

ARPANSA Regulatory Assessment of the Replacement Reactor Construction Application

9 July 2001 - Reactive Review Comments, Questions and Issues

PSAR Chapter 12 Operational Radiological Safety

Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
12.1.	12.1.1.1 Risks from radiation	<p>“[Radiation] is assumed to have a possibility of leading to the development of fatal cancers, non-fatal cancers or genetic effects.</p> <p>“According to the hormesis model, some radiation is beneficial to health,...</p> <p>“Supporters of this model cite the fact that people who live in mountainous areas, at high altitudes with high radiation levels, suffer less cancer than others. Although true,....”</p>	<p>ARPANSA considers the phrase “is assumed to have” is at best misleading given that radiation at certain doses (greater than 100 mSv, depending on effect) is known to cause certain cancers. ARPANSA follows world radiation protection practice (ICRP, IAEA) in assuming a linear, no threshold hypothesis for dose effects.</p> <p>Please provide references to the beneficial effects of radiation exposure in the context of the operational radiological safety of a nuclear reactor.</p> <p>Please provide appropriate references to the scientific literature supporting the phrase “suffer less cancer than others”.</p>
			<p>Response: Will rephrase words “is assumed to have a possibility of leading” to “may lead”.</p> <p>Some references from the Health Physics Journal are:</p> <ul style="list-style-type: none"> • HP Vol 75 No 4 1998 J Jagger Natural background and cancer death in Rocky Mountain States and Gulf Coast States. • HP Vol 80 No3 2001 K Mossman Deconstructing Radiation Hormesis • HP Vol 70 No5 1996 R Kathryn Pathway to a Paradigm: The Linear Non-Threshold Dose Response Model in Historical Context; The American Academy of Health Physics Centennial Hartman Oration.
12.2.	12.1.2 The ALARA Principle	Application of ALARA. Section 12.1.2 and Figure 12.1/1 purport to show steps in the procedure for determining whether doses are ALARA or not.	Information in 12.1.2 and table 12.1/1 indicates how it is established whether an ALARA assessment is required, not whether doses are ALARA. Please provide information on the procedure or methods to be followed if an ALARA assessment is deemed necessary.

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			<p>Response: The ANSTO Safety Directive on ALARA describes the criteria for determining if an ALARA assessment is required and outlines the assessment procedure.</p> <p>Section 12.3.2.10 describes the optimisation procedure of the main shielding structures if an ALARA assessment is deemed necessary.</p>
12.3.	12.1.1(c) Optimisation	2nd last line states: "Dose Constraints, agreed with ARPANSA, shall be used for the control of exposures".	<p>ARPANSA notes that the review of the ANSTO dose constraint is a current condition of ANSTO's existing facilities' licence 'within a timetable acceptable to the CEO'. Please indicate the status of this review and its impact on the RRR.</p>
			<p>Response: The review of dose constraints will be delivered to ARPANSA as part of ANSTO response to the existing facility licences. The Reactor Facility doses will be consistent with these constraints.</p>
12.4.	12.1.2.1 Operational Reference Levels	Operational Reference Levels appear to include a mixture of dose constraint, ALIs, DACs, surface contamination limits, ALARA, investigation levels.	<p>Introduction of another general term "Operational Reference Levels" is potentially confusing when other specific terms are used commonly in radiation protection practice. Please review the use of and, if necessary, provide a definition of "Operational Reference Levels".</p>
			<p>Response: Operational reference level is a generic term used for practical measurements of activity, surface and airborne contamination and dose and dose rates and is presented as such in Ch12. Where specific reference levels are used, their specific nomenclature will be and is used eg Derived Air Concentration, Notification level, etc. The definition of a reference level is given in IAEA Safety Series No 115 International Basic Safety Standards for the Protection against Ionizing Radiation and for the Safety Of Sources page 312.</p>

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12.5.	12.1.2.2 Operational Reference Levels for Emissions	Section 12.1.2.2 - second line states that the Operational Emission Limits will be established to "fulfil the public dose constraints".	<p>Please clarify the use of ALARA in this context.</p> <p>These operational reference levels (read Dose Constraint) are "set initially and adjusted as necessary as feedback from operational experience is acquired". Please explain what is intended by this statement.</p>
			<p>Response: The Reactor Facility design incorporates specific features to minimise airborne radioactive discharge. At the commencement of reactor operation, there will be no specific stack discharge data operational history to draw upon. The Airborne Discharge Authorisation for the Reactor Facility is subject to specific agreement by the ARPANSA CEO as part of the licence conditions. For ARPANSA consideration, it is proposed here that operational limits (cf correction levels) be based on public dose constraints and investigation levels (cf notification levels) be set at one tenth these values. As operational stack discharge data is acquired, this data can be reviewed for trend analysis and notification levels optimised similar to those levels set for other stacks on the LHSTC site under existing licence conditions.</p>
12.6.	12.1.2.1 Operational reference levels for doses	Source term for dose calculations from airborne releases	See item "12.2.2 Airborne and liquid sources" below.
			Response: see response to comment 12.13

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12.7.	12.1.2.2 Operational Reference Levels for Emissions	"All releases in excess of the Air Effluents Monitoring (AEM) system minimum detection limits will be recorded and reported on a monthly, quarterly and annual basis....."	Please identify the reporting procedures and to whom reports are made.
			Response: Insert " to ARPANSA" after "reported". The mechanisms for reporting to ARPANSA will be similar to those currently employed for other LHSTC stacks or as these evolve through ongoing discussion and agreement with ARPANSA.
12.8.	12.1.2.2 Operational reference levels for emissions	In the unlikely event of an abnormally high release, containment isolation will occur automatically	Please identify where in the PSAR 'containment isolation' is discussed or defined.
			Response: The Reactor Containment System, including the containment isolation system, is described in Chapter 7, Section 7.8.
12.9.	12.1.5 Audit and Review Programs	"The radiation protection activities are periodically reviewed to ensure compliance with legislation and consistency with international standards."	Please outline the procedure and arrangements for such reviews and indicate the likely frequency of them.
			Response: Periodic reviews will be undertaken within the Reactor Facility and Safety Division QA systems. The frequency is likely to be biannual (2 Yearly) or as required.
12.10.	12.1.5 Audit and Review Programs	"a) Any new practice or modification of a previous one will not generate a significant increase in the public or personnel exposure to radiation."	Please rephrase to indicate that new practices or modifications may be subject to ARPANSA approval.

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Question reference	Section number and name	Topic	ARPANSA Comment, Issue or Question and ANSTO's Response
			Response: Agreed; will incorporate "subject to ANSTO Safety and ARPANSA Approval".
12.11.	12.1.5 Audit and Review Programs	"b) Periodic evaluations of the effectiveness and efficiency of radiological protection activities are performed and conveniently documented."	The effectiveness and efficiency of radiological protection activities should be documented routinely.
			Response: Agreed; will delete "conveniently". It refers to the ease of retrieval of the information.
12.12.	12.1.5 Audit and Review Programs	"d) Staff suggestions have been evaluated and incorporated if they produce a reduction in radiological hazards." "e) Modifications of regulations/recommendations have been introduced in plant documentation and have been notified to the staff through retraining activities."	The use of the past tense ("have been") appears odd. Staff suggestions should continue to be evaluated and incorporated if they continue to produce a reduction in radiological hazards. Major changes may be subject to ARPANSA approval. Please review and rephrase.
			Response: Agreed. Will reword "have been" to "are" in d) and rephrase e) to "Modifications in national or international regulations or codes are reviewed as to applicability and where required incorporated into documentation following Safety and ARPANSA approval. Retraining activities are then undertaken to inform staff of modifications."
12.13.	12.2.2 and Table 12.2/5 Postulated Annual Releases; Figure 12.5/4	The noble gas fission product (Kr, Xe) releases to the reactor stack are postulated to total 650 GBq. Section 12.1.2 states that "reduction of radiation dose and emissions shall be achieved by the application of the ALARA principle".	The current airborne discharge authorisation for HIFAR does not permit any release of noble gas fission products. Please provide an analysis of the justification and optimisation of the postulated releases, including consideration of international best practice and the Environment Minister's conditions 7 and 8.

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			<p>Response: The postulated level of emissions for Kr and Xe isotopes from the Reactor Facility is not significant in terms of public exposure or other emissions of these radionuclides from elsewhere on site (the postulated Reactor Facility emissions are almost three orders of magnitude lower). These estimates have been provided for completeness and are consistent with values reported at similar facilities operating worldwide. Similar emissions of Kr and Xe isotopes from HIFAR are not detectable. The extremely low total dose to the public, from the Reactor Facility, clearly shows the commitment made by ANSTO to minimising all emissions.</p>
12.14.	12.3.1.1 Zoning and Access Control	"Forbidden areas: areas where potential dose rates are above radiation red area values or contamination risk is higher than contamination red area values. No access is allowed during normal operations."	<p>Under what circumstances and conditions is access allowed to Forbidden areas? Please outline any procedures for authorising access. Please describe the controls to prevent unauthorised access?</p>

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			<p>Response: The access is allowed to forbidden areas in specific plant conditions after appropriate radiological surveillance is performed to guarantee safe working conditions inside. The procedure to authorise access will include:</p> <ul style="list-style-type: none"> a) Verification of appropriate reactor status condition (e.g. shutdown) b) Radiological surveillance with portable equipment c) Staff wearing appropriate personal dosimeters d) Staff wearing appropriate protective clothing or gear as appropriate e) Prior planning of the tasks to be carried out in the area. <p>To prevent unauthorised access, the controls will include:</p> <ul style="list-style-type: none"> a) location of warning signs on access points into the forbidden area b) locked doors c) Procedures and training of facility staff.
12.15.	12.3.1.1 Zoning and Access Control	"Restricted areas: areas where, due to nuclear or conventional safety reasons, special provisions are required to allow the presence of personnel. In these areas dose rate values are normally low but can increase suddenly without local control (that is, no local actions produce or can prevent the dose rate variation)."	What measures are in place to monitor and protect personnel from sudden variations in dose rates?
			<p>Response: Area radiation and plant monitoring equipment give indication (including visual/audible alarms) prior to entry and locally regarding hazard level. Personal radiation monitoring equipment for individuals will also give dose rate level indication.</p>

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12.16.	12.3.1.1 Zoning and Access Control	"Showers and basins are distributed in the building (see Figure 12.3/17, Figure 12.3/18, Figure 12.3/19, Figure 12.3/20, Figure 12.3/21). An area for preliminary decontamination of tools and objects will be established in level +4 with appropriate basins and instruments. Further decontamination process could be continued, if necessary, in other LHSTC site facilities."	<p>Are the showers and basins specifically designed to handle decontamination, or are they just features of the amenities?</p> <p>The introduction of further decontamination facilities if necessary may not be adequate in emergency intervention. The establishment of decontamination facilities and procedures should be completed in preparedness for emergency decontamination and intervention situations.</p>
			<p>Response: Showers and basins are appropriate for the performance of personal decontamination activities.</p> <p>Noted, these aspects will be developed during the detail engineering phase and presented in the FSAR</p>
12.17.	12.3.1.1.3.1 Access to Reactor Building	"b) access from the Neutron Guide Hall to Reactor Beam Hall"	Please identify the security and access controls in this instance.
			Response: Chapter 4, Section 4.6.4 describes the access between Neutron Guide Hall and Reactor Beam Hall. Security controls will be provided in accordance with the Security Plan.
12.18.	12.3.1.4 Material Characteristics From the Radiological Viewpoint	"Shielding materials will be selected so that it is ensured that they will remain effective along the planned life of the facility, especially that they are capable of withstanding the radiation-induced damage. This issue is particularly important for beam shutters in relation to accumulated radiation dose and gamma heating phenomena."	Please identify or outline the methods and criteria for selection of shielding materials and their thicknesses.

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			Response: See Section 12.3.2
12.19.	12.5.1.1 Collective and individual doses in staff	For scheduled shut down and lightweight maintenance activities, the dose rate assigned is 25 μ Sv/h.	Please provide justification for this choice of dose rate.
			Response: This is based on INVAP operational experience during tasks that do not involve major maintenance.
12.20.	12.5.1.2 Dose in Experimental Related Areas	Contamination hazards – glove boxes and fume cupboards have been located in NAA and Chemistry Blue Labs.	Please reference information on monitoring and/or filtration of exhaust emissions.
			Response: Information on monitoring and filtration from glove box and fume cabinet exhaust is discussed in Chapter 10, Sections 10.4.7, Chapter 11, Section 11.4.3, and Section 12.4.6 together with Figure 12.4/8 and Figure 12.4/9.
12.21.	12.5.2 Dose Estimates for the Public	The wind frequency and atmospheric stability data used in the PC-CREAM modelling is for 10 metres whereas the release occurs from a 45 metre high stack.	Please provide further justification of the data and assumptions used.

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			<p>Response: The meteorological data is provided in the report "Estimates of Doses from Routine Airborne Emissions of Radionuclides from Lucas Heights Science and Technology Centre during 1997 and 1998" reference SD/SR/TN-99-11 rev 0, Appendix B (already supplied to ARPANSA). This provides a conservative estimate. A comparison of atmospheric stability distributions versus wind directions (data used in PC-Cream) using data from both 10m and 37/49m indicate the following :</p> <ol style="list-style-type: none"> 1. Atmospheric stabilities have been determined using the USEPA (1987) method. Because of higher wind speeds at 49m, the atmospheric stability distributions tend to be more towards the neutral category D, both in the day and more importantly at night when dispersion is limited by reduced atmospheric turbulence. 2. Wind direction distributions are similar at 10 and 49 m with some reduction in the influence of south winds at the higher altitude but very similar distributions of winds from the more important SSW through NW sectors. <p>The wind speed at different release heights is obtained by the PC CREAM using formula 3.10 of manual (EUR 15760). As indicated in the PC CREAM Manual, this is an acceptable assumption and approach.</p>
12.22.	12.6.1 Calculated and expected doses	... in most cases the expected doses are below the ALARA objective	Please provide justification for those cases where it is estimated that the ALARA objective will be exceeded.
			<p>Response: In those cases where doses are found to be above the ALARA objective, an ALARA assessment will be carried out. The results of the ALARA assessment will be provided in the FSAR:</p>

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12.23.	12.6.1 Reactor Facility Features	The Plant layout... a downgrade in the classification of certain areas is adequate and possible.	Please provide criteria and procedures for downgrading area classification.
			Response: The existing ANSTO Safety Directive "Radiological Classification of Areas and Access Control to Radiological Areas" describes this procedure for the review of radiological hazards undertaken by RPA/Health Physicist and Area Supervisor. This Safety Directive is applicable to the Reactor Facility.
12.24.	2.4.22 Radiation Protection Measures in the Design	"areas of continuous occupancy will have dose rates not greater than $5\mu\text{Sv h}^{-1}$ "	This could give an annual dose of 10 mSv. Please provide a justification and optimisation analysis for this dose rate.
			Response: There is a typographic error on dose rate. It should be $0.5 \mu\text{Sv/h}$. This erratum has already been identified and is included in the PSAR Errata Document (RRRP-7220-2BEIN-002-Rev0-Errata).
12.25.	12.1.3 Radiation Protection Plan	Quality of RRRP	Please identify the QA program and its status in relation to the RPP and its elements.
			Response: The Radiation Protection Plan will be an integral part of the Reactor Facility QA program that is discussed in Chapter 18. This will be presented in the FSAR.

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12.26.	12.1.3.1 Organisational Staffing and Responsibilities for Radiation Protection	The second last paragraph, and the last part of the last paragraph, seems to place the responsibility for controlling doses on the individual's shoulders. The last line states that employees "will ensure that any doses received does not exceed the given limits".	Please clarify the ANSTO management responsibilities with regard to radiation protection policies and practices, 'defence in depth', and ALARA, and the relationship of the management responsibilities to that of employees.
			Response: The Employer and Employee duties as specified in the National Standard for limiting occupational exposure to ionizing radiation (NOHSC: 1013 (195) or any updated version of this will apply.
12.27.	12.1.3.5 Effluent/Emission Monitoring	Part (b) talks about an 'adequate environmental model'	Please clarify what is 'adequate'?
			Response: Please replace "adequate environmental model" with the words "appropriate environmental model as agreed with ARPANSA".
12.28.	12.1.3.7 Classification of Areas, Persons and Tasks		Please clarify the approach intended for the classification of areas with regard to <u>potential</u> hazard.
			Response: The classification of areas with regard to potential hazards is discussed in Section 12.3 and is in accordance with the existing ANSTO Safety Directive - "Radiological Classification of Areas and Access Control to Radiological Areas".

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12.29.	12.1.4.3 Small Inter-Building Shielded Transfer Container	The last sentence of the first paragraph describes the shield thickness as 85 mm of Depleted Uranium.	Is the DU completely enclosed? Please identify or provide the assumptions and calculations used to derive the shield thickness.
			Response: Details will be determined during detail engineering phase and presented in the FSAR.
12.30.	12.4.1 Waste Management Principles	The waste management system for the Reactor Facility has been designed to guarantee the safety of the personnel involved in the activity and that of the general public and to minimise potential environmental impacts.	Please identify and describe the Waste Management Principles in this section. Where or how does the waste management system comply with the applicable IAEA WASSAC documents, ISO 14001 and or other documents reflecting international best practice in waste management?

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			<p>Response: The Waste Management Principles are consistent with IAEA, Safety Series No. 111-F, The Principles of Radioactive Waste Management (1995). The PSAR provides information with regards to its principles regarding waste reduction (Section 12.4.3.1 4th paragraph) and its process in ensuring all waste is tracked (Section 12.4.3.4). In addition, ANSTO has given a commitment that all processes related to the Reactor Facility and the rest of ANSTO activities will be compliant with the ISO 14001 standard. Included in this is the commitment (stated in ANSTO's Health Safety and Environment policy) to seek continual improvement in environmental performance. This commitment includes the management of waste that will use tracking to facilitate waste minimisation and reduce both short and long-term consequences of the waste. This has been incorporated into the design phase for the Reactor Facility (Chapter 14, Section 14.5.1.2.) and will carry through all stages in the RRRP. Handling and classification of waste will be as per the relevant national and international legislation, codes of practice and guidelines and will reflect international best practice where appropriate.</p>
12.31.	12.4.2 Monitoring, Control, Segregation and Classification	This section details the equipment and administrative arrangements used for monitoring, sampling, control, segregation and classification.	How is the monitoring regime determined? For example, where are the following identified and discussed: waste sources, background levels, acceptable levels in terms of dose and acceptable levels in terms of detection of operational problems?

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			<p>Response: Waste sources are discussed in Section 12.4.4 (solid), Section 12.4.5 (liquid) and Section 12.4.6 (gaseous). The selection of equipment requirements for the Reactor Facility was based on the activities and potential waste sources within the facility. Where appropriate, the selection of sampling methodology and sampling locations were directed by the necessity for monitoring of all emissions in a manner compliant with international best practice. Equipment requirements cover both routine and non-routine practices and emissions. Detection limits and monitoring requirements were also dictated by both the need to prove compliance with dose constraints and the stipulated investigation levels at a tenth of these values. The use of monitoring instruments will also be subject to a process of continual review as required by ISO14001.</p>
12.32.	12.4.3 Waste Generation	<p>States waste reduction principles will be applied through technical and administrative controls.</p> <p>Identifies types of identified waste. Quantifies amounts of waste.</p> <p>States waste shall be tracked. <i>All radioactive waste will be identified by an associated record, which will identify the source of the waste, its type volume and main radionuclides.</i></p>	<p>Please identify the origin of these waste principles.</p> <p>Characterisation of waste by radionuclide is only specified for airborne waste. Please provide characterisation for solid and liquid wastes. Please provide (or identify where to find) a description of liquid waste characterisation.</p> <p>What does “main radionuclides” mean? What is the process in place to identify that a radionuclide is not “main”?</p>

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			<p>Response: The process of waste reduction and minimisation is a general principle of environmental management and also is part of international best practice. The principles are consistent with IAEA, Safety Series No. 111-F, The Principles of Radioactive Waste Management (1995)</p> <p>A full radiological characterisation of solid waste was not provided in the PSAR as the final selection of materials will influence the radionuclides present. The exact radionuclide activities in the solid waste will not significantly effect exposure to workers or members of the public due to the waste handling facilities and procedures. For the most significant of the solid wastes, the spent resins, a full breakdown of the expected radionuclides is provided in the PSAR (Chapter 14, Table 14.2/2). For liquid wastes, the primary criteria used for discharge are given in Section 12.4.5.3.7 and are expressed as gross alpha, gross beta and tritium as is required for eventual discharge to the sewer (after further treatment) under the Trade Waste Agreement with Sydney Water.</p> <p>“Main radionuclides” relates to those radionuclides with importance with respect to either dose to workers or the public, performance of systems or with consequences for downstream processing.</p>
12.33.	12.4.4.2.3 Neutron Beam Guide Tubes	..'will be removed' after 10 yrs operation...	What is the planning arrangements for the removal? What are the estimated doses received by persons removing the tubes?

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			<p>Response: As indicated in section 11.5.1.1, there will be a specially designed shielded cask that will be placed in front of the shutter to allow the removal of irradiated components with a minimum of doses to operation staff.</p> <p>Further details on the operation and estimated doses will be provided in FSAR following the detail engineering phase.</p>
12.34.	12.4.4.6 Ventilation System Waste	It states that the filters 'are routinely replaced ... and are classified as low-level waste"	Please provide the basis and criteria for the replacement of the filters and the criteria used to classify them as low-level waste.
			<p>Response: Filters are replaced after a fixed time interval relating to recommendations from the manufacturer or when there is an indication of either heavy dust loading (high pressure drop across the filter), high dose rate at the filters or indication of poor performance (change in emission data). The HEPA filters will be classified as per other forms of low level waste, ie the radionuclides will be determined by gamma spectroscopy and the waste classified as per the code of practice in force in Australia at that time.</p>
12.35.	12.4.4.7 Solids and Sludge from the Ion Exchange Filters	The third paragraph talks about the discharges of 'solids in suspension' into the B or C line	Please identify the radionuclide characteristics of these solids and sludge and criteria for discharge.

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			<p>Response: All discharges into the B or C line will be monitored for gross alpha, gross beta and tritium as is required for eventual discharge to the sewer (after further treatment) under the Trade Waste Agreement with Sydney Water. For liquid wastes, the primary criteria used for discharge are given in Section 12.4.5.3.7</p>
12.36.	12.4.5 Liquid Waste Management.	<p><i>Collection systems allow for the segregation of liquid waste according to the radioactivity level and for monitoring and temporary storage of liquid wastes.</i></p> <p>Most waste is low level for discharge through B-line.</p> <p>Effluents, which contain heavy water, will be collected, stored and transferred separately.</p> <p>A waste stream monitor (WASMO) is installed in each of the temporary liquid waste storage tanks. This system allows detection of any discharge exceeding the admissible limits of that line.</p> <p><i>Details of discharge criteria are given in ANSTO's licence application to ARPANSA (ANSTO LA-01).</i></p>	<p>How is "low level liquid waste" defined?</p> <p>What are the criteria for sending liquid waste to the B and C line? To what extent will the liquid waste system use existing pumping systems and piping?</p> <p>Please clarify the interface with Waste Operations and Technology Development?</p> <p>What parameters is the WASMO measuring? What are the MDAs and sensitivity of this method for radionuclides to be detected? Where are the "admissible limits of that line" specified?</p> <p>Please provide the full reference to ANSTO LA-01 and where in that reference the details of the liquid discharge criteria are addressed.</p>

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			<p>Response: Low level liquid waste is defined as liquid waste that does not require shielding during handling and transportation, due to its low radionuclide content.</p> <p>The limits on the discharge of radioactivity to the B and C-lines are defined in the Draft Safety Directive (SD 5.7 – ANSTO's Radioactive Waste Management System). These limits are 100 kBq beta and 5 kBq alpha activity per approved operation for the C-line and 1 MBq beta and 50 kBq alpha activity per approved operation for the B-line.</p> <p>The Reactor Facility liquid waste system will connect directly to the existing HIFAR north delay tank system.</p> <p>Waste Operations and Technology Development is responsible for effluent and associated systems at the point of discharge from the Reactor Facility.</p> <p>The WASMO is a gamma detector that measures the gamma activity of the effluent streams from the Reactor Facility. Its capability will depend on the selected instrument.</p> <p>The reference for ANSTO LA-01 is 4Bb-6.1.3 Consent to Discharge Trade Waste Water (Category 3, Agreement No. 4423, between ANSTO and Sydney water Corporation).</p>
12.37.	12.4.5 cont Liquid Waste Management.	<p>Loss of Coolant Accident Drainage. Minimises the effects of flooding of the area.</p> <p>Criteria for Discharge to ANSTO's Liquid Waste Treatment Plant are described.</p>	<p>What areas are effected by this flooding?</p> <p>Please explain the basis of these criteria for discharge to WOTD. Please describe compliance with the current <i>Code of Practice for the Disposal of Radioactive Wastes by the User</i>.</p>

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			<p>Response: The areas affected by LOCA flooding are a) Core cooling pump rooms –5.02/-5.03/-5.04, b) Process room –5.06, Piping connection room –5.05, Active workshop for pumps –5.40, Manifold of Reactor and Service Pool cooling system –5.201 and Resins area –5.07.</p> <p>The water in the area will drain into the LOCA pool existent in Level –7.</p> <p>The electrical switchboards room –5.01 is protected by placing it at a level 185 mm height above level –5 floor level. Also, other switchboards in Level –5 are placed 185 mm above the floor.</p> <p>In the worst case LOCA, a maximum flooding of 100 mm measured at floor level in the process room and a maximum of 150 mm in the pump area are predicted.</p> <p>The discharge criterion for non-active discharge components is stipulated by the Sydney Water Corporation as detailed in the Trade Waste Water Agreement with ANSTO. The radioactive component discharge criteria are site operation guidelines outlined in SD 5.7 (DRAFT)</p> <p>The Code of practice was ‘prepared to supplement the radiation control legislation implemented by the appropriate statutory authority’. ANSTO complies with the responsibilities of the user outlined in section 3.2 of the Code in conjunction with compliance with the discharge criteria set by the Sydney Water Corporation</p>
12.38.	12.4.4.10 Spent Resins	The 5th paragraph talks about the temporary reclassification of a 'white area' as a 'blue area' for the transfer operation.	Please identify the basis that will be used for such a temporary reclassification.

Checked / agreed:

ARPANSA Regulatory Assessment of the Replacement Reactor Construction Application

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			Response: The existing ANSTO Safety Directive "Radiological Classification of Areas and Access Control to Radiological Areas" describes the procedure for the review/temporary reclassification of radiological hazards undertaken by RPA/Health Physicist and Area Supervisor. This Safety Directive is applicable to the Reactor Facility.
12.39.	12.4.5.1 Intro to Liquid waste management	Paragraph 10 talks about the tritium concentration in the Heavy Water varying "in a saw tooth form after a year".	Why does the tritium concentration vary in this way? What is the period of the "saw tooth" form?
			Response: ARPANSA reference should be to Section 12.4.7, not Section 12.4.5.1. The variation in tritium concentration is described in the PSAR Section 12.4.7. In essence, the ventilation circuit will be closed and tritium in air removed by condensation and molecular sieves. This results in a sharp drop in concentration. The tritium concentration will gradually build up until the next circuit closure resulting in the saw-tooth emission pattern. The period of the sawtooth will be dependant on the rate of tritium buildup and the total moisture content in the air (humidity) to ensure collection of sufficient water during the process.
12.40.	12.4.6 Gaseous Waste Management	Areas where gaseous radioactive waste is generated during normal operation have absolute filtration and where applicable activated charcoal absorption filters in the ventilation extract from them.	Please identify where the safety analysis is done for the Gaseous Waste System to adequately deal with a worst case accident scenario within the reactor.

Checked / agreed:

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			<p>Response: The information provided in Chapter 12 of the PSAR deals with emissions under routine operations.</p> <p>Considerations regarding the worst scenario for gaseous waste system are included in Chapter 16, Section 16.19.4.</p>
12.41.	12.6.2	proven calculation techniques and codes ...	<p>Statements in 12.3.2 discussing the shielding code MERCURE refer to a table without a specific reference provided and also refer "from INVAP experience in previous design". Please provide the reference for data used, including all the information from 'INVAP experience in previous design'.</p>
			<p>Response: The Tables provide an indication of the thickness required to obtain the required dose rate. It is INVAP's experience that in the later design stages (during Detail Engineering) some pipes or ducts are re-routed through the shielding thus decreasing the "real" thickness of the shielding. By adopting a thicker shield, the shielding requirements are achieved without affecting the layout. This extra thickness added to the calculated value, together with the selection of conservative inputs (e.g. lower material densities than expected) are used to provide confidence that the shielding is adequate. If major penetrations or channels need to be accommodated in the shield body, "compensating materials" (e.g. lead) embedded in the shield body will be used.</p> <p>INVAP experience is built on the use of this code to calculate several shielding requirements in different projects. As part of the commissioning of these projects, the adequacy of the shielding was assessed. Projects include Research Reactors RAE and ETRR-2, ASEC-Q a Spent Fuel storage facility for Argentina's CANDU Nuclear Power Plant, among others.</p>

Checked / agreed: