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16.43	16.1 Safety Analysis--Introduction	The design analysis (16.7 and 16.9) show that the reactor goes through a series of safe states following the occurrence of a range of Design Bases Events (DBE). Only three sequences involve fission product release and these do not involve any significant damage to the reactor core, namely: • failure underwater of 12 uranium-molybdenum rigs, • melting of 3 Fuel Assembly plates, • melting of a uranium metal rig in the hot cell.	The focus is on design basis events, although there is treatment of some selected beyond design basis events in Section 16.19. Also, as outlined below some DBEs might not have been considered (Low Power Operation). The level of credit given to the timescale for effective Second Shutdown System (SSS) needs further clarification and justification (see 16.3.3, 16.8.3.1, 16.8.7.3.3, 16.9.4.2,16.15.8). The extent of fuel damage is very low (3 fuel plates out of 336 or 1%).
			Response: The approach adopted for the presentation of the safety analysis in the PSAR concentrates upon the deterministic analysis of design basis events showing, to a high degree of confidence, the ability of the plant to withstand them. This is in accordance with IAEA methodology used internationally (SS 35-G1 Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report). However, the risks associated with very rare accidents representative of the plant have also been considered. The level of credit given to the timescale of operation of the Second Shutdown System is conservative. The discharge as a function of time has been obtained from measurements performed in a full-scale mock up, with single failure (five out of six valves open), and the errors estimated for the measurements have been included in the modelling of the SSS. The reactivity worth has been calculated using the MCNP code.

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			The extent of fuel damage is consistent with the accident scenarios presented in the PSAR and the PSA.
16.44	16.3 Identification of Initiating Events	The IAEA guidance for Safety Analysis has been followed and a comprehensive set of Postulated Initiating Events (PIEs) has been considered. These cover internal faults, external hazards, internal hazards and human factors. The screening criteria adopted are: • inapplicability to the design, • elimination by inherent design provisions, • sufficiently unlikely to occur.	The depth to which the PIES are examined and the screening out by including bounding accidents has been raised in the review of the RRR PSA. It appears that beyond design bases accidents (BDBA) are considered to be not credible, and thus not considered further in the deterministic analyses. This is not consistent since selected BDBA have been considered in section 16.19 as well as in the PSA. What was the basis for selecting the BDBAs considered in Section 16.19 and excluding others?
			Response: The design intent is that no significant core damage should occur for design basis initiating events and fault sequences. A nominal likelihood of 10^{-6} per year is assigned as a cut off frequency for design basis initiating events and fault sequences. Those events and sequences of lower likelihood are assessed to determine the level of residual risk they pose. The PSA included as part of the PSAR primarily considers accident frequency. The estimation of consequences of representative accident sequences in Chapter 16 is not inconsistent; it is an issue of what information is reported where. The BDBAs considered in Chapter 16 were selected on the basis that they formed the representative risks of the plant consistent with the design.
16.45	16.3.3 Methods of analysis	The design philosophy is that no significant damage should occur for design basis	See comments on the PSA and later comments under Ch.16 on the effects of failure of the FSS.

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		initiating events (DBIE or DBA). The analysis considers credible sequences, employs bounding methods, and uses approved computer codes. Failure of one safety system, such as the FRPS and FSS is deemed credible, but dual failure is not considered credible.	A significant issue is the credit given to the SSS during the 15 seconds before it is fully effective.
			Response: The SSS starts to be effective some 5s following its initiation as the heavy water drains from the reflector vessel. The negative reactivity it inserts after 5s is conservatively modelled in the analyses. See response to Question 16.43.
16.46	16.3.3.2.1.1 PARET-PC.	The PARET code has been developed by ANL-USA to calculate transients and accidents initiated by reactivity or power changes. The coolant may be either single phase or two-phase and the code contains flow instability, departure from nucleate boiling (DNBR) and single and two phase heat transfer coefficients. The code also provides a coupled thermal-hydraulic and points kinetic capability. This permits reactivity feedback and a voiding model which estimates voids produced by sub-cooled nucleate boiling	The ability to consider reactivity feed back is a key advantage of PARET. Please clarify whether its heat transfer correlations cover the "pulse cooling mode" described under some transient conditions, and justify use of the correlations. From 16.3.3.2.2.4 it appears that only two channels have been modelled (hot channel and average channel). Has the effect on reactivity, pressure differential and clad temperature of fuel channels across the core being periodically voided and filled been considered?
			Response: PARET heat transfer correlations suite is appropriate for different regimes and conditions that occur during a reactivity insertion transient. The use of the correlations built into the code has been justified by ANL in several reports that describe the core, its models and its validation against SPERT and other experiments. Given the characteristics of the transients analysed for the RRR and the operation of the FSS and the SSS, there is no

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			transient that exhibits periodic voiding and filling effects.
16.47	16.3.3.2.2 Data Input Modelling Assumptions	The following assumptions are important : • PARET uses an uncertainty value of 15% for the reactivity feedback coefficients. • RETRANO2 assumed no reactivity feedback. • The calculations are performed using a Beginning of Cycle core (BOC). • The assumed nominal power is 20 MW.	1. Has any end of life aspect been considered for the control rods (CR) since they lose about 20 % effectiveness over their 8 year life? 2. Also, are there any results from using RETRANO2 and PARET for the same sequence or transient? 3. A major difference between PARET and RETRAN is the absence of reactivity feedback in the RETRANO2 analyses. How has feedback been considered in the RETRANO2 runs covering the period between failure of the FSS and SSS success?
			Response: 1. Section 5.7.5.4.1.2 indicates that Control Rod depletion will result in a loss of 16% effectiveness over their expected 8 year life. This loss of effectiveness is considered through the 20% reduction in the worth of the FSS included in reactivity insertion worth calculations. 2. No, PARET was used for reactivity insertion transients and RETRAN02 was used for LOCA, LOFA, Loss of Heat Sink and Loss of Off-site Power. 3. Reactivity feedback coefficients have not been included in RETRAN. The delay for actuation of the SSS is 2.12s, and no significant contribution of feedback coefficients is expected. Given the range of transients analysed with RETRAN, none of them involves overpower, therefore there is no physical mechanism to produce a significant reactivity effect. The assumption of ignoring the feedback coefficients is conservative.

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16.48	16.3.3.3 Radiological Consequence Analysis	The design philosophy is that there should be no core damage for design basis accidents. In the case of accident sequences having some fission product release, the codes ORIGEN and PCCOSMA have been used to determine the fission product yield and radiation doses respectively.	There appears to be an inconsistency with Section 16.3.3, which refers to "no significant" core damage and not "no" core damage. The philosophy could be interpreted to mean that any sequences leading to core damage are excluded from consideration. What is the intent?
			Response: The intent is that no design basis accident should result in a release of radioactive material with the potential for offsite consequences. The wording 'no significant core damage' was used to indicate this intent. There are a number of fuel plate failures (primarily due to mechanical damage during handling) that are considered to lie within the design basis that lead to a limited release of radioactivity into the pool but that will have no offsite consequences.
16.49	16.4 Reactor characteristics	There are two independent RPS for the management of safety systems, the FRPS, which initiates the FSS and Containment Isolation, and the SRPS, which initiates the SSS.	If the FRPS fails, what initiates the containment isolation? Is there a back up hard-wired manual system operable from the main and emergency control rooms?
			Response: The Radiation Monitoring System consists of three sets of three monitors monitoring stack particulate activity, iodine activity and noble gas activity. Failure to signal containment isolation when necessary is considered highly unlikely. In the event that an isolation signal was not sent, sufficient alarms would sound, permitting manual operator isolation. This system is available in and can be operated from the Main Control Room and the Emergency Control Centre.

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16.50	16.4.3 Natural Circulation Cooling	The large amount of water in the reactor pool acts as a heat sink, with heat being lost to the containment atmosphere via evaporation.	The ultimate heat sink is the containment atmosphere if the containment is sealed. Has any consideration been given to providing a diverse heat removal and heat sink by upgrading the long-term pool decay heat removal system to the Cooling Tower pond?
			Response: The ultimate heat sink following reactor trip and initiation of natural circulation cooling is the reactor pool. The Containment Energy Removal System (CERS) ensures heat transferred to the containment atmosphere from the pool is rejected to the external atmosphere to maintain constant pressure, humidity and temperature conditions inside the containment and thus minimise leaks after isolation. The large pool water inventory is a passive, highly reliable heat removal mechanism. No failure mechanism is envisaged for this ultimate heat sink. Failure of the CERS will not prevent the pool from absorbing the decay heat.
16.51	16.7 Analysis of Loss of Electrical Power	All engineered safety features (ESFs) instrumentation and electrical equipment is on the standby power supply (SPS). Some ESFs are supplied by the uninterruptible power supply (ups), which has a 30 minute battery capacity. Although not ESFs, the long-term pool cooling pumps of the RSPCS and SCS are supplied from the SPS.	In view of their connection to the SPS, and their potential as a diverse decay heat removal system, why not classify the RSPCS and SCS long term pool cooling pumps as safety systems? (See 16.4.3 above)
			Response: See response to Question 16.50.
16.52	16.7.2 Loss of Normal Power Supply for up to 30 minutes.	Loss of normal power supplies for up to 30 minutes is an anticipated operational occurrence (not a PIE). It has no consequences due to design	See review comments on the PSA. This loss of power can be anticipated about twice per year. After pump coast-down (say 100 seconds) heat removal is dependent on

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		features.	natural circulation. The power in this period is much greater than the nominal design capacity for natural circulation operation (400KW), so how is the fuel clad protected? If it is associated with a failure of the FRPS-FSS (10^{-3} per year) then the event frequency is a PIE with a frequency of $2*10^{-3}$ per year. During this sequence the reactor will be at full power with significantly reduced flow and, if modelled with RETRANO2, there is no reactivity feedback allowance. How is the fuel clad protected.
			Response: The question on frequency is discussed in the response to the questions raised on the PSA (see response to Question PSA.57). For the loss of Normal Power transient, the flap valves open when the core decay power is approximately 1 MW. The subsequent temperature transient of the fuel and cladding does not reach ONB temperatures. The plant has the <u>capacity</u> to operate under conditions of natural circulation at this power level. An operational <u>limit</u> is placed on core power for normal operation under natural circulation conditions of 400 kW to keep clad temperatures below 110ºC, minimising corrosion and radiation doses at the reactor pool top due to plume effects. The loss of Normal Power together with failure of the FSS is considered to lie within the design basis and has been modelled deterministically, independent of consideration of likelihood. The failsafe characteristics of the FSS under loss of power render this scenario extremely unlikely. The loss of Normal Power will result in the decoupling of the electromagnets and the de-energising of the CRDs motors

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			that hold the control plates up and the plates will drop into the core. The quoted reliability of the FRPS is therefore inapplicable in this case. The results reported in the section show that clad and coolant temperatures are higher than in the case where the FSS successfully actuates but still within acceptable limits, even ignoring feedback effects (conservative assumption).
16.53	16.7.2 Loss of Normal Power Supply for up to 30 minutes	Loss of power supply for greater than 30 minutes is not considered, other than for the extreme case of 10 days (see Section 16.19)	The experience with HIFAR is to anticipate loss of off site power for up to 3 hours a number of times in the life of the plant (see HIFAR PSA). It is not clear why the PIE considered is for total loss of supply for only 30 minutes—if diesels do not start and run within a few attempts they are unlikely to start and run. Dependence would then be on return of off-site supply, since the UPS is sized for 30 minutes. Please comment. A reliability analysis for the power supply should be provided at the detailed design stage, covering both the off-site, SPS and UPS.
			Response: A small number of conservative fault sequences were considered sufficient for this initiating event. The sequence involving a total loss of supply for 30 minutes was considered to conservatively bound those involving the start up of the diesels. The diesels are expected to start up quickly. The conservative assumption was made that it takes 30 minutes to get them started. Any delay further than 30 minutes would see the failure of the UPS. These subsequent sequences are conservatively considered in the sequence involving total loss of power for 10 days (Section 16.19).

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			A reliability analysis for the power supply system will be developed during the detail engineering phase and will be provided in the FSAR.
16.54	16.7.5.3 Loss of Normal Power Supply for up to 10 days - Numerical Analysis.	RETRANO2 is used to analyse this PIE and the results are shown in Figs 16.7/1 to 16.7/16. The results are summarised in Table 16.7/1 for FSS actuation after 1.59 seconds and SSS actuation after 5.89 seconds. The maximum surface temperature is associated with SSS actuation and is 140 °C.	The results shown on Fig. 16.7/12 show two temperature peaks for the cladding at about 10 seconds and 128 seconds (about 125°C). This suggests that ONB is not reached, which is inconsistent with Ch.5 of the PSAR. Ch.5 suggests that ONB is reached if there is dependence on natural convection before 15 minutes. Please clarify these differences.
			Response: The statement in Chapter 5 regarding the reaching of ONB temperatures for onset of natural circulation within 15 minutes of shutdown, is made in Section 5.8.7.4 and is part of the Design Evaluation for steady state conditions, not for a transient. The actual analysis of the behaviour of the core for the loss of Normal Power sequence shows that ONB temperatures are, in fact, not reached for the average channel. In the case of the hot channel, however, within the limits of the calculations, ONB temperatures are considered to be reached but only for the case assuming actuation of the SSS and only for a short time (5 to 10s).
16.55	16.7.5.3 Loss of Normal Power Supply for up to 10 days - Numerical Analysis.	The results are summarised in Table 16.7/1 for FSS actuation after 1.59 seconds and SSS actuation after 5.89 seconds. The maximum surface temperature is associated with SSS actuation and is 140 °C.	Figure 16.7/7 shows the reactor power drop with time after SSS Actuation. The graph suggests that the reactor power is reduced from 20 MW at 5.89 s to 6 MW at 10 seconds. Is this an accurate representation of the effectiveness of the SSS? Have errors and uncertainties in SSS operation been taken into account?
			Response: Conservatively, it takes approximately 5s from initiation of the SSS trip to the point at which it begins to

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			insert negative reactivity. As soon as negative reactivity starts to be inserted, reactor power decreases. The representation of the SSS in the PSAR is conservative, taking into account uncertainties. It is based on the results obtained in the SSS mock up, and it includes single failure (five out of six valves open to drain the reflector vessel).
16.56	16.7.5.3.3 Reactor and Service Pools Cooling System	The effect of loss of power on the irradiation rigs is shown in Figures 16.7/13 to 16.7/16 for FSS actuation and SSS actuation.	The time scale for power reduction in Figure 16.7/13 represents operation of the FSS and not the SSS. Is this a labelling error? Has the SSS case been modelled?
			Response: This is a labelling error and should refer to the FSS. This will be amended in the next revision of the PSAR. The SSS case has been modelled and the results are shown in Figure 16.7/20 through 25.
16.57	16.8 Analysis of Excess Reactivity Insertion Events — Accidental drop of a fuel element	No fuel loading will be performed with the reactor in operation. All fuel movements will be performed with a shutdown margin of at least 3000 pcm. A fuel element that falls into the core will not be flat, but at an angle due to geometry. The reactivity insertion is 40 pcm.	Why has the dropping of a fuel element directly into a vacant space in the core has not been considered? Has the consequences of a new fuel element dropped onto the reflector tank with the reactor at power, as a result of procedural failure, been examined?
			Response: Fuel handling operations above the core will be performed only with the reactor shutdown. In this condition placement of a fuel assembly into a vacant core position of the shutdown reactor would not inject sufficient reactivity to send the reactor critical. No fuel movements will take place within the Reactor Pool while the reactor is critical. The inadvertent dropping and placement of a fuel assembly next to the chimney on the Reflector Vessel while the reactor is critical has not been analysed. Its distance from the compact core is considered

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			such that it would not represent a significant reactivity insertion. Also note that the Reflector Vessel is protected from dropped loads by screens. These are designed to withstand the impact from a silicon ingot; these being heavier than a fuel assembly.
16.58	16.8 Analysis of Excess Reactivity Insertion Events — inadvertent insertion and ejection of a fuel element	The design of the fuel clamp eliminates incorrect attachment or failure that could lead to FA ejection and re-insertion. It is not considered as a DBIE.	The fuel clamps are Safety Class 2 and Quality Class B. There are about 15,000 fuel movements anticipated in life of reactor. While structural failure is unlikely, could inadequate locking of the fuel clamp result from human factors? In view of the number of fuel movements and the reliance on administrative procedures is it reasonable to dismiss this sequence as a DBIE? What would be the consequences of a failed or incorrectly locked fuel clamp?
			Response: The design of the fuel clamp is such that there is positive indication to the operator securing the clamp that the fuel assembly has not been properly clamped. In addition to the operators clamping the fuel assembly in the Control Rod Drive room, a supervisor will check the clamping before return to power and operators above the pool will check the rigid connection to the core plate prior to startup. The fuel clamp either grips the fuel or it does not. Therefore, if the fuel has been clamped, then the design is such that the probability of later failure of the tool is remote. If the fuel is not clamped and the operator and the supervisor have failed to detect this condition, then the fuel will be dragged out from the core when the PCS pumps are started. The PCS pumps are started before the reactor is

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			taken to power. Damage to the fuel would be minor under such circumstances. There is an interlock that prevents start up of the PCS pumps if the reactor is in operation (critical). Therefore, there is no event where the fuel assembly can be dragged out while the reactor is at power, thus no cooling or reactivity transient could ensue.
16.59	16.8.2.2 Inadvertent Fast Insertion of Irradiation Fissile Material	This event refers to the inadvertent fast insertion of a uranium-molybdenum irradiation rig at full power. The reactivity inserted is estimated as 54 pcm at a rate of 108 pcm per sec. Design arrangement are in place to prevent loading iridium rigs in U-Mo positions and vice versa.	Please clarify whether iridium rigs will be loaded and unloaded at power, and what reactivity they could control.
			Response: Iridium rigs will be loaded and unloaded only when the reactor is shutdown. There is a limit in place of 200 pcm for rigs moveable at power and the iridium rigs have a potential reactivity worth of some 300 pcm (see Table 5.7/15).
16.60	16.8.3.1 Absorber Withdrawal - Start-up Accident	It is postulated that the reactor is almost critical and one CR is driven out of the core with flow assumed to be at its normal operational value. This is considered to bound continuous extraction during low power start up.	It is not clear to ARPANSA why this DBIE bounds the start-up at low power operation. It is understood that one mode of operation involves low power with natural circulation, to allow for neutronic tests. Please provide calculations for the transient at low power, indicating the flow rate, the thermal hydraulic parameters relevant for this mode, and the power and temperature reached for both FSS success, and FSS failure with SSS success.
			Response: The analysis of reactivity transients in low power operation mode is currently being performed. These results will be provided to ARPANSA.

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16.61	16.8.3.2 Absorber Withdrawal—Full Power Accident	The defence in depth (level 3) includes both the FRPS and SRPS. The former has trips on both high neutron flux and period. The SRPS has no period trip.	What is the justification for not including a period trip in the SRPS?
			Response: The intent is that the SRPS should not be a copy of the FRPS. The trip coverage of the SRPS, with failure of the FRPS and high neutron flux, is considered sufficient for the design to provide an appropriate level of safety.
16.62	16.8.3.2 Absorber Withdrawal—Full Power Accident	The movement of the CRDs is sequential, not simultaneous. Withdrawal simultaneously of more than one CR is not an RCMS action and is prevented by a hard-wired watchdog.	How is sequencing achieved and is it a proven system? Where else has it been used in similar reactors and where is it described in the PSAR? Is the hard-wired watchdog part of a redundant system that prevents simultaneous movement of more than one CR? Is the watchdog equally effective for manual as for automatic control, and during the low power operation? Are the sequencing and the watchdog parts of control or protection systems?
			Response: The inhibition to move more than a CR at a time is given by an independent device based on hardwired technology and placed between the controller and the drive mechanism. This device, the hardwired watchdog unit, has two functions: it inhibits the movement of the CR at a speed higher than the nominal speed and it inhibits the movement of a CR when another CR is being moved. Sequential movement is partially described in PSAR Chapter 5, Section 5.5.2.5. RCMS logic is described in Chapter 8, Section 8.7. The sequential movement for start up strategy is being defined in the detail engineering stage. This inhibition is effective during normal operation and low power operation.

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			The ETRR-2 is provided with this type of interlock as well as other reactors. The hardwired watchdog unit is part of the CRD motor unit. In order to have a simultaneous withdrawal, simultaneous failures of the controller and the hardwired watchdog are necessary. The probability of this failure is considered to be of the order of 10^{-8} per year.
16.63	16.8.3.3 Control Rod Drive or System Failure	This event refers to the uncoupling of a CR and upward dragging by the flow. The design minimises the hydraulic lifting force on the CRs. In the case of electro-magnetic failure the corresponding plate will fall into the core by gravity forces.	There is a positive rate trip on the FRPS. Is there any situation where a negative rate trip would be required, e.g. halving time trip similar to HIFAR?
			Response: The negative rate trip was installed in HIFAR to provide protection against failure of a Coarse Control Arm connecting rod which would see a CCA swinging down into the core (inserting negative reactivity) before swinging out (inserting positive reactivity). The protection allowed the safe shutdown of the reactor prior to the insertion of positive reactivity. Such a situation can not happen in the case of the RRR and therefore there is no need for a similar negative rate trip.
16.64	16.8.3.4 Inadvertent Control Rod Bank Extraction	There is no bank extraction mode in the RCMS logic, which prevents continuous withdrawal of more than one CR during reactor operation. The RCMS has a bank insertion mode and a bank extraction is prevented by the hard-wired watchdogs. A malfunction of the RCMS that would allow the extraction of more than one CR is beyond design bases.	The beyond design bases cases considered in section 16.19 do not consider the simultaneous extraction of more than one CR. Is this justified on the basis of a reliability analysis and levels of defence in depth of the RCMS (which is a safety category 2 system). What is the practice in other similar reactors - is bank withdrawal considered as a normal PIE?

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			Response: The failure of this system implies the failure of two independent hardwired devices. The probability of failure is less than 10^{-8} per year. In many other research reactors, such as HIFAR, bank withdrawal of control rods is the normal manner of operation. In the ETRR-2 reactor, bank rod withdrawal was not considered.
16.65	16.8.3.5 Inadvertent Extraction of a Fixed absorbing Irradiation Material	Handling of fixed rigs is neither necessary nor authorised during reactor operation.	Fixed rigs can control a considerable amount of reactivity (500 to 600 pcm). Is it intended to move such rigs at low power (400 kw)? What would be the consequences of unauthorised rig movement?
			Response: The highest worth rigs to be used in the RRR are the iridium rigs, having a reactivity worth of some 300 pcm (see Table 5.7/15). Any rig of worth greater than 200 pcm will only be moved when the reactor is shutdown. In the highly unlikely event of the removal of an iridium rig while the plant was at power, the reactor would safely shutdown tripping on either high neutron flux or high flux rate. This initiating event is being analysed during detail engineering phase. Results will be provided to ARPANSA.
16.66	16.8.3.6 Inadvertent Withdrawal of a Pneumatic Can with Excessive Absorbing Material	These cans normally control 40 pcm but a case is considered where a can controls more (240 pcm). This is considered the bounding case for rig withdrawal.	Is this case appears to be associated with a single can removal? What is the limitation on simultaneous withdrawal of a number of such rigs (there are 18 such rigs in the reactor)?
			Response: The reactivity worth assumed for the calculations was reached by postulating the maximum

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			reactivity worth allowed for movable targets by the regulation of the Argentina Nuclear Regulatory Body (ARN), 200pcm, plus a 20% margin. The worth of the insertion is not obtained by assuming the extraction of multiple cans. Simultaneous withdrawal of more than one can is inhibited by interlocks. This is described in Chapter 11, Section 11.4.2.
16.67	16.8.4 Increase in Moderation Effect—Cold Water Injection.	Three mechanisms for cold water injection are identified, namely spurious operation of the emergency Make Up Water System, start of PCS pump during low power operation, variations in the pool (heat sink) temperature. The consequences of these are bounded by the CR withdrawal at start-up.	Please justify that the consequences of these mechanisms are bounded by the CR withdrawal DBIE. Please provide the results of the cold water transients, particularly at low power and PCS operation. The transient should consider over-cooling of the heat exchangers, ie temperature at pond water temperature.
			Response: Start up of a PCS pump during low power operation (considered the worst possible case with sudden injection of 1000m^3/h) results in a reactivity insertion of 0.5 $ (359 pcm) with a ramp of 3.9$/s. This insertion results in a very mild transient and is bounded by the start-up accident presented in the PSAR.
16.68	16.8.5 Increase in Reflector Effect—Inadvertent refill of the reflector vessel	The refilling of the reflector vessel is slow (about 1 hour). Multiple failures of the protection system would be required for such an insertion.	The need to refill slowly raises the question of isolation valves between the Reflector Vessel and heavy water header tank. If there are such valves then there is a mechanism for the rapid insertion of heavy water into a partly empty Reflector Vessel. Do such valves exist?
			Response: There are no valves between the Reflector Vessel and the RC&PS expansion tank. The peristaltic pumps within the RC&PS are sized so as not to permit a rapid filling of the expansion tank and hence rapid

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			reactivity insertion into the Reflector Vessel (i.e. with both pumps operating, it takes over 2 hours to refill the Reflector Vessel). The pumps are interlocked such that they cannot operate with a standing trip signal from the SRPS present.
16.69	16.8.6 Fast Reactivity Insertion Accidents	A fast reactivity insertion accident has the potential to lead to the release of mechanical energy into the pool water and the pool structures. Recent reports conclude that such BORAX type accidents is incompatible with modern pool-type reactor design (see references). Mechanisms such as dragging of a CR from the core by either manual or flow are prevented by design, and there are no mechanisms for explosions in the Control Rod Drive Room. Also the CR bank removal is excluded by design. The fast reactivity insertion event is not considered as a DBIE.	The beyond design bases cases considered in section 16.19 do not consider any fast reactivity insertion accidents. Is this justified on the basis of a rigorous reliability analysis and levels of defence in depth? Please provide references 1, 3, 4 and 5, cited in support for not considering such an accident in the RRR. Note that a large reactivity insertion accident was selected as the Bases of the Reference Accident for siting purposes.
			Response: The design basis analysis considers the full range of potential reactivity insertions and analyses the bounding ones in terms of reactivity inserted and rate of reactivity inserted. The PSA carried out as detailed a reliability analysis as is possible at this time and concluded that the likelihood of reactivity induced core damage fault sequences was exceedingly small and that the sequences did not contribute to the residual risk of the reactor plant. The cited references will be provided. The selection of a reactivity insertion accident at the time of the siting licence application was made on the basis of only a generic outline of the type of plant to be built on the site. The design of the RRR is such that a large reactivity

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			insertion has been designed out and is therefore not applicable.
16.70	16.8.7.2 Design Bases Fault Sequences	Four DBIES ARE analysed: • inadvertent withdrawal of a CR during start-up • inadvertent withdrawal of a CR at full power • inadvertent fast insertion of a U-Mo irradiation target • inadvertent removal of a can with excess irradiation material from a pneumatic channel	See previous comments and questions on the justification for not considering some sequences as either DBIEs or beyond design bases in section 16.19.
			Response: See responses to questions provided.
16.71	16.8.7.3.3 Inadvertent Withdrawal of a CR during Start-up	The analysis is shown in Figures 16.8/1 to 16.8/5. Cladding temperature is shown to rise to 160 ^{o}C, and remains there for 9 seconds. This exceeds ONB, but does not result in any adverse consequences for the fuel plates. The most severe case is FSS failure and reliance on the SSS.	Please explain why the power starts to turn over at 50 MW, about 2 seconds after the SSS trip signal at 40.6 s. Is this solely due to the negative reactivity feedback from fuel, coolant and void in the channels? It would appear that no credit should be given to the SSS inserting reactivity until about 6.6 seconds after the trip initiation signal. Please clarify. Figure 16.8/1 shows a reactivity insertion of 719 pcm at 40 seconds. If the quoted rate of insertion of 30 pcm/s is used, the value should have been 1200 pcm. Please explain this apparent difference? The negative feed back over the 7.5 seconds from the trip can be estimated to be an effective injection of approximately 719+7.5*30=944 pcm, which causes the power to turn over. Are the feedback effects strong enough to inject this amount of reactivity?
			Response: The power turn over is due to the negative reactivity feedback effects from the fuel and coolant

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			temperature increase. The time delays associated with the failure of the FSS, the tripping of the SSS and initial draining of the expansion tank are such that it takes approximately 5s before the SSS starts to insert significant negative reactivity. This is explicitly considered in the core analysis. The curve of negative reactivity vs time for the SSS has been obtained on the basis of the results of the SSS full scale mock up, with single failure (five out of six valves open) and uncertainties in the measurements. The quoted rate of insertion is an average value that provides a simplification. The insertion rate used followed a reactivity worth similar to that given in Table 5.7/10. The calculations have been done assuming reactivity feedback, with a –15% uncertainty margin in the values of the feedback coefficients. The results of the analysis show that these feedback effects are sufficient to contribute to a reduction of the reactivity inserted during the transient.
16.72	16.8.7.3.3 Inadvertent Withdrawal of a CR during Start-Up		No consideration appears to have been given to the low power operation mode (400 kW) in which the reactor is cooled by natural circulation (10 kg/s). The power rise rate from an inadvertent withdrawal of a CR would be much the same as for the full power case, but the heat transfer capability would be much reduced. Has this case been modelled and what are the results? Would there be fuel damage even if the FSS were to be initiated successfully?
			Response: An analysis of the specific sequence at the low power mode is being performed and will be provided to

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			ARPANSA.
16.73	16.8.7.3.4 Inadvertent Withdrawal of a CR at Full Power	The rate of reactivity insertion is 20 pcm/s, which is 2/3 of the rate for the low power insertion. The results are shown in figs 16.8/6 and 16.8/7. The FSS is assumed to operate successfully.	Although the failure of the FSS is not presented, an extrapolation shows that a delay in tripping until SSS successful would not challenge any of the thermal hydraulic limits.
			Response: Comment noted.
16.74	16.8.7.3.6 Inadvertent Removal of a Can with Excess Irradiation Material from a Pneumatic Channel	The analysis assumes a QA procedure violation in the preparation of the target, resulting in a total reactivity insertion of 240 pcm. The reactivity ramp could be significant due to the high withdrawal speed.	The analysis of this sequence only considers successful FSS action. Please justify why failure of the FSS has not been considered in the PSAR? Has the latter sequence been modelled?
			Response: The sequence with FSS failure has been modelled. The maximum power is 31.1 MW at 2.22 seconds. The FSS is triggered at 0.033s due to overpower signal. An additional 2.12s is considered for the actuation of the SSS. The maximum cladding temperature for the hot channel is 135°C and the peak coolant temperature is 70°C
16.75	16.9.2.1.1 Loss of Flow (LOFA) Accidents—Pump Shaft Failure.	Pump shaft failure leads to a sudden reduction in flow (to 50%), and there would be no coast-down via the pump flywheel. The simultaneous failure of two pump seizures is considered beyond design basis.	Of concern with a pump seizure is the dissipation of the energy in the flywheel. If this is not disengaged from the shaft, significant loads will be imposed on the pump casing, anchorages and pipe flanges. How is the energy dissipated in order to prevent by pipe or pump damage causing a loss of coolant accident? If there is only one pump running, is there sufficient pressure to prevent bypass through a flap valve?
			Response: Pump shaft seizure is analysed as a bounding case of a flow reduction in the PCS. The instrumentation available at the pump and motor is such that the onset of failure will be quickly recognised and signalled to the operators allowing the pump to be shutdown before further

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			damage occurs. In the event of a pump seizure, the connection between the pump and the flywheel would fail, allowing the flywheel's continual safe rotation. One pump running will see a flow reduction in the PCS to some 1400 m^3/hr. The flap valves are designed to open when the flow decreases to 150 to 120 m^3/hr. No bypass through a flap valve will occur in the event of a single pump failure.
16.76	16.9.2.2.1 Coolant Flow Reduction due to Failure or Blockage in Primary Cooling System Piping or Component	One of the main protection devices preventing the entry of objects or debris into the PCS is a grill that covers the core. However, the grill is removed for fuel movements during shutdown.	Is the reactor operated at low power with the grill removed?
			Response: No. The grille will be locked shut whenever the reactor is taken to power in accordance with appropriate operating procedures.
16.77	16.9.2.2.2 Coolant Flow Reduction due to Core Bypass.	A core bypass could occur if any of the 4 flap valves opens during normal operation. The flap valves are a passive design and are kept closed by the pressure in the circuit caused by the running pumps.	Is it correct that multiple conditions are required to trip the SRPS, namely: low pressure drop in conjunction with no-end of stroke signal from two or more CRs or failure of the FSS? Why is there no unique trip on low pressure drop? The FSS failure on demand will cause an SRPS trip in any circumstance.
			Response: No. The SRPS will trip alone on low pressure drop signal, without the need of a no-end of stroke signal from two or more CRs or failure of the FSS. The logic set out in Chapter 8 Figure 8.2/23 is intended to prevent spurious tripping of the SRPS during low power operation in natural circulation. In addition to this trip, there is still the 'Failure of FSS' trip.

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16.78	16.9.2.2.2 Coolant Flow Reduction due to Core Bypass.	Reactor start-up to the low power mode takes place with the flap valves open. An RCMS interlock prevents power raise above 400 KW, and in any case if this were to fail the reactor would be tripped on low flow or low pressure drop.	In order to raise power to 400 kW on natural circulation, the flow and pressure differential trips would have to be dis-engaged. If the RCMS interlock to prevent operation above 400 kW fails, what causes the FRPS and SRPS trips? In view of its importance, why is this interlock not classed as Safety Class 1 and part of the FRPS?
			Response: The FRPS contains an interlock that negates the core flow and core delta-P trips whilst the reactor is in the Low Power operating mode. Failure of the RCMS interlock is thus taken into account by the design of the FRPS and the SRPS that trips the reactor on overpower (above 400kW).
16.79	16.9.3 Power-Flow Mismatch Events	Improper power distribution leads to occurrence of hot spots in the fuel. Ch.5 describes the thermal-hydraulic design and power peaking assumptions. The uncertainties considered are outlined in (a) to (h) and cover a range of effects, including burn-up and xenon effects.	Has the loss of xenon from a fuel plate, or number of fuel plates, as a consequence of fuel damage, been considered? The loss of xenon is a positive reactivity insertion.
			Response: Local failure of a fuel plate would see the release of very small amounts of fission products, including xenon. This release would quickly be picked up by the Failed Fuel Detector and the reactor shutdown. The amount of xenon lost from the core to the PCS would be very small and of minor importance as regards a consequent reactivity insertion. The release of the complete inventory of xenon from a fuel plate would be worth some 10 pcm. Any actual release would result in orders of magnitude lesser reactivity insertions.
16.80	16.9.3.5 System Pressure Deviation	The core outlet pressure is fixed by the water column above the core. Two pumps are normally	Is the third pump prevented from pumping additional water solely by an electrical interlock, or do valves need to be opened?

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	from Specified Limits.	operational and the third is on stand-by. An electrical interlock prevents the start-up of the third pump.	If valves need to be opened, how are they operated? If an electrical interlock only, then should these be part of the FRPS? Have the implications of spurious starting of a third pump been considered from a cold slug insertion and drag load on the CRs? During normal operation with the third pump on standby, is there any SCS flow through the third heat exchanger?
			Response: Inadvertent start up of the third pump while the other two are operating is prevented by a hardwired interlock. The isolation valves either side of the pump are normally open. Spurious start-up of a third PCS pump would result in marginal increase in the core flow (nominally of about 200-300 m^3h^{-1}) and shutdown of the reactor by the FRPS on high core delta-P. Given that the PCS flow is 2000 m^3/h, the dilution of the small cool water volume in the third pump will result in a very modest temperature variation. During normal operation, secondary coolant flows through all three heat exchangers. This, together with the resultant mixing of the flow, ensures that the potential for a cold slug being generated from spurious start up of the third pump is minimised.
16.81	16.9.4.2 Design Basis Fault Sequences (LOFA).	Two sequences are analysed: • pump shaft failure • simultaneous failure of both pump motors. The numerical analysis for pump shaft failure uses RETRANO2 and is shown in Figs 16.9.1 to 16.9.4	Please clarify whether FSS failure is considered for this event. Figs 16.9/2 and /4 suggest that FSS failure is considered, but the text indicates it is not.

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			Response: The likelihood of a pump shaft seizure together with failure of the FSS is considered so unlikely as to render it beyond the design basis. The sequence is considered as a BDBA in Chapter 16, Section 16.19. The text of the figures should refer to the FSS and not the SSS. This is a typographical error that will be amended in the next revision of the PSAR.
16.82	16.9.4.2 Design Basis Fault Sequences (LOFA).	Two sequences are analysed • pump shaft failure • simultaneous failure of both pump motors. The numerical analysis for the failure of the pump motors have been analysed using RETRANO2 and is shown in Figs 16.9/5 to 16.9/10	Fig 16.9/8 shows the difference in the power to be dissipated assuming FSS effective and SSS required. The additional energy to be dissipated between 6.3 s and 20 s is about 150 MJ. The assumption for SSS effectiveness in the first second after Reflector Vessel starts to drain is a factor, since the figure indicates that power drops to 50% within a second. Is this supported by the expected performance of the SSS?
			Response: Following the signal to trip, there is a time interval of some 3s before heavy water starts to drain from the Reflector Vessel. Significant negative reactivity will start to be inserted at about 5s, quickly shutting down the reactor. The performance of the SSS has been validated with the construction of a mock up. See response to Question 16.45
16.83	16.10 Analysis of Loss of Heat Sink Events.	The analysis refers to a loss of heat sink during full power operation. During shutdown and low power operation, the SCS is not required since the pool acts as the heat sink with natural circulation. The reduction in flow in the SCS results in slow variations in the PCS conditions. The results of the analysis, using RETRANO2, is summarised in Table 16.10/1 and Figs 16.9/1 to 16.9/14	There is no reactor trip parameter in the SCS, and reliance is placed on changes sensed by the FRPS (high core inlet or outlet temperature). There is no SCS trip connection to the FRPS. Is this an accepted practice for modern pool-type research reactors - HIFAR does have reactor trips in the SCS?

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			Response: No SCS trip is considered necessary. Any failure in the SCS will result in a very slow thermal transient being seen by the PCS. This will provide adequate time in which operators could make additional cooling available via the SCS or shut down the reactor in a controlled manner. If the operators failed to do anything, the FRPS would trip the reactor via the FSS on high core outlet temperature. In the event of a failure of the FRPS, the SRPS would trip the reactor via the SSS on high Reflector Vessel temperature. HIFAR has a trip on the SCS partly as a consequence of its much smaller thermal inertia.
16.84	16.10.2 Initiating Events for Loss of Heat Sink.	A valve in the SCS placed in the incorrect position has the potential to lead to SCS coolant flow reduction and effect the PCS.	Are all the in-line valves in the SCS operated manual, or are they motor operated valves that can be operated remotely?
			Response: All the in-line valves in the SCS are manually operated at the valve. There are no valves with remote actuation whose spurious closure could result in the loss of the SCS.
16.85	16.10.2.3 Rupture of SCS Boundary	In a number of places the PCS and SCS are run in the same room. In the case of the PCS pump rooms, a SCS rupture could lead to water falling on the pump motors. Much of the SCS piping is seismic class 1. However, the part of the SCS dealing with long term pool cooling is seismic class 2.	The fact that much of the SCS pipework is seismic class 2 means that failure can be anticipated for earthquakes at a return period of 1000 years. Does failure of the SCS have implications for the PCS, in particular where they run in the same rooms?
			Response: The equipment and piping of the SCS used for long term pool cooling and containment isolation are Seismic Class 1. The equipment and piping that is used for full power operation is Seismic Class 2. Seismic Class 2

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			equipment is designed to withstand the SL-1 event with a high likelihood. In case of loss of water from the SCS due to an earthquake, the spill would be collected in the LOCA pool. Failure of the SCS piping would not affect the PCS piping.
16.86	16.11 Analysis of Loss of Coolant Events (LOCA)	The analysis is divided into two groups: • LOCAs in the PCS, • LOCAs in the reactor and pool service pools, the RSPCS and connected systems. In the case of the PCS the pipe size ranges from 15 mm to 500 mm in diameter. The piping is designed for SL-2, ie it is Seismic Class 1. Failures of the PCS piping outside the reactor block are considered within the design bases.	The PCS pipework is of robust design, with tough material (AISI 304), low pressure and temperature, design to SL-2 so a guillotine type rupture appears unlikely. Has vibration and fatigue damage to the smaller diameter pipes been considered? It would have been anticipated that all the PCS components were stainless steel. Please justify the choice of cast iron, a brittle material, for construction of check valves in the primary circuit.
			Response: The consequences of vibration and fatigue damage to the smaller diameter pipes are bounded by the LOCA scenarios presented in Section 16.11 of the PSAR. The use of cast iron for valve bodies and pump casings (as identified in Chapter 6, Section 6.9.2 for the PCS) is not unusual in the nuclear and process industries. The operating regime under which it will function will ensure that the potential for brittle fracture is very low.
16.87	16.11 -1Analysis of Loss of Coolant Events (LOCA) - PCS	There are four pipes that could initiate a large LOCA: • 500mm diameter outlet PCS pipe and the common line for the heat exchanger (HX) outlet.	Please discuss the possibility of pump seizure and the stresses arising from the dissipation of energy in the flywheels, as a mechanism for large LOCA. How is this prevented?

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		• 350 mm diameter inlet PCS lines, pump suction lines and inlet and outlet lies from the HXs, • 300mm dimeter discharge lines from the PCS, • 100mm Interconnecting System pipe.	
			Response: Please refer to the response to Question 16.75.
16.88	16.11.2.2 Decay Tank Break	The tank design pressure and temperature are the same as those for the PCS piping.	What material and design codes are used for the Decay Tank? Is it designed to Seismic Class1?
			Response: The material and design codes used for the PCS Decay Tank are identified in Chapter 6, Section 6.2.7.1.3 and Section 6.1.1 respectively. The PCS Decay Tank is classified Seismic Class 1 as identified in Chapter 6, Section 6.2.2.
16.89	16.11.2.4 Heat Exchanger Breakage.	The HXs are composed of several plates that are sandwiched by frame plates at the ends. Tightened bolts join both frames and compress the HX plates and its gaskets.	Can catastrophic failure of the flywheel cause a LOCA in the PCS and SCS simultaneously by a HX is damaged? If so, this is a single event that can cause both a LOCA and containment bypass. Has this been considered as Design Basis Accident? Please discuss the design of the PCS pump and flywheel in view of pump seizure and flywheel failure mechanisms. Please provide any detailed design information.
			Response: The flywheel and its design requirements are described in Chapter 6, Section 6.2.7.1.1. The flywheels are made of mild steel. The main characteristic of this material is its ductility, without voids and cracks that can be expected in cast iron. The procurement of the flywheel will include a strict quality control that ensures total absence of cracks, in the surface and the interior. In addition, the pumps are instrumented for vibration. There

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			are vibration switches (RCMS) in the pump flywheel that will stop the pump in case of very high vibration in the pump. In addition, several alarms will alert the operator in the Main Control Room on the malfunction of the flywheel. Given the quality of the flywheel and the alarms and vibration switches, catastrophic failure of the flywheel is considered very unlikely. Detailed design information will be provided to ARPANSA.
16.90	16.11.3 LOCA in the Reactor and Service Pool Cooling System	The openings at the base of the reactor pool for the CRs, fuel element clamps etc are connected to the Control Rod Drive Room (CRR). The integrity of this room is considered sufficient to ensure the pool level will only drop by 3.43 m (well above the core and at levels which permit Natural circulation)	If the CRR does not contain the water there is a potential to drain the core. The CRR thus becomes a key safety structure, and its penetrations such as doors, pipes and cables are key components. Is there a design document for this room?
			Response: The Control Rod Drive Room is Safety Category 1, Seismic Class 1 and Quality Level A. It will withstand the ingress of water that might be brought about by seal failure. A document will be prepared during the detail engineering phase.
16.91	16.11.3.2 Reactor and Service Pools Cooling System	LOCAs in the reactor and pool service pools, the RSPCS and connected systems can involve either pipe or valve failure. The pipe diameters range from 25 mm to 200mm diameter.	The PCS pipework is of robust design, with tough material (AISI 304), low pressure and temperature, design to SL-2. A guillotine type rupture appears unlikely. Has vibration and fatigue damage to the smaller diameter pipes been considered? It would have been anticipated that all the RSPCS components were stainless steel. Please justify the choice of cast iron, a brittle material, for construction of check valves in the primary circuit.

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			Please discuss the design of the RSPCS pump and flywheel in view of pump seizure and flywheel failure mechanisms. Please provide any detailed design information.
			Response: The potential for vibration and fatigue damage to the smaller diameter pipes will be considered during the detail engineering phase. The use of cast iron for valve bodies and pump casings (as described in Chapter 6, Section 6.9.2 for the RSPCS) is not unusual in the nuclear and process industries. The operating regime under which it will function will ensure that the potential for brittle fracture is very low. The issues associated with flywheel failure have been discussed above. Further design information will be provided in the FSAR.
16.92	16.11.3.3 RSPCS Decay Tank Break	The tank design pressure and temperature are the same as those for the RSPCS piping.	What material and design codes are used for the Decay Tank? Is it designed to Seismic Class1?
			Response: The material and design codes used for the RSPCS Decay Tank are identified in Chapter 6, Section 6.3.6.6 and Section 6.1.1 respectively. The PCS Decay Tank is classified Seismic Class 1 as identified in Chapter 6, Section 6.3.2.
16.93	16.11.3.6 Failure of Beam Tubes	The beam penetrations have been provided with two static barriers to prevent a loss of coolant, and these are designed as Safety Category I components. In view of the design quality, LOCAS through the beam tubes are considered beyond design bases.	There is a need to consider the significance of any LOCA via the beam tube since this has the potential to drain the reactor pool. Please discuss the performance of these penetrations in beyond design bases earthquakes and provide any design information.
			Response: Design provisions, multiple barriers, material quality, and QA manufacturing and installation procedures

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			will ensure that the potential for a LOCA via the beam tubes is so remote as to place it beyond the design basis. The performance of the beam tubes in beyond design basis earthquakes is being considered and will be reported in the FSAR. The main components to be analysed for a scenario of LOCA through the beams are the plates that seal the openings on the reactor block face. These plates are designed to withstand the pressure of the water column inside the pool. This load is larger than the load that these plates would be submitted to in case of a severe earthquake. Therefore, the plates would not fail even in the case of a severe earthquake. The aluminium windows have already been tested and it has been demonstrated that they can resist pressures well in excess of the maximum pressure expected in case the beam tubes were to be flooded.
16.94	16.11.4 Design Basis Postulated Initiating Events	Two DBIES are defined, namely: • failure of the PCS piping, and • failure of the RSPCS piping. There are a whole range of trip parameters in the FRPS and SRPS that will trip the reactor. The pipe size breaks are limited to the range 100mm to 250mm diameter.	Please explain why the larger pipe sizes in the PCS have not been considered as LOCA initiators?
			Response: A parametric study of pipe failure sizes was conducted, considering sizes up to 350 mm. The likelihood of failures of this size is considered very small, given the benign operating temperatures and pressures of the system. Larger failure sizes are not considered

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			credible. See also Chapter 16, Section 16.11.
16.95	16.11.4 Design Basis Postulated Initiating Events	In both the LOCA DBIEs the water level drops until the siphon break is effective. Natural circulation is also established. The numerical analysis has been performed using RETRANO2. The results are summarised in Table 16.11/12 and Figs 16.11/1 to 16.11/18. The clad temperature is well below any thermal hydraulic limiting criterion	In both DBIES analysed, the core remains covered in water provided the siphon breaker works. Under this situation the fuel can be cooled using natural circulation. Unlike the PSA the failure of the FSS, SSS, and Siphon Breakers are not considered, so operation of the Emergency Water Make Up system is not required. Please justify the differing treatment of LOCAs between the PSA and Section 16.11.
			Response: The treatment of LOCAs set out in Section 16.1 is deterministic, based upon the operation of the various protection systems at their minimal level of operation. This is standard practice, the likelihood of an initiating event coupled with failure of a safety system being considered sufficiently low as not to warrant concern.. The PSA adopted a probabilistic approach, looking at the likelihoods associated with the various additional failures as part of the determination of their importance to the residual risk of the facility. The two methods adopted, deterministic and probabilistic, complement each other.
16.96	16.14 Analysis of Special Internal Events	Internal events in the facility not arising from the reactor PIEs include: • internal fire or explosion • internal flooding • loss of supporting systems • security incidents • improper access to restricted areas. There are no high energy piping systems in facility, so pipe whip, jet impingement are not considered.	Has the failure and fragmentation of the PCS and RSPCS flywheels been considered?

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			Response: Yes, please see the responses to Questions 16.89 and 16.91.
16.97	16.14.1 Internal Fire or Explosion	Chs.4 and 10 outline the details of the building and fire protection systems. The different causes of fire considered are: • failure of the electrical system • flammable gases inside the reactor building • flammable liquids within the reactor building • accidents during maintenance or repair involving welding • build-up of deuterium from the reflector. Fire has the potential to affect safety systems. The triple redundant instrumentation is protected against fire in accordance with IEEE Category 1 Classification, with barriers, physical separation etc. No consequence analysis is performed for a fire, although it is considered a DBIE	Why has a consequence analysis for a fire not been undertaken? Is there any mechanism where fire can be a PIE and initiate a reactivity insertion , loss of flow or loss of heat sink event. Will a fire hazard assessment be undertaken for the reactor building to examine the safety of the reactor and its operators?
			Response: The fire assessment is limited to determination of the safe shutting down of the reactor. Consequences other than that are not considered within the scope of Chapter 16. The design of the plant systems is such that damage by fire will result in shutdown of the reactor. No potential for a fire to result in a reactivity insertion has been identified. A fire does have the potential to cause a loss of flow or a loss of heat sink. The cases analysed represent bounding cases of these types of events. A detailed fire hazard assessment will be undertaken during the detail design stage as part of examining the risks

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			to operators and property and confirming the effect on the safety of the reactor. This will be reported in the FSAR.
16.98	16.14.2 Internal Flooding	Any spillage of water will be drained through the LOCA sumps located at each building level. Flooding will be detected by detectors in the LOCA sumps and LOCA pool.	Please address flooding from operation of the fire fighting system not discussed Can flooding from this source effect the operation of any safety systems (such as FRPS and SRPS)?
			Response: All ESFs are protected against water spread by sprinklers. Those areas where the RPS electronics modules, FSS, SSS are located either do not have sprinklers, rather gaseous fire suppression systems, or are protected from sprinkler water. Therefore, the operation of the FSS, SSS, FRPS and SRPS will not be jeopardised by the actuation of the sprinkler system.
16.99	16.15 Reactor Utilisation Initiating Events	The analysis applies to the effect that the irradiation facilities, neutron beams, and the Cold Neutron Source have on the behaviour of the reactor. The systems/facilities considered are: • Bulk Production Irradiation Facilities • Pneumatic Transfer System and neutron activation analysis • Transfer, loading and Pneumatic Cells • Large Volume Irradiation facilities • Cold Neutron Source • Neutron Beam Facilities.	The reactor is planned to be heavily utilised, so the opportunity for initiating events associated with utilisation will be frequent.
			Response: Procedures and practices will be developed during the detail engineering phase and put into place during operation that minimise the challenges to the reactor protection systems by utilisation events.

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16.100	16.15.2 Bulk Production Irradiation Facilities.	A set of irradiation positions is provided for bulk targets which may generate considerable nuclear heat. Rigs with a reactivity worth greater than 200 pcm can only be unloaded and loaded during reactor shutdown, and are fixed during operation.	Will all irradiation rigs controlling greater than 200 pcm be fixed rigs?
			Response: Yes. This will be an operational requirement that will also be incorporated into the OLCs.
16.101	16.15.2 Bulk Production Irradiation Facilities.	The design criterion for the heat flux equates to 125 kW from each rig. The heat flux criterion has been based on ANSTO tests.	Please provide ARPANSA with the test results and analyses which support the heat flux criterion for these rigs?
			Response: The design criterion is based upon 125 kW per rig which itself is based on the Rique and Siboul correlation for the Onset of Nucleate Boiling. The ANSTO tests referred to are those associated with the design of a rig to be used for the irradiations and not to a modification of the heat flux criterion. The results of the irradiations will be made available.
16.102	16.15.2 Bulk Production Irradiation Facilities.	Pump failure with actuation of the FSS and SSS is considered design basis, but shaft seizure with failure of the FSS is considered to be beyond design basis.	Pump shaft seizure and failure of FSS is considered in Section 16.19. There are no consequences for this BDBA event. Please justify shaft seizure with failure of the FSS being considered as beyond the design basis.
			Response: The instrumentation associated with the pump flywheel and motor for both the PCS and RSPCS pumps are such that the onset of degradation will be readily detected long before flywheel failure. The likelihood of this being allowed to proceed to shaft seizure is considered very low. The additional likelihood of failure of the FRPS is considered to render the fault sequence beyond the design basis.

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16.103	16.15.2 Bulk Production Irradiation Facilities.	The potential introduction of a uranium metal target into an iridium irradiation position is of interest. Iridium is a neutron absorber, while the molybdenum targets introduce positive reactivity. Different geometry is used to prevent such exchanges.	Is it correct that molybdenum metal targets introduce positive reactivity? Would this mean that a failure during rig removal would inject positive reactivity of about 200 pcm in the order of a second?
			Response: The uranium metal targets will be enriched to around 20% U-235. This will add a small amount of positive reactivity to the core. An irradiation rig with a full charge would insert some 60 pcm of positive reactivity (see Table 5.7/15). Removal of a U-Mo rig from its irradiation facility will result in a reactivity decrease. Dropping the rig back into its irradiation facility would see the reactivity removal negated, returning the core to its original reactivity.
16.104	16.15.3 Pneumatic Transfer System and Neutron Activation analysis	The potential exists for: • target loading being greater than required, • neutron flux being greater than anticipated, • irradiation times being longer than foreseen. These events are prevented by administrative procedures.	The consequence of these events would not affect reactor safety, but could cause contamination within the reactor rigs and surrounds. In view of the number of such irradiations, have such events been anticipated?
			Response: Such events have been considered in the safety case. The design allows for several features to cope with these events, i.e. hot cells, filters in the pneumatic system, etc. Procedures and practices to be developed during the detail engineering phase and put in place during operation will ensure that the likelihood of such events is minimised. The consequences of the events are also relatively minor, being restricted to the pool and hot cells.

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16.105	16.15.3 Pneumatic Transfer System and Neutron Activation analysis	A nitrogen system continuously cools the targets. There are two blowers, with one on standby. Loss of flow leads to a reactor power setback.	Please provide further details on the nitrogen cooling system, in particular in relation to possible reactivity additions. Can a single failure in the cooling system lead to the fast insertion or extraction of multiple targets?
			Response: The nitrogen cooling system is described in Chapter 11, Section 11.4.2.1.2. Reactivity worth is analysed in Chapter 16 Section 16.8. Any failure associated with the circulation of nitrogen will result in very minor reactivity effects. The system is designed to extract one target at a time. A single failure could not result in the extraction of multiple targets.
16.106	16.15.6 Cold Neutron Source	The Cold Neutron Source (CNS) is described in Ch.11. On total failure of the helium cooling system the deuterium vaporises and if it exceeds a set value the reactor trips (FRPS).	What happens if the reactor does not trip on a loss of helium cooling? Can the pressure lead to explosive failure of the vessel or pipe work? Is there a relief valve or rupture disc?
			Response: On loss of Normal Operation Mode cooling, the CNS deuterium will vaporise and the CNS will switch automatically to Standby Operation Mode. In this CNS standby mode large portions of the normal helium system are bypassed, the cooling system is simplified and the reactor can still operate at full power with the CNS remaining in a safe condition. In the hypothetical case that the standby cooling doesn´t operate in an appropriate manner, the reactor will be tripped by the FRPS/FSS automatically.

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			If the reactor is not tripped and the standby cooling is not successful the D_2 vessel would heat up and the aluminium cell would fail. In this case no relevant pressure increase would occur because the volume of the D_2 buffer tank is 20,000 litre and the volume of the cell is 20 litre. The pressure would not lead to an explosive failure. The lack of cooling could cause damage of the moderator cell due to overheating. The D_2 will be contained by the thimble and the blanketing system. The core will not be threatened. Relief is provided continuously by the buffer tank, that is connected to the thermosyphon by means of a normal open Safety Category 1 valve and a rupture disc. No fire loads are present in the buffer tank room to prevent heating and pressure rise.
16.107	16.15.6 Cold Neutron Source	The CNS contains hydrogen and its isotopes. This can form explosive mixtures with air or oxygen and possibly detonate. Three levels of protection are provided: • Level 1 - Exclude the formation of any deuterium-oxygen mixture. • Level 2 - Prevent the mixture from reacting. • Level 3 - Prevent the reactor from being damaged by a detonation. The first level relies on pure deuterium and the inert blanket (helium) to prevent any contact with air. If a deuterium-oxygen mixture exists it is detected. The protection is provided against static	Please provide further information on the CNS safety analyses. Will this be provided in a separate PSAR for the CNS?

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO Response
		electricity, ignition control devices, and by the inert blanket.	
			Response: A separate safety submission will be provided to cover the design and operation of the CNS.
16.108	16.15.6 Cold Neutron Source - Prevent the reactor from being damaged by a detonation	A containment vessel is provided that can withstand the detonation of a stoichiometric mixture of deuterium and oxygen. The pressure reached in the containment vessel is in the range 1.6 MPa to 3.8 MPa.	What would be the effect of failure of the containment vessel on the reactor and the reflector vessel? Is there any experience of explosive failures of Cold Sources world-wide?
			Response: The design, construction and operation of the CNS is such as to render the likelihood of failure of the containment vessel beyond the design basis. As part of the design verification the vessel will be tested to a pressure that emulates the conditions to be attained in case of detonation. The Mk 1 Cold Neutron Facility at the HFBR at Brookhaven national laboratory did detonate. An air leak in the connections of the hydrogen supply bottle was not detected because a cold trap that the gas passed through absorbed the air for a time. Eventually the air broke through the trap and condensed on the cold surfaces of the moderator chamber, as well as on the fill and vent lines. When the vent valve was opened to release the hydrogen, it is likely that there was supersonic gas flow through the exhaust line (which had been partially constricted by frozen air). This flow could have stripped electrons off the flowing gas, building up a charge and causing the spark to ignite the gas mixture. This incident demonstrated the adequacy of a detonation proof vessel. No damage or adverse effects to the core have been reported.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO Response
			The report on the detonation is available to the designers and the lessons learned from this incident as well as the corrective actions have been incorporated into the design of the RRR CNS.
16.109	16.15.8 .1 Loss of Flow in the Reactor and Service Pools Cooling System	This event has been analysed assuming FSS failure (SSS success) and FSS success. The results are shown in Figs 16.5/1 to 16.5/7 for the loss of flow. In the case of the SSS actuation (Fig 16/15/7) the rig clad temperature reaches ONB for a few seconds.	There is information to suggest that there will be no negative reactivity insertion until a gas space is initiated in the Reflector Vessel. This could be 6 seconds after the SSS trip signal is initiated. Has the SSS assumed to be effective before 6 seconds?
			Response: Please see responses to Questions 16.45 and 16.82. The dynamics of the draining of the SSS have been obtained from a full scale mock up of the SSS that has been built and tested in Bariloche.
16.110	16.17 External Events	External events are site dependent, but also contain aspects that are design dependent. Information is taken from the HIFAR Probabilistic Safety Analysis (HIFAR PSA) and the Environmental Impact Statement (EIS) for the reactor facility.	While the EIS and the HIFAR PSA are relevant, the documents that are most germane are the Application for an RRR Siting Licence, and the ARPANSA SER dealing with the siting (See Ch.3 PSAR)
			Response: Comment noted.
16.111	16.17 External Events	External events have been screened in order to select those requiring detailed analysis. The screening criteria used are: • The event is of equal or lesser damage potential than those events for which the plant is designed. • The event has a significantly lower frequency of occurrence than other events with similar consequences	See previous ARPANSA review of screening in relation to the RRR PSA. It is important to include events that are bounded by others in order to capture their frequency into the total frequency of events within the dose ranges in Table 2 of the RAPs. Please confirm that frequencies of events that have been screened out have been captured in the total frequencies and state how this has been achieved.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO Response
		• The event cannot occur close enough to the facility to affect it • The event is included in the definition of another event. The screening of external events is shown in Table 16.7/1. Items classed as level 1 are included within design basis. These are: • Bushfires • Site Flooding • Extreme Winds • Lightning • Low winter Temperature • High Summer Temperature • Seismic activity • Aircraft Crash • Nearby Industry • On Site Activity • Water supply quality • Ventilation air quality • Road and Rail Accidents.	
			Response: Chapter 16 presents a deterministic analysis of the response of the facility to the event. Please see response to Question PSA.33.
16.112	16.17.1 Aircraft Crash	See Ch.4 for details of the grillage design to protect against a small aircraft crash onto the reactor building. The grillage is designed to absorb the impact without damaging the reactor or	It is argued that the defence in depth levels ensure that fuel is not damaged, although the containment might sustain some damage. Has the consequential loss of off-site and on-site electrical

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO Response
		reactor building. Aircraft fires represent the bounding load on the reactor building.	supplies as well as loss of the pond cooling been considered for this event?
			Response: The loss of all electrical supplies in addition to the aircraft impact has implicitly been considered. Electrical power is not required to safely shut down the reactor and maintain adequate core cooling.
16.113	16.17.2 Bushfires	Large bushfires can be expected every ten years. The heat load is not a direct threat to the reactor building, but nevertheless bush fire management strategies are in place (see items (a) to (i)).	Some services could be lost (electrical, air, and water) during a major bushfire. What is ANSTO's policy regarding operation of the reactor if there is a major bushfire?
			Response: In the event of a major bushfire, ANSTO would adopt the prudent policy of shutting down the reactor.
16.114	16.17.4 Military activities	The HIFAR PSA estimated the likelihood of a military shell as 1 in 10 million years. Thus the impact of a military shell is beyond design basis.	See comments on PSA.
			Response: See response to PSA questions.
16.115	16.17.5 Onsite Activities	On-site activities have the potential to affect the reactor facility through over-pressure following explosions, fires, generation of missiles and releases of toxic gases.	No direct threat to the reactor has been identified. Could any on-site situations lead to the loss of services, or the need to isolate the containment and shut down the reactor?
			Response: No specific events have been identified leading to the loss of site services or the need to isolate the containment. Instead, the events have been considered generically in Section 16.14.3 as losses of supporting systems and their effects considered. In all cases, the plant is fully capable of being safely shut down in a timely manner were such an events to occur.
16.116	16.17.8 Earthquakes	An SL-2 seismic event, which corresponds to a return period of 10,000 years, is used for the	It is claimed that the water tanks are designed to SL-2. The design adequacy of the Water Tower was raised in the RRR

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO Response
		design of safety systems, structures and components (SSC) (see Chapter 2 for list). The design details are given in Ch.4 of the PSAR and covers SSC that are designed to SL-2 and those that are designed to SL-1 (corresponding to a 1000 year return period).	Siting Licence and ARPANSA agreed that it would be appropriate to address this in the PSAR. What is the appropriate seismic design basis for the site water tower, which provides essential water supply for fire, reactor water make up and cooling water for the containment energy removal system? Please justify the chosen design basis.
			Response: No reliance is placed on the water supply from the tower for very large seismic events. Water tanks at the RRR facility will be qualified to withstand the SL-2 event. Additional water is available from the cooling tower pond.
16.117	16.17.8 Earthquakes	The analysis in Ch.4 of the PSAR indicates that the reactor building and structures can withstand higher accelerations than 0.3 g by a factor of around 2.	In view of the low core damage frequency (CDF) of order 10^{-7} to 10^{-8} per year calculated for internal sequences, why have beyond design basis earthquakes not been considered in Section 16.19? Will the consequences of such low frequency events be calculated?
			Response: The design has followed international best practice in that earthquakes down to a likelihood of 10^{-4} per year have been considered. In addition, events down to a likelihood of 10^{-5} per year have also been considered to ensure that there are no 'cliff edge' effects in the design. No consequence analyses will be performed for lower likelihood events. The damage and destruction brought about by the earthquakes to the surrounding population would render such calculations essentially meaningless. The low core damage frequency arising from internal events reflects the robust nature of the design in being able to shut down the reactor and keep the core and rigs cool.
16.118	16.18 Human Factors	The reactor design is aimed at minimising the impact of human failure in : • Operating Procedures	See review comments on the PSA and questions raised in relation to the quantification of design, maintenance and testing human errors.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO Response
		• Human-Machine interface • Maintenance • Refuelling • Target Handling • Design Basis accidents • Accident Management All protective actions are automatic, no operator action is required during the first 30 minutes following an initiating event. Further discussion of human error and its quantification is in the PSA.	Has consideration been given to a HAZOP analysis for sharp end operations such as refuelling, target handling and maintenance?
			Response: The use of such techniques as HAZOP and Task Analysis will be used as part of the development of a safe set of operating procedures. Such development will be carried out during the detail engineering phase.
16.119	16.19 Beyond Design Basis Accident	Four Beyond Design Basis Accidents (BDBA) have been investigated in order to define an accident to be used for emergency arrangements. These are: • Failure of a PCS Pump Shaft with failure of the FSS, • RSPCS pump shaft failure with failure to detect the loss of flow, • partial blockage of cooling channels in a fuel assembly, • erroneous early removal of a U-Mo rig into the hot cells, • total blackout for 10 days.	These BDBAs are selective. For example there is no consideration of: • CR driven out at low power • more than one CR driven out • leakage from CRR during LOCA from bottom penetrations • beyond design basis seismic events Please justify exclusion of the above and other sequences from the BDBA sequences analysed.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO Response
			Response: An analysis will be provided for low power transients. The other events are considered in the PSA except where the frequency was so low as to render them incredible.
16.120	16.19.5 Total Plant Blackout for 10 days	This sequence involves the loss of offsite and on-site power for 10 days. Decay heat would be removed by evaporation of the pool water, without the use of the RSPCS for 10 days. Calculations have been done using RETRANO2. On the basis of the evaporation from the pool the tank level drops by 3 m in depth. The Fabrega correlation has been used for estimating the maximum heat flux.	Please justify use of the Fabrega correlation. Is this correlation modelled in RETRANO2 for the natural circulation cooling mode? Is pulse boiling expected at any stage in this accident sequence?
			Response: The Fabrega correlation is well accepted for the determination of DNB in research reactors during natural circulation. This correlation is not programmed in RETRAN, the calculation has been done for the conditions of the core after shutdown and assuming no sub-cooling at the core outlet (very conservative assumption). Pulsed boiling is not expected at any stage of this transient. The pulsed boiling cooling mode is primarily relevant only to those accident sequences where: a) the velocity is low (below 0.5 m/s) b) the coolant reaches saturation c) no natural circulation to the pool is available. The coolant does not reach saturation in this or any other transient presented in the PSAR. Failure to establish natural circulation is considered incredible since it assumes failure of four flap valves.

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO Response
16.121	16.20 Probabilistic Safety analysis Objectives	The PSAR Ch.16 is basically a deterministic safety analysis, and events not considered credible have been screened out. The remaining events are considered as DBIEs. The PSA looks at both DBIES and beyond design basis events.	See review comments on the PSA and in particular the findings on the Core Damage Frequency (CDF) estimates.
			Response: Comments noted. Responses are provided in the PSA section.