| **Question Reference** | Section number and name | **Topic for clarification** | **ARPANSA Comment, Issue or Question** |
|---|---|---|---|
| 5.1. | 5.1 (p.5.1-3)<br>Summary Description | Core critical point | At what thermal power is the reactor deemed to be 'critical'? |
| | | | Response: ANSTO understands the question to mean; at what fission thermal power is the reactor deemed to be 'critical' from the operational rather than the theoretical point of view. Reactor operation procedures that will be developed will specify the power where critical 'balances' would be performed. These features will be reported in the Final SAR. |
| 5.2. | 5.1 (p.5.1-3)<br>Summary Description | Burnup | In the table no burnup limit is specified. Given that U3Si2 fuel has been used for some years, working safe burnup limits based on experience must have been established for other reactors using this fuel, e.g. assuming the ETRR-2 and Peruvian reactors use U3Si2 fuel, what burnup limits apply to them? |
| | | | Response: The design of each reactor core, fuel plate, coolant channel geometry, coolant velocity etc will all contribute to the determination of a burn-up limit. It is stated in NUREG 1313 ( ANL/ RERTR/TM 10 ) that miniplates, plates and full scale fuel assemblies were irradiated up to very high peak burn-ups ( 80 to 96% ); far above that expected for normal reactor operation. There were no indications that the use of silicide fuel under these very stringent conditions might be precluded, so the limits to burn-up that can be imposed will mainly arise from Operational Conditions.<br>The necessity to establish such a limit will be further analysed during the Detail Engineering phase and will be reported in the Final SAR.<br>ETTR-2 and RP-10, the Peruvian reactor, do not use silicide fuel. |
| 5.3. | 5.1 (p.5.1-4) Summary Description | Rig reactivity | In the table, what is basis of the reactivity limit for 'non-fixed rigs'? |
| | | | Response: The reactivity limit for 'non-fixed rigs' is a requirement of the Argentine Nuclear Regulatory Authority. |
| 5.4. | 5.1 (p.5.1-4) Summary Description | CR reactivity | In the table, what is meant by 'Average reactivity insertion rate for CRs during operation'? Is it referring to automatic power control during at-power rig movements? |

| **Question Reference** | Section number and name | **Topic for clarification** | **ARPANSA Comment, Issue or Question** |
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| | | | Response: The "average reactivity insertion rate for CRs during operation" is the reactivity worth of the CR in question divided by the shortest time required to raise that CR from the fully inserted position up to the fully out position. It is a design quantity to determine the maximum speed of the CR drive in raising a CR plate. |
| 5.5. | 5.1 (p.5.1-4) Summary Description | Maximum power | In the table, the maximum reactor power is given as 31.3 MW corresponding to 'flow redistribution instability'. The OL&C need to specify a maximum reactor power which includes an appropriate safety margin. Please explain what is meant by "maximum reactor power"? |
| | | | Response: The referred sentence: "This has been adopted to allow adequate margin to the maximum reactor power (31.3 MW)", should read: "This has been adopted to allow adequate margin to the safety limit on the reactor power (31.3 MW)". This will be changed in the next revision of the PSAR. |
| 5.6. | 5.2.4.4.2 (p.5.2-4) Reactor Structures – crgb Description | Hydraulic forces on CRs | States that the CR cooling flow upthrust within the CRGB does not exceed the weight of the CR. However, if the upthrust is significant compared to the CR weight there is a reduced net force to cause the CRs to fall under gravity at full coolant flow. During a trip the coolant pumps lose power and coast down due to the fly wheels. Does this coast down affect the CR insertion time by gravity particularly in the accident case where the compressed air assist of CR insertion fails? What is the affect of starting a third primary coolant pump? |

| **Question Reference** | Section number and name | **Topic for clarification** | **ARPANSA Comment, Issue or Question** |
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| | | | Response: The velocity of the CR coolant flow is reduced during the coast down of the coolant pumps after a loss of power. This reduces the drag force on the CR. This condition is therefore bounded by a trip without loss of power when the first coolant pump is not turned off until after the reactor is shut down while the second pump is not turned off until 30 minutes after shutdown.<br>The lift force on the control plates is estimated to be between 22% and 32% of the control plate weight during normal operation.<br>The coast-down affects the CR insertion time in that the CR insertion time is shorter than that in case of no pump trip.<br>An interlock (See Chapter 6, Section 6.2.9.3.1) prevents the third pump from starting when two pumps are already in operation. Even when two pumps are in operation and one fails, the third pump cannot start so there is no situation where two pumps are in operation and the third is coasting down. |
| 5.7. | 5.2.4.5.2 (p.5.2-5)<br>Reactor Structures – FC Description | FA clamping | States in a table that the 'Maximum clamping force' is 1,960 N. What is meant by this as it does not appear to exceed the estimated drag force by a very large margin (only 11%, whereas a factor of 2 or 3 may be more appropriate)? Is the estimate of the drag force on a FA pessimistic, taking into account the imprecision of contributing parameters? Does the 'Maximum force that the FC can withstand' include the strength of the fastening to the FA? |
| | | | Response: The "maximum clamping force" is the force exerted by the clamp in the contact direction between the component and the FA fastener rod that it clamps to. The maximum force that the FC can withstand is the force at which the clamp would fail. Therefore a safety margin of a factor of 10 applies. |
| 5.8. | 5.2.5.1 (p.5.2-12)<br>Reactor Structures – Manufacturing and Installation | Core components performance testing | Will the Inlet Plenum and Core Chimney performance during Emergency Make-up Water System actuation be determined using a mock up at the manufacturer's works or after installation during commissioning? |
| | | | Response: The performance test will be carried out during commissioning. |
| 5.9. | 5.3.1.1 (p.5.3-1) | Cd burnable poison | Are the Cd burnable poison wires incorporated in the fuel plate or attached externally? |

| **Question Reference** | Section number and name | **Topic for clarification** | **ARPANSA Comment, Issue or Question** |
|---|---|---|---|
| | Fuel Assemblies – General Description | | If attached, how are they attached? Is there a significant corrosion problem having Cd metal in contact with the coolant? If incorporated into the fuel plate, how is installation verified? |
| | | | Response: Figure 5.3/3 shows the Cd wires as being separate from the fuel plate. As stated in Section 5.3.3.2.8, the Cd wires are located in the side plate grooves between the side and fuel plates.<br>Details on the Cd fixation method will be provided during detail engineering. They are mechanically fixed in place by swagging with their final position being verified by radiography.<br>The international experience shows that cadmium wires can be used with or without a clad, without important corrosion problems. A decision on this point will be taken during the detail engineering phase. |
| 5.10. | 5.3.2.1 (p.5.3-3)<br>Fuel Assemblies – Design Conditions | Fuel plate capability | What are the ‘general functional capabilities’ of the fuel plate stated in (c)? |
| | | | Response: General functional capabilities are described in Section 5.3.2.1 and are, for instance; that it should provide an adequate geometric and dimensional stability to ensure FA cooling and heat transfer; that it should withstand the action of hydrodynamic loads without major deformations and that it should withstand chemical, thermal, mechanical and irradiation effects without loosing its tightness.<br>Fulfillment of the limits stated in the Table on page 5.3 – 4 (Section 5.3.2.2) ensures that general functional capabilities are maintained. |
| 5.11. | 5.3.2.1 (p.5.3-4)<br>Fuel Assemblies – Design Conditions | Burnup | There is no design limit for fuel burnup, but burnup equivalent to 58% of 235U is mentioned. Will there be an OL&C for burnup and is it expected to be 58% or greater and will maximum permitted burnup included in the safety case? |

| **Question Reference** | Section number and name | **Topic for clarification** | **ARPANSA Comment, Issue or Question** |
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| | | | Response: The burnup of 58% of U-235 is given as an estimate of the highest attainable local burnup in normal operating conditions.<br>A burnup limit will be determined in the Detail Engineering Phase (see response to Question 5.2). |
| 5.12. | 5.3.2.2 (p.5.3-4)<br>Fuel Assemblies – Design Limits for Reactor Normal Operation | Coolant chemistry | In the table, design criteria/limits for coolant chemistry indicate 'pH neutral'. Is this intended to mean pH < 7 or pH > 7, or will a pH range be specified (as for HIFAR where a pD operating range is specified)? |
| | | | Response: The term neutral has been used to indicate a broad similarity in the concentrations of $H^+$ and $OH^-$ ions. This means that the pH will be close to 7 when the coolant is at a temperature of 25C. The operating range will be specified in the Final SAR. |
| 5.13. | 5.3.2.4 (p.5.3-6)<br>Fuel Assemblies – Heat Generation Effects | Fuel plate temperature | Maximum fuel plate surface temperature is given as 110ºC. What is the minimum saturation temperature of the coolant in contact with the top edge of an FA, i.e. the point along fuel element length where static pressure is the least? |
| | | | Response: The 110ºC given here is not the maximum fuel plate surface temperature; it is a limit placed on the fuel plate surface temperature.<br>The minimum saturation temperature at the downstream end of the meat is 120.0ºC. This is slightly lower than that at the top edge of the fuel plate because of an expansion-induced pressure recovery at the end of the fuel plates. |
| 5.14. | 5.3.2.4.1 (p.5.3-6)<br>Fuel Assemblies – Maximum Fuel Meat Temperature | Cd burnable poison | It is not clear what is meant by this para. Is the correct location of the Cd wire burnable poison included in the criteria for 'fuel integrity'? When the fuel meat temperature reaches 400ºC during a transient what is the maximum temperature of the Cd wire? |

| Question Reference | Section number and name | Topic for clarification | ARPANSA Comment, Issue or Question |
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| | | | Response: The Cd wire is not part of the fuel plate and therefore plays no part in the criteria for fuel integrity.<br>The Cd wire temperature has not been calculated during a reactor transient. In the very unlikely event that the maximum fuel meat temperature was to reach 400C during a transient the Cd wire temperature would be expected to be significantly less than the maximum fuel meat temperature.<br>The location of the Cd wires is addressed in the response to Question 5.9. |
| 5.15. | 5.3.2.4.2 (p.5.3-6)<br>Fuel Assemblies – Thermal Expansion and Stress | Coolant channel width | Does the minimum coolant channel width, of 90% of the nominal figure, have a basis, i.e. relative contributions of thermal expansion, irradiation swelling, growth of $Al_2O_3$ layer? What are the dimensional tolerances on the FAs for coolant channel width? |
| | | | Response: The minimum channel width is a nominal conservatively low value after allowing for all the contributions noted in the question.<br>Thermal expansion (Section 5.3.4.3.1) is no more than 0.003mm. No expansion expected from irradiation swelling (Section 5.3.2.6.1). Oxide layer thickness is expected to be the order of tens of microns at most.<br>The Table on page 5.3-4 indicates that the limit on the minimum coolant channel width is 90% of nominal value of 2.45 mm, ie the maximum reduction in channel width of 0.245 mm. This reduction is well in excess of the reduction due to the three processes above.<br>As stated in Section 5.3.3.1, dimensional tolerances will be defined in the Detail Engineering Phase. |
| 5.16. | 5.3.2.5 (p.5.3-7)<br>Fuel Assemblies – Hydraulic and Mechanical Loads | Coolant velocity | With regard to the margin between the best-estimate coolant velocity and the maximum velocity, what is the basis for the ‘uncertainties (typically 10%)’? |

| Question Reference | Section number and name | Topic for clarification | ARPANSA Comment, Issue or Question |
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| | | | Response: Given that there is a 10% tolerance in the coolant channel width, an approximation to the uncertainties to the coolant velocity is less than 10%.<br>Experimental data and previous experience on similar geometries and dimensions support this value. |
| 5.17. | 5.3.2.5 (p.5.3-8)<br>Fuel Assemblies – Hydraulic and Mechanical Loads | Fuel mechanical design | With regard to 'mechanical design protective actions', what is meant by (b)? What are the contact surfaces where excessive stresses could be developed? |
| | | | Response: Item (b) refers to the maximum force on the FA nozzle with which the end of the FC fastener rod contacts. This force is specified as 2000N in the table on page 5.3-4. It is the limit applied to prevent damage to the FA nozzle by the fastener rod. |
| 5.18. | 5.3.2.6.1 (p.5.3-8)<br>Fuel Assemblies – Swelling | Burnup | There are sufficient parameters mentioned to define a maximum burnup. What is the design maximum fuel burnup (to be part of the OL&C)? |
| | | | Response: The burnup limit will be determined in the Detail Engineering Phase See responses to Questions 5.2 and 5.11. |
| 5.19. | 5.3.2.6.3 (p.5.3-9)<br>Fuel Assemblies – Other Irradiation Effects | Burnup | How would fuel burnup be limited by (a), (b) and (a) + (b)? |
| | | | Response: These effects were taken into account in the fuel analysis presented in Section 5.3.4<br>a) It does not pose a limit because the increase in meat temperature is very low due to the lower thermal conductivity of the meat (see Section 5.3.4.2.5)<br>b) The yield stress increase of Al6061 due to irradiation acts in a positive conservative sense regarding mechanical properties of the fuel |

| **Question Reference** | Section number and name | **Topic for clarification** | **ARPANSA Comment, Issue or Question** |
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| 5.20. | 5.3.2.7 (p.5.3-9)<br>Fuel Assemblies – Chemical Environment Effects | Coolant chemistry | What are 'proper water conditions' to minimise Al corrosion, i.e. most importantly pH range (could be mentioned here as well as in Chap. 6)? |
| | | | Response: Coolant limits are defined in the table on page 5.3-4:<br>a) demineralised water. The level of non-dissolved solids: is given in Chapter 6, Section 6.4.5.1 as less than 0.5 ppm.<br>b) electric conductivity lower than 1$\mu$Scm$^{-1}$,<br>c) pH: Neutral (see response to Question 5.12)<br>Of the factors listed above electric conductivity is the most important factor in the prevention of pitting corrosion. An adequate conductivity value ensures that the pH value is within the limits. |
| 5.21. | 5.3.3.1 (p.5.3-10)<br>Fuel Assemblies – General Assembly | FA mechanical design | The side plates and the outer fuel plates are fixed to the end box by screws. What means are provided for preventing the screws becoming loose? Refer also to comment on 5.3.2.5 (p.5.3-8) and 5.3.3.2.6 (p.5.3-14). |
| | | | Response: The three screws used to hold the handling pin are secured by a weld spot. The screws used to hold the side/external fuel plates/end box together are locked in position by a centre punch at the edge of the screw head. See also response to Question 5.24. |
| 5.22. | 5.3.3.2.1 (p.5.3-11)<br>Fuel Assemblies – Fuel Plates | Fuel plate U loading | What is the reason for having two different FA 235U loadings for the first core and where are the two types situated in the core array (i.e. such information should be given here, or, a reference as to where it can be found in PSAR)? |
| | | | Response: Having two different FA U-235 loadings reduces the U-235 loading in the first and subsequent cores so as to limit the reactivity in the run up to the equilibrium core. Information on the first core is located in Section 5.7.5.5.1 rather than in Section 5.3 because it belongs to nuclear design (Section 5.7). |
| 5.23. | 5.3.3.2.2 (p.5.3-13)<br>Fuel Assemblies – End | FA mechanical design | What is meant by 'a perimetrical zone resting over the grid'? |

| **Question Reference** | Section number and name | **Topic for clarification** | **ARPANSA Comment, Issue or Question** |
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| | Box | | |
| | | | Response: It is a zone on the end box that extends around the perimeter of the FA such that when the FA is inserted into the core, it lies above the grid over which the FA positions are defined. |
| 5.24. | 5.3.3.2.6 (p.5.3-14)<br>Fuel Assemblies – Comb | FA mechanical design | The comb is fixed to the outer fuel plates by screws. What means are provided for preventing the screws becoming loose? Refer also to comment on 5.3.2.5 (p.5.3-8) and 5.3.3.1 (p.5.3-10). |
| | | | Response: The weld spot secures the screws. See response to Question 5.21. |
| 5.25. | 5.3.2.6.1 Swelling | ‘the percentage of porosity in the meat is 7% while the change in the volume of particles due to swelling is 6% at 50% burnup’ | What is the accuracy of burnup calculations?<br>What is the maximum local percentage of burnup (in a ‘hot’ spot)?<br>Does 7% provide enough margin of porosity taking into account the above questions? |
| | | | Response: The accuracy of burnup calculations is estimated to be around 10%.<br>The maximum local burnup is estimated to be 58% (see response to Question 5.11).<br>The comparison of a swelling rate of 6% at 50% burnup with 7% porosity in the fuel meat is a statement used to show that the calculation from DART makes sense. The DART code provides a more accurate estimate than a simple comparison between the swelling rate and porosity. |
| 5.26. | 5.6.2 Neutron Reflector | Reflector isotopic purity (Heavy water content) | Please provide calculational results showing degradation of D2O, preferably as a graph. How often will the D2O have to be replaced to maintain required purity? |
| | | | Response: The main process that will contribute to the degradation of $D_2O$ is expected to be leakage of $H_2O$ into the reflector tank.<br>Acceptable $D_2O$ purity levels are discussed in Sections 5.6.2 and 5.7.5.5.3. It is estimated that the $D_2O$ isotopic purity limit will be reached after 10 years of operation. |
| 5.27. | 5.7.4.1 General Design | ‘Should the reactivity of | Is it reasonable to employ ‘remaining plates’ for regulation? When the remaining |

| Question Reference | Section number and name | Topic for clarification | ARPANSA Comment, Issue or Question |
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| | bases | the safety and regulating plate prove insufficient, assistance will be provided by the remaining plates to compensate changes in the core reactivity throughout the operating cycle’ | plates are partially inserted in the core to compensate for reactivity changes during an operating cycle, how much reactivity is associated with this insertion?<br>Is it a proven approach used in other similar types of research reactors? |
| | | | Response: The use of control rods for both reactivity regulation and shutdown purposes is a design recognised and accepted throughout the world and in the IAEA Safety Series 35-S1.<br>From the data provided in Table 5.7/2 and Table 5.7/8 it can be seen that the four lateral plates together can insert approximately 5000 pcm at BOC.<br>The approach has been used in other research reactors, for example HIFAR. |
| 5.28. | 5.7.4.3 Thermal-hydraulics related Design bases | ‘The power peaking factor shall be lower than 3’ | Is this power peaking factor high compared to similar types of research reactors? Please explain. What uncertainties, including engineering and manufacturing uncertainties, are taken into account in calculating the power peaking factor? |

| **Question Reference** | Section number and name | **Topic for clarification** | **ARPANSA Comment, Issue or Question** |
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| | | | Response: The power peaking factor (PPF) is calculated to be 2.1 (Section 5.4.5.2). This value is similar to that for other compact core research reactors as indicated in Table 1.4/1 in Chapter 1. The limit adopted is 3 and this is consistent with all the thermal-hydraulic limits, see Section 5.8.<br>Systematic parametric analyses were done on the effects of different parameters on the PPF, including geometrical, Uranium-235 loading and burnable poison loading tolerances. As these parameters exhibit a statistical distribution over the design value, the composition of the effects is small on the PPF being the perturbations of 1% for the geometrical tolerances, 0.9% for fuel and burnable poison loading. Additional calculations were carried out in order to compute the perturbations for the cases of BOC and EOC condition (Section 5.7.5.2.1), CNS change of state (Section 5.7.5.2.2), presence (or not) of the fixed irradiation rigs (Section 5.7.5.2.2), and anticipated operational occurrences (Section 5.7.5.5.4). In all cases it was found that the perturbations are small and the values of the PPF remains well below the limit of 3.<br>It should also be noted that uncertainties (arising from both calculation and measurement) on the PPF are taken into account in the thermal and hydraulic design. |
| 5.29. | 5.7.4.7 Rate of Insertion of Reactivity Design Bases | 'FSS should insert a negative reactivity of 2000 pcm in less than 500 msec'<br>'SSS should insert a negative reactivity of 3000 pcm in less than 15 sec' | What is the basis for the 2000 pcm and 3000 pcm insertions design requirement while the FSS and SSS worth is much more, namely about 18000 pcm and 12000 pcm, respectively? Why is there a different shutdown margin for the FSS and SSS? Table 5.7/12 |

| Question Reference | Section number and name | Topic for clarification | ARPANSA Comment, Issue or Question |
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| | | | Response: Design bases for the FSS are consistent with the requirements of the Argentine Regulatory Authority (see response to Question 5.3).<br>For the case of the SSS they are realistic estimates of what can be achieved and will fulfil all the safety requirements (including accident conditions). Notice that the shutdown margin required for the SSS is equivalent to that required for the FSS with single failure (SM-1).<br>Not all of the reactivity worth of the FSS is available for insertion to shutdown the reactor. Some of the reactivity is required to balance the excess reactivity of the core and maintain the core critical. Initially sufficient reactivity should be inserted to stop any reactivity transient. In the long term there should be sufficient reactivity to maintain the reactor shutdown indefinitely.<br>The appropriate reference is Table 5.7/18 in which the various shutdown margins are given (SM1SS > 3000 pcm and SM2SS > 1000 pcm). |
| 5.30. | 5.7.5 Description | 'The bare core, without reflector and with all control rod plates removed, is sub-critical so the presence of the reflector is necessary to maintain the nuclear chain reaction' | What is the Keff of the bare core without reflector and with all control rod plates removed? |
| | | | Response: This value will of course depend on the core state being considered. Taking information from the PSAR – The reactivity worth of the SSS when drained is 11840 pcm (Table 5.7/12), the greatest excess reactivity core appears to be BOC, cold, No Xe, CR out, at 8270 pcm (Table 5.7/14). The reactivity difference would indicate a keff of ~0.96, under these conditions.<br>Note that the actuation of the SSS results in drainage of the reflector vessel down to the core mid level |
| 5.31. | 5.7.5.1 Nuclear Design Description, page 5.7- | 'Analyses carried out of the neutronics parameters | When is SAR Rev A scheduled for completion? Please provide the stated analyses undertaken to date for ARPANSA detailed review. |

| Question Reference | Section number and name | Topic for clarification | ARPANSA Comment, Issue or Question |
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| | 7 | for the pseudo equilibrium cores demonstrate compliance with the applicable design bases. Such detailed analyses for the anticipated strategy will be provided in the SAR Rev. A' | |
| | | | Response: Please note that SAR Rev A stands for the draft of the Final SAR.<br>This information will be presented with the commissioning program and procedures. ARPANSA will be informed of any analysis undertaken to date. |
| 5.32. | 5.7.5.2 Neutron Flux and Power distribution | 'The margin between the most stringent PPF, 2.039 at BOC, and the limit specified by criterion 6 (PPF<3) is 32%' | What is your total uncertainty in PPF calculations? Can you be sure about 4 significant digits? Did you conduct perturbation calculations to estimate the uncertainty of PPF associated with nuclear data uncertainty? See Section 5.7.4.3. Will the calculated PPF be verified by measurement during commissioning of the reactor? |
| | | | Response: Uncertainty in nuclear data has a negligible contribution to uncertainty in PPF since PPF is a relative quantity. The calculated PPF will be verified during commissioning tests. See also response to Question 5.28. |
| 5.33. | 5.7.5.2 Neutron Flux and Power distribution.<br>Figures 5.7/5-1 – 5.7/9-3 | Neutron fluxes at BOC and EOC | Please provide information on neutron fluxes along most rated FA, at the top, bottom and midplane should be provided both for BOC and EOC in graphical form. |
| | | | Response: Information is provided in Appendix 1. |
| 5.34. | 5.7.5.2.3 Neutron Flux and Power Distribution Anomalies | 'The perturbations on the neutron flux and power distributions, including the perturbation on the PPF, have been estimated | What is meant by 'estimated'? Please provide details of calculations.<br>Please amend "CR plates" to either "CR" or "control plates" to be consistent throughout the PSAR. |

| Question Reference | Section number and name | Topic for clarification | ARPANSA Comment, Issue or Question |
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| | | for the movement of the CR plates along the cycle’ | |
| | | | Response: The effect of loading and unloading rigs and of CR movements is ‘estimated’ by calculating the neutron flux and PPF for the various configurations and core states and taking the difference. In this way the perturbations resulting from the change can be calculated.<br>The issue of “CR” and “control plates” will be addressed in the next revision of the PSAR. |
| 5.35. | 5.7.5.4.2 Reactivity Variations | ‘During power operation the four lateral CR plates will be introduced into the core only a small percentage of their length’ | A figure should be provided together with a table showing the intended configuration. The percentage of insertion must be clearly specified for BOC for all five “CR plates”. |
| | | | Response: The preliminary CR strategy for start up would be (extraction percentages are indicated, perturbation on PPF is referred to the 2.1 value):<br>Shutdown: all CR 0%.<br>Cold, Xenon free: S1 35%, S2 35%, S3 82%, S4 82% and S5 0%. Perturbation on PPF is –0.14%<br>Hot, Xenon free: S1 45%, S2 45%, S3 82%, S4 82% and S5 0%. Perturbation on PFF is +4.17%.<br>Hot, Xenon in equilibrium: S1, S2, S3 and S4 88%, S5 0%. Perturbation on the PPF 0% (reference state). |
| 5.36. | 5.7.5.5.3 Heavy Water Degradation<br>Table 5/7/16 | Heavy water degradation | At EOC the relative negative reactivity effect of $D_2O$ degradation will be larger than that one for BOC. Therefore, the effect should be calculated as well for EOC. It cannot be concluded that ‘negative reactivity variations can be accommodated by the reactivity excess of the core and regulated by the CR plates’ unless the effect for EOC is calculated. A graph should be provided showing the effect of $D_2O$ degradation with time, both for BOC and EOC. |

| Question Reference | Section number and name | Topic for clarification | ARPANSA Comment, Issue or Question |
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| | | | Response: Table 5.7/16 indicates that the change in reactivity due to the degradation of $D_2O$ is only about 200 pcm based on a BOC model of the core. The effect for an EOC model is expected to be similar.<br>The degradation of the $D_2O$ with time depends on the leak rate of $H_2O$ into the $D_2O$ tank. Please also see response to Question 5.26. |
| 5.37. | 5.7.5.7.1 Xenon Transients | Xenon transients | The graph of xenon build-up in the case of the reactor trip at full power should be provided in the PSAR. |
| | | | Response: Reactivity worth of Xenon is reported in the PSAR (see Table 5.7/2). The Xenon build up will be calculated during the Detail Engineering phase and will be reported in the Final SAR. |
| 5.38. | 5.7.6.1 Calculation Methodologies<br>5.7.6.6.1 Nuclear Data Library | Calculation of uncertainties associated with nuclear data | Please explain what perturbation calculations have been conducted to estimate neutronics calculational uncertainties associated with nuclear data. |
| | | | Response: It is not usual to perform perturbation calculations for nuclear data in order to estimate uncertainties in the required values. The approach adopted is to benchmark the code and data by comparing calculations for similar systems with measurements. The difference between calculated and measured values is an estimate of the uncertainty. In addition, two different sources of nuclear data are used in some calculations in order to ensure that no undue errors are introduced by the nuclear data. Nuclear data sources used are indicated in Section 5.7.6.6.1. |
| 5.39. | 5.7.7 Design Evaluation | 'The neutronic design of the reactor Facility has been performed by following all applicable codes and standards and applying international practices' | Please explain what is meant by "applying international practices" in relation to conducting a full examination of calculational uncertainties of neutronics design? |

| Question Reference | Section number and name | Topic for clarification | ARPANSA Comment, Issue or Question |
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| | | | Response: ANSTO considers 'international best practice' the use of validated and benchmarked codes and neutron cross section libraries. These must be validated and benchmarked using problems relevant to the calculations being performed for the design of the RRR. |
| 5.40. | Table 5.7/11 | Abbreviations | What is meant by 'SM1SS' and 'SM-1' ? It should be clearly explained in the footnotes of the Table. |
| | | | Response: SM1SS and SM-1 are defined in Section 5.7.4.2. |
| 5.41. | 5.8.3.2 Power Peaking Factor Related Design Basis | Power peaking factor estimations | Please comment on discrepancies in estimation and actual measurements (after commissioning) of the Egyptian Reactor. Was the same methodology used for PPF estimation this time? Does the PPF uncertainty take into consideration nuclear data uncertainty? See also 5.7.4.3. |
| | | | Response: The measured power peaking factor for ETRR-2 was 2.42 and the calculated value was 2.72. This represents an overestimate of about 12%. This figure reflects the calculation and measurement uncertainties and is consistent with internationally reported values. |
| 5.42. | 5.8.3.6.1-5.8.3.6.3 | Terminology 'statistical analysis of uncertainty factors' | Needs to be clarified. In particular, what sort of 'uncertainty' is being referred to and how were they treated? |
| | | | Response: The statistical treatment of uncertainties is discussed in Section 5.8.5. As noted there, the treatment followed is a recommended approach. |
| 5.43. | 5.8.5 Uncertainties in Estimates | Calculational uncertainties | Lack of uncertainty of nuclear data analysis may effect thermal hydraulic uncertainties. Refer to 5.7.6.1 |
| | | | Response: The main effect of neutronic calculations on thermal and hydraulic performance is related to the PPF. This effect is taken in consideration by the conservative margin given by the PPF adopted to perform the thermal and hydraulic design. |
| 5.44. | Table 5.8/2 | Calculational uncertainties | Please explain what is meant by a 4% uncertainty of "change in atmospheric pressure"?<br>Uncertainties in heat transfer correlations of 25% should be justified. |

| Question Reference | Section number and name | Topic for clarification | ARPANSA Comment, Issue or Question |
|---|---|---|---|
| | | | Response: Barometric pressure variations are discussed in Chapter 3, Section 3.2.3.2. A 4% uncertainty is conservatively considerably beyond the maximum variation over a 5-year period.<br>For justification of uncertainties in heat transfer correlations, please refer to "Handbook of Single-Phase Convective Heat Transfer", Sdik Kakaç, Ramesh K. Shah and Win Aung, 1987, John Wiley & Sons, Inc.<br>The uncertainty for the heat transfer correlations is in practice much less than 25%. |
| 5.45. | 5.8.6 Thermal and Hydraulic Analysis of the Core for the Power State | 1900 $m^3s^{-1}$ | Is it a misprint ? Should read 1900 $m^3h^{-1}$ |
| | | | Response: Agreed. This will be corrected in the next revision of the PSAR |
| 5.46. | 5.8.11.1.1 Allowable Velocity | Strange unfinished piece of text at end of 2nd paragraph. | Please explain what is meant by 'Estimation of the Pressure Drop Distribution'. |
| | | | Response: Agreed. Already identified by an erratum |
| 5.47. | 5.8.12.2.1 Thermal Analysis | RSPCS | What is RSCPS? |
| | | | Response: This should be RSPCS. This will be amended in the next revision of the PSAR. |
| 5.48. | 5.9.2.1 Irradiation Damage | Irradiation damage | 1) The statement that the reactor facility components are expected to see a lifetime fluence of approximately $1x10^{23}$ n.cm-2 need to be verified. According to neutron fluxes supplied in Figure 5.7/3-5, some components may be exposed over 40 years time to much larger neutron fluence.<br>2) dpa should be calculated and compared with established criteria.<br>3) Irradiation damage from photons should be estimated.<br>4) The statements 'radiation-induced creep will be negligible' and 'irradiation-induced changes in the fracture toughness are not expected to be of significance' |

| **Question Reference** | Section number and name | **Topic for clarification** | **ARPANSA Comment, Issue or Question** |
|---|---|---|---|
| | | | should be quantified and compared with similar modern research reactors and experience.<br>5) The statement 'the design takes into account these effects and provides sufficient safety margins to ensure adequate component performance' should be justified. Safety margins should be listed in a table together with calculated data and compared with similar modern research reactor data available up to date. |
| | | | Response:<br>1) This section is for the stainless steel components. These will not experience high fluences due either to: i) being replaced after a period (eg control rod cladding), or ii) are a significant distance from the core (eg, the inlet plenum, upper chimney etc). The value of 1x$10^{23}$ n/$cm^2$ as a local maximum for the fluence is confirmed.<br>2) Not relevant for the stainless steels components<br>3) Not relevant for the stainless steels components<br>4) Not relevant for the stainless steel components. Note that the component that will experience the maximum fluence will be the control plate cladding (2 x $10^{22}$ n/$cm^2$ in 10 years) but it is anticipated that this component will be replaced every 8 years. For other stainless steel components the swelling/creep phenomenon will not be present, as neither the temperature nor the fluence will be high enough to activate this phenomenon. See also Appendix 2.<br>5) Please refer to Appendix 2. |
| 5.49. | 5.9.3.1 Irradiation Damage | Irradiation damage for Al | 1) The fluences used for estimations of damage (up to 2x$10^{23}$ n.cm-2) should be verified in a view of the previous question.<br>2) Calculations of dpa in Al should be provided. |

| Question Reference | Section number and name | Topic for clarification | ARPANSA Comment, Issue or Question |
|---|---|---|---|
| | | | Response:<br>1) Aluminium components are either replaced (fuel plate cladding) or are in low fluence regions. Fluence is confirmed to be lower than $10^{23}$ n/cm$^2$<br>2) Fluence information can be readily correlated with irradiation damage data for this material. |
| 5.50. | Table 5.9/6 | Irradiation damage for Al | Is the ratio of thermal to fast flux (1.7 and 2) in the Table what is predicted for the RRR? |
| | | | Response: The thermal-to-fast flux ratio in the RRR varies depending on location. It is at its lowest in the core region, increasing in the reflector tank as a function of distance from the core. A value in the fast flux irradiation facilities is approximately 13. Further information on the thermal-to-fast ratio can be found by referring to Figures 5.7/3 to 5.7/8. |
| 5.51. | 5.9.3.3 Corrosion | Corrosion of Al | The statement that oxidation rates are sufficiently low should be further substantiated. An appropriate comparison with HIFAR and HFBR should be arranged in a table. Perhaps some other research reactors should be listed. |
| | | | Response: Corrosion resistance for the aluminium alloys to be used in the RRR is very good in both the welded and un-welded condition, particularly when the pH and conductivity of the water is controlled as tightly as it will be in the RRR. Further information is provided in Appendix 3 and will be included in the next revision of the PSAR. |
| 5.52. | 5.9.4 Polymers | Polymers used in RRR | A table summarising all polymers should be given with estimated damage, frequency of change and location of use in the reactor. |
| | | | Response: A table summarising the properties of polymers in radiation fields is shown in Appendix 4. |
| 5.53. | 5.9.6 Zirconium Alloys | Suitability of zirconium for chimney | It is stated that zirconium alloy is 'the most suitable material for construction of the reactor Vessel and the Chimney'. Should the Chimney Structure be constructed from stainless steel as indicated Table 5.9/1? |

| Question Reference | Section number and name | Topic for clarification | ARPANSA Comment, Issue or Question |
|---|---|---|---|
| | | | Response: The part of the chimney that is made from stainless steel is the section above the reflector vessel. The lower section is part of the reflector vessel and is thus Zircaloy-4. The referred table will be clarified. |
| 5.54. | 5.9.6.1 Irradiation Damage | Irradiation damage of zirconium | 1. Neutron fluences used should be verified in view of questions 5.9.2.1.<br>2. dpa should be calculated for most vulnerable parts of the chimney and compared with established criteria. |
| | | | Response:<br>1) We confirm that $10^{23}$ n/cm$^2$ is the order of magnitude of the fluence expected in the Zicaloy-4 components.<br>2) This information will be prepared together during the Detail Engineering phase and reported in the Final SAR. |
| 5.55. | 5.10 Reactor Design Software | Additional information | ARPANSA expects that a full set of manuals of neutronics and thermal hydraulics codes used for the design of RRR will be available for ARPANSA reference. |
| | | | Response: It is suggested that the provision to ARPANSA of a full set of manuals for the codes is not appropriate both from the volume of materials viewpoint and potentially from a Copyright viewpoint. A full set of manuals will be available at ANSTO; ARPANSA staff will be able to view them when required. |
| 5.56. | 5.10.2.6.1 Experimental Measurements | Experimental Measurements | A summary of the results obtained should be provided in a table. |
| | | | Response: Agreed. The information, with results, is currently being consolidated and will be provided in a separate document shortly. |
| 5.57. | 5.10.2.6.2 Theoretical Verifications – Benchmark Calculations | Theoretical Verifications – Benchmark Calculations | What was the difference in the results obtained by different parties? A brief summary in a table should be provided. |

| Question Reference | Section number and name | Topic for clarification | ARPANSA Comment, Issue or Question |
|---|---|---|---|
| | | | Response: This information is contained in IAEA TECDOC-233. The only results relevant for the PSAR are those obtained by INVAP with the codes used for the design of the RRR. |
| 5.58. | 5.10.3 Thermal-Hydraulics Design Codes | Thermal-Hydraulics Design Codes | Please justify the use of so many Thermal-Hydraulics Design Codes. |
| | | | Response: There are many thermal-hydraulic parameters to calculate for both steady state and transient conditions. As such there are different physical models required. It is usual to have specialised codes for these various parameters and states.<br>ANSTO neither understands the concern nor considers that there are too many design codes. Reactor design conventionally uses a flow network code to determine how steady state flows are distributed between available channels, etc. INVAP's version of this is CAUDVAP. Reactor design conventionally uses a steady state code to determine local thermal-hydraulic phenomena for flows obtained from the network code. INVAP's version of this is TERMIC. As these codes are for forced flow conditions INVAP uses CONVEC for the important natural convection regime at shutdown conditions. PARET has been widely used to study reactivity transients; INVAP follows this practice. RETRAN has been widely used for coolant and flow transients. INVAP follows this practice. |
| 5.59. | 5.10.4.5 ANISN | ANISN | Statement that 'it used anisotropic cross section of any order and can calculate fluxes for any quadrature' is vague. Please clearly indicate what order of anisotropic cross section and what type of quadrature sets are used for design calculations. |
| | | | Response: This statement is an indication of the capabilities of the code. ANISN has been used with P3 and S16; for reactor block concrete thickness, CRD Room shielding and dose at the top of the pool for various water levels. The DORT core (2D) has been used typically with P5 and S12 with the exception of the case of the calculations for the shielding of the thermal shutter where P5 and S282 was used. |
| 5.60. | 5.10.4.6 DOT | DOT | The statement that DOT is a 1-D code should be verified. |

| **Question Reference** | Section number and name | **Topic for clarification** | **ARPANSA Comment, Issue or Question** |
|---|---|---|---|
| | | | Response: DOT is a 2-D code. It may be used for 1D or 2D calculations. |

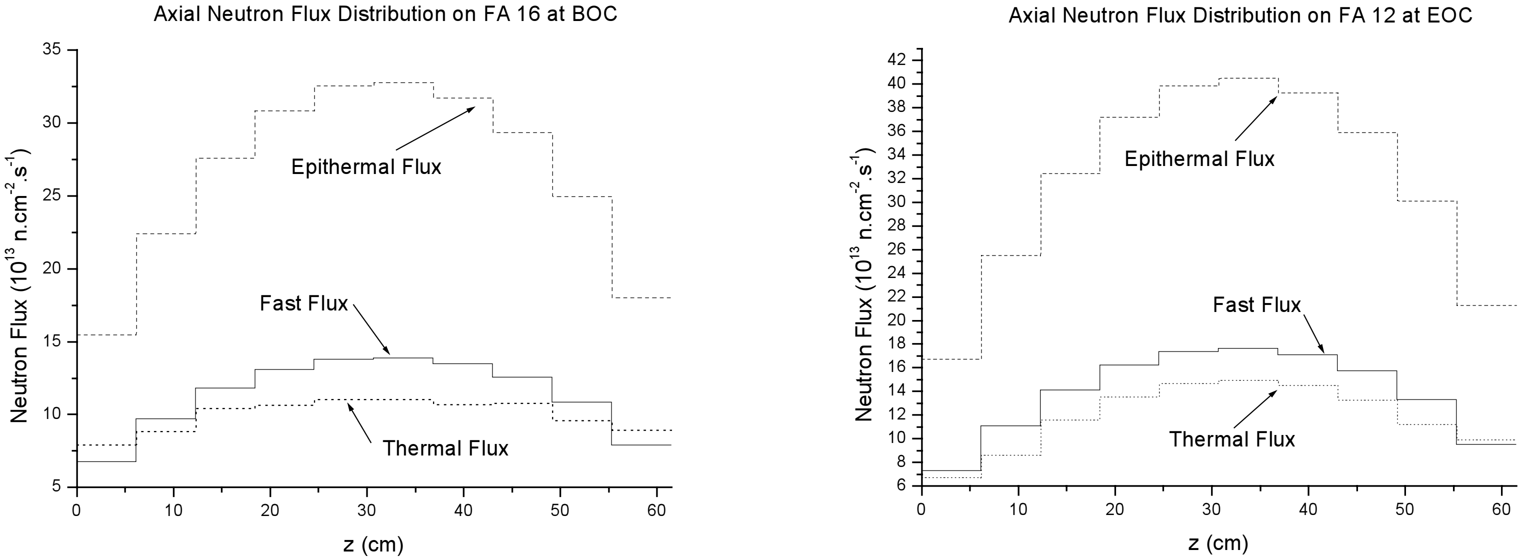
## APPENDIX 1: INFORMATION FOR QUESTION REFERENCE 5.33

Flux distributions is provided for the FAs having the maximum power density at BOC and EOC namely, FA 16 and FA 12 respectively.
The values presented in the tables and the graphics are the average neutron fluxes. Average neutron fluxes are computed in the volumes that results from the division of the FA active length in 10 equal parts. Layer 1 corresponds to the upper part of the FA and layer 10 to the lower part of the FA.
Three groups of energy are reported in the tables: fast flux (En > 0.821 MeV), epithermal flux (0.821 MeV < En < 0.625 eV) and Thermal flux (En < 0.625 eV).

| FA 16<br>Maximum power density at BOC<br>($10^{13}$ n.cm$^{-2}$.s$^{-1}$) | | | | |
|---|---|---|---|---|
| Layer | x (cm) | Fast | Epi | Thermal |
| 1 | 0<br>6.15 | 6.744 | 15.463 | 7.916 |
| 2 | 6.15<br>12.3 | 9.685 | 22.413 | 8.856 |
| 3 | 12.3<br>18.45 | 11.781 | 27.562 | 10.412 |
| 4 | 18.45<br>24.6 | 13.083 | 30.827 | 10.635 |
| 5 | 24.6<br>30.75 | 13.756 | 32.526 | 11.031 |
| 6 | 30.75<br>36.9 | 13.853 | 32.757 | 11.01 |
| 7 | 36.9<br>43.05 | 13.453 | 31.695 | 10.689 |
| 8 | 43.05<br>49.2 | 12.548 | 29.33 | 10.777 |
| 9 | 49.2<br>55.35 | 10.816 | 24.957 | 9.587 |
| 10 | 55.35<br>61.5 | 7.89 | 17.986 | 8.913 |

| FA 12<br>Maximum power density at EOC<br>($10^{13}$ n.cm$^{-2}$.s$^{-1}$) | | | | |
|---|---|---|---|---|
| Layer | x (cm) | Fast | Epi | Thermal |
| 1 | 0<br>6.15 | 7.315 | 16.732 | 6.698 |
| 2 | 6.15<br>12.3 | 11.101 | 25.497 | 8.625 |
| 3 | 12.3<br>18.45 | 14.119 | 32.416 | 11.609 |
| 4 | 18.45<br>24.6 | 16.203 | 37.194 | 13.556 |
| 5 | 24.6<br>30.75 | 17.335 | 39.824 | 14.646 |
| 6 | 30.75<br>36.9 | 17.624 | 40.482 | 14.959 |
| 7 | 36.9<br>43.05 | 17.108 | 39.232 | 14.512 |
| 8 | 43.05<br>49.2 | 15.708 | 35.889 | 13.284 |
| 9 | 49.2<br>55.35 | 13.304 | 30.132 | 11.206 |
| 10 | 55.35<br>61.5 | 9.496 | 21.299 | 9.888 |

a)


Axial Neutron Flux Distribution on FA 16 at BOC
Neutron Flux ($10^{13}$ n.cm$^{-2}$.s$^{-1}$)
z (cm)
Epithermal Flux
Fast Flux
Thermal Flux
Axial Neutron Flux Distribution on FA 12 at EOC
Neutron Flux ($10^{13}$ n.cm$^{-2}$.s$^{-1}$)
z (cm)
Epithermal Flux
Fast Flux
Thermal Flux

## APPENDIX 2: INFORMATION FOR QUESTION REFERENCE 5.48

Information on creep and swelling for other reactors.

| Reactor | Material | Applied Tensile Stress MPa | Temperature | Fluence , n/cm$^2$ | Swelling | Creep |
|---|---|---|---|---|---|---|
| EBR II (1) | 304 L | 0 - 188 | 395°C-530°C | $1.8 - 9.3 \times 10^{22}$ | 1 % | 2 % |
| EBR (2) | 316 – specimen | Pressurised Tube | 443°C-480°C | $\approx 1 \times 10^{24}$ | 10% | 4% |
| PWR – Control Plate Cladding(3) | 304 – 316 | - | 350°C - 750°C | $1\text{-}12 \times 10^{22}$ | 5%-10% | |

(1):Flinn, J. E., G. L. McVay and L. C. Walters. ”In Reactor Deformation of Solution Annealed Type 304L Stainless Steel.. J of N. Mat. 65, 210-223. 1977.
(2): Harris, D. R. “Irradiation Creep in Non-Fissile Metals and Alloys”. J. of N. Mat. 65, 157-173. 1977.
(3): Ursu, I.. “Physics and Technology of Nuclear Materials”. Book-Pergamon Press. 1985

## APPENDIX 3: INFORMATION FOR QUESTION REFERENCE 5.51

The corrosion resistance of the 1XXX, 5XXX and 6XXX series of aluminium alloys in water is very good[1] in both the welded and un-welded condition[2]. The pH and conductivity the pool water and heavy water in the RRR are controlled in order to minimise corrosion rates.

Binger and Marstiller[3] have analysed the use of aluminium alloys (1XXX, 3XXX, 5XXX and 6XXX series alloys) in the distilled water industry. Some of the installations had operated for 26 years (1100 alloy). The use of aluminium in this industry is testimony to the low corrosion rate that avoids contamination of the product. It can be concluded from this paper that the corrosion rate of aluminium in high purity water is minimal.

ANSTO's experience with late-life inspections of HIFAR indicates that the corrosion rate of aluminium in the reactor aluminium tank has been very low. Ultrasonic thickness measurements of the tank wall, re-entrant tubes, downcomers, plenum plate and the 2-Tan tube all indicate that the metal thickness is still within manufacturing tolerances; even after 40 years of service[4,5].

Griess et. al.[6] studied corrosion in aluminium alloys related to use in research reactors. They concluded from "Within the ranges investigated, pressure and flow rate were without effect, and the same results were obtained with 6061 and 1100 aluminium". Although the conditions used in these tests were more severe than those that will be experienced within the RRR, results from these tests provide an indication that the corrosion rates of both 1100 and 6061 would be acceptable. A subsequent extension to this work[7] included X-8001 alloy and provided correlations with corrosion behaviour of 1100 and 6061 alloys. This work further reinforced the acceptable corrosion characteristics of aluminium for use in nuclear applications.

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[1] Metals Handbook 9th Edition, Volume 13, Corrosion, Corrosion of Aluminium Alloys, pp. 583 – 609, 1987.

[2] Welding Aluminium: Theory and Practice, The Aluminium Association, Second Ed. 1991, ISBN 89-080539.

[3] W.W. Binger & C.M. Marstiller, Aluminium Alloys for Handling High Purity Water, Corrosion, Vol 13 (No. 9), 1957

[4] Final Report on the Inspection of the HIFAR Aluminium Tank. Task No. 83, HIFAR MSD, ANSTO Internal Report NTP/TN 1995, 1995

[5] Inspection of HIFAR Reactor Aluminium Tank, 2000 MSD Final Report, ANSTO Internal Report R00m021, 2000

[6] J.C. Griess, H.C. Savage, J.G. Rainwater, T.H. Mauney, and J.L, English , Effect of Heat Flux on the Corrosion of Aluminium by Water, Part III, Final Report on Tests Relative to the High-Flux Isotope Reactor, ORNL-3230, February 1964.

[7] J.C. Griess, H.C. Savage and J.L. English, Effect of Heat Flux on the Corrosion of Aluminium by Water. Part IV. Tests relative to the Advanced Test Reactor and Correlation with Previous Results, ORNL-3541, December 1964.

## APPENDIX 4: INFORMATION FOR QUESTION REFERENCE 5.52

| Place | Calculated gamma dose at 40 years | Maximum admissible dose for the material |
|---|---|---|
| Control rod seals | $3{,}4 \times 10^4$ rad | Buna N: 1 $E^7$ - 1 $E^8$ rad / Molythane: 1 $E^8$ -1 $E^9$ rad |
| O-rings at Flap Valves (Level 5,8) | $< 10^3$ rad | EPDM: 1 $E^8$ -1 $E^9$ rad |

Seals to be installed in components located inside the reactor pool will be metallic.

Checked / agreed: