IAEA
International Atomic Energy Agency

REPORT OF THE

PEER-REVIEW MISSION

ON THE COMMISSIONING & OPERATION OF THE OPAL RESEARCH REACTOR FOR ARPANSA

Sydney, Australia

28 February – 11 March 2005

PEER REVIEW MISSION

DEPARTMENT OF NUCLEAR SAFETY AND SECURITY

DIVISION OF NUCLEAR INSTALLATION SAFETY

INTERNATIONAL ATOMIC ENERGY AGENCY

ORIGINAL: ENGLISH
Distribution: Restricted

Mission date:	28 February – 11 March 2005
Location:	Sydney, Australia
Facility:	OPAL RESEARCH REACTOR
Organized by:	IAEA At the request of ARPANSA
Conducted by:	

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1. INTRODUCTION

1.1 BACKGROUND

In September 2004, the Australian Nuclear Science and Technology Organisation (ANSTO) submitted a Licence Application to the Chief Executive Officer of the Australian Radiation Protection and Nuclear Safety Agency (CEO of ARPANSA) seeking authorization for operation of the Replacement Research Reactor (RRR) under construction at the Lucas Heights Science and Technology Centre, Australia. In response to a request from the CEO of ARPANSA (22 October 2004), a peer review of the Licence Application was organized by the International Atomic Energy Agency (IAEA).

The Peer Review Team consisted of one IAEA staff member, T. Hargitai, and three external experts: Messrs. A. D’Arcy (South Africa), J.P. Boogaard (The Netherlands) and P. Gubel (Belgium). The peer-review mission was conducted at facilities of ANSTO and ARPANSA from 28 February to 11 March 2005.

A construction licence was issued by the CEO of ARPANSA on 2 April 2002, authorizing construction of the Replacement Research Reactor up to, but not including, loading of nuclear fuel into the reactor. These construction activities include post installation testing, pre-commissioning activities and cold commissioning tests. The main contractor is the Argentinean company INVAP SE. Several Australian companies are involved, mainly in fabricating components and civil engineering, as subcontractors to INVAP.

The facility, until recently, has been called the Replacement Research Reactor because it is intended to replace the existing HIFAR research reactor operated at the same site. In January 2005, the reactor was renamed as OPAL (Open Pool Australian Light water reactor). The OPAL research reactor is of a pool type with a rated power of 20MW. A Cold Neutron Source (CNS) in the reactor pool is considered to be an integral part of the facility although a separate safety report has been prepared for this source. The CNS will be commissioned together with the reactor facility.

In August 1998, the IAEA was invited by Environment Australia to review the Environmental Impact Statement (EIS) for the Replacement Research Reactor. The review report prepared by the IAEA for the EIS included several recommendations to be taken into consideration in the Preliminary Safety Analysis Report (PSAR) submitted in support of the application for the licence authorizing construction of the reactor. A second IAEA Expert Mission was conducted from 28 May to 8 June 2001 and examined the Preliminary Safety Analysis Report. A part of the present review constituted a follow up of issues raised in the review of the Preliminary Safety Analysis Report (see Appendix I).

The present Peer Review Mission concentrated on the identification and discussion of safety issues associated with the application for a licence to operate the reactor. A list of these issues, including any ensuing comments or recommendations, is given in Appendix II.

1.2 OBJECTIVES OF THE MISSION

The objectives of the mission were to:

- Advise ARPANSA on the adequacy of the proposed Operational Limits and Conditions (OLCs) and Conduct of Operations of the reactor.
- Advise ARPANSA on the adequacy of the commissioning program.
- Follow up on the activities of the licence applicant to address the recommendations raised by the previous Peer Review mission for the application for a licence to construct the reactor.

1.3 REVIEW SCOPE

The scope of the mission was as follows:

- Closure of issues raised by the previous mission.
- Review of the licence application including the pre-commissioning version of the Safety Analysis Report.
- Operational Limits and Conditions for operation (and commissioning).
- Operating procedures.
- Adequacy of the proposed Operating Organization.

1.4 BASIS AND REFERENCE FOR THE REVIEW

The review is based on the Licence Application, which was delivered on a CD to the IAEA and the external experts, only a few days before the start of the peer-review mission. Supporting documents were provided to the review team by ANSTO and INVAP during the mission. A list of these documents is given in Appendix V.

Discussions with counterpart specialists constituted a major part of the review basis. The interactions with the counterparts are summarized in the next section (Conduct of the Review).

The main reference for the review is the IAEA Safety Standards related to research reactor safety. These are:

- Safety Requirements for Research Reactors - NS-R-4 (working ID DS 272, supersedes Safety Series 35-S1 and S2)
- Safety Assessment of Research Reactors and Preparation of the Safety Analysis Report - Safety Series 35-G1
- Safety in the Utilisation and Modification of Research Reactors - Safety Series 35-G2
- Operational Limits and Conditions and Operating Procedures for Research Reactors – DS 261 (under preparation)
- Commissioning of Research Reactors – DS 259 (under preparation)
- Recruitment, Training and Qualification of Research Reactor Personnel – DS 325 (under preparation)

1.5 CONDUCT OF THE REVIEW

Subsequent to the agreement reached between the IAEA and ARPANSA on the Peer Review of the Licence Application, a preparatory meeting was arranged to decide on matters such as the scope of the review; number and profile of experts in the team; the review mission, venue and dates; draft agenda of the mission, action plan; and logistical matters. This preparatory meeting took place at the ARPANSA offices in Miranda from 30 November to 02 December 2004 and one IAEA staff member (T. Hargitai) participated.

In accordance with the action plan and draft agenda prepared during that meeting, the review process comprised the following steps:

- Transmittal of the Licence Application to the IAEA and the Peer Review experts in late January (it reached the team members a few days before the mission).
- Identification of issues by the Peer Review Team, and communication of the issues to and clarification with the ANSTO and INVAP counterparts.
- Discussion of the issues in working groups with counterparts and between Peer Review Team members at plenary meetings.
- Written response to issues provided to the Peer Review Team, by the counterparts, giving counterpart views of the issues and measures to address them.
- Discussion of the issues, the main conclusions and recommendations with ARPANSA.
- Preparation and submission of the draft report to ARPANSA on 11 March 2005.

The final Agenda of the mission and milestones in preparation and submission of the mission report to ARPANSA are provided in Appendix III.

1.6 WALK-DOWN OF THE FACILITY

On Monday afternoon (28 February) the team took the opportunity to enter the construction site of the reactor for a short walk-down of the facility under construction. During the walk-down the team visited all the accessible technological areas. Generally it could be stated that the civil construction work was almost finished, while the installation of systems, structures and components was approximately 80-85% completed.

The cooling systems, including the decay tanks and water treatment systems, were ready for testing. The auxiliary systems (diesel generators, compressors, chillers etc.) had been installed and some of these have gone through post installation tests. The instrumentation and control systems had been partly installed; the cabinets of the First and Second Reactor Protection Systems were in place; the control desk was in the control room awaiting installation. The ventilation systems had been almost entirely installed: the fans and filters including the piping were in place. The helium compressors and the deuterium tanks of the Cold Neutron Source had been installed while the cryogenic system (Cold Box) was being installed. The reactor pool liner was undergoing a cleaning process but the reactor internals including the control plate driving systems were entirely missing, mainly due to the delay in the construction of the Reflector Vessel.

Considering the short time that has passed from receiving the construction licence it could be stated that the contractor and its subcontractors have done a remarkable job and made enormous efforts to keep the deadlines.

There are only three comments worth mentioning:

- At this stage of the construction the inner area (say the containment) should be kept as a “clean area” to make cleanup of the systems easier;
- Around the pumps no means could be seen for avoiding contamination during pump maintenance, such as trays with connections to hot drains.
- On level -5m a lot of repairs could be seen on the floor. It was reported that cracks have appeared on the surface of the concrete of both the floor and sidewalls. The means of repair was elaborated, the cracks have been excavated to an appropriate depth and repaired.

1.7 STRUCTURE OF THE REPORT

The main body of the report comprises the introductory part and the Main Conclusions and Recommendations. The latter is a summary of the main findings as well as the most pertinent recommendations that are elaborated in more detail in Appendix II - Issues Identified.

Appendix I provides a follow-up and closure of the issues raised by the mission conducted in 2001 to peer review the application for a licence to construct the reactor.

Appendix II constitutes the main technical part of the report in which all issues identified by the Peer Review Team are discussed. During this review process 25 issues were identified. Each issue is firstly clarified by the Review Team stating why the team identified the matter as an issue. Then the views of the ANSTO and INVAP counterparts, and measures identified by them, in response to the issue, are given. Finally, conclusions of the Peer Review Team are given, stating whether the issue is considered to have been resolved or, otherwise, providing comments and recommendations which may assist in addressing the issue.

2. MAIN CONCLUSIONS AND RECOMMENDATIONS

Follow-up of the Issues Identified by the Previous Expert Mission

1. The Issues raised by the previous mission have been followed up. The actions taken by ANSTO to address the Issues were considered by the Peer Review Team to be satisfactory so the Review Team closed these Issues. Only one Issue has remained open; where the recommendation has been followed-up from hardware point of view but the change has not been reflected in the Safety Analysis Report.

Operational Limits and Conditions

2. The Open Pool Australian Light water reactor (OPAL) is a modern reactor whose design and sophisticated safety features are dedicated to reducing the probability of an accident due to a hardware failure. These safety features should be used during operation to the extent possible in order to minimize the probability of any release to the environment. It is the opinion of the Review Team that this could be better reflected in the Operational Limits and Conditions (OLC) by adaptation of the surveillance requirements in order to guarantee the availability of the safety features to the extent possible.

3. It is recognized that the surveillance requirements (SR) have been based on a Probabilistic Safety Analysis (PSA) and manufacturers recommendations, resulting in longer surveillance intervals compared to the practice in research reactors. However, it is recommended that the Surveillance Requirements be re-analysed paying special attention to the nature of a research reactor cycling through all four operational states at roughly monthly intervals, with more frequent changes taking place in the irradiation and experimental facilities.

4. The applicability of the surveillance requirements SR 3.0.3, by which it could be possible to extend the specified surveillance interval by 100 %, should be adapted to avoid an excessive extension of the surveillance interval without reporting a breach of the OLC to the Licensing Authority.

Conduct of Operations

5. It is recommended that the lines of authority, responsibilities and communications be defined clearly for all staff members involved in assuring safe operations. Special attention should be given to the lines of authority, responsibilities and communication of the reactor manager and the specified ANSTO officer in order to avoid any conflicts regarding safety.
   In order for the Reactor Manager to have full control of those activities that assure safe operation of the reactor, the Analysis and Engineering Groups should be integrated under the responsibility of the Reactor Manager.

6. The proposed minimum staff requirements identified in the OLCs are based on a job and tasks analyses for two limiting events. As described in dedicated IAEA documentation regarding OLCs, justification of the proposed minimum staffing requirements needed to assure safe operation of the facility should be based on all operational states (including transitions such as start-up), safety related operations regarding utilisation, incidents and accidents as well as emergency situations. All operational states should have been incorporated in the job and task analyses for the shift members in order to justify the proposed minimum staffing requirements.

Licensing Documentation

7. It is recommended that ARPANSA and ANSTO agree upon the licence application documentation, which still needs some internal assessment by ANSTO, and the time frame when the final versions will be submitted for additional or final review by ARPANSA. Upon completion of this process a final set of licensing documents should be defined on which the review will be based for the operating licence.
   A clear procedure should be put into place by ARPANSA for the submission of new versions of the licence application documentation.

APPENDIX I
FOLLOW-UP OF THE ISSUES IDENTIFIED BY THE PREVIOUS EXPERT MISSION

ISSUE NUMBER: 1

1. REVIEW AREA: PSAR Section 2.7 - Codes and Standards

2. ISSUE TITLE: Application of ASME Codes/Standards to Aluminium and Zircalloy Vessels and Piping

3. RECOMMENDATIONS:

R1. When attempting to pick and choose among various portions of codes to fit a particular application, care should be exercised to select only those portions such that a consistent set of assumptions is used throughout.

R2. The embrittlement of highly irradiated aluminium and Zircalloy materials is not covered by codes and standards. Potential changes in material properties should be carefully considered over the entire design life of those components that may be subjected to high neutron fluences.

4. FOLLOW-UP OF THE ISSUE:

R1. The aim of selecting standards for the construction of the various components of the facility was not to "pick and choose" but to select a logical and consistent set of codes that would provide an appropriate level of confidence in the integrity of the components. In accepting the recommendation, mixing of codes was minimized. In most cases only a single code was used.

R2. The change in material properties under irradiation has been acknowledged and has been considered in the design of the key components such as the reflector vessel. Under such conditions end-of-life material properties were used to check that the component behaves in a manner consistent with the design intention. In the specific case of the reflector vessel a wide surveillance programme has been developed in which samples of the original material will be irradiated and tested every five years. In the case of control blade replacement (10-12 years) the rod holding the control plate and getting higher fluence than the reflector vessel will be checked as well.

It should be noted that leaking of the reflector vessel is not a safety issue but a very important economical and ageing management issue.

The issue is considered to be closed.

ISSUE NUMBER: 2

1. REVIEW AREA: PSAR Section 3.2.6.2

2. ISSUE TITLE: Seismic Hazard

3. RECOMMENDATION:

R1. In order to demonstrate the conservative choice of seismic design basis parameters, an additional study should be undertaken by ANSTO for the seismic hazard assessment of the RRR site in line with the recommendations of the IAEA Safety Guide 50-SG-S1 (Rev. 1). The results of this study should be integrated into the PSAR. Alternatively, ANSTO may choose to demonstrate that the design basis adopted for the RRR is sufficiently conservative with respect to the most recent studies, in particular, those taking account of the uncertainties in an adequate manner (such as the Institute of Geological Nuclear Science - IGNS study in progress). In particular, the high frequency content of the calculated response spectra should be enveloped by the design response spectra both for horizontal and vertical directions.

4. FOLLOW-UP OF THE ISSUE:

R1. The seismic design-basis response spectrum has been modified so that it bounds both the original design-basis response spectrum and the revised IGNS response spectrum. As such the high frequency part is the same as the IGNS response spectrum anchored at 0.37g but very conservative methods were used in the assessment of these high frequency components.
In relation to the need for additional seismic investigations ANSTO retained paleoseismologists who undertook a defined site vicinity study in accordance of IAEA Safety Guide 50-SG-S1, Rev. 1. This study has been summarized in chapter 4 of the Safety Analysis Report submitted with the application for an operating licence.

The issue is considered to be closed.

ISSUE NUMBER: 3

1. REVIEW AREA: PSAR Section 4.3.3

2. ISSUE TITLE: External Hazards (other than Seismic)

3. RECOMMENDATION:

R1. The conservatism incorporated in the decision of not including a variety of potential human-induced events in the design basis, leading to an impact or blast type of load on the reactor building structure is not very clear especially because any possible adverse change in the present situation has not been considered. The grillage provides partial protection from only one type of aircraft crash but fails to serve as a protective envelope for blast or impact loads in general. It is recommended to review this decision on the basis of updated data and calculations as well as projections for the potential development in the area that may change the present situation unfavourably.

4. FOLLOW-UP OF THE ISSUE:

R1. The recommendation was included in the Assessment Report of ARPANSA (R102). Provisions of aircraft impact have been assessed and even an aircraft simulator-based experiment was performed showing that it was almost impossible to hit the facility with a big aircraft. Note that ANSTO has administrative control over the 1.6 km exclusion zone and, as such, will ensure that no development takes place within this zone that may change the present situation unfavourably. See also additional information on various threat scenarios included in the SAR.

The issue is considered to be closed.

ISSUE NUMBER: 4

1. REVIEW AREA: PSAR Section 4.4.3.3.2

2. ISSUE TITLE: Seismic Design of Reactor Building

3. RECOMMENDATIONS:

R1. The design basis response spectra and time histories should be used in accordance with IAEA Safety Standards and other applicable documents as well as internationally accepted practice suitable for nuclear facilities. In particular, loading combinations (with operational loads), combinations of directional components and the behaviour limits of materials should be selected with adequate conservatism. To the extent possible, the use of more than one standard for items interacting with each other should be avoided.

R2. The PSAR should clarify the use of standards and codes in design related calculations. The way in which different standards and codes have been used should be made easy to follow.

4. FOLLOW-UP OF THE ISSUE:

R1. The consolidated Seismic Evaluation Report includes clarification as to the use of appropriate codes and standards as well the conservatively selected loading combination and the combinations of directional components. The report was part of the approved seismic design.

R2. Has been addressed above under Issue Number 1.

The issue is considered to be closed.

ISSUE NUMBER: 5

1. **REVIEW AREA:** **PSAR Section 4.5.1.5.2.5 - Reactor Pool Leak Detection**
PSAR Section 4.5.1.5.2.2 - Penetrations to the Reactor Pool
PSAR Section 12.3.1.2 - Component Layout

2. **ISSUE TITLE:** **Replacement/Repair of Embedded Components**

3. **COMMENTS:**

C1. Flexible, spiral wound nuclear grade stainless steel tubing can be installed inside existing piping if a leak should develop in inaccessible sections of this piping.

C2. In a humid climate, generation of spurious leak detection signals by moisture condensation on the exterior surfaces of tanks could be an operational nuisance, but not a safety problem.

4. **FOLLOW-UP OF THE ISSUE:**

The Comments were noted.

The issue is considered to be closed.

ISSUE NUMBER: 6

1. **REVIEW AREA: PSAR Section 4.5.15**

2. **ISSUE TITLE: Seismic Design and Qualification of Systems and Components**

3. **RECOMMENDATIONS:**

R1. An enhanced seismic analysis should be made of the pool including all the internal structures, penetrations and possible sources of interactions. The analysis should use conservative assumptions in relation to support conditions and fluid-structure interaction issues. When components are assumed as isolated structures and analysed accordingly, justification should be provided for this assumption.

R2. All Seismic Class 1 items which are active or which have moving parts should be qualified by testing. These items include, but are not necessarily limited to, pumps, valves, relays, motor control centres and electrical cabinets (housing Seismic Class 1 items). Standard items that have qualification documents enveloping the associated floor response spectra need not be retested. Where applicable, earthquake experience data may be used to support seismic qualification. Testing should be performed using appropriate shaking tables and internationally accepted standards.

R3. The testing of the control plate drive mechanism (for FSS) is of particular importance because of the special significance of this system for seismic safety (the SSS is unable to shut the reactor down before the earthquake due to its long duration of the scram process). Adequate provisions should be made to test this system in order to ensure its safe performance in the event of an earthquake. Failing this, the reliability placed on the performance of this system in the safety analysis should be reviewed.

4. **FOLLOW-UP OF THE ISSUE:**

R1. The methodology for the detailed seismic analysis of the reactor pool including the treatment of the internal structures, penetrations and possible sources of interaction is discussed in the Consolidated Seismic Evaluation Report. The structural analysis, which included seismic issues as well, showed that the pool liner is qualified for sloshing effects. Moreover there is no component in the upper part that could be influenced by such effects.

R2. Although this recommendation was not addressed by ARPANSA, the requirements for the qualification of Seismic Class 1 components were identified and an appropriate qualification programme has been developed. On the qualified pipes, seismic analysis has been done; the flapper valves, cable trays, diesels and chillers are seismically qualified. Generally can be stated that Safety Category 1 components are qualified, while Safety Category 2 components are either qualified or appropriately tested before installation.

R3. Successful seismic testing has been performed on a full size control plate drive at INVAP site and viewed by ARPANSA.

The issue is considered to be closed.

ISSUE NUMBER: 7

1. REVIEW AREA: PSAR Section 5.2.6.5 - Reactor

2. ISSUE TITLE: Mechanical Stresses in the Reflector Vessel

3. RECOMMENDATION:

R1. During the final design phase the potential loads considered should be reviewed to assure that all credible loads are included and that they are properly combined in the analysis when appropriate and that sufficient safety margin exists.

4. FOLLOW-UP OF THE ISSUE:

R1. The review was performed and the final detailed analysis was submitted to ARPANSA and approved. The analysis included all credible loads and their proper combinations with proper safety margins. The loading cases included, but are not limited to, normal operational loads, seismic, heat loads, loads due to Zircalloy growth, loads applied during maintenance operations etc. See the request for approval (RFA) of the CEO of ARPANSA for manufacture of the Reflector Vessel.

The issue is considered to be closed.

ISSUE NUMBER: 8

1. REVIEW AREA: PSAR Section 5.6.2 - Neutron Reflector
PSAR Chapter 19 - Decommissioning
PSAR Chapter 20 - Emergency Planning and Preparedness

2. ISSUE TITLE: Tritium

3. RECOMMENDATION:

R1. It is recommended that this issue is addressed more explicitly in future documents especially with regard to maintenance and decommission, e.g. the radiological consequences due to tritium in the reflector system, particularly for personnel performing maintenance. The Emergency Planning and Preparedness should include special precautions to be taken when working with tritium-contaminated components.

4. FOLLOW-UP OF THE ISSUE:

R1. ARPANSA has not addressed this issue in the Assessment Report. Chapter 19 of the Safety Analysis Report deals with the tritium hazard during decommissioning, while Chapter 6 and 7 deal with the Engineering Safety Features to avoid tritium release and contamination while Chapter 16 - Safety Analysis, discusses the consequences of a release.

A response has been previously provided and contained in the IAEA Mission Report on the PSAR. The SAR contains effectively the same information.

Specific documentation associated with the maintenance of the Reflector Cooling and Purification System (i.e. the RC&PS – Maintenance Manual and a specific radiation protection procedure applicable to activities associated with RC&PS) is currently being prepared and will be subject to formal ANSTO review, verification and acceptance in accordance with project procedures. This documentation will explicitly address procedures and techniques that minimise the potential doses to operating personnel, including any special precautions to be taken when working with tritium-contaminated components.

With respect to the Emergency Planning and Preparedness, again the detailed procedures are currently being prepared and will be subject to formal ANSTO review, verification and acceptance in accordance with project Quality Management System.

It should be noted that the estimated annual stack discharge of tritium from the OPAL reactor is two orders of magnitude less than that for the current HIFAR operations (approximately 37 GBq per year versus about 4.32 TBq per year). In addition, it should also be noted that ANSTO has over 40 years operational experience in the safe handling of heavy water and tritium contaminated reactor plant.

The issue is considered to be closed.

ISSUE NUMBER: 9

1. REVIEW AREA: PSAR Section 5.8 - Thermal and Hydraulic Design

2. ISSUE TITLE: Validation of Thermal-Hydraulic Codes Used for the Reactor Core

3. COMMENT AND RECOMMENDATION:

C1. Direct measurement of the fuel cladding temperature with an instrumented fuel element is a good means for checking the validity of the thermal – hydraulic calculations without performing numerous sensitivity calculations for the various input data, has been reliably done before. This experimental validation is useful for the evaluation of the transients in which the cooling by natural circulation (an important safety issue) and for which the validity of the various calculation codes are questionable.

R1. Whether or not such a validation scheme is adopted, there should be some detailed justification concerning the validation of codes used for the safety analyses presented for the core thermal – hydraulics, as well as for the acceptance criteria used in results where channel boiling is evident.

4. FOLLOW-UP OF THE ISSUE:

C1. The comment has been noted by ANSTO. The feasibility of such an experimental validation with an instrumented fuel assembly has been examined. ANSTO notes that there are practical difficulties with measuring clad temperatures accurately, and also in interpreting the results. The devices used for physical measurement of cladding temperatures in the fuel plates perturb the physical and geometric properties of the fuel elements, which produce a large inaccuracy in the measured values. During Stage A commissioning instrumented dummy fuel assemblies will be used for flow distribution and pressure measurements.

R1. The issue was addressed in the Assessment Report of ARPANSA (R42). Licence Condition 4.10 requires the Licensee to develop and undertake a programme of work during construction validating the computational models used to demonstrate safety. (R 44, Approval Process)

The issue is considered to be closed.

ISSUE NUMBER: 10

1. REVIEW AREA: PSAR Section 5.9.7 - Materials Surveillance Plan

2. ISSUE TITLE: Materials Surveillance Program

3. COMMENT:

C1. Consider irradiating extra coupons in case a metallurgical problem is discovered and more frequent sampling becomes necessary.

4. FOLLOW-UP OF THE ISSUE:

C1. A material surveillance programme has been developed.

The issue is considered to be closed.

ISSUE NUMBER: 11

1. REVIEW AREA: PSAR Chapter 8 - Instrumentation and Control

2. ISSUE TITLE: Independence of Reactor Protection Systems

3. COMMENT:

C1. If the construction of the systems are carried out this way (and in accordance with IAEA guides etc. mentioned in Section 8.1.4.2) the independence of the reactor protection system from the monitoring systems is assured.
The separation between the two reactor protection systems is also assured in the way it has been described.

The issue was closed during the previous mission.

ISSUE NUMBER: 12

1. REVIEW AREA: PSAR Section 11.5.3, Cold Neutron Source

2. ISSUE TITLE: CNS Safety Analysis Assumptions

3. RECOMMENDATIONS:

R1. Even though the reactor may not be harmed by a detonation within the CNS, such an event would undoubtedly have negative consequences for the future of the neutron beam experiments. Every precaution should be taken to preclude this event from happening. Therefore, it is of outmost importance to guarantee that there are no “sneak” paths for accumulation and mixing of oxygen with the deuterium.

R2. The Cold Neutron Source forms a part of the facility; however a separate safety analysis is being prepared for the experimental utilisation of the CNS. It would be prudent to also have an in-depth multi-discipline safety review of the interfaces and operational interactions with the reactor. This review should address, for example, issues of operator training and awareness, human factors engineering, maintainability and radiological protection. This will provide additional assurance that all safety questions have been addressed regarding the interaction of this complex facility with the reactor.

4. FOLLOW-UP OF THE ISSUE:

R1. A detailed analysis has been performed which proved that a pressure wave couldn’t cause deformation blocking the cooling channels or damage the rigs. The deuterium is safely blanketed by helium that is monitored for the presence of deuterium.

R2. The Safety Analysis Report for the CNS was been made available to the Peer Review Team.

The issue is considered to be closed.

ISSUE NUMBER: 13

1. **REVIEW AREA:** **PSAR Section 8.4 - Instrumentation and Control**
   PSAR Section 16.12 - Safety Analysis

2. **ISSUE TITLE:** **Break of Reflector Drain Line in Reactor Pool**

3. **RECOMMENDATION:**

R1. While the probability of this event may be low, since a common postulated event could defeat redundant shutdown systems the scenario, along with the interactions of other structures, systems, and components, should be analysed and discussed in the PSAR and PSA.

4. **FOLLOW-UP OF THE ISSUE:**

R1. The issue was not addressed in the Assessment Report of ARPANSA. This sequence requires both the failure of Control Plates to drop into the core on demand and the failure of the Second Shutdown System discharge piping within the pool such that the Reflector vessel fails to drain on demand. The only common event postulated that could effect both the First Shutdown System and the Second Shutdown System is the seismic event. Note that this is the only event that simultaneously trips both systems. The pressure in the Reflector Vessel is lower than that of the pool. Light water breaking into the Reflector Vessel provides enough negative reactivity to shut down the reactor.

The issue is considered to be closed.

ISSUE NUMBER: 14

1. REVIEW AREA: PSAR Chapter16 - Safety Analysis

2. ISSUE TITLE: Treatment of the Onset of Nucleate Boiling (ONB) condition

3. RECOMMENDATIONS:

R1. It is recommended that the statement in section 16.8.7.3.3 that “the presence of nucleate boiling does not imply any damage to the clad . . .” in particular, and similar statements concluding the discussions of other safety analysis results throughout chapter 16, be carefully justified or quantified.

R2. It is recommended that the temperature for ONB be dynamically presented in all the relevant transient graphics so that the fuel clad temperature can be compared with its current value at any given time during the transient evolution rather than with its initial value.

4. FOLLOW-UP OF THE ISSUE:

R1. The issue was not addressed in the Assessment Report of ARPANSA. In any transient conditions, fuel damage will occur only if sufficiently high temperatures are reached. As with similar temperature induced damage, ONB is clearly a necessary but not sufficient condition for damage. It may be noted that the critical heat flux (CHF) or redistribution must be regarded as sufficient to cause overheating and hence fuel damage, if initiated during quasi-steady operation.
A detailed study has been performed of the transient considered.

R2. The CHF as a function of time has been plotted in various detailed transient analysis reports prepared in support of the Safety Analysis Report and submitted to ARPANSA.

The issue is considered to be closed.

ISSUE NUMBER: 15

1. REVIEW AREA: Chapter16 - Safety Analysis

2. ISSUE TITLE: Engineering Hot Spot Factors

3. RECOMMENDATION:

R1. It is recommended that a more direct reference to sections 5.8.5 and 5.8.7.2 and Table 5.8/9 be made, with some elaboration.

4. FOLLOW-UP OF THE ISSUE:

R1. The issue has been superseded by the overall evolution of the safety analyses. However, a fuller discussion of introducing conservatism to accident analyses will be incorporated into the Safety Analysis Report.

The issue is considered to be closed.

ISSUE NUMBER: 16

1. REVIEW AREA: PSAR Section 16.8.4.1 - Cold Water Injection

2. ISSUE TITLE: Bounding of Cold Water Injection by a Control Plate Withdrawal

3. RECOMMENDATION:

R1. It is recommended that the typographical correction and the justification along the lines of the last sentence of the Counterpart Views\an Measures be provided to support the statement that the initiating event is bounded by the Design Basis Initiating Event (DBIE) of Control Plate Withdrawal

4. FOLLOW-UP OF THE ISSUE:

R1. The issue was addressed in the Assessment Report of ARPANSA (R64). The design incorporates an interlock that prevents the start up of a stand-by Primary Cooling System (PCS) pump whilst the other two PCS pumps are operating. Thus, the potential of cold-water injection during power operation is minimized. The Safety Analysis Report should be clear that the start-up of the PCS pumps would result in a reactivity insertion of less than 400pcm and that this insertion is bounded by the withdrawal of a control plate during reactor start-up. The typographical error still exists in the Safety Analysis Report that will be corrected in the next revision of the report to be prepared following the completion of commissioning.

In addition, a Safety Analysis Report Errata document is currently being prepared identifying and correcting typographic and consistency errors that have been identified in the SAR subsequent to its formal issue as part of the Application. The above will be identified in this Errata document.

The issue is considered to be closed from hardware point of view.

ISSUE NUMBER: 17

1. REVIEW AREA: PSAR Section 16.9.2.2.3 - Core Blockage

2. ISSUE TITLE: Channel Blockage due to Fuel Assembly Damage

3. RECOMMENDATION:

R1. It is recommended that the addition of the operator "mishap" scenario be analysed as a possible event and that a passage similar to the counterpart view be included in the discussion of the event.

4. FOLLOW-UP OF THE ISSUE:

R1. The issue is covered by Safety Analysis Report Section 16.9.22.3.

The issue is considered to be closed.

ISSUE NUMBER: 18

1. REVIEW AREA: PSAR Section 16.14 - High Energy Piping Systems

2. ISSUE TITLE: Hazards Associated with High Energy Piping Systems

3. RECOMMENDATION:

R1. It is recommended that the design intention should be made clear rather than dismissing the matter as is done.

4. FOLLOW-UP OF THE ISSUE:

R1. The identification and analysis of internal hazards has been performed during the Detailed Engineering Phase. Safety Analysis Report Chapter 16, Section 16.14 has been revised according to the recommendation.

The issue is considered to be closed.

ISSUE NUMBER: 19

1. REVIEW AREA: PSAR Section 16.14.1 - Internal Fire or Explosion

2. ISSUE TITLE: Consequence Analysis for Internal Fire or Explosion

3. RECOMMENDATION:

R1. It is recommended to carry out a Fire-PSA as intended.

4. FOLLOW-UP OF THE ISSUE:

R1. The issue was addressed in the Assessment Report of ARPANSA (R22, R63). Due to legal obligations the Fire-PSA has been performed in connection of the design and installation of the fire protection systems. The Fire PSA has been included in the new version of the Safety Analysis Report.

The issue is considered to be closed.

ISSUE NUMBER: 20

1. REVIEW AREA: Section 16.14.3.3 - Loss of Communications Capabilities

2. ISSUE TITLE: Impact of Loss of Communications on the Safety of the Reactor

3. RECOMMENDATION:

R1. It is recommended that such a reference and clarification be provided.

4. FOLLOW-UP OF THE ISSUE:

R1. In Chapter 10 of the Safety Analysis Report a reference to the description of the communications systems together with the clarification has been incorporated. A Public Alarm System within the Post Accident Monitoring System (PAM) has been installed.

The issue is considered to be closed.

ISSUE NUMBER: 21

1. REVIEW AREA: Chapter 18

2. ISSUE TITLE: QA on Design Calculations

3. RECOMMENDATION:

R.1. More details should be provided on practical QA provisions including the eventual non-conformances treatment, the indication of important hold points during design, construction and commissioning activities. The management of the interfaces between the different parties should be clarified.

4. FOLLOW-UP OF THE ISSUE:

R1. Licence Condition 4.6.1 requires that the Licence Holder shall provide information establishing that each safety item has been constructed under a certified Quality Assurance programme.

The issue is considered to be closed.

ISSUE NUMBER: 22

1. REVIEW AREA: PSAR Appendix A

2. ISSUE TITLE: Probabilistic Safety Assessment (PSA)

3. RECOMMENDATIONS:

R1. PSA studies should continue to be updated in order to take account of the detailed technical design of the reactor, eventual modifications to this design and any experience feedback from incidents, which have occurred in similar installations.

R2. The seismic PSA should be checked when the hazard curve is finalised. Results should be compared with those of nuclear installations in similar seismotectonic regions of the world.

4. FOLLOW-UP OF THE ISSUE:

R1. PSA has been updated, independent assessment on common mode failures has been performed. The results obtained are similar to the previous results. ANSTO is taking over the calculations in order to have a "living PSA".

R2. See Issue Number 2.
The issue is considered to be closed.

APPENDIX II – ISSUES IDENTIFIED BY THE PEER-REVIEW TEAM

In the course of the review of the Safety Analysis Report and other documents the Review Team identified 25 specific issues which it considered in greater detail. The new issues continue the numbering of the issues raised by the previous mission and presented on the following pages.

ISSUE NUMBER: 23

1. REVIEW AREA: Stage A Commissioning Specific Plan
Section 3: Prerequisites

2. ISSUE TITLE: Ability to Record Data During Stage A Commissioning

3. ISSUE CLARIFICATION:

The listed prerequisites should include the availability of a Data Log system for recording of important parameters that are measured during the tests (e.g. in the First Reactor Protection System (FRPS), Second Reactor Protection System (SRPS), reactor, reflector and pool cooling systems and water purification systems, ventilation and containment parameters, etc).

4. COUNTERPART VIEWS AND MEASURES:

Each of the procedures to be executed as part of the Stage A Commissioning will state full details of the information to be collected as part of, and during the test. This information will be recorded in the Test Records. The test records are submitted to the Commissioning Manager for analysis and assessment on the approval of the test results.

Any abnormal system behaviour or event will be duly recorded in the log book in the Main Control Room and communicated to the Commissioning Team.

5. COMMENT:

C1. The explanation given by the counterpart appears to miss the point of the Issue. It is understood, and indeed normal (and expected) practice, that all the important parameters will be recorded during the numerous tests conducted during Stage A commissioning. From the explanation it appears as if the data will be recorded by hand.

The Issue was raised about the availability, during plant commissioning, of the electronic data acquisition system (maybe the operable Reactor Control and Monitoring System) that would later archive parameters throughout the plant during normal operation. Apart from the obvious benefit that using an electronic data acquisition system would have on manipulating/processing/reporting the plant commissioning results, it could also prove invaluable in resolving results that deviate from expectation or specification. Furthermore, it would present an excellent opportunity to thoroughly test, calibrate and commission the data acquisition system itself (which did not form part of the draft Stage A Commissioning Plan viewed by the Review Team).

The availability or not of an electronic data acquisition system is not a matter of safety, but rather one of convenience and the choice of the Commissioning Team.

The Review Team, having made the point, therefore considers the Issue to be closed.

ISSUE NUMBER: 24

1. REVIEW AREA: Stage A Commissioning Specific Plan, Section 4.3.9 – Normal Electrical Power Supply System Test:

2. ISSUE TITLE: Ordering of the Normal Power System Test

3. ISSUE CLARIFICATION:

This test of the normal electric power supplies is listed in the sequence of tests in the document after numerous tests that use the power supply – in particular those of the large consumers such as primary and secondary pumps, etc. The opinion is that this test should be carried out before tests are conducted on the power supply consumers.

4. COUNTERPART VIEWS AND MEASURES:

As stated in section 4 paragraph 4 of Stage A Commissioning Specific Plan "*The following list is not intended to represent the order in execution sequence of the tests*". A detailed schedule for the tests in Stage A Commissioning is being prepared as indicated in Section 5 of the plan. This detailed schedule, together with the actual commissioning procedures and instructions, will be subject to formal ANSTO review, verification and acceptance in accordance with project procedures.

5. COMMENT:

C1. Having missed the referred statement, the Review Team assumed that the sequence of tests in the document represented the sequence in which they will be carried out. However, the Team accepts that, in the detailed schedule, the sequencing will be logical.

The Issue is therefore closed.

ISSUE NUMBER: 25

1. REVIEW AREA: Sub-stage B1 Commissioning Specific Plan
Section 3: Prerequisites
Section 4.1.2: First Reactor Core Fuel Loading
Section 4.1.3: Approach to Criticality Procedure

2. ISSUE TITLE: Approach to First Criticality

3. ISSUE CLARIFICATION:

Item g) in section 3 poses the prerequisite for sub-stage B1 commissioning that the "core loading sequence and intermediate cores be defined".

Step 3 of the procedure for the first reactor core loading in 4.1.2.2 states that "the first core has two fuel assemblies less than the first critical core, as required by reference 5".

Step 2 in the procedure for the approach to criticality in 4.1.3.2 states that loading of fuel assemblies (starting with the sub-critical core of 4.1.2) "will start with those that are anticipated to have the greatest reactivity worth".

Appreciating the fact that this document is still in draft form and will probably be subjected to further review and revision by the Operating Organisation, the issues with these assertions are:

- There is no reference 5 in the list of references in section 1.4,
- There is no prerequisite for the status of the reflector tank (filled with D_2O or drained). What does it mean that the Second Shutdown System is in operable condition?
- The process described in these sections provides no information on the loading positions for the first core nor of the selection of fuel assembly masses in given positions in that and subsequent cores – at least in terms of concept if not in actuality.

In the opinion of the Review Team, the described approach, using the mass extrapolation method, could be flawed because the reflector cannot be arranged on the fuel boundary for each successive core. The result is that two parameters change as each additional fuel assembly is added – the total uranium-235 mass and the mean distance to the reflector.

4. COUNTERPART VIEWS AND MEASURES:

In relation to the first bullet point, this is a typographic error. The correct reference is Reference 4.

In relation to the second bullet point, the status of the reflector tank is that it is filled with heavy water. As indicated in section 1.3 of the Sub-Stage B1 Commissioning Specific Plan, 7th paragraph, "*the First and the Second Shutdown System will be operational during the approach to criticality*". In order for the Second Shutdown System to be operational, the reflector vessel will be full of heavy water (armed state), so that the drainage of the heavy water will serve to shut down the reactor if required.

In relation to the third bullet point, the following criteria have been considered in defining and planning the fuel loading sequence in the core during commissioning:

1) Compliance with the Argentine Regulatory Body regulation on commissioning of research reactors (ARN 4.8.2) that requires that: the first subcritical core must have at least two fuel assemblies less than the predicted first critical core configuration. In fact, in the strategy designed by INVAP, the first subcritical core has 4 fuel assemblies less than the critical core as predicted by calculations.
2) One external neutron source is loaded inside the core starting at the first subcritical core configuration that is assembled to ensure the fission counters are able to provide readings even in the less reactive Control Plate conditions (all Control Plates in the core).
3) The external neutron source location is chosen in order to ensure that it is the core until the last Fuel Assembly is loaded into the core, and that there is no direct line of sight between the external neutron source and the fission counters.
4) The fission counters positions are chosen so as to ensure the neutron fluxes in the detectors are appropriate to allow the fission counters to follow the core status and neutron flux evolution during the fuel loading sequences starting with the first subcritical core.
5) The selection of which fuel assembly to load next at each step is taken so as to have even reactivity additions in each step, and at the same time maintain a compact core array (preventing leaving void positions in the middle of the core). The criterion on having even reactivity addition at each step serves to prevent large changes in the counting rates of the fission counters.
6) The Fuel Assemblies used during commissioning include standard Fuel Assemblies (each with 484g U-235) and two special types of Fuel Assemblies (having 212g U-235 and 383g U-235). The reactor fuel management strategy has been designed in such a way that starting from the first core reshuffling, only standard type of Fuel Assemblies are loaded into the core.
7) Ensure that in every case, the shutdown margin is sufficient to compensate for the positive reactivity that would be added by the latest fuel assembly loaded into the core.
8) The uranium mass and burnable poison contents of the two special types of commissioning fuel have been chosen in such a manner that the 16 Fuel Assembly core will have all the parameters in compliance with the core nuclear design criteria as indicated in Safety Analysis Report section 5.7.4. This includes for example: Shutdown margins of the shutdown systems, Power Peaking Factor limit, safety reactivity factor, feedback coefficients, reactivity insertion rates, among others. In this way, the first 16 Fuel Assembly core can be operated at full power without further restrictions.
9) The distribution of fuel masses in the core configuration with 16 Fuel Assemblies was selected so as to allow a smooth transition into the intermediate cores up to the equilibrium core.

Detailed information is contained in the commissioning test procedures and the calculation reports that provide support to these procedures. For example, a Detailed Engineering design report details the fuel management strategy for first and intermediate U_3Si_2 cores with hafnium control plates. It provides details on the fuel assembly masses for the first core (containing sixteen Fuel Assemblies)

and all the subsequent cores up to reaching equilibrium core, specifically in relation to the following:

a. Fuel reshuffling strategy.
b. Control plate movement strategy and its effect on Power Peaking Factor.
c. Burnup.
d. Verification of neutronic design criteria: shutdown margins for First Shutdown System and Second Shutdown System, Power Peaking Factor, End of Cycle reactivity, actuation times of shutdown systems, reactivity insertion rates.

Additional analyses are currently being performed as required using the MCNP computer code and refined calculation models in support of the commissioning planning.

In relation to the fourth point, the core neutronic calculation models being used in support of the planning of the commissioning and the approach to criticality explicitly take into account the effect of the addition of the fuel assemblies to the core and the simultaneous change in the mean distance to the reflector.

MCNP calculations have been carried out to study and plan the sequence of steps for the fuel loading into the core and the approach to criticality. Based on these studies, the order of fuel loading into the core and the masses involved have been determined and the conservative analyses indicate that the critical mass approximation will start with the nine standard Fuel Assembly core. These calculations also provided information supporting the determination of the best locations for the fission counter detectors and the external neutron source during the approach to criticality.

Taking into account the calculated fission counters counting rate with all Control Plates withdrawn, the experimental procedure for the critical mass approximation has been simulated. The following table provides information on the results of the detailed core calculations which were carried out to support the planning of the approach to criticality. As shown, the estimated critical mass by the Fission Counter with the smallest prediction is always lower than the Critical Mass.

Initial Core		Estimated Critical Mass [g U235]			Critical Mass [g U235]	Mass of next FA to be loaded [g U235]
FA	Mass [g U235]	Fission Counter Nr1	Fission Counter Nr2	Fission Counter Nr3		
10	3187	4556	5556	**3697**	4480	383
11	3570	4779	**4067**	4650	4480	212
12	3782	4140	4199	**4001**	4480	484
13	4266	4462	4400	**4438**	4480	212

The commissioning procedures require that the personnel carrying out the approach to criticality act on the readings of the fission counter that at each step provides the smallest predicted critical mass. The first critical core based on the "as designed" Uranium loads of the Fuel Assemblies is predicted to have 14 Fuel Assemblies and 4480 g U235. All the critical masses estimations are conservative.

These calculations will be updated prior to commissioning and once the "as manufactured" uranium loads of the Fuel Assemblies are available. During commissioning between the initial nine Fuel Assembly core configuration and the 16 Fuel Assembly core configuration, one Fuel Assembly will be added at a time, with measurement of normalised counting rates at intermediate positions of the control plates.

5. COMMENT AND RECOMMENDATIONS:

C1. The counterpart's comprehensive response to this issue is appreciated. Presumably the typographical error will be corrected before the Sub-stage B1 Commissioning Specific Plan is issued.

R1. Since the Issue was raised about the absence of sufficient information in the Sub-stage B1 Commissioning Specific Plan to give the reader a sense for the processes, constraints and philosophy to be adopted in the approach to criticality, it is recommended that a summary of the above responses on the second and third points be included in that document – just the essence, not the detail.

R2. A full MCNP calculation of each sub-critical core (with explicit modelling of the light water geometry between the fuel and the heavy water reflector), and indeed of each core from the first critical core to the first full core in Sub-stage B2, would be an acceptable approach, provided that the experimental results of each core are fully understood and met in relation to the calculated results before the next Fuel Assembly is loaded.

This is, however, far from what is to be understood from the "mass extrapolation method" mentioned in the commissioning plan document. It is recommended that, as in R1 above, the essence of what is explained in the counterpart's response be given in the Specific Plan.

ISSUE NUMBER: 26

1. REVIEW AREA: Sub-stage B2 Commissioning Specific Plan
Section 4.1: First Full Core Configuration

2. ISSUE TITLE: Approach to Full Core Configuration

3. ISSUE CLARIFICATION:

Similar to Issue 25 Issue Clarification, second, third and fourth bullets:

- No prerequisite for the status of the reflector tank is given.
- No mention is made of the constraints/precautions/philosophy affecting the selection of uranium-235 mass and positioning of fuel assemblies added to the first critical core when approaching the first full core configuration.
- Estimation of the control plate positions for criticality should take account of both the uranium-235 mass increase and the reduction in distance to the reflector with each successive repeat of step b).

4. COUNTERPART VIEWS AND MEASURES:

In relation to the first bullet point, the operability requirement for the second shutdown system Commissioning Sub Stage B2 is the same as for Commissioning Sub-Stage B1, as described in our response to Issue 25.

In relation to the second bullet point, the constraints/precautions/philosophy identified by the Review Team are included in the calculation and assessment reports prepared in support of commissioning. It should also be noted that, in every case, the shutdown margin is sufficient to compensate the positive reactivity that would be added by the latest Fuel Assembly loaded into the core. For example, for the 15 Fuel Assembly core, there is a shutdown margin of some 20000 pcm, whilst the sixteenth fuel assembly will have 4500 pcm. For the cores featuring between 9 and 16 Fuel Assemblies detailed calculations are being prepared to predict the neutronic parameters of the intermediate cores (see also comments to Issue 25).

In relation to the third bullet point, we agree. The calculations undertaken to plan and analyse the sequence of steps during the approach to criticality take account of both the U-235 mass increase and the reduction in distance to the reflector with each successive fuel assembly added to the core. The calculations are carried out using the MCNP code, which takes into account these two effects on the estimation of the control plate positions for criticality (see also comments to Issue 25).

5. RECOMMENDATION:

R1. It is recommended that a similar explanation to that recommended for the Sub-Stage B1 Commissioning Specific Plan (see Comments and Recommendations for Issue 25) be incorporated into the Sub-Stage B2 Plan in relation to the transition from the first critical core to the first full core. This will greatly assist the reader to obtain a proper sense for the evolution of the transition.

ISSUE NUMBER: 27

1. REVIEW AREA: Sub-stage B2 Commissioning Specific Plan
Sections 4.13 and 4.14: Power Calibration of Detectors

2. ISSUE TITLE: Power Calibration of the Wide Range Neutron Detectors and the Compensated Ionisation Chambers

3. ISSUE CLARIFICATION:

- Probably the Wide Range Neutron Channels are meant. The Operational Limits and Conditions (OLC) Bases document explains that neutron detectors cannot be calibrated and their calibration has been specifically excluded from the OLC.
- This calibration should surely be carried out earlier in the B2 commissioning programme than indicated by the sequence of tests in the document, e.g. before tests to measure kinetic parameters (4.11), or tests to measure coefficients at different powers (e.g. 4.9, 4.10 and 4.12), since the accuracy of the results would depend on the accuracy of the power measured by these channels.

4. COUNTERPART VIEWS AND MEASURES:

The OLC and OLC Bases documents are applicable to the operation of the facility once the commissioning has been completed. The calibration of the nucleonic instrumentation referred in Stage B2 Commissioning Specific Plan consists of experimentally obtaining the correspondence between the detector signal (counts/s and/or mA) against the absolute neutron flux measurements performed using gold wire detectors and supported with correlations obtained from MCNP core models (please refer to explanation on methodology in section 4.5.2. of the Stage B2 Plan).

Also note that the order in which the different tests are listed in section 4 of the Sub-Stage B2 Commissioning Specific Plan is not the order in which they will be necessarily be executed. Please see comment in section 5 Sub-Stage B2 Commissioning Schedule.

5. COMMENTS:

C1. The counterpart misunderstood the first bullet of the Issue. The Issue was raised on the calibration of the neutron detectors (see Surveillance Requirement (SR) 3.3.6.2 in the Bases of the OLC) and not the entire channel. Common practice is to perform insulation measurements on the detectors including the cables and after that calibrate the channel.

C2. The Review Team assumed that the sequence of tests in the document represented the sequence in which they will be carried out. The Review Team accepts that in the detailed schedule the sequencing will be logical.

The Review Team, having made the point, therefore considers the Issue to be closed.

ISSUE NUMBER: 28

1. REVIEW AREA: Sub-stage B2 Commissioning Specific Plan

2. ISSUE TITLE: General Comments on Sub-stage B2 Document

3. ISSUE CLARIFICATION:

General comments on the Sub-Stage B2 Commissioning Plan document are:

- Section 4.10.2, point 2: Investigation of the feasibility to suddenly remove the Aluminium plates to simulate void feedback should consider not only the obvious danger of a short period/power excursion but also the danger of fuel clad damage due to seizing of Aluminium on Aluminium,
- Section 4.18: Shielding Measurements and section 4.21: Radiation Area Monitoring – also to be measured are the following:
  - Reactor operations floor and pool surface.
  - Primary piping routes, both embedded as well as in areas of personnel access during operation.
  - Pump and heat exchanger rooms.

  It is possible that these are included in meaning of the term "various points" in 4.21.1, but this is vague.
- Several references to "Power Mode" operations in the document should be changed to "Power State" to be consistent with the definition of the operational states.

4. COUNTERPART VIEWS AND MEASURES:

In relation to the first point, it is now anticipated that the reactivity worth measurement will be performed by removing the plates with the reactor shutdown (as proposed in the first paragraph of section 4.10.2). The specific test procedure covering void feedback coefficient evaluation will provide full details on the method to be used. It should be noted that the aluminium plates edges are manufactured not be sharp. The dimensions of the aluminium plates are such that they fit easily into the coolant channels of the fuel assemblies.

In relation to the second bullet point, all these points will be included in the detailed test procedures that are currently being prepared.

With reference to the third bullet point, this is a typographic error that will be corrected in the next revision of these plans.

5. COMMENT:

C1. The counterpart's response to this issue is appreciated.

The Review Team, having made the point, therefore considers the Issue to be closed.

ISSUE NUMBER: 29

1. REVIEW AREA: Stage C Commissioning Specific Plan
Section 4.1.6: Loss of Normal Power Supply Test

2. ISSUE TITLE: Synchronising the Reactor Trip with the Loss of Power

3. ISSUE CLARIFICATION:

From the description of the methodology for the Loss of Normal Power Test, points 1 to 3 in 4.1.6.2, it appears that the reactor will be tripped before the simulated loss of power. The behaviour of the reactor and plant in this case will differ from their behaviour when a loss of normal power occurs first, leading to a reactor trip. The test objective is the latter.

Furthermore, the list of recorded parameters in point 5 of section 4.1.6.2 does not include the requirement for continuously recording normal and emergency power supply voltages and the delay of the turnover. This information should be recorded to assist in the assessment of the transition from normal to emergency power for essential systems.

4. COUNTERPART VIEWS AND MEASURES:

The comment in relation to when the loss of normal power test should be performed is noted and will be taken into consideration.

The information referred to will be recorded as part of the detailed procedure for this test.

5. COMMENT:

C1. The counterpart's response to this issue is appreciated.

The Review Team, having made the point, therefore considers the Issue to be closed.

ISSUE NUMBER: 30

1. REVIEW AREA: SAR Chapter 13 - Conduct of Operations

2. ISSUE TITLE: General Comments on the SAR Chapter 13

3. ISSUE CLARIFICATION:

1. Sections 13.2.3.3 to 13.2.3.7 provide an outline of the duties of the various "Leaders" listed in the Organisation Chart. The important responsibilities for Core Management and In-service Inspection do not appear clearly in the lists of responsibilities distributed among these "Leaders".
2. Sections 13.3.2 to 13.3.6 provide the qualification requirements for the various "Leaders" listed in the Organisation Chart. The requirement in some cases for a university degree in science or engineering (e.g. the Reactor Operations Leader, the Reactor Maintenance Leader and the Reactor Utilisation Leader) may be too restrictive where persons with a lesser technical qualification and good appropriate experience may be just as competent for the task, if not more so.
3. Section 13.3.8 uses the old terminology "Health Physics" and "Health Physicist". Elsewhere the correct new terminology "Radiation Protection" and "Radiation Protection Officer" are used.
4. Section 13.6.1 "Facility Documents" lists the documents required for the safe operation of the facility, including the SAR and the OLC. Missing from this list is the document giving the Bases for the OLC, which, for the operators, is an indispensable extension of the OLC document. Documents that ensure safe operations should also be included, such as Modifications of Installations, Organisational Changes, Assessment of Safety Related Experiments etc.
5. Many of the acronyms emerging for operational use could be confusing, especially when used for giving verbal instructions over communications systems, and could potentially lead to operator misunderstanding and mistakes (e.g. FSS/SSS, FRPS/SRPS, RCPS/R&SPCS etc.).

4. COUNTERPART VIEWS AND MEASURES:

1. The correct responsibilities for Core Management and In-service Inspection will be checked and added accordingly.
2. The word "preferably" will be added to the requirement for a university degree where such a degree may not be absolutely essential for the Leader.
3. A search for the use of the old "Health Physics" term will be made and corrected throughout.
4. The OLC Bases document will be added to the list.
5. As part of our Safety Culture approach, communications training is an important part of the operator training for safe conduct of operations. This includes repeat back of messages and use of the phonetic alphabet for the reliable passing of information.

5. COMMENT:

C1. The counterpart's views and measures are sufficient to close this issue.

ISSUE NUMBER: 31

1. REVIEW AREA: SAR Section 13.2.3.9 - Shift Operations

2. ISSUE TITLE: Number of Accredited Operators

3. ISSUE CLARIFICATION:

A related Issue (Issue No 38) has been raised concerning the Operational Limits and Conditions (OLC) requirement for the minimum number of accredited operators and the period of continued operation allowed if this minimum is not met, and certain recommendations have been made in that regard. The present Issue concerns the <u>nominal</u> or <u>actual</u> staffing arrangements covered in the referenced section of the Safety Analysis Report and also in the draft Plant Operation Manual:

- From these references it would seem that the Operating Organisation intends to operate the facility always with the exact complement of accredited operators specified in the OLC as the minimum.
- In this case there would be a greater potential for the situation to arise, perhaps more often than should reasonably be tolerated, where the plant continues to operate, for the maximum time-out period allowed in the OLC, without the specified minimum accredited operators.
- Furthermore, in the draft Plant Operation Manual, the list of tasks assigned to one accredited operator "in the field" for a reactor start-up (procedure for transition from the Shutdown State to the Power State) is daunting. It should be expected that few start-ups are ever perfect and that this operator will be needed to deal with additional tasks not anticipated in the procedure.
- Since an authorised reactor operator shall be present in the control room during all reactor states it can be deduced that the additional shift member could perform the fuel handling alone. This does not conform to international best practice by which fuel handling is performed by at least two qualified shift members.

4. COUNTERPART VIEWS AND MEASURES:

The SAR considers only the minimum number of accredited operators required for safe operation of the reactor, as specified in the OLC. It is the intention of the operating organisation to man the shifts according to a task assessment that includes tasks other than those that are specifically safety related.

There is no nuclear safety (OLC) issue that we have identified that **requires** two operators. We consider that there is a clear distinction between OLC minimum staffing for nuclear safety and good practice operational staffing.

5. COMMENT AND RECOMMENDATIONS:

C1. It must be assumed that when the shift complement is at the minimum specified in the OLC, no fuel handling operations would be conducted.

R1. Since the minimum operator requirements specified in the OLC take precedence over anything stated in the referenced section of the SAR, it would be informative and not contrary to the context of the SAR to add the Operating Organisation's intention to this section.

R2. It is recommended that refuelling will be performed by two qualified shift members.

R3. The Operating Manual should certainly reflect the task loading of the intended complement of operating staff, and not only the minimum complement, as this document represents the nominal operating situation.

ISSUE NUMBER: 32

1. REVIEW AREA: Safety Analysis Report (SAR) in General

2. ISSUE TITLE: Post-construction Status of the SAR

3. ISSUE CLARIFICATION:

Much of the text of the Safety Analysis Report (SAR) is unchanged from the Preliminary Safety Analysis Report (PSAR) reviewed in May/June 2001, resulting in many statements that still make reference to future intentions regarding design and the safety of plant, systems and components, or refer to preliminary studies and assessments.

Since the construction phase is now nearing completion and the latest version of the SAR is meant to apply to the commencement of normal operations, there should no longer be vague references to matters regarding safety that are still unresolved.

The Review Team does not believe that there are important unresolved design or safety issues, but clearly the context and tense of the SAR needs to be carefully examined and corrected.

A number of specific examples were discussed with the counterparts, but it should be considered that they were only a few examples noticed by the Review Team while studying the particular scope of this mission.

4. COUNTERPART VIEWS AND MEASURES:

We acknowledge that there are a number of errors in the SAR, including incorrect use of tenses in some chapters. This will be corrected in the next revision of the SAR that will be prepared following the completion of commissioning. This revision will also incorporate appropriate resolution of any design and/or construction issues as well as the results of the commissioning.

In addition, a SAR Errata document is currently being prepared identifying and correcting typographic and consistency errors that have been identified in the SAR subsequent to its formal issue as part of the Application. The above error in relation to tense usage will be identified in this Errata document.

5. RECOMMENDATIONS:

R1. The Operating Organisation should undertake a project to fully review and revise the context of the SAR to be consistent with a reactor facility under normal operation, with issues of design and construction fully resolved. This should be done in such a way that any outstanding construction and as-built issues can be readily incorporated.

R2. During the discussions with the counterpart the intention was mentioned to finalise the SAR after hot commissioning in order to include commissioning results. The Review Team recommends finalising the SAR to reflect the as-built status of the facility as part of the Licence Application. It is expected that only very few modifications should be done later due to the commissioning results.

ISSUE NUMBER: 33

1. REVIEW AREA: **Operational Limits and Conditions (OLC)**

2. ISSUE TITLE: **General Comments on the OLC**

3. ISSUE CLARIFICATION:

The OLC form an envelope or boundary for reactor parameter values and system conditions within which safe operation of the facility should be assured.

Section 17.1.1 of the SAR states that these OLC are consistent with the IAEA Safety Series 35-G1 and the draft Safety Series Guide DS-261. However, Section 17.1.3 of the SAR states also that the OLC were developed using the guidance of the NUREG 1430 through 1434. The advantage is a uniform and concise format presenting the limiting conditions, their applicability, the required actions, the surveillance requirements and the allowable completion time. The presented OLC are quite complete and exhaustive and include all safety related instrumentation and systems. The information, in principle, covers much more than the minimum safe configuration of the facility. However, within this format the Safety System Settings and Limiting Conditions for Safe Operation are not presented as defined and elaborated in DS 261.

The bases and objectives of the OLC are presented in a separate document which has to be consulted for a full understanding of the limiting conditions, applicability and the related surveillance requirements. This together with a "complicated language" does not facilitate the accessibility of this document to the operators. An example of this is the special conditions described by LC 3.0.1 through LC 3.0.5 and SR 3.0.1 through SR 3.0.4. Clearly the correct use and interpretation of the OLC need an intensive training for all involved staff.

Specifically, the reporting conditions as described under SR 3.0.3 by which the surveillance interval can be extended by 100% without notice to the regulator, are against common practice. Obviously any breach of the OLC should be reported to the Licensing Authority within a well specified timeframe.

Some of the limiting conditions deal with parameters which cannot be verified by the operators. For example, LC 3.2.3 (Power Peaking Factor), LC 3.1.1 (Shutdown Margin), LC 3.1.2 (Reactivity Balance) and some of the specified actions under these limiting conditions are also outside the control of the operators.

It is not common practice that design features as under section 4.0 of the OLC form a part of the OLC and may be removed, with the exception of the conditions for the maximum capacity for fuel storage, which should become a limiting condition.

The defined reactor states as presented in the OLC do not cover the situation of an unloaded core (See draft Plant Operation Manual, section 5.1).

In contradiction of the recommendations of the safety Guide DS 261, the specific aspects of operational radiological safety are not covered by the OLC although this is covered by ARPANSA Standard Licence Conditions.

At the time of reviewing the OLC, these were still draft documents under QA check.

4. COUNTERPARTS VIEWS AND MEASURES:

The OLCs are consistent with the IAEA draft DS 261. However, DS 261 does not give detailed guidance on the format of the OLCs as it recognises (article 1.4) that there are various possible formats for their presentation. ANSTO chose to adopt the NUREG format as this is recognised by the Regulatory authorities in the US (NRC) and the UK (NII) as a clear, concise and comprehensive format that has been successfully used on over one hundred reactors.
There is a clear distinction between the Safety Limits (OLC 2) and the Limiting Safety System Settings (LSSS Allowable values) as in OLC 3.3.1 etc.

Trained operators find the format easy to follow as, in one or two pages, it clearly identifies the Limiting Condition, applicability, required actions and completion times and surveillance requirements. The OLC base is required to train the operators, but is not required for implementation. The separate OLC base is best practice.

LC 3.0.1 – 3.0.5 and SR 3.0.1- 3.0.4 are standard in the NUREG format and are fully accepted by various overseas regulators.

The OLC is a document used by a trained authorised operator. In addition to the inclusion of OLCs in systems and simulator training, a comprehensive OLC training program is scheduled before the start of cold commissioning.

SR 3.0.3 is included in the standard NUREG format as accepted by the NRC. Note that a risk assessment is required for any surveillance delayed greater than 24 hours.

It is usual practice that reactor performance programs are not owned by operators in the Main Control Room. The operator’s responsibility is to ensure that the program is performed in accordance with the surveillance requirements and if a condition is entered for any reason, that the required actions are completed.

Section 4 design features are included in the standard NUREG format. ARPANSA has made it clear that they expect at least some of these features to be retained.

The reactor pool spent fuel store has the design capacity for a full core unload but there are no routine operations identified during at least the first ten years of operations that would require a full core unload.

DS 261 recognises that the OLC depends on the legal framework of the country. ARPANSA require that various constraints and limitations listed in DS 261 are included under separate Licence Conditions. OLCs are specified in ARPANSA standard Licence Conditions 38-40. Radiological safety is not covered by the OLCs because it is covered by ARPANSA Standard Licence Conditions 30-34.

The QA check on the draft OLCs and bases has been completed and Revision 1 of the OLCs and Bases has been formally submitted to ARPANSA.

5. RECOMMENDATIONS AND COMMENTS:

R1. Limiting conditions should be easily verified by the operators to the extent possible or the responsibility for ensuring non-violation of the OLC should be identified.

R2. The applicability of the surveillance requirements SR 3.0.3 should be adapted to avoid an undesired and unlimited extension of the surveillance interval without reporting to the Licensing Authority.

R3. An intensive training program of all related staff (i.e. operators, maintenance, radiation protection....) with an appropriate examination of their correct understanding, should be put in place and completed before the beginning of the hot commissioning of the facility.

R4. The limiting conditions, including the minimum plant configuration, for the operational state with an unloaded core should be defined and incorporated in the OLC.

R5. It is recommended to incorporate the nominal trip set points as defined in “Determination of FRPS and SRPS set points” in the OLC, as specified in DS 261. Activation of trip settings is not regarded as breach of the OLC.

C1. It might be useful to the operators if an Index or Table would be added in an Appendix to the OLC associating the OLC applicable to each operating state in turn. This would eliminate a lot of paging through the document if the operator needs to know exactly what OLCs are applicable to a given operating state.

C2. In the OLC the allowable values are used. These values are derived from the analytical limits in which the response time and possible drift of the instruments between the surveillance intervals have been taken into account assuring sufficient acceptable margins. The allowable values and trip settings are clearly defined in “Determination of FRPS and SRPS set points”.

ISSUE NUMBER: 34

1. REVIEW AREA: **Operational Limits and Conditions (OLC)**

2. ISSUE TITLE: **Surveillance Requirement Frequency**

3. ISSUE CLARIFICATION:

Some of the surveillance requirements are proposed with very long time intervals of 12 months, 24 months...5 years. A number of these surveillance requirements are outside common practice for research reactors, taking into account the ever changing conditions encountered in these facilities (cycling through all four operational states at roughly monthly intervals, with more frequent changes taking place in the irradiation and experimental facilities).

For example, LC 3.1.3 (operability of the First Shutdown System) states that the control plate insertion time should be verified every 12 months. Common practice in Research Reactors is to perform this surveillance at least once after modification of the core (e.g. after a refuelling or after any work on the control plates/drives).

Very long intervals for surveillance can also be found in the OLC for the instrumentation channels. (For example SR 3.3.1 through 5 with time intervals ranging from 12 months to 5 years). An additional problem could arise from the fact that the neutron detectors are always excluded from any channel check or calibration (due to their lower drift, see Issue 27). Good practice is to regularly verify the correct insulation of the detectors and to calibrate the nuclear channels against thermal power in power state.

The same remark can be made for the surveillance requirements of more usual/classical components. For example SR 3.7.2.4 specifies a time interval of 2 years for the verification of battery capability of a UPS whereas it is well known that this could be a potential problem even with new, modern equipment.

Long surveillance intervals could barely be justified on the basis of a test program, a probabilistic analysis or a supplier recommendation, since the human factor coupled with the ever changing conditions, has turned out as the limiting factor in a research reactor facility. The sophisticated systems as proposed for this facility will lower the probability of an accident due to a hardware failure, but the residual risk will be mostly due to the human factor. Also, clearly experience gained with the operation of the facility will dictate adaptations/modifications to the surveillance requirements.

Appropriate surveillance requirements should also be applicable after each intervention or maintenance of a component or system.

4. COUNTERPART VIEWS AND MEASURES:

Many of the critical items (e.g. the control plates) have been extensively tested as prototypes and the surveillance requirements are supported by a detailed Probabilistic Safety Analysis (PSA). The risk of the human factor is increased with plant interference so that the human factor risk increases if the surveillance interval is decreased.

The Reactor Control and Monitoring System (RCMS) continually monitors the readings of all the instrumentation channels so that problems with an individual sensor may be quickly identified. In the case of neutron detectors, the insulation tests and calibration tests will be performed as good practice. Also, these instruments have a built in signal generator which enables the individual channel to be tested.

We have received the maintenance advice for the batteries and the manufacturer recommends an annual test. We will revise LC 3.7.2.4 to 12 months.

The experience gained during commissioning will be used to confirm or change the surveillance requirements.

When plant is returned to service after maintenance, post maintenance testing is always carried out to ensure the plant is operable.

5. RECOMMENDATIONS AND COMMENTS:

R1. It is recognised that the surveillance requirements have been based on a Probabilistic Safety Analysis and manufacturers recommendations. However it is still recommended to re-analyse the surveillance requirements paying special attention to the nature of a research reactor cycling through all four operational states at roughly monthly intervals, with more frequent changes taking place in the irradiation and experimental facilities.

R2. Surveillance requirements should be defined after an intervention or maintenance on a system;

C1. Operators should be actively encouraged to participate continuously in the surveillance of the equipment and systems and to report any deviation. The latter should be integrated in an overall documentation system.

C2. The IAEA International Conference on "Research Reactor Utilization, Safety, Decommissioning, Fuel and Waste Management", Santiago, Chile, 10-14 November 2003 made a number of recommendations on nuclear safety and physical security to the RR community and to the IAEA. One of them is as follows:

> *"To assure a more reliable outcome support activities for developing databases to assure more reliable results from PSA".*

The above recommendation was raised because there is no comprehensive and reliable PSA database available for Research Reactors.

C3. After further clarification by the counterpart it is noticed that a guide box for the control plates is part of the design affording more protection against deformations than normally applicable in open pool reactors. However it is still the opinion of the Review Team that proper operation of the control plates should be verified before every power stage.

ISSUE NUMBER: 35

1. REVIEW AREA: **Operational Limits and Conditions**

2. ISSUE TITLE: **Power Peaking Factor (PPF)**

3. ISSUE CLARIFICATION:

Operation with a Power Peaking Factor of three or less (PPF ≤ 3.0) ensures that fuel failures due to inadequate cooling do not occur during normal operation or during Design Basis Accidents or transients.

The proposed PPF limit is derived from fuel design analyses covering a chosen core management strategy. Any modification of the latter should be analysed, reviewed and reported.

Within a chosen core management strategy there is a need to define a limit for the misalignment between the control plates (when operating in manual mode) and to define the minimum surveillance requirements.

4. COUNTERPART VIEW AND MEASURES:

The fuel management strategy will be agreed with ARPANSA and any modification will have to be agreed under the existing ARPANS Regulation 51 and Regulation 52 as appropriate.

The PPF Verification Program addresses all the issues, but we will review the PPF analysis to determine whether any restrictions are necessary when the control plates are operated in manual mode. However, it should be noted that the effect on the PPF from the positions of the control plates has been analysed in a number of Detailed Engineering design reports that have previously been submitted to ARPANSA in support of request for approval under ARPANS Regulations 51 and 54. These reports have shown that for a large range of unbalanced control plate conditions, the PPF remains well bellow the limit of 3.0. Therefore, it is considered reasonable that control plate unbalance is not included in the OLCs. Not withstanding this, whenever the reactor is operating at a power level above 10 MW, a warning alarm will be triggered in the Main Control Room console if a control plate unbalance situation is detected.

5. RECOMMENDATIONS:

R1. It is recommended to verify whether all possible misalignment combinations of the control plates have been analysed, and introduce a specified limit for the misalignment between control plates (plus surveillance requirements) as appropriate.

R2. The Licensing Authority should approve any safety related modification of the core management strategy. A core management strategy programme should be incorporated into the administrative controls of the OLC.

ISSUE NUMBER: 36

1. REVIEW AREA: Operational Limits and Conditions

2. ISSUE TITLE: Acceptable Margins in the Limiting Conditions for Safe Operation

3. ISSUE CLARIFICATION:

Conformance to DS 261 requires that Limiting Safety System Settings (LSSS) shall be established for all operational states of the reactor. In determining a safety system setting, the process and measuring uncertainties shall be taken into account. Conformance to NS-R-4 requires that the safety margin will allow for, among other things, behaviour in system transients, the equipment response time and inaccuracy of the measuring devices. Consequently the safety system settings should have an acceptable margin in relation to the parameters as used in the Safety Analyses Report.

A number of the settings presented in the OLC are the values used in the Safety Analysis Report and the required margin to allow for behaviour in system transients, the equipment response time and inaccuracy of the measuring devices are not present.

Non-exhaustive examples of such settings are the shut down margin (LC 3.1.1), control plate drop time (SR 3.1.3.2), number of trip settings for First Reactor Protection System (Table 3.3.1-1), time delay on failure of first reactor protection system (LC 3.3.3) and filter efficiencies (SR 3.5.4.3; 3.5.4.4).

Limiting conditions to be determined for the OLC during the commission phase are presented between brackets.

4. COUNTERPART VIEWS AND MEASURES:

The Determination of Trip Settings document clearly demonstrates the margins between safety limits and trip settings. The analytical limit (AL) is derived from the safety analysis taking into account the system response time (e.g. plate insertion). The Allowable Value (LSSS), listed in the OLC, is derived from the AL with allowances for the instrument and process errors. The trip setpoint allows for instrument drift between calibrations. All of the AV listed in Table 3.3.1 of the OLCs included this margin to the safety limits.

With respect to the shutdown margin, uncertainties are already included.

The time delay on failure of the First Shutdown System (LC 3.3.3) is based on the control plate insertion time of 900 ms.

The Logic System function test OLC 3.3.1 overlaps the control plate insertion test 3.1.3.2 to provide complete testing of the First Shutdown System.

The operator is provided with all the information relating to safety limits, safety system settings, alarm values and normal operating conditions in the plant operating manuals. The OLC format ensures that the operator concentrates on the nuclear safety boundary and adding unnecessary detail undermines this goal.

5. COMMENT AND RECOMMENDATION:

C1. After further clarification by the counterpart regarding the analytical limits, allowable values and nominal set points, see also comments on issue 33, this issue except for the filter efficiencies is regarded as close.

R1. Additional surveillance requirements as rejection criteria for the filter efficiency for the Containment (LC 3.5.1) and the Hot Cell Exhaust Systems (LC 3.5.4) should be specified allowing sufficient margin to the efficiencies used in the SAR.

ISSUE NUMBER: 37

1. REVIEW AREA: Operational Limits and Conditions

2. ISSUE TITLE: Extension of Administrative Controls

3. ISSUE CLARIFICATION:

A set of administrative controls are incorporated into the OLC regarding the Operating Organisation and special programmes for Containment Performance Testing, Irradiation Requirements, Diesel Oil Testing, OLC Bases Control and Safety Function Determination Programme will be implemented. Although not formally stated, the initial programmes will also be issued to ARPANSA for formal approval.

For completeness the administrative controls to be agreed with ARPANSA could be extended with:

- A Core Management Strategy including relevant core calculations (see Issue No. 35)
- An Incident Reporting System in which clear guidelines should be given for which type of incidents and accidents should be reported (i.e. radiation incidents, fire, evacuation, violation of OLC, incidents of public interest, etc.) including the time frame and reporting format.
- An Overview of the safety related procedures and instructions to be issued to ARPANSA for information, review or reporting (i.e. modifications of systems and components) and a procedure for safety related changes of the organisation structure.

4. COUNTERPART VIEWS AND MEASURES:

The administrative controls for core management strategy are already included in OLC 5.3.2 Core Operational Limits Report.

Note that the programs (and all the OLCs), must be "acceptable to the CEO of ARPANSA" – they are not approved. ARPANSA approves changes.

Under the Australian legal system, the Incident Reporting System is covered in ARPANSA Standard Licence Conditions 18 and 19 and Safety Management arrangements are also covered by ARPANSA Standard Licence Conditions (e.g. 24 to 29 for modifications) and are therefore not included in the OLCs. Note also that *Plans and Arrangements for Managing Safety* have been submitted to ARPANSA in Part B of the Application for an Operating Licence.

5. COMMENT:

C1. It is noticed that the Issues raised are covered by Standard Licence Conditions.

The issue is considered to be closed.

ISSUE NUMBER: 38

1. REVIEW AREA: Operational Limits and Conditions

2. ISSUE TITLE: Staff Requirements for the Operating Organisation

3. ISSUE CLARIFICATION:

The proposed organisation is briefly addressed in the administrative controls of the OLC. In this proposal the Reactor Manager shall have responsibility for overall safe operations and shall have control over those activities necessary for safe operations and maintenance. On the other hand a specified ANSTO officer not reporting to the Reactor Manager shall have responsibility for overall nuclear safety of the facility. This does not conform to the requirements of NS-R-4.

In order to avoid any safety conflict the lines of authority, tasks and responsibilities should be well specified. In order to have control of those activities to assure safe operation by the Reactor Manager, the Analysis and Engineering Groups should be integrated under the responsibility of the reactor manager.

The minimum facility staff requirements for the POWER and PHYSICS-TESTS states to assure safe operations is defined as two authorised reactor operators and one additional shift member. For the REFUELLING and SHUT-DOWN states one authorised reactor operator and one additional shift member have been defined. The tasks, responsibilities and competences of the shift members shall be well defined and the additional shift member should be qualified as appropriate.

The job and task analyses are not yet completed to justify the proposed minimum requirements. In these analyses all operational states, safety related operations regarding utilisation, incidents and accidents as well as emergency situations should be incorporated.

The staff requirements only address normal operation after completion of commissioning. The staff for commissioning should be included and use should be made of experienced operators to the extent possible.

4. COUNTERPART VIEWS AND MEASURES:

The organisation, staff responsibilities and qualifications are described in detail in SAR Chapter 13 *Conduct of Operations* and in the *Plan for Maintaining Effective Control* (RRRP-7200-EDEAN-003) submitted in Part B of the Application for the Operating Licence.

The development of the final operating organisation is still in progress and we will take the Review Team's recommendations into consideration.

The Task Analysis to verify the minimum shift staffing requirements for the purpose of the OLC (nuclear safety) has been completed. Using an approach acceptable to ARPANSA, similar to the analysis performed for HIFAR, two limiting events (a fire and a LOCA) were examined. In all cases, two authorised operators would be sufficient to ensure safety.

These OLCs are for normal operation after commissioning. Separate OLCs apply during commissioning where appropriate.

Operating staff are already taking part in pre-commissioning testing and will be involved with all commissioning stages.

5. RECOMMENDATIONS:

R1. It is recommended to re-evaluate whether all operational states (including transitions such as start-up), safety related operations regarding utilisation, incidents and accidents as well as emergency situations have been incorporated in the job and task analyses for the shift members in order to justify the proposed minimum staffing requirements. As described in DS 261 the proposed organisation to assure safe operation of the facility should be based on these analyses and should be reviewed and approved by the CEO of ARPANSA.

R2. In order to have control by the Reactor Manager of those activities that assure safe operation, the Analysis and Engineering Groups should be integrated under the responsibility of the Reactor Manager.

It is recommended that the lines of authority, responsibilities and communications for all involved staff members to assure safe operations be defined. Special attention should be given to the lines of authority, responsibilities and communication of the Reactor Manager and the specified ANSTO officer in order to avoid any safety conflict.

R3. The staff for hot commissioning should be specified and should be included in the dedicated OLC for commissioning. Experienced operators should be involved in the hot commissioning to the extent possible.

ISSUE NUMBER: 39

1. REVIEW AREA: Operational Limits and Conditions

2. ISSUE TITLE: Containment Operability

3. ISSUE CLARIFICATION:

Containment operability is maintained by limiting leakage of the penetrations to a predetermined level together with sound operation of all penetration paths. The maximum overall allowable leak rate is 3 % per day. This will assure limitation of the release of fission products and radioactive gases to the environment in case of a postulated accident.

As presented in the administrative controls, containment performance operability will be verified in accordance with the containment performance programme. The operability of the containment should be a prerequisite for any state of the facility with risks of release of activity beyond the granted license. Inoperability of the containment that will challenge the maximum allowable leak rate shall require actions to restore the facility immediately into a state precluding uncontrolled release of activity. Actions to restore the containment to an "operable status" with a completion time of 7 days as specified by LC 3.5.1 does not conform to international best practice when the maximum allowable leak rate is challenged.

From the definition of "operable" regarding the containment, section 3.5.1, and the containment systems, section 3.5.2, both the limiting condition and the prescribed actions to be taken could be misunderstood. The definitions and the required actions should be precisely described in order to avoid misunderstanding of the objective to assure the containment function. The allowable completion times should be made dependent on the possible consequences of the inoperability. For smaller leak paths a completion time of 7 days is good international practice, but actions regarding inoperability of the containment challenging the allowable leak rate should assure that the facility will be put immediately into a state precluding the accidental release of any activity.

4. COUNTERPART VIEWS AND MEASURES:

The OPAL Reactor is provided with a dynamic containment to limit the release of radioactivity to the environment in the event of a beyond design basis accident. It is not required to mitigate design basis accidents, although it will contain any release that may occur within the containment as a result of such accidents. As such, the required operability is not as restrictive as would be for a nuclear power plant or other research reactors' containment where it is required to mitigate design basis accidents.

The completion time for action in LC 3.5.1 for when the containment is determined to be inoperable has been determined from the PSA for the OPAL Reactor using a risk-based method described in NUREG/CR-6141. This method indicated that a completion time of 550 hours (roughly 22 days) was acceptable for the inoperability of the containment without any significant increase in the overall risk associated with releases from the facility. To allow for conservatism and to ensure that standardised times were used throughout the OLCs, the actual completion time identified in LC 3.5.1 is 7 days.

The comments relating to the definition of "operable" are noted and will be taken into consideration to determine whether the wording could be improved or clarified. In addition, the point made regarding whether a graded approach to the actions depending on the specifics of any leak path identified will also be considered. However, ANSTO consider that if implemented, this would be reflected in the actions in LC 3.5.2, which addresses the operability of the individual containment penetrations, rather than LC 3.5.1, which addresses the overall operability of containment.

5. COMMENT AND RECOMMENDATIONS:

C1. Limiting Conditions LC 3.5.1 and LC 3.5.2 could be merged into one Limiting Condition.

R1. The definitions and the required actions should be precisely formulated in order to avoid any misunderstanding of the objective to assure the containment function.

R2. Actions regarding inoperability of the containment and/or inoperability of one or more flow paths challenging the allowable leak-rate should assure that the facility will be put immediately into a state precluding the accidental release of activity.

ISSUE NUMBER: 40

1. REVIEW AREA: Operational Limits and Conditions

2. ISSUE TITLE: Water Chemistry

3. ISSUE CLARIFICATION:

The core structures and cooling systems of the facility are made of different materials. In particular Zircalloy, Aluminium and Stainless Steel are the most important materials of the core structure and the coolant systems subjected to neutron flux. Specific requirements for the water chemistry are needed to avoid activation, long-term deposits and corrosion of materials, in particular, the fuel cladding. These requirements are not foreseen in the present Operational Limits and Conditions.

4. COUNTERPART VIEWS AND MEASURES:

Limits and Conditions are included in the operating manuals as water chemistry is important for long term component life. As such, ANSTO considers that water chemistry does not fall within the scope of criteria a), b) or c) identified in the Safety Analysis Report Chapter 17, Section 17.4.2 for inclusion as an OLC. However, in the light of the Review Team's comments, an OLC for water chemistry will be added under criterion d).

5. RECOMMENDATION:

R1. The minimum requirements for the water chemistry of each coolant system should be determined and the necessary limiting conditions and the surveillance requirements should be incorporated in the OLC.

ISSUE NUMBER: 41

1. REVIEW AREA: Operational Limits and Conditions

2. ISSUE TITLE: Pool Trip Level of Second Reactor Protection System

3. ISSUE CLARIFICATION:

As stated in the background for Limiting Condition 3.4.1 the reactor pool is the ultimate heat sink also for core cooling. To assure natural convection a Limiting Condition on the pool water level has been specified at + 7.6 m.

In table 3.3.3-1 the trip settings for the Second Reactor Protection System are presented in which a trip setting of + 5.8 m is defined to initiate a reactor shutdown in case of a loss of coolant accident (LOCA).

Since natural convection has to be assured, during the Power and Physics Test States, the trip level for the Second Reactor Protection System should also be initiated at least at + 7.6 m.

4. COUNTERPART VIEWS AND MEASURES:

The comment is acknowledged the recommendation will be taken into consideration.

5. RECOMMENDATION:

R1. It is recommended to define the trip setting for the Reactor Pool Water Level initiated by the Second Reactor Protection System at least at + 7.6 m, assuring natural circulation in case of a LOCA.

ISSUE NUMBER: 42

1. REVIEW AREA: Operational Limits and Conditions

2. ISSUE TITLE: Containment Energy Removal Systems (CERS)

3. ISSUE CLARIFICATION:

The Containment Energy Removal System provides an assured heat sink for all thermal loads inside the containment whenever the containment is isolated.

Limiting Condition LC 3.5.3 states that, with the reactor at power, the two trains can remain inoperable for 24 hours. The only justification found in the Bases is the low probability of a beyond design basis accident occurring during this period.

4. COUNTERPART VIEWS AND MEASURES:

Similar to what was stated in the ANSTO response to Issue 39, the completion time for action in Limiting Condition LC 3.5.3 for when both trains of the CERS are determined to be inoperable has been determined from the Probabilistic Safety Analysis for the OPAL Reactor using a risk-based method described in NUREG/CR-6141. This method indicated that a completion time of 33 hours was acceptable for the inoperability of the CERS without any significant increase in the overall risk associated with releases from the facility. To allow for conservatism and to ensure that standardised times were used throughout the OLCs, the actual completion time identified in Limiting Condition LC 3.5.1 is 24 hours.

It should be noted that in the event of unavailability of the CERS during normal operation, heat removal from the containment may be achieved by the normally operating Reactor Containment Ventilation System (RCVS) air supply and exhaust with no recirculation.

5. RECOMMENDATION:

R1. The 24 hour allowance for the inoperability of the two CERS trains should be better justified on the basis of the present thermal loads and permissible pressure and temperature increases in the containment in case of accident conditions, see also comments regarding the use of PSA at issue 34.

ISSUE NUMBER: 43

1. REVIEW AREA: Operational Limits and Conditions

2. ISSUE TITLE: Reflector Explosive Gas Concentration

3. ISSUE CLARIFICATION:

In Limiting Condition LC 3.6.2 maximum concentrations have been specified for deuterium and oxygen in order to ensure that these concentrations are maintained well below explosive levels. To assure effective operator action in case a limit is exceeded, the recombiner unit should be operable.

4. COUNTERPART VIEWS AND MEASURES:

An OLC specifies a Limiting Condition (in this case the gas concentration) and the operator carries out surveillance to ensure the limit is not reached. If the gas concentration unexpectedly reaches the limit, the operator has to shutdown the reactor. This is consistent with international best practice, even for power reactors.

5. COMMENT:

C1. A check for the operability of the recombiner unit should be incorporated in the operation procedures.

ISSUE NUMBER: 44

1. REVIEW AREA: Operational Limits and Conditions

2. ISSUE TITLE: Emergency Control Centre Ventilation (LC 3.6.5)

3. ISSUE CLARIFICATION:

The Emergency Control Centre (ECC) ventilation system provides a habitable environment in the ECC for operators during design basis accidents when the main control room is uninhabitable or unavailable. The ECC ventilation system consists of two independent trains each capable of independently fulfilling the full safety function of the system.

Limiting Condition LC 3.6.5 states that if one train is unavailable, one should immediately determine the operability of the second train. The action "Determine OPERABLE train is not inoperable due to common cause failure" is unique in the Operational Limits and Conditions, although this action could also be applicable to other redundant systems.

4. COUNTERPART VIEWS AND MEASURES

The action "Determine OPERABLE train is not inoperable due to common cause failure" is not unique in that that is a similar action associated with the standby diesel generators (Limiting Condition LC 3.7.1, Action B.3). In both cases, the Limiting Condition applies to a standby system that is not normally operating and that as such, without specifically verifying operability, it is not possible to determine whether a train is operable or not. Clarification will be incorporated into the Bases for both LC 3.6.5 and LC 3.7.1 to make this clear.

5. COMMENT:

C1. **After further clarification the Review Team consider this issue to be closed.**

ISSUE NUMBER: 45

1. REVIEW AREA: Operational Limits and Conditions

2. ISSUE TITLE: Hot Cell Exhaust System

3. ISSUE CLARIFICATION:

The function of Hot Cell Exhaust System is to control and treat airborne releases from the Hot Cell to the environment in the event of a Beyond Design Basis Accident.

Following Safety Analysis Report chapter 16.19.4, it is postulated that three U-Mo targets are erroneously removed from the decay rack before adequate cooling and are transported to the Hot Cell. Safety Analysis Report chapter 11.4.5.1 states that an interlock triggered by a radiation monitor will prevent the transfer of targets with insufficient cooling time to the Hot Cell.

An adequate protection against an early transfer of irradiated U-Mo targets to the Hot Cell is heavily dependent on administrative procedures. The proposed interlock based on the radiation level is in principle a good example of the "defence in depth" principle. Clearly the afforded protection is related to the measured radiation level and the decay time of the targets after irradiation. However calculations to predict the set point settings should be carefully checked by measurements and the adequate triggering level determined. When different numbers of targets and/or targets with different irradiation conditions (position and/or cooling time) and/or material compositions are being transferred through the transfer channel to the hot-cells, special attention should be given to the different set-point settings which have to be introduced for the different targets. Strict administrative procedures with independent checks should be introduced to avoid wrongly defined set-point settings, thereby introducing unreliable safety set-points.

4. COUNTERPART VIEWS AND MEASURES:

It should be noted that the design of the Molybdenum Safety Device (MSD); the formal name of the interlock on the Service Pool Elevator (SPE) referred to in the issue incorporates a collimator so as to ensure that the detectors only see one target at a time. Thus, the number of targets in a rig (between 1 and 6) has no effect on the operation of the interlock since it determines whether each individual target is too hot. In addition, the intention is to set a single MSD interlock set-point for the limiting irradiation target that might be transferred using the SPE. However, the Review Team's comment in relation to variable set points is noted and will be taken into consideration.

5. RECOMMENDATION:

R1. It is recommended to establish strict administrative procedures for the set-point settings with independent checks by which different numbers of targets and/or targets with different irradiation conditions (position and/or cooling time) and/or material compositions are taken into account.

ISSUE NUMBER: 46

1. REVIEW AREA: Operational Limits and Conditions

2. ISSUE TITLE: Second Shutdown System

3. ISSUE CLARIFICATION:

The Second Shutdown System is the backup system to shut down the reactor. This is done by automatically dumping the heavy water stored in the reflector vessel.

Limiting Condition LC 3.1.4 requires verification of the operability of each Second Shutdown System discharge valve and to proceed to a system functional test. Conditions for success should be specified in conformance to the analyses performed in the Safety Analysis Report (chapter 16.3.3.3.2.1 and 5.5.4.9)

4. COUNTERPART VIEWS AND MEASURES:

The Review Team's comments are noted and will be taken into consideration. ANSTO agrees that the success criteria for the Surveillance Requirements should be identified. However, it should be noted that the key performance indicator for the Second Shutdown System is the time it takes to insert sufficient negative reactivity into the reactor core. This cannot be measured directly but only implied by the drainage rate of the heavy water into the heavy water storage tank. As such, the Surveillance Requirement should identify a criterion for this drainage rate.

5. RECOMMENDATION:

R1. It is recommended to specify a maximum stroke time for the test of the Second Shutdown System discharge valves and specify a maximum dumping time for the functional test of the system in conformance with the analyses in the Safety Analysis Report.

ISSUE NUMBER: 47

1. REVIEW AREA: Application Documents

2. ISSUE TITLE: Definition of Final Set of Licensing Documentation

3. ISSUE CLARIFICATION:

For the preparation of the mission the full License Application has been sent to the experts. During the review and the discussions with the counterpart it appeared that the documents transmitted to ARPANSA for the formal application for a facility licence are still subject to review, modifications and improvements.

Due to these continuous improvements it could be misunderstood which documents will form the basis of the licensing documentation.

As also addressed in Issue 32, the Safety Analysis Report needs additional review in order to address the as-built status of the reactor.

The revision of the Operational Limits and Conditions sent to the experts was also still under internal review by ANSTO and the review during the mission has been carried out on a next revision which still had to be subject to Quality Assurance control.

Submission of documents still subjected to improvements and changes is not common practice for a formal licence application.

4. COUNTERPART VIEWS AND MEASURES:

All documentation prepared and submitted as part of the Application for an Operating Licence has been done in accordance with the requirements of the Replacement Research Reactor Project (RRRP) Quality Management System (QMS). In the case of Contractor supplied documentation (e.g. design documentation from INVAP), this means review, verification and acceptance in accordance with project procedure RRP 4.3 prior to submission. In the case of internal project documentation, this means an appropriate internal and, where appropriate, external review and sign off. The revision of documentation is subject to the same QMS and is appropriately controlled.

ANSTO understands that the Licence itself will identify the licensing documentation upon which approval is based, either directly or via a cross-referenced document. In the case of the Construction Licence, the licensing documentation was identified in the ARPANSA Regulatory Branch Assessment Report (RAR), which was referenced by Licence Condition 4.1.

It should be noted that many of the documents contained in the Application for an Operating Licence are “living” documents in that they are subject to ongoing review and revision throughout the life of the facility. This includes the Safety Analysis Report, which will first be revised after the completion of commissioning to reflect the results of the commissioning and the resolution of any design and/or construction issues, i.e. to fully reflect the "as-built" design. All changes to licensing documentation will be advised to ARPANSA in accordance with ARPANS Regulation 52 and where such changes have significant implications for safety, prior approval will be sought in accordance with ARPANS Regulation 51.

It should also be noted that in the case of the Operational Limits and Conditions, they have undergone an extensive internal ANSTO review and revision process since their original submission to ARPANSA as Part D of the Application for an Operating Licence. This review and revision arose as a result of a number of issues, including the further development of the operating organisation.

5. RECOMMENDATIONS:

R1. It is recommended that ARPANSA and ANSTO agree upon the documentation, which still needs some internal assessment by ANSTO and the time frame when the final versions will be submitted for additional or final review by ARPANSA. Upon completion of this process a final set of documents should be defined on which the review will be based for the licence.

R2. A clear procedure should be put in place by ARPANSA for the submission of new versions of the application documentation.

APPENDIX III: SCHEDULE

The peer review project progressed on the following schedule:

22 October 2004	IAEA receives request for peer review	ARPANSA
30 November 2004	Peer review planning meeting	ARPANSA/IAEA
2 December 2004	Agreement on documents availability date	ARPANSA
17 February 2005	Information on review team provided	ARPANSA/IAEA
28 February 2005	Introduction of peer review team	ARPANSA/IAEA
28 February 2005	Introduction of peer review team	ANSTO/IAEA
28 February 2005	Presentation of RRR design, safety	ARPANSA/IAEA
28 February 2005	Walk-down the facility	ARPANSA/ANSTO/IAEA
01 March 2005	Presentation on OLC& Conduct of Operations	ANSTO
01 March 2005	Review of material, prepare questions	IAEA
02-03 March 2005	Discussion on review topics	IAEA/ANSTO
04 March 2005	Prepare draft issue pages	IAEA
07 March 2005	Presentation of the issues	IAEA/ARPANSA/ANSTO
08-09 March 2005	Working groups clarify issues with the counterparts	IAEA/ANSTO
10 March 2005	Draft the Peer-review Mission Report	IAEA
11 March 2005	Present draft report to ARPANSA	IAEA/ARPANSA
11 March 2005	Exit meeting	IAEA/ARPANSA
11 March 2005	Meeting of the Nuclear Safety Committee	NSC/IAEA/ARPANSA
5 April 2005	Comments on draft report due to IAEA	ARPANSA
11 May 2005	Final report due to ARPANSA	IAEA

APPENDIX IV: LIST OF PARTICIPANTS

IV.1 ARPANSA – Counterparts

Don Macnab	Director, Regulatory Branch
Vince Diamond	Manager, Nuclear Installations
Guenael Le Cann	Assessment Engineer
Michael Kerr	Safety Engineer, Nuclear Installations

IV.2 ANSTO - Counterparts

Greg Whitbourn	Project Manager
Ross Miller	Assistant Project Manager
Mark Summerfield	Safety & Licensing Manager
Tony Irwin	Commissioning Reactor Manager
Ken Horlock	Nuclear Technology Project Engineer
Robert Godfrey	Operations Planning
Bob Harrison	Materials Advisor
Andrew Frikken	Test and Trials Manager
Shane Harrison	Construction, Buildings and Structures

IV.3 INVAP - Counterparts

Pablo M. Abbate	Design and Commissioning Manager
Nestor de Lorenzo	ILS Manager, Nuclear Projects

IV.4 IAEA – Working Groups

Group 1:
Reviewers: J. P. Boogaard
P. Gubel
Review area: Operational Limits and Conditions, Conduct of Operations

Group 2:
Reviewers: A. D'Arcy
T. Hargitai
Review area: Conduct of Operations, Commissioning, Follow-up of the issues raised by the previous mission

Note: Safety Analysis Report chapters reviewed by all members of the Review Team

IV.5 IAEA – Review Team CVs

Name:	Alan J D'Arcy	Office Phone:	+27 12 305 5017
Address:	SAFARI-1 Research Reactor	Cell Phone:	+27 83 304 7889
	South African Nuclear Energy Corporation	Email:	alan@necsa.co.za
	PO Box 582,		
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	SOUTH AFRICA		

Professional Qualifications:
Registered as a Professional Engineering Technologist (Mechanical Engineering) with the Engineering Council of South Africa (ECSA)
BSc Computer Science, University of South Africa, Pretoria (1990)

Relevant Experience:
23 years in nuclear reactor safety analysis (nuclear power plant and research reactors), of which the last 13 years has been research reactor -specific, with responsibility for: Management of nuclear safety and licensing for the SAFARI-1 Research Reactor; Safety analyses of the reactor and experimental facilities and generation of the Safety Analysis Report; Core and fuel management; Fuel design control; Safeguards; Operator training; Secretary and nuclear safety and training expert for the SAFARI-1 Reactor Safety Committee. Since 1999 have attended and participated in numerous IAEA activities such as Technical Committee Meetings, consultancies and expert missions related to research reactor safety.

Relevant Expertise:
- Conduct and review of safety analyses,
- Generation and review of SAR,
- Preparation of the OLC,
- Licence applications and liaison with the regulating authority,
- Reactor operation,
- Operator training.

Name:	Jeannot P. Boogaard	Office Phone:	+31 224 56 4832
	Manager QSE	Cell Phone:	+31 653 33 0255
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Professional Qualifications:
Physical Engineer, Delft Technical University, 1983

Relevant Experience:
More than 20 years of experiences in nuclear industry including 12 years as head of the maintenance group and deputy reactor manager HFR a 45 MW Research Reactor, including experiences with an extended upgrading and modifications program, including commissioning. In present position as Manager QSE overall responsibility for policy and implementation of Quality, Safety and Health Physics, Environmental Management and Security Management within NRG. Furthermore responsible for management control and formal administration of all licenses.

As project manager for the license applications of the nuclear activities of NRG (without the HFR) overall responsible for the safety analyses report, safety analyses performed and the Operational Limits and Conditions for the Low Flux Reactor, Hot Cell Laboratories and Molybdenum Productions Facilities as well as for the Decontamination and Waste Storage Facilities.

Since 1988 Dutch representative in several IAEA meetings for drafting and review of IAEA documents related to research reactor safety, operation, maintenance and management systems.

Relevant Expertise:
- Conduct of operations
- Operation, utilisation, maintenance and refurbishment
- Instrumentation and control
- License application, including preparation of safety analyses reports and safety analyses
- Preparation of Operational Limits and Conditions
- Management systems
- Emergency planning and preparedness

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Professional Qualifications:
Electrical Engineering (University of Louvain), 1964
Nuclear Sciences (University of Louvain), 1967

Relevant experience:
More than 35 years of research reactor experience including 20 years of experience as reactor manager of the BR3 plant (PWR plant of 40 MW) and of the MTR BR2 (120 MW capacity).
Head of the Refurbishment Program of the BR2 reactor (1995-1997) including a complete overhaul and modernization of the plant after 30 years intensive utilization.
Participated as an expert to various technical committees of the IAEA.

Consultant (October 1981-February 1982) to the Licensing Board of the Atomic Energy Commission of South-Africa.

Relevant expertise:

- -Conduct of operations
- -Operation and maintenance
- -Refurbishment programme
- -Safety assessment of research reactors and experiments
- -Utilization of research reactors
- -Fuel management

Name:	Tibor Ferenc Hargitai	Office Phone:	+431 2600 26176
	RR Safety Officer	Cell Phone:	+3630 4886122
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	A-1400 Vienna,		
	Austria		

Professional Qualifications:

Electric Engineer, Budapest Technical University, 1972

Nuclear Engineer, Budapest Technical University, 1990

Relevant Experience:

More than 30 years of research reactor relevant experience including 13 years of experience as Reactor Manager. As Reactor Manager, responsible for the re-commissioning of the Budapest Research Reactor after a major reconstruction in 1992. In the reconstruction responsible for the design and construction of the in core instrumentation. Responsible for design, construction, installation of a cold neutron source in 2000.

Since 2002 IAEA staff member working for the Nuclear Safety Nuclear Installation Division – Research Reactor Safety Section.

Relevant Expertise:

- Reactor diagnostics
- Instrumentation and Control
- Cold Neutron Source
- Conduct of operations
- Operation and maintenance
- License application, including preparation of safety analyses reports and safety analyses
- Preparation of Operational Limits and Conditions
- Emergency planning and preparedness

APPENDIX V

LIST OF ADDITIONAL DOCUMENTS

ANSTO Application for a Facility Licence

Part A: General Information:

- Application for a Facility Licence, Operation Authorisation for the Replacement Research Reactor

Part B: Plans and Arrangements for Managing Safety:

- Safety Management Plan
- Radiation Protection Plan
- Management of Radioactive Waste
- Ultimate Disposal or Transfer Plan
- Reactor facility Emergency Plan
- Environmental Management Plan

Part C: Safety Analysis Report and Associated Information

- List of Plant Manuals

Part D: Operational Limits and Conditions

Part E: Plans and Arrangements for Hot Commissioning

- Commissioning Plan

During the mission the following documents were provided additionally:

- Regulatory Branch Assessment Report on the Facility Licence Application F0118 to Construct a Controlled Facility, the Replacement Research Reactor
- CNS PSAR Rev 0
- Plant Operation Manual (draft, uncontrolled copy)
- Service Pool Elevator - Operation Manual (draft, uncontrolled copy)
- Reactor and Service Pools Cooling System -Operation Manual (draft, uncontrolled copy)
- Stage A Commissioning Specific Plan (draft, uncontrolled copy)
- Sub-Stage B1 Commissioning Specific Plan (draft, uncontrolled copy)
- Sub-Stage B2 Commissioning Specific Plan (draft, uncontrolled copy)
- Stage C Commissioning Specific Plan (draft, uncontrolled copy)