4.2 Reactivity design Bases	Criterion 4 states that the core shall be subcritical, with a shutdown margin of at least 1000pcm with any one of the CRs out of the core.	Maintenance of a CR introduces a potential for hazard, since one CR is out of core, and a human error could lead to the removal of an additional CR. The safest arrangement for such maintenance would be with the fuel removed from the core, and the SSS activated. What is the operational intention?
			Response: The design ensures that during core reshuffling and maintenance operations the core should be subcritical with a shutdown margin of 3000 pcm and of 1000 pcm with the most effective CR plate out (see Section 5.7.4.6). Criterion 4 (and 14) does not imply that it is intended to withdraw any CR during maintenance. Rather it is a criterion to ensure an appropriate shutdown margin. Compliance with these criteria have been verified during the preliminary design (see Table 5.7/18). See also Section 5.7.5.6 for the anticipated operational intentions during refuelling and maintenance.
5.99	5.7.4.2 Reactivity design Bases	In all 17 criterion are given for the reactivity design bases.	There is no information on how much reactivity is controlled by the burnable poisons for a typical BOC core loading. Is this information available?
			Response: These are design bases. It is not appropriate to include a reference to the reactivity controlled by burnable poison. Yes, this information is available.
5.100	5.7.5.1 Rate of Insertion of Reactivity –Design Bases	The fuel management strategy for the reference core comprises 3 chains, in which spent fuel is removed and fresh fuel inserted.	The fuel management strategy is very complex and involves the handling and shuffling of all fuel elements at every shutdown. The number of movements increases the likelihood of re-fuelling errors. More information is required on the fuel management strategy and the implications for the reactivity and thermal hydraulic criterion.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
			Response: As stated in Section 5.7.5.1 the strategy presented here is for calculational convenience. The actual strategy to be used will be determined during the Detail Engineering phase and will be presented in the FSAR. One of the goals in establishing the strategy will be to minimise the number of movements. Information on the effect of the fuel management strategy on the reactivity and PPF are presented in Sections 5.7.5.1 and 5.7.5.2.
5.101	5.7.5.2.1 Neutron Flux and Power distribution Measurements	Power distribution within the core will be measured using an appropriate technique such as foil wire activation analysis. Such measurements will be made after fresh fuel is loaded and at low power (400kw).	Low power operation appears to be necessary every shutdown after fuel shuffling. This mode of operation is with natural circulation, so the FRPS has a truncated level of protection. This mode of operation should be covered in Ch.16 and the PSA.
			Response: An analysis of reactivity transients in lower power operation mode is currently being performed. The results will be provided to ARPANSA. There is complete protection appropriate for this mode of operation.
5.102	5.7.5.3 Reactivity Feedback Coefficients and Kinetic Parameters	Table 5.7/6 and 5.7/7 shows the calculated reactivity feedback coefficients and kinetic parameters respectively.	The coefficients are all based on calculation and not measurements. They need to be compared with values from similar reactors and any supporting measurements. The void coefficient values in the core and the reflector are considerably greater than the temperature coefficients, and should be based on experimental results as far as possible.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
			Response: As stated in Section 5.10.2.6.2, calculated feedback coefficients have been compared with those obtained by other parties of the IAEA benchmark. The results show differences with other codes lower than 4% for the case of fuel temperature feedback coefficient and void coefficient. The power feedback coefficient has been measured in the ETRR-2 reactor and compared with the calculations. Measured $\alpha_P = -0.00059$ \$/kW Calculated $\alpha_P = -0.00052$ \$/kW Some verifications of the effect of void has been carried out also in the ETTR-2. During the RRR commissioning, tests will be carried out to measure the power feedback coefficient and the void feedback coefficient.
5.103	5.7.5.4.1.2 Control Rod Depletion	The reactivity changes and the efficiencies of the CR plates due to depletion of the absorber material (silver-indium-and cadmium) have been evaluated to estimate the CR plate lifetime. From table 5.7/11 the depletion is 16% after 8 years.	The intention is to replace all the CRs every 8 years during a major shutdown. Why is not necessary to replace S5 (the centre control rod) more frequently since it is always in the core.
			Response: The criteria for replacing all the CRs is determined by the shutdown margin requirements.
5.104	5.7.5.6 Criticality during Refuelling and Maintenance	The reflector vessel will be emptied for particular maintenance cases where the negative reactivity worth of the CRs is altered. It is anticipated that the FAs will be removed from the core prior to any CR replacement operation.	In view of the fact that all the fuel elements are removed from the core as part of the monthly re-shuffling all CR maintenance should be only with the core empty.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
			Response: It is not anticipated that all FAs will be removed during core reshuffling. The maximum number of FAs that will be out of the core during the reshuffling will be 2 (i.e. the spent FA and the FA being moved to another position). Note that the possibility of an error of the operator in the fuel loading is minimised in this way while the necessary shutdown margin is ensured by the five CRs being inserted in the core. It is intended to remove all FAs prior to any CR replacement operation. Most CR maintenance (or CRD maintenance) may be performed in the standard way (See response to Question 5.82).
5.105	5.7.5.7.2 Thermal-hydraulic Stability	The limits on the core configuration, power and PPF not exceeding 3 allows the uncoupling of neutronic and thermal hydraulic calculations.	It is not clear why the stated limits allows the uncoupling of neutronic and thermal hydraulic codes. Is this also the case for the accident sequences modelled in Ch.16 of the PSAR.
			Response: The stated limits allow a maximum power density to be assigned and the actual power density will be less than this value. Thermal-hydraulic calculations can then be performed assuming this maximum power density. The reactivity transient accident analyses are performed using coupled neutronic and thermal-hydraulic codes.
5.106	5.8 .3.2 Thermal Hydraulic Design-Power Peaking Factor	The Power Peaking Factor (PPF) quantifies the non-homogeneous spatial distribution of the heat flux over the core. A conservatively high basis for the PPF of 3 is adopted for every core operational configuration.	There is a need to compare with other pool reactors to see if the PPF of 3 is a reasonable basis for the RRR core.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
			Response: Please see Table 1.4/1-3 in Chapter 1. As noted in this section, actual PPF values can be calculated. As noted in Section 5.7.7, the conservatism of the assumed PPF value of 3 is demonstrated by the comparison with calculated RRR PPF values of about 2.1 (see for example Table 5.7/18).
5.107	5.8 .3.6 Thermal Hydraulic Design - Critical Heat Flux Design Basis	For burn-out, flow redistribution and departure from nucleate boiling (DNBR) a design basis value of 2 is chosen for all "nominal" operating conditions.	It is not clear what "nominal " operating conditions covers and this should be clarified. In Ch.6 of the PSAR what is assumed if these ratios are not satisfied. Is fuel damage assumed? Is the ratio of 2 maintained for all anticipated operational occurrences and design basis accidents?
			Response: "Nominal'' refers to the current best-estimates of normal operating conditions. These are summarised in Table 5.8/5 and row 4 of Table 5.8/19. Section 5.8.3 specifies thermal-hydraulic margins to different phenomena for whatever the nominal conditions are. The stated design ratio requirements are for normal operation and ensure no fuel damage. The accident and transient analyses of Chapter 16 also demonstrate that there is no fuel cladding damage. However, steady-state criteria (DNBR ratio > 2, etc) are not relevant to transient analyses. INVAP's experience indicates that by designing a core that under nominal conditions has a margin of 2 to this critical phenomena, it ensures there is an appropriate margin to the safety limits to enable that all operational transients and design basis accidents will not jeopardise the integrity of the fuel.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
5.108	5.8.4.1.1 Flow Instability Phenomena	Flow instabilities must be avoided in heated channels as flow oscillations affect the local heat transfer characteristics and may induce a premature burnout. As a general rule the operation of research reactors in the range 1 MW to 50 MW cooled and moderated by water at low pressures, is limited from the thermal point of view , by the flow redistribution phenomena. Fig 5.8/1 compares the correlation used in the RRR core against experimental data. In the case of DNBR the Mirshak correlation is used and is shown in Fig 5.8/2.	An independent set of calculations will be necessary to demonstrate that the flow instability margin and DNBR limit provides an adequate safety margin for all operational conditions, anticipated operational conditions, and design basis accidents.
			Response: Independent calculations are being performed by ANSTO. The design bases of Chapter 5 are not applicable to the transient phenomena of anticipated operational occurrences and design basis accidents, since, unlike the case for steady conditions, triggering the initiation of DNB or redistribution during transients may not result in unacceptably high temperatures.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
5.109	5.8.4.1.3 Low Flow Burn out Phenomenon	For low power(1.5 MW) and low flow velocities (<0.5m/s), or within the natural circulation regime, a periodic instability known as "pulsed boiling" can appear. In this mode the coolant inside the channel heats up, bubbles appear and pressure increases. Due to the pressure rise the coolant is flushed out at both ends of the coolant channel. Sub-saturated coolant re-enters the channel and the sequence starts again. Figures 5.8/4 and 5.8/5 show the results of measured burn out against the mass flux and correlation predictions.	It is not clear if the low flow burn out phenomenon have been included in the PARET modelling for the design basis and beyond design basis accidents described in Ch.16 of the PSAR. This includes LOCAs, LOFAs, reactivity transients, and the 15 seconds or so of operation at power, should the FSS fail on demand and before the reactor is eventually shutdown by the SSS. Note that in period before the SSS is effective the power would be much greater than 1.5 MW.
			Response: PARET examines reactivity accidents only. It does not examine LOCAs or LOFAs. PARET has been validated against the SPERT experiments, which include data from zero flow conditions. Irrespective of uncertainties in the DNB prediction, there is considerable margin to DNB in the LOCAs and LOFAs examined in Chapter 16; in all LOCA and LOFA transients the reactor is shutdown shortly after the occurrence of the initiating event. See also Appendix 1 on pulsed boiling.
5.110	5.8.7.3 Calculated Thermal-hydraulic Parameters for the Shutdown State	Table 5.8/10 shows the calculated thermal-hydraulic parameters for the shutdown state. The maximum heat flux is 5.1 watts/cm^2, the pulsed boiling heat flux is 8.5 watts/cm^2, the burn-out heat flux is 34.3 watts/cm^2, and the burn out ratio (BOR) is 6.7.	While the BOR may be 6.7 the margin to pulsed boiling is only 1.7. Thus it is possible the pulsed boiling mode is approached for normal shutdown conditions. Further work is necessary to demonstrate that the pulsed boiling mode, which is an instability will not damage the fuel plates (either from over heating or vibration). In transients the repeated voiding and filling of channels across the core will have a reactivity effect that needs examination.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
			Response: Pressure instabilities associated with low flow pulsed boiling are necessarily small since pulsed boiling occurs at flows where inertial forces are small. Pressure variations are expected to be much less than the pressure drop at normal flows. See, for example, Thermal Eng **35** (12), 689-693, 1988. Temperature variations for such instabilities are also small. Almost by definition, overheating prior to DNB cannot occur. If the reactor is shutdown the reactivity variation would be of no consequence. See also Appendix 1.
5.111	5.8.12.1 Bulk Production Irradiation Facilities	For design purposes a maximum heat flux of 110 watts/cm^2 is used for the molybdenum rigs, but this may be increased to 150 watts/cm^2 with future rig designs.	What is the basis for these heat flux values? The molybdenum cans in HIFAR are currently limited to 55 watts/cm^2, which is based on a suitable margin against the experimentally established burn out value of about 95 watts/cm^2.
			Response: These heat fluxes were set by ANSTO as a Contract requirement. HIFAR rig OLCs apply to the flows and geometry for HIFAR and are not relevant to the RRR. Limiting heat fluxes can be increased by, for example, increasing flow. The design limits given in Section 5.8.3.7 will be satisfied for the detailed rig design in the Detail Engineering phase.
5.112	5.8.13 Bulk Production Irradiation Facilities-Shutdown State	When the reactor is shutdown the RSPCS pumps remain operational for a period of time. The pumps are eventually shutdown and downward flow is maintained by pump coast-down. When the flow stops there is a flow reversal and natural circulation(upward flow) removes the decay heat.	The sequence is important, particularly for loss of power to the RSPCS pumps (see PSAR Ch.16). Do they have a long coast-down like the main PCS pumps?

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
			Response: As discussed in Section 16.7.5.3.3, the RSPCS flow coast-down following a loss of power is such that flow reversal occurs after some 120 seconds.
5.113	5.9.1 Reactor Materials	Three basic structural materials are used in the in the reactor facility, namely stainless steel (AISI 304 l), zirconium (Zircaloy –4), and aluminium (alloys 6061 and 5052).	There is no mention of other metals that may be in the reactor, such as the unclad cadmium wires used as a burnable poison. A list should be provided of all materials that come into contact with the reactor water (including materials within the pools).
			Response: This is provided in Table 5.9/1. On the Cd wires, please see responses to Questions 5.9, 5.77 and 5.79.
5.114	5.9.2.4 Dissimilar Metal Joints	Stainless steel to aluminium joints are acceptable within the operating temperature range and water quality requirements of the reactor.	This differs from the HIFAR philosophy. In HIFAR sacrificial plates are used in the primary circuit to connect aluminium and stainless steel components. What is the bases for not considering such plates in the RRR.
			Response: There are few components in the reactor pools built in aluminium. In the RRR the aluminium alloy used is series 6XXX which has better corrosion resistance characteristics that the aluminium alloy series 1XXX used in HIFAR. Strict water quality maintenance in the RRR will have a very important role in limiting the corrosion phenomenon. In addition, all aluminium components are removable for inspection. Several coupons of aluminium/stainless steel will be placed in the reactor pool as part of the material surveillance program to evaluate corrosion. See also response to Question 5.64.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
5.115	5.10.2 Nuclear Design Codes	Most of the design results have been obtained by using computer codes and programmes either developed by INVAP or other recognised organisations. The MTR-PC system was developed by INVAP to perform neutronic, thermal-hydraulic and shielding calculations. There has been extensive verification and validation against measurements from operating reactors.	In the case of the validation and verification of the code for natural circulation (CONVEC) there is no mention of verification validation against ETRR-2 results (see 5.10.3.3.3). why have the ETRR-2 results been excluded since they appear most relevant to the RRR.
			Response: Validation of CONVEC would require the collection of data on channel velocities, fuel cladding temperatures and pressure drop through the core, all of them parameters that cannot be measured with the instrumentation of a research reactor.
5.116	5.10.2 Nuclear Design Codes	Most of the design results have been obtained by using computer codes and programmes either developed by INVAP or other recognised organisations. The MTR-PC system was developed by INVAP to perform neutronic, thermal-hydraulic and shielding calculations.	Do ANSTO have access to or copies of any of these design codes. Please list those that ANSTO have.
			Response: ANSTO does not currently have access to the MTR-PC system.

APPENDIX 1: ADDITIONAL INFORMATION FOR QUESTIONS 5.109 and 5.110

PULSED BOILING and ONB HEAT FLUXES

The pulsed boiling heat flux presented in Tables 5.8/10 and 5.8/11, is derived from the burn-out heat flux correlation given in the same chapter, for low velocities:

$$q''_{BO} = D\,(0.023\ DT_{sub} + 4.56),\ \text{and}\ q''_{PB} = q''_{BO} / 4$$

This correlation is developed for the case in which, two-phase flow conditions under low flow condition in the coolant channel is reached. It means that, if the coolant were in a two phase flow condition, the burnout heat flux would be q''_{BO}.
However, if the coolant temperature is below the saturation temperature, the correlation represents no limiting value, as the system is in a single-phase condition.
To use this correlation under other coolant conditions as a design limit is a conservative assumption and suitable for design purposes. Doing that, it is possible to assure that even in case of an abnormal low flow condition during the natural circulation process, reaching the two-phase condition in the channel, no burnout will occur.

To verify the coolant temperature condition, a simple energy balance for the shutdown state gives the following results:

Power = the power necessary to be delivered at one coolant channel in order that the temperature at its outlet reaches the saturation temperature.
Power = Flow area x Coolant velocity x Coolant density x Cp x DT_{sub}
Power = (0.07 x 0.00245 x 0.13) m^3/s x 970 kg/m^3 x 4.18 kJ/kg/C x (120 – 40) C
Power = 7.2 kW

Note that a velocity = 0.13 m/s has been used as it is a value representative of the natural circulation regime

q" = Power / Heat transfer area of one channel
q" = 7200 W / (2 x 6.5 x 61.5) cm^2
q" = 9 W/cm^2

It implies that a **uniform** heat flux higher than 9 W/cm^2 is needed to reach the saturation temperature at the channel exit and the pulsed boiling phenomenon.

But the heat flux distribution has a **cosine** shape so, considering that the cosine has a peak to average ratio of 1.3, the equivalent maximum heat flux needed to reach the same uniform heat flux value is around 1.3 x 9 W/cm^2, that is, 11.7 W/cm^2. This value is around 20% above the maximum heat flux 100 seconds after shutdown. This means that all the hot channel is in single-phase.

In case of a Loss of Flow Accident, and from figures in Chapter 16, Section 16.9, it can be seen that the coolant outlet temperature, in the hot channel, is well below the saturation temperature, so no pulsed boiling condition can be reached, and, the maximum wall temperatures in the hot channel is lower than the ONB temperature.