Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
16.1.	16.3.3.2 Transient Analysis	'An uncertainty of 20% is considered in the reactivity worth of all insertions postulated in the reactivity transients'	Is this the uncertainty of insertion only or combined uncertainty of modelling assumptions?
			Response: This covers the uncertainties of design data, geometry and nuclear modelling of the system involved in the insertion. In addition, a 20% reduction is applied to the worth of the FSS.
16.2.	16.3.2.2.1 General	'For the safety analysis calculations carried out using PARET, an uncertainty of 15% was applied to the values of the reactivity feedback coefficients'	1) Why 15%? 2) Why were PARET calculations carried out with reactivity feedback while RETRAN02 calculations assumed no reactivity feedback? 3) Does the above mean that PARET calculations in the PSAR are less conservative that those carried out using RETRAN02?
			Response: 1. This uncertainty comes from comparison with IAEA benchmark calculations. 2. The taking into account of reactivity feedback is more important in the reactivity calculations than thermal hydraulic calculations. Reactivity transients lead to overpower and, as a consequence, the increase in temperature of the clad, fuel and coolant is higher than for flow transients and LOCAs. In all the transients modelled with RETRAN the reactor is shutdown due to process variables (low flow, high temperature, low or high pressure drop). They do not lead to overpower. 3. No. As explained in item 2 above, the different codes are used to look at different scenarios that exhibit strong differences in their phenomenology.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
16.3.	16.3.2.2.1 General	'Failure of the FSS implies that no end-of-stroke signal has been received by the Reactor Protection Systems from two or more control plates within one second following the actuation of the FSS'	1) Is it possible to have a situation when two CR failed to fall, but, a signal of a successful fall will be sent for one of the failed CR? 2) ARPANSA considers that failure of a single CR to fall should activate the SSS. 3) What is the minimum insertion of CRs required to shut the reactor down with adequate margin for various stages of operating cycle and CR burnup? 4) Has the following been considered: whether the logic of the FSS to activate the SSS in case of no end-of-stroke signal has been received by the Reactor Protection Systems from one or more control plates within one second following the actuation of the FSS?
			Response: 1. No credible situation is envisaged. Each CR has 3 limit switches that would have to be actuated by some mechanism and physically there is nothing else that could do this other than the CR reaching bottom. The case of a single switch failing is covered by the use of 2oo3 logic. 2. The safety case is predicated on the basis that 4 out of 5 control plates are demonstrably capable of shutting down the reactor in all situations. The reactivity worth and shutdown margin of the FSS with single failure is consistent with the application of the single failure criterion and international best practice. 3. The safety case has considered only the case of complete insertion of 4 CRs within the design time period. The FSS has been modelled with 20% depletion and single failure, worst possible case. In considering the operation of the SSS, no credit is taken for the insertion of any CRs.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
			4. We are not sure if we understand your point. The FRPS does not actuate the SSS on failure of the FSS. The signal from the FRPS to the SRPS starts a timer in the SRPS on actuation of the FS. SSS actuation is based entirely on the CR limit switches which are part of the SRPS. If the SRPS does not receive the limit switch signals indicating that at least 4 CRs have dropped within 1s, it proceeds to activate the SSS.
16.4.	16.7.5.3.2 Primary Cooling System	'Within the error of the calculations, nucleate boiling is expected to occur for some 5 to 10 s' after activation of the shutdown system.	1) What is the error of the calculations for this event? 2) Where it was estimated in the PSAR? 3) Is it possible to conduct similar calculations by another validated code to establish a benchmark for the case?
			Response: 1. The ONB correlation corresponds to the Net Vapour Generation Point (NVG point). However, before this point is reached, voids that grow on the heated wall actually travel in a narrow bubble layer close to the wall. Given that the temperature of the wall was very close to the ONB temperature (T_w=140C and T_{ONB}=143C) we conservatively assumed that the ONB point had been reached. The ONB is not a critical phenomenon. Relevant documentation will be supplied to ARPANSA in the matter of uncertainty analysis of the calculations. ~~2.~~ The error in the calculations has not been estimated in the PSAR. The conservative assumptions included in the modelling have been stated (20MW operation power, Power Peaking Factor =3, steady state correlations in the determination of the critical phenomena). As mentioned

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
			above, the "overestimation" of the presence of ONB is due to the presence of voids even before ONB is reached and the small difference between the wall temperature and the ONB temperature. 3. The use of a validated code employing verified, conservative, data is considered adequate as part of the production of the safety analysis. The use of a second code is not considered necessary as part of any benchmarking. That said, ANSTO will be carrying out its own calculations using RELAP as part of its adoption of the safety analyses. During the detail engineering phase, INVAP will also repeat the simulation for the transients in the SAR with RELAP.
16.5.	16.8.2.2 Inadvertent Fast Insertion of Irradiation Fissile Material	Handling of the U-Mo rigs.	Is it possible to accidentally insert two U-Mo rigs simultaneously? ARPANSA expects that a fast manual insertion of any rig should be considered as a step rather than ramp as it is a more conservative model of the dropping event.
			Response: Procedures will be put in place to require that U-Mo rigs be handled one at a time. In addition, U-Mo rigs will be handled with a custom made tool making the simultaneous insertion of two rigs impossible.
16.6.	16.8.3.1 Start-up accident	'The withdrawal of a CR inserts reactivity at a rate of 26.7 $pcms^{-1}$. The extraction of CR at a higher speed is prevented by the design of the CRD control system'. 'Inadvertent continuous withdrawal of a CR at start up is considered to be within the design basis and will be considered further'.	Please explain whether the design of the CRD control system meets the single failure criterion in preventing a higher speed of CR withdrawal? Please identify any proposed or previous testing of the CRD control system.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
			Response: For each CR, there is an independent device based on hardwired technology, placed between the CR driving system and the control system (i.e., the system that generates the signals to move the CRDs). This device inhibits the movement of a plate at a speed higher than the nominal value. An FMEA of the filter will be performed to demonstrate its fail-safe characteristics and compliance with the single failure criterion. Specific tests will be performed on the filter prior to integration of the system and in addition, the filter will be integrated to the CRD prototype for testing.
16.7.	16.8.3.2 Inadvertent Control Rod Withdrawal	'The movement of the CRDs is sequential, not simultaneous. Simultaneous withdrawal of all rods is not an RCMS action and is prevented by a hardwired watchdog'.	Please explain whether this is a proven technology and provide relevant testing information? Can a single failure defeat the sequential withdrawal restriction?
			Response: The system is fully hardwired. Each CRD has its own watchdog. Each filter receives input on the status of the other CRs. Failure of the watch dogs would send a signal – indicating its activation – to the remaining filters and inhibit the movement of the remaining CRs. A single failure can not result in the simultaneous withdrawal of CRs. This technology has been implemented in the ETRR-2.
16.8.	16.8.3.4 Inadvertent Control Bank Extraction	'Thus, a malfunction of the RCMS that could lead to the extraction of more than one CR is sufficiently unlikely as to render it beyond the design basis'.	Please estimate the probability of this 'sufficiently unlikely' event. Please explain whether the RCMS is a proven technology and provide relevant testing information?
			Response: 1. The failure of this system implies the failure of two

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
			independent hardwired devices. The probability of failure is less than 10^{-6} per year. 2. The RCMS is a proven technology. The Foxboro I/A series has been in existence since 1985 making it one of the pioneers in SCADA systems. The system is used extensively in the power (both nuclear and fossil), petro-chemical and railway industries. The Foxboro Company recently received a multimillion dollar retrofit order for the upgrade of the safety and control systems at KORI Unit 1 nuclear power plant, located in southeastern Korea. KORI was the first of the nine nuclear power plants currently in operation in Korea. This project illustrates Foxboro's ongoing commitment to its line of nuclear qualified analogue and digital instruments as well as its continued leadership in this industry. Foxboro's scope of supply includes the I/A Series control system in combination with Spec 200, and Spec 200 micro nuclear qualified instrumentation. Foxboro has more than 50,000 nuclear-certified modules supplied to 120 nuclear power plants around the world.
16.9.	16.8.3.6 Inadvertent Withdrawal of a Pneumatic Can with Excessive Absorbing Material.	Accidental withdrawal of a few pneumatic cans.	1) Please explain the basis for 40 pcm x 6 cans = 240 pcm reactivity being considered for withdrawal? 2) How many pneumatic cans may be accidentally simultaneously extracted as a manual control action after a high temperature alarm?
			Response: 1. The worth of the insertion is not obtained by assuming extraction of 6 cans. This number was reached by

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
			postulating the maximum reactivity worth allowed for movable targets by the regulation of the Argentina Nuclear Regulatory Body (ARN), 200pcm, plus a 20% margin. 2. The interlocks and control logic built into the transfer system do not allow accidental simultaneous extraction of more than one can at a time. For a description of the system see PSAR Chapter 11, Section 11.4.2.
16.10.	16.8.3.6 Inadvertent Withdrawal of a Pneumatic Can with Excessive Absorbing Material	Accidental withdrawal of a few pneumatic cans with the simultaneous failure of the FSS.	This event should be considered both with the successful actuation of the FSS and the failure of the FSS and the following actuation of the SSS.
			Response: The reactivity involved in this event is bounded by more severe cases analysed in Chapter 16. Nevertheless, the analysis has been performed and will be provided to ARPANSA.
16.11.	16.8.7 Design Basis Postulated Initiating Events	Inadvertent control bank extraction – Sufficiently unlikely to occur (Beyond Design Basis).	Needs to be verified. Refer to question 16.8.3.4.
			Response: There is no bank extraction mode for the RCMS. See responses to Questions 16.7 and 16.8. Also see PSAR Chapter 5, Sections 5.5.2.5 and 5.5.2.8 for a description of the inhibition on bank extraction.
16.12.	16.8.7.3.2 Calculation Methodology	Transient simulations with the PARET code.	1) As PARET is one-dimensional code, does it model correctly the hot channel by 1-D geometry? 2) How has PARET been validated for this situation? 3) Have transient simulations been conducted using a two-dimensional code?

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
			Response: 1. PARET is used to model both an average channel and a hot channel. Its use to model the hot channel by 1-D geometry is accepted practice supported by its performance in modelling experimental data. 2. See Section 16.3.2.1.1. It should be stressed that the US DOE has promoted extensive international use of PARET as part of the RERTR program. 3. The validation analysis and comparison with experimental data from the SPERT and CABRI experiences concludes that PARET is an appropriate tool to calculate reactivity accidents in small cores of research reactors.
16.13.	16.8.8 Conclusions	'From the results it can be concluded that none of the transients analysed results in damage to the core'	Table 16.8/1 recognises that ONB for the hot channel may be achieved, ARPANSA will require further analysis of this case by another code to establish a benchmark and gain confidence in the PSAR calculations. The issue of calculation uncertainty should be discussed here as well.
			Response: ONB is not a critical phenomenon. The Tong correlation for DNB in this transient gives a minimum DNB ratio larger than 1.5. This is consistent with thevalues adopted internationally for minimum DNBR in transients.. Application of steady state correlations to predict critical phenomena during transients is widely accepted as a conservative practice. The Tong correlation was selected to calculate the DNBR in PARET as it this correlation that best compares with SPERT and CABRI experimental results. We would refer ARPANSA to our response to Questions 16.4 and 16.3. INVAP are currently simulating reactivity transients that involve ONB with RELAP.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
16.14.	16.9.2.1.1 Pump Shaft Seizure	Simultaneous shaft seizure of both primary pumps.	1) Please explain the basis for the simultaneous shaft seizure of both primary pumps mentioned in the PSAR as being "highly unlikely"? Please provide the probability estimate for this event. 2) Has the falling of a foreign object into the core during refuelling been considered? 3) Has the situation of simultaneous shaft seizure of one primary pump and shut down of another pump (for example, due to a very high vibration in the bearings of the inertia flywheel) been considered?
			Response: 1. The chance of a pump seizure is considered to be unlikely due to the high quality of manufacture and the instrumentation provided on the pumps that will detect degradation prior to seizure. The independent coincidental seizure of both operating pumps is thus considered to be beyond the design basis. In addition, the PCS pumps are located in separate rooms thus ensuring that there is no credible mechanism by which a hypothetical shaft seizure in one pump can produce a shaft seizure in the other pump. The low probability of this event will be confirmed during the detail engineering phase and production of the draft FSAR.. 2. The falling of an object into the core during refuelling does not contribute to shaft seizure. The water leaving the core goes to the decay tank that acts as a very effective filter. 3. Yes, and the analysis shows no damage to core.
16.15.	16.9.2.2.3 Core Blockage	'Complete blockage of a fuel assembly nozzle is not credible …' (page 16.9-10).	Please provide the basis for the statement that this event is "not credible".

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
			Response: The argument for incredibility of complete blockage of a fuel assembly nozzle is that set out in the section. In essence, it requires the presence of a sufficiently large object downstream of the heat exchangers. No mechanism can be foreseen whereby a sufficiently large object could be generated.
16.16.	16.9.2.2.3 Core Blockage	' …the occurrence of significant core blockage is considered sufficiently unlikely as to render the event beyond the design basis' (page 16.9-10).	Please define what is meant by "significant core blockage" and provide the basis for the statement that this event is "beyond the design basis".
			Response: The term essentially refers to a degree of channel blockage that would lead to a challenge of the core safety limits. The argument for its occurrence being outside the design basis is that set out in the section.
16.17.	16.9.3.3 Improper Power Distribution Due to Unbalanced Rod Positions, Radioisotope Targets, or Erroneous Fuel Loading	Engineering Hot Spot Factors.	Engineering Hot Spot Factors should be considered here as the factors enhancing the events associated with the causes listed on page 16.9-11.
			Response: Should any of these events occur, the reactor will not be submitted to a transient. It will operate in a condition that deviates from normal operation parameters. The engineering hot spot factors have been included in the calculations of the effect of these situations on the steady state operation under these conditions, as described in Chapter 5, Section 5.7.5.
16.18.	16.10.3 Design Basis Postulated Initiating Events	Probability of pipe or heat exchanger blockage in the SCS loop vs PCS failure.	Please explain why pipe or heat exchanger blockage in the SCS loop is considered as a Design Basis PIE while core blockage due to PCS failure is considered as 'sufficiently unlikely' to occur?

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
			Response: The SCS is a Safety Category 2 system and includes cooling towers that are open to the atmosphere. The PCS is a Safety Category 1 system, access to which is strictly controlled.
16.19.	16.11.2.1 PCS Pipe Break or Valves Failure	'The siphon effect breakers are protected against blockage by a mesh with 10 mm openings' (page 16.11-2)	Can the 10mm mesh disintegrate due to corrosion? What would happen in a result of such disintegration?
			Response: The mesh will be made of Stainless Steel with appropriate thickness. Corrosion is not considered likely owing to the operating conditions. Corrosion sufficient to cause disintegration is considered so unlikely as to render it beyond the design basis. Even in the event of disintegration, any debris will be swept into the PCS decay tank where it would sediment out due to the low flow in the tank.
16.20.	16.11.4 Design Basis Postulated Initiating Events	Number of locations and break sizes analysed for PCS.	Please justify the sizes and number of breaks considered and the conservatism of the analysis.
			Response: Full break of large diameter pipes is considered very unlikely due to: • Operating conditions (mild temperature, low pressure, high water purity), • Absence of high energy lines, • Instrumentation on pumps (for early detection of abnormal vibration caused, for instance, by abnormal behaviour of the inertia flywheel,) • Seismic classification, • Design limits of the piping.

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			Therefore, break of small or medium size pipes or seal or valve failure leading to an equivalent break is postulated. Full break of the interconnection line is postulated due to its length and lay out. A break at the pump discharge (with the break of the same diameter as the interconnection line) has two purposes: a) To give an upper bound for the response of the reactor after the break of the interconnection line (given the limitation in the minimum time step that can be implemented in RETRAN, the break of the interconnection line could not be analysed directly and two bounding cases were adopted). b) To simulate a failure in a seal at the union between the pump and the piping, or the pie and the heat exchanger. The pump discharge line is the point of highest pressure in the Primary Cooling System. A LOCA through the pump drain line has been postulated because it is considered a moderately likely event.
16.21.	16.11.4.3.1.3 ND 100 mm at the PCS Pump Discharge Line	'…from this point onwards only sequences assuming success of the FSS will be analysed' (page 16.11-20).	The fact that the cases 'success of FSS' and 'failure FSS/success SSS' give close results for this particular break in the PCS is not a good justification for neglecting the case 'failure FSS/success SSS' for other break sizes and locations. It is not a conservative approach taking into consideration, in particular, that breaks may occur due to earthquake resulting in FSS failure. Therefore, it is recommended to consider all LOCA breaks with the 'failure FSS/success SSS' approach.
			Response: The curves of reactor power vs. time for a 100mm break at the pump discharge with failure of the FSS and success of the SSS is nearly identical to the one corresponding to the success of FSS. The difference in integrated energy dissipated

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			in both cases causes nearly no difference in the results for temperatures and mass flows for both transients. Since a 100 mm break at the pump discharge is the fastest transient analysed (where power differences in the first few seconds could be important), and nearly no difference in the variables analysed were observed, failure of the FSS was not analysed for the rest of the transients studied.
16.22.	16.11.4.3.1.6 Primary Cooling System Pump Outlet Break With Pumps On. Parametric Analysis.	Analysis methodology and obtained results	For this scenario, please justify why FSS failure followed by actuation of SSS has not been considered? Extrapolation of RETRAN2 results is not acceptable to ARPANSA. Another validated code should be used to calculate the case of a large break.
			Response: As explained in Section 16.11.4.3.1.6, very large breaks (of about the pipe diameter) at the pump discharge line could not be modeled with RETRAN02. When the break diameter is similar to the pipe diameter where the break occurs, the depressurization is so fast that the program fails to converge for the smallest time step achievable. The pump outlet pressure is very sensitive to the break diameter, because the rupture is suddenly exposed to a room at atmospheric pressure. Reductions in a factor of about 3 in the pressure at the pump discharge are observed for the larger diameters analysed. A calculation with constant pump outlet pressure would give highly overestimated results. Therefore, dependence of the pressure at the pump discharge with break diameter needs to be carefully studied to estimate flows and times for the breaks that cannot be calculated with RETRAN02 program.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
			Breaks up to 250 mm could be calculated with RETRAN02 for the ND 350 mm diameter pump outlet pipe. Therefore a function for Δp was interpolated for the points calculated with RETRAN02, and the results were extrapolated up to 350 mm diameter. Then the mass flows through the break were calculated from these extrapolated results, and with these mass flows the times for pool liquid level to reach siphon effect breaker level were calculated. These points showed the same trend exhibited by the times calculated up to 250 mm diameter. This was the only possible way to estimate the results for very large breaks at this stage. The results will be compared with the results of RELAP, another validated code able to converge for this diameter in the future.
16.23.	Table 16.11/2 Summary of Loss of Coolant Analyses for the PCS	Completeness of Table.	Table should include the cases of 350 mm D break in PCS with FSS success and FSS failure/SSS success.
			Response: Comment noted. Information will be provided to ARPANSA once the calculations are performed with RELAP.
16.24.	16.12 Analysis of Loss of Heavy Water Events	Possible leak of D2O.	1) What will happen in case of a small leak at the BOC, when the reactor has enough reactivity to compensate the leak? Please explain the words: 'compensation is limited'. 2) What is the maximum extraction of CR allowed, compensating for D_2O leak, before the reactor will be shut down?
			Response: 1. The loss of reactivity caused by the dilution of its contents with light water will be compensated for by the RCMS raising the CRs to some identified limit. Thereafter, no

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			further reactivity will be added and, assuming the complete absence of any alarms or reactor trip indication, the plant will shut itself down. Loss of heavy water will lead initially to a decrease in level in the expansion tank. As indicated in Chapter 6, Section 6.6.7.1.3 and Section 16.12.1 the FRPS trips the reactor on very low D_2O level in the heavy water expansion tank. 2. A D_2O leak will lead to FRPS reactor trip due to very low level in the expansion tank.
16.25.	16.12.2.3 Numerical Analysis	The effect of the heavy water loss on the reactivity of the core.	1 This effect of heavy water loss due to leakage should be discussed here and should refer to the part of PSAR where the consequences are estimated. 2 The effect of heavy water degradation from all causes on reactor reactivity should be discussed here and should refer to the part of PSAR where the consequences are estimated. 3 How will a small D_2O leak be detected?
			Response: 1. See Section 16.12.4. The effect of heavy water loss is not expected to lead to any offsite consequences. Only the operators might be expected to be exposed in some manner. The Health Physics arrangements are discussed in Chapter 12. 2. The effect of degradation of the purity of the heavy water is considered in Chapter 5, Section 5.7.5.5.3. The degradation is not considered to lead to any offsite consequences. 3. The mechanisms for the detection of leaks are set out in the Level 2 part of Section 16.12.1.
16.26.	16.13.3 Mechanical	'Should a fuel assembly be accidentally	Mechanical damage to FA during fuel shuffling manoeuvres may

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
	Damage to Core or Fuel Assembly	damaged during fuel shuffling manoeuvres, the operator will remove it to the storage rack inside the reactor Pool'.	be either ignored or unnoticed by the operator. A slightly damaged FA then may be installed into the core. This accidental event should be considered as DB PIE.
			Response: Fuel handling operations are performed by trained operators, acquainted with ANSTO's safety culture, using appropriate tools and in accordance to established procedures. Nevertheless mechanical damage to fuel elements during fuel handling operations is conservatively considered to be a design basis event but one that is bounded by fuel plate cladding failure during normal operation due to a manufacturing defect as discussed in the last paragraph of Section 16.13.3. It should also be noted that the PCS is provided with a failed fuel monitor that will detect such fuel damage where it is sufficient to result in a release.
16.27.	16.13.4 Criticality in Fuel Storage	Criticality assessment.	1) Has the criticality assessment of the new fuel storage considered the situation of fire combined with flooding? 2) Please clarify '±0.001' keff? Is it 3σ? If 'Yes', what is total calculation uncertainty (including modelling and nuclear data uncertainty)?
			Response: The fire suppression system in the fresh FA storage room is dry. Therefore, the simultaneous occurrence of fire and flooding due to independent causes is considered very unlikely. The value 0.001 is for sigma. For the preliminary design analysis (and PSAR) all the calculations were based on an infinite array of FA's. In this kind of model it is not possible to make a comparison with experiments in order to have an idea of the modelling uncertainty. However, a modelling uncertainty can be estimated as follows:

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			Loading uncertainty +/-350 pcm Mechanical uncertainty +235/-93 pcm. We have estimated a maximum modelling uncertainty of 600 pcm and 300 pcm for 3-sigma statical uncertainty. Then the total uncertainty is 900 pcm or 0.009. The modeling uncertainty can be absorbed by the impurity contents of the material. The calculations were done without impurities. The expected contents of impurities will decrease the core reactivity in 746 pcm. This result can be extrapolated to the effect of impurities on the flooded storage room.
16.28.	16.15.2.1 Excessive Power	Potential failure of a uranium metal target due to overpower. (page 16.15-2) 'Considering ANSTO's conservative target design (qualified by tests) and operational procedures … the potential failure of a uranium metal target due to overpower is considered to have been eliminated by design provisions'.	Please clarify this statement with regard to any proposed new target design. ARPANSA expects that if another type of uranium metal target is proposed then the new design should be a proven technology with experimental verification and justification by validated computer codes.
			Response: The detail engineering of the uranium metal target is in progress. The target will be based upon the experience gained with the target currently proposed for HIFAR. At present it is not envisaged that another type of target be used. If, in the future, it is decided to pursue the adoption of a new design, submissions will be made to ARPANSA.
16.29.	16.15.3.2 Excessive Target Heating Power	The maximum design power of a rig.	The adopted maximum design power of 140 W per target should be justified by thermal hydraulics calculations. Has it been done

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			in another part of PSAR?
			Response: Relevant documentation will be supplied to ARPANSA.
16.30.	16.15.3.2 Excessive Target Heating Power	Radiological impact of a target can failure.	Why is the radiological impact of this event bounded by the failure of a can in the transfer hot cell?
			Response: Failure of the can in its irradiation position will see radioactive material released at that position. In order that the material escape to the environment, it will have to travel along the pneumatic tube. Considerable retention would be expected in such a case. Failure of the can in the hot cell will see a greater inventory of radioactive material made available closer to any leak site.
16.31.	16.15.6.3.3 Third Level – Preventing Reactor Damage	Cold Neutron Source detonation.	It is not clear what will happen in case of a detonation. It should be discussed. Calculations should be included as well.
			Response: As explained in Section 16.15.6.3.3 and Chapter 11, Section 11.5.3 together with Figure 11.5/16, the vacuum containment is designed to withstand the detonation. The effect of detonation within the containment vessel of the CNS on the reactor is limited to a negative reactivity insertion. The safety case for the CNS will be presented as a separate document.
16.32.	16.15.6.4 Reactivity Influence	Cold Neutron Source failure	1. Reactivity is stated in % here, while the pcm units are used elsewhere in the PSAR. Use of reactivity units should be standardised. 2. What is meant by 'large amount of negative reactivity'? Please quantify. 3. Please explain how the Cold Neutron Source explosion may change reactivity of the core?

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
			Response: 1. Comment noted. This will be amended in the next revision of the PSAR. 2. As indicated in Chapter 5, Section 5.7.2.3, the reactivity insertion is –250 pcm. 3. Any detonation within the CNS will see the contents change rapidly from liquid to vapour. The reduced density of material will reduce the moderation of neutrons, essentially adding negative reactivity to the core.
16.33.	16.15.8.1.3 Numerical Analysis	Accident with a rig leading to ONB. 'The cladding temperature in the hot rig reaches the onset of nucleate boiling limit (127°C), transgressing it for a few seconds'.	What is the calculation uncertainty? Please provide details of calculation uncertainty estimations for this particular case.
			Response: Relevant documentation will be supplied to ARPANSA.
16.34.	16.15.9 Conclusions	Accident with a rig leading to ONB. '… surface temperature of the hot rigs exceeds the onset of nucleate boiling temperature for a short period of time. This will not compromise the integrity of the rig's cladding'.	Please verify this statement.
			Response: ONB is not a critical phenomenon. During the transient the minimum DNBR remains above 1.5 when calculated with a steady state correlation (conservative), therefore no damage is predicted. The presence of void lasts only a few seconds. Redistribution is not the dominant phenomena in transients due to their short duration. Therefore no damage to the cladding will occur.

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16.35.	16.17.8 Earthquake	Simultaneous failure of FSS and SSS	What is the likelihood of simultaneous failure of FSS and SSS due to an earthquake?
			Response: Both the FSS and the SSS are designed as seismic category 1 systems; thus they will operate appropriately in case of earthquake. The various likelihoods of failures are considered in the PSA.
16.36.	16.18 Human Error	Mechanical damage of FA due to mishandling of the FA handling tool .	Refer to 16.13.3 (Mechanical Damage to Core or Fuel Assembly)
			Response: Refer to response provided earlier.
16.37.	16.18 Human Error	'Fuel follows a one way path during its stay inside the reactor. During refuelling, all irradiated fuel in the storage rack is removed to the Service Pool before any fuel assembly is removed from the core. After the storage rack is empty, the spent fuel in the core is removed and placed in the storage rack. Reshuffling is then carried out'	Please make appropriate references to Chapter 10 fuel management sections to improve comprehension of this section.
			Response: Comment noted. This will be addressed in the next revision of the PSAR.
16.38.	16.20.3 Probabilistic Safety Analysis Results Core Damage Frequency	Credibility of the results obtained.	The quoted core damage frequencies obtained are to be justified.
			Response: The justification is contained within the full PSA.
16.39.	16.19 Beyond Design Basis Accident	'The purpose of this section is to investigate these sequences further with a view to defining an accident to be used for emergency planning purposes'.	The level of severity of accident used for emergency planning should be clarified with relation to the Reference Accident used in the Siting Application. Please confirm that the ANSTO policy is to maintain an off-site emergency responsibility capability.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
			Response: The Reference Accident was determined on the basis of limited available information of the proposed plant. Now that the design of the plant is known, the assumptions supporting the Reference Accident have been refined and incorporated into a detailed safety analysis. This analysis has shown that the degree of core damage and degradation of safety systems assumed for the Reference Accident is unwarranted. We confirm that ANSTO intends to maintain an offsite emergency response capability even though it is not considered necessary in the case of the Replacement Reactor.
16.40.	16.19 Beyond Design Basis Accident	Reflector tank tritium activity.	Please consider an accidental release of heavy water into the pool. How does it effect the reactor staff? What is the effect on reactivity?
			Response: Failure of the Reflector Vessel would see an inward flow of light water with no escape of heavy water.
16.41.	16.19.2 Reactor and Service Pool Cooling System Pump Shaft Seizure	Credibility of these analyses.	1) Please discuss the conservatism of the assumptions used in the analyses. 2) What is uncertainty in dose calculations performed with PC-COSYMA? 3) Validation issue for PC-COSYMA should be addressed.
			Response: 1. The conservative hypothesis include: • Shaft seizure • Failure of the Safety Category 1 flow meter • Containment isolation occurs two minutes after the beginning of release. • Maximum barometric pressure fluctuation

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
			• Worst atmospheric conditions maintained during the duration of the transient. 2. This is a deterministic analysis and therefore uses conservative assumptions on parameter values, as explained in the analysis information provided to ARPANSA. The cut-off was placed at 1% as shown in PC COSYMA Users Guide, page 4.124. For population numbers, best estimate values have been adopted according to the latest available census data. 3. PC COSYMA has been developed as part of the European Commission MARIA project by the UK National Radiological Protection Board (NRPB) and Germany's Forschunszentrum Karlsruhe. The COSYMA code has been developed under a formal QA system that involved all the steps from functional specifications to validation and verification of models and built in data.
16.42.	Table 16.19/6 and Table 16.19/8.	Credibility of calculation results.	1) What is the uncertainty of calculated results? 2) Please explain the meaninglessly low negative exponents in Table 16.19/8.
			Response: 1. See response to Question 16.41, item 1. 2. The data were directly extracted from the code output, with no elaboration. The numbers are effectively zero.