Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
			General Response: Chapter 2 sets out the safety objectives and engineering design requirements. Many of the questions and issues raised by ARPANSA relate to the realisation of these requirements in the design and safety analysis. These matters are dealt with substantively in later chapters to where cross reference is made here.
2.1.	2.2.2.1 Safety Criterion 1- Occupational Radiation Dose Limit	The constraint will be 15mSv per year.	In its review of ANSTO's operating nuclear installations ARPANSA has queried the 15mSv constraint (see question on section 12.1.1(c) Optimisation). Please explain why there is no numerical criteria for operator collective dose.
			Response: As is stated in the response to Question 12.3, ANSTO is currently reviewing the dose constraints in its response to the existing facility licences. The RRR facility doses will be consistent with these constraints.
2.2.	2.2.2.2 Safety Criterion 2-Public Radiation dose constraint	Dose to any member of the public (at the 1.6km buffer zone boundary) constrained to 0.1 mSv per year from the RRR operations. There is a reference to a regulatory discharge authorisation which will in practice to 0.01mSv per year.	It should be ensured that there is consistency between ANSTO's references to their commitment to the ALARA objective of 0.01 mSv, since it is from the LHSTC (as stated), while Chapter 1 suggests it is from the RRR.
			Response: Consistency will be ensured and the PSAR clarified to make this commitment clear.
2.3.	2.2.2.4 Safety Criterion 4—Safety Systems Reliability Targets	The Reliability of Safety Category 1 systems will be shown to be better than 10^{-3} failures per demand, and the total frequency of exceeding any of the safety limits for the reactor core will be 10^{-4} per year.	The Safety limits for the reactor core need to be more clearly identified. For example, do they include thermal hydraulic criterion as well as clad temperature limits? ARPANSA notes from the PSA that the FRPS only just satisfies the 10^{-3} target.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
			Response: The safety limits for the reactor core are discussed in Chapters 5 and 16 with respect to normal operation and accident conditions. They will be clearly identified in the OLCs as they are developed during detail engineering. Refer also to Chapter 17, Section 17.2.
2.4.	2.2.2.5 Safety Criterion 5—Maximum public dose from Design Basis Accidents.	The commitment is that the dose to individual members of the public will be less than any the minimum intervention levels recommended for any emergency.	Reference is required to what the emergency criteria minimum intervention levels are (or where discussed in the PSAR) and also reference to collective dose or land alienation commitment as outlined in the ARPANSA RAPs.
			Response: Reference will be provided.
2.5.	2.2.2.6—Safety Criterion 6—Risk Dose Criterion	Refers to Table 2 of the ARPANSA RAPS and the dose frequency objectives for accidental releases.	There is no recognition that Table 2 requires the total frequency in a dose range to be satisfied. The review of the PSA indicates that individual fault sequence frequencies have been plotted. Please clarify.
			Response: ANSTO is aware of the need to consider the total frequency in a dose band when comparing with the ARPANSA's RAPs. Chapter 2 will be amended to clarify this point in the next revision of the PSAR. For a discussion of the results of the PSA, please refer to Appendix A of the PSAR.
2.6.	2.3.1 Quality Assurance	Refers to project Quality Assurance program and the Design Plan. The references also include ARPANSA RG-5 (Design Guideline), IAEA SS 50-C/SG-Q, and IAEA DS272—draft report on safety requirements for research reactors.	QA is discussed in detail in Ch.18 and it is most important that the role of ARPANSA is appropriately recognised in the design, construction, and commissioning QA system.
			Response: We agree and will ensure that ARPANSA is fully involved in the design, construction and commissioning QA system.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
2.7.	2.3.2.3 Provision of Barriers	The listed barriers to release to the atmosphere are the fuel matrix, the fuel cladding, the primary coolant boundary, the pool water and the containment.	ARPANSA notes that there appears to be no reliance on an Emergency Core Cooling System (ECCS) based on the LOCA sequences identified in Ch.16 and the PSA.
			Response: Correct, there is no reliance on any ECCS for design basis LOCA sequences since the EMWS not an ESF and does not constitute a barrier. See Chapters 6 and 7 for further details.
2.8.	2.3.2.4 Proven Engineering Practice	In general the design is based on conservative design and feed back from experience. Where novel design features are used they will be based on extensive research and development (R & D).	Please reference where in the PSAR or supporting documents the novel features of the design are demonstrated by extensive R&D. ARPANSA expects this should include the flap valves, the reflector tank, the software based protection system (FRPS), the containment design, and the fuel element and chimney design.
			Responses: The novel design features are discussed in the appropriate chapters related to the functionality of the feature, specifically Chapters 5, 6, 7 and 8. Note that of those identified in the ARPANSA comment, only the reflector vessel is of "novel design", the other features all being developed from existing designs.
2.9.	2.3.2.5 Design Simplification	The design is to be kept simple as practicable and without unnecessary complexity.	Please clarify the following with respect to design simplification: the interface and mass transfer between the pool and the containment atmosphere, particularly following sealing of the containment; and in the interactions between the FSS and SSS.
			Response: The interface and mass transfer between the reactor pool and the containment atmosphere is discussed in Chapter 7. The interactions between the FSS and the SSS are discussed in Chapters 5, 8 and 16.
2.10.	2.3.2.6 Fail Safe Features	The fail-safe principle is incorporated in the design.	Please clarify the fail to safety for the SSS on the loss of power.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
			Response: The SSS is discussed in Chapter 5. The actuation valves are fail-safe in that they fail open upon loss of power supplies dumping the contents of the reflector vessel to the storage tank as discussed in Chapters 5 and 8.
2.11.	2.3.3 Inherent Safety Features.	Inherent safety features which ensure a change in a safe direction in response to PIEs will be used for reactor shutdown, decay heat removal, and containment	Please clarify the extent to which the containment function is inherently safe. For example, does it isolate automatically on loss of power?
			Response: The containment isolation valves are discussed in Chapter 7. They are fail-safe in that they close upon loss of power supplies as discussed in Chapter 7.
2.12.	2.3.5.1 Redundancy	Redundancy provided in safety systems to achieve reliability, meet the single failure criterion, and to meet reliability if there is a potential for undetected failure.	The arrangements for outage control (minimum safety plant configuration) is important. If a two redundant system used then the reactor may have to be shutdown within hours to meet redundancy requirements. The role of the standby power supply (two diesels is a case in point), particularly if there is automatic transfer of loads.
			Response: Agree that the arrangements for outage control are important and they will be reflected in the OLCs as discussed in the relevant functional chapter and Chapter 17.
2.13.	2.3.5.2 Single Failure Criterion (SFC)	There is a commitment to the single failure criterion, particularly in relation unavailability due to maintenance and testing. There is however an intent to justify areas of non compliance with the SFC	The reasons (a) to (e) given for justifying non compliance with the SFC are far ranging. Experience shows that this will be an area where safety and operation may come into conflict. The reasons given are primarily of a risk management nature.
			Response: Agree but note that these reasons for justify non-compliance are generally accepted within the international nuclear industry.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
2.14.	2.3.5.3 Diversity	Diversity will be applied to enhance reliability and reduce the potential for dependent failure.	Diversity is provided in reactor shutdown and protection system (FSS and SSS,--FRPS and SRPS). However, there is no redundancy in decay heat removal unless the shutdown pumps are made a Safety Category 1 system. Item (e) applies to decay heat removal by Natural circulation , viz " the component is a passive component of high quality, upon which reliance is not placed for an extended period"" In the case of Natural Circulation there is dependence for an extended period.
			Response: There is redundancy in decay heat removal by the provision of 4 flap valves and 2 flow paths. Since there is a single core, there are some parts where redundancy is not practicable (eg inlet plenum, chimney). These types of non-compliance with the single failure criterion are acknowledged in ARPANSA RG-5 (89).
2.15.	2.3.5.4 Independence	Independence is to be provided between process control and safety systems, and between redundant systems, sub-systems and components.	The maintenance of independence is always a design compromise. There are features of the design which appear to degrade independence, such as cross connections between the redundant trains of the electrical power supply; the sharing of control and safety in the case of the FSS; routing and layout of cabling; and the interconnection between the FRPS and SRPS. Independence is also an issue in operation in regard to common services, common maintenance, and a common environment. Will lack of independence be justified in the same way as the SFC exemptions above?.
			Response: Clarification will be provided generally along the same lines as for non-compliances with the single failure criterion to justify those situations where independence is degraded.
2.16.	2.3.6 Accident Prevention	The plant will be designed to be highly resistant to anticipated operational occurrences and design basis accidents	Anticipated Transients without Scram (ATWS) are a particular case in point, particularly failure of the FSS (FRPS) on frequent initiating events such as loss of power or loss of flow. The demonstration that the period before operation of the SSS is safe is a key safety matter.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
			Response: Agree. The response of the FSS (FRPS) and the SSS (SRPS) following frequent initiating events is discussed in Chapters 5, 8 and 16.
2.17.	2.3.7 Accident Management	The design will include accident management measures to limit the progression of design bases and beyond design base accidents and to mitigate their consequences. Features (a) to (h) are given as examples.	The regulatory assessment issue is the credit given to accident management features will depend on the extent to which they are designed as Safety Category 1 components and systems.
			Response: Comment noted although accident management features are not claimed as engineered "barriers" and are allocated to Safety Category 2.
2.18.	2.4.1.1 Shutdown System Requirements	Diversity in shutdown is claimed by having an FSS and an SSS, which are based on differing physical phenomena.	The analysis of the reactivity transients included in Ch.16 (and the justification for excluding more severe transients) is a key matter. There is a need to ensure the most reactive state is looked at (End of Cycle (EOC) and Beginning of Cycle (BOC)) as appropriate to the core fuel and Control Rod absorption. The possibility of loss of cadmium burn-able poisons should be considered.
			Response: As indicated in the comment, the reactivity transients are discussed in Chapter 16. Details are also provided in Chapter 5.
2.19.	2.4.1.2 Reactivity Control System Requirement	The reactivity control system utilises the same control rods as the FSS, but the shutdown feature will have priority over process control.	The sharing of control and safety in all the control rods is a matter that needs justification. Safety arguments may rest with the operation of the SSS. Can for example the argument for no inadvertent withdrawal of a number of rods be sustained if they are all involved in control during some modes of operation? (see Ch.16)
			Response: The use of the control absorbers for both control and protection functions is discussed in Chapters 5 and 16.

Checked / agreed:

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
2.20.	2.4.1.3 Reactivity Limits	Reactivity limits will be specified for each reactor irradiation facility, including fixed and movable rigs and experiments.	The Operational Limits and Conditions should identify the limits on reactivity controlled and the speed of withdrawal or insertion. The reactivity limits set by Argentine regulators (ARN) should be compared with those proposed for the RRR.
			Response: Agree, the limits on reactivity control and insertion/withdrawal limits will be identified in the OLCs at the draft FSAR stage on the basis of the safety analysis and the detail engineering.
2.21.	2.4.1.4 Shutdown Margins	Limits will be set for total shutdown margin, shutdown margin of the FSS, and shutdown margin of the SSS, assuming a single failure within the SSS and FSS	The single failure in the FSS case appears to be most reactive control rod does not drop. In the case of the SSS it appears to that only five out of six drain valves open. The margins should also include burn up of control rods, EOC, BOC, and errors in reactivity estimates (see HIFAR reactivity estimate assumptions).
			Response: The analysis of reactivity insertion accidents and the effects of various failures under various operational conditions are discussed in Chapters 5 and 16.
2.22.	2.4.2 Thermal-Hydraulic Design Criteria	The safety limit for fuel meat corresponds with the phenomenon of fuel clad blistering. For all normal operations the thermal –hydraulic design avoids the Onset of Flow Boiling (ONB)	Fuel blistering is a material property not a thermal hydraulic property. For accidents the appropriate thermal hydraulic criterion should be the temperature for the Departure From Nucleate Boiling (DNBR), or the Onset of Flow Instability. Ch.16 should demonstrate that these criteria are met, or assume fuel damage in the channels it is not met.
			Response: Blister formation on the fuel plates is the most limiting damage mechanism for fission product release. Fuel clad blistering is avoided by satisfying the thermal hydraulic design bases as given and discussed in Chapter 5, Section 5.8.

Checked / agreed:

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
2.23.	2.4.2 Thermal Hydraulic Design criteria	To avoid fuel plate vibration, which may result in local overheating, the maximum velocity will be limited to a value ,which ensures a conservative margin to fluid-structure instability effects	ARPANSA notes high velocity through the channels. Please provide more information on the fluid-structure instability velocity, and what margin is specified.
			Response: Details concerning the flow velocity and the associated critical velocity are discussed in Chapter 5.
2.24.	2.4.3 Reactor Core Integrity Requirements	The design conditions for the core components take into account loads from normal operation and design basis accidents. This is to ensure that the reactor can be shutdown, and fuel parameters maintained within specifications.	The integrity of the Reflector Tank (RT) should be included since its inner annulus forms the fuel chimney. The combination of stresses to be considered should include seismic, static water head, and helium expansion stresses associated with drain down (reflector dump).
			Response: The integrity of the reflector vessel is discussed in Section 4.5 and Chapter 5. The combination of stresses includes seismic, static water head and stresses associated with the partial and complete drain down of the vessel.
2.25.	2.4.3 Reactor Core Integrity Requirements	The pool design and operational restrictions prevent the dropping of heavy objects into the core.	The design and operational restrictions may not apply to shutdown (or low power operation), and the possibility of dropped loads should be considered for these modes of operation. Another matter is the extent of damage to the pool liner on a dropped load.
			Response: Pool design and operational restrictions on the movement of heavy objects in the reactor hall are discussed in Chapters 4 and 16. An assessment of the effects of credible loads on the pool liner will be carried out during the Detail Engineering phase.
2.26.	2.4.4 Protection against flow Instability.	A broad margin will be established between normal operation and flow instability phenomenon.	The onset of flow instability in normal operation, anticipated operational occurrences or accidents should assume there has been fuel damage in the affected channel.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
			Response: Agreed: Onset of flow instability is an accident condition. Please refer to Chapter 5, Section 5.8.4.1.1.
2.27.	2.4.4 Protection against flow Instability.	Power oscillations induced by the formation of vapour (bubbles) in the core is avoided by the prevention of boiling.	Attention needs to be given to other sources of bubbles, such as radiolysis, disassociation of air and the sucking in of air into the primary pipework.
			Response: The solubility of air in water decreases with increasing temperature and decreasing pressure. In the PCS, the decay tank is an area with lower velocity and higher temperature (velocity of the order of a few cm/s and 50ºC temperature). The terminal velocity of non-condensable bubbles in water is of the order of 10 - 30 cm/s. Therefore the tank would act as a degassing device, preventing any bubbles appearing due to dissociation in the PCS from being transported to the PCS piping. While the cold leg of the PCS can have more dissolved air in the water (the water at 40ºC admits 15% more moles of dissolved air than the concentration at 50ºC), non-condensable bubbles are not expected at the core inlet. Inside the core, due to the high velocity of the coolant (10m/s), nucleation of small bubbles with subsequent detachment and condensation is considered not credible. Even if voids were to result from the processes mentioned, the result would be a slightly reduced flow and hence slightly higher core exit temperature. This would not result in increased voidage and so cannot lead to the type of instability that leads to a flow excursion as occurs in catastrophic vapour generation. Sucking of air is not a credible event since the pressure inside the primary pipework is always above atmospheric pressure.
2.28.	2.4.5 Fuel Design Limits	Under all operational states and design basis accidents the fuel plate cladding retains its integrity and dimensional tolerances.	Thermal hydraulic phenomena should be regarded as surrogates for fuel cladding integrity

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
			Response: The limiting condition of fuel blistering and the thermal-hydraulic design bases that avoid reaching this condition are discussed in Chapter 5. Also see the response to Question 2.22.
2.29.	2.4.5.1 Fuel Design—Thermal effects	Design requirements will be set to ensure geometrical and dimensional stability of fuel with respect to thermal effects	It is not clear whether the thermal effects associated with DBA transients (Ch.16) maintain dimensional and geometrical stability.
			Response: The thermal effects associated with normal operational transients and DBA transients are discussed in Chapters 5 and 16.
2.30.	2.4.5.3 Fuel Design—Hydraulic and Mechanical Effects	Hydraulic and mechanical effects will be limited be having a flow velocity margin to the critical flow velocity.	Information is needed on how the critical velocity is established. It would be that it has been experimentally demonstrated over the range of operating temperatures and pressures.
			Response: The critical velocity has been established on the basis of the analysis discussed in Chapter 5 and operational experience from ETRR-2. This analysis relies on internationally accepted experimental data and correlations. The ability of the fuel to operate outside the hydraulic instability region will be demonstrated experimentally in the range of operation and accident conditions.
2.31.	2.4.5.4 Fuel Design—Radiation effects	The operational arrangements to ensure that blistering, swelling, significant changes in physical and mechanical properties do not occur are described.	A limiting factor with respect to radiation damage may well be the Reflector Tank. The inner wall will be exposed to high neutron flux for the 40 to 50 year life of the plant.
			Response: Agree and this will be taken into account in the detail engineering of the reflector vessel as discussed in Chapter 5.
2.32.	2.4.5.5 Fuel Design-Chemical effects	Corrosion of the fuel is avoided by the selection of materials, adequate cooling, and adequate water chemistry	The fuel will have low residence time, but this is not the case for other core components such control rods, guides for the control rods and other fixed in core components, such as the reflector tank. Corrosion could lead to degraded FSS performance.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
			Response: Agree and this will be taken into account in the detail engineering of the core components as discussed in Chapter 5 and the Reactor Coolant Purification System as discussed in Chapter 6.
2.33.	2.4.6. Design Criteria for the Reactor and service Pools	The reactor and service pool liners will be designed to AS1210: Pressure Vessels, but with supplementary inspection requirements.	Design to an Australian standard needs justification. What differences in wall thickness, weld requirements, material selection if designed to ASME III?
			Response: As indicated in Section 4.5, the pool liners will be designed to ASME III ND with supplementary inspection requirements, not AS1210. This section is incorrect and will be revised to reflect the correct position.
2.34.	2.4.6. Design Criteria for the Reactor and service Pools	The pool will be designed for the combination of static, dynamic, thermal and hydraulic loads under design basis accidents	The load combinations should be checked against ASME III, in particular in relation to seismic loads. There should also be a check against dropped loads and sloshing during an earthquake
			Response: See response to comment 2.33
2.35.	2.4.6. Design Criteria for the Reactor and service Pools	The pool liners will be designed to facilitate inspection sand to detect degradation of physical and mechanical properties during the lifetime of the reactor.	It is not clear how the pool can be inspected if the spent fuel remains in the pools. There would appear to be a need for an interim store elsewhere.
			Response: Spent fuel is not stored in the reactor pool if drain down of the reactor pool water level is required. If inspection is required, all fuel will be moved to the service pool and the transfer canal gate closed to allow the drain down of the reactor pool. See Chapters 4 and 10.
2.36.	2.4.6. Design Criteria for the Reactor and service Pools	Penetrations of the pool below the level of the reactor core will be provided with redundant seals.	The performance of these penetrations in large earthquake was a key issue in the Siting Approval for the RRR. Calculations are required to show seismic margins well beyond the SSE.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
			Response: Preliminary calculations with respect to the performance of the penetrations below the level of the core (principally the beam tube penetrations) are discussed in Section 4.5 and indicate that these penetrations are capable of surviving an earthquake well in excess of the SL-2 earthquake.
2.37.	2.4.7 Design Criteria for Structures within the Reactor and Service Pools	The material used within the pool is designed to minimise corrosion and thus possible activation of the water. All equipment is designed for the load combinations associated with normal operation and design basis accidents.	The structures within the reactor and service pools will be designed to withstand an SL-2 seismic event, and adequate margins for earthquakes with lower frequency than the SL-2 event. It is important to get a complete list of all such structures and components, and to identify for each the extent of seismic design.
			Response: Agreed. Please refer to Chapter 1, Figure 1.2/6 which shows the reactor pool in section and the main internal components. Table 2.5/2 also lists all components with a seismic categorisation in the pools.
2.38.	2.4.8.1 Cooling System General Requirements	In general Safety Class 1 vessels will be designed to AS1210, but with supplementary inspections. Safety Category 1 piping is designed to AS4041 (Class 1) or equivalent.	Need to look carefully at inspection requirements of Class 1 piping designed to AS4041. Past experience shows that 100% radiographic inspection of butt welds is not required. ARPANSA will require inspections beyond AS4041 to ensure all butt welds are radiographed. It is not clear how and where the ASME pressure vessel and piping codes are used. They are referenced in Chapter 2, so the components and structures involved should be made clear.
			Response: Agree and clarification will be provided as part of the detail engineering, particularly with respect to the identification as to the specific components and structure to which ASME standards (principally ASME III, ASME VIII and ASME B31.1) are being applied.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
2.39.	2.4.8.2 Primary Cooling System (PCS)	In normal operation the PCS provides sufficient forced cooling to maintain the fuel cladding integrity. Under design basis accident conditions it will provide a reliable means of residual heat removal. The reactor pool is considered as the ultimate heat sink for decay heat removal.	ARPANSA notes the design philosophy that makes the pool water the ultimate heat sink. Mass and heat transfer from the pool into the containment atmosphere could be argued to make this the ultimate heat sink. The existing decay heat removal diversity by means of forced convection to the cooling tower pond should be designed to Safety Category 1. This would also necessitate designing the Secondary Cooling Water system to safety category 1.
			Response: The CERS is provided to ensure that the containment design limits are not exceeded, not to act as an ultimate heat sink. As stated in the response to Question 2.14, redundancy is provided in the means of decay heat removal where practicable. It is difficult to envisage how to provide diversity for natural circulation since it is a passive system that relies on basic physical principles. Even in the event of failure of all four flap valves, adequate cooling would be achieved by boiling within the reactor core.
2.40.	2.4.8.3 Reactor and Service Pools Cooling System (RSPCS)	The RSPCS is designed to cool the irradiation targets during all operational and design basis accidents. On loss of forced convection there is a flow reversal during the transition from forced to natural convection	Chapter16 will be checked to ensure the irradiation rig cooling channels remain below the ONB for normal operation and anticipated occurrences. The behaviour during design basis events should also be investigated.
			Response: Comment noted. This is addressed in Chapter 16.
2.41.	2.4.8.4 Reflector Cooling and Purification system (RC&PS)	The RC&PS is designed to remove heat generated within the heavy water inside the reflector vessel under normal operating conditions. It also has purification systems to prevent the explosive build up of Deuterium/Oxygen mixtures.	It is not clear what is the heat removal capacity for the RC&PS. Is it the 1.2 MW given in Chapter 1 of the PSAR. It is not clear what cooling is required in anticipated transients and design basis accidents. Also what effect on this cooling system has the operation of the SSS dump. The flow path would appear to be interrupted.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
			Response: The heat removal capacity of the RC&PS is discussed in Chapter 6, as is the impact of the operation of the SSS on the RC&PS.
2.42.	2.4.8.5 Secondary Cooling System (SCS)	The SCS will remove heat from the PCS, the RSPCS and the RC&PS and discharge it to atmosphere via the Cooling Towers under all normal operating conditions.	Consideration should be given to the SCS as a diverse means of decay heat removal for design basis accidents.
			Response: See response to Question 2.39.
2.43.	2.4.8.6 Emergency Make up Water System (EMWS)	The EMWS is designed to maintain the core covered in water in the event of a beyond design basis (BDBA) Loss of Coolant Accident (LOCA) in order to prevent damage to the core. It is not considered an ESP.	Th EWMS is designed as Safety Category 2 system (Table 2.5/2-5). It is evident it has no redundancy, but since basically a passive system, used for BDBA situations the lack of redundancy is acceptable.
			Response: Agreed.
2.44.	2.4.9 Instrumentation and Control System Criteria (I&C)	The criteria and requirements applicable to I&C are specified. The I&C ranges from protection system (FRPS and SRPS) to seismic monitoring instrumentation.	It is not evident that other key I&C instrumentation such as that for fire detection are part of this I&C.
			Response: The I&C systems are discussed in Chapter 8 including the fire detection systems. Additional details about the design for fire protection are contained in Chapters 4 and 10.
2.45.	2.4.9.1 General Instrumentation and Control System Requirements	Process instrumentation and control systems will provide sufficient for monitoring the operation of the reactor and its facilities. The Reactor Control and Monitoring System (RCMS) is a key system since it controls the reactor without the need to invoke the reactor protection systems (FRPS and SRPS).	Need to determine the extent to which the general instrumentation is Safety Category 1, in particular the safety categorisation of the RCMS.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
			Response: As stated in Table 2.5/2, the RCMS is a Safety Category 2 system and the instrumentation associated with it is qualified appropriately on the basis that the RCMS is not a principal means of ensuring nuclear safety.
2.46.	2.4.9.2 .1 Reactor Protection Systems General Requirements	The FRPS and SRPS are diverse and independent systems and no single component failure in one can fail the other. The FRPS and SRPS are also fail safe. The FRPS and SRPS have important interlocks and trips which cannot inadvertently be changed.	If the FRPS and SRPS are independent why should any failures, including multiple failures affect the other? The independence of these systems is a key matter, and cross connections should no be used to prevent the early operation of the SRPS.
			Response: Do not understand comment. The FRPS and SRPS are independent such that failures within one cannot prevent the correct operation the other, as discussed in Chapter 8.
2.47.	2.4.9.2.2 First Reactor Protection System (FRPS)	The FRPS will provide signals for the FSS and other safety systems. It includes the cold Neutron Source Protection System (CNSPS) and Containment Isolation System. It is a computer based digital system and will be subject to hardware and software verification and validation.	ARPANSA will need to see how the verification and validation to IEEE (and IAEA guidance) is achieved. It needs to be established whether the FRPS also shuts down the reactor building ventilation as part of the CIS function.
			Response: The verification and validation of the FRPS is discussed in Chapter 8 and will be fully covered in the appropriate verification plans. The action of the FRPS on the CIS is addressed in Chapter 7.
2.48.	2.4.9.2.3 Second Reactor Shutdown System (SRPS)	Provides signals to the SSS only.	The PSA has indicated that the SRPS (hardware based) is ten times more reliable than the software based FRPS. It is thus surprising that it is not used for the range of safety functions allocated to the FRPS. See also interface with FSS to detect failure of rods (or is it rod) to insert.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
			Response: The FRPS and SRPS are discussed in detail in Chapter 8, including the relative reliability requirements. The use of a software based system offers advantages in terms of configuration of the system. See also response to Question 1.38.
2.49.	2.4.9.3 Reactor Shutdown Systems Instrumentation	FSS and SSS instrumentation will comply with IEEE Class 1 equipment and be to the same standard as the FRPS and SRPS.	The statement that FSS and SSS are to the same standards as FRPS and FRPS is confusing since FRPS is digital based and SRPS is hardware based. Will the FSS and SSS instrumentation be hardware based. It is also not clear what instrumentation is classed as FSS and SSS, and what impact the cross connection from FSS instrumentation to SRPS instrumentation has on independence.
			Response: The FSS and SSS instrumentation is described fully in Chapter 8.
2.50.	2.4.9.4 Reactor Nucleonic Instrumentation	Independent and diverse nucleonic instrumentation will be provided for the FPRS and SPRS.	ARPANSA will assess the level of diversity in nucleonics instrumentation.
			Response: Noted. Nucleonics instrumentation is discussed in Chapter 8.
2.51.	2.4.9.5 Post Accident Monitoring System (PAMS)	The PAMS is an ESP (or ESF) designed to IEEE standards. It will incorporate redundant trains and be supplied from the UPS.	See Chapter 10 of the PSAR for details of the parameters that are on the PAMS, and equipment capability to withstand the design and beyond design bases environmental conditions within the containment building.
			Response: The parameters monitored by the PAM Systems are identified in Chapter 8.
2.52.	2.4.9.6 Reactor Control and Monitoring System (RCMS).	The RCMS is not classed as an ESP, but it will be on the UPS. It will be a computer based system and subject to hardware and software validation.	A number of important interlocks are on this system so the reliance on software is important. It will be necessary to ensure interlocks, bypasses and control inputs are fail safe on software failure.

Checked / agreed:

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
			Response: Agree although as stated, the RCMS is not the principal means of ensuring safety but rather provides a significant additional contribution, hence its allocation to Safety Category 2. Fail safe characteristics are not claimed for the RCMS, since it is a Safety Category 2 system. Any malfunction or abnormal condition produced by the RCMS that could impact on reactor safety will be handled by the RPS.
2.53.	2.4.9.7 Radiation Protection System	Details are given in PSAR Ch.8 and Ch.12.	It is important to ensure that the radiation monitoring equipment system has a similar level of quality to the RCMS. It should also be on the UPS. It is not clear why it is not part of the RCMS.
			Response: The radiation monitoring equipment is discussed in Chapters 8 and 12, including the standards to which it will be designed and the provision of appropriate power supplies.
2.54.	2.4.9.7.2 Radiation Protection System—Air Waste Monitoring	Continuous on line sampling of all radioactive stack emissions is provided. Signals go the FRPS to effect containment isolation.	It is not clear if there is a reactor trip on a high stack emission signal and whether it trips the containment ventilation system. It is important that the sampling and monitoring arrangements satisfy an appropriate standard such as ANS/HPS N13-1—1999.
			Response: The reactor trip signals and logic are discussed in Chapter 8 whilst the sampling and monitoring arrangements are discussed in Chapter 7. Stack monitoring is described in Chapter 12, Section 12.4
2.55.	2.4.10 Engineered Safety Features Design Criteria (ESFs)	Ten systems (a) to (j) are specified as ESFs. All ESFs are classed as Safety Category 1 systems.	Reactive review comments have raised the question of other systems as ESFs, such as the EWMS, and the decay heat removal system to the cooling tower pond.
			Response: We note that your comment 2.43 indicates that the allocation of the EMWS as a Safety Category 2 system is acceptable. With respect to the decay heat removal, see our response to comment 2.39.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
2.56.	2.4.10.2 ESFs---Reliability Requirements	The reliability of the ESFs will be shown to be better than 10^{-3} failures per demand. Eight methods ranging from proven engineering to regular testing and inspection are used to support the reliability claim.	The demonstration of the reliability target should be supported for each system identified. The values used in the PSA may be generic and not specific to the RRR design.
			Response: There is a contractual requirement to demonstrate that the reliability of systems meets their identified requirements. The PSA will be revised to reflect the detail engineering and the demonstration of such reliability.
2.57.	2.4.10.3 Disabling and Bypassing of ESFs	Disabling and bypassing of an ESF will be justified and five reasons are given.	A minimum plant configuration is needed in the Operating Limits and Conditions to set out the conditions, outage times and modes of operation under which ESFs can be bypassed. A particular case is the Standby Power Supply s(SPS) since it may be involved in a number of the frontline ESFs
			Response: The OLCs will be developed during the detail engineering and will cover minimum plant configurations. Note that the SPS is only required in support of the CERS, not to achieve and maintain a safe shutdown state.
2.58.	2.4.10.3 Disabling and Bypassing of ESFs	Annunciation of ESFs being bypassed or disabled will be provided.	The means of annunciation needs to be specified and whether the annunciation is part of a Safety Class1 system.
			Response: The ESFs and their associated instrumentation are discussed in Chapters 7 and 8.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
2.59.	2.4.10.5 Independence of Process and Safety Systems	Process and safety systems are kept physically and functionally independent as far as practicable.	An example of where process and control systems are not independent is in the FSS. The central Control Rod is involved as the main power level control system, and in safe shutdown of the reactor (success criterion is 4oo5 drop). This is also true to a lesser extent for the four outer Control Rods since they have to be driven out at power raise and each is driven separately. The means of preventing the bank of 4 Control Rods being driven out needs to be reviewed closely.
			Response: The FSS and the independence between the control and protection functions is discussed in Chapter 5, Section 5.5.2.8, Chapter 8 and Chapter 16, Section 16.8.3.4. Note that there is no bank withdrawal of the control rods. Also see response to Question 1.36.
2.60.	2.4.11.1—General Requirements for the Reactor Containment System (RCS)	The design of the RCS is based on a conservative margin against the calculated pressure and temperature loads under design basis conditions. The RCS is an ESF.	It is not clear what building standard is used for the containment building, since it is clearly not a conventional pressure vessel.
			Response: The standards used for the reactor building and the containment are discussed in Chapters 4 and 7.The containment is not a pressure vessel. Since the design pressure load is small compared with other loads (seismic, self weight, live and dead loads), the standards applied correspond to structural design: ACI 318 1999 and ACI 349 1997 (see PSAR Chapter 4, Section 4.3.2.2). The containment pressure load is limited by the Containment Pressure Relief and Filtered Vent System and the Containment Vacuum Relief System (see PSAR Chapter 7, Section 7.8.3.6).
2.61.	2.4.11.2 Containment boundary	The containment boundary encloses the reactor systems containing radioactive material. The number of penetrations is kept to a minimum and will be provided with isolation valves, and be amenable to testing.	The containment design is novel in view of its dependence on energy removal, automatic vent, manual vent and isolation systems. There is a close interface with the ventilation system and this needs to be evaluated.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
			Response: The reactor containment system and the operation of the associated systems are discussed in Chapter 7.
2.62.	2.4.11.3 Containment Isolation System (CIS)	The CIS is part of the RCS and is connected to the FRPS. All piping and ducting penetrations that may potentially be open to the internal atmosphere have a closure inside and outside the containment.	There is a need to examine how the beam line penetrations are designed with respect to closures. The beam lines may be exposed to the containment atmosphere during maintenance and experimental set up.
			Response: The neutron beam penetrations are described in Chapters 4 and 11 and their interface with the containment is discussed in Chapter 7. Note that the beam lines are never exposed to the containment atmosphere except during replacement (if necessary) when all fuel would be removed and the reactor pool and reflector would be completely drained.
2.63.	2.4.11.4 Containment Energy Removal System (CERS)	The CERs is part of the RCS and capable of reducing the containment pressure and temperature in all design basis accidents.	The mass and energy transfer from the pool under containment sealed conditions, and under natural circulation conditions make the CERs effectively the ultimate heat sink.
			Response: See response to comment 2.39.
2.64.	2.4.11.5 Containment Pressure Relief and Filtered Vent system	This pressure relief system is provided for beyond design basis accidents which could result in over pressurisation of the containment The release will be filtered before going to the stack.	The pressure relief is automatic, but there is a need to confirm if there will also be a remote operation function from the MCR and ECR. Also information is required on the sizing of the filters with respect to fission product retention capacity.
			Response: The reactor containment system and the operation of the associated systems are discussed in Chapter 7. Remote manual operation is provided.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
2.65.	2.4.11.6 Containment Vacuum Relief System	This vacuum relief system is to prevent potential structural damage in beyond design basis accidents which result in under pressure of the containment.	The mechanism for under-pressurisation needs to be explained, as does the reference to beyond design basis.
			Response: The reactor containment system and the operation of the associated systems are discussed in Chapter 7, including the reasoning for the provision of the Containment Vacuum Relief System.
2.66.	2.4.12 Active area HVAC	The HVAC systems will be designed so as not to compromise the independence of redundant safety components and systems.	Isolation of the containment system HVAC on detection of fission products in the stack emissions was raised in the RRR Siting SER as a key factor in the Siting Reference accident. What happens to the fans when the CIS is initiated?
			Response: The reactor containment system and the operation of the associated systems (including the various ventilation systems) are discussed in Chapter 7.
2.67.	2.4.14 Sharing of Structures and systems important to safety	There will be some dependence on existing site services, but the RRR will have its own backup services for fire, electrical power, cooling towers, liquid waste, compressed air, telecommunications, and diesel fuel.	Please clarify if the diesel fuel tanks are housed within the reactor building, and whether any of the existing HIFAR liquid waste systems are shared with the RRR.
			Response: The Standby Power System (including the diesels) is discussed in Chapter 9 whilst the liquid waste management systems are discussed in Chapter 12. Note that the arrangement of the diesels is presently being reviewed due to an increase in their size but that the fuel tanks will not be located within the main building complex.
2.68.	2.4.15 Human Factors and Ergonomics	The design of the control rooms is to help operators assimilate information. Ergonomic consideration also given to the identification , maintenance of components such as valves and gauges.	Identification and tagging of components in the pool needs to be given close attention to ensure they can be confidently identified from the pool top area.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
			Response: Agreed. These aspects are discussed in Chapters 4, 5, 10, 11 and 13.
2.69.	2.4.16 Design Analysis Techniques	Design analysis will be carried out using validated techniques, models prototypes and computer codes. A range of methods of design validation are identified.	The detail design review of Chapters 4, 5, 6, 7, 8, etc should identify the validation undertaken. The support documents for the PSAR will be needed. The review against the RAPS and Design Guideline should also reveal the design analysis techniques used.
			Response: Comment noted.
2.70.	2.14.17.2.1 Cold Neutron Source	The moderator vessel of the CNS will be designed , manufactured and tested in accordance with ASME III. The instrumentation will be UPS.	The treatment of Cold Neutron Source accidents in Chapter 16 is important, in particular accident sequences that can affect the Reflector Tank and the reactor.
			Response: Noted. The CNS is discussed in Chapter 11 and its impact on reactor safety is also discussed in Chapter 16. The CNS will be the subject of a separate Safety Analysis Report.
2.71.	2.4.17.3 Irradiation Facilities	Facilities will be designed to avoid boiling on the surface of the irradiated facilities. All facilities will be secured to the Reflector Tank.	The movement of irradiation rigs at power is an important issue with respect to reactivity accidents. The amount of reactivity controlled by individual moveable rigs, and the total rig worth should be specified in the OLCs. (See guidance on rig worth from ARN).
			Response: Agree, the limits on the insertion, withdrawal and movement of rigs will be identified in the OLCs at the draft FSAR stage on the basis of the safety analysis and the detail engineering. See also the discussion in Chapter 11 and Chapter 17, Table 17.4/1.
2.72.	2.4.17.3.2 Interfaces between Irradiation, Transfer, and Processing Facilities	The provision of shielding to protect the operators is the key requirement, as is cooling of samples.	There will be a dependence on administrative control and accurate recognition of irradiation cans to ensure no errors are made. The gamma alarm system to prevent short cooled targets getting into the cells will need careful evaluation.
			Response: The utilisation of the reactor with respect to the irradiation rigs is discussed in Chapter 11.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
2.73.	2.4.18. Protection against Dependent Failures	There is a claim for diversity, independence and in-line testing capability to protect against dependent failure.	This is an important area which has been examined in the PSA. ARPANSA will review against the RAPS and DG-5.
			Response: Comment noted
2.74.	2.4.19 Capability of Surveillance and Maintenance of Safety Related Equipment.	The reactor facility will be designed to minimise the maintenance required on structures, systems and components (SSC). The SCC important to safety will incorporate features to enable surveillance and maintenance.	Design to minimise maintenance needs further explanation, noting that the requirement should be that there is adequate maintenance and that facilities and time are in place to achieve this within the proposed operating cycle of 2 days in 30 days for maintenance at shutdown.
			Response: A plant that requires little maintenance is considered to be well and simply designed. Minimisation and simplification of maintenance is considered a safety requirement. While maintenance will be minimised and simplified, adequate maintenance will still be performed.
2.75.	2.4.20 Electrical Power Supply Design Criteria	Three systems identified, normal power from off-site, standby power from diesels and uninterruptible power from UPS and batteries. The Standby Power Supply will comply with IEEE standards for Class1 equipment.	Please explain (in Chapter 9?) if there is any sharing of systems with HIFAR (eg Sub-stations). Also the interconnections between the off-site, standby and UPS need to be explained in terms of how independence and redundancy is achieved in supplies to redundant ESPs.
			Response: The Normal Power Supply and the Standby Power Supply are discussed in Chapter 9, including the interconnections between the systems and the connections to the off-site supplies. There is no sharing of electrical systems with HIFAR downstream of the main substation. Sharing of other systems is discussed in Chapter 2, Section 2.4.14.
2.76.	2.4.21.2 Auxiliary System Criteria	Interfaces between auxiliary systems (not required to shutdown or maintain fuel integrity) and ESPs.	More information is needed on these auxiliary systems and criteria, in particular a matrix diagram which shows the level of interaction and dependence. Under the above definition the containment systems could be classed as auxiliary.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
			Response: PSAR definition will be revised to clarify that the containment systems are not auxiliary systems since they contribute to containing the release of radioactive materials. The auxiliary systems are discussed in Chapter 10.
2.77.	2.4.22 Radiation Protection Measures in the Design.	Radiation shielding provided to meet ALARA, and areas of continuous occupancy will have dose rates greater than 5 microsievert per hour.	If 2000 hours per year occupancy then 5 microsievert per hour gives 10 mSv per year. This is not consistent with an ALARA target of 2 mSv per year. Please explain. See also question under 12.4.22 Radiation Protection Measures in the Design
			Response: The value 5 microsievert per hour quoted in the PSAR is an error and should be 0.5 microsievert per hour.
2.78.	2.4.22.2 Prevention of Inadvertent Criticality	High sensitivity nucleonic instrumentation will be provided to allow measurement of core neutron population.	It is not clear if there is any criticality detection instrumentation for areas outside the core, for example, for fresh and spent fuel storage.
			Response: There is no criticality detection, only detectors that would provide information after a criticality incident had occurred. Such instrumentation is discussed in Chapters 8 and 10 for areas outside the core.
2.79.	2.4.22.3 Waste Management	Waste management at the RRR will be consistent with ANSTO Waste Management arrangements.	The arrangement for temporary storage of solid and liquid waste within the reactor building needs evaluation.
			Response: Comment noted.
2.80.	2.5 Classification of Structures , Components and Systems	The Safety Classification identifies three safety categories for SSC, with SC1 forming a prime means of ensuring reactor safety. Table 2.5/1 and Table 2.5/2 lists all the SC1, SC2 and SC3 SSC.	As part of the detailed review ARPANSA will need to satisfy itself with the categorisation of all the SSC identified in these tables. The review to date has identified SSC where further consideration is needed, for example decay heat removal to pond and the EWMS.
			Response: Noted although refer to our response to comment 2.55.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
2.81.	2.5.1.1 Design Requirement for safety category 1 SSC	SC1 will be capable of performing their safety function under all normal operating conditions and anticipated operational occurrences. They also must fulfil their safety function under all design basis fault sequences and associated environmental conditions.	Some thought needs to be given to the performance of Structures, Components and Systems beyond the design basis to ensure no cliff edge effect. This seems to be recognised for the seismic case for the reactor building and key components. The PSA needs to be checked for claimed reliability and unavailability of Safety Category 1 Structures, Components and Systems for beyond design basis events.
			Response: Comment noted
2.82.	2.5.1.1 Design Requirement for safety category 1 SSC	The capability of SC1 SSC to fulfil the safety role will be demonstrated.	Some Safety Category 1 items are of novel design, and can be of quite complex design, construction and operation. Some of these SC1 items are the flap valves, the Reflector Tank, the Control Rods and their drives, the FRPS, and the containment energy removal system.
			Response: See response to Question 2.8.
2.83.	2.5.1.2 Design Requirement for safety category 2 SSC	SC2 will be capable of performing their safety function under all normal operating conditions and anticipated operational occurrences. They also must fulfil their safety function under all design basis fault sequences and associated environmental conditions.	Consideration needs to be given to the difference in standards, level of review between SC1 and SC2 Structures, Components and Systems. For example there is no reference to proven service, dependent failure analysis, failure modes and effect analysis, or single failure criterion with respect to SC2. Some guidance should be taken from the PSA and a dependency matrix in establishing the standards and level of review for SC2 Structures, Components and Systems.
			Response: Clarification will be provided to explain the difference in standards etc between Safety Category 1 and Category 2 SSCs and how these standards are determined for Category 2 SSCs.
2.84.	2.5.1.3 Design Requirement for safety category 3 SSC	Normal industrial standards will be applied, the minimum being the Australian Standard.	As part of the detailed review ARPANSA will need to satisfy itself with the categorisation of all the SC3 Structures, Components and Systems identified in the Tables2.5/1 and 2.5/2.
			Response: Comment noted.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
2.85.	2.5.2 Seismic Classification Methodology	Three levels of seismic categorisation are identified with Seismic Class1 SSC designed to SL-2, Seismic Class 2 SSC designed to the OBE, and Seismic Class3 designed to Australian standards for buildings. Table 2.5/2 identifies the SSC seismic classification, as well as its safety classification and quality classification.	As part of the detailed review ARPANSA will need to satisfy itself with the seismic classification of all the Structures, Components and Systems identified in these table. The seismic failure of SC 2 and SC 3 could affect SC1 systems
			Response: Comment noted.
2.86.	2.5.3 Quality classification Methodology	Four factors are used to set the quality requirement. These are Factor(a)—Safety, Factor (b)—Availability, Factor (c) –Complexity, and Factor (d) –Design experience. The Total quality ranking is the sum of a+b+c+d in accordance with numerical ratings for (a), (b), (c), (d).	All Safety Category 1 Structures, Components and Systems are Quality Level A, but some other Structures, Components and Systems can also be ranked as Quality level A depending on their relative numerical ranking. Table2.5/2 lists the quality level ranking which will require detailed ARPANSA review.
			Response: Comment noted.
2.87.	2.6.1 Design Criteria for the Resistance against Seismic Hazard	Two IAEA Safety Series (50 –SG-S1, 50-SG-D15) and USNRC Reg. Guides (1.61, 1.92,and 1.122) are adopted for seismic definitions and qualification. The IAEA TECDOC –348 is used for provisions relating to anchorages, piping supports.	The mixture of the IAEA Safety Series, NRC Guides and TECDOC –348 could lead to inconsistencies in the seismic design provisions, particularly when floor response spectra are used for SC1 and SC2 Structures, Components and Systems.
			Response: Comment noted, although the application of the appropriate guidance to the seismic design of the RRR is discussed in Chapter 4.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
2.88.	2.6.1 Design Criteria for the Resistance against Seismic Hazard	For the SL-1and SL-2 earthquakes used in design a response spectra known as the Elastic Design Response Spectra (EDRS) is used, and corresponds to 5% of critical damping. The EPDS is based on USNRC guidance (RG-1-60).	The EDRS, and not the AGSO time history accelogram and spectra used for the HIFAR seismic upgrade needs to be justified. Also Chapter3 of the PSAR and PSA uses the IGNS (1999) response spectra, which differs from the EDRS.
			Response: Noted. The justification for the choice of PGA and spectra is given in Chapter 4, Section 4.4. Chapter 3 only provided an outline of the previous studies and indicated that the RRR choice bounded previous studies over all appropriate frequency ranges.
2.89.	2.6.1.2 Basic Response Spectra (BRS)	The US NRC RG1-60 has been used for the BRS (scaled to 0.3g PGA for the SL-2 event) and 0.09g PGA for the SL-1 event).	The applicability of RG1-60 to SE Australia, has been questioned by AGSO and the IAEA. In particular there is a gross over estimate of accelerations at long period, and an underestimate at the shorter periods associated with many components.
			Response: A comparison of US NRC RG1-60 spectrum anchored at 0.3g and the IGNS spectrum anchored at 0.3g shows that the US NRC RG1-60 spectrum is conservative over the whole frequency range.
2.90.	2.6.1.3.1 Safe Shutdown Earthquake SL-2	A value of 7% of critical damping will be adopted to derive the EDSR from the BRS. This is base on a ductility allowance of 1.25.	The basis for using 7% ductility is not clear, since Chapter 4 states that the Seismic Class1 Structures, Components and Systems remain elastic well beyond the SL-2 earthquake.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
			Response: The wording in Section 2.6.1.3.1 is incorrect. It will be reworded in the next revision of the PSAR to reflect the position below: "The Reactor Facility Main Building will be designed to behave elastically during the SL-2 Earthquake. Therefore, no reduction factor is used to claim for any ductility supplied by the structure and the design is then based upon a Linear Elastic Design Response Spectrum (LEDRS). However, since the main seismic resisting system is a shear wall controlled reinforced concrete structure, it is reasonable to use a damping factor of ξ=7% of the critical**.** This value is well supported by Regulations: the IAEA TECDOC 348 (1995) allows the use of up to 10% for reinforced concrete structures with shear walls, and the US Regulatory Guide 1.61 specifies 4% for the OBE and 7% for the SSE design levels. The basic spectral ordinates of the LEDRS were derived for 5% damping. Thus, a correction has to be made for the 7% damping spectra. The IAEA TECDOC specifies a 0.88 correction factor. This figure agrees well when compared with the ratio of the Amplification Factors for acceleration control point B (for 9 Hz) taken from Table 1 of the US Regulatory Guide 1.60 (which is 2.27/2.61=0.87). These numbers correspond to the ordinates for 7% and 5% of critical damping respectively for the spectral control point B. Finally, when Calvi's expression is applied to justify the 0.88 factor, the equivalent ductility factor becomes 1.25. However, the design is based upon the LEDRS for 7% damping with no reduction for energy dissipation. Since levels of seismic acceleration always have a certain degree of uncertainty, the design of the structure will ensure that, for some critical regions, there will be some over-strength and reserve ductility for post-elastic behaviour beyond the SL-2 earthquake." This text replaces the first two paragraphs in Section 2.6.1.3.1.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
2.91.	2.6.1.3.2 Operating Basis Earthquake SL-1	A value of 4% of critical damping is used for the OBE, on the basis it remains elastic.	Since the SL-2 also ensures Structures, Components and Systems remain elastic why was a 4% damping value not used.
			Response: The damping values are taken from US NRC RG1-61.
2.92.	2.6.1.5 Ductility	All SSC are designed to remain elastic for the seismic level corresponding to them.	See 2.6.1.3 1 above and the confusion over ductility allowance in the SL-2 event and the resulting 7% damping.
			Response: Comment noted. See response to Question 2.90.
2.93.	2.6.1.6 Seismic Qualification	There is a reference to seismic qualification by analysis, or resting, or experience or indirect methods, or a combination of these.	Novel Safety Class 1 Structures, Components and Systems should be qualified by testing as far as is practicable. This should apply to the flap valves, the control rod drives and the Reflector Tank.
			Response: See response to Question 2.8 re identifying design features as novel, however it is confirmed that seismic qualification will be performed as appropriate on all relevant components. The qualification criterion of structures is dependent on their having passive or active safety functions. Depending on this classification, the qualification can be carried out by analytical verification, finite element analysis or testing.
2.94.	2.6.1.6.1 Seismic design Criteria—Civil Structures	A whole range of criteria apply the civil structures (a) to (j). These cover the support strata, foundations, symmetrical approach to shear walls, adjacent buildings and connections, and avoidance of brittle failure by shear or compression. US and Australian concrete codes are referenced.	The approach to load combinations for seismic, dead and live loads needs to be set out. In addition the superposition of two horizontal seismic components (N-S and E-W) as well as the vertical component needs to be set out.
			Response: The load combinations are described in Chapter 4 for civil structures. Seismic components will be combined according to the method indicated in AS 1170, for irregular buildings and civil structures.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
2.95.	2.6.1.7 Equipment and Components	The design loads from normal operation for vessels, structures etc will be combined with earthquake loads from the corresponding Floor Response Spectra.	Where an ASME code is used for vessels and pipes the load combinations should be based on the ASME requirements.
			Response: Agreed.
2.96.	2.6.1.7.1.4 Electrical and Electronic Components	The seismic qualification will be in accordance with IEEE requirements.	Areas of exception with respect to seismic qualification against IEEE will need justified.
			Response: Agreed. Such justification will be prepared where appropriate during detail engineering and will be provided in the FSAR.
2.97.	2.6.1.7.2.1.5 Reactor Hall Crane –Special Requirements	Designed as Seismic Class2, but also a special requirement to withstand the SL-2 event without collapse. Non elastic behaviour is permitted.	The behaviour of the crane and its collapse beyond the SL-2 event should be examined. It is noteworthy that the reactor building grillage is designed against a 10^{-7} per year event (aircraft crash), but the crane is not designed to survive earthquakes beyond 10^{-4} per year. Collapse of the crane could damage both the reactor pool and building.
			Response: The behaviour of the crane is discussed in Chapter 4. Note that the requirement of the reactor to withstand the impact of a light aircraft was a design basis requirement arising from a voluntary commitment by ANSTO to respond to public concern. It is unrelated to the frequency of the initiating event and as such the applicable criteria are different for the grillage and the crane.
2.98.	2.6.1.7.1.6 Anchors	The practical guidance given in TECDOC 348 will be adopted for equipment anchorages.	There is a draft IAEA document (1999—Design of nuclear facilities other than Nuclear Power Plants in relation to external events, with special emphasis on earthquakes). This is a revision of TECDOC 348 and should be used since it takes into account PC computer codes now readily available.
			Response: Agreed. The old TecDoc 348 was adopted to avoid referring to a draft document.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's response
2.99.	2.6.2 Design Criteria for Resistance against Aircraft impact	The design will follow IAEA SS 50-SG-D5 and assume an impact by a small executive jet (Cessna Model 500 Citation).	There needs to be some justification for the selection of this particular aircraft.
			Response: The design basis aircraft crash is discussed in Chapter 4. Note that the specific model identified is merely representative of a jet-powered light aircraft.
2.100.	2.6.3 Design Criteria for the Resistance against Wind and Tornado	The design will take into account the effect of high winds on the building.	No international or Australian Standards or codes are referenced. Please explain the basis of the wind loading design criteria.
			Response: The codes and standards used for the civil design are referenced in Chapter 4.
2.101.	2.6.4 Design Criteria for the Resistance against Floods	The design will take into account provisions for the effect of flooding near the facility.	No international or Australian Standards or codes are referenced. Please explain the basis of the flooding design criteria.
			Response: See response to Question 2.100.